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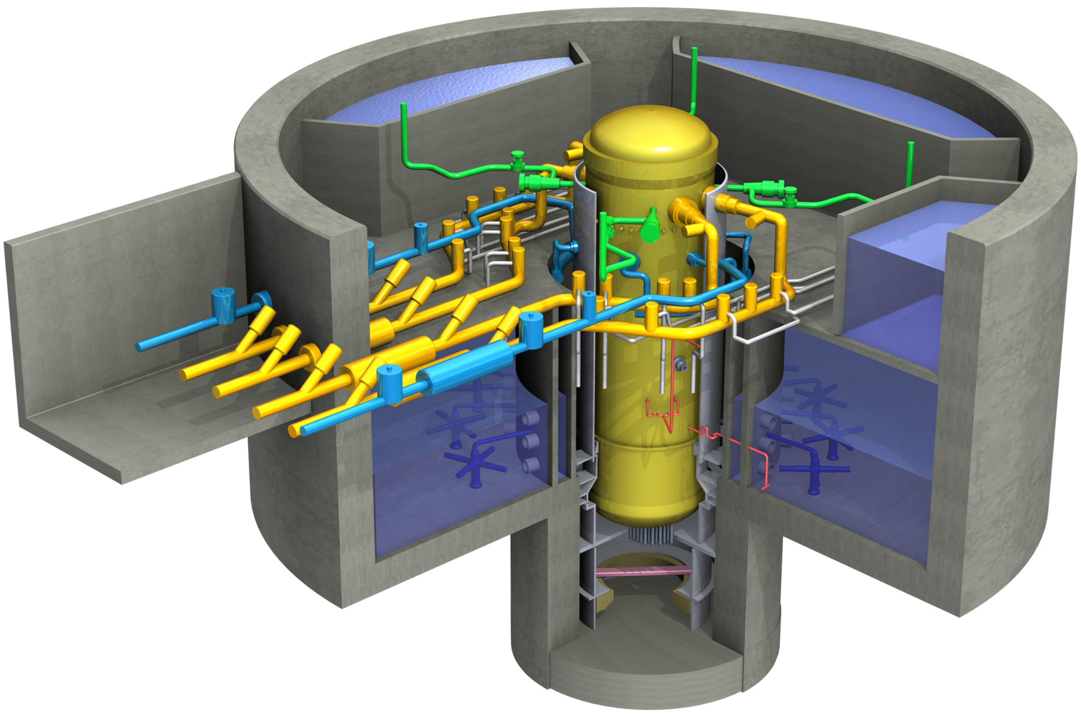

ESBWR Design Control Document
Tier 2

Chapter 6
Engineered Safety Features

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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 (ECCS), and (3) control room habitability systems (CRHS). The systems in each general group are:

(1) Fission Product Containment and Containment Cooling Systems.

  a. Containment System.

  b. Passive Containment Cooling System (PCCS).

(2) 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) Heating, Ventilation, and Air Conditioning (HVAC) Subsystem (CRHAVS).

6.1 DESIGN BASIS ACCIDENT ENGINEERED SAFETY FEATURE MATERIALS

Materials used in the Engineered Safety Feature (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 [RG] 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 NUREG-0800, Standard Review Plan (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-Accident (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 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 RG 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 RG 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 RG 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 (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 304 L stainless steel-lined 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) is 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 remains remain within an acceptable design basis range 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 RG 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 (DW) and wetwell (WW).

Consistent with the rationale of RG 1.54, the WW and attendant vertical vents are designated as a Service Level I area. All surfaces and equipment in this area are either uncoated, corrosion resistant stainless steel, or coated in accordance with RG 1.54 and referenced ASTM standards, as applicable.

The remainder of the containment, (i.e., the upper and lower DW and annulus areas) are physically separated from the Service Level I area, such that there is no practicable method by which coating debris may reach any strainer or screen in the suppression pool. Protective covers over the vertical vents prevent debris from being swept directly into the WW. Debris which might originate inside the annulus or be carried into the annulus and lower DW, has sufficient time to settle before water levels there can reach the spillover holes to the vent wells and WW. Due to these design features, the failure of coatings in the upper and lower DW, and areas in between, cannot adversely affect the operation of any post-accident fluid system

Regardless of service level designation, all field applied epoxy coatings inside containment meet the requirements of RG 1.54 and are qualified using the standard ASTM tests, as applicable to procurement, installation, and maintenance.

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 100 Gy (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 and cannot practicably reach any strainer or screen in the suppression pool.

6.1.3 Combined License (COL) Information

6.1-1-A Protective Coatings and Organic Materials (Deleted)

None.

6.1.4 References

None.

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	Div 2, Subsection CC	Plate	Carbon Steel	See Subsection 3.8.1.6.4
	Div 2, Subsection CC	Plate	Stainless Steel	See Subsection 3.8.1.6.4
Penetrations	Div 1, Subsection NE	Plate	Carbon Steel	See Subsection 3.8.2.6
	Div 1, Subsection NE	Pipe	Carbon Steel	See Subsection 3.8.2.6
GDCS and Suppression Pool Liner	Div 2, Subsection CC	Plate	Stainless Steel	See Subsection 3.8.1.6.4 and Subsection 3.8.3.6.5
(Deleted)				
Drywell Head, Personnel Lock, Equipment Hatch	Div 1, Subsection NE	See Subsection 3.8.2.6	See Subsection 3.8.2.6	See Subsection 3.8.2.6
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 and associated piping that are part of the containment pressure boundary	Div 1, Subsection NE	Forging	Stainless Steel	SA-182 Gr F304L
		Tube	Stainless Steel	SA-312 Gr XM-19
		Pipe	Stainless Steel	SA-312 Gr TP304L
Piping (in drywell)	Div 1, Subsection NC	Pipe	Stainless Steel	SA-312 Gr TP304L
Flanges	Div 1, Subsection NC	Forging	Stainless Steel	SA-182 Gr F304L
Nuts and Bolts	Div 1, Subsection NC	Bar	Stainless Steel	SA-194 Gr 8, SA-193 Gr B8

Table 6.1-1

Containment System Including PCCS, and ECCS Component Materials

Component	Applicable ASME Code Section III,	Form	Material	Specification (ASTM/ASME)
ADS				
DPV Body	Div 1, Subsection NB	See Table 5.2-4		
Safety Relief Valve (SRV) Body	Div 1, Subsection NB	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[1]
GDCS				
Check valve and downstream piping	Div 1, Subsection NB	See Table 5.2-4		
Piping-upstream of check valve	Div 1, Subsection NC	Pipe	Stainless Steel	SA-376 Gr TP304L or TP316L[1] SA-312 Gr TP304L or TP316L[1] SA-358 Gr TP304L or TP316L[1]
Fittings	Same as mating pipe	Forging	Stainless Steel	SA-182 Gr F304L or F316L[1] SA-403 WP 304L or WP 316L[1]
Flanges	Same as mating pipe	Forging	Stainless Steel	SA-182 Gr F304L or F316L[1]
Valves				
Gate, Squib, Check	Div 1, Subsection NB	See Table 5.2-4		
ICS				
Condenser	Div 1, Subsection NC	Pipe	Nickel Base Alloy	SB-167 per ASME Code Case N-580-1

Table 6.1-1

Containment System Including PCCS, and ECCS Component Materials

Component	Applicable ASME Code Section III,	Form	Material	Specification (ASTM/ASME)
		Header	Nickel Base Alloy	SB-564 per ASME Code Case N-580-1
Steam Piping	Div 1, Subsection NB	See Table 5.2-4		
Condensate Piping	Div 1, Subsection NB	See Table 5.2-4		
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
Injection valve and downstream piping	Div 1, Subsection NB	See Table 5.2-4		
Piping- upstream of injection valve	Div 1, Subsection NC	Pipe	Stainless Steel	SA-312 Gr TP316L[1]
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
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 E7018 ER80S-D2 ER70S-2

Table 6.1-1

Containment System Including PCCS, and ECCS Component Materials

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

[1] 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 NUREG-0800, 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 DW and 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 Design Basis Accident (DBA) is defined as the worst pipe break, which leads to maximum DW and WW pressure or temperature, and is postulated to occur simultaneously with loss of preferred power. For structural integrity evaluation, Safe Shutdown Earthquake (SSE) loads are combined with LOCA loads.

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

- The containment structure is designed to accommodate the maximum internal negative pressure difference between DW and WW, and the maximum external negative pressure difference relative to the reactor building (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 offsite doses greater than those set forth in 10 CFR 52.47.

- 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 through the PCCS condensers.
- 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, which is treated as a mechanical component, is an all steel, permanent installation with primary and secondary seals that are fabricated from stainless steel for corrosion resistance. The refueling bellows extends from the lower flange of the reactor vessel to the interior of the reactor cavity, and provides a 360° structural barrier to prevent leakage from the reactor cavity into the DW. This extension is also not part of the containment boundary, however, it provides a Seismic Category I seal between the upper DW 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.

Refueling Cavity Bellows Seal

As shown in Figure 6.2-35, the RPV is fabricated to include a refueling bellows skirt, and the refueling bellows is installed as a module that is welded to the lower horizontal flange of the RPV refueling bellows skirt. The connections to the drywell bulkhead are also fully welded using AWS standards. The final assembly contains welded connections that provide a permanent barrier to leakage and can only leak if there is a through-wall defect. A spring-loaded secondary seal is provided to prevent leakage into the drywell in the event a leak occurs through the primary seal. The bellows structure is vertically oriented with a corrugated configuration that is designed to be flexible under differential thermal expansion and seismic motion. The bellows material meets ASME/ASTM standards, and bellows examinations are in accordance with Section III of the ASME BPV Code. The bellows assembly is located below the RPV flange such that it cannot interfere with the removal of core components for refueling. Normally open leak detection connections are located on the dry side of the bellows for continuous leakage monitoring.

Cover plates are provided for the bellows to protect against objects (e.g., a fuel assembly) that may be dropped during refueling. The cover plates remain in place during operation, but are designed to be readily removable. In addition, the bellows assembly has a plate over the vertically oriented bellows, which provides further protection in the event the cover plates are removed for cleaning or inspection. Routine maintenance and inspection of the bellows is performed based on supplier recommendations.

As the bellows configuration is not susceptible to dropped objects, any seal failures would likely be related to normal wear or fatigue. All bellows materials are designed to have a 60-year design life. For shop welds, all welding procedures and welder qualifications are in accordance with Section IX of the ASME Code, with any exceptions approved prior to execution. Any leakage due to fatigue or weld failure would not result in a rapid drain down event and would be easily detectible and isolable. As mentioned above, leak detection is provided on the dry side of the bellows, and a second seal is in place that prevents leakage into the drywell in the event of bellows leakage.

Pool level on the refuel floor is constantly monitored, and alarms are provided in the event level drops. The buffer pool is equipped with safety-related water level sensors that alarm the operators if the pool level is below normal. Operators also have the ability to provide makeup water and suspend refueling operations, as needed. In addition to pool level alarms, the drywell sump provides alarms in the event excess water is present.

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 (see Figure 6.2-3 for pool arrangement) 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 DW pressure in Table 6.2-5 is below the design value shown in Table 6.2-1. The structure stresses are evaluated in Section 3G.5 considering the DW fluid temperature transients for multiple break locations.

Three vacuum breakers are provided between the DW and WW. The vacuum breaker is a process-actuated valve, similar to a check valve (Figure 6.2-28). 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. One out of the three vacuum breakers is required to perform vacuum relief function. The third vacuum breaker provides redundancy while the second vacuum breaker provides single failure protection for opening. On the upstream side of each vacuum breaker, pneumatically operated fail-as-is safety-related isolation valves are provided to isolate a leaking (not fully closed) or stuck open vacuum breaker. 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 fully close as detected by the proximity sensors, a control signal closes the upstream isolation valve to prevent bypass leakage through the vacuum breaker and therefore maintain the pressure suppression capability of the containment.

In addition to the proximity sensors, there are temperature sensors located on and in the vacuum breaker/vacuum breaker isolation valve assembly. See Figure 6.2-28 for approximate temperature sensor locations and sensor terminology. These sensors will detect a rise in temperature between the vacuum breaker and the end of the penetration on the wetwell side due to the hot DW gas leaking past a not fully closed vacuum breaker. When the difference between the cavity temperature, T_{cavity}, and the wetwell temperature, T_{ww}, exceeds a fraction of the difference between the drywell temperature, $T_{dw(1)}$, and wetwell temperature, T_{ww}, a signal is sent to the vacuum breaker isolation valve to close. The bulk drywell temperature, $T_{dw(2)}$, (Figure 6.2-28) is measured separately from the vacuum breaker/vacuum breaker isolation valve assembly and is used to detect LOCA conditions and acts as a permissive to allow the vacuum breaker isolation valve logic to function.

The corresponding bypass leakage area that the temperature sensors will detect to close a vacuum breaker isolation valve is a maximum analytical limit of 0.6 cm^2 ($A/\sqrt{K}$). Closing each vacuum breaker isolation valve at this bypass leakage assures the analytical limit of 2 cm^2 ($A/\sqrt{K}$) of total bypass leakage will not be exceeded in the unlikely scenario of three vacuum breakers not fully closing. This scenario assumes more than one single failure will occur which is beyond design basis accident requirements.

Each vacuum breaker isolation valve logic subsystem is located in physically separate divisional rooms or compartments that have appropriate fire barriers between them. The isolation valve can also be manually opened or closed. For more discussion on the logic control of the vacuum breaker isolation valves, see Subsection 7.3.6. The design WW-to-DW pressure difference and the vacuum breaker opening differential pressure are given in Table 6.2-1.

The vacuum breaker and vacuum breaker isolation valves are protected from pool swell loads by structural shielding/debris screen designed for pool swell loads determined based on the Mark II/III containment design. Both valves are located in the DW and connected to the WW gas space by a penetration through the diaphragm floor. The structural shielding/debris screen is located in the WW gas space at the inlet side of the penetration.

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) that are an integral part of the containment boundary located in large pools of water outside the containment 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 integral part 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 6.2.5.2.

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 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 5.2-3 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 5.2-3 and 6.2-5) connects the DW to the suppression pool.

The spillover function provides DW to WW connection for limiting suppression pool drawdown and the holdup volume in the DW by transferring water from the DW annulus to the suppression pool. Spillover is accomplished by twelve horizontal holes (200 mm nominal diameter (8 inch nominal diameter)) which are built into the vent wall connecting the drywell annulus with each vertical vent module. Each spillover hole is horizontally oriented with an elevation as shown in Figure 3G.1-57. If water, ascending through the DW annulus following a postulated LOCA, reaches the spillover holes, it flows 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 Reference 3B-1.

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, two nonsafety-related systems are available to provide long-term containment cooling. These systems are FAPCS and Reactor Water Cleanup (RWCU). Heat is removed via either the FAPCS or RWCU nonregenerative heat exchanger(s) to the Reactor Component Cooling Water System (RCCWS) and finally to the Plant Service Water System (PSWS). This FAPCS function is described in Subsection 9.1.3, while for RWCU, this function is described in Subsection 5.4.8. The nonregenerative heat exchanger capacity used for the containment pressure response analysis, is specified in Table 5.4-3.

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. Scaling analyses documented in Reference 6.2-9 show that the sub-scale integral test facilities, i. e., GIRAFFE/He and PANDA, adequately simulate the phenomena important to the post-LOCA long-term cooling of the ESBWR containment.

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. Specifically, the initial feedwater temperature is varied from 160ºC (320ºF) to 252.2ºC (486ºF) for the bounding Feedwater Line Break (FWLB) and Main Steam Line Break (MSLB) analyses with failure of one SRV. The calculations are run for 72 hours, and the maximum DW pressure remains below the design pressure as shown in Table 6.2-1 with adequate margin, similar to those shown in Table 6.2-5. The variation of the maximum DW pressure is small with respect to the initial feedwater temperature.

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). Nuclear Regulatory Commission (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, 6.2-4, and 6.2-11. 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.

The expected operating range of drywell temperature is from 46.1°C (115°F) to 57.2°C (135°F). Cooler initial temperature represents more initial inventory for the noncondensable gases, and consequently higher long-term containment pressure. Therefore, the analyses were performed at 46.1°C (115°F) as given in Table 6.2-6 to ensure conservative peak drywell pressure.

The lower bound on the relative humidity in the drywell is 20% as given in Table 6.2-6. The lower bound value was selected for the analyses, because a lower initial drywell relative humidity results in more noncondensable gases available to be transferred to the wetwell and higher containment pressures following the LOCA.

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 6.3-2 provide the performance parameters of the related ESF systems, which supplement the design conditions of Table 6.2-1, for containment performance evaluation. Table 15.2-23 provides the response time limits for initiation signals used/assumed in accident analyses.

ESBWR Core Decay Heat

The ESBWR core decay heat is generated based on the American National Standards Institute (ANSI)/American Nuclear Society (ANS)-5.1-1994 standard with additional terms to account for a more complete shutdown power assessment. The heat sources considered include decay heat from fission products, actinides and activation products; as well as fission power from delayed and prompt neutrons immediately after shutdown. The effect of neutron capture in fission products is considered. However, initial energy stored in the fuel assembly and heat from metal-water reaction during severe accident are not considered in the decay heat calculations.

The input parameters for the ESBWR decay heat calculation are derived based on the equilibrium core design. Additional safety margins are added to the core parameters in order to bound future cycle variations as well as other fuel product lines with similar parameters. The fuel type assumed in the ESBWR decay heat calculation is GE14E. A constant power irradiation for 3.8 years to reach end-of-cycle exposure of 32 GWd/ST (35.3 GWd/MT) is assumed prior to shutdown. The shutdown mode assumed in the ESBWR decay heat is consistent with a design-basis large break LOCA.

The decay heat modeling for the break LOCA events discussed in this chapter, including the listed fission power after shutdown or decay heat values is consistent, adequate and applicable to the entire LOCA break spectrum due to conservatisms included in its application (Figure 6.2-8c). These values represent the core average decay heat at the end-of-cycle.

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. 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. Additional TRACG outputs for the limiting break base case (main steam line break), e.g., the transient air mass profiles in different regions such as the gravity driven cooling system, DW head, and WW airspaces were generated. Also, additional parametric cases were performed to evaluate the impact of various model/plant parameters on the long-term containment pressure. The results of these additional outputs and parametric analyses are detailed in Appendix 6H.

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/PCCS pool, expansion pools, and the Equipment 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. Appendix 6D provides a detailed description of the passive heat sinks within containment as per RG 1.70. Appendix 6A, Figure 6A-1 shows the TRACG nodalization of the DW/WW walls as passive heat sinks.

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, DW and GDCS airspace pressure responses and PCCS heat removal for this analysis. Table 6.2-5 summarizes the results of this calculation. The calculated maximum DW 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 Subsection 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 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, DW and GDCS airspace pressure responses and PCCS heat removal for this analysis. Table 6.2-5 summarizes the results of this calculation. The calculated maximum DW 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, DW and GDCS airspace pressure responses and PCCS heat removal 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 DW 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 the Feedwater Line Break with failure of one Depressurization Valve (DPV), and Table 6.2-7f shows the sequence of events for the Feedwater Line Break with failure of one SRV. 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 6.2-13d6 and Figures 6.2-13e1 through 6.2-13h6 show the pressure, temperature, water level, DW and GDCS airspace pressure responses and PCCS heat removal for this analysis. Table 6.2-5 summarizes the results of this calculation. The calculated maximum DW pressure during the 72 hours following a LOCA for the bounding case is below the containment design pressure. The detailed discussion on the chronology of progressions of the Feedwater Line Break Bounding cases are given in Appendices 6E.1 and 6E.3.

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. 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. Table 6.2-7e shows the sequence of events for the Main Steam Line Break with failure of one DPV, and Table 6.2-7g shows the sequence of events for the Main Steam Line Break with failure of one SRV. A comparison of these two tables and the corresponding figures shows that these two cases are nearly the same in transient responses and in the calculated DW pressure. Both cases can be considered as the limiting event. These cases are discussed in the following paragraph.

Figures 6.2-14a1 through 6.2-14d3 and Figures 6.2-14f1 through 6.2-14i3 show the pressure, temperature, DW and GDCS airspace pressure responses and PCCS heat removal for these analyses. Table 6.2-5 summarizes the results of this calculation. The calculated maximum DW pressures during the 72 hours following a LOCA for these bounding cases are below the containment design pressure. The detailed discussion on the chronology of progressions of the Main Steam Line Break Bounding cases are given in Appendices 6E.2 and 6E.4.

A loss of all power generation buses is not the limiting assumption and the effects of continued feedwater injection is more limiting, as it can potentially add water to the wetwell and compress the wetwell air space. The ESBWR design incorporates features that mitigate this challenge by isolating reactor inventory sources outside of containment and provides a method of GDCS initiation based on LOCA condition detection. These features ensure that containment remains within design pressure for the entire 72-hour event duration. These features also ensure acceptable performance for the full spectrum of LOCA events within containment, with or without the assumption of loss of external injection capability. Additionally, although power generation buses are considered available to add feedwater or High Pressure Control Rod Drive (HP CRD) injection, no credit is given for heat removal systems powered by these buses. Table 6.2-7h shows the sequence of events for the Main Steam Line Break with failure of one SRV and with offsite power available. Figures 6.2-14j1 through 6.2-14m3 show the pressure, temperature, DW and GDCS airspace pressure responses and PCCS heat removal for this analysis. The noncondensable mass and the void fraction in the DW and GDCS are presented in Figures 6.2-14n1 through 6.2-14o3. The detailed discussion on the chronology of progression is given in Appendix 6E.5. The cases analyzed without offsite power and water addition assume higher initial pressure, and result in higher pressure as shown in Table 6.2-5. The highest value of Maximum DW Pressure in Table 6.2-5 is the calculated peak containment internal pressure for the design basis loss of coolant accident.

6.2.1.1.3.5.1 Post-LOCA Containment Cooling and Recovery Analysis

For post-LOCA containment cooling and recovery, Main Steam Line Break scenarios selected are one SRV failure and one DPV failure. The analysis with PARs and 4 of the 6 PCCS vent fans uses the failure with one SRV and the analysis with RWCU/SDC in suppression pool cooling mode followed by shutdown cooling mode uses the failure with one DPV. The post 72 hour analysis results are not sensitive to the event selection (failure of one DPV versus one SRV) due to the fact that these two cases are nearly the same in transient responses up to 72 hours and the containment pressure and temperature are rapidly reduced upon the activation of the nonsafety-related Structure, System, or Components (SSC).

After the first 72 hours of the accident, the following nonsafety-related SSCs are utilized to keep the reactor at safe stable shutdown conditions, to rapidly reduce containment pressure and temperature to a level where there is acceptable margin, and then to maintain these conditions indefinitely:

(1) SSCs to refill the IC/PCCS pools;

(2) PCCS Vent Fans;

(3) Passive Autocatalytic Recombiner System (PARS); and

(4) Power supplies to the PCCS Vent Fans and the IC/PCCS pool refill pumps.

Once a state of safe, stable reactor shutdown is reached, containment pressure and temperature are maintained with sufficient margin to containment design limits for a long period of time. Figure 6.2-14e1 through Figure 6.2-14e10a show key parameters for the long term pressure reduction and maintenance phase. PARS function at 72 hrs and 4 of 6 PCCS vent fans are credited in the calculation.

The containment pressure is reduced and is maintained at a reduced pressure after the 72 hour peak. Other non-safety related, non-Regulatory Treatment of Non-safety Systems (RTNSS) SSCs can be placed in service to bring the reactor to cold shutdown conditions and to further reduce the containment pressure and temperature. These SSCs include the FAPCS as the preferred method, and the RWCU/Shutdown Cooling (SDC) system in the unlikely event there is fuel damage (Subsections 9.1.3 and 5.4.8, respectively). The RWCU/SDC and the FAPCS system are not part of the primary success path for post-LOCA containment cooling. Calculations of RWCU/SDC performance are provided here to show its ability to cooldown the reactor and containment. In the unlikely event of fuel damage, where the RWCU/SDC system is used, the Reactor Building HVAC Accident Exhaust Filter Units are a required support system for limiting onsite and offsite dose.

Containment pressure and temperature responses which represent a postulated accident recovery evolution, with RWCU/SDC (fuel damage assumed) providing the cold shutdown function are shown in Figures 6.2-14e11 and 6.2-14e12. These response curves are based on the RWCU/SDC operating in suppression pool cooling mode for 24 hours, beginning seven days after a LOCA, followed by vessel injection via the normal RWCU/SDC midvessel suction line, with suction from the suppression pool. The heat removal for this mode of RWCU/SDC operation is provided by the non-regenerative heat exchanger (NRHX). A conservative heat exchanger capacity was assumed which is well within the capability of the RWCU/SDC NRHX. Table 6.2-48 lists the RWCU/SDC NRHX data used in the analysis. There is no requirement to start the recovery actions at seven days, since the reactor is already in a safe stable shutdown condition, and containment pressure and temperature are in a non-upward trending state, with sufficient margin to containment design limits.

The accident recovery analysis shows that after being in suppression pool cooling for 24 hours and then injecting into the reactor vessel for approximately 10 hours, the suppression pool has equilibrated with the reactor bulk water temperature at cold shutdown conditions.

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 (for example, GDCS) 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. Among the four design basis LOCA break types analyzed, a FWLB results in the highest peak DW pressure and a MSLB results in the lowest peak DW pressure during the initial 2000 seconds after the break. The peak pressure of a GDCS Injection Line or a Bottom Drain Line (BDL) break falls between those of the Feed Water Line (FWL) and MSLBs. DW temperatures for these four break types differ by less than 20ºC after 800 seconds. It is therefore adequate to analyze FWL and MSLB scenarios to provide diverse DW environmental conditions, which envelope other break locations, for determining the minimum DW pressure and the pressure differential between WW and DW as consequence of inadvertent initiation of DW spray. The results of the MSLB analysis show that the containment does not reach negative pressure relative to the RB, and the maximum WW-DW differential pressure is within the design capability. This calculation credits one of the three WW-DW vacuum breakers. Each vacuum breaker has an area of 9.67×10^{-2} m^2, (1.04 ft^2). An evaluation of the effect of DW spray on containment integrity for a main steam line break and a feedwater line break was performed to determine the maximum negative differential pressures (DW to WW, and DW to reactor building). This evaluation assumed that a DW spray flow rate of 127 m^3/hr (560 gpm) at a temperature of 293ºK (67.7ºF) is initiated at the worst possible moment for a DW spray, at the point in time when there is low inert gas content in the DW relative to the WW (i.e., when the DW pressure has peaked just prior to the opening of the DW-WW 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 DW spray piping system, a sized, flow restricting orifice has been established (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 DW and the suppression pool (WW) gas space, the leaking steam would produce undesirable pressurization of the containment. The design basis accident calculations assume a suppression pool bypass leakage of 2 cm^2 (2.16E-03 ft^2), ($A/\sqrt{K}$). Table 6.2-5 shows that this results in acceptable containment pressures. Additional calculations with the suppression pool bypass leakage assumption of 1 cm^2 (1.08E-03 ft^2), ($A/\sqrt{K}$) are presented in Appendix 6I.

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 Subsection 6.2.1.1.C) for automatic actuation of sprays, without the use of a containment spray system. The SRP states that the WW spray should be automatically actuated 10 minutes following a LOCA signal and an indication of pressurization of the WW 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 Subsection 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 Boiling Water Reactor (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 according to Technical Specification requirements.

6.2.1.1.5.4 Bypass Leakage Tests and Surveillance

There are provisions for leakage tests and surveillances to determine suppression pool bypass leakage, and to ensure that leakage does not substantially increase over the plant life.

After the pre-operational suppression pool bypass tests are performed, a periodic test is performed at a frequency of 24 months. Pre-operational and periodic local leakage rate testing of the vacuum breakers and vacuum breaker isolation valves are also performed. These tests and surveillances quantify, measure, or detect any degradation, and provide assurance that the suppression pool bypass leakage is maintained within the allowable value between tests.

6.2.1.1.5.4.1 (Deleted)

6.2.1.1.5.4.2 Local Leak Rate Testing of Drywell to Wetwell

Pre-operational and periodic visual examinations of the DW to WW penetrations are performed in accordance with inservice inspection or inservice testing requirements. A suppression pool bypass test is performed to detect leakage from the DW to the WW at a frequency of 24 months. Local leakage rate tests of the individual vacuum breakers and vacuum breaker isolation valves are performed according to Technical Specification requirements. 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 DW/WW 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 DW/WW 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 acceptance criteria for the suppression pool bypass test is a calculated bypass leakage area ($A/\sqrt{K}$) that is less than 50% of the bounding design basis accident allowable bypass area, which is 2.0 cm^2 (2.16E-03 ft^2), ($A/\sqrt{K}$). The calculated bypass leakage area is calculated at the upper 95% confidence level to account for instrument inaccuracies and uncertainties. The acceptance criteria for the individual vacuum breaker and vacuum breaker isolation valve local leakage rate tests is less than or equal to 15% of the bounding design basis accident allowable bypass area, and the acceptance criteria for the total leakage of all three vacuum breaker/vacuum breaker isolation valve pairs on a maximum pathway basis is less than or equal to 35% of the bounding design basis accident allowable bypass area.

Local leakage rate measurement inaccuracies are typically less than or equal to 2% of the full scale flow range. This value is insignificant when compared to the margins between the local leakage rate acceptance criteria and the leakage equivalent to the bounding design basis accident allowable bypass area.

6.2.1.1.5.4.4 Surveillance Test

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

6.2.1.1.5.5 Vacuum Breaker Valve and Isolation Valve Instrumentation and Tests

6.2.1.1.5.5.1 Position Indicators, Temperature Sensors, 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 breaker position indicator system has adequate sensitivity to detect an open vacuum breaker.

Redundant temperature sensors are placed within the cavity created by the vacuum breaker and vacuum breaker isolation valve assembly, and in close proximity to the vacuum breaker outlets. The temperature sensor system is designed to provide detection of a leaking vacuum breaker during a LOCA.

6.2.1.1.5.5.2 Vacuum Breaker Valves and Isolation Valves Operability Tests

The vacuum breakers are tested for free movement according to Technical Specification requirements. Vacuum breaker isolation valves are tested as specified in Table 3.9-8.

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 Section 3.8. The containment and internal structures are designed for these loads within the acceptance criteria specified in Section 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, see, Section 7.5.

6.2.1.1.10 Severe Accident Conditions

Severe Accident 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 severe accidents.

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) ICS

The isolation condensers support both reactor water level and pressure control and are the first defense against a severe accident. The ESBWR is equipped with four isolation condensers, which conserve RPV inventory in the event of RPV isolation. Basically, the isolation condensers take steam from the RPV and return condensate back to the RPV. The isolation condensers begin operation when the condensate lines open automatically on diverse signals including RPV level dropping to Level 2. After operation begins, the isolation condensers are capable of keeping the RPV level above the setpoint for ADS actuation. The design mitigates noncondensable buildup in the isolation condensers (that can impair heat removal capacity) by temporarily opening a small vent line connecting the isolation condensers 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 isolation condensers to prevent containment pressurization and containment pressure control function is served by the PCCS.

(2) ADS

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 DW. The resulting heatup of the upper DW atmosphere could overpressurize the containment or cause over temperature failure of the DW 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) PCCS Heat Exchangers

The basic design of the ESBWR ensures that any fission products from fuel damage following a severe accident are not released outside the plant. One such removal mechanism is the PCCS heat exchanger tubes. These tubes act like a filter for the aerosols. They essentially “filter out” any aerosols that are transported into the PCCS units along with the steam and noncondensable gas flow. Aerosols that are not retained, in the DW or the PCCS heat exchangers, get transported via the PCCS vent line to the suppression pool where they are efficiently scrubbed.

The PCCS 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 DW has been maximized to improve the potential for ex-vessel debris cooling. There is a drain sump incorporated into the lower DW floor intended to prevent water buildup on the floor. The location of the sump has been maximized to place it as far away from the RPV as possible. The sump has channels at floor level to allow water to flow into the sump. The channels are long enough that any molten debris from a severe accident will solidify before it exits the channels and reaches the sump.

(6) Manual Containment Overpressure Protection Subsystem

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. The vent path used to preclude containment overpressure failure is constructed to maintain structural integrity when exposed to the containment pressures present during severe accident conditions. The vent path incorporates manually operated isolation valves located outside of the Reactor Building.

(7) Deluge Lines Flooder System

The lower DW 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 DW floor temperature. The system consists of multiple lines that connect each of the GDCS water pools to the lower DW. The volume of water in the GDCS pools is capable of flooding the RPV and lower DW 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 DW floor to cool the ex-vessel core-melt debris from top and bottom sides. By flooding the lower DW 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) PCCS

The PCCS is designed to remove decay heat from the containment. The PCCS heat exchangers receive a steam-gas mixture from the DW atmosphere, condense the steam and return the condensate to the RPV via GDCS pools. The noncondensable gas is drawn to the suppression pool through a submerged vent line driven by the differential pressure between the DW and WW.

(9) Suppression Pool and Airspace

The suppression chamber is a large chamber with communication to the DW 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 DW 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 DW. 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.

(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.

(12) GDCS Pool Spillover Pipes

During a severe accident assuming the GDCS injection lines and equalizing lines fail to open, the PCCS condensate causes the GDCS pools to overflow. GDCS pool spillover pipes direct this overflow to the vertical vents and subsequently into the suppression pool. This prevents the flooding of the lower DW floor before the introduction of any core-melt debris. The GDCS pool spillover pipes are connected to the GDCS pools above the high water level and below the top of the pool walls. The piping is seismic Category II, and the pipes are sized to provide enough capacity to prevent any overflow.

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

- PCCS pool and heat exchangers provide passive containment cooling;
- ICS pool and heat exchangers provide natural circulation decay heat removal from RPV;
- GDCS (three pools, four divisions) with ADS (DPV, SRV) makes up the ECCS; GDCS deluge lines supply BiMAC for long-term coolability;
- Manual Containment Overpressure Protection Subsystem provides manual venting from the WW in a controlled manner; and
- 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.
- A factor of 1.2 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. The Reactor Shield Annulus subcompartment vent areas are always open. Insulation does not cause an impediment to venting and there are no blowout panels in this subcompartment.

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 RSW is designed for transient pressure loading conditions from the worst high energy line rupture inside the annulus region. For pipe break cases in this subcompartment, no credit was taken in the analysis to limit the break area due to presence of pipe restraints.

6.2.1.2.3 Design Evaluation

FW 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 (212°F) when the postulated pipe break occurs. The mass release rates are determined with Moody's Critical Flow Model for Homogenous Equilibrium Mixture (Reference 6.2-8). The subcompartment pressure responses were analyzed with TRACG (Reference 6.2-11). Prior to the time of peak pressure the vent flow is subsonic. The integrity of RSW is discussed in DCD Subsection 3G.1.5.4.2.3.

The break locations have been selected to maximize the mass and energy release into the subcompartment. Since instantaneous double-end guillotine breaks were postulated for all pipe breaks, Leak-Before-Break was not used to limit the break area. The mass and energy release rates are held constant for the analyses. For a feedwater line break, the critical flux is 9.389 x

10^4 kg/(s · m²), (19230 lb/(s · ft²)) from either end of the guillotine break, and the total mass release rate from both the RPV end and the reactor shield wall is 2854 kg/s, (6292 lb/s). For a RWCU line break, the critical flux is 4.868 x 10^4 kg/(s · m²), (9970 lb/(s · ft²)) from either end of the guillotine break, and the total mass release rate from both the RPV end and the reactor shield wall is 2395 kg/s, (5280 lb/s). Analyzed with TRACG, the peak subcompartment pressure responses were found to be below the design pressure for all postulated pipe break accidents.

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

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

- GDC 50, as it relates to the containment being designed with sufficient margin, requires that the containment and its associated systems can accommodate, without exceeding the design leakage rate and the containment design, 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 (Table 6.2-12d, Figures 6.2-9e1, 6.2-9e2, 6.2-10e1, and 6.2-10e2) 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. To maximize the energy release to the containment during the blowdown and reflood phases of a LOCA, the following conservative assumptions are used in the analyses.
    - All non-wall heat structures inside the DW and WW are conservatively ignored in the analyses.
    - The DW basemat and the top DW top slab (horizontal heat slabs) are expected to see some steam condensation during the early part of the LOCA. These horizontal heat slabs are conservatively ignored in the analyses.
    - The suppression pool basemat and the GDCS pool basemat are covered with water. The heat sink effect of these horizontal heat slabs is conservatively ignored in the analyses.
  - 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 four double-ended guillotine pipe break sizes and locations to assure that the worst case has been identified. These calculations are described in Appendix 6F.
- 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 NRC-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 are visually inspected at each refueling outage to determine and assure that they are free of foreign debris, and the valve disk is 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, are 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:

- DW temperature;
- DW pressure;
- Differential pressure from DW-to-WW and DW-to-RB;
- DW oxygen and hydrogen concentrations;
- DW radiation levels;
- WW temperature;
- WW pressure;
- Differential pressure between the WW and RB;
- WW oxygen and hydrogen concentrations;
- WW radiation levels;
- Suppression pool temperature;
- Suppression pool level;
- GDCS pools water level;
- Water level in DW;
- DW and WW 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 DW and WW as part of the Containment Monitoring System (CMS) 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 DW and WW. Each analyzer draws a sample from an appropriate area of the DW or WW. High oxygen and hydrogen concentration levels are recorded and alarmed in the control room.

Radiation detectors in the DW and WW 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 DW 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, 40 discussed in SRP 6.2.2, R4, 10 CFR 50.46(b)(5), 10 CFR 52.47(a)(2)(iv), and GDC 19 of 10 CFR 50 Appendix A. The plant meets the following containment cooling requirements.

- GDC 38 as it relates to:
  - The 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 (LOOP), the system function can be accomplished assuming a single failure; and
  - Leak detection, isolation and containment capabilities being incorporated in the design of the PCCS.
- GDC 39, as the PCCS is designed to permit periodic inspection of components.
- GDC 40, as the PCCS is designed to permit periodic testing to assure system integrity, and operability of the system and its active components.
- 10 CFR 50.46(b)(5), as the PCCS is designed to provide long term cooling following deflagrations or detonations within PCCS from hydrogen accumulation.
- 10 CFR 52.47(a)(2)(iv), and GDC 19 of 10 CFR 50 Appendix A, as the PCCS is designed to maintain containment pressure boundary following deflagrations or detonations within PCCS from hydrogen accumulation.

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 IC/PCCS pools, equipment 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 in the first 72 hours. 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 an integral part of the containment pressure boundary. Therefore, ASME Code Section III Class MC, Seismic Category I, and Tubular Exchanger Manufacturers Association 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 DW and the WW, which are parts of the ESBWR pressure suppression type containment system. The DW 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 closes.

The PCCS condenser, receives a steam-gas mixture supply directly from the DW. 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 condensers do 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 in the first 72 hours after a DBA, 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 sets of two steam condenser modules (Passive Containment Cooling Condensers), as shown Figure 6.2-16. Each PCCS condenser 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. The Equipment Storage pool and Reactor Well are designed to have sufficient water volume to provide makeup water to the IC/PCCS pools for at least the initial 72 hours after a LOCA without makeup. The Equipment Storage pool and Reactor Well are connected to Isolation Condenser/Passive Containment Cooling System (IC/PCCS) pools via pool cross-connect valves (see Figure 6.2-2), which open upon low level in IC/PCCS inner expansion pool. The PCCS relies on the water in the Equipment Storage pool and Reactor Well to perform its safety-related function for the first 72 hours of a DBA. The pool cross-connect valves reside within the ICS described in Subsections 5.4.6, 7.4.4, and 7.5.5. Long-term effectiveness of the PCCS (beyond 72 hours) credits pool makeup and an active gas recirculation system, which uses in-line fans to pull DW gas through the PCCS condensers.

The PCCS condensers are located in a large pool (IC/PCCS pool) positioned above the ESBWR DW.

Each PCCS condenser is configured as follows (Figures 3G.1-71a and 3G.1-71b).

A central steam supply pipe is provided which is open to the DW airspace at its lower end. The open end of this pipe is provided with a debris filter with holes no greater than 25 mm (1 inch). The maximum inlet velocity during a LOCA is estimated to be no greater than 106 m/s (348 ft/s). The steam supply 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 through the DW through a single passage per condenser module as shown on the figures.

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 vent line goes to the suppression pool and is submerged below the water level.

When the drywell pressure is higher than the combined wetwell pressure and vent line submergence, noncondensable gases vent to the suppression pool. When the drywell pressure is equal to or lower than the combined wetwell pressure and vent line submergence, noncondensable gases including hydrogen and oxygen (created by radiolytic decomposition in the core) accumulate in the lower drum of the PCCS condenser thereby producing a potentially flammable/detonable mixture. As such, the PCCS condensers are designed to withstand the overpressure and dynamic effects produced by deflagrations or detonations of these mixtures. Reference 6.2-14 provides details regarding the stress analysis of the condenser and supports.

A Passive Containment Cooling vent fan is teed off of each PCCS vent line and exhausts to the GDCS pool. The fan aids in the long-term removal of noncondensable gas from the PCCS for continued condenser efficiency. The minimum fan performance requirements are shown in Table 6.2-49. The fans are operated by operator action and are powered by a reliable power source which has a diesel generator backed up by an ancillary diesel, if necessary, without the need to enter the primary containment. The discharge of each PCCS vent fan is submerged below the GDCS pool water level to prevent backflow that could otherwise interfere with the normal venting of the PCCS. The vent fan discharge line terminates in a drain pan within the GDCS pool so that the gas seal is maintained after the GDCS pool drains. The vent fan discharge line is 24 cm (9.4 in) below the top of the drain pan lip with a tolerance of 1.4 cm (0.6) in . To further prevent reverse flow through an idle fan, a check valve is installed downstream of the fan. Since the PCCS condensers and vent piping have the potential for containing hydrogen and oxygen, the vent fans are designed and constructed so as to not be ignition sources for combustion in accordance with NFPA 69 and AMCA 99-03.

The PCCS condensers receive a steam-gas mixture supply directly from the DW. The PCCS condensers are initially driven by the pressure difference created between the 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. In order to ensure the PCCS can maintain the DW to WW differential pressure to a limit less than the value that causes pressure relief through the horizontal vents, the vent line discharge point is set at an elevation submerged below low water level and at least 0.85 m (33.5 in) and no greater than 0.900 m (35.4 in) above the top of the uppermost horizontal vent. The PCCS condensers are an integral part of the safety-related containment and do not have isolation valves.

The drain line is submerged in the GDCS pool to prevent 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. The drain line terminates in the same drain pan as the vent fan discharge to replace any evaporation loss in the drain pan after the GDCS pool drains.

Each PCCS condenser is located in a subcompartment of the IC/PCCS 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 PCCS 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/PCCS pool water.

IC/PCCS 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 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/PCCS expansion pool.

Cooling and cleanup of IC/PCCS 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 condensers 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/PCCS 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 WW through the vent line.

The PCCS vent fan can be started to assist the natural venting action to remove noncondensable gases that could accumulate in the PCCS condensers. TRACG studies have shown that the PCCS meets its design function without the use of the PCC vent fan for at least 72 hours.

6.2.2.3 Design Evaluation

The PCCS condenser is an integral part 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 MC 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 (ICS) 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/PCCS pool.

The PCCS components located in a subcompartment of the safety-related IC/PCCS pool are protected by the IC/PCCS pool subcompartment from the effects of missiles tornados to comply with 10 CFR 50, Appendix A, Criteria 2 and 4.

The PCCS condenser cannot fail in a manner that damages the safety-related IC/PCCS 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 (Reference 6.2-10 Section 5.3, and Reference 6.2-11, Section 13), which is reported in Subsection 6.2.1.1 and Section 15.4.

6.2.2.4 Testing and Inspection Requirements

The PCCS is an integral part of the containment, and it is periodically pressure tested as part of overall containment pressure testing (Subsection 6.2.6). Also, the PCCS condensers can be isolated using spectacle flanges for individual pressure testing during maintenance.

PCCS condenser removal for routine inspection is not required.

Refer to Reference 6.2-14 for inspection requirement for the PCCS condenser.

6.2.2.5 Instrumentation Requirements

The PCCS does not have instrumentation. Control logic is not needed for it's functioning. There are no sensing and power actuated devices except for the vent fans. 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/PCCS 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 consists of rooms/compartments, which are served by one of the three ventilation subsystems; Contaminated Area Ventilation Subsystem (CONAVS), Refueling and Pool Area HVAC Subsystem (REPAVS), and Clean Area Ventilation Subsystem (CLAVS). None of these compartmentalized areas communicate with each other.

Under accident conditions, the RB (CONAVS and REPAVS areas) automatically isolate on high radiation to provide a hold up volume for fission products. When isolated, the RB (CONAVS and REPAVS areas) can be serviced by the RB HVAC On-Line Purge Exhaust Filter units (CONAVS and REPAVS areas) and the RB HVAC Accident Exhaust Filter units (CONAVS areas) (Subsection 9.4.6). No credit is taken for the filters in dose consequence analyses (Subsection 15.4.4). While the RB HVAC Accident Exhaust Filter units are a defense in depth feature, operation of the system for a period of 30 days following a design basis accident was evaluated using the design basis LOCA RADTRAD model described in chapter 15 (subsection 15.4.4) to ensure the design basis LOCA analysis results remain bounding. The RB HVAC Accident Exhaust Filter dose consequence analysis was conservative in that no credit was taken for CONAVS area drawdown while operating the RB HVAC Accident Exhaust Filter unit over the design flowrate range in parallel with the RB design exfiltration. Credit of cleanup of the CONAVS area from operation of RB HVAC Accident Exhaust Filter units (95% carbon filter efficiency) was assumed. The radiological consequences presented in Chapter 15 for the design basis LOCA are bounding as long as operation of the RB HVAC Accident Exhaust Filter units is delayed at least eight hours post accident. Operation of the filter units at an earlier post accident time requires site specific radiological consequence evaluation.

With low leakage and stagnant conditions, the basic mitigating function is the hold up of fission products in the RB CONAVS area itself. The ESBWR design does not include a secondary containment; however credit is taken for the existence of the RB CONAVS area surrounding the primary containment vessel in radiological analyses. CONAVS areas envelope all containment penetrations except penetration for main steam and feedwater lines located in the main steam tunnel. The radiological dose consequences for LOCAs, based on an assumed containment leak rate of 0.35% per day and RB CONAVS area leakage rate of 141.6 l/s (300 cfm), show that offsite and control room doses after an accident are less than allowable limits, as discussed in Table 15.4-9.

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 6.5.2.3. Therefore the design criterion of GDC 43 is not applicable.

The effect of RB leakage less than the maximum leak rate used in the accident dose calculations has the potential to increase the radiation dose inside the RB following a design basis accident. The evaluation of the increased radiation levels to equipment is addressed through the environmental qualification program and any increased hazards during post-accident RB re-entry are addressed by the emergency planning program through emergency operating procedures.

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 Reactor Building HVAC System (RBVS) subsystems (CONAVS and REPAVS) automatically isolates upon detection of high radiation levels in their respective 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. The radiological analyses assume the RB CONAVS served areas form this boundary.

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 CONAVS portion of the RBVS (Subsection 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 RB/Fuel Buidling (FB) stack radiation monitors. If the radioactivity level rises above set levels, the discharge can be routed through RB HVAC online purge Exhaust Filter Unit 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. The RBVS is designed with safety-related isolation dampers and tested for isolation under various accident conditions.

HELBs 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 RB is equipped with passively acting pressure relief devices that allow the refuel floor to vent to the environment if cooling is lost to the auxiliary pools during an outage. The devices open at a high-pressure set point equivalent to the full tornado pressure drop described in Section 3.3.2.2.

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 HELB in any RWCU compartment is directed through adjoining compartments and pipe chase to HELB blowout panels on the side of the RB (not connected to the 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. Figure 6.2-18a to Figure 6.2-18c show the mass and energy release 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/PCCS 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 (Figure 1.2-9) and into the RB operating area with portions of the steam directed through the pool compartments to the RB/FB 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 (TB). 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 TB. The steam/air mixture resulting from a main steam line break is directed to the TB 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 TB is open.

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 at 102% power and 187.8ºC (370ºF) feedwater temperature. 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 35.2 kPaG (5.11 psig) 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 mass and energy blowdown from the postulated broken pipe terminates when system isolation valves are fully closed after receiving the pertinent isolation closure signal.

A conservative RWCU model based on RELAP5/Mod3.3 has been developed to evaluate the mass and energy release for the five break locations. Total blowdown duration is based on the assumption that the isolation valve starts to close at 46 seconds (1 second instrument time plus 45 seconds built in time delay in blowdown differential flow detection logic) after the break and the isolation valve is fully closed in 15 seconds. Mass and energy blowdown data are shown in Figure 6.2-18a to Figure 6.2-18c.

After the initial inventory depletion period, the steady RPV blowdown is choked at the venturi located upstream of the isolation valve since the venturi flow area is smaller than the isolation valve flow area. After the isolation valve starts closing, as soon as the valve area becomes equal to the venturi flow area, the break flow is choked at the isolation valve. The break flow stops when the isolation valve is fully closed.

The narrative of the event described above is applicable to all five cases analyzed since the breaks are all located downstream of the isolation valve and the dynamics of the break responses are similar. Descriptions of the break locations and break sizes are provided in Table 6.2-11.

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," Reference 6.2-13.

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. The RB (CONAVS area) is capable of periodic testing to ensure that the leakage rates assumed in the radiological analysis are met as required under Technical Specification 3.6.3.1. RB exfiltration testing is a positive pressure test of the CONAVS volume confirming that the test leak criteria bounds the analytical limit derived in the dose modeling. A nominal ¼ inch w.g. differential pressure bounds the effects of worst-case wind loading applied across a face of the RB. Numerous pressure measurements are taken at designated areas and interconnecting doors and dampers are opened to ensure uniform pressure is established within the contaminated areas of the RB (CONAVS area). The RB exfiltration test pressure is maintained for sufficient period of time to ensure steady state conditions are established (approximately ½ hr to 1 hr). This RB exfiltration test leak rate acceptance criteria are adjusted base on the actual CONAVS area test differential pressure applied to ensure that the impact of test parameter uncertainties are minimized (flow instrument uncertainty, CONAVS area temperature and pressure gradients).

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. RG 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.

Certain systems require alternative containment isolation arrangements that are an exception to the above requirements. These exceptions are listed in Table 1.9-6 and are qualified on a case-by-case basis.

- 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 52.47(a)(2)(iv) 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. Exemptions from these GDCs are listed in Table 1.9-6.
- Isolation valves for instrument lines that penetrate the DW/containment conform to the requirements of RG 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 meet the requirements of the ASME Boiler and Pressure Vessel Code, Section III, Class 1 or 2, 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.
- Penetrations with trapped liquid volume between the isolation valves have adequate relief for thermally-induced pressurization.
- Piping penetrations through the containment (that is, penetration themselves and not the pipes) are designed to the requirements of subsection NE, (MC component) of Section III of the ASME Code.

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 RG 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 power-operated valves are designed to fail to the closed position for containment isolation upon loss of the operator gas supply or electrical power with the exception of the following lines that are fail as-is:

- Isolation Condenser System steam supply.
- Isolation Condenser System condensate return.
- Fuel and Auxiliary Pools Cooling System suppression pool suction.
- Fuel and Auxiliary Pools Cooling System suppression pool return.

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 6.2-15 through 6.2-45.

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. Tables 6.2-15 through 6.2-45 show the pertinent data for the containment isolation valves, except for excess flow check valves, which are discussed in Subsection 6.2.4.2.2. Containment isolation valves are located as close to the containment as practical. Sufficient space is provided between the valves and the containment boundary to permit the following:

- Inservice inspection of non-isolable welds;
- 10 CFR 50 Appendix J leak testing;
- Cutout and replacement of isolation valves using standard pipe fitting tools and equipment;
- Local control; and
- Valve seat resurfacing in place.

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 power-operated valves with geared or bi-directional actuators (motorized or fluid-powered) remain in their last position upon failure of valve power. All power-operated valvess 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 RG 1.26 may be found in Subsection 3.2.2. General compliance or alternate approach assessment for RG 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 6.2-15 through 6.2-45.

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 FAPCS penetrations. Those FAPCS penetrations required for emergency operation include remote manual isolation valves or check valves. In addition, the SLC System and 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/PCCS 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 CFR 52.47. 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 offsite dose.

6.2.4.2.2 Instrument Lines Penetrating Containment

Sensing instrument lines penetrating the containment follow all the recommendations of RG 1.11, as follows:

- Each line includes a 8 mm (¼ inch) diameter orifice such that in the event of a piping or component failure, leakage is reduced to the maximum extent practical consistent with other safety requirements. The rate of coolant loss is within the makeup capability, the integrity and functional performance of secondary containment and associated safety systems is maintained and the potential offsite exposure is substantially below the limits of 10 CFR 52.47(a)(2)(iv).
- Each line is provided with a self-actuated excess flow check valve located outside containment, as close as practical to the containment. These check valves are designed to remain open as long as the flow through the instrument lines is consistent with normal plant operation; however, if the flow rate is increased to a value representative of a loss of piping integrity outside containment, the valves close. These valves reopen automatically when the pressure in the instrument line is reduced.

- The instrument lines are designated as Quality Group B up to and including the isolation valve, located and protected to minimize the likelihood of damage, protected or separated to prevent failure of one line from affecting the others, accessible for inspection and not so restrictive that the response time of the connected instrumentation is affected.

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 RGs 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 3.9.3.5. The power-operated and automatic isolation valves are 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 RGs 1.11 and 1.141, except as noted in Table 1.9-6.

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 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).

The plant operators apply administrative controls 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. "Non-powered" CIVs are manual valves, check valves, and also may include certain safety or relief valves. In general, only manual valves are configured to permit administrative control. Further, compliance with GDCs 55 through 57 requires that the manual CIVs be locked closed. Powered or non-powered CIVs in the ESBWR design that are defined as passive valves (Table 3.9-8) are designed to have their position administratively controlled or are prudently inhibited from being repositioned (for example, by inadvertent operator control action). For these valves, the Combined License (COL) Holder may use any of the administrative methods that apply, including but not limited to, wire locks, tab locks, chain or bar and padlocks, secured or covered switches, deenergized and locked-out electrical breakers, removed fuses, or closed-and-locked fluidic power supply valves, in conjunction with alignment control procedures. These administrative controls meet the requirements of RG 1.141 and satisfy the standards of ANS-56.2/ANSI N271-1976. Where applicable, and using technically reliable design(s), check valves are equipped with a means for position indication. Excess flow check valves, typically used in instrument line isolation, are also equipped with position indication devices. If a safety or relief type valve is used as a Containment Isolation Valve (CIV), a position indication device is included in the design to indicate that the valve is open, either by direct sensing of disk position (e.g., follower rod with inductive sensor) or indirect means (e.g., tailpipe thermal sensor).

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 (Figure 5.1-2). It has three containment isolation valves, the inboard isolation is a simple check valve with process-actuated closure, and the two outboard valves are gate valves with automatic closure. There is a branch connection to each feedwater line on the outboard side of the penetration and between the penetration and the inner outboard feedwater isolation valve. The branch connection is isolated by a testable check valve. An additional simple check valve is located outboard of the feedwater containment isolation valves for function redundancy. Two check valves redundantly isolate the feedwater line or branch connection line in the event of an outboard feedwater pipe or branch connection pipe rupture (feedwater HELB). Two gate valves, isolate the line in the event of an inboard feedwater pipe rupture or other LOCA, or vessel overfill event. The inboard and outboard isolation valves are located as close as practicable to the containment wall. More detail about the feedwater lines isolation configuration is provided in Subsection 5.4.5.

Isolation Condenser Condensate and Venting Lines

The containment isolation provisions for the ICS condensate, vent, and purge lines constitute an alternative design basis beyond what is described by GDC 55. Instead of one isolation valve outside the containment and one isolation valve inside the containment, the ICS influent lines rely upon two valves inside containment as well as a closed system outside the containment. The following rationale support this alternative design:

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 containment isolation for the vent lines is very similar in design to the condensate lines. Instead of automatic isolation valves inside containment the vent lines utilize two normally closed fail closed valves in series. The vent lines are 20 mm (0.75 inch) in diameter, and their inboard isolation valves are designed to ASME Code Section III, Class 2 Quality Group B, Seismic Category I. The IC purge line isolation valves and the pipes penetrating the containment are designed in accordance to ASME Code Section III, Class I Quality Group A, Seismic Category I. The purge line is a 20 mm (0.75 inch) line that utilizes a closed system outside containment, and a fail closed isolation valve in series with an excess flow check valve inside 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. It is more practical to locate both valves inside containment because a valve outside containment would be submerged in the IC/PCCS pool. The isolation valves shall be located as close to the containment boundary as possible, and the pipe between the outermost isolation valve and the containment shall be designed to the requirements of SRP 3.6.2 to minimize the chances of a break in this area. A break on any of these influent lines could be contained by either of the redundant isolation valves. Furthermore, a break between the isolation valves and the containment would still be contained by the closed system outside containment, and would require an additional break before a radioactive release could occur. Therefore, this design can accommodate a single failure.

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 Isolation Valves (MSIVs) are described in Subsection 5.4.5.

Isolation Condenser Steam Supply Lines

The containment isolation provisions for the ICS steam supply lines constitute an alternative design basis beyond what is described by GDC 55. Instead of one isolation valve outside the containment and one isolation valve inside the containment, the ICS effluent lines rely upon two valves inside containment as well as a closed system outside the containment. The following rationale support this alternative design:

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 two-out-of-four 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 provide a sufficient alternative to the isolation functions of US NRC 10 CFR 50, Appendix A, Criteria 55. It is more practical to locate both valves inside containment because a valve outside containment would be submerged in the IC/PCCS pool. The isolation valves shall be located as close to the containment boundary as possible, and the pipe between the outermost isolation valve and the containment shall be designed to the requirements of SRP 3.6.2 to minimize the chances of a break in this area. A break on the steam supply lines could be contained by either of the redundant isolation valves. Furthermore, a break between the isolation valves and the containment would still be contained by the closed system outside containment and would require an additional break before a radioactive release could occur.

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. A special isolation arrangement is required for the ICS and it has been shown to be an adequate alternative to the explicit requirements of GDC 55. In all other cases, two isolation barriers were shown to protect against the release of radioactive materials in accordance with GDC 55.

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 6.2-45 identify 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 DW spray, the suppression pool the GDCS pools, and to the Reactor Well Drain.

The Reactor Well drain line contains two manual valves inside the containment that are locked closed during normal operation. This arrangement is an exception to GDC 56, which requires that such lines contain one isolation valve outside and one isolation valve inside the containment. The alternative arrangement with both valves inside containment is necessary because a valve outside containment would be submerged in the reactor well making it inaccessible and less reliable. The isolation valves are located as close as possible to the containment and the piping between the outermost valve and the containment boundary shall be designed to conservative requirements to preclude breaks in this area.

In each of the remaining influent 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.

Chilled Water System

Isolation is provided for the Chilled Water System (CWS) cooling lines penetrating containment. It is assumed that the nonsafety-related Seismic Category II coolant boundary of the CWS or Drywell Cooling System heat exchanger may fail, opening to the containment atmosphere. Therefore, Criterion 56 is applied to the design of the CWS containment penetration. The CWS containment influent lines have a 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 so that they are not exposed to the harsh environment of the wetwell and drywell and are accessible for maintenance, inspection and testing during reactor operation. 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.
- Are protected against a HELB 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.

6.2.4.3.2.2 Effluent Lines from Containment

Tables 6.2-33 through 6.2-45 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. The CIVs are normally closed and fail as-is for improved reliability. "Fail as-is" valves are acceptable because the valves are normally closed, are only opened when it is necessary to provide cooling to the suppression pool and do not communicate with the DW atmosphere. This arrangement is an exception to GDC 56, which requires that such lines contain one isolation valve outside and one isolation valve inside the containment. Such an alternative arrangement is necessary because an inboard valve could potentially be under water under certain accident conditions. Leak detection is provided for CIVs on the suppression pool suction line and the valves are located as close as possible to the containment.

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 so that they are not exposed to the harsh environment of the wet well and dry well and are accessible for maintenance, inspection and testing during reactor operation. 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.
- Are protected against a HELB 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 of these two lines contains an inboard and outboard valve. These two valves are pneumatic, solenoid or equivalent power operated valves and are used for isolation. These isolation valves will fail as-is.

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. Exceptions were taken in the cases of the Reactor Well drain line, and suppression pool suction line in the Fuel and Auxiliary Pools Cooling System, and they have been shown to be an adequate alternative to the explicit requirements of GDC 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 6.4 mm (¼ inch) orifices and manual isolation valves, in compliance with RG 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 single failure evaluation of fluid system penetration isolation design is to demonstrate that the safety-related function of the system can be completed even assuming a 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 change state (open or close) by its safety-related logic scheme, a single failure is assumed in the analysis if the component does change state. Electrically-operated valves include those valves that are electrically piloted with air/nitrogen-operated actuators, as well as valves that are directly operated by an electromagnetic device (solenoid motor, motorized-gearbox, or electrohydraulic actuators). In addition, all electrically-operated valves that receive automatic actuation signals can also be remote-manually actuated from the main control room. Therefore, a single failure in any electrical or mechanical 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 from a component performing an unnecessary mechanical motion due to a spurious/incorrect signal or manual operating error. 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.

The isolation design for each penetration or penetration class also applies the guidance of standards ANS 58.9 and IEEE 379-2000 (Table 1.9-22), and complies as appropriate with RG 1.53 (refer to Tables 1.9-21 and 7.1-1, and to Subsections 7.3.3 and 7.5.2). Standard IEEE 379-2000 provides a suggested single-failure review method which includes:

- For each design event:
  - Determine the safety function to be performed;
  - Determine the protective action(s) available to accomplish the safety function;
  - Determine safety-related component that performs the protective action and satisfies the safety function; and
  - Verify independence between redundant safety-related components; or
  - Iterate design as required when independence is not verified, considering the electrical, mechanical and system logic failures potentially affecting the isolation design.
- Evaluate for interconnections between redundant circuits.
- Evaluate isolation system logic for common failures affecting isolation capability.
- Evaluate actuation devices for preferred mode on loss of power and for single-point failure mechanisms that might cause common failure of the isolation design.
- Evaluate support systems and auxiliary features, in particular the actuator power supplies (including backup electrical power, mechanical or process power and fluid accumulator stored energy supplies).
- Evaluate for nonsafety-related attachments or interfaces with the isolation design that could interfere with completion of the protective action.

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.

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 remains 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 Subsection 6.2.5.2.2 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 and RG 1.7. 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
- RG 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 Passive Autolytic Recombiner System, 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 Passive Autolytic Recombiner System 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 Passive Autolytic Recombiner System consists of passive autocatalytic recombiners (PARs) strategically located throughout the WW gas space and DW.

PARs are independently mounted components which are capable of recombining a stoichiometric mix of hydrogen and oxygen into water vapor. This recombination is facilitated through the use of a selective metal catalyst (typically palladium and hydrophobic polymer coating). This catalyst is located internal to the PARs typically constrained within a series of screens. These screens or plates are separated such that gas and vapor flow are not impeded to allow easy migration of the post-LOCA gasses through the PAR structure.

As the recombination process is exothermic, the heat created provides motive force to carry the vapor up and out of the PAR allowing more gasses to be drawn into the device to further the recombination process. As the gasses pass through the PAR (essentially a stainless steel cylinder or rectangle with an opening in either end), the hydrogen and oxygen pass by the catalyst and are selectively recombined into water vapor. The outer wall of the PAR continues above the level of the catalyst to provide a chimney effect, which further assists the convection driven flow of water vapor out of the device. This flow through the PAR aids in the overall containment mixing process by facilitating natural convection near and around the PAR locations.

The PAR device requires no external power or controls. They operate automatically when a stoichiometric mix of hydrogen and oxygen is realized. The oxygen and hydrogen molecules are adsorbed onto the surface of the catalyst due to the attractive forces of the catalyst. There they are recombined into water vapor and released. Although the recombination process is conservatively assumed to begin at 72 hours, it actually starts with as little as 1% by volume of an available stoichiometric mix of recombination gasses. And due to the hydrophobic coating of the catalyst, the recombination process does not require heating of the catalyst media enclosure (as is the case with the typical offgas recombiner). It occurs through the full range of operating and post-accident containment temperatures.

The PARS consists of sufficient capacity PAR units to effect a minimum safety factor of two for each containment compartment (Drywell and Wetwell), with respect to any efficiency loss primarily due to introduced catalytic poisons. The number and size of PARs to be utilized in each containment compartment will be selected based on the nominal hydrogen depletion rate of each individual PAR unit such that the total depletion rate is twice the maximum hydrogen generation rate at 72 hours. The maximum hydrogen generation rate at 72 hours is 0.32 kg/h

(0.71 lb/hr). As the hydrogen generation rate is dropping from that point forward in the accident, the amount of total hydrogen never exceeds this 72 hour maximum due to the operation of the PARs. The number and size of PARs specified will provide the minimum safety factor of two for each containment compartment (Drywell and Wetwell). There will be a minimum capacity of the equivalent of one full size PAR unit specified for each containment compartment, however due to other design considerations, more, smaller capacity units (with equivalent total capacity) will be specified. The nominal hydrogen depletion rates for the full size PAR will be a minimum of 0.8 kg/h (1.8 lb/hr). The half, quarter, and eighth size PARs have nominal depletion rates as a direct ratio to the full size PAR. Additionally PARs are sited to incorporate items such as protection from jet impingement, protection from containment spray and cooling fan discharge, protection from flooding and pool swell, discharged exhaust impacts and accessibility for testing. The PAR units are placed in azimuthally diverse locations which support adequate containment atmosphere mixing.

There is the potential for combustible gases to accumulate inside the PCCS condensers at concentrations that are higher than the overall containment. See Subsection 6.2.2.2.2 for further description and design features for these higher concentrations in the PCCS condensers.

6.2.5.1.1 Containment Purging Under Accident Conditions

In accordance with 10 CFR 50.34(f)(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.

The Containment Inerting System (CIS) is used to establish and maintain an inert atmosphere within the 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 and maintenance during reactor low power operation. The system is designed to permit deinerting the containment for safe operator access and minimizing personnel exposure.

6.2.5.2 Containment Inerting System

The objective of the CIS 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 Design Bases

The 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 four hours and less than 2% oxygen by volume in the next eight hours following an outage.

- The CIS is designed to maintain the containment oxygen concentration below the maximum permissible limit (4%) during normal power operations 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 rate of 0.4% per day based on containment operating pressure.
- The inerting auxiliary steam vaporizer is sized to provide at least 2.5 times the containment (WW and DW) free volume of nitrogen within the allotted four 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. (Subsection 6.2.1.1.10.2).

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.

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 DW 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 DW and suppression pool airspace.

The CIS includes an exhaust line from the lower DW on the opposite side of containment from the injection points. The discharge line connects to the RBVS exhaust before being diverted to the RB/Fuel Building (FB) vent stack. The RBVS 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) (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 DW and the WW. 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 DW and the WW 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 Main Control Room (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 RBVS is injected into the DW and WW air space through the inerting injection line. The incoming air displaces containment gases (mostly nitrogen) into the exhaust line. The RBVS exhaust fans, filters, and radiation detectors remove the vented gases and then they are diverted to the RB/FB vent stack. The CIS is capable of reaching a volumetric oxygen concentration of ≥ 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 inservice 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 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 I 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 DW and wetwell air space. The functions of the CMS are:

- To monitor hydrogen and oxygen concentrations and gross gamma radiation levels in the DW and WW 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 DW 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 provided in the main control room. 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 DW or WW, 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 provided in the main control room. Alarms 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 DW and WW. The signals are provided in the main control room. 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 noncondensable gases throughout the containment atmosphere and minimize the accumulation of high concentrations of combustible gases in local areas.

Adequate mixing within the ESBWR containment system is assured based on the configuration of the ESBWR containment coupled with the dynamics of the design basis loss-of-coolant-accident (LOCA) and the mitigating components within the containment volume. The containment atmospheres (DW and WW) are inerted with nitrogen. At the start of an accident normal DW ventilation can be assumed to be in operation, therefore the inerted atmosphere is thoroughly mixed at that point in time. Although the normal DW ventilation system ceases to operate at the onset of the accident, the accident itself (LOCA) creates a highly turbulent condition in which mixing is assured. Steam expansion also serves to create large mixing flows. Molecular diffusion and natural convection continues the mixing process providing reasonable assurance that adequate mixing exists throughout the accident coping period. Natural convection is promoted by temperature gradients existing in the DW and the cascading effect of the water exiting through the break. Because the ESBWR core remains covered during DBAs, only a minimal amount of hydrogen is generated by radiolysis. This is base on the fuel temperature remaining below the metal-water reaction initiation temperature.

The relatively large open volume of the ESBWR containment enhances the structure’s ability for continued mixing. In consideration of the differential components temperatures inside containment (coupled with the relative low temperatures of the outer DW walls), local convection around these components coupled with the natural chimney effect of the open shafts in the containment volume provides substantial motive force furthering the mixing process.

Aside from the DW and WW, there are only two other subcompartments within the containment, the Drywell Head Region and the Reactor Shield Annulus. The Drywell Head Region contains no high energy piping. As such, and based on the location of this region above the drywell proper, in the unlikely event that hydrogen migrates toward this area it would be quickly displaced by the rising steam from the break. There is reasonable assurance that it would not be possible for it to collect in this region. The Reactor Shield Annulus volume is that area between the reactor shield wall and the reactor vessel. In the unlikely event that the LOCA occurs at the nozzle of any of the vessel's high energy piping (which does pass though this region), any noncondensable effluent from the break would quickly be dispersed by the escaping steam and swept down into the WW by steam along with the DW noncondensable gases.

The dynamic effects of a LOCA in the ESBWR containment serves to drive the noncondensables gases from the DW into the WW. This initial blowdown is due to pressurization of the DW atmosphere by steam exiting the vessel. This steam forces (blowdown) noncondensable gases through (under) the WW water volume. This is quite a dynamic evolution that thoroughly mixes the WW atmosphere, albeit an atmosphere much richer in hydrogen and oxygen than that remaining in the DW.

Subsequent to the reactor depressurization, the PCCS condensers will continue to operate. The PCCS condensers vent noncondensable gas to the WW (suppression pool), so the concentration of hydrogen is higher in the WW than in the DW.

Another consideration with respect to the mixing process is the incorporation of PARs into both the DW and WW. PARs create convective air currents, which further serve to drive both the recombination process along with mixing both in the DW and WW atmospheres. A description of PARS is given in Subsection 6.2.5.1.

The containment design features that reduce the likelihood of combustible gas deflagrations resulting from localized buildup of combustible gases during degraded core accidents are listed in Subsection 19.3.2.

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 is not 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 which forces evacuated atmosphere through the suppression pool to scrub out fission products. For containment overpressure protection during severe accident conditions, only this line is used. This line has two air-operated valves in series outside containment. The valves have the capability to be operated with bottled air for local operation during accident conditions. The downstream piping is tied into the RB/FB vent stack. The piping downstream of the valves is designed for the maximum expected operating pressure following the accident.

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 is sufficient time to implement Severe Accident Management actions. It is desirable to have at least a 24 hour period following an accident to allow for Severe Accident Management implementation. This section discusses the rate at which post-accident oxygen is 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 Severe Accident Management 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 Subsection 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. Although the process of steam condensation has the effect of concentrating the radiolytically generated hydrogen and oxygen within the ICS and PCCS condensers these components have been designed to accommodate the loads resulting from combustion.

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

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 is 102% of rated;
- G (O_2) = 0.25 molecules/100eV;
- Initial containment O_2 concentration = 4%;
- Allowed containment O_2 concentration = 5%;
- Stripping of DW noncondensable gases to WW 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 is sufficient time available to activate the emergency response organization and implement the Severe Accident Management actions necessary to preclude a combustible gas deflagration.

Also, the potential combustion of radiolytically generated gas inside the PCCS and ICS condensers has been considered as described in Reference 6.2-14.

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 RG 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 10 CFR 50, 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 (ILRT) (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.

For the purposes of 10 CFR 50, Appendix J testing, the value of P_a (defined as the calculated peak containment internal pressure related to the design basis LOCA) is selected to be the containment design pressure as specified in Table 6.2-1.

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, 3.8.2, 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 eight 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 four 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. ILRT results are satisfactory if the upper confidence level 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 MSIV leakage rate. The treatment of MSIV leakage pathway separately in radiological dose analysis in Subsection 15.4.4.5.2.4 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. 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 upper confidence limit 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 upper confidence limit and as-left minimum pathway leakage rate 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 minimum pathway leakage rate for that leakage path must also be added to the Type A upper confidence limit. 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 are 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 are performed at least three times during each 10 year service period. In case Option B is selected, the test interval is per RG 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. If two consecutive periodic ILRTs fail to meet the acceptance criteria prior to corrective action, an ILRT is performed at each plant shutdown for major refueling or approximately every 24 months (whichever occurs first), until two consecutive ILRTs meet the acceptance criteria, after which time the previously established periodic retest schedule may be resumed. The 24 month minimum period is an exemption from 10 CFR 50, Appendix J.

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 previously established periodic retest schedule may be resumed.

The following additional criteria are met for ILRTs 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.

If 10 CFR 50 Appendix J Option B is implemented, all 10 CFR 50 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 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) are 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 three days of being opened. If the air-lock is to be opened more frequently than once every three days, the air-lock is tested at least once every three 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 L_a. Air-locks are tested at initial fuel loading, and at least once every six months thereafter. In case Option B is selected, the test interval is per RG 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 6.2-15 through 6.2-45.

Type C tests (like Type B tests) are performed by local pressurization using either the pressure-decay or flowmeter method. The test pressure is applied in the same direction as when the valve is required to perform its safety function, unless it can be shown that results from tests with pressure applied in a different direction are equivalent or conservative. For the pressure-decay method, test volume is pressurized with air or nitrogen to at least P_a. As an exemption from 10 CFR 50 Appendix J, Option A, 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 L_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 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.

Per ANSI/ANS-56.8-1994 (for Option A) and NEI 94-01, Revision 0 (for Option B), a Type C local leakage rate test may not be performed for the following cases:

- Primary containment boundaries that do not constitute potential primary containment atmospheric pathways during and following a DBA;
- Boundaries sealed with a qualified seal system; or
- Test connection vents and drains between primary containment isolation valves that are one inch or less in size, administratively secured closed and consist of a double barrier.

Per ANSI/ANS-56.8-1994, a qualified seal system is "a system that is capable of sealing the leakage with a liquid at a pressure no less than 1.1 P_{ac} [equivalent to P_a in 10 CFR 50, Appendix J] for at least 30 days following the DBA." Type C valves with a qualified seal system are periodically tested to prove functionality by pressurizing the line with the sealing fluid to a pressure of not less than 1.1 P_a. The measured leakage is excluded when determining the combined leakage rate, 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
- The installed isolation valve seal-water system fluid inventory is sufficient to assure the sealing function for at least 30 days at a pressure of 1.1 P_a.

Unless there is essentially an unlimited supply of sealing fluid, valve-specific leakage rate limits are assigned, based on analyses to assure fluid inventory for 30 days at a pressure of 1.10 P_a assuming the most limiting single failure of any active component, and included in the Technical Specifications.

The following exemption from 10 CFR 50 Appendix J Option A or Option B is taken for Type C test for MSIVs:

- The measured leakage rate of MSIV in a Type C test is 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.

Instrument lines that penetrate containment conform to RG 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 (Subsection 6.2.6.3) 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 is followed by either a Type A, B, or C test (as applicable) for the area affected by the modification.

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

6.2.6.5 (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 RG 1.26, which assigns correspondence of Group B Quality Standards to ASME Code Section III Class 2.

6.2.8 COL Information

6.2-1-H (Deleted)

6.2.9 References

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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, Revision 1, Class I (Non-proprietary), May 2009.

6.2-8 Moody, F.J. "Maximum Discharge Rate of Liquid-Vapor Mixtures from Vessels," General Electric Company, Report No. NEDO-21052-A, May 1979.

6.2-9 GE Hitachi Nuclear Energy, "ESBWR Scaling Report," NEDC-33082P, Revision 2, Class III (Proprietary), April 2008; NEDO-33082, Revision 2, Class I (Non-proprietary), April 2008.

6.2-10 TRACG Qualification for Simplified Boiling Water Reactor (SBWR), NEDC-32725P, Rev. 1, Vol. 1 and 2, August 2002.

6.2-11 GE Hitachi Nuclear Energy "ESBWR Safety Analysis - Additional Information," NEDE-33440P, Revision 2, Class III (Proprietary), March 2010; NEDO-33440, Revision 1, Class I (Non-proprietary), March 2010.

6.2-12 Idel'chik, I.E., Barouch, A. "Handbook of hydraulic resistance: coefficients of local resistance and of friction," National Technical Information Service, 1960.

6.2-13 SMSAB-02-04, "CONTAIN Code Qualification Report/User Guide for Auditing Subcompartment Analysis Calculations," Office of Nuclear Regulatory Research, September 2002 (ADAMS Accession Number ML023220288).

6.2-14 GE Hitachi Nuclear Energy, "ESBWR PCCS Condenser Structural Evaluation," NEDE-33572P, Class II (Proprietary), Revision 0, March 2010; NEDO-33572, Revision 0, Class I (Non-proprietary), March 2010.

Table 6.2-1

Containment Design Parameters

<u>**Design Conditions**</u>**:**	
Upper and Lower Drywell	
Design Pressure	310 kPaG [45 psig]
Design Temperature	171°C (340°F)
Internal minus External Differential Pressure	-20.7 kPaD [-3.0 psid]
DW minus WW Differential Pressure	241 kPaD [35 psid] to -20.7 kPaD [-3.0 psid]
Inerting Gas	Nitrogen (with ≤ 3% Oxygen by Volume)
Wetwell	
Design Pressure	310 kPaG [45 psig]
Design Temperature	121°C (250°F)
Inerting Gas	Nitrogen (with ≤ 3% Oxygen by Volume)
Horizontal Vent System	
Design Pressure	310 kPaG [45 psig]
Design Temperature	171°C (340°F)
Containment Leak Rates	
Maximum Containment Leakage Excluding MSIV Leakage	0.35% of Weight of Containment Free Volume per 24 hours at Pressure 310 kPaG [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 (WW Pressure minus DW Pressure)	3.07 kPaD [0.445 psid]
Vacuum Breaker Closing Differential Pressure (WW Pressure minus DW Pressure)	2.21 kPaD [0.320 psid]

Table 6.2-2

Containment Conditions During Normal Operation

Upper and Lower Drywell	
Pressure during Normal Operation	
Nominal	104.1 kPa (15.1 psia)
Maximum	106.9 kPa (15.5 psia)
Minimum	101.3 kPa (14.7 psia)
Temperature during Normal Operation	
Nominal	57.2°C (135°F)
Maximum	65.5°C (150°F)
Minimum	46.1°C (115°F)
Relative Humidity during Normal Operation	
Nominal	50%
Drywell Pressure Scram Initiation Setpoint	13.8 kPaG (2.0 psig)
Wetwell	
Pressure during Normal Operation	
Nominal	104.1 kPa (15.1 psia)
Maximum	106.9 kPa (15.5 psia)
Minimum	101.3 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 (nominal design value)	6016 m^3 (~212500 ft^3)
Lower Drywell Free Gas Volume (nominal design value)	1190 m^3 (~42020 ft^3)
Wetwell	
Free Gas Space Volume atHigh Water Level (nominal design value)	5350 m^3 (188900 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 (8600 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	581 m^3 (20500 ft^3)
Total Water Volume (for pool at 180 degrees) at Normal Water Level	779 m^3 (27500 ft^3)
Non-Drainable Water Volume (per pool for pools at 90 and 270 degrees)	83 m^3 (2900 ft^3)
Non-Drainable Water Volume (for pool at 180 degrees)	111 m^3 (3920 ft^{3-})
Pool Surface Area (per pool for pools at 90 and 270 degrees)	88 m^2 (950 ft^2)
Pool Surface Area (for pool at 180 degrees)	118 m^2 (1270 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 Normal Water Level (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[(1)] m^2 (ft^2)	Single Failure	Maximum DW Pressure[(5)] kPa (psia)	Maximum DW Pressure[(5)] kPaG (psig)	Margin[(6)] to Design Pressure of 310 kPaG (45 psig) (%)	Short-term Bulk DW Temperature[(7)] ºC (ºF)	Long-term Bulk DW Temperature[(7)] ºC (ºF)	Long-term WW Temperature[(7)] ºC (ºF)	Long-term Suppression Pool Temperature[(7)] ºC (ºF)
Based on standard (Nominal) TRACG evaluation model:									
Steam Line Inside Containment[(3)]	0.09832 (1.058)	1 DPV	365.75 (53.05)	264.39 (38.35)	15%	182.91 (361.24)	140.47 (284.85)	122.73 (252.91)	75.56 (168.00)
Feedwater Line[(2)]	0.07420 (0.7986)	1 DPV	338.87 (49.15)	237.52 (34.45)	23%	169.88 (337.78)	139.33 (282.80)	118.45 (245.21)	72.00 (161.60)
GDCS Injection Line	0.004561 (0.04910)	1 DPV	314.46 (45.61)	213.11 (30.91)	31%	164.16 (372.49)	137.39 (279.30)	114.55 (238.19)	63.98 (147.16)
Bottom Head Drain Line	0.004052 (0.04361)	1 DPV	324.95 (47.13)	223.60 (32.43)	28%	164.75 (328.55)	138.20 (280.76)	115.69 (240.26)	64.75 (148.57)
Based on bounding values:									
Steam Line Inside Containment[(3)]	0.09832 (1.058)	1 DPV	396.25 (57.47)	294.90 (42.77)	5%	173.58 (344.44)	143.35 (290.03)	125.94 (258.69)	74.81 (166.66)
Feedwater Line[(2)]	0.07420 (0.7986)	1 DPV	367.88 (53.36)	266.53 (38.66)	14%	169.47 (337.05)	142.13 (287.83)	121.95 (251.51)	71.08 (189.94)
Steam Line Inside Containment[(3)]	0.09832 (1.058)[(1)]	1 SRV	397.45 (57.65)	296.10 (42.95)	5%	173.57 (344.43)	143.46 (290.22)	126.55 (259.79)	74.59 (166.26)
Feedwater Line[(2)]	0.07420 (0.7986)	1 SRV	369.63 (53.61)	268.28 (38.91)	14%	167.94 (334.30)	142.24 (288.03)	123.16 (253.69)	69.61 (157.30)
Steam Line Inside Containment, Off Site Power[(4)]	0.09832 (1.058)[(1)]	1 SRV	394.12 (57.16)	292.76 (42.46)	6%	174.97 (346.95)	143.17 (289.71)	128.39 (263.10)	74.33 (165.79)
(Deleted)									

(1) The break area is from the RPV side of the break.

(2) For the feedwater line break, the total break area from the TB side is limited at the two parallel venturi sections, with flow area of 0.04997 m^2 (0.53787 ft^2) each. The break area from the RPV side of the break is limited by the feedwater sparger piping, which has a flow area of 0.07420 m^2 (0.79862 ft^2). 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 10 s).

(3) Main Steam Line Break, at Level 34, 2 GDCS vent paths. The break area from the RPV side of the break is limited by the Main Steam Line (MSL) nozzle, which has a flow area of 0.09832 m^2 (1.05831 ft^2).

(4) Main Steam Line Break with offsite power, see Table 6.2-6 for initial conditions. Other bounding cases are based on the initial DW pressure and WW pressure of 110.3 kPa (16.0 psia).

(5) Maximum DW pressure calculated during the 72 hours following a LOCA.

(6) Minimum pressure margin calculated during the 72 hours following a LOCA.

(7) The containment structure is evaluated for the fluid temperatures in Appendix 3G.5.

Table 6.2-5a

Bounding Estimate of the ESBWR Containment Pressure[1]

(1) TRACG Calculated Pressure and Margin

Case #	Break Location	Single Failure	Peak DW Pressure (kPa)	GDCS NC Gas Mass at 72 hrs (kg)	DW Head NC Gas Mass at 72 hrs (kg)	DW NC Gas Mass at 72 hrs (kg)	WW NC Gas Mass at 72 hrs (kg)	DW Pressure (psig)	TRACG Calculated DW Pressure Margin at 72 hr [based on design limit of 310 kPaG (45 psig)], %
1	Steam Line Inside Containment	1 DPV	396.25	27.01	121.05	3.85	14875.00	42.77	5
2	Feedwater Line	1 DPV	367.88	712.66	50.20	22.13	14247.76	38.66	14
3	Steam Line Inside Containment	1 SRV	397.45	13.61	124.21	3.22	14884.95	42.95	5
4	Feedwater Line	1 SRV	369.63	702.43	13.88	3.29	14313.51	38.91	14
5	Steam Line Inside Containment, Off Site Power[2]	1 SRV	394.12	10.98	145.40	3.21	14417.78	42.46	6

(2) Pressure Margin (Purge all residual noncondensable gas from DW to WW)

Case #	Break Location	Single Failure	Total NC Gas Mass remaining in DW regions at 72 hrs (kg)	Total NC Gas Mass in WW (with All NC Gas purged into WW at 72 hrs) (kg)	Adjusted WW NC Gas Mass Ratio	Adjusted DW Pressure at 72 hr (kPa)	Adjusted DW Pressure at 72 hr (psia)	Adjusted DW Pressure at 72 hr (psig)	Adjusted DW Pressure Margin at 72 hr [based on design limit of 310 kPaG (45 psig)], %
1	Steam Line Inside Containment	1 DPV	151.92	15026.92	1.010	400.30	58.06	43.36	4
2	Feedwater Line	1 DPV	784.99	15032.75	1.055	388.15	56.30	41.60	8
3	Steam Line Inside Containment	1 SRV	141.04	15025.99	1.009	401.22	58.19	43.49	3
4	Feedwater Line	1 SRV	719.60	15033.11	1.050	388.22	56.31	41.61	8
5	Steam Line Inside Containment, Off Site Power[2]	1 SRV	159.59	14577.37	1.011	398.48	57.79	43.09	4

Table 6.2-5a

Bounding Estimate of the ESBWR Containment Pressure[1]

(3) Pressure Margin with additional noncondensable gas from pneumatic valves and nitrogen dissolved in the SLC system liquid

Case #	Break Location	Single Failure	Additional NC Gas Mass from Containment Valve Systems and SLC Liquid[3] (kg)	Total NC Gas Mass in WW (with All NC Gas purged into WW at 72 hrs) (kg)	New WW NC Gas Mass Ratio	Adjusted DW Pressure at 72 hr (kPa)	Adjusted DW Pressure at 72 hr (psia)	Adjusted DW Pressure at 72 hr (psig)	Adjusted DW Pressure Margin at 72 hr [based on design limit of 310 kPaG (45 psig)], %
1	Steam Line Inside Containment	1 DPV	184.50	15211.42	1.023	405.22	58.77	44.07	2
2	Feedwater Line	1 DPV	184.50	15217.25	1.068	392.91	56.99	42.29	6
3	Steam Line Inside Containment	1 SRV	184.50	15210.49	1.022	406.14	58.91	44.21	2
4	Feedwater Line	1 SRV	184.50	15217.61	1.063	392.98	57.00	42.30	6
5	Steam Line Inside Containment, Off Site Power[2]	1 SRV	184.50	14761.87	1.024	403.53	58.53	43.83	3

(3a) Pressure margin with reduced initial NC gas mass from reduced DW and WW Free Volume[4]

Case #	Break Location	Single Failure	Mass of NCG Removed from WW due to Volume Readjustments (kg)	Total NC Gas Mass in WW (with All NC Gas purged into WW at 72 hrs) (kg)	New WW NC Gas Mass Ratio	Adjusted DW Pressure at 72 hr (kPa)	Adjusted DW Pressure at 72 hr (psia)	Adjusted DW Pressure at 72 hr (psig)	Adjusted DW Pressure Margin at 72 hr [based on 310.3 kPaG (45 psig)], %
1	Steam Line Inside Containment	1 DPV	-230.63	14980.80	1.007	399.07	57.88	43.18	4
2	Feedwater Line	1 DPV	-230.63	14986.62	1.052	386.96	56.12	41.42	8
3	Steam Line Inside Containment	1 SRV	-230.63	14979.87	1.006	399.99	58.01	43.31	4
4	Feedwater Line	1 SRV	-230.63	14986.98	1.047	387.02	56.13	41.43	8

Table 6.2-5a

Bounding Estimate of the ESBWR Containment Pressure[1]

5	Steam Line Inside Containment, OSP[2]	1 SRV	-223.02	14538.85	1.008	397.43	57.64	42.94	6

(3b) Pressure Margin with Reduced WW Free Volume[5]

Case #	Break Location	Single Failure				Adjusted DW Pressure at 72 hr (kPa)	Adjusted DW Pressure at 72 hr (psia)	Adjusted DW Pressure at 72 hr (psig)	Adjusted DW Pressure Margin at 72 hr [based on 310.3 kPaG (45 psig)], %
1	Steam Line Inside Containment	1 DPV				405.64	58.83	44.13	2
2	Feedwater Line	1 DPV				393.33	57.05	42.35	6
3	Steam Line Inside Containment	1 SRV				406.57	58.97	44.27	2
4	Feedwater Line	1 SRV				393.39	57.06	42.36	6
5	Steam Line Inside Containment, OSP[2]	1 SRV				403.97	58.59	43.89	2

(4) Targeted Pressure Margin at 1%

					DW Pressure (kPa)	DW Pressure (kPaG)	DW Pressure (psia)	DW Pressure (psig)	DW Pressure Margin [based on design limit of 310 kPaG (45 psig)], %
	Pressures at 1% Margin				408.51	307.16	59.25	44.55	1

1 Containment pressure adjustment assuming all noncondensable (NC) gases are purged into WW at 72 hrs, including NC gases remaining in DW regions, NC gases dissolved in the SLC system liquid, and NC gases from safety/non-safety related pneumatic containment valves during the event.

2 Main Steam Line Break with Offsite Power (OSP), see Table 6.2-6 for initial conditions. Other bounding cases are based on the initial DW pressure and WW pressure of 110.3 kPa (16.0 psia).

3 Additional noncondensable gases from pneumatic valves (142.9 kg) and nitrogen dissolved in the SLC system liquid (41.6 kg) total 184.5 kg.

4 Containment pressure recalculated considering a change in DWand WW NC gas masses (maximum DW volume and minimum WW volume per Table 6.2-6.)

5 Containment pressure recalculated considering a reduction in the available WW volume (minimum per Table 6.2-6).

Table 6.2-6

Plant Initial Conditions Considered in the Containment DBA Cases

No.	Plant Parameter	Nominal Value	Bounding Value
1	RPV Core Power	100%	102%
2	WW Relative Humidity	100%	100%
3	PCCS Pool Level	4.8 m (15.8 ft)	4.8 m (15.8 ft)
4	PCCS Pool Temperature	43.3°C (110°F)	43.3°C (110°F)
5	DW Pressure	101.3 kPa (14.7 psia)	106.9 kPa (15.5 psia)
6	DW Temperature	46.1°C (115°F)	46.1°C (115°F)
7	WW Pressure	101.3 kPa (14.7 psia)	106.9 kPa (15.5 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 (17.9 ft)	5.50 m (18.1 ft)
12	GDCS Pool Level	6.60 m (21.7 ft)	6.60 m (21.7 ft)
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 (1 ft)
16	Maximum DW Volume	7245 m^3** (255900 ft^3)	7245 m^3** (25590 ft^3)
17	Minimum WW Volume	5300 m^3** (187200 ft^3)	5300 m^3** (187200 ft^3)
18	RPV Dome Vapor and Saturation Temperature	287.4°C (549.3°F)	288.4°C (551.0°F)
19	RPV Lower Plenum Liquid Temperature	272.3°C (522.2°F)	272.2°C (522.0°F)

*NWL = Normal Water Level (Table 15.2-1)
** Values used in Table 6.2-5a

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 (1.076 ft)	✓	
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 SV	✓	
10	Containment vents	10 to 12	✓	
11	Spill-over connection (DW annulus to vertical vent module)	Changed from ten horizontal holes to twelve horizontal holes; hole inlet elevation raised to approximately 2.5 m (8.2 ft) above the suppression pool normal water level.	✓	
12	SLC System 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.	✓	
15	Integrated TRACG input deck	Combined the RPV and containment input decks into one consistent, detailed deck.		✓

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
16	Chimney modeling	Changed from one chimney in containment P&T nodalization and three chimneys in RPV water level nodalization, to a total of five 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 DW	Changed the modeling from TEE component cells to VSSL component cells (improved nodalization).		✓
19	DW head air space	Modeled in the combined nodalization (improved nodalization).		✓
20	Top of WW airspace	One additional axial level added at an elevation near the top of WW (improved nodalization).		✓
21	IC/PCCS 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.
1	FWL isolation starts, to be completely isolated in 15 seconds
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 seconds, Middle Vent: 2.4 seconds, Bottom Vent: 3.2 seconds.
3	IC initiated from loss of power generation bus with 3 seconds signal delay time, drain valves start to open at 15 seconds later.
5	Level 3 is reached (scram signal from Level 3 is not credited in this analysis).
12	Level 2 is reached (MSIV closure and ICS initiation signals are not credited in this analysis).
16	Low MSL pressure setpoint reached, MSIV closure initiated at 0.7 second later.
20	Reactor isolated on low MSL pressure setpoint.
334	Level 1 is reached.
344	Level 1 signal confirmed; ADS/GDCS/SLCS timer initiated; SRV actuated.
394	DPV actuation begins at 50 seconds after confirmed Level 1 signal; SLCS flow starts on DPV actuation.
494	GDCS timer (150 seconds after confirmed Level 1 signal) timed out. GDCS injection valves open.
527	Vessel pressure decreases below maximum injection pressure of GDCS. GDCS flow into the vessel begins. Chimney and downcomer water levels start to rise.
694	SLCS flow depleted.

Table 6.2-7

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

Time (sec)	**Events**
135000 (~ 37.6 hrs)	PCCS pool level drops below the elevation of 29.6 m (97.1 ft.); 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 IC/PCCS expansion pools.
From ~900 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 338.9 kPa (49.2 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 DW pressure setpoint reached, scram signal from high DW 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: 1.4 seconds, Middle Vent: 1.7 seconds, Bottom Vent: 2.3 seconds.
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.
16	Level 2 is reached (MSIV closure and ICS initiation signals are not credited in this analysis).
458	Level 1 is reached.
468	Level 1 signal confirmed; ADS/GDCS/SLC timer initiated; SRV actuated.
518	DPV actuation begins at 50 seconds after confirmed Level 1 signal; SLC flow starts on DPV actuation.
618	GDCS timer (150 seconds after confirmed Level 1 signal) times out. GDCS injection valves open.
621	Vessel pressure decreases below maximum injection pressure of GDCS. GDCS flow into the vessel begins. Chimney and downcomer water levels start to rise.
818	SLCS flow depleted.

Table 6.2-7a

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

Time (sec)	Events
134626.1 (~ 37.4 hrs)	PCCS pool drops below the elevation of 29.6 m (97.1 ft); top ¼ portion of the PCCS tube length becomes uncovered; connection valves open to allow the water from the equipment 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 365.75 kPa (53.05 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 ICS.
2	Loss of normal auxiliary power confirmed; reactor scram initiated; rod insertion started at 0.25 second later.
3	ICS initiated from loss of power generation bus with 3 seconds signal delay time, drain valves started to open at 15 seconds later.
3	High Drywell pressure setpoint reached, scram signal from high drywell pressure is not credited in this analysis.
7	Level 3 is reached (scram signal from Level 3 is not credited in this analysis).
16	Level 2 is reached (MSIV closure and ICS initiation signals are not credited in this analysis).
20	Low MSL pressure setpoint reached, MSIV closure initiated at 0.7 second later.
24	Reactor isolated on low MSL pressure set point.
Vent clearing time	Top Vent: 32.9 seconds, Middle Vent: never cleared, Bottom Vent: never cleared.
219	Level 1 is reached.
229	Level 1 signal confirmed; ADS/GDCS/SLC timer initiated; SRV actuated.
279	DPV actuation begins at 50 seconds after confirmed Level 1 signal; SLC flow starts on DPV actuation.
379	GDCS timer (150 seconds after confirmed Level 1 signal) timed out. GDCS injection valves open.
505	Vessel pressure decreases below maximum injection pressure of GDCS. GDCS flow into the vessel begins. Chimney and downcomer water levels start to rise.
580	SLC flow depleted.

Table 6.2-7b

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

(Nominal Case)

Time (sec)	Events
1322127 (~ 36 hrs)	PCCS pool drops below the elevation of 29.6 m (97.1 ft); top ¼ portion of the PCCS tube length becomes uncovered; connection valves open to allow the water from the equipment storage pool to flow into the PCCS pools.
From ~825 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 314.46 kPa (45.61 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.
6	High Drywell pressure setpoint reached, scram signal from high drywell pressure is not credited in this analysis.
7	Level 3 is reached (scram signal from Level 3 is not credited in this analysis).
16	Level 2 is reached (MSIV closure and ICS initiation signals are not credited in this analysis).
20	Low MSL pressure setpoint reached, MSIV closure initiated at 0.7 second later.
24	Reactor isolated on Low MSL pressure setpoint
Vent clearing time	Top Vent: 286.6 seconds, Middle Vent: 427.5 seconds, Bottom Vent: never cleared.
362	Level 1 is reached.
372	Level 1 signal confirmed; ADS/GDCS/SLC timer initiated; SRV actuated.
422	DPV actuation begins at 50 seconds after confirmed Level 1 signal; SLC flow starts on DPV actuation.
522	GDCS timer (150 seconds after confirmed Level 1 signal) timed out. GDCS injection valves open.
652	Vessel pressure decreases below maximum injection pressure of GDCS. GDCS flow into the vessel begins. Chimney and downcomer water levels start to rise.
723	SLC flow depleted.

Table 6.2-7c

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

Time (sec)	Events
137759 (~38.3 hrs)	PCCS pool drops below the elevation of 29.6 m (97.1 ft); top ¼ portion of the PCCS tube length becomes uncovered; connection valves open to allow the water from the equipment 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 324.95 kPa (47.13 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.
1	FWL isolation starts, to be completely isolated in 15 seconds.
2	Loss of normal auxiliary power confirmed; reactor scram initiated; rod insertion starts 0.25 second later.
Vent clearing time	Top Vent: 2.0 seconds, Middle Vent: 2.6 seconds, Bottom Vent 3.4 seconds.
3	IC initiated from loss of power generation bus with 3 seconds signal delay time, drain valves starts to open 15 seconds later.
6	Level 3 is reached (scram signal from Level 3 is not credited in this analysis).
11	Level 2 is reached (MSIV closure and ICS initiation signals are not credited in this analysis).
19	Low MSL pressure setpoint reached, MSIV closure initiated at 0.7 second later.
23	Reactor isolated on low MSL pressure setpoint.
350	Level 1 is reached.
360	Level 1 signal confirmed; ADS/GDCS/SLC timer initiated; SRV actuated.
410	DPV actuation begins at 50 seconds after confirmed Level 1 signal; SLC flow starts on DPV actuation.
510	GDCS timer (150 seconds after confirmed Level 1 signal) times out. GDCS injection valves open.
579	Vessel pressure decreases below maximum injection pressure of GDCS. GDCS flow into the vessel begins. Chimney and downcomer water levels start to rise.
711	SLC flow depleted.
122600 (~ 34 hrs)	PCCS pool drops below the elevation of 29.6 m (97.1 ft); 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 IC/PCCS expansion pools.

Table 6.2-7d

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

Time (sec)	**Events**
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 367.9 kPa (53.4 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 ICS.
<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: 1.5 seconds, Middle Vent: 1.9 seconds, Bottom Vent: 2.5 seconds.
3	ICS initiated from loss of power generation bus with 3 seconds signal delay time, drain valves start to open 15 seconds later.
8	Level 3 is reached (scram signal from Level 3 is not credited in this analysis).
8	Low MSL pressure setpoint reached, MSIV closure initiated at 0.7 second later.
12	Reactor isolated on low MSL pressure setpoint.
20	Level 2 is reached (MSIV closure and ICS initiation signals are not credited in this analysis).
491	Level 1 is reached.
501	Level 1 signal confirmed; ADS/GDCS/SLC timer initiated; SRV actuated.
551	DPV actuation begins at 50 seconds after confirmed Level 1 signal; SLC flow starts on DPV actuation.
651	GDCS timer (150 seconds after confirmed Level 1 signal) times out. GDCS injection valves open.
698	Vessel pressure decreases below maximum injection pressure of GDCS. GDCS flow into the vessel begins. Chimney and downcomer water levels start to rise.
853	SLC flow depleted.
121203 (~ 33.6 hrs)	PCCS pool drops below the elevation of 29.6 m (97.1 ft.); top ¼ portion of the PCCS tube length becomes uncovered; connection valves open to allow the water from the equipment storage pool to flow into the IC/PCCS 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 396.25 kPa (57.47 psia).

Table 6.2-7f
Operational Sequence of ECCS for a Feedwater Line Break with Failure of one SRV (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 ICS.
<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: 2.11 seconds, Middle Vent: 2.63 seconds, Bottom Vent: 3.48 seconds.
3	ICS initiated from loss of power generation bus with 3 seconds signal delay time, drain valves start to open 15 seconds later.
6	Level 3 is reached (scram signal from Level 3 is not credited in this analysis).
11	Level 2 is reached (MSIV closure and ICS initiation signals are not credited in this analysis).
19	Low MSL pressure setpoint reached, MSIV closure initiated at 0.7 second later.
23	Reactor isolated on low MSL pressure setpoint.
351	Level 1 is reached.
361	Level 1 signal confirmed; ADS/GDCS/SLC timer initiated; SRV actuated.
411	DPV actuation begins at 50 seconds after confirmed Level 1 signal; SLC flow starts on DPV actuation.
511	GDCS timer (150 seconds after confirmed Level 1 signal) times out. GDCS injection valves open.
571	Vessel pressure decreases below maximum injection pressure of GDCS. GDCS flow into the vessel begins. Chimney and downcomer water levels start to rise.
712	SLC flow depleted.
122918 (~ 34 hrs)	PCCS pool drops below the elevation of 29.6 m (97.1 ft); 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 IC/PCCS expansion pools.

Table 6.2-7f

Operational Sequence of ECCS for a Feedwater Line Break with Failure of one SRV (Bounding Case)

Time (sec)	**Events**
From ~720 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 369.6 kPa (53.61 psia).

Table 6.2-7g

Operational Sequence of ECCS for a Main Steam Line Break with Failure of One SRV(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 ICS.
<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: 1.52 seconds, Middle Vent: 1.90 seconds, Bottom Vent: 2.50 seconds.
3	IC initiated from loss of power generation bus with 3 seconds signal delay time, drain valves start to open 15 seconds later.
8	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).
12	Reactor isolated on low MSL pressure setpoint.
20	Level 2 is reached (MSIV closure and IC initiation signals are not credited in this analysis).
491	Level 1 is reached.
501	Level 1 signal confirmed; ADS/GDCS/SLC timer initiated; SRV actuated.
551	DPV actuation begins at 50 seconds after confirmed Level 1 signal; SLC flow starts on DPV actuation.
651	GDCS timer (150 seconds after confirmed Level 1 signal) times out. GDCS injection valves open.
692	Vessel pressure decreases below maximum injection pressure of GDCS. GDCS flow into the vessel begins. Chimney and downcomer water levels start to rise.
852	SLC flow depleted.
121401 (~ 33.7 hrs)	PCCS pool drops below the elevation of 29.6 m (97.1 ft); 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 IC/PCCS expansion pools.

Table 6.2-7g

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

Time (sec)	**Events**
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 397.45 kPa (57.65 psia).

Table 6.2-7h

Operational Sequence of ECCS for a Main Steam Line Break with Failure of One SRV (Bounding Case, with Offsite Power)

Time (sec)	Events
0	Guillotine break of main steam line inside containment.
<1	Drywell Pressure High setpoint is reached, reactor scram initiated; rod insertion starts 0.25 second later.
≤1	Drywell Pressure High-High setpoint is reached; feedwater isolation occurs 15 seconds later.
Vent clearing time	Top Vent: 1.52 seconds, Middle Vent: 1.91 seconds, Bottom Vent 2.52 seconds.
9	Low MSL pressure setpoint reached. MSIV closure initiated at 0.7 second later.
10	Level 3 is reached (scram signal from Level 3 is not credited in this analysis).
13	Reactor isolated on low MSL pressure setpoint. (1)
16	Feedwater isolated on Drywell pressure High-High.
23	Level 2 is reached (MSIV closure and IC initiation signals are not credited in this analysis). HP CRD injection begins on Level 2.
890	Level 1 is reached.
900	Level 1 signal confirmed; ADS/GDCS/SLC timer initiated; SRV actuated.
950	DPV actuation begins at 50 seconds after confirmed Level 1 signal; SLC flow starts on DPV actuation.
1050	GDCS timer (150 seconds after confirmed Level 1 signal) times out. GDC injection valves open.

Table 6.2-7h

Operational Sequence of ECCS for a Main Steam Line Break with Failure of One SRV (Bounding Case, with Offsite Power)

Time (sec)	Events
1053	Vessel pressure decreases below maximum injection pressure of GDCS. GDCS flow into the vessel begins. Chimney and downcomer water levels start to rise.
1250	SLC flow depleted.
2178	HP CRD terminated on low GDCS Pool Level
123763 (~34.4 hrs)	PCCS pool drops below the elevation of 29.6 m (97.1 ft); 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 IC/PCCS expansion pools.
From ~0 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 394.12 kPa (57.16 psia).

(1) TRACG analysis assumes ICS initiation at 3 sec, it actually initiates at 13 sec on MSIV closure. This difference has no effect on the long term containment conditions

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	Critical Flow* (PIRT84)	1.0	Normal	9.5%	- 2 sigma	0.81
2	Decay Heat Multiplier	1.0	Normal	~0.05	+ 2 sigma	Decay Heat + 2 sigma
3	Surface Heat Transfer** (PIRT07)	100	Uniform	1 to 200	Lower bound	1
4	Passive Containment Cooling inlet Loss (k/A^2)	1065 m^{-4} 9.192 ft^{-4}	Normal	260.0 m^{-4} 2.244 ft^{-4}	+ 2 sigma	1585 m^{-4} 13.68 ft^{-4}
5	Passive Containment Cooling Heat Transfer (PIRT78)	1.0	Normal	7.9% (bias – 6.0%)	- 2 sigma	0.902
6	Vacuum Breaker Loss (k/A^2)	747 m^{-4} 6.45 ft^{-4}	Normal	93.6 m^{-4} 0.81 ft^{-4}	+ 2 sigma	934.1 m^{-4} 8.06 ft^{-4}

* A model multiplier (PIRT84) of 1.0 is applied to the critical flow for the nominal cases for all breaks. The choked flow uncertainty in the TRACG model (Reference 6.2-1) has been determined to be 1 sigma (9.5%) uncertainty, or + 2 sigma (+0.19) variance. For long-term containment pressure, uncertainty analysis shows that a smaller critical flow multiplier results in higher DW pressure. Hence for all bounding cases, a smaller model multiplier (PIRT84) of 0.81 (=1-0.19) is applied to the critical flow model to account for the lower uncertainty range (-2 sigma value) for the choked flow.

** Free surface to vapor heat transfer in WW.

Table 6.2-9

ESBWR Design Features for Severe Accident Control

Design Feature	Function: Prevention / Mitigation	Purpose/Description
IC	Prevention	Controls reactor pressure. First line of defense against accidents.
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 DW kept dry	Mitigation	Containment isolation with minimum leakage. High retention of aerosols. Fuel-Coolant Interactions and Ex-Vessel Steam Explosions minimized.
Lower DW configuration	Mitigation	Lower DW 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.
PCCS heat exchangers	Mitigation	Filter aerosols - minimize offsite dose.
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 DW airspace volume to handle noncondensable gas release in severe accident.
Inerted containment with nitrogen	Prevention /Mitigation	Prevents hydrogen detonation within drywell and wetwell.
GDCS Spillover Pipes	Prevention	Prevents steam explosion.

Table 6.2-10

Passive Containment Cooling Design Parameters

Number of PCCS Condensers	Six (6)
Heat Removal Capacity for Each Condenser	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 (216°F).
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	Break Size mm (in)
1	Break in RWCU/SDC NRHX Room	300 (12)
2	Break in NRHX Valve Room	150 (6)
3	Break in Regenerative Heat Exchanger Room	150 (6)
4	Break in RWCU/SDC Pump Rooms	200 (8)
5	Break in RWCU/SDC Filter/Demineralizer Room	150 (6)

Table 6.2-12
Subcompartment Vent Path Designation
(SI Units)

Flow Path No.	Type	Cell From	Cell To	P (m)	DH (m)	Lp/ DH	T	K FORW	K REVER	K AVERA	K CONTAIN	Flow Condition	Flow Direction	Blow Out Pressure (kPaG)	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	BOTH	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	BLOW OUT PANEL	12	17	9.00	2.22	0.45	1.05	2.25	1.78	2.25	1.13	SUBSONIC	FORWARD	16.547	TO ATMOSPHERE
15	BLOW OUT PANEL	13	17	9.00	2.22	0.45	1.05	2.25	1.78	2.25	1.13	SUBSONIC	FORWARD	16.547	TO ATMOSPHERE
16	BLOW OUT PANEL	14	17	9.00	2.22	0.45	1.05	2.25	1.78	2.25	1.13	SUBSONIC	FORWARD	16.547	TO ATMOSPHERE

Table 6.2-12
Subcompartment Vent Path Designation
(SI Units)

Flow Path No.	Type	Cell From	Cell To	P (m)	DH (m)	Lp/ DH	T	K FORW	K REVER	K AVERA	K CONTAIN	Flow Condition	Flow Direction	Blow Out Pressure (kPaG)	Comments
17	BLOW OUT PANEL	15	17	9.00	2.22	0.45	1.05	2.25	1.78	2.25	1.13	SUBSONIC	FORWARD	16.547	TO ATMOSPHERE
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
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	DELETED														
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
(SI Units)

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	17	1.00E+08	5.00	19.50	5.00	1.00	99999.00	100.00	0.24	0.86	31.00	31.00
15	13	197	17	1.00E+08	5.00	19.50	5.00	1.00	99999.00	100.00	0.24	0.86	31.00	31.00
16	14	197	17	1.00E+08	5.00	19.50	5.00	1.00	99999.00	100.00	0.24	0.86	31.00	31.00
17	15	197	17	1.00E+08	5.00	19.50	5.00	1.00	99999.00	100.00	0.24	0.86	31.00	31.00
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
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

Table 6.2-12 (Continued)
Subcompartment Vent Path Designation
(SI Units)

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)
22	DELETED													
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

F0 – Path Area
F1 – Cross-Section Area of From Cell
F2 – Cross-Section Area of To Cell
P – Path Perimeter
L1 – Length of From Cell
L2 – Length of To Cell
Lp – Passage Length
DH – Hydraulic Diameter
T – Flow Coefficient per Diagram 4-11 of Idel'Chik Handbook (Reference 6.2-12)
K FORW. – Direct Loss Pressure Coefficient per Diagram 4-11 of Idel'Chik Handbook (Reference 6.2-12)
K REVER. – Inverse Loss Pressure Coefficient per Diagram 4-11 of Idel'Chik Handbook (Reference 6.2-12)
K AVERA. – (K FORW.+K REVER.)/2
K CONTAIN – K AVERA./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 Subsection 4.2.1 of SMSAB-02-04, CONTAIN Code Qualification Report/User Guide for Auditing Subcompartment Analysis Calculations).

Table 6.2-12 (Continued)
Subcompartment Vent Path Designation
(English Units)

Flow Path No.	TYPE	CELL FROM	CELL TO	P (ft)	DH (ft)	Lp/DH	T	K FORW.	K REVER.	K AVERA.	K CONTAIN	FLOW COND.	FLOW DIRECTION	BLOW-OUT PRES (psig)	COMMENTS
1	DOOR	1	2	26.25	6.56	1.00	0.24	1.56	1.61	1.58	0.79	SUBSONIC	BOTH	NO	TWO WAY PATH
2	DOOR	2	3	26.25	6.56	0.50	0.97	1.51	1.24	1.38	0.69	SUBSONIC	FORWARD	1.5	
3	DOOR	2	3	26.25	6.56	0.50	0.97	1.52	1.26	1.39	0.70	SUBSONIC	FORWARD	1.5	
4	DOOR	3	4	26.25	6.56	0.35	1.13	1.25	1.24	1.24	0.62	SUBSONIC	FORWARD	1.5	
5	DOOR	3	5	26.25	6.56	0.25	1.19	1.31	1.32	1.31	0.66	SUBSONIC	FORWARD	1.5	
6	DOOR	6	7	26.25	6.56	1.00	0.24	1.56	1.61	1.58	0.79	SUBSONIC	BOTH	NO	TWO WAY PATH
7	DOOR	7	5	26.25	6.56	0.50	0.97	1.52	1.26	1.39	0.70	SUBSONIC	FORWARD	1.5	
8	DOOR	7	5	26.25	6.56	0.50	0.97	1.51	1.24	1.38	0.69	SUBSONIC	FORWARD	1.5	
9	DOOR	8	4	26.25	6.56	1.00	0.24	1.43	1.47	1.45	0.72	SUBSONIC	FORWARD	1.5	
10	DOOR	9	10	26.25	6.56	1.00	0.24	1.49	1.48	1.49	0.74	SUBSONIC	FORWARD	1.5	
11	DOOR	10	5	26.25	6.56	0.35	1.13	1.25	1.24	1.24	0.62	SUBSONIC	FORWARD	1.5	
12	DOOR	10	4	26.25	6.56	0.25	1.19	1.24	1.24	1.24	0.62	SUBSONIC	FORWARD	1.5	
13	DELETED														
14	BLOW OUT PANEL	12	17	29.53	7.28	0.45	1.05	2.25	1.78	2.25	1.13	SUBSONIC	FORWARD	2.4	TO ATMOSPHERE
15	BLOW OUT PANEL	13	17	29.53	7.28	0.45	1.05	2.25	1.78	2.25	1.13	SUBSONIC	FORWARD	2.4	TO ATMOSPHERE
16	BLOW OUT PANEL	14	17	29.53	7.28	0.45	1.05	2.25	1.78	2.25	1.13	SUBSONIC	FORWARD	2.4	TO ATMOSPHERE

Table 6.2-12 (Continued)
Subcompartment Vent Path Designation
(English Units)

Flow Path No.	TYPE	CELL FROM	CELL TO	P (ft)	DH (ft)	Lp/DH	T	K FORW.	K REVER.	K AVERA.	K CONTAIN	FLOW COND.	FLOW DIRECTION	BLOW-OUT PRES (psig)	COMMENTS
17	BLOW OUT PANEL	15	17	29.53	7.28	0.45	1.05	2.25	1.78	2.25	1.13	SUBSONIC	FORWARD	2.4	TO ATMOSPHERE
18	OPEN SPACE	3	12	32.81	6.56	0.50	0.97	0.47	0.90	0.69	0.34	SUBSONIC	BOTH	NO	TWO WAY PATH
19	OPEN SPACE	5	14	32.81	6.56	0.50	0.97	0.48	0.93	0.71	0.35	SUBSONIC	BOTH	NO	TWO WAY PATH
20	OPEN SPACE	10	15	32.81	6.56	0.50	0.97	0.48	0.93	0.70	0.35	SUBSONIC	BOTH	NO	TWO WAY PATH
21	OPEN SPACE	4	13	32.81	6.56	0.50	0.97	0.47	0.90	0.69	0.34	SUBSONIC	BOTH	NO	TWO WAY PATH
22	DELETED														
23	HATCH	11	18	44.62	11.15	0.29	1.16	2.09	1.41	1.75	0.87	SUBSONIC	FORWARD	1.5	
24	DOOR	18	19	26.25	6.56	0.30	1.16	1.97	1.97	1.97	0.98	SUBSONIC	FORWARD	1.5	
25	DOOR	19	20	26.25	6.56	0.15	1.25	2.07	2.07	2.07	1.04	SUBSONIC	FORWARD	1.5	
26	DOOR	20	5	26.25	6.56	1.00	0.24	1.22	0.82	1.02	0.51	SUBSONIC	FORWARD	1.5	
27	DOOR	21	20	26.25	6.56	0.15	1.25	2.07	2.07	2.07	1.04	SUBSONIC	FORWARD	1.5	
28	DOOR	21	22	26.25	6.56	0.30	1.16	1.97	1.97	1.97	0.98	SUBSONIC	FORWARD	1.5	
29	DOOR	18	22	26.25	6.56	0.15	1.25	2.07	2.07	2.07	1.04	SUBSONIC	FORWARD	1.5	
30	DOOR	22	23	26.25	6.56	1.00	0.24	1.49	1.63	1.56	0.78	SUBSONIC	FORWARD	1.5	
31	DOOR	23	24	26.25	6.56	1.00	0.24	1.59	1.48	1.53	0.77	SUBSONIC	FORWARD	1.5	

Table 6.2-12 (Continued)
Subcompartment Vent Path Designation
(English Units)

Flow Path No.	CELL FROM	VOLUME (ft^3)	CELL TO	VOLUME (ft^3)	F1 (ft^2)	L1 (ft)	F0 (ft^2)	Lp (ft)	F2 (ft^2)	L2 (ft)	A/L per ANSI/ANS-56.10-1982	A/L per SMSAB-02-04	EL. IN (ft)	EL. OUT (ft)
1	1	12289.52	2	9605.6	1141.3	6.56	43.06	6.56	585.34	8.2	5.81	2.03	-34.45	-34.45
2	2	9570.28	3	11795.11	103.76	16.4	43.06	3.28	208.5	17.06	3.15	2.03	-34.45	-34.45
3	2	9570.28	3	11795.11	105.92	18.04	43.06	3.28	208.5	10.5	3.38	2.03	-34.45	-34.45
4	3	11795.11	4	16668.54	111.94	36.09	43.06	2.3	114.42	36.09	1.44	1.90	-34.45	-34.45
5	3	11795.11	5	12077.63	118.08	31.17	43.06	1.64	115.5	39.37	1.54	1.90	-34.45	-34.45
6	6	12466.09	7	9570.28	1141.3	6.56	43.06	6.56	585.34	8.2	5.81	2.03	-34.45	-34.45
7	7	9570.28	5	12077.63	105.92	18.04	43.06	3.28	208.5	16.4	3.08	2.03	-34.45	-34.45
8	7	9570.28	5	12077.63	103.76	18.04	43.06	3.28	208.5	13.12	3.18	2.03	-34.45	-34.45
9	8	5332.52	4	16668.54	471.35	3.28	43.06	6.56	354.35	8.2	5.48	2.46	-34.45	-34.45
10	9	5332.52	10	18328.33	457.47	3.28	43.06	6.56	492.88	8.2	5.68	2.46	-34.45	-34.45
11	10	18328.33	5	12077.63	111.94	36.09	43.06	2.3	114.42	31.17	1.54	1.87	-34.45	-34.45
12	10	18328.33	4	16668.54	110.33	37.73	43.06	1.64	110.33	37.73	1.38	1.67	-34.45	-34.45
13	DELETED												0	0
14	12	6957	17	3.53E+09	53.82	63.98	53.82	3.28	1076380.24	328.08	0.79	2.82	101.71	101.71
15	13	6957	17	3.53E+09	53.82	63.98	53.82	3.28	1076380.24	328.08	0.79	2.82	101.71	101.71
16	14	6957	17	3.53E+09	53.82	63.98	53.82	3.28	1076380.24	328.08	0.79	2.82	101.71	101.71
17	15	6957	17	3.53E+09	53.82	63.98	53.82	3.28	1076380.24	328.08	0.79	2.82	101.71	101.71
18	3	11795.11	12	6957	1031.83	6.56	53.82	3.28	53.82	63.98	0.79	2.82	-24.28	-21
19	5	12077.63	14	6957	1603.07	6.56	53.82	3.28	53.82	63.98	0.79	2.82	-24.28	-21
20	10	18328.33	15	6957	1457.54	6.56	53.82	3.28	53.82	63.98	0.79	2.82	-24.28	-21

Table 6.2-12 (Continued)
Subcompartment Vent Path Designation
(English Units)

Flow Path No.	CELL FROM	VOLUME (ft^3)	CELL TO	VOLUME (ft^3)	F1 (ft^2)	L1 (ft)	F0 (ft^2)	Lp (ft)	F2 (ft^2)	L2 (ft)	A/L per ANSI/ANS-56.10-1982	A/L per SMSAB-02-04	EL. IN (ft)	EL. OUT (ft)
21	4	16668.54	13	6957	1056.26	6.56	53.82	3.28	53.82	63.98	0.79	2.82	-24.28	-21
22	DELETED												0	0
23	11	3319.58	18	16174.13	269.1	7.22	124.43	3.28	11610.49	5.91	18.64	8.37	-6.56	-3.28
24	18	16174.13	19	16174.13	247.57	39.37	43.06	1.97	247.57	39.37	2.76	1.71	0	0
25	19	16174.13	20	5403.15	261.24	26.25	43.06	0.98	261.24	3.61	7.28	2.46	0	0
26	20	5403.15	5	12077.63	86.11	9.19	43.06	6.56	454.45	4.1	3.74	2.46	-34.45	-34.45
27	21	16174.13	20	5403.15	261.24	26.25	43.06	0.98	261.24	3.28	7.35	2.46	0	0
28	21	16174.13	22	16174.13	247.57	39.37	43.06	1.97	247.57	39.37	2.76	1.71	0	0
29	18	16174.13	22	16174.13	261.24	26.25	43.06	0.98	261.24	26.25	4.46	1.71	0	0
30	22	16174.13	23	4308.39	2221.67	10.83	43.06	6.56	385.35	4.1	5.97	2.66	0	0
31	23	4308.39	24	1024126.3	385.35	4.1	43.06	6.56	1151.74	12.63	5.74	2.66	0	0

F0 – Path Area
F1 – Cross-Section Area of From Cell
F2 – Cross-Section Area of To Cell
P – Path Perimeter
Lp – Passage Length
L1 – Length of From Cell
L2 – Length of To Cell
DH – Hydraulic Diameter
T – Flow Coefficient per Diagram 4-11 of Idel'Chik Handbook (Reference 6.2-12)
K FORW. – Direct Loss Pressure Coefficient per Diagram 4-11 of Idel'Chik Handbook (Reference 6.2-12)
K REVER. – Inverse Loss Pressure Coefficient per Diagram 4-11 of Idel'Chik Handbook (Reference 6.2-12)
K AVERA. – (K FORW.+K REVER.)/2
K CONTAIN – K AVERA./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 Subsection 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 (ft^3)	Calculated Peak Pressure kPa G[1] (psig)	Initial Conditions		
							Pressure Pa (psia)	Temperature ºC (ºF)	Relative Humidity (%)
6.2-18	1	CASE 1 CASE 3	RWCU /Shutdown Cooling Heat Exchanger Room A	1151	348 (12290)	29.37 (4.26)	1.013e5 (14.7)	43 (109.4)	0
6.2-18	2	CASE 2	RWCU /Shutdown Cooling Valve Room A	1150	271 (9570)	22.22 (3.22)	1.013e5 (14.7)	43 (109.4)	0
6.2-18	3	NO	Corridor A El. -11500 mm	1100	334 (11795)	20.13 (2.93)	1.013e5 (14.7)	43 (109.4)	0
6.2-18	4	NO	Corridor B El. -11500 mm	1101	472 (16667)	19.48 (2.82)	1.013e5 (14.7)	43 (109.4)	0
6.2-18	5	NO	Corridor D El. -11500 mm	1103	342 (12078)	25.30 (3.67)	1.013e5 (14.7)	43 (109.4)	0
6.2-18	6	CASE 1 CASE 3	RWCU /Shutdown Cooling Heat Exchanger Room B	1161	353 (12466)	35.20 (5.11)	1.013e5 (14.7)	43 (109.4)	0
6.2-18	7	CASE 2	RWCU /Shutdown Cooling Valve Room B	1160	271 (9570)	27.61 (4)	1.013e5 (14.7)	43 (109.4)	0
6.2-18	8	CASE 4	RWCU /Shutdown Cooling Pump Room A	1152	151 (5333)	19.03 (2.76)	1.013e5 (14.7)	43 (109.4)	0
6.2-18	9	CASE 4	RWCU /Shutdown Cooling Pump Room B	1162	151 (5333)	18.92 (2.74)	1.013e5 (14.7)	43 (109.4)	0
6.2-18	10	NO	Corridor C El. -11500 mm	1102	519 (18328)	18.81 (2.73)	1.013e5 (14.7)	43 (109.4)	0
6.2-18	11	CASE 5	RWCU /Shutdown Cooling Filter/Demineralizer Vault A1	1251	94 (3320)	11.44 (1.66)	1.013e5 (14.7)	43 (109.4)	0
6.2-18	12	NO	HELB Vent Chase A		197 (6957)	19.80 (2.87)	1.013e5 (14.7)	43 (109.4)	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 (ft^3)	Calculated Peak Pressure kPa G[1] (psig)	Initial Conditions		
							Pressure Pa (psia)	Temperature °C (°F)	Relative Humidity (%)
6.2-18	13	NO	HELB Vent Chase B		197 (6957)	18.81 (2.73)	1.013e5 (14.7)	43 (109.4)	0
6.2-18	14	NO	HELB Vent Chase D		197 (6957)	22.0 (3.19)	1.013e5 (14.7)	43 (109.4)	0
6.2-18	15	NO	HELB Vent Chase C		197 (6957)	18.59 (2.70)	1.013e5 (14.7)	43 (109.4)	0
	16		DELETED						
6.2-18	17	NO	Atmosphere	-----	1.0E8 (3.5E9)	-----	1.013e5 (14.7)	40 (104)	0
6.2-18	18	NO	Filter/Demineralizer Access Room	1306	458 (16174)	11.0 (1.60)	1.013e5 (14.7)	43 (109.4)	0
6.2-18	19	NO	RWCU /Shutdown Cooling Heat Exchanger Access Room A	1304	458 (16174)	11.0 (1.60)	1.013e5 (14.7)	43 (109.4)	0
6.2-18	20	NO	Interior Stairwell A	1195	153 (5403)	11.22 (1.63)	1.013e5 (14.7)	43 (109.4)	0
6.2-18	21	NO	RWCU /Shutdown Cooling Heat Exchanger Access Room B	1305	458 (16174)	N/A[(2)]	1.013e5 (14.7)	43 (109.4)	0
6.2-18	22	NO	Control Rod Drive Pump Access Room	1307	458 (16174)	11.0 (1.60)	1.013e5 (14.7)	43 (109.4)	0
6.2-18	23	NO	Controlled Equipment Removal Access Room	1308	122 (4308)	11.0 (1.60)	1.013e5 (14.7)	43 (109.4)	0
6.2-18	24	NO	FUEL BUILDING	-----	29000 (1024125)	10.23 (1.48)	1.013e5 (14.7)	43 (109.4)	0

(1) Includes a 10% margin
(2) Not affected by HELB

Table 6.2-12b

(Deleted)

Table 6.2-12c
Heat Sink Descriptions
(SI Units)

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
	5	Floors Cell 3	Floor3	Floor	Slab	95.87	4	Concrete	43

Table 6.2-12c

Heat Sink Descriptions

(SI Units)

Cell	Heat Sink	Description	Name	Type	Shape	Surface (m^2)	Thickness (m)	Compound	Initial Temperature (ºC)
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
	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

Table 6.2-12c
Heat Sink Descriptions
(SI Units)

Cell	Heat Sink	Description	Name	Type	Shape	Surface (m^2)	Thickness (m)	Compound	Initial Temperature (ºC)
	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
	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

Table 6.2-12c
Heat Sink Descriptions
(SI Units)

Cell	Heat Sink	Description	Name	Type	Shape	Surface (m^2)	Thickness (m)	Compound	Initial Temperature (°C)
	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
	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

Table 6.2-12c
Heat Sink Descriptions
(SI Units)

Cell	Heat Sink	Description	Name	Type	Shape	Surface (m^2)	Thickness (m)	Compound	Initial Temperature (°C)
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	DELETED								
17	No heat sinks								
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

Table 6.2-12c
Heat Sink Descriptions
(SI Units)

Cell	Heat Sink	Description	Name	Type	Shape	Surface (m^2)	Thickness (m)	Compound	Initial Temperature (°C)
	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
	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

Table 6.2-12c
Heat Sink Descriptions
(SI Units)

Cell	Heat Sink	Description	Name	Type	Shape	Surface (m^2)	Thickness (m)	Compound	Initial Temperature (°C)
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
	3	Floors Cell 23	Floor23	Floor	Slab	30.92	1	Concrete	43
24	No heat sinks								

Table 6.2-12c (Continued)
Heat Sink Descriptions
(English Units)

Cell	Heat Sink	Description	Name	Type	Shape	Surface (ft^2)	Thickness (ft)	Compound	Initial Temperature (ºF)
1	1	Roof of 600 mm	Roof1	Roof	Slab	832.37	1.97	Concrete	109.4
	2	M1 wall of 2000 mm (25º)	S1-1	Wall	Slab	401.6	3.28	Concrete	109.4
	3	M2 wall of 2000 mm (47º)	S2-1	Wall	Slab	755.41	3.28	Concrete	109.4
	4	M3 of 1400 mm	S3-1	Wall	Slab	203.44	4.59	Concrete	109.4
	5	M4 of 1400 mm	S4-1	Wall	Slab	222.81	4.59	Concrete	109.4
	6	M5 Containment of 1400 mm	S5-1	Wall	Slab	813.43	4.59	Concrete	109.4
	7	Floors Cell 1	Floor1	Floor	Slab	1095.98	13.12	Concrete	109.4
2	1	Roof of 1000 mm	Roof2	Roof	Slab	836.79	3.28	Concrete	109.4
	2	Wall of 1000 mm common with cell 3	S1-2	Wall	Slab	977.69	1.64	Concrete	109.4
	3	Wall of 2.0 m common with V1 (arc of 26º)	S2-2	Wall	Slab	417.64	3.28	Concrete	109.4
	4	Floors Cell 2	Floor2	Floor	Slab	836.79	13.12	Concrete	109.4
3	1	Roof of 1000 mm	Roof3	Roof	Slab	1031.94	3.28	Concrete	109.4
	2	Wall of 2000 mm with outside	S1-3	Wall	Slab	2063.44	6.56	Concrete	109.4
	3	Wall of 1000 mm common with cell 2	S2-3	Wall	Slab	1108.25	1.64	Concrete	109.4
	4	Wall of 2.0 m common with V1 (arc of sum 39º)	S3-3	Wall	Slab	570.7	3.28	Concrete	109.4
	5	Floors Cell 3	Floor3	Floor	Slab	1031.94	13.12	Concrete	109.4
4	1	Roof of 1000 mm	Roof4	Roof	Slab	1304.16	3.28	Concrete	109.4
	2	Wall of 2000 mm with outside	S1-4	Wall	Slab	804.06	6.56	Concrete	109.4
	3	Wall of 2000 mm with fuel building	S2-4	Wall	Slab	926.77	6.56	Concrete	109.4
	4	Wall of 2000 mm common with cell 8	S3-4	Wall	Slab	292.67	3.28	Concrete	109.4
	5	Wall of 2000 mm common with sump pumps	S4-4	Wall	Slab	365.87	6.56	Concrete	109.4
	6	Wall of 750 mm common with valve room	S5-4	Wall	Slab	873.6	2.46	Concrete	109.4
	7	Floors Cell 4	Floor4	Floor	Slab	1457.54	13.12	Concrete	109.4
5	1	Roof of 1000 mm	Roof5	Roof	Slab	900.29	3.28	Concrete	109.4

Table 6.2-12c (Continued)
Heat Sink Descriptions
(English Units)

Cell	Heat Sink	Description	Name	Type	Shape	Surface (ft^2)	Thickness (ft)	Compound	Initial Temperature (ºF)
	2	Wall of 2000 mm with outside	S1-5	Wall	Slab	1951.6	6.56	Concrete	109.4
	3	Wall of 1000 mm with cell 7	S2-5	Wall	Slab	1117.19	1.64	Concrete	109.4
	4	Wall of 2000 mm common with cell 6	S3-5	Wall	Slab	570.7	3.28	Concrete	109.4
	5	Floors Cell 5	Floor5	Wall	Slab	1056.26	13.12	Concrete	109.4
6	1	Roof of 600 mm	Roof6	Roof	Slab	728.61	1.97	Concrete	109.4
	2	M1 wall of 2000 mm (arc of 25º)	S1-1	Wall	Slab	1156.47	3.28	Concrete	109.4
	3	M2 wall of 900 mm	S2-6	Wall	Slab	203.44	2.95	Concrete	109.4
	4	M3 of 1000 mm	S3-6	Wall	Slab	223.35	3.28	Concrete	109.4
	5	M5 Containment of 1400 mm	S4-6	Wall	Slab	894.05	4.59	Concrete	109.4
	6	Floors Cell 6	Floor6	Floor	Slab	992.11	13.12	Concrete	109.4
7	1	Roof of 1000 mm	Roof7	Roof	Slab	836.79	3.28	Concrete	109.4
	2	Wall of 1000 mm common with cell 5	S1-7	Wall	Slab	977.69	1.64	Concrete	109.4
	3	Wall of 2.0 m common with cell 2 (arc of 26º)	S2-7	Wall	Slab	417.64	3.28	Concrete	109.4
	4	Floors Cell 7	Floor7	Floor	Slab	836.79	13.12	Concrete	109.4
8	1	Roof of 1000 mm	Roof8	Roof	Slab	467.48	3.28	Concrete	109.4
	2	Wall of 250 mm	S1-8	Wall	Slab	464.25	0.82	Concrete	109.4
	3	Wall of 2.0 m common with V5 (arc of 20º)	S2-8	Wall	Slab	292.67	3.28	Concrete	109.4
	4	Wall of 2000 mm with internal room (arc of 11º)	S3-8	Wall	Slab	160.92	6.56	Concrete	109.4
	5	Containment of 600 mm	S4-8	Wall	Slab	370.06	1.97	Concrete	109.4
	6	Floors Cell 8	Floor8	Floor	Slab	467.48	13.12	Concrete	109.4
9	1	Roof of 1000 mm	Roof9	Roof	Slab	467.48	3.28	Concrete	109.4
	2	Wall of 250 mm	S1-9	Wall	Slab	464.25	0.82	Concrete	109.4
	3	Wall of 2.0 m common with cell 5 (arc of 20º)	S2-9	Wall	Slab	292.67	3.28	Concrete	109.4
	4	Wall of 2000 mm with internal room (arc of 11º)	S3-9	Wall	Slab	160.92	6.56	Concrete	109.4

Table 6.2-12c (Continued)
Heat Sink Descriptions
(English Units)

Cell	Heat Sink	Description	Name	Type	Shape	Surface (ft^2)	Thickness (ft)	Compound	Initial Temperature (°F)
	5	Containment Wall of 600 mm	S4-9	Wall	Slab	370.06	1.97	Concrete	109.4
	6	Floors Cell 9	Floor9	Floor	Slab	467.48	13.12	Concrete	109.4
10	1	Roof of 1000 mm	Roof10	Roof	Slab	1603.07	3.28	Concrete	109.4
	2	Wall of 2000 mm with outside and stairs	S1-10	Wall	Slab	1003.09	6.56	Concrete	109.4
	3	Wall of 2000 mm with fuel building	S2-10	Wall	Slab	836.36	6.56	Concrete	109.4
	4	Wall of 2000 mm common with cell 9/11	S3-10	Wall	Slab	614.62	3.28	Concrete	109.4
	5	Wall of 750 mm common with valve room	S4-10	Wall	Slab	887.38	2.46	Concrete	109.4
	6	Floors Cell 10	Floor10	Floor	Slab	1603.07	13.12	Concrete	109.4
11	1	M1 wall of 2000 mm	S1-11	Wall	Slab	306.56	6.56	Concrete	109.4
	2	M2 wall of 700 mm	S2-11	Wall	Slab	208.39	2.3	Concrete	109.4
	3	M3 wall of 1500 mm	S3-11	Wall	Slab	231.53	4.92	Concrete	109.4
	4	M4 of 1000 mm	S4-11	Wall	Slab	208.39	3.28	Concrete	109.4
	5	Floors Cell 11	Floor11	Floor	Slab	269.1	3.28	Concrete	109.4
12	1	Wall of 2000 mm with outside	S1-16	Wall	Slab	921.82	6.56	Concrete	109.4
	2	Wall of 1500 mm with outside	S2-16	Wall	Slab	2104.77	4.92	Concrete	109.4
	3	Wall of 1000 mm with outside	S3-16	Wall	Slab	516.67	3.28	Concrete	109.4
13	1	Wall of 2000 mm with outside	S1-13	Wall	Slab	921.82	6.56	Concrete	109.4
	2	Wall of 1500 mm with outside	S2-13	Wall	Slab	2104.77	4.92	Concrete	109.4
	3	Wall of 1000 mm with outside	S3-13	Wall	Slab	516.67	3.28	Concrete	109.4
14	1	Wall of 2000 mm with outside	S1-14	Wall	Slab	921.82	6.56	Concrete	109.4
	2	Wall of 1500 mm with outside	S2-14	Wall	Slab	2104.77	4.92	Concrete	109.4
	3	Wall of 1000 mm with outside	S3-14	Wall	Slab	516.67	3.28	Concrete	109.4
15	1	Wall of 2000 mm with outside	S1-15	Wall	Slab	921.82	6.56	Concrete	109.4
	2	Wall of 1500 mm with outside	S2-15	Wall	Slab	2104.77	4.92	Concrete	109.4

Table 6.2-12c (Continued)
Heat Sink Descriptions
(English Units)

Cell	Heat Sink	Description	Name	Type	Shape	Surface (ft^2)	Thickness (ft)	Compound	Initial Temperature (°F)
	3	Wall of 1000 mm with outside	S3-15	Wall	Slab	516.67	3.28	Concrete	109.4
16	DELETED								
17	No heat sinks								
18	1	Roof of 2000 mm	Roof18	Roof	Slab	1590.48	6.56	Concrete	109.4
	2	Wall of 2000 mm	S1-18	Wall	Slab	1028.38	6.56	Concrete	109.4
	3	Wall of 600 mm	S2-18	Wall	Slab	735.28	1.97	Concrete	109.4
	4	Floors Cell 18	Floor18	Floor	Slab	1590.48	1.97	Concrete	109.4
19	1	Roof of 2000 mm	Roof19	Roof	Slab	1590.48	6.56	Concrete	109.4
	2	Wall of 2000 mm	S1-19	Wall	Slab	1028.38	6.56	Concrete	109.4
	3	Wall of 600 mm	S2-19	Wall	Slab	735.28	1.97	Concrete	109.4
	4	Floors Cell 19	Floor19	Floor	Slab	1590.48	1.97	Concrete	109.4
20	1	Roof of 2000 mm	Roof20	Roof	Slab	158.66	6.56	Concrete	109.4
	2	Wall of 1400 mm	S1-20	Wall	Slab	1229.67	2.3	Concrete	109.4
	3	Wall of 300 mm	S2-20	Wall	Slab	526.46	0.49	Concrete	109.4
	4	Wall of 2000 mm	S3-20	Wall	Slab	327.98	6.56	Concrete	109.4
	5	Floors Cell 20	Floor20	Floor	Slab	132.61	14.11	Concrete	109.4
21	1	Roof of 2000 mm	Roof21	Roof	Slab	1590.48	6.56	Concrete	109.4
	2	Wall of 2000 mm	S1-21	Wall	Slab	1028.38	6.56	Concrete	109.4
	3	Wall of 600 mm	S2-21	Wall	Slab	735.28	1.97	Concrete	109.4
	4	Floors Cell 21	Floor21	Floor	Slab	1590.48	1.97	Concrete	109.4
22	1	Roof of 2000 mm	Roof22	Roof	Slab	1590.48	6.56	Concrete	109.4
	2	Wall of 2000 mm	S1-22	Wall	Slab	1028.38	6.56	Concrete	109.4
	3	Wall of 600 mm	S2-22	Wall	Slab	735.28	1.97	Concrete	109.4
	4	Floors Cell 22	Floor22	Floor	Slab	1590.48	1.97	Concrete	109.4

Table 6.2-12c (Continued)
Heat Sink Descriptions
(English Units)

Cell	Heat Sink	Description	Name	Type	Shape	Surface (ft^2)	Thickness (ft)	Compound	Initial Temperature (°F)
23	1	Roof of 1000 mm	Roof23	Roof	Slab	332.82	3.28	Concrete	109.4
	2	Wall of 2000 mm common with other rooms	S1-23	Wall	Slab	335.3	3.28	Concrete	109.4
	3	Floors Cell 23	Floor23	Floor	Slab	332.82	3.28	Concrete	109.4
24	No heat sinks								

Table 6.2-12d

RPV Sensible Heat Data

Item	Mass, kg (lb)	C_p ,J/kg-K (Btu/lbm-°F)	TRACG Vessel Location (See Figures 6.2-6, 6.2-7)
Bottom Head	122016 (268999)	515 (0.123)	L1, R1, 2, 3, 4
Cylinder	823810 (1816190)	515 (0.123)	L2:L20, R4
Top Head	142473 (314099)	515 (0.123)	L21, R1, 2, 3, 4

Table 6.2-13

Reactor Coolant Pressure Boundary Influent Lines Penetrating Drywell

Influent Line		Inside Drywell	Outside Drywell
1	Feedwater	CV	(1) POV (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.
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 6.2-45

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

a_1	=	Reactor Building
a_2	=	Main Steam Tunnel

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

b_1	=	Pool open to reactor building
b_2	=	External environment
b_3	=	Main Condenser
b_4	=	Isolation Condenser pool
b_5	=	Reactor building
b_6	=	Closed loop outside containment
b_7	=	Radwaste System
b_8	=	Feedwater System

(c) (Deleted)

(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/PCCS pool high radiation
K	IC lines high flow
L	Low main condenser vacuum
M	High differential mass flow in the RWCU/SDC train
N	Standby Liquid Control System operating
P	Remote manual
Q	Process actuated
R	Local manual (By Hand)
T	High HVAC radiation exhaust from refueling area or from Reactor Building.
U	Feedwater lines differential pressure high
W	Containment water level high
X	Reactor vessel low-low water level – Level 0.5
Y	DW pressure High-High

(e) (Deleted)

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*	750 mm (30 in)	750 mm (30 in)	50 mm (2 in)
Type C Leakage Test	Yes	Yes	Yes
(Deleted)			
Leakage Through Packing[(a)]	N/A	a_2	a_2
Leakage Past Seat[(b)]	b_3	b_3	b_3
Location	Inboard	Outboard	Outboard
(Deleted)			
(Deleted)			
Normal Position	Open	Open	Open
Shutdown Position	Closed	Closed	Open
Post-Accident Position	Closed	Closed	Open/Closed
Power Fail Position	Closed	Closed	Closed
Containment Isolation 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, 3	Div. 2, 4	Div. 1, 2, 3

* Nominal pipe size diameter

Note: For explanation of codes, see legend in Table 6.2-15. See Table 3.9-8 for valve and actuator types.

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*	750 mm (30 in)	750 mm (30 in)	50 mm (2 in)
Type C Leakage Test	Yes	Yes	Yes
(Deleted)			
Leakage Through Packing[(a)]	N/A	a_2	a_2
Leakage Past Seat[(b)]	b_3	b_3	b_3
Location	Inboard	Outboard	Outboard
(Deleted)			
(Deleted)			
Normal Position	Open	Open	Open
Shutdown Position	Closed	Closed	Open
Post-Accident Position	Closed	Closed	Open/Closed
Power Fail Position	Closed	Closed	Closed
Containment Isolation 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, 3	Div. 2, 4	Div. 1, 2, 3

* Nominal pipe size diameter

Note: For explanation of codes, see legend in Table 6.2-15. See Table 3.9-8 for valve and actuator types.

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*	750 mm (30 in)	750 mm (30 in)	50 mm (2 in)
Type C Leakage Test	Yes	Yes	Yes
(Deleted)			
Leakage Through Packing[(a)]	N/A	a_2	a_2
Leakage Past Seat[(b)]	b_3	b_3	b_3
Location	Inboard	Outboard	Outboard
(Deleted)			
(Deleted)			
Normal Position	Open	Open	Open
Shutdown Position	Closed	Closed	Open
Post-Accident Position	Closed	Closed	Open/Closed
Power Fail Position	Closed	Closed	Closed
Containment Isolation 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, 3	Div. 2, 4	Div. 1, 2, 3

* Nominal pipe size diameter

Note: For explanation of codes, see legend in Table 6.2-15. See Table 3.9-8 for valve and actuator types.

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*	750 mm (30 in)	750 mm (30 in)	50 mm (2 in)
Type C Leakage Test	Yes	Yes	Yes
(Deleted)			
Leakage Through Packing[(a)]	N/A	a_2	a_2
Leakage Past Seat[(b)]	b_3	b_3	b_3
Location	Inboard	Outboard	Outboard
(Deleted)			
(Deleted)			
Normal Position	Open	Open	Open
Shutdown Position	Closed	Closed	Open
Post-Accident Position	Closed	Closed	Open/Closed
Power Fail Position	Closed	Closed	Closed
Containment Isolation 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, 3	Div. 2, 4	Div. 1, 2, 3

* Nominal pipe size diameter

Note: For explanation of codes, see legend in Table 6.2-15. See Table 3.9-8 for valve and actuator types.

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 (3 in)	80 mm (3 in)
Type C Leakage Test	Yes	Yes
(Deleted)		
Leakage Through Packing[(a)]	N/A	a_2
Leakage Past Seat[(b)]	b_3	b_3
Location	Inboard	Outboard
(Deleted)		
(Deleted)		
Normal Position	Open	Open
Shutdown Position	Open	Open
Post-Accident Position	Closed	Closed
Power Fail Position	Closed	Closed
Containment Isolation 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

* Nominal pipe size diameter

Note: For explanation of codes, see legend in Table 6.2-15. See Table 3.9-8 for valve and actuator types.

Table 6.2-21

Containment Isolation Valve Information for the Nuclear Boiler System Feedwater Line A

Penetration Identification	B21-MPEN-0006			
Valve No.	**F102A**	**F100A**	**F101A**	**F111A**
Applicable Basis	GDC 55	GDC 55	GDC 55	GDC 55
Tier 2 Figure	5.1-2	5.1-2	5.1-2	5.1-2
ESF	No	No	No	No
Fluid	Water	Water	Water	Water
Line Size*	550 mm (22 in)	550 mm (22 in)	550 mm (22 in)	300 mm (12 in)
Type C Leakage Test	Yes	Yes	Yes	Yes
(Deleted)				
Leakage Through Packing[a]	N/A	a_2	a_2	a_2
Leakage Past Seat[b]	b_8	b_8	b_8	b_5
Location	Inboard	Outboard	Outboard	Outboard
(Deleted)				
(Deleted)				
Normal Position	Open	Open	Open	Open
Shutdown Position	Open/Closed	Closed	Closed	Open/Closed
Post-Accident Position	Open/Closed	Closed	Closed	Open/Closed
Power Fail Position	N/A	Closed	Closed	Closed
Containment Isolation Signal[d]	Q	U+H, W+H[1], X,	U+H, W+H[1], X,	Q

Table 6.2-21

Containment Isolation Valve Information for the Nuclear Boiler System Feedwater Line A

Penetration Identification	B21-MPEN-0006			
Valve No.	**F102A**	**F100A**	**F101A**	**F111A**
		Y(1)	Y(1)	
Primary Actuation	Flow to open/close	Auto-closed	Auto-closed	Flow to open/close
Secondary Actuation	N/A	Remote manual	Remote manual	N/A
Closure Time (sec)	N/A	N/A on reverse-flow, < 15 sec on auto-isolation	N/A on reverse-flow, < 15 sec on auto-isolation	N/A
Power Source	N/A	Div. 1, 3	Div. 2, 4	N/A

(1) Isolates coolant in-flow to containment
* Nominal pipe size diameter
Note: For explanation of codes, see legend on Table 6.2-15. See Table 3.9-8 for valve and actuator types.

Table 6.2-22

Containment Isolation Valve Information for the Nuclear Boiler System Feedwater Line B

Penetration Identification	**B21-MPEN-0007**			
Valve No.	**F102B**	**F100B**	**F101B**	**F111B**
Applicable Basis	GDC 55	GDC 55	GDC 55	GDC 55
Tier 2 Figure	5.1-2	5.1-2	5.1-2	5.1-2
ESF	No	No	No	No
Fluid	Water	Water	Water	Water
Line Size*	550 mm (22 in)	550 mm (22 in)	550 mm (22 in)	300 mm (12 in)
Type C Leakage Test	Yes	Yes	Yes	Yes
(Deleted)				
Leakage Through Packing[(a)]	N/A	a_2	a_2	a_2
Leakage Past Seat[(b)]	b_8	b_8	b_8	b_5
Location	Inboard	Outboard	Outboard	Outboard
(Deleted)				
(Deleted)				
Normal Position	Open	Open	Open	Open
Shutdown Position	Open/Closed	Closed	Closed	Open/Closed
Post-Accident Position	Open/Closed	Open/Closed	Open/Closed	Open/Closed
Power Fail Position	N/A	Closed	Closed	Closed
Containment Isolation Signal[(d)]	Q	U+H, W+H[(1)], X,	U+H, W+H[(1)], X,	Q

Table 6.2-22

Containment Isolation Valve Information for the Nuclear Boiler System Feedwater Line B

Penetration Identification	B21-MPEN-0007			
Valve No.	**F102B**	**F100B**	**F101B**	**F111B**
		Y(1)	Y(1)	
Primary Actuation	Flow to open/close	Auto-closed	Auto-closed	Flow to open/close
Secondary Actuation	N/A	Remote manual	Remote manual	N/A
Closure Time (sec)	N/A	N/A on reverse flow, < 15 sec on auto-isolation	N/A on reverse flow, < 15 sec on auto-isolation	N/A
Power Source	N/A	Div. 1, 3	Div. 2, 4	N/A

(1) Isolates coolant in-flow to containment

* Nominal pipe size diameter

Note: For explanation of codes, see legend on Table 6.2-15. See Table 3.9-8 for valve and actuator types.

Table 6.2-23

Containment Isolation Valve Information for the Isolation Condenser System Loop A

Penetration Identification	B32-MPEN-0001[2]		B32-MPEN-0005[2]	
Valve Number	**F001A**	**F002A**	**F003A**	**F004A**
Valve Location	Steam Supply	Steam Supply	Condensate Return	Condensate Return
Applicable Basis	GDC 55[1]	GDC 55[1]	GDC 55[1]	GDC 55[1]
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 (14 in)	350mm (14 in)	200mm (8 in)	200mm (8 in)
Type C Leakage Test	Yes	Yes	Yes	Yes
(Deleted)				
Leakage Through Packing[a]	N/A	N/A	N/A	N/A
Leakage Past Seat[b]	b_6[3]	b_6[3]	b_6[3]	b_6[3]
Location	Inboard	Inboard	Inboard	Inboard
(Deleted)				
(Deleted)				
Normal Position	Open	Open	Open	Open
Shutdown Position	Open	Open	Open	Open
Post-Accident Position	Open[4]	Open[4]	Open[4]	Open[4]
Power Fail Position	As is	As is	As is	As is
Containment Isolation 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

(1) With respect to meeting the requirements of US NRC 10 CFR 50, Appendix A, General Design Criterion 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 CFR 50, App.A, Criteria 55 and 56.

(2) Two in series valves
(3) Closed barrier outside containment (IC piping outside containment is Quality Group B)
(4) Except on IC pipe or tube failure
* Nominal pipe size diameter
Note: For explanation of codes, see legend on Table 6.2-15. See Table 3.9-8 for valve and actuator types.

Table 6.2-24

Containment Isolation Valve Information for the Isolation Condenser System Loop A

Penetration Identification	B32-MPEN-0009(2)		B32-MPEN-0009(3)				B32-MPEN-0001(2)	
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(1)	GDC 55(1)	GDC 55(1)	GDC 55(1)	GDC 55(1)	GDC 55(1)	GDC 55(1)	GDC 55(1)
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/Stea m /Non Cond Gases
Line Size*	20mm (0.75 in)	20mm (0.75 in)	20mm (0.75 in)	20mm (0.75 in)	20mm (0.75 in)	20mm (0.75 in)	20mm (0.75 in)	20mm (0.75 in)
Type C Leakage Test	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes
(Deleted)								
Leakage Through Packing(a)	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A
Leakage Past Seat(b)	b_6(4)	b_6(4)	b_6(4)	b_6(4)	b_6(4)	b_6(4)	b_6(4)	b_6(4)
Location	Inboard	Inboard	Inboard	Inboard	Inboard	Inboard	Inboard	Inboard
(Deleted)								

Table 6.2-24

Containment Isolation Valve Information for the Isolation Condenser System Loop A

Penetration Identification	B32-MPEN-0009[(2)]		B32-MPEN-0009[(3)]				B32-MPEN-0001[(2)]	
Valve Number	**F007A**	**F008A**	**F009A**	**F010A**	**F011A**	**F012A**	**F013A**	**F014A**
(Deleted)								
Normal Position	Closed	Closed	Closed	Closed	Closed	Closed	Open	Open
Shutdown Position	Closed	Closed	Closed	Closed	Closed	Closed	Open	Open
Post-Accident Position	Closed	Closed	Closed	Closed	Open/Closed	Open/Closed	Open/Close	Open
Power Fail Position	Closed	Closed	Closed	Closed	N/A	Open	Closed	N/A
Containment Isolation Signal[(d)]	P	P	P	P	Q	P	I, K	Q
Primary Actuation	Remote manual	Remote manual	Remote manual	Remote manual	Pressure	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)	< 15	< 15	< 15	< 15	< 15	< 15	< 15	< 15
Power Source	Div. 1	Div. 1	Div. 2, 4	Div. 2, 4	N/A	Div. 1	Div. 1, 2, 3	N/A

(1) The piping and valve arrangement for these lines meet the requirement 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.
(2) Two in-series valves
(3) Two in-series valves (F009/F010) in parallel with two in series valves (F011/F012)
(4) Closed barrier outside containment
* Nominal pipe size diameter
Note: For explanation of codes, see legend on Table 6.2-15. See Table 3.9-8 for valve and actuator types.

Table 6.2-25

Containment Isolation Valve Information for the Isolation Condenser System Loop B

Penetration Identification	B32-MPEN-0002[(2)]		B32-MPEN-0006[(2)]	
Valve Number	F001B	F002B	F003B	F004B
Valve Location	Steam Supply	Steam Supply	Condensate Return	Condensate Return
Applicable Basis	GDC 55[(1)]	GDC 55[(1)]	GDC 55[(1)]	GDC 55[(1)]
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 (14 in)	350mm (14 in)	200mm (8 in)	200mm (8 in)
Type C Leakage Test	Yes	Yes	Yes	Yes
(Deleted)				
Leakage Through Packing[(a)]	N/A	N/A	N/A	N/A
Leakage Past Seat[(b)]	b_6[(3)]	b_6[(3)]	b_6[(3)]	b_6[(3)]
Location	Inboard	Inboard	Inboard	Inboard
(Deleted)				
(Deleted)				
Normal Position	Open	Open	Open	Open
Shutdown Position	Open	Open	Open	Open
Post-Accident Position	Open[(4)]	Open[(4)]	Open[(4)]	Open[(4)]
Power Fail Position	As is	As is	As is	As is
Containment Isolation 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

(1) 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 CFR 50, App. A, Criteria 55 and 56.

(2) Two in series valves

(3) Closed barrier outside containment (IC piping outside containment is Quality Group B Design)
(4) Except on IC pipe or tube failure
* Nominal pipe size diameter
Note: For explanation of codes, see legend on Table 6.2-15. See Table 3.9-8 for valve and actuator types.

Table 6.2-26

Containment Isolation Valve Information for the Isolation Condenser System Loop B

Penetration Identification	B32-MPEN-0010(2)		B32-MPEN-0010(3)				B32-MPEN-0002(2)	
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(1)	GDC 55(1)	GDC 55(1)	GDC 55(1)	GDC 55(1)	GDC 55(1)	GDC 55(1)	GDC 55(1)
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 (0.75 in)	20mm (0.75 in)	20mm (0.75 in)	20mm (0.75 in)	20mm (0.75 in)	20mm (0.75 in)	20mm (0.75 in)	20mm (0.75 in)
Type C Leakage Test	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes
(Deleted)								
Leakage Through Packing(a)	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A
Leakage Past Seat(b)	b_6(4)	b_6(4)	b_6(4)	b_6(4)	b_6(4)	b_6(4)	b_6(4)	b_6(4)
Location	Inboard	Inboard	Inboard	Inboard	Inboard	Inboard	Inboard	Inboard

Table 6.2-26

Containment Isolation Valve Information for the Isolation Condenser System Loop B

Penetration Identification	B32-MPEN-0010(2)		B32-MPEN-0010(3)				B32-MPEN-0002(2)	
Valve Number	**F007B**	**F008B**	**F009B**	**F010B**	**F011B**	**F012B**	**F013B**	**F014B**
(Deleted)								
(Deleted)								
Normal Position	Closed	Closed	Closed	Closed	Closed	Closed	Open	Open
Shutdown Position	Closed	Closed	Closed	Closed	Closed	Closed	Open	Open
Post-Accident Position	Closed	Closed	Closed	Closed	Open/Closed	Open/Closed	Open/Close	Open
Power Fail Position	Closed	Closed	Closed	Closed	N/A	Open	Closed	N/A
Containment Isolation Signal(d)	P	P	P	P	Q	P	I, K	Q
Primary Actuation	Remote manual	Remote manual	Remote manual	Remote manual	Pressure	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)	< 15	< 15	< 15	< 15	< 15	< 15	< 15	< 15
Power Source	Div. 2	Div. 2	Div. 1, 3	Div. 1, 3	N/A	Div. 2	Div. 2, 3, 4	N/A

(1) 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.
(2) Two in series valves
(3) Two in series valves (F009/F010) in parallel with two in series valves (F011/F012)
(4) Closed barrier outside containment (IC piping outside containment is Quality Group B Design)
* Nominal pipe size diameter

Note: For explanation of codes, see legend on Table 6.2-15. See Table 3.9-8 for valve and actuator types.

Table 6.2-27

Containment Isolation Valve Information for the Isolation Condenser System Loop C

Penetration Identification	B32-MPEN-0003[2]		B32-MPEN-0007[2]	
Valve Number	**F001C**	**F002C**	**F003C**	**F004C**
Valve Location	Steam Supply	Steam Supply	Condensate Return	Condensate Return
Applicable Basis	GDC 55[1]	GDC 55[1]	GDC 55[1]	GDC 55[1]
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 (14 in)	350 mm (14 in)	200 mm (8 in)	200 mm (8 in)
Type C Leakage Test	Yes	Yes	Yes	Yes
(Deleted)				
Leakage Through Packing[a]	N/A	N/A	N/A	N/A
Leakage Past Seat[b]	b_6[3]	b_6[3]	b_6[3]	b_6[3]
Location	Inboard	Inboard	Inboard	Inboard
(Deleted)				
(Deleted)				
Normal Position	Open	Open	Open	Open
Shutdown Position	Open	Open	Open	Open
Post-Accident Position	Open[4]	Open[4]	Open[4]	Open[4]
Power Fail Position	As is	As is	As is	As is
Containment Isolation 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

(1) 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 intent of the isolation guidelines of 10 CFR 50, App.A, Criteria 55 and 56.
(2) Two in series valves
(3) Closed barrier outside containment (IC piping outside containment is Quality Group B)
(4) Except on IC pipe or tube failure
* Nominal pipe size diameter

Note: For explanation of codes, see legend on Table 6.2-15. See Table 3.9-8 for valve and actuator types.

Table 6.2-28

Containment Isolation Valve Information for the Isolation Condenser System Loop C

Penetration Identification	B32-MPEN-0011[2]		B32-MPEN-0011[3]				B32-MPEN-0003[2]	
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[1]	GDC 55[1]	GDC 55[1]	GDC 55[1]	GDC 55[1]	GDC 55[1]	GDC 55[1]	GDC 55[1]
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 (0.75 in)	20mm (0.75 in)	20mm (0.75 in)	20mm (0.75 in)	20mm (0.75 in)	20mm (0.75 in)	20mm (0.75 in)	20mm (0.75 in)
Type C Leakage Test	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes
(Deleted)								
Leakage Through Packing[a]	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A
Leakage Past Seat[b]	b_6[4]	b_6[4]	b_6[4]	b_6[4]	b_6[4]	b_6[4]	b_6[4]	b_6[4]
Location	Inboard	Inboard	Inboard	Inboard	Inboard	Inboard	Inboard	Inboard
(Deleted)								

Table 6.2-28

Containment Isolation Valve Information for the Isolation Condenser System Loop C

Penetration Identification	B32-MPEN-0011[2]		B32-MPEN-0011[3]				B32-MPEN-0003[2]	
Valve Number	**F007C**	**F008C**	**F009C**	**F010C**	**F011C**	**F012C**	**F013C**	**F014C**
(Deleted)								
Normal Position	Closed	Closed	Closed	Closed	Closed	Closed	Open	Open
Shutdown Position	Closed	Closed	Closed	Closed	Closed	Closed	Open	Open
Post-Accident Position	Closed	Closed	Closed	Closed	Open/Closed	Open/Closed	Open/Close	Open
Power Fail Position	Closed	Closed	Closed	Closed	N/A	Open	Closed	N/A
Containment Isolation Signal[d]	P	P	P	P	Q	P	I, K	Q
Primary Actuation	Remote manual	Remote manual	Remote manual	Remote manual	Pressure	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)	< 15	< 15	< 15	< 15	< 15	< 15	< 15	< 15
Power Source	Div. 3	Div. 3	Div. 2, 4	Div. 2, 4	N/A	Div. 3	Div. 3, 4, 1	N/A

(1) 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.
(2) Two in series valves
(3) Two in series valves (F009/F010) in parallel with two in series valves (F011/F012)
(4) Closed barrier outside containment (IC piping outside containment is Quality Group B)
* Nominal pipe size diameter
Note: For explanation of codes, see legend on Table 6.2-15. See Table 3.9-8 for valve and actuator types.

Table 6.2-29

Containment Isolation Valve Information for the Isolation Condenser System Loop D

Penetration Identification	B32-MPEN-0004(2)		B32-MPEN-0008(2)	
Valve Number	**F001D**	**F002D**	**F003D**	**F004D**
Valve Location	Steam Supply	Steam Supply	Condensate Return	Condensate Return
Applicable Basis	GDC 55(1)	GDC 55(1)	GDC 55(1)	GDC 55(1)
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 (14 in)	350 mm (14 in)	200 mm (8 in)	200 mm (8 in)
Type C Leakage Test	Yes	Yes	Yes	Yes
(Deleted)				
Leakage Through Packing(a)	N/A	N/A	N/A	N/A
Leakage Past Seat(b)	b_6(3)	b_6(3)	b_6(3)	b_6(3)
Location	Inboard	Inboard	Inboard	Inboard
(Deleted)				
(Deleted)				
Normal Position	Open	Open	Open	Open
Shutdown Position	Open	Open	Open	Open
Post-Accident Position	Open(4)	Open(4)	Open(4)	Open(4)
Power Fail Position	As is	As is	As is	As is
Containment Isolation 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

(1) 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, Appendix A, Criteria 55 and 56.

(2) Two in series valves

(3) Closed barrier outside containment (IC piping outside containment is Quality Group B)

(4) Except on IC pipe or tube failure

* Nominal pipe size diameter

Note: For explanation of codes, see legend on Table 6.2-15. See Table 3.9-8 for valve and actuator types.

Table 6.2-30

Containment Isolation Valve Information for the Isolation Condenser System Loop D

Penetration Identification	B32-MPEN-0012[(2)]		B32-MPEN-0012[(3)]				B32-MPEN-0004[(2)]	
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[(1)]	GDC 55[(1)]	GDC 55[(1)]	GDC 55[(1)]	GDC 55[(1)]	GDC 55[(1)]	GDC 55[(1)]	GDC 55[(1)]*
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 (0.75 in)	20mm (0.75 in)	20mm (0.75 in)	20mm (0.75 in)	20mm (0.75 in)	20mm (0.75 in)	20mm (0.75 in)	20mm (0.75 in)
Type C Leakage Test	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes
(Deleted)								
Leakage Through Packing[(a)]	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A
Leakage Past Seat[(b)]	b_6[(4)]	b_6[(4)]	b_6[(4)]	b_6[(4)]	b_6[(4)]	b_6[(4)]	b_6[(4)]	b_6[(4)]
Location	Inboard	Inboard	Inboard	Inboard	Inboard	Inboard	Inboard	Inboard
(Deleted)								
(Deleted)								
Normal Position	Closed	Closed	Closed	Closed	Closed	Closed	Open	Open

Table 6.2-30

Containment Isolation Valve Information for the Isolation Condenser System Loop D

Penetration Identification	B32-MPEN-0012(2)		B32-MPEN-0012(3)				B32-MPEN-0004(2)	
Valve Number	**F007D**	**F008D**	**F009D**	**F010D**	**F011D**	**F012D**	**F013D**	**F014D**
Shutdown Position	Closed	Closed	Closed	Closed	Closed	Closed	Open	Open
Post-Accident Position	Closed	Closed	Closed	Closed	Open/Closed	Open/Closed	Open	Open
Power Fail Position	Closed	Closed	Closed	Closed	N/A	Open	Closed	N/A
Containment Isolation Signal(d)	P	P	P	P	Q	P	I, K	Q
Primary Actuation	Remote manual	Remote manual	Remote manual	Remote manual	Pressure	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)	< 15	< 15	< 15	< 15	< 15	< 15	< 15	< 15
Power Source	Div. 4	Div. 4	Div. 1, 3	Div. 1, 3	N/A	Div. 4	Div. 4, 1, 2	N/A

(1) 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.
(2) Two in series valves
(3) Two in series valves (F009/F010) in parallel with two in series valves (F011/F012)
(4) Closed barrier outside containment (IC piping outside containment is Quality Group B)
* Nominal pipe size diameter

Note: For explanation of codes, see legend on Table 6.2-15. See Table 3.9-8 for valve and actuator types.

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 (10 in)	250 mm (10 in)	150 mm (6 in)	150 mm (6 in)	250 mm (10 in)	250 mm (10 in)	150 mm (6 in)	150 mm (6 in)
Type C Leakage Test	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes
(Deleted)								
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
(Deleted)								
(Deleted)								
Normal Position	Open	Open	Open	Open	Open	Open	Open	Open
Shutdown Position	Open	Open	Open	Open	Open	Open	Open	Open
Post-Accident Position	Closed	Closed	Closed	Closed	Closed	Closed	Closed	Closed
Power Fail Position	Closed	Closed	Closed	Closed	Closed	Closed	Closed	Closed
Containment Isolation 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

* Nominal pipe size diameter

Note: For explanation of codes, see legend on Table 6.2-15. See Table 3.9-8 for valve and actuator types.

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 (.75 in)	20 mm (.75 in)	20 mm (.75 in)	20 mm (.75 in)
Type C Leakage Test	Yes	Yes	Yes	Yes
(Deleted)				
Leakage Through Packing[(a)]	N/A	a_1	N/A	a_1
Leakage Past Seat[(b)]	b_7	b_7	b_7	b_7
Location	Inboard	Outboard	Inboard	Outboard
(Deleted)				
(Deleted)				
Normal Position	Closed	Closed	Closed	Closed
Shutdown Position	Open/Closed	Open/Closed	Open/Closed	Open/Closed
Post-Accident Position	Open/Closed	Open/Closed	Open/Closed	Open/Closed
Power Fail Position	Closed	Closed	Closed	Closed
Containment Isolation 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

* Nominal pipe size diameter

Note: For explanation of codes, see legend on Table 6.2-15. See Table 3.9-8 for valve and actuator types.

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 (3 in)	80 mm (3 in)	80 mm (3 in)	80 mm (3 in)
Type C Leakage Test	Yes	Yes	Yes	Yes
(Deleted)				
Leakage Through Packing[(a)]	N/A	a_1	a_1	a_1
Leakage Past Seat[(b)]	b_5	b_5	b_5	b_5
Location	Inboard	Outboard	Outboard	Outboard
(Deleted)				
(Deleted)				
Normal Position	Closed	Closed	Closed	Closed
Shutdown Position	Closed	Closed	Closed	Closed
Post-Accident Position	Open/Close	Open/Close	Open	Open
Power Fail Position	N/A	N/A	As is	As is
Containment Isolation Signal[(d)]	Q	Q	N/A[(2)]	N/A[(2)]
Primary Actuation	Flow	Flow	N/A[(2)]	N/A[(2)]
Secondary Actuation	N/A	N/A	N/A[(2)]	N/A[(2)]
Closure Time (sec)	N/A	N/A	N/A[(2)]	N/A[(2)]
Power Source	N/A	N/A	N/A[(2)]	N/A[(2)]

(1) The disk/inlet-fitting cap is hermetically sealed and when valve is actuated, the cap is sheared to permanently open the flow path.
(2) Not relevant to the valve isolation function.
* Nominal pipe size diameter
Note: For explanation of codes, see legend on Table 6.2-15. See Table 3.9-8 for valve and actuator types.

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 (3 in)	80 mm (3 in)	80 mm (3 in)	80 mm (3 in)
Type C Leakage Test	Yes	Yes	Yes	Yes
(Deleted)				
Leakage Through Packing[(a)]	N/A	a_1	a_1	a_1
Leakage Past Seat[(b)]	b_5	b_5	b_5	b_5
Location	Inboard	Outboard	Outboard	Outboard
(Deleted)				
(Deleted)				
Normal Position	Closed	Closed	Closed	Closed
Shutdown Position	Closed	Closed	Closed	Closed
Post-Accident Position	Open/Close	Open/Close	Open	Open
Power Fail Position	N/A	N/A	As is	As is
Containment Isolation Signal[(d)]	Q	Q	N/A[(2)]	N/A[(2)]
Primary Actuation	Flow	Flow	N/A[(2)]	N/A[(2)]
Secondary Actuation	N/A	N/A	N/A[(2)]	N/A[(2)]
Closure Time (sec)	N/A	N/A	N/A[(2)]	N/A[(2)]
Power Source	N/A	N/A	N/A[(2)]	N/A[(2)]

(1) The disk/inlet-fitting cap is hermetically sealed and when valve is actuated, the cap is sheared to permanently open the flow path.
(2) Not relevant to the valve isolation function.
* Nominal pipe size diameter
Note: For explanation of codes, see legend on Table 6.2-15. See Table 3.9-8 for valve and actuator types.

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 (10 in)	250 mm (10 in)	250 mm (10 in)	250 mm (10 in)
Type C Leakage Test	No(4)	No(4)	Yes	Yes
(Deleted)				
Leakage Through Packing(a)	a_1	a_1	a_1	N/A
Leakage Past Seat(b)	b_6	b_6	b_6	b_6
Location	Outboard	Outboard	Outboard	Inboard
(Deleted)				
(Deleted)				
Normal Position	Closed(1)	Closed(1)	Closed(1)	Closed(1)
Shutdown Position	Closed(1)	Closed(1)	Closed(1)	N/A
Post-Accident Position	Closed(2)	Closed(2)	Closed(3)	Closed
Power Fail Position	As-is	As-is	As-is	N/A
Containment Isolation 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
Closure Time (sec)	<30	<30	<30	N/A
Power Source	Div. 1, 3	Div. 2, 4	Div. 1, 3	N/A

(1) The valve is open occasionally for the suppression pool cooling and cleanup function.
(2) The valve is open remote manually for performing low pressure coolant injection (LPCI), DW Spray, or Suppression Pool Cooling function if required.
(3) The valve is opened remote manually for performing Suppression Pool Cooling function if required.
(4) The FAPCS suppression pool suction lines are always filled with water, since the suction lines are located below the suppression pool water level and are sealed from the containment atmosphere.
* Nominal pipe size diameter
Note: For explanation of codes, see legend on Table 6.2-15. See Table 3.9-8 for valve and actuator types.

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 (10 in)	250 mm (10 in)	250 mm (10 in)	250 mm (10 in)
Type C Leakage Test	No(2)	No(2)	Yes	Yes
(Deleted)				
Leakage Through Packing(a)	a_1	a_1	a_1	N/A
Leakage Past Seat(b)	b_6	b_6	b_6	b_6
Location	Outboard	Outboard	Outboard	Inboard
(Deleted)				
(Deleted)				
Normal Position	Closed(1)	Closed(1)	Closed(1)	Closed(1)
Shutdown Position	Closed(1)	Closed(1)	Closed(1)	N/A
Post-Accident Position	Closed(3)	Closed(3)	Closed(4)	Closed
Power Fail Position	As-is	As-is	As-is	N/A
Containment Isolation Signal(d)	P	P	P	Q
Primary Actuation	Remote manual	Remote manual	Remote manual	Self
Secondary Actuation	Local manual	Local manual	Local manual	N/A
Closure Time (sec)	<30	<30	<30	N/A
Power Source	Div. 1, 3	Div. 2, 4	Div. 2, 4	N/A

(1) The valve is open occasionally for the suppression pool cooling and cleanup function.
(2) The FAPCS suppression pool suction lines are always filled with water, since the suction lines are located below the suppression pool water level and are sealed from the containment atmosphere.
(3) The valve is opened remote manually for performing LPCI, DW Spray, or Suppression Pool Cooling function if required.
(4) The valve is opened remote manually for performing Suppression Pool Cooling function if required.
* Nominal pipe size diameter
Note: For explanation of codes, see legend on Table 6.2-15. See Table 3.9-8 for valve and actuator types.

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 (10 in)	250 mm (10 in)	250 mm (10 in)	250 mm (10 in)
Type C Leakage Test	Yes	Yes	Yes	Yes
(Deleted)				
Leakage Through Packing[(a)]	N/A	a_1	a_1	N/A
Leakage Past Seat[(b)]	b_6	b_6	b_6	b_6
Location	Inboard	Outboard	Outboard	Inboard
(Deleted)				
(Deleted)				
Normal Position	Closed[(1)]	Closed[(1)]	Closed[(1)]	Closed[(1)]
Shutdown Position	Closed	Closed	Closed	N/A
Post-Accident Position	Closed	Closed	Closed	Closed
Power Fail Position	Closed	Closed	Closed	N/A
Containment Isolation Signal[(d)]	B,C,H	B,C,H	B,C,H	Q
Primary Actuation	Automatic	Automatic	Automatic	Flow
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

(1) The valve is open occasionally for GDCS pools cooling and cleanup function
* Nominal pipe size diameter
Note: For explanation of codes, see legend on Table 6.2-15. See Table 3.9-8 for valve and actuator types.

Table 6.2-35

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

Penetration Identification	G21-MPEN-0001		G21-MPEN-0008	
Valve No.	**F309**	**F310**	**F212**	**F213**
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 (10 in)	250 mm (10 in)	250 mm (10 in)	250 mm (10 in)
Type C Leakage Test	Yes	Yes	Yes	Yes
(Deleted)				
Leakage Through Packing [(a)]	a_1	N/A	N/A	N/A
Leakage Past Seat [(b)]	b_6	b_6	b_6	b_6
Location	Outboard	Inboard	Inboard	Inboard
(Deleted)				
(Deleted)				
Normal Position	Closed	Closed	Closed	Closed
Shutdown Position	Closed	N/A	Open / Closed	Open / Closed
Post-Accident Position	Closed	Closed	Closed	Closed
Power Fail Position	Closed	N/A	N/A	N/A
Containment Isolation Signal[(d)]	P	Q	R	R
Primary Actuation	Remote manual	Flow	Local Manual	Local Manual
Secondary Actuation	N/A	N/A	N/A	N/A
Closure Time (sec)	<35	N/A	N/A	N/A
Power Source	Div. 1, 2, 3	N/A	N/A	N/A

* Nominal pipe size diameter

Note: For explanation of codes, see legend on Table 6.2-15. See Table 3.9-8 for valve and actuator types.

Table 6.2-36

Containment Isolation Valve Information for the Containment Inerting System

Penetration Identification	T31-MPEN-0004				T31-MPEN-0003(1)			
Valve No.	**F012**	**F011**	**F013**	**F016**	**F010**	**F011**	**F014**	**F015**
Applicable Basis	GDC 56	GDC 56	GDC 56	GDC 56	GDC 56	GDC 56	GDC 56	GDC 56
Tier 2 Figure	6.2-29	6.2-29	6.2-29	6.2-29	6.2-29	6.2-29	6.2-29	6.2-29
ESF	No	No	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	Air/N_2	Air/N_2
Line Size+	350 mm (14 in)	500 mm (20 in)	200 mm (8 in)	200 mm (8 in)	400 mm (16 in)	500 mm (20 in)	25 mm (1 in)	25mm (1 in)
Type C Leakage Test	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes
(Deleted)								
Leakage Through Packing(a)	a_1	a_1	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	b_2/b_5	b_2/b_5
Location	Outboard	Outboard	Outboard	Outboard	Outboard	Outboard	Outboard	Outboard
(Deleted)								
(Deleted)								
Normal Position	Closed	Closed	Closed	Closed	Closed	Closed	Closed(2)	Closed(2)
Shutdown Position	Closed(2)	Closed(2)	Closed	Closed	Closed(2)	Closed(2)	Closed	Closed
Post-Accident Position	Closed	Closed	Closed	Closed	Closed	Closed	Closed	Closed
Power Fail Position	Closed	Closed	N/A	N/A	Closed	Closed	Closed	Closed
Containment Isolation Signal(d)	B,C,H,T	B,C,H,T	N /A	N/A	B,C,H,T	B,C,H,T	B,C,H,T	B,C,H,T

Table 6.2-36

Containment Isolation Valve Information for the Containment Inerting System

Penetration Identification	**T31-MPEN-0004**				**T31-MPEN-0003[1]**			
Valve No.	**F012**	**F011**	**F013**	**F016**	**F010**	**F011**	**F014**	**F015**
Primary Actuation	Automatic	Automatic	Local Manual	Local Manual	Automatic	Automatic	Automatic	Automatic
Secondary Actuation	Remote manual	Remote manual	N/A	N/A	Remote manual	Remote manual	Remote manual	Remote manual
Closure Time (sec)	< 30	< 30	N/A	N/A	< 30	< 30	< 5	< 5
Power Source	Div. 2, 4	Div. 1, 3	N/A	N/A	Div. 2, 4	Div. 1, 3	Div. 2, 4	Div. 1, 3

(1) Two valves in series (F011/F010) in parallel with two in series valves (F015/F014)
(2) Open during the early stage of "inerting/de-inerting" modes to purge resident air/N2 after which the valves are closed
* Nominal pipe size diameter
Note: For explanation of codes, see legend on Table 6.2-15. See Table 3.9-8 for valve and actuator types.

Table 6.2-37

Containment Isolation Valve Information for the Containment Inerting System

Penetration Identification	T31-MPEN-0002[(1)]			
Valve No.	**F008**	**F007**	**F024**	**F023**
Applicable Basis	GDC 56	GDC 56	GDC 56	GDC 56
Tier 2 Figure	6.2-29	6.2-29	6.2-29	6.2-29
ESF	No	No	No	No
Fluid	Air/N_2	Air/N_2	Air/N_2	Air/N_2
Line Size*	500 mm (20 in)	350 mm (14 in)	25 mm (1 in)	25 mm (1 in)
Type C Leakage Test	Yes	Yes	Yes	Yes
(Deleted)				
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
(Deleted)				
(Deleted)				
Normal Position	Closed	Closed	Open	Open
Shutdown Position	Open	Open	Closed	Closed
Post-Accident Position	Closed	Closed	Closed	Closed
Power Fail Position	Closed	Closed	Closed	Closed
Containment Isolation Signal[(d)]	B,C,H,T	B,C,H,T	B,C,H,T	B,C,H,T
Primary Actuation	Automatic	Automatic	Automatic	Automatic
Secondary Actuation	Remote manual	Remote manual	Remote manual	Remote manual
Closure Time (sec.)	< 30	< 30	< 5	< 5
Power Source	Div. 1, 3	Div. 2, 4	Div. 2, 4	Div. 1, 3

(1) Valve F008 in series with F007, valve F024 in series with F023

* Nominal pipe size diameter

Note: For explanation of codes, see legend on Table 6.2-15. See Table 3.9-8 for valve and actuator types.

Table 6.2-38

Containment Isolation Valve Information for the Containment Inerting System

Penetration Identification	T31-MPEN-0001[1]			
Valve No.	**F025**	**F023**	**F008**	**F009**
Applicable Basis	GDC 56	GDC 56	GDC 56	GDC 56
Tier 2 Figure	6.2-29	6.2-29	6.2-29	6.2-29
ESF	No	No	No	No
Fluid	Air/N_2	Air/N_2	Air/N_2	Air/N_2
Line Size*	25 mm (1 in)	25 mm (1 in)	500 mm (20 in)	350 mm (14 in)
Type C Leakage Test	Yes	Yes	Yes	Yes
(Deleted)				
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
(Deleted)				
(Deleted)				
Normal Position	Open	Open	Closed	Closed
Shutdown Position	Closed	Closed	Open	Open
Post-Accident Position	Closed	Closed	Closed	Closed
Power Fail Position	Closed	Closed	Closed	Closed
Containment Isolation Signal[d]	B,C,H,T	B,C,H,T	B,C,H,T	B,C,H,T
Primary Actuation	Automatic	Automatic	Automatic	Automatic
Secondary Actuation	Remote manual	Remote manual	Remote manual	Remote manual
Closure Time (sec)	< 5	< 5	< 30	< 30
Power Source	Div. 2, 4	Div. 1, 3	Div. 1, 3	Div. 2, 4

(1) Valve F008 in series with F009, valve F025 in series with F023

* Nominal pipe size diameter

Note: For explanation of codes, see legend on Table 6.2-15. See Table 3.9-8 for valve and actuator types.

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	6.2-34	6.2-34	6.2-34	6.2-34
ESF	No	No	No	No
Fluid	Water	Water	Water	Water
Line Size*	150 mm (6 in)	150 mm (6 in)	150 mm (6 in)	150 mm (6 in)
Type C Leakage Test	Yes	Yes	Yes	Yes
(Deleted)				
Leakage Through Packing[(a)]	a_1	N/A	N/A	a_1
Leakage Past Seat[(b)]	b_2	b_2	b_2	b_2
Location	Outboard	Inboard	Inboard	Outboard
(Deleted)				
(Deleted)				
Normal Position	Open	Open	Open	Open
Shutdown Position	Open/Closed	Open/Closed	Open/Closed	Open/Closed
Post-Accident Position	Closed	Closed	Closed	Closed
Power Fail Position	Closed	Closed	Closed	Closed
Containment Isolation 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

* Nominal pipe size diameter

Note: For explanation of codes, see legend on Table 6.2-15. See Table 3.9-8 for valve and actuator types.

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	6.2-34	6.2-34	6.2-34	6.2-34
ESF	No	No	No	No
Fluid	Water	Water	Water	Water
Line Size*	150 mm (6 in)	150 mm (6 in)	150 mm (6 in)	150 mm (6 in)
Type C Leakage Test	Yes	Yes	Yes	Yes
(Deleted)				
Leakage Through Packing[(a)]	a_1	N/A	N/A	a_1
Leakage Past Seat[(b)]	b_2	b_2	b_2	b_2
Location	Outboard	Inboard	Inboard	Outboard
(Deleted)				
(Deleted)				
Normal Position	Open	Open	Open	Open
Shutdown Position	Open/Closed	Open/Closed	Open/Closed	Open/Closed
Post-Accident Position	Closed	Closed	Closed	Closed
Power Fail Position	Closed	Closed	Closed	Closed
Containment Isolation 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

* Nominal pipe size diameter

Note: For explanation of codes, see legend on Table 6.2-15. See Table 3.9-8 for valve and actuator types.

Table 6.2-40

Containment Isolation Valve Information for the High Pressure Nitrogen 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	9.3-4	9.3-4	9.3-4	9.3-4
ESF	No	No	No	No
Fluid	Air/N_2	Air/N_2	N_2	N_2
Line Size*	50 mm (2 in)	50 mm (2 in)	50 mm (2 in)	50 mm (2 in)
Type C Leakage Test	Yes	Yes	Yes	Yes
(Deleted)				
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
(Deleted)				
(Deleted)				
Normal Position	Open	Open/Closed	Open	Open/Closed
Shutdown Position	Open/Closed	Open/Closed	Open/Closed	Open/Closed
Post-Accident Position	Closed	Closed	Closed	Closed
Power Fail Position	Closed	N/A	Closed	N/A
Containment Isolation Signal[(d)]	C,H	Q	C,H	Q
Primary Actuation	Automatic	Flow	Automatic	Flow
Secondary Actuation	Remote manual	N/A	Remote manual	N/A
Closure Time (sec.)	< 15	N/A	< 15	N/A
Power Source	Div. 2, 4	N/A	Div. 2, 4	N/A

* Nominal pipe size diameter

Note: For explanation of codes, see legend on Table 6.2-15. See Table 3.9-8 for valve and actuator types.

Table 6.2-41

Containment Isolation Valve Information for the Makeup Water System

Penetration Identification	P10-MPEN-0001	
Valve No.	**F016**	**F015**
Applicable Basis	GDC 56	GDC 56
Tier 2 Figure	6.2-31	6.2-31
ESF	No	No
Fluid	Water	Water
Line Size*	< 50 mm (2 in)	< 50 mm (2 in)
Type C Leakage Test	Yes	Yes
(Deleted)		
Leakage Through Packing[(a)]	N/A	a_1
Leakage Past Seat[(b)]	b_6	b_6
Location	Inboard	Outboard
(Deleted)		
(Deleted)		
Normal Position	Closed	Closed
Shutdown Position	Closed	Closed
Post-Accident Position	Closed	Closed
Power Fail Position	N/A	N/A
Containment Isolation Signal[(d)]	Q	R
Primary Actuation	Flow	Local manual
Secondary Actuation	N/A	N/A
Closure Time (sec.)	N/A	N/A
Power Source	N/A	N/A

* Nominal pipe size diameter

Note: For explanation of codes, see legend on Table 6.2-15. See Table 3.9-8 for valve and actuator types.

Table 6.2-42

Containment Isolation Valve Information for the Process Radiation Monitoring System

Penetration Identification	D11-MPEN-0001[1]		D11-MPEN-0002[2]	
Valve No.	**F001**	**F002**	**F003**	**F004**
Applicable Basis	GDC 56	GDC 56	GDC 56	GDC 56
Tier 2 Figure	6.2-30	6.2-30	6.2-30	6.2-30
ESF	No	No	No	No
Fluid	Air/N_2	Air/N_2	Air/N_2	Air/N_2
Line Size*	25 mm (1 in)	25 mm (1 in)	25 mm (1 in)	25 mm (1 in)
Type C Leakage Test	Yes	Yes	Yes	Yes
(Deleted)				
Leakage Through Packing[a]	N/A	a_1	N/A	a_1
Leakage Past Seat[b]	b_2	b_2	b_2	b_2
Location	Inboard	Outboard	Inboard	Outboard
(Deleted)				
(Deleted)				
Normal Position	Open	Open	Open	Open
Shutdown Position	Open	Open	Open	Open
Post-Accident Position	Open	Open	Open	Open
Power Fail Position	As-is	As-is	As-is	As-is
Containment Isolation 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
Closure Time (sec.)	< 5	< 5	< 5	< 5
Power Source	Div. 2, 4	Div. 1, 3	Div. 2, 4	Div. 1, 3

(1) Valve F001 in series with F002
(2) Valve F003 in series with F004
* Nominal pipe size diameter
Note: For explanation of codes, see legend on Table 6.2-15. See Table 3.9-8 for valve and actuator types.

Table 6.2-43

Containment Isolation Valve Information for the Equipment and Floor Drain System

Penetration Identification	U50-MPEN-0001		U50-MPEN-0002	
Valve No.	**F001**	**F002**	**F003**	**F004**
Applicable Basis	GDC 56	GDC 56	GDC 56	GDC 56
Tier 2 Figure	6.2-32	6.2-32	6.2-32	6.2-32
ESF	No	No	No	No
Fluid	Water	Water	Water	Water
Line Size*	< 50 mm (2 in)	< 50 mm (2 in)	< 50 mm(2 in)	< 50 mm (2 in)
Type C Leakage Test	Yes	Yes	Yes	Yes
(Deleted)				
Leakage Through Packing[(a)]	N/A	a_1	N/A	a_1
Leakage Past Seat[(b)]	b_7	b_7	b_7	b_7
Location	Inboard	Outboard	Inboard	Outboard
(Deleted)				
(Deleted)				
Normal Position	Closed	Closed	Closed	Closed
Shutdown Position	Closed	Closed	Closed	Closed
Post-Accident Position	Closed	Closed	Closed	Closed
Power Fail Position	Closed	Closed	Closed	Closed
Containment Isolation Signal[(d)]	B, C, H	B, C, H	B, C, H	B, C, H
Primary Actuation	Automatic	Automatic	Automatic	Automatic
Secondary Actuation	Remote manual	Remote manual	Remote manual	Remote manual
Closure Time (sec.)	N/A	N/A	N/A	N/A
Power Source	Div. 2, 4	Div. 1, 3	Div. 2, 4	Div. 1, 3

* Nominal pipe size diameter

Note: For explanation of codes, see legend on Table 6.2-15. See Table 3.9-8 for valve and actuator types.

Table 6.2-44

Containment Isolation Valve Information for the Service Air System

Penetration Identification	P51-MPEN-0001	
Valve No.	**F001**	**F002**
Applicable Basis	GDC 56	GDC 56
Tier 2 Figure	6.2-33	6.2-33
ESF	No	No
Fluid	Air	Air
Line Size*	< 50 mm (2 in)	< 50 mm (2 in)
Type C Leakage Test	Yes	Yes
(Deleted)		
Leakage Through Packing[(a)]	a_1	N/A
Leakage Past Seat[(b)]	b_6	b_6
Location	Outboard	Inboard
(Deleted)		
(Deleted)		
Normal Position	Closed	Closed
Shutdown Position	Open	Open
Post-Accident Position	Closed	Closed
Power Fail Position	N/A	N/A
Containment Isolation Signal[(d)]	R	R
Primary Actuation	Local manual	Local manual
Secondary Actuation	N/A	N/A
Closure Time (sec.)	N/A	N/A
Power Source	N/A	N/A

* Nominal pipe size diameter

Note: For explanation of codes, see legend on Table 6.2-15. See Table 3.9-8 for valve and actuator types.

Table 6.2-45

Containment Isolation Valve Information for the Containment Monitoring System

Penetration Identification	T62-MPEN-0001 through 0006	
Valve No.	**Various**	**Various**
Applicable Basis	GDC 56	GDC 56
Tier 2 Figure	7.5-1	7.5-1
ESF	No	No
Fluid	Air/N_2	Air/N_2
Line Size*	< 50 mm (2 in)	< 50 mm (2 in)
Type C Leakage Test	Yes	Yes
(Deleted)		
Leakage Through Packing[(a)]	a_1	N/A
Leakage Past Seat[(b)]	b_2	b_2
Location	Outboard	Inboard
(Deleted)		
(Deleted)		
Normal Position	Open	Open
Shutdown Position	Open	Open
Post-Accident Position	Open	Open
Power Fail Position	Open	Open
Containment Isolation Signal[(d)]	B, C, H	B, C, H
Primary Actuation	Automatic	Automatic
Secondary Actuation	Remote manual	Remote manual
Closure Time (sec.)	N/A	N/A
Power Source	Div. 2, 4	Div. 1, 3

* Nominal pipe size diameter

Note: For explanation of codes, see legend on Table 6.2-15. See Table 3.9-8 for valve and actuator types.

Table 6.2-46

(Deleted)

Table 6.2-47

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

Penetration Number (1)	Description	(Deleted)	(Deleted)	Penetration Type (4)	Leak Test Type (5)
Piping Penetrations					
B21: Nuclear Boiler System (NBS)					
B21-MPEN-0001	Main Steam Line A			A	A, C
B21-MPEN-0002	Main Steam Line B			A	A, C
B21-MPEN-0003	Main Steam Line C			A	A, C
B21-MPEN-0004	Main Steam Line D			A	A, C
B21-MPEN-0006	Feedwater Line A			A	A, C
B21-MPEN-0007	Feedwater Line B			A	A, C
B21-MPEN-0005	Main Steam Drain Header			A	A, C
B21-MPEN-0008	RPV Water Level			I	A
B21-MPEN-0009	RPV Water Level			I	A
B21-MPEN-0010	RPV Water Level			I	A
B21-MPEN-0011	RPV Water Level			I	A
B21-MPEN-0012	RPV Water Level			I	A
B21-MPEN-0013	RPV Water Level			I	A
B21-MPEN-0014	RPV Water Level			I	A
B21-MPEN-0015	RPV Water Level			I	A
B21-MPEN-0016	RPV Water Level			I	A
B21-MPEN-0017	RPV Water Level			I	A
B21-MPEN-0018	RPV Water Level			I	A
B21-MPEN-0019	RPV Water Level			I	A
B21-MPEN-0020	Spare Mechanical Penetration [penetration is capped]			S	A
B21-MPEN-0021	Spare Mechanical Penetration [penetration is capped]			S	A
B21-MPEN-0022	Spare Mechanical Penetration [penetration is capped]			S	A
B21-MPEN-0023	Spare Mechanical Penetration [penetration is capped]			S	A
B21-MPEN-0024	Main Steam Line A Flow Restrictor Instr Line 1			I	A
B21-MPEN-0025	Main Steam Line A Flow Restrictor Instr Line 2			I	A

Table 6.2-47

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

Penetration Number [1]	Description	(Deleted)	(Deleted)	Penetration Type [4]	Leak Test Type [5]
B21-MPEN-0026	Main Steam Line B Flow Restrictor Instr Line 1			I	A
B21-MPEN-0027	Main Steam Line B Flow Restrictor Instr Line 2			I	A
B21-MPEN-0028	Main Steam Line C Flow Restrictor Instr Line 1			I	A
B21-MPEN-0029	Main Steam Line C Flow Restrictor Instr Line 2			I	A
B21-MPEN-0030	Main Steam Line D Flow Restrictor Instr Line 1			I	A
B21-MPEN-0031	Main Steam Line D Flow Restrictor Instr Line 2			I	A
B21-MPEN-0032	Feedwater Line A Instrumentation			I	A
B21-MPEN-0033	Feedwater Line B Instrumentation			I	A
B21-MPEN-0034	RPV Flange Seal Leakage Monitor			I	A
B21-MPEN-0035	RPV Top Head Vent Instrument Line			I	A
B21-MPEN-0036	Main Steam Line A Flow High Pressure Instr Line A			I	A
B21-MPEN-0037	Main Steam Line B Flow High Pressure Instr Line B			I	A
B21-MPEN-0038	Main Steam Line C Flow High Pressure Instr Line C			I	A
B21-MPEN-0039	Main Steam Line D Flow High Pressure Instr Line D			I	A
B32: Isolation Condenser System (ICS)					
B32-MPEN-0001	Train A Steam Supply Line			B	A, C
B32-MPEN-0001	Train A Purge Line From Steam Supply Line			B	A, C
B32-MPEN-0005	Train A Condensate Return			B	A, C
B32-MPEN-0009	Train A Vent Line A From Upper Header (1CA)			B	A, C
B32-MPEN-0009	Train A Vent Line A From Lower Header (1CA)			B	A, C
B32-MPEN-0013	Train A Steam Line Flowrate Instrumentation			I	A

Table 6.2-47

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

Penetration Number (1)	Description	(Deleted)	(Deleted)	Penetration Type (4)	Leak Test Type (5)
B32-MPEN-0014	Train A Steam Line Flowrate Instrumentation			I	A
B32-MPEN-0015	Train A Steam Line Flowrate Instrumentation			I	A
B32-MPEN-0016	Train A Steam Line Flowrate Instrumentation			I	A
B32-MPEN-0017	Train A Condensate Line Flowrate Instrumentation			I	A
B32-MPEN-0018	Train A Condensate Line Flowrate Instrumentation			I	A
B32-MPEN-0019	Train A Condensate Line Flowrate Instrumentation			I	A
B32-MPEN-0020	Train A Condensate Line Flowrate Instrumentation			I	A
B32-MPEN-0002	Train B Steam Supply Line			B	A, C
B32-MPEN-0002	Train B Purge Line From Steam Supply Line			B	A, C
B32-MPEN-0006	Train B Condensate Return			B	A, C
B32-MPEN-0010	Train B Vent Line A From Upper Header (1CB)			B	A, C
B32-MPEN-0010	Train B Vent line A From Lower Header (1CB)			B	A, C
B32-MPEN-0021	Train B Steam Line Flowrate Instrumentation			I	A
B32-MPEN-0022	Train B Steam Line Flowrate Instrumentation			I	A
B32-MPEN-0023	Train B Steam Line Flowrate Instrumentation			I	A
B32-MPEN-0024	Train B Steam Line Flowrate Instrumentation			I	A
B32-MPEN-0025	Train B Condensate Line Flowrate Instrumentation			I	A
B32-MPEN-0026	Train B Condensate Line Flowrate Instrumentation			I	A
B32-MPEN-0027	Train B Condensate Line Flowrate Instrumentation			I	A
B32-MPEN-0028	Train B Condensate Line Flowrate Instrumentation			I	A
B32-MPEN-0003	Train C Steam Supply Line			B	A, C

Table 6.2-47

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

Penetration Number (1)	Description	(Deleted)	(Deleted)	Penetration Type (4)	Leak Test Type (5)
B32-MPEN-0003	Train C Purge Line From Steam Supply Line			B	A, C
B32-MPEN-0007	Train C Condensate Return			B	A, C
B32-MPEN-0011	Train C Vent Line A From Upper Header (1CC)			B	A, C
B32-MPEN-0011	Train C Vent Line A From Lower Header (1CC)			B	A, C
B32-MPEN-0029	Train C Steam Line Flowrate Instrumentation			I	A
B32-MPEN-0030	Train C Steam Line Flowrate Instrumentation			I	A
B32-MPEN-0031	Train C Steam Line Flowrate Instrumentation			I	A
B32-MPEN-0032	Train C Steam Line Flowrate Instrumentation			I	A
B32-MPEN-0033	Train C Condensate Line Flowrate Instrumentation			I	A
B32-MPEN-0034	Train C Condensate Line Flowrate Instrumentation			I	A
B32-MPEN-0035	Train C Condensate Line Flowrate Instrumentation			I	A
B32-MPEN-0036	Train C Condensate Line Flowrate Instrumentation			I	A
B32-MPEN-0004	Train D Steam Supply Line			B	A, C
B32-MPEN-0004	Train D Purge Line From Steam Supply Line			B	A, C
B32-MPEN-0008	Train D Condensate Return			B	A, C
B32-MPEN-0012	Train D Vent Line A From Upper Header (1CD)			B	A, C
B32-MPEN-0012	Train D Vent Line A From Lower Header (1CD)			B	A, C
B32-MPEN-0037	Train D Steam Line Flowrate Instrumentation			I	A
B32-MPEN-0038	Train D Steam Line Flowrate Instrumentation			I	A
B32-MPEN-0039	Train D Steam Line Flowrate Instrumentation			I	A
B32-MPEN-0040	Train D Steam Line Flowrate Instrumentation			I	A

Table 6.2-47

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

Penetration Number (1)	Description	(Deleted)	(Deleted)	Penetration Type (4)	Leak Test Type (5)
B32-MPEN-0041	Train D Condensate Line Flowrate Instrumentation			I	A
B32-MPEN-0042	Train D Condensate Line Flowrate Instrumentation			I	A
B32-MPEN-0043	Train D Condensate Line Flowrate Instrumentation			I	A
B32-MPEN-0044	Train D Condensate Line Flowrate Instrumentation			I	A
C12: Fine Motion Control Rod Drive System (FMCRD)					
C12-MPEN-0001	FMCRD: 23 Hydraulic Lines (2)			M	A
C12-MPEN-0002	FMCRD: 22 Hydraulic Lines (2) + 1 SPARE			M	A
C12-MPEN-0003	FMCRD: 22 Hydraulic Lines (2) + 1 SPARE			M	A
C12-MPEN-0004	FMCRD: 23 Hydraulic Lines (2)			M	A
C12-MPEN-0005	FMCRD: 23 Hydraulic Lines (2)			M	A
C12-MPEN-0006	FMCRD: 22 Hydraulic Lines (2) + 1 SPARE			M	A
C12-MPEN-0007	FMCRD: 23 Hydraulic Lines (2)			M	A
C12-MPEN-0008	FMCRD: 22 Hydraulic Lines (2) + 1 SPARE			M	A
C12-MPEN-0009	FMCRD: 22 Hydraulic Lines (2) + 1 SPARE			M	A
C12-MPEN-0010	FMCRD: 23 Hydraulic Lines (2)			M	A
C12-MPEN-0011	FMCRD: 22 Hydraulic Lines (2) + 1 SPARE			M	A
C12-MPEN-0012	FMCRD: 22 Hydraulic Lines (2) + 1 SPARE			M	A
C41: Standby Liquid Control (SLC) System					
C41-MPEN-0001	Borated Liquid Injection (Train A)			B	A, C
C41-MPEN-0002	Borated Liquid Injection (Train B)			B	A, C

Table 6.2-47

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

Penetration Number (1)	Description	(Deleted)	(Deleted)	Penetration Type (4)	Leak Test Type (5)
D11: Process Radiation Monitoring System (PRMS)					
D11-MPEN-0001	Fission Product Rad Monitoring Extraction Line			I	A
D11-MPEN-0002	Fission Product Rad Monitoring Return Line			I	A
E50: Gravity Driven Cooling System (GDCS)					
E50-MPEN-0001	GDCS Pool A Water Level			I	A
E50-MPEN-0002	GDCS Pool A Water Level			I	A
E50-MPEN-0003	GDCS Pool B/C Water Level			I	A
E50-MPEN-0004	GDCS Pool B/C Water Level			I	A
E50-MPEN-0005	GDCS Pool D Water Level			I	A
E50-MPEN-0006	GDCS Pool D Water Level			I	A
G21: Fuel and Auxiliary Pools Cooling System (FAPCS)					
G21-MPEN-0001	Drywell Spray Discharge Line			C	A, C
G21-MPEN-0002	Suppression Pool Return Line A			C	A, C
G21-MPEN-0003	GDCS Pool Return Line			C	A, C
G21-MPEN-0004	Suction Line from GDCS Pool			C	A, C
G21-MPEN-0005	Suction Line A from Suppression Pool			C	A, C
G21-MPEN-0006	Suppression Pool Return Line B			C	A, C
G21-MPEN-0007	Suction Line B from Suppression Pool			C	A, C
G21-MPEN-0008	Reactor Well Drain Line			C	A, C
G31: Reactor Water Cleanup and Shutdown Cooling (RWCU/SDC) System					
G31-MPEN-0001	RPV Mid-Vessel Line (Train A)			A	A, C
G31-MPEN-0002	RPV Mid-Vessel Line (Train B)			A	A, C
G31-MPEN-0003	RPV Bottom Drain Line (Train A)			B	A, C
G31-MPEN-0004	RPV Bottom Drain Line (Train B)			B	A, C

Table 6.2-47

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

Penetration Number (1)	Description	(Deleted)	(Deleted)	Penetration Type (4)	Leak Test Type (5)
G31-MPEN-0005	Sample Line (Train A)			B	A, C
G31-MPEN-0006	Sample Line (Train B)			B	A, C
P10: Makeup Water System (MWS)					
P10-MPEN-0001	Demin Water Drywell Distribution			C	A, C
P25: Chilled Water System (CWS)					
P25-MPEN-0001	CWS Supply Line Train A			B	A, C
P25-MPEN-0003	CWS Supply Line Train B			B	A, C
P25-MPEN-0002	CWS Return Line Train A			B	A, C
P25-MPEN-0004	CWS Return Line Train B			B	A, C
P51: Service Air System (SAS)					
P51-MPEN-0001	Service Air Supply			C	A, C
P54: High Pressure Nitrogen Supply System (HPNSS)					
P54-MPEN-0001	Supply to MSIV Accumulators			B	A, C
P54-MPEN-0002	Supply to ADS and IC CIV Accumulators			B	A, C
T11: Containment Vessel: Equipment & Personnel Access Hatches					
T11-SPEN-0001	LDW Equipment Hatch			Hatch	A, B
T11-SPEN-0002	LDW Personnel Airlock			Air Lock	A, B
T11-SPEN-0003	Wetwell Access Hatch			Hatch	A, B
T11-SPEN-0004	UDW Equipment Hatch			Hatch	A, B
T11-SPEN-0005	UDW Personnel Airlock			Air Lock	A, B
T11: Containment Vessel: Temporary Services During Outages & Spare Penetrations					
T11-MPEN-0001	Temporary Services During Outages			E	A, B
T11-MPEN-0002	Temporary Services During Outages			E	A, B
T11-MPEN-0003	Temporary Services During Outages			E	A, B
T11-MPEN-0004	Temporary Services During Outages			E	A, B
T11-MPEN-0005	Temporary Services During Outages			E	A, B
T11-MPEN-0006	Spare Mechanical Penetration [penetration is capped]			S	A

Table 6.2-47

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

Penetration Number (1)	Description	(Deleted)	(Deleted)	Penetration Type (4)	Leak Test Type (5)
T11-MPEN-0007	Spare Mechanical Penetration [penetration is capped]			S	A
T11-MPEN-0008	Spare Mechanical Penetration [penetration is capped]			S	A
T11-EPEN-0001	Spare Electrical Penetration			E	A, B
T11-EPEN-0002	Spare Electrical Penetration			E	A, B
T11-EPEN-0003	Spare Electrical Penetration			E	A, B
T11-EPEN-0004	Spare Electrical Penetration			E	A, B
T11-EPEN-0005	Spare Electrical Penetration			E	A, B
T11-EPEN-0006	Spare Electrical Penetration			E	A, B
T11-EPEN-0007	Spare Electrical Penetration			E	A, B
T11-EPEN-0008	Spare Electrical Penetration			E	A, B
T31: Containment Inerting System (CIS)					
T31-MPEN-0001	Upper Drywell Injection Line			C	A, C
T31-MPEN-0002	Suppression Pool Airspace Injection Line			C	A, C
T31-MPEN-0003	Main Exhaust Line (Lower Drywell)			C	A, C
T31-MPEN-0004	Second Exhaust Line (Suppression Pool Airspace)			C	A, C
T31-MPEN-0005	Containment Pressure Test (Lower Drywell) [penetration is capped]			C	A
T62: Containment Monitoring System (CMS)					
T62-MPEN-0001	H2-O2 & Drywell Gas Sample Line From Upper Drywell (Loop A)			C	A, C
T62-MPEN-0002	H2-O2 & Drywell Gas Sample Return Line to Upper Drywell (Loop A)			C	A, C
T62-MPEN-0003	H2-O2 & Drywell Gas Sample Line From Wetwell Airspace (Loop A)			C	A, C

Table 6.2-47

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

Penetration Number (1)	Description	(Deleted)	(Deleted)	Penetration Type (4)	Leak Test Type (5)
T62-MPEN-0004	H2-O2 & Drywell Gas Sample Return Line to Wetwell Airspace (Loop A)			C	A, C
T62-MPEN-0005	H2-O2 & Drywell Gas Sample Line From Upper Drywell (Loop B)			C	A, C
T62-MPEN-0006	H2-O2 & Drywell Gas Sample Return Line to Upper Drywell (Loop B)			C	A, C
T62-MPEN-0007	H2-O2 & Wetwell Gas Sample Line From Wetwell Airspace (Loop B)			C	A, C
T62-MPEN-0008	H2-O2 & Wetwell Gas Sample Return Line to Wetwell Airspace (Loop B)			C	A, C
T62-MPEN-0009	Suppression Pool Water Level Monitoring-Wide Range			I	A
T62-MPEN-0010	Suppression Pool Water Level Monitoring-Wide Range			I	A
T62-MPEN-0011	Suppression Pool Water Level Monitoring-Wide Range			I	A
T62-MPEN-0012	Suppression Pool Water Level Monitoring-Wide Range			I	A
T62-MPEN-0013	Suppression Pool Water Level Monitoring-Narrow Range			I	A
T62-MPEN-0014	Suppression Pool Water Level Monitoring-Narrow Range			I	A
T62-MPEN-0015	Suppression Pool Water Level Monitoring-Narrow Range			I	A
T62-MPEN-0016	Suppression Pool Water Level Monitoring-Narrow Range			I	A

Table 6.2-47

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

Penetration Number (1)	Description	(Deleted)	(Deleted)	Penetration Type (4)	Leak Test Type (5)
T62-MPEN-0017	Drywell Pressure Monitoring-Wide Range (Post-Accident Monitoring)			I	A
T62-MPEN-0018	Drywell Pressure Monitoring-Wide Range (Post-Accident Monitoring)			I	A
T62-MPEN-0019	Drywell Pressure Monitoring-Wide Range (Diverse Protection System)			I	A
T62-MPEN-0020	Drywell Pressure Monitoring-Wide Range (Diverse Protection System)			I	A
T62-MPEN-0021	Drywell Pressure Monitoring-Wide Range (Diverse Protection System)			I	A
T62-MPEN-0022	Drywell Pressure Monitoring-Wide Range (Diverse Protection System)			I	A
T62-MPEN-0023	Drywell Pressure Monitoring – Narrow Range			I	A
T62-MPEN-0024	Drywell Pressure Monitoring – Narrow Range			I	A
T62-MPEN-0025	Drywell Pressure Monitoring – Narrow Range			I	A
T62-MPEN-0026	Drywell Pressure Monitoring – Narrow Range			I	A
T62-MPEN-0027	Wetwell Vapor Pressure Monitoring			I	A
T62-MPEN-0028	Wetwell Vapor Pressure Monitoring			I	A
T62-MPEN-0029	Drywell/Wetwell Differential Pressure Monitoring (UDW)			I	A
T62-MPEN-0030	Drywell/Wetwell Differential Pressure Monitoring (WA)			I	A

Table 6.2-47

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

Penetration Number (1)	Description	(Deleted)	(Deleted)	Penetration Type (4)	Leak Test Type (5)
T62-MPEN-0031	Drywell/Wetwell Differential Pressure Monitoring (LDW)			I	A
T62-MPEN-0032	Drywell/Wetwell Differential Pressure Monitoring (WA)			I	A
T62-MPEN-0033	Lower Drywell Post-LOCA Water Level Monitoring Line A			I	A
T62-MPEN-0034	Lower Drywell Post-LOCA Water Level Monitoring Line B			I	A
T62-MPEN-0035	Upper Drywell Post-LOCA Water Level Monitoring Line A			I	A
T62-MPEN-0036	Upper Drywell Post-LOCA Water Level Monitoring Line B			I	A
U50: Equipment and Floor Drain System (EFDS)					
U50-MPEN-0001	Drywell LCW Sump Discharge Line (3)			B	A, C
U50-MPEN-0002	Drywell HCW Sump Discharge Line (3)			B	A, C
Electrical Penetrations					
R31: Raceway System					
R31-EPEN-0001	Div 1 Electrical Penetration			E	A, B
R31-EPEN-0002	Non-Div Electrical Penetration			E	A, B
R31-EPEN-0003	Non-Div Electrical Penetration			E	A, B
R31-EPEN-0004	Div 2 Electrical Penetration			E	A, B
R31-EPEN-0005	Non-Div Electrical Penetration			E	A, B
R31-EPEN-0006	Non-Div Electrical Penetration			E	A, B
R31-EPEN-0007	Div 3 Electrical Penetration			E	A, B
R31-EPEN-0008	Non-Div Electrical Penetration			E	A, B
R31-EPEN-0009	Non-Div Electrical Penetration			E	A, B

Table 6.2-47

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

Penetration Number (1)	Description	(Deleted)	(Deleted)	Penetration Type (4)	Leak Test Type (5)
R31-EPEN-0010	Div 4 Electrical Penetration			E	A, B
R31-EPEN-0011	Non-Div Electrical Penetration			E	A, B
R31-EPEN-0012	Non-Div Electrical Penetration			E	A, B
R31-EPEN-0013	Non-Div Electrical Penetration			E	A, B
R31-EPEN-0014	Div 1 Electrical Penetration			E	A, B
R31-EPEN-0015	Non-Div Electrical Penetration			E	A, B
R31-EPEN-0016	Electrical Penetration			E	A, B
R31-EPEN-0017	Div 2 Electrical Penetration			E	A, B
R31-EPEN-0018	Non-Div Electrical Penetration			E	A, B
R31-EPEN-0019	Non-Div Electrical Penetration			E	A, B
R31-EPEN-0020	Div 3 Electrical Penetration			E	A, B
R31-EPEN-0021	Non-Div Electrical Penetration			E	A, B
R31-EPEN-0022	Div 4 Electrical Penetration			E	A, B
R31-EPEN-0023	Div 1 Electrical Penetration			E	A, B
R31-EPEN-0024	Div 1 Electrical Penetration			E	A, B
R31-EPEN-0025	Div 2 Electrical Penetration			E	A, B
R31-EPEN-0026	Div 2 Electrical Penetration			E	A, B
R31-EPEN-0027	Div 3 Electrical Penetration			E	A, B
R31-EPEN-0028	Div 3 Electrical Penetration			E	A, B
R31-EPEN-0029	Div 4 Electrical Penetration			E	A, B
R31-EPEN-0030	Div 4 Electrical Penetration			E	A, 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) UDW – UPPER DRYWELL
ST – STEAM TUNNEL
TS – TOP SLAB
LDW – LOWER DRYWELL
WA – WETWELL AIRSPACE
WP – WETWELL POOL
HCW - HIGH CONDUCTIVITY WASTE

LCW - LOW CONDUCTIVITY WASTE

(4) Penetration type:

Type A = Penetration with thermal sleeve for High Energy Pipelines; (Main Steam & Feedwater Lines) (Figure 3.8-6)

Type B = Penetration with thermal sleeve for Low / High Energy Flow (Figures 3.8-6 and 3.8-7)

Type C = Embedded penetration without thermal sleeve (Cold Type for flow Tmax<93°C(200°F)) (Figure 3.8-8)

Type E = Penetration with flanges (Electrical, Maintenance, etc) (Figure 3.8-10)

Type I = Instrumentation and Radiation Monitoring.

Type M = Multiple penetration with sleeve (Figure 3.8-9)

Type S = Spare Mechanical Penetration

(5) All penetrations are subject to the Type A, ILRT

All penetrations excluded from Type B testing are welded penetrations and do not include any resilient seals in their design.

Table 6.2-48

RWCU/SDC NRHX Parameters Assumed in Post-LOCA Containment Cooling and Recovery Analysis

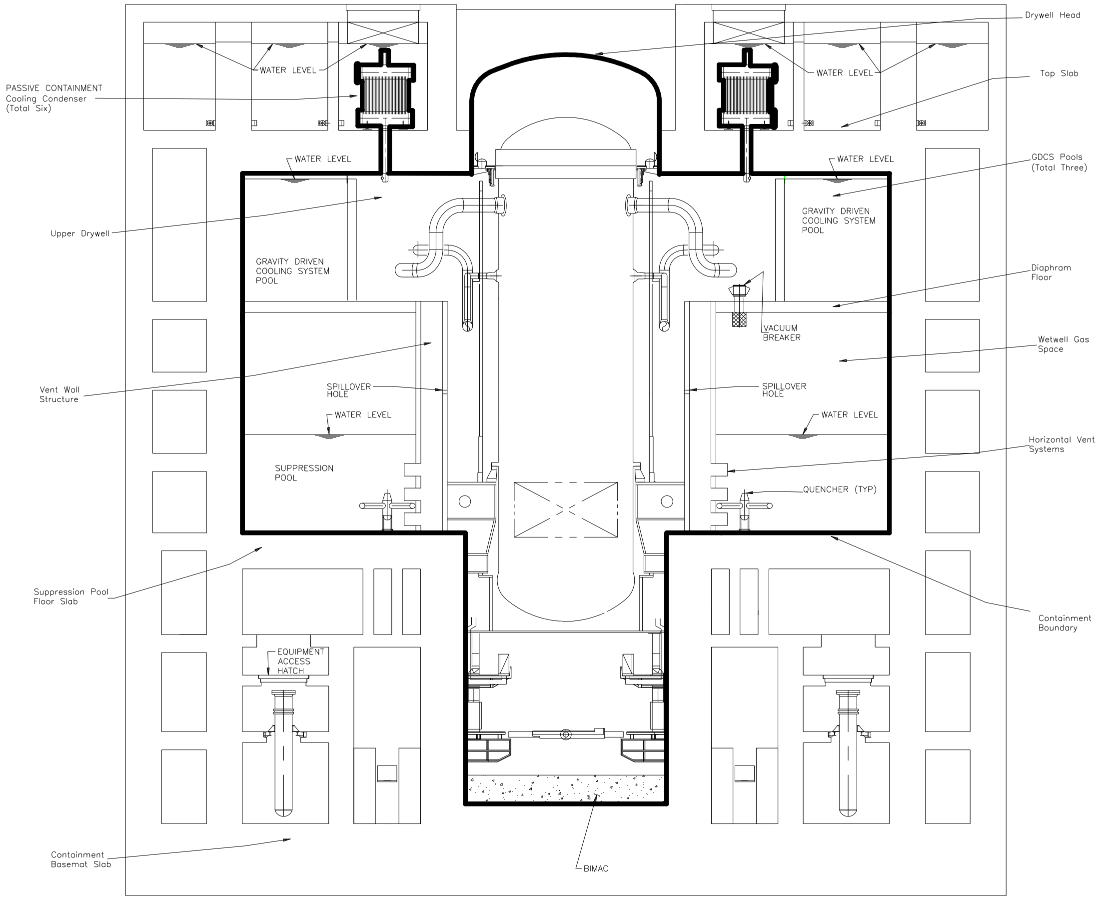
NRHX Parameter	Value Assumed
Shell Side Flow Rate	1590 m^3/hr (7000 gpm)
Tube Side Flow rate (Suppression Pool Cooling mode)	605 m^3/hr (2660 gpm)
Tube Side Flow rate (Reactor Vessel injection mode)	605 m^3/hr (2660 gpm)
Shell Side Inlet temperature	38.3°C (101°F)
Heat Exchanger K value	4.6E+05 J/sec °C (8.7E+05 Btu/hr °F)

Table 6.2-49

PCCS Vent Fan Minimum Performance Requirements[1, 2]

Normalized Fan Head $\Delta P/\rho$ m^2/s^2 (ft^2/s^2)	Flow m^3/s (CFM)
2,410 (25,900)	0.071 (150)
2,380 (25,600)	0.141 (300)
2,290 (24,600)	0.283 (600)
2,050 (22,100)	0.472 (1,000)
1,880 (20,200)	0.566 (1,200)
1,390 (15,000)	0.741 (1,570)

1. The inlet losses for the PCCS are described in Table 6.2-8, Item 4. The outlet losses shall not exceed a k/A^2 value of 1500 m^{-4} (174,000 ft^{-4}).
2. The range of fluid densities associated with values in the Table is 1.81 kg/m^3 to 3.81 kg/m^3



Note: The components attached to the PCCS condenser are an integral part of the containment boundary above the DW. The IC/PCCS pool structure and cooling water are located outside the containment boundary.

Figure 6.2-1. Containment System

Figure 6.2-2. IC/PCCS Pools Configuration

Figure 6.2-3. GDCS Pools Configuration

Figure 6.2-4. (Deleted)

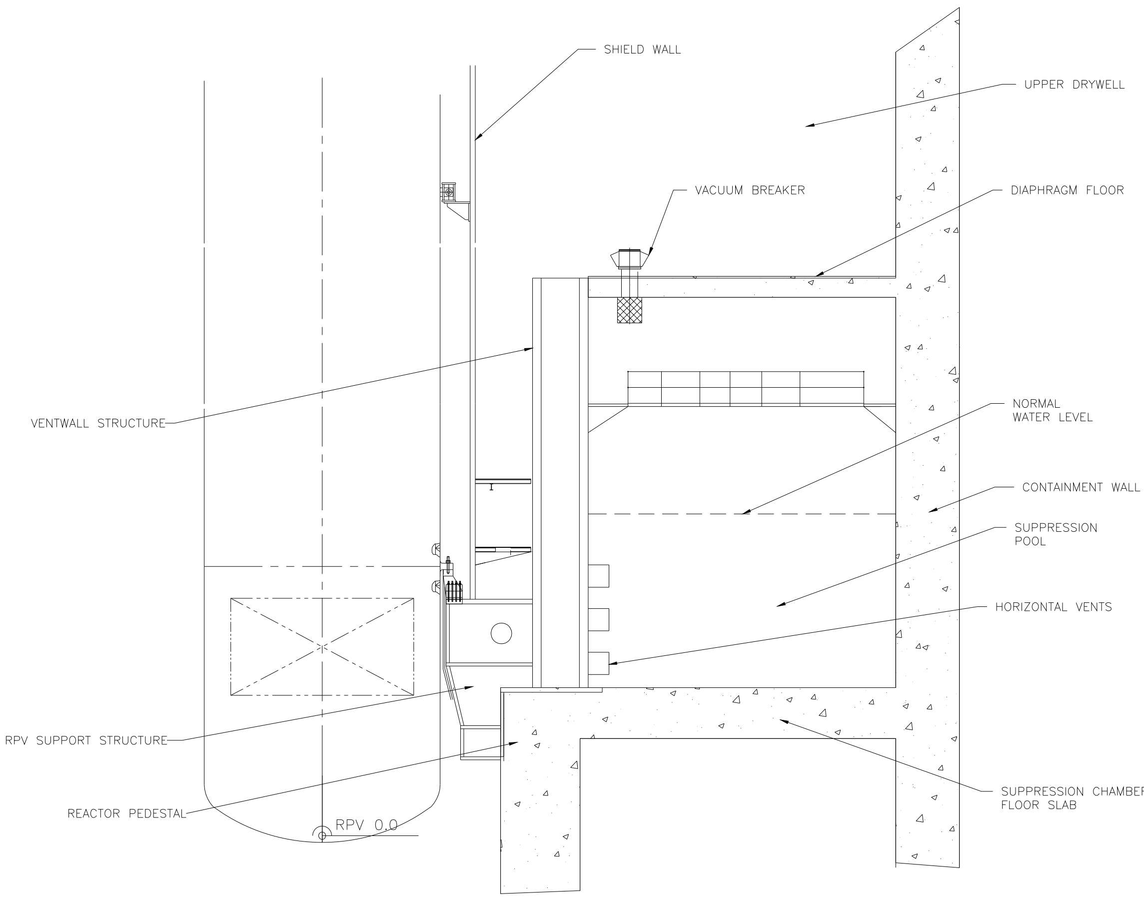


Figure 6.2-5. Horizontal Vent System Configuration

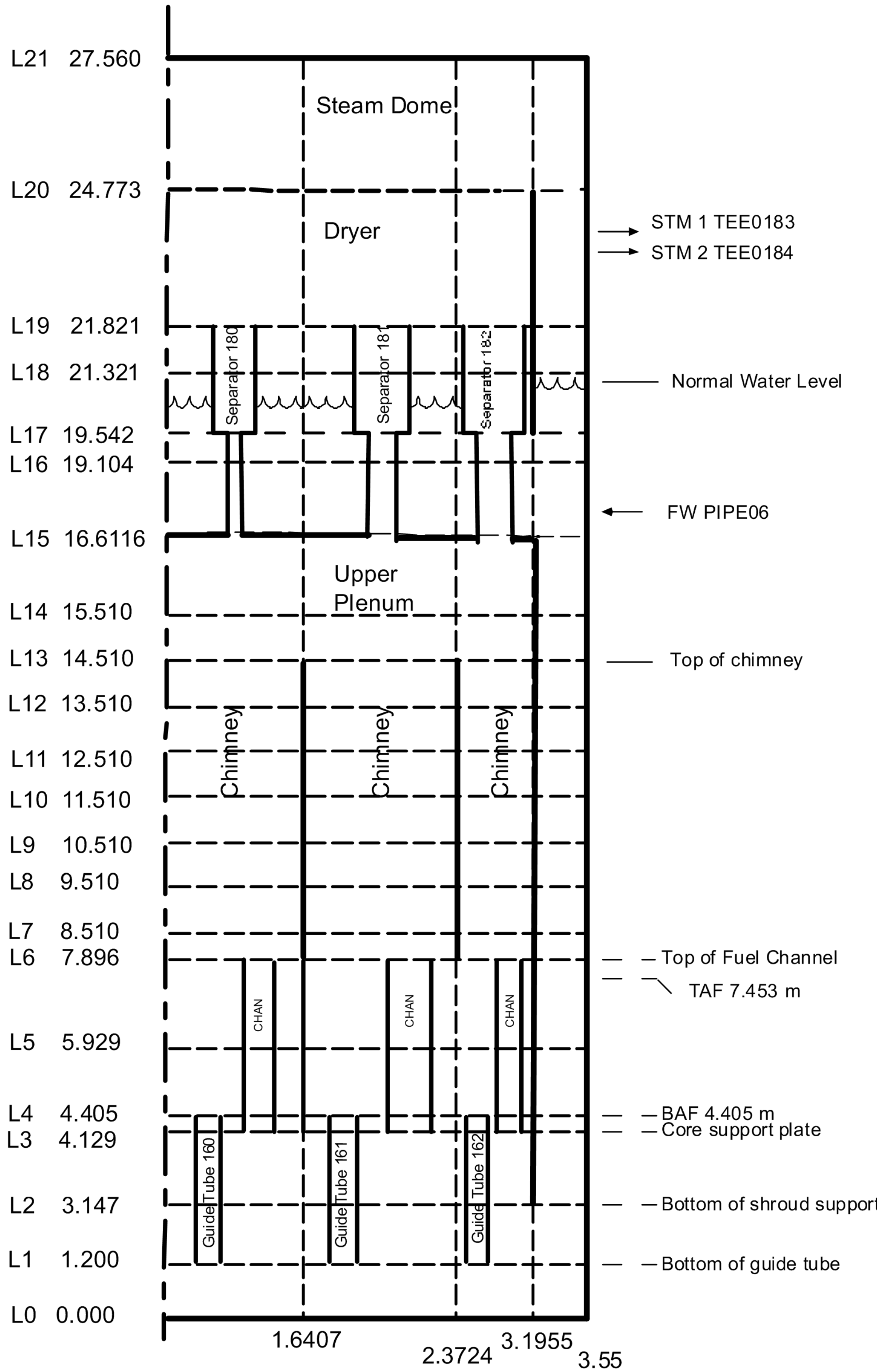


Figure 6.2-6. TRACG Nodalization of the ESBWR RPV

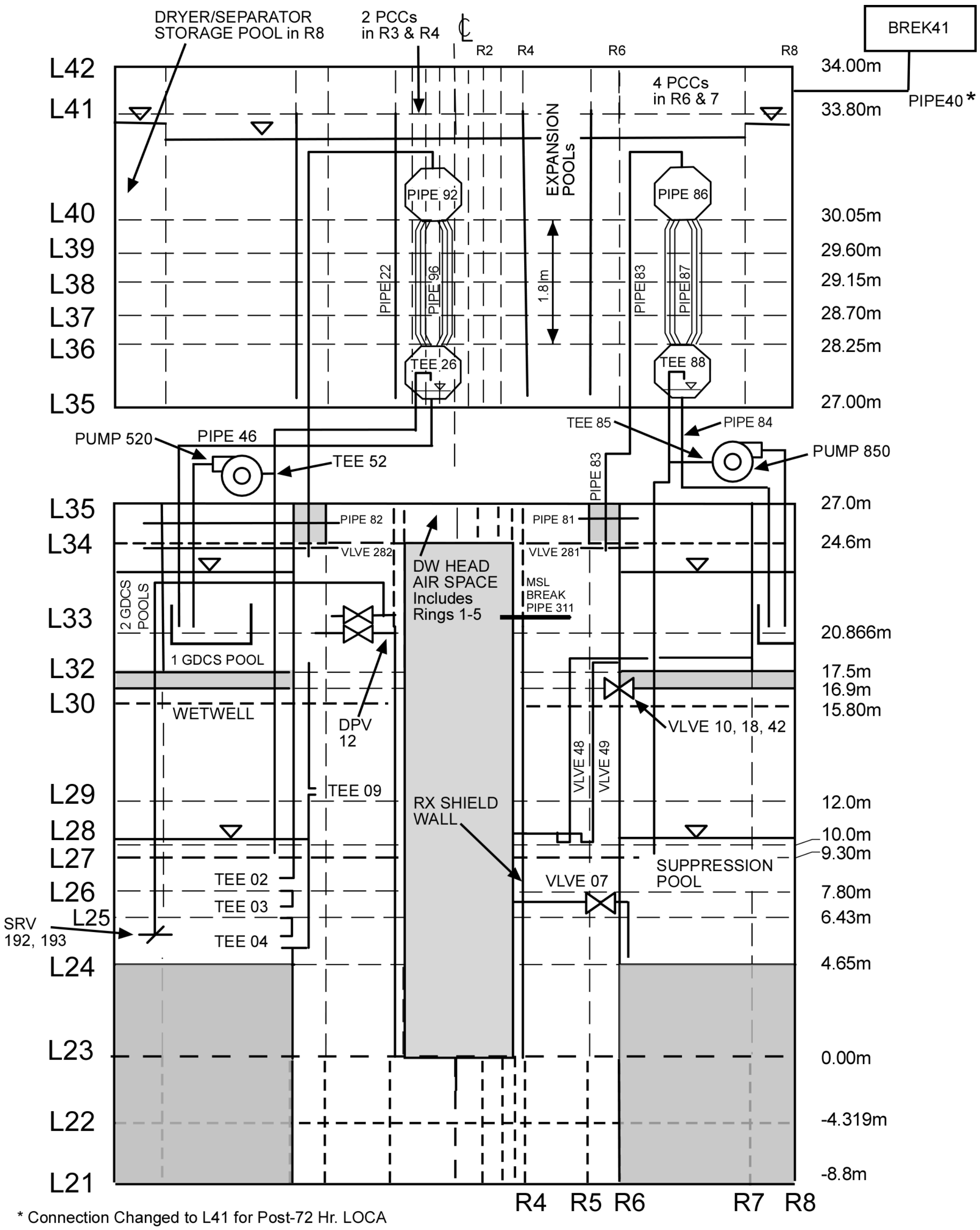


Figure 6.2-7. TRACG Nodalization of the ESBWR Containment

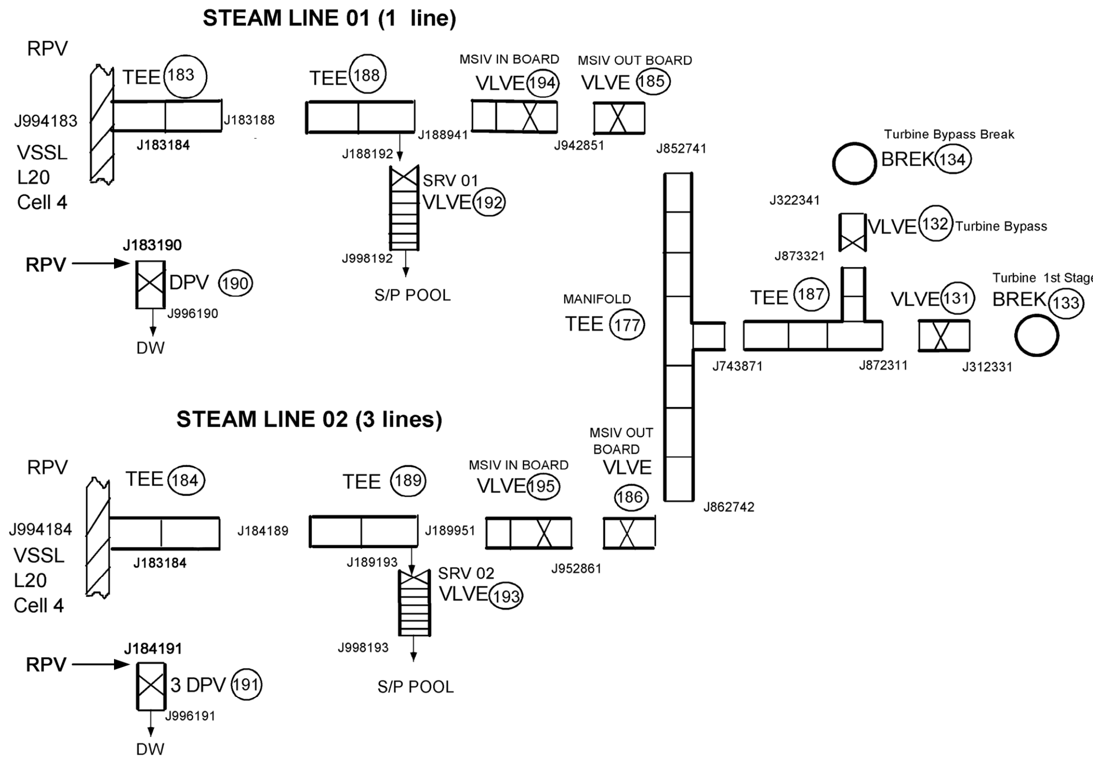
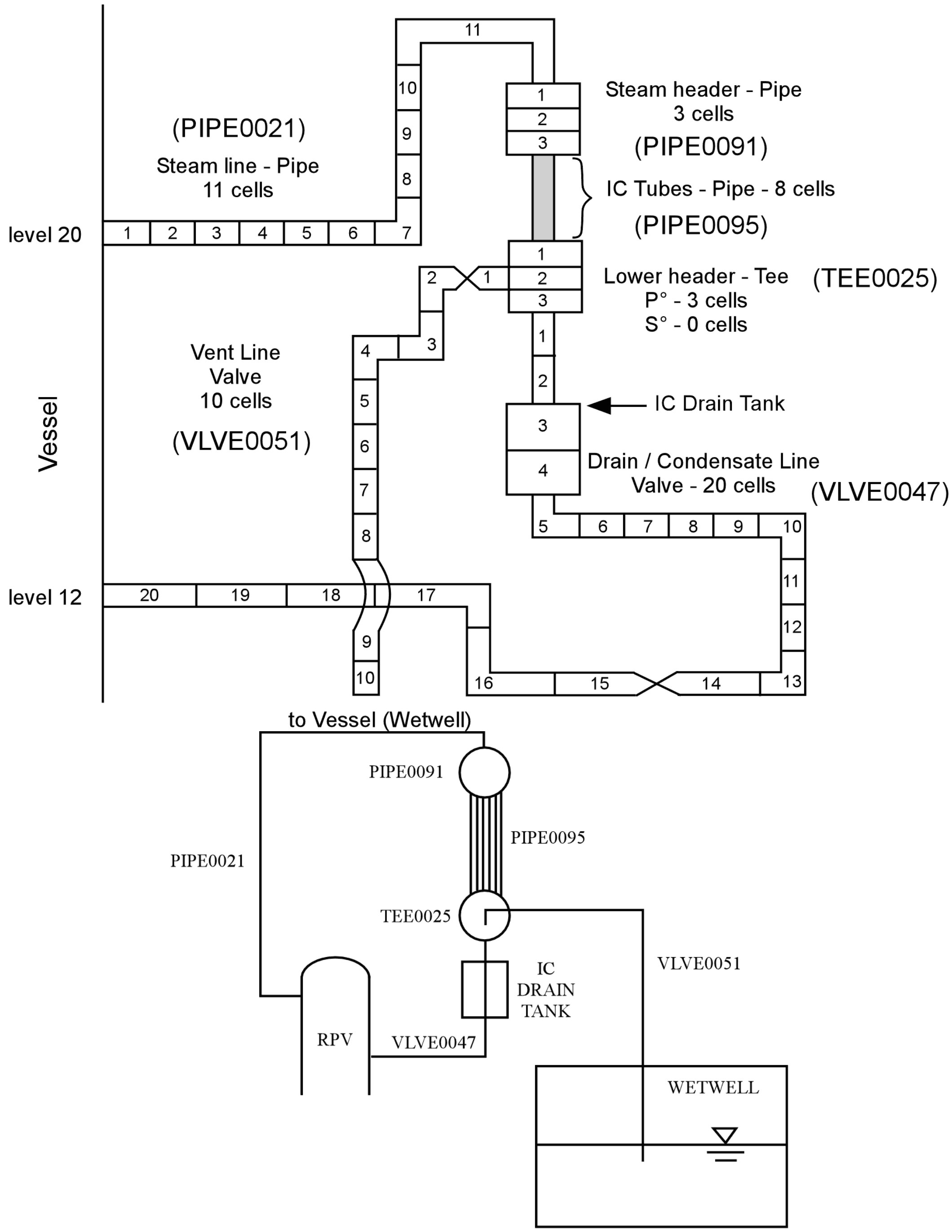


Figure 6.2-8. TRACG Nodalization of the ESBWR Main Steam Lines and DPVs



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

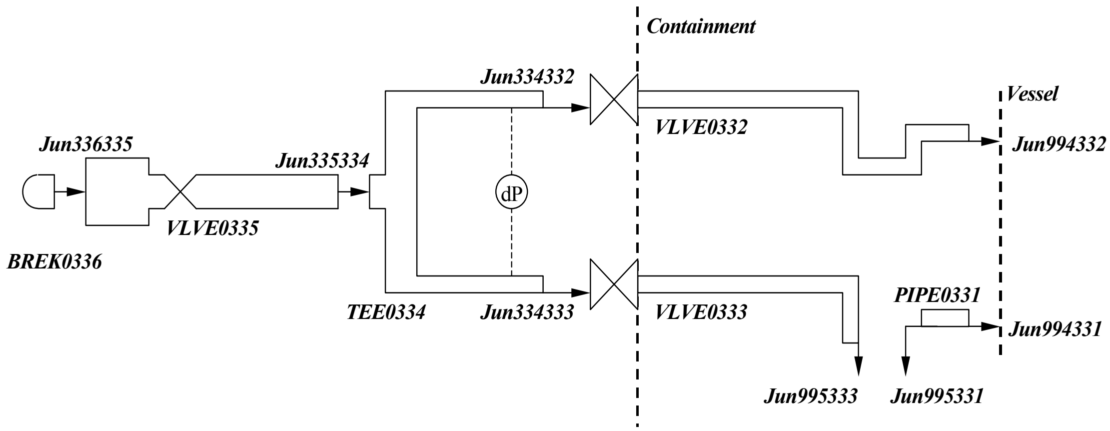


Figure 6.2-8b. TRACG Nodalization of the ESBWR Feedwater Line System

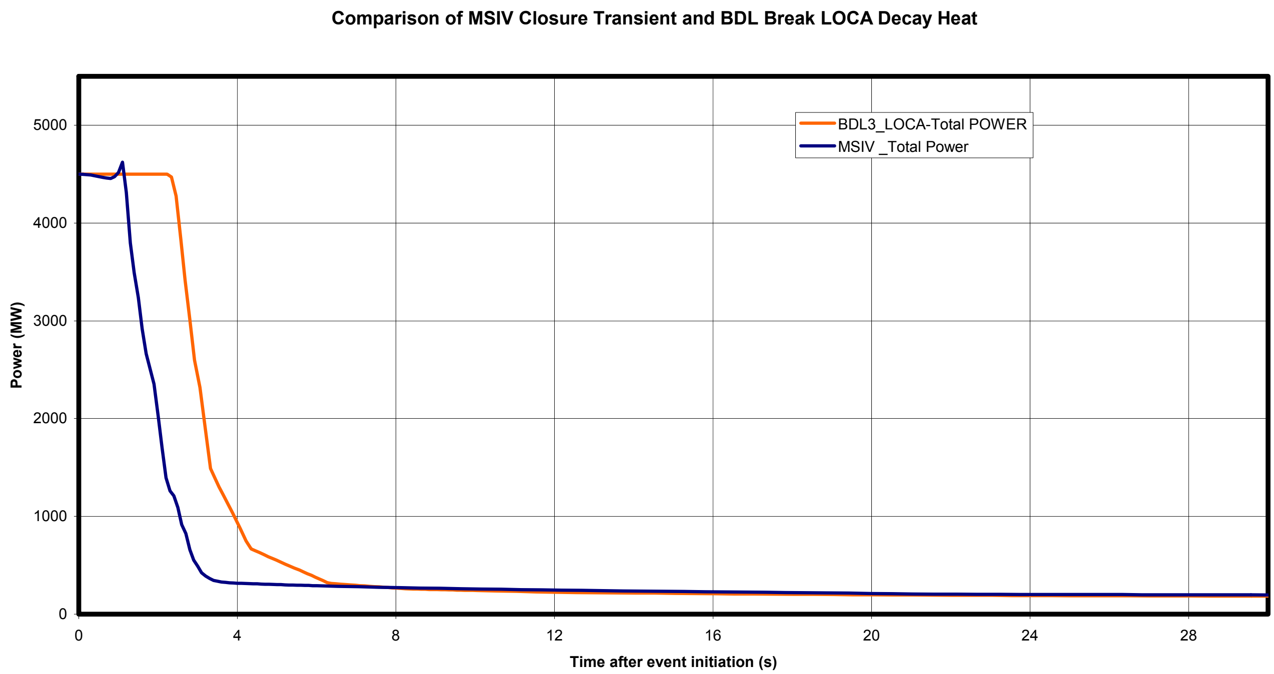


Figure 6.2-8c. ESBWR End-of-Cycle Core Average Decay Heat

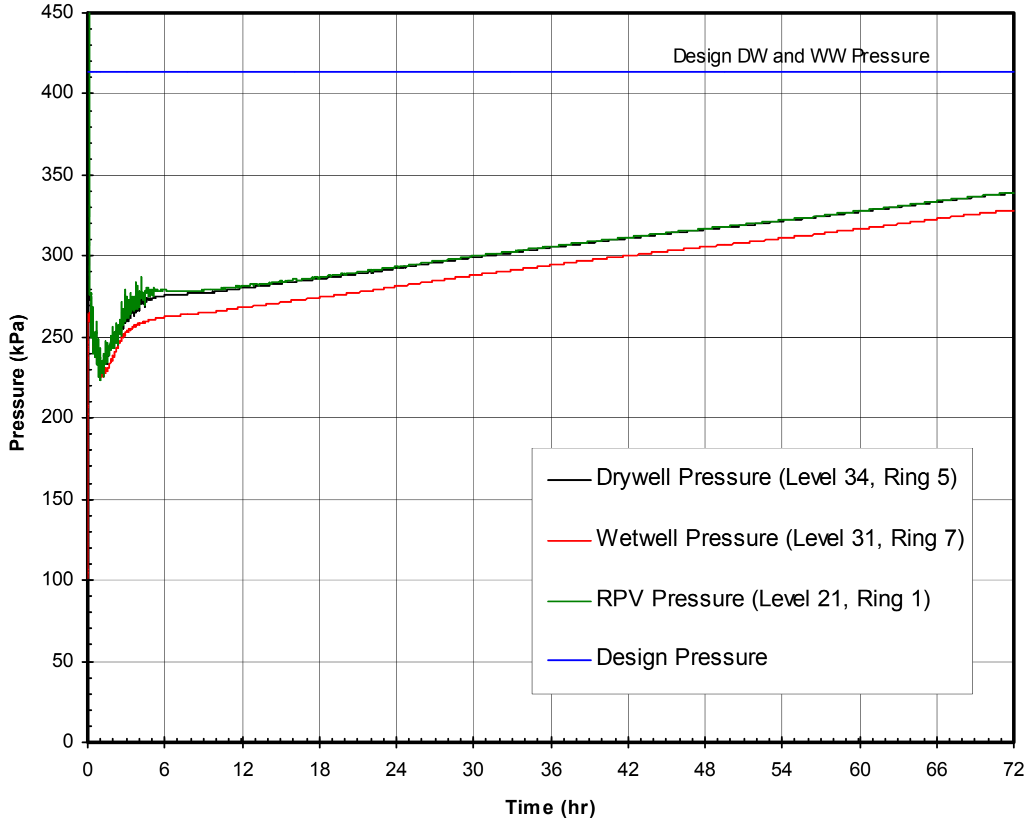


Figure 6.2-9a1. Feedwater Line Break (Nominal Case) – Containment Pressures (72 hrs)

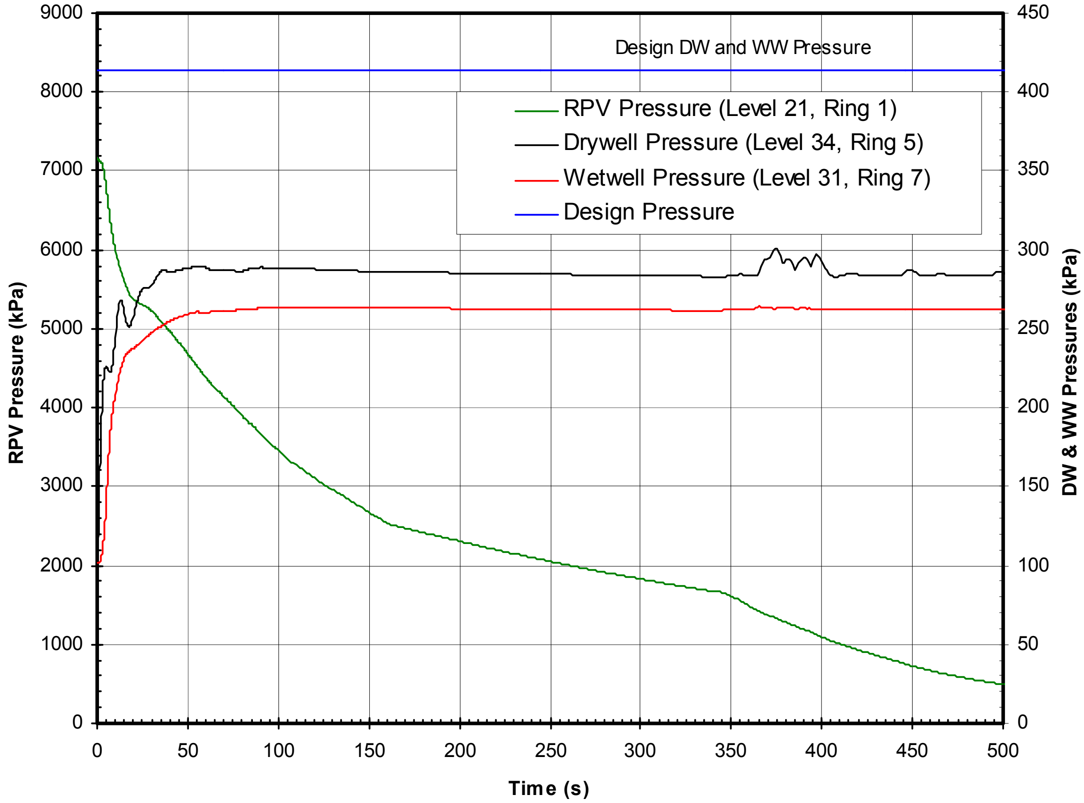


Figure 6.2-9a2. Feedwater Line Break (Nominal Case) – Containment Pressures (500 s)

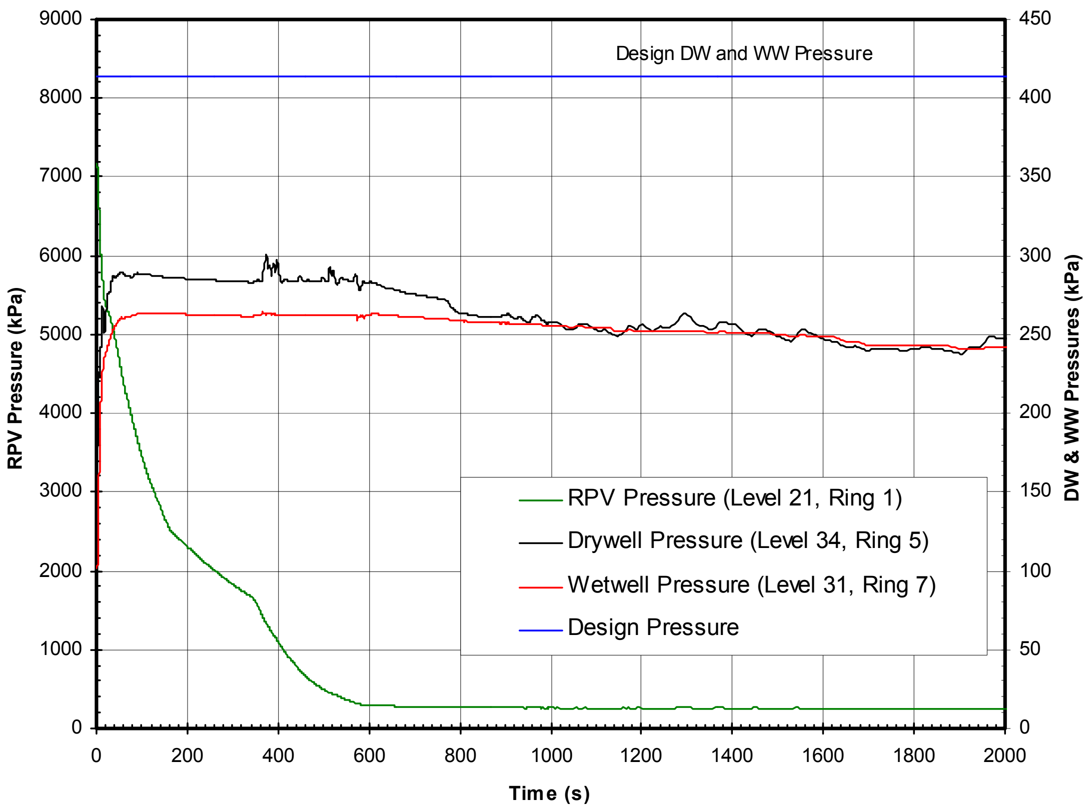


Figure 6.2-9a3. Feedwater Line Break (Nominal Case) – Containment Pressures (2000 s)

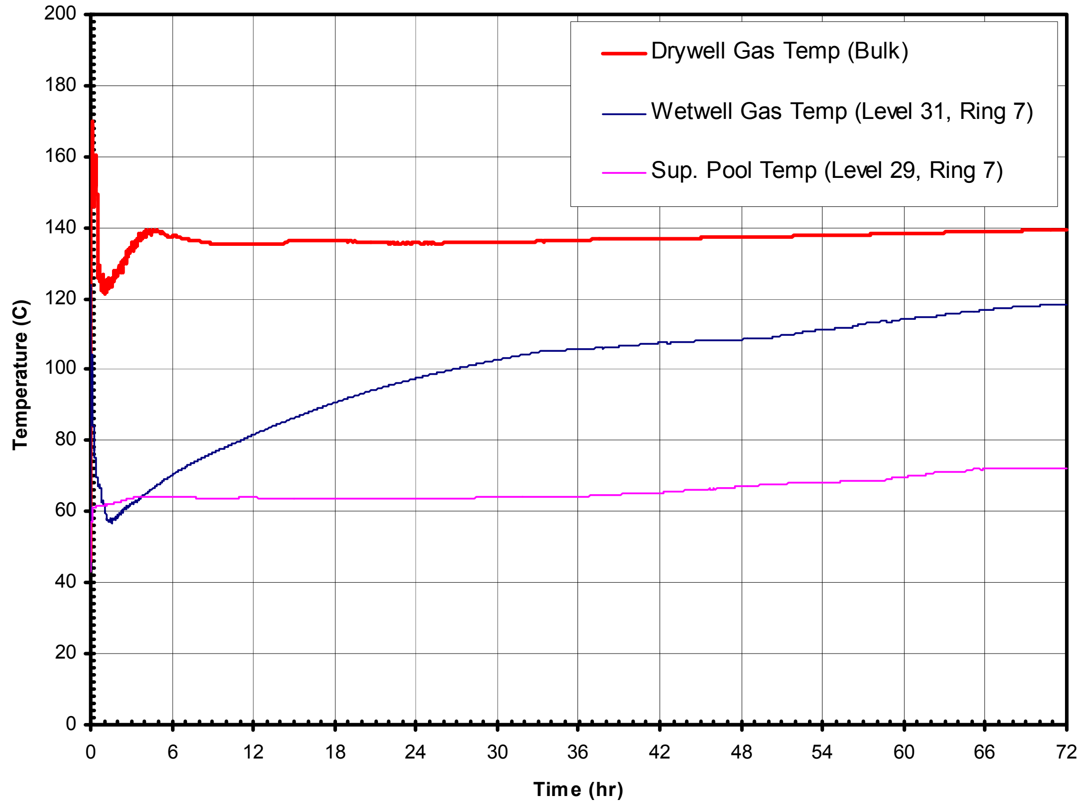


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Figure 6.2-9b1. Feedwater Line Break (Nominal Case) – Containment Temperatures (72 hrs)

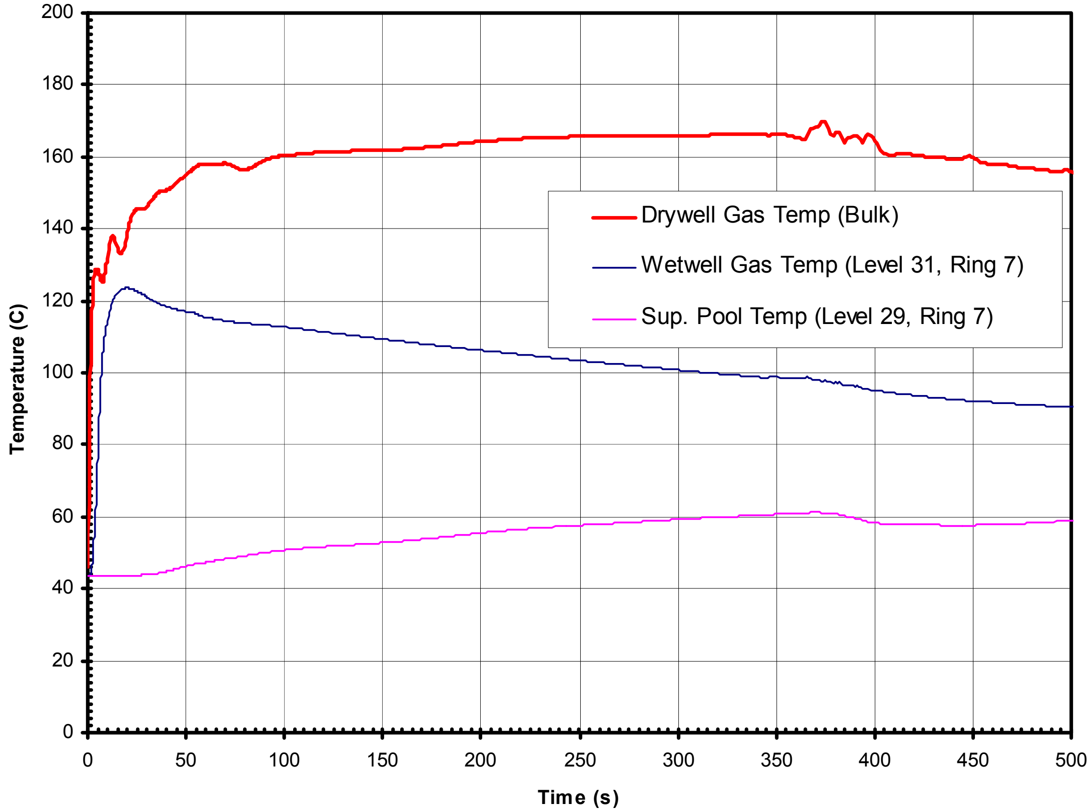

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Figure 6.2-9b2. Feedwater Line Break (Nominal Case) – Containment Temperatures (500 s)

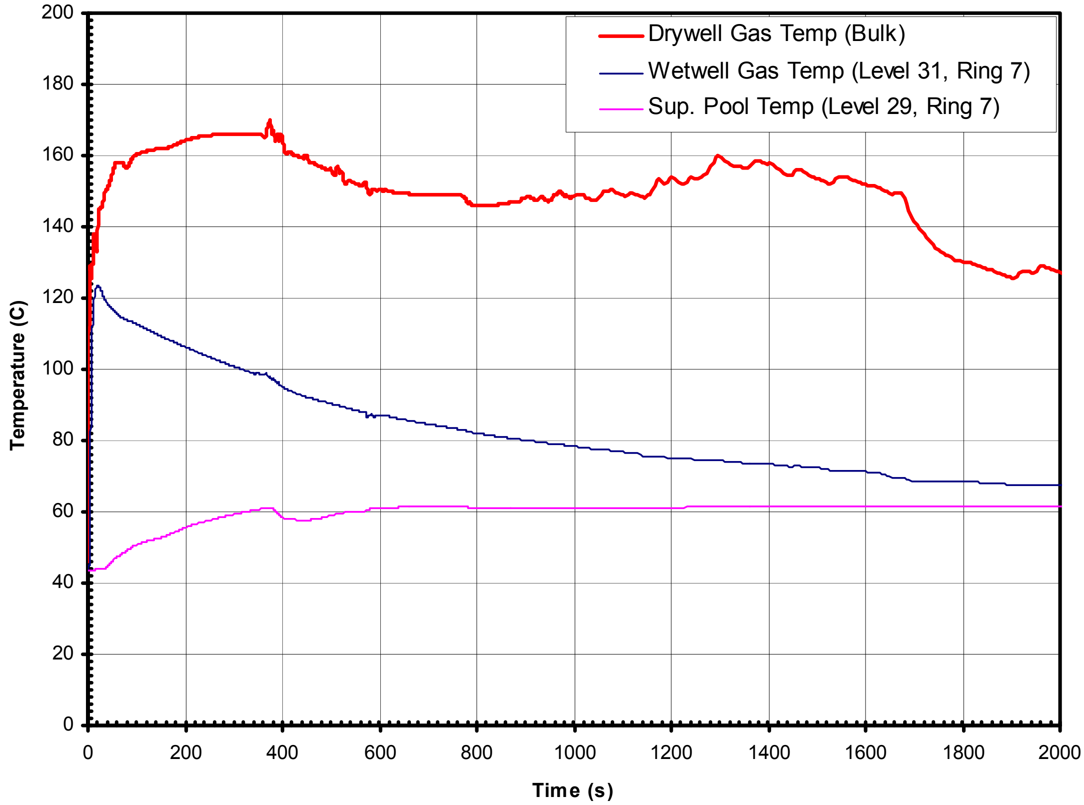

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Figure 6.2-9b3. Feedwater Line Break (Nominal Case) – Containment Temperatures (2000 s)

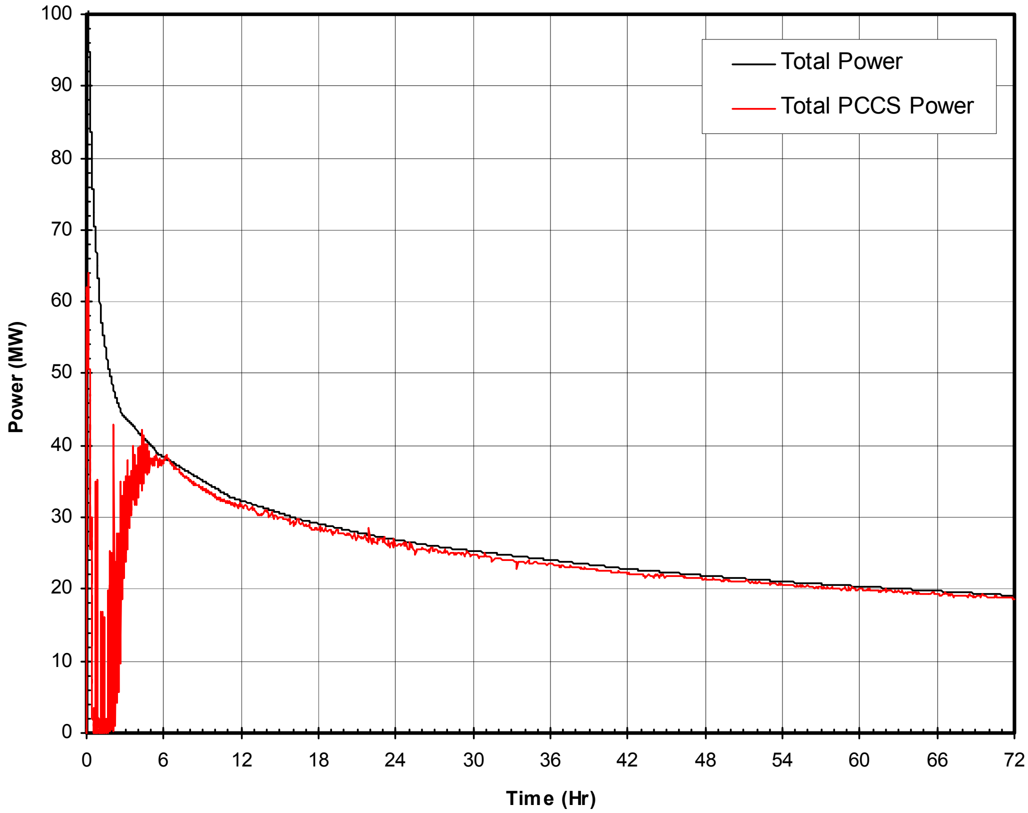

Figure 6.2-9c1. Feedwater Line Break (Nominal Case) – PCCS Heat Removal versus Decay Heat (72 hrs)

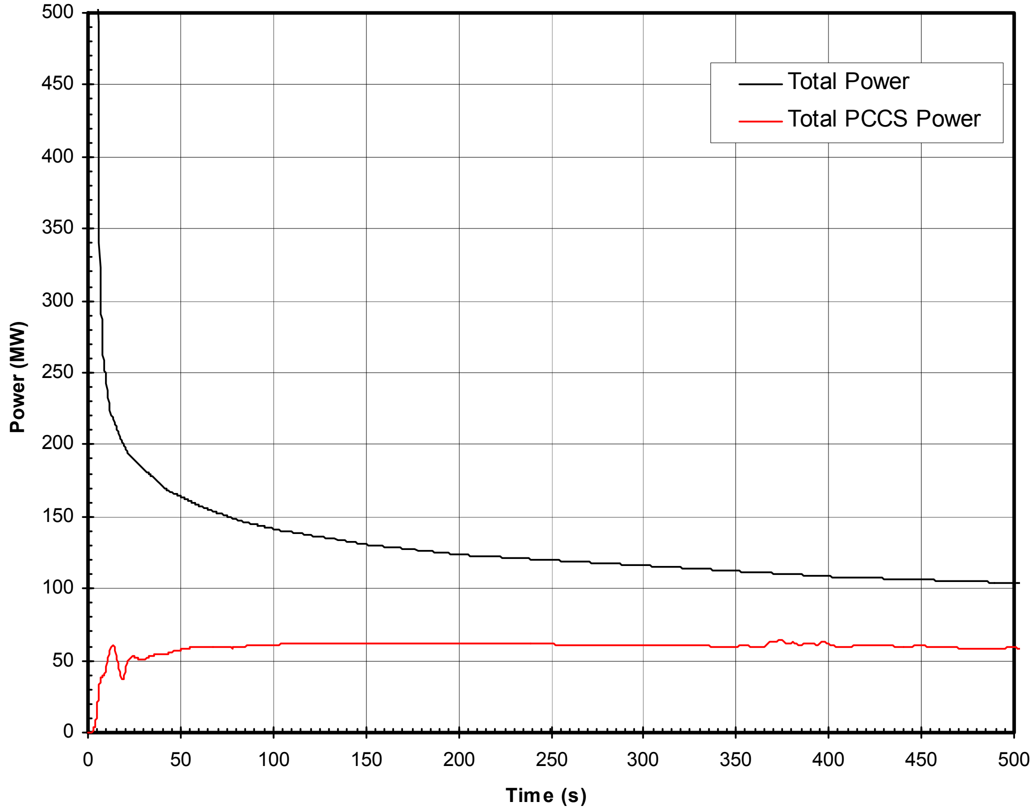

Figure 6.2-9c2. Feedwater Line Break (Nominal Case) – PCCS Heat Removal versus Decay Heat (500 s)

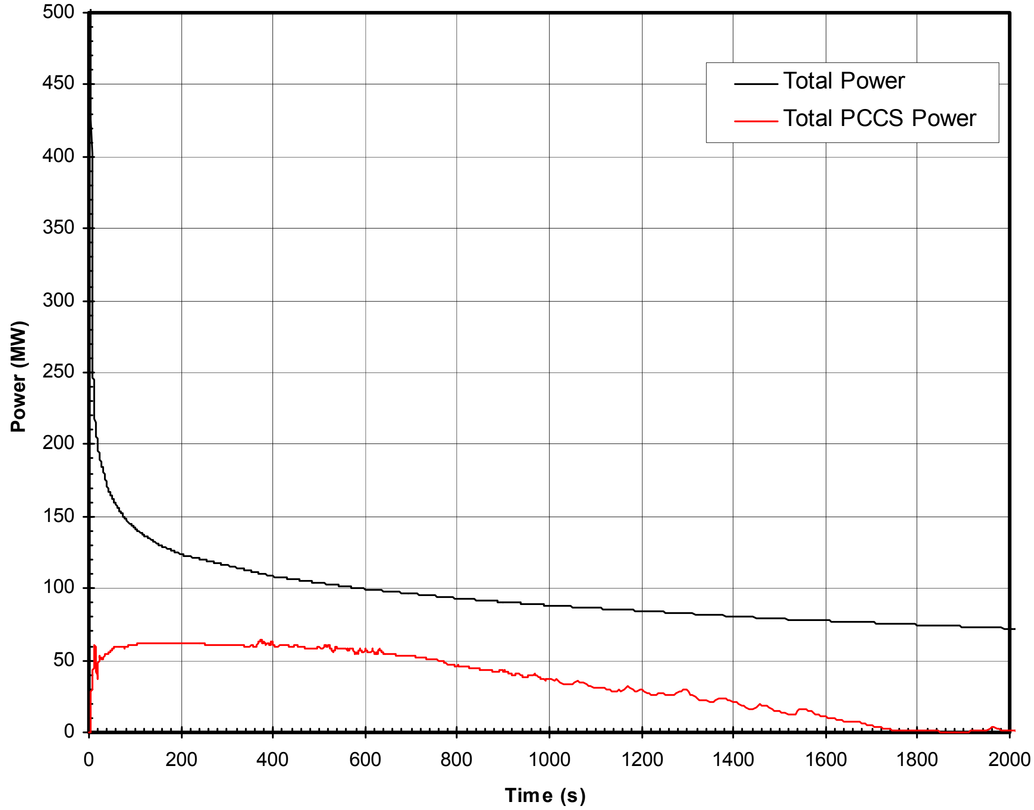

Figure 6.2-9c3. Feedwater Line Break (Nominal Case) – PCCS Heat Removal versus Decay Heat (2000 s)

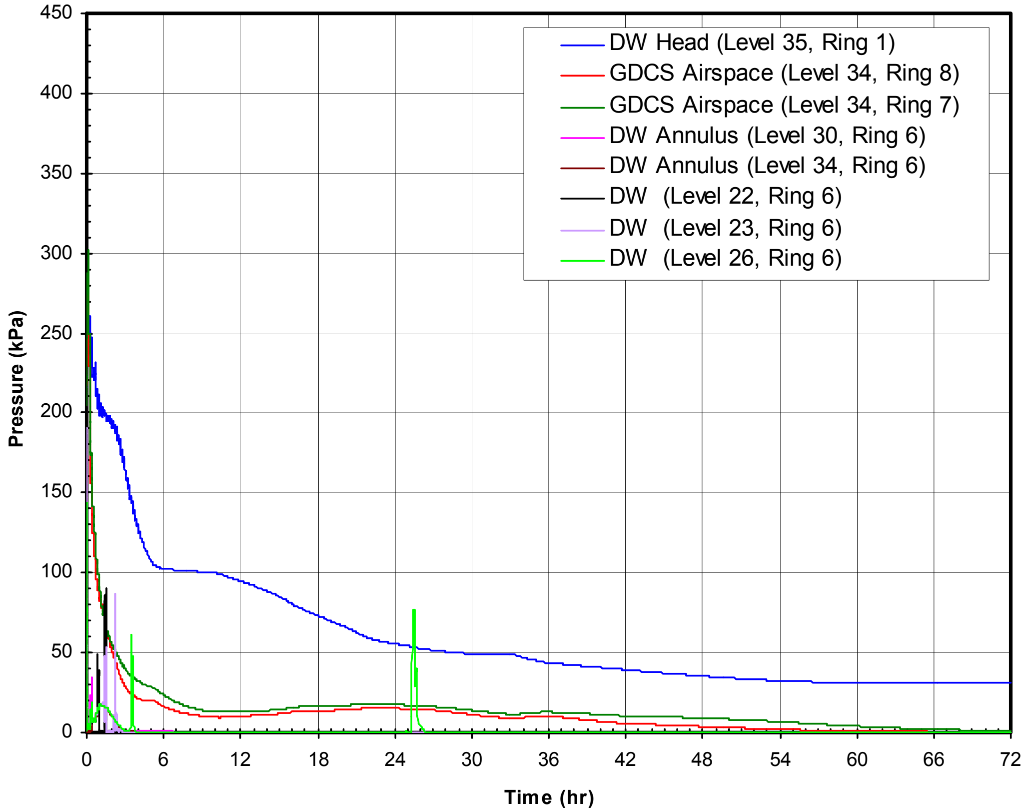

Figure 6.2-9d1. Feedwater Line Break (Nominal Case) - Drywell and GDCS Airspace Pressures (72 hrs)

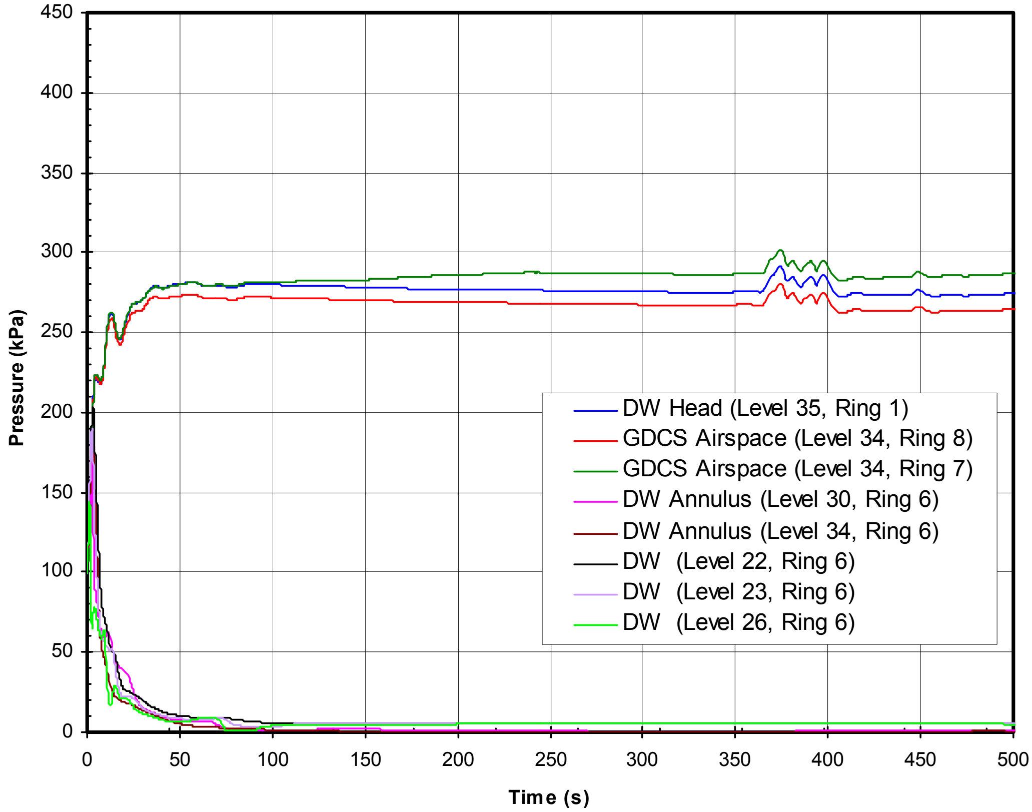

Figure 6.2-9d2. Feedwater Line Break (Nominal Case) - Drywell and GDCS Airspace Pressures (500 s)

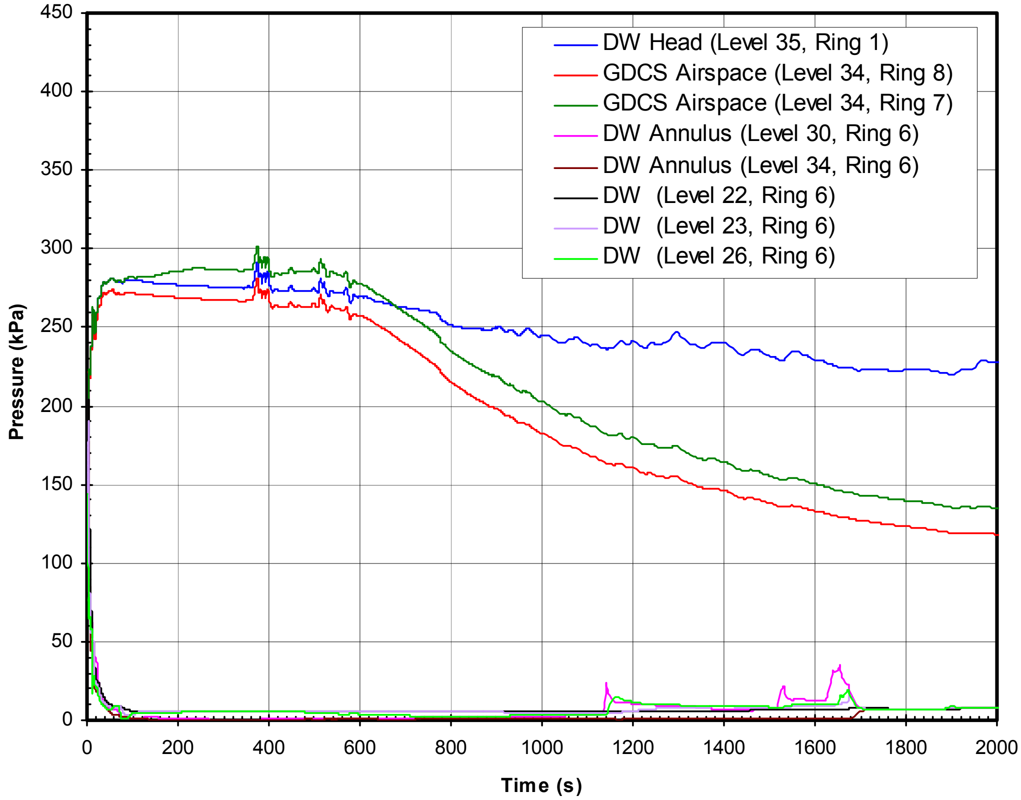

Figure 6.2-9d3. Feedwater Line Break (Nominal Case) - Drywell and GDCS Airspace Pressures (2000 s)

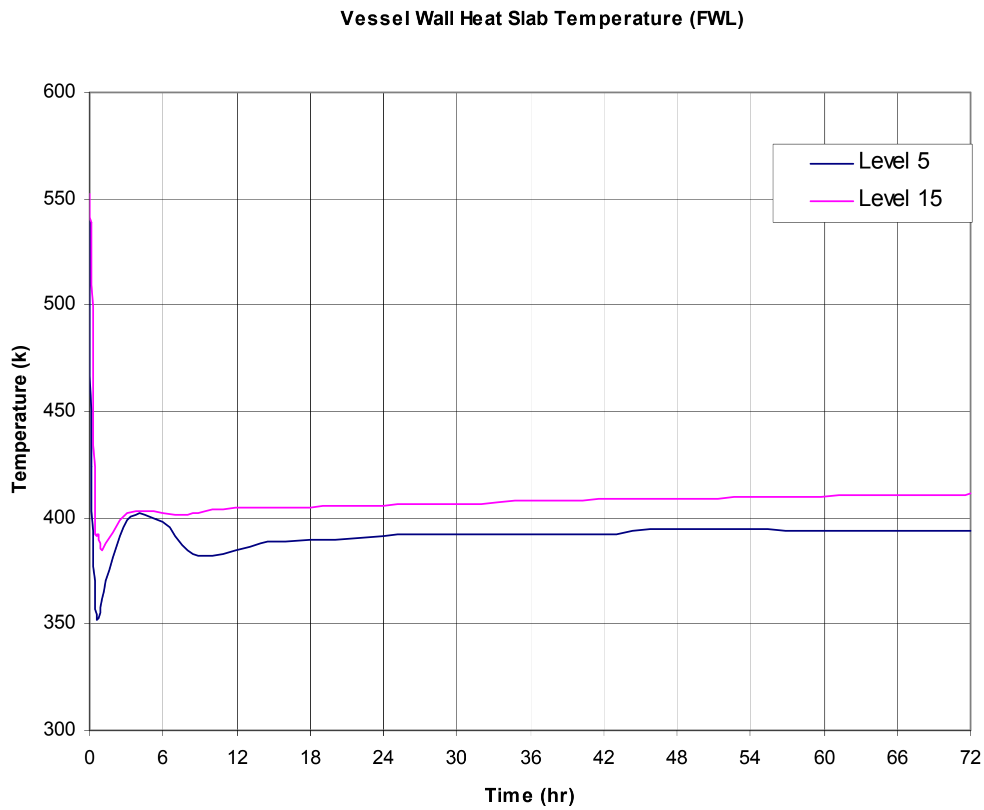

Figure 6.2-9e1. Vessel Wall Heat Slab Temperature (FWL) (72 hrs)

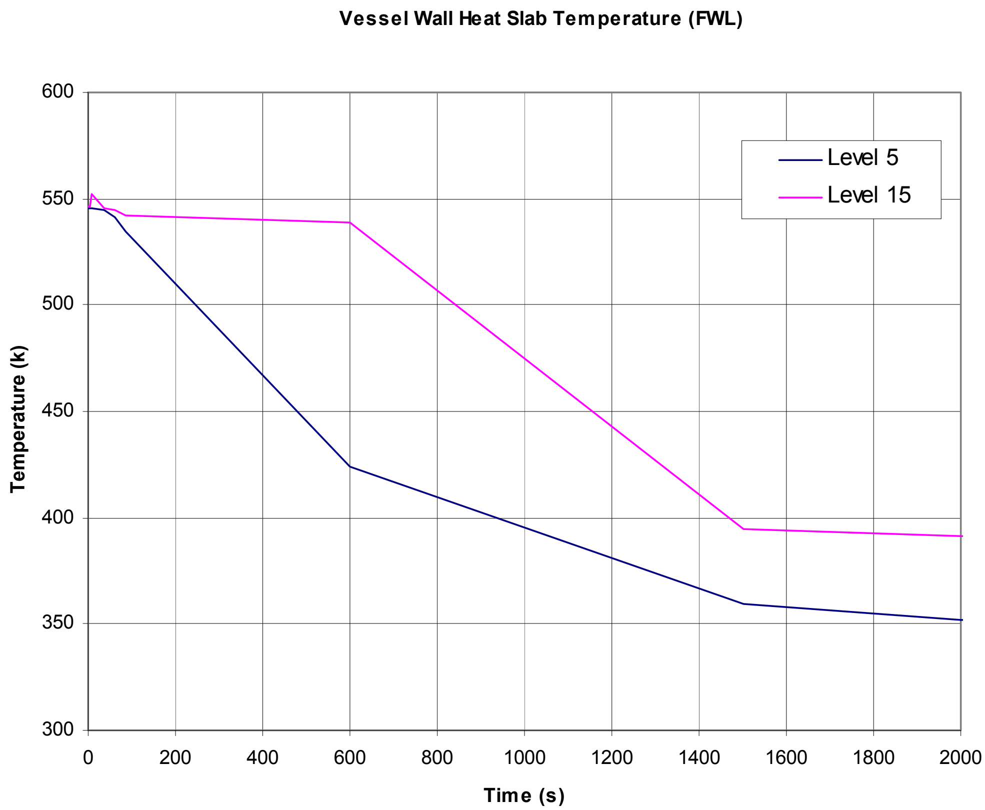

Figure 6.2-9e2. Vessel Wall Heat Slab Temperature (FWL) (2000 s))

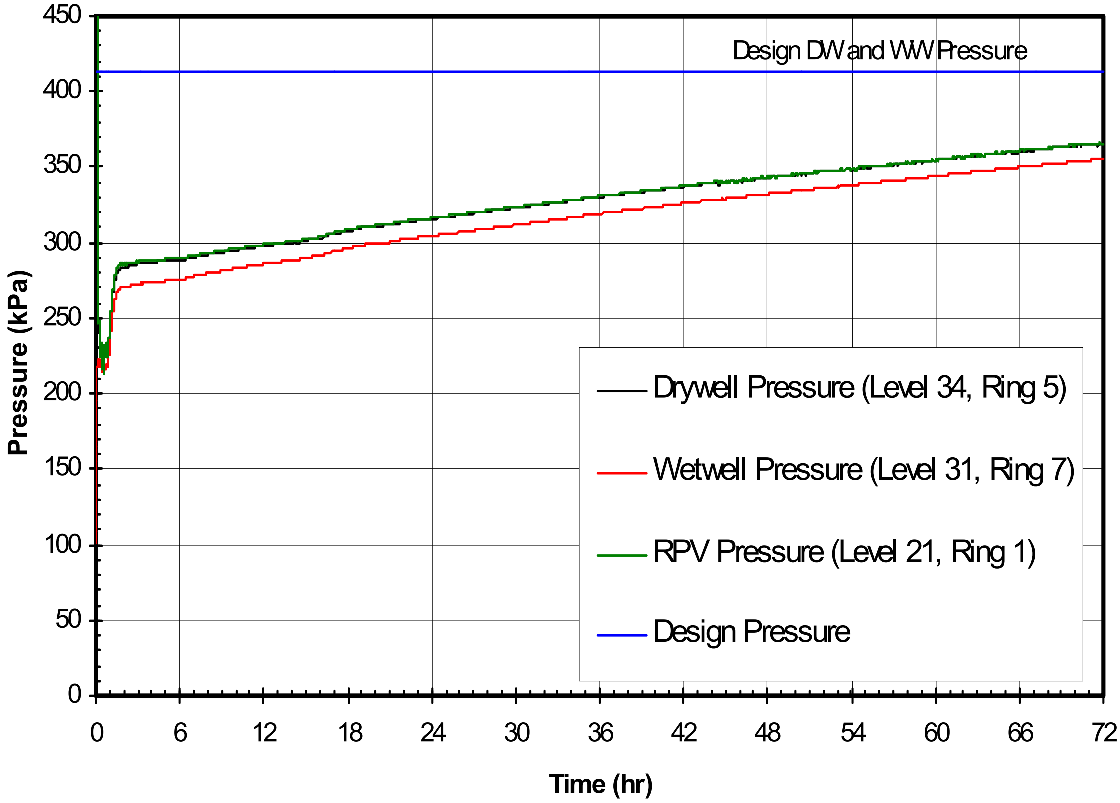

Figure 6.2-10a1. Main Steam Line Break (Nominal Case) – Containment Pressures (72 hrs)

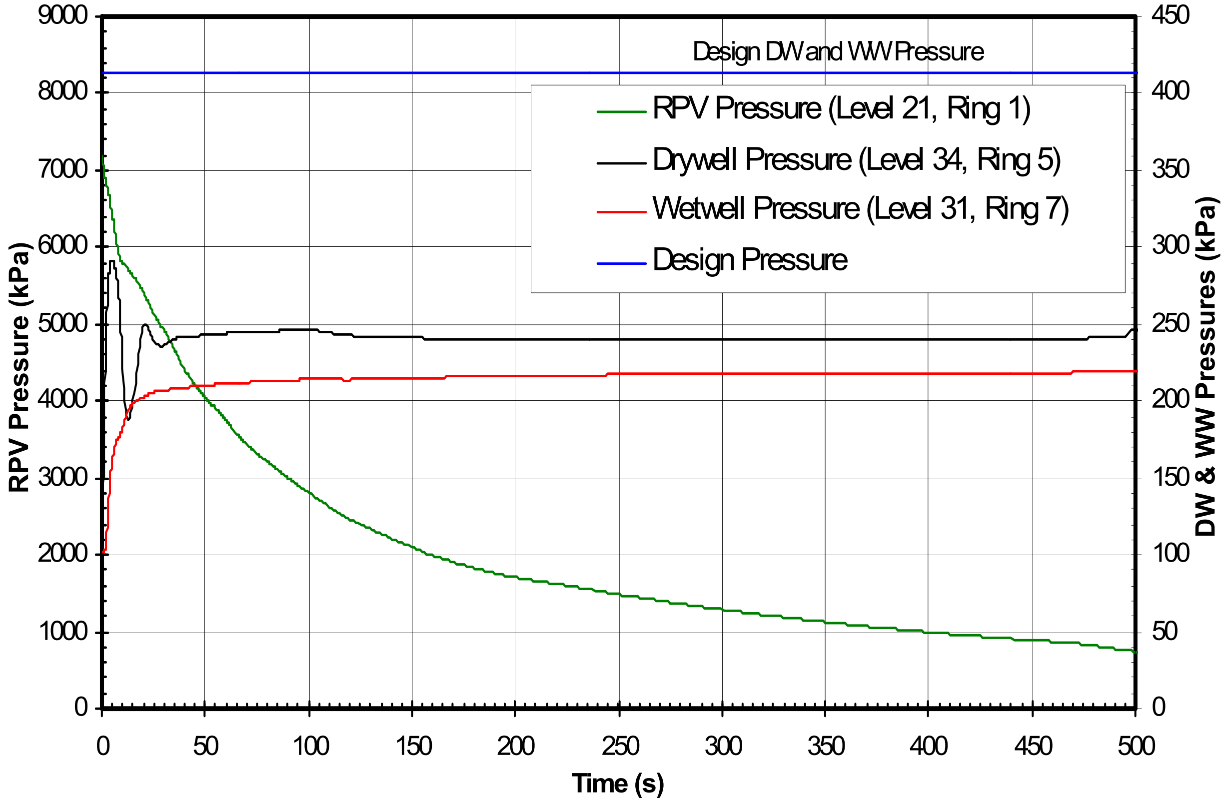

Figure 6.2-10a2. Main Steam Line Break (Nominal Case) – Containment Pressures (500 s)

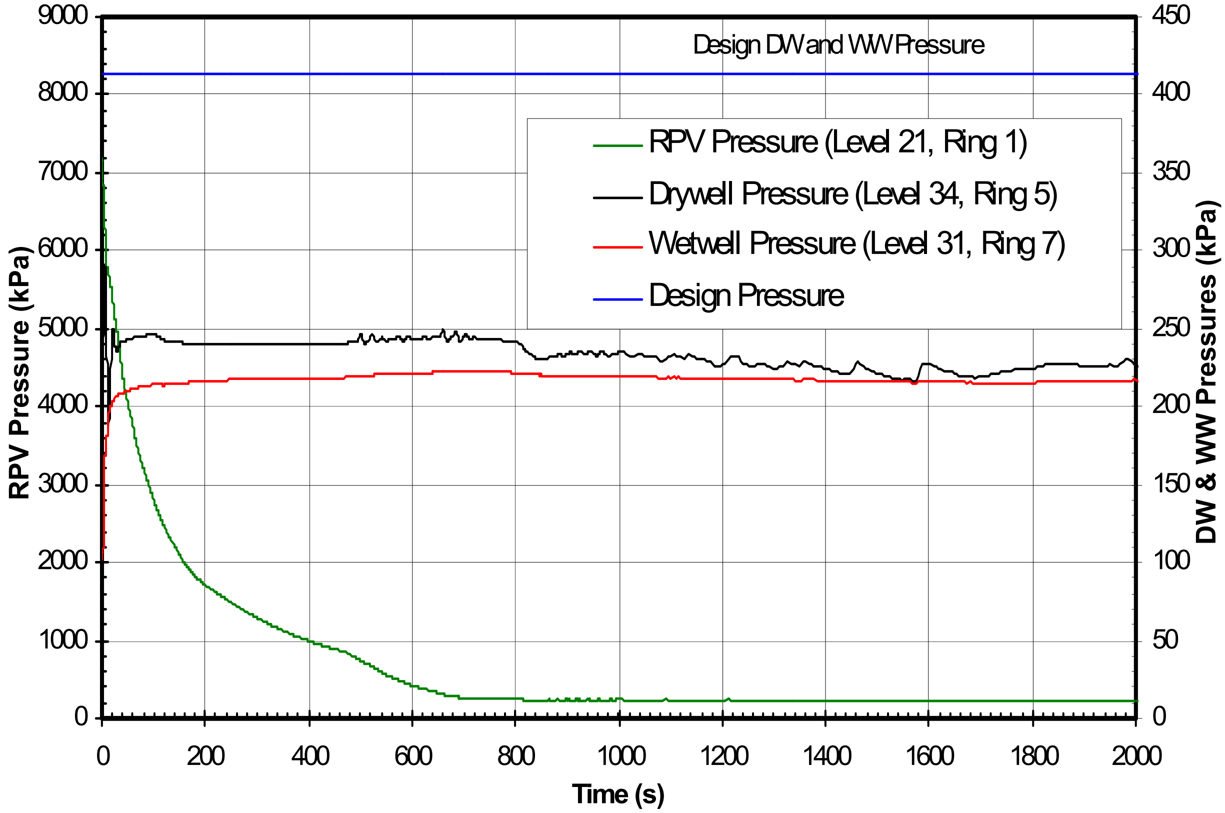

Figure 6.2-10a3. Main Steam Line Break (Nominal Case) - Containment Pressures (2000 s)

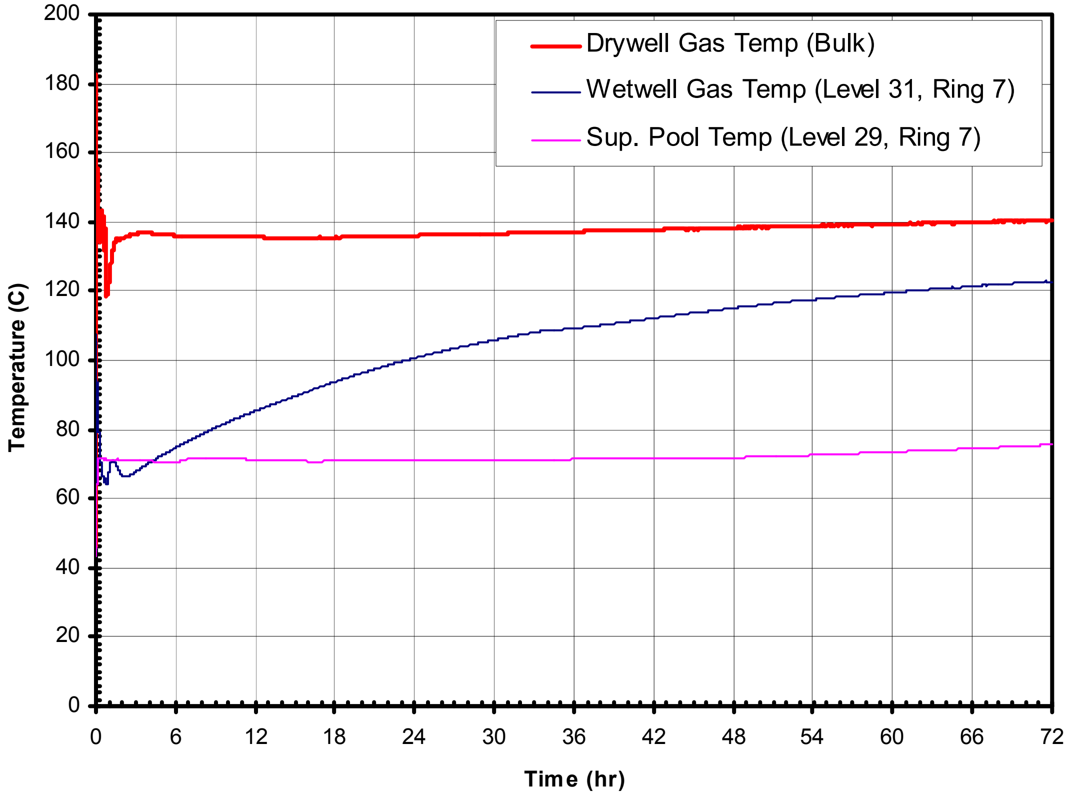

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Figure 6.2-10b1. Main Steam Line Break (Nominal Case) – Containment Temperatures (72 hrs)

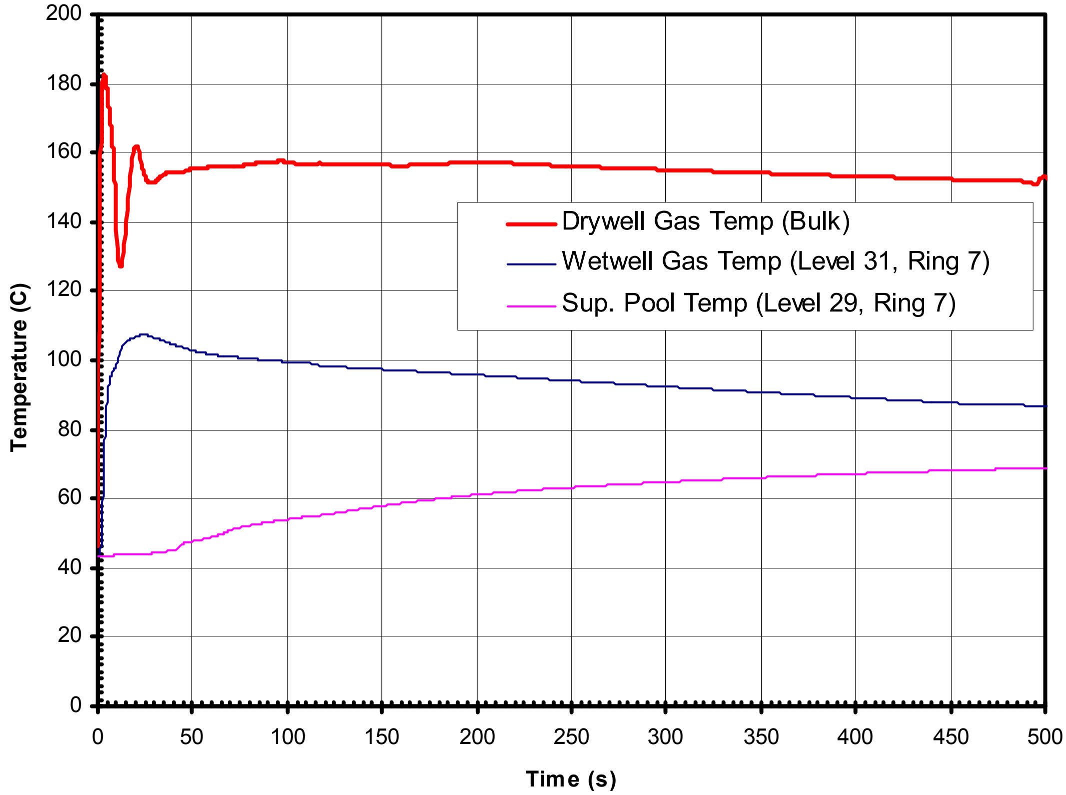

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Figure 6.2-10b2. Main Steam Line Break (Nominal Case) – Containment Temperatures (500 s)

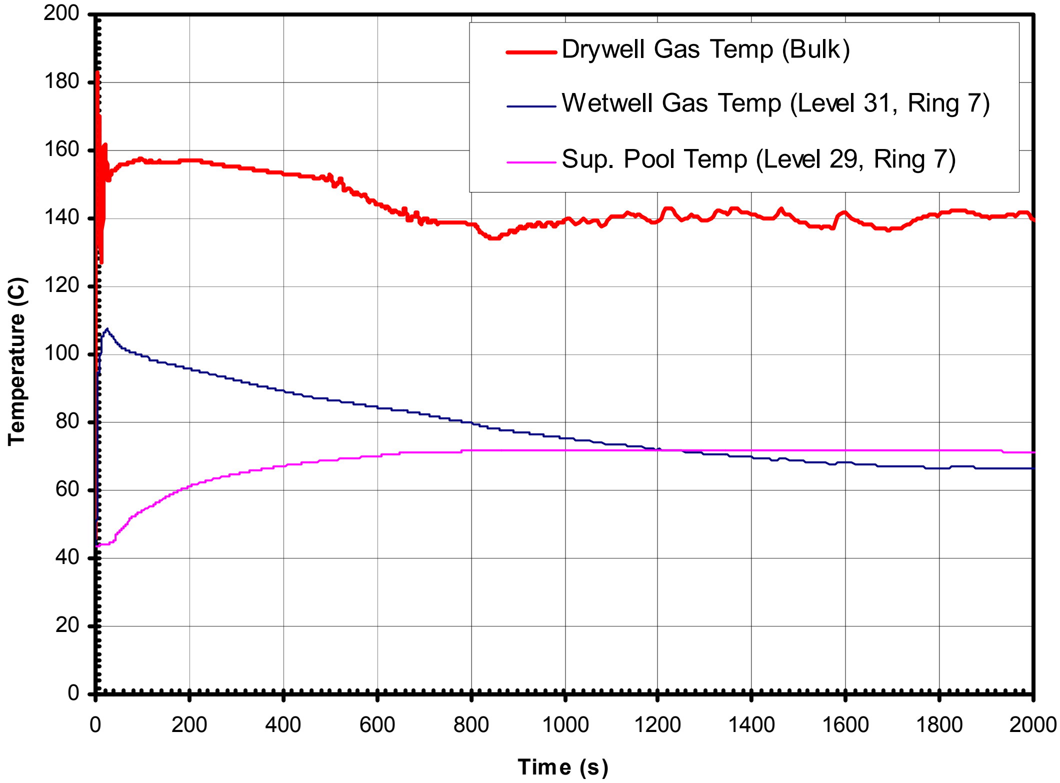

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Figure 6.2-10b3. Main Steam Line Break (Nominal Case) – Containment Temperatures (2000 s)

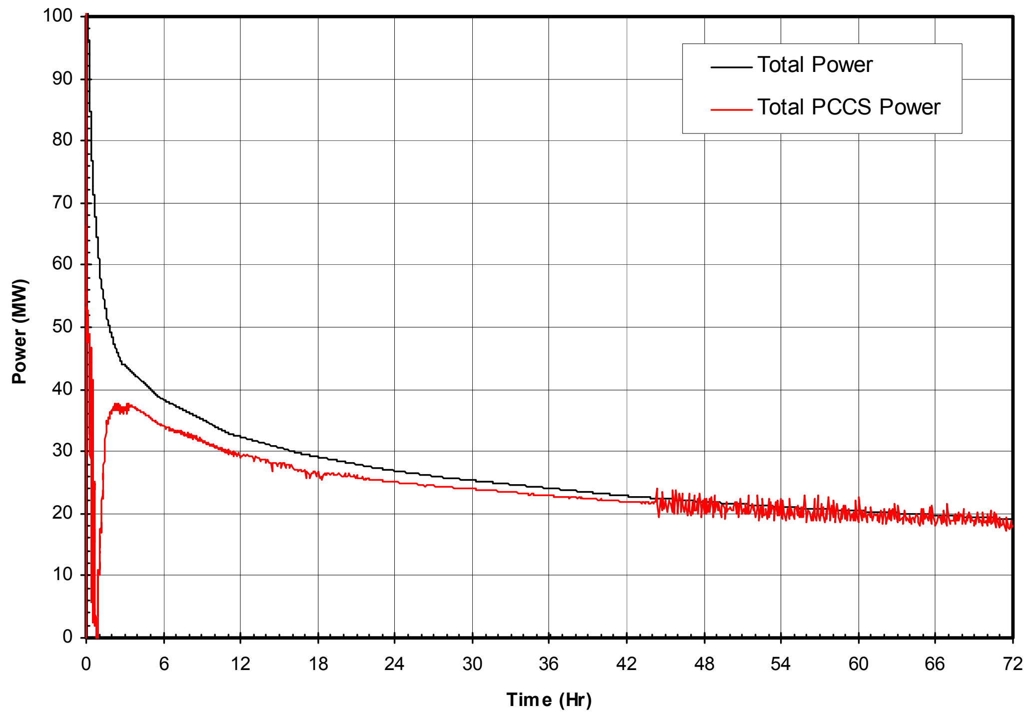

Figure 6.2-10c1. Main Steam Line Break (Nominal Case) – PCCS Heat Removal versus Decay Heat (72 hrs)

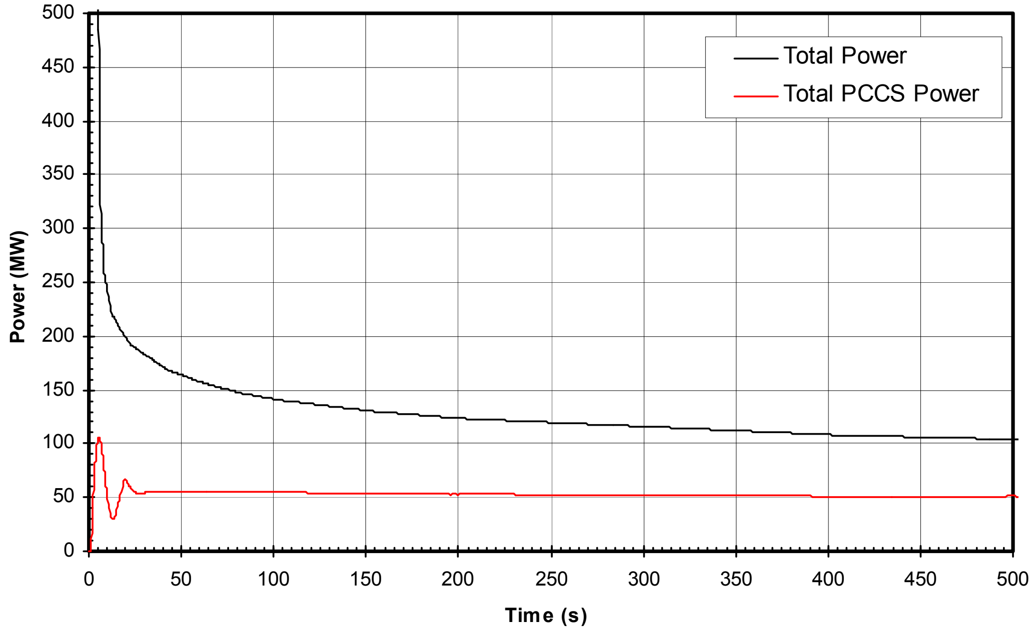

Figure 6.2-10c2. Main Steam Line Break (Nominal Case) – PCCS Heat Removal versus Decay Heat (500 s)

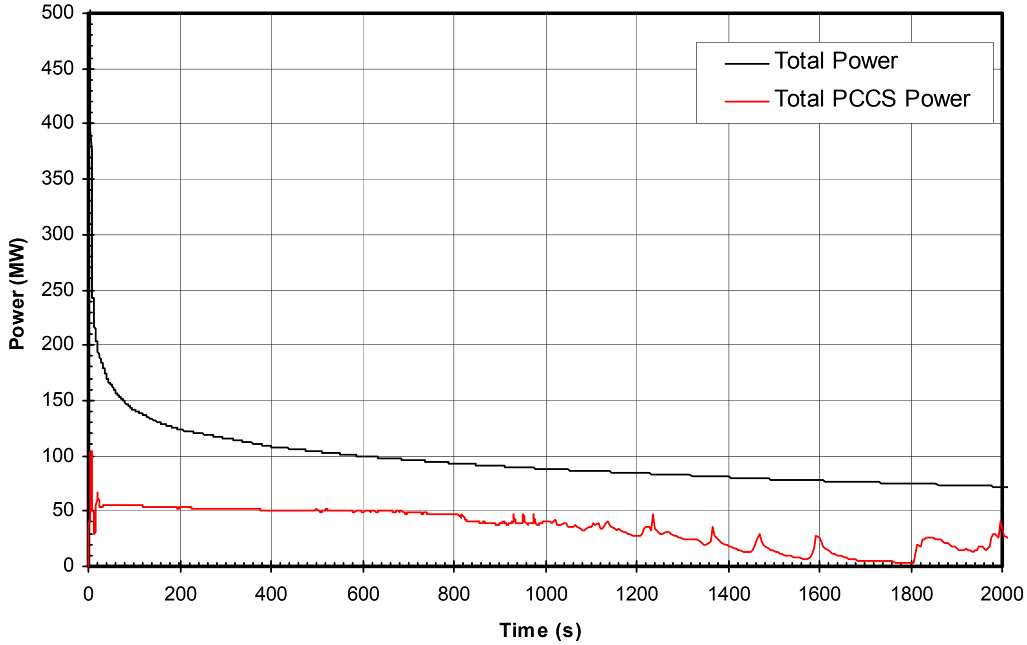

Figure 6.2-10c3. Main Steam Line Break (Nominal Case) – PCCS Heat Removal versus Decay Heat (2000 s)

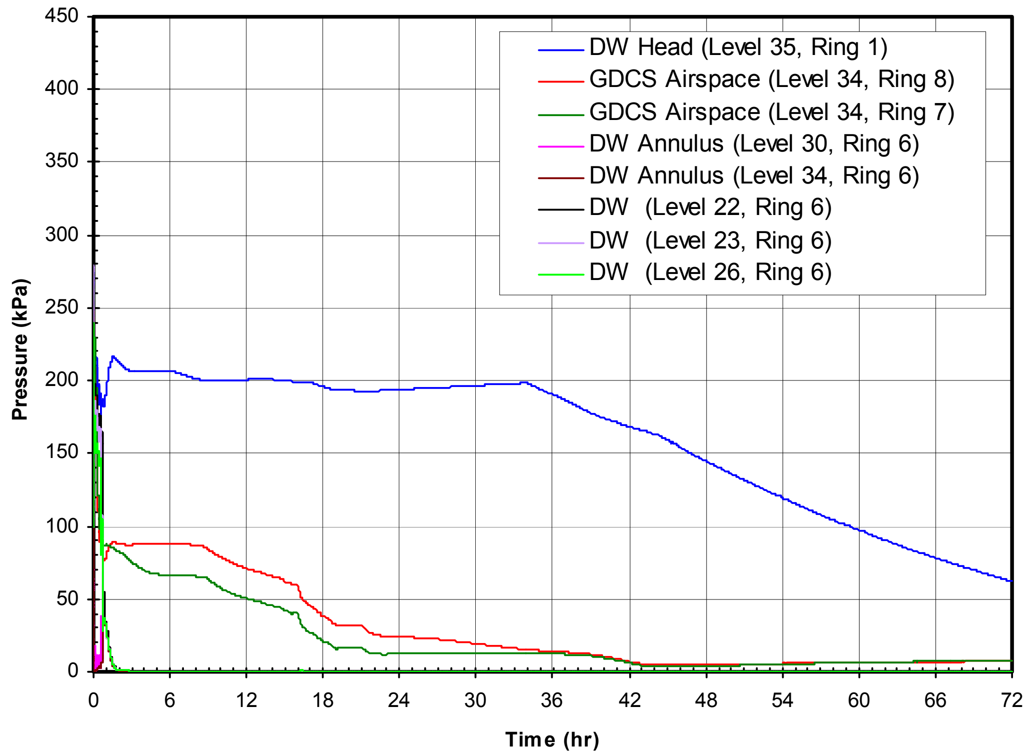

Figure 6.2-10d1. Main Steam Line Break (Nominal Case) - Drywell and GDCS Noncondensable Gas Pressures (72 hrs)

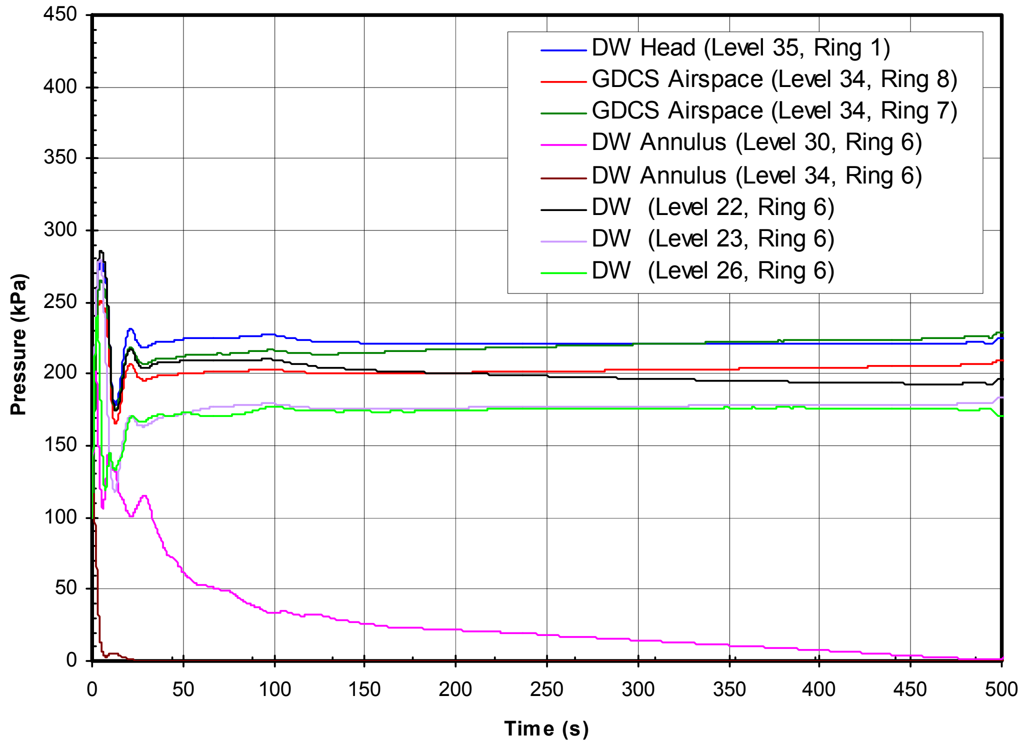

Figure 6.2-10d2. Main Steam Line Break (Nominal Case) - Drywell and GDCS Noncondesables Gas Pressures (500 s)

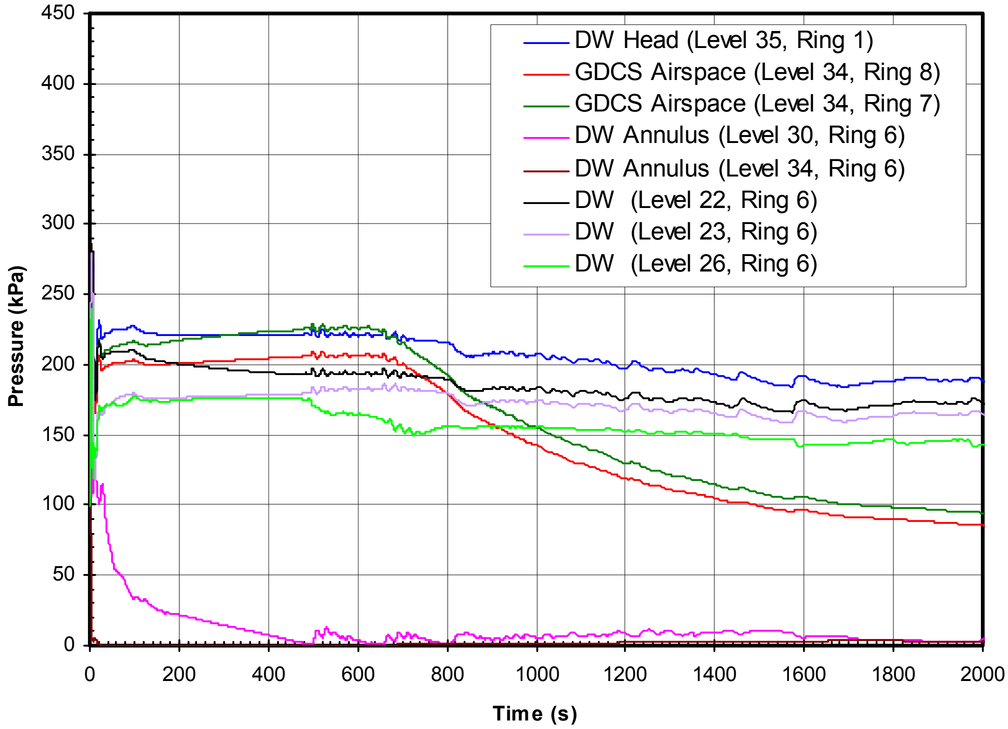

Figure 6.2-10d3. Main Steam Line Break (Nominal Case) - Drywell and GDCS Noncondensable Gas Pressures (2000 s)

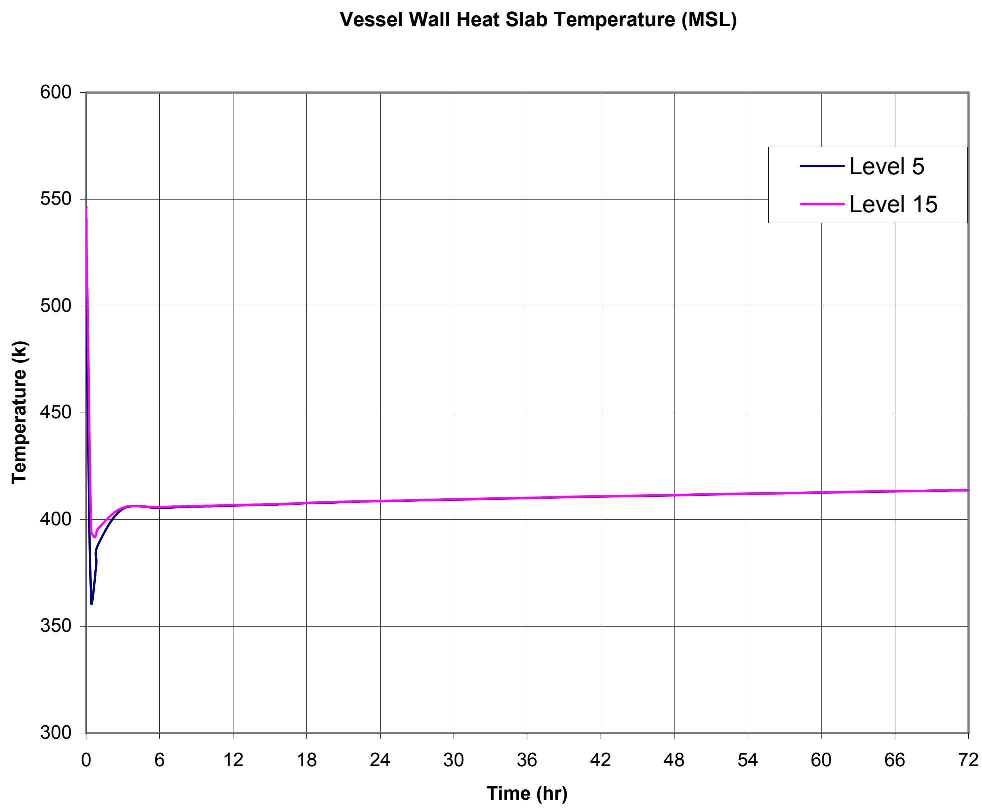

Figure 6.2-10e1. Vessel Wall Heat Slab Temperature (MSL) (72 hrs)

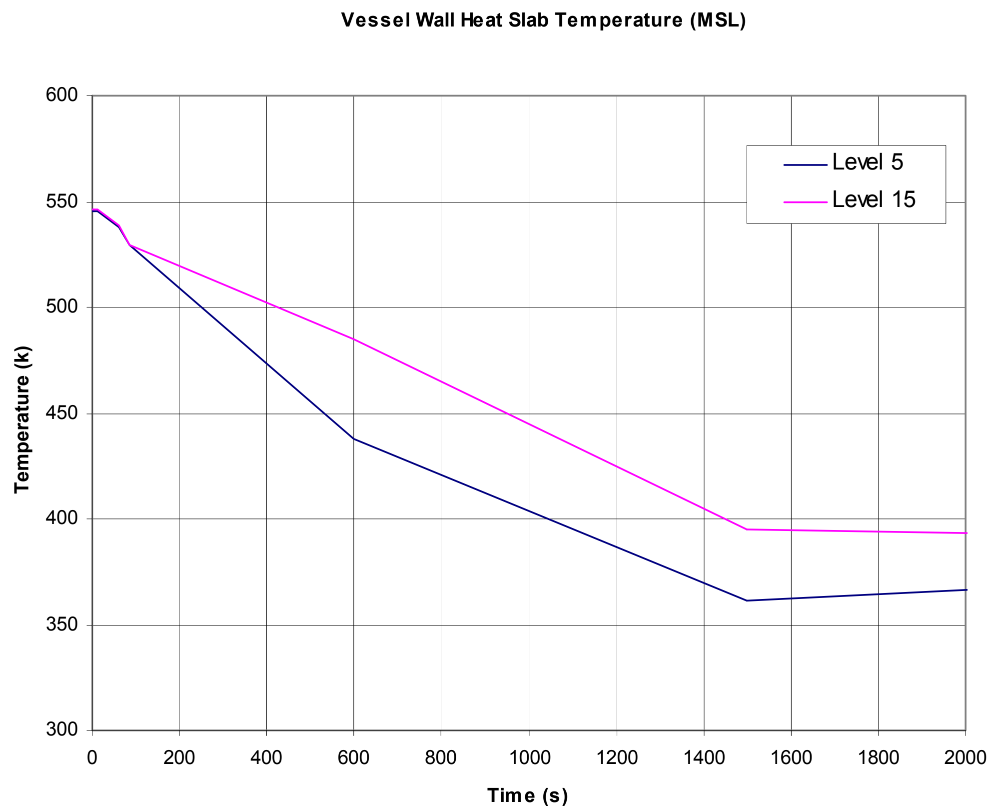

Figure 6.2-10e2. Vessel Wall Heat Slab Temperature (MSL) (2000 s)

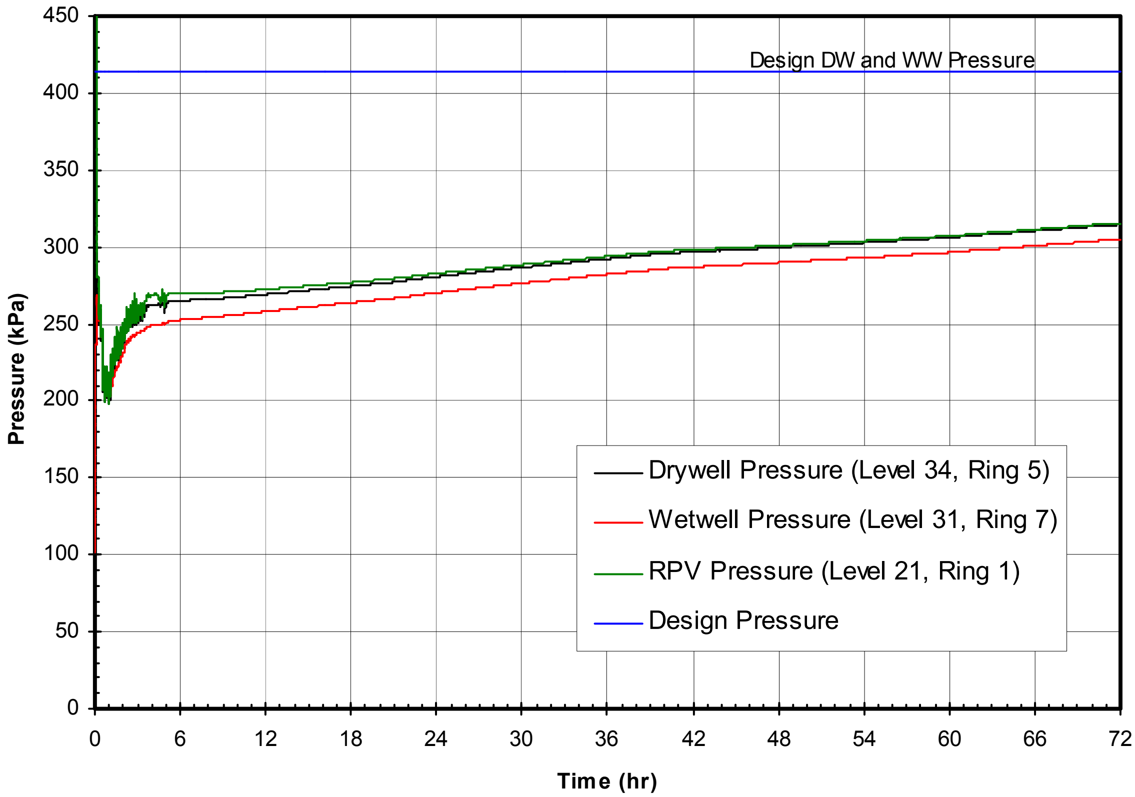

Figure 6.2-11a1. GDCS Line Break (Nominal Case) – Containment Pressures (72 hrs)

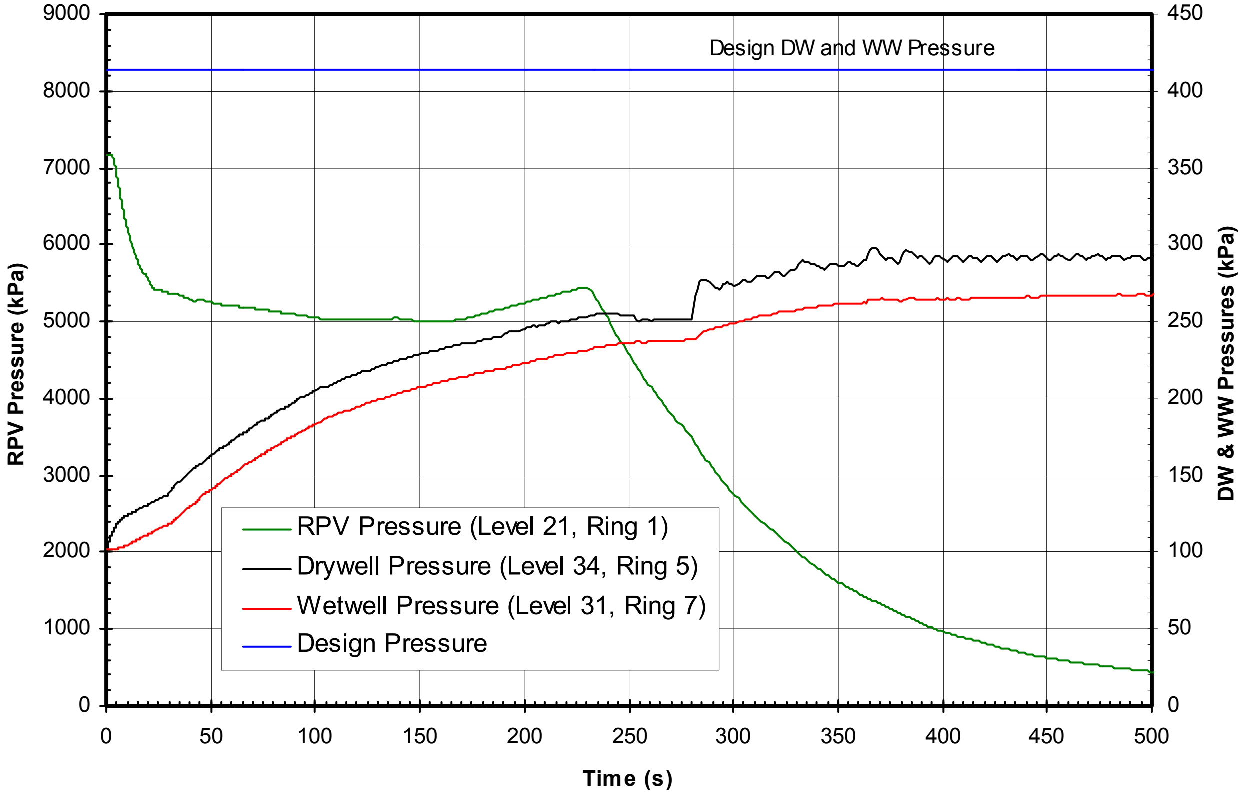

Figure 6.2-11a2. GDCS Line Break (Nominal Case) – Containment Pressures (500 s)

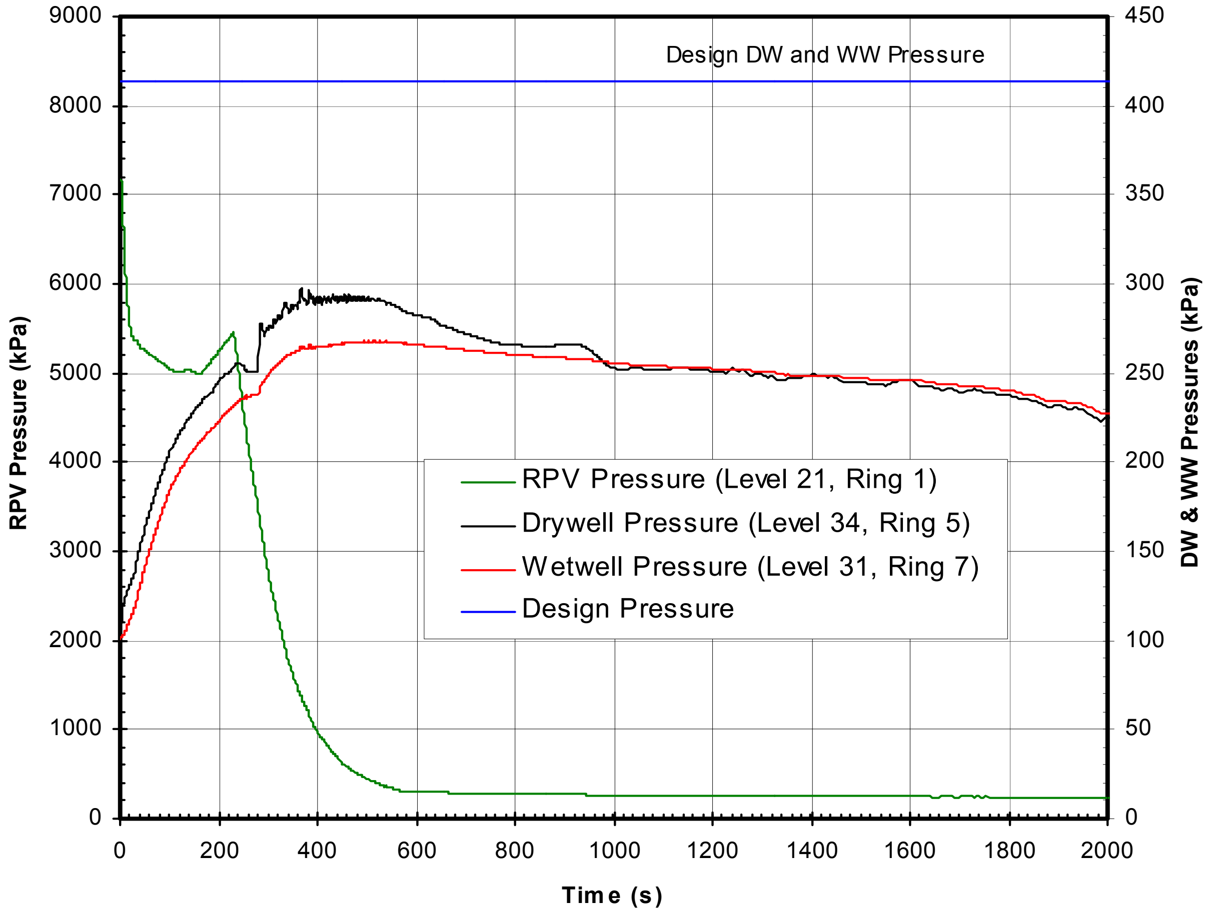

Figure 6.2-11a3. GDCS Line Break (Nominal Case) – Containment Pressures (2000 s)

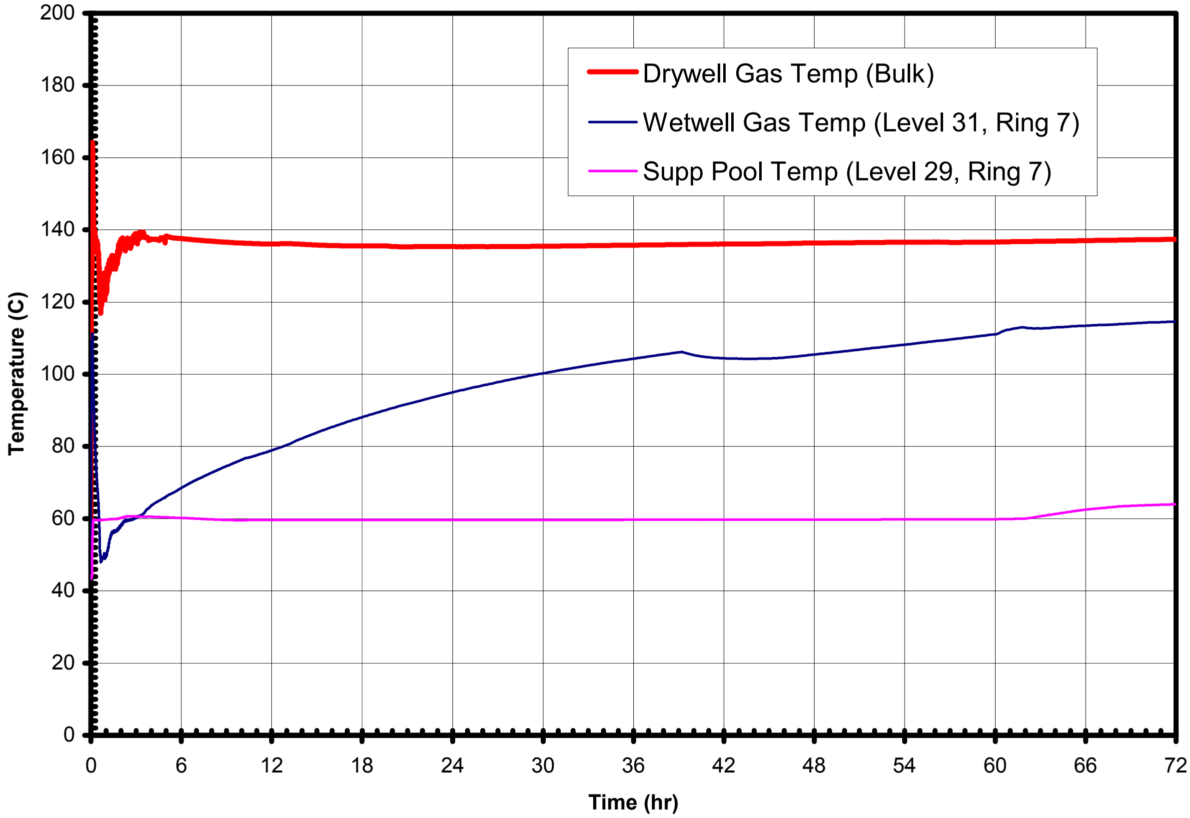

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Figure 6.2-11b1. GDCS Line Break (Nominal Case) – Containment Temperatures (72 hrs)

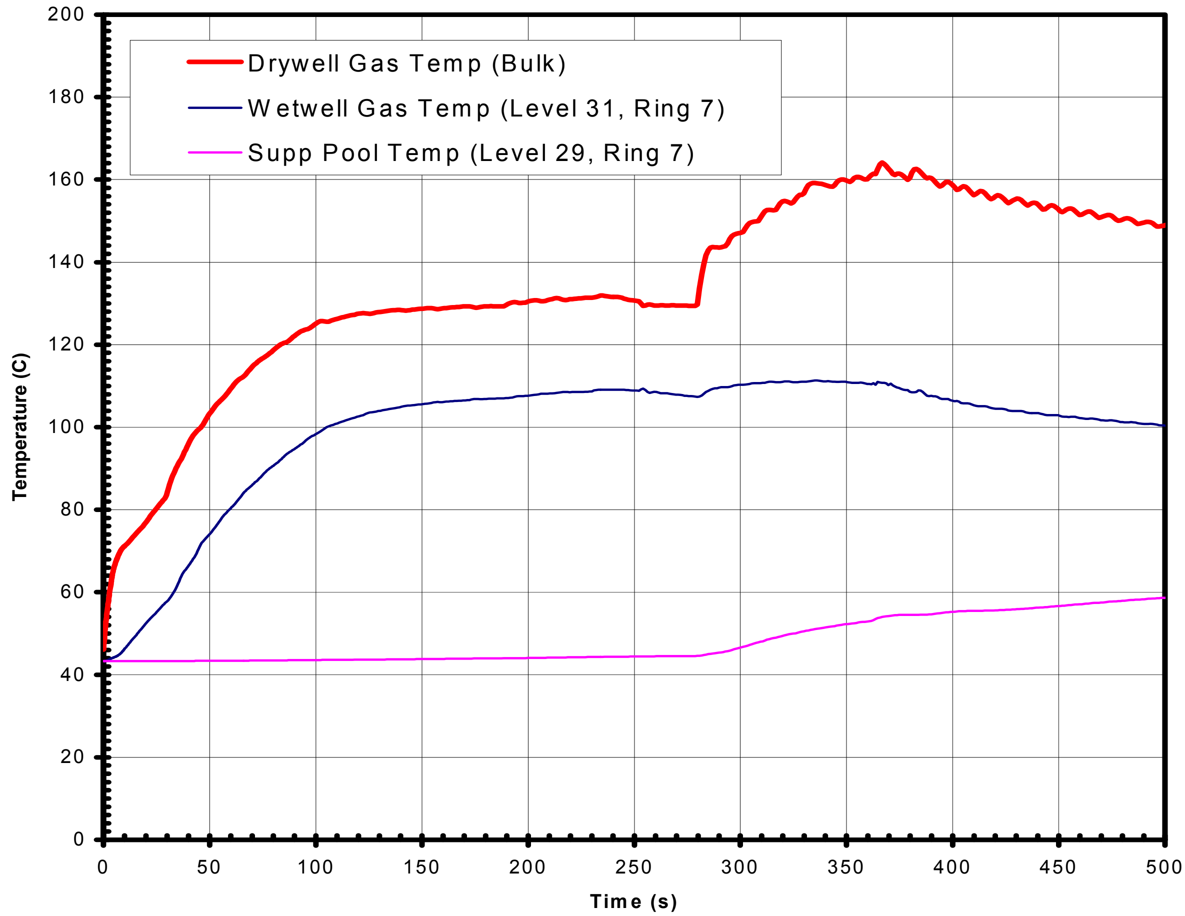

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Figure 6.2-11b2. GDCS Line Break (Nominal Case) – Containment Temperatures (500 s)

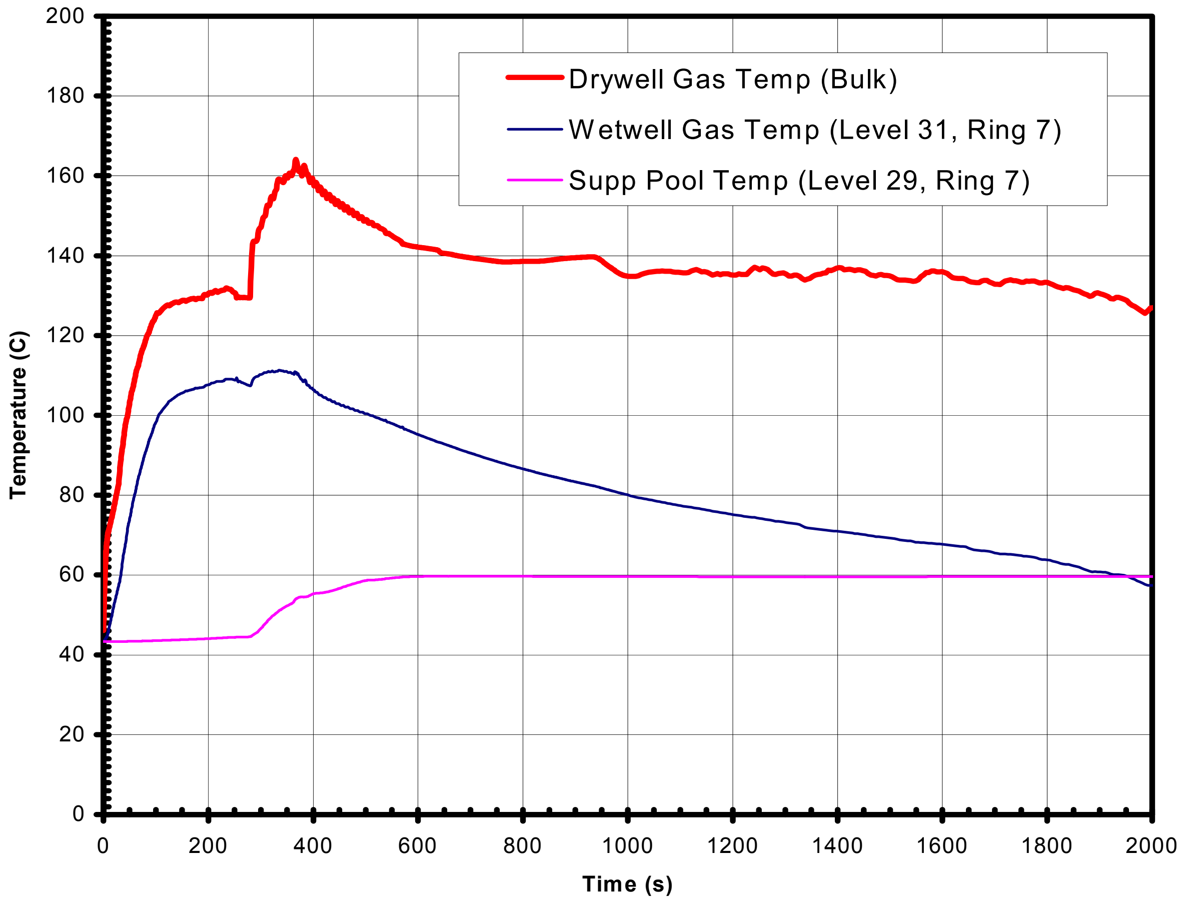

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Figure 6.2-11b3. GDCS Line Break (Nominal Case) – Containment Temperatures (2000 s)

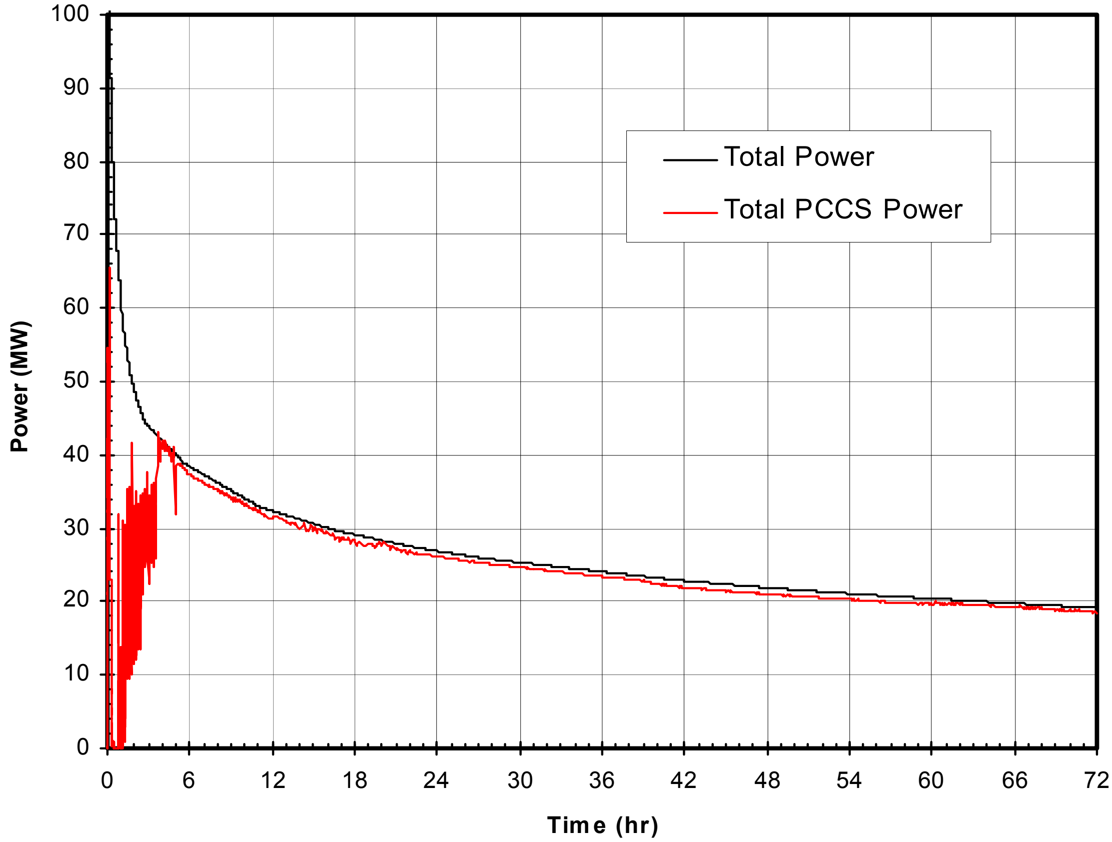

Figure 6.2-11c1. GDCS Line Break (Nominal Case) – PCCS Heat Removal versus Decay Heat (72 hrs)

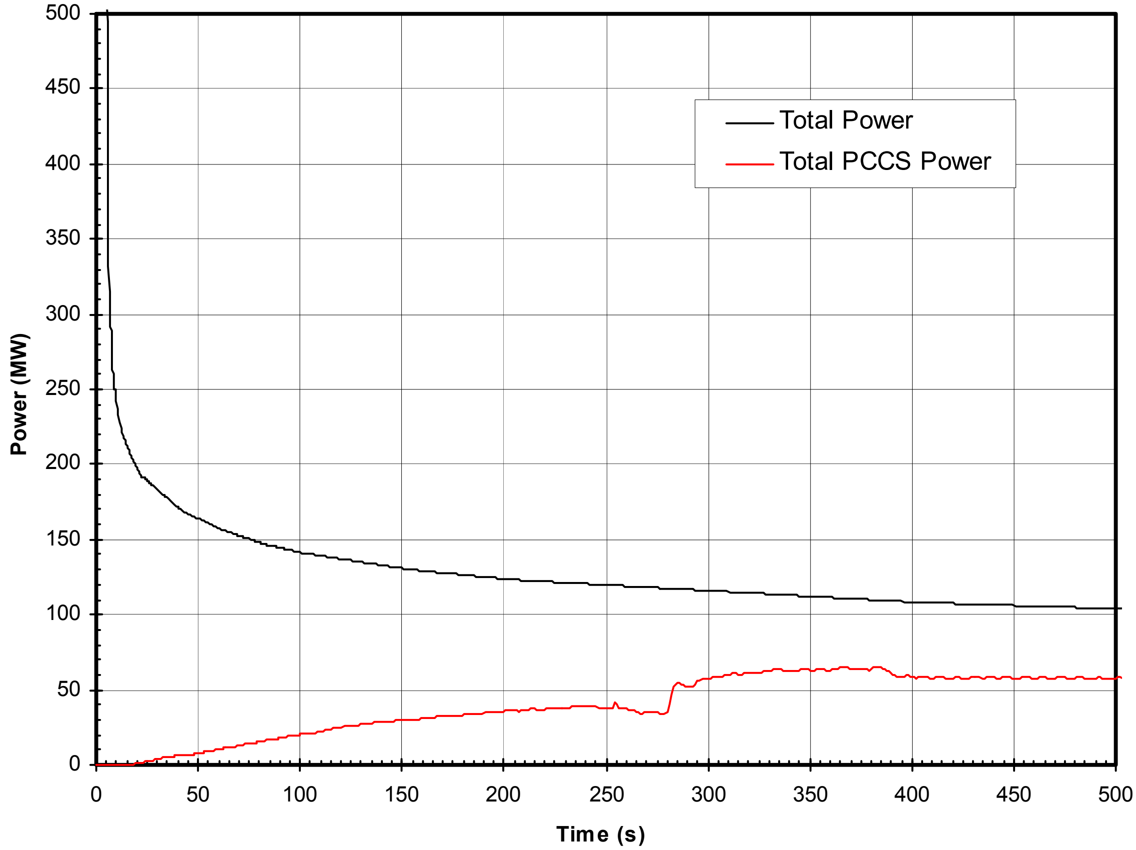

Figure 6.2-11c2. GDCS Line Break (Nominal Case) – PCCS Heat Removal versus Decay Heat (500 s)

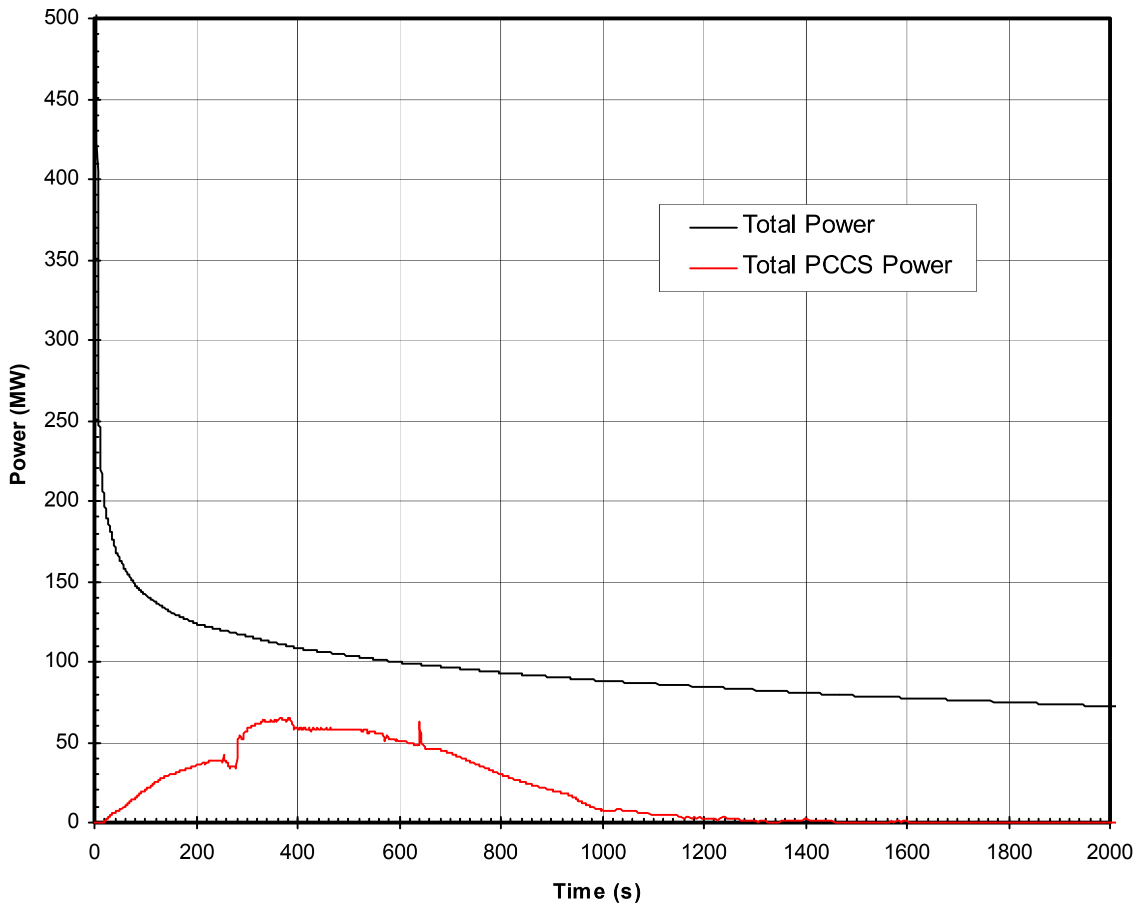

Figure 6.2-11c3. GDCS Line Break (Nominal Case) – PCCS Heat Removal versus Decay Heat (2000 s)

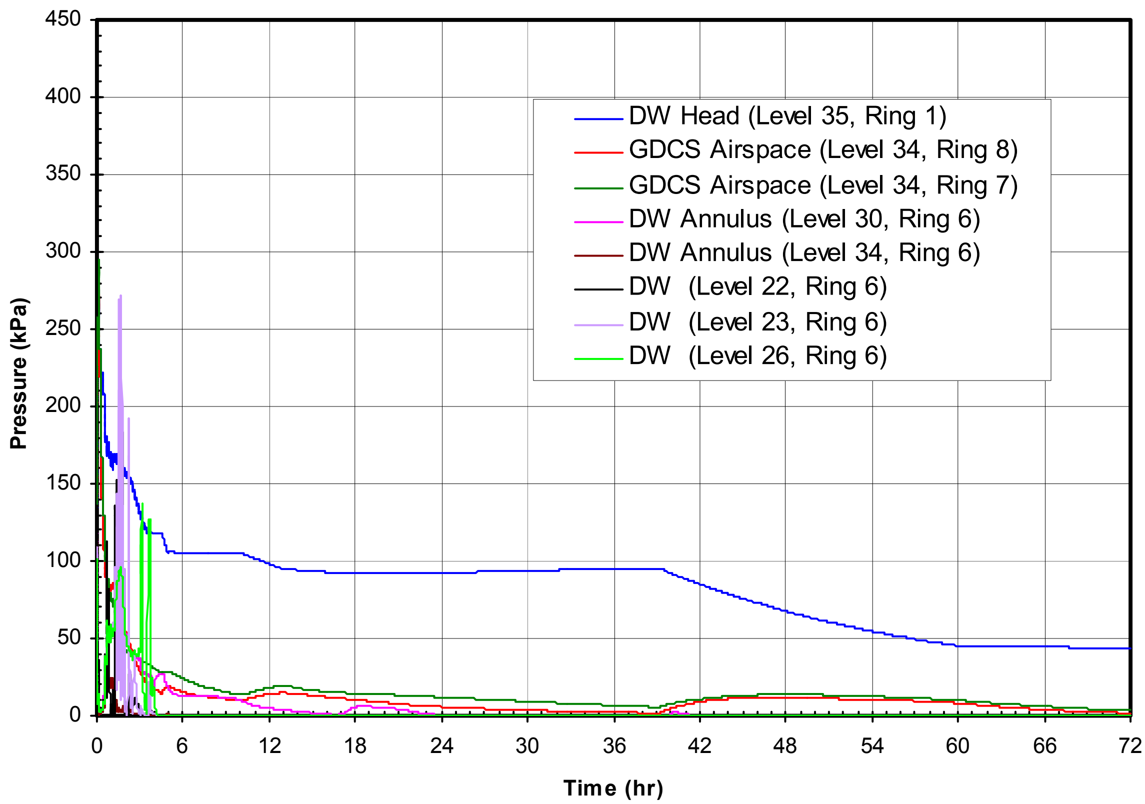

Figure 6.2-11d1. GDCS Line Break (Nominal Case) – Drywell and GDCS Noncondensable Gas Pressures (72 hrs)

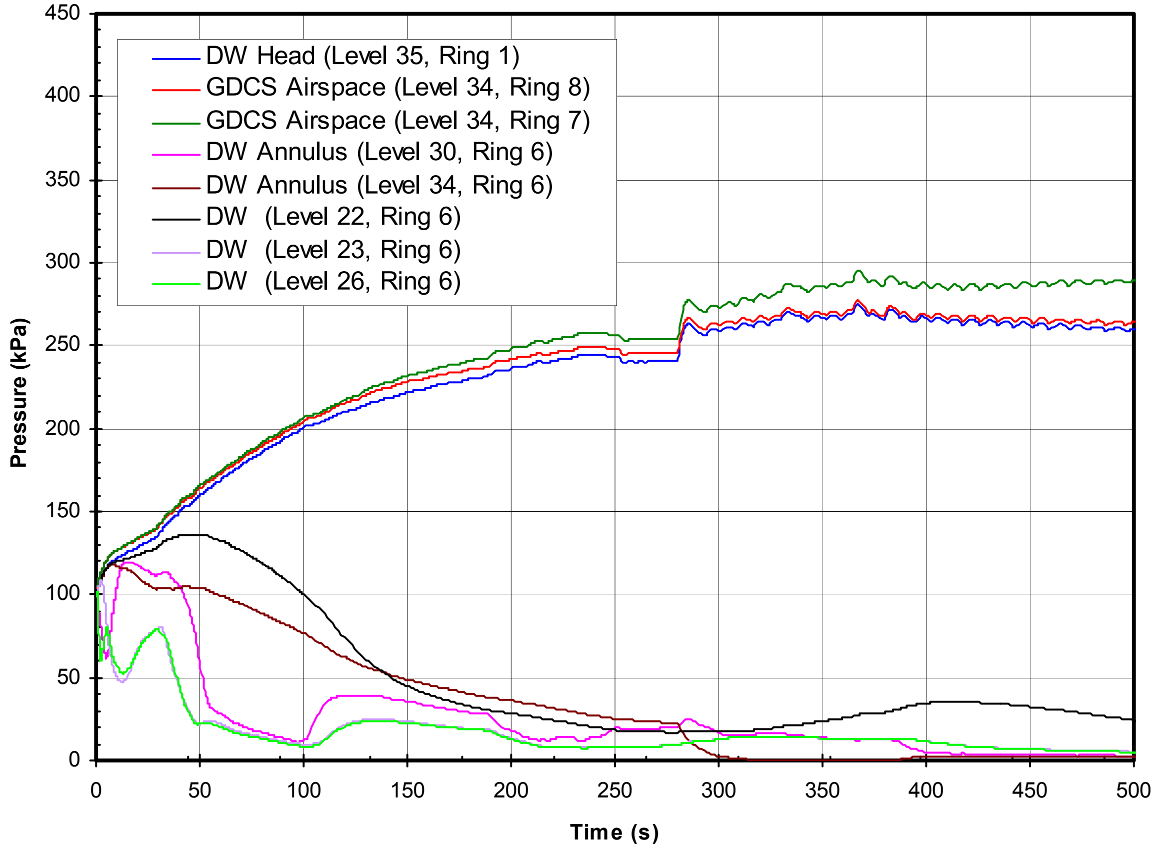

Figure 6.2-11d2. GDCS Line Break (Nominal Case) – Drywell and GDCS Noncondensable Gas Pressures (500 s)

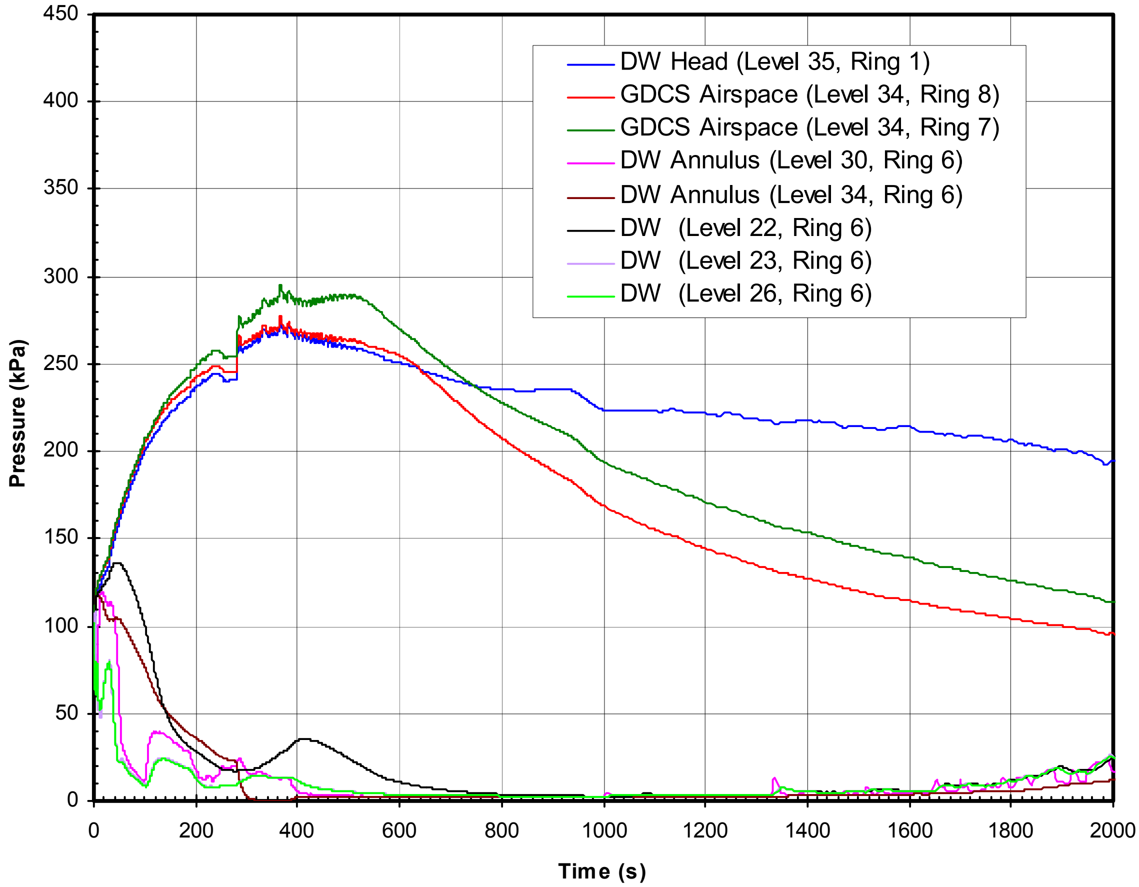

Figure 6.2-11d3. GDCS Line Break (Nominal Case) – Drywell and GDCS Noncondensable Gas Pressures (2000 s)

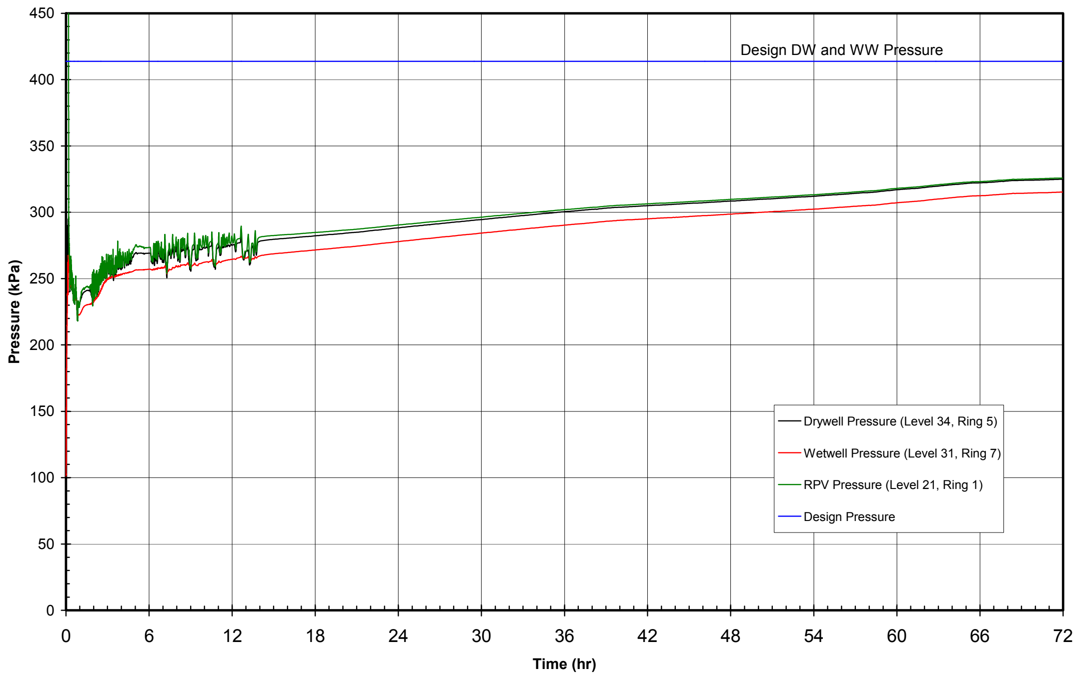

Figure 6.2-12a1. Bottom Drain Line Break (Nominal case) - Containment Pressures (72 hrs)

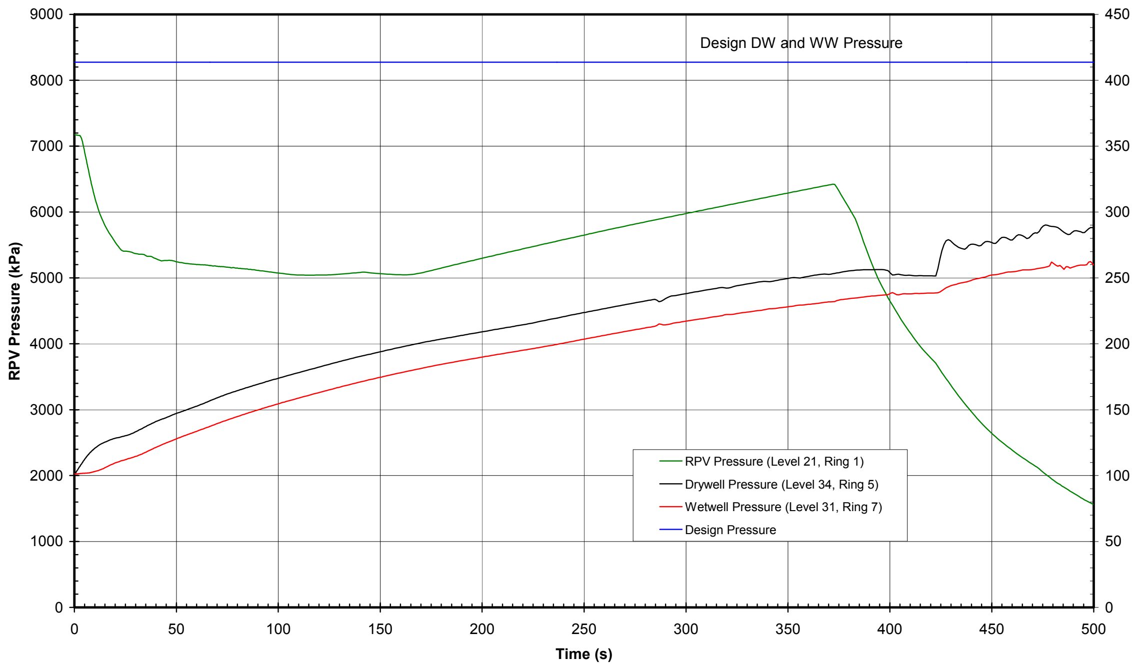

Figure 6.2-12a2. Bottom Drain Line Break (Nominal Case) - Containment Pressures (500s)

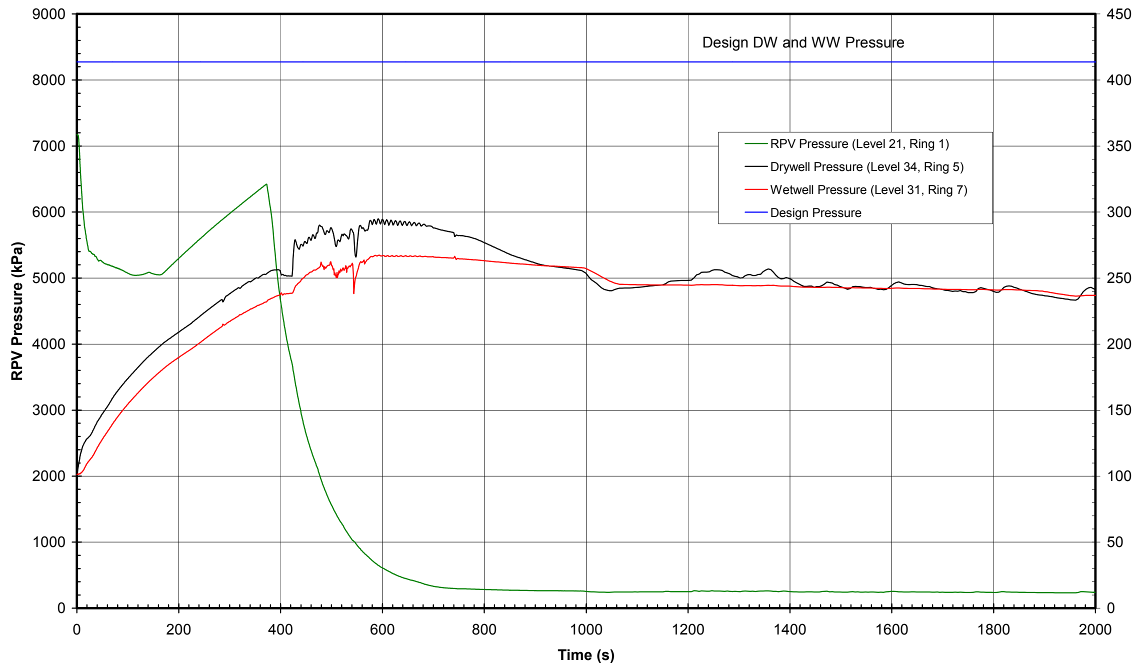

Figure 6.2-12a3. Bottom Drain Line Break (Nominal Case) - Containment Pressures (2000 s)

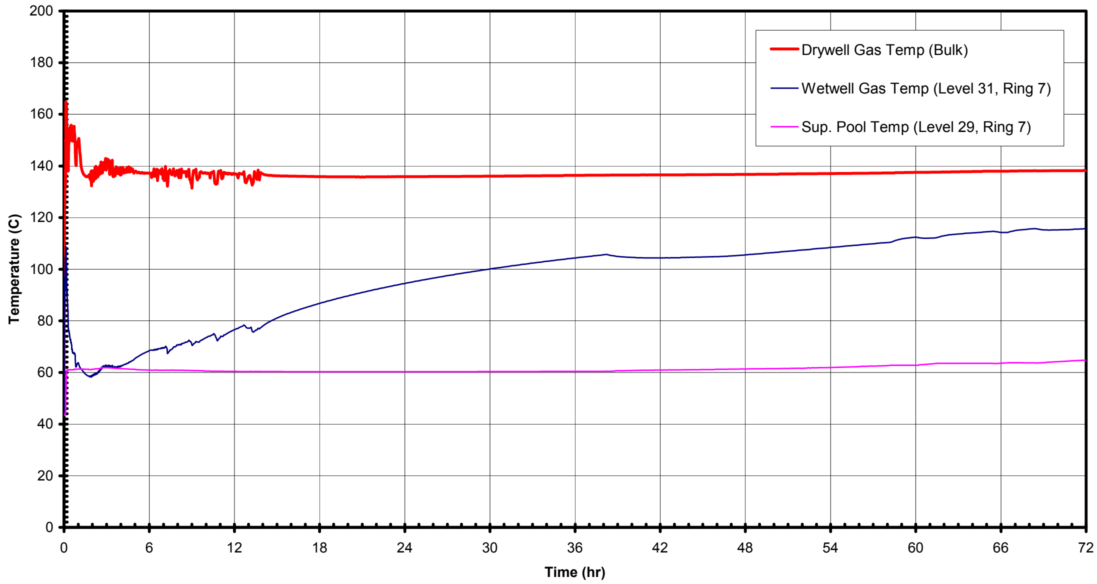

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Figure 6.2-12b1. Bottom Drain Line Break (Nominal Case) - Containment Temperatures (72 hrs)

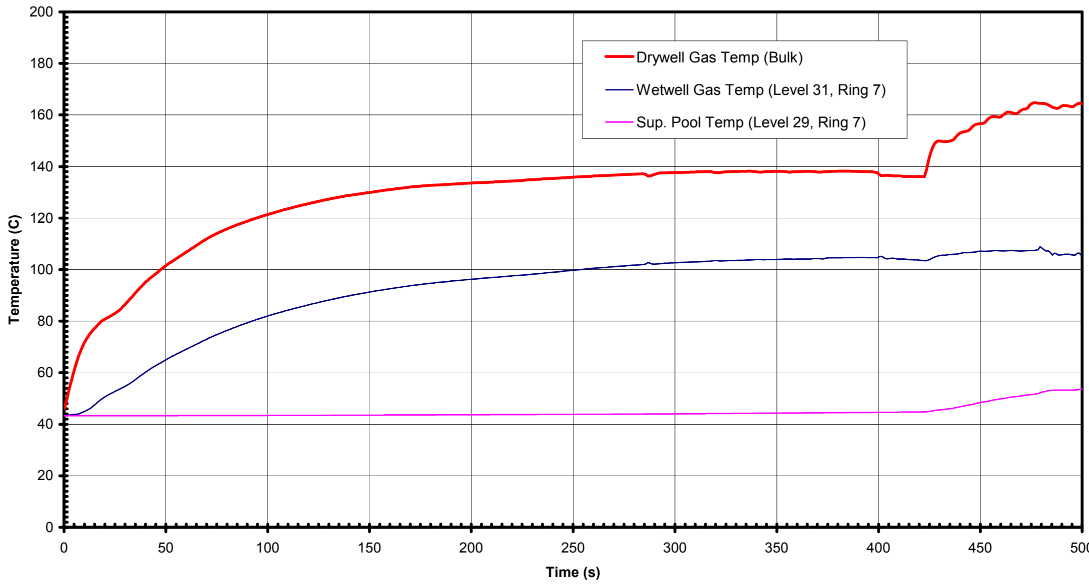

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Figure 6.2-12b2. Bottom Drain Line Break (Nominal Case) - Containment Temperatures (500 s)

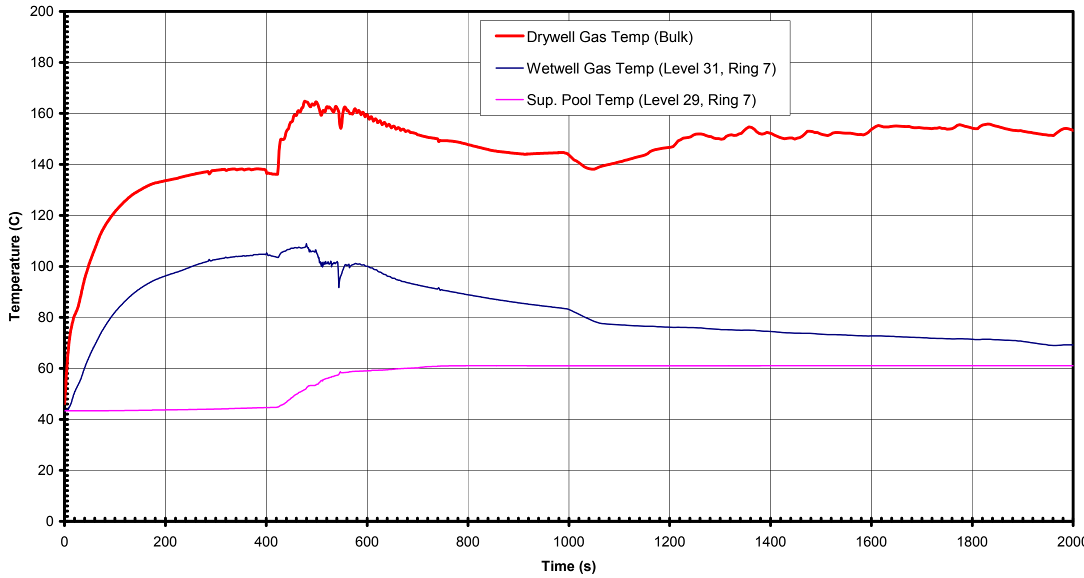

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Figure 6.2-12b3. Bottom Drain Line Break (Nominal Case) - Containment Temperatures (2000 s)

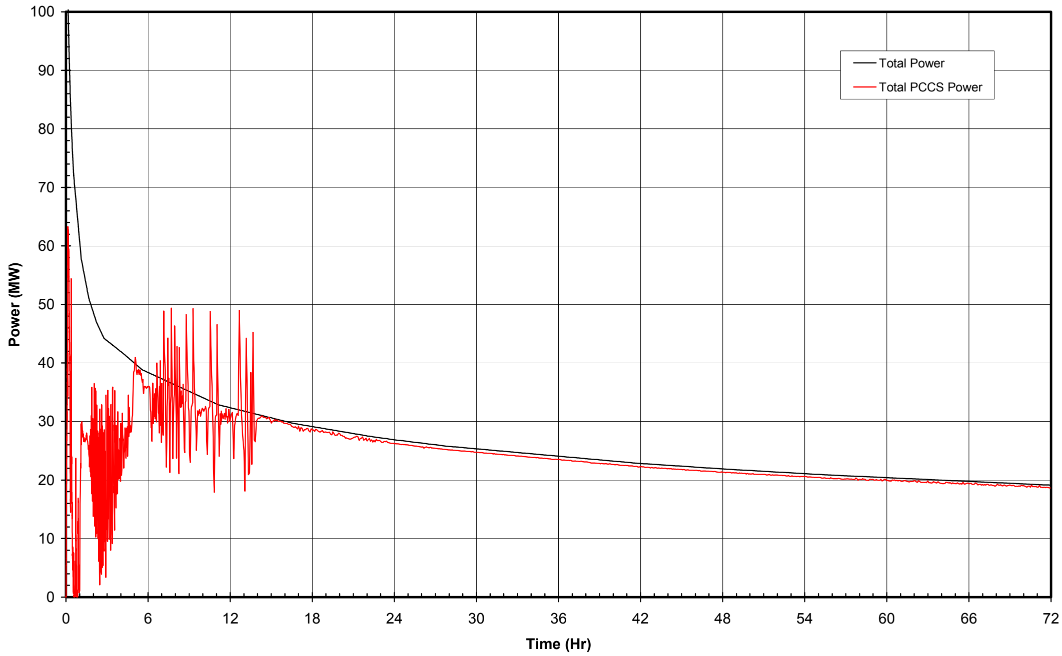


Figure 6.2-12c1. Bottom Drain Line Break (Nominal Case) - PCCS Heat Removal versus Decay Heat (72 hrs)

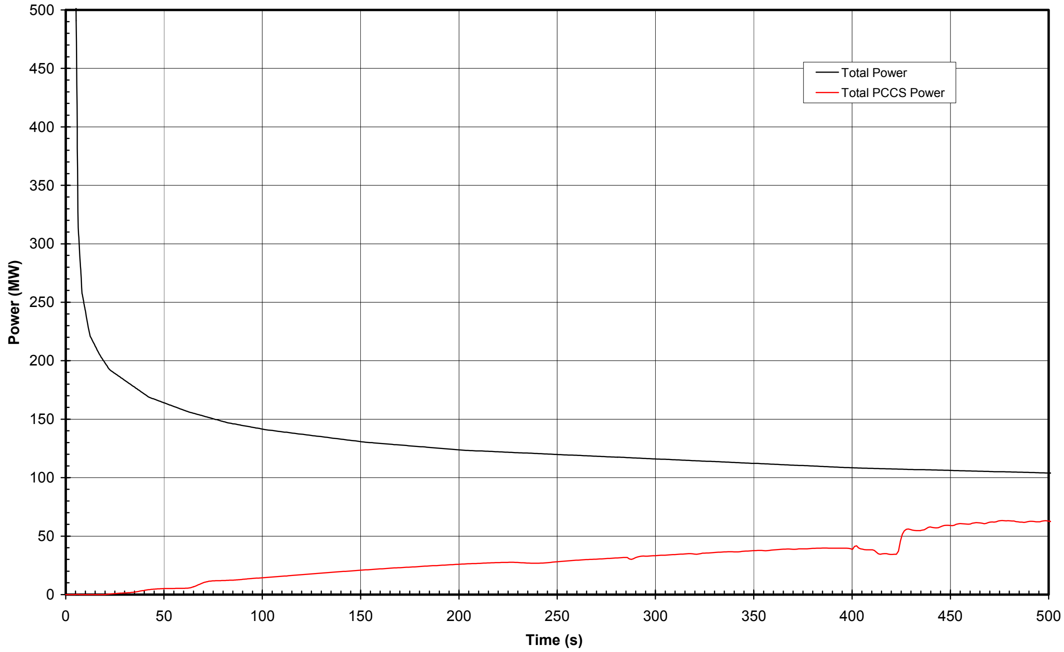


Figure 6.2-12c2. Bottom Drain Line Break (Nominal Case) - PCCS Heat Removal versus Decay Heat (500 s)

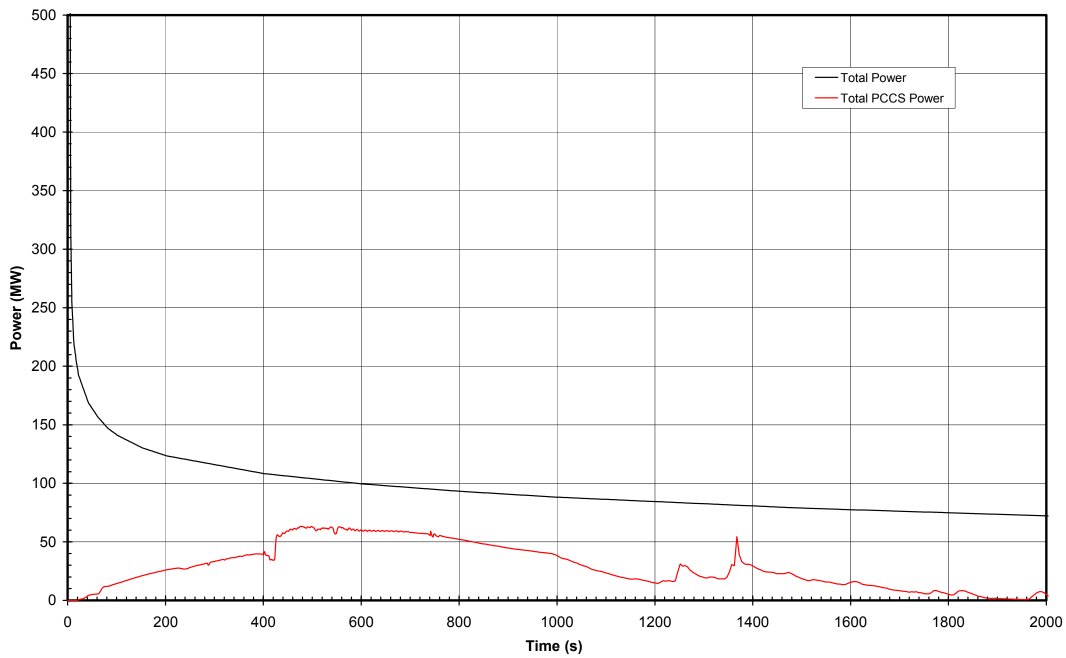


Figure 6.2-12c3. Bottom Drain Line Break (Nominal Case) - PCCS Heat Removal versus Decay Heat (2000 s)

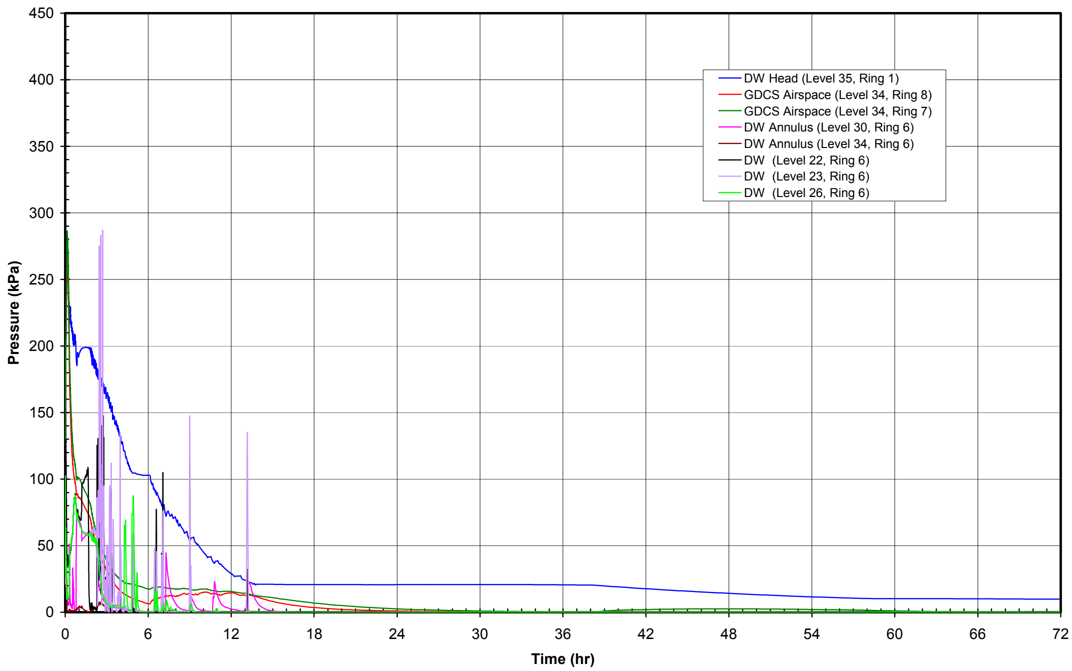


Figure 6.2-12d1. Bottom Drain Line Break (Nominal Case) - Drywell and GDCS Noncondensable Pressures (72 hrs)

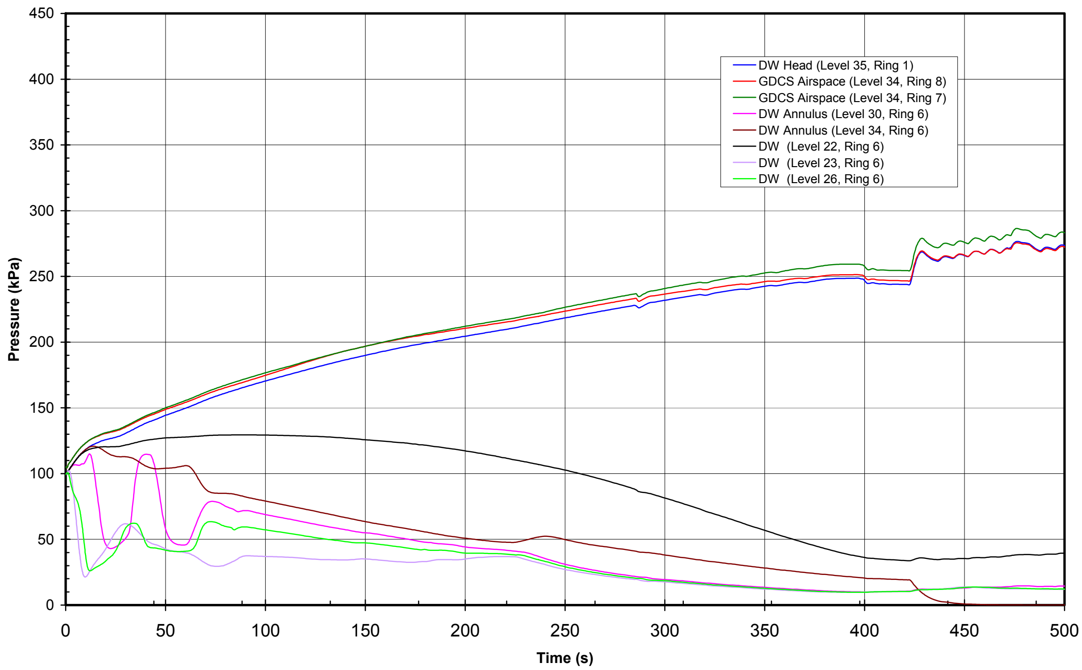


Figure 6.2-12d2. Bottom Drain Line Break (Nominal Case) - Drywell and GDCS Noncondensable Pressures (500 s)

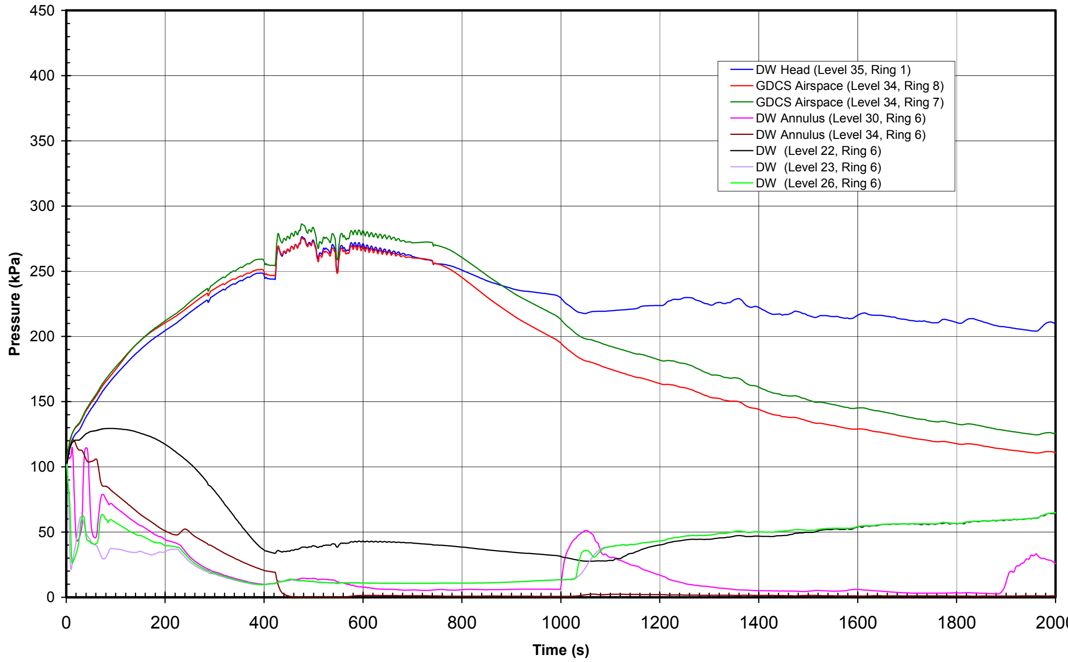


Figure 6.2-12d3. Bottom Drain Line Break (Nominal Case) - Drywell and GDCS Noncondensable Pressures (2000 s)

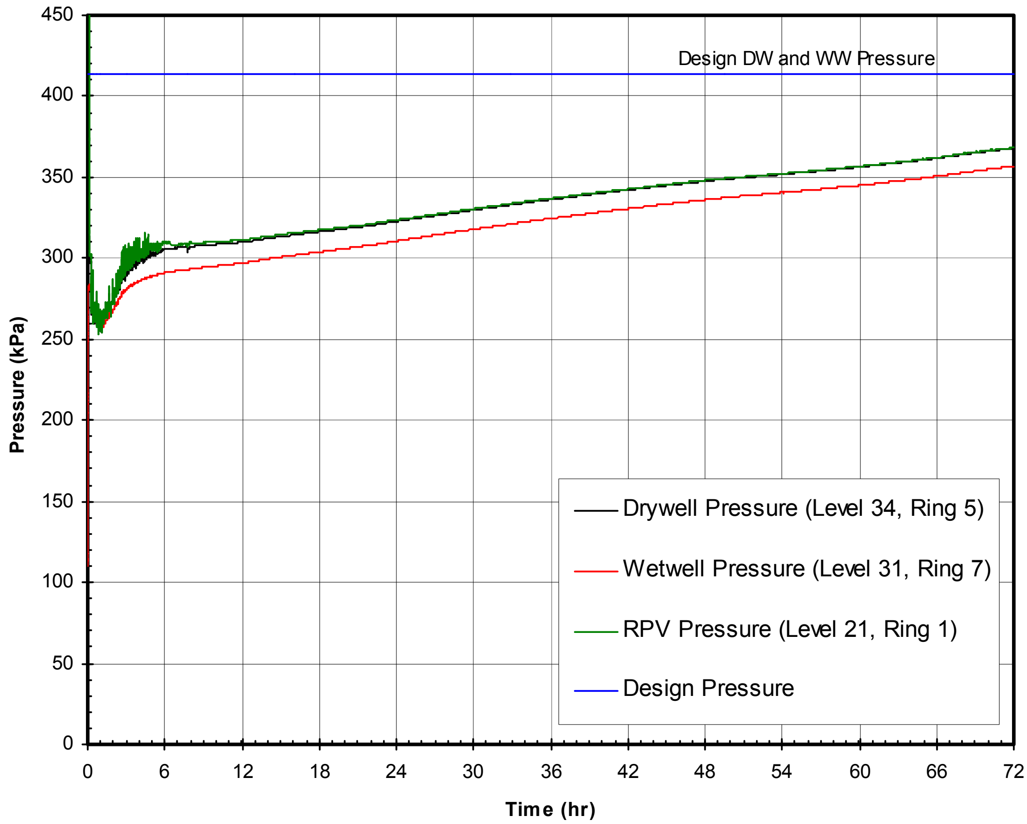


Figure 6.2-13a1. Feedwater Line Break, 1 DPV Failure (Bounding Case) – Containment Pressures (72 hrs)

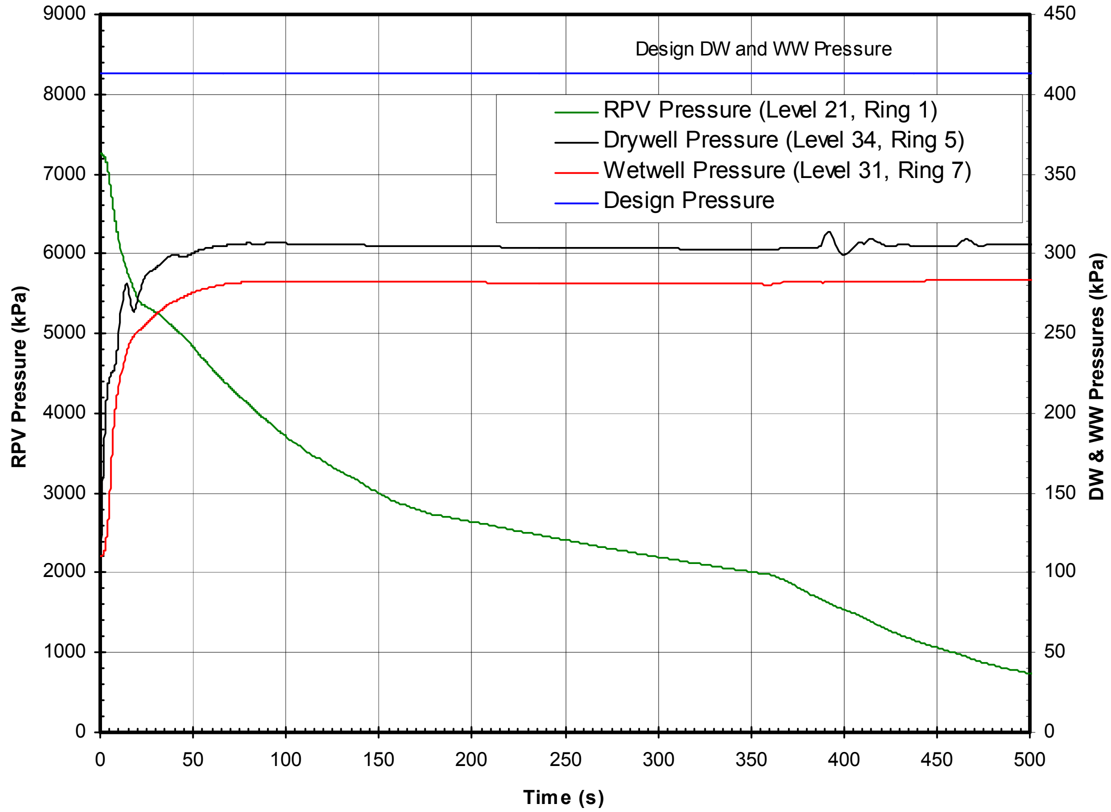


Figure 6.2-13a2. Feedwater Line Break, 1 DPV Failure (Bounding Case) – Containment Pressures (500 s)

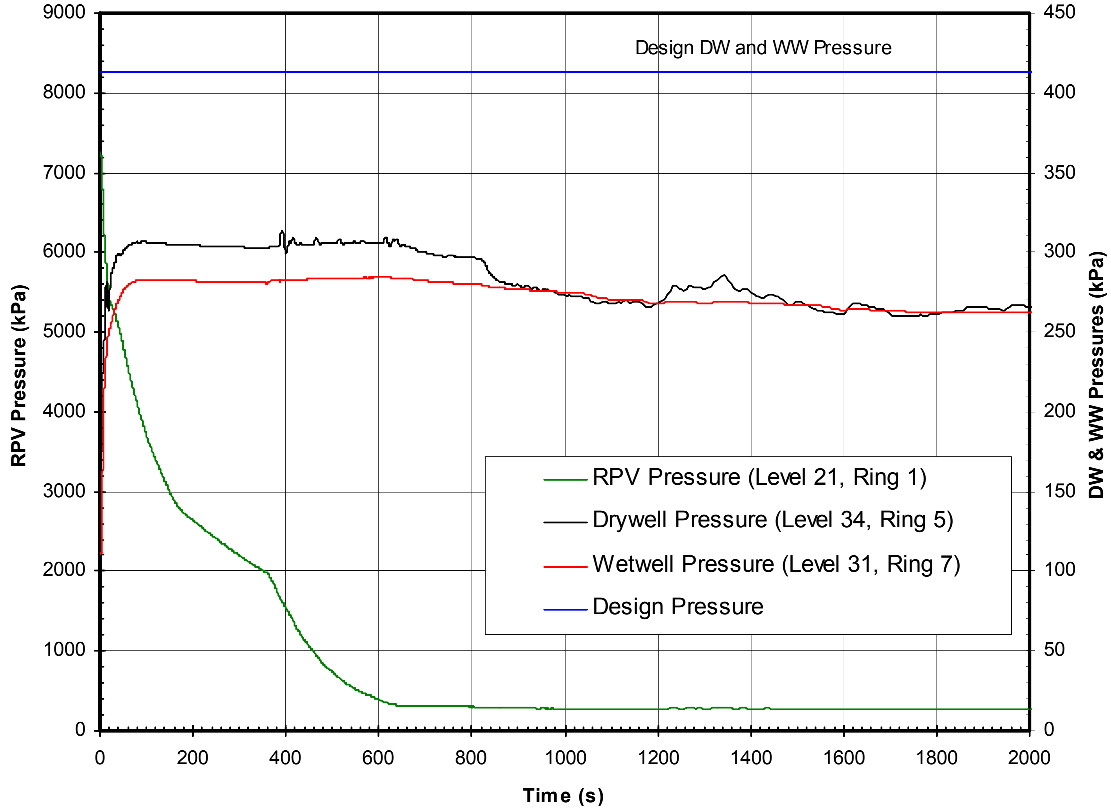
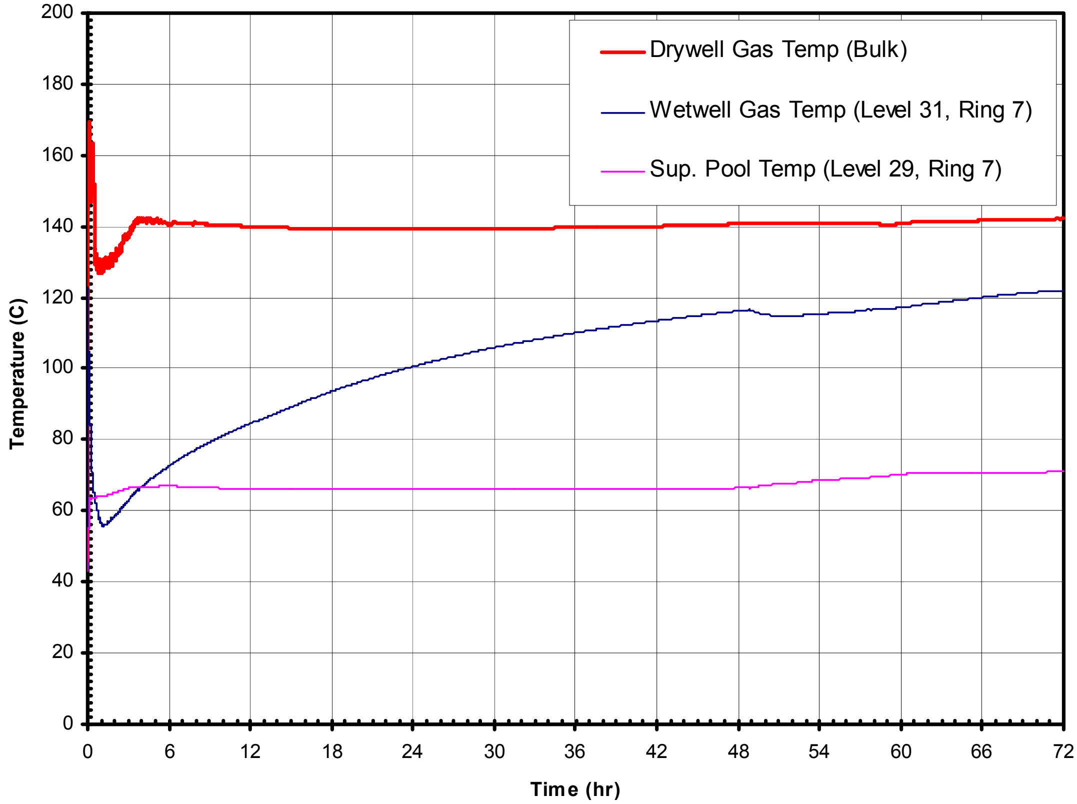
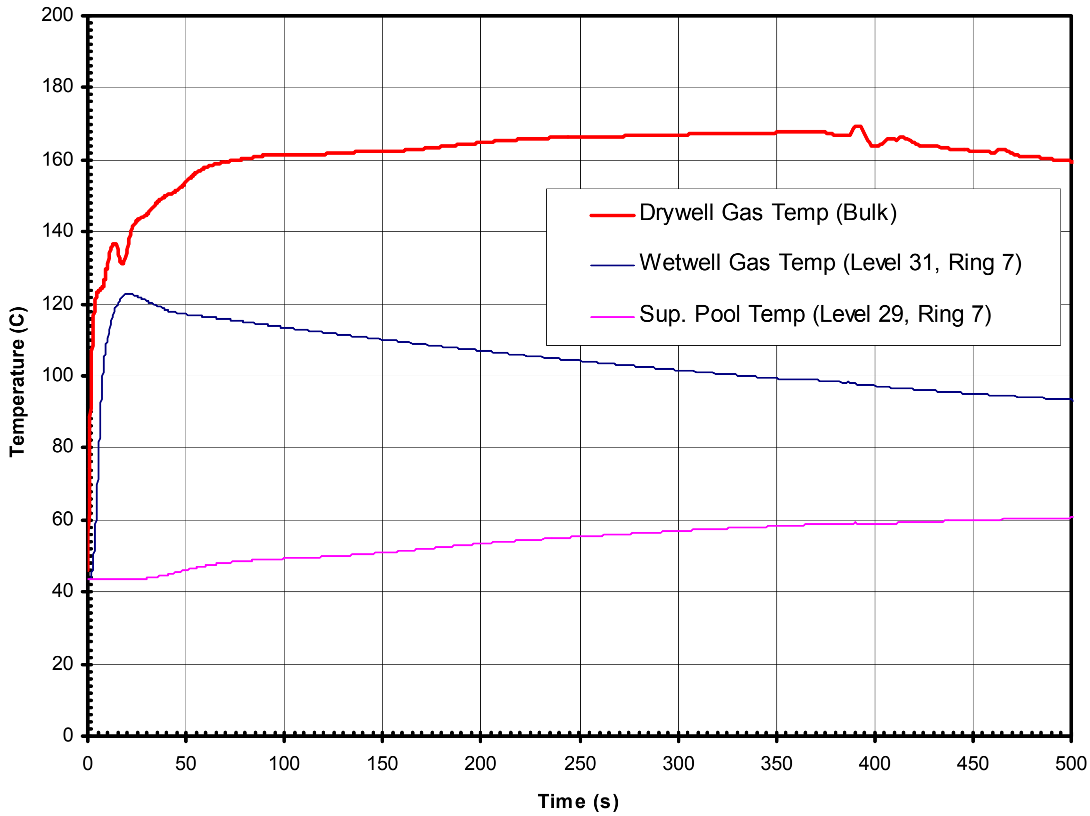


Figure 6.2-13a3. Feedwater Line Break, 1 DPV Failure (Bounding Case) – Containment Pressures (2000 s)



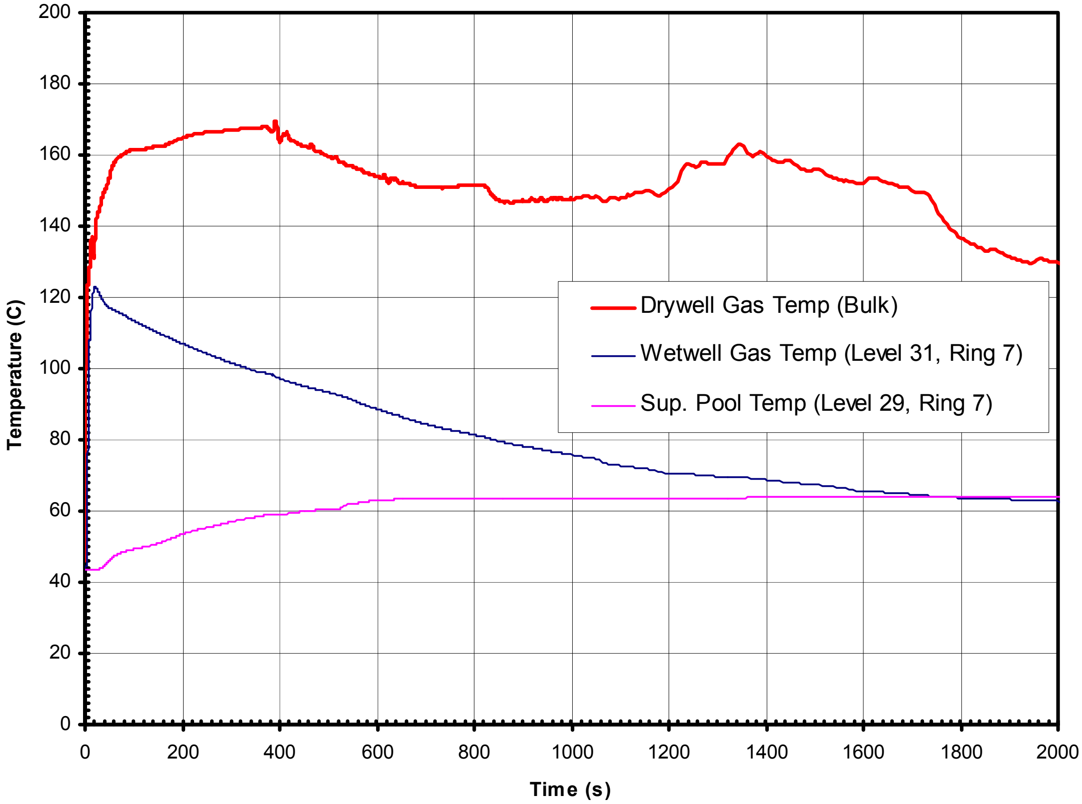
LEGEND: Sup. = Suppression

Figure 6.2-13b1. Feedwater Line Break, 1 DPV Failure (Bounding Case) – Containment Temperatures (72 hrs)



LEGEND: Sup. = Suppression

Figure 6.2-13b2. Feedwater Line Break, 1 DPV Failure (Bounding Case) – Containment Temperatures (500 s)



LEGEND: Sup. = Suppression

Figure 6.2-13b3. Feedwater Line Break, 1 DPV Failure (Bounding Case) – Containment Temperatures (2000 s)

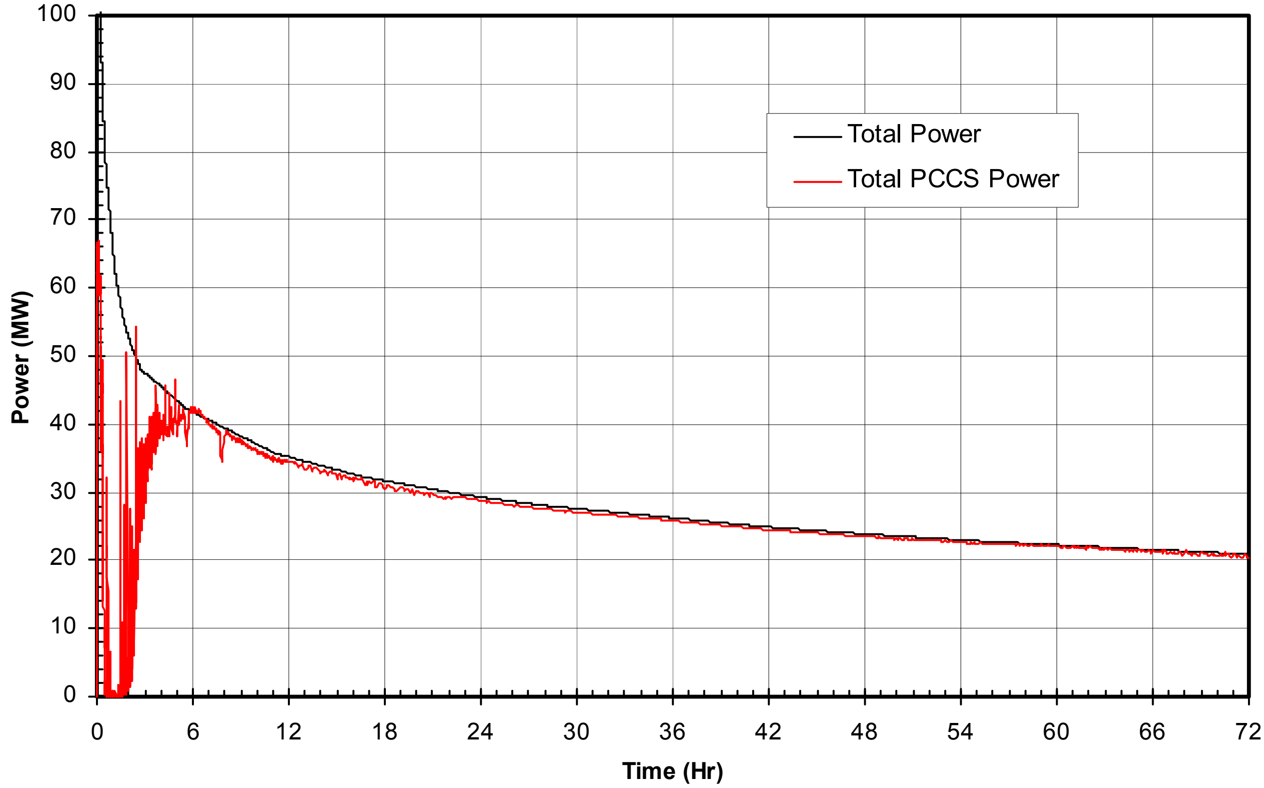


Figure 6.2-13c1. Feedwater Line Break, 1 DPV Failure (Bounding Case) – PCCS Heat Removal versus Decay Heat (72 hrs)

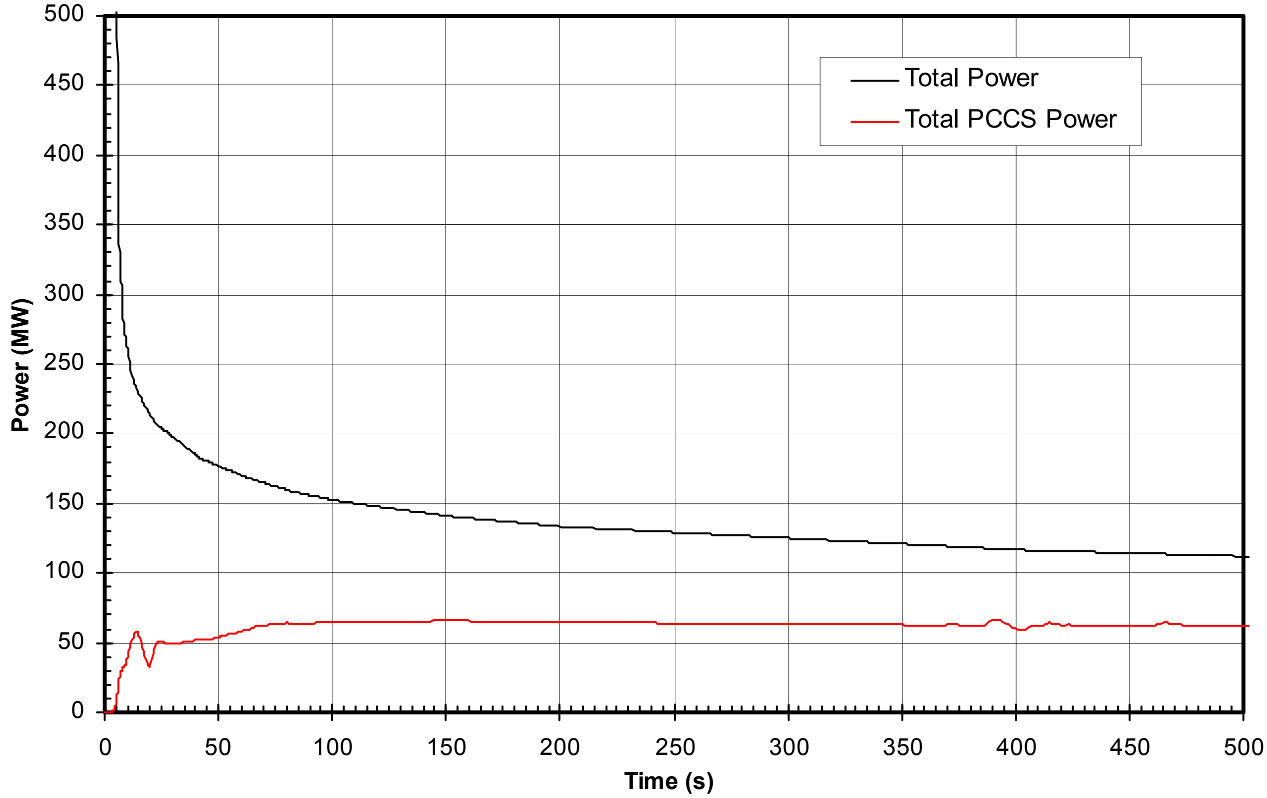


Figure 6.2-13c2. Feedwater Line Break, 1 DPV Failure (Bounding Case) – PCCS Heat Removal versus Decay Heat (500 s)

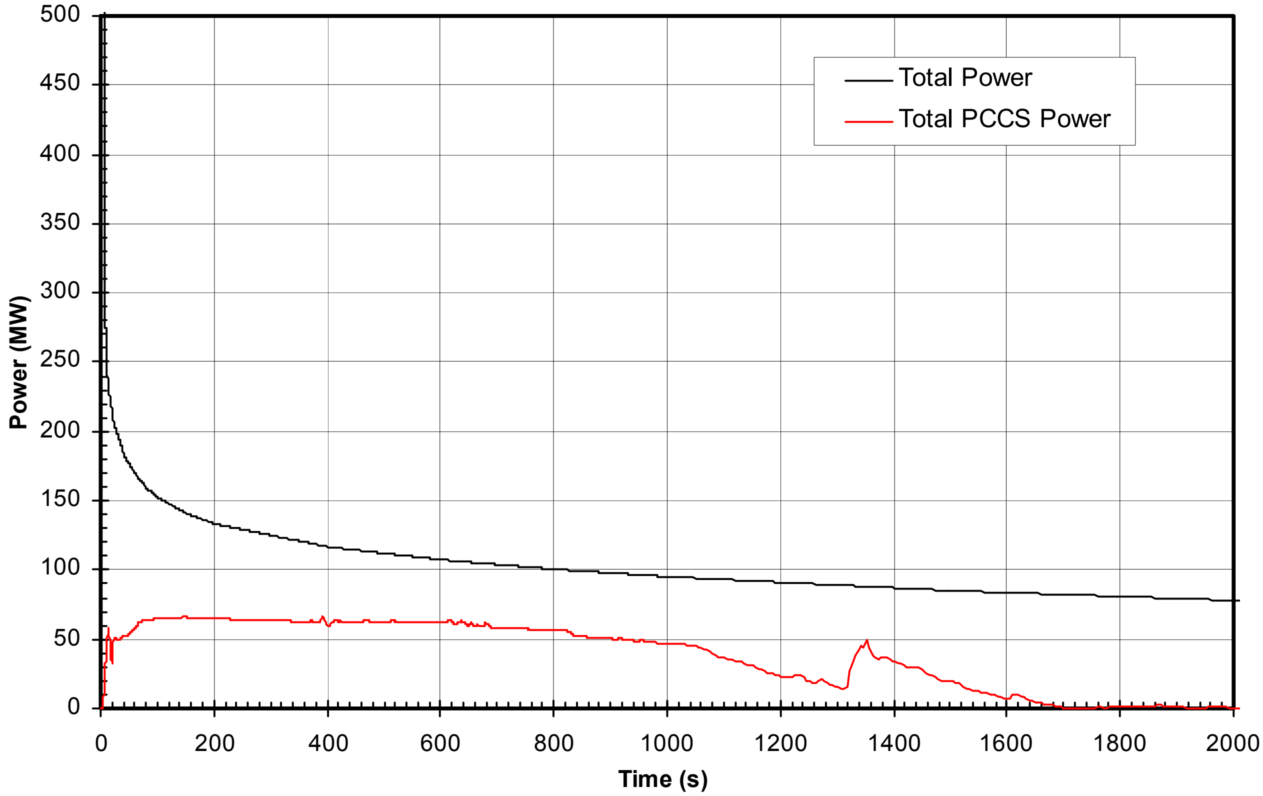


Figure 6.2-13c3. Feedwater Line Break, 1 DPV Failure (Bounding Case) – PCCS Heat Removal versus Decay Heat (2000 s)

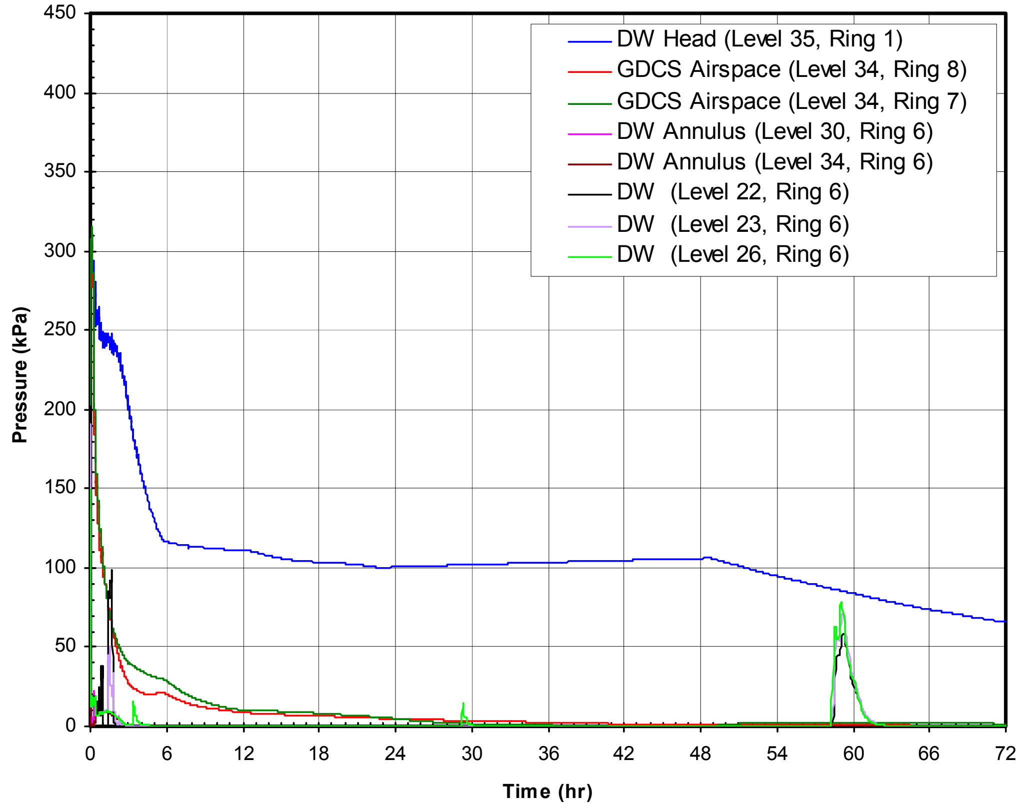


Figure 6.2-13d1. Feedwater Line Break, 1 DPV Failure (Bounding Case) - Drywell and GDCS Noncondensable Pressures (72 hrs)

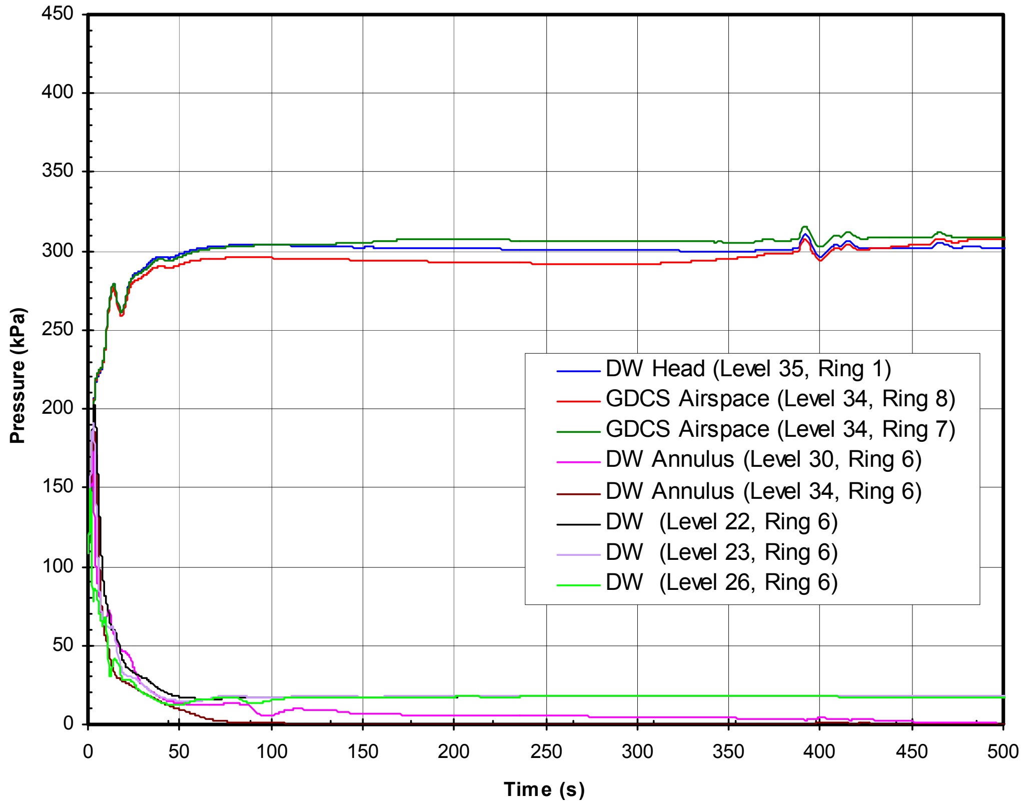


Figure 6.2-13d2. Feedwater Line Break, 1 DPV Failure (Bounding Case) - Drywell and GDCS Noncondensable Pressures (500 s)

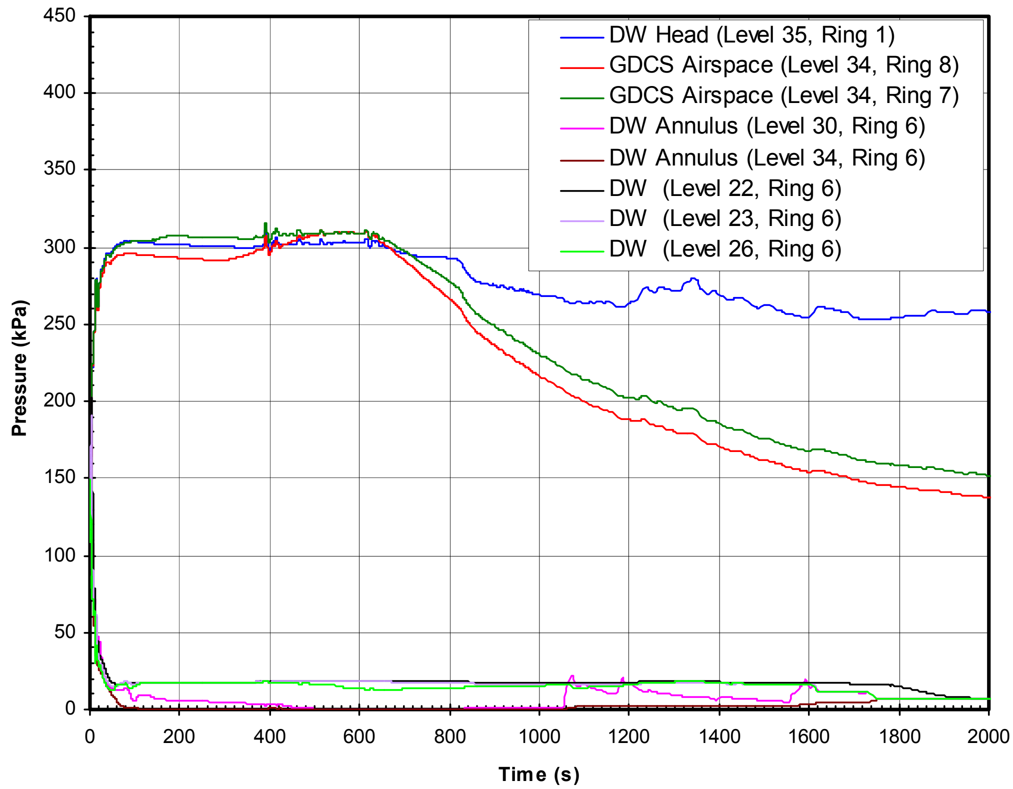


Figure 6.2-13d3. Feedwater Line Break, 1 DPV Failure (Bounding Case) - Drywell and GDCS Noncondensable Pressures (2000 s)

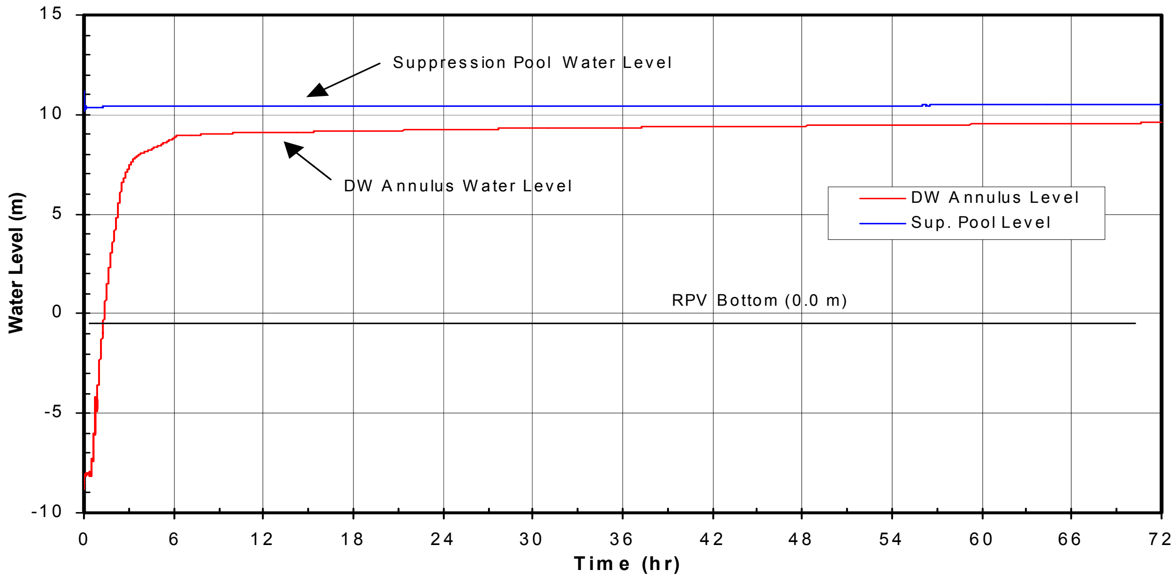


Figure 6.2-13d4. Feedwater Line Break, 1 DPV Failure (Bounding) Drywell Annulus and Suppression Pool Levels (72 hrs)

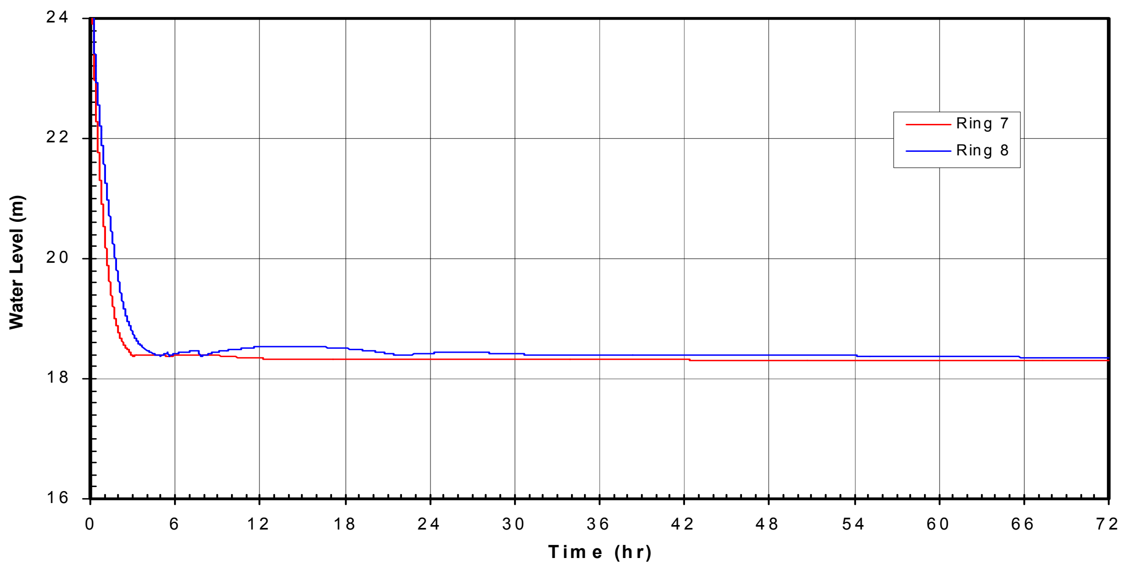


Figure 6.2-13d5. Feedwater Line Break, 1 DPV Failure (Bounding) GDCS Pool Levels (72 hrs)

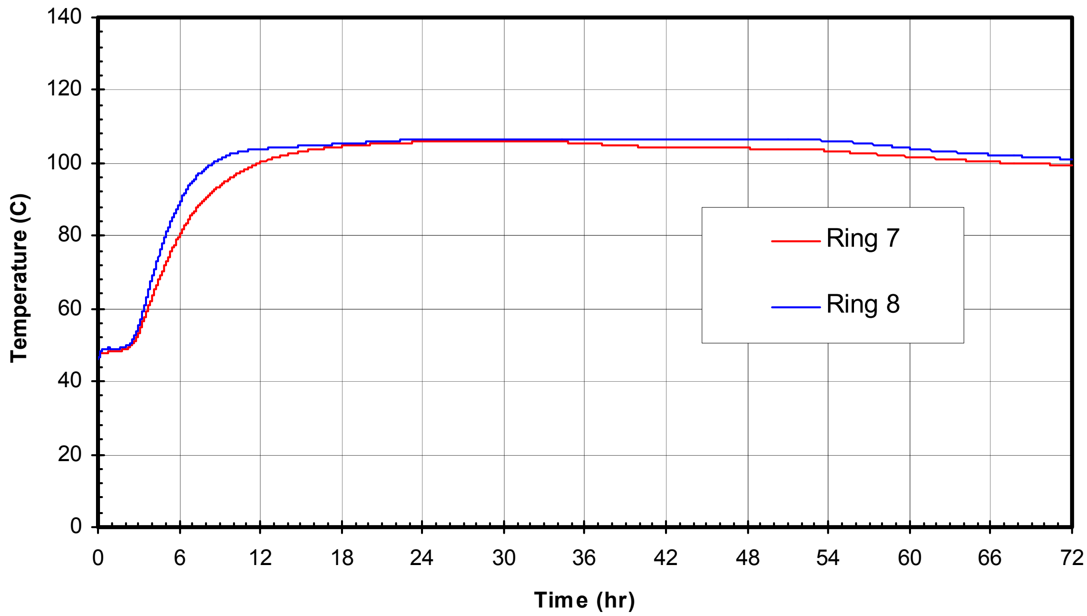


Figure 6.2-13d6. Feedwater Line Break, 1 DPV Failure (Bounding) GDCS Pool Temperature (72 hrs)

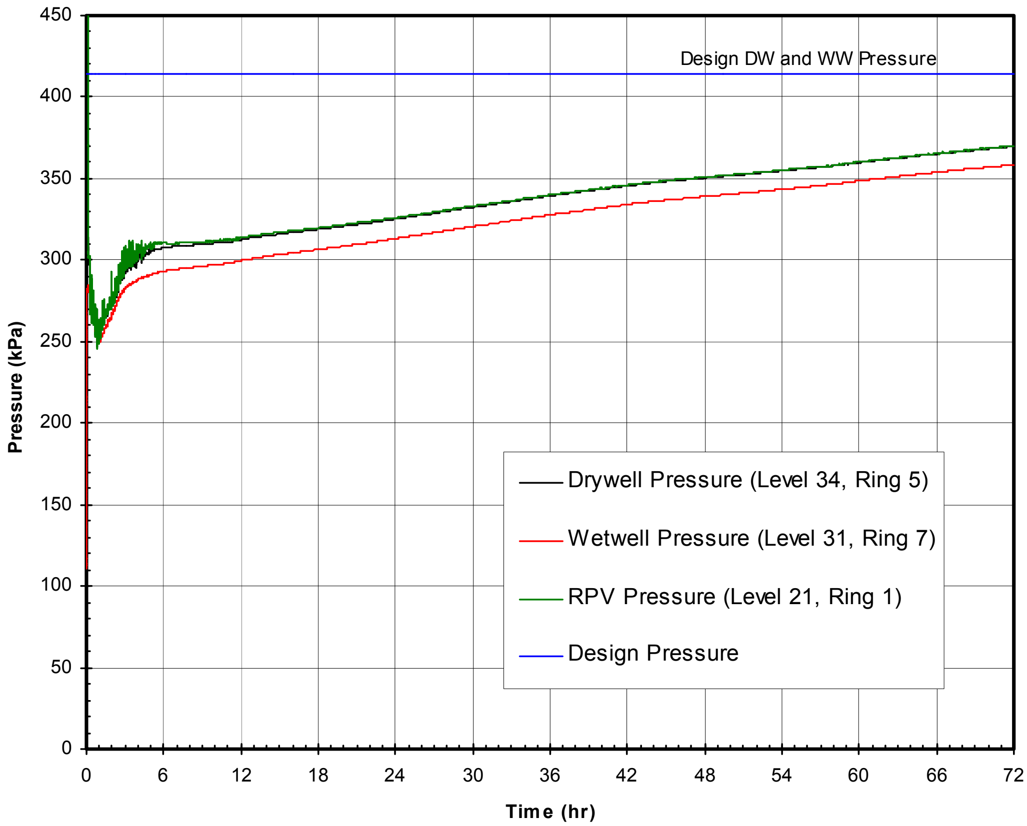


Figure 6.2-13e1. Feed Water Line Break, 1 SRV Failure (Bounding Case) - Containment Pressures (72 hrs)

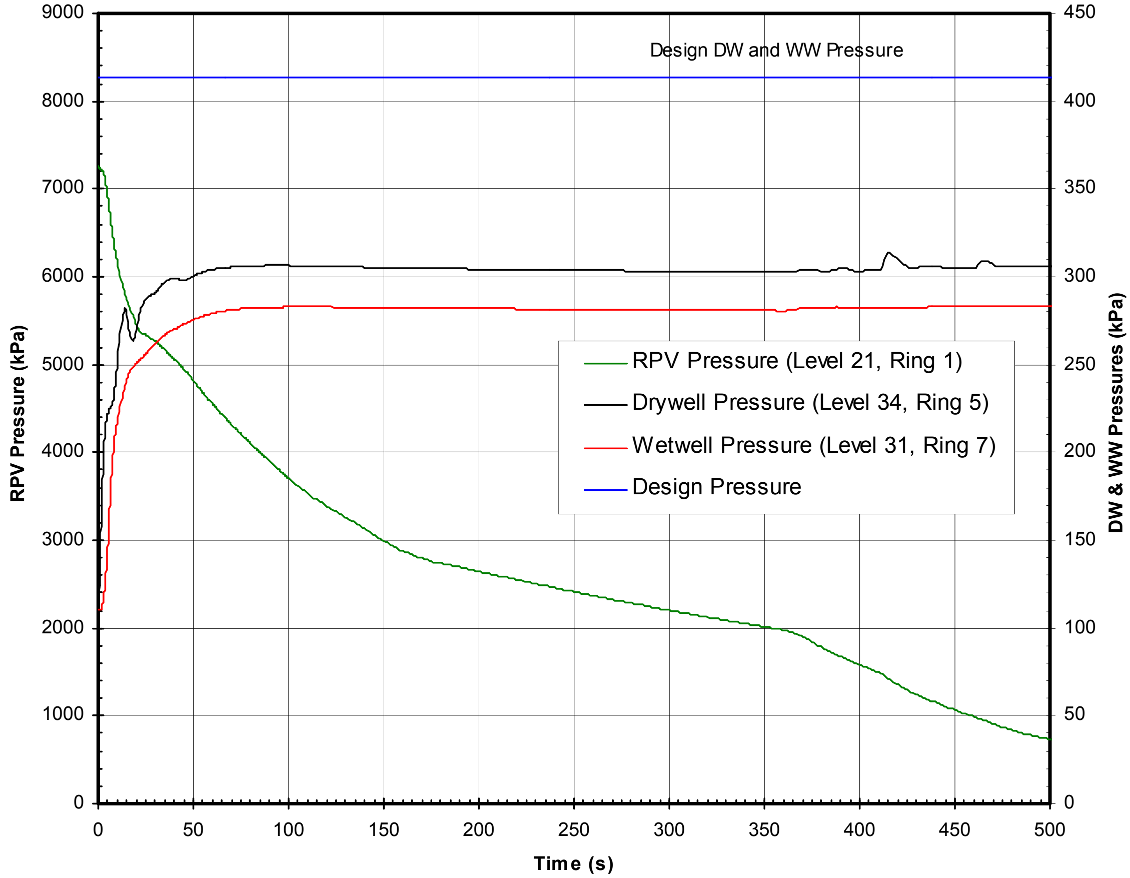


Figure 6.2-13e2. Feed Water Line Break, 1 SRV Failure (Bounding Case) – Containment Pressures (500 s)

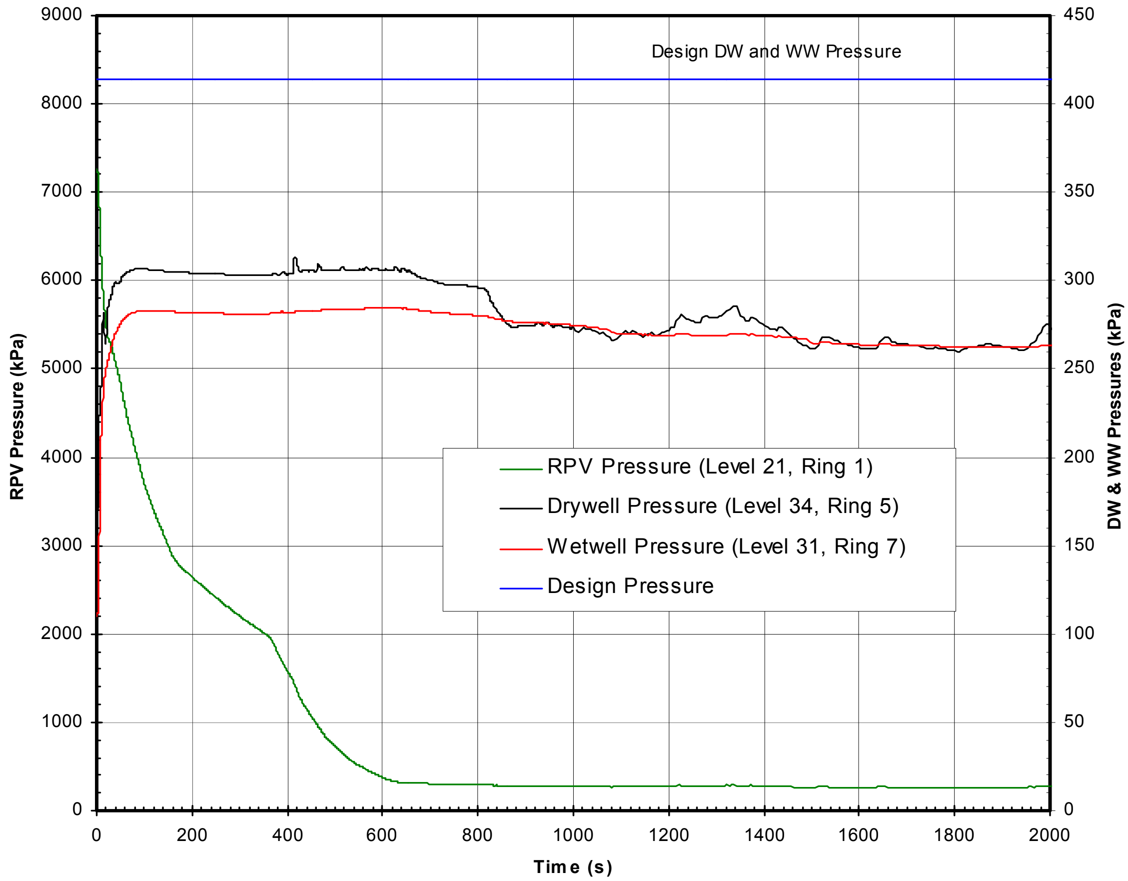


Figure 6.2-13e3. Feed Water Line Break, 1 SRV Failure (Bounding Case) – Containment Pressures (2000 s)

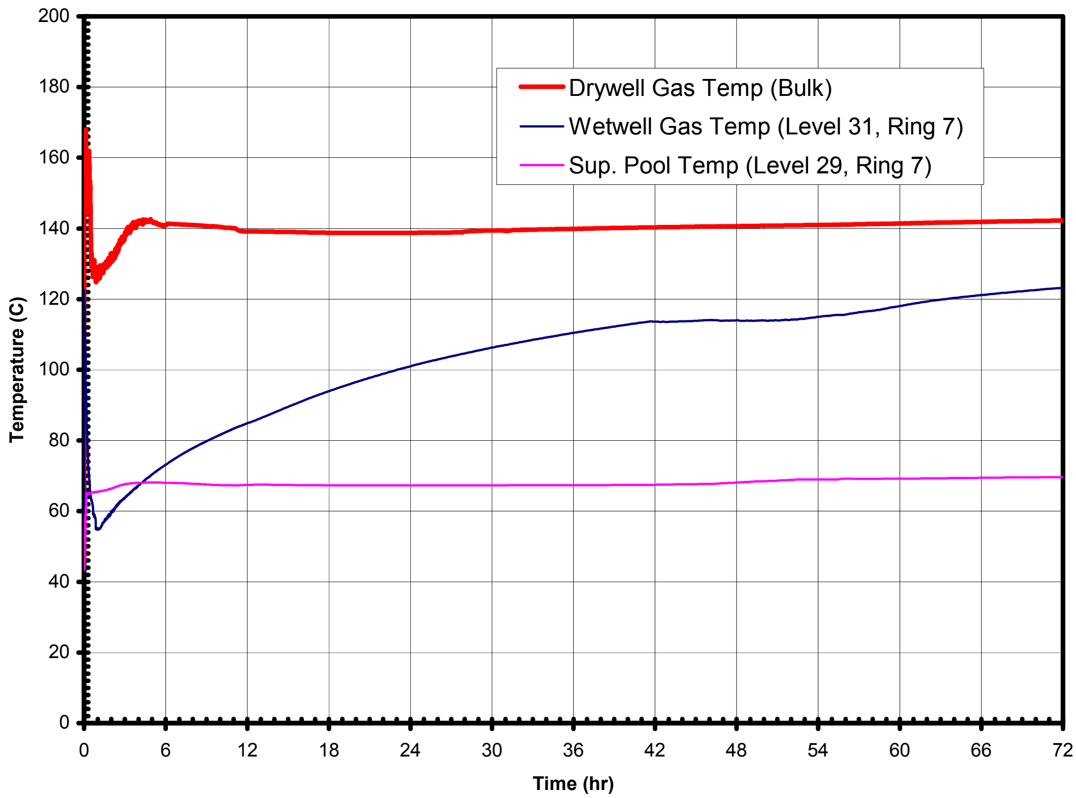


Figure 6.2-13f1. Feed Water Line Break, 1 SRV Failure (Bounding Case) – Containment Temperatures (72 hrs)

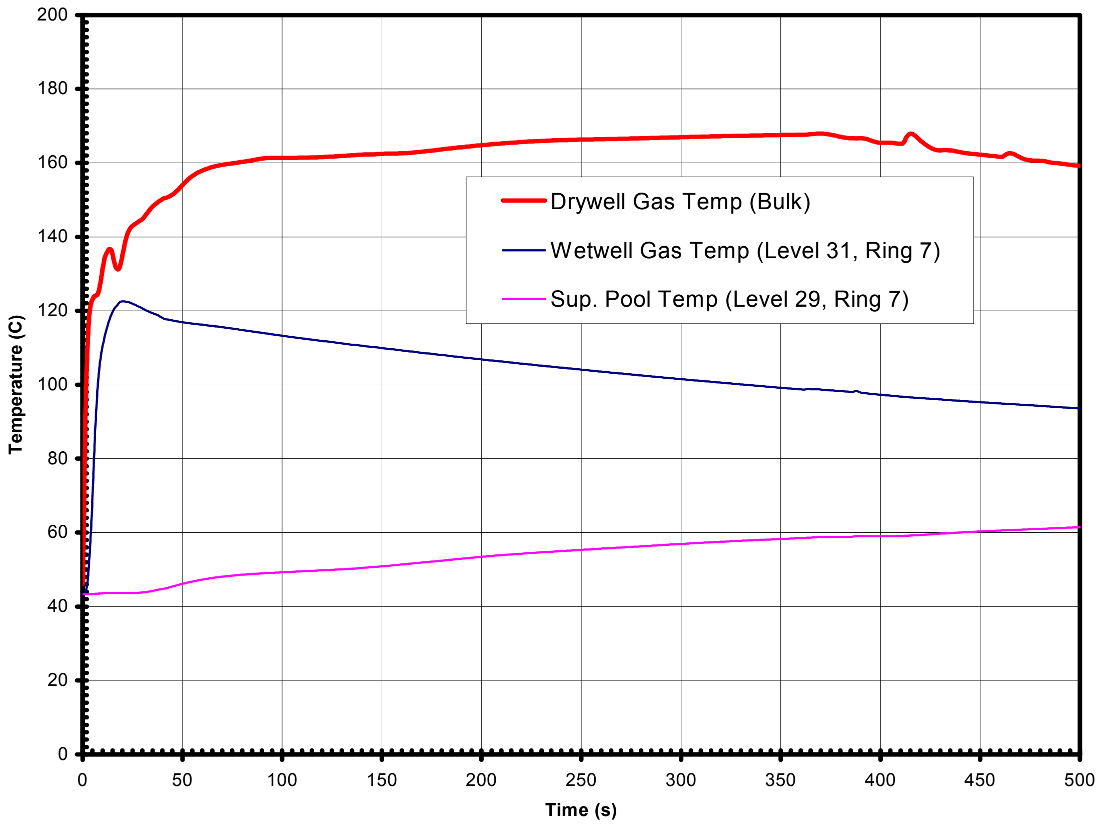


Figure 6.2-13f2. Feed Water Line Break, 1 SRV Failure (Bounding Case) – Containment Temperatures (500 s)

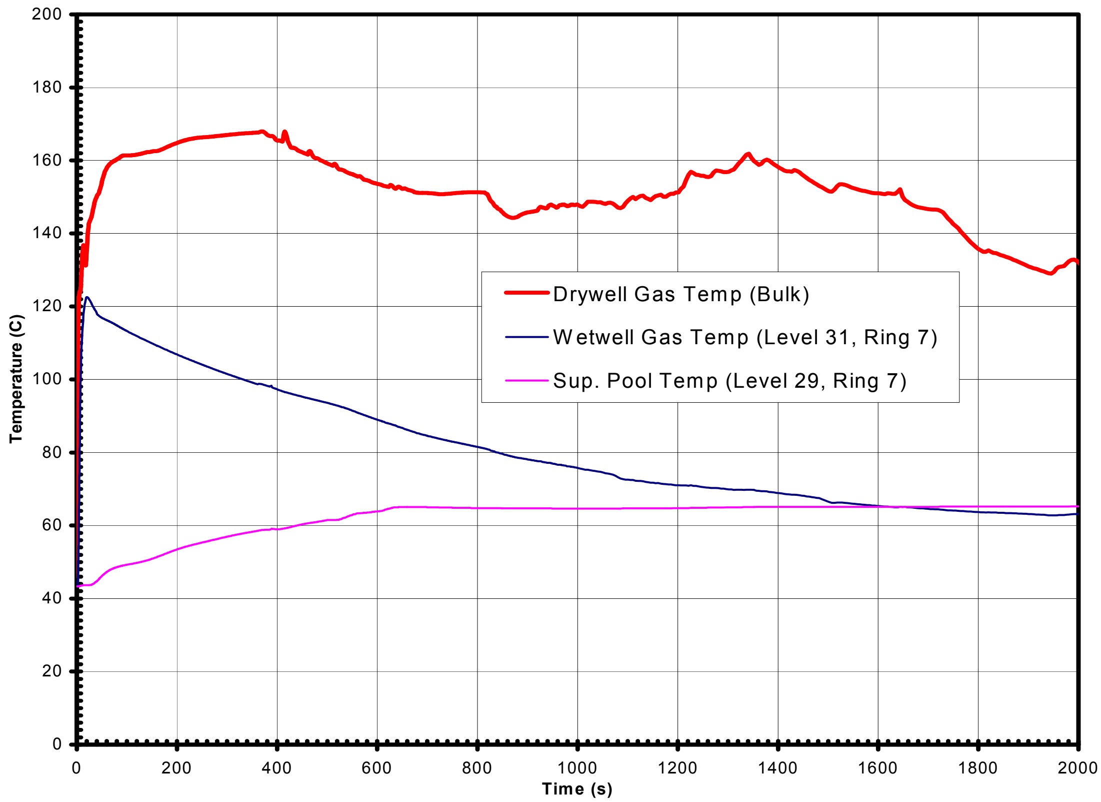


Figure 6.2-13f3. Feed Water Line Break, 1 SRV Failure (Bounding Case) – Containment Temperatures (2000 s)

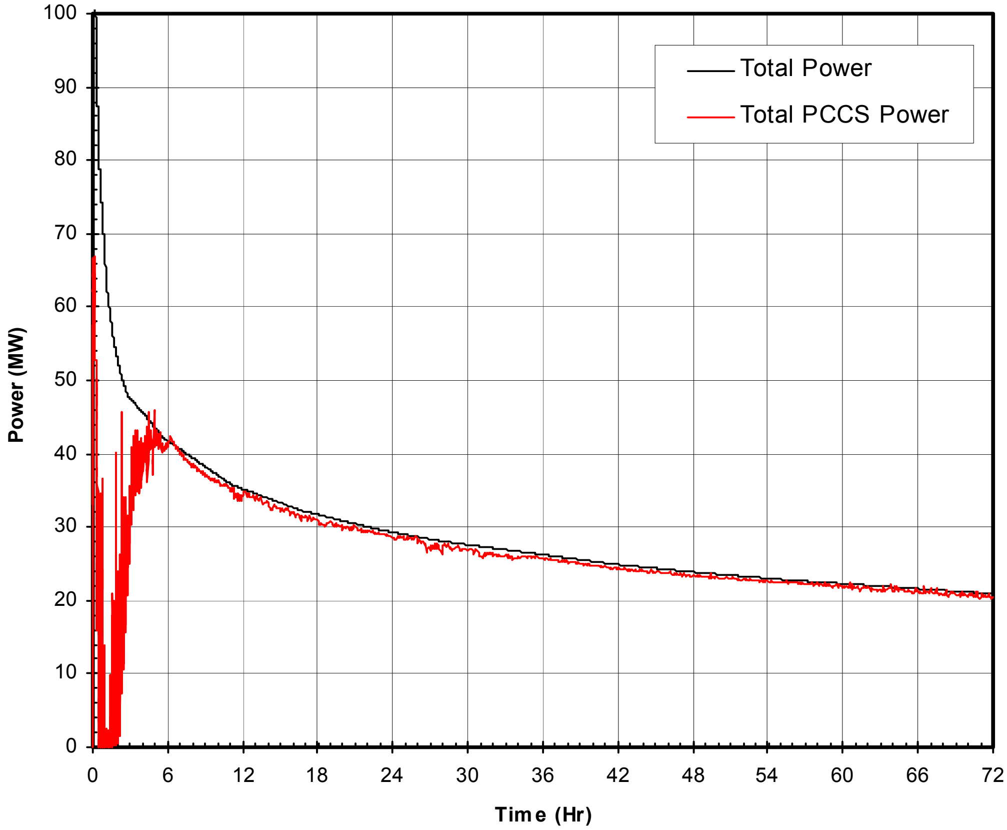


Figure 6.2-13g1. Feed Water Line Break, 1 SRV Failure (Bounding Case) – PCCS Heat Removal versus Decay Heat (72 hrs)

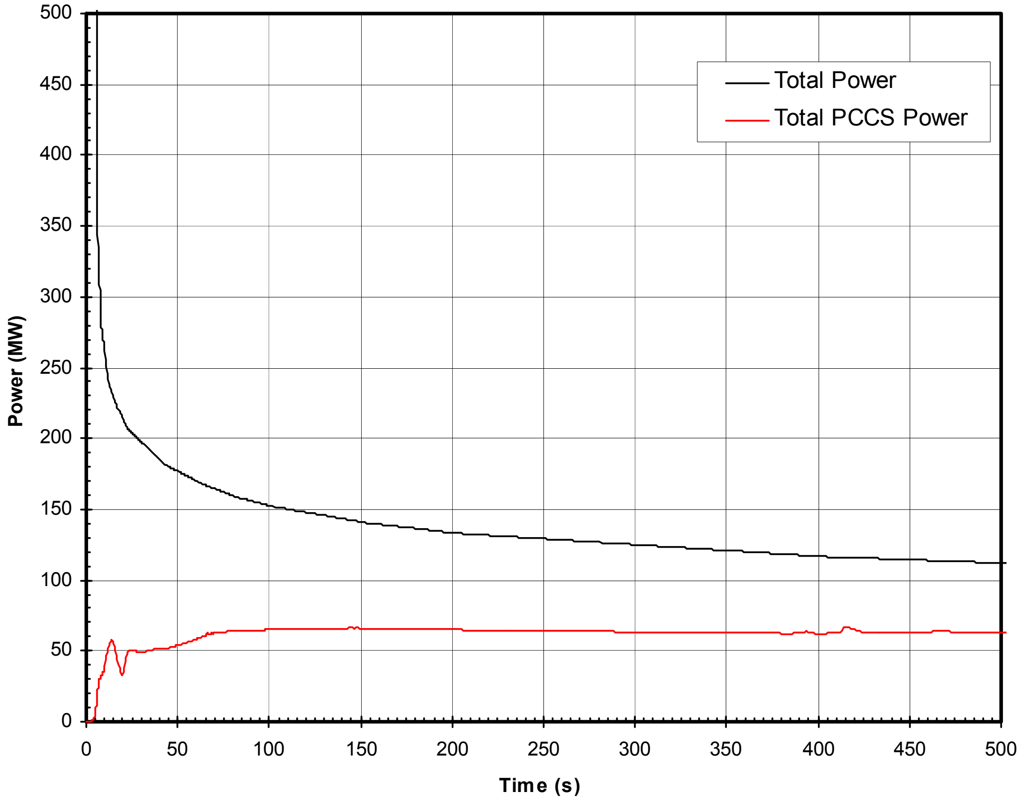


Figure 6.2-13g2. Feed Water Line Break, 1 SRV Failure (Bounding Case) – PCCS Heat Removal versus Decay Heat (500 s)

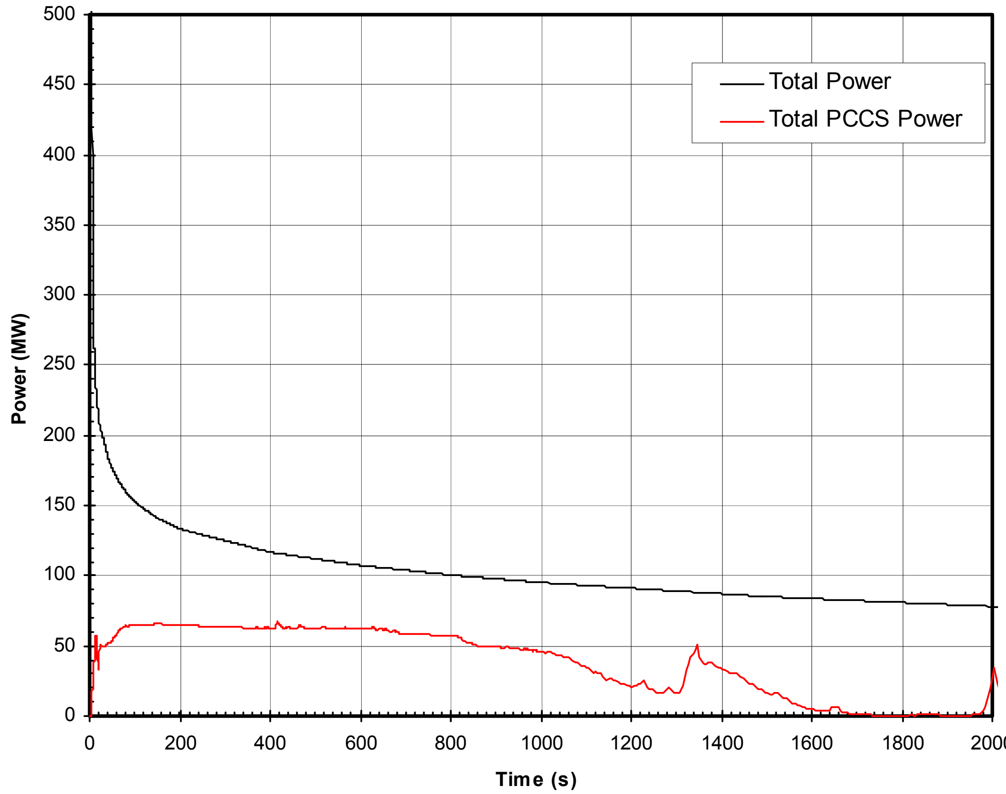


Figure 6.2-13g3. Feed Water Line Break, 1 SRV Failure (Bounding Case) – PCCS Heat Removal versus Decay Heat (2000 s)

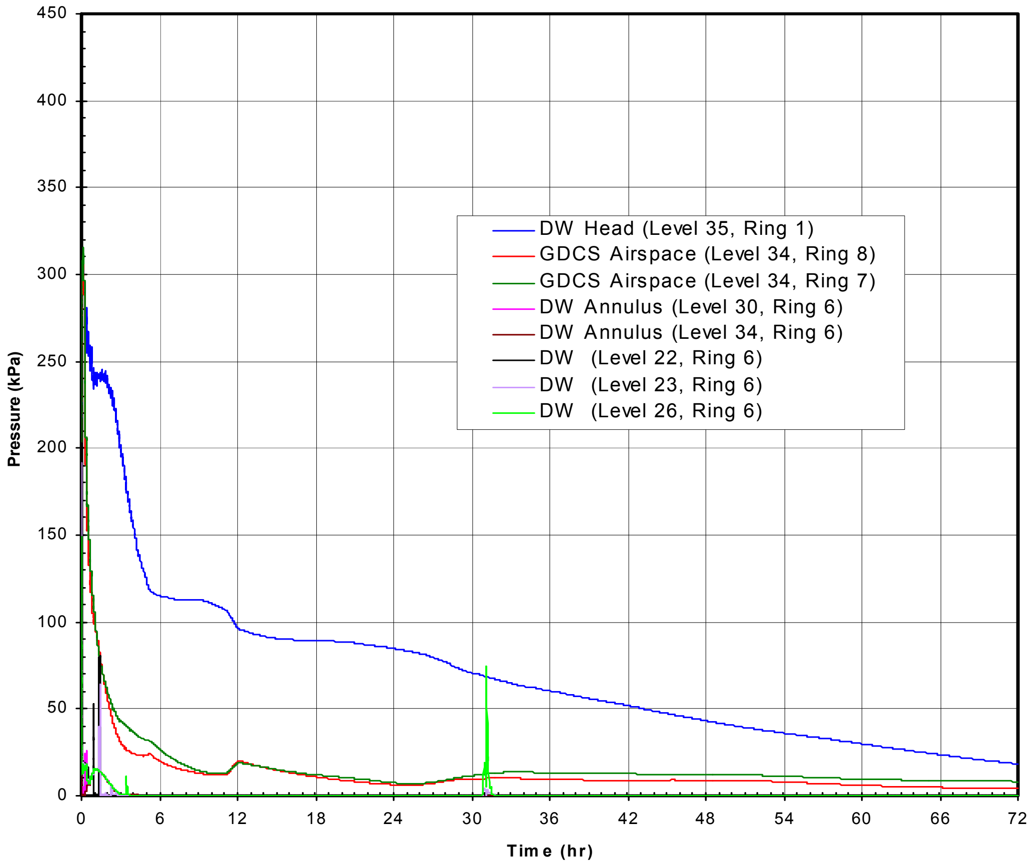


Figure 6.2-13h1. Feed Water Line Break, 1 SRV Failure (Bounding Case) – DW and GDCS Noncondensable Gas Pressures (72 hrs)

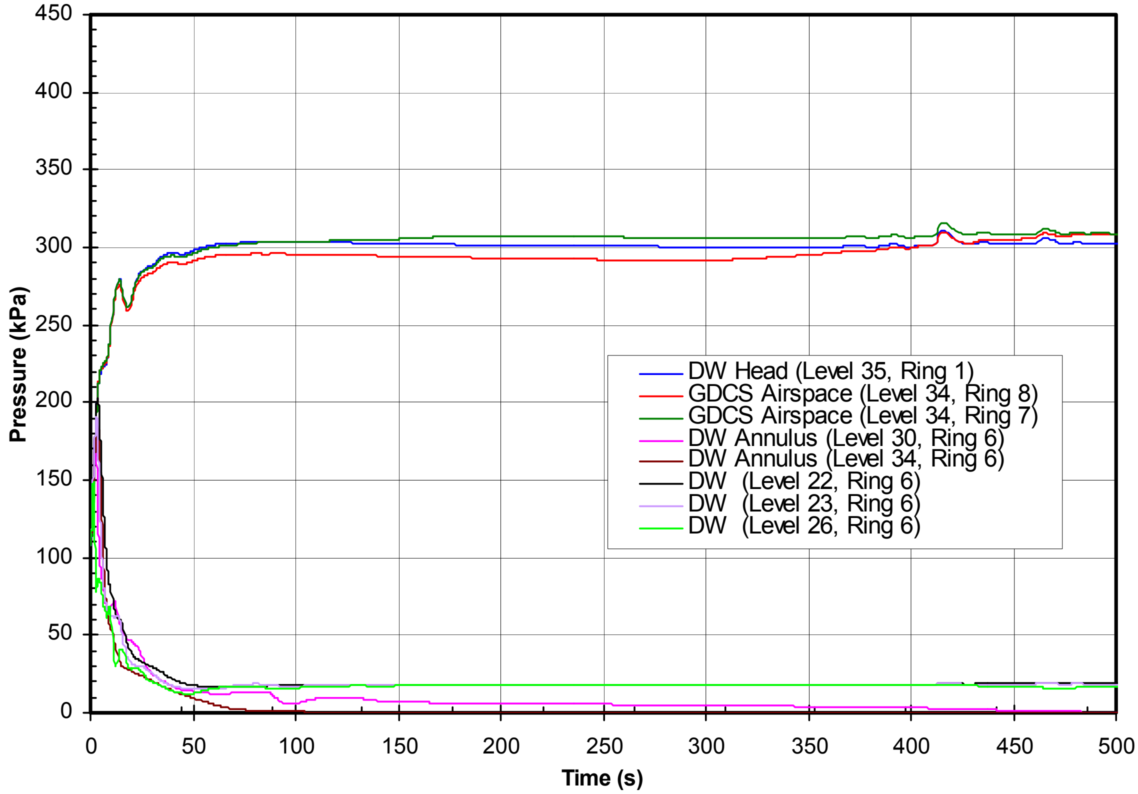


Figure 6.2-13h2. Feed Water Line Break, 1 SRV Failure (Bounding Case) – DW and GDCS Noncondensable Gas Pressures (500 s)

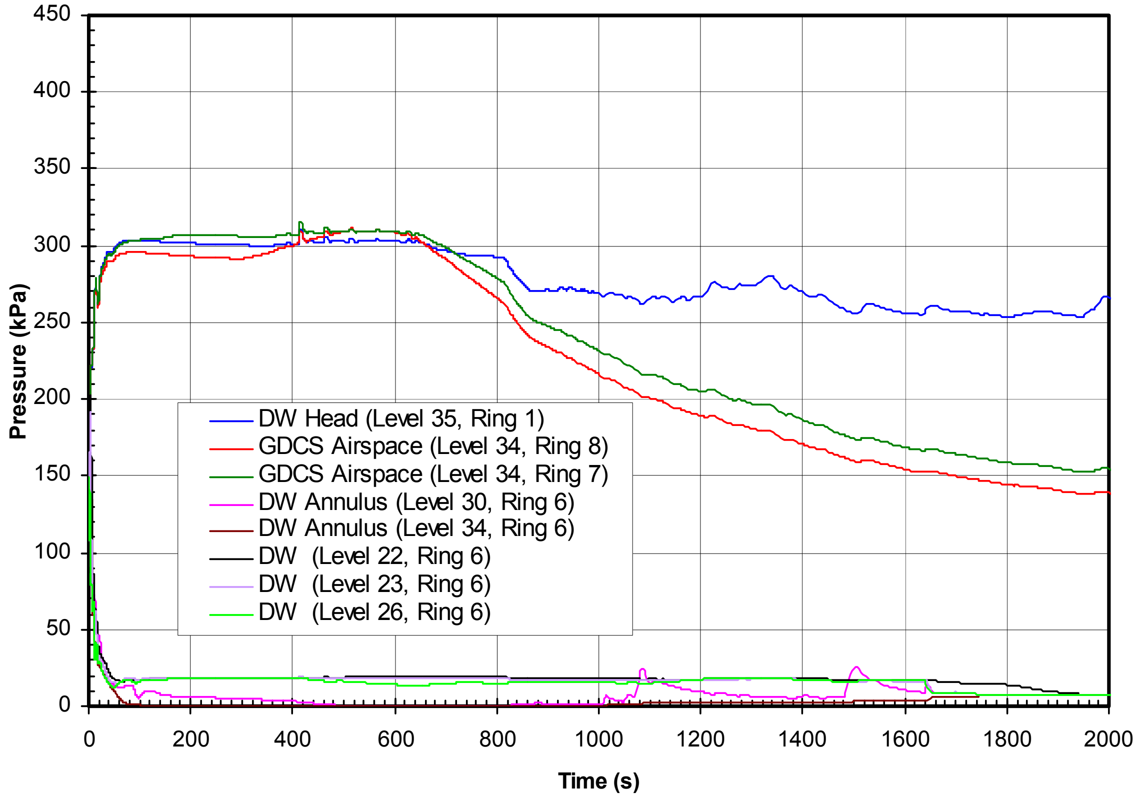


Figure 6.2-13h3. Feed Water Line Break, 1 SRV Failure (Bounding Case) – DW and GDCS Noncondensable Gas Pressures (2000 s)

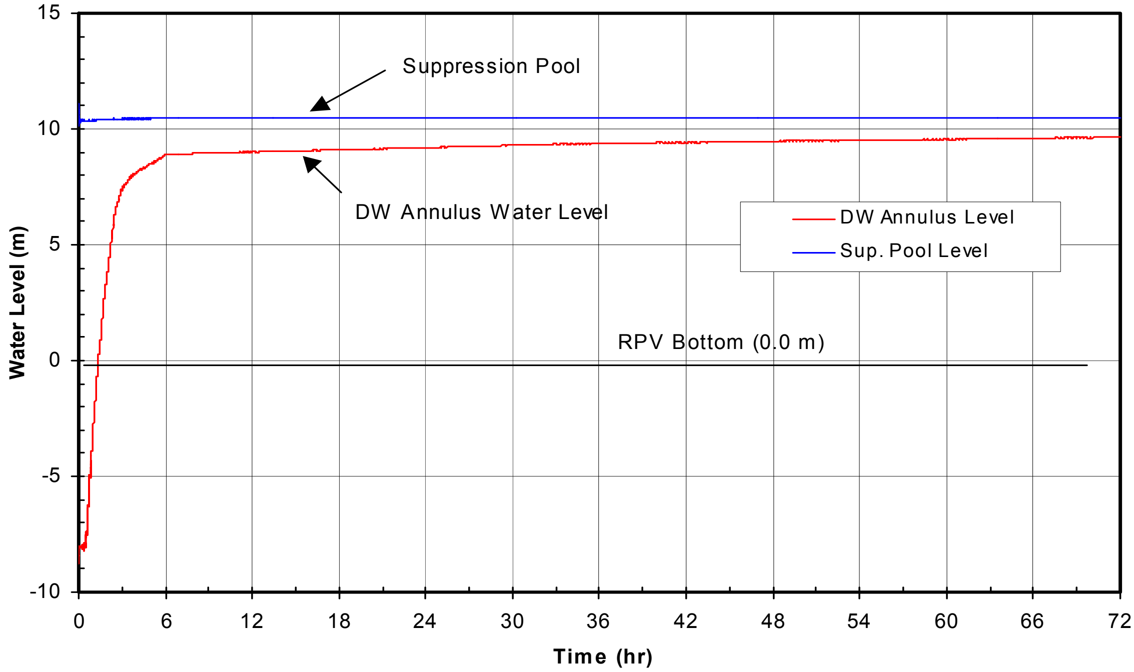


Figure 6.2-13h4. Feedwater Line Break, 1 SRV Failure (Bounding) DW Annulus and Suppression Pool Levels (72 hrs)

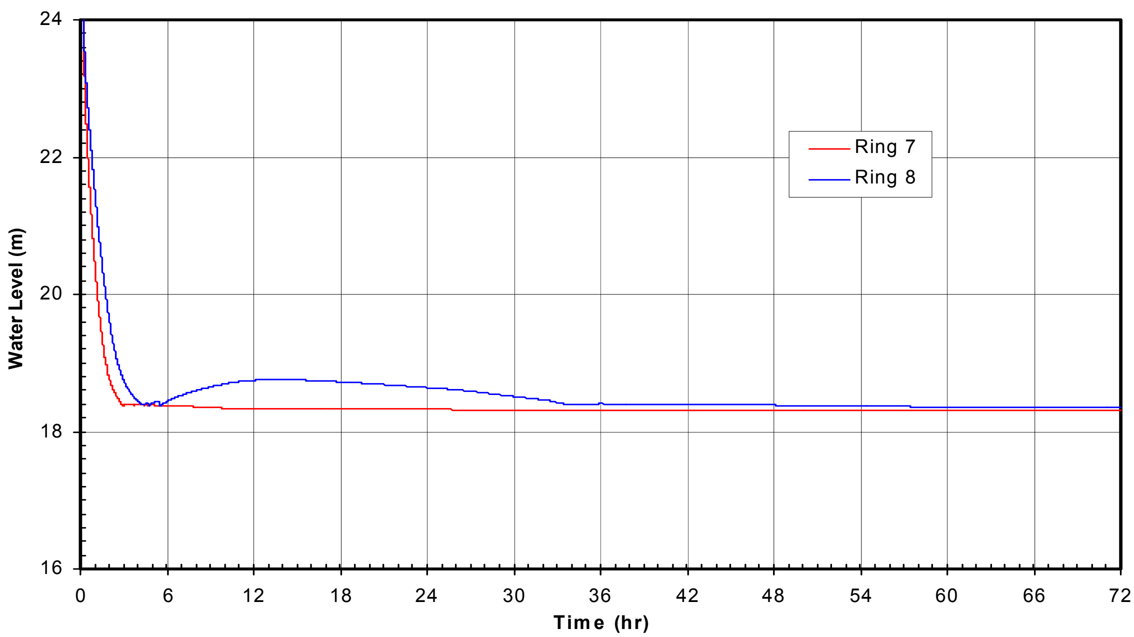


Figure 6.2-13h5. Feedwater Line Break, 1 SRV Failure (Bounding) GDCS Pool Levels (72 hrs)

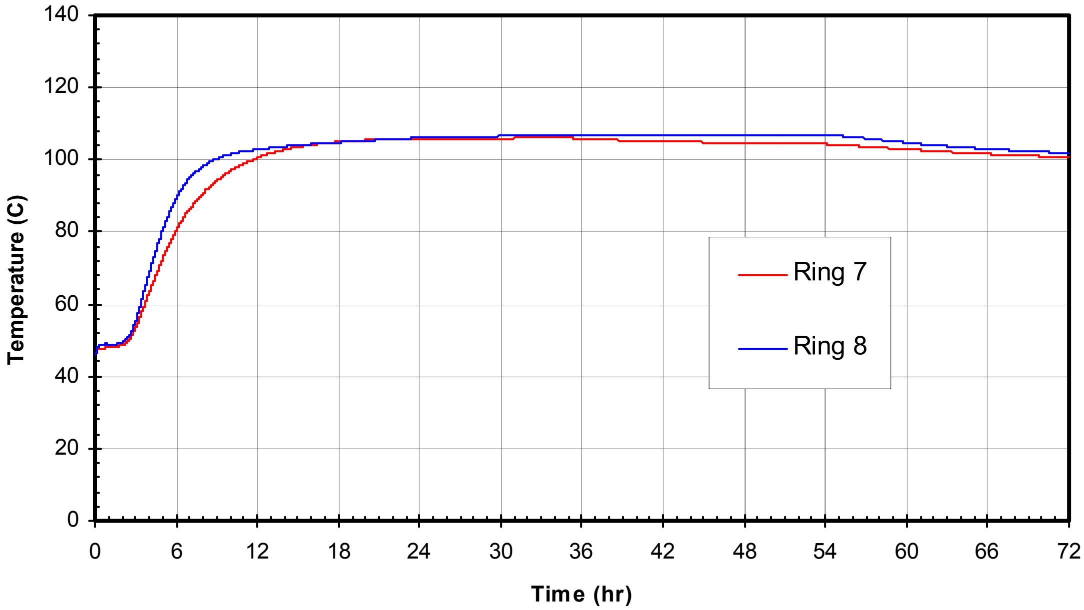


Figure 6.2-13h6. Feedwater Line Break, 1 SRV Failure (Bounding) GDCS Pool Temperature (72 hrs)

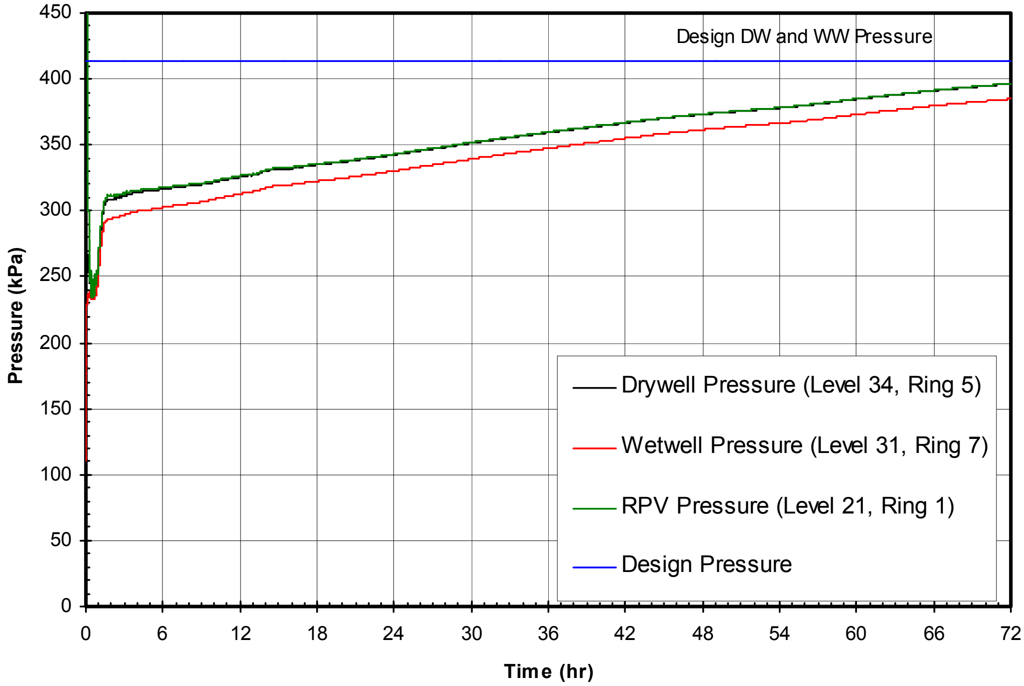


Figure 6.2-14a1. Main Steam Line Break, 1 DPV Failure (Bounding Case) - Containment Pressures (72 hrs)

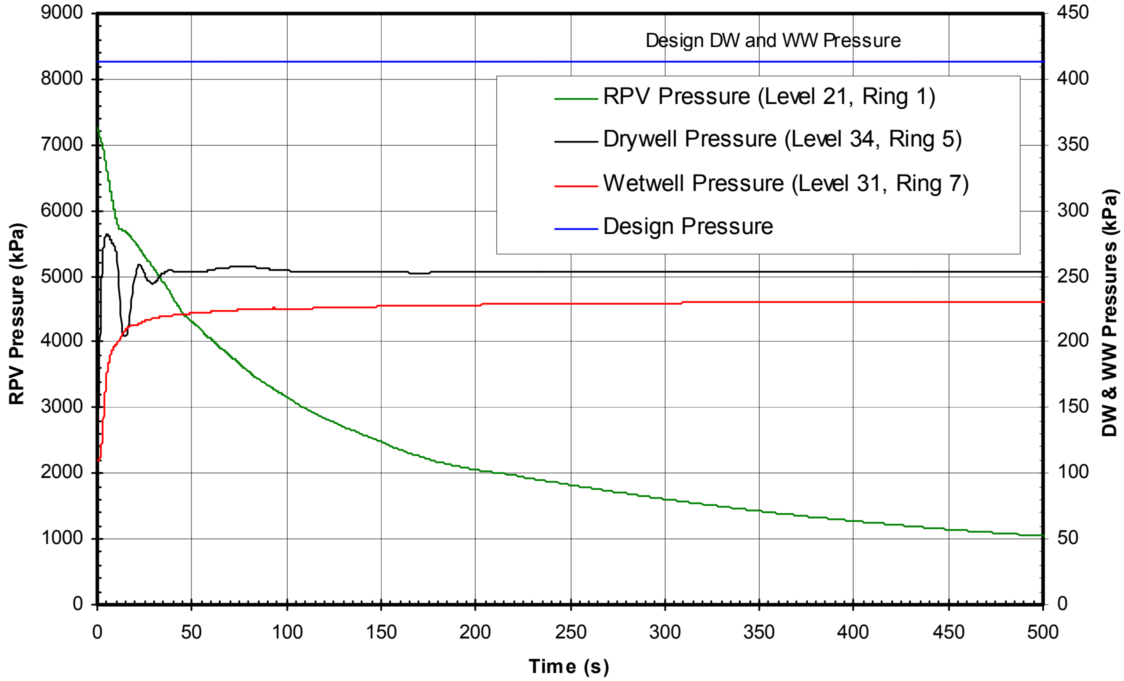


Figure 6.2-14a2. Main Steam Line Break, 1 DPV Failure (Bounding Case) – Containment Pressures (500 s)

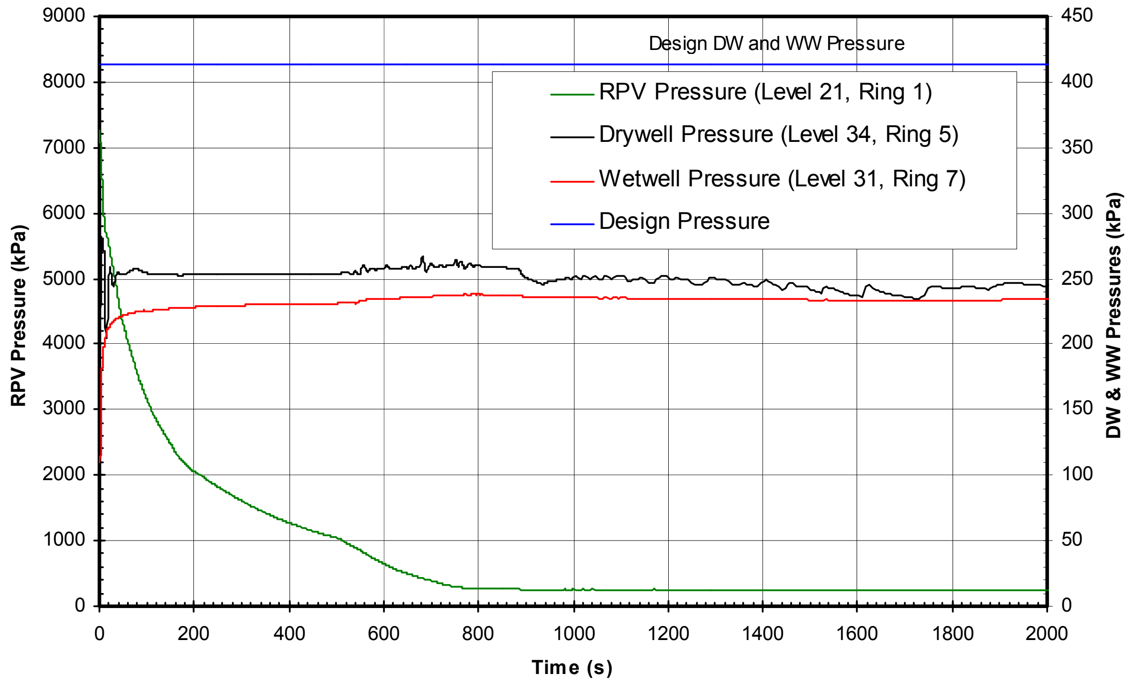


Figure 6.2-14a3. Main Steam Line Break, 1 DPV Failure (Bounding Case) – Containment Pressures (2000 s)

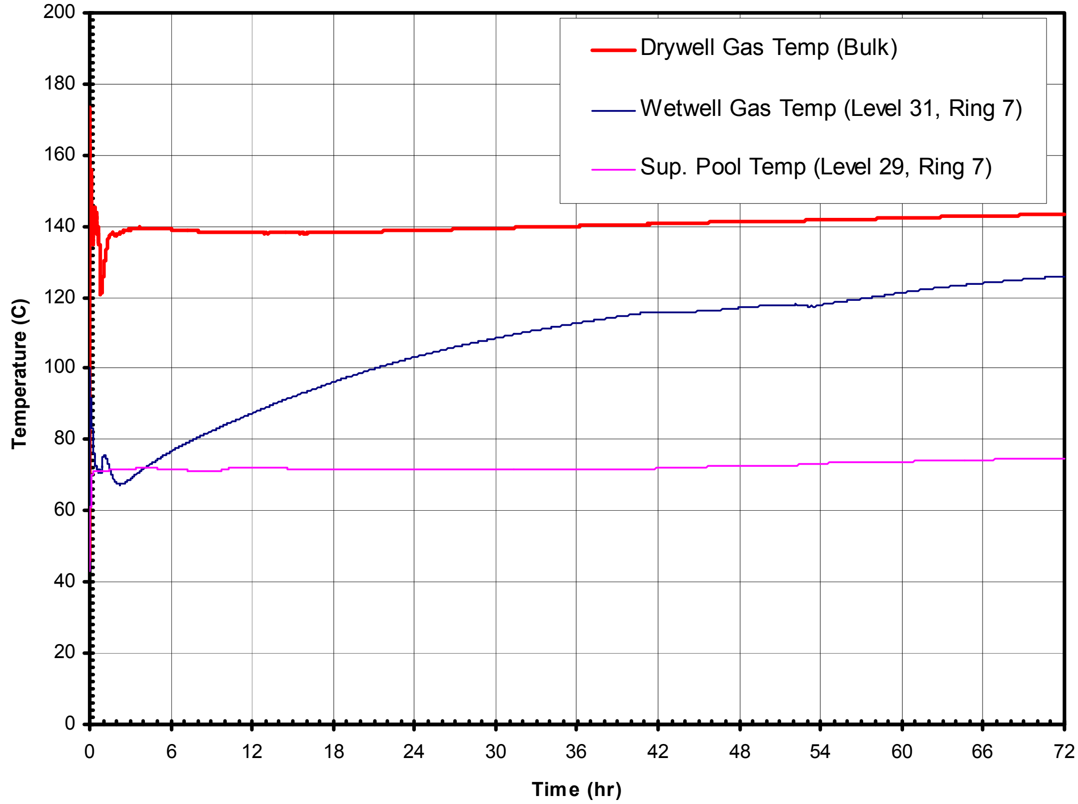


Figure 6.2-14b1. Main Steam Line Break, 1 DPV Failure (Bounding Case) – Containment Temperatures (72 hrs)

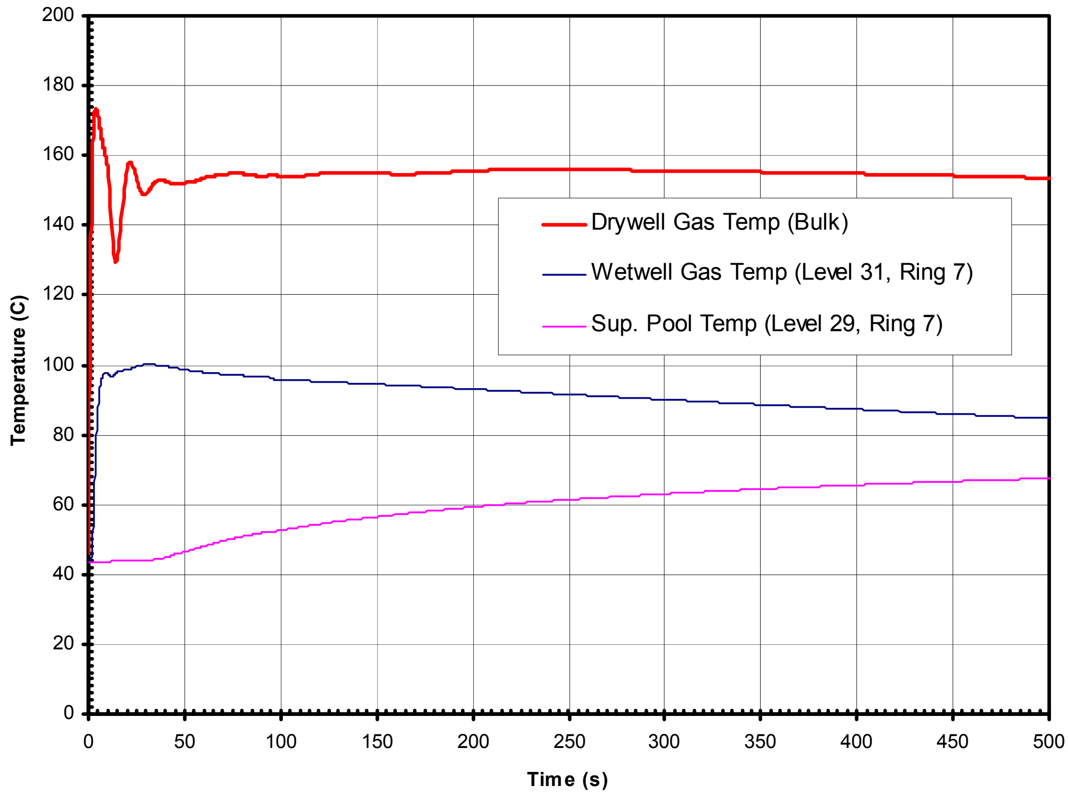


Figure 6.2-14b2. Main Steam Line Break, 1 DPV Failure (Bounding Case) – Containment Temperatures (500 s)

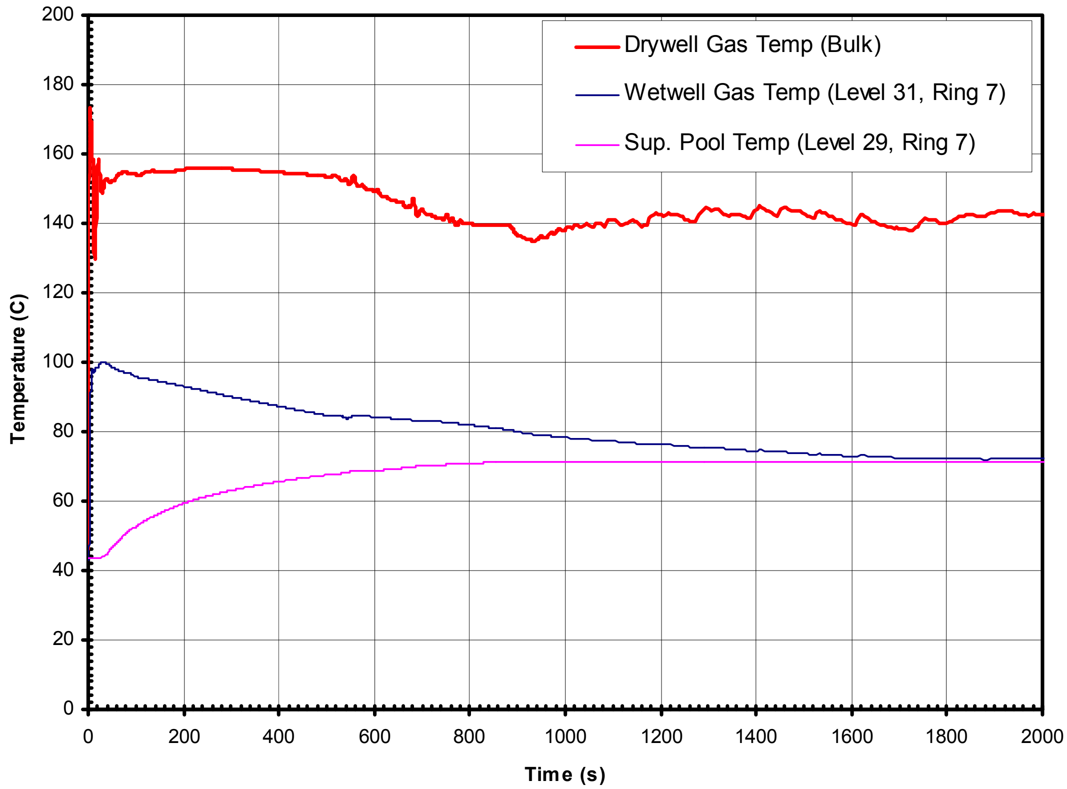


Figure 6.2-14b3. Main Steam Line Break, 1 DPV Failure (Bounding Case) – Containment Temperatures (2000 s)

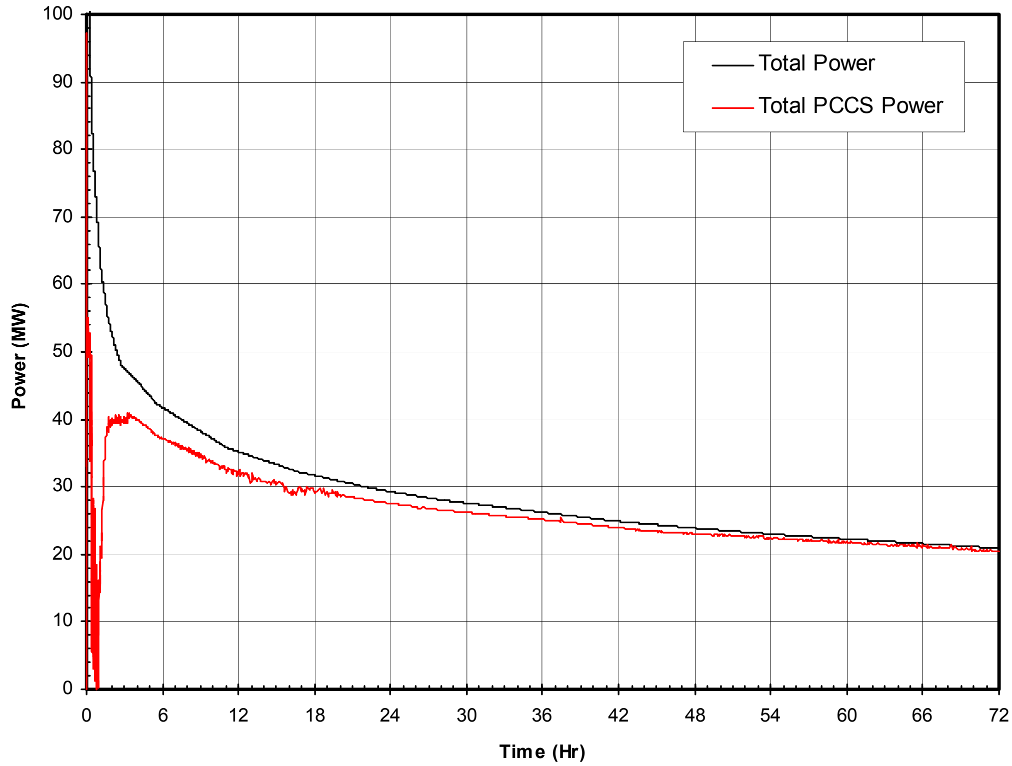


Figure 6.2-14c1. Main Steam Line Break, 1 DPV Failure (Bounding Case) – PCCS Heat Removal versus Decay Heat (72 hrs)

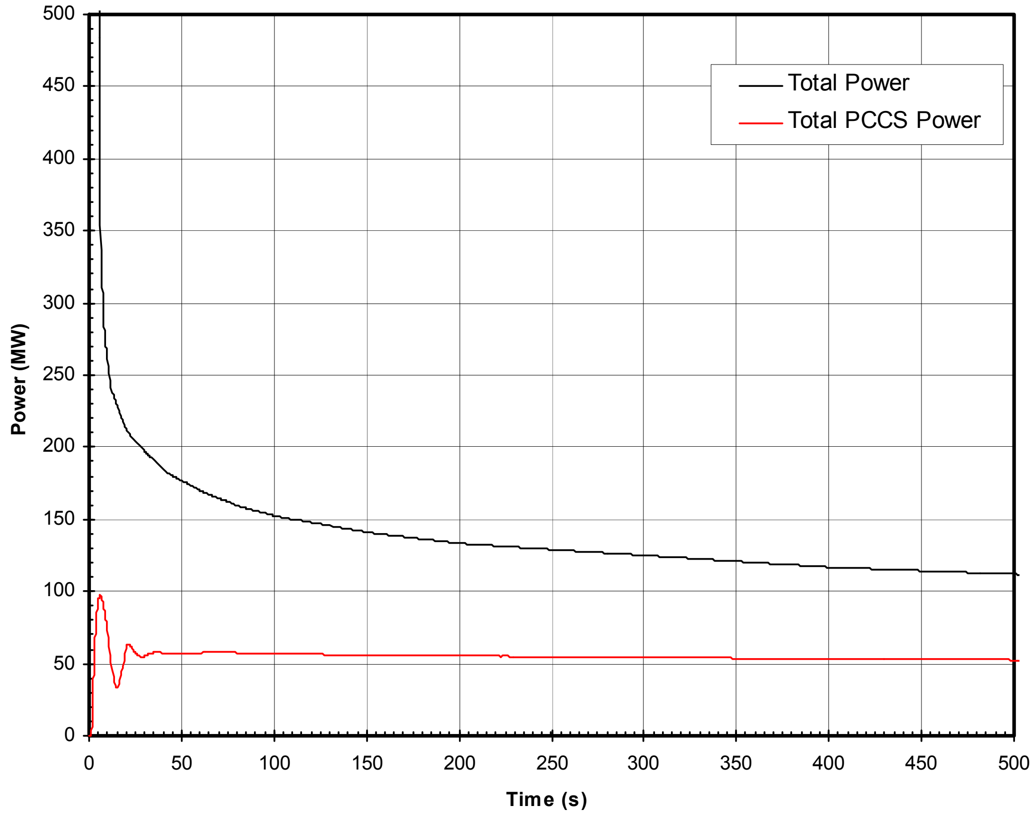


Figure 6.2-14c2. Main Steam Line Break, 1 DPV Failure (Bounding Case) – PCCS Heat Removal versus Decay Heat (500 s)

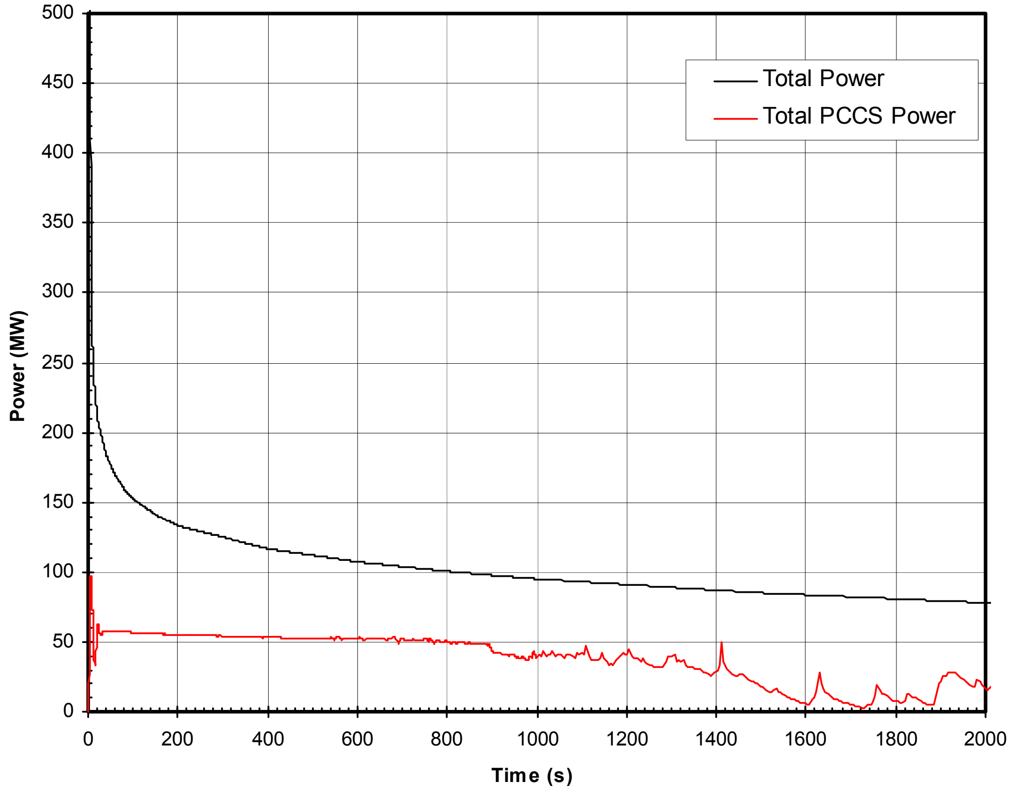


Figure 6.2-14c3. Main Steam Line Break, 1 DPV Failure (Bounding Case) – PCCS Heat Removal versus Decay Heat (2000 s)

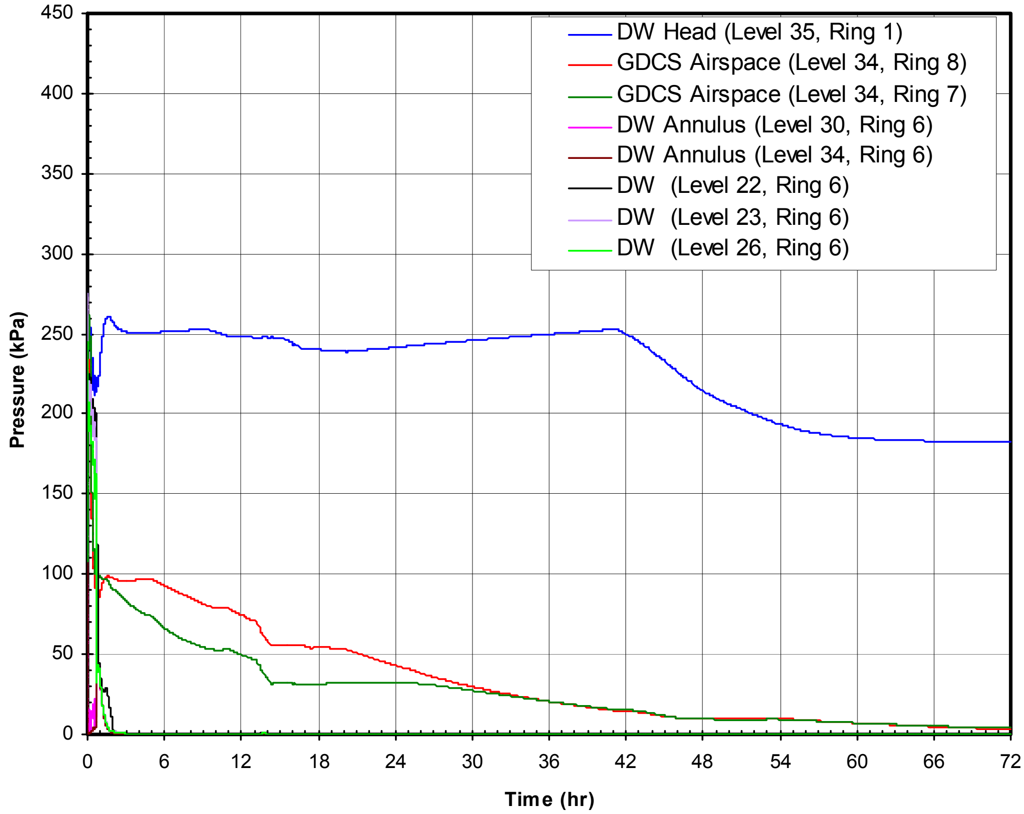


Figure 6.2-14d1. Main Steam Line Break, 1 DPV Failure (Bounding Case) - Drywell and GDCS Noncondensable Gas Pressures (72 hrs)

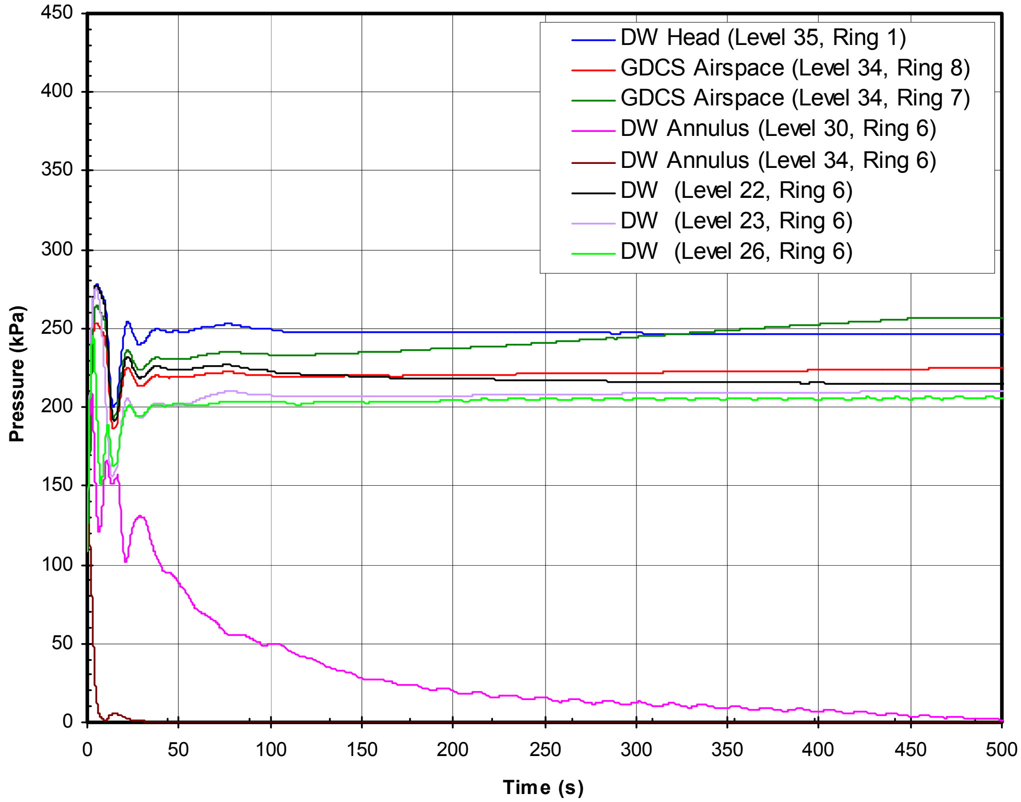


Figure 6.2-14d2. Main Steam Line Break, 1 DPV Failure (Bounding Case) - Drywell and GDCS Noncondensable Gas Pressures (500 s)

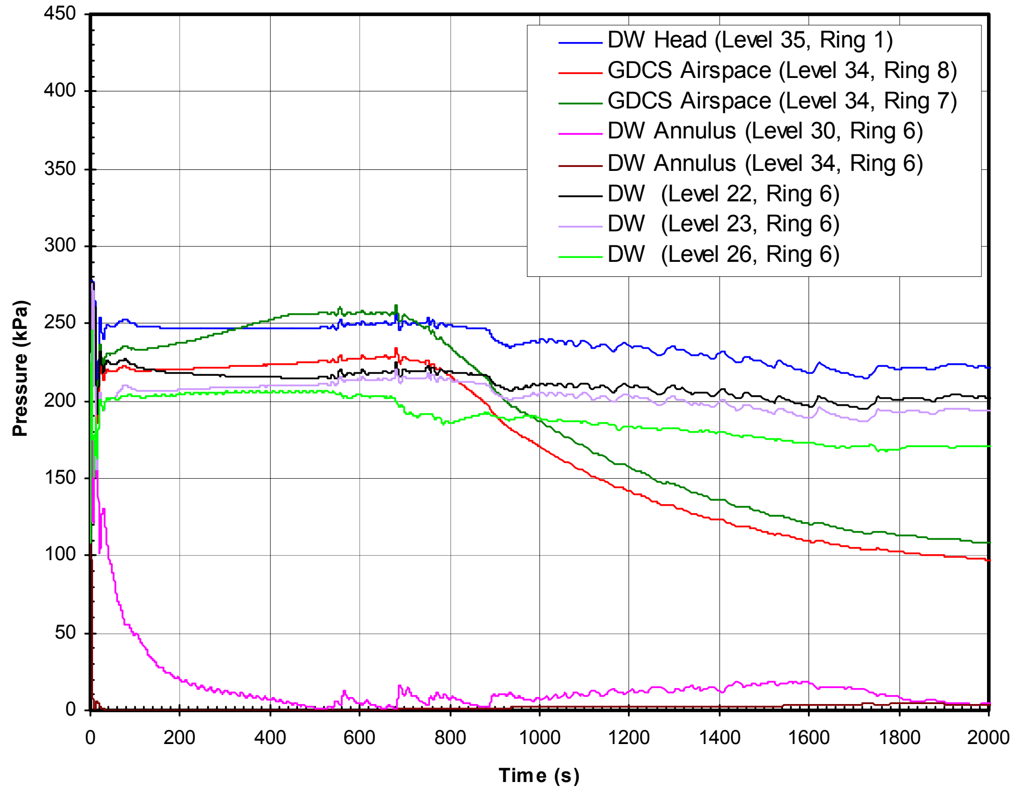


Figure 6.2-14d3. Main Steam Line Break, 1 DPV Failure (Bounding Case) - Drywell and GDCS Noncondensable Gas Pressures (2000 s)

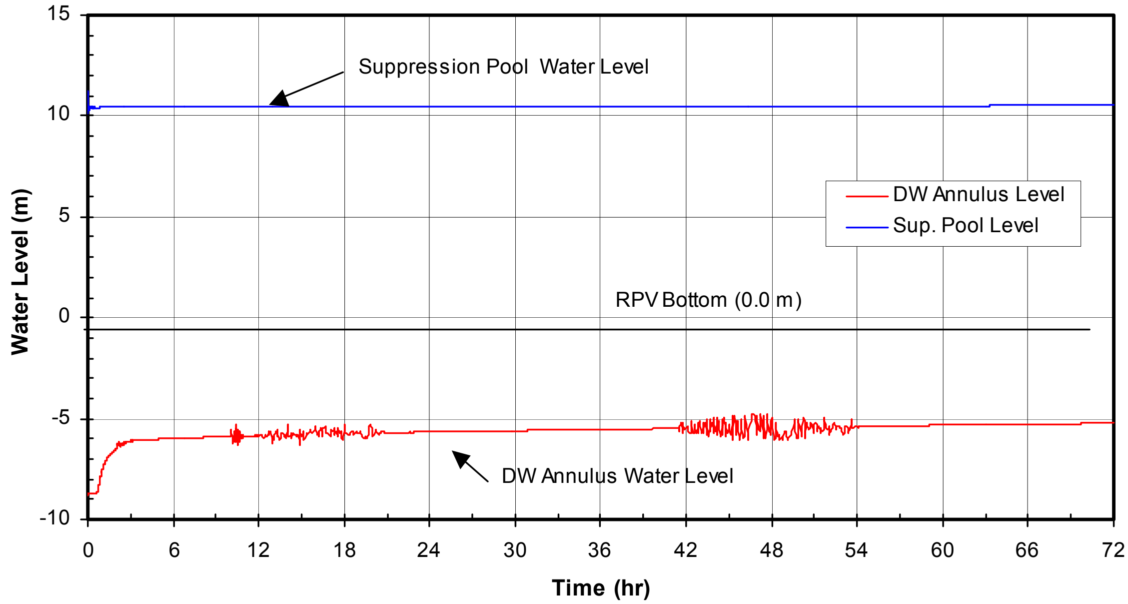


Figure 6.2-14d4. Main Steam Line Break, 1 DPV Failure (Bounding) – Drywell Annulus and Suppression Pool Levels (72 hrs)

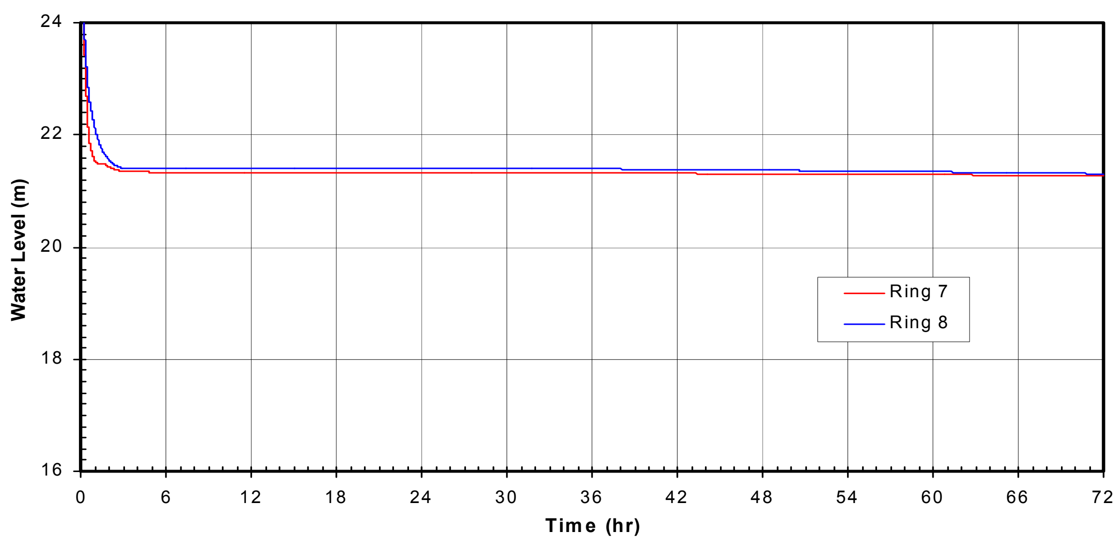


Figure 6.2-14d5. Main Steam Line Break, 1 DPV Failure (Bounding) – GDCS Pool Levels (72 hrs)

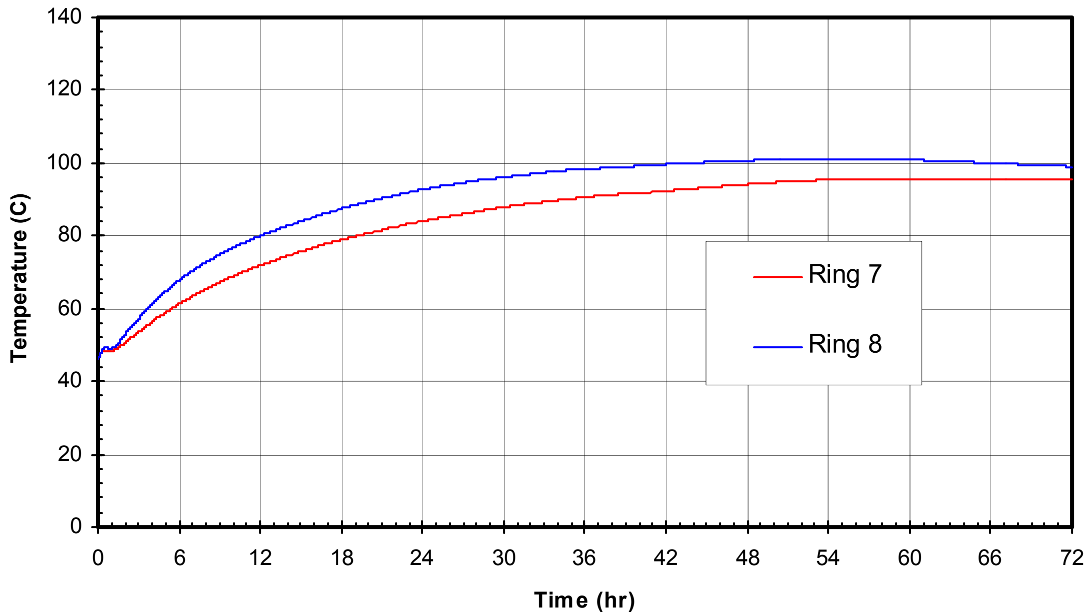


Figure 6.2-14d6. Main Steam Line Break, 1 DPV Failure (Bounding) – GDCS Pool Temperature (72 hrs)

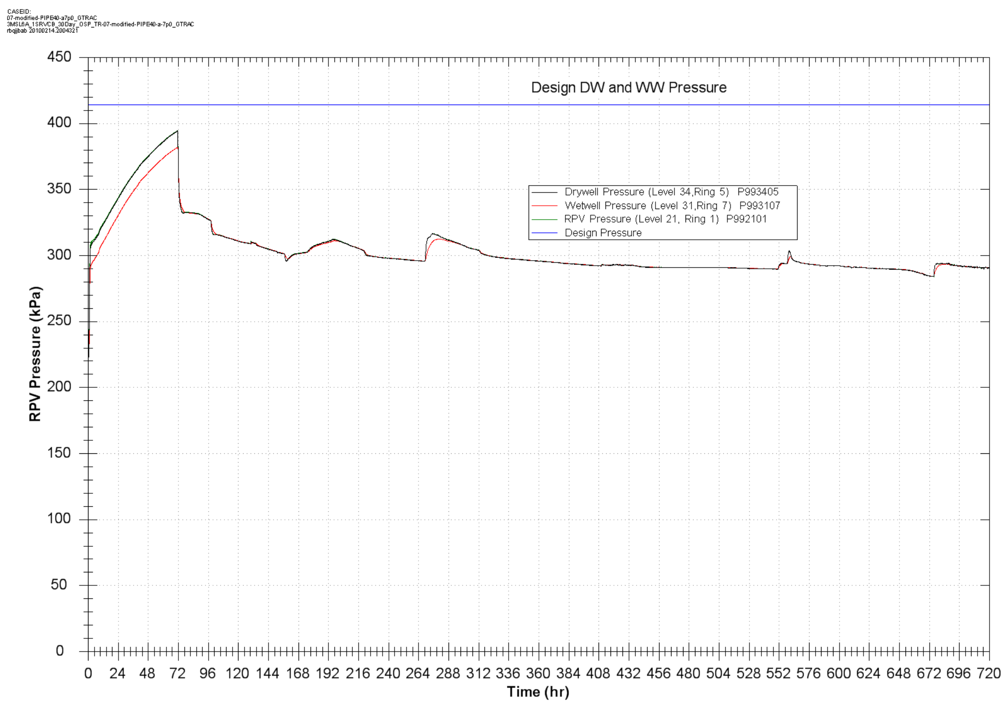


Figure 6.2-14e1. Drywell, Wetwell and RPV Pressures (720 hr)

Note: DW pressure perturbations at 269, & 551 hrs are caused by step increases in PCC fan discharge submergence to maintain 10 inches or greater submergence.

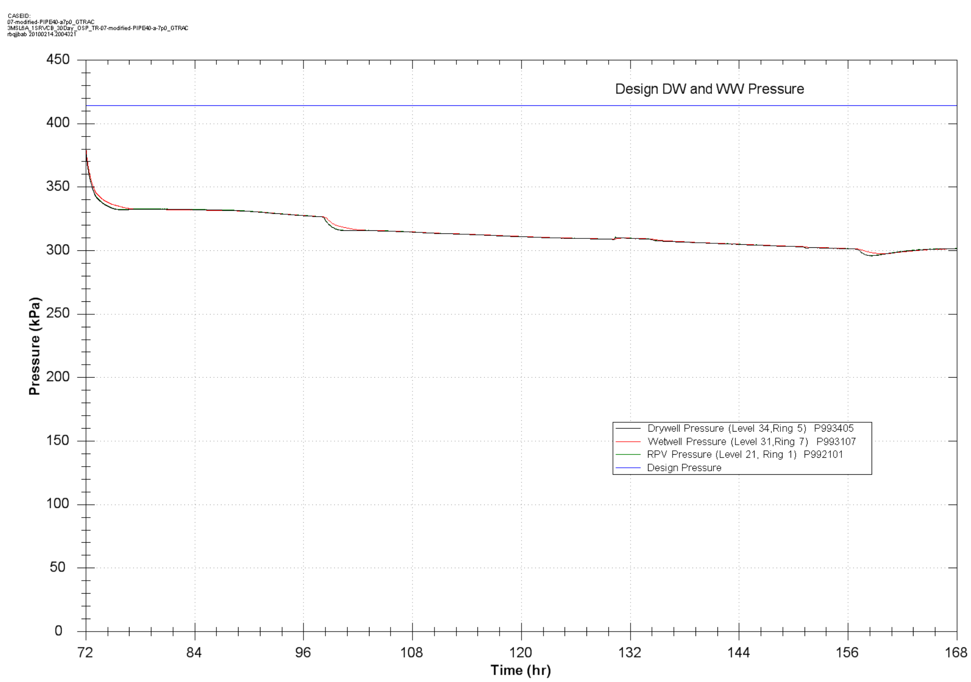


Figure 6.2-14e2. Drywell, Wetwell and RPV Pressures (72 – 144 hr)

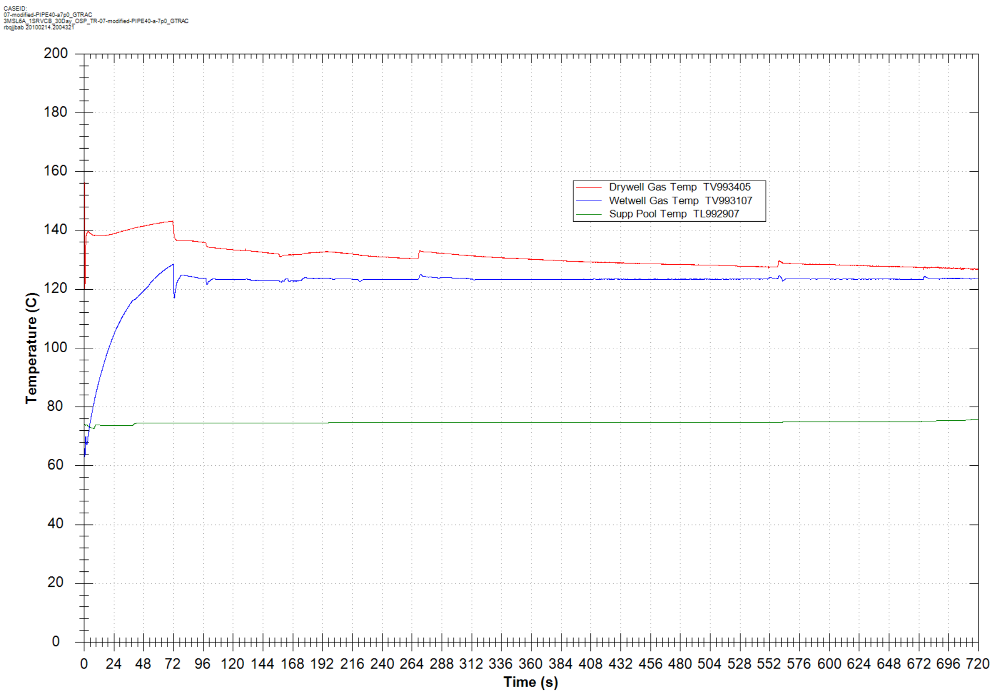


Figure 6.2-14e3. Drywell, Wetwell and Suppression Pool Temperatures (720 hr)

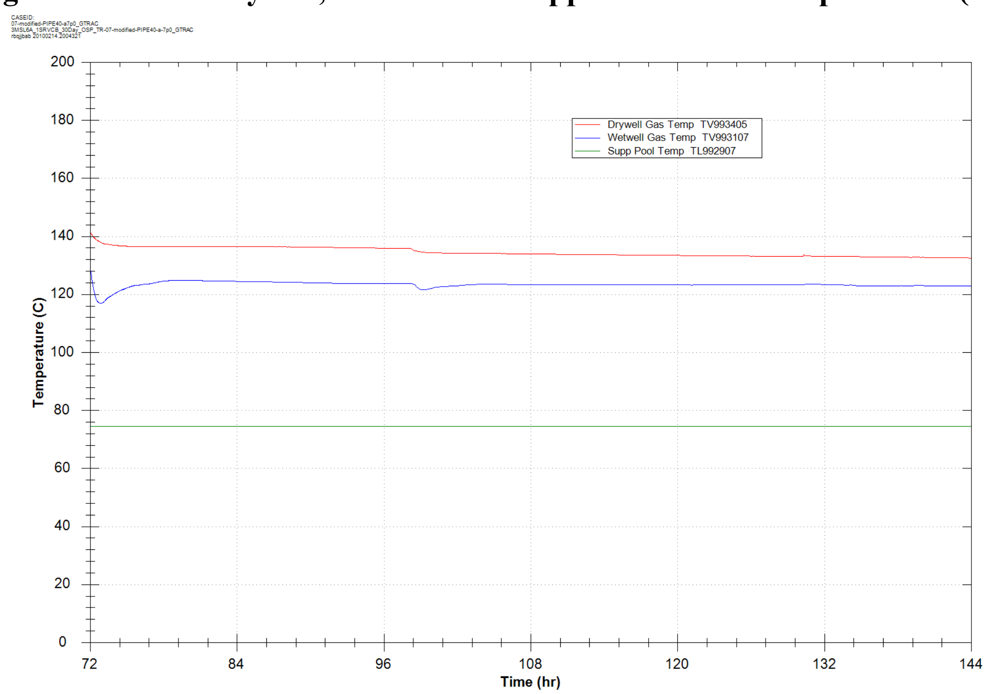


Figure 6.2-14e4. Drywell, Wetwell and Suppression Pool Temperatures (72 – 144 hr)

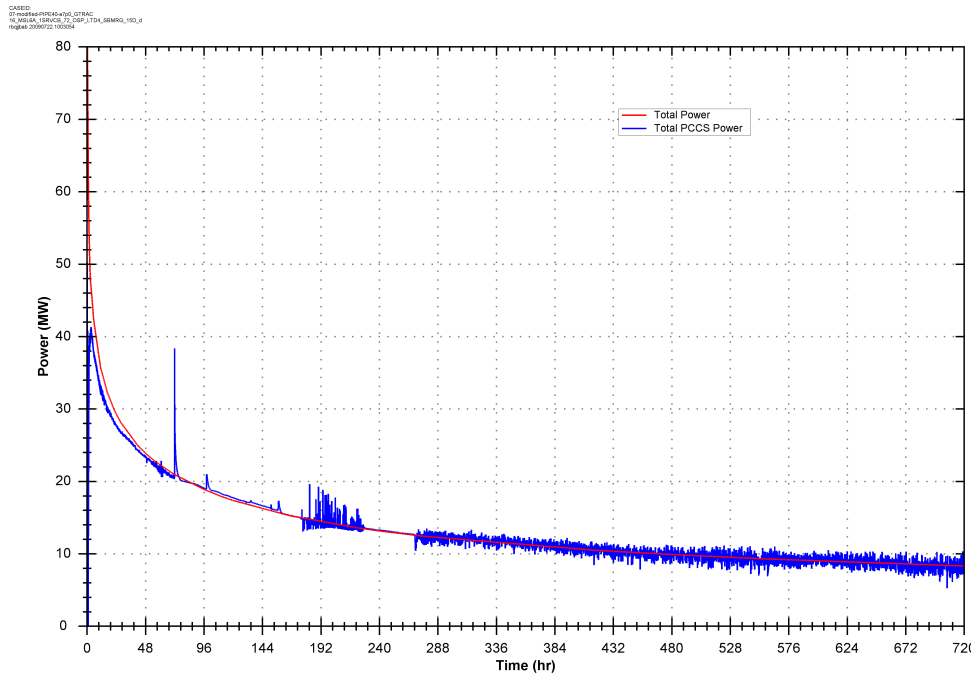


Figure 6.2-14e5. Total Power and Total PCCS Power (720 hr)

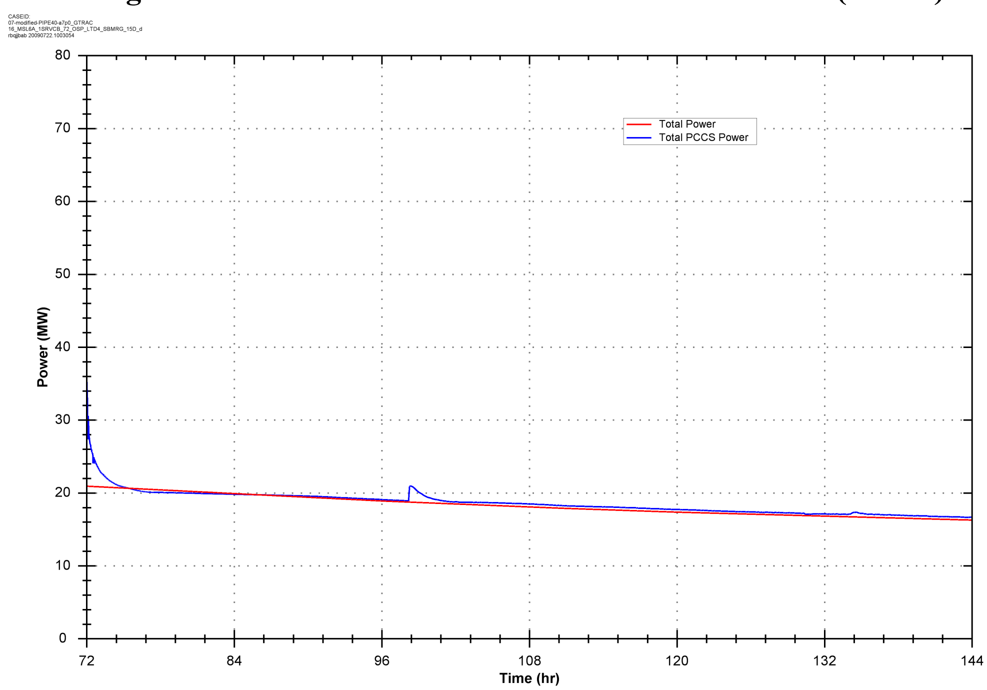


Figure 6.2-14e6: Total Power and Total PCCS Power (72 – 144 hr)

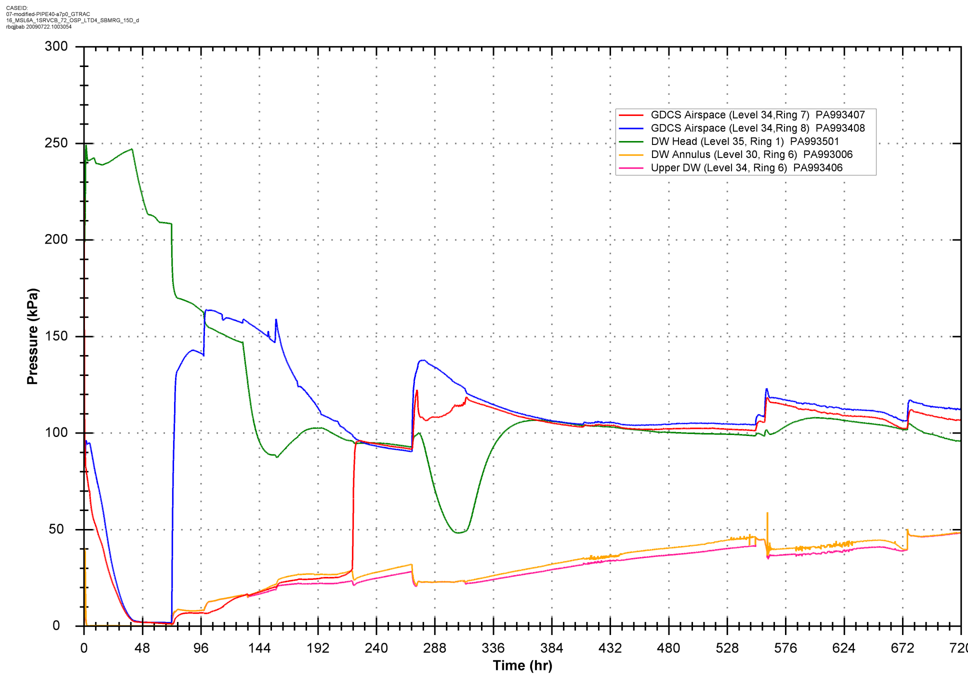


Figure 6.2-14e7. DW and GDCS Noncondensable Pressures (720 hr)

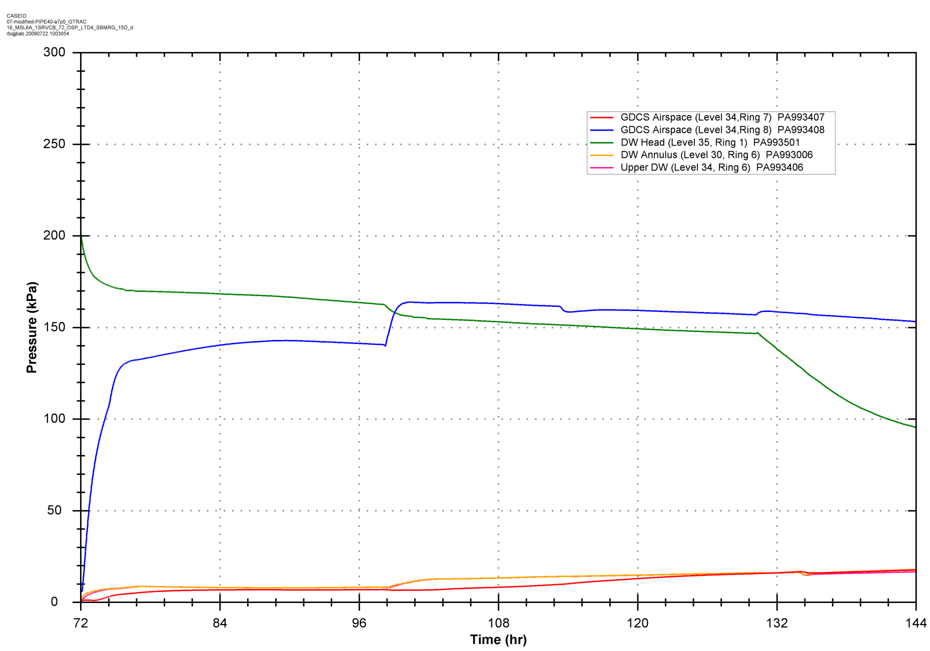


Figure 6.2-14e8. DW and GDCS Noncondensable Pressures (72 – 144 hr)

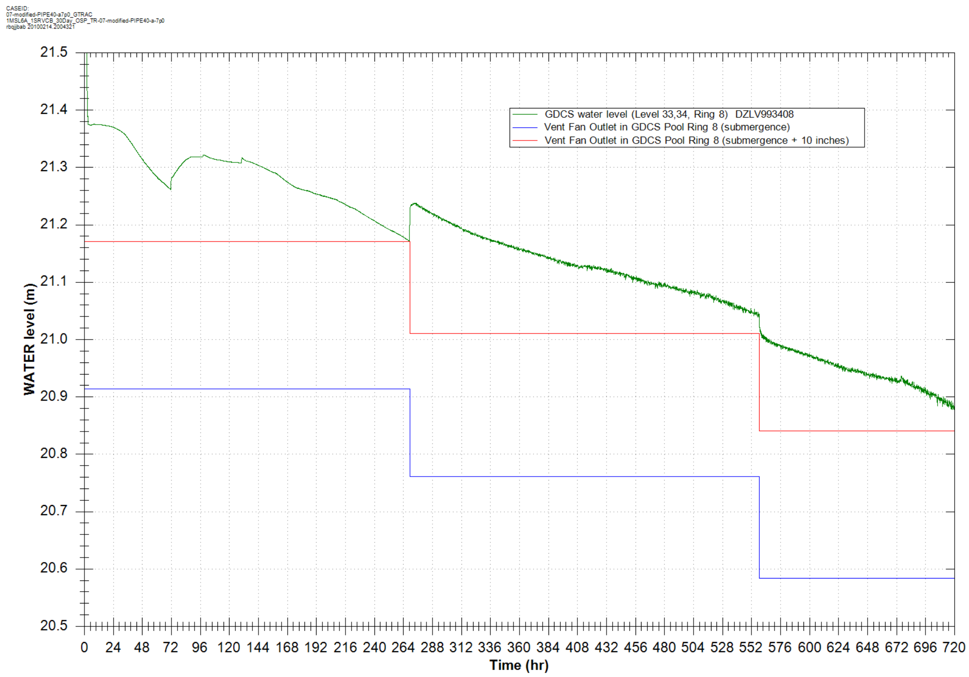


Figure 6.2-14e9. GDCS Pool Water Level (720 hr)

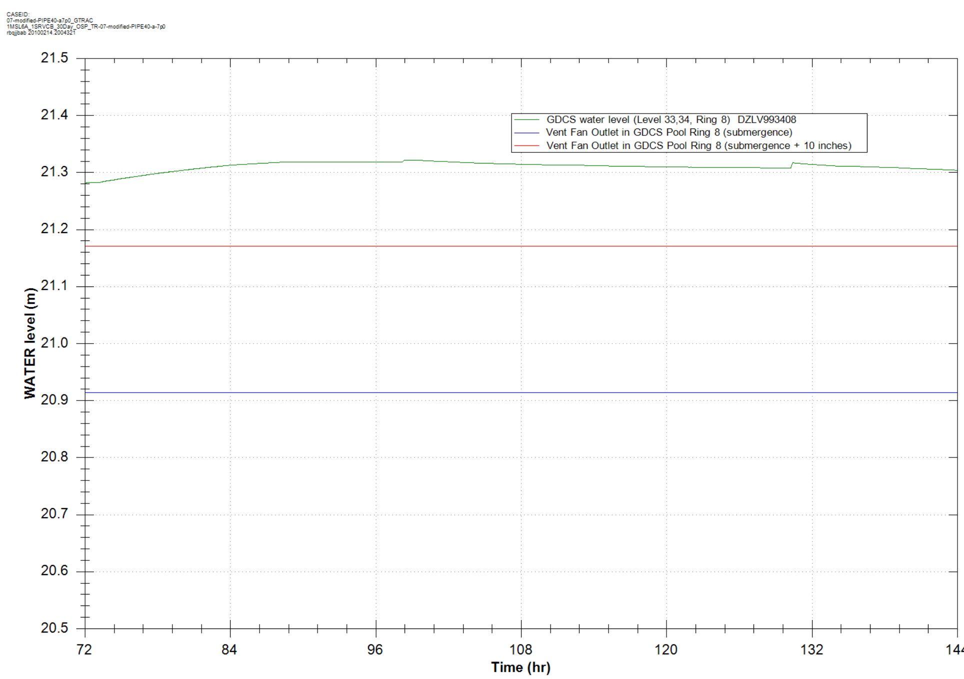


Figure 6.2-14e9a. GDCS Pool Water Level (72 – 144 hr)

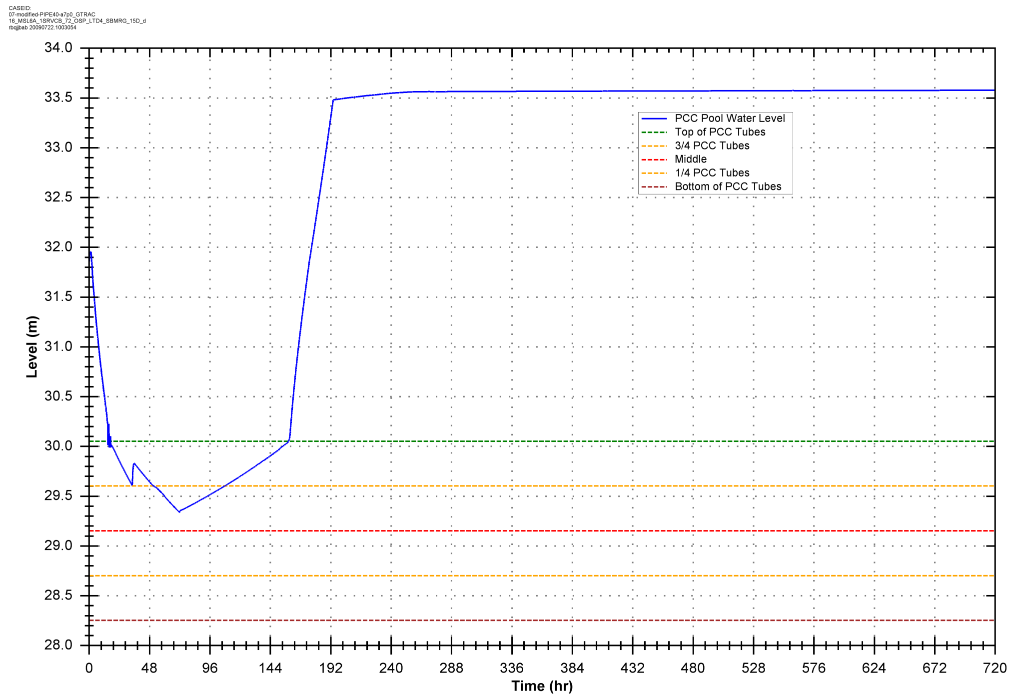


Figure 6.2-14e10. PCCS Pool Water Level (720 hr)

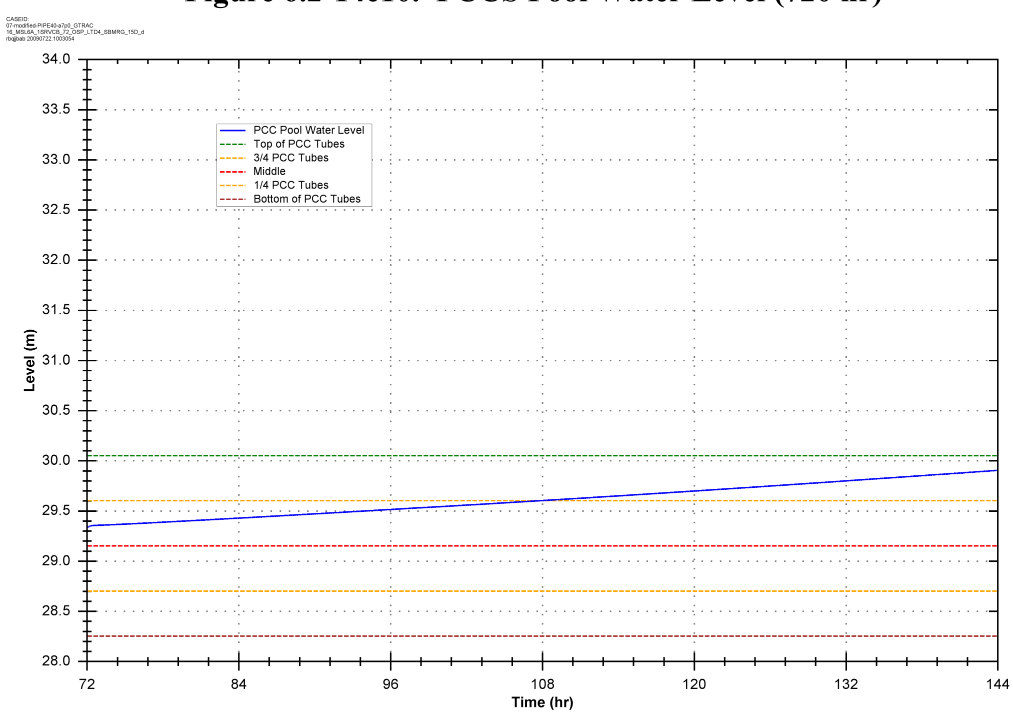


Figure 6.2-14e10a. PCCS Pool Water Level (72 – 144 hr)

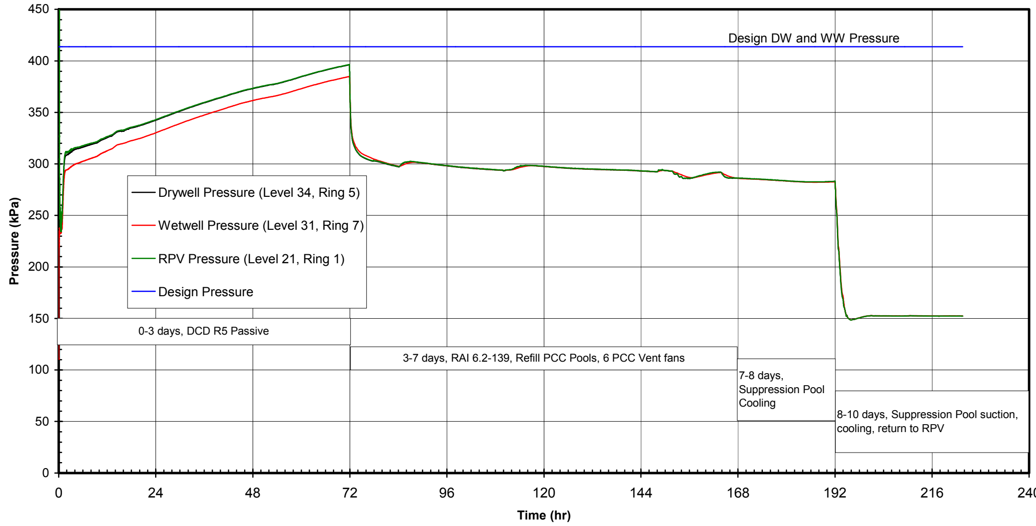


Figure 6.2-14e11. Containment Pressure Response – Post-LOCA Containment Cooling and Recovery

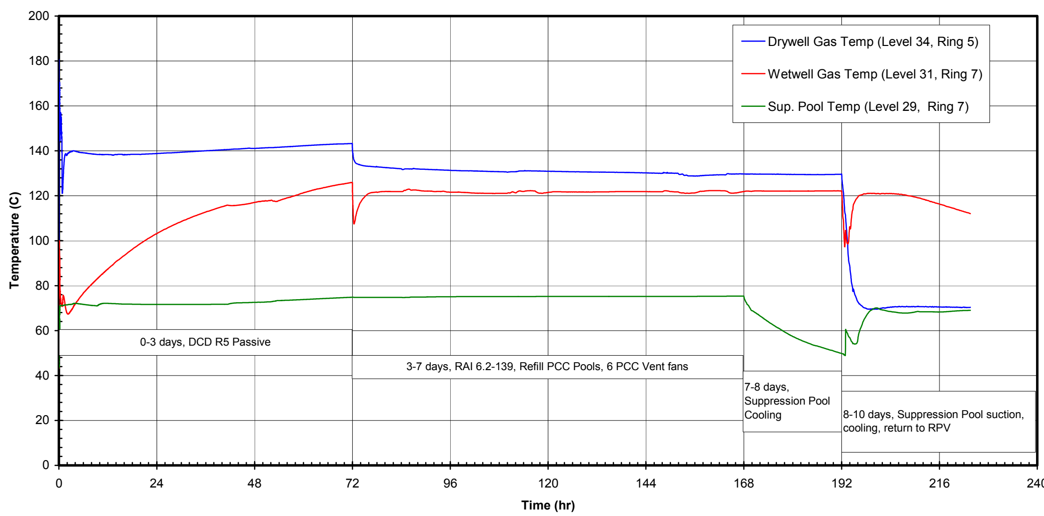


Figure 6.2-14e12. Containment Temperature Response – Post-LOCA Containment Cooling and Recovery

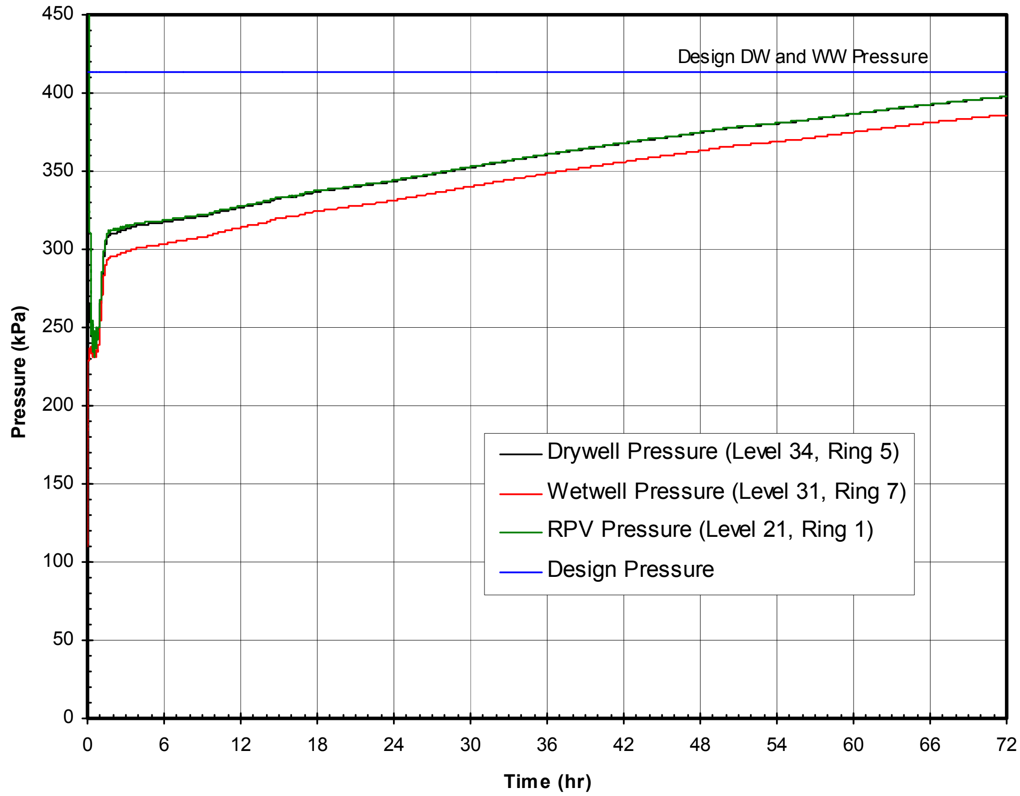


Figure 6.2-14f1. Main Steam Line Break, 1 SRV Failure (Bounding Case) – Containment Pressures (72 hrs)

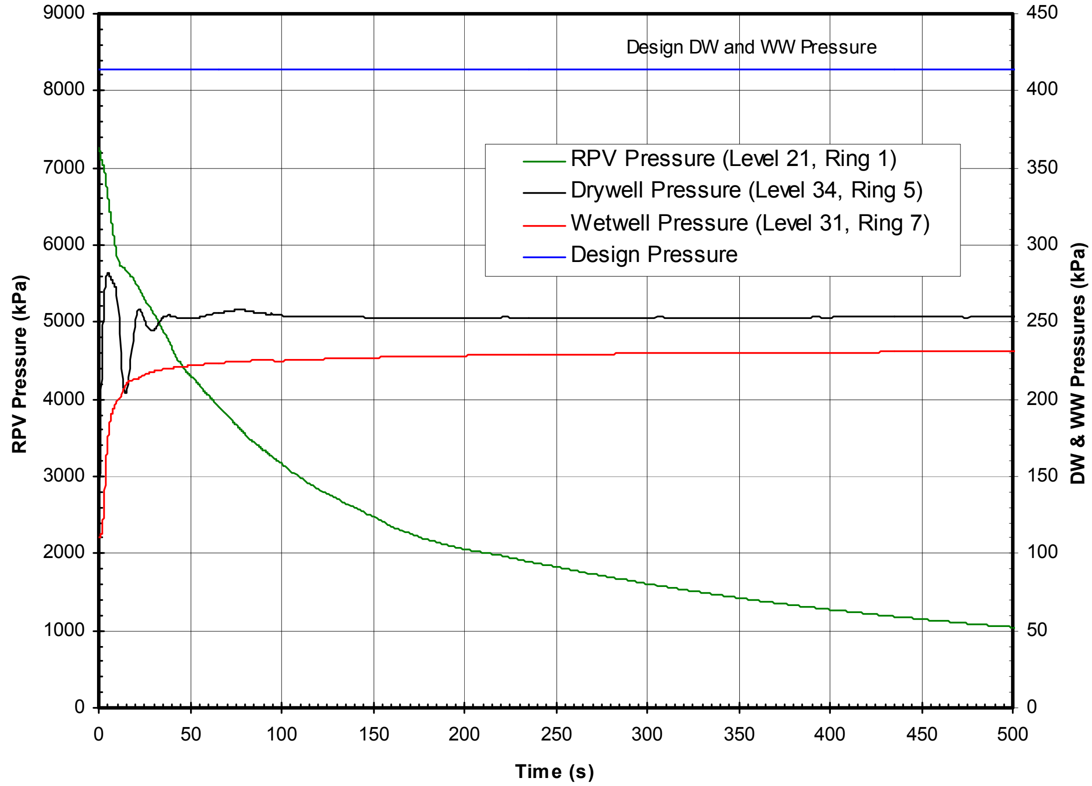


Figure 6.2-14f2. Main Steam Line Break, 1 SRV Failure (Bounding Case) – Containment Pressures (500 s)

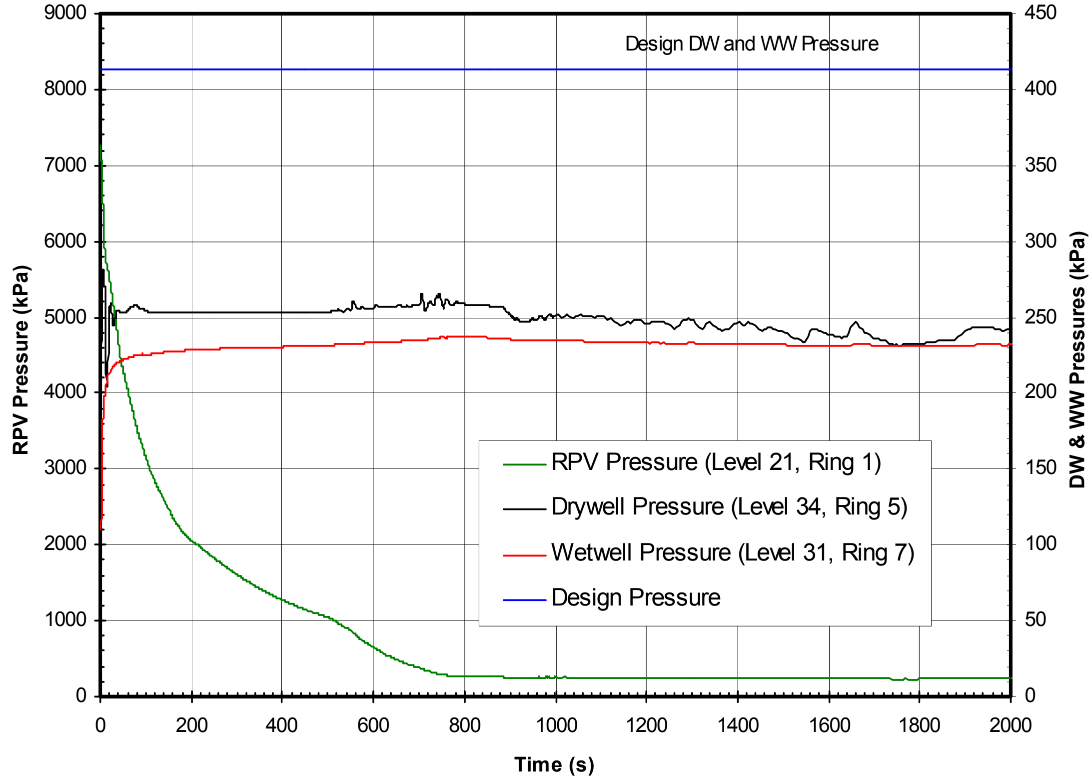


Figure 6.2-14f3. Main Steam Line Break, 1 SRV Failure (Bounding Case) – Containment Pressures (2000 s)

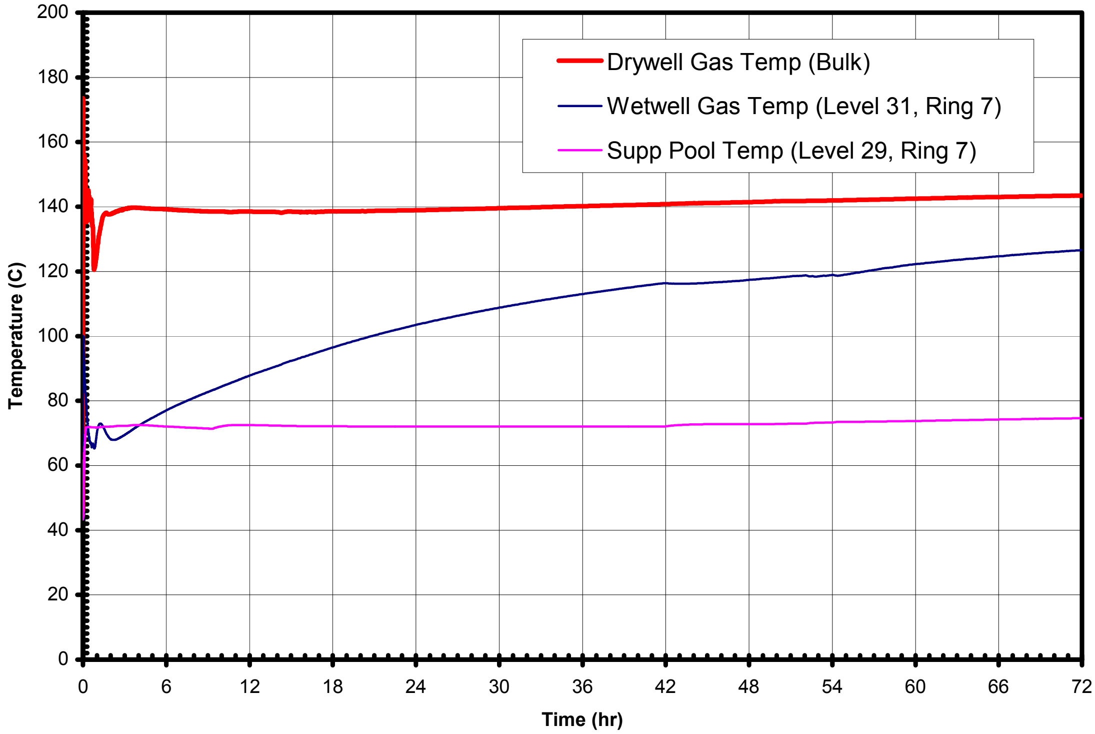


Figure 6.2-14g1. Main Steam Line Break, 1 SRV Failure (Bounding Case) – Containment Temperatures (72 hrs)

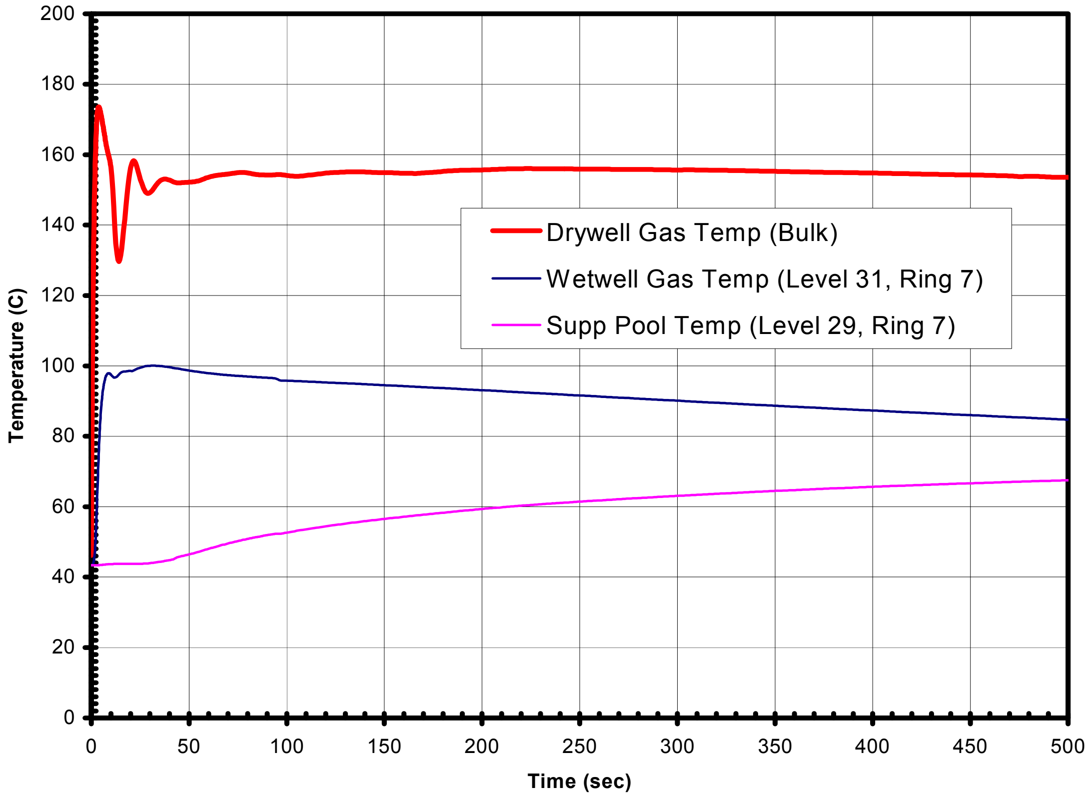


Figure 6.2-14g2. Main Steam Line Break, 1 SRV Failure (Bounding Case) – Containment Temperatures (500 s)

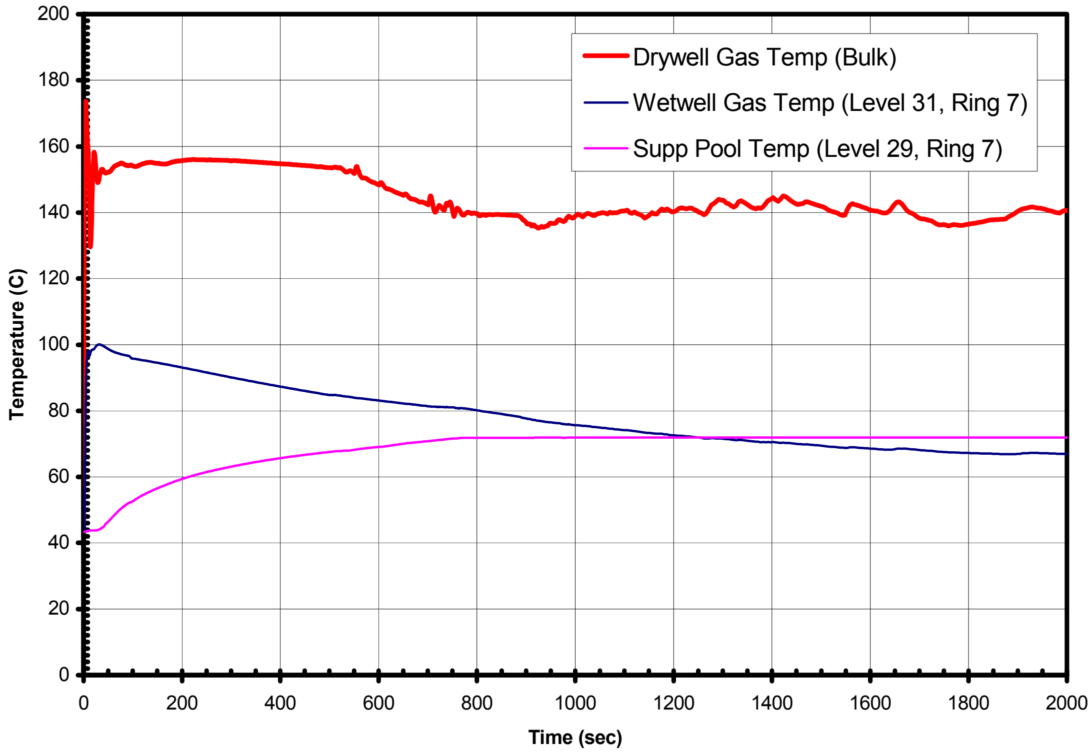


Figure 6.2-14g3. Main Steam Line Break, 1 SRV Failure (Bounding Case) – Containment Temperatures (2000 s)

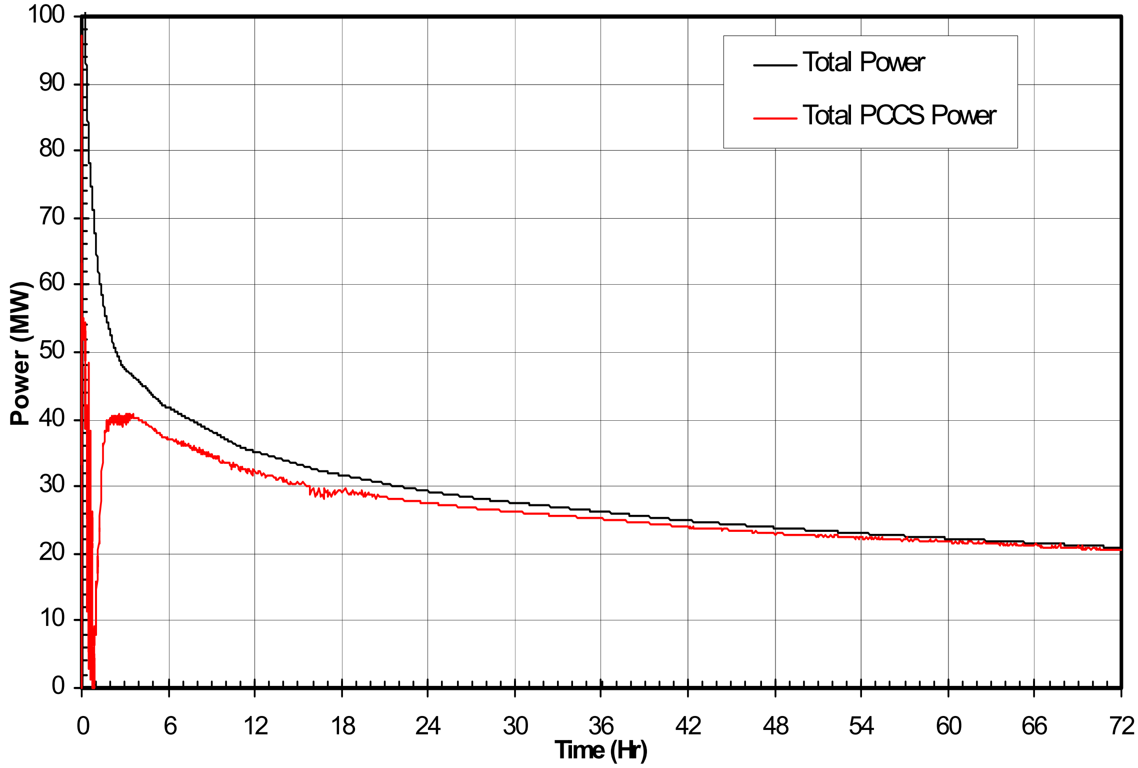


Figure 6.2-14h1. Main Steam Line Break, 1 SRV Failure (Bounding Case) – PCCS Heat Removal versus Decay Heat (72 hrs)

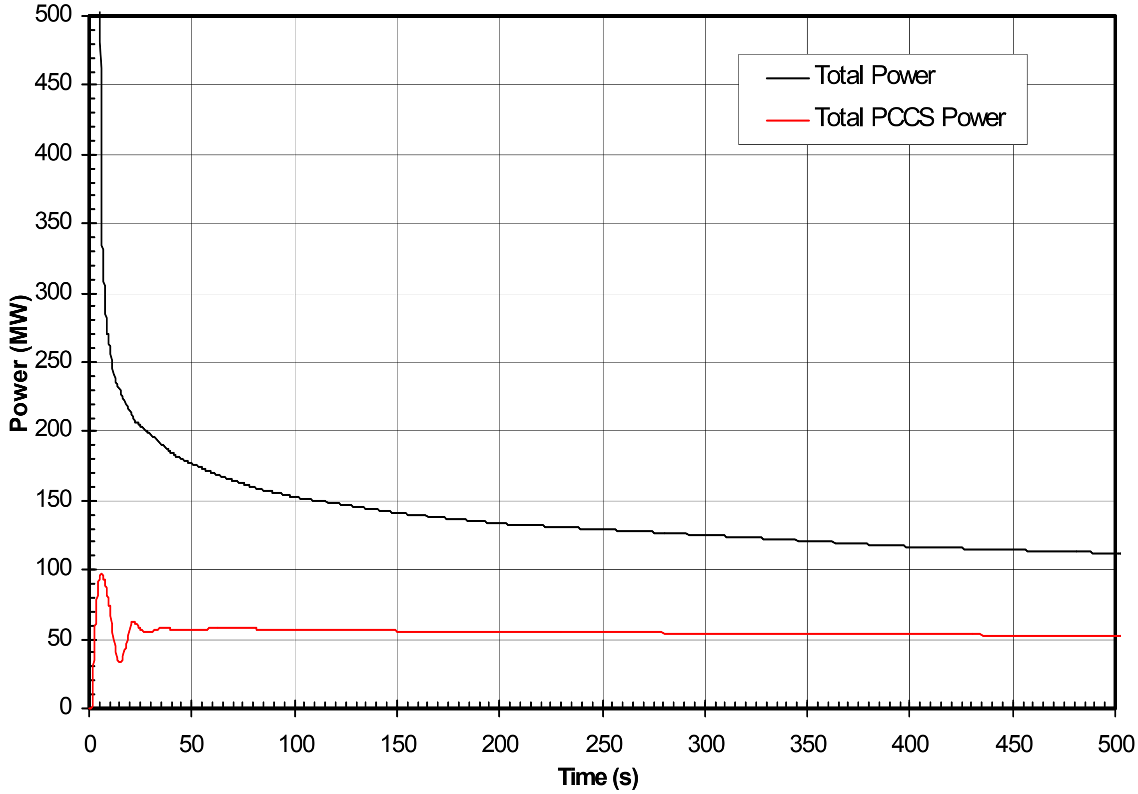


Figure 6.2-14h2. Main Steam Line Break, 1 SRV Failure (Bounding Case) – PCCS Heat Removal versus Decay Heat (500 s)

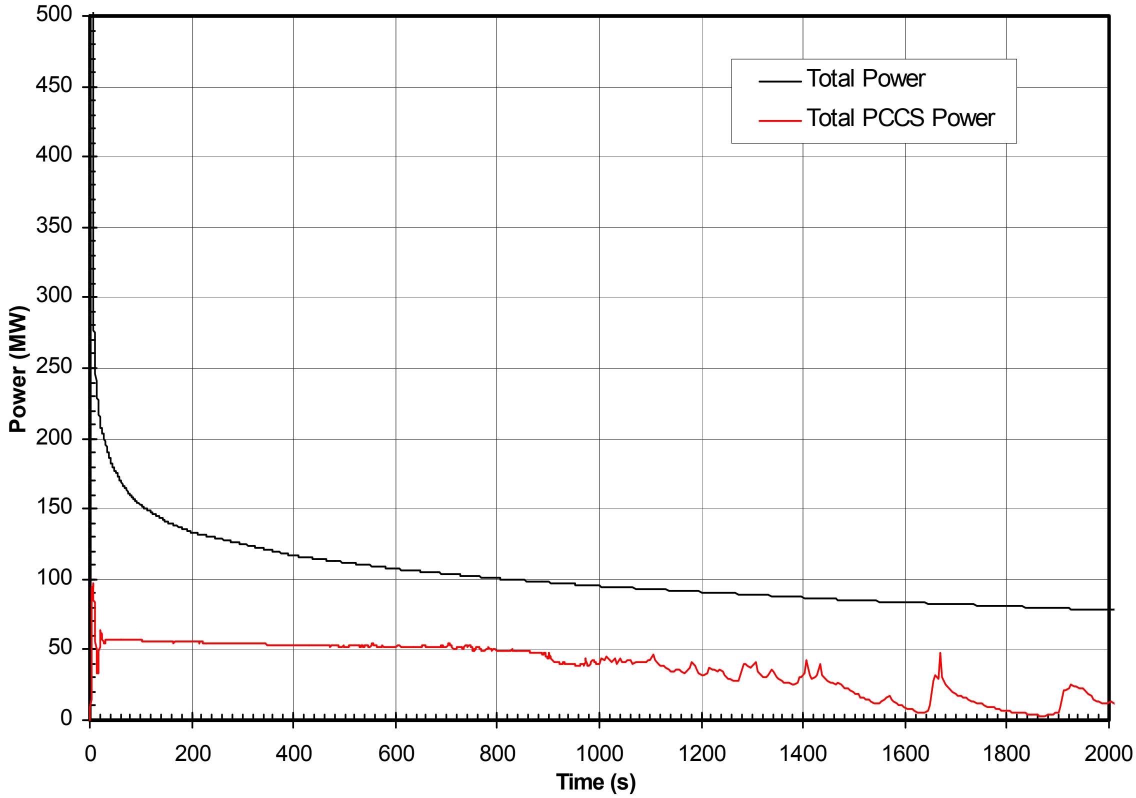


Figure 6.2-14h3. Main Steam Line Break, 1 SRV Failure (Bounding Case) – PCCS Heat Removal versus Decay Heat (2000 s)

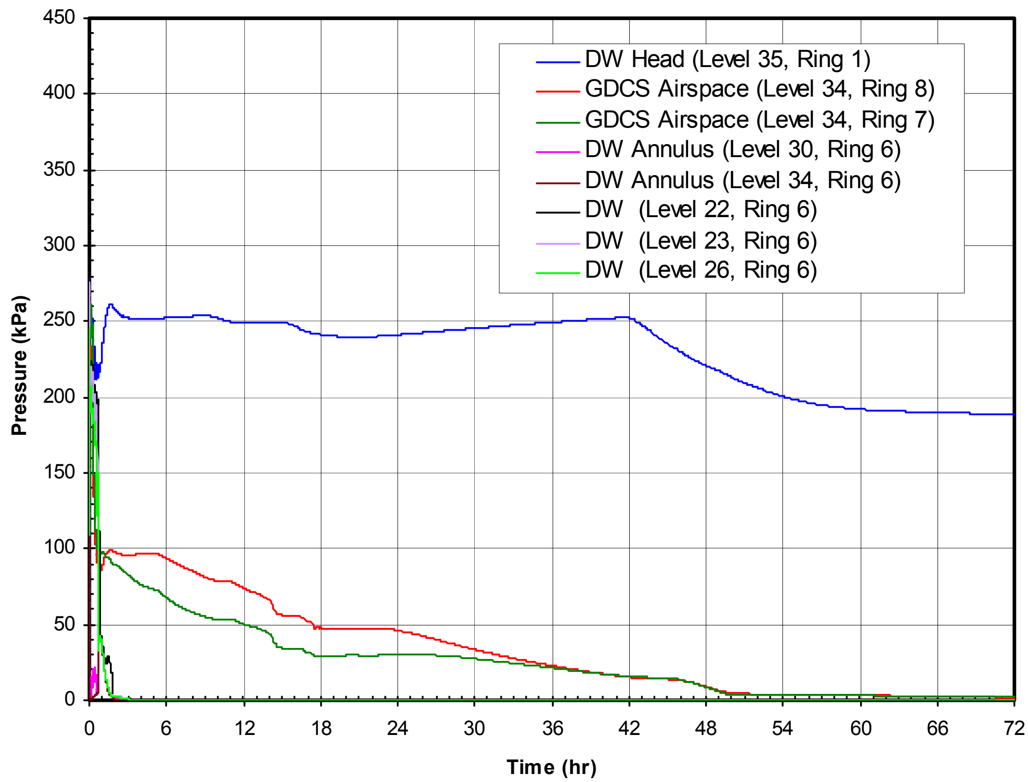


Figure 6.2-14i1. Main Steam Line Break, 1 SRV Failure (Bounding Case) – Drywell and GDCS Noncondensable Gas Pressures (72 hrs)

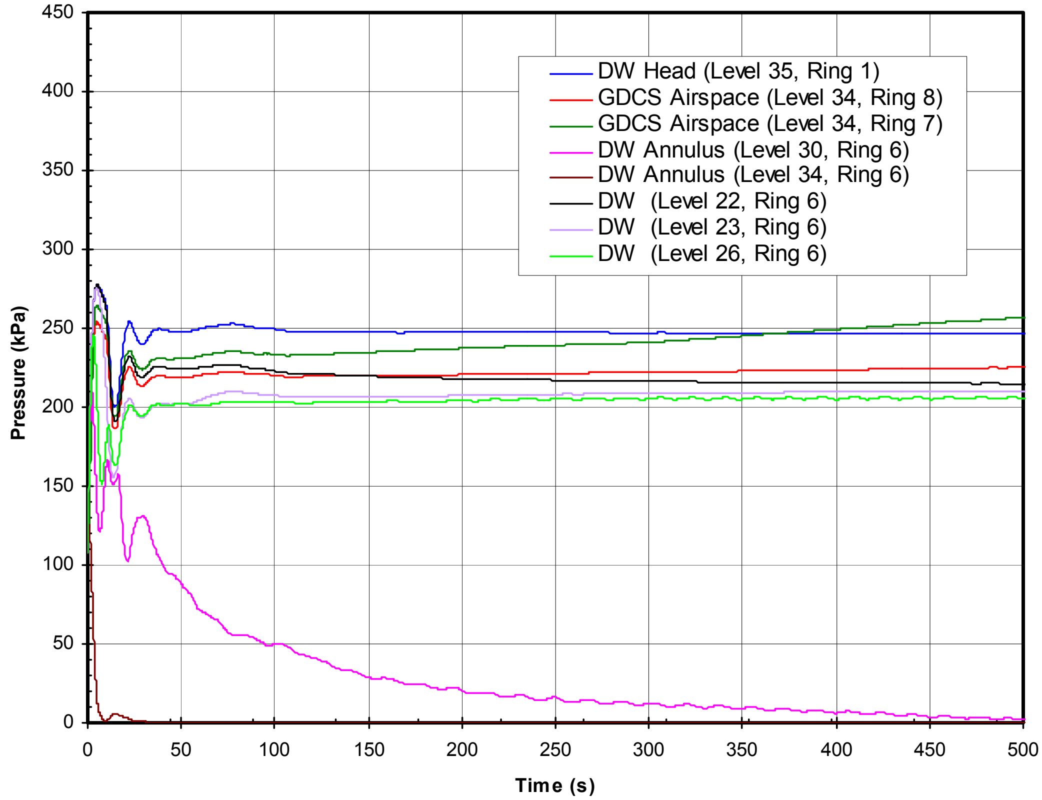


Figure 6.2-14i2. Main Steam Line Break, 1 SRV Failure (Bounding Case) – Drywell and GDCS Noncondensable Gas Pressures (500 s)

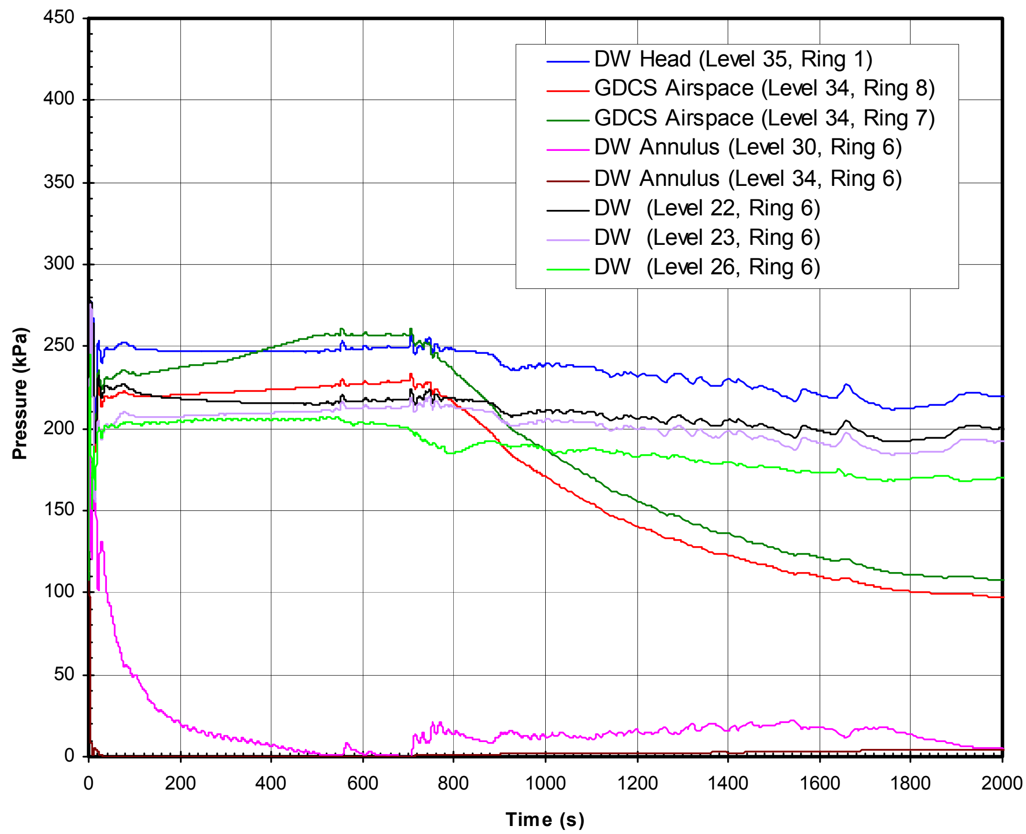


Figure 6.2-14i3. Main Steam Line Break, 1 SRV Failure (Bounding Case) – Drywell and GDCS Noncondensable Gas Pressures (2000 s)

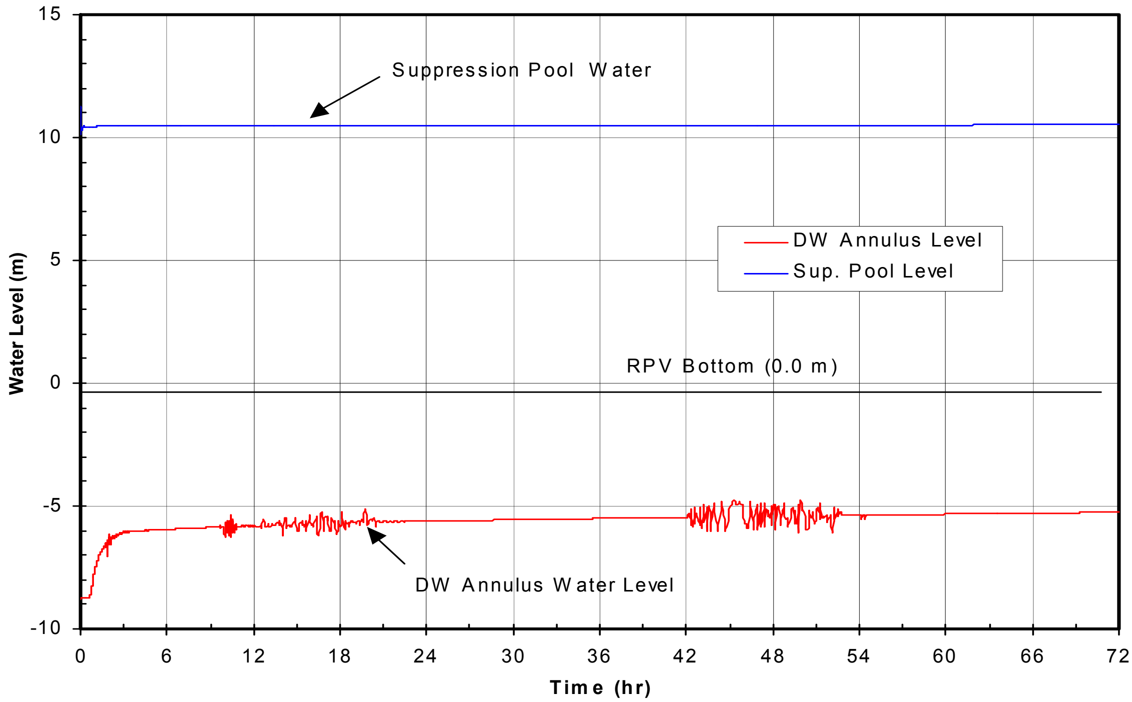


Figure 6.2-14i4. Main Steam Line Break, 1 SRV Failure (Bounding) Drywell Annulus and Suppression Pool Levels (72 hrs)

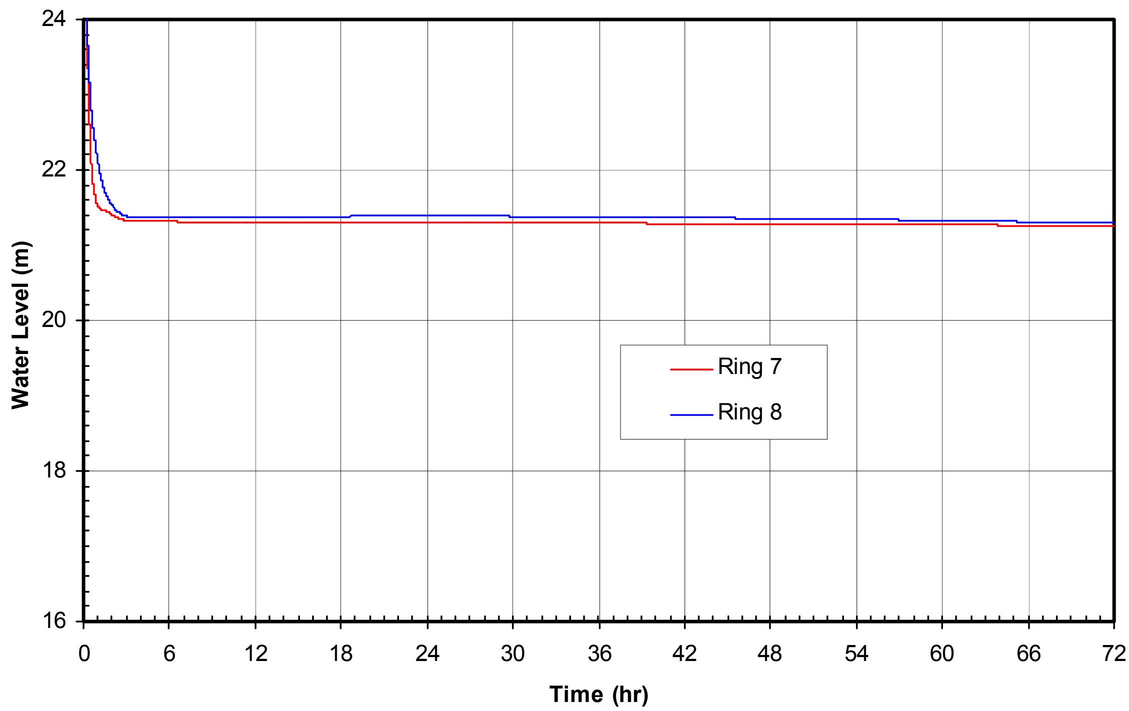


Figure 6.2-14i5. Main Steam Line Break, 1 SRV Failure (Bounding) GDCS Pool Levels (72 hrs)

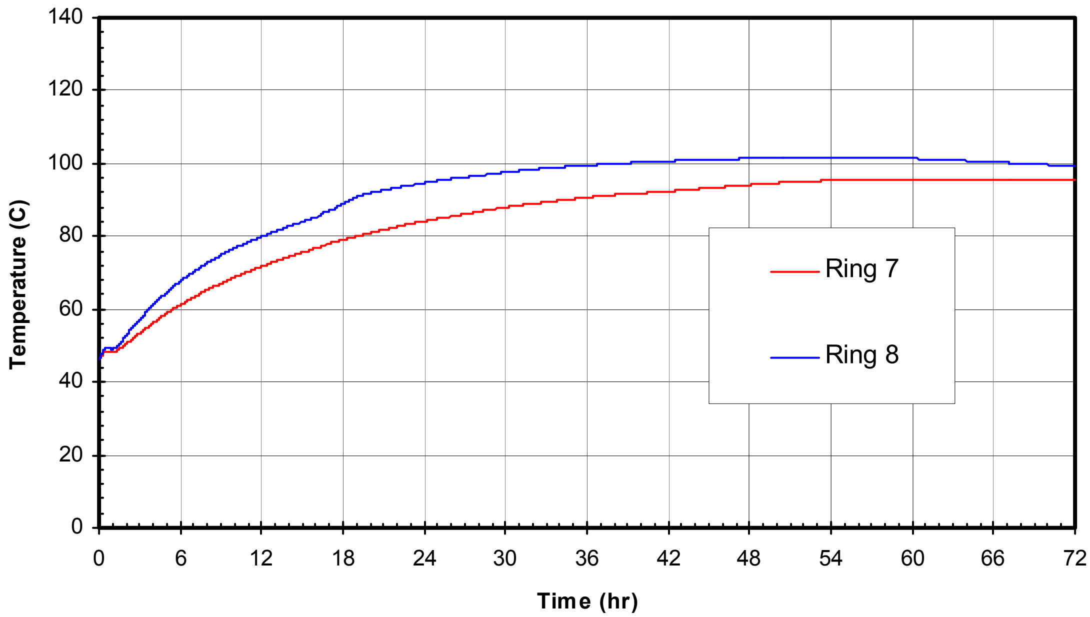


Figure 6.2-14i6. Main Steam Line Break, 1 SRV Failure (Bounding) GDCS Pool Temperature (72 hrs)

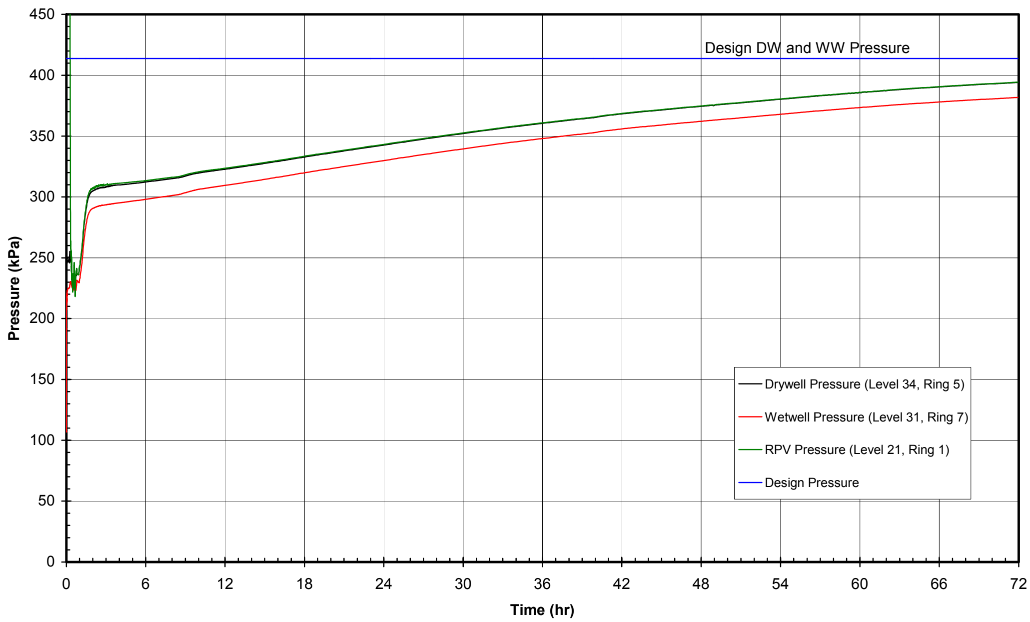


Figure 6.2-14j1. Main Steam Line Break, 1 SRV Failure (Bounding Case, with Offsite Power) – Containment Pressures (72 hrs)

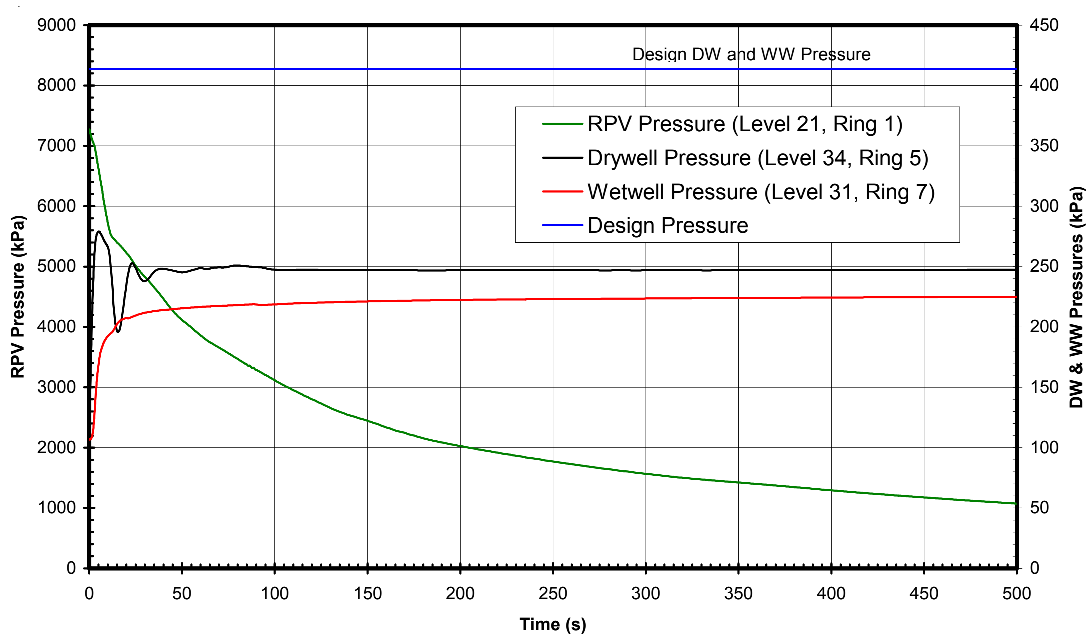


Figure 6.2-14j2. Main Steam Line Break, 1 SRV Failure (Bounding Case, with Offsite Power) – Containment Pressures (500 s)

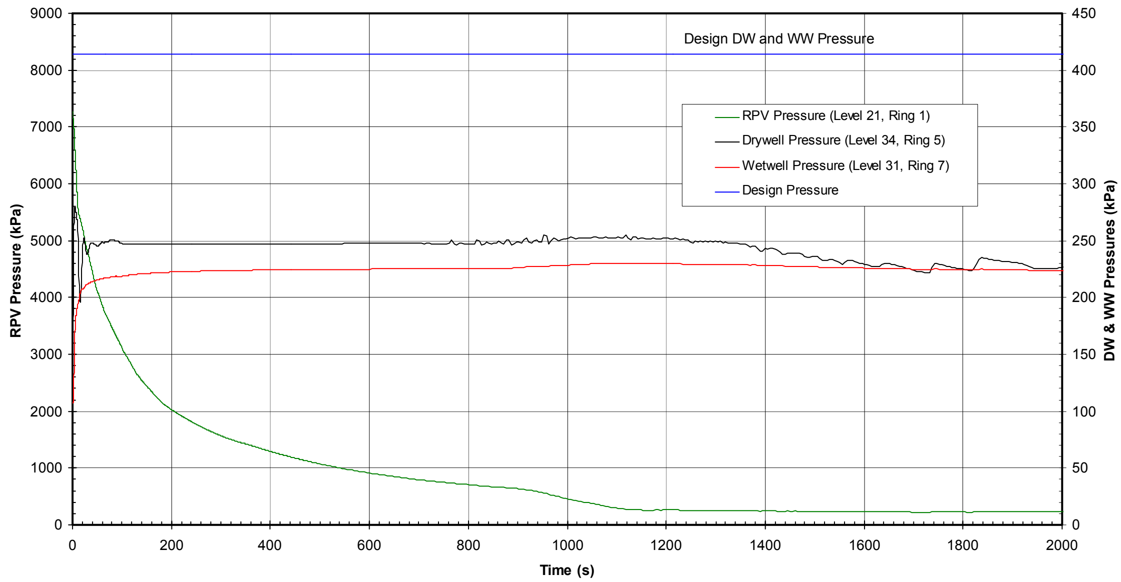


Figure 6.2-14j3. Main Steam Line Break, 1 SRV Failure (Bounding Case, with Offsite Power) – Containment Pressures (2000 s)

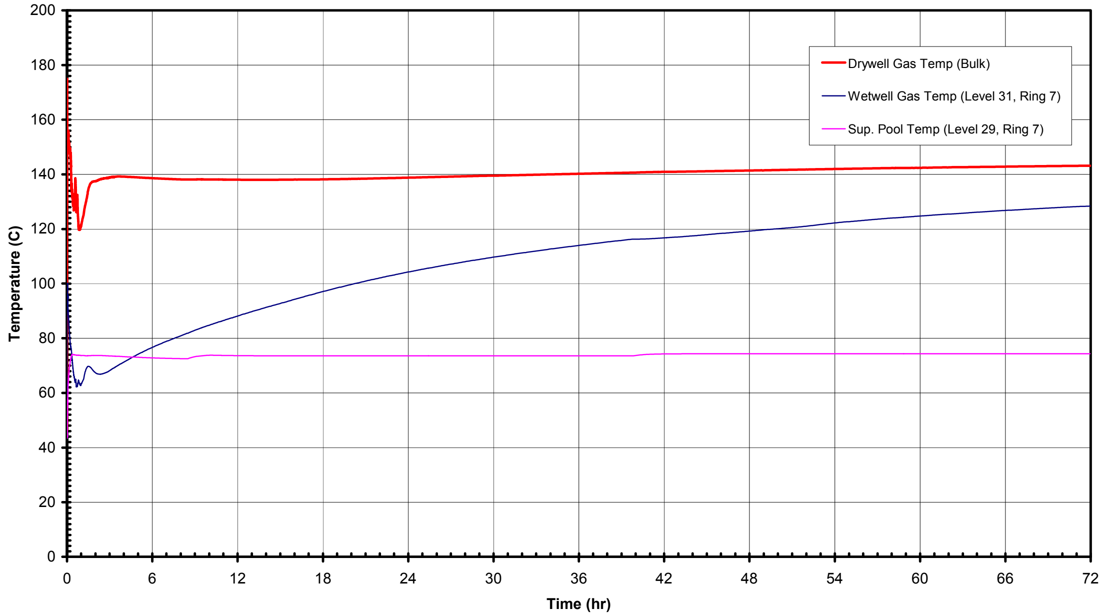


Figure 6.2-14k1. Main Steam Line Break, 1 SRV Failure (Bounding Case, with Offsite Power) – Containment Temperatures (72 hrs)

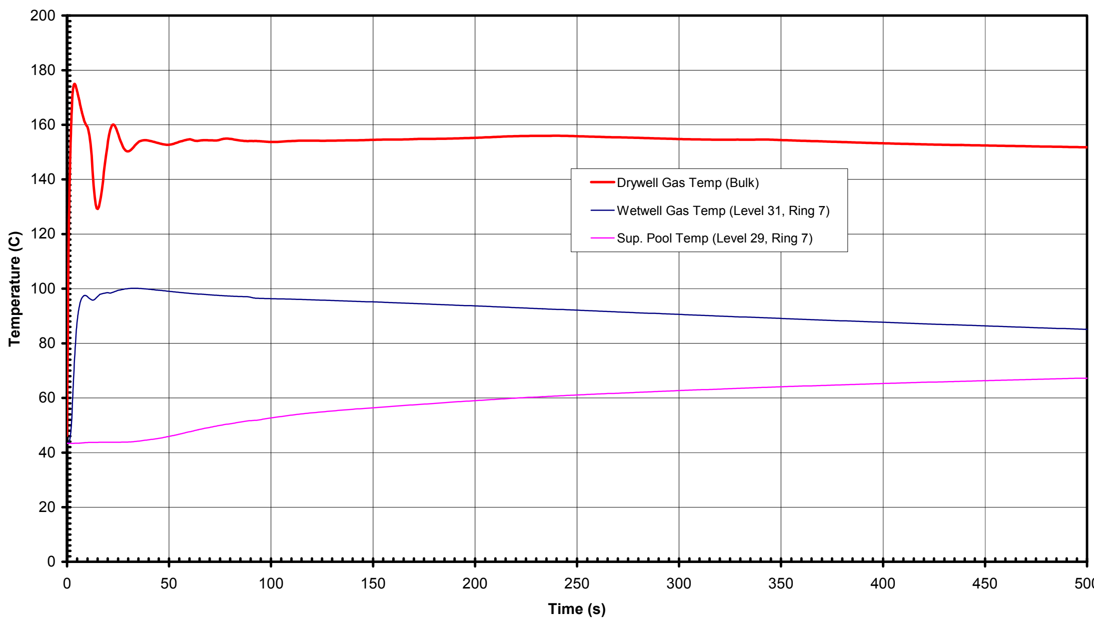


Figure 6.2-14k2. Main Steam Line Break, 1 SRV Failure (Bounding Case , with Offsite Power) – Containment Temperatures (500 s)

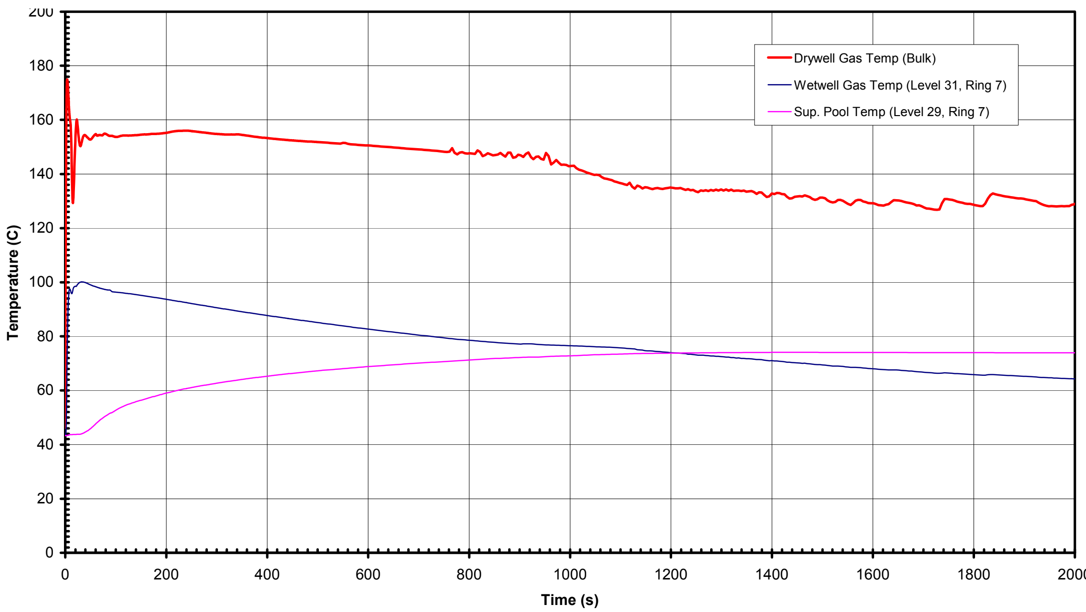


Figure 6.2-14k3. Main Steam Line Break, 1 SRV Failure (Bounding Case, with Offsite Power) – Containment Temperatures (2000 s)

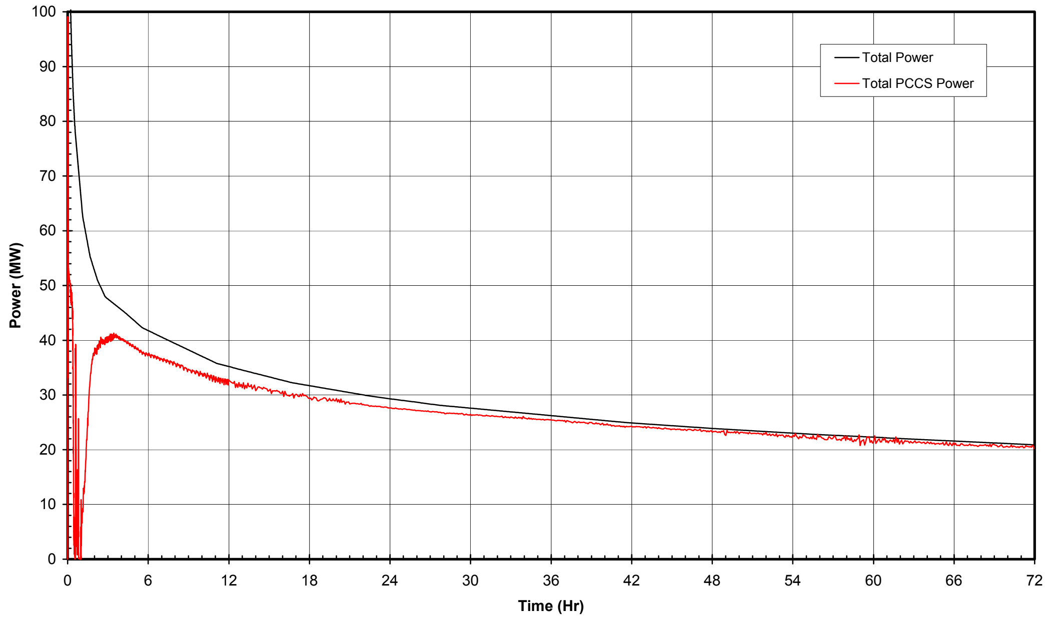


Figure 6.2-14l1. Main Steam Line Break, 1 SRV Failure (Bounding Case, with Offsite Power) – PCCS Heat Removal versus Decay Heat (72 hrs)

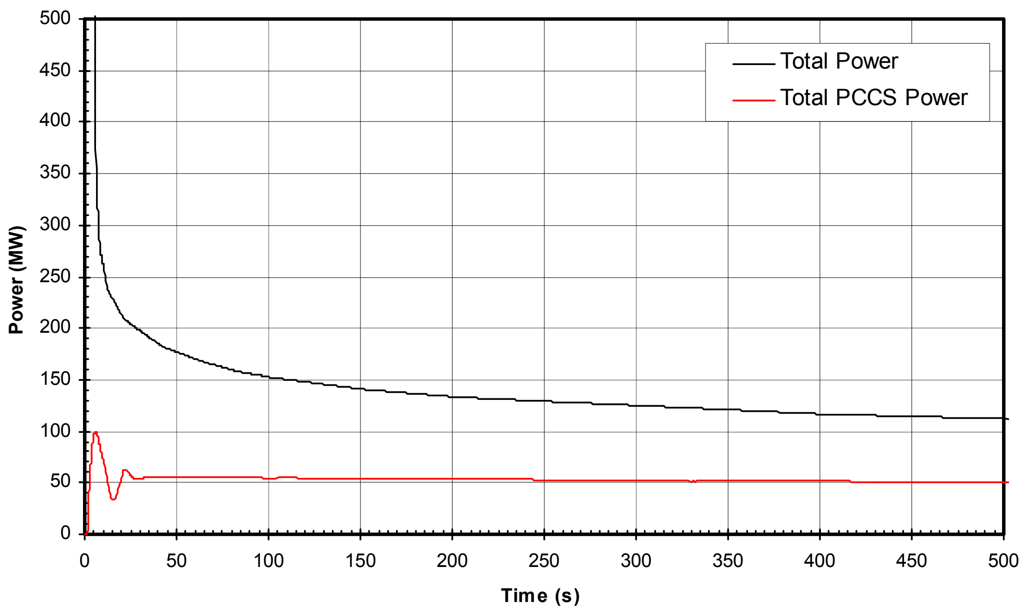


Figure6.2-14l2. Main Steam Line Break, 1 SRV Failure (Bounding Case, with Offsite Power) – PCCS Heat Removal versus Decay Heat (500 s)

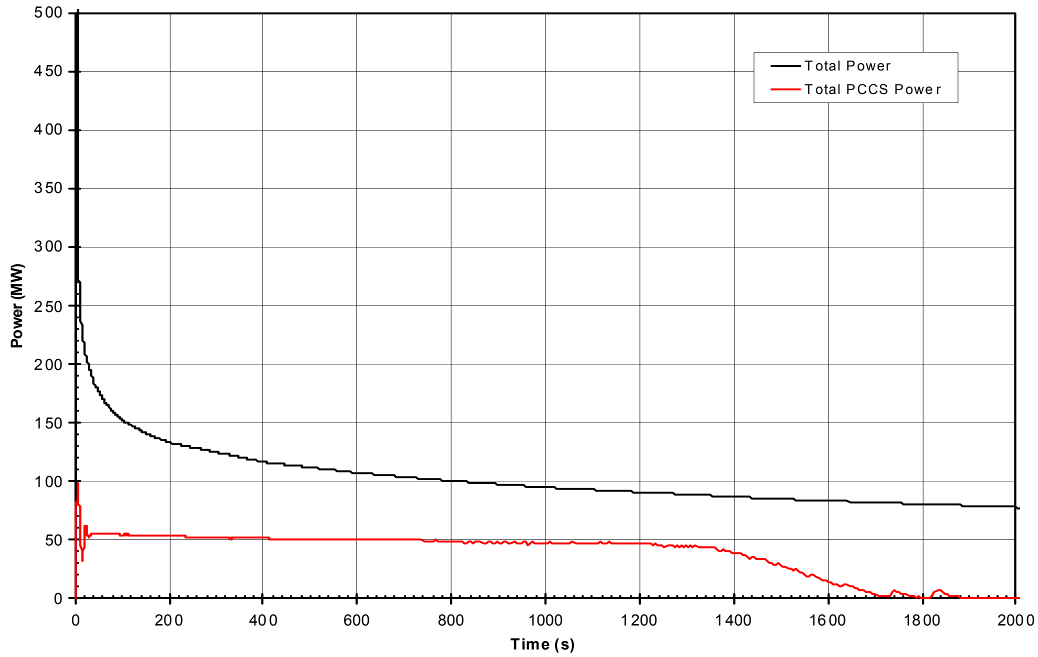


Figure6.2-14l3. Main Steam Line Break, 1 SRV Failure (Bounding Case, with Offsite Power) – PCCS Heat Removal versus Decay Heat (2000 s)

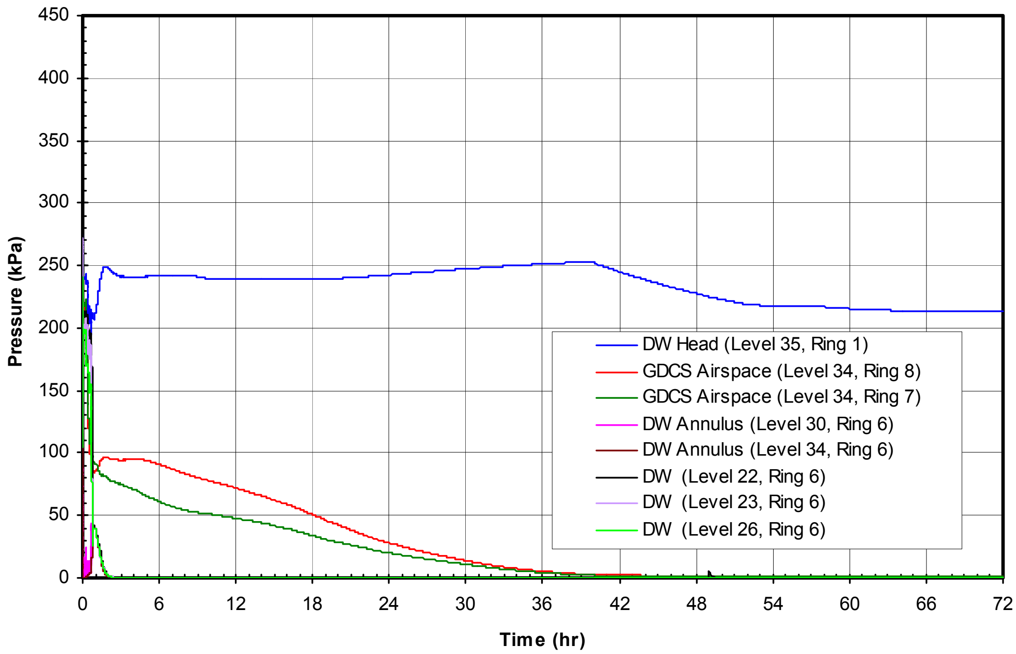


Figure 6.2-14m1. Main Steam Line Break, 1 SRV Failure (Bounding Case, with Offsite Power) – Drywell and GDCS Noncondensable Gas Pressure (72 hrs)

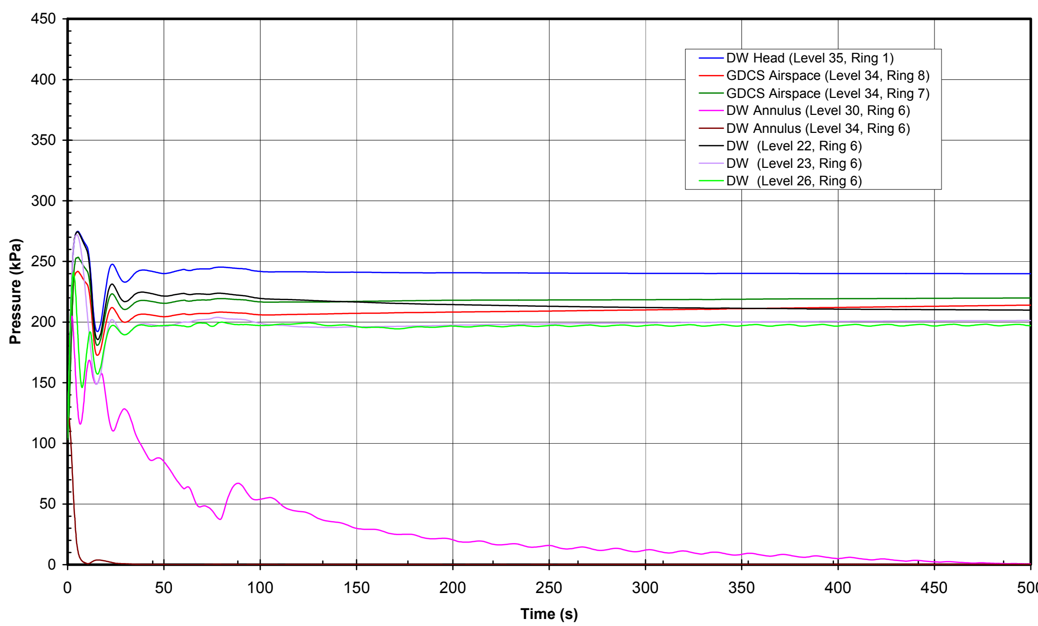


Figure 6.2-14m2. Main Steam Line Break, 1 SRV Failure (Bounding Case, with Offsite Power) – Drywell and GDCS Noncondensable Gas Pressures (500 s)

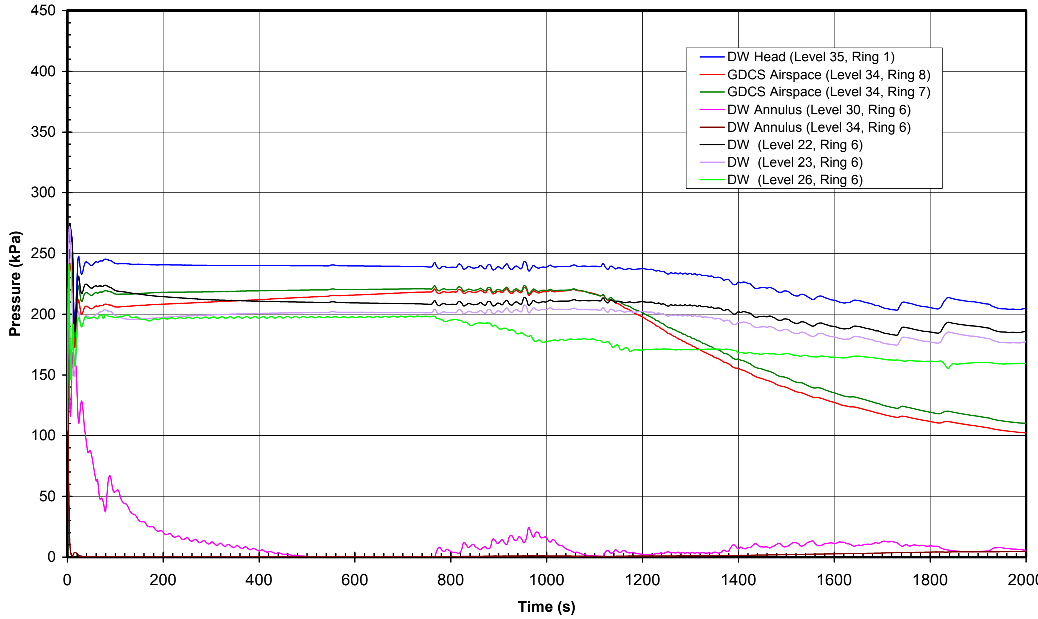


Figure 6.2-14m3. Main Steam Line Break, 1 SRV Failure (Bounding Case, with Offsite Power) – Drywell and GDCS Noncondensable Gas Pressures (2000 s)

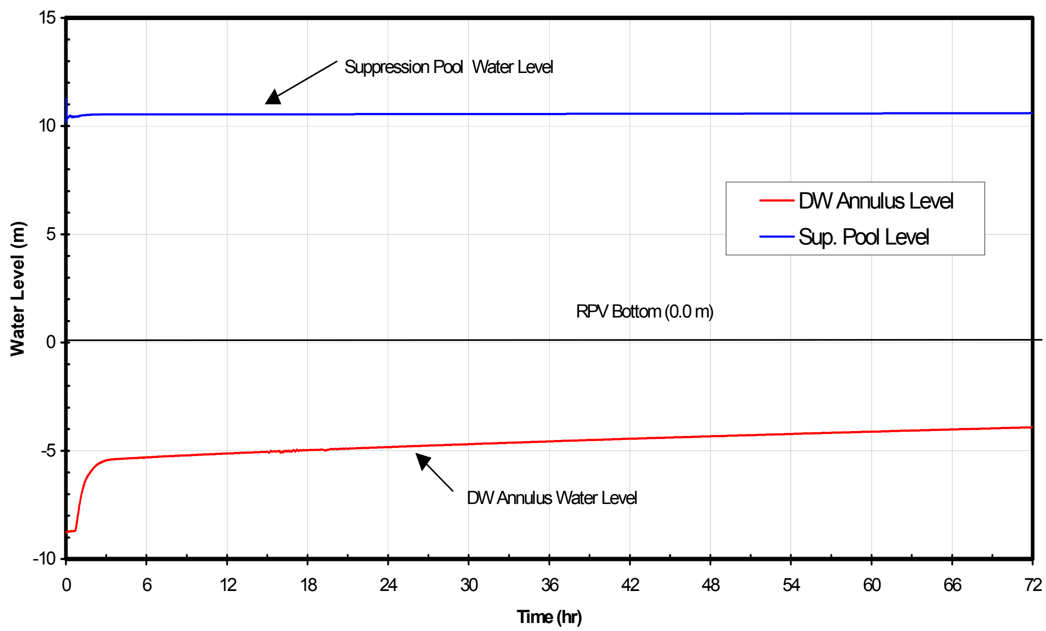


Figure 6.2-14m4. Main Steam Line Break, 1 SRV Failure (Bounding Case, with Offsite Power) Drywell Annulus and Suppression Pool Levels (72 hrs)

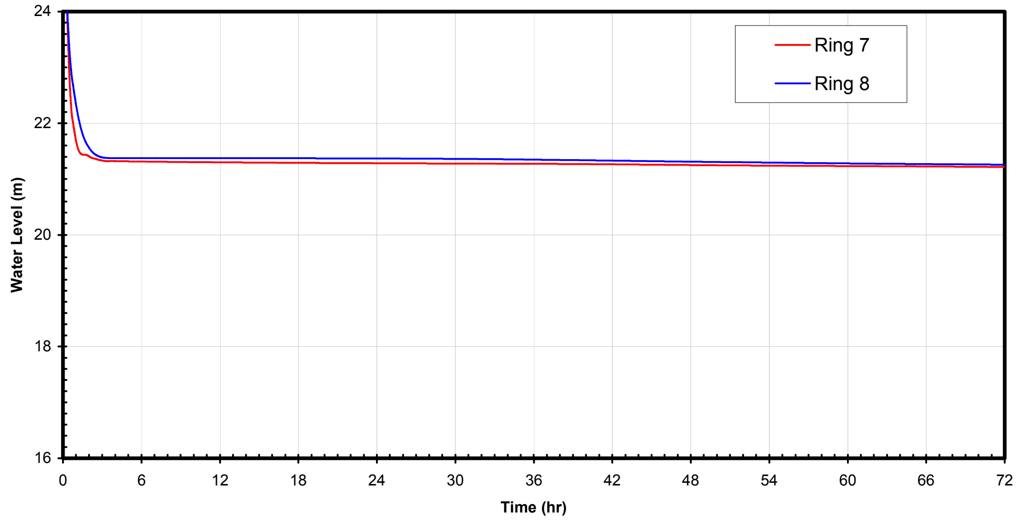


Figure 6.2-14m5. Main Steam Line Break, 1 SRV Failure (Bounding Case, with Offsite Power) GDCS Pool Levels (72 hrs)

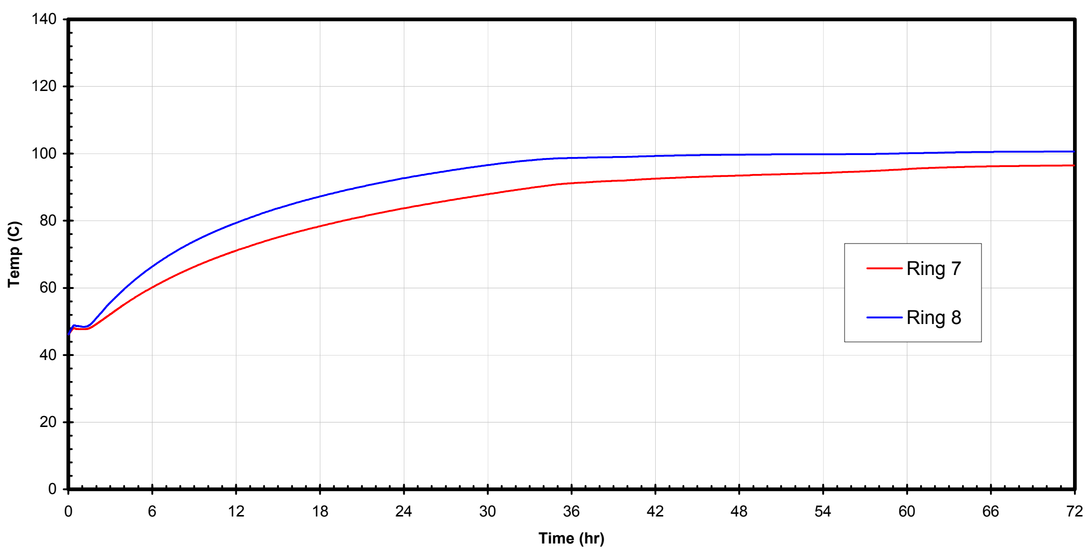


Figure 6.2-14m6. Main Steam Line Break, 1 SRV Failure (Bounding Case, with Offsite Power) GDCS Pool Temperature (72 hrs)

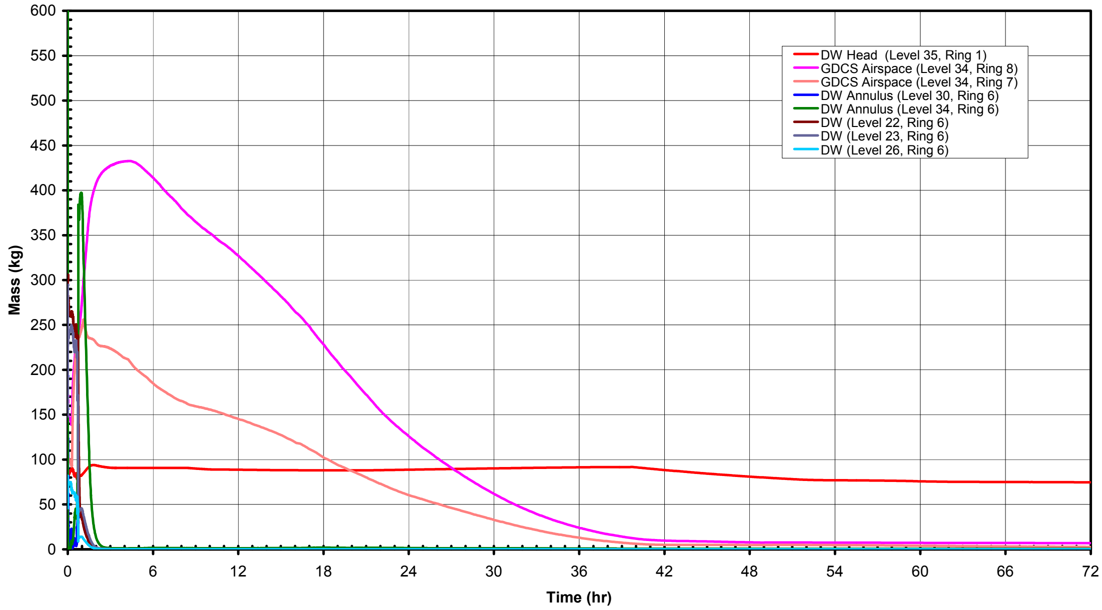


Figure 6.2-14n1. Main Steam Line Break, 1 SRV Failure (Bounding Case, with Offsite Power) –Drywell and GDCS Noncondensable Gas Mass (72 hrs)

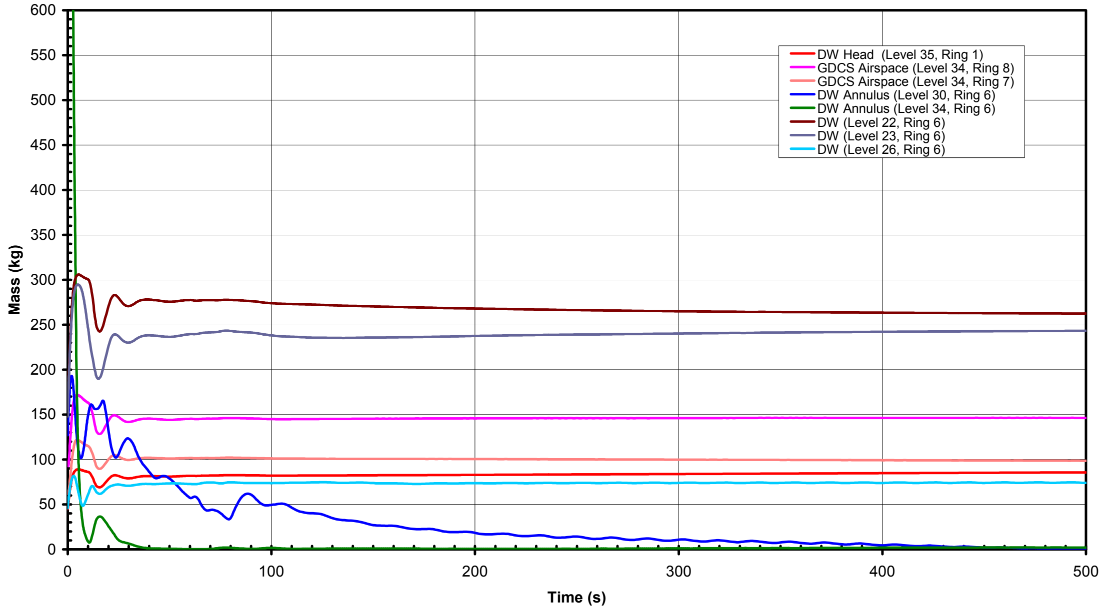


Figure 6.2-14n2. Main Steam Line Break, 1 SRV Failure (Bounding Case, with Offsite Power) –Drywell and GDCS Noncondensable Gas Mass (500 s)

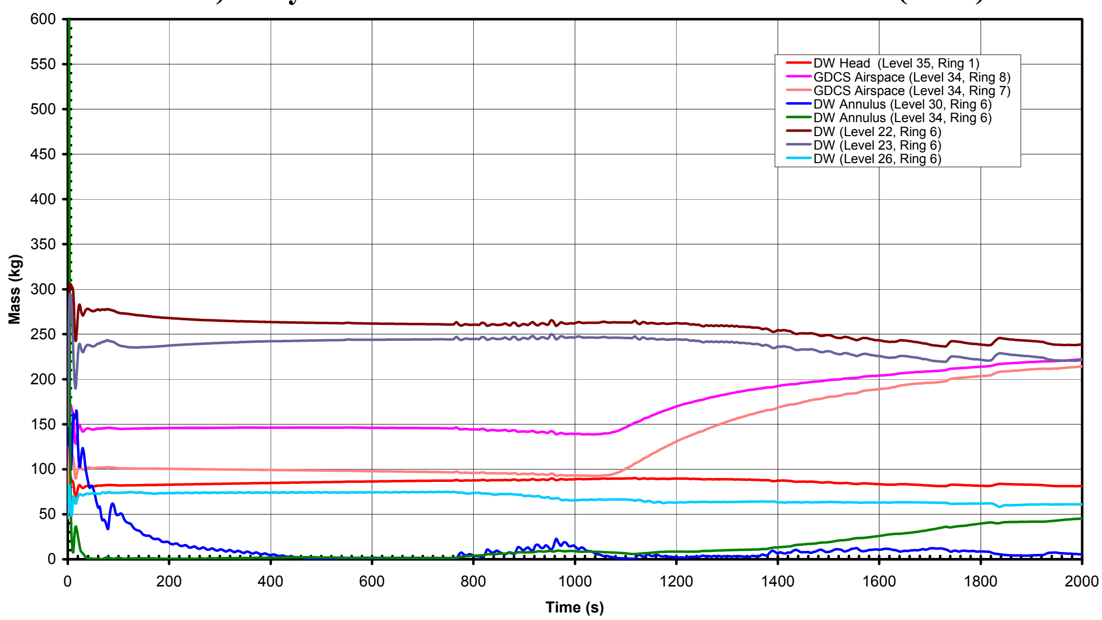


Figure 6.2-14n3. Main Steam Line Break, 1 SRV Failure (Bounding Case, with Offsite Power) –Drywell and GDCS Noncondensable Gas Mass (2000 s)

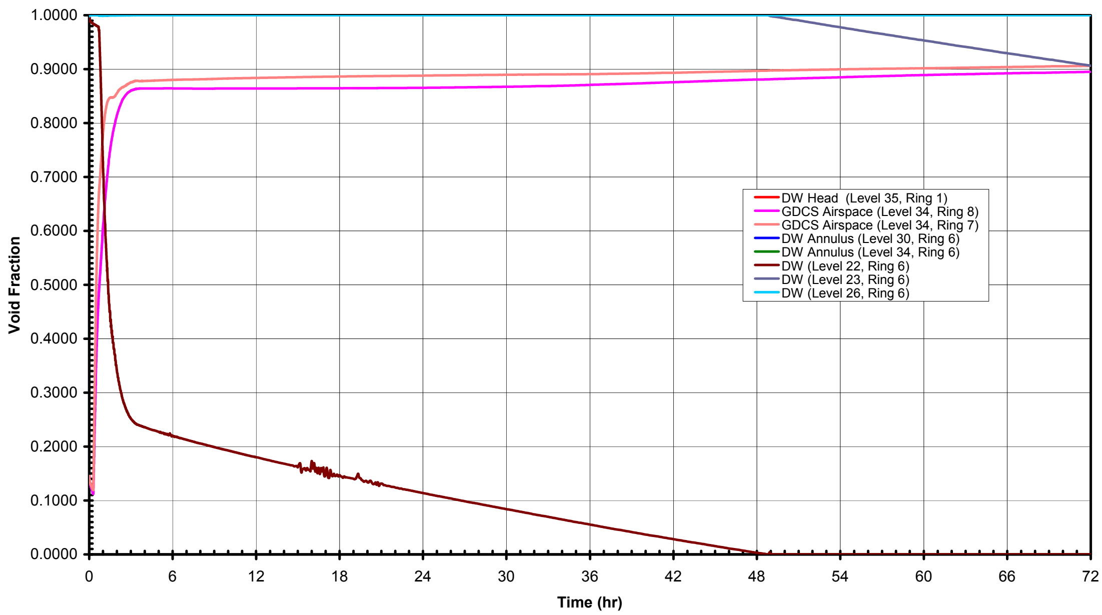


Figure 6.2-14o1. Main Steam Line Break, 1 SRV Failure (Bounding Case, with Offsite Power) –Drywell and GDCS Pool Void Fraction (72 hrs)

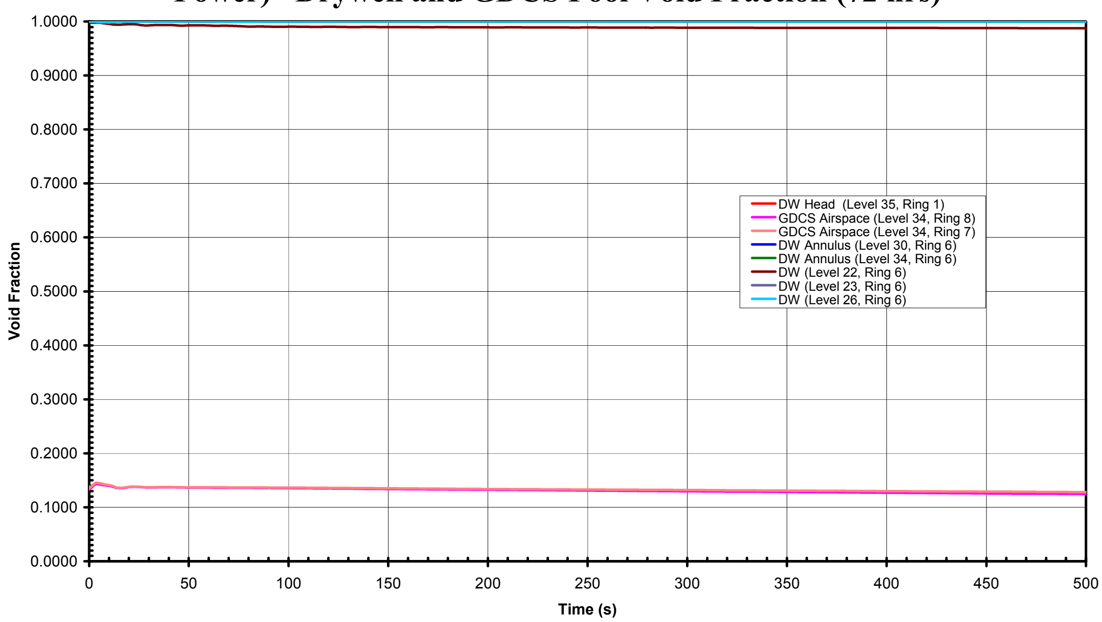


Figure 6.2-14o2. Main Steam Line Break, 1 SRV Failure (Bounding Case, with Offsite Power) –Drywell and GDCS Pool Void Fraction (500 s)

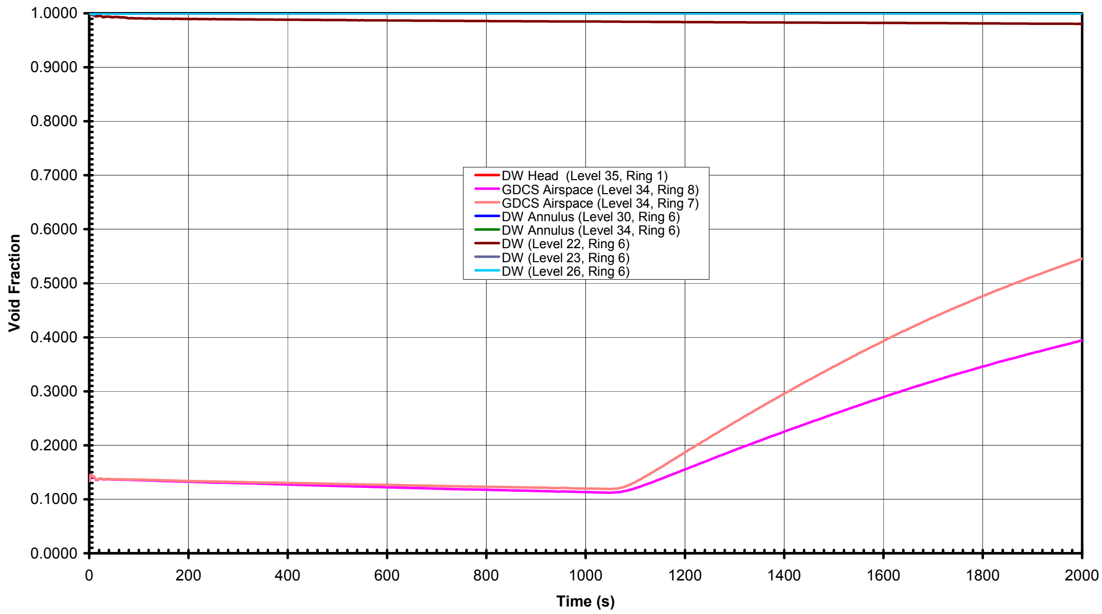


Figure 6.2-14o3. Main Steam Line Break, 1 SRV Failure (Bounding Case, with Offsite Power) –Drywell and GDCS Pool Void Fraction (2000 s)

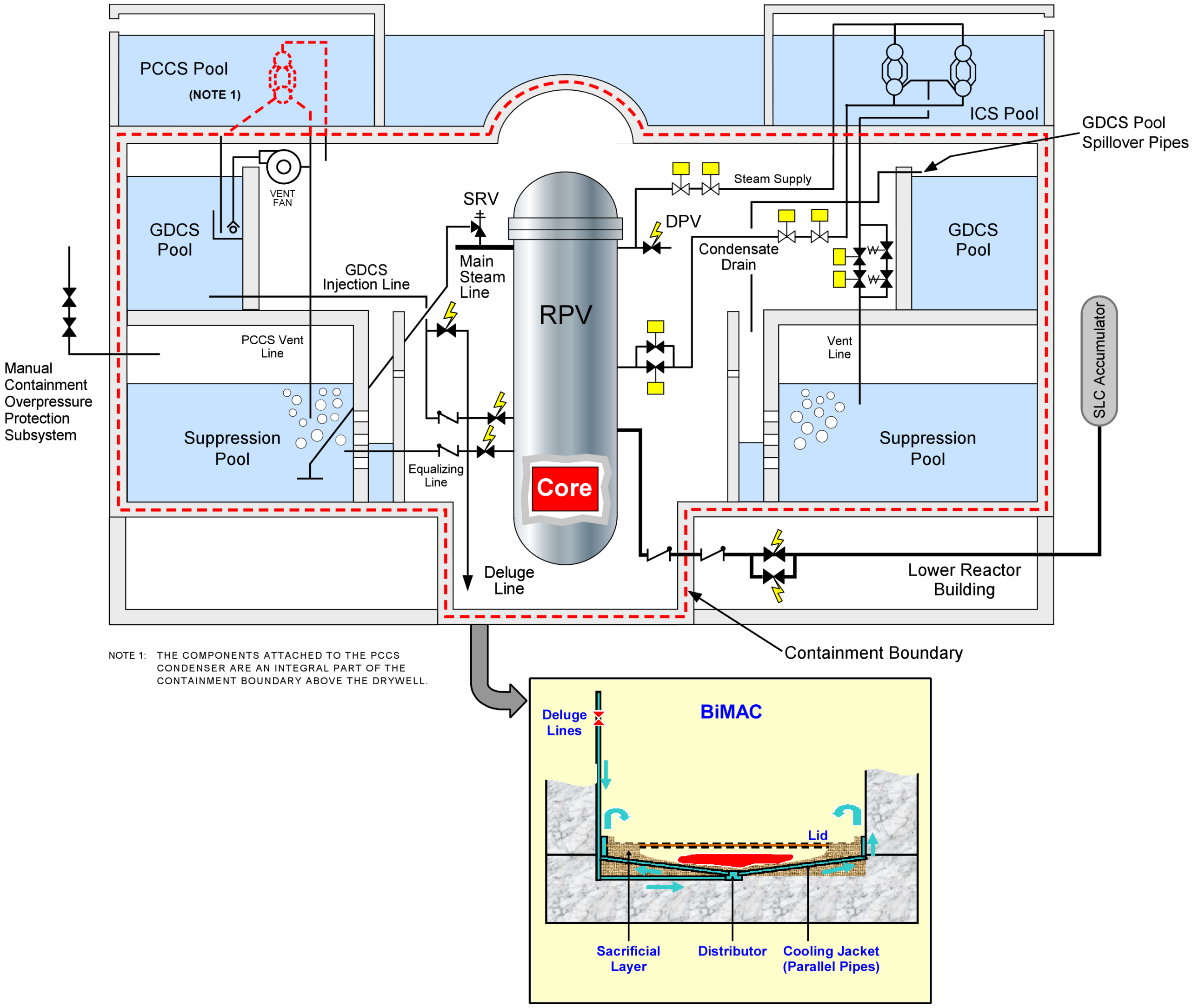


Figure 6.2-15. Summary of Severe Accident Design Features

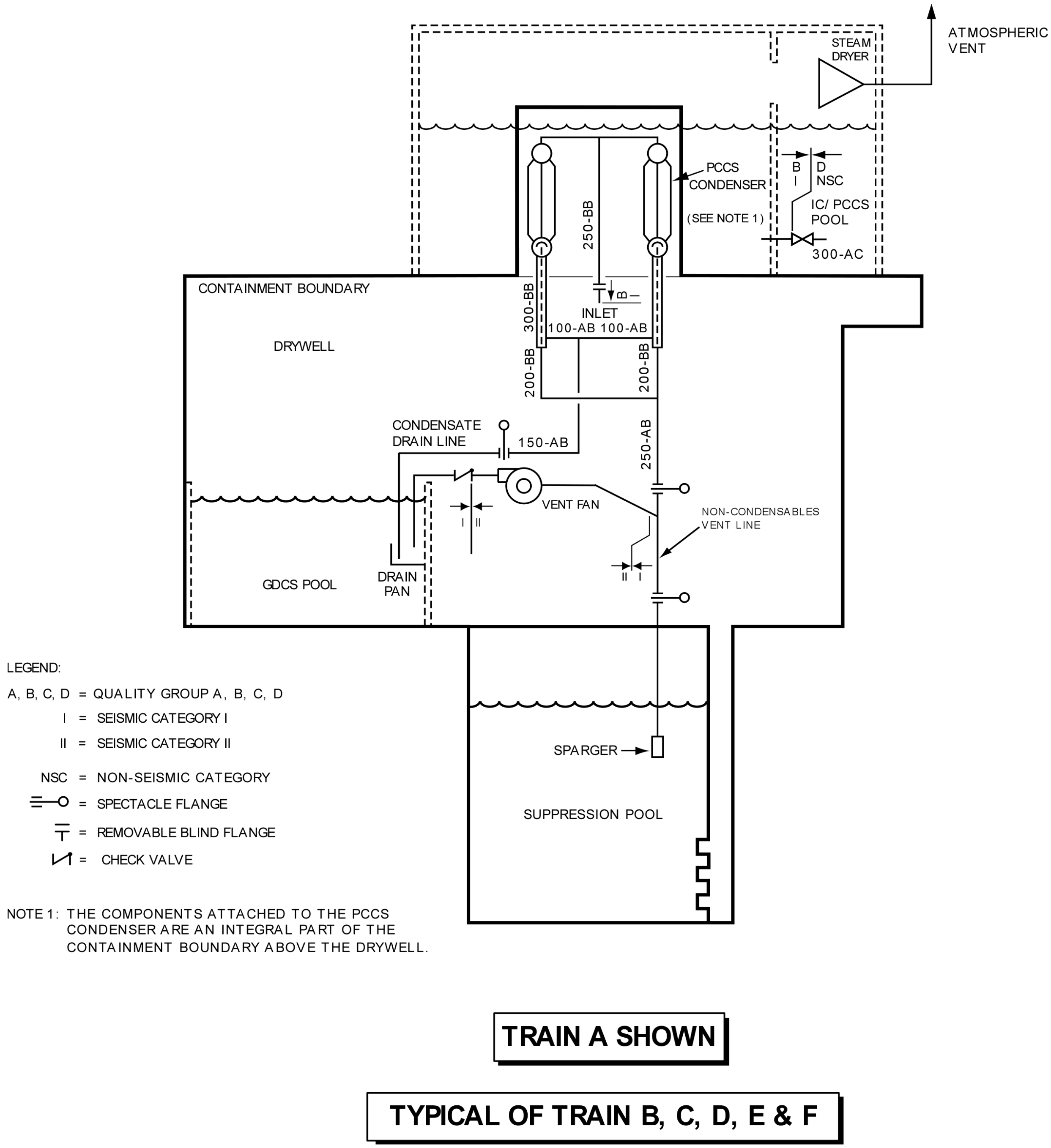


Figure 6.2-16. PCCS Schematic Diagram

PCC – Passive Containment Cooling System

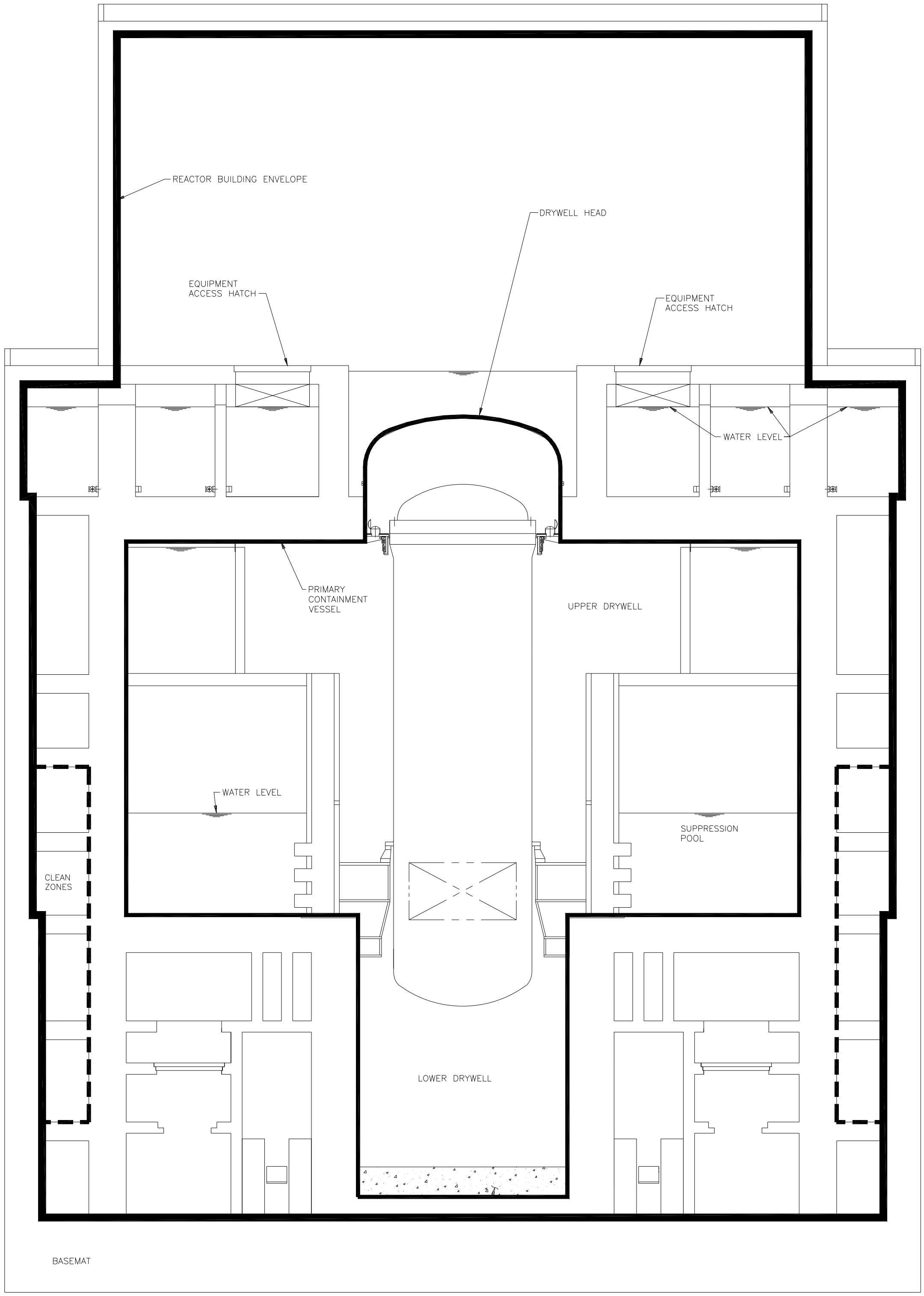


Figure 6.2-17. Reactor Building Envelope

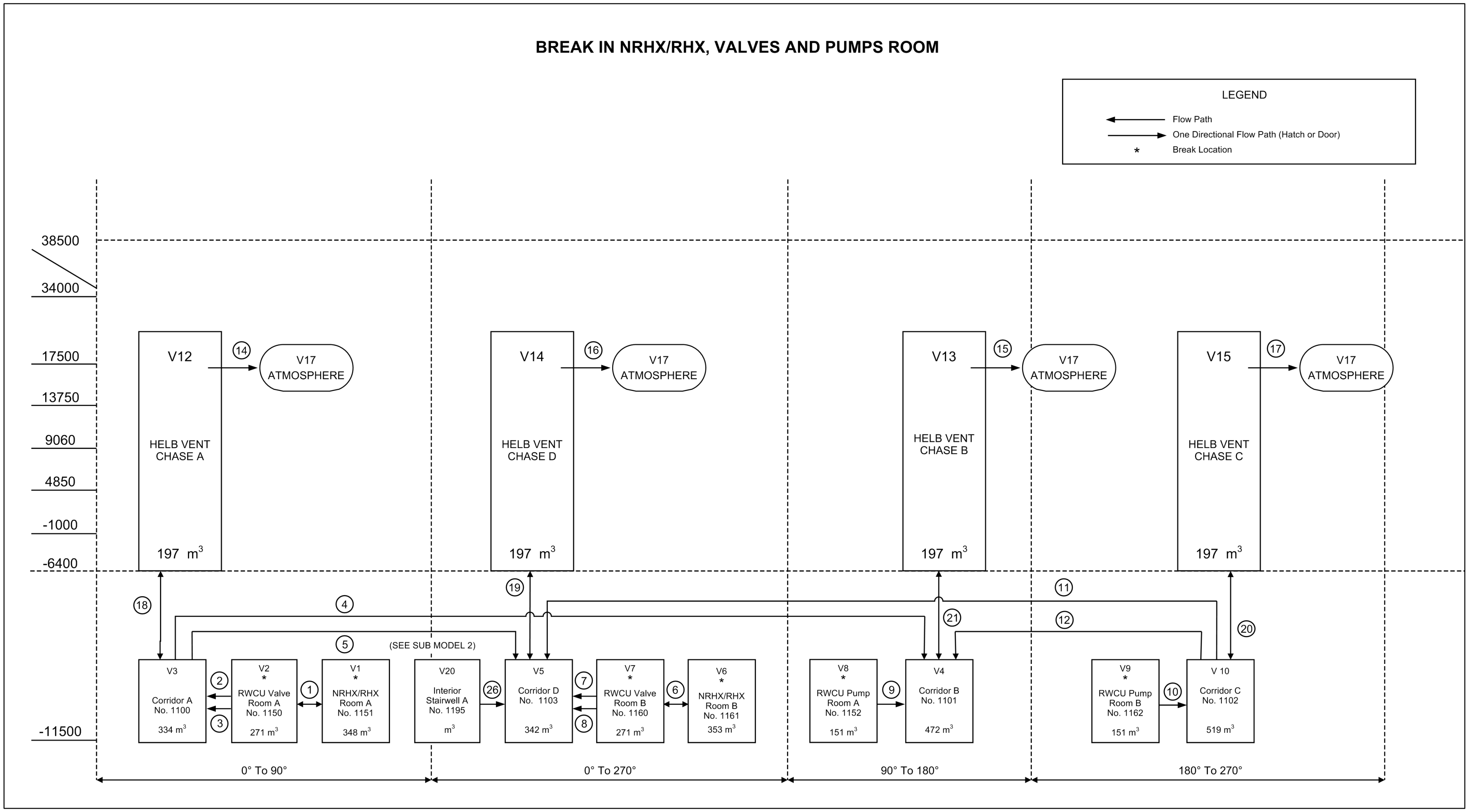


Figure 6.2-18. RWCU System Compartment Pressurization Analysis

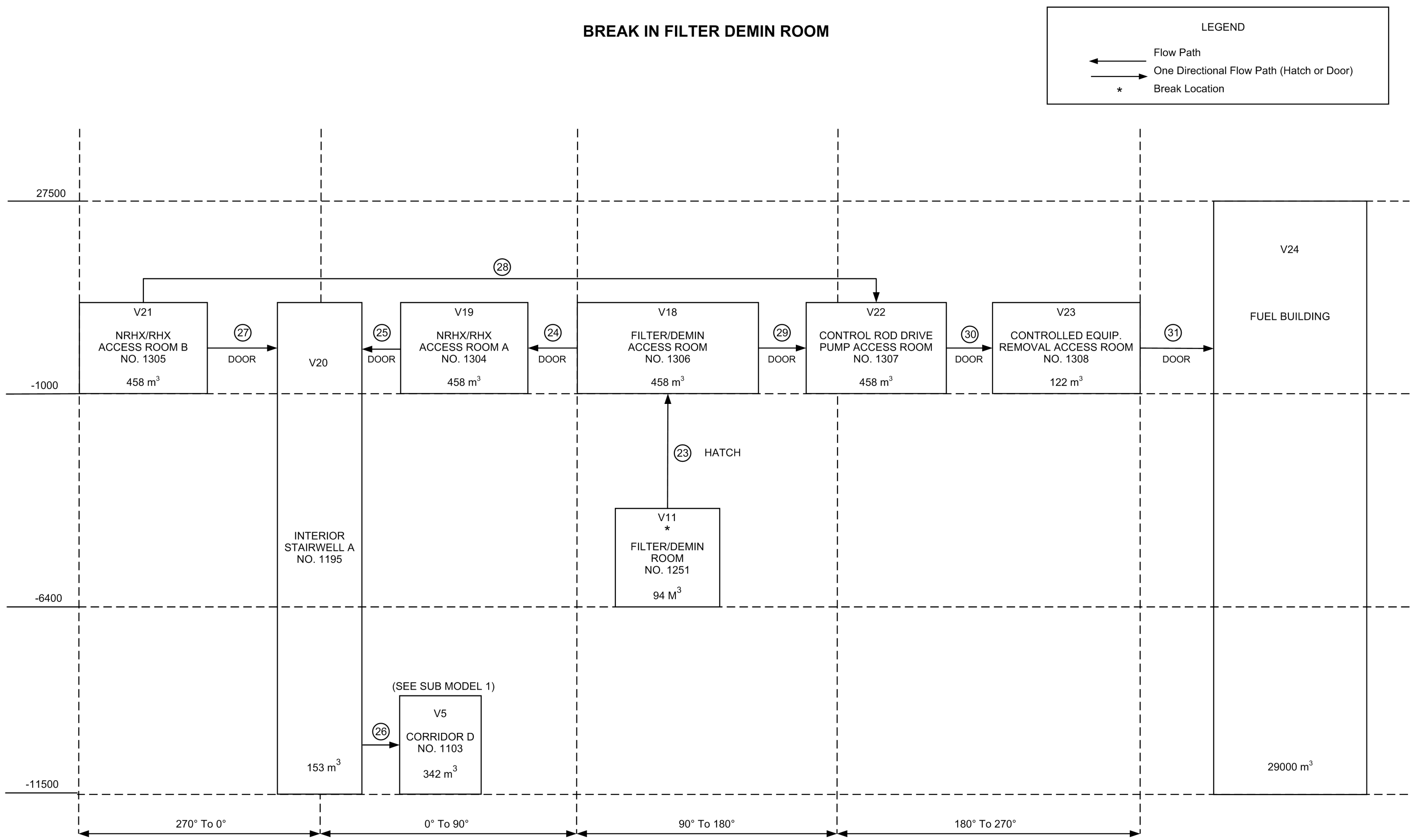


Figure 6.2-18. (Continued) RWCU/SDC System Compartment Pressurization Analysis (Sub-Model 2)

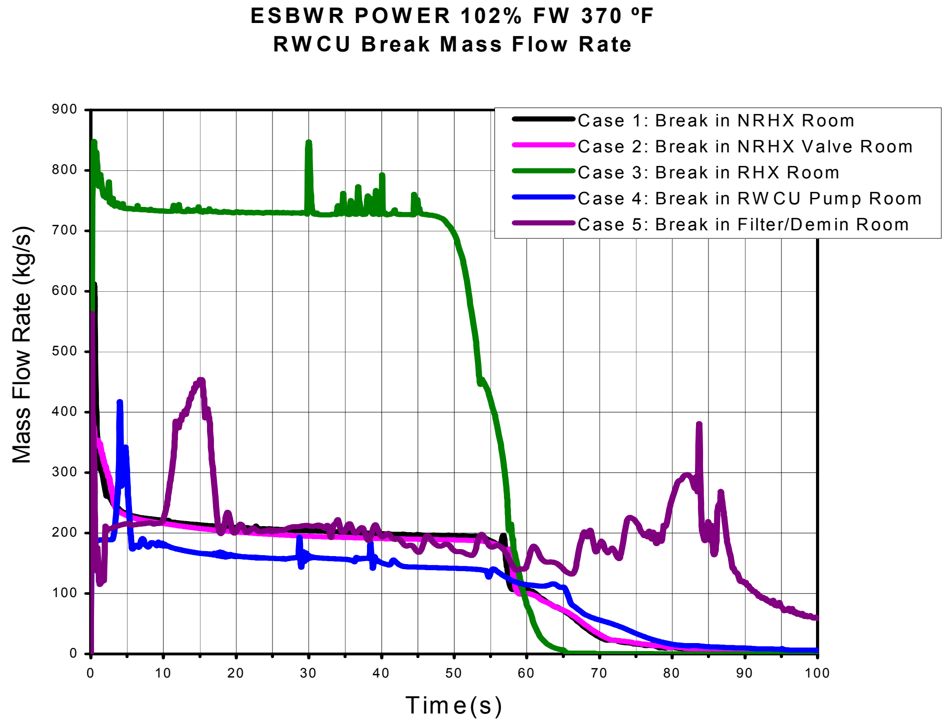


Figure 6.2-18a. RWCU Pipe Break Mass Flow Rate

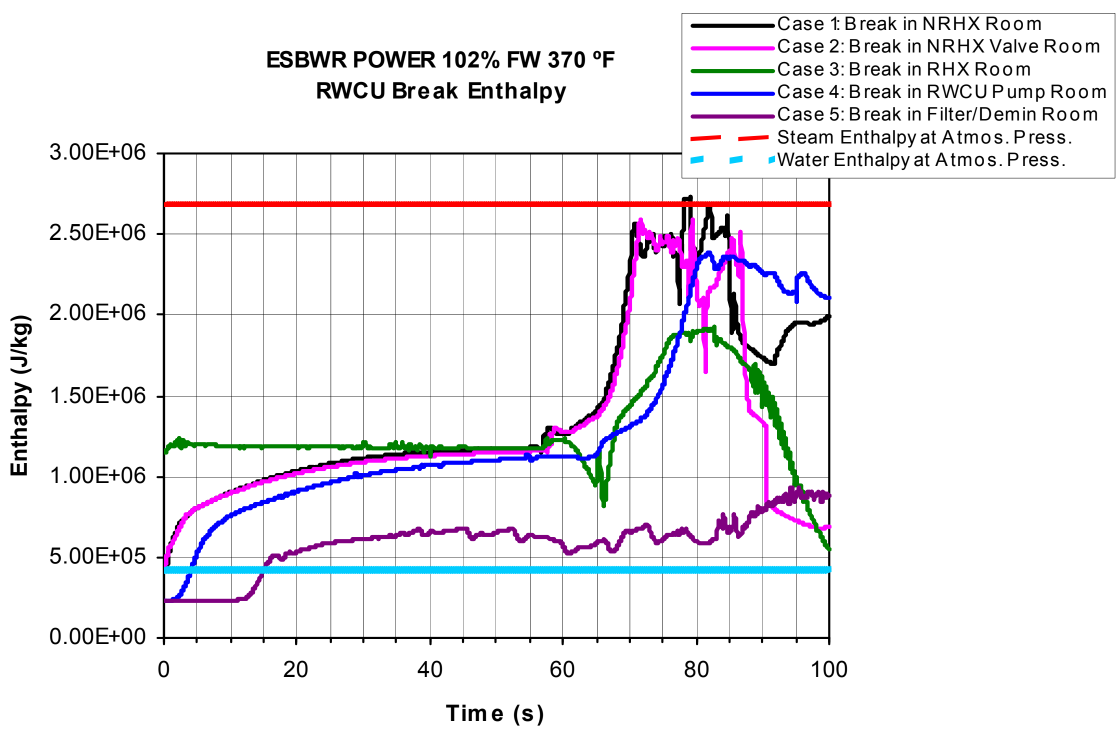


Figure 6.2-18b. RWCU Pipe Break Enthalpy

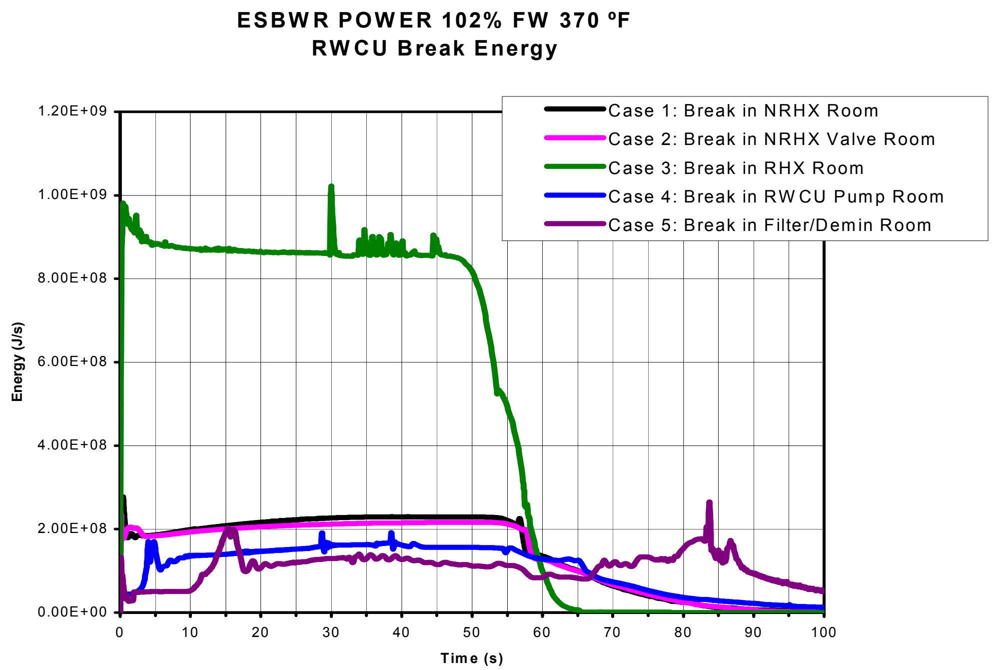


Figure 6.2-18c. RWCU Pipe Break Energy Release

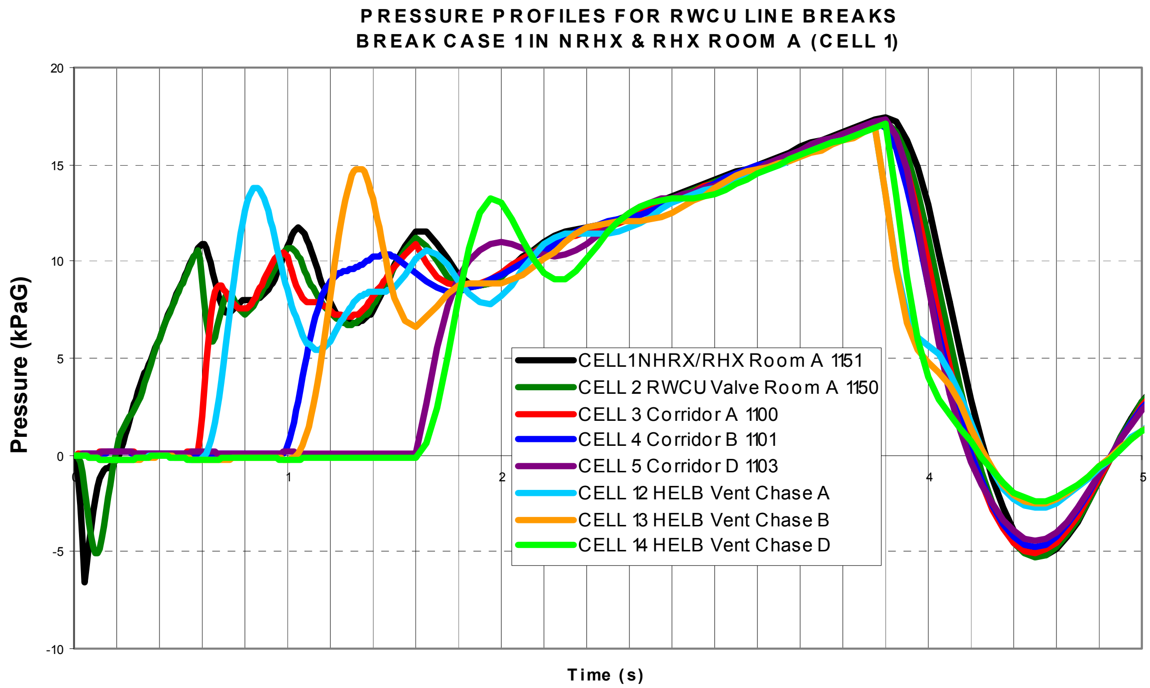


Figure 6.2-19. Pressure Histories due to Break Case 1 in Cell 1 (Sub-Model 1)

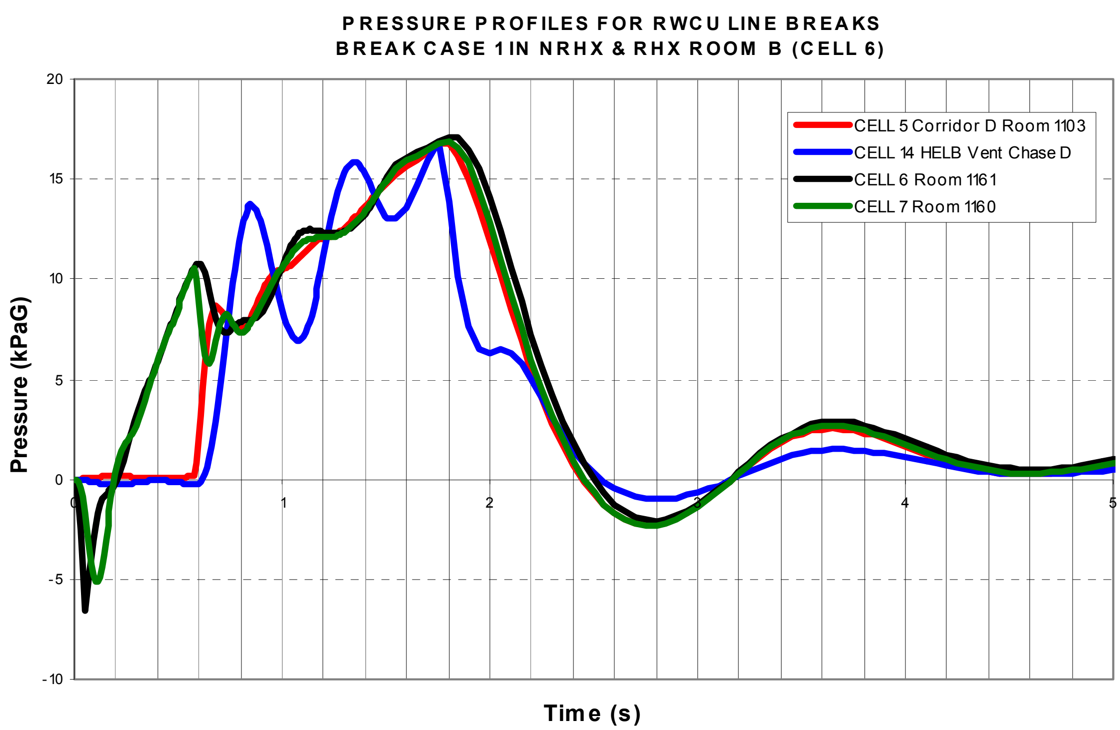


Figure 6.2-20. Pressure Histories due to Break Case 1 in Cell 6 (Sub-Model 1)

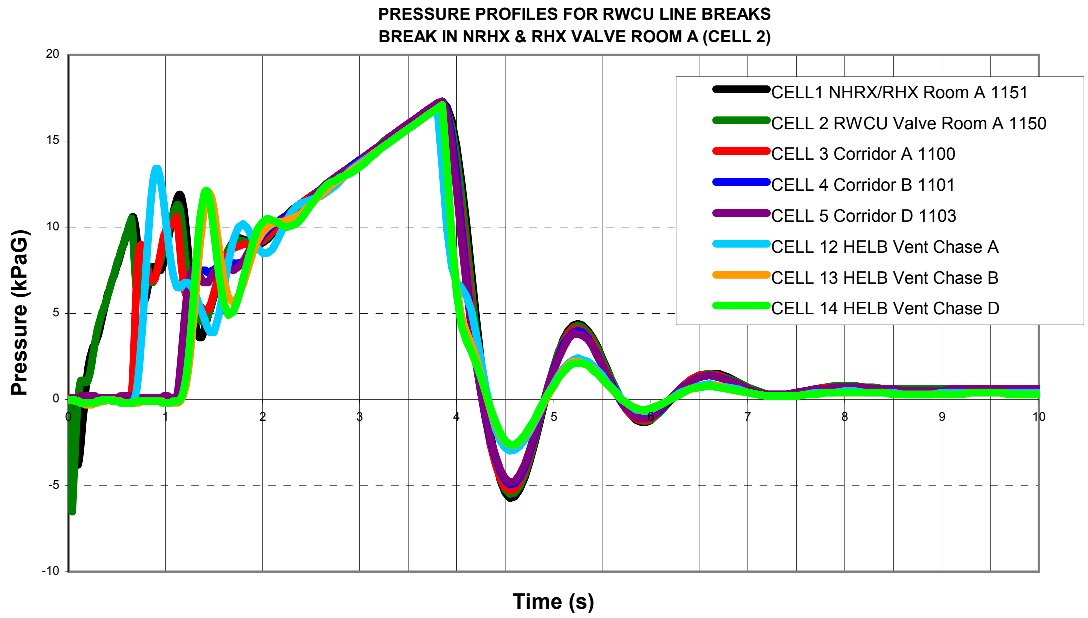


Figure 6.2-21. Pressure Histories due to Break Case 2 in Cell 2 (Sub-Model 1)

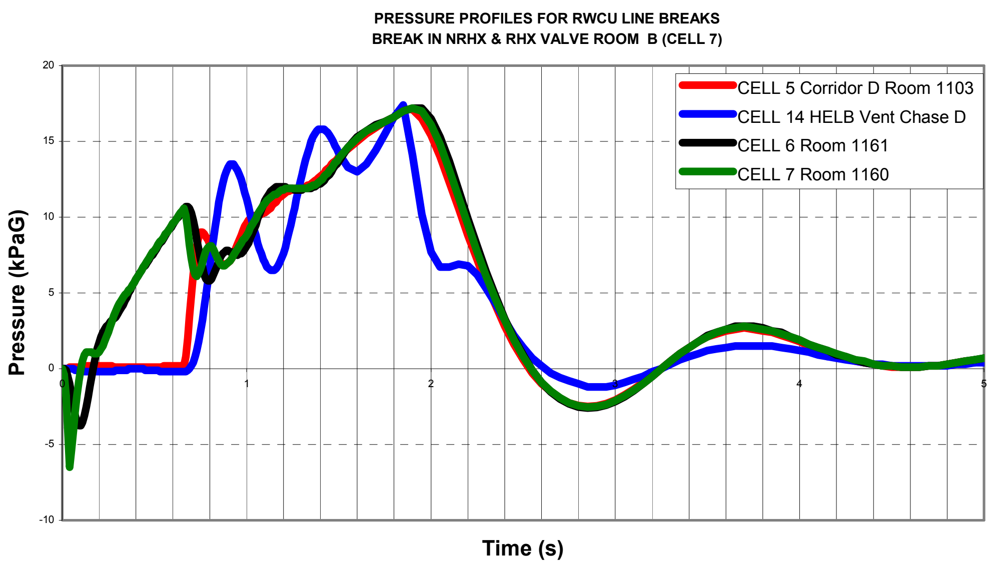


Figure 6.2-22. Pressure Histories due to Break Case 2 in Cell 7 (Sub-Model 1)

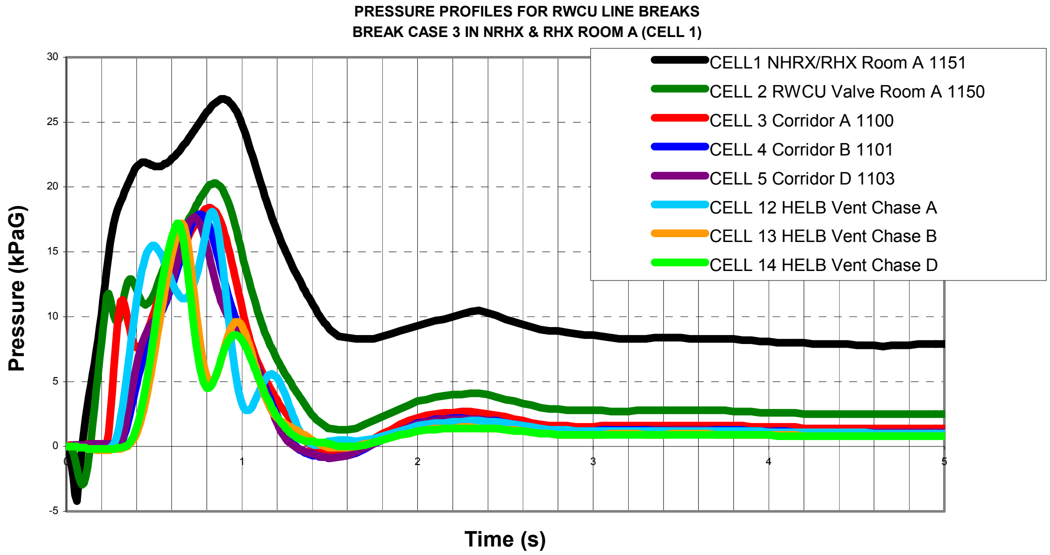


Figure 6.2-23. Pressure Histories due to Break Case 3 in Cell 1 (Sub-Model 1)

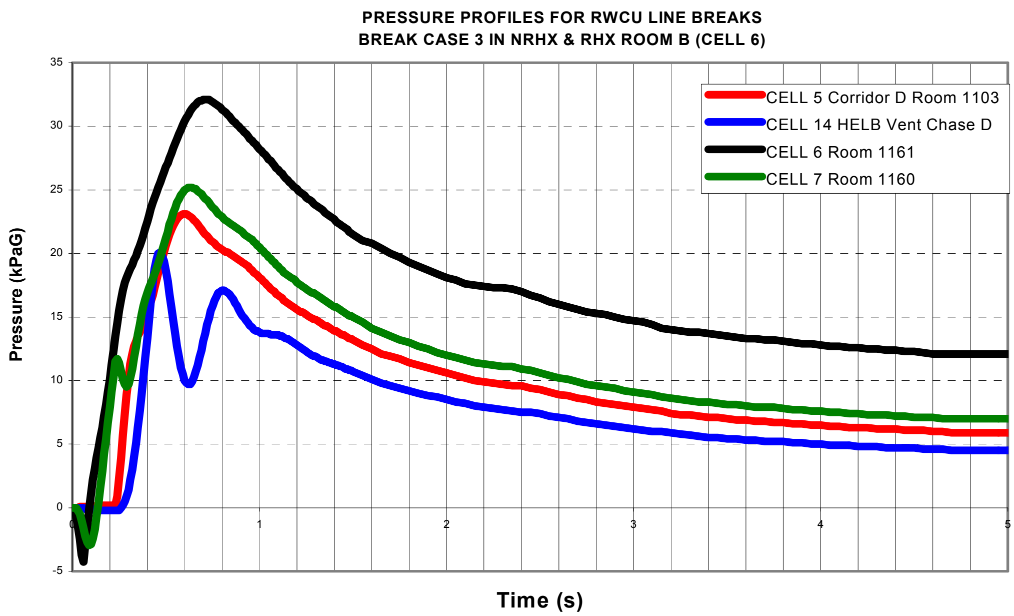


Figure 6.2-24. Pressure Histories due to Break Case 3 in Cell 6 (Sub-Model 1)

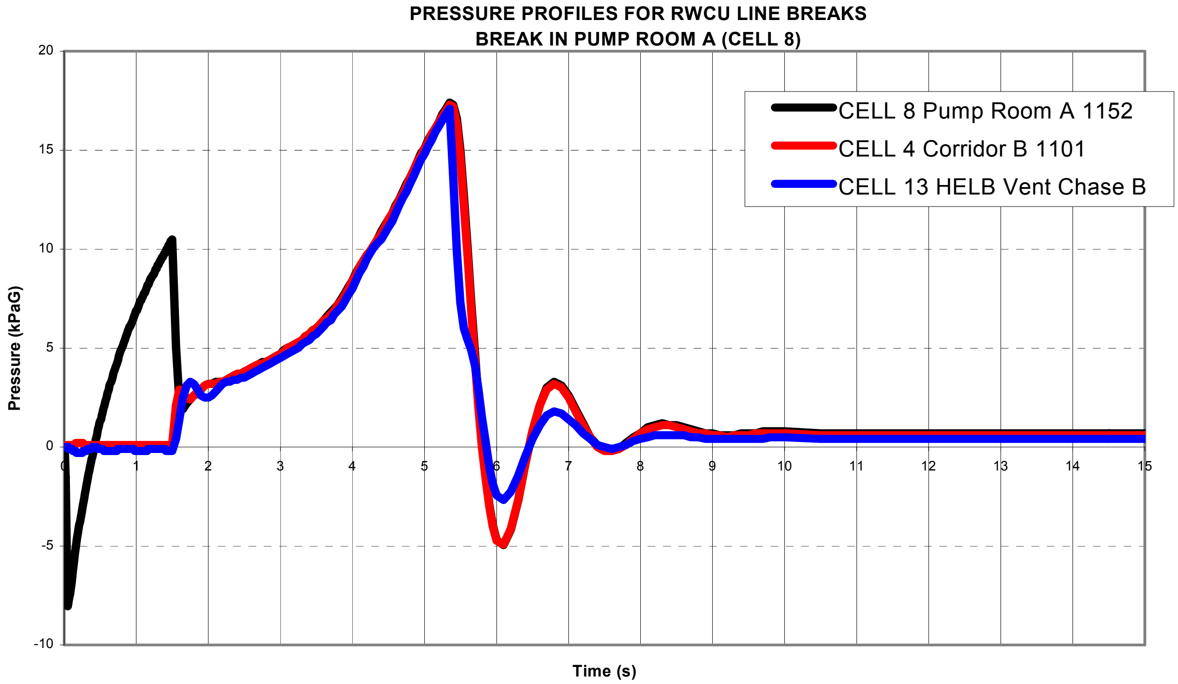


Figure 6.2-25. Pressure Histories due to Break Case 4 in Cell 8 (Sub-Model 1)

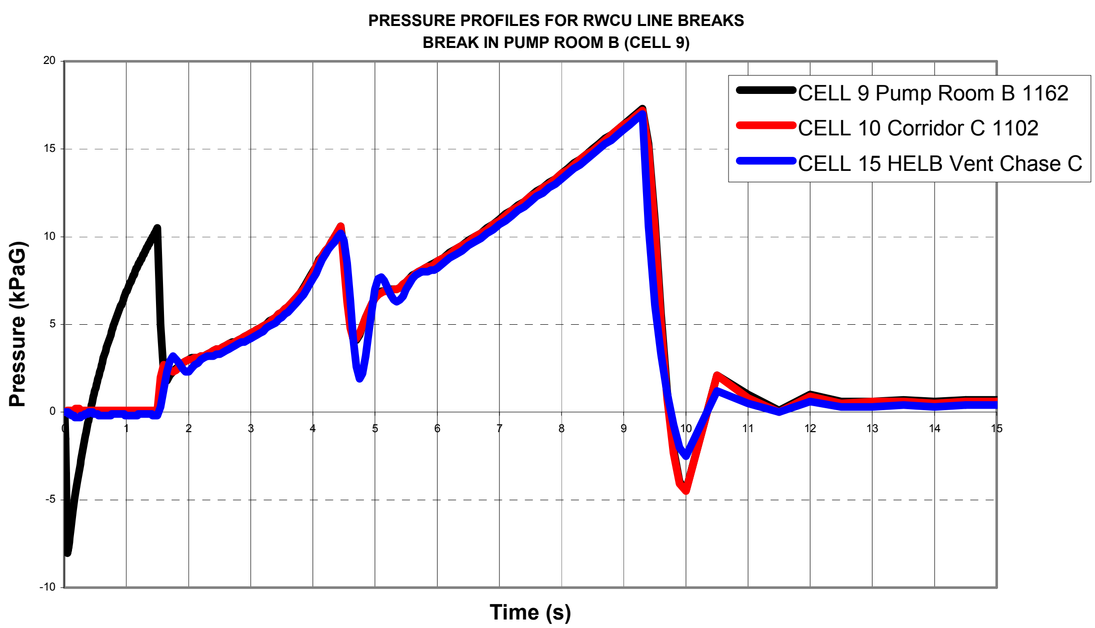


Figure 6.2-26. Pressure Histories due to Break Case 4 in Cell 9 (Sub-Model 1)

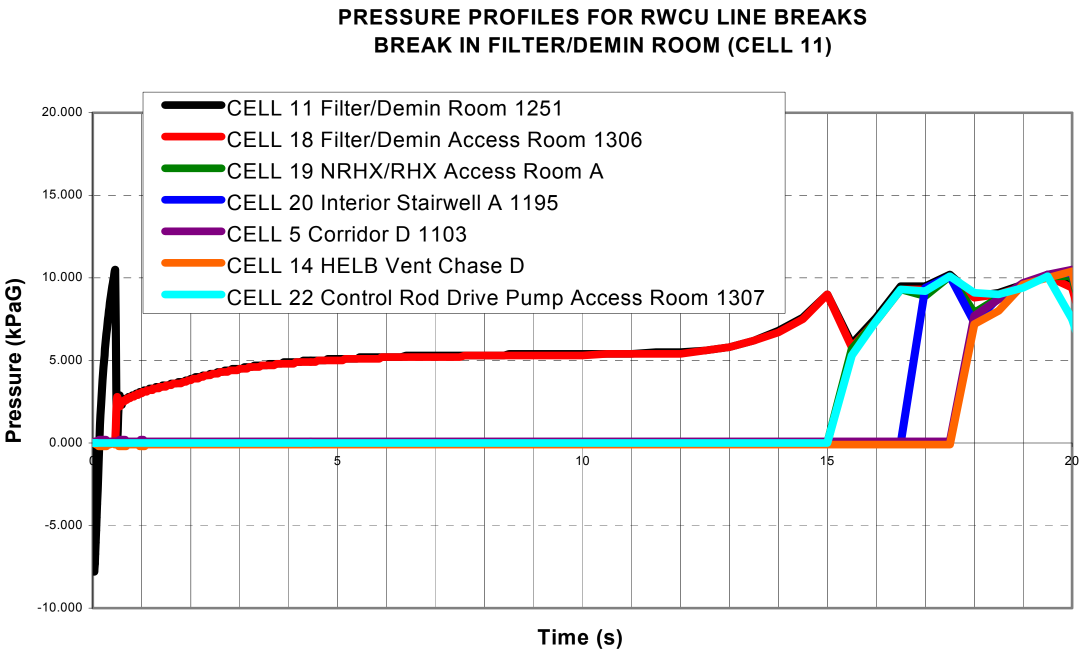


Figure 6.2-27. Pressure Histories due to Break Case 5 in Cell 11 (Sub-Model 2)

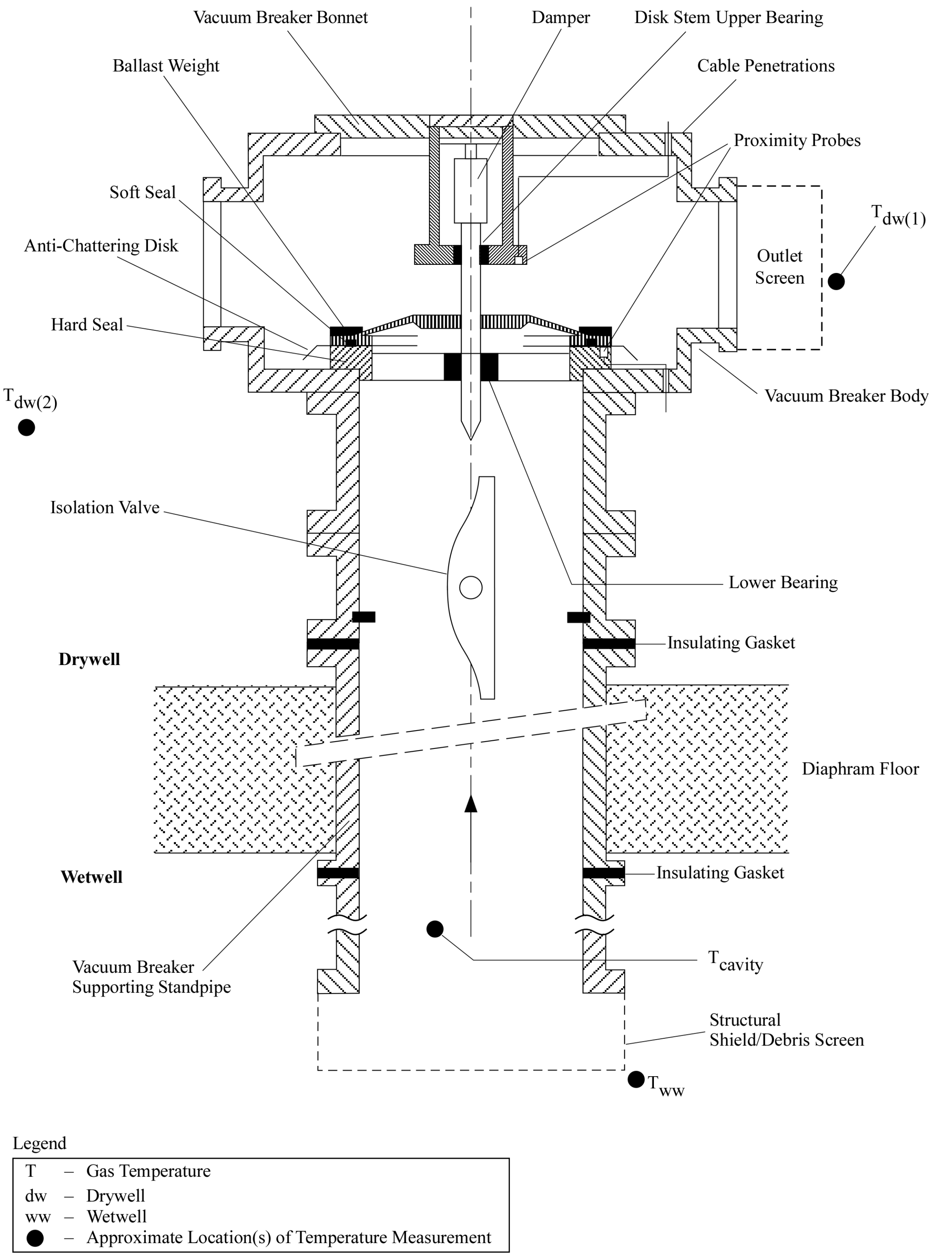


Figure 6.2-28. Wetwell-to-Drywell Vacuum Breaker

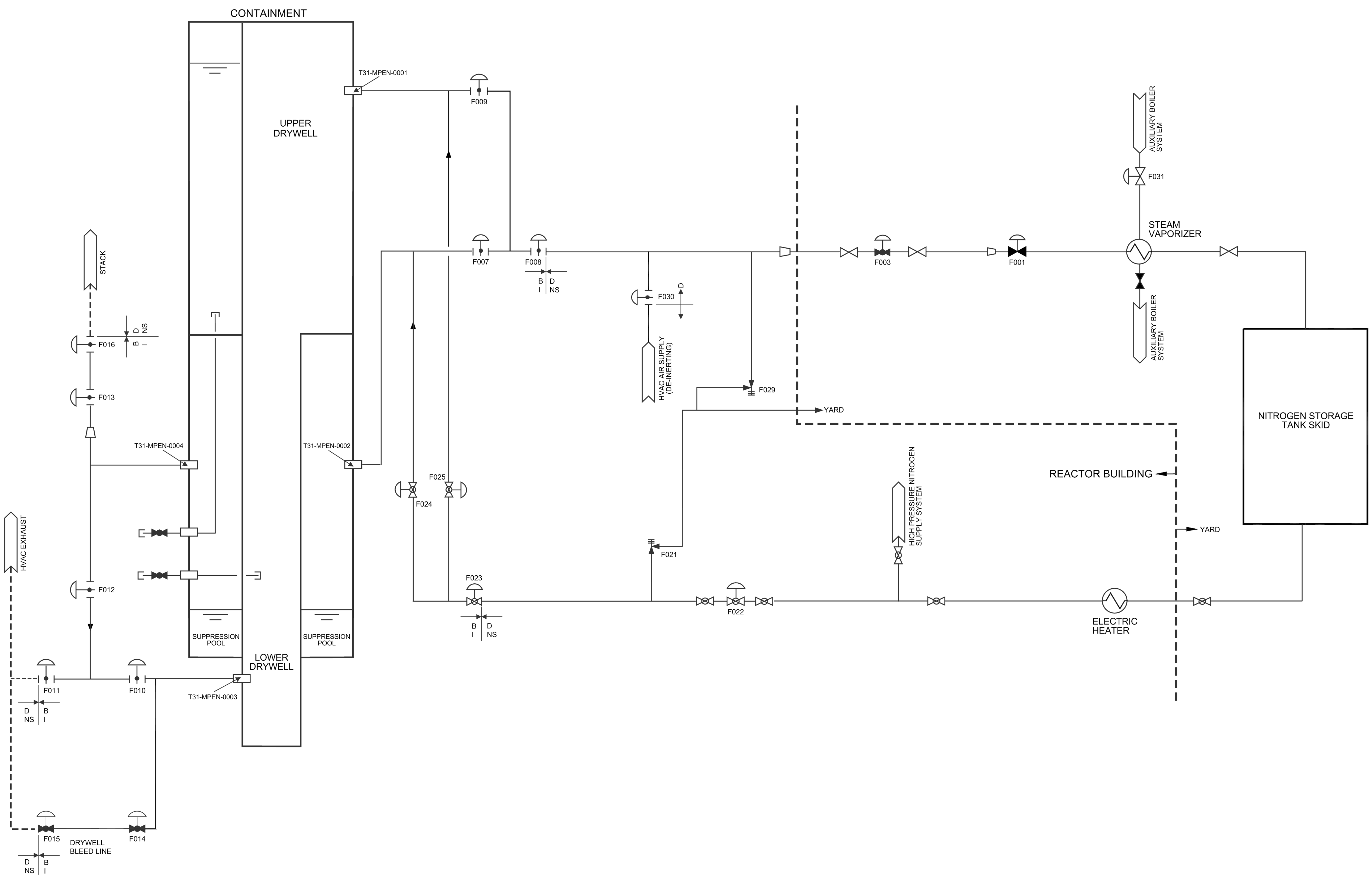


Figure 6.2-29. CIS Simplified System Diagram

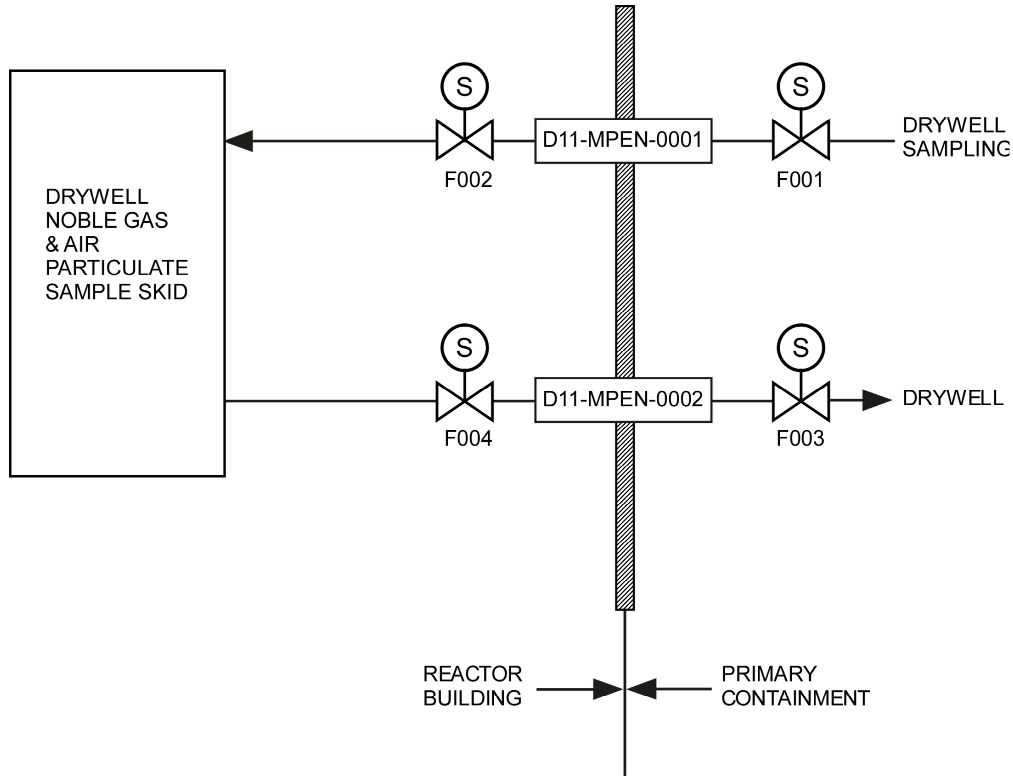


Figure 6.2-30. Drywell Fission Product Radiation Monitoring Subsystem

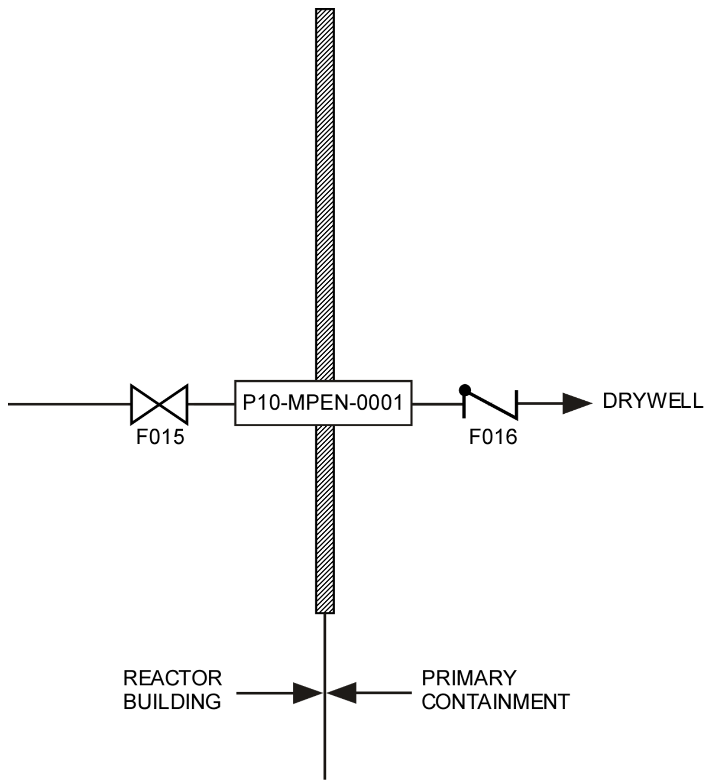


Figure 6.2-31. Containment Isolation Valves for Makeup Water System

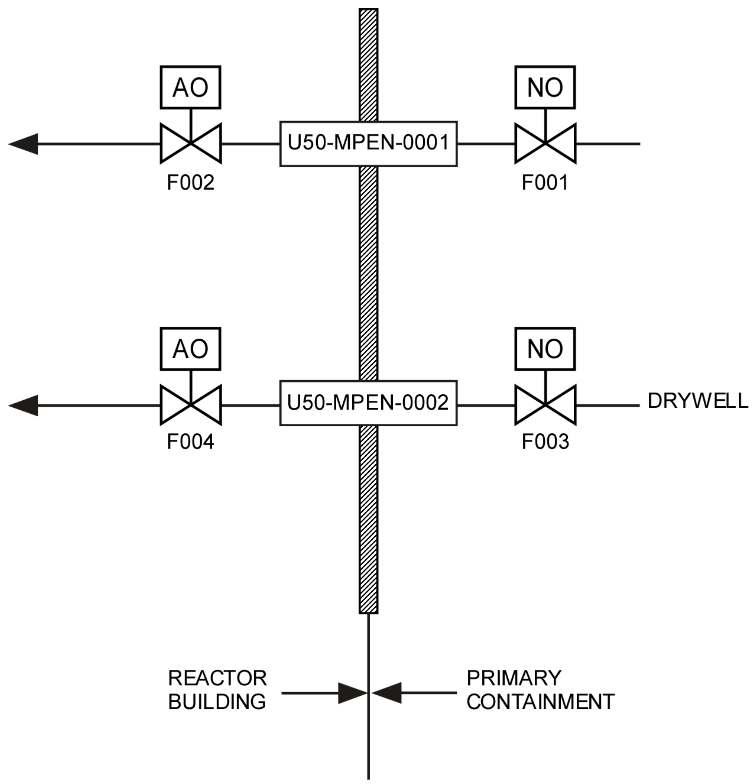


Figure 6.2-32. Containment Isolation Valves for Equipment and Floor Drain System

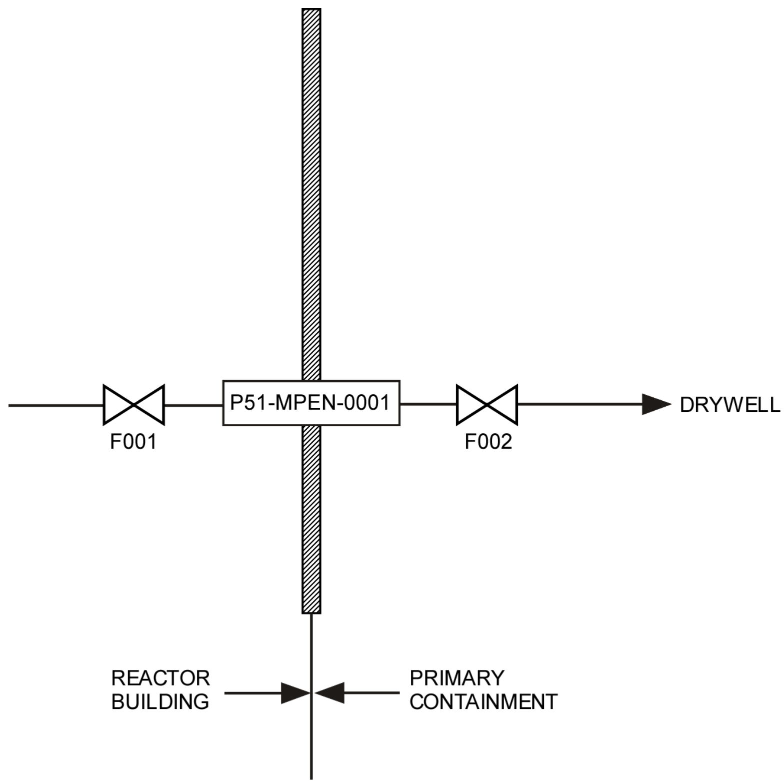


Figure 6.2-33. Containment Isolation Valves for Service Air System

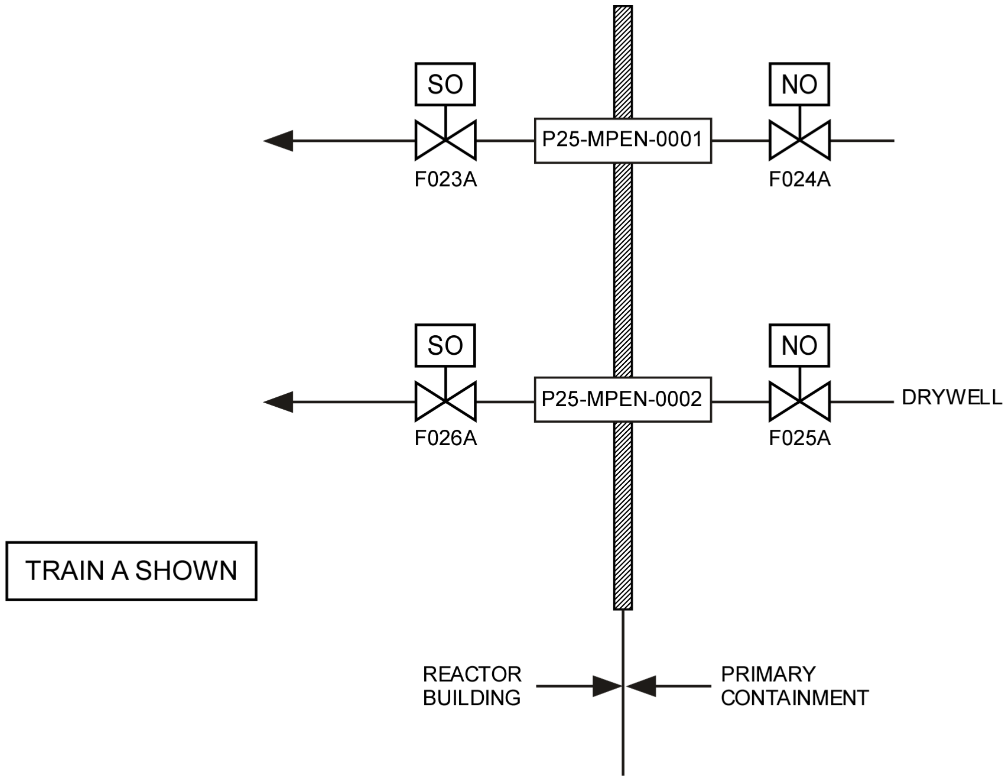


Figure 6.2-34. Containment Isolation Valves for Chilled Water System

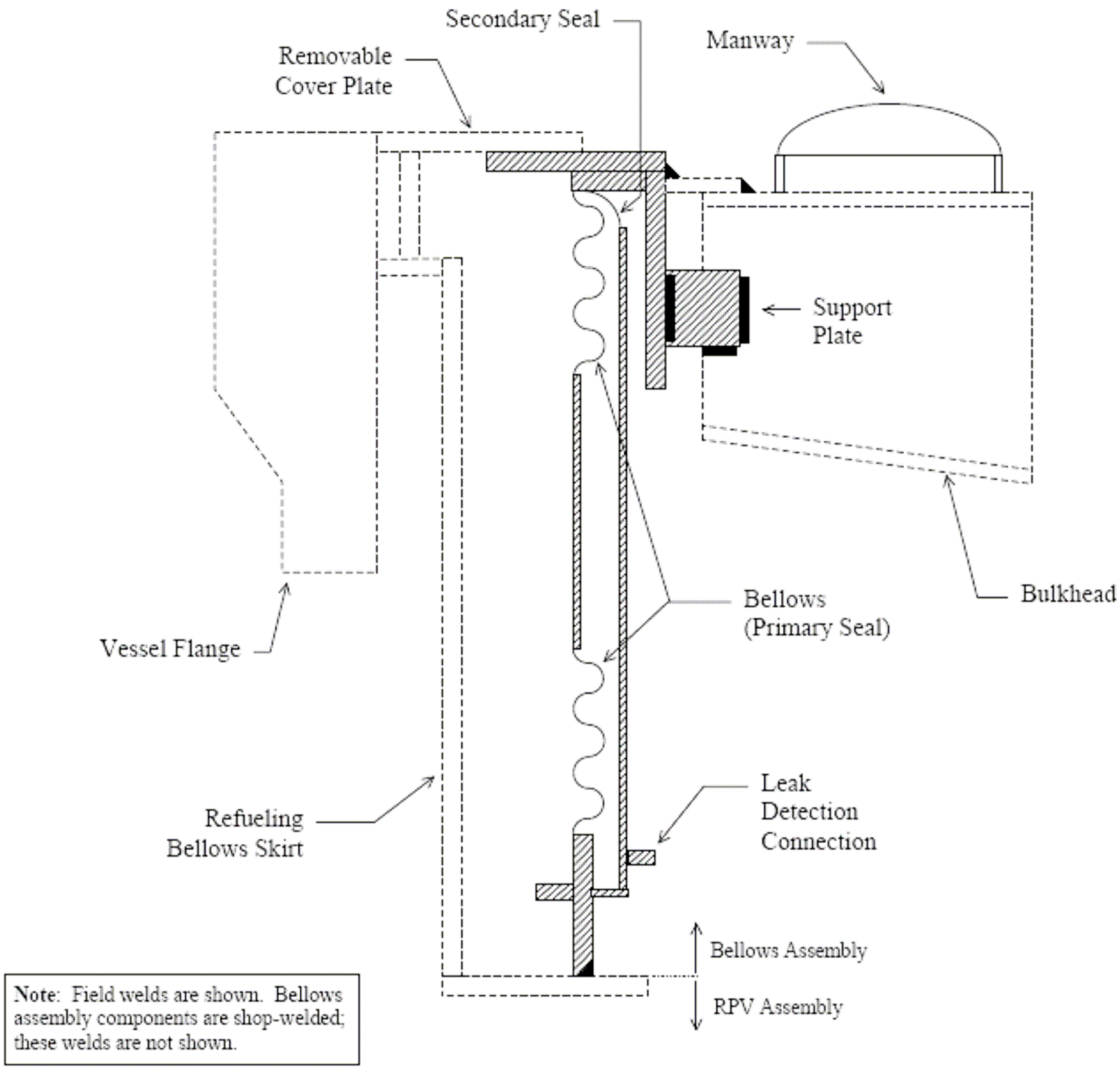


Figure 6.2-35. Refueling Cavity Bellows Assembly

6.3 EMERGENCY CORE COOLING SYSTEMS

Relevant to ESBWR ECCS, this section addresses or references other sections that address the applicable requirements of GDC 2, 4, 5, 17, 27, 35, 36 and 37, 10 CFR 50.46, and Three Mile Island (TMI) action plan items in 10 CFR 50.34(f), discussed in NUREG-0800 SRP 6.3.

The ESBWR ECCS meets the requirements of GDC 2 as it relates to the seismic design of 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 Subsections 8.1.5.2.4 and 8.3.1.2.1.

Regardless if the core has stuck control rods or not, the ECCS maintains the vessel water level above the top of the core for all abnormal events. 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 do 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 6.2.1.1.3, 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 GDCS and 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 GDCS, ICS, the SLC system, and the ADS function of the Nuclear Boiler System (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 the performance analysis provided in Subsection 6.3.3. Table 15.2-23 provides the response time limits for initiation signals used/assumed in accident analyses.

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 Systems 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 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 two RPV nozzles.
- Long-term cooling requirements call for the removal of decay heat from DW via the Passive Containment Cooling System (Subsection 6.2.2).
- Systems that interface with, but are not part of, the ECCS are designed and operated such that failures in the interfacing systems do not propagate 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 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 DW) and the ECCS equipment located outside the DW 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 DW 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 NBS. The depressurization function is accomplished through the use of SRVs and DPVs.

Isolation Condenser System

The ICS provides additional liquid inventory upon opening of the condensate return valves to initiate the system. The ICS also provides initial depressurization of the reactor before ADS in event of loss of feedwater, 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).

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 open the deluge valves manually, two control switches of the "arm/fire" type located in the MCR are 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 a LOCA.

The system also drains the GDCS pools to the lower DW 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 DW cavity floor. This action is accomplished by detection of elevated temperatures registered by thermocouples in the lower DW cavity, and by logic circuits that actuate squib-type valves on independent pipelines draining GDCS pool water to the lower DW 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 DW temperature is less than a preset value.

The GDCS requires no external electrical power source or operator intervention. The GDCS injection lines initiate on a sustained RPV Level 1 signal or a sustained Drywell Pressure High signal, and the GDCS equalizing lines initiate on a sustained RPV Level 1 signal (Table 7.3-4). These signals initiate 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 the GDCS pools. 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 1.00 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 DW, the flow through the deluge lines is required to flood the lower DW 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 DW.

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 DW 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 DW region directly via four GDCS injection drain lines (one each from two pools and two from the third pool) through the 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 are 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 DW. 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 1.00 meter (3.28 ft) 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 horizontal vent and the centerline of the GDCS equalizing line RPV nozzle.

The GDCS deluge lines provide a means of flooding the lower DW 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 DW 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 DW region basemat temperature indicative of molten fuel on the lower DW 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 DW 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 the lower DW. The 200 mm (8 inch) injection line continues after the 100 mm (4 inch) deluge line connection from the upper DW region through the DW annulus where the 200 mm (8 inch) line branches into two 150 mm

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

(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 pressure and temperature. Figure 6.3-1 illustrates the arrangement of GDCS piping.

The RPV injection line nozzles and the equalizing line nozzles all contain integral flow limiters with a venturi shape for pressure recovery. The maximum throat diameter of the nozzles in the short-term system is 76.2 mm (3 in) and the maximum throat diameter of the nozzles in the long-term system is 50.8 mm (2 in). The short-term and long-term flow limiters have nominal reactor side outlet ratio of length to diameter (L/D) values of 4.41 and 6.59, respectively. 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 DW 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 caused 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 allows 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 DW 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 DW region. Flow then drains to the lower DW via permanently open DW 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 screen 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 is covered by a perforated steel plate to prevent debris from entering the pool and potentially blocking the coolant flow through the fuel. Protection against the dynamic effects associated with postulated pipe ruptures is described in Section 3.6. The maximum hole diameters in the perforated steel plate are 38 mm (1.5 inch). A splash guard is provided at the opening to minimize any sloshing of GDCS pool water into the DW following a 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 by nonsafety-related power.

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 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 injection lines are initiated following a sustained RPV Level 1 signal or a sustained Drywell Pressure High signal, and the GDCS equalizing lines are initiated following a sustained RPV Level 1 signal (Table 7.3-4). The signals start two sets of timers in each division; injection valve timer for initiation of the short-term water injection lines and a longer equalization timer which creates a permissive signal (in combination with RPV water level below Level 0.5 or 1.00 m (3.28 ft.) above TAF) for initiation of the long-term injection lines. After the injection valve timer expires, 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 sustained RPV Level 1 signal and when RPV level reaches Level 0.5, which is 1.00 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 DW (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 DW and break flow losses in the mid-term. In the long-term the equalizing lines provide for evaporation losses to the DW.

The GDCS is designed to mitigate the consequences of a hypothetical severe accident with molten core material on the lower DW floor. The lower DW basemat is divided into 30 cells, with two thermocouples (channels A and B) installed in each cell, to sense the presence of molten fuel on the lower DW floor. A temperature greater than the setpoint sensed by channel A thermocouples in any two adjacent cells, coincident with channel B thermocouples also sensing a temperature greater than the 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 DW 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 DW. This water aids in cooling the molten core.

GDCS Injection and Equalizing Line Sloping

The GDCS injection lines downstream of the GDCS pools have a minimum downward slope of 1:48 (one unit of rise per 48 units of run) to each GDCS injection line squib valve. Downstream of each GDCS injection line squib valve, the lines have a minimum upward slope of 1:48 to each RPV injection line nozzle.

The GDCS equalizing lines downstream of the suppression pool have a minimum downward slope of 1:48 to each GDCS equalizing line squib valve. Downstream of each GDCS equalizing line squib valve, the lines have a minimum upward slope of 1:48 to each RPV equalizing line nozzle.

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 flow coefficient, C_v, that permits 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.

Valve actuation occurs via squib valve initiators, in which a pyrotechnic booster charge is ignited and hot gases are produced. The logic for initiation is described in Subsection 7.3.1.2.2. 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.

The squib valve is designed to meet the following requirements:

- The valve is designed such that, in the event of squib actuation, no internal fragments (not inherently trapped within the valve) are produced of a size that if transported downstream could, by themselves or collectively, credibly represent a threat of blockage at the venturi throat.
- The valve is designed such that, in the event of squib actuation, no missiles are generated that could impact the operation of any system valves, components or instrumentation within the DW.
- The valve provides remote indication of "valve opened" and "valve closed" status.
- The valve has a Cv greater than 94.72 m3/hr/kPa$^{1/2}$ (1095 gpm/psi$^{1/2}$) at full GDCS flow. The valve manufacturer performs a full flow test and provides test data to verify the minimum required Cv.
- Once the valve is open, it remains permanently open.

GDCS Check Valve

The GDCS check valves are long duration submersible, piston check valves. The valves meet the requirements for minimum fully open flow coefficient in the forward flow direction and maximum fully open flow coefficient in the reverse flow direction. The reverse flow coefficient addresses the case in which a check valve sticks in the fully open position following a LOCA. Type testing is performed to verify the valve meets the reverse flow coefficient requirement. The results of the testing and a comparison of the measured flow coefficient to the maximum value is documented in a report. 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.

The check valves will either be installed in a horizontal piping run and held normally open by a spring or installed in a vertical piping run and held normally open by gravity. In either case, the net force keeping the valve open will be minimized to a value sufficient to ensure the valve is open with no differential pressure. One possible configuration is a nozzle check valve in a vertical pipe, oriented such that gravity opens the valve. To minimize the reverse flow/differential pressure required to close the valve, valve opening would be resisted by a light spring, sized such that the valve is fully open with no differential pressure (disk weight is equal to the spring force at fully open).

When the GDCS squib valves actuate, reactor pressure is significantly below the normal operating pressure but above the GDCS injection pressure. The GDCS check valves will remain closed until the reactor pressure is just above the GDCS injection pressure. As reactor pressure drops further, GDCS injection will begin at a low flow rate and gradually increase, such that chatter is not expected to occur.

The GDCS check valves are located upstream of the GDCS injection squib valves, which are located at the bottom of a U-shaped pipe loop, and open block valves downstream of the squib valves. During normal operation, the injection line squib valves are closed and pipe legs on both sides of the squib valves are filled with water. The water solid pipe legs from the squib valves to the reactor pressure vessel inlet nozzles prevent noncondensable gases from entering into the injection lines. The GDCS injection lines from the squib valves to the GDCS pools are self-venting back to the pools, which are at the highest elevation of the system. Test lines in the system stay filled with liquid up to the test line isolation valves.

The design process for the GDCS check valves will evaluate the loads on the valve disk during normal and design basis conditions to ensure the valves remain open under normal operating conditions (zero differential pressure) and will close under low reverse differential pressure/flow conditions. The design process will also include an evaluation of the closure loads, including potential water hammer effects, on the valve, piping and other applicable components under design basis conditions and following an inadvertent actuation of the GDCS squib valves, to ensure damage from deceleration of GDCS fluid will not occur. Valve qualification will verify applicable design requirements are met and will also address the effect of the installed orientation on valve performance.

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.

The deluge valve is designed to meet the following requirements:

- The valve remains closed with zero leakage under all normal operating and design basis conditions, during anticipated operational occurrences and following the inadvertent opening of a GDCS injection line squib valve while the RPV is at normal operating pressure.
- The valve is designed such that, in the event of squib actuation, no missiles are generated that could impact the operation of any system valves, components or instrumentation within the DW.
- The valve is designed to survive the severe accident environment and still perform its intended function.
- The valve provides remote indication of "valve opened" and "valve closed" status.
- The valve has a C_v greater than 11.2 $m^3/hr/kPa^{1/2}$ (130 $gpm/psi^{1/2}$) at full flow. The valve manufacturer performs a full flow test and provides test data to verify the minimum required valve C_v.

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, and

- Seismic Category: I.

All piping and valves (including supports) connecting the GDCS pools and suppression pool to the check valve, and all piping and valves (including supports) connecting GDCS pools to the lower DW are classified as follows:

- Safety-Related,
- Quality Group: B, and
- 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 a sustained RPV Level 1 signal and to a sustained Drywell Pressure High 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 continuously self-tested. 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.

Pre-operational and periodic testing of the igniters and booster subassemblies of the GDCS squib valves are performed. These tests quantify, measure, or detect any degradation, and provide assurance that the GDCS squib valves will perform their safety-related function.

The igniters and booster subassemblies of the GDCS squib valves are removed from the valves and tested in sequential sets. The initial qualified life for the boosters and ignitors is four years. Replacement is done without opening of the reactor coolant pressure boundary. Subsequently in the laboratory, the removed charges are tested to confirm end of life capability to function upon demand. Periodic testing is conducted during the refueling and maintenance outage at the end of each plant operating cycle. Pre-operational testing is conducted prior to initial plant start up.

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 DW and connect to the high and low pressure sides of two level transmitters. Each line penetrating the DW contains a series of isolation valves. The output of the level transmitter is sent to the SSLC and the Control Rod Drive (CRD) system for processing. 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 GDCS pool water level signal is also sent to FAPCS.

During a LOCA after GDCS initiation, a low water level in two of the three GDCS pools isolates HP CRD flow. This prevents excessive makeup water being pumped into the containment during a LOCA. In addition to GDCS pool water level low, there are other signals that isolate HP CRD flow. These are described in Subsection 4.6.1.2.5.

Controls

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

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 DW 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 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.

Non-safety, 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 Safety System Logic and Control/Engineered Safety Features (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 ten SRVs and eight DPVs. The SRVs are described in Subsection 5.2.2. The DPVs are described in Subsection 5.4.13.

System Operation

The ADS automatically actuates in response to a sustained RPV Level 1 signal or a sustained Drywell Pressure High signal (Tables 7.3-2 and 7.3-3). 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 Tables 7.3-2 and 7.3-3.

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.

Although the nominal and bounding containment performance analyses are performed at an initial condition of 46ºC (115ºF) for the GDCS pool water temperature (DCD Tier 2, Table 6.2-6), additional analyses assuming GDCS pool water temperature as high as 65.5ºC (150ºF) were performed. Using TRACG for the limiting cases in Sections 6.2 (main steam line break) and 6.3 (ICS drain line break) demonstrates that higher initial GDCS pool water temperatures do not have a significant impact on the containment and ECCS performance. These analyses demonstrate the relative insensitivity of the calculated peak containment pressure and temperature and reactor pressure vessel long-term water level after a DBA for increased GDCS pool water initial temperature.

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 9.3.5.4.

6.3.2.10.5 Instrumentation Requirements

Refer to Subsection 9.3.5.5.

6.3.3 ECCS Performance Evaluation

Performance of the ECCS 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 performance for the entire spectrum of postulated break sizes. Scaling analyses documented in Reference 6.3-4 show that the sub-scale integral test facilities, i. e., GIST and GIRAFFE/SIT, adequately simulate the phenomena important to the ESBWR ECCS performance after a postulated pipe break.

The analyses are based upon the bundle design discussed 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 decrease), the limiting breaks were evaluated for the initial core and results are discussed in Reference 6.3-3. Specifically, the initial feedwater temperature is varied from 160ºC (320ºF) to 252ºC (486ºF) for the bounding ICS return line break analyses with failure of one SRV or one GDCS injection valve or one DPV. For all cases, the reactor core remains covered with adequate margin with no cladding heat up, similar to that shown in Table 6.3-5. Also, the variation of minimum chimney static head level and the minimum downcomer collapsed water level is small with respect to the initial feedwater temperature. As explained in Section 1.2 of Reference 6.3-3, the nuclear characteristic of the core (initial vs. equilibrium) is not an important parameter during LOCA. Therefore, the results presented in Table 6.3-5 and Reference 6.3-3 are valid for all cycles.

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 Maximum Linear Heat Generation Rate (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 6.3.2.7.4 and 6.3.4. 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 Performance

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

Criterion 1: Peak Cladding Temperature (PCT)

"The calculated maximum fuel element cladding temperature shall not exceed 2200°F," which is equivalent to 1204°C. Conformance to Criterion 1 is shown for the system response analyses within Subsection 6.3.3.7 and specifically in Table 6.3-5 (Summary of LOCA Analysis Results).

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 6.3-10a (IC Drain 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 SLC System is designed to be used during an Anticipated Transient Without Scram (ATWS), and the ECCS system ICS is designed to avoid unnecessary use of other ESFs for residual heat removal. Both, SLC System 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 four 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.

The pipe breaks sizes and elevations (relative to the bottom of the vessel) for all vessel penetrations including main steam lines, DPV/IC line, feedwater line, RWCU/SDC line, IC return line, GDCS injection line, GDCS equalizing line, and bottom drain line are listed in Table 6.3-5a. The PCCS condensate return line is not included since it is connected to the GDCS pool.

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, conformance 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 drain 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. Important variables from these analyses are shown in Table 6.3-5.

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, once the break is isolated, the isolation condensers, High Pressure Control Rod Drive flow or the ADS/GDCS systems 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 as shown in Table 6.3-5.

For each bundle design in a plant, conformance 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 IC drain line break and the GDCS injection line break were evaluated. The results of these calculations are given in Table 6.3-5. The IC drain 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.

The IC drain line break with failure of one GDCS injection valve is the most limiting case for level above the core. Only three IC's are credited in LOCA analysis, and in the IC drain line break model, two of the IC's are lumped together while the third is modeled individually. It is the individually modeled IC that suffers the line break. The effect is such that only two of the IC's are seen to have a mitigating effect on the accident. Furthermore, the IC drain line break size is four times as large as the GDCS injection line break. This increase in area results in a larger loss of inventory from the RPV as is reflected in the lower collapsed downcomer water level. The collapsed downcomer level presented in Table 6.3-5 indicates that the IC drain line break with failure of one GDCS injection valve is the most limiting. Results of nominal cases are also shown in Table 6.3-5.

6.3.3.8 ECCS-LOCA Performance Analysis Conclusions

The ECCS-LOCA performance 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 or startup test program. As applicable, each component is tested for power source, range, setpoint, limit switch setting, etc. Subsection 6.3.2.7.4 contains additional details on GDCS testing, and Subsection 7.3.5.4 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 RG 1.68, Appendix A, except that the mechanical components of the ECCS squib type valves are 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 DW can be visually inspected only during periods of access to the DW.

6.3.5 Instrumentation Requirements

Design details including redundancy and logic of the ECCS instrumentation are discussed in Subsection 7.3.1.

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-603 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 (Deleted)

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, Revision 1, Class I (Non-proprietary), May 2009.

6.3-4 GE Hitachi Nuclear Energy, "ESBWR Scaling Report," NEDC-33082P, Revision 2, Class III (Proprietary), April 2008; NEDO-33082, Revision 2, Class I (Non-proprietary), April 2008.

Table 6.3-1

Significant Input Variables to the ECCS-LOCA Performance Analysis

Variable	Units	Value
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 Maximum Sensor Response Time	--- sec	Loss of power to two feedwater pump buses 2
B. Emergency Core Cooling System Parameters		
B.1a RPV Level 1 Signal		
Variable	**Units**	**Value**
Initiating Signal: Level 1	meters (above TAF) [ft] (above TAF)	4.05[(2)] [13.28]
Maximum Allowable Time Delay to Confirm RPV Level 1 Signal	sec	10
B.1b Alternate ADS/GDCS Initiation Signal		
Variable	**Units**	**Value**
Initiating Signal: DW Pressure	kPaG [psig]	13.8 [2.0]
Maximum Allowable Time Delay to Confirm Alternate ADS/GDCS Initiation Signal	hour	1
B.2a Gravity-Driven Cooling System (Short-Term)		
Variable	**Units**	**Value**
Initiating Signal	—	Confirmed initiating signal (See B.1 a, B.1b)
GDCS Injection valve timer delay	sec	150
Minimum drainable inventory per GDCS pool		See Table 6.3-2
Minimum elevation of GDCS pool surfaces above the RPV nozzles		See Table 6.3-2
GDCS drain line loss coefficient (k/A^2)	$1/m^4$ [$1/ft^4$]	12.587×10^3 [1.458×10^6]

Table 6.3-1

Significant Input Variables to the ECCS-LOCA Performance Analysis

B.2b Gravity-Driven Cooling System (Long-Term)		
Variable	**Units**	**Value**
Permissive Signal Delay Time after Level 1	min	30
Initiation Signal Level 0.5 after Permissive Signal	meters (above TAF) [ft] (above TAF)	1.00 [3.28]
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]
Isolation Condenser Water Inventory	—	Credited
B.4 Standby Liquid Control System		
Variable	**Units**	**Value**
Initiating Signal	—	DPV actuation from B.1a
Liquid Volume per Tank	m^3 [ft^3]	7.8 [275.4]
B.5 Automatic Depressurization Subsystem		
Variable	**Units**	**Value**
Initiating Signal	—	Confirmed initiating signal (See B.1a, B.1b)
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(1)	kg/s MPa (gauge) [lbm/hr] [psig]	1,380 8.618 [1.093 x 10^7] [1250]
Total Number of Depressurization Valves	—	8

Table 6.3-1

Significant Input Variables to the ECCS-LOCA Performance Analysis

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 [18.6×10^6] [1085]
B.6 Hydraulic Control Units		
Variable	**Units**	**Value**
Water Added During Scram	m^3 [ft^3]	8.9 [314]
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)	—	1.22
D. Feedwater Isolation		
Variable	**Units**	**Value**
DW Pressure High-High	KPaG [psig]	27.6 [4.0]

(1) ECCS/LOCA performance was analyzed with ADS capacity of 1440 kg/s (1.143×10^7 lbm/hr). Sensitivity study has confirmed that the variations of chimney and downcomer minimum levels are both under 0.1 m (0.33 ft.), for ADS capacity between 1440 and 1380 kg/sec (1.143×10^7 and 1.095×10^7 lbm/hr). The minimum chimneys water levels are well above the TAF.

(2) Level 1 analytical limit, a conservatively lower value is used in the calculations.

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 (21.3 ft)	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 DW region	256 m^3/hr (1127 gpm)*
Minimum available suppression pool water inventory 1 meter (3.3 feet) above TAF with 1.0 m (3.3 ft) 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

(Deleted)

Table 6.3-5

Summary of ECCS-LOCA Performance Analyses

Break Location	Break Size[1] m^2 (ft^2)	Minimum Chimney Static Head[3] Level with reference to Vessel Zero Per Active Single Failure m (ft)			PCT[4]	Maximum Local and Core Wide Oxidations (%)[5]	Minimum Downcomer Collapsed Water Level Above Vessel Zero Per Active Single Failure m (ft)			Change in MCPR From Start of Event	Change in RPV Pressure 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)	8.47 (27.80)	8.36 (27.43)	8.76 (28.74)	No heatup	<1.0	7.46 (24.47)	7.47 (24.50)	7.43 (24.38)	Increases	Decreases
Feedwater Line[2]	0.07420 (0.7986)	8.37 (27.47)	8.26 (27.09)	8.35 (27.3)	No heatup	<1.0	6.36 (20.87)	6.69 (21.95)	6.69 (21.95)	Increases	Decreases
GDCS Injection Line	0.004561 (0.04910)	8.69 (28.52)	8.90 (29.19)	8.73 (28.64)	No heatup	<1.0	6.13 (20.09)	6.20 (20.34)	6.31 (20.71)	Increases	Decreases
Bottom Head Drain Line	0.004052 (0.04361)	8.35 (27.39)	8.62 (28.29)	8.42 (27.63)	No heatup	<1.0	5.97 (19.59)	6.27 (20.56)	6.26 (20.52)	Increases	Decreases
ICS Drain Line	0.01824 (0.1963)	8.40 (27.55)	8.55 (28.04)	8.56 (28.08)	No heatup	<1.0	5.95 (19.52)	6.04 (19.82)	5.85 (19.19)	Increases	Decreases
Based on bounding values:											
IC Drain Line	0.01824 (0.1963)	8.33 (27.33)	8.23 (27.00)	8.31 (27.26)	No heatup	<1.0	4.91 (16.11)	5.12 (16.78)	5.02 (16.49)	Increases	Decreases
GDCS Injection Line	0.004561 (0.04910)	8.82 (28.93)	8.34 (27.36)	8.87 (29.09)	No heatup	<1.0	5.65 (18.53)	5.60 (18.37)	5.59 (18.34)	Increases	Decreases

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

2 For the feedwater line break, the total break area from the TB side is limited at the two parallel venturi sections, with flow area of 0.04997 m^2 (0.5379 ft^2) each.

3 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 (25.91ft)) of bottom of chimney.

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

5 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 6.3-5a

Summary of ECCS Line Break Sizes and Elevations

Name	Qty	Elevation m (ft)*	Break Size m^2 (ft^2)	Notes
Main Steam Line	4	22.84 (74.93)	0.09832 (1.058)	Limited by venturi throat area
DPV/IC Line	4	21.91 (71.88)	0.08320 (0.8956)	Limited by venturi throat area (16” Sch.160)
Feedwater Line	6	18.915 (62.057)	0.07420 (0.7986)	Limited by FW nozzle area
RWCU/SDC Line	2	17.215 (56.480)	0.06558 (0.7059)	12” Sch.80 pipe
IC Return Line**	4	13.025 (42.733)	0.01824 (0.1963)	Limited by venturi throat area
GDCS Injection Line	8	10.453 (34.295)	0.004561 (0.04910)	Limited by venturi throat area
GDCS Equalizing Line	4	8.563 (28.09)	0.002027 (0.02182)	Limited by venturi throat area
Bottom Drain Line	4	0.0	0.004052 (0.04361)	2 nozzles

* Elevation referenced to RPV bottom.

** Isolation Condenser System condensate return line is referred throughout the chapter as “drain line” or “return line.”

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 SRV	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 4th 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 (s)	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 not credited in this analysis.
1	Feedwater Isolation signal activated. Feedwater valves are fully closed 15 seconds later.
2	Loss of normal auxiliary power confirmed. Reactor scram initiated. Rod insertion started at 0.25 seconds later.
Vent Clearing time	Top Vent: 1.95 seconds, Middle Vent: 2.40 seconds, Bottom Vent: 3.16 seconds
3	IC initiated from loss of power generation bus with 3 s signal delay time. Drain valves started to open at 15 seconds later.
5	Level 3 is reached (scram signal from Level 3 is not credited in this analysis).
12	Level 2 is reached (MSIV closure and IC initiation signals are not credited in this analysis).
17	Low MSL pressure setpoint reached. MSIV closure initiated in 0.7 seconds later.
21	Reactor is isolated on low MSL pressure setpoint.
332	Level 1 is reached.
342	Level 1 confirmed. ADS/GDCS/SLC timer initiated. SRV actuated.
392	DPV actuation begins at 50 seconds after confirmed Level 1 signal. SLC flow starts on DPV actuation.
492	GDCS timer (150 seconds after confirmed Level 1 signal) runs out. GDCS injection valves open.
514	Vessel pressure decreases below maximum injection pressure of GDCS. GDCS flow into the vessel begins. Chimney and downcomer water level start to rise.
573	Minimum chimney water level (8.26 m (27.1 ft)) is reached.
692	SLC flow depleted.
From 1000 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 an Inside Steam Line Break with Failure of One GDCS Injection Valve (Nominal Calculation)

Time (s)	Events
0	Guillotine break of MSL 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 DW pressure setpoint reached. Scram signal not credited in this analysis.
2	Loss of normal auxiliary power confirmed. Reactor scram initiated. Rod insertion started at 0.25 seconds later.
Vent Clearing time	Top Vent: 1.41 seconds, Middle Vent: 1.78 seconds, Bottom Vent: 2.34 seconds
3	IC initiated from loss of power generation bus with 3 seconds signal delay time. Drain valves started to open at 15 seconds later.
6	Low MSL pressure setpoint reached. MSIV closure initiated in 0.7 seconds 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 (MSIV closure and IC initiation signals are not credited in this analysis).
456	Level 1 is reached.
466	Level 1 confirmed. ADS/GDCS/SLC timer initiated. SRV actuated.
516	DPV actuation begins at 50 seconds after confirmed Level 1 signal. SLC flow starts on DPV actuation.
616	GDCS timer (150 s after confirmed Level 1 signal) runs out. GDCS injection valves open.
617	GDCS flow into the vessel begins.
627	Minimum chimney water level (8.36 m (27.43 ft)) is reached.
816	SLC flow depleted.
From 1000 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 Injection Line Break with Failure of One GDCS Injection Valve (Nominal Calculation)

Time (s)	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 seconds 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.
3	High DW pressure setpoint reached. Scram signal not credited in this analysis.
Vent Clearing time	Top Vent: 32.94 seconds, Middle Vent: 284.69 seconds, Bottom Vent: never cleared
7	Level 3 is reached (scram signal from Level 3 is not credited in this analysis).
16	Level 2 is reached (MSIV closure and IC initiation signals are not credited in this analysis).
20	Low MSL pressure setpoint reached. MSIV closure initiated in 0.7 seconds later.
24	Reactor is isolated on low MSL pressure setpoint.
220	Level 1 is reached.
231	Level 1 confirmed. ADS/GDCS/SLC timer initiated. SRV actuated.
281	DPV actuation begins at 50 seconds after confirmed Level 1 signal. SLC flow starts on DPV actuation.
380	GDCS timer (150 seconds after confirmed Level 1 signal) runs out. GDCS injection valves open.
488	Vessel pressure decreases below maximum injection pressure of GDCS. GDCS flow into the vessel begins. Chimney and downcomer water level start to rise.
580	SLC flow depleted.
608	Minimum chimney water level (8.90 m (29.19 ft)) 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 (s)	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 seconds 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.
6	High Drywell pressure setpoint reached. Scram signal not credited in this analysis.
Vent Clearing time	Top Vent: 285.38 seconds, Middle Vent: 426.48 seconds, Bottom Vent: never cleared
7	Level 3 is reached (scram signal from Level 3 is not credited in this analysis).
17	Level 2 is reached (MSIV closure and IC initiation signals are not credited in this analysis).
20	Low MSL pressure setpoint reached, MSIV closure initiated in 0.7 seconds later.
24	Reactor is isolated on low MSL pressure setpoint.
362	Level 1 is reached.
372	Level 1 confirmed. ADS/GDCS/SLC timer initiated. SRV actuated.
422	DPV actuation begins at 50 seconds after confirmed Level 1 signal. SLC flow starts on DPV actuation.
522	GDCS timer (150 s after confirmed Level 1 signal) runs out. GDCS injection valves open.
552	Minimum chimney water level, 8.623 m (28.29 ft) is reached.
633	Vessel pressure decreases below maximum injection pressure of GDCS. GDCS flow into the vessel begins. Chimney and downcomer water level

Table 6.3-10

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

Time (s)	Events
	start to rise.
722	SLC flow depleted.
From 1006 to 2000	RPV water level remains higher than Level 0.5. Therefore equalizing line valves do not open for this event.

Table 6.3-10a

Operational Sequence of ECCS for a IC Drain Line Break with Failure of One GDCS Valve (Bounding Case)

Time (s)	Events
0	Guillotine break of IC 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.
1	High DW 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 seconds 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.
Vent clearing time	Top Vent: 3.3 seconds, Middle Vent: 4.9 seconds, Bottom Vent: never cleared.
6	Level 3 is reached (scram signal from Level 3 is not credited in this analysis).
17	Level 2 is reached (MSIV closure and IC initiation signals are not credited in this analysis).
20	Low MSL pressure setpoint reached, MSIV closure initiated at 0.7 seconds later.
24	Reactor isolated on low MSL pressure setpoint.
397	Level 1 is reached.
407	Level 1 signal confirmed. ADS/GDCS/SLCS timer initiated. SRV actuated.
457	DPV actuation begins at 50 seconds after confirmed Level 1 signal; SLC system flow starts on DPV actuation.
557	GDCS timer (150 seconds after confirmed Level 1 signal) timed out. GDCS injection valves open.
616	Vessel pressure decreases below maximum injection pressure of GDCS. GDCS flow into the vessel begins.
703	Minimum chimney water level (8.23 m (27.02 ft)) is reached.

Table 6.3-10a

Operational Sequence of ECCS for a IC Drain Line Break with Failure of One GDCS Valve (Bounding Case)

Time (s)	Events
757	SLC system flow depleted.
From 884 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.	Peak LHGR	44.0 kW/m (13.4 kW/ft)	44.8 kW/m (13.7 kW/ft)
5.	Initial MCPR	1.25	1.22
6.	Initial Downcomer Water Level	NWL	NWL – 0.3m (NWL – 0.98 ft)
7.	Significant TRACG Modeling Parameters**	Nominal	Bounding

* Represents upper 95% or higher probability value.
** Reference 6.3-2, Table 2.5-2.

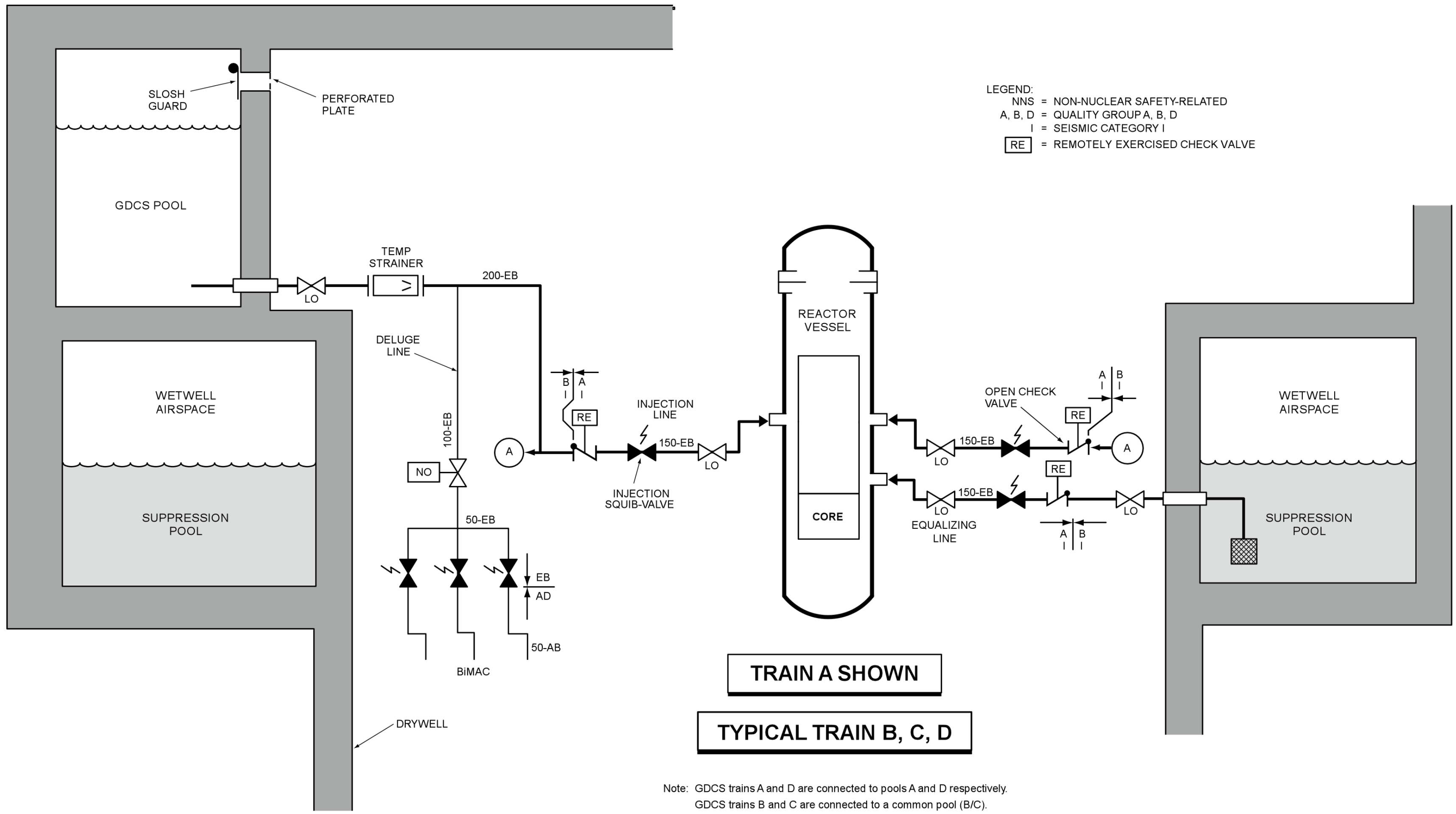
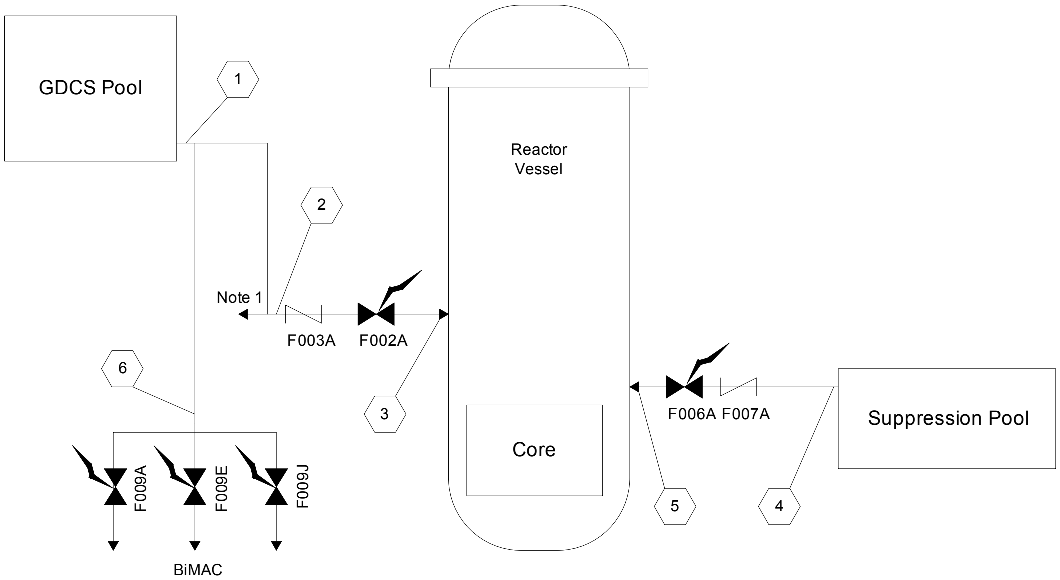


Figure 6.3-1. GDCS Configuration



Mode A: Standby						
Position ⬡	1	2	3	4	5	6
Temperature °C (°F)	43 (110)	43 (110)	288 (550)	43 (110)	288 (550)	43 (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)	162 (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 19.

Figure 6.3-1a. GDCS Typical Process Flows

Figure 6.3-2. (Deleted)

Figure 6.3-3. (Deleted)

Figure 6.3-4. (Deleted)

Figure 6.3-5. (Deleted)

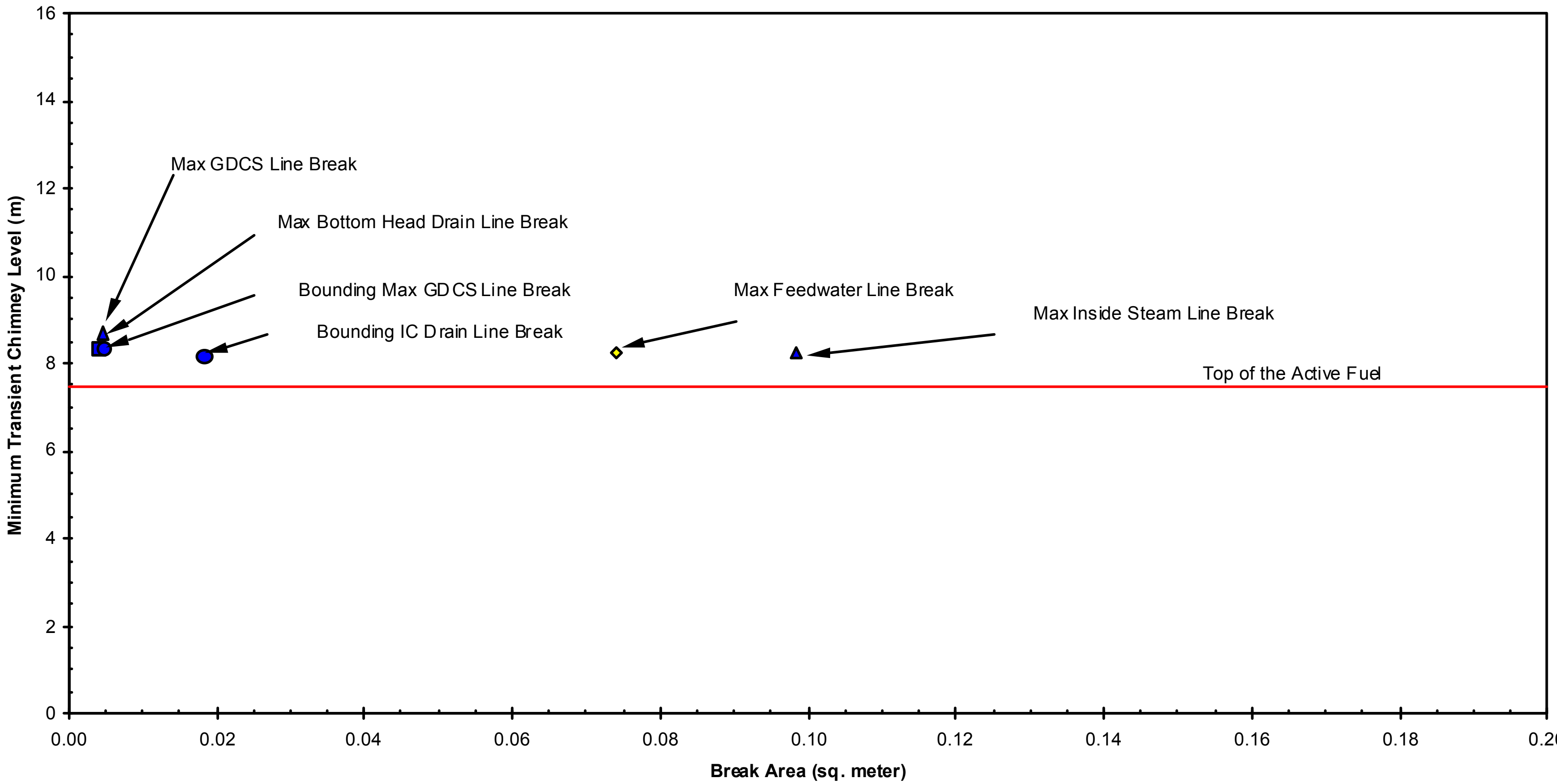


Figure 6.3-6. Minimum Transient Chimney Water Level vs. Break Area

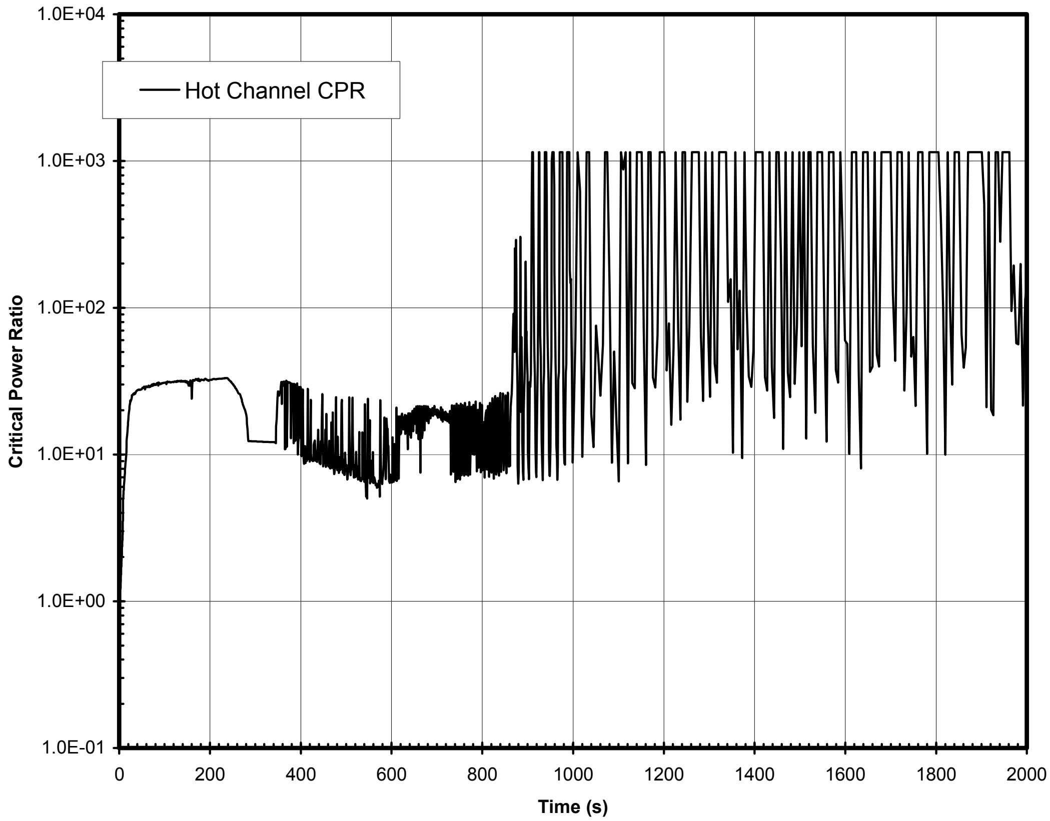


Figure 6.3-7a. MCPR, Feedwater Line Break (Nominal Case), 1 GDCS Valve Failure (2000 s)

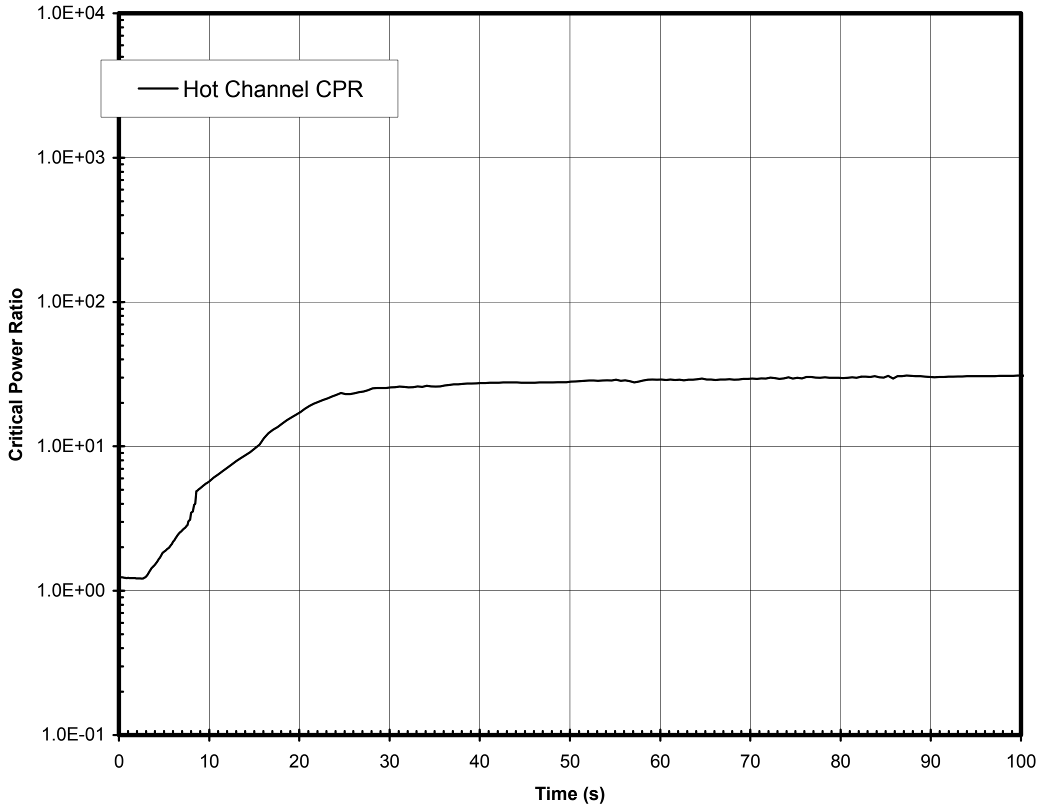


Figure 6.3-7b. MCPR, Feedwater Line Break (Nominal Case), 1 GDCS Valve Failure (100 s)

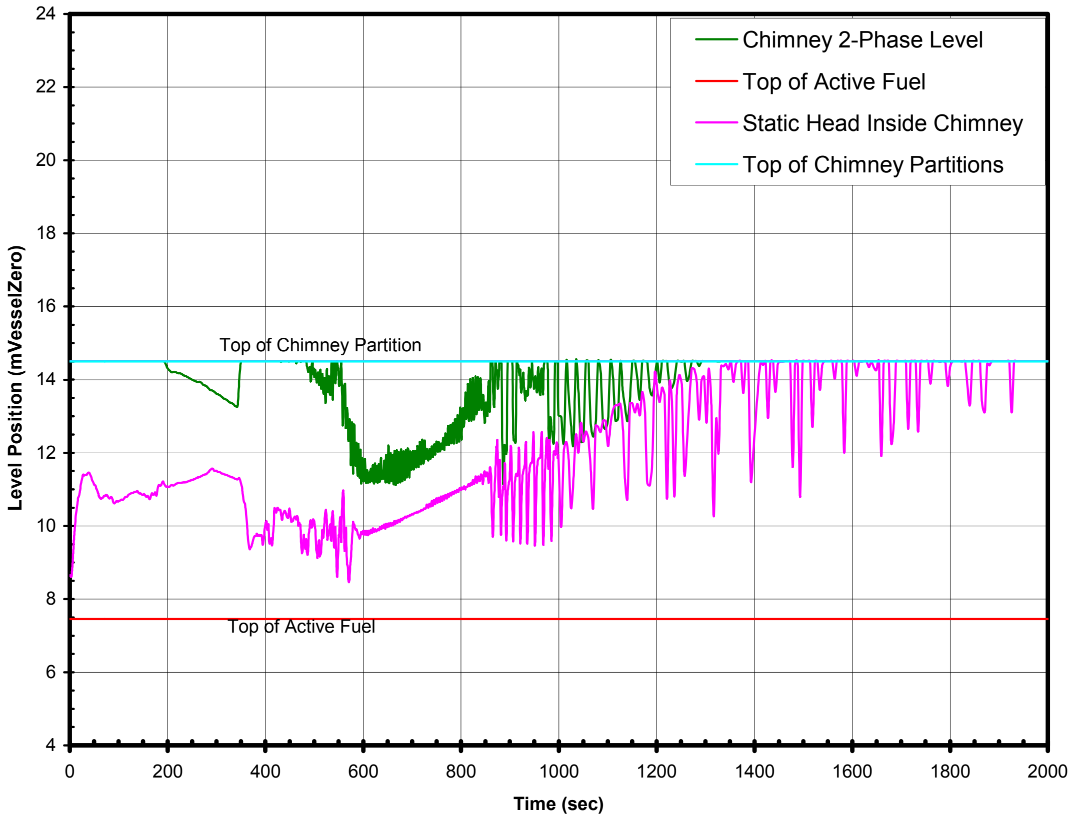


Figure 6.3-8a. Chimney Water Level, Feedwater Line Break (Nominal Case), 1 GDCS Valve Failure (2000 s)

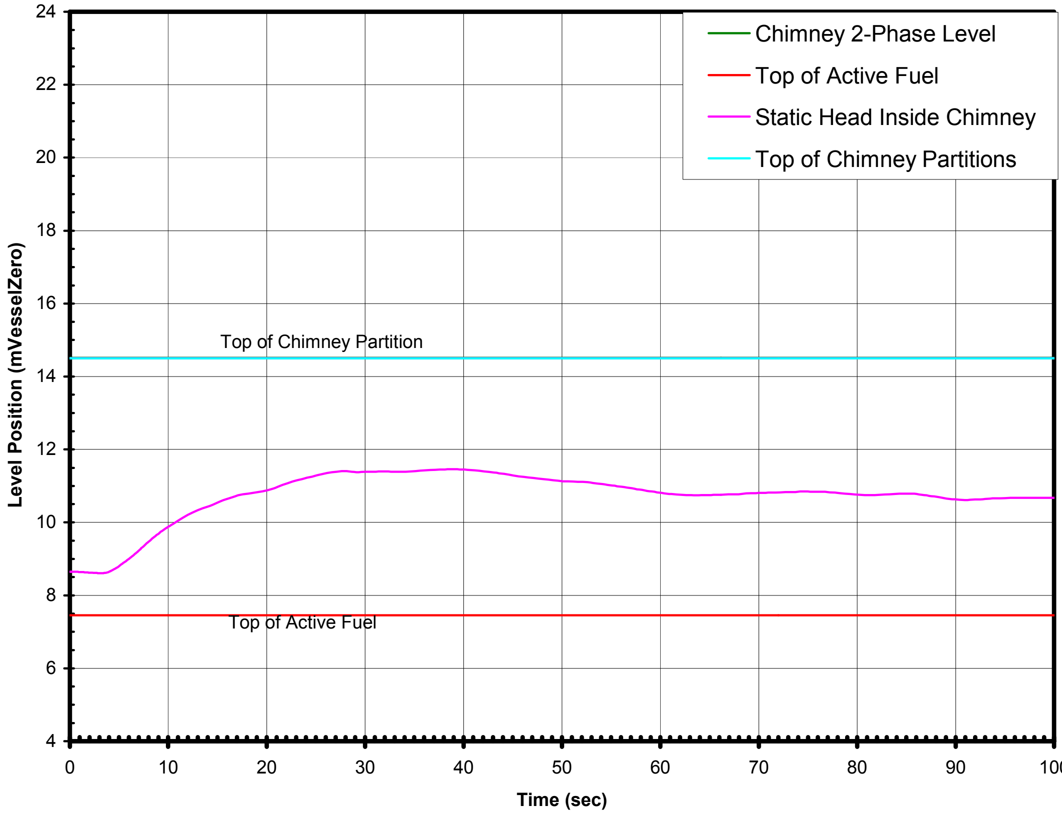


Figure 6.3-8b. Chimney Water Level, Feedwater Line Break (Nominal Case), 1 GDCS Valve Failure (100 s)

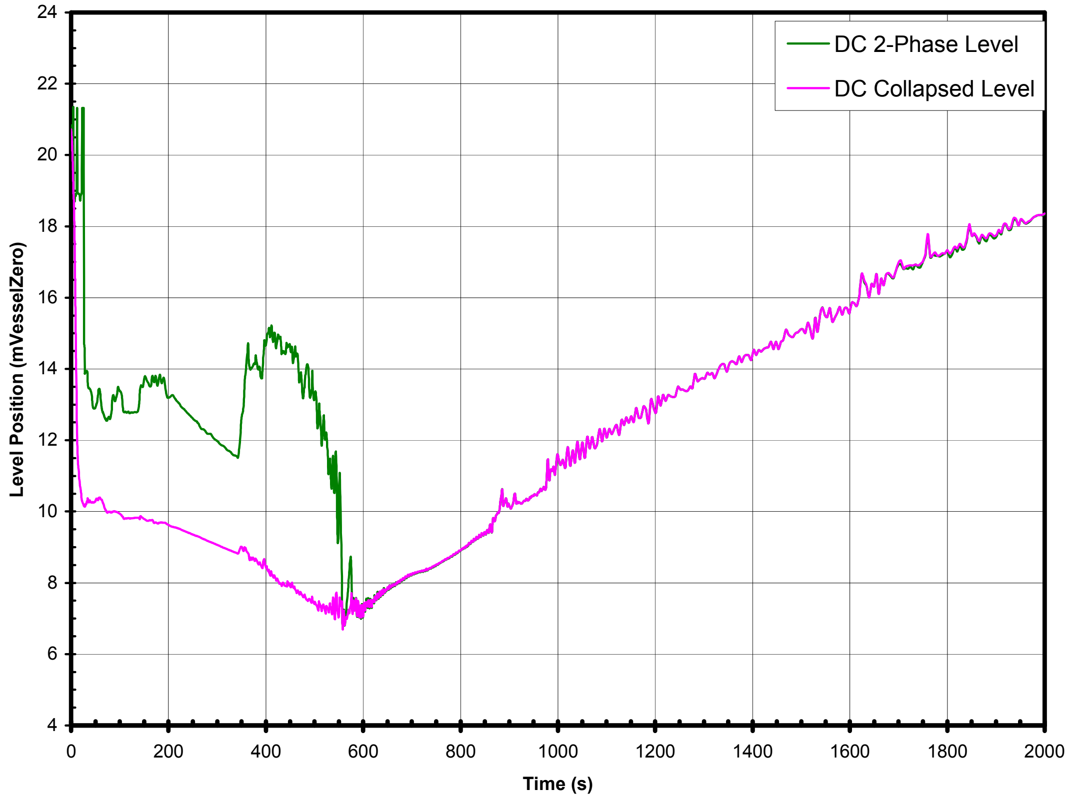


Figure 6.3-9a. Downcomer Water Level, Feedwater Line Break (Nominal Case), 1 GDCS Valve Failure (2000 s)

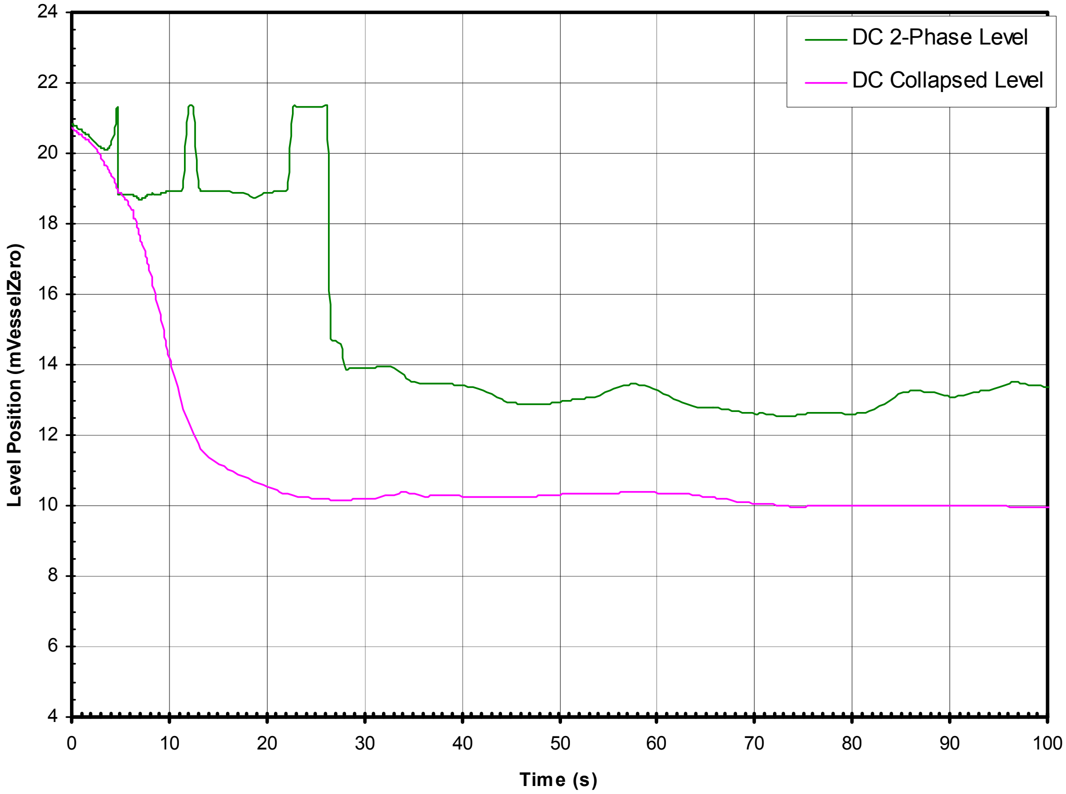


Figure 6.3-9b. Downcomer Water Level, Feedwater Line Break (Nominal Case), 1 GDCS Valve Failure (100 s)

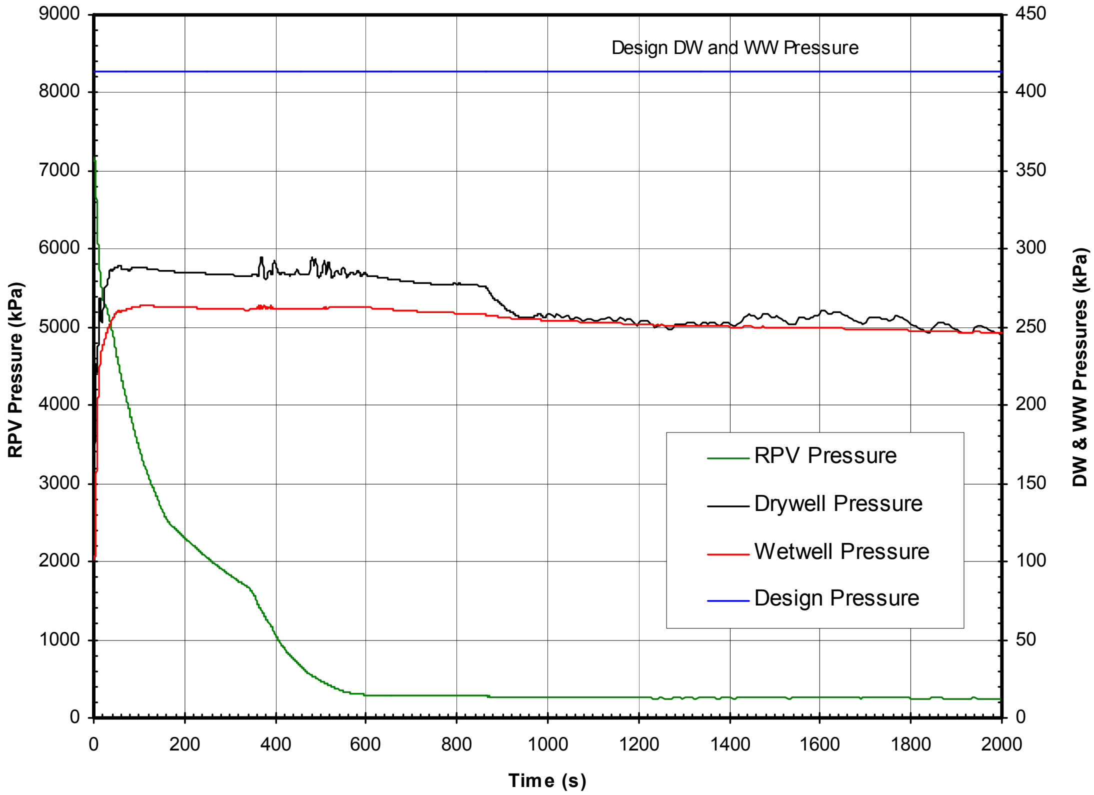


Figure 6.3-10a. System Pressures, Feedwater Line Break (Nominal Case), 1 GDCS Valve Failure (2000 s)

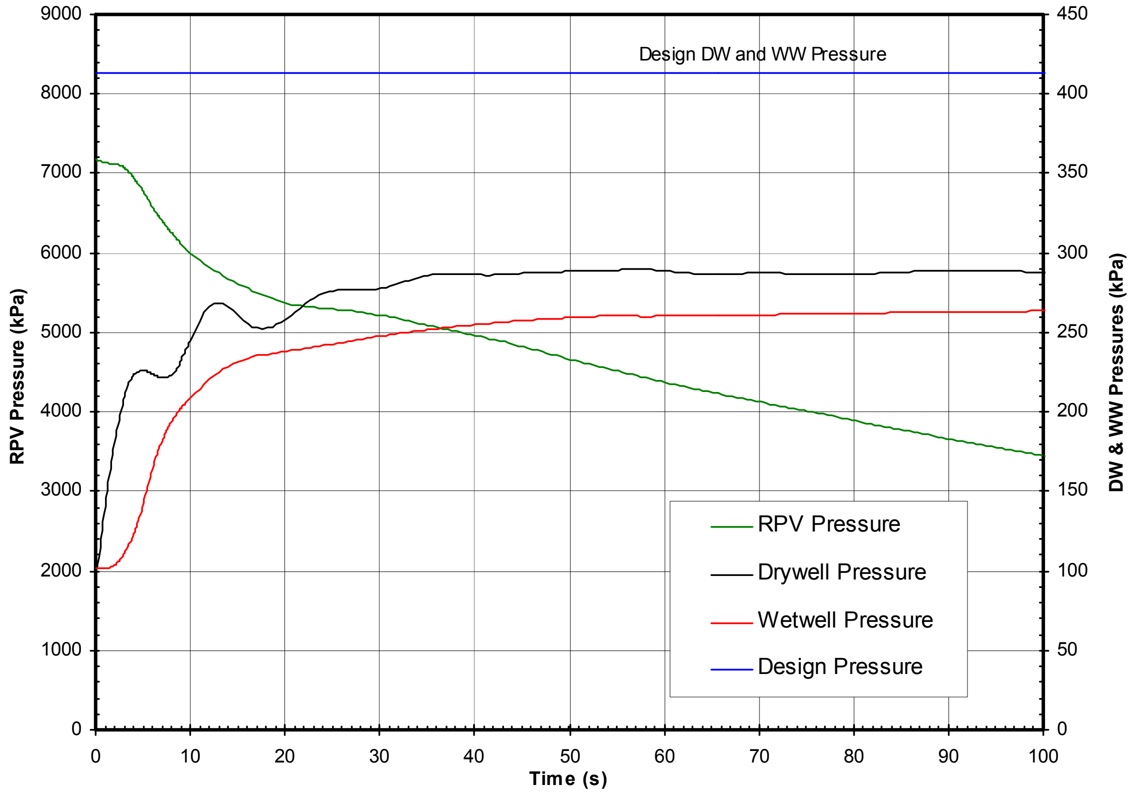


Figure 6.3-10b. System Pressures, Feedwater Line Break (Nominal Case), 1 GDCS Valve Failure (100 s)

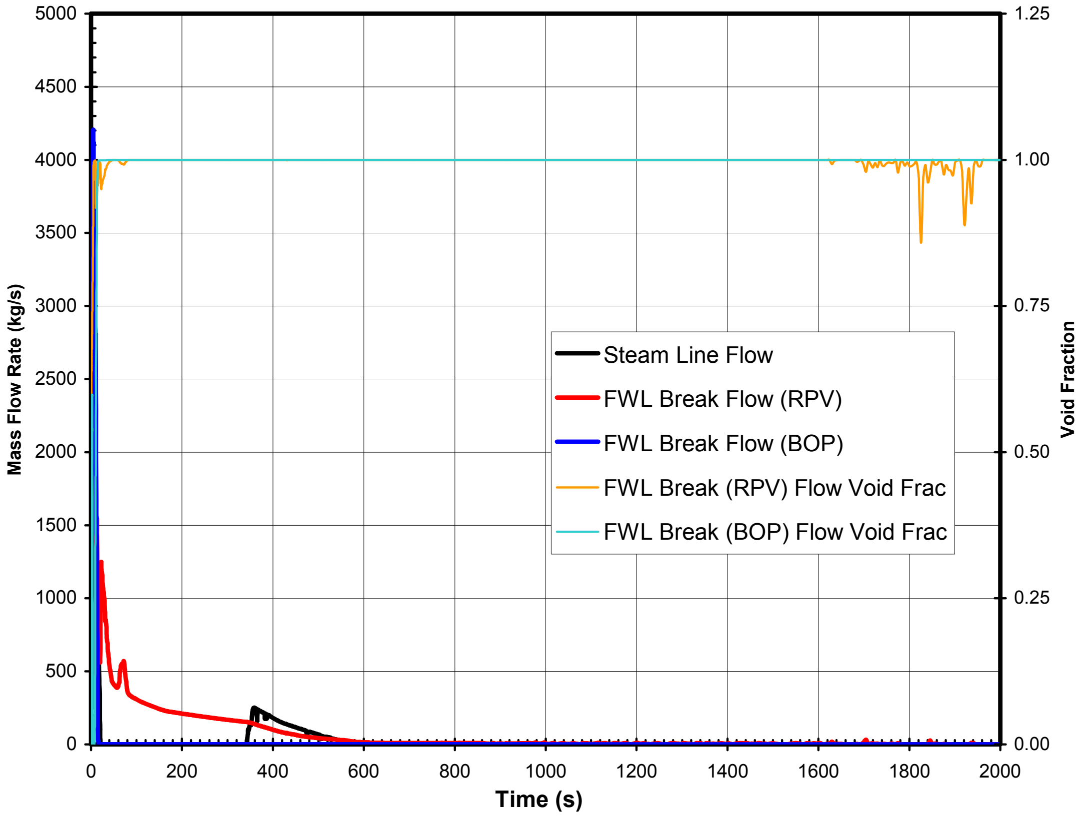


Figure 6.3-11a. Steam Line and Break Flows, Feedwater Line Break (Nominal Case), 1 GDCS Valve Failure (2000 s)

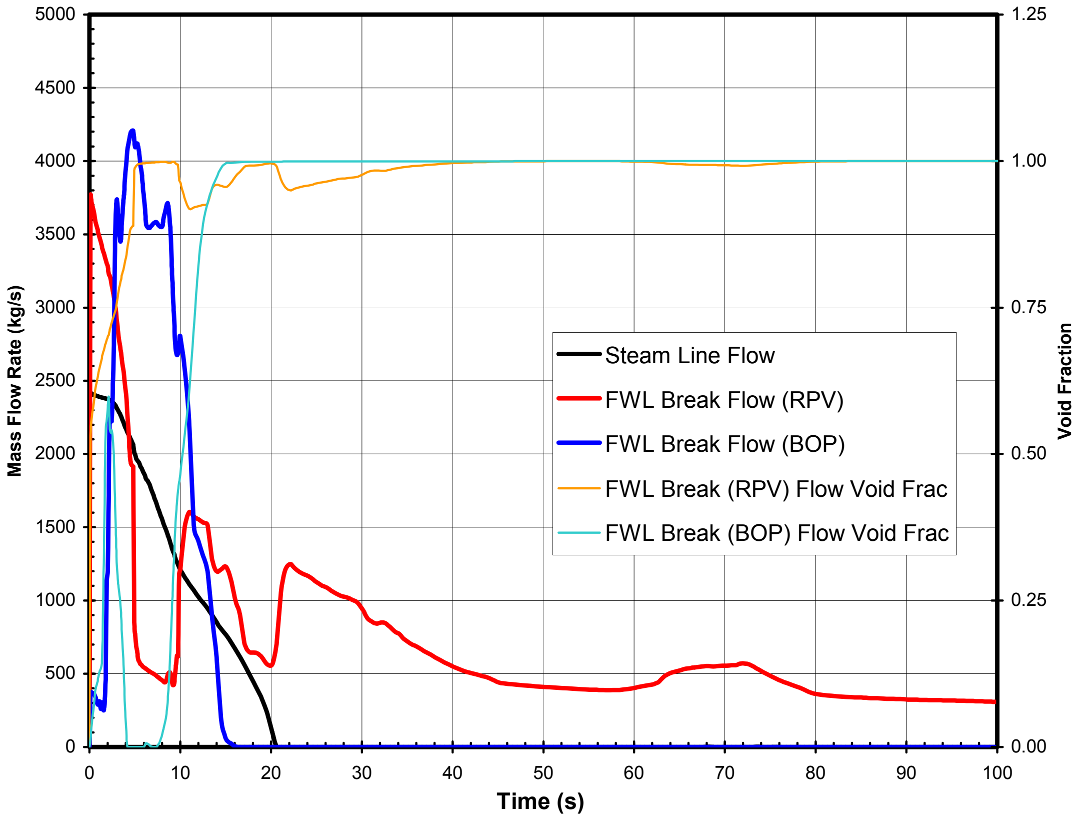


Figure 6.3-11b. Steam Line and Break Flows, Feedwater Line Break (Nominal Case), 1 GDCS Valve Failure (100 s)

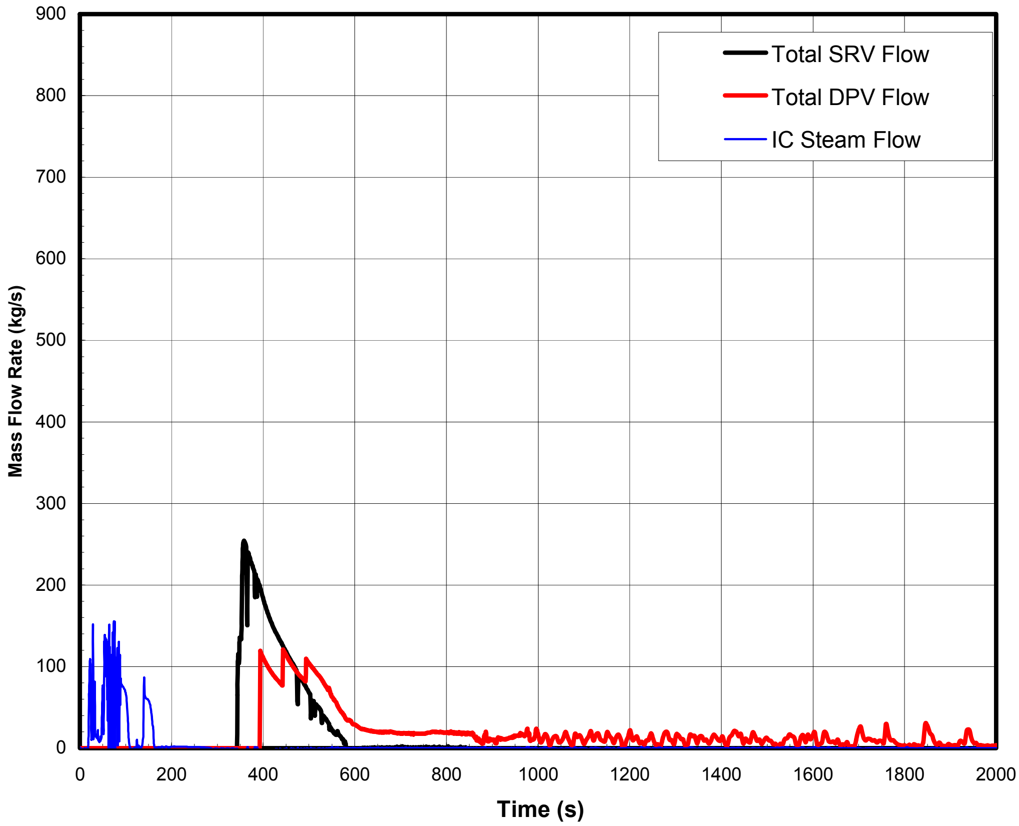


Figure 6.3-12a. ADS Flows, Feedwater Line Break (Nominal Case), 1 GDCS Valve Failure (2000 s)

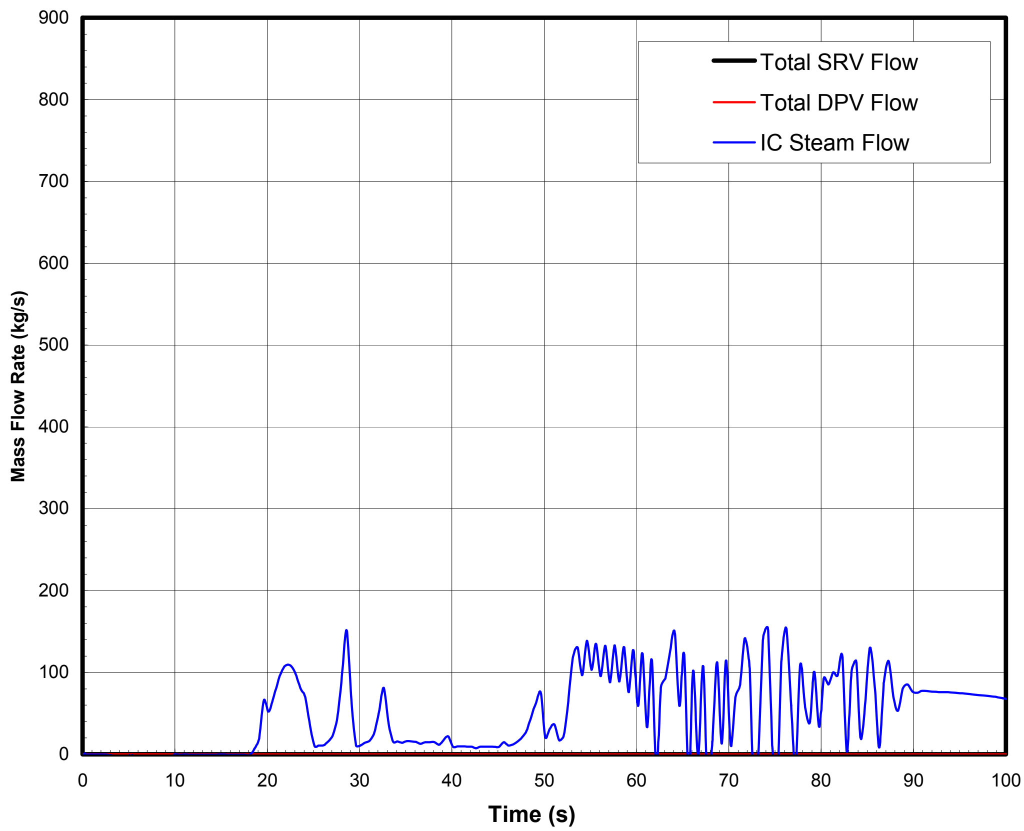


Figure 6.3-12b. ADS Flows, Feedwater Line Break (Nominal Case), 1 GDCS Valve Failure (100 s)

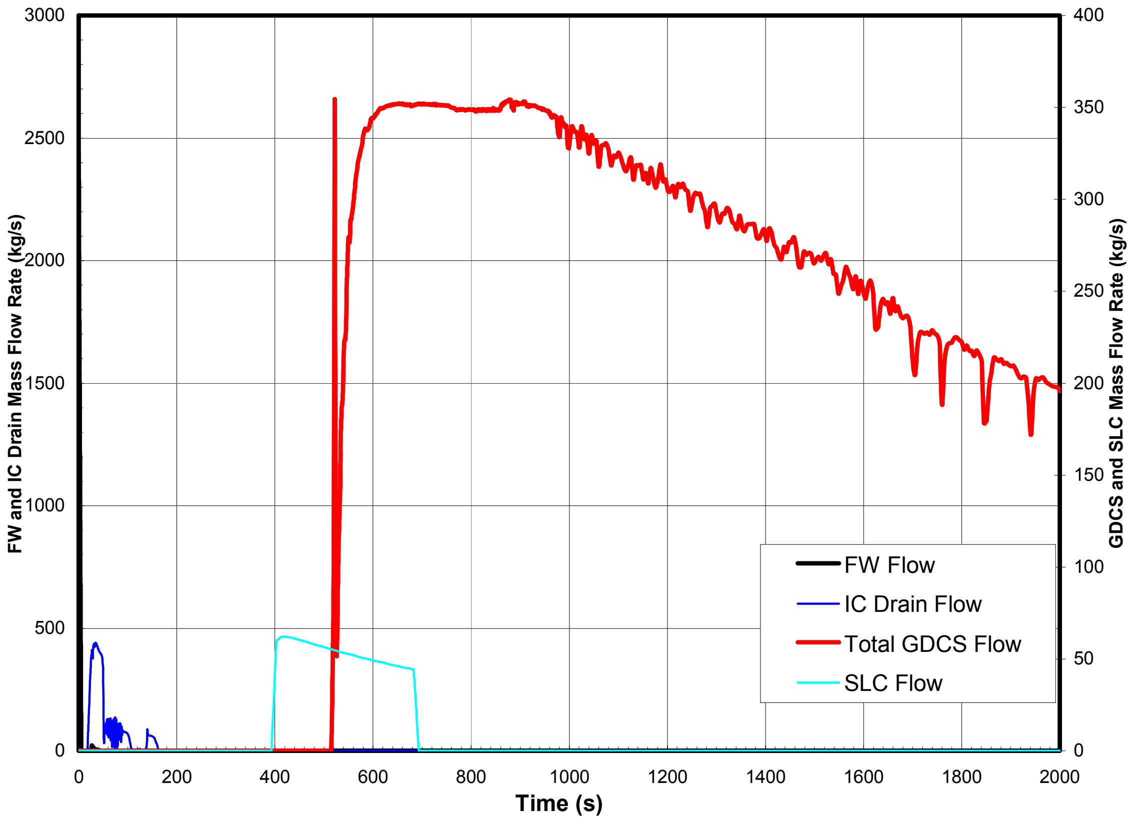


Figure 6.3-13a. Flows Into Vessel, Feedwater Line Break (Nominal Case), 1 GDCS Valve Failure (2000 s)

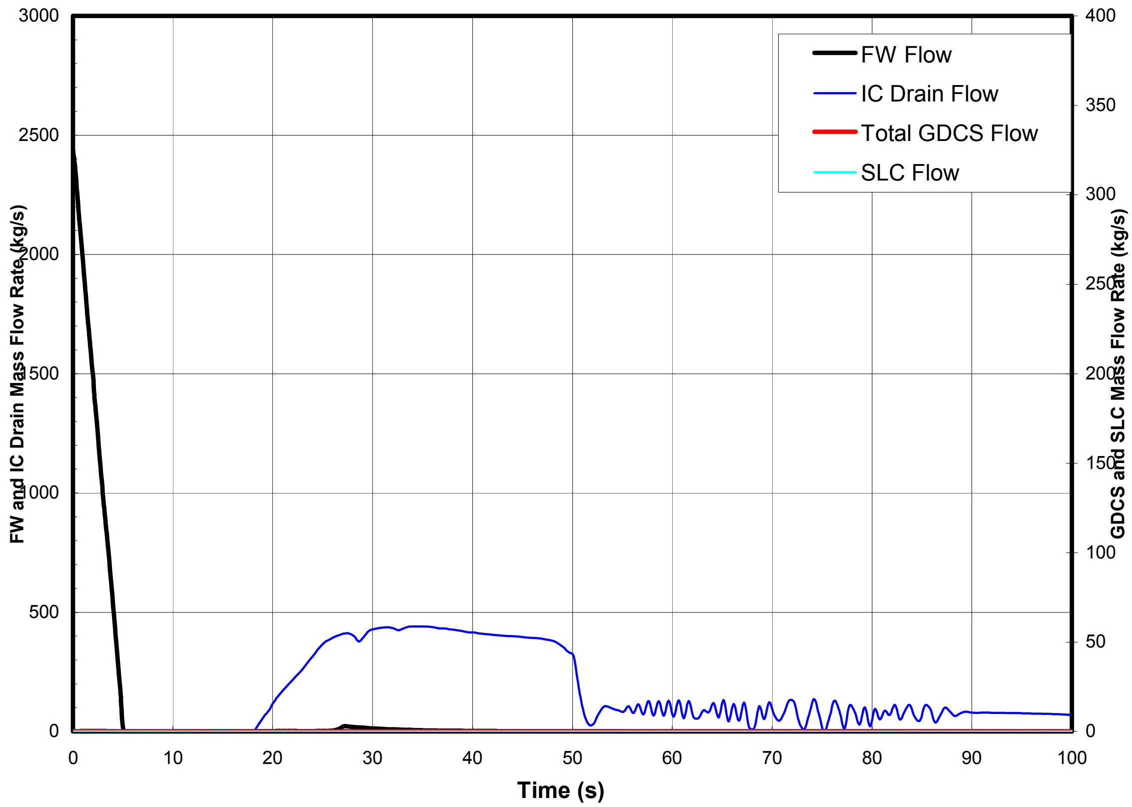
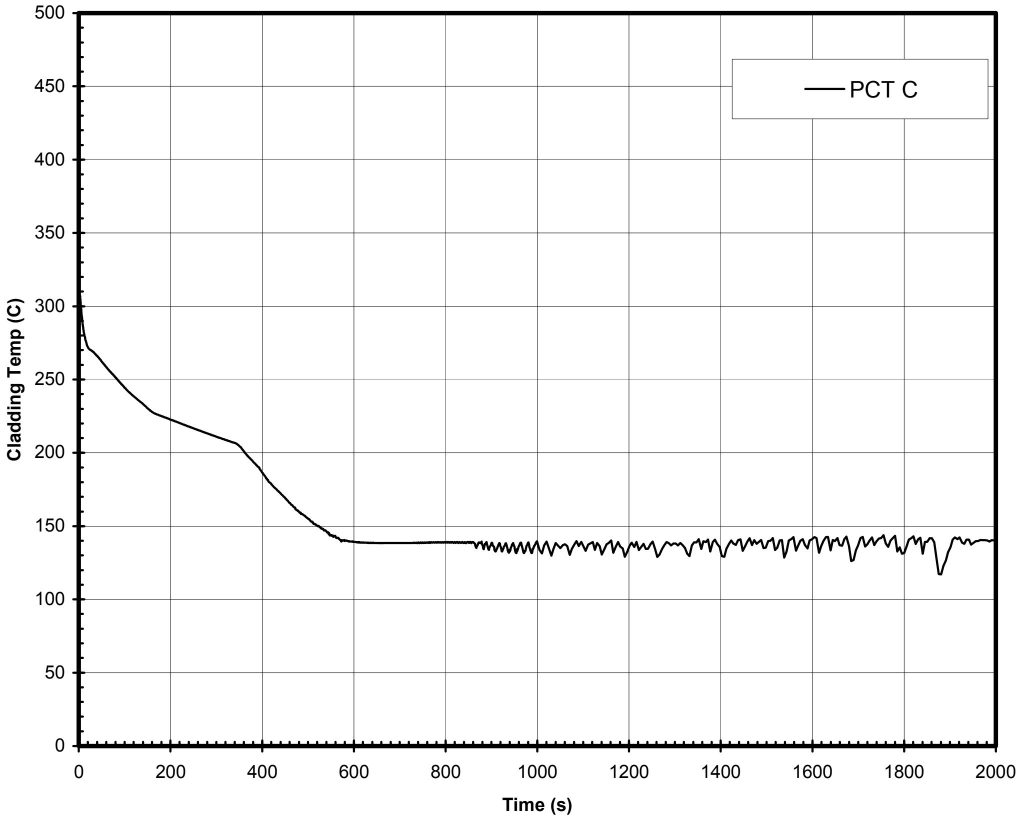
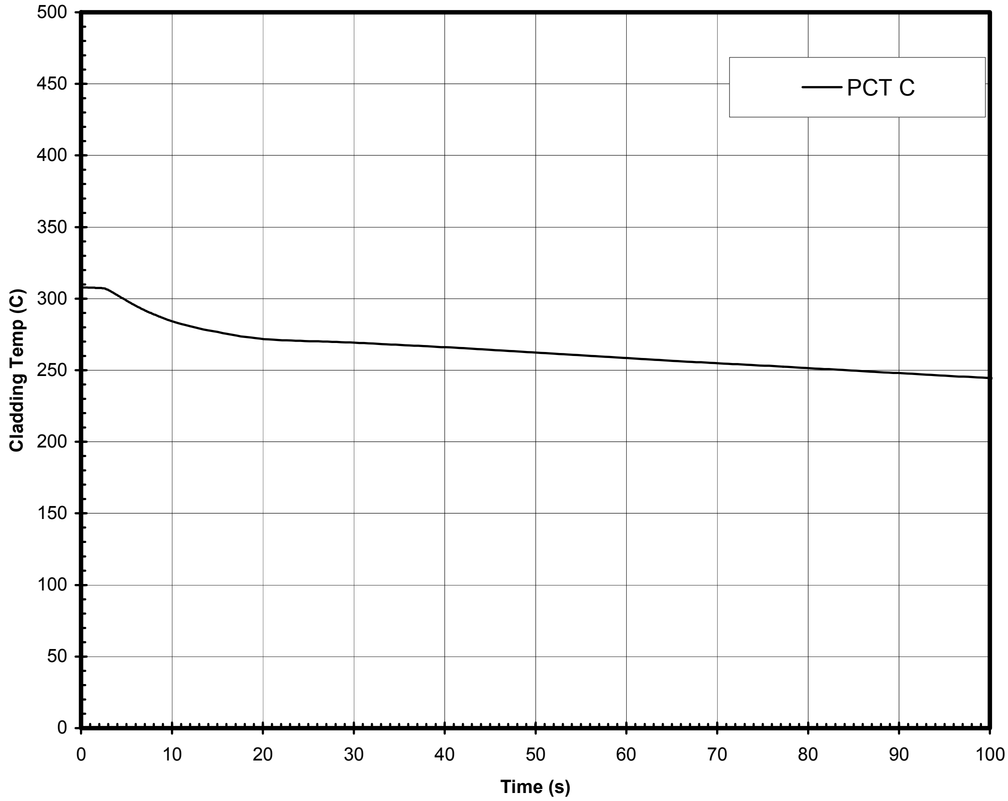


Figure 6.3-13b. Flows Into Vessel, Feedwater Line Break (Nominal Case), 1 GDCS Valve Failure (100 s)



Legend: PCT C = Peak Cladding Temperature, ^{o}C

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



Legend: PCT C = Peak Cladding Temperature, °C

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

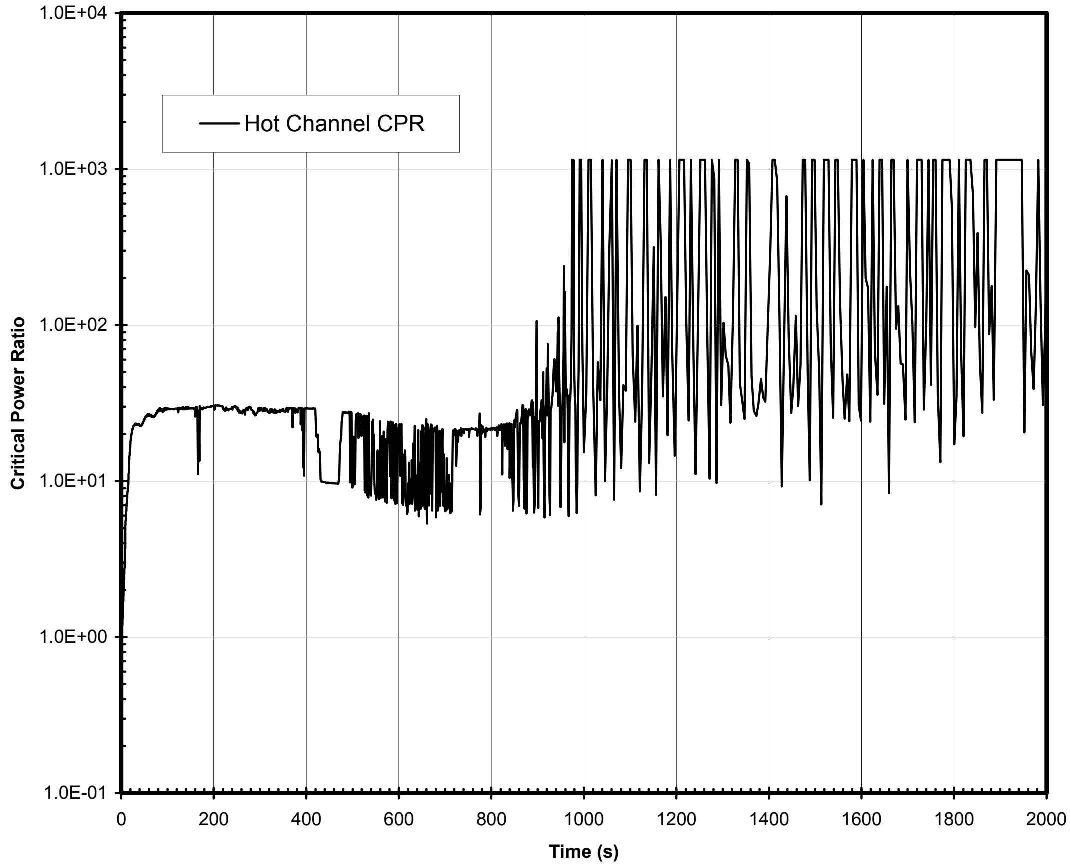


Figure 6.3-15a. MCPR, Inside Steam Line Break (Nominal Case), 1 GDCS Valve Failure (2000 s)

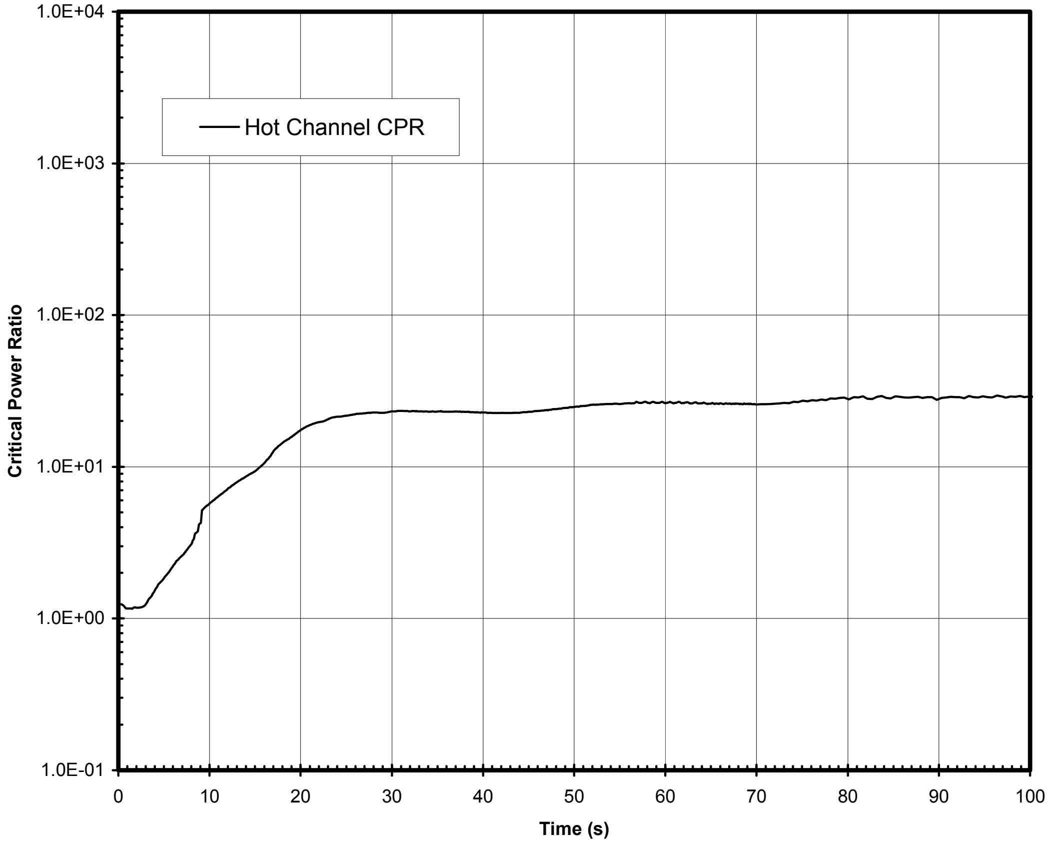


Figure 6.3-15b. MCPR, Inside Steam Line Break (Nominal Case), 1 GDCS Valve Failure (100 s)

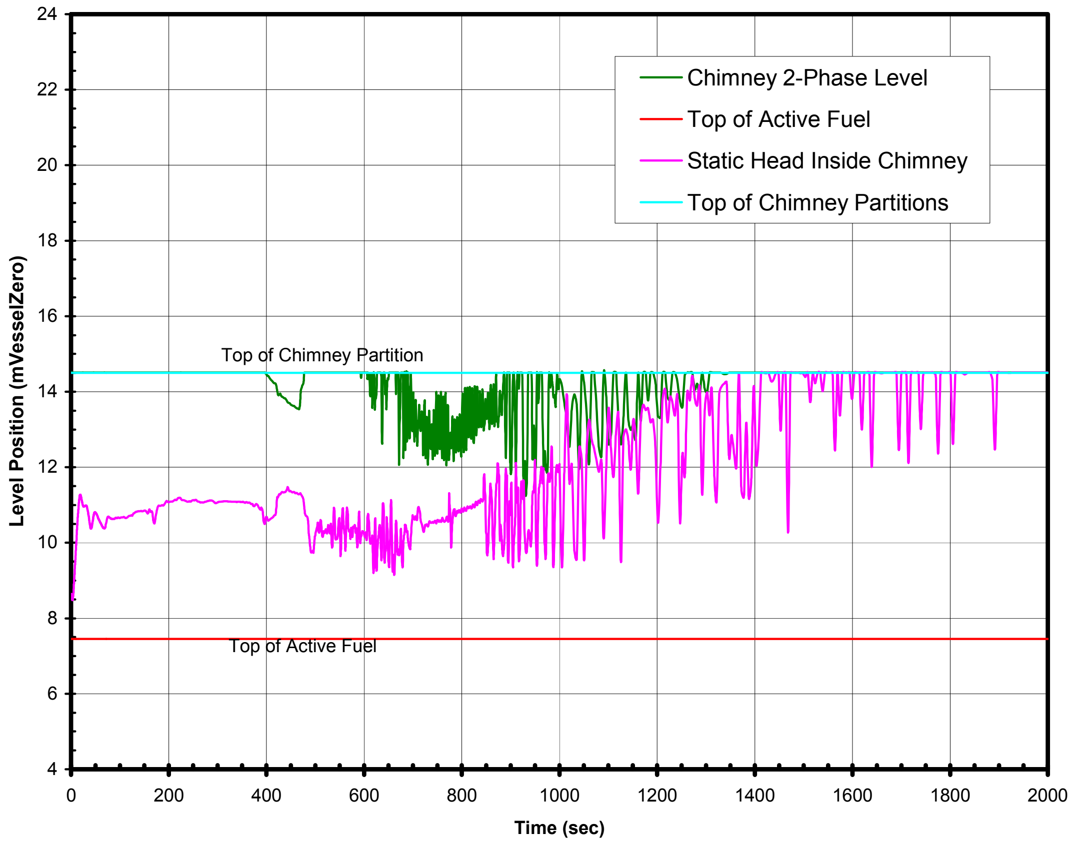


Figure 6.3-16a. Chimney Water Level, Inside Steam Line Break (Nominal Case), 1 GDCS Valve Failure (2000 s)

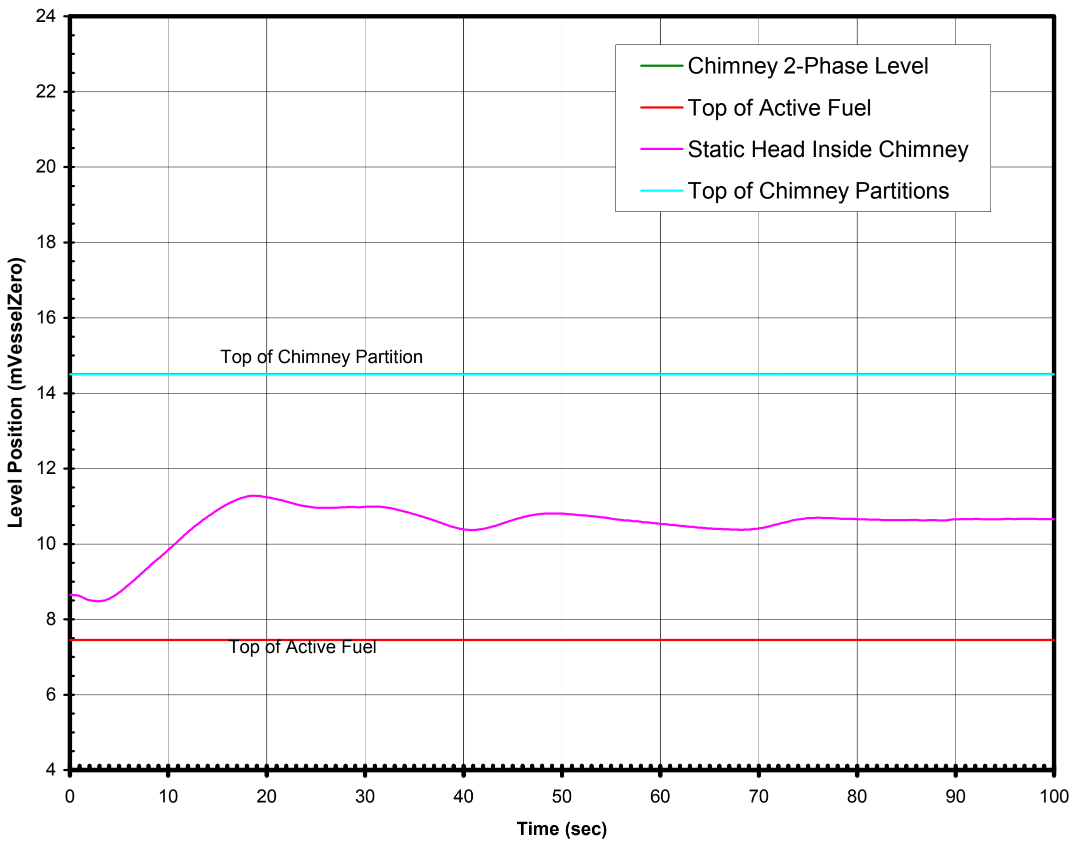


Figure 6.3-16b. Chimney Water Level, Inside Steam Line Break (Nominal Case), 1 GDCS Valve Failure (100 s)

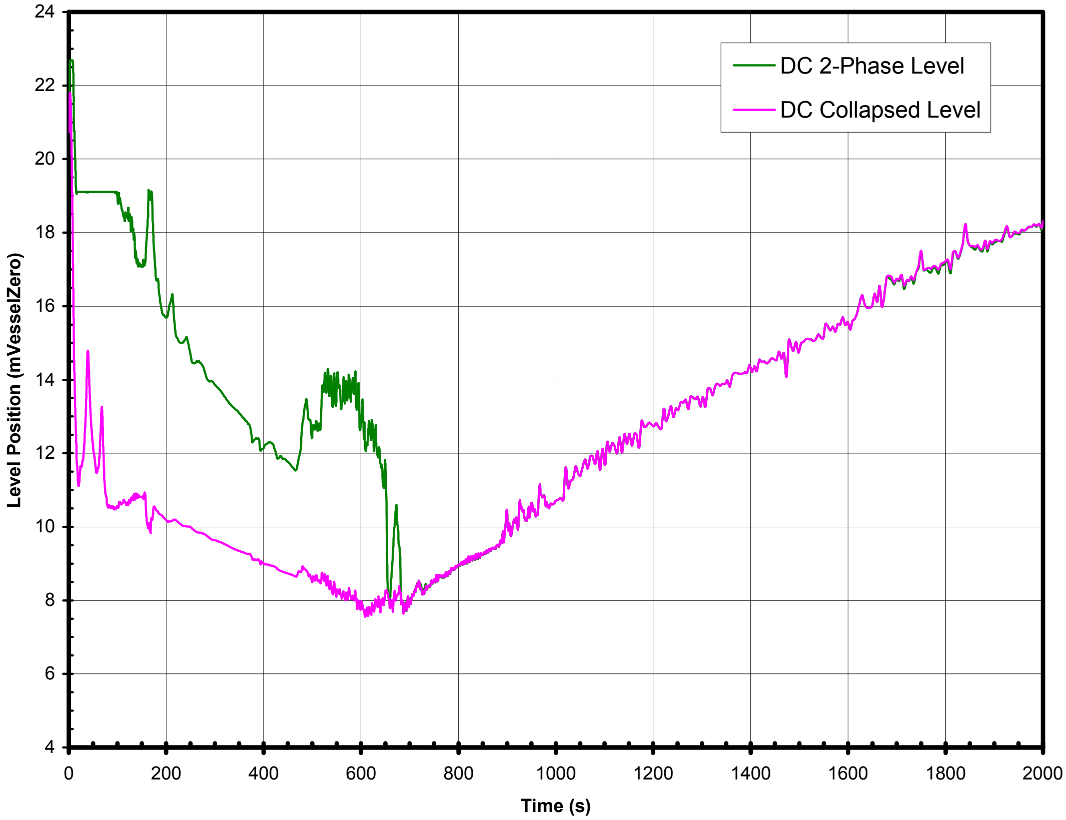


Figure 6.3-17a. Downcomer (DC) Water Level, Inside Steam Line Break (Nominal Case), 1 GDCS Valve Failure (2000 s)

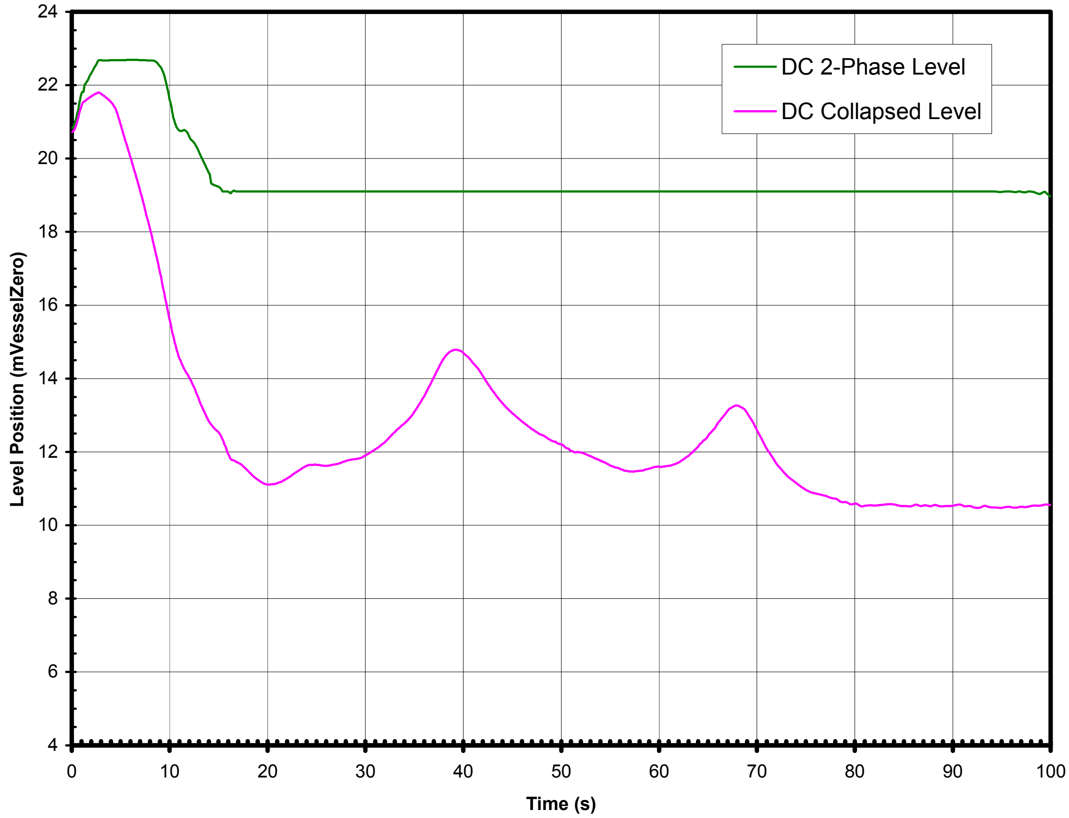


Figure 6.3-17b. Downcomer (DC) Water Level, Inside Steam Line Break (Nominal Case), 1 GDCS Valve Failure (100 s)

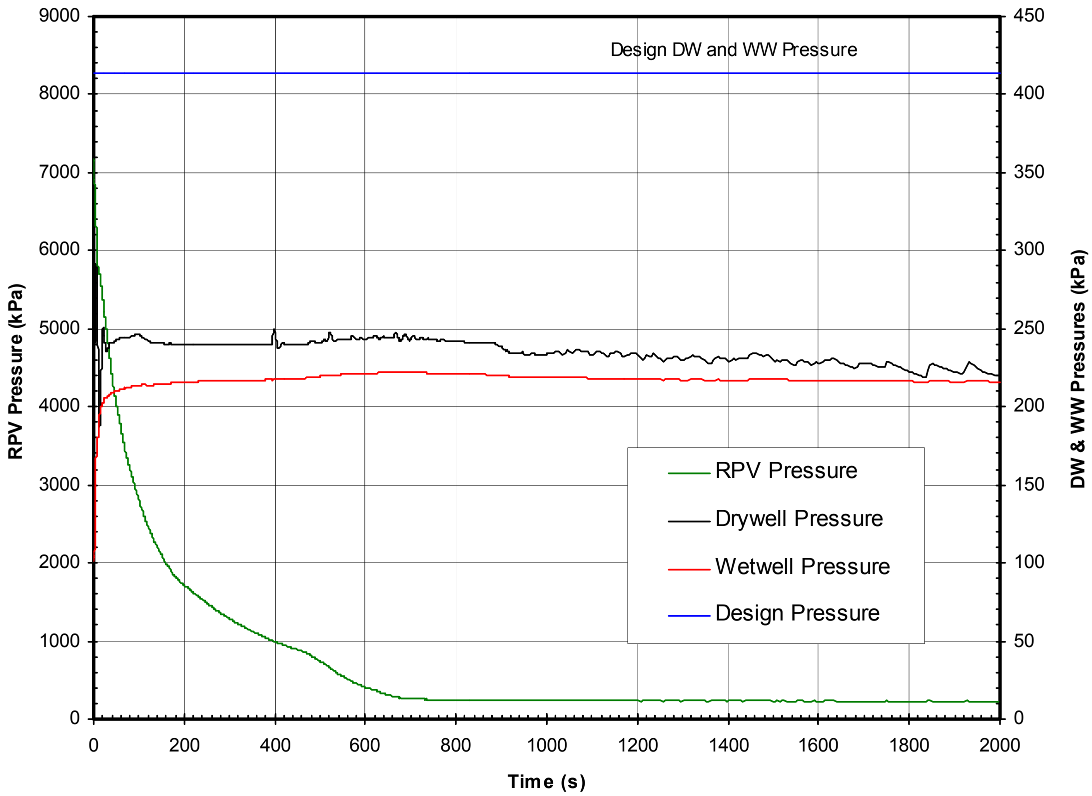


Figure 6.3-18a. System Pressures, Inside Steam Line Break (Nominal Case), 1 GDCS Valve Failure (2000 s)

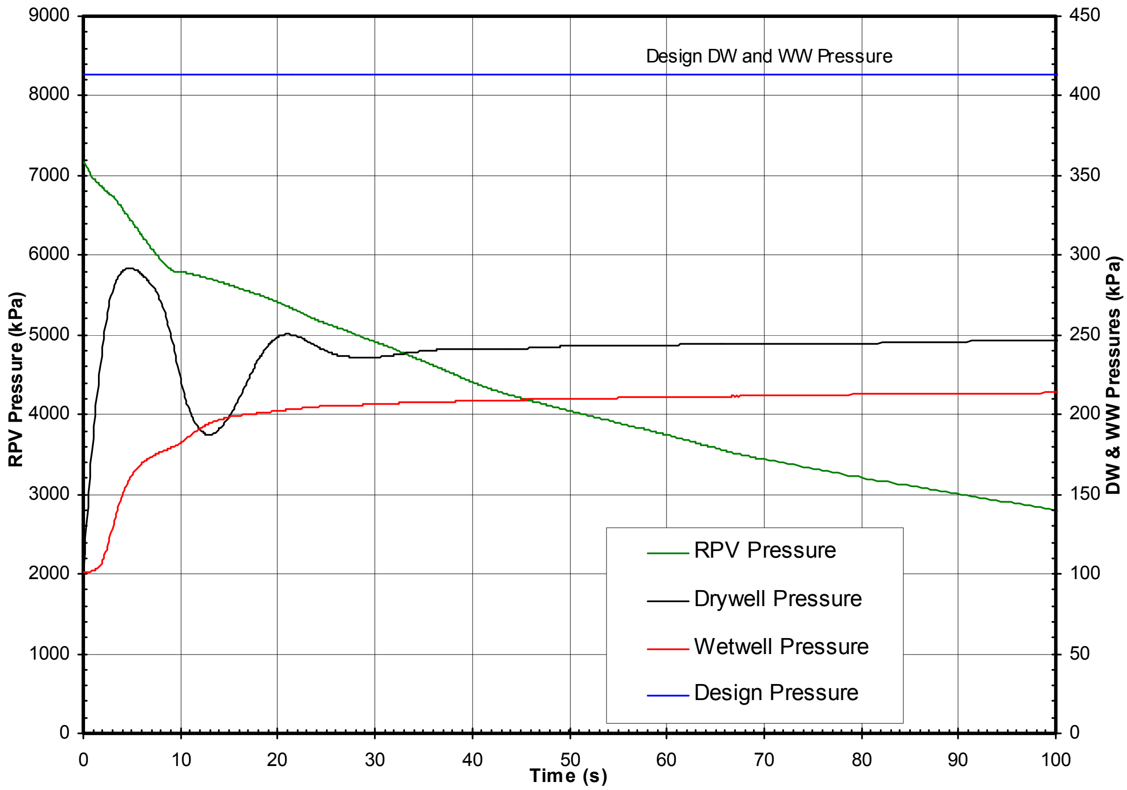


Figure 6.3-18b. System Pressures, Inside Steam Line Break (Nominal Case), 1 GDCS Valve Failure (100 s)

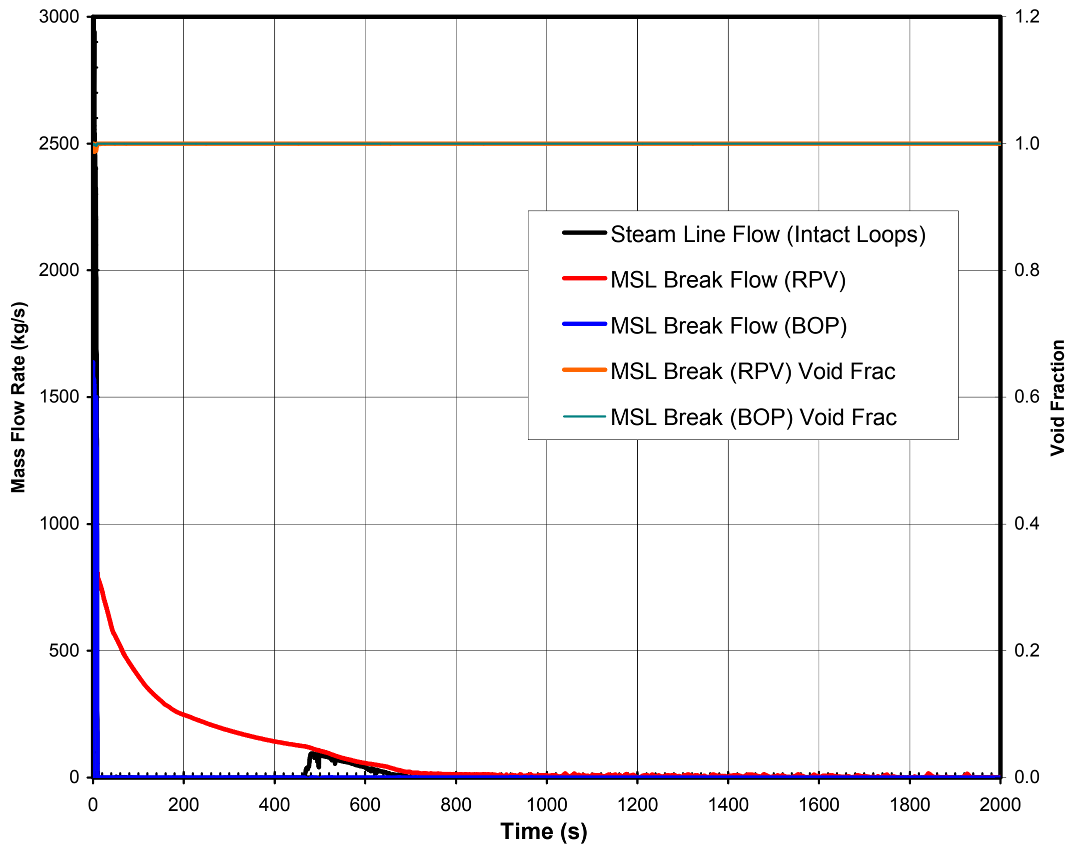


Figure 6.3-19a. Steam Line and Break Flow with Void Fraction, Inside Steam Line Break (Nominal Case), 1 GDCS Valve Failure (2000 s)

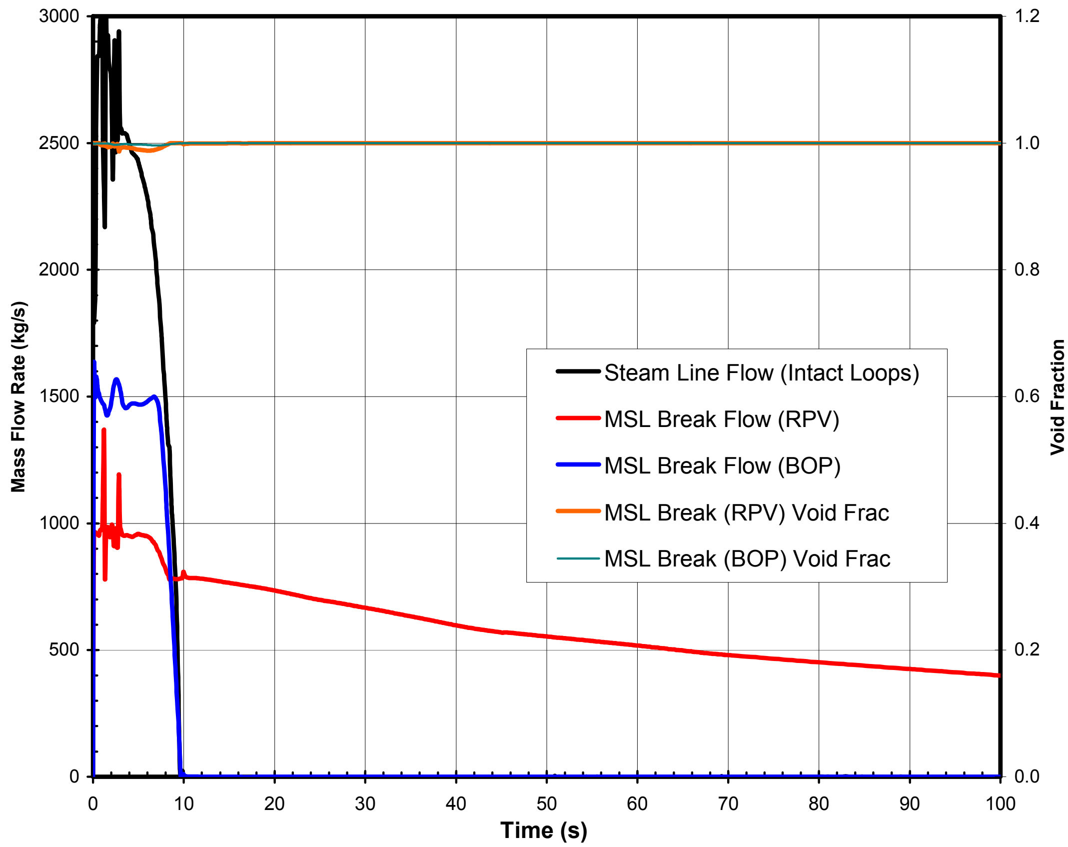


Figure 6.3-19b. Steam Line and Break Flow with Void Fraction, Inside Steam Line Break (Nominal Case), 1 GDCS Valve Failure (100 s)

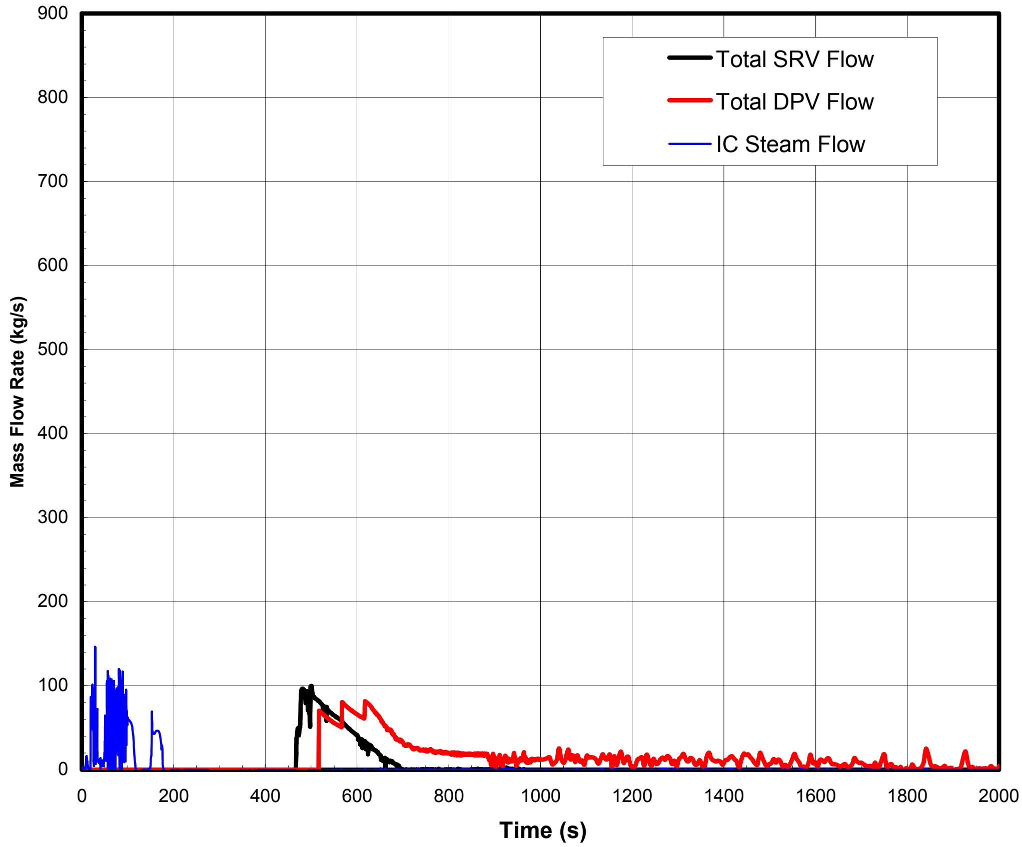


Figure 6.3-20a. ADS Flow, Inside Steam Line Break (Nominal Case), 1 GDCS Valve Failure (2000 s)

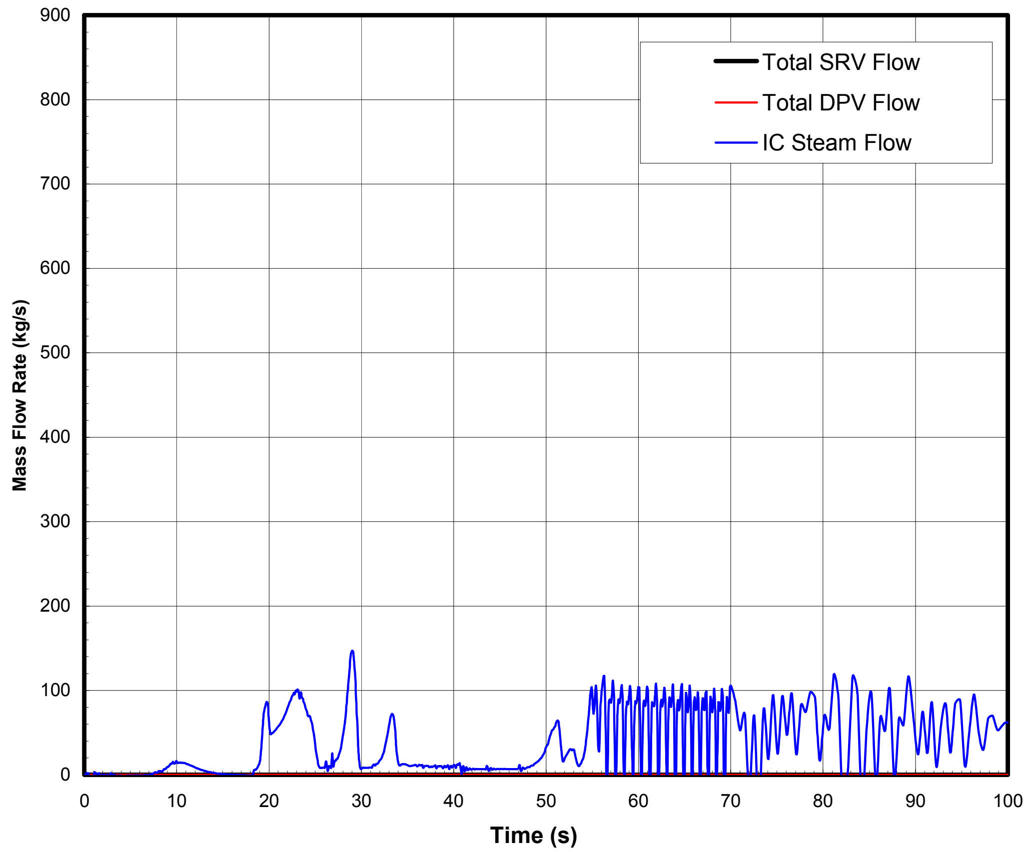


Figure 6.3-20b. ADS Flows, Inside Steam Line Break (Nominal Case), 1 GDCS Valve Failure (100 s)

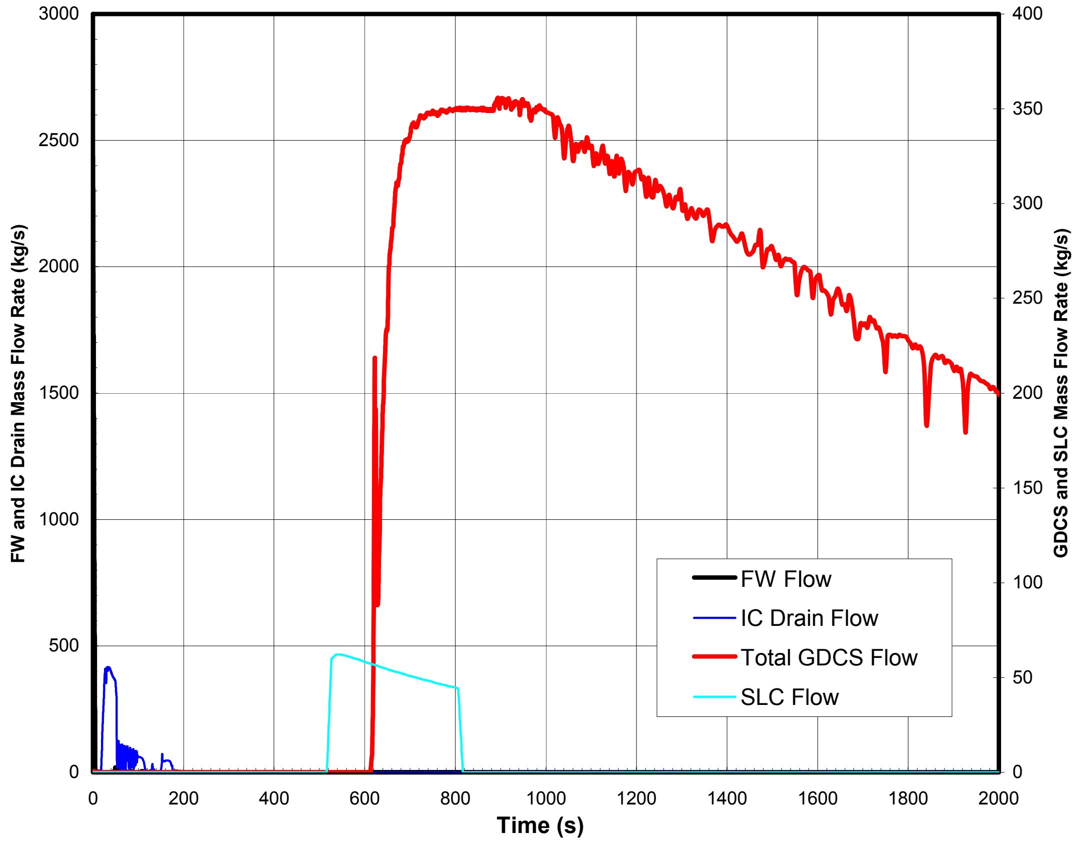


Figure 6.3-21a. Flows Into Vessel, Inside Steam Line Break (Nominal Case), 1 GDCS Valve Failure (2000 s)

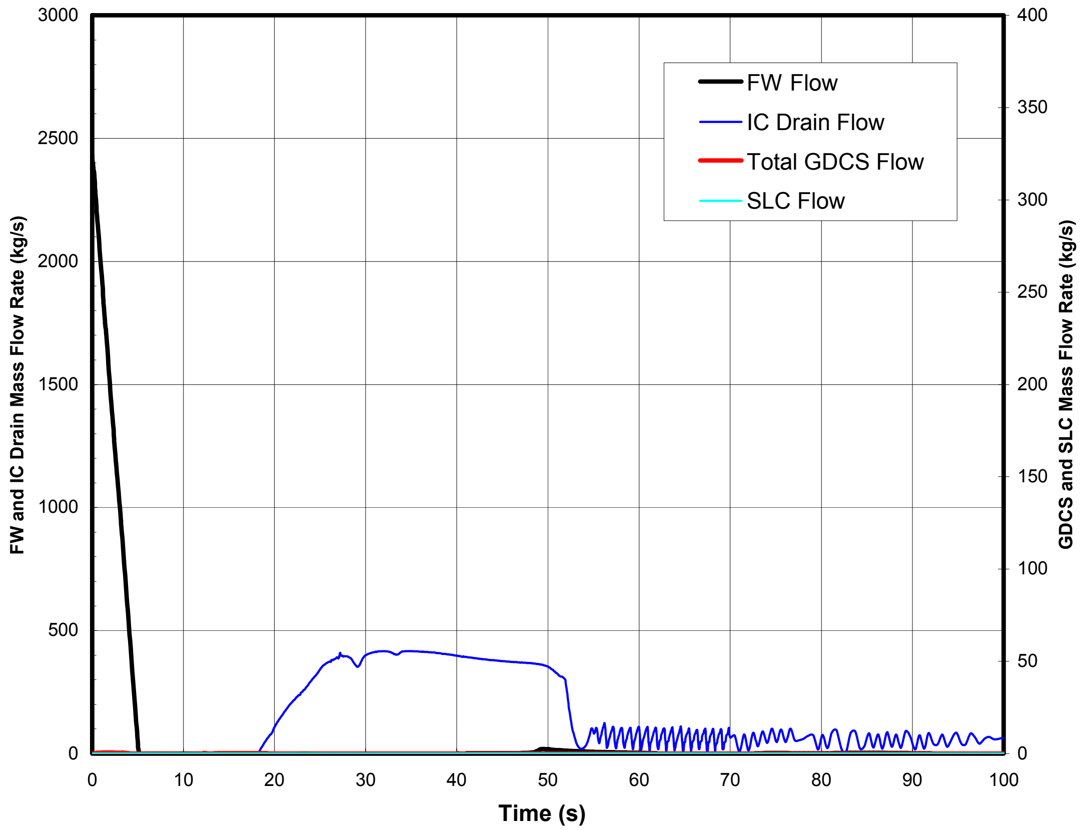
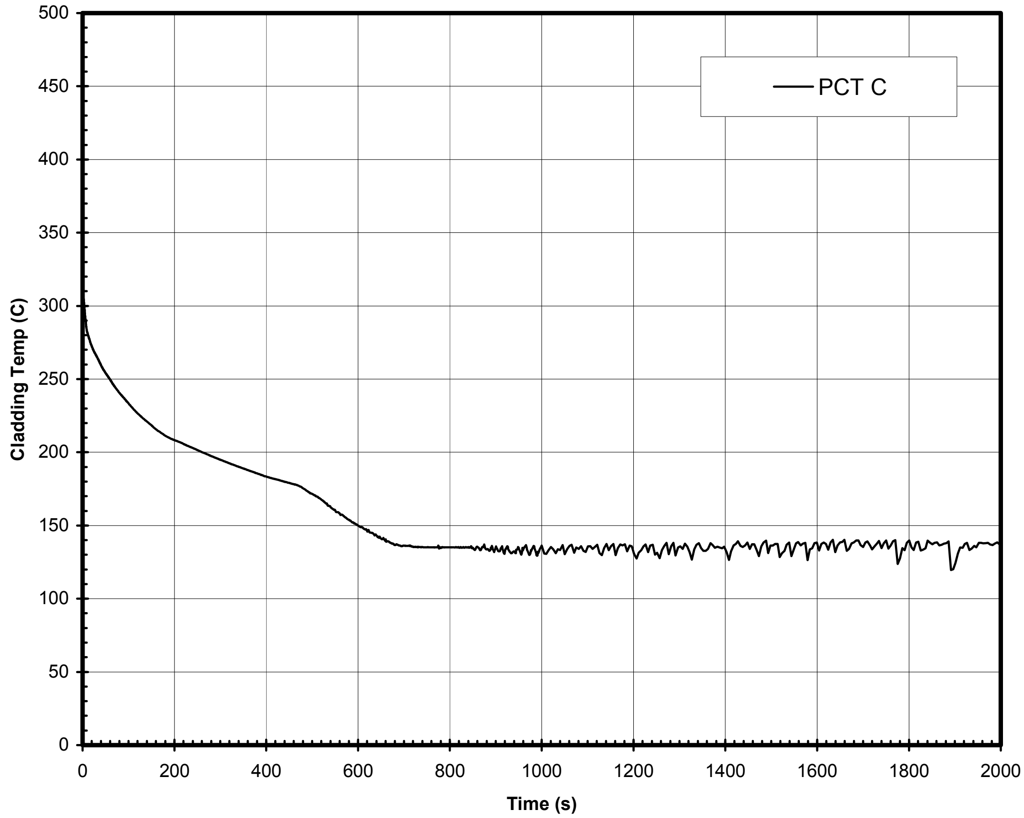
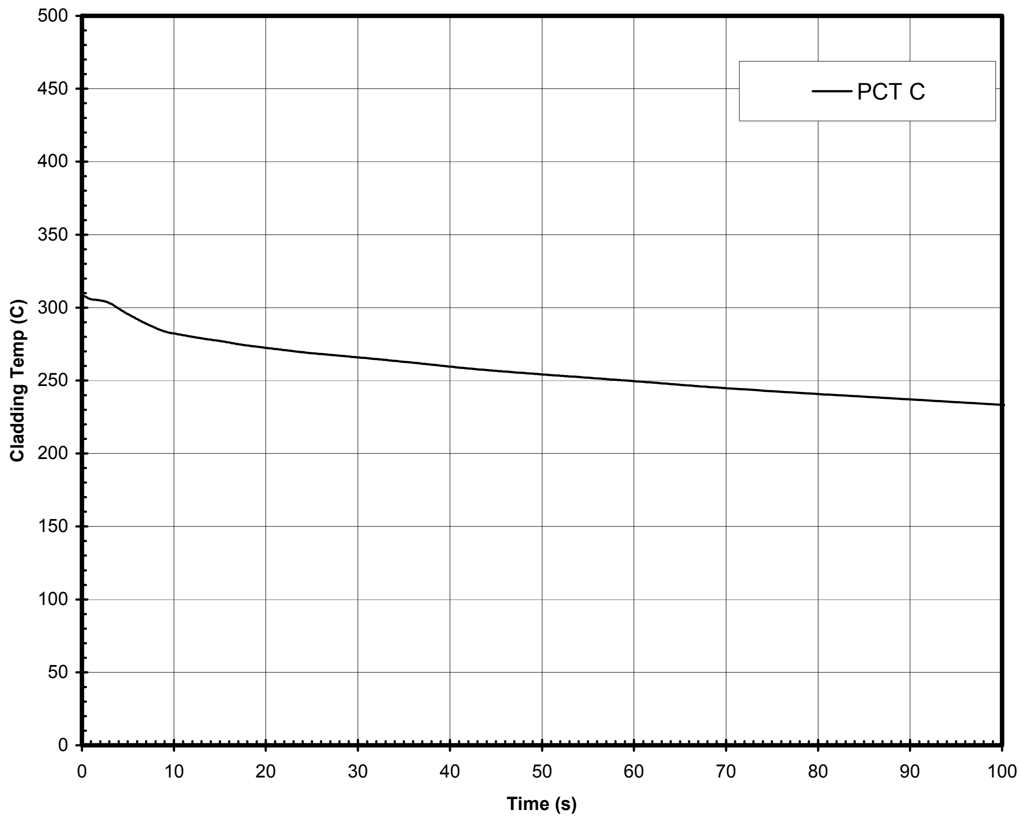


Figure 6.3-21b. Flows Into Vessel, Inside Steam Line Break (Nominal Case), 1 GDCS Valve Failure (100 s)



Legend: PCT C = Peak Cladding Temperature, °C

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



Legend: PCT C = Peak Cladding Temperature, °C

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

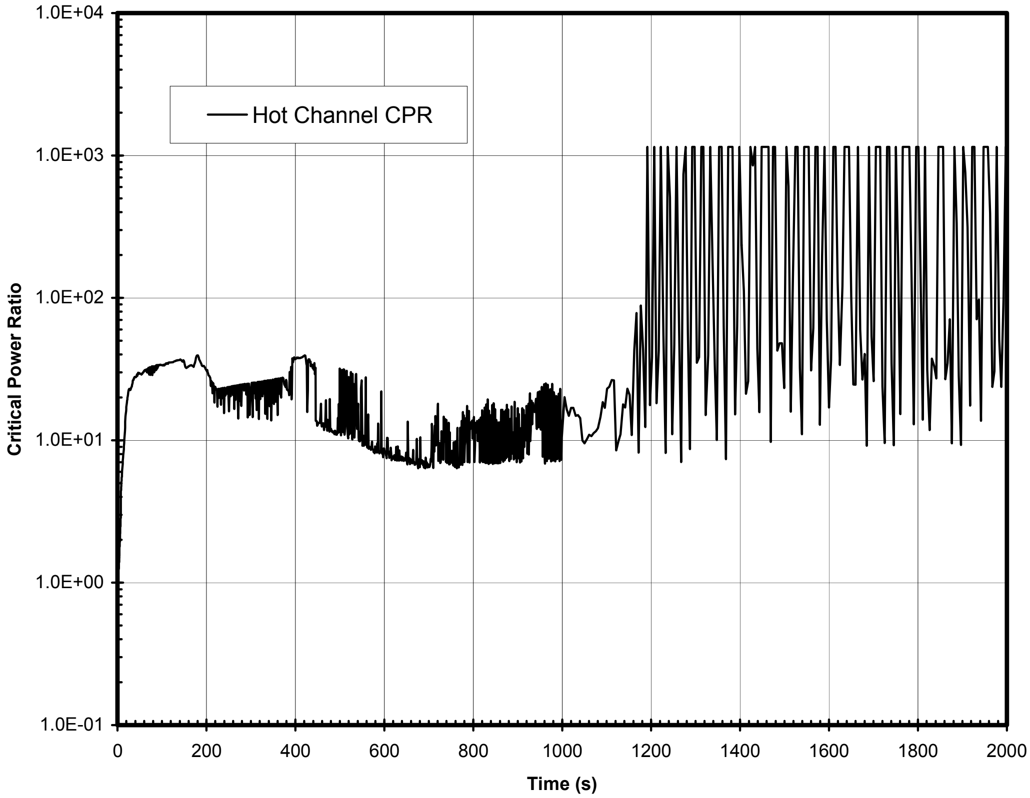


Figure 6.3-23a. MCPR, Bottom Drain Line Break (Nominal Case), 1 GDCS Valve Failure (2000 s)

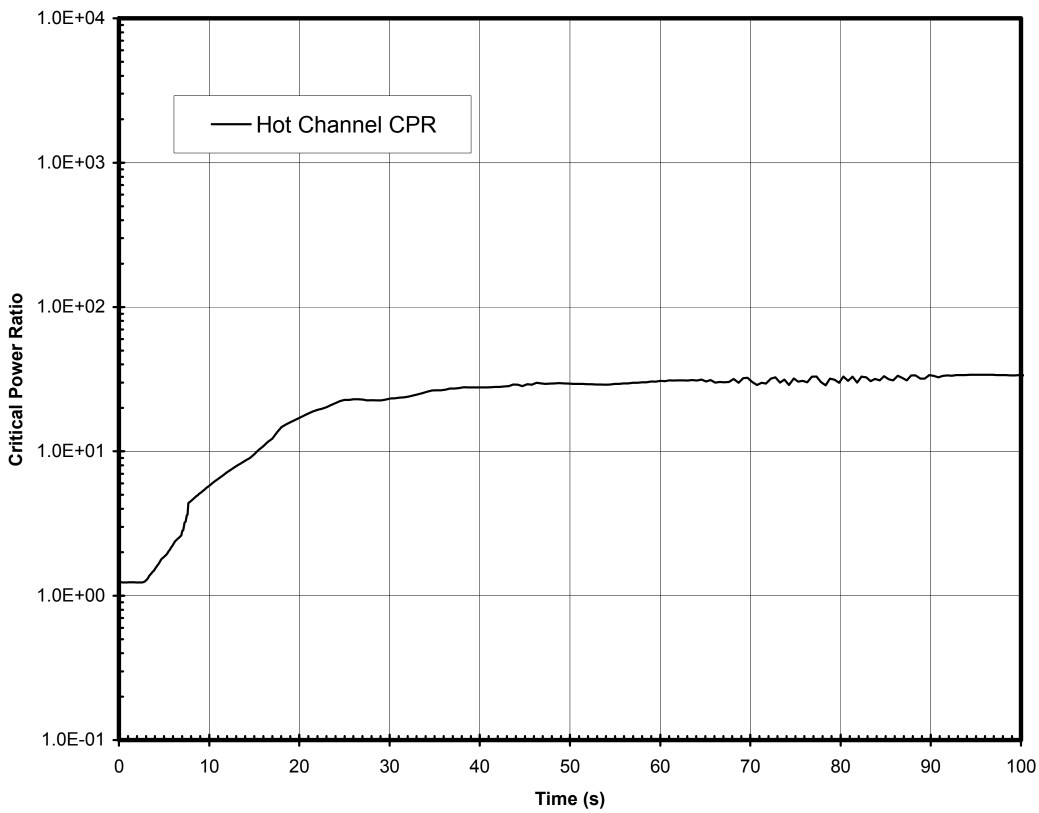


Figure 6.3-23b. MCPR, Bottom Drain Line Break (Nominal Case), 1 GDCS Valve Failure (100 s)

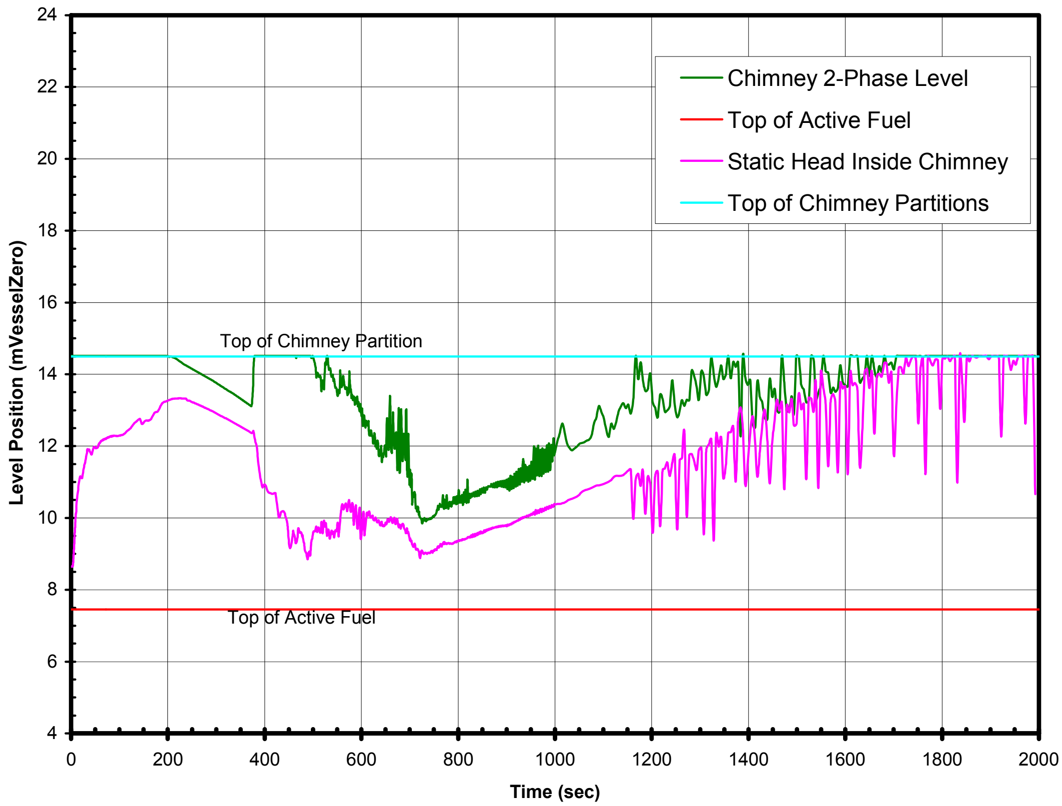


Figure 6.3-24a. Chimney Water Level, Bottom Drain Line Break (Nominal Case), 1 GDCS Valve Failure (2000 s)

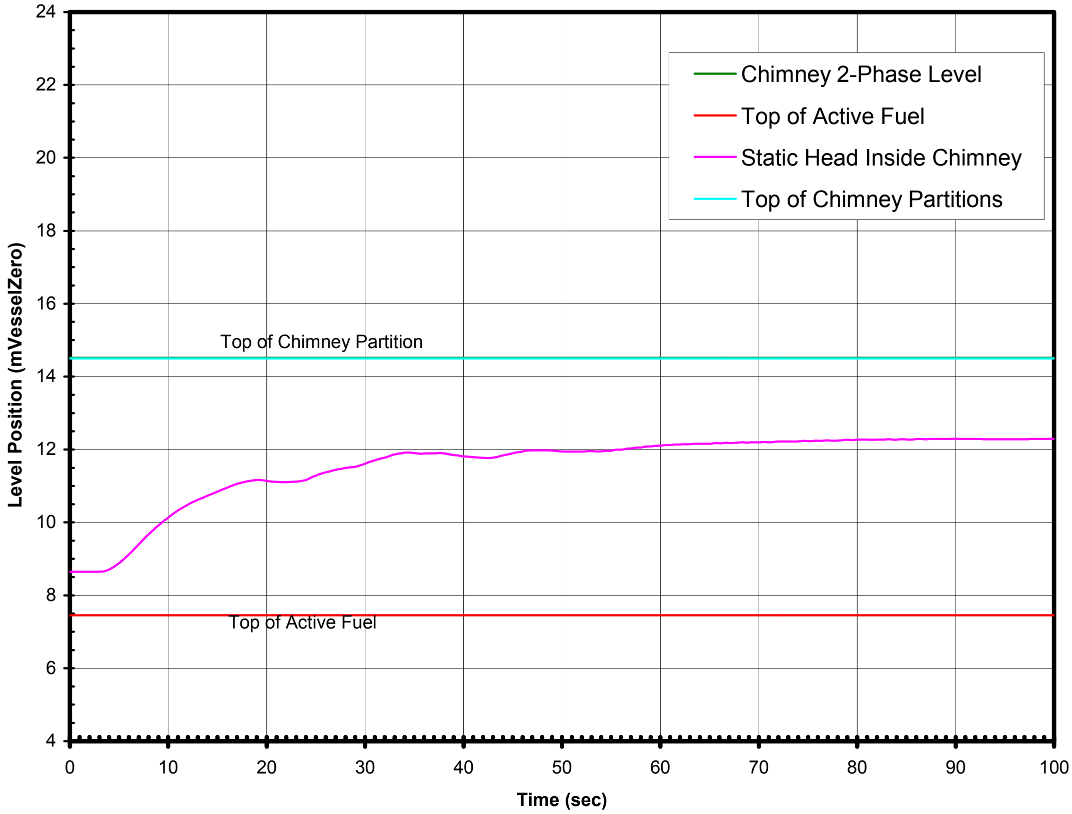


Figure 6.3-24b. Chimney Water Level, Bottom Drain Line Break (Nominal Case), 1 GDCS Valve Failure (100 s)

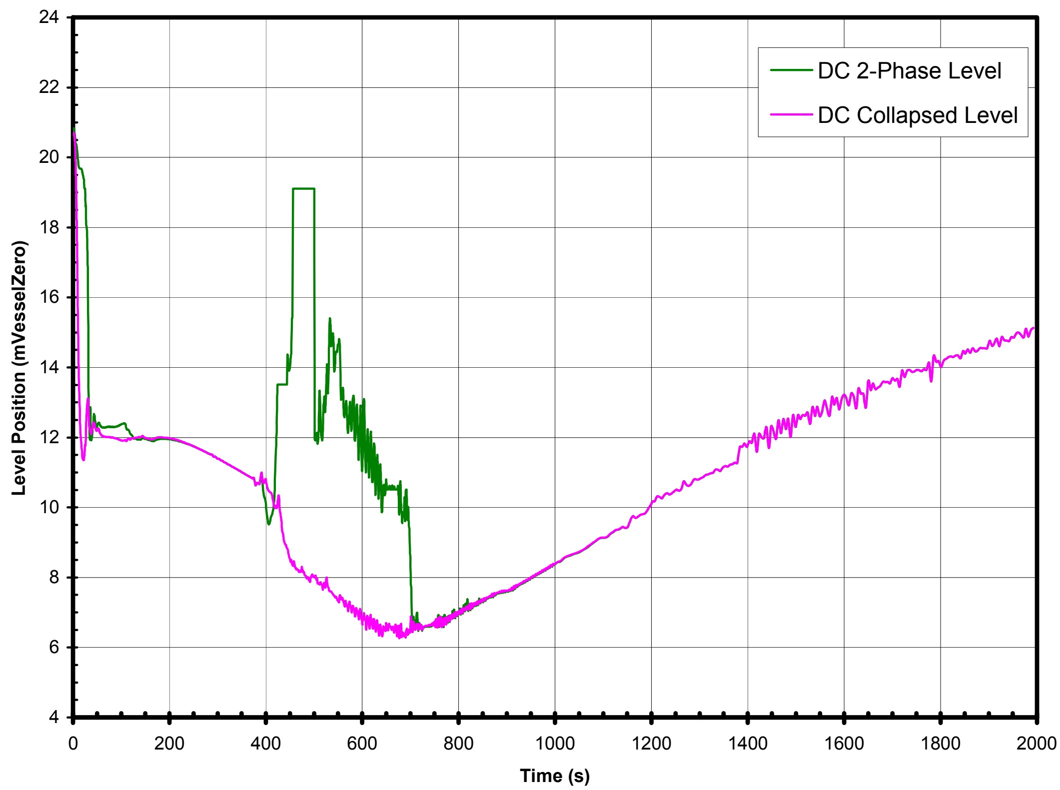


Figure 6.3-25a. Downcomer (DC) Water Level, Bottom Drain Line Break (Nominal Case), 1 GDCS Valve Failure (2000 s)

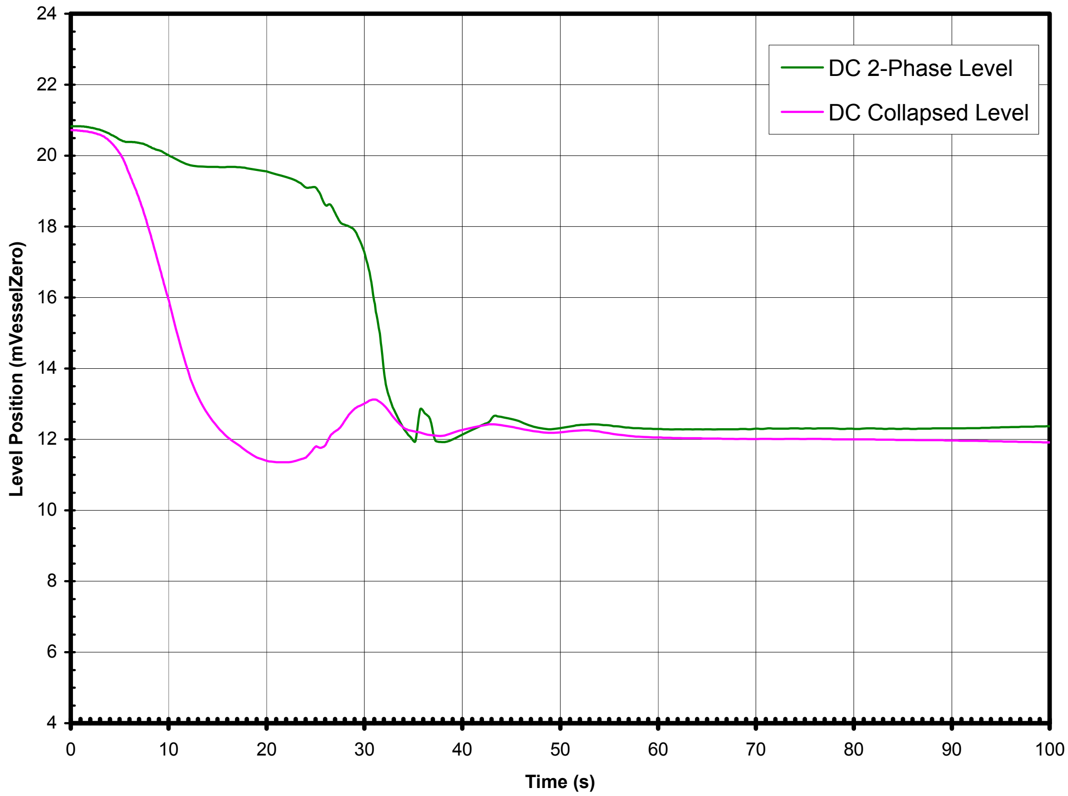


Figure 6.3-25b. Downcomer (DC) Water Level, Bottom Drain Line Break (Nominal Case), 1 GDCS Valve Failure (100 s)

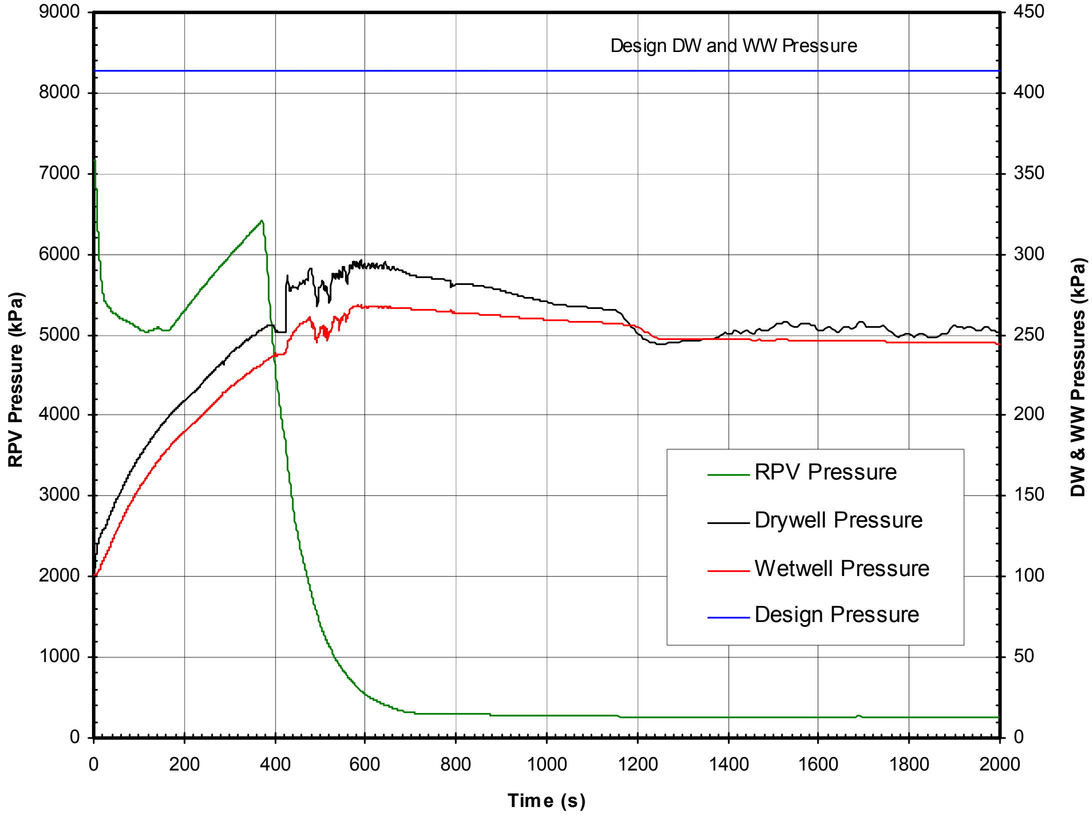


Figure 6.3-26a. System Pressures, Bottom Drain Line Break (Nominal Case), 1 GDCS Valve Failure (2000 s)

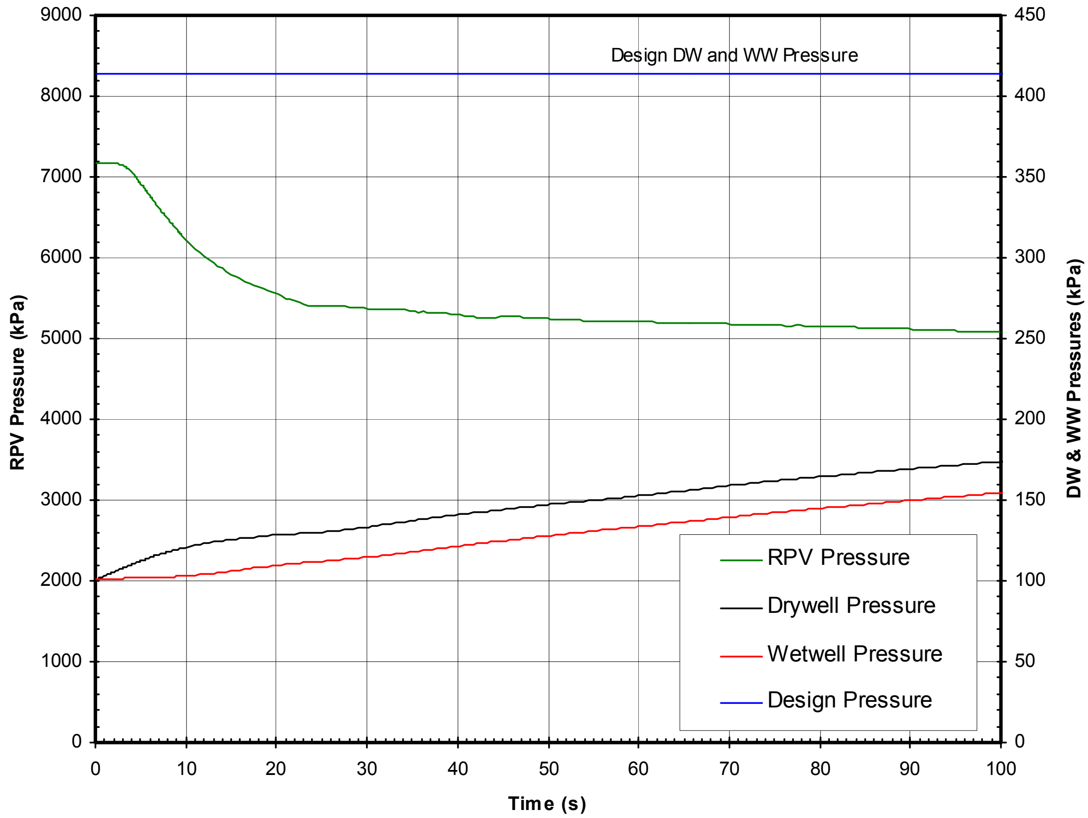


Figure 6.3-26b. System Pressures, Bottom Drain Line Break (Nominal Case), 1 GDCS Valve Failure (100 s)

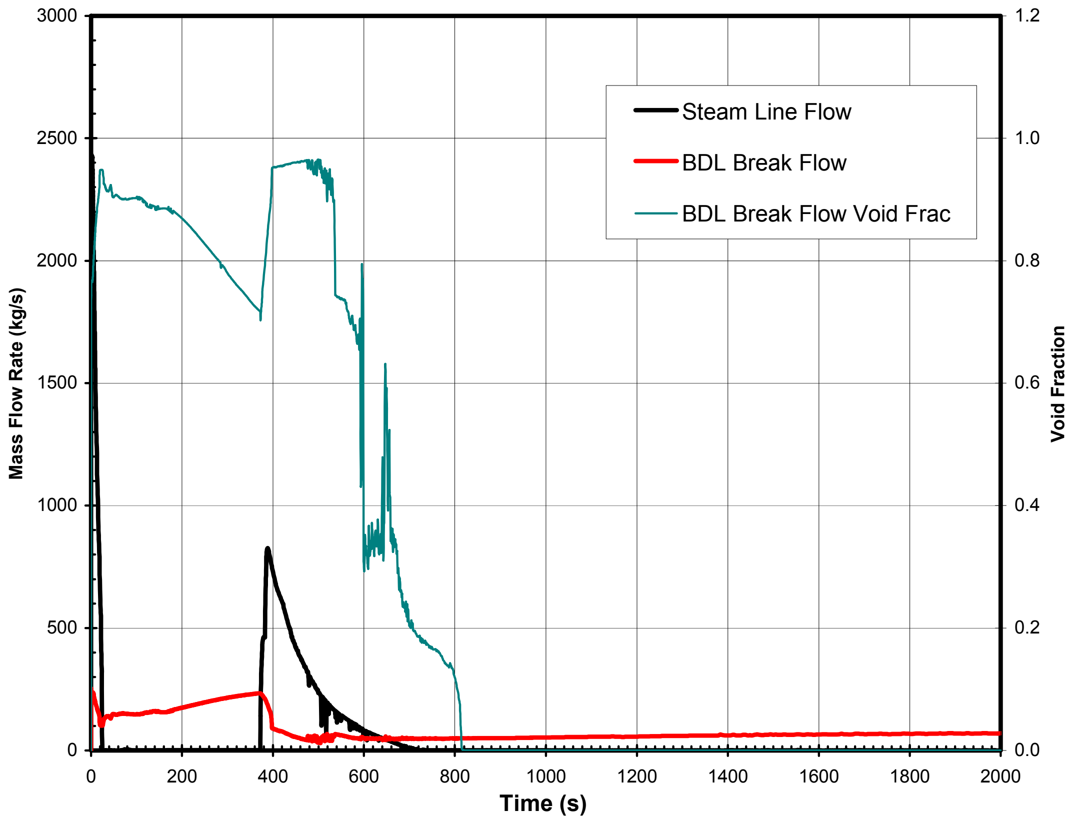


Figure 6.3-27a. Steam Line and Break Flow with Void Fraction, Bottom Drain Line Break (Nominal Case), 1 GDCS Valve Failure (2000 s)

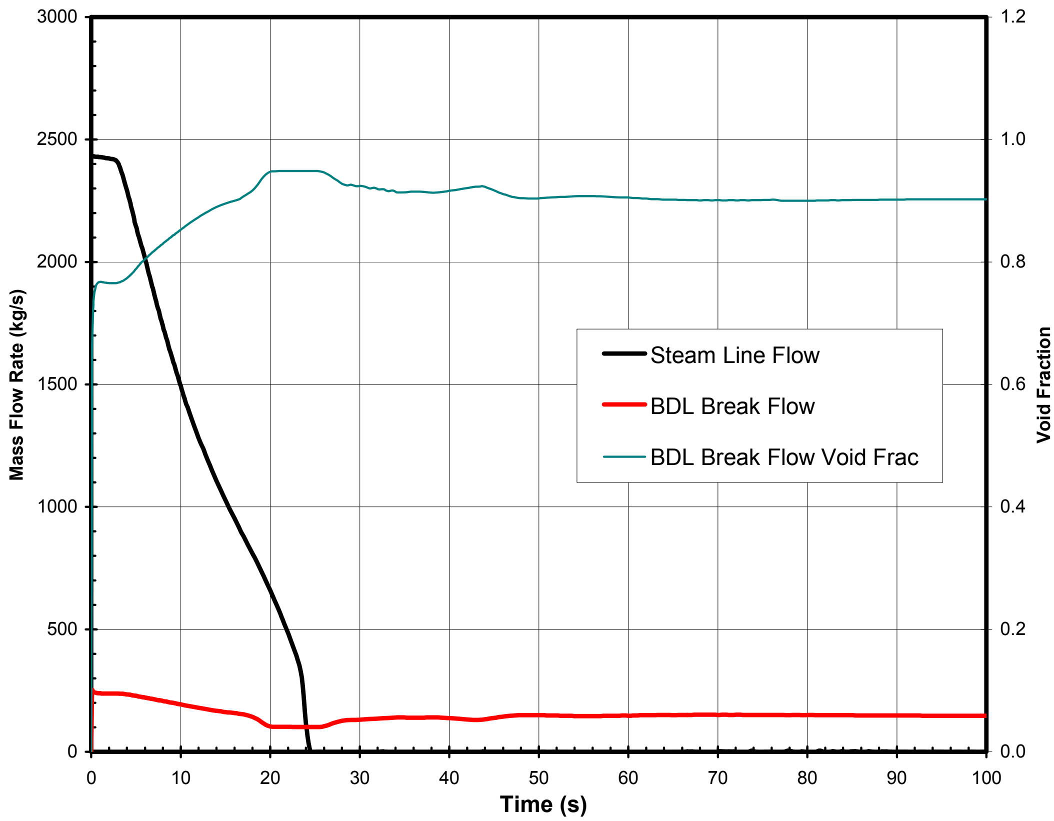


Figure 6.3-27b. Steam Line and Break Flow with Void Fraction, Bottom Drain Line Break (Nominal Case), 1 GDCS Valve Failure (100 s)

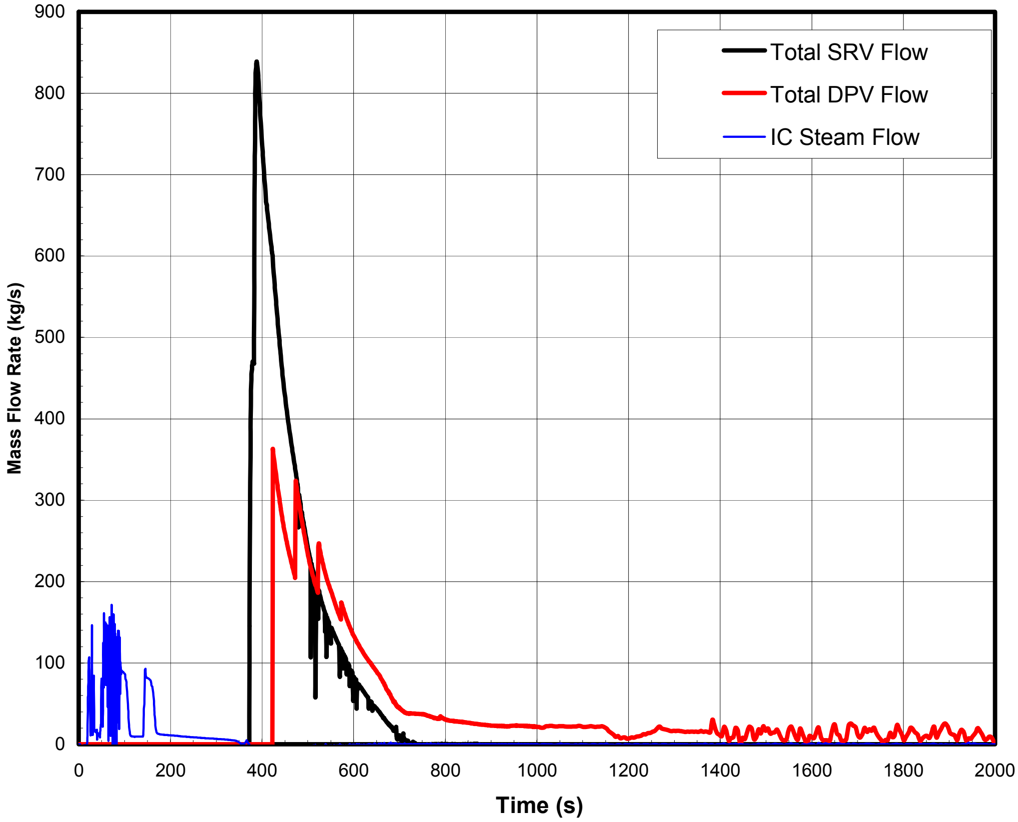


Figure 6.3-28a. ADS Flow Bottom Drain Line Break (Nominal Case), 1 GDCS Valve Failure (2000 s)

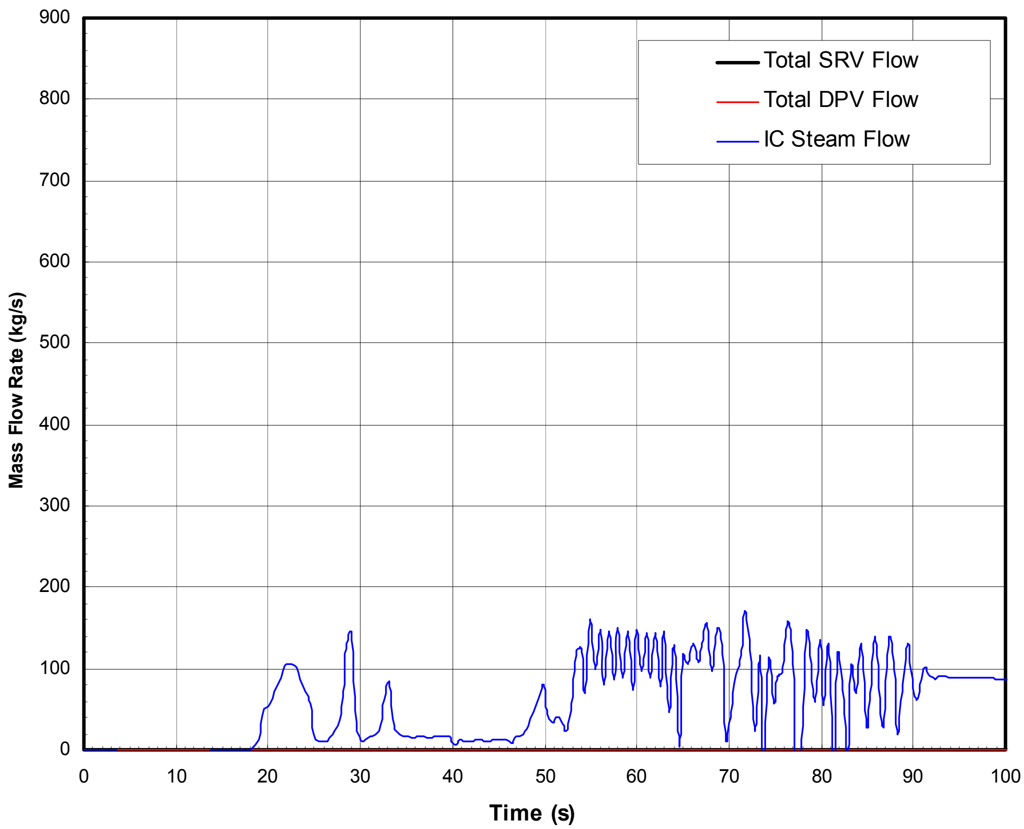


Figure 6.3-28b. ADS Flows, Bottom Drain Line Break (Nominal Case), 1 GDCS Valve Failure (100 s)

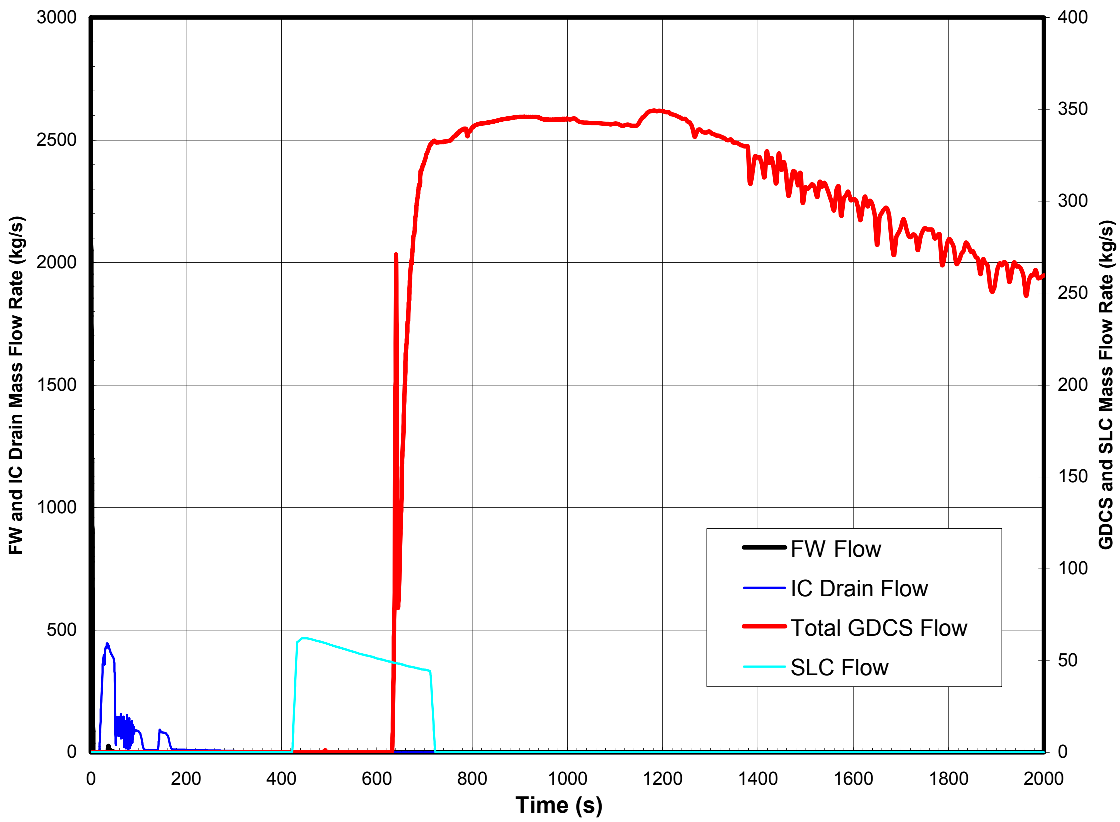


Figure 6.3-29a. Flows Into Vessel, Bottom Drain Line Break (Nominal Case), 1 GDCS Valve Failure (2000 s)

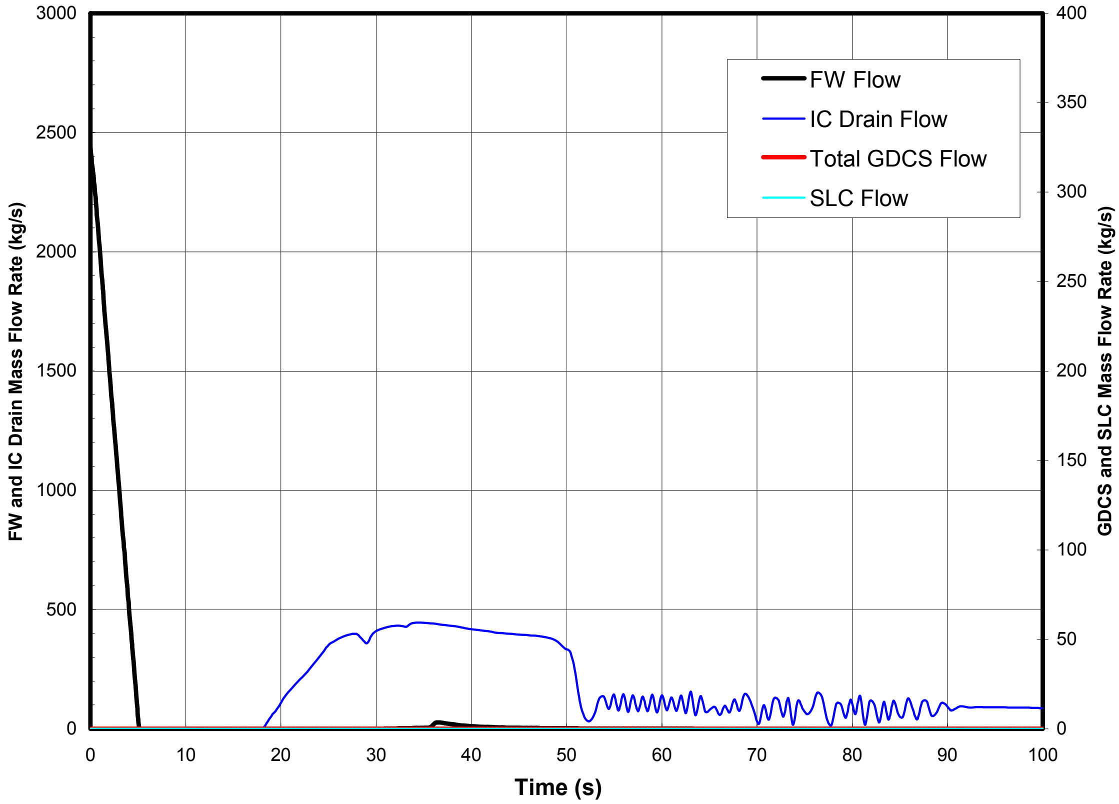
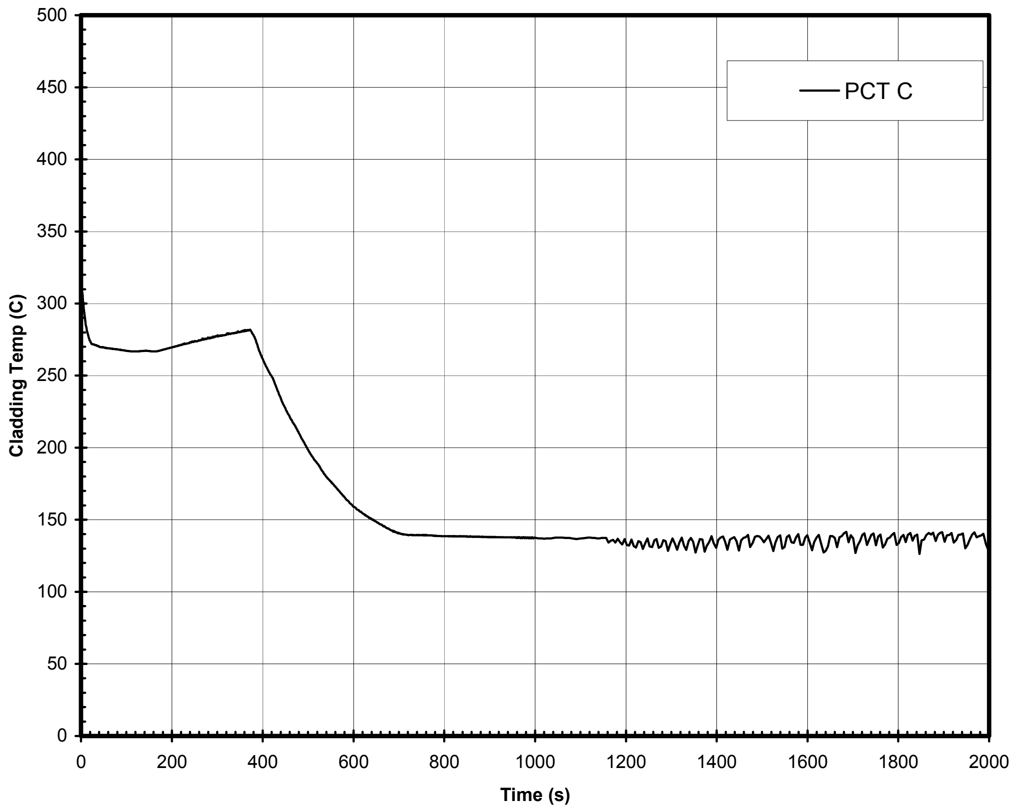
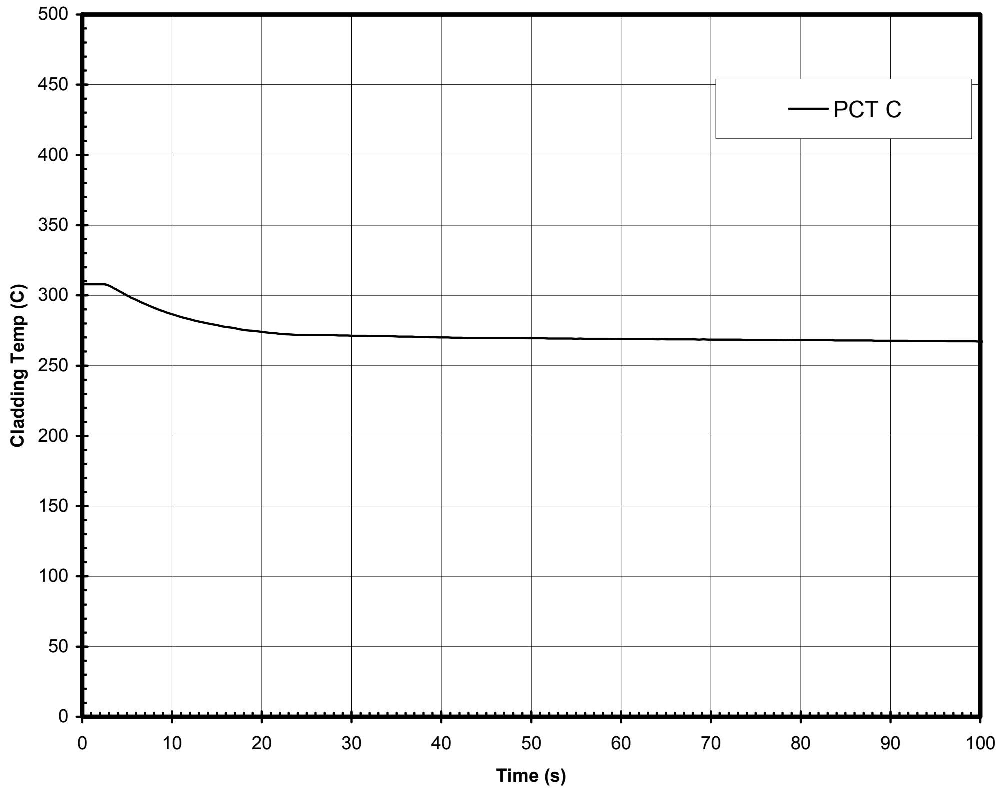


Figure 6.3-29b. Flows Into Vessel, Bottom Drain Line Break (Nominal Case), 1 GDCS Valve Failure (100 s)



Legend: PCT C = Peak Cladding Temperature, ^{o}C

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



Legend: PCT C = Peak Cladding Temperature, °C

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

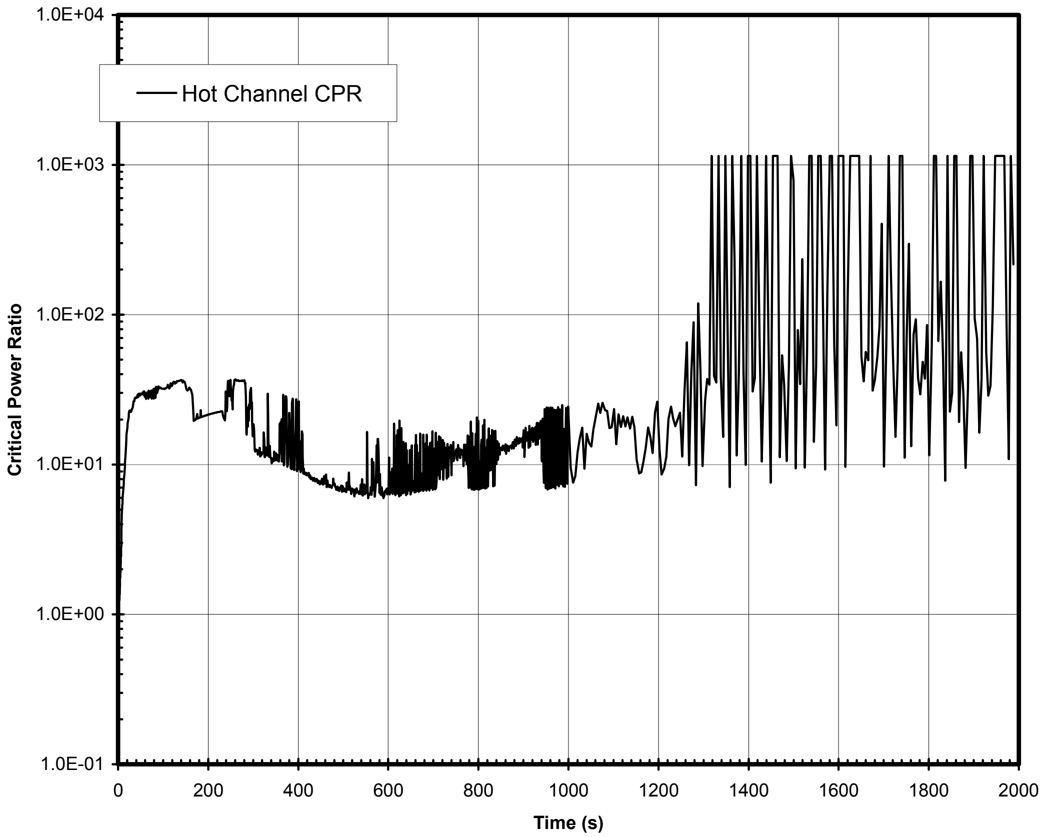


Figure 6.3-31a. MCPR, GDCS Injection Line Break (Nominal Case), 1 GDCS Valve Failure (2000 s)

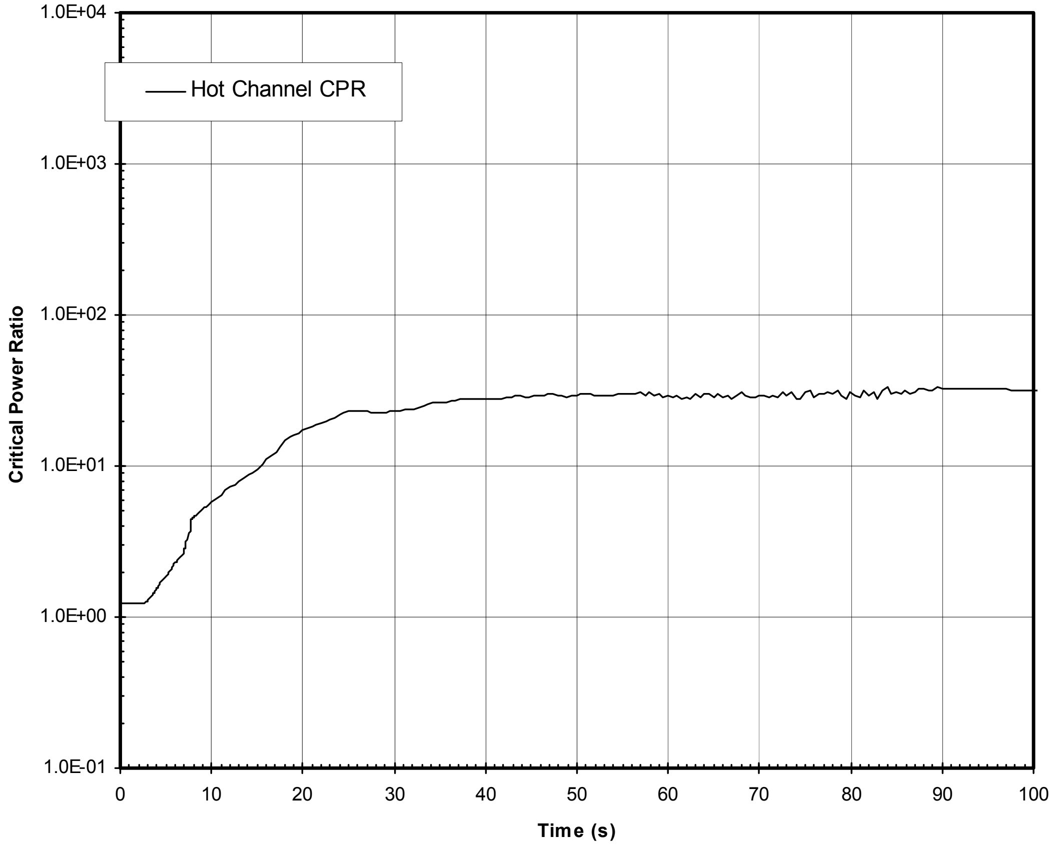


Figure 6.3-31b. MCPR, GDCS Injection Line Break (Nominal Case), 1 GDCS Valve Failure (100 s)

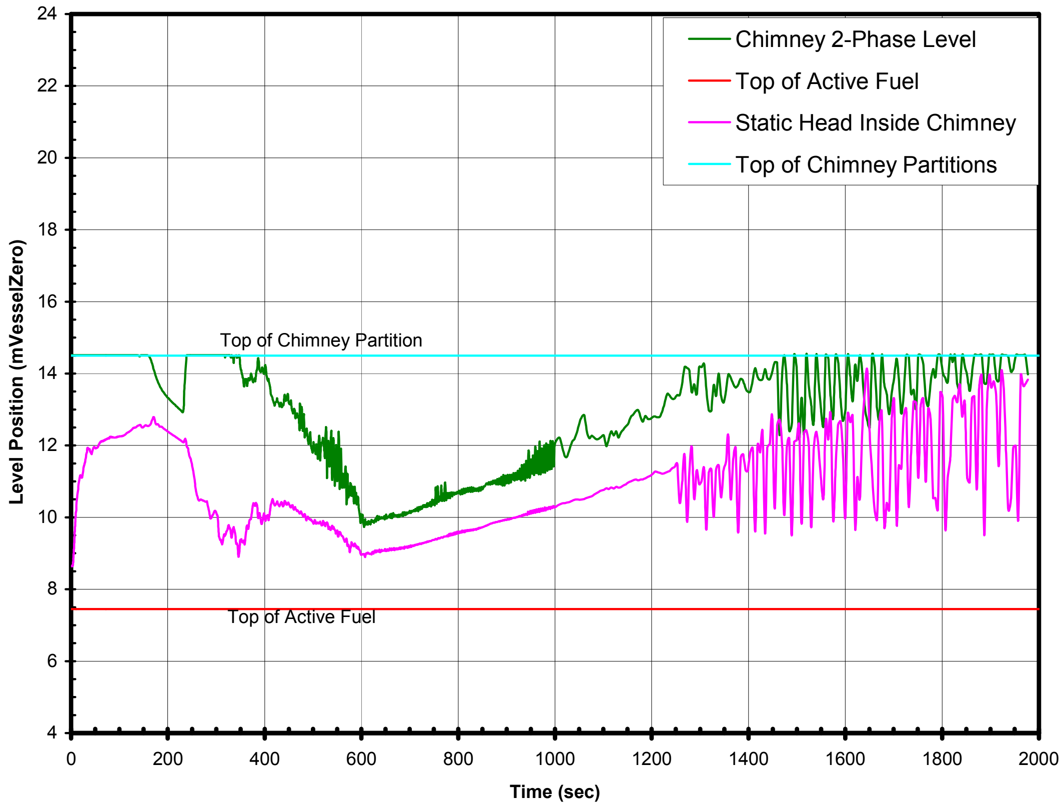


Figure 6.3-32a. Chimney Water Level, GDCS Injection Line Break (Nominal Case), 1 GDCS Valve Failure (2000 s)

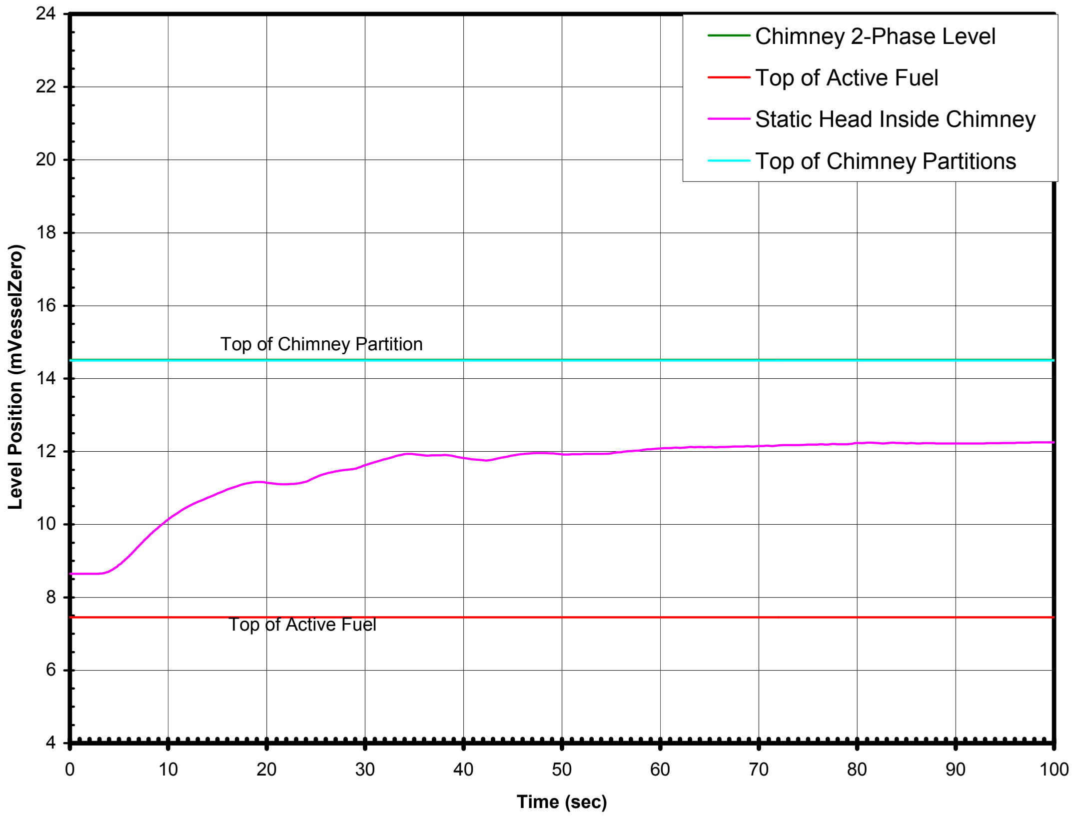
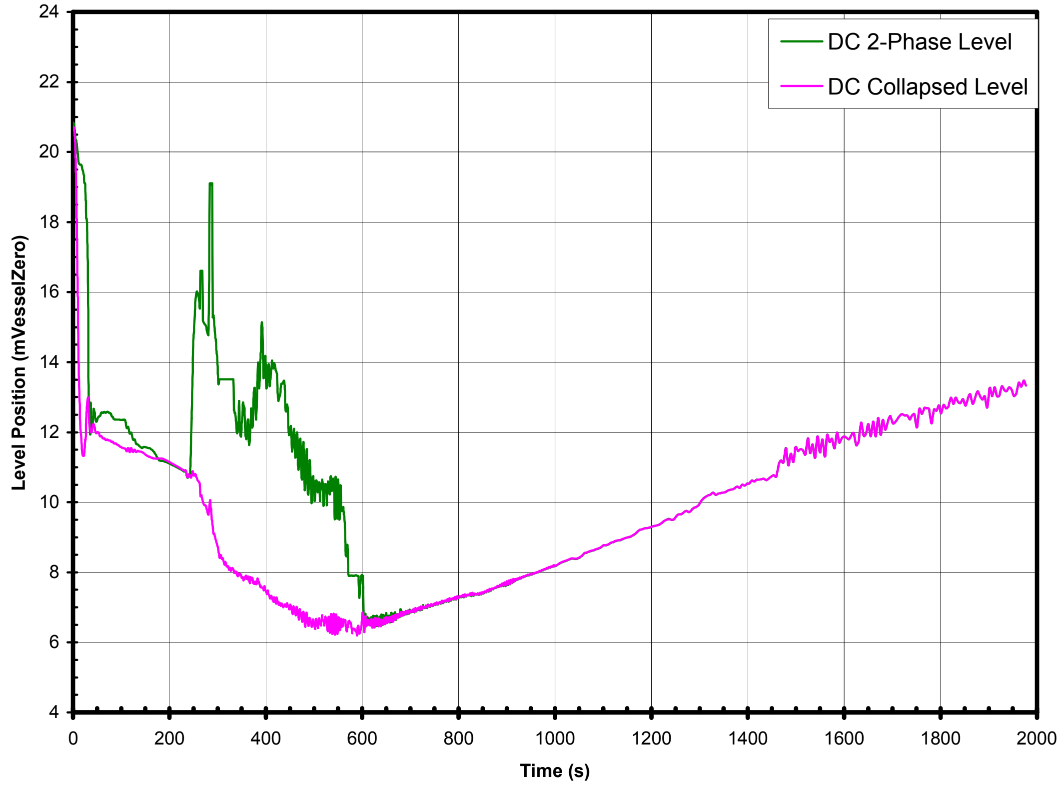
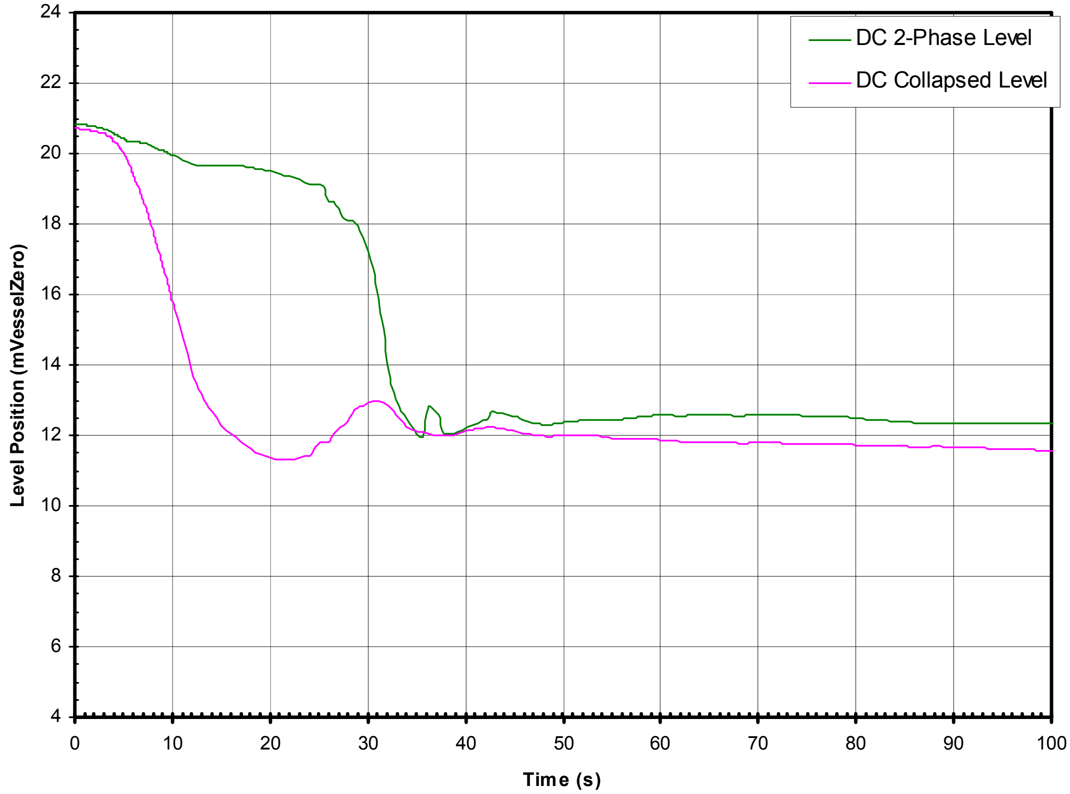


Figure 6.3-32b. Chimney Water Level, GDCS Injection Line Break (Nominal Case), 1 GDCS Valve Failure (100 s)



LEGEND: DC = Downcomer

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



LEGEND: DC = Downcomer

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

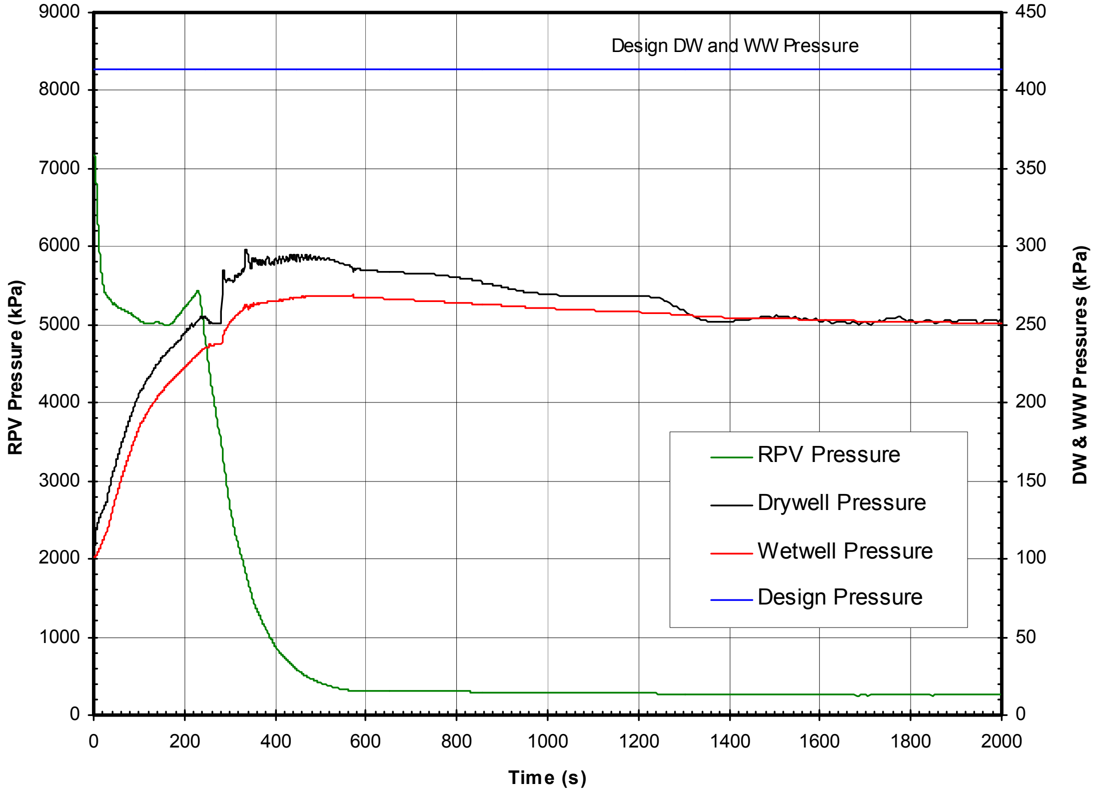


Figure 6.3-34a. System Pressures, GDCS Injection Line Break (Nominal Case), 1 GDCS Valve Failure (2000 s)

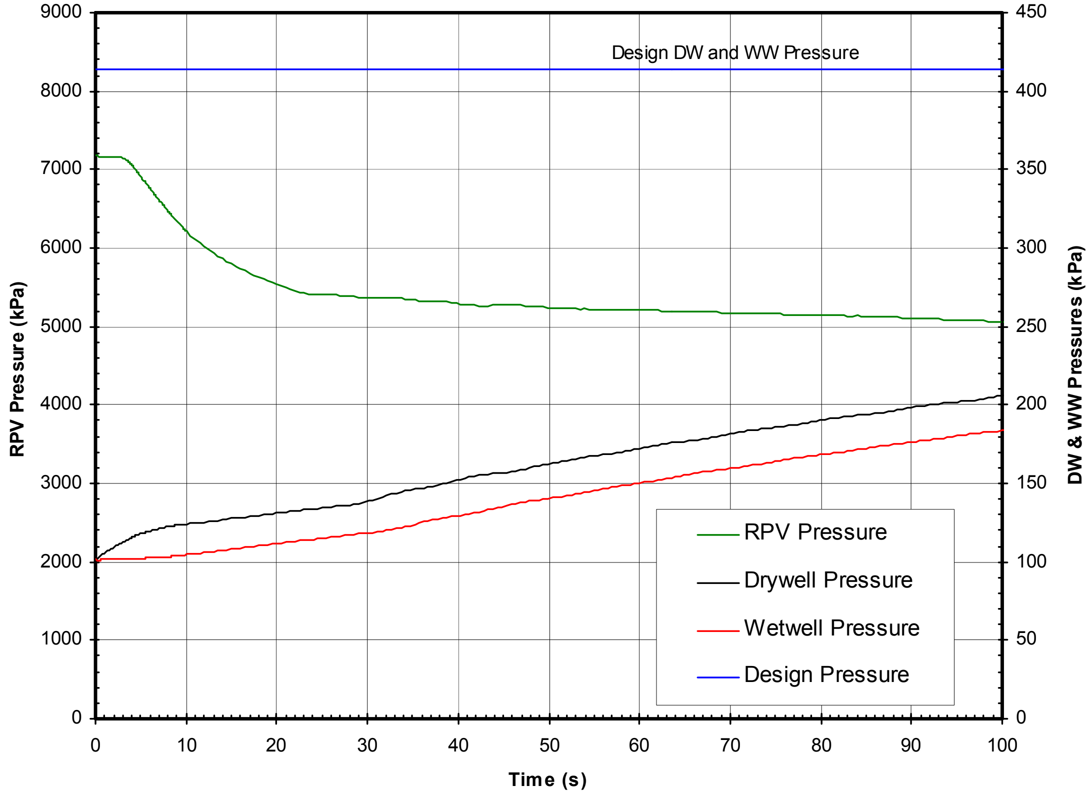
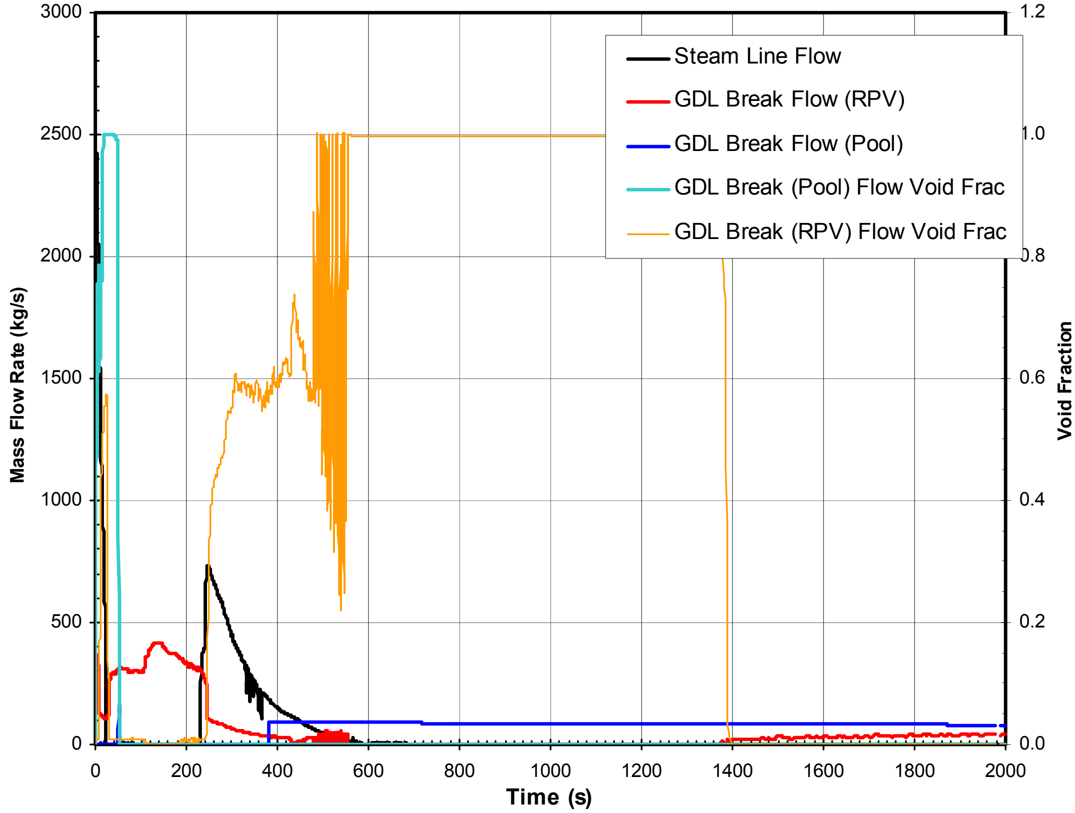


Figure 6.3-34b. System Pressures, GDCS Injection Line Break (Nominal Case), 1 GDCS Valve Failure (100 s)



LEGEND: GDL = Gravity Drain Line

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

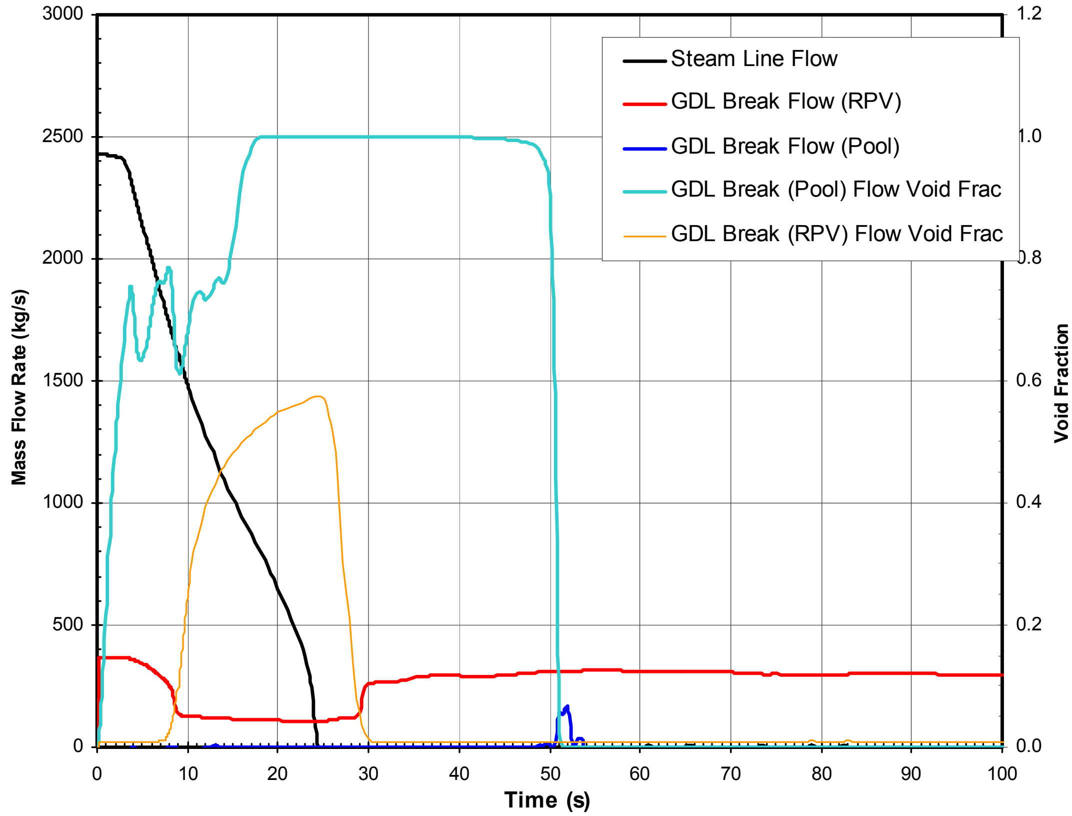


Figure 6.3-35b. Steam Line and Break Flow with Void Fraction, GDCS Injection Line Break (Nominal Case), 1 GDCS Valve Failure (100 s)

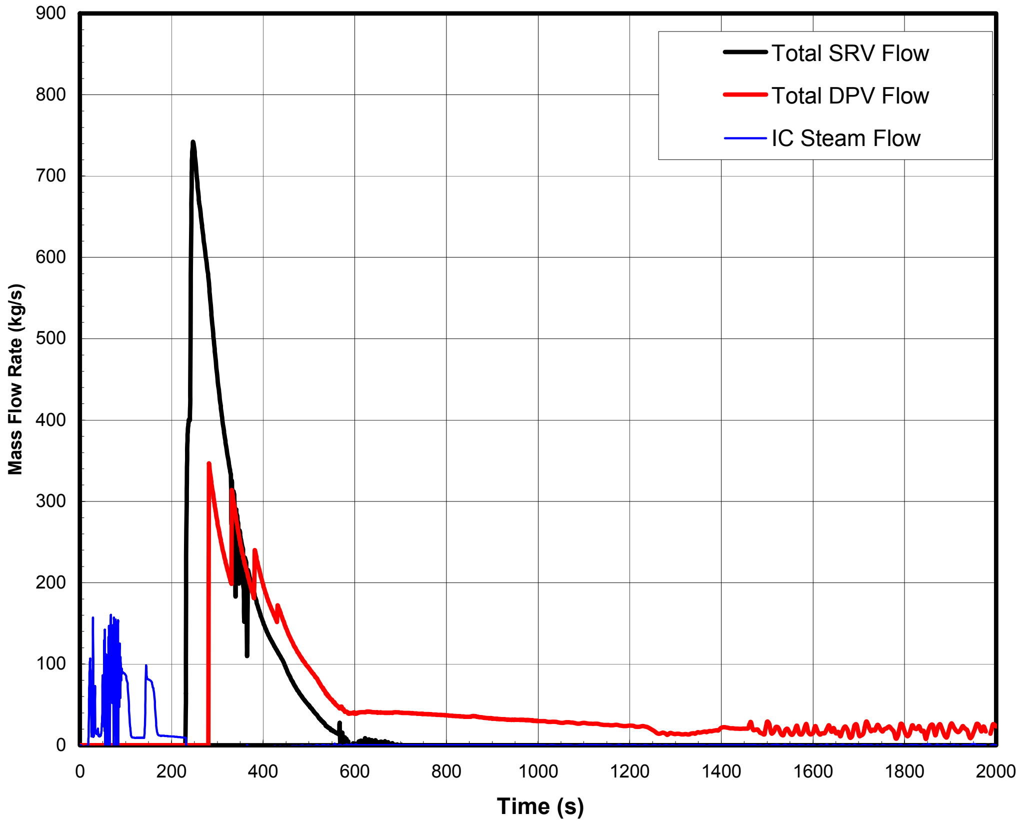


Figure 6.3-36a. ADS Flow GDCS Injection Line Break (Nominal Case), 1 GDCS Valve Failure (2000 s)

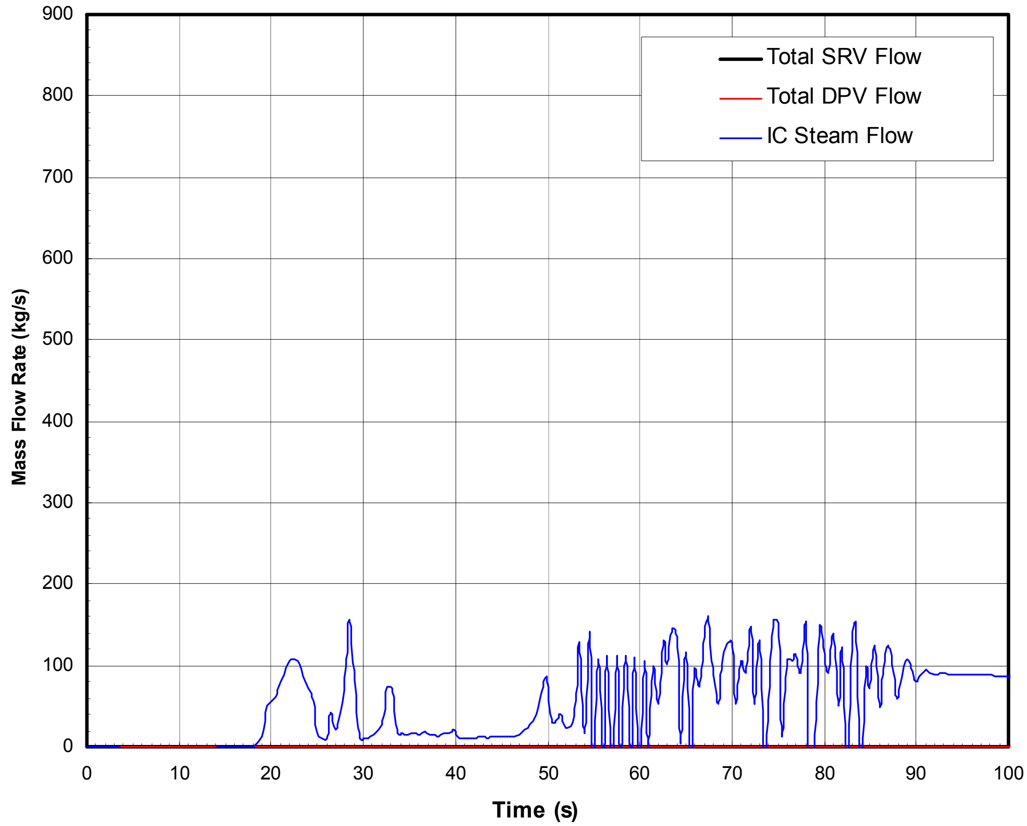


Figure 6.3-36b. ADS Flows, GDCS Injection Line Break (Nominal Case), 1 GDCS Valve Failure (100 s)

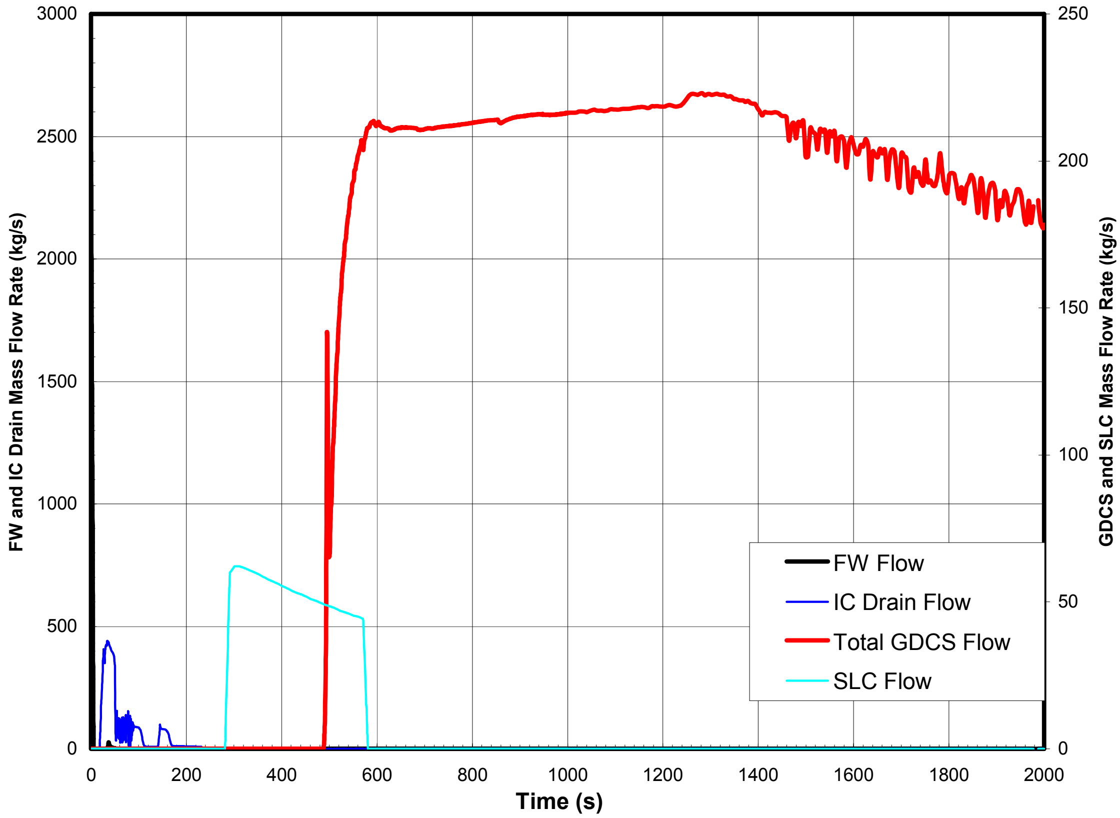


Figure 6.3-37a. Flows Into Vessel, GDCS Injection Line Break (Nominal Case), 1 GDCS Valve Failure (2000 s)

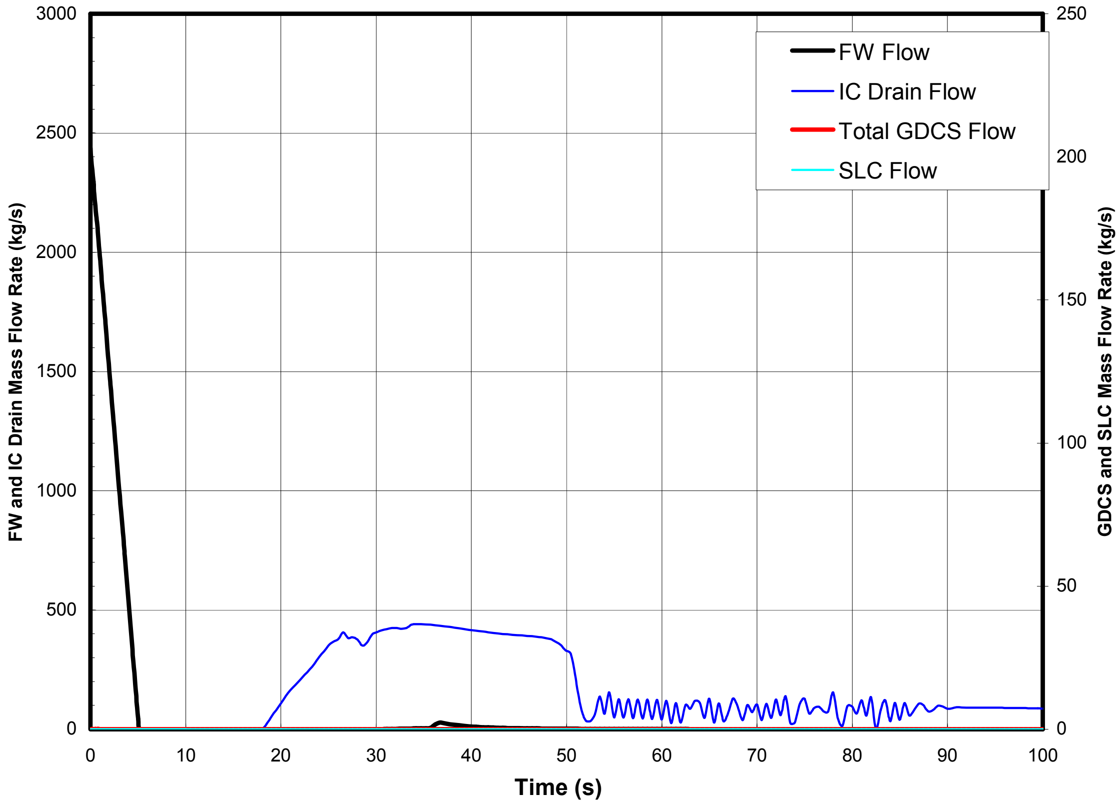


Figure 6.3-37b. Flows Into Vessel, GDCS Injection Line Break (Nominal Case), 1 GDCS Valve Failure (100 s)

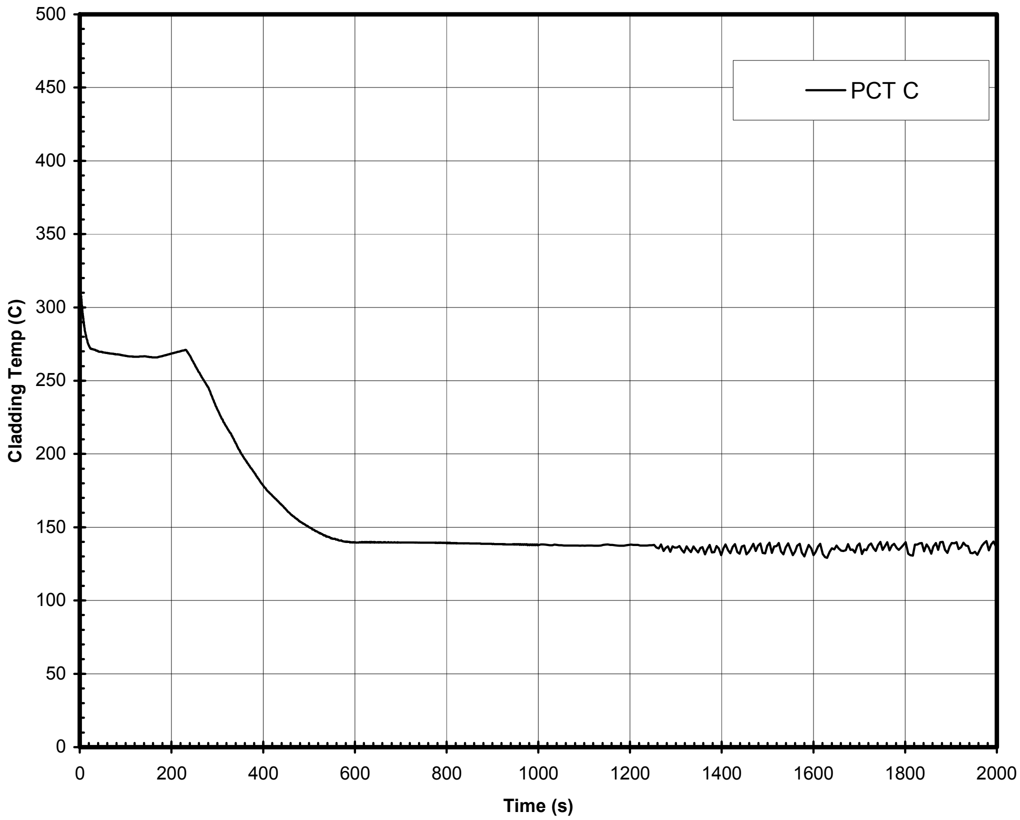


Figure 6.3-38a. Peak Cladding Temperature (PCT), GDCS Injection Line Break (Nominal Case), 1 GDCS Valve Failure (2000 s)

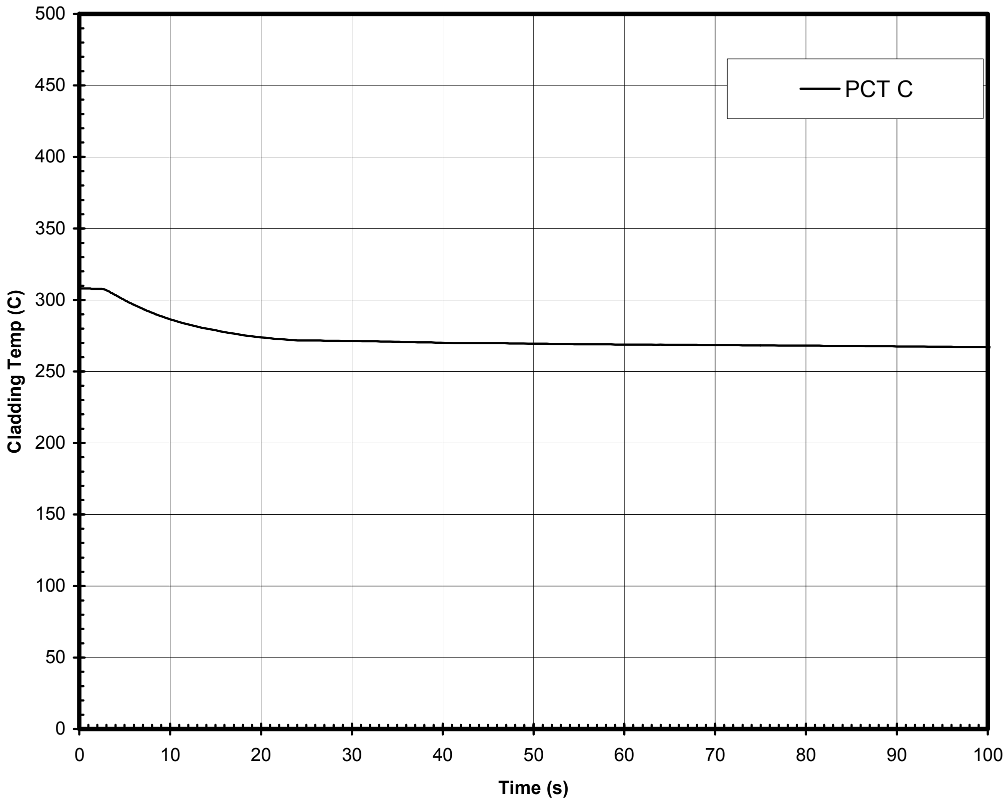


Figure 6.3-38b. Peak Cladding Temperature (PCT), GDCS Injection Line Break (Nominal Case), 1 GDCS Valve Failure (100 s)

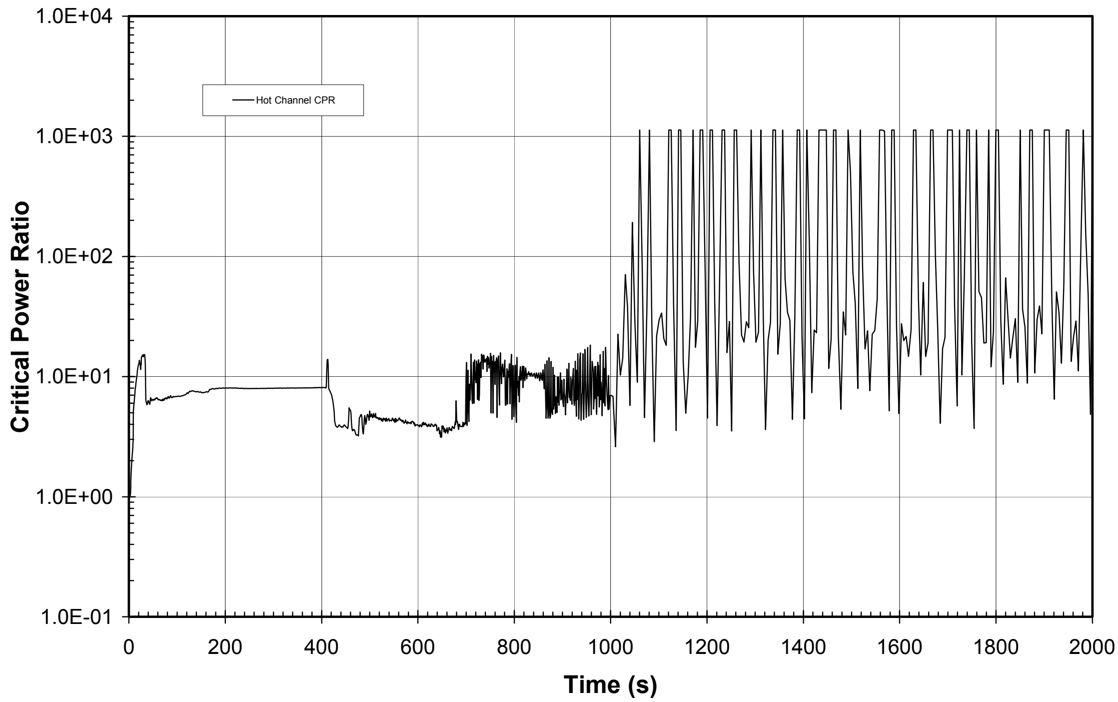


Figure 6.3-38A-a. MCPR, IC Drain Line Break (Bounding Case), 1 GDCS Valve Failure (2000 s)

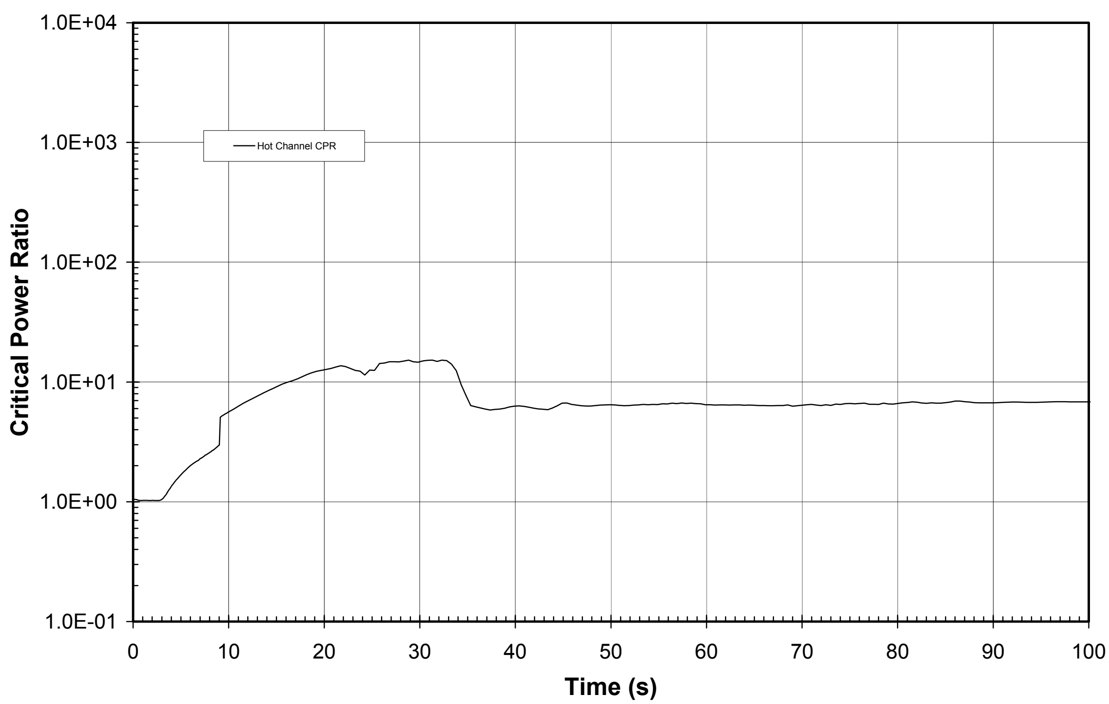


Figure 6.3-38A-b. MCPR, IC Drain Line Break (Bounding Case), 1 GDCS Valve Failure (100 s)

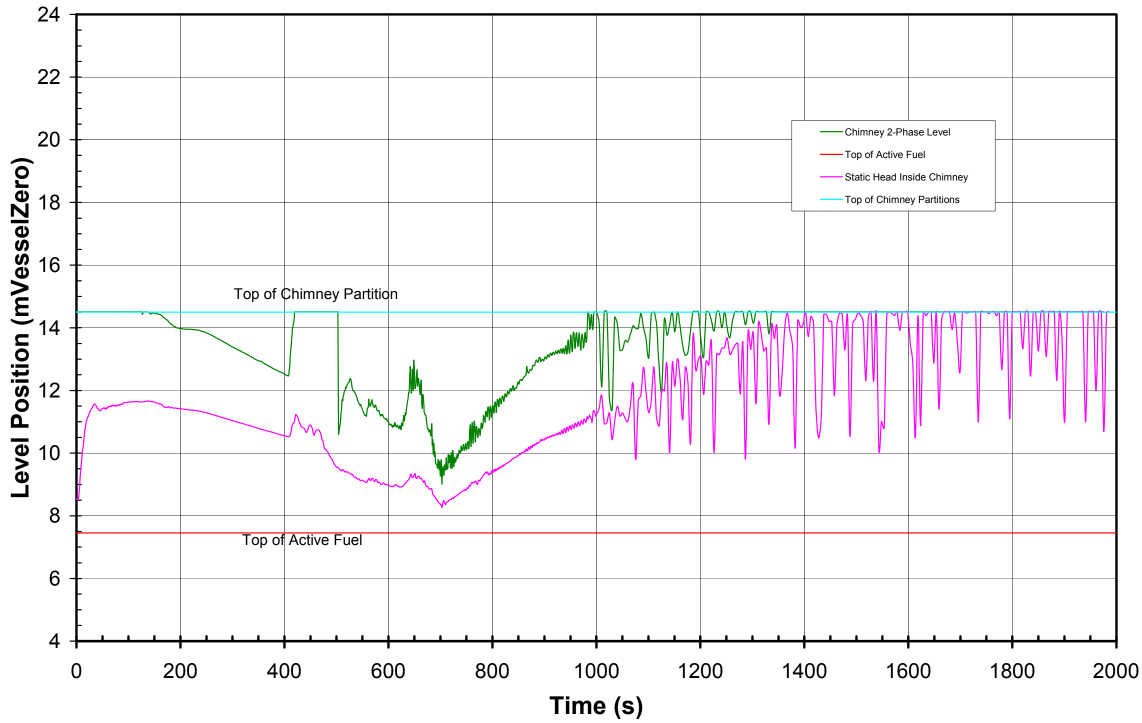


Figure 6.3-38B-a. Chimney Water Level, IC Drain Line Break (Bounding Case), 1 GDCS Valve Failure (2000 s)

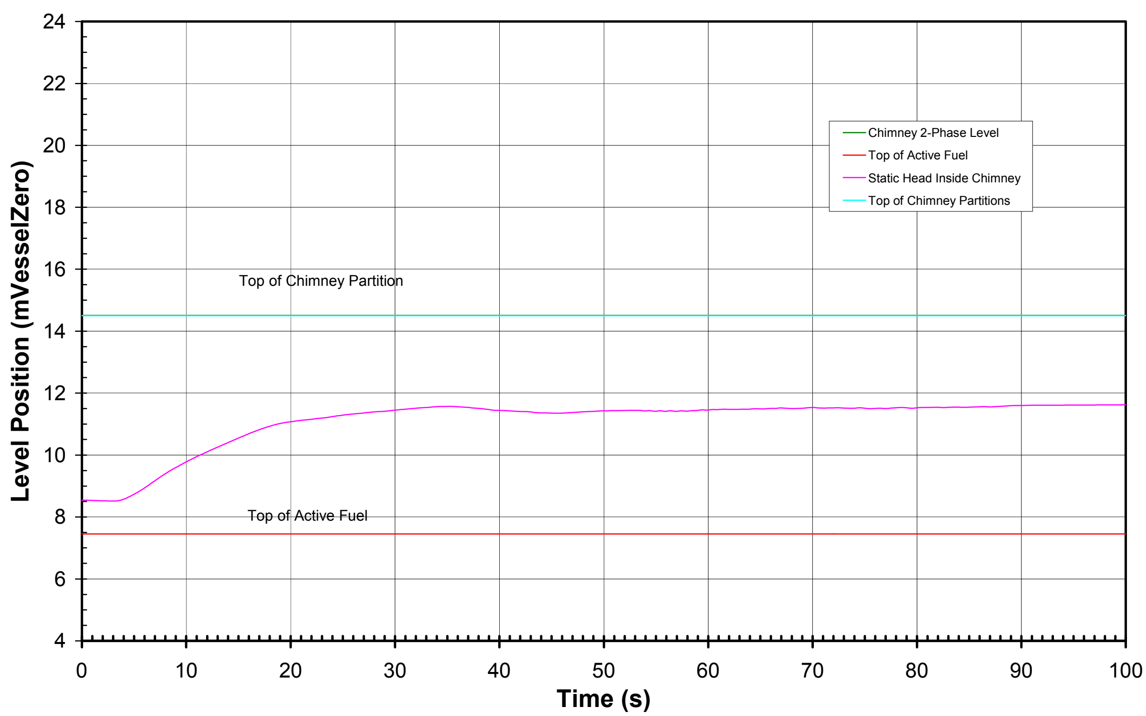


Figure 6.3-38B-b. Chimney Water Level, IC Drain Line Break (Bounding Case), 1 GDCS Valve Failure (100 s)

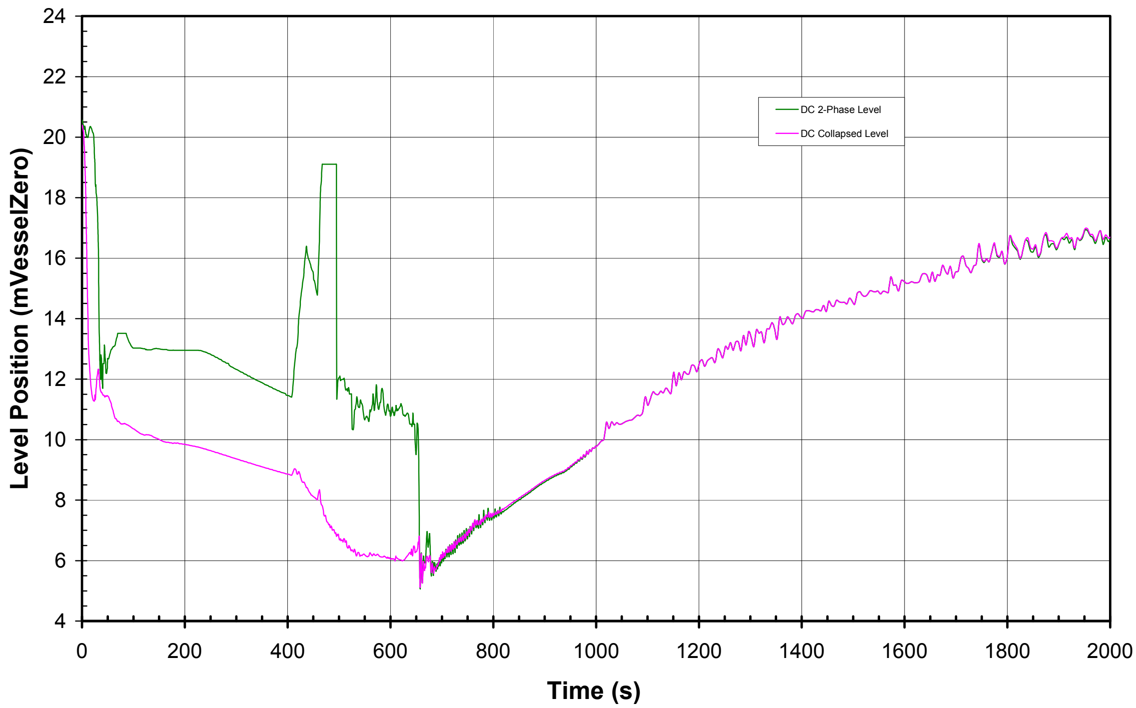


Figure 6.3-38C-a. Downcomer Water Level, IC Drain Line Break (Bounding Case), 1 GDCS Valve Failure (2000 s)

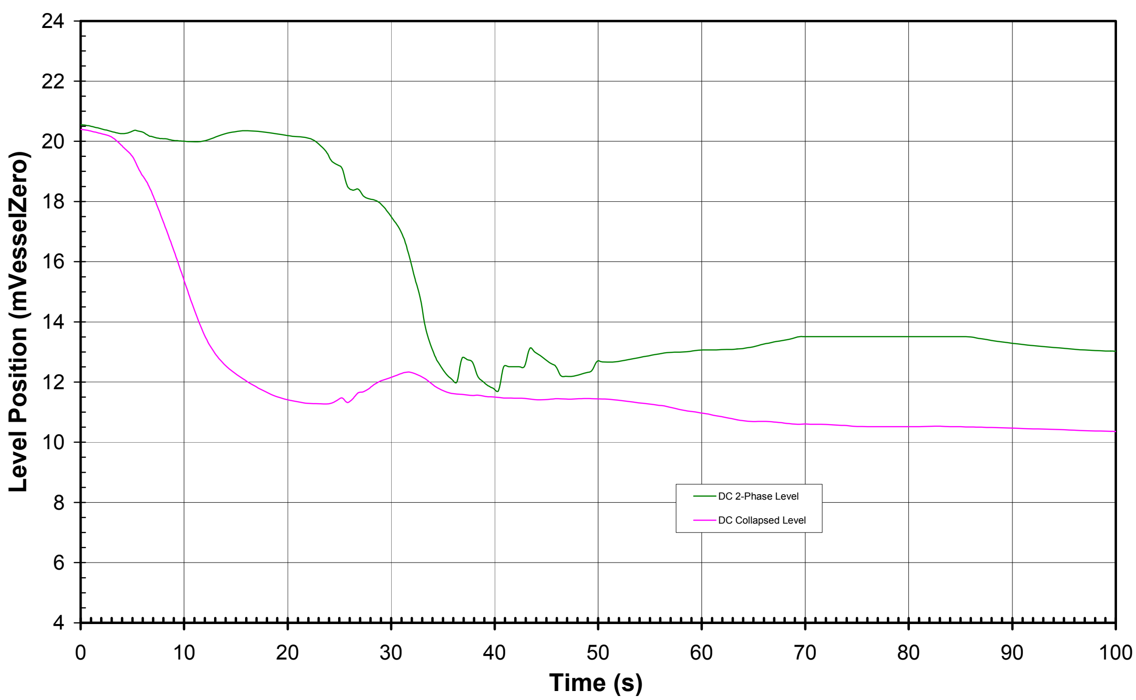


Figure 6.3-38C-b. Downcomer Water Level, IC Drain Line Break (Bounding Case), 1 GDCS Valve Failure (100 s)

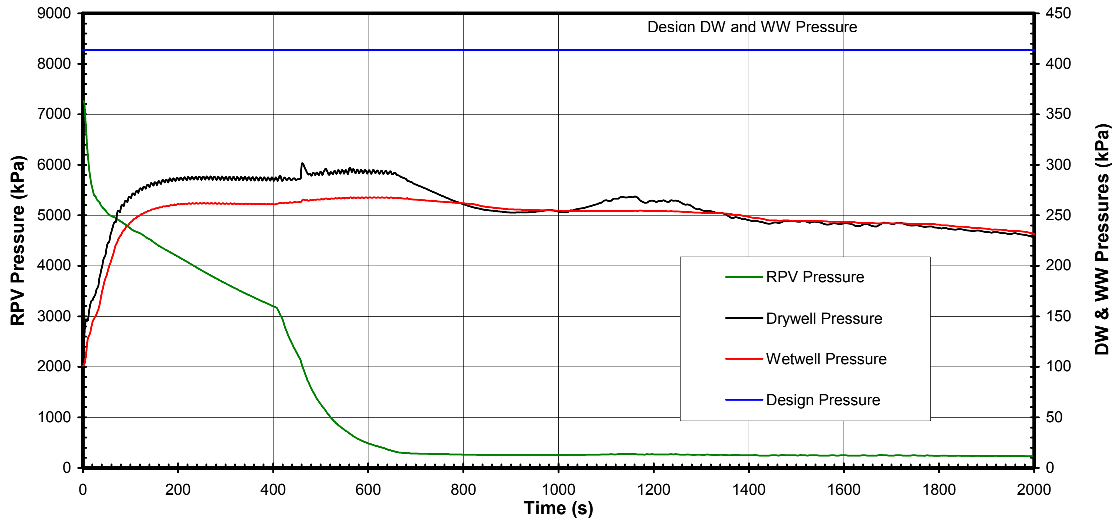


Figure 6.3-38D-a. System Pressures, IC Drain Line Break (Bounding Case), 1 GDCS Valve Failure (2000 s)

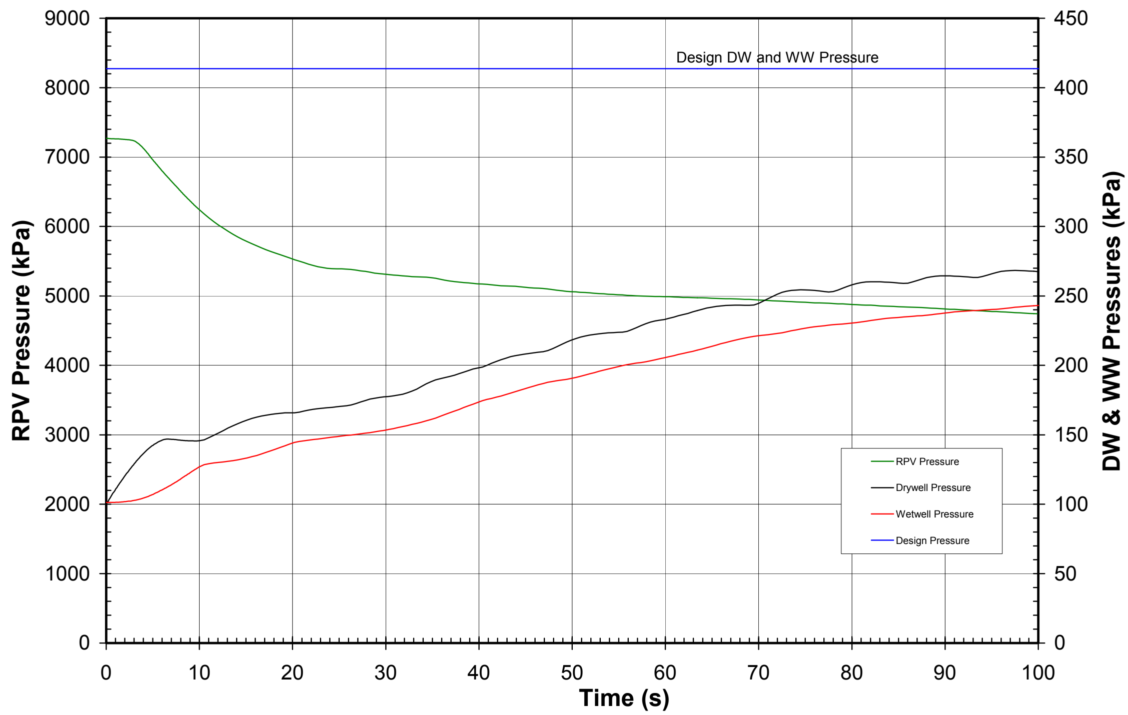
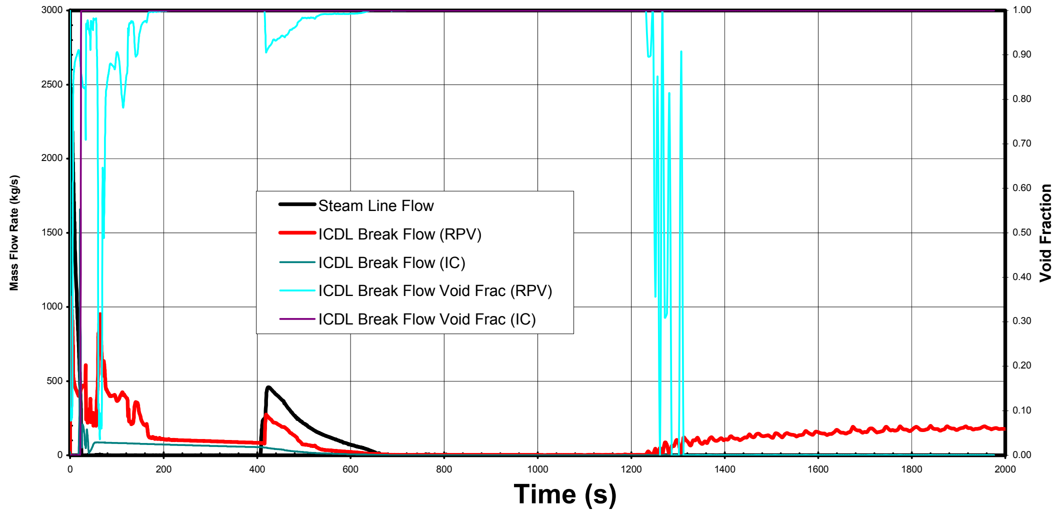


Figure 6.3-38D-b. System Pressures, IC Drain Line Break (Bounding Case), 1 GDCS Valve Failure (100 s)



LEGEND: ICDL = Isolation Condenser Drain Line

Figure 6.3-38E-a. Break Flows and Void Fractions, IC Drain Line Break (Bounding Case), 1 GDCS Valve Failure (2000 s)

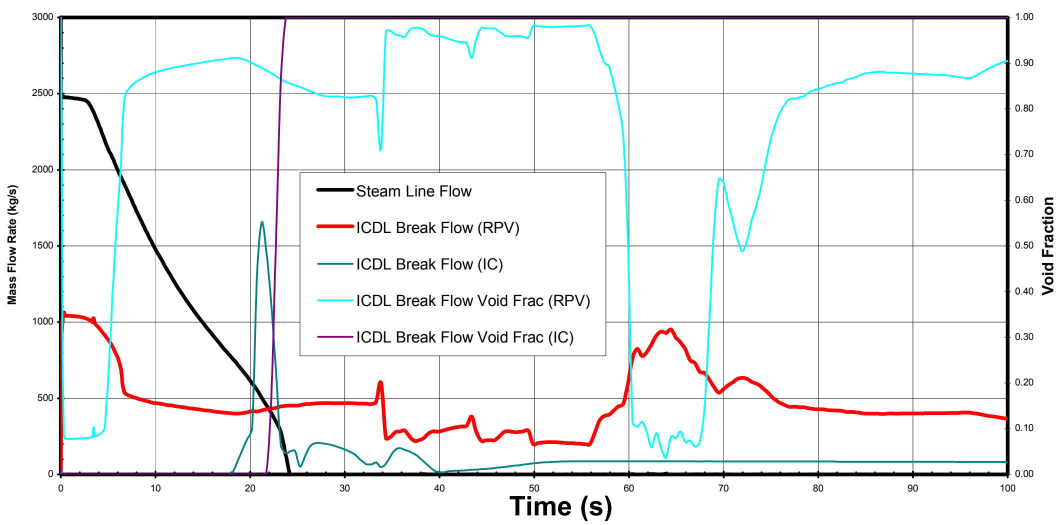


Figure 6.3-38E-b. Break Flows and Void Fractions, IC Drain Line Break (Bounding Case), 1 GDCS Valve Failure (100 s)

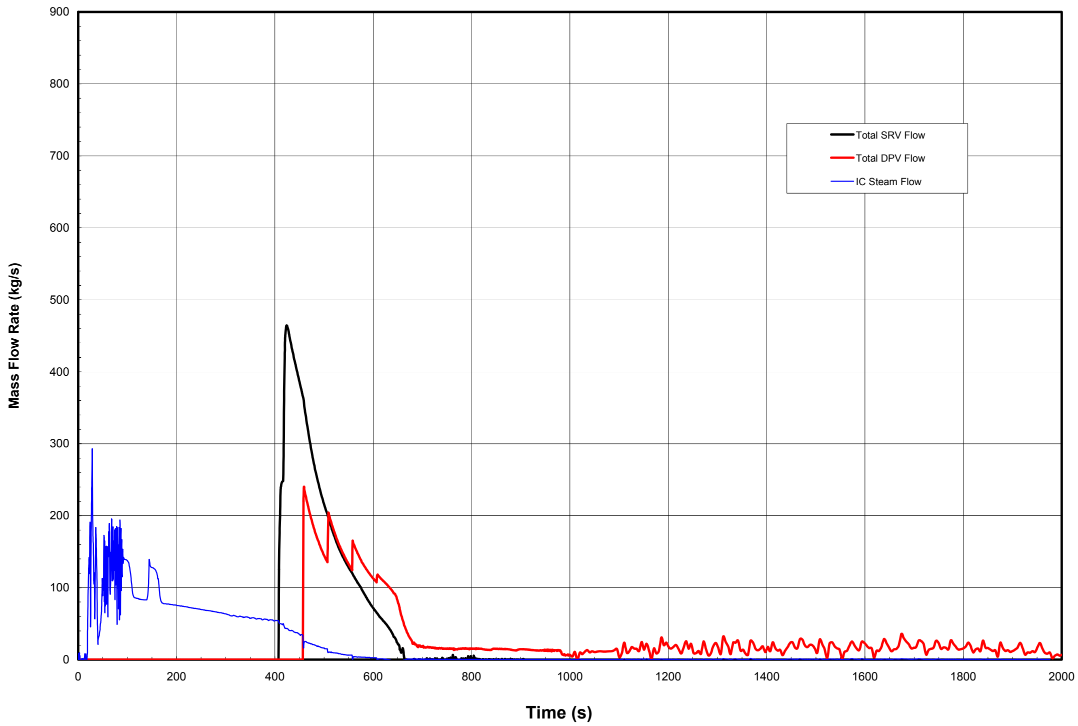


Figure 6.3-38F-a. ADS Flow, IC Drain Line Break (Bounding Case), 1 GDCS Valve Failure (2000 s)

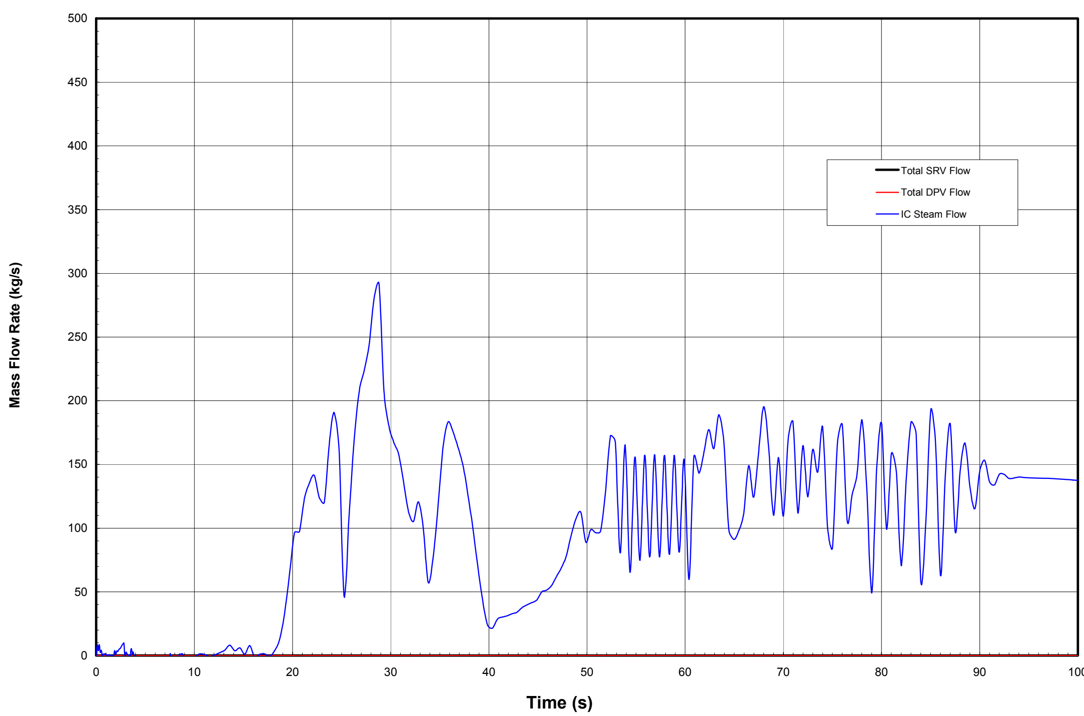


Figure 6.3-38F-b. ADS Flow, IC Drain Line Break (Bounding Case), 1 GDCS Valve Failure (100 s)

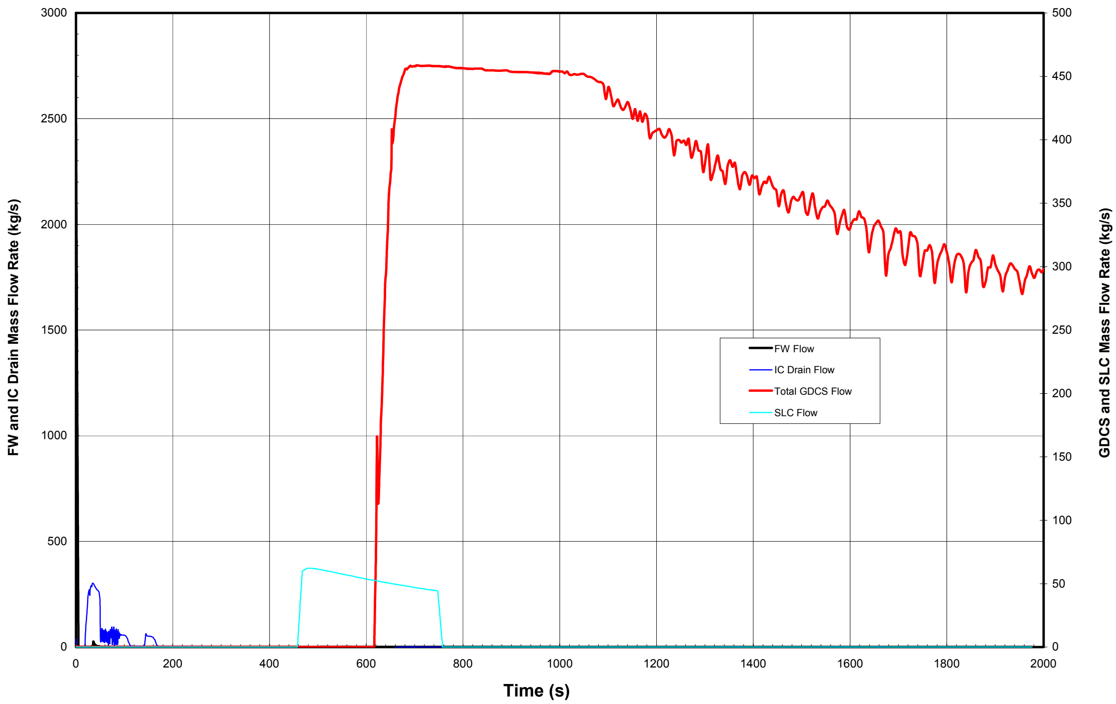


Figure 6.3-38G-a. Flows Into Vessel, IC Drain Line Break (Bounding Case), 1 GDCS Valve Failure (2000 s)

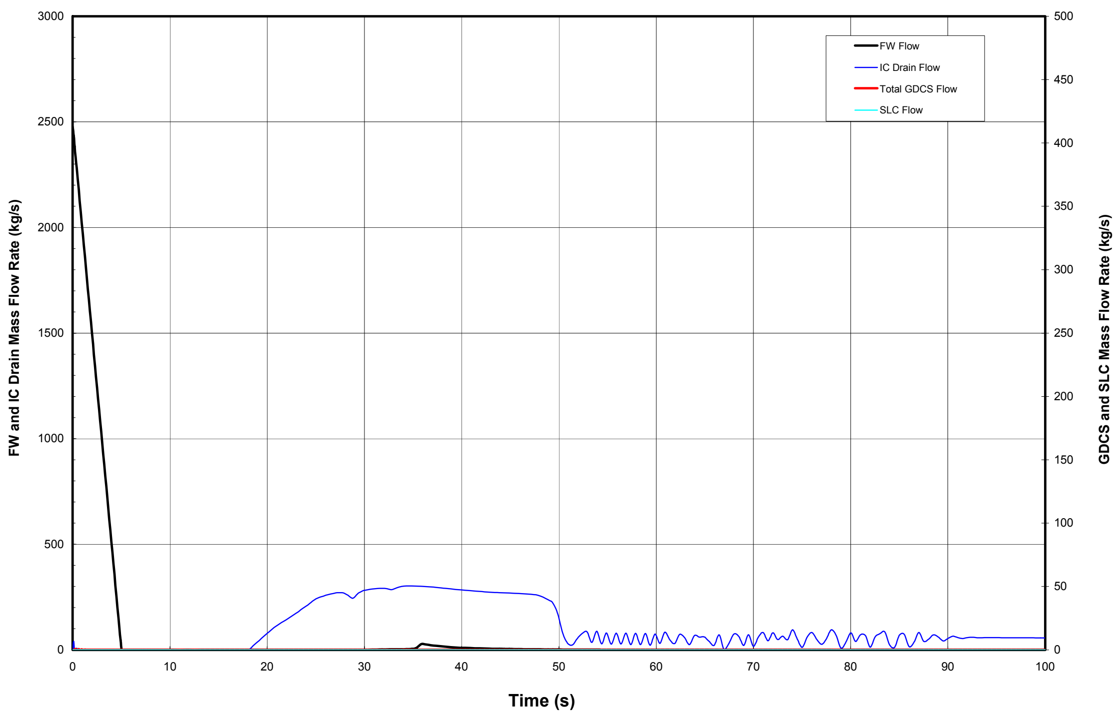


Figure 6.3-38G-b. Flows Into Vessel, IC Drain Line Break (Bounding Case), 1 GDCS Valve Failure (100 s)

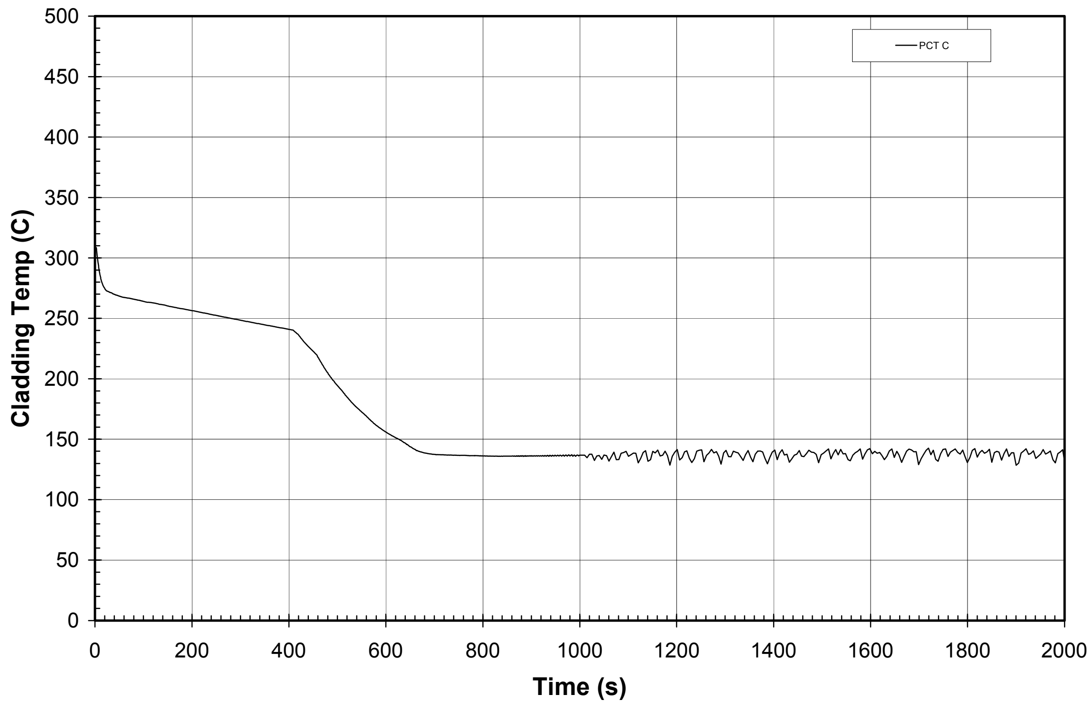


Figure 6.3-38H-a. Peak Cladding Temperature (PCT), IC Drain Line Break (Bounding Case), 1 GDCS Valve Failure (2000 s)

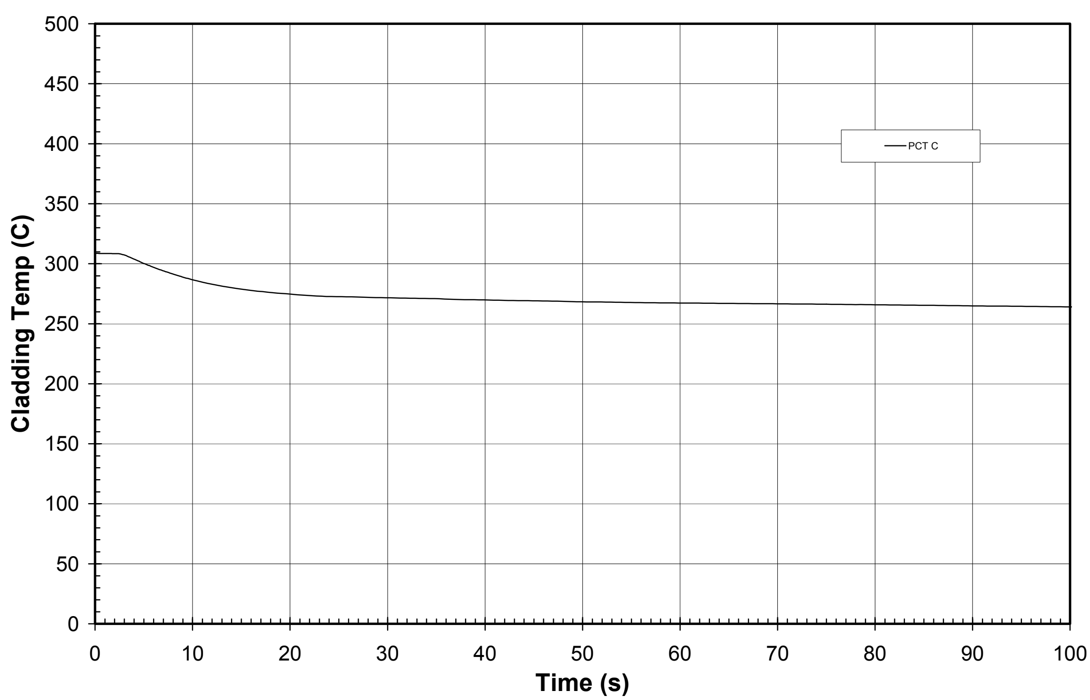


Figure 6.3-38H-b. Peak Cladding Temperature (PCT), IC Drain Line Break (Bounding Case), 1 GDCS Valve Failure (100 s)

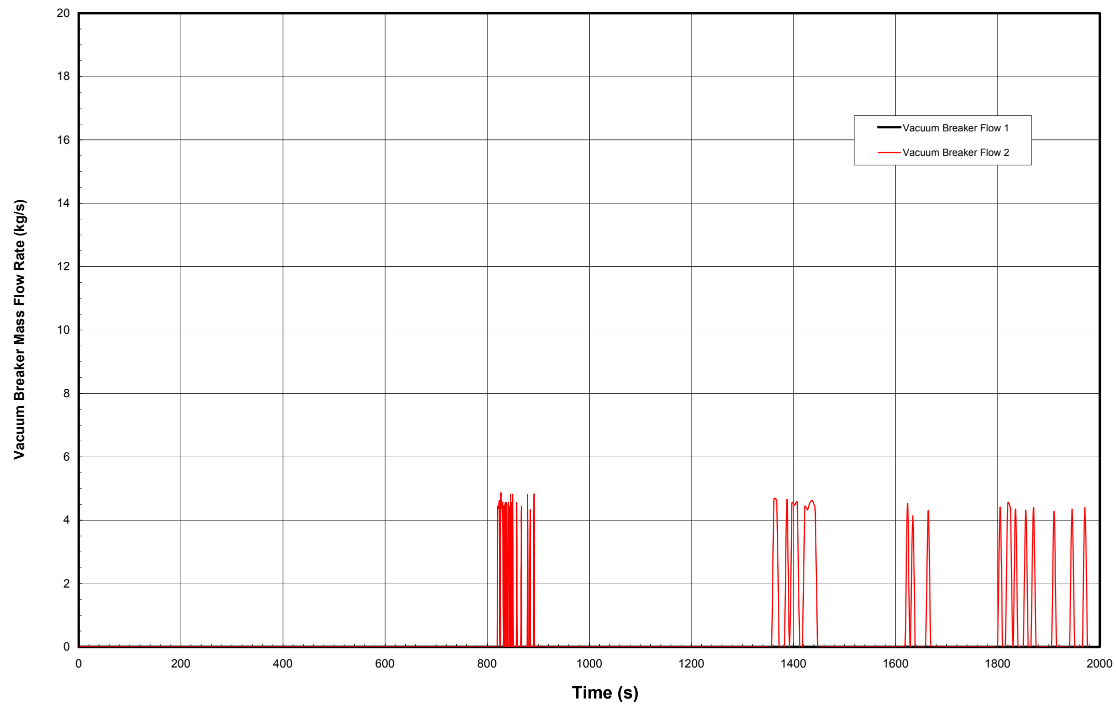
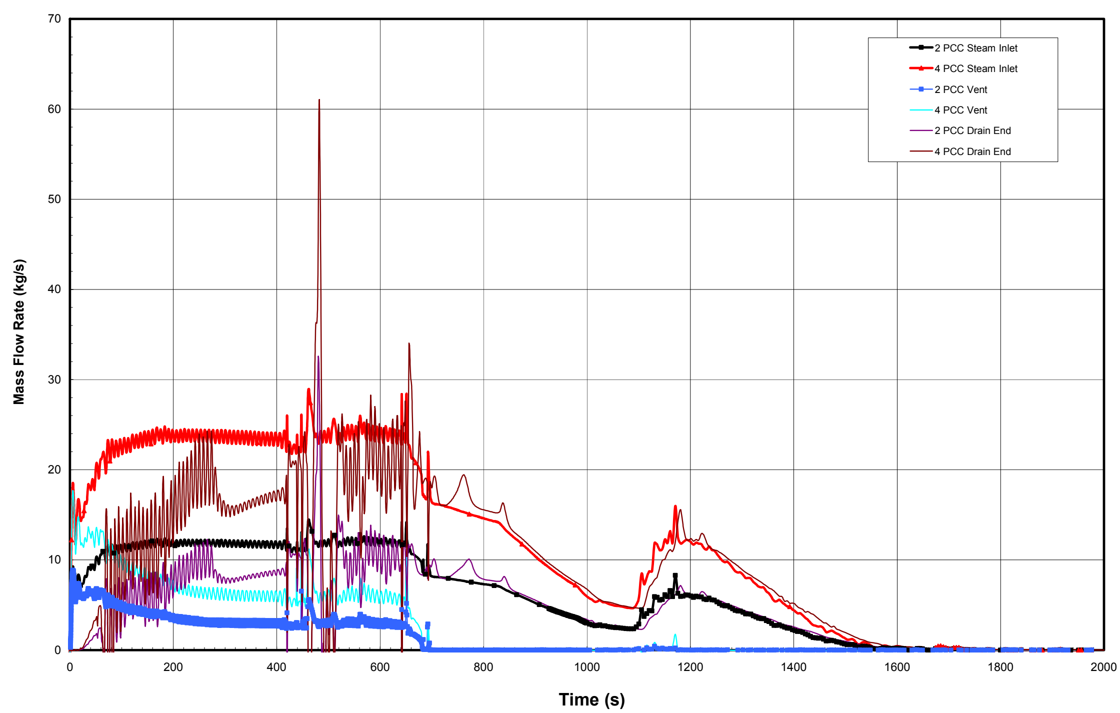
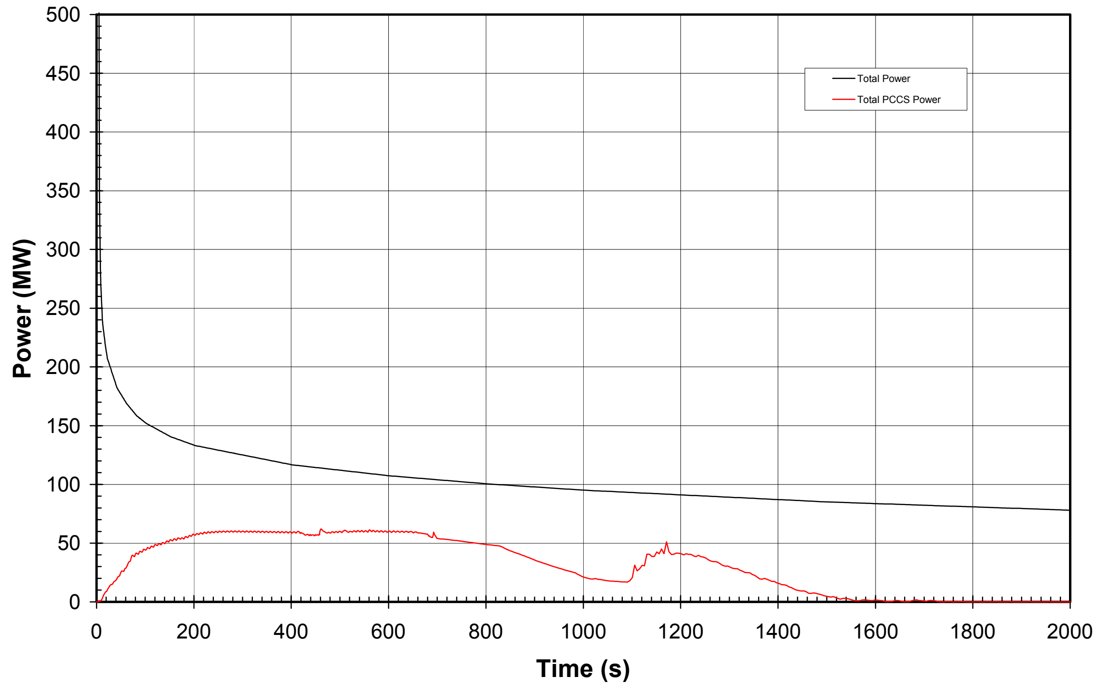


Figure 6.3-38I. Vacuum Breaker Flows, IC Drain Line Break (Bounding Case), 1 GDCS Valve Failure (2000 s)



LEGEND: PCC = Passive Containment Cooling

Figure 6.3-38J. Passive Containment Cooling Flows, IC Drain Line Break (Bounding Case), 1 GDCS Valve Failure (2000 s)



LEGEND: PCC = Passive Containment Cooling

Figure 6.3-38K. Passive Containment Cooling Power, IC Drain Line Break (Bounding Case), 1 GDCS Valve Failure (2000 s)

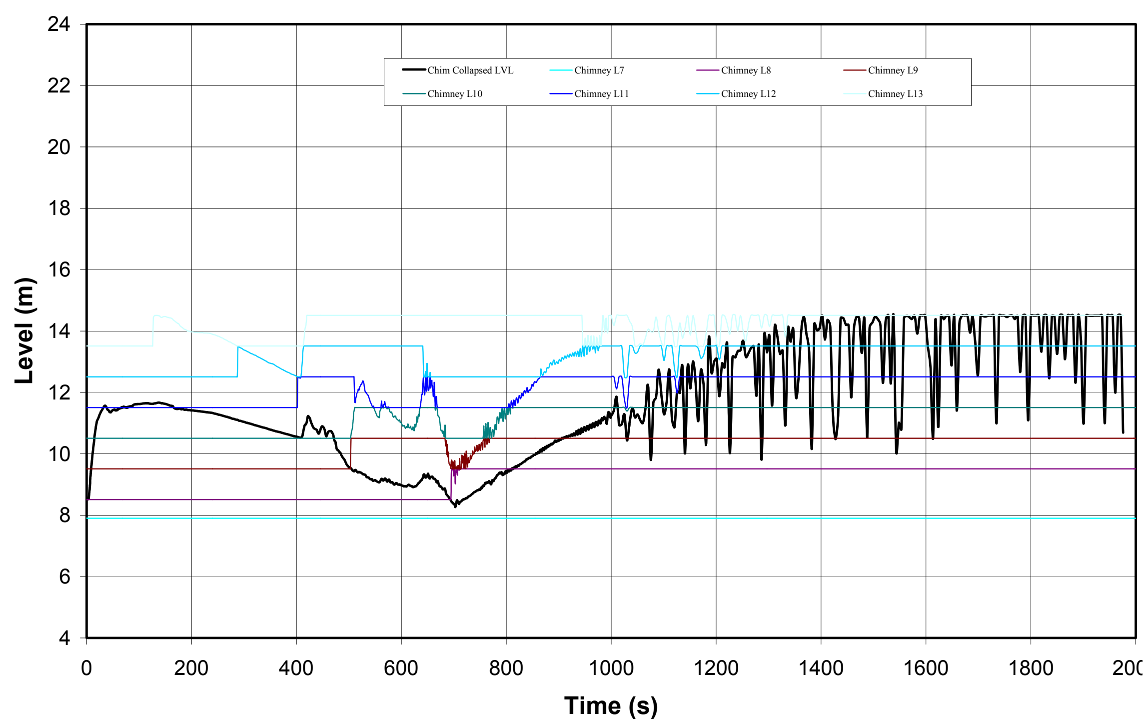


Figure 6.3-38L. Chimney Water Level, IC Drain Line Break (Bounding Case), 1 GDCS Valve Failure (2000 s)

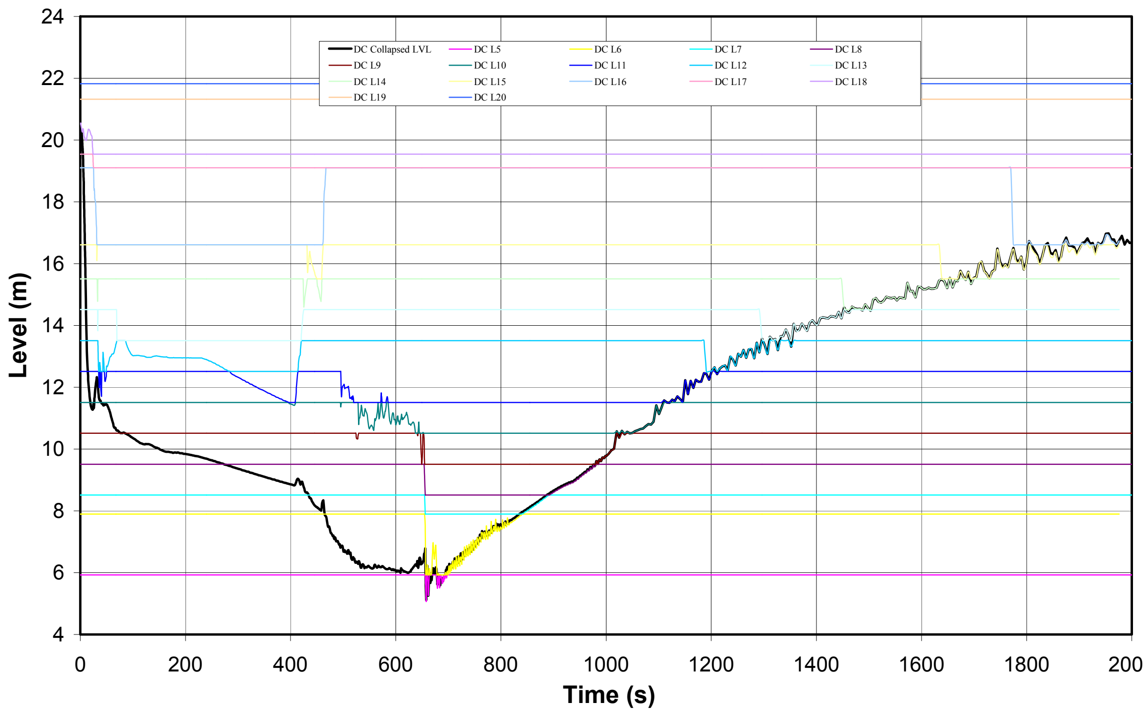


Figure 6.3-38M. Downcomer Water Level, IC Drain Line Break (Bounding Case), 1 GDCS Valve Failure (2000 s)

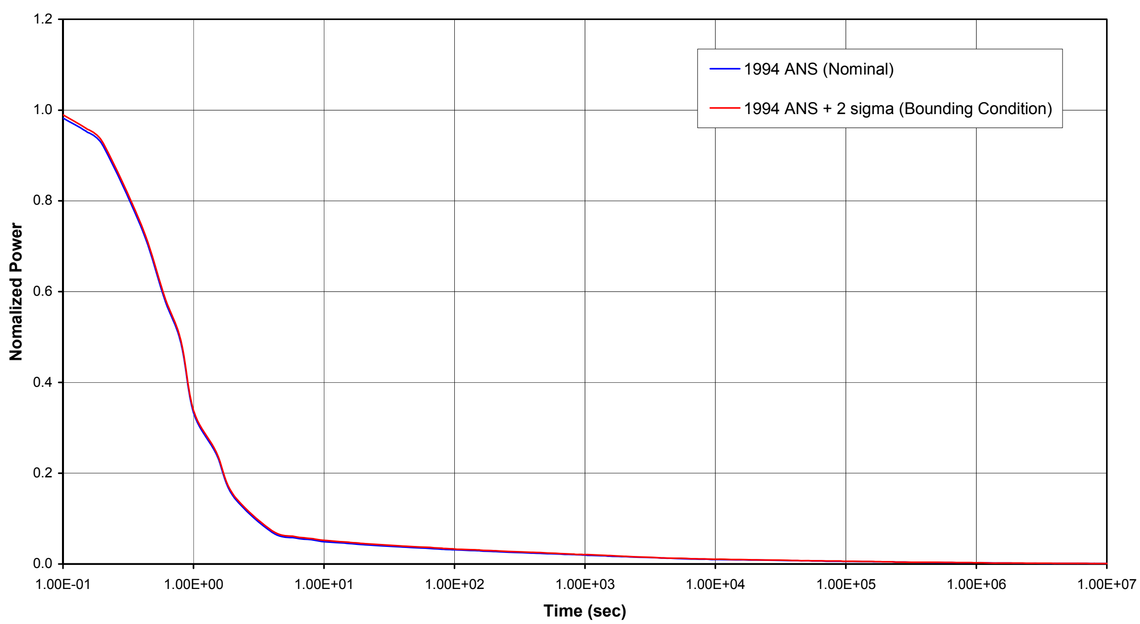


Figure 6.3-39. Normalized Shutdown Power

6.4 CONTROL ROOM HABITABILITY SYSTEMS

The 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);
- Process Radiation Monitoring System (PRMS);
- Lighting System; and
- 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 Loadings	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 Building HVAC System	Subsection 9.4.1
Fire Protection System	Subsection 9.5.1
Lighting System	Subsection 9.5.3
Electric 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 Radiation Monitoring System	Section 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 alternating current (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 PRMS 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 100% capacity fans each, including High Efficiency Particulate Air (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. The CRHA is equipped with a variable orifice relief device to ensure an equal amount of air is exhausted from the CRHA, as supplied. When AC power is unavailable, the CRHA is passively cooled by the CRHA passive heat sink.

The Process Radiation Monitoring System (PRMS) provides radiation monitoring of the CRHA environment and outside air intake. The FPS provides smoke detection and appropriate alarms. 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 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 the ESBWR CRHS, 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 Revison 3. 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 Subsection 1.2.1.

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 Section 15.0 and Subsections 3.1.2, 6.4.5, and 9.4.1 for discussions on conformance with GDC 19, and to Table 1.11-1 for a discussion on conformance with Generic Issues B-36 and B-66.

- The main control room is designed to withstand the effects of a Safe Shutdown Earthquake (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. (Reference 6.4-4). The Emergency Filter Unit System maintains CO_2 concentration in the CRHA to less than 5000 ppm.
- 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. 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 Design Basis Accident (DBA).
- CRHA habitability requirements are satisfied without the need for individual breathing apparatus or special protective clothing.

- The CRHA EFUs and associated fans and ductwork, the CRHA envelope structures, the CRHA heat sink, doors, isolation dampers 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 normal AC power.
- 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 PRMS 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 in-leakage. 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 American Society of Mechanical Engineers (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 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.

The EFU delivery and a variable orifice relief device discharge system is optimized to ensure that there is adequate fresh air delivered and mixed in the CRHA. This is accomplished by using multiple supply registers, which distribute the incoming supply air with the control room air volume, and a remote exhaust to prevent any short cycling. The EFU delivered supply air is distributed in the MCR area of the CRHA. The EFU operation results in turning over the CR volume approximately seven to nine times per day. This diffusion design (mixing and displacement) in conjunction with the known convective air currents (due to heat loads/sinks) and personnel movement ensures that occupied zone temperature is within acceptable limits, buildup of contaminants (e.g., CO_2) is minimal and a freshness of air is maintained.

The occupied zone of the MCR region is normally occupied by personnel, and is generally considered to be between the raised floor and 2 m (6.6 ft) above the floor. Short Cycling refers to the design condition where the outside air transits the served space and exhausts to the outside without mixing. This occurs when the outside air inlet and room exhaust are in close proximity. The CRHA has the fresh air supplied at a high elevation and the exhaust removed low, below the floor, so they are not in close proximity to each other.

6.4.3 Control Room Habitability Area

The CRHA boundary is located on elevation –2000 mm in the Control Building. The layout of the CRHA, which includes the MCR, is shown on Figure 3H-1, Control Room Habitability Area. The CRHA envelope includes the following areas:

- Admin Area (Room 3270)
- RE/Shift Technical Advisor Office (Room 3271)
- Shift Supervisor Office (Room 3272)
- Kitchen (Room 3273)
- Main Control Room (Room 3275)
- Restroom A (Room 3201)
- Restroom B (Room 3202)
- Main Control Room Storage Room (Room 3204)
- Electrical Panel Board Room (Room 3205)
- Gallery (Room 3206)
- Auxiliary Equipment Operators (AEOs) Workshop (Room 3207)
- Air Handling Unit (AHU) Room (Room 3208)

These areas constitute the operation control area, which can be isolated and remain habitable for the duration of a DBA 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 sinks consist of the thermal mass of the concrete that makes up the ceilings and walls of these rooms. The CRHA heat sinks consist of the following: the CRHA walls, floor, ceiling, and interior walls, and access corridors; adjacent Safety-Related Distributed Control and Instrumentation System (Q-DCIS) and Nonsafety-Related Distributed Control and Information System (N-DCIS) equipment rooms and electrical chases; and, CRHA HVAC equipment rooms and HVAC chases. The Control Building concrete characteristics with the material properties of the concrete used in the thermal analysis are provided in Table 3H-14.

After the 72-hour period, the EFU maintains the habitability of the CRHA when RTNSS power supplies are available. The recirculation AHU with supporting auxiliary cooling units is required to remove heat to support main control room habitability post 72 hours.

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 radiological analysis presented in Chapter 15 crediting the control room protective features 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 on Figures 3H-1 and 9A.2-3.

Release Points

Radiological release parameters are described in Sections 15.3 and 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. The exhaust from the CRHA is via a Variable Orifice Relief Device, which is safety-related and its location is optimized to ensure proper scavenging of air from the control room in an amount equal to the supply. Backflow prevention through the controlled leak path, the variable orifice relief device, is not required since the CRHA is at a positive pressure during normal and emergency operation.

- 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 RG 1.52. Each EFU design incorporates 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 American National Standards Institute/Air Management and Control Association (ANSI/AMCA) 210, 301, 302, 303, and 410.

- Isolation Dampers and Valves (including Variable Orifice Relief Device)

The CRHA pressure boundary includes penetrations, dampers or valves, interconnecting duct or piping, and related test connections and manual valves. The isolation dampers or valves are classified as Safety Class 3 (Table 3.2-1) and Seismic Category I. The dampers 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 or valves is demonstrated by pressure testing of the CRHA and in-leakage testing in accordance with American Society for Testing and Materials (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 ANSI/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, is 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 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 62 Pa (1/4 inch w.g.) pressure differential. The bulk penetration sealing material is gypsum cement or equivalent, with epoxy or equivalent sealants applied to complement penetration sealing.
- The CRHA utilizes internal recirculation AHUs that preclude any AHU ductwork external to the CRHA envelope.

The following isolation dampers / components are safety-related and penetrate the CRHA boundary envelope as shown on Figure 6.4-1:

a. Smoke purge intake CRHA isolation dampers, two dampers.

b. Normal Outside Air Intake Supply CRHA isolation dampers, two dampers.

c. Restroom exhaust CRHA isolation dampers, two dampers.

d. Smoke purge exhaust CRHA isolation dampers, two dampers.

e. EFU supply CRHA isolation dampers, two dampers per division, total of four.

f. Variable Orifice Relief Device.

The control room makeup air flow is sized for leakage from the control room boundary when the control room is pressurized to a positive pressure differential of 31 Pa (1/8 inch w. g). An analysis of the control room boundary was performed based on the planned leaktight design features in accordance with the requirements of Standard Review Plan (SRP) Section 6.4, Acceptance Criteria 3.B. This analysis included boundary leakage paths in the control room envelope such as CRHA doors, dampers, and penetrations for piping, electrical conduit, duct and HVAC equipment. Based on the control room total volume and design/construction features employed, the results of the analysis support the feasibility of maintaining the tested differential pressure with the design makeup air flow rate.

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 AHUs 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 upon loss of normal AC power , 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 with the exception of the variable orifice relief device. This relief device is in service exhausting CRHA air during normal and emergency operation. 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, Reference 6.4-3 (COL 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, or
- Extended Loss of Normal 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, 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 EFUs 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 (⅛ inch w.g.) positive differential pressure with respect to the adjacent areas. Excess air is exhausted from the CRHA via the variable orifice relief device. This device is a locked in place orifice or valve set up to maintain CRHA pressure at the delivered flow. The EFU system air flow rate is also sufficient to supply the fresh air requirement of 10.5 l/s (22 cfm) per person for up to 21 occupants (Reference 6.4-4).

Airflow in Emergency Mode

The mixing of the EFU supplied inlet air with the general CRHA air is performed via the following mechanisms:

1. Supply / Inlet registers mixing - The mixing is continuous as EFU provided outside air is delivered to the CRHA. For each cfm delivered it generates mixing with the CR air as it exits the supply registers. This is the most common type of space air diffusion called a Mixing System. The supply air jet is delivered by the air inlet registers, which create an air jet that then mixes the outside air with the room air by entrainment (induction), which helps to reduce the jet velocity and equalize the supply air temperature as it enters the CRHA.
2. Displacement (Ventilation) Supply / Exhaust - As air is supplied to the CRHA, an amount is similarly exhausted from the space. This exhaust air is designed to be at a remote location to ensure no short cycling and a properly scavenged control room.
3. Equipment and Personnel Convective Plumes due to air differential temperature / density – The higher temperature of the air surrounding operating equipment and personnel, generates convective air plumes which rise out of the occupied zone, along with any pollutants (body odors, etc.). The rising air is replaced by cooler air from below.
4. Personnel Movement – It is assumed that a certain activity level by the CRHA occupants which derived the airflow requirements. This activity generates mixing of the CRHA air.
5. Molecular Dispersion - For Contaminants, CO_2 and others, the movement of CO_2 and other molecules across a space is via molecular dispersion.

The airflow developed in the ESBWR control room during worst case (outside air temp of 117ºF) accident conditions when the CRHA is isolated and the EFU is in operation with passive cooling is as follows and is illustrated in Figure 6.4-2.

The EFU is operating providing 466 cfm clean outside airflow into the CRHA. This is delivered to the occupied MCR area, primarily since this area has the personnel on duty and houses the active electronic equipment. This supply air exits the ductwork at supply air diffusers (4), which perform mixing, mechanism 1) above. Depending upon the delivered air temperature, the combined mixed volume either rises or drops. At the worst case outside air condition of 117ºF, modeling shows this air mixture rises above the ceiling with a larger quantity of MCR heated air; the balance driven primarily by the equipment and personnel convective plumes, mechanism 3) above. The combined, rising air above the ceiling tiles draws the same quantity of air into the MCR space from the area below the raised floor volume, mechanism 2). This cooler, slow moving air slowly spreads over the raised floor and displaces the warmer, stale air toward the ceiling, where it leaves the room. The MCR with the high ceiling becomes thermally stratified, i.e., warmer stale air is concentrated above the occupied zone and cool, fresher air is concentrated in the occupied zone. When the cool air encounters a heat source, such as a person or heat generating equipment, the air heats up and buoyantly rises out of the occupied zone.

The hot air that collects above the suspended ceiling, with CO_2 and body generated odors, spills over into the adjacent rooms due to the air density difference due to differential temperature where heat is released to the cooler walls and concrete. Cooler lower temperature air in these adjacent rooms drops to the raised floor level where air continues to drop thru to the common space below the floor. Discharge flow of 466 cfm of this air, exits the main control room at a remotely opposite location from EFU supply to prevent any short cycle of the supply air and ensure a constant turnover of the CRHA air. This air then is drawn into the MCR and a circuit is complete.

A positive pressure is maintained in the CRHA. There is no buildup of any CO_2 in any of these areas since the areas are scavenged continuously by the EFU supply with exhaust airflow of 466 cfm. The exhaust is remote to the supply at one of the adjacent rooms lower common area.

With a source of AC power available, the EFU can operate and is controlled indefinitely through Q-DCIS. In the event that normal 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 post 72 hrs, each EFU fan is powered via an electrical bus supplied by one (1) of two (2) ancillary diesel generators. 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 6.4-1 and 6.4-2) over a 72-hour period. The CRHA isolation dampers fail closed on a loss of normal AC power or instrument air.

Backup power to the safety-related Control Room EFU fans (post 72 hours) if normal AC power is not available is provided by two (2) ancillary diesel generators. These generators are required to support operation of the Control Room EFU beyond 72 hours after an accident. For a period between 7 days and the duration of the design basis accident, the safety-related function of the EFU can be powered from either offsite power, onsite diesel generator powered Plant Investment Protection (PIP) bus, or by continued use of the ancillary diesel generators. The requirements for the ancillary generators are described in Appendix 19A.

Upon a loss of normal AC power, the initial ranges of temperature/relative humidity in the CRHA are 21.1-23.3°C (70-74°F) and 25%-60% RH. The CRHA temperature / humidity values calculated during the 72 hours following a design basis accident equate to less than 32.2°C (90°F) Wet Bulb Globe Temperature (WBGT) index. The 32.2°C (90°F) WBGT index value is the acceptability limit for minimizing performance decrements, potential harm, and preserving well-being and effectiveness of the control room staff for an unlimited duration. (Reference 6.4-5). During the first two hours of loss of normal AC power, most of the equipment in the MCR remains powered by the nonsafety-related battery supply. Anytime during a loss of normal AC power, once either ancillary diesel is available, the environmental conditions are maintained indefinitely. This is accomplished via the continued operation of a CRHA recirculation AHU and auxiliary cooling unit supplied with each recirculation AHU. Power is provided during the initial two hours from the same nonsafety-related battery supply that powers the non-safety MCR equipment. At any time during the initial two hours, or after, power can be provided by an ancillary diesel. If this cooling function is lost, the N-DCIS components in the MCR are automatically de-energized. This is accomplished via safety-related temperature sensors with two-out-of-four logic that automatically trip the power to selected N-DCIS components in the MCR, thus removing the heat load due to these sources.

The remaining CRHA equipment heat loads are dissipated passively to the CRHA heat sinks. The CRHA heat sinks limit the temperature rise to that listed in Table 6.4-1 by passively conducting heat into the heat sinks. The CRHA heat sinks consist of the following: the CRHA walls, floor, ceiling, and interior walls, and access corridors, adjacent Q-DCIS and N-DCIS equipment rooms and electrical chases; and CRHA HVAC equipment rooms and HVAC chases. The Control Building thermal analysis, including the CRHA, is presented in Subsection 3H.3.2. The temperatures presented in the analysis are acceptable for human performance and equipment qualification.

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 scenarios where the EFU provides filtered air to pressurize the CRHA. Doses are calculated for the following accidents:

1000 Fuel Rod Failure Dose Results	Table 15.3-16
Radwaste System Failure Accident Dose Results	Table 15.3-19
LOCA Inside Containment Analysis Total Effective Dose Equivalent (TEDE) Results	Table 15.4-9
Main Steamline Break Accident Analysis Results	Table 15.4-13
Feedwater Line Break (Analysis) Results	Table 15.4-16
Small Line Carrying Coolant Outside Containment Break Accident Results	Table 15.4-19
RWCU/SDC Line Break Accident Results	Table 15.4-23

The dose analyses are performed in accordance with the requirements of RG 1.183. For all events, the control room dose is within the dose acceptance limit of 5.0 rem (50 mSv) total effective dose equivalent (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. The control room ventilation inlet distances from potential release points are maximized to the extent possible. However, the separation distances in SRP Section 6.4 are not always met. Not meeting these distances is acceptable because the dose analyses developed for the CRHA used actual plant layout of the Control Building intake louvers and potential release points. 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 and ancillary 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 and ancillary diesel generators 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 is in accordance with RG 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 (Subsection 6.4.9).

During emergency operation, the CRHA emergency habitability system passive heat sink is designed to limit the temperature inside the CRHA to 33.9°C (93°F). This maintains the CRHA within the limits for reliable human performance (References 6.4-1 and 6.4-2) 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 input parameters assumed in the Control Building Heatup Analyses are listed in Table 3H-14. The EFU portion of the CRHAVS provides a minimum 220 l/s (466 cfm) of ventilation air to the main control room and is sufficient to pressurize the control room to at least a positive 31 Pa (⅛ inch w.g.) differential pressure with respect to the adjacent areas. This flowrate also supplies the recommended fresh air supply of 10.5 l/s (22 cfm) per person for a maximum occupancy of 21 persons (Reference 6.4-4).

The normal and emergency (EFU) filter unit outside air intake flows are adjusted as required to maintain a minimum flow and, in conjunction with a controlled leak path, maintain a 31 Pa (⅛ inch w.g.) minimum positive pressure in the CRHA relative to adjacent areas. Flow instrumentation is provided for the fans and AHUs to ensure airflow is maintained above the minimum required flow. A low airflow alarm is provided. CRHAVS differential pressure transmitters are provided to monitor CRHA pressure with respect to adjacent areas and ensure the pressure is maintained above the minimum positive pressure. A low CRHA differential pressure alarm is provided. A variable leakage device, located under the raised floor to facilitate air circulation and mixing, is provided with sufficient adjustment to maintain the required airflow and CRHA positive pressure relative to adjacent areas under all normal and emergency conditions requiring operation of the CRHA AHU or EFU. Periodic monitoring of the CRHA air intake flows and positive CRHA differential pressure is performed during operation of the CRHA AHU or EFU. 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, or
- Extended Loss of Normal 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. 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 Subsection 6.4.4, Subsection 7.3.4.2, and in Subsection 9.4.1.2. The EFUs automatically start during a radiological event, independent of the loss of normal AC power. Through the use of redundant EFU components and dampers, one EFU and supply path to the CRHA would be available during a loss of normal AC power with failure of up to two divisions of safety-related power to provide CRHA breathing air and pressurization during a loss of AC concurrent with a radiological event. Local, audible alarms warn the operators to shut the self-closing doors, if for some reason they are open.

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. The COL Applicant will determine the protective measures to be instituted to ensure adequate protection for control room operators as recommended under RG 1.78. These protective measures include features to (1) provide capability to detect releases of toxic or hazardous materials, (2) isolate the control room if there is a release, (3) make the control room sufficiently leak tight, and (4) provide equipment and procedures for ensuring the use of breathing apparatus by the control room operators (COL 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. Design changes to the CRHA will ensure key design assumptions are met such as:

- Heat sink / Heat source assumptions
- Air flow assumptions
- Heat transfer values

This will ensure that CRHA calculations and methodologies are maintained and updated throughout the life of the plant.

Preoperational Inspection and Testing

Preoperational testing of the CRHAVS is performed to verify that the minimum air flow rate of 220 l/s (466 cfm) is sufficient to maintain pressurization of the main control room envelope of at least 31 Pa (⅛ inch w.g.) with respect to the adjacent areas. The variable orifice relief device is set during this evolution to ensure an equal amount of air is exhausted from the CRHA. 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 below regulatory limits. Air quality within the CRHA environment is confirmed to be within the guidelines of American Society of Heating, Refrigeration and Air Conditioning Engineers (ASHRAE) Standard 62.1 requirements for continued occupancy via meeting the fresh air supply requirement of 10.5 l/s (22 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 are 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 leak tightness on the envelope walls, doors, and boundaries. Testing to demonstrate the integrity of the CRHA envelope is performed in accordance with RG 1.197 and ASTM E741.

The Control Room EFU supplies air with a design flow rate of 220 l/s (466 cfm), and it is designed to maintain the control room envelope at a positive pressure with respect to adjacent compartments during normal operation and radiological events. An intake filter efficiency of 99% is assumed for particulate, elemental, and organic iodine species. The system does not include filtered recirculation and the design incorporates leak tightness design requirements (Section 6.4.3). Although the control room is maintained at a positive pressure, the dose analysis assumes a 5.66 l/s (12.0 cfm) unfiltered inleakage. Based on the ESBWR CRHA design and ventilation system operation the acceptance criteria for inleakage associated with the CRHA will be no greater than the amount of unfiltered leakage assumed in the Dose Consequence Analysis minus 2.36 1/s (5 cfm) which is the amount of unfiltered inleakage allocated for ingress and egress. This leakage estimate value is conservative.

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 RG 1.52.

Periodic surveillance testing of safety-related CRHA isolation dampers and the EFU components are carried out per Institute of Electrical and Electronic Engineers (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 Subsection 7.3.4. 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 Subsection 11.5.3. 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 room temperature (nonsafety-related temperature detection);
- High room temperature (safety-related temperature detection);
- Low room temperature;
- 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 station blackout (SBO);
- Area high radiation in the CRHA; and
- High radiation in the Outside Air Intake duct.

If the redundant, nonsafety-related CRHAVS cooling is lost, and the CRHA temperature increases, safety-related sensors provide a trip signal via SSLC/ESF to de-energize selected nonsafety N-DCIS equipment located in the CRHA. Safety-related temperature sensors monitoring CRHA temperatures provide the logic to trip selected N-DCIS loads in the CRHA. A common alarm is provided to indicate a high CRHA air temperature and a potential high thermal heat sink temperature. Further, this high temperature alarm setting is set below the N-DCIS trip setpoint. This early detection of rising CRHA and heat sink temperatures provides for early operator attention and action prior to tripping selected N-DCIS loads in the main control room and ensures appropriate actions will occur prior to experiencing temperatures in excess of those assumed in the CRHA heatup calculation. CRHA heat sink temperatures are assumed to be within the specified limit if the average of the air temperatures in the heat sink has been within the specified limit. The temperature response of the CRHA heat sink area materials is slower than the response of the average air temperature on increasing temperature, i.e., a loss of normal cooling. Since the CRHA air temperature will respond quicker than the materials in the heat sink (notably concrete), this approach is conservative. If the average of the CRHA air temperatures exceed the specified limit, restoration of the CRHA heat sinks is verified by administrative evaluation considering the length of time and extent of the CRHA heat sink average air temperature excursion outside of limits, or by direct measurement of the CRHA heat sink area structural materials temperatures.

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, Reference 6.4-3 (Subsection 6.4.4) including statements under 6.4.7.

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. The COL Applicant will determine the protective measures to be instituted to ensure adequate protection for control room operators as recommended under RG 1.78. These protective measures include features to (1) provide capability to detect releases of toxic or hazardous materials, (2) isolate the control room if there is a release, (3) make the control room sufficiently leak tight, and (4) provide equipment and procedures for ensuring the use of breathing apparatus by the control room operators (Subsection 6.4.5).

6.4.10 References

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

6.4-2 MIL-STD-1472F, Human Engineering.

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

6.4-4 ASHRAE Standard 62.1/2007, Ventilation for Acceptable Indoor Air Quality.

6.4-5 Human-System Interface Design Review Guidelines, NUREG-0700, May 2002.

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)
	Winter: -40.0°C (-40°F) Dry bulb
Inside Design temperatures and humidity:	
CRHA (normal operation)	21.1°C (70°F) to 23.3°C (74°F) and 25% to 60% RH
CRHA (SBO and Accident Conditions)	Maximum 33.9°C (93°F) temperature for the first 72 hours into the event, RH not controlled
Pressurization	> 31 Pa (⅛ inch w.g.) positive differential
CRHAVS EFUs:	
CRHAVS Breathing air supply capacity	≥10.5 l/s (22 cfm) person for up to 21 persons (≥220 l/s total) (≥466 cfm total) for 72 hours Note: CRHA heat up analysis assumes 11 control room occupants for CRHA thermal loading (Table 3H-12)
Quantity	2 - 100% capacity each
Capacity	Flow – ≥220 l/s per unit (≥466 cfm)
Type	Metal housing containing medium efficiency pre-filter, HEPA filter, carbon filter, and post filter
Differential Pressure	500 Pa (2.0” w.g.) maximum
Medium efficiency filter minimum ASHRAE efficiency	40%
HEPA filter minimum efficiency	99.97% DOP
Post-filter minimum efficiency	95% DOP
Carbon Adsorber Requirements	10.2 cm (4 inch) minimum 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 TB, West side
Hydrogen	Generator Gas Control System	H_2 Storage Area - Outside the TB, 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
Diesel	Diesel Generator Fuel Oil System	Diesel Generator Building and Yard Areas

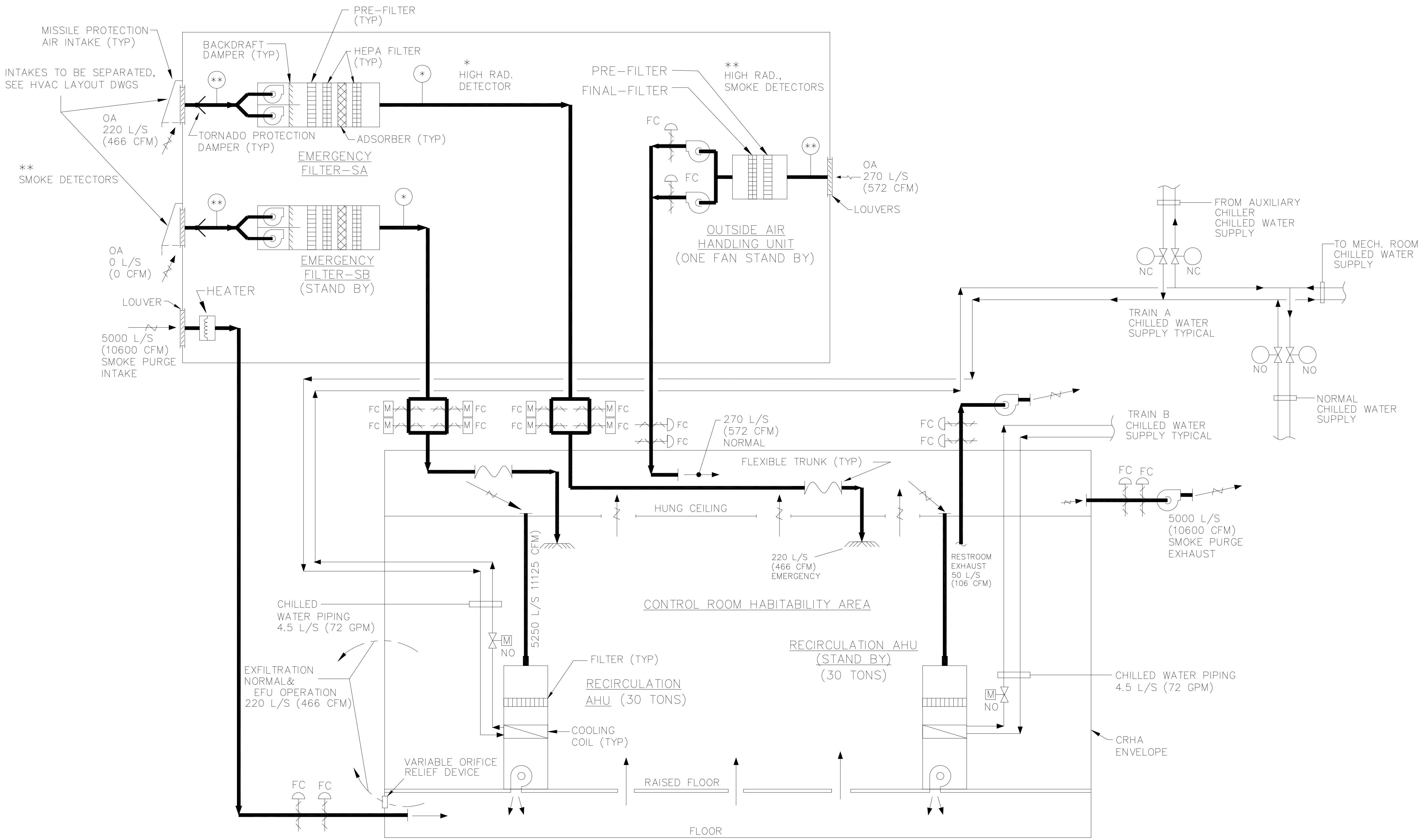


Figure 6.4-1. CRHAVS Schematic Diagram

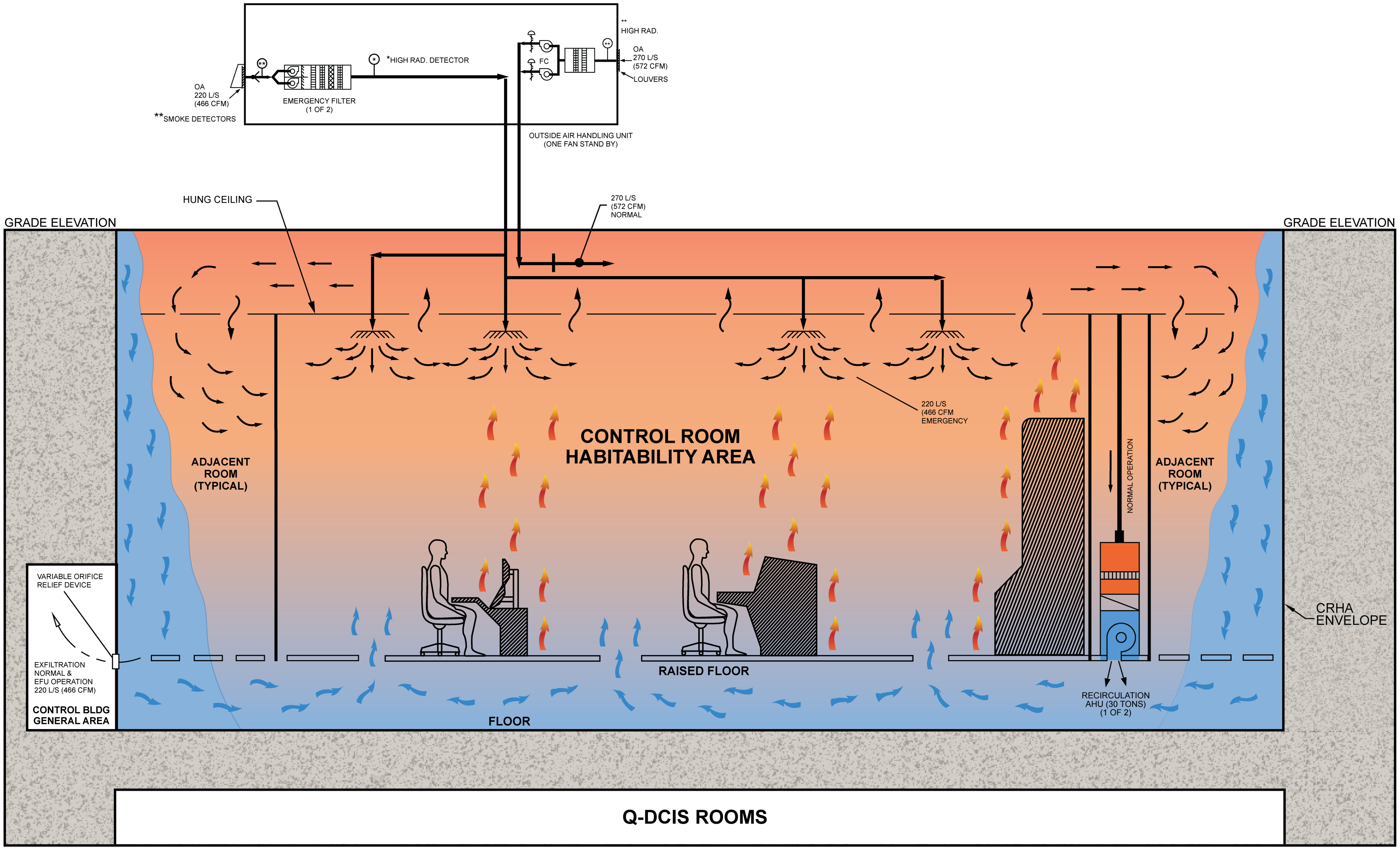


Figure 6.4-2, Control Room Habitability Area Airflows Emergency Operation - FOR ILLUSTRATIVE PURPOSES ONLY[ms1868]

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 SRP 6.5.1.

6.5.1 Containment Spray Systems

The ESBWR contains DW containment sprays, which can be initiated manually 72 hours after a LOCA to cooldown the containment to aid in post-accident recovery or to mitigate the effects of a beyond design basis severe accident. The spray system is nonsafety-related, and no credit is taken for removal of fission products in design basis accident evaluations due to this spray system. Therefore, the acceptance criteria in SRP 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 (Section 12.2), the radwaste building for processing of liquid and solid radioactive waste streams (Chapter 11), and the TB which contains one liquid treatment system and the offgas gaseous waste treatment system (Chapter 11). The RB is further divided into three distinct areas: one clean, and two potentially contaminated 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 (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 DW, with the DW being further divided into upper and lower DW 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 DW and from the DPVs through a series of vertical and horizontal vents between the DW and the suppression pool water, and from the Safety Relief Valves (SRVs) through a series of downcomer pipes through quenchers in the suppression pool water. During such an event, releases would then be subject to 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 DW 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 are controlled as described in Subsection 6.2.5.

6.5.2.3 Reactor Building

The RB completely surrounds the containment and the RBVS is divided into clean and contaminated areas (Subsection 9.4.6). Under normal conditions, the contaminated areas (CONAVS and REPAVS areas) air flow is maintained from clean to potentially contaminated areas and then routed via the respective RBVS subsystem to the RB/FB vent stack. Under accident conditions, the RB (CONAVS and REPAVS areas) automatically isolates on high radiation to provide a holdup volume for fission products. When isolated, the RB (CONAVS and REPAVS areas) can be serviced by the RBVS Accident Exhaust Filter units (Subsection 9.4.6). Under high energy release conditions such as a HELB, the overpressure is routed to the HELB chase in which blow out panels relieve the overpressure to the environment. The RBVS system performs no safety-related function, other than the building ventilation isolation function, but credit is taken for holdup in the RB as discussed in Subsection 15.4.4.5.2.

The controlled (CONAVS served)area of the RB surrounds most of the containment (except feedwater and MSIV containment penetrations located in the main steam tunnel) and provides a barrier for airborne leakage of fission products resulting from containment leakage including containment penetrations. 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 CONAVS area of 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 high efficiency particulate air (HEPA) and charcoal filtration systems of the RBVS Accident Exhaust Filter Unit (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/PCCS 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. Main steamline and feedwater line leakage is discussed in Subsection 15.4.4.5.2.

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 radwaste building 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 TB 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 TBVS to the TB stack. The TBVS 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 DW 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 through testing as described 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 Inservice Inspection (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 Inspection (PSI), ISI 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.

According to the ASME B&PV Code, Section XI, either UT or radiographic examination (RT) may be used for ISI of B&PV Code Class 1 and 2 austenitic and dissimilar metal (DM) welds. The COL Applicant is responsible for developing a plan and providing a full description of its use during construction, PSI, ISI, and design activities for components that are not included in the referenced certified design, to preserve accessibility to piping systems, to enable nondestructive examinations (NDE) of ASME Code Class 2 austenitic and DM welds during ISI (COL 6.6-2-A).

PSI and ISI programs for Class 2 and 3 components and piping are 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 development of PSI and ISI programs is the responsibility of the COL Holder, and shall be based on the ASME Code, Section XI, edition and addenda approved in 10 CFR 50.55a(b) twelve months before initial fuel load. The COL Applicant is responsible for providing a full description of the PSI/ISI programs and augmented inspection programs for Class 2 and 3 components and piping by supplementing, as necessary, the information in Section 6.6. The COL Applicant will also provide milestones for program implementation (COL 6.6-1-A).

6.6.1 Class 2 and 3 System Boundaries

The Class 2 and 3 system boundaries for both the PSI and ISI programs and the system pressure test program item boundaries include all or part of the following:

- Nuclear Boiler System.
- Isolation Condenser System.
- Control Rod Drive system.
- Standby Liquid Control system.
- Gravity Driven Cooling System.
- Fuel and Auxiliary Pools Cooling System.

- Reactor Water Cleanup/Shutdown Cooling system.
- Chilled Water System.
- 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 RG 1.26, for Quality Group B 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 Subsection 6.6.1 within the Class 3 boundary, based on RG 1.26 for Quality Group C 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 RG 1.183), that exceed 5 mSv (0.5 rem) TEDE.

6.6.2 Accessibility

All items within the Class 2 and 3 boundaries are designed to provide access for the examinations required by ASME B&PV Code, Section XI, IWC-2500 and IWD-2500.

[*The ESBWR design includes specific access requirements, in accordance with 10 CFR 50.55a(g)(3), to support preferred UT or optional RT examinations. The design of each component and system takes into account the NDE method, UT or RT, that will be used to fulfill PSI and ISI examination requirements and will take into full consideration the operational and radiological concerns associated with the method selected to ensure that the performance of the required examination will be practical during commercial operation of the plant. Additionally, the design procedural requirements for the 3D layout of the plant include acceptance criteria regarding access for inspection equipment and personnel.*]* However, with respect to any design activities for components that are not included in the referenced ESBWR certified design, it is the responsibility of the COL Applicant to preserve accessibility to piping systems to enable NDE of ASME Code Class 2 austenitic and DM welds during ISI (COL 6.6-2-A).

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

The design and physical arrangement of piping, pumps, valves, and supports 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 ASME B&PV Code, Section XI, Subsection IWF. The design of the nuclear power plant structures, systems, and components provides access for the performance of Inservice Testing (IST) and ISI as required by the applicable ASME Code. Working platforms are provided in some areas to facilitate servicing of pumps and valves. Removable thermal insulation is provided on welds and components, which require frequent access for examination or are located in high radiation areas. Welds are located to permit ultrasonic examination from at least one side, but where component geometry permits, access from both sides is provided.

Restrictions: For piping systems and portions of piping systems subject to volumetric examination, the following piping designs are generally not used:

- Valve to valve;
- Valve to reducer;
- Valve to tee;
- Elbow to elbow;
- Elbow to tee;
- Nozzle to elbow;

- Reducer to elbow;
- Tee to tee; 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.

Text sections that are bracketed and italicized with an asterisk following the brackets are designated as Tier 2. Prior NRC approval is required to change.

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 PSI and ISI programs state 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 ASME Section XI, 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-B.

6.6.3.2 Examination Methods

6.6.3.2.1 Visual Examination

Visual examination methods VT-1, VT-2 and VT-3 are used in accordance with ASME Section XI, IWA-2210. In addition, VT-2 examinations meet the requirements of ASME Section XI, 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 and liquid penetrant 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 (500 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 magnetic particle and liquid penetrant 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 150 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 500 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 ISI. 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 ISI. 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 ISI intervals for Class 2 systems conform to Inspection Program B as described in ASME Section XI, IWC-2412. Except where deferral is permitted by ASME Section XI, Table IWC-2500-1, the percentages of examinations completed within each period of the interval correspond to ASME Section XI, 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 ISI intervals for Class 3 systems conform to Inspection Program B as described in ASME Section XI, IWD-2412. Except where deferral is permitted by ASME Section XI, Table IWD-2500-1, the percentages of examinations completed within each period of the interval corresponds to ASME Section XI, 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-3000. Examination results are evaluated in accordance with ASME Section XI, IWC-3000 for Class 2 components, with repairs based on the requirements of ASME Section XI, 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 ASME Section XI, IWA-4000.

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

6.6.6 System Pressure Tests

The requirements for system leakage and hydrostatic pressure tests are described in this Section. System leakage and hydrostatic testing are required in accordance with ASME Code Section XI. Regardless of which test method is chosen, system leakage and hydrostatic pressure tests meet all applicable requirements of ASME Code Section XI: IWA-5000 and IWC-5000 for Class 2 components; and IWA-5000 and IWD-5000 for Class 3 components.

6.6.6.1 System Leakage Test

As required by ASME 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-5221 on Class 3 systems. The test includes all Class 2 or 3 pressure retaining components and piping within the boundaries defined by IWC-5222 and IWD-5240. 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-5222. 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 ASME Section XI, IWA-4540. The test includes all Class 2 or 3 pressure retaining components and piping within the boundaries defined by ASME Section XI, IWC-5222 and IWD-5240 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, Subsections 6.6.3.2, each inspection interval as defined within 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.

Maintenance and Inspection Program

Equipment requiring special maintenance and inspections shall also be part of the augmented inservice inspection program. For example, equipment important to turbine overspeed protection and referred to in COL Information item 10.2-1-A, should be included.

6.6.8 Code Exemptions

As provided in ASME BPVC 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

ASME Section XI requirements can be modified by invoking approved ASME Section XI Code Cases. Approved Code Cases for inservice inspection are listed in RG 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 are 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 BPVC Section XI edition and addenda for this program description are as specified in Table 1.9-22. The COL Holder will define the applicable edition 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 is responsible for providing a full description of the PSI/ISI programs and augmented inspection programs for Class 2 and 3 components and piping by supplementing, as necessary, the information in Section 6.6. The augmented inservice inspection program shall also cover erosion-corrosion and equipment requiring special maintenance and inspections as described in 6.6.7. The COL Applicant will also provide milestones for program implementation (Section 6.6).

6.6-2-A PSI/ISI NDE Accessibility Plan Description

The COL Applicant is responsible for developing a plan and providing a full description of its use during construction, PSI, ISI, and for design activities for components that are not included in the referenced certified design, to preserve accessibility to piping systems, to enable NDE of ASME Code Class 2 austenitic and DM welds during ISI (Section 6.6).

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 (Reference 6A.1-1). Since the approval, there have been some configuration changes in the ESBWR, primarily the location of the GDCS pools in the DW space rather than being a part of the WW volume. None of these invalidate the Phenomena Identification and Ranking Table (PIRT), 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.

The volumes in various regions inside the reactor pressure vessel are calculated from the design drawings. The heat structures for the reactor internal components (such as steam dryer, chimney partitions, shroud, top guide, fuel support, core plate, and feedwater spargers) are simulated by appropriate lumped heat masses or double-sided heat slabs in the TRACG model. The vessel bottom head, top head and vertical cylindrical wall are simulated by appropriate lumped heat masses. The volumes and heat masses of steam separators, guide tubes, fuel and fuel assemblies are modeled as separate TRACG components and are based on drawings.

The volumes of other piping and systems that are external to the vessel (such as main steam lines, feedwater lines, IC system, PCCS system, and GDCS lines) are also calculated from drawings. However, the heat masses of these structures are not simulated in the DCD TRACG model.

The same DCD TRACG model is used for analyses reported in both Sections 6.2 and 6.3.

For ECCS/LOCA analyses, the focus or figure of merit is the minimum chimney static head during the first 2000 seconds following the LOCA initiation. The vessel volumes and heat masses are two of the few key parameters that would affect the short-term water inventory, and they are simulated in the TRACG model in detail.

For Containment/LOCA analyses, the focus or figure of merit is the peak drywell pressure. The limiting case is a Main Steam Line Break bounding case (Table 6.2-5). Again, the vessel internal heat masses are simulated in detail to calculate the short-term stored energy and the short-term peak drywell pressure.

The heat structures of piping and systems external to the vessel are not simulated in the DCD TRACG model. Results of parametric study (NEDC-33083P-A, Response to RAI 298) show that the heat slab in the drywell has a very small impact on the containment response. An increase of 25% vertical DW heat slab area results in a change of 0.04 psi in the long-term containment pressure.

For a specified set of power and dome pressure, the TRACG model is performed with a steady-state initialization run to assure appropriate initial conditions (such as pressures, temperatures, void fractions, flows, etc.) prior to the LOCA initiation. The reactor dome temperature corresponds to the saturation temperature at the specified dome pressure. Table 6.2-6 includes these temperatures.

6A.1 References

6A.1-1 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.

Table 6A-1

Evaluation of TRACG Application Procedure

		Implementation in NEDC-33083P-A(1) (Reference 6A.1-1) [Sections, Tables, and Figures referenced within column refer to Reference 6A.1-1]	Implementation in ESBWR Design Certification(2)
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 TRACG VSSL component with eleven levels and two rings. The WW is modeled with the VSSL component with two levels and two rings. The Lower Drywell (LDW) is modeled with a 1-D 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 twelve levels and two rings. The WW is modeled with the VSSL component with three levels and two rings. The Lower Drywell is modeled as part of VSSL component with two levels and six rings. The new nodalization results in a quicker and more complete clearing of noncondensable gas from the LDW.
2	Heat Slabs	Only the vertical heat structures representing the inner and outer WW walls are modeled for the containment. The area associated with the horizontal diaphragm floor between the WW and DW is lumped in with the inner WW wall.	Same, see Figure 6A-1.
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.

Table 6A-1

Evaluation of TRACG Application Procedure

		Implementation in NEDC-33083P-A(1) (Reference 6A.1-1) [Sections, Tables, and Figures referenced within column refer to Reference 6A.1-1]	Implementation in ESBWR Design Certification(2)
Bounding Models			
4	Suppression Pool Stratification: Region above highest source of mass and energy to pool is assumed to be stratified.	Subsection 3.3.1.1.1	Same(3)
5	WW gas space: Top level assumed to be stratified due to Vacuum Breaker (VB) leakage.	Subsection 3.3.1.1.2	Same(3)
6	Noncondensable gases transport to WW: All noncondensable moved to WW; rate of transfer varied to maximize containment pressure.	Subsection 3.3.1.1.3 Hideout volume combined with LDW. Rate of release varied. Results insensitive to release rate.	With the current nodalization, noncondensable 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 DW to purge residual noncondensable in this space.
7	Location of MSL break elevation – vary to obtain highest containment pressure.	Subsection 3.3.1.1.3	Same. Break location in the upper portion of the UDW is slightly more limiting.

Table 6A-1

Evaluation of TRACG Application Procedure

		Implementation in NEDC-33083P-A(1) (Reference 6A.1-1) [Sections, Tables, and Figures referenced within column refer to Reference 6A.1-1]	Implementation in ESBWR Design Certification(2)
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) (Decay Heat. + 2σ)
10	Suppression Pool surface heat transfer, lower bound.	Table 3.4-1	Same. (Table 6.2-8) (PIRT07 = 1.0)
11	PCCS inlet loss: +2σ	Table 3.4-1	Same. (Table 6.2-8) (1585 m^{-4} (13.68 ft^{-4}))
12	PCCS heat transfer: -2σ	Table 3.4-1	Same. (Table 6.2-8) (PIRT78 = 0.902)
13	VB loss: +2σ	Table 3.4-1	(Table 6.2-8) (934 m^{-4} (8.06 ft^{-4}))
14	PCCS tube crud factor of 4.5e-5 m^2-K/W	Subsection 3.7.2	Same(3)
15	Noncondensable gas properties.	Air properties are used.	Nitrogen properties are used.

Table 6A-1

Evaluation of TRACG Application Procedure

		Implementation in NEDC-33083P-A(1) (Reference 6A.1-1) [Sections, Tables, and Figures referenced within column refer to Reference 6A.1-1]	**Implementation in ESBWR Design Certification(2)**
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 = (Noncondensable gas degradation factor) * (KSPw for Re_l <1000 smoothed into 0.55*BURM for Re_l > 2000)	Same(3)
17	Other parameters shown to have low sensitivity: nominal values.	Table 3.4-1	Same(3)
18	Radiolytic gas generation.	Consistent with SRP 6.2.5 and RG 1.7.	Same(3)
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 (7 kPa) in peak pressure.

Table 6A-1

Evaluation of TRACG Application Procedure

		Implementation in NEDC-33083P-A(1) (Reference 6A.1-1) [Sections, Tables, and Figures referenced within column refer to Reference 6A.1-1]	**Implementation in ESBWR Design Certification(2)**
Plant Parameter Inputs			
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	PCCS pool level: NWL	Table 3.5-1	Same (Table 6.2-6)
24	PCCS pool temperature: 316.5 K (569.7ºR)	Table 3.5-1	Same (Table 6.2-6)
25	DW pressure: 110.3 kPa (16.00 psia)	Table 3.5-1	Same (Table 6.2-6)
26	DW temperature: 319.3 K (574.4ºR)	Table 3.5-1	Same (Table 6.2-6)
27	WW pressure: 110.3 kPa (16.00 psia)	Table 3.5-1	Same (Table 6.2-6)
28	WW temperature: 316.5 K (569.7ºR)	Table 3.5-1	Same (Table 6.2-6)
29	Suppression Pool temperature: 316.5 K (569.7ºR)	Table 3.5-1	Same (Table 6.2-6)
30	GDCS pool temperature:	Table 3.5-1 (316.5 K (110ºF))	319.3 K (115ºF), (Configuration changed, GDCS connected to DW) (Table 6.2-6)

Table 6A-1

Evaluation of TRACG Application Procedure

		Implementation in NEDC-33083P-A[(1)] (Reference 6A.1-1) [Sections, Tables, and Figures referenced within column refer to Reference 6A.1-1]	Implementation in ESBWR Design Certification[(2)]
31	Suppression Pools level: NWL+0.05 m (2 in)	Table 3.5-1 (5.50 m (18.0 ft))	Same (5.50 m (18.1 ft)) (Table 6.2-6)
32	GDCS pool level:	Table 3.5-1 (NWL+0.05 m (2 in) = 6.75 m (22.1 ft), to minimize the draindown volume and the effective WW volume)	NWL 6.60 m (21.7 ft) (Configuration changed, GDCS connected to DW).
33	DW relative humidity: 20%	Table 3.5-1	Same[(3)]
34	RPV initial pressure: 7.274 Mpa (1055 psia)	Table 3.5-1	Same[(3)]
35	RPV initial water level: NWL+0.3 m (1 ft)	Table 3.5-1	Same[(3)]

(1) Description given in each row of column pertains to Reference 6A.1-1.
(2) Description given in each row of column pertains to ESBWR DCD Tier 2.
(3) “Same means ESBWR DCD Tier 2 is the same as Reference 6A.1-1 for the feature described in first two columns.

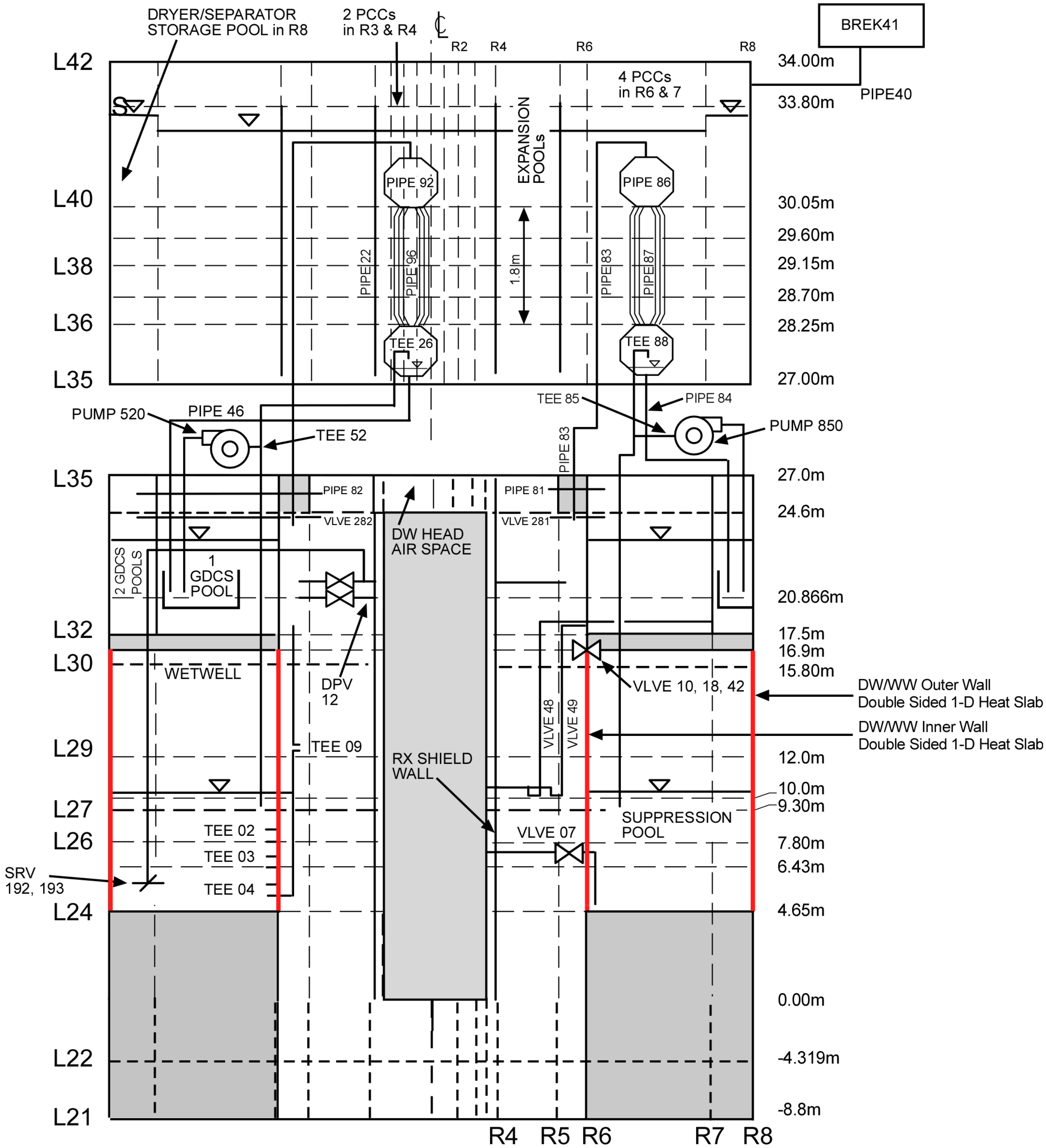


Figure 6A-1. TRACG Nodalization showing Containment Heat Slabs

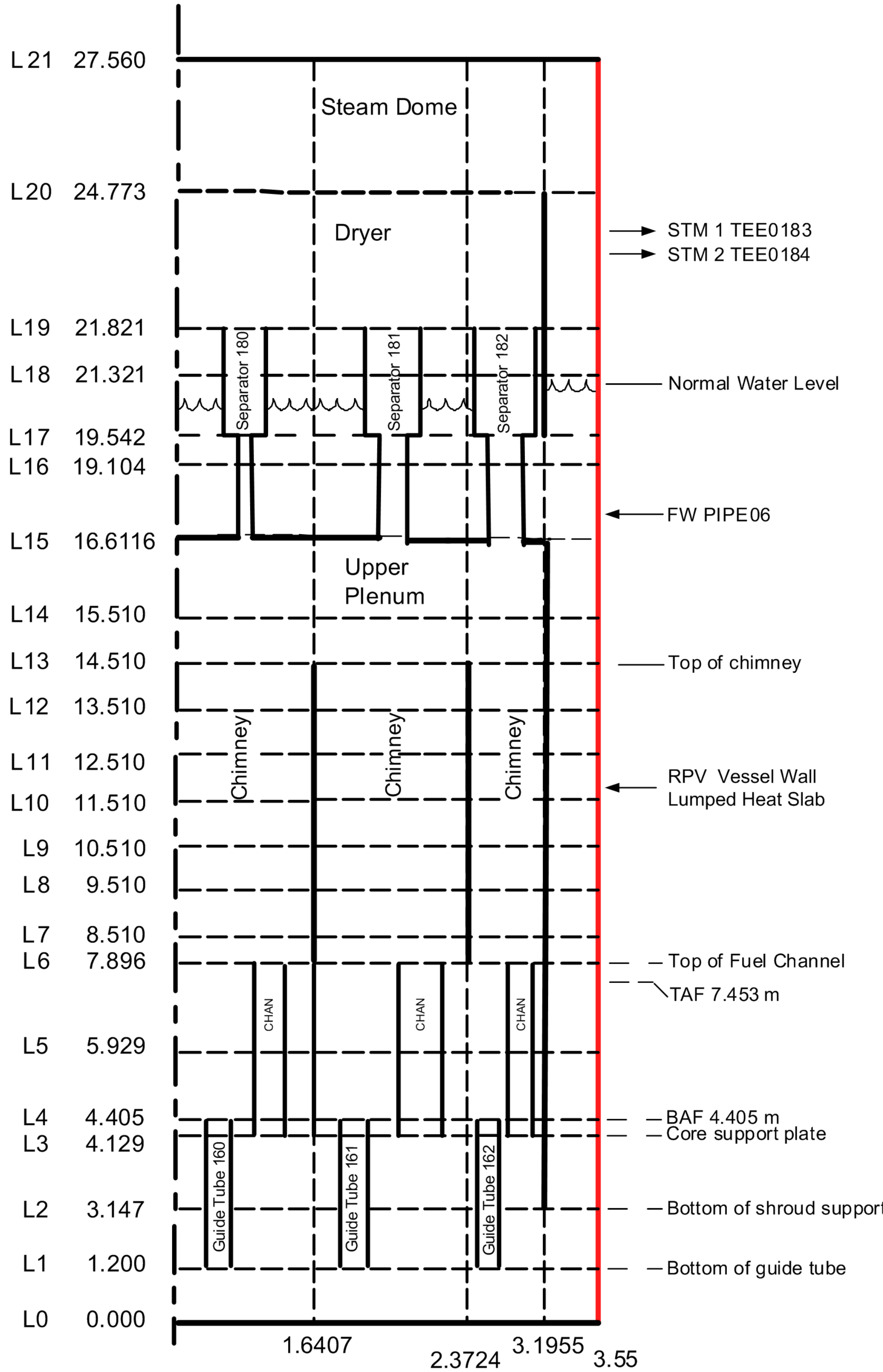


Figure 6A-2. RPV Nodalization showing Vessel Wall Heat Slab

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

The analysis for the ESBWR containment evaluation follows the application methodology outlined in Reference 6B.1-1. The TRACG nodalization approach in the licensing analysis is similar to that used in Reference 6B.1-1. 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 (SER) of Reference 6B.1-1. 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.1-1.

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.1-1. These tie-back calculations include a total of three cases used in Reference 6B.1-1, 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.1-1, and the impacts due to nodalization changes on the minimum chimney static head level (+0.1 to -0.16 m (+4 to -6.3 in)) and on the long-term DW pressure (< 2 kPa (0.3 psia)) are judged to be small by comparing to the margins.

At the time of this evaluation and in the process of running parametric cases for the feedwater line break (FWLB), it was discovered that the early peak in DW pressure for the FWLB was sensitive to the time step size. Also, three other input errors were identified. Sensitivity studies were performed and the impact of these input errors on the key output parameters was identified to be small. In addition, two model enhancements were implemented into the input deck.

During the development and evolution of the ESBWR design, changes have been made to the original nodalization presented in Reference 6B.1-1. In order to accurately analyze the ESBWR for reactor pressure vessel level and containment pressure responses after a LOCA, the TRACG model is maintained up to date, reflecting the final design. Due to this fact, the nodalization diagrams in Reference 6B.1-1 are intended to document previous models, and the changes are needed to maintain the final nodalization representative of the ESBWR. The final nodalization used in certification analyses is discussed in DCD Tier 2, Sections 6.2 and 6.3. Figures 6B-3 and 6B-4 represent a departure from the nodalization presented in Reference 6B.1-1.

The following paragraphs discuss the results of these tie-back calculations, including the effect on the noncondensable gases holdup, mixing and stratification. Also included in the following discussions are the descriptions and impact of the input errors and two model enhancements.

Combined Nodalization

In the TRACG Application Report (Reference 6B.1-1), two nodalizations are used for the calculations, Figure 6B-1 (from Figure 2.7-1 in Reference 6B.1-1) for the ECCS analyses (short-term) and Figure 6B-2 (from Figure 3.7-1 in Reference 6B.1-1) for the Containment (long-term) analyses. In the SER for Reference 6B.1-1, 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.1-1). (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 DW. And in the second step, component TEE35 is replaced with additional vessel levels and cells to model the LDW.

Model COMB-5 (with TEE35 to model the LDW)

Figure 6B-3 shows the combined nodalization in the first step (Model COMB-5, with TEE35 as the LDW). 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 DW, suppression pool, WW, 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 IC/PCCS 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 LDW)

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 DW. 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 LDW. 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 COMB-6 are used to model the DW, WW, 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 WW. The main purpose of this additional axial layer is to simulate the thin layer near the top of the WW.

The noncondensable gases trapped inside this thin layer (between the I-beams) 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 IC/PCCS 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 TEE63) 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 Subsection 3.7.2 in Reference 6B.1-1, 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 passive containment cooling condensation powers between these two Cases. For “MSLB-NL2_V40,” because of the exaggerated downcomer level response, the condensation power shows spiky drops at the time of the spiky increases in the downcomer 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 nine 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 passive containment cooling 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 passive containment cooling condensation power is greater than the decay heat after nine to ten hours. From that time on, the added energy to the suppression pool water due the movement of noncondensable gases from the DW to the WW is not significant.

Figure 6B-8 compares the suppression pool water temperature at the surface. For both cases, the pool surface temperatures increase during the first nine hours. However, the pool surface temperature for Case “MSLB-CB5” is higher than that for Case “MSLB-NL2_V40” by about 4ºC (7.2ºF) at the end of nine hours (because the PCCS condensation power is lower for the Case “MSLB-CB5”).

Figure 6B-9 shows the DW partial noncondensable gas pressures for Case "MSLB-CB5," and Figure 6B-10 shows the DW partial noncondensable gas pressures for Case "MSLB-NL2_V40." During the first 18 hours, Case "MSLB-CB5" retains more noncondensable gases in the DW than that for Case "MSLB-NL2_V40." It takes 54 hours to purge all DW noncondensable gases for Case "MSLB-CB5," while it takes 48 hours for Case "MSLB-NL2_V40." The long-term noncondensable gas distribution depends on the noncondensable 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 nine hours, the DW pressure for Case "MSLB-NL2_V40" is higher than that for Case "MSLB-CB5," due to lesser noncondensable gases remaining in the DW. After all the noncondensable gases have been purged into the WW (48 hours for Case "MLSB-NL2_V40," 54 hours 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 9.0 kPa (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 DW partial noncondensable gas pressures for Case "MSLB-CB5," and Figure 6B-12 shows the DW partial noncondensable gas pressures for Case "MSLB-CB6." During the first 18 hours, Case "MSLB-CB5" retains more noncondensable gases in the DW than that for Case "MSLB-CB6." It takes 54 hours to purge all DW noncondensable gases for Case "MSLB-CB5," while it takes 21 hours for Case "MSLB-CB6." The DW pressure reaches the maximum value when all the noncondensable gases have been purged in the WW (Figure 6B-14). The difference in the timing for purging all noncondensable 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 noncondensable gases have been purged into the WW (21 hours 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 noncondensable 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 noncondensable gases have been purged into the WW (48 hours for Case "MSLB-NL2_V40," 54 hours for Case "MSLB-CB5," and 21 hours 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 2 kPa (0.3 psia) lower than that for Case "MSLB-CB5," when all noncondensable 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 0.5ºC (0.9ºF) lower than that for Case "MSLB-CB5."

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

Time Step Size Sensitivity and Input Error Corrections

In the process of running parametric cases for the FWLB, it was discovered that the early peak in DW pressure for the FWLB was sensitive to the time step size. In addition, three input errors were identified. The following paragraphs describe these subjects and the impacts on the peak DW pressure.

Time Step Size Sensitivity of FWL Case

The nominal FWL LOCA case (DCD Tier 2 Revision 1) was rerun using the PC version of TRACG and served as the base case (Case E0 in Table 6B-1) for the time step sensitivity study, and also for the study of input error corrections and impacts. Figure 6B-19 shows the drywell pressure response for the first 100 seconds. The peak DW pressure is 342 kPa at 78.16 seconds. A closer examination on the total flow to the main vent (Figure 6B-20) reveals that there is a sudden reduction in main vent flow at 71 seconds, and subsequently a reduction in the SRV flow (Figure 6B-21). These flow reductions cause the increase in DW pressure prior to the DPV opening.

A pressure spike occurring at that time in the suppression pool near the second horizontal vent exit causes the sudden reduction in main vent and SRV flows. This pressure disturbance is the result of numerical problem, commonly known as the Water Packing.' 'Water Packing' generally occurs when steam is condensing in the subcooled water in a confined volume. Usually, this numerical problem can be avoided by using smaller time step size during the time period when the 'Water Packing' problem is likely to occur.

A parametric case (E0a) was performed with smaller time step size. The time step size is reduced from 0.05 seconds in the base case (E0) to 0.025 seconds in the parametric case. Figures 6B-22 through 6B-24 show the results of this parametric case. The 'Water Packing' problem and the sudden reduction in flows have been avoided. The DW pressure increases following the DPV opening. The peak DW pressure is 306 kPa at 78.33 seconds. A time step size of 0.025 seconds is adopted for other follow-on FWLB cases.

Correction of the 'water packing' error for the FWLB, causes the peak pressure to be reduced below the MSLB peak pressure, making MSLB the limiting case.

Input Error Corrections

After the completion of the TRACG calculations for the DCD Tier 2 Revision 1, three errors were identified in the TRACG input decks. The following paragraphs describe these errors.

(1) Vacuum break flow area

The ESBWR design uses three vacuum breakers. Assuming one vacuum breaker is out-of-service for the LOCA analyses, there should be two vacuum breakers available for the LOCA transient. In the input deck, the vacuum breaker areas are controlled by the control system. The input for the valve opening area fraction is incorrectly inputted as the actual valve area (instead of the opening fraction) in the control input cards. The error causes the total effective vacuum breaker flow area to be about one vacuum breaker (instead of two). Parametric cases have been performed and the results show that the impact of this modeling error on the key parameters (such as minimum chimney level and the maximum DW pressure) is insignificant.

(2) SLC system flow input table

SLC system flow is modeled by a FILL component in the TRACG model. The flow rate is specified by a time versus FILL velocity table, using 34 pairs of time-velocity data with 10 seconds interval. The FILL velocity at 260 seconds (time after the SLC initiation) was inadvertently entered as 30.8 m/s instead of 26.0 m/s. This input error causes a minor blip in the SLC system flow.

(3) Axial power input for the non-existing part of the part-length rod

Each CHAN component uses four rod-groups to model the fuel rods. Group 2 models the Part-Length rods. Axial nodes 1 to 28 simulate the actual length of the fuel rod. Axial nodes 29 to 32 simulate the part of the rod that is not there (the non-existent part of the part-length rod). To model the non-existing part, the axial power (CPOW) for the last four axial nodes (29 to 32) is inputted as zero, and the gap conductance (HGAP) is inputted as 1.0. This modeling sets zero power to the non-existing part of the part-length rod and essentially no energy transferred from the non-existing part of the rod to the bundle fluid.

There are six CHAN components in the TRACG input decks. The CPOW entries for the non-existing part of the part-length rod in CHAN0112 and CHAN0212 were incorrectly inputted as non-zero values. The impact of this input error is as follow. The energy modeled in the last four nodes of Rod Group 2 is essentially kept inside the fuel pellets and no energy is transferred radially to the bundle fluid. This is because HGAP = 1.0 for these nodes. TRACG does not model the axial heat conduction in the fuel rod. So the energy in the last four nodes does not transfer axially downward to other nodes. This error causes abnormally high fuel pellet temperature for the last four nodes, which are the non-existent part of the part-length rod and do not participate in the transient.

Sensitivity Study on the Impact of Input Errors

Additional sensitivity cases were performed to evaluate the impact of these input errors, using the FWLB as the reference case. The E0a (time step size = 0.025 seconds) was modified to incorporate the error correction one at a time. Table 6B-1 summarizes the peak DW pressure for these cases. Comparison among Cases 2 to 6 shows that the impact of each error correction on the Peak DW pressure is less than 0.3 kPa, which is acceptably small when comparing to the margin to the containment design pressure.

Model Enhancement

The model enhancement is the WW gas stratification model and the improved nodalization in the DCD analyses (i.e., one additional axial level added at an elevation near the top of the WW).

The WW gas stratification model has been documented and discussed in Subsection 3.3.1.1.2 in Reference 6B.1-1. The improved nodalization has been documented as Item 20, in Table 6.2-6a, DCD Tier 2, Revision 4. The purpose of the wetwell gas stratification model is to trap the hot noncondensable gas and steam flowing into the top of the WW from the DW through the bypass leakage. The vapor additive friction loss coefficients on the top of the WW cells (Level 31) were changed to high values to simulate the gas stratification. The temperatures and pressures in the top cells remain high throughout the transient.

Another model enhancement is by adding a large negative loss at the top of the horizontal vent exit to reduce the high vent flow oscillations. This large negative loss coefficient was later removed from the input decks for both the ECCS / LOCA and Containment/LOCA analyses (for analyses performed after DCD Tier 2, Revision 3).

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 4ºC (7.2ºF) 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 9 kPa (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 DW noncondensable gases into the WW. However, the effect of purging time period on the peak, long-term DW pressure is small and is less than 2 kPa (0.3 psia) (Figure 6B-14). The relatively small effect is because the total PCCS condensation power is greater than the decay heat after approximately nine to ten hours. From that time on, the added energy to the suppression pool water due to the movement of noncondensable gases from the DW to the WW is not significant. The DW pressure reaches the maximum value when all the noncondensable gases have been purged in the WW (Figure 6B-14). The difference in the timing for purging all noncondensable gases and the subsequent VB 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 (+4 to –6.3 in)) and on the long-term DW pressure (< 2 kPa (0.3 psia)) are judged to be small by comparing to the margins. Furthermore, considering that the MSLB and FWLB containment responses for these events are very similar in nature during the blow down period, the GDCS period and the long-term PCCS cooling period results, as noted in Figures 6.2-13a1 through Figures 6.2-13c3 and Figures 6.2-14a1 through Figures 6.2-14c3, are also applicable to the FWLB and with minimum impact to the respective noncondensable gas migration.

A slightly more noticeable and conservative impact due to nodalization changes is the inclusion of two pipes in the TRACG nodalization. These two pipes simulate the connection between the GDCS air space and the upper DW. This nodalization change increases the DW pressure as a result of a more effective migration of noncondensable gases from the GDCS air space to the

DW, and eventually to the WW. The current TRACG nodalization of the ESBWR Containment includes these changes as seen in Figure 6.2-7. These changes result in a MSLB DW pressure increase of about 1% when compared to a single pipe and a pressure margin reduction from 19% to about 14% for the FWLB.

The difference in the effect on DW pressure between these two events is attributed to their corresponding scenarios. During the initial part of the FWLB, the lower DW accumulates more water, displacing the noncondensable gases and forcing them to migrate faster to the WW area than during the MSLB; as seen in Figure 6.2-13d4 and Figure 6.2-14d4. During this period some amount of noncondensable gases migrate and hide in the GDCS drained volume by a greater amount and at a faster rate than during the MSLB, due to earlier and greater inventory drainage from the GDCS pool to the RPV. At the same time this hideout is minimized by the carry over of noncondensable gases to the WW, thereby increasing the containment pressure, as seen in Figure 6.2-13d5, Figure 6.2-14d5, Table 6.2-7e and Table 6.2-7d.

These two main scenarios contribute to their different noncondensable gases migration pattern, such that the difference in noncondensable gases migration patterns is attributed mainly to different system event responses and not to nodalization changes.

Even though these two scenarios have a different noncondensable gas migration, the two-pipe nodalization stated above and the assumption that all DW noncondensable gases migrate to the WW airspace at 72 hours adds sufficient conservatism to their DW pressure. The addition of two pipes promotes migration of noncondensable gases from the DW to the WW resulting in a conservatively high DW pressure increase. Furthermore this migration enhancement is maximized when all the containment noncondensable gases, including those added to the containment from pneumatic supplies are assumed to relocate to the WW, see Appendix 6E.2.

Table 6B-2 provides a summary of differences from the ESBWR description given in Reference 6B.1-2 and the ESBWR design in the DCD.

6B.1 References

6B.1-1 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.

6B.1-2 GE Nuclear Energy, “TRACG Model Description,” NEDE-32176P, Revision 4, Class III, (Proprietary), January 2008; NEDO-32176, Revision 4, Class I (Non-proprietary), January 2008.

Table 6B-1

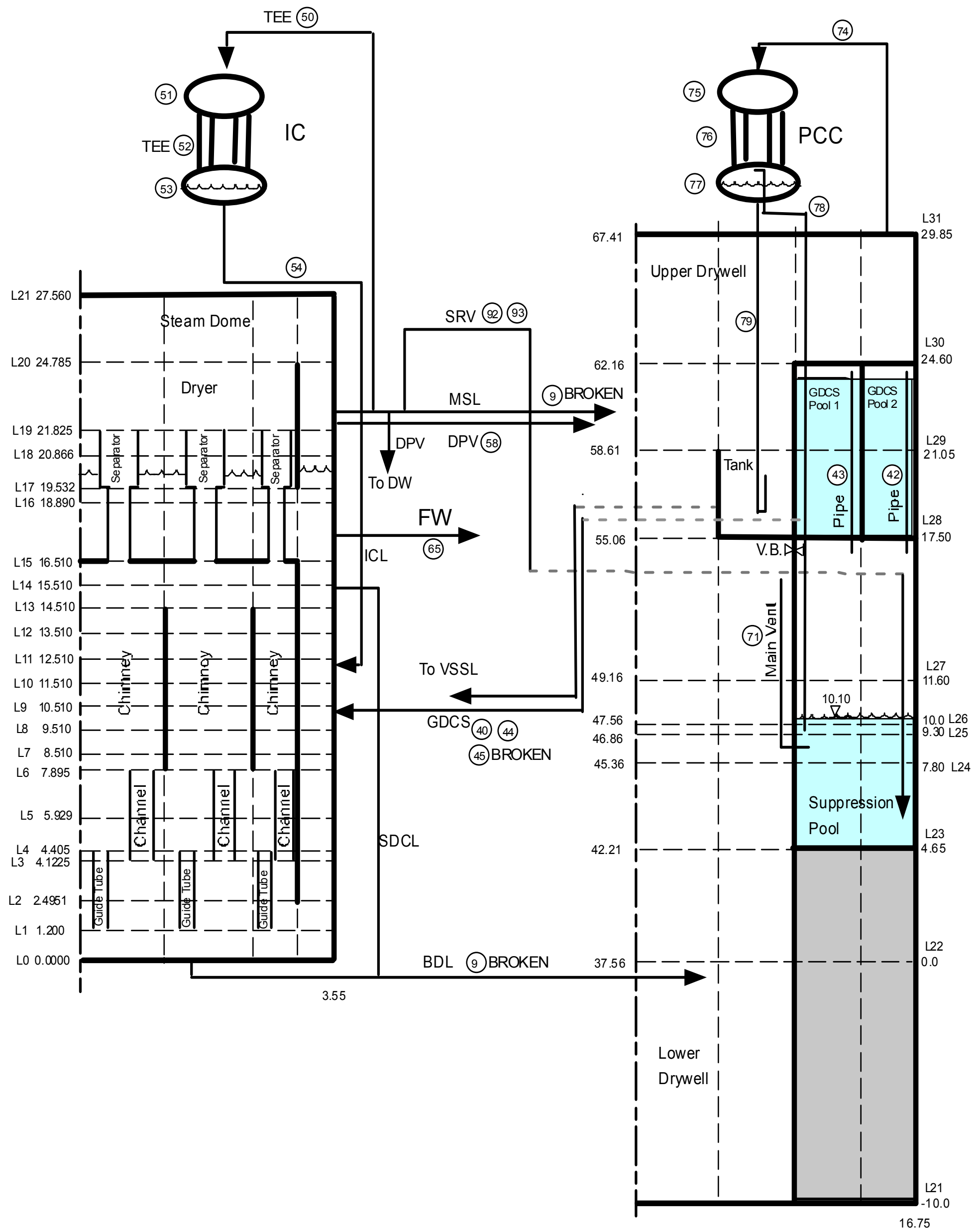
Summary of Peak DW Pressure for Error Correction Cases

Case #	Case Description	Peak DW Pressure, kPa (psia)	Time at Peak DW Pressure (sec)
(1)	E0 (Base Case)	342.0 (49.60)	78.16
(2)	E0a (0.025 seconds)	306.0 (44.38)	78.33
(3)	E1 (Vacuum Breaker)	306.3 (44.43)	78.00
(4)	E2 (SLC system)	306.0 (44.38)	78.33
(5)	E3 (Part-length rod power)	306.1 (44.40)	78.24
(6)	E123 (Combined E1, E2, E3)	306.2 (44.41)	77.80

Table 6B-2

Summary of Differences from Reference 6B.1-2

	Item	NEDE 32176P		DCD Tier 2		Notes
		Subsection	Detail	Subsection	Detail	
1	Containment Nodalization Figure	7.11	Figure 7-43	6.2	Figure 6.2-7	Most updated nodalization can be found in the DCD Tier 2.
2	Number of PCCS Heat Exchangers	7.11.5	4	6.2.2.1	6	PCCS vent fans are modeled in DCD Tier 2. Vent fans were not modeled in NEDE-32176P.
3	Number of ICS Heat Exchangers	7.11.5	6	Table 6.2-6a	4	None.
4	Number of GDCS Equalization lines connecting to the RPV downcomer.	7.11.7.8	3	Table 6.3-5a	4	Current GDCS and RPV design allows for four GDCS equalizing lines.
5	Number of Vacuum Breakers modeled as operating.	7.11.8	2	6.2.1.1.4	1	DCD Tier 2 assumes only one functioning Vacuum Breaker.



ESBWR TRACG LOCA MODEL

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

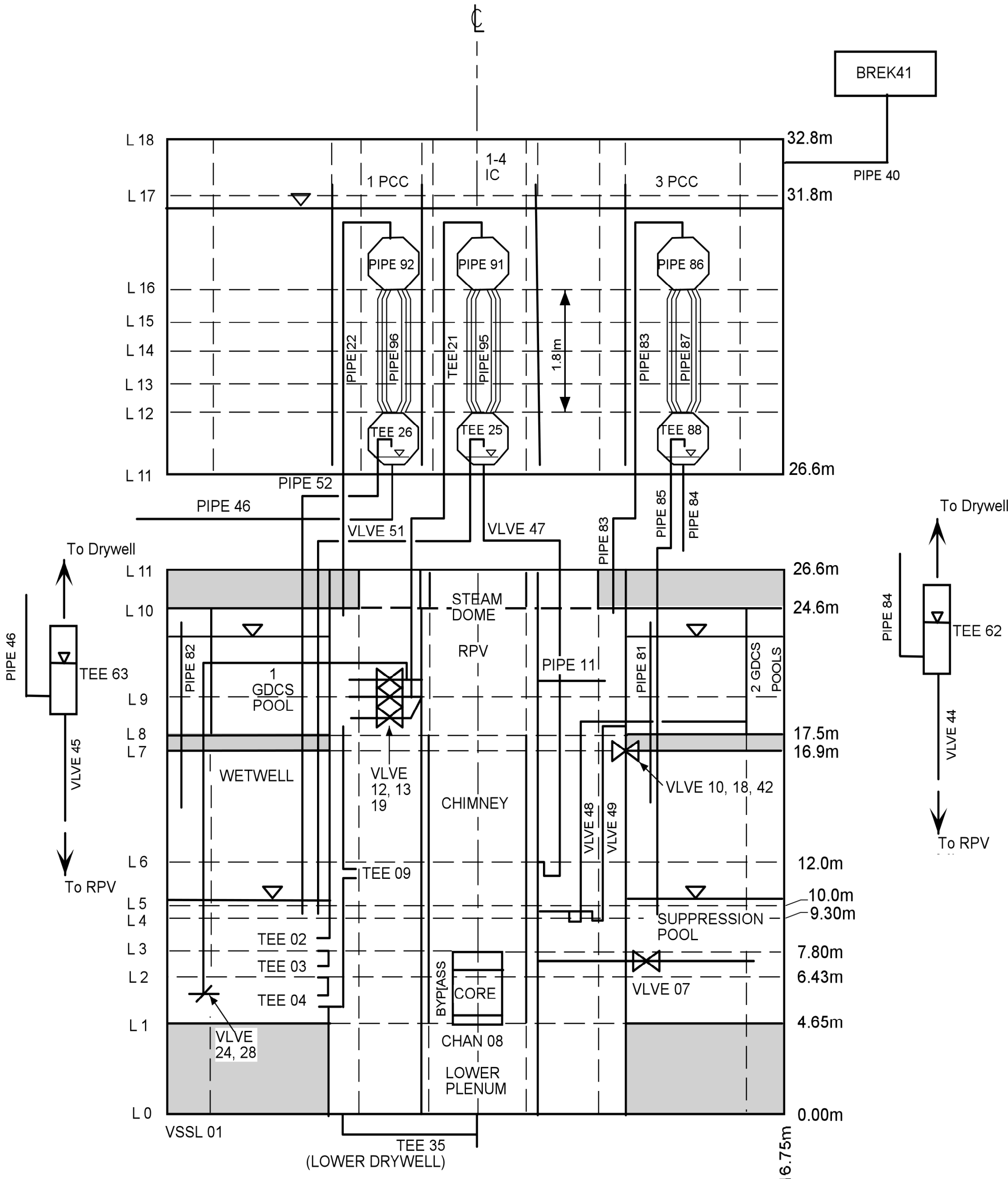


Figure 6B-2. TRACG Nodalization for ESBWR Containment Analysis (NEDC-33083P-A)

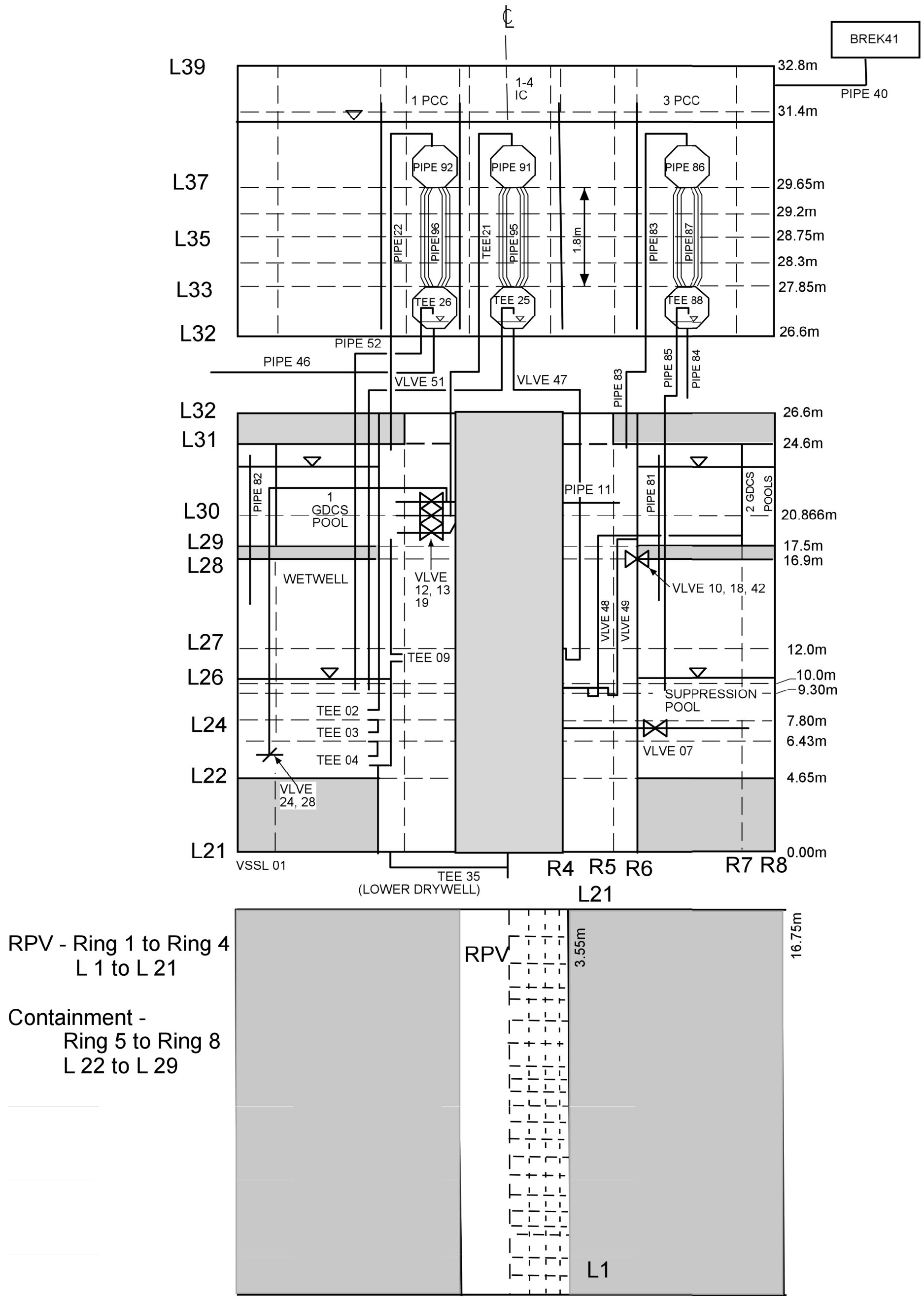


Figure 6B-3. TRACG Combined Nodalization (Model COMB-5, with TEE35 as Lower Drywell)

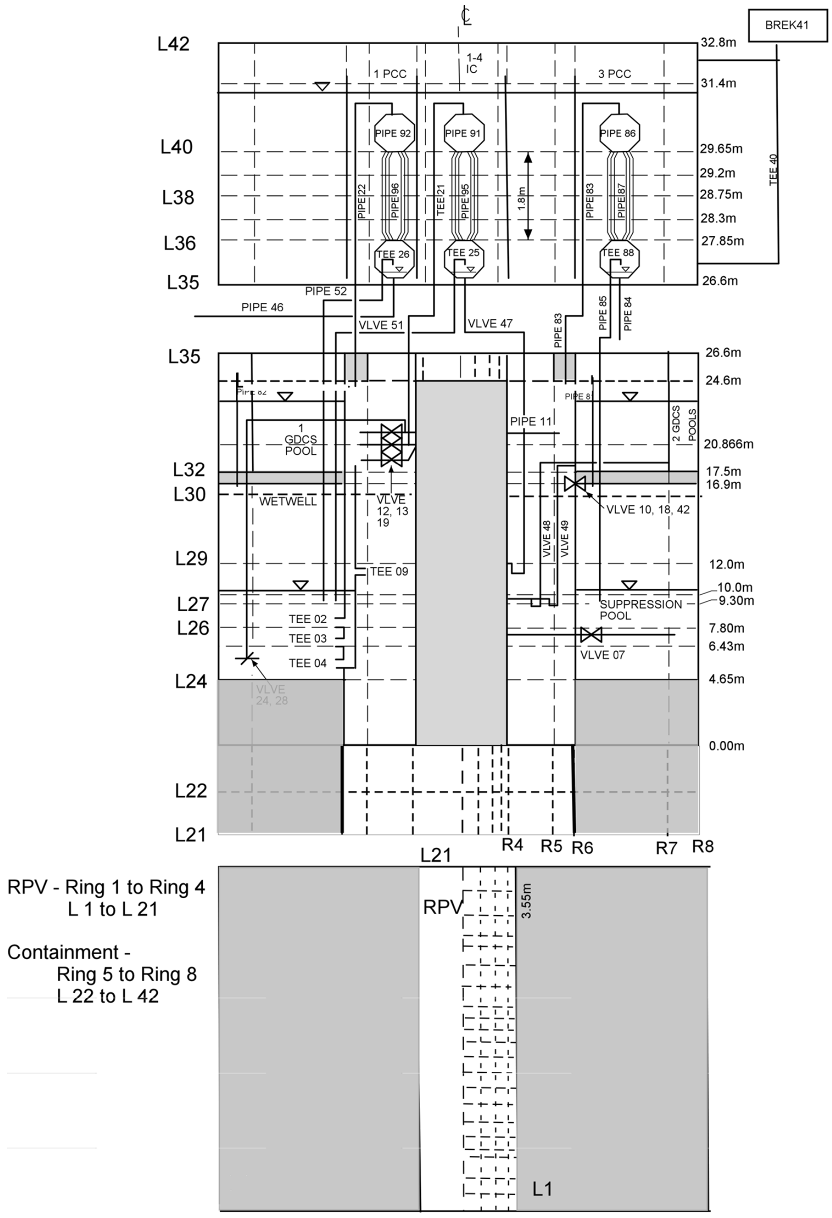


Figure 6B-4. TRACG Combined Nodalization
(Model COMB-6, without TEE35 as Lower Drywell)

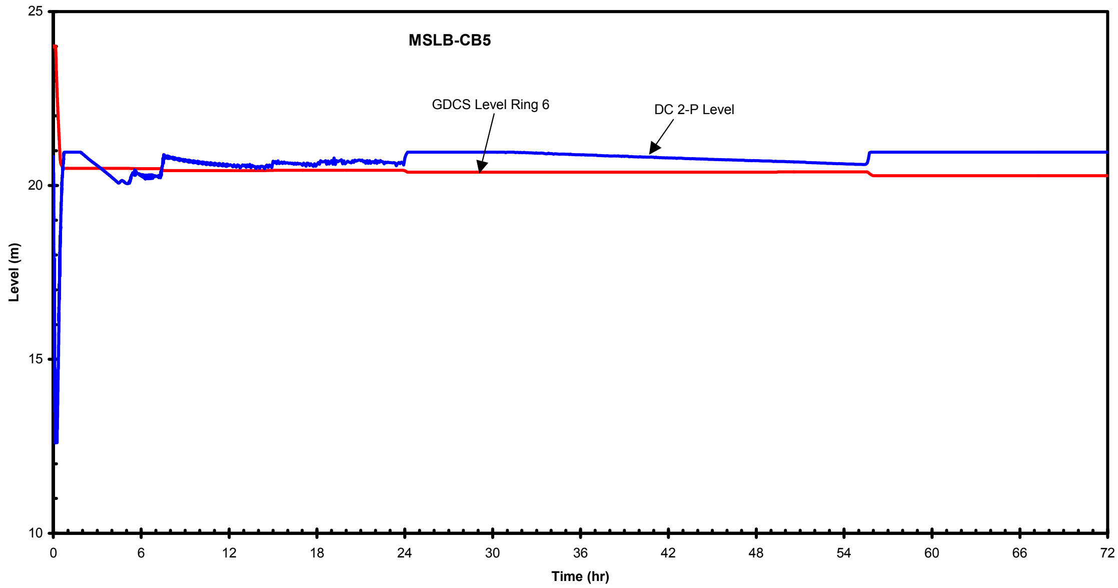


Figure 6B-5. MSLB-CB5 – GDCS and Downcomer Water Levels

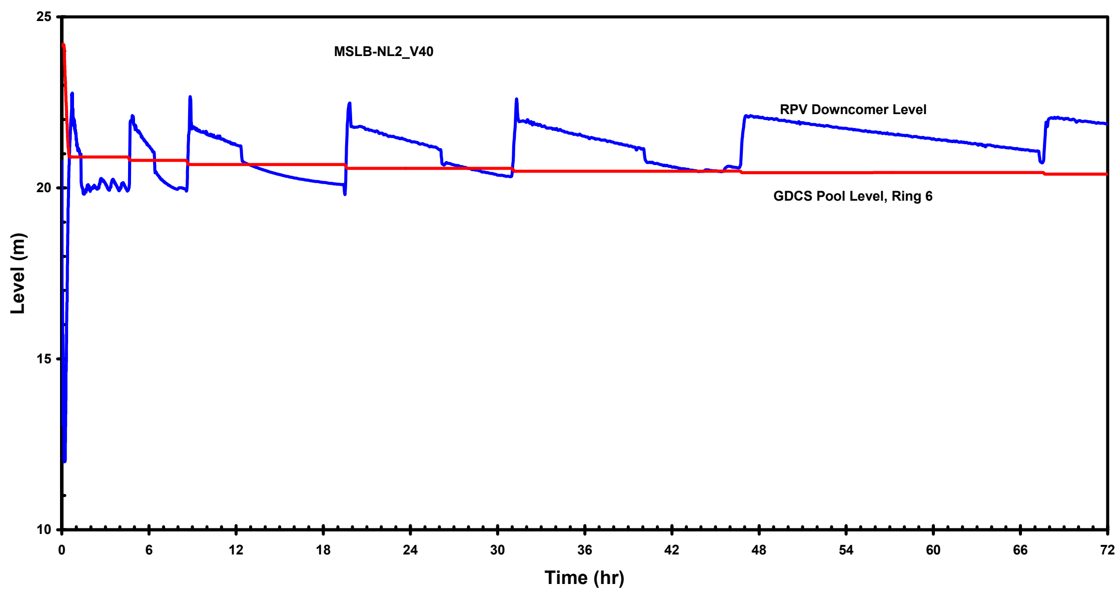
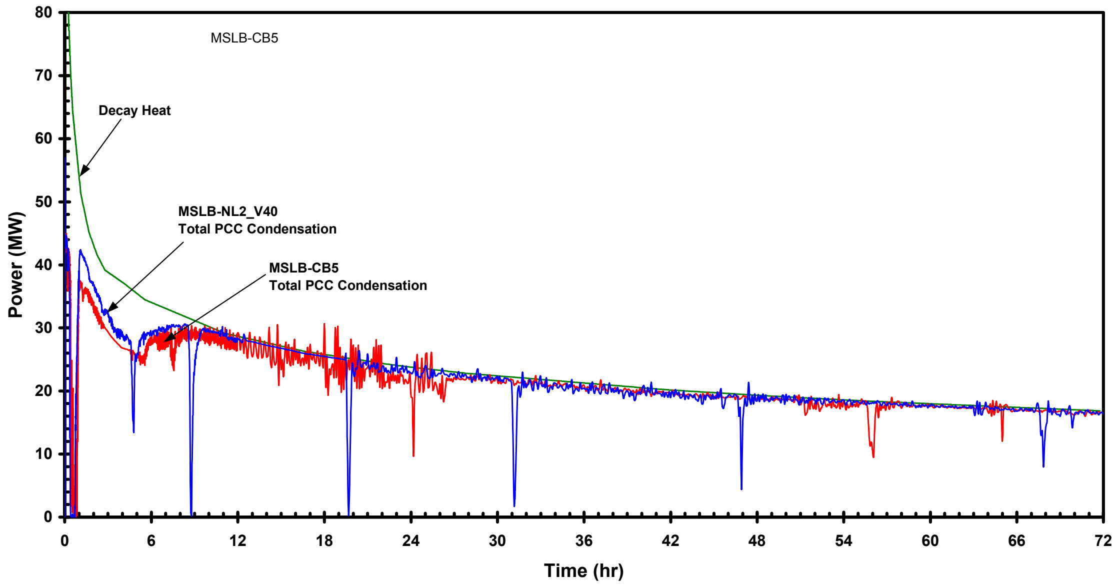


Figure 6B-6. MSLB-NL2_V40 – GDCS and Downcomer Water Levels



LEGEND: PCC = Passive Containment Cooling

**Figure 6B-7. MSLB-CB5 vs. MSLB-NL2_V40 –
Total Passive Containment Cooling Condensation Powers**

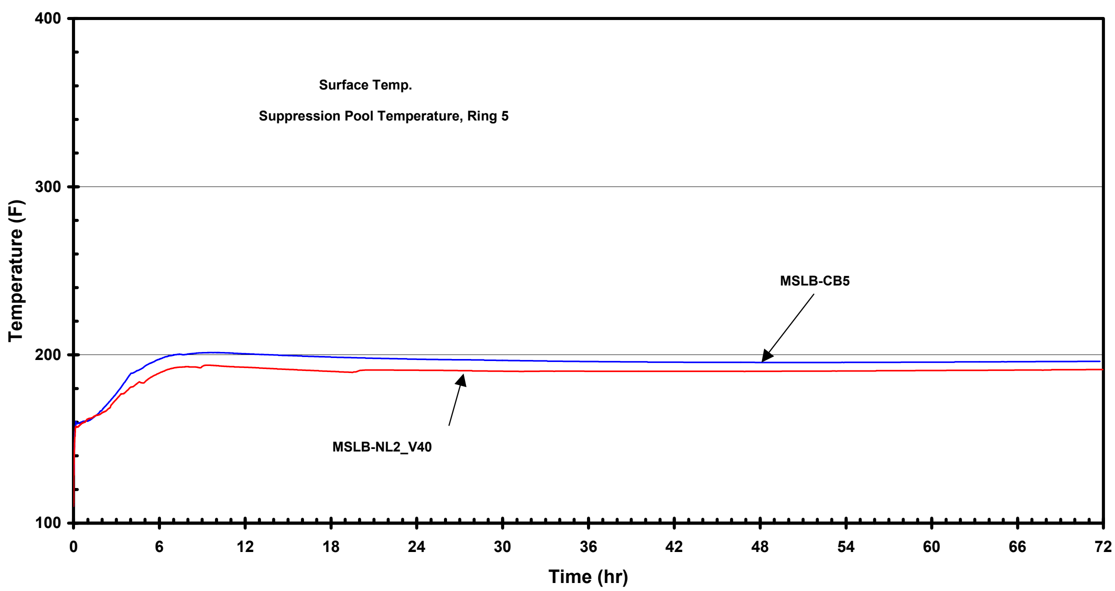


**Figure 6B-8. MSLB-CB5 vs. MSLB-NL2_V40 –
Suppression Pool Surface Temperatures**

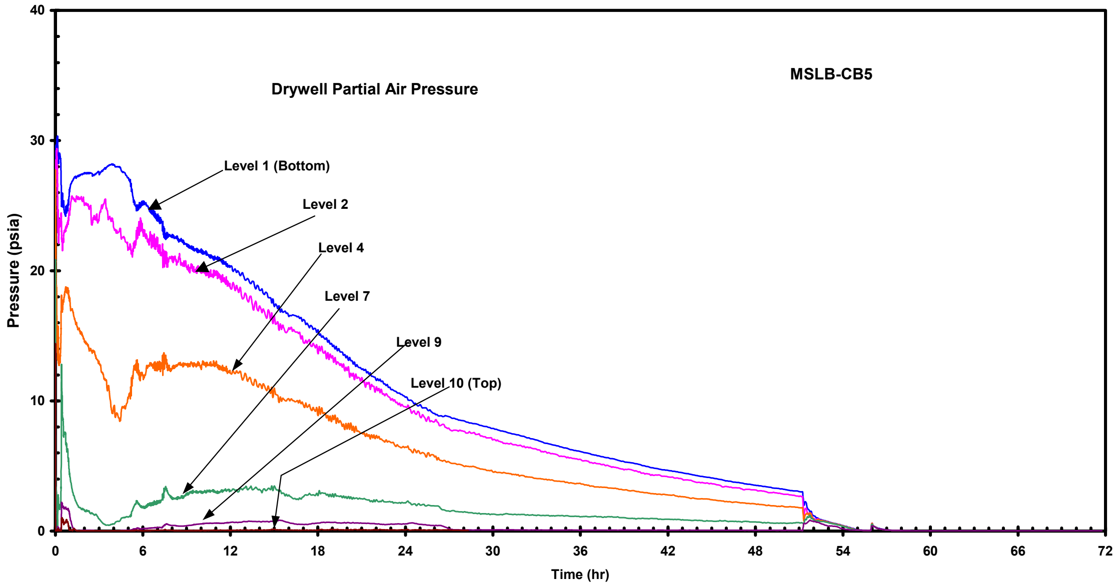


Figure 6B-9. MSLB-CB5 – Drywell Partial Noncondensable Gas Pressures

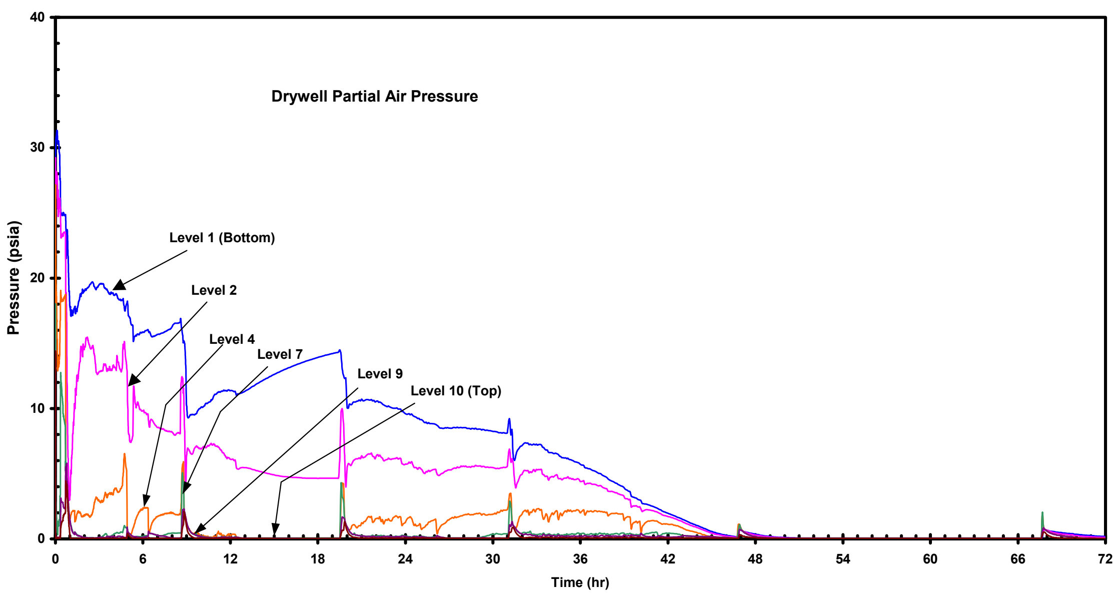


Figure 6B-10. MSLB-NL2_V40 – Drywell Partial Noncondensable Gas Pressures

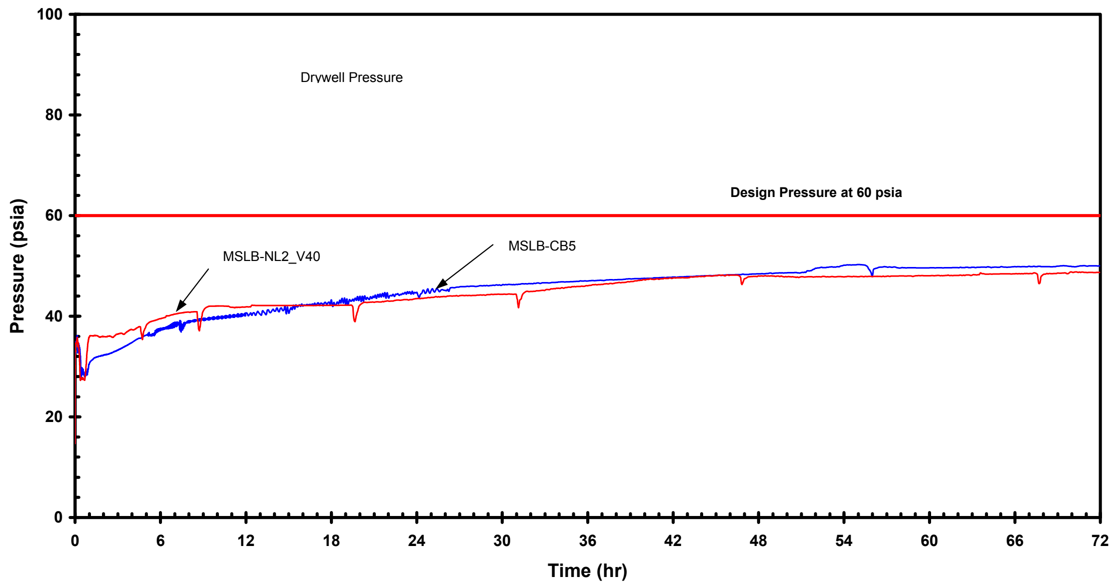


Figure 6B-11. MSLB-CB5 vs MSLB-NL2_V40 – Drywell Pressures

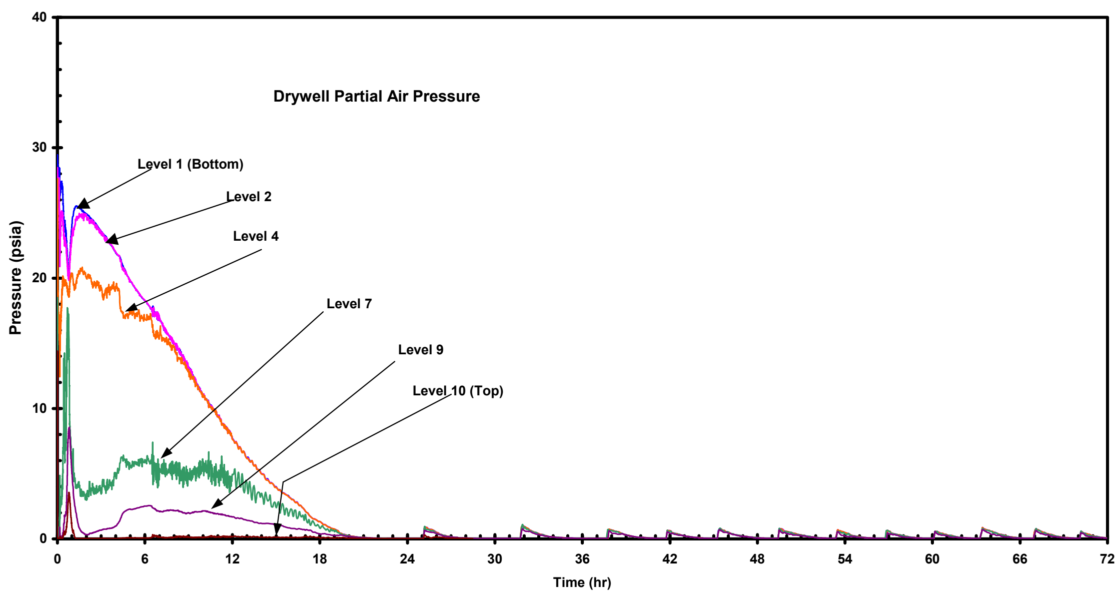


Figure 6B-12. MSLB-CB6 – Drywell Partial Noncondensable Gas Pressures

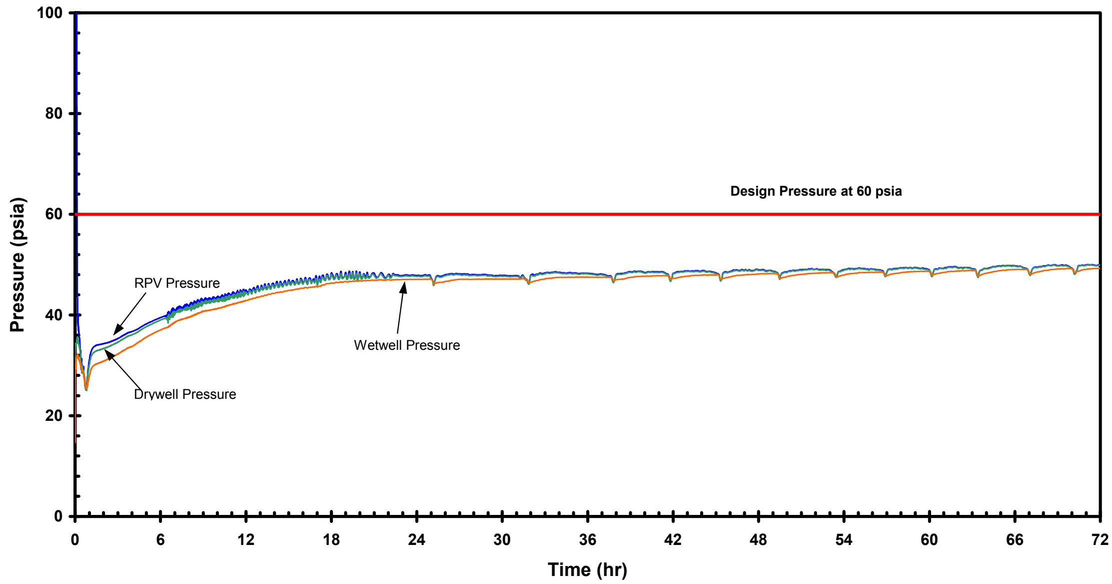


Figure 6B-13. MSLB-CB6 – RPV, Drywell and Wetwell Pressures

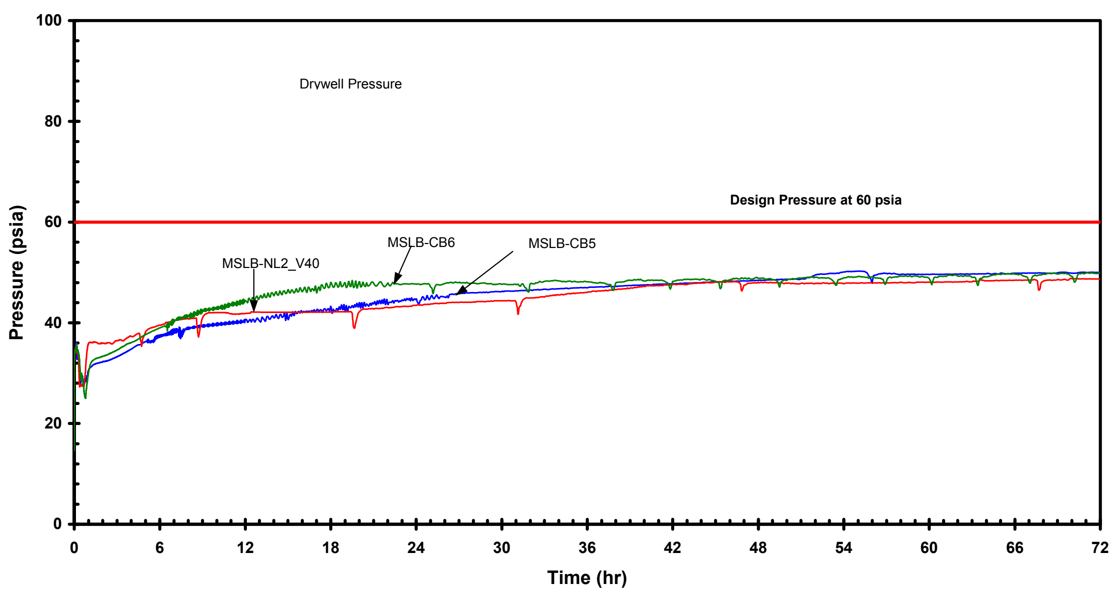


Figure 6B-14. Drywell Pressure Comparison – MSLB-CB6, MSLB-CB5 and MSLB-NL2_V40

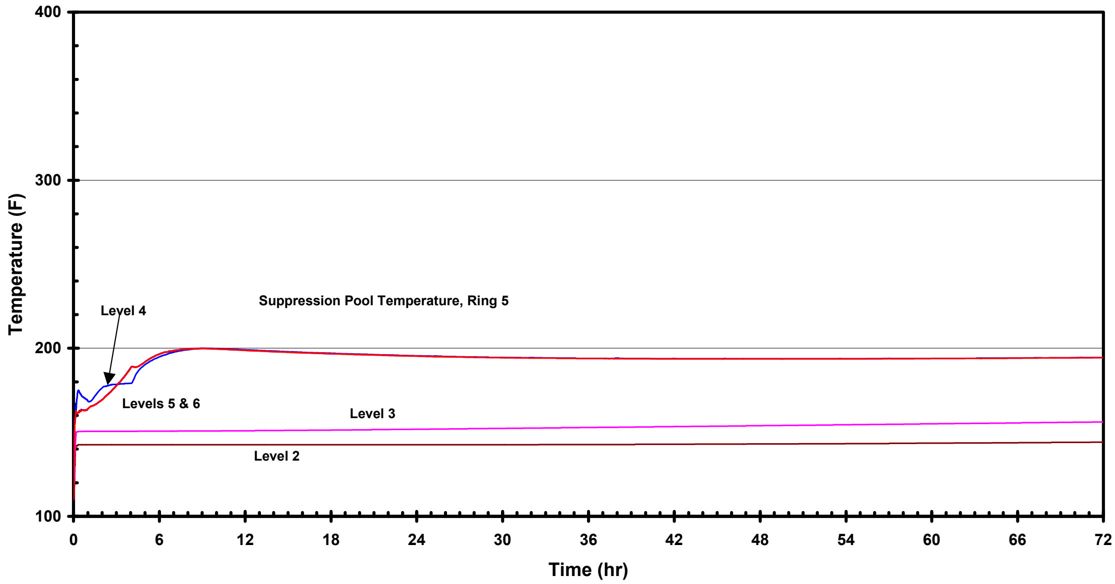


Figure 6B-15. MSLB-CB6 – Suppression Pool Temperatures

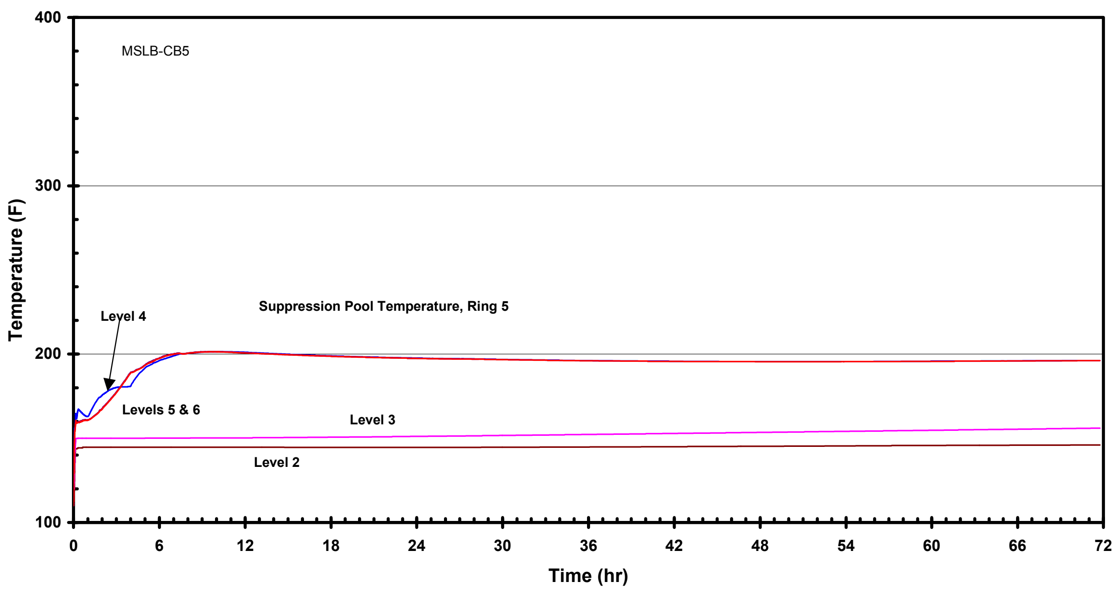
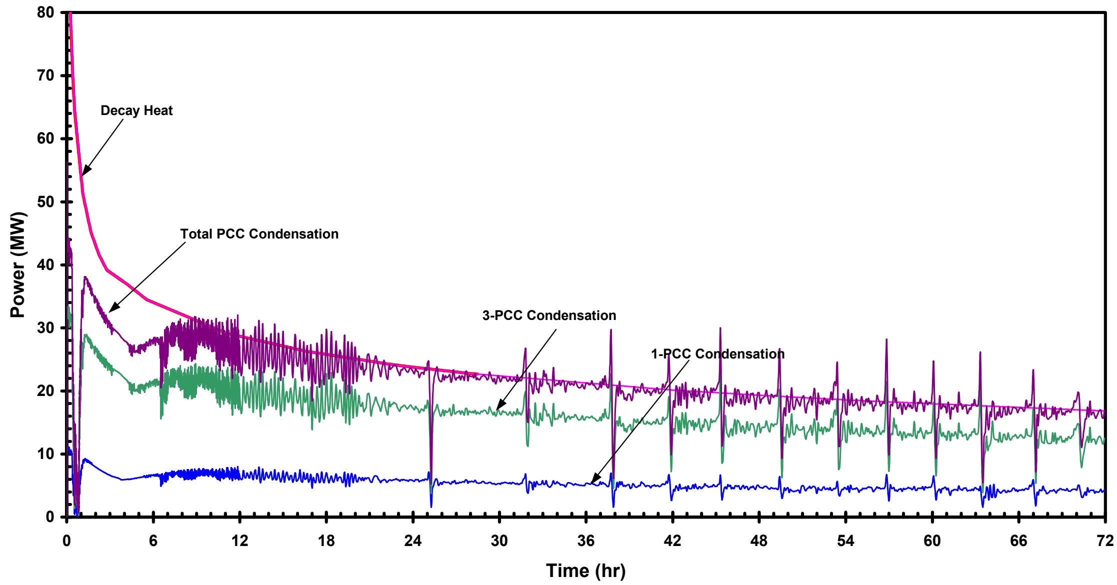
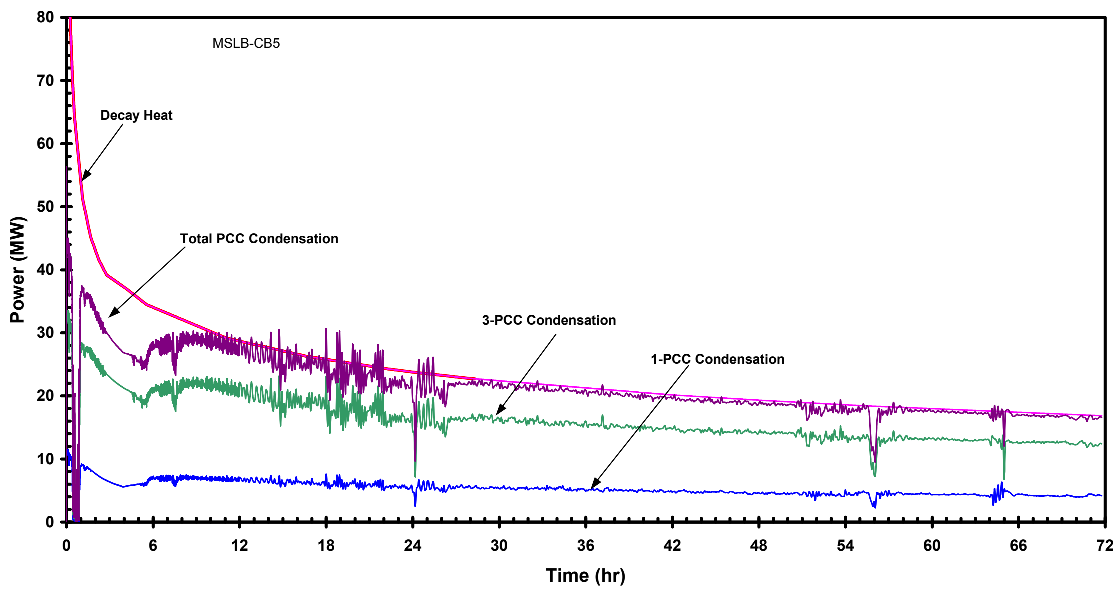


Figure 6B-16. MSLB-CB5 – Suppression Pool Temperatures



LEGEND: PCC = Passive Containment Cooling

Figure 6B-17. MSLB-CB6 – Passive Containment Cooling Condensation Power



LEGEND: PCC = Passive Containment Cooling

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

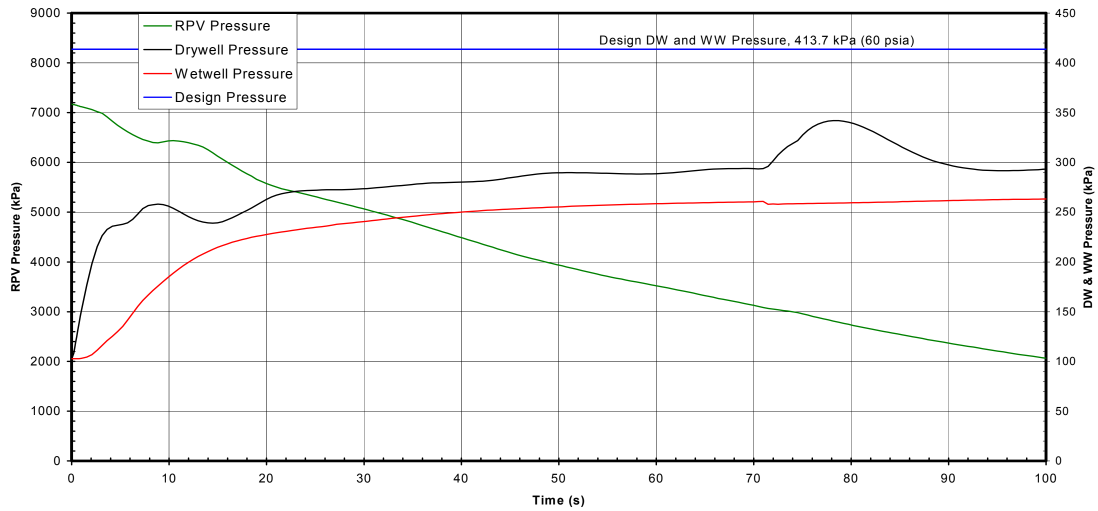


Figure 6B-19. E0 (Base Case) – Drywell Pressure Response

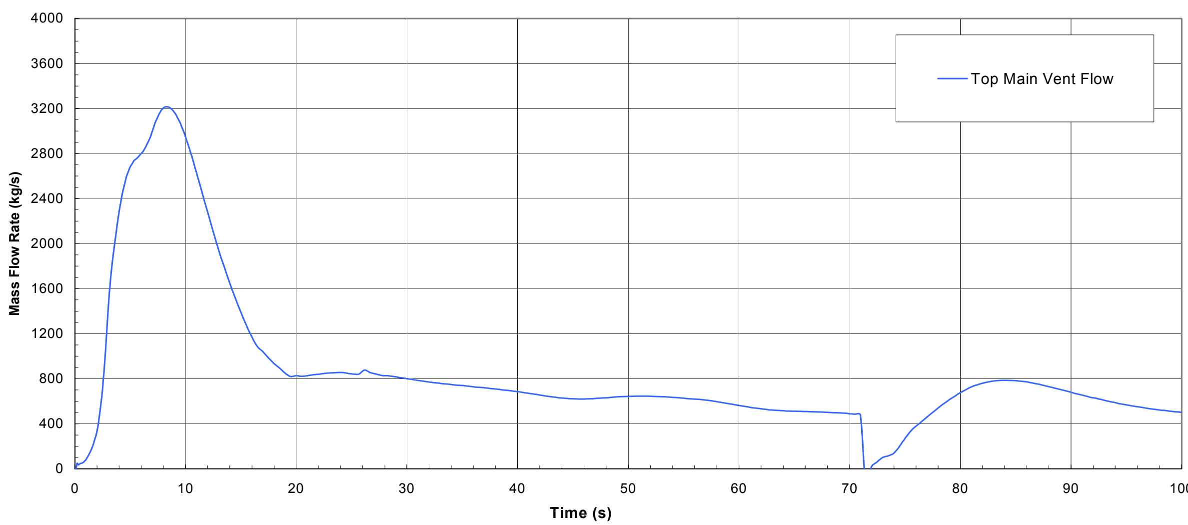


Figure 6B-20. E0 (Base Case) – Top Main Vent Flow

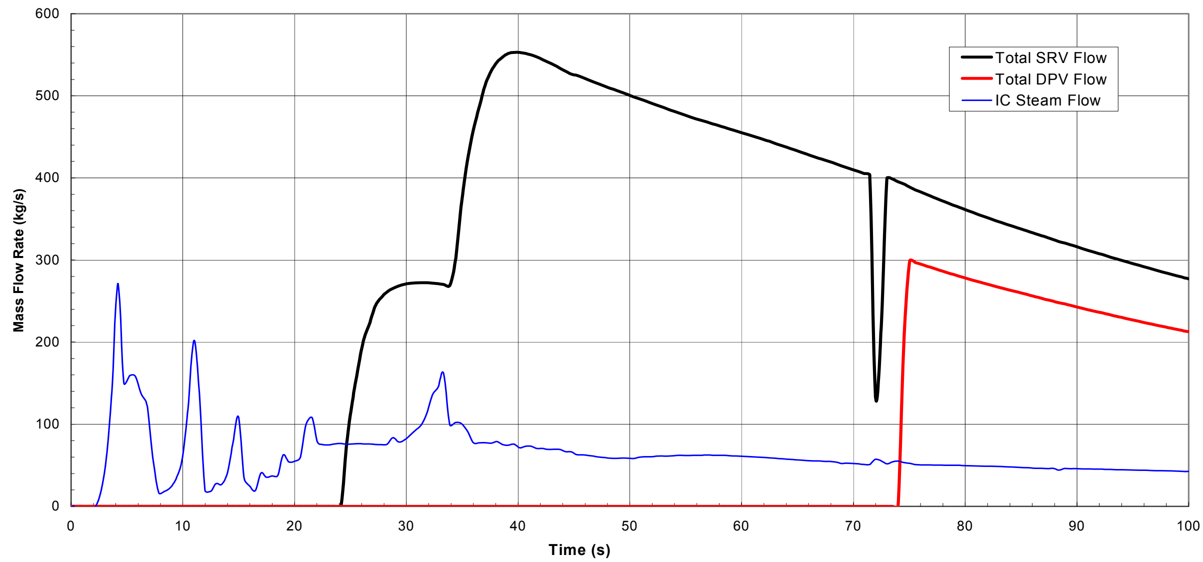


Figure 6B-21. E0 (Base Case) – DPV and SRV Flows

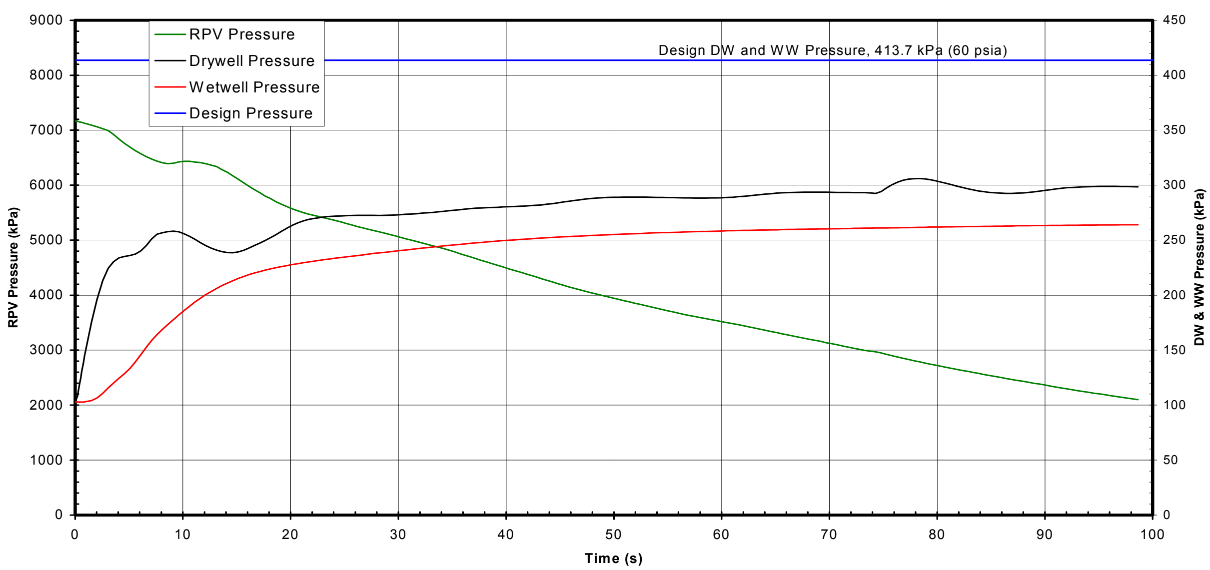


Figure 6B-22. E0a (Time Step) – Drywell Pressure Response

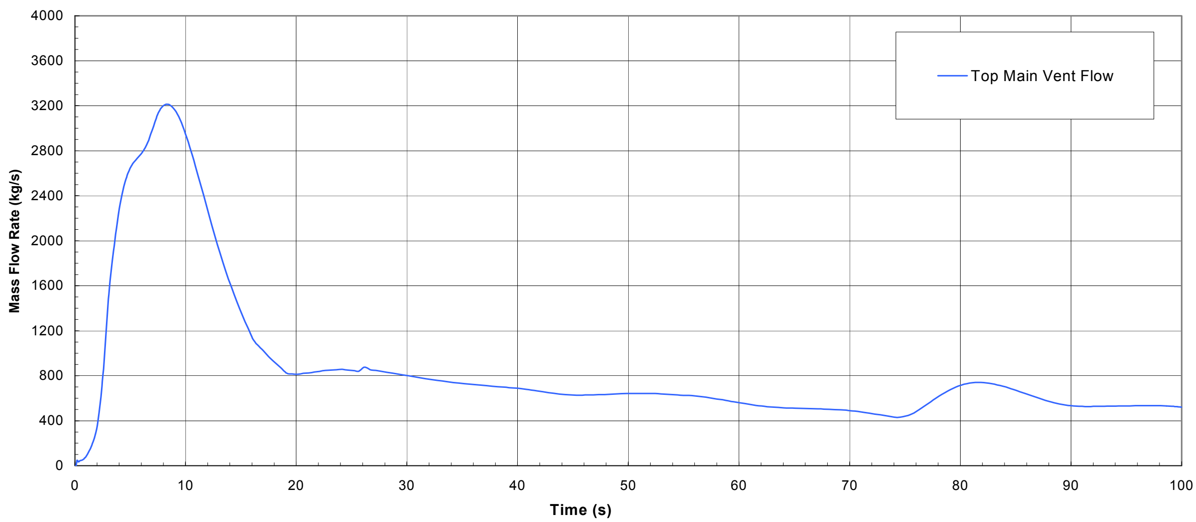


Figure 6B-23. E0a (Time Step) – Top Main Vent Flow

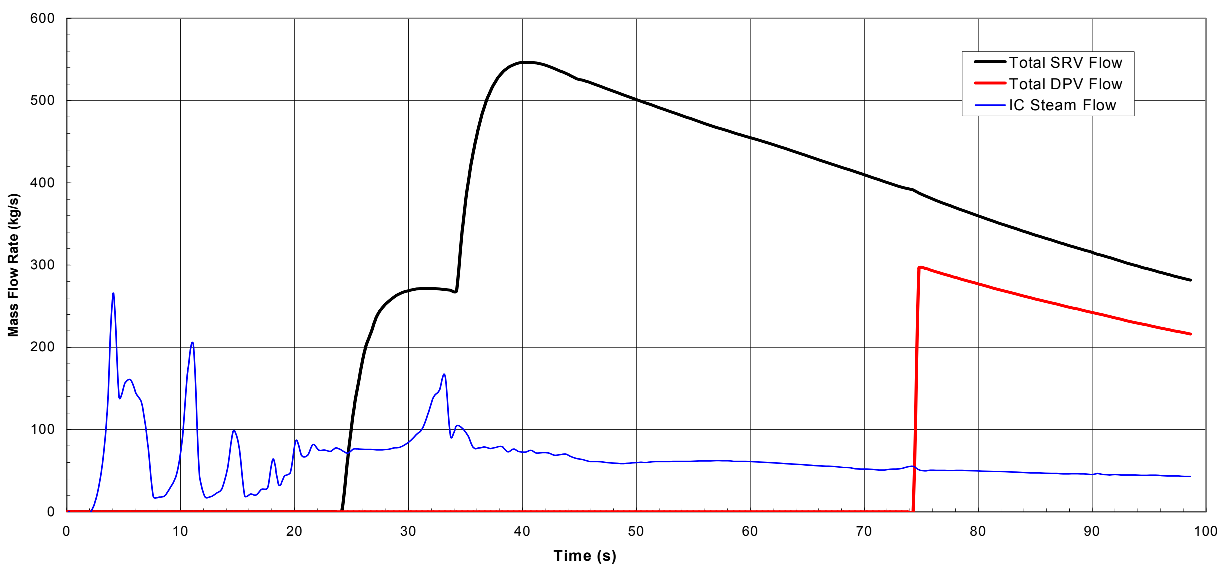


Figure 6B-24. E0a (Time Step) – DPV and SRV Flows

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 nodalization that integrates the responses between the containment and the reactor vessel, for the ECCS performance analyses. This evaluation considers the SRP Subsection 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, Subsection 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 (SRP Subsection 6.2.1.5), which could affect the ECCS performance. However, the GDCS, the DW/WW and the RPV are interconnected in the ESBWR design. 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, show that the impacts on the containment back pressure are less than 20 kPa (2.9 psi). The effects of this change in containment back pressure on the minimum chimney collapsed level are less than 0.1 m (4 in) (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 (0.0631 m^3/s (1000 gpm) DW 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 WW and suppression pool volumes are artificially increased by a factor of 1.5, 2.0 and 3.0. Larger WW volume results in smaller DW 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 WW 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 WW pressure decreases with the additional DW steam condensing heat sinks such as the DW spray and additional DW heat structures; it also decreases with larger WW volume. With the additional passive and active heat sinks, the maximum WW pressure decreases by 17 kPa (2.5 psi) from the base case value.

Figure 6C-1 shows the effect of the maximum WW pressure on the GDCS injection timing. The initiation time for the GDCS flow increases linearly as the WW pressure reduces. With the additional passive and active heat sinks, the initiation time increases by seven seconds from the base case value.

Figure 6C-2 shows the effect of the maximum WW 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 (4 in) 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.

6C.1 References

None.

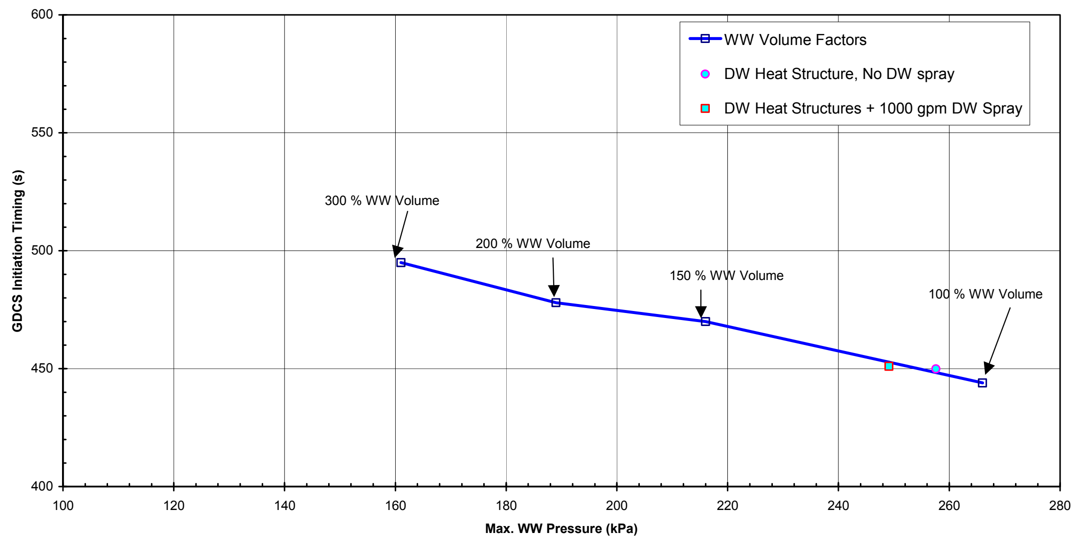


Figure 6C-1. Effect of Wetwell Pressure on the GDCS Initiation Timing (GDCS Injection Line Break, 1 GDCS Injection Valve Failure)

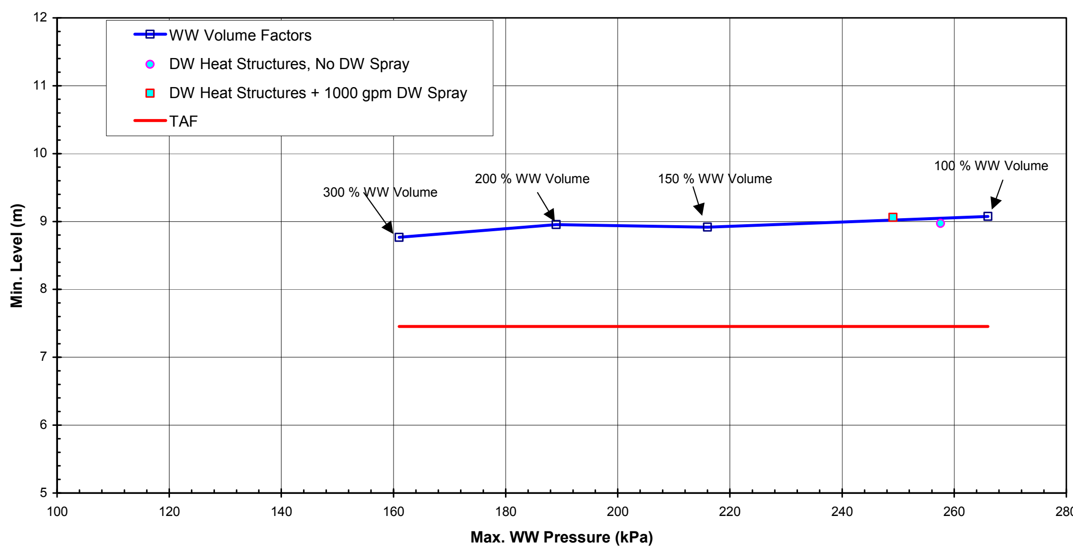


Figure 6C-2. Effect of Wetwell Pressure on the Minimum Chimney Collapsed Level (GDCS Injection Line Break, 1 GDCS Injection Valve Failure)

6D. CONTAINMENT PASSIVE HEAT SINK DETAILS

Table 6-11, Item A, in RG 1.70, requests a listing of all structures, components, and equipment used as passive heat sinks according to RG 1.70 Table 6-4A. The ESBWR containment was conservatively modeled in TRACG by excluding all piping, equipment and miscellaneous structures. The TRACG containment volume was reduced by 1% to account for piping, equipment and miscellaneous structures. The passive heat sinks that were modeled for the ESBWR are shown in Table 6D-1.

Table 6-11, Item B, in RG 1.70 requests detailed passive heat sink data. The information to be provided and the format are given in RG 1.70 Tables 6-4B, 6-4C, and 6-4D. The containment DW and WW inner and outer walls in TRACG are modeled as one-dimensional heat slabs. The modeling of these heat slabs is listed in Table 6D-2.

The thermophysical properties of the DW and WW walls are listed in Table 6D-3. RG 1.70 Table 6-11, Item C, requests a graphical display of the condensing heat transfer coefficients as functions of time for the design basis accident. TRACG output data containing heat transfer coefficients are available to compare with NRC code calculations of heat transfer coefficients.

The passive heat sinks that are modeled in the ESBWR are the DW/WW inner and outer walls. The walls are modeled as double-sided, one-dimensional heat slabs, which conduct heat in the radial direction from a TRACG cell to the next cell radially outward, or to an ambient temperature. The heat slabs are located between Levels 25 and 31 of the containment. The DW/WW inner wall connects Ring 6 to Ring 7. The DW/WW outer wall connects Ring 8 to the reactor building ambient conditions. The ambient temperature of the reactor building is 308 K (94.7ºF). Figure 6A-1 illustrates the nodalization of the DW/WW inner and outer walls. Table 6D-4 provides a further breakdown of the heat transfer area by level.

6D.1 References

None.

Table 6D-1

Listing of Passive Heat Sinks

Item 7 from RG 1.70, Table 6-4A	Internal Separation Walls and Floors (DW/WW)

Note: No other types of heat sinks listed in RG 1.70, Table 6-4A are modeled.

Table 6D-2

Modeling of Passive Heat Sinks

Passive Heat Sink	Material	Material Thickness, m (ft)	Thickness Group	Surface Area, m^2 (ft^2)
Drywell/Wetwell Inner Wall	Concrete	0.6 (1.97)	b	900 (9687)
Drywell/Wetwell Outer Wall	Concrete	2.0 (6.6)	b	1386 (14919)

Table 6D-3

Thermophysical Properties of Passive Heat Sink Materials

Material	Density, kg/m³ (lb/ft^3)	Specific Heat, J/(kg-K) (Btu/(lb-°F))	Thermal Conductivity, W/(m-K) (Btu-ft/(h-ft^2-°F))
Concrete	2322.6767 (145)	879.228 (0.21)	1.3845872 (0.8)

Table 6D-4

Total Heat Transfer Area by Containment Level

Level	Drywell/Wetwell Inner Wall, m² (ft^2)	Drywell/Wetwell Outer Wall, m² (ft^2)
25	73.81 (794.5)	201.31 (2166.9)
26	56.81 (611.5)	154.94 (1667.8)
27	62.2 (670)	169.65 (1826.1)
28	29.03 (312.5)	79.17 (852.2)
29	214.94 (2313.6)	226.19 (2434.7)
30	287.58 (3095.5)	429.77 (4626.0)
31	175.62 (1890.4)	124.41 (1339.1)
Total Area	899.99 (9687.4)	1385.44 (14912.8)

6E. TRACG LOCA CONTAINMENT RESPONSE ANALYSIS

The chronology of progressions of FWLB LOCA (Bounding Case) and Main Steam Line Break (MSLB) LOCA (Bounding Case) containment responses are discussed in detail in this appendix.

6E.1 Feedwater Line Break LOCA, 1 DPV Failure (Bounding Case)

Following the postulated FWLB LOCA, the DW pressure increases rapidly leading to the clearing of the Passive Containment Cooling System (PCCS) and main vents. The DW pressure increase is terminated at around 60 seconds (Figure 6.2-13a2), when most of the noncondensable gases in the DW annulus have been purged into the WW (Figure 6.2-13d3). The DW pressure drops to 255 kPa following the initial spike then increases gradually and continues to 72 hours (Figure 6.2-13a1).

As shown in Figure 6.2-13a2, the peak DW pressure is approximately 314 kPa (45.5 psia) at about 391 seconds, prior to the GDCS flow initiation, shortly after the opening of the depressurization valves (DPVs) (Table 6.2-7d). This peak pressure is below the design pressure of 310 kPaG (45 psig) with a large margin. The GDCS flow initiates at approximately 510 seconds. The subcooled GDCS water continues flowing into the vessel, reducing the steaming from the reactor pressure vessel (RPV) and the DW pressure. At approximately 946 seconds, the DW pressure drops below the WW pressure, causing the opening of the vacuum breakers and allowing some noncondensable gases to flow back into the DW. Consequently, the system pressures drop to a value of approximately 276 kPa (40 psia) (Figure 6.2-13a3).

Figures 6.2-13d1 through 6.2-13d3 show the noncondensable gas pressures in the DW annulus, the DW head gas space, and the GDCS pool gas space. Figure 6.2-13d3 shows that most of the noncondensable gases in the DW annulus are purged into the WW within 100 seconds. At around 1060 seconds, some noncondensable gases flow back to the DW annulus (Figure 6.2-13d3) after the opening of the vacuum breakers.

The increase of DW pressure due to the opening of the vacuum breakers occurs between 1200 and 1600 seconds. Subsequently, decay heat overcomes the subcooling in the GDCS water and steaming resumes (at ~1800 seconds, Figure 6.2-13a3). The resumption of RPV steaming causes the DW pressure to increase again starting from 1826 seconds. The DW pressure reaches the long-term peak of 368 kPa (53 psia) at 72 hours (Figure 6.2-13a1).

After 1826 seconds, the DW pressure is higher than the WW pressure. The PCCS moves the steam and noncondensable gas mixture from the DW and purges the noncondensable gases into the WW. Most of the noncondensable gases that returned to the DW annulus because of vacuum breaker openings are purged back into the WW in approximately 4 hours (Figure 6.2-13d1).

Figure 6.2-13c1 compares the total heat removal by the PCCS with the decay heat. After the first 6 hours, the PCCS is able to remove all the decay heat with some margin to spare. From this point on, all the decay heat generated by the core is transferred to the Isolation Condenser /Passive Containment Cooling System (IC/PCCS) pools, which are located outside of the containment.

Figures 6.2-13b1 through 6.2-13b3 show the DW gas temperature, WW gas temperature, and suppression pool surface temperature. As shown in Figure 6.2-13b2, the bulk DW temperature reaches a maximum of 169.5°C (337ºF) at approximately 391.5 seconds. By the end of 72 hours, the DW temperature is maintained at 142.1°C (288ºF), well below the DW design temperature of 171°C (340°F), while the WW gas temperature remains below the WW temperature design limit of 121°C (250°F) at 72 hours.

Figure 6.2-13d4 compares the water levels in the DW annulus and suppression pool. The DW annulus water level rises due to the break flow discharges from the RPV and the broken feedwater piping. In approximately 6.5 hours, the DW annulus water level reaches the quasi-equilibrium elevation of 8.96 meters (29 ft). At this elevation, the DW annulus water level is approximately 3 meters (10 ft) below the spillover holes. The hot water in the DW annulus remains in the DW and does not enter into the bottom of the suppression pool via the spillover holes.

Figure 6.2-13d5 shows the GDCS pool levels. The water level drops below L33 (Figure 6.2-7) in approximately 1.3 hours, after the initiation of GDCS flow. This creates a bottom layer of noncondensable gas space, which is approximately 6 meters (20 ft) below the connection pipes. The noncondensable gas masses stored in the top two levels (L34 and L35) are purged to minimal values in a few hours through the connection pipes.

6E.2 Main Steam Line Break LOCA, 1 DPV Failure (Bounding Case)

Following the postulated LOCA, the DW pressure increases rapidly leading to clearing of the PCCS and main vents. The containment pressure responses of the DW, WW, and RPV are shown in Figures 6.2-14a1 through 6.2-14a3. During the initial blowdown phase, the noncondensable gases in the upper regions of the DW are cleared rapidly to the WW through the main vents, as shown by the DW and GDCS gas pressures in Figures 6.2-14d2 and 6.2-14d3. The DW pressure starts to turn around and decrease at approximately 888 seconds after the GDCS flow initiates, as shown in Figure 6.2-14a3. The GDCS flow fills the reactor vessel to the elevation of the break, and then spills into the DW. The heat transfer process during the GDCS cooling period is characterized by condensation of steam in the vessel and DW. The DW pressure rises until flow is established through the PCCS, where condensate from the PCCS is recycled back into the vessel through the GDCS pool in the DW. The long-term (72 hours) response of the containment pressure is shown in Figure 6.2-14a1. The DW pressure levels off at around 2 hours and increases at a gradual pace. The peak DW pressure reaches 396 kPa (57.5 psia) at the end of 72 hours, but remains below the design pressure of 310 kPaG (45 psig).

Due to the presence of noncondensable gases, the effectiveness of condensation heat transfer in the PCCS is impaired for the first 48 hours, as shown by the PCCS heat removal versus decay heat in Figure 6.2-14c1. For the first 48 hours, the decay heat exceeds the PCCS heat removal capacity. Then after the noncondensable gases are purged from the GDCS and DW gas spaces, the PCCS heat removal capacity matches the decay heat. However, due to the production of radiolytic gases, there will be a source of noncondensable gases in the DW; these gases can degrade the PCCS heat removal capacity. Hence, both the DW and WW pressures increase between 48 hours and 72 hours.

The TRACG containment model predicts some noncondensable gases remain in the DW at 72 hours. As a conservative approach to determine the maximum DW pressure, an ideal gas calculation is performed to calculate the effect of purging these remaining gases to the WW. The calculation accounts for the total noncondensable gas in the containment, the noncondensable gas dissolved in the sodium pentaborate solution, and the safety/nonsafety-related pneumatic containment valves during the MSLB event. The increase in WW and DW pressure from additional noncondensable gas is based on the ideal gas equation. The result of this noncondensable addition is shown in Table 6.2-5a.

Figures 6.2-14b1 through 6.2-14b3 show the DW, WW and suppression pool gas space temperature responses at different elevations. Initially, all elevations heat up due to the main vent and break discharge. After the main vents close, only the upper levels are impacted by the PCCS vent discharge. The bulk DW gas temperature peaks at 173.6°C (344.5ºF) immediately following the blowdown, which is slightly above the DW design temperature of 171°C (340°F), due to adiabatic compression by the steam discharging from the break. In the long-term (72 hours), the temperature remains below 171°C (340°F).

Figure 6.2-14d4 compares the water levels in the DW annulus and suppression pool. The DW annulus water level rises due to the break flow discharges from the main steam line. The DW annulus water level remains in the lower DW and does not enter into the bottom of the suppression pool via the spillover holes.

For MSLB, the GDCS pool level (Figure 6.2-14d5) drops to the elevation of the DPVs and stays above L33 (Figure 6.2-7). As no gas mass is stored below L33, gas masses stored in the top two levels (L34 and L35) are purged to minimal values in a few hours through the connection pipes. Approximately 60% (or 1200 m^3 (42380 ft^3)) of the initial GDCS cold water 46.1°C (115°F) remains inside the GDCS pools. The hot PCCS condensate mixes with the cold GDCS pool water. The temperature of the GDCS water injected into the RPV is lower than that of the FWLB. A portion of the decay heat is used to heat up the incoming colder GDCS water, which is shown in Figure 6.2-14c3 as the difference between the decay heat and the total PCCS condensation power. It takes approximately 60 hours to bring the mixture temperature to approximately 95°C (Figure 6.2-14d6).

Both the initial blowdown and the ensuing suppression pool heat up are larger in the MSLB than the FWLB, which leads to higher pool water temperature and higher suppression pool water level and smaller WW gas space volume. All these effects result in higher DW pressure in the MSLB than FWLB.

6E.3 Feedwater Line Break LOCA, 1 SRV Failure (Bounding Case)

Following the postulated FWLB LOCA, the DW pressure increases rapidly leading to the clearing of the Passive Containment Cooling System (PCCS) and main vents. The DW pressure increase is terminated at around 65 seconds (Figure 6.2-13e2), when most of the noncondensable gases in the DW annulus have been purged into the WW (Figure 6.2-13h3). The DW pressure drops to 250 kPa following the initial spike then increases gradually and continues to 72 hours (Figure 6.2-13e1).

As shown in Figure 6.2-13e2, the peak DW pressure is approximately 314 kPa (45.5 psia) at 416 seconds, prior to the GDCS flow initiation, shortly after the opening of the depressurization valves (DPVs) (Table 6.2-7f). This peak pressure is below the design pressure of 310 kPaG (45 psig) with a large margin. The GDCS flow initiates at approximately 511 seconds. The subcooled GDCS water continues flowing into the vessel, reducing the steaming from the reactor pressure vessel (RPV) and the DW pressure. At approximately 855 seconds, the DW pressure drops below the WW pressure, causing the opening of the vacuum breakers and allowing some noncondensable gases to flow back into the DW. Consequently, the system pressures drop to a value of approximately 277 kPa (40 psia) (Figure 6.2-13e3).

Figures 6.2-13h1 through 6.2-13h3 show the noncondensable gas pressures in the DW annulus, the DW head gas space, and the GDCS pool gas space. Figure 6.2-13h3 shows that most of the noncondensable gases in the DW annulus are purged into the WW within 100 seconds. At around 1181 seconds, some noncondensable gases flow back to the DW annulus (Figure 6.2-13h3) after the opening of the vacuum breakers.

The increase of DW pressure due to the opening of the vacuum breakers occurs between 1100 and 1500 seconds. Subsequently, decay heat overcomes the subcooling in the GDCS water and steaming resumes (at ~1800 seconds, Figure 6.2-13e3). The resumption of RPV steaming causes the DW pressure to increase again starting from 1845 seconds. The DW pressure reaches the long-term peak of 370 kPa (54 psia) at 72 hours (Figure 6.2-13e1).

After 1845 seconds, the DW pressure is higher than the WW pressure. The PCCS moves the steam and noncondensable gas mixture from the DW and purges the noncondensable gases into the WW. Most of the noncondensable gases that returned to the DW annulus because of vacuum breaker openings are purged back into the WW in approximately 4 hours (Figure 6.2-13h1).

Figure 6.2-13g1 compares the total heat removal by the PCCS with the decay heat. After the first 6 hours, the PCCS is able to remove all the decay heat with some margin to spare. From this point on, all the decay heat generated by the core is transferred to the IC/PCCS pools, which are located outside of the containment.

Figures 6.2-13f1 through 6.2-13f3 show the DW gas temperature, WW gas temperature, and suppression pool surface temperature. As shown in Figure 6.2-13f2, the bulk DW temperature reaches a maximum of 168°C (334ºF) at approximately 415 seconds. By the end of 72 hours, the DW temperature is maintained at 142.2°C (288ºF), well below the DW design temperature of 171°C (340°F).

Figure 6.2-13h4 compares the water levels in the DW annulus and suppression pool. The DW annulus water level rises due to the break flow discharges from the RPV and the broken feedwater piping. In approximately 6 hours, the DW annulus water level reaches the quasi-equilibrium elevation of 9 meters (29 ft). At this elevation, the DW annulus water level is approximately 3 meters (10 ft) below the spillover holes. The hot water in the DW annulus remains in the DW and does not enter into the bottom of the suppression pool via the spillover holes.

Figure 6.2-13h5 shows the GDCS pool levels. The water level drops below L33 (Figure 6.2-7) in approximately 3 hours, after the initiation of GDCS flow. This creates a bottom layer of noncondensable gas space, which is approximately 6 meters (20 ft) below the connection pipes. The noncondensable gas masses stored in the top two levels (L34 and L35) are purged to minimal values in a few hours through the connection pipes.

6E.4 Main Steam Line Break LOCA, 1 SRV Failure (Bounding Case)

Following the postulated LOCA, the DW pressure increases rapidly leading to clearing of the PCCS and main vents. The containment pressure responses of the DW, WW, and RPV are shown in Figures 6.2-14f1 through 6.2-14f3. During the initial blowdown phase, the noncondensable gases in the upper regions of the DW are cleared rapidly to the WW through the main vents, as shown by the DW and GDCS gas pressures in Figures 6.2-14i2 and 6.2-14i3. The DW pressure starts to turn around and decrease at approximately 900 seconds after the GDCS flow initiates, as shown in Figure 6.2-14f3. The GDCS flow fills the reactor vessel to the elevation of the break, and then spills into the DW. The heat transfer process during the GDCS cooling period is characterized by condensation of steam in the vessel and DW. The DW pressure rises until flow is established through the PCCS, where condensate from the PCCS is recycled back into the vessel through the GDCS pool in the DW. The long-term (72 hours) response of the containment pressure is shown in Figure 6.2-14f1. The DW pressure levels off at around 2 hours and increases at a gradual pace. The peak DW pressure reaches 397 kPa (58 psia) at the end of 72 hours, but remains below the design pressure of 310 kPaG (45 psig).

Due to the presence of noncondensable gases, the effectiveness of condensation heat transfer in the PCCS is impaired for the first 48 hours, as shown by the PCCS heat removal versus decay heat in Figure 6.2-14h1. For the first 48 hours, the decay heat exceeds the PCCS heat removal capacity. Then after the noncondensable gases are purged from the GDCS and DW gas spaces, the PCCS heat removal capacity matches the decay heat. However, due to the production of radiolytic gases, there will be a source of noncondensable gases in the DW; these gases can degrade the PCCS heat removal capacity. Hence, both the DW and WW pressures increase between 48 and 72 hours.

The TRACG containment model predicts some noncondensable gases remain in the DW at 72 hours. As a conservative approach to determine the maximum DW pressure, an ideal gas calculation is performed to calculate the effect of purging these remaining gases to the WW. The calculation accounts for the total noncondensable gas in the containment, the noncondensable gas dissolved in the sodium pentaborate solution, and the safety/nonsafety-related pneumatic containment valves during the MSLB event. The increase in WW and DW pressure from additional noncondensable gas is based on the ideal gas equation. The result of this noncondensable addition is shown in Table 6.2-5 and Table 6.2-5a.

Figures 6.2-14g1 through 6.2-14g3 show the DW, WW and suppression pool gas space temperature responses at different elevations. Initially, all elevations heat up due to the main vent and break discharge. After the main vents close, only the upper levels are impacted by the PCCS vent discharge. The bulk DW gas temperature peaks at 173.6°C (344.5ºF) immediately following the blowdown, which is slightly above the DW design temperature of 171°C (340°F), due to adiabatic compression by the steam discharging from the break. In the long-term (72 hours), the temperature remains below 171°C (340°F).

Figure 6.2-14i4 compares the water levels in the DW annulus and suppression pool. The DW annulus water level rises due to the break flow discharges from the main steam line. The DW annulus water level remains in the lower DW and does not enter into the bottom of the suppression pool via the spillover holes.

For MSLB, the GDCS pool level (Figure 6.2-14i5) drops to the elevation of the DPVs and stays above L33 (Figure 6.2-7). As no gas mass is stored below L33, gas masses stored in the top two levels (L34 and L35) are purged to minimal values in a few hours through the connection pipes. Approximately 60% (or 1200 m^3 (42380 ft^3)) of the initial GDCS cold water 46.1°C (115°F) remains inside the GDCS pools. The hot PCCS condensate mixes with the cold GDCS pool water. The temperature of the GDCS water injected into the RPV is lower than that of the

FWLB. A portion of the decay heat is used to heat up the incoming colder GDCS water, which is shown in Figure 6.2-14h3 as the difference between the decay heat and the total PCCS condensation power. It takes approximately 60 hours to bring the mixture temperature to approximately 95°C (Figure 6.2-14i6).

Both the initial blowdown and the ensuing suppression pool heat up are larger in the MSLB than the FWLB, which leads to higher pool water temperature and higher suppression pool water level and smaller WW gas space volume. All these effects result in higher DW pressure in the MSLB than FWLB.

6E.5 Main Steam Line Break LOCA, 1 SRV Failure (Bounding Case, with Offsite Power)

Following the postulated LOCA, the DW pressure increases rapidly leading to clearing of the PCCS and main vents. The containment pressure responses of the DW, WW, and RPV are shown in Figures 6.2-14j1 through 6.2-14j3. During the initial blowdown phase, the noncondensable gases in the upper regions of the DW are cleared rapidly to the WW through the main vents, as shown by the DW and GDCS gas pressures in Figures 6.2-14m2 and 6.2-14m3. The DW pressure starts to turn around and decrease at approximately 1300 seconds after the GDCS flow initiates, as shown in Figure 6.2-14j3. The GDCS flow fills the reactor vessel to the elevation of the break, and then spills into the DW. The heat transfer process during the GDCS cooling period is characterized by condensation of steam in the vessel and DW. The DW pressure rises until flow is established through the PCCS, where condensate from the PCCS is recycled back into the vessel through the GDCS pool in the DW. The long-term (72 hours) response of the containment pressure is shown in Figure 6.2-14j1. The DW pressure levels off at around 2 hours and increases at a gradual pace. The peak DW pressure reaches 394 kPa (57 psia) at the end of 72 hours, but remains below the design pressure of 310 kPaG (45 psig).

Due to the presence of noncondensable gases, the effectiveness of condensation heat transfer in the PCCS is impaired for the first 48 hours, as shown by the PCCS heat removal versus decay heat in Figure 6.2-14l1. For the first 48 hours, the decay heat exceeds the PCCS heat removal capacity. Then after the noncondensable gases are purged from the GDCS and DW gas spaces, the PCCS heat removal capacity matches the decay heat. However, due to the production of radiolytic gases, there will be a source of noncondensable gases in the DW; these gases can degrade the PCCS heat removal capacity. Hence, both the DW and WW pressures increase between 48 hours and 72 hours.

The TRACG containment model predicts some noncondensable gases remain in the DW at 72 hours. As a conservative approach to determine the maximum DW pressure, an ideal gas calculation is performed to calculate the effect of purging these remaining gases to the WW. The calculation accounts for the total noncondensable gas in the containment, the noncondensable gas dissolved in the sodium pentaborate solution, and the safety/nonsafety-related pneumatic containment valves during the MSLB event. The increase in WW and DW pressure from additional noncondensable gas is based on the ideal gas equation. The result of this noncondensable addition is shown in Table 6.2-5a.

Figures 6.2-14k1 through 6.2-14k3 show the DW, WW and suppression pool gas space temperature responses at different elevations. Initially, all elevations heat up due to the main vent and break discharge. After the main vents close, only the upper levels are impacted by the PCCS vent discharge. The bulk DW gas temperature peaks at 175.0°C (347.0ºF) immediately following the blowdown, which is slightly above the DW design temperature of 171°C (340°F), due to adiabatic compression by the steam discharging from the break. In the long-term (72 hours), the temperature remains below 171°C (340°F).

Figure 6.2-14m4 compares the water levels in the DW annulus and suppression pool. The DW annulus water level rises due to the break flow discharges from the main steam line. The DW annulus water level remains in the lower DW and does not enter into the bottom of the suppression pool via the spillover holes.

For MSLB, the GDCS pool level (Figure 6.2-14m5) drops to the elevation of the DPVs and stays above L33 (Figure 6.2-7). As no gas mass is stored below L33, gas masses stored in the top two levels (L34 and L35) are purged to minimal values in a few hours through the connection pipes. Approximately 60% (or 1200 m^3 (42380 ft^3)) of the initial GDCS cold water 46.1°C (115°F) remains inside the GDCS pools. The hot PCCS condensate mixes with the cold GDCS pool water. The temperature of the GDCS water injected into the RPV is lower than that of the FWLB. A portion of the decay heat is used to heat up the incoming colder GDCS water, which is shown in Figure 6.2-14l3 as the difference between the decay heat and the total PCCS condensation power. It takes approximately 60 hours to bring the mixture temperature to approximately 95°C (Figure 6.2-14m6).

Both the initial blowdown and the ensuing suppression pool heat up are larger in the MSLB than the FWLB, which leads to higher pool water temperature and higher suppression pool water level and smaller WW gas space volume. All these effects result in higher DW pressure in the MSLB than FWLB.

6E.6 References

None.

Figure 6E-1. (Deleted)

Figure 6E-2. (Deleted)
Figure 6E-3. (Deleted)

Figure 6E-4. (Deleted)

6F. BREAK SPECTRA OF BREAK SIZES AND BREAK ELEVATIONS

6F.1 Spectrum of Break Sizes

Parametric cases were performed prior to DCD Revision 2 with different break areas (40%, 60%, 80% and 100% of the DEG break area) for the feedwater line break and the main steam line break. The results of these parametric cases are discussed below:

Main Steam Line Break - Parametric Study on the Break Areas

The base case (MSL-8F_1DPV-72) considers a single failure of one DPV and nominal conditions (Table 6.2-6), and assumes 100% DEG break.

Parametric cases were performed with the break area varied (for both ends of the break pipe) from 100% break area to 40%, 60%, and 80%, respectively. The peak DW pressures for these cases are summarized in Table 6F1-1. The transient DW pressures of these cases are compared and shown in Figure 6F1-1.

The peak DW pressures for these cases occur near the end of the calculation (72 hrs), and the cases with larger break area calculate slightly higher peak DW pressures. The base case (100% break area) calculates a peak DW pressure of 323 kPa (46.85 psia). It should be noted that the corresponding margin to the design pressure of 310 kPaG (45.0 psig) is 29%.

The long-term noncondensable gas distribution depends on the noncondensable gas circulation in the DW annulus, which affects the initial removal and subsequent removals of the noncondensable gases returning to the DW due to the vacuum breaker openings. The noncondensable gas circulation pattern depends on the DW annulus geometry, and the strength and location of the steam source. To cap the effect and the uncertainty of the noncondensable gas distribution on the DW pressure, sensitivity cases were performed by changing the location of the steam source. Results of these sensitivities are discussed in Appendix 6F.2.

Feedwater Line Break - Parametric Study on the Break Areas

The base case (FWL-8D_1SRV-72) considers a single failure of one SRV and nominal conditions (Table 6.2-6), and assumes 100% DEG break.

Parametric cases were performed with the break area varied (for both ends of the break pipe) from 100% break area to 40%, 60%, and 80%, respectively. The peak DW pressures for these cases are summarized in Table 6F1-2. The transient DW pressures of these cases are compared and shown in Figures 6F1-2a and 6F1-2b.

The peak DW pressure for these cases occurs at around 78 seconds, shortly after the DPV opening. The change in peak DW pressure is less than 0.3 kPa (.04 psia) (< 0.1% of the peak value) when the break size changes from 100% to 40%.

6F.2 Spectrum of Break Elevations

Sensitivity cases were performed with different break discharge location to the DW annulus for the main steam line break. The break elevation varies between the RPV bottom elevation and the MSL elevation. The results of these parametric cases are discussed in the following.

Main Steam Line Break - Parametric Study on the Break Discharge Elevation

The base case (MSL-8F_1DPV-72) considers a single failure of one DPV and nominal conditions (Table 6.2-6), and assumes 100% DEG break.

In the base case (MSL-8F_1DPV-72), the broken MSL from the RPV discharges steam into the DW at Level 34 (DCD Tier 2 prior to DCD Revision 2, Figure 6.2-7, TRACG nodalization). Sensitivity cases were performed with different elevation for the break pipe discharge location. For the sensitivity cases, the break location changes from Level 34 to Level 23 (RPV bottom), to Level 25, and to Level 31. The peak DW pressures for these cases are summarized in Table 6F2-1. The transient DW pressures of these cases are compared and shown in Figure 6F2-1.

Figure 6F2-1 shows that the base case with highest break location (MSL-8F_1DPV-72) generates the limiting DW pressure. The peak DW pressure from this case also bounds those from cases with different break area discussed in Appendix 6F.1. It should be noted that, for this limiting case, the margin to the design pressure of 310 kPaG (45.0 psig) is 29%, at the end of the 72-hour transient.

Many parameters affect the gas circulation pattern, which in turn affects the noncondensable gas distribution and the containment pressure. The limiting long-term containment pressure is determined by performing parametric cases with a different elevation for the break pipe discharge location.

6F.3 References

None.

Table 6F1-1

Summary of Peak DW Pressures for the MSL Break Area Study

Case ID	DEG Break Size	Max. DW Pressure kPa (psia)	Time at Max. (hr)
MSL-8F_1DPV-72 (Base Case)	100%	323.0 (46.85)	71.7
MSL-8F_1DPVP8-72	80%	316.8 (45.95)	71.9
MSL-8F_1DPVP6-72	60%	312.7 (45.35)	71.6
MSL-8F_1DPVP4-72	40%	312.1 (45.27)	71.4

Table 6F1-2

Summary of Peak DW Pressures for the FWL Break Area Study

Case ID	DEG Break Size	Max. DW Pressure kPa (psia)	Time at Max. (sec)
FWL-8D_1SRV-72 (Base Case)	100%	306.2 (44.41)	77.8
FWL-8D_1SRVP8-72	80%	306.0 (44.38)	77.9
FWL-8D_1SRVP6-72	60%	305.9 (44.37)	78.2
FWL-8D_1SRVP4-72	40%	305.9 (44.37)	78.3

Table 6F2-1

Summary of Peak DW Pressures for the MSL Break Elevation Study

Case ID	Break Location* in DW	Max. DW Pressure kPa (psia)	Time at Max. (hr)
MSL-8F_1DPV-72 (Base Case)	Level 34	323.0 (46.85)	71.7
MSL-8F_1DPVL31-72	Level 31	316.3 (45.88)	71.5
MSL-8F_1DPVL25-72	Level 25	316.4 (45.89)	71.2
MSL-8F_1DPVL23-72	Level 23	314.6 (45.63)	71.3

* DCD Revision 3, Figure 6.2-7, TRACG nodalization

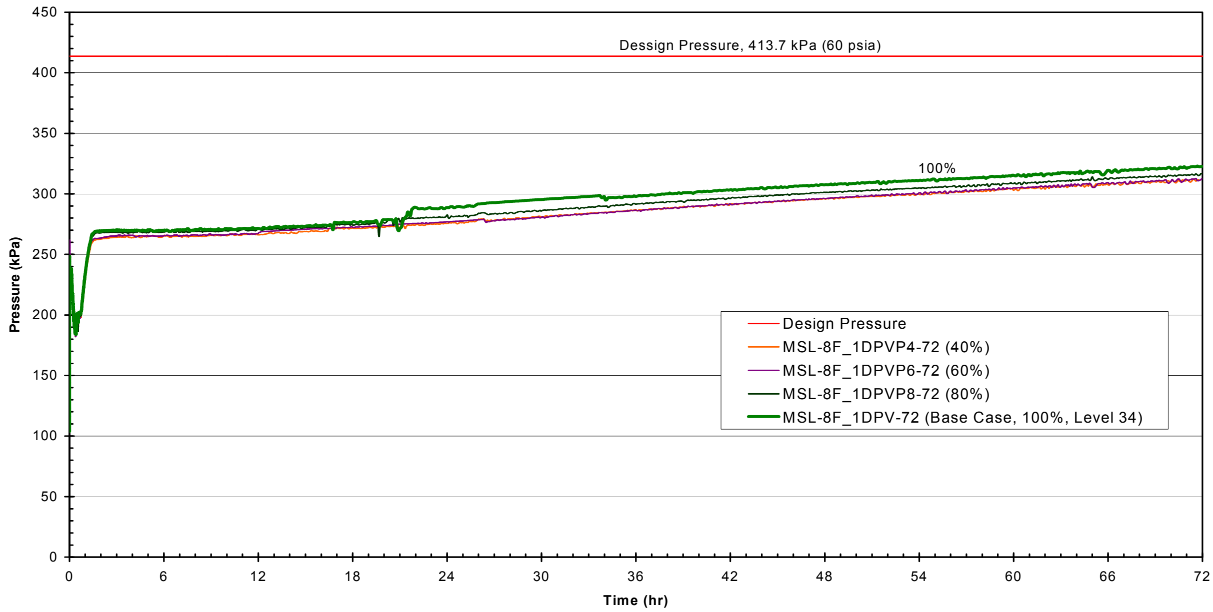


Figure 6F1-1. MSLB - Effect of Break Areas on Transient DW Pressures

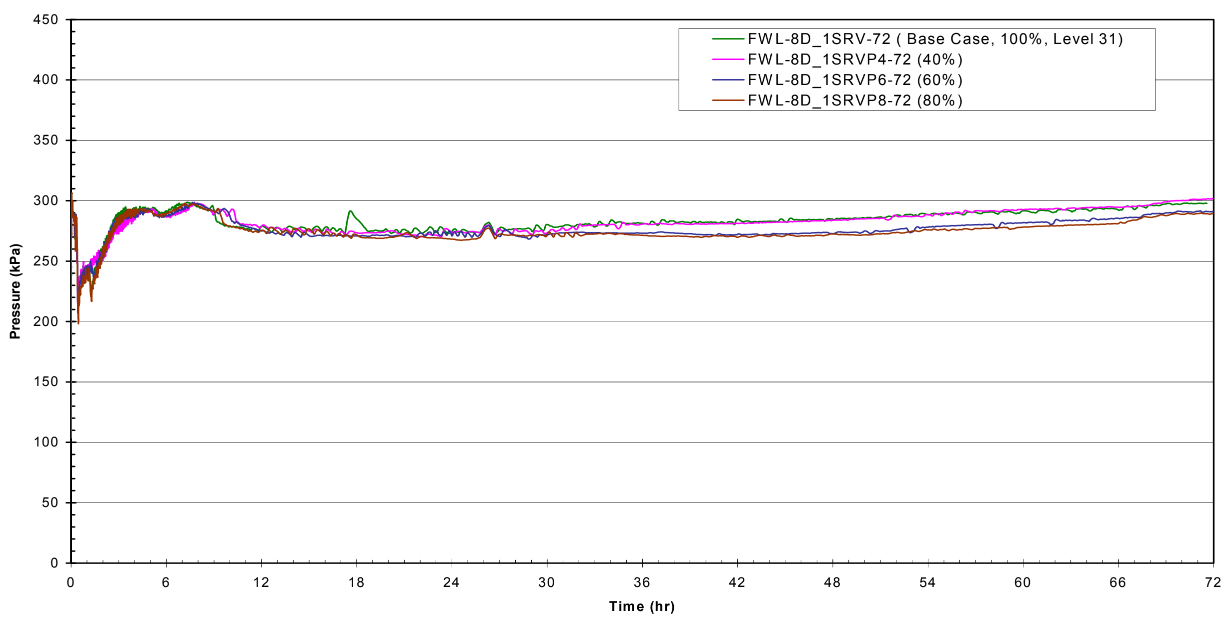


Figure 6F1-2a. Feedwater Line Break - Parametric Study on the Break Areas (0-72 hrs)

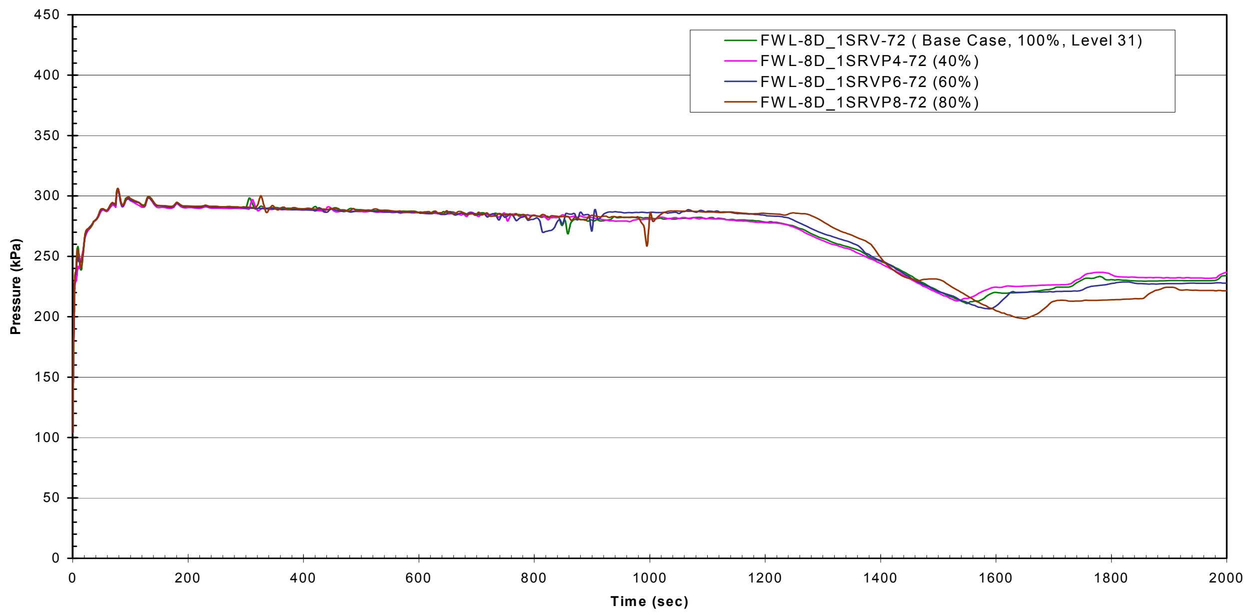


Figure 6F1-2b. Feedwater Line Break - Parametric Study on the Break Areas (0-2000 sec)

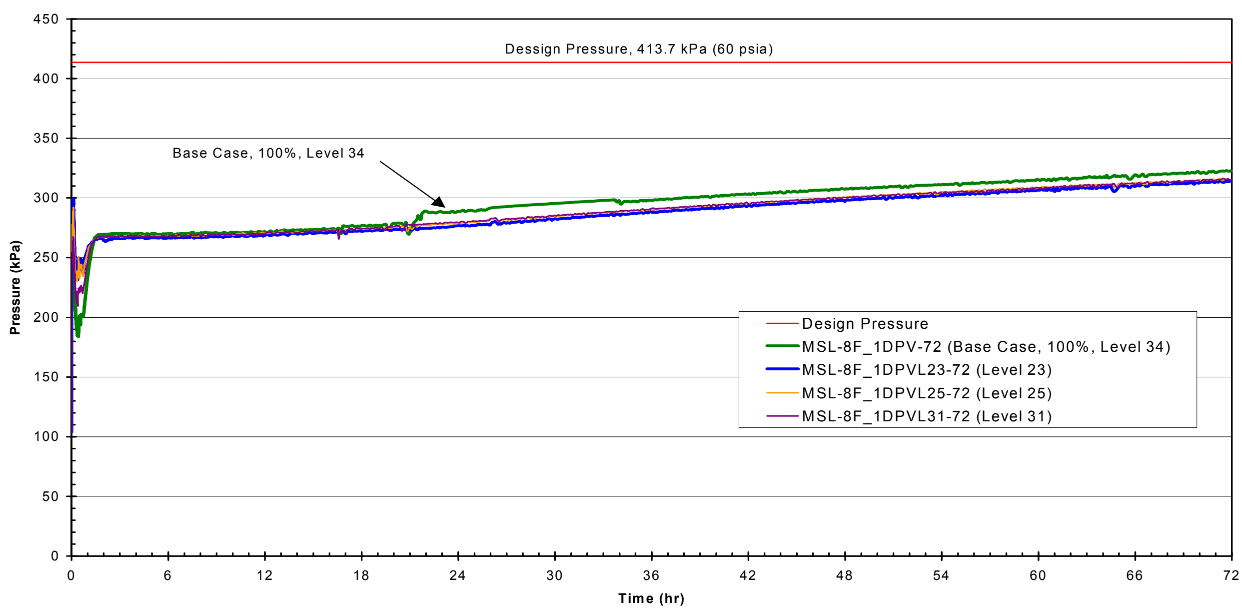


Figure 6F2-1. Main Steam Line Break - Parametric Study on the Break Elevation

6G. TRACG LOCA SER CONFIRMATION ITEMS

This appendix presents responses to TRACG LOCA SER confirmation items (Reference 6G-1); 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. Materials included in this appendix are taken from Reference 6G-1 that is based on the results from TRACG analyses performed at the time of the request. Subsequently, the containment and LOCA analyses have been refined and results are updated in the DCD. Therefore, some of the materials presented in the SER confirmation responses are no longer appropriate to be included in the DCD Chapter 6.

The RPV level response for the long-term PCCS period is discussed in Section 6G.1. The phenomena identification and ranking table is presented in Section 6G.2 and Table 6G-1. Major design changes from pre-application review design to DCD design are shown in Table 6G-2 and Table 6G-3.

6G.1 Reactor Pressure Vessel Level Response for the Long-Term PCCS Period

The ESBWR test and analysis program description (TAPD) (Reference 6G-2) defines three phases of the ESBWR LOCA transient (Figure 6G-1). These are the Blowdown Phase, GDCS Phase, and the Long-Term PCCS Phase. The LOCA scenarios and phenomena identification and ranking table (PIRT) were discussed in the TAPD for all three phases for the containment response and for the first two phases for the RPV response. The purpose of this section is to cover the RPV level responses for the long-term phase.

At the end of the GDCS phase, the chimney and downcomer levels have been restored by the GDCS flow to elevations that are well above the short-term minimum values. The end of the GDCS phase refers to a time when the GDCS tanks have drained most of their initial inventory into the RPV and through the break to the DW, and a quasi-steady equilibrium has been established between the levels in the RPV downcomer and the GDCS pools. The maximum inventory in the lower DW is limited by the "spillover pipes." These are pipes in the vertical sections of the main vent pipes between the DW and the suppression pool. As the level in the lower DW reaches this elevation, water spills over into the spillover pipes and is diverted to the bottom of the suppression pool. The long-term RPV levels are determined primarily by the break elevations. For low breaks, lower than the elevation of the spillover holes (upper openings of the spillover pipes), a manometric balance is established between the levels in the RPV downcomer and the lower DW. For these breaks, the asymptotic level position in the downcomer and DW is close to the elevation of the spillover holes. For high breaks, the long-term level is close to the break elevation. Because the long-term inventory distribution between the GDCS pools, RPV and DW is somewhat different for high elevation breaks (feedwater, steam line) and lower breaks (GDCS, bottom drain line), the scenarios for the individual break locations are discussed separately. In all cases, the long-term level remains higher than the short-term minimum level reached in the blowdown phase. The equalization line does not open for any of these breaks.

6G.1.1 Main Steam Line Break

The water inventory distribution shortly after the initiation of the LOCA and in the long-term are shown in Figure 6G-2. At the end of the GDCS period, the RPV is filled to the elevation of the

DPVs. The GDCS pools have also drained to this level, leaving about two-thirds of their initial inventory in the pools. Flow through the break discharges to the DW and accumulates in the lower DW. Because of the large inventory retained in the RPV and GDCS pools, the amount of water in the lower DW is small. In the long-term period, the steam generated by the decay heat is condensed in the PCCS and returned to the RPV via the GDCS pools. A small amount of steam condenses on the DW surfaces and does not return to the RPV. This does not appreciably affect the RPV water level over 72 hours. Figure 6G-3 shows a TRACG calculation for 12 hours that illustrates this behavior. (The TRACG chimney level is bounded by the elevation of the top of the chimney, which is 14.5 m (47.6 ft). The level in the separators would be close to the downcomer level in the long-term phase.)

The parameters that affect the long-term behavior are the capacity of the GDCS pool relative to the RPV volume, the heat removal capacity of the PCCS relative to decay heat, and to a smaller degree the condensation on DW surfaces relative to the condensation in the PCCS.

6G.1.2 Feedwater Line Break

The feedwater line break is also a high elevation break (at 18.9 m (62.0 ft) from vessel bottom) and the inventory distribution sketched in Figure 6G-2 also applies to this break. During the first hour of the transient, the RPV fills to the elevation of the feedwater lines, with flow draining out of the break. The GDCS pools have also drained to this level. Flow through the break discharges to the DW and accumulates in the lower DW. Because of the large break flow from the upstream side of the feedwater lines and the direct contact heat exchanger, the level in the DW rises significantly. The GDCS pools continue to drain and the level in the pools and in the RPV downcomer drops to the elevation of the entrance to the GDCS drain line (close to the bottom of the GDCS pools) after several hours. During this period, the PCCS cannot condense all the steam generated by decay heat and the inventory lost to steaming is partially offset by the continued draining of the GDCS pools. Once the GDCS pools are drained, the level in the downcomer falls faster until the PCCS capacity catches up with the decay heat and the RPV water level is maintained a little below the feedwater line elevation. Figure 6G-4 shows a TRACG calculation for 12 hours that illustrates this behavior. In the long-term period, a small amount of steam condenses on the DW surfaces and does not return to the RPV. This does not appreciably affect the RPV water level over 72 hours. The water from the break accumulates in the lower DW. The inventory addition from the feedwater lines causes the level in the DW to get high enough to return to the suppression pool via the spillover holes in the vertical vent pipes. However, the DW level remains well below the RPV break location.

The parameters that affect the long-term behavior are the capacity of the GDCS pool relative to the RPV volume, the heat removal capacity of the PCCS relative to decay heat, and to a smaller degree the condensation on DW surfaces relative to the condensation in the PCCS.

6G.1.3 Bottom Drain Line (BDL) Break

The water inventory distribution shortly after the initiation of the LOCA and in the long-term for a break in the bottom drain line are shown in Figure 6G-5. For low breaks, the inventory balance between the lower DW and the RPV becomes important. During the first hour of the transient, the RPV fills almost to the elevation of the steam lines, despite the small break flow from the bottom of the vessel. Flow through the break discharges to the DW and accumulates in the lower DW. In a few hours, the GDCS pools have drained completely. At this point, the level in the downcomer starts dropping faster as inventory continues to be lost to break flow and steaming with reduced compensation from GDCS drain flow. Concurrently, the water accumulating in the lower DW builds up to the elevation of the spillover holes (10.65 m (34.94 ft)). Figure 6G-6 shows a TRACG calculation for 12 hours that illustrates this behavior. The levels inside the RPV and in the DW come together and reach an equilibrium position where the difference in the two levels is the head loss for the decay heat-generated steam flowing through the DPVs. In the long-term period, the steam generated by the decay heat is condensed in the PCCS and returned to the RPV via the GDCS pools. A small amount of steam condenses on the DW surfaces and does not return to the RPV. This does not affect the RPV water level over 72 hours.

The parameters that affect the long-term behavior are the capacity of the GDCS pool relative to the lower DW volume, the pressure drop through the DPVs, the heat removal capacity of the PCCS relative to decay heat and to a smaller degree the condensation on DW surfaces relative to the condensation in the PCCS.

6G.1.4 GDCS Line Break

The GDCS line break is located above the top of the core. However, the elevation is slightly below the location of the spillover holes. Thus, the water in the DW can build up and communicate with the RPV through the break, similar to the situation in Figure 6G-5. During the first hour of the transient, the RPV fills almost to the elevation of the steam lines. Flow through the break discharges to the DW and accumulates in the lower DW. In a few hours, the GDCS pools have drained completely. At this point, the level in the downcomer starts dropping faster as inventory continues to be lost to break flow and steaming with reduced compensation from GDCS drain flow. Concurrently, the water accumulating in the lower DW builds up to the elevation of the spillover holes. Figure 6G-7 shows a TRACG calculation for 12 hours that illustrates this behavior. The levels inside the RPV and in the DW come together and reach an equilibrium position where the difference in the two levels is the head loss for the decay heat generated steam flow through the DPVs. This difference in levels is larger than for the BDL break, because the GDCS break is larger and the RPV and DW levels come together earlier when the decay heat is higher. In the long-term period, the steam generated by the decay heat is condensed in the PCCS and returned to the RPV via the GDCS pools. A small amount of steam condenses on the DW surfaces and does not return to the RPV. This does not affect the RPV water level over 72 hours.

The parameters that affect the long-term behavior are the capacity of the GDCS pool relative to the lower DW volume, the pressure drop through the DPVs, the heat removal capacity of the PCCS relative to decay heat and to a smaller degree the condensation on DW surfaces relative to the condensation in the PCCS.

6G.2 Phenomena Identification and Ranking Table

The PIRT is shown in Table 6G-1. It reflects the discussion of the scenarios and asymptotic level positions reached in the long-term. The key parameters are plant parameters such as the GDCS pool volume and lower DW volume. For high breaks, the level is maintained close to the break elevation as long as the PCCS can handle the decay heat after a few hours. For the low breaks, the level is maintained close to the elevation of the spillover holes by a simple manometric balance. These phenomena are readily handled by TRACG.

6G.3 References

6G-1 MFN 05-105, "TRACG LOCA SER confirmatory Items (TAC#MC8168)," October 6, 2005.

6G-2 NEDC-33079P, Rev. 1, "ESBWR Test and Analysis Program Description," March 2005; NEDO-33079, Revision 1, Class I (Non-proprietary), November 2005.

Table 6G-1

Phenomena Identification and Ranking Table (PIRT)

Governing Phenomena for ESBWR LOCA Long-Term PCCS Phase						
Break		**LOCA (Focus: Water Level in Shroud and Downcomer)**			
		MSL	**FWL**	**BDL**	**GDCS**
	Phenomena	**H = High, M = Medium, L = Low**			
A	**Region: Lower plenum (sub-focus: inventory loss)**				
	Break flow from BDL	N/A	N/A	M	N/A
C	**Region: Core/bundle (sub-focus: steam generation)**				
C24	Decay heat	H	H	H	H
E	**Region: Downcomer (sub-focus: inventory loss)**				
E8	Break flow	N/A	L	N/A	M
F	**Region: Chimney and upper plenum (sub-focus: inventory)**				
F1	Void distribution/two-phase level	L	L	L	L
EQ	**Region: Equalization Line (sub-focus: flow rate)**				
EQ	Equalization line flow	N/A	N/A	N/A	N/A
DPV	**Region: Depressurization valves (sub-focus: pressure drop)**				
DPV1	Break flow through DPVs	L	L	H	H
DPV2	Pressure drop through DPVs	L	L	H	H
PC	**Region: PCCS (sub-focus: condensation capacity)**				
PC2	Condensation capacity	M	M	H	H
DW	**Region: Drywell (sub-focus: inventory)**				
DW3	Condensation on surfaces	L	L	L	L
DWV	Lower DW volume vs. elevation	L	L	H	H
GDCS	**Region: GDCS pools (sub-focus: inventory)**				
GDV	GDCS pool volume vs. elevation	H	H	H	H
RPV	**Region: RPV (sub-focus: inventory)**				
RPV1	RPV volume vs. elevation	H	H	H	H

Table 6G-2

Major Design Changes from Pre-Application Review Design to DCD Design, Parameter

Parameter	Pre-App. Design	DCD Design	Reason for change	Impact on LOCA analysis	Justification for the Applicability of TRACG
Core Power, MWt	4000	4500	Power uprate – improved economics.	Higher core exit and chimney void fraction.	No new phenomena introduced, power density unchanged, selected system capacities increased. TRACG applies to new design.
No. of bundles	1020	1132	Increased to maintain power density.	Geometry change, see shroud diameter increase.	No new phenomena introduced. TRACG applies to new design.
Change in core shroud size	Base	+0.328 m (1.076 ft)	Increased to accommodate additional bundles.	Loss of liquid volume in downcomer (26%). Larger initial level drop.	Additional water sources included in analysis to maintain margin to core uncovery.
Core lattice	F lattice with wide blades	N lattice, standard blades	Simplification – similar to current BWR cores.	No significant LOCA effect.	No new phenomena introduced. TRACG applies to new design.
No. of CRDs	121	269	Result of going back to N lattice.	No significant LOCA effect.	No new phenomena introduced. TRACG applies to new design.
GDCS pool and airspace location	Wetwell	Drywell	Simplification. Additional containment pressure margin not needed.	Tested configuration for SBWR. Loss of containment pressure margin accommodated by reduced suppression pool heatup.	TRACG applicable to both configurations; testing included both.

Table 6G-2

Major Design Changes from Pre-Application Review Design to DCD Design, Parameter

Parameter	Pre-App. Design	DCD Design	Reason for change	Impact on LOCA analysis	Justification for the Applicability of TRACG
PCCS	4 x 13.5 MW	6 x 11 MW	Increased power level.	Percent increase larger than core power increase. Reduces pool heatup.	Heat exchanger consistent with tested prototype. TRACG applicable to the larger number of PCCS units.
ICS	4 x 30 MW	4 x 33.75 MW	Increased power level.	Maintains 3% capacity.	Tube geometry consistent with prototype, small increase in manifold length. TRACG applicable to the longer IC manifold.
Pressure relief system	12 ADS valves	10 ADS valves + 8 S/V	Increased relief capacity.	Minor impact on minimum water level.	TRACG critical flow model is independent of the number of valves; code is applicable to current design.
Containment vents	10	12	Reduced blowdown mass fluxes in vents.	Minor effect on LOCA pressure and temperature.	Reduces vent flow rate, within TRACG application range.

Table 6G-2

Major Design Changes from Pre-Application Review Design to DCD Design, Parameter

Parameter	Pre-App. Design	DCD Design	Reason for change	Impact on LOCA analysis	Justification for the Applicability of TRACG
Feedwater System		30 sec delay on L2; scram on Loss of Feedwater (LOFW); safety-related FW pump trip on FW line differential pressure.	Time delay on L2 to avoid unnecessary isolations and IC initiation when FW available. Early scram on LOFW helps initial level drop. FW pump trip terminates FW pumping additional mass and energy into containment via broken FW line.	Scram on LOFW is a slight benefit for small breaks. FW pump trip has no impact on LOCA analysis because loss of normal AC power is assumed.	TRACG control system capable of modeling design change.
Turbine bypass capacity	33%	110% option.	Flexibility.	No LOCA effect, slight reduction in number of scrams/year and improved reliability of on-site AC.	TRACG control system capable of modeling design change.
PCC drain tanks	In drywell	Eliminated PCCS drains to GDCS pools.	Simplification.	Tested configuration for SBWR.	TRACG applicable to new configuration.
Suppression Pool volume	3610 m^3 (127500 ft^3)	4424 m^3 (156200 ft^3)	DW/GDCS pool & WW diameter increased to provide improved equipment clearances. Additional benefit: larger suppression pool size.	Reduces pool heatup.	No new phenomena introduced. TRACG code applicable to changed volume.
DW/WW volume ratio	1.31	1.33	Ratio was not exactly maintained in containment diameter increase.	Small increase in containment pressure.	No new phenomena introduced. TRACG code applicable to changed

Table 6G-2

Major Design Changes from Pre-Application Review Design to DCD Design, Parameter

Parameter	Pre-App. Design	DCD Design	Reason for change	Impact on LOCA analysis	Justification for the Applicability of TRACG
					volume.
Spillover connection (DW annulus to suppression pool)	Holes	Pipes discharging to suppression pool at elevation of bottom horizontal vent.	Enhanced suppression pool mixing.	Reduces pool heatup and WW pressure. Pipes are closed until after the RPV blowdown to prevent any change in hydrodynamic loads.	No new phenomena introduced. Discharge location consistent with bottom horizontal vent. TRACG code applicable.
Lower DW free volume to top of active fuel elevation	1564 m^3 (55230 ft^3)	1190 m^3 (42020 ft^3)	Lower drywell volume reduced.	Improved long-term LOCA response in bottom drain line and GDCS breaks.	No new phenomena introduced. TRACG code applicable to changed volume.
SLC System activated on ADS	No	Yes	Compensate for larger initial level drop.	Improves LOCA minimum water level.	TRACG models are applicable to liquid flow into bypass.

Table 6G-3

Major Design Changes from Pre-Application Review Design to DCD Design, Modeling

Changes in LOCA Analysis	Effect
Integrated TRACG base deck with consistent, detailed nodalization for both RPV and containment.	More accurate analysis. Addresses SER confirmation item.
Credit for water added by HCUs during scram.	More accurate analysis.
Credit for IC inventory for RPV analysis.	More realistic analysis. Tested in GIRAFFE/SITs.
Credit for SLC system inventory for RPV analysis following DPV actuation.	Additional source of water driven by accumulators. Inventory is discharged into core bypass region similar to BWR5/6 LPCI.
Included FW mass outside of containment in LOCA evaluation model of FW line break.	Additional mass and energy considered to maximize containment pressure.
FW line break limiting for RPV level.	Minimum water levels for FWL, GDCS line and BDL breaks are close, because of larger initial water level drop and earlier depressurization. FWLB is a break in the downcomer similar to GDCS line break, but at a higher elevation. No new phenomena are introduced.
FW line break is limiting for containment pressure.	Direct consequence of large inventory addition from FW line and FW heaters. FW line inventory was discussed in earlier reviews and left as an item to be resolved at DCD. A conservative estimate is used in the DCD analysis.
Peak pressure is reached in the early part of the transient for FW line break.	Peak pressure is reached early because of large discharge from upstream side of FWLB. Significant margin is still maintained to design limits. TRACG qualification shows early pressure rise is calculated conservatively. Results consistent with M3CPT.

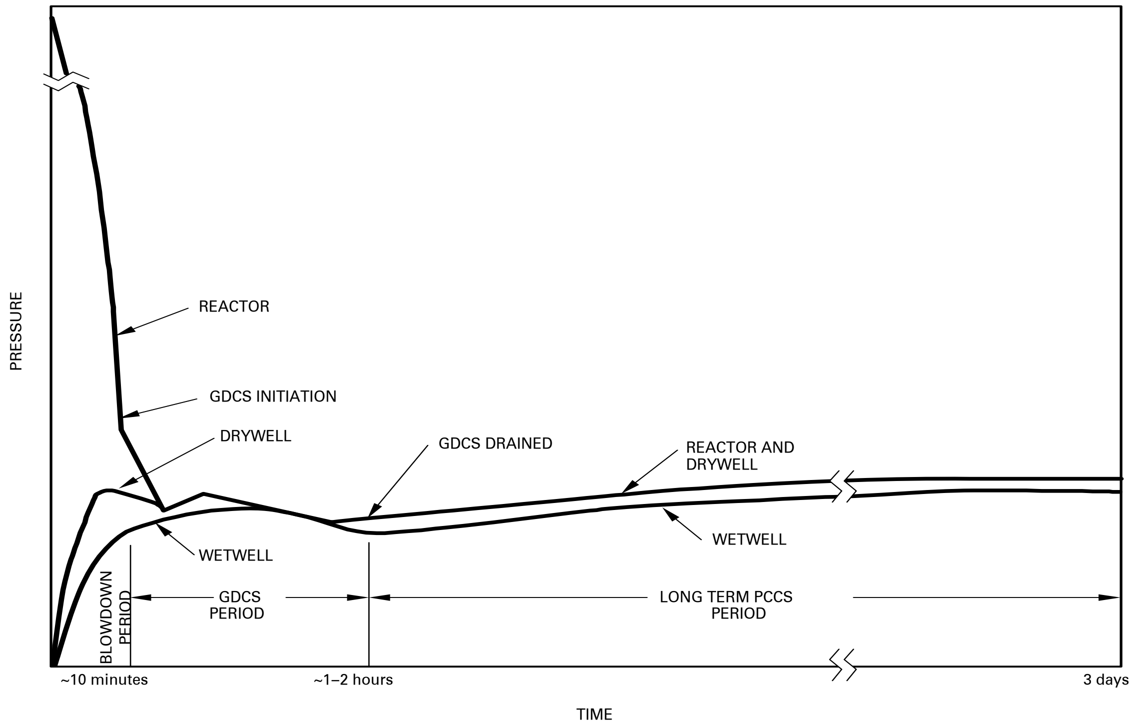


Figure 6G-1. Phases of the LOCA Transient

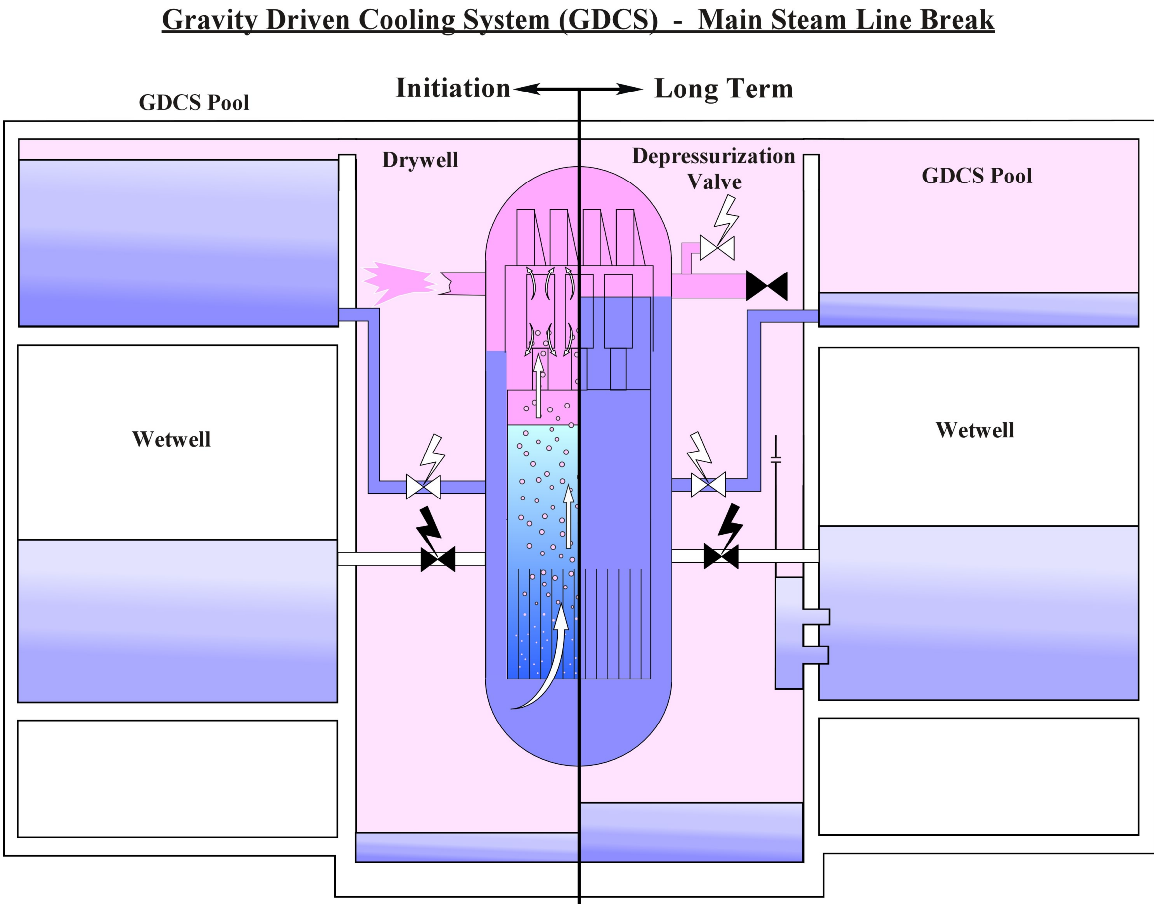
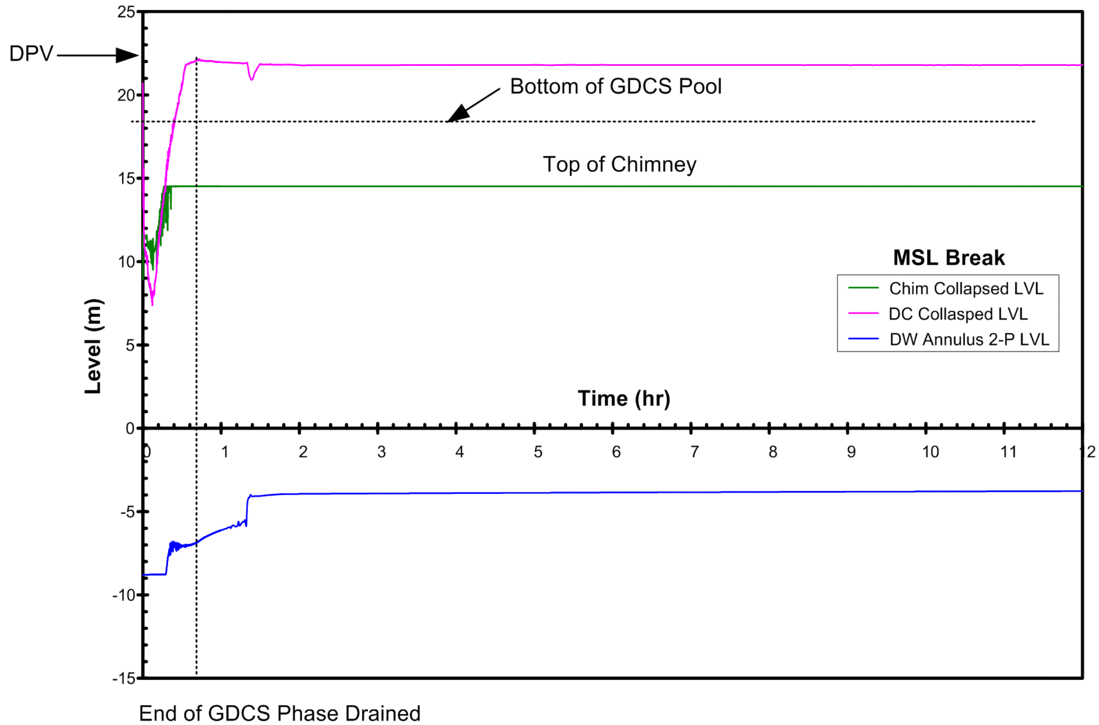
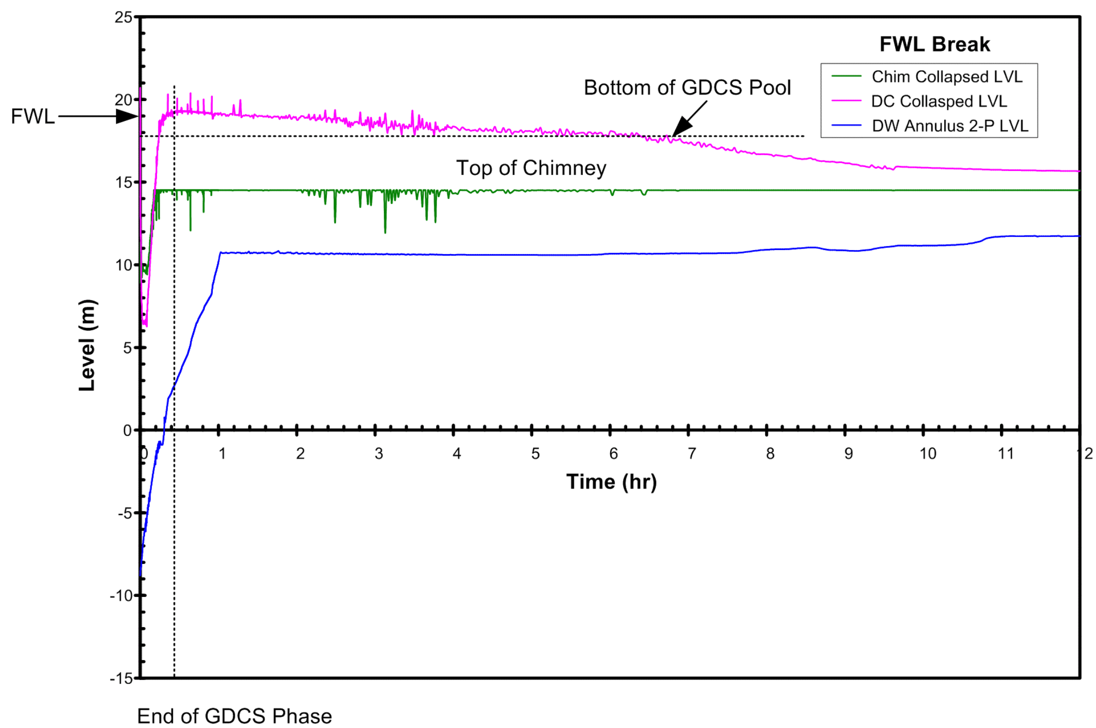


Figure 6G-2. Inventory Distributions for Main Steam Line Break



LEGEND: LVL = Level
DC = Downcomer

Figure 6G-3. RPV and Drywell Water Levels for MSLB (12 hours)



LEGEND: LVL = Level
DC = Downcomer

Figure 6G-4. RPV and Drywell Water Levels for FWLB (12 hours)

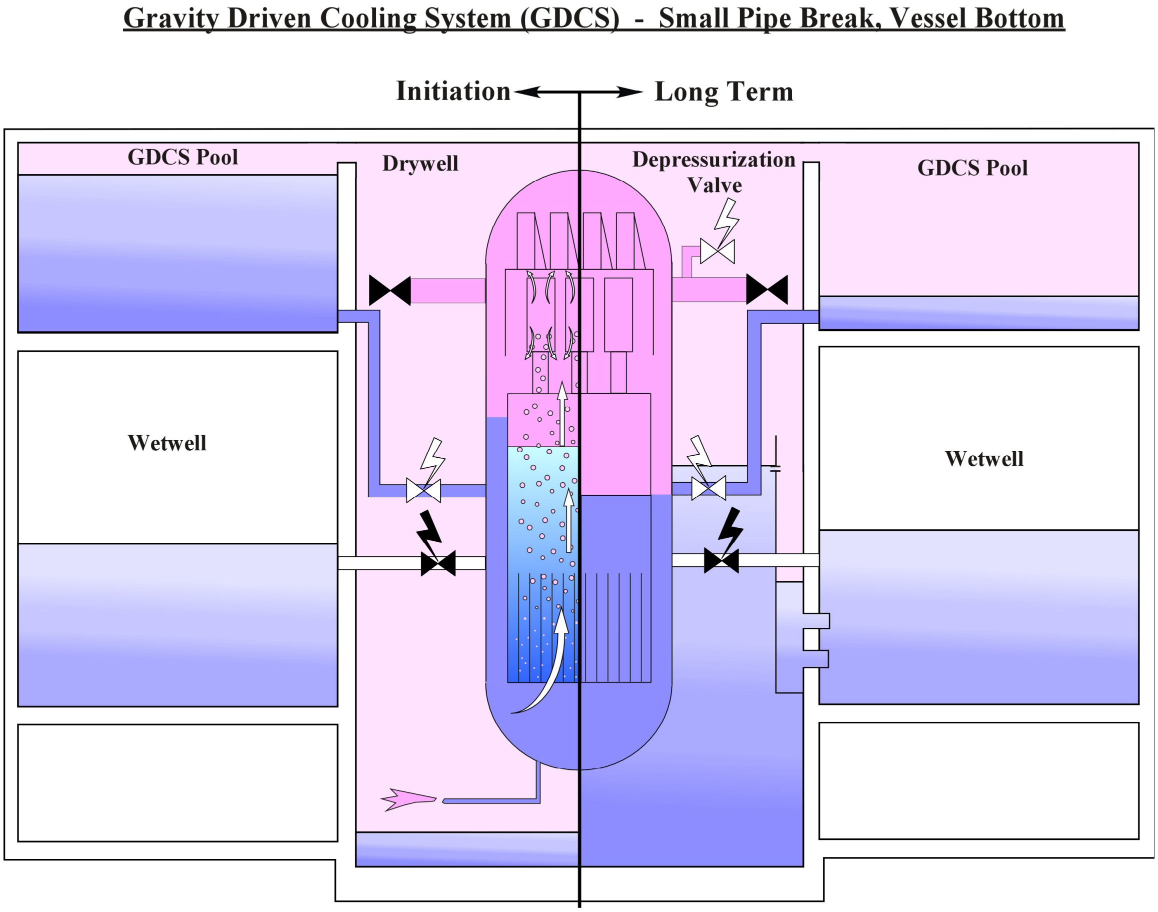
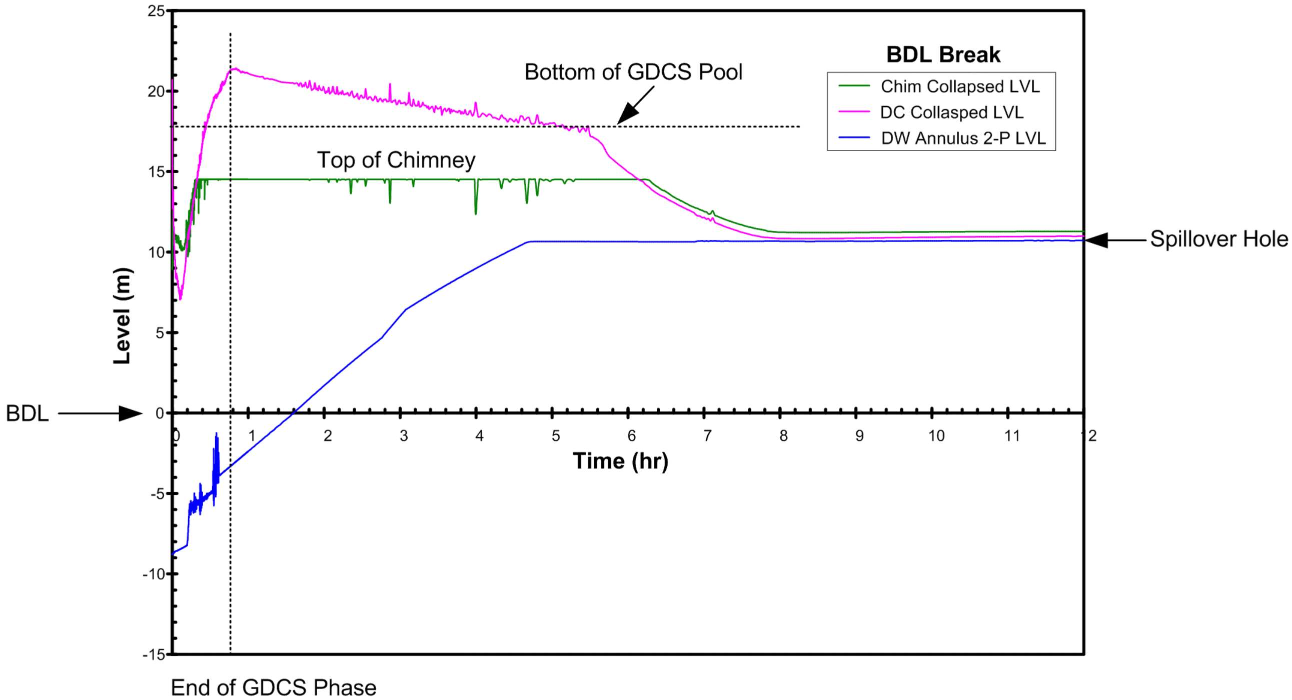
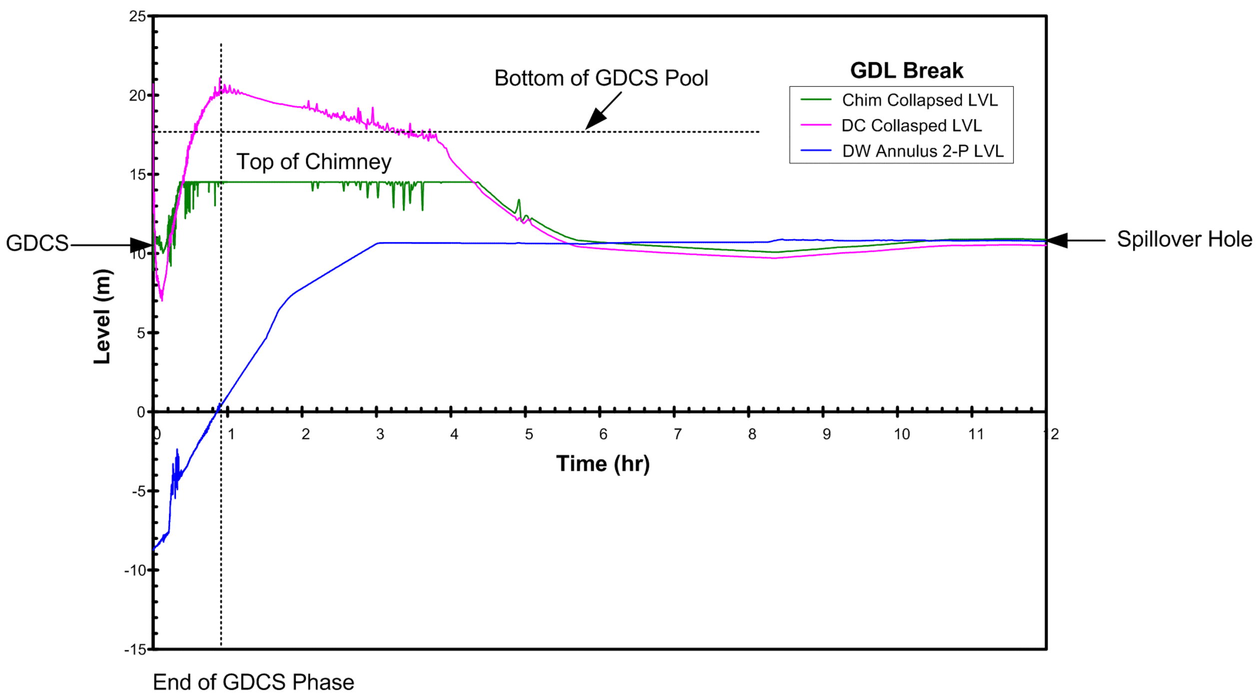


Figure 6G-5. Inventory Distributions for Bottom Drain Line Break



LEGEND: LVL = Level
DC = Downcomer

Figure 6G-6. RPV and Drywell Water Levels for BDL Break (12 hours)



LEGEND: LVL = Level
DC = Downcomer

Figure 6G-7. RPV and Drywell Water Levels for GDCS Line Break (12 hours)

6H. ADDITIONAL TRACG OUTPUTS AND PARAMETRIC CASES

This appendix discusses the limiting DBA of the main steam line break base case that assumes a single failure of 1 DPV and bounding conditions and 100% double-ended guillotine break. This case is referenced as Case A in the following sections, and was documented in DCD Tier 2, Rev. 3, Subsection 6.2.1.1.3.5 and Table 6.2-6, DCD Tier 2 Rev. 3. Subsequently, additional TRACG outputs for this limiting break case, e.g., the transient air mass profiles in different regions, were generated. Also, additional parametric cases were performed to evaluate the impact of various model/plant parameters on the long-term containment pressure. This Appendix summarizes these additional TRACG outputs for the limiting break case and the results from the additional parametric studies. Section 6H.1 provides the additional TRACG outputs for the limiting DBA break case. These are the transient air mass profiles in different regions. Section 6H.2 discusses the results from the additional parametric studies, performed to evaluate the impact of various model/plant parameters on the long-term containment pressure.

Section 6H.3 discusses the impact of operation at less than 100% power on the containment conditions, with DBA conditions resulting in maximum calculated containment pressure or temperature.

6H.1 Transient Air Mass Profiles for the Limiting DBA (Main Steam Line) Break Case

This section provides the additional TRACG outputs for this limiting break case, e.g., the transient air mass profiles in different regions. Figures 6H-1 to 6H-3 show the air mass profiles in the GDCS, DW head and WW airspaces. After 20 hours into the transient, all the air mass in the GDCS and DW head airspaces is essentially purged and transferred into the WW airspace. After that time, the air mass in the WW airspace continues to increase gradually due to the generation of radiolytic gases in the core during the transient.

6H.2 Description of Parametric Cases on the Main Steam Line Break

Additional parametric cases were performed to evaluate the impact of various model/plant parameters on the long-term containment pressure. This section summarizes the additional TRACG outputs for the limiting break case and the results of the additional parametric studies.

Table 6H-1 summarizes the eight cases that are discussed and compared with the base case. The base case is the limiting DBA of the main steam line break accident (DCD Tier 2, Rev. 3, Subsection 6.2.1.1.3.5). The parametric cases (E through L) use the same nodalization and conditions as those used in the base case, except the parameters that are noted in the third column in the table.

The following paragraphs discuss the results of these parametric cases.

6H.2.1 Effect of Wetwell Stratification (Case A versus Case E)

The base Case A (with bounding conditions) assumes stratification in the top level of the WW airspace due to vacuum breaker leakage (DCD Tier 2, Rev. 3, Table 6A-1, Item 5). The parametric Case E turns off the stratification model in the WW. Figure 6H-4 compares the DW pressures from these two cases. Without the WW stratification, the calculated peak DW pressure is 10.39 kPa (1.507 psia) lower than that for the base case at 72 hours.

6H.2.2 Effect of Suppression Pool Stratification (Case A versus Case F)

The base Case A (with bounding conditions) assumes stratification in the suppression pool in the region above the highest source of mass and energy to the pool (DCD Tier 2, Rev. 3, Table 6A-1, Item 4). The parametric Case F turns off the stratification model in the suppression pool. Figure 6H-5 compares the DW pressures from these two cases. Without the suppression pool stratification, the calculated peak DW pressure is 20.79 kPa (3.015 psia) lower than that for the base case at 72 hours.

6H.2.3 Effect of IC Heat Transfer (Case A versus Case G)

The base Case A (with bounding conditions) assumes no credit for the heat transfer in the ICs (DCD Tier 2, Rev. 3, Table 6A-1, Item 19). The parametric Case G takes credit for the heat transfer in the ICs. Figure 6H-6 compares the DW pressures from these two cases. With the credit for the heat transfer in the ICs, the calculated peak DW pressure is 0.37 kPa (0.054 psia) lower than that for the base case at 72 hours.

6H.2.4 Effect of Single Failure: 1 DPV versus 1 SRV (Case A versus Case H)

The base Case A (with bounding conditions) assumes a single failure of 1 DPV. The parametric Case H assumes a single failure of 1 SRV. Figure 6H-7 compares the DW pressures from these two cases. The calculated peak DW pressure for the case with a single failure of 1 SRV is 0.79 kPa (0.11 psia) lower than that for the base case at 72 hours.

6H.2.5 Effect of Containment Outer wall Heat Transfer Area (Case A versus Cases I and J)

The parametric cases decrease the containment outer wall (in the WW airspace and suppression regions) heat transfer area by 10% (Case I) and 25% (Case J). Figures 6H-8 and 6H-9 compare the DW pressures from these cases with that from the base case. With 10% reduction in the outer wall heat transfer area, the calculated peak DW pressure at 72 hours is 1.07 kPa (0.155 psia) higher than that for the base case. With 25% reduction in the outer wall heat transfer area, the calculated peak DW pressure at 72 hours is 6.69 kPa (0.970 psia) higher than that for the base case. The increase in the calculated peak DW pressure at 72 hours is small compared to the margin to the design pressure.

6H.2.6 Effect of Containment Inner wall Heat Transfer Area (Case A versus Case K)

The parametric case increases the containment inner wall (in the vent wall between the DW and the WW) heat transfer area by 25% (Case K). Figure 6H-10 compares the DW pressures from this case with that from the base case. With 25% increase in the inner wall heat transfer area, the calculated peak DW pressure at 72 hours is 3.64 kPa (0.528 psia) lower than that for the base case. The decrease in the calculated peak DW pressure is small compared to the margin to the design pressure.

6H.2.7 Effect of Noncondensable Gases: Air versus Nitrogen (Case A versus Case L)

The base Case A (with bounding conditions) uses air properties for the noncondensable gases inside the containment (DCD Tier 2, Rev. 3, Table 6A-1, Item 15). The parametric Case L uses nitrogen properties for the noncondensable gases inside the containment. Figure 6H-11 compares the DW pressures from these two cases. The difference in the calculated peak DW pressure at 72 hours is small (~ 0.53 kPa (0.077 psia)) compared to the margin to the design pressure.

6H.2.8 Summary of Results from the Parametric Cases on the Main Steam Line Break

Eight additional parametric cases were performed to evaluate the impact of various model/plant parameters on the long-term containment pressure. Table 6H-1 describes the parameters used in these eight parametric cases. This table also summarizes the calculated peak DW pressures at 72 hours and the comparison to the base case.

Results from these parametric cases show the following:

(1) The bounding models (WW stratification and suppression pool stratification) are conservative. The calculated DW pressures are reduced by 10 to 20 kPa (1.5 to 2.9 psia) without these models, or the margins to the design pressure are improved by 3 to 6.5%.

(2) The calculated long-term DW pressure is not sensitive to the credit of IC heat transfer, the assumption of single failure (1 DPV vs. 1 SRV), or the assumption of noncondensable gas properties (air vs. nitrogen).

(3) The effect of the containment wall heat transfer areas on the calculated long-term DW pressure is small. For +/- 25% wall areas, the impact on the margin is small (-2% to +1%) compared to the base value of 9.4%.

6H.3 Effect of Operating at Less than 100% Power on the Containment Conditions

The suppression pool average temperature during normal operation is < 110 °F, and the maximum pool temperature of 110 °F is used in the safety analyses. Per Technical Specifications the reactor is required to reduce thermal power to ≤ 1% of rated thermal power when the suppression pool temperature > 110 °F, and the reactor will be switched to shutdown mode immediately when the suppression pool temperature > 120 °F.

For operation at less than 100% power, the integrated blowdown energy (the decay heat is the key contributor) from the primary system to the containment during a DBA will be smaller comparing to the case operating at 100% power.

For the feedwater line break, the maximum drywell pressure occurs at about 40 seconds during the blowdown period (Figure 6.2-9a2). Smaller integrated blowdown energy is expected to result in lower peak drywell pressure.

The long-term containment pressure depends on the wetwell partial steam pressure, which depends on the amount of blowdown energy that is discharged into the suppression pool. Figure 6.2-9c1 compares the total PCC heat removal versus the decay heat. For the first ~6 hours of the transient, the decay heat is greater than the PCC removal power. The amount that is not removed by the PCC will end up in the suppression pool and will increase the pool temperature accordingly. For operation at less than 100% power, the decay heat is lower and the equilibrium point between the decay heat and the PCC removal power will occur sooner. Consequently, there will be less heatup on the pool water, resulting in lower drywell pressure. Accordingly, it is expected that the DBA conditions will result in the maximum calculated containment pressure or temperature.

6H.4 References

None.

Table 6H-1

Summary of Parametric Cases on the Main Steam Line Break

Case #	Case ID	Comment	Calculated Peak DW Pressure(1) kPa (psia)	P_{DW} Difference (Parametric – Base) kPa (psia)	Margin to 45.3 psig (%)
Base Case					
A	MSL3_1DPVCB _NL2Pa-72	DCD Tier 2, Rev. 3, Subsection 6.2.1.1.3.5	384.18 (55.721)	0.00 (0.0)	9.4
Parametric Cases					
E	MSL3_1DPVCB _NL2P_NSTR-72	Turn-off WW stratification	373.79 (54.214)	-10.39 (-1.507)	12.8
F	MSL3_1DPVCB _NL2Pb-72	Turn-off suppression pool stratification	363.39 (52.705)	-20.79 (-3.015)	16.1
G	MSL3_1DPVCB _NL2PIC-72	Turn-on IC heat transfer	383.81 (55.667)	-0.37 (-0.05)	9.6
H	MSL3_1SRVCB _NL2P-72	MSL break with failure of 1 SRV	383.39 (55.606)	-0.79 (-0.11)	9.7
I	MSL3_1DPVCB _NL2P_M10-72	Decrease containment outer wall heat transfer area by 10%	385.25 (55.876)	+1.07 (+0.155)	9.1
J	MSL3_1DPVCB _NL2P_M-72	Decrease containment outer wall heat transfer area by 25%	390.87 (56.691)	+6.69 (+0.970)	7.3
K	MSL3_1DPVCB _NL2P_P-72	Increase vent wall (DW-WW) heat transfer area by 25%	380.54 (55.193)	-3.64 (-0.528)	10.6
L	MSL3_1DPVCB _NL2P_N2-72	Change noncondensable gas from air (in the base case) to nitrogen (N2)	384.71 (55.797)	+0.53 (+0.77)	9.3

(1) The peak DW pressure calculated during the transient period from 0 to 72 hrs.

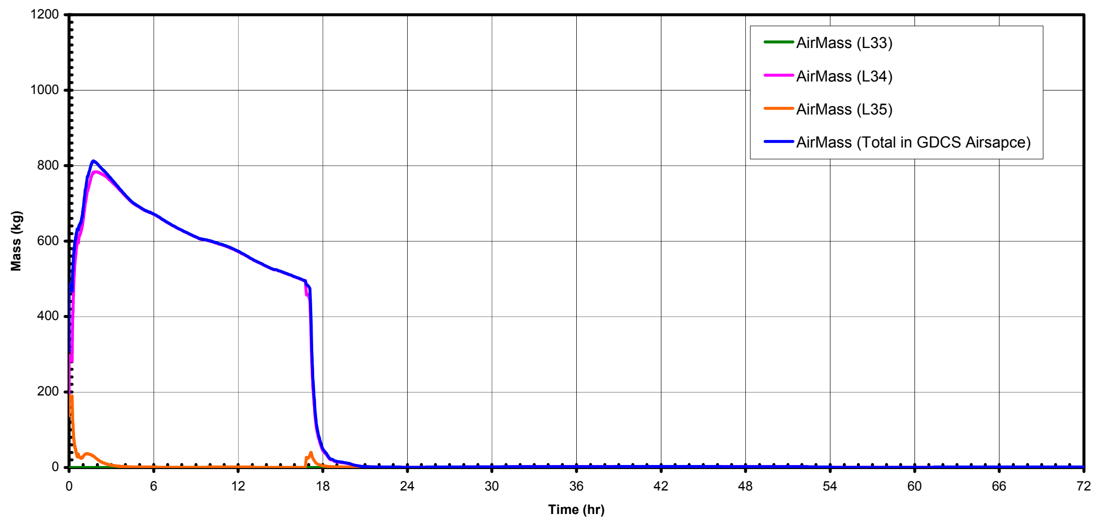


Figure 6H-1. Air Mass Profiles in the GDCS Airspace (Case A: MSL3_1DPVCB_NL2Pa-72)

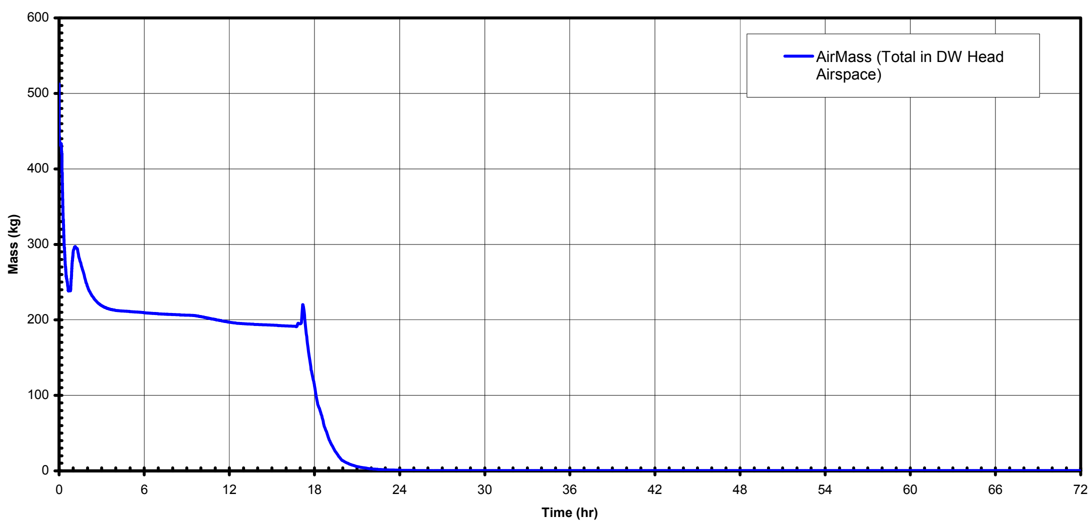


Figure 6H-2. Air Mass Profile in the Drywell Head Airspace (Case A: MSL3_1DPVCB_NL2Pa-72)

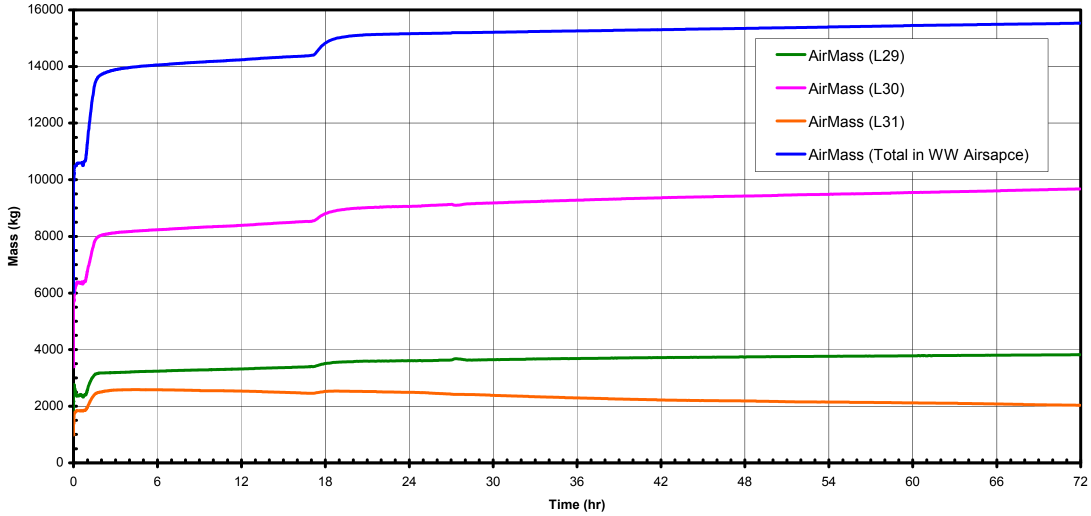


Figure 6H-3. Air Mass Profiles in the Wetwell Airspace (Case A: MSL3_1DPVCB_NL2Pa-72)

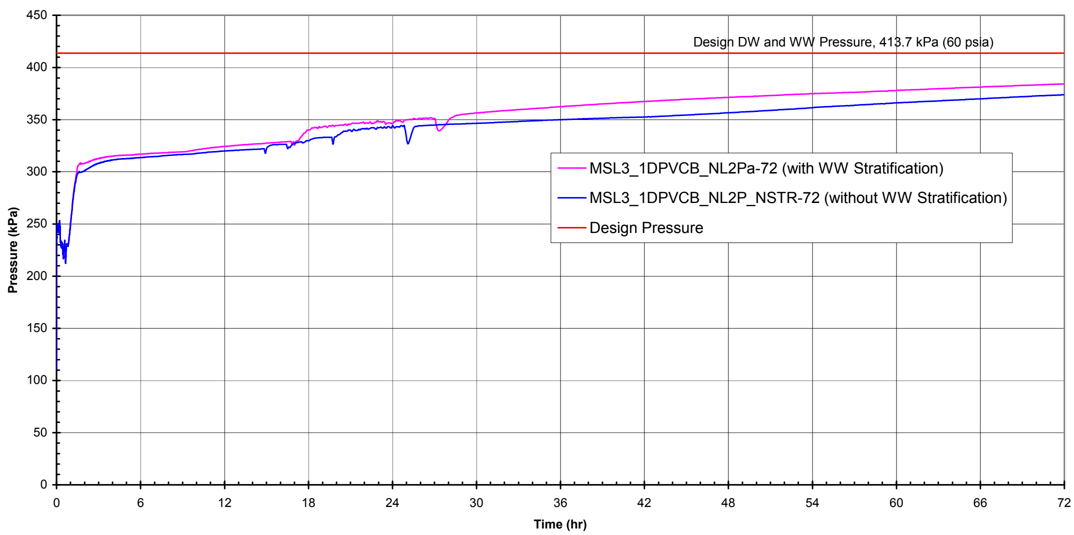


Figure 6H-4. Comparison of DW Pressures (Case A vs. Case E: Effect of WW Stratification)

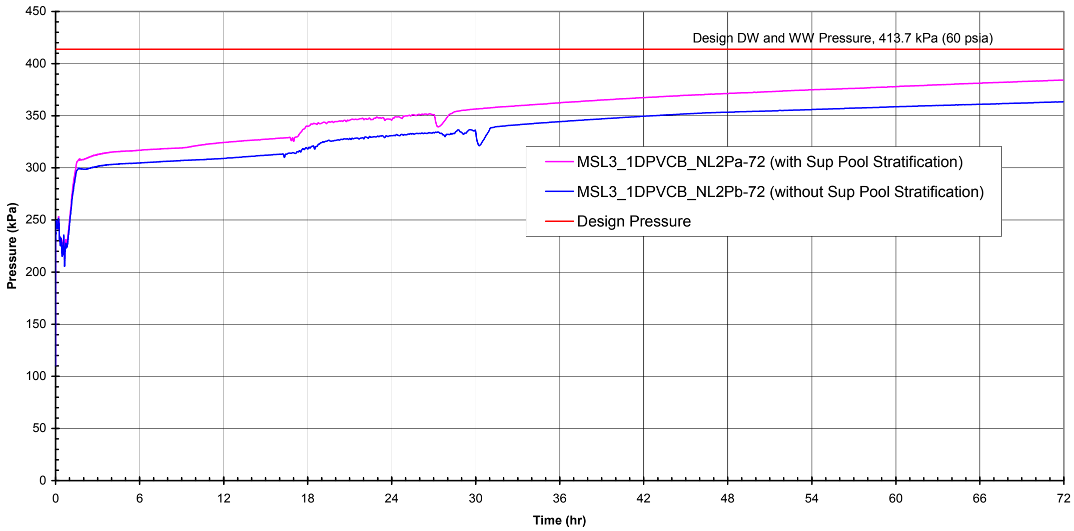


Figure 6H-5. Comparison of DW Pressures
(Case A vs. Case F: Effect of Suppression Pool Stratification)

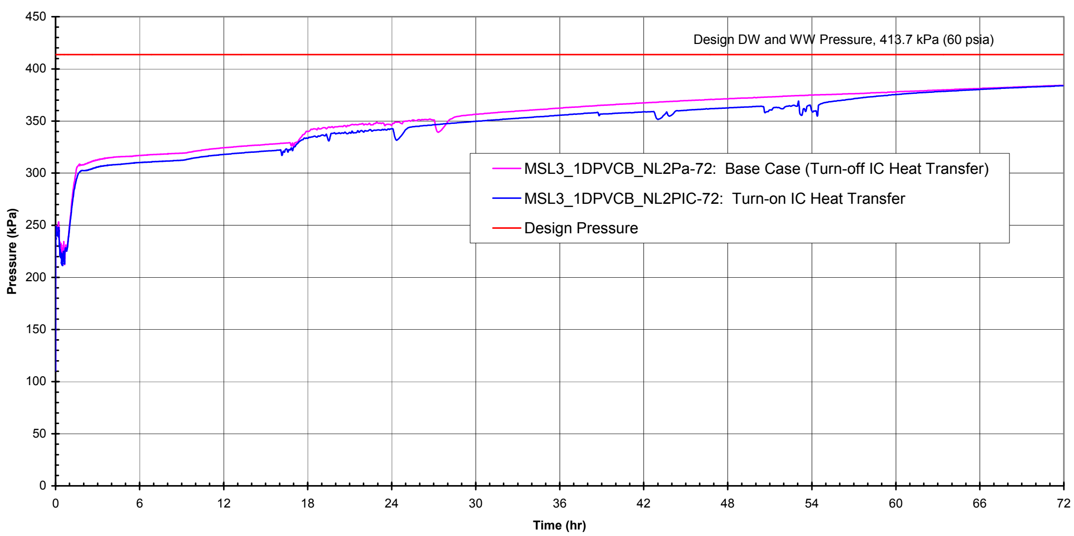


Figure 6H-6. Comparison of DW Pressures
(Case A vs. Case G: Effect of IC Heat Transfer)

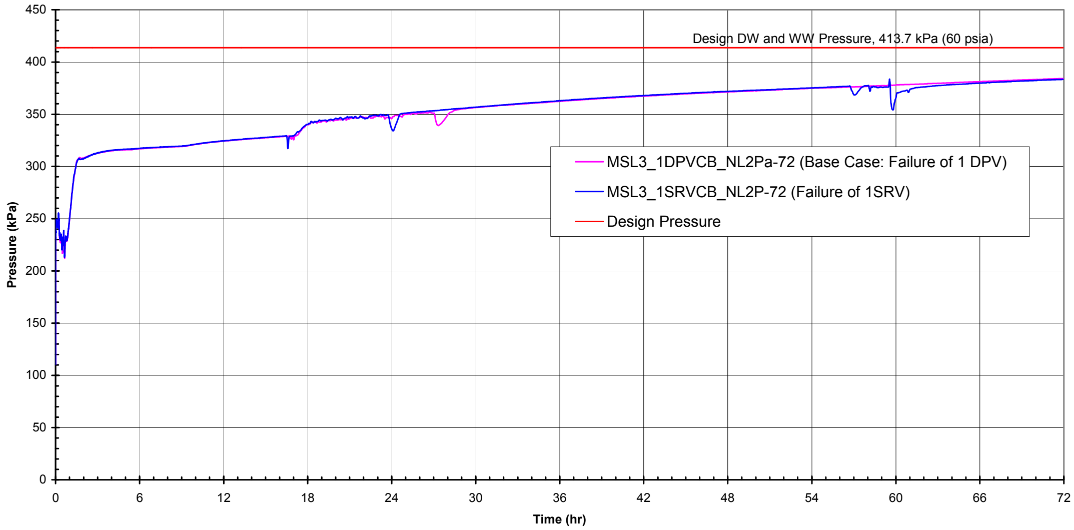


Figure 6H-7. Comparison of DW Pressures
(Case A vs. Case H: Effect of Single Failure, 1 DPV versus 1 SRV)

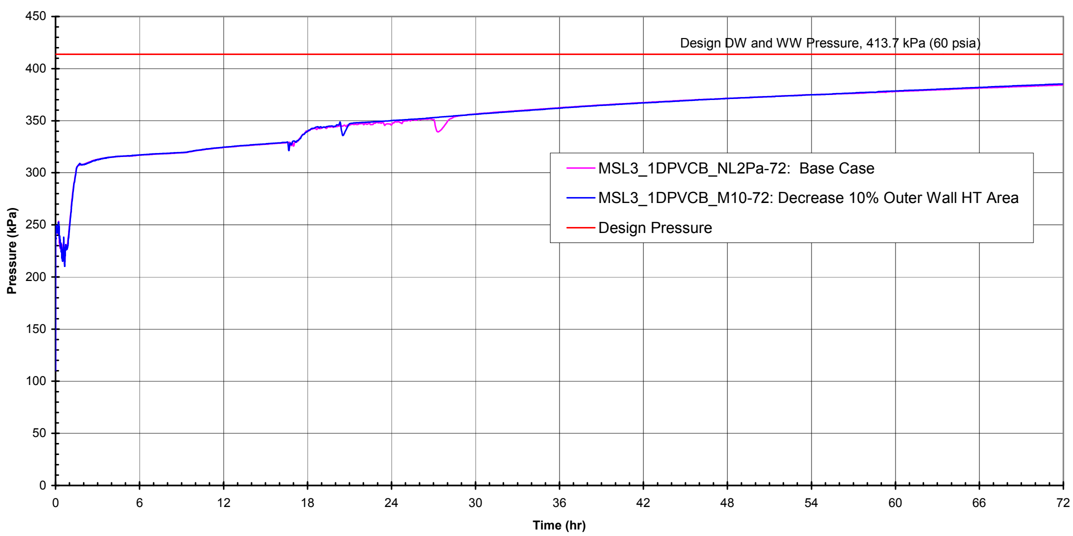


Figure 6H-8. Comparison of DW Pressures
(Case A vs. Case I: Effect of Outer Wall Heat Transfer Area –10%)

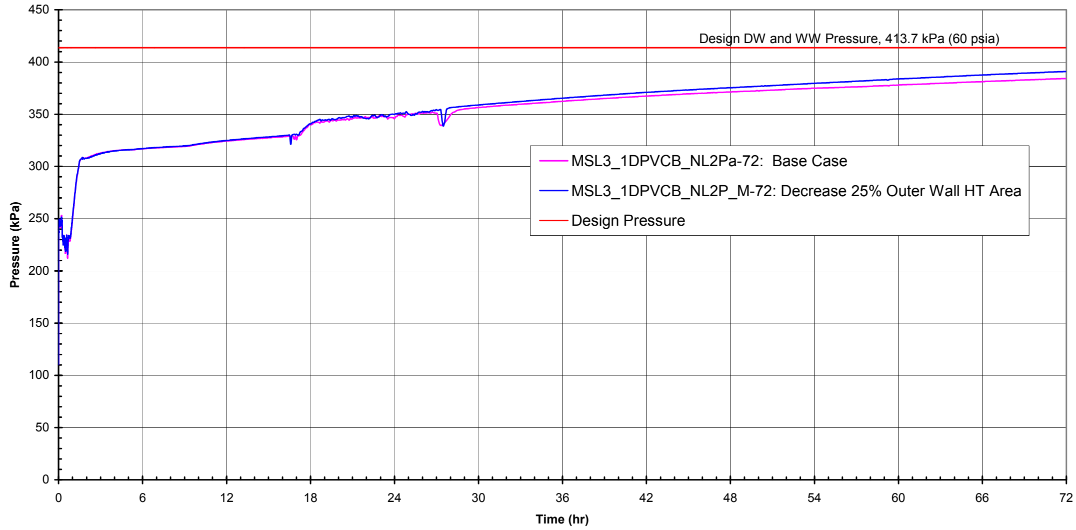


Figure 6H-9. Comparison of DW Pressures (Case A vs. Case J: Effect of Outer Wall Heat Transfer Area –25%)

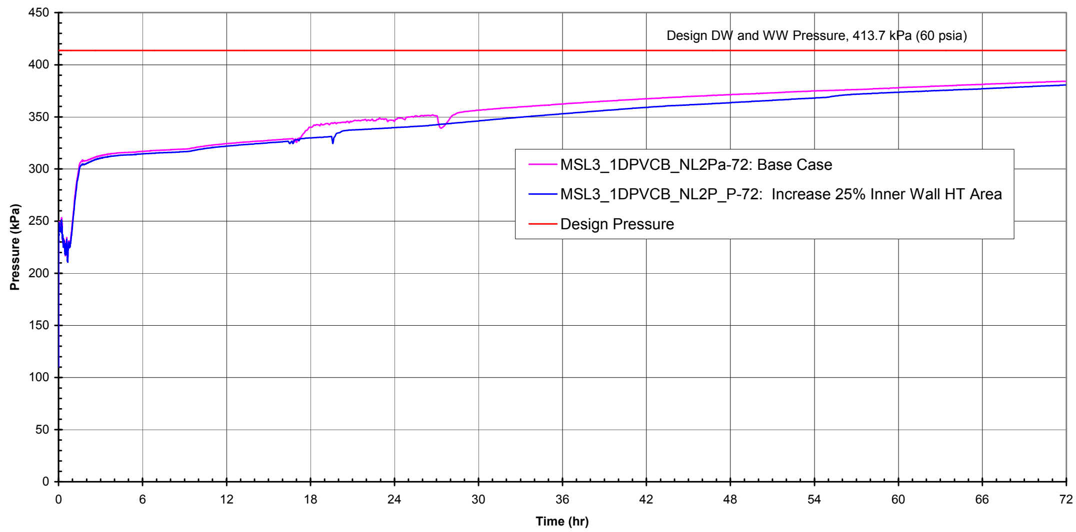


Figure 6H-10. Comparison of DW Pressures (Case A vs. Case K: Effect of Inner Wall Heat Transfer Area +25%)

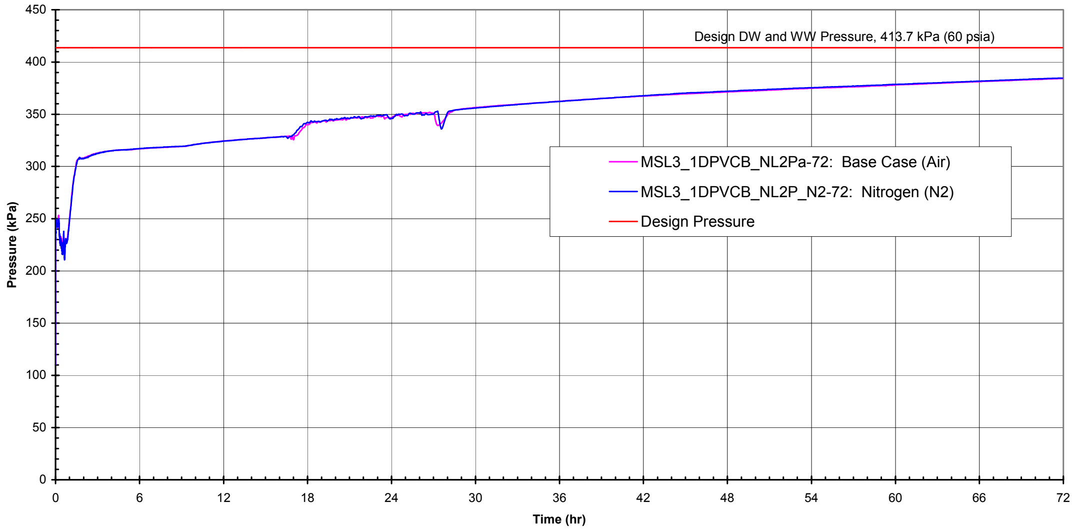


Figure 6H-11. Comparison of DW Pressures (Case A vs. Case L: Effect of Noncondensable Gases, Air versus Nitrogen)

6I. RESULTS OF THE CONTAINMENT DESIGN BASIS CALCULATIONS WITH SUPPRESSION POOL BYPASS LEAKAGE ASSUMPTION OF 1 CM^2 (1.08 E-03 FT^2)

This appendix presents the results of the containment design basis calculations with suppression pool bypass leakage assumption of 1 cm^2 (1.08 E-03 ft^2). This ties the results presented in Section 6.2 with 2 cm^2 (2.16 E-03 ft^2) DW-to-WW bypass leakage to 1 cm^2 (1.08 E-03 ft^2) calculations on the same bases. The Appendix 6I 1 cm^2 (1.08 E-03 ft^2) cases in turn tie-back to previous documented ESBWR results (previous to Revision 5). The 1 cm^2 (1.08 E-03 ft^2) results also provide a sensitivity to a more realistic DW-to-WW bypass leakage of 1 cm^2 (1.08 E-03 ft^2).

Two 1 cm^2 (1.08 E-03 ft^2) cases of main steam line break are provided, one with 1 DPV failure and the other one with 1 SRV failure, with the identical design changes implemented in Revision 5 design basis calculations except for the suppression pool bypass leakage of 1 cm^2 (1.08 E-03 ft^2). Table 6I-1 summarizes the results of these calculations. Sequence of events for the analyzed cases are presented in Table 6I-2 and Table 6I-3. Figure 6I-1a1 through Figure 6I-2d3 show the containment pressure, temperature, DW and GDCS airspace pressure responses and PCCS heat removal.

Table 6I-1

Summary of Containment-LOCA Performance Analyses

Break Location	Break Size[1] m^2 (ft^2)	Single Failure	Maximum DW Pressure[3] kPa (psia)	Maximum DW Pressure[3] kPaG (psig)	Margin[4] to Design Pressure of 310 kPaG (45 psig) (%)	Short-term Bulk DW Temperature ºC (ºF)	Long-term Bulk DW Temperature ºC (ºF)	Long-term WW Temperature ºC (ºF)	Long-term Suppression Pool Temperature ºC (ºF)
Based on bounding values:									
Steam Line Inside Containment[2]	0.09832 (1.058)	1 DPV	374.56 (54.33)	273.21 (39.63)	11.9%	173.6 (344.5)	141.3 (286.3)	115.24 (239.43)	73.36 (164.05)
Steam Line Inside Containment[2]	0.09832 (1.058)	1 SRV	373.47 (54.17)	272.12 (39.47)	12.3%	173.6 (344.5)	141.1 (286.0)	114.98 (238.96)	73.10 (163.59)

(1) The break area is from the RPV side of the break.

(2) Main Steam Line Break, at Level 34, 2 GDCS vent paths. The break area from the RPV side of the break is limited by the MSL nozzle, which has a flow area of 0.09832 m^2 (1.05831 ft^2).

(3) Maximum DW pressure calculated during the 72 hours following a LOCA.

(4) Minimum pressure margin calculated during the 72 hours following a LOCA.

Table 6I-2

Operational Sequence of ECCS for a MSLB with Failure of One DPV

(Bounding Case)

Time (s)	Events
0	Guillotine break of MSL 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 DW pressure setpoint reached. Scram signal from high DW pressure is not credited in this analysis.
2	Loss of normal auxiliary power confirmed. Reactor scram initiated. Rod insertion starts 0.25 seconds later.
Vent clearing time	Top Vent: 1.52 seconds, Middle Vent: 1.65 seconds, Bottom Vent: 2.11 seconds.
3	IC initiated from loss of power generation bus with 3 seconds signal delay time. Drain valves start to open 15 seconds later.
8	Low MSL pressure setpoint reached. MSIV closure initiated at 0.7 seconds later.
8	Level 3 is reached (scram signal from Level 3 is not credited in this analysis).
12	Reactor isolated on low MSL pressure setpoint.
20	Level 2 is reached (MSIV closure and IC initiation signals are not credited in this analysis).
490	Level 1 is reached.
500	Level 1 signal confirmed. ADS/GDCS/SLC system timer initiated. SRV actuated.
550	DPV actuation begins at 50 seconds after confirmed Level 1 signal. SLC system flow starts on DPV actuation.
650	GDCS timer (150 seconds after confirmed Level 1 signal) times out. GDCS injection valves open.
697	Vessel pressure decreases below maximum injection pressure of GDCS. GDCS flow into the vessel begins. Chimney and downcomer water levels start to rise.

Table 6I-2

Operational Sequence of ECCS for a MSLB with Failure of One DPV

(Bounding Case)

Time (s)	Events
851	SLC system flow depleted.
121200 (~ 33.7 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 IC/PCCS expansion pools.
From ~490 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 374.56 kPa (54.32 psia).

Table 6I-3

Operational Sequence of ECCS for a MSLB with Failure of One SRV

(Bounding Case)

Time (s)	Events
0	Guillotine break of MSL 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 DW pressure setpoint reached. Scram signal from high DW pressure is not credited in this analysis.
2	Loss of normal auxiliary power confirmed. Reactor scram initiated. Rod insertion starts 0.25 seconds later.
Vent clearing time	Top Vent: 1.52 seconds, Middle Vent: 1.57 seconds, Bottom Vent: 1.89 seconds
3	IC initiated from loss of power generation bus with 3 seconds signal delay time. Drain valves start to open 15 seconds later.
8	Low MSL pressure setpoint reached. MSIV closure initiated at 0.7 seconds later.
8	Level 3 is reached (scram signal from Level 3 is not credited in this analysis).
12	Reactor isolated on low MSL pressure setpoint.
20	Level 2 is reached (MSIV closure and IC initiation signals are not credited in this analysis).
491	Level 1 is reached.
501	Level 1 signal confirmed. ADS/GDCS/SLC system timer initiated. SRV actuated.
551	DPV actuation begins at 50 seconds after confirmed Level 1 signal. SLC system flow starts on DPV actuation.
651	GDCS timer (150 seconds after confirmed Level 1 signal) times out. GDCS injection valves open.
694	Vessel pressure decreases below maximum injection pressure of GDCS. GDCS flow into the vessel begins. Chimney and downcomer water levels start to rise.
851	SLC system flow depleted.

Table 6I-3

Operational Sequence of ECCS for a MSLB with Failure of One SRV

(Bounding Case)

Time (s)	Events
121430 (~ 33.7 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 IC/PCCS expansion 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 373.47 kPa (54.2 psia).

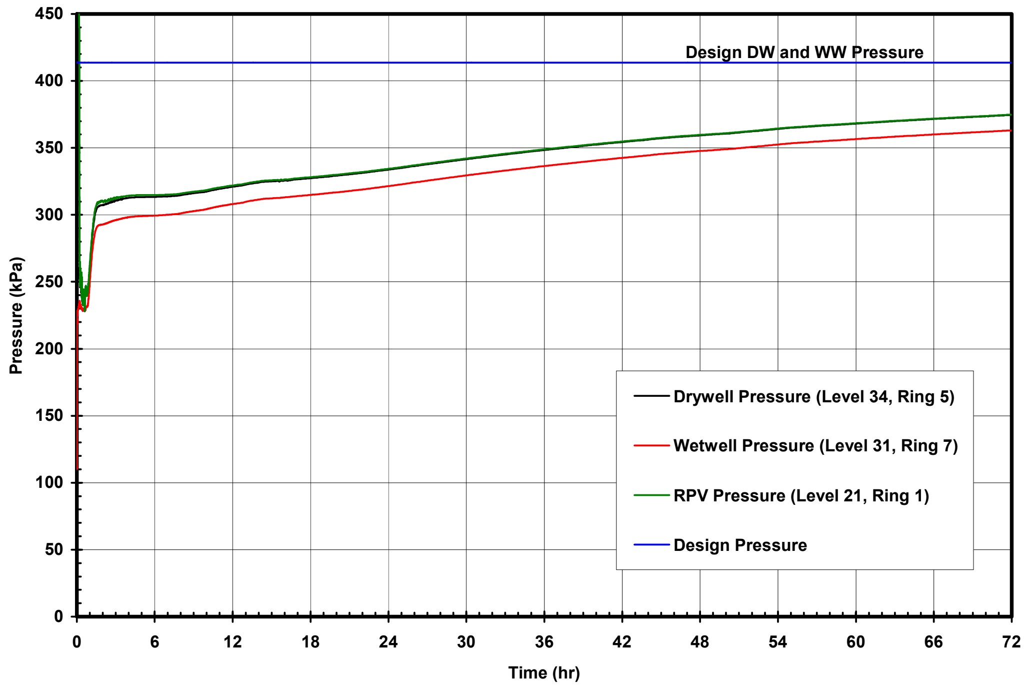


Figure 6I-1a1. Main Steam Line Break, 1 DPV Failure (Bounding Case) – Containment Pressures (72 hrs)

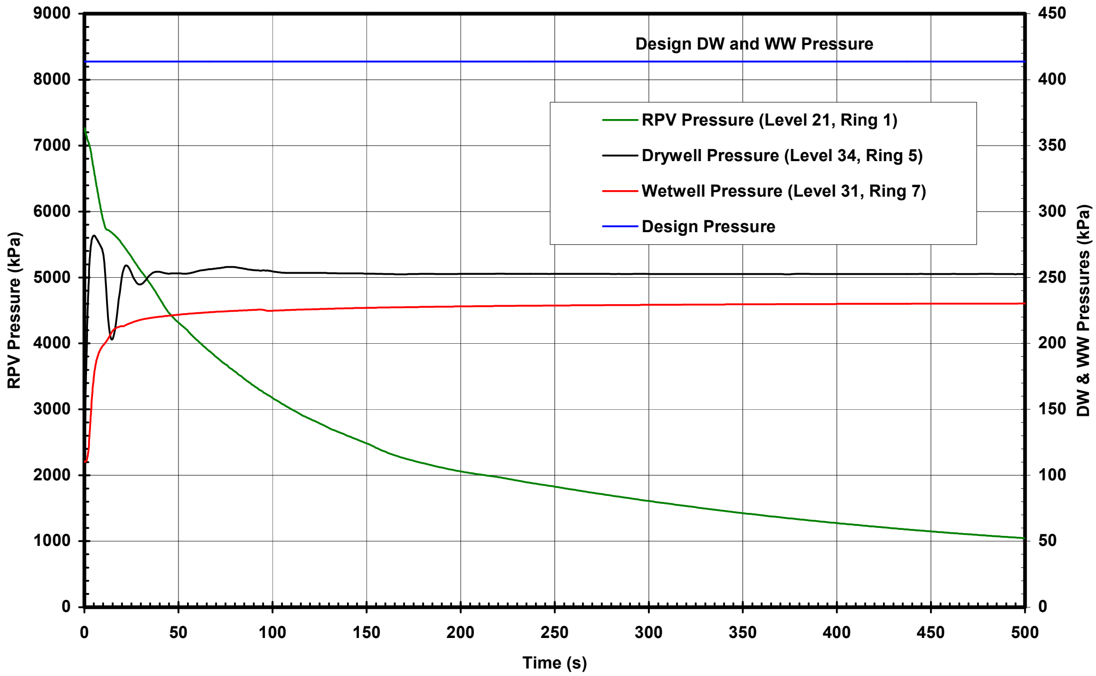


Figure 6I-1a2. Main Steam Line Break, 1 DPV Failure (Bounding Case) – Containment Pressures (500 s)

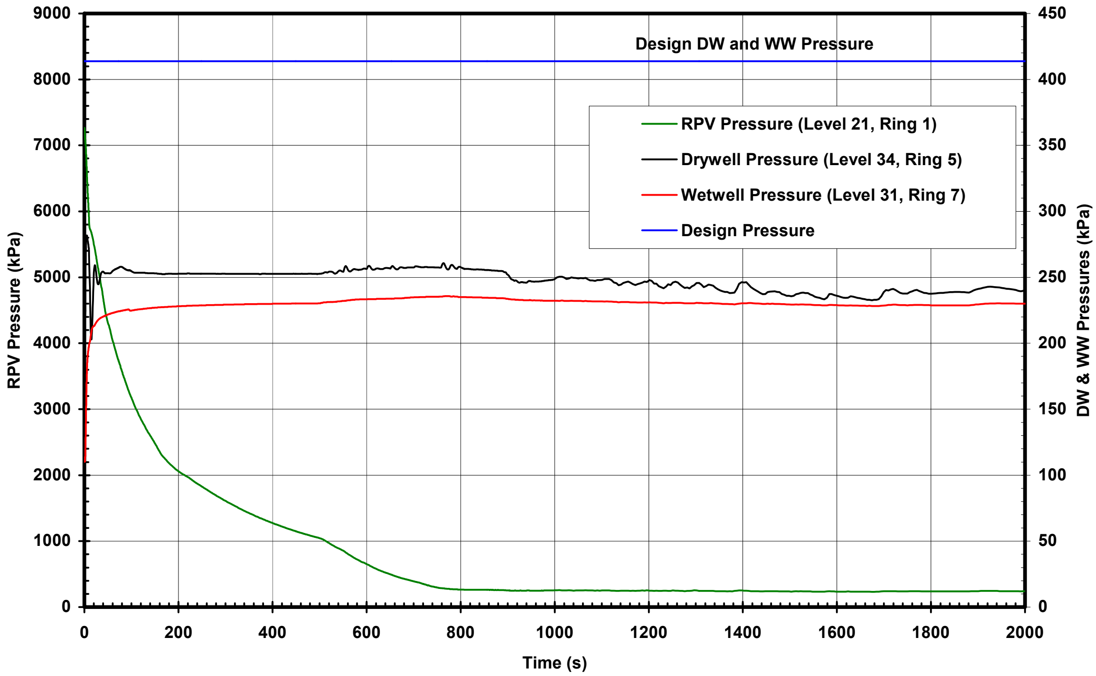
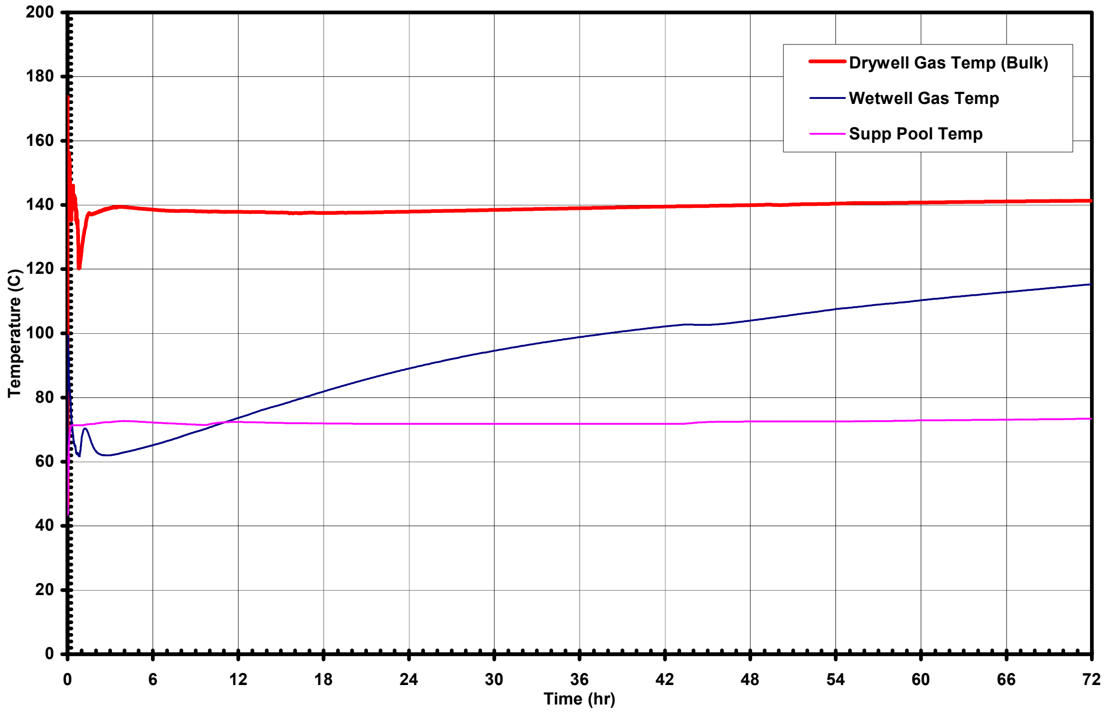
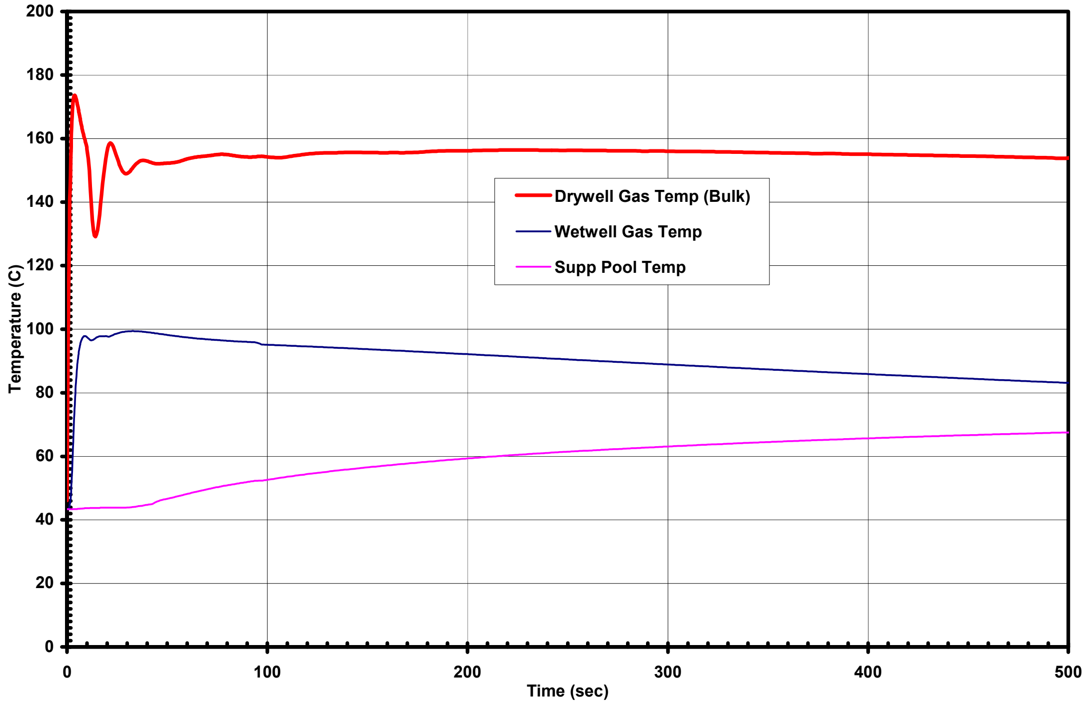


Figure 6I-1a3. Main Steam Line Break, 1 DPV Failure (Bounding Case) – Containment Pressures (2000 s)



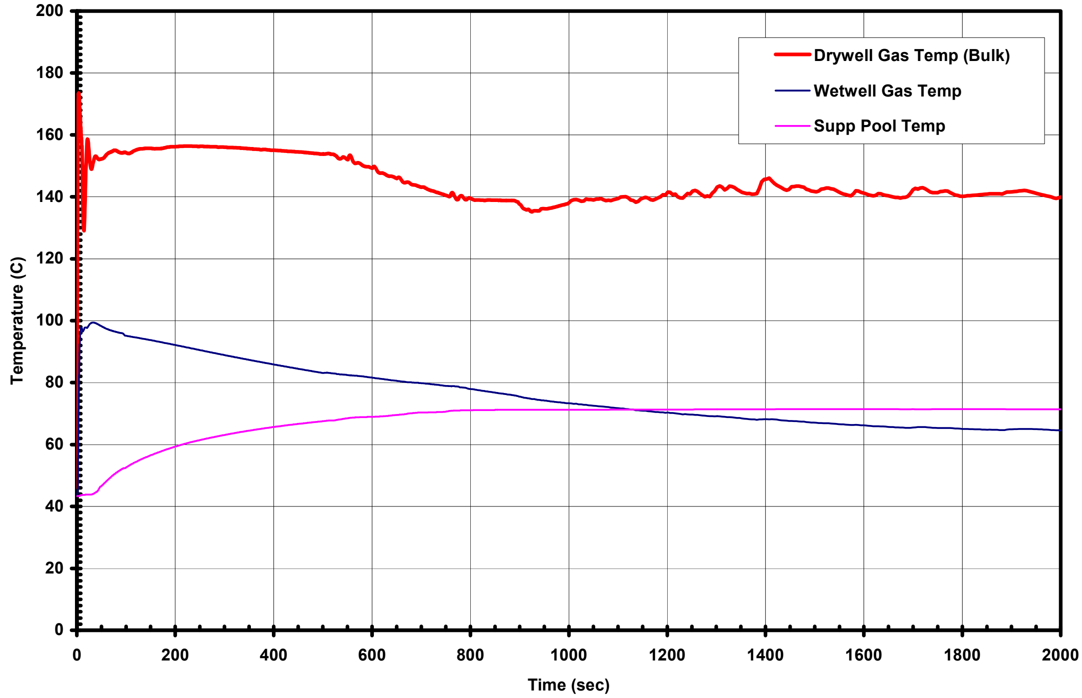
LEGEND: Supp = Suppression

Figure 6I-1b1. Main Steam Line Break, 1 DPV Failure (Bounding Case) – Containment Temperatures (72 hrs)



LEGEND: Supp = Suppression

Figure 6I-1b2. Main Steam Line Break, 1 DPV Failure (Bounding Case) – Containment Temperatures (500 s)



LEGEND: Supp = Suppression

Figure 6I-1b3. Main Steam Line Break, 1 DPV Failure (Bounding Case) – Containment Temperatures (2000 s)

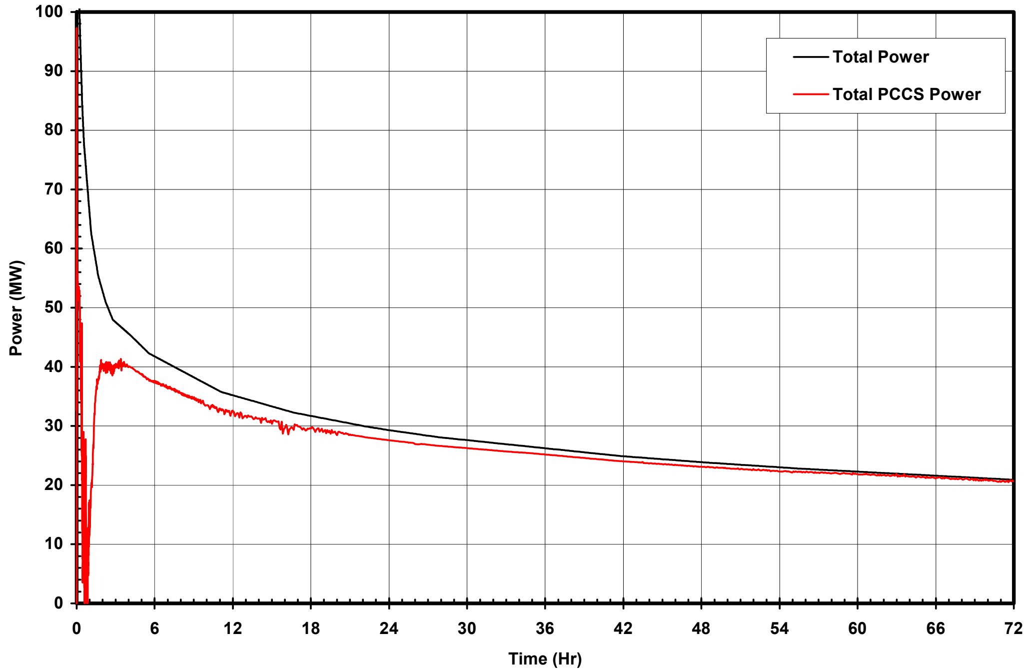


Figure 6I-1c1. Main Steam Line Break, 1 DPV Failure (Bounding Case) – PCCS Heat Removal versus Decay Heat (72 hrs)

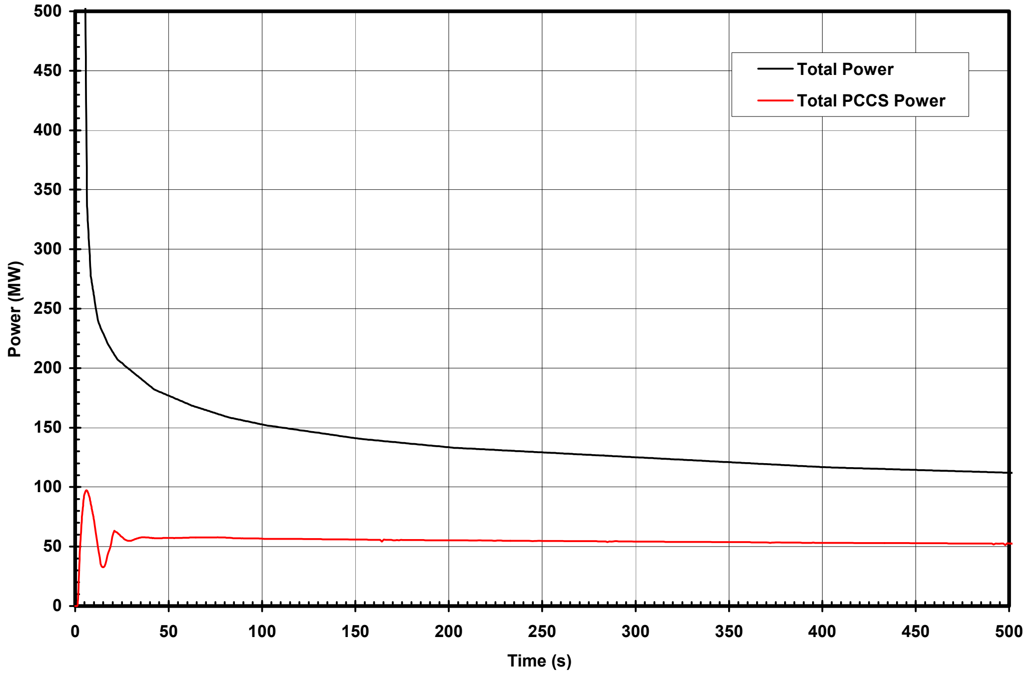


Figure 6I-1c2. Main Steam Line Break, 1 DPV Failure (Bounding Case) – PCCS Heat Removal versus Decay Heat (500 s)

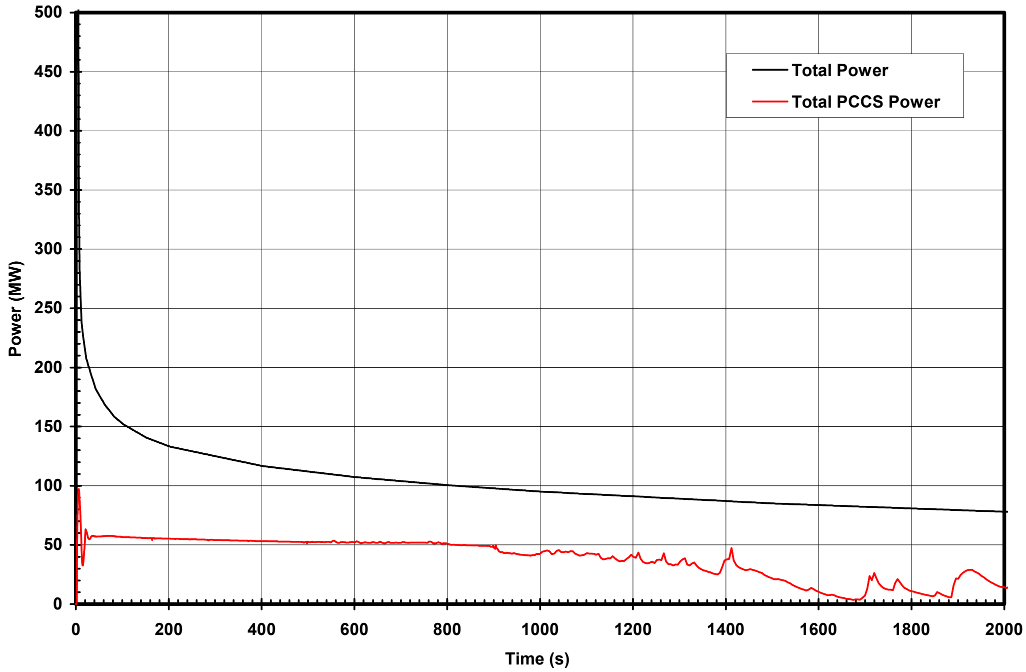


Figure 6I-1c3. Main Steam Line Break, 1 DPV Failure (Bounding Case) – PCCS Heat Removal versus Decay Heat (2000 s)

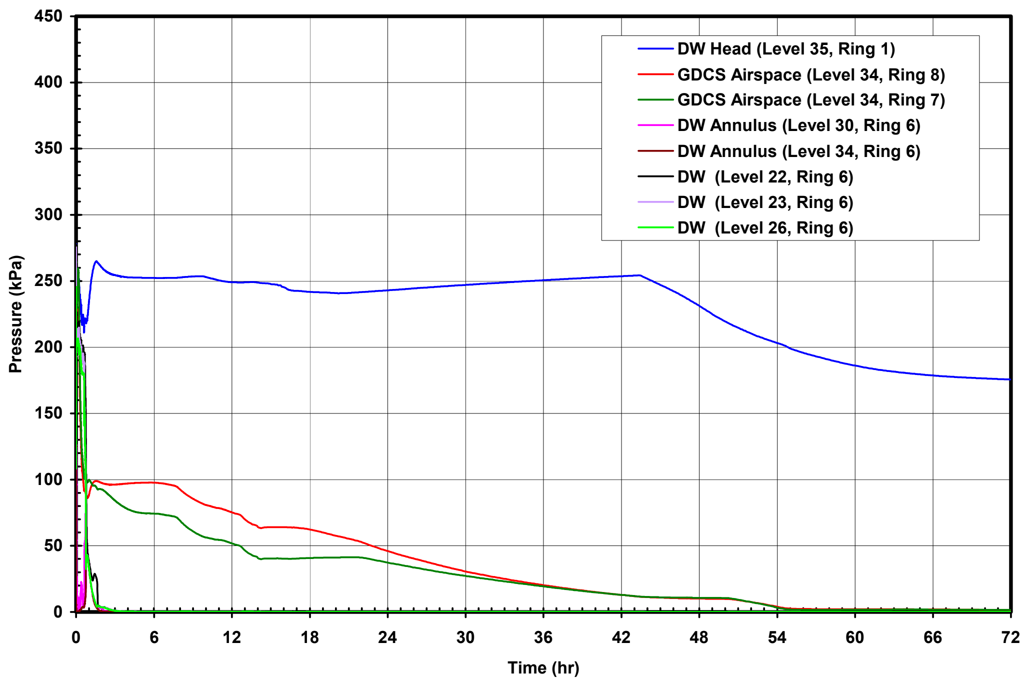


Figure 6I-1d1. Main Steam Line Break, 1 DPV Failure (Bounding Case) – Drywell and GDCS Noncondensable Gas Pressures (72 hrs)

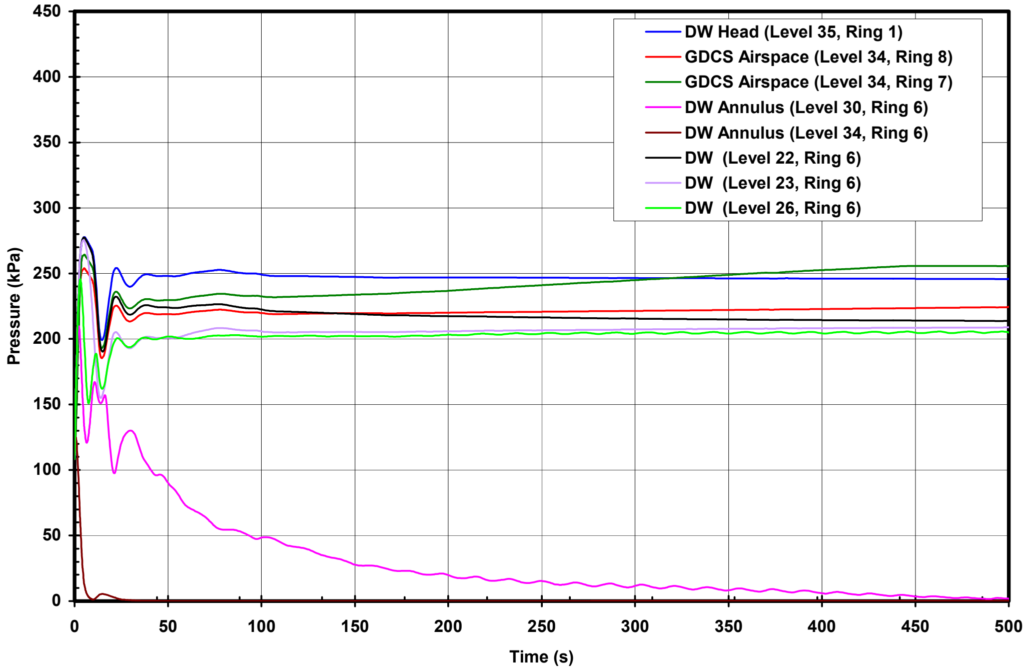


Figure 6I-1d2. Main Steam Line Break, 1 DPV Failure (Bounding Case) – Drywell and GDCS Noncondensable Gas Pressures (500 s)

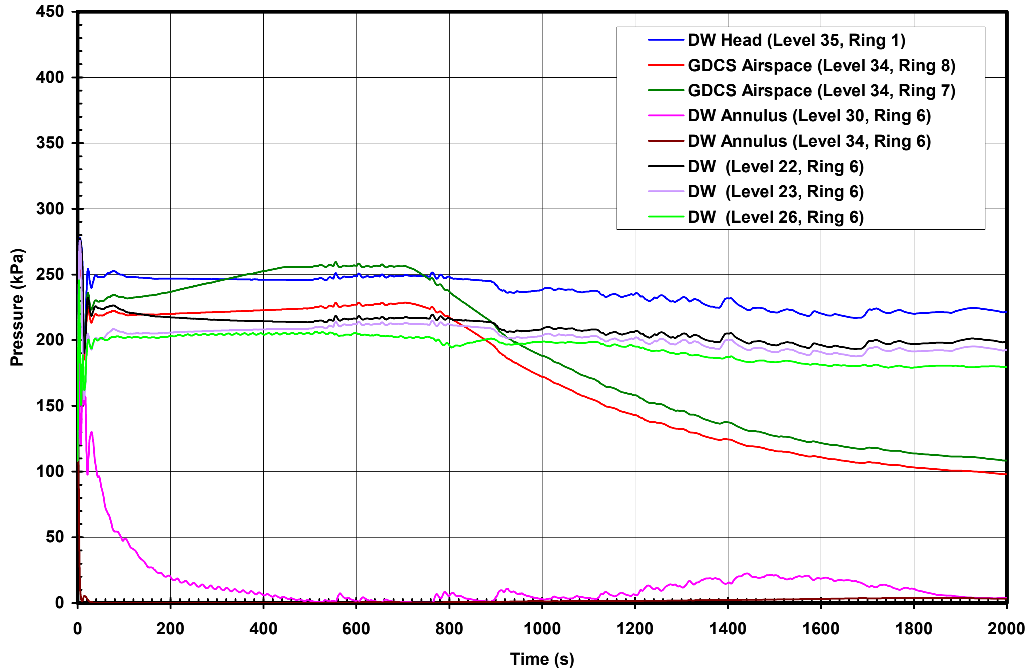


Figure 6I-1d3. Main Steam Line Break, 1 DPV Failure (Bounding Case) – Drywell and GDCS Noncondensable Gas Pressures (2000 s)

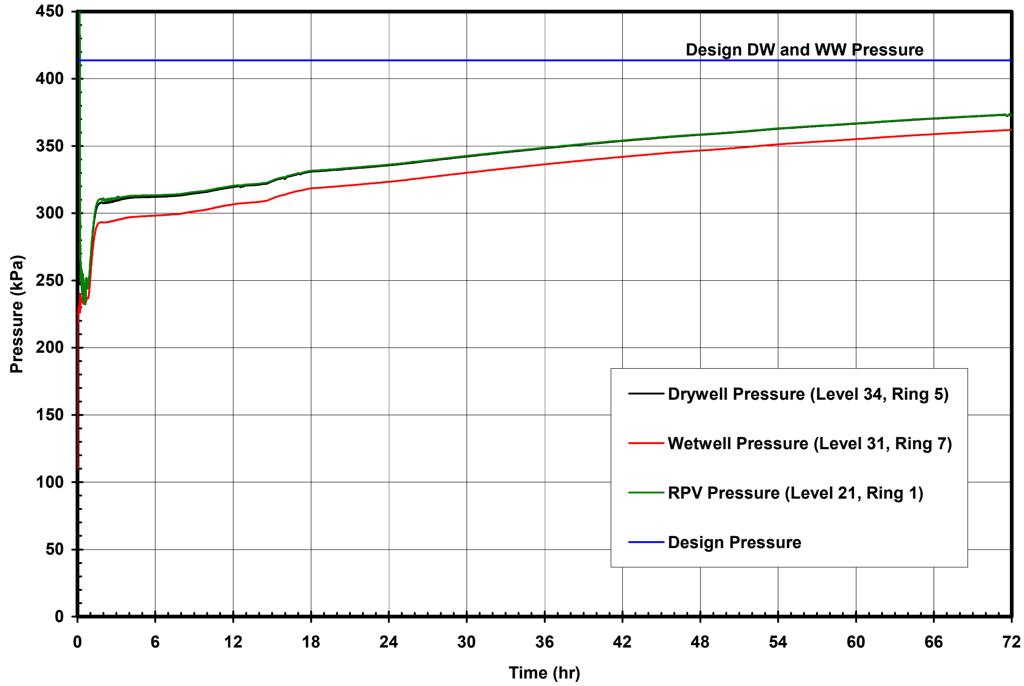


Figure 6I-2a1. Main Steam Line Break, 1 SRV Failure (Bounding Case) – Containment Pressures (72 hrs)

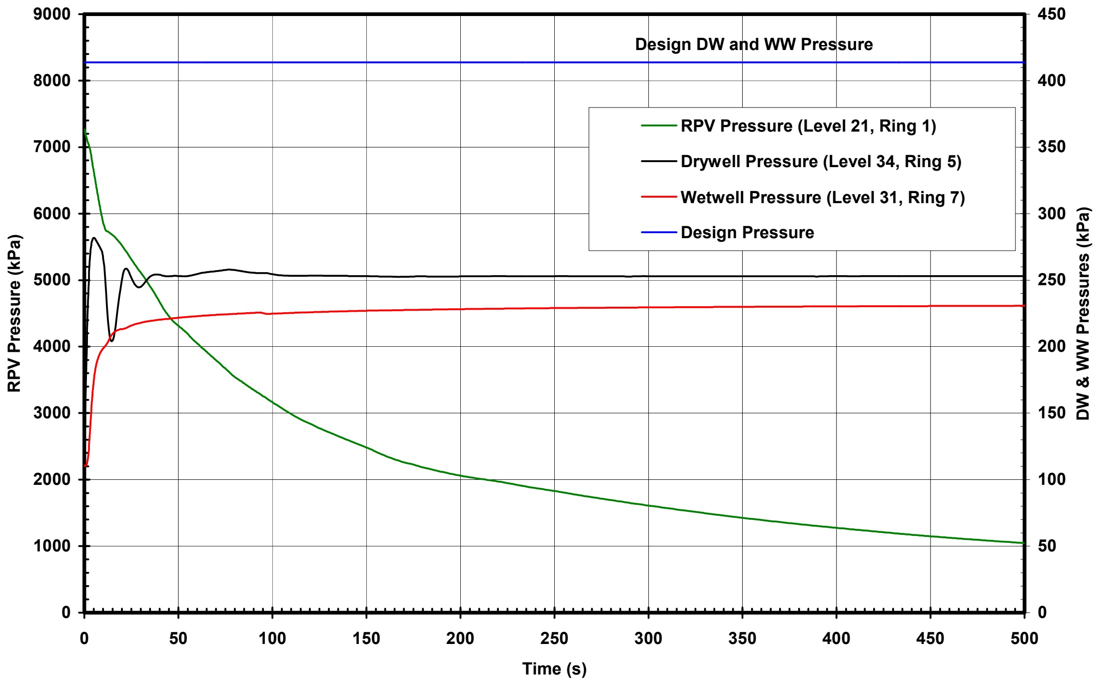


Figure 6I-2a2. Main Steam Line Break, 1 SRV Failure (Bounding Case) – Containment Pressures (500 s)

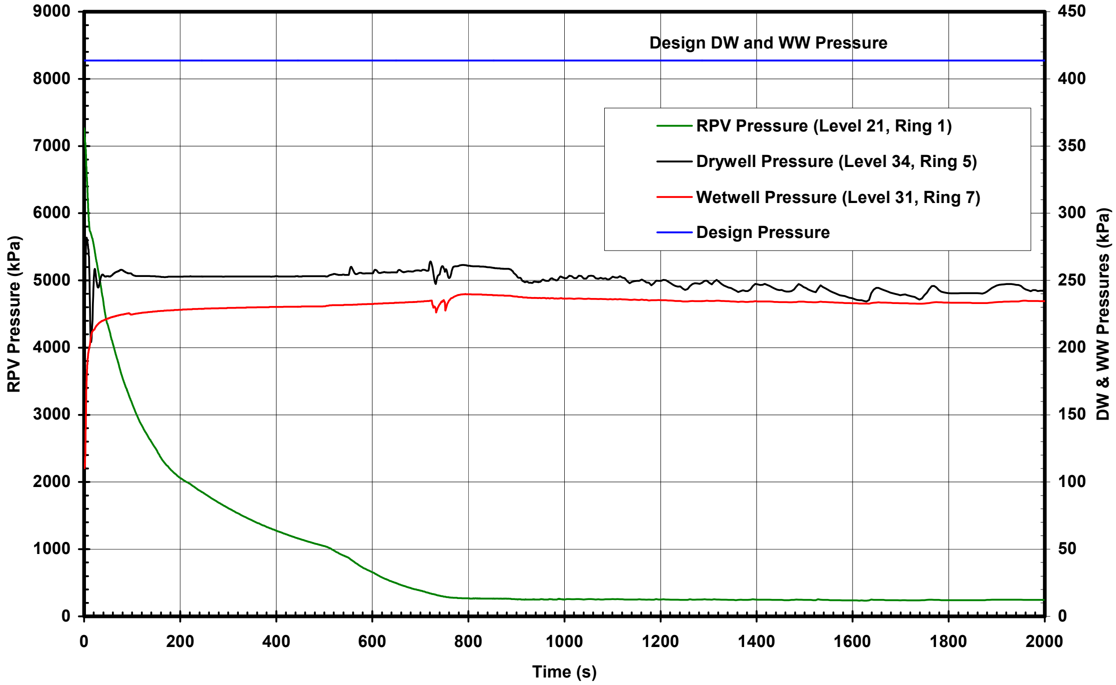
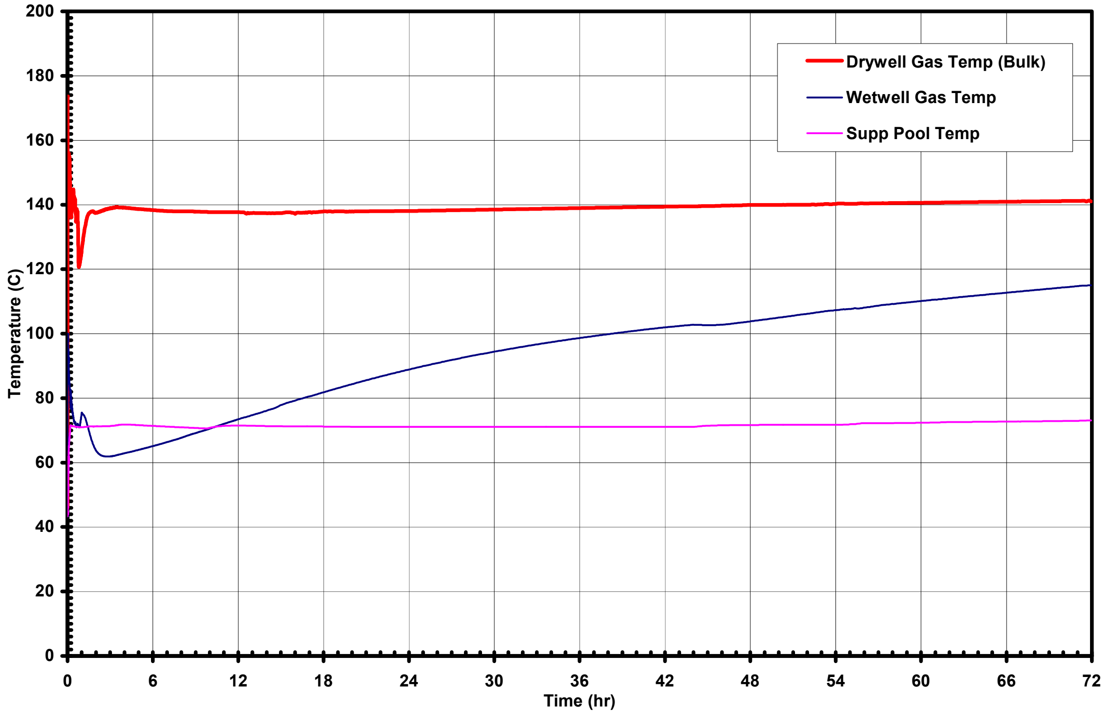
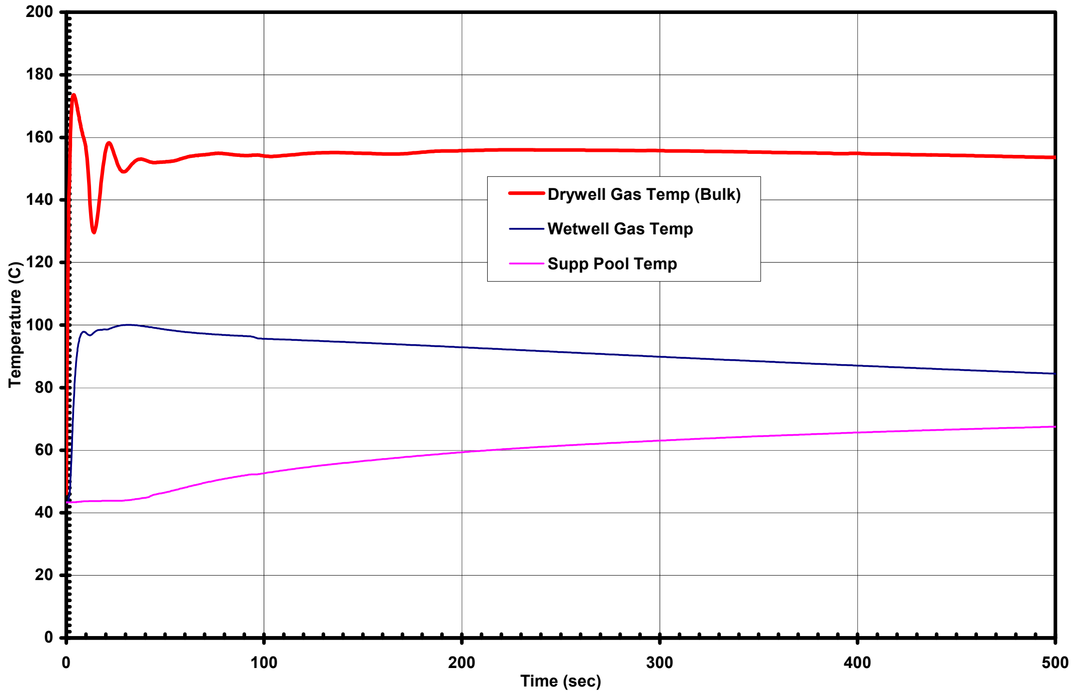


Figure 6I-2a3. Main Steam Line Break, 1 SRV Failure (Bounding Case) – Containment Pressures (2000 s)



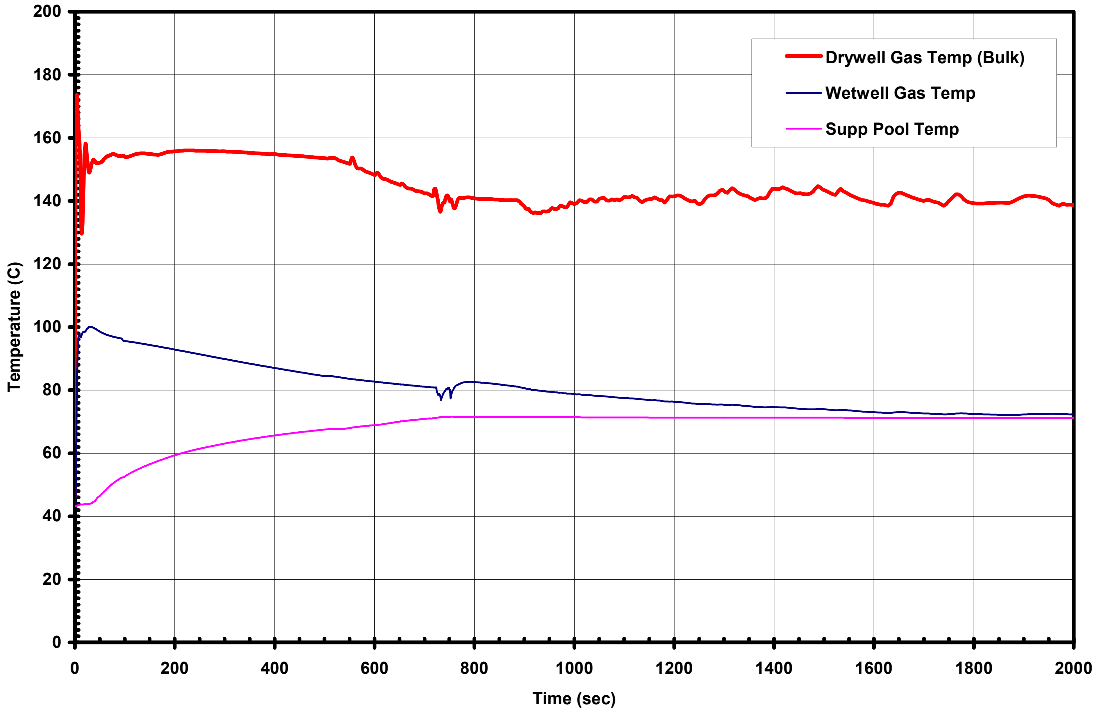
LEGEND: Supp = Suppression

Figure 6I-2b1. Main Steam Line Break, 1 SRV Failure (Bounding Case) – Containment Temperatures (72 hrs)



LEGEND: Supp = Suppression

Figure 6I-2b2. Main Steam Line Break, 1 SRV Failure (Bounding Case) – Containment Temperatures (500 s)



LEGEND: Supp = Suppression

Figure 6I-2b3. Main Steam Line Break, 1 SRV Failure (Bounding Case) – Containment Temperatures (2000 s)

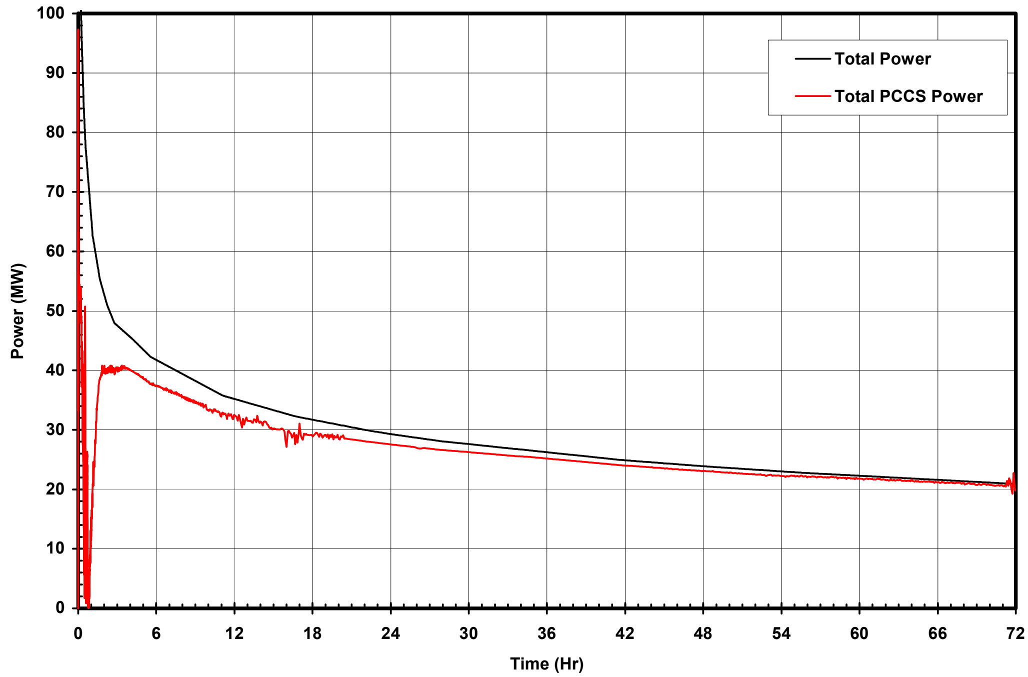


Figure 6I-2c1. Main Steam Line Break, 1 SRV Failure (Bounding Case) – PCCS Heat Removal versus Decay Heat (72 hrs)

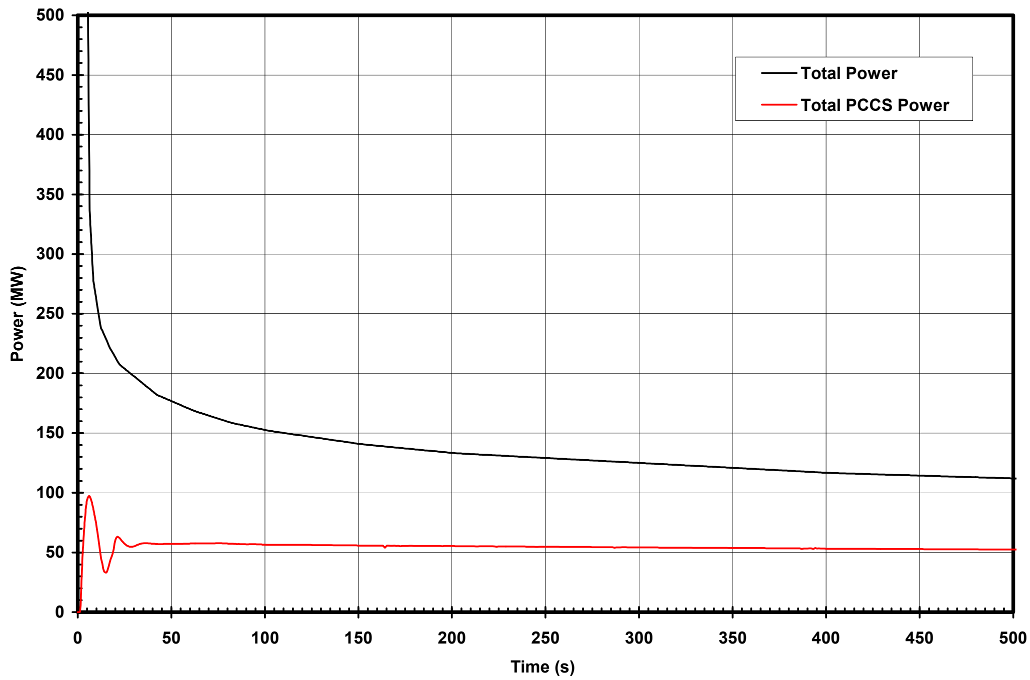


Figure 6I-2c2. Main Steam Line Break, 1 SRV Failure (Bounding Case) – PCCS Heat Removal versus Decay Heat (500 s)

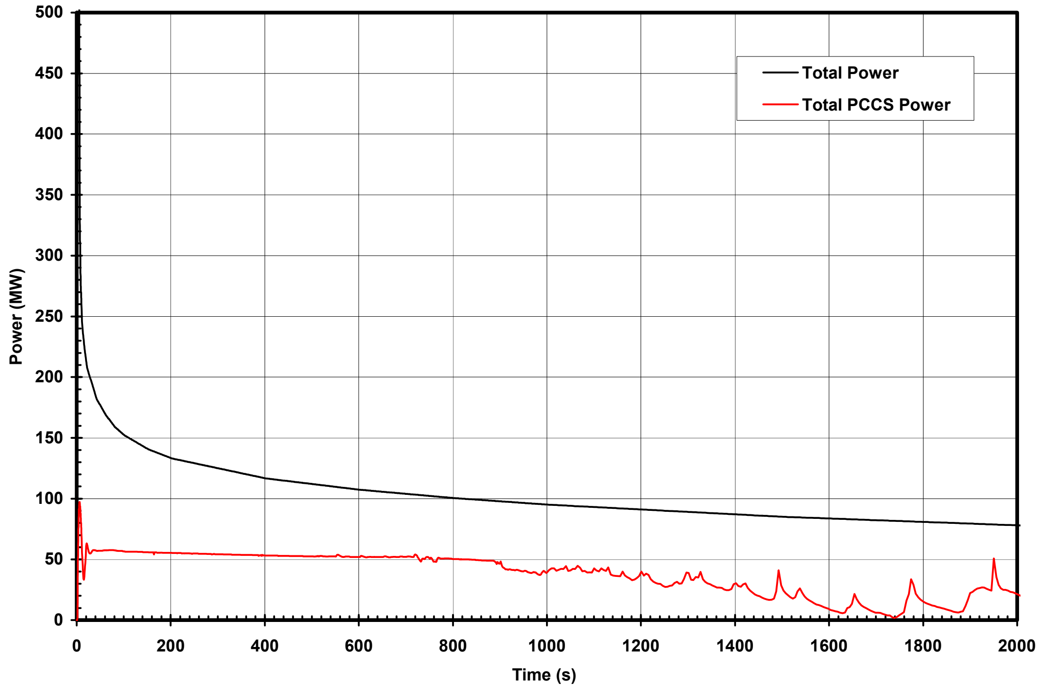


Figure 6I-2c3. Main Steam Line Break, 1 SRV Failure (Bounding Case) – PCCS Heat Removal versus Decay Heat (2000 s)

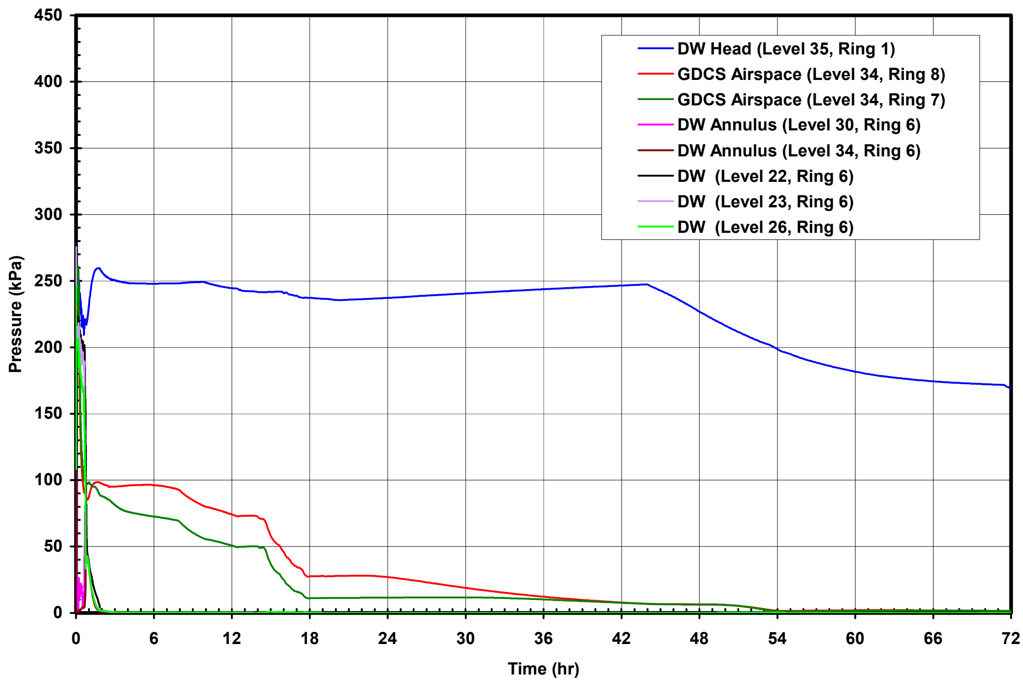


Figure 6I-2d1. Main Steam Line Break, 1 SRV Failure (Bounding Case) – Drywell and GDCS Noncondensable Gas Pressures (72 hrs)

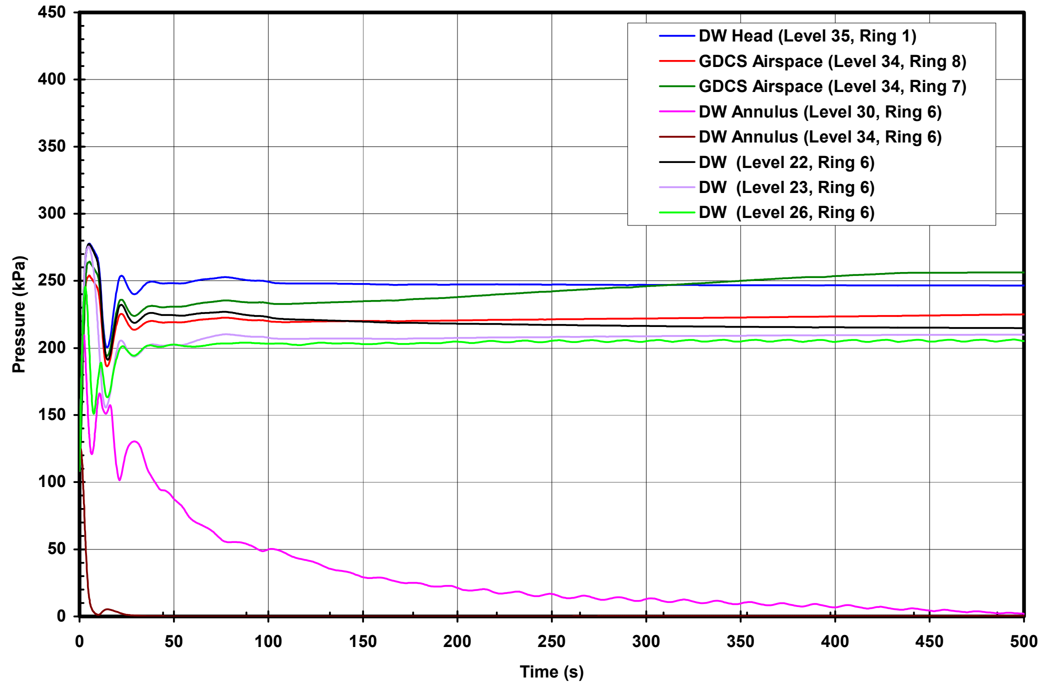


Figure 6I-2d2. Main Steam Line Break, 1 SRV Failure (Bounding Case) – Drywell and GDCS Noncondensable Gas Pressures (500 s)

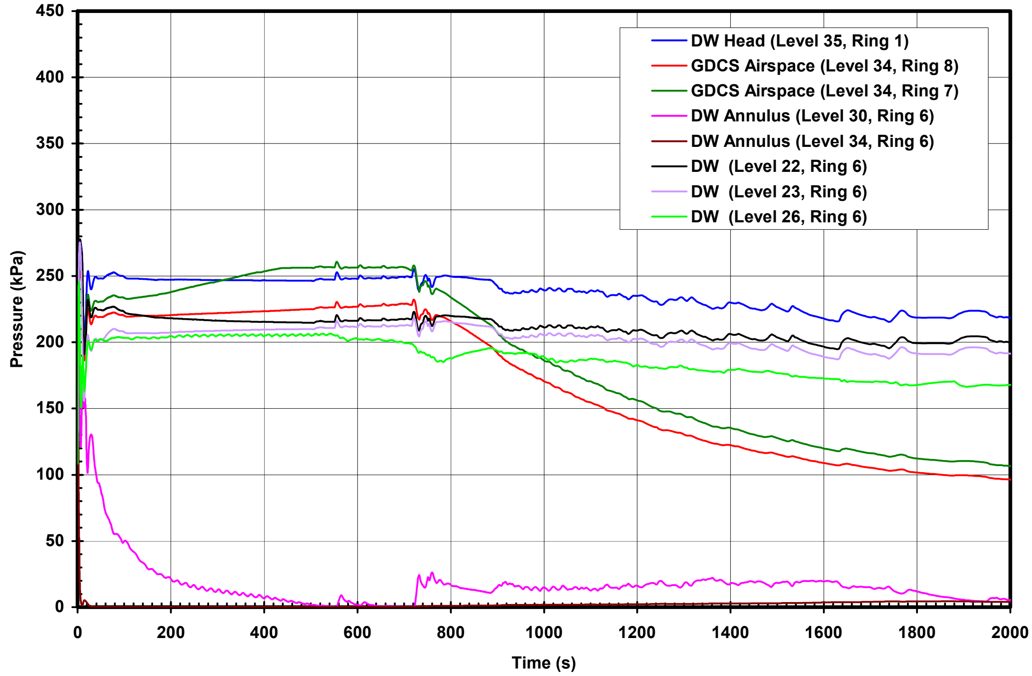


Figure 6I-2d3. Main Steam Line Break, 1 SRV Failure (Bounding Case) – Drywell and GDCS Noncondensable Gas Pressures (2000 s)