6.2 Containment Systems

6.2.1 Containment Functional Design

6.2.1.1 Pressure Suppression Containment

6.2.1.1.1 Design Bases

The ABWR pressure suppression primary containment system, which is comprised of the drywell and wetwell and supporting systems, is designed to have the following functional capabilities:

(1) The containment structure has the capability to maintain its functional integrity during and following the peak transient pressures and temperatures which would occur following any postulated loss-of-coolant accident (LOCA). A design basis accident (DBA) is defined as the worst LOCA pipe break (which leads to maximum containment and drywell pressure and/or temperature) and is further postulated to occur simultaneously with loss of offsite power and a safe shutdown earthquake (SSE).

The containment structure is designed for the full range of loading conditions consistent with normal plant operation and accident conditions, including the LOCA-related design loads in and above the suppression pool.

The containment structure is designed to accommodate the negative pressure difference between the drywell and wetwell and relative to the Reactor Building (R/B) surrounding.

(2) The containment structure and isolation system, with concurrent operation of other accident mitigation systems, is designed to 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 10CFR100.

(3) Capability for rapid closure or isolation of all pipes or ducts which penetrate the containment boundary is provided to maintain leakage within acceptable limits.

(4) The containment structure can withstand coincident fluid jet forces associated with the flow from the postulated rupture of any pipe within the containment.

(5) The containment structure is designed to accommodate flooding to a sufficient depth above the active fuel to permit safe removal of the fuel assemblies from the reactor core after the postulated DBA.

(6) The containment structure is 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.

(7) The containment structure provides means to channel the flow from postulated pipe ruptures in the drywell to the pressure suppression pool.

(8) The containment system is designed to 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 structure to confirm the leaktight integrity of the containment.

(9) The Atmospheric Control System (ACS) establishes and maintains the containment atmosphere to less than 3.5% by volume oxygen during normal operating conditions to maintain an inert atmosphere.

6.2.1.1.2 Design Features

The containment structure consists of the following major components (shown in Figures 1.2-2 through 1.2-12):

(1) A drywell (DW), which is comprised of two volumes:

(a) An upper drywell (UD) volume surrounding the reactor pressure vessel (RPV) and housing the steam and feedwater lines and other connections of the reactor primary coolant system, safety/relief valves (SRVs) and the drywell HVAC coolers.

(b) A lower drywell (LD) volume housing the reactor internal pumps, fine motion control rod drives (FMCRD) and undervessel components and servicing equipment. The UD is a cylindrical, reinforced concrete structure with a removable steel head and a reinforced concrete diaphragm floor. The cylindrical RPV pedestal, which is connected rigidly to the diaphragm floor, separates the LD from the wetwell. It is a prefabricated steel structure filled with concrete after erection. Ten drywell connecting vents (DCVs), approximately 1m x 2m in cross-section, are built into the RPV pedestal and connect the UD and LD. The DCVs are extended downward via 1.2m inside diameter steel pipes, each of which has three horizontal 0.7m diameter vent outlets into the suppression pool.

(2) A wetwell, which is comprised of an air volume and suppression pool filled with water to rapidly condense steam from a reactor vessel blowdown via the SRVs or from a break in a major pipe inside the drywell through the vent system. The wetwell boundary is a cylindrical reinforced concrete wall which is continuous with the UD boundary. A reinforced concrete mat foundation supports the entire containment system and enclosed structures, systems and components, and extends to support the Reactor Building surrounding the containment.

(3) The containment structure includes a steel liner to reduce fission product leakage to allowable levels. All normally wetted surfaces of the liner in the suppression pool are stainless steel. Penetrations through the liner for the drywell head, equipment hatches, personnel locks, piping, and electrical and instrumentation lines are provided with seals and leaktight connections. The allowable leakage is 0.5% per day from all sources, excluding MSIV leakage.

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

6.2.1.1.2.1 Drywell

The drywell is designed to withstand the pressure and temperature transients associated with the rupture of any primary system pipe inside the drywell and also the rapid reversal in pressure when the steam in the drywell is condensed by the ECCS flow following post-LOCA flooding of the RPV.

A vacuum breaker system has been provided between the drywell and wetwell. The purpose of the wetwell-to-drywell vacuum relief system is to prevent backflooding of the suppression pool water into the lower drywell and to protect the integrity of the diaphragm floor (D-F) slab between the drywell and wetwell, and the drywell structure and liner. Redundant vacuum relief systems are provided to protect against failure of a single system. The design drywell-to-wetwell pressure difference is + 172.6 kPaD and –13.73 kPaD. The vacuum breaker system is also designed to withstand the high temperature associated with the break of a small line in the drywell which does not result in rapid depressurization of the RPV.

The maximum drywell temperature occurs in the case of a steamline break (173.2°C). Although this exceeds the ABWR drywell design temperature (171.1 °C), it only exceeds it by 2.1 °C and only for about 2 seconds. Due to thermal inertia, components in the drywell would not have sufficient time to reach the design limit temperature.

The maximum drywell pressure occurs in case of a feedwater line break (281.8 kPaG). The design pressure for the drywell (309.9 kPaG) includes approximately 10% margin.

No vacuum breaker system is required for the primary containment-to-Reactor Building negative pressure, which is predicted to be maximum 11.8 kPaG, between the wetwell and the Reactor Building, compared to the design negative pressure of 13.7 kPaG.

A heating and cooling system is provided to maintain drywell temperatures during normal operation within acceptable limits for equipment operation, as described in Subsection 9.4.9.

The drywell is protected against the dynamic effects of plant-generated missiles (Section 3.5), and the jet forces and pipe whip associated with postulated line breaks (Section 3.6). Protection is provided by the massive structure of the drywell and by providing restraints that prevent pipes from impacting on the drywell walls (see Subsection 3.8.3.1 for additional information).

Both upper drywell and lower drywell are provided with an equipment hatch for removal of equipment during maintenance and an air lock for entry of personnel into the drywell. These access openings are sealed under normal plant operation and are only opened when the plant is shut down for refueling and/or maintenance.

During normal operation, a nitrogen makeup subsystem automatically supplies nitrogen to the wetwell and drywell to maintain a slightly positive pressure to preclude air inleakage from the Reactor Building. Before personnel can enter the drywell, it is necessary to deinert the drywell atmosphere. The ACS, supported by the purge supply and exhaust system, provides for deinerting as discussed in detail in Subsection 6.2.5.2.

6.2.1.1.2.2 Wetwell

The suppression pool water is located inside the wetwell annular region between the cylindrical RPV pedestal wall and the outer wall of the wetwell. The horizontal vent system communicates the drywell to the suppression pool. The nominal submergence to the centerline of the top row of horizontal vents is 3.5m. The vertical spacing between the centerlines of the horizontal vents is 1.37m. The centerline of the bottom horizontal vent is 0.76m above the bottom of the suppression pool.

In the event of a pipe break within the drywell, the increased pressure inside the drywell forces a mixture of air, steam and water through the drywell connecting vents (DCVs) and horizontal vents into the suppression pool, where the steam is rapidly condensed. The noncondensable gases transported with the steam escape to and are contained in the free air volume of the wetwell. There is sufficient water volume in the suppression pool to provide a minimum of 0.61 meters of submergence over the top to the upper row of horizontal vents when water is removed from the pool during post-LOCA drawdown by the ECCS. This drawdown floods the RPV to the steamlines, floods the lower drywell to its drain to the DCV, and provides for water in transit from the break on its gravity drain back to the suppression pool.

The wetwell chamber design pressure is 309.9 kPaG and design temperature is 104°C.

Performance of the pressure suppression pool concept in condensing steam under water (main steamlines through the SRVs) has been demonstrated by the horizontal vent system tests as described in Appendix 3B.

The SRVs discharge steam from the relief valves through their exhaust piping and quenchers into the suppression pool. The quencher locations within the suppression pool are identified in Figures 1.2-3c, 1.2-13i and 3B-3. Operation of the SRVs is intermittent and closure of the valves with subsequent condensation of steam in the exhaust piping can produce a partial vacuum, thereby sucking suppression pool water into the exhaust pipes. Vacuum relief valves are provided on the exhaust piping to control the maximum SRV discharge bubble pressure resulting from high water levels in the SRV discharge pipe.

Under normal plant operating conditions, the maximum suppression pool water and wetwell airspace temperature is 35°C or less. The continued release of decay heat after the initial

blowdown following an isolation event or LOCA may result in suppression pool temperatures as high as 99.6°C. The Residual Heat Removal (RHR) System is available in the Suppression Pool Cooling mode to control the pool temperature. Heat is removed via the RHR heat exchanger(s) to the Reactor Building Cooling Water (RCW) System and finally to the Reactor Service Water (RSW) System. The RHR System is described in Subsection 5.4.7.

6.2.1.1.3 Design Evaluation

6.2.1.1.3.1 Summary Evaluation

The key design parameter and the maximum calculated accident parameters for the pressure suppression containment are shown in Table 6.2-1.

The maximum drywell pressure would occur during a feedwater line break. The maximum drywell temperature condition would result from a main steamline break. All of the analyses assume that the primary system and containment system are initially at the nominal operating conditions.

6.2.1.1.3.2 Containment Design Parameters

Table 6.2-2 provides a listing of the key design parameters of the primary containment system including the design characteristics of the drywell, suppression pool and the pressure suppression vent system.

Table 6.2-2a provides the performance parameters of the related ESF systems which supplement the design conditions of Table 6.2-2 for containment cooling purposes during post-blowdown long-term accident operation. Performance parameters given include those applicable to full capacity operation and reduced capacities assumed for containment analyses. Analyses calculating long-term containment response following a feedwater line break and main steamline break used containment cooling system only, and containment sprays were not used.

6.2.1.1.3.3 Accident Response Analysis

The containment functional evaluation is based upon the consideration of several postulated accident conditions which would result in the release of reactor coolant to the containment. These accidents include:

(1) An instantaneous guillotine rupture of a feedwater line

(2) An instantaneous guillotine rupture of a main steamline

(3) Small break accidents

The containment design pressure and temperature were established based on enveloping the results of this range of analyses.

For the ABWR pressure suppression containment system, the peak containment pressure following a LOCA is relatively insensitive to variations in the size of the assumed primary system rupture. This is because the peak occurs late in the blowdown and is determined in very large part by the transfer of the noncondensible gases from the drywell to the wetwell airspace. In addition, there is an approximately 10% margin between the peak calculated value and the containment design pressure that will easily accomodate small variations in the calculated maximum value.

Tolerances associated with fabrication of the RPV nozzles have been taken into account in this analysis.

6.2.1.1.3.3.1 Feedwater Line Break

Immediately following a double-ended rupture in one of the two main feedwater lines just outside the vessel (Figure 6.2-1), the flow from both sides of the break will be limited to the maximum allowed by critical flow considerations. The effective flow area on the RPV side is 0.08399 m^2. Reverse RPV flow in the second FW line is prevented by check valves shown in Figure 6.2-1.

The feedwater system side of the feedwater line break (FWLB) was modeled by adding a time variant feedwater mass flow rate and enthalpy directly to the drywell airspace. The time histories of the mass flow and enthalpy were determined from the operating characteristics of a typical feedwater system.

In order to provide further assurance of conservatism, FWLB mitigation is added to the ABWR design. The system is described in Subsection 7.3.1.1.2. The specific enthalpy time history, assuming the break flow of Figure 6.2-3, is shown in Figure 6.2-4. Initial reactor power is assumed to be 102% NBR.

6.2.1.1.3.3.1.1 Assumptions for Short-Term Response Analysis

The response of the Reactor Coolant System and the Containment System during the short-term blowdown period of the accident has been analyzed using the following assumptions:

(1) The initial conditions for the FWLB accident maximize the containment pressure response. That is:

   (a) The reactor is operating at 102% of the rated thermal power, which maximizes the post-accident decay heat.

   (b) The initial suppression pool mass is at the high water level.

   (c) The initial wetwell air space volume is at the high water level.

   (d) The suppression pool temperature is the operating maximum value.

(2) The feedwater line is considered to be severed instantaneously. This results in the most rapid coolant loss and depressurization of the vessel, with coolant being discharged from both ends of the break.

(3) Scram occurs in less than one second from receipt of the high drywell pressure signal.

(4) The main steam isolation valves (MSIVs) start closing at 0.5 s after the accident. The turbine stop valves are closed in 0.2 seconds after reactor trip/turbine trip (RT/TT). By assuming rapid closure of these valves, the RPV is maintained at a high pressure, which maximizes the calculated discharge of high energy water into the drywell.

(5) The vessel depressurization flow rates are calculated using Moody's critical flow model (Reference 6.2-2).

(6) Influence of the ECCS systems is minimal since the time interval analyzed for short-term is approximately the same time as the response time of associated systems injections into the RPV.

(7) Drywell and wetwell airspaces are homogeneous mixtures of inert atmosphere, vapor and liquid water.

(8) The wetwell airspace temperature is allowed to exceed the suppression pool temperature as determined by a mass and energy balance on the airspace.

(9) Wetwell and drywell wall and structure heat transfer are ignored.

(10) Actuation of SRVs is modeled.

(11) Wetwell-to-drywell vacuum breakers are modeled but do not open.

(12) Drywell and wetwell sprays and RHR cooling mode are not modeled.

(13) Not Used.

(14) Initial drywell conditions are 0.107 MPa, 57°C, and 20% relative humidity.

(15) Initial wetwell airspace conditions are 0.107 MPa, 35°C and 100% relative humidity.

(16) The drywell is modeled as a single node. All break flow into the drywell is homogeneously mixed with the drywell inventory.

(17) Not Used.

6.2.1.1.3.3.1.2 Assumptions for Long-Term Cooling Analysis

Following the blowdown period, the ECCS discussed in Section 6.3 provides water for core flooding, containment spray, and long-term decay heat removal. The containment pressure and temperature response during this period was analyzed using the following assumptions:

(1) There are two HPCF Systems, one RCIC System, and three RHR Systems in the ABWR. All motor operated systems (HPCF and RHR) are assumed to be available to maximize pump heat into the suppression pool. A single failure of one RHR heat exchanger was assumed for conservatism.

(2) The ANSI/ANS-5.1 decay heat plus 2-sigma uncertainty is used. Fission energy, fuel relaxation heat, and pump heat are included.

(3) The suppression pool volume corresponds to the low water level.

(4) After 30 minutes, one RHR heat exchanger is activated to remove energy via recirculation cooling of the suppression pool and one RHR heat exchanger is activated to remove energy via drywell sprays with the RCW System and ultimately to the RSW System.

(5) The maximum service water temperature is assumed to be 35°C. This is a conservative assumption that maximizes the suppression pool temperature.

(6) The lower drywell flooding is not modeled. Water which is from the lower drywell is assumed to be mixed with the suppression pool to calculate the bulk average temperature.

(7) Structural heat sinks are modeled in the containment system.

6.2.1.1.3.3.1.3 Short-Term Accident Responses

The calculated containment pressure and temperature responses for a feedwater line break are shown in Figures 6.2-6 and 6.2-7, respectively.

The containment pressure response (Figure 6.2-6) covers the pool swell phase of the short-term containment response. The drywell pressure peaks soon after the bubble breakthrough as the break flow continues to push the drywell air into the wetwell. The wetwell pressure also continues to climb after this phase as the air carryover from the drywell continues.

6.2.1.1.3.3.1.4 Long-Term Accident Responses

In order to assess the adequacy of the containment system following the initial blowdown transient, an analysis was made of the long-term temperature and pressure response following the accident. The analysis assumptions are those discussed in Subsection 6.2.1.1.3.3.1.2.

Figure 6.2-8 shows temperature time histories for the suppression pool, wetwell, and drywell temperatures. Peak temperatures are bounded by those predicted for the main steamline break event as described in Subsections 6.2.1.1.3.3.2.3 and 6.2.1.1.3.3.2.4.

6.2.1.1.3.3.2 Main Steamline Break

A schematic of the ABWR main steamlines, with a postulated break in one of the main steamlines, is shown in Figure 6.2-9. The main steamline (MSL) break is a double-ended break with one end fed by the RPV directly through the broken line, and the other fed by the RPV through the unbroken main steamlines until the MSIVs are closed. Once the MSIVs are closed, the break flow is only from the RPV through the broken line.

Each MSL contains a flow limiter built into the MSL nozzle on the RPV with a throat area of 0.09848 m^2 as shown in Figure 6.2-9. This flow limiter provides the effective break area for the vessel side.

Flow from the condenser side of the break continues for 0.5 seconds, at which time the MSIVs begin to close on high flow signal. A valve stroke time of 4.5 seconds is used for the MSIV closure. Flow from the condenser side of the break is ramped down to zero between 0.5 and 5.0 seconds.

6.2.1.1.3.3.2.1 Assumptions for Short-Term Response Analysis

The response of the reactor coolant system and the containment system during the short-term blowdown period of the MSLB accident is analyzed using the assumptions listed in the above subsection and Subsection 6.2.1.1.3.3.1.1 for the feedwater line break, except feedwater mass flow rate for a MSL break was assumed to be 130 % NBR for the case where no operator action is assumed to control water level. Additional cases were run with feedwater mass flow rate regulated to control RPV water level or with no feedwater flow based on an assumed loss of offsite power.

6.2.1.1.3.3.2.2 Assumption for Long-Term Cooling Analysis

The containment pressure and temperature response during the period following blowdown is analyzed using the assumptions listed in Subsection 6.2.1.1.3.3.1.2.

6.2.1.1.3.3.2.3 Short-Term Accident Response

Figures 6.2-12 and 6.2-13 show the pressure and temperature responses of the drywell and wetwell during the blowdown phase of the steamline break.

The maximum drywell temperature (173.2 °C) is predicted to occur for the steamline break. MSLB with two-phase blowdown starting when the RPV water level is at or below the main steamline nozzle provides the highest peak drywell air temperature. The peak drywell temperature is 173.2°C, which is above the design value of 171.1°C, and is the limiting one as compared to the FWLB peak temperature. As noted in Section 6.2.1.1.2.1, this peak calculated

drywell temperatrue exceeds the design limit for only 2 seconds. The peak drywell pressure for the MSLB remains below that for the FWLB, which becomes the most limiting. The peak drywell pressure is below the design pressure. The MSLB is the limiting event for peak drywell temperature. The FWLB is the most limiting for drywell pressure.

6.2.1.1.3.3.2.4 Long-Term Accident Response

The long-term drywell pressure response following the MSLB accident remains below that for the feedwater line break. The long-term drywell temperature response remains below that for the peak achieved in the short term for the steam line break shown in Figure 6.2-13a. Peak wetwell pressure, suppression pool temperature, and wetwell temperature are predicted for the steamline break long term response as shown in Figures 6.2-12b, 6.2-13b, and 6.2-13c.

The peak pool temperature (99.6°C) is reached at 6600 seconds (1.833 hours). This is less than the suppression pool temperature value of 100°C which is used in the net positive suction head available (NPSHA) calculations for RHR and HPCF.

6.2.1.1.3.4 Accident Analysis Models

6.2.1.1.3.4.1 Short-Term Pressurization Model

The analytical models, assumptions and methods used to evaluate the containment response during the reactor blowdown phase of a LOCA are similar to those for the feedwater line break.

6.2.1.1.3.4.2 Long-Term Cooling Model

Once the RPV blowdown phase of the LOCA is over, a fairly simple model of the drywell and wetwell may be used. During the long-term post-blowdown transient, the RHR cooling system flow path is a closed loop and the suppression pool mass will be constant.

The analytical models, assumptions and methods used to evaluate the containment response during the long-term cooling phase of a LOCA are described in Reference 6.2-3.

6.2.1.1.4 Negative Pressure Design Evaluation

During normal plant operation, the inerted wetwell and the drywell volumes remain at or slightly above atmospheric conditions. However, certain events in the containment cause depressurization transients that can create a negative pressure differential across the diaphragm floor and lower drywell access tunnels (negative means the wetwell pressure is greater than the drywell pressure) and a negative pressure differential across the drywell and the wetwell walls (negative means the Reactor Building pressure is greater than the containment pressure). Vacuum relief function is necessary in order to limit these negative pressure differentials within design values. The events which cause the containment depressurization are:

(1) The drywell/wetwell sprays are inadvertently actuated during normal operation.

(2) The drywell is depressurized following a LOCA.

(3) The wetwell spray is actuated subsequent to a stuck open relief valve (SORV).

Drywell depressurization following a LOCA results in the severest pressure transient in the drywell; this transient is therefore used in sizing the Wetwell-to-Drywell Vacuum Breaker System (WDVBS). The most severe depressurization in the wetwell is caused by wetwell spray actuation subsequent to a stuck open relief valve. The analysis of this transient shows that the Primary Containment Vacuum Breaker System (PCVBS) is not required.

6.2.1.1.4.1 Wetwell-to-Drywell Negative Differential Pressure

The WDVBS is sized to keep the differential pressure between the drywell and wetwell within the negative design values for the PCV, diaphragm floor, and tunnels during all operating and accident transients.

Without the WDVBS, the post-LOCA drywell pressure may decrease to the saturation pressure (20.6-27.5 kPaA) of the drywell spray flow or the break flow out of the RPV, and the wetwell pressure may be still around 275.6 kPaA, creating the negative pressure differential close to 275.6 kPaD. The primary purpose of the WDVBS is to prevent such a large negative pressure differential between the drywell and wetwell. In addition, the WDVBS can hold the drywell pressure above the negative design pressure of the PCV liner. This is achieved by the transfer of air from the wetwell to the drywell.

The specific requirements to be met by the WDVBS are:

(1) The drywell-to-wetwell negative differential pressure shall be less than 13.7 kPaG. This limits the negative pressure differential across the diaphragm floor, tunnels, and the pedestal.

(2) The drywell-to-Reactor Building negative pressure shall be less than 13.7 kPaG.

This requirement protects the PCV liner on the drywell portion of the containment.

Drywell depressurization is caused by two major events:

(1) Post-LOCA drywell depressurization

(2) Inadvertent drywell spray actuation during normal operation

The former causes a much larger depressurization in the drywell; this depressurization would become the most severe if a break occurred in a feedwater line. Hence, the feedwater line break post-LOCA transient is the limiting event for the sizing of the WDVBS. Following the break, the pressurization of the drywell causes the air in the drywell to be purged into the wetwell airspace, leaving the drywell full of steam. Subsequent condensation of this steam by cold ECCS flow through the break results in the depressurization of the drywell. This depressurization follows the general trend shown in Figure 6.2-16.

As the RPV is reflooded, the ECCS flow begins to cascade down through the break and into the drywell, causing the initial drywell depressurization (Region I in Figure 6.2-16). The pressure differential between the drywell and wetwell causes suppression pool water to flow into the drywell, accelerating the drywell depressurization rate even further (Region II). When the pressure difference between the drywell and wetwell reaches a predetermined setpoint, the WDVBS opens, allowing the flow of air back into the drywell, thus slowing down its depressurization, and eventually reaching a steady state (Region III). As can be observed, the maximum negative pressure differential between the wetwell and drywell occurs during the depressurization of the drywell and can be controlled by proper sizing of the WDVBS.

Drywell-to-Reactor Building negative pressure differentials can exist during both drywell depressurization and the steady-state condition which takes place as the drywell pressure approaches the wetwell pressure. The drywell and wetwell pressures decrease slightly below the initial containment pressure because the steam condenses due to the drywell spray or the cold break flow as the air is evenly distributed in the PCV.

Limiting conditions are selected such that the initial drywell depressurization is the most severe. This determines the WDVBS size to meet the drywell-to-wetwell negative design pressure requirement. The following case was found to be the limiting one:

(1) No decay heat in the RPV, as would be the case during a hot standby condition (not reactor isolation)

(2) No drywell sprays

(3) Maximum break flow

(4) No pool cooling

(5) Maximum allowable wetwell temperature prior to LOCA

(6) ECCS flow taken from the condensate storage tank at 15.6°C

(7) The ECCS is comprised of 2 HPCFs, 1 RCIC, and 3 RHR LPFLs (no single failure in ECCS)

Additionally, the limiting event and conditions were considered for the PCV negative design pressure requirement on the drywell part during steady-state operations. The limiting event is the same as the one above and all conditions are the same except that the wetwell spray was activated.

The following assumptions were made during the analysis model:

(1) Suppression pool and wetwell airspace temperatures prior to the LOCA are 35°C.

(2) Minimum condensate storage tank temperature is 15.6°C.

(3) Maximum combination of HPCF, RCIC and LPFL flows is 4316 m^3/h and remains at this value throughout the event.

(4) Any liquid flow into the drywell remains in the drywell airspace.

When the drywell pressure first drops below the wetwell pressure, the following conditions exist in the containment:

(1) Pressure in the drywell is 271.6 kPaA.

(2) Pressure in the wetwell is 273.6 kPaA.

(3) Drywell temperature is 130.1°C.

(4) Wetwell temperature is 98.4°C.

(5) Relative humidity in the drywell is 100%.

(6) Relative humidity in the wetwell is 12.9%.

(7) Height of water in the suppression pool is 7.62m.

(8) Suppression pool temperature is 51.5°C.

(9) Height of water in the horizontal vent vertical pipes is 7.50m.

Other physical parameters of importance to this transient are:

(1) Surface area of the suppression pool (wetwell side) = 506.6m^2.

(2) Total flow area of drywell connecting vents = 11.3m^2.

(3) Lower drywell volume = 1644.4m^3.

(4) Upper drywell volume = 5493.7m^3.

(5) Air volume ratio (wetwell/drywell) = 0.81.

(6) Vacuum breakers start opening at 0.69 kPaD, and fully open at 3.43 kPaD.

The vacuum breaker size is characterized by the ratio $A/\sqrt{k}$, where A is the actual flow area of the vacuum breaker and k its pressure loss coefficient. When $A/\sqrt{k} \geq 0.77m^2$, the calculated negative pressure differential is 9.8 kPaD between the wetwell and drywell. The pressure-time histories are shown in Figure 6.2-17. Thus, a WDVBS effective area of 0.77m^2

is adequate to satisfy the drywell-to-wetwell negative design pressure requirements of 13.7 kPaD.

With the WDVBS size determined above, the PCV negative design pressure on the drywell side is checked. This analysis utilizes the wetwell spray in order to minimize the wetwell/drywell pressure. Figure 6.2-18 shows the pressure-time histories for the wetwell and drywell. It should be noted that no drastic depressurization occurs because the WDVBS has sufficient size to prevent the initial rapid depressurization in the drywell. In addition, the wetwell airspace contains a large amount of air and the wetwell spray capacity is less than 15% of the drywell break flow capacity. The lowest containment pressure, and thus the maximum PCV-to-Reactor Building negative pressure, occurs during the steady-state end of the transient. The final pressure becomes lower than the initial containment pressure because the drywell/wetwell sprays decrease the vapor partial pressure and cool the air in the PCV as the WDVBS equalizes the pressure in the drywell to that in the wetwell.

The maximum negative pressure is 5.9 kPaG for the drywell and wetwell, which satisfies the PCV negative design pressure requirement of 13.7 kPaG.

With a typical vacuum breaker diameter of 50.8 cm and a flow loss coefficient (k) of 3, the required number of wetwell-to-drywell vacuum breakers is eight, which considers one single failure in the WDVBS. The total flow area for eight vacuum breakers is 1.53m^2.

Vacuum breakers are intended to be swing check type valves which open passively due to negative differential pressure (wetwell gas space pressure greater than the drywell pressure) across the valve disk, and require no external power to actuate them. These valves are installed horizontally locating in wetwell gas space, one valve per penetration (through pedestal wall) opening into lower drywell. This position location protects vacuum breaker valves from being subjected to cyclic pressure loading during LOCA steam condensation period. Position location of these valves, both axially and azimuthally, are shown in Figures 1.2.3c and 1.2.13k.

In view that these vacuum breaker valves are located in the wetwell gas space, they can be subjected to loads due to pool swell during early phase of a loss-of-coolant accident. The containment design will provide features, as appropriate, which will protect these valves from applicable loads due to pool swell. For example, the design may include features which protect the valves by designing catwalk structure below the valves as a solid plate of sufficient area assuring complete structural shielding of vacuum breakers which are located (approximately) 1m above the catwalk platform from possible direct pool swell impact loads, as well as protection from possible water fallback associated with flow around edges of solid catwalk area. The COL applicant will review the issue of providing appropriate structural features protecting these valves from pool swell loads and propose to the NRC staff an appropriate design for assuring that these valves are protected adequately. The structural shielding will be designed for pool swell loads determined based on the methodology approved for Mark II/III designs. For design of structural shielding features, pool swell loads to the maximum practical extent will be defined. See Subsection 6.2.7.4 for COL license information.

6.2.1.1.4.2 Wetwell-to-Reactor Building Negative Differential Pressure

Since the WDVBS meets the PCV negative design pressure requirement on the drywell, additional analyses were performed to determine need for the PCVBS to satisfy the PCV negative design pressure requirement on the wetwell.

The wetwell-to-Reactor Building negative pressure shall be less than 13.7 kPaG to protect the PCV liner in the wetwell.

The following wetwell depressurization Events [(1),(2) and (3)] which may result in negative differential pressure in the wetwell were considered:

(1) Drywell and wetwell spray actuation during normal operation

(2) Wetwell spray actuation subsequent to stuck open relief valve (SORV)

(3) Drywell and wetwell spray actuation following a LOCA

The depressurization results presented in the previous subsection indicate that maximum negative pressure in the wetwell for the Event (3) conditions is expected to be 5.9 kPaG, which satisfies the PCV negative design pressure requirement of 13.7 kPaG without the PCVBS. Events (1) and (2) were analyzed to determine the limiting maximum negative pressure in the wetwell, and conclude whether or not the PCVBS is required.

(1): Drywell and wetwell spray actuation during normal operation

The key assumptions and initial conditions used in analyzing this event are:

(1) Inert gas behaves as a perfect gas.

(2) Initial drywell temperature is 57.2°C.

(3) Initial wetwell temperature is 35°C.

(4) Initial containment (drywell and wetwell) pressure is 101.1 kPaA.

(5) Initial drywell relative humidity is 20%.

(6) Initial wetwell relative humidity is 100%.

(7) Wetwell and drywell spray water source is the suppression pool.

(8) Drywell spray flow rate is 954 m^3/h.

(9) Wetwell spray flow rate is 160 m^3/h.

(10) Initial suppression pool temperature is 35°C.

(11) WDVBS area is $0.77m^2$.

(12) No PCVBS modeled.

Recognizing that drywell initial relative humidity and suppression pool initial temperature (suppression pool is the water source for sprays), an additional case representing a non-mechanistic and conservative combination of these two input parameters was also analyzed. The two cases which were analyzed for this event are:

(a) Initial conditions and assumptions as listed above.

(b) Same as case a, except drywell initial relative humidity of 60%, and suppression pool initial temperature of 23.9°C.

The calculated maximum negative differential pressure in the wetwell for cases a and b is found to be 6.9 kPaG and 11.8 kPaG, respectively. These results show that the containment design satisfies the PCV negative design pressure requirement of 13.7 kPaG, without PCVBS.

Event (2): Wetwell Spray Actuation Subsequent to SORV.

The effect of SRV discharge to the suppression pool is to heat the wetwell airspace, thus increasing its pressure. When the pressure in the wetwell becomes greater than the drywell pressure, the WDVBS allows the flow of air from the wetwell to the drywell, thereby pressurizing both volumes. The wetwell pressure and temperature peak when the reactor decay heat decreases below the heat removal from the continued pool cooling and wetwell spray. The wetwell temperature and pressure decrease, but the drywell pressure remains at its peak value. When the pressure difference between the two volumes becomes greater than the hydrostatic head of water above the top vent, air flows back into the wetwell airspace, slowing down the wetwell depressurization rate. The pressure differential between the drywell and wetwell is maintained constant at the hydrostatic head above the top row of horizontal vents. The final pressure in the wetwell is lower than the Reactor Building (R/B) pressure because more air is transferred to the drywell during wetwell pressurization than is received during wetwell depressurization.

The following assumptions are made in analyzing the above event:

(1) Inerted gas behaves as a perfect gas.

(2) Temperature in the drywell remains at 57.2°C throughout the transient by means of the drywell cooler.

(3) Initial wetwell temperature is 35°C.

(4) Initial containment pressure is 101.1 kPaA.

(5) Maximum suppression pool temperature is 97.2°C.

(6) Wetwell spray is from the suppression pool.

(7) Initial wetwell spray temperature is 35°C.

(8) Capacity of the RHR heat exchanger is 0.371 MJ/s·°C.

(9) Maximum wetwell temperature is determined by the maximum wetwell spray temperature and the pool surface heat transfer to the wetwell airspace.

(10) Convective heat transfer coefficient between the suppression pool and the wetwell airspace is 41.0 kJ/h·m^2·°C.

(11) Mixture of steam and air in the drywell is homogeneous such that the ratio of its partial pressures remain constant after the peak pressure is attained.

(12) Air content of the horizontal vent flow mixture increases the wetwell pressure.

(13) Drywell pressure is equal to the wetwell pressure when the peak pressure is reached.

(14) Wetwell vapor pressure is equal to the saturation pressure at the wetwell temperature due to the wetwell spray.

(15) Initial relative humidity in the drywell is 20%.

(16) Initially, the suppression pool is at the High Water Level point.

(17) Wetwell spray flow rate is 114 m^3/h.

An analysis was conducted with no PCVBS, and the maximum negative differential pressure between the wetwell and the Reactor Building was determined to be 11.8 kPaD. This shows that the SORV is a much more severe event than the Event (3) (during which the maximum negative differential pressure is 5.9 kPaG) and Event 1 (during which the maximum differential pressure is 9.8 kPaG) transients. Therefore, the PCV negative pressure requirement of 13.7 kPaG on the wetwell side can be met without PCVBS.

6.2.1.1.5 Steam Bypass of the Suppression Pool

6.2.1.1.5.1 Introduction

The concept of the pressure suppression reactor containment is that any steam released from a pipe rupture in the primary system will be condensed by the suppression pool and will not have an opportunity to produce a significant pressurization effect on the containment. This is accomplished by channeling the steam into the suppression pool through a vent system. If a leakage path were to exist between the drywell and the wetwell gas space, the leaking steam would produce undesirable pressurization of the containment. To mitigate the consequences of

any steam which bypasses the suppression pool, operator will actuate containment sprays 30 minutes after containment pressure reaches to a defined value.

The following presents the results of calculations performed to determine the allowable leakable capacity between the drywell and wetwell gas space.

6.2.1.1.5.2 Criteria

The allowable bypass leakage is defined as the amount of steam which could bypass the suppression pool without exceeding the containment design pressure. In calculating this value, a stratified drywell atmosphere model is used to ensure a conservative result. A stratified model will allow steam only flow through the bypass leakage area, thus maximizing heatup of the wetwell gas space.

6.2.1.1.5.3 Bypass Capability Without Containment Sprays and Heat Sinks

Large primary system ruptures generate high pressure differentials across the assumed leakage paths which, in turn, give proportionately higher leakage flow rates. However, large primary system breaks also rapidly depressurize the reactor and terminate the blowdown. Once this has occurred, there will no longer be a pressure differential across the drywell leakage path, so the containment pressurization due to steam bypass leakage will cease. Since leakage into the wetwell gas space is of limited duration, the allowable area of the steam bypass leakage paths is expected to be large.

As the size of the assumed primary system rupture decreases, the magnitude of the differential pressure across any leakage path also decreases. However, smaller breaks are expected to result in an increasingly longer reactor blowdown period, which, in turn, results in longer duration of the steam bypass leakage flow. The limiting case is a sufficiently small primary system break which will not automatically result in reactor depressurization. For this case it is assumed that the response of the plant operator is to shut the reactor down in an orderly manner at 55.6°C per hour cooldown rate. This would result in the reactor being depressurized and the break flow being terminated within approximately 6 hours. During this 6-h period, the blowdown flow from the reactor primary system would have swept all the drywell noncondensable gas over into the wetwell gas space. This continuous pressure differential, combined with a 6-h duration, is expected to result in the most severe drywell-to-wetwell steam bypass leakage requirement.

Based on the above description of a limiting case, a simplified analysis was performed to determine the allowable leakage path area. Consistent with the above description, this analysis assumed that plant operator initiates and completes a normal plant shutdown (at a rate of 55.6°C/h) in 6 hours, and there is continuous steam bypass leakage over the entire 6-h period. A stratified atmosphere model, which assumed steam only flow through the leakage path, was used to ensure conservative result. For an added conservatism, no credit for structural heat sinks and actuation of drywell/wetwell sprays was taken.

Simplified end-point calculations were done to determine maximum allowable area of the leakage paths. Key steps included in this procedure are:

(1) Compute, M_{NC}, mass of noncondensable gas initially in the drywell and the wetwell gas space.

(2) Compute, ΔP_V, pressure difference between drywell and wetwell gas space needed to keep water level depressed to the top of upper row of vents.

(3) Compute, P_{WM}, the maximum allowable pressure in the wetwell gas space.

$$P_{WM} = [P_{DES} - \Delta P_V],$$

where

P_{DES} is the containment design pressure.

(4) Compute $(P_{WM})_{AIR}$, and $(P_{WM})_{STEAM}$ components of P_{WM}. Assume that wetwell gas space temperature is equal to accident maximum pool temperature, and there is complete carryover of drywell noncondensable gas into the wetwell gas space.

$$P_{WM} = [(P_{WM})_{AIR} + (P_{WM})_{STEAM}]$$

(5) Compute, M_S, mass of steam corresponding to $(P_{WM})_{STEAM}$. This defines allowable steam bypass leakage mass into the wetwell gas space.

(6) Compute leakage path flow rate of steam, M_{dot}, as follows:

$$M_{dot} = [(A/\sqrt{K})\sqrt{(2g_c(\Delta Pv)/v)}]$$

where

v = drywell steam specific volume, and

K = total loss coefficient of the flow path.

(7) Compute the maximum allowable leakage path area, $A/\sqrt{K}$, as follows:

$$A/\sqrt{K} = [(M_{dot})/\sqrt{2g_c(\Delta Pv)/v}]$$

$$= [(M_S/\Delta t)/\sqrt{2g_c(\Delta P_V)/v}]$$

where

Δt = Accident duration

Using the procedure outlined above and assuming an accident duration of 6 hours, the maximum allowable leakage path area under these circumstances is determined to be an effective flow area $A/\sqrt{K}$ of 5 cm^2. See Appendix 6E for additional bypass considerations.

6.2.1.1.5.4 Bypass Capability With Containment Spray and Heat Sinks

An analysis has been performed which evaluates the bypass capability of the containment for a spectrum of break sizes considering containment sprays and containment structural heat sinks as means of mitigating the effects of steam bypass of the suppression pool.

The containment system design provides two RHR spray loops, and each loop consists of both wetwell and drywell sprays. In operation of RHR in spray mode, the wetwell and drywell sprays activate simultaneously. Per loop, the design flow rate of drywell spray is about 800 m^3/hour, and that of wetwell spray is about 114 m^3/hour. In this analysis it is assumed that spray is to be initiated no sooner than 30 minutes after the wetwell gas space pressure is reached to 103.0 kPaG. This assumed value of spray initiation pressure set point, which is higher than the EPGs pressure set point of 71.6 kPaG, is expected to produce slightly conservative results. The suppression pool water passes through the RHR heat exchanger and is then injected into the drywell and wetwell spray headers located respectively in the upper region of drywell and wetwell gas space. The spray will rapidly condense the stratified steam, creating a homogeneous air-steam mixture in the containment. Structural heat sinks (drywell and wetwell boundary surfaces) were considered with variable convective heat transfer coefficients based on Uchida correlation. The reactor vessel shutdown rate was assumed to be 55.6°C/h, and the maximum design service water temperature was used. This shutdown rate corresponds to the maximum rate which does not thermally cycle the reactor vessel. This analysis results in an allowable maximum steam bypass leakage capability of $A/\sqrt{K}$ of 50 cm^2, meeting the criterion that calculated maximum containment pressure remain below the containment design pressure. Allowable leakage capacity vs primary system break area is shown in Figure 6.2-42.

The key assumptions for allowable steam bypass calculations utilizing structural heat sinks are summarized as follows:

(1) Following the occurrence of a pipe line break within the drywell, air is purged through the vents into the wetwell.

(2) Flow through the postulated leakage path is pure steam. For a given leakage path, if the leakage flow consists of mixture of liquid and vapor, the total leakage mass flow rate is higher, but the steam flowrate is less than for the case of pure steam leakage. Since the steam entering the wetwell air space results in the additional pressurization, this is considered as a conservative assumption.

(3) The containment sprays are manually actuated 30 minutes after the wetwell airspace pressure reaches to 103.0 kPaG.

(4) Credit for wetwell spray only was taken. Considering that wetwell spray is more effective in mitigating consequences of steam bypass leakage, credit for drywell spray was not taken to produce conservative results.

(5) The efficiency of the sprays is dependent upon the local steam-to-air ratio. A conservative constant value of 0.7 was used in this analysis.

(6) Heat is transferred to exposed drywell/wetwell concrete walls (with steel liner) in the drywell and wetwell gas space regions. The Uchida convective heat transfer coefficients used are based on the local steam-to-air ratio.

(7) No energy is assumed to leave the containment except through the RHR heat exchangers.

The following is an illustration of the methods employed in calculating steam condensing capability under typical post-LOCA conditions. The condensation capability is calculated using the following equation:

$$M_c = M_s \text{ x } N_s \text{ x } [(T_c - T_s)/H_{fg}] \text{ x } C_p$$

where

M_c = steam condensation rate

M_s = spray flow rate

N_s = spray efficiency

T_c = containment temperature

T_s = spray temperature at the spray nozzles

H_{fg} = latent heat of vaporization of water

C_p = constant pressure specific heat of water

The spray water temperature is calculated from:

$$T_s = T_p - KHX \text{ x } [(T_p - T_{sw}) / (M_s \text{ x } C_p)]$$

where

T_p = suppression pool temperature

KHX = RHR heat exchanger effectiveness

T_{sw} = service water temperature

Containment sprays have a significant effect on the allowable steam bypass capability. Use of sprays increases the maximum allowable bypass leakage by an order of magnitude and represents an effective backup means of condensing bypass steam. See Appendix 6E for additional bypass consideration.

6.2.1.1.5.5 Suppression Pool Bypass During Severe Accidents

The only mode of suppression pool bypass that presents any significant risk during a severe accident is vacuum breaker leakage. Vacuum breaker leakage results in the passage of gas from the drywell into the wetwell airspace. Vapor suppression and fission product scrubbing by the suppression pool are not available to the gas and vapor which pass through the vacuum breakers. The consequences associated with vacuum breaker leakage can be mitigated by use of containment sprays.

Large amounts of leakage can occur as a result of catastrophic failure of valve components or a valve sticking open. Lesser amounts of leakage can result from normal wear and tear including degradation of the valve seating surfaces. For sufficiently large amounts of leakage during a severe accident without containment heat removal, the time to COPS activation or containment overpressurization can be reduced and the amount of fission products released can be increased.

The probability that the vacuum breakers will leak or stick open will be minimized by using materials selected for wear resistance and using high quality seating surfaces. Additionally, the position switches which provide annunciation in the control room can sense a gap between the disk and the seating surface. If the gap is less than 9 mm, aerosols generated as a result of core damage can form a plug and terminate bypass flow. The severe accident analysis assumes the position switch can sense this gap.

6.2.1.1.5.6 Justification for Deviation From SRP Requirements

6.2.1.1.5.6.1 Actuation of Wetwell Sprays

It is recognized that provision of manual, and not automatic, spray actuation of wetwell sprays in the ABWR design is a deviation from the SRP requirement (Appendix A to SRP Section 6.2.1.1.C) of automatic actuation of sprays. The SRP states that the wetwell spray should be automatically actuated 10 minutes following a LOCA signal and an indication of pressurization of the wetwell to quench steam bypassing the suppression pool. However, in determining maximum allowable steam bypass leakage area for ABWR design, analyses assume and take credit for operator actuation of wetwell sprays 30 minutes (instead of 10 minutes) following a LOCA signal and after the wetwell gas space pressure reaches to 103.0 kPaG, though ABWR EPGs permit actuation of wetwell sprays when wetwell gas space pressure reaches to 71.6 kPaG.

Given this conservative analysis assumption, provision of manual wetwell spray actuation is considered sufficiently adequate to provide mitigation for consequences due to steam bypass

leakage during a LOCA event. There appears to be no technical merit in upgrading the current ABWR design to facilitate automatic actuation of wetwell sprays. Therefore, this deviation from the SRP requirement of automatic actuation of wetwell sprays is considered technically adequate. Plant safety is not compromised by providing manual, and not automatic, actuation of wetwell sprays in the current ABWR design.

6.2.1.1.5.6.2 Not Used

6.2.1.1.5.6.3 Vacuum Valve Operability Tests

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

6.2.1.1.5.7 Bypass Leakage Tests and Surveillance

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

6.2.1.1.5.7.1 High-Pressure Leak Test

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

6.2.1.1.5.7.2 Low-Pressure Leak Test

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

6.2.1.1.5.7.3 Acceptance Criteria for Leakage Tests

The acceptance criteria for both high- and low-pressure leakage tests shall be a measured bypass leakage area $A/\sqrt{K}$ which is less than 10% of the suppression pool steam bypass capability $A/\sqrt{K}$ specified in Subsection 6.2.1.1.5.4.

6.2.1.1.5.7.4 Surveillance Test

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

6.2.1.1.5.8 Vacuum Relief Valve Instrumentation and Tests

6.2.1.1.5.8.1 Position Indicators and Alarms

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

6.2.1.1.5.8.2 Vacuum Valves Operability Tests

As described in 6.2.1.1.5.7.4, the vacuum relief valves will be tested for free movement during each refueling outage. There will be no operability tests at monthly intervals, see Subsection 6.2.1.1.5.6.3 for justification.

6.2.1.1.6 Suppression Pool Dynamic Loads

During a LOCA and events such as SRV actuation, steam released from the primary system is channeled into the suppression pool where it is condensed. These actuation events impose hydrodynamic loading conditions on the containment system structures. The containment and its internal structures are designed to withstand all loading conditions associated with these events. These hydrodynamic loads are combined with those from the postulated seismic events in the load combinations specified Subsections 3.8.1.3 and 3.8.3.3. A detailed description and definition of hydrodynamic loading conditions for structure design is provided in Appendix 3B. These loading conditions are briefly summarized in the following paragraphs.

6.2.1.1.6.1 LOCA Loads

During a postulated loss-of-coolant accident (LOCA) inside the drywell, wetwell region will be subjected to the following three sequential hydrodynamic loading conditions of significance to structure design:

- Pool Swell loads
- Condensation Oscillation (CO) loads
- Chugging (CH) loads

Following a postulated LOCA and after the water is cleared from the vents, air/steam mixture from the drywell flows into the suppression pool creating a large bubble at vent exit as it exits into the pool. Bubble at vent exit expands to suppression pool hydrostatic pressure, as the air/steam mixture flow continues from the pressurized drywell. The water ligament above the expanding bubble is accelerated upward which gives rise to pool swell phenomena lasting, typically for a couple of seconds. During this pool swell phase, the wetwell region is subjected to:

(a) loads on suppression pool boundary and drag loads on structures initially submerged in the pool

(b) loads on wetwell gas space

(c) impact and drag loads on structures above the initial pool surface

The CO period of a postulated LOCA follows the pool swell transient period. During the CO period the steam condensation process at the vent exit induces periodic transient loads on the suppression pool boundary and structures initially submerged in the pool. Figure 6.2-43 shows a typical CO loading condition.

The CH period of a postulated LOCA follows the CO period, and it occurs during periods of low vent steam mass flux. During the chugging period the steam condensation process at the vent exit imposes loads on the suppression pool boundary and structures initially submerged in the pool. Figure 6.2-44 shows a typical CH loading condition.

6.2.1.1.6.2 SRV Actuation Loads

During the actuation of SRV, air (initially contained in the SRV discharge line) after it exits into the suppression pool and oscillates as Rayleigh bubble while rising to the pool free surface. The oscillating air bubble produces hydrodynamic loads on the pool boundary and drag loads on structures submerged in the pool. After the air has been expelled, steam exits steadily and condenses in the pool. This condensing steady SRV steam flow has been found to produce negligible loading on the pool boundary. Figure 6.2-45 shows a typical graphical representation of the dynamic loading due to SRV actuation.

6.2.1.1.7 Asymmetric Loading Conditions

Asymmetric loads are included in the load combination specified in Subsections 3.8.1.3, 3.8.2.3 and 3.8.3.3. The containment and internal structures are designed for these loads within the acceptance criteria specified in Subsections 3.8.1.5, 3.8.2.5 and 3.8.3.5. Since internal

structures are not subject to external design or tornado winds, they are not designed for these loads.

Localized pipe forces, pool swell and SRV actuation are asymmetric pressure loads which act on the containment and internal structure (see Subsection 6.2.1.1.6 for magnitudes of pool swell and SRV loads).

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. The magnitude of these loads is discussed in Section 3.7.

6.2.1.1.8 Containment Environment Control

The drywell ventilation system maintains temperature, pressure and humidity in the containment and its subcompartments at the normal design conditions. The safety-related containment heat removal systems described in Subsection 6.2.2 maintain required containment atmosphere conditions during accidents. Since the loss of the drywell ventilation system does not result in exceeding the design environmental conditions for the safety-related equipment inside the containment, the drywell system is not classified as safety-related.

6.2.1.1.9 Post-Accident Monitoring

Refer to Subsections 6.2.1.7, 7.2, 7.3, 7.5, and 7.6.1 for discussion of instrumentation inside the containment which may be used for monitoring various containment parameters under post-accident conditions.

6.2.1.1.10 Severe Accident Considerations

6.2.1.1.10.1 Overall Containment Performance

The containment structure provides for holdup and delay of fission product release should a core damage event occur. Core damage can only occur when all sources of core cooling are lost. Containment leakage during a severe accident is expected to be the same magnitude as the allowable containment leakage.

Long term containment pressurization is governed by the generation of decay heat and non-condensable gases. The primary source of non-condensable gas generation is metal-water reaction of the zirconium in the core. Non-condensable pressure buildup is accommodated by a relatively large containment volume and a high containment pressure capability. The steam produced by decay heat is absorbed in the suppression pool resulting in a very slow containment pressurization and ample time for fission product removal.

The limiting pressure bearing structure in the containment boundary is the drywell head. The Service Level C drywell head allowable pressure is 666.9 kPaG. This pressure capability is adequate to withstand the non-condensable gasses generated by reacting 100% of the zirconium in the active fuel region of the core with water. The median ultimate strength of the containment

is 921.8 kPaG. Ultimate strength capability is important for very rapid containment challenges such as direct containment heating and rapid steam generation. Evaluation of both of these phenomena indicate early containment failure is unlikely. Containment failure due to slower pressurization challenges are largely prevented by the Containment Overpressure Protections System as described in Subsection 6.2.5.2.6.

6.2.1.1.10.2 Inerted Containment

One of the important severe accident consequences is the generation of combustible gasses. Combustion of these gasses could increase the containment temperature and pressure. The containment will be inerted during operation to minimize the impact from the generation of these gasses.

6.2.1.1.10.3 Lower Drywell Design

The design details of the lower drywell are important in the containment response to a severe accident. The key features are as described below.

(1) Sacrificial Concrete

The floor of the lower drywell includes a 1.5 m layer of concrete above the containment liner. This is to ensure that debris will not come in direct contact with the containment boundary upon discharge from the reactor vessel. This added layer of concrete will protect the containment from possible early failure.

(2) Basaltic Concrete

The sacrificial concrete in the lower drywell will be a low gas content concrete. The selection of concrete type is yet another example of how the ABWR design has striven not only to provide severe accident mitigation, but to also address potential uncertainties in severe accident phenomena. Here, the uncertainty is whether or not the ex-vessel core debris can be cooled by flooding the lower drywell. For scenarios in which water in the lower drywell is unable to cool the core debris, the concrete type selected has approximately 4 weight percent calcium carbonate which will result in a very low gas generation rate. This translates into a long time to pressurize the containment. This is important because time is one of the key factors in aerosol removal.

(3) Pedestal

The pedestal is formed by two concentric steel shells with webbing between them. The space between the shells is filled with concrete. The thickness of the concrete between the shells is 1.64 m. A parametric study of core concrete interaction was performed which indicated a very small potential for pedestal failure in the event of

continued interaction. Furthermore, any potential failure will not occur for approximately one day.

(4) Sump Protection

The lower drywell sumps are protected by corium shields such that core debris will not enter them. This maximizes the upper surface area between the debris and the water and maximizes the potential to quench the core debris. The shields are made of alumina which is impervious to chemical attack from core-concrete interaction. The walls of the floor drain sump shield have channels which permit water flow, but which will not permit debris flow. The equipment drain sump shield has no such channels. The height and depth of the shields has been specified to ensure the debris will not enter the sumps in the long term. Further discussion of the corium shields may be found in Subsection 6.2.1.1.10.4

(5) Floor Area

The floor area of the lower drywell has been maximized to improve the potential for debris cooling. The lower drywell floor has an area of 79 m^2 available for core debris spreading which meets the ALWR Utility Requirements Document criterion of 0.02 m^2/MWt.

There may be a limited amount of metallic structures in this area, but this will not limit debris spreading or coolability due to the relatively low melting point of metals and their high thermal conductivity.

(6) Wetwell-Drywell Connecting Vents

The flow area between the lower and upper drywell is adequate to allow venting of gases generated in the lower drywell. The connecting vents flow area is 11.25 m^2. This is important when considering the steam generation rates associated with fuel-coolant-interactions in the lower drywell.

The presence of a significant amount of water in the lower drywell prior to a presumed vessel failure could lead to an increased risk of a steam explosion. After core debris enters the lower drywell, overflow of the suppression pool can prevent or mitigate core-concrete interaction. The interconnection between the lower drywell and the wetwell is at elevation -4.55 m, 8.6 m above the floor of the suppression pool. Thus, approximately 7.2E5 kg of water must be added from outside the containment for the suppression pool to overflow into the lower drywell.

The path from the lower to the upper drywell includes several 90 degree turns. This tortuous path enables core debris to be stripped from the gas flow prior to transport into the upper drywell minimizing the consequences from high pressure melt ejection. Also important when considering high pressure core melt scenarios, the

configuration of the connecting vents will result in the transport of some core debris directly into the suppression pool. This is preferable to transport into the upper drywell and would result in the debris being quenched with only a slight increase in the suppression pool temperature.

(7) Solid Vessel Skirt

The vessel skirt does not have any penetrations which would allow the flow of water from the upper drywell directly to the lower drywell. This, in combination with the other design feature described above, ensures a very low probability that water is in the lower drywell before the time of vessel failure. Thus, large scale fuel-coolant interactions are precluded.

6.2.1.1.10.4 Corium Shield

During a hypothetical severe accident in the ABWR, molten core debris may be present on the lower drywell floor. The EPRI ALWR Requirements Document specifies a floor area of at least 0.02 m^2/MWt to promote debris coolability. This has been interpreted in the ABWR design as a requirement for an unrestricted LD floor area of 79 m^2.

The ABWR has two drain sumps in the periphery of the LD floor which could collect core debris during a severe accident if ingression is not prevented. If ingression occurs, a debris bed will form in the sump which has the potential to be deeper than the bed on the LD floor. Debris coolability becomes more uncertain as the depth of a debris bed increases.

The two drain sumps have different design objectives. One, the floor drain sump, is designed to collect any water which falls on the LD floor. The other, the equipment drain sump, collects water leaking from valves and piping. Both sumps have pumps and instrumentation which allow the plant operators to determine water leakage rates from various sources. Plant shutdown is required when leakage rate limits are exceeded for a certain amount of time. A more complete discussion on the water collection system can be found in Subsection 5.2.5.

A protective layer of refractory bricks — a corium shield — will be built around the sumps to prevent corium ingression. The shield for the equipment drain sump (EDS) is solid except for the inlet and outlet piping which goes through its roof. The shield for the floor drain sump(FDS) is similar except that it has channels at floor level to allow water which falls onto the LD floor to flow into the sump. The length of the channels will be long enough so that any molten debris which reaches the inlet will freeze before it exits and spills into the sump. The width and number of the channels will be chosen so that the required water flow rate during normal reactor operation is achievable.

The solid walls of the sump shields only have to be thick enough to withstand ablation, if any is expected to occur for the chosen wall material. The walls of the FDS shield with channels in them must be thicker so that molten debris flowing through the channels has enough residence time to ensure debris solidification.

Both shields extend above the LD floor to an elevation greater than the expected maximum height of core debris. Thus, no significant amount of debris will collect on the shield roofs. The walls of both shields extend below the LD Floor to prevent debris from tunneling under the walls and entering the sumps.

Both shields have provisions in their roofs to allow water to flow into the sumps when the lower drywell is flooded. The provisions are located next to the pedestal wall so that the debris which relocates from the vessel can not directly enter the sumps due to geometrical constraints. Additionally, the provision for the roof of the EDS shield wall not affect the normal water collection capabilites of the EDS.

To prevent the debris which falls on the lower drywell floor from directly entering the FDS shield, the channels in the FDS shield are in the walls which face away from the center of the lower drywell. The FDS shield wall which faces the center of the lower drywell is solid and does not contain any channels.

6.2.1.1.10.4.1 Success Criteria

The shield walls must satisfy the following requirements:

(1) Melting Point of Shield Material Above Initial Contact Temperature

The shield wall material will have a melting temperature that is greater than the interface between the debris and the shield wall.

(2) Channel Length

The length of the channels in the FDS shield must be long enough to ensure that a plug forms in the channel before debris spills into the sump. The freezing process is expected to take on the order of seconds or less to complete.

(3) Shield Height Above Lower Drywell Floor

The shield height above the lower drywell floor ensures long term debris solidification and prevents debris from collecting on top of the shields. The freezing process will be complete during the time frame when the shield walls are behaving as semi-infinite solids.

(4) Shield Depth Below Lower Drywell Floor

The shield walls extend to the floors of the sumps to prevent tunneling of debris into the sumps.

(5) Chemical Resistance of Shield Walls

The wall material chosen for the corium shields must have good chemical resistance to siliceous slags and reducing environments. Resistance can be determined to a first degree by comparing the Gibb's free energy of the oxides which make up the shield wall and the oxides present in core debris.

(6) Channel Flow Area

The total flow area of the shield channels shall be great enough to allow water flow rates stated in the Technical Specifications without causing excessive water pool formation in the lower drywell.

(7) Seismic Adequacy

The seismic adequacy of the corium shields will be determined in the detailed design phase. Adequacy should be easily met because the shields are at the lowest point in the containment. Missile generation is not an issue because the shields are not near any vital equipment.

(8) Channel Height

The channel height is small enough to ensure freezing.

6.2.1.1.10.4.2 Corium Shield Design

The corium shields are constructed of alumina. The height of the shields above the floor is 0.4m. The floor drain sump has channels 1 cm high. The channel length must be at least 0.5 m. The channel walls extend to the floor of the sumps.

6.2.1.1.10.4.3 Design Evaluation

Alumina has a melt point which is greater than the contact temperature of the core debris. It is resistant to reduction reactions with the metals which make up core debris. The height of the shields meets the requirements to ensure long term debris solidification and to prevent material from accumulating on the roof of the shields. Similarly, the depth of the channeled wall will ensure long term debris solidification. The height and length of the channel for the floor drain sump will ensure debris freezing.

The details of the analyses leading to these conclusions may be found in Appendix 19ED.

6.2.1.2 Containment Subcompartments

6.2.1.2.1 Design Bases

The design of the containment subcompartments is based upon the 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 into an analysis to obtain the pressure/temperature transient response for each subcompartment.

6.2.1.2.2 Design Features

The upper drywell, lower drywell and wetwell subcompartment volumes are covered in depth in Subsection 6.2.1.1. The remaining containment subcompartment volumes are:

(1) **Drywell Head Region**—The drywell head region is covered with a removable steel head which forms part of the containment boundary. The drywell bulkhead connects the RPV flange to the containment and represents the interface between the drywell head region and the drywell.

The DBA for the drywell head region is the double-ended circumferential break of the 150A RPV head spray line of the CUW System at the connection to the RPV head nozzle. The other high-energy line in the drywell head region is the 50A main steam vent line. The RPV head spray line is chosen as the DBA for this subcompartment due to the higher mass and energy release rates from a postulated break of this line.

(2) **Reactor Shield Annulus**—The reactor shield annulus exists between the reactor shield wall (RSW) and the RPV. The RSW is a concrete cylinder surrounding the RPV. The reactor shield wall is supported by the reactor pedestal and extends to a height 0.1m below the containment top slab.

Several high-energy lines (such as, main steam lines, feedwater lines, RHR shutdown cooling suction lines, HPCF and LPFL injection lines, etc.) connect to the RPV and extend through the reactor shield wall. There are penetrations for other piping, vents and instrumentation lines and personnel access holes in the shield wall.

In order to determine transient pressure loading inside the annulus for structure design evaluation, a double-ended pipe break in high-energy lines at vessel nozzle safe end inside the annulus was postulated. A double-ended pipe break at the RHR shutdown cooling suction nozzle (representing a flow area of about 750 cm^2) was determined to be the DBA for the reactor shield annulus sub-compartment pressurization. This break type resulted in the largest mass and energy blowdown inside the annulus. No main steam line break inside the annulus was postulated, because the RPV steam outlet nozzle safe end connection to the main steam line is outside the annulus region.

Transient pressure loading condition inside the annulus due to the DBA pipe break was determined. The total venting flow area from the annulus region to the outside drywell region which was assumed and used in this calculation, comprised of

(a) the annular clearance area due to the 0.1m clearance between the top of the shield wall and containment top slab, and

(b) the combined area of penetration door openings.

6.2.1.2.3 Design Evaluation

The reactor shield wall structure, and the reactor pressure vessel and its internals design considered and accounted for the transient pressure loads due to the DBA pipe break inside the reactor shield annulus.

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

The environmental conditions created by any high-energy line break (HELB) are analyzed according to Regulatory Guide 1.89. The first step in such analysis is to calculate the mass and energy release rate from the high-energy line break (HELB).

Figure 6.2-3 and 6.2-4 show the break flow rate and specific enthalpy for the feedwater line break flow coming from the feedwater system side. Figure 6.2-23 shows the same information for the feedwater line break flow coming from the RPV. Figure 6.2-25 shows the same information for the main steamline break flow.

When the break size is small and the reactor pressure stays at approximately 7.31 MPaG, the critical flow table (such as in Reference 6.2-2) may be used. If the long-term performance is required or the break size is in the intermediate break range, then the reactor pressure does not stay constant. In this case, the transient is analyzed by using computer codes to determine the mass and energy release rate based on Reference 6.2-5.

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

Not Applicable

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

Not Applicable

6.2.1.6 Testing and Inspection

6.2.1.6.1 Preoperational Testing

Preoperational testing and inspection programs for the containment and associated structures, systems and components are described in Subsections 3.8.1.7, 3.8.2.7, 3.8.3.7, 6.2.6 and

Chapter 14. These programs demonstrate the structural integrity and desired leaktightness of the containment and associated structures, systems, and components.

6.2.1.6.2 Post-Operational Leakage Rate Test

For descriptions of the containment integrated leak rate test (ILRT) and other post-operational leakage rate tests (10CFR50, Appendix J, Tests Type A and B), see Subsection 6.2.6.

Accessible portions of the WDVBS will be visually inspected at each refueling outage to determine that they are free of foreign debris. The maximum allowable leakage for each valve shall be per ASME OM Code, Appendix I of 10CFR50.

6.2.1.6.3 Design Provisions for Periodic Pressurization

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

6.2.1.7 Instrumentation Requirements

Instrumentation is provided to monitor the following containment parameters:

(1) Drywell temperature

(2) Drywell pressure

(3) Differential pressure both drywell-to-wetwell and drywell-to-Reactor Building

(4) Drywell oxygen concentrations

(5) Drywell radiation levels

(6) Wetwell air space temperature

(7) Wetwell pressure

(8) Differential pressure between the wetwell and Reactor Building

(9) Wetwell oxygen concentrations

(10) Wetwell radiation levels

(11) Suppression pool temperature

(12) Suppression pool level

(13) Separate inerting flow indication to both the drywell and wetwell

(14) Drywell pressure and nitrogen makeup flow monitoring and recording

(15) Wetwell nitrogen makeup flow

(16) Open/close position indicators for wetwell-to-drywell vacuum breaker valves, and alarm in control room.

Drywell pressure is an input to containment isolation, ADS, HPCF, RCIC, RHR Division A, B and C initiation logic. The logic circuitry is located in the control room. Pressure indicators for both the drywell and wetwell are part of the containment inerting system, which maintains containment at a pressure higher than the secondary containment pressure.

Wetwell-to-drywell differential pressure is monitored for proper functioning of the WDVBS.

Drywell space temperatures are inputs to the Leak Detection and Isolation System (LDS). Thermocouples are mounted at appropriate elevations of the drywell to monitor the drywell temperatures. Temperature, pressure and radiation are monitored for environmental conditions of equipment in the containment during normal, abnormal and accidental conditions.

Four 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 and alarms are located in the control room. The sensors are divided into subgroups for normal indications as an input to RHR initiation logic and for post-LOCA pool monitoring.

Two oxygen analyzers are provided for the drywell, and two for the wetwell. Each analyzer draws a sample from an appropriate area of the drywell or wetwell. Oxygen concentration alarms and recorders are located in the control room.

Radiation detectors in the drywell and wetwell areas provide inputs to the radiation monitors, recorders and high level alarms located in the control room.

Refer to Section 7.2 for a description of drywell pressure as an input to the Reactor Protection System, and Section 7.3 for a description of drywell pressure, wetwell-to-drywell differential pressure and suppression pool level as inputs to the ESF systems. Suppression pool temperature monitoring, drywell temperature monitoring and inerting flow indication and makeup flow monitoring and recording are discussed in Section 7.6. 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.

Containment design features as related to debris formation have an important relationship to the ECCS's ability to provide containment cooling. A primary source of debris in containment is the thermal insulation. If insulation is dislodged and enters the wetwell, it can cause plugging of the ECCS suction strainers, which can impede ECCS performance and containment cooling.

The ABWR design includes the necessary provisions to prevent debris from impairing the ability of the RCIC, HPCF, and RHR systems to perform their required post-accident functions. Specifically, the ABWR does the following:

(1) The design is resistant to the transport of debris to the suppression pool.

(2) The SPCU system will provide early indication of any potential problem.

(3) The ECCS suction strainers meet the current regulatory requirements unlike the strainers at the incident plants.

(4) The equipment installed in the drywell and wetwell minimize the potential for generation of debris.

In addition to the ABWR design features, the control of the suppression pool cleanliness is a significant element of minimizing the potential for strainer plugging.

6.2.1.7.1 Suppression Pool Cleanliness Program

6.2.1.7.1.1 Purpose

This operational program is to ensure that the primary containment is free from debris that could become dislodged in an accident and be transported to the ECCS suction strainers and interfere with their proper functioning during a design basis event.

6.2.1.7.1.2 Scope

This program applies to the primary containment, including the drywell and suppression pool, for the ABWR. This program has design, maintenance and operational elements. This program is comprised of: (1) design change control to ensure that material whose susceptibility to damage resulting in uncontrolled debris is limited and cannot be replaced with material with greater susceptibility: (2) restricted access to primary containment during reactor operations and refueling periods: (3) suppression pool cleanup system operation to maintain Suppression Pool (S/P) cleanliness; (4) foreign material exclusion and housekeeping requirements to ensure that foreign material that could be detrimental to ECCS strainer operation if left in primary containment is removed prior to containment close out; and (5) drywell, S/P, and strainer inspections following outages to ensure that no debris is present prior to the containment being closed out in preparation for operation.

The program is based on ABWR Operating Experience, Electric Power Research Institute (EPRI) guidelines contained in EPRI TR 1016315, "Nuclear Maintenance Applications Center:

Foreign Material Exclusion Guidelines" and Institute of Nuclear Power Operations (INPO) guidance in INPO 07-008, "Guidelines for Achieving Excellence in Foreign Material Exclusion (FME)."

6.2.1.7.1.3 Responsibilities

The operations and maintenance organizations have overall responsibility for the procedures that implement this program. There is a suppression pool cleanliness program owner, whose responsibility is to have overview of all aspects of this program, including reviewing procedures, training station personnel, being aware of industry operating experience, and on an ongoing basis assessing the overall effectiveness of the program.

6.2.1.7.1.4 Standards

There will be no fibrous or calcium silicate insulation inside the primary containment. All insulation will be RMI-type which will not pass through the ECCS suction strainers. Design change control will ensure that the RMI is not replaced with fibrous or calcium silicate insulation.

The primary containment will be designated as a Foreign Material Exclusion (FME) Zone 1 in accordance with the INPO Definition. This is an area where loss of FME could result in personnel injury, nuclear fuel failure, reduced safety system or station availability, or an outage extension or significant cost for recovery and is the highest level of FME defined by INPO. All activities associated with suppression pool cleanliness will be done in accordance with a Quality Assurance Program.

The S/P cleanup system will be operated as necessary to maintain the water chemistry in the S/P comparable to that required for refueling water.

The primary containment atmosphere is inerted during reactor operations. Therefore, access to the primary containment is effectively prohibited.

6.2.1.7.1.5 Key Elements of the Suppression Pool Cleanliness Program

During refueling outages, the containment is a FME Zone 1 area. In addition, strict house keeping controls are in place to ensure that only needed material is brought into containment and that work areas are restored to their original conditions following completion of the work. Prior to entry into the containment during scheduled or unscheduled outages, all material will be accounted for and documented.

Following each refueling outage, a detailed visual inspection is performed of the primary containment to identify and remove any loose debris. This detailed inspection is controlled by plant procedures in accordance with the Procedure Development Program. All debris identified will be documented and entered into the corrective action program for trending and potential action.

In addition a remote visual inspection will be performed of the Residual Heat Removal (RHR), Reactor Core Isolation Cooling (RCIC), and High Pressure Core Flooder (HPCF) suction strainers and the S/P floor to ensure there is no debris present. This inspection will be focused on the presence of debris in the suction strainers but will also look for any structural gaps that would allow debris to bypass the strainer flow holes. Results of these inspections will be documented in the procedure and in the corrective action program. Debris that is identified will be removed and any strainer structure gaps will be assessed and repaired if necessary.

The S/P cleanup system will normally be operated in alignment with a train of the fuel pool cleanup filter/demineralizers to ensure S/P water quality. Floating debris and sediment in the suppression pool not removed by the Suppression Pool Cleanup System will be removed during refueling outages.

In the unlikely event of a primary containment entry during the operating cycle, a close-out inspection will be performed prior to the return to operation.

6.2.1.7.1.6 Acceptance Criteria

Procedures related to suppression pool cleanliness will have defined acceptance criteria that must be met prior to closing containment and returning to power. Acceptance criteria will be absence of debris in the primary containment non suppression pool areas. For the strainers themselves, the acceptance criteria will be that the strainer inlets are not restricted, the strainer screens are not plugged, and the strainer structure does not have any structural gaps. For the suppression pool, the acceptance criteria will be the absence of debris and sediment.

There is a documented close-out of containment following completion of all cleanliness inspections and prior to resumption of power operation.

6.2.1.7.1.7 Procedural Controls

Station procedures that implement the suppression pool cleanliness program will be developed in accordance with the Procedure Development Plan described in Section 13.5. These procedures will address control of materials, access to the containment, inspection and cleanup of containment, inspection and cleanup of the strainers, and inspection and cleanup of the suppression pool.

6.2.1.7.1.8 Implementation

The suppression pool cleanliness program will be implemented prior to the initiation of the startup test program.

6.2.1.7.1.9 Corrective Action Program

Adverse conditions from the containment and strainer inspections will be documented in the corrective action program to ensure they are properly addressed and to allow trending and analysis of results.

6.2.1.7.1.10 Audits

Periodic audits will be performed on this program.

6.2.1.7.1.11 Operating Experience

Operating experience at other plants will be periodically assessed for lessons learned that could be applied to the program.

6.2.2 Containment Heat Removal System

6.2.2.1 Design Bases

The Suppression Pool Cooling mode and the wetwell and drywell spray features of the Containment Heat Removal System (CHRS)are integral parts of the RHR System. The purpose of the CHRS is to prevent excessive containment temperatures and pressures, thus maintaining containment integrity following a LOCA. To fulfill this purpose, the CHRS meets the following safety design bases:

(1) The system limits the long-term bulk temperature of the suppression pool to 97.2°C when considering the energy additions to the containment following a LOCA. These energy additions, as a function of time, are provided in the previous section.

(2) The single-failure criterion applies to the system.

(3) The system is designed to safety grade requirements, including the capability to perform its function following a safe shutdown earthquake (SSE).

(4) The system maintains operation during those environmental conditions imposed by the LOCA.

(5) Each active component of the system is testable during normal operation of the nuclear power plant.

6.2.2.2 Containment Cooling System Design

The Containment Cooling System (CCS) encompasses several of the RHR operating modes, including the Low Pressure Flooder (LPFL) mode, the Suppression Pool Cooling (SPC) mode, and the Containment Spray modes (drywell and wetwell). Containment cooling starts as soon as the LPFL injection flow begins. The SPC mode cools the containment. The containment sprays cool the drywell and wetwell by condensing steam and the condensate running back into the suppression pool. All water that leaves the suppression pool is cooled by the RHR heat exchangers during the three operational modes indicated above. For each of the three loops, water is drawn from the suppression pool, pumped through an RHR heat exchanger and injected into the reactor vessel for the LPFL mode. Also, for each of the three loops of the SPC mode, water is drawn from the suppression pool, pumped through an RHR heat exchanger and delivered to the suppression pool. On two of the loops (B&C), a portion of the water returned

to the suppression pool may be passed through wetwell spray headers. These two loops also have a manual feature for providing drywell spray. Water from the RCWS is pumped through the heat exchanger shell side to exchange heat with the processed water. Three cooling loops are provided, each being mechanically and electrically separate from the other to achieve redundancy. A piping and instrumentation diagram (P&ID) is provided in Section 5.4. The process diagram, including the process data, is provided for all design operating modes and conditions.

All portions of the CCS mode are designed to withstand operating loads and loads resulting from natural phenomena. All operating components can be tested during normal plant operation so that reliability can be assured. Construction codes and standards are covered in Subsection 5.4.7.

The LPFL mode is automatically initiated from ECCS signals or manually initiated. The SPC mode is started manually or automatically. The RHR System must be realigned for suppression pool cooling by the plant operator after the reactor vessel water level has been recovered. The RHR pumps are already operating. Suppression pool cooling is initiated in any of the three loops by manually closing the LPFL injection valve and opening the pool return valve. For automatic initiation of suppression pool cooling, all three RHR loops are initiated. In the event that a single failure has occurred, and the action which the plant operator is taking does not result in system initiation, then the operator will place the other totally redundant system into operation by following the same initiation procedure. If the operator chooses to utilize the containment sprays, he must close the LPFL injection valves and open the spray valves. The drywell spray mode may be initiated manually only after a high drywell pressure permissive occurs.

Preoperational tests are performed to verify individual component operation, individual logic element operation, and system operation up to the containment spray spargers. A sample of the sparger nozzles is bench tested for flow rate versus pressure drop to evaluate the original hydraulic calculations. Finally, the spargers are tested by air and visually inspected to verify that all nozzles are clear (see Subsection 5.4.7.4 for further discussion of preoperational testing).

6.2.2.3 Design Evaluation of the Containment Cooling System

6.2.2.3.1 System Operation and Sequence of Events

In the event of the postulated LOCA, the short-term energy release from the reactor primary system will be dumped to the suppression pool. Subsequent to the accident, fission product decay heat will result in a continuing energy input to the pool. The RHR SPC mode will remove this energy which is released into the primary containment system, thus resulting in acceptable suppression pool temperatures and containment pressures.

In order to evaluate the adequacy of the RHR System, the following is assumed:

(1) With the reactor initially operating at 102% of rated power, a LOCA occurs.

(2) A single failure of a RHR heat exchanger is the most limiting single failure.

(3) The ECCS flows assumed available are 2 HPCF, 1 RCIC, and 2 LPFL (RHR).

(4) Containment cooling is initiated after 30 minutes.

Analysis of the net positive suction head (NPSH) available to the RHR and HPCF pumps in accordance with the recommendations of Regulatory Guides 1.1 and 1.82 is provided in Tables 6.2-2b and 6.2-2c, respectively.

General compliance for Regulatory Guide 1.26 may be found in Subsection 3.2.2.

6.2.2.3.2 Summary of Containment Cooling Analysis

When calculating the long-term post-LOCA pool temperature transient, it is assumed that the initial suppression pool temperature and the RHR service water temperature are at their maximum values. This assumption maximizes the heat sink temperature to which the containment heat is rejected and thus maximizes the containment temperature. In addition, the RHR heat exchanger is assumed to be in a fully fouled condition at the time the accident occurs. This conservatively minimizes the heat exchanger heat removal capacity. Even with the degraded conditions outlined above, the maximum temperature is maintained below the design limit specified in Subsection 6.2.2.1.

It should be noted that, when evaluating this long-term suppression pool transient, all heat sources in the containment are considered with no credit taken for any heat losses other than through the RHR heat exchanger. These heat sources are discussed in Subsection 6.2.1.3.

It can be concluded that the conservative evaluation procedure described above clearly demonstrates that the RHR System in the SPC mode limits the post-LOCA containment temperature transient.

6.2.2.3.3 Severe Accident Considerations

The containment spray features of the RHR System can reduce the amount of radioactive material released to the environment in the event core damage occurs. The benefits provided by the sprays are condensing steam, scrubbing of fission products in the containment airspace, and supplying water to ex-vessel core debris. The conditions for activation of the containment sprays are described in the Emergency Procedure Guidelines in Appendix 18A.

The water sprayed into the upper drywell absorbs heat from the RPV outer surfaces and the debris which relocates into the upper drywell, if any, upon vessel failure at high pressure. Cooling of the upper drywell prevents any potential for overtemperature failure of seals in the

large operable penetrations (e.g., the drywell head, equipment hatches and personnel airlocks). Water which collects on the upper drywell floor is directed into the wetwell through the connecting vents.

The containment sprays provide significant mitigation of suppression pool bypass. The incoming water absorbs heat and condenses steam. While the heat absorption is not as efficient nor as extensive as what would occur if the suppression pool was not bypassed, the time to Containment Overpressure Protection System (COPS) activation or containment failure can be delayed significantly. This delay results in a significant reduction in the radioactive release due to fission product decay.

The water sprayed into the containment also scrubs fission products which are in the containment airspace. Scrubbing reduces the amount of radioactive material which is available for release from the containment.

6.2.2.4 Test and Inspections

The Containment Cooling System (CCS) is required to have scheduled maintenance. The system testing and inspection will be performed periodically during the plant normal operation and after each plant shutdown. Functional testing will be performed on all active components and controls. The system reference characteristics will be established during preoperational testing to be used as base points for checking measurements obtained from the system tests during the plant operation.

The preoperational test program of the CCS is described in Subsection 14.2.12. The following functional tests will be performed. The RHR pump will be tested through the suppression pool cooling loop operation by measuring flow and pressure. Each pump will be tested individually.

Containment spray spargers will be tested during reactor shutdown by air, and by visual inspection to verify that all the nozzles are clear. RHR heat exchangers will be checked for effectiveness by measuring inlet and outlet temperatures at the tube and shell sides.

All motor- and air-operated valves required for safety are capable of being exercised and their operation demonstrated. The layout and arrangement of critical equipment outside the drywell is designed to permit access for appropriate equipment used in testing and inspecting system integrity. Relief valves on the low pressure lines are removable for testing.

Periodic inspection and maintenance of the main system pumps, pump motors, and heat exchangers are conducted in accordance with the manufacturer’s instructions.

During the normal plant operation, the pumps, heat exchangers, valves, piping, instrumentation, wiring and other components outside the containment can be inspected visually at any time. Testing frequencies are generally correlated with testing frequencies of the associated controls and instrumentation. When a pump or valve control is tested, the operability of that pump or valve and its associated instrumentation is tested by the same action. When a system is tested,

operation of the components is indicated by installed instrumentation. Relief valves are removed as scheduled at refueling outages for bench tests and setting adjustment.

6.2.2.5 Instrumentation Requirements

Details of the instrumentation are provided in Section 7.3. The SPC mode of the RHR System is manually or automatically initiated.

6.2.3 Secondary Containment Functional Design

The secondary containment boundary, as shown in Figure 6.2-26, completely surrounds the primary containment vessel (PCV) except for the basemat and, together with the clean zone, comprises the Reactor Building. The secondary containment encloses all penetrations through the primary containment and all those systems external to the primary containment that may become a potential source of radioactive release after an accident. During normal plant operation, the secondary containment areas are kept at a negative pressure with respect to the environment and clean zone by the HVAC System. Following an accident, the Standby Gas Treatment System (SGTS) provides this function. These systems are described in Subsections 9.4.5 and 6.5.1, respectively.

Fission products that may leak from the primary to secondary containment are processed by the SGTS before being discharged to the environment. The HVAC exhaust systems and SGTS are located within the secondary containment to assure collection of any leakage. The secondary containment provides detection of the level of radioactivity released to the environment during abnormal and accident plant conditions. Personnel or material entrances to the secondary containment consist of airlocks with interlocked doors or hatches.

There are basically three types of potential leakage paths for the release of fission product during and following an accident. These leakage paths are shown in Figure 6.2-27. Potential leak paths that can bypass the secondary containment are shown in Table 6.2-10.

6.2.3.1 Design Bases

(1) Secondary containment is provided to collect fission products which may leak from the primary containment following a DBA. This collection allows filtration by the SGTS prior to release to the environment. The secondary containment region completely surrounds the primary containment vessel.

(2) During a DBA, the secondary containment and supporting systems such as the SGTS, is designed to limit the thyroid and whole body doses to less than 10CFR100 guidelines at the site boundary and low population zone and to less than 10CFR50 Appendix A, General Design Criterion 19, doses for the control room operator.

(3) The mechanical, electrical, instrumentation, and structural components of the secondary containment design are protected, as necessary, from internally- and externally-generated missiles, dynamic effects associated with pipe whip and jet forces, and environmental conditions from an accident, and are designed to Seismic Category I requirements. These items and equipment of supporting systems required to function after an accident are designed for single active failure, loss of offsite power (LOOP) coincident with an accident, 30-day accident duration for radiological analysis and 100-day duration for operational capability. No credit is assumed for non-Seismic Category I piping, power supplies and equipment.

(4) The design allowable inleakage at the secondary containment-environs boundary is within the capability of the SGTS to maintain the pressure inside secondary containment at –6.4 mm water gauge relative to the environs, under design exterior wind conditions. This prevents exfiltration such that 10CFR100 guidelines will not be exceeded following a DBA.

Post-accident pressure transients do not cause 10CFR100 guidelines to be exceeded because of exfiltration.

(5) Automatic shutoff of the normal HVAC air supply and ventilation exhaust and of other systems after a LOCA or on detection of high radiation in effluent prevents airborne leakage from escaping the secondary containment.

(6) All openings through the secondary containment boundary, such as personnel and equipment doors, remain closed after a DBA by interlocks or administrative control. These doors are provided with position indicators and alarms having readout and annunciation capability in the control room.

(7) Liquid leakage from the secondary containment to the clean zone or to the environment is controlled by means of water loop seals, automatic shutoff valves in series, or piping upgrade to safety class.

(8) All operating systems that transport liquid from the secondary containment to the clean zone or the environment are automatically isolated during an accident. These systems cannot be automatically initiated following an accident.

(9) The exhaust unit of the secondary containment HVAC maintains the secondary containment air flow pattern from areas of normally low contamination to those with a potentially high level.

(10) High-energy pipe breaks within the secondary containment are detected by the Leak Detection and Isolation System (LDS). Blowout panels are provided where necessary to relieve the thermal and pressure buildup in the various subcompartments.

(11) All effluents from the secondary containment areas are monitored for gamma radiation level prior to being released to the environment.

(12) Adequate instrumentation and control room indications are provided to monitor all important secondary containment parameters in order to maintain the plant within the Technical Specification limits and provide information for operator actions.

(13) The secondary containment, in conjunction with supporting systems, will be periodically tested to assure that performance requirements can be met.

6.2.3.2 System Design

The secondary containment is a reinforced concrete building that forms an envelope surrounding the PCV above the basemat. The secondary containment has isolation systems on piping, doors and other penetrations. This permits maintaining the secondary containment volume at a slightly negative pressure so all PCV leakage can be collected and treated before release. Details of structural design and arrangement of compartments for various systems are described in Section 3.8.

The boundary of the secondary containment is shown in the following figures:

6.2-28	Containment Boundaries in the Reactor Building-Plan Section A-A(0–180°)
6.2-29	Containment Boundaries in the Reactor Building-Plan Section B-B(90°–270°)
6.2-30	Containment Boundaries in the Reactor Building-Plan at El –8200 mm
6.2-31	Containment Boundaries in the Reactor Building-Plan at El –1700 mm
6.2-32	Containment Boundaries in the Reactor Building-Plan at El –4800/8500 mm
6.2-33	Containment Boundaries in the Reactor Building-Plan at El 12300 mm
6.2-34	Containment Boundaries in the Reactor Building-Plan at El 18100 mm
6.2-35	Containment Boundaries in the Reactor Building-Plan at El 23500 mm

6.2-36 Containment Boundaries in the Reactor Building-Plan at El 31700 mm

Secondary containment design and performance data are provided in Table 6.2-2d.

During normal operation, the secondary containment system is operated at a slightly negative pressure relative to the atmosphere. This assures that any leakage from the primary containment will be collected and can be treated before release if its radioactivity level is above prescribed limits. The secondary containment HVAC System operates on a feed-and-bleed principle with internal recirculation. Air flow is from clean to potentially contaminated areas.

The building effluents are monitored for radioactivity by the Area Radiation Monitoring System (ARMS). If the radioactive level rises above set levels, the secondary containment discharge can be routed through the SGTS for treatment before release. The operation of the secondary containment SGTS and HVAC are discussed in detail in Subsections 6.5.1 and 9.4.5, respectively.

During normal operation, the secondary containment is the envelope that forces collection of airborne radioactive material from fuel storage pools, CUW, FPCCU, SPCU and other potentially radioactive sources in the secondary containment. The HVAC exhaust systems and SGTS are also located within the secondary containment to assure collection of any leakage. The RHR and HPCF pump seals and valve packings and RCIC System components are a potential source of radioactive release and are located within the secondary containment.

During refueling operations, the drywell head is removed and the secondary containment becomes the containment envelope. Therefore, entry into the secondary containment is provided via double door vestibules or, in the case of the main equipment hatch, a double door entry. This assures the integrity of the secondary containment envelope with effluent monitoring and treatment of airborne radioactive material resulting from normal plant or refueling operations or from abnormal events such as a fuel drop accident.

The airborne fission product is contained by maintaining all portions of the secondary containment at a negative 6.4 mm of water gauge relative to the lowest pressure boundary outside the secondary containment. This negative pressure is achieved following an accident by the SGTS.

The airborne fission product leakage from the primary containment is processed by the SGTS. The SGTS achieves a 99.95% removal of halogen (stable and radioactive) and a 99.95% of airborne particulates prior to discharge to the environment. This removal efficiency will be periodically tested in accordance with regulatory requirements. The dose limit evaluation takes credit for 99% airborne halogen and particulates for this type of leakage. A 99% removal credit is allowed even though the design will achieve 99.95% removal capability.

The potentially high level contaminated areas are the following:

(1) CUW System Rooms

(2) RCIC System Room

(3) HPCF System Rooms

(4) Fuel Pool Cooling and Cleanup System Rooms

(5) RHR System Rooms

(6) Suppression Pool Cleanup System Rooms

(7) SGTS Filter Compartments

(8) Spent Fuel Storage Pool

The liquid leakage from the secondary containment to the clean zone or the environment is controlled, as required, by means of water loop seals, automatic shutoff valves in series, or piping upgrade to safety class. All system operations that transport liquid from the secondary containment to the clean zone or to the environment will be automatically shut off during an accident and not be automatically initiated following an accident.

A postulated high-energy pipe break in the secondary containment is accommodated so as not to exceed the environmental qualification limits of the equipment required for plant shutdown. Blowout panels are installed as required in rooms where high-energy pipe breaks are postulated, and the panels relieve the thermal temperature and pressure buildup in the room.

High-energy pipe breaks in the secondary containment will cause a failure to maintain negative pressure in the secondary containment. This is acceptable, since there is no significant release of radioactivity from this accident event because fuel is not damaged and the plant is shut down promptly.

All effluents processed by the SGTS from the secondary containment areas are monitored for gamma radiation level prior to their release to the environment.

The ECCS, RCIC, CUW, FPCCU and SPCU System rooms of the secondary containment collect throughline leakage of fission products. The pump rooms are of reinforced concrete construction (see Subsection 3.8.4 for design details). Following accidents which require secondary containment integrity (i.e., do not open the blowout panels to the steam tunnel), the normal room ventilation subsystems are isolated and the SGTS begins to exhaust the air (through its filter) from the rooms, maintaining the pressure at –6.4 mm water gauge or less with respect to the environs. No mixing of fission products with the room volumes is assumed.

High-energy pipe breaks in the secondary containment compartments do not require secondary containment integrity. Following breaks of this type, pressure will be relieved by blowout vent openings and panels within the secondary containment or to the steam tunnel and Turbine Building.

The fuel storage and handling areas are part of secondary containment, where throughline leakage of fission products is collected. These areas are constructed of reinforced concrete (see Subsection 3.8.4 for design details). The secondary containment boundaries are the concrete walls and ceiling of the refueling floor and the stainless steel-lined upper pool.

Following accidents requiring secondary containment integrity, the normal Reactor Building (R/B) ventilation system is isolated, and the SGTS begins exhausting the secondary containment air. The SGTS thus maintains the pressure at –6.4 mm, or less under all wind conditions up to the wind speed at which diffusion becomes great enough to assure site boundary exposures less than those calculated for DBAs even if exfiltration occurs (See Subsection 6.5.1.3.2). (Above that wind speed, when exfiltration does occur, 10CFR100 guidelines will not be exceeded because of the increased atmospheric dispersion which may be assumed.) No mixing is assumed for fission products within the secondary containment volume. There are no high-energy lines in the fuel handling and storage areas, whose failure would result in pressurization or loss of secondary containment integrity.

Penetrations between secondary containment and the environs are of four different types:

(1) Piping penetrations

(2) Architectural openings (doors, hatches, and blowout panels)

(3) HVAC duct penetrations

(4) Electrical penetrations

Each of these categories is discussed below separately. Most piping which forms a part of the secondary containment boundary is designed to at least Seismic Category I and ASME Section III, Class 3 requirements. Some lines have no special isolation provisions and are not ASME Section III or Seismic Category I if an analysis shows that exfiltration would not occur in the event of failure of that pipe (i.e., the –6.4 mm water gauge pressure differential would be maintained).

For architectural openings, the inleakage is based on –6.4 mm water gauge pressure differential. All doors have a vestibule with a second (outer) door. HVAC and electrical penetrations are designed to minimize leaks, and the HVAC System is designed and tested for isolation under accident conditions.

Table 6.2-9 provides a listing of secondary containment openings. All piping and cable tray penetrations will be sealed with a sealing compound for leakage and fire protection. All doors

are vestibule type with card reader access security systems that are monitored (Subsection 13.6.3.4). The HVAC penetrations are designed to close on a design basis accident (see Subsection 9.4.3 on R/B HVAC). The required testing procedure and frequency can be found in the plant Technical Specifications.

6.2.3.3 Design Evaluation

The design of the secondary containment boundaries is described in the preceding subsection. Evaluation of this design, such that all regulatory requirements are met, is given in the following subsections:

(1) 6.5.1 (Standby Gas Treatment System)

(2) 9.4.5 (Reactor Building HVAC System)

6.2.3.3.1 Compartment Pressurization

6.2.3.3.1.1 Design Bases

The design of secondary containment compartments with respect to pressurization due to a pipe rupture is based upon the worst-case DBA rupture of a high energy line postulated to occur in each compartment through which a high energy line passes (for details regarding the pipe rupture location and configuration, see Subsection 3.6.2). The pipe rupture producing the highest mass and energy release rate, in conjunction with a worst case single active component failure was chosen for the pressurization analysis of each component. For this analysis, a worst case single active component failure is defined as the failure to close an isolation valve which separates the reactor pressure vessel from the high energy pipe break in the secondary containment. The design pressure for the compartment structure design will include some margin over the calculated peak differential pressure. The design margin is intended to make allowance for changes (piping, equipment layout arrangement) in the as-built compartment design.

6.2.3.3.1.2 Design Features

The following paragraphs are brief descriptions of the compartments analyzed for pressurization. Figures 1.2-3 through 1.2-10 show compartment configurations, and component and equipment locations. The schematic layout of the compartments, with the interconnecting vent paths and blowout panels, which are modeled and analyzed for various line breaks are shown in Figures 6.2-37a through 6.2-37h.

6.2.3.3.1.2.1 Reactor Core Isolation Cooling (RCIC) Compartment

The RCIC compartment is located in the secondary containment at El –8200 mm, in the 0–90° quadrant of the R/B. The design basis break for the RCIC compartment is determined to be the single-ended break of the 150A steam supply line to the RCIC turbine. This line is a high-energy line out to the normally closed isolation valve inside the RCIC compartment. It supplies

high-energy steam to the RCIC turbine in the event of reactor vessel isolation. In the event of a postulated design basis high-energy line break (HELB), the steam/air mixture from that compartment is directed into adjoining compartments, corridors and stairways, and is eventually purged into the refueling floor connected to the outside atmosphere.

6.2.3.3.1.2.2 Reactor Water Cleanup (CUW) Equipment Rooms and Pipe Spaces

The CUW equipment (pump, heat exchanger, filter/demineralizer, valves) and pipe spaces are located in the 0 - 270 degree quadrant of the reactor building, with floor elevations ranging from elevation -8200 mm to elevation 12300 mm. The design basis pipe break for the CUW System compartment network is determined to be a 200 A double-ended break of the cleanup water suction line from the RPV. This high energy piping, which connects the CUW equipment, originates at the reactor pressure vessel. After being routed through the CUW System, this line is directed back to the RPV through special pipe spaces and the steam tunnel. In the event of a postulated design basis high energy line break in a compartment, the steam/air mixture from that compartment is directed into adjoining compartments, corridors and stairways, and eventually purged into the turbine building and refueling floor connected to the outside atmosphere.

6.2.3.3.1.2.3 Main Steam Tunnel

The Reactor Building main steam tunnel is located between the primary containment vessel and the Turbine Building at elevation 12300 mm and 0° azimuthal position. The DBA for the steam tunnel is the double-ended break of one of the 700A main steamlines. These lines originate at the RPV and are routed through the main steam tunnel to the Turbine Building. In the event of a postulated design basis HELB, the steam/air mixture from the main steam tunnel is purged into the Turbine Building.

6.2.3.3.1.3 Design Evaluation

The compartment response to the postulated high energy line break was calculated using the engineering computer program SCAM. A detailed discussion of methodology and assumptions used in this program can be found in Reference 6.2-4.

The initial conditions for the analysis include the assumption of 102% rated reactor power and the compartment pressures, temperatures and relative humidity as tabulated in Table 6.2-3. Blowout panels are used in place of open vent pathways when the environmental conditions of one compartment must be isolated from the environment in another compartment. The blowout panels are assumed to open fully against a differential pressure of 3.45 kPa, and are assumed to remain open.

For the postulated high energy line break, the blowdown mass and energy release rates from the break were determined using Moody's homogeneous equilibrium model for critical flow described in Reference 6.2-2. The blowdown mass and energy release rate for the postulated High Energy Line Break (HELB) in a given compartment compromised of initial inventory

depletion followed by steady critical flow from the ruptured pipe. After the inventory depletion period, break flow, limited by critical flow consideration, continues until the isolation valve is fully closed. Heat transfer between the flowing components and the subcompartment walls is neglected. This is a conservative assumption, since removal of heat would lower calculated pressures in the subcompartments.

The following paragraphs describe the key assumptions and calculation of mass and energy release rates for the postulated HELB in the RCIC, CUW and Main Steam Tunnel compartments.

6.2.3.3.1.3.1 RCIC Compartment

For RCIC a single-ended pipe break, as noted earlier, was postulated. The mass and energy blowdown release rate comprised only of flow from the RPV side. The flow from the other side of the break was assumed to be negligible. The blowdown flow comprised of initial inventory depletion followed by steady critical flow from the RPV. In computing the critical flow rate, flow loss factors between RPV and break location were ignored for conservatism. Tabulated values of mass and energy release rate for the postulated break is shown in Table 6.2-4b. The total blowdown duration of 41 seconds, as obvious from tabulated values, is based on assumption that the isolation valve starts closing at 11 seconds (1 second instrument response time and 10 seconds built in logic time delay) after the break and is fully closed in 30 seconds. Considering that the isolation valve is a gate valve, non-linear flow area changes with respect to time were used during the valve closure period.

Figure 6.2-37a shows the compartment nodalization scheme used for the pressurization analysis model for different break cases. Table 6.2-3 shows the free volume, initial environmental conditions and DBA characteristics for the compartments which were analyzed. Table 6.2-4 tabulates subcompartment vent path characteristics. The calculated peak pressure for the RCIC compartments are tabulated in Table 6.2-3. Graphs showing the compartment bounding pressure and temperature response as a function of time due to the postulated high energy line breaks are shown in Figures 6.2-37i and 6.2-37j, respectively. These results form basis for evaluating the effect of high energy line breaks on structures and safety related equipment.

6.2.3.3.1.3.2 CUW Compartment

For CUW a double-ended pipe break, as noted earlier, was postulated. The mass and energy blowdown release rate comprised of flow from both the RPV and BOP sides of the break location. The flow from the RPV side comprised of initial inventory depletion followed by steady critical flow. The flow from the BOP side is the depletion of inventory between the break location and the closest check valve. Flow loss factors due to pipe friction, and other mechanical devices such as valves, elbows, tees, etc. were accounted for only in determining steady critical flow rate, and not in determining the initial inventory depletion flow rate. Table 6.2-4a tabulates the flow loss factor considered for different postulated pipe break locations.

After the initial inventory depletion period, the steady RPV blowdown is choked at the venturi FE-001 (Figure 5.4-12, sheet 1 of 4) located upstream of the isolation valve, since the venturi flow area is smaller than the isolation valve flow area. After the isolation valve start closing, as soon as valve flow area becomes equal to the venturi flow area, flow will be choked at the isolation valve. The break flow stops when the isolation valve is fully closed.

Compartment pressurization analyses were done for postulated pipe breaks in different compartments. Tabulated mass and energy release rate for the postulated break cases are shown in Table 6.2-4b. The total blowdown duration of 76 seconds, as obvious from the tabulated values, is based on the assumption that the isolation valve starts to close at 46 seconds (1 second instrument response 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 30 seconds. Considering that the isolation valve is a gate valve, non-linear flow area changes with respect to time were used during the valve closure period.

Figure 6.2-37c shows compartment nodalization scheme for the pressurization analyses for different break cases. Table 6.2-3 shows the free volume, initial environmental conditions and DBA characteristics for the compartments which were analyzed. Table 6.2-4 tabulates subcompartment vent path characteristics. The calculated peak pressures for the CUW compartments are tabulated in Table 6.2-3. Graphs showing the compartment bounding pressure and temperature response as a function of time due to the postulated high energy line breaks are shown in Figures 6.2-37k and 6.2-37l, respectively. These results form basis for evaluating the effect of high energy line breaks on structures and safety related equipment.

6.2.3.3.1.3.3 Main Steam Tunnel

Double ended pipe breaks in main steamline (MSL) and feedwater line (FWL) were postulated. The mass and energy release rate comprised of flow from both the RPV and BOP sides. The blowdown flow comprised of initial inventory depletion followed by steady critical flow from the RPV and BOP sides. In calculating the critical flow rate flow loss factors were ignored for conservatism.

Tabulated values of mass and energy release rate for the postulated MSL and FWL breaks are shown in Table 6.2-4b. The total blowdown duration of 5.5 seconds for the MSL break, as obvious from the tabulated values, is based on the assumption that the main steam isolation valve (MSIV) starts closing at 0.5 seconds after the break and is fully closed in 5 seconds. The duration of 5.5 seconds is the longest closing time for the MSIVs. For the FWL break, the total blowdown duration of 120 seconds is determined by the feedwater flow from the BOP side, see Figure 6.2-3.

Figure 6.2-37b shows compartment nodalization scheme for the pressurization analyses for different postulated break cases. Table 6.2-3 shows the free volume, initial environmental conditions and DBA characteristics for the compartment analyzed. Table 6.2-4 tabulates subcompartment vent path characteristics. The calculated peak pressures for the main steam

tunnel compartments were limited by the MSL break and they are tabulated in Table 6.2-3. In comparison, the calculated peak differential pressure for the FWL break was found to be 26.5 kPaG. Graphs showing the compartment bounding pressure and temperature response as a function of time due to the postulated high energy line breaks are shown in Figures 6.2-37m and 6.2-37n, respectively. These results form basis for evaluating the effect of high energy line breaks on structures and safety related equipment.

As seen from the pressure transient results in Figure 6.2-37m, the peak pressure condition is reached very early in the transient, well before isolation valve starts closing. The compartment pressure drops as flow approaches to steady condition via interconnecting compartments. This suggests that the compartment peak pressure will not be influenced by valve closure characteristic, including an unlikely event in which valve failed to close automatically upon receipt of an isolation signal.

6.2.3.3.1.4 Equipment Qualification Temperature Values

The minimum set of thermal environmental conditions for safety-related systems and equipment are presented in Appendix 3I. These conditions are conservative envelope of the values due to high energy line breaks, with no credit taken for heat transfer into the compartment structural heat sinks. Heat transfer into structural heat sinks would result in lower calculated temperature values in the compartments.

Simplified conservative compartment transient cooling analyses were performed to determine adequacy of the equipment qualification (EQ) temperature values given in Appendix 3I. Credit for heat transfer into the structural heat sinks was taken in these analyses. Transient temperature response results from these analyses were compared with the EQ temperature values to confirm the adequacy of these values. These analyses are described in the following subsections.

6.2.3.3.1.4.1 Compartment Transient Cooling Analyses

The transient cooling analyses evaluated a double-ended guillotine break in CUW rooms (at EL -8200), and considered and analyzed two separate cases.

Case 1: It is assumed that isolation valve successfully close automatically upon receipt of an isolation signal. Closure of isolation valve terminates further depletion of vessel inventory into the subcompartment and, consequently, terminating further heating of the compartment environment. This case represents the design basis condition, and it is termed as an “Isolated Case”.

Case 2: It is postulated that isolation valve failed to close automatically upon receipt of an isolation signal. Isolation valve is closed through operator actions, which then terminates further depletion of vessel inventory into the subcompartment. This case represents beyond the design basis condition, and it is termed as an “Unisolated Case”. For sensitivity study purposes, two blowdown conditions were postulated and evaluated: 1) Operator actions close valve in 1/2

hour after pipe break accident; and 2) Operator actions close valve in 1 hour after pipe break accident.

6.2.3.3.1.4.1.1 Isolated Case: Design Basis Accident Condition

In this analysis it is assumed that isolation valve close automatically upon receipt of isolation signal, and mass and energy blowdown into the subcompartment terminates in 76 seconds after pipe break accident. This blowdown duration is comprised of sensor response time (1 s), time delay (45 s), and valve closing time (30 s). This is consistent with that used in subcompartment pressurization analyses. Mass and energy blowdown was confined to EL -8200 compartments only, taking no credit for flow communications with the higher level floors. All boundary walls and internal walls/floors were modeled as structural heat sinks. Natural convection plus radiation heat transfer mechanism between compartment environment and the structural heat sinks was assumed, for conservatism.

The calculated compartment transient temperature response is presented and compared with the EQ temperature values in Figure 6.2-37o. These results confirm that the EQ temperature values conservatively envelope the calculated transient temperature conditions.

6.2.3.3.1.4.1.2 Unisolated Case: Beyond the Design Basis Accident Condition

In these analyses, it is assumed that isolation valve failed to close automatically upon receipt of an isolation signal, but closed through operator actions. Break flow is confined to the entire secondary containment volume. All boundary walls and internal walls/floors were modeled as structural heat sinks. Natural convection plus radiation heat transfer mechanism between the compartment environment and the structural heat sinks was assumed, for conservatism.

The calculated transient temperature response for 1/2 hour and 1 hour operator actions time are presented and compared with the EQ temperature values in Figure 6.2-37p. These results show that the EQ temperature values (representative of the design basis condition) envelopes the transient temperature conditions for both 1/2 hour and 1 hour operator actions time.

6.2.3.4 Tests and Inspections

Testing and inspection of the integrity of secondary containment will be made as part of the testing of the SGTS (Subsection 6.5.1).

Status lights and alarms for door opening of secondary containment will be tested periodically by their operation, with observation of lights and alarms. Leakage testing and inspection of all other architectural openings will be made as they are utilized periodically.

6.2.3.5 Instrumentation Requirements

By their nature, electrical penetrations of secondary containment do not have any instrumentation requirements. Piping and HVAC penetrations instrumentation requirements are discussed as part of each system’s description in this SAR.

Certain doors are fitted with status indication lights.

6.2.4 Containment Isolation System

The primary objective of the Containment Isolation System (CIS) is to provide protection against releases of radioactive materials to the environment as a result of accidents occurring in the systems inside the containment. The objective is accomplished by isolation of lines or ducts that penetrate the containment vessel. Actuation of the CIS is automatically initiated at specific limits defined for reactor plant operation. After the isolation function is initiated, it goes through to completion.

6.2.4.1 Design Bases

6.2.4.1.1 Safety Design Bases

(1) 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 10CFR50 or 10CFR100 limits. Leaktightness of the valves shall be verified by Type C test.

(2) Capability for rapid closure or isolation of all pipes or ducts that penetrate the containment is provided by means that provide a containment barrier in such pipes or ducts sufficient to maintain leakage within permissible limits.

(3) The design of isolation valving for lines penetrating the containment follows the requirements of General Design Criteria 54 through 57 to the greatest extent practicable consistent with safety and reliability.

(4) Isolation valves for instrument lines that penetrate the drywell/containment conform to the requirements of Regulatory Guide 1.11.

(5) Isolation valves, actuators and controls are protected against loss of their safety function from missiles and postulated effects of high- and moderate-energy line ruptures.

(6) Design of the containment isolation valves and associated piping and penetrations meets the requirements for Seismic Category I components.

(7) Containment isolation valves and associated piping and penetration meet the requirements of the ASME Boiler and Pressure Vessel Code, Section III, Class 1, 2, or MC, in accordance with their quality group classification.

(8) The design of the Control Systems for automatic containment isolation valves ensures that resetting the isolation signal shall not result in the automatic reopening of containment isolation valves.

6.2.4.1.2 Design Requirements

The Containment Isolation System, in general, closes fluid penetrations that support systems not required for emergency operation. Fluid penetrations supporting ESF systems have remote manual isolation valves which can be closed from the control room, if required.

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

Protection of CIS 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. Air-operated containment isolation valves are designed to fail to the required position for containment isolation upon loss of the instrument air supply or electrical power.

The CIS is designed to Seismic Category I requirements. Classification of equipment and systems is found in Table 3.2-1. Figure 6.2-38 identifies the quality group classifications and containment isolation provisions.

The criteria for the design of the containment and reactor vessel isolation control system are listed in Subsection 7.1.2. The bases for assigning certain signals for containment isolation are listed and explained in Subsections 7.3.1 and 7.6.1.

On signals of high drywell pressure or low water level in the reactor vessel, all isolation valves that are part of systems not required for emergency shutdown of the plant are closed. The same signals initiate the operation of systems associated with the ECCS. The isolation valves that are part of the ECCS can be closed remote-manually from the control room or closed automatically, as appropriate.

6.2.4.2 System Design

The Containment Isolation System consists of the valves and controls required for the isolation of lines penetrating the containment. Figure 6.2-38 identifies the containment isolation provisions. Table 6.2-7 shows the pertinent data for the containment isolation valves. A detailed discussion of the controls associated with the CIS is included in Subsections 7.3.1.1.2 and 7.3.1.1.11.

Power-operated containment isolation valves have indicating switches in the control room to show whether the valve is open or closed. Loss of power to each motor-operated valve (MOV) is detected and annunciated. Air-operated containment isolation valves are designed to fail in a safe position upon loss of air or power to the solenoid pilot valve. 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.

Two main steam isolation valves (MSIVs) in series are used on each of the four steamlines to assure containment isolation when needed. One valve is as close as possible to the inside of the drywell, and the other is just outside the containment. Each MSIV is spring loaded and operated by pneumatic pressure that opens and closes the valve. Operating gas is supplied to the valves from the plant nitrogen or instrument air system. A pneumatic accumulator provides backup operating gas. Spring force closes the valve if gas pressure is not available. A detailed description of valve design is contained in Subsection 5.4.5.

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

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

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

Containment isolation valves are either automatically actuated by the containment isolation signals or are remote-manually operated, as appropriate. Primary and secondary modes of operation are assigned to these actuations, respectively.

Isolation valve closure will be assured by using the latest state of the art technology in valve design. Valve actuators will be sized based on demonstrated valve design and established disk friction factors. Adequate thrust capability will be developed with sufficient margin in the actuator and the valve as appropriate to demonstrate acceptability of the valve design for its application.

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 10CFR100. For system lines which can provide an open path from the containment to the environment, a discussion of valve closure time bases is provided in Chapter 15.

6.2.4.2.2 Instrument Lines Penetrating Containment

Sensing instrument lines penetrating the containment follow all the recommendations of Regulatory Guide 1.11. Each line has a 6.35 mm orifice inside the drywell, as close to the beginning of the instrument line as possible, and a manually-operated isolation valve just outside the containment.

6.2.4.2.3 Compliance with General Design Criteria and Regulatory Guides

In general, all requirements of General Design Criteria 54, 55, 56, and 57, and Regulatory Guide 1.11 are met in the design of the Containment Isolation System. A case-by-case analysis of all such penetrations is given in Subsection 6.2.4.3.2.

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

Protection is provided for isolation valves, actuators and controls against damage from missiles. All potential sources of missiles are evaluated. Where possible hazards exist, protection is afforded by separation, missile shields or by location outside the containment. Tornado missile protection is afforded by the fact that all containment isolation valves are inside the missile-proof Reactor Building. Internally-generated missiles are discussed in Subsection 3.5.1, and the conclusion is reached that there are no potentially damaging missiles generated. Dynamic effects from pipe break (jet impingement and pipe whip) are discussed in Section 3.6. The arrangement of containment isolation valves inside and outside the containment affords sufficient physical separation such that a high-energy pipe break will not preclude containment isolation. The CIS piping and valves are designed in accordance with Seismic Category I requirements as defined in Section 3.7 using the techniques of Subsection 3.9.2.

Section 3.11 presents a discussion of the environmental conditions, both normal and accidental, for which the Containment Isolation System is designed. The section discusses the qualification tests required to assure the performance of the isolation valves under particular environmental conditions.

Containment isolation valves are designed in accordance with the requirements of ASME Code Section III. Where necessary, a dynamic system analysis which covers the impact effect of rapid valve closures under operating conditions is included in the design specifications of piping systems involving containment isolation valves. Valve operability assurance testing is discussed in Subsection 3.9.3.

6.2.4.2.5 Valve Operability and Leakage Control

Provisions for demonstrating the operability of isolation valves are discussed in Subsection 3.9.3. Subsection 6.2.6 describes leakage rate testing of containment isolation barriers. The power-operated and automatic isolation valves will be cycled during normal operation to assure their operability.

6.2.4.2.6 Redundancy and Modes of Valve Actuations

The main objective of the Containment Isolation System is to provide environmental protection by preventing releases of radioactive materials. This is accomplished by complete isolation of system lines penetrating the primary containment. Redundancy is provided in all design aspects to satisfy the requirement of GDC 54 that no active failure of a single valve or component prevents containment isolation.

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

Isolation valve arrangements with appropriate instrumentation are shown in the P&IDs. The isolation valves 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 leaktightness. The design specifications require each isolation valve to be operable under the most severe operating conditions that it may experience. Each isolation valve is afforded protection by separation and/or adequate barriers from the consequences of potential missiles.

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

Provisions for administrative controls and/or locks ensure that the position of all manual isolation valves is maintained and known. The position of all power-operated control valves is indicated in the control room. Discussion of instrumentation and controls for the isolation valves is included in Chapter 7.

6.2.4.3 Design Evaluation

6.2.4.3.1 Introduction

The main objective of the Containment Isolation System is to provide protection by preventing releases to the environment of radioactive materials. This is accomplished by complete isolation of system lines penetrating the primary containment. Redundancy is provided in all design aspects to satisfy the requirement that any active failure of a single valve or component does not prevent containment isolation.

Mechanical components are redundant, such as isolation valve arrangements to provide backup in the event of accident conditions. Isolation valve arrangements satisfy requirements specified in General Design Criteria 54, 55, 56, and 57, and Regulatory Guide 1.11, as shown on Figure 6.2-38.

The isolation valves 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 leaktightness.

Each isolation valve is qualified operable under the most severe operating conditions that it might experience. Each isolation valve is afforded protection by separation and/or adequate barriers from the consequences of potential missiles.

Electrical redundancy is provided in isolation valve arrangements, which eliminates dependency on one power source to attain isolation. Electrical cables for isolation valves in the same line have been routed separately. Cables have been selected and based on the specific environment to which they may be subjected, such as magnetic fields, high radiation temperature, and high humidity.

Provisions for administrative control and/or locks ensure that the position of all nonpowered isolation valves is maintained and known. For all power-operated valves the position is indicated in the main control room. Discussion of instrumentation and controls for the isolation valves is included in Chapter 7.

6.2.4.3.2 Evaluation Against General Design Criteria

6.2.4.3.2.1 Evaluation Against Criterion 55

The reactor coolant pressure boundary (RCPB), as defined in 10CFR50, Section 50.2, consists of the reactor pressure vessel (RPV), pressure-retaining appurtenances attached to the vessel, valves and pipes which extend from the RPV up to and including the outermost isolation valve. The lines of the RCPB that penetrate the containment include provisions for isolation of the containment, thereby precluding any significant release of radioactivity. Similarly, for lines that do not penetrate the containment, but which from a portion of the RCPB, the design ensures that isolation of the RCPB can be achieved.

6.2.4.3.2.1.1 Influent Lines

Influent lines, which penetrate the containment directly to the RCPB, are 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 6.2-5 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.

6.2.4.3.2.1.1.1 Feedwater Line

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

the check valves prevent significant loss of reactor coolant inventory and offer immediate isolation. The use of check valves as feedwater isolation valve allows reactor makeup from the Feedwater System, the RCIC System, and the RHR System operating in the LPFL mode following a postulated LOCA inside the reactor containment. A motor-operated gate valve is provided upstream of the outboard check valve for long-term leakage control.

See Subsection 6.2.4.3.2.1.1.5 for isolation of CUW return line connecting the feedwater line outside containment.

6.2.4.3.2.1.1.2 RHR Injection Line

Satisfaction of isolation criteria for the RHR injection line is accomplished by use of a remote-manually operated gate valve and check valve (Table 6.2-5). Both types of valves are normally closed, with the gate valve receiving an automatic signal to open at the appropriate time to assure that fuel temperature design limits are not exceeded in the event of a LOCA. The normally closed check valve protects against containment overpressurization in the event of a break in the line between the check valve and containment wall by preventing high-energy reactor water from entering the primary containment.

6.2.4.3.2.1.1.3 HPCF Line

The HPCF line penetrates the drywell to inject directly into the reactor pressure vessel. Isolation is provided by an air testable check valve, located inside the containment with position indicated in the main control room, and remote-manually actuated gate valve located as close as practicable to the exterior wall of the containment. Long-term leakage control is maintained by this gate valve. If a LOCA occurs, this gate valve will receive an automatic signal to open.

6.2.4.3.2.1.1.4 Standby Liquid Control System Line

The Standby Liquid Control System (SLCS) line penetrates the containment and connects to the HPCF line inside the upper drywell to form a common line which discharges directly into the RPV. In addition to a simple check valve inside the containment, a check valve, together with a MOV, are located outside the drywell. Since the SLCS line is a normally closed, nonflowing line, rupture of this line is extremely improbable. However, should a break occur subsequent to the opening of the MOV, the check valves insure isolation.

6.2.4.3.2.1.1.5 Reactor Water Cleanup System Line (Reactor Vessel Head Spray)

The Reactor Water Cleanup (CUW) System returns water to the RPV through two paths. The normal path during plant operation returns water from the filter/demineralizers to the feedwater lines outside the containment. During the postulated LOCA, it is desirable to terminate any CUW leakage. Isolation of the return line is provided by the Feedwater System check valve and CUW check valve and motor-operated valve. The motor- operated valve provides long-term leakage control.

The CUW System return path during initial shutdown cooling operation is through the head spray nozzle on the top of the RPV. The CUW System head spray line enters the containment in the upper drywell and has a MOV outside and a check valve inside the containment. The motorized valve is normally closed during plant operation but is given a close signal by the Leak Detection System when a containment isolation signal is given.

6.2.4.3.2.1.1.6 Recirculation Pump Seal Purge Water Supply Line

The recirculation pump seal water line extends from the recirculation pump motor housing through the containment and connect to the CRD supply just outside containment (Figure 5.4-4). Since the seal purge water line forms a part of the RCPB, the consequences of its failure have been evaluated.

The evaluations for previous similar designs show that the consequences of breaking the line are less severe than those of failing an instrument line. The recirculation pump seal water line is 20A Quality Group B from the manual shutoff valve located close to the recirculation pump motor housing through the excess flow check valve (located outside the containment). From the excess flow check valve to the CRD connection, the line is Quality Group D. An orifice is located inside the containment and if the line is postulated to fail and the excess flow check valve is assumed not to close (single active failure), the flow rate through the broken line is calculated to be substantially less than permitted for a broken instrument line. Therefore, this configuration provides sufficient isolation capability for postulated failure of the line.

6.2.4.3.2.1.2 Effluent Lines

Effluent lines which form part of the RCPB and penetrate containment are equipped with at least two isolation valves; one inside the containment and one outside, located as close to the containment wall as practicable.

Table 6.2-6 contains those effluent lines that comprise the reactor coolant pressure boundary and which penetrate the containment.

6.2.4.3.2.1.2.1 Main Steam and Drain Lines and RCIC Steamline

The main steamlines which extend from the reactor pressure vessel to the main turbine and condenser system, penetrate the primary containment. The main steam drain lines connect the low points of the steamlines, penetrate the primary containment and are routed to the condenser hotwell. The RCIC turbine steamline connects to the main steamline in the upper drywell and penetrates the primary containment. For these lines, isolation is provided by automatically actuated block valves, one inside and one just outside the containment.

6.2.4.3.2.1.2.2 RHR Shutdown Cooling Line

Three RHR shutdown cooling lines connect to the reactor vessel and penetrate the primary containment. Isolation is provided by two automatically actuated block valves, one inside and the other outside the containment.

6.2.4.3.2.1.2.3 Reactor Water Cleanup System Suction Line

The CUW takes its suction from the bottom head of the RPV and from the RHR "B" shutdown cooling suction line. The CUW suction line is isolated by two automatic motor-operated gate valves on the inside and outside of the containment. Should a break occur in the CUW System, the check valves would prevent backflow from the RPV and the isolation valves would prevent forward flow from the RPV.

CUW pumps, heat exchangers and filter/demineralizers are located outside the drywell.

6.2.4.3.2.1.3 Conclusion on Criterion 55

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

In addition to meeting the isolation requirements stated in Criterion 55, the pressure-retaining components which comprise the RCPB 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 RCPB are designed in accordance with ASME Boiler and Pressure Vessel Code Section III, Class 1.

It is therefore concluded that the design of piping system which comprises the reactor coolant pressure boundary and penetrates the containment satisfies Criterion 55.

6.2.4.3.2.2 Evaluation Against Criterion 56

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

Although a word-for-word comparison with Criterion 56 in some cases is not practical, it is possible to demonstrate adequate isolation provisions on some other defined basis.

6.2.4.3.2.2.1 Influent Lines to Suppression Pool

Figure 6.2-38 identifies the isolation provisions in the influent lines to the suppression pool.

6.2.4.3.2.2.1.1 HPCF and RHR Test and Pump Minimum Flow Bypass Lines

The HPCF and RHR test and pump minimum flow bypass lines have isolation capabilities commensurate with the importance to safety of isolating these lines. Each line has a motor-operated valve located outside the containment. Containment isolation requirements are met on

the basis that the lines are low-pressure lines constructed to the same quality standards commensurate with their importance to safety. Furthermore, the consequences of a break in these lines result in no significant safety consideration. All of the lines terminate below the minimum drawdown level in the suppression pool.

The test return lines are also used for suppression pool return flow during other modes of operation. In this manner, the number of penetrations is reduced, thus minimizing the potential pathways for radioactive material release. Typically, pump minimum flow bypass lines join the respective test return lines downstream of the test return isolation valve. The bypass lines are isolated by MOVs in series with a restricting orifice.

6.2.4.3.2.2.1.2 RCIC Turbine Exhaust and Pump Minimum Flow Bypass Lines

The RCIC turbine exhaust line, which penetrates the containment and discharges to the suppression pool, is equipped with a normally open, motor-operated, remote-manually actuated gate valve located as close to the containment as possible. In addition, there is a simple check valve upstream of the gate valve, which provides positive actuation for immediate isolation in the event of a break upstream of this valve. The gate valve in the RCIC turbine exhaust is designed to be locked open in the control room and is interlocked to preclude opening of the inlet steam valve to the turbine until the turbine exhaust valve is in its full-open position. The RCIC pump minimum flow bypass line is isolated by a normally closed, remote manually actuated valve outside containment.

6.2.4.3.2.2.1.3 SPCU Discharge Line

The Suppression Pool Cleanup (SPCU) System discharge line to the suppression pool (i.e., containment penetration, piping and isolation valves) is designed to Seismic Category I, ASME Section III, Class 2 requirements.

6.2.4.3.2.2.2 Effluent Lines from Suppression Pool

Figure 6.2-38 identifies the isolation provisions in the effluent lines from the suppression pool.

6.2.4.3.2.2.2.1 RHR, RCIC and HPCF Lines

The RHR, RCIC, and HPCF suction lines contain motor-operated, remote-manually actuated gate valves which provide assurance of isolating these lines in the event of a break. These valves also provide long-term leakage control. In addition, the suction piping from the suppression pool must be available for long-term usage following a design basis LOCA, and, as such, is designed to the quality standards commensurate with its importance to safety. The RHR discharge line fill system suction lines have manual valves for operational purposes. These systems are isolated from the containment by the respective RHR pump suction valves from the suppression pool.

6.2.4.3.2.2.2.2 SPCU Suction Line

The SPCU System suction line has two isolation valves. However, because the penetration is under water, both isolation valves are located outside the containment. The first valve is located as close as possible to the containment, and the second is located to provide adequate separation from the first.

6.2.4.3.2.2.2.3 ACS Lines to Containment

The Atmospheric Control System (ACS) has both influent and effluent 550A lines which penetrate the containment. Both isolation valves on these lines are outside of the containment vessel to provide accessibility to the valves. The valves are located as close as practical to the containment vessel. The piping from the containment to and including both valves is an extension of the primary containment boundary and is designed in accordance with the ASME Boiler and Pressure Vessel Code, Section III, Class 2 requirements. The arrangement of the isolation valves and connecting piping is such that a single active failure of an inboard valve, or a single active or passive failure in the connecting piping or an outboard valve, cannot prevent isolation of the ACS containment penetrations. The ACS containment isolation valve closure time is ≤ 20 seconds. These valves close on the following signals: high drywell pressure, RPV low water level 3, and high radioactivity in the purge and vent exhaust line. The SRP 6.2.4 states that the 5-second closure speed is necessary to assure that the purge and vent valves would have closed before the onset of fuel failures following a LOCA. The ACS purge and vent valves are normally closed during plant operation and are allowed to open only during the inerting (startup) and de-inerting (shutdown) process where the reactor is at greater than 15% power. The likelihood of LOCA during inerting/de-inerting is very low. If a LOCA does occur, these valves will have closed before the onset of fuel failure. Note that the onset of fuel failure is when the core is uncovered and that reactor water level 3 (when ACS valves isolate) is 3.8m above the core. In the event of a radioactivity leak during inerting/ de-inerting, the radiation detectors at the purge and vent exhaust line will detect the condition and isolate the ACS containment isolation valves. Note that the exhaust radiation detectors are very sensitive and are set at a lower setpoint compared to the ones inside containment to have an effective early detection. For the ACS, a more reliable isolation valve is necessary to ensure containment integrity. A fast closing valve is less reliable than valves with moderate speed. The difference between 5 and 20 seconds is considered to be insignificant. Thus, the risk is judged to be sufficiently small and that the 20-second closure time, is deemed sufficient and reliable.

The ACS also has two 50A makeup line isolation valves which are normally open during normal reactor operation to provide nitrogen makeup into the containment. If these isolation valves are placed in the normally closed position, nitrogen makeup will not be possible without opening. In either position, these valves need to open to provide nitrogen makeup. The normally open position provides automatic nitrogen makeup without frequent cycling that could cause damage to valves. In the event of a LOCA or an event requiring primary containment isolation, these valves automatically close upon receipt of the following signals: high drywell pressure, low water level, high radioactivity in the purge and vent exhaust line. These valves are

redundant and meet ESF requirements as described above for the 550A influent and effluent lines.

6.2.4.3.2.2.2.4 Conclusion on Criterion 56

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

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

6.2.4.3.2.3 Evaluation Against Criterion 57

Lines penetrating the primary containment, which are governed by neither Criterion 55 nor Criterion 56, comprise the closed system isolation valve group.

Influent and effluent lines of this group are isolated by automatic or remote-manual isolation valves located as close as possible to the containment boundary.

6.2.4.3.2.4 Evaluation Against Regulatory Guide 1.11

Instrument lines that connect to the RCPB and penetrate the containment have 6.35 mm orifices and manual isolation valves, in compliance with Regulatory Guide 1.11 requirements.

6.2.4.3.3 Evaluation of Single Failure

A single failure can be defined as a failure of a component (e.g., a pump, valve, or a utility such as offsite power) to perform its intended safety functions as a part of a safety system. The purpose of the evaluation is to demonstrate that the safety function of the system will be completed even with that single failure. Appendix A to 10CFR50 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 as well as mechanical systems are designed to meet the single-failure criterion, regardless of whether the component is required to perform a safety action. Even though a component, such as an electrically-operated valve, is not designed to receive a signal to change state (open or closed) in a safety scheme, it is assumed as a single failure if the system component changes state or fails. Electrically-operated valves include valves that are electrically piloted but air operated, as well as valves that are directly operated by an electrical device. In addition, all electrically-operated valves that are automatically actuated can also be manually actuated from the main control room. Therefore, a single failure in any electrical system is analyzed, regardless of whether the loss of a safety function is caused by a component

failing to perform a requisite mechanical motion or a component performing an unnecessary mechanical motion.

6.2.4.3.4 Evaluation of Containment Purge and Vent Valves Isolation Barrier Design

Protection of the containment purge system CIVs from the effects of flood and dynamic effects of pipe breaks will be provided in accordance with Sections 3.4 and 3.6. The CIVs are air-operated with pilot AC solenoid valves. The power to the AC solenoid valve is supplied from the Vital AC distribution system to the Input/Output (I/O) device from the valve. Both the supply and return lines for the AC are fused at the I/O device so that faults are isolated and do not propagate back up into the portions of the Vital AC system common with other systems. This is also discussed in the Fire Hazard Analysis in Section 9A.5.

6.2.4.3.5 Evaluation of Simultaneous Venting of Drywell and Wetwell

The large (550A) purge and vent lines for the ACS, shown in Figure 6.2-39 are not used for purge or venting during normal reactor operation. The isolation valves in these lines are normally closed, they fail in the closed position, they receive an automatic closure signal in the event of a LOCA and they are not needed for pressure control of the containment during normal operation. Administrative controls are used to prevent opening of these valves except at the beginning and end of an operating cycle.

Pressure control of the containment during operation is maintained by a single, small (50A) nitrogen supply line, and a single, small (50A) vent line. The supply line is divided and provides makeup nitrogen to both drywell and wetwell. The small vent line is attached to the 550A drywell purge exhaust line and bypasses the closed 550A valve (F004). There is no equivalent vent line from the wetwell. Therefore, the drywell and wetwell are not vented simultaneously during operation and the system has only one supply and one exhaust line as required by BTP CSB 6-4.

6.2.4.3.6 Evaluation of Containment Purge System Against Criterion 54

The containment purge system has redundant CIVs each powered from independent electrical division. The CIVs are arranged such that any single failure will not compromise the integrity of the containment. These valves are designed to fail in closed position upon loss of air or loss of electric power to the pilot solenoid valve. With the exception of the makeup valves (50A), all containment purge system CIVs are in closed position during normal reactor operation. The purge and vent valves are open only during the inerting and de-inerting modes. All containment purge system CIVs automatically close upon receipt of containment isolation signal. Also, these valves are outside containment and accessible should manual actuation be required. Since this arrangement has adequate redundancy, and independence and is not unduly vulnerable to common mode failures, it is not necessary to have redundant and independent CIVs as would be required by Criterion 54.

6.2.4.4 Test and Inspections

The Containment Isolation System, except ECCS injection valves that interface with the reactor coolant system, is scheduled to undergo periodic testing during reactor operation. The ECCS injection valves can only be tested during reactor shutdown, due to inter-system LOCA considerations. 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.

Air-testable check valves are provided on influent emergency core cooling lines of the HPCF and RHR Systems whose operability is relied upon to perform a safety function.

A discussion of testing and inspection of isolation valves is provided in Subsection 6.2.1.6. Instruments are periodically tested and inspected. Test and/or calibration points are supplied with each instrument. Leakage integrity tests shall be performed on the containment isolation valves with resilient material seals at least once every three months.

6.2.5 Combustible Gas Control in Containment

The Atmospheric Control System (ACS) is provided to establish and maintain an inert atmosphere within the primary containment during all plant operating modes except during shutdown for refueling or equipment maintenance and during limited periods of time to permit access for inspection at low reactor power.

6.2.5.1 Design Bases

Since there is no design requirement for the ACS in the absence of a LOCA and since there is no design basis accident in the ABWR that results in core uncovery or fuel failures, the following requirements mechanistically assume that a LOCA producing the design basis quantities of hydrogen and oxygen has occurred. Following are criteria that serve as the bases for design:

(1) The hydrogen generation from metal-water reaction is defined in Regulatory Guide 1.7.

(2) The hydrogen and oxygen generation from radiolysis is defined in Regulatory Guide 1.7.

(3) The ACS establishes an inert atmosphere throughout the primary containment following an outage or other occasions when the containment has been purged with air to an oxygen concentration greater than 3.5%.

(4) The ACS maintains the primary containment oxygen concentration below the maximum permissible limit per Regulatory Guide 1.7 during normal, abnormal, and accident conditions in order to assure an inert atmosphere.

(5) The ACS also maintains a slightly positive inert gas pressure in the primary containment during normal, abnormal and accident conditions to prevent air (oxygen) leakage into the inerted volumes from the secondary containment, and provides non-safety-related monitoring of the oxygen concentration in the primary containment to assure a breathable mixture for safe personnel access. Essential safety-related monitoring is provided by the Containment Atmospheric Monitoring System (CAMS), as described in Chapter 7.

(6) The drywell and the suppression chamber will be mixed uniformly after the design basis LOCA due to natural convection and molecular diffusion. Mixing will be further promoted by operation of the containment sprays.

(7) Not Used.

(8) The ACS is designed to maintain an inert primary containment after the design-bases LOCA, assuming a single-active failure. The backup purge function need not meet this criterion.

(9) Components of the ACS inside the Reactor Building are protected from postulated missiles and pipe whip, as required to assure proper action.

(10) The ACS has the capability to withstand the dynamic effects associated with a safe shutdown earthquake without loss of isolation function.

(11) The system is designed so that all components exposed to the primary containment atmosphere (i.e., inboard isolation valves) are capable of withstanding the temperature, humidity, pressure, and radiation transients resulting from a LOCA.

(12) The ACS is non-safety class except as necessary to assure primary containment integrity (penetrations, isolation valves). The ACS is designed and built to the requirements specified in Section 3.2.

(13) The ACS includes a liquid nitrogen storage tank, vaporizer and heater along with the valves and piping carrying nitrogen to the containment, valves and piping from the containment to the SGTS and HVAC exhaust line, dedicated containment overpressure relief line with attached valves and rupture disk, non-safety oxygen monitoring, and all related instruments and controls. The ACS does not include any structures or housing supporting the aforementioned equipment or any ducting in the primary containment. Figure 6.2-39 shows the system P&ID.

The nitrogen supplied from the ACS shall be oil-free with a moisture content of less than 2.5 ppm. Filters are provided to remove particulates larger than 5 micrometers.

(14) The system is designed to facilitate periodic inspections and tests. The ACS can be inspected or tested during normal plant conditions.

(15) The primary containment purge system will aid in the long-term post-accident cleanup operation. The primary containment atmosphere will be purged through the SGTS to the outside environment. Nitrogen makeup will be available during the purging operation.

(16) The ACS is also designed to release containment pressure before uncontrolled containment failure could occur.

6.2.5.2 System Design

6.2.5.2.1 General

The ACS system is designed to control the environment within the primary containment. Any oxygen evolution from radiolysis is very slow such that natural convection and molecular diffusion is sufficient to provide mixing. Spray operation will provide further assurance that the drywell or wetwell is uniformly mixed.

The ACS provides and maintains an inert atmosphere in the primary containment during plant operation. The system is not designed as a continuous containment purging system. The ACS exhaust line isolation valves are closed when an inert condition in the primary containment has been established. The nitrogen supply makeup lines, compensating for leakage, provide a makeup flow of nitrogen to the containment. If a LOCA signal is received, the ACS valves close. Nitrogen purge from the containment occurs during shutdown for personnel access. Purging is accomplished with the containment inlet and exhaust isolation valves opened to the selected exhaust path and the nitrogen supply valves closed. Nitrogen is replaced by air in the containment (see Item (3) Shutdown-Deinerting below this subsection). The system has the following features:

(1) Atmospheric mixing is achieved by natural processes. Mixing will be enhanced by operation of the containment sprays, which are used to control pressure in the primary containment.

(2) The ACS primary containment nitrogen makeup maintains an oxygen-deficient atmosphere ($\leq$3.5% by volume) in the primary containment during normal operation.

(3) The redundant oxygen analyzer system (CAMS) measures oxygen in the drywell and suppression chamber. Oxygen concentrations are displayed in the main control room.

In addition, the ACS provides overpressure protection to relieve containment pressure, as required, through a pathway from the wetwell airspace to the stack. The pathway is isolated during normal operation by a rupture disk.

The following modes of ACS operation are provided:

(1) **Startup**—Inerting: Liquid nitrogen is vaporized with steam or electric heaters to a temperature greater than -7°C and is injected into the wetwell and the drywell. The nitrogen will be mixed with the primary containment atmosphere by the drywell coolers in the drywell and, if necessary, by the sprays in the wetwell.

(2) **Normal**—Maintenance of Inert Condition: A nitrogen makeup system automatically supplies nitrogen to the wetwell and upper drywell to maintain a slightly positive pressure in the drywell and wetwell to preclude air leakage from the secondary to the primary containment. An increase in containment pressure is controlled by venting through the drywell bleed line.

(3) **Shutdown**—Deinerting: Air is provided to the drywell and wetwell by the Reactor Building HVAC purge supply fan. Exhaust is through the drywell and wetwell exhaust lines to the plant vent, through the HVAC or SGTS, as required. During shutdown, purge air provides containment access ventilation.

(4) Overpressure Protection: If the wetwell pressure increases to about 617.8 kPaG (Subsection 19E.2.8.1), the rupture disk will open. The overall containment pressure decreases as venting continues. Closing the two 250A air-operated butterfly valves re-establishes containment isolation as required.

(5) ACS, except COPS, primary containment isolation valves, if open, (they are normally closed) are automatically closed if the drywell high pressure, or reactor low water level 3 setpoint is reached or if high radiation is detected in the exhaust flow. (See Table 5.2-6)

The following interfaces with other systems are provided:

(1) Residual Heat Removal System (RHR): The RHR System provides post-accident suppression pool cooling, as necessary, following heat dumps to the pool, including the exothermic heat of reaction released by the design basis metal-water reaction. This heat of reaction is very small and has no real effect on pool temperature or RHR heat exchanger sizing. The wetwell spray portion of the RHR may be activated during a LOCA help mixing by reducing pocketing. Wetwell spray would also serve to accelerate deaeration of the suppression pool water, though the impact of the dissolved oxygen on wetwell airspace oxygen concentration is very small.

(2) Drywell Cooling System: Provides circulation to all portions of the upper and lower drywell, the drywell head area, and the vessel support skirt area to accomplish the mixing necessary for completion of either the inerting or deinerting process and provides representative oxygen samples to the CAMS oxygen sensors. Should the arrangement of the RPV insulation leave a significant gap between itself and the

RPV, forced circulation will be provided to that area. Portions of the drywell will be inerted to sufficiently below 3.5% such that the bulk average oxygen concentration does not exceed 3.5% oxygen

(3) HVAC System: (1) supplies the drywell and wetwell exhaust flow during inerting, deinerting, and shutdown venting (2) accommodates drywell bleedoff flows during startup, (3) provides sufficient air flow to limit the concentration of any nitrogen leaking from the primary containment into the secondary containment, and (4) supplies air for purging the primary containment during deinerting and shutdown venting. Nitrogen leaking from the primary containment is insignificant and does not impact HVAC design.

The two outdoor air intakes of the Control Room habitability HVAC System are located far apart to protect personnel in the control room in the event of a nitrogen pipe or storage tank rupture. Similarly, intakes for all HVAC systems are located to minimize the introduction of nitrogen from such ruptures into occupied areas of the plant.

(4) High Pressure Nitrogen Gas Supply System: Serves all pneumatically-operated components in the primary containment because the containment is inerted. The pneumatic devices in the primary containment or those which could leak into the primary containment are supplied with nitrogen for the purpose of preventing oxygen addition to the inerted volumes. The High Pressure Nitrogen Gas Supply System is supplied from the ACS nitrogen storage tank and a bank of nitrogen storage cylinders.

(5) Standby Gas Treatment System: Processes any drywell bleedoff, inerting, and deinerting exhaust flows, as required by offsite release constraints.

(6) Containment Atmospheric Monitoring System: Monitors oxygen levels in the wetwell and drywell to confirm the primary containment oxygen level is kept within limits.

Radiation monitoring in the plant vent, part of Process Radiation Monitoring, detects high radiation during deinerting.

There are no potential sources of oxygen in the containment other than that resulting from radiolysis of the reactor coolant. Consideration of potential sources of leakage of oxygen into the containment included not only normal plant conditions but also postulated LOCA conditions. Potential sources of leakage are instrument air systems, service air lines, and inflatable door seals. Nitrogen is substituted for service and instrument air whenever leakage into the inerted containment could be postulated.

6.2.5.2.2 Inerting Equipment

The inerting subsystem is capable of reducing the wetwell and drywell oxygen concentrations from atmospheric conditions to less than 3.5% in less than four hours. The inerting vaporizers are sized to provide at least 2.5 times the containment (wetwell and drywell) free volume of nitrogen within the allotted four hours. The specified oxygen limit of 3.5% by volume must be adjusted for initial containment conditions, instrumentation errors, operator and equipment response time, and equipment performance to ensure that the actual oxygen concentration does not exceed 3.5% by volume during normal operation. The actual oxygen concentration shall not exceed 5% by volume during an accident when the hydrogen concentration is greater than 4%. The inert containment can be deinerted to allow safe personnel access without breathing apparatus in less than four hours.

Each penetration and pipe carrying nitrogen is sloped as necessary to prevent condensation collection and line blockage and shall be protected against entry of debris.

All pipe volumes where liquid or very cold nitrogen could be trapped between closed valves have relief valves. All relief valves exhaust outside the Reactor Building. Means are provided to spray nitrogen to the nitrogen storage tank vapor space (to decrease tank pressure) and the liquid volume (to increase tank pressure). Tank level and pressure indication are provided at the tank. Means for startup full-scale testing of the inerting and makeup portions of the system without nitrogen injection to the containment are provided. During startup, the test discharges shall be temporarily piped away from the control panel and storage and vaporization equipment to avoid excessive noise from the open discharge. Strainers are provided in the liquid portion of the makeup and inerting lines. Means are provided to feed the makeup circuit from either the liquid or vapor portion of the nitrogen storage. Pressure is automatically maintained in the nitrogen storage tank during nitrogen discharge by a circuit with another ambient heat exchanger fed by a pressure control valve. The inerting and makeup portions of the system do not rely on pumps to perform their function. Means are provided to manually vent the tank vapor space to control pressure. Means are provided to drain the storage tank. The vessel bottom is sloped or dished to facilitate this draining.

Pressure relief for the nitrogen storage tank is provided at 10% above the upper limit of the normal range of operating pressures. Rupture disks, set 20% above the upper limit but not higher than the design pressure of the vessel, are provided. Redundant pressure relief valves are provided so that protection is immediately available should a disk rupture and then be isolated. Penetrations through the nitrogen storage tank insulation are minimized to reduce heat gain. The length of piping through the insulation is maximized to the extent practicable to reduce heat gain.

During plant startup, drywell and wetwell atmospheric oxygen concentration will be less than 3.5% by volume within 24 hours after thermal power has reached to 15% of plant rating. Prior to plant shutdown, twenty-four hours of operation above 3.5% oxygen at greater than 15% of plant rating is allowed. All piping outside the outboard primary containment isolation valves

carrying nitrogen are protected from overpressurization by relief valves ducted to the outdoor atmosphere.

6.2.5.2.3 Nitrogen Makeup

(1) The nitrogen makeup equipment is sized to maintain a positive pressure in the drywell and wetwell during the maximum drywell cooldown rate not caused by spray actuation.

(2) Automatic addition of nitrogen is physically limited to less than the maximum drywell bleed capacity.

6.2.5.2.4 Drywell Bleed

Primary containment bleed capability is provided in accordance with Regulatory Guide 1.7 and as an aid in cleanup following an accident. During normal plant operation, the bleed line also functions, in conjunction with the nitrogen purge line, to maintain primary containment pressure at about 5.2 kPaG and oxygen concentration below 3.5% by volume. This is accomplished by makeup of the required quantity of nitrogen into the primary containment through the makeup line or relieving pressure through the bleed line. The drywell bleed line is manually operable from the control room. Flow through the bleed line will be directed through either the SGTS or the secondary containment HVAC (SCHVAC), and be monitored by the SGTS and SCHVAC flow and radiation instrumentation. All ACS primary containment isolation valves are automatically closed when high radiation is detected in the exhaust flow (Table 5.2-6).

The drywell bleed line is located above an elevation which would be covered by post-LOCA flooding for unloading the fuel.

6.2.5.2.5 Pressure Control

(1) In general, during startup, normal, and abnormal operation, the wetwell and drywell pressure is maintained greater than 0 kPaG to prevent leakage of air (oxygen) into the primary containment from secondary containment but less than the nominal 13.7 kPaG scram setpoint. Sufficient margin is provided such that normal containment temperature and pressure fluctuations do not cause either of the two limits to be reached considering variations in initial containment conditions, instrumentation errors, operator and equipment response time, and equipment performance.

(2) Nitrogen makeup automatically maintains a 5.2 kPaG positive pressure to avoid leakage of air from the secondary into the primary containment.

(3) The drywell bleed sizing is capable of maintaining the primary containment pressure less than 8.6 kPaG during the maximum containment atmospheric heating which could occur during plant startup.

6.2.5.2.6 Overpressure Protection

6.2.5.2.6.1 General

(1) The system is designed to passively relieve the wetwell vapor space pressure at 617.8 kPaG (Subsection 19E.2.8.1). The system valves are capable of being closed from the main control room using AC power and pneumatic air.

(2) The vent system is sized so that residual core thermal power in the form of steam can be passed through the relief piping to the stack.

(3) The initial driving force for pressure relief is assumed to be the expected pressure setpoint of the rupture disk.

(4) The rupture disk is designed to prevent flow in the containment overpressure relief piping until a specified rupture pressure is reached. It is constructed of stainless steel or a material of similar corrosion resistance.

(5) A number of rupture disks are procured at the same time and made from the same sheet to provide uniformity of relief pressure.

(6) The rupture disk is part of the primary containment boundary and is able to withstand the containment design pressure (309.9 kPa) with no leakage to the environment. It is also capable of withstanding full vacuum in the wetwell vapor space without leakage. The disk ruptures at 617.8 kPa due to overpressurization during a severe accident as required to assure containment structural integrity. As potential backup to a leaking, fractured or improperly sealed rupture disk, the two valves upstream of the disk can be closed. These valves are safety-related and are subjected to all testing required for normal isolation valves. The solenoids in these valves are powered by Vital AC (VAC). These valves are capable of closing against pressures up to 617.8 kPaG.

(7) The piping material is carbon steel. The design pressure is 1030 kPaG, and the design temperature is 171°C.

6.2.5.2.6.2 Containment Overpressure Protection System

ABWR has a very low core damage frequency. Furthermore, in the unlikely event of an accident resulting in core damage, the fission products are typically trapped in the containment and there is no release to the environment. Nonetheless, in order to mitigate the consequences of a severe accident which results in the release of fission products and to limit the effects of uncertainties in severe accident phenomena, ABWR is equipped with a containment overpressure protection system (COPS). This system is intended to provide protection against the rare sequences in which structural integrity of the containment is challenged by

overpressurization. It has been determined that these rare sequences comprise a small percentage of the hypothesized severe accident sequences.

The COPS is part of the atmospheric control system and consists of two 200 mm diameter overpressure relief rupture disks mounted in series on a 250 A line which connects the wetwell airspace to the stack. The second rupture disk, located at the inlet to the plant stack, has a very low set point, less than 0.03 MPaD. The setpoint of the inner rupture disk, located near the containment boundary, will be selected such that the COPS opens when the wetwell pressure is 0.72 MPaA. The COPS provides a fission product release point at a time prior to containment structural failure. Thus, the containment structure will not fail. By engineering the release point in the wetwell airspace, the escaping fission products are forced through the suppression pool. In a core damage event initiated by a transient in which the vessel does not fail, fission products are directed to the suppression pool via the SRVs, scrubbing any potential release. In a severe accident with core damage and vessel failure or in a LOCA which leads to core damage, the fission products will be directed from the vessel and drywell through the drywell connecting vents and into the suppression pool again ensuring any release is scrubbed. Eventually, if containment pressure cannot be controlled, the rupture disk opens. Any fission product release to the environment is greatly reduced by the scrubbing provided by the suppression pool.

In the absence of the COPS, unmitigated overpressurization of the containment would result in failure of the drywell head for most severe accident scenarios (some high-pressure core melt sequences result in fission product leakage through the moveable penetrations in the drywell rather than drywell head failure).

6.2.5.2.6.3 Pressure Setpoint Determination

Several factors were considered in determining the optimum pressure setpoint for the rupture disk. The results of the previous analysis show that it is desirable to avoid drywell head failure. This can be assured by providing a rupture disk pressure setpoint below the pressure that would begin to challenge the structural integrity of the containment. However, as the pressure setpoint is reduced, the time to containment failure and fission product release is also reduced. Thus, the setpoint of the rupture disk must optimize these competing factors: minimizing the probability of drywell head failure while maximizing time before fission product release to the environment.

The service level C capability of the containment serves as one indication of a lower bound for the structural integrity of the containment. As shown in Appendix 19F, the service level C for the ABWR is 0.77 MPaA, limited by the drywell head. Thus, it is desirable to set the rupture disk setpoint below this value.

The distribution of drywell head failure pressure and the distribution of rupture disk burst pressure were also considered in determining the burst pressure. The drywell head failure pressure is assumed to have a lognormal distribution with a median failure pressure equal to its ultimate strength of 1.025 MPa. The variability of rupture disk opening pressures is best modeled with a normal or Gaussian distribution. Typical high quality rupture disk exhibits a

tolerance of ±5% of the mean opening pressure. Tests have shown that this ±5% tolerance spans ±2 to ±2.5 standard deviations of the rupture disk population. This analysis of the containment overpressure protection system conservatively assumes that only ±2 standard deviations are included within the ±5% tolerance. Because the setpoint of the other rupture disk is very low, the variability of the pressure is neglected in comparison to the variability of the inner high pressure disk.

A critical parameter in determining the risk of drywell head failure before rupture disk opening is the pressure difference between the drywell and wetwell. Late in an accident the drywell is at higher pressure than the wetwell. For a given rupture disk setpoint, the probability of drywell head failure increases as the pressure difference increases. The maximum drywell to wetwell pressure difference is 0.1 MPa. This pressure difference occurs for cases in which firewater spray was activated after vessel failure but terminated before containment failure. Cases without firewater spray have pressure differences of no more than 0.05 MPa.

A COPS system setpoint of 0.72 MPaA at 93°C was chosen. The residual risk of drywell head failure may be calculated by combining the two distributions with an offset corresponding to the pressure difference between the wetwell and the drywell. A 0.72 MPaA setpoint results in a small probability of drywell head failure prior to rupture disk opening for a 0.1 MPa drywell to wetwell pressure difference. For a drywell to wetwell pressure difference of 0.05 MPa, the drywell head failure probability prior to rupture disk opening is smaller. This is judged to be an acceptable level of risk.

6.2.5.2.6.4 Variability in Rupture Disk Setpoint

Nickel was chosen as the material for the rupture disk for evaluation purposes due to its relative insensitivity to changes in temperature. At temperatures above room temperature the opening pressure of a typical nickel rupture disk will decrease by about 2% for a 56°C increase in temperature. Thus, in order to estimate the uncertainty due to variations in the temperature of the ABWR rupture disk, a sensitivity study was performed in which the pressure setpoint of the rupture disk was varied.

The nominal pressure setpoint of the rupture disk is 0.72 MPaA at 93°C. Two cases were examined using MAAP ABWR in this sensitivity study. For both cases the LCLP-PF-R sequence was used as the base case. First, the rupture disk pressure setpoint was reduced to 0.708 MPaA which corresponds to a rupture disk temperature of 149°C; and second, the pressure setpoint was increased to 0.735 MPaA which corresponds to a temperature of 38°C. This temperature range, from 38°C to 149°C, bounds all anticipated rupture disk temperatures.

The elapsed time to rupture disk opening was within 0.8 hours of the base case value of 20.2 hours for both cases tested. Higher rupture disk temperatures (i.e., lower pressure setpoints) reduce the time to rupture disk opening and lower rupture disk temperatures (i.e., higher pressure setpoints) increase the time to rupture disk opening. There were no significant changes

in fission product release. For both cases the CsI release fraction at 72 hours remained less than 1E-7.

Another parameter affected by the variation in the rupture disk temperature is the probability of drywell head failure prior to rupture disk opening in a severe accident. Using the rupture disk and drywell head failure distributions, it was determined that the probability of drywell head failure prior to rupture disk opening increased slightly for the case with rupture disk temperature of 38°C. With a rupture disk temperature of 149°C, the probability decreased slightly. The rupture disk temperature variation has a similar effect on the severe accident sequences in which the firewater spray system is activated. The probability of drywell head failure prior to rupture disk opening increases slightly for the case with rupture disk temperature of 38°C and decreases slightly for the case with rupture disk temperature of 149°C.

The results of this sensitivity study show that variations in rupture disk temperature, which cause small variations in rupture disk opening pressure, have a minor effect on the performance of the ABWR containment overpressure protection system.

6.2.5.2.6.5 Sizing of Rupture Disk

The size of the rupture disk has also been optimized. If the rupture disk is too small, it could be incapable of venting enough steam to prevent further containment pressurization. On the other hand, if the rupture disk is too large, level swell in the suppression pool could introduce water into the COPS piping. If this were to occur, the piping could be damaged or there could be carryover of waterborne fission products from the containment.

A 200A rupture disk was selected. This is sufficient to allow 35 kg/s of steam flow at the opening pressure of 0.72 MPaA and corresponds to an energy flow of about 2.4% rated power. The minimum acceptable flow rate is 28 kg/s of steam flow at the same pressure. For virtually all severe accident sequences, the rupture disk would not be called upon until about 20 hours after scram. The decay heat level at this time is less than 0.5%. Thus, there is ample margin in the sizing of the rupture disk for severe accidents.

An additional accident was considered in the selection of the rupture disk size. In the event of an ATWS with the additional failure of the standby liquid control system, the operator is directed to lower water level to control power. Analysis has shown that the RHR system is capable of removing the energy generated by the ATWS from the containment (Subsection 19.3.1.3.1). If the additional failure of containment heat removal is assumed, a simple calculation indicates that the rupture disk area is just sufficient to limit the containment pressure below service level C.

Calculations were also performed to investigate the potential effects of pool swell and fission product carryover at the time of COPS operation. These analyses (Subsection 19E.2.3.5) indicate that pool swell does not threaten the integrity of the COPS piping and that no significant entrainment of fission products will occur due to carryover.

6.2.5.2.6.6 Performance for Inadvertent Actuation COPS

The potential for increased risk due to the rupture disk opening early has been considered. It is assumed that recovery of RHR capability is sufficient to terminate containment pressurization and prevent drywell head failure. In the few hours between rupture disk opening and hypothetical drywell head failure for the LCLP-FS sequence, the probability of recovering RHR capability is very small (Subsection 19.3.2.7). This represents the probability that the COPS was opened unnecessarily since RHR would have been recovered in this time period.

There is a small probability that RHR will be recovered before the time at which containment would fail if the rupture disk setpoint has been surpassed. In light of this fact and given the difference in magnitude of the fission product release, it is clearly preferable to direct the fission products through the rupture disk.

6.2.5.2.6.7 Suppression Pool Bypass

A comparison of performance for cases with suppression pool bypass flow through an open vacuum breaker valve was also considered. Cases were run with bypass effective area varying from 5 to 2030 cm^2. A fully open vacuum breaker has an effective area of 2030 cm^2. The dominant loss of all core coolant with vessel failure at low pressure sequence was considered with passive flooder operation since previous analysis has shown that the firewater system is capable of mitigating bypass.

No credit was taken for aerosol plugging of the bypass leakage in this analysis; and, therefore, the results are conservative. Also, it was assumed that the bypass leakage was present from the beginning of the accident sequence. As the bypass area increases, the fraction of fission product aerosols which pass through the suppression pool decreases. Thus, the benefit of a wetwell release of fission products is significantly reduced as the bypass area increases.

For bypass effective areas less than 50 cm^2, CsI releases at 72 hours from the COPS cases were smaller than from the corresponding drywell head failure cases. However, the differences in CsI releases at 72 hours were only factors of 2 to 4 rather than several orders of magnitude. The time difference between drywell head failure and rupture disk opening was 4 to 8 hours for these small bypass areas. For bypass effective areas greater than 50 cm^2 CsI release fractions at 72 hours are on the order of 10% for both the drywell head failure cases and the COPS cases. On the other hand, the time difference between rupture disk opening and drywell head failure is only 2 to 4 hours for these larger bypass areas. These relatively small time differences will not significantly affect the magnitude of the offsite dose. Attachment 19EE has a complete discussion of suppression pool bypass flow through vacuum breaker valves.

6.2.5.2.6.8 Potential Impact of Hydrogen Burning and Detonation

Hydrogen burning and detonation are not a concern for the ABWR containment because the containment is inerted with nitrogen. There could be a potential for burning in the COPS system

and the stack after the rupture disk opens. However, due to the design and operation of the COPS system, this issue does not have an impact on risk.

Hydrogen burning and detonation will be precluded in the piping associated with the COPS system. The piping will be inerted during operation with the rupture disk located at the inlet of the stack. This, combined with initial purging of the piping, will ensure that the inertion of the containment will extend out to the stack, and prevent burning of hydrogen in the portion of the COPS system which is within the Reactor Building. Therefore, there will be no concern of the leading edge of the containment atmosphere mixing with the gas in the piping and causing a burn. After passing of the leading edge of the gas flow, the mixture in the piping will be identical to that in the containment. The gas flow through the system will prevent the backflow of air into the COPS piping.

Hydrogen burning could occur in the plant stack as the gas flow enters the stack. The stack is a non-seismic structure located on top of the Reactor Building. Because of this configuration, the Reactor Building has been designed to withstand the loads associated with the collapse of the plant stack. Furthermore, no credit is taken in the analysis for the plant stack to reduce the offsite dose by providing for an elevated release. All releases were presumed to occur at the elevation of the top of the Reactor Building. Therefore, hydrogen burning or detonation in the stack will have no impact on the consequences of a severe accident as modeled in this analysis.

No burning will occur within the COPS piping. Furthermore, no credit was taken for the plant stack to reduce the source term to the environment and the Reactor Building can withstand the collapse of the plant stack. Therefore, hydrogen burn or detonation in the COPS system will have no impact on risk and no further consideration of this phenomenon is required.

6.2.5.2.6.9 Summary

A wetwell pressure setpoint of 617.8 kPaG for the overpressure relief rupture disk meets the design goal. The probability of containment structural failure is minimized while maximizing the time to fission product release in a severe accident. The small probability of containment structural failure if the pressure reaches the rupture disk setpoint in a severe accident, combined with the already low core damage frequency and reliable containment heat removal, produces an extremely low probability of significant fission product release. In addition, the elapsed time to rupture disk opening is greater than 24 hours for most severe accident sequences.

6.2.5.2.7 Not Used

6.2.5.3 Design Evaluation

The ACS is designed to maintain the containment in an inert condition except for nitrogen makeup needed to maintain a positive containment pressure and prevent air (0_2) leakage from the secondary into the primary containment.

The primary containment atmosphere will be inerted with nitrogen during normal operation of the plant. Oxygen concentration in the primary containment will be maintained below 3.5% by volume measured on a dry basis.

During normal operation, nitrogen makeup and containment pressure control are accomplished using only the 50A supply lines. The large valves (550A) in the containment ventilation lines are closed and flow to the plant stack through the overpressure protection line (250A) is prevented by the rupture disk.

The following conditions assure that the large (550A) containment purge and vent lines will be isolated following a LOCA:

(1) The valves remain closed at all times during normal operation and will only be opened for inerting or de-inerting at the beginning and end of a shutdown.

(2) The valves and piping provide redundancy such that no single failure can prevent isolation of the purge and vent lines.

(3) In the event of a LOCA, the valves receive an isolation signal.

(4) The valves fail in the closed position. If electrical power to the solenoids is lost or the pneumatic pressure fails, the valves will close.

Following an accident, hydrogen concentration will increase due to the addition of hydrogen from the specified design-basis metal-water reaction. Hydrogen concentration will also increase due to radiolysis. Any increase in hydrogen concentration is of lesser concern because the containment is inerted. Due to dilution, additional hydrogen moves the operating point (O_2 Concentration) of the containment atmosphere farther from the envelope of flammability.

Containment oxygen concentration also increases due to radiolysis. During plant operation, there are no other sources of oxygen in the containment.

In the ABWR, there are no design basis events that result in core uncovery or core heatup sufficient to cause significant metal-water reaction. Therefore, per Regulatory Guide 1.7, the design basis metal/water reaction is that equivalent to the reaction of the active clad to a depth of 0.0058 mm. This is equivalent to 0.72% of the active clad. Radiolysis is calculated based on Regulatory Guide 1.7 source terms.

Overpressure relief is provided to passively relieve the containment pressure, as required, by venting the wetwell atmosphere to the plant stack. Venting the wetwell airspace to the plant stack precludes an uncontrolled containment failure. Venting from the wetwell, as opposed to the drywell, takes advantage of the decontamination factor provided by the suppression pool. Venting to the stack provides a monitored, elevated release.

Unintended opening of the COPS rupture disk is highly unlikely. Unintended operation at a lower pressure, such as during a design basis accident, would not significantly affect offsite doses, since no fuel failures would be expected. Failure of the rupture disk would be required for this unintended operation. In addition, the butterfly valves could be closed if inline radiation monitoring indicated unexplained flow in the relief line.

6.2.5.4 Tests and Inspections

Complete process systems are pressure tested to the maximum practicable extent. Piping systems will be hydrostatically tested in their entirety, utilizing available valves or temporary plugs. Hydrostatic testing of piping systems will be performed at a pressure 1.5 times the design pressure, but in no case at less than 519.8 kPaG. The test pressure will be held for a minimum of 30 minutes. Pneumatic testing may be substituted for hydrostatic testing in accordance with the applicable codes.

Preoperational testing will demonstrate the ability of the ACS to meet design requirements. Each valve will be exercised both opened and closed and position indication verified. Trip and alarm logic signals will also be checked. The tests assure correct functioning of all controls, instrumentation, compressors, recombiners, piping and valves. System reference characteristics, such as pressure differentials and flow rates, are documented during the preoperational tests and are used as base points for measurements in subsequent operational tests.

During plant operation, the ACS, its valves, instrumentation, wiring and other components outside the containment can be inspected visually at any time. Testing frequencies of the ACS components are generally correlated with testing frequencies of the associated controls and instrumentation. When a valve control is tested, the operability of that valve and its associated instrumentation are generally tested by the same action. In addition, inservice inspection and testing of all ASME Section III, Class 3 components is done in accordance with Subsections 6.6.5 and 3.9.6, respectively.

Preoperational tests of the ACS are conducted during the final stages of plant construction prior to initial startup.

The overpressure protection concept was designed to minimize any adverse impact on normal operation or maintenance. Initially, several rupture disks from a batch of rupture disk could be tested to verify the opening characteristics and setpoint. The disk would be replaced every five years according to normal industry practice. The installation of the disk would not impact containment leakage tests, since disk integrity is expected to be essentially perfect.

The overpressure protection valves would be tested during preoperational testing and periodically during inservice testing (Subsection 3.9.6), to verify their normally open position and their ability to close using AC power and pneumatic air.

6.2.5.5 Instrumentation Requirements

Separate inerting flow indication to both the drywell and wetwell are provided. Drywell pressure and makeup flow are monitored and recorded in the main control room. Additional drywell pressure instrumentation, with a lower setpoint, is provided in addition to the redundant, safety-grade drywell pressure instrumentation of the Nuclear Boiler System. If drywell pressure exceeds a given setpoint, the nitrogen makeup flow is shut off as the inerting valves are closed. The temperature of the makeup and inerting vaporizers nitrogen outlet are monitored. Low makeup vaporizer nitrogen outlet temperature alarms (only) in the main control room. Auxiliary steam feeding the main inerting vaporizer(s) is controlled to regulate the inerting vaporizer nitrogen outlet temperature. Low inerting vaporizer nitrogen outlet temperature sounds a local alarm and low-low temperature isolates the main inerting line. It is intended that the local panel be attended full-time during all main inerting operations. All locally-mounted instruments are easily read from the local ACS panel. Keylocked switches in the main control room are provided to override the containment isolation signal to the valves, providing nitrogen makeup to the drywell and wetwell and the small 50A pipe size drywell vent line. Position indication in the main control room is provided for all remotely-operated valves.

Backup purge and the addition of makeup nitrogen is initiated at the operator’s discretion.

Design details and logic of the instrumentation are discussed in Chapter 7.

As discussed in Subsection 6.2.5.2, oxygen monitoring is provided in the wetwell and drywell by the CAMS. This monitoring function, when used during normal operation, determines when the primary containment is inert and nitrogen purging may be terminated. It also determines when primary containment is de-inerted and personnel re-enter procedures may be initiated.

The CAMS oxygen monitors assure safe personnel entry into the primary containment after shutdown. In addition, CAMS assures that the primary containment is in an inert condition during startup, normal and abnormal operation conditions. This system has a measurement range of 0 to 25% (by volume) at 100% relative humidity. The minimum and maximum inlet temperature to the oxygen monitor will be 10°C and 65°C, respectively. Two sample points are provided in both the drywell and wetwell, high and low in their respective compartments and in opposing quadrants. Each airlock can also be sampled.

The sample lines are sized and sloped to assure draining condensation to the containment.There are no loops in the sample lines which could collect water and block flow. The oxygen monitors provide indication outside of the primary containment where necessary (for example, at and in each airlock) to assure safe operator access into first the airlock and then the containment.

The CAMS oxygen analyzing system is provided to indicate the concentration of oxygen inside the containment during reactor operation, and to aid in maintaining the oxygen concentration below a safety limit prescribed in the plant Technical Specifications. The oxygen analyzing system readings are not used as a basis for determining when drywell entry criteria are not met.

The only role of this system related to drywell purging for re-entry is to indicate when oxygen levels are high enough to start taking the samples that will be used for determining compliance with entry criteria.

6.2.5.6 Personnel Safety

Entry into a nitrogen atmosphere is particularly hazardous due to the fact that the body cannot easily detect relative changes in the nitrogen content of the air. Low oxygen causes blood chemistry changes that can lead to an automatic increase in breathing rate, leading to hyperventilation. The individual can lose consciousness in 20 to 40 seconds and be totally unable to save himself.

A general procedure which outlines the critical items to be included in any procedure controlling purged drywell entry is provided below. This procedure is intended to be a framework of minimum requirements for drywell entry and for general guidance. The COL applicant will provide specific, detailed site procedures and administrative controls to meet the specific needs of each particular physical plant and administrative setup.

General Procedure Drywell Entrance Control Following De-inerting

(1) Inerting and de-inerting of the drywell shall be in conformance with applicable Technical Specifications.

(2) Personnel access to the drywell is normally prohibited at all times when the drywell has an oxygen-deficient atmosphere, unless an emergency condition arises, in which case the procedure outlined in Subsection 6.2.5.6(8) should be followed.

(3) The status of the drywell atmosphere shall be posted at the drywell entrance at all times, and the entrance locked, except when cleared for entry.

(4) Suitable authorization, control and recording procedures shall be established and remain in effect throughout the entry process.

(5) Prior to initial entry, the drywell shall be purged with air in accordance with operating procedure until drywell samples indicate that the following conditions are met:

   (a) Oxygen: Greater than 16.5% content by volume.

   (b) Hydrogen: Less than 14% of the lower limit of flammability, or a limit of 0.57% hydrogen by volume. (The lower flammability limit is 4.1% hydrogen content by volume.)

   (c) Carbon Monoxide: Less than 100 ppm.

   (d) Carbon Dioxide: Less than 5000 ppm.

(e) Airborne Activity: Less than applicable limits in 10CFR20, or equivalent.

(6) During the purge, drywell atmosphere samples shall be drawn from a number of locations when the drywell oxygen analyzer indicates an oxygen concentration of 16.5% or greater.

Samples shall be analyzed for oxygen, hydrogen, carbon monoxide, carbon dioxide and airborne activity.

When the results of two successive samples taken at least one-half hour apart are found to be within the conditions in Subsection 6.2.5.6(5), initial entry may be authorized.

(7) Criteria for entry are:

(a) The initial entry will require a minimum of two (2) persons.

(b) Initial entry will require, in addition to normal protective clothing and protective equipment consisting of self-contained breathing apparatus (such as Scott Air Pack), portable air sampling and monitoring equipment, and portable radiation survey meters.

(c) A means of communication shall be established.

(8) Under certain conditions, the Station Superintendent may deem that an emergency condition exists which would justify drywell entry with an oxygen-deficient atmosphere.

When it has been determined from the results of the initial entry survey and samples that the entire drywell atmosphere meets the required conditions, the drywell may be cleared for general access and the drywell status posted at the drywell entrance.

6.2.6 Containment Leakage Testing

This section includes criteria for the containment integrated leakage rate test (Type A test), containment penetration leakage rate test (Type B test) and containment isolation valve leakage rate test (Type C test) that complies with 10CFR50, Appendix J and General Design Criteria 52, 53 and 54 in 10CFR50, Appendix A.

Testing requirements for piping penetration isolation barriers and valves have been established using the intent of General Design Criterion 54, as interpreted in 10CFR50 Appendix J.

Structural integrity tests of the containment, as described in Subsection 3.8.1, will be satisfactorily completed prior to performance of the preoperational integrated leakage rate tests.

Periodic Type A, B and C tests will be performed to assure that leakage through the containment and systems and components that penetrate primary containment do not exceed allowable leakage rate values specified in the standard technical specifications. Maintenance and repairs will be performed during the service life of the containment including repairs on systems and components penetrating the containment to restore any leakage paths to acceptable values.

6.2.6.1 Containment Integrated Leakage Rate Test

6.2.6.1.1 Initial Integrated Leak Rate Test

After completion of construction of the primary reactor containment, including installation of all portions of mechanical, fluid, electrical and instrumentation systems penetrating the containment pressure boundary, and upon satisfactory completion of all structural integrity tests, the preoperational integrated leakage rate Type A tests will be conducted to meet the requirements of 10CFR50 Appendix J.

6.2.6.1.1.1 Objectives

Objectives of the initial integrated leak rate test (ILRT) follow:

(1) Verify that the total measured integrated leakage rate, L_{am}, does not exceed the containment design basis accident (DBA) leakage rate, L_a*, which is 0.5% (excluding MSIV leakage) by weight of the contained atmosphere in 24 h, at a calculated peak containment internal pressure, P_a, related to the DBA.

(2) Calculate a maximum allowable leakage rate, L_t, at reduced pressure, P_t, which will be used during subsequent integrated leakage rate tests.

(3) Obtain data which may be used to develop the leakage rate characteristics and history of the containment system.

(4) Demonstrate by a verification test the accuracy of the integrated leakage rate instrumentation to satisfactorily determine the containment integrated leakage rate.

6.2.6.1.1.2 Preoperational Test Procedure

The preoperational test will be conducted in two phases per “Preoperational Leakage Rate Tests”. The first phase of the test will be performed with the containment vessel pressurized to pressure P_t, not less than 0.50 P_a to measure a leakage rate, identified as L_{tm}. The second phase is then conducted at pressure P_a resulting in a measured leakage rate identified as L_{am}. The absolute method shall be employed for determining the leakage rate (ANSI N45.4 Subsection 5.2.1 and Section 7.9). Test duration of each phase shall be sufficient for pressure and temperature stabilization. To ensure uniform temperature distribution, fans will be provided to

* See Appendix J of 10CFR50 for definition of all terms.

circulate air in the containment during the test. Prior to commencement of the tests, the test prerequisites described in Subsections 6.2.6.1.2.1 and 6.2.6.1.3 will be met.

6.2.6.1.1.3 Supplement Verification Test

The accuracy of the leakage rate tests is verified by using a supplemental method of leakage measurement. Verification is obtained by superimposing a controlled and measurable leak on the normal containment leakage rate or other methods of demonstrated equivalency. The difference between the total leakage and the superimposed known-leakage results in the actual leakage rate. This leakage rate is a check against its accuracy and is acceptable provided the correlation between the supplemental test data and integrated leak test data demonstrates an agreement within ± 25%. Conduct of the verification test is normally accomplished after completion of each test phase of the ILRT. Complete descriptive details are found in Appendix C of ANSI N45.4.

6.2.6.1.1.4 Instrumentation Requirements

Instrumentation provided to monitor the containment leakage rate testing is designed, calibrated and tested to accurately ensure that the containment atmosphere parameters can be precisely measured.

6.2.6.1.1.5 Acceptance Criteria

The initial allowable leakage rate (L_{tm}) at test pressure P_t shall not exceed 75% of the maximum allowable test leakage rate (L_t), where L_t is defined as follows:

$$L_t = L_a \frac{L_{tm}}{L_{am}} \text{ for values of } \frac{L_{tm}}{L_{am}} \leq 0.7 \qquad (6.2\text{-}1)$$

$$L_t = L_a \left(\frac{P_t}{Pa}\right)^{\frac{1}{2}} \quad \text{for values of } \frac{L_{tm}}{L_{am}} > 0.7 \qquad (6.2\text{-}2)$$

The leakage L_{am} shall be less than 0.75 L_a and not greater than the design leakage rate (L_d).

6.2.6.1.2 Periodic Leakage Rate Tests

Leakage rate tests are conducted periodically in conformance to Appendix J of 10CFR50 to ensure that the integrity of the containment is maintained and to determine if any leakage increase has developed since the previous ILRT. The tests are performed at regular intervals, after major repairs or upon indication of excessive leakage, as specified in the standard Technical Specification for the ABWR.

6.2.6.1.2.1 Integrated Leakage Rate Test (ILRT, Type A)

Type A tests are conducted periodically, following the initial preoperational tests, at test pressure P_t only. Except for the elimination of the P_a pressure test, all ILRTs follow the same format as the initial ILRT, as outlined in Subsection 6.2.6.1.1.

In addition to the normal test prerequisites, the following requirements are mandatory prior to all periodic Type A tests:

(1) A detailed visual examination of critical areas and general inspection of the accessible interior and exterior surfaces of the containment structure and components shall be performed to uncover any evidence of structural deterioration which may affect either the structural integrity or leaktightness of the containment. If there is evidence of significant structural deterioration, Type A tests shall not be performed until corrective action is taken in accordance with approved repair procedures. If leak repairs of testable components are performed, the reduction in leakage shall be measured (at test pressure P_t) and added to the Type A test result.

Except for inspections and actions taken above, no preliminary leak detection surveys and repairs shall be performed prior to the conduct of the Type A test.

(2) Closure of containment isolation valves shall be accomplished by normal mode of actuation and without preliminary exercises or adjustments. All malfunctions and subsequent corrective actions shall be reported to the NRC.

6.2.6.1.2.2 Acceptance Criteria

The measured leakage rate L_{tm} shall not exceed 0.75 L_t as established by the initial ILRT.

(1) If during a Type A test (including the supplemental test) potentially excessive leakage paths are identified which will interfere with satisfactory completion of the test, or which result in the Type A test not meeting the acceptance criteria, the Type A test shall be terminated and the leakage through such paths shall be measured using local leakage testing methods. Repairs and/or adjustments to equipment shall be made and a Type A test performed. The corrective action taken and the change in leakage rate determined from the tests and overall integrated leakage determined from the local leak and Type A tests shall be included in the report submitted to the NRC.

(2) If any Type A test fails to meet the acceptance criteria, prior to corrective action, the test schedule applicable to subsequent Type A tests shall be subject to NRC review and approval.

(3) If two consecutive periodic Type A tests fail to meet the applicable acceptance criteria, prior to corrective action (notwithstanding the established periodic retest schedule), a Type A test shall be performed at each plant shutdown for major refueling, or approximately every 18 months, whichever occurs first, until two consecutive Type A tests meet the acceptance criteria, after which time the previously established periodic retest schedule may be resumed.

6.2.6.1.2.3 Test Frequency

After initial ILRT, a set of three Type A tests shall be performed at approximately equal intervals during each 10-year service period, with the third test of each set coinciding with the end of each 10-year major inservice inspection shutdown. In addition, any major modification or replacement of components of the primary reactor containment performed after the initial ILRT shall be followed by either a Type A or a Type B test of the area affected by the modification, with the affected area to meet the applicable acceptance criteria. The basis for the frequency of testing is established in accordance with 10CFR50 Appendix J.

6.2.6.1.3 Additional Criteria for Integrated Rate Test

(1) Those portions of fluids systems that are part of the reactor coolant pressure boundary, that are open directly to the primary reactor containment atmosphere under post-accident conditions and become an extension of the boundary of the primary reactor containment, shall be opened or vented to the containment atmosphere prior to or during the Type A test. Portions of closed systems inside the containment that penetrate the primary containment and are not relied upon for containment isolation purposes following a LOCA shall be vented to the containment atmosphere.

(2) All vented systems shall be drained of water to the extent necessary to ensure exposure of the system primary containment isolation valves to the containment air test pressure.

(3) Those portions of fluid systems that penetrate the primary containment, that are external to containment and are not designed to provide a containment isolation barrier, shall be vented to the outside atmosphere, as applicable, to assure that full post-accident differential pressure is maintained across the containment isolation barrier.

(4) Systems that are required to maintain the plant in a safe condition during the Type A test shall be operable in their normal mode and are not vented.

(5) Systems that are normally filled with water and operating under post-LOCA conditions need not be vented.

(6) ILRT results from items 4 and 5 above shall be added to the ILRT results.

6.2.6.2 Containment Penetration Leakage Rate Test (Type B)

6.2.6.2.1 General

Containment penetrations whose designs incorporate resilient seals, bellows, gaskets, or sealant compounds, airlocks and lock door seals, equipment and access hatch seals, and electrical canisters, and other such penetrations are leak tested during preoperational testing and at periodic intervals thereafter in conformance to Type B leakage rate tests defined in 10CFR50 Appendix J. A list of all containment penetrations is provided in Table 6.2-8. The leak tests ensure the continuing structural and leak integrity of the penetrations.

To facilitate local leak testing, a permanently installed system may be provided, consisting of a pressurized gas source (nitrogen or air) and the manifolding and valving necessary to subdivide the testable penetrations into groups of two to five. Each group is then pressurized, and if any leakage is detected (by pressure decay or flow meter), individual penetrations can be isolated and tested until the source and nature of the leak is determined. All Type B tests are performed at containment peak accident pressure, Pa. The local leak detection tests of Type B and Type C (Subsection 6.2.6.3) must be completed prior to the preoperational or periodic Type A tests.

See Subsection 6.2.7.5 for COL license information pertaining to containment penetration leak rate testing (Type B).

6.2.6.2.2 Acceptance Criteria

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

All Type B tests are performed at containment peak accident pressure P_a. The acceptance criteria are given in Chapter 16.

6.2.6.2.3 Retest Frequency

In compliance with the requirement of Section III.D.2(a) of 10CFR50 Appendix J, Type B tests (except for air locks) are performed during each reactor shutdown for major fuel reloading, or other convenient intervals, but in no case at intervals greater than two years.* Airlocks opened during periods when containment integrity is required will be tested in manual mode within three days of being opened. If the airlock is to be opened more frequently than once every three days, the airlock will be tested at least once every three days during the period of frequent openings. Airlocks will be tested at initial fuel loading, and at least once every six months

* In compliance with the requirement of Section III.D.2(b)(iii) of Appendix J to 10CFR50

thereafter. Airlocks may be tested at full power so as to avoid shutting down. These airlocks contain no inflatable seals.

Main control room readout of time to next test, test completion and test results is provided. An alarm sounds if the specified interval passes without a test being affected. No direct, safety-related function is served by the seal test instrumentation system.

6.2.6.2.4 Design Provisions for Periodic Pressurization

In order to assure the capability of the containment to withstand the application of peak accident pressure at any time during plant life for the purpose of performing ILRTs, close attention is given to certain design and maintenance provisions. Specifically, the effects of corrosion on the structural integrity of the containment are compensated for by the inclusion of a 60-year service life corrosion allowance, where applicable. Other design features that have the potential to deteriorate with age, such as flexible seals, are carefully inspected and tested as outlined in Subsection 6.2.6.2.2. In this manner, the structural and leakage integrity of the containment remains essentially the same as originally accepted.

6.2.6.3 Containment Isolation Valve Leakage Rate Test (Type C)

6.2.6.3.1 General

Type C tests will be performed on all containment isolation valves required to be tested per 10CFR50 Appendix J. All testing is performed pneumatically, except hydraulic testing may be performed on isolation valve Type C tests using water as a sealant provided that the system line for the valve is not a potential containment atmosphere leak path.

Type C tests (like Type B test) are performed by local pressurization using either 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. The rate of decay of pressure of the known test volume is monitored to calculate leakage rate. For the flowmeter method, required pressure is maintained in the test volume by making up air, nitrogen or water (if applicable) through a calibrated flowmeter. The flowmeter fluid flow rate is the isolation valve (or Type B test volume) leakage rate.

All isolation valve seats which are exposed to containment atmosphere subsequent to a LOCA are tested with air or nitrogen at containment peak accident pressure, P_a.

MSIVs and isolation valves isolated from a sealing system will use a test pressure of at least P_a.

Those valves which are in lines designed to be, or remain, filled with a liquid for at least 30 days subsequent to a loss-of-coolant accident are leakage rate tested with that liquid. The liquid

leakage measured is not converted to equivalent air leakage nor added to the Type B and C test total.

All test connections, vent lines, or drain lines consisting of double barrier (e.g., two valves in series, one valve and a cap, or one valve and a flange), that are connected between isolation valves and form a part of the primary containment boundary need not be Type-C tested due to their infrequent use and multiple barriers as long as the barrier configurations are maintained using an administrative control program. These lines are surveillance inspected at cold shutdown and at 31 day intervals (internal and external to primary containment respectively) as required by the Technical Specifications.

For Type C testing of containment penetrations, all testing will be done in the correct direction unless it can be shown that testing in the reverse direction is equivalent, or more conservative. The correct direction for this design is defined as flow from inside the containment to outside the containment.

6.2.6.3.2 Acceptance Criteria

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

6.2.6.4 Scheduling and Reporting of Periodic Tests

The periodic leakage rate test schedules for Type A, B and C tests are described in Chapter 16.

Type B and C tests may be conducted at any time during normal plant operations or during shutdown periods, as long as the time interval between tests for any individual Type B or C tests does not exceed two years. 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, replacement of component which is part of the primary reactor containment boundary, or resealing a seal welded door, performed after the preoperational leakage rate test will be followed by either a Type A, B, or C test, as applicable for the area affected by the modification. Type A, B and C test results shall be submitted to the NRC in the summary report approximately three months after each test.

Included in the leak rate test summary report will be a report detailing the containment inspection, a report detailing any repairs necessary to pass the tests, and the leak rate test results.

6.2.6.5 Special Testing Requirements

The maximum allowable leakage rate into the secondary containment and the means to verify that the inleakage rate has not been exceeded, as well as the containment leakage rate to the environment, are discussed in Subsections 6.2.3 and 6.5.1.3.

6.2.7 COL License Information

6.2.7.1 Not Used

6.2.7.2 Administrative Control Maintaining Containment Isolation

The COL applicant shall maintain the primary containment boundary by administrative controls in accordance with Subsection 6.2.6.3.1.

6.2.7.3 Not Used

6.2.7.4 Wetwell-to-Drywell Vacuum Breaker Protection

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

6.2.7.5 Containment Penetration Leakage Rate Test (Type B)

The COL applicant shall perform Type B leakage rate tests in conformance with 10CFR50 Appendix J for containment penetrations whose designs incorporate resilient seals, bellows, gaskets, or sealant compounds, airlocks and lock door seals, equipment and access hatch seals, and electrical canisters, and other such penetrations. (See Subsection 6.2.6.2.1)

6.2.8 References

6.2-1 W.J. Bilanin, “The G.E. Mark III Pressure Suppression Containment Analytical Model”, June 1974 (NEDO-20533).

6.2-2 F.J. Moody, “Maximum Discharge Rate of Liquid-Vapor Mixtures from Vessels”, General Electric Company, Report No. NEDO-21052, September 1975.

6.2-3 W.J. Bilanin, “The G.E. Mark III Pressure Suppression Containment Analytical Model”, Supplement 1, September 1975 (NEDO-20533-1).

6.2-4 WCAP-17065-P, "Westinghouse Secondary Containment Analysis Methodology", Revision 0, April 2010.

6.2-5 “Implementation of ABWR Methodology using GOTHIC for STP 3 and 4 Containment Design Analyses”, WCAP-17058-P, Westinghouse Electric Company, LLC, June 2009.

6.2-6 “Nuclear Maintenance Applications Center: Foreign Material Exclusion Guidelines”, EPRI TR 1016315, Electric Power Research Institute, July 2008.

6.2-7 "Guidelines for Achieving Excellence in Foreign Materials Exclusion (FME)", INPO 07-008, Institute of Nuclear Power Operations, December 2007.

Table 6.2-1 Containment Parameters

	Design Parameter	Design Value	Calculated Value[1]
1.	Drywell pressure	309.9 kPaG	281.8 kPaG
2.	Drywell temperature	171.1°C	173.2°C[2]
3.	Wetwell pressure	309.9 kPaG	217.2 kPaG
4.	Wetwell temperature		
	• Gas Space	104 °C	98.6°C
	• Suppression pool	100°C	99.6°C
5.	Drywell-to-wetwell differential pressure	+172.6 kPaD – 13.7 kPaD	+148.3 kPaD – 10.7 kPaD

1 Calculated values from Ref. 6.2-5.

2 Calculated drywell maximum temperature exceeds design temperature for only 2 seconds. See discussion in Subsection 6.2.1.1.2.1.

Table 6.2-2 Containment Design Parameters

	Drywell	Wetwell
A. Drywell and Wetwell*		
1. Internal Design Pressure (kPaG)	309.9	309.9
2. Negative Design Pressure (kPaG)	–13.7	–13.7
3. Design Temperature (°C)	171.1	104
4. Net Free Volume (m^3)	7350	5960
5. Maximum allowable leak rate† (%/day)	0.5	0.5
6. Minimum Suppression Pool Water Volume (m^3)	—	3580
7. Suppression pool depth (m)		
Low Level	—	7
High Level	—	7.1
B. Vent System		
1. Number of Vents		30
2. Nominal Vent Diameter (m)		0.7
3. Total Vent Area (m^2)		11.55
4. Vent Centerline Submergence		
Low Level, (m)		
Top Row		3.5
Middle Row		4.9
Bottom Row		6.2
5. Vent Loss Coefficient		
(Varies with number of vents open)		‡

* Items A.1, A.2, A.3 and A.5 apply to related structures including lower drywell access tunnels, drywell equipment hatches, drywell personnel locks and drywell head.

† Corresponds to calculated peak containment pressure related to the design basis accident conditions. Excludes MSIV leakage.

‡ Provided in Section 6.1 of Reference 6.2-5.

Table 6.2-2a Engineered Safety Systems Information for Containment Response Analyses

	Full Capacity	Containment Analysis Value
A. Containment Spray		
1. Number of RHR Pumps	1(*)	1(*)
2. Number of Lines	1(*)	1(*)
3. Number of Heat Exchangers	1(†)	1(†)
4. Drywell Flow Rate (kg/h)	0.84×10^6	0.84×10^6
5. Wetwell Flow Rate (kg/h)	1.14×10^5	1.14×10^5
B. Containment Cooling System		
1. Number of RHR Pumps	3	2
2. Pump Capacity (m^3/h/pump)	954	954
3. RHR Heat Exchangers		
a. Type–U-tube,		
b. Number	3	2
c. Heat Transfer Area (m^2/unit)	‡	‡
d. Overall Heat Transfer Coefficent (Btu/h—m^2-°C/unit)	‡	‡
e. Reactor Cooling Water Flowrate (kg/h)	1.2×10^6	1.2×10^6
f. Maximum Cooling Water Inlet Temperature (°C)	35	35

* Two redundant loops available with one pump each.

† One header each for drywell and wetwell.

‡ The RHR heat exchanger characteristic has been defined by an overall K coeficient based on a temperature difference and the heat rate. The defining equation is:

Q = (K) (ΔT)

$$Q, \frac{\text{kcal}}{\text{s}} = \left(K, \frac{\text{kcal}}{(\text{s}°\text{C})}\right)\left(\Delta(T, °C)\right)$$

The K value is 427 kJ/s°C.
The applicable temperature difference occurs form the RHR heat exchanger's reactor side inlet to the ultimate heat sink temperature. Thus, K is a characteristic of the combined RHR and reactor cooling water system's heat exchangers.

Table 6.2-2b Net Positive Suction Head (NPSH) Available to RHR Pumps

A. Suppression pool is at its minimum depth, El. –3740 mm.

B. NPSH Reference Level is at El. –7200 mm.

C. Suppression pool water is at its maximum temperature for the given operating mode, 100°C.

D. Pressure is atmospheric above the suppression pool.

E. Minimum suction strainer area as committed to by Appendix 6C methods.

NPSH available $= H_{ATM} + H_S - H_{VAP} - (H_F + H_{ST})$

where:

H_{ATM} = Atmospheric head

H_S = Static head

H_{VAP} = Vapor pressure head

H_F = Maximum Frictional head excluding strainer frictional head

H_{ST} = Strainer frictional head

Minimum Expected NPSH

RHR Pump Runout is 1130 m^3/h.

Maximum suppression pool temperature is 100°C.

H_{ATM} =10.77m

H_S =3.46m

H_{VAP} =10.77m

NPSH available = $10.77 + 3.46 - 10.77 - (H_F + H_{ST}) = 3.46 - (H_F + H_{ST})$

NPSH required = 2.0m

Margin =$1.46 - (H_F+H_{ST})$ = NPSH available – NPSH required

* NPSH Reference level is 1m above the pump floor level.

** The final system design will meet the required NPSH with adequate margin.

Table 6.2-2c Net Positive Suction Head (NPSH) Available to HPCF Pumps

A. Suppression pool is at its minimum depth, El. –3740 mm.

B. Centerline of pump suction NPSH Reference level is at El. –7200 mm.

C. Suppression pool water is at its maximum temperature for the given operating mode, 100°C.

D. Pressure is atmospheric above the suppression pool.

E. Minimum suction strainer area as committed to by Appendix 6C methods.

NPSH available = $H_{ATM} + H_S - H_{VAP} - (H_F + H_{ST})$

where:

H_{ATM} = Atmospheric head

H_S = Static head

H_{VAP} = vapor pressure head

H_F = Maximum Frictional head excluding strainer frictional head

H_{ST} = Strainer frictional head

Minimum Expected NPSH

HPCF Pump Runout is 890 m^3/h.

Maximum suppression pool temperature is 100°C

H_{ATM} = 10.77m

H_S = 3.46m

H_{VAP} = 10.77m

NPSH available = 10.77 + 3.46 – 10.77 – $(H_F + H_{ST})$ = 3.46 – $(H_F + H_{ST})$

NPSH required = 1.7m

Margin = 1.76 – $(H_F + H_{ST})$ = NPSH available – NPSH required

* NPSH Reference level is 1m above the pump floor level

** The final system design will meet the required NPSH with adequate margin.

Table 6.2-2d Secondary Containment Design and Performance Data

Description	Unit	Value
A.Secondary Containment Design		
1. Free Volume	m^3	8.5×10^4
2. Pressure, mm of water, gauge	mm H_2O	–6.4
3. Leak Rate at Post-accident pressure (% of Secondary Containment Free Volume)	%/day	50
4. Exhaust Fans		
Number	—	2
Type	—	Centrifugal
5. Filters		
a. Basic specification		
Number of filter train	—	1
Type	—	Dust
b. Component specification		
(1) Prefilter		
Number of set	—	1
Type	—	Dry
(2) HEPA filters		
Number of set	—	2
Type (Material)		Glass fiber
(3) Charcoal adsorber		
Number of set	—	1
Type	—	Deep bed
B.Transient Analysis		
1. Initial conditions		
a. Primary Containment		
(1) Pressure	kPa	106.5
(2) Temperature	°C	57.2

Table 6.2-2d Secondary Containment Design and Performance Data (Continued)

Description	Unit	Value	
(3) Outside air temperature			
Summer operation	°C	46.1	
Winter operation	°C	–40.0	
b. Secondary Containment			
(1) Pressure	mm H_2O	–6.4	
(2) Temperature			
Max value in summer	°C	40	
Min value in winter	°C	10	
2. Thickness of Secondary Containment Wall thickness range form	m	0.3–1.5	
3. Thickness of Primary Containment Wall			
a. Concrete Wall	m	2.0	
b. Liner Plate	mm	6.4	
C.Thermal Characteristics		Drywell	Wetwell
1. Primary Containment Wall			
a. Coefficient of Linear Expansion			
Concrete Wall	m/m°C	0.33×10^{-5}	0.33×10^{-5}
Liner Plate	m/m°C	0.37×10^{-5}	0.51×10^{-5}
b. Modulus of Elasticity			
Concrete Wall	MPa	10.3	14.5
Liner Plate	MPa	191	182.7
c. Thermal Conductivity			
Concrete Wall	W/m-K	0.93	0.93
Liner Plate	kJ/h-cm^3°C	187.6	187.6
d. Thermal Capacitance			
Concrete Wall	kJ/h-m^3°C	2023	2023
Liner Plate	kJ/h-m^3°C	3935	3935

Table 6.2-2d Secondary Containment Design and Performance Data (Continued)

Description	Unit	Value
2. Secondary Containment Wall		
a. Thermal Conductivity	W/m-K	438.3
b. Thermal Capacitance	W/m-K	940
3. Heat Transfer Coefficients		
a. Primary Containment—Atmosphere to Primary Containment Wall	kJ/·cm^3·°C	5.03 x 10^{-4}
b. Primary Containment Wall to Secondary Containment Atmosphere	kJ/h·cm^3·°C	12.55 x 10^{-4}
c. Secondary Containment Wall to Secondary Containment Atmosphere	kJ/h·cm^3·°C	12.55 x 10^{-4}
d. Primary Containment Emissivity	—	0.95
e. Secondary Containment Emissivity	—	0.95

Table 6.2-3 Subcompartment Nodal Description

		Initial Conditions				Design Basis Accident Break Characteristics				
Vol. ID	Description	Volume m^3	Temp °C	Pressure kPaA	Humidity %	Break# Location Volume ID	Break Line Identification	Calc Peak Pressure kPaG	Design$ Pressure (Margin) kPaG	Margin %
SS1	Steam Tunnel Reactor Build.	1948	60	101	10.0	SS1	Main Steam	58.8	75.5	28
SS2	Steam Tunnel Betw. RB. &TB.	244	60	101	10.0	SS2	Main Steam	33.3	75.5	127
SS3	Steam Tunnel Inside TB.	850	60	101	10.0	*	‡	‡	‡	-
SS4	Steam Tunnel Inside TB.	178	60	101	10.0	‡	‡	‡	‡	-
SS5	Turbine Building	144982	40	101	10.0	‡	‡	‡	‡	-
SA1	RCIC Pump & Turbine Room	524	40	101	10	SA1	RCIC (Steam)	37.3	103.0	178
SA2	RHR Pump & Heat Exchanger !	686	40	101	10	SA3	RCIC (Steam)	34.3	103.0	202
SA3	ECCS – Div A B1F, B2F, 183F PS	279	40	101	10	SA3	RCIC (Steam)	35.3	103.0	193
SA4	EL –8200 Corridor * !	1954	40	101	10	SA1	RCIC (Steam)	19.6	34.3	72
SA5	EL –1700 Corridor * !	4021	40	101	10	SA1/SA3	RCIC (Steam)	18.6	34.3	85
SA6	Staircase A * !	438	40	101	10	SA1	RCIC (Steam)	18.6	20.6	11
SA7	Staircase B * !	394	40	101	10	SA1	RCIC (Steam)	18.6	20.6	12
SA8	Ground and Refuling Floors * !	28317	40	101	10	SA1/SA3	RCIC (Steam)	13.7	137.7	0
SR1	Steam Tunnel/Turbine Bldg. 1	148202	40	101	10	SR12	CUW	2.9	N/A	N/A
SR5	CUW Pipe Entr/Exit Room	108	40	101	10	SR5	CUW	32.4	103.0	219
SR4	CUW Regener. Heat Exchanger Valve Room	144	40	101	10	SR8	CUW	38.2	103.0	172
SR2	CUW Pipespace	36	40	101	10	SR2	CUW	45.1	103.0	131

Table 6.2-3 Subcompartment Nodal Description (Continued)

		Initial Conditions				Design Basis Accident Break Characteristics				
Vol. ID	Description	Volume m^3	Temp °C	Pressure kPaA	Humidity %	Break# Location Volume ID	Break Line Identification	Calc Peak Pressure kPaG	Design$ Pressure (Margin) kPaG	Margin %
SR12	CUW Non-Regener. Heat Exchanger Valve Room & CUW Pump Valve Room	100	40	101	10	SR12	CUW	44.1	103.0	133
SR8	CUW Non-Regener. & Regen. HX. Rooms	346	40	101	10	SR8	CUW	43.1	103.0	141
SR9	EL - 8200 Corridor * !	1954	40	101	10	SR11	CUW	30.4	34.3	14
SR11	CUW Pump Room A & B	249	40	101	10	SR11	CUW	39.2	103.0	165
SR13	CUW Filter/Demin. Rm. B	51	40	101	10	SR13	CUW	68.6	103.0	51
SR14	CUW Filter/Demin. Rm. A	51	40	101	10	SR14	CUW	74.5	103.0	39
SR15	CUW Filter/Demin. Valve Room A & B	421	40	101	10	SR15	CUW	32.4	103.0	216
SR6	Stair Case A/B * !	438	40	101	10	SR4	CUW	16.7	20.6	23
SR7	El -1700 Corridor * !	4021	40	101	10	SR15	CUW	18.6	34.3	82
SR10	Ground and Refuling Floors * !	28317	40	101	10	SR15	CIW	13.7	13.7	0

*
‡ High Energy Line Break analysis inside Turbine Building is not required.
\# Break in subcompartment causing maximum peak pressure
$ Design pressures are to be used in conjunction with appropriate dynamic load factors for structural evaluation.
! No RCIC or CUW High Energy Line passes through the compartment.

Table 6.2-4 Subcompartment Vent Path Description

Vent Path ID	From Volume Node ID	To Volume Node ID	Flow Choked or Unchoked	Flow Sonic or Subsonic	Vent Area (m^2)	Vent Length (m)	Head Loss Coefficient Forward	Head Loss Coefficient Reverse	Blowout Opening Pressure (DP) (kPaG)
FA1	SA1	SA3	Unchoked	Subsonic	15.89	0.5	1.61	1.07	*
FA2	SA2	SA3			7.43	0.5	1.69	1.34	*
FA3	SA1	SA4			0.56	0.5	1.24	1.25	*
FA4	SA2	SA4			0.56	0.5	1.23	1.25	*
FA5	SA4	SA7			2.04	0.3	1.67	$	10.3
FA6	SA4	SA6			2.04	0.3	1.68	$	10.3
FA7	SA5	SA4			1.86	0.5	1.44	1.44	*
FA8	SA5	SA4			1.86	0.5	1.44	1.44	*
FA9	SA5	SA7			2.04	0.3	1.66	$	10.3
FA10	SA5	SA6			2.04	0.3	1.66	$	10.3
FA11	SA5	SA8			1.86	14.0	0.42	0.42	*
FA12	SA5	SA8			1.86	25.2	0.42	0.42	*
FA13	SA6	SA8			2.04	0.3	1.54	$	10.3
FA14	SA8	SA7			2.04	0.3	0.02	$	10.3
FA15	SA8	SA9			2.32	0.3	1.45	$	13.8
FA16	SA3	SA5			2.04	3.0	0.45	$	10.3
FR1	SR5	SR1			9.29	2.0	0.78	$	3.4
FR2	SR2	SR5			3.72	0.5	0.02	1.54	*
FR3	SR2	SR4			3.72	0.9	0.02	0.73	*
FR4	SR8	SR4			2.32	0.9	1.31	1.24	*
FR5	SR8	SR4			11.61	0.9	1.42	0.90	3.4
FR6	SR8	SR9			2.04	0.3	1.34	$	10.3
FR7	SR9	SR8			2.04	0.9	1.24	$	3.4
FR8	SR9	SR6			2.04	0.3	1.68	$	10.3
FR9	SR7	SR8			2.04	0.3	1.41	$	3.4
FR10	SR9	SR7			1.86	0.5	1.44	1.44	*
FR11	SR9	SR7			1.86	0.5	1.44	1.44	*

Table 6.2-4 Subcompartment Vent Path Description (Continued)

Vent Path ID	From Volume Node ID	To Volume Node ID	Flow Choked or Unchoked	Flow Sonic or Subsonic	Vent Area (m^2)	Vent Length (m)	Head Loss Coefficient Forward	Head Loss Coefficient Reverse	Blowout Opening Pressure (DP) (kPaG)
FR12	SR7	SR6			2.04	0.3	1.66	$	10.3
FR13	SR10	SR3			2.32	0.3	1.45	$	13.8
FR14	SR6	SR10			2.04	0.3	1.54	$	10.3
FR15	SR5	SR10			2.04	0.9	0.92	$	10.3
FR16	SR4	SR7			2.04	0.3	1.33	$	10.3
FR17	SR7	SR10			1.86	14.0	0.42	0.42	*
FR18	SR7	SR10			1.86	25.2	0.42	0.42	*
FR19	SR12	SR9			2.04	0.3	0.69	$	10.3
FR20	SR11	SR9			2.04	0.3	1.69	$	10.3
FR21	SR11	SR9			3.34	0.3	1.68	$	3.4
FR22	SR14	SR13			2.79	0.9	1.30	$	3.4
FR23	SR14	SR13			2.79	0.9	1.30	$	3.4
FR24	SR13	SR2			2.79	0.9	1.30	$	3.4
FR25	SR13	SR2			2.79	0.9	1.30	$	3.4
FR26	SR15	SR7			2.04	0.3	1.60	$	10.3
FR27	SR15	SR7			3.34	0.3	1.61	$	3.4
FS1	SS1	SS2			29.69	0.3	1.58	0.49	*
FS2	SS2	SS3			29.69	0.3	0.49	1.72	*
FS3	SS3	SS4			16.19	0.3	1.76	0.48	*
FS4	SS2	SS5			26.00	27.7	0.56	0.62	*
FS5	SS4	SS5			16.19	0.3	0.48	1.61	*
FS6	SR5	ATM.			9.29	0.9	0.52	$	3.4
FS7	SR5	ATM.			9.29	0.9	0.52	$	3.4
FS8	SR5	ATM.			9.29	0.9	0.52	$	3.4

Notes:

$ Indicates one-directional blow-out panel. Reverse loss coefficient not applicable.

* Indicates flowpath without blowout panel.

Table 6.2-4a Flow Loss Factor

BREAK NODE	PIPE ID (m)	PIPE LENGTH (m)	PIPE FRICTION FACTOR	PIPE LOSS COEFFICIENT	MECHANICAL LOSS COEFFICIENT	OVERALL* LOSS COEFFICIENT
SS1				LOSSES NOT CONSIDERED		
SS2			TYP	LOSSES NOT CONSIDERED	TYP	
SS3				NO BREAK POSTULATED		
SS4				NO BREAK POSTULATED		
SS5				NO BREAK POSTULATED		
SA1				LOSSES NOT CONSIDERED		
SA2				NO HIGH ENERGY LINES PRESENT		
SA3				LOSSES NOT CONSIDERED		
SA4				NO HIGH ENERGY LINES PRESENT		
SA5				NO HIGH ENERGY LINES PRESENT		
SA6				NO HIGH ENERGY LINES PRESENT		
SA7				NO HIGH ENERGY LINES PRESENT		
SR1				NO BREAK POSTULATED		
SR5	0.1909	66	0.015	5.2	1.7	6.9
SR3				NO HIGH ENERGY LINES PRESENT		
SR4	0.1909	89	0.015	7.0	3.4	10.4
SR5	0.1909	56	0.015	4.4	1.7	6.9
SR2				NO HIGH ENERGY LINES PRESENT		
SR12				NO HIGH ENERGY LINES PRESENT	57.1	64.7
SR8	0.1909	93	0.015	7.3	3.7	11.0

Table 6.2-4a Flow Loss Factor (Continued)

BREAK NODE	PIPE ID (m)	PIPE LENGTH (m)	PIPE FRICTION FACTOR	PIPE LOSS COEFFICIENT	MECHANICAL LOSS COEFFICIENT	OVERALL* LOSS COEFFICIENT
SR9	←			NO HIGH ENERGY LINES PRESENT		→
SR10				NO HIGH ENERGY LINES PRESENT		
SR11	0.1909	171		13.4	92.6	100.0
SR12	0.1909	98	0.015	7.7	57.1	64.7
SR13	0.1909	210	0.015	16.5	203.3	100.0
SR14	0.1909	210	0.015	16.5	203.3	100.0
SR15	0.1909	210	0.015	16.5	203.3	100.0

* Overall Loss Coefficient is limited to 100

Table 6.2-4b Mass and Energy Release Rate

Time (s)	Mass Flowrate (kg/s)	Enthalpy (Joule/gram)	Enthalpy Release Rate (kJ/s)
Break in subcompartment SA1 RCIC Pump & Turbine Room and in subcompartment SA3 ECCS Division A B1F,B2F, & B3F Pipespace (Figure 6.2-37a)			
0.00	189.9	2754.57	5.23E+05
11.00	189.9	2754.57	5.23E+05
17.00	170.8	2754.57	4.70E+05
23.00	140.4	2754.57	3.87E+05
41.00	0.0	2754.57	0.00E+00
1.00E+08	0.0	2754.57	0.00E+00
Break in subcompartment SR5 CUW Pipe Return (Figure 6.2-37c)			
0.00	782.4	1224.67	9.58E+05
3.12	782.4	1224.67	9.58E+05
3.12	655.8	1224.67	8.03E+05
9.85	655.8	1224.67	8.03E+05
9.85	376.3	1002.25	3.77E+05
59.65	376.3	1002.25	3.77E+05
59.65	376.3	923.62	3.48E+05
64.05	376.3	923.62	3.48E+05
70.56	232.1	923.62	2.14E+05
70.56	120.3	1224.67	1.47E+05
76.00	0.0	1224.67	0.00E+00
1.00E+08	0.0	0.00	0.00E+00
Break in subcompartment SR4 Regenerative Heat Exchanger Valve Room & Pipespace (Figure 6.2-37c)			
0.00	782.4	1224.67	9.58E+05
4.92	782.4	1224.67	9.58E+05
4.92	621.3	1224.67	7.61E+05
8.05	621.3	1224.67	7.61E+05
8.05	341.9	979.92	3.35E+05
57.85	341.9	979.92	3.35E+05

Table 6.2-4b Mass and Energy Release Rate (Continued)

Time (s)	Mass Flowrate (kg/s)	Enthalpy (Joule/gram)	Enthalpy Release Rate (kJ/s)
57.85	341.9	893.14	3.05E+05
64.05	341.9	893.14	3.05E+05
68.76	251.1	893.14	2.24E+05
68.76	139.4	1224.67	1.71E+05
76.00	0.0	1224.67	0.00E+00
1.00E+08	0.0	0.00	0.00E+00
Break in subcompartment SR2 CUW Pipespace (Figure 6.2-37c)			
0.00	782.4	1224.67	9.58E+05
3.66	782.4	1224.67	9.58E+05
3.66	649.5	1224.67	7.95E+05
9.31	649.5	1224.67	7.95E+05
9.31	370.0	998.53	3.70E+05
59.12	370.0	998.53	3.70E+05
59.12	370.0	918.50	3.40E+05
64.05	370.0	918.50	3.40E+05
70.02	240.9	918.50	2.21E+05
70.02	129.1	1224.67	1.58E+05
76.00	0.0	1224.67	0.00E+00
1.00E+08	0.0	0.00	0.00E+00
Break in subcompartment SR12 Non-Regenerative Heat Exchanger Valve Room & CUW Pump pipe space (Figure 6.2-37e)			
0.00	503.0	1058.32	5.32E+05
12.97	503.0	1058.32	5.32E+05
12.97	204.4	815.20	1.67E+05
60.29	204.4	815.20	1.67E+05
60.29	92.6	1224.67	1.13E+05
64.05	92.6	1224.67	1.13E+05
76.00	0.0	1224.67	0.00E+00
1.00E+08	0.0	0.00	0.00E+00

Table 6.2-4b Mass and Energy Release Rate (Continued)

Time (s)	Mass Flowrate (kg/s)	Enthalpy (Joule/gram)	Enthalpy Release Rate (kJ/s)
Break in subcompartment SR8 CUW Regenerative Heat Exchanger & Non-Regenerative Heat Exchanger (Figure 6.2-37c)			
0.00	782.4	1224.67	9.58E+05
5.14	782.4	1224.67	9.58E+05
5.14	617.1	1224.67	7.56E+05
7.83	617.1	1224.67	7.56E+05
7.83	337.7	976.90	3.30E+05
57.63	337.7	976.90	3.30E+05
57.63	337.7	888.95	3.00E+05
64.05	337.7	888.95	3.00E+05
68.54	252.7	1224.67	3.10E+05
68.54	141.0	1224.67	1.73E+05
76.00	0.0	1224.67	0.00E+00
1.00E+08	0.0	0.00	0.00E+00
Break in subcompartment SR11 CUW Pump A & B Rooms (Figure 6.2-37d)			
0.00	223.5	1224.67	2.74E+05
17.02	223.5	1224.67	2.74E+05
17.02	111.8	1224.67	1.37E+05
34.69	111.8	1224.67	1.37E+05
34.69	111.8	1224.67	1.37E+05
36.77	111.8	1224.67	1.37E+05
36.77	391.2	1224.67	4.79E+05
49.73	391.2	1224.67	4.79E+05
49.73	68.5	1224.67	8.39E+04
64.05	68.5	1224.67	8.39E+04
76.00	0.0	1224.67	0.00E+00
1.00E+08	0.0	1224.67	0.00E+00

Table 6.2-4b Mass and Energy Release Rate (Continued)

Time (s)	Mass Flowrate (kg/s)	Enthalpy (Joule/gram)	Enthalpy Release Rate (kJ/s)
Break in subcompartment SR13 and SR14 CUW Filter/Demin B Room CUW Filter/Demin A or B (Figure 6.2-37f & g)			
0.00	194.8	590.00	1.15E+05
9.90	194.8	590.00	1.15E+05
9.90	503.0	1167.67	5.87E+05
30.55	503.0	1167.67	5.87E+05
30.55	180.2	1065.77	1.92E+05
64.05	180.2	1065.77	1.92E+05
64.05	180.2	1065.77	1.92E+05
76.00	111.8	968.52	1.08E+05
136.42	111.8	968.52	1.08E+05
136.42	0.0	0.00	0.00E+00
1.00E+08	0.0	0.00	0.00E+00
Break in subcompartment SR15 CUW Filter/Demin A & B Valve Rooms (Figure 6.2-37h)			
0.00	503.0	999.46	5.03E+05
9.90	503.0	999.46	5.03E+05
9.90	503.0	1167.67	5.87E+05
30.55	503.0	1167.67	5.87E+05
30.55	180.2	1065.77	1.92E+05
64.05	180.2	1065.77	1.92E+05
64.05	180.2	1065.77	1.92E+05
76.00	111.8	968.52	1.08E+05
136.42	111.8	968.52	1.08E+05
136.42	0.0	0.00	0.00E+00
1.00E+08	0.0	0.00	0.00E+00
Break in subcompartment SS1 & SS2 (steam tunnel) Main Steamline Break (Figure 6.2-37b)			
0.0000	5142.9	2770.86	1.43E+07
0.0059	5142.9	2770.86	1.43E+07
0.0610	3596.4	2770.86	9.97E+06

Table 6.2-4b Mass and Energy Release Rate (Continued)

Time (s)	Mass Flowrate (kg/s)	Enthalpy (Joule/gram)	Enthalpy Release Rate (kJ/s)
0.0996	3581.9	2770.86	9.92E+06
0.3965	3537.9	2770.86	9.80E+06
0.5215	3503.9	2773.18	9.72E+06
0.9941	3111.1	2773.18	8.63E+06
1.0020	6354.2	1419.16	9.02E+06
1.1270	6172.8	1419.16	8.76E+06
1.2129	6047.2	1419.16	8.58E+06
1.6504	5417.2	1419.16	7.69E+06
1.9980	4921.1	1419.16	6.98E+06
2.4980	4215.2	1419.16	5.98E+06
2.8105	3779.8	1419.16	5.36E+06
3.4355	2915.6	1419.16	4.14E+06
3.9355	2222.7	1419.16	3.15E+06
4.4355	1522.3	1423.82	2.17E+06
5.1855	454.6	1428.47	6.49E+05
5.4980	2.9	1430.79	4.09E+03
5.5000	0.0	1430.79	0.00E+00
Break in subcompartment SS1 and SS2 (steam tunnel) Feedwater Line Break (Figure 6.2-37b)			
0.00	2462.1	931.1	2.29E+06
3.75	2462.1	931.1	2.29E+06
3.75	3474.1	931.1	3.23E+06
18.40	3474.1	931.1	3.23E+06
42.10	3474.1	653.6	2.27E+06
67.90	3474.1	653.6	2.27E+06
100.60	3474.1	146.5	5.09E+05
120.00	3474.1	146.5	5.09E+05
120.00	0.0	146.5	0.00E+00

Table 6.2-5 Reactor Coolant Pressure Boundary (RCPB) Influent Lines Penetrating Drywell

Drywell	Inside Drywell	Outside Drywell
Influent Line		
1. Feedwater	CV	AOCV & MOV
2. RHR Injection	TCV	MOV
3. HPCF	TCV	MOV
4. Standby liquid control	CV	MOV
5. Reactor water cleanup, reactor vessel head spray	CV	MOV
6. Recirculating internal pump seal purge water supply	N/A	EFCV

Note: CV—Check Valve
AOCV—Air-operated check valve
MOV—Motor-operated valve
TCV—Testable check valve
EFCV—Excess flow check valve

Table 6.2-6 Reactor Coolant Pressure Boundary (RCPB) Effluent Lines Penetrating Drywell

Inside Drywell	Outside Drywell	Drywell
Effluent Line		
1. Main steam	AOV	GOV
2. Main steam drain	MOV	MOV
3. RCIC steam supply	MOV	MOV
4. RHR shutdown cooling supply	MOV	MOV
5. CUW pump suction	MOV	MOV

Note: MOV—Motor-operated valve
GOV— Gas operated valve. N_2 to open, and N_2 and/or spring to close.
AOV—Air operated valve. Air to open, and Air and/or spring to close.

Table 6.2-7 Containment Isolation Valve Information*

MPL	System	Page
B21	Nuclear Boiler	Page 6.2-132 thru Page 6.2-134
B31	Reactor Recirculation	Page 6.2-117
C41	Standby Liquid Control	Page 6.2-118
D23	Containment Atmospheric Monitoring	Page 6.2-119 thru Page 6.2-120
E11	Residual Heat Removal	Page 6.2-121 thru Page 6.2-128
E22	High Pressure Core Flooder	Page 6.2-129 thru Page 6.2-131
E31	Leak Detection & Isolation	Page 6.2-156
E51	Reactor Core Isolation Cooling	Page 6.2-136 thru Page 6.2-146
G31	Reactor Water Cleanup	Page 6.2-146 thru Page 6.2-148
G51	Suppression Pool Cleanup	Page 6.2-149
K17	Radwaste	Page 6.2-157
P11	Makeup Water (Purified)	Page 6.2-155
P21	Reactor Building Cooling Water	Page 6.2-151
P24	HVAC Normal Cooling Water	Page 6.2-151
P51	Service Air	Page 6.2-152
P52	Instrument Air	Page 6.2-153
P54	High Pressure Nitrogen Gas Supply	Page 6.2-154
T31	Atmospheric Control	Page 6.2-140 thru Page 6.2-145
See notes at the end of this table.		

* This table responds to NRC Questions 430.35, 430.50b. 430.50c, 430.50d and 430.50f regarding containment isolation provisions for fluid system lines and for fluid instrument lines penetrating containment within the scope of the ABWR Standard Plant. Locked closed isolation valves are identified on the P&IDs. The containment information is presented separately for each system for the MPL numbers given below.

Table 6.2-7 Containment Isolation Valve Information Reactor Recirculation System RIP Purge

Valve No.	B31-F008A-H/J/K
Tier 2 Figure	5.4-4
Applicable Basis	RG 1.11
Fluid	Demin. Reactor Water
Line Size	20A
ESF	No (d,m)
Leakage Class	(a)
Location	0
Type C Leak Test	No (d, m)
Valve Type	Excess Flow Check
Operator	N/A
Primary Actuation	Self
Secondary Actuation	N/A
Normal Position	Open
Shutdown Position	Open
Post-accident Position	Open
Power Fail Position	Open
Containment Isolation Signal (c)	N/A
Closure Time (s)	Instantaneous
Power Source (Div)	N/A
See notes at the end of this table.	

Table 6.2-7 Containment Isolation Valve Information
Standby Liquid Control System

Valve No.	C41-F008	C41-F006A	C41-F006B
Tier 2 Figure	9.3-1	9.3-1	9.3-1
Applicable Basis	GDC 55	GDC 55	GDC 55
Fluid	Boron/Water	Boron/Water	Boron/Water
Line Size	40A	40A	40A
ESF	No	No	No
Leakage Class	(a)	(a)	(a)
Location	I	O	O
Type C Leak Test	Yes	Yes	Yes
Valve Type	Swing Check	Globe	Globe
Operator	N/A	Motor	Motor
Primary Actuation	Self	Electrical	Electrical
Secondary Actuation	N/A	Manual	Manual
Normal Position	Close	Close	Close
Shutdown Position	Close	Close	Close
Post-accident Position	Close	Close	Close
Power Fail Position	As is	As is	As is
Containment Isolation Signal (c)	N/A	N/A	N/A
Closure Time (s)	Instantaneous	24	24
Power Source (Div)	N/A	I	II
See notes at the end of this table.			

Table 6.2-7 Containment Isolation Valve Information
Containment Atmospheric Monitoring

Valve No.	D23-F001A/B	D23-F004A/B	D23-F005A/B	D23-F006A/B	D23-F007A/B	D23-F008A/B
Tier 2 Figure	7.6-7 (Sheet 2)	7.6-7 (Sheet 2)	7.6-7 (Sheet 2)	7.6-7 (Sheet 2)	7.6-7 (Sheet 2)	7.6-7(Sheet 2)
Applicable Basis	GDC 56 RG 1.11	GDC 56	GDC 56	GDC 56	GDC 56	GDC 56
Fluid	DW Atmosphere	DW Atmosphere	DW Atmosphere	WW Atmosphere	WW Atmosphere	WW Atmosphere
Line Size	20A	20A	20A	20A	20A	20A
ESF	No	No	No	No	No	No
Leakage Class	(a)	(a)	(a)	(a)	(a)	(a)
Location	O	O	O	O	O	O
Type C Leak Test	No (m)	No (f)	No (f)	No (f)	No (f)	No (f)
Valve Type	Gate	Globe	Globe	Globe	Globe	Globe
Operator	Solenoid	Solenoid	Solenoid	Solenoid	Solenoid	Solenoid
Primary Actuation	Electrical	Electrical	Electrical	Electrical.	Electrical	Electrical
Secondary Actuation	N/A	N/A	N/A	N/A	N/A	N/A
Normal Position	Open	Close/Open	Close/Open	Close/Open	Close/Open	Close/Open
Shutdown Position	Close	Close	Close	Close	Close	Close
Post-accident Position	Open	Open	Open	Open	Open	Open
Power Fail Position	Open	Close	Close	Close	Close	Close
Containment Isolation Signal[c]	RM	RM	RM	RM	RM	RM
Closure Time (s)	N/A	N/A	N/A	N/A	N/A	NIA
Power Source (Div)	I/II	I/II	I/II	I/II	I/II	I/II
See notes at the end of this table.						

Table 6.2-7 Containment Isolation Valve Information
Containment Atmospheric Monitoring

Valve No.	D23-F009A/B	D23-F0010A/B	D23-F0011A/B	D23-F0012A/B	D23-F0013A/B	D23-F0014A/B
Tier 2 Figure	7.6-7 (Sheet 2)	7.6-7 (Sheet 2)	7.6-7 (Sheet 2)	7.6-7 (Sheet 2)	7.6-7 (Sheet 2)	7.6-7 (Sheet 2)
Applicable Basis	GDC 56	GDC 56	GDC 56	GDC 56	GDC 56	GDC 56
Fluid	DW Atmosphere	DW Atmosphere	DW Atmosphere	WW Atmosphere	WW Atmosphere	WW Atmosphere
Line Size	20A	20A	20A	20A	20A	20A
ESF	No	No	No	No	No	No
Leakage Class	(a)	(a)	(a)	(a)	(a)	(a)
Location	O	O	O	O	O	O
Type C Leak Test	No (f)	No (f)	No (f)	No (f)	No (f)	No (f)
Valve Type	Globe	Globe	Globe	Globe	Globe	Globe
Operator	N/A	N/A	N/A	N/A	N/A	N/A
Primary Actuation	Manual	Manual	Manual	Manual	Manual	Manual
Secondary Actuation	N/A	N/A	N/A	N/A	N/A	N/A
Normal Position	Open	Open	Open	Open	Open	Open
Shutdown Position	Open	Open	Open	Open	Open	Open
Post-accident Position	Open	Open	Open	Open	Open	Open
Power Fail Position	N/A	N/A	N/A	N/A	N/A	N/A
Containment Isolation Signal[c]	N/A	N/A	N/A	N/A	N/A	N/A
Closure Time (s)	N/A	N/A	N/A	N/A	N/A	N/A
Power Source (Div)	N/A	N/A	N/A	N/A	N/A	N/A

See notes at the end of this table.

Table 6.2-7 Containment Isolation Valve Information Residual Heat Removal System Wetwell Spray

Valve No.	E11-F019B	E11-F019C
Tier 2 Figure	5.4-10 (Sheet 5)	5.4-10 (Sheet 7)
Applicable Basis	GDC 56	GDC 56
Fluid	Water	Water
Line Size	100A	100A
ESF	Yes	Yes
Leakage Class	(a)	(a)
Location	O	O
Type C Leak Test	No (g)	No (g)
Valve Type	Gate	Gate
Operator	Motor	Motor
Primary Actuation	Electrical	Electrical
Secondary Actuation	Manual	Manual
Normal Position	Close	Close
Shutdown Position	Close	Close
Post-accident Position	Close/Open	Close/Open
Power Fail Position	As is	As is
Containment Isolation Signal[c]	RM	RM
Closure Time (s)	20	20
Power Source (Div)	II	III
See notes at the end of this table.		

Table 6.2-7 Containment Isolation Valve Information Residual Heat Removal System Drywell Spray

Valve No.	E11-F017B	E11-F018B	E11-F017C	E11-F018C
Tier 2 Figure	5.4-10 (Sheet 5)	5.4-10 (Sheet 5)	5.4-10 (Sheet 7)	5.4-10 (Sheet 7)
Applicable Basis	GDC 56	GDC 56	GDC 56	GDC 56
Fluid	Water	Water	Water	Water
Line Size	250A	250A	250A	250A
ESF	Yes	Yes	Yes	Yes
Leakage Class	(a)	(a)	(a)	(a)
Location	O	O	O	O
Type C Leak Test	No (g)	No (g)	No (g)	No (g)
Valve Type	Globe	Gate	Globe	Gate
Operator	Motor	Motor	Motor	Motor
Primary Actuation	Electrical	Electrical	Electrical	Electrical
Secondary Actuation	Manual	Manual	Manual	Manual
Normal Position	Close	Close	Close	Close
Shutdown Position	Close	Close	Close	Close
Post-accident Position	Close/Open	Close/Open	Close/Open	Close/Open
Power Fail Position	As is	As is	As is	As is
Containment Isolation Signal(c)	RM	RM	RM	RM
Closure Time (s)	50	50	50	50
Power Source (Div)	II	II	III	III
See notes at the end of this table.				

Table 6.2-7 Containment Isolation Valve Information
Residual Heat Removal System Minimum Flow Line

Valve No.	E11-F021A	E11-F021B	E11-F021C
Tier 2 Figure	5.4-10 (Sheet 3)	5.4-10 (Sheet 4)	5.4-10 (Sheet 8)
Applicable Basis	GDC 56	GDC 56	GDC 56
Fluid	Water	Water	Water
Line Size	100A	100A	100A
ESF	Yes	Yes	Yes
Leakage Class	(a)	(a)	(a)
Location	O	O	O
Type C Leak Test	No (h)	No (h)	No(h)
Valve Type	Gate	Gate	Gate
Operator	Motor	Motor	Motor
Primary Actuation	Electrical	Electrical	Electrical
Secondary Actuation	Manual	Manual	Manual
Normal Position	Open	Open	Open
Shutdown Position	Close	Close	Close
Post-accident Position	Close	Close	Close
Power Fail Position	As is	As is	As is
Containment Isolation Signal (c)	RM	RM	RM
Closure Time (s)	20	20	20
Power Source (Div)	I	II	III
See notes at the end of this table.			

Table 6.2-7 Containment Isolation Valve Information Residual Heat Removal System S/P Cooling

Valve No.	E11-F008A	E11-F031A	E11-F008B	E11-F031B	E11-F008C	E11-F031C
Tier 2 Figure	5.4-10 (Sheet 3)	5.4-10 (Sheet 3)	5.4-10 (Sheet 4)	5.4-10 (Sheet 4)	5.4-10 (Sheet 6)	5.4-10 (Sheet 6)
Applicable Basis	GDC 56	GDC 56	GDC 56	GDC 56	GDC 56	GDC 56
Fluid	Water	Water	Water	Water	Water	Water
Line Size	250A	100A	250A	100A	250A	100A
ESF	Yes	Yes	Yes	Yes	Yes	Yes
Leakage Class	(a)	(a)	(a)	(a)	(a)	(a)
Location	O	O	O	O	O	O
Type C Leak Test	No (j)	No (j)	No (j)	No (j)	No (j)	No (j)
Valve Type	Globe	Globe	Globe	Globe	Globe	Globe
Operator	Motor	Motor	Motor	Motor	Motor	Motor
Primary Actuation	Electrical	Electrical	Electrical	Electrical	Electrical	Electrical
Secondary Actuation	Manual	Manual	Manual	Manual	Manual	Manual
Normal Position	Close	Close	Close	Close	Close	Close
Shutdown Position	Close	Close	Close	Close	Close	Close
Post-accident Position	Close	Close	Close	Close	Close	Close
Power Fail Position	As is	As is	As is	As is	As is	As is
Containment Isolation Signal (c)	RM, CX, K	RM	RM, CX, K	RM	RM, CX, K	RM
Closure Time (s)	50	20	50	20	50	20
Power Source (Div)	I	I	II	II	III	III
See notes at the end of this table.						

Table 6.2-7 Containment Isolation Valve Information Residual Heat Removal System S/P Suction (LPFL)

Valve No.	E11-F001A	E11-F001B	E11-F001C
Tier 2 Figure	5.4-10 (Sheet 3)	5.4-10 (Sheet 4)	5.4-10 (Sheet 6)
Applicable Basis	GDC 56	GDC 56	GDC 56
Fluid	Water	Water	Water
Line Size	450A	450A	450A
ESF	Yes	Yes	Yes
Leakage Class	(a)	(a)	(a)
Location	O	O	O
Type C Leak Test	No (i)	No (i)	No (i)
Valve Type	Gate	Gate	Gate
Operator	Motor	Motor	Motor
Primary Actuation	Electrical	Electrical	Electrical
Secondary Actuation	Manual	Manual	Manual
Normal Position	Open	Open	Open
Shutdown Position	Close	Close	Close
Post-accident Position	Open	Open	Open
Power Fail Position	As is	As is	As is
Containment Isolation Signal [c]	RM	RM	RM
Closure Time (s)	90	90	90
Power Source (Div)	I	II	III
See notes at the end of this table.			

Table 6.2-7 Containment Isolation Valve Information
Residual Heat Removal System Inboard Shutdown Cooling

Valve No.	E11-F010A	E11-F010B	E11-F010C
Tier 2 Figure	5.4-10 (Sheet 2)	5.4-10 (Sheet 2)	5.4-10 (Sheet 2)
Applicable Basis	GDC 55	GDC 55	GDC 55
Fluid	Water	Water	Water
Line Size	350A	350A	350A
ESF	Yes	Yes	Yes
Leakage Class	(a)	(a)	(a)
Location	I	I	I
Type C Leak Test	No (n)	No (n)	No (n)
Valve Type	Gate	Gate	Gate
Operator	Motor	Motor	Motor
Primary Actuation	Electrical	Electrical	Electrical
Secondary Actuation	Manual	Manual	Manual
Normal Position	Close	Close	Close
Shutdown Position	Open/Close	Open/Close	Open/Close
Post-accident Position	Close	Close	Close
Power Fail Position	As is	As is	As is
Containment Isolation Signal (c)	RM, A, U, Z	RM, A, U, Z	RM, A, U, Z
Closure Time (s)	70	70	70
Power Source (Div)	I	II	III

See notes at the end of this table.

Table 6.2-7 Containment Isolation Valve Information
Residual Heat Removal System
Outboard Shutdown Cooling

Valve No.	E11-F011A	E11-F011B	E11-F011C
Tier 2 Figure	5.4-10 (Sheet 2)	5.4-10 (Sheet 2)	5.4-10 (Sheet 2)
Applicable Basis	GDC 55	GDC 55	GDC 55
Fluid	Water	Water	Water
Line Size	350A	350A	350A
ESF	Yes	Yes	Yes
Leakage Class	(a)	(a)	(a)
Location	O	O	O
Type C Leak Test	No (n)	No (n)	No (n)
Valve Type	Gate	Gate	Gate
Operator	Motor	Motor	Motor
Primary Actuation	Electrical	Electrical	Electrical
Secondary Actuation	Manual	Manual	Manual
Normal Position	Close	Close	Close
Shutdown Position	Open/Close	Open/Close	Open/Close
Post-accident Position	Close	Close	Close
Power Fail Position	As is	As is	As is
Containment Isolation Signal[c]	RM, A, U, Z	RM, A, U, Z	RM, A, U, Z
Closure Time (s)	70	70	70
Power Source (Div)	II	III	I
See notes at the end of this table.			

Table 6.2-7 Containment Isolation Valve Information
Residual Heat Removal System
Injection and Testable Check

Valve No.	E11-F005B	E11-F006B	E11-F005C	E11-F006C
Tier 2 Figure	5.4-10 (Sheet 5)	5.4-10 (Sheet 5)	5.4-10 (Sheet 7)	5.4-10 (Sheet 7)
Applicable Basis	GDC 55	GDC 55	GDC 55	GDC 55
Fluid	Water	Water	Water	Water
Line Size	250A	250A	250A	250A
ESF	Yes	Yes	Yes	Yes
Leakage Class	(a)	(a)	(a)	(a)
Location	O	I	O	I
Type C Leak Test	No (k)	No (k)	No (k)	No (k)
Valve Type	Gate	Check	Gate	Check
Operator	Motor	N/A	Motor	N/A
Primary Actuation	Electrical	Self	Electrical	Self
Secondary Actuation	Manual	N/A	Manual	N/A
Normal Position	Close	Close	Close	Close
Shutdown Position	Close	Close	Close	Close
Post-accident Position	Close/Open	Close/Open	Close/Open	Close/Open
Power Fail Position	As is	N/A	As is	N/A
Containment Isolation Signal(c)	RM	N/A	RM	N/A
Closure Time (s)	10	Instantaneous	10	Instantaneous
Power Source (Div)	II	N/A	III	N/A
See notes at the end of this table.				

Table 6.2-7 Containment Isolation Valve Information
High Pressure Core Flooder System
S\P Suction

Valve No.	E22-F006B	E22-F006C
Tier 2 Figure	6.3-7 (Sheet 2)	6.3-7 (Sheet 2)
Applicable Basis	GDC 56	GDC 56
Fluid	Water	Water
Line Size	400A	400A
ESF	Yes	Yes
Leakage Class	(a)	(a)
Location	O	O
Type C Leak Test	No(i)	No(i)
Valve Type	Gate	Gate
Operator	Motor	Motor
Primary Actuation	Electrical	Electrical
Secondary Actuation	Manual	Manual
Normal Position	Close	Close
Shutdown Position	Close	Close
Post-Accident Position	Close/Open	Close/Open
Power Fail Position	As is	As is
Containment Isolation Signal [c]	RM	RM
Closure Time (s)	80	80
Power Source (Div)	II	III
See notes at the end of this table.		

Table 6.2-7 Containment Isolation Valve Information High Pressure Core Flooder System Test and Minimum Flow

Valve No.	E22-F009B	E22-F010B	E22-F009C	E22-F010C
Tier 2 Figure	6.3-7 (Sheet 2)	6.3-7 (Sheet 2)	6.3-7 (Sheet	6.3-7 (Sheet 2)
Applicable Basis	GDC 56	GDC 56	GDC 56	GDC 56
Fluid	Water	Water	Water	Water
Line Size	200A	80A	200A	80A
ESF	Yes	Yes	Yes	Yes
Leakage Class	(a)	(a)	(a)	(a)
Location	O	O	O	O
Type C Leak Test	No(h)	No(h)	No(h)	No(h)
Valve Type	Globe	Gate	Globe	Gate
Operator	Motor	Motor	Motor	Motor
Primary Actuation	Electrical	Electrical	Electrical	Electrical
Secondary Actuation	Manual	Manual	Manual	Manual
Normal Position	Close	Close	Close	Close
Shutdown Position	Close	Close	Close	Close
Post-Accident Position	Close	Close	Close	Close
Power Fail Position	As is	As is	As is	As is
Containment Isolation Signal (c)	RM	RM	RM	RM
Closure Time (s)	20	20	20	20
Power Source (Div)	II	II	III	III
See notes at the end of this table.				

Table 6.2-7 Containment Isolation Valve Information High Pressure Core Flooder System Injection

Valve No.	E22-F003B	E22-F004B	E22-F003C	E22-F004C
Tier 2 Figure	6.3-7 (Sheet 1)	6.3-7 (Sheet 1)	6.3-7 (Sheet 1)	6.3-7 (Sheet 1)
Applicable Basis	GDC 55	GDC 55	GDC 55	GDC 55
Fluid	Water	Water	Water	Water
Line Size	200A	200A	200A	200A
ESF	Yes	Yes	Yes	yes
Leakage Class	(a)	(a)	(a)	(a)
Location	O	I	O	I
Type C Leak Test	No(k)	No(k)	No(k)	No(k)
Valve Type	Gate	Check	Gate	Check
Operator	Motor	Self	Motor	Self
Primary Actuation	Electrical	N/A	Electrical	N/A
Secondary Actuation	Manual	N/A	Manual	N/A
Normal Position	Close	Close	Close	Close
Shutdown Position	Close	Close	Close	Close
Post-Accident Position	Close/Open	Close/Open	Close/Open	Close/Open
Power Fail Position	As is	N/A	As is	N/A
Containment Isolation Signal (c)	N/A	N/A	N/A	N/A
Closure Time (s)	36	Instantaneous	36	Instantaneous
Power Source (Div)	II	N/A	III	N/A
See notes at the end of this table.				

Table 6.2-7 Containment Isolation Valve Information
Nuclear Boiler System
Main Steam Lines A, B, C and D

Valve No.	B21-F008A/B C/D	B21-F009A/B C/D
Tier 2 Figure	5.1-3 (Sheet 3)	5.1-3 (Sheet 3)
Applicable Basis	GDC 55	GDC 55
Fluid	Steam	Steam
Line Size	700A	700A
ESF	No	No
Leakage Class	(b)	(b)
Location	I	O
Type C Leak Test	Yes(e)	Yes(e)
Valve Type	Globe	Globe
Operator	Pneumatic	Pneumatic
Primary Actuation	N_2 to open N_2 and/or Spring to close	Air to open Air and/or Spring to close
Secondary Actuation	N/A	N/A
Normal Position	Open	Open
Shutdown Position	Close	Close
Post-Accident Position	Close	Close
Power Fail Position	Close	Close
Containment Isolation Signal [c]	C, E, F, H, N, BB, RM	C, E, F, H, N, BB, RM
Closure Time (s)	3-4.5	3-4.5
Power Source (Div)	I/II	I/II

See notes at the end of this table.

Table 6.2-7 Containment Isolation Valve Information
Nuclear Boiler System
Main Steam Line Drains

Valve No.	B21-F011	B21-F012
Tier 2 Figure	5.1-3 (Sheet 3)	5.1-3 (Sheet 3)
Applicable Basis	GDC 55	GDC 55
Fluid	Steam/Water	Steam/Water
Line Size	80A	80A
ESF	No	No
Leakage Class	(b)	(b)
Location	I	O
Type C Leak Test	Yes(e)	Yes(e)
Valve Type	Gate	Gate
Operator	Motor	Motor
Primary Actuation	Electrical	Electrical
Secondary Actuation	Manual	Manual
Normal Position	Open	Open
Shutdown Position	Close	Close
Post-Accident Position	Close	Close
Power Fail Position	As is	As is
Containment Isolation Signal (c)	C, E, F, H, N, BB, RM	C, E, F, H, N, BB, RM
Closure Time (s)	15	15
Power Source (Div)	II	I
See notes at the end of this table.		

Table 6.2-7 Containment Isolation Valve Information Nuclear Boiler System Feedwater Line A and B

Valve No.	B21-F004A/B	B21-F003A/B
Tier 2 Figure	5.1-3 (Sheet 4)	5.1-3 (Sheet 4)
Applicable Basis	GDC 55	GDC 55
Fluid	Water	Water
Line Size	550A	550A
ESF	Yes	Yes
Leakage Class	(b)	(b)
Location	I	O
Type C Leak Test	Yes	Yes
Valve Type	Check	Spring Check
Operator	Self	Pneumatic
Primary Actuation	N/A	Air to open
Secondary Actuation	N/A	N/A
Normal Position	Open	Open
Shutdown Position	Open/Close	Open/Close
Post-Accident Position	Open/Close	Open/Close
Power Fail Position	N/A	N/A
Containment Isolation Signal (c)	N/A	N/A
Closure Time (s)	Instantaneous	Instantaneous
Power Source (Div)	N/A	N/A
See notes at the end of this table.		

Table 6.2-7 Containment Isolation Valve Information
Nuclear Boiler System
Instrument Lines

Valve No.	Various
Tier 2 Figure	5.1-3 (Sheets 2,3,6,7,8)
Applicable Basis	RG 1.11
Fluid	Air/Water
Line Size	20A
ESF	N/A
Leakage Class	(b)
Location	O
Type C Leak Test	No(m)
Valve Type	Excess Flow Check
Operator	Self
Primary Actuation	N/A
Secondary Actuation	N/A
Normal Position	Open
Shutdown Position	Open
Post-Accident Position	Open
Power Fail Position	Open
Containment Isolation Signal (c)	N/A
Closure Time (s)	Instantaneous
Power Source (Div)	N/A
See notes at the end of this table.	

Table 6.2-7 Containment Isolation Valve Information Reactor Core Isolation Cooling System Steam Supply

Valve No.	E51-F035	E51-F048	E51-F036
Tier 2 Figure	5.4-8 (Sheet 2)	5.4-8 (Sheet 2)	5.4-8 (Sheet 2)
Applicable Basis	GDC 55	GDC 55	GDC 55
Fluid	Steam	Steam	Steam
Line Size	150A	25A	150A
ESF	Yes	Yes	Yes
Leakage Class	(a)	(a)	(a)
Location	I	I	O
Type C Leak Test	Yes(e)	Yes(e)	Yes
Valve Type	Gate	Globe	Gate
Operator	Motor	Motor	Motor
Primary Actuation	Electrical	Electrical	Electrical
Secondary Actuation	Remote Manual	Remote Manual	Remote Manual
Normal Position	Open	Close	Open
Shutdown Position	Close	Close	Close
Post-Accident Position	Open/Close	Open/Close	Open/Close
Power Fail Position	As is	As is	As is
Containment Isolation Signal (c)	S, T, RM,Z	S, T, RM,Z	S, T, RM,Z
Closure Time (s)	<30	<30	<30
Power Source (Div)	I	I	II

See notes at the end of this table.

Table 6.2-7 Containment Isolation Valve Information
Reactor Core Isolation Cooling System
Minimum Flow and Test Return

Valve No.	E51-F011	E51-F009
Tier 2 Figure	5.4-8 (Sheet 1)	5.4-8 (Sheet 1)
Applicable Basis	GDC 56	GDC 56
Fluid	Water	Water
Line Size	50A	100A
ESF	Yes	Yes
Leakage Class	(a)	(a)
Location	O	O
Type C Leak Test	No(h)	No(h)
Valve Type	Globe	Globe
Operator	Motor	Motor
Primary Actuation	Electrical	Electrical
Secondary Actuation	Remote Manual	Remote Manual
Normal Position	Close	Close
Shutdown Position	Close	Close
Post-Accident Position	Close	Close
Power Fail Position	As is	As is
Containment Isolation Signal [(c)]	RM	RM
Closure Time (s)	<5	<60
Power Source (Div)	I	I
See notes at the end of this table.		

Table 6.2-7 Containment Isolation Valve Information Reactor Core Isolation Cooling System S/P Suction

Valve No.	E51-F006
Tier 2 Figure	5.4-8 (Sheet 1)
Applicable Basis	GDC 56
Fluid	Water
Line Size	200A
ESF	Yes
Leakage Class	(a)
Location	O
Type C Leak Test	No(i)
Valve Type	Gate
Operator	Motor
Primary Actuation	Electrical
Secondary Actuation	Remote Manual
Normal Position	Close
Shutdown Position	Close
Post-Accident Position	Close/Open
Power Fail Position	As is
Containment Isolation Signal (c)	RM
Closure Time (s)	<30
Power Source (Div)	I
See notes at the end of this table.	

Table 6.2-7 Containment Isolation Valve Information Reactor Core Isolation Cooling System Turbine Exhaust

Valve No.	E51-F039	E51-F038
Tier 2 Figure	5.4-8 (Sheet 1)	5.4-8 (Sheet 1)
Applicable Basis	GDC 56	GDC 56
Fluid	Steam	Steam
Line Size	350A	350A
ESF	Yes	Yes
Leakage Class	(a)	(a)
Location	O	O
Type C Leak Test	Yes(e)	Yes
Valve Type	Gate	Check
Operator	Motor	Self
Primary Actuation	Electrical	N/A
Secondary Actuation	Manual	N/A
Normal Position	Locked Open	Close
Shutdown Position	Open	Close
Post-Accident Position	Close	Close
Power Fail Position	As is	N/A
Containment Isolation Signal [c]	RM	N/A
Closure Time (s)	<70	Instantaneous
Power Source (Div)	I	N/A
See notes at the end of this table.		

Table 6.2-7 Containment Isolation Valve Information Atmospheric Control System

Valve No.	T31-F001	T31-F002	T31-F003	T31-F004	T31-F005	T31-F006	T31-F007
Tier 2 Figure	6.2-39 (Sheet 1)	6.2-39 (Sheet 1)	6.2-39 (Sheet 1)	6.2-39 (Sheet 1)	6.2-39 (Sheet 1)	6.2-39 (Sheet 1	6.2-39 (Sheet 1)
Applicable Basis	GDC 56	GDC 56	GDC 56	GDC 56	GDC 56	GDC 56	GDC 56
Fluid	Air	Air or N_2	Air or N_2	Air or N_2	Air or N_2	Air or N_2	Air or N_2
Line Size	550A	550A	550A	550A	50A	550A	250A
ESF	Yes	Yes	Yes	Yes	Yes	Yes	Yes
Leakage Class	(b)	(b)	(b)	(b)	(b)	(b)	(b)
Location	O	O	O	O	O	O	O
Type C Leak Test	Yes	Yes(e)	Yes(e)	Yes(e)	Yes(e)	Yes(e)	Yes(e)
Valve Type	Butterfly	Butterfly	Butterfly	Butterfly	Globe	Butterfly	Butterfly
Operator	Pneumatic	Pneumatic	Pneumatic	Pneumatic	Pneumatic	Pneumatic	Pneumatic
Primary Actuation	Electric	Electric	Electric	Electric	Electric	Electric	Electric
Secondary Actuation	Manual	Manual	Manual	Manual	Manual	Manual	Manual
Normal Position	Close	Close	Close	Close	Close	Close	Open
Shutdown Position	Close	Close	Close	Close	Close	Close	Open
Post-Accident Position	Close	Close	Close	Close	Close	Close	Open
Power Fail Position	Close	Close	Close	Close	Close	Close	Open
Containment Isolation Signal [(c)]	A, K, XX, YY, RM	A, K, XX, YY, RM	A, K, XX, YY, RM	A, K, XX, YY, RM	A, K, XX, YY, RM	A, K, XX, YY, RM	RM
Closure Time (s)	<20	<20	<20	<20	<15	<20	<20
Power Source (Div)	I	II	II	II	II	II	II

See notes at the end of this table.

Table 6.2-7 Containment Isolation Valve Information
Atmospheric Control System

Valve No.	T31-F008	T31-F009	T31-F025	T31-F039	T31-F040	T31-F041
Tier 2 Figure	6.2-39 (Sheet 1)	6.2-39 (Sheet 1)	6.2-39 (Sheet 1)	6.2-39 (Sheet 1)	6.2-39 (Sheet 1)	6.2-39 (Sheet 1)
Applicable Basis	GDC 56	GDC 56	GDC 56	GDC 56	GDC 56	GDC 56
Fluid	Air or N_2	Air or N_2	N_2	N_2	N_2	N_2
Line Size	550A	550A	400A	50A	50A	50A
ESF	Yes	Yes	Yes	Yes	Yes	Yes
Leakage Class	(a)	(a)	(a)	(a)	(a)	(a)
Location	O	O	O	O	O	O
Type C Leak Test	Yes(b)	Yes(b)	Yes(b)	Yes(b)	Yes	Yes
Valve Type	Butterfly	Butterfly	Butterfly	Globe	Globe	Globe
Operator	Pneumatic	Pneumatic	Pneumatic	Pneumatic	Pneumatic	Pneumatic
Primary Actuation	Electric	Electric	Electric	Electric	Electric	Electric
Secondary Actuation	Manual	Manual	Manual	Manual	Manual	Manual
Normal Position	Close	Close	Close	Open	Open	Open
Shutdown Position	Close	Close	Close	Close	Close	Close
Post-Accident Position	Close	Close	Close	Close	Close	Close
Power Fail Position	Close	Close	Close	Close	Close	Close
Containment Isolation Signal (c)	A, K, XX, YY, RM	A, K, XX, YY, RM	A, K, XX, YY, RM	A, K, XX, YY, RM	A, K, XX, YY, RM	A, K, XX, YY, RM
Closure Time (s)	<20	<15	<20	<15	<15	<15
Power Source (Div)	I	I	I	I	II	II
See notes at the end of this table.						

Table 6.2-7 Containment Isolation Valve Information Atmospheric Control System

Valve No.	T31-F731	T31-F733A/B	T31-F735A-D	T31-F010	T31-F011
Tier 2 Figure	6.2-39 (Sheet 3)	6.2-39 (Sheet 3)	6.2-39 (Sheet 3)	6.2-39 (Sheet 1)	6.2-39 (Sheet 1)
Applicable Basis	RG 1.11	RG 1.11	RG 1.11	GDC 56	GDC 56
Fluid	DW Atmosphere	DW Atmosphere	DW Atmosphere	Air or N_2	Air or N_2
Line Size	20A	20A	20A	250A	550A
ESF	No	No	No	Yes	Yes
Leakage Class	(a)	(a)	(a)	(a)	(a)
Location	O	O	O	O	O
Type C Leak Test	No(m)	No(m)	No(m)	Yes(e)	Yes(e)
Valve Type	Gate	Gate	Gate	Butterfly	Butterfly
Operator	Solenoid	Solenoid	Solenoid	Pneumatic	Pneumatic
Primary Actuation	Electric	Electric	Electric	Electric	Electric
Secondary Actuation	N/A	N/A	N/A	Manual	Manual
Normal Position	Open	Open	Open	Open	Close
Shutdown Position	Open	Open	Open	Open	Close
Post-Accident Position	Open	Open	Open	Open	Close
Power Fail Position	Open	Open	Open	Open	Close
Containment Isolation Signal [c]	RM	RM	RM	RM	A, K XX, YY, RM
Closure Time (s)	N/A	N/A	N/A	<20	<20
Power Source (Div)	N/A	N/A	N/A	I	III

See notes at the end of this table.

Table 6.2-7 Containment Isolation Valve Information Atmospheric Control System

Valve No.	T31-F737A-B	T31-F739A-D	T31-F741A-D	T31-F743A-D
Tier 2 Figure	6.2-39 (Sheet 3)	6.2-39 (Sheet 2)	6.2-39 (Sheet 3)	6.2-39 (Sheet 2)
Applicable Basis	RG 1.11	RG 1.11	RG 1.11	RG 1.11
Fluid	WW Atmosphere	WW Atmosphere	SP H_20	WW Atmosphere
Line Size	20A	20A	20A	20A
ESF	No	No	No	No
Leakage Class	(a)	(a)	(a)	(a)
Location	O	O	O	O
Type C Leak Test	No(m)	No(m)	No(m)	No(m)
Valve Type	Gate	Gate	Gate	Gate
Operator	Solenoid	Solenoid	Solenoid	Solenoid
Primary Actuation	Electric	Electric	Electric	Electric
Secondary Actuation	N/A	N/A	N/A	N/A
Normal Position	Open	Open	Open	Open
Shutdown Position	Open	Open	Open	Open
Post-Accident Position	Open	Open	Open	Open
Power Fail Position	Open	Open	Open	Open
Containment Isolation Signal [(c)]	RM	RM	RM	RM
Closure Time (s)	N/A	N/A	N/A	N/A
Power Source (Div)	N/A	N/A	N/A	N/A

See notes at the end of this table.

Table 6.2-7 Containment Isolation Valve Information Atmospheric Control System

Valve No.	T31-F745A-D	T31-F801A/B	T31-F803A/B
Tier 2 Figure	6.2-392 (Sheet 2)	6.2-39 (Sheet 3)	6.2-39 (Sheet 3)
Applicable Basis	RG 1.11	RG 1.11	RG 1.11
Fluid	SP H_20	DW Atmosphere	DW Atmosphere
Line Size	20A	20A	20A
ESF	No	No	No
Leakage Class	(b)	(b)	(b)
Location	O	O	O
Type C Leak Test	No(m)	No(m)	No(m)
Valve Type	Gate	Gate	Gate
Operator	Solenoid	Solenoid	Solenoid
Primary Actuation	Electric	Electric	Electric
Secondary Actuation	N/A	N/A	N/A
Normal Position	Open	Open	Open
Shutdown Position	Open	Open	Open
Post-Accident Position	Open	Open	Open
Power Fail Position	Open	Open	Open
Containment Isolation Signal (c)	RM	RM	RM
Closure Time (s)	N/A	N/A	N/A
Power Source (Div)	N/A	N/A	N/A
See notes at the end of this table.			

Table 6.2-7 Containment Isolation Valve Information Atmospheric Control System

Valve No.	T31-F805A/B	T31-D001	T31-D002
Tier 2 Figure	6.2-39 (Sheet 3)	6.2-39 (Sheet 1)	6.2-39 (Sheet 1)
Applicable Basis	RG 1.11	GDC 56	GDC 56
Fluid	WW Atmosphere	WW Atmosphere	WW Atmosphere
Line Size	20A	250A	250A
ESF	No	Yes	Yes
Leakage Class	(a)	N/A	N/A
Location	O	O	O
Type C Leak Test	No(m)	No(p)	No(p)
Valve Type	Gate	Rupture Disk	Rupture Disk
Operator	Solenoid	Self	Self
Primary Actuation	Electric	N/A	N/A
Secondary Actuation	N/A	N/A	N/A
Normal Position	Open	Close	Close
Shutdown Position	Open	Close	Close
Post-Accident Position	Open	Open	Open
Power Fail Position	Open	N/A	N/A
Containment Isolation Signal(c)	RM	N/A	N/A
Closure Time (s)	N/A	N/A	N/A
Power Source (Div)	N/A	N/A	N/A
See notes at the end of this table.			

Table 6.2-7 Containment Isolation Valve Information Reactor Water Cleanup System

Valve No.	G31-F002	G31-F003	G31-F017	G31-F018
Tier 2 Figure	5.4-12 (Sheet 1)	5.4-12 (Sheet 1)	5.4-12 (Sheet 1)	5.4-12 (Sheet 1)
Applicable Basis	GDC 55	GDC 55	GDC 55	GDC 55
Fluid	RPV H_2O	RPV H_2O	RPV H_2O	RPV H_2O
Line Size	200A	200A	150A	150A
ESF	Yes(e)(t)	Yes(t)	Yes(t)	Yes(t)
Leakage Class	(a)	(a)	(a)	(a)
Location	I	O	O	I
Type C Leak Test	Yes(e)(t)	Yes(t)	Yes(t)	Yes(t)
Valve Type	Gate	Gate	Gate	Check
Operator	Motor	Motor	Motor	Self
Primary Actuation	Electrical	Electrical	Electrical	N/A
Secondary Actuation	Manual	Manual	Manual	N/A
Normal Position	Open	Open	Close	Close
Shutdown Position	Open	Open	Close	Close
Post-Accident Position	Close	Close	Close	Close
Power Fail Position	As is	As is	As is	N/A
Containment Isolation Signal[(c)]	B,F,V,Z,AA	B,F,V,Z,CC,AA	B,F,V,Z,CC,AA	N/A
Closure Time (s)	<30	<30	<30	Instantaneous
Power Source (Div)	II	I	I	N/A
See notes at the end of this table.				

Table 6.2-7 Containment Isolation Valve Information Reactor Water Cleanup System

Valve No.	G31-F071	G31-F072
Tier 2 Figure	5.4-12 (Sheet 1)	5.4-12 (Sheet 1)
Applicable Basis	GDC55	GDC55
Fluid	RPV H_2O	RPV H_2O
Line Size	20A	20A
ESF	No	No
Leakage Class	(a)	(a)
Location	I	O
Type C leak Test	Yes	Yes
Valve Type	Globe	Globe
Operator	Pneumatic	Pneumatic
Primary Actuation	Electrical	Electrical
Secondary Actuation	Manual	Manual
Normal Position	Close	Close
Shutdown Position	Close	Close
Post-accident Position	Close	Close
Power Fail Position	Close	Close
Containment Isolation Signal(c)	C,E,F,H,N,BB,RM	C,E,F,H,N,BB,RM
Closure Time(s)	<15	<15
Power Source (Div)	II	I
See notes at the end of this table.		

Table 6.2-7 Containment Isolation Valve Information Reactor Water Cleanup System

Valve No.	G31-F700A/B	G31-F701A/B	G31-F702A/B	G31-F703A/B
Tier 2 Figure	5.4-12 (Sheet 1)	5.4-12 (Sheet 1)	5.4-12 (Sheet 1)	5.4-12 (Sheet 1)
Applicable Basis	GDC 55 RG 1.11	GDC 55 RG 1.11	GDC 55 RG 1.11	GDC 55 RG 1.11
Fluid	RPV H_2O	RPV H_2O	RPV H_2O	RPV H_2O
Line Size	20A	20A	20A	20A
ESF	No	No	No	No
Leakage Class	(a)	(a)	(a)	(a)
Location	O	O	O	O
Type C Leak Test	No(m)	No(m)	No(m)	No(m)
Valve Type	Globe	Globe	XS Check	XS Check
Operator	Manual	Manual	Self	Self
Primary Actuation	N/A	N/A	N/A	N/A
Secondary Actuation	N/A	N/A	N/A	N/A
Normal Position	Open	Open	Open	Open
Shutdown Position	Open	Open	Open	Open
Post-Accident Position	Close	Close	Close	Close
Power Fail Position	N/A	N/A	N/A	N/A
Containment Isolation Signal(c)	N/A	N/A	N/A	N/A
Closure Time (s)	N/A	N/A	Instantaneous	Instantaneous
Power Source (Div)	N/A	N/A	N/A	N/A

See notes at the end of this table.

Table 6.2-7 Containment Isolation Valve Information
Suppression Pool Cleanup System

Valve No.	G51-F001	G51-F002	G51-F006	G51-F007
Tier 2 Figure	9.5-1	9.5-1	9.5-1	9.5-1
Applicable Basis	GDC 56	GDC 56	GDC 56	GDC 56
Fluid	Water	Water	Water	Water
Line Size	200A	200A	250A	250A
ESF	No	No	No	No
Leakage Class	(a)	(a)	(a)	(a)
Location	O	O	O	O
Type C Leak Test	No(q)	No(q)	No(r)	No(r)
Valve Type	Gate	Gate	Check	Gate
Operator	Motor	Motor	Self	Motor
Primary Actuation	Electrical	Electrical	N/A	Electrical
Secondary Actuation	Manual	Manual	N/A	Manual
Normal Position	Close	Close	Open	Close
Shutdown Position	Open/Close	Open/Close	Open/Close	Open/Close
Post-Accident Position	Close	Close	Close	Close
Power Fail Position	As is	As is	N/A	As is
Containment Isolation Signal[(c)]	A,K,X,RM	A,K,X,RM	A,K,X,RM	A,K,X,RM
Closure Time (s)	<30	<30	Inst.	<30
Power Source (Div)	II	I	N/A	II
See notes at the end of this table.				

Table 6.2-7 Containment Isolation Valve Information Reactor Building Cooling Water System

Valve No.	P21-F075A /F076A	P21-F081A /F080A	P21-F075B /F076B	P21-F081B /F080B
Tier 2 Figure	9.2-1 (Sheet 3)	9.2-1 (Sheet 3)	9.2-1 (Sheet 6)	9.2-1 (Sheet 6)
Applicable Basis	GDC 56	GDC 56	GDC 56	GDC 56
Fluid	Water	Water	Water	Water
Line Size	200A	200A	200A	200A
ESF	Yes	Yes	Yes	Yes
Leakage Class	(a)	(a)	(a)	(a)
Location	O/I	O/I	O/I	O/I
Type C Leak Test	No(t)	No(t)	No(t)	No(t)
Valve Type	Gate/Check	Gate/Gate	Gate/Check	Gate/Gate
Operator	Motor/NA	Motor/Motor	Motor/NA	Motor/Motor
Primary Actuation	Electrical	Electrical	Electrical	Electrical
Secondary Actuation	Manual	Manual	Manual	Manual
Normal Position	Open	Open	Open	Open
Shutdown Position	Open	Open	Open	Open
Post-Accident Position	Close/Open	Close/Open	Close/Open	Close/Open
Power Fail Position	As is	As is	As is	As is
Containment Isolation Signal(c)	CX,K	CX,K	CX,K	CX,K
Closure Time (s)	80/Instantaneous	80/80	80/Instantaneous	80/80
Power Source (Div)	I/N/A	I/II	I/N/A	I/II
See notes at the end of this table.				

Table 6.2-7 Containment Isolation Valve Information HVAC Normal Cooling Water System

Valve No.	P24-F053	P24-F054	P24-F142	P24-F141
Tier 2 Figure	9.2-2	9.2-2	9.2-2	9.2-2
Applicable Basis	GDC 56	GDC 56	GDC 56	GDC 56
Fluid	Water	Water	Water	Water
Line Size	100A	100A	100A	100A
ESF	No	No	No	No
Leakage Class	(a)	(a)	(a)	(a)
Location	O	I	O	I
Type C Leak Test	Yes(t)	Yes(t)	Yes(t)	Yes(t)
Valve Type	Gate	Check	Gate	Gate
Operator	Motor	Self	Motor	Motor
Primary Actuation	Electrical	N/A	Electrical	Electrical
Secondary Actuation	HW	N/A	HW	N/A
Normal Position	Open	Open	Open	Open
Shutdown Position	Open	Open	Open	Open
Post-Accident Position	Close	N/A	Close	Close
Power Fail Position	As is	N/A	As is	As is
Containment Isolation Signal[(c)]	CX,K, RM	N/A	CX,K, RM	CX,K, RM
Closure Time (s)	25	Instantaneous	25	25
Power Source (Div)	I	N/A	I	II

See notes at the end of this table.

Table 6.2-7 Containment Isolation Valve Information
Service Air System

Valve No.	P51-F131	P51-F132
Tier 2 Figure	9.3-7 (Sheet 2)	9.3-7 (Sheet 2)
Applicable Basis	GDC 56	GDC-56
Fluid	Air	Air
Line Size	25A	25A
ESF	No	No
Leakage Class	(b)	(b)
Location	O	I
Type C Leak Test	Yes	Yes
Valve Type	Globe	Check
Operator	HW	Self
Primary Actuation	Manual	N/A
Secondary Actuation	N/A	N/A
Normal Position	Close	Close
Shutdown Position	Open	Open
Post-Accident Position	Close	Close
Power Fail Position	N/A	N/A
Containment Isolation Signal(c)	N/A	N/A
Closure Time (s)	N/A	Instantaneous
Power Source (Div)	N/A	N/A
See notes at the end of this table.		

Table 6.2-7 Containment Isolation Valve Information Instrument Air System

Valve No.	P52-F276	P52-F277
Tier 2 Figure	9.3-6 (Sheet 1)	9.3-6 (Sheet 1)
Applicable Basis	GDC 56	GDC 56
Fluid	N_2	N_2
Line Size	50A	50A
ESF	No	No
Leakage Class	(b)	(b)
Location	O	I
Type C Leak Test	Yes	Yes
Valve Type	Globe	Check
Operator	Motor	Self
Primary Actuation	Electrical	N/A
Secondary Actuation	HW	N/A
Normal Position	Open	Open
Shutdown Position	Open	Open
Post-Accident Position	Close	Close
Power Fail Position	As is	N/A
Containment Isolation Signal[(c)]	C	N/A
Closure Time (s)	20	Instantaneous
Power Source (Div)	I	N/A
See notes at the end of this table.		

Table 6.2-7 Containment Isolation Valve Information High Pressure Nitrogen Gas Supply System

Valve No.	P54-F007A/F008A	P54-F007B/F008B	P54-F200/F208
Tier 2 Figure	6.7-1	6.7-1	6.7-1
Applicable Basis	GDC 56	GDC 56	GDC 56
Fluid	N_2	N_2	N_2
Line Size	50A	50A	50A
ESF	Yes	Yes	Yes
Leakage Class	(a)	(a)	(a)
Location	O/I	O/I	O/I
Type C Leak Test	No(s)	No(s)	No(s)
Valve Type	Globe/Check	Globe/Check	Globe/Check
Operator	Motor/Self	Motor/Self	Motor/Self
Primary Actuation	Electrical/N/A	Electrical/N/A	Electrical/N/A
Secondary Actuation	HW/N/A	HW/N/A	HW/N/A
Normal Position	Open	Open	Open
Shutdown Position	Open	Open	Open
Post-Accident Position	Close	Close	Close
Power Fail Position	As Is/N/A	As Is/N/A	As Is/N/A
Containment Isolation Signal[c]	N/A	N/A	N/A
Closure Time (s)	30 / Instantaneous	30 / Instantaneous	30 / Instantaneous
Power Source (Div)	I/N/A	II/N/A	I/N/A
See notes at the end of this table.			

Table 6.2-7 Containment Isolation Valve Information Makeup Water System (Purified)

Valve No.	P11-F141	P11-F142
Tier 2 Figure	9.2-5 (Sheet 2)	9.2-5 (Sheet 2)
Applicable Basis	GDC 56	GDC 57
Fluid	Water	Water
Line Size	50A	50A
ESF	No	No
Leakage Class	(b)	(b)
Location	O	I
Type C Leak Test	Yes	Yes
Valve Type	Globe	Check
Operator	Manual	Self
Primary Actuation	N/A	N/A
Secondary Actuation	N/A	N/A
Normal Position	Close	Close
Shutdown Position	Open	Open
Post-Accident Position	Close	Close
Power Fail Position	N/A	N/A
Containment Isolation Signal[(c)]	N/A	N/A
Closure Time (s)	N/A	Instantaneous
Power Source (Div)	N/A	N/A
See notes at the end of this table.		

Table 6.2-7 Containment Isolation Valve Information
Leak Detection & Isolation System

Valve No.	E31-F002	E31-F003	E31-F004	E31-F005	E31-F009/ F010	E31-F702/ F704 A/B/C/D
Tier 2 Figure	5.2-8 (Sheet 9)	5.2-8 (Sheet 9)	5.2-8 (Sheet 9)	5.2-8 (Sheet 9)	5.2-8 (Sheet 8)	5.2-8 (Sheet 6)
Applicable Basis	GDC 56	GDC 56	GDC 56	GDC 56	GDC 57	RG 1.11
Fluid	Air	Air	Air	Air	Water	Steam
Line Size	32A	32A	32A	32A	20A	20A
ESF	Yes	Yes	Yes	Yes	Yes	No
Leakage Class	(a)	(a)	(a)	(a)	(a)	(a)
Location	O	O	O	O	O	O
Type C Leak Test	Yes(e)	Yes(e)	Yes(e)	Yes(e)	Yes(e)	No(m)
Valve Type	Globe	Globe	Globe	Globe	Globe	Excess Flow Check
Operator	Solenoid	Solenoid	Solenoid	Solenoid	N/A	Self
Primary Actuation	Electric	Electric	Electric	Electric	Manual	N/A
Secondary Actuation	N/A	N/A	N/A	N/A	N/A	N/A
Normal Position	Open	Open	Open	Open	Close	Open
Shutdown Position	Close	Close	Close	Close	Close	Open
Post-Accident Position	Close	Close	Close	Close	Close	Open
Power Fail Position	Close	Close	Close	Close	N/A	Open
Containment Isolation Signal(c)	B,K,RM	B,K,RM	B,K,RM	B,K,RM	N/A	N/A
Closure Time (s)	<15	<15	<15	<15	N/A	Instan-tane-ous
Power Source (Div)	I	II	II	I	N/A	N/A
See notes at the end of this table.						

Table 6.2-7 Containment Isolation Valve Information
Radwaste System

Valve No.	K17-F003	K17-F004	K17-F103	K17-F104
Tier 2 Figure	11A.2-2 (Sheet 29)	11A.2-2 (Sheet 29)	11A.2-2 (Sheet 30)	11A.2-2 (Sheet 30)
Applicable Basis	GDC 56	GDC 56	GDC 56	GDC 56
Fluid	LCW H_20	LCW H_20	HCW H_20	HCW H_20
Line Size	65A	65A	65A	65A
ESF	No	No	No	No
Leakage Class	(b)	(b)	(b)	(b)
Location	I	O	I	O
Type C Leak Test	No(w)	No(w)	No(w)	No(w)
Valve Type	Gate	Gate	Gate	Gate
Operator	Motor	Motor	Motor	Motor
Primary Actuation	Electrical	Electrical	Electrical	Electrical
Secondary Actuation	Manual	Manual	Manual	Manual
Normal Position	Open	Open	Open	Open
Shutdown Position	Open	Open	Open	Open
Post-Accident Position	Close	Close	Close	Close
Power Fail Position	As Is	As Is	As Is	As Is
Containment Isolation Signal(c)	A,K,RM	FF,A,K,RM	A,K,RM	FF,A,K,RM
Closure Time (s)	<30	<30	<30	<30
Power Source (Div)	II	I	II	I
See notes at the end of this table.				

Table 6.2-7 Containment Isolation Valve Information Neutron Monitoring System

Valve No.	C51-J004A	C51-J004B	C51-J004C	C51-J011
Tier 2 Figure	7.6-2 (Sheet 3)	7.6-2 (Sheet 3)	7.6-2 (Sheet 3)	7.6-2 (Sheet 3)
Applicable Basis	GDC57	GDC57	GDC57	GDC57
Fluid	N_2	N_2	N_2	N_2
Line Size	OD15	OD15	OD15	20A
ESF	No	No	No	No
Leakage Class	(a)	(a)	(a)	(a)
Location	O	O	O	O
Type C leak Test	Yes	Yes	Yes	Yes
Valve Type	Ball	Ball	Ball	Globe
Operator	Motor	Motor	Motor	Solenoid
Primary Actuation	Electrical	Electrical	Electrical	Electrical
Secondary Actuation	N/A	N/A	N/A	N/A
Normal Position	Close	Close	Close	Close
Shutdown Position	Close	Close	Close	Close
Post-accident Position	Close	Close	Close	Close
Power Fail Position	Close	Close	Close	Close
Containment Isolation Signal(c)	A,K	A,K	A,K	A,K
Closure Time(s)	<3	<3	<3	Instantaneous
Power Source (Div)	N/A	N/A	N/A	N/A
See notes at the end of this table.				

Notes:

(a) Termination Region: Secondary Containment

(b) Termination Region: Main Condenser, Turbine Building, Bypass Leakage Barrier: Redundant Primary Containment Isolation Valves.

(c) Isolation Signal Codes

Signal	Description
A	Reactor vessel low water level–Level 3.
B	Reactor vessel low water level–Level 2.
C	Reactor vessel low water level–Level 1.5.
CX	Reactor vessel low water level–Level 1.
E	Line break–main steamline (steamline high steam flow).
F	Line break–main steamline (steamline high tunnel temperature).
H	Line break–main steamline (steamline high turbine building temperature.)
J	Turbine Building high temperature.
K	High drywell pressure.
L	RHR injection valve low pressure.
N	Low main condenser vacuum.
S	Line break in RCIC System steamline to turbine (low steamline pressure).
T	High pressure RCIC turbine exhaust.
U	Reactor high pressure.
V	Close-through electrical interlocks with other valves or pump motors.
W	Condensate storage tank low level.
X	Suppression pool low level.
XX	High Radiation–R/B HVAC exhaust air

Signal	Description
Y	RCIC
YY	High Radiation–Refueling floor exhaust air
Z	Equipment area temperature high
AA	Differential mass flow high
BB	Low main steamline pressure at inlet to turbine (RUN mode only).
CC	Line break in Reactor Water Cleanup System (high space temperature).
DD	Containment pressure.
EE	High differential flow in the reactor water cleanup system.
FF	High radiation–process line
RM	Remote manual switch from control room (All automatic initiated isolation valves are capable of remote manual operation from the control room).

(d) This line is filled with water and pressurized higher than 110% of the post-accident peak containment pressure. Line is small and postulated failure is considered less severe than instrument line.

(e) Leakage testing may be performed in the reverse direction in the absence of test connections and/or isolable test boundaries in the upstream side of the valve relative to the leakage flow direction (i.e., from inside to outside primary containment). The results are conservative or equivalent to the normal direction as described below:

(1) For globe valves including MSIVs, testing in the reverse direction is conservative, since the test pressure tends to lift the plug from the seat.

(2) For gate valves and butterfly valves, leakage characteristics for these types of valves are similar in both directions provided seat construction is designed for sealing on either side.

(f) These lines are CAM System sample lines that continuously monitor (sample) post-accident containment atmosphere. These lines are safety grade closed loop extension of the primary containment. Sampled gases (or leakage if any) are returned to the primary

containment. In addition, these lines are subject to periodic Type-A test whose leaktight integrity can be verified.

(g) The RHR drywell and wetwell spray lines are always filled with water in the outboard side, thereby providing water seal. The seal is maintained at a pressure higher than 110% of the post-accident peak containment pressure by jockey pumps and/or hydrostatic head; thus precluding leakage.

Furthermore, these valves are required to open post-LOCA to provide containment cooling function. When this function is activated, flow direction is towards the containment.

(h) The ECCS (RHR, HPCF and RCIC) test return and minimum flow lines terminate below the suppression pool water level and are sealed from the containment atmosphere by the suppression pool water. The outboard side of the valve (away from the containment) is always filled with water and pressurized higher than 110% of the post-accident peak containment pressure as in(g) above.

(i) The ECCS (RHR, HPCF and RCIC) 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.

(j) The RHR suppression pool cooling discharge line is the same line used for system flow or pump flow testing. See (h) above.

(k) The ECCS (RHR, HPCF and RCIC) injection lines are always filled withwater up to the outboard isolation valves, thereby forming a water seal. These water seals are kept pressurized higher than 110% of the post-accident containment pressure as in (g) above. Furthermore, these valves are subject to ASME/ANSI OMa-1988 Addenda to ASME/ANSI OM-1987, Part 10, leak rate tests.

(l) RCIC vacuum pump discharge line terminates below the suppression pool water level and is sealed from the containment atmosphere.

(m) Instrument lines that penetrate the primary containment conform to Regulatory Guide 1.11. The lines that connect to the reactor 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 maintenance valves and excess flow check isolation valves or equivalent. These lines are normally open, and are considered an extension of the primary containment whose integrity is continuously demonstrated during normal operation. In addition, these lines are subject to periodic Type A tests. Leaktight integrity is also verified during functional and surveillance activities as well as visual observations during operator tours.

(n) The outboard side of the RHR shutdown cooling suction valves are sealed with water since RHR pump and suction lines are located below the suppression pool water level.

This is a closed-loop water seal, since RHR is a closed loop system always filled with water.

(o) Furthermore, these valves are subject to ASME leak rate tests as in (k) above.

(p) Rupture discs are normally closed and sealed from leakage. The opening setpoint of these rupture discs is higher than primary containment test pressures. Additionally, these rupture discs are subject to the Type A test.

(q) SPCU suction line is always filled with water, since it is located below the suppression pool water level and is sealed from the containment atmosphere.

(r) SPCU return line terminates below the suppression pool water level and is sealed from the containment atmosphere.

(s) The outboard side of these valves is always pressurized with nitrogen gas at a pressure higher than the post-accident peak containment pressure. The nitrogen supply in these lines is required for post-accident mitigating function.

(t) The outboard side of these valves is always filled with water and pressurized above 110% post-accident peak containment pressure. These lines are kept charged with cooling water for cooling emergency equipment necessary for post-accident mitigation.

(u) Line will be drained and tested with air.

(v) Not Used.

(w) These lines terminate below the drywell sumps water level and are sealed from the containment atmosphere.

(x) The outboard side of these valves are provided with a water leg. In addition, these valves are subject to ASME leak tests as in (k) above.

(y) Not applicable.

Table 6.2-8 Primary Containment Penetration List*

Penetration Number	Name	Elevation (mm)	Azimuth (deg)	Offset (mm)	Diameter (mm)	Barrier Type	Testing†‡
X-1	U/D Equipment Hatch	19170	130	0	2600	Door	B
X-2	U/D Personnel Hatch	19170	230	0	2400	Door	B
X-3	ISI Hatch	19500	145	0	200	Door	B
X-4	Wetwell Access Hatch	6400	45	0	2000	Door	B
X-5	L/D Personnel Hatch	-650	0	0	2400/4300	Door	B
X-6	L/D Equipment Hatch	-900	180	0	2400/4300	Door	B
X-10A	Mainsteam Line	16300	0	1400	1200	Valve	C
X-10B	Mainsteam Line	16300	0	4200	1200	Valve	C
X-10C	Mainsteam Line	16300	0	-4200	1200	Valve	C
X-10D	Mainsteam Line	16300	0	-1400	1200	Valve	C
X-11	Mainsteam Drain	13650	0	5200	500	Valve	C
X-12A	Feedwater Line	13810	0	2800	950	Valve	C
X-12B	Feedwater Line	13810	0	-2800	950	Valve	C
X-22	Borated Water Injection	15250	275	0	450	Valve	A
X-30B	Drywell Spray	14680	260	-3400	200	Valve	A
X-30C	Drywell Spray	14680	100	3400	200	Valve	A
X-31A	HPCF (B)	14630	260	0	600	Valve	A
X-31B	HPCF (C)	14630	100	0	600	Valve	A
X-32A	LPFL (B) RHR (B)	14610	260	-2000	650	Valve	A
X-32B	LPFL (C) RHR (C)	14610	100	-1800	650	Valve	A
X-33A	RHR Suction (A)	14550	80	-800	750	Valve	A
X-33B	RHR Suction (B)	14550	260	1800	750	Valve	A
X-33C	RHR Suction (C)	14550	100	2000	750	Valve	A
X-37	RCIC Turbine Steam	14414	80	1200	550	Valve	C
X-38	RPV Head Spray	14450	310	1500	550	Valve	C

* This table provided in response to Questions 430.49d & e.
† All penetrations will be subject to the Type A test. Those penetrations subject to Type B testing are also tested in the Type A test.
‡ All penetrations excluded from Type B testing are welded penetrations and do not include resilient seals in their design.

Table 6.2-8 Primary Containment Penetration List* (Continued)

Penetration Number	Name	Elevation (mm)	Azimuth (deg)	Offset (mm)	Diameter (mm)	Barrier Type	Testing†‡
X-50	CUW Pump Feed	14480	310	0	600	Valve	C
X-60	MUWP Suction	13500	290	0	200	Valve	C
X-61	RCW Suction (A)	13700	45	-3000	200	Valve	A
X-62	RCW Return (A)	13700	45	-2000	200	Valve	A
X-63	RCW Suction (B)	13500	225	3400	200	Valve	A
X-64	RCW Return (B)	13500	225	2400	200	Valve	A
X-65	HNCW Supply	13500	225	250	350	Valve	C
X-66	HNCW Return	13500	225	1400	350	Valve	C
X-69	SA	19000	42	0	90	Valve	C
X-70	IA	19000	46	0	200	Valve	C
X-71A	ADS Accumulator (A)	19000	50	0	200	Valve	A
X-71B	ADS Accumulator (B)	19000	296.5	1000	200	Valve	A
X-72	SRV Accumulator	19000	296.5	2000	200	Valve	A
X-80	Drywell Purge Suction	13700	68	0	550	Valve	C
X-81	Drywell Purge Exhaust	19000	216	0	550	Valve	C
X-82	Spare	14850	225	-600	150	Welded Cap	A
X-90	Spare	20100	50	0	400	Welded Cap	A
X-91	Spare	20100	296.5	1000	300	Welded Cap	A
X-92	Spare	14700	45	-1000	300	Welded Cap	A
X-93	Spare	14700	135	-500	400	Welded Cap	A
X-94	Spare	16400	300	-500	400	Welded Cap	A
X-95	Spare	9400	45	-400	400	Welded Cap	A

* This table provided in response to Questions 430.49d & e.
† All penetrations will be subject to the Type A test. Those penetrations subject to Type B testing are also tested in the Type A test.
‡ All penetrations excluded from Type B testing are welded penetrations and do not include resilient seals in their design.

Table 6.2-8 Primary Containment Penetration List* (Continued)

Penetration Number	Name	Elevation (mm)	Azimuth (deg)	Offset (mm)	Diameter (mm)	Barrier Type	Testing†‡
X-100A	RIP Power	13500	55	-1100	450	O-ring	B
X-100B	RIP Power	13500	180	2725	450	O-ring	B
X-100C	RIP Power	13500	180	-6550	300	O-ring	B
X-100D	RIP Power	13500	280	0	450	O-ring	B
X-100E	RIP Power	13500	180	-2725	450	O-ring	B
X-100F	RIP Power	13500	280	1350	450	O-ring	B
X-101A	LP Power	16400	45	0	450	O-ring	B
X-101B	LP Power	16400	180	125	450	O-ring	B
X-101C	LV Power	16400	180	-1425	300	O-ring	B
X-101D	FMCRD Power	19000	279.81	-1350	300	O-ring	B
X-101E	FMCRD Power	19000	260.5	-1350	300	O-ring	B
X-101F	FMCRD Power		279.5	1350	300	O-ring	B
X-101G	FMCRD Power	19000	99	1350	300	O-ring	B
X101J	LV Power		180	5250	300	O-ring	B
X101K	LV Power		45	3900	300	O-ring	B
X-102A	I & C	16400	45	-1350	300	O-ring	B
X-102B	I & C	16400	180	1425	450	O-ring	B
X-102C	I & C		180	-2725	300	O-ring	B
X-102D	I & C	16100	280	1350	300	O-ring	B
X-102E	I & C	19000	99	-1350	300	O-ring	B
X-102F	I & C	19000	279.5	-1350	300	O-ring	B
X-102G	I & C	13500	180	-1175	300	O-ring	B
X102H	I & C	13500	180	-5250	300	O-ring	B
X102-J	I & C	13500	55	1100	300	O-ring	B
X-103A	I & C	Note 1	340.5	0	150	O-ring	B
X-103B	I & C	Note 1	211	0	150	O-ring	B
X-103C	I & C	Note 1	134	0	150	O-ring	B
X-103D	1 & C	Note 1	295	5600	150	O-ring	B
X-103E	1 & C	Note 1	211	1350	300	O-ring	B

* This table provided in response to Questions 430.49d & e.
† All penetrations will be subject to the Type A test. Those penetrations subject to Type B testing are also tested in the Type A test.
‡ All penetrations excluded from Type B testing are welded penetrations and do not include resilient seals in their design.

Table 6.2-8 Primary Containment Penetration List* (Continued)

Penetration Number	Name	Elevation (mm)	Azimuth (deg)	Offset (mm)	Diameter (mm)	Barrier Type	Testing†‡
X-104A	FMCRD Position Indicator	19000	81	0	300	O-ring	B
X-104B	FMCRD Position Indicator	19000	260.5	0	300	O-ring	B
X-104C	FMCRD Position Indicator	20100	99	1350	450	O-ring	B
X-104D	FMCRD Position Indicator	20100	279.5	1350	450	O-ring	B
X-104E	FMCRD Position Indicator	19000	81	1350	300	O-ring	B
X-104F	FMCRD Position Indicator	19000	260.5	1350	450	O-ring	B
X-104G	FMCRD Position Indicator	19000	99	0	300	O-ring	B
X-104H	FMCRD Position Indicator	19000	279.5	0	450	O-ring	B
X-105A	Neutron Detection	20100	81	1350	450	O-ring	B
X-105B	Neutron Detection	20100	260.5	1350	450	O-ring	B
X-105C	Neutron Detection	20100	99	-1350	450	O-ring	B
X-105D	Neutron Detection	20100	279.5	-1350	450	O-ring	B
X-106A	Div I Instrumentation	16400	45	1350	300	O-ring	B
X-106B	Div II Instrumentation	13500	180	125	300	O-ring	B
X-106C	Div III Instrumentation	16400	180	-6200	300	O-ring	B
X-106D	Div IV Instrumentation	16100	280	0	300	O-ring	B
X-106F	Div NON Instrumentation	16400	180	2725	300	O-ring	B
X-106G	Div NON Instrumentation	16400	45	2700	300	O-ring	B
X-106H	Div NON Instrumentation	14700	55	1000	300	O-ring	B

* This table provided in response to Questions 430.49d & e.
† All penetrations will be subject to the Type A test. Those penetrations subject to Type B testing are also tested in the Type A test.
‡ All penetrations excluded from Type B testing are welded penetrations and do not include resilient seals in their design.

Table 6.2-8 Primary Containment Penetration List* (Continued)

Penetration Number	Name	Elevation (mm)	Azimuth (deg)	Offset (mm)	Diameter (mm)	Barrier Type	Testing†‡
X-106J	Div NON Instrumentation	20100	260.5	-1350	300	O-ring	B
X-107A	Group B Instr	16400	180	-4950	300	O-ring	B
X-107B	Power and Control	13500	180	1425	450	O-ring	B
X-110	Spare	20100	99	0	300	Welded Cap	A
X-111	Spare	20100	260.5	0	300	O-ring	B
X-112	Spare	20100	279.5	0	300	O-ring	B
X-130A	I & C	13500	45	0	300	Valve	A
X-130B	I & C	13500	212	0	300	Valve	A
X-130C	I & C	13500	124	0	300	Valve	A
X-130D	I & C	13500	295	0	300	Valve	A
X-140A	I & C	12935	45	-2500	250	Valve	A
X-140B	I & C	13500	300	0	300	Valve	A
X-141A	I & C	13500	63.5	0	300	Valve	A
X-141B	I & C	13500	275	0	300	Valve	A
X-142A	I & C	20100	38	0	90	Valve	A
X-142B	I & C	20100	244	0	90	Valve	A
X-142C	I & C	20100	116	0	90	Valve	A
X-142D	I & C	20100	296.5	2000	90	Valve	A
X-143A	I & C	14700	45	0	90	Valve	A
X-143B	I & C	14700	212	0	90	Valve	A
X-143C	I & C	14700	124	0	90	Valve	A
X-143D	I & C	14700	300	0	90	Valve	A
X-144A	I & C	12700	45	0	90	Valve	A
X-144B	I & C	12700		0	90	Valve	A
X-144C	I & C	12700	124	0	90	Valve	A
X-144D	I & C	12700	300	0	90	Valve	A
X-146A	I & C	19000	38	0	300	Valve	A

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† All penetrations will be subject to the Type A test. Those penetrations subject to Type B testing are also tested in the Type A test.
‡ All penetrations excluded from Type B testing are welded penetrations and do not include resilient seals in their design.

Table 6.2-8 Primary Containment Penetration List* (Continued)

Penetration Number	Name	Elevation (mm)	Azimuth (deg)	Offset (mm)	Diameter (mm)	Barrier Type	Testing†‡
X-146B	I & C	19000	248	0	300	Valve	A
X-146C	I & C	19000	112	0	300	Valve	A
X-146D	I & C	19000	296.5	0	300	Valve	A
X-147	I & C	20100	248	0	90	Valve	A
X-160	LDS Monitor	20100	46	0	250	Valve	A
X-161A	CAMS I & C	20100	42.75	0	250	Welded Cap	A
X-161B	CAMS I & C	20100	292.5	0	250	Welded Cap	A
X-162A	Sample/Return Drywell Gas	19000	116	0	250	Valve	A
X-162B	Sample/Return Drywell Gas	19000	244	0	250	Valve	A
X-170	I & C	13400	310	0	200	Valve	A
X-171	SAMP	16400	45	-2700	250	Valve	A
X-177	I & C	15900	135	-500	250	Valve	A
X-200B	Wetwell Spray	8900	258	0	100	Valve	A
X-200C	Wetwell Spray	8900	102	0	100	Valve	A
X-201	RHR Pump Suction (A)	-7200	36	0	450	Valve	A
X-202	RHR Pump Suction (B)	-7200	216	0	450		A
X-203	RHR Pump Suction (C)	-7200	144	0	450	Valve	A
X-204	RHR Pump Test (A)	800	85	0	250	Valve	A
X-205	RHR Pump Test (B)	800	265	0	250	Valve	A
X-206	RHR Pump Test (C)	800	95	0	250	Valve	A
X-210	HPCF Pump Suction (B)	-7200	252	0	400	Valve	A

* This table provided in response to Questions 430.49d & e.
† All penetrations will be subject to the Type A test. Those penetrations subject to Type B testing are also tested in the Type A test.
‡ All penetrations excluded from Type B testing are welded penetrations and do not include resilient seals in their design.

Table 6.2-8 Primary Containment Penetration List* (Continued)

Penetration Number	Name	Elevation (mm)	Azimuth (deg)	Offset (mm)	Diameter (mm)	Barrier Type	Testing†‡
X-211	HPCF Pump Suction (C)	-7200	108	0	400	Valve	A
X-213	RCIC Turbine Exhaust	5848	60	0	550	Valve	C
X-214	RCIC Pump Suction	-7050	72	0	200	Valve	A
X-216	SPCU Pump Suction	-7450	283	0	200	Valve	A
X-217	SPCU Return	1700	340	0	250	Valve	A
X-240	Wetwell Purge Suction	9200	45	1200	550	Valve	C
X-241	Wetwell Purge Exhaust	9200	221	0	550	Valve	C
X-242	Spare	1500	225	-1000	150	Welded Cap	A
X-250	Spare	8500	45	0	400		A
X-251	Spare	9000	213	0	400		A
X-252	Spare	1500	50	0	300	Welded Cap	A
X-254	Spare	2650	225	-1000	300	Welded Cap	A
X-320	I & C	8900	74	0	90	Valve	A
X-321A	I & C	2200	112	0	300	Valve	A
X-321B	I & C	2200	248	0	300	Valve	A
X-322A	I & C	400	78	0	90	Valve	A
X-322B	I & C	400	258	0	90	Valve	A
X-322C	I & C	400	102	0	90	Valve	A
X-322D	I & C	400	282	0	90	Valve	A

* This table provided in response to Questions 430.49d & e.
† All penetrations will be subject to the Type A test. Those penetrations subject to Type B testing are also tested in the Type A test.
‡ All penetrations excluded from Type B testing are welded penetrations and do not include resilient seals in their design.

Table 6.2-8 Primary Containment Penetration List* (Continued)

Penetration Number	Name	Elevation (mm)	Azimuth (deg)	Offset (mm)	Diameter (mm)	Barrier Type	Testing†‡
X-322E	I & C	Note 1	75	0	90	Valve	A
X-322F	I & C	Note 1	254	0	90	Valve	A
X-322G	I & C	Note 1	106	0	90	Valve	A
X-322H	I & C	Note 1	282	0	90	Valve	A
X-323A	I & C	-5200	30	0	90	Valve	A
X-323B	I & C	-5200	210	0	90	Valve	A
X-323C	I & C	-5500	138	0	90	Valve	A
X-323D	I & C	-5200	304	0	90	Valve	A
X-323E	I & C	-7500	58	0	90	Valve	A
X-323F	I & C	-7500	230	0	90	Valve	A
X-323G	I & C	-7500	130	0	90	Valve	A
X-323H	I & C	-7500	302	0	90	Valve	A
X-331A	CAMS Gamma Det.	9700	76.5	0	250	Welded Cap	A
X-331B	CAMS Gamma Det.	9700	231	0	250	Welded Cap	A
X-332A	CAMS Sampling Ret.	9700	97	0	300	Valve	A
X-332B	CAMS Sampling Ret.	9700	261	0	300	Valve	A
X-342	I & C	9500	266	0	90	Valve	A
X-600A	TIP Drive	1693	0	-700	40	Valve	A
X-600B	TIP Drive	1693	0	0	40	Valve	A
X-600C	TIP Drive	1693	0	700	40	Valve	A
X-600D	TIP Drive Purge	1693	0	420	40	Valve	A
X-700A	RIP Purge Water Supply	265	180	-1750	20	Valve	A
X-700B	RIP Purge Water Supply	265	180	-1610	20	Valve	A
X-700C	RIP Purge Water Supply	-515	180	-1750	20	Valve	A

* This table provided in response to Questions 430.49d & e.
† All penetrations will be subject to the Type A test. Those penetrations subject to Type B testing are also tested in the Type A test.
‡ All penetrations excluded from Type B testing are welded penetrations and do not include resilient seals in their design.

Table 6.2-8 Primary Containment Penetration List* (Continued)

Penetration Number	Name	Elevation (mm)	Azimuth (deg)	Offset (mm)	Diameter (mm)	Barrier Type	Testing†‡
X-700D	RIP Purge Water Supply	515	180	-1610	20	Valve	A
X-700E	RIP Purge Water Supply	-765	180	-1610	20	Valve	A
X-700F	RIP Purge Water Supply	-265	180	-1470	20	Valve	A
X-700G	RIP Purge Water Supply	-15	180	-1330	20	Valve	A
X-700H	RIP Purge Water Supply	-15	180	-1470	20	Valve	A
X-700J	RIP Purge Water Supply	-15	180	-1610	20	Valve	A
X-700K	RIP Purge Water Supply	-15	180	-1750	20	Valve	A
X-710	CRD Insertion (Total 102)	1285	180	1680	32	Valve	A
X-740	Spare	85	180	1750	100	Welded Cap	A
X-750A	I&C (Core Diff Press.)	1135	180	-910	20	Valve	A
X-750B	I&C (Core Diff Press.)	985	180	1330	20	Valve	A
X-750C	I&C (Core Diff Press.)	1285	180	-910	20	Valve	A
X-750D	I&C (Core Diff Press.)	985	180	1470	20	Valve	A
X-751A	I&C (RIP Diff Press.)	985	180	-1470	20	Valve	A
X-751B	I&C (RIP Diff Press.)	1285	180	910	20	Valve	A
X-751C	I&C (RIP Diff Press.)	985	180	-1330	20	Valve	A
X-751D	I&C (RIP Diff Press.)	1135	180	910	20	Valve	A

* This table provided in response to Questions 430.49d & e.
† All penetrations will be subject to the Type A test. Those penetrations subject to Type B testing are also tested in the Type A test.
‡ All penetrations excluded from Type B testing are welded penetrations and do not include resilient seals in their design.

Table 6.2-8 Primary Containment Penetration List* (Continued)

Penetration Number	Name	Elevation (mm)	Azimuth (deg)	Offset (mm)	Diameter (mm)	Barrier Type	Testing†‡
X-780A	Spare	235	180	-1190	20	Welded Cap	A
X-780B	Spare	235	180	1190	20	Welded Cap	A
X-610	CRD Insertion (Total 103)	1285	0	1680	32	Valve	A
X-620	Low Conductivity Drain	-700	0	-1750	65		A
X-621	High Conductivity Drain	-450	0	-1750	150		A
X-650A	I&C (Core Diff Press.)	985	0	1330	20	Valve	A
X-650B	I&C (Core Diff Press.)	1285	0	-910	20	Valve	A
X-650C	I&C (Core Diff Press.)	985	0	1470	20	Valve	A
X-650D	I&C (Core Diff Press.)	1135	0	-910	20	Valve	A
X-651A	I&C (RIP Diff Press.)	1285	0	910	20	Valve	A
X-651B	I&C (RIP Diff Press.)	985	0	-1330	20	Valve	A
X-651C	I&C (RIP Diff Press.)	1135	0	910	20	Valve	A
X-651D	I&C (RIP Diff Press.)	985	0	-1470	20	Valve	A
X-680A	Spare	85	0	-1750	20	Welded Cap	A
X-680B	Spare	85	0	1750	20		B

* This table provided in response to Questions 430.49d & e.
† All penetrations will be subject to the Type A test. Those penetrations subject to Type B testing are also tested in the Type A test.
‡ All penetrations excluded from Type B testing are welded penetrations and do not include resilient seals in their design.

Note 1: Penetration is located such that the bottom of penetration sleeve is above pool swell impact zone (7700mm).

Table 6.2-9 Secondary Containment Penetration List*

Penetration Number	Name	Elevation (mm)	Diameter (mm)
1	RCW (B)	-8200	600
2	RCW (B)	-8200	600
3	HPCF	-8200	600
4	SS	-8200	50
5	RD (LCW)	-8200	80
6	RD (SD)	-8200	65
7	RD (HCW)	-8200	150
8	TV	-8200	250
9	RCW (A)	-8200	600
10	RCW (A)	-8200	600
11	RCW (C)	-8200	550
12	RCW (C)	-8200	550
13	HPCF	-8200	600
14	MUWC	-8200	250
15	CRD	-8200	150
16	CRD	-8200	50
17	SPH	-8200	150
18	RCW (B)	-1700	150
19	RCW (B)	-1700	150
20	RCW (B)	-1700	200
21	RCW (B)	-1700	200
22	MS	-1700	80
23	SA	-1700	65
24	IA	-1700	50
25	FP	-1700	150
26	RCW (A)	-1700	150
27	RCW (A)	-1700	150
28	RCW (A)	-1700	200
29	RCW (A)	-1700	200
30	HSR	-1700	150
31	RCW (C)	-1700	100

Table 6.2-9 Secondary Containment Penetration List* (Continued)

Penetration Number	Name	Elevation (mm)	Diameter (mm)
32	RCW (C)	-1700	100
33	RCW (C)	-1700	200
34	RCW (C)	-1700	200
35	HS	4800	150
36	MS	4800	80
37	LCW (FPC)	4800	150
38	LCW (CUW)	4800	150
39	RCIC	4800	50
40	MS (4)	16191	700
41	FDW (2)	13810	600
42	HVAC Exhaust	27200	†
43	HVAC Supply	31700	†
44	Controlled Access(2)	12300	‡
45	Equipment Lock	12300	‡
46	Railroad Car Door	12300	‡
47	HS	12300	150
48	Deleted		
49	Deleted		
50	HNCW	12300	250
51	HNCW	12300	250
52	MUWP	4800	150
53	AC	4800	50
54	AC	4800	250
55	Deleted		
56	Deleted		
57	Cabletrays	23500	
58	Cabletrays	12300	
59	Cabletrays	4800	

Table 6.2-9 Secondary Containment Penetration List* (Continued)

Penetration Number	Name	Elevation (mm)	Diameter (mm)
60	Not Used	—	—
61	Not Used	—	—
62	AFI	-1700	150
63	AFI (Drain Line)	-8200	20

* This table is provided in response to Question 430.34.

† These HVAC openings have safety-related isolation valves with both local monitoring and remote (in control room) monitoring.

‡ These doors are monitored in the control room as per Subsection 13.6.3.4.

Table 6.2-10 Potential Bypass Leakage Paths*

Penetration Number	Name	Diameter (mm)	Termination Region †	Leakage Barriers‡	Potential Bypass Path
X-1	U/D Equipment Hatch	2600	S	C/M-J	No
X-2	U/D Personnel Hatch	2400	S	C/M-J	No
X-3	ISI Hatch	200	S	C/M-J	No
X-4	Wetwell Access Hatch	2000	S	C/M-J	No
X-5	L/D Personnel Hatch	2400/4300	S	C/M-J	No
X-6	L/D Equipment Hatch	2400/4300	S	C/M-J	No
X-10A	Main Steamline	1200	E	E/D/G	Yes
X-10B	Main Steamline	1200	E	E/D/G	Yes
X-10C	Main Steamline	1200	E	E/D/G	Yes
X-10D	Main Steamline	1200	E	E/D/G	Yes
X-11	Main Steamline Drain	500	E	E/D/G	Yes
X-12A	Feedwater Line	950	E	E/D/L	Yes
X-12B	Feedwater Line	950	E	E/D/L	Yes
X-22	Borated Water Injection	450	S	E/C/L	No
X-30B	Drywell Spray	200	S	E/C/L	No
X-30C	Drywell Spray	200	S	E/C/L	No
X-31A	HPCF (B)	600	S	E/C/L	No
X-31B	HPCF (C)	600	S	E/C/L	No
X-32A	LPFL (B) RHR (B)	650	S	E/C/L	No
X-32B	LPFL (C) RHR (C)	650	S	E/C/L	No
X-33A	RHR Suction (A)	750	S	C/L	No
X-33B	RHR Suction (B)	750	S	C/L	No
X-33C	RHR Suction (C)	750	S	C/L	No
X-37	RCIC Turbine Steam	550	S	C/G	No
X-38	RPV Head Spray	550	S	E/C/L	No
X-50	CUW Pump Feed	600	S	E/C/L	No
X-60	MUWP Suction	200	S	C/L	No
X-61	RCW Suction (A)	200	E	E/D/H	No
X-62	RCW Return (A)	200	E	E/D/H	No
X-63	RCW Suction (B)	200	E	E/D/H	No
X-64	RCW Return (B)	200	E	E/D/H	No

Table 6.2-10 Potential Bypass Leakage Paths* (Continued)

Penetration Number	Name	Diameter (mm)	Termination Region †	Leakage Barriers‡	Potential Bypass Path
X-65	HNCW Suction	350	E	E/D/H	No
X-66	HNCW Return	350	E	E/D/H	No
X-69	SA	90	E	E/D	No
X-70	IA	200	E	E/D	No
X-71A	ADS Accumulator (A)	200	S	C/K	No
X-71B	ADS Accumulator (B)	200	S	C/K	No
X-72	Relief Valve Accumulator	200	S	C/K	No
X-82	Spare	150	S	E/C/J	No
X-90	Spare	400	P	B/A	No
X-91	Spare	300	P	B/A	No
X-92	Spare	300	P	B/A	No
X-93	Spare	400	P	B/A	No
X-94	Spare	400	S	B/A	No
X-95	Spare	400	S	B/A	No
X-100A	IP Power	450	S	C/J	No
X-100B	IP Power	450	S	C/J	No
X-100C	IP Power	300	S	C/J	No
X-100D	IP Power	450	S	C/J	No
X-100E	IP Power	450	S	C/J	No
X-100F	RIP Power	450	S	C/J	No
X-101A	LP Power	450	S	C/J	No
X-101B	LP Power	450	S	C/J	No
X-101C	LP Power	300	S	C/J	No
X-101D	FMCRD Power	300	S	C/J	No
X-101E	FMCRD Power	300	S	C/J	No
X-101F	FMCRD Power	300	S	C/J	No
X-101G	FMCRD Power	300	S	C/J	No
X-101J	LV Power	300	S	C/J	No
X-101K	LV Power	300	S	C/J	No
X-102A	I & C	300	S	C/J	No

Table 6.2-10 Potential Bypass Leakage Paths* (Continued)

Penetration Number	Name	Diameter (mm)	Termination Region †	Leakage Barriers‡	Potential Bypass Path
X-102B	I & C	450	S	C/J	No
X-102C	I & C	300	S	C/J	No
X-102D	I & C	300	S	C/J	No
X-102E	I & C	300	S	C/J	No
X-102F	I & C	300	S	C/J	No
X-102G	I & C	300	S	C/J	No
X-102H	I & C	300	S	C/J	No
X-102J	I & C	300	S	C/J	No
X-103A	I & C	150	S	C/J	No
X-103B	I & C	150	S	C/J	No
X-103C	I & C	150	S	C/J	No
X-103D	I & C	150	S	C/J	No
X-103E	I & C	300	S	C/J	No
X-104A	FMCRD Pos. Indicator	300	S	C/J	No
X-104B	FMCRD Pos. Indicator	300	S	C/J	No
X-104C	FMCRD Pos. Indicator	450	S	C/J	No
X-104D	FMCRD Pos. Indicator	450	S	C/J	No
X-104E	FMCRD Pos. Indicator	300	S	C/J	No
X-104F	FMCRD Pos. Indicator	450	S	C/J	No
X-104G	FMCRD Pos. Indicator	300	S	C/J	No
X-104H	FMCRD Pos. Indicator	450	S	C/J	No
X-105A	Neutron Detection	450	S	C/J	No
X-105B	Neutron Detection	450	S	C/J	No
X-105C	Neutron Indicator	450	S	C/J	No
X-105D	Neutron Indicator	450	S	C/J	No
X-106A	Div I Instrumentation	300	S	C/J	No
X-106B	Div II Instrumentation	300	S	C/J	No
X-106C	Div III Instrumentation	300	S	C/J	No
X-106D	Div IV Instrumentation	300	S	C/J	No
X-106F	Div IV Instrumentation	300	S	C/J	No
X-106G	Div IV Instrumentation	300	S	C/J	No

Table 6.2-10 Potential Bypass Leakage Paths* (Continued)

Penetration Number	Name	Diameter (mm)	Termination Region †	Leakage Barriers‡	Potential Bypass Path
X-106H	Div IV Instrumentation	300	S	C/J	No
X-106J	Div IV Instrumentation	300	S	C/J	No
X-107A	Group B Instr	300	S	C/J	No
X-107B	Power and Control	300	S	C/J	No
X-110	Spare	300	S	E/C/J	No
X-111	Spare	300	P	B/A	No
X-112	Spare	300	P	B/A	No
X-130A	I & C	300	S	C/J	No
X-130B	I & C	300	S	C/J	No
X-130C	I & C	300	S	C/J	No
X-130D	I & C	300	S	C/J	No
X-140A	I & C	250	S	C/J	No
X-140B	I & C	300	S	C/J	No
X-141A	I & C	300	S	C/J	No
X-141B	I & C	300	S	C/J	No
X-142A	I & C	90	S	C/J	No
X-142B	I & C	90	S	C/J	No
X-142C	I & C	90	S	C/J	No
X-142D	I & C	90	S	C/J	No
X-143A	I & C	90	S	C/J	No
X-143B	I & C	90	S	C/J	No
X-143C	I & C	90	S	C/J	No
X-143D	I & C	90	S	C/J	No
X-144A	I & C	90	S	C/J	No
X-144B	I & C	90	S	C/J	No
X-144C	I & C	90	S	C/J	No
X-144D	I & C	90	S	C/J	No
X-146A	I & C	300	S	C/J	No
X-146B	I&C	300	S	C/J	No
X-146C	I&C	300	S	C/J	No

Table 6.2-10 Potential Bypass Leakage Paths* (Continued)

Penetration Number	Name	Diameter (mm)	Termination Region †	Leakage Barriers‡	Potential Bypass Path
X-146D	I&C	300	S	C/J	No
X-147	I&C	90	S	C/J	No
X-160	LDS Monitor (drywell sample line)	250	S	E/C/H	No
X-161A	CAMS I&C	250	S	C/J	No
X-161B	CAMS I&C	250	S	C/J	No
X-162A	Sample/Return Drywell Gas	250	S	C/J	No
X-162B	Sample/Return Drywell Gas	250	S	C/J	No
X-170	I&C	200	S	C/J	No
X-171	I&C	250	S	C/J	No
X-177	I&C	250	S	C/J	No
X-200B	Wetwell Spray	100	S	C/H	No
X-200C	Wetwell Spray	100	S	C/H	No
X-201	RHR Pump Suction (A)	450	S	C/H	No
X-202	RHR Pump Suction (B)	450	S	C/H	No
X-203	RHR Pump Suction (C)	450	S	C/H	No
X-204	RHR Pump Test (A)	250	S	C/H	No
X-205	RHR Pump Test (B)	250	S	C/H	No
X-206	RHR Pump Test (C)	250	S	C/H	No
X-210	HPCF Pump Suction (B)	400	S	C/H	No
X-211	HPCF Pump Suction (C)	400	S	C/H	No
X-213	RCIC Turbine Exhaust	550	S	C/G	No
X-214	RCIC Pump Suction	200	S	C/H	No
X-216	SPCU Pump Suction	200	S	C/H	No
X-217	SPCU Pump Return	250	S	C/H	No
X-240	Wetwell Purge Suction	550	E	E/C/J	Yes
X-241	Wetwell Purge Exhaust	550	E	E/C/J	Yes

Table 6.2-10 Potential Bypass Leakage Paths* (Continued)

Penetration Number	Name	Diameter (mm)	Termination Region †	Leakage Barriers‡	Potential Bypass Path
X-242	Spare	150	S	E/C/J	No
X-250	Spare		P	B/A	No
X-251	Spare		P	B/A	No
X-252	Spare	300	S	E/C/J	No
X-254	Spare	300	S	B/A	No
X-320	I&C	90	S	C/J	No
X-321B	I&C	300	S	C/J	No
X-322A	I&C	90	S	C/J	No
X-322B	I&C	90	S	C/J	No
X-322C	I&C	90	S	C/J	No
X-322D	I&C	90	S	C/J	No
X-322E	I&C	90	S	C/J	No
X-322F	I&C	90	S	C/U	No
X-323A	I&C	90	S	C/J	No
X-323B	I&C	90	S	C/J	No
X-323C	I&C	90	S	C/J	No
X-323D	I&C	90	S	C/J	No
X-323E	I&C	90	S	C/J	No
X-323F	I&C	90	S	C/J	No
X-331A	CAMS Gamma Det.	250	S	C/J	No
X-331B	CAMS Gamma Det.	250	S	C/J	No
X-332A	CAMS Sampling Ret.	300	S	C/J	No
X-332B	CAMS Sampling Ret.	300	S	C/J	No
X-342	I&C	90	S	C/J	No
X-610	CRD Insertion (Total 103)	32	S	C/L	No
X-620	LCW Drain	65	S	C/J	No

Table 6.2-10 Potential Bypass Leakage Paths* (Continued)

Penetration Number	Name	Diameter (mm)	Termination Region †	Leakage Barriers‡	Potential Bypass Path
X-621	HCW Drain	150	S	C/J	No
X-650A	I&C Core Diff Press.	20	S	C/J	No
X-650B	I&C Core Diff Press.	20	S	C/J	No
X-650C	I&C Core Diff Press.	20	S	C/J	No
X-650D	I&C Core Diff Press.	20	S	C/J	No
X-651A	I&C RIP Diff Press.	20	S	C/J	No
X-651B	I&C RIP Diff Press.	20	S	C/J	No
X-651C	I&C RIP Diff Press.	20	S	C/J	No
X-651D	I&C RIP Diff Press.	20	S	C/J	No
X-600A	TIP Drive	40	S	C/J	No
X-600B	TIP Drive	40	S	C/J	No
X-600C	TIP Drive	40	S	C/J	No
X-600D	TIP Drive Purge	40	S	C/K	No
X-680A	Spare	20	S	C/J	No
X-680B	Spare	20	S	C/J	No
X-700A	RIP Purge Water Supply	15	S	C/H	No
X-700B	RIP Purge Water Supply	15	S	C/H	No
X-700C	RIP Purge Water Supply	15	S	C/H	No
X-700D	RIP Purge Water Supply	15	S	C/H	No
X-700E	RIP Purge Water Supply	15	S	C/H	No
X-700F	RIP Purge Water Supply	15	S	C/H	No
X-700G	RIP Purge Water Supply	15	S	C/H	No
X-700H	RIP Purge Water Supply	15	S	C/H	No
X-700J	RIP Purge Water Supply	15	S	C/H	No
X-700K	RIP Purge Water Supply	15	S	C/H	No
X-710	CRD Insertion (Total 102)	32	S	C/L	No
X-740	Spare	100	S	B/A	No
X-750A	I&C (Core Diff Press.)	20	S	C/J	No
X-750B	I&C (Core Diff Press.)	20	S	C/J	No
X-750C	I&C (Core Diff Press.)	20	S	C/J	No
X-750D	I&C (Core Diff Press.)	20	S	C/J	No

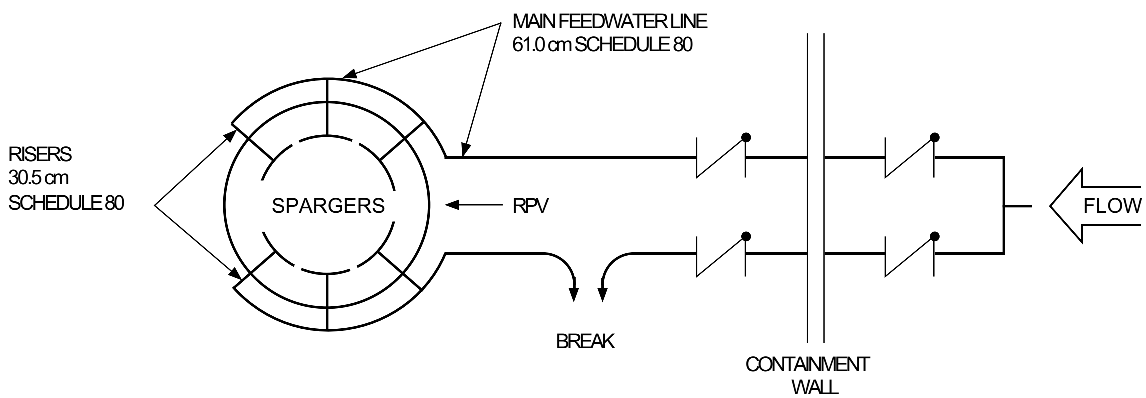
Table 6.2-10 Potential Bypass Leakage Paths* (Continued)

Penetration Number	Name	Diameter (mm)	Termination Region †	Leakage Barriers‡	Potential Bypass Path
X-751A	I&C (RIP Diff Press.)	20	S	C/J	No
X-751B	I&C (RIP Diff Press.)	20	S	C/J	No
X-751C	I&C (RIP Diff Press.)	20	S	C/J	No
X-751D	I&C (RIP Diff Press)	20	S	C/J	No
X-780A	Spare	20	S	B/A	No
X-780B	Spare	20	S	B/A	No

* This table is provided in response to Question 430.52b.

† E - Environment
P - Primary containment
S - Secondary containment

‡ A) Penetration is capped
B) Terminates at Primary Containment wall
C) Terminates inside Secondary Containment
D) Terminates outside Secondary Containment
E) Redundant containment isolation valves
F) Water seal plus third stop check valve
G) Leakage handled and accounted for alternate leakage control system (condenser)
H) Closed loop
J) SGTS - Standby gas treatment system collects and treats
K) Continuous gas pressure
L) Continuous water pressure
M) Testable double seal



FLOW AREAS:		
	MAIN FEEDWATER LINE	= 0.24 m^2
	RPV NOZZLE	= 0.12 m^2
	RISERS	= 0.07 m^2
	SPARGER OPEN AREA	= 0.03 m^2

Figure 6.2-1 A Break in a Feedwater Line

Figure 6.2-2 Not Used

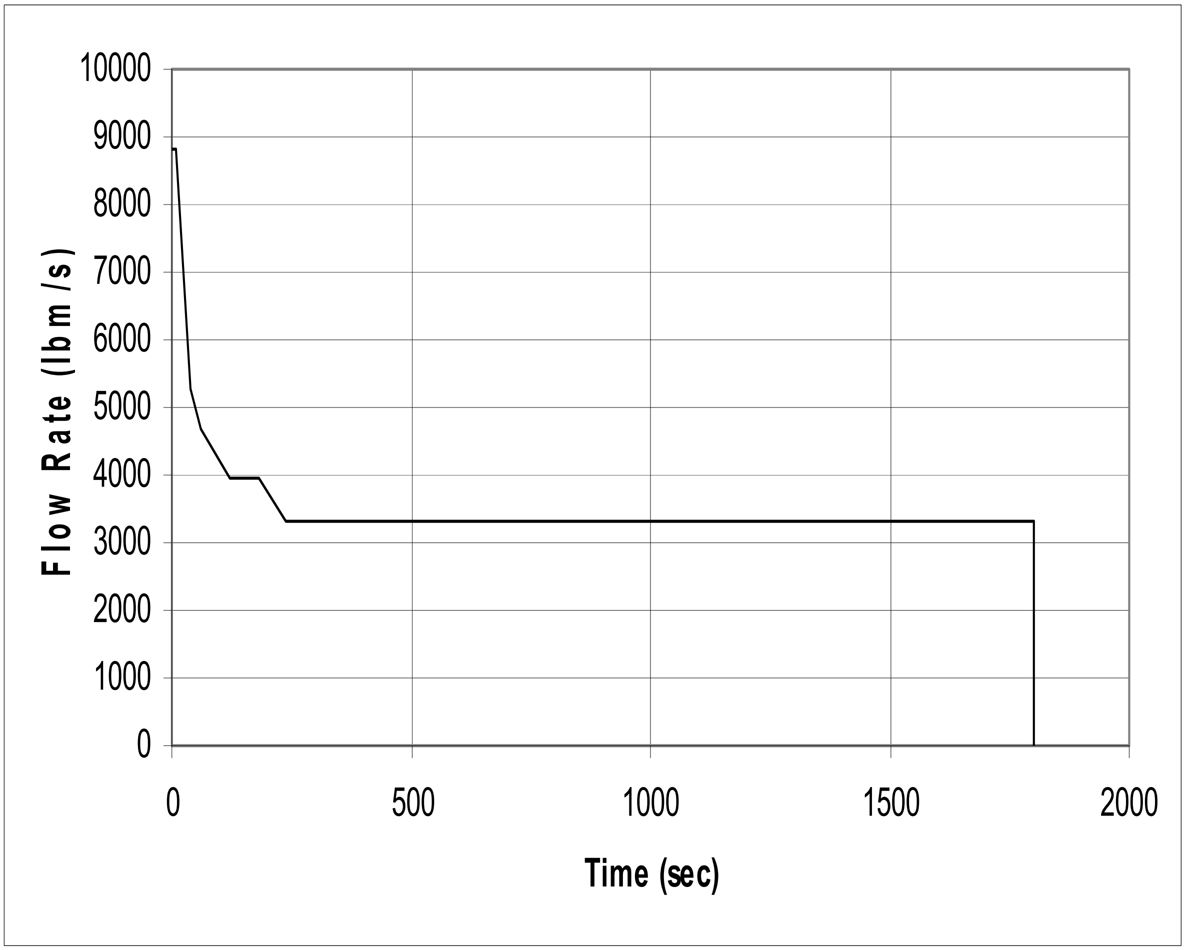


Figure 6.2-3 Feedwater Line Break Flow—Feedwater System Side of Break

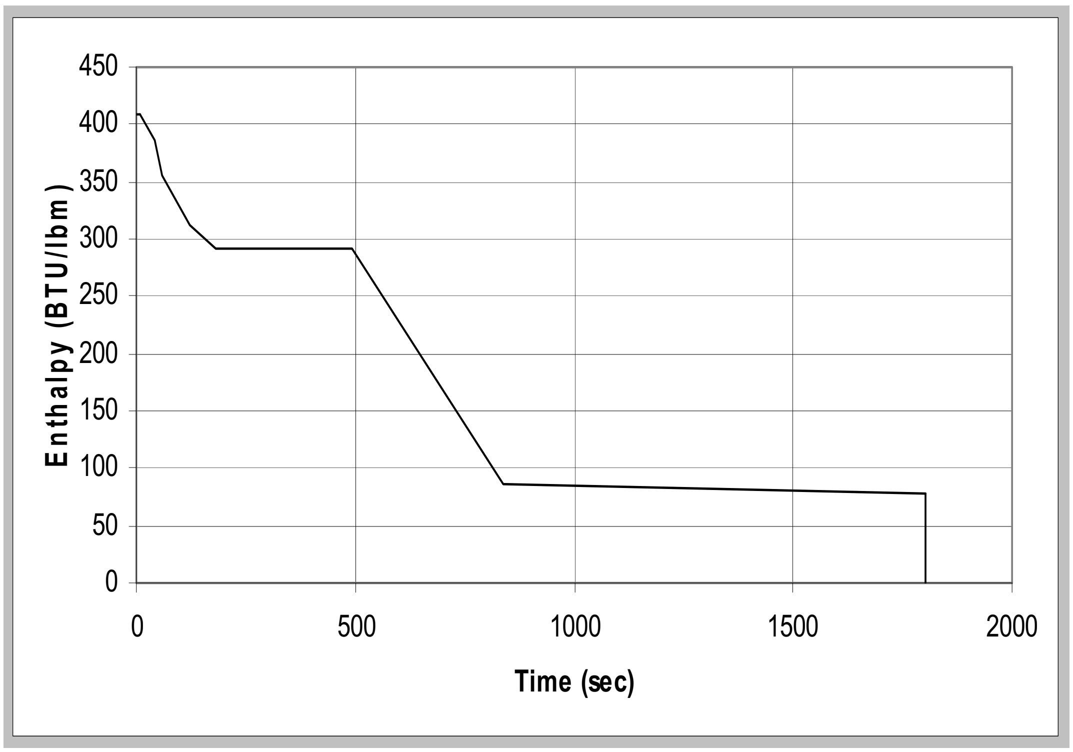


Figure 6.2-4 Feedwater Line Break Flow Enthalpy—Feedwater System Side of Break

Figure 6.2-5 Not Used

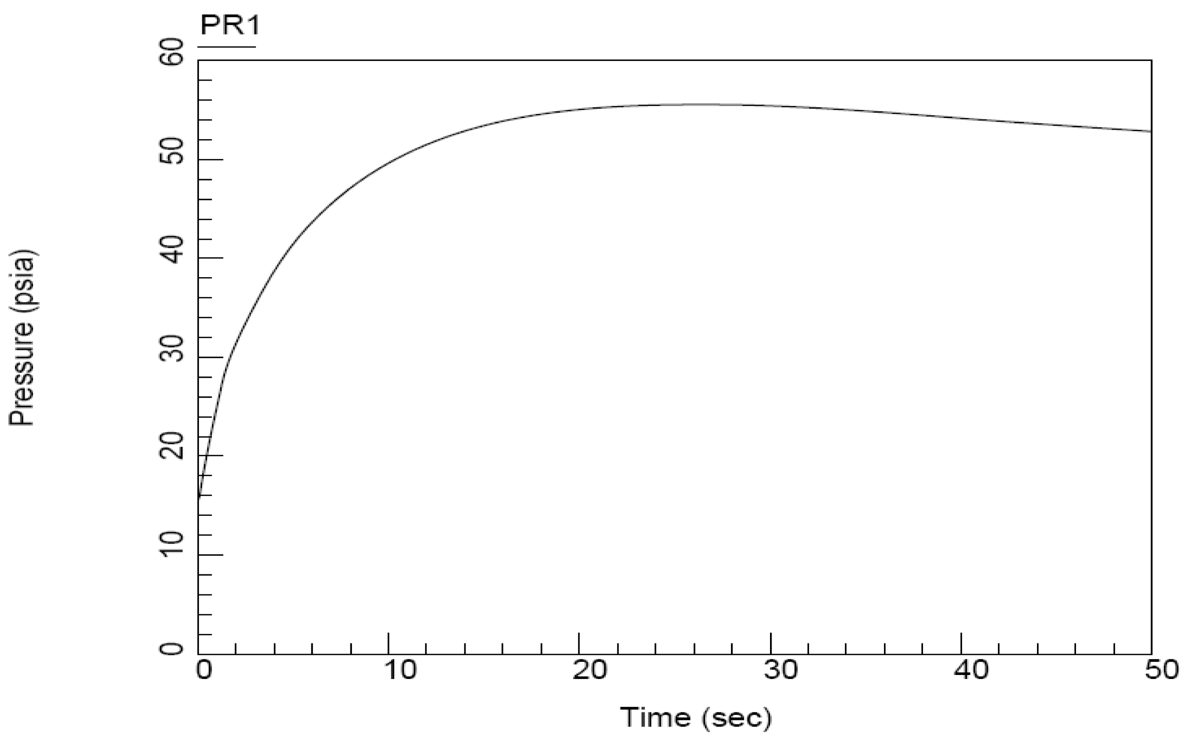


Figure 6.2-6a Drywell Pressure Response for Feedwater Line Break

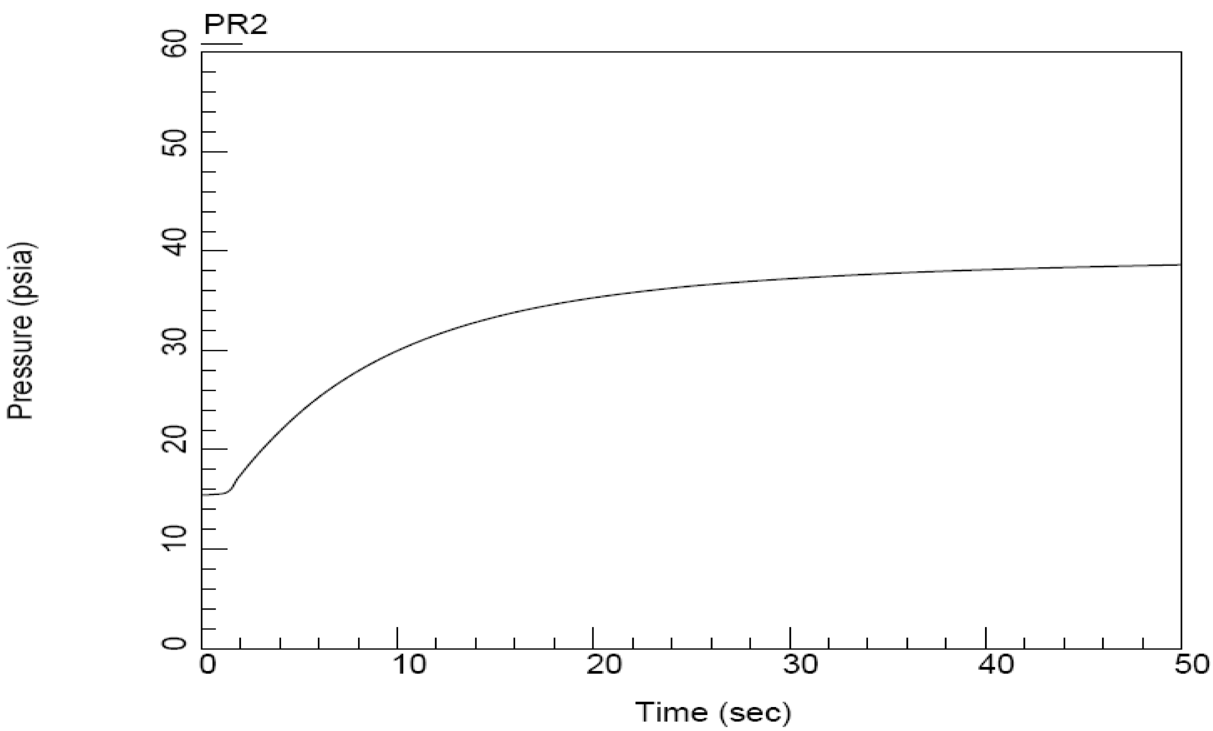


Figure 6.2-6b Wetwell Pressure Response for Feedwater Line Break

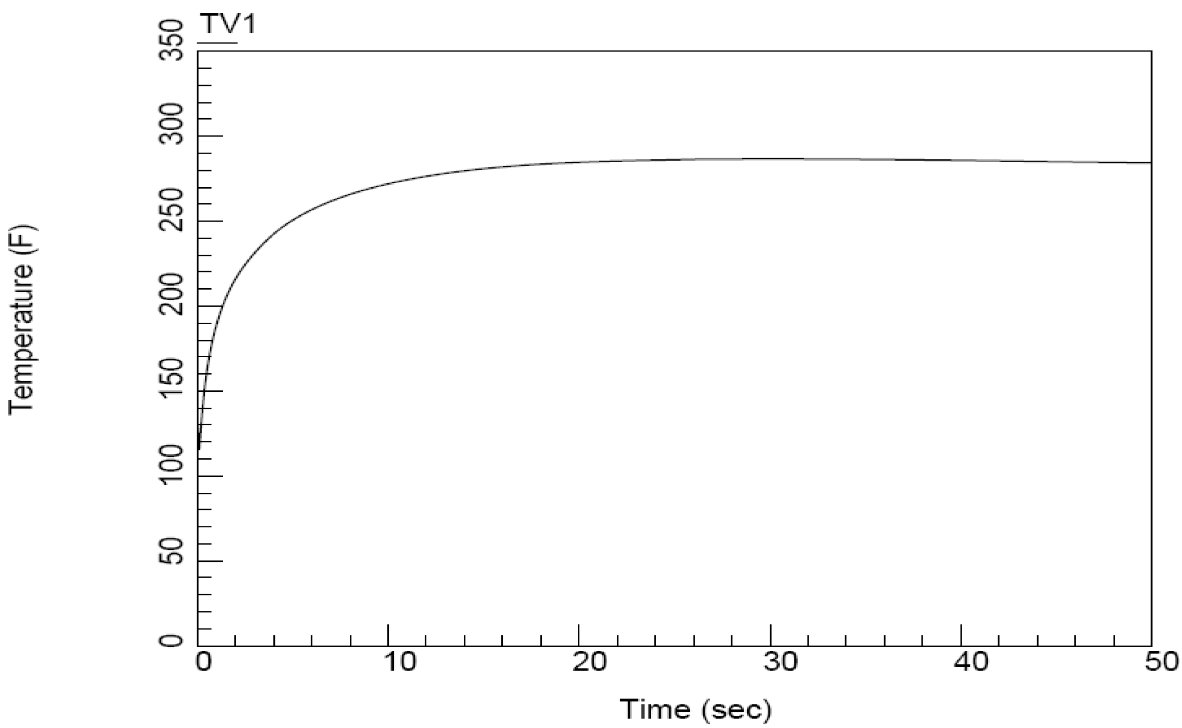


Figure 6.2-7a Temperature Response of Drywell for Feedwater Line Break

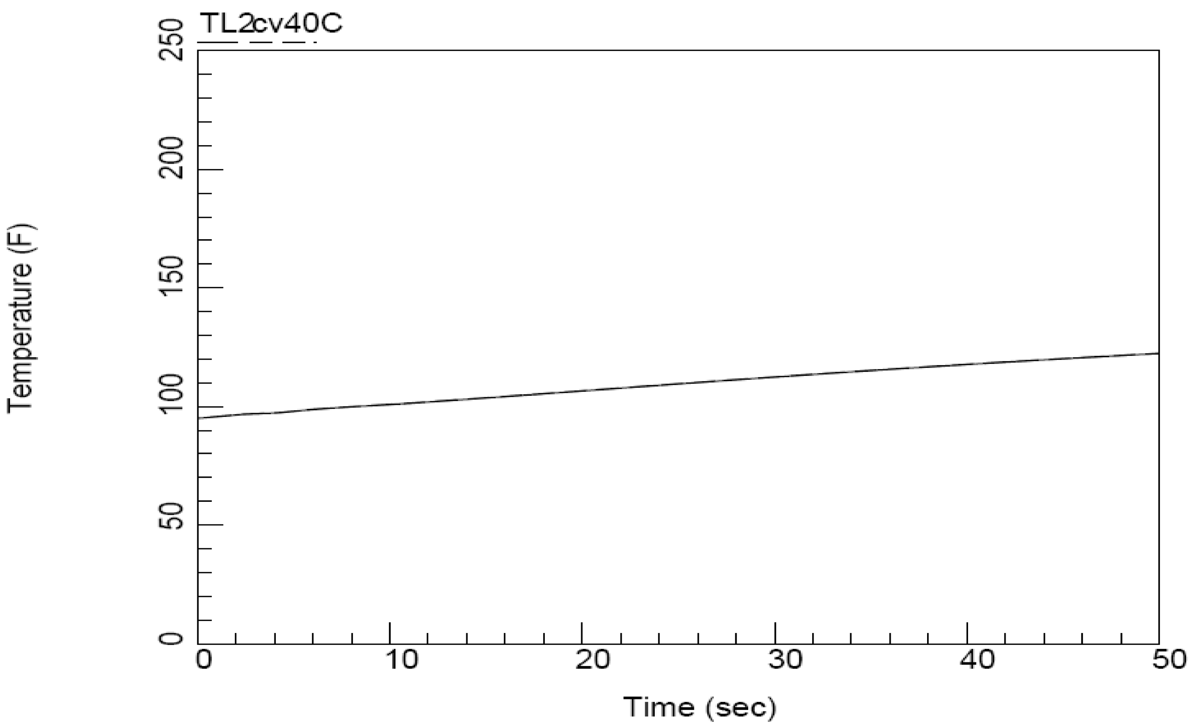


Figure 6.2-7b Temperature Response of Wetwell for Feedwater Line Break

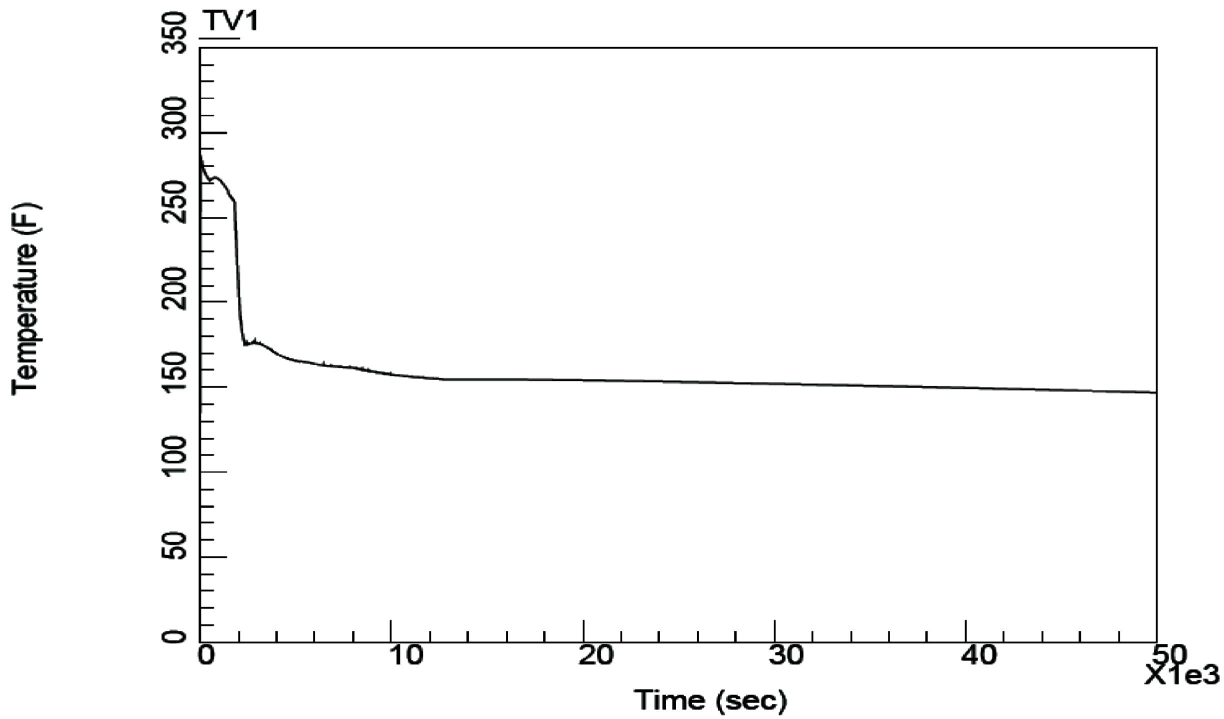


Figure 6.2-8a Drywell Temperature Time History After a Feedwater Line Break

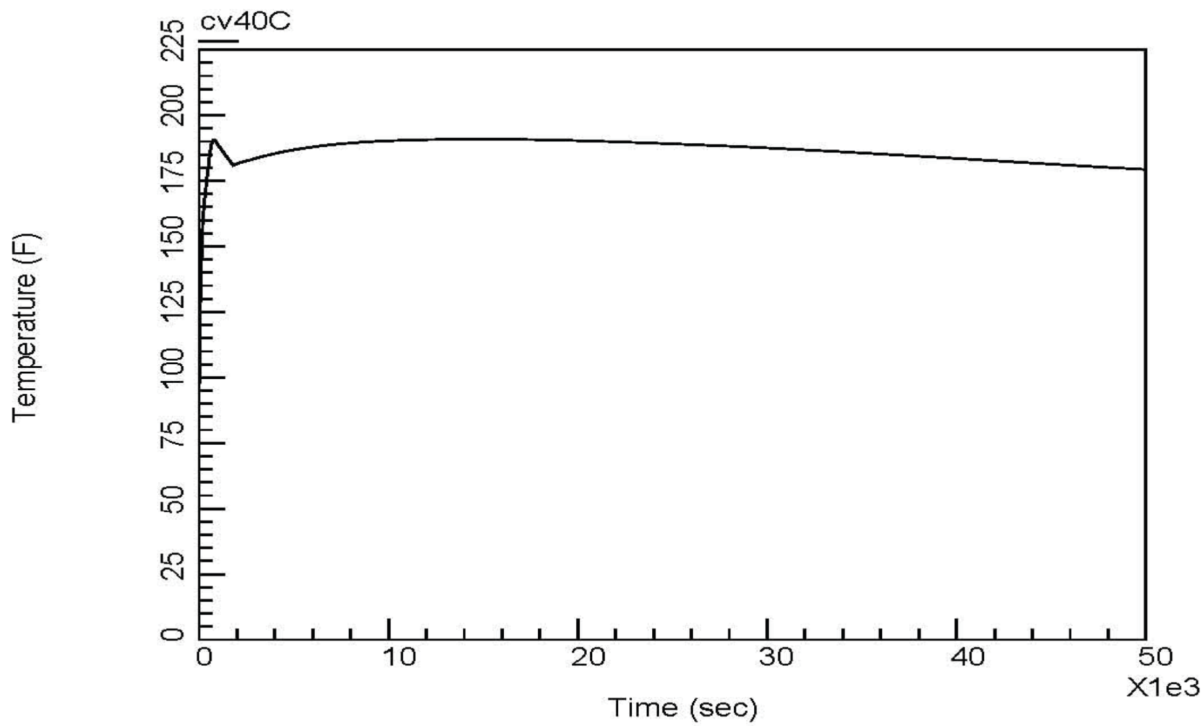


Figure 6.2-8b Suppression Pool Temperature Time History After a Feedwater Line Break

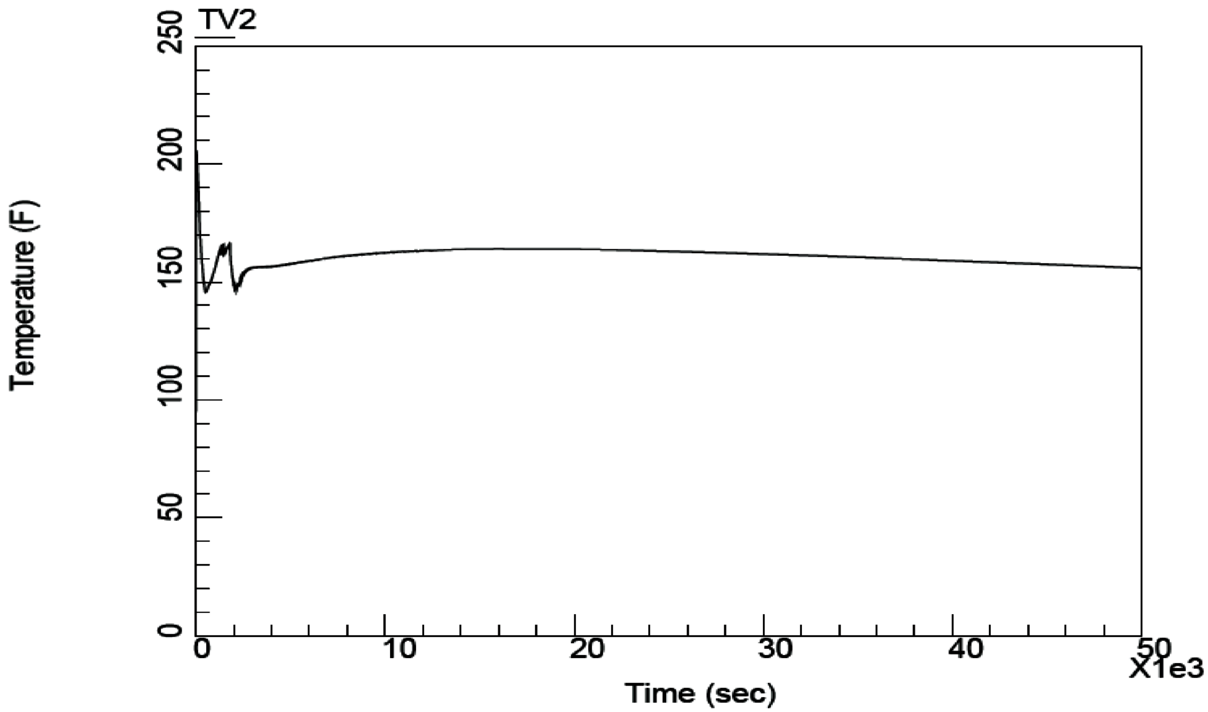


Figure 6.2-8c Wetwell Temperature Time History After a Feedwater Line Break

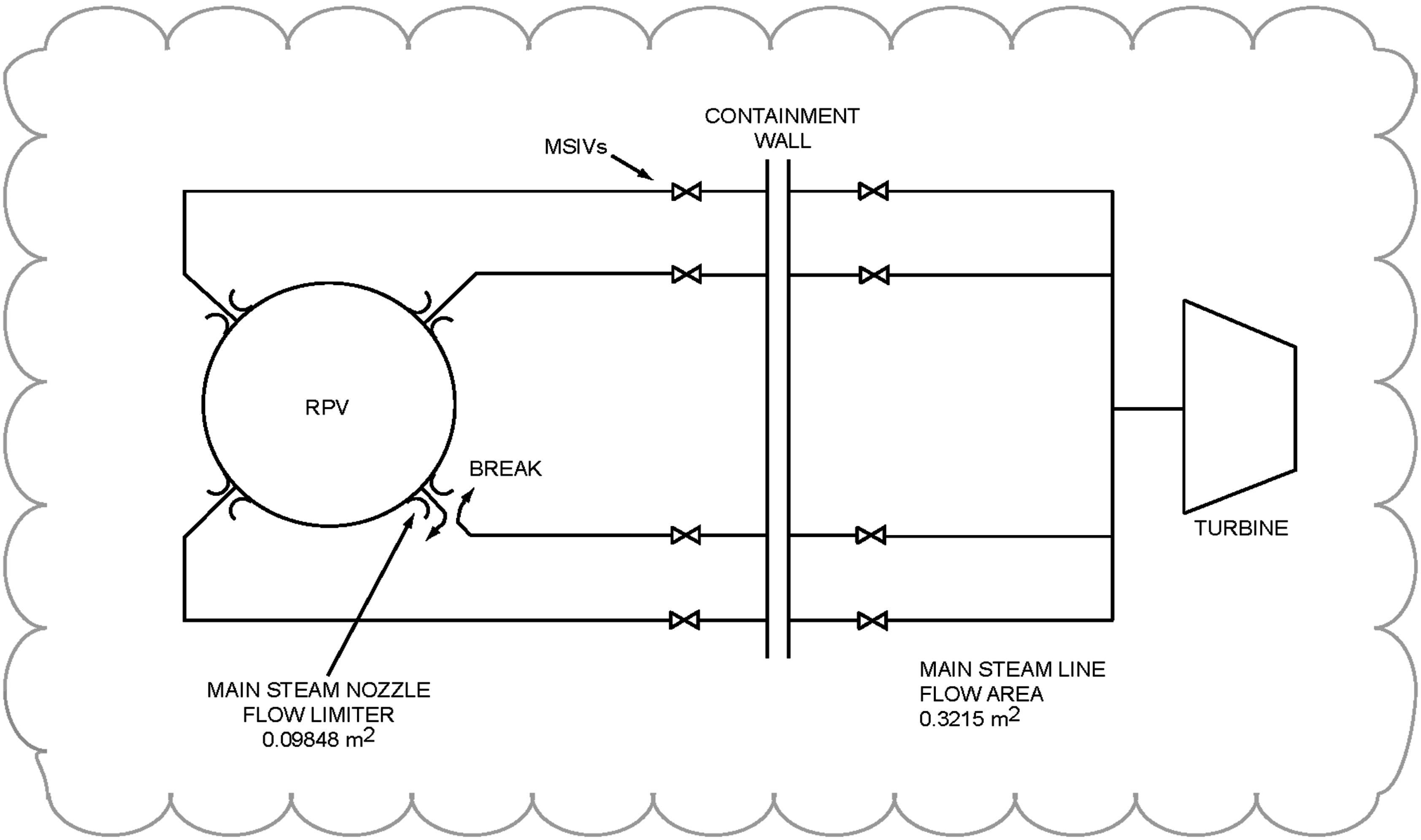


Figure 6.2-9 ABWR Main Steamlines with a Break

Figure 6.2-10 Not Used

Figure 6.2-11 Not Used

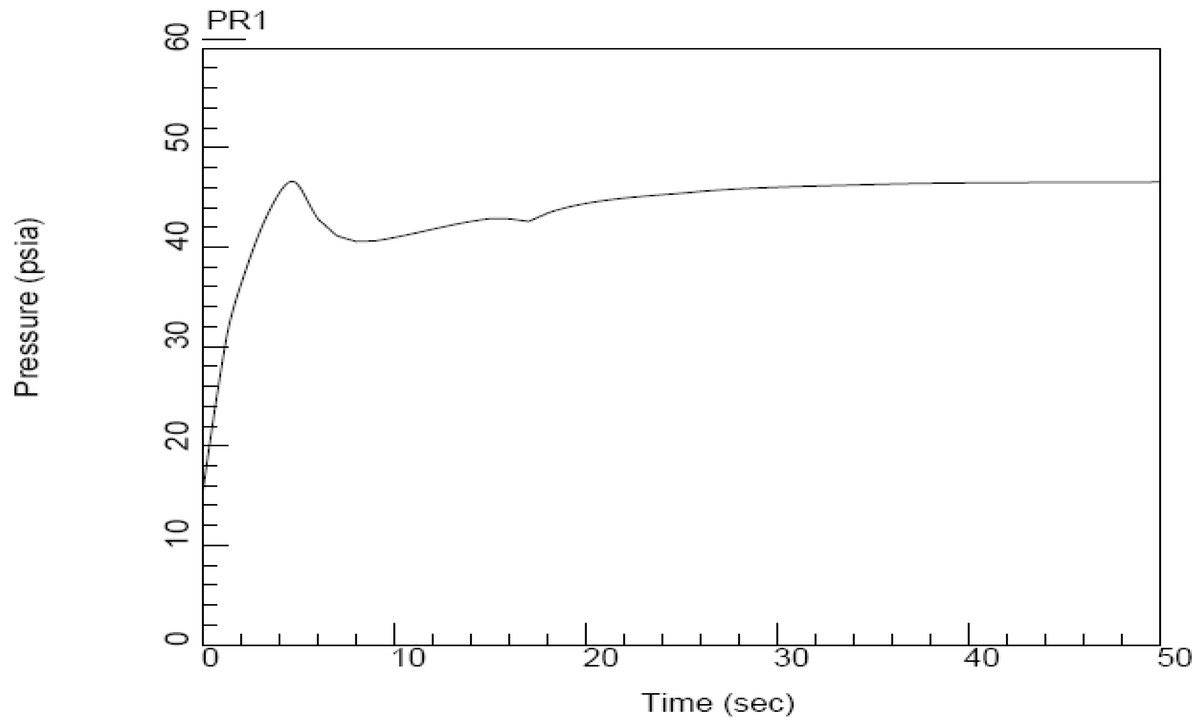


Figure 6.2-12a Drywell Pressure Time History for MSLB

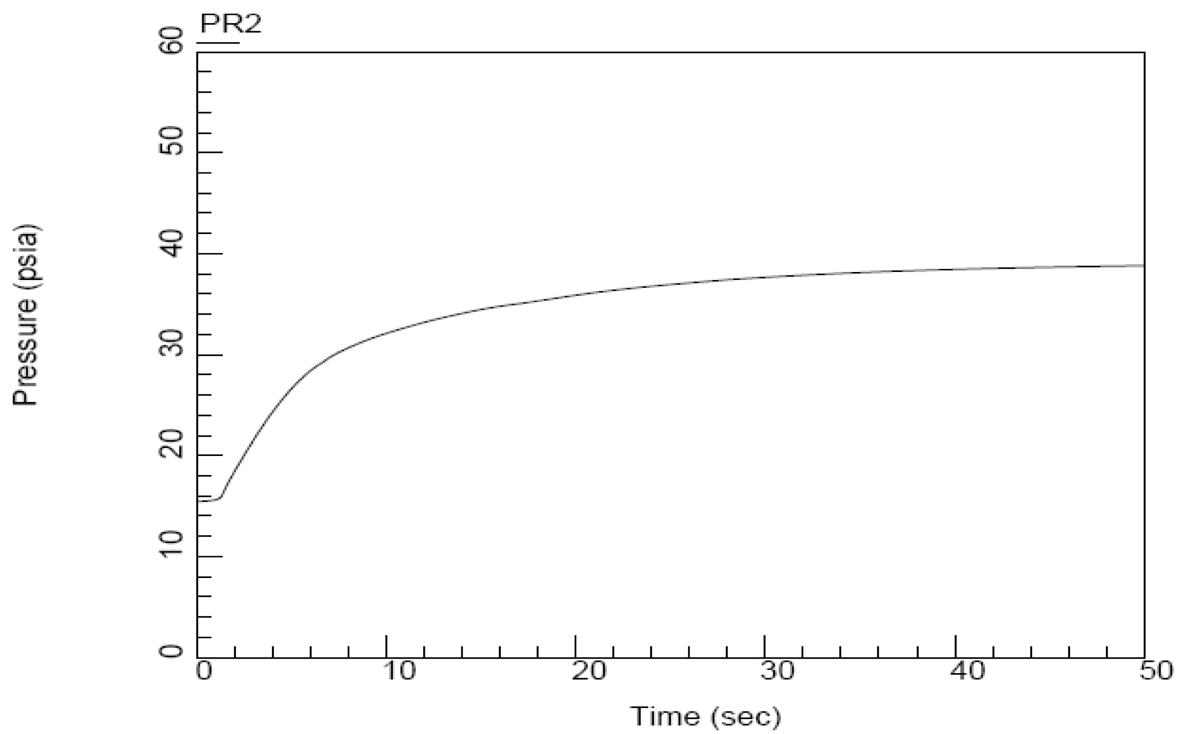


Figure 6.2-12b Wetwell Pressure Time History for MSLB

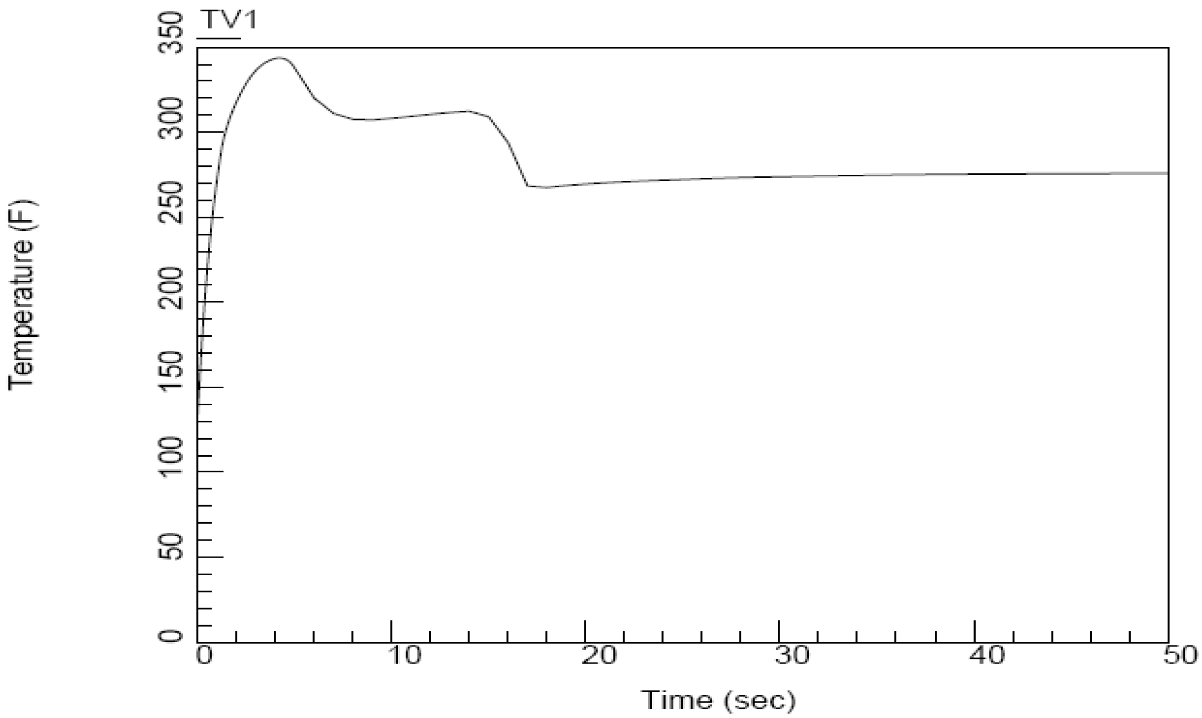


Figure 6.2-13a Drywell Temperature Time History for MSLB

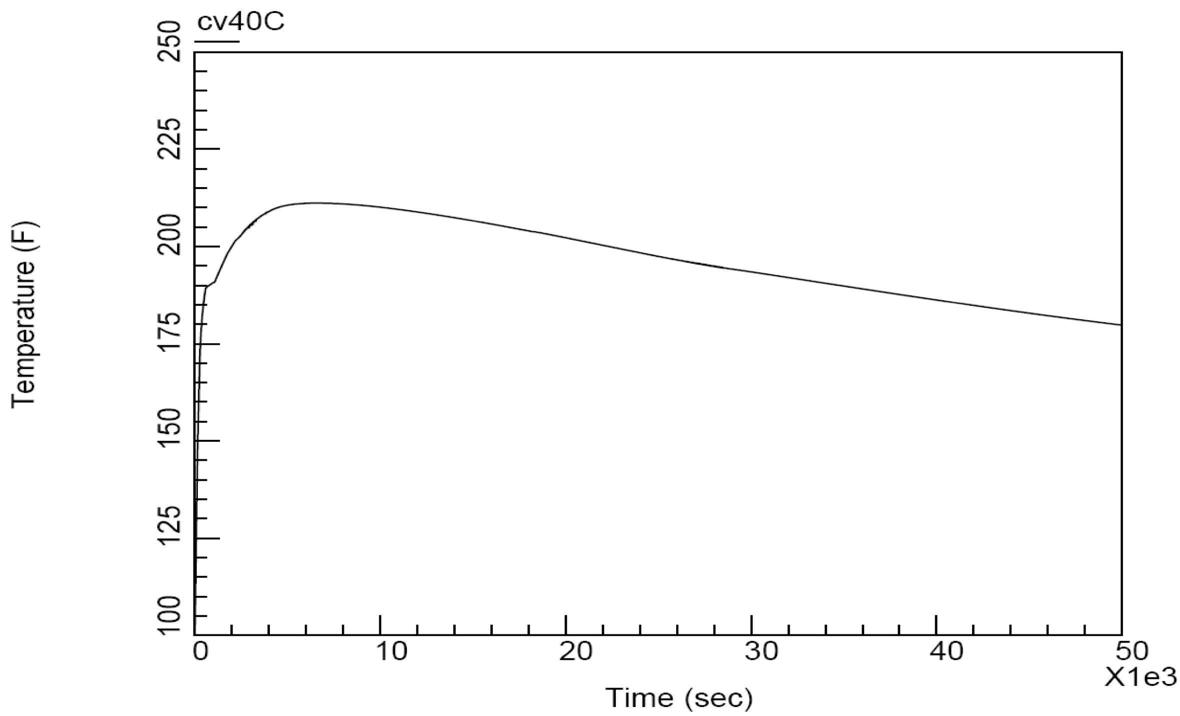


Figure 6.2-13b Suppression Pool Temperature Time History for MSLB

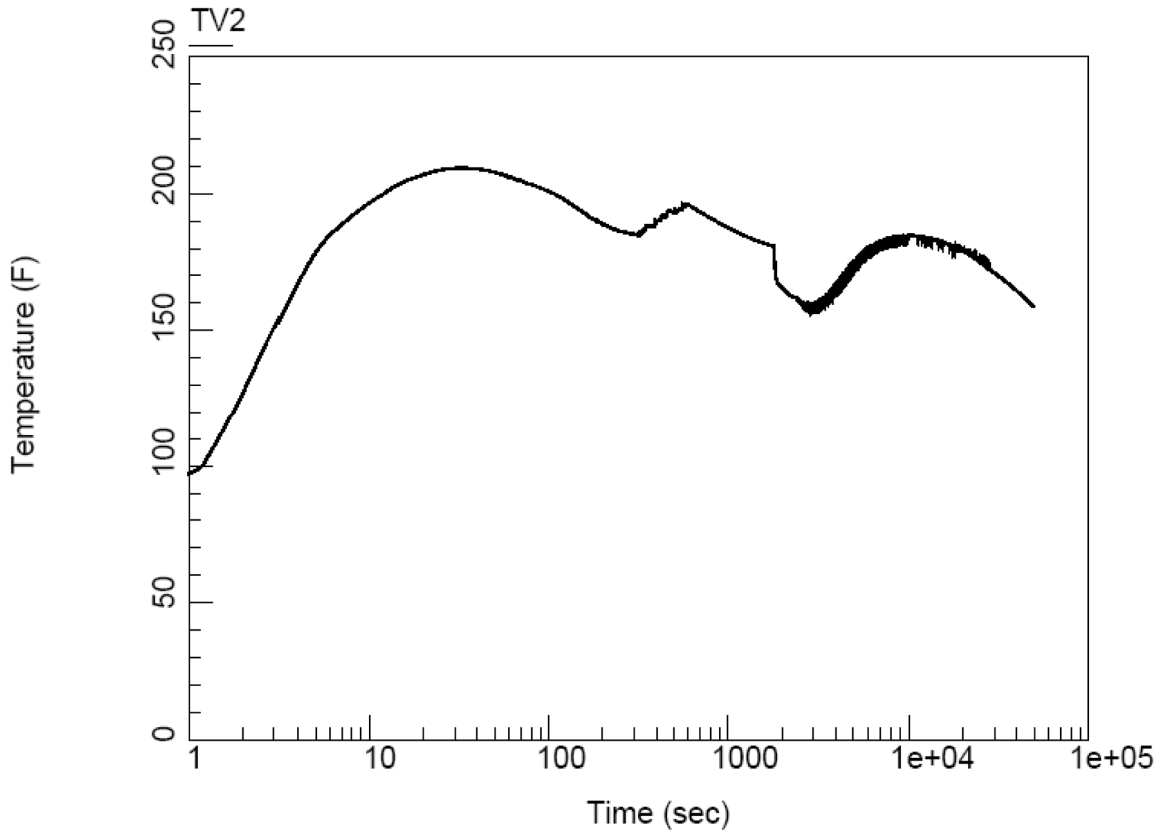


Figure 6.2-13c Wetwell Temperature Time History for MSLB

Figure 6.2-14 Not Used

Figure 6.2-15 Not Used

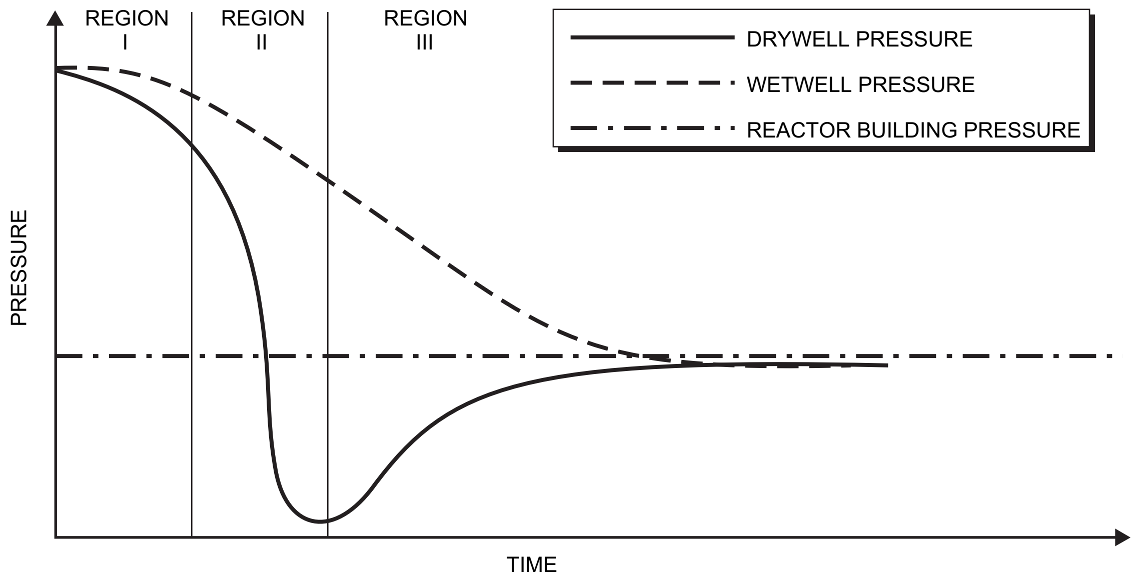


Figure 6.2-16 General Pressure Trends in the Containment During a Post-LOCA Depressurization Transient

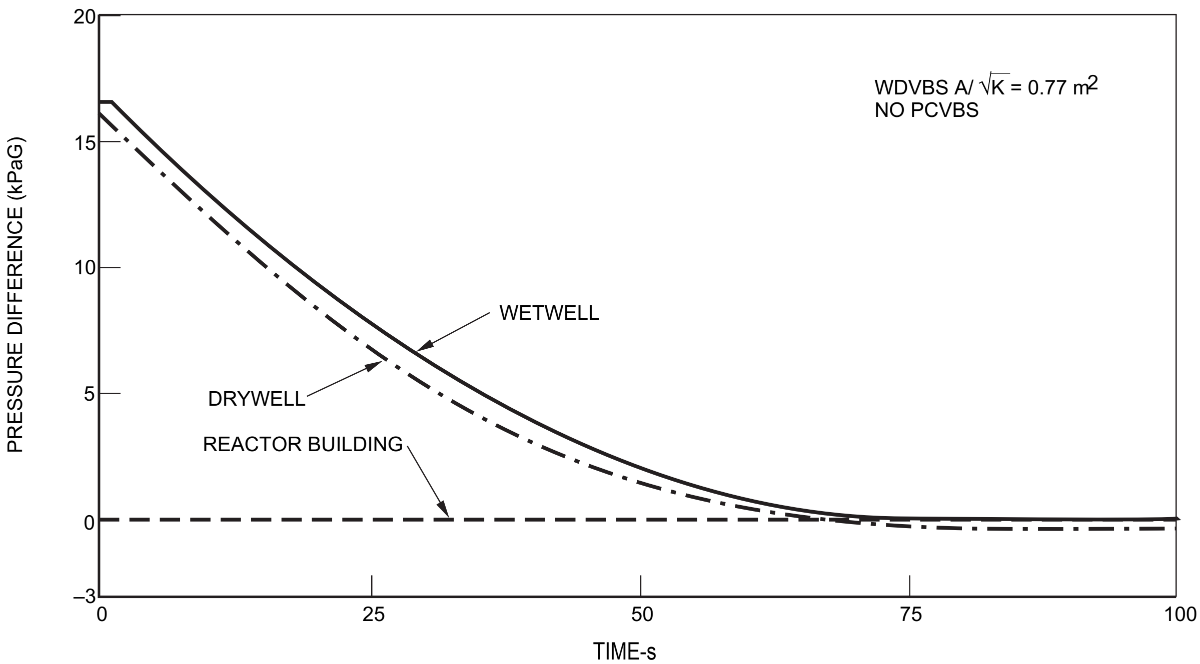


Figure 6.2-17 Differential Pressures in Wetwell and Drywell Relative to Reactor Building for Vacuum Breaker Size of .771 m^2

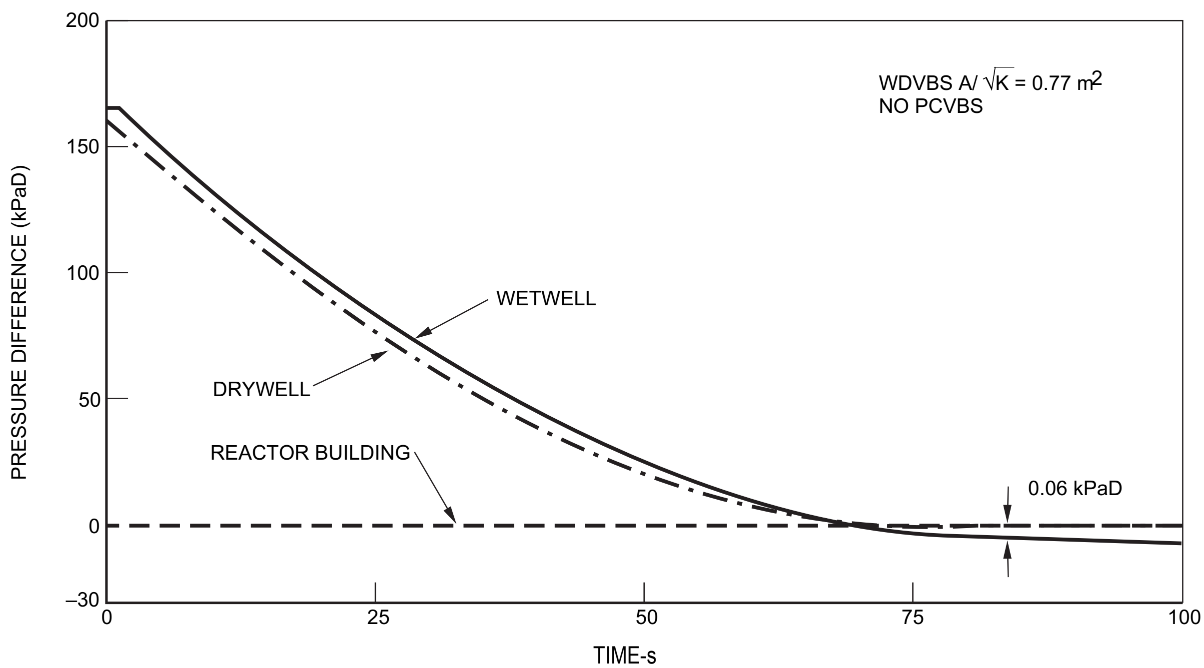


Figure 6.2-18 Differential Pressures in Wetwell and Drywell Relative to Reactor Building with Wetwell Spray for Vacuum Breaker Size of .771 m^2

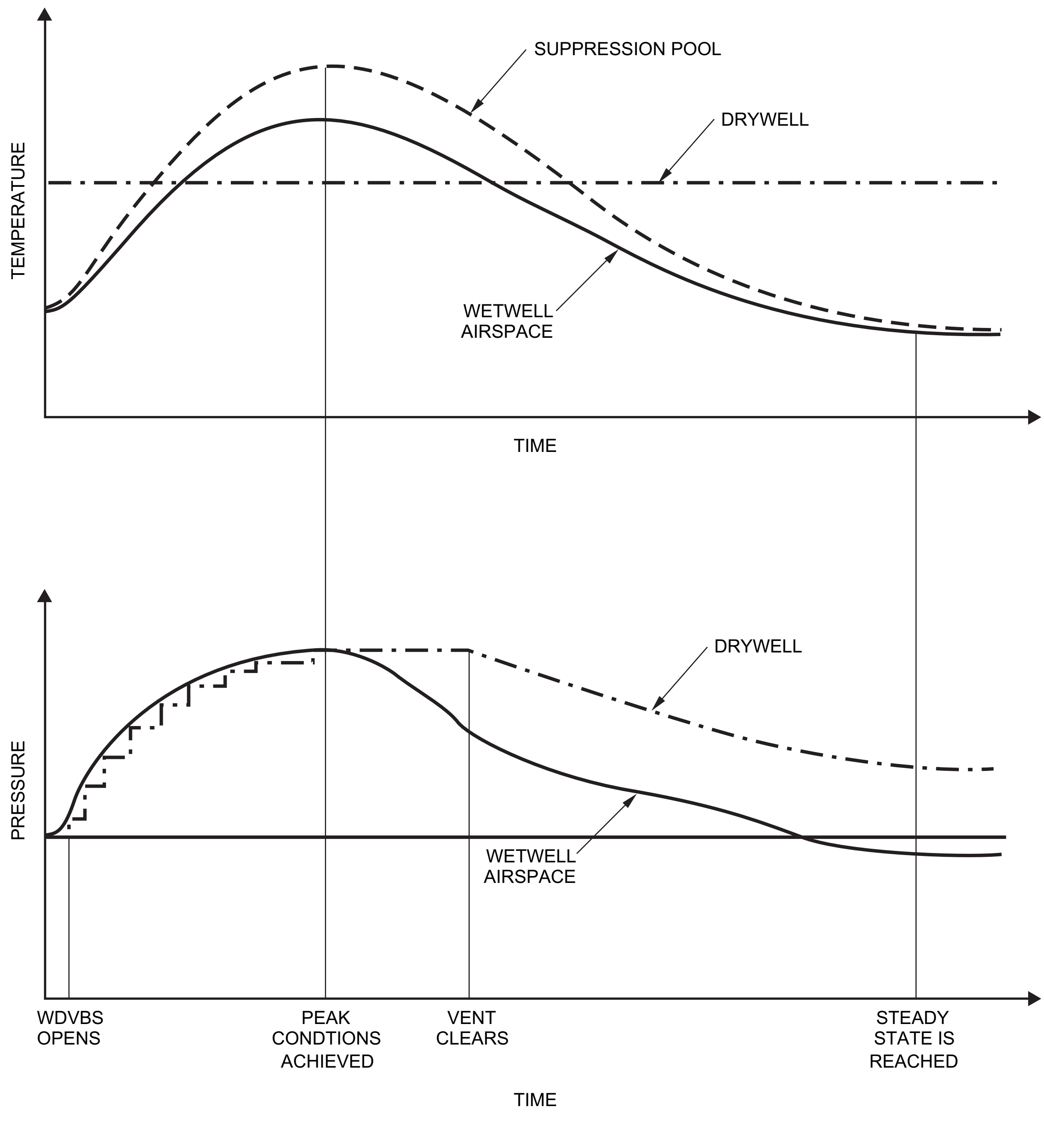


Figure 6.2-19 Temperature and Pressure Time Histories in the Containment During Stuck Open Relief Valve Transient

Figure 6.2-20 Not Used

Figure 6.2-21 Not Used

Figure 6.2-22 Not Used

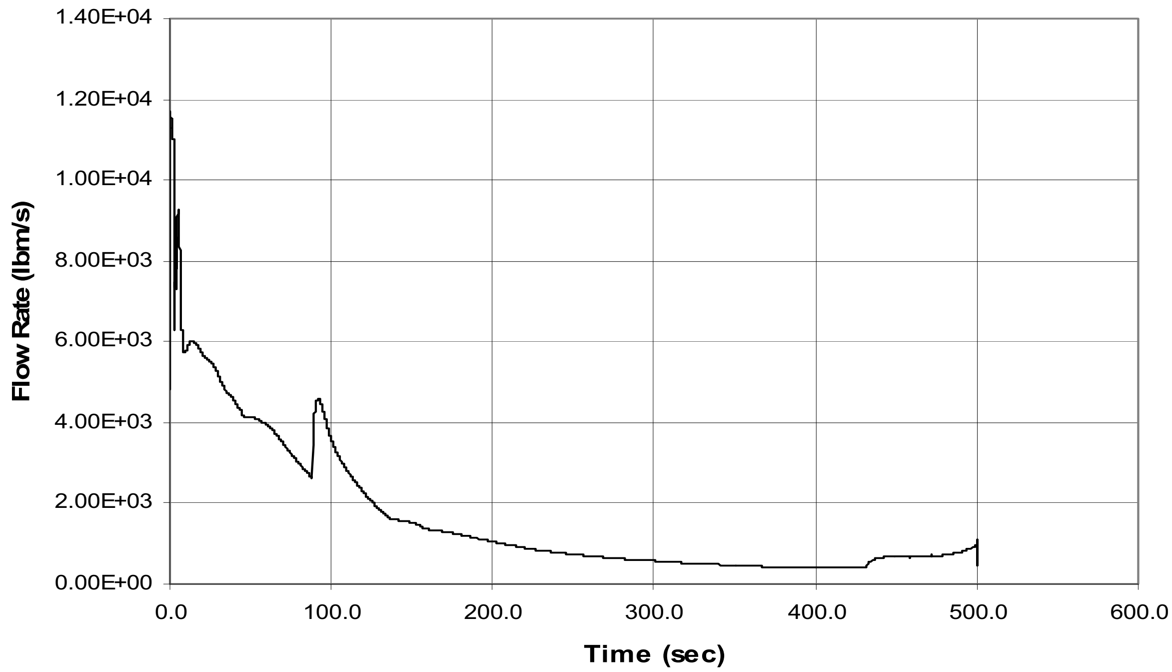


Figure 6.2-23a Break Flow Rate for the Feedwater Line Break Flow coming from the RPV Side

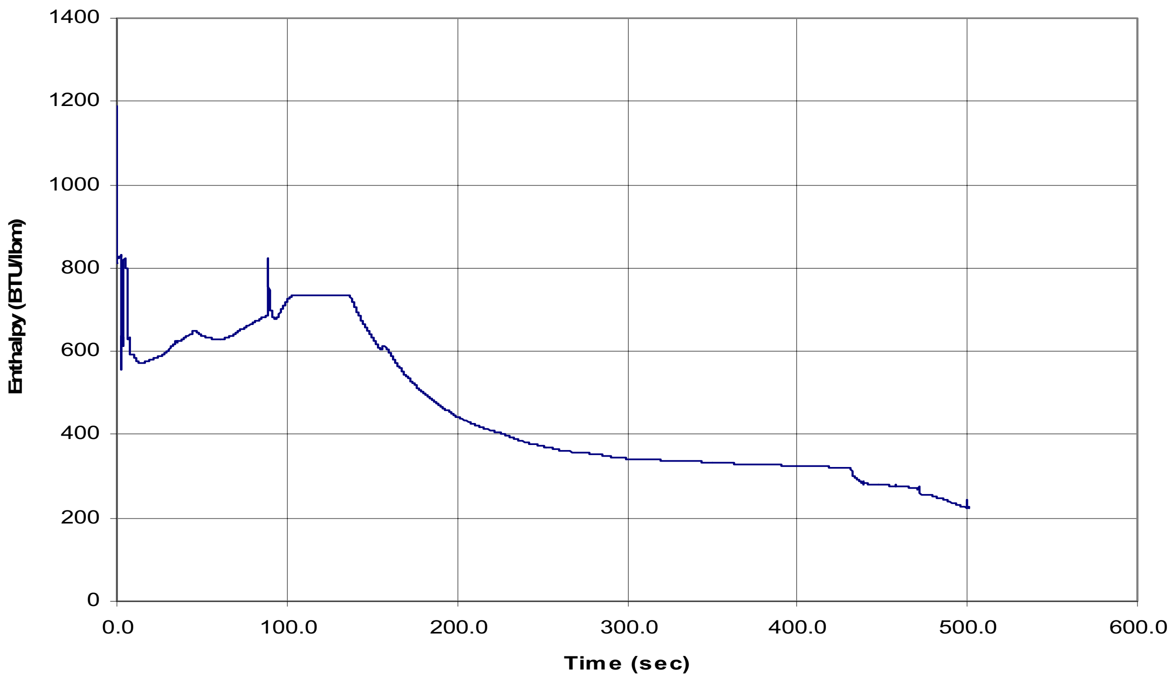


Figure 6.2-23b Break Flow Specific Enthalpy for the Feedwater Line Break Flow coming from the RPV Side

Figure 6.2-24 Not Used

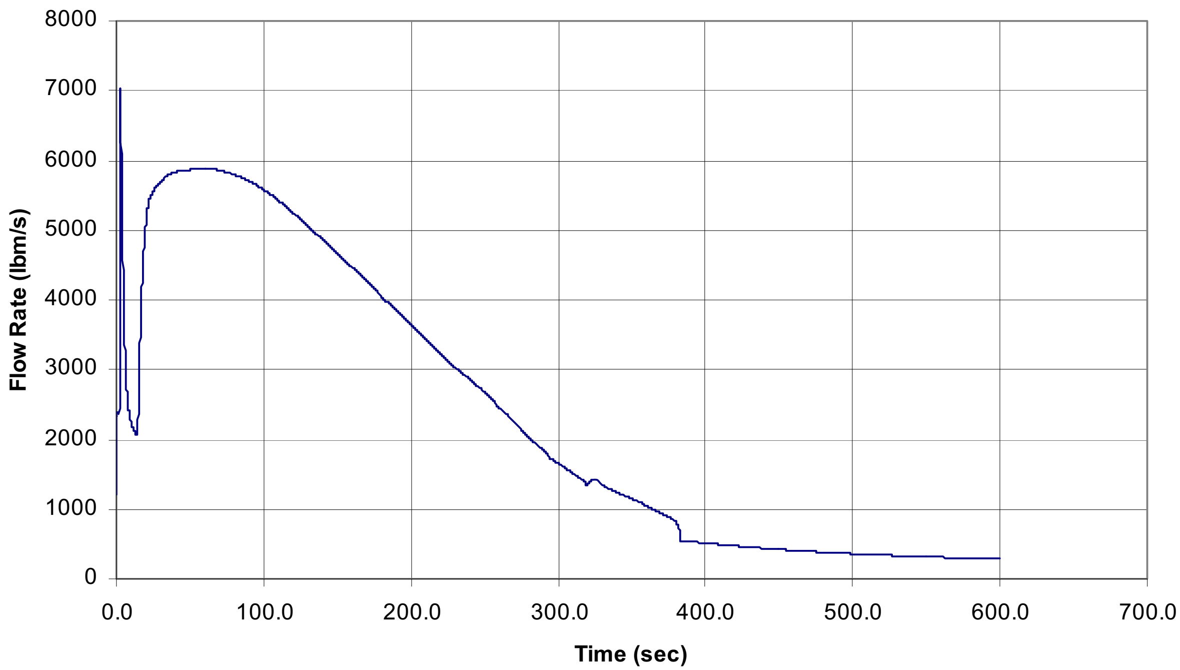


Figure 6.2-25a MSLB Short Term Break Flow (RPV Side)

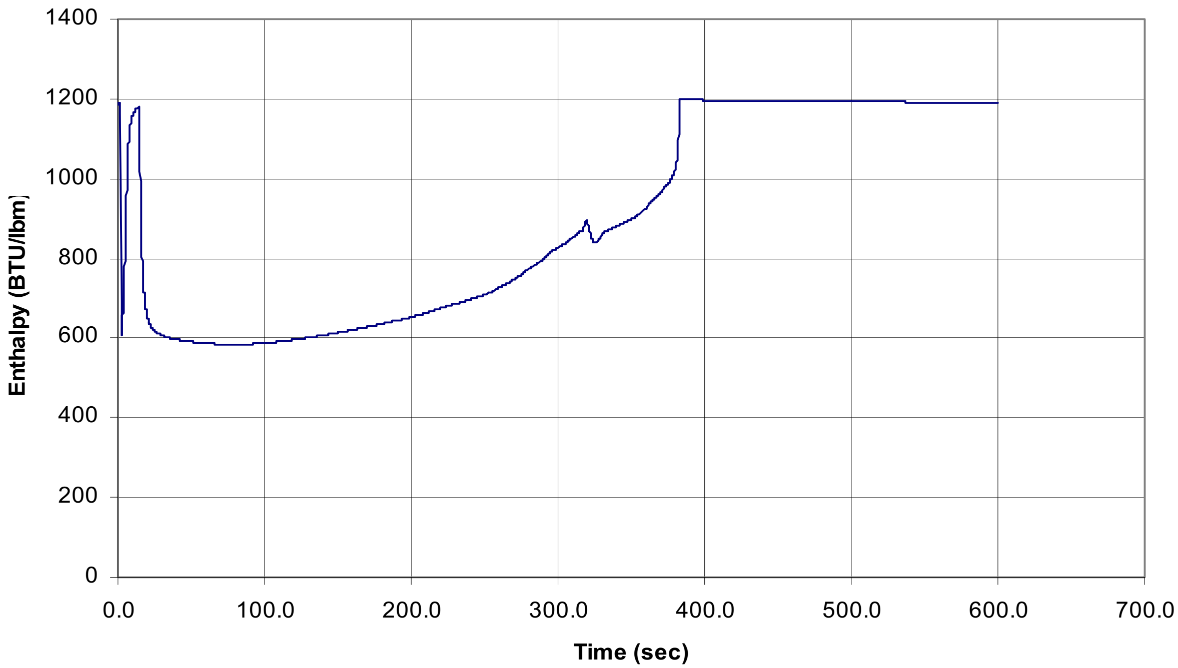


Figure 6.2-25b MSLB Short Term Break Flow (RPV Side)

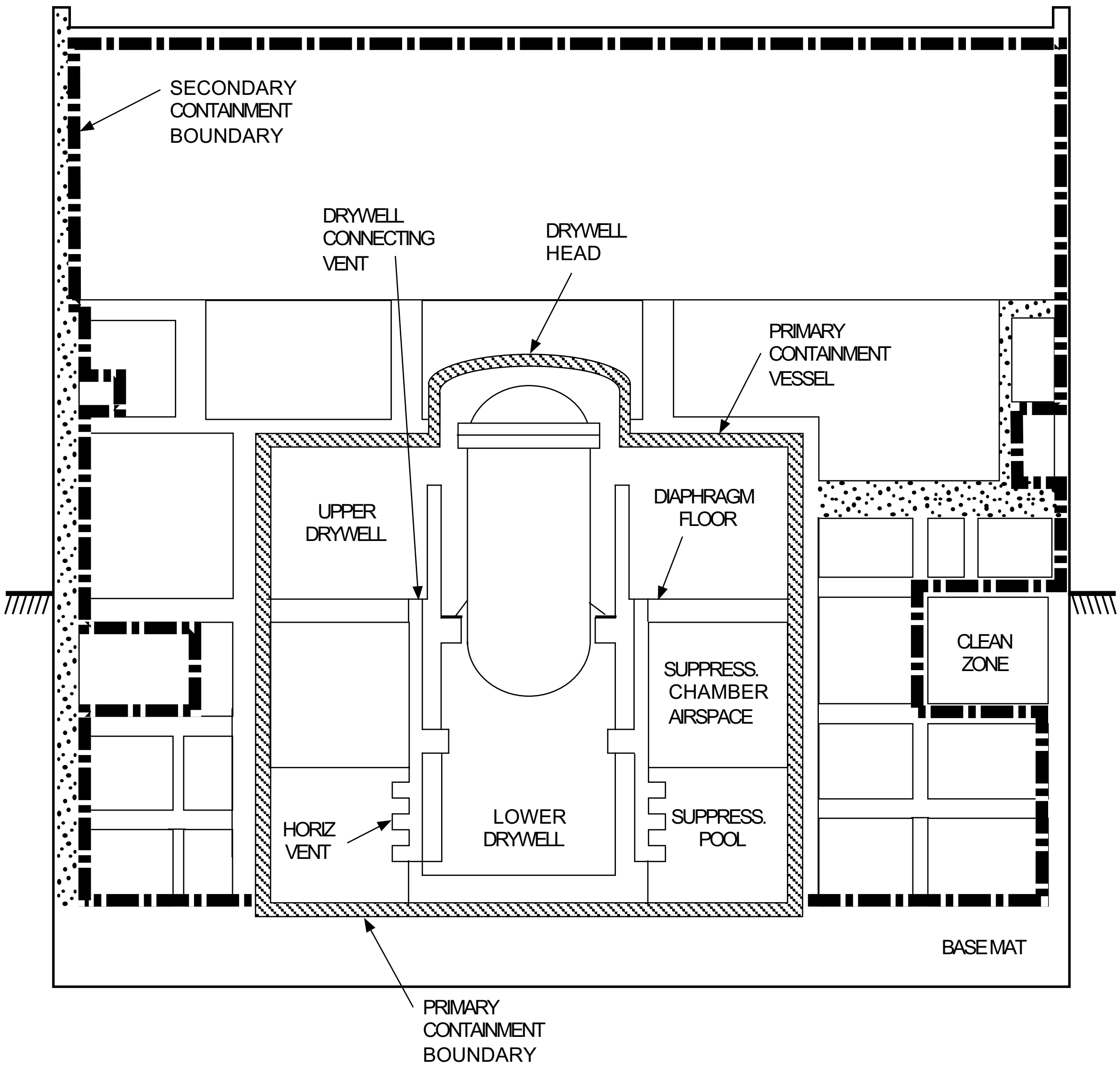
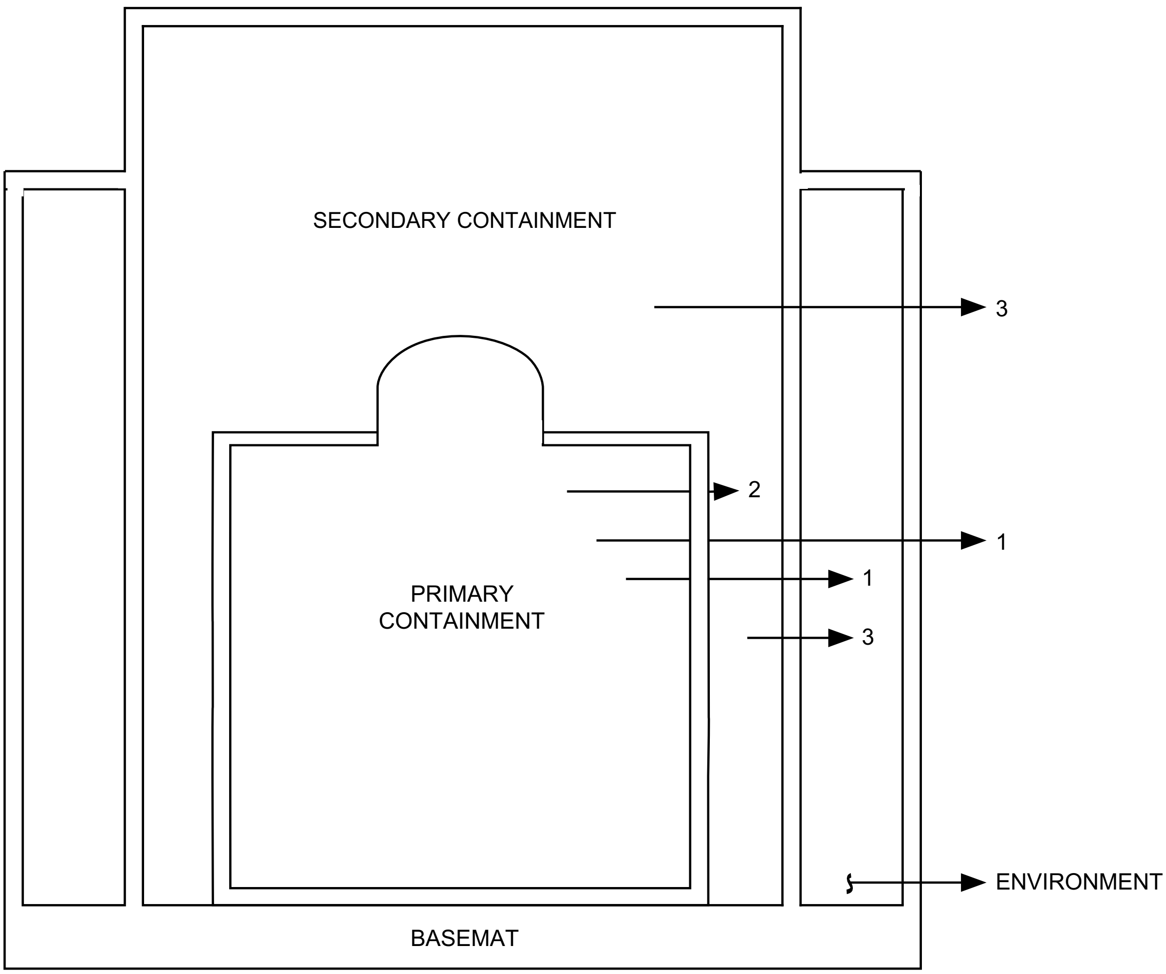


Figure 6.2-26 ABWR Containment Boundary Nomenclature



LEAKAGE FROM:

1. PRIMARY CONTAINMENT TO ENVIRONMENT OR CLEAN ZONE
2. PRIMARY CONTAINMENT TO SECONDARY CONTAINMENT
3. SECONDARY CONTAINMENT TO CLEAN ZONE OR THE ENVIRONMENT

Figure 6.2-27 Three Basic Types of Leakage Paths

		Refer to Figures :
Figure 6.2-28	Containment Boundaries in the Reactor Building—Plan Section A-A (0°–180°)	1.2-2
Figure 6.2-29	Containment Boundaries in the Reactor Building—Plan Section B-B (90°–270°)	1.2-2a
Figure 6.2-30	Containment Boundaries in the Reactor Building—Plan at Elevation –8200 mm	1.2-4
Figure 6.2-31	Containment Boundaries in the Reactor Building—Plan at Elevation –1700 mm	1.2-5
Figure 6.2-32	Containment Boundaries in the Reactor Building—Plan at Elevation 4800/8500 mm	1.2-6
Figure 6.2-33	Containment Boundaries in the Reactor Building—Plan at Elevation 12300 mm	1.2-8
Figure 6.2-34	Containment Boundaries in the Reactor Building—Plan at Elevation 18100 mm	1.2-9
Figure 6.2-35	Containment Boundaries in the Reactor Building—Plan at Elevation 23500 mm	1.2-10
Figure 6.2-36	Containment Boundaries in the Reactor Building—Plan at Elevation 31700 mm	1.2-12

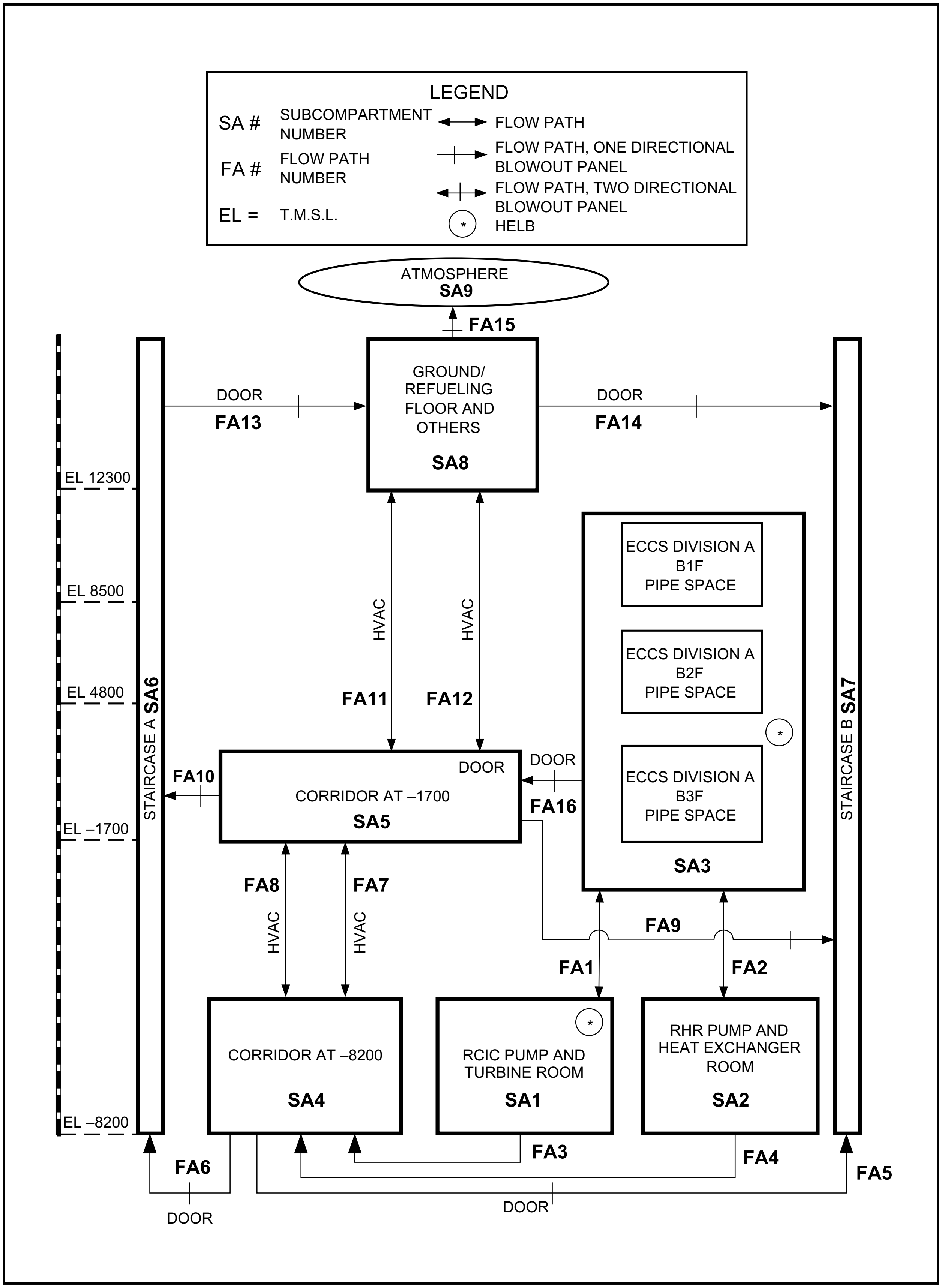


Figure 6.2-37a Secondary Containment Schematic Flow Diagram (ECCS/RCIC)

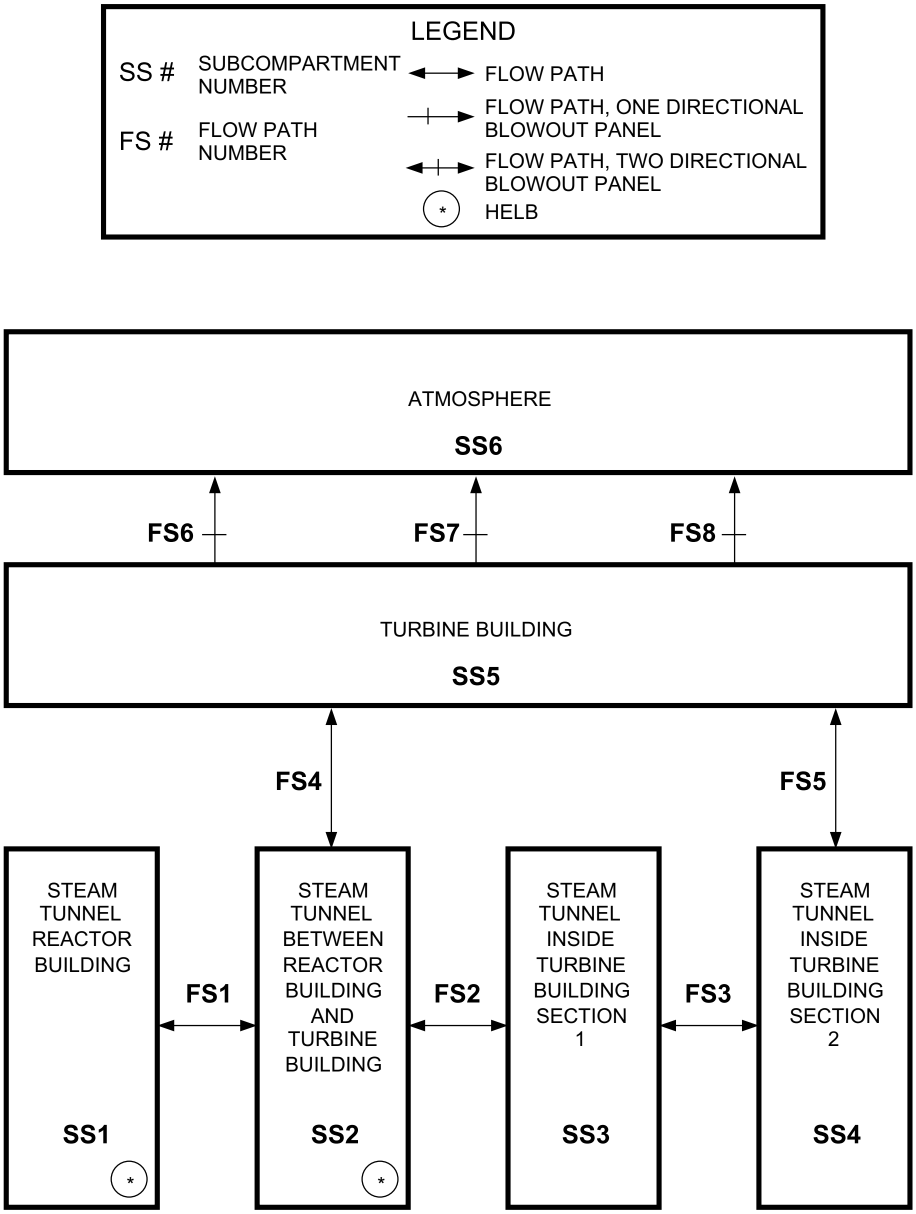


Figure 6.2-37b
Secondary Containment Schematic Flow Diagram (Main Steam/Feedwater)

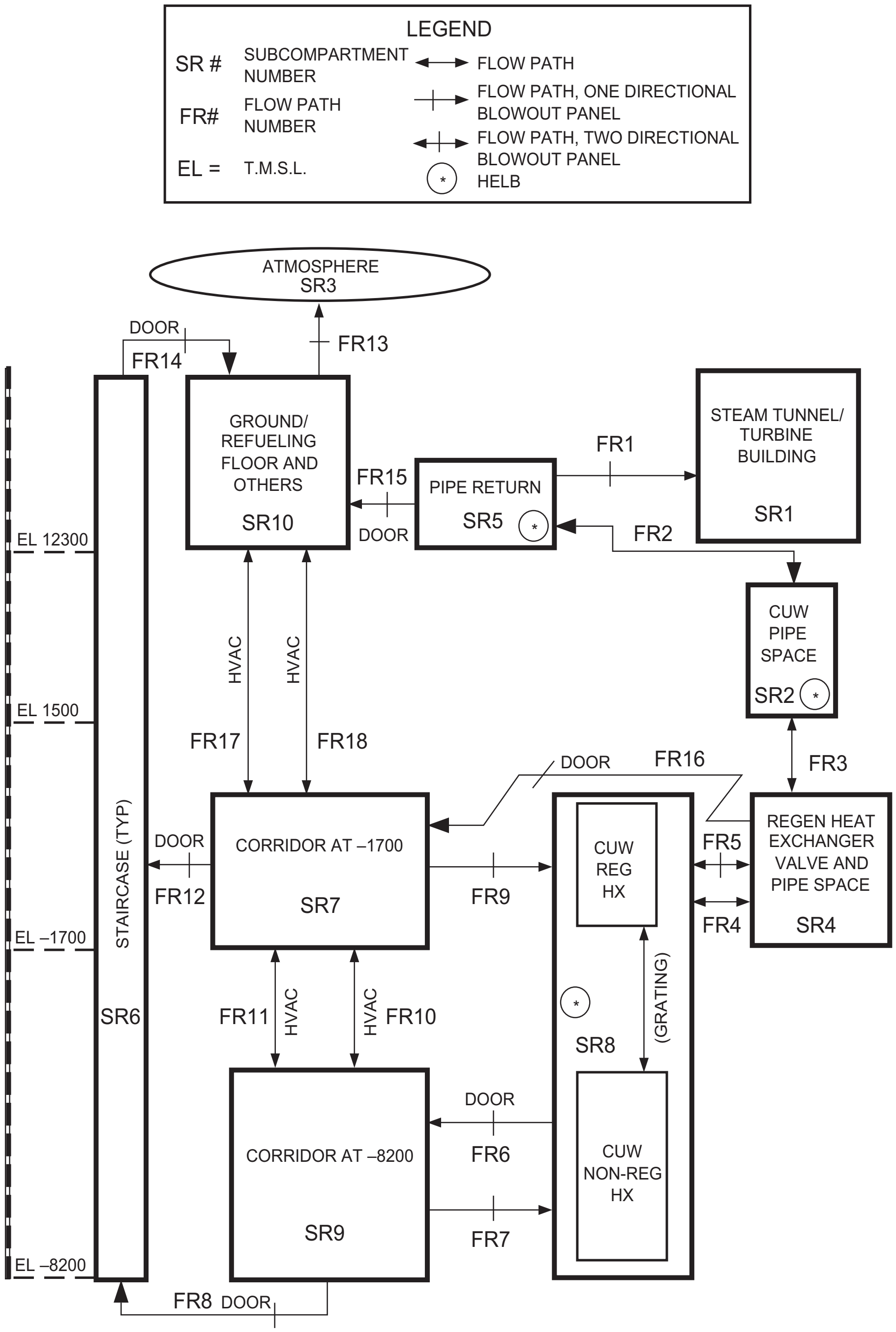


Figure 6.2-37c Secondary Containment Schematic Flow Diagram (CUW)

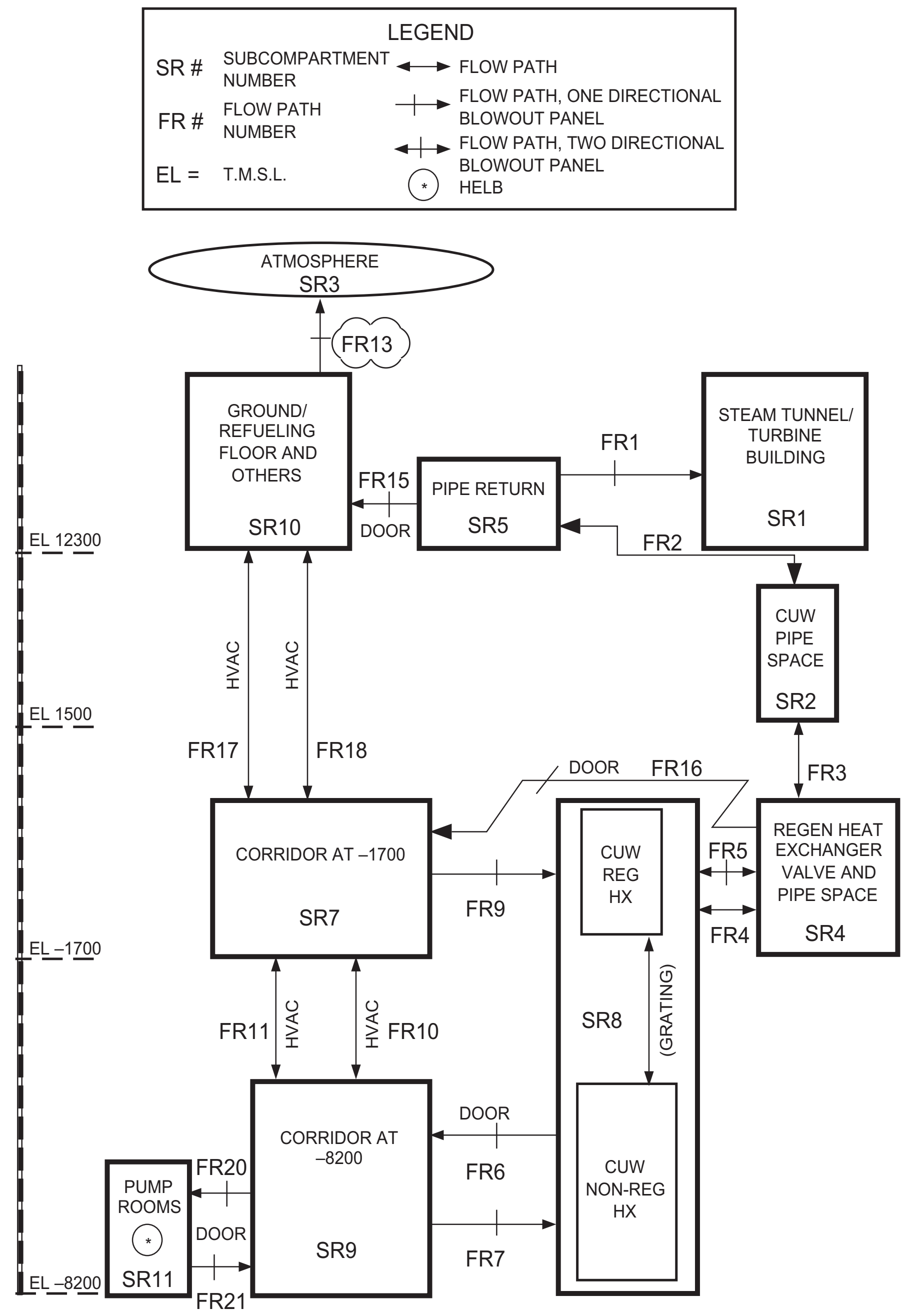


Figure 6.2-37d Secondary Containment Schematic Flow Diagram (CUW) (Continued)

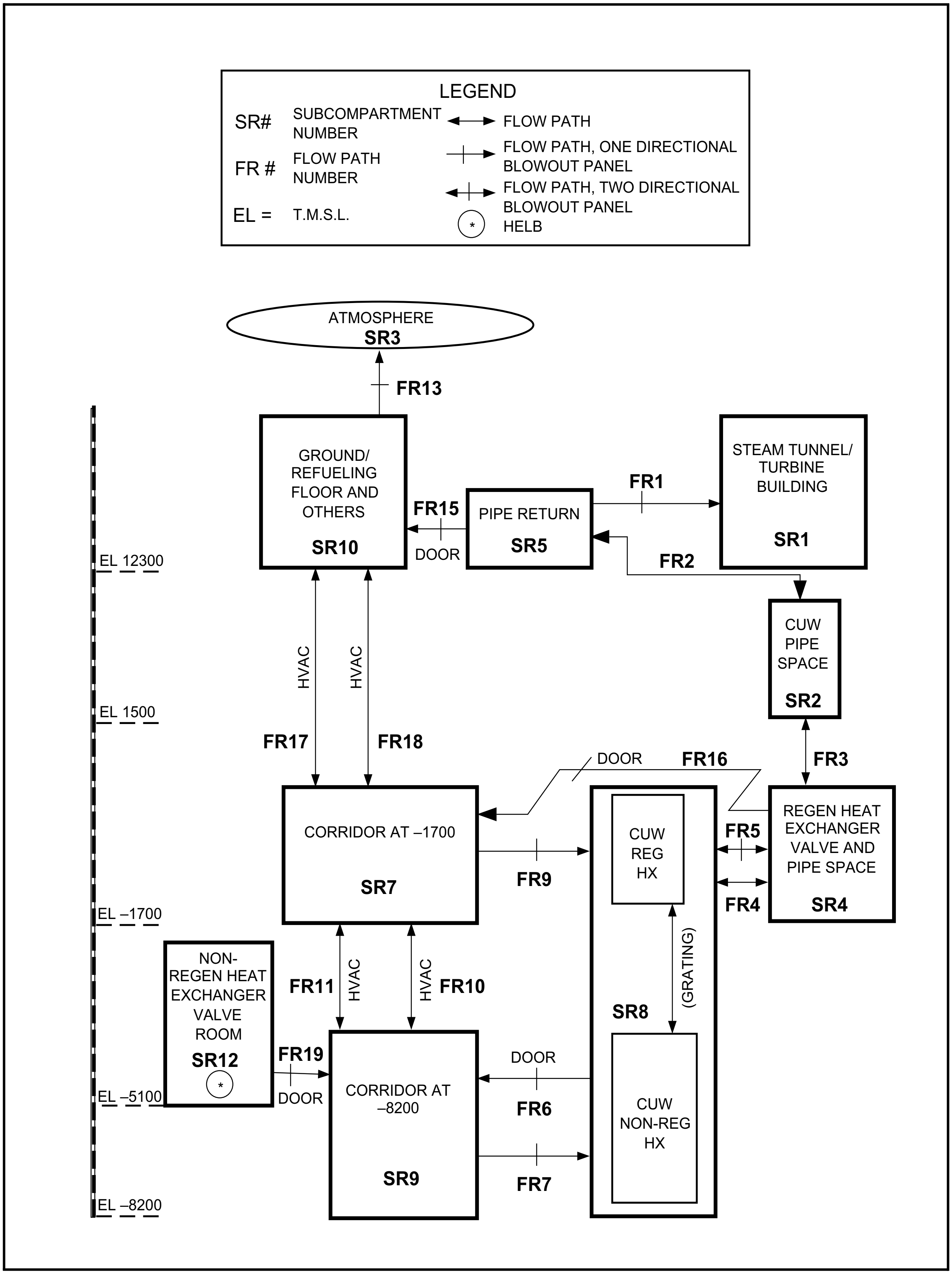


Figure 6.2-37e Secondary Containment Schematic Flow Diagram (CUW) (Continued)

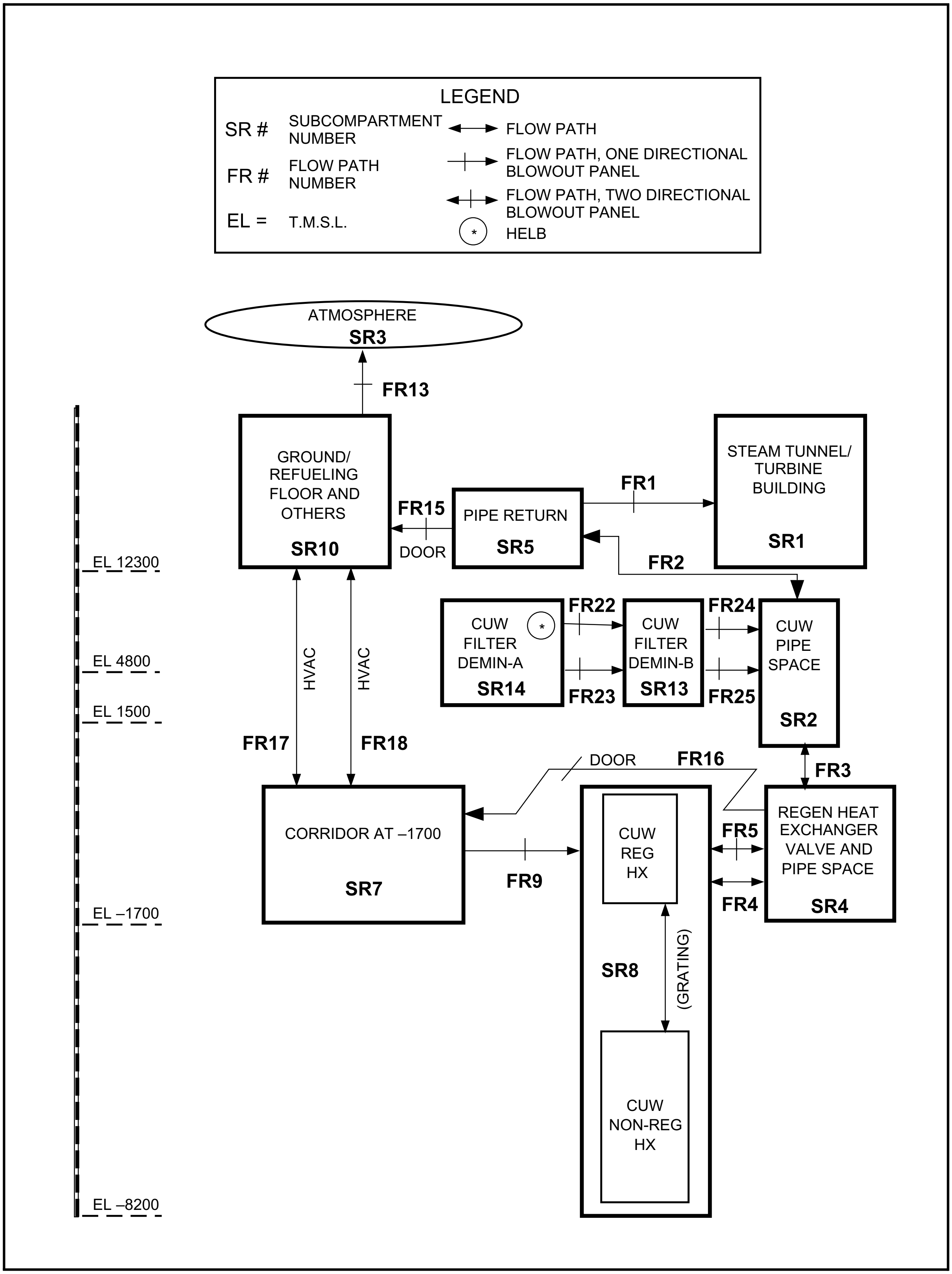


Figure 6.2-37f Secondary Containment Schematic Flow Diagram (CUW) (Continued)

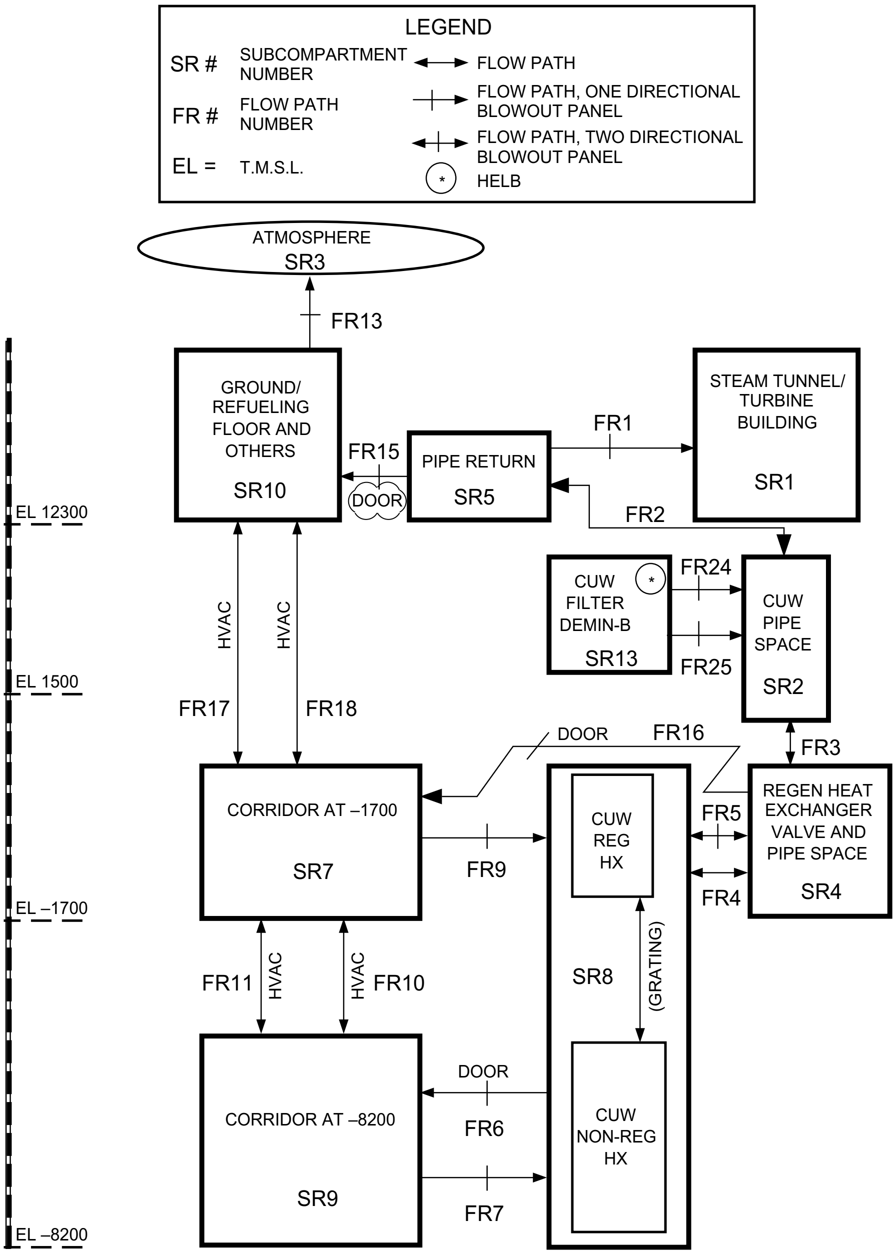


Figure 6.2-37g Secondary Containment Schematic Flow Diagram (CUW) (Continued)

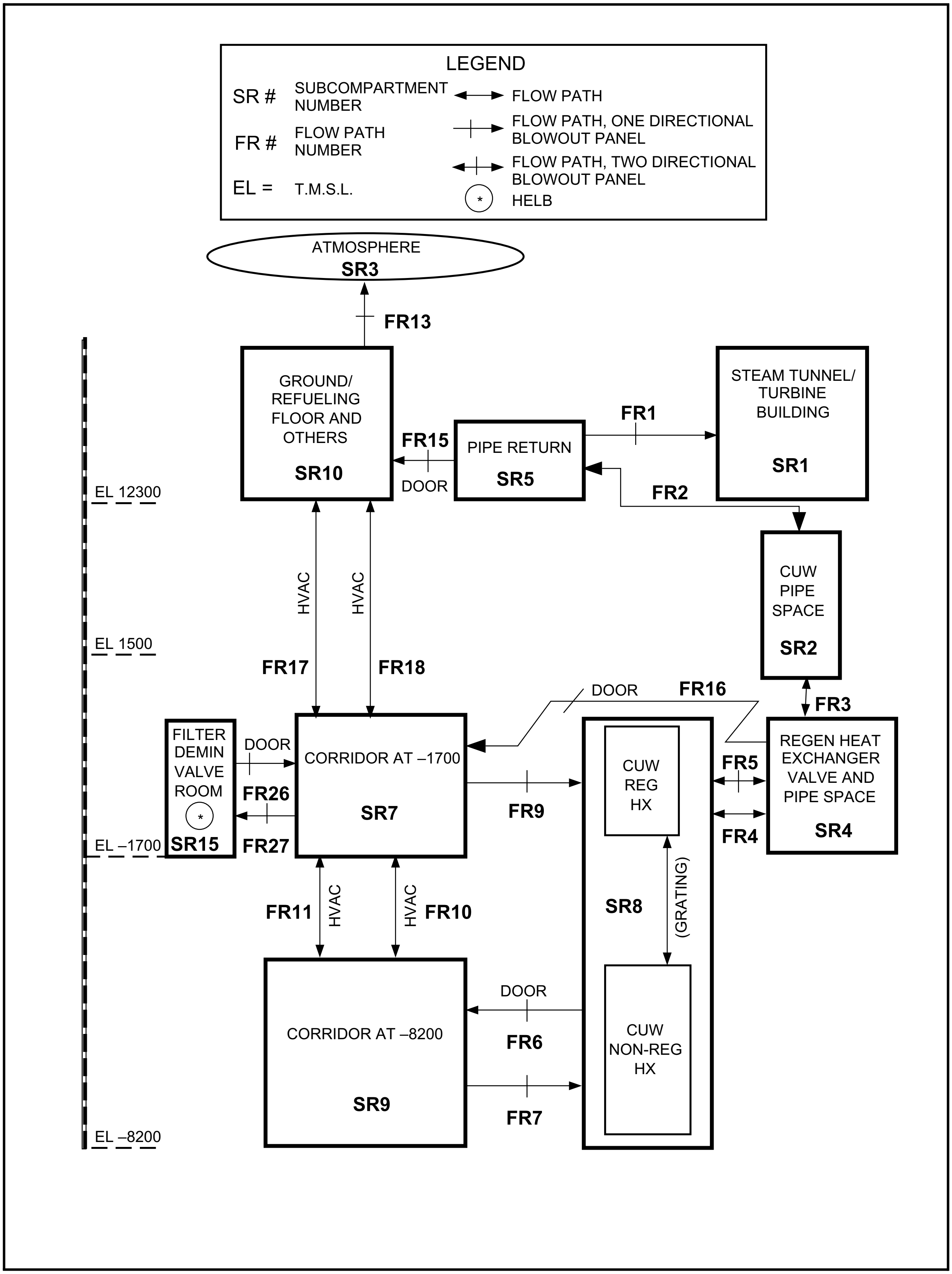


Figure 6.2-37h Secondary Containment Flow Schematic Diagram (CUW) (Continued)

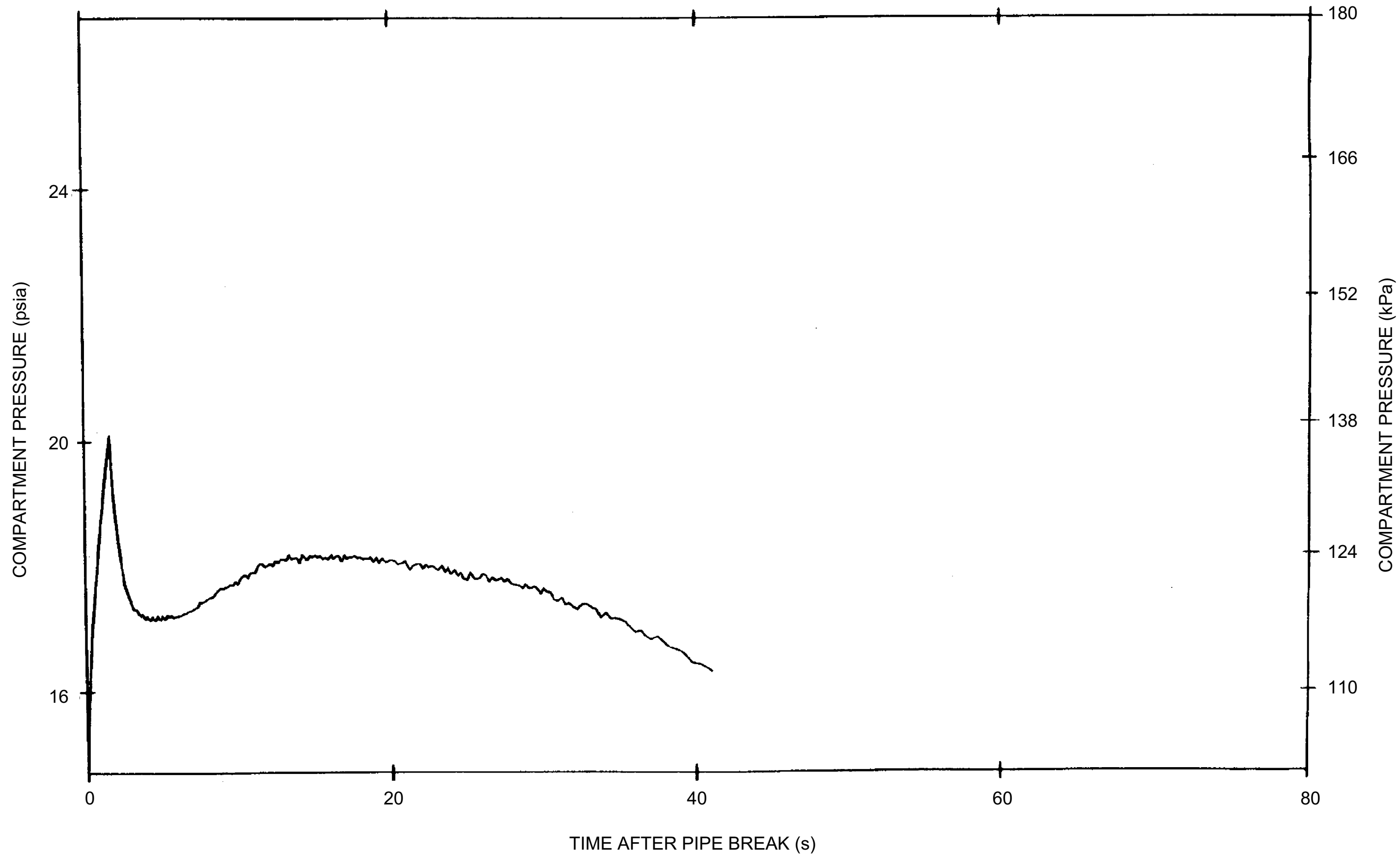


Figure 6.2-37i Pressure Transient Due to High Energy Line Break in RCIC Compartments-Isolated Cases

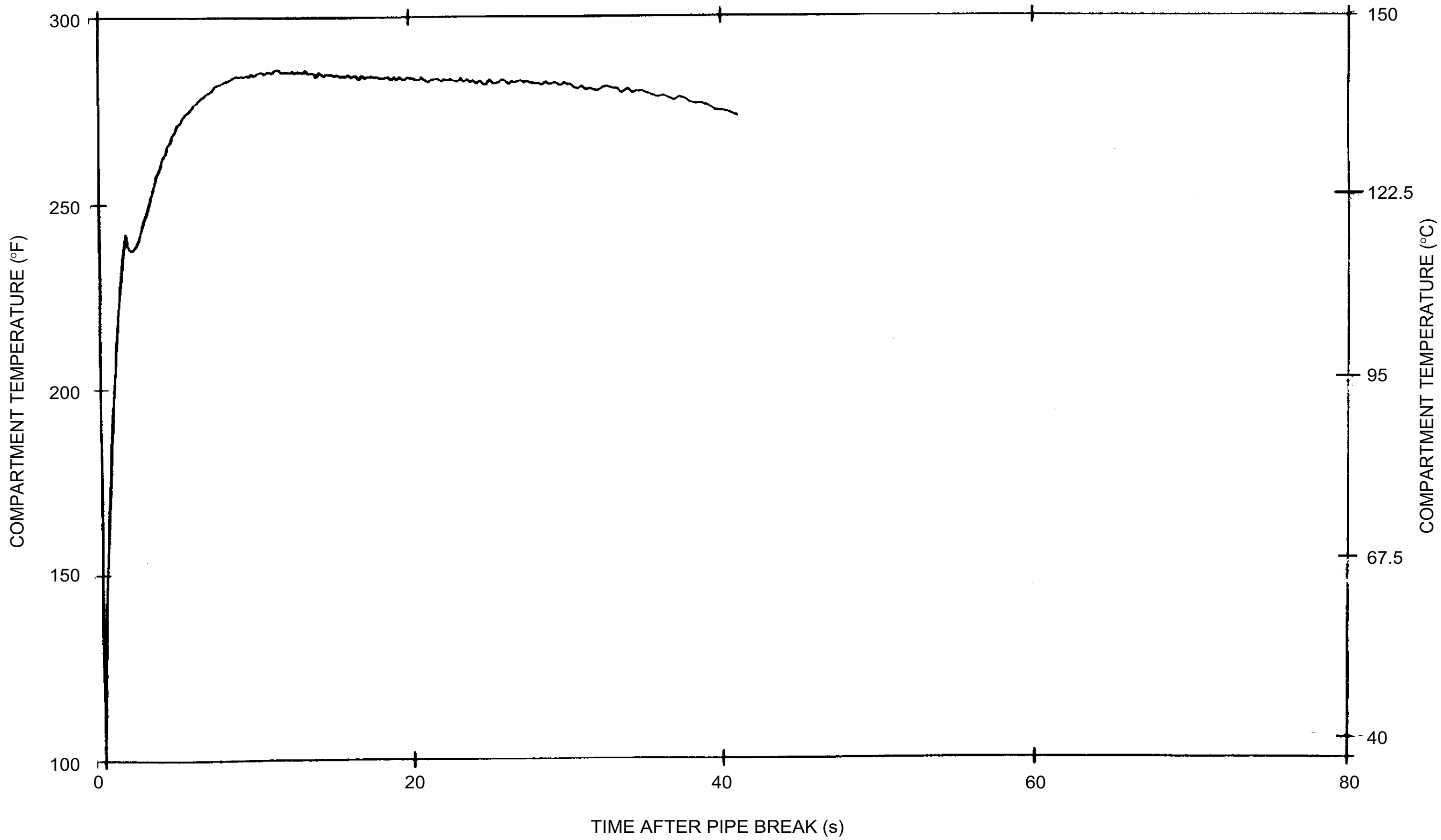


Figure 6.2-37j Temperature Transient Response Due to High Energy Line Break in RCIC Compartments

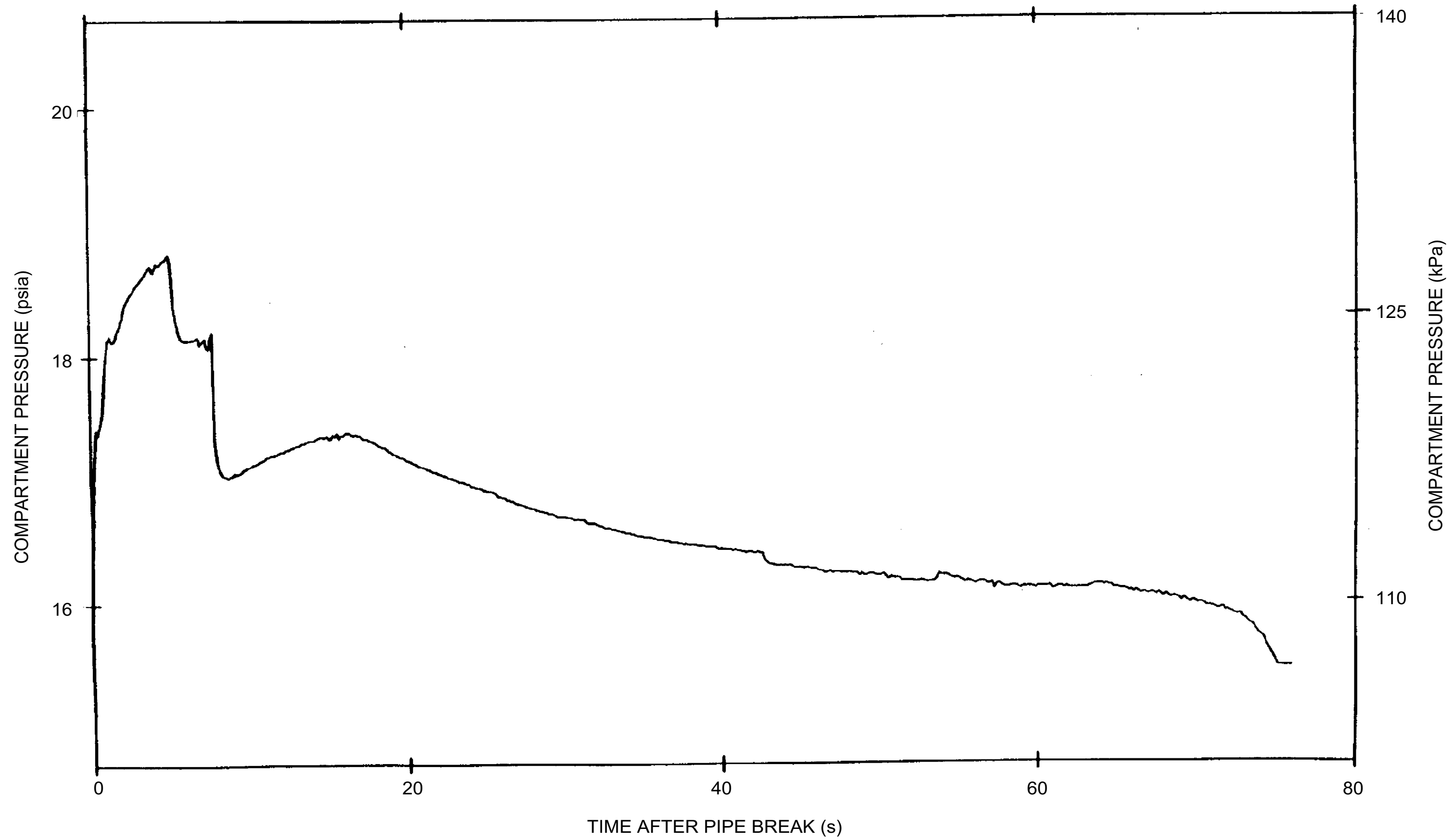


Figure 6.2-37k Pressure Transient Due to High Energy Line Breaks in CUW Compartments

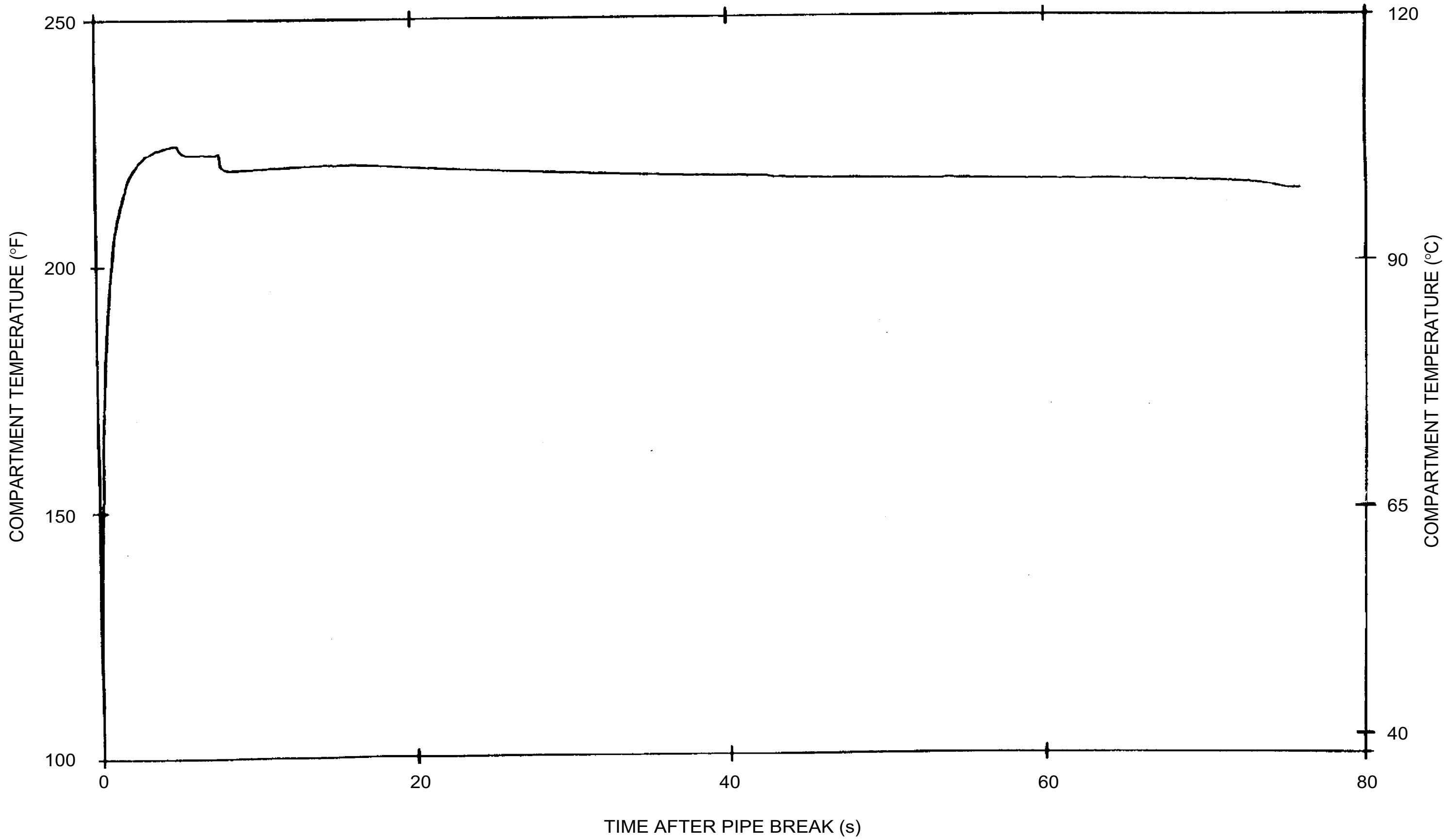


Figure 6.2-37l Temperature Transient Due to High Energy Line Breaks in CUW Compartments

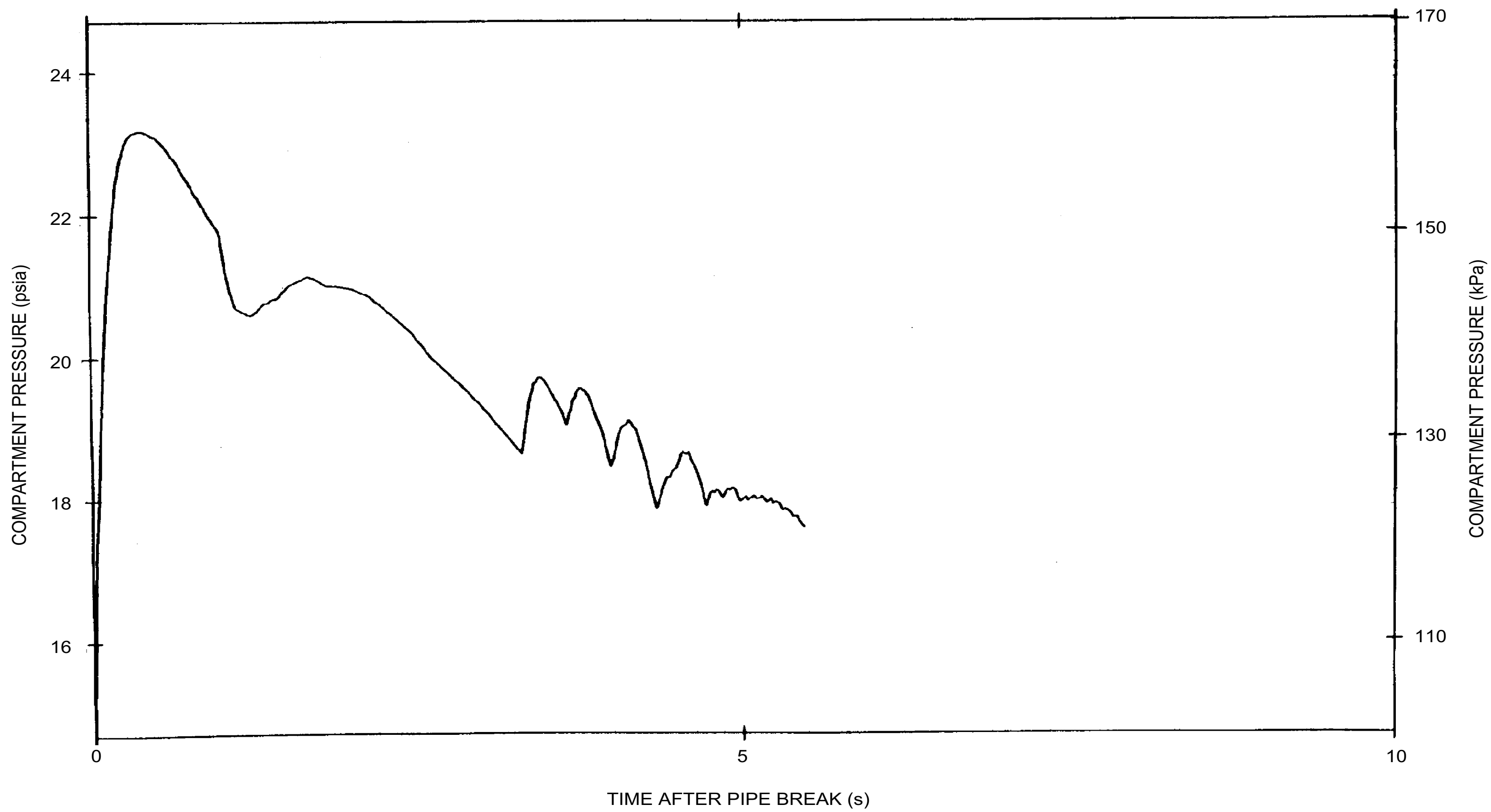


Figure 6.2-37m   Pressure Transient Due to High Energy Line Break in Steam Tunnel Compartments

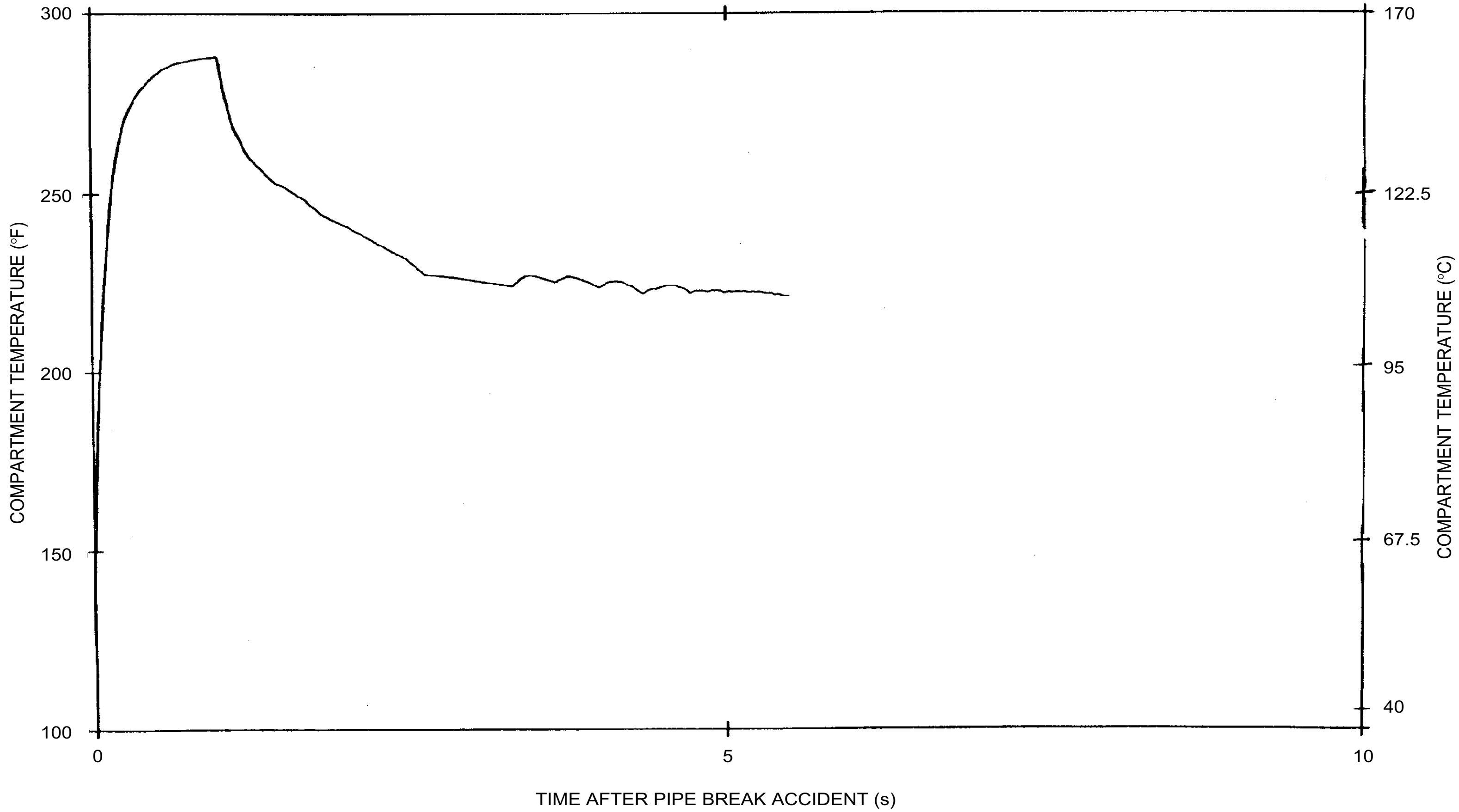


Figure 6.2-37n Temperature Transient Due to High Energy Line Break in Steam Tunnel Compartments

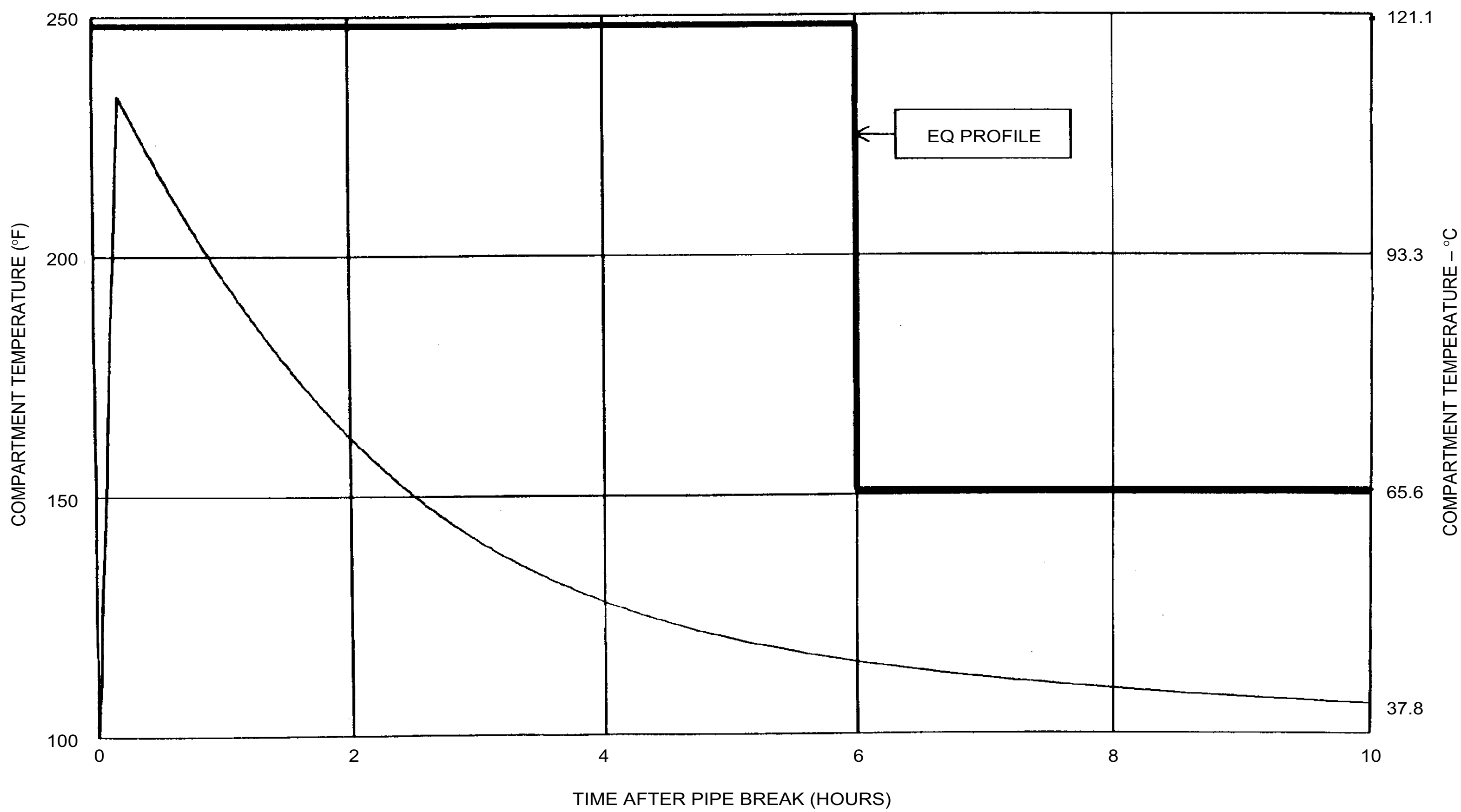


Figure 6.2-37o Compartment Temperature Response Due to High Energy Line Break in CUW Rooms

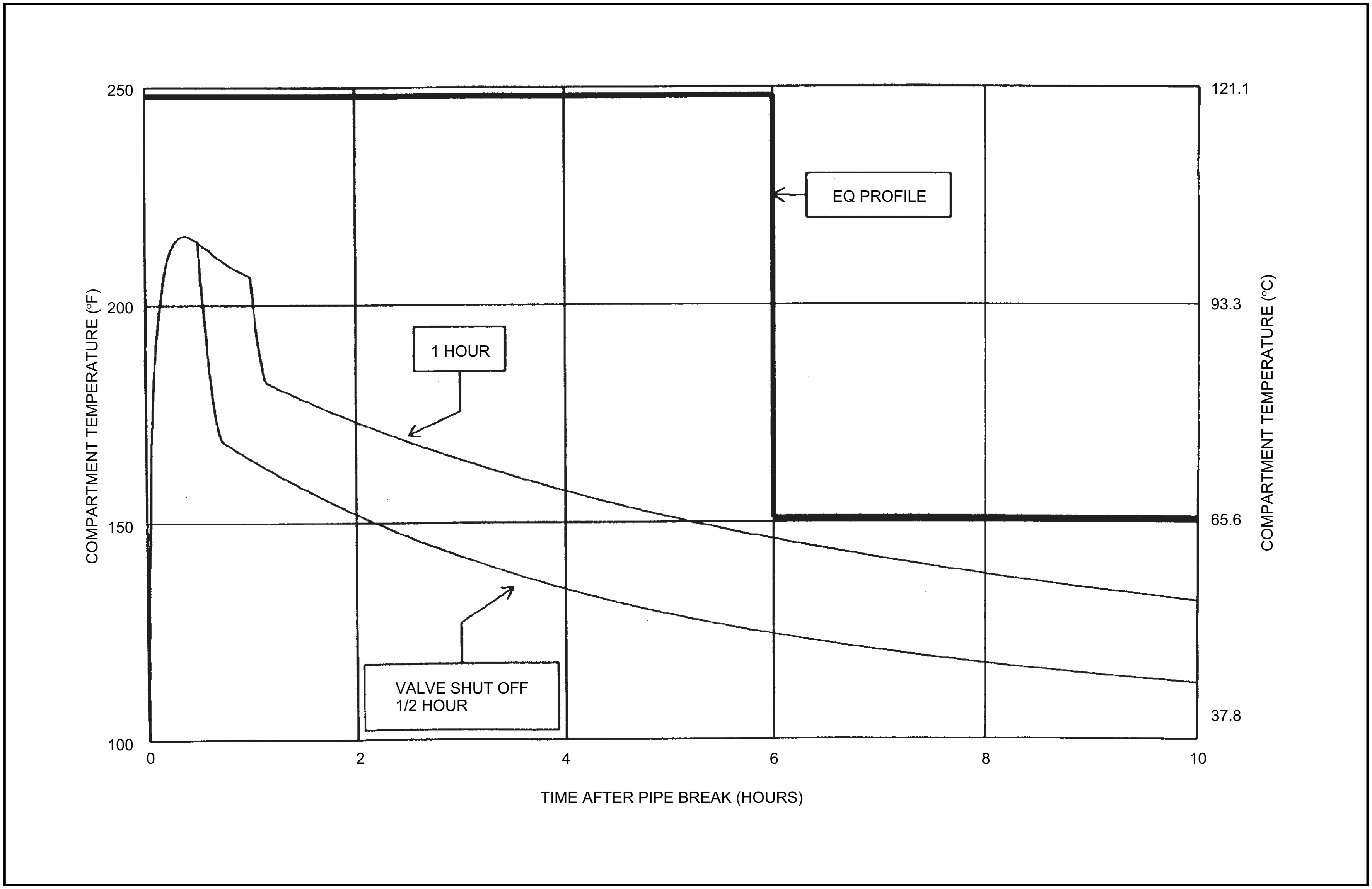


Figure 6.2-37p Compartment Transient Temperature Response Due to High Energy Line Break in CUW-Unisolated Case

The following figures are located in Chapter 21:

Figure 6.2-41 Not Used

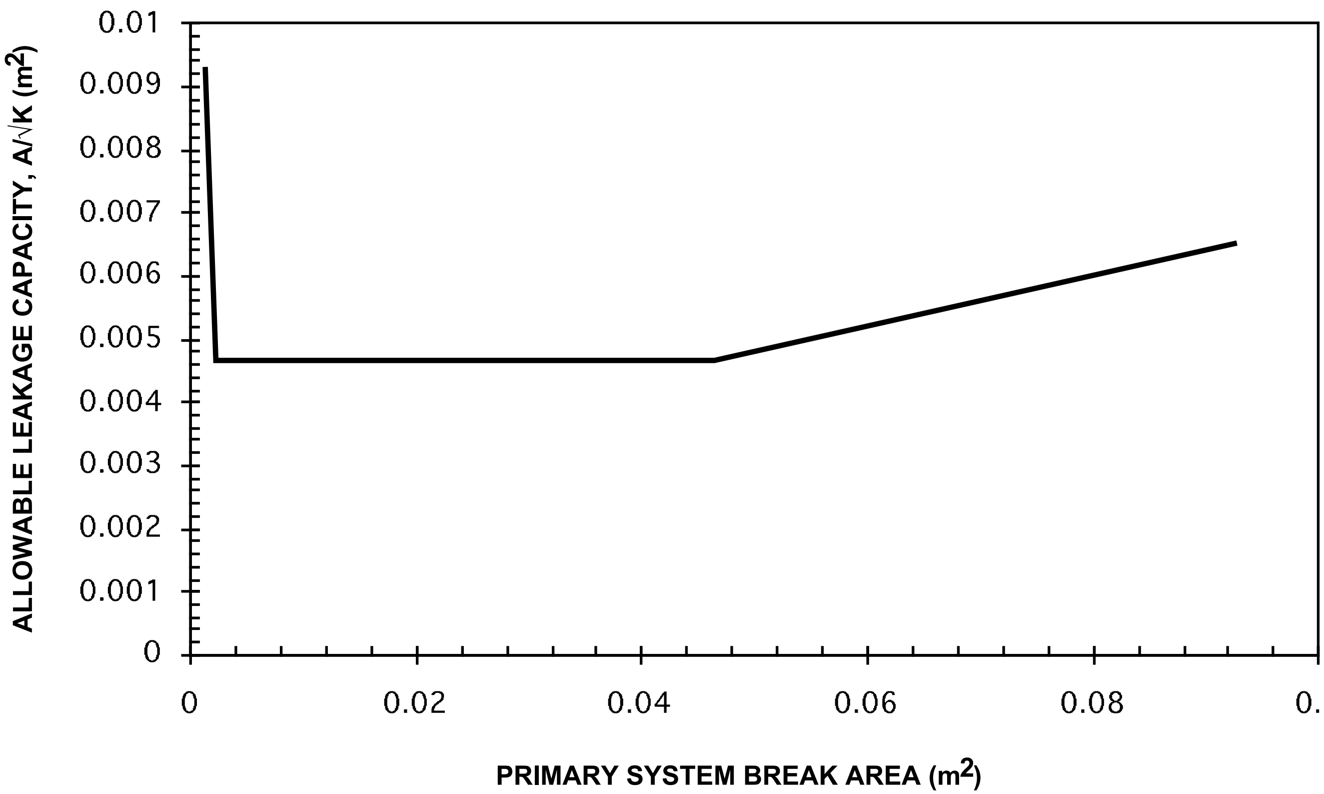


Figure 6.2-42 Allowable Steam Bypass Leakage Capacity

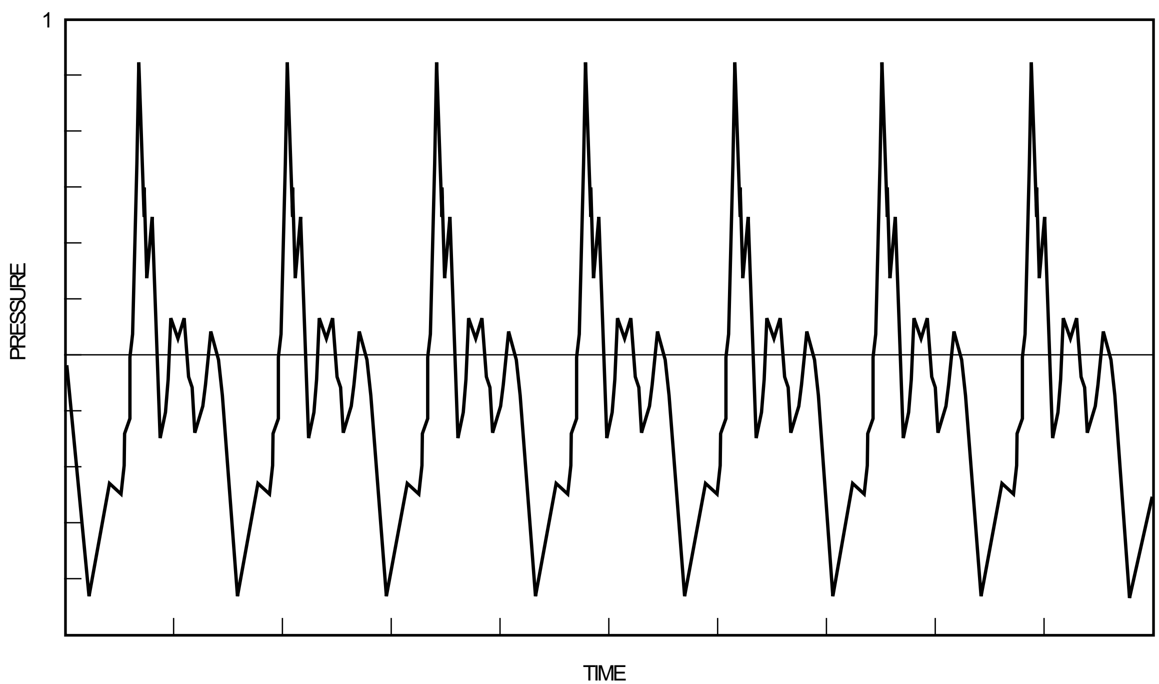


Figure 6.2-43 Typical Pressure Fluctuation Due to CO (See Figure 3B-22)

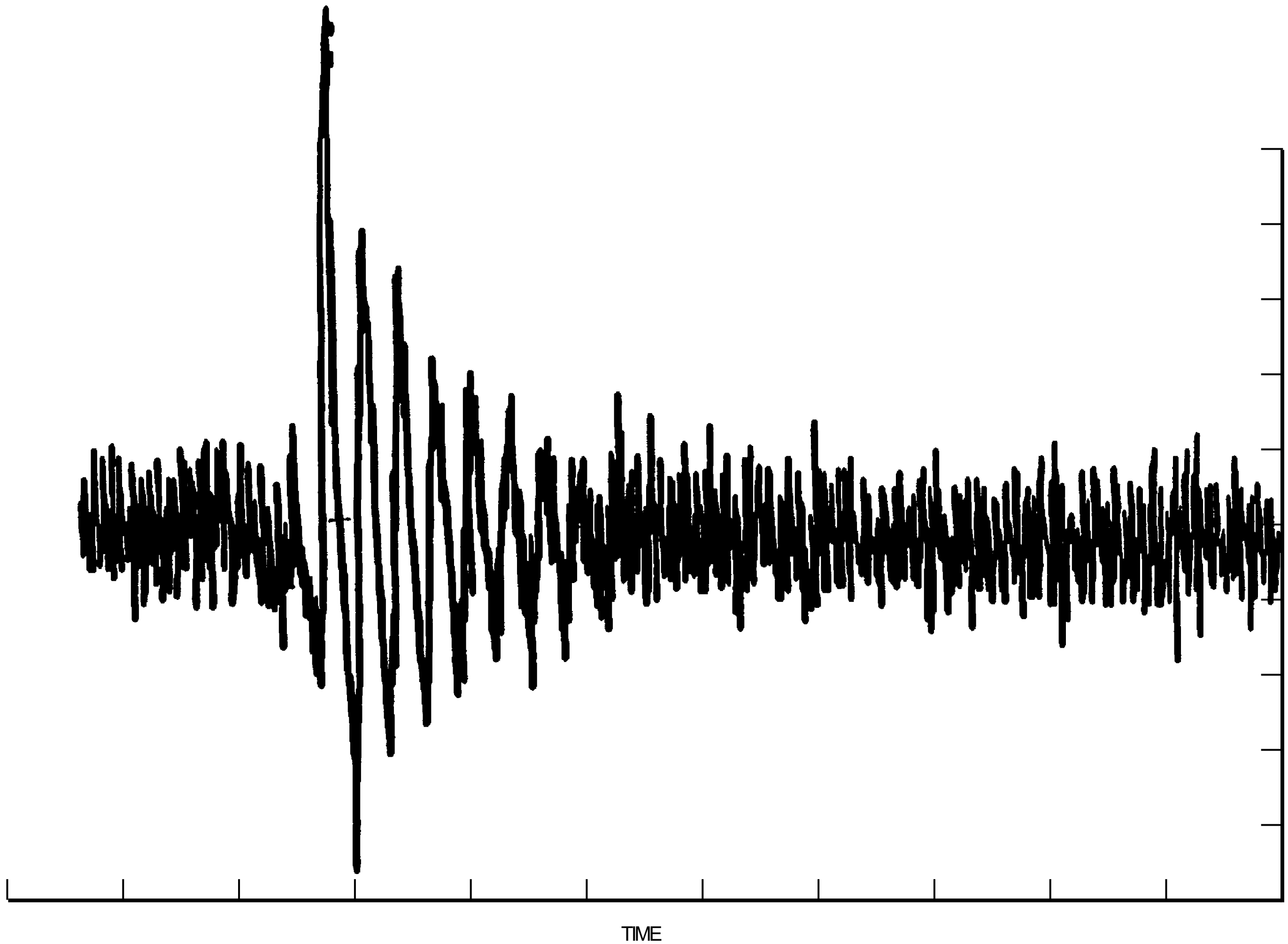


Figure 6.2-44 Typical Pressure Fluctuation Due To CH (See Figure 3B-24)

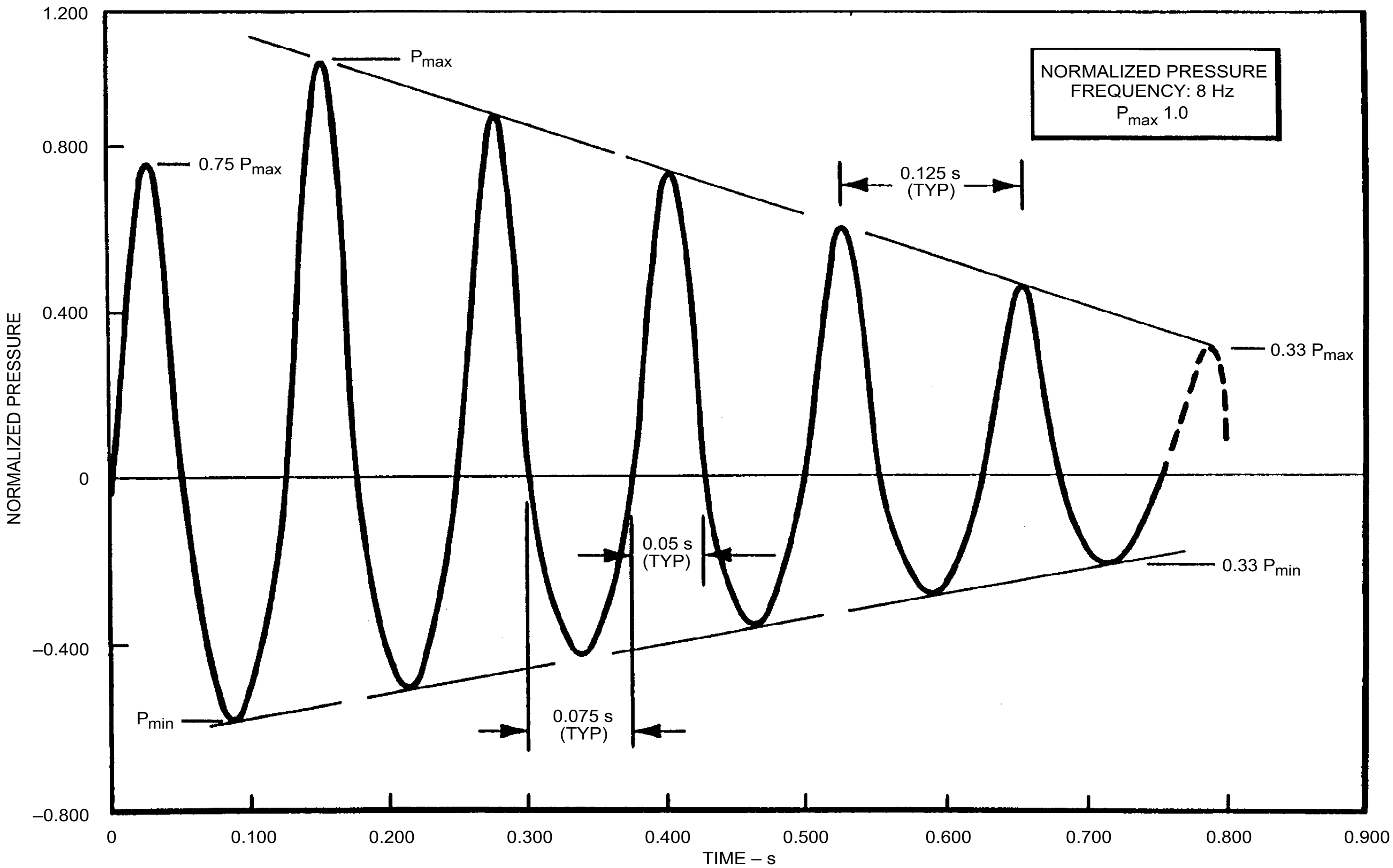


Figure 6.2-45 Quencher Bubble Pressure Time History