6.2 Containment Systems

6.2.1 Containment Functional Design

6.2.1.1 Containment Structure

6.2.1.1.1 Design Basis

The containment system is designed such that for all break sizes, up to and including the double-ended severance of a reactor coolant pipe or secondary side pipe, the containment peak pressure is below the design pressure. A summary of the results is presented in Table 6.2.1.1-1.

This capability is maintained by the containment system assuming the worst single failure affecting the operation of the passive containment cooling system (PCS). For primary system breaks, loss of offsite power (LOOP) is assumed. For secondary system breaks, offsite power is assumed to be available when it maximizes the mass and energy released from the break. Additional discussion of the assumptions made for secondary side pipe breaks may be found in subsection 6.2.1.4.

The single failure postulated for the containment pressure/temperature calculations is the failure of one of the valves controlling the cooling water flow for the PCS. Failure of one of these valves would lead to cooling water flow being delivered to the containment vessel through one of two delivery headers. This results in reduced cooling flow for PCS operation. No other single failures are postulated in the containment analysis.

The containment integrity analyses for the AP600 employs a multivolume lumped parameter model to study the long-term containment response to postulated Loss of Coolant Accidents (LOCA) and Main Steam Line Break (MSLB) accidents.

The analyses presented in this section are based on assumptions that are conservative with respect to the containment and its heat removal systems, such as minimum heat removal, and maximum initial containment pressure.

The containment design for the Safe Shutdown Earthquake (SSE) is discussed in subsection 3.8.2.

The minimum containment backpressure used in the Passive Core Cooling System (PXS) analysis is discussed in subsection 6.2.1.5.

6.2.1.1.2 Design Features

The operation of the PCS is discussed in subsection 6.2.2. The arrangement of the containment and internal structures is described in Section 1.2.

The reactor coolant loop is surrounded by structural walls of the containment internal structures. These structural walls are a minimum of 2-feet - 6-inches thick and enclose the reactor vessel, steam generators, reactor coolant pumps, and the pressurizer.

The containment vessel is designed and constructed in accordance with the ASME Code, Section III, Subsection NE, Metal Containment, including Addenda through 1989, as described in subsection 3.8.2.

Structural steel non-pressure retaining parts such as ladders, walkways, and handrails are designed to the requirements for steel structures defined in subsection 3.8.4.

The design features provide adequate containment sump levels following a design basis event as described in subsection 3.4.

Containment and subcompartment atmospheres are maintained during normal operation within prescribed pressure, temperature, and humidity limits by means of the containment air recirculation system (VCS), and the central chilled water system (VWS). The recirculation system cooling coils are provided with chilled water for temperature control. The filtration supply and exhaust subsystem can be utilized periodically to purge the containment air for pressure control. Periodic inspection and maintenance verify functional capability.

6.2.1.1.3 Design Evaluation

The Westinghouse-GOTHIC (WGOTHIC) computer code (Reference 20) is a computer program for modeling multiphase flow in a containment transient analysis. It solves the conservation equations in integral form for mass, energy, and momentum for multicomponent flow. The momentum conservation equations are written separately for each phase in the flow field (drops, liquid pools, and atmosphere vapor). The following terms are included in the momentum equation: storage, convection, surface stress, body force, boundary source, phase interface source, and equipment source.

To model the passive cooling features of the AP600, several assumptions are made in creating the plant decks. The external cooling water does not completely wet the containment shell, therefore, both wet and dry sections of the shell are modeled in the WGOTHIC analyses. The analyses use conservative coverage fractions to determine evaporative cooling.

Heat conduction from the dry to wet section is considered in the analysis. The combination of passive containment cooling system coverage area and heat conduction from the dry to wet sections is explained in Chapter 7 of Reference 20. The analyses conservatively assume that the external cooling water is not initiated until 337 seconds into the transient, allowing time to initiate the signal and to fill the headers and weirs and to develop the flow down the containment side walls. The effects of water flowing down the shell from gravitational forces are explicitly considered in the analysis.

The containment initial conditions of pressure, temperature, and humidity are provided in Table 6.2.1.1-2.

For the LOCA events, two double-ended guillotine reactor coolant system pipe breaks are analyzed. The breaks are postulated to occur in either a hot or a cold leg of the reactor coolant system. The hot leg break results in the highest blowdown peak pressure. The cold leg break results in the higher post-blowdown peak pressure. The cold leg break analysis includes the long term contribution to containment pressure from the sources of stored energy, such as the steam generators. The LOCA mass and energy releases described in subsection 6.2.1.3 are used for these calculations.

For the MSLB event, a representative pipe break spectrum is analyzed. Various break sizes and power levels are analyzed with the WGOTHIC code. The MSLB mass and energy releases described in subsection 6.2.1.4 are used for these calculations.

The results of the LOCA and MSLB postulated accidents are provided in Table 6.2.1.1-1. A comparison of the containment integrity acceptance criteria to General Design Criteria is provided in Table 6.2.1.1-3.

The containment pressure response for the peak pressure steam line break case is provided in Figure 6.2.1.1-1. The temperature response for this case is provided in Figure 6.2.1.1-2. Figures 6.2.1.1-3 and 6.2.1.1-4 provide the containment pressure and temperature response for the peak temperature steam line break case.

The passive internal containment heat sink data used in the WGOTHIC analyses is presented in Reference 20, Section 4. Data for both metallic and concrete heat sinks are presented. The containment pressure and temperature responses to a double-ended cold leg guillotine are presented in Figures 6.2.1.1-5 and 6.2.1.1-6 for the 24 hour portion of the transient and Figures 6.2.1.1-7 and 6.2.1.1-8 for the 72 hour transient. The containment pressure and temperature response to a double-ended hot leg guillotine break are presented in Figures 6.2.1.1-9 and 6.2.1.1-10. The physical properties of the materials corresponding to the heat sink information is presented in Table 6.2.1.1-8.

The instrumentation provided inside containment to monitor and record the containment pressure and temperature is found in Section 7.5.

6.2.1.1.4 External Pressure Analysis

Certain design basis events and credible inadvertent systems actuation have the potential to result in containment external pressure loads. Evaluations of these events show that a loss of all ac power sources during extreme cold ambient conditions has the potential for creating the worst-case external pressure load on the containment vessel. This event leads to a reduction in the internal containment heat loads from the reactor coolant system and other active components, thus resulting in a temperature reduction within the containment and an accompanying pressure reduction. Evaluations are performed to determine the maximum external pressure to which the containment may be subjected during a postulated loss of all ac power sources.

The evaluations are performed with the assumption of a -40°F ambient temperature with a steady 48 mph wind blowing to maximize cooling of the containment vessel. The initial internal containment temperature is conservatively assumed to be 120°F, creating the largest possible temperature differential to maximize the heat removal rate through the containment vessel wall. A negative 0.2 psig initial containment pressure is used for this evaluation. A conservative maximum initial containment relative humidity of 100 percent is used to produce the greatest reduction in containment pressure due to the loss of steam partial pressure by condensation. It is also conservatively assumed that no air leakage occurs into the containment during the transient.

Evaluations are performed using WGOTHIC with conservatively low estimates of the containment heat loads and conservatively high heat removal through the containment vessel consistent with the limiting assumptions stated above. Results of these evaluations demonstrate that at one hour after the event the net external pressure is well within the 3.0 psid design external pressure. This is sufficient time for operator action to prevent the containment pressure from dropping below the design external pressure, based on the PAM's containment pressure indications (four containment pressure instruments) and the ability to mitigate the pressure reduction by opening either set of containment ventilation purge isolation valves, which are powered by the 1E batteries.

The limiting case containment pressure transient is shown in Figure 6.2.1.1-11.

6.2.1.2 Containment Subcompartments

6.2.1.2.1 Design Basis

Subcompartments within containment are designed to withstand the transient differential pressures of a postulated pipe break. These subcompartments are vented so that differential pressures remain within structural limits. The subcompartment walls are challenged by the differential pressures resulting from a break in a high energy line. Therefore, a high energy line is postulated, with a break size chosen consistent with the position presented in Section 3.6, for analyzing the maximum differential pressures across subcompartment walls.

Section 3.6 describes the application of the mechanistic pipe break criteria, commonly referred to as leak-before-break (LBB), to the evaluation of pipe ruptures. This eliminates the need to consider the dynamic effects of postulated pipe breaks for pipes which qualify for LBB. However, the analyses of containment pressure and temperature, emergency core cooling, and environmental qualification of equipment are based on double-ended guillotine (DEG) reactor coolant system breaks and through-wall cracks.

6.2.1.2.1.1 Summary of Subcompartment Pipe Break Analyses

Each subcompartment is analyzed for effects of differential pressures resulting from the break of the most limiting line in the subcompartment which has not been evaluated for LBB.

The subcompartment analysis demonstrates that the wall differential pressures resulting from the most limiting high energy line break within the subcompartments are within the design capability.

6.2.1.2.2 Design Features

The plant general arrangement drawings shown in Section 1.2 include descriptions of the containment sub-compartments and surrounding areas. The general arrangement drawings are used in assembling the subcompartment analysis model.

Vent paths considered in the analyses are shown in the general arrangement drawings and consist of floor gratings and openings through walls. In the AP600 subcompartment analyses, no credit is taken for vent paths that become available only after the occurrence of the postulated break (such as blowout panels, doors, hinged panels and insulation collapsing).

6.2.1.2.3 Design Evaluation

The TMD computer code (Reference 2) is used in the subcompartment analysis to calculate the differential pressures across subcompartment walls. The TMD code has been reviewed by the NRC and approved for use in subcompartment differential pressure analyses.

Specific information relative to details on the analysis, such as noding diagrams, volumes, vent areas, and initial conditions, are provided in Section 6 of Reference 26.

The methodology used to generate the short term mass and energy releases is described in subsection 6.2.1.3.1.

The initial atmospheric conditions used in the TMD subcompartment analysis are selected so that the calculated differential pressures are maximized. These conditions are chosen according to criteria identified in subsection 6.2.1.2 of NUREG-0800 and include the maximum allowable air temperature, minimum absolute pressure, and zero percent relative humidity.

The containment and subcompartment atmospheres during normal operating conditions are maintained within prescribed pressure, temperature, and humidity limits by means of the containment air recirculation system (VCS), and the central chilled water system (VWS). The recirculation system cooling coils are provided with chilled water to provide sufficient temperature control. The filtration supply and exhaust subsystem can be utilized to purge the containment air for pressure control. Periodic inspection and maintenance are performed to verify functional capability.

6.2.1.2.3.1 Flow Equation

The flow equations used by the TMD code to calculate the flow between nodes are described in Reference 2. These flow equations are based on the unaugmented critical flow model, which demonstrate conservatively low critical flow velocity predictions compared to experimental test data. Due to the TMD calculation methods presented in subsection 1.3.1 of Reference 2, 100 percent entrainment results in the highest calculated differential pressures and therefore this degree of entrainment is conservatively assumed in the subcompartment analysis.

6.2.1.2.3.2 Pipe Breaks

The subcompartment analysis for the steam generator compartment is performed assuming a double-ended guillotine break in a 3-inch inside diameter reactor cooling system hot leg or cold leg pipe or a 4-inch double-ended steam generator blowdown line, or a 4-inch pressurizer spray line break. The breaks can be assumed to occur between the 84-foot elevation and the 135-foot elevation of the steam generator compartment. Because the TMD code assumes homogeneous mixtures within a node, the specific location of the break within the node is not critical to the differential pressure calculation. No flow restrictions exist that limit the flow out of the break.

The analysis for the pressurizer compartment pipe and valve room is performed assuming a double-ended guillotine break in a 4-inch inside diameter reactor coolant system spray line. This break envelopes the branch lines that could be postulated to rupture in this area. The break is assumed to occur between the 107-foot elevation and the 163-foot elevation of the pressurizer compartment or the 118-foot to 135-foot elevations of the pressurizer spray valve room.

The analysis for the steam generator vertical access area is performed assuming a double-ended guillotine break in a 3-inch inside diameter reactor coolant system cold-leg pipe. This break envelopes the branch lines that could be postulated to rupture in this area. The break is assumed to occur between the 83-foot elevation and the 103-foot elevation of the steam generator vertical access area compartment.

The analysis for the maintenance floor and operating deck compartments are performed assuming a one square foot rupture of a main steam line pipe. This break envelopes the branch lines that could be postulated to rupture in these areas. The break is assumed to occur between the 107-foot elevation and the 135-foot elevation of the maintenance floor compartment and between the 135-foot elevation and the 256-foot elevation of the operating deck region.

The analysis for the main chemical and volume control system room is performed assuming a single-ended guillotine break in a 3-inch diameter reactor coolant system cold-leg pipe. This break envelopes the branch lines that could be postulated to rupture in this area. The break is assumed to occur between the 91-foot elevation and the 105-foot elevation of the chemical and volume control system room compartment.

The analysis for the pipe tunnel in the chemical and volume control system room is performed assuming a double-ended guillotine break in a 4-inch diameter steam generator blowdown line. This double-ended break envelopes the branch lines that could be postulated to rupture in this area. The break is assumed to occur between the 100-foot elevation and the 105-foot elevation of the chemical and volume control system room pipe tunnel.

An evaluation of rooms which could have either a main or startup feedwater line break was performed. No significant pressurization of the regions is predicted to occur because the postulated breaks are located in regions which are open to the large free volume of containment. For these regions, the main or startup feedwater line breaks are not limiting.

6.2.1.2.3.3 Node Selection

The nodalization for the sub-compartments is analyzed in sufficient detail such that nodal boundaries are at the location of flow obstructions or geometrical changes within the subcompartment. These discontinuities create pressure differentials between adjoining nodes. There are no significant discontinuities within each node, and hence the pressure gradient is negligible within any node.

6.2.1.2.3.4 Vent Flowpath Flow Conditions

The flow characteristics for each of the subcompartments are such that, at no time during the transient does critical flow exist through vent paths.

6.2.1.3 Mass and Energy Release Analyses for Postulated Pipe Ruptures

Mass and Energy releases are documented in this section for two different types of transients.

The first section describes the methodology used to calculate the releases for the subcompartment differential pressure analysis using the TMD code (referred to as the short term analysis). These releases are used for the subcompartment response in subsection 6.2.1.2.

The second section describes the methodology used to determine the releases for the containment pressure and temperature calculations using the WGOTHIC code (Reference 20) (referred to as the long term analysis). These releases are used for the containment integrity analysis in subsection 6.2.1.1.

The short term analysis considers only the initial stages of the blowdown transient, and takes into consideration the application of LBB methodology. LBB is discussed in subsection 3.6.3. Since LBB is applicable to reactor coolant system piping that is 6 inches in diameter and greater, the mass and energy release analysis for sub-compartments postulates the complete DEG severance of 3-inch and 4-inch pipe. The mass and energy release postulated for a ruptured steam line is for a one square foot break.

Conversely, the limiting break size for containment integrity analysis considers as its LOCA design basis the complete DEG severance of the largest reactor coolant system pipe.

The containment system receives mass and energy releases following a postulated rupture of the reactor coolant system. The release rates are calculated for pipe failure at two locations: the hot leg and the cold leg. These break locations are analyzed for both the short-term and the long-term transients. Because the initial operating pressure of the reactor coolant system is approximately 2250 psi, the mass and energy are released extremely rapidly when the break occurs. As the water exits from the broken pipe, a portion of it flashes to steam because of the differences in pressure and temperature between the reactor coolant system and containment. The reactor coolant system depressurizes rapidly since break flow exits from both sides of the pipe in a DEG severance.

6.2.1.3.1 Short Term Mass and Energy Release Data

The AP600 short term LOCA mass and energy releases are predicted for the first ten seconds of the blowdown from a postulated DEG break of the largest non-LBB high energy line in each compartment. The density of the fluid released from a postulated pipe rupture has a direct effect on the magnitude of the differential pressures that results across subcompartment walls. A DEG rupture that is postulated in the cold leg piping is typically the most limiting scenario. This analysis provides mass and energy releases for a 3-inch DEG rupture in the cold leg and in the hot leg.

The modified Zaloudek correlation (Reference 3) is used to calculate the critical mass flux from a 3-inch double-ended cold leg guillotine (DECLG) break and a 3-inch double-ended hot leg guillotine (DEHLG) break. This maximum mass flux is conservatively assumed to remain constant at the initial AP600 full power steady state conditions and the enthalpy is varied to determine the energy release rates. Conservative enthalpies are obtained from the SATAN-VI blowdown transients for ruptures of the largest reactor coolant system cold leg and hot leg piping in the AP600 design. This assumption maximizes the mass released, which is conservative for the subcompartment analysis.

The mass release for the 4-inch pressurizer spray line break is determined with the Fauske break flow model in NOTRUMP. The steam generator blowdown releases for a 4-inch line are calculated with the critical mass flux method.

The initial conditions and inputs to the modified Zaloudek correlation are given in Table 6.2.1.3-1. The short term LOCA double-ended guillotine mass and energy release data is provided in Tables 6.2.1.3-2 and 6.2.1.3-3 for the cold and hot legs, respectively. The short-term non-LOCA mass and energy release data are provided in Tables 6.2.1.3-4 and 6.2.1.3-5. The pressurizer spray line mass and energy releases are shown in Table 6.2.1.3-6. The short term LOCA single-ended mass and energy release data are provided in Table 6.2.1.3-7.

6.2.1.3.2 Long Term Mass and Energy Release Data

A long term LOCA analysis calculational model is typically divided into four phases: blowdown, which includes the period from the accident initiation (when the reactor is in a steady-state full power operation condition) to the time that the broken loop pressure equalizes to the containment pressure; refill, which is the time from the end of the blowdown to the time when the passive core cooling system (PXS) refills the vessel lower plenum; reflood, which begins when the water starts to flood the core and continues until the core is completely quenched; and post-reflood, which is the period after the core has been quenched and energy is released to the reactor coolant system primary system by the reactor coolant system metal, core decay heat, and the steam generators.

The long-term analysis considers the blowdown, reflood, and post-reflood phases of the transient. The refill period is conservatively neglected so that the releases to the containment are conservatively maximized.

The AP600 long-term LOCA mass and energy releases are predicted for the blowdown phase for postulated DECLG and DEHLG breaks. The blowdown phase mass and energy releases are calculated using the NRC approved SATAN-VI computer code (Reference 4). The post blowdown phase mass and energy releases are calculated considering the energy released from the available energy sources described below. The energy release rates are conservatively modeled so that the energy is released quickly. The higher release rates result in a conservative containment pressure calculation. The releases are provided in Tables 6.2.1.3-9 and 6.2.1.3-10.

6.2.1.3.2.1 Mass and Energy Sources

The following are accounted for in the long-term LOCA mass and energy calculation:

- Decay heat
- Core stored energy
- Reactor coolant system fluid and metal energy
- Steam Generator fluid and metal energy
- Accumulators core make-up tanks (CMTs), and the in-containment refueling water storage tank (IRWST)
- Zirconium-water reaction

The methods and assumptions used to release the various energy sources during the blowdown phase are given in Reference 4.

The following parameters are used to conservatively analyze the energy release for maximum containment pressure:

- Maximum expected operating temperature
- Allowance in temperature for instrument error and dead band
- Margin in volume (+1.4 percent)

- Allowance in volume for thermal expansion (+1.6 percent)
- 100 percent full power operation
- Allowance for calorimetric error (+2.0 percent of full power)
- Conservatively modified coefficients of heat transfer
- Allowance in core stored energy for effect of fuel densification
- Margin in core stored energy (+15.0 percent)
- Allowance in pressure for instrument error and dead band
- Margin in steam generator mass inventory (+10.0 percent)
- One percent of the Zirconium surrounding the fuel is assumed to react

6.2.1.3.2.2 Description of Blowdown Model

A description of the SATAN-VI model that is used to determine the mass and energy released from the reactor coolant system during the blowdown phase of a postulated LOCA is provided in Reference 4. Significant correlations are discussed in this reference.

6.2.1.3.2.3 Description of Post-Blowdown Model

The remaining reactor coolant system and SG mass and energy inventories at the end of blowdown are used to define the initial conditions for the beginning of the reflood portion of the transient. The broken and unbroken loop SG inventories are kept separate to account for potential differences in the cooldown rate between the loops. In addition, the mass added to the reactor coolant system from the IRWST is returned to containment as break flow so that no net change in system mass occurs.

Energy addition due to decay heat is computed using the 1979 ANS standard (plus 2 sigma) decay heat table from Reference 4. The energy release rates from the reactor coolant system metal and steam generators are modelled using exponential decay rates. This modelling is consistent with analyses for current generation design analyses that are performed with the models described in Reference 4.

The accumulator, CMT, and IRWST mass flow rates are computed from the end of blowdown to the time the tanks empty. The rate of reactor coolant system mass accumulation is assumed to decrease exponentially during the reflood phase. More CMT and accumulator flow is spilled from the break as the system refills. The break flow rate is determined by subtracting the reactor coolant system mass addition rate from the sum of the accumulator, CMT and IRWST flow rates.

Mass which is added to, and which remains in, the vessel is assumed to be raised to saturation. Therefore, the actual amount of energy available for release to the containment for a given time period is determined from the difference between the energy required to raise the temperature of the incoming flow to saturation and the sum of the decay heat, core stored energy, reactor coolant system metal energy and SG mass and metal energy release rates. The energy release rate for the available break flow is determined from a comparison of the total energy available release rate and the energy release rate assuming that the break flow is

100-percent saturated steam. Saturated steam releases maximize the calculated containment pressurization.

6.2.1.3.2.4 Single Failure Analysis

The assumptions for the containment mass and energy release analysis are intended to maximize the calculated release. A single failure could reduce the flow rate of water to the RCS, but would not disable the passive core cooling function. For example, if one of the two parallel valves from the CMT were to fail to open, the injection flow rate would be reduced and, as a result, the break mass release rate would decrease. Therefore, to maximize the releases, the AP600 mass and energy release calculations conservatively do not assume a single failure. The effects of a single failure are taken into account in the containment analysis of subsection 6.2.1.1.

6.2.1.3.2.5 Metal-Water Reaction

Consistent with 10 CFR 50, Appendix K criteria, the energy release associated with the zirconium-water exothermic reaction has been considered. The LOCA peak cladding temperature analysis, presented in Chapter 15, that demonstrates compliance with the Appendix K criteria demonstrates that no appreciable level of zirconium oxidation occurs. This level of reaction has been bounded in the containment mass and energy release analysis by incorporating the heat of reaction from 1 percent of the zirconium surrounding the fuel. This exceeds the level predicted by the LOCA analysis and results in additional conservatism in the mass and energy release calculations.

6.2.1.3.2.6 Energy Inventories

Inventories of the amount of mass and energy released to containment during a postulated LOCA are provided in summary Tables 6.2.1.3-2 through 6.2.1.3-7.

6.2.1.3.2.7 Additional Information Required for Confirmatory Analysis

System parameters and hydraulic characteristics needed to perform confirmatory analysis are provided in Table 6.2.1.3-8 and Figures 6.2.1.3-1 through 6.2.1.3-4.

6.2.1.4 Mass and Energy Release Analysis for Postulated Secondary-System Pipe Rupture Inside Containment

Steam line ruptures occurring inside a reactor containment structure may result in significant releases of high-energy fluid to the containment environment, possibly resulting in high containment temperatures and pressures. The quantitative nature of the releases following a steam line rupture is dependent upon the configuration of the plant steam system, the containment design as well as the plant operating conditions and the size of the rupture. This section describes the methods used in determining the containment responses to a variety of postulated pipe breaks encompassing wide variations in plant operation, safety system performance, and break size. The spectrum of breaks analyzed is listed in Table 6.2.1.4-1.

6.2.1.4.1 Significant Parameters Affecting Steam Line Break Mass and Energy Releases

Four major factors influence the release of mass and energy following a steam line break: steam generator fluid inventory, primary-to-secondary heat transfer, protective system operation and the state of the secondary fluid blowdown. The following is a list of those plant variables which have significant influence on the mass and energy releases:

- Plant power level
- Main feedwater system design
- Startup feedwater system design
- Postulated break type, size, and location
- Availability of offsite power
- Safety system failures
- Steam generator reverse heat transfer and reactor coolant system metal heat capacity.

The following is a discussion of each of these variables.

6.2.1.4.1.1 Plant Power Level

Steam line breaks are postulated to occur with the plant in any operating condition ranging from hot shutdown to full power. Since steam generator mass decreases with increasing power level, breaks occurring at lower power generally result in a greater total mass release to the containment. Because of increased energy storage in the primary plant, increased heat transfer in the steam generators and additional energy generation in the nuclear fuel, the energy released to the containment from breaks postulated to occur during power operation may be greater than for breaks occurring with the plant in a hot shutdown condition. Additionally, steam pressure and the dynamic conditions in the steam generators change with increasing power. They have significant influence on both the rate of blowdown and the amount of moisture entrained in the fluid leaving the break following a steam break event.

Because of the opposing effects of changing power level on steam line break releases, no single power level can be identified as a worst case initial condition for a steam line break event. Therefore, several different power levels spanning the operating range as well as the hot shutdown condition are analyzed.

6.2.1.4.1.2 Main Feedwater System Design

The rapid depressurization that occurs following a rupture may result in large amounts of water being added to the steam generators through the main feedwater system. Rapid closing isolation valves are provided in the main feedwater lines to limit this effect. The piping layout downstream of the isolation valves determine the volume in the feedwater lines that cannot be isolated from the steam generators. As the steam generator pressure decreases, some of the fluid in this volume will flash into the steam generator, providing additional secondary fluid that may exit out the rupture.

The feedwater addition that occurs prior to closing of the feedwater line isolation valves influences the steam generator blowdown in several ways. First, the rapid addition increases the amount of entrained water in large-break cases by lowering the bulk quality of the steam generator inventory. Second, because the water entering the steam generator is subcooled, it lowers the steam pressure, thereby reducing the flow rate out the break. Finally, the increased flow rate causes an increase in the heat transfer rate from the primary-to-secondary system, resulting in greater energy being released out the break. Since these are competing effects on the total mass and energy release, no worst case feedwater transient can be defined for all plant conditions. In the results presented, the worst effects of each variable have been used. For example, moisture entrainment for each break is calculated assuming conservatively small feedwater additions so that the entrained water is minimized or zero. Determination of total steam generator inventory is based on conservatively large feedwater additions, as explained in subsection 6.2.1.4.3.2.

The unisolated feedwater line volumes between the steam generator and the isolation valves serve as a source for additional high-energy fluid to be discharged through the pipe break. This volume is accounted for in the mass and energy release data presented in subsection 6.2.1.4.3.2.

6.2.1.4.1.3 Startup Feedwater System Design

Within the first minute following a steam line break, the startup feedwater system may be initiated on any one of several protection system signals. The addition of startup feedwater to the steam generators increases the secondary mass available for release to the containment, as well as the heat transferred to the secondary fluid. The effects on the steam generator mass are maximized in the calculation described in subsection 6.2.1.4.3.2 by assuming full startup feedwater flow to the faulted steam generator starting at time zero from the safeguard system(s) signal or low steam generator level reactor trip and continuing until automatically terminated.

6.2.1.4.1.4 Postulated Break Type, Size and Location

Postulated Break Type

Two types of postulated pipe ruptures are considered in evaluating steam line breaks.

First is a split rupture in which a hole opens at some point on the side of the steam pipe but does not result in a complete severance of the pipe. A single, distinct break area is fed uniformly by both steam generators until steam line isolation occurs. The blowdown flow rates from the individual steam generators are interdependent, since fluid coupling exists between the steam lines. Because flow limiting orifices are provided in each steam generator, the largest split rupture can have an effective area prior to isolation, that is no greater than the throat area of the flow restrictor times the number of steam generators. Following isolation, the effective break area for the steam generator with the broken line can be no greater than the flow restrictor throat area.

The second break type is the double-ended guillotine rupture in which the steam pipe is completely severed and the ends of the break displace from each other. Guillotine ruptures are characterized by two distinct break locations, each of equal area, but being fed by different steam generators. The largest guillotine rupture can have an effective area per steam generator no greater than the throat area of one steam line flow restrictor.

Postulated Break Size

Break area is also important when evaluating steam line breaks. It controls the rate of releases to the containment, and influences the steam pressure decay and the amount of entrained water in the blowdown flow. The data presented in this section include releases for four breaks at each of four initial power levels. Included are three double-ended ruptures and one split rupture, as follows:

- A full double-ended pipe rupture downstream of the steam line flow restrictor. For this case, the actual break area equals the cross-sectional area of the steam line, but the blowdown from the steam generator with the broken line is controlled by the flow restrictor throat area (1.388 square feet nominal). The reverse flow from the intact steam generator is controlled by the smaller of the pipe cross section, the steam stop valve seat area, or the total flow restrictor throat area in the intact steam generator. The reverse flow has been conservatively assumed to be controlled by the flow restrictor in the intact loop steam generator.

- An intermediate size double-ended rupture having an area of 0.4 square feet.

- A small double-ended rupture having an area of 0.1 square feet.

- A split rupture representing the largest break which can not generate a steam line isolation signal from the primary protection equipment. Steam and feedwater line isolation signals are generated by high containment pressure signals for this type of break.

Table 6.2.1.4-1 lists the spectrum of secondary system pipe ruptures analyzed.

Postulated Break Location

Break location affects steam line blowdown due to the pressure losses which occur in the length of piping between the steam generator and the break. The effect of the pressure loss is to reduce the effective break area seen by the steam generator. Although this reduces the rate of blowdown, it would not significantly change the total release of energy to the containment. Therefore, piping loss effects are conservatively ignored in the blowdown results. The release point is conservatively modeled at the maximum elevation of the main steam line piping.

6.2.1.4.1.5 Availability of Offsite Power

The effects of the assumption of the availability of offsite power are enveloped in the analysis.

Offsite power is assumed to be available where it maximizes the mass and energy released from the break because of the following:

- The continued operation of the reactor coolant pumps until automatically tripped as a result of core makeup tank (CMT) actuation. This maximizes the energy transferred from the reactor coolant system to the steam generator.

- The continued operation of the feedwater pumps and actuation of the startup feedwater system until they are automatically terminated. This maximizes the steam generator inventories available for release.

- The AP600 is equipped with the passive safeguards system including the CMT and the passive residual heat removal (PRHR) heat exchanger. Following a steam line rupture, these passive systems are actuated when their setpoints are reached. This decreases the primary coolant temperatures. The actuation and operation of these passive safeguards systems do not require the availability of offsite power.

  When the PRHR is in operation, the core-generated heat is dissipated to the in-containment refueling water storage tank (IRWST) via the PRHR heat exchanger. This causes a reduction of the heat transfer from the primary system to the steam generator secondary system and causes a reduction of mass and energy releases via the break.

Thus, the availability of ac power in conjunction with the passive safeguards system (CMT and PRHR) maximizes the mass and energy releases via the break. Therefore, blowdown occurring in conjunction with the availability of offsite power is more severe than cases where offsite power is not available.

6.2.1.4.1.6 Safety System Failures

In addition to assuming a loss of system pressure, the following single active failures are considered:

- Failure of one main steam isolation valve
- Failure of one main feedwater isolation valve

Initial analyses determined that the main feedwater isolation valve failure is not limiting. The spectrum of cases presented in this section all assume the failure of one main steam isolation valve.

6.2.1.4.1.7 Steam Generator Reverse Heat Transfer and Reactor Coolant System Metal Heat Capacity

Once steam line isolation is complete, the steam generator in the intact steam loop becomes a source of energy that can be transferred to the steam generator with the broken line. This energy transfer occurs through the primary coolant. As the primary plant cools, the temperature of the coolant flowing in the steam generator tubes drops below the temperature of the secondary fluid in the intact unit, resulting in energy being returned to the primary coolant. This energy is then available to be transferred to the steam generator with the broken steam line.

Similarly, the heat stored in the metal of the reactor coolant piping, the reactor vessel, and the reactor coolant pumps is transferred to the primary coolant as the plant cooldown progresses. This energy also is available to be transferred to the steam generator with the broken line.

The effects of both the reactor coolant system metal and the reverse steam generator heat transfer are included in the results presented.

6.2.1.4.2 Description of Blowdown Model

The steamline blowdown is calculated with the AP600 version of LOFTRAN (Reference 31). This is a version of LOFTRAN (Reference 6) which has been modified to include simulation of the AP600 passive residual heat removal heat exchanger, core makeup tanks, and associated protection and safety monitoring system actuation logic. Documentation of the code changes for the passive models is provided in Reference 31. The methodology for the steamline break analysis is based on Reference 5.

6.2.1.4.3 Containment Response Analysis

The WGOTHIC Computer Code (Reference 20) is used to determine the containment responses following the steam line break. The containment response analysis is described in subsection 6.2.1.1.

6.2.1.4.3.1 Initial Conditions

The initial containment conditions are discussed in subsection 6.2.1.1.3.

6.2.1.4.3.2 Mass and Energy Release Data

Using References 5, 6 and 31 as a basis, mass and energy release data are developed to determine the containment pressure-temperature response for the spectrum of breaks analyzed. Tables 6.2.1.4-2 and 6.2.1.4-3 provide the mass and energy release data for the cases that produce the highest containment pressure and temperature in the containment response analysis. Table 6.2.1.4-4 provides plant data used for the cases used in the mass and energy releases determination.

The rate of startup feedwater addition represents the maximum flow rate limited by a cavitating venturi to a fully depressurized steam generator. Actual isolation is dependent on signals generated by the protection and safety monitoring system. Feedwater isolation for the split breaks is based on the time required to reach the containment pressure setpoint that generates the isolation signal. The feedwater flow rates before automatic isolation assumed in the analyses are based on input for the AP600 steam generator and main feed system design.

6.2.1.4.3.3 Containment Pressure-Temperature Results

The results of the containment pressure-temperature analyses for the postulated secondary system pipe ruptures that produce the highest peak containment pressure and temperature are presented in subsection 6.2.1.1.3.

6.2.1.5 Minimum Containment Pressure Analysis for Performance Capability Studies of Emergency Core Cooling System (PWR)

The containment backpressure used for the AP600 cold leg guillotine and split breaks for the emergency core cooling system (ECCS) analysis presented in subsection 15.6.5 is described. The minimum containment backpressure for emergency core cooling system performance during a loss-of-coolant accident is computed using the WGOTHIC computer code. Subsection 6.2.1.1 demonstrates that the AP600 containment pressurizes during large break LOCA events. An analysis is performed to establish a containment pressure boundary condition applied to the WCOBRA/TRAC code (Reference 8). A single-node containment model is used to assess containment pressure response. Containment internal heat sinks used heat transfer correlations of 4 times Tagami during the blowdown phase followed by 1.2 times Uchida for the post-blowdown phase. The calculated containment backpressure is provided in Figure 6.2.1.5-1. Results of the WCOBRA/TRAC analyses demonstrate that the AP600 meets 10 CFR 50.46 requirements (Reference 7).

6.2.1.5.1 Mass and Energy Release Data

The mass and energy releases to the containment during the blowdown portion only of the double-ended cold-leg guillotine break (DECLG) transient are presented in Table 6.2.1.5-1, as computed by the WCOBRA/TRAC code.

The mathematical models which calculate the mass and energy releases to the containment are described in subsection 15.6.5. A break spectrum analysis is performed (see references in subsection 15.6.5) that considers various break sizes and Moody discharge coefficients for the double-ended cold leg guillotines and splits. Mixing of steam and accumulator water injected into the vessel reduces the available energy released to the containment vapor space, thereby minimizing calculated containment pressure. Note that the mass/energy releases during the reflood phase of the subject break are not considered. This produces a conservatively low containment pressure result for use as a boundary condition in the WCOBRA/TRAC large break LOCA analysis.

6.2.1.5.2 Initial Containment Internal Conditions

Initial containment conditions were biased for the emergency core cooling system backpressure analysis to predict a conservatively low containment backpressure. Initial containment conditions include an initial pressure of 14.7 psia, initial containment temperature of 90°F, and a relative humidity of 99 percent. An air annulus temperature of 0°F is assumed. The initial through-thickness metal temperature of the containment shell is assumed to also be 0°F.

6.2.1.5.3 Other Parameters

Containment parameters, such as containment volume and passive heat sinks, are biased to predict a conservative low containment backpressure. The containment volume used in the calculation is conservatively set to 1.1 times the free volume of the AP600 containment Evaluation Model. Passive heat sink surface areas were increased by a factor of 2.1 times the values presented in Reference 20. Material properties were biased high (density, conductivity, and heat capacity) as indicated in CSB 6-1 (Reference 8). No air gap was modeled between the steel liner and base concrete of jacketed concrete heat sinks. The outside surface of the containment shell was maintained at 0°F throughout the calculation. To further minimize containment pressure, containment purge was assumed to be in operation at time zero and air is vented through both the 15-inch diameter (16-inch, Sch. 40 piping) containment purge supply and exhaust lines until the isolation valves have fully closed. These valves were modeled to close 22 seconds after the 8 psig closure setpoint was reached.

6.2.1.6 Testing and Inspection

This section describes the functional testing of the containment vessel. Testing and in-service inspection of the containment vessel are described in subsection 3.8.2.6. Isolation testing and leak testing are described in subsection 6.2.5. Testing and inspection are consistent with regulatory requirements and guidelines.

The valves of the passive containment cooling system are stroke tested periodically. Subsection 6.2.2 provides a description of testing and inspection.

The baffle between the containment vessel and the shield building is equipped with removable panels and clear observation panels to allow for inspection of the containment surface. See subsection 3.8.2 for the requirements for in-service inspection of the steel containment vessel. Subsection 6.2.2 provides a description of testing and inspection to be performed.

Testing is not required on any subcompartment vent or on the collection of condensation from the containment shell. The collection of condensate from the containment shell and its use in leakage detection are discussed in subsection 5.2.5.

6.2.1.7 Instrumentation Requirements

Instrumentation is provided to monitor the conditions inside the containment and to actuate the appropriate engineered safety features, should those conditions exceed the predetermined levels. The instruments measure the containment pressure, containment atmosphere radioactivity, and containment hydrogen concentration. Instrumentation to monitor reactor coolant system leakage into containment is described in subsection 5.2.5.

The containment pressure is measured by four independent pressure transmitters. The signals are fed into the engineered safety features actuation system, as described in subsection 7.3.1. Upon detection of high pressure inside the containment, the appropriate safety actuation signals are generated to actuate the necessary safety-related systems. Low pressure is alarmed but does not actuate the safety-related systems.

The physically separated pressure transmitters are located inside the containment. Section 7.3 provides a description.

The containment atmosphere radiation level is monitored by four independent area monitors located above the operating deck inside the containment building. The measurements are continuously fed into the engineered safety features actuation system logic. Section 11.5 provides information on the containment area radiation monitors. The engineered safety features actuation system operation is described in Section 7.3.

The containment hydrogen concentration is measured by hydrogen monitors, as described in subsection 6.2.4. Hydrogen concentrations are monitored by sensors distributed throughout containment to provide a representative indication of containment hydrogen concentration.

The sensors also indicate the specific areas evaluated for potential hydrogen pockets. These indications are used by the plant operators to control ignitors and monitor hydrogen concentrations. High hydrogen concentration is alarmed in the main control room.

6.2.2 Passive Containment Cooling System

The passive containment cooling system (PCS) is an engineered safety features system. Its functional objective is to reduce the containment temperature and pressure following a loss of coolant accident (LOCA) or main steam line break (MSLB) accident inside the containment by removing thermal energy from the containment atmosphere. The passive containment cooling system also serves as the means of transferring heat to the safety-related ultimate heat sink for other events resulting in a significant increase in containment pressure and temperature.

The passive containment cooling system limits releases of radioactivity (post-accident) by reducing the pressure differential between the containment atmosphere and the external environment, thereby diminishing the driving force for leakage of fission products from the containment to the atmosphere. This subsection describes the safety design bases of the safety-related containment cooling function. Nonsafety-related containment cooling, a function of the containment recirculation cooling system, is described in subsection 9.4.6.

The passive containment cooling system also provides a source of makeup water to the spent fuel pool in the event of a prolonged loss of normal spent fuel pool cooling.

6.2.2.1 Safety Design Basis

- The passive containment cooling system is designed to withstand the effects of natural phenomena such as ambient temperature extremes, earthquakes, winds, tornadoes, or floods.

- Passive containment cooling system operation is automatically initiated upon receipt of a Hi-2 containment pressure signal.

- The passive containment cooling system is designed so that a single failure of an active component, assuming loss of offsite or onsite ac power sources, will not impair the capability of the system to perform its safety-related function.

- Active components of the passive containment cooling system are capable of being tested during plant operation. Provisions are made for inspection of major components in accordance with the intervals specified in the ASME Code, Section XI.

- The passive containment cooling system components required to mitigate the consequences of an accident are designed to remain functional in the accident environment and to withstand the dynamic effects of the accident.

- The passive containment cooling system is capable of removing sufficient thermal energy including subsequent decay heat from the containment atmosphere following a design basis event resulting in containment pressurization such that the containment pressure remains below the design value with no operator action required for 72 hours.

- The passive containment cooling system is designed and fabricated to appropriate codes consistent with Regulatory Guides 1.26 and 1.32 and in accordance with Regulatory Guide 1.29 as described in Section 1.9.

6.2.2.2 System Design

6.2.2.2.1 General Description

The passive containment cooling system and components are designed to the codes and standards identified in Section 3.2; flood design is described in Section 3.4; missile protection is described in Section 3.5. Protection against dynamic effects associated with the postulated rupture of piping is described in Section 3.6. Seismic and environmental design and equipment qualification are described in Sections 3.10 and 3.11. The actuation system is described in Section 7.3.

6.2.2.2.2 System Description

The passive containment cooling system is a safety-related system which is capable of transferring heat directly from the steel containment vessel to the environment. This transfer of heat prevents the containment from exceeding the design pressure and temperature following a postulated design basis accident, as identified in Chapters 6 and 15. The passive containment cooling system makes use of the steel containment vessel and the concrete shield building surrounding the containment. The major components of the passive containment cooling system are: the passive containment cooling water storage tank (PCCWST) which is incorporated into the shield building structure above the containment; an air baffle, located between the steel containment vessel and the concrete shield building, which defines the cooling air flowpath; air inlets and an air exhaust, also incorporated into the shield building structure; and a water distribution system, mounted on the outside surface of the steel containment vessel, which functions to distribute water flow on the containment. A passive containment cooling ancillary water storage tank and two recirculation pumps are provided for onsite storage of additional passive containment cooling system cooling water, to transfer the inventory to the passive containment cooling water storage tank, and to provide a back-up supply to the fire protection system (FPS) seismic standpipe system as discussed in subsection 9.5.1.

A normally isolated, manually-opened flow path is available between the passive containment cooling system water storage tank and the spent fuel pool.

A recirculation path is provided to control the passive containment cooling water storage tank water chemistry and to provide heating for freeze protection. Passive containment cooling water storage tank filling operations and normal makeup needs are provided by the demineralized water transfer and storage system discussed in subsection 9.2.4.

The system piping and instrumentation diagram is shown in Figure 6.2.2-1. System parameters are shown in Table 6.2.2-1. A simplified system sketch is included as Figure 6.2.2-2.

6.2.2.2.3 Component Description

The mechanical components of the passive containment cooling system are described in this subsection. Table 6.2.2-2 provides the component design parameters.

Passive Containment Cooling Water Storage Tank - The passive containment cooling water storage tank is incorporated into the shield building structure above the containment vessel. The inside wetted walls of the tank are lined with stainless steel plate. It is filled with demineralized water and has a useable volume of greater than 531,000 gallons for passive containment cooling functions. The passive containment cooling system functions as the safety-related ultimate heat sink. The passive containment cooling water storage tank is seismically designed and missile protected.

The surrounding reinforced concrete supporting structure is designed to ACI 349 as described in subsection 3.8.4.3. The welded seams of the plates forming part of the leak tight boundary are examined by liquid penetrant after fabrication to confirm that the boundary does not leak.

The tank also has redundant level measurement channels and alarms for monitoring the tank water level and redundant temperature measurement channels to monitor and alarm for potential freezing. To maintain system operability, a recirculation loop that provides chemistry and temperature control is connected to the tank.

The tank is constructed to provide sufficient thermal inertia and insulation such that draindown can be accomplished without heater operation.

In addition to its containment heat removal function, the passive containment cooling water storage tank also serves as a source of makeup water to the spent fuel pool and a seismic Category I water storage reservoir for fire protection following a safe shutdown earthquake.

The PCCWST suction pipe for the fire protection system is configured so that actuation of the fire protection system will not infringe on the 531,000 gallons volume allocated to the passive containment cooling function.

Passive Containment Cooling Water Storage Tank Isolation Valves - The passive containment cooling system water storage tank outlet piping is equipped with two sets of redundant isolation valves. The air-operated butterfly valves are normally closed and open upon receipt of a Hi-2 containment pressure signal. These valves fail-open, providing a fail-safe position, on the loss of air or loss of 1E dc power. The normally-open motor-operated gate valves are located upstream of the butterfly valves. They are provided to allow for testing or maintenance of the butterfly valves.

The storage tank isolation valves, along with the passive containment cooling water storage tank discharge piping and associated instrumentation between the passive containment cooling water storage tank and the downstream side of the isolation valves, are contained within a temperature-controlled valve room to prevent freezing. Valve room heating is provided to maintain the room temperature above 50°F.

Flow Control Orifices - Orifices are installed in each of the four passive containment cooling water storage tank outlet pipes. They are used, along with the different elevations of the outlet pipes, to control the flow of water from the passive containment cooling water storage tank as a function of water level. The orifices are located within the temperature-controlled valve room.

Water Distribution Bucket - A water distribution bucket is provided to deliver water to the outer surface of the containment dome. The redundant passive containment cooling water delivery pipes and auxiliary water source piping discharge into the bucket, below its operational water level, to prevent excessive splashing. A set of circumferentially spaced distribution slots are included around the top of the bucket. The bucket is hung from the shield building roof and suspended just above the containment dome for optimum water delivery. The structural requirements for safety-related structural steel identified in subsection 3.8.4 apply to the water distribution bucket. ANSI/ASCE-8-90 (Reference 24) is used for design and analysis of stainless steel cold formed parts. The water distribution bucket is fabricated from one or more of the materials included in Table 3.8.4-6, ASTM-A240 austenitic stainless steel, or ASTM-A276 austenitic stainless steel.

Water Distribution Weir System - A weir-type water delivery system is provided to optimize the wetted coverage of the containment shell during passive containment cooling system operation. The water delivered to the center of the containment dome by the water distribution bucket flows over the containment dome, being distributed evenly by slots in the distribution bucket. Vertical divider plates are attached to the containment dome and originate at the distribution bucket extending radially along the surface of the dome to the first distribution weir. The divider plates limit maldistribution of flow which might otherwise occur due to variations in the slope of the containment dome. At the first distribution weir set, the water in that sector is collected and then redistributed onto the containment utilizing channeling walls and collection troughs equipped with distribution weirs. A second set of weirs are installed on the containment dome at a greater radius to again collect and then redistribute the cooling water to enhance shell coverage. The system includes channeling walls and collection troughs, equipped with distribution weirs. The distribution system is capable of functioning during extreme low- or high-ambient temperature conditions. The structural requirements for safety-related structural steel and cold formed steel structures identified in subsection 3.8.4 apply to the water distribution weir system. ANSI/ASCE-8-90, (Reference 24) is used for design and analysis of stainless steel cold formed parts. The water distribution weir system is fabricated from one or more of the materials included in Table 3.8.4-6, ASTM-A240 austenitic stainless steel, or ASTM-A276 austenitic stainless steel.

Air Flow Path - An air flow path is provided to direct air along the outside of the containment shell to provide containment cooling. The air flow path includes a screened shield building inlet, an air baffle that divides the outer and inner flow annuli, and a chimney to increase buoyancy. Subsection 3.8.4.1.3 includes information regarding the air baffle. The general arrangement drawings provided in Section 1.2 provide layout information of the air flow path.

Passive Containment Cooling Ancillary Water Storage Tank – The passive containment cooling ancillary water storage tank is a cylindrical steel tank located at ground level near the auxiliary building. It is filled with demineralized water and has a useable volume of greater than 400,000 gallons for makeup to the passive containment cooling water storage tank. The tank is analyzed, designed and constructed using the method and criteria for Seismic Category II building structures defined in subsections 3.2.1 and 3.7.2. The tank is designed and analyzed for Category 5 hurricanes including the effects of sustained winds, maximum gusts, and associated wind-borne missiles.

The tank has a level measurement, an alarm for monitoring the tank water level and a temperature measurement channel to monitor and alarm for potential freezing. To maintain system operability, an internal heater, controlled by the temperature instrument, is provided to maintain water contents above freezing. Chemistry can be adjusted by passive containment cooling water storage tank recirculation loop.

The tank is insulated to assure sufficient thermal inertia of the contents is available to prevent freezing for 7 days without heater operation. The transfer piping is maintained dry also to preclude freezing.

Chemical Addition Tank - The chemical addition tank is a small, vertical, cylindrical tank that is sized to inject a solution of hydrogen peroxide to maintain a passive containment cooling water storage tank concentration for control of algae growth.

Recirculation Pumps - Each recirculation pump is a 100 percent capacity centrifugal pump with wetted components made of austenitic stainless steel. The pump is sized to recirculate the entire volume of PCCWST water once every week. Both pumps are operated in parallel to meet fire protection system requirements.

Recirculation Heater - The recirculation heater is provided for freeze protection. The heater is sized based on heat losses from the passive containment cooling water storage tank and recirculation piping at the minimum site temperature, as defined in Section 2.3.

6.2.2.2.4 System Operation

Operation of the passive containment cooling system is initiated upon receipt of two out of four Hi-2 containment pressure signals. Manual actuation by the operator is also possible from either the main control room or remote shutdown workstation. System actuation consists of opening the passive containment cooling water storage tank isolation valves. This allows the passive containment cooling water storage tank water to be delivered to the top, external surface of the steel containment shell. The flow of water, provided entirely by the force of gravity, forms a water film over the dome and side walls of the containment structure.

The flow of water to the containment outer surface is initially established at approximately 440 gpm for short-term containment cooling following a design basis loss of coolant accident. The flow rate is reduced over a period of 72 hours to a value of approximately 62.7 gpm. This flow provides the desired reduction in containment pressure over time and removes decay heat. The flow rate change is dependent only upon the decreasing water level in the passive containment cooling water storage tank. Prior to 72 hours after the event, operator actions are taken to align the passive containment ancillary water storage tank to the suction of the passive containment cooling system recirculation pumps to replenish the cooling water supply to the passive containment cooling water storage tank. Sufficient inventory is available within the passive containment cooling ancillary water storage tank to maintain the 62.7 gpm flow rate for an additional 4 days.

To adequately wet the containment surface, the water is delivered to the distribution bucket above the center of the containment dome which subsequently delivers the water to the containment surface. A weir-type water distribution system is used on the dome surface to distribute the water for effective wetting of the dome and vertical sides of the containment shell. The weir system contains radial arms and weirs located considering the effects of tolerances of the containment vessel design and construction. A corrosion-resistant paint or coating for the containment vessel is specified to enhance surface wetability and film formation.

The cooling water not evaporated from the vessel wall flows down to the bottom of the inner containment annulus into annulus drains. The redundant annulus drains route the excess water out of the upper annulus. The annulus drains are located in the shield building wall slightly above the floor level to minimize the potential for clogging of the drains by debris. The drains are horizontal or have a slight slope to promote drainage. The drains are always open (without isolation valves) and each is sized to accept maximum passive containment cooling system flow. The outside ends of the drains are located above catch basins or other storm drain collectors.

A path for the natural circulation of air upward along the outside walls of the containment structure is always open. The natural circulation air flow path begins at the shield building inlet, where atmospheric air enters horizontally through openings in the concrete structure. Air flows past a set of fixed louvers and is forced to turn 90 degrees downward into an outer annulus. This outer shield building annulus is encompassed by the concrete shield building on the outside and a removable baffle on the inside. At the bottom of the baffle wall, curved vanes aid in turning the flow upward 180 degrees into the inner containment annulus. This inner annulus is encompassed by the baffle wall on the outside and the steel containment vessel on the inside. Air flows up through the inner annulus to the top of the containment vessel and then exhausts through the shield building chimney.

As the containment structure heats up in response to high containment temperature, heat is removed from within the containment via conduction through the steel containment vessel, convection from the containment surface to the water film, convection and evaporation from the water film to the air, and radiation from the water film to the air baffle. As heat and water vapor are transferred to the air space between the containment structure and air baffle, the air becomes less dense than the air in the outer annulus. This density difference causes an increase in the natural circulation of the air upward between the containment structure and the air baffle, with the air finally exiting at the top center of the shield building.

The passive containment cooling water storage tank provides water for containment wetting for 72 hours following system actuation. Operator action can be taken to replenish this water supply from the passive containment cooling ancillary water storage tank or to provide an alternate water source directly to the containment shell through an installed safety-related seismic piping connection. In addition, water sources used for normal filling operations can be used to replenish the water supply.

The arrangement of the air inlet and air exhaust in the shield building structure has been selected so that wind effects aids the natural air circulation. The air inlets are placed at the top, outside of the shield building, providing a symmetrical air inlet that reduces the effect of wind speed and direction or adjacent structures. The air/water vapor exhaust structure is elevated above the air inlet to provide additional buoyancy and reduces the potential of exhaust air being drawn into the air inlet. The air flow inlet and chimney regions are both designed to protect against ice or snow buildup and to prevent foreign objects from entering the air flow path.

Inadvertent actuation of the passive containment cooling system is terminated through operator action by closing either of the series isolation valves from the main control room. Subsection 6.2.1.1.4 provides a discussion of the effects of inadvertent system actuation.

The passive containment cooling system provides for makeup water to the spent fuel pool to provide for continued spent fuel pool inventory and heat removal. The passive containment cooling water storage tank provides makeup to the spent fuel pool when the inventory is not required for passive containment cooling system operation. An installed long term makeup connection for the passive containment cooling system and the spent fuel pool is provided as a part of the passive containment cooling system. The passive containment cooling ancillary water storage tank and the passive containment cooling system recirculation pumps may also be utilized for makeup to the spent fuel pool.

6.2.2.3 Safety Evaluation

The safety-related portions of the passive containment cooling system are located within the shield building structure. This building (including the safety-related portions of the passive containment cooling system) is designed to withstand the effects of natural phenomena such as earthquakes, winds, tornadoes, or floods. Components of the passive containment cooling system are designed to withstand the effects of ambient temperature extremes.

The portions of the passive containment cooling system which provide for long term (post 72-hour) water supply for containment wetting are located in Seismic Category I or Seismic Category II structures excluding the passive containment ancillary water storage tank and associated valves located outside of the auxiliary building. The water storage tank and the anchorage for the associated valves are Seismic Category II. The features of these structures which protect this function are analyzed and designed for Category 5 hurricanes including the effects of sustained winds, maximum gusts, and associated wind-borne missiles.

Operation of the containment cooling system is initiated automatically following the receipt of a Hi-2 containment pressure signal. The use of this signal provides for system actuation during transients, resulting in mass and energy releases to containment, while avoiding unnecessary actuations. System actuation requires the opening of either isolation valve, with no other actions required to initiate the post-accident heat removal function since the cooling air flow path is always open. Operation of the passive containment cooling system may also be initiated from the main control room and from the remote shutdown work station. A description of the actuation system is contained in Section 7.3.

The active components of the passive containment cooling system, the isolation valves, are located in two redundant pipe lines. Failure of a component in one train does not affect the operability of the other mechanical train or the overall system performance. The fail-open, air-operated valves require no electrical power to move to their safe (open) position. The normally open motor-operated valves are powered from separate redundant Class 1E dc power sources. Table 6.2.2-3 presents a failure modes and effects analysis of the passive containment cooling system.

Capability is provided to periodically test actuation of the passive containment cooling system. Active components can be tested periodically during plant operation to verify operability. The system can be inspected during unit shutdown. Additional information is contained in subsections 3.9.6 and 6.2.2.4, as well as in the Technical Specifications.

The passive containment cooling system components located inside containment, the containment pressure sensors, are tested and qualified to perform in a simulated design basis accident environment. These components are protected from effects of postulated jet impingement and pipe whip in case of a high-energy line break.

The containment pressure analyses are based on an ambient air temperature of 115°F dry bulb and 80°F coincident wet bulb. The passive containment cooling water storage tank water temperature basis is 120°F. Results of the analyses are provided in subsection 6.2.1.

6.2.2.4 Testing and Inspection

6.2.2.4.1 Inspections

The passive containment cooling system is designed to permit periodic testing of system readiness as specified in the Technical Specifications.

The portions of the passive containment cooling system from the isolation valves to the passive containment cooling water storage tank are accessible and can be inspected during power operation or shutdown for leaktightness. Examination and inspection of the pressure retaining piping welds is performed in accordance with ASME Code, Section XI. The design of the containment vessel and air baffle retains provisions for the inspection of the vessel during plant shutdowns.

6.2.2.4.2 Preoperational Testing

Preoperational testing of the passive containment cooling system is verified to provide adequate cooling of the containment. The flow rates are confirmed at the minimum initial tank level, an intermediate step with all but one standpipe delivering flow and at a final step with all but two standpipes delivering to the containment shell. The flow rates are measured utilizing the differential pressure across the orifices within each standpipe and will be consistent with the following minimum flow rates:

- 442 gpm at the minimum operating water level
- 123.5 gpm at a level after the first standpipe is uncovered
- 72.5 gpm at a level after the second standpipe is uncovered

The containment coverage will be measured at the base of the upper annulus in addition to the coverage at the spring line for the full flow case using the PCS water storage tank delivering to the containment shell and a lower flow case with both PCS recirculation pumps delivering to the containment shell. For the low flow case, a throttle valve is used to obtain a low flow rate less than the full capacity of the PCS recirculation pumps. This flow rate is then re-established for subsequent tests using the throttle vlave. These benchmark values will be used to devleop acceptance criteria for the Technical Specifications. The full flow condition is selected since it is the most important flow rate from the standpoint of peak containment pressure and the lower flow rate is selected to verify wetting characteristics at less than full flow conditions.

The standpipe elevations are verified to be at the values specified in Table 6.2.2-2.

The inventory within the tank is verified to provide 72 hours of operation from the minimum initial operating water level with a minimum flow rate over the duration in excess of 62.7 gpm. The flow rates are measured utilizing the differential pressure across the orifices within each standpipe.

The containment vessel exterior surface is verified to be coated with an inorganic zinc coating.

The passive containment cooling air flow path will be verified at the following locations:

- Air inlets
- Base of the outer annulus
- Base of the inner annulus
- Discharge structure

With either a temporary water supply or the passive containment cooling ancillary water storage tank connected to the suction of the recirculation pumps and with either of the two pumps operating, the flow rate to the passive containment cooling water storage tank will be in excess of 62.7gpm. Temporary instrumentation or changes in the passive containment cooling water storage tank level will be utilized to verify the flow rates. The capacity of the passive containment cooling ancillary water storage tank is verified to be adequate to supply 62.7 gpm for a duration of 4 days.

The passive containment cooling water storage tank provides makeup water to the spent fuel pool. When aligned to the spent fuel pool the flow rate is verified to exceed 50 gpm. Installed instrumentation will be utilized to verify the flow rate. The volume of the passive containment cooling water storage tank is verified to exceed 400,000 gallons.

Additional details for preoperational testing of the passive containment cooling system are provided in Chapter 14.

6.2.2.4.3 Operational Testing

Operational testing is performed to:

- Demonstrate that the sequencing of valves occurs on the initiation of Hi-2 containment pressure and demonstrate the proper operation of remotely operated valves.
- Verify valve operation during plant operation. The normally open motor-operated valves, in series with each normally closed air-operated isolation valve, are temporarily closed. This closing permits isolation valve stroke testing without actuation of the passive containment cooling system.
- Verify water flow delivery, consistent with the accident analysis.
- Verify visually that the path for containment cooling air flow is not obstructed by debris or foreign objects.
- Test frequency is consistent with the plant technical specifications (subsection 16.3.6) and inservice testing program (subsection 3.9.6).

6.2.2.5 Instrumentation Requirements

The status of the passive containment cooling system is displayed in the main control room. The operator is alerted to problems with the operation of the equipment within this system during both normal and post-accident conditions.

Normal operation of the passive containment cooling system is demonstrated by monitoring the recirculation pump discharge pressure, flow rate, water storage tank levels and temperatures, and valve room temperature. Post-accident operation of the passive containment cooling system is demonstrated by monitoring the passive containment cooling water storage tank level, passive containment cooling system cooling water flow rate, containment pressure and external cooling air discharge temperature.

The information on the activation signal-generating equipment is found in Chapter 7.

The protection and safety monitoring system providing system actuation is discussed in Chapter 7.

6.2.3 Containment Isolation System

The major function of the containment isolation system of the AP600 is to provide containment isolation to allow the normal or emergency passage of fluids through the containment boundary while preserving the integrity of the containment boundary, if required. This prevents or limits the escape of fission products that may result from postulated accidents. Containment isolation provisions are designed so that fluid lines which penetrate the primary containment boundary are isolated in the event of an accident. This minimizes the release of radioactivity to the environment.

The containment isolation system consists of the piping, valves, and actuators that isolate the containment. The design of the containment isolation system satisfies the requirements of NUREG 0737, as described in the following paragraphs.

6.2.3.1 Design Basis

6.2.3.1.1 Safety Design Basis

A. The containment isolation system is protected from the effects of natural phenomena, such as earthquakes, tornadoes, hurricanes, floods, and external missiles (General Design Criterion 2).

B. The containment isolation system is designed to remain functional after a safe shutdown earthquake (SSE) and to perform its intended function following the postulated hazards of fire, internal missiles, or pipe breaks (General Design Criteria 3 and 4).

C. The containment isolation system is designed and fabricated to codes consistent with the quality group classification, described in Section 3.2. Conformance with Regulatory Guide 1.26, 1.29, and 1.32 is described in subsection 1.9.

D. The containment isolation system provides isolation of lines penetrating the containment for design basis events requiring containment integrity.

E. Upon failure of a main steam line, the containment isolation system isolates the steam generators as required to prevent excessive cooldown of the reactor coolant system or overpressurization of the containment.

F. The containment isolation system is designed in accordance with General Design Criterion 54.

G. Each line that penetrates the containment that is either a part of the reactor coolant pressure boundary or that connects directly to the containment atmosphere, and does not meet the requirements for a closed system (as defined in paragraph H below), is provided with containment isolation valves according to General Design Criteria 55 and 56.

H. Each line that penetrates the containment, that is neither part of the reactor coolant pressure boundary nor connected directly to the atmosphere of the containment, and that satisfies the requirements of a closed system is provided with a containment isolation valve according to General Design Criterion 57. A closed system is not a part of the reactor coolant pressure boundary and is not connected directly to the atmosphere of the containment. A closed system also meets the following additional requirements:

- The system is protected against missiles and the effects of high-energy line break.

- The system is designed to Seismic Category I requirements.
- The system is designed to ASME Code, Section III, Class 2 requirements.
- The system is designed to withstand temperatures at least equal to the containment design temperature.
- The system is designed to withstand the external pressure from the containment structural acceptance test.
- The system is designed to withstand the design basis accident transient and environment.

I. The containment isolation system is designed so that no single failure in the containment isolation system prevents the system from performing its intended functions.

J. Fluid penetrations supporting the engineered safety features functions have remote manual isolation valves. These valves can be closed from the main control room or from the remote shutdown workstation, if required.

K. The containment isolation system is designed according to 10 CFR 50.34, so that the resetting of an isolation signal will not cause any valve to change position.

6.2.3.1.2 Power Generation Design Basis

The containment isolation system has no power generation design basis. Power generation design bases associated with individual components of the containment isolation system are discussed in the section describing the system of which they are an integral part.

6.2.3.1.3 Additional Requirements

The AP600 containment isolation system is designed to meet the following additional requirements:

A. The containment isolation elements are designed to minimize the number of isolation valves which are subject to Type C tests of 10 CFR 50, Appendix J. Specific requirements are the following:

- The number of pipe lines which provide a direct connection between the inside and outside of primary containment during normal operation are minimized.
- Closed systems outside of containment that may be open to the containment atmosphere during an accident are designed for the same conditions as the containment itself, and are testable during Type A leak tests.

- The total number of penetrations requiring isolation valves are minimized by appropriate system design. For example:

  - In the component cooling system, a single header with branch lines inside of containment is employed instead of providing a separate penetration for each branch line.

  - Consistent with other considerations, such as containment arrangement and exposure of essential safety equipment to potentially harsh environments, the equipment is located inside and outside of containment so as to require the smallest number of penetrations.

- Consistent with current practice, Type C testing is not required for pressurized water reactor main steam, feedwater, startup feedwater, or steam generator blowdown isolation valves. The steam generator tubes are considered to be a suitable boundary to prevent release of radioactivity from the reactor coolant system following an accident. The steam generator shell and pipe lines, up to and including the first isolation valve, are considered a suitable boundary to prevent release of containment radioactivity.

B. Personnel hatches, equipment hatches, and the fuel transfer tube are sealed by closures with double gaskets.

C. Containment isolation is actuated on a two-out-of-four logic from within the protection and safety monitoring system. The safeguards signals provided to each isolation valve are selected to enhance plant safety. Provisions are provided for manual containment isolation from the main control room.

D. Penetration lines with automatic isolation valves are isolated by engineered safety features actuation signals.

E. Isolation valves are designed to provide leaktight service against the medium to which the valves are exposed in the short and long-term course of any accident. For example, a valve is gas-tight if the valve is exposed to the containment atmosphere.

F. Isolation valves are designed to have the capacity to close against the conditions that may exist during events requiring containment isolation.

G. Isolation valve closure times are designed to limit the release of radioactivity to within regulation and are consistent with standard valve operators, except where a shorter closure time is required.

H. The position of each power-operated isolation valve (fully closed or open), whether automatic or remote manual, is indicated in the main control room and is provided as input to the plant computer. Such position indication is based on actual valve position, for example, by a limit switch which directly senses the actual valve stem position, rather than demanded valve position.

I. Normally closed manual containment isolation valves have provisions for locking the valves closed. Locking devices are designed such that the valves can be locked only in the fully closed position. Administrative control provides verification that manual isolation valves are maintained locked closed during normal operation. Position locks provide confidence that valves are placed in the correct position prior to locking.

J. Automatic containment isolation valves are powered by Class 1E dc power. Air-operated valves fail in the closed position upon loss of a support system, such as instrument air or electric power.

K. Valve alignments used for fluid system testing during operation are designed so that either: containment bypass does not occur during testing, assuming a single failure; or exceptions are identified, and remotely operated valves provide timely isolation from the control room. Containment isolation provisions can be relaxed during system testing. The intent of the design is to provide confidence that operators are aware of any such condition and have the capability to restore containment integrity.

L. A diverse method of initiating closure is provided for those containment isolation valves associated with penetrations representing the highest potential for containment bypass. Diverse actuation is discussed in Section 7.7.

M. Containment penetrations with leaktight barriers, both inboard and outboard, are designed to limit pressure excursion between the barriers due to heatup of fluid between the barriers. The penetration will either be fitted with relief or check valves to relieve internal pressure or one of the valves has been designed or oriented to limit pressures to an acceptable value. For example, a penetration which incorporates two air-operated globe valves – one of the globe valves will be oriented such that pressure between the two valves will lift the plug from the seat to relieve the pressure, then reseat.

6.2.3.2 System Description

6.2.3.2.1 General Description

Piping systems penetrating the containment have containment isolation features. These features serve to minimize the release of fission products following a design basis accident. SRP Section 6.2.4 provides acceptable alternative arrangements to the explicit arrangements given in General Design Criteria 55, 56 and 57. Table 6.2.3-1 lists each penetration and provides a summary of the containment isolation characteristics. The Piping and Instrumentation Diagrams of the applicable systems show the functional arrangement of the containment penetration, isolation valves, test and drain connections. Section 1.7 contains a list of the Piping and Instrumentation Diagrams.

As discussed in subsection 6.2.3.1, the AP600 containment isolation design satisfies the NRC requirements including post-Three Mile Island requirements. Two barriers are provided -- one inside containment and one outside containment. Usually these barriers are valves, but in some cases they are closed piping systems not connected to the reactor coolant system or to the containment atmosphere.

The AP600 has fewer mechanical containment penetrations (including hatches) and a higher percentage of normally closed isolation valves than current plants. The majority of the penetrations that are normally open incorporate fail closed isolation valves that close automatically with the loss of support systems such as instrument air. Table 6.2.3-1 lists the AP600 containment mechanical penetrations and the isolation valves associated with them. Provisions for leak testing are discussed in subsection 6.2.5.

For those systems having automatic isolation valves or for those provided with remote-manual isolation, subsection 6.2.3.5 describes the power supply and associated actuation system. Power-operated (air, motor, or pneumatic) containment isolation valves have position indication in the main control room.

The actuation signal that occurs directly as a result of the event initiating containment isolation is designated in Table 6.2.3-1. If a change in valve position is required at any time following primary actuation, a secondary actuation signal is generated which places the valve in an alternative position. The closure times for automatic containment isolation valves are provided in Table 6.2.3-1.

The containment air filtration system is used to purge the containment atmosphere of airborne radioactivity during normal plant operation, as described in subsection 9.4.7. The system is designed in accordance with Branch Technical Position CSB 6-4 using 16-inch supply and exhaust lines and containment isolation valves. These valves close automatically on a containment isolation signal.

Section 3.6 describes dynamic effects of pipe rupture. Section 3.5 discusses missile protection, and Section 3.8 discusses the design of Category I structures including any structure used as a protective device. Lines associated with those penetrations that are considered closed systems inside the containment are protected from the effects of a pipe rupture and missiles. The actuators for power-operated isolation valves inside the containment are either located above the maximum containment water level or in a normally nonflooded area. The actuators are designed for flooded operation or are not required to function following containment isolation and designed and qualified not to spuriously open in a flooded condition.

Other defined bases for containment isolation are provided in SRP Section 6.2.4.

6.2.3.2.2 Component Description

Codes and standards applicable to the piping and valves associated with containment isolation are those for Class B components, as discussed in Section 3.2. Containment penetrations are classified as Quality Group B and Seismic Category I.

Section 3.11 provides the normal, abnormal, and post-loss-of-coolant accident environment that is used to qualify the operability of power-operated isolation valves located inside the containment.

The containment penetrations which are part of the main steam system and the feedwater system are designed to meet the stress requirements of NRC Branch Technical Position MEB 3-1, and the classification and inspection requirements of NRC Branch Technical Position ASB 3-1, as described in Section 3.6. Section 3.8 discusses the interface between the piping system and the steel containment.

As discussed in subsection 6.2.3.5, the instrumentation and control system provides the signals which determine when containment isolation is required. Containment penetrations are either normally closed prior to the isolation signal or the valves automatically close upon receipt of the appropriate engineered safety features actuation signal.

6.2.3.2.3 System Operation

During normal system operation, approximately 25 percent of the penetrations are not isolated. These lines are automatically isolated upon receipt of isolation signals, as described in subsections 6.2.3.3 and 6.2.3.4 and Chapter 7. Lines not in use during power operation are normally closed and remain closed under administrative control during reactor operation.

6.2.3.3 Design Evaluation

A. Engineered safeguards and containment isolation signals automatically isolate process lines which are normally open during operation. The containment isolation system uses diversity in the parameters sensed for the initiation of redundant train-oriented isolation signals. The majority of process lines are closed upon receipt of a containment isolation signal. This safeguards signal is generated by any of the following initiating conditions.

   - Low pressurizer pressure
   - Low steam-line pressure
   - Low T_{cold}
   - High containment pressure
   - Manual containment isolation actuation

   The component cooling water lines penetrating containment provide cooling water to the reactor coolant pumps and chemical and volume control system and liquid radwaste system heat exchangers. The reactor coolant pumps are interlocked to trip following a safeguards actuation (S) signal but will continue to operated (if in service) following a containment isolation (T) signal. In order to provide reliable cooling to the reactor coolant pumps the component cooling lines are isolated on a safeguards actuation signal rather than on a containment isolation signal. The safeguards actuation signal is generated by any of the following conditions.

- Low pressurizer pressure
- Low steam line pressure
- Low reactor coolant inlet temperature
- High containment pressure
- Manual initiation

The chemical and volume control system charging line, normal residual heat removal system reactor coolant and IRWST cooling lines, and containment air filtration system containment purge lines are isolated on high containment radiation signals. Closure of the containment air filtration system isolation valves is based on providing rapid response to elevated activity conditions in containment to limit offsite doses and is initiated on either a high radiation signal or a containment isolation signal consistent with the requirements of NUREG-0737 (Reference 22) and NUREG-0718 Rev 2 (Reference 23). The isolation of the chemical and volume control system charging line on a high radiation signal and normal residual heat removal system cooling lines on a high radiation or safeguards actuation signal with provisions to reset safeguards actuation signal for the normal residual heat removal system valves permits a defense in depth response to a postulated accident by providing for normal residual heat removal system and chemical and volume control system operation unless there is a high radiation level present.

The remainder of the containment isolation valves are closed on parameters indicative of the need to isolate.

B. Upon failure of a main steam line, the steam generators are isolated, and the main steam-line isolation valves, main steam-line isolation bypass valves, power operated relief block valves, and the main steam-line drain are closed to prevent excessive cooldown of the reactor coolant system or overpressurization of the containment.

The two redundant train-oriented steam-line isolation signals are initiated upon receipt of any of the following signals:

- Low steam-line pressure
- High steam pressure negative rate
- High containment pressure
- Manual actuation
- Low T_{cold}

The main steam-line isolation valves, main steam line isolation valve bypass valves, main feedwater isolation valves, steam generator blowdown system isolation valves, and piping are designed to prevent uncontrolled blowdown from more than one steam generator.

The main steam-line isolation valves and main feedwater isolation valves close fully within 5 seconds after an isolation is initiated. The blowdown rate is restricted by steam flow restrictors located within the steam generator outlet steam nozzles in each blowdown path. For main steam-line breaks upstream of an isolation valve, uncontrolled blowdown from more than one steam generator is prevented by the main steam-line isolation valves on each main steam line.

Failure of any one of these components relied upon to prevent uncontrolled blowdown of more than one steam generator does not permit a second steam generator blowdown to occur. No single active component failure results in the failure of more than one main steam isolation valve to operate. Redundant main steam isolation signals, described in Section 7.3, are fed to redundant parallel actuation vent valves to provide isolation valve closure in the event of a single isolation signal failure.

The effects on the reactor coolant system after a steam-line break resulting in single steam generator blowdown and the offsite radiation exposure after a steam line break outside containment are discussed in Chapter 15. The containment pressure transient following a main steam-line break inside containment is discussed in Section 6.2.

C. The containment isolation system is designed according to General Design Criterion 54. Leakage detection capabilities and leakage detection test program are discussed in subsection 6.2.5. Valve operability tests are also discussed in subsection 3.9.6. Redundancy of valves and reliability of the isolation system are provided by the other safety design bases stated in Section 6.2. Redundancy and reliability of the actuation system are covered in Section 7.3.

The use of motor-operated valves that fail as-is upon loss of actuating power in lines penetrating the containment is based upon the consideration of what valve position provides the plant safety. Furthermore, each of these valves, is provided with redundant backup valves to prevent a single failure from disabling the isolation function. Examples include: a check valve inside the containment and motor-operated valve outside the containment or two motor-operated valves in series, each powered from a separate engineered safety features division.

D. Lines that penetrate the containment and which are either part of the reactor coolant pressure boundary, connect directly to the containment atmosphere, or do not meet the requirements for a closed system are provided with one of the following valve arrangements conforming to the requirements of General Design Criteria 55 and 56, as follows:

- One locked-closed isolation valve inside and one locked-closed isolation valve outside containment

- One automatic isolation valve inside and one locked-closed isolation valve outside containment

- One locked-closed isolation valve inside and one automatic isolation valve outside containment. (A simple check valve is not used as the automatic isolation valve outside containment.)

- One automatic isolation valve inside and one automatic isolation valve outside containment. (A simple check valve is not used as the automatic isolation valve outside containment).

Isolation valves outside containment are located as close to the containment as practical. Upon loss of actuating power, air-operated automatic isolation valves fail closed.

E. Each line penetrating the containment that is neither part of the reactor coolant pressure boundary nor connected directly to the containment atmosphere, and that satisfies the requirements of a closed system, has at least one containment isolation valve. This containment isolation valve is either automatic, locked-closed, or capable of remote-manual operation. The valve is outside the containment and located as close to the containment as practical. A simple check valve is not used as the automatic isolation valve. This design is in compliance with General Design Criterion 57.

F. The containment isolation system is designed according to seismic Category I requirements as specified in Section 3.2. The components (and supporting structures) of any system, equipment, or structure that are non-seismic and whose collapse could result in loss of a required function of the containment isolation system through either impact or resultant flooding are evaluated to confirm that they will not collapse when subjected to seismic loading resulting from a safe shutdown earthquake.

Air-operated isolation valves fail in the closed position upon loss of air or power. Containment isolation system valves required to be operated after a design basis accident or safe shutdown earthquake are powered by the Class 1E dc electric power system.

6.2.3.4 Tests and Inspections

6.2.3.4.1 Preoperational Testing

Preoperational testing is described in Chapter 14. The containment isolation system is testable through the operational sequence that is postulated to take place following an accident, including operation of applicable portions of the protection system and the transfer between normal and standby power sources.

The safety related function of containment boundary integrity is verified by an integrated leakage rate test. The integrated leakage rate is verified to be less than L_a as defined in Table 6.5.3-1. The integrated containment leakage rate system is utilized to measure the containment leak rate for determination of the integrated leakage rate. The containment isolation valves are verified to close within the time specified in Table 6.2.3-1.

The piping and valves associated with the containment penetration are designed and located to permit pre-service and in-service inspection according to ASME Section XI, as discussed in subsection 3.9.6 and Section 6.6.

6.2.3.4.2 In-service Testing

Each line penetrating the containment is provided with testing features to allow containment leak rate tests according to 10 CFR 50, Appendix J, as discussed in subsection 6.2.5.

6.2.3.5 Instrumentation and Control Application

Instrumentation and control necessary for containment isolation, and the sensors used to determine that containment isolation is required, are described in Section 7.3.

Engineered safeguards actuation signals which initiate containment isolation will be initiated using two out of four logic. Containment isolation signals can also be initiated manually from the main control room. Containment isolation valves requiring isolation close automatically on receipt of a safeguards actuation signal.

Containment isolation valves that are equipped with power operators and are automatically actuated may also be controlled individually from the main control room. Also, in the case of certain valves with actuators (for example, sampling containment isolation valves), a manual override of an automatic isolation signal is installed to permit manual control of the associated valve. The override control function can be performed only subsequent to resetting of the actuation signal. That is, deliberate manual action is required to change the position of containment isolation valves in addition to resetting the original actuation signal. Resetting of the actuation signal does not cause any valve to change position. The design does not allow ganged reopening of the containment isolation valves. Reopening of the isolation valves is performed on a valve-by-valve basis, or on a line-by-line basis. Safeguards actuation signals take precedence over manual overrides of other isolation signals. For example, a containment isolation signal causes isolation valve closure even though the high containment radiation signal is being overridden by the operator. Containment isolation valves with power operators are provided with open/closed indication, which is displayed in the main control room. The valve mechanism also provides a local mechanical indication of valve position.

Power supplies and control functions necessary for containment isolation are Class 1E, as described in Chapters 7 and 8.

6.2.4 Containment Hydrogen Control System

Following a loss of coolant accident (LOCA), hydrogen may be produced inside the reactor containment by reaction of the zirconium fuel cladding with water, by radiolysis of water, by corrosion of materials of construction, and by release of the hydrogen contained in the reactor coolant system. The containment hydrogen control system is provided to limit the hydrogen concentration in the containment so that containment integrity is not endangered.

Two situations are postulated, a design basis case and a severe accident case. In the design basis case there is a limited reaction of less than 1 percent of fuel cladding zirconium with water to form hydrogen. For this case there is an initial release of hydrogen due to the reaction of fuel cladding with water and the release of hydrogen contained in the reactor coolant system. This initial hydrogen release to containment is not sufficient to approach the flammability limit of 4 volume percent. However, hydrogen continues to evolve to the containment due to radiolysis of water and the corrosion of materials in the containment. The flammability limit will eventually be reached unless mitigating action is taken. The function of the containment hydrogen control system is to prevent the hydrogen concentration from reaching the flammability limit.

In the severe accident case it is assumed that 100 percent of the fuel cladding reacts with water. Although hydrogen production due to radiolysis and corrosion occurs, the cladding reaction with water dominates the production of hydrogen for this case. The hydrogen generation from the zirconium-steam reaction could be sufficiently rapid that it may not be possible to prevent the hydrogen concentration in the containment from exceeding the lower flammability limit. The function of the containment hydrogen control system for this case is to promote hydrogen burning soon after the lower flammability limit is reached in the containment. Initiation of hydrogen burning at the lower level of hydrogen flammability prevents accidental hydrogen burn initiation at high hydrogen concentration levels and thus provides confidence that containment integrity can be maintained during hydrogen burns and that safety-related equipment can continue to operate during and after the burns.

The containment hydrogen control system serves the following functions:

- Hydrogen concentration monitoring
- Hydrogen control during and following a design basis loss of coolant accident (provided by passive autocatalytic recombiners [PARs])
- Hydrogen control during and following a degraded core or core melt scenarios (provided by hydrogen igniters).

6.2.4.1 Design Basis

A. The safety related portion of the hydrogen control system is protected from the effects of natural phenomena, such as earthquakes, tornadoes, hurricanes, floods, and external missiles (General Design Criterion 2).

B. The safety related portion of the hydrogen control system is designed to remain functional after a safe shutdown earthquake (SSE) and to perform its intended function following the postulated hazards of fire, internal missiles, or pipe breaks (General Design Criteria 3 and 4). Missile protection is discussed in section 3.5, pipe break protection in 3.6 and fire protection in 9.5.1 and appendix 9A.

C. The hydrogen control system is designed to provide containment atmosphere cleanup (hydrogen control) in accordance with General Design Criterion 41, 42 and 43.

D. The hydrogen control system is designed in accordance with the requirements of 10 CFR 50.44 and 10 CFR 50.34(f) and meets the NRC staff's position related to hydrogen control of SECY-93-087.

E. The hydrogen control system is designed in compliance with the recommendations of NUREG 0737 and 0660 as detailed in subsection 1.9.

F. The hydrogen control system is designed in accordance with the recommendations of Regulatory Guide 1.7 as discussed in appendix 1A. The containment recirculation system discussed in subsection 9.4.7 provides the controlled purge capability for the containment as specified in position C.4 of Regulatory Guide 1.7

G. The hydrogen control system is designed and fabricated to codes consistent with the quality group classification, described in Section 3.2. Conformance with Regulatory Guide 1.26, 1.29, and 1.32 is described in subsection 1.9.

H. The hydrogen control system complies with the intent of Regulatory Guide 1.82 "The Water Sources For Long-Term Recirculation Cooling Following A Loss-Of-Coolant Accident" as it could be applied to concerns for blockage of recombiner air flow paths.

6.2.4.1.1 Containment Mixing

Containment structures are arranged to promote mixing via natural circulation. The physical mechanisms of natural circulation mixing that occur in the AP600 are discussed in Appendix 6A and summarized below. For a postulated break low in the containment, buoyant flows develop through the lower compartments due to density head differences between the rising plume and the surrounding containment atmosphere, tending to drive mixing through lower compartments and into the region above the operating deck. There is also a degree of mixing within the region above the operating deck, which occurs due to the introduction of and the entrainment into the steam-rich plume as it rises from the operating deck openings. Thus, natural forces tend to mix the containment atmosphere.

Two general characteristics have been incorporated into the design of the AP600 to promote mixing and eliminate dead-end compartments. The compartments below deck are large open volumes with relatively large interconnections, which promote mixing throughout the below deck region. All compartments below deck are provided with openings through the top of the compartment to eliminate the potential for a dead pocket of high-hydrogen concentration. In addition, if forced containment air-circulation is operated during post-accident recovery, then nonsafety-related fan coolers contribute to circulation in containment.

In the event of a hydrogen release to the containment, passive autocatalytic recombiners act to recombine hydrogen and oxygen on a catalytic surface (see subsection 6.2.4.2.2). The enthalpy of reaction generates heat within a passive autocatalytic recombiner, which further drives containment mixing by natural circulation. Catalytic recombiners reduce hydrogen concentration at very low hydrogen concentrations (less than 1 percent) and very high steam concentrations, and may also promote convection to complement passive containment cooling system natural circulation currents to inhibit stratification of the containment atmosphere (Reference 17). The implementation of passive autocatalytic recombiners has a favorable impact on both containment mixing and hydrogen mitigation.

6.2.4.1.2 Survivability of System

The portion of the containment hydrogen control system required for the design basis loss of coolant accident is designed to withstand the dynamic effects associated with postulated accidents, the environment existing inside the containment following the postulated accident, and a safe shutdown earthquake.

The environmental qualification of the PAR's and hydrogen monitors are performed in accordance with the specifications of Section 3.11 using the methodology defined in Appendix 3D. Within Reference 27, the NRC has concluded that the chemical environment for environmental qualification should include potential poisons. Specifically, that the PAR's should be environmentally qualified to include the source term constituents which were conservatively assumed to yield the radioactivity dose rates for environmental qualification. The AP600 PAR's will be qualified pursuant to the guidance of Reference 27 and utilizing the poisons resulting from a core melt event through in-vessel releases as discussed in Reference 28. This approach is conservative since the level of potential poisons assumed is inconsistent with the functional purpose of the PAR in design basis hydrogen mitigation. The magnitude of poisons is based on the concentration consistent with a source term derived from a severe accident core melt scenario rather than the regulatory criteria for design basis accidents.

In addition to the source term constituents, environmental qualification of the PAR's and the hydrogen monitors will include exposure to phosphates and silicon oil. The source of phosphate source is from trisodium phosphate (TSP) utilized for post accident sump pH adjustment (see subsection 6.3). The TSP is dissolved in the post accident cooling water following a design basis accident. The release mechanism to the containment atmosphere post LOCA is via the ADS stage 4 discharge following the onset of the recirculation phase of PXS operation. The source of silicon oil is from a postulated failed steam generator hydraulic snubber. The containment hydrogen control equipment provided to mitigate severe accident conditions (igniter subsystem) is designed to function under the event environment including the effects of combustion of hydrogen in containment.

6.2.4.1.3 Single Failure Protection

The hydrogen monitoring function is designed to accommodate a single failure. The hydrogen recombination subsystem consists of qualified passive devices which are not susceptible to single failures. However to provide margin and increased containment coverage two global containment PAR's are provided and credit for only a single unit is assumed in the hydrogen analysis. The location of the PAR's are such that the units are not susceptible to debris blockage as identified in Regulatory Guide 1.82. The global PAR's are at an approximate elevation of 164 feet and not immediately above the loop compartments. The IRWST PAR is located above the operating deck above the IRWST well away from any pipe failure locations that could produce design basis accidents. The CVS compartment PAR is well away from any pipe failure locations that could produce a design basis accident and is located in the upper region of the compartment. This location assures that if the PAR is flooded there will be no hydrogen concern in the compartment. All other units are above the maximum flooding elevation. Based on the locations of the PAR's they are not susceptible to debris blockage. The hydrogen ignition system, since it is provided only to address a low-probability severe accident, is designed to accommodate probable component and system failures.

6.2.4.1.4 Validity of Hydrogen Monitoring

The hydrogen monitoring function monitors hydrogen concentrations of various locations within the containment.

6.2.4.1.5 Hydrogen Control for Design Basis Accident

The containment volume average hydrogen concentration is prevented from exceeding 4 volume percent. This limit eliminates the potential for flammable conditions.

6.2.4.1.6 Hydrogen Control for Severe Accident

The containment hydrogen concentration is limited by operation of the distributed hydrogen ignition subsystem. Ignition causes deflagration of hydrogen (burning of the hydrogen with flame front propagation at subsonic velocity) at hydrogen concentrations between the flammability limit and 10 volume percent and thus prevents the occurrence of hydrogen detonation (burning of hydrogen with supersonic flame front propagation).

6.2.4.2 System Design

6.2.4.2.1 Hydrogen Concentration Monitoring Subsystem

The hydrogen concentration monitoring subsystem consists of two groups of eight hydrogen sensors each. The sensors are placed in various locations throughout the containment free volume including the upper dome and containment compartments.

The system contains a total of three sensors designated as Class 1E and thirteen sensors designated as non-Class 1E. The three Class 1E sensors are seismic Category 1 and serve to provide a post accident monitoring function for design basis accidents. See Section 7.5 for additional information. The sensors designated as non-Class 1E provide a defense in depth function of monitoring local hydrogen concentrations. The sensors are environmentally qualified as specified in Section 6.2.4.1.2 and Section 3.11.

The 1E hydrogen sensors are powered by a Class 1E power source and the non-Class 1E hydrogen sensors are powered by a non-Class 1E power source. The Class 1E instrument channels are independent of the non-Class 1E instrument channels. Sensor parameters are provided in Table 6.2.4-1. Hydrogen concentration is continuously indicated in the main control room. Additionally, high hydrogen concentration alarms are provided in the main control room.

The sensors are designed to provide a rapid response detection of changes in the containment hydrogen concentration. The response time of the sensor is at least 90 percent in 10 seconds.

6.2.4.2.2 Hydrogen Recombination Subsystem

The hydrogen recombination subsystem is designed to accommodate the hydrogen production rate anticipated for a design-basis loss of coolant accident. The hydrogen recombination subsystem consists of two passive autocatalytic recombiners installed inside the containment above the operating deck at an approximate elevation of 162 feet and 13 feet inboard from the containment shell. The locations provide placement within a homogeneously mixed region of containment as supported by subsection 6.2.4.1.1 and Appendix 6A. The location is in a predominately upflow natural convection region. Additionally, the PARs are located azmuthally away from potential high upflow regions such as the direct plume above the loop compartment.

A third PAR is located at one of the vent paths from the IRWST and is utilized to limit the accumulation of hydrogen within the IRWST during normal and post-accident operation.

A fourth PAR is located in the chemical and volume control system compartment to limit the accumulation of hydrogen within the compartment as a result of radiolysis and corrosion within the compartment from a partially flooded condition following a design basis LOCA.

The passive autocatalytic recombiners are simple and passive in nature without moving parts and independent of the need for electrical power or any other support system. The recombiners are safety-related equipment. They are seismic Category I and are qualified for the post-loss of coolant accident environment. The recombiners require no power supply and are self-actuated by the presence of the reactants (hydrogen and oxygen).

Normally, oxygen and hydrogen recombine by rapid burning only at elevated temperatures (greater than about 1100°F [600°C]). However, in the presence of catalytic materials such as the palladium group, this "catalytic burning" occurs even at temperatures below 32°F (0°C). Adsorption of the oxygen and hydrogen molecules occurs on the surface of the catalytic metal because of attractive forces of the atoms or molecules on the catalyst surface. Passive autocatalytic recombiner devices use palladium or platinum as a catalyst to combine molecular hydrogen with oxygen gases into water vapor. The catalytic process can be summarized by the following steps (Reference 15):

1. Diffusion of the reactants (oxygen and hydrogen) to the catalyst
2. Reaction of the catalyst (chemisorption)
3. Reaction of intermediates to give the product (water vapor)
4. Desorption of the product

5. Diffusion of the product away from the catalyst

The reactants must get to the catalyst before they can react and subsequently the product must move away from the catalyst before more reactants will be able to react.

The passive autocatalytic recombiner device consists of a stainless steel enclosure providing both the structure for the device and support for the catalyst material. The enclosure is open on the bottom and top and extends above the catalyst elevation to provide a chimney to yield additional lift to enhance the efficiency and ventilation capability of the device. The catalyst material is either constrained within screen cartridges or deposited on a metal plate substrate material and supported within the enclosure. The spaces between the cartridges or plates serve as ventilation channels for the throughflow. During operation, the air inside the recombiner is heated by the recombination process, causing it to rise by natural convection. As it rises, replacement air is drawn into the recombiner through the bottom of the passive autocatalytic recombiner and heated by the exothermic reaction, forming water vapor, and exhausted through the chimney where the hot gases mix with containment atmosphere. The device is a molecular diffusion filter and thus the open flow channels are not susceptible to fouling.

Passive autocatalytic recombiners begin the recombination of hydrogen and oxygen almost immediately upon exposure to these gases when the catalyst is not wetted. If the catalyst material is wet, then a short delay is experienced in passive autocatalytic recombiner startup (References 19 and 29). The delay is short with respect to the time that the PARs have to control hydrogen accumulation rates (days to weeks) following a design basis accident. The recombination process occurs at room or elevated temperature during the early period of accidents prior to the buildup of flammable gas concentrations. Passive autocatalytic recombiners are effective over a wide range of ambient temperatures, concentrations of reactants (rich and lean, oxygen/hydrogen less than 1 percent) and steam inerting (steam concentrations greater than 50 percent). Although the passive autocatalytic recombiner depletion rate reaches peak efficiency within a short period of time, the rate varies with hydrogen concentration and containment pressure, (Reference 19).

Reference 19 provides passive autocatalytic recombiner performance estimates appropriate (depletion rates) for a design basis accident, while a best-estimate depletion rate is appropriate for severe accident hydrogen control scenarios where realistic estimates of system performance are appropriate due to the low probability of occurrence. A conservative or lower bound estimate of depletion rate may be used for a design basis accident analysis. The conservative depletion rate accounts for effects such as instrumentation error, curve fitting, and startup delay. This rate (with one of the two containment passive autocatalytic recombiners available) is used for the analysis results presented in Figure 6.2.4-1, "Passive Autocatalytic Recombiner Sensitivity Study - Dry Conditions, Impact on Containment H_2 Concentrations."

The depletion rate assumed in the analysis is based on a generic passive autocatalytic recombiner application as described in Reference 19, and is expected to be representative of a number of vendor's recombiners. The calculated containment hydrogen concentration presented in Figures 6.2.4-1 and 6.2.4-2 is based on the assumptions and analysis discussed in subsection 6.2.4.3. The results demonstrate abundant margin for system performance. Further, the hydrogen concentration following an accident with only one of the two available passive autocatalytic recombiners operating within containment demonstrates significant margin to maintaining hydrogen concentrations below the recommendations of Regulatory Guide 1.7, Control of Combustible Gas Concentrations in Containment Following a Loss-of-Coolant Accident.

The equations predicting the depletion rate are as follows:

- For H_2 concentrations less than 0.2 percent depletion rate (kg/hr) = 0.0
- For H_2 concentrations equal to or greater than 0.2 percent depletion rate (kg/hr) = 78,800 x [0.029883 x ([C-0.2]/100)^2 + (0.001009 x [C-0.2]/100) x P]/(T + 273)

where C = volume average H_2 concentration at passive autocatalytic recombiner inlet
P = total pressure (bars)
T = gas temperature at passive autocatalytic recombiner inlet (°C)

The conditions under which the passive autocatalytic recombiners are assumed to operate for a design basis accident for defining the lower bound hydrogen depletion rate per Reference 19 are:

- Inlet gas temperatures ranging from 100 to 330°F
- Pressures ranging from 1 to 4 atmospheres
- Hydrogen concentrations up to 5 volume percent
- Steam concentrations ranging from near zero to 75 percent
- Condensing steam environment
- No significant levels of potential catalyst poisons

The basis for defining the hydrogen depletion rate is testing conducted by Battelle Frankfurt of both full scale and segment model NIS passive autocatalytic recombiner units. The results of the tests and their use in the definition of a hydrogen depletion rate equation appropriate for a design basis accident is provided in References 18 and 19. Subsequent test conducted by EPRI and EdF (Reference 29) support the conclusions of Battelle testing. Reference 19 assumed no significant levels of catalyst poisons (for example, iodine, carbon monoxide, cable fire combustion products and tellurium) would be present following a design basis event. This assumption is consistent with the regulatory limits imposed on clad damage levels of 10 CFR 50.46 and 50.44 for a loss of coolant accident. The existence of significant levels of poisons would normally mandate consideration of events and hydrogen generation rates for which other design attributes of the hydrogen control system are specifically provided. Events which generate high levels of iodine and tellurium, for example, are the result of gross fuel clad damage and cladding/water reactions.

However, in accordance with Reference 27, the PAR's will be environmentally qualified in the presence of catalyst poisons which would be present following a design basis event that results in a source term defined in NUREG 1465 for an event progressing through the stages of reactor coolant, gap and early in-vessel releases. Further, based on industry test data and catalyst poison literature, the hydrogen recombination subsystem performance for a DBA will be evaluated with the reduction in performance anticipated as a result of the effects of poisons and inhibitors as determined in Reference 28. The fractions of core inventory released through reactor coolant, gap and early in-vessel stages of core damage event are identified in NUREG 1465. Reference 28 considers the releases of NUREG 1465 and addresses the potential effects of poisons and inhibitors on the performance of the PAR's. The approach in Reference 28 is to compile existing information and data as a basis for establishing a generic bounding value of a deactivation reduction factor for design and qualification of PAR's. The approach combines qualitative information based on established chemical and physical principles with quantitative information from testing of catalysts systems subjected to a wide range of inhibitors and poisons. The sources of test data include (1) tests on PAR's conducted by two suppliers over the past several years, (2) tests on the same two types of PAR's conducted recently in France by EPRI/EdF/CEA and (3) tests on catalyst pellet-bed filters conducted in a laboratory about 25 years ago and additional testing described in Reference 28. The report concludes that "Even if the accident were to progress to beyond a DBA to substantial in-vessel damage, PAR recombination capacity would be reduced by no more than 25%..."

To illustrate the margin available and the tolerance to catalytic poisons and inhibitors, Figure 6.2.4-2 demonstrates the effects of the presence of elevated concentrations of poisons and inhibitors in containment. The curve demonstrates the effects of the conservative depletion rate discussed above in combination with a 25 percent penalty due to poisons and inhibitors. The curve remains below 2.2 volume percent hydrogen concentration. The margin of difference between the regulatory limit of 4 percent and the projected concentration may be considered to represent the maximum potential gradient between global hydrogen concentration and an isolated postulated hypothetical volume of containment atmosphere.

The environments in which safety-related components of the hydrogen control system are designed and qualified to function are discussed in Section 3.11 and subsection 6.2.4.1.2. The pressure, temperature, and chemical environment conditions for which components are designed to function have been based on analysis of the design basis event and the systems response with additional considerations identified in Reference 27. The radiation environments have, in contrast, been the result of a deterministic application of the accident source term. As specified in NUREG 1465, to determine the accident source term for regulatory purposes, the staff examined a range of severe accidents that have been analyzed for light water reactors. The environmental qualification guidance and practice is conservatively based on the effects of radiation due to a severe accident source term.

The passive autocatalytic recombiner testing and reporting of test data, conducted under the NIS quality assurance program has demonstrated proof of principle as appropriate for design certification. An evaluation and summary of the quality assurance program for the Battelle tests is provided in Reference 21. Procurement will be in accordance with the COL applicants QA program.

A summary of component data for the hydrogen recombiners is provided in Table 6.2.4-2.

6.2.4.2.3 Hydrogen Ignition Subsystem

The hydrogen ignition subsystem is provided to address the possibility of an event that results in a rapid production of large amounts of hydrogen such that the rate of production exceeds the capacity of the recombiners. Consequently, the containment hydrogen concentration will exceed the flammability limits. This massive hydrogen production is postulated to occur as the result of a degraded core or core melt accident (severe accident scenario) in which up to 100 percent of the zirconium fuel cladding reacts with steam to produce hydrogen.

The hydrogen ignition subsystem consists of 64 hydrogen igniters strategically distributed throughout the containment. Since the igniters are incorporated in the design to address a low-probability severe accident, the hydrogen ignition system is not Class 1E. Although not class 1E, the igniter coverage, distribution and power supply has been designed to minimize the potential loss of igniter protection globally for containment and locally for individual compartments. The igniters have been divided into two power groups. Power to each group will be normally provided by offsite power, however should offsite power be unavailable, then each of the power groups is powered by one of the onsite non-essential diesels and finally should the diesels fail to provide power then approximately 4 hours of igniter operation is supported by the non-Class 1E batteries for each group. Assignment of igniters to each group is based on providing coverage for each compartment or area by at least one igniter from each group.

The locations of the igniters are based on evaluation of hydrogen transport in the containment and the hydrogen combustion characteristics. Locations include compartmented areas in the containment and various locations throughout the free volume, including the upper dome.

For enclosed areas of the containment at least two igniters are installed. The separation between igniter locations is selected to prevent the velocity of a flame front initiated by one igniter from becoming significant before being extinguished by a similar flame front propagating from another igniter. The number of hydrogen igniters and their locations are selected considering the behavior of hydrogen in the containment during severe accidents. The likely hydrogen transport paths in the containment and hydrogen burn physics are the two important aspects influencing the choice of igniter location.

The primary objective of installing an igniter system is to promote hydrogen burning at a low concentration and, to the extent possible, to burn hydrogen more or less continuously so that the hydrogen concentration does not build up in the containment. To achieve this goal, igniters are placed in the major regions of the containment where hydrogen may be released, through which it may flow, or where it may accumulate. The criteria utilized in the evaluation and the application of the criteria to specific compartments is provided in Table 6.2.4-6. The location of igniters throughout containment is provided in Figures 6.2.4-5 through 6.2.4-11. The location of igniters is also summarized in Table 6.2.4-7 identifying subcompartment/regions and which igniters by power group provide protection. The locations identified are considered approximations ($\pm$ 2.5 feet) with the final locations governed by the installation details.

The igniter assembly is designed to maintain the surface temperature within a range of 1600° to 1700°F in the anticipated containment environment following a loss of coolant accident. A spray shield is provided to protect the igniter from falling water drops (resulting from condensation of steam on the containment shell and on nearby equipment and structures). Design parameters for the igniters are provided in Table 6.2.4-3.

6.2.4.2.4 Containment Purge

Containment purge is not part of the containment hydrogen control system. The purge capability of the containment air filtration system (see subsection 9.4.7) can be used to provide containment venting prior to post-loss of coolant accident cleanup operations.

6.2.4.3 Design Evaluation (Design-Basis Accident)

6.2.4.3.1 Hydrogen Production and Accumulation

Following a loss of coolant accident, hydrogen may be added to the reactor containment atmosphere by reaction of the zirconium fuel cladding with water, by radiolysis of water, by corrosion of materials of construction, and by release of the hydrogen contained in the reactor coolant system. The assumptions used in calculating the hydrogen release to containment are listed in Table 6.2.4-4.

6.2.4.3.1.1 Zirconium-Water Reaction

Zirconium fuel cladding reacts with steam according to the following equation:

$$Zr + 2\ H_2O \rightarrow ZrO_2 + 2\ H_2 + \text{heat}$$

There is 8.5 standard cubic feet (SCF) of hydrogen produced for each pound of zirconium that is reacted.

The extent of the zirconium-water reaction is dependent on the effectiveness of the core cooling. An evaluation of the AP600 design shows that there is no zirconium-water reaction during a design basis accident. The NRC model presented in Regulatory Guide 1.7 conservatively assumes that the cladding oxidizes to a depth of 0.00023 inch. For the 0.0225-inch cladding thickness used for AP600 fuel, this constitutes 1.09 percent of the zirconium. The hydrogen produced by the reaction of zirconium is 3000 standard cubic feet. This hydrogen is assumed to be released to the containment atmosphere at the beginning of the accident.

6.2.4.3.1.2 Radiolysis of Water

Water radiolysis is a complex process involving reactions of numerous intermediates. However, the overall radiolytic process may be described by the equation:

$$H_2O \rightleftarrows H_2 + \frac{1}{2} O_2$$

Post-accident conditions in the containment create two distinct radiolytic environments. One environment exists inside the reactor vessel where radiolysis occurs due to energy emitted by decaying fission products in the fuel and absorbed by the cooling water. The second environment exists outside the reactor vessel, in the post-accident cooling solution itself, where radiolysis occurs due to the absorption of decay energy emitted by the fission products retained in the solution. The two basic differences between the core environment and the solution environment that affect the rate of hydrogen production are the fraction of energy absorbed by the water and the type of flow regime.

The rate of hydrogen production from radiolysis depends on the rate of energy absorption by the solution. Analysis of energy deposition in the reactor core where decaying fission products are retained in the fuel shows that beta radiation constitutes roughly 50 percent of the total decay energy. Since the beta radiation is absorbed by the fuel and the fuel clad, this energy is not available to the solution to contribute to the radiolysis of water. Additionally, most of the gamma radiation energy is absorbed by the fuel, fuel cladding, and other components; or it passes through the water without being absorbed. The solution in the reactor vessel would absorb approximately seven percent of the gamma radiation energy. However, consistent with Regulatory Guide 1.7, it is assumed that 10 percent of the core gamma energy is absorbed by the water.

For the post-accident cooling solution in which the fission products released from the core are assumed to be dissolved, energy is emitted directly into the solution. All of the beta radiation is assumed to be absorbed by the water. Since the mass of water is relatively large compared to the penetrating capability of gamma radiation, it is also assumed that 100 percent of the gamma radiation energy is absorbed by the water.

The radiolytic decomposition of water is a reversible reaction. In the reactor vessel, where the products of radiolysis are continuously flushed away by the circulation of cooling solution, there is little chance for hydrogen and oxygen to accumulate. Consequently, recombination of hydrogen and oxygen is assumed not to occur because significant quantities of the two reactants are not available.

The post-accident cooling solution in the sump, however, is a deep and relatively static environment where the products of radiolysis are lost from solution primarily by molecular diffusion. Tests simulating post-accident sump conditions demonstrate that there is significant reverse reaction in the sump. Hence, there is an apparent reduction in the quantity of hydrogen produced per unit energy absorbed by the water.

The results of Westinghouse and Oak Ridge National Laboratory studies indicate maximum hydrogen yields of 0.44 molecules per 100 eV for core radiolysis and 0.3 molecules per 100 eV for solution radiolysis. The results of these studies are published in References 10, 11, and 12.

The analysis performed for the AP600 assumes a hydrogen yield of 0.5 molecules per 100 eV for both the core and the solution radiolysis cases. This value is conservative relative to the referenced studies and is consistent with the guidance of Regulatory Guide 1.7.

In a design basis loss of coolant accident there is expected to be no damage to the core and thus no release of activity from the core to the sump solution. The source term used for determining radiolysis production of hydrogen is conservatively based on guidance of Regulatory Guide 1.7 which states that 100 percent of noble gases, 50 percent of iodines, and 1 percent of other nuclides are assumed to be released from the core even though it is inconsistent with the limited amount of fuel cladding reaction that is determined to take place. Appendix 15A provides the core fission product inventory at shutdown.

Table 6.2.4-4 contains a summary of the assumptions used in the analysis of hydrogen produced from radiolysis. Production rate of hydrogen as a function of time is shown graphically in Figure 6.2.4-3 and the production of hydrogen is shown in Figure 6.2.4-4.

6.2.4.3.1.3 Corrosion of Metals

In the environment that would exist inside the containment following a postulated loss of coolant accident, aluminum and zinc corrode to form hydrogen gas. Table 6.2.4-4 lists the inventory of aluminum and zinc inside the containment.

Aluminum corrosion may be described by the overall reaction:

$$2\ Al + 3\ H_2O \rightarrow Al_2O_3 + 3\ H_2$$

About 21.4 standard cubic feet of hydrogen gas is produced for each pound of aluminum corroded.

The corrosion of zinc is described by the following reaction:

$$Zn + 2\ H_2O \rightarrow Zn(OH)_2 + H_2$$

About 5.9 standard cubic feet of hydrogen gas is produced for each pound of zinc corroded.

The corrosion rates for both aluminum and zinc are dependent on the post-accident temperature and pH conditions that the materials are subjected to. Table 6.2.4-5 provides the time-temperature cycle considered in the analysis of aluminum and zinc corrosion and also the corrosion rates for the metals at these temperatures.

Production rate of hydrogen as a function of time is shown graphically in Figure 6.2.4-3 and the production of hydrogen is shown in Figure 6.2.4-4.

6.2.4.3.1.4 Initial Reactor Coolant Hydrogen Inventory

During normal operation of the plant, hydrogen is dissolved in the reactor coolant and is also contained in the pressurizer vapor space. Following a loss of coolant accident, this hydrogen is assumed to be immediately released to the containment atmosphere. Table 6.2.4-4 lists the assumptions used for determining the amount of hydrogen from this source. The total hydrogen released to the containment as a result of this source is 1171 standard cubic feet.

6.2.4.3.2 Hydrogen Mixing

The AP600 is designed to prevent the accumulation of hydrogen in compartments. If there is the possibility of accumulation in compartments, venting is provided to allow the hydrogen to escape to the larger containment volume. Mixing of the containment air mass is accomplished through natural processes as a result of the passive cooling of the containment (see subsection 6.2.2) that induces a recirculating air flow in the containment. The release rate for a design basis accident is sufficiently slow that mixing is effectively assured. Additional details are provided in subsection 6.2.4.1.1 and Appendix 6A.

6.2.4.3.3 Hydrogen Recombination

Assuming no hydrogen removal, the concentration of hydrogen in the containment atmosphere increases with time as shown in Figure 6.2.4-1. The curve shows that the flammability limit of 4 volume percent is not reached until after 12 days. Hydrogen recombination begins prior to reaching this limit. The passive autocatalytic recombiners are brought into service by the presence of the reactants. The available passive autocatalytic recombiner test data as discussed in Reference 19 supports passive autocatalytic recombiner startup within 7 hours of reaching 1 volume percent hydrogen concentration in containment. Subsequent to passive autocatalytic recombiner startup the conservative lower bound equation for depletion rates provided in Reference 19 has been used to predict containment concentrations considering only the PARs between the 150 and 175 foot elevations. Figure 6.2.4-1 shows the impact of operation of one of the two recombiners on containment hydrogen concentration. The hydrogen concentration never exceeds 1.5 percent which indicates ample margin in the hydrogen recombiner capacity. Figure 6.2.4-2 evaluates the containment hydrogen concentration using the same lower bound equation for a single PAR's depletion rate but with additional conservatism to account for catalyst poisons or evaluation of margin. The curve identified as "Worst Case PAR Depletion Rate with Catalyst Poisons" has been reduced as a result of the effects of poisons and inhibitors which could be released to containment following a core melt scenario progressing through reactor coolant, gap, and early in-vessel releases. In accordance with Reference 28 the lower bound depletion rate has been reduced by 25 percent to conservatively account for the effects of potential poisons and inhibitors.

A further demonstration of the passive autocatalytic recombiner's available capacity margin is provided by calculation of containment concentrations with artificially reduced depletion rates. Figure 6.2.4-2 provides the impact of one of two available passive autocatalytic recombiners operating at 20, 10 and 1 percent of the conservative lower bound capacity. The curves provide indication of the abundant hydrogen control margin. Also provided in Figure 6.2.4-2 are the results of a calculation assuming no recombination until the hydrogen concentration reaches 3.5 volume percent in containment. Although a zero depletion rate until concentration reaches a 3.5 percent threshold is excessively conservative, the results emphasize the abundant margin.

6.2.4.4 Design Evaluation (Severe Accident)

Although a severe accident involving major core degradation or core melt is not a design basis accident, the containment hydrogen control system contains design features to address this potential occurrence. The hydrogen monitoring subsystem has sufficient range to monitor concentrations up to 20 percent hydrogen. The hydrogen ignition subsystem is provided so that hydrogen is burned off in a controlled manner, preventing the possibility of deflagration with supersonic flame front propagation which could result in large pressure spikes in the containment.

The hydrogen released to the containment due to initial inventory of hydrogen in the coolant would be the same as for the design basis case (see subsection 6.2.4.3.1.4).

The hydrogen production due to corrosion of aluminum and zinc or to radiolysis of water is not of concern for evaluating the containment hydrogen control system for the severe accident since hydrogen production from these sources takes place at a relatively slow rate and over a long period of time.

It is assumed that 100 percent of the active fuel cladding zirconium reacts with steam. This reaction may take several hours to complete. The igniters initiate hydrogen burns at concentrations less than 10 percent by volume and prevent the containment hydrogen concentration from exceeding this limit. Further evaluation of hydrogen control by the igniters is presented in the AP600 Probabilistic Risk Assessment.

6.2.4.5 Tests and Inspections

6.2.4.5.1 Preoperational Inspection and Testing

Hydrogen Monitoring Subsystem

Pre-operational testing is performed either before or after installation but prior to plant startup to verify performance.

Hydrogen Recombination Subsystem

The performance of the autocatalytic recombiner plates (or cartridges) is tested by the manufacturer for each lot or batch of catalyst material. The number of plates tested is based on the guidance provided in ANSI/ASQC Z1.4-1993, "Sampling Procedures and Tables for Inspection by Attributes," (formerly Military Standard 105), required to achieve Inspection Level III quality level.

Pre-operational testing is performed following vendor production testing and installation but prior to plant startup to verify PAR performance. The PAR's are verified to provide a hydrogen depletion rate of greater than or equal to the minimum depletion rate identified in Table 6.2.4-2. It is also verified that two PAR's are installed within containment at an elevation of between 150 and 175 feet with the PAR centerline at least 10 feet from the containment shell. It is also verified that a PAR is located in the exhaust of an IRWST vent and within the chemical and volume control system compartment.

A sample of the PAR cartridges or plates are selected and removed from each passive autocatalytic recombiner and surveillance bench tests are performed on the removed specimens to confirm continued satisfactory performance. The specimen is placed in a performance test apparatus and exposed to a known standard air/hydrogen sample. The test instrumentation will be designed to assess PAR performance and the time to reach a threshold recombination start to measure degradation in catalytic action. The overall PAR performance verification will be based on vendor testing recommendations and may include among other means, recombiner internal or exhaust temperature measurement or exhaust sample concentration measurement. Should internal temperature measurement be utilized as the measured recombination parameter, location of the sensor must be consistent for all samples and with vendor test recommendations to assure consistency between tests. The recombiner start verification will be based on a time dependent measurement of the recombination rate parameter or other instrumentation verifying the recombination start. The vendor manufacturing acceptance data or accepted industry standards will be utilized as acceptance data provided it represents performance in excess of the required rate specified in Table 6.2.4-2.

Hydrogen Ignition Subsystem

Pre-operational testing and inspection is performed after installation of the hydrogen ignition system and prior to plant startup to verify operability of the hydrogen igniters. It is verified that 64 igniter assemblies are installed at the locations defined by Figures 6.2.4-5 through 6.2.4-11. Operability of the igniters is confirmed by verification of the surface temperature in excess of the value specified in Table 6.2.4-3. This temperature is sufficient to ensure ignition of hydrogen concentrations above the flammability limit.

Pre-operational inspection is performed to verify the location of openings through the ceilings of the passive core cooling system valve/accumulator rooms. The primary openings must be at least 19 feet from the containment shell. Primary openings are those that constitute 98% of the opening area. Other openings must be at least 3 feet from the containment shell.

Pre-operational inspection is performed to verify the orientation of the vents from the IRWST that are located along the side of the IRWST next to the containment. The discharge of each of these IRWST vents must be oriented generally away from the containment shell.

6.2.4.5.2 In-service Testing

Hydrogen Monitoring Subsystem

The system is normally in service. Periodic testing and calibration are performed to provide ongoing confirmation that the hydrogen monitoring function can be reliably performed.

Hydrogen Recombination Subsystem

Periodic inspection and testing are performed on the PAR's. The testing is in accordance with the inservice testing program (Section 3.9.6) as required by the technical specification surveillance testing (Section 16.3.6.10). The verification is performed by testing a sample of the catalyst plates as specified in 6.2.4.5.1.

Hydrogen Ignition Subsystem

Periodic inspection and testing are performed to confirm the continued operability of the hydrogen ignition system. Operability testing consists of energizing the igniters and confirming the surface temperature exceeds the value specified in Table 6.2.4-3.

6.2.4.5.2 In-service Testing

Hydrogen Monitoring Subsystem

The system is normally in service. Periodic testing and calibration are performed to provide ongoing confirmation that the hydrogen monitoring function can be reliably performed.

Hydrogen Recombination Subsystem

Periodic inspection and testing are performed on the passive autocatalytic recombiners. The testing is in accordance with subsection 3.9.6 and is performed by testing a sample of the catalyst plates as specified in subsection 6.2.4.5.1.

Hydrogen Ignition Subsystem

Periodic inspection and testing are performed to confirm the continued operability of the hydrogen ignition system. Operability testing consists of energizing the igniters and confirming the surface temperature exceeds the value specified in Table 6.2.4-3.

6.2.4.6 Combined License Information

This section has no requirement to be provided in support of the Combined License application.

6.2.5 Containment Leak Rate Test System

The reactor containment, containment penetrations and isolation barriers are designed to permit periodic leak rate testing in accordance with General Design Criteria 52, 53, and 54. The containment leak rate test system is designed to verify that leakage from the containment remains within limits established in the technical specifications, Chapter 16.

6.2.5.1 Design Basis

Leak rate testing requirements are defined by 10 CFR 50 Appendix J, "Primary Reactor Containment Leakage Testing for Water Cooled Power Reactors," (Reference 14) which classifies leak tests as Types A, B and C. The system design provides testing capability consistent with the testing requirements of ANSI-56.8 (Reference 13). The system design accommodates the test methods and frequencies consistent with requirements of 10 CFR 50 Appendix J, Option A or Option B.

6.2.5.1.1 Safety Design Basis

The containment leak rate test system serves no safety-related function other than containment isolation, and therefore has no nuclear safety design basis except for containment isolation. See subsection 6.2.3 for the containment isolation system.

6.2.5.1.2 Power Generation Design Basis

The containment leak rate test system is designed to verify the leak tightness of the reactor containment. The specified maximum allowable containment leak rate is 0.10 weight percent of the containment air mass per day at the calculated peak accident pressure, P_a, identified in subsection 6.2.1. The system is specifically designed to perform the following tests in accordance with the provisions of ANSI-56.8 (Reference 13):

- Containment integrated leak rate testing (Type A): The containment is pressurized with clean, dry air to a pressure of P_a. Measurements of containment pressure, dry bulb temperature, and dew point temperature are used to determine the decrease in the mass of air in the containment over time, and thus establish the leak rate.

- Local leak rate testing of containment penetrations with a design that incorporates features such as resilient seals, gaskets, and expansion bellows (Type B): The leakage limiting boundary is pressurized with air or nitrogen to a pressure of P_a and the pressure decay or the leak flow rate is measured.

- Local leak rate testing of containment isolation valves (Type C): The piping test volume is pressurized with air or nitrogen to a pressure of P_a and pressure decay or the leak flow rate is measured. For valves sealed with a fluid such as water, the test volume is pressurized with the seal fluid to a pressure of not less than 1.1 P_a.

The containment leak rate test system piping is also designed for use during the performance of the containment structural integrity test. The instrumentation used for the structural integrity test may be different than that used for the integrated leak rate test.

6.2.5.1.3 Codes and Standards

The containment leak rate test system is designed to conform to the applicable codes and standards listed in Section 3.2. The containment leak testing program satisfies 10 CFR 50, Appendix J requirements.

6.2.5.2 System Description

6.2.5.2.1 General Description

The containment leak rate test system is illustrated on Figure 6.2.5-1. Unless otherwise indicated on the figure, piping and instrumentation is permanently installed. Fixed test connections used for Type C testing of piping penetrations are not shown on Figure 6.2.5-1. These connections are not part of the containment leak rate test system and are shown on the applicable system piping and instrument diagram figure.

Air compressor assemblies used for Type A testing are temporarily installed and are connected to the permanent system piping. The number and capacity of the compressors is sufficient to pressurize the containment with air to a pressure of P_a at a maximum containment pressurization rate of about 5 psi/hour. The compressor assemblies include additional equipment, such as air coolers, moisture separators and air dryers to reduce the moisture content of the air entering containment.

Temperature and humidity sensors are installed inside containment for Type A testing. Data acquisition hardware and instrumentation is available outside containment. Instrumentation not required during normal plant operation may be installed temporarily for the Type A tests.

The system is designed to permit depressurization of the containment at a maximum rate of 10 psi/hour.

Portable leak rate test panels are used to perform Type C containment isolation valve leak testing using air or nitrogen. The panels are also used for Type B testing of penetrations, for which there is no permanently installed test equipment. The panels include pressure regulators, filters, pressure gauges and flow instrumentation, as required to perform specific tests.

6.2.5.2.2 System Operation

Containment Integrated Leak Rate Test (Type A)

An integrated leak rate test of the primary reactor containment is performed prior to initial plant operation, and periodically thereafter, to confirm that the total leakage from the containment does not exceed the maximum allowable leak rate. The allowable leak rate specified in the test criteria is less than the maximum allowable containment leak rate, in accordance with 10 CFR 50, Appendix J.

Following construction of the containment and satisfactory completion of the structural integrity test, described in subsection 3.8.2.7, a preoperational Type A test is performed as described in Chapter 14. Additional Type A tests are conducted during the plant life, at intervals in accordance with the technical specifications, Chapter 16.

- **Pretest Requirements**

Prior to performing an integrated leak rate test, a number of pretest requirements must be satisfied as described in this subsection.

A general inspection of the accessible interior and exterior surfaces of the primary containment structure and components is performed to uncover any evidence of structural deterioration that could affect either the containment structural integrity or leak tightness. If there is evidence of structural deterioration, corrective action is taken prior to performing the Type A test. The structural deterioration and corrective action are reported in accordance with 10 CFR 50, Appendix J. Except as described above, during the period between the initiation of the containment inspection and the performance of the Type A test, no repairs or adjustments are made so that the containment can be tested in as close to the "as-is" condition as practical.

Containment isolation valves are placed in their post-accident positions, identified in Table 6.2.3-1, unless such positioning is impractical or unsafe. Test exceptions to post-accident valve positioning are identified in Table 6.2.3-1 or are discussed in the test report. Closure of containment isolation valves is accomplished by normal operation and with no preliminary exercising or adjustments (such as tightening of a valve by manual handwheel after closure by the power actuator). Valve closure malfunctions or valve leakage that requires corrective action before the test is reported in conjunction with the Type A test report.

Those portions of fluid systems that are part of the reactor coolant pressure boundary and are open directly to the containment atmosphere under post-accident conditions and become an extension of the boundary of the containment, are opened or vented to the containment atmosphere prior to and during the test.

Portions of systems inside containment that penetrate containment and could rupture as a result of a loss of coolant accident are vented to the containment atmosphere and drained of water to the extent necessary to provide exposure of the containment isolation valves to containment air test pressure and to allow them to be subjected to the full differential test pressure, except that:

- Systems that are required to maintain the plant in a safe condition during the Type A test remain operable and are not vented.

- Systems that are required to establish and maintain equilibrium containment conditions during Type A testing remain operable and are not vented.

- Systems that are normally filled with water and operating under post-accident conditions are not vented.

Systems not required to be vented and drained for Type A testing are identified in Table 6.2.3-1. The leak rates for the containment isolation valves in these systems, measured by Type C testing, are reported in the Type A test report.

Tanks inside the containment are vented to the containment atmosphere as necessary to protect them from the effects of external test pressure and/or to preclude leakage which could affect the accuracy of the test results. Similarly, instrumentation and other components that could be adversely affected by the test pressure are vented or removed from containment.

The containment atmospheric conditions are allowed to stabilize prior to the start of the Type A test consistent with the guidance of ANSI-56.8. The containment recirculation cooling system and central chilled water system are operated as necessary prior to, and during, the test to maintain stable test conditions.

- **Test Method**

The Type A test is conducted in accordance with ANSI-56.8, using the absolute method. The test duration is established consistent with ANSI-56.8 following the stabilization period. Periodic measurements of containment pressure, dry bulb temperatures and dew point temperatures (water vapor pressure) are used to determine the decrease in the mass of air in the containment over time. A standard statistical analysis of the data is conducted consistent with recommendations of ANSI-56.8.

The accuracy of the Type A test results is then verified by a supplemental verification test. The supplemental verification test is performed using methodology consistent with the recommendations described in ANSI-56.8.

Test criteria for the Type A test are given in the technical specifications. If any Type A test fails to meet the criteria, the test schedule for subsequent tests is adjusted in accordance with 10 CFR 50, Appendix J as defined in the Containment Leakage Rate Testing Program.

During the period between the completion of one Type A test and the initiation of the containment inspection for the subsequent Type A test, repairs or adjustments are made to components identified as exceeding individual leakage limits, as soon as practical after such leakage is identified.

Containment Penetration Leak Rate Tests (Type B)

The following containment penetrations receive preoperational and periodic Type B leak rate tests in accordance with ANSI-56.8 with test intervals as defined by NEI 94-01 (Reference 30):

- Penetrations whose design incorporates resilient seals, gaskets or sealant compounds
- Air locks and associated door seals
- Equipment and access hatches and associated seals
- Electrical penetrations

Containment penetrations subject to Type B tests are illustrated in Figure 6.2.5-1.

The fuel transfer tube penetration is sealed with a blind flange inside containment. The flanged joint is fitted with testable seals as shown in Figure 3.8.2-4. The two expansion bellows used on the fuel transfer tube penetration are not part of the leakage-limiting boundary of the containment.

The personnel hatches (airlocks) are designed to be tested by internal pressurization. The doors of the personnel hatches have testable seals as shown in Figure 3.8.2-3. Mechanical and electrical penetrations on the personnel hatches are also equipped with testable seals. The hatch cover flanges for the main equipment and maintenance hatches have testable seals as shown in Figure 3.8.2-2. Containment electrical penetrations have testable seals as shown in Figure 3.8.2-6.

Type B leak tests are performed by local pressurization using the test connections shown on Figure 6.2.5-1. Unless otherwise noted in Table 6.2.3-1, the test pressure is not less than the calculated containment peak accident pressure, P_a. Either the pressure decay or the flowmeter test method is used. These test methods and the test criteria are presented below for Type C tests.

Containment Isolation Valve Leak Rate Tests (Type C)

Containment isolation valves receive preoperational and periodic Type C leak rate tests in accordance with ANSI-56.8 with test intervals as defined by NEI 94-01 (Reference 30). A list of containment isolation valves subject to Type C tests is provided in Table 6.2.3-1. Containment isolation valve arrangement and test connections provided for Type C testing are illustrated on the applicable system piping and instrument diagram figure.

Type C leak tests are performed by local pressurization. Each valve to be tested is closed by normal means without any preliminary exercising or adjustments. Piping is drained and vented as needed and a test volume is established that, when pressurized, will produce a differential pressure across the valve. Table 6.2.3-1 identifies the direction in which the differential pressure is applied.

Isolation valves whose seats may be exposed to the containment atmosphere subsequent to a loss of coolant accident are tested with air or nitrogen at a pressure not less than P_a. Valves in lines which are designed to be, or remain, filled with a liquid for at least 30 days subsequent to a loss of coolant accident are leak rate tested with that liquid at a pressure not less than 1.1 times P_a. Isolation valves tested with liquid are identified in Table 6.2.3-1.

Isolation valves are tested using either the pressure decay or flowmeter method. For the pressure decay method the test volume is pressurized with air or nitrogen. The rate of decay of pressure in the known volume is monitored to calculate the leak rate. For the flowmeter method pressure is maintained in the test volume by supplying air or nitrogen through a calibrated flowmeter. The measured makeup flow rate is the isolation valve leak rate.

The leak rates of penetrations and valves subject to Type B and C testing are combined in accordance with 10 CFR 50, Appendix J. As each Type B or C test, or group of tests, is completed the combined total leak rate is revised to reflect the latest results. Thus, a reliable summary of containment leaktightness is maintained current. Leak rate limits and the criteria for the combined leakage results are described in the technical specifications.

Scheduling and Reporting of Periodic Tests

Schedules for the performance of periodic Type A, B, and C leak rate tests are in accordance with the technical specifications, Chapter 16 as specified in the Containment Leakage Rate Testing Program. Provisions for reporting test results are described in the Containment Leakage Rate Testing Program.

Type B and C tests may be conducted at any time that plant conditions permit, provided that the time between tests for any individual penetration or valve does not exceed the maximum allowable interval specified in the Containment Leakage Rate Testing Program.

Special Testing Requirements

AP600 does not have a subatmospheric containment or a secondary containment. There are no containment isolation valves which rely on a fluid seal system. Thus, there are no special testing requirements.

6.2.5.2.3 Component Description

The system pressurization equipment is temporarily installed for Type A testing. In addition to one or more compressors, this hardware includes components such as aftercoolers, moisture separators, filters and air dryers. The hardware characteristics may vary from test to test.

The flow control valve in the pressurization line is a leaktight valve capable of throttling to a low flow rate.

6.2.5.2.4 Instrumentation Applications

For Type A testing, instruments are provided to measure containment absolute pressure, dry bulb temperature, dew point temperature, air flow rate, and atmospheric pressure. Data acquisition equipment scans, processes and records data from the individual sensors. For Type B and C testing, instruments are provided to measure pressure, dry bulb temperature, and flow rate.

The quantity and location of Type A instrumentation and permanently installed Type B instrumentation, is indicated on Figure 6.2.5-1. The type, make and range of test instruments may vary from test to test. The instrument accuracy must meet the criteria of Reference 13.

6.2.5.3 Safety Evaluation

The containment leak rate test system has no safety-related function, other than containment isolation and therefore requires no nuclear safety evaluation, other than containment isolation which is described in subsection 6.2.3.

6.2.5.4 Inservice Inspection/Inservice Testing

There are no special inspection or testing requirements for the containment leak rate test system. Test equipment is inspected and instruments are calibrated in accordance with ANSI-56.8 criteria and the requirements of the test procedure.

6.2.6 Combined License Information for Containment Leak Rate Testing

The Combined License applicant is responsible for developing a "Containment Leakage Rate Testing Program" which will identify which Option is to be implemented under 10 CFR 50, Appendix J. Option A defines a prescriptive-based testing approach whereas option B defines a performance-based testing program.

6.2.7 References

1. DELETED

2. "Ice Condenser Containment Pressure Transient Analysis Methods," WCAP-8077, March, 1973 (Proprietary), WCAP-8078 (Non-Proprietary).

3. Shepard, R. M., et al., "Westinghouse Mass and Energy Release Data for Containment Design," WCAP-8264-P-A, June 1975 (Proprietary), and WCAP-8312-A, Revision 2, August 1975 (Non-Proprietary).

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6. Burnett, T. W. T., "LOFTRAN Code Description," WCAP-7907-P-A (Proprietary) and WCAP-7907-A (Non-Proprietary), June 1984.

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8. Branch Technical Position CSB6-1, "Minimum Containment Pressure Model for PWR ECCS Performance Evaluation."

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15. Thomas C. L. Catalytic Processes and Proven Catalysts, Academic Press, 1970.

17. J. Rohde, et al., "Hydrogen Mitigation by Catalytic Recombiners and Ignition During Severe Accidents," Third International Conference on Containment Design and Operation, Canadian Nuclear Society, Toronto, Ontario, October 19-21, 1994.

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19. EPRI Report, "NIS passive autocatalytic recombiner Depletion Rate Equation for Evaluation of Hydrogen Recombination During AP600 Design Basis Accident," EPRI ALWR Program, November 15, 1995.

20. WCAP-14407 (Proprietary) and WCAP-14408 (Non-Proprietary) WGOTHIC Application to AP600," Revision 3, April 1998.

21. EPRI Report, "Evaluation of Quality Assurance Applied to Battelle Tests of NIS Passive Autocatalytic Recombiner," EPRI ALWR Program, October 1995.

22. NUREG-737, "Clarification of TMI Action Plan Requirements," October, 1980

23. NUREG-718. Rev. 2, "Licensing Requirements for Pending Applications for Construction Permits and Manufacturing License," January, 1982.

24. ANSI/ASCE-8-90, Specification for the Design of Cold Formed Stainless Steel Structural Members

25. NUREG-0800, Section 6.2.1.1.A, "PWR Dry Containments, Including Subatmospheric Containments."

26. "AP600 Accident Analyses-Evaluation Models," WCAP-14601, Revision 2 (Proprietary), June 1998.

27. Thomas T. Martin, "AP600 Use of Passive Autocatalytic Recombiners (PARs) for Design Basis Hydrogen Control" to Mr. Nicholas Liparulo, April 1, 1997.

28. EPRI Report, "The Effects of Inhibitors and Poisons on the Performance of Passive Autocatalytic Recombiners for Combustible Gas Control in ALWRs," May 22, 1997.

29. EPRI Report TR-107517, Volumes 1, 2, and 3, "Generic Model Tests of Passive Autocatalytic Recombiners (PARs) for Combustible Gas Control in Nuclear Power Plants," June 1997.

30. Nuclear Energy Institute Report, NEI 94-01, "Industry Guidelines for Implementing Performance Based Option of 10 CFR 50, Appendix J," Revision 0.

31. Carlin, E. L. and U. Bachrach, "LOFTRAN and LOFTTR2 AP600 Code Applicability Document," WCAP-14234, Revision 1 (Proprietary), June 1997.

Table 6.2.1.1-1

SUMMARY OF CALCULATED PRESSURES AND TEMPERATURES

Break	Peak Pressure (psig)	Available[1] Margin (psi)	Peak Temperature (°F)
Double-ended hot leg guillotine	39.6	5.4	391.1
Double-ended cold leg guillotine	43.4	1.6	281.2
Full main steamline DER, 102% power, MSIV failure	43.7	1.3	370.9
Full main steamline DER, 30% power, MSIV failure	44.1	0.9	368.2

1. Design Pressure is 45 psig

Table 6.2.1.1-2

INITIAL CONDITIONS

Internal Temperature (°F)	120
Pressure (psia)	15.7
Relative Humidity (%)	0
Net Free Volume (ft^3)	1.7 E+06
External Temperature (°F)	115 dry bulb 80 wet bulb

Table 6.2.1.1-3

RESULTS OF POSTULATED ACCIDENTS

Criterion	Acceptance Criterion Value	Lumped DEHLG LOCA Value	Lumped DECLG LOCA Value	102% Power MSLB Value	30% Power MSLB Value	External Pressuri-zation Value
GDC 16 & GDC 50 10% Margin to Design Pressure	< 45.0 psig	39.6	43.4	43.7	44.1	--
GDC 38 Rapidly Reduce Containment Pressure	< 22.5 psig	--	19.2 for 24 hrs	--	--	--
GDC 38 & 50 External Pressure	< 3 psid	--	--	--	--	2.04
GDC 38 & GDC 50 Containment Heat Removal Single Failure	Most Severe	One Train of PCS Water Supply	One Train of PCS Water Supply	One Train of PCS Supply	One Train of PCS Supply	

Tables 6.2.1.1-4 through 6.2.1.1-7 DELETED

Table 6.2.1.1-8

PHYSICAL PROPERTIES OF PASSIVE HEAT SINKS

Material	Density (lbm/ft^3)	Thermal Conductivity (Btu/hr-ft-°F)	Specific Heat (Btu/lbm-°F)	Dry Emis.	Wet Emis.
Epoxy	105	0.1875	0.35	0.81	0.95
Carbon Steel	490.7	23.6	0.107	0.81	0.95
Concrete	140.	0.83	0.19	0.81	0.95
Stainless Steel	501.	9.4	0.12	0.81	0.95
Carbo Zinc	207.5	1.21	0.15	0.81	0.95
Oxidized Carbo Zinc	207.5	0.302	0.15	0.81	0.95
Carbo Zinc-PCS Inside Surface	207.5	0.302	0.15	1e-10	1e-10
Air @ 0°F	0.0864	0.0131	0.240	1e-10	1e-10
Air @ 250°F	0.056	0.0192	0.242	1e-10	1e-10
Air @ 500°F	0.0414	0.0246	0.248	1e-10	1e-10

Table 6.2.1.2-1 (Sheet 1 of 6)

LISTING OF LINES NOT LBB QUALIFIED
AND THE CALCULATED MAXIMUM DIFFERENTIAL PRESSURES

AP600 Room #	Possible[1] Pipe Rupture	Design Differential Pressure (psi)	Maximum Differential[2] Pressure (psi)	Table for M&E Data
11104	None	5.0	NA	NA
11105	None	5.0	NA	NA
11201	4" Pressurizer Spray	5.0	< 4.0	6.2.1.3-6
11202	None	5.0	NA	NA
11204	3" Regen HX to SG	5.0	< 2.9	6.2.1.3-2
	3" Purification from CL to Regen HX		< 2.9	6.2.1.3-2
11205	None	5.0	NA	NA
11206	None	5.0	NA	NA
11207	None	5.0	NA	NA
11208	None	5.0	NA	NA
11209 North	None	5.0	NA	NA

Table 6.2.1.2-1 (Sheet 2 of 6)

LISTING OF LINES NOT LBB QUALIFIED
AND THE CALCULATED MAXIMUM DIFFERENTIAL PRESSURES

AP600 Room #	Possible[1] Pipe Rupture	Design Differential Pressure (psi)	Maximum Differential[2] Pressure (psi)	Table for M&E Data
11209 Center	3"Purification from Prz Spray	5.0	< 4.2	6.2.1.3-7
	3"Purification to PRHR Return		< 4.2	6.2.1.3-7
	3" Regen HX to Letdown HX		< 4.2	6.2.1.3-7
	3" RHR HX		< 4.2	6.2.1.3-7
	3" Regen HX to RNS pump		< 4.2	6.2.1.3-7
11209 South	3" Regen HX to Letdown HX	5.0	< 4.3	6.2.1.3-7
11209 Pipe Tunnel	3"Purification from Prz Spray to Regen HX	7.5	< 6.2	6.2.1.3-7
	3"Purification from Regen HX to PRHR Return	7.5	< 6.2	6.2.1.3-7

Table 6.2.1.2-1 (Sheet 3 of 6)

LISTING OF LINES NOT LBB QUALIFIED AND THE CALCULATED MAXIMUM DIFFERENTIAL PRESSURES

AP600 Room #	Possible[(1)] Pipe Rupture	Design Differential Pressure (psi)	Maximum Differential[(2)] Pressure (psi)	Table for M&E Data
	4" SG Blowdown		< 6. 75	6.2.1.3-5
11300	None	5.0	NA	NA
11301	3" Purification	5.0	< 4.0	6.2.1.3-2 6.2.1.3-3
11302	None	5.0	NA	NA
11303	4" Pressurizer Spray	5.0	< 3.7	6.2.1.3-6
11304	3" Purification to PRHR return	5.0	< 3.6	6.2.1.3-2
	2" CVS Purification to Prz Spray		< 3.6	Bounded by larger break size
11305	None	5.0	NA	NA
11400	6" Startup Feedwater	5.0	NA	NA
11401	4" SG Blowdown	5.0	< 2.9	6.2.1.3-5

Table 6.2.1.2-1 (Sheet 4 of 6)

LISTING OF LINES NOT LBB QUALIFIED
AND THE CALCULATED MAXIMUM DIFFERENTIAL PRESSURES

AP600 Room #	Possible[1] Pipe Rupture	Design Differential Pressure (psi)	Maximum Differential[2] Pressure (psi)	Table for M&E Data
11402	4" SG Blowdown	5.0	< 2.9	6.2.1.3-5
11403	3" Letdown	5.0	< 4.5	6.2.1.3-3
	2" Aux Spray		< 4.5	Bounded by larger break size
	4" Prz Spray at 4 x 2 TEE		< 4.5	6.2.1.3-6
	4" Prz Spray at Anchor		< 4.5	6.2.1.3-6
11500	None	5.0	NA	NA
11501	None	5.0	NA	NA
11502	None	5.0	NA	NA
11503	4" Pressurizer Spray	5.0	< 4.0	6.2.1.3-6
11504	None	5.0	NA	NA

Table 6.2.1.2-1 (Sheet 5 of 6)

LISTING OF LINES NOT LBB QUALIFIED
AND THE CALCULATED MAXIMUM DIFFERENTIAL PRESSURES

AP600 Room #	Possible[1] Pipe Rupture	Design Differential Pressure (psi)	Maximum Differential[2] Pressure (psi)	Table for M&E Data
11601	16" Main Feedwater	5.0	NA	NA
	6" Startup Feedwater		NA	NA
11602	16" Main Feedwater	5.0	NA	NA
	6" Startup Feedwater		NA	NA
11603	4" ADS	5.0	NA	NA
11701	None	5.0	NA	NA

Table 6.2.1.2-1 (Sheet 6 of 6)

LISTING OF LINES NOT LBB QUALIFIED
AND THE CALCULATED MAXIMUM DIFFERENTIAL PRESSURES

AP600 Room #	Possible[1] Pipe Rupture	Design Differential Pressure (psi)	Maximum Differential[2] Pressure (psi)	Table for M&E Data
11702	None	5.0	NA	NA
11703	4" ADS	5.0	NA	NA

__Notes:__

1. "None" indicates that there are no High Energy Lines >1" in diameter that are not qualified to LBB.

2. Structures are designed to a pressurization load of 5.0 psig; except the CVS room pipe tunnel which is designed to a pressurization load of 7.5 psig. See DCD Section 3.8.3.5.

3. "NA" indicates that no calculation was performed because no rupture was postulated or that the line was postulated to rupture in a region with a large free volume so compartment differential pressures would be negligible.

Table 6.2.1.3-1

SHORT-TERM MASS AND ENERGY INPUTS

Vessel Outlet Temperature (°F)	597.0
Vessel Inlet Temperature (°F)	528.6
Initial RCS Pressure (PSIA)	2300.0
Zaloudek Coefficient (CK1)	1.018
Zaloudek Coefficient (C1)	0.9

Table 6.2.1.3-2

SHORT-TERM 3-INCH COLD-LEG BREAK MASS AND ENERGY RELEASES

Time (sec)	Mass (lbm/sec)	Energy (Btu/sec)
0.0	0.0	0.0
0.001	3186.8	1.7084E+6
0.05	3186.8	1.7084E+6
1.000	3186.8	1.7084E+6
5.000	3186.8	1.6591E+6
7.000	3186.8	1.6225E+6
10.00	3186.8	1.6005E+6

Table 6.2.1.3-3

SHORT-TERM 3-INCH HOT-LEG
BREAK MASS AND ENERGY RELEASES

Time (sec)	Mass (lbm/sec)	Energy (Btu/sec)
0.0	0.0	0.0
0.001	2514.2	1.5623E+6
0.05	2514.2	1.5623E+6
1.000	2514.2	1.5640E+6
5.000	2514.2	1.6947E+6
7.000	2514.2	1.7966E+6
10.00	2514.2	1.8406E+6

Table 6.2.1.3-4

MAIN STEAM LINE BREAK MASS AND ENERGY

Time (sec)	**Mass (lbm/sec)**	**Energy (Btu/sec)**
0	2100	2504900
3.65	2100	2504900
4.65	6610	3980500
5.65	6970	4084400
6.65	7060	4109600
7.65	7060	4109600
8.65	7020	4099700
9.65	6940	4075900
10.65	6820	4040900
11.65	6680	4001300
12.65	6520	3951800
13.65	6340	3896600
14.65	6190	3852700
15.65	6000	3792600
16.65	5830	3732800
17.65	5680	3689700

Table 6.2.1.3-5

4" SG BLOWDOWN LINE MASS AND ENERGY RELEASES

Time (sec)	Total Mass (lbm/sec)	Energy (Btu/sec)
0.0	0.0	0.0
0.001	1932.0	10.48 E+5
0.5	1932.0	10.48 E+5
0.51	966.0	5.24 E+5
172.5	966.0	5.24 E+5
172.6	0.0	0.0

Table 6.2.1.3-6

PRESSURIZER SPRAY LINE BREAK RELEASES

Time (sec)	Mass (lbm/sec)	Energy (Btu/sec)
0	3006.872	1794802
0.0503	2957.944	1768521
0.102	2941.763	1759619
0.501	2856.777	1711344
0.763	2854.027	1707538
1	2860.371	1708709
1.075	2860.858	1708365
2	2766.115	1650733
3	2666.345	1590401
4	2564.804	1529641
5	2459.947	1467666

Table 6.2.1.3-7

SHORT TERM 3-INCH SINGLE-ENDED COLD-LEG BREAK MASS AND ENERGY RELEASES

Time (sec)	Mass (lbm/sec)	Energy (Btu/sec)
0.0	0.0	0.0
0.001	1593.4	8.5420E+05
0.050	1593.4	8.5420E+05
1.001	1593.4	8.5420E+05
5.000	1593.4	8.2955E+05
7.000	1593.4	8.1125E+05
10.00	1593.4	8.0025E+05

Table 6.2.1.3-8

BASIS FOR LONG-TERM ANALYSIS

Number of Loops	2
Active Core Length (ft)	12.0
Core Power, license application (MWt)	1933
Nominal Vessel Inlet Temperature (°F)	535.1
Nominal Vessel Outlet Temperature (°F)	600.0
Steam Pressure (psia)	821.0
Rod Array	17 x 17
Accumulator Temperature (°F)	120.0
Containment Design Pressure (psia)	59.7

Table 6.2.1.3-9 (Sheet 1 of 5)

**LONG-TERM DECLG BREAK
POST-BLOWDOWN MASS AND ENTHALPY RELEASES**

Time (sec)	Mass (lbm/sec)	Enthalpy (Btu/lbm)	Mass (lbm/sec)	Enthalpy (Btu/lbm)
0.00	0.00	0.00	0.00	0.00
0.05	15786.30	530.77	19284.30	532.59
0.10	26421.10	530.90	18445.50	531.40
0.25	27548.80	532.12	18238.00	531.82
0.40	30050.80	532.57	19214.10	532.61
0.60	28706.50	532.75	18401.60	533.69
0.70	30173.00	532.68	21548.40	534.22
0.80	30429.10	532.63	20146.30	535.01
1.00	30346.80	532.52	21069.80	536.06
1.40	30335.60	534.91	19790.00	537.94
1.60	29284.30	538.03	16356.80	538.59
1.80	30748.00	542.91	17495.40	539.03
1.90	28021.40	546.11	17741.70	539.21
2.20	25688.70	557.49	18312.90	539.71
2.40	25055.20	565.82	16570.70	540.00
2.70	19340.90	578.52	16827.60	540.54
2.90	16650.20	584.49	15809.00	540.99
3.30	14333.50	591.86	14307.30	542.02
3.40	14165.30	592.59	18868.90	542.33
4.00	12577.20	600.64	17602.20	545.52
4.20	12449.10	614.37	17530.90	546.76
4.30	11560.00	622.66	21076.90	547.11
4.50	11102.30	618.27	20749.00	549.13
4.80	11135.20	575.24	15792.00	550.73
5.00	11095.40	553.27	14564.00	550.80
5.25	12075.40	541.78	11728.20	550.80
5.50	11425.00	534.95	7595.10	550.72
5.75	11453.30	531.34	6477.90	550.53
6.25	11499.00	529.50	6756.00	549.63

Table 6.2.1.3-9 (Sheet 2 of 5)

LONG-TERM DECLG BREAK POST-BLOWDOWN MASS AND ENTHALPY RELEASES

Time (sec)	Mass (lbm/sec)	Enthalpy (Btu/lbm)	Mass (lbm/sec)	Enthalpy (Btu/lbm)
7.00	10284.70	529.08	5669.40	546.80
7.75	9620.00	528.01	5047.20	542.08
8.00	9962.50	530.79	4942.50	540.52
8.50	8013.90	559.19	4795.80	537.82
9.00	8205.80	547.13	4504.40	535.23
10.00	4320.80	808.28	4423.60	530.70
10.50	4457.00	710.61	4027.60	528.93
11.80	4408.30	708.80	3798.50	529.66
12.00	3585.40	827.97	3788.30	532.75
12.50	4222.80	644.79	3530.30	538.48
13.30	4152.00	653.54	3730.70	559.68
13.50	3987.90	658.47	1736.70	567.63
14.00	4072.60	622.82	1527.30	579.85
14.50	4051.70	629.00	2492.00	603.17
15.00	3773.50	634.87	2643.10	639.74
15.80	3888.00	585.06	2534.00	718.23
17.50	3628.40	519.54	2046.80	825.04
20.80	3387.70	484.99	1723.70	798.28
22.00	2973.80	485.34	1704.80	790.06
26.00	1481.10	338.13	377.60	922.14
28.80	64.70	295.21	0.00	1167.00
28.95	0.00	1167.40	7.08	1167.40
31.04	0.00	1167.40	40.02	1167.40
41.36	0.00	1167.40	184.74	1167.40
45.61	10.31	691.25	220.70	1167.40
51.72	69.45	266.68	218.49	1167.40
55.90	103.69	243.91	216.82	1167.40
60.04	133.50	234.53	215.13	1167.40
64.21	158.06	230.53	213.96	1167.40
70.62	189.99	227.88	212.12	1167.40
74.76	206.85	227.42	210.91	1167.40

Table 6.2.1.3-9 (Sheet 3 of 5)

LONG-TERM DECLG BREAK
POST-BLOWDOWN MASS AND ENTHALPY RELEASES

Time (sec)	Mass (lbm/sec)	Enthalpy (Btu/lbm)	Mass (lbm/sec)	Enthalpy (Btu/lbm)
81.10	227.51	227.80	209.07	1167.40
85.41	238.99	228.41	207.94	1167.40
89.55	248.08	229.20	206.86	1167.40
95.77	258.89	230.62	205.20	1167.40
99.91	264.30	231.69	204.10	1167.40
102.03	266.64	232.25	203.66	1167.40
112.05	271.00	235.93	201.69	1167.40
122.20	270.17	240.00	199.72	1167.40
132.26	266.17	244.12	197.76	1167.40
142.31	260.92	247.97	195.75	1167.40
152.48	253.45	252.08	193.74	1167.40
162.53	245.22	256.07	192.05	1167.40
172.77	235.98	260.24	190.35	1167.40
182.90	226.65	264.31	188.67	1167.40
193.03	219.17	267.51	186.85	1167.40
203.21	219.63	267.02	184.43	1167.40
223.50	218.65	266.27	180.31	1167.40
243.76	219.69	264.95	171.25	1167.40
264.17	214.51	265.42	167.52	1167.40
284.42	208.82	266.07	163.94	1167.40
304.42	203.34	266.63	160.49	1167.40
324.42	195.61	268.37	157.29	1167.40
344.42	187.83	270.26	154.20	1167.40
364.64	12.99	656.39	175.06	1167.40
384.64	16.65	587.86	170.33	1167.40
404.64	20.12	543.95	165.78	1167.40
424.64	23.02	515.91	161.77	1167.40
444.64	49.89	407.39	133.79	1167.40
464.64	52.36	400.01	130.22	1167.40
484.64	54.68	393.07	126.77	1167.40
504.64	56.89	386.51	123.46	1167.40

Table 6.2.1.3-9 (Sheet 4 of 5)

LONG-TERM DECLG BREAK POST-BLOWDOWN MASS AND ENTHALPY RELEASES

Time (sec)	Mass (lbm/sec)	Enthalpy (Btu/lbm)	Mass (lbm/sec)	Enthalpy (Btu/lbm)
524.64	58.96	380.28	120.27	1167.40
544.64	60.92	374.34	117.19	1167.40
564.64	62.77	368.65	114.23	1167.40
584.64	64.52	363.17	111.37	1167.40
604.64	66.17	357.91	108.61	1167.40
654.64	69.55	345.88	102.43	1167.40
704.64	72.42	334.73	96.78	1167.40
754.64	74.81	324.37	91.60	1167.40
804.64	76.79	314.65	86.84	1167.40
854.64	78.24	305.69	82.60	1167.40
904.64	79.36	297.25	78.70	1167.40
954.64	80.20	289.27	75.09	1167.40
1004.64	80.74	281.76	71.76	1167.40
1504.75	223.22	149.04	47.12	1167.40
2004.75	211.76	127.84	37.58	1167.40
3504.75	156.70	104.74	27.09	1167.40
4004.75	157.03	100.94	25.08	1167.40
6004.84	153.75	96.34	21.54	1167.40
7504.95	134.76	96.64	20.33	1167.40
8004.95	132.14	96.53	19.75	1167.40
10005.00	128.76	96.18	18.53	1167.40
15005.00	130.22	95.25	16.61	1167.40
20005.80	114.62	95.62	15.38	1167.40
26007.30	35.36	111.52	14.64	1167.40
30007.90	0.0	1168.30	16.32	1168.30
36008.10			15.38	1168.30
40000.00			14.76	1168.30
60000.00			13.20	1168.30

Table 6.2.1.3-9 (Sheet 5 of 5)

LONG-TERM DECLG BREAK
POST-BLOWDOWN MASS AND ENTHALPY RELEASES

Time (sec)	Mass (lbm/sec)	Enthalpy (Btu/lbm)	Mass (lbm/sec)	Enthalpy (Btu/lbm)
80000.00			12.19	1168.30
100000.00			11.42	1168.30
150000.00			10.10	1168.30
200000.00			9.20	1168.30
400000.00			7.15	1168.30
600000.00			6.07	1168.30
800000.00			5.38	1168.30
1000000.00			4.89	1168.30
1500000.00			4.11	1168.30
2000000.00			3.62	1168.30
4000000.00			2.57	1168.30

Table 6.2.1.3-10 (Sheet 1 of 2)

LONG-TERM DEHLG BREAK
BLOWDOWN MASS AND ENTHALPY RELEASES

Time (sec)	Mass (lbm/sec)	Enthalpy (Btu/lbm)
0.00	0.0	0.0
0.05	71107.2	619.4
0.10	69007.6	620.8
0.20	53663.5	637.6
0.25	48799.4	637.0
0.30	48512.1	634.6
0.50	46323.8	626.9
0.90	45759.7	614.7
1.30	43842.4	614.2
1.60	40909.6	615.1
2.10	41561.1	607.6
2.30	40918.9	602.6
2.90	41652.2	577.7
3.00	31464.3	688.4
3.10	35047.5	629.1
3.30	38017.0	615.0
3.60	38102.2	621.8
3.90	35045.8	625.9
4.40	31654.6	610.5
4.60	31149.8	606.9
5.25	31080.8	586.8
5.50	30494.5	580.1
6.00	26489.7	580.6
6.25	25396.8	579.8
6.75	25088.5	570.9
7.25	23247.5	563.1
7.75	16200.5	601.2
8.25	15895.9	586.6

Table 6.2.1.3-10 (Sheet 2 of 2)

LONG-TERM DEHLG BREAK
BLOWDOWN MASS AND ENTHALPY RELEASES

Time (sec)	Mass (lbm/sec)	Enthalpy (Btu/lbm)
8.50	15171.6	588.3
9.00	12573.3	610.0
10.30	8949.0	674.4
12.80	4225.1	983.7
13.80	3119.2	1088.1
15.50	1797.6	1211.5
16.50	1574.9	1073.8
17.00	978.2	1254.9
19.50	436.6	1264.1
20.30	307.4	1250.5
21.50	457.8	1253.2
22.30	307.4	1243.3
27.30	382.7	1081.0
27.50	116.9	1272.9
30.30	456.9	823.4
32.00	425.7	985.4
33.50	213.3	1280.8
34.50	594.4	1232.2
34.80	270.1	1205.8
35.30	652.8	1037.8
36.00	383.6	1244.8
36.20	53.4	1243.4

Table 6.2.1.4-1

SPECTRUM OF SECONDARY SYSTEM PIPE RUPTURES ANALYZED

Power Level	102%	70%	30%	0%
Full DER, MSIV failure	Full DER	Full DER	Full DER	Full DER
Full DER, MFWIV failure	Full DER	Full DER	Full DER	Full DER
Small DER, MSIV failure (ft^2)	0.4 0.1	0.4 0.1	0.4 0.1	0.4 0.1
Split Rupture (ft^2)	0.6[a]	0.598[a]	0.587[a]	0.571[a]

[a]As total area of two loops.

DER = double ended rupture
MSIV = main steam line isolation valve
MFWIV = main feedwater isolation valve

Table 6.2.1.4-2 (Sheet 1 of 10)

MASS AND ENTHALPY RELEASE DATA FOR THE CASE OF MAIN STEAM LINE FULL DOUBLE ENDED RUPTURE FROM 30% POWER LEVEL WITH FAULTED LOOP MAIN STEAM LINE ISOLATION VALVE FAILURE THAT PRODUCES HIGHEST CONTAINMENT PRESSURE

Initial steam generator mass (lbm)	: 164530
Mass added by feedwater flashing (lbm)	: 10390
Mass added from initial steamline header blowdown (lbm)	: 9970
Initial steam pressure (psia)	: 976.5
Feedwater line isolation at (sec)	: 7.92
Steam line isolation at (sec)	: 7.92

Time (sec)	Mass (lbm/sec)	Enthalpy (Btu/lbm)
0.0	12575	1193
0.1	12575	1193
0.2	12547	1193
0.3	12525	1193
0.4	12503	1193
0.5	12483	1193
0.6	12462	1193
0.7	12442	1193
0.8	12423	1193
0.9	12404	1193
1.0	12385	1193
1.1	5667	1194
1.2	5621	1194
1.3	5584	1194
1.4	5547	1195
1.5	5511	1195
1.6	5475	1195
1.7	5440	1195
1.8	5405	1195
1.9	5371	1196
2.0	5337	1196

Table 6.2.1.4-2 (Sheet 2 of 10)

MASS AND ENTHALPY RELEASE DATA FOR THE CASE OF MAIN STEAM LINE FULL DOUBLE ENDED RUPTURE FROM 30% POWER LEVEL WITH FAULTED LOOP MAIN STEAM LINE ISOLATION VALVE FAILURE THAT PRODUCES HIGHEST CONTAINMENT PRESSURE

Time (sec)	Mass (lbm/sec)	Enthalpy (Btu/lbm)
2.1	5305	1196
2.2	5272	1196
2.3	5241	1196
2.4	5210	1196
2.5	5179	1197
2.6	5149	1197
2.7	5119	1197
2.8	5089	1197
2.9	5060	1197
3.0	5032	1197
3.1	5004	1198
3.2	4975	1198
3.3	4948	1198
3.4	4921	1198
3.5	4894	1198
3.6	4868	1198
3.7	4843	1198
3.8	4818	1198
3.9	4792	1198
4.0	4767	1199
4.1	4743	1199
4.2	4719	1199
4.3	4696	1199
4.4	4673	1199
4.5	4650	1199
4.6	4628	1199

Table 6.2.1.4-2 (Sheet 3 of 10)

MASS AND ENTHALPY RELEASE DATA FOR THE CASE OF MAIN STEAM LINE FULL DOUBLE ENDED RUPTURE FROM 30% POWER LEVEL WITH FAULTED LOOP MAIN STEAM LINE ISOLATION VALVE FAILURE THAT PRODUCES HIGHEST CONTAINMENT PRESSURE

Time (sec)	Mass (lbm/sec)	Enthalpy (Btu/lbm)
4.7	4606	1199
4.8	4584	1199
4.9	4562	1199
5.0	4542	1200
5.1	4520	1200
5.2	4500	1200
5.3	4480	1200
5.4	4459	1200
5.5	4439	1200
5.6	4421	1200
5.7	4401	1200
5.8	4382	1200
5.9	4363	1200
6.0	4344	1200
6.1	4326	1200
6.2	4308	1200
6.3	4291	1201
6.4	4273	1201
6.5	4256	1201
6.6	4238	1201
6.7	4222	1201
6.8	4205	1201
6.9	4189	1201
7.0	4172	1201
7.1	4156	1201
7.2	4140	1201

Table 6.2.1.4-2 (Sheet 4 of 10)

MASS AND ENTHALPY RELEASE DATA FOR THE CASE OF MAIN STEAM LINE FULL DOUBLE ENDED RUPTURE FROM 30% POWER LEVEL WITH FAULTED LOOP MAIN STEAM LINE ISOLATION VALVE FAILURE THAT PRODUCES HIGHEST CONTAINMENT PRESSURE

Time (sec)	Mass (lbm/sec)	Enthalpy (Btu/lbm)
7.3	4125	1201
7.4	4109	1201
7.5	4094	1201
7.6	4079	1201
7.7	4065	1201
7.8	4049	1201
7.9	4036	1201
8.0	4021	1201
8.1	4007	1202
8.2	3997	1202
8.3	3984	1202
8.4	3973	1202
8.5	3962	1202
8.6	3950	1202
8.7	3940	1202
8.8	3929	1202
8.9	3918	1202
9.0	3907	1202
9.1	1872	1203
9.2	1868	1203
9.3	1864	1203
9.4	1860	1203
9.5	1856	1203
9.6	1852	1203
9.7	1848	1203
9.8	1844	1203

Table 6.2.1.4-2 (Sheet 5 of 10)

MASS AND ENTHALPY RELEASE DATA FOR THE CASE OF MAIN STEAM LINE FULL DOUBLE ENDED RUPTURE FROM 30% POWER LEVEL WITH FAULTED LOOP MAIN STEAM LINE ISOLATION VALVE FAILURE THAT PRODUCES HIGHEST CONTAINMENT PRESSURE

Time (sec)	Mass (lbm/sec)	Enthalpy (Btu/lbm)
9.9	1840	1203
10	1836	1203
11	1796	1203
12	1757	1203
13	1718	1204
14	1679	1204
15	1639	1204
16	1601	1204
17	1561	1204
18	1517	1204
19	1470	1204
20	1422	1204
21	1385	1204
22	1340	1204
23	1295	1204
24	1252	1204
25	1211	1204
26	1171	1204
27	1133	1204
28	1097	1204
29	1062	1204
30	1029	1204
31	998	1204
32	968	1203
33	939	1203
34	912	1203

Table 6.2.1.4-2 (Sheet 6 of 10)

MASS AND ENTHALPY RELEASE DATA FOR THE CASE OF MAIN STEAM LINE FULL DOUBLE ENDED RUPTURE FROM 30% POWER LEVEL WITH FAULTED LOOP MAIN STEAM LINE ISOLATION VALVE FAILURE THAT PRODUCES HIGHEST CONTAINMENT PRESSURE

Time (sec)	Mass (lbm/sec)	Enthalpy (Btu/lbm)
35	888	1203
36	866	1203
37	845	1202
38	824	1202
39	805	1202
40	786	1202
41	768	1201
42	751	1201
43	735	1201
44	719	1201
45	704	1201
46	689	1200
47	676	1200
48	662	1200
49	650	1200
50	637	1199
55	584	1198
60	541	1197
65	508	1196
70	481	1195
75	459	1194
80	442	1194
85	428	1193
90	417	1193
95	408	1192
100	399	1192

Table 6.2.1.4-2 (Sheet 7 of 10)

MASS AND ENTHALPY RELEASE DATA FOR THE CASE OF MAIN STEAM LINE FULL DOUBLE ENDED RUPTURE FROM 30% POWER LEVEL WITH FAULTED LOOP MAIN STEAM LINE ISOLATION VALVE FAILURE THAT PRODUCES HIGHEST CONTAINMENT PRESSURE

Time (sec)	Mass (lbm/sec)	Enthalpy (Btu/lbm)
105	392	1192
110	385	1191
115	378	1191
120	372	1191
125	365	1190
130	358	1190
135	351	1190
140	344	1189
145	337	1189
150	330	1189
155	324	1188
160	318	1188
165	312	1188
170	306	1187
175	301	1187
180	297	1187
185	292	1186
190	288	1186
195	285	1186
200	281	1186
205	278	1185
210	275	1185
215	272	1185
220	270	1185
225	267	1185
230	265	1185

Table 6.2.1.4-2 (Sheet 8 of 10)

MASS AND ENTHALPY RELEASE DATA FOR THE CASE OF MAIN STEAM LINE FULL DOUBLE ENDED RUPTURE FROM 30% POWER LEVEL WITH FAULTED LOOP MAIN STEAM LINE ISOLATION VALVE FAILURE THAT PRODUCES HIGHEST CONTAINMENT PRESSURE

Time (sec)	Mass (lbm/sec)	Enthalpy (Btu/lbm)
235	263	1184
240	261	1184
245	259	1184
250	256	1184
255	254	1184
260	252	1184
265	250	1183
270	248	1183
275	246	1183
280	244	1183
285	242	1183
290	240	1183
295	237	1182
300	235	1182
306	233	1182
312	230	1182
318	227	1181
342	225	1181
330	222	1181
336	220	1181
324	218	1181
348	215	1180
354	213	1180
360	211	1180
366	208	1180
372	206	1180

Table 6.2.1.4-2 (Sheet 9 of 10)

MASS AND ENTHALPY RELEASE DATA FOR THE CASE OF MAIN STEAM LINE FULL DOUBLE ENDED RUPTURE FROM 30% POWER LEVEL WITH FAULTED LOOP MAIN STEAM LINE ISOLATION VALVE FAILURE THAT PRODUCES HIGHEST CONTAINMENT PRESSURE

Time (sec)	Mass (lbm/sec)	Enthalpy (Btu/lbm)
378	204	1179
384	202	1179
390	200	1179
396	198	1179
402	196	1179
408	194	1178
414	192	1178
420	190	1178
426	188	1178
432	186	1178
438	184	1177
444	183	1177
450	181	1177
456	179	1177
462	177	1177
468	175	1176
474	173	1176
480	168	1176
486	163	1175
492	157	1174
498	152	1174
504	145	1173
510	138	1172
516	130	1170
522	118	1168
528	102	1166

Table 6.2.1.4-2 (Sheet 10 of 10)

MASS AND ENTHALPY RELEASE DATA FOR THE CASE OF MAIN STEAM LINE FULL DOUBLE ENDED RUPTURE FROM 30% POWER LEVEL WITH FAULTED LOOP MAIN STEAM LINE ISOLATION VALVE FAILURE THAT PRODUCES HIGHEST CONTAINMENT PRESSURE

Time (sec)	Mass (lbm/sec)	Enthalpy (Btu/lbm)
534	84	1162
540	63	1156
546	9	1151
552	0	1151

Table 6.2.1.4-3 (Sheet 1 of 8)

MASS AND ENTHALPY RELEASE DATA FOR THE CASE OF MAIN STEAM LINE FULL DOUBLE ENDED RUPTURE FROM 102% POWER LEVEL WITH FAULTED LOOP MAIN STEAM LINE ISOLATION VALVE FAILURE THAT PRODUCES HIGHEST CONTAINMENT TEMPERATURE

Initial steam generator mass (lbm)	: 137052
Mass added from initial steamline header blowdown (lbm)	: 7810
Mass added by feedwater flashing (lbm)	: 9564
Initial steam pressure (psia)	: 843.2
Feedwater line isolation at (sec)	: 8.56
Steam line isolation at (sec)	: 8.56

Time (sec)	Mass (lbm/sec)	Enthalpy (Btu/lbm)
0.0	10886	1197
0.1	10886	1197
0.2	10871	1197
0.3	10860	1197
0.4	10849	1197
0.5	10839	1197
0.6	10829	1197
0.7	10819	1197
0.8	10809	1197
0.9	10800	1197
1.0	4962	1197
1.1	4938	1197
1.2	4919	1198
1.3	4900	1198
1.4	4882	1198
1.5	4863	1198
1.6	4846	1198
1.7	4828	1198
1.8	4811	1198
1.9	4794	1198
2.0	4777	1198

Table 6.2.1.4-3 (Sheet 2 of 8)

MASS AND ENTHALPY RELEASE DATA FOR THE CASE OF MAIN STEAM LINE FULL DOUBLE ENDED RUPTURE FROM 102% POWER LEVEL WITH FAULTED LOOP MAIN STEAM LINE ISOLATION VALVE FAILURE THAT PRODUCES HIGHEST CONTAINMENT TEMPERATURE

Time (sec)	Mass (lbm/sec)	Enthalpy (Btu/lbm)
2.1	4760	1198
2.2	4743	1198
2.3	4727	1199
2.4	4711	1199
2.5	4695	1199
2.6	4680	1199
2.7	4664	1199
2.8	4649	1199
2.9	4634	1199
3.0	4619	1199
3.1	4605	1199
3.2	4591	1199
3.3	4577	1199
3.4	4563	1199
3.5	4550	1199
3.6	4536	1199
3.7	4522	1199
3.8	4510	1200
3.9	4497	1200
4.0	4485	1200
4.1	4472	1200
4.2	4459	1200
4.3	4448	1200
4.4	4436	1200
4.5	4424	1200
4.6	4413	1200

Table 6.2.1.4-3 (Sheet 3 of 8)

MASS AND ENTHALPY RELEASE DATA FOR THE CASE OF MAIN STEAM LINE FULL DOUBLE ENDED RUPTURE FROM 102% POWER LEVEL WITH FAULTED LOOP MAIN STEAM LINE ISOLATION VALVE FAILURE THAT PRODUCES HIGHEST CONTAINMENT TEMPERATURE

Time (sec)	Mass (lbm/sec)	Enthalpy (Btu/lbm)
4.7	4403	1200
4.8	4392	1200
4.9	4382	1200
5.0	4372	1200
5.1	4361	1200
5.2	4351	1200
5.3	4341	1200
5.4	4332	1200
5.5	4322	1200
5.6	4312	1200
5.7	4303	1200
5.8	4294	1200
5.9	4285	1201
6.0	4275	1201
6.1	4267	1201
6.2	4258	1201
6.3	4249	1201
6.4	4241	1201
6.5	4233	1201
6.6	4224	1201
6.7	4216	1201
6.8	4208	1201
6.9	4200	1201
7.0	4192	1201
7.1	4184	1201
7.2	4177	1201

Table 6.2.1.4-3 (Sheet 4 of 8)

MASS AND ENTHALPY RELEASE DATA FOR THE CASE OF MAIN STEAM LINE FULL DOUBLE ENDED RUPTURE FROM 102% POWER LEVEL WITH FAULTED LOOP MAIN STEAM LINE ISOLATION VALVE FAILURE THAT PRODUCES HIGHEST CONTAINMENT TEMPERATURE

Time (sec)	Mass (lbm/sec)	Enthalpy (Btu/lbm)
7.3	4170	1201
7.4	4162	1201
7.5	4154	1201
7.6	4147	1201
7.7	4140	1201
7.8	4133	1201
7.9	4126	1201
8.0	4119	1201
8.1	4112	1201
8.2	4105	1201
8.3	4099	1201
8.4	4092	1201
8.5	4085	1201
8.6	4078	1201
8.7	4072	1201
8.8	4067	1201
8.9	4061	1201
9.0	4055	1201
9.1	4050	1201
9.2	4044	1201
9.3	4038	1202
9.4	4033	1202
9.5	4026	1202
9.6	1975	1202
9.7	1972	1202
9.8	1970	1202

Table 6.2.1.4-3 (Sheet 5 of 8)

MASS AND ENTHALPY RELEASE DATA FOR THE CASE OF MAIN STEAM LINE FULL DOUBLE ENDED RUPTURE FROM 102% POWER LEVEL WITH FAULTED LOOP MAIN STEAM LINE ISOLATION VALVE FAILURE THAT PRODUCES HIGHEST CONTAINMENT TEMPERATURE

Time (sec)	Mass (lbm/sec)	Enthalpy (Btu/lbm)
9.9	1967	1202
10	1965	1202
11	1936	1202
12	1901	1203
13	1861	1203
14	1818	1203
15	1772	1203
16	1726	1204
17	1680	1204
18	1632	1204
19	1579	1204
20	1525	1204
21	1482	1204
22	1429	1204
23	1377	1204
24	1327	1204
25	1279	1204
26	1234	1204
27	1190	1204
28	1149	1204
29	1111	1204
30	1074	1204
31	1040	1204
32	1009	1204
33	979	1203
34	950	1203

Table 6.2.1.4-3 (Sheet 6 of 8)

MASS AND ENTHALPY RELEASE DATA FOR THE CASE OF MAIN STEAM LINE FULL DOUBLE ENDED RUPTURE FROM 102% POWER LEVEL WITH FAULTED LOOP MAIN STEAM LINE ISOLATION VALVE FAILURE THAT PRODUCES HIGHEST CONTAINMENT TEMPERATURE

Time (sec)	Mass (lbm/sec)	Enthalpy (Btu/lbm)
35	924	1203
36	899	1203
37	875	1203
38	852	1202
39	833	1202
40	815	1202
41	797	1202
42	781	1202
43	764	1201
44	749	1201
45	734	1201
46	720	1201
47	707	1201
48	694	1200
49	682	1200
50	670	1200
55	621	1199
60	584	1198
65	555	1197
70	534	1197
75	517	1196
80	504	1196
85	493	1196
90	483	1195
95	473	1195
100	464	1195

Table 6.2.1.4-3 (Sheet 7 of 8)

MASS AND ENTHALPY RELEASE DATA
FOR THE CASE OF MAIN STEAM LINE FULL DOUBLE
ENDED RUPTURE FROM 102% POWER LEVEL WITH FAULTED
LOOP MAIN STEAM LINE ISOLATION VALVE FAILURE THAT
PRODUCES HIGHEST CONTAINMENT TEMPERATURE

Time (sec)	Mass (lbm/sec)	Enthalpy (Btu/lbm)
105	455	1194
110	445	1194
115	435	1193
120	425	1193
125	416	1193
130	406	1192
135	397	1192
140	388	1191
145	380	1191
150	372	1191
155	365	1190
160	358	1190
165	352	1190
170	346	1189
175	341	1189
180	336	1189
185	331	1189
190	327	1188
195	323	1188
200	320	1188
205	316	1188
210	313	1188
215	310	1187
220	307	1187
225	304	1187
230	301	1187

Table 6.2.1.4-3 (Sheet 8 of 8)

MASS AND ENTHALPY RELEASE DATA FOR THE CASE OF MAIN STEAM LINE FULL DOUBLE ENDED RUPTURE FROM 102% POWER LEVEL WITH FAULTED LOOP MAIN STEAM LINE ISOLATION VALVE FAILURE THAT PRODUCES HIGHEST CONTAINMENT TEMPERATURE

Time (sec)	Mass (lbm/sec)	Enthalpy (Btu/lbm)
235	299	1187
240	296	1187
245	293	1186
250	291	1186
255	288	1186
260	286	1186
265	283	1186
270	281	1186
275	278	1185
280	276	1185
285	273	1185
290	271	1185
295	268	1185
300	266	1185
306	256	1184
312	244	1183
318	230	1182
324	214	1180
330	194	1178
336	168	1176
342	135	1171
348	95	1164
354	3	1157
360	0	1158

Table 6.2.1.4-4

PLANT DATA USED FOR MASS AND ENERGY RELEASES DETERMINATION

Plant data for all cases:

Power, Nominal Rating (MWt)	1940
Nominal RCS Flow (GPM)	194200
Nominal Full Load T_{avg} (°F)	565.9
Nominal RCS Pressure (psia)	2250
Nominal Steam Temperature (°F)	518.3
Nominal Feedwater Enthalpy (BTU/lbm)	413.8

Table 6.2.1.5-1 (Sheet 1 of 2)

MINIMUM CONTAINMENT PRESSURE MASS AND ENERGY RELEASES

Time (sec)	Mass Release (lbm/sec)	Energy Release (Btu/sec)
0.01125	5099.869	2670001
0.51	51055.6	26504774
1.01	51273.84	26680200
1.51	49860.99	26033554
2.01	46206.26	24306452
2.51	41412.38	22093776
3.01	35793.19	19400192
3.51	30418.92	16703766
4.01	26853.12	15011002
4.51	24604.13	13976796
5.01	22754.77	13127462
5.51	20785.91	12302017
6.01	19052.57	11643622
6.51	18187.08	11179082
7.01	17869	10899948
7.51	17087.62	10497992
8.01	15744.36	9907718
8.51	14096.94	9156074
9.01	13299.04	8703237
9.51	12671.68	8305148
10.01	11523.11	7748078
11.01	10783.73	7150337
12.01	9524.35	6427988
13.01	8396.749	5768359

Table 6.2.1.5-1 (Sheet 2 of 2)

MINIMUM CONTAINMENT PRESSURE MASS AND ENERGY RELEASES

Time (sec)	Mass Release (lbm/sec)	Energy Release (Btu/sec)
14.01	7543.894	5163321
15.01	6243.189	4418082
16.01	5291.53	3756248
17.01	4250.511	3093208
18.01	3642.385	2505180
19.01	2792.771	2093299
20.01	2480.567	1713555
21.011	2149.12	1480193
22.01	1839.571	1264136
23.01	1799.34	1082463
24.01	1651.465	973969.9
25.01	1452.628	820180.9
26.01	1458.903	741079.4
27.01	1062.866	589754.2
28.01	813.4185	475906
29.01	826.8542	395359.7
30.01	547.0674	303974.3
31.01	433.1182	234913.5
32.01	168.5994	132740.8
33.01	0.	0.

Table 6.2.2-1

PASSIVE CONTAINMENT COOLING SYSTEM PERFORMANCE PARAMETERS

PCCWST[1] initial inventory duration - Minimum	72 hours
PCCWST useable capacity for PCS (gal) - Minimum	531,000
PCCWST useable capacity for FPS[2] (gal) - Minimum	18,000
Injection flow rate (gpm) - Initial - Minimum	442
Injection flow rate (gpm) - Flow at 72 hours - Minimum	62.7
Injection flow rate (gpm) - Final at 7 days - Minimum	62.7
Injection duration (days) - Minimum	7
PCCWST minimum temperature (°F)	40
PCCWST maximum temperature (°F)	120
Upper annulus drain rate (per drain) - Minimum	450 gpm
PCCAWST[4] long-term makeup rate - Minimum	62.7 gpm
PCCAWST long-term makeup duration - Minimum	4 days

Containment Wetting Coverage

PCCWST Elevation (Note 3) (feet)	Minimum Flow (gpm)	Wetted Coverage (Note 3) (percentage of circumference)
23.70	442	90
20.65	123.5	51
13.05	72.5	30

Notes:
1. PCCWST = passive containment cooling water storage tank
2. FPS = fire protection system
3. PCCWST Elevation is measured as feet above the lowest tank standpipe entrance. Wetted coverage is measured as the linear percentage of the containment shell circumference wetted measured at the upper spring line.
4. PCCAWST = passive containment cooling ancillary water storage tank

Table 6.2.2-2

COMPONENT DATA
PASSIVE CONTAINMENT COOLING SYSTEM
(Nominal)

Passive Containment Cooling Water Storage Tank	
Volume (gal) - Minimum	531,000
Design temperature (°F)	125
Design pressure (psig)	Atmospheric
Material	Concrete with stainless steel liner
Standpipe Elevations Above Lowest Standpipe	
Top standpipe (ft) - Nominal	21.6
Second standpipe (ft) - Nominal	14.0
Third standpipe (ft) - Nominal	6.1
Bottom standpipe (ft)	0.0
Passive Containment Ancillary Cooling Water Storage Tank	
Volume (gal) - Nominal	425,000
Design temperature (°F)	125
Design pressure (psig)	Atmospheric
Material	Carbon steel
Water Distribution Bucket	
Volume (gal) - Nominal	42
Design temperature (°F)	150
Design pressure (psig)	Atmospheric
Material	Stainless steel
Water Distribution Collection Troughs and Weirs	
Design temperature (°F)	N/A
Design pressure (psig)	Atmospheric
Material	Stainless steel
Passive Containment Cooling Recirculation Pump	
Quantity	2
Type	Centrifugal
Design capacity (gpm)	100
Design total differential head (ft)	300

Table 6.2.2-3

FAILURE MODE AND EFFECTS ANALYSIS - PASSIVE CONTAINMENT COOLING SYSTEM ACTIVE COMPONENTS

Component	Failure Mode	PCS Operation Phase	Effect on System Operation	Failure Detection Method	Remarks
Air-operated butterfly valve PCS-PL-V001A (PCS-PL-V001B analogous)	Failure to open on demand	Passive containment cooling water delivery to containment	Failure blocks flow of containment cooling water through train A (B) of PCS which reduces system redundancy. No safety effect on system operation Minimum containment cooling requirements will be met by the flow of cooling water through operation of train B (train A).	Valve position indication (closed to open position change) in main control room and at the remote shutdown work station	Valve is normally closed during power operations. Valve opens on actuation by a Hi-2 containment pressure signal or loss of air or loss of 1E power.
Motor-operated gate valve PCS-PL-V002A (PCS-PL-V002B analogous)	Spurious valve closure	Passive containment cooling water delivery to containment	Spurious closure blocks flow of containment cooling water through train A (B) of PCS which reduces system redundancy. No safety effect on system operation. Minimum containment cooling requirements will be met by the flow of cooling water through operation of train B (train A).	Valve position indication (open to closed position change) in main control room and at the remote shutdown work station	Valve is normally open during power operations. Valve receives confirmatory open signal on Hi-2.
Air-operated butterfly valve PCS-PL-V001A (PCS-PL-V001B analogous)	Spurious valve opening	Normal idle condition	Failure initiates flow of containment cooling water through train A (B) of PCS when not required. No safety effect on system operation. Flow through train A (B) will be terminated through operator action by closing the series isolation valves via the main control room.	Valve position indication (closed to open) in main control room or at the remote shutdown work station. Also by PCS flow indication and decreasing PCCWST level.	Valve is normally closed during power operations to isolate PCS water.

Table 6.2.3-1 (Sheet 1 of 4)

Containment Mechanical Penetrations and Isolation Valves

	Containment Penetration			Isolation Device					Test		
System	Line	Flow	Closed Sys IRC	Valve/Hatch Identification	DCD Subsection	Position N-S-A	Signal	Closure Times	Type[1] & Note	Medium	Direction
CAS	Service air in	In	No	CAS-PL-V204 CAS-PL-V205	9.3.1	C-O-C C-O-C	None None	N/A N/A	C,5	Air	Forward
	Instrument air in	In	No	CAS-PL-V014 CAS-PLV015	9.3.1	O-O-C O-O-C	T None	std. N/A	C,5	Air	Forward
CCS	IRC loads in	In	No	CCS-PL-V200 CCS-PLV201	9.2.2	O-O-C O-O-C	S None	std. N/A	C,5	Air	Forward
	IRC loads out	Out	No	CCS-PLV208 CCS-PLV207	9.2.2	O-O-C O-O-C	S S	std. std.	C,5	Air	Forward
CVS	Spent resin flush out	Out	No	CVS-PL-V041 CVS-PL-V040 CVS-PL-V042	9.3.6	C-C-C C-C-C C-C-C	None None None	N/A N/A N/A	C	Air	Forward
	Letdown	Out	No	CVS-PL-V047 CVS-PL-V045	9.3.6	C-O-C C-O-C	T T	std. std.	C	Air	Forward
	Charging	In	No	CVS-PL-V090 CVS-PL-V091 CVS-PL-V100	9.3.6	C-O-C C-O-C C-C-C	HR,PL2, S+PL1, SGL HR,PL2, S+PL1, SGL None	std. std. N/A	C	Air	Forward
	H_2 injection to RCS	In	No	CVS-PL-V092 CVS-PL-V094	9.3.6	C-C-C C-C-C	T None	std. N/A	C	Air	Forward
DWS	Demin. water supply	In	No	DWS-PL-V244 DWS-PL-V245	9.2.4	C-O-C C-O-C	None None	N/A N/A	C,5	Air	Forward
FHS	Fuel transfer	N/A	No	FHS-FT-01	6.2.5	C-O-C	None	N/A	B	Air	Forward
FPS	Fire protection standpipe sys.	In	No	FPS-PL-V050 FPS-PL-V052	9.5.1	C-C-C C-C-C	None None	N/A N/A	C,5	Air	Forward
PSS	RCS/PSX/CVS samples out	Out	No	PSS-PL-V011 PSS-PL-V010A,B	9.3.3	C-C-C C-C-C	T T	std. std.	C	Air	Forward
	Cont. air samples out	Out	No	PSS-PL-V046 PSS-PL-V008	9.3.3	O-C-C O-C-C	T T	std. std.	C	Air	Forward
	RCS/Cont. air sample return	In	No	PSS-PL-V023 PSS-PL-V024	9.3.3	O-C-C O-C-C	T None	std. N/A	C	Air	Forward

Table 6.2.3-1 (Sheet 2 of 4)

Containment Mechanical Penetrations and Isolation Valves

	Containment Penetration			Isolation Device					Test		
System	Line	Flow	Closed Sys IRC	Valve/Hatch Identification	DCD Subsection	Position N-S-A	Signal	Closure Times	Type[1] & Note	Medium	Direction
PXS	N_2 to accumulators	In	No	PXS-PL-V042 PXS-PL-V043	6.3	O-O-C C-C-C	T None	std. N/A	C	Air	Forward
RNS	RCS to RHR pump	Out	No	RNS-PL-V002A/B RNS-PL-V023 RNS-PL-V022 RNS-PL-V021 RNS-PL-V061 PXS-PL-V208A	5.4.7 5.4.7 5.4.7 5.4.7 5.4.7 6.3	C-O-C C-O-C C-O-C C-C-C C-O-C C-C-C	HR, S HR, S HR, S None T None	std. std. std. N/A std. N/A	6 C C,4 C C C	Air	-- Reverse Forward Reverse Forward Forward
	RHR pump to RCS	In	No	RNS-PL-V011 RNS-PL-V013	5.4.7	C-O-C C-O-C	HR, S None	std. N/A	C,4 C,4	Air	Forward
SFS	IRWST/Ref. cav. SFP pump discharge	In	No	SFS-PL-V038 SFS-PL-V037	9.1.3	C-O-C C-O-C	T None	std. N/A	C,5	Air	Forward
	IRWST/Ref. cav. purif. out	Out	No	SFS-PL-V035 SFS-PL-V034	9.1.3	C-O-C C-O-C	T T	std. std.	C,5	Air	Forward
SGS	Main steamline 01	Out	Yes	SGS-PL-V040A SGS-PL-V027A[(8)] SGS-PL-V030A,31A,32A SGS-PL-V036A SGS-PL-V240A	10.3	O-C-C O-O-C C-C-C O-O-C C-C-C	MS LSL None MS MS	5 sec std. N/A std. std.	A,2	N_2	Forward
	Main steamline 02	Out	Yes	SGS-PL-V040B SGS-PL-V027B[(8)] SGS-PL-V030B,31B,32B SGS-PL-V036B SGS-PL-V240B	10.3	O-C-C O-O-C C-C-C O-O-C C-C-C	MS LSL None MS MS	5 sec std. N/A std. std.	A,2	N_2	Forward
	Main feedwater 01	In	Yes	SGS-PL-V057A	10.3	O-C-C	MF	5 sec	A,2	H_2O	Forward
	Main feedwater 02	In	Yes	SGS-PL-V057B	10.3	O-C-C	MF	5 sec	A,2	H_2O	Forward
	SG blowdown 01	Out	Yes	SGS-PL-V074A	10.3	O-O-C	PRHR	std.	A,2	H_2O	Forward
	SG blowdown 02	Out	Yes	SGS-PL-V074B	10.3	O-O-C	PRHR	std.	A,2	H_2O	Forward
	Startup feedwater 01	In	Yes	SGS-PL-V067A	10.3	C-O-C	LTC, SGL	std.	A,2	H_2O	Forward
	Startup feedwater 02	In	Yes	SGS-PL-V067B	10.3	C-O-C	LTC, SGL	std.	A,2	H_2O	Forward

Table 6.2.3-1 (Sheet 3 of 4)

Containment Mechanical Penetrations and Isolation Valves

System	Containment Penetration			Isolation Device					Test		
	Line	Flow	Closed Sys IRC	Valve/Hatch Identification	DCD Subsection	Position N-S-A	Signal	Closure Times	Type[1] & Note	Medium	Direction
VFS	Cont. air filter supply	In	No	VFS-PL-V003 VFS-PL-V004	9.4.7	C-O-C C-O-C	T, HR,DAS T, HR,DAS	20 sec 20 sec	C,5	Air	Forward Forward
	Cont. air filter exhaust	Out	No	VFS-PL-V010 VFS-PL-V009 VFS-PL-V008	9.4.7	C-O-C C-O-C C-C-C	T,HR,DAS T,HR,DAS N/A	20 sec 20 sec N/A	C,5	Air	Forward Forward Forward
VWS	Fan Coolers out	Out	No	VWS-PL-V086 VWS-PL-V082	9.2.7	O-O-C O-O-C	T T	std. std.	C,3,4,5	Air	Forward
	Fan coolers in	In	No	VWS-PL-V058 VWS-PL-V062	9.2.7	O-O-C O-O-C	T N/A	std. std.	C,3,4,5	Air	Forward
WLS	Reactor coolant drain tank gas	Out	No	WLS-PL-V068 WLS-PL-V067	11.2	C-C-C C-C-C	T T	std. std.	C	Air	Forward
	Normal cont. sump	Out	No	WLS-PL-V057 WLS-PL-V055	11.2	C-C-C C-C-C	T,DAS T,DAS	std. std.	C	Air	Forward
SPARE		N/A	No	P40	6.2.5	C-C-C	N/A	N/A	B	Air	Forward
SPARE		N/A	No	P41	6.2.5	C-C-C	N/A	N/A	B	Air	Forward
SPARE		N/A	No	P42	6.2.5	C-C-C	N/A	N/A	B	Air	Forward
CNS	Main equipment hatch	N/A	No	CNS-MY-Y01	6.2.5	C-C-C	None	N/A	B	Air	Forward
	Maintenance hatch	N/A	No	CNS-MY-Y02	6.2.5	C-C-C	None	N/A	B	Air	Forward
	Personnel hatch	N/A	No	CNS-MY-Y03	6.2.5	C-C-C	None	N/A	B	Air	Forward
	Personnel hatch	N/A	No	CNS-MY-Y04	6.2.5	C-C-C	None	N/A	B	Air	Forward

Table 6.2.3-1 (Sheet 4 of 4)

Containment Mechanical Penetrations and Isolation Valves

Explanation of Heading and Acronyms for Table 6.2.3-1

System:	Fluid system penetrating containment
Containment Penetration:	These fields refer to the penetration itself
Line:	Fluid system line
Flow:	Direction of flow in or out of containment
Closed Sys IRC:	Closed system inside containment as defined in DCD Section 6.2.3.1.1
Isolation Device:	These fields refer to the isolation devices for a given penetration
Valve/Hatch ID:	Identification number on P&ID or system figure
Subsection Containing Figure:	Safety analysis report containing the system P&ID or figure
Position N-S-A:	Device position for N (normal operation)
	S (shutdown)
	A (post-accident)
Signal:	Device closure signal
	MS: Main steamline isolation
	LSL: Low steamline pressure
	MF: Main feedwater isolation
	LTC: Low T_{cold}
	PRHR: Passive residual heat removal actuation
	T: Containment isolation
	S: Safety injection signal
	HR: High containment radiation
	DAS: Diverse actuation system signal
	PL2: High 2 pressurizer level signal
	S+PL1: Safety injection signal plus high 1 pressurizer level
	SGL: High steam generator level

Closure Time:
Required valve closure stroke time

std:	Industry standard for valve type (≤ 60 seconds)
N/A:	Not Applicable

Test:	These fields refer to the penetration testing requirements
Type:	Required test type
	A: Integrated Leak Rate Test
	B: Local Leak Rate Test -- penetration
	C: Local Leak Rate Test -- fluid systems
Note:	See notes below
Medium:	Test fluid on valve seat
Direction:	Pressurization direction
	Forward: High pressure on containment side
	Reverse: High pressure on outboard side

Notes:

1. Containment leak rate tests are designated Type A, B, or C according to 10CFR50, Appendix J.
2. The secondary side of the steam generator, including main steam, feedwater, startup feedwater, blowdown and sampling piping from the steam generators to the containment penetration, is considered an extension of the containment. These systems are not part of the reactor coolant pressure boundary and do not open directly to the containment atmosphere during post-accident conditions. During Type A tests, the secondary side of the steam generators is vented to the atmosphere outside containment to ensure that full test differential pressure is applied to this boundary.
3. The central chilled water system remains water-filled and operational during the Type A test in order to maintain stable containment atmospheric conditions.
4. The containment isolation valves for this penetration are open during the Type A test to facilitate testing. Their leak rates are measured separately.
5. The inboard valve flange is tested in the reverse direction.
6. These valves are not subject to a Type C test. Upstream side of RNS hot leg suction isolation valves is not vented during local leak rate test to retain double isolation of RCS at elevated pressure. Valve is flooded during post accident operation.
7. The inboard globe valve is tested in the reverse direction. The test is conservative since the test pressure tends to unseat the valve disc, whereas containment pressure would tend to seat the disc.
8. Refer to DCD Table 15.0-4b for PORV block valve closure time.

Table 6.2.4-1

COMPONENT DATA - HYDROGEN SENSORS (NOMINAL)

Number	16
Range (% hydrogen)	0 - 20
Response time	90% in 10 seconds

Table 6.2.4-2

COMPONENT DATA - HYDROGEN RECOMBINER (NOMINAL)

Number	
Full Size PAR	2
Partial size PAR	2
Inlet hydrogen concentration range for design basis events (volume percent)	0 - 4
Average efficiency (percent)	85
Depletion rate	Reference 19
Minimum acceptable depletion rate*	
Full Size PAR	1 scfm
Partial Size PAR	0.25 scfm

* Determined at prevailing conditions of 120°F, 3.5 volume percent of hydrogen and atmospheric pressure.

Table 6.2.4-3

COMPONENT DATA - HYDROGEN IGNITER (NOMINAL)

Number	64
Surface Temperature (°F)	1600 to 1700

Table 6.2.4-4 (Sheet 1 of 3)

ASSUMPTIONS USED TO CALCULATE HYDROGEN PRODUCTION FOLLOWING A LOSS OF COOLANT ACCIDENT

General

Core thermal power (MWt)	1,972
Containment free volume (ft^3)	1.73×10^6

Zirconium-Water Reaction

Weight of zirconium fuel cladding (lb)	34,788
Percent zirconium-water reaction (%)	1.09

Radiolysis of Water in Reactor Vessel

Percentage of core fission product inventory in core	
Noble gases	0
Iodines	50
Remainder	99
Energy absorption by core cooling solution	
Percent of gamma energy absorbed	10
Percent of beta energy absorbed	0
Molecules of hydrogen produced per 100 eV energy absorbed by solution	0.5

Radiolysis of Water in Sump

Percentage of core fission product inventory in the sump solution	
Noble gases	0
Iodines	50
Remainder	1
Energy absorption by core cooling solution	
Percent of gamma energy absorbed	100
Percent of beta energy absorbed	100
Molecules of hydrogen produced per 100 eV energy absorbed by solution	0.5

Table 6.2.4-4 (Sheet 2 of 3)

ASSUMPTIONS USED TO CALCULATE HYDROGEN PRODUCTION FOLLOWING A LOSS OF COOLANT ACCIDENT

Corrosion of Materials

Aluminum inventory in containment

Component	Weight (lb)	Surface (sq. ft)
Excore detectors	25	8
Miscellaneous valve parts	230	86
CRDM connectors	190	42
Paint	140	18,000
Contingency	370	171
Other non-NSSS items	500	100
Total aluminum	1,455	

Table 6.2.4-4 (Sheet 3 of 3)

ASSUMPTIONS USED TO CALCULATE HYDROGEN PRODUCTION FOLLOWING A LOSS OF COOLANT ACCIDENT

Zinc inventory in containment

Component	Weight (lb)	Surface (sq. ft)
Cable trays	310	2,100
Conduit	500	3,500
Hangers	24	170
Junction boxes	100	730
Paint	1,200	72,000
Gratings	680	41,000
HVAC ductwork	840	5,900
Stairs	13	800
Pipe supports	510	30,000
Contingency	1,050	39,000
Total zinc	5,227	

Aluminum corrosion rate	See Table 6.2.4-5
Zinc corrosion rate	See Table 6.2.4-5
Containment temperature	See Table 6.2.4-5
Solution pH	7 - 9.5

Initial Reactor Coolant Hydrogen Inventory

Hydrogen concentration in reactor coolant (cc at STP per kg)	40
Reactor coolant mass (lb)	353,000

Table 6.2.4-5

POST-ACCIDENT CONTAINMENT TEMPERATURE AND ASSOCIATED CORROSION RATES FOR ALUMINUM AND ZINC

Interval (sec)	Temperature (°F)	Al Corrosion (lb/ft^2-hr)	Zn Corrosion (lb/ft^2-hr)
0 - 2,500	267	0.024	0.00030
2,500-5,000	260	0.019	0.00027
5,000-10,000	245	0.012	0.00021
10,000-20,000	230	0.0070	0.00016
20,000-60,000	228	0.0065	0.00015
60,000-80,000	220	0.0049	0.00013
80,000-108,000	218	0.0045	0.00013
108,000-173,000	225	0.0058	0.00034
>173,000	220	0.0049	0.00013

Table 6.2.4-6 (Sheet 1 of 3)

IGNITER LOCATION

Criteria

- A sufficient number of igniters are placed in the major transport paths (including dominant natural circulation pathways) of hydrogen so that hydrogen can be burned continuously close to the release point. This prevents hydrogen from preferentially accumulating in a certain region of the containment.

- Igniters (minimum of 2) are located in major regions or compartments where hydrogen may be released, through which it may flow, or where it may accumulate.

- It is preferable to ignite a hydrogen-air mixture at the bottom so that upward flame propagation can be promoted at lean hydrogen concentrations. Igniters within each subcompartment are located in the vicinity of, and above, the highest potential release location within the subcompartment.

- In compartments with relatively small openings in the ceiling, the potential may exist for the hydrogen-air mixture to rise and to collect near the ceiling. Therefore, one or more igniters are placed near the ceiling of such compartments. Igniter coverage is provided within the upper 10 percent of the vertical height subcompartments or 10 feet from the ceiling whichever is less. In cases where the highest potential release point is low in the compartment, both this and the previous criteria are considered.

- To the extent possible, igniters are placed away from walls and other large surfaces so that a flame front created by ignition at the bottom of a compartment can travel unimpeded up to the top.

- A sufficient number of igniters are installed in long, narrow compartments (corridors) so that the flame fronts created by the igniters need to travel only a limited distance before they merge. This limits the potential for significant flame acceleration.

- Igniter coverge are provided to contol combustion in areas where oxygen rich air may enter into an inerted region with combustible hydrogen levels during an accident scenario.

- Igniters are located above the flood level, if possible. Those which may be flooded have redundant fuses to protect the power supply.

- In locations where the potential hydrogen release location can be defined, i.e. above the IRWST spargers, at IRWST vents, etc igniter coverage is provided as close to the source as feasible.

- Provisions for installation, maintenance, and testing is be considered.

Table 6.2.4-6 (Sheet 2 of 3)

IGNITER LOCATION

Implementation

- **Reactor cavity** – Hydrogen releases within the reactor cavity will flow either through the vertical access tunnel, through the opening around the RCS hot and cold legs into the loop compartments or if the refueling cavity seal ring fails then potentially through the refueling cavity. The potential flow paths have at least four igniters with at least two powered by each of two power groups. No igniters have been located within the reactor cavity since this region would always be flooded, adequate igniter coverage is available in hydrogen pathways from the reactor cavity and any maintenance or inspection would result in elevated personnel exposure.

- **Loop Compartments** – Hydrogen releases from the hot or cold legs or from the reactor cavity would flow up through the loop compartment to the dome region. Igniter coverage provided within the loop compartment consists of a total of four igniters at two different elevations covering the perimeter of the compartment and with two igniters powered by one power group and two by the second power group. Additional coverage is provided above the loop compartments at elevation 162' with four igniters above each loop compartment and powered by different power groups.

- **Pressurizer Compartment** – Hydrogen releases within the pressurizer compartment would flow up through the compartment toward the dome region. Igniter coverage is provided within the compartment consists of a total of four igniters at two different elevations covering the perimeter of the compartment with two igniters powered by one power group and two by the second power group. Additional coverage is provided above the pressurizer compartment at elevation 162' with two igniters above powered by different power groups.

- **Tunnel Connection Loop Compartments** – The tunnel between the loop compartments and extending downward into the reactor coolant drain tank cavity is provided with four igniters for hydrogen control. Releases within the reactor cavity or from the loop compartment may flow through this vertical access tunnel. Igniter coverage is provided over the width of the tunnel at three separate elevations and are powered by different power groups.

- **Refueling Cavity** – Hydrogen releases from the reactor cavity or from the potentially from the reactor coolant loops may flow up past the refueling cavity seal ring and through the refueling cavity to the dome region. Igniter coverage provided within the refueling consists of a total of four igniters at two different elevations covering the perimeter of the compartment with two igniters powered by one power group and two by the second power group. Additional coverage is provided above the refueling cavity at elevation 162' with four igniters powered by different power groups.

- **Southeast Valve and Accumulator Rooms** – Hydrogen releases within the southeast valve or accumulator rooms will rise with the mass and energy releases to near the ceiling and exit either through the stairwell on the west wall or through piping penetration holes in the ceiling. The hydrogen control protection is provided by two igniters, one located near the ceiling of each of the adjoining rooms. The igniters are powered by different power groups and provide backup control for each other.

Table 6.2.4-6 (Sheet 3 of 3)

IGNITER LOCATION

- **East Valve, Northeast Accumulator, and Northeast Valve Room** – Hydrogen releases within the east valve, northeast accumulator or valve rooms will rise with the mass and energy releases to near the ceiling and exit either through the enlarged vent area surrounding the discharge piping from the core makeup tank located at the 107" 2"" elevation and through other piping penetration holes in the ceiling. The hydrogen control protection is provided by three igniters, one located near the ceiling of each of the adjoining rooms. The igniters are powered by different power groups and provide backup control for each other.

- **North CVS Equipment Room** – Hydrogen releases within the CVS equipment room will rise from the piping or equipment located on the CVS module to near the ceiling, pass over the outer barrier wall and flow up through the stairwell or ceiling grating. Hydrogen control is provided by two igniters located near the ceiling of the equipment room between the equipment module and the major relief paths from the compartment. The igniters are powered by different power groups.

- **IRWST** – Hydrogen releases into the IRWST are controlled by the distribution of igniters internal to the IRWST and within the vents from and into the IRWST. Two igniters on different power groups are located within the IRWST below the tank roof of the IRWST and above the spargers. In the event of hydrogen releases via the spargers, the igniters directly above the release points will provide the most immediate point of recombination. Should the environment within the IRWST be inerted or otherwise not be ignited by the assemblies above the sparger, the hydrogen will be ignited as it exhausts from the IRWST at any of four of the vents fitted with igniter assemblies. Two of the four igniters are powered by one power group and two by the second power group. Finally, in the event that the IRWST is hydrogen rich and air is drawn into the IRWST the mixture will become flammable. In order to provide this recombination, the two inlet vents on the other side of the IRWST from the sparger and primary exhaust vents are fitted an igniter each.

- **Lower Compartment Area** – Hydrogen releases within the lower compartment will rise with the mass and energy releases to near the ceiling and exit either through the north stairwell or along the circumferential gap between the operating deck and the containment shell. The hydrogen control protection is provided by eleven igniters spread over the potential release areas and located either just above the mezzanine deck elevation or near the ceiling. This approach provides wide coverage over the entire compartment area at two separate elevations. The igniters are split between the two separate power groups.

- **Upper Compartment** – Hydrogen control is provided at three separate levels within the upper compartment. At the 162 foot elevation, 10 igniters are distributed over the area primarily above the major release flow paths including the loop compartments, refueling cavity, pressurizer compartment and above the stairwell from the lower compartment area. The igniters are split between the two power groups. At 210 foot elevation, an igniter is provided in each quadrant at the mid region of the upper compartment with two igniter on each of the two power groups. At the upper region elevation 235 four additional igniters are located to initiate recombination of hydrogen not ignited at either the source or along its flow path. The four igniters are split between the two power groups.

Table 6.2.4-7

SUBCOMPARTMENT/AREA IGNITER COVERAGE

Subcompartment	Igniter Coverage (Elevation)[1]	
	Power Group 1	Power Group 2
Reactor Cavity	1(El 91') 3 (El 95') 13, 5, 55 (El 120') 58 (El 132') 8, 12 (El 139')	4 (El 95') 2 (El 99') 11, 7, 56 (El 120') 57 (El 132') 6, 14 (El 139')
Loop Compartment 01	13 (El 120') 12 (El 139')	11 (El 120') 14 (El 139')
Loop Compartment 02	5 (El 120') 8 (El 139')	7 (El 120') 6 (El 139')
Pressurizer Compartment	49 (El 154') 60 (El 135')	50 (El 154') 59 (El 135')
Tunnel connecting Loop Compartments	1 (El 91') 3 (El 95') 31 (El 120')	4 (El 95') 2 (El 99') 30 (El 120')
Southeast Valve Room	21 (El 105')	20 (El 105')
Southeast Accumulator Room	21 (El 105')	20 (El 105')
East Valve Room	18 (El 105')	19 (El 105')
Northeast Accumulator Room	18 (El 105')	17, 19 (El 105')
Northeast Valve Room	18 (El 105')	17 (El 105')
North CVS Equipment Room	34 (El 105')	33 (El 105')
Lower Compartment Area (CMT and Valve area)	22 (El 133') 27, 28, 29, 31, 32 (El 120')	23, 24, 25 (El 133') 26, 30 (El 120')
IRWST Compartment	35, 37 (El 135')	36, 38 (El 135')
IRWST Interior	9 (El 133')	10 (El 133')
IRWST Inlet	16 (El 133')	15 (El 133')
Refueling Cavity	55 (El 120') 58 (El 132')	56 (El 120') 57 (El 132')
Upper Compartment		
Lower Region	39, 42, 44, 43, 47 (El 162')	40, 41, 45, 46, 48 (El 162')
Mid Region	51, 54 (El 210')	52, 53 (El 210')
Upper Region	61, 63 (El 235')	62, 64 (El 235')

<u>Note:</u>
1. Elevations are approximate.

Table 6.2.5-1 DELETED

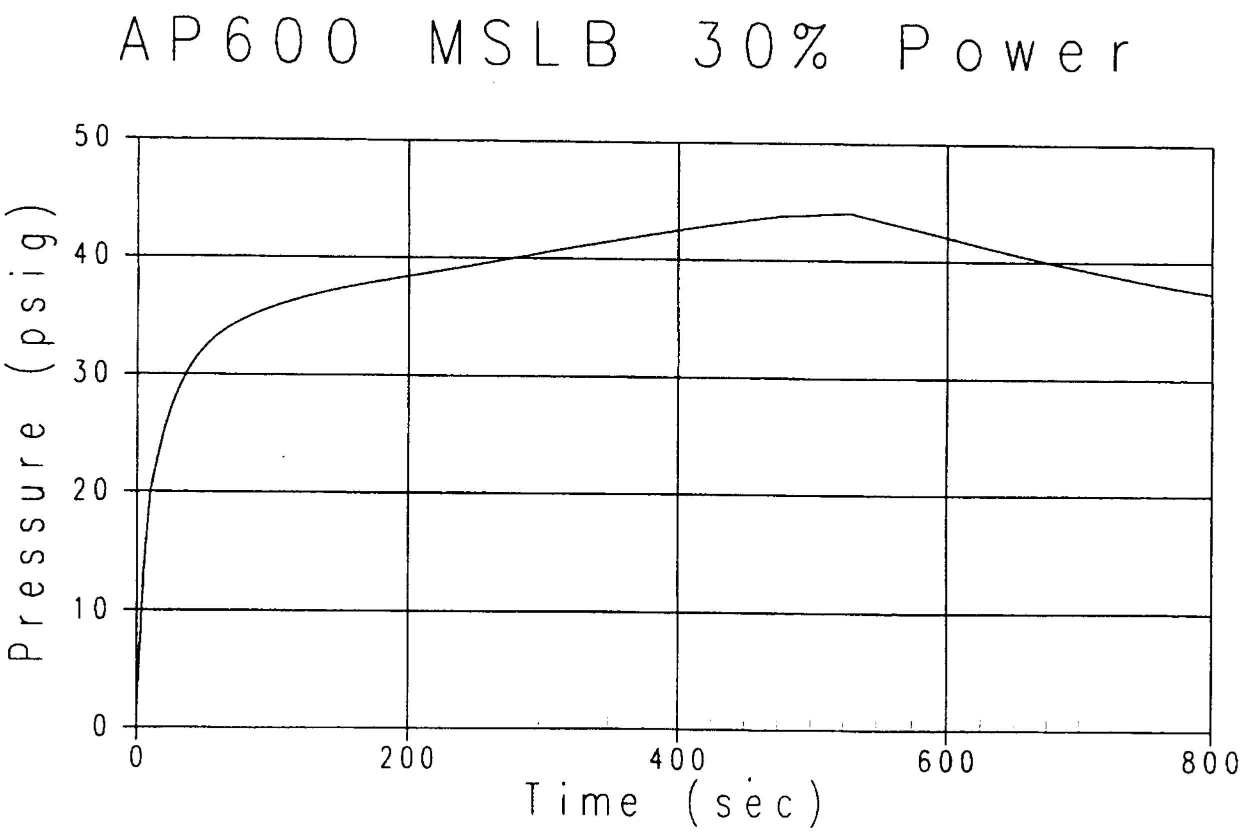


Figure 6.2.1.1-1

MSLB Full DER 30% Power Case

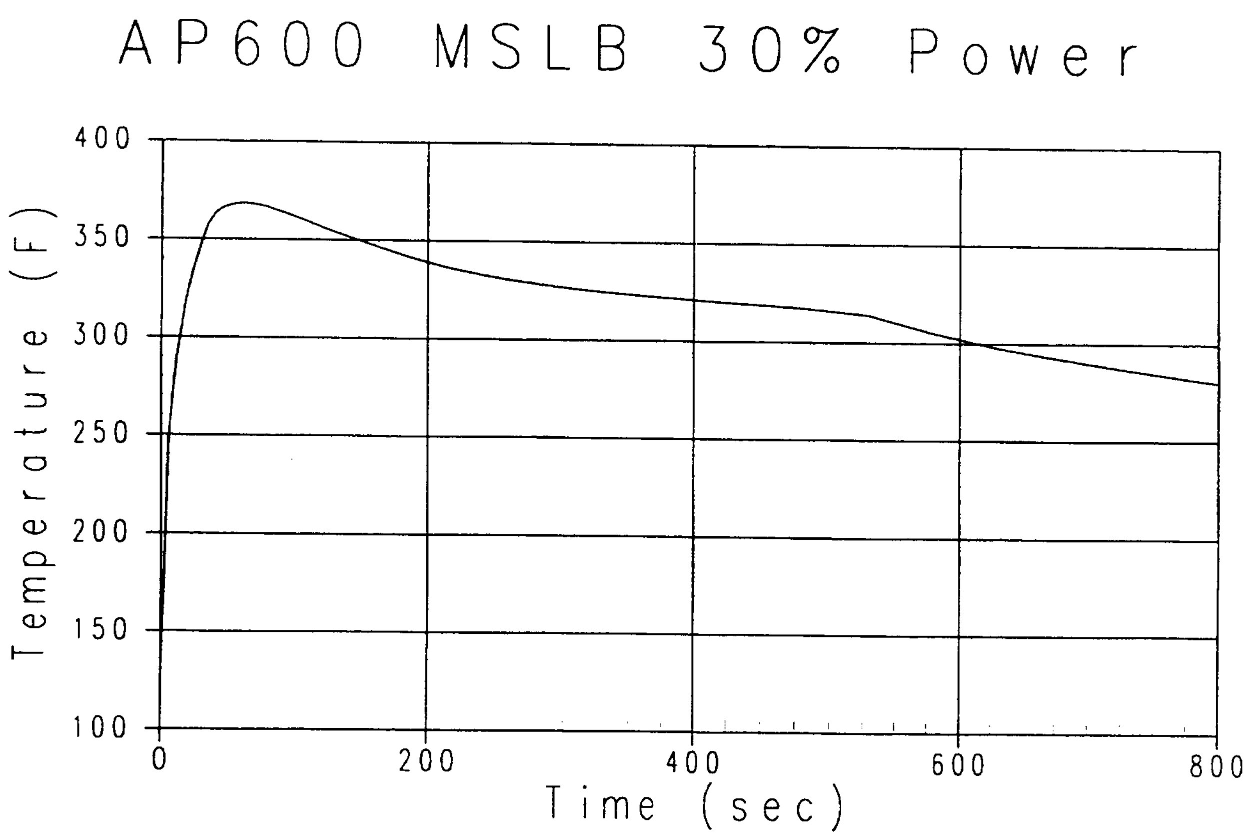


Figure 6.2.1.1-2

MSLB Full DER 30% Power Case

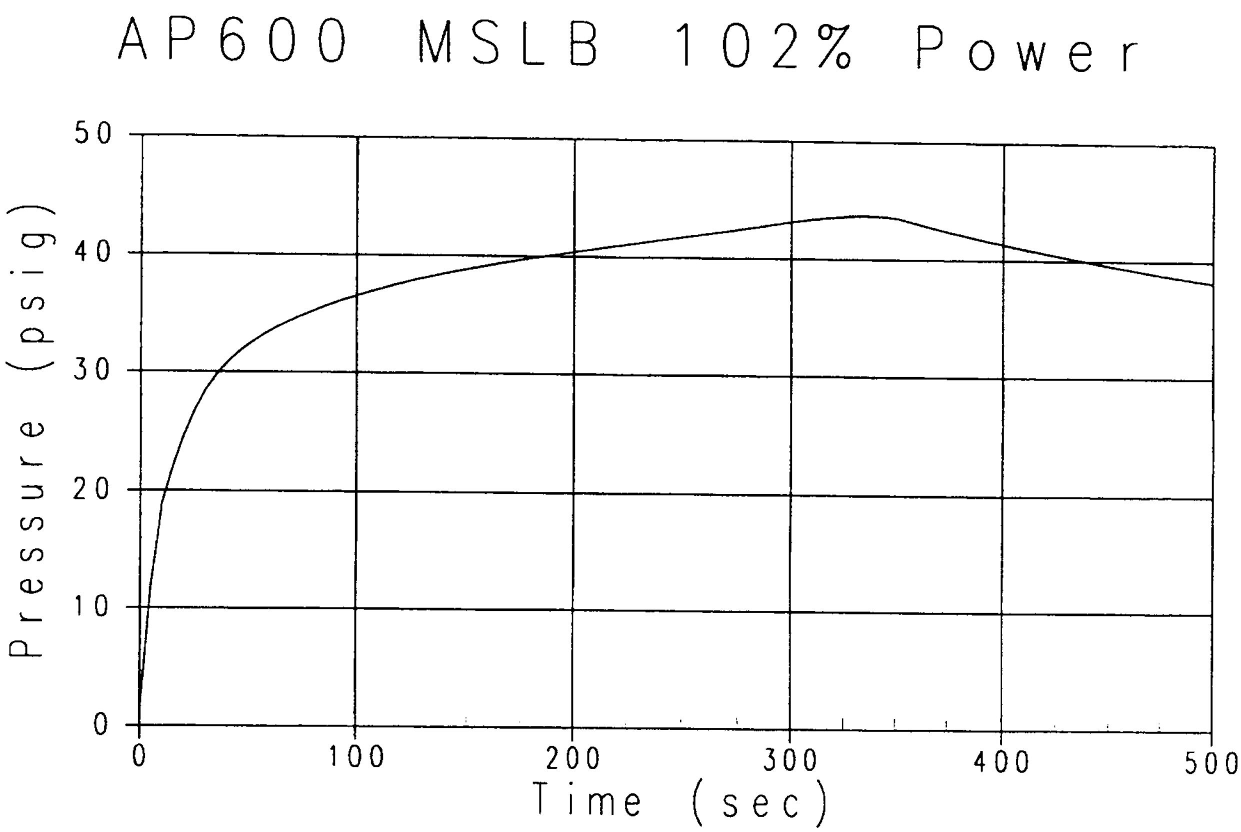


Figure 6.2.1.1-3

MSLB Full DER 102% Power Case

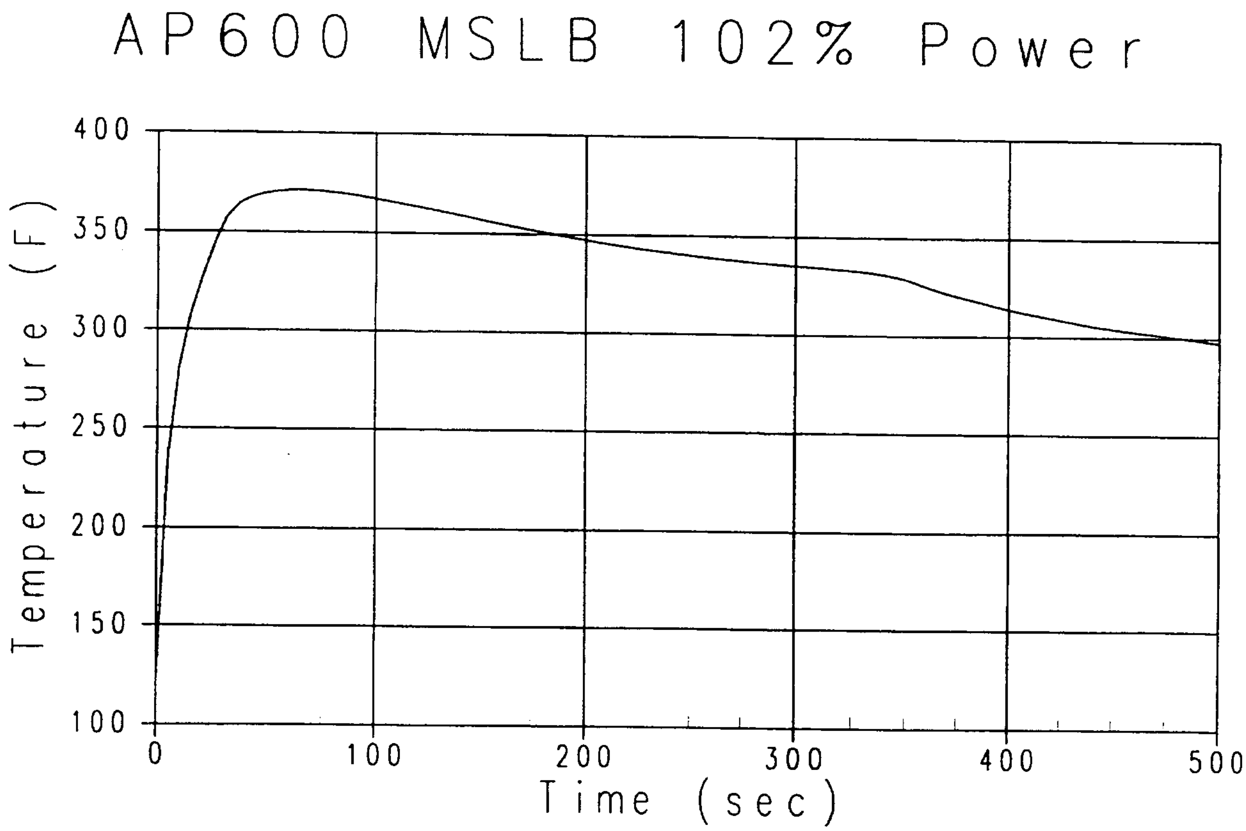


Figure 6.2.1.1-4

MSLB Full DER 102% Power Case

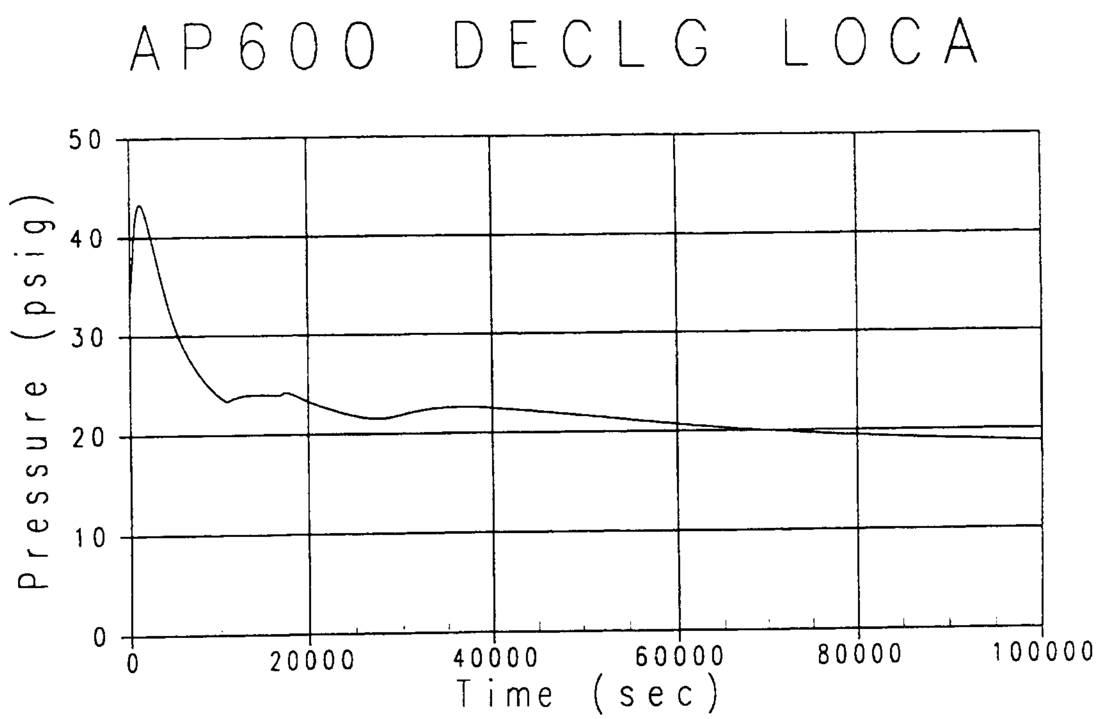


Figure 6.2.1.1-5

DECLG LOCA Case for 24 Hours

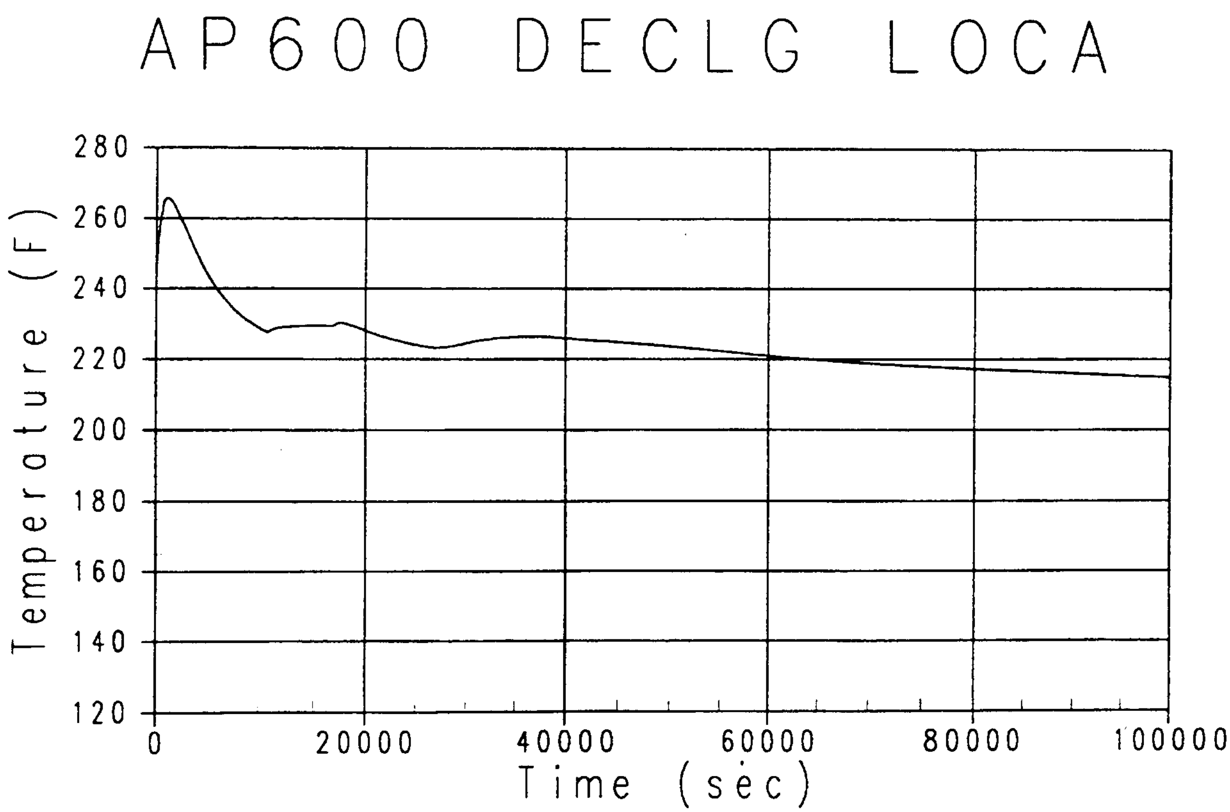


Figure 6.2.1.1-6

DECLG LOCA Case for 24 Hours

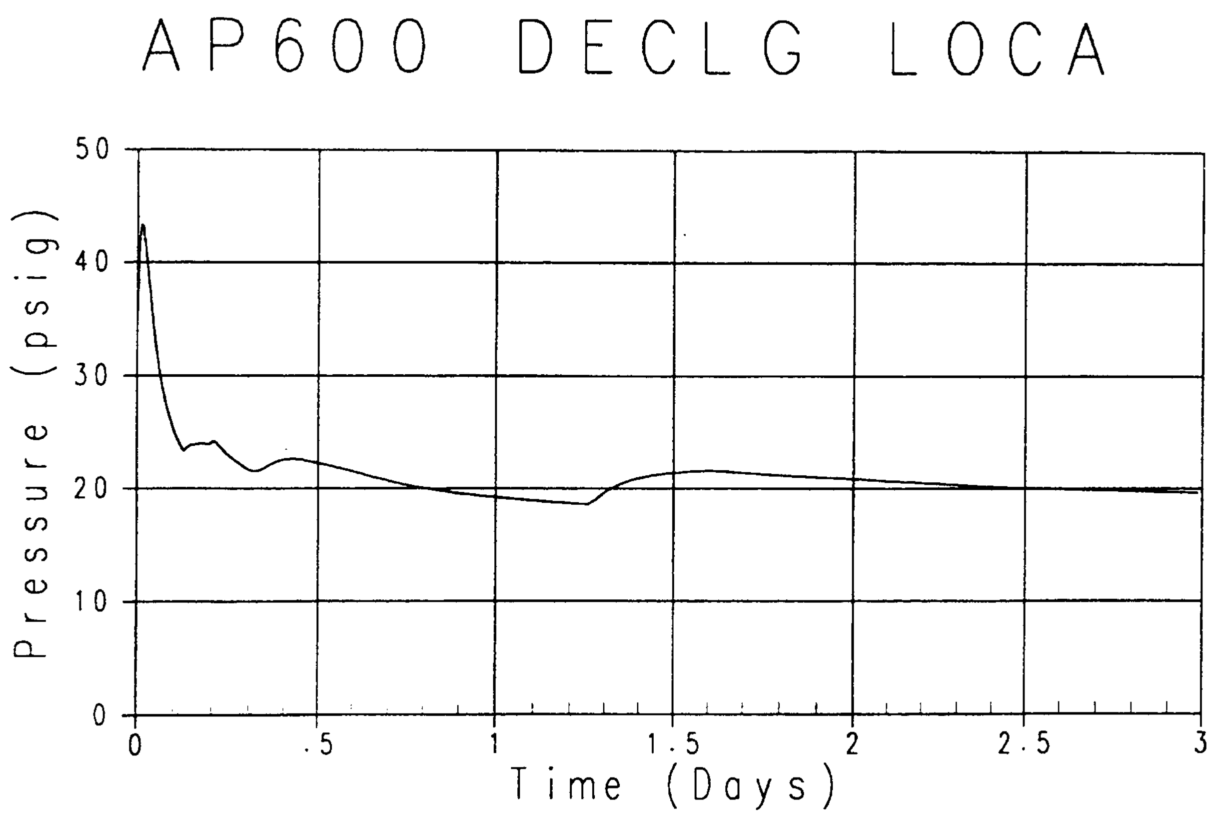


Figure 6.2.1.1-7

DECLG LOCA Case for 3 Days

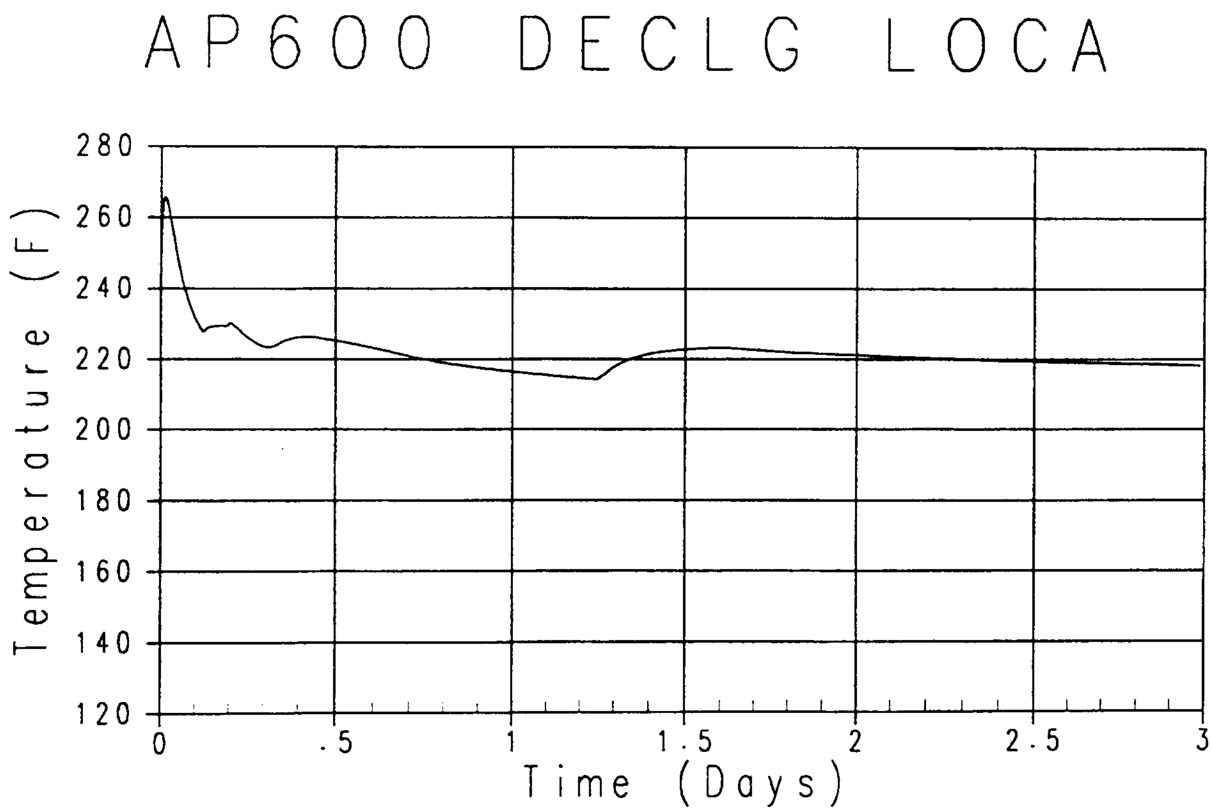


Figure 6.2.1.1-8

DECLG LOCA Case for 3 Days

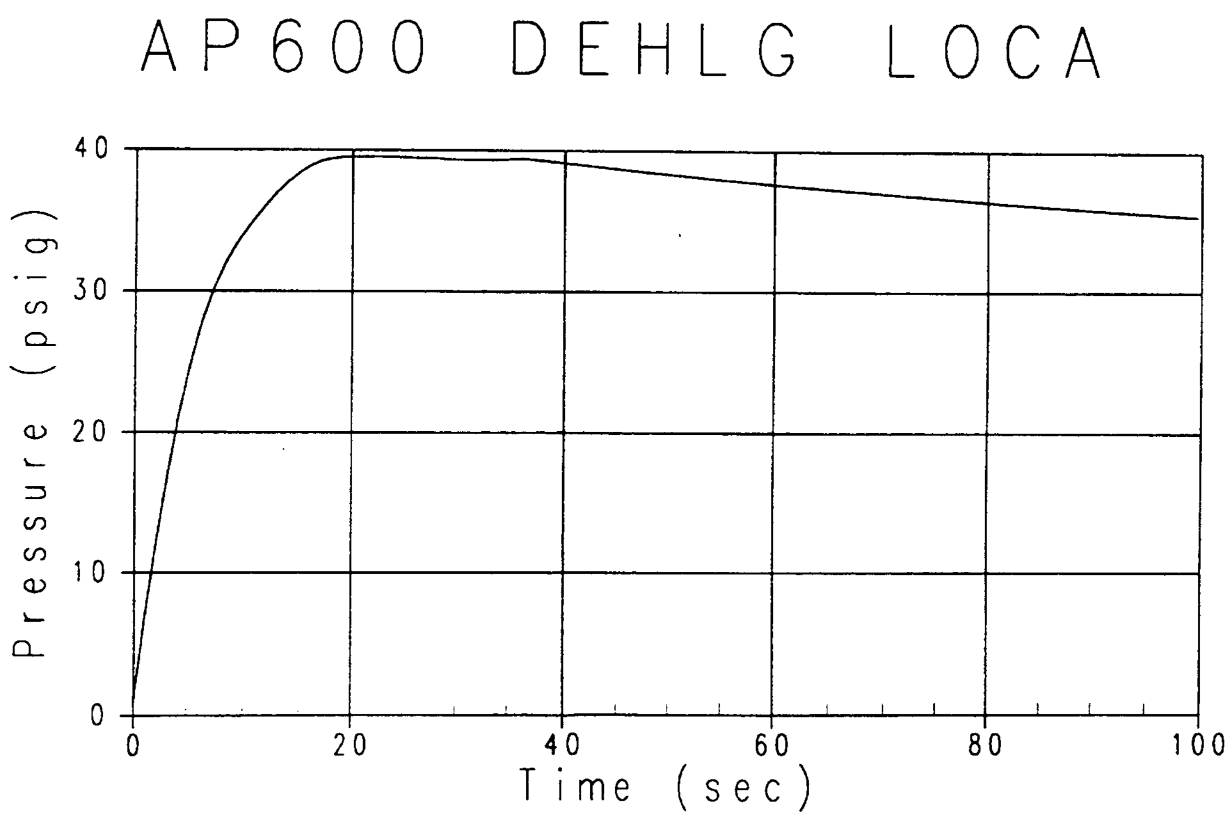


Figure 6.2.1.1-9

DEHLG LOCA Case

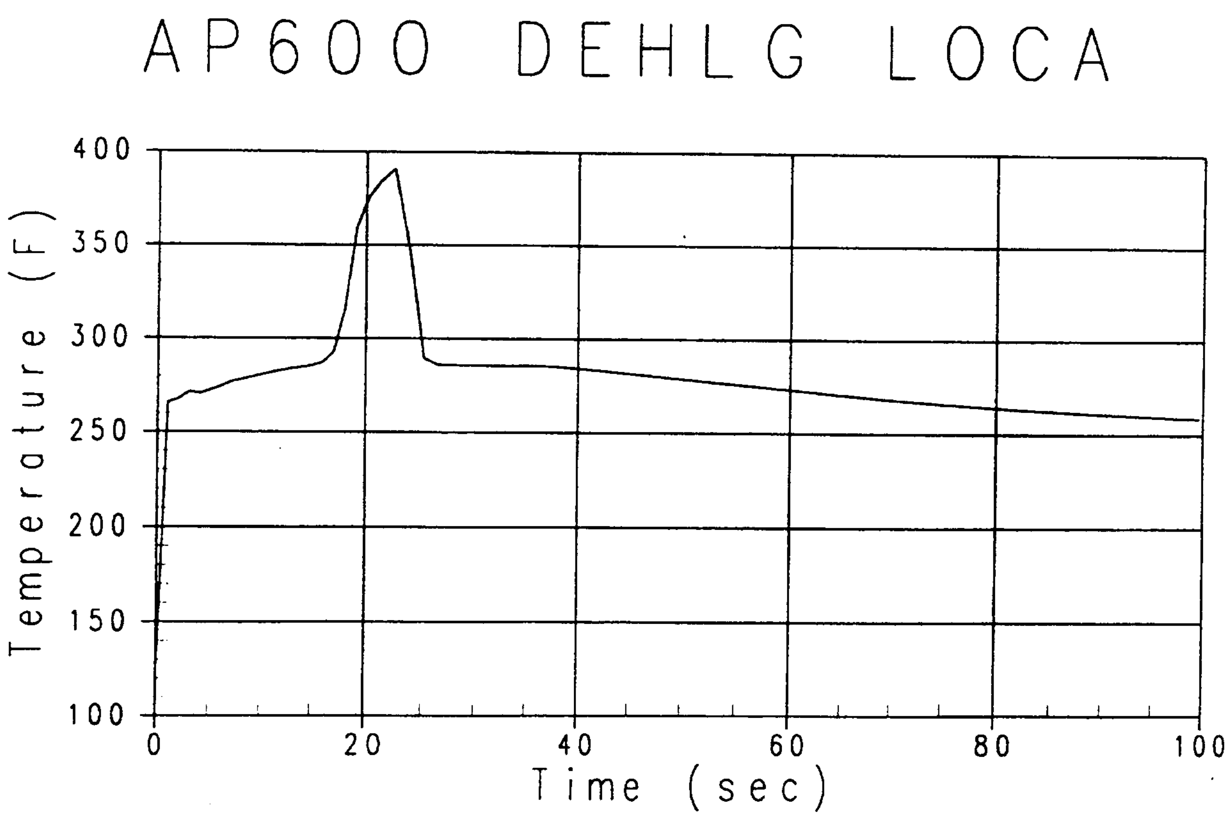


Figure 6.2.1.1-10

DEHLG LOCA Case

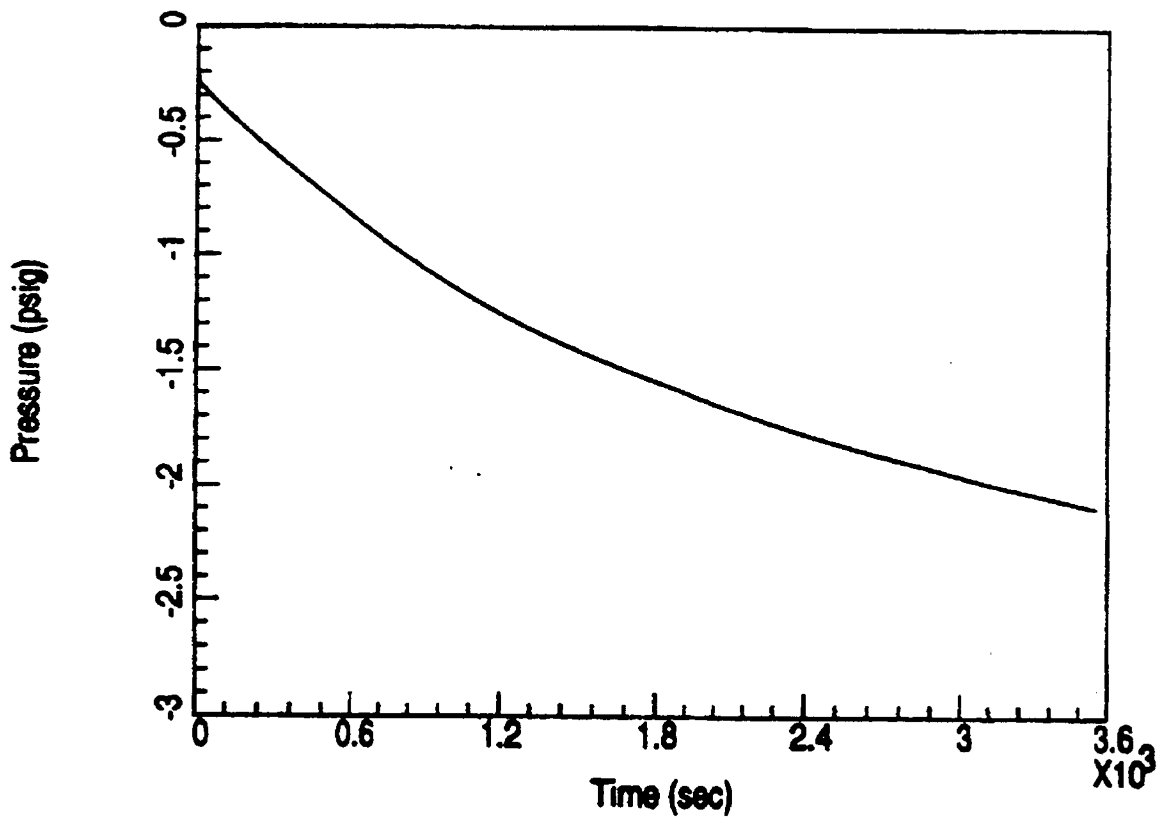


Figure 6.2.1.1-11

External Pressure Analysis Containment Pressure vs. Time

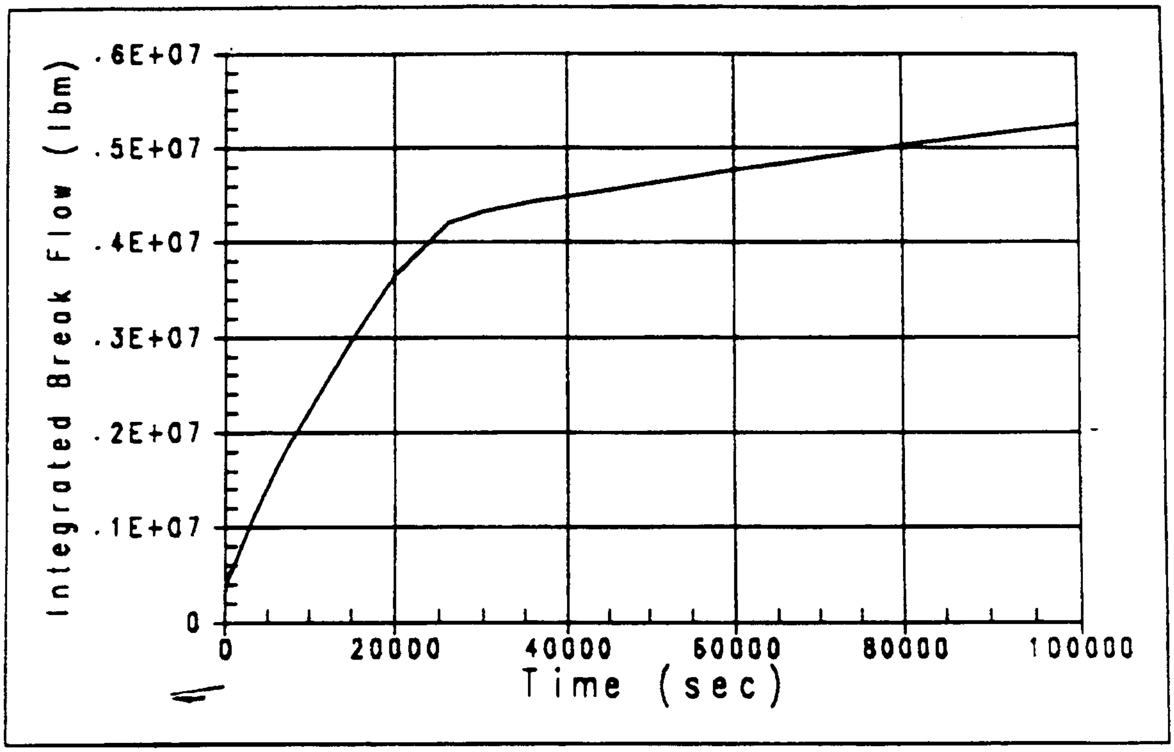


Figure 6.2.1.3-1

Integrated Mass Released for DECLG Break

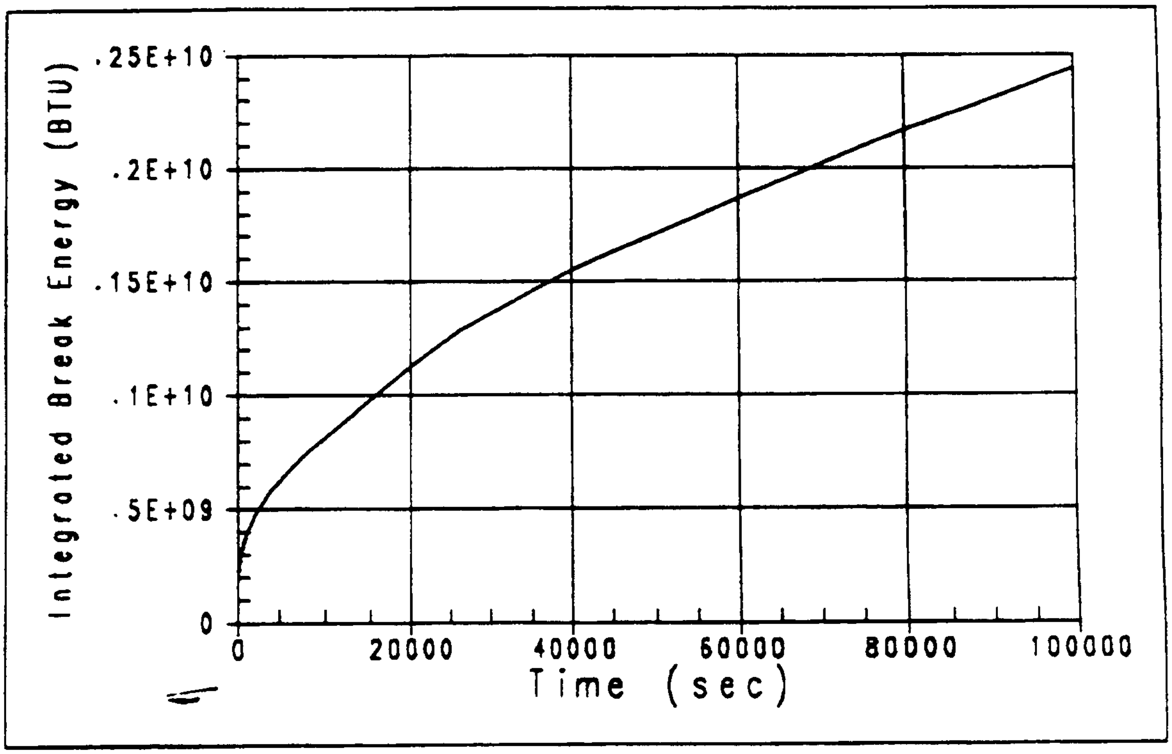


Figure 6.2.1.3-2

Integrated Energy Released for DECLG Break

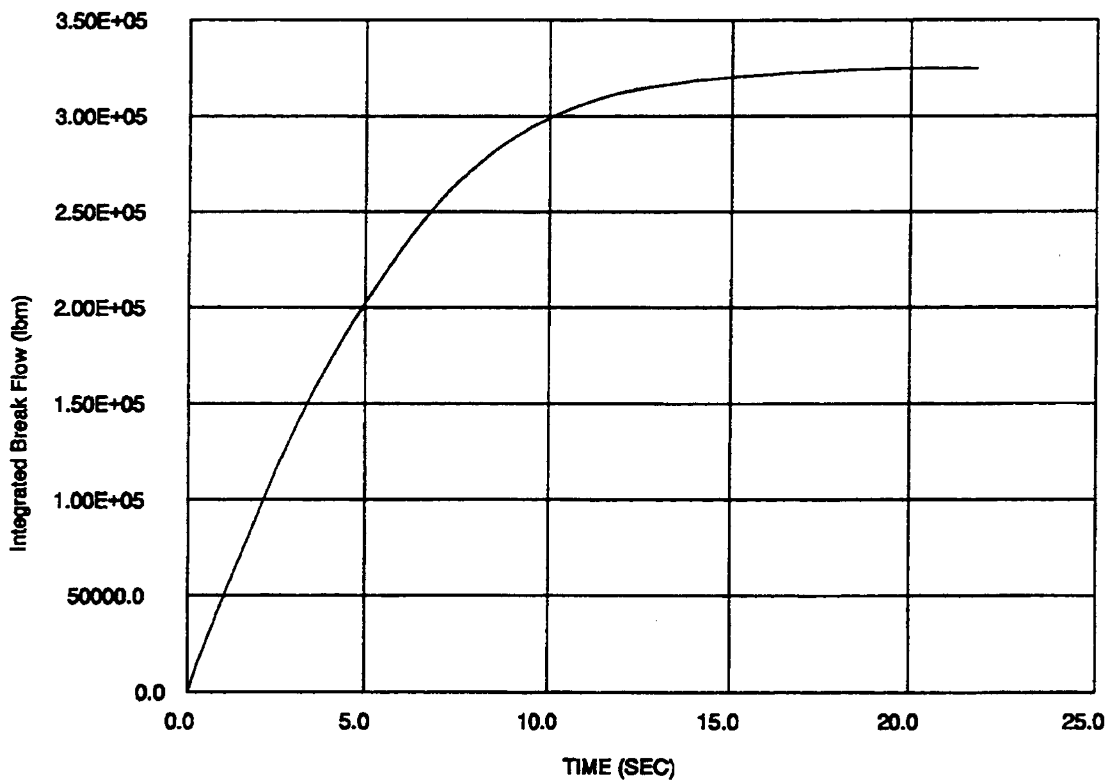


Figure 6.2.1.3-3

Integrated Mass Released for DEHLG Break

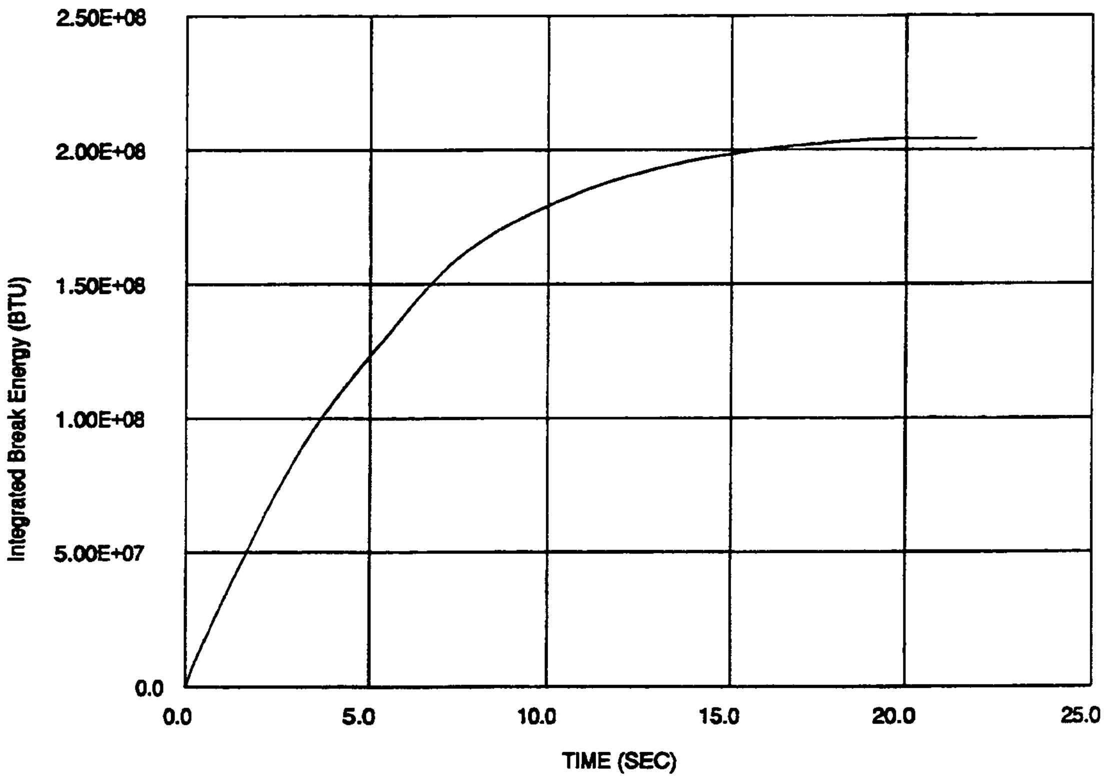


Figure 6.2.1.3-4

Integrated Energy Released for DEHLG Break

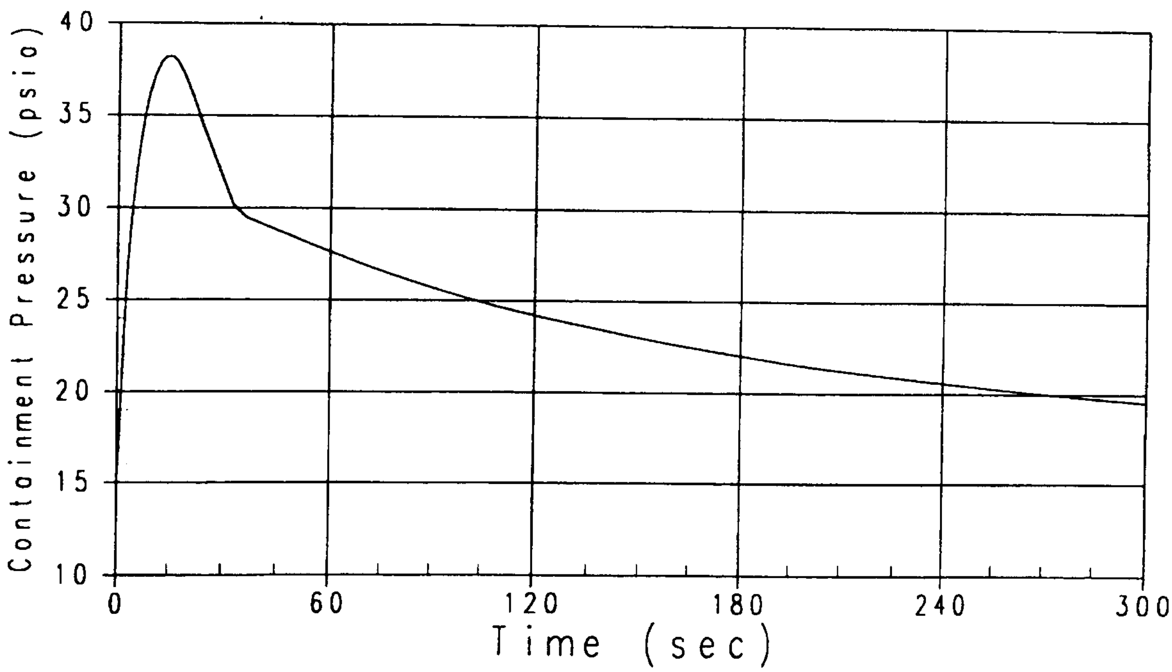
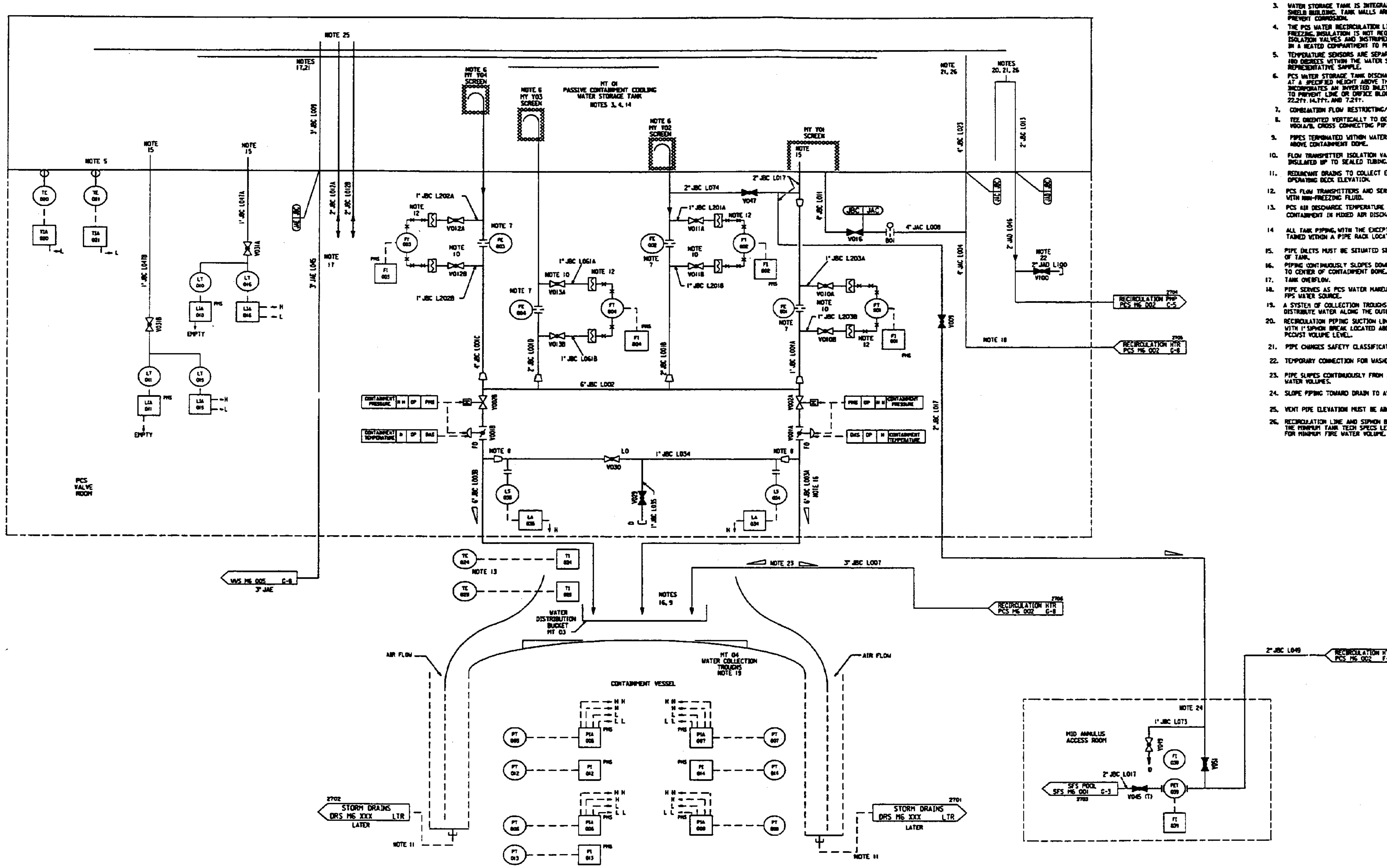


Figure 6.2.1.5-1

WGOTHIC-Predicted AP600 Containment Response to DECL Breaks



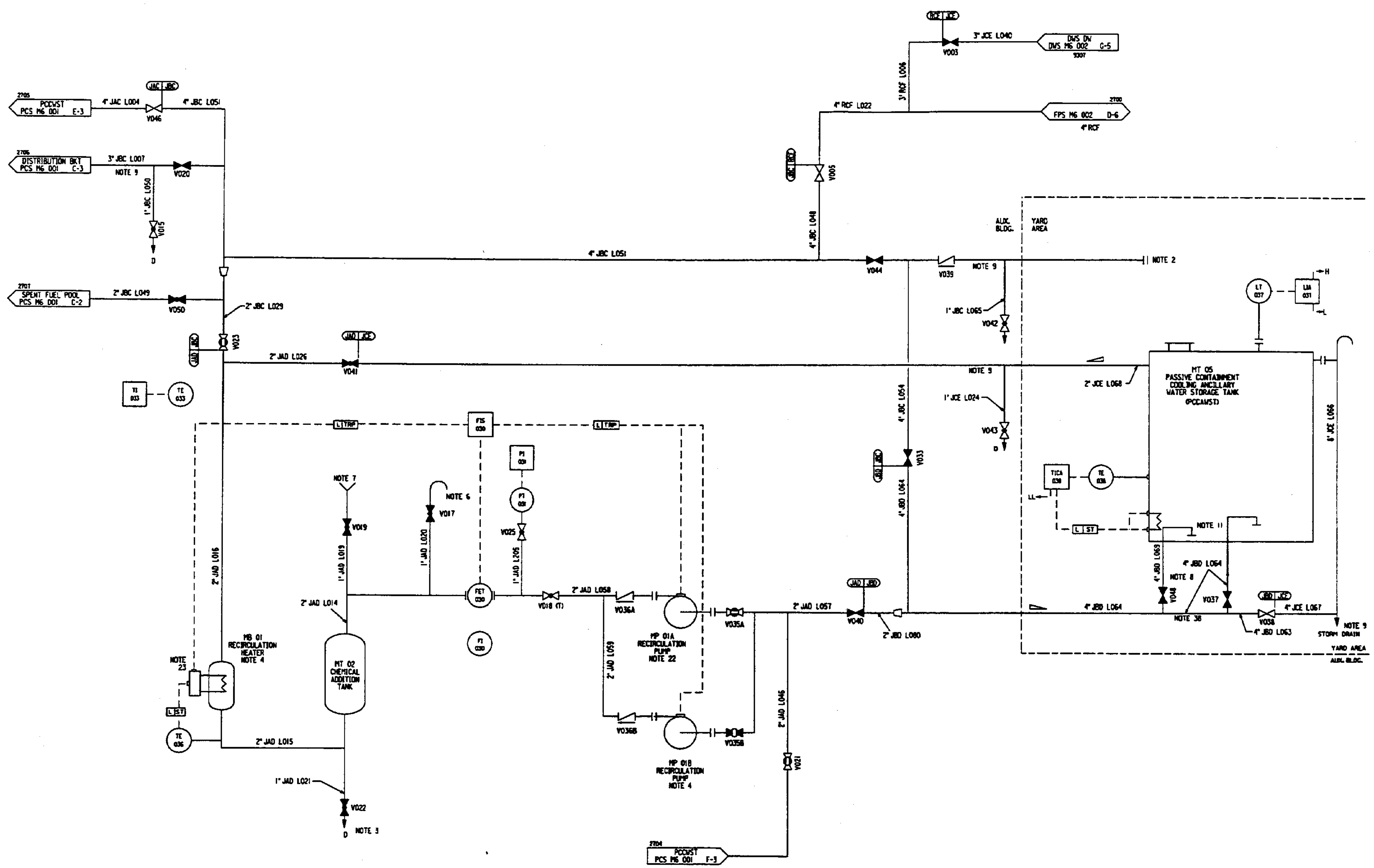
NOTES:

1. THE SYSTEM LOCATOR CODE "PCS" HAS BEEN OMITTED FROM ALL COMPONENT NUMBERS. THE COMPONENT CODE HAS BEEN OMITTED FROM ALL COMPONENTS EXCEPT EQUIPMENT. REFER TO THE P&ID LEGEND DRAWING GW M6 001, 002 AND 003 FOR ADDITIONAL INFORMATION REGARDING COMPONENT NUMBERING.
3. WATER STORAGE TANK IS INTEGRAL TO THE TOP OF THE CONTAINMENT SHIELD BUILDING. TANK WALLS ARE STAINLESS STEEL LINED TO PREVENT CORROSION.
4. THE PCS WATER RECIRCULATION LINES ARE INSULATED TO PREVENT FREEZING. INSULATION IS NOT REQUIRED FOR VENT, DRAIN AND ISOLATION VALVES AND INSTRUMENTATION WHICH ARE LOCATED IN A HEATED COMPARTMENT TO PREVENT FREEZING.
5. TEMPERATURE SENSORS ARE SEPARATED BY APPROXIMATELY 180 DEGREES WITHIN THE WATER STORAGE TANK TO PROVIDE REPRESENTATIVE SAMPLE.
6. PCS WATER STORAGE TANK DISCHARGE LINES ARE EACH TERMINATED AT A SPECIFIED HEIGHT ABOVE THE TANK BOTTOM. EACH PIPE INCORPORATES AN INVERTED INLET AND IS ENCLOSED WITHIN A SCREEN TO PREVENT LINE OR ORIFICE BLOCKAGE. ELEVATIONS ABOVE TANK BOTTOM ARE 22.2ft. 14.7ft. AND 7.2ft.
7. COMBINATION FLOW RESTRICTING/FLOW MEASUREMENT ORIFICE.
8. TEE ORIENTED VERTICALLY TO DETECT VALVE WATER LEAKAGE FROM V001A/B. CROSS CONNECTING PIPING CONFIGURED TO DRAIN LEAKAGE.
9. PIPES TERMINATED WITHIN WATER DISTRIBUTION BUCKET SUSPENDED ABOVE CONTAINMENT DOME.
10. FLOW TRANSMITTER ISOLATION VALVES AND IMPULSE PIPING INSULATED UP TO SEALED TUBING.
11. REDUNDANT DRAINS TO COLLECT EXCESS PCS WATER AT OPERATING DECK ELEVATION.
12. PCS FLOW TRANSMITTERS AND SENSING LINES ARE FILLED WITH NON-FREEZING FLUID.
13. PCS AIR DISCHARGE TEMPERATURE ELEMENTS LOCATED ABOVE CONTAINMENT IN MIXED AIR DISCHARGE STREAM.
14. ALL TANK PIPING, WITH THE EXCEPTION OF THE DRAIN PENETRATIONS, ARE CONTAINED WITHIN A PIPE RACK LOCATED INSIDE THE TANK ALONG ITS OUTER WALL.
15. PIPE INLETS MUST BE SITUATED SIX INCHES ABOVE BOTTOM OF TANK.
16. PIPING CONTINUOUSLY SLOPES DOWNWARD FROM VALVES V001A/B TO CENTER OF CONTAINMENT DOME.
17. TANK OVERFLOW.
18. PIPE SERVES AS PCS WATER MAKEUP LINE AS WELL AS EMERGENCY FPS WATER SOURCE.
19. A SYSTEM OF COLLECTION TROUGHS AND WEIRS SERVES TO EVENLY DISTRIBUTE WATER ALONG THE OUTER CONTAINMENT SHELL SURFACE.
20. RECIRCULATION PIPING SUCTION LINE TERMINATES NEAR TANK BOTTOM WITH 1" SIPHON BREAK LOCATED ABOVE THE MINIMUM REQUIRED PCCWST VOLUME LEVEL.
21. PIPE CHANGES SAFETY CLASSIFICATION AT TANK PENETRATION POINT.
22. TEMPORARY CONNECTION FOR WASHDOWN OF CONTAINMENT VESSEL.
23. PIPE SLOPES CONTINUOUSLY FROM A SINGLE HIGH POINT TO AVOID ANY TRAPPED WATER VOLUMES.
24. SLOPE PIPING TOWARD DRAIN TO AVOID POTENTIAL FOR FREEZING.
25. VENT PIPE ELEVATION MUST BE ABOVE OVERFLOW PIPE ELEVATION.
26. RECIRCULATION LINE AND SIPHON BREAK ARE EACH TERMINATED ABOVE THE MINIMUM TANK TECH SPECS LEVEL AND BELOW THE LEVEL REQUIRED FOR MINIMUM FIRE WATER VOLUME.

Figure 6.2.2-1 (Sheet 1 of 2)

Passive Containment Cooling System Piping and Instrumentation Diagram

(REF) PCS 002



NOTES:

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2. PIPING SLOPES DOWNWARD TOWARD DRAIN. FLANGED CONNECTION LOCATED AT GROUND LEVEL FOR CONVENIENT USE OF ALTERNATE WATER SOURCE (POST 72 HOUR MAKEUP) AND, FLANGE FITTED WITH FIRE NOZZEL FITTING PER LOCAL FIRE REGULATIONS.
3. DRAIN LOCATED TO PROVIDE FOR CHEMICAL ADDITION TANK AND RECIRCULATION HEATER DRAIN.
4. PUMP AND HEATER TRIP ON LOW FLOW.
5. TANK AND THERMOSTATIC CONTROL ARE PROVIDED BY HEATER VENDOR.
6. SAMPLE CONNECTION.
7. OPEN FUNNEL PROVIDES FOR CHEMICAL ADDITION TO CHEMICAL ADDITION TANK.
8. TANK AND VALVES V037 AND V048 INSULATED TO AVOID FREEZING WITH OUT POWER TO HEATER FOR SEVEN DAYS.
9. SLOPE PIPING TOWARD DRAIN TO AVOID POTENTIAL FOR FREEZING.
10. LINES L014, L015, L019, L016, L020, L026, L046, L053, L057, L058, L059, L060, L063, L064, L069, AND L205 ARE DESIGNED TO WITHSTAND A SAFE SHUTDOWN EARTH QUAKE.
11. ELEVATION DIFFERANCE FOR DRAIN TERMINATION SHALL PROVIDE FOR MINIMUM FIRE WATER VOLUME.

Figure 6.2.2-1 (Sheet 2 of 2)

Passive Containment Cooling System Piping and Instrumentation Diagram

(REF) PCS 002

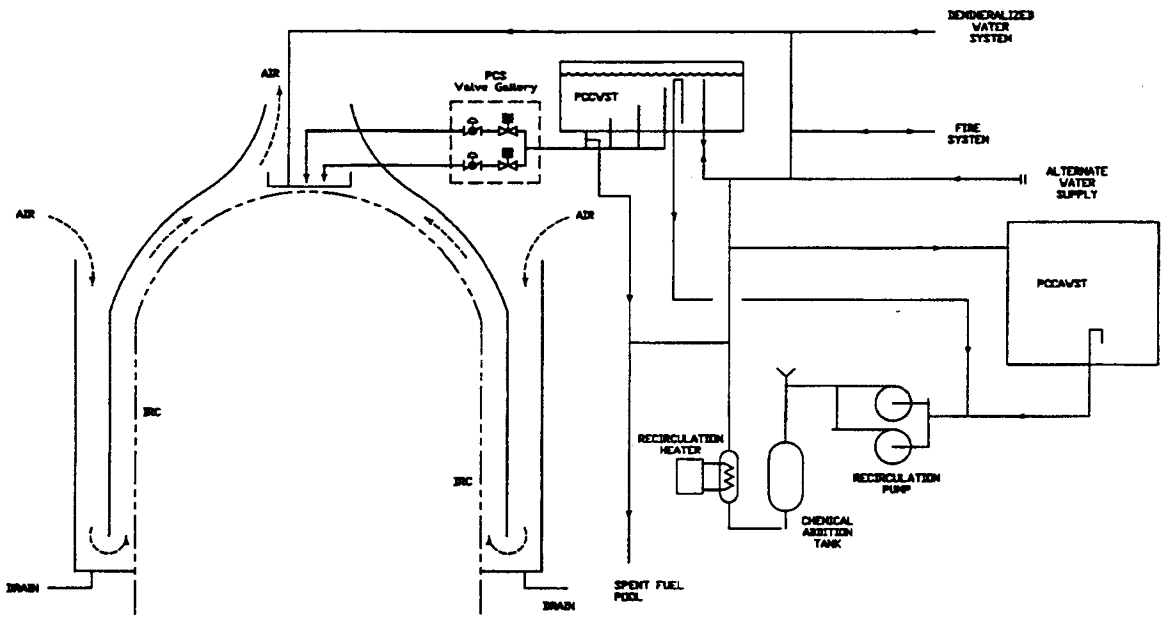

Figure 6.2.2-2

Simplified Sketch of Passive Containment Cooling System

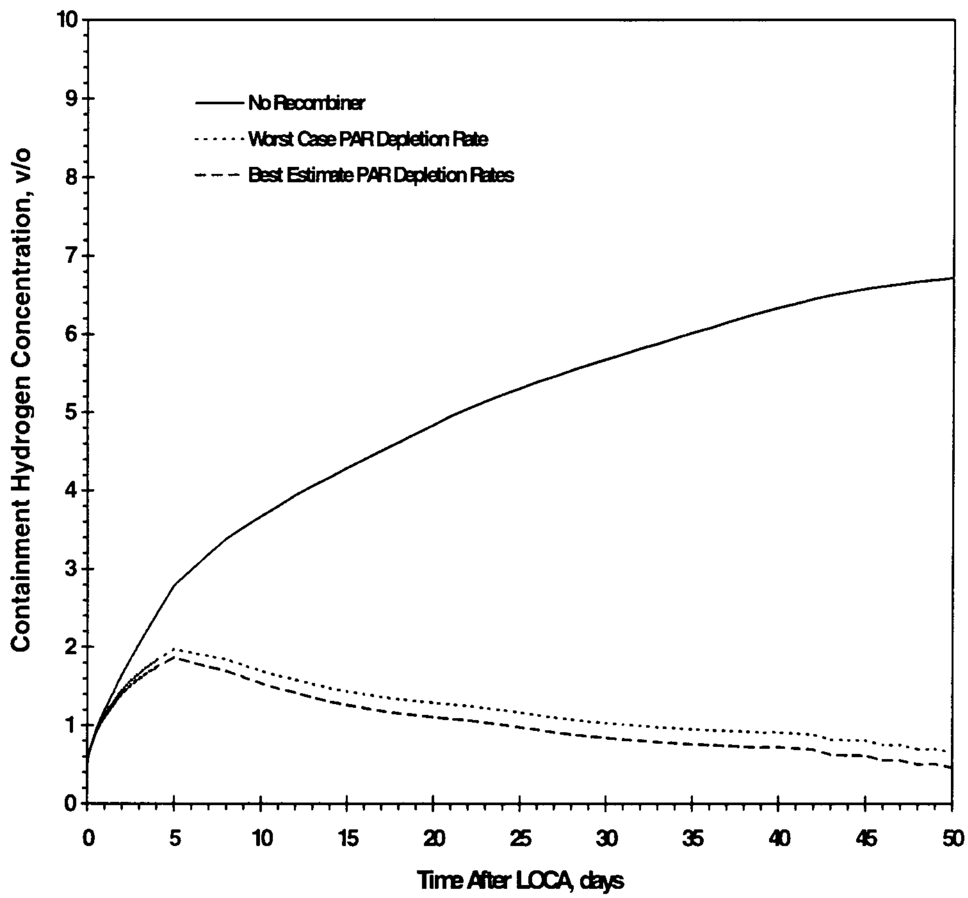

Figure 6.2.4-1

Hydrogen Concentration

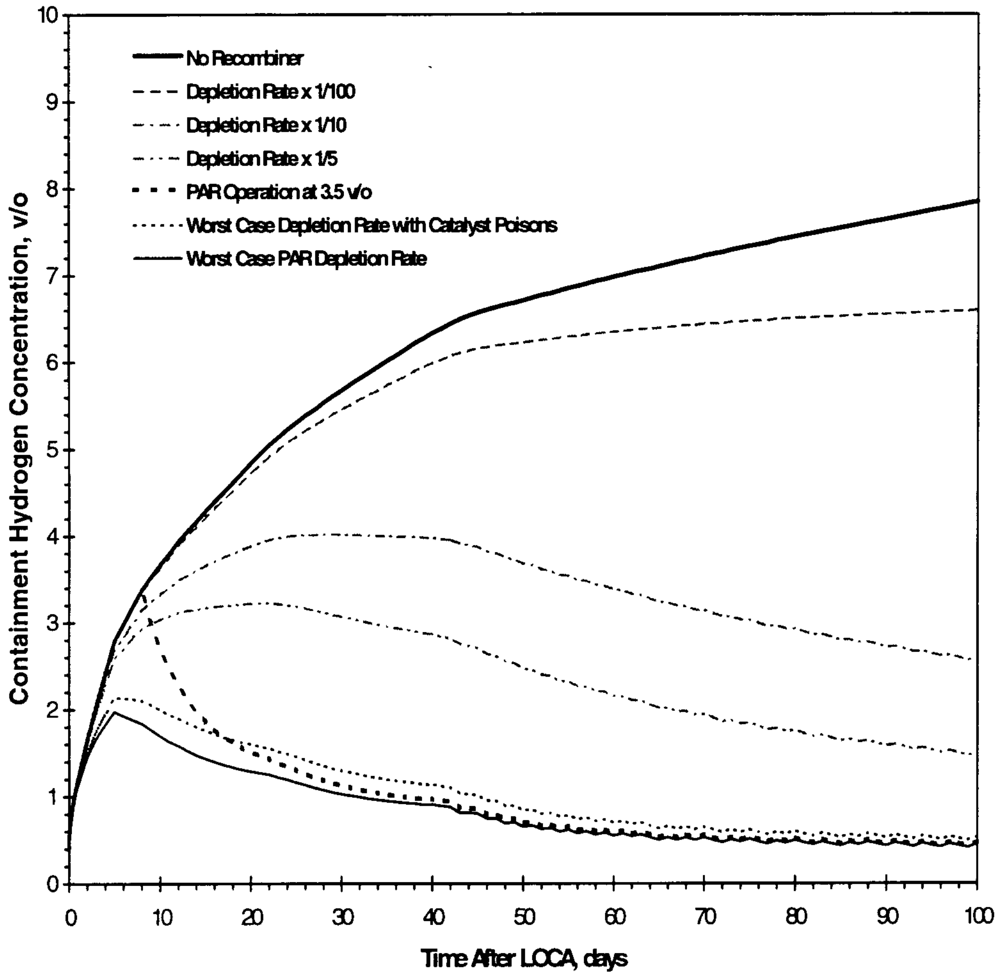

Figure 6.2.4-2

Hydrogen Concentration in Containment

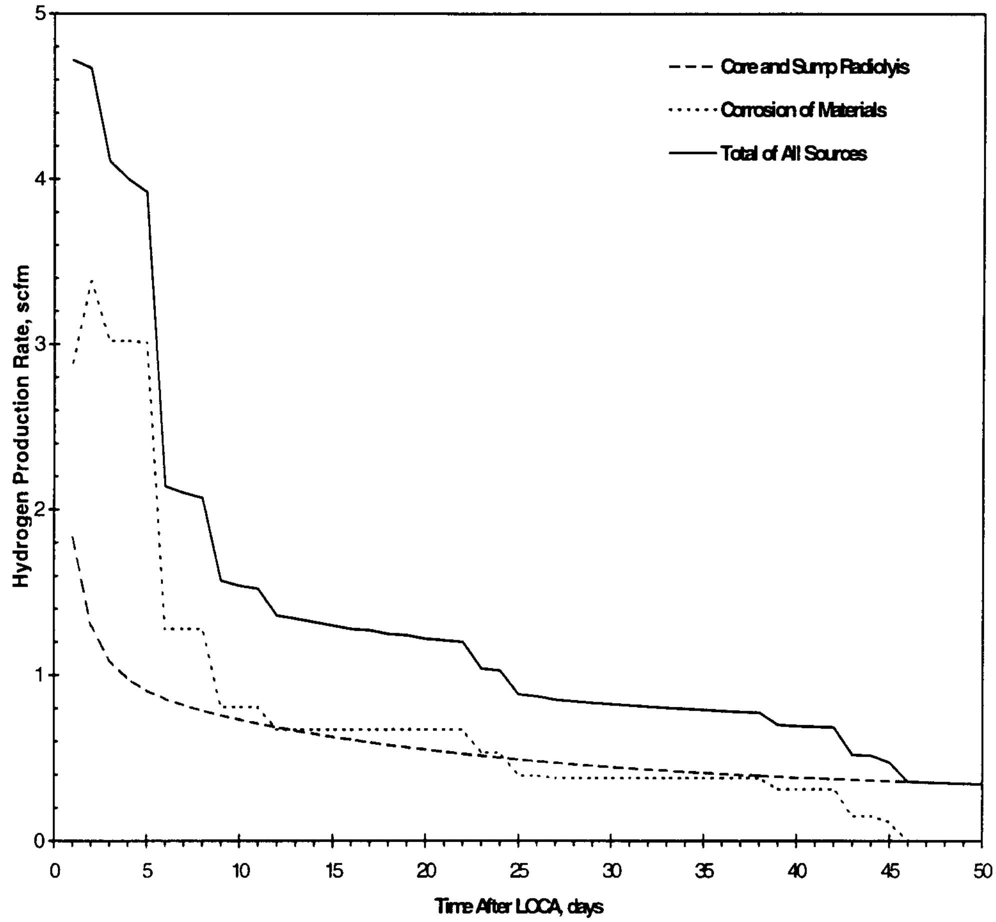

Figure 6.2.4-3

Hydrogen Production Rate

PAR SENSITIVITY STUDY - DRY CONDITIONS
IMPACT ON CONTAINMENT H2 CONCENTRATIONS

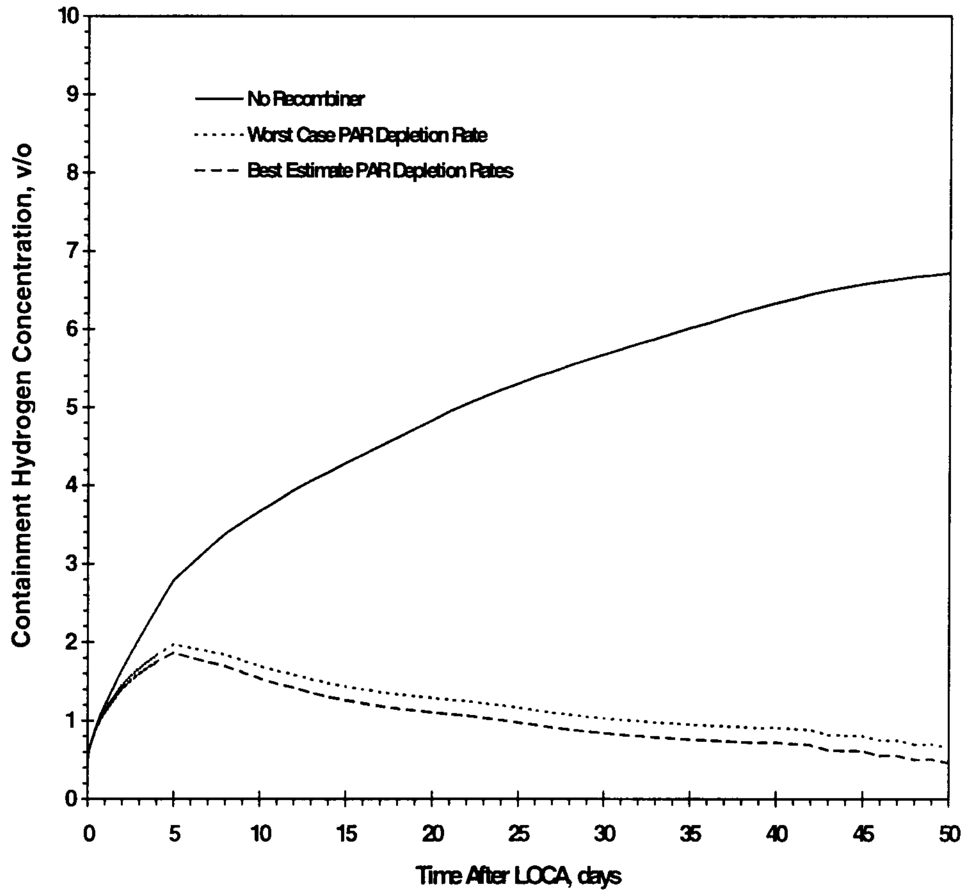

Figure 6.2.4-4

Hydrogen Production in Containment

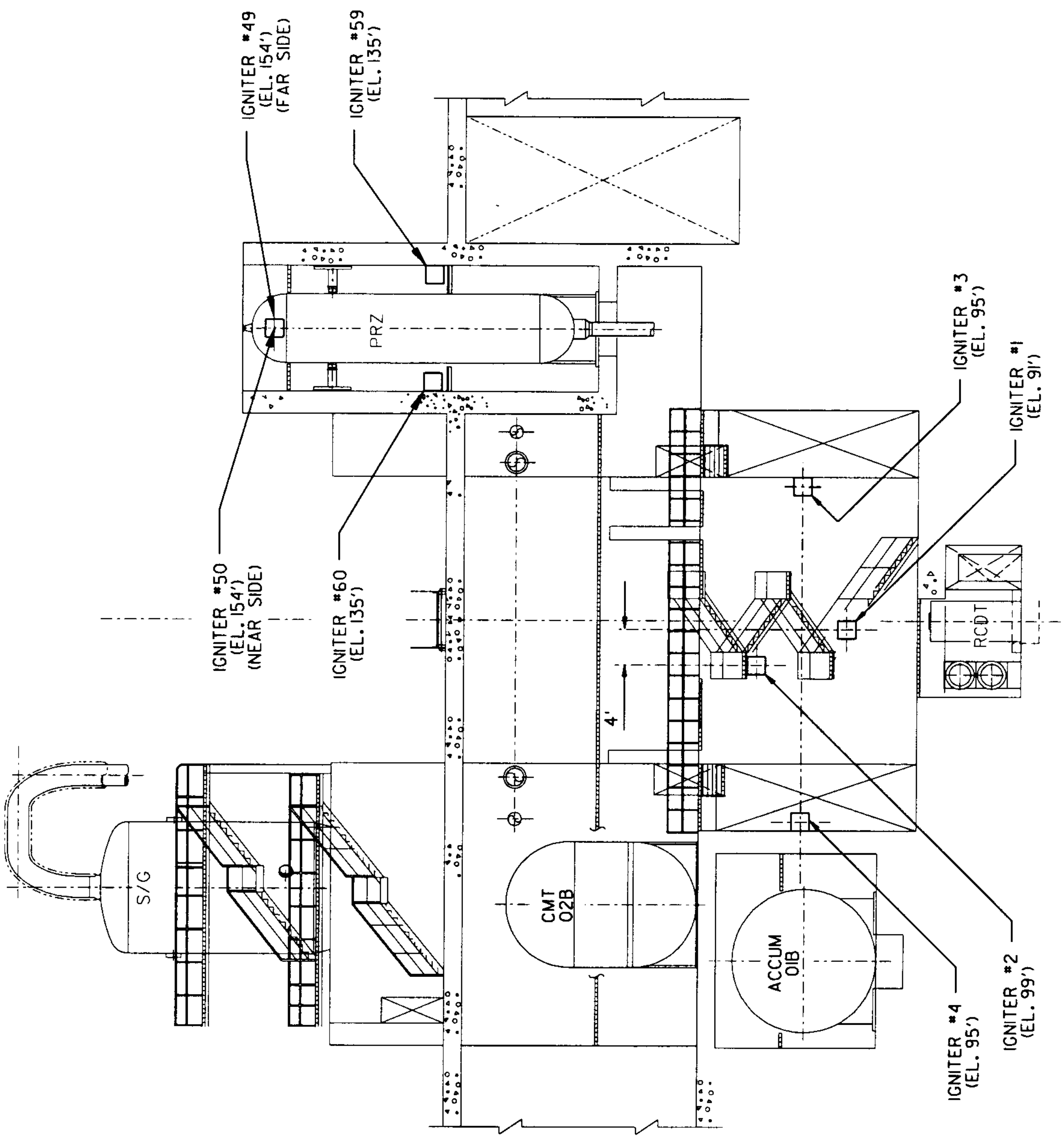

Figure 6.2.4-5

Hydrogen Igniter Locations - Section View

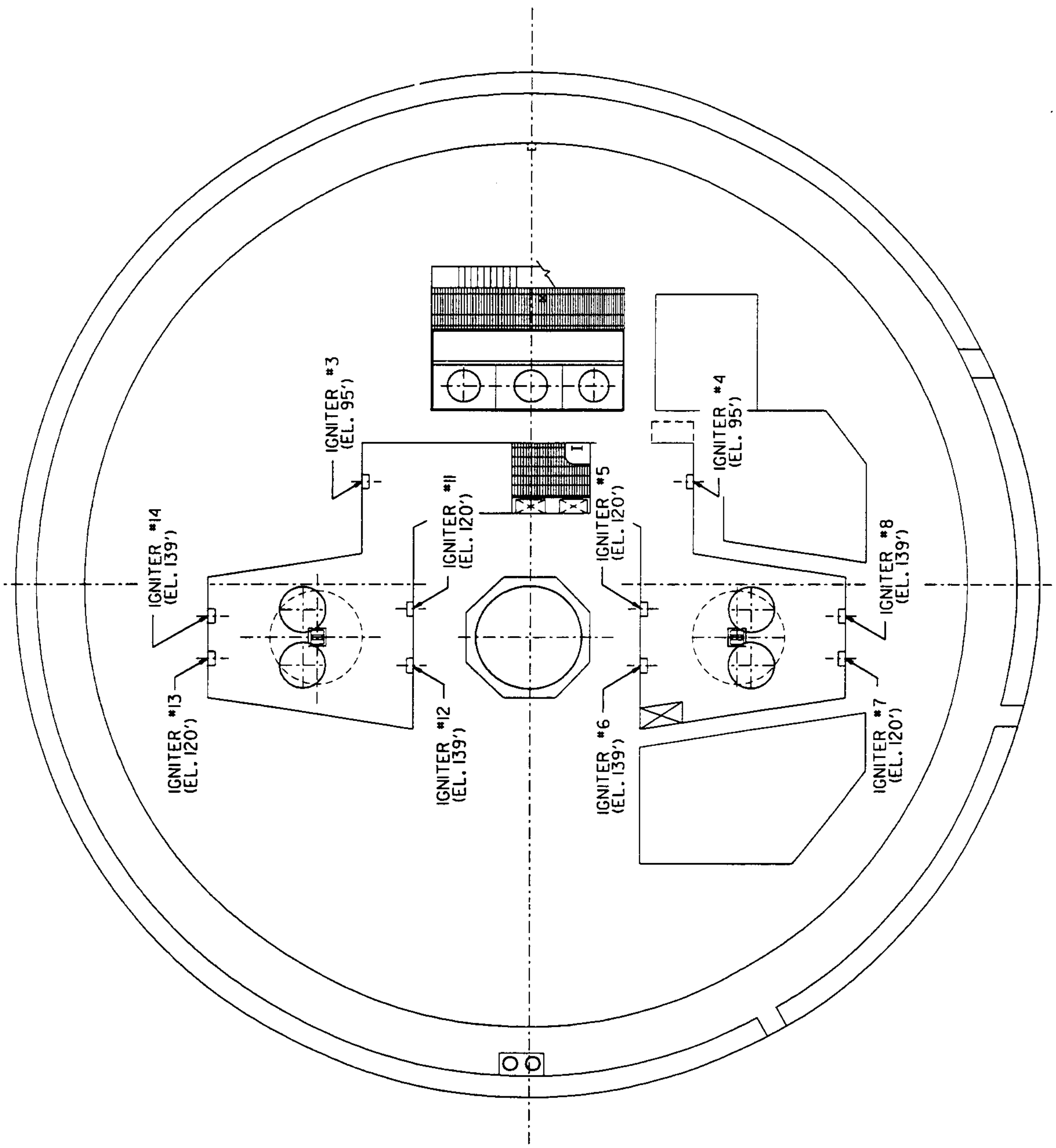

Figure 6.2.4-6

Hydrogen Igniter Locations
Plan View Elevation 82′-6″

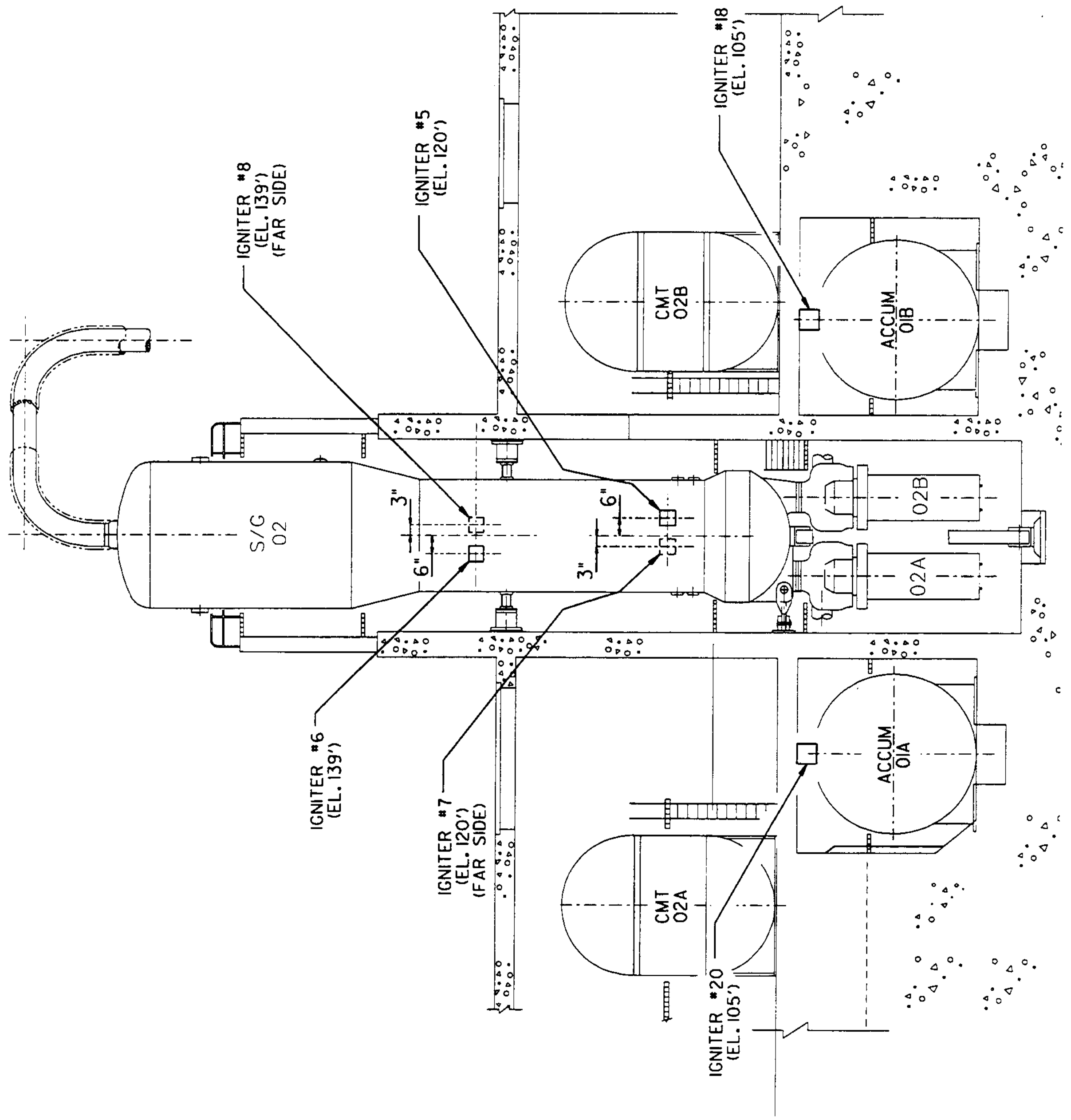

Figure 6.2.4-7

Hydrogen Igniter Locations - Section View

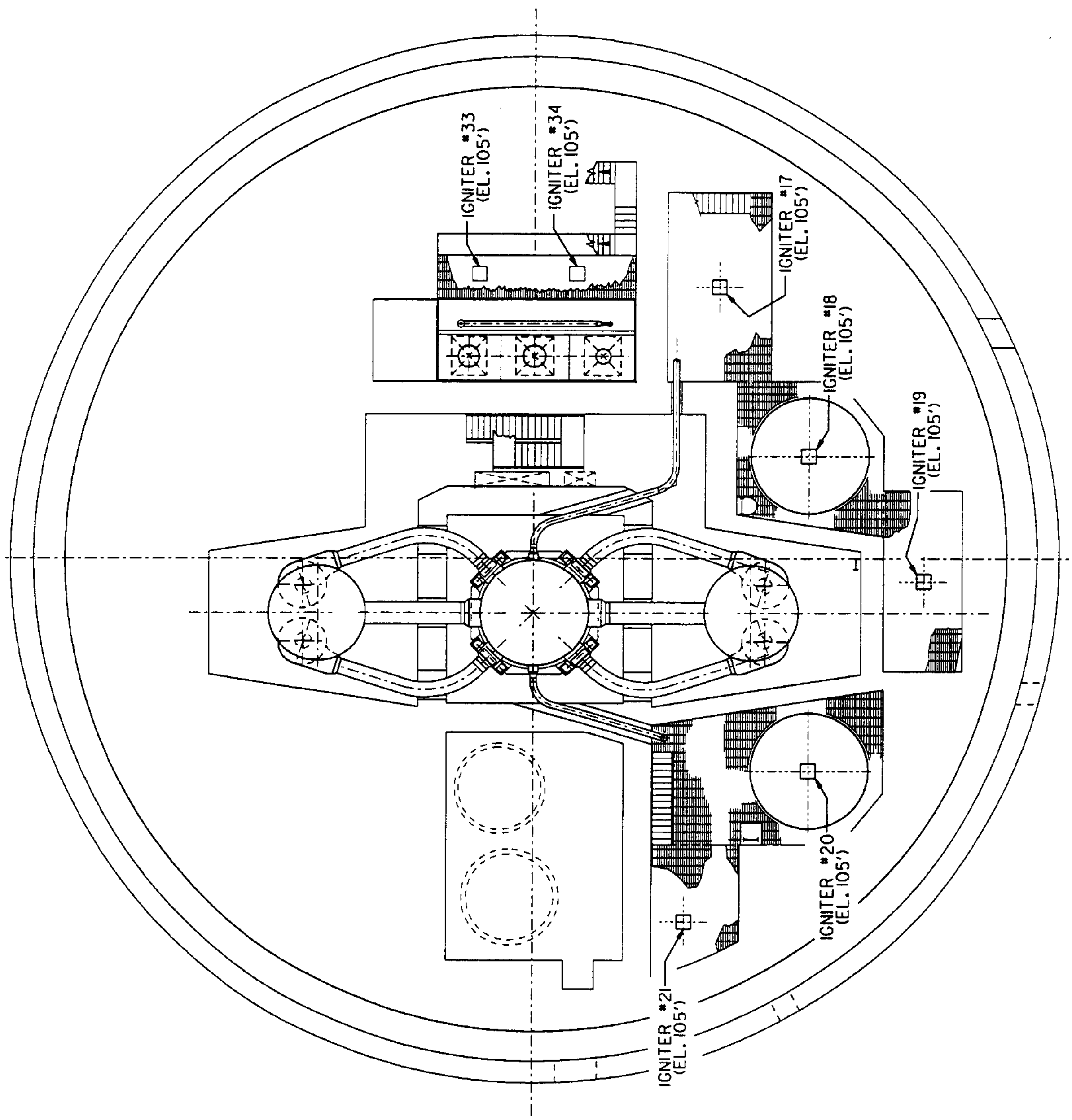

Figure 6.2.4-8

Hydrogen Igniter Locations
Plan View Elevation 96'-6"

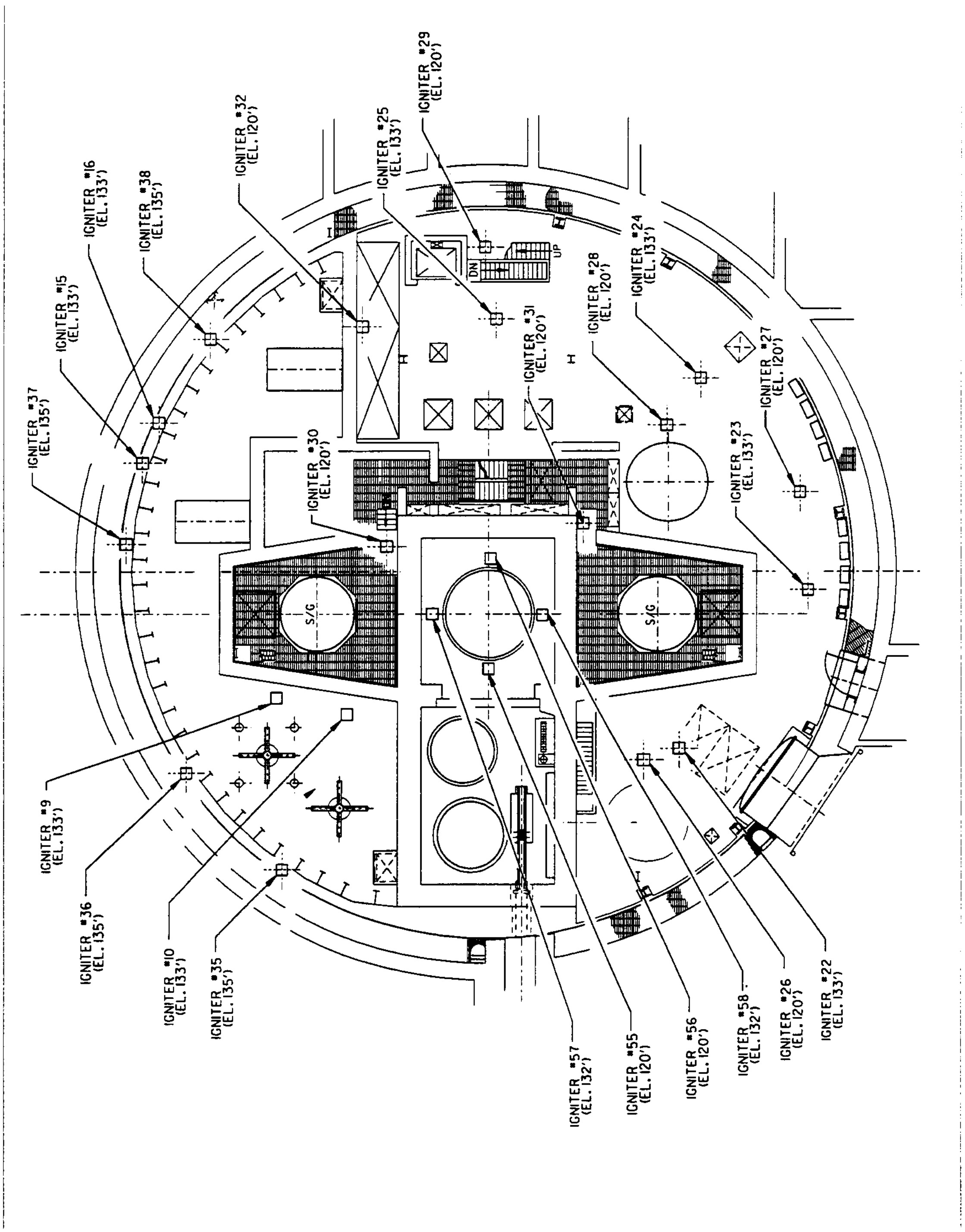

Figure 6.2.4-9

Hydrogen Igniter Locations
Plan View Elevation 100′ & 107′-2″

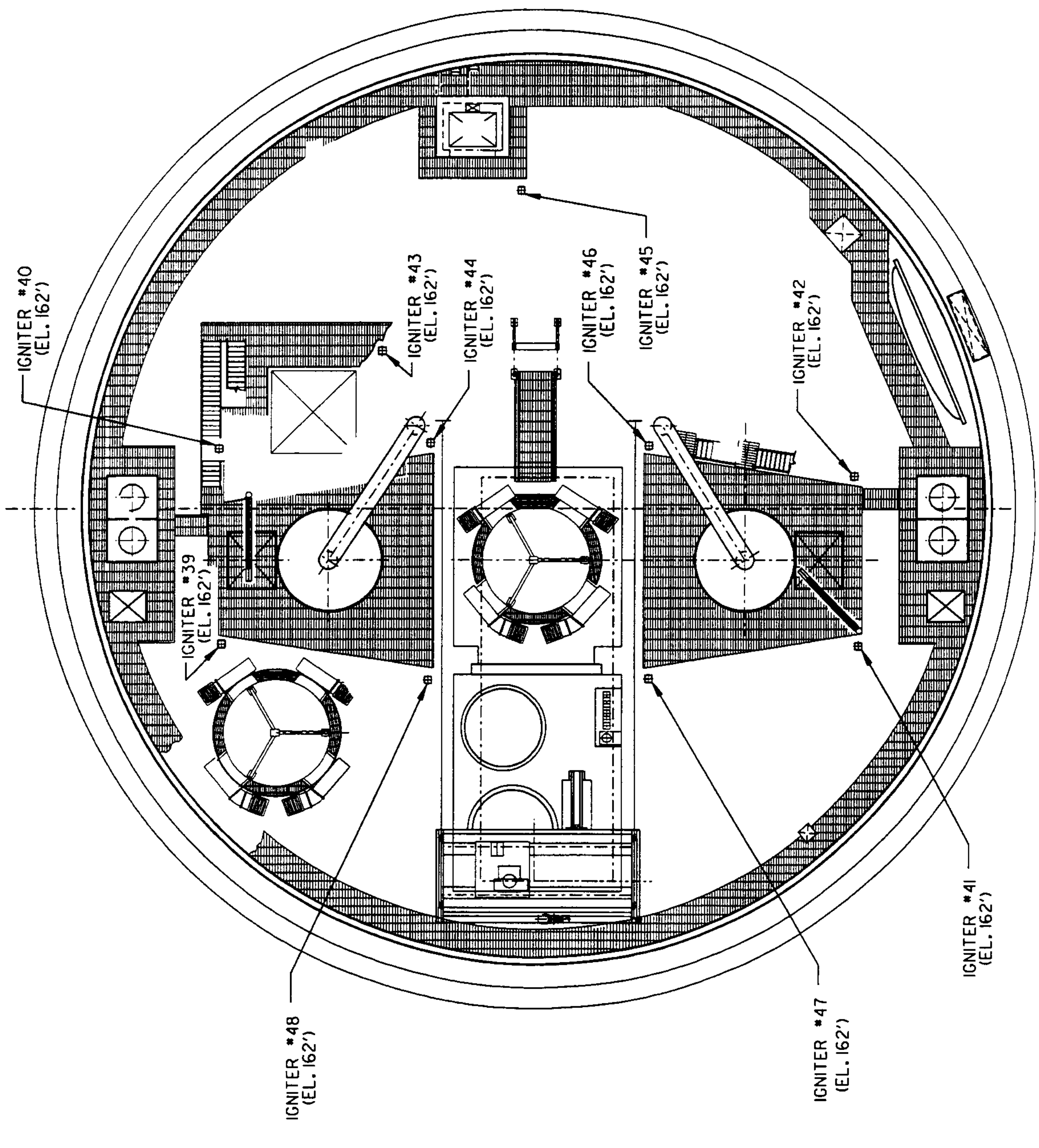

Figure 6.2.4-10

Hydrogen Igniter Locations
Plan View Elevation 160′-6″ & 153′-0″

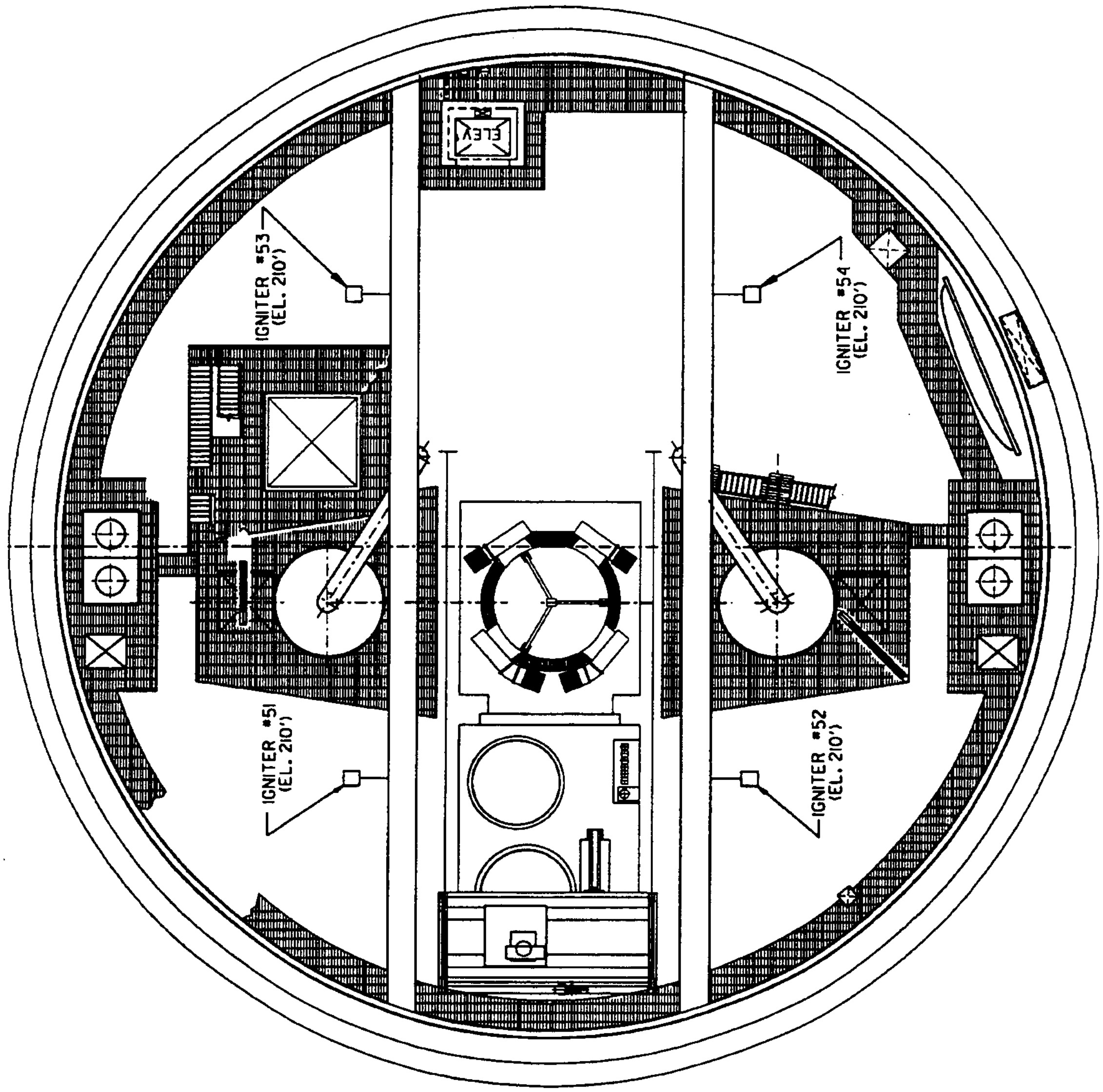

Figure 6.2.4-11

Hydrogen Igniter Locations
Plan View Elevation 210″

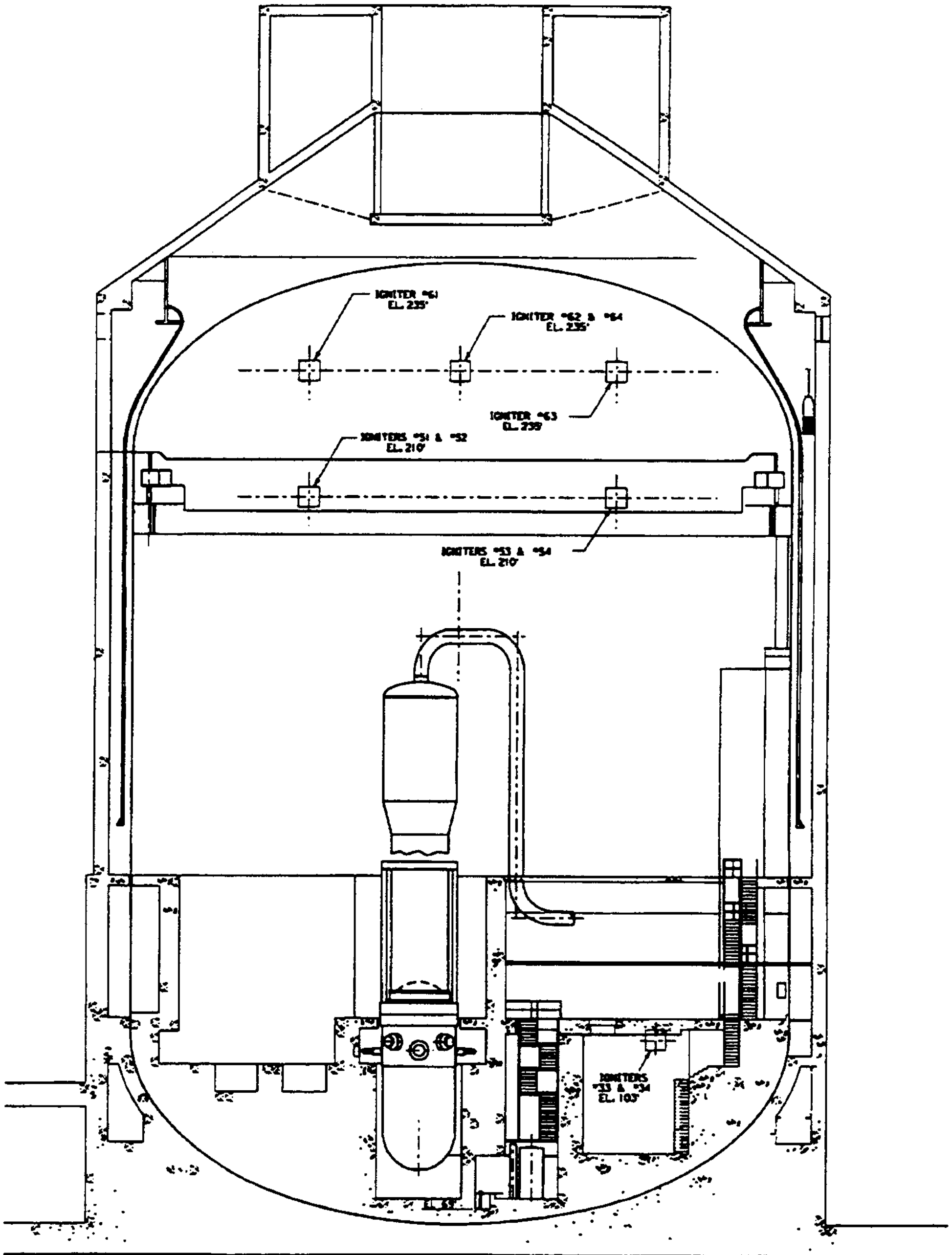

Figure 6.2.4-12

Igniter Locations in the Upper Compartment at Elevations 210′ & 235′ Sectional View

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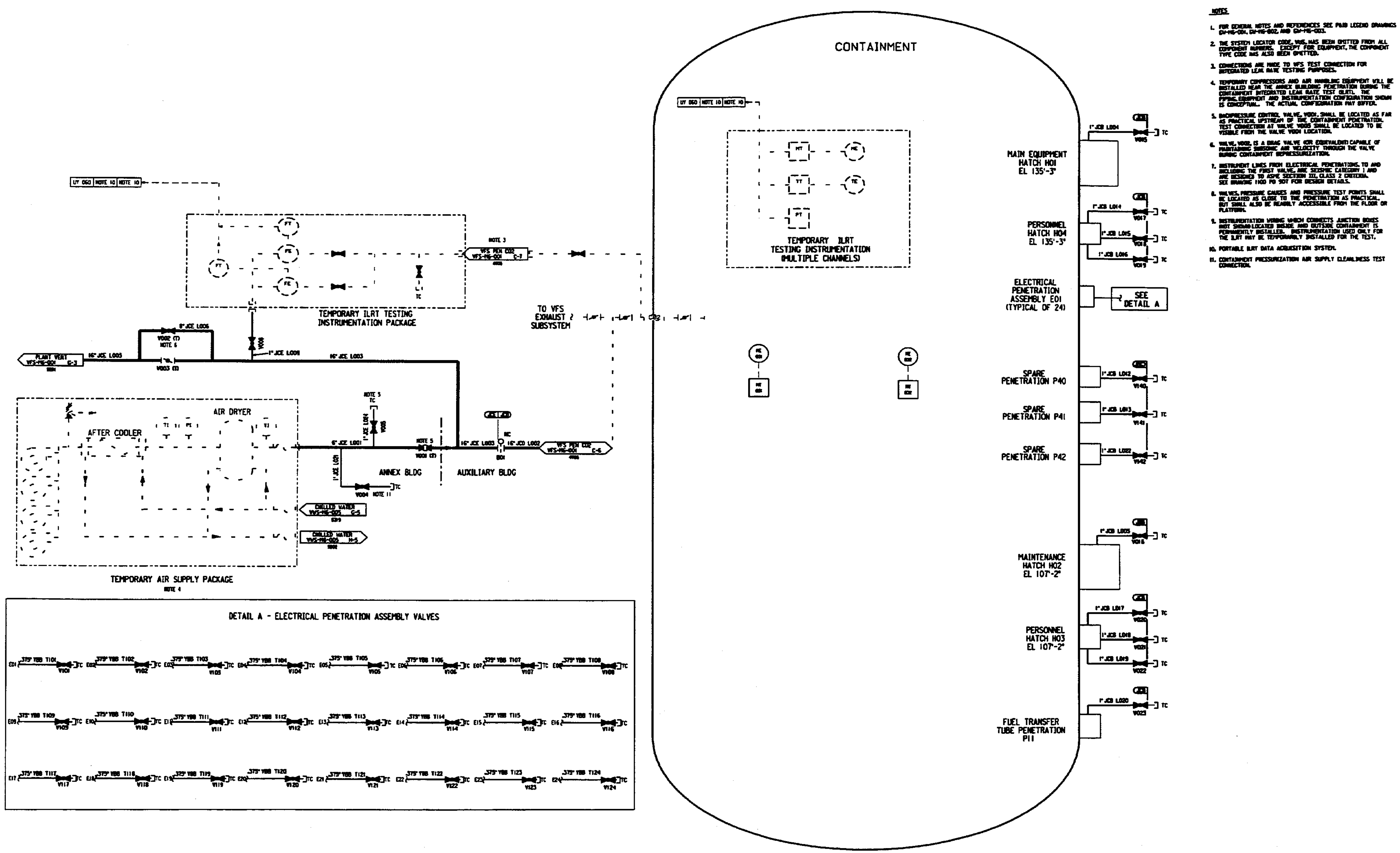

Figure 6.2.5-1

Containment Leak Rate Test System Piping and Instrumentation Diagram