# CHAPTER 6: ENGINEERED SAFETY FEATURES

## 6.1 GENERAL

Engineered safety feature (ESF) systems are provided to mitigate the consequences of postulated accidents. The following ESF systems are discussed in this chapter:

a. Containment structures
    1. Primary
    2. Secondary.

b. Containment systems

c. Emergency core cooling system
    1. High pressure coolant injection system
    2. Automatic depressurization system
    3. Core spray system
    4. Low pressure coolant injection mode of residual heat removal system.

d. Main control room habitability systems.

In addition to the ESF systems discussed in this chapter, other ESF systems discussed elsewhere are provided to limit the consequences of postulated accidents. The ESF systems are covered in Chapter 6 and those other locations referenced in Table 6.1-1.

The information provided herein demonstrates the following:

a. The concepts upon which the operation of each system is predicated have been proven by tests under simulated accident conditions and/or by conservative extrapolations from present knowledge and experience

b. Component reliability, system independency, redundancy, and separation of components or portions of systems ensure that the feature will accomplish its intended purpose and will function for the period required

c. Provisions for test, inspection, and surveillance have been made to ensure that the feature will be dependable and effective upon demand

d. The material used will withstand the postulated accident environment, including radiation levels, and the radiolytic decomposition products which may occur will not interfere with ESF systems.

TABLE 6.1-1 ENGINEERED SAFETY FEATURES DISCUSSED IN OTHER CHAPTERS OF FERMI 2 UFSAR

| Engineered Safety Fearures | UFSAR Location |
|---|---|
| Chapter 4 | |
| Control rod velocity limiter | 4.5.2 |
| Control rod drive housing supports | 4.5.3 |
| Chapter 5 | |
| Main steam line flow restrictors | 5.5.4 |
| Main steam line isolation valves | 5.5.5 |
| Chapter 7 | |
| Main steam line monitoring system | 7.3.2, 11.4.3 |
| Chapter 8 | |
| Onsite power systems | 8.3 |
| AC power systems | 8.3.1 |
| DC power systems | 8.3.2 |
| Chapter 9 | |
| Emergency equipment cooling water and emergency equipment service water systems | 9.2.2 |
| Ultimate heat sink | 9.2.5 |
| RHR service water system | 9.2.5.1 |
| RHR complex reservoir | 9.2.5.2.1 |
| Mechanical draft cooling towers | 9.2.5.2.2 |
| ESF cooling and ventilation units | 9.4.2 |

## 6.2 CONTAINMENT SYSTEMS

### 6.2.1 Containment Functional Design

On September 9, 1992, the NRC issued Amendment 87 to the Fermi 2 operating license authorizing a change in the thermal power limit from 3293 MWt to 3430 MWt, a 4.2 percent increase in the thermal power and a 5 percent increase in steam flow. The Fermi 2 Power Uprate Program followed GE Nuclear Energy guidelines and evaluations for BWR power plants (References 1, 2, 3, and 4).

On February 10, 2014, the NRC issued Amendment 196 to the Fermi 2 operating license authorizing a change in the thermal power limit from 3430 MWt to 3486 MWt, a 1.64 percent increase in thermal power and a 1.88 percent increase in steam flow. This changed the net electrical capacity from 1150 MWe to approximately 1170 MWe. This power uprate was performed in accordance with 10 CFR 50, Appendix K and reflects the improvement in feedwater flow measurement. The Fermi 2 Measurement Uncertainty Recapture (MUR) power uprate followed the GE generic guidelines and evaluations for BWR plants provided in GEH Topical Report NEDC-32938P-A, “Generic Guidelines and Evaluations for General Electric Boiling Water Reactor Thermal Power Optimization,” Revision 2, May 2003 (Reference 30). The analyses performed at 102% of the pre-MUR licensed thermal power (3430 MWt) remain applicable at the MUR uprated thermal power (3486 MWt) because the 2% uncertainty factor discussed in Regulatory Guide 1.49 is effectively reduced by the improvement in feedwater flow measurements.

Short-term and long-term containment analyses results are reported in Subsection 6.2.1.3. The short-term analysis is directed primarily at determining the drywell pressure responses during the initial blowdown of the reactor vessel inventory to the containment following a large break inside the drywell. The long-term analysis is directed primarily at the pool temperature response, considering the decay heat addition to the pool.

#### 6.2.1.1 Design Bases

The containment system design meets the following safety design bases:

a. The containment systems shall have the capability to withstand the peak transient pressures and temperatures that could occur due to a postulated design-basis accident (DBA), intermediate-break accident (IBA), or small-break accident (SBA). The assumptions and criteria used to conservatively predict the short-term pressure and temperature response of the containment system drywell and suppression chamber during these accident conditions are provided in the Mark I Owners Group Load Definition Report (Reference 5), the Fermi 2 Plant Unique Load Definition Report (Reference 6), and in NUREG-0661 (Reference 7). The reevaluation of containment response for power uprate is provided in References 3 and 4. The long-term response of the drywell and suppression chamber is described in Subsection 6.2.1.3.3.

   No one accident results in the simultaneous occurrence of the maximum values of pressure and temperature (drywell design pressure and temperature, suppression chamber design pressure and temperature)

b. The containment systems shall accommodate the effects of metal/water reactions and other chemical reactions following the postulated DBA to values consistent with Regulatory Guide 1.7

c. The containment shall have the capability to maintain its functional integrity indefinitely after a postulated DBA, IBA, or SBA

d. The containment design shall permit filling the containment system drywell with water to a level above the reactor core

e. The containment system shall be protected against missiles from internal or external sources and excessive motion of pipes that could directly or indirectly endanger the integrity of the containment

f. The containment shall withstand jet forces associated with the flow from the postulated rupture of any pipe within the containment

g. The containment shall limit leakage during and following a postulated accident to values less than leakage rates that would result in offsite doses greater than the limits established in 10 CFR 50.67 or 10 CFR 100

h. It shall be possible to periodically conduct such leakage tests as may be appropriate to confirm the integrity of the containment at calculated peak pressure resulting from the accident condition that produces the maximum pressure response (DBA)

i. There shall be means to direct the flow from postulated pipe ruptures to the pressure suppression pool, to distribute such flow uniformly throughout the pool, to condense the steam portion of the flow rapidly, and to limit the pressure differentials between the drywell and the wetwell during the various postaccident cooling modes. The hydrodynamic events of pool swell, condensation oscillation, and chugging that occur during these flow and steam condensation regimes are defined by NUREG-0661 (Reference 7) and the Mark I Owners Group Load Definition Report (Reference 5). The design basis of the containment system includes the loading conditions associated with these hydrodynamic events

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

k. There shall be the means for stable steam condensation of safety/relief valve (SRV) discharges into the suppression pool during transient and accident plant conditions. The containment system design basis includes the SRV actuation events, associated hydrodynamic loading conditions, and pool temperature limits described in NUREG-0661 (Reference 7), NUREG-0783 (Reference 8), and the Mark I Owners Group Load Definition Report (Reference 5)

l. During the DBA, with the minimum emergency core cooling system (ECCS) pumps operating, and the available service water at the design maximum temperature, the long-term peak pool temperature shall not exceed the design temperature.

### 6.2.1.2 System Design

There are two passive provisions for containment of possible postaccident airborne contamination, the primary containment system and the secondary containment system. A perspective drawing illustrating these systems and their relationship is presented in Figure 5.1-4.

In addition to these two passive containment systems, the gases in either the primary or secondary containment can be exhausted through the standby gas treatment system (SGTS). This arrangement ensures that any accident-related discharge will be filtered by the SGTS before release. The SGTS is discussed in Subsection 6.2.3.

#### 6.2.1.2.1 Primary Containment

The primary containment is a pressure suppression system. It consists of a drywell that houses the reactor pressure vessel (RPV); reactor coolant recirculating loops, and other branch connections of the reactor coolant system; a pressure suppression chamber that stores a large volume of water; a vent system connecting the drywell and the pressure suppression chamber water; a vacuum relief system; isolation valves; and service equipment.

In the event of a process system piping failure within the drywell, reactor water and steam would be released into the drywell. The resulting increased drywell pressure would force a mixture of air, steam, and water through the vents into the pool of water that is stored in the suppression chamber. The steam would condense in the suppression pool, resulting in a rapid pressure reduction in the drywell. The hydrodynamic events of pool swell, condensation oscillation, and chugging associated with the venting and steam condensation processes are described in NUREG-0661 (Reference 7) and the Mark I Owners Group Load Definition Report (Reference 5). Noncondensable gases trans-ferred to the suppression chamber pressurize the chamber and are subsequently vented back to the drywell to equalize the pressure between the two vessels. Cooling of the primary containment under accident conditions is provided by the containment cooling and spray modes of the residual heat removal (RHR) system, as discussed in Subsection 6.2.2. Appropriate isolation valves are actuated to ensure containment of radioactive materials that might otherwise be released from the primary containment.

Detailed design information of the primary containment is given in Subsection 3.8.2 and in References 9 and 10. The information given there includes the dynamic loads that could be imposed on the torus, the vent system, the torus internal structures, and the torus attached piping following a LOCA. Also given there is a description of the methods used to determine these loads and how these loads were incorporated in the structural and attached piping design. A summary of important design parameters of the primary containment is presented in Table 6.2-1. The more important features of the primary containment system are described below.

#### 6.2.1.2.1.1 Drywell

The drywell is a steel pressure vessel with a spherical lower portion, 68 ft in diameter, and a cylindrical upper portion, 38 ft 10 in. in diameter. The overall height is approximately 114 ft 8 in. The drywell design pressure is 56 psig at a temperature of 281°F. The design temperature is 340°F with a coincident pressure of 25 psig.

The design, fabrication, inspection, and testing of the drywell vessel comply with requirements of the ASME Boiler and Pressure Vessel (B&PV) Code Section III, Nuclear Vessels, 1968 Edition with Summer 1969 Addenda, Subsection B, Requirements for Class B Vessels, which pertain to containment vessels for nuclear power plants. The steel head and shell of the drywell are fabricated of SA-5l6GR70 steel plate, firebox quality, aluminum-killed to SA-300 requirements. Thermal stress in the steel shell due to temperature gradients is considered in the design. Special procedures not required by code have been used in the fabrication of the steel drywell shell. For seams exceeding 1-1/4-in. thickness, the plate was heated to a minimum temperature of 200°F prior to welding. For seams 1-1/4 in. or less, the plate was heated to a minimum temperature of l00°F if the ambient temperature was below 40°F.

Charpy V-notch impact tests were performed on specimens of all plate and forged materials.

Plates, forgings, and pipes of the drywell have an initial nil ductility transition (NDT) temperature of approximately 0°F when tested in accordance with the appropriate code for these materials. It can be reasonably expected that the drywell will not be pressurized or subjected to a substantial stress at temperatures below 30°F.

The drywell is enclosed in reinforced concrete for shielding purposes. Resistance to deformation and buckling of the drywell is provided over areas where the concrete backs up the steel shell. Above Elevation 572 ft 1 in., the drywell is separated from the reinforced concrete by a gap of approximately 2 in. This gap is filled with a compressible polyurethane material to allow for movement between the drywell and concrete. The bottom portion of the shell is totally embedded in concrete and therefore is not subject to significant thermal stresses. The transition zone (below Elevation 572.5 ft) is backed by compacted sand to allow for thermal expansion and to aid in the drainage of condensate that may accumulate in the gap outside the drywell. Sand in the four drain lines at azimuths 0, 90, 180, and 270 degrees have been removed up to the pipe upstream of the 90 degree elbow. Sand in the sand cushion or transition zone is still intact.

Provisions for protection of the drywell against earthquakes, missiles, and pipe whip, which could damage the primary containment, are discussed in Chapter 3.

#### 6.2.1.2.1.2 Pressure Suppression Chamber

The pressure suppression chamber is a steel pressure vessel, in the shape of a torus, below and encircling the drywell. It has a major diameter of 112 ft 6 in. and a cross-sectional diameter of 30 ft 6 in. It contains a total volume of approximately 251,980 $ft^3$. The suppression chamber is supported vertically by inside and outside columns and by a saddle support that spans the inside and outside columns. The support system transmits dead weight and seismic and hydrodynamic loading to the reinforced-concrete foundation slab of the reactor building. Space is provided outside the chamber for inspection and maintenance.

The pressure suppression chamber is designed for a temperature of 281°F and a pressure of 56 psig. The suppression chamber was originally designed to the same material and code requirements as the drywell vessel. The suppression chamber has been subsequently reevaluated for the effects of LOCA-related loads and SRV-discharge-related loads defined by the NRC Safety Evaluation Report NUREG-0661, the GE Reports NEDO-21888 (Mark I Containment Program Load Definition Report) and NEDC-31897P-1 (Generic Guidelines for General Electric Boiling Water Reactor Power Uprate). The criteria set forth in NUREG-0661 have been applied as the basis for acceptance of the analysis methods and the suppression chamber design. A detailed discussion of these reevaluations and their results is provided in the Fermi 2 Plant Unique Analysis Report (References 9 and 10) and in the Power Uprate Safety Analysis (Reference 3). All materials have an initial NDT temperature of approximately 0°F.

Where safety/relief valves terminate inside the suppression chamber, T-quencher devices are provided to aid in mitigating the associated SRV discharge loads in the suppression chamber. Reference 5 contains a description of the T-quencher design and its performance.

6.2.1.2.1.3 Vent Systems

Eight vent pipes connect the drywell and the pressure suppression chamber. Each pipe has a diameter of 6 ft 0 in. The vent pipes are designed for an internal pressure of 56 psig at 281°F. They will withstand an external pressure of 2 psig. Jet deflectors are provided in the drywell at the inlet of each vent pipe to prevent possible damage from jet forces, which might accompany a pipe break in the drywell. The vent pipes are fabricated of SA-516GR70 steel plate, firebox quality, aluminum-killed to SA-300 requirements, and comply with requirements of the ASME B&PV Code Section III, Subsection B. The pipes are enclosed with sleeves and provided with expansion joints to accommodate differential motion between the drywell and suppression chamber.

These vent pipes connect to a vent header in the form of a torus located in the air space of the suppression chamber. The vent header is nominally 1/4-in. thick and has an inside diameter of 4 ft 3 in. Near the vent line-vent header intersection, the vent header has an inside diameter of 6 ft 0 in. Conical transition segments connect the smaller and larger diameter portions of the vent header.

The vent header and downcomer system inside the torus was designed, fabricated, and erected in accordance with ASME B&PV Code Section III, 1968 Edition through winter 1969 addenda, Class B requirements but it is not leak tested.

Projecting downward from the header are 80 downcomer pipes, each 24 in. in diameter and terminating below the surface of the water in the suppression chamber pool. The pool water level is maintained to ensure a 3.00- to 3.33-ft submergence of the downcomer pipes. The header is designed to meet the same temperature and pressure requirements as the vent pipes.

The vent system has also been evaluated for the effects of LOCA-related loads and SRV-discharge-related loads defined by NUREG-0661 and NEDO-21888. As with the suppression chamber discussed above, a detailed discussion of these evaluations is provided in References 3 and 9.

Vacuum breakers discharge from the suppression chamber into the vent header system. Vacuum breaker sizing is based on the Moss Landing (Reference 11) test configuration.

Both the drywell and the pressure suppression chamber can be vented to the atmosphere through the SGTS or reactor building ventilation system.

#### 6.2.1.2.1.4 Pipe Penetrations

Primary containment penetrations are designed for peak transient conditions to be expected during a LOCA. They will withstand, or are shielded from, the forces caused by impingement of fluid from the rupture of the largest local pipe or connection. Specific evaluations of the suppression chamber penetrations to address the requirements of NUREG-0661 are described in Reference 10.

These penetrations are designed to accommodate, without failure, any combination of thermal and mechanical stresses, which may be encountered during all modes of operation. (Refer to Subsection 3.8.2.1.3.)

Primary containment system piping penetrations are enumerated in Table 6.2-2. Electrical penetrations are listed in Table 6.2-3.

Relative movement between the containment penetrations and the drywell is accommodated by using bellows-type expansion joints (Figure 6.2-1). For this type of penetration, a sleeve passes through concrete and is welded to the primary containment vessel. The process line that passes through the penetration is anchored to allow only radial thermal expansion. A guard pipe surrounds the process line to protect the bellows and maintain containment integrity should the process line fail within the penetration. Insulation and air gaps are provided to reduce radiant heating of the guard pipe and the penetration sleeve and bellows. The dual-ply bellows arrangement permits periodic leak testing of these penetrations at a pressure equal to the primary containment DBA pressure (see Subsection 6.2.1.4) as well as continuous monitoring capability.

Figure 6.2-1 presents the containment penetration configuration for a typical process line of the reactor coolant pressure boundary (RCPB). As it passes through the drywell containment vessel and the concrete biological shield, the process line is enclosed in a guard pipe that is attached to it through a multiple head fitting. This fitting is a one-piece forging with integral flues or nozzles made to SA-105, Grade II requirements, and designed to meet all requirements of the ASME B&PV Code Section III, Class l. The guard pipe design is based on 90 percent of the material yield stress when pressurized to 1250 psi due to process line rupture. The process line penetration sleeve is welded to a bellows which in turn is welded to the guard pipe. The bellows assembly accommodates the differential thermal expansion and seismic movements between the process pipe and the drywell in the three mutually perpendicular directions.

Pipe penetrations for those applications not requiring provisions for relative movement between pipe and containment shell are illustrated in Figures 6.2-2 and 6.2-3.

The design of the penetrations takes into account the simultaneous stresses associated with normal thermal expansion, live and dead loads, seismic loads, and loads associated with LOCAs within the drywell. For all of these conditions, including appropriate combinations of these loads, the resultant combined stresses in the pipe and penetration components do not exceed design limits allowable by applicable codes. If, in addition, the jet force loadings resulting from random failures of the steam pipe are included, the resultant stresses in the pipe and penetration do not exceed allowable code stresses for fault conditions.

Cold piping and ventilation duct penetrations are welded directly to the sleeves. Bellows and guard pipes are not necessary in these applications because the thermal stresses are small and accounted for in the design of the weld joints.

#### 6.2.1.2.1.5 Electrical Penetrations

Figure 6.2-4 shows a typical electrical penetration used for transmitting electric power, and instrumentation and control signals from the reactor building into the primary containment. Separation of divisions is obtained by locating penetrations on the semi-peripheries of the containment at Elevation 604 ft. The division boundary is the east-west diameter. Division I is on the north half; Division II, the south half.

One group of six penetrations is used to transmit power to two 7100-hp, three-phase, 3920-V reactor coolant recirculation pump motors.

One group of two penetrations is used for low-voltage power, motor control three-phase, 480-V, 208-V, and single-phase 120-V, and 125-V-dc loads.

One group of two penetrations is used for 120-V signals for limit and level switches. These penetrations also contain an isolated penetration within a penetration for the reactor protection system (RPS).

One group of six penetrations is used for low-voltage instrumentation cable to transmit control and temperature signals for control rod position from reactor to recorders and computer.

One penetration is used for analog signals, to be used for vibration tests and miscellaneous primary signals.

One group of two penetrations for low-voltage shielded instrumentation thermocouple extension lead wire is used to transmit RPV and other equipment temperature signals to recording and readout equipment.

One group of four penetrations is used for neutron monitoring. The penetrations include the following coaxial and triaxial cables per penetration:

a. Three triaxial - for intermediate-range monitors
b. Two triaxial - for source range monitor
c. 48 coaxial - for local power range monitor.

All penetrations are sized for a 12-in.-diameter nozzle and are hermetically sealed, with provisions for continuous leak detection at design pressure. The penetrations are factory assembled, prewired and tested, and do not require field welding for installation due to the flange mount design. Radiation shielding is integral, thus minimizing radiation shine, and eliminating overhanging moments which would occur if shielding were mounted externally.

Edison made a review of the primary containment electrical penetrations to determine that the electrical penetration assemblies were designed to withstand, without the loss of mechanical integrity, the maximum available fault current versus time conditions that could occur, given single random failures of circuit overload devices as recommended by Regulatory Guide 1.63, Revision 1.

In making the review, the following assumption was primary: The $I^2t$ characteristics of the penetration conductors as furnished by Conax Corp. were used as a basis for determining integrity. The $I^2t$ curves as furnished by Conax Corp. were conservative in nature and the $I^2t$ curve points were not necessarily the points of damage to the mechanical integrity of the penetrations.

The following positions are in line with the guidelines set forth in Regulatory Guide 1.63, which were taken by Edison, based on the results of this review.

a. For low-energy penetrations, maximum fault current does not approach the $I^2t$ of the penetration conductor. No backup or redundant protection is provided

b. On low-voltage power penetrations where maximum fault current versus time will exceed the $I^2t$ of the penetration conductor (considering single random failure of the primary protection), backup protection is provided by one of the two following methods:

    1. If adequate backup protection can be obtained from the feeder position and the fault can be cleared in sufficient time to prevent reaching the $I^2t$ of the penetration conductor - no additional redundant protective devices are provided

    2. Where the feeder position cannot provide adequate clearing time, an additional protective device, fuse or breaker as necessary, is provided.

There are six medium voltage power penetrations, and they are used for the reactor recirculation pump motor M-G set output from the generator to the pump motors. In these cases the primary protection is provided by tripping the main M-G set motor drive incoming circuit breaker positions. Backup protection is provided by tripping the generator field breakers. Proper relaying ensures operation of the field breaker.

Loads to the primary containment not necessary for reactor operation (i.e., lighting and welding) are maintained in a deenergized condition.

6.2.1.2.1.6 Traversing In-Core Probe Penetrations

A total of five traversing in-core probe (TIP) guide tubes and two spare penetrations pass through the primary containment. Penetrations of these guide tubes through the primary containment are sealed with a Class I drywell penetration seal weld which meets the requirements of the ASME B&PV Code Section III. These seals also meet the intent of Section III of the Code even though the Code has no provisions for qualifying the procedures or performance.

#### 6.2.1.2.1.7 Personnel and Equipment Access Lock

One personnel access lock is provided for access to the drywell (Figure 6.2-5). The lock has two gasketed doors in series. The inner door has a double seal gasket and the outer door a single gasket. The doors are designed and constructed to withstand the drywell design pressure. The doors are mechanically interlocked to ensure that at least one door is locked. The locking mechanisms are designed so that a tight seal will be maintained when the doors are subjected to either internal or external pressure. The seals are capable of being tested for leakage. Two equipment access hatches and a control rod drive (CRD) removal hatch are in the spherical portion of the drywell. These hatches have double testable seals and are bolted in place (Figure 6.2-6).

#### 6.2.1.2.1.8 Access To the Pressure Suppression Chamber

Access from the reactor building to the pressure suppression chamber is provided at two locations. These are two 4-ft-diameter manhole entrances with double-gasketed bolted covers connected to the chamber by 4-ft-diameter steel pipes. These access ports are bolted closed when the primary containment is secured.

#### 6.2.1.2.1.9 Access for Refueling Operations

The head or top portion of the drywell vessel is removed during refueling operations. This head is held in place by studs and is sealed with a double seal. It is closed when the primary containment is required and is opened only when the primary coolant temperature is below 212°F and the pressure suppression system is not required to be operational.

A double seal on the head flange is provided to permit checking leaktightness after the drywell head has been replaced.

#### 6.2.1.2.1.10 Venting and Vacuum Relief System

The primary containment is designed for an external pressure of 2 psi. It can be vented through the SGTS or the reactor building ventilation system to limit pressure fluctuations caused by temperature changes during various operating modes. For normal operation, this can be accomplished through the small dedicated lines of the containment atmospheric control system that controls the venting or makeup of nitrogen. During normal operation, the primary containment is maintained at a slightly positive pressure by the Nitrogen Inerting System as described in Subsection 9.3.6.1. Containment pressure is monitored as described in Subsection 7.6.1.12.3.1. The same penetrations that are used for makeup nitrogen are also used to vent the containment for pressure control. The large ventilation purge connections are normally closed while the reactor is at a temperature greater than 212°F, except for inerting or purging. Vacuum breakers are between the drywell and the suppression chamber. Automatic vacuum relief devices are used to prevent the external primary containment pressure from exceeding the design value. The drywell vacuum relief valves draw gas from the pressure suppression chamber, and the pressure suppression chamber vacuum relief device draws air from the reactor building.

A vacuum breaker in series with an air-operated normally closed butterfly valve is used in each of two lines from the suppression chamber to the reactor building atmosphere. One valve (a pilot-operated butterfly valve) is actuated by a differential pressure signal. The second valve is a self-actuating vacuum breaker, opening at a maximum differential pressure of 0.5 psid. The valves are sized to provide sufficient mass flow rate to equalize the pressure between the suppression chamber and the reactor building in case of an inadvertent operation of the suppression chamber spray. The flow rate calculation assumed that the vacuum breaker valves failed to open until the differential pressure reached 1.0 psid. The two separate lines are redundant in that either can provide adequate venting.

The vacuum breakers connecting the suppression chamber and the drywell are sized on the basis of the pressure suppression system test program conducted for Bodega Bay at Moss Landing (Reference 11). The vacuum breaker flow area is proportional to the flow area of the vents connecting the drywell and suppression pool. Their chief purpose is to prevent excessive water-level variation in the portion of the vent discharge line that is submerged in suppression pool water. The tests relating to vacuum breaker sizing were conducted by simulating a small system rupture, which tended to cause vent water-level variation as a preliminary step in the large rupture test sequence. The vacuum breaker capacity selected on this test basis is more than adequate (typically by a factor of four) to limit the suppression chamber-drywell pressure differential during postaccident drywell cooling operations to within containment system design values.

The Fermi 2 vacuum breakers are described in Table 6.2-4. The number of suppression-chamber-to-drywell vacuum breaker valves was chosen so that 25 percent (three of 12) could fail to open and adequate venting would still be provided.

The vacuum breaker valves are provided with a magnetic latch that holds the valve disk against the seat so that vibration does not cause the valve to chatter. The close limit switches, located near the bottom of the valve body, are actuated directly by the pallet. This design allows a precise adjustment of the limit switch setpoint to a very slight opening of the pallet. The transfer point of the switch from the closed to open position is measured electrically using an ohm meter or other continuity device. With the switch properly adjusted, the maximum distance the valve may be unseated and still indicate the closed position is 0.03 in. After limit-switch adjustment, the opening gap of the pallet at the switch is verified to be less than or equal to 0.03 in. Inspection of vacuum breaker instrumentation during reactor refueling will include verification of the opening gap for switch actuation. The bypass opening for the suppression-chamber-to-drywell vacuum breaker corresponding to a 0.03-in. disk opening is 0.009 $ft^2$, well within the maximum allowable leakage area of 0.25 $ft^2$ discussed in Subsection 6.2.1.3.6.

A suppression-chamber-to-drywell vacuum breaker valve similar to the Fermi 2 vacuum breaker valves has also been tested by the Mark I Owners Group in the full-scale test facility (FSTF). During several FSTF tests, the pressure fluctuations in the vent system produced during downcomer chugging caused the vacuum breaker to cycle open and closed. The measured FSTF pressure data have been used to evaluate the expected structural performance of the Fermi 2 vacuum breaker valves. The results of this evaluation are described in the report, Mark 1 Wetwell to Drywell Differential Pressure Load and Vacuum Breaker Response for the Fermi Atomic Power Plant Unit 2, by Continuum Dynamics, Inc., submitted to the NRC by Edison letter NE-85-0707 (Reference 12).

The secondary containment to torus vacuum breaker open and closed valve disk positions are indicated by lights on the main control room panel H11-P808. The drywell-to-torus vacuum breakers are provided with open and closed position indicators on panel H11-P808, and a second set of closed position indicators on panel H11-P817. The drywell-to-torus closed indicating circuits are powered by Class 1E power supplies and are wired to meet the requirements of IEEE 279-1971.

There is no annunciation of the valve position. The position switches and circuits do not control or affect the operation of the vacuum breakers. Any single failure of the indicating circuits or switches will not prevent proper action of the vacuum breakers.

The drywell-to-torus and the secondary containment to torus vacuum breakers are equipped with pneumatic actuators operated by pushbuttons from the main control room. The purpose of these actuators is to enable verification of the operability of the vacuum breakers by observing the response of limit switches. The operability of the vacuum breakers will be verified as required by the Technical Specifications.

The actuators are sized such that they have insufficient power to open the vacuum breakers if a backflow differential pressure exists. The drywell-to-torus vacuum breakers and test actuator supports are designed to Category I criteria. The drywell-to-torus vacuum breaker nitrogen supply components downstream of the testing actuator solenoids are designed to Category II/I criteria. The drywell-to-torus vacuum breaker test actuator solenoids meet QA1 and Category I seismic requirements and are environmentally qualified because they form part of the primary containment inboard closed boundary associated with penetrations X204A – M. The secondary containment-to-torus vacuum breakers and test actuators (including actuator supports) are also designed to seismic Category I criteria.

A negative pressure analysis was performed to demonstrate the adequacy of the containment vacuum relief system (Reference 13).

The most severe negative pressures in containment would result from events that challenge the vacuum relief system. The events are associated with operation of the containment spray mode of the RHR system under accident and transient conditions which result in high depressurization rates.

The bounding accident events involve actuation of the drywell spray following a steam leak in the drywell (small-break accident) and following a DBA. All intermediate-line break events are enveloped by these cases. The inadvertent drywell spray actuation during plant operation has been evaluated. The inadvertent drywell spray scenario is an event characterized by multiple operator errors and was not part of the original License application and review. The confirmatory evaluation of this event takes credit for both reactor building to suppression chamber vacuum breakers being operable and assumes the initial drywell ambient temperature of 145°F as described in License Amendment 20. The assumed scenarios and respective bases that lead up to the initial condition for these three cases and the analysis of these three cases are described in Reference 13.

The drywell and torus pressure/temperature responses resulting from these three cases were calculated using a computer program for the calculation of mass and energy balances at successive time intervals using basic thermodynamic, flow, and ideal gas law equations.

The mass flow of spray water through each loop increases in proportion to the opening flow characteristic of the outboard drywell spray isolation valve E1150F016A(B). The model employed assumes a linear valve flow characteristic that is scaled appropriately to accurately model the actual flow as a function of valve position. A linear ramp assuming 60 sec to reach maximum flow accurately reproduces the flow characteristic for a spray isolation valve having a 98-sec open stroke time. In order to model the flow characteristic of spray isolation valves having shorter opening stroke times, the time used to calculate the linear coefficient of mass flow acceleration is based on the 60-sec value scaled by the ratio of the actual minimum value of the valve open stroke time to 98 sec.

Many conservative assumptions are made in the calculational model. The spray is not accounted for in the drywell mass balances and only serves as a heat sink. The addition of water mass to the control volume atmospheres would tend to increase pressure and some vaporization of the spray would be expected. The butterfly valve opening setpoint was arbitrarily set at 0.5 psi. The actual setpoint is 0.25 psi. Any delay in butterfly valve opening time tends to increase depressurization.

The small break accident case was determined to be the most severe of the three bounding cases considered. A resulting drywell pressure of (-1.87 psid) was predicted for this case. This value is below the design pressure for the containment structures of (-2.00) psid.

#### 6.2.1.2.2 Secondary Containment System

The reactor building completely encloses the reactor and its pressure suppression primary containment.

This building provides secondary containment when the primary containment is closed and in service, and provides primary containment when the primary containment is open, as it is during refueling. The reactor building houses the refueling and reactor servicing equipment; new- and spent-fuel storage facilities, and reactor auxiliary and service equipment, including the reactor core isolation cooling (RCIC) system; reactor cleanup demineralizer system, standby liquid control system (SLCS), CRD system equipment, emergency core cooling system (ECCS), and electrical equipment components.

The reactor building includes the "tunnel" containing the outboard main steam isolation valves (MSIVs), the main steam lines up to the turbine building, the feedwater lines, and the outboard feedwater line isolation valves. The tunnel is equipped with hinged doors which, upon pressure buildup due to a break in one of these lines, will relieve the steam pressure to the first and second floors of the turbine building. The net volume of the secondary containment is $2.8 \times 10^6$ $ft^3$.

The reactor building is a Category I structure designed and constructed in accordance with all applicable local and state building code requirements.

Substructures and exterior walls of the building up to the refueling floor consist of poured-in-place, reinforced concrete. The building structure above the refueling floor is a steel frame covered with insulated metal siding and is sealed against leakage. The building is designed for an external pressure of 0.295 psig and for low inleakage and outleakage (depending on wind conditions) during reactor operation.

6.2.1.2.2.1 Reactor Building Penetrations

Access openings for personnel and equipment are equipped with weather-strip-type seals, except for the railroad bay entry, for airtightness to meet secondary containment negative building pressure requirements. The railroad bay entry doors have inflatable seals which provide the airtightness requirements as well as site flood protection. The railroad bay rail pockets have seals which provide the airtightness requirements as well as site flood protection. Personnel entrances to the secondary containment are at the following locations:

a. The reactor core isolation cooling system/core spray pump room at Elevation 551 ft 0 in.

b. The auxiliary building basement from the CRD pump room at Elevation 551 ft 0 in.

c. Between the turbine and auxiliary building at Elevation 564 ft 0 in.

d. Outdoor entry to the reactor building at 583 ft 6in.

e. Railroad bay entry to the reactor building at 583 ft 6 in.

f. Between the reactor building and the auxiliary building at Elevation 613 ft 6 in.

g. Between the reactor building refueling floor and the auxiliary building at Elevation 684 ft 6 in.

h. Between the reactor building refueling floor and the auxiliary building at Elevation 701 ft 0 in.

All of these entries have a vestibule with double doors to maintain secondary containment integrity. The double doors are administratively controlled to prevent both doors from being open at the same time, thus maintaining secondary containment integrity. Additionally, as an administrative aid, the doors have either interlocks to prevent the opening of one door until the other door is closed or one of the doors is key locked closed. The interlock feature is not considered QA1 safety related. Failure of these interlock circuits would not cause the doors to open on their own accord. Keys for the locked closed doors are administratively controlled by the Shift Manager. In the case of the railroad bay airlock, the doors have inflatable seals which are considered active components. Therefore, to meet single failure criteria and maintain secondary containment integrity, the inner door seal is supplied from Division I of non-interruptible control air and the outer door seal is supplied from Division II of non-interruptible control air. The railroad bay airlock doors also have low seal pressure alarms which are monitored in the main control room.

Penetrations for piping and ducts are designed for leakage characteristics consistent with containment requirements for the entire building. Electrical cables and instrument leads pass through ducts sealed into the building wall.

#### 6.2.1.2.2.2 Reactor Building Ventilation Systems

The reactor building has two ventilation systems: the normal ventilation system and the SGTS. During normal power operation, shutdown, or refueling, the normal ventilation system provides outside filtered air to all levels and building equipment rooms. This system provides a minimum of one reactor building free volume change of air per hour. Air flows from the filtered supply to uncontaminated areas, to potentially contaminated areas, and then to the release vent (a short stack) on the reactor building roof.

The fans for the normal ventilation system are automatically shut down in the event a high radiation level in the building exhaust ducts is detected by the radiation monitoring system (RMS), or if there is high pressure in the drywell, low RPV water level, or high static pressure in the building, or if high radiation is detected by the east or west fuel pool radiation monitors. The normal ventilation may be isolated manually from the control room.

Shutting down the fans closes the dual ventilation duct isolation dampers. The fans are controlled from the main control room.

During emergencies when the normal ventilation system is not operating, the reactor building is ventilated through the SGTS. The SGTS filters and exhausts the atmosphere of the reactor building via the roof vent.

#### 6.2.1.2.2.3 Bypass Leakage Paths

One purpose of the secondary containment (reactor building) is to collect and filter leakage from the primary containment prior to release to the environment and thereby reduce offsite doses after a LOCA. This purpose is accomplished by

a. Minimizing reactor building leakage

b. Maintaining the reactor building at a negative pressure

c. Passing all exhaust from the reactor building through the SGTS after a LOCA.

A study has been made to evaluate the secondary containment system and determine all potential paths that could result in a fraction of the primary containment leakage going directly to the environment (i.e., without passing through the SGTS). The study encompasses three areas

a. Lines that are connected to the primary containment and pass through the secondary containment

b. Electrical penetrations

c. Reactor building leakage.

Primary Containment Lines

Lines that are connected to the primary containment and pass through the secondary containment are potential paths for leakage of radioactivity directly from the primary containment to the environment, bypassing the SGTS. The containment penetrations through which potential bypass leakage paths are possible are identified in Table 6.2-2.

All the bypass leakage paths listed in Table 6.2-2 will not contribute more leakage than 10 percent $L_A$, where $L_A$ is the maximum allowable leak rate in the Type A containment integrated leak rate test (see Subsection 6.2.4.4). The radiological impacts of MSIV leakages of up to 100 scfh per steam line, and up to 250 scfh of total MSIV leakage are analyzed separately from $L_A$ controlled leakages. Fermi 2 uses air or water sealing systems that eliminate leakage through certain valves:

a. The torus water management system suction lines (penetrations X-213A and B) are sealed with water in the torus

b. The high pressure coolant injection system suction line from suppression chamber (penetration X-225) and reactor core isolation cooling system suction line from suppression chamber (penetration X-226) are sealed with water in the torus.

The bypass leakage program will maintain a running total of leak rate measurements through all other bypass leakage paths as listed in Table 6.2-2 and will compare it with the maximum allowable. Valve maintenance will be performed when necessary.

With the exception of two leakage paths, all the valves in the bypass leakage program are containment isolation valves, and, as such, leak rates will be measured in accordance with 10 CFR 50, Appendix J, Type C tests (see Subsection 6.2.4.4). These paths accordingly are protected by redundant and diversely powered isolation valves. In the case of the reactor vessel instrument line backfill system leakage through the CRD piping when the CRD pressure is lost, certain noncontainment isolation valves are used in the program to meet criteria equivalent to those met by the other leakage paths. These valves will be tested in accordance with Section XI, Category A, of the ASME Code.

In summary, the following valves are encompassed in the bypass leakage program for Fermi 2:

| System | Valve | Test |
|---|---|---|
| Reactor Feedwater | B2100F010A | Appendix J, Type C |
| | B2100F010B | Appendix J, Type C |
| | B2100F076A | Appendix J, Type C |
| | B2100F076B | Appendix J, Type C |
| Steam line drain | B2103F016 | Appendix J, Type C |
| | B2103F019 | Appendix J, Type C |
| HPCI | E4150F006 | Appendix J, Type C |
| | E4150F002 | Appendix J, Type C |
| | E4150F003 | Appendix J, Type C |
| | E4150F600 | Appendix J, Type C |
| RCIC | E5150F013 | Appendix J, Type C |
| | E5150F007 | Appendix J, Type C |
| | E5150F008 | Appendix J, Type C |
| Drywell sumps | G1154F600 | Appendix J, Type C |
| | G1100F003 | Appendix J, Type C |
| | G1154F018 | Appendix J, Type C |
| | G1100F019 | Appendix J, Type C |
| Reactor Vessel Instrument Line Backfill | B2100F248A | Section XI, Category A |
| | B2100F248B | Section XI, Category A |
| | B2100F249A | Section XI, Category A |
| | B2100F249B | Section XI, Category A |
| Emergency Equipment Cooling Water System (EECW) | P4400F282A | Appendix J, Type C |
| | P4400F606A | Appendix J, Type C |
| | P4400F616 | Appendix J, Type C |
| | P4400F607A | Appendix J, Type C |
| | P4400F282B | Appendix J, Type C |
| | P4400F606B | Appendix J, Type C |
| | P4400F615 | Appendix J, Type C |
| | P4400F607B | Appendix J, Type C |
| Post Accident Sampling System (PASS) | P34F403A | Appendix J, Type C |
| | P34F404A | Appendix J, Type C |
| | P34F403B | Appendix J, Type C |
| | P34F404B | Appendix J, Type C |
| | P34F401A | Appendix J, Type C |
| | P34F401B | Appendix J, Type C |
| | P34F408 | Appendix J, Type C |
| | P34F410 | Appendix J, Type C |
| | P34F405B | Appendix J, Type C |
| | P34F406B | Appendix J, Type C |
| | P34F405A | Appendix J, Type C |
| | P34F406A | Appendix J, Type C |

The EECW penetrations are normally open. The listed valves are Remote Manual Isolation valves that are closed by the Operators responding to alarm response procedures. The EECW leakage detection equipment and other EECW system indications will provide the required information to the Operators. The analysis of the available sealing water in the EECW/RBCCW systems indicate that over two hours is available prior to required Operator actions to close these valves. Closure of these valves will ensure that this path will not exceed measured bypass leakage.

Leakage through the primary containment exhaust lines is collected by the SGTS and is not discharged through the exhaust fans. The large purge/inert lines and the small "on-line" pressure control lines are tied to both the reactor building ventilation system and the SGTS. High radiation in the reactor building heating, ventilation, and air conditioning (RBHVAC) exhaust isolates these valves and starts the SGTS. A suction line to the SGTS is connected to the inerting supply line as shown in Figure 9.3-14. This line collects any leakage past the containment isolation valve and processes it through the SGTS.

Category I design requirements are met (1) on the main steam piping from the reactor, up to and including the third set of MSIVs, and (2) on all branch piping, up to and including the first valve that is either normally closed or capable of automatic closure during all modes of normal nuclear steam supply system (NSSS) operation.

Electrical Penetrations

Electrical cables exit from the primary containment via penetrations sealed at both internal and external ends; the external end is within the secondary containment. The cables leaving these penetrations run in cable trays. Thus there are no electrical wiring conduits or ducts that go directly from the primary containment to the environment, bypassing the secondary containment.

Reactor Building Leakage

The reactor building, under both normal and emergency conditions, is maintained at a negative pressure so that leakage is inward. The reactor building is maintained at 0.25 in. plus or minus 0.125 in. water gage. However, due to the kinetics of gas at high velocities, the pressure on the leeward side of the building will be negative at high wind speeds. Consequently, above a threshold wind speed, air could be drawn from the reactor building, bypassing the SGTS.

An exfiltration/infiltration analysis has been made on the building to determine inward and outward leakage rates as a function of wind speed. The analysis was based on the following:

a. The SGTS maintains the building at l/4 in. $H_2O$ negative pressure

b. Leakage to the environment occurs only through the metal siding and only when the pressure differential across the siding is outward

c. The rate of leakage is 0.015 $ft^3$/minute/$ft^2$ at 1/4 in. $H_2O$ and varies as the square root of pressure differential. The leakage rate is the same for positive and negative differentials

d. The wind force acts on two sides of the building; the other two sides are at a negative pressure

e. The positive and negative pressures due to wind are based on the equation

$$P = 0.002558\ S\ (GV)^2$$

where

| | | | | |
|---|---|---|---|---|
| P | = | wind pressure (lb/ft$^2$) | | |
| S | = | shape factor | = | 0.9 windward s |
| G | = | gust factor | = | 1.1 |
| V | = | wind velocity (mph) | | |

The study shows the threshold wind velocity for any leakage outward from the building is 30 mph. The study also shows that the net leakage (inward) through the siding is not a strong function of wind velocity; consequently, the operating parameters of the SGTS are independent of wind velocity.

Since there is siding only above the refueling floor, this leakage path is not directly from the primary containment to the environment, but rather from the secondary containment the reactor building. The estimate of the fraction of primary leakage bypassing the SGTS will be conservative if this fraction is assumed to be equal to the fraction of building leakage to total discharge from the reactor building. This statement can be expressed by the following equation:

$$B = \frac{S}{S+G}$$

Where:

| | | |
|---|---|---|
| B | = | fraction of primary leakage bypassing SGTS |
| S | = | outward leakage rate of siding (function of wind speed) (scfm) |
| G | = | discharge rate of SGTS (scfm) |

The results of this study are summarized in the following table:

| | | Reactor Building Leakage* | |
|---|---|---|---|
| Wind Velocity (mph) | Fraction of Time per Year** | Total Outward From Siding (scfm) | Fraction of Primary Leakage Bypassing SGTS |
| 0 | 0.65 | 0 | 0 |
| 10 | 0.34 | 0 | 0 |
| 20 | 0.01 | 0 | 0 |
| 30 | 0.001 | 52 | 0.017 |
| 40 | -- | 246 | 0.076 |
| 50 | -- | 370 | 0.110 |

*The radiological dose from exfiltration will result in inconsequential increases, i.e., less than 1 percent, in the total calculated doses since the fraction of time the leakage occurs is so very small. In addition, if an atmospheric dispersion parameter ($\chi/Q$), which is inversely proportional to wind speed, is calculated for the higher wind speeds associated with exfiltration, it will further decrease the dose values.

**Winds of 15-minute duration as measured from the 10-m level on the 60-m tower during the 12-month period from June 1, 1974, to May 31, 1975.

### 6.2.1.3 Design Evaluation

#### 6.2.1.3.1 Introduction

In the design of the primary containment vessel, certain extreme conditions were hypothesized; the design then proceeded so that maximum stress levels under these conditions did not exceed the maximum allowable values specified in the appropriate code.

The key parameters of stress are vessel temperature, pressure, and hydrodynamic loads. The containment vessel for Fermi 2 was designed under ASME B&PV Code Section III, Nuclear Vessels (1968), including Summer 1969 Addenda. This code specifies that the internal pressure used for design conditions shall not be less than 90 percent of the maximum containment internal pressure, and that the design temperature shall not be less than the maximum containment temperature at the coincident maximum containment pressure.

The containment vent system and suppression shell, supports, internals, and attachments have been reevaluated (References 9 and 10) to include the hydrodynamic loading events and analysis methods defined by GE Topical Report NEDO-21888 (Mark I Containment Program Load Definition Report) and the NRC Safety Evaluation Report, NUREG-0661. The appropriate edition of, Section III of the ASME Code and service-level limits specified in NUREG-0661, have been applied in the reevaluation.

The maximum drywell pressure occurs during the reactor blowdown phase of a LOCA. It is dependent upon the rate at which primary system energy and fluid enter the drywell. The largest pipe in the primary coolant system is the 28-in.-diameter main recirculation line. The instantaneous guillotine rupture of this pipe is the DBA for the containment design pressure. The same pressure is conservatively used for suppression chamber design.

The most severe drywell temperature condition would occur as a result of a small primary system rupture above the reactor water level that results in the blowdown of reactor steam to the drywell. Because of the nature of the blowdown process, this would produce high-temperature steam in the drywell.

The blowdown phase of an intermediate-size break was also evaluated to demonstrate that breaks smaller than the rupture of the largest primary system pipe can be accommodated safely without any of the containment design parameters being exceeded.

In Subsections 6.2.1.3.2 through 6.2.1.3.8, the various extreme conditions that have been hypothesized and analyzed as part of the original licensing basis for the containment design are described as modified by power uprate. The initial conditions, assumptions, and break flow model applied in these analyses maximize the containment temperatures and pressures that could be expected during postulated LOCAs. The discussions of the analysis results in these subsections include the conservatively predicted short-term and long-term response of the containment. However, as part of the Fermi 2 Mark I containment long-term program (References 9 and 10), and the subsequent reevaluation of limiting events for the Power Uprate Safety Analysis (Reference 3), the spectra of postulated pipe breaks have been reinvestigated to determine the worst loading conditions for each of the affected containment structural elements. The loading conditions associated with the long-term program analyses included pool swell, condensation oscillation, chugging, and safety/relief valve discharge. To establish a conservative load basis, the initial conditions, assumptions, and models differed, in some cases, from those used in the original licensing-basis containment analyses. The load bases and application methods used in the Mark I containment analyses are completely described in Reference 5 and have been accepted by the NRC in NUREG-0661 (Reference 7). The plant-unique load definition (Reference 6) describes the pressure and temperature responses of the drywell, vent system, and suppression chamber volumes used in the Fermi 2 containment longterm program analyses. Since the long-term program-related loads occur early in the postulated LOCA events, Reference 6 only describes the short-term containment responses (less than 1100 sec). The break flow model used in the plant-unique load definition analyses is described in Reference 14.

6.2.1.3.2 Recirculation Line Break - Short-Term Response

The instantaneous guillotine rupture of a main recirculation line results in the maximum flow rate of primary system fluid and energy into the drywell. This in turn results in the maximum containment differential pressure. Figure 6.2-7 is a diagram showing the location of a recirculation line break.

Immediately following the rupture, the flow out both sides of the break will be limited to the maximum allowed by critical-flow considerations. Figure 6.2-7 shows a schematic view of the flow paths to the break. In the side adjacent to the suction nozzle, the flow will correspond to critical flow in the nozzle pipe cross section. In the side adjacent to the injection nozzle, the flow will correspond to critical flow at the 10 jet pump nozzles associated with the broken loop. In addition, there is a 4-in. cleanup line cross tie that will add to the critical flow area, yielding a total of approximately 4.1 $ft^2$.

The short-term analysis was performed for the limiting DBA/LOCA which assumes a double-ended guillotine break of a recirculation suction line that results in the maximum flow rate of primary system fluid and energy into the drywell. The analysis predicted the peak drywell pressure at 49.9 psig which is less than the containment allowable design limit of 62 psig. The peak drywell pressure of 49.9 psig is bounded by the Technical Specification value of 56.5 psig which has not been changed.

The short-term analysis covers the blowdown period during which the maximum drywell pressure and differential pressure between the drywell and wetwell occurs. The analysis assumed 102 percent power (102 percent of 3430 MWt, 3499 MWt) and was done using the M3CPT computer code which is used to model short-term containment pressure and temperature response. The M3CPT code is based on References 14 and 15 and has been reviewed and accepted by the NRC (Reference 7) during the Mark I Long Term Program (LTP) for application to the Mark I plants, including Fermi 2. The inputs for the short-term analysis (M3CPT code) are shown in Table 6.2-1, Section II.

Figure 6.2-8 shows the blowdown flow rates from the primary system to the containment. Table 6.2-5 shows the primary system energy distribution at the time of the break. (Reference 31)

The calculated primary containment pressure and temperature responses to this DBA/LOCA are shown in Figures 6.2-9 and 6.2-10.

The calculated peak drywell pressure is 49.9 psig. After the discharge of primary coolant from the RPV into the drywell, the temperature of the suppression chamber water approaches

135°F and the suppression chamber pressure stabilizes at approximately 25 psig. The drywell pressure stabilizes at a slightly higher pressure, the difference being equal to the downcomer submergence. During the RPV depressurization phase, most of the noncondensable gases in the drywell initially are forced into the suppression chamber. However, the noncondensables will redistribute between the drywell and suppression chamber via the vacuum breaker system as the drywell pressure is decreased by steam condensation.

The LPCI and/or core spray system removes decay heat and stored heat from the core, thereby controlling core heatup and limiting any metal/water reaction. The RPV is flooded to the height of the jet pump nozzles and the excess flow discharges through the recirculation line break into the drywell. This flow of water transports the core decay heat out of the RPV, through the broken recirculation line in the form of hot water that flows into the suppression chamber via the drywell-to-suppression chamber vent pipes. Steam flow is negligible. This flow, in addition to heat losses to the drywell walls, offers considerable cooling to the drywell atmosphere and causes a depressurization of the containment as the steam in the drywell is condensed.

The LPCI/RHR pumps that are used to flood the core are also used as the containment spray and cooling pumps. Prior to activation of the containment cooling mode (arbitrarily assumed at 20 minutes after the accident), all of the LPCI pump flow may be used only to flood the core. After 20 minutes, the RHR pump flow will have to be diverted from the RPV to the containment cooling mode. This is a manual operation. Actually, the containment spray need not be activated at all to keep the containment pressure below the containment peak allowable pressure. As discussed above, the peak drywell pressure is less than the containment design limit of 62 psig.

#### 6.2.1.3.3 Recirculation Line Break Long Term Response

The primary purpose of this analysis is to calculate the peak suppression pool temperature following a DBA/LOCA. The GE SUPERHEX (SHEX) code is used to predict the long-term containment response following a DBA/LOCA event.

The limiting case assumes that one RHR loop is operating in the containment cooling mode at partial pumping capacity. This includes one RHR heat exchanger, one RHR main system pump, and two service water pumps. During this mode of operation the RHR pump draws suction from the suppression pool and discharges flow through the RHR heat exchangers where it is cooled and then injected back into the suppression pool. Core cooling is provided by the core spray system and the RHR/LPCI pump prior to activation of the containment cooling mode at 20 minutes after the accident.

The long term analysis using the SHEX code with conservative input values yielded a peak post DBA/LOCA pool temperature of 196.5°F. This temperature shows margin remains to the controlling limit of 198°F which comes from NPSH requirement for pumps taking suction from the suppression pool with no credit for containment pressure per Regulatory Guide 1.1 (Subsection 6.3.2.14).

The input parameters for the long term response are shown in Table 6.2-1, Section III. Figures 6.2-11, 6,2-12, and 6.2-13 show the drywell and wetwell airspace pressure response, the drywell and wetwell airspace temperature response, and the suppression pool temperature response, respectively. The accident chronology is shown in Table 6.2-7. The conservatisms built into some of the inputs are described below.

Service Water

a. The Technical Specification limit for cooling tower reservoir temperature is 80°F. An energy balance calculation was used to determine the post-LOCA RHRSW temperature increase as a function of time from the initial condition of 80°F to the cooling tower maximum design temperature of 90°F. The temperature profile, which is non-linear, was conservatively bounded by a linear profile with the initial temperature of 80°F increasing in a linear way to 90°F over an 8-hour period. (Note: The current maximum analyzed service water supply temperature is below the assumed maximum 90°F).

b. The minimum technical specification RHR reservoir water level was used. This is conservative because it minimizes the heat capacity of the reservoir and maximizes the reservoir heatup.

c. Evaporative and drift losses were used to reduce reservoir inventory during the heatup period.

d. Complete mixing is assumed in the reservoir. This is conservative because hot water is discharged into the cooling towers and is sprayed down to the surface of the reservoir. Cooler water is drawn from the bottom of the reservoir where the pump suctions are located. No credit is taken for temperature stratification which lowers the reservoir discharge temperature profile.

Suppression Pool Volume

A pool volume of 117,161 $ft^3$ is used for the long-term containment analysis. The technical specification minimum value is 121,080 $ft^3$. This lower pool volume of 117,161 $ft^3$ adds conservatism to the calculated pool temperature, since a lower initial pool volume results in higher calculated values for pool temperature.

Initial Pool Temperature

The initial pool temperature of 95°F was used in the analysis. The 95°F is the Technical Specification limit for normal operation.

Feedwater Addition

For conservatism the analysis includes all water in the feedwater system that can contribute to higher calculated pool temperatures. This was achieved by adding all feedwater in the feedwater system during normal operation that has a temperature greater than the maximum expected pool temperature. This translates to all feedwater through feedwater heaters nos. 3, 4, 5, and 6.

In addition, a conservative calculation of the energy in the feedwater piping is added to the RPV/containment system. This water mass and energy addition assures that the pool temperature calculation conservatively reflects the effect of feedwater temperature on suppression pool temperature.

Initiation Time for Containment Cooling

The plant emergency operating procedures require that containment cooling be started for any suppression pool temperature greater than 95°F (that is within the first few minutes of a DBA/LOCA). However, the UFSAR does not take credit for operator action for the first 10 minutes into the accident. Added conservatism is built into the analysis by assuming containment cooling is initiated at 20 minutes resulting in a higher pool temperature than will be obtained with the 10 minute initiation time. Also the RHR heat exchangers providing cooling to the suppression pool water are assumed to be fouled, adding more conservatism.

Decay Heat

The decay heat based on the ANS 5.1 model (Reference 16) as described in Appendix B of Reference 17 has been used for the containment long-term analysis. This decay heat includes contributions due to fission heat induced by delayed neutrons, decay heat from fission products, decay heat from actinides (heavy elements), and decay heat from irradiated structural materials. For conservatism additional margin which corresponds to two standard deviations (10%) was added to the decay heat as described in Reference 17, Appendix B.

6.2.1.3.4 Intermediate Breaks

Intermediate breaks were not reanalyzed for power uprate since they were not the limiting case. The analysis presented below is based on the original power of 3358 MWt (102 percent of 3293 MWt).

The failure of a recirculation line results in the most severe pressure loading on the drywell structure. However, as part of the containment performance evaluation, the consequences of intermediate breaks are also analyzed. This classification covers those breaks for which operation of the ECCS will occur during the blowdown and which result in reactor depressurization. These breaks can involve either reactor steam or liquid blowdown. This section describes the consequences to the containments of a 0.l-ft$^2$ break below the RPV water level. This break area was chosen as being representative of the intermediate-break-area range. Figures 6.2-15 and 6.2-16 show the drywell and suppression chamber pressure and temperature response.

Following the 0.1-ft$^2$ break, the drywell pressure increases at 0.5 psi/sec. This drywell pressure transient is sufficiently slow so that the dynamic effect of the water in the vents is negligible and the vents will clear when the drywell-to-suppression chamber differential pressure is equal to the vent submergence pressure. For this containment design, the distance between the pool surface and the bottom of the vents is 3 ft 4 in. maximum. Thus, the water level in the vent will reach this point when the drywell-to-suppression chamber pressure differential reaches 1.5 psi, i.e., approximately 3 sec after the 0.1-ft$^2$ break occurs. At this time, air, steam, and water will start to flow from the drywell to the suppression pool; the steam will be condensed and the air will enter the suppression chamber free space. After 3 sec there will be a constant pressure differential between the drywell and the suppression chamber. The continual purging of drywell air to the suppression chamber will result in a gradual pressurization of the latter. By approximately 300 sec, all the drywell air will have been swept over to the suppression chamber and the pressure increase terminated. After this time, the drywell and wetwell pressures will remain relatively constant and all the steam being released to the drywell will be condensing in the pool. Some continuing containment pressurization will occur because of the continued pool heatup. The ECCS will be initiated by the 0.1-$ft^2$ break via high drywell pressure and will provide emergency cooling of the core. The operation of these systems is such that the reactor will be depressurized in approximately 600 sec.

This will terminate the blowdown phases of the transient. The drywell will be at approximately 25 psig and the suppression chamber at approximately 23 psig.

In addition, the suppression pool temperature will be the same as from the recirculating line break because essentially the same amount of primary system energy would be released during the blowdown. After reactor depressurization, the flow through the break will condense the drywell steam and will eventually cause the drywell and suppression chamber pressures to equalize in the same manner as following a recirculation line rupture.

The subsequent long-term suppression pool and containment heatup transient that follows is essentially the same as for the recirculation break without containment spray.

From this description, it can be concluded that the consequences of an intermediate break are less severe than a recirculation line rupture over short time periods and essentially the same over a long time period.

Additionally, as discussed in Subsection 6.2.1.3.1, the containment response due to intermediate breaks has also been calculated using the bases provided in References 5 and 7. The corresponding short term containment response is reported in Reference 6. These predicted results also support the above conclusions.

#### 6.2.1.3.5 Small Breaks

Small breaks were not reanalyzed for power uprate since they were not the limiting case. The analysis presented below is based on the original power of 3358 Mwt (102 percent of 3293 MWt).

This subsection discusses the containment transient associated with small primary system blowdowns. The sizes of primary system blowdowns in this category are those blowdowns that will not result in reactor depressurization due either to loss of reactor fluid or automatic operation of the ECCS equipment. The underlying assumption is that, following the manifestation of a break of this size, the reactor operators will initiate an orderly shutdown and depressurization of the plant.

The thermodynamic process associated with the blowdown of primary system fluid is one of constant enthalpy. If the primary system break is below the water level, the blowdown flow will consist of reactor water. Upon depressurizing from reactor pressure to the drywell pressure, approximately one-third of this water will flash to steam and two-thirds will remain as liquid. Both phases will be at saturated conditions corresponding to the drywell pressure. Thus, if the drywell is at atmospheric pressure, the steam and liquid associated with a liquid blowdown would be at 212°F. Similarly, if the containment is assumed to be at its maximum allowable pressure, the reactor liquid would blow down to approximately 309°F steam and water. If the primary system rupture is located so that the blowdown flow consists of reactor steam, the resultant steam temperature in the containment is significantly higher than the temperature associated with liquid blowdown. This is because a constant enthalpy decompression of high-pressure, saturated steam will result in a superheat condition. For example, decompression of 1000 psia steam to atmospheric pressure will result in 298°F superheated steam (86°F of superheat).

The conclusion is that a small reactor steam leak will impose the most severe temperature conditions on the drywell structures and the safety equipment in the drywell. The superheat temperature for large steam-only blowdowns would be the same as for small breaks, but the duration of the high temperature condition would be less. This is because the larger breaks will depressurize the reactor more rapidly than the orderly reactor shutdown that is assumed to terminate the small break.

For drywell design evaluation, the following sequence of events was assumed to occur. With the reactor and containment operating at the maximum normal conditions defined in Table 6.2-1, a small break occurs that allows blowdown of reactor steam to the drywell. The resulting pressure increase in the drywell will lead to a high-drywell-pressure signal that will scram the reactor and activate the containment isolation system. The drywell pressure will continue to increase at a rate dependent upon the size of the assumed steam leak. This pressure increase will depress the water level in the vents until the level reaches the bottom of the vents. At this time, air and steam will start to enter the suppression pool. The steam will be condensed and the air will pass to the suppression chamber free space. The latter will result in a gradual pressurization of the containment at a rate dependent upon the air carryover rate. Eventually, the entrainment of the drywell air in the steam flow through the vents will result in all the drywell air being carried over to the suppression chamber. At this time, pressurization of the containment will cease and the system will reach an equilibrium condition with the drywell pressure at 25 psig and the suppression chamber at approximately 23 psig. The drywell will be full of superheated steam. Continued blowdown of reactor steam will be condensed in the pool.

The reactor operators will be alerted to the incident by the high-drywell-pressure signal and the reactor scram. For the purposes of evaluating the duration of the superheat condition in the drywell, it is assumed that their response is to shut the reactor down in an orderly manner using the safety/relief valves, or main condenser, and limiting the reactor cooldown rate to 100°F per hour. This will result in the reactor primary system being depressurized within 6 hr. At this time, the blowdown flow to the drywell will cease and the superheat condition will be terminated. If the plant operators elect to cool down and depressurize the reactor primary system more rapidly than at 100°F per hour, then the drywell superheat condition will be shorter.

The temperature resulting from the blowdown is determined by finding the combination of primary system pressure and containment pressure that produces the maximum superheat temperature. These are 450 psia, 35 psig, and 340°F, respectively. This temperature is assumed to exist for the initial 3 hr of the blowdown.

Additionally, as described in Subsection 6.2.1.3.1, the containment response due to small breaks has also been calculated using the bases provided in References 5 and 7. The corresponding short-term response is reported in Reference 6. Assumed operator actions that will minimize cyclic loads on suppression chamber and vent system structures are discussed in Reference 9.

#### 6.2.1.3.6 Steam Bypass

The Fermi 2 containment has been examined to determine what leakage between the drywell and suppression chamber can be tolerated as a function of primary system break area; i.e., what leakage will result in a peak pressure equal to the maximum allowable pressure for the system. For this calculation, the following assumptions were made:

a. Flow through the postulated leakage path is pure steam. For a given leakage path, postulating that the leakage flow consisted of a mixture of liquid and vapor would increase the total leakage mass flow rate but would decrease the steam flow rate. Since it is the steam entering the suppression chamber free space that is resulting in the containment pressurization, this is a conservative assumption

b. There is no condensation of the leakage flow on either the suppression pool surface or the torus and vent system structures. Since any condensation results in less steam being in the suppression chamber free space, this is a conservative assumption. In practice, there would be condensation, especially for the larger primary system breaks when there will be vigorous agitation at the pool surface during blowdown.

Leakage capacity is expressed in terms of A, the area of the leakage flow path, and K, the geometric loss coefficient. These terms are interrelated such that the allowable leakage capacity for a system is expressed in units of $A/\sqrt{K}$.

The calculation shows that the limiting leakage capacity occurs for a primary system break area of 0.4 $ft^2$. For this break area, the allowable leakage capacity is 0.147. Typically, the geometric loss factor K would be three or greater; thus, the maximum allowable leakage area would be about 0.25 $ft^2$. This corresponds to a 7-in. line.

Primary system breaks greater than about 0.4 $ft^2$ will result in rapid system depressurization, and, for the given primary allowable leakage area, would result in the containment pressure being less than the maximum allowable pressure at the end of the reactor blowdown period.

Primary system breaks less than about 0.4 $ft^2$ will not result in rapid primary system depressurization and some operator action is required to terminate the pressure rise in the containment. The operators have several options available to them. If the source of the leakage is undefined, they would probably depressurize the primary system via either the main condenser or relief valves, or they could activate the suppression chamber or drywell sprays.

#### 6.2.1.3.7 Small-Break Temperature Consideration

The Fermi 2 containment vessel was designed in accordance with the ASME B&PV Code Section III, Nuclear Vessels (1968), including the Summer 1969 Addenda. The primary containment design parameters, as shown in Part I of Table 6.2-1, were chosen on the basis of conditions discussed in the Fermi 2 PSAR. The design-basis conditions have since changed, as discussed in Subsection 6.2.1.3.2. However, no change in design pressure was necessary.

A small steam leak inside the primary containment, followed by an orderly shutdown and RPV depressurization, presents a different drywell atmosphere temperature transient. This situation is discussed in Subsection 6.2.1.3.5. The drywell temperature is calculated to be 340°F for 3 hr, and 320°F for 6 hr. During this period the calculated maximum drywell pressure is 35 psig, and during the following 24-hr period the temperature is 250°F with a pressure maximum of 25 psig. The containment vendor has analyzed the containment capability and found it adequate for these conditions.

#### 6.2.1.3.8 Line Breaks in Sacrificial Shield Annulus

##### 6.2.1.3.8.1 Description of System Configuration

The sacrificial shield is approximately 49-ft high cylindrical shell, with a 25 ft 7 in. inside diameter, a 29 ft 1 in. outside diameter, and a thickness of 1 ft 9 1/4 in. It has steel liners on its exterior and interior surfaces, and is meridionally stiffened by 12 vertical steel columns. The steel liner plates are welded to the columns, and the annular space between these plates is filled with concrete. The wall is rigidly attached to the reactor support at the bottom and attached to the drywell and RPV at the top by means of stiff leg supports and snubbers, respectively. The RPV sits inside the sacrificial shield with annular clearance of approximately 18 in. Of this 18 in., approximately 3 in. is occupied by insulation and 3 in. by a ventilation space between the shield and the insulation. This leaves a 12-in. annular space between the insulation and the RPV.

A detailed description of the sacrificial shield is given in Subsections 3.8.3.1.1 and 3.8.3.3.1. Openings are provided in the shield for the passage of lines from the RPV to the drywell. Those openings which lie within an area 9 ft above and 16 ft below the centerline of the core are required to be shielded and are equipped with shielding doors; these doors are locked closed and will not open during a pipe break within the annulus. The openings above and below this band have no shielding requirements; they are covered with a light-weight rupture diaphragm designed to help relieve the annulus pressure should a break occur.

The nozzles of the RPV are connected to the main piping using a short transition piece called a safe-end. The postulated break is the weld at either end of the safe-end. There are 26 penetrations in the wall, of which 17 occur where shield doors are required. Of these 17 lines, the major ones are the two 28-in. diameter recirculation outlet lines and the ten 12-in. recirculation inlet lines. The safe-end welds for these nozzles lie within the thickness of the shield wall or in the annular space. These two sets of lines were considered the critical cases, because the rest of the lines either are smaller, or may vent directly to the drywell because of the absence of any shield doors. One more case was considered: the feedwater line safe- end break. Because this line is located at the top of the sacrificial shield, forces generated during a postulated line break have a large moment which can lead to high stresses. The analysis requires modeling the system to predict what forces and pressures are generated following a postulated failure of safe- ends from these three lines and then using these in a structural design assessment.

Subsection 3.9.1.5 presents the GE analysis of the loads on the reactor vessel and internals due to a line break in the sacrificial shield annulus. Part of that work also includes computation of forces and moments for the RPV pedestal, RPV anchor bolts, and stabilizer truss. For these components, Sargent & Lundy used the larger of the stresses computed from their analysis or the GE analysis. This procedure has been incorporated in Revision 2 to SL-3647, dated March 14, 1980 (Reference 18).

The conclusion of Reference 18 states that the existing design of the sacrificial shield, reactor pedestal, stabilizer truss, RPV, and shield anchor bolts can safely accommodate the effects of annulus pressurization resulting from a postulated safe-end break.

6.2.1.3.8.2 Summary of Study

A study was performed in 1973 using state-of-the-art methods. A detailed report of that study was filed with the AEC in response to Open Item No. 12 and Question 12.4, Amendment 11 of the PSAR (Refer to Reference 15 in Section 3.8). During the review of the original FSAR, the NRC questioned whether certain aspects of the model used to predict the pressure distribution were adequately conservative and requested that the calculation be repeated using models currently available.

The recalculation was broken down into three tasks: calculation of mass energy release, calculation of annulus pressure distribution history, and the structural design assessment.

Mass Energy Release

This task was performed using a method developed by GE. The method assumes that the initial fluid velocity is zero. After the break, a finite time is required to accelerate the fluid to steady-state velocities; this is called the inventory period. The flow rate during this period is computed by two methods; one includes the effect of inventory and subcooling on flow in the pipe, the other accounts for the finite break opening time. The smaller of the two flow rates at any time is used. Both methods produce maximum flow rates based on different limiting areas. The transfer from one curve to the other represents a change in the point where the flow is choked. Following the inventory period, the flow is assumed to be choked at the limiting cross-sectional flow area. Mass flux is calculated using the Moody steady-slip flow model with subcooling. Results of this calculation are in Reference 20.

Annulus Pressurization

The computation of pressure distribution in the annulus following these breaks was based on the use of the computer code COMPARE. The model for Fermi 2 used 42 nodes in the annulus and four nodes in the drywell. The analysis considered movement of insulation at penetrations and the resulting venting of fluid to the drywell. The code was modified to account for variable junction area as a function of time. A Moody multiplier of 0.6 was used for all junctions except that from the break to the annulus, where 1.0 was used. All junctions had an inertia term, and sub-critical flow was calculated on the basis of a solution to the momentum equation with constant density. Reference 20 is the report of this work.

Structural Design Assessment

The structure was analyzed using the Sargent & Lundy thin-shell- of-revolution computer code, DYMAX. The Fermi 2 model for this study consisted of 76 nodes. Reference 18 is the report of this work.

The loads included in the study were:

a. Annulus pressurization

b. Jet impingement

c. Pipe-whip reaction

d. Dead load

e. Thermal effect due to accident

f. Seismic effect due to operating-basis earthquake (OBE) and safe-shutdown earthquake (SSE).

The structural components assessed were:

a. Sacrificial shield

b. Reactor pedestal

c. Stabilizer truss

d. Reactor anchor bolts

e. Sacrificial shield anchor bolts.

### 6.2.1.4 Inspection and Testing

#### 6.2.1.4.1 Primary Containment

The Fermi 2 containment has been designed and constructed as a Class B vessel in compliance with Section III of the ASME Code, 1968 Edition, including the Summer 1969 Addenda. The containment vent system and suppression shell, supports, internals, and attachments have also been reevaluated (References 9 and 10) to include the hydrodynamic loading events and analysis methods defined by GE Topical Report NEDO-21888 (Mark I Containment Program Load Definition Report) and the NRC Safety Evaluation Report, NUREG-0661. The appropriate edition of Section III of the ASME Code and service-level limits specified in NUREG-0661 have been applied in the reevaluation. All inspections and tests prescribed by these editions of the Code have been successfully completed.

Containment boundary integrity has been verified during the construction of the Fermi 2 plant using the reference-vessel method. This method involves measuring the pressure differential between the containment vessel and a reference system of copper vessels that are interconnected with copper tubing and located in the upper and lower portions of the drywell and in the suppression chamber. This initial test, begun March 3, 1973, was performed in accordance with 10 CFR 50, Appendix J, and ANSI N45.4-1972, "Leakage Rate Testing of Containment Structures for Nuclear Reactors." The leak rate was determined to be 0.079 ±0.035 percent per 24 hr at 56 psig.

A preoperational, integrated leak-rate test using the absolute method was performed at the peak containment pressure calculated from DBA considerations in accordance with 10 CFR 50, Appendix J. This integrated leak-rate test was a Type A test. The test was conducted over a minimum of 8 hr with at least 20 value data points.

Type A tests will be performed periodically throughout the life of the plant in accordance with Fermi 2 Technical Specifications.

Permanently installed piping penetrations are provided through the containment structure for both the compressor system and the pressure-indication piping required for the Type A tests. An electrical penetration is provided for the leads to the temperature instrumentation in the containment.

Containment atmosphere-circulation fans are operable during the elevated-pressure conditions of the Type A tests to minimize local variations in temperature and humidity.

Personnel entry into the containment is not required during the Type A tests. Therefore, no provisions have been made for this kind of activity.

6.2.1.4.1.1 Penetrations

Airlock doors, access hatches, and the drywell head are equipped with double seals and instrument taps which permit pressurization of the space between them to verify seal integrity.

Piping penetrations with bellows seals which allow relative movement between pipe and containment wall are provided with double bellows and a space between them which can be pressurized. These penetrations are equipped with test fittings necessary to facilitate pressurization and testing of the penetration boundaries without pressurizing the entire containment.

Electrical penetrations are equipped with double seals and test connections and are capable of being tested at containment design pressure without pressurization of the containment.

6.2.1.4.1.2 Isolation Valves

Tests will be performed on isolation valves including reactor-building-to-torus vacuum breakers to verify their operability, pressure boundary integrity, and seat-and-stem leaktightness. Design provisions have been made when possible to accommodate the specific leak-test requirements of 10 CFR 50, Appendix J, Type C tests. Alternative methods are used where necessary and technically justifiable. The alternative methods are identified and discussed in Table 6.2-2.

Edison performed opening force tests on the Fermi 2 vacuum breakers during preoperational testing and will include these tests in the inservice testing. With the valve air cylinder properly adjusted using a predetermined air pressure, the pallet will close smoothly, without banging, in 2 to 5 sec. The closing time of the pallet is measured and set as part of the normal valve inspection and adjustment during refueling. This testing, along with the opening force measurement, provides assurance that the pallet is not binding and the valve will open with the proper opening time.

6.2.1.4.1.3 Pressure Suppression System

Drywell-to-suppression-chamber gross-leak tests will be conducted periodically as defined in the Technical Specifications to ensure that bypass of the pressure suppression feature of the containment has not developed. The test will be based on determination of the rate of change of pressure in the suppression chamber and drywell at a drywell-to-suppression-chamber differential pressure of 1 psi. In addition, individual drywell-to-torus vacuum breakers will be inspected and their position-switch setting will be verified. During plant operation, these valves will be periodically exercised to verify the operability of the valve and the closed-position instrumentation. These tests are documented in the Technical Specifications.

6.2.1.4.1.4 Test Frequencies

Test frequencies for the Type A test will be in accordance with Fermi 2 Technical Specifications. Type B and C test frequencies are based on the requirements of 10 CFR 50, Appendix J, Option B.

Data, data reduction, and test acceptability requirements for all tests are described in ANSI/ANS 56.8-2002 or other alternative testing methods that have been approved by the NRC and are based on the requirements of 10 CFR 50, Appendix J, Option B.

If the result of any test indicates that leakage exceeds the limits established in the Technical Specifications, repairs shall be made and a retest performed. In addition, for unsuccessful Type A tests, the provisions of 10 CFR 50, Appendix J, Option B, shall apply.

6.2.1.4.2 Secondary Containment

The reactor building leakage rate may be tested by complete isolation of the building except for the effluent from the SGTS. The SGTS is placed in operation and the system will maintain a constant flow. The building inleakage is small enough to ensure that the building negative pressure exceeds the value required by the Technical Specifications. The rate at which air is exhausted through the system is an accurate measure of building inleakage.

Visual inspection of reactor building penetrations will be possible. Penetration leakage is determined as a part of the gross reactor building inleakage as discussed above.

Frequency of these tests and inspections as defined in the Technical Specifications is based upon expected lifetime of the various seals, components, and penetrations and anticipated failure modes. The test and inspection schedule is intended to ensure that gross failures do not occur and that such failures, should they occur, are discovered and corrected within a reasonable time.

6.2.1.5 Instrumentation

6.2.1.5.1 Primary Containment

The primary containment monitoring system is designed to make available to the plant operators sufficient information to permit normal operation, to assist the operator in assessing the consequences of an accident or an incident, and to determine the effectiveness of control actions taken to mitigate the effects of the postulated event.

Functions of the primary containment monitoring system include multipoint measurement and recording of hydrogen and oxygen concentrations, gaseous radiation levels, pressure, temperatures, and water levels in the drywell and pressure suppression chamber. Suppression chamber water temperatures and drywell vessel wall and atmospheric temperatures are also measured and recorded. This system provides information for operator control of suppression pool cooling. Details of the primary containment monitoring system, its subsystems, sensors, and logic are described in Subsections 7.1.2 and 7.6.1.

Radiation monitors and pressure transmitters and the logic associated with the initiation of primary containment isolation as well as actuation of the ECCS and other engineered safety feature (ESF) systems are described in Subsections 7.1.2 and 7.3.2. The primary containment high-range radiation monitors are discussed in Subsection 11.4.3.

#### 6.2.1.5.2 Secondary Containment

Secondary containment pressure is normally controlled by the reactor/auxiliary building ventilation system. Pressure sensors outside the building are arranged so that the lowest pressure on the building (due to wind) is compared with the building internal pressure which is maintained at 0.25 in. of water below the lowest outside pressure. The building fans are shut down in the event that a differential pressure of approximately ±2 in. occurs. Time-delay relays prevent spurious shutdown of the ventilation system caused by wind gusts.

The secondary containment is isolated on the same signals that actuate the SGTS; i.e., high drywell pressure, level 2 low reactor water level, reactor building ventilation exhaust radioactivity high, fuel pool area ventilation exhaust radioactivity high, or a manual pushbutton in the control room.

The SGTS is also actuated, and the secondary containment isolated, upon Loss of Offsite Power (LOOP). A LOOP causes a failure of the radiation monitors located in the reactor building ventilation exhaust system and in the fuel pool ventilation exhaust system which initiates a downscale trip signal. The radiation monitors' downscale trip signal isolates the reactor building ventilation (RBHVAC) exhaust system and initiates the SGTS system.

The systems whose signals initiate secondary containment isolation are discussed in Subsections 7.3.2 and 7.6.1.

### 6.2.1.6 Materials

Organic materials used in the Fermi 2 primary and secondary containments have been selected for extended life during normal operation and for resistance to expected accident environmental conditions. Thermal insulations used are inorganic and are not sensitive to high radiation fields, steam, or high temperature.

Table 6.2-8 lists the type of protective coatings used, their thicknesses, and their locations within the primary and secondary containments.

Table 6.2-9 lists organic materials used for wiring insulation in the primary and secondary containments.

Table 6.2-10 lists other organic materials of significant quantity and the amounts used in the primary and secondary containments.

Evaluations of these materials have been made. It has been determined that they will satisfactorily endure accident environmental conditions and that their expected products of decomposition, if any, will not adversely affect the operability of any ESF system.

The following paragraphs describe the coatings and paint used within the primary containment, including pertinent information regarding the following:

a. Identification of material used, location, and function

b. Physical and chemical characteristics

c. Performance under accident conditions including washdown, radiation, steam, temperature, and jet impingement effects

d. Data on effect of any coating material that may be dissolved or carried by the fluids that flow in the spray systems of the ECCS that may affect the functioning of the systems

e. Effect of coating on core and heat exchanger heat-transfer surfaces

f. Clogging and other effects on fluid flows in Class 1 systems from coatings.

Additional information is available in Reference 21.

<u>Reactor Vessel Support Pedestal</u>

The inside and outside surfaces of the reactor vessel support pedestal are coated with Ameron Nu-klad surfacer 110 AA primer and one finish coat of Ameron polyamide epoxy 66. Damaged areas of Ameron Nu-klad 110 AA are repaired with Ameron Nu-klad 111. The function of this coating system is to protect and seal the pedestal surfaces against attack by either demineralized (aggressive) water or radiation contamination and to facilitate washdown.

The physical and chemical characteristics of the Ameron Nu-klad surfacer 110 AA primer are excellent adhesion to clean concrete and good adhesion to steel, resistance to attack by demineralized water or hot condensate, excellent abrasion resistance, considerable radiation resistance, excellent chemical resistance, and indefinite repairability. Both primer and finish are modified epoxy. Ameron polyamide epoxy 66 has properties similar to those of the primer. Both coatings withstand temperatures to 200°F continuously and to 300°F intermittently.

Required DBA testing has been performed, and the coating system is capable of withstanding the rigors of a LOCA. A washdown removes contamination.

The coating effect on the core and heat-transfer surfaces is negligible because the coating system is nonleachable.

No clogging or other effects on fluid flow in Class 1 systems are expected since the coating is nonleachable and has excellent adhesion.

The Ameron 66 top coat has been applied in accordance with the recommendations of Regulatory Guide 1.54 and ANSI N101.4 and the coating system has met the pull-test requirements of ANSI N5.12. The coating of the reactor vessel support pedestal and other concrete surfaces of the drywell have been designated as a QA Level 1, safety-related activity. The coating system as described above is a qualified coating.

Drywell Concrete Floors and Walls

The concrete surfaces of the drywell floors and walls are coated with Ameron Nu-klad surfacer 110 AA primer and a top coat of Ameron polyamide epoxy 66. The function, physical and chemical characteristics, and other properties of this coating are discussed under "Reactor Vessel Support Pedestal" above.

Sacrificial Shield Wall

The exterior surface of the sacrificial shield is coated with Carboline Carbozinc 11 and repaired with Carboline Carbozinc 11 SG. This is a self-curing, zinc-filled, inorganic, two-part basic zinc silicate complex that readily accepts top coats. The function of this coating is to provide long-term protection against corrosion, attack by radiation or radioactive water, and to facilitate washdown.

The physical characteristics are a hard surface resistant to aggressive water, very good impact resistance, and a temperature use range up to 750°F continuous and 800°F intermittent. Flexibility is fair. Chemical characteristics are insolubility in water and resistance to aggressive water and solvents. Relatively wide application temperatures (0-200°F) and humidity ranges (to 95 percent) are permissible.

Contaminants on the coated surface can be easily washed down with water. The coating has high radiation resistance, resists steam to 180°F, and has excellent temperature resistance up to 750°F.

The coating has no effect on core heat transfer or heat exchanger heat-transfer surfaces since it is not soluble.

Carbozinc 11 and Carbozinc 11 SG coatings have been subjected to extensive DBA testing for a variety of application techniques and were found acceptable for use in BWR environments under LOCA conditions.

Report DECO 12 2191 notes that some particle separation could occur under accident conditions in areas subjected to continuous scouring by water and steam spray. Such scouring would occur only in the immediate vicinity of a pipe break. In such areas, the coating is not lost in large flakes, however, but rather in particles less than 20 microns in size. ECCS suction strainer head loss calculations include the recommended Utility Resolution Guidance (NEDO-32686) for qualified paint assumed to degrade from a direct steam jet impingement.

Most of the initial Carbozinc 11 coatings in the primary containment were applied in accordance with the original 1969 specification, prior to the issuance of Regulatory Guide 1.54 and ANSI N101.4. The industry standard at that time was to apply Carbozinc 11 in accordance with the manufacturer's recommendations. This type of coating has been successfully used in operating BWRs and for years has withstood a variety of adverse conditions.

In 1984, the commercial name of the Carbozinc 11 coating was changed to Carbozinc 11 SG. Consequently, in cases where repairs to the original Carbozinc 11 coating were needed after 1984, Carbozinc 11 SG was used.

Drywell Interior Steel

All exposed interior surfaces of the drywell pressure boundary, including the drywell jet deflectors and surfaces in contact with concrete, are coated with Carboline Carbozinc 11 and repaired with Carbozinc 11 SG. The function of this coating system is to protect the surfaces from corrosion, from attack by aggressive water, radioactive water, or radiation, and to facilitate washdown.

Those coatings which cover the drywell pressure boundary are maintained under Fermi 2 QA Level I criteria to ensure long-term corrosion protection for the pressure boundary. This coating is not considered to be in full compliance with ANSI N101.4.

Drywell Interior Structural Steel

The primary structural steel within the drywell is coated with Carboline Carbozinc 11 and repaired with Carboline Carbozinc 11 SG. The purpose of the coating is to provide long-term protection against corrosion and to facilitate washdown.

The Drywell dado region was recoated with Carboline Carboguard 890 N, and is classified as an acceptable coating (as defined in ASTM D4538).

Substantial modifications were made to the primary structural members in two separate phases due to load reevaluations that resulted in varying degrees of surface preparation. Welding and nondestructive examination procedures necessitated removing existing coatings at tie-ins and welded connections. Due to completed installation of equipment, generally very tight working quarters, and complex components placement, sandblasting and recoating of steel members were not routinely completed.

Surfaces of Suppression Chamber

The interior surfaces of the suppression chamber, including the exterior surfaces of the downcomers and vent header, the exterior surfaces of the vent pipes, vent header supports, ring girders, catwalks, monorail, stiffeners, supporting steel, piping, hangers, and penetration nozzles, are coated with the Wisconsin Protective Coating Plasite 7155 system above elevation 558’-2” and with the Carboline Carboguard 6250 N system below elevation 558’-2”. The Carboguard coating overlaps the Plasite coating at the intersection of the coatings. The interior surface of the downcomers is coated with Plasite 7155. The Plasite coating is a water-resistant phenolic coating cross-linked with epoxy resin and polymerized with an alkaline curing agent. The Carboguard coating is a solventless epoxy novolac coating designed to handle exposures inside nuclear containment facilities. The function of these coatings is to provide long-term protection from corrosion and radiation, and to facilitate washdown.

These coatings resist temperatures up to 400°F intermittently, develop good hardness and abrasion resistance, can withstand cyclic thermal shock, and provide a broad range of long-term chemical resistance.

The Plasite coating was applied in accordance with Regulatory Guide 1.54, ANSI N101.4, meets pull-test requirements of ANSI N5.12, Section 6.4, has been DBA tested, and is considered a fully qualified coating capable of withstanding accident conditions. Its application is a safety-related, QA Level 1 activity.

The Carboguard tie-in band that overlaps the Plasite coating is considered unqualified. There are additional minor areas in the wetwell with unqualified Carboguard coating. The unqualified coatings are tracked as indicated in Table 6.2-8. The unqualified coating amounts have been evaluated and are within established limits for unqualified coatings inside containment. The remainder of the Carboguard 6250 N coating is in accordance with Regulatory Guide 1.54 and ANSI N101.4 except that later ASTM standards endorsed by Revision 2 of Regulatory Guide 1.54 were used for test panel preparation, radiation qualification testing, chemical resistance qualification testing, and test panel evaluation. The coating, except for the tie-in band, is considered a fully qualified coating capable of withstanding accident conditions. Its application is a safety-related, QA Level 1 activity.

The interior surface of the vents from the drywell shell down to a transition point approximately 20 in. from the vent header is coated with Carboline Carbozinc 11 SG coating system. The remainder of the interior surface of the vents (from the transition point to the vent header) and the interior of the vent header are coated with a qualified Carboline Carboguard 6250 N coating system. A small qualified overlap band of Carboguard 6250 N over Carbozinc 11 SG exists at the transition point between the two coating systems. The interior surface of the vacuum breaker extensions and downcomers are coated with Plasite 7155 coating system described above in this section. The Carboline Carboguard 6250 N coating overlaps the Plasite 7155 coating on the inside surface of each downcomer and vacuum breaker penetration in the vent header. The Carboguard 6250 N coating in this overlap band is classified as an acceptable coating (as defined in ASTM D4538).

Touch-up repairs to the suppression chamber interior coating under submerged or dry conditions are made using compatible safety-related coatings complying with the original requirements and standards.

Miscellaneous Coatings

Coatings on miscellaneous equipment and components in the drywell are discussed below. These coatings were included in the evaluation of the Fermi 2 primary containment coatings, and will not impair plant operation under normal or abnormal conditions.

a. Galvanized Surfaces

   The drywell cooling system ducting and dampers are completely galvanized without any further coatings. At welded joints, the galvanized surface was ground off to clean metal, and in some locations these ground areas were touched up with Galvanox I or Galvanox V, zinc-rich coatings similar in properties to Carboline Carbozinc 11. In addition, all electrical conduit, terminal boxes, cable trays, and supporting unistruts are galvanized. The only exceptions are some large flexible conduits made of stainless steel

b. Hangers and Supports

   Hanger and support components, including clamps, rods, spring cans, snubber attachments, pipe-whip restraint components, and secondary support steel, were originally coated with Carboline Carbozinc 11. Significant changes in the hanger and support design resulted in addition of secondary support steel, change-out of hanger components, and welding of attachments. Coating repair and touch-up of these areas is not safety related

c. Piping

Most of the piping within the drywell is insulated with reflective metallic insulation panels (Mirror Insulation), consisting of removable sections and having an outer cover of stainless steel. Encapsulated NUKON or encapsulated silicon (Min-K) is used where clearance restrictions exist, i.e., drywell penetrations and spaces between pipe whip restraints and pipe. Normally, cold fluid system piping is not insulated or coated. The uninsulated carbon steel piping was shop coated with a protective varnish. Tight mill scale and some rust is apparent on the piping surfaces. The varnish and mill scale are considered unqualified coatings

d. Unidentified and Unqualified Coatings

These coatings consist largely of manufacturer's shop coatings and primers such as red lead, aluminum base, enamels, polymer, and phenolic paints. These coatings are present on valve bodies, yokes and bonnets, motor and air operators, handwheels, electric motors, etc. Another category of unqualified coatings consists of identification marking and banding of electrical conduit, terminal boxes, and trays.

Coatings of this category that have thicknesses of 3 mils or less are postulated to fail in small particles and will not clog strainers.

Unqualified coatings greater than 3 mils DFT have either been removed, and the surfaces have been recoated with Carbozinc 11 where appropriate (see Reference 21 for additional information); or have been evaluated for use in the primary containment. Design calculations have been prepared to evaluate the addition of materials to the primary containment. These are updated as necessary as part of the plant's response to NRC Generic Letter 98-04.

## 6.2.2 Primary Containment Heat Removal System

### 6.2.2.1 Design Bases

Containment heat removal is provided by operating the RHR system in the suppression pool cooling mode or the containment spray mode. The system meets the following safety design bases:

a. The source of coolant inventory shall be located within the containment so as to establish a closed cooling water path

b. A closed-loop flow path between the suppression pool and the RHR heat exchangers shall be established so that the heat-removal capability of these heat exchangers can be utilized

c. This system, in conjunction with other ESF systems, shall have diversity and redundancy such that no single failure can result in its inability to cool the containment adequately

d. Each active component shall be testable during operation of the nuclear system. Testing is described in Section 5.5.7.5.

### 6.2.2.2 System Design

The containment cooling subsystem is an integral part of the RHR system, as described in Subsection 5.5.7. Redundancy is achieved by having two complete containment cooling systems.

Consideration of the fouling of heat exchangers and the selection of temperatures for heat exchanger design is discussed in Subsection 5.5.7.

### 6.2.2.3 Design Evaluation

The discussion in this subsection has been updated for power uprate conditions.

In the event of the postulated LOCA, the short-term energy release from the reactor primary system will be dumped to the suppression pool. Subsequent to the accident, fission product decay heat will result in a continuing energy dump to the pool. Unless this energy is removed from the containment system, it will eventually result in unacceptable suppression pool temperatures and containment pressures. The containment cooling mode of the RHR system is used to remove heat from the suppression pool, the suppression chamber, and the drywell.

When the RHR system is in the containment cooling mode, the pumps draw water from the suppression pool, pass it through the RHR heat exchangers, and inject it back into the suppression pool or into the containment via sprays.

The adequacy of the RHR system has been evaluated considering, two sequences of events with different assumed single active failures. Both scenarios assume the occurrence of a LOCA coincident with a loss of offsite power with the reactor initially at maximum power. The original licensing and design basis scenario assumes a loss of offsite power occurs and the single failure of one divisional power supply for the duration of the accident. Immediately following the accident, the ECCS initiates automatically as designed in response to the accident initiation signals.

Under the original scenario, due to the assumed loss of offsite power and one division of onsite power, two core spray pumps and two RHR pumps will be operating. (Section 6.3 describes the ECCS equipment.) Twenty minutes later the plant operators activate one RHR heat exchanger in order to start containment heat removal. This involves shutting down one of the two LPCI pumps and starting up the service water pumps for the heat exchanger. Once containment cooling has been established (including RHR cooling towers), no further operator actions are required.

Subsequent to the original plant analysis it was determined that a single failure of an RHRSW isolation valve to open would result in the same available suppression pool cooling capability (namely one RHR heat exchanger) but would result in additional operating ECCS pumps – four RHR pumps and four core spray pumps; thus, resulting in additional ECCS pump heat to the suppression pool. Consistent with the original containment analysis, this scenario assumes plant operators activate the remaining operable RHR heat exchanger and the associated division of the Ultimate Heat Sink [See Section 9.2.5] twenty minutes after the initiating event.

The evaluations of both scenarios use the SUPERHEX (SHEX) code to calculate the long-term containment response for Fermi 2 with power uprate (Reference 22). SUPERHEX evolved from two previously approved codes (Reference 14 and 15) and was shown to give equivalent pool temperature response to the predecessor code. The long-term analysis for Fermi 2 with the SUPERHEX computer code using conservative inputs yields a peak post DBA/LOCA pool temperature of 196.5°F. This temperature shows margin remains to the controlling limit of 198°F which comes from NPSH requirement for pumps taking suction from the suppression pool with no credit for containment pressure per Regulatory Guide 1.1.

The input parameters used for SUPERHEX for the long-term containment response analyses for Fermi 2 with power uprate are identified in Table 6.2-1.

<u>Service Water Temperature</u>

The original containment analysis used a constant RHR service water (RHRSW) temperature of 90°F which is the maximum design cooling tower outlet temperature. The Technical Specifications prohibit operation with the cooling tower reservoir temperature above 80°F. An energy balance calculation was used to determine the post LOCA RHRSW temperature increase as a function of time from the initial condition of 80°F to the cooling tower maximum design temperature of 90°F. The temperature profile, which is non-linear, was conservatively bounded by a linear profile which was used in the containment analysis (Table 6.2-1). The following are the important assumptions used in the energy balance. (Note: The current maximum analyzed service water supply temperature is below the assumed maximum 90°F).

a. The maximum Technical Specification reservoir temperature of 80°F was used as an initial condition.

b. The maximum design cooling tower outlet temperature of 90°F was used.

c. The minimum Technical Specification RHR reservoir water level was used. This is conservative because it minimizes the heat capacity of the reservoir and maximizes the reservoir heatup.

d. Evaporative and drift losses were used to reduce reservoir inventory during the heatup period.

e. Complete mixing was assumed in the reservoir. This is conservative because hot water is discharged into the cooling towers and is sprayed down to the surface of the reservoir. Cooler water is drawn from the bottom of the reservoir where the pump suctions are located. No credit was taken for temperature stratification which would have lowered the reservoir discharge temperature profile.

<u>Suppression Pool Volume</u>

The initial suppression pool volume used for the power uprate long-term containment analysis was 117,161 $ft^3$ which is less than the pool volume of 121,080 $ft^3$ that corresponds to the Technical Specification minimum value. The lower pool volume of 117,161 $ft^3$ adds conservatism to the calculated pool temperature since a lower initial pool volume results in higher calculated values for pool temperature.

Initial Pool Temperature

The initial pool temperature for the containment analysis was set at 95°F which is the Technical Specification limit for normal operation.

Feedwater Addition

All water in the feedwater system which could contribute to higher calculated pool temperatures was added to the RPV and containment system for the power uprate analysis. This was achieved by adding all feedwater which is in the feedwater system during normal operation that has a temperature greater than the maximum expected pool temperature. This translates to all feedwater through Feedwater Heaters Nos. 3, 4, 5, and 6.

In addition, a conservative calculation of the energy in the feedwater piping is added to the RPV/containment system. This water mass and energy addition assures that the pool temperature calculation conservatively reflects the effect of feedwater addition on suppression pool temperature.

Initiation Time for Containment Cooling

The long-term containment response analysis has assumed that the containment cooling is initiated at twenty minutes.

Decay Heat

The original analysis identified decay heat values used for the long-term containment analysis which correspond to the May-Witt decay heat model values after 60 seconds. For the power uprate analysis a more realistic decay heat has been included. This decay heat is based on the ANS 5.1 model (Reference 16) and is described in Appendix B of Reference 17. This decay heat includes contributions due to fission heat induced by delayed neutrons, decay heat from fission products, decay heat from actinides (heavy elements), and decay heat from irradiated structural materials. For conservatism additional margin which corresponds to two standard deviations (10%) was added on the decay heat as described in Reference 17, Appendix B, for the Fermi 2 long-term containment power uprate analysis.

Suppression Pool Temperature Response

The suppression pool temperature response has also been evaluated following several other plant transient events in which steam is discharged to the suppression pool. General Electric Report NEDC-24388-P (Reference 23) describes transient events, the assumed RHR system modes of operation, and the predicted pool temperature results. The report concludes that the peak pool temperatures in the vicinity of SRV discharge quencher devices are below the limit established to ensure stable steam condensation.

### 6.2.2.4 Testing and Inspections

The preoperational and operational testing and the periodic inspection of components of the containment heat removal system are described in Subsection 5.5.7.5.

### 6.2.2.5 Instrumentation Requirements

The containment spray and the suppression pool cooling modes of the RHR system are manually initiated. Once initiated, containment cooling performance is monitored by monitoring pump performance, flow and pressure, and coolant temperature.

### 6.2.2.6 Materials

Materials used are reviewed and evaluated with regard to radiolytic and pyrolytic decomposition and attendant effects on safe operation of the system. For example, fluorocarbon plastic (Teflon) is not permitted in environments that attain temperatures greater than 300°F, or radiation exposures above $10^4$ rads. Only inorganic thermal insulation, which does not decompose due to radiation or temperature, is used in these environments. An inorganic zinc primer is used on all exterior surfaces of carbon steel components that are treated. All paints used are suitable for the temperature conditions expected.

## 6.2.3 Secondary Containment Air Purification and Cleanup System

The SGTS is designed to minimize the release-related offsite dose rates by permitting the venting and purging of both the primary and the secondary containment atmospheres under accident or abnormal conditions, and at the same time containing any airborne particulate or halogen contamination that might be present.

### 6.2.3.1 Design Bases

Under postaccident conditions, it is possible that the primary containment atmosphere could become contaminated with radioactive particulates and halogens. Any air from this volume finding its way to the secondary containment is therefore likely to be similarly contaminated. The SGTS is designed to permit controlled ventilation of this area by maintaining it under slightly negative pressure with respect to the outside atmosphere to ensure that any air leaving is filtered to remove particulates and halogens. The system is also capable of filtering gases exhausted from the primary containment and the HPCI barometric condenser. The system is designed to function under postaccident conditions of high radiation levels, temperatures, and relative humidity.

The SGTS flow rate is sufficient to provide a secondary containment air volume change at least once per day and to maintain the reactor building at approximately negative 1/4-in. water pressure for accident and abnormal conditions.

Particulate- and halogen-removal capability permits venting of the primary and secondary containment volumes following an accident while maintaining offsite dose rates well within the guidelines set by 10 CFR 100. For those design basis accidents reanalyzed per Regulatory Guide 1.183, SGTS limits offsite dose within the limits of 10 CFR 50.67.

The SGTS is designed to operate with influent air temperatures up to 135°F and relative humidity up to 100 percent. The system is periodically tested such that a decontamination efficiency of 99 percent can be assumed for removal of all forms of gaseous and particulate iodine. System retention capacity, originally based on the requirements of Regulatory Guide 1.3 and TID-14844, and amounts of up to 1300 gm (Reference 24), is currently evaluated against the 2.5 mg/g Regulatory Guide 1.52 limit for 30-day post-accident iodine accumulation based on the Regulatory Guide 1.183 Alternative Source Term. (Reference 26)

The SGTS is a Quality Level I, Category I ESF system meeting all applicable portions of IEEE 344; IEEE 308; ORNL-NSIC 65; UC-80 Reactor Technology, "Design, Construction and Testing of High Efficiency Air Filtration Systems for Nuclear Application;" ASME B&PV Code Section IX, "Welding Qualifications" (1971); Air Moving and Conditioning Association (AMCA), "Standard Test Code for Air Moving Devices" and "Standards Handbook;" and Savannah River Laboratory Report DP-812, "Application of Moisture Separators and Particulate Filters in Reactor Containment."

The SGTS meets the intent and functional objective requirements of Regulatory Guide 1.52. Some detail design requirements of this guide, however, are not met because system fabrication was commenced before the guide was issued. All areas of noncompliance have been reviewed and in each case it has been determined that design and hardware changes required to bring these areas into compliance would not improve the system performance or capability to meet the design objectives. (See Appendix A, Subsection A.1.52.)

### 6.2.3.2 System Design

The SGTS is a 100 percent-redundant ESF system and is shown schematically in Figure 6.2-20. Major system components are listed and briefly described in Table 6.2-11. The system is designed to meet reactor building containment tests.

The SGTS consists of two separate and parallel 100 percent capacity trains. In addition to its associated ducts, controls, instrumentation, isolation valves, and protection systems, each train consists of the following items listed sequentially and in the direction of air flow:

a. A moisture separator to remove entrained water droplets, thus minimizing water loading of the prefilter. The moisture separator meets design requirements specified in Savannah River Laboratory Report DP-812

b. A prefilter to reduce the loading on the absolute filter. The prefilter is fire resistant and capable of operation at temperatures up to 250°F

c. An electric heater to reduce the relative humidity of the influent air to 70 percent or less under the "worstcase" conditions

d. A high-efficiency particulate air (HEPA) filter with a design DOP filtration efficiency of 99.97 percent for particles 0.3 μm in diameter or larger. Four parallel filter elements, each rated at 1000 scfm, are provided. These elements meet the intent of Military Specification MIL-F-51068-C. They are Underwriters Laboratories (UL) approved, fire resistant, and suitable for service under the temperatures, mass, and heat loading expected. The filters are mounted and sealed in a welded steel frame to ensure against possible bypass flow. The filters are tested periodically for bypass leakage such that a 99 percent decontamination efficiency can be assumed for removal of particulate iodine

e. A deep-bed, gasketless, all-welded construction adsorber containing activated carbon

f. A HEPA filter identical to the one described above to trap charcoal fines and decay daughters entrained by the air stream

g. An exhaust fan designed for 4000 $ft^3$/minute

h. A cooling air fan installed in parallel with the exhaust fan, designed and built to the same standards and codes as the exhaust fan. The purpose of this blower is to provide cooling air flow to the charcoal filter in order to maintain charcoal temperature below 310°F under design loading conditions, in the event of high charcoal adsorber bed temperature.

Piping connections and valving exist between the SGTS and the secondary containment building ventilation system, the primary containment drywell, the suppression chamber, and the HPCI turbine barometric condenser vacuum pump discharge.

When the cooling air fan is in use, suction is taken from a roof vent. Discharge under both modes is to a vent located on the reactor building roof.

Full access and interior compartment lights with external light switches are provided for the spaces between filter train components where required to facilitate inspection, testing, and replacement of components.

Injection nozzles, sample points, and pressure taps are provided to facilitate periodic inservice inspection tests.

### 6.2.3.3 Design Evaluation

#### 6.2.3.3.1 General

The SGTS is designed to permit controlled venting of the primary or secondary containment following an accident or abnormal occurrence which might cause abnormally high airborne contamination in these areas.

Achievement of acceptable offsite dose rates following a DBA depends on the proper functioning of the SGTS. Therefore, the system, along with its power supplies and surrounding structures, has been designed to meet ESF system standards. All necessary equipment and surrounding structures are of Category I design. The equipment is powered from essential buses which will supply power to the SGTS in the event of a loss of offsite power. All power and control circuits meet the requirements of IEEE 279. Redundant active components are provided where necessary to ensure that a single failure does not impair or prevent system operation. An SGTS failure analysis is presented in Table 6.2-12.

The SGTS removal efficiency was successfully tested (Reference 24) for radioactive and nonradioactive forms of iodine and for particulate matter 0.3 mm or larger. The thyroid dose at the site boundary and low-population zone has been calculated on the basis of iodine-removal efficiency of 99 percent. Credit for 99 percent removal efficiency is dependent on in-place testing per Regulatory Guide 1.52, as stated in the Technical Specifications.

### 6.2.3.3.2 Secondary Containment Pressurization During Design Basis LOCA

The pressure of the secondary containment volume after a LOCA has been studied. The analysis included infiltration and thermal loads from the primary containment, operating equipment, and emergency lighting.

The SGTS is designed to maintain a secondary containment pressure of -0.25 in. of water, thus ensuring that any airborne radioactive material in the secondary containment is not released to the surrounding atmosphere without passing through the SGTS filters. In the event of a design-basis LOCA, loss of offsite power is assumed; consequently, there is a delay from the start of the event to the activation of the SGTS and the emergency area coolers.

During the delay, the secondary containment pressure increases because of heat generated by emergency equipment and other sources. Upon initiation of the SGTS and emergency area coolers, a short time is required to reduce the secondary containment pressure to a negative pressure at or below -0.25 in. of water.

The purpose of the calculation was to generate the secondary containment pressure response during a design-basis LOCA and to determine the period of time when the secondary containment pressure is above -0.25 in. of water. The method of analysis and the assumptions and results are described in the following paragraphs.

Method of Analysis and Assumptions

The computer code GOTHIC (Reference 25) was used to generate the secondary containment pressure response.

All major assumptions are given below:

a. No credit was taken for exfiltration from the secondary containment

b. Infiltration to the secondary containment was included in the pressure response analysis

c. No heat transfer was allowed to the outdoor atmosphere

d. Heat transfer to interior secondary containment walls, floors, and ceilings was included

e. Heat transfer from the torus room to the secondary containment is based on flow through the pressure relieving doors in the corner room basement walls

f. Only one SGTS filter train is available with a minimum volumetric flow rate of 3800 cfm

g. Offsite power is lost at the start of the design-basis LOCA event

h. The activation of the SGTS is delayed by 33 sec and the activation of the emergency area coolers is delayed by 38 sec (see Table 8.3-5)

i. The RHR, core spray, and RCIC pump rooms in the reactor building subbasement are treated separately from the main secondary containment volume. These rooms have their own emergency coolers to handle emergency equipment and lighting heat loads. Because the heat loads and cooling are confined to partially enclosed volumes at the very bottom of the secondary containment, the area coolers will absorb the heat loads within the confines of the corner rooms

j. The heat loads from the RHR, core spray, and RCIC pump rooms will not affect the main secondary containment volume before the initiation of the area coolers. The RHR pumps are activated 13 sec after the start of the design-basis LOCA event (see Table 8.3-5). The emergency coolers are activated at 38 sec. For the heat loads to affect the main volume, the pumps, piping, and subsequently the corner room atmospheres must heat up. After the corner room atmospheres have heated up, the only mode of heat transfer to the main volume is natural convection. Considering that natural convection is a rather slow process, no significant heat transfer to the main secondary containment volume from the corner rooms is expected during the 25 sec from the initiation of the RHR pumps to the initiation of emergency cooling

k. An outdoor temperature of 95°F was used in the analysis

l. The reactor building closed cooling water system is inoperable and both divisions of the emergency equipment cooling water system are operating

m. All ECCS equipment starts

n. The fuel pool cooling and cleanup system, the reactor water cleanup system, and the recirculation pump motor-generator set cooling system are shut down

o. The fuel pool is at an operating temperature of 125°F.

   Any increase in fuel pool temperature in the range of 125°F to 130°F will have negligible effects on the results of the analysis

p. An initial secondary containment pressure of 0.0 in. water gage was assumed.

Results

The secondary containment response due to a design-basis LOCA is shown in Figure 6.2-21. During the first 33 sec, the pressure increases to a slightly positive value. With the activation of the SGTS at 33 sec and the activation of the area coolers at 38 sec, the pressure decreases slightly.

At approximately 50 seconds, pressure-relieving doors on the common wall between the torus room and the corner rooms open and allow heated torus room air to enter the rest of the secondary containment. This step input of heat into the secondary containment appears as a sharp pressure spike in Figure 6.2-21.

The pressure then decreases past -0.25 in. of water to a steady-state secondary containment pressure. Less than 1020 sec elapses from the start of the design-basis LOCA event to the point where the secondary containment pressure decreases to and subsequently stays below -0.25 in. of water. For conservatism, the 1020 sec (17 minutes) is maintained for the LOCA dose assessment (Subsection 15.6.5.5.2).

6.2.3.4 Tests and Inspections

The SGTS and its components are thoroughly tested in a program consisting of the following classifications:

a. Predelivery and component qualification tests

b. Onsite preoperational acceptance tests

c. Operational surveillance tests.

Written test procedures establish acceptance criteria for all test results. Operational test results are recorded and compared with previous performance records, thus enabling early prediction of end of component life and appropriate corrective action.

For the various components of the system, the following predelivery qualification tests were applied:

a. Equipment Train Housing - Leak tests at +20 in. of water internal pressure. Magnetic-particle or liquid-penetrant testing of all welds and discontinuities which could cause bypass leakage around the HEPA filters or adsorber beds

b. Demister - Qualification test or objective evidence to demonstrate compliance with requirements specified in Savannah River Laboratory Report DP-812

c. HEPA Filters - Qualification test to demonstrate a minimum of 99.97 percent efficiency when measured using a 0.3 mm DOP aerosol in conformance with MIL-STD-282

d. HEPA Filter Frames - Soap-bubble leak test across filterless covered bank

e. Adsorber Beds - Available objective evidence demonstrates acceptable flow-pressure characteristics and channeling effects

f. Adsorbent -

   1. Ignition test

   2. Methyl iodide removal test

   3. Hardness test

   4. Impregnant content test.

To demonstrate the integrity of the potassium iodide impregnated charcoal, required factory tests have been performed by the manufacturer prior to acceptance

g. Fans - Fan tests in accordance with the latest revision of AMCA Standard 210, "Air Moving and Conditioning Association Test Code for Air Moving Devices," to establish characteristic curves

h. Prefilter - Objective evidence and certification that NBS efficiency specified is attained

i. Valves, Dampers, and Actuators - Shop tests demonstrating seal effectiveness and ability to perform intended functions under the anticipated conditions.

Onsite preoperational tests for the SGTS are listed in Subsection 14.1.3.2.47.

Onsite periodic testing will be performed. Items such as design conditions of flow, drawdown time, and differential pressure will be verified during these routine tests performed in compliance with the Technical Specifications.

### 6.2.3.5 Instrumentation and Controls

Each SGTS unit and its controls, power supplies, valves, dampers, and auxiliary equipment are designed and installed so that they are both physically and electrically independent. The system conforms to single-failure criteria outlined in IEEE 279.

A separate control system is provided for each SGTS unit, including all items necessary for control and for determining the status of all components. The SGTS instrumentation is presented in Figure 6.2-20, a brief summary of which is presented below.

Differential pressure indicators are provided to measure the pressure drop across each filter and charcoal bed. Differential pressure switches are provided to signal abnormal conditions.

Each adsorber bed is equipped with the following controls:

a. Charcoal adsorber bed high-temperature-detection temperature element to actuate $CO_2$ injection

b. Charcoal adsorber bed overheat temperature element to actuate standby cooling fan

c. Charcoal adsorber bed temperature controller to operate dryer (heater).

Fire protection for the adsorber bed is provided by a $CO_2$ system which is actuated automatically by adsorber bed high temperature. Actuation of the system is signaled in the main control room.

Every isolation valve is supplied with position switches to provide positive indication of valve status.

High-temperature cutouts are provided as an integral part of the single-stage electric heaters. Local temperature indication is provided upstream and downstream of the electric heaters.

Flow signals are transmitted to the main control board for indication and record. The flow transmitter directly controls the flow-control valve.

Manual switches are provided on the main control panel for each fan.

A continuous isokinetic sample is taken from the discharge of the operating filter train and processed through radiation detectors, a particulate filter, and an adsorber bed, and is returned to the SGTS roof vent.

High radiation levels are indicated by audible and visible alarms in the main control room.

The SGTS electrical equipment and instrumentation required to function in a postaccident harsh environment are environmentally qualified and in compliance with NUREG-0588.

When the SGTS filter units are shut down (auto standby mode with no actuation signal), all valves are closed and exhaust fans are deactivated. The charcoal adsorber blanket heater may or may not be on, depending on the charcoal temperature.

Standby cooling fans and associated valves will automatically be activated on high charcoal bed temperature. The system is actuated and put into service automatically in response to any one of the following signals:

a. Auto standby mode

1. High drywell pressure
2. Low reactor water level
3. Reactor building ventilation exhaust radioactivity high
4. Fuel pool area ventilation exhaust radioactivity high.

b. Manual mode.

On actuation in the auto standby mode, both trains are started. The SGTS can be manually started by placing the control switch for the selected train in the run position.

The exhaust fans start, associated isolation valves open, normal reactor building ventilation system is tripped, and valves are automatically realigned to exhaust into the SGTS.

Adsorber-blanket heaters are automatically shut down if they were operating prior to system startup.

Activation of the SGTS in either mode is accompanied by audible and visible alarms in the main control room. The operator would then manually shut down one of the operating trains, leaving the other to perform as intended.

In the event of failure of the operating train for any reason, that train would be shut down and isolated by the operator in the main control room, and the redundant filter train would manually be put into service.

Main control room visible and audible alarms include the following:

a. Both SGTS trains "Auto Start"
b. High relative humidity ahead of charcoal bed
c. Low system flow rate (interlocked with primary blower "Run" signal)
d. High airborne contamination at the roof vent
e. Failure of either train to start up and operate on signal
f. Cooling fan "Auto Start"
g. Carbon dioxide fire protection system actuation.

Functions that can be accomplished manually from the main control room include the following:

a. Startup or shutdown of either or both SGTS trains
b. Startup of alternative SGTS train upon failure of operating train
c. Startup or shutdown of either standby cooling air fan

d. Isolation of SGTS train upon manual shutdown command from main control room.

Automatic functions include the following:

a. Startup of both SGTS trains and proper alignment of isolation valves

b. Activation of adsorber heater on low temperature

c. Startup of standby cooling air fan system when adsorber temperature exceeds its setpoint

d. $CO_2$ injection on high adsorber bed temperature

e. $CO_2$ shutoff when adsorber bed temperature is below its setpoint.

6.2.3.6 Materials

Materials for fabrication, coating, and sealing the SGTS are chosen because of their capability for a satisfactory normal service life of 40 years, and 6 months of service under post-LOCA conditions at the maximum cumulative radiation exposure, without any adverse effects on service, performance, operation, or appearance. All materials of construction, including metal components, seals, gaskets, lubricants, and finishes, such as paints, are compatible with these objectives and are capable of satisfactory service under the expected radiation exposure.

Gaskets and seal pads are unicellular, ozone-resistant, oil-resistant neoprene or silicone-rubber sponge, Grade SCE-43, in accordance with ASTM Dl056.

Only adhesives listed and approved in AEC Health and Safety Bulletin 306, dated March 31, 1971, or Military Specification MIL-F-5l068C, dated June 8, 1970, are used.

Organic compounds included in the filter train are as follows.

a. Charcoal

b. HEPA filter media binder - The total weight of media binder per filter element is approximately 4 lb, or a total of 32 lb per equipment train

c. Filter adhesive - Approximately 1 liquid qt of fire-retardant neoprene adhesive is used to manufacture each HEPA filter

d. HEPA filter, pre-filter and coverplate gaskets - Filter and coverplate gaskets are unicellular neoprene per ASTM Dl056, Grade SCE-43

e. Door and access port gaskets - Door and access port gaskets are unicellular neoprene per ASTM D735-SCE-516, or ASTM D2000-BC-516

f. All painted metals (inside and out) are coated with 0.003-in. MOBIL 13R56B primer and 0.003-in. MOBIL VALCHEM Series 89 white top coat. Stainless components are not painted

g. Wire Coatings and Insulation - Approximately 15 lb of Cerro Products "Rockbestos" silicone rubber is used. Of this amount, less than 0.5 lb is inside the SGTS. Approximately 10 lb of EPR neoprene is used, none of which is inside the SGTS.

## 6.2.4 Containment Isolation System

### 6.2.4.1 Design Bases

The containment isolation system consists of valves and controls required for the isolation of lines penetrating the primary containment. The primary objective of this system is to provide protection against release of radioactive materials to the environment as a result of accidents occurring to the nuclear steam supply system (NSSS), auxiliary systems, and support systems. This objective is accomplished by automatic isolation of appropriate lines that penetrate the primary containment. The containment isolation system is actuated automatically when specific limits are reached.

The containment isolation system, in general, closes fluid penetrations that support those systems not required for emergency operation. Fluid penetrations supporting ESF systems have remote manual isolation valves which may be closed from the main control room, if necessary. The automatic isolation valves close on receipt of an isolation signal from a sensor. For example, the main steam isolation valves (MSIVs) may be closed by signals indicating low water level in the reactor, main steam line tunnel high temperature, high steam flow, low steam line pressure, or low condenser vacuum. Isolation signals for each valve are specified in Table 6.2-2.

It is neither necessary nor desirable that every isolation valve close simultaneously with a common isolation signal. For example, if a process pipe were to rupture in the drywell, it would be important to close all lines that are open to the drywell and some effluent process lines, such as the main steam lines. However, under these conditions it would be essential that the containment and core cooling systems be operable. Therefore, several specific signals are used for isolation of various process and safety systems.

The design of isolation valving for lines penetrating the containment conforms to the intent of 10 CFR 50, Appendix A, General Design Criteria (GDC) 54, 55, 56, and 57. Redundancy and physical separation are provided in the electrical and mechanical design to ensure that no single failure in the containment isolation system prevents the system from performing its intended functions.

Where a penetration is part of a redundant train in an ESF system, isolation valves for that train may receive power from a single electrical division. This is desirable so that a single failure of an electrical division cannot disable both trains of the ESF system. In these cases a redundant mechanical barrier (i.e., closed systems beyond the isolation valves) exists so that containment isolation is not lost as a result of a single electrical failure.

Protection of primary containment isolation system components from missiles, and the integrity of these components to withstand seismic occurrences without loss of operability, was considered in the design of this system. The containment isolation system is Category I.

On signals of high drywell pressure or low water level in the reactor vessel all isolation valves that are part of systems not required for emergency shutdown of the plant are closed. The same signals initiate the operation of systems associated with the emergency core cooling system (ECCS). Isolation valves that are part of the ECCS may be closed remote manually from the control room.

Criteria for the design of the containment isolation control system are listed in Subsection 7.1.2.1.2. The bases for assigning certain signals for primary containment isolation are listed and explained in Chapter 7.

### 6.2.4.2 System Design

The containment isolation system is designed to provide a minimum of one protective barrier between the reactor and the environs under all postulated conditions. A detailed discussion of the controls associated with the containment isolation system is included in Subsection 7.3.2. Table 6.2-2 specifies the plant protection system signals that initiate closure of the containment isolation valves.

#### 6.2.4.2.1 Design Requirements

Containment isolation valves were designed in accordance with the requirements of the ASME B&PV Code Section III, in effect at the time of purchase as required by 10 CFR 50, Section 50.55. Where necessary, a dynamic system analysis, which includes the impact effect of rapid valve closure under operating conditions, is included in the design specifications of piping systems that require containment isolation valves. Quality Assurance (QA) procedures are followed to ensure compliance with these specifications.

All containment isolation valves are located inside either the drywell or the secondary containment. Both structures are of Category I design and are protected against damage from missiles. The primary containment vessel is enclosed completely in a reinforced-concrete structure having a thickness of 4 to 7 ft. This concrete structure provides a major mechanical barrier for protection against missiles that may be generated external to the primary containment. Protection against damage from missiles is provided for isolation valves, actuators, and controls. Refer to Section 3.5 for a discussion of missile protection. Section 3.6 contains a discussion of protection provided against the dynamic effects of pipe whip, while Section 3.7 contains a discussion of the design analyses performed on containment penetration piping.

Each containment isolation valve is designed to ensure its performance under all anticipated environmental conditions including maximum differential pressure, extreme seismic occurrences, steam-laden atmosphere, high temperature, and high humidity. Section 3.11 presents a discussion of the environmental conditions, both normal and accident, for which the containment isolation system is designed.

Closed systems used as an isolation barrier, either inside or outside the primary containment, meet the following requirements:

a. The systems are protected against postulated missiles and pipe whip

b. The systems are designed to Category I

c. The systems are at least Quality Group B, except for specific instrument line applications noted in Table 6.2-2 (Note 12)

d. The systems are designed to at least the maximum temperature and pressure of the containment.

In addition, closed systems inside the containment meet the following requirements:

a. They are designed to withstand external pressure from the containment structural acceptance test

b. They are designed to withstand the design-basis accident and accompanying environment

c. They do not communicate with either the reactor coolant system or the containment atmosphere.

Power-operated containment isolation valves have limit switches that indicate valve position in the main control room. Containment isolation valves are designed to fail in the safe position. Containment isolation valves are either automatically actuated by the signals shown in Table 6.2-2 or are remote manually operated. Some containment isolation check valves inside containment are provided with supplemental air operators to verify free disk movement during opening and closing and zero pressure differentials across the valves. This arrangement provides a means by which to periodically verify valve operability.

Containment isolation valves that are remote manually operated are required to be provided with a leakage detection capability or be administratively closed (Standard Review Plan [SRP] 6.2.4). Table 6.2-13 lists the remote manual containment isolation valves that have a leak detection capability.

Remote manual containment isolation valves that are locked closed (and are thus under administrative control) are as follows.

| Penetration | Valve |
|---|---|
| X-12 | V8-3407 |
| X-21 | V5-2006 |
| | V5-2007 |

The only other containment isolation valves with a remote manual primary actuation mode are the $N_2$ supply to the drywell-to-torus vacuum breakers, penetrations X-204A-M (valves V4-2036, V4-2065, V4-2075, V4-2077, V4-2082, V4-2084, V4-2086, V4-2088, V4-2090, V4-2092, V4-2094, and V4-2096). (Table 6.2-2 provides the F valve numbers.) These valves are locked closed to comply with Technical Specification 3.6.1.3.2 and are opened during the testing of the drywell-to-torus vacuum breakers. These valves are under administrative control and considered locked closed as defined in SRP 6.2.4 to preclude the possibility of their being inadvertently opened during normal reactor operations. Thus, as all remote manual containment isolation valves are either provided with leak detection capability or locked closed, Fermi 2 meets the guidance set forth in SRP 6.2.4.

6.2.4.2.2 Conformance To General Design Criteria

As stated in Subsection 6.2.4.1, the design of isolation valving for lines penetrating the containment follows the intent of GDC 54 through 57. Isolation valving for instrument lines that penetrate the containment follows the guidance of Regulatory Guide 1.11. Those cases where literal interpretation of GDC 54 through 57 has not been followed are included in the discussions in the following subsections.

6.2.4.2.2.1 General Design Criterion 54

General Design Criterion 54 in 10 CFR 50 states

> Piping systems penetrating primary reactor containment shall be provided with leak detection, isolation, and containment capabilities having redundancy, reliability, and performance capabilities which reflect the importance to safety of isolating these piping systems. Such piping systems shall be designed with a capability to test periodically the operability of the isolation valves and associated apparatus and to determine if valve leakage is within acceptable limits.

Criterion 54 Conformance

All piping penetrations meet the intent of GDC 55, 56, or 57. In doing so, they also conform to the intent of GDC 54 to the extent that all piping systems penetrating the primary containment are provided with leak detection, isolation, and containment capabilities which reflect the importance to safety of isolating these piping systems. In addition, each piping penetration is designed to be tested periodically in accordance with 10 CFR 50, Appendix J, as described in Table 6.2-2. Specifically, the following systems have containment isolation provisions consistent with the provisions of GDC 54.

Traversing In-Core Probe (TIP) System (Penetrations X-35B, C, D, E, F

The TIP system detector signal and drive cable neither comprise a portion of the reactor coolant pressure boundary nor directly communicate with the primary containment atmosphere. Thus, GDC 55 and 56 are not directly applicable to this specific class of lines. The basis on which TIP system lines are designed is described more closely in GDC 54, which states, in effect, that systems penetrating the primary containment are to be provided with isolation capabilities commensurate with the importance of isolating the system. Thus, even though the failure of TIP system lines presents no safety hazard, additional conservatism is provided in TIP system isolation capabilities, which reflects the intent of GDC 55.

The TIP system detector signal and drive cable are stored outside the primary containment behind a normally closed ball valve and an explosively actuated shear valve. The valves are located outside the containment for inspection and maintenance accessibility, and the position of each is indicated in the control center. The ball valve remains closed at all times except during operation of the associated TIP system channel. Prior to use of the TIP system, the ball valve is manually opened. All five TIP machines may be used simultaneously, however any one guide tube is used, at most, only a few hours per year.

After TIP system cable retraction, the ball valve is manually closed. Should a containment isolation signal be received while the TIP system cable is inserted, the cable will withdraw automatically, and this will be followed by automatic closure of the ball valve.

The function of the shear valve is to ensure the integrity of the containment in the unlikely event that the ball valve should fail to close or the drive cable should fail to retract from the guide tube during the time containment isolation is required. The valve is designed to shear the TIP drive cable and seal the drive tube upon command from the control center. In addition to valve position, the condition of each shear valve dc firing circuit is monitored continuously in the control center. Additional testing requirements are discussed in Note 17 to Table 6.2-2.

Control Rod Drive Insert and Withdrawal Lines (Penetrations X-37A, B, C, D and X-38A, B, C, D)

Control rod drive (CRD) insert and withdrawal lines penetrate the primary containment, but they neither directly communicate with the containment atmosphere nor comprise part of the reactor coolant pressure boundary. Thus, GDC 55 and 56 are not directly applicable to this class of lines. The basis on which the CRD lines are designed is described more closely in GDC 54, which requires such systems to have isolation capabilities commensurate with the importance of isolating the system. Since these lines are necessary for the scram function, the reliability of their operation is of utmost concern. Thus, isolation valves should not be incorporated in the design of this system. The probability of reliable and timely operation is enhanced by simplicity of design and by minimizing, where possible, the introduction of possible failure mechanisms. Even though multiple breaks postulated and analyzed in Section 4.0 pose no threat to public health and safety, CRD insert and withdrawal isolation capabilities were designed to reflect the conservative intent of GDC 55.

Both the CRD insert and withdrawal lines are provided with normally closed, fail-closed, solenoid-operated directional control valves, which open only during routine movement of their associated control rod. The normally closed, fail-open, air-operated scram inlet and exhaust valves open only when required to effect a rapid reactor shutdown (scram). In addition, manual shutoff valves are provided for positive isolation in the unlikely event of a pipe break within a hydraulic control unit. (These units and the valves described above are located outside the containment to satisfy testing, inspection, and maintenance requirements.) In addition, each CRD insert line is provided with an automatically actuated flange ball check valve inside containment; the flange ball check valve is part of the CRD mechanism.

During post-LOCA, the scram inlet and outlet valves will remain open if the scram cannot be reset. Therefore, due to CRD seal leakage, the scram discharge volume (SDV) could experience reactor vessel pressure. To ensure the integrity of the SDV, it will be included in the Type A tests.

#### 6.2.4.2.2.2 General Design Criterion 55

General Design Criterion 55 in 10 CFR 50 states:

> Each line that is part of the reactor coolant pressure boundary and that penetrates primary reactor containment shall be provided with containment isolation valves as follows, unless it can be demonstrated that the containment isolation provisions for a specific class of lines such as instrument lines, are acceptable on some other defined basis:
>
> (1) One locked closed isolation valve inside and one locked closed isolation valve outside containment; or
>
> (2) One automatic isolation valve inside and one locked closed isolation valve outside containment; or

(3) One locked closed isolation valve inside and one automatic isolation valve outside containment. A simple check valve may not be used as the automatic isolation valve outside containment; or

(4) One automatic isolation valve inside and one automatic isolation valve outside containment. A simple check valve may not be used as the automatic isolation valve outside containment.

Isolation valves outside containment shall be located as close to containment as practical and upon loss of actuating power, automatic isolation valves shall be designed to take the position that provides greater safety.

Other appropriate requirements to minimize the probability or consequences of an accidental rupture of these lines or of lines connected to them shall be provided as necessary to assure adequate safety. Determination of the appropriateness of these requirements, such as higher quality in design, fabrication, and testing, additional provisions for inservice inspection, protection against more severe natural phenomena, and additional isolation valves and containment, shall include consideration of the population density, use characteristics, and physical characteristics of the site environs.

Criterion 55 Conformance

The reactor coolant pressure boundary (RCPB) (as defined in 10 CFR 50, Section 50.2[v]) consists of the reactor pressure vessel (RPV), pressure-retaining appurtenances attached to the RPV, and valves and pipes that extend from the RPV up to and including the outermost isolation valve. The lines of the RCPB that penetrate the primary containment are capable of isolation, thereby precluding any significant release of radioactivity. Similarly, lines that do not penetrate the primary containment but form a portion of the RCPB (such as connecting lines up to and including the second isolation valve) are designed to ensure that isolation of the reactor pressure boundary can be achieved.

6.2.4.2.2.2.1 Influent Lines

Influent lines that penetrate the primary containment and connect directly to the RPV are equipped with two isolation valves: one inside the containment, the other outside the containment. Both valves are located as close to the containment as practical. Influent lines which comprise part of the RCPB are listed below and discussed in detail in the remainder of this section.

| Penetration No. | System |
|---|---|
| X-9A | Feedwater<br>HPCI supply |
| X-9B | Feedwater<br>RCIC supply<br>RWCU return |
| X-13(A, B) | RHR pump discharge to recirculation loops |
| X-16(A, B) | Core spray pump discharge to core spray spargers |

| Penetration No. | System |
|---|---|
| X-42 | Standby liquid control system |
| X-49A and X-51A | Recirculation pump seal purge |

Feedwater System

The feedwater line penetrating the primary containment is part of the RCPB. This penetration is supplied with one automatic isolation valve inside and one automatic isolation valve outside the containment. The isolation valve inside the containment is a check valve. The isolation valve outside the containment is an air-operated, spring-to-close, positive-acting check valve.

Should a break occur in the feedwater line, the valves will prevent significant loss of fluid inventory and offer immediate isolation. During the postulated LOCA, it is desirable to maintain reactor coolant makeup from all sources of supply. For this reason, the outer containment isolation valve does not automatically isolate on a signal from the containment isolation system. However, the valve is capable of remote closure from the control room to provide long-term leakage protection when, based on operator judgment, continued makeup from the feedwater system is no longer necessary. A second check valve is located outside the containment--between the air-operated isolation valve and the containment wall--for added isolation capability.

RWCU, HPCI, and RCIC Systems

Influent lines that use the feedwater piping and penetrations in order to transfer fluid to the RPV consist of the reactor water cleanup (RWCU) return, and reactor core isolation cooling (RCIC), high-pressure coolant injection (HPCI) supply, and standby feedwater. Each of these lines can be isolated by the feedwater check valve inside the containment. The RCIC and HPCI supply lines each have an isolation valve outside the containment. These valves are normally closed, dc power-operated, remote manually actuated gate valves. The RWCU return line has a motor operated, normally open, ac power-operated gate valve as its isolation valve outside the containment. This valve is capable of remote closure from the control room. Two check valves are provided between the isolation valve and the containment wall. Should a break occur in the RWCU line, these check valves will prevent significant loss of fluid inventory from the feedwater side.

RHR and Core Spray Systems

The residual heat removal (RHR) pump discharge lines to the recirculation system (low-pressure coolant injection and shutdown cooling modes) and the core spray pump discharge lines have testable check valves inside the containment that provide for immediate isolation in the event of a break upstream of these valves. The outer containment isolation valves are remote manually actuated gate valves. However, no licensing credit is taken for the containment isolation feature of the RHR inboard check valves (see Reference 25a). Each valve will receive an automatic opening signal in the event of the postulated LOCA.

Standby Liquid Control System

The standby liquid control line uses a check valve as the isolation valve inside, as well as outside, the primary containment. General Design Criterion 55 states that a simple check valve may not be used as the automatic isolation valve outside the containment; however, should insertion of the liquid poison become necessary, it is imperative that the injection line be open. In the design of this system, it has been accepted practice to omit an automatic valve that opens on signal, as this introduces a possible failure mechanism. As a means of providing assurance for reliable and timely actuation, an explosive valve is used.

In this manner, the availability of the line is ensured. Because the standby liquid control line is a normally closed and non-flowing line, rupture of this line is a very remote possibility.

Recirculation Pump Seal Purge System

The recirculation pump seal purge lines use two air-operated globe valves, one inside the containment and one outside the containment. The valves isolate automatically on high drywell pressure or low vessel water level (level 2).

6.2.4.2.2.2.2 Effluent Lines

With the exception of the postaccident pressurized reactor coolant sample lines, effluent lines that form part of the RCPB and penetrate the primary containment are equipped with two isolation valves, one inside the containment and the other outside the containment. Both valves are located as close to the containment as practical. Effluent lines that comprise part of the RCPB are listed below, and are discussed in detail in the remainder of this section.

| Penetration No. | Section |
|---|---|
| X-7(A,B,C,D) | Main steam lines |
| X-8 | Main steam line drains |
| X-10 | Steam to RCIC turbine |
| X-11 | Steam to HPCI turbine |
| X-12 | RHR pump suction for recirculation piping (shutdown cooling mode) |
| X-28Cf | Postaccident pressurized reactor coolant sample |
| X-29A | Reactor water sample line |
| X-40Dd | Postaccident pressurized reactor coolant sample |
| X-43 | Reactor water cleanup suction |

Main Steam System

The MSIVs are air-operated, automatically actuated, Y-pattern globe valves. Two valves are provided in each line: one inside and one outside the containment. There is a third valve in each line outside the containment that is a gate valve.

The main steam drain line is provided with two automatic, motor-operated gate valves: one inside and one outside the containment. These valves are closed during normal reactor operation.

RCIC System

Both isolation valves in the RCIC steam supply line are normally open, remote manually actuated gate valves. These valves close automatically on indication of an RCIC system piping failure.

HPCI System

The isolation valves in the HPCI steam supply line consist of two gate valves and a 1-in. globe valve. All are remote manual motor-operated valves. The isolation valve inside the containment is open normally. The normally open, 1-in. globe valve bypasses the normally closed system supply valve outside the containment to keep the HPCI steam supply line warm. All HPCI steam supply line valves close automatically on indication of an HPCI system piping failure.

RHR System

The RHR shutdown cooling suction line is provided with two normally closed, automatically actuated, motor-operated gate valves and a locked-closed bypass valve. The bypass valve provides assurance that the normal shutdown cooling method will be available if the normally used valve fails. There is also a 3/4-in. bypass line with two check valves in series, which allows heated water trapped inside the RHR line to be relieved to the reactor vessel.

Reactor Coolant Sample System (Non-Postaccident)

The reactor water sample line is provided with two automatic, air-operated, fail-closed isolation globe valves: one inside and one outside the containment. These valves are closed during normal reactor operation, but receive an automatic closure signal in case they are open when containment isolation is required.

Postaccident Pressurized Reactor Coolant Sample

The two postaccident reactor coolant sample lines are connected to jet pump instrumentation lines outside the containment. Each line is provided with a solenoid-operated globe isolation valve outside the containment. These valves are closed during normal reactor operation and are opened only during postaccident conditions.

RWCU System

The RWCU suction line is provided with two normally open, automatic, motor-operated gate valves. These valves will close on receipt of a containment isolation or RWCU system piping failure signal.

Leak detection is provided for each line that has remote manual containment isolation valves and is evaluated against GDC 55.

#### 6.2.4.2.2.3 General Design Criterion 56

General Design Criterion 56 in 10 CFR 50 states

> Each line that connects directly to the containment atmosphere and penetrates primary reactor containment shall be provided with containment isolation valves as follows, unless it can be demonstrated that the containment isolation provisions for a specific class of lines, such as instrument lines, are acceptable on some other defined basis:
>
> (1) One locked closed isolation valve inside and one locked closed isolation valve outside containment; or
>
> (2) One automatic isolation valve inside and one locked closed isolation valve outside containment; or
>
> (3) One locked closed isolation valve inside and one automatic isolation valve outside containment. A simple check valve may not be used as the automatic isolation valve outside containment; or
>
> (4) One automatic isolation valve inside and one automatic isolation valve outside containment. A simple check valve may not be used as the automatic isolation valve outside containment.

Isolation valves outside containment shall be located as close to the containment as practical and upon loss of actuating power, automatic isolation valves shall be designed to take the position that provides greater safety.

Criterion 56 Conformance

The lines that penetrate the primary containment and communicate with the containment atmosphere can be grouped into two categories: (1) pipes that penetrate the primary containment and connect directly to the suppression pool; and (2) pipes that penetrate the primary containment and connect directly to the drywell atmosphere.

6.2.4.2.2.3.1 Lines Connecting To the Suppression Pool

Lines in this category are listed below:

| Penetration No. | System |
|---|---|
| X-205(A,B) | Torus to secondary containment vacuum breakers |
| X-205C | Suppression pool $N_2$ and air purge inlet |
| X-205D | Suppression pool exhaust and $N_2$ inlet |
| X-206(A, B, C, D, E, F) | Suppression pool water level and pressure instrumentation |
| X-210(A, B) | RHR minimum flow line<br>RHR heat exchanger thermal relief<br>RHR test line<br>Torus water management return<br>RHR suction thermal relief<br>RHR heat exchanger discharge header thermal relief<br>Postaccident liquid sample return<br>RHR warmup and return |
| X-211(A, B) | RHR to suppression pool spray |
| X-212 | RCIC turbine exhaust line |
| X-213(A, B) | Torus water management supply |
| X-214 | RCIC vacuum breaker line |
| X-215 | HPCI vacuum breaker line<br>Combustible gas control suction<br>Postaccident gaseous sample return |

| Penetration No. | System |
|---|---|
| X-217 | Grab sample line |
| X-218 | Combustible gas control return |
| X-219 | Combustible gas control suction |
| X-220 | HPCI turbine exhaust line |
| X-221 | HPCI turbine exhaust drain |
| X-222 | RCIC vacuum pump discharge |
| X-223(A, B, C, D) | RHR pump suction<br>RHR pump suction header thermal relief |
| X-224(A, B) | Core spray pump suction |
| X-225 | HPCI pump suction |
| X-226 | RCIC pump suction |
| X-227(A, B) | Core spray pump suction thermal relief<br>Core spray pump discharge header thermal relief<br>Core spray pump minimum flow line<br>Core spray pump test line<br>Torus water management return<br>HPCI minimum flow line<br>RCIC minimum flow line |
| X-230 | Primary containment monitoring system<br>Post accident suppression pool atmosphere sample |
| X-231 | Primary containment monitoring system |
| X-231 | Postaccident suppression poolatmosphere sample |

As stated in GDC 56, two isolation valves--one inside and one outside the containment--are required in lines that penetrate the primary containment and connect directly to the containment atmosphere. However, GDC 56 allows for alternatives to these explicit isolation requirements where the acceptable basis for each alternative is defined. The following are alternatives to explicit conformance with GDC 56. Notes in Table 6.2-2 identify the alternative basis to which each penetration is designed.

Two Isolation Valves Outside Containment

The primary containment radiation monitor system (PCRMS) is associated with Division I of the primary containment atmosphere monitoring system (PCAMS). The nonessential

PCRMS has two isolation valves on the inlet and two isolation valves on the outlet. These isolation valves are a normally open spring-to-close solenoid operated globe valve and an air operated ball valve. These inlet and outlet lines are connected to the containment atmosphere via PCAMS piping during normal operation. The isolation valves receive a containment isolation signal on a LOCA (see Subsection 6.2.4.2.2.3.2).

For lines that connect to the suppression pool, an isolation valve located inside the containment would necessitate placement of the valve either under water or in a high-humidity, nonaccessible area. Such placement would subject these valves to an extremely hostile environment, which could compromise their reliability and prevent routine inspection and maintenance. Thus, as an alternative to the explicit requirements of GDC 56 for lines in ESF or ESF-related systems, both isolation valves are located outside, and as close to, the containment wall as practical.

Relief Valves as Isolation Valves

Relief valves are provided in the RHR, core spray, HPCI, RCIC, and combustible gas control (CGC) systems as overpressureprotection devices. These valves are required for the design of Class B systems according to the ASME B&PV Code, Subsection NC-7000. The valves are installed in a manner that ensures their correct operation and reliability. Further, the Code requires that no stop valves or other devices be placed (in relation to a pressure relief device) so that it could impair the overpressure protection offered by the relief valve itself. Relief valves installed in these lines provide this required level of protection, and, if required to operate, would route the diverted fluid to the suppression pool.

Because of the orientation required, each of these relief valves is an isolation valve for the applicable penetration. The piping and valve designs are Quality Group B, Category I, and will withstand temperatures and pressures at least equal to the containment design pressure and temperature. Should the postulated LOCA occur, containment pressure would be felt on the downstream side of the relief valve, and would act in conjunction with the spring pressure setting of the relief valve to further enhance seating.

Remote Manual Isolation Valves

Remote manual valves are used as containment isolation valves in ESF and ESF-related systems. These systems include RHR, core spray, HPCI, RCIC, and reactor building closed cooling water (RBCCW) Emergency Equipment Cooling Water (EECW) systems. In each case, leak detection is provided.

Closed Systems Outside the Containment

The RHR, core spray, HPCI, and RCIC systems are closed-loop systems outside the containment. These systems can accommodate a single active failure and still maintain containment integrity. The systems are designed to Category I standards, are classified as Quality Group B, and will maintain their integrity should the containment experience its design temperature and pressure transient. Thus, as an alternative to the explicit requirements of GDC 56 for such lines in ESF or ESF-related systems, a single isolation valve is used outside the containment to enhance system reliability.

Lines that are not Quality Group B but that connect to these closed-loop systems are itemized in Table 6.2-14. By necessity, some of the valves in these lines are located near system pumps and are subject to missile damage should the pump fail. Should this occur, the system would be isolated either manually or automatically, and, therefore, failure of these valves as a result of missile damage would not constitute a breach of the primary containment.

Other Systems

The CGC, purge and inerting systems each use two isolation valves in series outside the suppression pool. Installing one of these valves inside the suppression pool could compromise reliability and prevent routine inspection and maintenance. These systems are built to the same quality standards as the primary containment and are protected against postulated missiles and pipe whip. The CGCS PCIVs are permanently de-energized and locked-closed.

The vacuum breakers to the secondary containment are essential for primary containment integrity. Isolation is provided through a power-to-close, spring-to-open butterfly valve and a testable check valve. Power from divisional electrical buses is applied to the butterfly valve to keep the valves closed, except when air is required to relieve a vacuum inside the primary containment. The butterfly valve will open on loss of power or degraded voltage but closes automatically once power is restored or voltage recovers. During a LOCA concurrent with a Loss of Offsite Power (LOP), the butterfly valves will de-energize and open until power is restored to the divisional electrical buses. Upon restoration of power, the butterfly valves will re-energize and reposition, closing the valves. During a LOCA concurrent with a low grid voltage, insufficient voltage during the time Core Spray and RHR pumps start may cause the Division I butterfly valve to pen. Once nominal voltage is restored after RHR and Core Spray pump starts, the butterfly valve will close. In either scenario, the time the butterfly valve is open is less than the 108 second allowed stroke time for containment isolation valves established in the accident analysis. The vacuum breaker testable check valves provide containment isolation and remain closed during the accident unless negative differential pressure exists. These lines and valves are Category I, Quality Group B and are located in missile-free areas.

6.2.4.2.2.3.2 Lines Connecting To the Drywell

Lines in this category are listed below and discussed in the remainder of this section. The lines are Category I and Quality Group B at least through the outermost containment isolation valve.

| Penetration No. | System |
|---|---|
| X-15 | CGC suction |
| X-17 | Abandon RHR head spray |
| X-18 | Drywell floor drain sump pump discharge |
| X-19 | Drywell equipment drain sump pump discharge |
| X-20 | Demineralized service water to drywell |

| Penetration No. | System |
|---|---|
| X-22 | Control air and $N_2$ to drywell |
| X-23 | RBCCW/EECW supply |
| X-24 | RBCCW/EECW return |
| X-25 | Drywell exhaust |
| X-26 | Drywell $N_2$ and air inlet |
| X-27(a, b, c, d, e) | Containment atmosphere sample and postaccident drywell atmosphere sample (X-27b only) |
| X-27f | Drywell pressure instrumentation |
| X-29B (b,c) | Reactor protection system |
| X-29Be | Drywell instrumentation |
| X-31Ba | Drywell on line pressure control |
| X-34(A, B) | RBCCW/EECW supply and return |
| X-36 | $N_2$ to dry |
| X-39(A, B) | RHR to containment spray header |
| X-44 | CGC suction |
| X-47(a,b) | Reactor protection system |
| X-47e | Drywell instrumentation Nitrogen inerting instrumentation |
| X-48(a,b,c,d,e,f) | Containment atmosphere sample and postaccident drywell atmosphere sample (X-48f only) |

Regulatory Guide 1.7 was revised in March 2007 to reflect the amended 10 CFR 50.44. The Combustible Gas Control System (CGCS) has been retired in place with its electrical circuits de-energized and fluid process piping isolated from primary containment with redundant locked-closed isolation valves. The valves are located external to the primary containment, and are accessible for inspection and testing during normal reactor operation.

Penetration X-17 for the abandoned RHR head spray line now conforms to the requirements of GDC-56 since the line is no longer directly connected to the RPV. Two normally closed motor operated valves are located in this line, one inside containment and one outside containment.

The drywell equipment and floor drain sump pump discharge lines each have a motor-operated, normally open gate valve inside the containment, and an air-operated, normally closed gate valve outside the containment. These valves receive containment isolation signals on the postulated LOCA. Rupture disc overpressure protection is installed to limit the pressure rise from LOCA heatup of the isolated penetrations per GL 96-06. The rupture disc discharges into a small discharge tank which provides a sealed closed barrier for containment isolation.

Demineralized service water line has an isolation valve inside containment and a spectacle flange assembly with blank installed outside containment. Control air and nitrogen lines have isolation valves inside and outside the drywell. The demineralized service water isolation valve is the only manual valve in this group. The valve remains locked closed at all times during reactor operation.

The drywell exhaust and air purge lines have isolation valves inside and outside the containment. The valves are either automatically or remote manually actuated. Leak detection is provided to inform the control room operator when closure of the remote manual valves is required.

The RHR pump discharge to the containment spray lines contains two isolation valves outside the containment. Since the spray header is integral to the drywell wall, placing an isolation valve inside the containment could compromise the structural integrity of the containment spray headers.

The RBCCW/EECW supply lines each have a check valve inside the containment and a motor-operated gate valve outside the containment. These motor-operated gate valves are remote, manually actuated and close on a high drywell pressure EECW initiation signal. The RBCCW/EECW return lines each have a remote, manually actuated, diverse electrically powered motor-operated gate valve inside and outside the containment.

The drywell instrumentation, nitrogen-inerting instrumentation, reactor protection system, and containment atmosphere sample systems are closed-loop systems outside the containment. These systems can accommodate a single active failure and still maintain containment integrity. The systems are designed and installed as Quality Group B, up to and including the isolation valves. The balance of the instrument piping is designed to meet Quality Group B design criteria. These design criteria include stress analysis with consideration given to deadweight, thermal, and seismic conditions. The systems are seismically supported.

Nuclear-grade materials are used throughout the fabrication of the piping system. They will maintain their integrity should the containment experience its design temperature and pressure transient. Thus, as an alternative to the explicit requirements of GDC 56 for such lines in ESF or ESF-related systems, a single air-operated isolation valve or solenoid-operated isolation valve is used outside the containment to enhance system reliability.

The lines that connect the nonessential PCRMS to Division I of the closed outside containment loop of the PCAMS have two isolation valves outside containment for both the inlet and outlet of the PCRMS. The PCRMS utilizes common piping of PCAMS; therefore, the valves are outside containment and placed as close as practical to the PCAMS piping loop. All other requirements of GDC 56 are met.

The drywell postaccident atmosphere sample lines contain two solenoid-operated globe isolation valves outside the containment. These lines are connected to the normal containment atmosphere sample system lines outside the containment. These valves are closed during normal reactor operation and are opened only during postaccident conditions.

6.2.4.2.2.4 General Design Criterion 57

General Design Criterion 57 in 10 CFR 50 states

> Each line that penetrates primary reactor containment and is neither part of the reactor coolant pressure boundary nor connected directly to the containment atmosphere shall have at least one containment isolation valve which shall be either automatic, or locked closed, or capable of remote manual operation. This valve shall be outside the containment and located as close to the containment as practical. A simple check valve may not be used as the automatic isolation valve.

Criterion 57 Conformance

Penetrations X-204 (A through M) for the drywell-to-torus vacuum breaker nitrogen supply and their associated isolation valves conform to the locked closed requirements of GDC 57 to comply with Technical Specification 3.6.1.3.2. A locked closed, air-operated globe valve as defined is SRP 6.2.4 is located in each line outside the containment.

6.2.4.2.3 Containment Isolation Dependability

Fermi 2 meets the NRC requirements developed for reliable containment isolation as follows.

a. The containment isolation design complies with the recommendation of SRP 6.2.4 in that there is diversity in the parameters sensed for the initiation of containment isolation. Safety-grade signals are provided for the detection of abnormal conditions in the reactor coolant system and containment; these are low reactor vessel water level and high drywell pressure

   Several lines are not isolated on the high-drywell- pressure signal in order to retain system availability for small breaks or leaks. Justification for these cases is given under Comments in Table 6.2-15

b. Essential and nonessential systems containing piping systems that penetrate the containment are identified in Table 6.2-16. Those systems identified as essential are regarded as indispensable or are backup systems in the event of a LOCA. The nonessential systems have been judged to be not required in LOCA situations

c. Nonessential lines that are a possible open path out of the containment are automatically isolated by the containment isolation signals, by check valves that prevent flow out of the containment, or by manual valves that are normally closed. Normally closed valves are under administrative control to ensure that valves are closed during startup, power, hot-standby, and hot-shutdown modes of operation

   For instrument lines connected to the RCPB, each line is equipped with a flow-restricting orifice located as close as practical to the point of connection to the RCPB. A manual shutoff valve is located outside the containment and is located as close as practical to the containment wall. An excess-flow check valve is provided immediately downstream of the manual valve. This design and installation follows the guidance of Regulatory Guide 1.11

d. The resetting of containment isolation signals does not result in the automatic reopening of containment isolation valves. Deliberate operator action is required to reopen a containment isolation valve once the containment isolation signals are reset

e. Drywell high pressure initiates the containment isolation of nonessential systems and lines. The Technical Specifications specify the drywell high-pressure trip-point setting

f. The Fermi 2 purge valves satisfy the operability criteria set forth in Branch Technical Position (BTP) CSB 6-4, Revision 1, and Staff Interim Position dated October 23, 1979. The Fermi 2 position relative to BTP CSB 6-4 is provided in Subsection 6.2.5.2.5. Fermi 2 complies with the Staff Interim Position as follows: (1) the purge valves are intended to be operated only for inerting, deinerting, or pressure control in accordance with the Technical Specifications; and (2) the Fermi 2 valves are operable for DBA flows

g. Containment purge and ventilation isolation valves close automatically upon the detection of high airborne radiation in the reactor building exhaust line. This high-radiation isolation signal is in addition to the diverse containment isolation signals.

#### 6.2.4.2.4 Valve Closure Times

Proper valve closing time is achieved by appropriate selection of valve type, operator type, and operator size. Isolation valve closing times were verified during the functional performance tests prior to reactor startup and are periodically retested at intervals specified in the Technical Specifications. The design of piping systems penetrating the reactor containment includes provisions for operability and leakage testing of isolation valves.

Motive power for the valves on process lines that require two valves is supplied from physically independent power sources to provide a high probability that no single event could interrupt power to both closure devices. Loss of valve actuation power is detected and annunciated in the main control room.

In general, isolation valves located outside the primary containment receive dc power from the Division II power supply, or alternate division ac power, while those located inside the primary containment receive ac power from the Division I power supply.

#### 6.2.4.2.5 Instrument Lines Penetrating the Primary Containment

All instrument lines connected to the reactor coolant pressure boundary are Category I and Quality Group A. Physical separation is provided for redundant instrument lines to the extent practical, so that the failure of one line will not induce failure in another. The response time for all sensors connected to instrument lines is not affected by the valves or orifices in the line. The design and installation of instrument lines follows the guidance of Regulatory Guide 1.11 (Safety Guide 11).

The instrument-sensing lines listed below penetrate the primary containment and connect to the RCPB.

| Number of lines | Instrument Description |
|---|---|
| 24 | Jet pump flow |
| 1 | Jet pump |
| 14 | RPV level/pressure* |
| 8 | Recirculation inlet to RPV DP |
| 2 | Recirculation system pressure |
| 8 | Recirculation system flow |
| 4 | Recirculation Pump DP |
| 4 | Recirculation pump seal pressure |
| 4 | Steam flow to HPCI turbine |
| 4 | Steam flow to RCIC turbine |
| 16 | Main steam flow |
| 2 | Feedwater pressure** |

* The portion of the instrument line passing through the containment is part of a penetration assembly that is part of the containment and thus is Quality Group B, consistent with the Containment Quality Group.

Two check valves are provided in series for the isolation of each division of the reactor vessel instrument-sensing line backfill system from the RPV level/pressure instrument reference legs.

** These lines do not penetrate the containment. They tap in between the containment and the outer isolation valve.

Each line, except for the feedwater pressure-sensing line, is equipped with a flow-restricting orifice located as close as practical to the point of connection to the RCPB. No such device is necessary for the feedwater pressure-sensing lines because they tap in outside the containment, and the isolation valve inside the containment (check valve) serves the function of the restricting orifice. A manual shutoff valve is located outside the primary containment and is installed as close as practical to the containment wall or pipe (in the case of feedwater). An excess-flow check valve is provided immediately downstream of the manual valve. The excess-flow check valve will close automatically in the event of a line break downstream. Indicating lights on a control room panel monitor excess-flow-check-valve position. These valves may be reopened by actuation of a solenoid valve, which is operated from a local control panel, after repairs are made. This design and installation follows the guidance of Regulatory Guide 1.11. There are no instrument lines that penetrate both the primary and the secondary containments.

The postulated break of an instrument line attached to the RCPB is discussed and evaluated in Subsection 15.6.2. Leakage from such a rupture upstream of the excess-flow check valve is minimized by the restricting orifice in the line. The integrity and functional performance of the secondary containment and standby gas treatment systems are not impaired by this event, and the calculated potential offsite exposures are substantially below the guidelines of 10 CFR 100.

Each instrument line except the jet pump instrument lines is provided with a 0.25-in.-diameter orifice in addition to the excess-flow check valve. The jet pump lines are 0.25 in. diameter from the RPV nozzles to the jet pump taps. This orifice will restrict the coolant loss to a value whose equivalent steam volume is much less than the capacity of one standby gas treatment system (SGTS) train. Therefore, pressurization of the secondary containment will not result from an instrument line break and a failure of the associated excess-flow check valve to isolate the ruptured line. Coolant lost from such a break is inconsequential when compared to the makeup capabilities of the feedwater or RCIC system.

#### 6.2.4.2.6 Leak Detection

For systems penetrating the primary containment, major leaks in the pipe are located by increased temperature, radiation, sump level, changes in pressure, differential pressure, process line flow, etc. These indications are monitored in the control room to alert the operator when remote manual valves should be closed. In addition, certain indications of leakage will cause automatic valves to close in response to a system accident.

Leak detection is further discussed in Sections 5.2 and 7.6.

#### 6.2.4.2.7 Leak Rate Testing

The reactor containment and containment penetrations are designed to permit periodic leak rate testing in accordance with GDC 52 and 53, and Appendix J to 10 CFR 50. See also Subsection 6.2.4.4.

Testing requirements for piping penetration isolation barriers and valves have been established by using the intent of GDC 54 as interpreted in Appendix J to 10 CFR 50. Exceptions taken to Appendix J Type C tests are described in Table 6.2-2.

The primary containment isolation system is scheduled to undergo periodic testing during reactor operation. The functional capabilities of power-operated isolation valves are tested remote manually from the main control room. By observing position indicators and changes in the operation of the affected system, the closing ability of a particular isolation valve is demonstrated. Testable check valves are provided on influent lines whose operability is relied upon.

Test capabilities, incorporated in the primary containment system to permit leak testing of containment isolation valves, are separated into two categories. The first category consists of pipelines that open into the containment and do not terminate in closed loops outside the containment, but do contain two isolation valves in series. Test taps are provided between the two valves to permit leakage monitoring. The second category consists of pipelines that connect to the reactor cooling system and that also contain two isolation valves in series. A leakoff line is provided between the two valves, and a drain line is provided downstream of the outboard valve. This arrangement permits leakage monitoring of both the inboard and outboard valves. Valves subject to Type C testing are shown in Table 6.2-2.

Excess-flow check valves can be tested by opening a test drain valve downstream of each valve and verifying proper operation.

As these valves are outside the primary containment and are accessible, periodic visual inspection can be performed in addition to the operational check.

The only systems circulating contaminated water after a postulated LOCA are the core spray system (to cool the reactor core) and the RHR system (to remove the heat from the emergency coolant).

The potential sources for leakage are the pump mechanical seals. The available data indicate the leakage from the pump seals to be essentially zero. This is based on the manufacturer's design criteria, its technical manuals, and industrywide experience. Therefore, specifying a leakage limit would be quite arbitrary.

Only a seal failure could result in any significant leakage. This leakage would be indicated by the operation of the sump pumps in either one of the equipment drain sumps or the floor drain sumps located in each of the four corner rooms of the reactor building subbasement. Sump pump startup is indicated in the control room. Following sump pump startup and operator investigation, the leaking emergency core cooling system (ECCS) pump would be isolated.

Following a postulated LOCA, either one LPCI pump or two core spray pumps are required for core cooling and one RHR pump is required for long-term containment cooling. Should seal failure occur in one of these, there is sufficient redundancy to allow the leaking pump to be removed from service and isolated. Four RHR pumps and four core spray pumps are provided.

Radioactivity releases and resultant doses from this postulated seal leak would be negligible.

#### 6.2.4.2.8 Environmental Qualification Tests

Qualification tests required to ensure the performance of the isolation valves under adverse environmental conditions are discussed in Section 3.11.

### 6.2.4.3 Design Evaluation

One of the basic purposes of the primary containment system is to provide a minimum of one protective barrier between the reactor and the environs. To fulfill its role as a barrier, the primary containment is designed to remain intact before, during, and subsequent to any failure involving process systems either inside or outside the primary containment. Where process lines penetrate the primary containment, the penetration has the same integrity as the primary containment structure itself. In addition, the process line isolation valves perform the containment isolation function for leakage through the process lines.

Since a rupture of a large line connected to the reactor coolant system and penetrating the primary containment may be postulated, isolation valves for lines of this type are required to be located within the primary containment. These isolation valves are required to close automatically on various indications of reactor coolant loss. A certain degree of additional reliability is added if a second valve, located outside and as close as practical to the primary containment, is included. This second valve also closes automatically. A single active failure can be accommodated since a second valve is available to perform the containment isolation function. By physically separating the two valves, there is less likelihood that a failure of one valve would cause failure of the second. Series valves of this type are provided with independent power sources.

As an example, the ability of the main steam line penetrations and the associated steam line isolation valves to fulfill the containment isolation objective for several break conditions in the steam line is shown by consideration of various assumed main steam line break locations.

a. The failure occurs inside the drywell, upstream of the inner isolation valve. Steam from the reactor is released into the drywell, and the resulting sequence is similar to that of the design-basis accident (DBA) except that the pressure transient is less severe since the reactor blowdown rate is slower. Both isolation valves close on receipt of a signal indicating low water level in the RPV. This action provides two barriers within the steam pipe passing through the penetration and prevents further flow of steam to the turbine. Thus, when the two isolation valves close subsequent to this postulated failure, containment integrity is attained and the reactor is effectively isolated from the external environment

b. The failure occurs inside the drywell, and it is assumed that the inner isolation valve is inoperable. Again, reactor steam will blow into the primary containment. The outer isolation valve will close on receipt of a signal indicating low water level in the RPV, and the reactor will become isolated within the primary containment, as delineated above

c. The failure occurs downstream of the inner isolation valve either inside the drywell or within the guard pipe. Both isolation valves will close on receipt of a signal indicating low water level in the RPV. The guard pipe is designed to accommodate such a failure without damage to the drywell penetration bellows. In addition, the design of the pipeline both supports and protects its welded juncture to the drywell vessel. Thus, the RPV is isolated within the primary containment by the inner isolation valve, and the primary containment integrity is main-tained by closure of the outer isolation valve. It should be noted that this condition provides two barriers between the reactor core and the external environment

d. The failure occurs outside the primary containment between the penetration and the outer isolation valve. Steam will blow directly into the pipe tunnel until the isolation valves are closed automatically. Closure of the inner isolation valve places a barrier between the reactor core and the external environment. This barrier serves to isolate the reactor and maintain containment integrity

e. The failure occurs outside the primary containment, and it is assumed that the outer isolation valve is inoperable. Containment isolation is established by the inner isolation valve, and containment integrity is maintained as described in Item d., above

f. The failure occurs outside the primary containment between the outer isolation valve and the turbine. Steam will blow directly into the pipe tunnel or the turbine building until both isolation valves are closed automatically. This action isolates the reactor, completes containment integrity, and places two barriers in series between the reactor core and the outside environment

Exceptions to the arrangement of isolation valves described above for lines connected directly to the primary containment atmosphere or reactor coolant system are made only in cases in which the above arrangement would lead to a less desirable situation because of required operation or maintenance of the system in which the valves are located.

Isolation valves must be closed before significant amounts of fission products are released from the reactor core during the DBA. Because the amount of radioactive material in the reactor coolant is small, a sufficient limitation of fission product release will be accomplished if the isolation valves are closed before the coolant drops to a level below the top of the reactor core. For a discussion of closure times for Class A and Class B isolation valves, refer to Section 7.3.2.2.

Valves, sensors, and other automatic devices essential to containment isolation are provided with means for periodic testing of their functional performance. Such tests provide reasonable assurance that the primary containment isolation system will perform properly when required.

#### 6.2.4.4 Leak Rate Testing

A testing program has been implemented to measure containment leakage rates prior to initial operation of the unit, and to test the primary containment periodically throughout the operating life. The purpose of the testing is to verify that the leakage rate is within allowable limits given in the Technical Specifications and in the Inservice Testing Program for Pumps and Valves (Subsection 5.2.8.7).

The testing program includes performance of Type A tests to measure the overall integrated leakage rates, Type B tests to detect and measure local leakage from certain components, and Type C tests to measure valve leakage rates.

The leakage tests are performed in accordance with the Fermi 2 Primary Containment Leakage Rate Testing Program as defined in the Technical Specifications. The program, which is based on the requirements of 10 CFR 50 Appendix J, Option B, retains certain previously approved exemptions, and utilizes the approach as defined in Regulatory Guide 1.163 (see Appendix A, Subsection A.1.163).

##### 6.2.4.4.1 Type A Tests

Type A tests are intended to measure the primary reactor containment overall integrated leakage rate after the containment has been completed and is ready for operation and at periodic intervals thereafter.

After the preoperational leakage rate test, testing will be scheduled in accordance with Fermi 2 Technical Specifications.

The Type A test will be performed using the Absolute Method or other alternative testing methods that have been approved by the NRC, and verification will be achieved by the Superimposed Leak Method, as described in ANSI/ANS 56.8-2002.

Prior to Type A testing, all lines are either isolated or drained and vented to reflect their status following a postulated LOCA. This ensures that Type A test results accurately reflect the most restrictive LOCA conditions. Systems that are provided with isolation capabilities to satisfy GDC 55 or 56 are either normally open to the containment atmosphere or will be vented to the containment during Type A tests. Exceptions to this are systems that must be in operation during the test.

The primary containment is pressurized and depressurized using existing system piping and equipment to the extent possible. Appropriate pressure controls are provided to attain the test pressure and for controlled depressurization to the plant vent stack via existing adsorber filters. Pressurization is carried out under conditions that will minimize containment air humidity and temperature.

Temperature-sensing devices are distributed throughout the containment and at different parts of the structure wherever local temperature variations are expected in the course of the test. Fans are used for air circulation as required to equalize temperatures.

Measurements are taken during each test period to provide a sufficient amount of data to determine leakage rates for the following tests.

a. Preoperational Leakage Rate Test

   1. Peak Pressure Test

      A test was performed at pressure $P_a$ (where $P_a$ is the calculated peak containment internal pressure related to the DBA) to measure the leakage rate $L_{am}$ (where $L_{am}$ is the total measured leakage rate at pressure $P_a$ obtained from testing the containment with equipment and systems in a state as close as practical to that which would exist under DBA conditions)

   2. Acceptance Criteria

      $L_{am}$ shall be no greater than $L_d$ (where $L_d$ is the design leakage rate at pressure $P_a$, as specified in the Technical Specifications), which conforms to the requirement of 10 CFR 50 Appendix J, Option B that $L_{am}$ shall be less than 0.75 $L_a$ (where $L_a$ is the maximum allowable leakage at pressure $P_a$). See Table 6.2-1 for pressure and leakage values

   3. Results

      The preoperational leak rate test was concluded on December 7, 1984. The calculated leak rates at the 95 percent confidence level were below the acceptance criterion of 0.375 weight percent/day. The Appendix J acceptance criterion at 95 percent confidence level is 0.75 $L_a$ = (0.75)(0.50 weight percent/day) = 0.375 weight percent/day. The accuracy of the test was verified by means of a supplemental test.

b. Periodic Leak Rate Tests

1. The peak pressure tests shall be conducted at $P_a$

2. Acceptance Criteria - same as Item 2 above.

The accuracy of Type A tests will be verified by a supplemental test. The verification is intended to be conducted by the Superimposed Leak Method.

Results from the supplemental test are acceptable provided the difference between the supplemental test data and the Type A data is within 0.25 $L_a$.

If this should not be the case, the reason shall be determined, corrective action taken, and a successful supplemental test performed.

6.2.4.4.2 Type B Test

The Type B test is intended to detect local leaks and to measure leakage across each pressure-containing or leakage-limiting boundary for the following primary containment penetrations:

a. Contained penetrations whose design incorporates resilient seals, gaskets, or sealant compounds, piping penetrations fitted with expansion bellows, and electrical penetrations fitted with flexible metal seal assemblies

b. Air-lock door seals, including door-operating mechanism penetrations that are part of the containment pressure boundary

c. Doors with resilient seals or gaskets, except for seal-welded doors.

Table 6.2-2 lists those penetrations that require Type B testing. A detailed description of those penetrations is found in Subsection 6.2.1.2.1.

Type B tests (except the test for the air lock) shall be performed and scheduled in accordance with the Primary Containment Leakage Rate Testing Program as described in the Technical Specifications, based on 10 CFR 50 Appendix J, Option B. Air locks shall be tested at 30-month intervals or after maintenance is performed on the air lock. Additionally, the interior and exterior door seals of the air locks shall be tested after each air-lock opening in accordance with the Primary Containment Leakage Rate Testing Program as described in the Technical Specifications, based on 10 CFR 50 Appendix J, Option B.

All components subject to Type B testing are equipped with test connections to allow pressurization with a test medium.

Soap-bubble testing at design pressure $P_a$ will be used, if necessary, to provide a sensitive and rapid method for qualitative determination of leakage over large areas. The quantitative leakage measurements are made by pressurizing the component to be tested with air or nitrogen to design pressure $P_a$ and measuring the amount of gas required to maintain that pressure.

The personnel access lock and equipment access doors are tested for leakage in accordance with approved written procedure, specifically, the Type B test procedures. The drywell personnel access lock has two mechanically interlocked, gasketed doors. These are designed and fabricated to withstand drywell design pressure.

The Type B test for the personnel access lock is conducted in three steps:

a. The exterior door seals are tested by connecting the local leak-rate test (LLRT) panel to a pressure tap, which has been provided, pressurizing the space between the door's testable gasket to design pressure, and measuring the leak rate

b. The interior door seals are tested in a manner similar to that of the external door seals

c. The space between the shut interior and exterior doors is tested by connecting the LLRT panel to a pressure tap, which has been provided, pressurizing to design pressure, and measuring the leak rate. Prior to conducting this step, tie-downs are installed on the interior door to ensure proper seating of the interior door's testable gasket when pressure is applied in a direction that is not normally expected. By design, both the interior and exterior doors seal with internal pressure, thereby providing a better seal as the drywell pressure increases.

When the tie-downs are installed on the interior door, the air lock cannot be operated from within.

The tie-downs are adjusted to permit compression of the gasket until the door is about 1/16 in. away from the frame. The forces exerted on the door during the leak-rate test (Type B) are the sum of the forces caused by the mechanical tie-downs and the forces attributable to the test pressure.

#### 6.2.4.4.3 Type C Test

Table 6.2-2 lists all containment isolation valves that require Type C testing, plus a sketch of piping configurations and test connections.

Type C tests will be performed and scheduled in accordance with the Primary Containment Leakage Rate Testing Program as described in the Technical Specifications, based on 10 CFR 50 Appendix J, Option B.

The boundaries for each test will be established with consideration for minimizing the test volume. Test connections for venting, draining, and pressurization are provided on penetration piping that includes valves requiring Type C testing. To the extent practicable, the piping between the containment penetrations and the test connection isolation valves is minimized.

The tests shall be performed by local pressurization applied in the same direction that the valve would be required to perform its safety function, unless it has been determined that applying the pressure in the opposite direction will provide equal or more conservative results.

Valves listed in Table 6.2-2 as being Type C tested, except those having a water seal which can be maintained for at least 30 days after an accident requiring containment isolation, shall be pressurized with air or nitrogen to the design pressure $P_a$. Valves that have a water seal shall be pressurized with that fluid to a pressure of not less than 1.10 $P_a$.

Type C LLRT testing is not required for containment isolation valves that are located in piping of systems which penetrate the Torus and terminate below the minimum water level in the Torus when the systems are closed both inside and outside of containment. The Torus is designed and operated so that it is always filled with water. The supply of water in the Torus is assured during all design basis post-accident modes of operation. Consequently, the subject isolation valves will remain “sealed” by the water.

The water seal inside the Torus, in conjunction with the design of the piping associated with the penetrations, is a passive post-accident containment bypass leakage barrier. It precludes any direct communication between the post-accident Primary Reactor Containment atmosphere and the subject Containment Isolation Valves, thereby eliminating the possibility of post-accident containment bypass leakage. The torus is assured to maintain its level 30 days post accident, as described in Section 6.2.4.4.3. As such, the torus is not a “seal-water fluid system” as intended by Appendix J. Therefore, 10 CFR 50, Appendix J, Type C water leak rate testing for the lines and valves is not appropriate and is not necessary to ensure post-accident, containment integrity.

The combined leakage rate for all penetrations and valves subject to Type B and C tests shall be less than 0.60 $L_a$. Leakage from those valves that are sealed with a fluid from a seal system may be excluded when determining the combined leakage rate provided that

a. The fluid leakage limit is based on a radiological analysis of the plant site

b. The installed isolation valve seal system fluid inventory is sufficient to ensure sealing function for at least 30 days at a pressure of 1.10 $P_a$.

Test, vent, and drain (TVD) connections on Class 1 systems which are a part of the containment boundary are provided with at least two isolation valves and are sealed with a threaded pipe cap except for the vents on the RHR return piping inside the drywell which are provided with one isolation valve and a threaded cap. All other TVD connections which are a part of the containment boundary are provided with at least one isolation valve and sealed with a threaded pipe cap. Test, vent, and drain connections shall be under administrative control, and they shall be subject to periodic surveillance to verify their integrity and to verify the effectiveness of the administrative controls in ensuring closure.

There are six types of valves that are not tested in the accident direction: globe valves, gate valves, ball valves, relief valves, stop check valves, and butterfly valves. Each valve type is discussed separately in the following paragraphs.

Globe Valves (Note 2, Table 6.2-2)

All globe valves that are tested in the reverse direction have test pressure applied beneath the disk. The seating force of a globe valve is the vector sum of the actuator force and the fluid force on the valve plug. For all globe valves being considered, accident pressure is above the seat and is thus acting in the same direction as the actuator force, tending to close the valve. When a valve of this configuration is tested in the reverse direction (pressure under the seat), test pressure will be acting in opposition to the actuator force, thus tending to unseat the valve. Therefore, the resultant force on the seating surface will be less when test pressure is applied in the reverse direction than when pressure is applied in the accident direction. As there is only one seating surface where the fluid pressure is applied in a direction opposite to the actuator force, leakage will tend to increase due to the reduced seating load. Because leakage during a test in the reverse direction tends to be greater than when fluid pressure is applied in the accident direction, a test in the reverse direction is conservative.

Gate Valves (Note 4, Table 6.2-2)

All gate valves that are not tested in the accident direction are wedge-disk-type gate valves. In lieu of testing these valves in the accident direction, Edison tests them through the bonnet. The gate valve may be tested through a body/bonnet tap. This valve has a tap through which the body/bonnet area is pressurized. Leakage is measured through both seating surfaces along with leakage through the bonnet and packing. Compared with testing in the accident direction, this method of leakage testing is more conservative.

Butterfly Valves (Note 11, Table 6.2-2)

Twenty-seven butterfly valves serve as containment isolation valves and are subject to Type C leak testing. Twelve of the valves are inboard isolation valves, and the remaining 15 are outboard isolation valves.

During Type C testing, the pipe volume or test volume between the inboard and outboard valves will be pressurized. Pressurizing between the valves is necessary because test volumes cannot be established on the containment side of the inboard isolation valve given the present valve and line configurations. Thus, the inboard valves will have the test differential pressure applied in the reverse direction to the accident pressure.

Of the ten inboard valves located outside the primary containment, eight of the valves will have their pipe-to-valve flanges nearest containment seal welded to ensure a leaktight pressure boundary. Because of this seal weld, the inboard valves will have to be maintained in place. In order to change the valve seat, access from the disk side of the valve is necessary. Therefore, for all inboard valves that are located outside the primary containment, the valve disk must face away from the primary containment. There are 10 inboard valves located outside the primary containment. With this orientation, stem leakage of the inboard valves is not measured while pressurizing the test volume. Additionally, two of the inboard containment isolation valves are located inside the primary containment and are flanged into place.

For Type C testing, the stem leakage is measured by pressurizing to $P_a$ through the stem vent and adding this stem leakage to the test volume leakage. The valve manufacturer has stated that the leakage through the stem is not dependent on the direction of the differential pressure. Consequently, pressurizing through the stem vent will yield stem leakage results that are conservative or equivalent to applying the pressure differential in the accident direction.

There are two inboard isolation valves on the inside of the containment. These valves have the disk facing the containment so that the valve seats are accessible. Pressurizing the test volume between the inboard and outboard valves will provide the stem leakage along with the seat leakage.

All the outboard valves have the disk facing toward the containment; thus test and accident differential pressures are in the same direction.

Relief Valves (Note 29, Table 6.2-2)

In addition to the safety/relief valves on the main steam lines, there are 17 relief valves that blow down to the pressure suppression chamber and therefore are classified as isolation valves. During a LOCA, containment pressure will be acting over the relief valve seat. Therefore, the direction of the accident pressure differential will tend to seat the valves.

Of the 17 relief valves, 15 of them will not be Type C LLRT tested. These 15 relief valves are located in piping of systems which penetrate the Torus and terminate below the minimum water level in the Torus. The Torus is designed and operated so that it is always filled with water. The supply of water in the Torus is assured during all design basis, post-accident modes of operation. Consequently, the subject isolation valves will remain “sealed” by the water.

The water seal inside the Torus, in conjunction with the design of the piping associated with the penetrations, is a passive, post-accident containment bypass leakage barrier. It precludes any direct communication between the post-accident Primary Reactor Containment atmosphere and the subject CIVs, thereby eliminating the possibility of post-accident containment bypass leakage. The Torus is assured to maintain its level 30 days post accident, as described in Section 6.2.4.4.3. As such, the torus is not a “seal-water fluid system” as intended by Appendix J. Therefore, 10 CFR 50, Appendix J, Type C water leak rate testing for the lines and valves is not appropriate and is not necessary to ensure post-accident containment integrity.

The two remaining valves in the CGC system will be in-situ tested in the accident direction at a pressure of $P_a$.

Stop Check Valves (Table 6.2-2)

There are four stop check valves in the HPCI and RCIC systems. All of these stop check valves have uncoupled globe style disks and motor operators.

Operating procedures provide instructions for closing these stop check isolation valves following a post-LOCA event when the HPCI and RCIC systems are no longer needed.

These four stop check valves will not be LLRT Type C tested. All four are located in piping of systems which penetrate the Torus and terminate below the minimum water level in the Torus. The Torus is designed and operated so that it is always filled with water. The supply of water in the Torus is assured during all design basis, post-accident modes of operation. Consequently, the subject isolation valves will remain “sealed” by the water.

The water seal inside the Torus, in conjunction with the design of the piping associated with the penetrations, is a passive, post-accident containment bypass leakage barrier. It precludes any direct communication between the post-accident Primary Reactor Containment atmosphere and the subject Containment Isolation Valves thereby eliminating the possibility of post-accident containment bypass leakage. The torus is assured to maintain its level 30 days post accident, as described in Section 6.2.4.4.3. As such, the torus is not a “seal-water fluid system” as intended by Appendix J. Therefore, 10 CFR 50, Appendix J, Type C water leak rate testing for the lines and valves is not appropriate and is not necessary to ensure post-accident, containment integrity.

Ball Valves (Note 13, Table 6.2-2)

There are 23 ball valves used for containment isolation; 10 of these are tested in the forward direction and 13 are tested in the reverse direction. All of these valves were manufactured by Jamesbury and are air operated and spring assisted to fail in the closed position.

Valves of this type have the same sealing characteristics in either direction. Consequently, test results obtained in the present configuration (i.e., reverse direction) are equivalent to testing in the accident direction. Additionally, these valves have a "corner seal" design on the stem and stem packing. This design eliminates stem leakage.

The spring assist merely rotates the ball valve in its seat. It does not increase the seat pressure; therefore, the spring assist has no effect on the leakage regardless of the test direction.

### 6.2.5 Primary Containment Combustible Gas Control

The NRC amended 10 CFR 50.44, “Standards for combustible gas control system in light-water-cooled power reactors” on October 16, 2003 to eliminate the requirements for hydrogen recombiners. The hydrogen recombiner Technical Specification requirements were subsequently removed by License Amendment 159, dated March 15, 2004. Regulatory Guide 1.7 was revised in March 2007 to reflect the amended 10 CFR 50.44. The Combustible Gas Control System (CGCS) has been retired in place with its electrical circuits de-energized and fluid process piping isolated from primary containment with redundant locked-closed isolation valves. Combustible gas control of the primary containment is provided by inerting the primary containment with nitrogen.

General Design Criterion 41 of 10 CFR 50, Appendix A, requires that systems be provided to control the concentration of hydrogen or oxygen and other substances that might potentially be released to the containment atmosphere. Title 10 CFR 50, Section 50.44, establishes the standards for these systems. In Fermi 2, no substances of a combustible nature (other than hydrogen and oxygen) would potentially be released in significant amounts to the containment atmosphere under LOCA conditions. To ensure that containment integrity is not potentially impaired due to buildup of combustible gases following a LOCA, Fermi 2 has an inert containment atmosphere with mixing capability. Hydrogen and oxygen concentrations are monitored. A purge system that uses the reactor building ventilation system or the SGTS is available. The purge system is not required to be a qualified system.

#### 6.2.5.1 Deleted

#### 6.2.5.2 System Design

##### 6.2.5.2.1 Deleted

#### 6.2.5.2.2 Design Features

The CGCS is retired, but all components remain in place as shown in the piping and instrumentation diagram in Figure 6.2-23. The primary containment isolation valves associated with the CGCS have been manufactured, fabricated, and tested in accordance with the requirements of the ASME B&PV Code Section III, Class 2, 1971 edition, summer 1973 addenda.

#### 6.2.5.2.3 Hydrogen/Oxygen Monitoring

Because Fermi 2 has an inerted primary containment atmosphere during reactor operation, the oxygen concentration, in the event of a LOCA, is the limiting parameter. The hydrogen and oxygen concentrations are continuously monitored, and are displayed in the main control room. Grab samples are obtained on a weekly basis to ensure the correct operation of the monitoring system. Samples are also taken prior to containment entry. Subsection 7.6.1.12 contains a description of the hydrogen/oxygen monitoring system. To ensure representative sampling, multiple ports allow gas to be drawn into the monitoring system from several locations in the containment. An alarm indicates when the oxygen concentration reaches a preset level.

#### 6.2.5.2.4 Deleted

#### 6.2.5.2.5 Containment Purge

Containment purge capability is provided for the purpose of removing fission product activity from the containment atmosphere and pressure control. Containment purge can also be utilized for combustible gas control following a significant beyond design-basis accident. Piping and valves are provided, connecting the containment atmospheres to the SGTS or reactor building heating, ventilation, and air conditioning (HVAC) system as shown in Figure 9.3-14. The purge system is comprised of the large purge piping used for purging and inerting and a smaller on-line purge system used for nitrogen vent/makeup and pressure control. Isolation valves and piping at the primary containment boundary meet the requirements of Section III ASME B&PV Code, Class 2, and are designed in conformance with Category I requirements. The SGTS treats the containment atmosphere prior to its release to the environment.

The drywell air purge inlet and vent outlet lines are 24 in. in diameter while the suppression chamber purge and vent lines are 20 in. in diameter. Both suppression chamber and drywell outboard isolation valves are supplied with a 6-in. bypass for use when the larger valve is to remain closed. The drywell bypass valve and suppression chamber bypass valve will isolate automatically.

During a power increase and drywell temperature increase, the drywell vent bypass line is opened periodically to maintain a constant drywell pressure. The drywell vent bypass line is also used to alleviate pressure buildup due to leakage from pneumatic solenoid valves. The purge system is not used during normal reactor operation to reduce airborne activity in the primary containment.

Containment vent line effluents are directed to the reactor building ventilation exhaust duct or to the SGTS for release. See Figure 6.2-20. The purge lines can open to the secondary containment volume, which is processed by the SGTS.

Because purging is initiated under the reactor operator's control, and the effluent from the SGTS is monitored for radioactivity, the incremental dose at the low-population zone during the purging will be controlled to ensure that the purge dose does not cause the total dose (LOCA plus purge dose) to exceed the limit specified in 10 CFR 100. High-radiation monitors prior to the reactor building HVAC exhaust fans isolate the containment purge valves and initiate the SGTS. The purge/inert valves comply with BTP CSB 6-4 of SRP 6.2.4. as follows:

a. The design basis for the valves includes the higher post-LOCA pressures

b. The operation of Fermi 2 containment purge and vent valves is in accordance with the Technical Specifications and is consistent with the guidance of the BTP for use of a single supply and exhaust line

c. The nitrogen purge supply valves for the torus and drywell are 6-in. and 10-in. valves, respectively. The exhaust line from both the torus and drywell are provided with 6-in. valves in parallel with the outboard isolation valve

d. Automatic isolation occurs on low reactor water level (level 2), high drywell pressure, or high radiation. The air-operated isolation valves fail closed on loss of air. The motor-operated isolation valves fail as is, but are only used in series with an air-operated isolation valve. Table 6.2-2 defines which of the above criteria are applicable to each specific isolation valve. The valves are listed in this table under penetration numbers X-25, X-26, X-205C, X-205D

e. The purge and vent valve closure times are consistent with the 5-sec requirement of the BTP

f. Debris screens have been provided for the purge valves inside the drywell to prevent debris from becoming entrained in the valves

g. The purge and vent system is not relied on for temperature and humidity control. The drywell cooling system is described in Subsection 9.4.5, and the vent/makeup of nitrogen for the containment is described in Subsection 9.3.6

h. Isolation valve testing of specific purge and inlet isolation valves is indicated in Table 6.2-2. The testing program is described in Subsection 6.2.4.4. The operability of the isolation function and the purge valve leakage rate are verified in accordance with the Technical Specifications

i. The radiological consequences of a LOCA while purging have been evaluated both specifically for Fermi 2, and generically by the NRC. Both a Fermi 2 specific analysis and the NRC's "Generic Evaluation of the Radiological Consequences of Accidents While Purging or Venting at Power-Multi-Plant Action Item B-24" indicate that while venting or purging at power, the dose contribution through open valves is small

j. The SGTS is downstream of the purge system isolation valves. Operation of the SGTS while purging will be limited and controlled to protect the SGTS from loss of function from the environment created by the escaping air and steam. The Technical Specifications delineate the limits on the use of the SGTS while purging or venting. This limit is further controlled by the Technical Specifications, which require that only one division of the SGTS be used.

k. Fermi 2 net positive suction head (NPSH) requirements for emergency core cooling system (ECCS) pumps are in conformance with Regulatory Guide 1.1. The Regulatory Guide allows no credit for positive containment pressure in the NPSH calculations. Therefore, a reduced containment pressure due to purging has no safety consequence on ECCS pump NPSH margins.

6.2.5.2.5.1 Hardened Torus Vent System

A hardened torus vent system has been installed at Fermi 2 under the 10 CFR 50.59 process in response to NRC Generic Letter 89-16, "Installation of Hardened Wetwell Vent".

During severe accidents which are outside the design basis, plant emergency procedures direct the operators to vent the wetwell airspace to prevent exceeding the primary containment pressure limit. Venting permits controlled releases by preventing permanent damage to the drywell. In addition, venting from the wetwell scrubs fission products from the effluent and reduces radioactive releases. The benefits of venting over a rupture of the drywell are reduced radiological consequences. The purpose of a hardened wetwell vent system is to provide a reliable design consistent with the safety objective of the plant emergency procedures.

The vent is sized to meet or exceed the BWR Owners Group (BWROG)/NRC general design criteria which require that under the conditions of (1) a constant heat input at a rate equal to 1.1 percent of rated thermal power and (2) containment pressure is equal to the primary containment pressure limit (PCPL), the exhaust flow through the vent is sufficient to prevent the containment pressure from increasing.

The hardened torus vent system consists of a 10-inch, Schedule 40, carbon-steel pipe routed from the 24-inch standby gas treatment system (SGTS) inlet header on the fifth floor Reactor Building through the Reactor Building siding into a stack which discharges at an elevated location. The 10-inch pipe contains two torus vent secondary containment fail closed isolation valves (TVSCIV), T4600F420 and T4600F421. The TVSCIVs air-operated butterfly valves (AOVs) are normally supplied by Division II non-interruptable control air supply (NIAS). The AC solenoid valves are normally powered by the reactor protection system (RPS) and divisionally separated. The inboard AOV is powered by Division I RPS and the outboard AOV is powered by the Division II RPS. Spectacle flanges, to facilitate maintenance of the AOVs, are installed upstream and downstream of the AOVs, with one outboard spectacle flange located outside the Reactor Building. Controls and position indications for the AOVs are located in the control room and are keylocked to prevent inadvertent positioning.

The piping from the first spectacle flange downstream from the existing header up to and including the second spectacle flange is Class D, QA Level I, and Seismic Category I. This is consistent with the original classification of SGTS. From the second spectacle flange through the remainder of the stack is QA Level 1M and Seismic II/I. The TVSCIVs maintain secondary containment integrity and are Class D, QA Level 1, Seismic Category I, and fail safe. The valves have been environmentally qualified to NUREG-0588 Category 2B (Mechanical) for pressure boundary integrity purposes. The leak tightness of the TVSCIVs is ensured by performing the secondary containment drawdown test at regular intervals.

Air supply for the primary containment isolation valves T4600F400, F401, and F412 in the SGTS has been changed from interruptable air supply (IAS) to Division 2 NIAS to improve venting reliability.

The pilot AC solenoid valves for the TVSCIVs are supplied by NIAS and are Class D, QA Level I, Seismic Category I, and have been environmentally qualified to NUREG-0588 Category 2B (Mechanical) and 2C (Electrical) to maintain the pressure boundary integrity of NIAS. The limit switches are QA Level non-Q, Seismic Category II/I.

A radiation monitor is installed on the 4$^{th}$ floor of the Auxiliary Building to enable monitoring of any radiological releases when the vent is open. The monitor is QA Level 1M, Seismic Category II/I, and has indication and alarm in the control center to alert the operators of a radiological release. The monitor also has an interface with the Integrated Plant Computer System (IPCS). Arrangement details are shown in Figure 11.4-4. The details of the radiation monitoring system are described in Subsection 11.4.3.11.3.

The torus hardened vent system components which require electrical power are the radiation monitor, solenoid valves, and the controls of the hardened vent air operated isolation valves. There are two TVSCI valves in series that are keylock switch controlled and fail closed. To preclude any inadvertent opening of the vent line to the atmosphere and jeopardizing secondary containment integrity due to a single failure, the two TVSCIV pilot solenoid valves are powered by different Divisions of RPS. The radiation monitor is powered as described in Subsection 11.4.3.11.3.

The hardened vent system is designed to be used for events that are outside the design basis of the plant. Therefore, the system does not comply with the design basis described in Subsection 6.2.5.1. The RPS power supply is selected to power the above components for reliable operation of the system. The RPS branch circuits feeding the hardened vent system components are adequately protected through properly coordinated safety grade fuses. Since RPS is a fail-safe system and the branch circuits used in the hardened vent system are properly protected, any single failure in the hardened vent system cannot prevent the RPS' ability to scram the reactor when it is needed. The power supply to each of these valves is divisionally separated and each valve control circuit is defeated through a normally open contact of a qualified keylocked selector switch; thus no single failure can inadvertently open the vent path nor can it prevent the ability of the RPS system from performing the scram action when it is needed. Furthermore, the RPS power to non-safety grade torus hardened vent system components is consistent with Fermi 2 design practices and by design any potential of full scram due to single failure or non-Q component failure in the hardened vent system is avoided. Therefore, the RPS system's intended design function to safely shut down the reactor is not compromised.

Beyond Design Basis Events:

As part of the response to the Fukushima Event, the NRC issued Order EA-13-109 and Interim Staff Guidance (ISG) JLD-ISG-2013-02 which requires Licensees:

a. Provide a reliable Hardened Containment Vent System (HCVS) to assist in preventing core damage when heat removal capability is lost.

b. Ensure that venting functions are also available during sever accident conditions. Sever accident conditions involving extensive core damage include elevated temperatures, pressures, radiation levels and combustible gas concentrations, such as hydrogen and carbon monoxide. This includes accidents involving a breach of the reactor vessel by molten core debris.

To comply with this order Fermi has modified the existing HCVS as follows:

a. Installed an alternate pneumatic gas supply to containment isolation valves, TVSCIVs and boundary valves associated with the Hardened Vent to allow control during and after a severe event.

b. Provided required panels, 130 VDC power supply, 120 VAC power supply, required instruments, and indications at panels outside and inside the control room.

c. Modified the HCVS exhaust stack to lengthen the stack, install a check valve to preclude backflow of air into the pip, and install a weather shroud.

During normal plant operation and during Design Basis Accidents the Hardened Containment Vent Equipment is de-energized and/or isolated. Upon declaration of a Hardened Containment Venting Scenario, the necessary hardened vent equipment is activated to support mitigation of the event. The intent being to address the station needs for the first 24 hours until the FLEX equipment can be brought on-line.

The provision of these modifications establishes alternate means of providing motive force (compressed gas) and electric power to assure the capability of the Hardened Vent to remain operable during and after a severe accident.

6.2.5.3 Safety Evaluation

The corrosion of containment materials was considered as a potential source of hydrogen. The corrosion of aluminum, zinc, and zinc-base paints located either in the drywell or torus was evaluated for a potential source of hydrogen. It was determined that these potential sources were insignificant for the following reasons:

1. The containment spray solution, if used, does not contain any chemical additives. The pH of the spray solution is 6.5 to 7.0

2. Aluminum corrosion is highly pH dependent. The Oak Ridge National Laboratory (ORNL) experiments described in Reference 29 have determined that at high pH (approximately 9.3), the corrosion of aluminum was about 100 times greater than at a pH 6.5 to 7.5, which was shown to be negligible

3. Although the corrosion of zinc does not exhibit the same pH dependence as aluminum, the corrosion of both zinc and aluminum is highly temperature dependent. The post-LOCA time/temperature profile in the drywell and torus is much less severe than that experienced in typical BWR

containments. The magnitude, as well as duration, of elevated temperature, is short-lived as shown in Figures 6.2-27, 6.2-29, 6.2-30, and 6.2-16

Because of these reasons, the corrosion of aluminum and zinc is relatively insignificant and does not represent a significant source of hydrogen.

6.2.5.4 Deleted

6.2.5.5 Deleted

6.2.5.6 Materials

There are no materials in the CGC system subject to radiolytic or pyrolytic decomposition under the conditions that would exist following a postulated LOCA. The principal materials used are

a. The heated components forming the containment boundary of the system are type 304 (or equivalent) stainless steel in accordance with the appropriate ASME material specifications, and Section III, Class 2 requirements

b. Unheated components forming the containment boundary conform with Section III, Class 2 of the ASME Code. Carbon steel, per SA-106, Grade B or SA-333, Grade 6, is used for piping, SA-216 for castings, and code allowable carbon steels for plate, forgings, weld rod, and other components, as appropriate.

6.2.6 Main Steam Isolation Valve Leakage Control System

Note: As a result of the re-analysis of the Loss-of-Coolant Accident (LOCA) using an Alternative Source Term (AST) methodology, it is no longer necessary to credit the Main Steam Isolation Valve Leakage Control System (MSIVLCS) for post-LOCA activity leakage mitigation.

6.2.6.6 Design of Main Steam System Piping and Valves

The main steam piping system, from the outboard MSIV to the appropriate anchor positions of all branch lines downstream of the third MSIV, is seismically qualified. The main portion of the main steam system is located in the turbine building, which is seismically qualified to withstand the effects of an operating-basis earthquake (OBE) or a safe-shutdown earthquake (SSE) event.

The main steam system has been seismically analyzed to ensure its integrity after either an OBE or an SSE event. The section of main steam piping analyzed begins at the anchor outside the primary containment and ends at the anchor in each of the branch lines downstream of the third MSIV. The seismic analysis of this portion of the main steam piping and included valves verifies that piping structural and pressure integrity will be maintained, and that included valves will remain in the elastic stress range after either an OBE or an SSE event.

6.2.6.6.1 Main Steam Lines

The main steam lines and branch connections downstream from the outboard containment isolation valve are classified as Group D, where these sections of pipes shall meet all pressure integrity requirements of Group D.

6.2.6.6.2 Valves in Branch Lines Connected To Main Steam Lines

The block valve(s) in branch lines connected to the main steam lines downstream of the outboard MSIV shall meet all the pressure integrity requirements of Group D.

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TABLE 6.2-1 CONTAINMENT PARAMETERS[a]

I. General Information

| | Drywell | Torus |
|---|---|---|
| A. Calculated peak pressure, Pa, psig | 49.9 | 28.3 |
| B. Maximum allowable pressure, psig | 62 | 62 |
| C. Design temperature, °F | 340 | 281 |
| D. Free volume, $ft^3$ | 163,730 | 130,900 |
| E. Design leak rate, Ld, percent/day | 0.5 | 0.5 |
| F. Maximum allowable leak rate, La, percent/day | 0.5 | 0.5 |

II. Initial Conditions Short-Term Analysis (M3CPT Code)

| INPUT PARAMETER | VALUE |
|---|---|
| Core Thermal Power, Mwt | 3,499 (102% of 3430 MWt) |
| RPV Dome Pressure, psia | 1,063 |
| Core Inlet Enthalpy, Btu/lbm | 531.1 |
| Initial Liquid Mass in RPV, lbm | 640,500 |
| Feedwater Addition to RPV | 0. |
| Drywell volume, $ft^3$ | 163,730 |
| Initial Drywell Pressure, psig | 0.75 |
| Initial Drywell Rel., Humidity, % | 20 |
| Initial Drywell Temperature, °F | 145 |
| Vent Flow Area, $ft^2$ | 240.9 |
| Vent Flow Loss Coefficient | 5.51 |
| Vent Submergence, ft | 3.33 |
| Suppression Pool Volume, $ft^3$ | 124,220 |

TABLE 6.2-1 CONTAINMENT PARAMETERS[a]

| INPUT PARAMETER | VALUE |
|---|---|
| Wetwell Airspace Volume, $ft^3$ | 127,760 |
| Suppression Pool Temperature, °F | 95 |
| Wetwell Airspace Pressure, psig | 0.75 |

III. Initial Conditions Long-Term Analysis (SUPERHEX Code)

| INPUT PARAMETER | SUPERHEX VALUE |
|---|---|
| Core Thermal Power, Mwt | 3,499<br>(102% of 3430 MWt) |
| Vessel Dome Pressure, psia | 1,063 |
| Feedwater Addition, lbm | 607,638 |
| Decay Heat | ANS/5.1 + 2σ |
| Drywell Free Volume, $ft^3$ | 163,730 |
| Supression Pool Volume, $ft^3$ | 117,161 |
| Initial Suppression Pool Temp, °F | 95 |
| Initial Wetwell Air Temp, °F | 95 |
| Initial Wetwell Relative Humidity, % | 100 |
| Wetwell Airspace Free Volume, $ft^3$ | 134,819 |
| RHR HXR K, Btu/sec - °F | 321 (original analysis, loss of one division of AC)<br>366 (loss of one division of RHRSW, only)<br>See Section 6.2.2.3 |
| RHR Service Water Temperature, °F | 80 – 90 |
| RHR Pump Heat, Hp | 2,100 |
| LPCS Pump Heat, HP | 1,600 |

TABLE 6.2-1 CONTAINMENT PARAMETERS[a]

| | |
|---|---|
| Time to turn on RHR, minutes | 20 |
| Initial Drywell Relative Humidity, % | 20 |
| Initial Drywell Pressure, psia | 15.45 |
| Initial Drywell Temperature, °F | 145 |
| Initial Wetwell Pressure, psia | 15.45 |
| IV. Engineered Safety Features Systems Information | |
| | Full Capacity |
| High-pressure coolant injection | |
| No. of pumps | 1 |
| No. of lines | 1 |
| Flow rate, gpm | 5,000 |
| Core spray | |
| No. of pumps | 4 |
| No. of lines | 2 |
| Flow rate (rated), gpm/line | 6,350 |
| No. of spargers | 2 |
| Low pressure coolant injection mode of RHR system | |
| No of pumps | 4 |
| No. of lines | 2 |
| Flow rate, gpm/line | 25,860 |
| Heat exchangers (RHR system) | |
| Type – inverted U-tube, single pass shell, multi-pass tubes, vertical mounting | |
| Number | 2 |

TABLE 6.2-1 CONTAINMENT PARAMETERS[a]

| | |
|---|---|
| Heat transfer areas, $ft^2$ | 7,320 |
| Overall heat transfer coefficient | 321 (original analysis – loss of one division of AC.)<br>366 (loss of one division of RHRSW, only.)<br>See Section 6.2.2.3 |
| Flow of pumps, gpm | |
| Shell-side | 10,000* with one RHR pump |
| Tube-side | 9,000** |
| Source of cooling water | RHR service water |
| Flow begins | Manual, approximately 1200 sec (20 minutes) |
| Automatic depressurization system | |
| Total number of safety/relieve valves | 15 |
| No. actuated on ADS | 5 |
| Drywell spray (RHR system) | |
| No. of pumps | 4 |
| No. of lines | 2 |
| Flow rate gpm/line | |
| 1 pump | 9,500 |
| Suppression pool spray (RHR system) | |
| No. of pumps | 4 |
| No of lines | 2 |
| No. of headers | 1 |
| Flow rate, gpm/line | |

TABLE 6.2-1 CONTAINMENT PARAMETERS[a]

| | |
|---|---|
| 1 pump | 500 |

[*] RHR heat exchanger performance maintained to assure credited overall heat transfer coefficient based on an RHR heat exchanger flow of 9250 gpm.

[**] RHRSW pump flow reduces below 9,000 gpm with time due to the RHR reservoir evaporative and drift losses.

| | |
|---|---|
| V. Assumptions Used in Pressure Transient Analysis | |
| Feedwater valve closure time | Instantaneous |
| MSIV closure time, seconds | 3.5 |
| Scram time, seconds | 1 |
| Liquid carryover, percent | 100 |
| VI. General Information for the Pressure Suppression Type Containment | |
| Drywell | Value |
| Maximum code allowable pressure, psig | 62 |
| Internal design pressure, psig | 56 |
| External design pressure, psig | 2 |
| Design temperature, °F | 340 |
| Suppression Pool | |
| Maximum code allowable pressure, psig | 62 |
| Internal design pressure, psig | 56 |
| External design pressure, psig | 2 |
| Design temperature, °F | 281 |
| Drywell free volume, including vent system (minimum), $ft^3$ | 163,730 |
| Suppression pool free (air) volume, $ft^3$ | |

TABLE 6.2-1 CONTAINMENT PARAMETERS[a]

| | |
|---|---|
| Analytic | 134,819 |
| Tech Spec | 130,900 |
| Suppression pool water volume, $ft^3$ | |
| Analytic | 117,161 |
| Tech Spec | 121,080 |
| Vent submergence, ft | |
| Minimum, ft | 3 |
| Maximum, ft | 3.33 |
| Vent loss coefficient | 5.51 |
| Pool cross sectional area, $ft^2$ | 731 |
| Pool depth (normal), ft | 14 ft 6 in. |
| No. of vents | 8 |
| Nominal vent diameter, ft | 6 |
| Nominal vent line area, $ft^2$ | 226 |
| No. of downcomers | 80 |
| Nominal downcomer diameter, ft | 2 |
| Drywell free volume/pressure suppression chamber free volume | 1.25 |
| Deleted | |
| Containment heat removal capability per loop, using 85°F service water and 165 °F pool temperature; 1 RHR and 2 service water pumps, Btu/hr | $66.5 \times 10^6$ |

VII. Recirculating Line Break Accident Initial Conditions and Calculated Response

| | Value |
|---|---|
| Effective accident break area (total), $ft^2$ | 4.1 |

TABLE 6.2-1 CONTAINMENT PARAMETERS[a]

| | |
|---|---|
| Componentsof effective break area | |
| Recirculation line (area), $ft^2$ | 3.5 |
| Equalizer line (area), $ft^2$ | N/A |
| RWCU line (area), $ft^2$ | 0.07 |
| Jet pumps (area), $ft^2$ | 0.55 |
| Break area/ vent area $\frac{4.1}{226}$ = | 0.018 |
| Reactor pressure vessel and attached piping initial liquid volume, $ft^3$ | 13,706 |
| Drywell | |
| Initial temperature, °F | 145 |
| Initial pressure, psig | 0.75 |
| Relative humidity, percent | 20 |
| Suppression pool | |
| Initial temperature, °F | 95 |
| Initial pressure, psig | 0.75 |
| Relative humidity, percent | 100 |
| RHR complex reservoir initial temperature, °F | 80 – 90[b] |
| Calculated peak drywell pressure, psig | 49.9 |
| Calculated drywell margin, percent | 19.5[c] |
| Calculated peak suppression pool pressure, psig | 28.3 |
| Calculated suppression pool margin, percent | 54.35c |
| Calculated peak deck differential pressure margin, psig | N/A |
| Calculated deck differential pressure margin, percent | N/A |
| Peak pool temperature during blowdown, °F | ≈135 |

TABLE 6.2-1 CONTAINMENT PARAMETERS[a]

| | |
|---|---|
| Long-term peak pool temperature from accident, °F (with degraded containment cooling system) | 196.5 |

[a] This list of parameter and results corresponds to those referred to in Subsection 6.2.1.2, Primary Containment System Design.

[b] RHR service water varies linearly from 80 °F to 90 °F over a period of 8 hours.

[c] Percent below maximum allowable pressure of 62 psig.

TABLE 6.2-2 SUMMARY OF PRIMARY CONTAINMENT PENETRATIONS AND ASSOCIATED ISOLATION VALVES

| PENETRATION DATA | | | | ISOLATION VALVE DATA | | | | | | | | | | | | | | | | | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| | | | | | | | | | | | | Valve Position | | | | | | | | | |
| Penetration Detail | P&ID Number | Design Criteria | Bypass Leakage Path | System Title | Valve Number | Valve Type | Actuator Type | Primary Actuation Mode | Secondary Actuation Mode | Containment Isolation Signal(s) | Accident Isolation Signal(s) | Normal | Shutdown | Post-LOCA (Accident) | Power Failure | ILRT | Engineered Safety Feature | Quality Group | Type C Test | Leak Detection | Remarks |
| X-1A | 6C721-2304 | -- | -- | Equipment Access Hatch | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | Type B Test |
| X-1B | 6C721-2304 | -- | -- | Equipment Access Hatch | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | Type B Test |
| X-2 | 6C721-2304 | -- | -- | Personnel Airlock | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | Type B Test |
| X-3 | -- | -- | -- | Drywell Head | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | Type B Test |
| X-5A | 6C721-2304 | -- | -- | Vent Pipe | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | Type A Test, Note 1 |
| X-5B | 6C721-2304 | -- | -- | Vent Pipe | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | Type A Test, Note 1 |
| X-5C | 6C721-2304 | -- | -- | Vent Pipe | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | Type A Test, Note 1 |
| X-5D | 6C721-2304 | -- | -- | Vent Pipe | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | Type A Test, Note 1 |
| X-5E | 6C721-2304 | -- | -- | Vent Pipe | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | Type A Test, Note 1 |
| X-5F | 6C721-2304 | -- | -- | Vent Pipe | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | Type A Test, Note 1 |
| X-5G | 6C721-2304 | -- | -- | Vent Pipe | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | Type A Test, Note 1 |
| X-5H | 6C721-2304 | -- | -- | Vent Pipe | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | Type A Test, Note 1 |
| X-6 | 6C721-2304 | -- | -- | Control Rod Drive Removal Hatch | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | Type B Test |

TABLE 6.2-2 SUMMARY OF PRIMARY CONTAINMENT PENETRATIONS AND ASSOCIATED ISOLATION VALVES

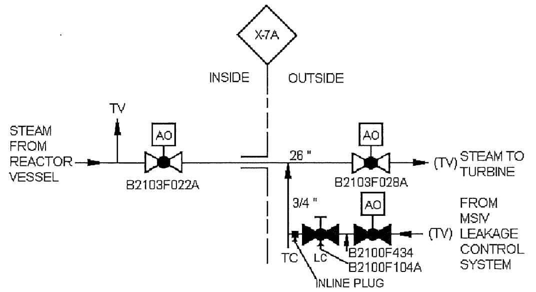
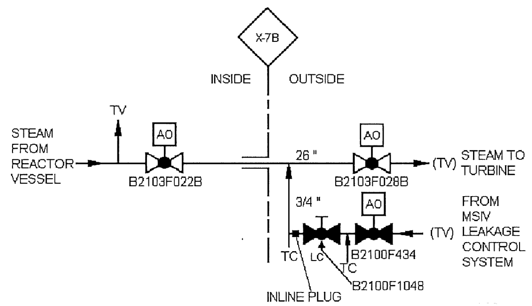
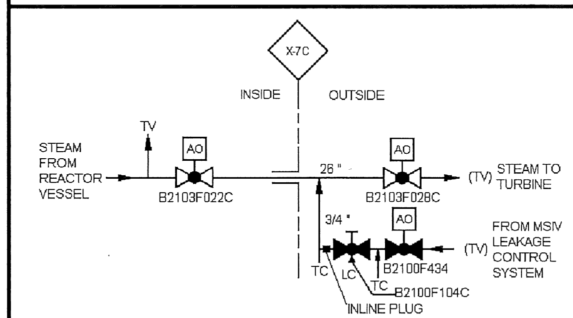
| PENETRATION DATA | | | | ISOLATION VALVE DATA | | | | | | | | | | | | | | | | | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| | | | | | | | | | | | | Valve Position | | | | | | | | | |
| Penetration Detail | P&ID Number | Design Criteria | Bypass Leakage Path | System Title | Valve Number | Valve Type | Actuator Type | Primary Actuation Mode | Secondary Actuation Mode | Containment Isolation Signal(s) | Accident Isolation Signal(s) | Normal | Shutdown | Post-LOCA (Accident) | Power Failure | ILRT | Engineered Safety Feature | Quality Group | Type C Test | Leak Detection | Remarks |
|   | 6M721-2089 | 55 | Yes | Main Steam Line A | B2103F022A (V17-2003) | GLB | AO | A | RM | A | E, F, G, J, P | O | C | C | C | C | No | A | Yes | Yes | Notes 2 and 3 |
| | | | | | B2103F028A (V17-2007) | GLB | AO | A | RM | A | E, F, G, J, P | O | C | C | C | C | No | A | Yes | Yes | Note 3 |
| | 6M721-3045 | -- | No | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | Note 42 |
|   | 6M721-2089 | 55 | Yes | Main Steam Line B | B2103F022B (V17-2001) | GLB | AO | A | RM | A | E, F, G, J, P | O | C | C | C | C | No | A | Yes | Yes | Notes 2 and 3 |
| | | | | | B2103F028B (V17-2005) | GLB | AO | A | RM | A | E, F, G, J, P | O | C | C | C | C | No | A | Yes | Yes | Note 3 |
| | 6M721-3045 | -- | No | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | **Note 42** |
|   | 6M721-2089 | 55 | Yes | Main Steam Line C | B2103F022C (V17-2002) | GLB | AO | A | RM | A | E, F, G, J, P | O | C | C | C | C | No | A | Yes | Yes | Notes 2 and 3 |
| | | | | | B2103F028C (V17-2006) | GLB | AO | A | RM | A | E, F, G, J, P | O | C | C | C | C | No | A | Yes | Yes | Note 3 |
| | 6M721-3045 | -- | No | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | Note 42 |
| | | | | | | | | | | | | | | | | | | | | | |
| | | | | | | | | | | | | | | | | | | | | | |

TABLE 6.2-2 SUMMARY OF PRIMARY CONTAINMENT PENETRATIONS AND ASSOCIATED ISOLATION VALVES

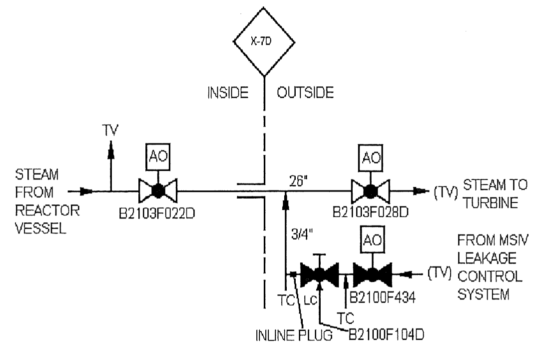
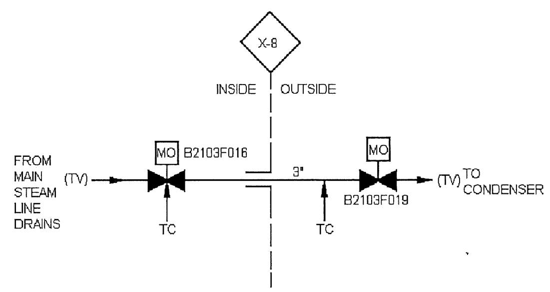
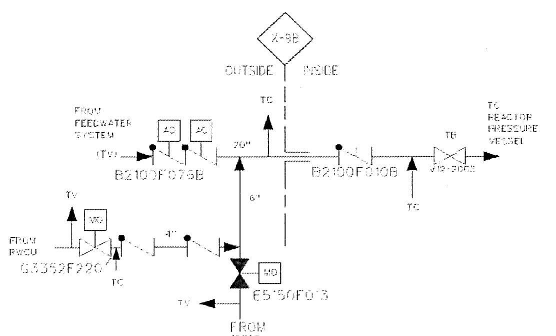
| PENETRATION DATA | | | | ISOLATION VALVE DATA | | | | | | | | | | | | | | | | | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| | | | | | | | | | | | | Valve Position | | | | | | | | | |
| Penetration Detail | P&ID Number | Design Criteria | Bypass Leakage Path | System Title | Valve Number | Valve Type | Actuator Type | Primary Actuation Mode | Secondary Actuation Mode | Containment Isolation Signal(s) | Accident Isolation Signal(s) | Normal | Shutdown | Post-LOCA (Accident) | Power Failure | ILRT | Engineered Safety Feature | Quality Group | Type C Test | Leak Detection | Remarks |
|   | 6M721-2089 | 55 | Yes | Main Steam Line D | B2103F022D (V17-2004) | GLB | AO | A | RM | A | E, F, G, J, P | O | C | C | C | C | No | A | Yes | Yes | Notes 2 and 3 |
| | | | | | B2103F028D (V17-2008) | GLB | AO | A | RM | A | E, F, G, J, P | O | C | C | C | C | No | A | Yes | Yes | Note 3 |
| | 6M721-3045 | -- | No | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | Note 42 |
|   | 6M721-2089 | 55 | Yes | Main Steam Line Drains | B2103F016 (V30-0259) | GAT | MO | A | RM | A | E, F, G, J, P | C | O | C | AIS | C | No | A | Yes | Yes | Note 4 |
| | | | | | B2103F019 | GAT | MO | A | RM | A | E, F, G, J, P | C | O | C | AIS | C | No | A | Yes | Yes | |
|  | 6M721-2023 | 55 | Yes | Feedwater | B2100F010B (V12-2007) | CHK | -- | RF | -- | -- | -- | O | C | C | -- | C | R | A | Yes | Yes | -- |
| | | | | | B2100F076B (V12-2001) | CHK | AO | RF | RM | -- | -- | O | C | C | C | C | R | A | Yes | Yes | Note 5 |
| | 6M721-2044 | 55 | Yes | Reactor Core Isolation Cooling | E5150F013 (V8-2228) | GAT | MO | RM | M | -- | Z | C | C | O | AIS | C | R | A | Yes | Yes | Note 6 |
| | 6M721-2046 | 55 | No | Reactor Water Clean-up | G3352F220 (V30-0322) | GAT | MO | A | RM | B | W | O | C | C | AIS | C | No | A | Yes | Yes | Note 35 |
| | | | | | | | | | | | | | | | | | | | | | |
| | | | | | | | | | | | | | | | | | | | | | |

TABLE 6.2-2 SUMMARY OF PRIMARY CONTAINMENT PENETRATIONS AND ASSOCIATED ISOLATION VALVES

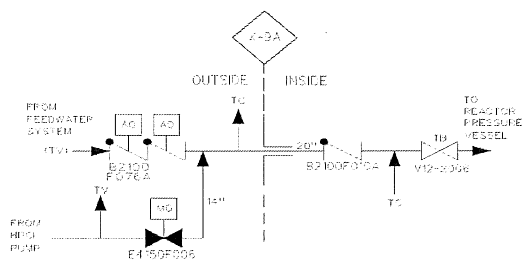
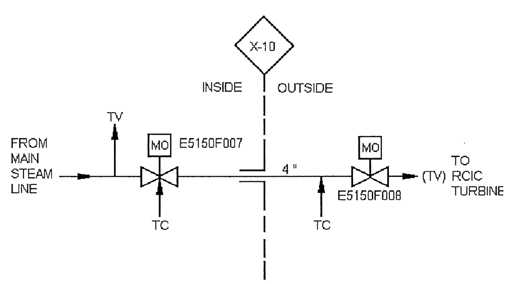
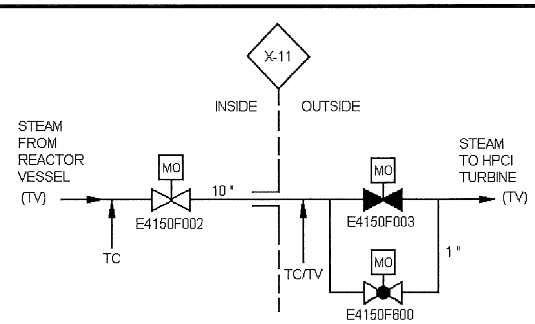
| PENETRATION DATA | | | | ISOLATION VALVE DATA | | | | | | | | Valve Position | | | | | | | | | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| Penetration Detail | P&ID Number | Design Criteria | Bypass Leakage Path | System Title | Valve Number | Valve Type | Actuator Type | Primary Actuation Mode | Secondary Actuation Mode | Containment Isolation Signal(s) | Accident Isolation Signal(s) | Normal | Shutdown | Post-LOCA (Accident) | Power Failure | ILRT | Engineered Safety Feature | Quality Group | Type C Test | Leak Detection | Remarks |
|  | 6M721-2023 | 55 | Yes | Feedwater | B2100F010A (V12-2008) | CHK | -- | RF | -- | -- | -- | O | C | C | -- | C | R | A | Yes | Yes | -- |
| | | | | | B2100F076A (V12-2002) | CHK | AO | RF | RM | -- | -- | O | C | C | C | C | R | A | Yes | Yes | Note 5 |
| | 6M721-2035 | 55 | Yes | High Pressure Coolant Injection | E4150F006 (V8-2194) | GAT | MO | RM | M | -- | Z | C | C | O | AIS | C | Yes | A | Yes | Yes | Note 7 |
|  | 6M721-2044 | 55 | Yes | Steam to Reactor Core Isolation Cooling Turbine | E5150F007 (V17-2030) | GAT | MO | RM | M | -- | Y | O | C | O | AIS | C | R | A | Yes | Yes | Notes 4, 6, and 31 |
| | | | | | E5150F008 (V17-2031) | GAT | MO | RM | M | -- | Y | O | C | O | AIS | C | R | A | Yes | Yes | Notes 6 and 31 |
|  | 6M721-2035 | 55 | Yes | Steam to High Pressure Coolant Injection Turbine | E4150F002 (V17-2020) | GAT | MO | RM | M | -- | X | O | C | O | AIS | C | Yes | A | Yes | Yes | Notes 4, 7, and 31 |
| | | | | | E4150F003 (V17-2021) | GAT | MO | RM | M | -- | X | C | C | O | AIS | C | Yes | A | Yes | Yes | Notes 4 and 7 |
| | | | | | E4150F600 (V17-2088) | GLB | MO | RM | M | -- | X | O | C | C | AIS | C | Yes | A | Yes | Yes | Notes 7 and 31 |
| | | | | | | | | | | | | | | | | | | | | | |
| | | | | | | | | | | | | | | | | | | | | | |

TABLE 6.2-2 SUMMARY OF PRIMARY CONTAINMENT PENETRATIONS AND ASSOCIATED ISOLATION VALVES

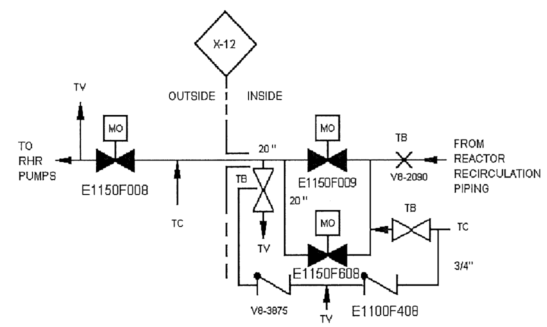
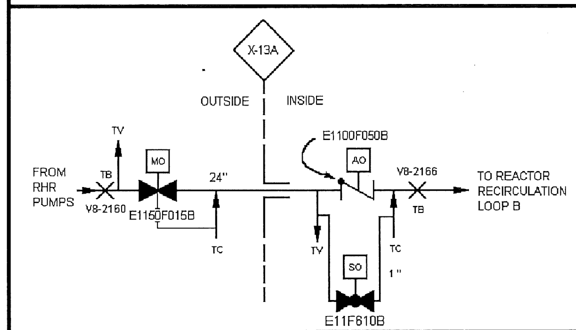
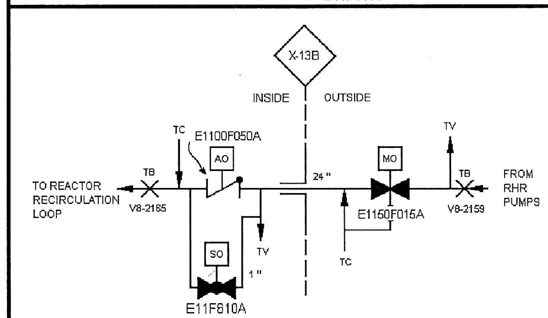
| PENETRATION DATA | | | | ISOLATION VALVE DATA | | | | | | | | | | | | | | | | | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| | | | | | | | | | | | | Valve Position | | | | | | | | | |
| Penetration Detail | P&ID Number | Design Criteria | Bypass Leakage Path | System Title | Valve Number | Valve Type | Actuator Type | Primary Actuation Mode | Secondary Actuation Mode | Containment Isolation Signal(s) | Accident Isolation Signal(s) | Normal | Shutdown | Post-LOCA (Accident) | Power Failure | ILRT | Engineered Safety Feature | Quality Group | Type C Test | Leak Detection | Remarks |
|  | 6M721-2083 | 55 | No | Residual Heat Removal Pump Suction From Recirculation Piping | E1150F009 (V8-2091) | GAT | MO | A | RM | C | L | C | O | C | AIS | C | No | A | Yes | No | Note 8 |
| | | | | | E1150F608 (V8-3407) | GAT | MO | RM | M | -- | -- | LC | LC | LC | AIS | C | No | A | Yes | No | Note 9 |
| | | | | | E1150F008 (V8-2092) | GAT | MO | A | RM | C | L | C | O | C | AIS | C | No | A | Yes | No | Note 8 |
| | | | | | E1100F408 (V8-3874) | CHK | SA | RF | -- | -- | -- | C | C | C | -- | C | No | A | Yes | | |
|  | 6M721-2083 | 55 | No | Residual Heat Removal Pump Discharge to Recirculation Loop | E1100F050B (V8-2164) | CHK | SA | RF | -- | -- | -- | C | O | O | -- | C | Yes | A | No | Yes | Note 36 |
| | | | | | E1150F015B (V8-2162) | GAT | MO | RM | M | -- | Z | C | O | O | AIS | C | Yes | A | No | Yes | Notes 7, 12, 37 and 38 |
| | | | | | E11F610B (V13-7688) | GLB | SO | RM | M | -- | -- | C | C | C | C | C | No | A | No | Yes | Notes 7 and 36 |
|  | 6M721-2084 | 55 | No | Residual Heat Removal Pump Discharge to Recirculation Loop | E1100F050A (V8-2163) | CHK | SA | RF | -- | -- | -- | C | O | O | -- | C | Yes | A | No | Yes | Note 36 |
| | | | | | E1150F015A (V8-2161) | GAT | MO | RM | M | -- | Z | C | O | O | AIS | C | Yes | A | No | Yes | Notes 7, 12, 37, and 38 |
| | | | | | E11F610A (V13-7687) | GLB | SO | RM | M | -- | -- | C | C | C | C | C | No | A | No | Yes | Notes 7 and 36 |
| X-14 | -- | -- | -- | Spare | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | Type A Test |
| | | | | | | | | | | | | | | | | | | | | | |
| | | | | | | | | | | | | | | | | | | | | | |

TABLE 6.2-2 SUMMARY OF PRIMARY CONTAINMENT PENETRATIONS AND ASSOCIATED ISOLATION VALVES

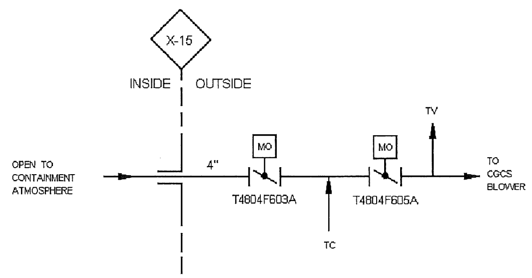
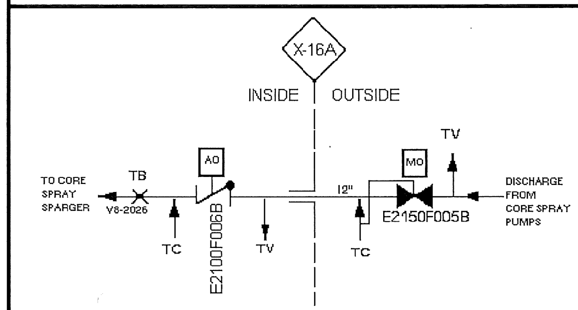
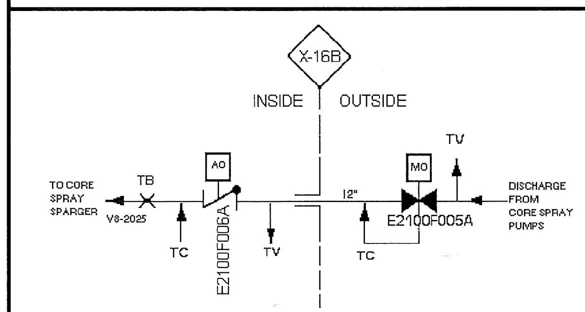
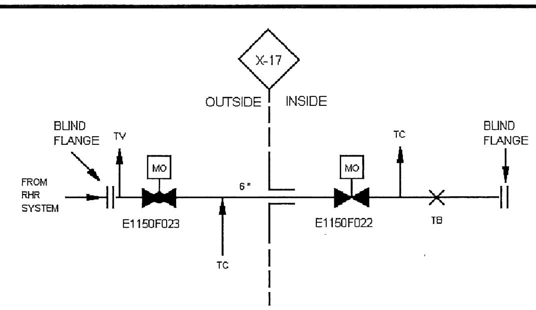
| PENETRATION DATA | | | | ISOLATION VALVE DATA | | | | | | | | | | | | | | | | | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| | | | | | | | | | | | | Valve Position | | | | | | | | | |
| Penetration Detail | P&ID Number | Design Criteria | Bypass Leakage Path | System Title | Valve Number | Valve Type | Actuator Type | Primary Actuation Mode | Secondary Actuation Mode | Containment Isolation Signal(s) | Accident Isolation Signal(s) | Normal | Shutdown | Post-LOCA (Accident) | Power Failure | ILRT | Engineered Safety Feature | Quality Group | Type C Test | Leak Detection | Remarks |
|  | 6M721-2087 | 56 | No | Combustible Gas Control System Suction | T4804F603A (V4-2144) | BFY | M | M | -- | -- | -- | LC | LC | LC | LC | LC | No | B | Yes | No | Notes 9, 10, 11, and 45 |
| | | | | | T4804F605A (V4-2154) | BFY | M | M | -- | -- | -- | LC | LC | LC | LC | LC | No | B | Yes | No | Notes 9 and 45 |
|  | 6M721-2034 | 55 | No | Core Spray Pump Discharge | E2100F006B (V8-2024) | CHK | SA | RF | -- | -- | -- | C | C | O | -- | C | Yes | A | Yes | Yes | |
| | | | | | E2150F005B (V8-2022) | GAT | MO | RM | M | -- | -- | C | C | O | AIS | C | Yes | A | Yes | Yes | Notes 7 and 38 |
|  | 6M721-2034 | 55 | No | Core Spray Pump Discharge | E2100F006A (V8-2023) | CHK | SA | RF | -- | -- | -- | C | C | O | -- | C | Yes | A | Yes | Yes | |
| | | | | | E2150F005A (V8-2021) | GAT | MO | RM | M | -- | -- | C | C | O | AIS | C | Yes | A | Yes | Yes | Notes 7 and 38 |
|  | 6M721-2083 | 56 | No | Residual Heat Removal Discharge to Head Spray | E1150F023 (V8-2171) | GLB | MO | A | RM | C | L | C | C | C | AIS | C | No | B | Yes | No | Notes 8 and 33 |
| | | | | | E1150F022 (V8-2172) | GAT | MO | A | RM | C | L | C | C | C | AIS | C | No | B | Yes | No | Notes 8 and 33 |
| | | | | | | | | | | | | | | | | | | | | | |

TABLE 6.2-2 SUMMARY OF PRIMARY CONTAINMENT PENETRATIONS AND ASSOCIATED ISOLATION VALVES

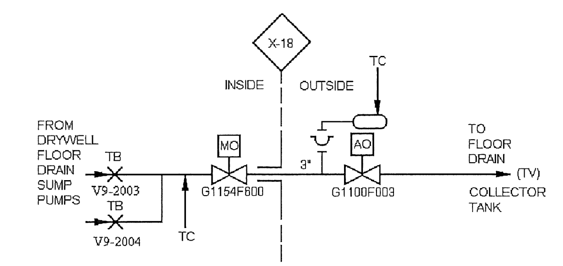
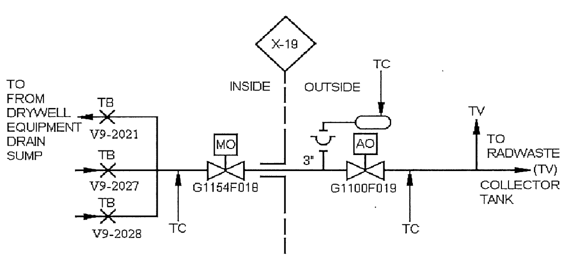
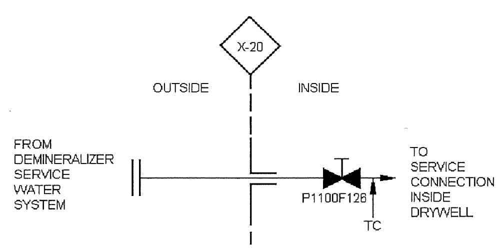
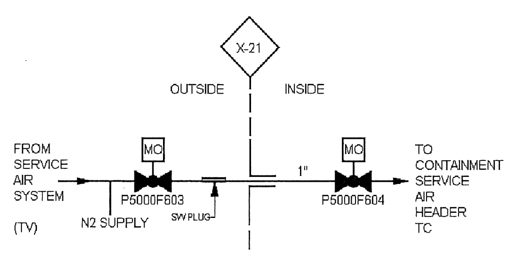
| PENETRATION DATA | | | | ISOLATION VALVE DATA | | | | | | | | | | | | | | | | | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| | | | | | | | | | | | | Valve Position | | | | | | | | | |
| Penetration Detail | P&ID Number | Design Criteria | Bypass Leakage Path | System Title | Valve Number | Valve Type | Actuator Type | Primary Actuation Mode | Secondary Actuation Mode | Containment Isolation Signal(s) | Accident Isolation Signal(s) | Normal | Shutdown | Post-LOCA (Accident) | Power Failure | ILRT | Engineered Safety Feature | Quality Group | Type C Test | Leak Detection | Remarks |
|  | 6M721-2032 | 56 | Yes | Drywell Floor Drain Sump Pump Discharge | G1154F600 (V9-2044) | GAT | MO | A | RM | C, K | -- | O | O | C | AIS | C | No | B | Yes | No | -- |
| | | | | | G1100F003 (V9-2005) | GAT | AO | A | RM | C, K | -- | O | C | C | C | C | No | B | Yes | No | -- |
|  | 6M721-2032 | 56 | Yes | Drywell Equipment Drain Sump Pump Discharge | G1154F018 | GAT | MO | A | RM | C, K | -- | O | O | C | AIS | C | No | B | Yes | No | -- |
| | | | | | G1100F019 (V9-2023) | GAT | AO | A | RM | C, K | -- | O | C | C | C | C | No | B | Yes | No | -- |
|  | 6M721-2678 | 56 | No | Demineralized Service Water to Drywell Connection | P1100F126 (V8-3120) | GAT | M | M | -- | -- | -- | LC | LC | LC | -- | LC | No | B | Yes | No | Note 9<br>Flange Type B Tested |
|  | 6M721-2085 | -- | No | Service Air to Drywell | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | Note 41<br>Type A Test |

TABLE 6.2-2 SUMMARY OF PRIMARY CONTAINMENT PENETRATIONS AND ASSOCIATED ISOLATION VALVES

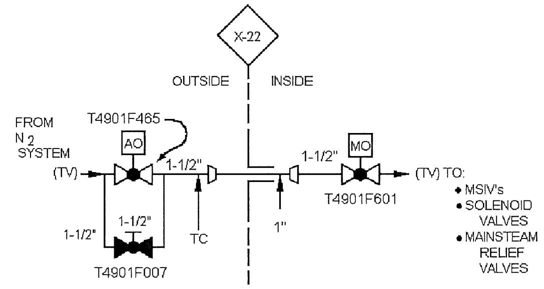
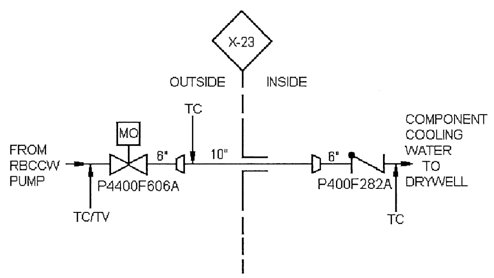
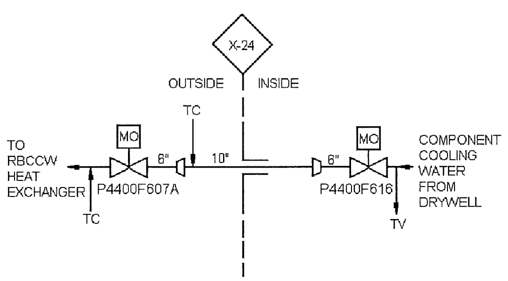
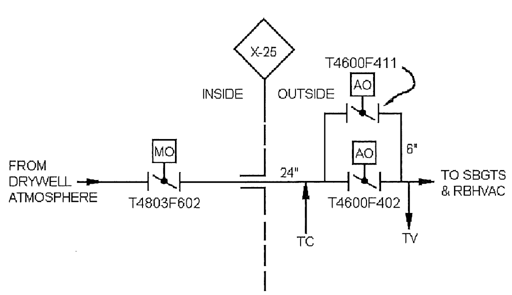
| PENETRATION DATA | | | | ISOLATION VALVE DATA | | | | | | | | | | | | | | | | | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| | | | | | | | | | | | | Valve Position | | | | | | | | | |
| Penetration Detail | P&ID Number | Design Criteria | Bypass Leakage Path | System Title | Valve Number | Valve Type | Actuator Type | Primary Actuation Mode | Secondary Actuation Mode | Containment Isolation Signal(s) | Accident Isolation Signal(s) | Normal | Shutdown | Post-LOCA (Accident) | Power Failure | ILRT | Engineered Safety Feature | Quality Group | Type C Test | Leak Detection | Remarks |
|  | 6M721-5007 | 56 | No | Nitrogen to Drywell | T4901F601 | GLB | MO | A | RM | B, K | -- | O | O | C | AIS | C | R | B | Yes | Yes | Notes 2 and 32 |
| | | | | | T4901F465 | GLB | AO | A | RM | B, K | -- | O | O | C | C | C | R | B | Yes | Yes | Note 32 |
| | | | | | T4901F007 | GLB | M | M | -- | -- | -- | LC | LC | LC | LC | LC | No | B | Yes | Yes | Note 9 |
|  | 6M721-5444 | 56 | Yes | Reactor Building Component Cooling Water / Emergency Equipment Cooling Water Supply | P4400F282A | CHK | SA | RF | -- | -- | -- | O | O | O | -- | O | Yes | B | Yes | Yes | -- |
| | | | | | P4400F606A | GAT | MO | RM | M | -- | -- | O | O | O | AIS | O | Yes | B | Yes | Yes | Note 34 |
|  | 6M721-5444 | 56 | Yes | Reactor Building Component Cooling Water / Emergency Equipment Cooling Water Return | P4400F616 | GAT | MO | RM | M | -- | -- | O | O | O | AIS | O | Yes | B | Yes | Yes | Note 4 |
| | | | | | P4400F607A | GAT | MO | RM | M | -- | -- | O | O | O | AIS | O | Yes | B | Yes | Yes | |
|  | 6M721-3445<br><br>7M721-2709 | 56 | No | Drywell Exhaust and Air Purge | T4803F602 | BFY | MO | A | RM | B, K, R | -- | C | O | C | AIS | C | R | B | Yes | No | Note 11 |
| | | | | | T4600F402 | BFY | AO | A | RM | B, K, R | -- | C | O | C | C | C | R | B | Yes | No | -- |
| | | | | | T4600F411 | BFY | AO | A | RM | B, K, R | -- | C | O | C | C | C | R | B | Yes | No | -- |

TABLE 6.2-2 SUMMARY OF PRIMARY CONTAINMENT PENETRATIONS AND ASSOCIATED ISOLATION VALVES

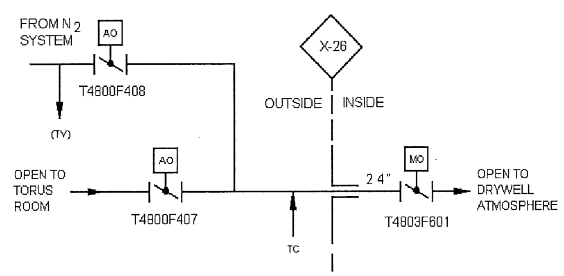
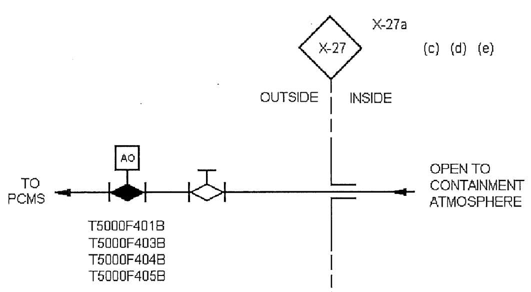
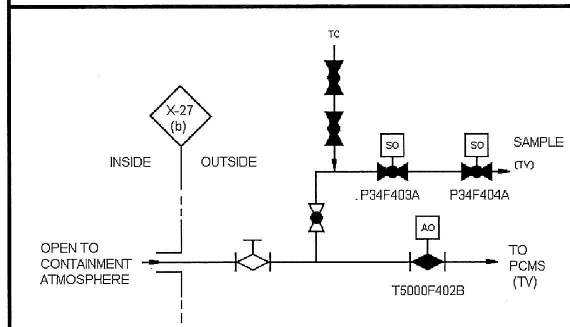
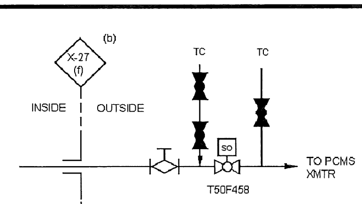
| PENETRATION DATA | | | | ISOLATION VALVE DATA | | | | | | | | | | | | | | | | | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| | | | | | | | | | | | | Valve Position | | | | | | | | | |
| Penetration Detail | P&ID Number | Design Criteria | Bypass Leakage Path | System Title | Valve Number | Valve Type | Actuator Type | Primary Actuation Mode | Secondary Actuation Mode | Containment Isolation Signal(s) | Accident Isolation Signal(s) | Normal | Shutdown | Post-LOCA (Accident) | Power Failure | ILRT | Engineered Safety Feature | Quality Group | Type C Test | Leak Detection | Remarks |
|  | 6M721-3445 | 56 | No | Drywell Air Purge Inlet | T4803F601 (VR3-3011) | BFY | MO | A | RM | B, K, R | -- | C | O | C | AIS | C | R | B | Yes | No | Note 11 |
| | | | | | T4800F407 (VR3-3012) | BFY | AO | A | RM | B, K, R | -- | C | O | C | C | C | R | B | Yes | No | -- |
| | 7M721-2709 | | | | T4800F408 (V4-2060) | BFY | AO | A | RM | B, K, R | -- | C | C | C | C | C | R | B | Yes | No | -- |
|  | 6I721-2679-1 | 56 | No | Containment Atmosphere Sample | Typical of Four T5000F401B (V5-2159) T5000F403B (V5-2161) T5000F404B (V5-2162) T5000F405B (V5-2163) | BAL | AO | RM | M | -- | -- | C | O | O | C | O | No | B | Yes | Yes | Notes 12 and 13 |
|  | 6I721-2679-1 | 56 | Yes | Containment Atmosphere Samples | T5000F402B (V5-2160) | BAL | AO | RM | M | -- | -- | C | O | O | C | O | No | B | Yes | Yes | Note 12 |
| | | | | | P34 F403A (V13-7364) | GLB | SO | RM | -- | -- | -- | C | C | C | C | C | No | B | Yes | Yes | Note 10 |
| | 6I721-2400-10 | 56 | Yes | | P34 F404A (V13-7374) | GLB | SO | RM | -- | -- | -- | C | C | C | C | C | No | B | Yes | Yes | Note 10 |
|  | 6I721-2679-1 | 56 | -- | Containment Atmosphere Monitoring System | T50F458 | GLB | SO | RM | -- | -- | -- | O | O | O | AIS | O | No | B | Yes | -- | Note 12 |

TABLE 6.2-2 SUMMARY OF PRIMARY CONTAINMENT PENETRATIONS AND ASSOCIATED ISOLATION VALVES

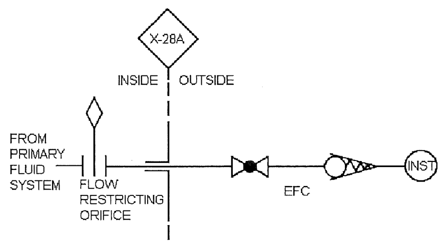
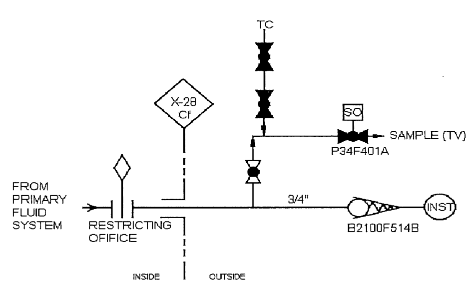
| PENETRATION DATA | | | | ISOLATION VALVE DATA | | | | | | | | | | | | | | | | | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| | | | | | | | | | | | | Valve Position | | | | | | | | | |
| Penetration Detail | P&ID Number | Design Criteria | Bypass Leakage Path | System Title | Valve Number | Valve Type | Actuator Type | Primary Actuation Mode | Secondary Actuation Mode | Containment Isolation Signal(s) | Accident Isolation Signal(s) | Normal | Shutdown | Post-LOCA (Accident) | Power Failure | ILRT | Engineered Safety Feature | Quality Group | Type C Test | Leak Detection | Remarks |
|  | 6M721-2090 | 55 | No | Jet Pump Flow Instrumentation | | EFC | SA | HF | -- | -- | -- | O | O | O | -- | O | No | A | No | Yes | Note 14, Type A Test |
| X-28B | -- | -- | -- | Spare | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | Type A Test |
| X-28C a,b,c,d,e | 6M721-2090 | 55 | No | Jet Pump Flow Instrumentation | Typical of Five | EFC | SA | HF | -- | -- | -- | O | O | O | -- | O | No | A | No | Yes | Note 14, Type A Test<br>See Penetration Detail X-28A |
|  | 6M721-2090 | 55 | No | Jet Pump Flow Instrumentation | B2100F514B (V13-2329) | EFC | SA | HF | -- | -- | -- | O | O | O | -- | O | No | A | No | Yes | Note 14, Type A Test |
| | 6I721-2400-10 | | Yes | Postaccident Pressurized Reactor Coolant Sample | P34 F401A | GLB | SO | RM | -- | -- | -- | C | C | C | C | C | No | B | Yes | Yes | -- |
| X-28D | 6M721-2090 2833 | 55 | | Jet Pump Instrumentation (1) and Recirculation Inlet ΔP (4) | Typical of Five | EFC | SA | HF | -- | -- | -- | O | O | O | -- | O | No | A | No | Yes | Notes 15 and 16, Type A Test<br>See Penetration Detail X-28A |
| X-28E | -- | -- | -- | Spare | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | Type A Test |
| X-28F | -- | -- | -- | Spare | -- | -- | -- | -- | -- | -- | -- | C | C | -- | C | -- | -- | -- | -- | -- | Type A Test |
| X-28G | -- | -- | -- | Spare | -- | -- | -- | -- | -- | -- | -- | C | C | -- | C | -- | -- | -- | -- | -- | Type A Test |
| | | | | | | | | | | | | | | | | | | | | | |

TABLE 6.2-2 SUMMARY OF PRIMARY CONTAINMENT PENETRATIONS AND ASSOCIATED ISOLATION VALVES

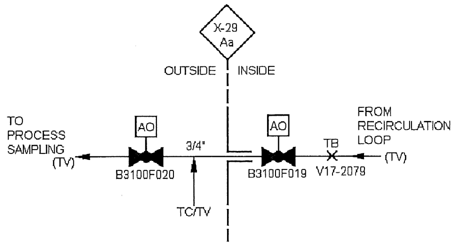
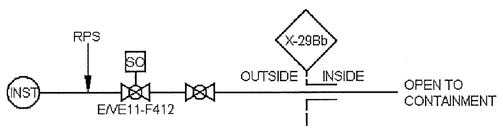
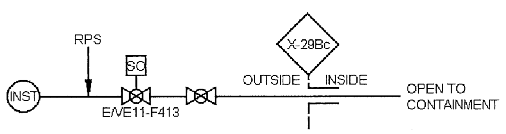
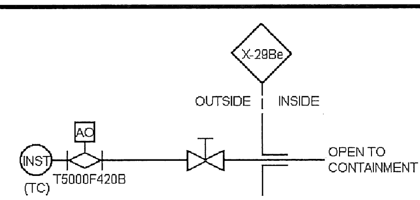
| PENETRATION DATA | | | | ISOLATION VALVE DATA | | | | | | | | | | | | | | | | | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| | | | | | | | | | | | | Valve Position | | | | | | | | | |
| Penetration Detail | P&ID Number | Design Criteria | Bypass Leakage Path | System Title | Valve Number | Valve Type | Actuator Type | Primary Actuation Mode | Secondary Actuation Mode | Containment Isolation Signal(s) | Accident Isolation Signal(s) | Normal | Shutdown | Post-LOCA (Accident) | Power Failure | ILRT | Engineered Safety Feature | Quality Group | Type C Test | Leak Detection | Remarks |
|  | 6M721-2833 | 55 | No | Reactor Water Sample | B3100F019 (V17-2077) | GLB | AO | A | RM | B, K | D | C | C | C | C | C | No | A | Yes | No | Note 2 |
| | | | | | B3100F020 | GLB | AO | A | RM | B, K | D | C | C | C | C | C | No | A | Yes | No | -- |
|  | 6M721-2083 | 56 | No | Reactor Protection System | E/VE11-F412 (V5-2546) | GLB | SO | RM | -- | -- | -- | O | O | O | AIS | O | No | B | Yes | Yes | Notes 2, 12, and 15 |
|  | 6M721-2083 | 56 | No | Reactor Protection System | E/VE11-F413 (V5-2547) | GLB | SO | RM | -- | -- | -- | O | O | O | AIS | O | No | B | Yes | Yes | Notes 2, 12, and 15 |
|  | 6M721-2083<br>6I721-2679-1 | 56 | No | Drywell Instrumentation | T5000F420B (V5-2231) | BAL | AO | RM | M | -- | -- | O | O | O | C | O | No | B | Yes | Yes | Notes 12, 13, 15, and 43 |
| X-29Bd | 6M721-2090 | 55 | No | Reactor Pressure Vessel Level Instrumentation | B2100F509 (V13-2320) | EFC | SA | HF | -- | -- | -- | O | O | O | -- | O | No | A | No | Yes | Note 16<br>See Penetration Detail X-28A |
| X-30A | 6M721-2833<br>2046 | 55 | No | RPV Pressure (1), Recirculation Pressure (1), and Recirculation Loop Flow (2) | Typical of Four | EFC | SA | HF | -- | -- | -- | O | O | O | -- | O | No | A | No | Yes | Notes 15 and 16<br>See Penetration Detail X-28A |
| X-30B | 6M721-2833 | 55 | No | Recirculation Pump Inst. (2), Recirculation Pump ΔP (2), Recirculation Loop Flow (2) | Typical of Six | EFC | SA | HF | -- | -- | -- | O | O | O | -- | O | No | A | No | Yes | Note 16<br>See Penetration Detail X-28A |
| X-31A | -- | -- | -- | Spare | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | Type A Test |
| | | | | | | | | | | | | | | | | | | | | | |

TABLE 6.2-2 SUMMARY OF PRIMARY CONTAINMENT PENETRATIONS AND ASSOCIATED ISOLATION VALVES

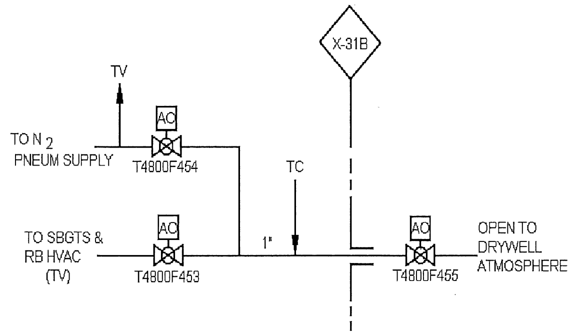
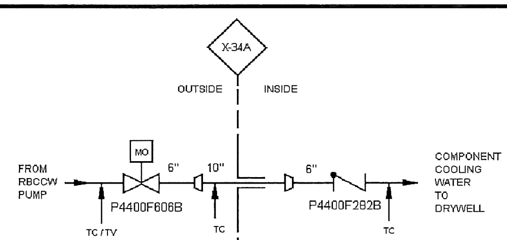
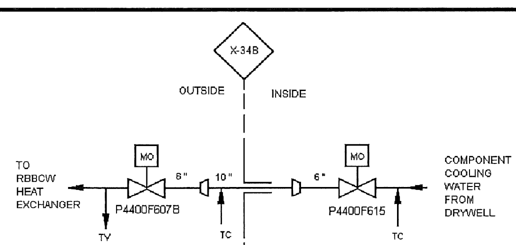
| PENETRATION DATA | | | | ISOLATION VALVE DATA | | | | | | | | | | | | | | | | | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| | | | | | | | | | | | | Valve Position | | | | | | | | | |
| Penetration Detail | P&ID Number | Design Criteria | Bypass Leakage Path | System Title | Valve Number | Valve Type | Actuator Type | Primary Actuation Mode | Secondary Actuation Mode | Containment Isolation Signal(s) | Accident Isolation Signal(s) | Normal | Shutdown | Post-LOCA (Accident) | Power Failure | ILRT | Engineered Safety Feature | Quality Group | Type C Test | Leak Detection | Remarks |
|   | 6M721-3445 | 56 | No | Drywell On-Line Pressure Control | T4800F455 | GLB | AO | A | RM | B, K, R | -- | O | C | C | C | C | R | B | Yes | No | Note 2 |
| | | | | | T4800F454 | GLB | AO | A | RM | B, K, R | -- | O | C | C | C | C | R | B | Yes | No | -- |
| | | | | | T4800F453 | GLB | AO | A | RM | B, K, R | -- | O | C | C | C | C | R | B | Yes | No | -- |
| X-32A | 6M721-2090 | 55 | No | Reactor Pressure Vessel Pressure | B2100F516C | EFC | SA | HF | -- | -- | -- | O | O | O | -- | O | No | A | No | Yes | Notes 15 and 16 See Penetration Detail X-28A |
| X-32B | 6M721-2833 2035 | 55 | No | Steam Flow to High-Pressure Coolant Injection (2) and Recirculation Loop Flow (4) | Typical of Six | EFC | SA | HF | -- | -- | -- | O | O | O | -- | O | No | A | No | Yes | Note 16 See Penetration Detail X-28A |
| X-33A | 6M721-2833 | 55 | No | Recirculation Pump ΔP (2), Recirculation Pump Inst. (2), and Recirculation Pressure (1) | Typical of Five | EFC | SA | HF | -- | -- | -- | O | O | O | -- | O | No | A | No | Yes | Notes 15 and 16 See Penetration Detail X-28A |
| X-33B | 6M721-2090 2035 2034 | 55 | No | RPV Pressure (1), Steam Flow to High Pressure Coolant Injection (2), and Feedwater Pressure (1) | Typical of Four | EFC | SA | HF | -- | -- | -- | O | O | O | -- | O | No | A | No | Yes | Notes 15 and 16 See Penetration Detail X-28A |
|   | 6M721-5357 | 56 | Yes | Reactor Building Component Cooling Water/Emergency Equipment Cooling Water Supply | P4400F282B | CHK | SA | RF | -- | -- | -- | O | O | O | -- | O | Yes | B | Yes | Yes | -- |
| | | | | | P4400F606B | GAT | MO | RM | M | -- | | O | O | O | AIS | O | Yes | B | Yes | Yes | Note 34 |
|   | 6M721-5357 | 56 | Yes | Reactor Building Component Cooling Water/Emergency Equipment Cooling Water Return | P4400F615 | GAT | MO | RM | M | -- | | O | O | O | AIS | O | Yes | B | Yes | Yes | Note 4 |
| | | | | | P4400F607B | GAT | MO | RM | M | -- | | O | O | O | AIS | O | Yes | B | Yes | Yes | |
| X-35 A, B, C, D, E, F, G | 6I721-2837-6 | 54 | No | TIP System Flanges | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | No | -- | No | Yes | Type B Test |

TABLE 6.2-2 SUMMARY OF PRIMARY CONTAINMENT PENETRATIONS AND ASSOCIATED ISOLATION VALVES

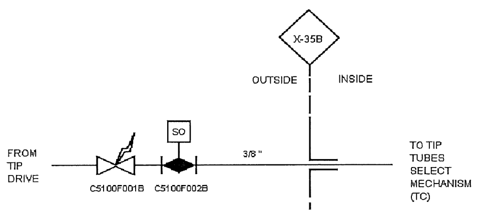
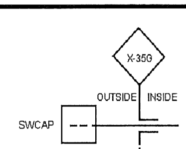
| PENETRATION DATA | | | | ISOLATION VALVE DATA | | | | | | | | | | | | | | | | | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| | | | | | | | | | | | | Valve Position | | | | | | | | | |
| Penetration Detail | P&ID Number | Design Criteria | Bypass Leakage Path | System Title | Valve Number | Valve Type | Actuator Type | Primary Actuation Mode | Secondary Actuation Mode | Containment Isolation Signal(s) | Accident Isolation Signal(s) | Normal | Shutdown | Post-LOCA (Accident) | Power Failure | ILRT | Engineered Safety Feature | Quality Group | Type C Test | Leak Detection | Remarks |
| X-35A | 61721-2837-6 | -- | -- | TIP System (Spare) | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | Type A Test |
|  | 61721-2837-6 | 54 | No | TIP System | C5100F002B (C5102J004B) | BAL | SO | -- | -- | C, K | -- | C | C | C | C | C | No | -- | Yes | Yes | Note 17 |
| | | | | | C5100F001B | SHR | EX | RM | -- | -- | -- | O | O | O | O | O | No | -- | No | Yes | Note 17 |
| X-35C | 61721-2837-6 | 54 | No | TIP System | C5100F002A (C5102J004A) | BAL | SO | -- | -- | C, K | -- | C | C | C | C | C | No | -- | Yes | Yes | Note 17 |
| | | | | | C5100F001A | SHR | EX | RM | -- | | -- | O | O | O | O | O | No | -- | No | Yes | See Penetration Detail X-35B |
| X-35D | 61721-2837-6 | 54 | No | TIP System | C5100F002C (C5102J004C) | BAL | SO | -- | -- | C, K | -- | C | C | C | C | C | No | B | Yes | Yes | Note 17 |
| | | | | | C5100F001C | SHR | EX | RM | -- | | -- | O | O | O | O | O | No | B | No | Yes | See Penetration Detail X-35B |
| X-35E | 61721-2837-6 | 54 | No | TIP System | C5100F002E (C5102J004E) | BAL | SO | -- | -- | C, K | -- | C | C | C | C | C | No | B | Yes | Yes | Note 17 |
| | | | | | C5100F001E | SHR | EX | RM | -- | | -- | O | O | O | O | O | No | B | No | Yes | See Penetration Detail X-35B |
| X-35F | 61721-2837-6 | 54 | No | TIP System | C5100F002D (C5102J004D) | BAL | SO | -- | -- | C, K | -- | C | C | C | C | C | No | B | Yes | Yes | Note 17 |
| | | | | | C5100F001D | SHR | EX | RM | -- | | -- | O | O | O | O | O | No | B | No | Yes | See Penetration Detail X-35B |
|  | 61721-2837-6 | -- | No | TIP System (Spare) | -- | -- | -- | -- -- | -- -- | -- | -- -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | Note 18<br>Type A Test |
| | | | | | | | | | | | | | | | | | | | | | |

TABLE 6.2-2 SUMMARY OF PRIMARY CONTAINMENT PENETRATIONS AND ASSOCIATED ISOLATION VALVES

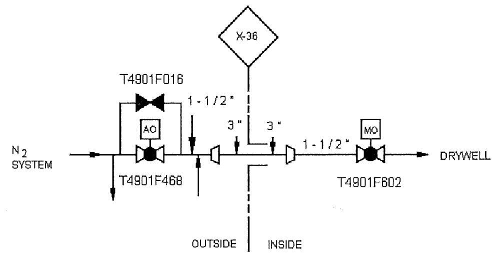
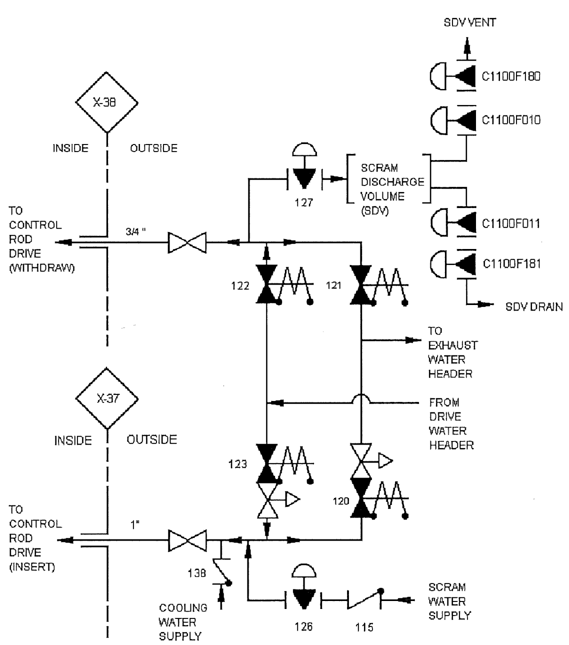
| PENETRATION DATA | | | | ISOLATION VALVE DATA | | | | | | | | | | | | | | | | | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| | | | | | | | | | | | | Valve Position | | | | | | | | | |
| Penetration Detail | P&ID Number | Design Criteria | Bypass Leakage Path | System Title | Valve Number | Valve Type | Actuator Type | Primary Actuation Mode | Secondary Actuation Mode | Containment Isolation Signal(s) | Accident Isolation Signal(s) | Normal | Shutdown | Post-LOCA (Accident) | Power Failure | ILRT | Engineered Safety Feature | Quality Group | Type C Test | Leak Detection | Remarks |
|   | 6M721-5007 | 56 | No | Nitrogen to Drywell | T4901F468 (V4-2187) | GLB | AO | A | RM | B, K | -- | O | O | C | C | C | Yes | B | Yes | Yes | Note 32 |
| | | | | | T4901F602 (V4-2188) | GLB | MO | A | RM | B, K | -- | O | O | C | AIS | C | Yes | B | Yes | Yes | Notes 2 and 32 |
| | | | | | T4901F016 (V8-4140) | GLB | M | -- | -- | -- | -- | LC | LC | LC | LC | LC | No | B | Yes | Yes | Note 9 |
|   | 6M721-2081 | 54 | No | Control Rod Drive Insert and Withdrawal Lines | -- | BCK | SA | RF | -- | -- | -- | O | O | C | -- | O | Yes | B | No | No | Note 19 |
| | | | | | 115 | BCK | SA | RF | -- | -- | -- | O | C | C | C | C | Yes | B | No | No | This Information Applies to Penetrations X-37 (A, B, C, D) and X-38 (A, B, C, D) |
| | | | | | 121 | GAT | SO | A | RM | -- | -- | C | C | C | C | C | Yes | B | No | No | |
| | | | | | 123 | GAT | SO | A | RM | -- | -- | C | C | C | C | C | Yes | B | No | No | |
| | | | | | 120 | GAT | SO | A | RM | -- | -- | C | C | C | C | C | Yes | B | No | No | |
| | | | | Typical of 185 Units | 122 | GAT | SO | A | RM | -- | -- | C | C | C | C | C | Yes | B | No | No | |
| | | | | | 126 | GLB | AO | A | RM | -- | -- | C | C | O | O | O | Yes | B | No | No | |
| | | | | | 127 | GLB | AO | A | RM | -- | -- | C | C | O | O | O | Yes | B | No | No | |
| | | | | | 138 | BCK | SA | RF | -- | -- | -- | O | C | C | C | C | Yes | B | No | No | |
| | | | | | C1100F010 (V8-2073) | REG | AO | A | RM | | | O | O | C | C | C | Yes | B | Yes | No | |
| | | | | | C1100F011 | REG | AO | A | RM | | | O | O | C | C | C | Yes | B | Yes | No | |
| | | | | | C1100F180 (V8-3876) | REG | AO | A | RM | | | O | O | C | C | C | Yes | B | Yes | No | |
| | | | | | C1100F181 | REG | AO | A | RM | | | O | O | C | C | C | Yes | B | Yes | No | |
| | | | | | | | | | | | | | | | | | | | | | |
| | | | | | | | | | | | | | | | | | | | | | |

TABLE 6.2-2 SUMMARY OF PRIMARY CONTAINMENT PENETRATIONS AND ASSOCIATED ISOLATION VALVES

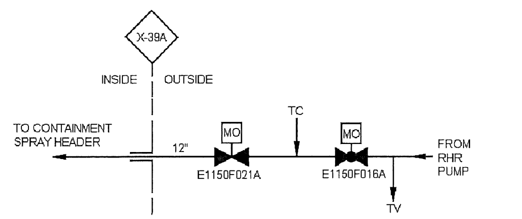
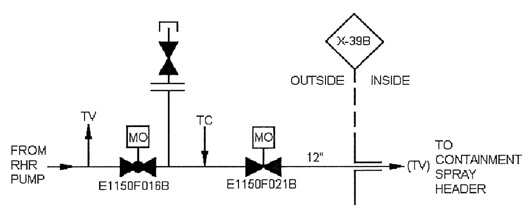
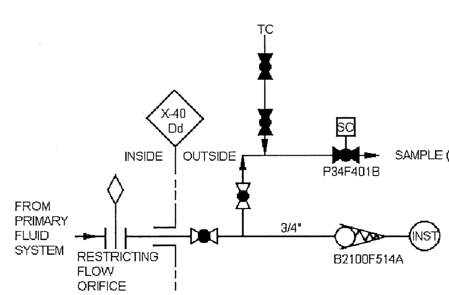
| PENETRATION DATA | | | | ISOLATION VALVE DATA | | | | | | | | | | | | | | | | | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| | | | | | | | | | | | | Valve Position | | | | | | | | | |
| Penetration Detail | P&ID Number | Design Criteria | Bypass Leakage Path | System Title | Valve Number | Valve Type | Actuator Type | Primary Actuation Mode | Secondary Actuation Mode | Containment Isolation Signal(s) | Accident Isolation Signal(s) | Normal | Shutdown | Post-LOCA (Accident) | Power Failure | ILRT | Engineered Safety Feature | Quality Group | Type C Test | Leak Detection | Remarks |
|  | 6M721-2084 | 56 | No | Residual Heat Removal to Containment Spray Header | E1150F021A (V8-2169) | GAT | MO | A | RM | A, K | -- | C | C | O | AIS | C | Yes | B | Yes | No | Notes 4, 10, 20 and 21 |
| | | | | | E1150F016A (V8-2167) | GLB | MO | A | RM | A, K | -- | C | C | O | AIS | C | Yes | B | Yes | No | Note 20 |
|  | 6M721-2083 | 56 | No | Residual Heat Removal to Containment Spray Header | E1150F021B (V8-2170) | GAT | MO | A | RM | A, K | -- | C | C | O | AIS | C | Yes | B | Yes | No | Notes 4, 10, 20 and 21<br>Note 20 |
| | | | | | E1150F016B (V8-2168) | GLB | MO | A | RM | A, K | -- | C | C | O | AIS | C | Yes | B | Yes | No | Flange to be Type B Tested |
| X-40A | 6M721-2833 2090 | 55 | No | Recirculation Inlet ΔP (4) and Reactor Pressure Vessel Pressure (2) | Typical of Six | EFC | SA | HF | -- | -- | -- | O | O | O | -- | O | No | A | No | Yes | Note 16<br>See Penetration Detail X-28A |
| X-40B | 6M721-2089 2090 | 55 | No | Reactor Pressure Vessel Pressure (1) and Main Steam Flow (4) | Typical of Five | EFC | SA | HF | -- | -- | -- | O | O | O | -- | O | No | A | No | Yes | Notes 15 and 16<br>See Penetration Detail X-28A |
| X-40C | 6M721-2090 | 55 | No | Jet Pump Flow Instrumentation | Typical of Six | EFC | SA | HF | -- | -- | -- | O | O | O | -- | O | No | A | No | Yes | Note 14<br>See Penetration Detail X-28A |
| X-40D a,b,c,e,f | 6M721-2090 | 55 | No | Jet Pump Flow Instrumentation | Typical of Five | EFC | SA | HF | -- | -- | -- | O | O | O | -- | O | No | A | No | Yes | Note 14<br>See Penetration Detail X-28A |
|  | 6M721-2090 | 55 | Yes | Jet Pump Flow Instrumentation and Postaccident Reactor Coolant Sample | B2100F514A (V13-2328) | EFC | SA | HF | -- | -- | -- | O | O | O | -- | O | No | A | No | Yes | Note 14 |
| | 6I721-2400-10 | | | | P34F401B | GLB | SO | RM | -- | -- | -- | C | C | C | C | C | No | B | Yes | Yes | -- |
| | | | | | | | | | | | | | | | | | | | | | |
| X-41 | -- | -- | -- | Spare | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | Type A Test |

TABLE 6.2-2 SUMMARY OF PRIMARY CONTAINMENT PENETRATIONS AND ASSOCIATED ISOLATION VALVES

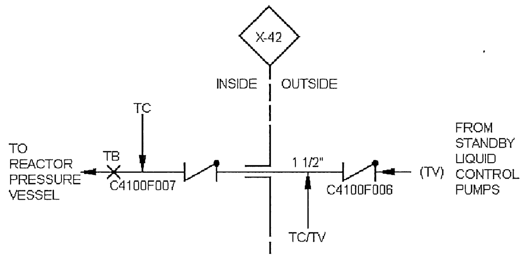
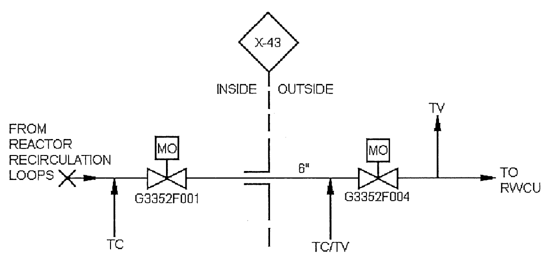
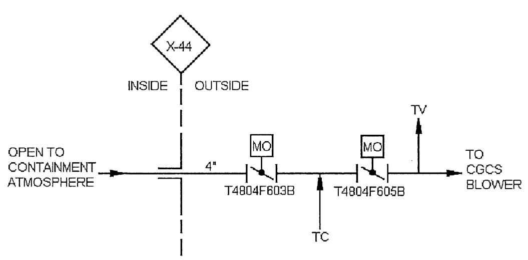
| PENETRATION DATA | | | | ISOLATION VALVE DATA | | | | | | | | | | | | | | | | | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| | | | | | | | | | | | | Valve Position | | | | | | | | | |
| Penetration Detail | P&ID Number | Design Criteria | Bypass Leakage Path | System Title | Valve Number | Valve Type | Actuator Type | Primary Actuation Mode | Secondary Actuation Mode | Containment Isolation Signal(s) | Accident Isolation Signal(s) | Normal | Shutdown | Post-LOCA (Accident) | Power Failure | ILRT | Engineered Safety Feature | Quality Group | Type C Test | Leak Detection | Remarks |
|   | 6M721-2082 | 55 | No | Standby Liquid Control | C4100F007 (VR4-2037) | CHK | SA | RF | -- | -- | -- | C | C | C | -- | C | R | A | Yes | No | -- |
| | | | | | C4100F006 (VR4-2036) | CHK | SA | RF | -- | -- | -- | C | C | C | -- | C | R | A | Yes | No | Note 22 |
|   | 6M721-2046 | 55 | No | Reactor Water (Cleanup From Recirculation Piping) | G3352F001 (V8-2252) | GAT | MO | A | RM | B | W | O | O | C | AIS | C | No | A | Yes | Yes | Alternate-Note 4 |
| | | | | | G3352F004 (V8-2253) | GAT | MO | A | RM | B | W | O | O | C | AIS | C | No | A | Yes | Yes | Note 35 |
|   | 6M721-2087 | 56 | No | Combustible Gas Control System Suction | T4804F603B (V4-2143) | BFY | M | M | -- | -- | -- | LC | LC | LC | LC | LC | No | B | Yes | No | Notes 9, 10, 11, and 45 |
| | | | | | T4804F605B (V4-2153) | BFY | M | M | -- | -- | -- | LC | LC | LC | LC | LC | No | B | Yes | No | Notes 9 and 45 |
| X-45 | -- | -- | -- | Spare | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | Type A Test |
| X-46A | 6M721-2089 | 55 | No | Main Steam Flow | Typical of Four | EFC | SA | HF | -- | -- | -- | O | O | O | -- | O | No | A | No | Yes | Notes 15 and 16 See Penetration Detail X-28A |
| X-46B | 6M721-2089 | 55 | No | Main Steam Flow | Typical of Four | EFC | SA | HF | | -- | -- | O | O | O | -- | O | No | A | No | Yes | Notes 15 and 16 See Penetration Detail X-28A |
| | | | | | | | | | | | | | | | | | | | | | |

TABLE 6.2-2 SUMMARY OF PRIMARY CONTAINMENT PENETRATIONS AND ASSOCIATED ISOLATION VALVES

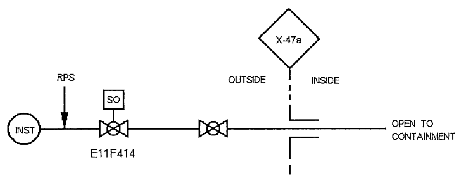
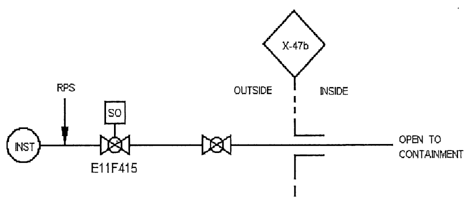
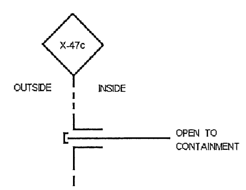
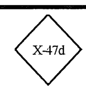
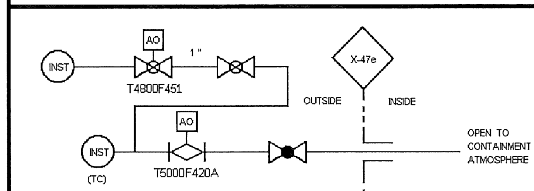
| PENETRATION DATA | | | | ISOLATION VALVE DATA | | | | | | | | | | | | | | | | | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| | | | | | | | | | | | | Valve Position | | | | | | | | | |
| Penetration Detail | P&ID Number | Design Criteria | Bypass Leakage Path | System Title | Valve Number | Valve Type | Actuator Type | Primary Actuation Mode | Secondary Actuation Mode | Containment Isolation Signal(s) | Accident Isolation Signal(s) | Normal | Shutdown | Post-LOCA (Accident) | Power Failure | ILRT | Engineered Safety Feature | Quality Group | Type C Test | Leak Detection | Remarks |
|  | 6M721-2084 | 56 | No | Reactor Protection System | E11F414 (V5-2548) | GLB | SO | RM | -- | -- | -- | O | O | O | AIS | O | No | B | Yes | Yes | Notes 2, 12, and 15 |
|  | 6M721-2084 | 56 | No | Reactor Protection System | E11F415 (V5-2549) | GLB | SO | RM | -- | -- | -- | O | O | O | AIS | O | No | B | Yes | Yes | Notes 2, 12, and 15 |
|  | 6M721-3445 | | No | | | | | | | | | | | | | | | | | | Note 15 |
|  | 6M721-2090 | 55 | No | Reactor Pressure Vessel Level | B21F507 (V13-2318) | EFC | SA | HF | -- | -- | -- | O | O | O | -- | O | No | A | No | Yes | Notes 15 and 16<br>See Penetration Detail X-28A |
|  | 6M721-2084<br><br>6I721-2679-1 | 56 | No | Drywell Pressure Nitrogen Inerting Instrumentation | T5000F420A (V5-2230) | BAL | AO | RM | M | -- | -- | O | O | O | C | O | No | B | Yes | Yes | Notes 2, 12, 13, and 15 |
| | | | | | | | | | | | | | | | | | | | | | |
| | | | | | | | | | | | | | | | | | | | | | |

TABLE 6.2-2 SUMMARY OF PRIMARY CONTAINMENT PENETRATIONS AND ASSOCIATED ISOLATION VALVES

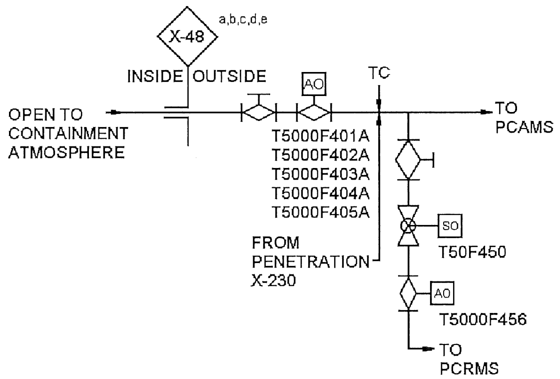
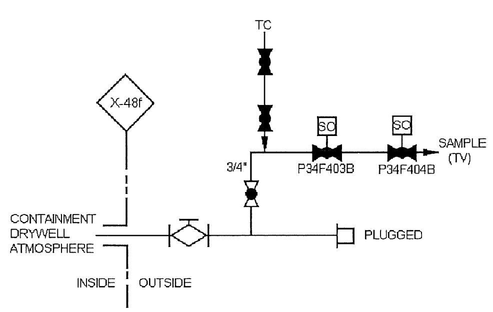
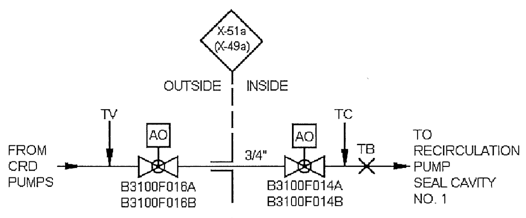
| PENETRATION DATA | | | | ISOLATION VALVE DATA | | | | | | | | | | | | | | | | | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| | | | | | | | | | | | | Valve Position | | | | | | | | | |
| Penetration Detail | P&ID Number | Design Criteria | Bypass Leakage Path | System Title | Valve Number | Valve Type | Actuator Type | Primary Actuation Mode | Secondary Actuation Mode | Containment Isolation Signal(s) | Accident Isolation Signal(s) | Normal | Shutdown | Post-LOCA (Accident) | Power Failure | ILRT | Engineered Safety Feature | Quality Group | Type C Test | Leak Detection | Remarks |
|  | 6I721-2679-1 | 56 | No | Containment Atmosphere Samples | T5000F401A (V5-2151) | BAL | AO | RM | M | -- | -- | O | O | O | C | O | No | B | Yes | Yes | See Penetration Detail X-27 |
| | | | | | T5000F402A (V5-2152) | | | | | | | | | | | | | | | | |
| | | | | | T5000F403A (V5-2153) | | | | | | | | | | | | | | | | |
| | | | | | T5000F404A (V5-2154) | | | | | | | | | | | | | | | | |
| | | | | | T5000F405A (V5-2155) | | | | | | | | | | | | | | | | |
| | | | | | T50F450 (V5-3083) | GLB | SO | A | -- | B, K | B, K | O | O | C | C | C | No | B | Yes | Yes | Note 10 |
| | | | | | T5000F456 (V5-2235) | BAL | AO | A | M | B, K | B, K | O | O | C | C | C | No | B | Yes | Yes | Note 10 |
|  | 6I721-2679-1 | 56 | Yes | Containment Atmosphere Samples | | | | | | | | | | | | | | | | | |
| | 6I721-2400-10 | | | | P34F403B (V13-7365) | GLB | SO | RM | -- | -- | -- | C | C | C | C | C | No | B | Yes | Yes | Note 10 |
| | | | | | P34F404B (V13-7375) | GLB | SO | RM | -- | -- | -- | C | C | C | C | C | No | B | Yes | Yes | Note 10 |
|  | 6M721-2833 | 55 | No | Recirculation Pump Seal Purge | B3100F014B (V8-3590) | GLB | AO | A | M | B, K | -- | O | O | C | C | C | No | B | Yes | Yes | Note 15 |
| | | | | | B3100F016B (V8-3768) | GLB | AO | A | M | B, K | -- | O | O | C | C | C | No | B | Yes | Yes | Note 15 |
| | | | | | B3100F014A (V8-3710) | GLB | AO | A | M | B,K | -- | O | O | C | C | C | No | B | Yes | Yes | Note 15 |
| | | | | | B3100F016A (V8-3767) | GLB | AO | A | M | B,K | -- | O | O | C | C | C | No | B | Yes | Yes | Note 15 |
| X-50 | -- | -- | -- | Spare | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | Type A Test |
| X-52 | 6M721-2089 2044 | 55 | No | Main Steam Flow (4) and Steam Flow to Reactor Core Isolation Cooling (2) | Typical of Six | EFC | SA | HF | -- | -- | -- | O | O | O | -- | O | No | A | No | Yes | Note 16 See Penetration Detail X-28A |

TABLE 6.2-2 SUMMARY OF PRIMARY CONTAINMENT PENETRATIONS AND ASSOCIATED ISOLATION VALVES

| PENETRATION DATA | | | | ISOLATION VALVE DATA | | | | | | | | | | | | | | | | | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| | | | | | | | | | | | | Valve Position | | | | | | | | | |
| Penetration Detail | P&ID Number | Design Criteria | Bypass Leakage Path | System Title | Valve Number | Valve Type | Actuator Type | Primary Actuation Mode | Secondary Actuation Mode | Containment Isolation Signal(s) | Accident Isolation Signal(s) | Normal | Shutdown | Post-LOCA (Accident) | Power Failure | ILRT | Engineered Safety Feature | Quality Group | Type C Test | Leak Detection | Remarks |
| X-53 | 6M721-<br>2044<br>2034<br>2090 | 55 | No | Steam Flow to Reactor Core Isolation Cooling (2), Feedwater Pressure (1), and Reactor Pressure Vessel Pressure (1) | Typical of Four | EFC | SA | HF | -- | -- | -- | O | O | O | -- | O | No | A | No | Yes | Notes 15 and 16<br>See Penetration Detail X-28A |
| X-54Ad | 6M721-<br>2090 | 55 | Yes | Reactor Level, Pressure | B2100F506<br>(V13-2317) | EFC | SA | HF | -- | -- | -- | O | O | O | -- | O | No | A | No | Yes | Notes 15 and 16<br>See Penetration Detail X-28A |
| X-54B | 6M721-<br>2090 | 55 | Yes | Reactor Level, Pressure | B2100F508<br>(V13-2397) | EFC | SA | HF | -- | -- | -- | O | O | O | -- | O | No | A | No | Yes | Notes 15 and 16<br>See Penetration Detail X-28A |
| X-55Ae | 6M721-<br>2090 | 55 | No | Reactor Level, Pressure | B2100F510<br>(V13-2321) | EFC | SA | HF | -- | -- | -- | O | O | O | -- | O | No | A | No | Yes | Notes 15 and 16<br>See Penetration Detail X-28A |
| X-55Bb<br>X-55Be | 6M721-<br>2090 | 55 | No | Reactor Level, Pressure | B2100F512<br>(V13-2323)<br>B2100F511<br>(V13-2396) | EFC | SA | HF | -- | -- | -- | O | O | O | -- | O | No | A | No | Yes | Notes 15 and 16<br>See Penetration Detail X-28A |
| X-100A | 6E721-2831-<br>8 | -- | -- | Neutron Monitor | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | Type B Test |
| X-100B | 6E721-2831-<br>8 | -- | -- | Low Voltage Switching Reactor Protection System | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | Type B Test |
| X-100F | 6E721-2831-<br>8 | -- | -- | Spare | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | Type A Test |
| X-100G | 6E721-2831-<br>8 | -- | -- | Neutron Monitor | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | Type B Test |
| X-101A | 6E721-2831-<br>8 | -- | -- | Recirculation Pump Power, 5kV | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | Type B Test |
| X-101B | 6E721-2831-<br>8 | -- | -- | Recirculation Pump Power, 5kV | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | Type B Test |
| X-101C | 6E721-2831-<br>8 | -- | -- | Recirculation Pump Power, 5kV | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | Type B Test |
| X-101D | 6E721-2831-<br>8 | -- | -- | Recirculation Pump Power, 5kV | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | Type B Test |
| X-101E | 6E721-2831-<br>8 | -- | -- | Recirculation Pump Power, 5kV | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | Type B Test |

TABLE 6.2-2 SUMMARY OF PRIMARY CONTAINMENT PENETRATIONS AND ASSOCIATED ISOLATION VALVES

| PENETRATION DATA | | | | ISOLATION VALVE DATA | | | | | | | | | | | | | | | | | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| | | | | | | | | | | | | Valve Position | | | | | | | | | |
| Penetration Detail | P&ID Number | Design Criteria | Bypass Leakage Path | System Title | Valve Number | Valve Type | Actuator Type | Primary Actuation Mode | Secondary Actuation Mode | Containment Isolation Signal(s) | Accident Isolation Signal(s) | Normal | Shutdown | Post-LOCA (Accident) | Power Failure | ILRT | Engineered Safety Feature | Quality Group | Type C Test | Leak Detection | Remarks |
| X-101F | 6E721-2831-8 | – | – | Recirculation Pump Power, 5kV | – | – | – | – | – | – | – | – | – | – | – | – | – | – | – | – | Type B Test |
| X-102A | 6E721-2831-8 | – | – | Neutron Monitor | – | – | – | – | – | – | – | – | – | – | – | – | – | – | – | – | Type B Test |
| X-102B | 6E721-2831-8 | – | – | Low Voltage Switching/ Reactor Protection System | – | – | – | – | – | – | – | – | – | – | – | – | – | – | – | – | Type B Test |
| X-105B | 6E721-2831-8 | – | – | Low Voltage Switching/ Reactor Protection System | – | – | – | – | – | – | – | – | – | – | – | – | – | – | – | – | Type B Test |
| X-105C | 6E721-2831-8 | – | – | Low Voltage Switching/ Reactor Protection System | – | – | – | – | – | – | – | – | – | – | – | – | – | – | – | – | Type B Test |
| X-103A | 6E721-2831-8 | – | – | Drywell Thermocouples | – | – | – | – | – | – | – | – | – | – | – | – | – | – | – | – | Type B Test |
| X-103B | 6E721-2831-8 | – | – | Neutron Monitor | – | – | – | – | – | – | – | – | – | – | – | – | – | – | – | – | Type B Test |
| X-104A | 6E721-2831-8 | – | – | Control Rod Drive Position Indicator | – | – | – | – | – | – | – | – | – | – | – | – | – | – | – | – | Type B Test |
| X-104B | 6E721-2831-8 | – | – | Control Rod Drive Position Indicator | – | – | – | – | – | – | – | – | – | – | – | – | – | – | – | – | Type B Test |
| X-104C | 6E721-2831-8 | – | – | Control Rod Drive Position Indicator | – | – | – | – | – | – | – | – | – | – | – | – | – | – | – | – | Type B Test |
| X-104D | 6E721-2831-8 | – | – | Control Rod Drive Position Indicator | – | – | – | – | – | – | – | – | – | – | – | – | – | – | – | – | Type B Test |
| X-104E | 6E721-2831-8 | – | – | Control Rod Drive Position Indicator | – | – | – | – | – | – | – | – | – | – | – | – | – | – | – | – | Type B Test |
| X-104F | 6E721-2831-8 | – | – | Control Rod Drive Position Indicator | – | – | – | – | – | – | – | – | – | – | – | – | – | – | – | – | Type B Test |
| X-105A | 6E721-2831-8 | – | – | Low Voltage Power (480 V) | – | – | – | – | – | – | – | – | – | – | – | – | – | – | – | – | Type B Test |

TABLE 6.2-2 SUMMARY OF PRIMARY CONTAINMENT PENETRATIONS AND ASSOCIATED ISOLATION VALVES

| PENETRATION DATA | | | | ISOLATION VALVE DATA | | | | | | | | | | | | | | | | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| | | | | | | | | | | | | Valve Position | | | | | | | | |
| Penetration Detail | P&ID Number | Design Criteria | Bypass Leakage Path | System Title | Valve Number | Valve Type | Actuator Type | Primary Actuation Mode | Secondary Actuation Mode | Containment Isolation Signal(s) | Accident Isolation Signal(s) | Normal | Shutdown | Post-LOCA (Accident) | Power Failure | ILRT | Engineered Safety Feature | Quality Group | Type C Test | Leak Detection | Remarks |
| X-105D | 6E721-2831-8 | – | – | Low Voltage Power (480 V) | – | – | – | – | – | – | – | – | – | – | – | – | – | – | – | – | Type B Test |
| X-106A | 6E721-2831-8 | – | – | Spare | – | – | – | – | – | – | – | – | – | – | – | – | – | – | – | – | Type A Test |
| X-106B | 6E721-2831-8 | – | – | Low Level Signal Vibration Test | – | – | – | – | – | – | – | – | – | – | – | – | – | – | – | – | Type B Test |
| X-200A | – | – | – | Torus Access Hatch | – | – | – | – | – | – | – | – | – | – | – | – | – | – | – | – | Type B Test |
| X-200B | – | – | – | Torus Access Hatch | – | – | – | – | – | – | – | – | – | – | – | – | – | – | – | – | Type B Test |
| X-201A | 6C721-2305 | – | – | Vent Line Bellows | – | – | – | – | – | – | – | – | – | – | – | – | – | – | – | – | Type B Test |
| X-201B | 6C721-2305 | – | – | Vent Line Bellows | – | – | – | – | – | – | – | – | – | – | – | – | – | – | – | – | Type B Test |
| X-201C | 6C721-2305 | – | – | Vent Line Bellows | – | – | – | – | – | – | – | – | – | – | – | – | – | – | – | – | Type B Test |
| X-201D | 6C721-2305 | – | – | Vent Line Bellows | – | – | – | – | – | – | – | – | – | – | – | – | – | – | – | – | Type B Test |
| X-201E | 6C721-2305 | – | – | Vent Line Bellows | – | – | – | – | – | – | – | – | – | – | – | – | – | – | – | – | Type B Test |
| X-100C | 6E721-2831-8 | – | – | Blanked Off Electrical Penetration (Spare) | – | – | – | – | – | – | – | – | – | – | – | – | – | – | – | – | Type B Test, Double O-Ring Testable Seal |
| X-100E | 6E721-2831-8 | – | – | Blanked Off Electrical Penetration (Spare) | – | – | – | – | – | – | – | – | – | – | – | – | – | – | – | – | Type B Test, Double O-Ring Testable Seal |
| | | | | | | | | | | | | | | | | | | | | | |
| X-100D | 6E721-2831-8 | – | – | Low Level Signal Vibration Test | – | – | – | – | – | – | – | – | – | – | – | – | – | – | – | – | Type B Test |

TABLE 6.2-2 SUMMARY OF PRIMARY CONTAINMENT PENETRATIONS AND ASSOCIATED ISOLATION VALVES

| PENETRATION DATA | | | | ISOLATION VALVE DATA | | | | | | | | | | | | | | | | | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| | | | | | | | | | | | | Valve Position | | | | | | | | | |
| Penetration Detail | P&ID Number | Design Criteria | Bypass Leakage Path | System Title | Valve Number | Valve Type | Actuator Type | Primary Actuation Mode | Secondary Actuation Mode | Containment Isolation Signal(s) | Accident Isolation Signal(s) | Normal | Shutdown | Post-LOCA (Accident) | Power Failure | ILRT | Engineered Safety Feature | Quality Group | Type C Test | Leak Detection | Remarks |
| X-107B | 6E721-2831-8 | – | – | Drywell Thermocouples | – | – | – | – | – | – | – | – | – | – | – | – | – | – | – | – | Type B Test |
| X-107A | 6E721-2831-8 | – | – | Spare | – | – | – | – | – | – | – | – | – | – | – | – | – | – | – | – | Type A Test |
| X-201F | 6C721-2305 | – | – | Vent Line Bellows | – | – | – | – | – | – | – | – | – | – | – | – | – | – | – | – | Type B Test |
| X-201G | 6C721-2305 | – | – | Vent Line Bellows | – | – | – | – | – | – | – | – | – | – | – | – | – | – | – | – | Type B Test |
| X-201H | 6C721-2305 | – | – | Vent Line Bellows | – | – | – | – | – | – | – | – | – | – | – | – | – | – | – | – | Type B Test |
| X-202A | 6C721-2305 | – | – | Vacuum Breaker (Inside Torus) | – | – | – | – | – | – | – | – | – | – | – | O | – | – | – | – | Note 23 |
| X-202B | 6C721-2305 | – | – | Vacuum Breaker (Inside Torus) | – | – | – | – | – | – | – | – | – | – | – | O | – | – | – | – | Note 23 |
| X-202C | 6C721-2305 | – | – | Vacuum Breaker (Inside Torus) | – | – | – | – | – | – | – | – | – | – | – | O | – | – | – | – | Note 23 |
| X-202D | 6C721-2305 | – | – | Vacuum Breaker (Inside Torus) | – | – | – | – | – | – | – | – | – | – | – | O | – | – | – | – | Note 23 |
| X-202E | 6C721-2305 | – | – | Vacuum Breaker (Inside Torus) | – | – | – | – | – | – | – | – | – | – | – | O | – | – | – | – | Note 23 |
| X-202F | 6C721-2305 | – | – | Vacuum Breaker (Inside Torus) | – | – | – | – | – | – | – | – | – | – | – | O | – | – | – | – | Note 23 |
| X-202G | 6C721-2305 | – | – | Vacuum Breaker (Inside Torus) | – | – | – | – | – | – | – | – | – | – | – | O | – | – | – | – | Note 23 |
| | | | | | | | | | | | | | | | | | | | | | |
| X-202H | 6C721-2305 | – | – | Vacuum Breaker (Inside Torus) | – | – | – | – | – | – | – | – | – | – | – | O | – | – | – | – | Note 23 |

TABLE 6.2-2 SUMMARY OF PRIMARY CONTAINMENT PENETRATIONS AND ASSOCIATED ISOLATION VALVES

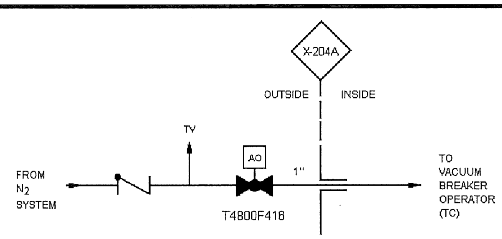
| PENETRATION DATA | | | | ISOLATION VALVE DATA | | | | | | | | | | | | | | | | | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| | | | | | | | | | | | | Valve Position | | | | | | | | | |
| Penetration Detail | P&ID Number | Design Criteria | Bypass Leakage Path | System Title | Valve Number | Valve Type | Actuator Type | Primary Actuation Mode | Secondary Actuation Mode | Containment Isolation Signal(s) | Accident Isolation Signal(s) | Normal | Shutdown | Post-LOCA (Accident) | Power Failure | ILRT | Engineered Safety Feature | Quality Group | Type C Test | Leak Detection | Remarks |
| X-202J | 6C721-2305 | -- | -- | Vacuum Breaker (Inside Torus) | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | O | -- | -- | -- | -- | Note 23 |
| X-202K | 6C721-2305 | -- | -- | Vacuum Breaker (Inside Torus) | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | O | -- | -- | -- | -- | Note 23 |
| X-202L | 6C721-2305 | -- | -- | Vacuum Breaker (Inside Torus) | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | O | -- | -- | -- | -- | Note 23 |
| X-202M | 6C721-2305 | -- | -- | Vacuum Breaker (Inside Torus) | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | O | -- | -- | -- | -- | Note 23 |
|  | 6M721-3445 | 57 | No | Drywell to Torus Vacuum Breaker Nitrogen Supply | T4800F416 (V4-2036) | GLB | AO | RM | M | -- | -- | LC | LC | LC | C | C | No | B | Yes | No | Note 9 |
| X-204B | 6M721-3445 | 57 | No | Drywell to Torus Vacuum Breaker Nitrogen Supply | T4800F417 (V4-2065) | GLB | AO | RM | M | -- | -- | LC | LC | LC | C | C | No | B | Yes | No | See Penetration Detail X-204A |
| X-204C | 6M721-3445 | 57 | No | Drywell to Torus Vacuum Breaker Nitrogen Supply | T4800F418 (V4-2075) | GLB | AO | RM | M | -- | -- | LC | LC | LC | C | C | No | B | Yes | No | See Penetration Detail X-204A |
| X-204D | 6M721-3445 | 57 | No | Drywell to Torus Vacuum Breaker Nitrogen Supply | T4800F419 (V4-2077) | GLB | AO | RM | M | -- | -- | LC | LC | LC | C | C | No | B | Yes | No | See Penetration Detail X-204A |
| X-204E | 6M721-3445 | 57 | No | Drywell to Torus Vacuum Breaker Nitrogen Supply | T4800F420 (V4-2082) | GLB | AO | RM | M | -- | -- | LC | LC | LC | C | C | No | B | Yes | No | See Penetration Detail X-204A |
| X-204F | 6M721-3445 | 57 | No | Drywell to Torus Vacuum Breaker Nitrogen Supply | T4800F421 (V4-2084) | GLB | AO | RM | M | -- | -- | LC | LC | LC | C | C | No | B | Yes | No | See Penetration Detail X-204A |
| X-204G | 6M721-3445 | 57 | No | Drywell to Torus Vacuum Breaker Nitrogen Supply | T4800F422 (V4-2086) | GLB | AO | RM | M | -- | -- | LC | LC | LC | C | C | No | B | Yes | No | See Penetration Detail X-204A |
| X-204H | 6M721-3445 | 57 | No | Drywell to Torus Vacuum Breaker Nitrogen Supply | T4800F423 (V4-2088) | GLB | AO | RM | M | -- | -- | LC | LC | LC | C | C | No | B | Yes | No | See Penetration Detail X-204A |

TABLE 6.2-2 SUMMARY OF PRIMARY CONTAINMENT PENETRATIONS AND ASSOCIATED ISOLATION VALVES

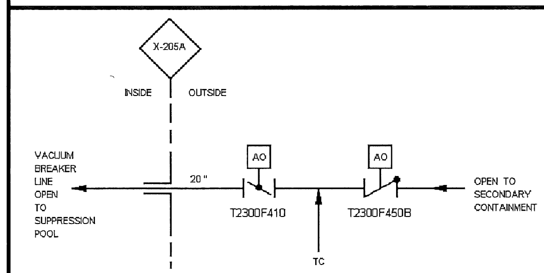
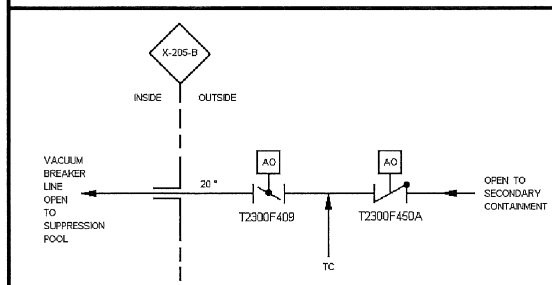
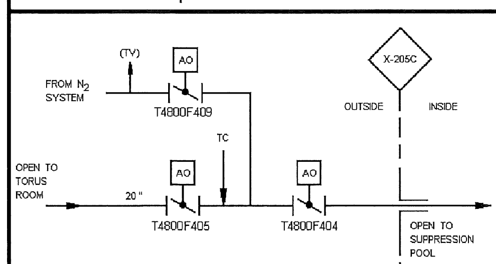
| PENETRATION DATA | | | | ISOLATION VALVE DATA | | | | | | | | | | | | | | | | | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| | | | | | | | | | | | | Valve Position | | | | | | | | | |
| Penetration Detail | P&ID Number | Design Criteria | Bypass Leakage Path | System Title | Valve Number | Valve Type | Actuator Type | Primary Actuation Mode | Secondary Actuation Mode | Containment Isolation Signal(s) | Accident Isolation Signal(s) | Normal | Shutdown | Post-LOCA (Accident) | Power Failure | ILRT | Engineered Safety Feature | Quality Group | Type C Test | Leak Detection | Remarks |
| X-204J | 6M721-3445 | 57 | No | Drywell to Torus Vacuum Breaker Nitrogen Supply | T4800F424 (V4-2090) | GLB | AO | RM | M | -- | -- | LC | LC | LC | C | C | No | B | Yes | No | See Penetration Detail X-204A |
| X-204K | 6M721-3445 | 57 | No | Drywell to Torus Vacuum Breaker Nitrogen Supply | T4800F425 (V4-2092) | GLB | AO | RM | M | -- | -- | LC | LC | LC | C | C | No | B | Yes | No | See Penetration Detail X-204A |
| X-204L | 6M721-3445 | 57 | No | Drywell to Torus Vacuum Breaker Nitrogen Supply | T4800F426 (V4-2094) | GLB | AO | RM | M | -- | -- | LC | LC | LC | C | C | No | B | Yes | No | See Penetration Detail X-204A |
| X-204M | 6M721-3445 | 57 | No | Drywell to Torus Vacuum Breaker Nitrogen Supply | T4800F427 (V4-2096) | GLB | AO | RM | M | -- | -- | LC | LC | LC | C | C | No | B | Yes | No | See Penetration Detail X-204A |
|   | 6M721-3445-1 | 56 | No | Secondary Containment to Torus Vacuum Breaker | T2300F410 (V21-2016) | BFY | AO | RM | -- | -- | H | C | C | C | O | C | Yes | B | Yes | No | Notes 10, 11, and 24 |
| | | | | | T2300F450B (V21-2014) | CHK | SA | RF | RM | -- | -- | C | C | C | -- | C | Yes | B | Yes | No | Notes 10 and 25 |
|   | 6M721-3445-1 | 56 | No | Secondary Containment to Torus Vacuum Breaker | T2300F409 (V21-2015) | BFY | AO | RM | -- | -- | H | C | C | C | O | C | Yes | B | Yes | No | Notes 10, 11, and 24 |
| | | | | | T2300F450A (V21-2013) | CHK | SA | RF | RM | -- | -- | C | C | C | -- | C | Yes | B | Yes | No | Notes 10 and 25 |
|   | 6M721-3445-1<br>7M721-2709 | 56 | No | Suppression Pool Air Purge Inlet | T4800F404 (VR3-3013) | BFY | AO | A | RM | B, K, R | -- | C | O | C | C | C | R | B | Yes | No | Notes 10 and 11<br>Flanges type B tested |
| | | | | | T4800F405 (VR3-3014) | BFY | AO | A | RM | B, K, R | -- | C | O | C | C | C | R | B | Yes | No | Note 10 |
| | | | | | T4800F409 (VR3-2061) | BFY | AO | A | RM | B, K, R | -- | C | C | C | C | C | R | B | Yes | No | Note 10 |

TABLE 6.2-2 SUMMARY OF PRIMARY CONTAINMENT PENETRATIONS AND ASSOCIATED ISOLATION VALVES

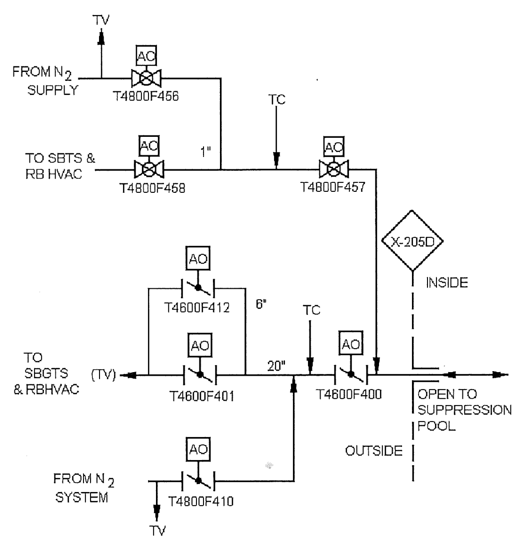
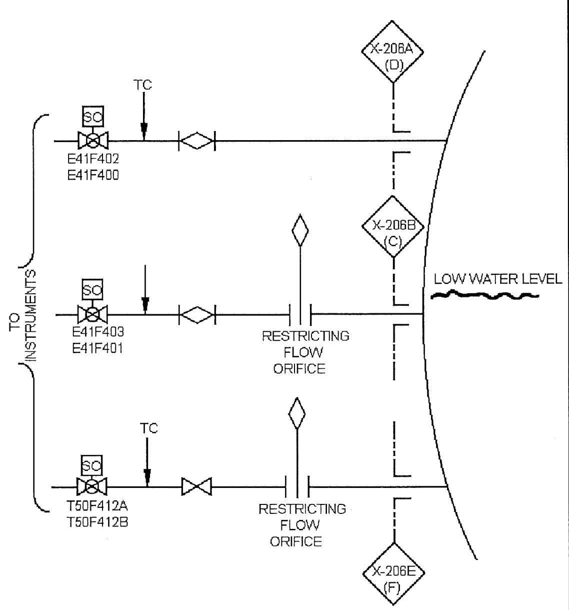
| PENETRATION DATA | | | | ISOLATION VALVE DATA | | | | | | | | | | | | | | | | | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| | | | | | | | | | | | | Valve Position | | | | | | | | | |
| Penetration Detail | P&ID Number | Design Criteria | Bypass Leakage Path | System Title | Valve Number | Valve Type | Actuator Type | Primary Actuation Mode | Secondary Actuation Mode | Containment Isolation Signal(s) | Accident Isolation Signal(s) | Normal | Shutdown | Post-LOCA (Accident) | Power Failure | ILRT | Engineered Safety Feature | Quality Group | Type C Test | Leak Detection | Remarks |
|  | 7M721-2709 | 56 | No | Suppression Pool Exhaust and Nitrogen Inlet | T4600F400 (VR3-3015) | BFY | AO | A | RM | B, K, R | -- | C | O | C | C | C | R | B | Yes | No | Notes 10, 11, and 44 Flanges type B tested |
| | 6M721-3445-1 | | | | T4600F401 (VR3-3016) | BFY | AO | A | RM | B, K, R | -- | C | O | C | C | C | R | B | Yes | No | Note 10 and 44 |
| | | | | | T4600F412 (VR3-3019) | BFY | AO | A | RM | B, K, R | -- | C | C | C | C | C | R | B | Yes | No | Note 10 |
| | | | | | T4800F410 (V4-2063) | BFY | AO | A | RM | B, K, R | -- | C | C | C | C | C | R | B | Yes | No | Note 10 |
| | | | | | T4800F456 (VR3-2826) | GLB | AO | A | RM | B, K, R | -- | O | C | C | C | C | R | B | Yes | No | Note 10 |
| | | | | | T4800F457 (VR3-2827) | GLB | AO | A | RM | B, K, R | -- | O | C | C | C | C | R | B | Yes | No | Notes 2 and 10 |
| | | | | | T4800F458 (VR3-2828) | GLB | AO | A | RM | B, K, R | -- | O | C | C | C | C | R | B | Yes | No | Note 10 |
|  | 6I721-2679-1 | 56 | No | Torus Pressure and Liquid Level Instrumentation | E41F402 (V5-2552) | GLB | SO | RM | M | -- | -- | O | O | O | AIS | O | No | B | Yes | Yes | Note 12 |
| | | | | | E41F403 (V5-2553) | GLB | SO | RM | M | -- | -- | O | O | O | AIS | O | No | B | Yes | Yes | Notes 12 and 26 |
| | | | | | E41F401 (V5-2551) | GLB | SO | RM | M | -- | -- | O | O | O | AIS | O | No | B | Yes | Yes | Notes 12 and 26 |
| | | | | | E41F400 (V5-2550) | GLB | SO | RM | M | -- | -- | O | O | O | AIS | O | No | B | Yes | Yes | Note 12 |
| | | | | | T50F412A (V5-2555) | GLB | SO | RM | M | -- | -- | O | O | O | AIS | O | No | B | Yes | Yes | Notes 12 and 26 |
| | | | | | T50F412B (V5-2556) | GLB | SO | RM | M | -- | -- | O | O | O | AIS | O | No | B | Yes | Yes | Notes 12 and 26 |
| X-207A | 6C721-2305 | -- | -- | Drain Line (Inside Torus) | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | Note 23 |

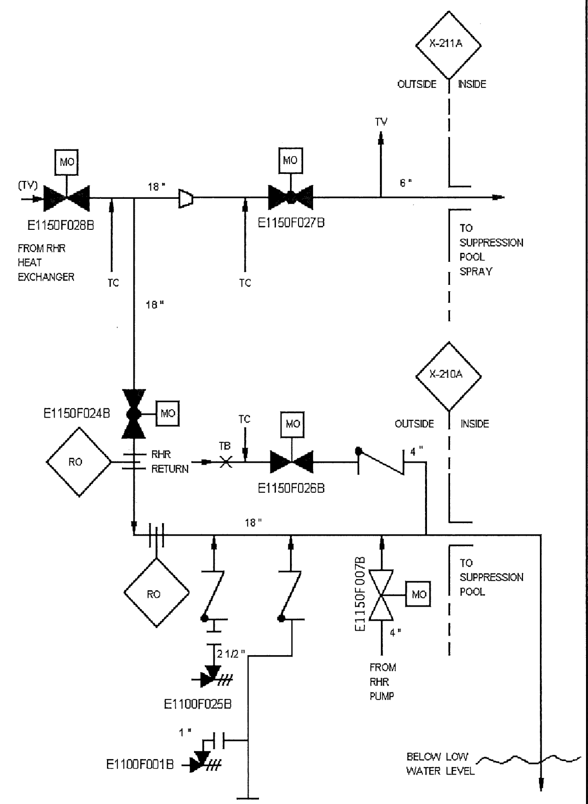
TABLE 6.2-2 SUMMARY OF PRIMARY CONTAINMENT PENETRATIONS AND ASSOCIATED ISOLATION VALVES

| PENETRATION DATA | | | | ISOLATION VALVE DATA | | | | | | | | | | | | | | | | | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| | | | | | | | | | | | | Valve Position | | | | | | | | | |
| Penetration Detail | P&ID Number | Design Criteria | Bypass Leakage Path | System Title | Valve Number | Valve Type | Actuator Type | Primary Actuation Mode | Secondary Actuation Mode | Containment Isolation Signal(s) | Accident Isolation Signal(s) | Normal | Shutdown | Post-LOCA (Accident) | Power Failure | ILRT | Engineered Safety Feature | Quality Group | Type C Test | Leak Detection | Remarks |
| X-207B | 6C721-2305 | -- | -- | Drain Line (Inside Torus) | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | Note 23 |
| X-207C | 6C721-2305 | -- | -- | Drain Line (Inside Torus) | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | Note 23 |
| X-207D | 6C721-2305 | -- | -- | Drain Line (Inside Torus) | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | Note 23 |
| X-207E | 6C721-2305 | -- | -- | Drain Line (Inside Torus) | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | Note 23 |
| X-207F | 6C721-2305 | -- | -- | Drain Line (Inside Torus) | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | Note 23 |
| X-207G | 6C721-2305 | -- | -- | Drain Line (Inside Torus) | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | Note 23 |
| X-207H | 6C721-2305 | -- | -- | Drain Line (Inside Torus) | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | Note 23 |
| X-208A | 6C721-2305 | -- | -- | Electromagnetic Relief Valve Discharge (Inside Torus) | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | Note 23 |
| X-208B | 6C721-2305 | -- | -- | Electromagnetic Relief Valve Discharge (Inside Torus) | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | Note 23 |
| X-208C | 6C721-2305 | -- | -- | Electromagnetic Relief Valve Discharge (Inside Torus) | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | Note 23 |
| X-208D | 6C721-2305 | -- | -- | Electromagnetic Relief Valve Discharge (Inside Torus) | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | Note 23 |
| X-208E | 6C721-2305 | -- | -- | Electromagnetic Relief Valve Discharge (Inside Torus) | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | Note 23 |
| X-208F | 6C721-2305 | -- | -- | Electromagnetic Relief Valve Discharge (Inside Torus) | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | Note 23 |
| X-208G | 6C721-2305 | -- | -- | Electromagnetic Relief Valve Discharge (Inside Torus) | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | Note 23 |

TABLE 6.2-2 SUMMARY OF PRIMARY CONTAINMENT PENETRATIONS AND ASSOCIATED ISOLATION VALVES

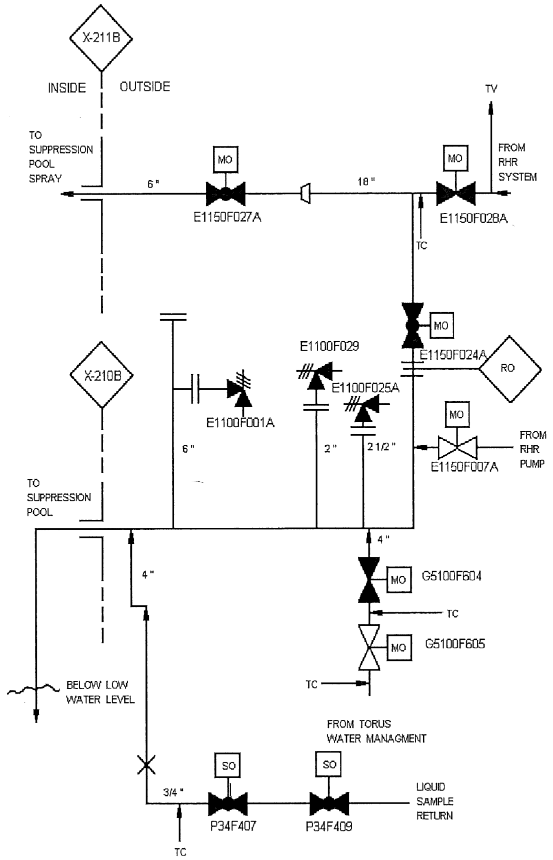
| PENETRATION DATA | | | | ISOLATION VALVE DATA | | | | | | | | | | | | | | | | | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| | | | | | | | | | | | | Valve Position | | | | | | | | | |
| Penetration Detail | P&ID Number | Design Criteria | Bypass Leakage Path | System Title | Valve Number | Valve Type | Actuator Type | Primary Actuation Mode | Secondary Actuation Mode | Containment Isolation Signal(s) | Accident Isolation Signal(s) | Normal | Shutdown | Post-LOCA (Accident) | Power Failure | ILRT | Engineered Safety Feature | Quality Group | Type C Test | Leak Detection | Remarks |
| X-208H | 6C721-2305 | -- | -- | Electromagnetic Relief Valve Discharge (Inside Torus) | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | Note 23 |
| X-208J | 6C721-2305 | -- | -- | Electromagnetic Relief Valve Discharge (Inside Torus) | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | Note 23 |
| X-208K | 6C721-2305 | -- | -- | Electromagnetic Relief Valve Discharge (Inside Torus) | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | Note 23 |
| X-208L | 6C721-2305 | -- | -- | Electromagnetic Relief Valve Discharge (Inside Torus) | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | Note 23 |
| X-208M | 6C721-2305 | -- | -- | Electromagnetic Relief Valve Discharge (Inside Torus) | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | Note 23 |
| X-208N | 6C721-2305 | -- | -- | Electromagnetic Relief Valve Discharge (Inside Torus) | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | Note 23 |
| X-208O | 6C721-2305 | -- | -- | Electromagnetic Relief Valve Discharge (Inside Torus) | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | Note 23 |
| X-208P | 6C721-2305 | -- | -- | Electromagnetic Relief Valve Discharge (Inside Torus) | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | Note 23 |
| X-209A | -- | -- | -- | Thermocouple | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | Type B Test |
| X-209B | -- | -- | -- | Spare | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | Type A Test |
| X-209C | -- | -- | -- | Torus Thermocouple | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | Type B Test |
| X-209D | -- | -- | -- | Spare | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | Type A Test |

TABLE 6.2-2 SUMMARY OF PRIMARY CONTAINMENT PENETRATIONS AND ASSOCIATED ISOLATION VALVES




| PENETRATION DATA | | | | ISOLATION VALVE DATA | | | | | | | | | | | | | | | | | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| | | | | | | | | | | | | Valve Position | | | | | | | | | |
| Penetration Detail | P&ID Number | Design Criteria | Bypass Leakage Path | System Title | Valve Number | Valve Type | Actuator Type | Primary Actuation Mode | Secondary Actuation Mode | Containment Isolation Signal(s) | Accident Isolation Signal(s) | Normal | Shutdown | Post-LOCA (Accident) | Power Failure | ILRT | Engineered Safety Feature | Quality Group | Type C Test | Leak Detection | Remarks |
| | 6M721-2083 | 56 | No | Residual Heat Removal Minimum Flow | E1150F007B (V8-4679) | GAT | MO | RM | M | -- | Z | O | C | C | AIS | C | Yes | B | No | Yes | Notes 12 and 39 |
| | | 56 | No | Residual Heat Removal Heat Exchanger Discharge Header Thermal Relief | E1100F025B | REL | SA | -- | -- | -- | -- | C | C | C | -- | C | Yes | B | No | Yes | Notes 27, 28 and 39 |
| | | 56 | No | Residual Heat Removal Test Line | E1150F024B (V8-2136) | GLB | MO | A | RM | A,K | -- | C | C | C | AIS | C | Yes | B | Yes | Yes | Notes 2, 10 and 26 |
| | | 56 | No | Residual Heat Removal to Suppression Pool Spray | E1150F027B (V8-2158) | GLB | MO | A | RM | A,K | -- | C | C | C | AIS | C | Yes | B | Yes | Yes | Notes 2 and 10 |
| | | | | | E1150F028B (V8-2156) | GAT | MO | A | RM | A,K | -- | C | C | O | AIS | C | Yes | B | Yes | Yes | -- |
| | | 56 | No | Residual Heat Removal Heat Exchanger Thermal Relief | E1100F001B (V22-2642) | REL | SA | -- | -- | -- | -- | C | C | C | -- | C | Yes | B | No | Yes | Notes 27, 28, and 39 |
| | | 56 | No | Residual Heat Removal Warmup and Flush Line | E1150F026B (V8-2152) | GAT | MO | RM | M | -- | -- | C | C | C | AIS | C | Yes | B | No | Yes | Notes 39, and 40 Flanges Type B Tested |

TABLE 6.2-2 SUMMARY OF PRIMARY CONTAINMENT PENETRATIONS AND ASSOCIATED ISOLATION VALVES




| PENETRATION DATA | | | | ISOLATION VALVE DATA | | | | | | | | | | | | | | | | | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| | | | | | | | | | | | | Valve Position | | | | | | | | | |
| Penetration Detail | P&ID Number | Design Criteria | Bypass Leakage Path | System Title | Valve Number | Valve Type | Actuator Type | Primary Actuation Mode | Secondary Actuation Mode | Containment Isolation Signal(s) | Accident Isolation Signal(s) | Normal | Shutdown | Post-LOCA (Accident) | Power Failure | ILRT | Engineered Safety Feature | Quality Group | Type C Test | Leak Detection | Remarks |
| | 6M721-4100 | 56 | No | Torus Water Managment System | G5100F605 (V8-4680) | GAT | MO | A | RM | B,K | M | O | C | C | AIS | C | No | B | Yes | Yes | Note 26 |
| | | | | | G5100F604 (V8-3849) | GAT | MO | A | RM | B,K | M | C | C | C | AIS | C | No | B | Yes | Yes | Notes 4 and 26 |
| | 6M721-2083 | 56 | No | Residual Heat Removal Suction Thermal Relief | E1100F029 (V22-2033) | REL | SA | -- | -- | -- | -- | C | C | C | -- | C | Yes | B | No | Yes | Notes 27, 28, and 39 |
| | 6M721-2084 | 56 | No | Residual Heat Removal Heat Exchanger Discharge Header Thermal Relief | E1100F025A | REL | SA | -- | -- | -- | -- | C | C | C | -- | C | Yes | B | No | Yes | Notes 27, 28, and 39 |
| | | 56 | No | Residual Heat Removal Heat Exchanger Relief | E1100F001A (V22-2643) | REL | SA | -- | -- | -- | -- | C | C | C | -- | C | Yes | B | No | Yes | Notes 27, 28, and 39 |
| | | 56 | No | Residual Heat Removal Minimum Flow | E1150F007A (V8-2133) | GAT | MO | RM | M | -- | Z | O | O | C | AIS | C | Yes | B | No | Yes | Notes 7, 12, and 39 |
| | | 56 | No | Residual Heat Removal Test Line | E1150F024A (V8-2135) | GLB | MO | A | RM | A,K | -- | C | C | C | AIS | C | Yes | B | Yes | Yes | Notes 2, 10, and 26 |
| | | 56 | No | Residual Heat Removal Suppression Pool Spray | E1150F028A (V8-2155) | GAT | MO | A | RM | A,K | -- | C | C | C | AIS | C | Yes | B | Yes | Yes | -- |
| | | | | | E1150F027A (V8-2157) | GLB | MO | A | RM | A,K | -- | C | C | C | AIS | C | Yes | B | Yes | Yes | Note 10 |
| | 6I721-2400-10 | 56 | No | Liquid Sample Return | P34F407 (V13-7368) | GLB | SO | RM | -- | -- | -- | C | C | C | C | C | No | B | Yes | Yes | Notes 10 and 26 |
| | | | | | P34F409 (V13-7378) | GLB | SO | RM | -- | -- | -- | C | C | C | C | C | No | B | Yes | Yes | Notes 10 and 26<br>Flanges Type B Tested |

TABLE 6.2-2 SUMMARY OF PRIMARY CONTAINMENT PENETRATIONS AND ASSOCIATED ISOLATION VALVES

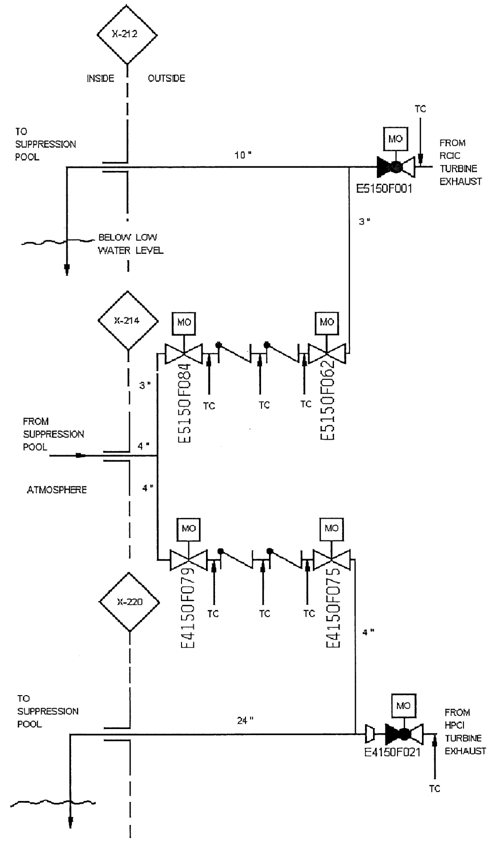
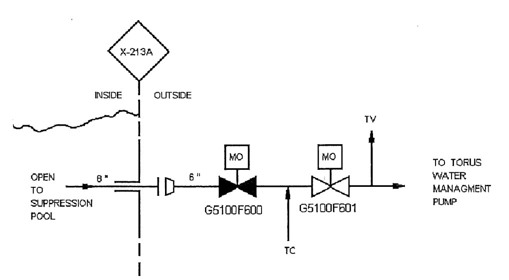
| PENETRATION DATA | | | | ISOLATION VALVE DATA | | | | | | | | | | | | | | | | | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| | | | | | | | | | | | | Valve Position | | | | | | | | | |
| Penetration Detail | P&ID Number | Design Criteria | Bypass Leakage Path | System Title | Valve Number | Valve Type | Actuator Type | Primary Actuation Mode | Secondary Actuation Mode | Containment Isolation Signal(s) | Accident Isolation Signal(s) | Normal | Shutdown | Post-LOCA (Accident) | Power Failure | ILRT | Engineered Safety Feature | Quality Group | Type C Test | Leak Detection | Remarks |
|  | 6M721-2044 | 56 | No | Reactor Core Isolation Cooling Turbine Exhaust Line | E5150F001 (V11-2002) | SCK | MO | RF | RM | -- | -- | O | O | O | AIS | C | R | B | No | Yes | Notes 6,12 and 39 |
| | | | | Reactor Core Isolation Cooling Vacuum Breaker Line | E5150F062 (V11-2020) | GAT | MO | A | RM | K&Y(4) | -- | O | O | O | AIS | C | R | B | Yes | No | Note 6 |
| | | | | | E5150F084 (V11-2026) | GAT | MO | A | RM | K&Y(4) | -- | O | O | O | AIS | C | R | B | Yes | No | Notes 4 and 6 |
| | 6M721-2035 | | | High-Pressure Coolant Injection Vacuum Breaker Line | E4150F075 (V11-2013) | GAT | MO | A | RM | K&X4) | -- | O | O | O | AIS | C | Yes | B | Yes | No | Note 7 |
| | | | | | E4150F079 (V11-2019) | GAT | MO | A | RM | K&X(4) | -- | O | O | O | AIS | C | Yes | B | Yes | No | Notes 4 and 7 |
| | | | | High-Pressure Coolant Injection Turbine Exhaust Line | E4150F021 (V11-2006) | SCK | MO | RF | RM | -- | -- | O | O | O | AIS | C | Yes | B | No | Yes | Notes 7, 12 and 39 |
|  | 6M721-4100 | 56 | No | Torus Water Management Suction | G5100F601 (V8-3834) | GAT | MO | A | RM | B,K | M | O | C | C | AIS | C | No | B | Yes | No | Notes 10 and 26 |
| | | | | | G5100F600 (V8-3832) | GAT | MO | A | RM | B,K | M | C | C | C | AIS | C | No | B | Yes | No | Notes 4, 10, and 26 |
| | | | | | | | | | | | | | | | | | | | | | |

TABLE 6.2-2 SUMMARY OF PRIMARY CONTAINMENT PENETRATIONS AND ASSOCIATED ISOLATION VALVES

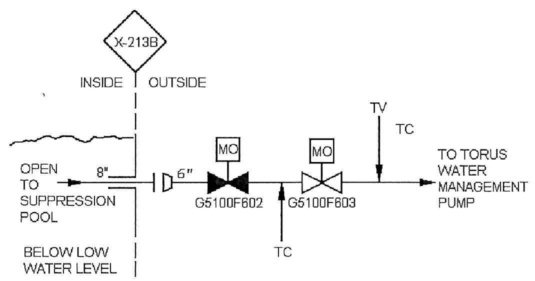
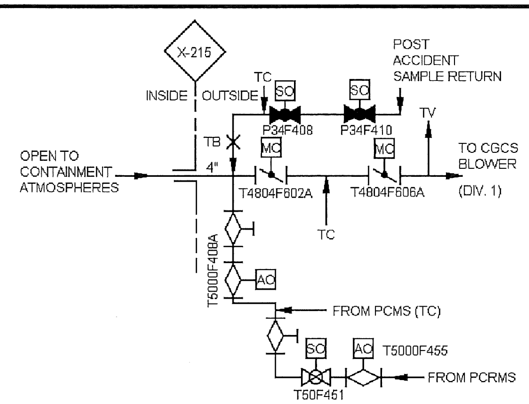
| PENETRATION DATA | | | | ISOLATION VALVE DATA | | | | | | | | | | | | | | | | | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| | | | | | | | | | | | | Valve Position | | | | | | | | | |
| Penetration Detail | P&ID Number | Design Criteria | Bypass Leakage Path | System Title | Valve Number | Valve Type | Actuator Type | Primary Actuation Mode | Secondary Actuation Mode | Containment Isolation Signal(s) | Accident Isolation Signal(s) | Normal | Shutdown | Post-LOCA (Accident) | Power Failure | ILRT | Engineered Safety Feature | Quality Group | Type C Test | Leak Detection | Remarks |
|  | 6M721-4100 | 56 | No | Torus Water Management Suction | G5100F602 (V8-3831) | GAT | MO | A | RM | B, K | M | C | C | C | AIS | C | No | B | Yes | No | Notes 4, 10, and 26 |
| | | | | | G5100F603 (V8-3833) | GAT | MO | A | RM | B, K | M | O | C | C | AIS | C | No | B | Yes | No | Notes 10 and 26 |
| X-214 | -- | -- | -- | Vacuum Breaker Line, High-Pressure Coolant Injection/ Reactor Core Isolation Cooling | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | See Penetration Detail X-212 |
|  | 6M721-2087 | 56 | No | Combustible Gas Control System Suction and Gaseous Sample Returns | T4804F602A (V4-2142) | BFY | M | M | -- | -- | -- | LC | LC | LC | LC | LC | No | B | Yes | No | Notes 9, 10, 11, and 45 |
| | 6I721-2679-1 | 56 | No | | T5000F408A (V5-2158) | BAL | AO | RM | M | -- | -- | O | O | O | C | O | No | B | Yes | Yes | Notes 12 and 13 |
| | 6M721-2087 | 56 | No | | T4804F606A (V4-2156) | BFY | M | M | -- | -- | -- | LC | LC | LC | LC | LC | No | B | Yes | No | Notes 9 and 45 |
| | 6I721-2400-10 | 56 | Yes | | P34F408 (V13-7369) | GLB | SO | RM | -- | -- | -- | C | C | C | C | C | No | B | Yes | Yes | Note 10 |
| | | 56 | Yes | | P34F410 (V13-7379) | GLB | SO | RM | -- | -- | -- | C | C | C | C | C | No | B | Yes | Yes | Note 10 |
| | 6I721-2679-1 | 56 | No | | T50F451 (V5-3084) | GLB | SO | A | -- | B, K | B, K | O | O | C | C | C | No | B | Yes | Yes | Note 10 |
| | 6I721-2679-1 | 56 | No | | T5000F455 (V5-2239) | BAL | AO | A | M | B, K | B, K | O | O | C | C | C | No | B | Yes | Yes | Note 10 |
| X-216A | -- | -- | -- | Spare | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | Type A Test |
| X-216B | -- | -- | -- | Spare | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | Type A Test |
| X-217 | -- | -- | -- | Spare | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | Type A Test |
| | | | | | | | | | | | | | | | | | | | | | |

TABLE 6.2-2 SUMMARY OF PRIMARY CONTAINMENT PENETRATIONS AND ASSOCIATED ISOLATION VALVES

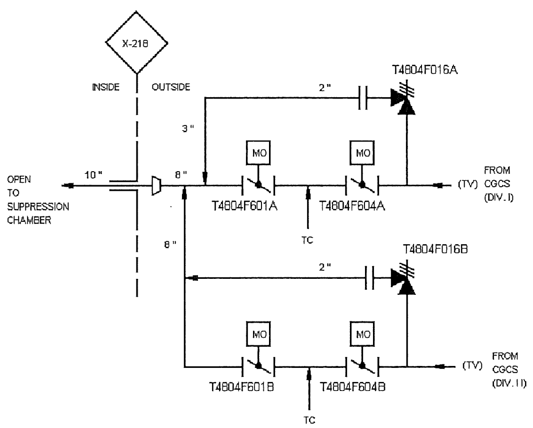
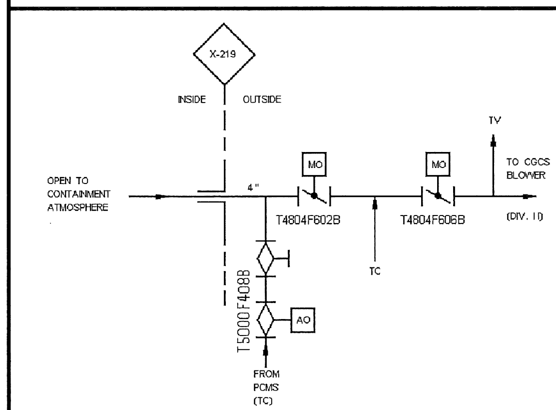
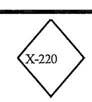
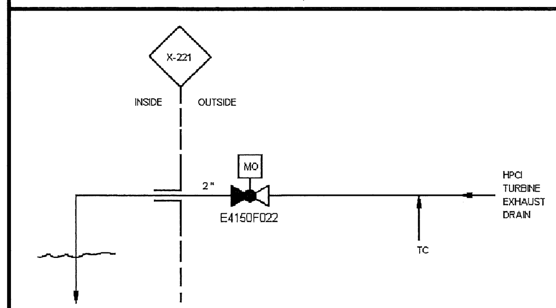
| PENETRATION DATA | | | | ISOLATION VALVE DATA | | | | | | | | | | | | | | | | | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| | | | | | | | | | | | | Valve Position | | | | | | | | | |
| Penetration Detail | P&ID Number | Design Criteria | Bypass Leakage Path | System Title | Valve Number | Valve Type | Actuator Type | Primary Actuation Mode | Secondary Actuation Mode | Containment Isolation Signal(s) | Accident Isolation Signal(s) | Normal | Shutdown | Post-LOCA (Accident) | Power Failure | ILRT | Engineered Safety Feature | Quality Group | Type C Test | Leak Detection | Remarks |
|  | 6M721-2087 | 56 | No | Combustible Gas Control System Return | T4804F601A (V4-2140) | BFY | M | M | -- | -- | -- | LC | LC | LC | LC | LC | No | B | Yes | No | Notes 9, 10, 11, and 45 |
| | | | | | T4804F604A (V4-2148) | BFY | M | M | -- | -- | -- | LC | LC | LC | LC | LC | No | B | Yes | No | Notes 9 and 45 |
| | | | | | T4804F016A (V22-2122) | REL | SA | -- | -- | -- | -- | C | C | C | -- | C | Yes | B | Yes | Yes | Notes 27 and 28 |
| | | | | | T4804F601B (V4-2139) | BFY | M | M | -- | -- | -- | LC | LC | LC | LC | LC | No | B | Yes | No | Notes 9, 10, 11, and 45 |
| | | | | | T4804F604B (V4-2149) | BFY | M | M | -- | -- | -- | LC | LC | LC | LC | LC | No | B | Yes | No | Notes 9 and 45 |
| | | | | | T4804F016B (V22-2121) | REL | SA | -- | -- | -- | -- | C | C | C | -- | C | Yes | B | Yes | Yes | Notes 27 and 28 |
|  | 6M721-2087 | 56 | No | Combustible Gas Control System Suction | T4804F602B (V4-2141) | BFY | M | M | -- | -- | -- | LC | LC | LC | LC | LC | No | B | Yes | No | Notes 9, 10, 11, and 45 |
| | 6I721-2679-1 | 56 | No | | T5000F408B (V5-2166) | BAL | AO | RM | M | -- | -- | O | O | O | C | O | No | B | Yes | Yes | Notes 12 and 13 |
| | 6M721-2087 | 56 | No | | T4804F606B (V4-2155) | BFY | M | M | -- | -- | -- | LC | LC | LC | LC | LC | No | B | Yes | No | Notes 9 and 45 |
|  | -- | -- | -- | High-Pressure Coolant Injection Turbine Exhaust | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | See Penetration Detail X-212 |
|  | 6M721-2035 | 56 | No | High-Pressure Coolant Injection Turbine Exhaust Drain | E4150F022 (V11-2008) | SCK | MO | RF | RM | -- | -- | O | O | O | AIS | C | Yes | B | No | Yes | Notes 7, 12 and 39 |

TABLE 6.2-2 SUMMARY OF PRIMARY CONTAINMENT PENETRATIONS AND ASSOCIATED ISOLATION VALVES

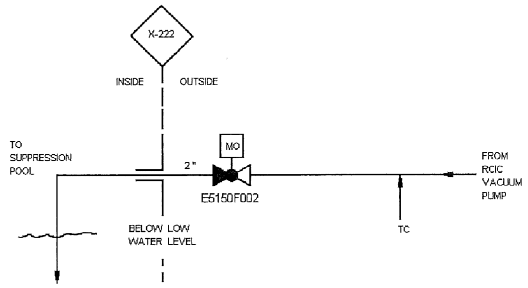
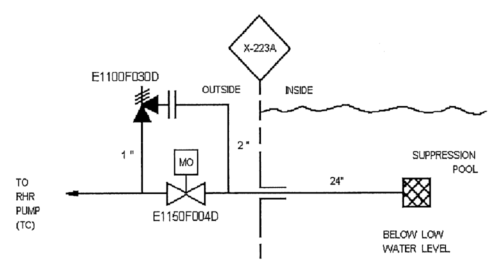
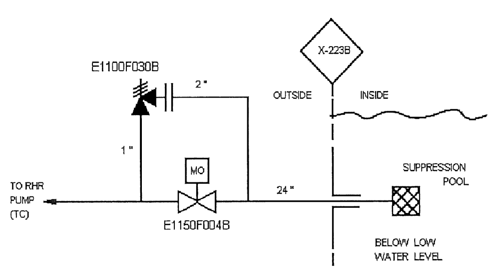
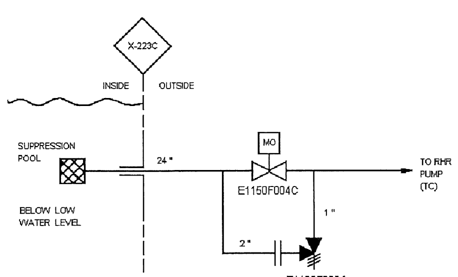
| PENETRATION DATA | | | | ISOLATION VALVE DATA | | | | | | | | | | | | | | | | | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| | | | | | | | | | | | | Valve Position | | | | | | | | | |
| Penetration Detail | P&ID Number | Design Criteria | Bypass Leakage Path | System Title | Valve Number | Valve Type | Actuator Type | Primary Actuation Mode | Secondary Actuation Mode | Containment Isolation Signal(s) | Accident Isolation Signal(s) | Normal | Shutdown | Post-LOCA (Accident) | Power Failure | ILRT | Engineered Safety Feature | Quality Group | Type C Test | Leak Detection | Remarks |
|  | 6M721-2044 | 56 | No | Reactor Core Isolation Cooling Vacuum Pump Discharge | E5150F002 (V8-2235) | SCK | MO | RF | RM | -- | -- | O | O | O | AIS | C | R | B | No | Yes | Notes 6, 12 and 39 |
|  | 6M721-2083 | 56 | No | Residual Heat Removal Pump Suction | E1150F004D (V8-2100) | GAT | MO | RM | M | -- | -- | O | O | O | AIS | O | Yes | B | No | Yes | Notes 7, 12, and 39 |
| | | | | Residual Heat Removal Pump Suction Header Thermal Relief | E1100F030D (V22-2035) | REL | SA | -- | -- | -- | -- | C | C | C | -- | C | Yes | B | No | Yes | Notes 27, 28, and 39 |
|  | 6M721-2083 | 56 | No | Residual Heat Removal Pump Suction | E1150F004B (V8-2102) | GAT | MO | RM | M | -- | -- | O | O | O | AIS | O | Yes | B | No | Yes | Notes 7, 12, and 39 |
| | | | | Residual Heat Removal Pump Suction Header Thermal Relief | E1100F030B (V22-2037) | REL | SA | -- | -- | -- | -- | C | C | C | -- | C | Yes | B | No | Yes | Notes 27, 28, and 39 |
|  | 6M721-2084 | 56 | No | Residual Heat Removal Pump Suction | E1150F004C (V8-2101) | GAT | MO | RM | M | -- | -- | O | O | O | AIS | O | Yes | B | No | Yes | Notes 7, 12, and 39 |
| | | | | Residual Heat Removal Pump Suction Header Thermal Relief | E1100F030C (V22-2036) | REL | SA | -- | -- | -- | -- | C | C | C | -- | C | Yes | B | No | Yes | Notes 27, 28, and 39 |

TABLE 6.2-2 SUMMARY OF PRIMARY CONTAINMENT PENETRATIONS AND ASSOCIATED ISOLATION VALVES

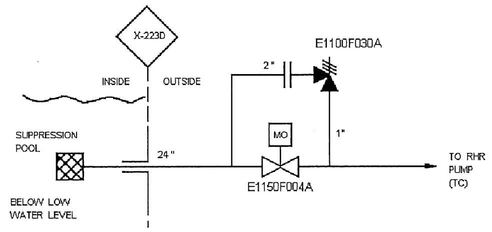
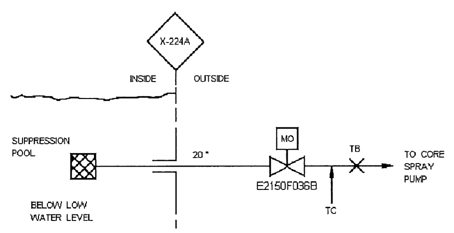
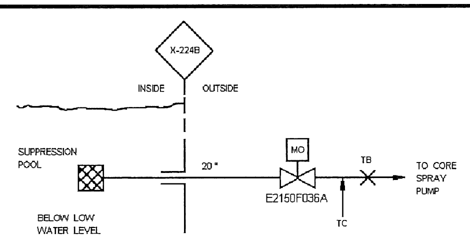
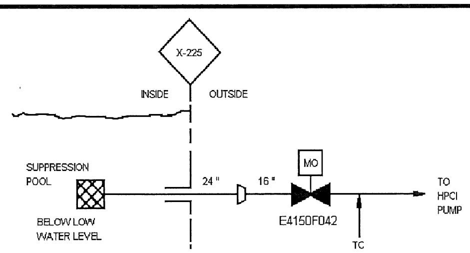
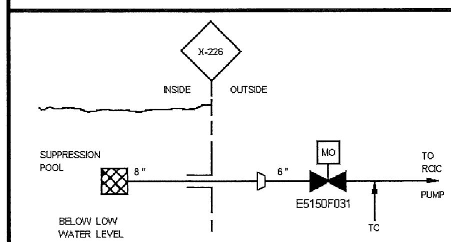
| PENETRATION DATA | | | | ISOLATION VALVE DATA | | | | | | | | | | | | | | | | | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| | | | | | | | | | | | | Valve Position | | | | | | | | | |
| Penetration Detail | P&ID Number | Design Criteria | Bypass Leakage Path | System Title | Valve Number | Valve Type | Actuator Type | Primary Actuation Mode | Secondary Actuation Mode | Containment Isolation Signal(s) | Accident Isolation Signal(s) | Normal | Shutdown | Post-LOCA (Accident) | Power Failure | ILRT | Engineered Safety Feature | Quality Group | Type C Test | Leak Detection | Remarks |
|  | 6M721-2084 | 56 | No | Residual Heat Removal Pump Suction | E1150F004A (V8-2099) | GAT | MO | RM | M | -- | -- | O | O | O | AIS | O | Yes | B | No | Yes | Notes 7, 12, and 39 |
| | | | | Residual Heat Removal Pump Suction Header Thermal Relief | E1100F030A (V22-2034) | REL | SA | -- | -- | -- | -- | C | C | C | -- | C | Yes | B | No | Yes | Notes 27, 28, and 39 |
|  | 6M721-2034 | 56 | No | Core Spray Pump Suction | E2150F036B (V8-2008) | GAT | MO | RM | M | -- | -- | O | O | O | AIS | O | Yes | B | No | Yes | Notes 7, 12, and 39 |
|  | 6M721-2034 | 56 | No | Core Spray Pump Suction | E2150F036A (V8-2007) | GAT | MO | RM | M | -- | -- | O | O | O | AIS | O | Yes | B | No | Yes | Notes 7, 12, and 39 |
|  | 6M721-2035 | 56 | No | High-Pressure Coolant Injection Pump Suction | E4150F042 (V8-2202) | GAT | MO | RM | M | -- | X | C | C | O | AIS | C | Yes | B | No | Yes | Notes 7, 12, 31, and 39 |
|  | 6M721-2044 | 56 | No | Reactor Core Isolation Cooling Pump Suction | E5150F031 (V8-2225) | GAT | MO | RM | M | -- | -- | C | C | O | AIS | C | R | B | No | Yes | Notes 6, 12, and 39 |

TABLE 6.2-2 SUMMARY OF PRIMARY CONTAINMENT PENETRATIONS AND ASSOCIATED ISOLATION VALVES

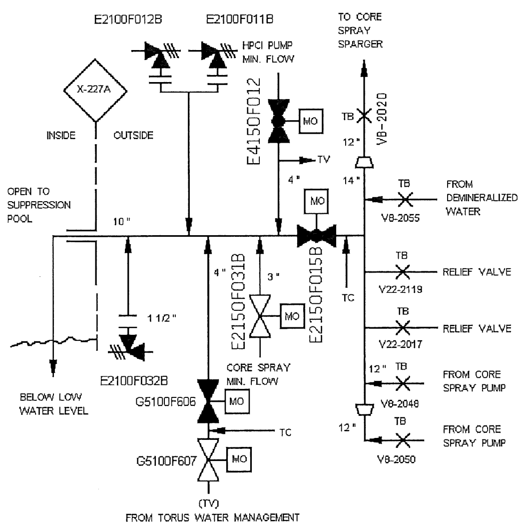
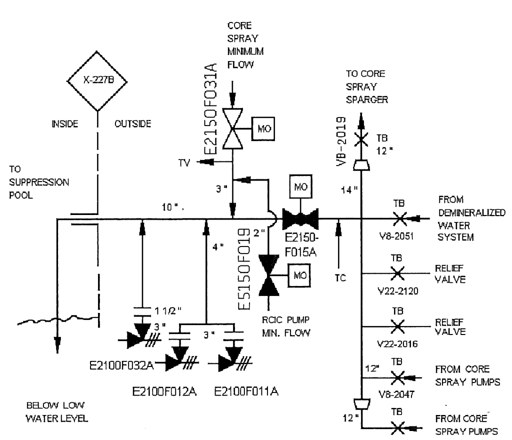
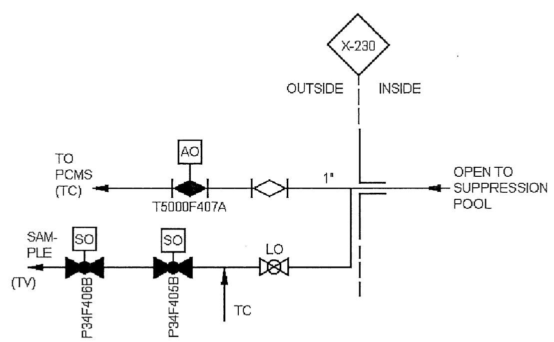
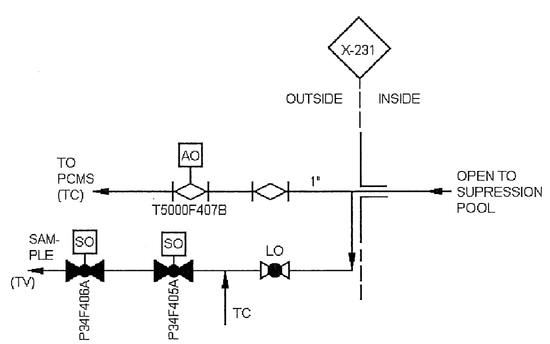
| PENETRATION DATA | | | | ISOLATION VALVE DATA | | | | | | | | | | | | | | | | | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| | | | | | | | | | | | | Valve Position | | | | | | | | | |
| Penetration Detail | P&ID Number | Design Criteria | Bypass Leakage Path | System Title | Valve Number | Valve Type | Actuator Type | Primary Actuation Mode | Secondary Actuation Mode | Containment Isolation Signal(s) | Accident Isolation Signal(s) | Normal | Shutdown | Post-LOCA (Accident) | Power Failure | ILRT | Engineered Safety Feature | Quality Group | Type C Test | Leak Detection | Remarks |
|  | 6M721-2034 | 56 | No | Core Spray Pump Suction Thermal Relief | E2100F032B (V22-2004) | REL | SA | -- | -- | -- | -- | C | C | C | -- | C | Yes | B | No | Yes | Notes 12, 27, 28, and 39 |
| | | 56 | No | Core Spray Pump Discharge Header Relief | E2100F012B (V22-2017) | REL | SA | -- | -- | -- | -- | C | C | C | -- | C | Yes | B | No | Yes | Notes 12, 26, 27, 28, and 39 |
| | | | | | E2100F011B (V22-2119) | REL | SA | -- | -- | -- | -- | C | C | C | -- | C | Yes | B | No | Yes | Notes 12, 27, 28, and 39 |
| | | 56 | No | Core Spray Pump Minimum Flow | E2150F031B (V8-2032) | GAT | MO | RM | M | -- | Z | O | O | C | AIS | C | Yes | B | No | Yes | Notes 7, 12, and 39 |
| | | 56 | No | Core Spray Pump Test Line | E2150F015B (V8-2034) | GLB | MO | A | M | A,K | -- | C | C | C | AIS | C | Yes | B | No | Yes | Notes 12 and 39 |
| | 6M721-4100 | 56 | No | Torus Water Managment System | G5100F606 (V8-4679) | GAT | MO | A | RM | B,K | M | C | C | C | AIS | C | No | B | Yes | No | Notes 4 and 26 |
| | | | | | G5100F607 (V8-4682) | GAT | MO | A | RM | B,K | M | O | C | C | AIS | C | No | B | Yes | No | Note 26 |
| | 6M721-2035 | 56 | No | High-Pressure Coolant Injection Minimum Flow | E4150F012 (V8-2196) | GLB | MO | RM | M | -- | Z | C | C | C | AIS | C | Yes | B | No | Yes | Notes 7, 12, and 39 |
|  | 6M721-2034 | 56 | No | Core Spray Pump Suction Thermal Relief | E2100F032A (V22-2019) | REL | SA | -- | -- | -- | -- | C | C | C | -- | C | Yes | B | No | Yes | Notes 12, 27, 28, and 39 |
| | | 56 | No | Core Spray Pump Discharge Header Relief | E2100F012A (V22-2016) | REL | SA | -- | -- | -- | -- | C | C | C | -- | C | Yes | B | No | Yes | Notes 12, 27, 28, and 39 |
| | | | | | E2100F011A (V22-2120) | REL | SA | -- | -- | -- | -- | C | C | C | -- | C | Yes | B | No | Yes | Notes 12, 27, 28, and 39 |
| | | 56 | No | Core Spray Pump Test Line | E2150F015A (V8-2033) | GLB | MO | A | RM | A,K | -- | C | C | C | AIS | C | Yes | B | No | Yes | Notes 12 and 39 |
| | 6M721-2044 | 56 | No | Reactor Core Isolation Cooling Minimum Flow | E5150F019 | GAT | MO | RM | M | -- | Z | C | C | C | AIS | C | R | B | No | Yes | Notes 6, 12, and 39 |
| | 6M721-2034 | 56 | No | Core Spray Minimum Flow | E2150F031A (V8-4683) | GAT | MO | RM | M | -- | Z | O | O | C | AIS | C | Yes | B | No | Yes | Notes 7, 12, and 39 |
| X-228A | -- | -- | -- | Torus - Low Voltage Switching | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | Type B Test |
| X-228B | -- | -- | -- | Torus - Low Voltage Switching | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | Type B Test |
| X-228C | -- | -- | -- | Torus - Low Voltage Switching | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | Type B Test |

TABLE 6.2-2 SUMMARY OF PRIMARY CONTAINMENT PENETRATIONS AND ASSOCIATED ISOLATION VALVES

| PENETRATION DATA | | | | ISOLATION VALVE DATA | | | | | | | | | | | | | | | | | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| | | | | | | | | | | | | Valve Position | | | | | | | | | |
| Penetration Detail | P&ID Number | Design Criteria | Bypass Leakage Path | System Title | Valve Number | Valve Type | Actuator Type | Primary Actuation Mode | Secondary Actuation Mode | Containment Isolation Signal(s) | Accident Isolation Signal(s) | Normal | Shutdown | Post-LOCA (Accident) | Power Failure | ILRT | Engineered Safety Feature | Quality Group | Type C Test | Leak Detection | Remarks |
| X-228D | -- | -- | -- | Torus - Low Voltage Switching | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | Type B Test |
| X-229 | -- | -- | -- | Spare | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | Type A Test |
|  | 6I721-2679-1 | 56 | No | Primary Containment Monitoring System Suction Division I | T5000F407A (V5-2157) | BAL | AO | RM | M | -- | -- | C | C | C | C | O | No | B | Yes | Yes | Notes 12, 13 and 31 (See Penetration X-48) |
| | | | | and | | | | | | | | | | | | | | | | | |
| | 6I721-2400-10 | 56 | Yes | Suppression Pool Postaccident Atmosphere Sample Suction | P34F405B (V13-7367) | GLB | SO | RM | -- | -- | -- | C | C | C | C | C | No | B | Yes | Yes | -- |
| | | | | | P34F406B (V13-7377) | GLB | SO | RM | -- | -- | -- | C | C | C | C | C | No | B | Yes | Yes | -- |
|  | 6I721-2679-1 | 56 | No | Primary Containment Monitoring System Suction Division II | T5000F407B (V5-2165) | BAL | AO | RM | M | -- | -- | O | C | C | C | O | No | B | Yes | Yes | Notes 12 and 13 |
| | | | | and | | | | | | | | | | | | | | | | | |
| | 6I721-2400-10 | 56 | Yes | Suppression Pool Postaccident Atmosphere Sample Suction | P34F405A (V13-7366) | GLB | SO | RM | -- | -- | -- | C | C | C | C | C | No | B | Yes | Yes | -- |
| | | | | | P34F406A (V13-7376) | GLB | SO | RM | -- | -- | -- | C | C | C | C | C | No | B | Yes | Yes | -- |
| | | | | | | | | | | | | | | | | | | | | | |
| | | | | | | | | | | | | | | | | | | | | | |

# TABLE 6.2-2 SUMMARY OF PRIMARY CONTAINMENT PENETRATIONS AND ASSOCIATED ISOLATION VALVES

## CODES AND SYMBOLS

### Penetration Details

Standard mechanical symbols are employed to represent piping details. Each penetration detail provides the following information:

| Symbol | Description |
|---|---|
| 1. ◇ | - Penetration details in numerical order |
| 2. (TC) | - Test connection for Type C testing |
| 3. (TV) | - Test vent for Type C testing |
| 4. TB | - Test Barrier |

### Valve Type

The following codes are used to identify valve type:

CHK - Check
GAT - Gate
GLB - Globe
BFY - Butterfly
REG - Regulating
BAL - Ball
REL - Relief
EFC - Excess Flow Check
SCK - Stop Check
BCK - Ball Check
SHR - Shear

### Actuator Type

The following codes are used to identify valve actuator type:

AO - Air Operator
SO - Solenoid Operator
MO - Motor Operator
M - Manual
SA - Self Actuated
EX - Explosive

### Valve Position

The following codes are used to identify different valve positions:

O - Open
C - Closed
AIS - As Is
LC - Locked Closed
LO - Locked Open

### Miscellaneous

A dash (-) indicates that technical information is not applicable to this column

This column references the appropriate General Design Criteria of 10 CFR 50, Appendix A (or other defined basis) with which the penetration is in compliance.

### Bypass Leakage

Bypass leakage paths are identified in this table by a "Yes or a "No". Bypass leakage paths are further discussed in Section 6.2.1.2.2.3.

### Primary/Secondary Actuation Modes

These columns indicate the nature of the containment isolation signal as follows:

A - Automatic
RM - Remote Manual
RF - Reverse Flow
M - Manual
HF - High Flow

### Engineered Safety Feature

Valves in Engineered Safety Feature systems are identified in this column by a "yes" entry. An "R" entry identifies a valve in an Engineered Safety Feature-related system. Such Systems are not required to function following the design basis loss-of-coolant accident. However, if the system is available, it can be used to accomplish a function similar to an Engineered Safety Feature system.

### Containment/Accident Isolation Signals

The following codes are used to abbreviate isolation signals:

| Signal | Description |
|---|---|
| A | Reactor Vessel Low Level 1 |
| B | Reactor Vessel Low Level 2 |
| C | Reactor Vessel Low Level 3 |
| D | Main Steam Line High Radiation |
| E | Main Steam Line High Flow |
| F | Main Steam Line Tunnel High Temperature |
| G | Main Steam Line Low Pressure |
| H | Torus Pressure ≥ Secondary Containment Pressure |
| J | Low Condenser Vacuum |
| K | High Drywell Pressure |
| L | High Reactor Vessel Pressure |
| M | High-High Sump Level Torus Area or High-High Drywell Floor Drain Sump Level |
| N | High Sump Level or High Sump Temperatures |
| P | Turbine Building High Temperatures |
| R | Reactor Building Exhaust Radiation High |
| W | RWCU System Line<br>1) SLCS Initiation (Outboard Valve Only)<br>2) High RWCU Differential Flow<br>3) High RWCU Area Temperature<br>4) High RWCU Area Ventilation Differential Temperature<br>5) Reactor Vessel Low Water Level 2 |
| X | HPCI System Steam Lines<br>1) HPCI Space High Temp.<br>2) High Steam Flow<br>3) High Turbine Exhaust Pressure<br>4) HPCI Steam Line Low Pressure |
| Y | RCIC System Steam Lines<br>1) RCIC Space High Temp<br>2) High Turbine Exhaust Pressure<br>3) High Steam Flow<br>4) RCIC Steam Line Low Pressure |
| Z | Closes Through Electrical Interlocks With Other System Valves of Pump Motors |

## NOTES

| Note | Description |
|---|---|
| 1. | This piping consists of the eight vent pipes that connect the drywell and pressure suppression chamber. As part of the primary containment structure, they are Type A tested. |
| 2. | Globe valve tested in the reverse direction. Results obtained in this test configuration are conservative since test pressure tends to unseat the valve disk. |
| 3. | These valves will be tested at a differential pressure of 25 psi with the reactor at atmospheric pressure (i.e., 25 psig). |
| 4. | Gate valve tested through the bonnet. This valve has a bonnet tap through which the bonnet area is pressurized. Leakage is measured through both seating surfaces along with leakage through the bonnet. Compared with testing in the accident direction, the bonnet test leakage is conservative. |
| 5. | Air-operated, spring-to-close, positive-acting check valve. Can be closed by remote manual operation from the control room when system isolation is required. |
| 6. | Remote manual containment isolation valve in an ESF-related system. Provisions are made to detect leakage from this line outside the containment. See Table 5.2-11. |
| 7. | Remote manual containment isolation valve in an ESF system. Provisions are made to detect leakage from this line outside the containment. See Table 5.2-11. |
| 8. | Valve isolates when reactor pressure exceeds 75 psig. |
| 9. | Manual or remote manual valve that is locked closed and remains closed after a LOCA. |
| 10. | Two containment isolation valves located outside the containment. Due to the design of both the containment and the system, it is not practical to locate one of the two valves inside the containment. Both valves are located outside the containment as close as practical to the containment wall. |
| 11. | Butterfly valve tested in the reverse direction. Reverse flow tests are designed to provide equivalent or conservative results compared with testing in the accident direction. In cases where stem leakage is not measured by the leakage out of the test volume, stem leakage is determined by testing through the stem vent and this leakage is added to the test volume leakage. Additional tests on purge system butterfly valves are set forth in the Technical Specifications. |
| 12. | Single isolation valve and closed system outside the containment. This line contains a single isolation valve based on<br>a. The line is in an ESF or ESF-related system<br>b. System reliability is greater with one isolation valve<br>c. The system is a closed system outside the containment<br>d. A single active failure can be accommodated with one isolation valve in the line.<br>The specific closed system requirements met by this system outside the containment include missile protection, Category I, and Quality Group B design standards.<br>For instrumentation piping, the systems are designed and installed as Quality Group B, up to and including the isolation valves. The balance of the instrument piping is designed to meet Quality Group B design criteria. These design criteria include stress analysis with consideration given to dead-weight, thermal, and seismic conditions. The systems are seismically supported. Nuclear grade material is used throughout the fabrication of the piping system.<br>The design temperature and pressure ratings of the systems are greater than those of the containment. |
| 13. | Ball valve tested in the reverse direction. Results obtained in this configuration are equivalent to testing in the accident direction, since valves of this type have the same sealing characteristics in either direction. |
| 14. | Jet pump flow instrumentation lines are provided with manual globe valves and excess-flow check valves outside the containment. Also, flow is restricted to a 1/4-in. orifice at the nozzle. Therefore, these instrumentation lines are designed in accordance with the recommendations of Regulatory Guide 1.11 (Safety Guide 11). All instrument line penetrations will be Type A tested. |
| 15. | Instrumentation penetration. Standard instrumentation penetrations are provided with six instrument tubes. In this table, only those tubes that are utilized by instrument lines are addressed. All other instrument tubes associated with the penetration are spares and are Type A tested. |
| 16. | Instrument lines of this type are provided with a flow-restricting orifice inside the containment and a manual globe valve and excess-flow check valve outside the containment in accordance with the recommendations of Regulatory Guide 1.11 (Safety Guide 11). All instrument line penetrations will be Type A tested. |
| 17. | The TIP system lines do not communicate freely with the containment atmosphere or the reactor coolant. General Design Criteria 55 and 56 are not directly applicable to this specific class of lines. The basis to which these lines are designed is more closely described by GDC 54, which states in effect that the isolation capability of a system should be commensurate with the safety importance of that isolation. Furthermore, even though the failure of the TIP system lines presents no safety consideration, the TIP system guide tubes have redundant isolation capabilities. The safety features have been reviewed by the NRC for BWR/4 (Duane Arnold), BWR/5 (Nine Mile Point) and BWR/6 (GESSAR), and it was concluded that the design of the containment isolation system meets the objectives and intent of the GDC.<br>The TIP guide tube assembly and the portion of the tubing between the guide tube assembly and the containment flange are considered to be instruments and as a result are not classified as ASME components but are purchased and installed as safety-related assemblies.<br>A valve system is provided with a valve on each guide tube entering the primary containment. These valves are closed except when the TIP system is in operation (open an average of 15 hr/month). A ball valve and a cable-shearing valve are mounted in the guide tubing just outside the primary containment. They prevent the loss of containment integrity. The ball valve position is indicated in the control room. The shear valve is used only if containment isolation is required when the TIP is beyond the ball valve and when power to the TIP system fails. The shear valve, which is controlled by a manually operated keylock switch, can cut the cable and close off the guide tube. The shear valves are actuated by detonation squibs. The continuity of the squib circuits is monitored by indicator lights in the main control room. Identical-design shear valves are shop tested by statistical |

TABLE 6.2-2 SUMMARY OF PRIMARY CONTAINMENT PENETRATIONS AND ASSOCIATED ISOLATION VALVES

| Note | Description |
|---|---|
| | sampling methods to ensure operability and leaktightness.<br>The Technical Specification testing requirements for the TIP shear and ball valves are<br>a. Verifying the continuity of the explosive charges at least every 31 days<br>b. Initiating one of the explosive charges once every fuel cycle. The replacement charge shall be from the same manufactured batch as the one fired or from another batch that has been certified by having one of -that batch successfully fired<br>c. Replacing all charges according to the manufacturer's recommended lifetime for the charges<br>d. Performing Type C tests on the ball valves in accordance with a performance based leak testing program in Technical Specification 5.5.12. |
| 18. | A plant modification, EDP-4940, changed the routing of the nitrogen purge supply line to the TIP system. The nitrogen source is taken from a primary containment pneumatic system line inside the drywell. Penetration X-35G thus becomes a spare, the pipe penetrating the drywell is capped and welded, and GDC 56 no longer applies. |
| 19. | The control rod drive (CRD) insert and withdrawal line. Each of the 185 CRD withdrawal lines is separated from the RPV by a redundant seal design in the CRD units. Each of the 185 CRD insert lines has a CRDM flange ball check valve that isolates the line from the RPV following a scram. The redundant seal system, the CRDM flange ball check valve, and a manual isolation valve provide adequate isolation in the event of a line break in the hydraulic control unit (HCU) or the scram discharge volume (SDV).<br>During a scram, the 185 outlet scram valves open and the four SDV vent and drain valves close. If the scram system is not reset, thus closing the scram inlet and outlet valves, CRD seal leakage could slowly pressurize the SDV to the RPV pressure. Therefore, the SDV vent and drain valves along with 185 drive and 185 cooling water ball check valves will be Type A tested.<br>Leakage from the CRD system into the reactor building is detected for the full spectrum of leakage rates. Small leaks will be detected by observation during daily inspection rounds of the control unit areas by operators. Large leaks will be detected by duty timers on the reactor building floor drain sump pumps. A large leak of reactor coolant from any insert line will be automatically isolated by the ball check valve in the CRD housing. Leaks of CRD supply water will be indicated by increased flow as continuously recorded in the control room. The CRD directional control valves are normally closed and are automatically closed upon a reactor scram signal. Excessive leakages through the scram valves will be detected by duty timers on the sump pumps. |
| 20. | The residual heat removal (RHR) system discharge line to the containment spray header receives an automatic containment isolation signal. The operator may, however manually override the isolation signal as needed to reduce containment temperature and pressure. |
| 21. | Due to system configuration, the test pressure is not in the same direction as the pressure existing when the valve is required to perform its containment isolation function. The valve will be tested in the correct direction during the Type A tests. |
| 22. | The standby liquid control system (SLCS) has been designed to reflect the importance of the functions it may perform. The probability of reliable and timely actuation of this system is enhanced by inclusion of fewer valves and simplicity of design. The use of a check valve outside the containment is consistent with these system design requirements. |
| 23. | This is a penetration of the vent pipe inside the torus and thus is not a bona fide containment penetration. It is included in this table for completeness only. |
| 24. | This butterfly valve is normally closed and opens automatically to prevent formation of negative pressure in the torus. This butterfly valve closes automatically upon increasing torus pressure and remains closed during containment high-pressure conditions. Upon loss of power or degraded voltage, the butterfly valve will open but closes automatically once power is restored or voltage recovers. |
| 25. | Secondary containment to torus vacuum breaker. This vacuum breaker opens automatically to prevent formation of a negative pressure in the torus. This line is essential to ensure primary containment structural integrity. The probability of system operation is enhanced by using fewer valves and by the simplicity of design. The use of a check valve outside containment is consistent with these system design requirements. |
| 26. | The flow path associated with this penetration inside containment terminated below the low water level in the suppression pool. A water seal is assured during normal plant operation and for more than 30 days following an accident requiring containment isolation. It is not credible that these isolation valves will be exposed to the containment atmosphere at any time following the accident.<br>These penetration containment isolation valves will be Type C seat leak tested using water or air as the test medium. In some cases, due to system configuration, the Type C test pressure will not be in the same direction as the pressure existing when the valve is required to perform the containment isolation function. The valves will be a part of the periodic Type A tests where the test pressure is applied in the correct direction. |
| 27. | Relief valve used as a containment isolation valve. The construction and orientation of this valve is such that increased containment pressure acts in conjunction with spring pressure to increase the seating force of the valve and tends to reduce leakage. Thus the relief valve setpoint has no bearing on containment isolation. |
| 28. | Flange on relief valve discharge to be Type B tested. |
| 29. | Due to system configuration, the test pressure on the relief valve is not in the same direction as the pressure existing when the relief valve is required to perform its containment isolation function. The test pressure is applied under the relief valve seat. This is conservative since it tends to unseat the relief valve. |
| 30. | Not used. |
| 31. | These valves do not close on the containment isolation signals, but automatically close on accident isolation signal as identified in Table 6.2-2. |
| 32. | These valves close on the containment isolation signals but are also provided with a manual override to these signals to reopen the valves. This is done to provide divisional control air/nitrogen for the controls of pneumatic equipment/instruments inside the drywell. |
| 33. | Flanged portion of the residual heat removal head spray piping, between reactor pressure vessel and the refueling floor bulkhead penetration, is permanently removed.<br>Remaining line within the drywell is blind flanged. |
| 34. | Valve closes on a high drywell pressure signal to isolate the drywell heat load. |
| 35. | SLCS initiation signal is not a containment isolation signal. This signal prevents removal of liquid poison in the event of standby liquid control system actuation. |
| 36. | Penetrations X-13A and X-13B will have a 30-day water seal during and following a postulated LOCA. Therefore valves E1100-F050A, F610A, F050B, and F610B are not considered containment isolation valves and therefore are not subject to Type C testing (see TS Amendment 98). |
| 37. | 30-day water seal for penetrations X-13A and X-13B (during and following a postulated LOCA) requires external water leakage, through valves E1150-F015A and F015B, to be less than 5 ml/min. at 1.1 Pa, i.e., 62.2 psig. As these valves will be subjected to a more conservative PIV test that demonstrates external leakage less than 5 ml/min. at a pressure of 1045 psig, TS Amendment 98 exempts these valves from Type C air test. |
| 38. | Flexible wedge gate valve tested in the accident direction and through a body\bonnet tap from a single test connection. Leakage is measured past the outboard seating surface. |
| 39. | Single isolation valves on a closed system both inside and outside of containment.<br>The flow path associated with this penetration inside containment terminates below the low water level in the suppression pool. A water seal is assured during normal plant operation and for more than 30 days following an accident requiring containment isolation. It is not credible that these isolation valves will be exposed to the containment atmosphere at any time following the accident.<br>Also many of the valves are required to be open post-accident to fulfill their safety function.<br>Type C LLRT testing is not required when the valve is located on a closed system and on a line which terminates below the minimum water level of the suppression chamber. |
| 40. | Valve is interlocked to inhibit opening on low reactor water level signal (Level 1) to ensure all RHR flow is directed to the reactor vessel. |
| 41. | A plant modification, EDP 33297, modified the service air piping to the drywell. The pipe penetrating the drywell is plugged and welded and GDC 56 no longer applies. |
| 42. | A plant modification, EDP 35267, plugged the Main Steam Isolation Valve Leak Control System at the Main Steam Line interface. Therefore, GDC 55 no longer applies. |
| 43. | In case of a loss of power event, T5000F420B can be reopened using a DC solenoid valve with a dedicated nitrogen supply system. |
| 44. | In case of a Beyond Design Basis External Event, T4600F400 and T4600F401 can be reopened using a 3-way piloted shuttle valve with a dedicated nitrogen supply system. |
| 45. | The Combustible Gas Control System (CGCS) has been retired in place with its electrical circuits de-energized and fluid process piping isolated from primary containment with redundant isolation valves. These valves are locked closed and can only be operated locally. |

TABLE 6.2-3 ELECTRICAL PENETRATION SCHEDULE

| Penetration Number | Penetration Type |
|---|---|
| T2301-X-100A | Neutron monitor |
| T2301-X-100B | “ |
| T2301-X-100F | “ |
| T2301-X-100G | “ |
| T2301-X-101A | Medium voltage power (5kV) |
| | Recirculating pump power |
| T2301-X-101B | “ |
| T2301-X-101C | “ |
| T2301-X-101D | “ |
| T2301-X-101E | “ |
| T2301-X-101F | “ |
| T2301-X-102A | Low-voltage switching/RPS |
| T2301-X-102B | “ |
| T2301-X-102C | “ |
| T2301-X-102D | “ |
| T2301-X-103A | Thermocouples |
| T2301-X-103B | “ |
| T2301-X-104A | Control rod drive position indicators |
| T2301-X-104B | “ |
| T2301-X-104C | “ |
| T2301-X-104D | “ |
| T2301-X-104E | “ |
| T2301-X-104F | “ |
| T2301-X-105A | Low-voltage power (480 V) |
| T2301-X-105D | “ |
| T2301-X -106A | Low level signal vibration test |
| T2301-X-106B | “ |

TABLE 6.2-4 DRYWELL TO SUPPRESSION CHAMBER VACUUM BREAKER VALVE DATA

| | | |
|---|---|---|
| 1. | Number of drywell-to-suppression chamber vacuum breaker valves | 12 |
| 2. | Valve size | 20 in. seat x 18 in. flanged outlet |
| 3. | Valve location | |
| | Elevation | Valve centerline 562 ft 8.5 in. |
| | Position | Two valves on each drywell support chamber downcomer with azimuth locations 22° - 30', 67° - 30' 112° - 30', 247° - 30', 292° - 30', and 337° - 30' |
| 4. | Differential pressure to open | 0.5 psid |
| 5. | Valve manufacturer | GPE Controls of Morton Grove, Illinois |
| 6. | Design temperature | 350 °F |
| 7. | Design pressure | 62 psig |
| 8. | Hydrostatic test pressure | 87 psig |
| 9. | Valve position indication | Limit switches, circuitry, and indicator lights. Closed limit switches are redundant |
| 10. | Main control room panel number | Indicating lights are on panel H11-P808 and H11-P817 |

TABLE 6.2-5 PRIMARY SYSTEM ENERGY DISTRIBUTION AT THE TIME A RECIRCULATION LINE BREAK ACCIDENT OCCURS

| Fluid Energy[a] | Energy ($10^6$ Btu) |
|---|---|
| 1. Steam | 30.6 |
| 2. Liquid | 346.8 |
| Sensible Energy | |
| 1. Reactor pressure vessel | 103 |
| 2. Reactor internals (less core) | 78.9 |
| 3. Core | 7.5 |

[a] All energy values are based on a 32 °F datum. Fuel energy is based on a datum of 285 °F.

TABLE 6.2-6 HAS BEEN INTENTIONALLY DELETED

TABLE 6.2-7 ACCIDENT CHRONOLOGY DESIGN BASIS RECIRCULATION LINE BREAK ACCIDENT

| | | Time (sec) Minimum ECCS Available |
|---|---|---|
| 1. | Vents cleared | 0.2 |
| 2. | Drywell reaches peak pressure | 4.6 |
| 3. | Maximum positive differential pressure occurs | 4.6 |
| 4. | Initiation of the ECCS | 60** |
| 5. | Vessel reflooded | 220* |
| 6. | Introduction of RHR heat exchanger | 1200 |
| 7. | Containment reaches peak secondary pressure | $1 \times 10^4$ (2.8 hr) |

* This value taken from the containment analysis models; it is onlysignificant in confirming that core reflooding occurs before pool cooling or other RHR functions are needed.

** The containment analysis was based on a 30-sec maximum analyzed HPCI response time. The HPCI design basis has been subsequently revised to incorporate a 60-sec maximum system response time. The containment analysis is not impacted by increasing the HPCI response time from 30 sec to 60 sec. The short-term containment analysis calculates a peak containment pressure before the HPCI injects. The long-term calculation assumes one RHR loop is operating in the containment cooling mode at partial pumping capacity. Core cooling is provided by the core spray system and the RHR/LPCI pump and no credit was taken for the HPCI system (UFSAR Section 6.2.1.3.3) for long-term core makeup.

TABLE 6.2-8 PRIMARY AND SECONDARY CONTAINMENTS SURFACE COATING SCHEDULE PRIMARY CONTAINMENT[f]

| Type of Coating | Location | Approx. Average DFT[a] (mils) | Approx. Total Surface ($ft^2$) |
|---|---|---|---|
| Carbozinc 11 | Drywell interior steel<br>Interior structural steel hangers and supports<br>Vent line interior | 7 | 120,000 |
| Plasite 7155[bc] | Torus interior | 12 | 38,000 |
| Carboguard 6250 N[b,c,e] | Torus interior<br>Vent header interior<br>Vent line interior tie-in to vent header | 40 | 34,300 |
| Ameron 66 and Surfacer[b] | RPV support pedestal<br>Drywell concrete floors<br>Drywell concrete walls | 1/16 in. plus10 mils | 7,380 |
| Unqualified Paints[e] | Miscellaneous | Note e | Note e |
| Carboguard 890N | Drywell dado region | 6 | 232 |

SECONDARY CONTAINMENT

| Location | Area($ft^2$) | Primer | Thickness, Approx. (in.) | Top Coat | Thickness, Approx. (in) |
|---|---|---|---|---|---|
| Drywell exterior steel | 36,200 | Carbozinc 11 | 0.002 ± 0.002<br>0.001 | None | - |
| Torus exterior steel | 84,000 | Carbomastic 15 | 0.002 - 0.009 | None | - |
| Secondary containment concrete | 109,200 | Carboline 295 | 0.020 - 0.040 | Carboline 288 | 0.008 |
| Secondary containment structural steel | 29,000 | Type II red lead[d] | 0.001 to 0.00153 | Cook's Amercote Enamel[d] | 0.002 |

[a] DFT = Dry Film Thickness.

[b] Qualified Coating; other coatings are unqualified.

[c] Other compatible touch-up coatings are used inside the torus.

[d] Other compatible coatings per Specification 3071-055 per painting system PS-2 are used on top or in lieu of Cook's Amercote Enamel and Type II red lead.

[e] For current unidentified and unqualified coating totals, see the design calculations for the Torus strainers.

[f] Coating materials listed in this table are estimated quantities of significant coating materials inside containment. Actual coating materials and quantities inside containment are managed as indicated in design calculations for the Torus strainers.

TABLE 6.2-9 WIRING INSULATION

| | Type | Approximate Amount (lb) |
|---|---|---|
| Primary Containment Cable | | |
| Power and control cable | EPR Hypalon Silicone Rubber | 5,340<br>25 |
| Instrument | Cross-linked Polyethylene or Polyolefin | 5 |
| Thermocouple | Polyamide Capton | 18 |
| Secondary Containment Cable | | |
| Power | EPR Hypalon and Neoprene | 137,000[a] |
| Control | EPR Hypalon and Neoprene | 14,000[a] |
| Instrument | Cross-linked Polyethylene by Raychem | 99,000[a] |
| Thermocouple | Polyamide Capton | 25 |

[a] In addition to the amounts shown here in cable trays, there is an approximate additional 15 percent in conduit.

TABLE 6.2-10 OTHER ORGANIC COMPOUNDS

| | Commercial Name | Compound | Quantity (approx.) |
|---|---|---|---|
| Primary Containment | | | |
| Shell to concrete joints | Dow Corning/ DOWSIL 790* | Silicone rubber | 1900 $in.^3$ |
| Concrete floor to wall joints | Carboline 225 | Epoxy polysulfide | 400 $in.^3$ |
| Secondary Containment | | | |
| Steel wall panel gaskets | Blanchard Foam Guard | Polyvinyl chloride | 40 $ft^3$-340 lb to 620 lb |
| Steel wall panel caulking | 3M THIOKOL Weatherban | Polysulfide rubber | 40 $ft^3$-580 lb |
| Concrete floor joints | Carboline 225 | Epoxy polysulfide | 14 $ft^3$ |

* Note: Dow Corning 790 is retained as a historic information as Dow Corning 790 product name was rebranded to DOWSIL 790.

TABLE 6.2-11 STANDBY GAS TREATMENT SYSTEM MAJOR COMPONENT DESCRIPTION

| | |
|---|---|
| Filter Train | |
| Type | Multiple filters for removal or particulates, elemental iodine, and organic iodine from air |
| Quantity | Two 100 percent-capacity trains |
| Capacity, scfm air | 4000 each |
| Demister (Each Train) | |
| Type | Impingement |
| Quantity | One |
| Water removal rate, lb/min | 20 |
| Static resistance at design flow in. $H_2O$ | 1 max at 4000 scfm with 0.005 lb/ft$^3$ of free moisture |
| Prefilter (Each Train) | |
| Type | Dry disposable cartridge |
| Quantity | One bank |
| Capacity, scfm air | 4000 |
| Media | Glass fiber |
| Efficiency, percent | 85 (NBS dust spot) |
| Heater (Each Train) | |
| Type | Electric, open, single-stage, on-off |
| Quantity | One |
| Capacity, kW | 24 |
| Accessories | Overload cutout |
| HEPA Filters (Each Train) | |
| Type | High-efficiency, dry |

TABLE 6.2-11 STANDBY GAS TREATMENT SYSTEM MAJOR COMPONENT DESCRIPTION

| | |
|---|---|
| Quantity | Two banks (one before and one after charcoal absorber) |
| Elements per bank | Four |
| Capacity, scfm air | 4000 |
| Media | Waterproof, glass |
| Separator material | Aluminum |
| Frame material | Steel |
| Charcoal Adsorber Bed (Each Train) | |
| Type | Deep bed |
| Quantity | One |
| Capacity, scfm air | 4000 |
| Media | Impregnated Carbon |
| Quantity of media, lb | 1250-1500 lbm (nominal) |
| Efficiency | Lab tested to ensure 99.9 percent removal efficiency for methyl iodide. Installed and tested in the adsorber housing such that an overall decontamination efficiency of 99 percent is assumed for removal of all forms of gaseous iodine. |
| Charcoal volume, $ft^3$ | 50 (approximately) |
| Charcoal density, $lb/ft^3$ | 23.7 $Min^m$ |
| Depth of bed, in. | 6 |
| Face velocity, ft/minute | 40 |
| Residence time, seconds | 0.75 |
| Ignition temperature, °F | 625 (approximately) |
| Iodine desorption temperature, °F | 355 |

TABLE 6.2-11 STANDBY GAS TREATMENT SYSTEM MAJOR COMPONENT DESCRIPTION

| | |
|---|---|
| Charcoal loading, mg iodine/g carbon (30-day accident duration) | 2.5 (approximately) |

Media Particle Size Distribution USS Mesh

| | |
|---|---|
| 8 | 3 percent |
| 12 | 51 percent |
| 16 | 40 percent |
| 18 | 5 percent |
| Fines | 1 percent |

SGTS Exhaust Blower (Each Train)

| | |
|---|---|
| Quantity and type | One Centrifugal (with inlet vanes) |
| Capacity, scfm | 4000 |
| Static pressure, in. $H_2O$ | 20 |
| Drive | V-belt |
| Motor, hp | 25 |

Standby Cooling Air Fan

| | |
|---|---|
| Quantity and type | One centrifugal |
| Capacity, scfm air | 1000 |
| Static pressure, in. $H_2O$ | 18 |
| Drive | V-belt |
| Motor, hp | 10 |

TABLE 6.2-12 STANDBY GAS TREATMENT SYSTEM EQUIPMENT FAILURE ANALYSIS

| Component | Failure | Failure Detected By | Action |
|---|---|---|---|
| SGTS primary blower | Monitor burnout, drive shaft break, trip, etc. | Flow monitor – low pressure switch | Main control board alarm<br>Operating equipment train shutdown (manual)<br>Redundant train startup (manual) |
| Electric heating coil | Element overheat | High temperature cutout on coil | Circuit trip |
| Electric heating coil | Element burnout | Temperature indicator or moisture detector upstream of adsorber | High moisture alarm<br>Operating equipment train shutdown (manual)<br>Redundant train startup (manual) |
| Standby cooling fan | No start or failure results in high charcoal adsorber temperature | Temperature switches | Alarm sounds in main control room (automatic if setpoint achieved)<br>$CO_2$ is auto backup to cooling fan if charcoal bed temperature raises to 310 °F |
| Flow-control valve | Falls in open position | High ΔP indicator across filters, demisters, and adsorber | Main control board alarm |
| | | High building vacuum alarm | Operating equipment train shut-down (manual)<br>Redundant train startup (manual)<br>Isolation valves positioned (automatic) |
| Isolation valve | Falls in open position | Local indicator light | No automatic action.<br>Requires backflow prevention and building isolation accomplished by series valves |

TABLE 6.2-12 STANDBY GAS TREATMENT SYSTEM EQUIPMENT FAILURE ANALYSIS

| Component | Failure | Failure Detected By | Action |
|---|---|---|---|
| | Falls in closed position | Flow monitor – low-pressure switch | Main control board alarm<br>Operating equipment train shutdown (manual)<br>Redundant train startup (manual)<br>Isolation valve positioned (automatic) |
| HEPA filter | High particulate loading | High ΔP indication | Operating equipment train shutdown (manual)<br>Redundant train startup (manual)<br>Isolation valves positioned (automatic) |
| Charcoal filter | High temperature | Temperature elements | Alarm sounds start cooling fan |

TABLE 6.2-13 REMOTE MANUALLY OPERATED CONTAINMENT ISOLATION VALVES WITH LEAK DETECTION CAPABILITY

| Penetration | Valve | Penetration | Valve |
|---|---|---|---|
| | | X-27 | T5000F401B |
| | | | T5000F402B |
| | | | T5000F403B |
| | | | T5000F404B |
| | | | T5000F405B |
| | | | T50-F458 |
| | | | P34F403A |
| | | | P34F404A |
| X-9A | E4150F006 | | |
| X-9B | E5150F013 | X-29Be | T5000F420B |
| X-10 | E5150F007 | X-34A | P4400F606B |
| | E5150F008 | X-34B | P4400F615 |
| | | | P4400F607B |
| X-11 | E4150F002 | | |
| | E4150F003 | X-35B-F | TIP shear valves |
| | E4150F600 | | |
| X-13A | E1150F015B | | |
| | E1150F610B | X-40Dd | P34F401B |
| X-13B | E1150F015A | | |
| | E1150F610A | | |
| | | X-47e | T5000F420A |
| X-16A | E2150F005B | X-48 | T5000F401A |
| | | | T5000F402A |
| X-16B | E2150F005A | | T5000F403A |
| | | | T5000F404A |
| | | | T5000F405A |
| | | X-219 | T5000F408B |
| X-23 | P4400F606A | | |
| X-24 | P4400F616 | | |
| | P4400F607A | X-223A | E1150F004D |
| | | X-223B | E1150F004B |
| X-48 | P34F403B | | |
| | P34F404B | X-223C | E1150F004C |

TABLE 6.2-13 REMOTE MANUALLY OPERATED CONTAINMENT ISOLATION VALVES WITH LEAK DETECTION CAPABILITY

| Penetration | Valve | Penetration | Valve |
|---|---|---|---|
| | | X-223D | E1150F004A |
| | | X-224A | E2150F036B |
| | | X-224B | E2150F036A |
| X-206A | E41F402 | X-225 | E4150F042 |
| X-206B | E41F403 | X-226 | E5150F031 |
| X-206C | E41F401 | X-227A | E2150F031B |
| | | | E4150F012 |
| X-206D | E41F400 | | |
| | | X-227B | E5150F019 |
| | | | E2150F031A |
| X-210A | E1150F007B | | |
| | E1150F026B | | |
| X-210B | E1150F007A | X-230 | T5000F407A |
| | P34F407 | | P34F405B |
| | P34F409 | | P34F406B |
| | | X-231 | T5000F407B |
| X-215 | T5000F408A | | P34F405A |
| | P34F408 | | P34F406A |
| | P34F410 | | |
| | | X-206E | T50F412A |
| | | X-206F | T50F412B |
| X-28Cf | P34F401A | | |
| X-29Bb | E11F412 | | |
| X-29Bc | E11F413 | | |
| X-47a | E11F414 | | |
| X-47b | E11F415 | | |

TABLE 6.2-14 PRIMARY CONTAINMENT PENETRATION PIPE LINES CONNECTING CLOSED-LOOP QUALITY GROUP B SYSTEMS TO QUALITY GROUP D SYSTEMS

| System | Line | Line Diameter (in.) | Separation Valve |
|---|---|---|---|
| Emergency core cooling – high pressure coolant injection | Turbine exhaust drain to barometric condenser | 1 | SOV |
| | Interstage tap to barometric condenser | 2 | MOV |
| | Pressure source from the condensate system through the Torus Water Management System (TWMS) supplying HPCI pump discharge piping | 3/4 | CV, CV |
| | Condensate to radwaste | 1 | CV, AOV, AOV (1) |
| | Suction from condensate storage | 14 | CV, MOV (1) |
| | Discharge to condensate storage | 10 | MOV, AOV (1) |
| | Steam drain to main condenser | 1 | AOV, AOV (1) |
| Emergency core cooling – core spray | Suction from condensate storage | 16 | LCV |
| | “Keep full” line from demineralizer water system | 3, 3 | CV, CV |
| Emergence core cooling - residual heat removal | “Keep full” line from demineralizer water system | 4, 4 | CV, CV |
| | Supply from RHRSW system | 12 | TC, MOV (1) |
| | RHR drain to radwaste | 4 | MOV, MOV (1) |
| | To fuel pool cleanup | 8 | LCV |
| | From fuel pool cleanup | 8 | LCV |
| | From chemical clean | 4, 4 | MV (NC) |
| | To process sampling system | 1/2 | AOV, AOV (1) |
| | From FLEX supply piping Div 1 | 8 | LCV |
| | From FLEX supply piping Div 2 | 8 | LCV |
| Reactor core isolation cooling | Turbine exhaust drain to barometric condenser | 3/4 | MV, MV, MV |

TABLE 6.2-14 PRIMARY CONTAINMENT PENETRATION PIPE LINES CONNECTING CLOSED-LOOP QUALITY GROUP B SYSTEMS TO QUALITY GROUP D SYSTEMS

| System | Line | Line Diameter (in.) | Separation Valve |
|---|---|---|---|
| | Discharge to lube oil cooler | 2 | MOV |
| | Condensate to radwaste | 1 | CV, AOV, AOV (1) |
| | Suction from condensate storage | 6 | CV, MOV (1) |
| | Steam drain to main condenser | 1 | SOV, SOV (1) |
| Combustible gas control | None | | |

Symbols:

SOV = solenoid-operated valve

MOV = motor-operated valve

CV = check valve

AOV = air-operated valve

LCV = locked-closed valve

TC = testable check valve

MV = manual valve

(1) = on isolation panel

(NC) = normally closed

# TABLE 6.2-15 ESSENTIAL/NONESSENTIAL LINES

| Containment Penetration Number | System/Line | Valve Number | Classification | Containment Isolation Signals[a] | Comments |
|---|---|---|---|---|---|
| X-7A | Main steam line A<br>MSIV leakage control | B2103F022A<br>B2103F028A | Nonessential<br>Nonessential | | These lines provide a heat- sink path for the reactor pressure vessel. It is desirable to keep the MSIVs open for this function during postulated small leaks or breaks. Therefore, high drywell pressure has been deliberately omitted from isolation of main steam lines. The MSIVs and the main steam line drains also isolate on signals D, E, F, G, J, P, and RM. |
| X-7B | Main steam line B<br>MSIV leakage control | B2103F022B<br>B2103F028B | Nonessential<br>Nonessential | | |
| X-7C | Main steam line C<br>MSIV leakage control | B2103F022C<br>B2103F028C | Nonessential<br>Nonessential | | |
| X-7D | Main steam line D<br>MSIV leakage control | B2103F022D<br>B2103F028D | Nonessential<br>Nonessential | | |
| X-8 | Main steam line drains | B2103F016<br>B2103F019 | Nonessential<br>Nonessential | | |
| X-9A | Feedwater line A | B2100F010A<br>B2100F032A | Essential<br>Essential | | The portion of the feed-water line that is Class 1 is essential. During the postulated LOCA, it is desirable to maintain reactor coolant makeup from all sources of supply. |
| | | B2100F076A | Essential | | This valve is provided for long-term leaktightness only. Remote manual control is provided in the control room to close the valve upon indication of loss of feedwater flow. |
| X-9A | High-pressure coolant Injection | E4150F006 | Essential | | Automatically opens and closes with HPCI pump operation. |
| X-9B | Feedwater line B | B2100F010B<br>B2100F032B | Essential<br>Essential | | The portion of the feedwater line that is Class 1 is essential. During the postulated LOCA, it is desirable to maintain reactor coolant makeup from all sources of supply. |
| | | B2100F076B | Essential | | This valve is provided for long-term leaktightness only. Remote manual control is provided in the control room to close the valve upon indication of loss of feedwater flow. |
| X-9B | Reactor core isolation Cooling | E5150F013 | Essential (safety system) | | Automatically opens and closes with RCIC pump operation. |
| X-9B | Reactor water cleanup | G3352F220 | Nonessential | | Inadvertent isolation of this line due to inclusion of the high drywell pressure signal is undesirable, as it results in reactor coolant chemistry problems, fuel leaks, and RPV bottom thermal problems.<br>RWCU is desirable for post-accident sampling of reactor coolant.<br>The system includes break detection mechanisms that will automatically isolate on unbalanced flow or high temperature. Therefore, isolation on high drywell pressure is not needed. |
| X-10 | Steam to RCIC turbine | E5150F007<br>E5150F008 | Essential<br>Essential | | |

# TABLE 6.2-15 ESSENTIAL/NONESSENTIAL LINES

| Containment Penetration Number | System/Line | Valve Number | Classification | Containment Isolation Signals[a] | Comments |
|---|---|---|---|---|---|
| X-11 | Steam to HPCI turbine | E4150F002<br>E4150F003<br>E4150F600 | Essential<br>Essential<br>Essential | | |
| X-12 | RHR/RHR pump suction from recirculation piping | E1150F009<br>E1150F608<br>E1150F008 | Nonessential<br>Nonessential<br>Nonessential | | High drywell pressure has been deliberately omitted from this line's isolation initiation to avoid the loss of the shutdown cooling mode of RHR for small breaks or leaks. |
| X-13A | RHR/RHR pump discharge to recirculation loop | E1100F050B<br>E1150F015B | Essential<br>Essential | | Not a containment isolation valve. |
| X-13B | RHR/RHR pump discharge to recirculation loop | E1100F050A<br>E1150F015A | Essential<br>Essential | | Not a containment isolation valve. |
| X-15 | Combustible gas control system suction | T4804F603A<br>T4804F605A | Nonessential<br>Nonessential | | The CGCS PCIVs are permanently de-energized and locked closed. |
| X-16A | Core spray pump discharge | E2100F006B<br>E2150F005B | Essential<br>Essential | | |
| X-16B | Core spray pump discharge | E2100F006A<br>E2150F005A | Essential<br>Essential | | |
| X-17 | RHR/RHR head spray (piping within the drywell is blanked off) | E1150F023<br>E1150F022 | Nonessential<br>Nonessential | | High drywell pressure was deliberately omitted from this line's isolation initiation to avoid the loss of the head spray mode of RHR for small breaks or leaks. |
| X-18 | Radwaste system/drywell floor drains sump pump discharge | G1100F003<br>G1154F600 | Nonessential<br>Nonessential | | |
| X-19 | Radwaste system/drywell equipment drains sump pump discharge | G1100F019<br>G1154F018 | Nonessential<br>Nonessential | | |
| X-20 | Demineralized service water to drywell | P1100F126 | Nonessential | | |
| X-22 | Station and control air/ nitrogen inerting system/ drywell equipment pneumatic supply Division I | T4901F465<br>T4901F601<br>T4901F007 | Essential<br>Essential<br>Essential | | Manual override is available to operator. |
| X-23 | Reactor building closed cooling water and emergency equipment cooling water systems supply | P4400F606A<br>P4400F282A | Essential<br>Essential | | Closes on high drywell pressure |
| X-24 | Reactor building closed cooling water and emergency cooling water systems return | P4400F616<br>P4400F607A | Essential<br>Essential | | |
| X-25 | Reactor building HVAC/ drywell exhaust and air purge | T4600F402<br>T4803F602<br>T4600F411 | Nonessential<br>Nonessential<br>Nonessential | | |
| X-26 | Nitrogen inerting system and reactor building HVAC/ drywell air purge inlet | T4800F408<br>T4803F601<br>T4800F407 | Nonessential<br>Nonessential<br>Nonessential | | |
| X-27a | PCMS containment atmosphere sample | T5000F401B | Essential | | |

# TABLE 6.2-15 ESSENTIAL/NONESSENTIAL LINES

| Containment Penetration Number | System/Line | Valve Number | Classification | Containment Isolation Signals[a] | Comments |
|---|---|---|---|---|---|
| X-27b | PCMS containment atmosphere sample | T5000F402B | Essential | | |
| X-27c | PCMS containment atmosphere sample | T5000F403B | Essential | | |
| X-27d | PCMS containment atmosphere sample | T5000F404B | Essential | | |
| X-27e | PCMS containment atmosphere sample | T5000F405B | Essential | | |
| X-27f | Drywell pressure instrumentation | T50-F458 | Essential | | |
| X-27b | PASS[b]/containment drywell atmosphere sample | P34F403A<br>P34F404A | Nonessential<br>Nonessential | | Administrative control utilized. |
| X-28Cf | PASS/pressurized reactor coolant sample | P34F401A | Nonessential | | Administrative control utilized. Orifice in line inside containment. |
| X-29Aa | Process sample/reactor recirculation water sample | B3100F019<br>B3100F020 | Nonessential<br>Nonessential | | |
| X-29Bc | PCMS/drywell instrumentation | E11F413 | Essential | | |
| X-29Bb | PCMS/drywell instrumentation | E11F412 | Essential | | |
| X-29Be | PCMS/drywell instrumentation | T5000F420B | Essential | | |
| X-30Aa | RWCU/RPV pressure | G33F583 | Essential | | |
| X-31Ba | Nitrogen inerting system/ drywell nitrogen makeup and vent | T4800F453<br>T4800F454<br>T4800F455 | Nonessential<br>Nonessential<br>Nonessential | | |
| X-32Ba | Steam flow to HPCI (instrumentation) | E4100F503 | Essential | | |
| X-32Bb | Steam flow to HPCI (instrumentation) | E4100F502 | Essential | | |
| X-33Bc | Core spray/RPV pressure (instrumentation) | E21F500A | Essential | | |
| X-33Ba | Steam flow to HPCI (instrumentation) | E4100F501 | Essential | | |
| X-33Bb | Steam flow to HPCI (instrumentation) | E4100F500 | Essential | | |
| X-34A | RBCCW and emergency cooling water systems supply | P4400F606B<br>P4400F282B | Essential<br>Essential | | Closes on high drywell pressure |
| X-34B | RBCCW and emergency cooling water systems return | P4400F607B<br>P4400F615 | Essential | | |
| X-35B | NMS/TIP system | | Nonessential | | System normally isolated and closed inside containment. |
| X-35C | NMS/TIP system | | Nonessential | | |

# TABLE 6.2-15 ESSENTIAL/NONESSENTIAL LINES

| Containment Penetration Number | System/Line | Valve Number | Classification | Containment Isolation Signals[a] | Comments |
|---|---|---|---|---|---|
| X-35D | NMS/TIP system | | Nonessential | | |
| X-35E | NMS/TIP system | | Nonessential | | |
| X-35F | NMS/TIP system | | Nonessential | | |
| X-35G | TIP system spare | | Nonessential | | |
| X-36 | Station and control air/ nitrogen inerting system/ drywell equipment pneumatic supply, Division II | T4901F468<br>T4901F602<br>T4901F016 | Essential<br>Essential<br>Essential | | Manual override is available to operator. |
| X-37A,B,C,D | CRD/control rod drive insertion line | None | Essential | | |
| X-38A,B,C,D | CRD/control rod drive withdrawal line | None | Essential | | |
| X-39A | RHR/RHR to containment spray header | E1150F021A<br>E1150F016A | Essential<br>Essential | | |
| X-39B | RHR/RHR to containment spray header | E1150F021B<br>E1150F016B | Essential<br>Essential | | |
| X-40Dd | PASS/pressurized reactor coolant sample | P34F401B | Nonessential | | Administrative control utilized. Orifice in line inside containment. |
| X-42 | SLCS/standby liquid control | C4100F007<br>C4100F006 | Essential<br>Essential | | |
| X-43 | RWCU/reactor water (cleanup from recirculation piping) | G3352F001<br>G3352F004 | Nonessential<br>Nonessential | | Inadvertent isolation of this line due to inclusion of the high-drywell-pressure signal is undesirable, as it results in reactor coolant chemistry problems, fuel leaks, and RPV bottom thermal problems.<br><br>RWCU is desirable for postaccident sampling of reactor coolant.<br><br>The system includes break-detection mechanisms that will automatically isolate on unbalanced flow or high temperature. Therefore, isolation on high drywell pressure is not needed. |
| X-44 | CGCS/combustible gas control system suction | T4804F603B<br>T4804F605B | Nonessential<br>Nonessential | | The CGCS PCIVs are permanently de-energized and locked closed. |
| X-47a | PCMS/drywell instrumentation | E11F414 | Essential | | |
| X-47b | PCMS/drywell instrumentation | E11F415 | Essential | | |
| X-47e | PCMS/drywell pressure | T5000F420A | Essential | | |
| X-48b | PCMS/containment atmosphere sample | T5000F402A | Essential | | See also containment penetration "PCRMS." |
| X-48c | PCMS/containment atmosphere sample | T5000F403A | Essential | | See also containment penetration "PCRMS." |
| X-48d | PCMS/containment atmosphere sample | T5000F404A | Essential | | See also containment penetration "PCRMS." |
| X-48e | PCMS/containment atmosphere sample | T5000F405A | Essential | | See also containment penetration "PCRMS." |

# TABLE 6.2-15 ESSENTIAL/NONESSENTIAL LINES

| Containment Penetration Number | System/Line | Valve Number | Classification | Containment Isolation Signals[a] | Comments |
|---|---|---|---|---|---|
| X-48f | PASS/containment drywell atmosphere sample | P34F403B<br>P34F404B | Nonessential | | Administrative control utilized. |
| X-49a | Reactor recirculation/ recirculation pump seal purge | B3100F016A<br>B3100F014A | Nonessential<br>Nonessential | | High-pressure line with globe valves inside and outside containment, and an orifice in the line to prevent backflow. |
| X-51a | Reactor recirculation/ recirculation pump seal purge | B3100F016B<br>B3100F014B | Nonessential<br>Nonessential | | High-pressure line with globe valves inside and outside containment, and an orifice in the line to prevent backflow. |
| X-52e | Steam flow to RCIC (instrumentation) | E51F506 | Essential | | |
| X-52f | Steam flow to RCIC (instrumentation) | E51F505 | Essential | | |
| X-53a | Steam flow to RCIC (instrumentation) | E51F503 | Essential | | |
| X-53b | Steam flow to RCIC (instrumentation) | E51F504 | Essential | | |
| X-53c | Core spray/RPV pressure (instrumentation) | E21F500B | Essential | | |
| X-204A | Nitrogen inerting system/drywell to torus vacuum breaker nitrogen supply, vacuum breaker valve T2300F400A | T4800F416 | Nonessential | | Electrically de-energized |
| X-204B | Nitrogen inerting system/drywell to torus vacuum breaker nitrogen supply, vacuum breaker valve T2300F400B | T4800F417 | Nonessential | | Electrically de-energized |
| X-204C | Nitrogen inerting system/drywell to torus vacuum breaker nitrogen supply, vacuum breaker valve T2300F400C | T4800F418 | Nonessential | | Electrically de-energized |
| X-204D | Nitrogen inerting system/drywell to torus vacuum breaker nitrogen supply, vacuum breaker valve T2300F400D | T4800F419 | Nonessential | | Electrically de-energized |
| X-204E | Nitrogen inerting system/drywell to torus vacuum breaker nitrogen supply, vacuum breaker valve T2300F400E | T4800F420 | Nonessential | | Electrically de-energized |
| X-204F | Nitrogen inerting system/ drywell to torus vacuum breaker nitrogen supply, vacuum breaker valve T2300F400F | T4800F421 | Nonessential | | Electrically de-energized |
| X-204G | Nitrogen inerting system/ drywell to torus vacuum breaker nitrogen supply, vacuum breaker valve T2300F400G | T4800F422 | Nonessential | | Electrically de-eneregized |

# TABLE 6.2-15 ESSENTIAL/NONESSENTIAL LINES

| Containment Penetration Number | System/Line | Valve Number | Classification | Containment Isolation Signals[a] | Comments |
|---|---|---|---|---|---|
| X-204H | Nitrogen inerting system/ drywell to torus vacuum breaker nitrogen supply, vacuum breaker valve T2300F400H | T4800F423 | Nonessential | | Electrically de-energized |
| X-204J | Nitrogen inerting system/ drywell to torus vacuum breaker nitrogen supply, vacuum breaker valve T2300F400J | T4800F424 | Nonessential | | Electrically de-energized |
| X-204K | Nitrogen inerting system/ drywell to torus vacuum breaker nitrogen supply, vacuum breaker valve T2300F400K | T4800F425 | Nonessential | | Electrically de-energized |
| X-204L | Nitrogen inerting system/ drywell to torus vacuum breaker nitrogen supply, vacuum breaker valve T2300F400L | T4800F426 | Nonessential | | Electrically de-energized |
| X-204M | Nitrogen inerting system/ drywell to torus vacuum breaker nitrogen supply, vacuum breaker valve T2300F400M | T4800F427 | Nonessential | | Electrically de-energized |
| X-205A | Primary containment system/ to secondary containment to torus vacuum breaker | T2300F450B<br>T2300F410 | Essential<br>Essential | | Provisions for administrative control ensure that the valve is not inadvertently positioned open by the operator. This does not prevent automatic operation to control primary containment vacuum formation. |
| X-205B | Primary containment system/secondary containment to torus vacuum breaker | T2300F450A<br>T2300F409 | Essential<br>Essential | | Provisions for administrative control ensure that the valve is not inadvertently positioned open by the operator. This does not prevent automatic operation to control primary containment vacuum formation. |
| X-205C | Nitrogen inerting system and reactor building HVAC/suppression pool air purge inlet | T4800F409<br>T4800F404<br>T4800F405 | Nonessential<br>Nonessential<br>Nonessential | | |
| X-205D | Nitrogen inerting system and reactor building HVAC/suppression pool exhaust air purge to standby gas treatment | T4600F400<br>T4600F401<br>T4600F412 | Nonessential<br>Nonessential<br>Nonessential | | |
| | Torus nitrogen inerting inlet | T4800F410 | Nonessential | | |
| | Torus nitrogen makeup and Vent | T4800F456<br>T4800F457<br>T4800F458 | Nonessential<br>Nonessential<br>Nonessential | | |
| X-206A | PCMS/liquid level indicators | E41F402 | Essential | | Accident monitoring instrumentation. |
| X-206B | PCMS/liquid level indicators | E41F403 | Essential | | |
| X-206C | PCMS/liquid level indicators | E41F401 | Essential | | |
| X-206D | PCMS/liquid level indicators | E41F400 | Essential | | Valves fail as is. |
| X-206E | PCMS/liquid level indicators | T50F412A | Essential | | |

# TABLE 6.2-15 ESSENTIAL/NONESSENTIAL LINES

| Containment Penetration Number | System/Line | Valve Number | Classification | Containment Isolation Signals[a] | Comments |
|---|---|---|---|---|---|
| X-206E | PCMS/suppression pool liquid level indicators | | Essential | | |
| X-206F | PCMS/liquid level indicators | T50F412B | Essential | | |
| X-206F | PCMS/suppression pool liquid level indicators | | Essential | | |
| X-210A | RHR/RHR minimum flow | E1150F007B | Essential | | |
| | RHR heat exchanger discharge header thermal relief | E1100F025B | Essential | | |
| | RHR/RHR test line | E1150F024B | Essential | | Manual override available to operator. |
| | RHR/RHR heat exchanger thermal relief | E1100F001B | Essential | | |
| | RHR warmup line | E1150F026B | Nonessential | | |
| X-210B | PASS/containment liquid sample return | P34F407<br>P34F409 | Nonessential<br>Nonessential | | Administrative control utilized. |
| X-210B | TWMS | G5100F604<br>G5100F605 | Nonessential<br>Nonessential | | |
| X-210B | RHR/suction thermal relief | E1100F029 | Nonessential | | |
| | RHR/heat exchanger discharge header thermal relief | E1100F025A | Essential | | |
| | RHR/heat exchanger relief | E1100F001A | Essential | | |
| | RHR/minimum flow | E1150F007A | Essential | | |
| | RHR/test line | E1150F024A | Essential | | |
| X-211A | RHR/RHR suppression pool Spray | E1150F027B<br>E1150F028B | Essential<br>Essential | | |
| X-211B | RHR/RHR suppression pool Spray | E1150F028A<br>E1150F027A | Essential<br>Essential | | |
| X-212 | RCIC turbine exhaust line | E5150F001 | Essential | | |
| X-213A | TWMS suction | G5100F600<br>G5100F601 | Nonessential<br>Nonessential | | |
| X-213B | TWMS suction | G5100F602<br>G5100F603 | Nonessential<br>Nonessential | | |
| X-214 | HPCI vacuum breaker line | E4150F075<br>E4150F079 | Essential<br>Essential | | |
| X-214 | RCIC vacuum breaker line | E5150F062<br>E5150F084 | Essential<br>Essential | | |
| X-215 | PCMS return Division I | T5000F408A | Essential | | See also containment penetration "PCRMS." |
| X-215 | CGCS/combustible gas control system suction | T4804F602A<br>T4804F606A | Nonessential<br>Nonessential | | The CGCS PCIVs are permanently de-energized and locked closed. |
| X-215 | PASS/containment gaseous sample return | P34F408<br>P34F410 | Nonessential<br>Nonessential | | Administrative control utilized. |

# TABLE 6.2-15 ESSENTIAL/NONESSENTIAL LINES

| Containment Penetration Number | System/Line | Valve Number | Classification | Containment Isolation Signals[a] | Comments |
|---|---|---|---|---|---|
| X-218 | CGCS/combustible gas control system return | T4804F601A<br>T4804F604A<br>T4804F016A<br>T4804F601B<br>T4804F604B<br>T4804F016B | Nonessential<br>Nonessential<br>Nonessential<br>Nonessential<br>Nonessential<br>Nonessential | | The CGCS PCIVs are permanently de-energized and locked closed. |
| X-219 | CGCS/combustible gas control system suction | T4804F602B<br>T4804F606B | Nonessential<br>Nonessential | | The CGCS PCIVs are permanently de-energized and locked closed. |
| X-219 | PCMS return Division II | T5000F408B | Essential | | |
| X-220 | HPCI turbine exhaust line | E4150F021 | Essential | | |
| X-221 | HPCI turbine exhaust drain | E4150F022 | Essential | | |
| X-222 | RCIC vacuum pump discharge | E5150F002 | Essential | | |
| X-223A | RHR/RHR pump suction | E1150F004D | Essential | | |
| X-223A | RHR/RHR pump suction header thermal relief | E1100F030D | Essential | | |
| X-223B | RHR/RHR pump suction | E1150F004B | Essential | | |
| X-223B | RHR/RHR pump suction header thermal relief | E1100F030B | Essential | | |
| X-223C | RHR/RHR pump suction | E1150F004C | Essential | | |
| X-223C | RHR/RHR pump suction header thermal relief | E1100F030C | Essential | | |
| X-223D | RHR/RHR pump suction | E1150F004A | Essential | | |
| X-223D | RHR/RHR pump suction header thermal relief | E1100F030A | Essential | | |
| X-224A | Core spray pump suction | E2150F036B | Essential | | |
| X-224B | Core spray pump suction | E2150F036A | Essential | | |
| X-225 | HPCI pump suction | E4150F042 | Essential | | |
| X-226 | RCIC pump suction | E5150F031 | Essential | | |
| X-227A | TWMS discharge | G5100F606<br>G5100F607 | Nonessential<br>Nonessential | | |
| X-227A | HPCI minimum flow | E4150F012 | Essential | | |
| X-227A | Core spray pump suction thermal relief | E2100F032B | Essential | | |
| X-227A | Core spray pump discharge header relief | E2100F012B<br>E2100F011B | Essential | | |
| X-227A | Core spray pump minimum flow | E2150F031B | Essential | | |
| X-227A | Core spray pump test line | E2150F015B | Nonessential | | |
| X-227B | RCIC minimum flow | E5150F019 | Essential | | |
| X-227B | Core spray pump suction thermal relief | E2100F032A | Essential | | |
| X-227B | Core spray pump discharge header relief | E2100F012A<br>E2100F011A | Essential<br>Essential | | |

# TABLE 6.2-15 ESSENTIAL/NONESSENTIAL LINES

| Containment Penetration Number | System/Line | Valve Number | Classification | Containment Isolation Signals[a] | Comments |
|---|---|---|---|---|---|
| X-227B | Core spray pump test line | E2150F015A | Nonessential | | |
| X-227B | Core spray minimum flow | E2150F031A | Essential | | |
| X-230 | PASS/suppression pool Atmosphere sample | P34F405B<br>P34F406B | Nonessential<br>Nonessential | | Administrative control utilized |
| X-230 | PCMS suction Division I | T5000F407A | Essential | | |
| X-231 | PCMS suction Division II | T5000F407B | Essential | | |
| X-231 | PASS/suppression pool atmosphere sample | P34F405A<br>P34F406A | Nonessential<br>Nonessential | | Administrative control utilized. |
| PCRMS | Primary containment radiation monitor | T50F450<br>T5000F456<br>T50F451<br>T5000F455 | Nonessential<br>Nonessential<br>Nonessential<br>Nonessential | | Sample suction X-48.<br>Sample suction X-48.<br>Sample return X-215.<br>Sample return X-215. |

[a] Containment Isolation Signals are contained in UFSAR Table 6.2-2.

[b] PASS is postaccident sampling system.

TABLE 6.2-16 ESSENTIAL/NONESSENTIAL SYSTEMS

| System | Classification | Comments |
|---|---|---|
| Main Steam | Nonessential | Not required for shutdown. |
| Feedwater | Nonessential | Not required for shutdown. Portion that is Class 1 is essential. |
| Reactor core isolation cooling | Essential | Necessary for core cool-down following isolation from the turbine condenser and feedwater makeup. |
| Reactor water cleanup | Nonessential | Not required during and immediately following an accident. |
| High pressure coolant injection | Essential | Safety system. |
| Core spray | Essential | Safety system. |
| Standby liquid Control | Essential | Should be available as a post LOCA pH control system and backup to CRD system. |
| Drywell floor/equipment drains | Nonessential | Not necessary for core cooldown. |
| Torus water management | Nonessential | Not required for reactor shutdown cooling. |
| Primary containment monitoring system | Essential | Required for postaccident monitoring of containment atmosphere hydrogen concentration. |
| Primary containment radiation monitoring system | Nonessential | Not required during or immediately after an accident. |
| Residual heat removal | | |
| Heat exchangers | Essential | Main heat sink during isolation. |

TABLE 6.2-16 ESSENTIAL/NONESSENTIAL SYSTEMS

| System | Classification | Comments |
|---|---|---|
| Shutdown cooling | Nonessential | Nonessential, but desirable to use if available. Not redundant, but safety grade. |
| Drywell/suppression pool spray | Essential | Necessary to control pressure. |
| LPCI function | Essential | Safety function. |
| Keep-filled system | Nonessential | Not required after accident. |
| Control rod drive | Essential | Necessary for shutdown. No credit taken for reflood, but is desirable. |
| Emergency equipment cooling water | Essential | Necessary to cool safety system pumps and motors. |
| Station and control air | | |
| Pneumatic supply to primary containment | Essential | For safety/relief valves on steam lines and ADS accumulators. |
| Demineralized service water | Nonessential | Not assumed available in ECCS analysis. |
| Nitrogen inerting | Nonessential | Not required during and immediately after accident. |
| Reactor building closed cooling water | Nonessential | Used for normal operation only. |
| Reactor recirculation | Nonessential | Not required because core can be cooled by natural circulation. |
| Traversing in-core probe | Nonessential | Not required for reactor shutdown cooling. |

TABLE 6.2-16 ESSENTIAL/NONESSENTIAL SYSTEMS

| System | Classification | Comments |
|---|---|---|
| Primary containment (vacuum breakers between secondary containment and suppression pool) | Essential | Vacuum breakers automatically open to prevent formation of excessive negative pressure in the suppression pool chamber. They close automatically upon increasing suppression pool chamber pressure and remain closed during all containment high-pressure conditions. |
| Reactor building heating, ventilation and air conditioning | Nonessential | Reactor building purge and vent functions are nonessential. Essential cooling is provided by equipment outside primary containment. |

Figure Intentionally Removed
Refer to Plant Drawing M-2501

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FIGURE 6.2-1

PRIMARY CONTAINMENT SYSTEM
PROCESS LINE FLEXIBLE PENETRATION

Figure Intentionally Removed
Refer to Plant Drawing M-2502

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**FIGURE 6.2-2**

**PRIMARY CONTAINMENT SYSTEM**
**PROCESS LINE PENETRATIONS**

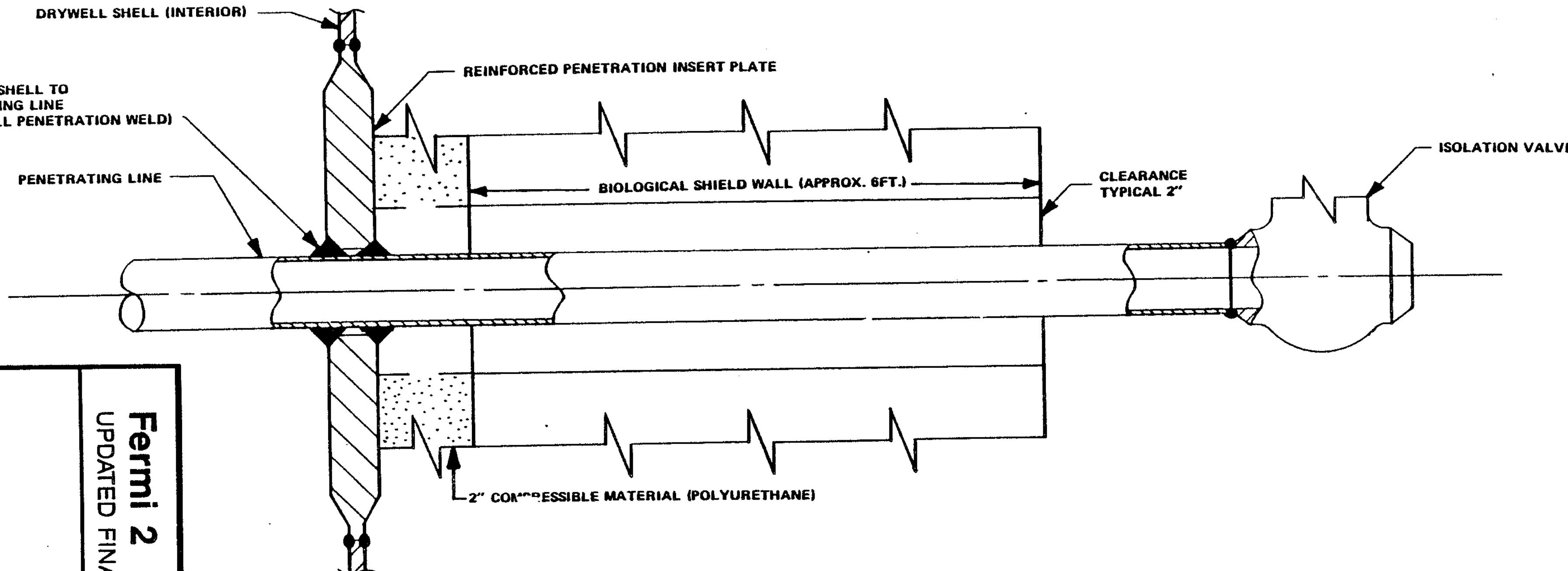


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FIGURE 6.2-3

PRIMARY CONTAINMENT SYSTEM
NON-RCPB PROCESS LINE PENETRATION

Figure Intentionally Removed
Refer to Plant Drawing E-2831-08

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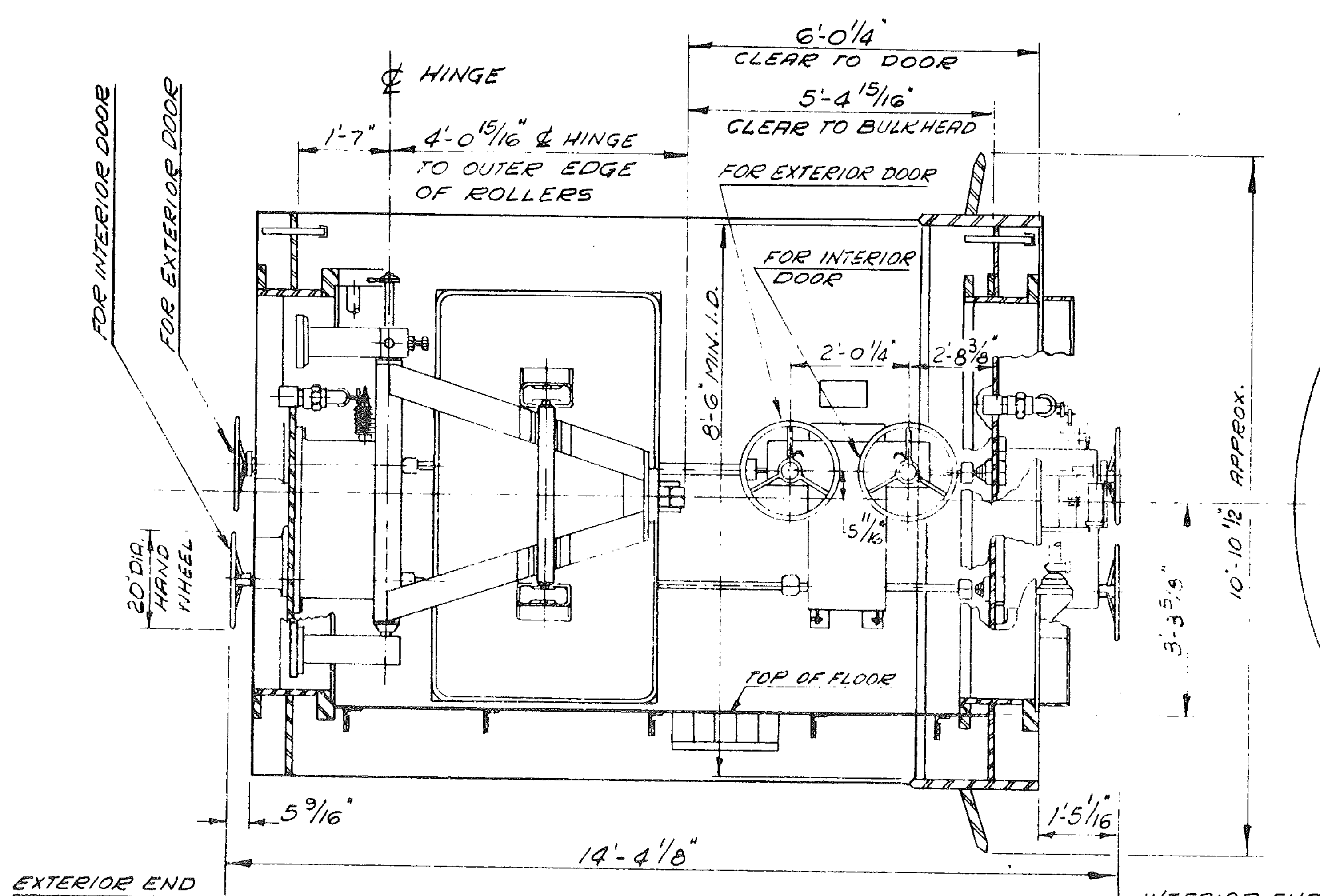
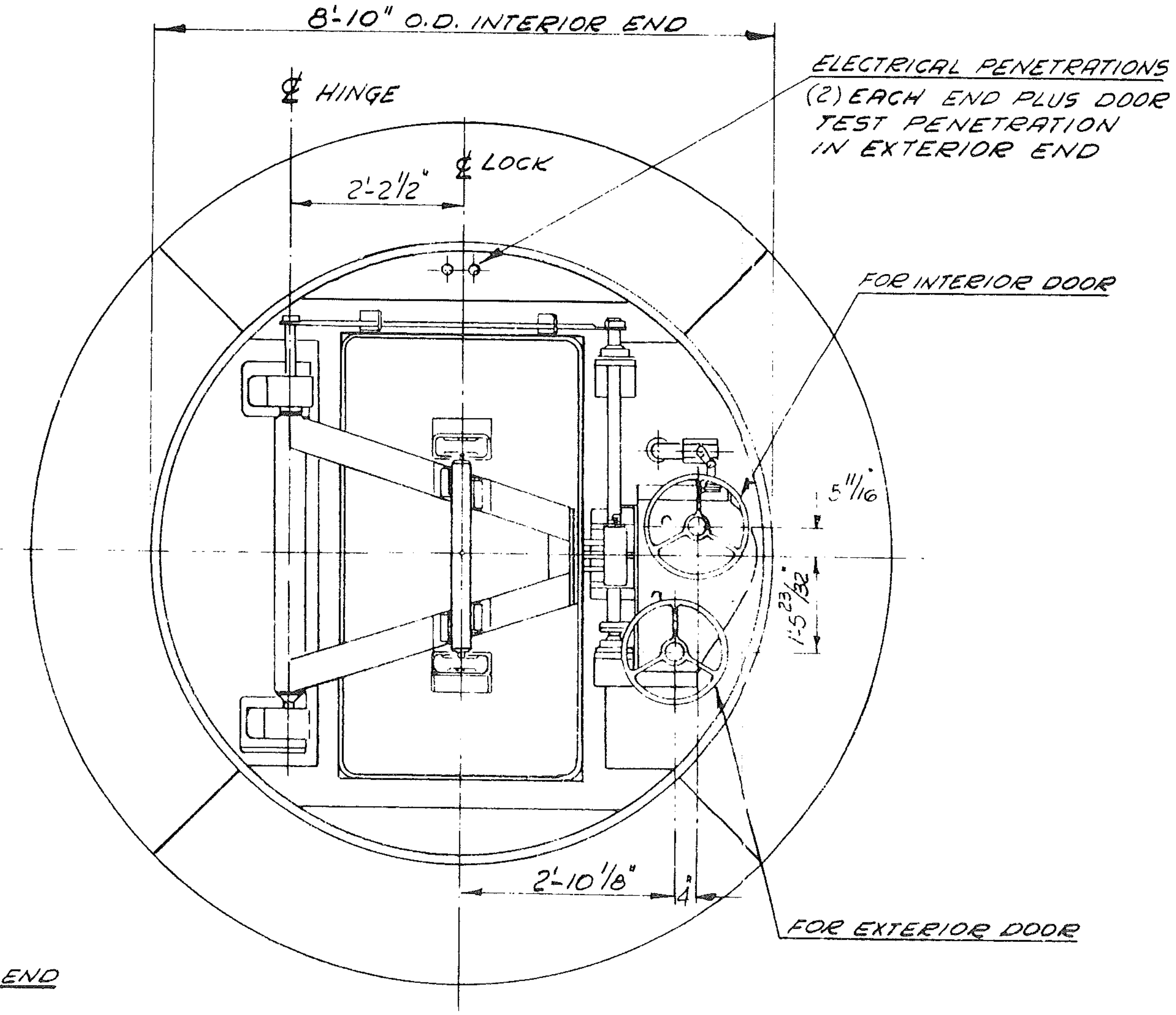
FIGURE 6.2-4

TYPICAL PRIMARY CONTAINMENT
SYSTEM ELECTRICAL PENETRATION




SECTIONAL ELEVATION OF PERSONNEL LOCK




INTERIOR END VIEW

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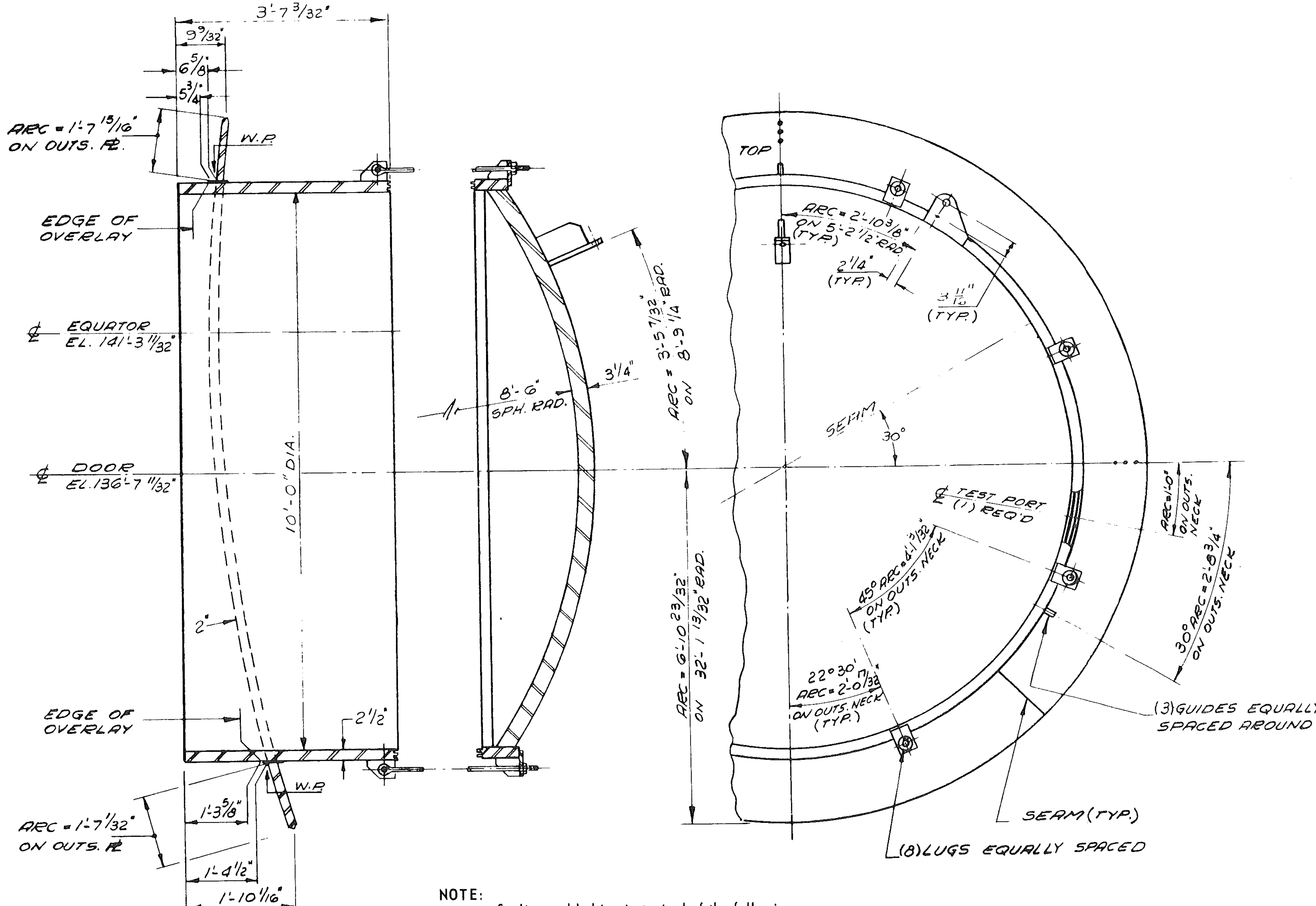
FIGURE 6.2-5

PRIMARY CONTAINMENT
PERSONNEL HATCH




NOTE:

Sealing and bolting is typical of the following

1. Drywell head.
2. Equipment access hatch in cylindrical portion of Drywell.
3. Rod removal hatch in the spherical portion of Drywell.
4. Access to the pressure suppression chamber

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FIGURE 6.2-6

PRIMARY CONTAINMENT EQUIPMENT HATCH

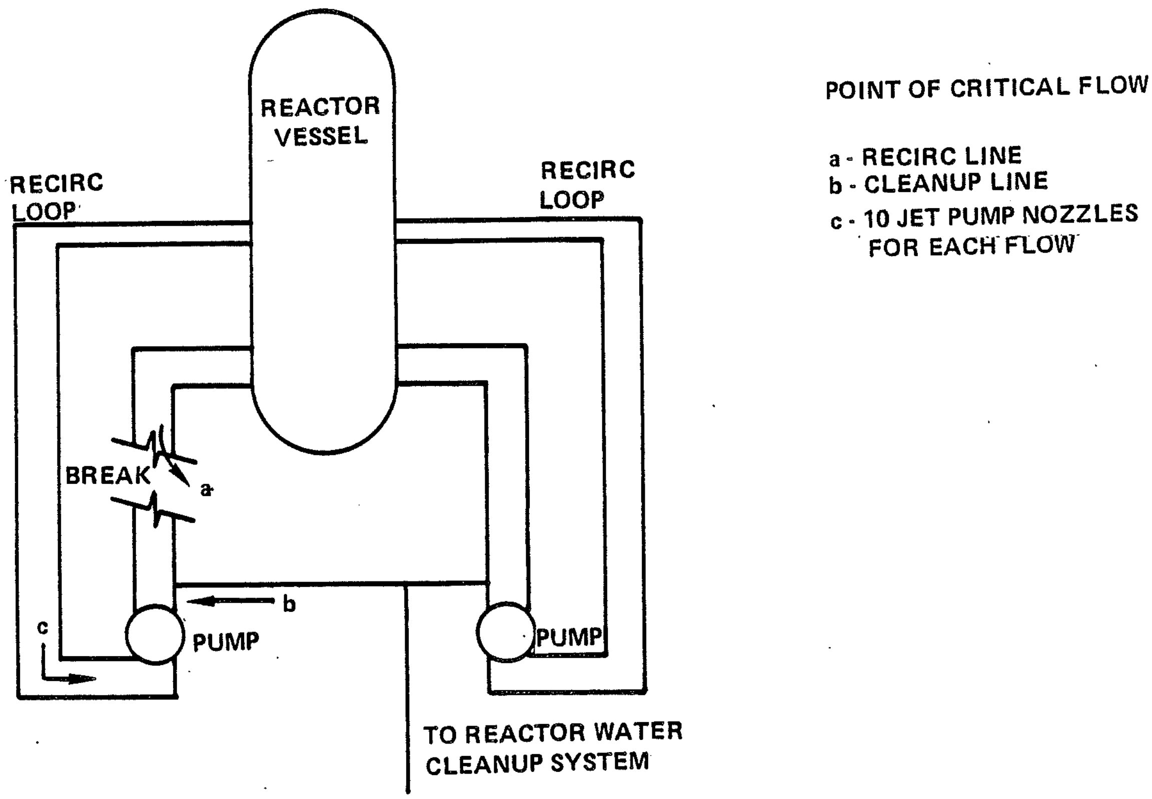


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FIGURE 6.2-7

DIAGRAM SHOWING LOCATION OF RECIRCULATION LINE BREAK

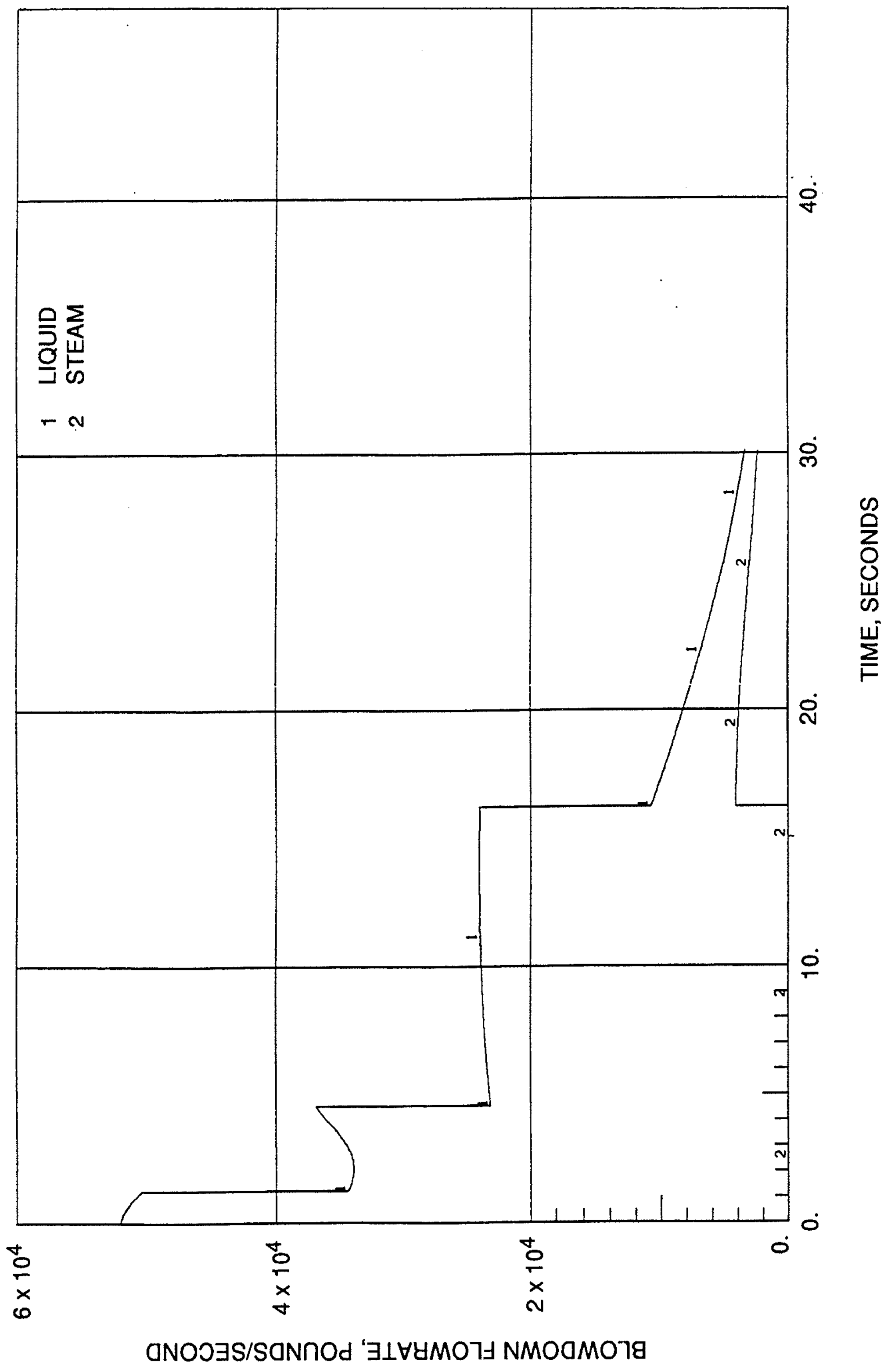


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**FIGURE 6.2-8**

**RECIRCULATION BREAK BLOWDOWN FLOWRATES (3499 MWT)**

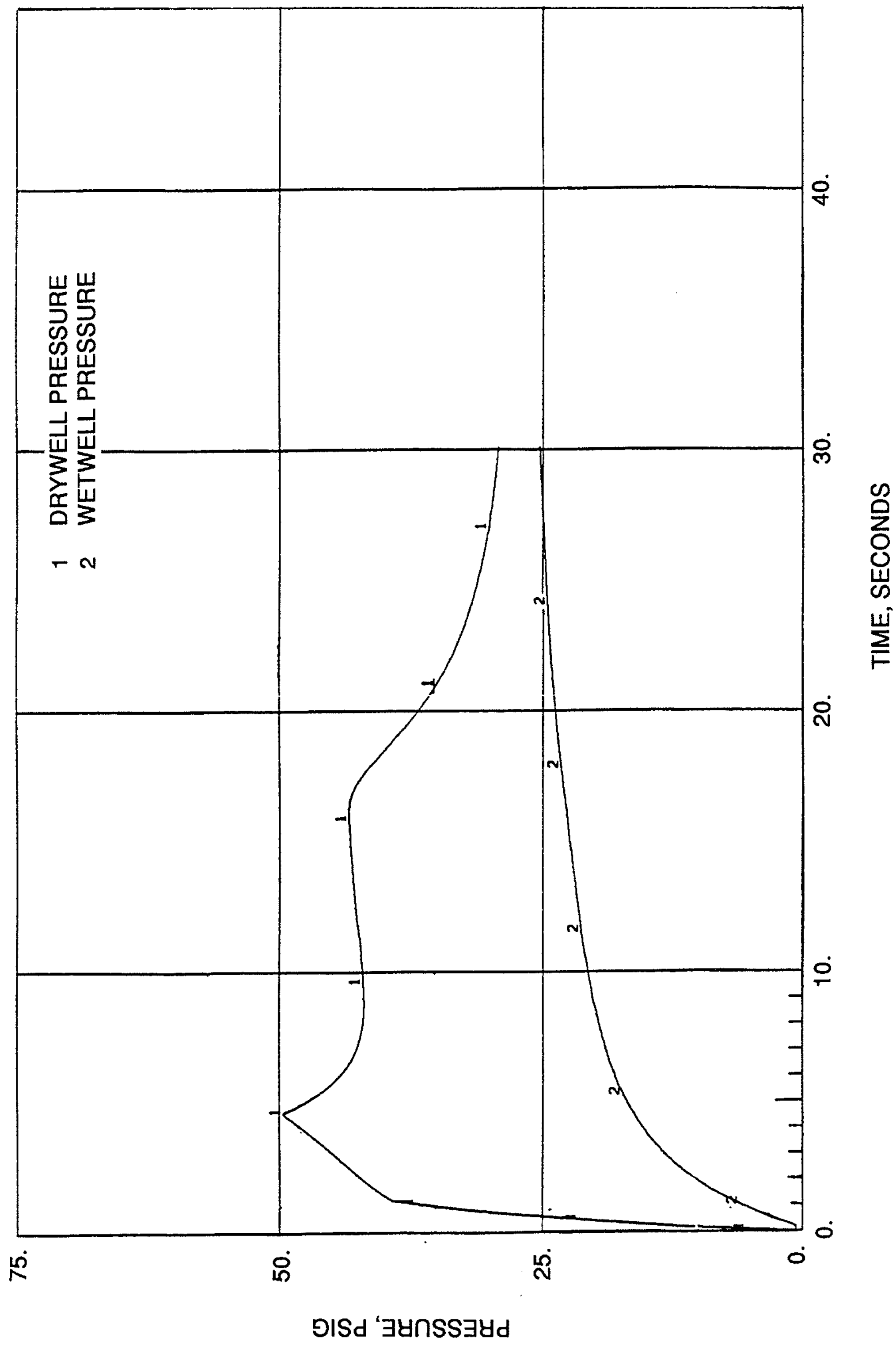


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**FIGURE 6.2-9**

**RECIRCULATION LINE BREAK PRIMARY CONTAINMENT INITIAL PRESSURE TRANSIENT (3499 MWT)**

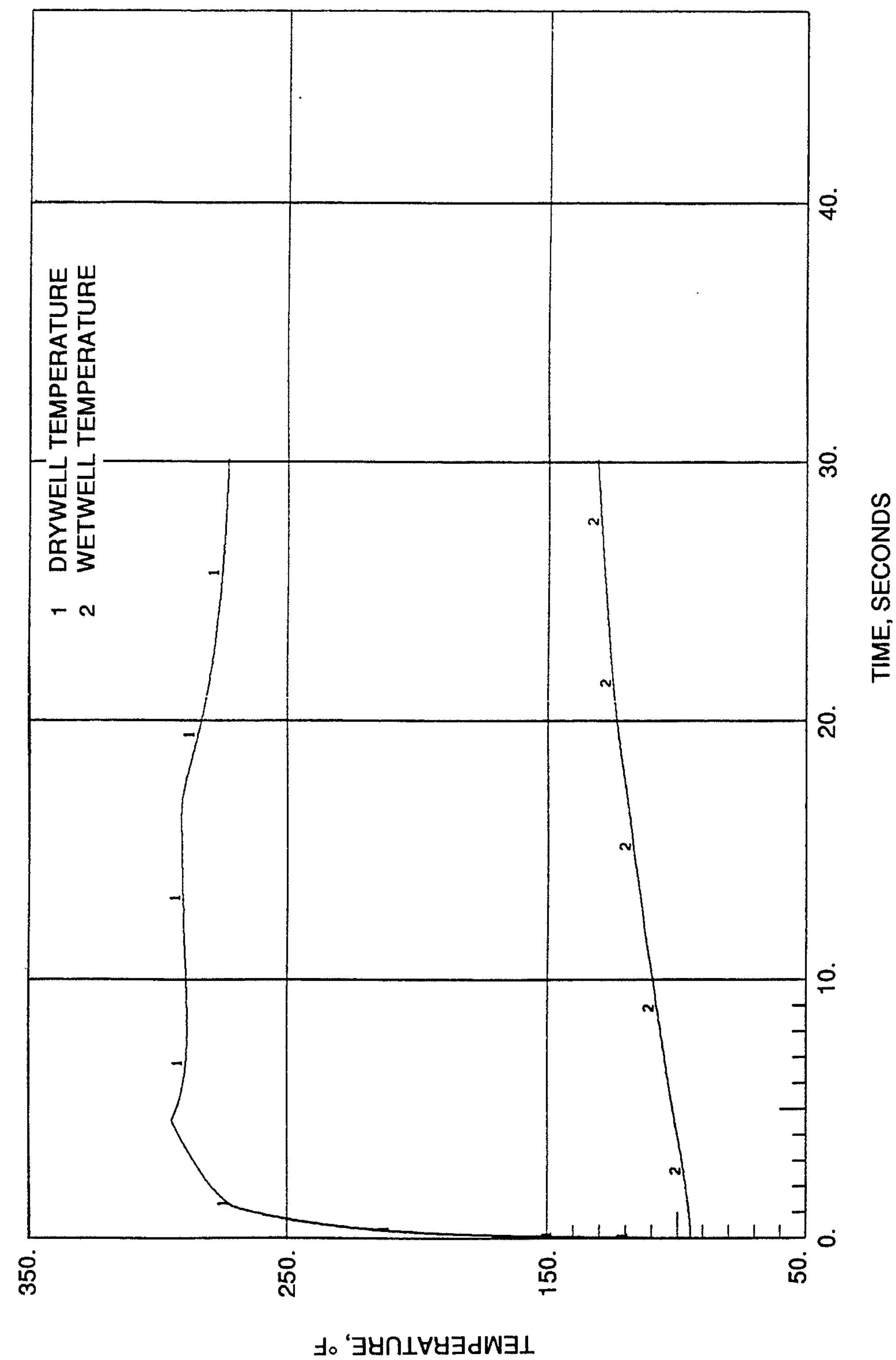


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**FIGURE 6.2-10**

**RECIRCULATION LINE BREAK PRIMARY CONTAINMENT INITIAL TEMPERATURE TRANSIENT (3499 MWT)**

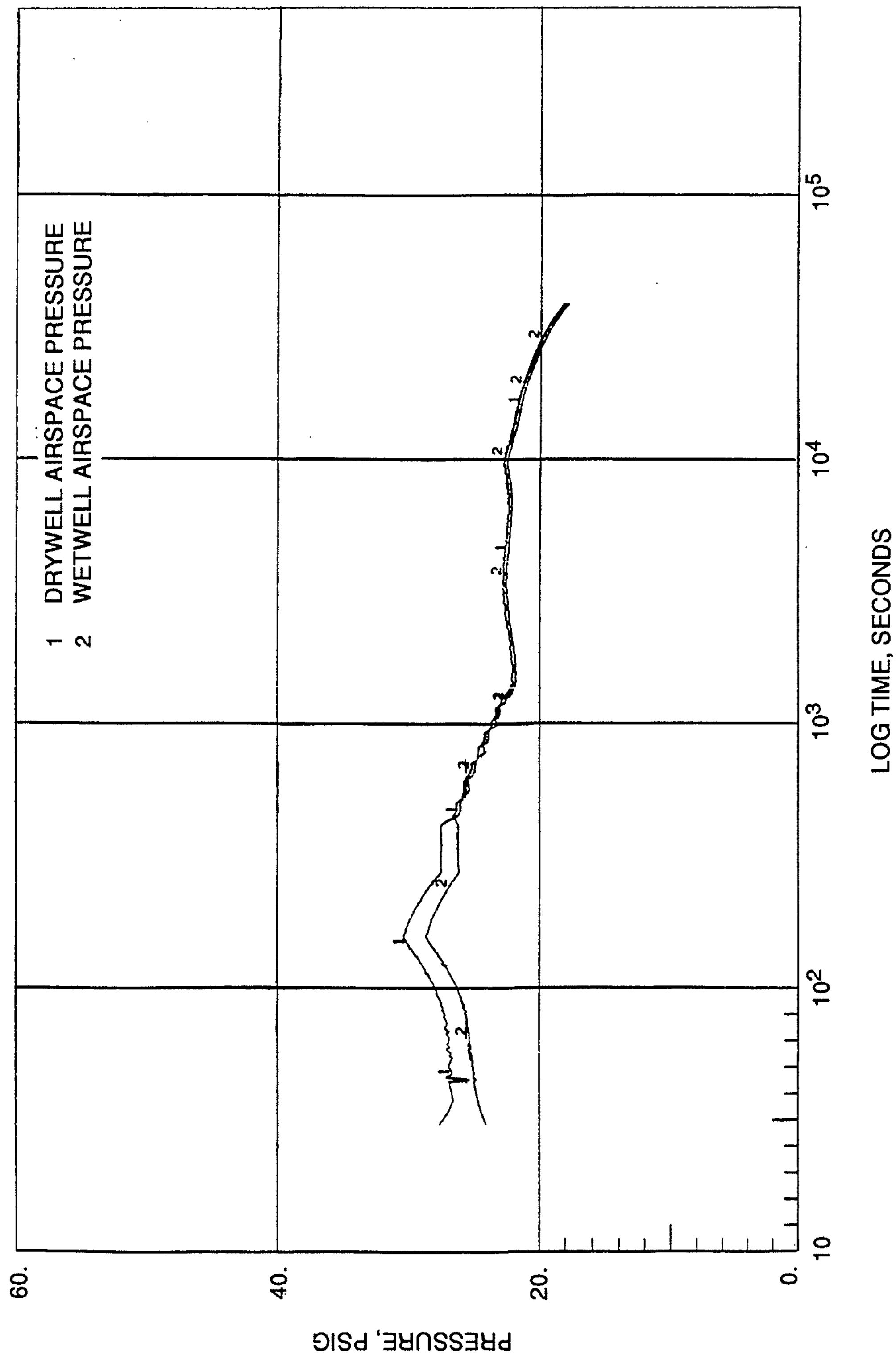


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**FIGURE 6.2-11**

**PRIMARY CONTAINMENT PRESSURE**
**LONG TERM RESPONSE**
**(3499 MWT)**

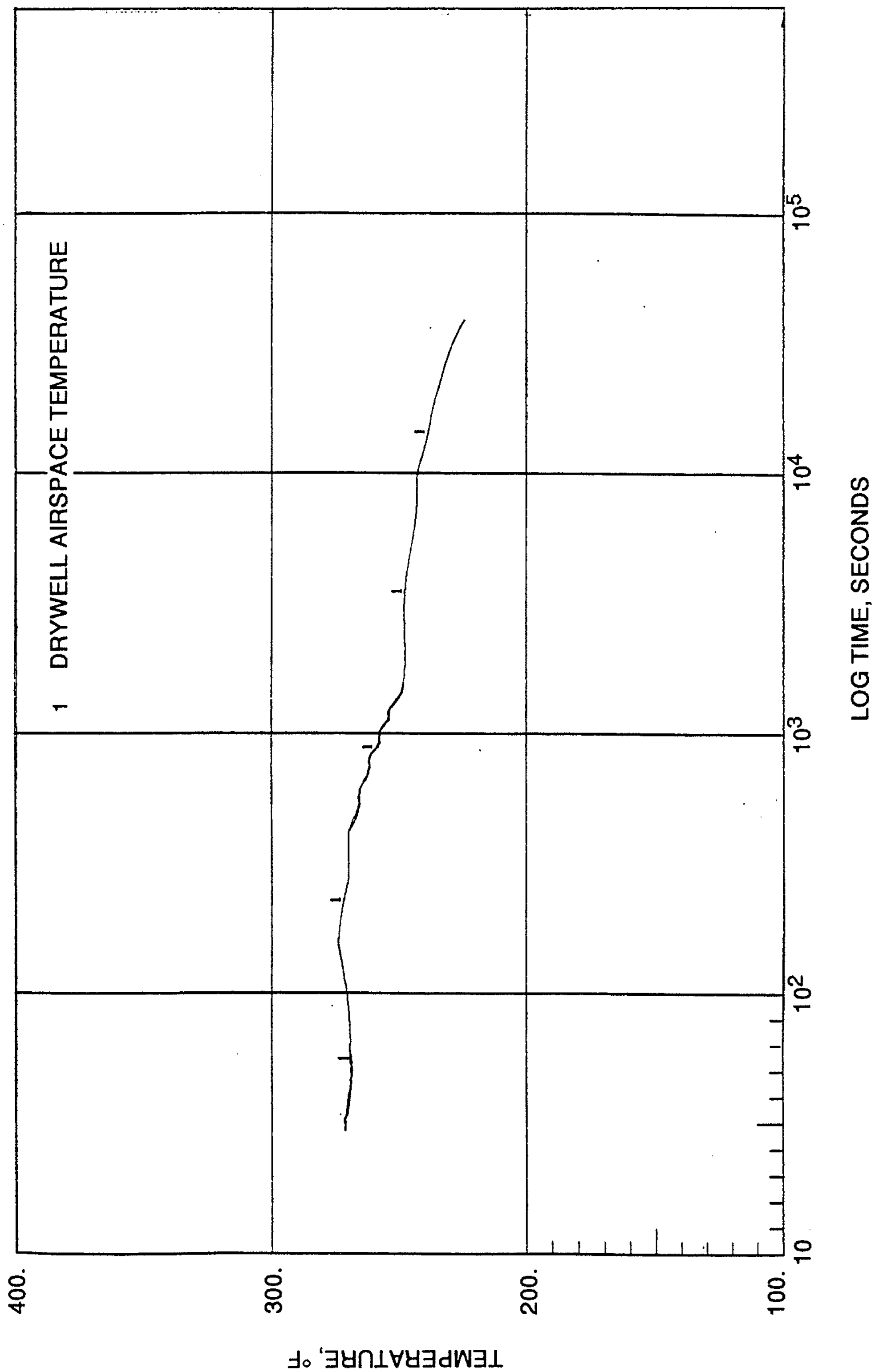


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**FIGURE 6.2-12**

**DRYWELL TEMPERATURE**
**LONG TERM RESPONSE**
**(3499 MWT)**

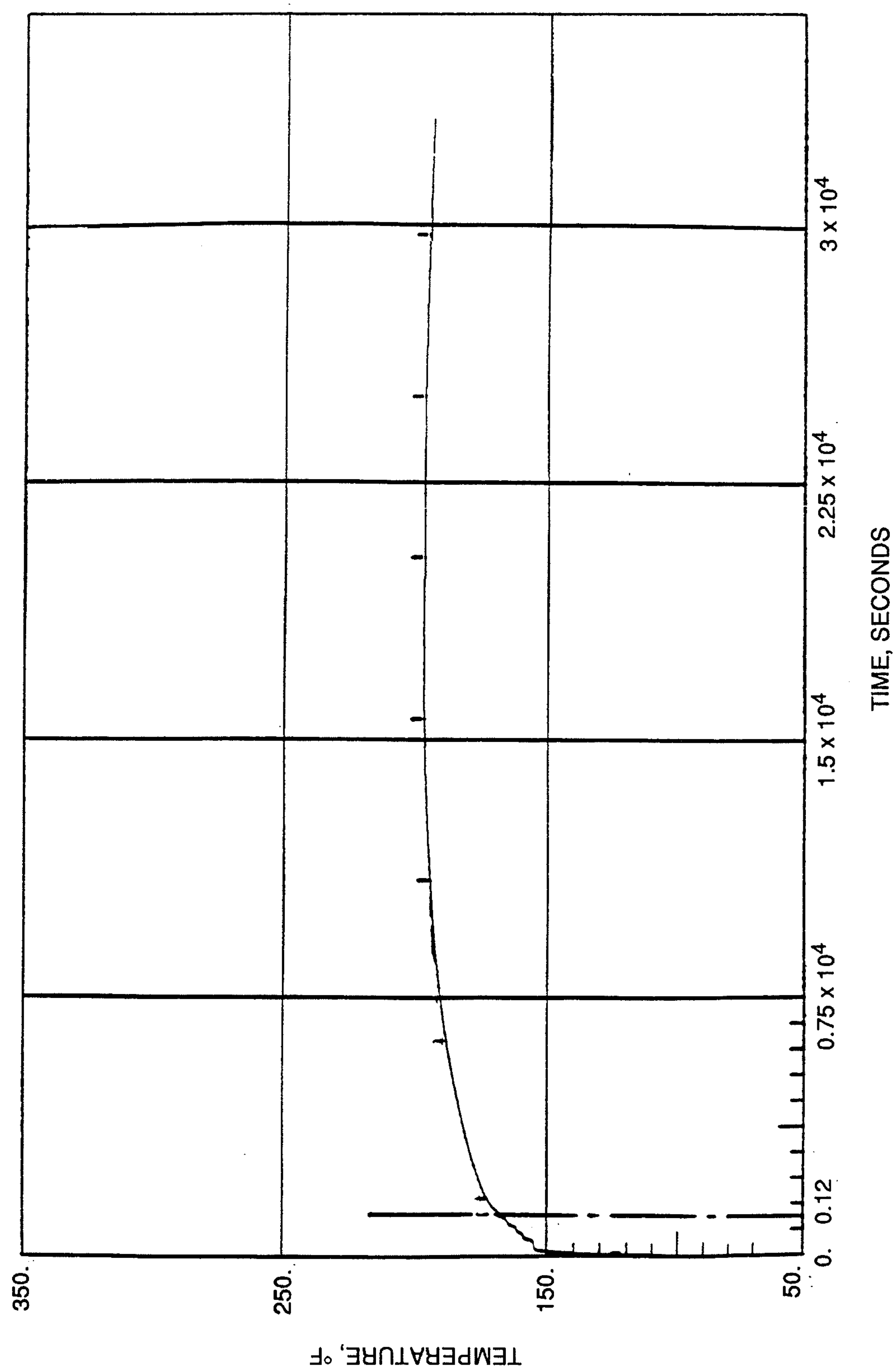


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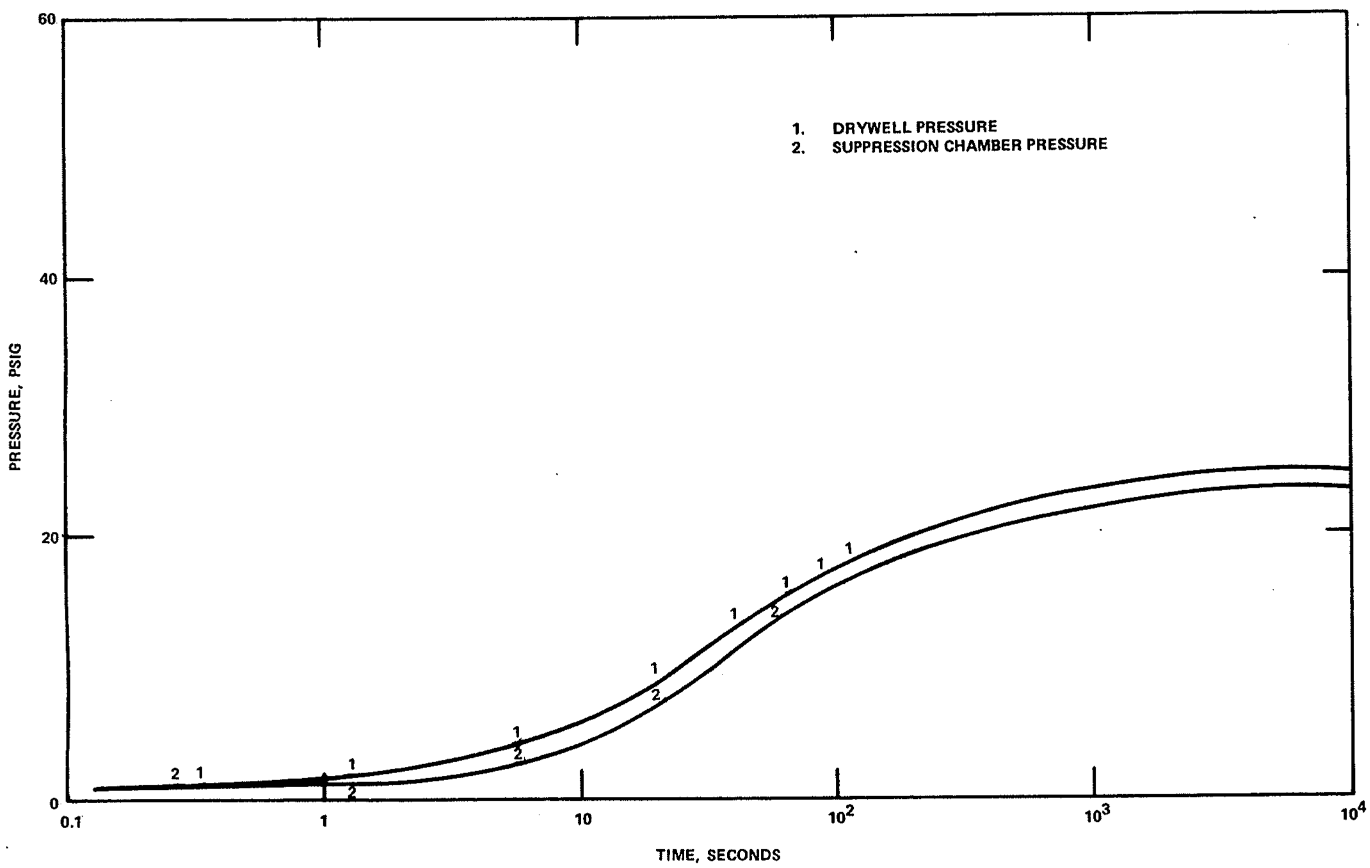
**FIGURE 6.2-13**

**SUPPRESSION POOL TEMPERATURE LONG TERM RESPONSE (3499 MWT)**

FIGURE 6.2-14

HAS BEEN INTENTIONALLY DELETED




*NOTE:

THE CONTAINMENT TEMPERATURE RESPONSE DUE TO THE SBA HAS ALSO BEEN RE-EVALUATED USING THE BASES PROVIDED IN REFERENCES 1 AND 3. THE PREDICTED SHORT-TERM RESPONSE IS REPORTED IN REFERENCE 2.

*EDISON NOTE TO GE DRAWING

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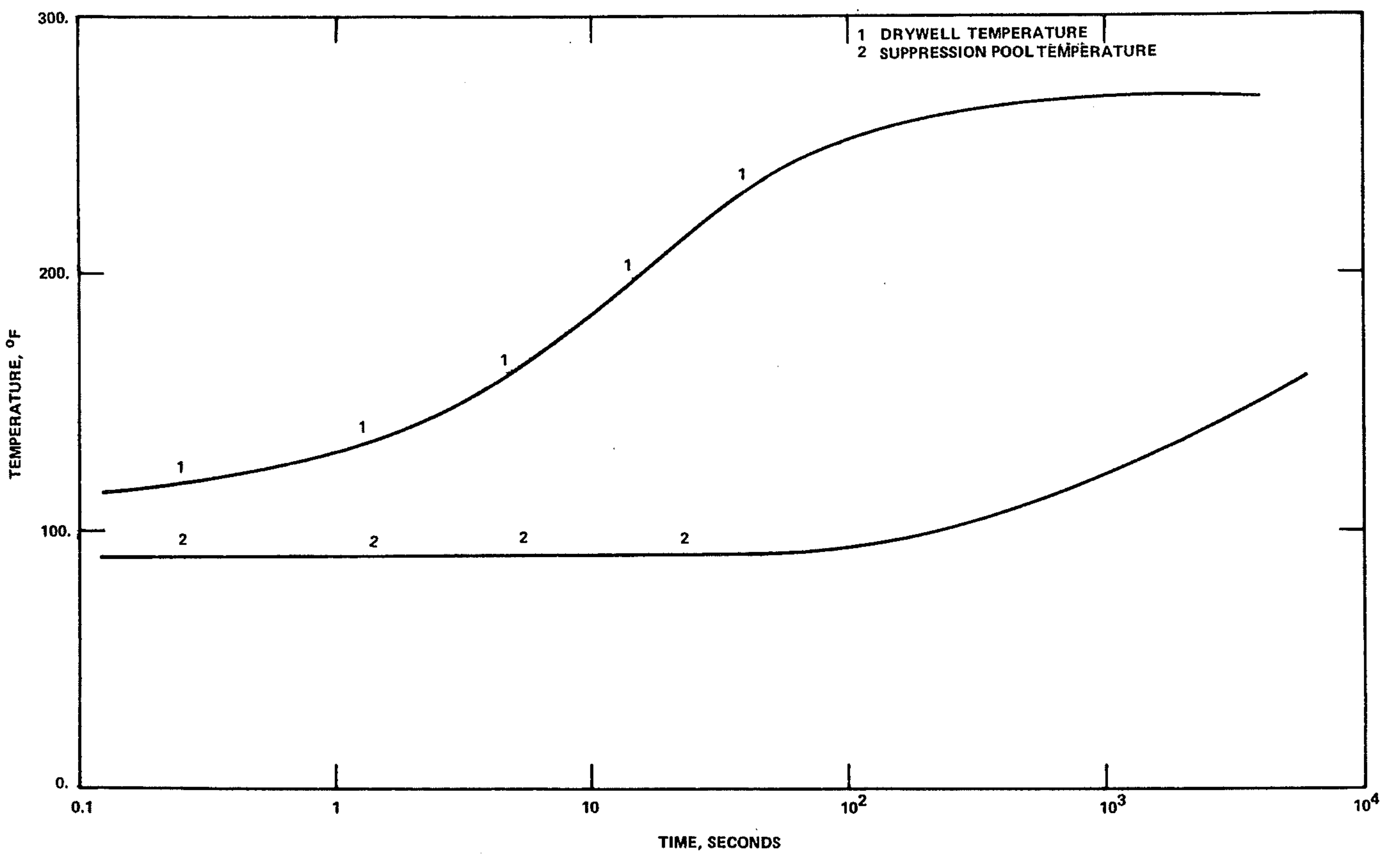
FIGURE 6.2-15

0.1 $FT^2$ LIQUID BREAK
PRIMARY CONTAINMENT PRESSURE RESPONSE
(BASED ON ORIGINAL POWER OF 3358 MWT)




*NOTE:

THE CONTAINMENT TEMPERATURE RESPONSE DUE TO THE SBA HAS ALSO BEEN RE-EVALUATED USING THE BASES PROVIDED IN REFERENCES 1 AND 3. THE PREDICTED SHORT-TERM RESPONSE IS REPORTED IN REFERENCE 2.

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FIGURE 6.2-16

0.1 $FT^2$ LIQUID BREAK
PRIMARY CONTAINMENT TEMPERATURE RESPONSE
(BASED ON ORIGINAL POWER OF 3,358 MWT)

FIGURES 6.2-17 THROUGH 6.2-19

HAVE BEEN INTENTIONALLY DELETED

Figure Intentionally Removed
Refer to Plant Drawing I-2649-01

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FIGURE 6.2-20

STANDBY GAS TREATMENT SYSTEM
P&ID

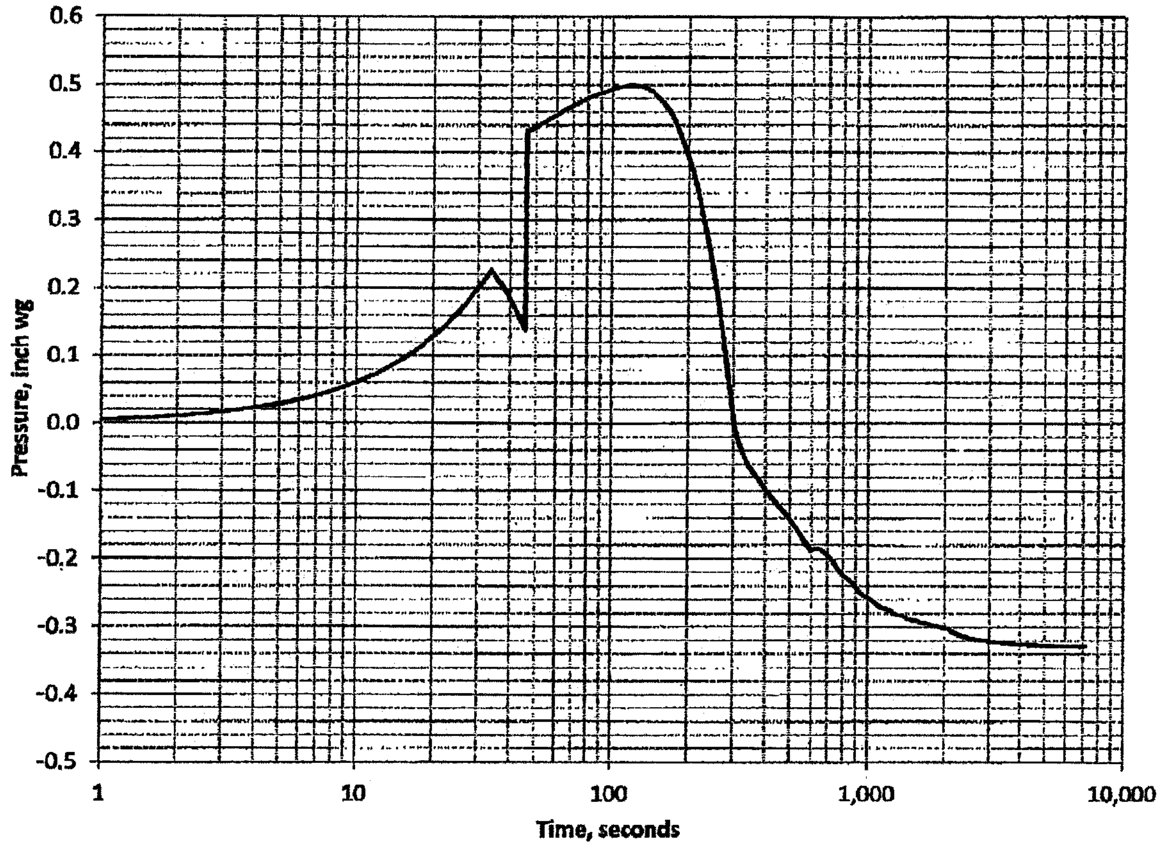


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FIGURE 6.2-21

SECONDARY CONTAINMENT RESPONSE
DUE TO A DBA-LOCA

FIGURE 6.2-22
HAS BEEN INTENTIONALLY DELETED

Figure Intentionally Removed
Refer to Plant Drawing M-2087

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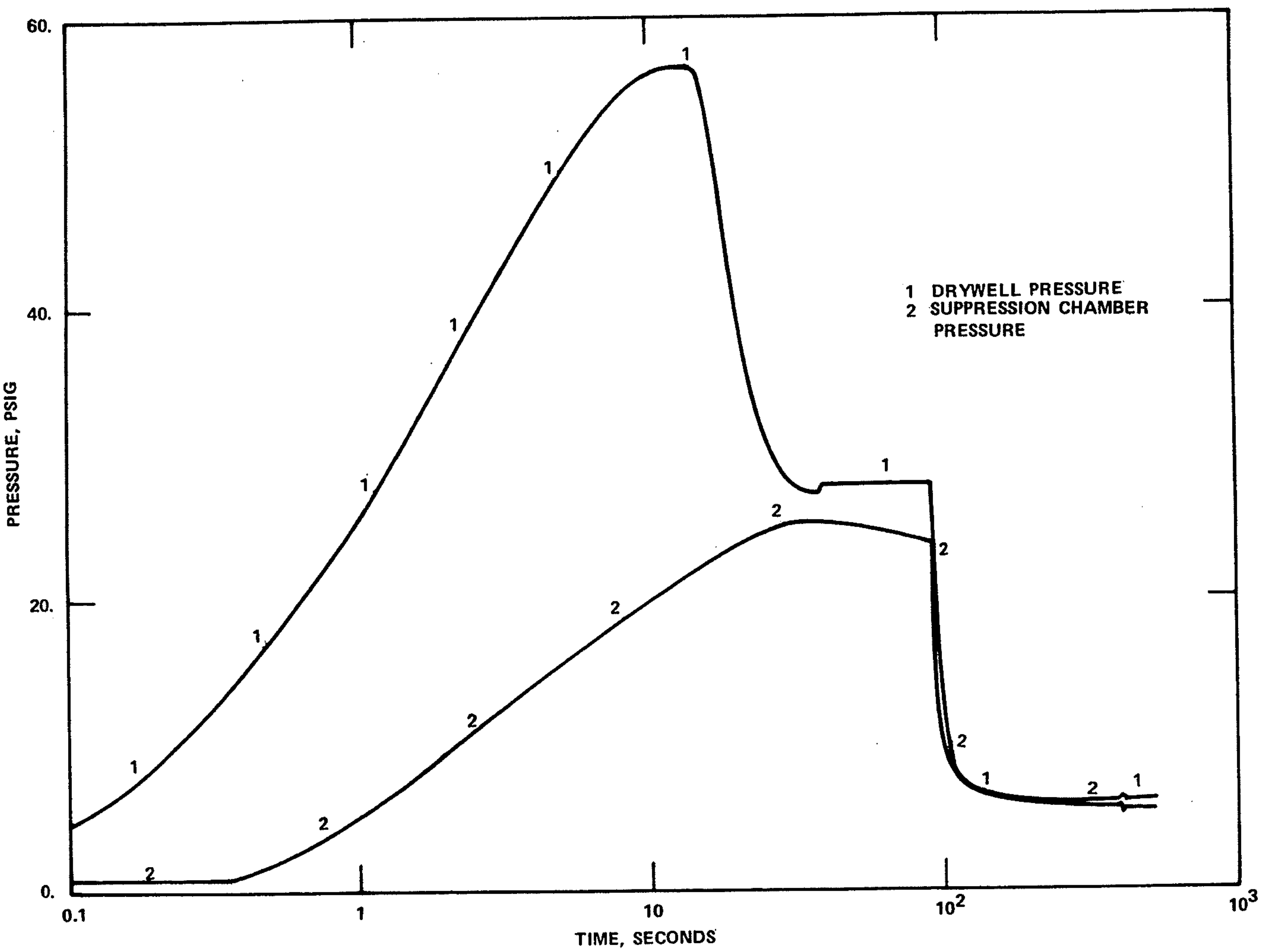
**FIGURE 6.2-23**

**POST-LOCA RECOMBINER P&ID**

FIGURE 6.2-24
HAS BEEN INTENTIONALLY DELETED

FIGURE 6.2-25
HAS BEEN INTENTIONALLY DELETED




*NOTE:

THE CONTAINMENT PRESSURE RESPONSE DUE TO THE RECIRCULATION LINE BREAK HAS ALSO BEEN RE-EVALUATED USING THE BASES PROVIDED IN REFERENCES 1 AND 3. THE PREDICTED SHORT-TERM RESPONSE IS REPORTED IN REFERENCE 2.

*EDISON NOTE TO GE DRAWING

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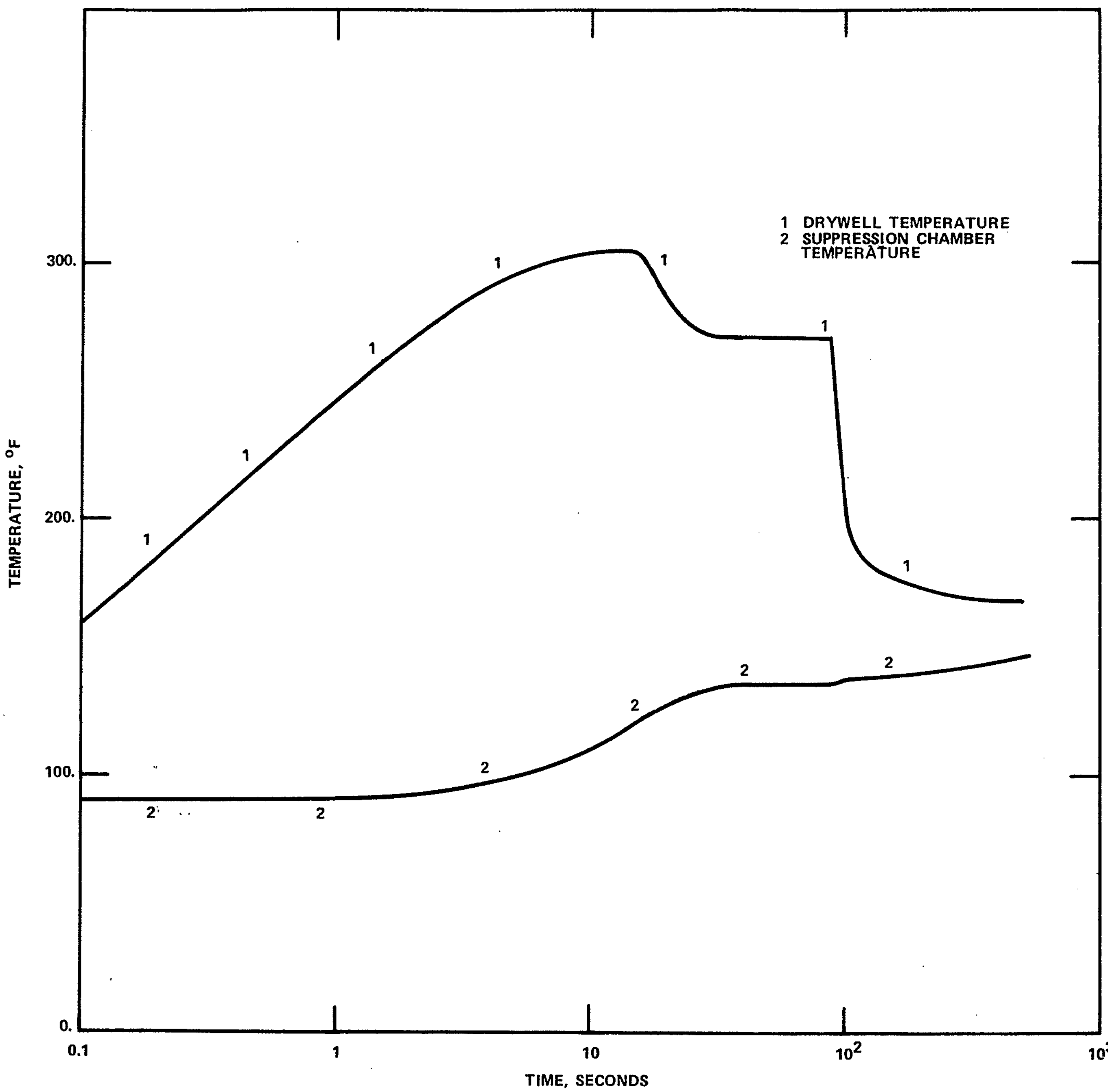
**FIGURE 6.2-26**

**RECIRCULATION LINE BREAK PRIMARY CONTAINMENT INITIAL PRESSURE TRANSIENT (3358 MWT)**




*NOTE:

THE CONTAINMENT TEMPERATURE RESPONSE DUE TO THE RECIRCULATION LINE BREAK HAS ALSO BEEN RE-EVALUATED USING THE BASES PROVIDED IN REFERENCES 1 AND 3. THE PREDICTED SHORT-TERM RESPONSE IS REPORTED IN REFERENCE 2.

*EDISON NOTE TO GE DRAWING

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**FIGURE 6.2-27**

**RECIRCULATION LINE BREAK PRIMARY CONTAINMENT INITIAL TEMPERATURE TRANSIENT (3358 MWT)**

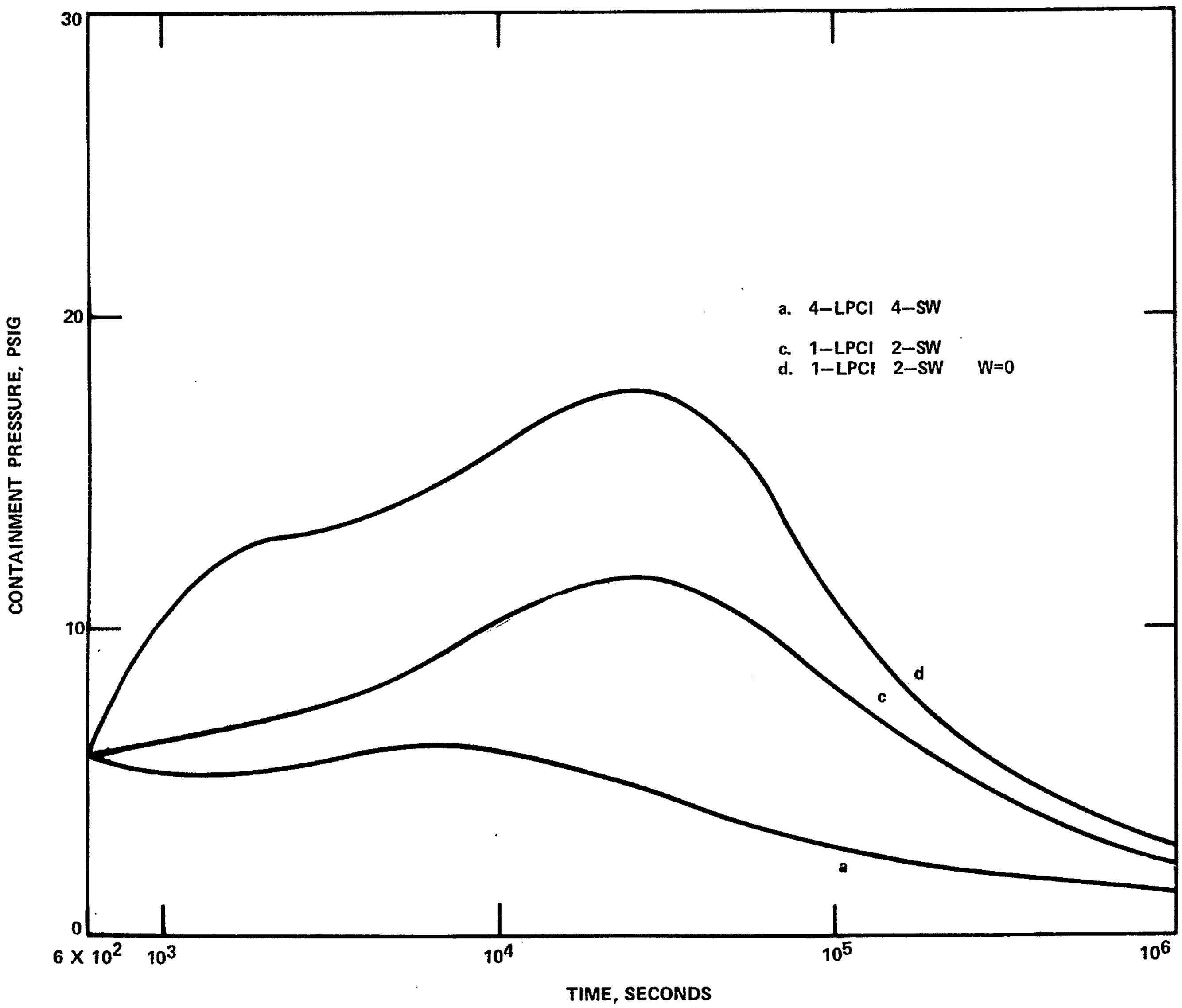


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**FIGURE 6.2-28**

**PRIMARY CONTAINMENT PRESSURE LONG TERM RESPONSE (3358 MWT)**

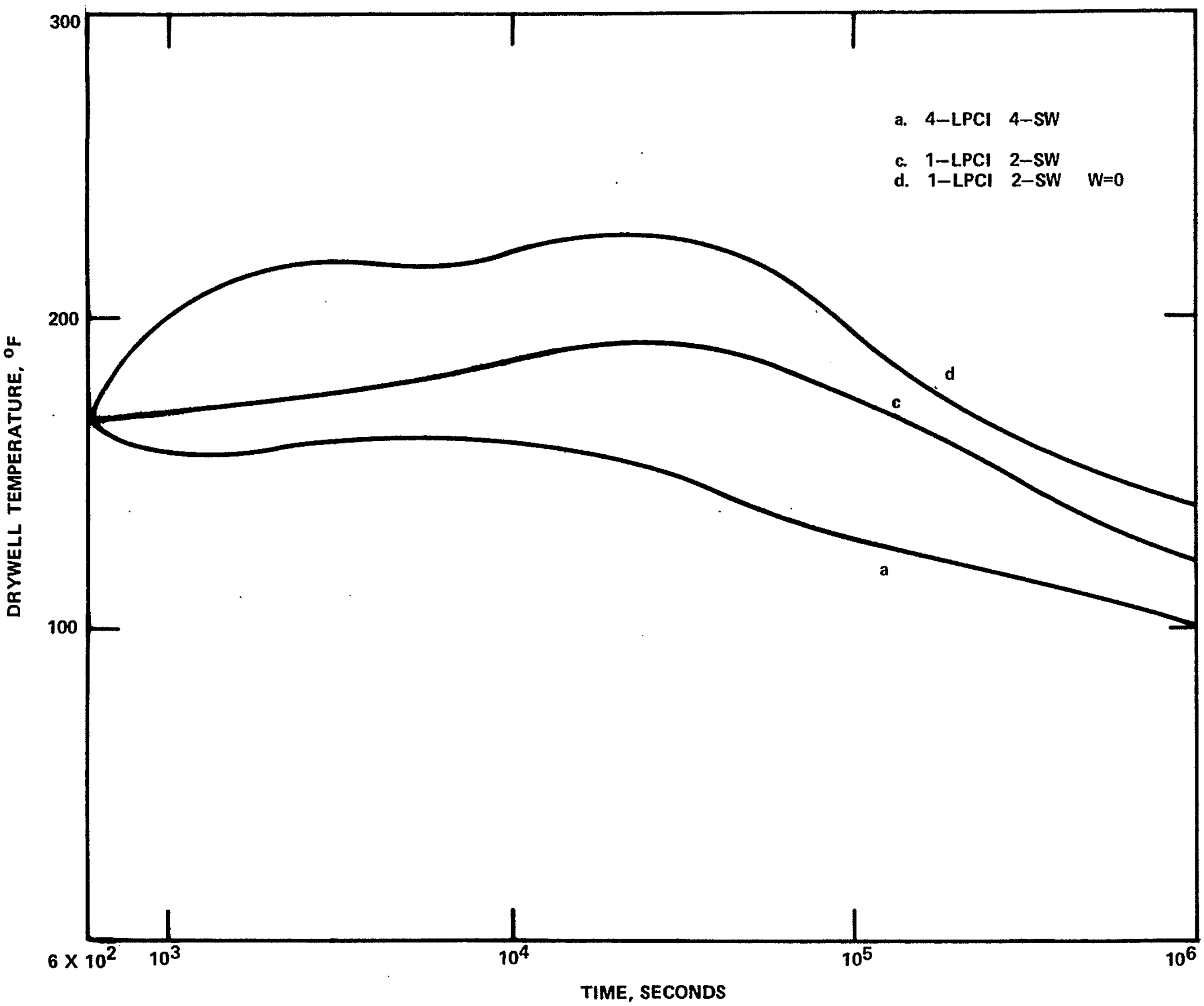


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**FIGURE 6.2-29**

**DRYWELL TEMPERATURE**
**LONG TERM RESPONSE**
**(3358 MWT)**

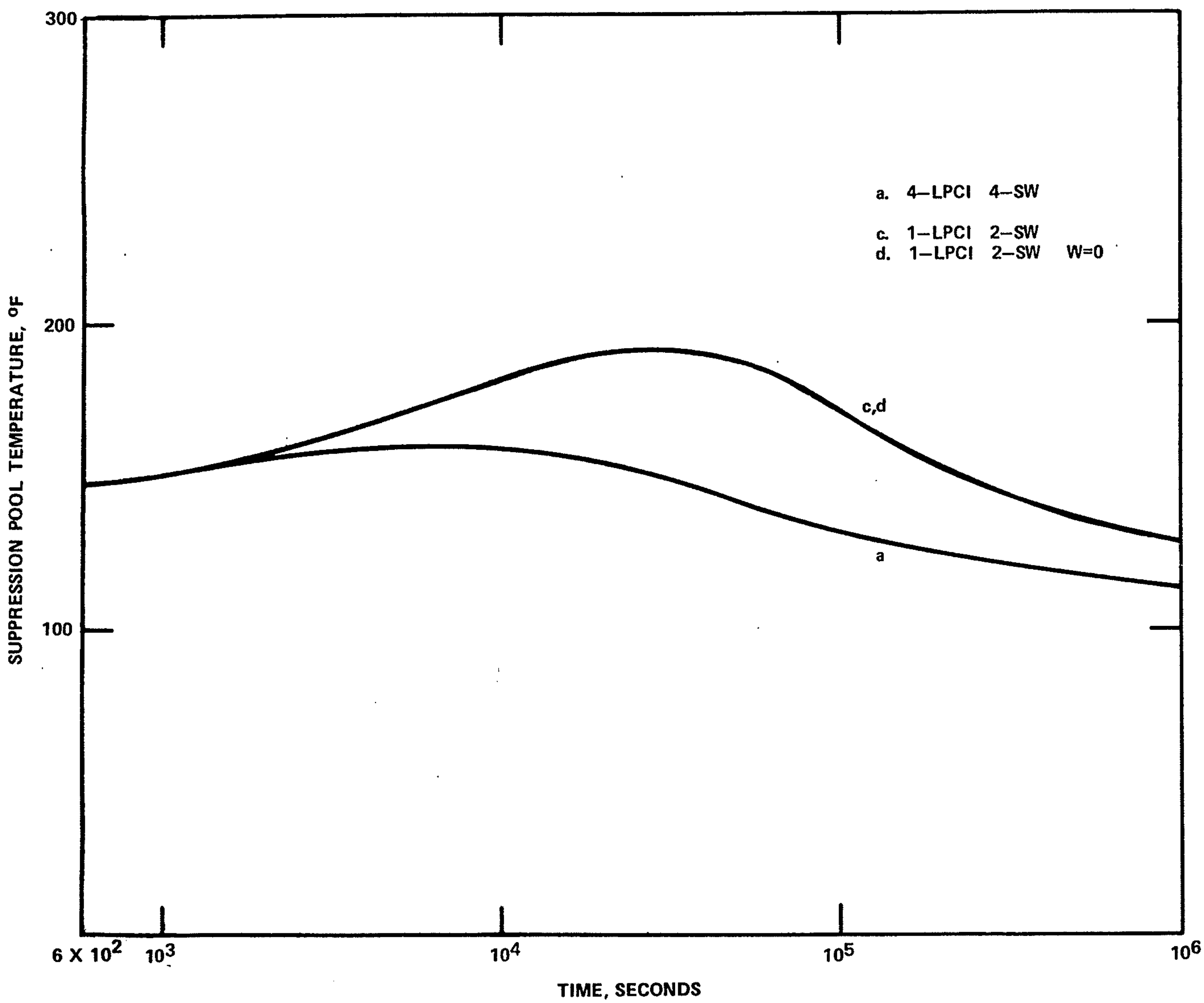


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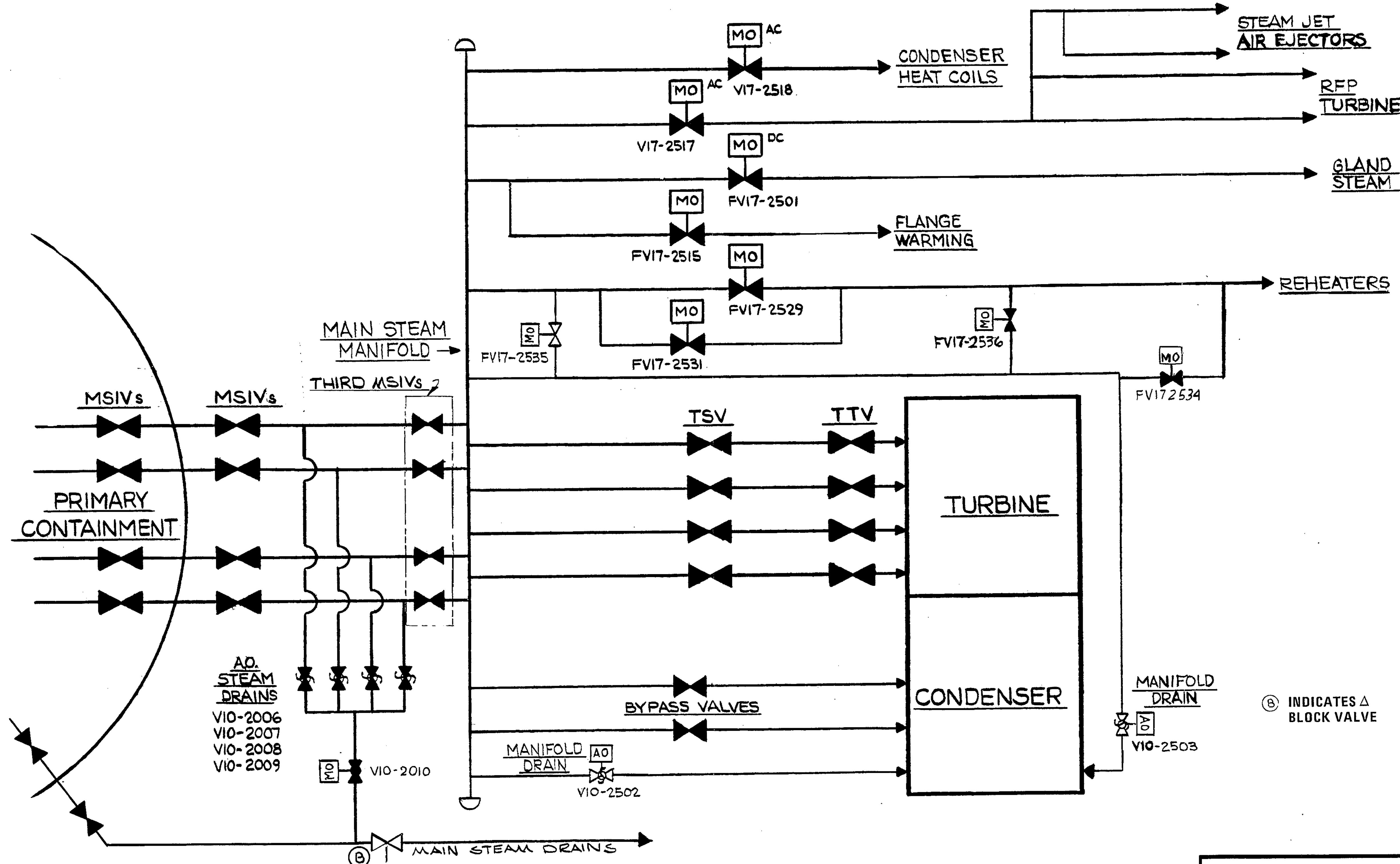
**FIGURE 6.2-30**

**SUPPRESSION POOL TEMPERATURE LONG TERM RESPONSE (3358 MWT)**




**Fermi 2**

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FIGURE 6.2-31

MAIN STEAM SYSTEM PIPING AND VALVES

REV 6 3/93

Figure Intentionally Removed
Refer to Plant Drawing M-3045

Fermi 2

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FIGURE 6.2-32

MAIN STEAM ISOLATION
VALVE LEAKAGE CONTROL SYSTEM

## 6.3 EMERGENCY CORE COOLING SYSTEMS

Four systems are provided to protect the core against various sizes of hypothetical pipe breaks. Three of these inject emergency core cooling water into the reactor and one is a reactor pressure vessel (RPV) automatic depressurization system (ADS). The three injection systems consist of the high-pressure coolant injection (HPCI), low-pressure coolant injection (LPCI), and core spray system. The protection afforded by these systems meets the NRC criteria given in 10CFR 50.46.

### 6.3.1 Design Bases

The objective of the emergency core cooling systems (ECCS), in conjunction with the containment, is to limit the release of radioactive materials should a LOCA occur, so that resulting radiation exposures are kept within the guideline values given in 10 CFR 50.67 or 10 CFR 100 as applicable.

Safety design bases for the subsystems of the ECCS are given in the following subsections.

#### 6.3.1.1 Range of Coolant Ruptures and Leaks

The ECCS provides adequate core cooling in the event of any break or leak in the piping of the nuclear system process barrier up to and including the double-ended break of the largest line connected to the RPV. The selection of break sizes and break locations is discussed in Subsection 6.3.3.7.3.

#### 6.3.1.2 Fission Product Decay Heat

In the event of a LOCA, the ECCS removes delayed neutron fission heat, residual stored heat, and radioactive decay heat from the reactor core at a rate that limits the maximum fuel cladding temperature to a value less than the 10 CFR 50.46 limit of 2200°F. The amount of heat to be removed is discussed in Subsection 6.2.1.3.8.

#### 6.3.1.3 Reactivity Required for Cold Shutdown

The reactor is designed to be in the cold-shutdown condition with the control rod of highest reactivity worth fully withdrawn and all other control rods fully inserted. Refer to Subsection 4.3.2 for a complete discussion.

#### 6.3.1.4 Capability To Meet Functional Requirements

The following functional requirements are met:

a. The ECCS is provided with sufficient capacity, diversity, reliability, and redundancy to cool the reactor core under all accident conditions

b. The ECCS is initiated automatically by conditions that indicate the potential inadequacy of the normal core cooling

c. The ECCS is capable of startup and operation regardless of the availability of offsite power supplies and the normal generating system of the plant

d. Action taken to effect containment integrity does not negate the ability to achieve core cooling. All ECCS pumps are designed to operate without benefit of containment pressure

e. The components of the ECCS within the RPV are designed to withstand the transient mechanical loadings during a LOCA so that the required core cooling flow is not restricted

f. The equipment of the ECCS is designed to withstand the physical effects of a LOCA so that the core can be effectively cooled. Effects considered are missiles, fluid jets, pipe whip, high temperature, pressure, humidity, and seismic acceleration

g. A reliable supply of water for the ECCS is provided. The prime source of liquid for cooling the reactor core after a LOCA is a stored source located within the containment. The source is located so that a closed cooling water path is established during ECCS operation

h. The flow rate and sensing networks of each ECCS are testable during reactor shutdown. All active components are testable during normal operation of the nuclear system.

## 6.3.2 System Design

### 6.3.2.1 Emergency Core Cooling System Design

The bounds within which system parameters must be maintained and the acceptable inoperable components are discussed in the Technical Specifications.

The ECCS, containing four separate subsystems, is designed to satisfy the following performance objectives:

a. To limit the peak cladding temperature to 2200°F, to prevent a cladding metal/water reaction in excess of 1 percent of the cladding, and to maintain long-term coolability of the core in the event of a mechanical failure of the piping or the nuclear system process barrier, up to and including a break equivalent to the largest nuclear steam supply system (NSSS) pipe

b. To provide this protection by at least two independent, automatically actuated cooling systems

c. To function with or without external (offsite) power sources

d. To permit testing of the ECCS by acceptable methods, including, wherever practical, testing during power plant operations.

The aggregate of the ECCS is designed to protect the reactor core against fuel clad damage in excess of the limits set forth in 10 CFR 50.46 across the entire spectrum of line break accidents.

The operational capabilities of the various subsystems of the ECCS meet the functional requirements and performance objectives described below. Table 6.3-1 lists the types of LOCAs and the ECCS that would operate in response to each.

During the first 10 minutes following the initiation of operation of the ECCS, the functional requirement is satisfied for all combinations of single active component failures and single pipe breaks, including pipe breaks in any ECCS subsystem which might partially or completely disable that subsystem.

After the first 10 minutes following the initiation of operation of the ECCS, and in the event of an active or passive component failure in the ECCS or its essential support system, long-term core and containment cooling is provided by any one LPCI or core spray loop delivering water to the RPV and by one residual heat removal (RHR) pump supported by one RHR heat exchanger with 100 percent service water flow. Containment cooling, using one RHR pump supported by one RHR heat exchanger, can be delayed up to twenty minutes following the DBA LOCA.

The power for operation of the core spray and LPCI is from regular ac power sources. Upon loss of the regular power, operation is from onsite standby ac power sources. Standby sources have sufficient diversity and capacity so that all core spray and LPCI requirements are satisfied. One core spray loop and one LPCI loop are powered from one ac division and the other core spray loop and LPCI loop are powered from a second and separate ac division. Four diesel generators are the site backup power supplies, with two diesel generators and two buses per division.

With the exception of LPCI while lined up in shutdown cooling and RPV pressure is less than or equal to the cut in pressure, all systems start automatically. The starting signal comes from independent and redundant sensors of drywell pressure and low RPV water level. Refer to Subsection 7.3.1 for a complete discussion of the ECCS instrumentation and starting and control logic.

Piping and instrumentation diagrams for the subsystems and components that constitute the ECCS are provided and referenced under the discussion of the subsystem or component.

#### 6.3.2.2 Equipment and Component Descriptions

The four types of core cooling systems (HPCI, ADS, core spray, and LPCI) are described in this section with reference to the appropriate piping and instrumentation diagrams and system process diagrams.

##### 6.3.2.2.1 High Pressure Coolant Injection System

The HPCI system is provided to ensure that the reactor core is adequately cooled to meet the design bases in the event of a small break in the nuclear system and loss of coolant that does not result in rapid depressurization of the RPV. Liquid breaks up to approximately 0.1 $ft^2$ break area and steam breaks up to approximately 0.5 $ft^2$ break area are within the capability of the HPCI system alone. This permits the plant to be shut down while maintaining sufficient RPV water inventory until the RPV is depressurized. The HPCI system continues to operate until RPV pressure is below the maximum pressure at which LPCI operation or core spray system operation can maintain core cooling.

The HPCI system consists of a steam turbine assembly driving a constant-flow pump assembly and system piping, valves, controls, and instrumentation. The HPCI piping and instrumentation diagram is shown in Figure 7.3-1. The HPCI system process and valve lineup diagrams are shown in Figures 6.3-1 through 6.3-5. The schematic drawing is shown in Figure 6.3-1.

The principal HPCI equipment is installed in the reactor building. The turbine-pump assembly is located in a shielded area to ensure that personnel access to adjacent areas is not restricted during operation of the HPCI system and to be protected from the physical effects of design-basis accidents (DBAs) such as pipe whip, flooding, and high temperature.

The pump assembly is located below the level of the condensate storage tank and below the water level in the suppression pool to ensure positive suction head to the pumps.

Two sources of water are available. The HPCI system initially injects water from the condensate storage tank (see Figure 6.3-2). When the water level in the tank falls below setpoint level or when suppression pool level is high, the pump suction is auto-matically transferred to the suppression pool. This transfer may also be made from the main control room using remote controls. The transfer requires the opening of normally closed valves F041 and F042 in the pump suction line leading from the suppression pool. The opening of these valves automatically closes valve F004 in the pump suction line leading from the condensate storage tank. When the pump suction has been transferred to the suppression pool, a closed loop is established for recirculation of water escaping from a break (see Figure 6.3-3).

Injection water is piped to the reactor feedwater pipe at a T-connection.

The HPCI turbine is driven by steam from the RPV which, after reactor shutdown, is generated by decay and residual heat. The steam is extracted from a main steam line upstream of the main steam isolation valves (MSIVs). The HPCI inboard isolation valve (F002) and the bypass valve around the HPCI outboard isolation valve (F600) in the steam line to the HPCI turbine are normally open to keep the piping to the turbine at elevated temperatures. This permits rapid startup of the HPCI system. Signals from the HPCI control system open (with oil pressure available) or close the turbine control/stop valve.

A condensate drain pot is provided upstream of the turbine steam admission valve to prevent the HPCI steam supply line from filling with water. The drain pot normally routes the condensate to the main condenser, but upon receipt of a HPCI initiation signal or a loss of non-interruptible control air pressure, isolation valves on the condensate line automatically close.

The turbine has two devices for controlling power. One is a speed governor that limits turbine speed to its maximum operating level, and the other is a control governor with automatic speed setpoint control that is positioned by a demand signal from a flow controller to maintain constant flow over the pressure range of HPCI operation.

As reactor steam pressure decreases, the HPCI governor valves open further to pass the steam flow required to provide the necessary pump flow. The capacity of the system is selected to provide sufficient core cooling to prevent excessive clad temperatures while the pressure in the RPV is above the pressure at which core spray and LPCI become effective.

Startup of the HPCI system is completely independent of ac power. Only dc power from the station battery and steam extracted from the nuclear system are necessary. The HPCI controls automatically start the system and bring it to design flow rate within 60 sec from receipt of a primary containment (drywell) high-pressure signal or an RPV low water level signal. This time interval for HPCI injection is used in the Fermi 2 TRACG-LOCA analyses that demonstrate conformance to 10 CFR 50.46 (Reference 42).

High-pressure coolant injection operation automatically actuates the following valves:

a. HPCI pump discharge shutoff valve

b. HPCI steam supply shutoff valves

c. HPCI turbine stop valve

d. HPCI turbine control valves

e. HPCI steam line drain isolation valves

f. HPCI pump suction valve from condensate storage.

Startup of the hydraulic oil pump and proper functioning of the hydraulic control system is required to open the turbine valves. Operation of the barometric condenser components is functionally illustrated in Figure 7.3-2 and their failure does not prevent the HPCI system from fulfilling its core cooling objective. The same initiating signal automatically starts the turbine oil pump, and when sufficient oil pressure is developed, the stop valve begins to open. Contacts actuated by the HPCI turbine stop and turbine steam supply valve limit switches initiate the speed control ramp generator which slowly increases the control valve position from closed to the value demanded by the flow controller. As a result, the turbine smoothly accelerates from rest to the speed at which rated pump flow is developed. When rated flow is established, the flow controller signal adjusts the setting of the control governor so that rated flow is maintained as nuclear system pressure decreases.

A minimum flow bypass is provided for pump protection (see Figure 6.3-4). The bypass valve (F012) automatically opens when a low flow combined with a high discharge pressure signal is sensed. It automatically closes on a high-flow signal or if the closing of either the turbine stop valve or steam inlet valve is sensed. Pump discharge pressure is sensed by PS N027, and flow is sensed by FS N006. When the bypass is open, flow is directed to the suppression pool.

A full-flow functional test of the HPCI can be performed during plant operation by drawing suction from the condensate storage tank and returning the water to the tank through a full-flow test line (see Figure 6.3-5). During this test, a signal to initiate the HPCI automatically stops the test mode and starts the water injection to the feedwater line. This transfer from the test mode to the accident mode requires the closing of the normally closed (but open for the test mode) valves F008 and F0ll located in the test line connecting the pump discharge and the condensate storage tank.

A cross connection is provided from the HPCI Test Line piping to the GSW piping to be used as part of the Flexible and Diverse Coping Strategy (FLEX) to mitigate Beyond Design Basis External Events (BDBEE) in response to NRC Order EA-12-049.

Exhaust steam from the HPCI turbine is discharged to the suppression pool. A drain pot at the low point in the exhaust line collects moisture present in the steam. Collected moisture is discharged to the suppression pool or bypassed to the barometric condenser.

The HPCI turbine gland seals are vented to the barometric seal condenser. Noncondensable gases from the barometric condenser are pumped to the standby gas treatment system (SGTS).

A redundant system of check valves and isolation valves has been installed as a vacuum breaker line that connects the air space in the suppression pool with the HPCI turbine exhaust line. This eliminates any possibility of water from the suppression pool being drawn into the HPCI turbine exhaust line. The two isolation valves (electrically separated) in series in this vacuum breaker line operate automatically via a combination of low reactor pressure and high drywell pressure. Test connections are provided on either side of the two check valves.

The system component classifications plus additional requirements are described in Chapter 3. The pump is designed and tested in accordance with the standards of the Hydraulic Institute.

The system is designed for a service life of 40 years, accounting for corrosion, erosion, and material fatigue. The various operations of the HPCI components are summarized below.

The HPCI turbine is shut down automatically by any of the following signals:

a. Turbine overspeed--prevents damage to the turbine casing

b. Reactor pressure vessel high water level--indicates that core cooling requirements are satisfied

c. High-pressure coolant injection pump low suction pressure--prevents damage to the pump due to loss of flow

d. High-pressure coolant injection turbine exhaust high pressure--indicates a turbine or turbine control malfunction.

If an initiation signal is received after the turbine is shut down, the system will restart automatically if no shutdown signals exist.

Because the steam supply line to the HPCI turbine is part of the nuclear system process barrier, certain signals automatically isolate this line, causing shutdown of the HPCI turbine. Automatic shutoff of the steam supply is described in Section 7.3. However, automatic depressurization and the low-pressure systems of the ECCS act as backup, and automatic shutoff of the steam supply does not negate the ability of the ECCS to satisfy the safety objective.

In addition to the automatic operational features of the system, provisions are included for remote manual startup, operation, and shutdown (provided automatic initiation or shutdown signals do not exist). Remote controls for valve and turbine operation are provided in the main control room. The controls and instrumentation of the HPCI system are described, illustrated, and evaluated in detail in Section 7.3.

#### 6.3.2.2.2 Automatic Depressurization System

In case the capability of the feedwater pumps, control rod drive (CRD) pumps, reactor core isolation cooling (RCIC) system, and HPCI system is not sufficient to maintain the reactor water level, the ADS functions to reduce the reactor pressure to a value low enough (<300 psig) to allow the LPCI and core spray systems to pump water to the RPV in time to cool the core consistent with the design bases.

The ADS uses five of the 15 safety/relief valves of the nuclear system pressure relief system to achieve the automatic blowdown to the suppression pool. The capacity of each relief valve is about 900,000 lb/hr at set pressure. The ADS starts operating soon enough after failure of the HPCI and dumps steam fast enough to ensure that the LPCI and core spray systems begin to operate and cool the fuel adequately.

To activate, the ADS must have drywell high pressure (2 psig) and RPV low water level (level 1) signals. Simultaneous occurrence of these drywell high pressure and RPV low water level conditions initiate a time delay of 120 seconds to allow the HPCI system time to recover level. After that time delay, ADS safety relief valves will operate if at least one RHR pump or both core spray pumps in either division are running (developing pressure). RPV low water level (level 1) signal also activates a bypass timer set for 8 minutes. This bypass time delay is provided to bypass the drywell pressure high logic circuit. If for some reason the drywell high pressure is not detected, the RPV low water level signal alone will activate the ADS safety relief valves after the 8 minute bypass time delay, plus the original 120 second time delay, provided that appropriate discharge pressure signals are present. The values shown above are based on analysis (Reference 34); refer to Technical Requirements Manual Table 3.5.1-1 for operating setpoints.

Opening of the relief valve requires pneumatic pressure to the valve's diaphragm actuator. This pneumatic supply is controlled by a solenoid-operated pilot valve. For the ADS to function, this valve control system must be operable, and there must be a pneumatic supply. The accumulator associated with the relief valves used with the ADS has sufficient capacity to allow for five operations of the pilot valves to cover interruptions if the pneumatic supplies are switched from the normal to the emergency backup sources. The relief valve pneumatic supply and backup supply systems are capable of performing their function for the long-term period of 100 days following an accident as required by NUREG-0737, Item II.K.3.28. (See also Section 5.2.2.2.3 for a description of the accumulator system.) The accumulator and pneumatic supply systems are capable of performing their design function during and following exposure to a harsh environment and/or a seismic event. In the automatic depressurization mode, the relief valves do not reset to normal safety/relief valve setpoints on low RPV pressure. To ensure proper cooling under all circumstances, including a postulated failure of ADS, reactor pressure relief can still be provided by operation of the non-ADS safety/relief valves. This ensures that the low pressure systems can be actuated with a HPCI failure and one additional single failure of the ADS, since any single failures affecting ADS will not impair remote operation of eight non-ADS safety relief valves.

The ADS valves stay open once activated until the reactor pressure is 50 psi higher than the containment pressure. These valves close when the reactor pressure decays to less than 50 psi above the containment pressure, and reopen when the reactor pressure is 100 psi above the containment pressure. Thus, the maximum pressure that can exist during the long-term period following a LOCA is 100 psi plus containment pressure.

The design, description, and evaluation of the pressure relief valves are discussed in detail in Subsection 5.2.2. See Section 7.3 for details on instrumentation and control.

The relief valve setpoints cannot be tested while they are in place on the steam lines. The safety/relief valves are designed to allow removal for bench testing of the setpoints during shutdown.

6.3.2.2.3 Core Spray System

The core spray system protects the core in the event of a large break in the nuclear system if the feedwater pumps, the CRD pumps, the RCIC, and the HPCI systems are unable to maintain RPV water level.

The protection provided by the core spray system also extends to a small break if the feedwater pumps, CRD pumps, RCIC, and HPCI systems are all unable to maintain the RPV water level and the ADS has operated to lower the RPV pressure so that LPCI and the core spray system provide core cooling.

Two independent loops are provided as a part of the core spray system. Each loop consists of two single-stage, in-line water pumps with suction and discharge connected in parallel and each pump driven by an 800-hp electric motor; a spray sparger in the RPV above the core; piping and valves to convey water from the suppression pool to the pumps and to the sparger; and the associated controls and instrumentation. Figures 6.3-7 through 6.3-11 show the schematic process and valve lineup diagrams of the core spray system. The piping and instrumentation diagram is shown in Figure 7.3-7.

In case of low water level in the RPV or high pressure in the drywell, the core spray system automatically starts and the pumps in the two core spray loops are signaled to start after a 5-sec delay on auxiliary ac power. This signal also starts without delay the diesel generator set and the LPCI system. In case auxiliary ac power is lost, the pumps start in sequence (with time delay) on standby ac power. Pump suction valves F001A, B, C, and D are normally locked open to ensure a positive suction head for the pumps. The test bypass motor-controlled valves F0l5A and B, normally closed, are signaled to close if open. When the reactor pressure is permissive (<500 psig), valves F004A and B (normally open) and F005A and B are signaled to open automatically. The pumps take water from the suppression pool and discharge to the sparger ring and nozzle spray. This condition is shown in Figure 6.3-8.

When the system is actuated, water is taken from the screened suction line in the suppression pool. Flow then passes through a normally open, motor-operated valve. A keylock switch is installed in the control circuit with position indication available in the main control room. This valve is located in the core spray pump suction line as close to the suppression pool as practical. It can be closed by a remote manual switch from the main control room to isolate the system from the suppression pool in case of a leak from the core spray system.

The four core spray pumps are located in the reactor building below the water level in the suppression pool to ensure positive pump suction. The pumps, piping, controls, and instrumentation of each loop are separated and protected so that any single physical event, or missile generated by rupture of any pipe in any system within the containment drywell, cannot make both core spray loops inoperable. The switchgear for each loop is in a separate emergency bus room for the same reason.

A vent line with two normally closed valves is provided from the pump casing for filling the pump with water. A shaft seal drain is provided, which drains to the radwaste system, along with the vent line. Leakage from the drain line is measured during primary containment leakage tests.

A low-flow bypass line is provided from the pump discharge to below the surface of the suppression pool. The bypass flow is required to prevent the pump from overheating when pumping against a closed discharge valve. An orifice limits the bypass flow. A manual valve that is normally locked open is used to close the bypass line for maintenance. A motor-operated valve on the bypass line closes upon receipt of a signal from a flow switch in the main discharge line.

Two relief valves, set for 500 psig, protect each low pressure core spray system loop upstream of the outboard shutoff valve from reactor pressure. The relief valves discharge to the suppression pool.

Two motor-operated valves are provided to isolate the core spray system from the nuclear system when the core spray pump is not running. These valves admit core spray water to the inboard check valve when signaled to open at approximately 500-psig RPV pressure. Both valves are installed outside the drywell to facilitate operation and maintenance, but as close as practical to the drywell to limit the length of line exposed to reactor pressure. The valve nearer the containment is normally closed to back up the inside check valve for containment purposes. The outboard valve is normally open to limit the equipment needed to operate in an accident condition. A test line is normally closed with two normally closed valves and a pipe cap to ensure containment.

A check valve is provided in each core spray line just inside the primary containment to prevent loss of reactor coolant outside containment in case the core spray line breaks. A normally locked-open manual valve is provided downstream of the inside check valve to shut off the core spray system from the reactor during shutdown conditions for maintenance of the upstream valves. The two core spray system pipes enter the RPV through nozzles l20° apart. Each pipe then divides into a semicircular header with a downcomer at each end which turns through the shroud near the top. A semicircular sparger is attached to each of the four outlets to make two practically complete circles, one above the other inside the shroud head. Short elbow nozzles are spaced around the spargers to spray the water radially into the tops of the fuel assemblies.

Core spray piping upstream of the outboard shutoff valves F004A and B is designed for the lower pressure and temperature of the core spray pump discharge. The outboard shutoff valve and downstream piping are designed for RPV pressure and temperature. The pumps, piping, and valves are designed to meet requirements described in Chapter 3. The pumps are also designed and tested in accordance with the standards of the Hydraulic Institute. Pump operability testing under the plant technical specifications and the in-service testing program ensures pump operation at or above the minimum required performance assumed in the plant safety analyses. Pump test acceptance criteria developed for this purpose are required to include consideration of the lowest allowed emergency diesel generator (EDG) operating frequency based on the maximum allowed frequency control tolerance as well as pressure and flow test instrument accuracies.

The core spray pumps and all automatic valves can be operated individually by manual switches in the main control room. Operating information is provided in the main control room with pressure indicators, flowmeters, and indicator lights. Automatic signals to start the system preempt all other signals while the system is in auto mode. In the manual condition, the pump or valve will be under total operator control. The manual condition is indicated to the operator in the main control room.

6.3.2.2.3.1 Core Spray Test Mode During Plant Operation

A test line for the rated core spray system flow rate is provided to route the suppression pool water from the pump discharge to the suppression chamber without entering the reactor pressure vessel (see Figure 6.3-9). During reactor operation, the core spray injection valves are normally closed. The pumps are started by the operator using the remote manual control in the main control room. Valves F015A and B in the test lines are opened partially to achieve the rated flow through the test lines. This mode of operation permits testing of pump operation and ensures that rated flow is achieved. It also permits testing of control and operation of components of the low pressure section of the core spray system.

6.3.2.2.3.2 Core Spray Test During Plant Shutdown

To provide for system testing during a plant shutdown (see Figure 6.3-10), a connection is provided from the condensate storage tank to the pump suction. This condensate is used for flow testing of the spray nozzles inside the pressure vessel. A normally closed manual valve is provided between the condensate storage tank and the pump suction line to minimize the possibility of communication of the condensate to the suppression pool and to avoid extension of the primary containment.

During plant shutdown, the core spray system can be tested by manually opening valves F002A and B. The pumps are started by remote manual control, and valves F004A and B and F005A and B are opened by remote manual switch. System operation including sparger rings can be tested in this manner during shutdown conditions. Any system maintenance or repairs may be made on the core spray system during plant shutdown by manually closing valves F007A and B, which are normally locked open.

System operability is determined by performing a system functional test which includes simulated automatic actuation of the system throughout its emergency operating sequence and verifying that each automatic valve in the flow path actuates to its correct position. Actual injection of coolant into the reactor vessel may be excluded from this test, which is performed every 24 months. Such testing was performed during initial plant startup and periodic performance is not needed as the continued operability of the LPCS header is assured by means of the break detection logic discussed in Subsection 7.3.1.2.3.9.

6.3.2.2.3.3 Core Spray Minimum Flow Bypass Mode

The pump discharge line is provided with a low-flow bypass line to protect the core spray pumps from overheating during operation at high vessel pressure (see Figure 6.3-11). Flow-measuring element FE N002 is coupled to flow switch FS N006, which at a nominal flow rate of less than 2100 gpm, signals bypass line motor-operated valve F031 to open automatically. Water from the suppression pool is then routed through the bypass lines back to the suppression pool. As soon as flow is established in the pump discharge lines, the signal from FS N006 signals the minimum bypass motor-operated valves F03lA and B to close automatically.

#### 6.3.2.2.4 Low Pressure Coolant Injection System

In case of low water level in the reactor or high pressure in the containment drywell, the LPCI mode of operation of the RHR system pumps water into the RPV in time to cool the core consistent with the design bases. The core spray system starts from the same signals and operates independently to achieve the same objective. The isolation valves for these two systems are opened when reactor pressure is less than 500 psig, but injection flow does not occur until the differential pressure across the check valves permits. This occurs when the RPV pressure is less than 300 psig.

Low-pressure coolant injection operation provides protection to the core for the case of a large break in the nuclear system when the feedwater pumps, the CRD pumps, and RCIC and HPCI systems are unable to maintain RPV water level. Manual override of LPCI operation is prevented by two keylocked switches. Conditions for manual override of LPCI operation are described in the Fermi 2 Containment Control Emergency Procedures.

Protection provided by LPCI also extends to a small break in which the feedwater pumps, CRD pumps, and RCIC and HPCI systems are all unable to maintain the RPV water level, and the ADS has operated to lower the reactor vessel pressure so that LPCI and core spray systems start to provide core cooling.

In the event of a break in one of the two reactor recirculation system loops, logic is provided to sense the broken loop and to inject the LPCI flow into the unbroken loop. Thus, the flows from the two LPCI system loops are interconnected by valving. Since electrical power to each LPCI loop is isolated (Divisions I and II), it is necessary to have a swing bus arrangement that permits the valves of an LPCI loop that has been disabled by a single failure of a divisional electrical supply to be energized. This feature preserves the ability of the LPCI to cross-connect flow and inject into the unbroken recirculation loop.

The Fermi 2 LPCI valve logic also provides for closing only the valve in the discharge side of the unbroken reactor recirculation loop as opposed to earlier logic that closed both the discharge and suction valves in the unbroken reactor recirculation system loop. The logic change provides for continued depressurization (permitting coolant injection by the core spray system) in the event that the single failure is the LPCI logic that selects the broken reactor recirculation system loop.

##### 6.3.2.2.4.1 Accident Mode

Valves F048A and F048B open automatically. Also, valves F015 and F017 in the loop corresponding to the undamaged recirculation loop receive a signal to open automatically upon receipt of reactor vessel low pressure (<500 psig) signal. See Figure 6.3-14.

The system pumps take water from the suppression pool and pump it into the core region of the reactor vessel through the undamaged recirculation loop. The system pumps are rated at 10,000 gpm per pump. The rated flow of 30,000 gpm is delivered with three-pump operation at 20 psid pressure difference between the reactor vessel and the primary containment. A redundancy of pumps is provided so there is one more pump available than the number required for the rated flow in the LPCI. The core is flooded to an adequate height and the level maintained by the LPCI operating alone with three of four pumps operating.

Soon after the LOCA (assuming offsite power is also lost), the high-drywell-pressure or reactor-low-water signal initiates the selected valves in the LPCI and core spray systems to open. These valves receive power as soon as it is restored by the emergency diesel generators. The LPCI system pumps are also signaled to start. The core spray system pumps start with 5-sec delay, i.e., within 18 sec of the accident. The LPCI injection valves are fully open and the recirculation loop discharge valve (in the undamaged loop) closes within 77 seconds. This interval is the maximum allowable time from the high drywell pressure initiating signal to pump at rated speed and ready to inject flow to the vessel with emergency power that is used in Fermi 2 TRACG-LOCA analyses (Reference 42).

6.3.2.2.4.2 Low Pressure Coolant Injection Loop Selection Logic

This system is described in Subsection 7.3.1.2.4.

6.3.2.2.4.3 Low Pressure Coolant Injection Test Mode

A design flow (10,000 gpm per pump) functional test of the RHR system pumps is performed for each pair of pumps during plant operation by taking suction from the suppression pool and discharging through the test lines back to the suppression pool (see Figure 6.3-15). Discharge valve F015 to the reactor recirculation line remains closed, and reactor operation is undisturbed. The upstream and downstream valves for the containment spray headers (F016 and F021) are tested or exercised individually by remote manual switches in the main control room.

The control system is designed to provide automatic return from test mode to operating mode if LPCI injection is required during testing.

6.3.2.2.4.4 Low Pressure Coolant Injection Minimum Flow Bypass Mode

This mode of operation is provided to protect the system pumps from overheating at low flow rates, by routing the pump flow through the minimum bypass lines to the suppression chamber. A single motor-operated valve F007 in each bypass line automatically opens upon sensing low flow after either RHR pump within the associated division is started. This valve automatically closes whenever the flow from either of the associated main system pumps is above the low-flow setting. One switch (N021) is used for each loop. (See Figure 6.3-16).

Low-pressure coolant injection pump and piping equipment is described in detail in Subsection 5.5.7, which also describes the other functions served by the same pumps if not needed for the LPCI function.

6.3.2.2.5 Emergency Core Cooling System Discharge Line Fill System

One design requirement of any core cooling system is that cooling water flow to the RPV be initiated rapidly when the system is called on to function. This quick-start system characteristic is provided by quick-opening valves and quick-start pumps. By always keeping the HPCI, LPCI, and core spray pump discharge lines full, the lag between the signal for pump start and the initiation of flow into the RPV can be minimized. If for some reason these lines were empty when the systems were called for, not only would the lag time be increased, but also the lines would be subjected to unnecessarily large momentum forces associated with accelerating fluid into an empty pipe. To prevent draining of the ECCS-discharge lines, a fill system is provided to keep the core spray lines charged with demineralized water and the RHR lines charged with condensate water by a pressure regulating valve. A system is provided to maintain the HPCI pump discharge piping between the normally closed injection valve and pump discharge check valve charged with condensate water.

Since the core cooling pumps are located in the subbasement of the reactor building, approximately 75 ft below the point where the discharge piping enters the RPV, check or stop-check valves are provided near the pumps to prevent backflow from emptying the lines into the suppression pool. These valves will leak slightly, producing a small backflow that will eventually empty the discharge piping. The core spray lines are kept charged with demineralized water and the RHR lines are kept charged with condensate water by a pressure regulating valve. The HPCI pump lines are charged by the head (gravity) from the condensate storage tanks. The HPCI pump discharge piping valves up to the normally closed injection valve are also kept charged with condensate water. Alarms are provided for RHR and core spray fill line low pressure.

The demineralized water storage system supplies water to the reactor building, turbine building, radwaste building, auxiliary steam boilers, and core spray fill systems from a common manifold. The demineralized water supply from the manifold to the fill system is controlled by the pressure regulating valve. The water supply to the manifold is pumped from the demineralized storage system by a demineralized water jockey pump (DWJP) and, as required, by one or two demineralized water transfer pumps (DWTP).

The condensate water storage system supplies water to the RHR keep fill system from a reactor building second floor header. The condensate water supply from the header to the fill system is controlled by a pressure regulating valve. The water supply to the header is from the condenser pumps through a 4-inch supply line located downstream of the polishing demineralizers. During a plant shutdown, condensate is supplied by the condensate storage jockey pump and as required by the normal hotwell supply Pump.

When the demand for demineralized water is less than 20 gpm, the DWJP maintains the manifold pressure at 82 psig. If the quantity of water supplied by the DWJP exceeds 20 gpm, the manifold pressure drops. When the pressure reaches a defined setpoint, a pressure switch on the common manifold activates and one of the two DWTPs automatically starts. Should the demineralized water demand exceed the capability of the DWJP and one DWTP, the manifold pressure would continue to drop. At a lower setpoint, a second switch would activate to start the second DWTP and activate an alarm in the main control room.

The RHR fill system consists of two fill lines (branching from a single valve) supplied with condensate water from the header and terminates at the connections to the RHR pump discharge lines.

The core spray fill system consists of two fill lines (involving one pressure control valve in each line) supplied with demineralized water from the common manifold and terminates at the connections to the core spray pump discharge lines.

Vent and drain connections have been incorporated at all high and low points in the RHR and core spray piping systems. Prior to initial start (such as after maintenance to the RHR or core spray system) the discharge lines are filled by manually venting the high-point vents of the RHR and core spray discharge lines to avoid any trapped air pockets in the discharge lines. The pressure control valve keeps the lines filled after initial filling.

The suction and discharge piping to the RCIC system is kept full up to the normally closed RCIC injection valve by static head from the condensate storage tanks and by appropriate high-point venting during initial fill and as required periodically by the Technical Specifications. The elevation of the injection valve is lower than the low level of the condensate storage tanks.

The HPCI piping from the discharge of the pump, through the check valves up to the normally closed injection valve is kept full from the static head of the Condensate Storage Tank (CST), and by appropriate high-point venting during initial fill and as required periodically by Technical Specifications. The elevation of the injection valve is below the minimum level of the CST, and any leakage from the system is made up by CST water. The relative height of the feedwater line connection for HPCI is such that the water in the feedwater lines keeps the remaining portion of the HPCI discharge line full of water.

However, the HPCI system discharge piping near the injection valve to the feedwater system absorbs heat from feedwater via conduction and valve leakage, that has the potential to form a localized steam void. When the HPCI turbine-driven pump is started, rapid pressurization of this line causes the void to collapse and produce a momentum transient which stresses the piping and related supports.

The momentum transient that is present during HPCI start has been analyzed and shown not to produce any damage to the HPCI piping, components, and supporting structure. However, the condensate water storage system is utilized to maintain the HPCI discharge piping between the normally closed injection valve and pump discharge check valve charged with condensate water to prevent possible void formation and minimize momentum transient effects. The pressure of the condenser pumps, connected to the HPCI discharge piping by a supply line located downstream of the condensate polishing demineralizers, maintains the piping charged with condensate water to prevent steam void formation.

In addition, the HPCI pump discharge piping just upstream of the injection valve is provided with cooling fins to remove heat and provide additional subcooling margin in the area of void formation.

#### 6.3.2.2.6 Emergency Equipment Cooling Water System

The emergency equipment cooling water system (EECWS) consists of two (redundant) systems that supply cooling water to emergency equipment that is automatically operable on high drywell pressure, low reactor building closed cooling water system (RBCCWS) differential pressure, or on a loss of offsite ac power or that may be manually initiated upon failure of the RBCCWS. In addition, the EECWS may be manually initiated with the non-essential loads subsequently restored to facilitate RBCCW heat exchanger cleaning, to enhance drywell cooling during high lake water (GSW) temperature, for testing, or to provide RHR Reservoir freeze protection during extreme cold weather. The system diagram is presented in Figure 9.2-2. The EECWS is designed to provide equipment cooling and ventilation space cooling for HPCI, RCIC, RHR, and core spray systems.

Each of the two supply and return cooling loops (Division I and Division II) consists of one circulating pump of sufficient capacity to circulate water through the system and return the cooling water to a full capacity heat exchanger. The demineralized water in the system is cooled by the emergency equipment service water system (EESWS).

Both the EECWS and the EESWS are discussed in Subsection 9.2.2.

#### 6.3.2.2.7 Emergency Core Cooling System Suction Lines

The two core spray, four RHR, and HPCI suction lines have remote manual motor-operated gate valves.

The piping is classified as Class 2 piping in accordance with ASME Boiler and Pressure Vessel (B&PV) Code Section III, 1971, and has been stress analyzed for thermal and deadweight flexibility and seismic dynamic response. This analysis established nozzle loads on the torus connections which in turn are analyzed in accordance with ASME B&PV Code Section III-B. As these connections are situated below the torus, they are protected against possible missiles originating from the slab above the torus or any high-pressure lines situated above the torus.

To prevent the loss of torus water due to an ECCS suction line break, a leak-detection system is provided. Any postulated break in the ECCS pump suction line is detected and the appropriate isolation valve can be activated to isolate the break.

The maximum distance between the containment nozzle and the center line of the isolation valve occurs on the four 24-in. RHR suction lines. This distance is 11 ft 8 in.

Leak detection for the ECCS suction lines is provided by a system that measures the rate of change of the liquid level in the sump of the floor drain. The operator can isolate the leaking line and verify that the leak is stopped by observing the sump level.

### 6.3.2.3 Applicable Codes and Classification

All ECCS piping, components, and system designs comply, as a minimum, with applicable codes, code cases, and addenda in effect at the time the equipment was procured. These systems are designed and constructed in accordance with Category I criteria and Quality Assurance Level 1.

The RHR/LPCI, HPCI, and core spray systems are each divided into two classes.

The Class 1 portion of each system includes all piping and components that are a part of the reactor system pressure boundary out to and including the second isolation valve.

The Class 1 portions of the RHR/LPCI, core spray systems, and HPCI are designed and constructed in accordance with Subsection NB of ASME B&PV Code Section III, 1971 or later issue and addenda of this code in effect at the date of purchase order, and conform with 10 CFR 50.55a, whichever is more restrictive.

The remaining portions of the RHR/LPCI, HPCI, and core spray systems are designated Class 2 and are designed and constructed in accordance with Subsection NC of ASME B&PV Code Section III, 1971 or later addenda in effect at the date of purchase order. The only exceptions to the foregoing are

a. The RHR/LPCI and core spray pumps were purchased in 1970 and therefore meet the requirements of Section B ASME Code for pumps and valves for nuclear power (1968 draft issue)

b. The shell sides of the RHR heat exchangers are designed and constructed in accordance with ASME B&PV Code Section III (1968), Class C, and Tubular Exchanger Manufacturers Association (TEMA) Class C Standard

c. The tube sides of the RHR heat exchangers are designed and constructed in accordance with ASME Section VIII, Division I, and TEMA Class C Standard

d. Relief valve code and standards are defined in Chapter 5

e. The HPCI turbine is a non-ASME component per ASME B&PV Section III, 1971 Edition, Article NE-1130

f. The HPCI barometric condenser is designed and constructed in accordance with the ASME B&PV Code Section VIII.

#### 6.3.2.4 Materials Specifications and Compatibility

Subsection 5.2.3 discusses general material considerations. Refer to Table 5.2-6 for a presentation of the specifications that generally apply to the selection of materials used in the ECCS. Nonmetallic materials such as lubricants, seals, packings, paints, primers, and insulation, as well as metallic materials, are selected as a result of an engineering review and evaluation for compatibility with other materials in the system and the surroundings with concern for chemical, radiolytic, mechanical, nuclear radiation, and temperature effects.

Subsection 6.3.2.19 contains a discussion by commercial name of materials in the primary containment that may conceivably interfere with ECCS performance as a result of their deterioration under LOCA conditions.

Materials for the principal components are listed in Table 6.3-2.

#### 6.3.2.5 Design Pressures and Temperatures

The design pressures and temperatures at various points in the system during each of several modes of operation of the ECCS subsystems can be obtained from the information blocks on the following process diagrams: Figures 6.3-1 through 6.3-5 for the HPCI system, Figures 6.3-7 through 6.3-11 for the core spray system, and Figures 6.3-14 through 6.3-16 for the LPCI. The operational characteristics of the ADS valves are presented in Chapter 5.

#### 6.3.2.6 Coolant Quantity

The HPCI system normally takes suction from the condensate storage tank. This tank is designed so that the last 150,000 gal is reserved for use by the HPCI or RCIC systems by the tank’s standpipe design. Not all of this 150,000 gal is usable since the suction is switched to the suppression pool automatically upon a condensate storage tank low level (equivalent to about 45,000 gal of water in the tank based on a nominal trip setpoint). However, while the plant is operating, the condensate storage tank is maintained at a normal level considerably higher than that required to provide 150,000 usable gallons. The HPCI suction is also switched to the suppression pool upon suppression pool high level signal (approximately 3.5

in. above normal water level) or at any time manually. The suppression pool contains approximately 880,000 gal of water. The core spray and LPCI systems take suction from the suppression pool.

The residual heat removal service water (RHRSW) system serves as the ultimate heat sink. Its design includes two 3,465,000-gal reservoirs (described in detail in Section 9.2). They are sized in accordance with the recommendations of Regulatory Guide 1.27.

#### 6.3.2.7 Pump Characteristics

The HPCI pump is driven by a high-pressure turbine fed by reactor steam. The rated horsepower of the turbine at high speed (4100 rpm) is 4750 hp. The turbine produces 1000 hp at low speed (2100 rpm).

The core spray pump is driven by an open, drip-proof, induction motor rated at 800 hp. Power required is 102 amp at 4160 V ac.

The RHR pumps are each driven by a type-K, squirrel cage, induction motor rated at 2000 hp. Power required is 255 amp at 4l60 V ac.

#### 6.3.2.8 Heat Exchanger Characteristics

There are no heat exchangers in the closed cooling water path associated with the emergency core cooling subsystems. The heat exchangers in the RHR system are discussed in Section 5.5.

#### 6.3.2.9 Emergency Core Cooling System Flow Diagram

A schematic diagram, the flow rates, and the pressures of the various ECCS subsystems can be obtained from the information blocks on the following process diagrams: Figures 6.3-1 through 6.3-11, and Figures 6.3-14 through 6.3-16. These parameters are presented for several modes of operation, including LOCA and test conditions.

#### 6.3.2.10 Relief Valves and Vents

The RHR/LPCI and the core spray systems are not designed to withstand normal reactor system pressures. Relief valves are provided to protect the low-pressure portions of these subsystems against possible overpressurization due to valve leakage and pump heat input. Pressurized portions of the HPCI system are designed for service at reactor pressure and therefore do not require overpressure protection.

The design basis for the relief valves to protect the core spray and RHR systems from overpressurization is given in Subsection 5.5.13.

#### 6.3.2.11 System Reliability

The ECCS reliability has been achieved through an evolutionary process. Initially a proposed system configuration was submitted for evaluation. A reliability model of the proposed system was constructed and an estimate of the system success probability was made. Reliability models were then constructed for alternative ECCS configurations and a comparative trade-off study yielded the most reliable system configuration. Upon completion of the final design, a formal reliability analysis was performed to

a. Determine the expected system availability (average reliability)

b. Set safe system test intervals and allowable repair times

c. Qualitatively evaluate the system for conformance to the original design concepts, as well as existing industry standards and criteria for reactor protection and safety systems.

System availability is evaluated and selected test intervals and allowable repair times were determined by well-established reliability/availability methods. The qualitative analysis includes a functional system failure modes and effects analysis (FMEA). The FMEA results are used to verify conformance to industry criteria, develop reliability models, and ensure that the original design redundancy and diversity have been retained.

Availability, as applied to the ECCS, is defined as the probability that the system is operable when required. The ECCS availability is a function of the component system test intervals and the failure rates of the component parts used in the systems. The component parts used in the ECCS have low failure rates, as evidenced by historical field operating experience. The ECCS availability required to ensure adequate plant safety is established as a system design requirement. To ensure adherence to the availability design requirement, the periodic test intervals and allowable repair times for inoperable systems are defined in the Technical Specifications.

The power sources required for successful system operation are arranged in redundant configurations such that the power availability is not a limiting factor in determining the overall system success probability.

### 6.3.2.12 Protection Provisions

The ECCS piping and components are designed to accommodate the effects of movement, missiles, thermal stresses, the effects of the LOCA, and the safe-shutdown earthquake (SSE).

The reactor coolant pressure boundary (RCPB) has been analyzed for four categories of design transients: normal, upset, emergency, and faulted conditions. These categories are generally described in the ASME B&PV Code Section III, 1968 Edition. Subsection 5.2.1 contains details of this analysis.

Protection of the mechanical, instrumentation, and electrical portions of the engineered safety feature (ESF) system and reactor protection systems (RPS) against environmental conditions is discussed in Section 3.11.

Subsection 6.3.2.2.5 describes the features protecting against water-hammer effects in ECCS discharge lines.

The components of the core spray system, the HPCI, the LPCI, and the RHR systems are protected from becoming functionally inoperative as a result of flooding the lower levels of the reactor building due to excessive leakage from the ECCS complex. Section 3.4 describes design protection against flooding.

Response 3.1.2, Amendment 11 of the Fermi 2 PSAR discusses thermal stresses generated by high-temperature and high-pressure jet streams impinging on spherical plate sections. Also presented is an analysis of the uplift force on the RPV associated with a main steam line break. The methods used to provide assurance that thermal stresses do not cause damage to the ECCS are described in Subsection 6.3.3.9.

The ECCS is protected against the effects of the pipe whip, which might result from piping failures up to and including the LOCA, by separation barriers, pipe-whip restraints, or energy-absorbing materials. One or more of these three methods have been applied to provide protection against cascading damage to piping and components of the ECCS that could otherwise result in a reduction of ECCS effectiveness to an unacceptable level. Section 3.6 describes the design protection and analysis performed for pipe whip.

The ECCS piping and components located outside the containment are protected from internally and externally generated missiles by the reinforced-concrete structure of the ECCS pump rooms. In addition, the watertight construction of the ECCS pump rooms below grade level protects against damage by flooding. Section 3.5 describes design protection against postulated missile damage.

#### 6.3.2.13 Provisions for Performance Testing

a. High pressure coolant injection system

1. A full-flow test line is provided to route water from and to the condensate storage tank without entering the RPV
2. Instrumentation is provided to indicate system performance during normal and test operations
3. All motor-operated and air operated valves are capable of manual operation, either local or remote for test purposes
4. Drains are provided to leak test the major system valves.

b. Core spray system

1. A full-flow test line is provided to route water from and to the suppression pool without entering the RPV
2. In the event the torus is unavailable to provide suction, a test line from the condensate storage tank provides reactor grade water to test pump discharge to simulate injection into the RPV. Direct injection to the vessel is not performed (Subsection 6.3.2.2.3.2)
3. Instrumentation is provided to indicate system performance during normal and test operations
4. All motor-operated valves, air-operated valves, and check valves are capable of manual operation for test purposes
5. Relief valves are removable for bench testing.

c. Low pressure coolant injection system

1. Discharge test lines are provided for the four pumps to route suppression pool water back to the suppression pool without entering the RPV
2. Instrumentation is provided to indicate system performance during normal and test operations
3. All motor-operated valves, air-operated valves, and check valves are capable of manual operation for test purposes
4. Shutdown cooling lines taking suction from the recirculation system are provided to allow testing of pump discharge into the RPV during normal plant shutdown
5. All relief valves are removable for bench testing.

d. Automatic depressurization system

Actual operation of each safety/relief valve pilot valve associated with the ADS was verified during the Preoperational Test Program.

### 6.3.2.14 Net Positive Suction Head

The RHR/LPCI and core spray pump systems are designed to ensure adequate net positive suction head (NPSH) margin availability under all combinations of foreseeable adverse conditions. The point of minimum margin for all pumps occurs at the peak suppression pool temperature, calculated on the basis of conservative assumptions. No dependence is placed upon positive containment pressure. The Regulatory Position stated in Regulatory Guide 1.1, dated November 2, 1970, is met.

The conditions assumed for calculating the peak suppression pool temperature and the available NPSH margin are as follows:

a. Reactor at 3499 MWt (102% of 3430 MWt)

b. Suppression pool volume is 117,161 $ft^3$

c. Initial suppression pool water temperature 95°F

d. Temperature of RHRSW reservoir varies linearly from 80°F to 90°F over 8 hours and stabilizes at 90°F

e. Pump suction strainers plugged to maximum design per UFSAR Section A.1.82

f. All of the energy in the RPV is absorbed by the suppression pool water following a LOCA

g. Pumps operating at rated flows

h. The heat loads considered were pump heating; Zr-$H_2O$ reaction; peak sensible heat in RPV, steam, all water in the feedwater system, recirculation loops, fuel; and decay heat

i. The primary containment long-term response to a recirculation line break LOCA scenario in Subsection 6.2.1.3.3

j. Bounding minimum RHR heat exchanger heat transfer coefficient is 366 BTU/sec-°F

k. RHR system is placed in the suppression pool cooling mode 20 minutes after LOCA

The analysis yielded a peak suppression pool temperature of 196.5°F which occurs approximately 5 hours after LOCA. This temperature is less than the suppression pool temperature of 198°F used in the RHR and the core spray NPSH margin calculations described below. The NPSH margins for the RHR and core spray pump systems using a peak suppression pool temperature of 198°F and other conservative calculational methods are positive, even allowing for instrument tolerances to cause flow to be above nominal design values. The 198°F pool temperature for NPSH is the controlling limit for the bulk pool temperature.

A hydraulic analysis has been performed for each RHR operating mode shown in the General Electric process diagram (Figure 6.3-14, Sheet 2). The NPSH required was obtained by using the most conservative RHR pump performance curve and corresponds to the flow rate through the pump. In calculating the NPSH available for each mode, the design conditions on the process diagram were used and the suction strainers were assumed to be plugged to maximum design per UFSAR Section A.1.82. The results of the RHR hydraulic analysis are as follows, assuming pumps with maximum allowed degradation:

| Mode | Description | GE Nominal Design Flow per Pump (gpm) | NPSH Required (ft absolute) | NPSH Available (ft absolute) |
|---|---|---|---|---|
| A | LPCI accident, three pumps, RPV pressure equals 20 psig | 10,000 | 15 | 30 |
| B | LPCI accident, two pumps, RPV pressure equals 20 psig | 13,000 | 17 | 27 |
| C2 | Containment spray, one pump, RPV pressure equals 0 psig | 10,000 | 12.8 | 13.9* |
| D2 | Suppression pool Cooling, one pump | 10,000 | 12.8 | 31<br>14* |
| E | Shutdown cooling, two pumps, RPV pressure equals 97 psig | 9,625 | 13 | 49 |

| | | | | |
|---|---|---|---|---|
| F | Shutdown cooling, two pumps, RPV pressure equals 0 psig | 9,750 | 13 | 76 |
| G | LPCI accident, two pumps, RPV pressure equals 0 psig | 13,000 | 17 | 19** |
| H | LPCI test, one pump | 10,300 | 13 | 42 |
| J2, J4 | Minimum flow bypass, two pumps | 480 | 8 | 44 |

* Suppression pool at 198°F

** Margin consistent with GE process diagram fluid temperature. Long-term operation at the higher suppression pool design maximum temperature of 198°F requires throttling flow consistent with EOP procedures consistent with UFSAR Section 6.3.2.17.

From the above cases, it is apparent that Modes C2 and G have the lowest NPSH margin. Since NPSH margin is typically reduced at higher flow rates, Modes C2 and G were also examined using LPCI pumps operating on the vendor supplied pump curve (not degraded) and with 2% over-frequency applied. Under this condition, while NPSH margin is reduced to less than 1 ft for the worst case pump, the margin remains positive without containment overpressure.

The RHR pump suction strainers are 11-gage perforated plate stacked discs with 1/8-in.-diameter holes. Therefore, the largest particles that could pass through the strainers are 1/8 in. in diameter.

After a postulated LOCA, debris released to the suppression pool would tend to sink to the bottom of the pool. The accumulation of this debris on the RHR pump suction strainers is minimized because the strainer is located at a 45° angle above the bottom of the pool, with a 10° mitre bend between the suction flange and the strainer flange. Any particles that could pass through the strainer perforations are not of sufficient size to affect RHR pump suction flow.

Local heating in the suppression pool is a phenomenon that occurs when high-energy steam is released into the suppression pool for quenching. The most severe local heating occurs during a safety/relief valve discharge. In general, local-to-bulk temperature differences at the time of maximum temperatures are about 15° for cases where two RHR loops are assumed available and about 30° for cases where one RHR loop is assumed available. The pool temperature during normal plant conditions is limited by Technical Specifications so that localized heating from safety/ relief valve discharges will not nullify the NPSH of the ECCS pumps, even for prolonged operation of the valves.

Local heating also occurs during the RPV depressurization stage of a postulated LOCA when steam and noncondensable gases are blown through the downcomers into the suppression pool. Local heating during this event will be significantly less than the temperature achieved during safety/relief valve blowdown because of lower energy in the steam and high turbulence in the water.

As described in Subsection 6.2.1.3.2, at about 100 sec after the design-basis accident (DBA), hot water only will be discharged out of the break; at 1200 sec, the suppression pool temperature has reached 168°F. Thus, up to this point, there would be a very large margin on NPSH even if local heating were significant. At the time the suppression pool reaches its peak temperature, approximately 5 hr after LOCA, local heating cannot be significant for the small $\Delta T$ in the water being discharged is quickly blended with the pool water.

The preceding conditions describe the containment system after full blowdown following a large break. Consequently, they are not applicable to the HPCI system. The HPCI pump is located below the level of the condensate storage tank (CST), from which suction is normally taken, and below the water level in the suppression pool.

A low-water level in the condensate storage tank (2 ft 8 in. above the bottom of the tank) or a high-water level in the suppression pool (3.5 in. above normal level), would cause the two normally closed, motor-operated gate valves located in the suppression pool suction line to automatically open. The normally open, motor-operated gate valve located in the condensate storage tank suction line would remain open until the two suppression pool suction valves fully open. At that time, the condensate storage tank suction valve would close. In this way, NPSH is maintained.

In the case of an ECCS passive failure such as pump seals or valve seals, the operator has adequate time to take corrective action and isolate the failure before the NPSH would become inadequate for the remaining ECCS pumps due to reduced suppression pool level.

Adequate minimum submergence is available to prevent vortex formation and air ingestion during operation. With the exception of suction from the CST, minimum required submergence is computed in accordance with NUREG/CR-2772 as endorsed in Reg. Guide 1.82 Rev. 2. Under the RHR RUNOUT scenario, available submergence is sufficient to ensure minimal air entrainment and still meet NPSH requirements under Regulatory Guide 1.82, Rev. 2 for the minimum required compliment of pumps. The predicted CST suction minimum submergence is acceptable based on analytical analysis and the use of mechanical vortex suppression assemblies. The analytical method establishes the minimum submergence for straight (non-circular) spill over flow into the suction piping in accordance with Reference 2. For circular flow or vortex flow considerations, mechanical vortex suppression assemblies are installed that eliminate flow vortexes from introducing the potential for air entrainment into the pump suction. The pertinent data are summarized below:

| System | Suction Source | Available Submergence (ft) | Predicted Submergence for Incipient Air Ingestion (ft) |
|---|---|---|---|
| RHR | Suppression pool | 8.7 | 7.35 (LPCI)<br>9.0 (LPCI RUNOUT)† |
| Core spray | Suppression pool | 9.2 | 3.9 |
| HPCI | Suppression pool | 8.2 | 1.2 |
| HPCI | Condensate storage tank | 1.45 (including 0.44 foot silt block) | 0.98 |

† Under the limiting RUNOUT condition, the minimum submergence of the most limited of four available RHR pumps is not sufficient to ensure zero air ingestion. Here, the LPCI function is not credited to meet 10CFR50.46 ECCS performance requirements. With consideration of the standard applied NPSH penalties one or more RHR pumps have sufficient NPSH. This ensures RHR is available for subsequent use to provide long-term containment cooling in suppression pool cooling mode.

A postulated minimum level of 14 in. below the torus centerline was used to determine the level of suppression pool submergence.

The precaution taken to preclude vortex formation in the HPCI-RCIC condensate storage tank suction is to transfer suction to the suppression pool on low tank level. This is supplemented by the installation of a vortex suppression assembly over the suction intake for the HPCI-RCIC systems to mechanically preclude vortex formation.

#### 6.3.2.15 Residual Heat Removal Pump Runout Evaluation

The Fermi 2 design was reexamined also to determine whether a failure in the LPCI logic could disable the RHR pumps. The single failures that potentially could disable the RHR pumps have been identified as the following.

Case A The LPCI logic correctly selects the unbroken loop, but a single failure causes inadvertent opening of LPCI injection valve E11-F015 into the broken loop. This condition results in four RHR pumps injecting into both recirculation loops with one loop broken

Case B All four RHR pumps start for LPCI injection, but a single failure causes the LPCI loop selection logic to select the wrong (broken) loop. This condition results in all four RHR pumps injecting into the broken recirculation loop

Case C The LPCI logic correctly selects the unbroken loop, but a single failure causes the recirculation pump discharge valve B31-F031 to remain unclosed. This condition results in four RHR pumps injecting into the vessel through one recirculation loop inlet and discharge lines.

Under the above conditions, the RHR pump operation was examined for cavitation, pump motor overload, and emergency diesel generator overload.

#### 6.3.2.15.1 Analysis Assumptions

As shown in Figure 6.2-13, the initial LOCA blowdown causes an almost immediate temperature increase to approximately 135°F in the torus with a continued increase to 168°F after 20 minutes. A water temperature of 168°F was assumed for the entire time period and for each of the cases (A, B, and C) listed above.

As described in Subsection 6.3.2.1, after 10 minutes the operator begins the post-LOCA manual control of the RHR system, which includes throttling the RHR system and initiating containment cooling. However, the operator can delay the containment cooling for up to 20 minutes. Therefore, it is assumed that the RHR pump runout condition occurs during the 0- to 20-minute part of the DBA.

Although the LOCA blowdown will cause a pressure increase in the primary containment, the drywell and torus pressure assumed for the analysis is 0 psig.

Reactor vessel pressure was determined from the LPCI process diagram Figures 6.3-14 through 6.3-16. A vessel pressure of 20 psig given in mode A was assumed, because the runout condition lasts only during the short term portion of the LOCA analysis. The RHR suction strainers were assumed to be plugged.

For Case B, no jet pump resistance is available as the broken injection loop bypasses the normal injection path through the jet pumps. For Cases A and C, an equivalent jet pump resistance value was determined and used in the analysis.

The reactor core level was assumed to flood to two-thirds of the core height.

The Technical Specifications allow only 7 days of continuous plant operation with an inoperable LPCI pump. Therefore, the analysis did not consider any pumps to be out of service for maintenance.

#### 6.3.2.15.2 Calculation Procedure and Results

From the description of cases A, B, and C, it is clear that case A is the limiting case. This is because case A allows all four pumps to pump into parallel paths consisting of both the normal injection path and to the broken loop. Case A bounds case B, since case B simply eliminates the parallel path to the desired injection line. Case C is the least limiting. Like case B, case C involves only one flow path, but it would inject against a greater residual pressure in the reactor and intact recirculation loop piping. Therefore, only case A is analyzed in detail. The vendor-certified RHR pump performance curves provided a record of pump performance test data up to a pumping condition of 14,000 gpm. Operating conditions beyond this point were assessed using extrapolated vendor pump performance curves and data from the preoperational testing for RHR pumps. The results for each case are discussed below.

Case A

Case A results in four RHR pumps pumping into both LPCI loops, with one loop broken. Under the most limiting scenario for minimum overall available NPSH, the RHR pumps are required to operate at approximately 15,500 gpm. The analysis of the available NPSH margins for this condition determines that adequate margins are available for torus water temperatures less than 168°F consistent with the time available for operators to take action to establish containment cooling within 20 minutes in accordance with the plant design as described in Section 6.3.2.14. For the Case A pump operating condition, pump motor and emergency diesel generator overloads will not be experienced.

6.3.2.15.3 Conclusion

A failure of the LPCI logic will not result in RHR pump operating conditions that would allow pump cavitation, pump motor overload, or emergency diesel generator overload. The limiting case failure does allow the pumps to operate at a point that is not part of the manufacturer's performance test data. Extrapolated performance data were used as a basis for the analysis conclusion. The extrapolated data has been confirmed by expanding the scope of the preoperational testing for the RHR pumps to provide performance data for the 14,000-gpm to 15,300-gpm range. No design changes to the RHR system are required; therefore, the performance of the RHR system, with respect to process diagram requirements, has not changed and there is no impact on the Appendix K analysis discussed in Subsection 6.3.3.

6.3.2.16 Motor-Operated Valves and Controls

The LPCI and the core spray systems are not designed to withstand reactor system pressures. Provisions have therefore been made to ensure that these ESF systems are not subjected to damaging pressures. These provisions include appropriate relief valves, discussed in Subsection 6.3.2.10, and isolation valves with system interlocks and alarms that are discussed below. Refer to Section 7.3 for a further discussion of controls for these valves. The LPCI/RHR system is isolated from the reactor system by the following:

| Valves | Line Isolation | Isolation Signal |
|---|---|---|
| F008 | RHR pump suction | Pressure trip unit B31-N611B signals “close” above permissive pressure |
| F009 | RHR pump suction | Pressure trip unit B31-N611A signals “close” above permissive pressure |
| F022 | RPV head spray (RHR pump discharge) | Pressure trip unit B31-N611A signals “close” above permissive pressure |
| F015A, B | RHR pump discharge to RPV | B21-N690A, B and logic signal prevent opening above interlock pressure |
| F050A, B | RHR pump discharge to RPV | Check valve (air operator on valve for testing) |

If either trip unit B31-N611A or B31-N611B fails in a nonconservative manner, the system is still protected because the other unit sends a "close" signal to the valve in series with the valve controlled by the failed trip unit. In the event of valve leaks, the system is protected by the pressure relief valves as outlined in Subsection 6.3.2.10.

Provisions have been made to allow for thermal expansion of water trapped between valves F008 and F009 (penetration X-12), by way of a line that returns the trapped water to the RPV.

The core spray system is isolated from the reactor system by the following

| Valves | Line Isolation | Logic |
|---|---|---|
| F005A, B | Core spray pump discharge | Manual control room pushbutton to close |
| F006A, B | Core spray pump discharge | Check valve |

Check valves are backed up by normally closed motor-operated valves. In the event of operator failure and check valve leak, the system is protected by relief valves as outlined in Subsection 6.3.2.10.

All motor-operated ECCS valves have position indication in the control room.

#### 6.3.2.17 Manual Actions

With the exception of LPCI while RHR is in the shutdown cooling mode, the initiation of the ECCS is completely automatic. No operator action is required for the initiation of postaccident modes of operation. When RHR is lined up in the shutdown cooling mode and RPV pressure is less than or equal to the cut in pressure, manual operation is required to permit LPCI to align and initiate. This includes manually lining up the suction path from the torus for the loop which is in shutdown cooling. No manual valve is required to change position to accomplish a safety-related mode of any ECCS. Manual valves generally do not have position indication in the control room. Administrative procedures require the position of any critical manual valve to be verified and recorded after each time it is operated and the position of critical manual valves to be verified and recorded during a refuel outage prior to plant operation following refueling. Thus, all critical manual valves are under rigid administrative control.

Following is a list of manual valves critical to the operation of ECCS that are controlled by administrative procedures:

| Safety System | Valve Number(s) |
|---|---|
| HPCI system | P1100-F042 |
| LPCI mode of RHR | E1100-F034 A (B, C, D) |
| Core Spray | E2100-F001 A (B, C, D)<br>E2100-F037 A (B, C, D) |

Operators are instructed and trained to observe the values and rates of change of the plant parameters that have the greatest significance for plant safety (e.g., RPV water level, containment pressure, torus temperature, radiation monitors, operation of ECCS, standby gas treatment system, emergency diesel generator loads, etc.). From these parameters, together with his training and use of the symptom-oriented emergency operating procedures, the operator is able to logically evaluate the condition of the plant and is prepared to take appropriate action at the end of the initial interval.

A timer is used in each ADS logic. The time-delay setting before actuation of the ADS is long enough that the HPCI system has time to operate, yet not so long that the LPCI and core spray systems are unable to adequately cool the fuel if the HPCI system fails to start. Manual reset circuits are provided for the ADS initiation signal. By resetting this signal manually, the delay timers are recycled. The operator can use the reset pushbuttons to delay or prevent automatic opening of the relief valves if such delay or prevention is necessary.

A manual inhibit switch is also provided for each ADS trip system. These switches allow the operator to inhibit ADS operation without repeatedly pressing the reset pushbuttons. Operation of the manual inhibit switch will activate a white indicating light and an annunciator to alert the operator of the inhibit action. Enabling the inhibit function will not terminate an ADS logic actuation after the 120 second time delay has elapsed. At this point, only the reset pushbutton can be used to affect the ADS operation. Guidance is contained in the Emergency Operating Procedures.

6.3.2.18 Process Instrumentation

Sufficient instrumentation is available to the operator in the main control room to assist him in accurately assessing the post-LOCA conditions if LOCA should occur. Basically, these indications are of two varieties: those that indicate the pressures, temperatures, and levels in the RPV and containment, and those that provide indication of operations of the ECCS position of valves and circuit breakers, flows, temperatures, and pressures of ECCS.

The most significant instruments in the first category are

a. Reactor pressure vessel level
b. Reactor pressure vessel pressure
c. Containment pressure
d. Containment temperature
e. Suppression pool level
f. Suppression pool temperature.

The most significant instruments in the second category are as follows.

a. LPCI flow and pressure
b. Core spray flow and pressure
c. HPCI flow and pressure.

Other available instrumentation is listed on the piping and instrumentation diagram included with the description of the above system in Chapter 5. Discussion of instrumentation also appears in Chapter 7. See Subsection 7.5.1.4.2 for a detailed listing of the process information available in the main control room that permits accurate assessment of postaccident conditions.

6.3.2.19 Materials

Materials used in or on the ECCS are reviewed and evaluated with regard to radiolytic and pyrolytic decomposition and attendant effects on safe operation of the ECCS. For example, fluorocarbon plastic (Teflon) is not permitted in environments that attain temperatures greater than 300°F or radiation exposures above $10^4$ rads.

Organic materials used in the Fermi 2 primary and secondary containments have been selected for extended life during normal operations for their resistance to expected accident environmental conditions. Thermal insulation used is inorganic and is not sensitive to high radiation fields, steam, or high temperature.

Evaluations of the protective coatings used within the containment (Subsection 6.2.1.6) have been made. It has been determined that they will satisfactorily endure accident environmental conditions and their expected products of decomposition, if any, will not adversely affect the operability of any ESF system.

6.3.2.20 Maintenance and Operability

The capability of the ECCS to provide core cooling is verified by regularly scheduled functional tests on each component and system. Subsection 6.3.4 discusses these tests and the testing program.

The configuration of the ECCS systems has placed most of the components in concrete cubicles so that maintenance on any component has a minimum of complications due to radiation from the primary system or from other components (see Figures 1.2-6 and 1.2-8). Drains for all pumps, heat exchangers, and low points in piping runs are piped directly to radwaste collection points. Flushing and makeup are provided from the demineralized water or CSTs.

Because of these features, maintenance of ECCS components during a long-term LOCA mode of operation may be possible depending upon which component has failed. However, special facilities for this situation have not been provided, since the system designs inherently account for component failures without overall loss of intended function (usually by redundancy of systems, see Subsection 3.12.2.2). In addition, the following design provisions have been included to increase system operational reliability during a LOCA:

a. All components essential to ECCS operation are capable of continued operation under LOCA conditions of pressure, temperature, and radiation

b. Suction strainers have been provided on all ECCS pumps to prevent pump seizure due to entrained foreign particulates

c. Adequate fouling factors have been included in the determination of the design heat transfer capacities of the RHR heat exchangers.

The core spray and RHR pumps and motors are designed for the operating life of the plant (40 years) and for a postulated single continuous operation of 100 days for an accident during that 40-year operating life.

The following table shows the maximum expected accumulated operating time of these pumps for the life of the plant (40 years):

| Mode of Operation | RHR (hr) | Core Spray (hr) |
|---|---|---|
| In-shop testing | 4 | 4 |
| Preoperational testing | 168 | 168 |
| Monthly testing | 480 | 480 |
| Yearly testing | 40 | 40 |
| Post LOCA | 2,400 | 2,400 |
| Shutdown | 28,800 | N/A |
| Total | 31,892 | 3,092 |

The severe operating conditions to which the HPCI pumps are exposed are temperatures to 212°F, radiation, and dynamic loads from seismic and hydrodynamic effects. The pumps are mainly fabricated of metallic materials that will not be degraded by the temperature and radiation environment. The nonmetallic gaskets and seals are made of materials with a demonstrated resistance to the environment. The dynamic load inputs are addressed analytically and evaluated against appropriate criteria to ensure operation of the pump while undergoing dynamic loading. The above ensures that the expected service life will exceed the expected operating time of 500 hr. (Surveillance tests are performed once a month for 40 years equaling 480 tests plus a possible 20 real starts equaling 500 operating hours.)

CS pumps are analyzed for the effects of dynamic loads resulting from seismic and hydrodynamic effects. Operability under the worst loadings is ensured by the operability assurance program described in Section 3.9.4.3.

### 6.3.3 Emergency Core Cooling System Performance Evaluation

The performance of the ECCS was determined originally by applying the 10 CFR 50, Appendix K, evaluation models and then showing conformance to the acceptance criteria of 10 CFR 50, Section 50.46. Reference 3 provided a complete description of the LOCA events and the methods used to perform the original calculations.

The original methodology was updated (Reference 4) for the power uprate program (Reference 5) for GE11 (Reference 19), and for the GE14 fuel introduction (Reference 16). The LOCA analysis was then revised using the SAFER/PRIME-LOCA analytical model and methodology (Reference 39). Then the TRACG-LOCA evaluation model replaced the SAFER/PRIME-LOCA for the ECCS-LOCA analysis for the GNF3 fuel introduction (Reference 42). The updated methodology and a description of the LOCAs are summarized here.

The ECCS performance is evaluated for the entire spectrum of break sizes for postulated LOCAs. The discussion includes information on the radiological consequences of the following events:

a. Feedwater piping break, Subsection 15.6.6

b. Spectrum of BWR steam system piping failures outside containment, Subsection 15.6.4

c. Loss-of-coolant accidents, Subsection 15.6.5.

Cycle-specific reload information is in Reference 15.

### 6.3.3.1 Emergency Core Cooling System Bases for Technical Specifications

The maximum average planar linear heat generation rates (MAPLHGR) calculated in this performance analysis provide the basis for Technical Specifications designed to ensure conformance with the acceptance criteria of 10 CFR 50, Section 50.46. Minimum ECCS functional requirements are specified in Subsections 6.3.3.4 and 6.3.3.5; testing requirements are discussed in Subsection 6.3.4. Limits on minimum suppression pool water level are discussed in Section 6.2.

The plant is licensed for average power range monitor (APRM) rod block monitor (RBM) Technical Specification (ARTS) improvement program (Reference 6, 7, 8, and 20) and has both power and flow dependent limits imposed on the operating limit MAPLHGR (Reference 8 and 20). The flow dependent MAPLHGR, $MAPLHGR_f$, is determined from the product of the standard MAPLHGR and a flow dependent term, $MAPFAC_f$, which is defined as a function of the core flow rate and positioning of the scoop tube on the recirculation pump motor. The plant specific $MAPFAC_f$ versus flow curve is shown in the Core Operating Limits Report (COLR).

The power dependent operating limit MAPLHGR, $MAPLHGR_p$, is determined from the product of the standard MAPLHGR and the power dependent term, $MAPFAC_p$. For powers between 25 percent rated and the bypass point for the turbine stop valve/turbine control valve fast closure scram signal (29.5 percent rated), there are two values for $MAPFAC_p$, one for core flows >50 percent rated and one for core flows ≤50 percent rated, as shown in the COLR. Once the power exceeds this bypass point, the $MAPFAC_p$ is determined from a single curve which must be multiplied by the standard MAPLHGR to produce the reduced power operating limit MAPLHGR, $MAPLHGR_p$.

The operating limit MAPLHGR to be used becomes the most limiting value of either $MAPLHGR_f$ or $MAPLHGR_p$.

### 6.3.3.2 Acceptance Criteria for Emergency Core Cooling System Performance

The applicable acceptance criteria, quoted from 10 CFR 50, Section 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light-Water-Cooled Nuclear Power Reactors," are listed here. A detailed description of the methods used to show compliance are in Subsection 6.3.3.7 and Reference 1.

a. <u>Criterion 1, peak cladding temperature</u> - "The calculated maximum fuel element cladding temperature shall not exceed 2200°F." Conformance to Criterion 1 is shown in Table 6.3-4

b. <u>Criterion 2, maximum cladding oxidation</u> - "The calculated total local oxidation of the cladding shall nowhere exceed 0.17 times the total cladding thickness before oxidation." Conformance to Criterion 2 is shown in Table 6.3-4

c. <u>Criterion 3, maximum hydrogen generation</u> - "The calculated total amount of hydrogen generated from the chemical reaction of the cladding with water or steam shall not exceed 0.01 times the hypothetical amount that would be generated if all the metal in the cladding cylinder surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react." Conformance to Criterion 3 is shown in Table 6.3-4

d. <u>Criterion 4, coolable geometry</u> - "Calculated changes in core geometry shall be such that the core remains amenable to cooling." As described in Reference 1, conformance to Criterion 4 is demonstrated by conformance to Criteria 1 and 2

e. <u>Criterion 5, long-term cooling</u> - "After any calculated successful initial operation of the ECCS, the calculated core temperature shall be maintained at an acceptably low value and decay heat shall be removed for the extended period of time required by the long-lived radioactivity remaining in the core." Conformance to Criterion 5 is demonstrated in Reference 9. Briefly summarized, the core remains covered to at least the jet pump suction elevation, and the uncovered region is cooled by spray cooling and/or by steam generated in the covered part of the core.

### 6.3.3.3 <u>Single-Failure Considerations</u>

The functional consequences of potential operator errors, single failures, and the potential for submerging valve motors in the ECCS are discussed in Subsection 6.3.2. This Subsection includes information on errors that could cause any manually controlled, electrically operated valve in the ECCS to move to a position that could adversely affect the ECCS. There it was shown that all potential single failures are no more severe than one of the single failures identified in Table 6.3-5.

It is therefore only necessary to consider each of these single failures in the emergency-core-cooling-system performance analyses.

The specific analysis (Reference 42) included break sizes ranging from the minimum size that meets the definition of a LOCA to 200% of the largest applicable pipe cross-sectional area. Different single failure assumptions were investigated in order to identify the limiting case. Non-recirculation line breaks were found to be non-limiting. The feedwater line break accident analysis assumes operator actions are required to depressurize the reactor during a Division I battery failure. This assumption was reviewed and accepted by the NRC per the Ref. 24 SER.

The TRACG LOCA analysis (Reference 42) indicates that the small recirculation line breaks with Division II DC Power Source (Div II Battery) failure are limiting. This analysis was performed at maximum core thermal power, including uncertainty allowance (see Table 6.3-6, Section A Plant Parameters).

6.3.3.4 System Performance During the Accident

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

a. Receiving an initiation signal

b. A small lag time (to open all valves and have the pumps up to rated speed)

c. Finally the ECCS flow entering the vessel.

Key ECCS actuation setpoints and time delays for all the ECCSs are provided in Table 6.3-6. The minimization of the delay from the receipt of signal until the ECCS pumps have reached rated speed is limited by the physical constraints on accelerating the diesel generators and pumps. The delay time resulting from valve motion in the case of a high-pressure system provides a suitably conservative allowance for valves available for this application. In the case of the low-pressure system, the time delay for valve motion is such that the pumps are at rated speed before the vessel pressure reaches the pump shutoff pressure.

Simplified piping and instrumentation and functional control diagrams for the ECCS are provided in Subsection 6.3.2. The operational sequence of ECCS for the DBA is shown in Table 6.3-7.

Operator action is not required, except as a monitoring function and as noted in Section 6.3.3.3, during the short-term cooling period following a LOCA. During the long-term cooling period, the operator will take action as specified in Subsection 6.2.2.3 to place the containment cooling system into operation.

6.3.3.5 Use of Dual Function Components for Emergency Core Cooling System

With the exception of the LPCI system, the systems of the ECCS are designed to accomplish only one function: to cool the reactor core following a loss of reactor coolant. To this extent, components or portions of these systems (except for pressure relief) are not required for operation of other systems that have emergency core-cooling functions, or vice versa. Because either the ADS initiating signal or the overpressure signal opens the safety/relief valve, no conflict exists.

The LPCI subsystem, however, uses the RHR pumps and some of the RHR valves and piping. When the reactor water level is low, the LPCI subsystem has priority through the valve control logic over the other RHR subsystems for containment cooling. When RHR is lined up in the shutdown cooling mode and RPV pressure is less than or equal to the cut in pressure, manual operator action is required for LPCI injection. Immediately following a LOCA, the RHR system is directed to the LPCI mode.

6.3.3.6 Limits on Emergency Core Cooling System Parameters

The limits on the ECCS parameters are discussed in Subsection 6.3.3.1 and Subsection 6.3.3.7.1.

Any number of components in any given system may be out of service, up to and including the entire system. The maximum allowable out-of-service time is a function of the level of redundancy and the specified test intervals. The limiting conditions for operation and surveillance requirements are given in the Technical Specifications.

### 6.3.3.7 Emergency Core Cooling System Analyses for Loss-of- Coolant Accident

#### 6.3.3.7.1 Loss-of-Coolant Accident Analysis Procedures and Input Variables

The procedures approved for LOCA analysis conformance calculations were originally performed and approved in accordance with the methodology described in Reference 3. This methodology has been updated in accordance with the procedures in detail in Reference 40, commonly referred to as SAFER/PRIME methodology. The SAFER/PRIME methodology has been replaced with the procedures in Reference 41, commonly referred to as TRACG-LOCA, for GNF3 and GE14 fuel types. The new methodology, which is an ECCS evaluation model developed to analyze BWR LOCA in accordance with 10 CFR 50.46, is a best estimate plus uncertainty type of evaluation model. Potentially limiting break locations, initial conditions, and ECCS performance are determined using inputs that correspond to the "nominal" trial associated with the statistical analysis in the break spectrum calculations. Statistical analyses are performed for at least the most limiting small break, intermediate break, and double-ended guillotine break (DEGB).

Two primary computer models were used to determine the LOCA response for plant Fermi 2 using the TRACG-LOCA method. These models are PRIME and TRACG, which are described below.

a. DELETED

b. DELETED

c. PRIME

   The PRIME model provides the parameters to initialize the fuel rod fission gas inventory and rod internal pressure at the onset of a postulated LOCA for input to TRACG. PRIME also provides the initial pellet-cladding gap conductance and other parameters used by TRACG to calculate the transient gap conductance.

d. TRACG

   TRACG calculates the system response of the reactor and the detailed fuel rod heat transfer over a complete spectrum of hypothetical break sizes and locations. TRACG is compatible with the PRIME fuel rod model for gap conductance and fission gas release. A simplified form of the PRIME fuel thermal conductivity model is built into TRACG. TRACG calculates the core and vessel water levels, system pressure response, ECCS performance, and other thermal-hydraulic phenomena occurring in the reactor as a function of time. TRACG conservatively models the sources of heat in the core such as fission power, decay heat, and metal-water reaction. TRACG realistically models all regimes of heat transfer to calculate the transient cladding temperatures and oxidation.

The significant input variables used by the LOCA codes are listed in Table 6.3-6.

6.3.3.7.2 Accident Description

A detailed description of the LOCA calculation is provided in Reference 40 and is supplemented by Reference 42. With the TRACG-LOCA methodology, the limiting break is the limiting small recirculation suction line break. The limiting single failure is the one which results in the highest PCT. This is the failure of the Division II DC power (battery).

Table 6.3-8 provides a listing of figures which summarizes LOCA results.

6.3.3.7.3 TRACG-LOCA Break Spectrum Calculations

The break spectrum calculations were performed in Reference 42 to determine all potentially limiting initial conditions, single failures, break locations, and break size combinations. All calculations were performed for both GNF3 and GE14 fuel, and all calculations were performed with a loss of offsite power coincident with the break. All break spectra were calculated assuming a maximum core thermal power corresponding to the current licensed thermal power, plus power uncertainty, of 3,499 MW with an initial dome pressure of 1045 psia.

Only three limiting single failures are evaluated for the standard LOCA analysis, which are Division I Battery, Division II Battery and LPCI Injection Valve. The other three single failures, which are Diesel Generator (DG), HPCI and One ADS Valve, result in more ECCS systems available than at least one of the three limiting single failures and, therefore, are not considered in the break spectrum calculations.

For Division I Battery Failure, the core spray line (CSL), feedwater line (FWL), recirculation suction line (RSL), recirculation discharge line (RDL), Main Steam Line (MSL) and Reactor Water Cleanup (RWCU) breaks are considered. The limiting break sizes are 0.3185, 0.3154, and 0.3743 $ft^2$ for RDL, RSL and FWL, respectively. The CSL, MSL and RWCU are clearly non-limiting compared to the recirculation line breaks.

For Division II Battery Failure, the CSL, FWL, RSL, RDL, MSL and RWCU breaks are considered. The limiting break sizes are 0.1280, 0.1056, and 0.4491 $ft^2$ for RDL, RSL and FWL, respectively. The CSL, MSL and RWCU are clearly non-limiting compared to the recirculation line breaks.

For LPCI Injection Valve Failure, the RSL, RDL and CSL breaks are considered. The limiting break sizes are 0.7924 and 0.7848 $ft^2$ for RDL and RSL, respectively. Other breaks are not included for this failure because they are not limiting.

The Double Ended Guillotine Break (DEGB) peak cladding temperature (PCT), vessel pressure, and water level for GNF3 are provided in Figures 6.3-79, 6.3-80, and 6.3-81. The PCT, vessel pressure, and water level at the most limiting break for GNF3 are provided in Figures 6.3-82, 6.3-83, and 6.3-84.

6.3.3.7.4 TRACG-LOCA Statistical Analyses

Based on the break spectra calculations, potentially limiting breaks were chosen for statistical analysis in Reference 42. The results of the statistical analyses are shown in Table 6-3.4, with the overall maximum peak cladding temperature, maximum local oxidation (MLO), and core wide oxidation (CWO) for GE14 and GNF3 fuel types.

#### 6.3.3.7.5 Compliance Evaluations

The licensing basis PCT, maximum local fuel cladding oxidation (MLO), and total fraction of fuel cladding oxidized in the core (CWO) for GE14 and GNF3 are determined based on the results from the statistical analyses. The licensing basis PCT, MLO and CWO values are identified in Table 6.3-4.

#### 6.3.3.7.6 Operating Mode Considerations

The ECCS performance (Reference 42) was also evaluated for the following operating mode considerations:

a. Maximum Extended Operating Domain (MEOD) - The MEOD and Maximum Extended Load Line Limit Analysis (MELLLA) provide an expanded operating rod line and an increased core flow range operating domain as shown in Figure 4.4-3.

b. Partial Feedwater Heating (PFH) - The Feedwater Heaters Out-of-Service (FWHOOS) and Final Feedwater Temperature Reduction (FFWTR) mode of the PFH mode of operation (References 6 and 7).

c. Single Loop Operation (SLO) - SLO is permitted when operation is below 66.1% of rated power with recirculation pump speed limited to 75% (References 12 and 14).

d. Out-of-Service Equipment - The Fermi-2 Technical Specifications allows the turbine bypass, moisture separator reheater and several SRVs to be inoperable without requiring a plant shutdown. The unavailability of the turbine bypass and moisture separator has no impact on the results of the ECCS analysis because no credit for these systems has been taken in the ECCS evaluation. The availability of SRVs does not impact the calculated licensing basis PCT results since the limiting break events produce only a mild pressurization during the early time period of the event.

### 6.3.3.8 Loss-of-Coolant Accident Analysis Conclusions

Having shown compliance with the applicable acceptance criteria of Subsection 6.3.3.2, it is concluded that the ECCS will perform its function in an acceptable manner and meet all of the 10 CFR 50, Section 50.46 acceptance criteria.

### 6.3.3.9 Thermal Shock Considerations

The ECCS pumps starting at some time after the accident are at ambient (greater than 40°F) and could be heated rapidly as they draw their suctions from the suppression pool.

The HPCI pump and piping system considers a rapid rise in suction temperature from ambient (greater than 40°F) to the maximum operating temperature. The suction is normally from the condensate storage tank (less than 100°F), but can be switched to the suppression pool. If the reactor is not depressurized, the suppression pool temperature rises slowly, providing ample time for the operator to either depressurize and use the LPCI and/or core spray, or to cool the suppression pool with the containment cooling subsystem of the RHR system.

The design of the ECCS pumps (except HPCI), therefore, considers the differences in the rate of expansion between stationary and rotating parts in order to ensure operability during the transients (sudden change in water temperature from 40° to 170°F).

The piping design similarly considers this thermal shock. The steam line in the HPCI turbine is kept warm since it is normally open from the reactor with a drain pot at the turbine end of the line. A design requirement for the turbine itself is for rapid start, i.e., admission of hot steam to a cold turbine. The turbine vendor has considered the possible thermal shock effects in his design. The turbine exhaust increases rapidly from ambient (greater than 40°F) to operating (300°F) temperature, which is considered in both turbine and piping design.

The output of these ECCS subsystems into the reactor introduces relatively cold water into a hot RPV, and thermal shock is considered in the design of the reactor vessel, its nozzles, and the feedwater lines. The LPCI discharges via the hot recirculation line, so this thermal shock is also considered in the recirculation system piping design.

Section 5.2 contains a summary of results of the cold water injection thermal stress analyses.

### 6.3.4 Inspection and Testing

Each active component of the ECCS required to operate in a DBA is designed to be operable for test purposes during normal operation of the nuclear steam supply system (NSSS).

Regular tests are performed on the system to verify operability. If a test shows some element of the system to be inoperative, repairs are made to return the system to fully operative status. A failure of the system occurring between tests may have serious consequences, depending on whether or not a need to function occurs before the next test is performed. There is, therefore, a direct relationship between the system unreliability, the rate of occurrence of system failures from all causes, and the testing interval for the system.

It has been shown that the test frequency as well as the failure rate affect system reliability. There are practical limits on test frequency, such as the possibility of wearing out system components with too much testing. The test frequency outlined in the Technical Specifications is based upon these considerations.

The HPCI system, ADS, and core spray system have no normal process uses, and therefore are tested periodically to provide assurance that the ECCS will operate to effectively cool the reactor core in an accident. The four LPCI pumps may be placed in use as part of the RHR system and, if so, their status is known from normal process uses. However, the LPCI pumps should be tested no less frequently than the rest of the ECCS. Other parts of the LPCI, such as the two testable check valves inside the primary containment drywell and the four shutoff valves outside the drywell, are intended for use only in an accident, so they are also tested periodically.

Preoperational tests of the ECCS were conducted during the final stages of plant construction prior to initial startup. These tests ensured the proper functioning of all controls and instrumentation, pumps, piping, and valves. System reference characteristics such as pressure differentials and flow rates were documented during the preoperational tests and will be used as base points for measurements obtained in the subsequent operational tests.

During plant operations, the pumps, valves, piping, instrumentation, wiring, and other components outside the primary containment can be visually inspected at any time. Components inside the primary containment can be inspected when the drywell is open for access. When the RPV is open, for refueling or other purposes, the spargers and other internals can be inspected. The testing frequencies of most components of the ECCS are correlated with the testing frequencies of the associated controls and instrumentation. When a pump or valve control is tested, the operability of the pump or valve and the associated instrumentation is also tested by the same action.

When the system is tested, the operation of most of the components is indicated in the main control room. There are exceptions that require local observation at the component and may require special tests for which there are special provisions and methods.

Pressure-operated relief valves may leak after operation and it is not advisable to overpressurize the system for test, so relief valves are removed as scheduled at refueling outages for bench tests and setting adjustments. Bench tests of automatic depressurization valves are discussed in Subsection 5.2.2.

A pressure-operated control valve such as the one upstream of the HPCI system barometric condenser is functionally tested and adjusted in place, in accordance with the valve manufacturer's manual and the system specification for pressure setting. A test pressure connection is provided to check and adjust the setting.

Reverse flow and excess flow check valves in the ECCS are tested periodically in accordance with the Technical Specifications and the Inservice Testing Program.

Test lines are provided between pairs of containment isolation valves in the ECCS to measure leakage when the containment is pressurized for tests. The test line is also used to pressurize between the closed valves to identify which one is leaking. Allowable valve leakage is in accordance with Section 6.2 and the Technical Specifications.

Allowable valve seat leakage during shop hydrostatic tests for nuclear Class 1, 2, and 3 gate, globe, and ball valves associated with these systems is 2 $cm^3$/hr/in. of seat diameter during hydrostatic test at design pressure. Leakage for check and stop- check valves is 10 $cm^3$/hr/in. diameter of valve seat at the design differential pressure across the valve. Valve packing leakage during the hydrostatic test is specified as "no visible leakage."

Pumps for the ECCS are equipped with face-type mechanical shaft seals.

A design flow functional test of the HPCI system up to the normally closed pump discharge valve is performed during normal plant operation by pumping water from the condensate storage tank and back through the full-flow test return line. The HPCI system turbine pump is driven at its rated output by steam from the reactor. The suction valves from the suppression pool and discharge valves to the reactor feedwater line remain closed.

The HPCI system test conditions are tabulated on the HPCI system process diagram, Figures 6.3-1 through 6.3-5. If an initiation signal occurs while the HPCI system is being tested, the system returns to the automatic startup mode and supplies water to the reactor.

The HPCI may be tested at full flow with condensate at any time except when the reactor vessel water level is low, the condensate level in the condensate storage tank is below the reserve level, or the valves from the suppression pool to the pump are open.

During the HPCI flow test, the minimum-flow bypass valve opens/closes as required per the logic. The turbine steam valves and the flow test valves to the condensate storage tank are opened to support the HPCI flow test.

To ensure proper operation of the valves when pumping from the suppression pool, the HPCI suction valve auto transfer test is performed to meet Technical Specification requirements. Credit is taken for the RHR and CS testing from the suppression pool as an indication of strainer performance/degradation.

The RHR, CS and HPCI suppression pool suction strainers are inspected periodically in accordance with the plant preventive maintenance program.

Each loop of the core spray systems may be tested during reactor operation. The test conditions are tabulated on the core spray system process diagram, Figures 6.3-7 through 6.3-11. The normal system test does not inject cold water into the reactor because the testable check valve is held closed by the reactor pressure which is higher than core spray pump pressure. To test the injection portion of the system, using demineralized water, the reactor must be shut down and depressurized. This prevents unnecessary thermal stresses.

To test the core spray pumps at rated flow, the pump suction valve from the suppression pool is open, the pump is started using the remote manual switches in the main control room and the test bypass valve is opened to the suppression pool. Proper operation is determined by observing the instruments in the main control room. The core spray system outside the drywell is checked for leaks.

The two motor-operated injection valves outside the drywell and the air-operated testable check valve inside the drywell are tested as described in the Fermi 2 Inservice Testing Program.

If an initiation signal occurs during the test, the core spray system is signaled to start and the system returns to the automatic startup mode and is ready to deliver water to the reactor.

Similarly, LPCI pumps and valves are tested periodically during reactor operations. With the injection valves closed and the return line open to the suppression pool, full-flow pumping capability is demonstrated. The injection valves are tested, and the testable check valves are operated, as described previously for the core spray valves. The system test conditions during reactor shutdown are shown on the RHR/LPCI system process diagram, Figures 6.3-14 through 6.3-16. The portion of the LPCI outside the drywell is inspected for leaks during tests. Controls and instrumentation are tested as described in Section 7.3.

On receipt of an LPCI initiation signal during tests, the valves in the test bypass lines and in the shutdown cooling system are closed automatically to ensure that the LPCI pump discharge is routed properly to the reactor vessel.

Detailed specifications for ECCS component testing are contained in Chapter 14 and the Technical Specifications.

The valves performing an isolation function between high-pressure and low-pressure portions of systems connected to the RCS are tested in accordance with the Technical Specifications.

Table 6.3-9 lists the valves that perform an isolation function between high-pressure and low-pressure portions of systems connected to the RCS. These pressure isolation valves meet the requirements of the ASME Code Section XI, Pump and Valve Testing Program, and are categorized as A or AC. The testing program for the valves, which is referenced in the Technical Specifications, consists of the following methods:

a. Exercise the valve and verify the valve position during refueling and after maintenance before the return to service in accordance with IWV-3300 or IWV-3522(b)

b. Exercise the valve (full stroke) for operability during the cold-shutdown mode as time permits, but not more frequently than once every 3 months

c. Measure the full-stroke time (not for check valves)

d. Leak test the valve seat before reaching power operation following refueling and after valve maintenance before the return to service.

These valves will not be routinely exercised every 3 months during plant operation as required by IWV-3410 because of the following.

a. Such tests remove one of the two barriers protecting the low-pressure portion of the ECCSs

b. The operators on testable check valves cannot overcome the force on the valve with reactor pressure on one side.

Instead, the valves will be exercised during cold-shutdown periods as time permits (but not more frequently than once every 3 months). If there is excessive leakage through the normally closed gate and check valves, the operator will be alerted by the high pressure alarm indicated in Table 6.3-10. The operator will then be procedurally required to close the normally open gate valve from the control room to effect isolation.

### 6.3.5 Instrumentation Requirements

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

#### 6.3.5.1 High Pressure Coolant Injection Actuation Instrumentation

The HPCI is automatically actuated by the following sensed variables: (1) RPV low water level; or, (2) drywell high pressure.

In addition, the HPCI can be manually actuated from the main control room.

#### 6.3.5.2 Automatic Depressurization System Actuation Instrumentation

The ADS is automatically actuated when the RPV low water level is coincident with drywell high pressure. A time delay is incorporated as discussed in Chapter 7. In addition, two core spray pumps or an RHR pump must be running. Each ADS valve can be manually actuated from the main control room.

#### 6.3.5.3 Core Spray Actuation Instrumentation

The core spray is automatically actuated by the RPV low water level or drywell high pressure. In addition, the core spray can be manually actuated from the main control room.

#### 6.3.5.4 Low Pressure Coolant Injection Actuation Instrumentation

The LPCI is automatically actuated by the RPV low water level or drywell high pressure. In addition, the LPCI can be manually actuated from the main control room.

Emergency Procedures contain adequate caution to deter the operator from premature LPCI flow diversion. The Emergency Procedures caution the operator against diversion unless adequate core cooling is assured. The containment cooling modes of the RHR are secondary to core cooling requirements except in those instances outside the design envelope involving multiple failures, for which maintenance of containment integrity is required to minimize risk to the environment.

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# TABLE 6.3-1 SHUTDOWN COOLING AND EMERGENCY CORE COOLING SYSTEM OPERATION

| Accident or condition | Required Operation | RHR or ECCS Subsystems and Components Used | Redundancy Provided Within System | Backup System(s) |
|---|---|---|---|---|
| Shutdown Cooling | For a normal shutdown and cooldown, the main condenser is used to condense decay-heat-generated steam until the condenser vacuum is lost. Makeup water is provided as for condenser isolation. When the condenser is no longer effective, primary system cooling is continued by taking water from one of the recirculation loops, through the RHR heat exchangers, and back to the recirculation loop using the RHR pumps. Should the reactor be isolated from the condenser by operation of the isolation valve (not a normal operation) steam is first dumped to the suppression pool rather than to the condenser (below). | Main condenser<br>RHR heat exchangers,<br>RHR main pumps | Two RHR heat exchangers (one heat exchanger sufficient)<br>Four RHR pumps (two pumps sufficient) | RHR cooling subsystem backs up the main condenser used for shutdown cooling |
| Isolation of condenser (occurs when a reactor scram is accompanied by containment isolation) | Upon the closing of the main steam line valve following a scram, automatic operation of relief valves causes steam to be dumped to the suppression pool, the RHR removes heat from the pool.<br>The single RCIC steam-driven pump takes water from condensate storage and discharges to the feedwater line. Signal: low reactor vessel water level. | Relief valves<br>RHR heat exchangers<br>RCIC pump<br>Condensate (reserve storage) | Total of 15 safety/relief valves available (nine sufficient)<br>Two RHR heat exchangers (one sufficient) | 1. RCIC[a]<br>2. HPCI[a]<br>3. Control rod drive water systema<br>4. Core spray and LPCI[a] |
| Small leaks (accident condition) | First Level Feedwater system and control rod drive system can provide some makeup water. | | | Second level HPCI and/or RCIC |
| | Second Level The single HPCI steam-driven pump takes water from condensate storage and discharges to a feedwater line. Signal: low reactor vessel water level or high drywell pressure. The decay-heat-generated steam flows to the HPCI turbine and is exhausted to the suppression pool. | HPCI steam-driven pump, station battery (no ac required, 5000gpm) | RCIC pump (600gpm)<br>Water can also come from suppression pool | Third level |
| | Third Level Automatic depressurization system vents steam to the suppression pool. With decreased pressure LPCI and core spray systems can provide water signals: low reactor vessel level and high drywell pressure and core spray or RHR pump running. | Five safety/relief valves<br>Three of four pumps of LPCI<br>Two loops of core spray system (alternate)<br>Standby ac power supply | Manual actuation of any of 15 safety/relief valves | At low pressure (RV approximately 300 psi) both LPCI and core spray can operate.<br>SRVs |
| Large leaks (accident condition) | Core spray system pumps water from the suppression pool to core. Signal: low reactor vessel water level or high drywell pressure. | Pumps with electric motors, spray sparger, standby ac bus; different for each spray loop (6350 gpm per core spray system and 10,000 gpm per LPCI/RHR pump) | Two independent core spray systems standby ac bus available | LPCI |
| | LPCI operates. Three of the four RHR main pumps take water from the suppression pool and delivers it to a recirculation loop. Signal: low reactor vessel level or high drywell pressure. | Pumps and motors. Power from standby ac bus (30,000 gpm for 3 LPCI pumps plus 12,700 gpm for two core spray systems) | Three of the four RHR pumps required. All have ac standby as backup power source (four pumps are signaled to start) | Core spray |

[a] Systems used for reactor water inventory control.

TABLE 6.3-2 MATERIALS FOR THE PRINCIPAL EMERGENCY CORE COOLING SYSTEM COMPONENTS

| Item | Supplier | Impeller | Casing | Shaft |
|---|---|---|---|---|
| LPCI pump | Byron Jackson | Martensite SS | Carbon steel | Austenite SS |
| Core spray pump | Byron Jackson | Martensite SS | Carbon steel | Austenite SS |
| HPCI pump | Byron Jackson | Martensite SS | Carbon steel | Martensite SS |
| HPCI turbine | Terry | Low-alloy steel | Carbon steel | Low-alloy steel |
| ADS safety/relief valves | Target Rock two-stage walves | N/A | N/A | N/A |

TABLE 6.3-3 HAS BEEN INTENTIONALLY DELETED

TABLE 6.3-4 SUMMARY OF RESULTS OF LOSS-OF-COOLANT ACCIDENT ANALYSIS

| | Parameters | Results | Results | Acceptance Criteria |
|---|---|---|---|---|
| 1. | Fuel Type | GE14 Fuel | GNF3 Fuel | |
| 2. | Limiting Break | Recirculation Suction Small Break | Recirculation Suction Small Break | |
| 3. | Limiting Failure | Division II DC Power (Battery) | Division II DC Power (Battery) | |
| 4. | Peak Cladding Temperature (Licensing Basis) | < 1980 °F | < 2150 °F | ≤ 2200 °F |
| 5. | Maximum Local Oxidation | < 6.0 % | < 9.5 % | ≤ 13 %[a] |
| 6. | Core-Wide Metal-Water Reaction | < 0.02 % | < 0.02 % | ≤ 1.0 % |
| 7. | Coolable Geometry | Items 4 and 5 | Items 4 and 5 | PCT ≤ 2200 °F and Maximum Local Oxidation ≤ 13 %[a] |
| 8. | Long Term Cooling | Core flooded above top of active fuel | Core flooded above top of active fuel | Core temperature acceptably low and long-term decay heat removed |

[a] The MLO calculated by TRACG-LOCA is limited to 13% to ensure the 10CFR50.46 limit of 17% is satisfied.

TABLE 6.3-5 EMERGENCY CORE COOLING SYSTEM SINGLE - FAILURE EVALUATIONS[a]

| Assumed Failure[b] | Suction Break Systems Remaining[c,d] |
|---|---|
| LPCI valve | All ADS, 2 core spray, HPCI |
| Divisional diesel generators (EDG) | All ADS, 1 core spray, HPCI, 2 LPCI |
| Battery (Division I) | HPCI, 2 LPCI, 1 core spray |
| Battery (Division II) | All ADS, 1 core spray, 2 LPCI |
| HPCI | All ADS, 4 LPCI, 2 core spray |
| One ADS valve | All ADS minus one, 2 core spray, HPCI, 4 LPCI |

[a] This table shows the single active failures considered in the ECCS performance evaluation.

[b] Other postulated failures are not specifically considered because they all result in at least as much ECCS capacity as one of the above assumed failures.

[c] Systems remaining, as indentified in this table, are with concurrent loss-of-offsite power and are applicable to all non-ECCS line breaks. For a LOCA from an ECCS line break, the systems remaining are those listed, less the ECCS in which the break is assumed.

[d] Analyses performed with one ADS valve assumed unavailable in addition to the single failure (Table 6.3-6).

## TABLE 6.3-6 ECCS ANALYSIS SIGNIFICANT INPUT VARIABLES AND INITIAL CONDITION

| Variable | Units | Value |
|---|---|---|
| A. Plant parameters | | |
| Core Thermal Power, plus uncertainty | MWth | 3499 |
| Nominal Vessel Dome Pressure | psia | 1045 |
| Maximum Core Recirculation Flow | mlb/hr | 105 |
| Rated normal feedwater temperature | °F | 426.5 |
| Reduced feedwater temperature | °F | 376.5 |
| Nominal downcomer water level (above vessel zero) | inches | 563.5 |
| B. Emergency Core Cooling System Parameters | | |
| Low-Pressure Coolant Injection (LPCI) System | | |
| Vessel Pressure at Which Flow Credited to Commence | psid | 264 |
| Minimum Flow at Vessel Pressure of: | psid | 20 |
| Two pumps | gpm | 21,850 |
| Three pumps | gpm | 26,260 |
| Four pumps | gpm | 27,625 |
| Initiating Signals and Setpoints: | | |
| Low Water Level | ft above TAF* | 1.02 |
| or | | |
| High Drywell Pressure | psig | 2.0 |
| Assumed Injection Valve Stroke Time | sec | 30 |
| Maximum Vessel Pressure At Which LPCI Injection Valve Can Open | psig | 350 |
| Maximum Allowable Time from Drywell Pressure Initiating Signal to Pump at Rated Speed and Ready to Inject Flow to Vessel with Emergency Power | sec | 77 |
| Minimum Break Size for Which Loop Selection Logic Assumed to Select Unbroken Loop | $ft^2$ | 0.15 |

TABLE 6.3-6 ECCS ANALYSIS
SIGNIFICANT INPUT VARIABLES AND INITIAL CONDITION

| Variable | Units | Value |
|---|---|---|
| Low Pressure Core Spray (CS) System | | |
| Vessel Pressure at Which Flow May Commence | psid | 280 |
| Minimum Rated Flow at Vessel Pressure of: | psid | 100 |
| One Loop | gpm | 5,625 |
| Initiating Signals and Setpoints: | | |
| Low Water Level | ft above TAF* | 1.02 |
| or | | |
| High Drywell Pressure | psig | 2.0 |
| Runout Flow at Vessel Pressure of: | psid | 0 |
| One Loop | gpm | 7,013 |
| Assumed Injection Valve Stroke Time | sec | 15 |
| Maximum Vessel Pressure at Which LPCS Injection Valve Can Open | psig | 350 |
| Maximum Allowable Time from Drywell Pressure Initiating Signal to Pump at Rated Speed and Ready to Inject flow to Vessel with Emergency Power | sec | 47 |
| High Pressure Coolant Injection (HPCI) System | | |
| Vessel Pressure at Which Flow May Commence | psia | 1135 |
| Minimum Rated Flow at Vessel Pressure of: | psia<br>gpm | 1135 to 165**<br>5000 |
| Initiating Signals and Setpoints: | | |
| Low Water Level | ft above TAF* | 7.6 |
| or | | |
| High Drywell Pressure | psig | 2.0 |
| Maximum Allowable Time from Drywell Pressure Initiating Signal to Rated Flow Available and Injection Valve Wide Open | sec | 60 |
| Automatic Depressurization System (ADS) | | |
| Total Number of Valves Installed | -- | 5 |
| Number of Valves Assumed in Analysis | -- | 4 |

## TABLE 6.3-6 ECCS ANALYSIS SIGNIFICANT INPUT VARIABLES AND INITIAL CONDITION

| Variable | Units | Value |
|---|---|---|
| Minimum Flow Capacity of any 4 Valves at | mlb/hr | 3.48 |
| Vessel Pressure | psig | 1090 |
| Initiating Signals and Setpoints: | | |
| Low Water Level | ft above TAF* | 1.02 |
| and | | |
| High Drywell Pressure | psig | 2.0 |
| Time Delay After Initiating Signal | sec | 120 |

* TAF (Top of Active Fuel) = 366.3 inches from vessel zero

** HPCI pump is designed to produce a flow of 5000 gpm at an RPVpressure of 1184 psia, which exceeds LOCA input.

TABLE 6.3-7 OPERATIONAL SEQUENCE OF EMERGENCY CORE COOLING SYSTEMS FOR DESIGN-BASIS ACCIDENT[a]

| Time (sec) | Events |
|---|---|
| 0 | Design-basis LOCA assumed to start; normal auxiliary power assumed to be lost. |
| 0 | Drywell high pressure and reactor low water level reached; scram; HPCI, LPCS, LPCI signaled to start on high drywell pressure. |
| 3 | Reactor low-low water level reached. Main steam isolation valves close; HPCI receives second signal to start, all diesel generators signaled to start. |
| 7 | Reactor low-low-low water level reached. Second signal to start LPCI and LPCS; autodepressurization sequence begins. |
| <13 | All diesel generators ready to load; open HPCI injection valve; begin energizing LPCI pump motors. |
| 18 | Begin energizing LPCS pump motors. |
| ≤47 | LPCS pumps at rated flow; LPCS injection valves open, completing the LPCS startup. |
| ≤77 | LPCI pumps at rated flow; LPCI injection valves open, completing the LPCI startup. |
| See Figure 6.3-20 | Core effectively reflooded assuming worst single failure; heatup terminated |
| ≥10 minutes | Operator shifts to containment cooling. |

[a] For the purpose of all but the next-to-last entry on this table, all ECCS equipment is assumed to function as designed. Performance analysis calculations consider the effects of single equipment failures. (See Subsections 6.3.2.5 and 6.3.3.3.)

TABLE 6.3-8 KEY TO FIGURES IN SECTION 6.3.3.7

| Break Size<br>Variable | Large Recirculation<br>Line Break, DEGB | Small Recirculation<br>Line Break |
|---|---|---|
| Peak Clad Temperature Vs. Time | Figure 6.3-79 (GNF3) | Figure 6.3-82 (GNF3) |
| RPV Pressure Vs. Time | Figure 6.3-80 (GNF3) | Figure 6.3-83 (GNF3) |
| Water Level Vs. Time | Figure 6.3-81 (GNF3) | Figure 6.3-84 (GNF3) |

These curves indicate the trends of the variables post-LOCA.

TABLE 6.3-9 PRESSURE ISOLATION VALVES

| System | P&ID | Valve Numbers | Type | Size (in.) | Function |
|---|---|---|---|---|---|
| RHR | 6M721-2083 | E11-F015A, B | Gate | 24 | Discharge to recirculation system |
| | 6M721-2084 | E11-F050A, B | Check | 24 | Discharge to recirculation system |
| | | E11-F008 | Gate | 20 | Suction from recirculation system |
| | | E11-F009 | Gate | 20 | Suction from recirculation system |
| | | E11-F608 | Gate | 20 | Suction from recirculation system |
| Core spray | 6M721-2034 | E21-F005A, B | Gate | 12 | Discharge to core spray sparger |
| | | E21-F006A, B | Check | 12 | Discharge to core spray sparger |
| HPCI | 6M721-2035 | E41-F006 | Gate | 14 | Discharge to feedwater line |
| | | E41-F005 | Check | 14 | Discharge to feedwater line |
| RCIC | 6M721-2044 | E51-F013 | Gate | 6 | Discharge to feedwater line |
| | | E51-F014 | Check | 6 | Discharge to feedwater line |

TABLE 6.3-10 PRESSURE ISOLATION PROTECTION AND MONITORING

| System/Line Needing Protection | Relief Valve Overpressure Protection | Control Room Alarm | Control Room Indicator | Local Indicator |
|---|---|---|---|---|
| RHR discharge | F025A, B, 1-1/2 in. | E11-N022A, B at 400 psig | E11-R003A, B, C, D, 0-600 psig | -- |
| RHR suction | F030A, B, C, D, F029, 1 in. | -- | E11-R002A, B, C, D, 30 in. Hg, 150 psig | -- |
| Core spray discharge | E2100F012A (V22-2016), E2100F012B (V22-2017), E2100F011B (V22-2119), E2100F011A (V22-2120) | E21-N007A, B at 440 psig | -- | E21-R600A, B, 0-600 psig |
| HPCI | E41-F020 (V22-2044), 1-1/2 in. | E41-N031 at 70 psig | -- | E41-R004, 30 in. Hg, 100 psig |
| RCIC suction | E51-F017 (V22-2002), 1 in. | E51-N030 at 70 psig | -- | E51-R002, 30 in. Hg, 85 psig |

Figure Intentionally Removed
Refer to Plant Drawing M-5860

Figure Intentionally Removed
Refer to Plant Drawing M-5872

**Fermi 2**

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FIGURE 6.3-2

HIGH PRESSURE COOLANT INJECTION SYSTEM
HIGH PRESSURE INJECTION
ACCIDENT CONDITION

Figure Intentionally Removed
Refer to Plant Drawing M-5873

**Fermi 2**

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FIGURE 6.3-3

HIGH PRESSURE COOLANT INJECTION SYSTEM
HIGH PRESSURE INJECTION MODE USING
SUPRESSION POOL AS BACKUP SOURCE

Figure Intentionally Removed
Refer to Plant Drawing M-5874

**Fermi 2**

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FIGUE 6.3-4

HIGH PRESSURE COOLANT INJECTION SYSTEM
MINIMUM FLOW MODE

Figure Intentionally Removed
Refer to Plant Drawing M-5875

**Fermi 2**

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FIGURE 6.3-5

HIGH PRESSURE COOLANT INJECTION SYSTEM
TEST MODE DURING PLANT OPERATION

FIGURE 6.3-6
HAS BEEN DELETED

Figure Intentionally Removed
Refer to Plant Drawing M-5861

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FIGURE 6.3-7

CORE SPRAY SYSTEM PROCESS DIAGRAM

Figure Intentionally Removed
Refer to Plant Drawing M-5868



FIGURE 6.3-8

CORE SPRAY - ACCIDENT CONDITION

Figure Intentionally Removed
Refer to Plant Drawing M-5869



FIGURE 6.3-9

CORE SPRAY SYSTEM TEST MODE DURING PLANT OPERATION USING SUPPRESSION POOL

Figure Intentionally Removed
Refer to Plant Drawing M-5870

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FIGURE 6.3-10

CORE SPRAY SYSTEM TEST MODE DURING PLANT SHUTDOWN USING CONDENSATE STORAGE TANK

Figure Intentionally Removed
Refer to Plant Drawing M-5871

**Fermi 2**

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**FIGURE 6.3-11**

**CORE SPRAY SYSTEM MINIMUM FLOW BYPASS MODE – SUCTION FROM SUPPRESSION POOL**

FIGURE 6.3-12
HAS BEEN DELETED

FIGURE 6.3-13
HAS BEEN DELETED

Figure Intentionally Removed
Refer to Plant Drawing M-5857

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**FIGURE 6.3-14, SHEET 1**

**LOW PRESSURE COOLANT INJECTION SYSTEM PROCESS DIAGRAM**

Figure Intentionally Removed
Refer to Plant Drawing M-5690

**Fermi 2**

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FIGURE 6.3-14, SHEET 2

LOW PRESSURE COOLANT INJECTION SYSTEM
PROCESS DIAGRAM

Figure Intentionally Removed
Refer to Plant Drawing M-5866

**Fermi 2**
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FIGURE 6.3-15

LOW PRESSURE COOLANT INJECTION SYSTEM
TEST MODE DURING PLANT OPERATION

Figure Intentionally Removed
Refer to Plant Drawing M-5867

**Fermi 2**
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FIGURE 6.3-16

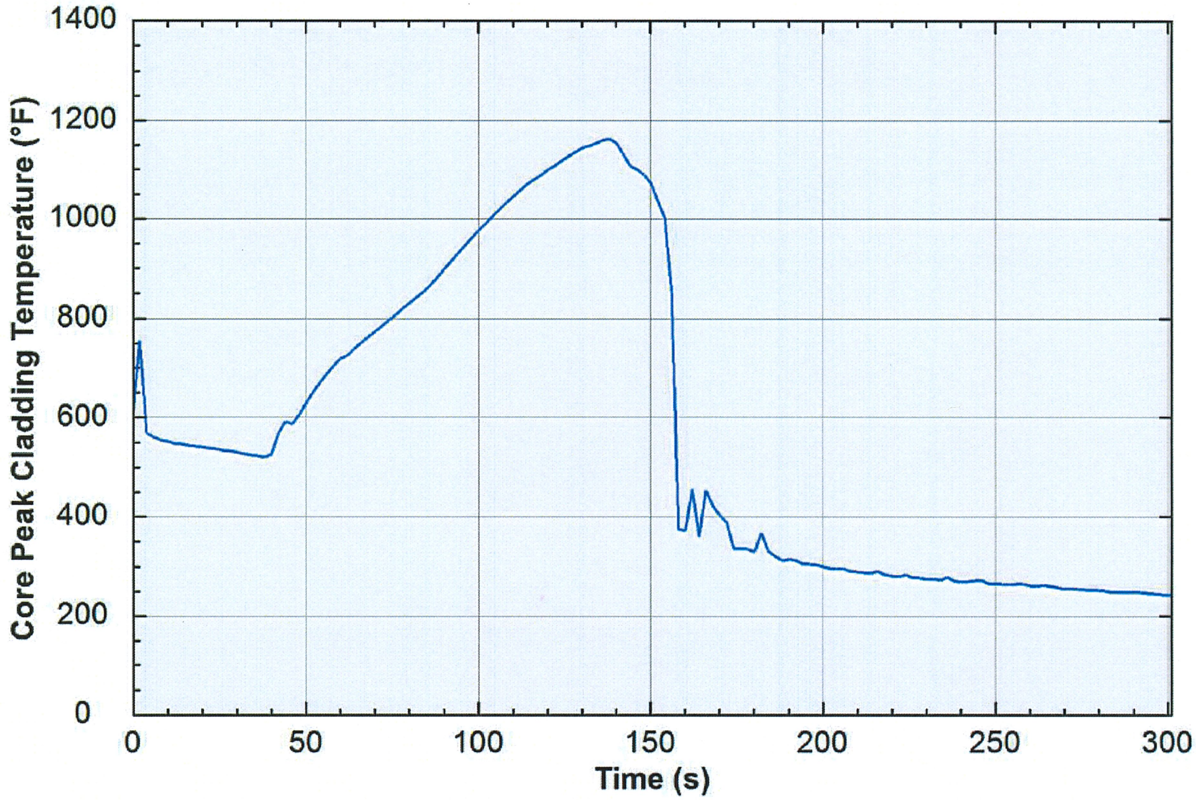
LOW PRESSURE COOLANT INJECTION SYSTEM
MINIMUM FLOW BYPASS MODE

FIGURE 6.3-17
HAS BEEN DELETED

FIGURES 6.3-18 THROUGH 6.3-23
HAVE BEEN DELETED

FIGURE 6.3-24 THROUGH FIGURE 6.3-78

HAVE BEEN DELETED




NEDC-33919P, Revision 0, Figure 9-33

**Fermi 2**
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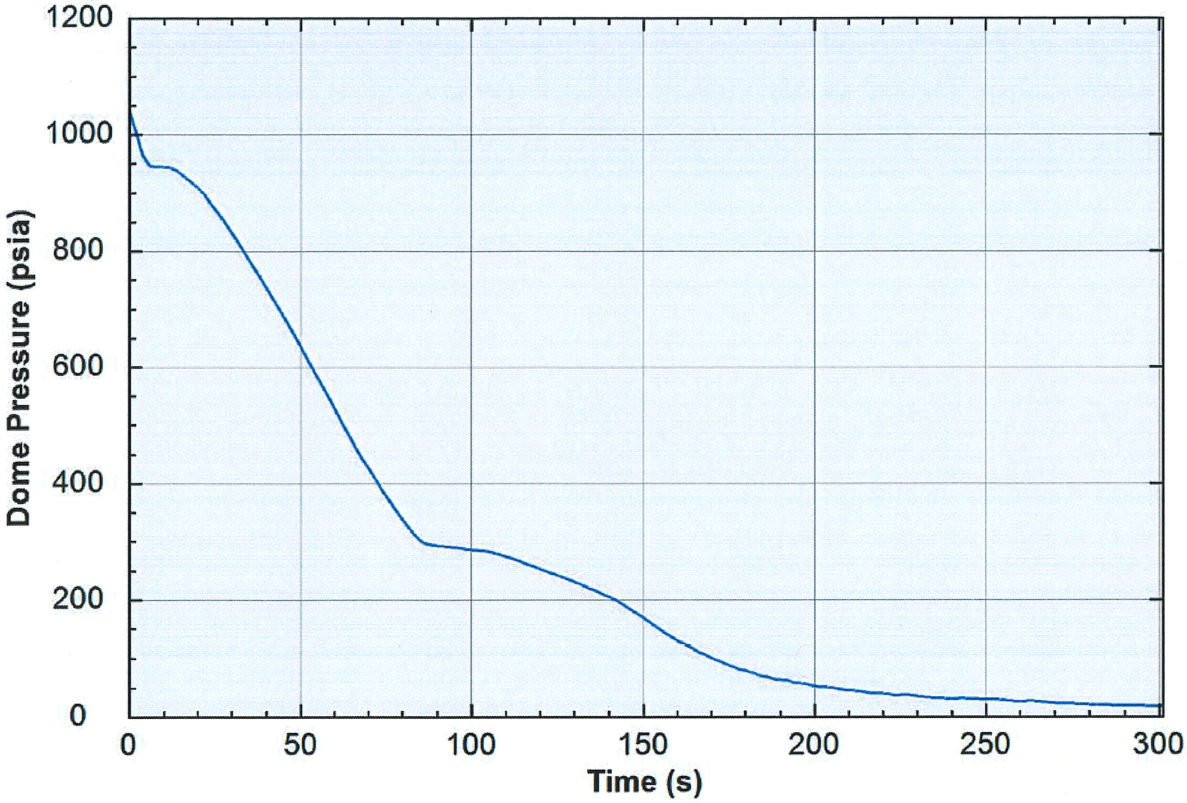
FIGURE 6.3-79

FERMI 2 GNF3 NOMINAL DEGB RDL BREAK FOR CLTP INITIAL CONDITIONS FOR DIV II BATTERY FAILURE – OVERALL CORE PCT




NEDC-33919P, Revision 0, Figure 9-29

**Fermi 2**

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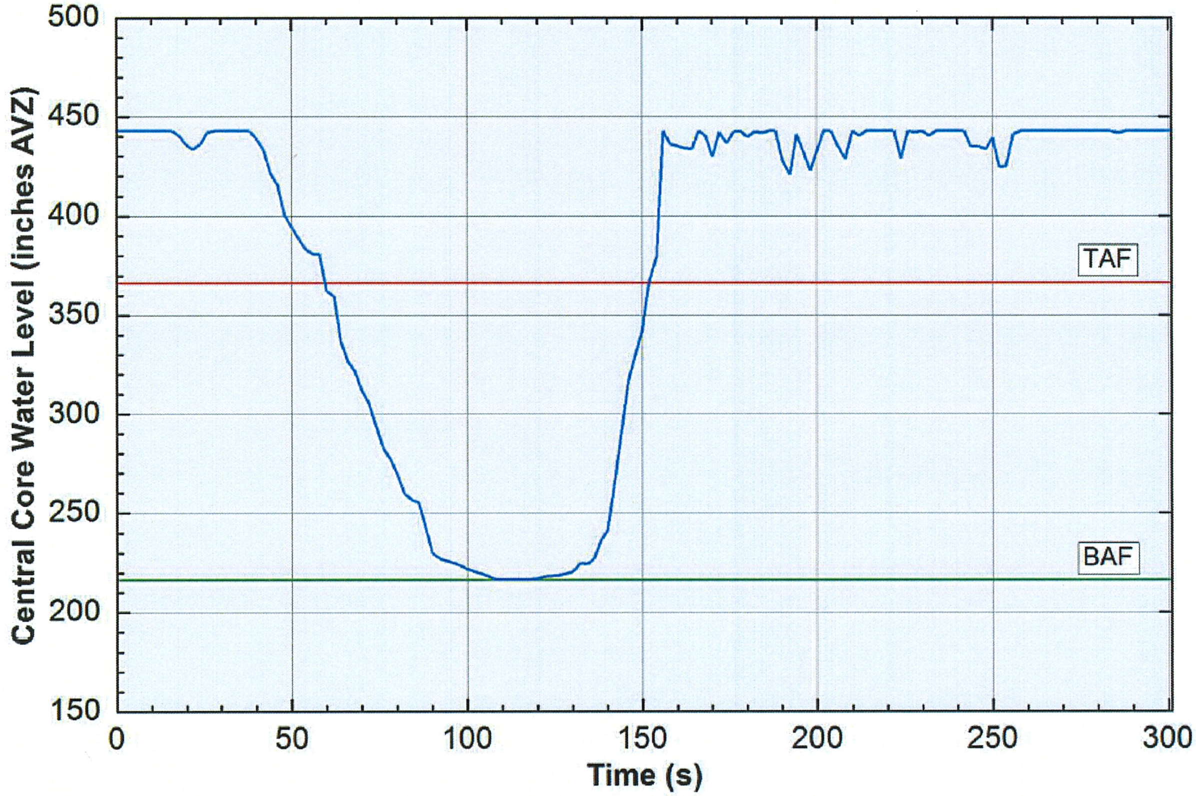
FIGURE 6.3-80

FERMI 2 GNF3 NOMINAL DEGB RDL BREAK FOR CLTP INITIAL CONDITIONS FOR DIV II BATTERY FAILURE – REACTOR PRESSURE




NEDC-33919P, Revision 0, Figure 9-31

**Fermi 2**
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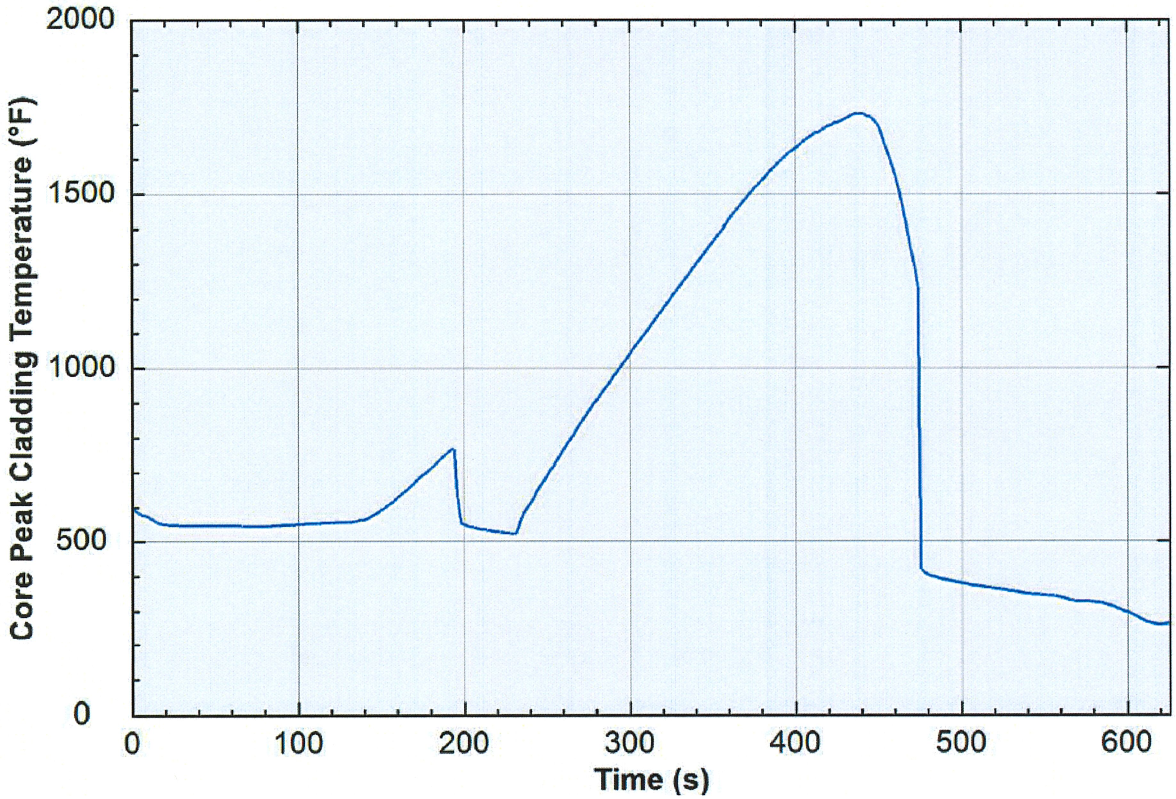
FIGURE 6.3-81

FERMI 2 GNF3 NOMINAL DEGB RDL BREAK FOR CLTP INITIAL CONDITIONS FOR DIV II BATTERY FAILURE – CENTRL CORE (BYPASS AND UPPER PLENUM) WATER LEVEL




NEDC-33919P, Revision 0, Figure 9-6

**Fermi 2**
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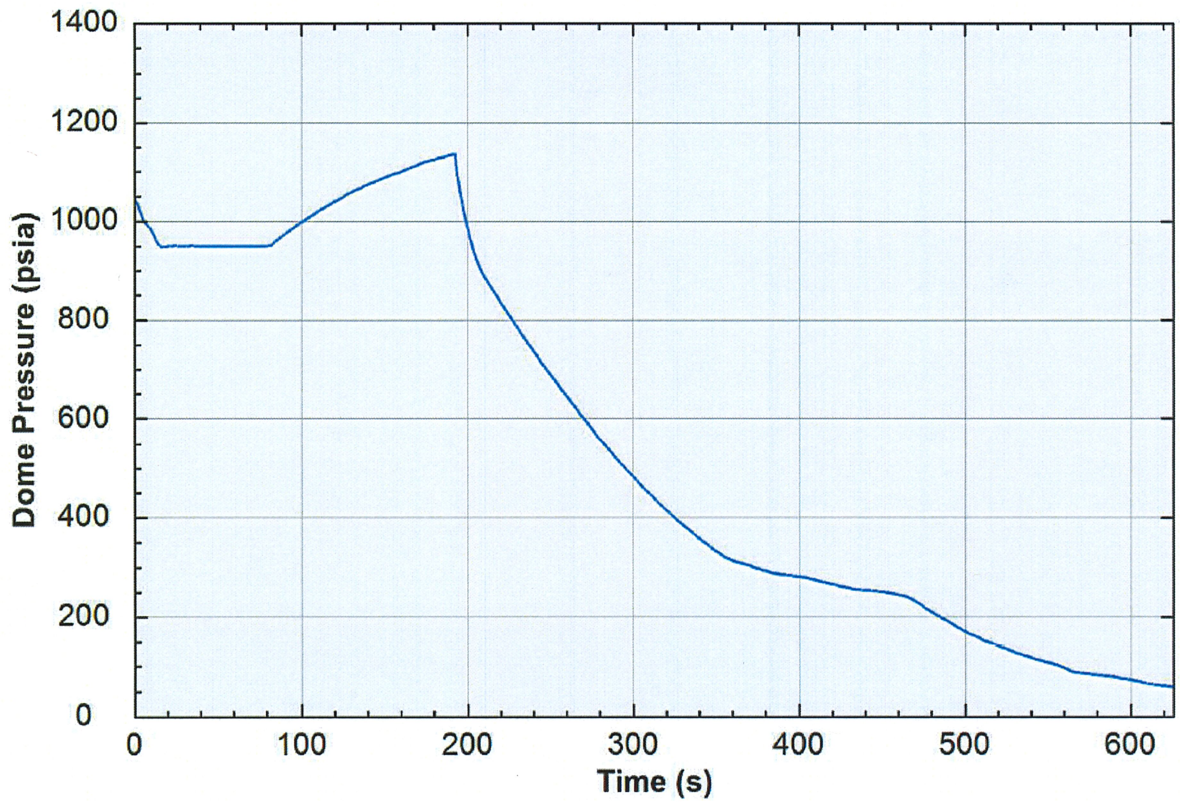
FIGURE 6.3-82

FERMI 2 GNF3 NOMINAL SMALL RSL BREAK FOR MELLLA INITIAL CONDITIONS FOR DIV II BATTERY FAILURE – OVERALL CORE PCT




NEDC-33919P, Revision 0, Figure 9-2

**Fermi 2**
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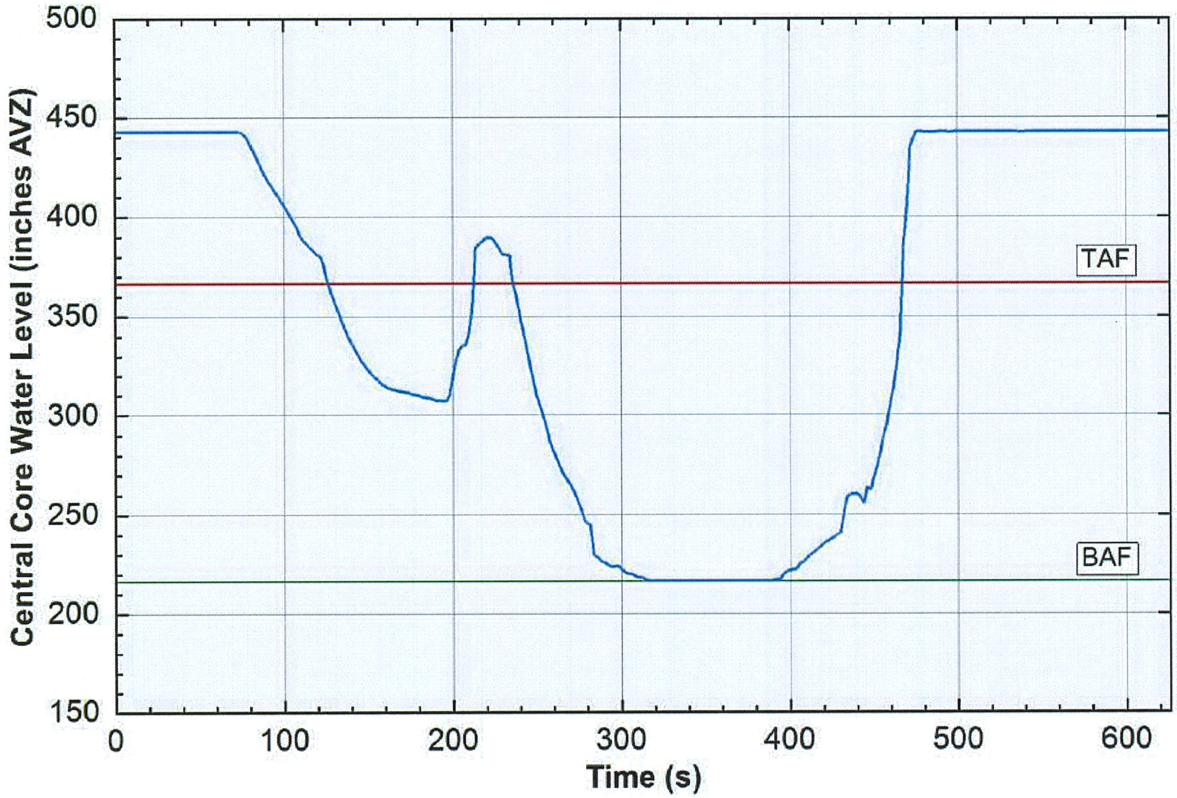
FIGURE 6.3-83

FERMI 2 GNF3 NOMINAL SMALL RSL BREAK FOR MELLLA INITIAL CONDITIONS FOR DIV II BATTERY FAILURE – REACTOR PRESSURE




NEDC-33919P, Revision 0, Figure 9-4

**Fermi 2**
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FIGURE 6.3-84

FERMI 2 GNF3 NOMINAL SMALL RSL BREAK FOR MELLLA INITIAL CONDITIONS FOR DIV II BATTERY FAILURE – CENTRAL CORE (BYPASS AND UPPER PLENUM) WATER LEVEL

## 6.4 HABITABILITY SYSTEMS

Control center habitability systems ensure that the main control room can be occupied under normal, accident, and postaccident conditions. Habitability systems include the systems, components, facilities, supplies, and equipment required for safe habitation of the main control room.

### 6.4.1 Habitability Systems Design Bases

The bases for the functional design of the habitability systems are given below. The bases result in systems that ensure compliance with General Design Criterion 19, 10 CFR 50, Appendix A. Under any design-basis conditions, the environment within the main control room is safe, with personnel protected from radiation, fire, toxic gases, and noxious substances.

#### 6.4.1.1 Radiation Shielding and Air Filtration

The total radiation dose to main control room personnel is the dose received while occupying the control center. Doses received while in the main control room are due to the radiation that penetrates the biological shielding and to the isotopes that enter the control center through the ventilation system or through inleakage. Source terms and individual contributions to total doses are given in Chapter 15, along with a further discussion of the assumptions, the physical models, and the methods of analysis.

Sufficient radiation shielding and air filtration are provided to ensure that radiation exposures of main control room personnel do not exceed 5 rem whole-body, or its equivalent to any part of the body, for the duration of a design-basis accident (DBA). For the DBA-LOCA and the Fuel Handling Accident, the dose to main control room personnel does not exceed 5 rem TEDE.

Following is a list of principal assumptions used in determining the control center personnel doses:

a. The plant personnel occupying the control center at the time the LOCA occurs remain in the control center for a period of 24 hr after that occurrence

b. Control center personnel shift changes occur twice per day, starting 24 hr after the occurrence

c. The occupancy factor is 1 for 0-1 day, 0.6 for 1-4 days, and 0.4 for 4-30 days

d. The breathing rates of the main control room personnel are $3.47 \times 10^{-4}$ $m^3$/sec as specified by the International Committee on Radiation Protection (ICRP) (Reference 1)

e. In the event of a LOCA, the control center mode in operation is automatically shut down and the emergency makeup air filtration system is placed in operation.

f. When emergency makeup air is supplied to the main control room, the rate of introduction of outside air is 1800 cfm maximum

g. Radiation monitors in the reactor/auxiliary (i.e., fuel pool ventilation exhaust ducting) building detect airborne radiation concentrations above those specified in the Technical Specifications and cause the control center air conditioning system to automatically switch to its emergency mode of operation

h. Filter trains are provided for emergency makeup air as well as recirculated air. The filter trains are located outside the main control room

i. Charcoal filters (described in Table 9.4-1) have a assigned decontamination efficiency of 95 percent for removal of all forms of iodine; 99 percent efficiency could be claimed for the recirculation charcoal adsorber according to Regulatory Guide 1.52, but only 95 percent efficiency is claimed to avoid the more frequent testing and replacement of charcoal

j. Control center filter banks are in service throughout the course of the LOCA, filtering outside air makeup l800 cfm maximum and recirculated air 1200 cfm for a total filtered airflow of 3000 cfm

k. The mechanisms for introduction of radioisotopes into the main control room are

    1. Intake through filter trains during periods of air makeup

    2. Infiltration of outside air or exchange of inside-outside air due to opening and closing of main control room doors at shift-change times. The total quantity of unfiltered inleakage is not more than 173 cfm.

l. A radiation monitor in the control center air intake ducting, before filtration by the emergency makeup air intake and recirculation filter trains, provides radiation level information to the operators.

The assumed atmospheric dispersion factors and radioactive source terms used for each accident analysis are listed in Chapter 15.

Table 15.6.5-4 presents the doses to the control room operator from occupancy of the control room for the 30-day course of a LOCA. Table 12.1-14 presents the direct doses through the concrete walls and ceilings for the 30-day course of a LOCA as experienced by main control room personnel.

#### 6.4.1.2 Physical Environment

Systems and controls are provided to ensure that the environment in the control center is safe and comfortable. The thermal environmental conditions are within the comfort range specified in American Society of Heating, Refrigerating and Air-Conditioning Engineers (ASHRAE) Comfort Standard 55-66 (Reference 2), with a nominal dry bulb temperature of 75°F and a relative humidity not exceeding 60 percent, except for the mechanical equipment room (MER) and SGTS room which are discussed in Subsection 9.4.1.1. The emergency operating modes of the air conditioning system are designed to meet single-failure criteria and ensure l00 percent backup for the entire system, with the exception of the common ductwork and filters. The smoke/Halon dampers to the relay room, cable spreading room or computer room will close due to a single active failure in the Halon fire protection system.

Sufficient time is available to take manual action to reestablish airflow. Ventilation capability is provided by the air conditioning system in both the normal mode and the emergency air makeup mode of operation. Main control room environmental conditions, including radiation levels, are monitored. The air volume of the control center envelope is approximately 252,731 $ft^3$, which is sufficient to allow closing of the makeup air intake for a period of more than 28 days without exceeding permissible carbon dioxide concentrations when three workers occupy the main control room.

### 6.4.1.3 Fire Protection

Noncombustible and flame-retardant materials are used where practical, and equipment, electrical components, and control instrumentation are designed to minimize fire hazards. Fire and smoke detection and alarm systems are provided as required. Applicable NFPA codes and standards used for guidance are listed in Subsection 9.5.1.

Portable fire extinguishers are located in the main control room. The equipment is adequate to control fires that could originate inside the main control room. The air conditioning system has a purging capability to expedite the discharge of smoke from the main control room.

The control center can be isolated to prevent admission of smoke or noxious fumes resulting from a postulated fire outside the main control room. The control center is designed to be protected against exterior fire exposure by the 3-hr-rated walls. Personnel are not harmed and safety-related equipment is not damaged by the proper use of the portable fire extinguishers in the main control room.

Personnel training ensures that plant operators are cognizant of the proper use of fire extinguishers and know the emergency procedures to be taken in the event of fire.

### 6.4.1.4 Personnel Protection and First Aid

The control center contains emergency safety breathing apparatus for personnel use, as well as first aid supplies for immediate emergency use.

### 6.4.1.5 Utilities and Sanitation

Normal communications, lighting, kitchen, and sanitary facilities are provided in the control center to ensure habitability. The onsite power system supplies power for the main control room habitability systems when offsite power is not available.

## 6.4.2 System Design

### 6.4.2.1 Radiation Shielding

Accessibility to the main control room during normal operation is unlimited, with sufficient shielding provided to ensure that normal radiation levels are below 0.3 mrem/hr. In addition to its function during normal operation, the main control room shielding reduces direct radiation doses from the LOCA to levels that permit controlled occupancy by operating personnel following that accident.

Control center shielding is ordinary concrete. The actual floor of the main control room is 1 ft thick but, as a result of other structures, it has an effective thickness of 8 ft 4 in.; the outside (north) wall is 2 ft thick; the roof of the main control room is 5 ft thick over one portion and 1 ft thick over the remaining portion, with the total effective concrete thickness over the main control room varying, however, between 6 ft 6 in. and 10 ft 6 in.; the wall facing the reactor is 4 ft 4 in. thick, with an additional 7 ft of concrete biological shielding surrounding the reactor.

Section 12.1 includes a layout drawing of the control center, as well as a scaled isometric view of the main control room and its associated shielding. Section 12.1 also presents detailed descriptions of shield thicknesses, justifications for the thicknesses of shielding provided, descriptions of the geometric and physical models used, and information relative to the assumptions and data used in the design.

#### 6.4.2.2 Radiation Monitoring System

The functional design of the radiation monitoring system (RMS) provides adequate and reliable radiological data for the evaluation of habitability of the main control room.

The outputs from area and process radiation monitors associated with main control room habitability are displayed, alarmed, and recorded, if necessary, in the main control room.

The area radiation monitor provides measurements of dose rates in the main control room. Location and the design criteria used in their selection are described in Subsection 12.1.4 along with operational characteristics, including type of detector, sensitivity, range, method of calibration, and setpoints.

Process monitors continuously monitor the levels of radioactivity in main control room ventilation systems. Inlet makeup air is monitored upstream of the filters in the emergency air makeup and recirculation systems. Airborne radioactivity monitoring is described in Subsection 12.2.4, which gives the locations and design criteria of the fixed instruments, as well as the criteria used to determine the necessity for and the location of the equipment. That section also provides information on the operational characteristics of the monitors, the detector type, the sensitivities, the ranges, and setpoints and their bases.

Personnel dosimetry under normal and under accident conditions is described in detail in Subsection 12.3.4.

#### 6.4.2.3 Air Conditioning System

##### 6.4.2.3.1 System Description

The control center air conditioning system (CCACS) is described in Section 9.4, which includes system, water control, and airflow diagrams.

The air conditioning system provides year-round comfort and safety from airborne radioactivity for control center personnel. The individual components of the Category I system are designed for an operational life of 40 years, accounting for corrosion and material fatigue. Electrical power for motor operation is supplied from the reactor building engineered safety feature (ESF) buses, maintaining separation and redundancy, and common-mode failure is prevented by physical separation.

The system is capable of maintaining the control center at a nominal temperature of 75°F and at a maximum relative humidity of 60 percent during normal operation, except for the mechanical equipment room (MER) and SGTS room which are discussed in Subsection 9.4.1.1. Noise level in the main control room, when measured in accordance with Appendix E1 of NUREG-0700, Guidelines for Control Room Design Reviews, does not exceed 65 dB(A). The noise level in the washroom and kitchen, on the same measurement basis, also does not exceed 65 dB(A). Conditioned air is supplied directly to the main control room, while the kitchen and washroom are conditioned by exhausting air that is drawn from the main control room.

There are four operating modes for the CCACS as follows:

a. Normal mode: A minimum of 2769 cfm outside air mixes with recirculated ventilating air, bypassing the emergency makeup and recirculation filters

b. Purge mode: 100 percent outside air is circulated through the control center and exhausted to the atmosphere to purge any smoke or fumes within the control center

c. Recirculation mode: A maximum of 1800 cfm outside air is filtered and mixes with 1200 cfm recirculated air; it is filtered again and mixed with recirculating ventilation air to prevent intrusion and to provide continuous removal of contaminants during a radiation-release emergency

d. Chlorine mode: All outside intakes are closed to prevent ingress during a chlorine-release emergency. Ventilating air is recirculated with 1200 cfm passing through the emergency recirculation filter.

Normal Operation Mode

During normal operation, the control center air conditioning system serves the main control room and several other areas. The supply airflow to the control center is 31,510 cfm and the return airflow is 30,440 cfm. Normal makeup air is passed through an electronic air cleaner and a roll-type filter.

The master selector switches in the main control room activate all components in the Division I or Division II system. The mixture of return and outside air is filtered, then cooled, heated, and dehumidified, as required, by a multizone air conditioning supply unit. Each zone thermostat modulates zone mixing dampers to obtain the supply-air temperature necessary to satisfy the zone cooling or heating requirements. Positive pressure is maintained in the control center by throttling the exhaust air-modulating damper. Exhaust fans are provided in the kitchen and washrooms.

Manual override is provided such that an operator in the main control room is able to select the purge mode, which opens the outside air dampers and closes the return air dampers.

Recirculation Mode

Upon an automatic isolation signal from the reactor protection system or the RMS, the CCACS is automatically transferred to the recirculation mode. Under emergency conditions, airflow rate into the control center is 31,510 cfm including 1800 cfm maximum makeup air to offset supply air lost through room leakage (maintaining the control center positive pressure of 1/4 ± 1/8-in. water gage). The kitchen and washroom individual exhaust ducts each contain dual isolation valves that are closed under emergency conditions.

During an emergency, the control center is isolated from all other areas of the plant. All air supplies to the standby gas treatment rooms and the normal operation of air intake and exhaust ducts are dampered closed.

The multizone air-handling unit, the chiller, chilled water pump and the return air fan continue to operate as during normal operation. The return air damper assumes a full-open position. Cooling water is supplied from the emergency equipment cooling water (EECW) system. The fan in the mechanical equipment room fan-coil cooling unit is also energized under room thermostat control. Chilled- water flow through the cooling coil of the unit continues unimpeded as during normal operation.

The emergency recirculation air fan is energized and the dampers on the emergency intake air duct are opened and the kitchen and washrooms exhaust fans are deactivated.

The emergency recirculation air filter train consists of a prefilter, a high-efficiency particulate air (HEPA) filter, a charcoal filter, another HEPA filter, and redundant fans. The emergency makeup air filter train consists of a filtering-type demister, two electric heaters, a HEPA filter, a charcoal filter, and another HEPA filter. Detailed descriptions of the filter trains are presented in Subsection 9.4.1, with summary descriptions of the train components given below:

a. The demister removes entrained water droplets and serves as a prefilter for the downstream HEPA filter. The demister meets design requirements specified in Savannah River Laboratory Report DP-812

b. The electric heater reduces the relative humidity of influent air under worst conditions to 70 percent or less

c. The HEPA filters have a design DOP filtration efficiency of 99.97 percent for particles 0.3 μm in diameter or larger. The elements meet the requirements of ANSI N509-1980. They are UL-approved fire resistant and suitable for service under the temperatures and mass peak loadings expected. The filters are installed and field tested such that a 95 percent decontamination efficiency can be assumed for removal of particulate iodine.

d. The charcoal adsorber in the emergency makeup air filter train is a deep-bed unit, as is that in the recirculation air filter train. These units contain impregnated activated carbon. Representative samples of the carbon are lab tested prior to installation and periodically while in service to demonstrate that the carbon is 99 percent efficient in removing methyl iodide. The carbon is installed and field tested for by-pass leakage such that a 95 percent decontamination efficiency can be assumed for removal of all forms of gaseous iodine.

e. Downstream HEPA filters are identical to the upstream HEPA units and serve to trap charcoal fines and decay daughters entrained in the air stream.

The CCACS design provides redundant alarms for main control room high/low pressure to ensure that a positive pressure in the main control room is maintained at all times. Additionally, alarms are provided in the control room to alert operators for large pressure drops across the CCHVAC emergency make-up and recirculation filters indicating the filter airflow is degrading.

Chlorine Mode

In the event that chlorine gas is detected, control room personnel will place CCHVAC in the chlorine mode, whereupon the normal intake and discharge isolation dampers would close and the emergency intake isolation dampers would remain closed; all other dampers and equipment would function as described in the recirculation mode. In this mode, airflow is circulated throughout the control center at the emergency flow rate, but the outside air intake and exhaust ducts are closed by dampers.

For all operating modes, damper position indications in the main control room allow continuous monitoring of the system performance and confirm all remote manual control actions taken.

Purge Mode

The air conditioning system has a smoke purge mode. In this mode, fresh air is brought into the main control room and no air is recirculated. This mode is initiated automatically when the gaseous fire suppression system actuates, or it can be initiated manually by the operator.

Ionization-type smoke detectors provide an alarm indicating conditions that require isolation of the control center.

For heat and smoke removal from the control center complex in the event of a fire in any of the air-conditioning zones, the fire- detection system will activate alarms in the control center. The control room operator can remote manually initiate the purge mode.

#### 6.4.2.3.2 Control Center Air Intakes

The control center air conditioning system consists of two 100 percent-capacity air-conditioning supply units, an air- distribution system, and an emergency filtration system. The control center is heated, cooled, and pressurized by a recirculating air system. Figures 9.4-1 and 9.4-2 show the ventilating air circulating flow path, rates, and dampers and their positions for the different operating modes. The emergency filtration system processes control center recirculated air and makeup air through charcoal filters if the control center is subjected to airborne radioactive contamination. This system consists of two separate emergency air intakes. The intake that draws from the area having the lowest level of contamination is manually selected for operation.

The physical orientation between the normal and emergency intake openings and the potential source points of radiation are described in Section 2.3.4.2.4.

Each emergency intake has two parallel paths containing redundant dampers. One path in each intake contains two Division I isolation dampers and one Division I modulating damper, which maintains the control center at approximately $1/4 \pm 1/8$-in. water gage positive pressure in the recirculation mode. The other path contains identical dampers powered by Division II.

For normal operation, a separate normal intake supply is used, allowing the makeup and emergency filters to remain on standby with full filtering capacity available for emergencies. Two air- operated isolation dampers are provided on the normal air intake duct and on the system exhaust vent. One damper in each duct is designated as a Division I damper; the other damper in each duct is designated as a Division II damper. Each damper will close within 5 sec after an isolation signal is initiated and is designed to achieve "bubble-tight" full shutoff.

Two return air fans are provided and each fan is sized to return 95 percent of the total air supplied to the control center. One fan is for Division I and the other for Division II. In the normal mode, the exhaust air damper is modulated to maintain approximately 1/4 ± 1/8 in. of water difference between the lower of the outside ambient pressure or the turbine building pressure and the control center pressure when the system is in the normal operating mode.

#### 6.4.2.4 Fire Protection System

The fire protection system is described in detail in Subsection 9.5.l. Portable fire extinguishers are provided at strategic locations in the main control room. High-sensitivity, ionization-type detectors for combustion products are located in the ceiling space. The fire protection system and specific construction materials are identified in the fire hazards analysis referenced in Subsection 9.5.1.

#### 6.4.2.5 Personnel Protective Equipment and First Aid and Emergency Supplies

Operator respiratory protection in the main control room consists of a mask-hose apparatus connected to a bottled air supply. The supply is 3600 $ft^3$, which is adequate for 30 manhours of heavy work, with a 20 percent contingency. The size of the supply is based on the data supplied in Reference 3. This handbook indicates that an adult man performing heavy work requires 39.3 to 45.2 liters of air per minute. The size of the supply is based on the larger of these figures.

The supply consists of a rack outside the main control room containing 12 air cylinders (300-$ft^3$) connected to a manifold. This supply is piped to a five connection manifold in the main control room. Located at the manifold are five individual dual purpose airline/self-contained breathing apparatus (SCBA) respirator units with a length of hose adequate to permit operator movement throughout the main control room area.

The dual purpose airline/SCBA respirator may be attached to either the emergency air system via the manifold, or function independently via the on-board air supply. This provides the operators the capability to move about, and exit the main control room. Two spare bottles per SCBA are also maintained adjacent to the main control room. These units are of the type tested and approved by the Bureau of Mines and by the National Institute for Occupational Safety and Health (NIOSH), and supply fresh air for a period of approximately 20 minutes.

Possible first aid needs are met by a kit in the main control room.

A five-man-day supply of food is stored in the main control center complex and can be used for an emergency. In addition, sufficient potable water is reserved to provide a five-man-day supply during an emergency. Under the conditions that would exist in the event of long-duration accidents, the main control room is accessible for shift changes so that additional food and water can be brought to the main control room.

### 6.4.2.6 Utilities and Sanitation

The plant communications system is described in Subsection 9.5.2. This diverse system, which includes telephones, portable two-way radios, an intercom system, and a public address system, provides assurance that there is a means of communication between the main control room and plant or offsite areas.

Subsection 9.5.3 contains a description of the normal lighting system and the emergency lighting system. The design criteria and failure analysis ensure that these systems, in conjunction with the power supply system (Section 8.3), will provide adequate lighting for the main control room.

The kitchen area of the main control room contains an electric range, a refrigerator, a water heater, and cooking and eating utensils. The main control room washroom contains toilets, washing facilities, housekeeping supplies, and waste containers.

## 6.4.3 Design Evaluation

Operating systems that serve to ensure main control room habitability are discussed in detail in the following sections and subsections.

a. Control center air conditioning system - 9.4.l

b. Fire protection system - 9.5.l

c. Communications system - 9.5.2

d. Lighting system - 9.5.3

e. Onsite power systems - 8.3

f. Radiation monitoring systems - 12.1.4 and 12.2.4.

As the referenced sections and subsections state, the systems or portions of systems essential for main control room habitability meet the seismic, the component redundancy, and the power-supply redundancy requirements that ensure satisfactory performance under normal and accident conditions.

Summary evaluations of the designs of the systems that contribute to main control room habitability are provided in the following subsections.

### 6.4.3.1 Radiation Monitoring System

The design of the radiation monitoring equipment essential for main control room habitability meets all of the functional requirements given in Subsection 6.4.2.2. The monitor locations, types, sensitivities, ranges, and setpoints ensure that necessary information is available to main control room personnel and that those main control room habitability systems, which are actuated automatically, will receive initiation signals.

The portable radiation monitoring equipment applicable to main control room habitability is readily available as required. The equipment is described in Subsection 12.3.2, and personnel dosimetry is discussed in Subsection 12.3.4.

#### 6.4.3.2 Air Conditioning System - Control of Main Control Center Thermal Environment

The CCACS is operated on a continuous basis to maintain a safe and comfortable thermal environment in the main control room. The state of readiness of this system is indicated by the system performance, as reflected in the main control room temperature and relative humidity. With the exception of the common ductwork and filters, the system has 100 percent backup and meets single- failure criteria. The smoke/Halon dampers to the relay room, cable spreading room or computer room will close due to a single active failure in the Halon fire protection system. Sufficient time is available to take manual action to reestablish airflow. The air conditioning system emergency mode and all essential components are designed to Category I requirements.

The air-conditioning functions provided include cooling, heating, humidification, air filtration, forced air circulation, exhaust, and positive pressure control. In normal operation, air filtration is provided by an electronic air cleaner and a fiberglass media roll filter. After the mixture of return and outside air is filtered, it is cooled and heated by the multizone air-handling unit.

The air conditioning system, sized in accordance with ASHRAE recommendations, is designed for an ambient temperature of 95°F dry bulb and 75°F wet bulb during summer operation and -10°F dry bulb for winter operation. The ambient temperature range specified prevails 99 percent, or more, of the total time at the plant location.

The total system flow is 31,510 cfm, of which 11,350 cfm is supplied directly to the main control room. The supply of conditioned air ensures that the thermal environment within the main control room permits habitation under any weather or plant conditions.

#### 6.4.3.3 Air Conditioning System - Control of Main Control Room Airborne Radioactivity

During an emergency, the control center is isolated from all other areas of the plant. All air supply to the standby gas treatment rooms and the normal-operation air intake and exhaust ducts are dampered or valved closed. The multizone air-handling unit and the return air fan continue to operate as during normal operation, with the emergency air-handling system placed in operation by automatic or manual opening of the emergency air intake dampers and energizing the emergency recirculation air fan.

The 1800 cfm maximum emergency outside air required for pressurization and personnel physiology is drawn through the emergency outside air intake filter train. A mixture of filtered outside air and the emergency recirculation air is passed through the emergency recirculation filter train. The kitchen and washroom exhaust air ducts will be closed during emergency operation.

Airborne and fuel pool radioactivity levels in the reactor/ auxiliary building ventilation and exhaust air ducts are monitored. If the activity exceeds acceptable levels, isolation dampers or valves in the control center normal intake and exhaust air ducts and in the air conditioning equipment and standby gas treatment system (SGTS) room air ducts are actuated, placing the control center air conditioning system in an emergency recirculation mode. In this mode, air is brought in through the emergency makeup air filter train (1800 cfm maximum), mixed with 1200 cfm minimum recirculated air, and put through a 3000 cfm recirculation filter train.

Redundant and separate isolation dampers or valves are installed in all ducts of the air conditioning system that affect the isolation of the main control room from other building areas; the emergency intake air duct and the kitchen and washroom exhaust air ducts are also equipped with redundant isolation dampers.

The system design, the isolation capabilities, and the efficiencies of the components used in the emergency filter trains ensure that airborne radioactivity in the main control room does not rise to levels that prohibit habitability.

#### 6.4.3.4 Air Conditioning System - Control of Main Control Room Chemical Environment

Adverse chemical effects on the main control room environment could result from the following three events:

a. A chlorine accident off the plant site

b. A fire outside the main control room

c. A fire inside the main control room

There are shipments of hazardous chemicals by rail and road routes within a 5-mile radius of the plant. The closest transportation line lies about 3.5 miles from the plant. At this distance, a release of a hazardous chemical is not a threat to Fermi 2 control room habitability. In accordance with the provisions of Regulatory Guide 1.78, Revision 1, control center habitability was analyzed for the rupture of a 90-ton chlorine railroad tankcar.

It was determined that the probability of a chlorine railcar accident and a spill resulting in a control room toxic concentration meets the Regulatory Guide criterion for not considering such scenario to be a credible event (Reference 6).

The CCACS ensures that the toxic or noxious substances that might result from one of the above events do not prevent occupancy of the main control room.

Upon manual initiation of chlorine mode, the (100 percent recirculation) chlorine mode of operation of the air conditioning system commences; in this configuration, there is no makeup airflow and approximately 1200 cfm of the main control room airflow is passed through the recirculation air filter train for cleanup.

The safety of main control room operators is further ensured by the provision of self-contained breathing apparatus units in the main control room, as described in Subsection 6.4.2.5. Storage provisions for the breathing apparatus and procedures for use permit operators to don appropriate respirators upon detection of toxic gases or chlorine odors. The emergency plan includes instructions for immediate donning of breathing apparatus on detection of chlorine release, and the training of main control room personnel includes rehearsal and the procedures necessary for rapid utilization of the equipment.

A fire-detection system is provided throughout the control center. The system consists of ionization or photoelectric detectors for alarming the presence of smoke or for actuating the automatic gaseous fire-suppression systems where provided.

In the chlorine mode, introduction of smoke and/or noxious fumes from outside fires into the main control room is prevented; in the unlikely event of a UL Class A fire inside the main control room, the smoke purge system is used to remove the products of the fire from the main control room.

For smoke purging, the normal air conditioning system can be operated on a zero-recirculation basis, with a greatly increased outside air intake. As the zero-recirculation terminology implies, airflow in the control center areas is on a once-through basis.

In summary, the CCACS is highly flexible, providing modes of operation that ensure acceptable air quality.

#### 6.4.3.5 Fire Protection

A description of the fire protection system for the main control room is identified in Subsection 9.5.1.

#### 6.4.3.6 Personnel Protection

Self-contained breathing apparatus is provided for emergency use in the main control room. The apparatus is selected according to the guidelines of ANSI Z88.2 (Reference 4). A respiratory protective program meeting the requirements of Occupational Safety and Health Administration (OSHA) 1910.134 (Reference 5) has been established and will be maintained, thereby ensuring the effectiveness of the provisions for personnel protection.

#### 6.4.3.7 Utilities and Sanitation

Several communications channels are maintained open under all conditions. Most of the communications systems are in routine use. Those systems not frequently used are subjected to periodic maintenance and testing to ensure their state of readiness. The provision of diverse and redundant systems ensures a reliable communications capability.

Lighting is provided in the main control room at all times. The installation of normal and emergency systems, with power-source redundancy, ensures that the main control room is adequately illuminated. The normal lighting system is proven operable during regular operating periods and will continue to operate under most accident conditions; in the event that the normal system is inoperative, the emergency system provides illumination.

Kitchen and sanitary facilities are proven operable under normal conditions and will continue to function under accident conditions.

### 6.4.4 Testing and Inspection

### 6.4.4.1 Radiation Monitoring System

Testing and inspection of the RMS ensure that each functional requirement of the system is met. The RMS is tested in conjunction with the CCACS to ensure that the monitors perform the desired functions.

The area and process radiation monitors are readily accessible for testing, inspection, and calibration. The testing of the monitors does not interfere with normal operation of the habitability systems for the main control room. Portable equipment such as air samplers, personnel dosimeters, and other radiation analysis equipment applicable to main control room habitability is tested and inspected periodically as required.

Specific details of the measures taken to ensure the operability of radiation monitoring equipment are given in Subsections 12.1.4, 12.2.4, and 12.3.4.

### 6.4.4.2 Control Center Air Conditioning System

The CCACS is subjected to those tests and inspections required to ensure its capability to perform its designed functions throughout the lifetime of the plant. As indicated in the preceding subsection, those portions of the system that interact directly with other systems are subjected to testing concurrent with the other systems.

The system and its components are tested in accordance with the codes and standards to which they are designed, and with the tests and inspections specified in Section 9.4, the Technical Specifications, and the Technical Requirements Manual. The compliance of the emergency makeup air and emergency recirculation filter trains and their components with Regulatory Guide 1.52 is described in Subsection A.1.52. Testing of the filter trains and their components involves

a. Predelivery and component qualification tests

b. Onsite preoperational acceptance tests

c. Operational surveillance tests

The quantity of air supplied for pressurization of the control center is determined by performing a duct traverse measurement at the installed test ports in the ductwork. The static pressure differential in the control center complex is measured to verify that a pressure of 1/4 ± 1/8-in. water gage is maintained by the CCACS operating in the emergency mode.

Should a component or material in the CCACS fail to meet the required level of performance, the component or material is replaced. Should the system fail to meet performance standards in any mode of operation, the component(s) adversely affecting the system performance is replaced. The modes of operation considered for the main control room are normal mode, recirculation mode (radiation emergency), chlorine mode (complete isolation), and purge mode (smoke removal with zero recirculation).

### 6.4.4.3 Main Control Room Fire Protection System

Fire protection for the main control room is ensured by fire extinguishers inside the main control room, fire-detection equipment, the smoke purge capability of the CCACS, and the isolation provisions of that system. Inspection and testing requirements are provided in Subsection 9.5.1.

#### 6.4.4.4 Other Control Center Habitability Systems

Self-contained breathing apparatus is inspected to ensure that pressures are at least equal to those required to supply air for the minimum acceptable breathing period. If cylinder pressure is insufficient, the cylinder is recharged or replaced. Regulators in the air packs are periodically inspected to verify operability; units that do not function properly are repaired or replaced.

The communications and lighting systems are proven operable, in part, by normal use, with backup or emergency facilities tested periodically by individual tests or intentional disabling of the primary system.

Kitchen and sanitation facilities are known to be operable through normal use.

### 6.4.5 Instrumentation

The individual system design sections of the UFSAR contain descriptions of the instrumentation used for monitoring and actuating those portions of the systems vital to main control room habitability. Design details and logic of the instrumentation are discussed in Chapter 7.

## 6.4 HABITABILITY SYSTEMS

## REFERENCES


1. Report of ICRP Committee II on Permissible Dose for Internal Radiation (1959), Health Physics Journal, Vol. 3 (June 1960).
2. American Society of Heating, Refrigerating and Air- Conditioning Engineers (ASHRAE), 55-66, Standard on Thermal Comfort Conditions.
3. P. L. Altman, J. F. Gibson, and C. C. Wang, Handbook of Respiration, W. B. Saunders Company, 1958.
4. American National Standards Institute, ANSI F88.2-1969, Respiratory Protection.
5. Occupational Safety and Health Administration, Title 29, Part 1910.134, Respiratory Protection.
6. License Amendment No. 147, “Elimination of the Chlorine Detection Function from the Control Center Heating, Ventilation and Air Conditioning System”, dated June 26, 2002.

Figure Intentionally Removed
Refer to Plant Drawing A-2100

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UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE 6.4-1

PLOT PLAN - GRADE ELEVATION 583 FT