5.2 PRIMARY CONTAINMENT SYSTEM

5.2.1 Safety Objective

The safety objective of the Primary Containment System is to provide the capability, in the event of the postulated loss-of-coolant accident, to limit the release of fission products to the plant environs so that offsite doses would be within the guideline values of 10 CFR 50.67.

5.2.2 Safety Design Basis

1. The Primary Containment System shall have the capability to withstand the peak transient pressure which could occur due to the postulated loss-of-coolant accident, i.e., a mechanical failure of the Reactor Primary System equivalent to the circumferential rupture of one of the main recirculation pipes.

2. The containment design basis for metal-water reactions and other chemical reactions subsequent to the postulated loss-of-coolant accident shall be consistent with the performance objectives of the Reactor Core Standby Cooling System.

3. The Primary Containment System shall have the capability to maintain the functional integrity of the system indefinitely after the postulated loss-of-coolant accident.

4. The containment design shall be adequate to permit filling the primary containment vessel with water above the reactor core.

5. The Primary Containment System shall be designed to provide means to rapidly condense the steam portion of the flow from the postulated rupture of a recirculation line so that the peak transient pressure shall be substantially less than containment design pressure.

6. The Primary Containment System shall be designed to provide means to conduct the flow from postulated pipe ruptures to the pressure suppression pool, to distribute such flow uniformly throughout the pool, and to limit pressure differentials between the drywell and the pressure suppression chamber during the various post-accident cooling modes.

7. The Primary Containment System shall have the capability of limiting leakage during and following the postulated accident to values which are substantially less than leakage rates which would result in offsite doses approaching the reference doses in 10 CFR 50.67.

8. The Primary Containment System shall have the capability to conduct periodically such leakage tests as may be appropriate to confirm the integrity of the containment at the peak transient pressure resulting from the postulated accident.

9. The Primary Containment System shall have the capability to withstand jet forces associated with the flow from the postulated rupture of any pipe within the containment.

10. The Primary Containment System shall provide the capability for rapid closure or isolation of all pipes or ducts which penetrate the primary containment by means which provide a containment barrier in such pipes or ducts as effective as is required to maintain leakage within permissible limits.

11. The primary containment shall have the capability of being purged with nitrogen to reduce and maintain the containment atmosphere to less than 4 percent oxygen.

12. The primary containment shall have the capability to withstand the hydrodynamic loading resulting from a postulated LOCA or main steam relief valve actuation.

5.2.3 Description

5.2.3.1 General

Each unit employs a pressure suppression containment system which houses the reactor vessel, the reactor coolant recirculating loops, and other branch connections of the Reactor Primary System. The pressure suppression system consists of a drywell, a pressure suppression chamber (alternatively referred to as the torus or wetwell) which stores a large volume of water, a connecting vent system between the drywell and the pressure suppression chamber, isolation valves, containment cooling systems, equipment for establishing and maintaining a pressure differential between the drywell and pressure suppression chamber, and other service equipment. In the event of a process system piping failure within the drywell, reactor water and steam would be released into the drywell air space. The resulting increased drywell pressure would then force a mixture of air, steam, and water through the vents into the pool of water which is stored in the pressure suppression chamber. The steam would condense rapidly and completely in the pressure suppression pool, resulting in rapid pressure reduction in the drywell. Air that is transferred to the pressure suppression chamber pressurizes the chamber and is subsequently vented to the drywell to equalize the pressure between the two vessels. Cooling systems are provided to remove heat

from the drywell and from the water in the pressure suppression chamber, thus cooling the primary containment, when required, under accident conditions. Appropriate isolation valves are actuated during this period to ensure containment of radioactive materials within the primary containment which might be released from the reactor during the course of the accident. If long-term cooling capability is lost, resulting in a pressure increase that would jeopardize the structural integrity of the primary containment, the hardened containment venting system (HCVS) would relieve the corresponding pressure increase. The Primary Containment System is designed as a Class I system (except the drywell pressure suppression chamber pressure differential subsystem -Reference Section 5.2.3.9) in accordance with Appendix C, "Structural Qualification of Subsystems and Components," to the BFNP FSAR.

5.2.3.2 Drywell

The drywell is a steel pressure vessel with a spherical lower portion 67 feet in diameter, and a cylindrical upper portion 38 feet 6 inches in diameter. The overall height is approximately 115 feet. The design, fabrication, inspection and testing of the drywell vessel comply with requirements of the ASME Boiler and Pressure Vessel Code, 1965 edition, Section III, Class B, which pertain to containment vessels for nuclear power plants. The steel head and shell of the drywell are fabricated of SA-516 plate.

The drywell is designed for a maximum internal pressure of 62 psig coincident with a temperature of 281°F, plus the dead, live, and seismic loads imposed on the shell. Thus, in accordance with the ASME Boiler and Pressure Vessel Code, Section III, the drywell design pressure is 56 psig. Thermal stresses in the steel shell due to temperature gradients were taken into account in the design.

Special precautions not required by codes were taken in the fabrication of the steel drywell shell. The plate was preheated to a minimum temperature of 200°F prior to welding of all seams whose thickness exceeds 1-1/4 inches, regardless of surrounding air temperature. Preheat at a minimum of 100°F was applied prior to welding of all seams 1-1/4 inches or less in thickness if the ambient temperature fell below 50°F. Charpy V-notch specimens were used for impact testing of plate and forging material to give assurance of proper material properties. Plates, forgings and pipe associated with the drywell have an initial NDT temperature of 0°F or lower when tested in accordance with the appropriate code for the materials. It is contended that the drywell will not be pressurized or subjected to substantial stress at temperatures below 30°F.

The drywell is enclosed in reinforced concrete for shielding purposes to provide additional resistance to deformation and buckling of the drywell over areas where the concrete backs up the steel shell. Above the transition zone, the drywell is

separated from the reinforced concrete by a gap of approximately 2 inches. This gap is filled with polyurethane foam.

Irradiation tests have shown that no change in the resilient characteristics will take place for exposures up to 10^8R.

The polyurethane foam filler is 2-1/4 inches maximum thickness polyester-based, flexible foamed slab having the following physical properties.

Density	0.116 lb/ft^3
Compression set	10 percent at 50 percent compressibility
Compressibility	35 percent at 1.0 psi maximum
Resilience	20 percent minimum
Service temperature	285°F maximum

The sizing of the expansion gap, in which the foam slab was placed, was based on an ultimate steel shell temperature of 281°F and an internal pressure of 56 psi following a postulated reactor loss-of-coolant accident. The maximum pressure and thermal growth as related to compression of the foam slab occurs at the bottom of the juncture of the spherical lower portion and cylindrical upper portion. The total vertical growth was calculated to be 1.02 inches and the horizontal growth 0.36 inches. For the normal operating condition at a temperature of 150°F and pressure of 2.0 psi, the growth at the bottom of the knuckle was calculated to be 0.33 inches vertical and 0.14 inches horizontal.

In the event of a loss-of-coolant accident, the foam slab would be compressed about 50 percent and the maximum permanent set would be less than 0.1 inch. There is no objection to the substitution of space for foamed slab. Shielding over the top of the drywell is provided by removable, segmented, reinforced concrete shield plugs.

In addition to the drywell head, one double door airlock, one bolted CRD removal hatch, and two bolted equipment hatches are provided for access to the drywell. The doors are mechanically interlocked so that neither door may be operated unless the other door is closed and locked. The drywell head and hatch covers are bolted in place and sealed with gaskets. The seals on the hatches and drywell head are designed with double gaskets with intermediate leak taps and are thus capable of being tested for leakage.

The drywell is not normally entered during MODE 1, but access is permissible during MODE 2 or MODE 3. Provisions are made to supply breathing apparatus to personnel while in the drywell, if necessary. During normal reactor operation, the drywell is essentially at atmospheric pressure. The design temperature (during normal operation) is 150°F bulk volumetric average DW temperature at 100°F RBCCW supply water. This is the maximum value used for the initial drywell temperature at the beginning of a LOCA event. The actual operating temperature is less than this value and is specified in the plant operating procedures. This temperature is maintained by recirculating the drywell atmosphere across forced draft air cooling units which, in turn, are cooled by the Reactor Building Closed Cooling Water System. (See Subsection 10.6.)

Provisions made for protection of the drywell against missiles and pipe whipping which could damage the primary containment are discussed in paragraph 5.2.4.6.

The primary containment vessels are designed to permit flooding as a means of post-accident recovery following a loss-of-coolant accident. A water volume of approximately 250,000 cubic feet is necessary to accomplish containment flooding to above core level. It is necessary to vent the containment vessels during the reflooding process to prevent overpressurization. The standby coolant supply used to flood the primary containment is discussed in Subsection 4.8, "Residual Heat Removal System."

The exposed portions of the interior of the drywell were originally sandblasted and provided with a protective coating consisting of an inorganic zinc primer (Amercoat Dimetcoat 6) with an epoxy topcoat (Amercoat 66). Any repairs or replacement of this protective coating system are performed using other coating system(s) that are design basis accident qualified to ANSI N101.2-1972 and approved by TVA.

It is expected that this coating system will satisfactorily withstand temperatures and pressures of the steam environment postulated during a design basis loss-of-coolant accident as described in Section 14.0 of the FSAR. Test panels were exposed to steam-water atmospheres under comparable or greater temperatures and pressures with negligible deterioration. In some cases discoloration and small (<1/8 inch) blisters were observed.

5.2.3.3 Pressure Suppression Chamber and Vent System

5.2.3.3.1 General Description

The vent system, which connects the drywell and pressure suppression chamber, conducts flow from the drywell to the pressure suppression chamber without

excessive resistance and distributes this flow effectively and uniformly in the pool following a postulated pipe rupture in the drywell.

The pressure suppression chamber receives this flow, condenses the steam portion of this flow, and contains non-condensable gases and fission products driven into the pressure suppression chamber. The pressure suppression chamber-to-drywell vacuum breakers limit the pressure differential between the drywell and pressure suppression chamber. The pressure suppression chamber is designed for the same leakage rate as the drywell.

Large vent pipes form a connection between the drywell and the pressure suppression chamber. A total of eight circular vent pipes are provided, each having a diameter of 6.75 feet.

The vent pipes are designed for an internal pressure of 56 psig (the ASME Boiler and Pressure Vessel Code, Section III, allows a maximum internal pressure of 62 psig) coincident with a temperature of 281°F and are designed to withstand an external pressure of 2 psi above internal pressure. Jet deflectors are provided in the drywell at the entrance of each vent pipe to prevent possible damage to the vent pipes from jet forces which might accompany a pipe break in the drywell. The vent pipes are fabricated of SA-516 steel, and comply with requirements of the ASME Boiler and Pressure Vessel Code, Section III, Subsection B. Expansion joints are provided on each vent pipe to accommodate differential motion between the drywell and pressure suppression chamber.

The pressure suppression chamber is a steel pressure vessel in the shape of a torus below and encircling the drywell, with a centerline diameter of approximately 111 feet and a cross-sectional diameter of 31 feet. As a result of the torus modifications on all three units, the maximum water volume in the torus is approximately 131,400 ft^3 (see note on Table 5.2-1). The pressure suppression chamber is held by supports which transmit vertical and seismic loading to the reinforced concrete foundation slab of the Reactor Building. Space is provided outside the pressure suppression chamber for inspection and maintenance. The eight drywell vents are connected to a 4-foot, 9-inch diameter vent header in the form of a torus which is contained within the airspace of the pressure suppression chamber. Projecting downward from the header are 96 downcomer pipes, 24 inches in diameter, and terminating approximately 3 feet 10 inches (maximum, see note on Table 5.2-1) below the water surface of the pool. T-quenchers have been added to replace the ramshead discharge devices at the end of the main steam relief valve discharge pipes to assure the controlled release and condensation of steam and reduce stresses on the torus and tailpipe assemblies. The vent header and vent pipes have the same temperature and pressure design requirements as the pressure suppression chamber. Vacuum breakers discharge from the pressure suppression chamber into the drywell to limit the pressure differential

and to prevent a backflow of water from the pressure suppression pool into the vent header system. Vacuum breaker sizing is based on Moss Landing test configurations.

The system to establish and maintain a controlled pressure differential between the drywell and pressure suppression chamber during normal operations is described in paragraph 5.2.3.9.

The pressure suppression chamber is designed to the same material and code requirements as the steel drywell vessel. All attachments to the torus are by full penetration welds.

The HCVS can mitigate the consequences of a severe accident that would cause the pressure of the torus to exceed 56 psig by venting primary containment. The HCVS connects the torus of each unit to independent vents which discharges above the Unit 1, 2, and 3 Reactor Buildings.

During each refueling and each shutdown for required maintenance inside the containment, the containment is purged to restore a normal air atmosphere and to reduce the amount of gaseous and airborne radioactivity present. These purges are accomplished through the ventilation purge connections and are normally passed through a containment purge filter train (HEPA and charcoal filters) before release through the normal reactor building ventilation system. A vent from the primary containment is provided which will normally be closed, but which will permit the vent discharge to be routed to the Standby Gas Treatment System so that release of gases from the primary containment is controlled, and so that effluents are filtered and monitored before dispersal through the stack.

A 30-inch ECCS suction header with a wall thickness of 1/2-inch minimum circumscribes the pressure suppression chamber at El. 525 feet 4 inches. Four 30-inch tees are used to connect the suction header to the pressure suppression chamber. The suction header is supported vertically and horizontally by brackets attached to the 16 cradles.

Four strainers on connecting lines between the suction header and the pressure suppression chamber have been provided. The strainers are a stacked disk design having a large surface area to accommodate debris that may be generated by the dynamic forces in a LOCA and other debris that may be resident in the containment such as sludge and paint chips. The strainer design is governed by debris generation assumptions in accordance with the Boiling Water Reactor Operating Group Utility Resolution Guidance for ECCS Suction Strainer Blockage (GE NEDO-32686, R0, Dated November 1, 1996). The strainers are designed to

provide acceptable head loss while saturated with reflective metal insulation from primary system piping combined with other debris.

The suction header and its connecting pipes are designed, constructed, tested, and inspected in accordance with the same requirements as the pressure suppression chamber. Additional safety is provided by locating the four connecting pipes in unused portions of the pressure suppression chamber so that they will not be directly subjected to the water jet issuing from the downcomers. The suction header is designed to accommodate a temperature differential between itself and the pressure suppression chamber. Hydraulic snubbers are used to support the suction header to provide seismic supports that will prevent any abrupt lateral movement due to earthquake, but will not offer resistance to relatively slow thermal expansion. The suction lines from the RHR, HPCI, and Core Spray systems are supplied from this header. The RCIC System is also supplied from this header via the Core Spray System supply piping.

The interior carbon steel surface of the pressure suppression chamber and all other exposed carbon steel surfaces within the pressure suppression chamber were originally coated for corrosion protection with Valspar Hi-Build Epoxy 78:00. This coating has passed test criteria for a design basis accident (DBA) for carbon steel substrate as outlined in ANSI N101.2-1972. Any repairs or replacement of this protective coating system are performed using other coating system(s) that are design basis accident qualified to ANSI N101.2-1972 and approved by TVA. The RCIC and HPCI turbine exhaust spargers may be uncoated carbon steel.

Additionally the following stainless steel components and structures within the Unit 2 and Unit 3 pressure suppression chambers have been found to be coated:

1. T-quenchers
2. Main Vent Bellows
3. Miscellaneous support steel on the torus walkway
4. Electrical Conduits
5. Electrical Junction Boxes
6. Small bore piping and valve bodies

Based on the qualification of the Valspar 78 coating, the controlled surface preparation and application of the coating on the stainless steel components by qualified individuals, the in-place adhesion testing of the coating, and the degree of resiliency exhibited by the coating to different removal methods, TVA has concluded that Valspar 78 will behave the same on stainless steel as it will on carbon steel when applied properly. Therefore, no disbonding of the coating is expected during design basis accident conditions.

For Unit 1, coatings on stainless steel components in the pressure suppression chamber were either removed or are accounted for as unqualified coatings.

For all three units, unqualified coatings are maintained within margins with respect to ECCS Suction Strainer sizing assumptions.

5.2.3.3.2 Description of the Pressure Suppression Pool (Torus) Modifications

The original design of the BFNP containment system considered postulated accident loads previously associated with a loss-of-coolant accident (LOCA), seismic loads, dead loads, jet-impingement loads, hydrostatic loads (due to water in the pressure suppression chamber), overload pressure test loads, and construction loads.

However, since the establishment of the original design criteria, additional loading conditions were identified. These conditions were discovered as a result of GE performing large scale testing of the Mark III containment design and in-plant testing of Mark I containments. New pressure suppression pool loads which had not explicitly been included in the containment design bases were identified. These loads result from dynamic effects of drywell air and steam being rapidly released to the pressure suppression pool (torus) during a postulated LOCA and during main steam relief valve discharge associated with plant transient operating conditions. As a result, the Mark I Owners Group was established, with GE serving as the lead technical organization.

The efforts of the Owners Group were to be accomplished in two phases: a short term program (STP) and a long term program (LTP). The objective of the STP was to verify that the Mark I containment and related structures were capable of sustaining the additional hydrodynamic loads before the more detailed results of the LTP were available. The STP structural acceptance criteria used to evaluate the design of the torus and related structures were based on providing adequate margins of safety, i.e., a safety-to-failure factor of 2, to justify continued operation of the plant. A report on torus main steam relief valve piping and vent header support modifications (Kaiser Engineer Report No. 75-83-R, dated October, 1975) was submitted to the NRC on December 18, 1975.

The NRC staff's conclusions relative to the STP are documented in NUREG-0408, dated December, 1977. Subsequently, exemptions relating to the structural factor of safety requirements of 10 CFR 50.55(a) were granted by the NRC, while the more comprehensive LTP was being conducted.

The Mark I Owners Group developed the "Mark I Containment Program, Program Action Plan" and submitted it to the NRC in March, 1977. As a result of discussions with NRC, Revision 1 to the Program Action Plan was submitted February 11, 1977 (GE Topical Report EW 7610.09).

TVA initiated analysis activity for the BFNP torus integrity long term program in March, 1977.

GE and the Mark I Owners Group members worked together to develop load definitions and structural acceptance criteria (for the LTP) that were generically acceptable to the NRC. A partially complete Load Definition Report was issued by GE in December, 1978. In March, 1979, the initial Load Definition Report (GE Topical Report NEDO-21888, "Mark I Containment Program Load Definition Report") was submitted to the NRC for review. A series of experimental and analytical programs were conducted by the Mark I Owners Group to provide the bases for the generic load definition and structural acceptance criteria. The NRC issued NUREG 0661 (Safety Evaluation Report for the Long Term Program) in July, 1980.

This Safety Evaluation Report included Revision 1 of the NRC acceptance criteria for the Mark I containment LTP. GE subsequently revised and published application guides for load definitions in September, 1980 (including revisions to NEDO-21888 and NEDO-24583). The approved structural acceptance criteria (NUREG-0661, Appendix A), include 27 different load combinations for which the entire torus, vent system, torus internals, and attached piping must be analyzed. BFN unique analyses have been completed and associated modifications to the torus and related structures, systems, and components have been installed. These modifications meet the requirements of NUREG-0661.

In addition, a plant-unique main steam relief valve (MSRV) discharge test has been performed as part of the BFNP unique analysis, as requested by the NRC in NUREG-0661. This test confirmed the methods used to calculate containment loads from the various MSRV discharge cases. These tests were performed after all torus modifications significantly affecting measured torus motion effects were in their final configuration.

The final configuration of torus integrity modifications is described in the Browns Ferry Nuclear Plant Torus Integrity Long Term Program Plant Unique Analysis Report first submitted to NRC in January 1984 and supplemented by submittals in September 1984 and January 1985. See FSAR Sections C.2.5, C.3.5, and C.5.3 for additional information on this subject.

The operation of the units in their partially modified state has been evaluated. The modifications improve the margin of safety of the torus.

5.2.3.4 Penetrations

5.2.3.4.1 General

In order to maintain designed containment integrity, containment penetrations have the following design characteristics:

a. They are capable of withstanding the peak transient pressure which could occur due to the postulated rupture of any pipe inside the drywell.

b. They are capable of withstanding the forces caused by impingement of the fluid from the rupture of the largest local pipe or connection without failure.

c. They are capable of accommodating the thermal and mechanical stresses which may be encountered during all modes of operation without failure.

The number and size of the principle primary containment penetrations and associated isolation valves are shown in Table 5.2-2.

5.2.3.4.2 Pipe Penetrations

Four types of pipe penetrations are utilized as required by stress conditions. Type A is used where stress levels would exceed the allowable design limits if a bellows were not used. The design permits leak-testing of the bellows during plant operation. Type B is used where stress levels would not exceed the allowable design limits if a bellows were not used. These types of penetrations are illustrated in Figure 5.2-3. Where stress levels are within the allowable design limits without the use of Types A or B, the penetration assemblies illustrated in Figures 5.2-4 or 5.2-4a are suitable.

The piping penetrations for which movement provisions are made are the high temperature lines such as the steam lines and other reactor system lines. The penetration sleeve passes through the concrete and is welded in the primary containment vessel. The process line that passes through the penetration is free to move axially, and a bellows expansion joint is provided to accommodate the movement. A guard pipe immediately surrounds the process line and is designed to protect the bellows and maintain the penetration seal should the process line fail within the penetration. Insulation and air gaps are provided to reduce thermal stress.

If necessary, these lines are anchored outside the containment to limit the movement of the line relative to the containment. The bellows accommodates the relative movement between the pipe and the containment shell. This design is

utilized to assure reasonable integrity of the flexing penetration during plant operation.

The steam line, as it passes through the drywell containment vessel and the concrete biological shield, is enclosed in a guard pipe that is attached to the main steam line through a multiple head fitting. This fitting is a one-piece forging with integral flues or nozzles and is designed to meet all requirements of the ASME Boiler and Pressure Vessel Code, Section III. The forging is radiographed and ultrasonically tested as specified by this code. The guard pipe and fittings are designed to the same pressure requirements as the steam line. The steam line penetration sleeve is welded to the drywell and extends through the biological shield, where it is welded to a bellows which, in turn, is welded to the guard pipe. The bellows assembly accommodates the relative thermal expansion of the steam pipe and drywell. The steam pipe is guided through pipe supports at each end of the penetration assembly to allow steam pipe movement parallel to the penetration and to limit pipe reactions of the penetration to allowable stress levels. Two isolation valves are provided.

The external valve is located as close to the drywell penetration sleeve as practical, and the inside valve is located downstream of the reactor vessel main steam relief valves.

The design of the penetration takes into account the simultaneous stresses associated with normal thermal expansion, live and dead loads, seismic loads, and loads associated with a loss-of-coolant accident within the drywell. For failure of the steam pipe taken at random, the design takes into account the loadings given above in addition to the jet force loadings resulting from the failure. For these conditions, the resultant stresses in the pipe and penetration components do not exceed the code allowable design stress.

The cold piping and ventilation duct penetrations are welded directly to the sleeves. (See Figure 5.2-4.) Bellows and guard pipes are not necessary in this design, since the thermal stresses are small and are accounted for in the design of the weld joints. Small, low-pressure lines which do not connect to the reactor system are run through larger pipes sealed by welded end plates. The plates are drilled, tapped, and equipped with compression fittings. (See Figure 5.2-4a.) Connections are provided for leak testing of this type penetration.

5.2.3.4.3 Electrical Penetration

The electrical penetrations include electrical power, signal and instrument leads. A typical penetration is shown in Figure 5.2-5. The penetration assembly consists of header plates welded into the assembly at each end to form a double pressure barrier.

The electrical conductors are hermetically sealed into the header plates with insulating material to form a leak tight configuration which is leak tested after installation and provides a means for periodic testing. The penetration assembly is welded to the containment nozzle by a single weld ring.

5.2.3.4.4 TIP Penetrations

TIP guide tubes pass from the Reactor Building through the primary containment. The insertion guide tubes pass through double O-ring sealed, flanged penetrations on the primary containment. The guide tubes are connected in the flanged penetration by means of brazing, which meets the requirements of the ASME Boiler and Pressure Vessel Code, Section VIII. These seals would also meet the intent of Section III of the code, even though the code has no provisions for qualifying the procedures or performances. The flanged TIP penetrations are bolted to flanged nozzles which are welded to the primary containment.

5.2.3.4.5 Personnel and Equipment Access Locks

One personnel access lock is provided for access to the drywell. The lock has two sealed doors in series, and 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 at all times when primary containment is required. The locking mechanisms will hold the doors tight against the seals, and door design will ensure a tight seal when the doors are subjected to design accident pressure. The space between the air-lock doors can be pressurized to test for leakage through the door seals.

An access hatch with double testable seals is provided on the drywell head. This hatch is bolted in place.

Two 12-foot diameter equipment access hatches with double, testable seals are also provided. These hatches are bolted in place. One 2-foot diameter CRD removal hatch with double, testable seals is provided. The hatch is bolted in place.

5.2.3.4.6 Access to the Pressure Suppression Chamber

Access to the pressure suppression chamber is provided at three locations from the Reactor Building. There are two 4-foot diameter and one about 3 1/2-foot diameter hatches with double testable seals and bolted covers. The access hatches will be bolted closed when primary containment is required and will be opened only when the primary coolant temperature is $\leq 212°F$ and the pressure suppression function of the torus is not required to be operational.

5.2.3.4.7 Access for Refueling Operations

The top portion of the drywell vessel is removed during refueling operations. The head is held in place by bolts and is sealed with a double seal arrangement. The head is bolted closed when primary containment is required, and will be opened only when the primary coolant temperature is ≤ 212°F and the pressure suppression function of the torus is not required to be operational.

The double seal on the head flange provides a method for determining the leak tightness after the drywell head has been installed.

5.2.3.5 Isolation Valves

The criteria governing isolation valves for the various categories of penetrations are as follows. (See Subsection 7.3, "Primary Containment and Reactor Vessel Isolation Control System".)

a. Pipes or ducts which penetrate the primary containment and which connect to the reactor primary system, or are open to the drywell free air space, are provided with at least two isolation valves in series.

   Excluding check valves and closed manual valves, valves in this category are designed to close automatically from selected signals and shall be capable of remote-manual actuation from the control room. (See Table 5.2-2.)

b. The two valves are physically separated. On lines connecting to the reactor primary system, one valve is located inside the primary containment and the second outside the primary containment as close to the primary containment wall as practical.

c. Lines that penetrate the primary containment, and which neither connect to the reactor primary system nor are open into the primary containment, are provided with at least one valve which may be located outside the primary containment. Valves in this category are capable of manual actuation from the control room.

d. Motive power for the valves on process lines which require two valves are physically independent sources to provide a high probability that no single accidental event could interrupt motive power to both closure devices. Loss of power to each electrical division is detected and annunciated.

e. For design basis breaks in a main steam line downstream of the outboard main steam isolation valve, isolation valve closure time is such that the valve will be closed prior to the start of uncovering the fuel.

f. Valves, sensors, and other automatic devices essential to the isolation of the containment are provided with means to periodically test the functional performance of the equipment. Such tests include demonstration of proper working conditions, correct setpoint of sensors, proper speed of responses, and operability of fail-safe features.

g. The control circuits for the isolation valves are designed so as to prevent the valves from automatically reopening when primary containment isolation logic is reset.

h. The leakage from HCVS Primary Containment Isolation Valves FCV-064-0221 and FCV-064-0222 is designed, tested, and maintained to ensure the Control Room and offsite dose limits of 10 CFR 50.67 are not exceeded following a design basis accident.

The following are exceptions to the above isolation valve criteria.

a. Automatic isolation valves are not provided on the outlet lines from the pressure suppression chamber to the core spray and RHR pumps. These lines return to the containment and are required to be open during post-accident conditions for operation of these systems.

b. No automatic isolation valves are provided on the Control Rod Drive Hydraulic System lines. These lines are isolated by means of the normally closed hydraulic system control valves located in the Reactor Building, and by means of check valves comprising a part of the drive mechanisms.

c. TIP isolation valves and small diameter instrument lines.

d. Automatic isolation signals are not provided to the two isolation valves for the HCVS. These valves are closed during all design basis accidents/events. Both valves will be maintained closed by separate remote key-locked permissive control switches in the Main Control Room to prevent power from being supplied to the solenoid valves which serve the valve operators. They fail closed on loss of electrical power and/or pneumatic supply. These valves are only opened for surveillance or to operate the HCVS (purposely violating primary containment) to preserve the structural integrity of the torus.

Table 5.2-2 is a listing of the principal isolation valves. The table indicates the number and service of the valves, the motive power which actuates the valve, and the closure time of the valve.

The main steam lines have air-powered valves. Studies have shown this arrangement to have a high reliability with respect to functional performance. These valves are closed automatically by the signals indicated in Table 5.2-2. (See Subsection 4.6, "Main Steam line Isolation Valves.")

Influent lines, such as the feedwater lines which connect to the reactor vessel, have one check valve inside and one check valve or motor-operated isolation valve outside the primary containment. An AC operator is chosen for the motor-operated valves, since the motor is simpler in construction and is assessed as having higher overall reliability than a DC motor for the same service. The check valves close automatically by reverse flow through the pipe.

TIP System guide tubes are provided with an isolation valve which closes automatically upon receipt of proper signal and after the TIP cable and fission chamber have been retracted. Manual operator intervention to reset the insertion logic for the TIP system is required in the event a Group 8 isolation signal causes the TIP ball valves to isolate upon withdrawal of the probe. This feature ensures containment integrity is maintained in the event of design basis accident. In series with this isolation valve, an additional, or back-up, isolation shear valve is included. Both valves are located outside the drywell. The function of the shear valve is to assure integrity of the containment even in the unlikely event that the present isolation valve should fail to close or the chamber drive cable should fail to retract, if it should be extended in the guide tube during the time that containment isolation is required. This valve is designed to shear the cable and seal the guide tube, if necessary, upon a manual actuation signal. Valve position (full open or full closed) of the automatic closing valves is indicated in the control room. Closing of the shear valves will be performed by operator action from the control room. Each shear valve will be operated independently. The valve is an explosive-type valve, DC operated, with monitoring of each actuating circuit provided.

In the event of a containment isolation signal, the TIP System receives a command to retract the traveling probes for the five machines. Upon full retraction, the isolation valves are then closed automatically. If a traveling probe were jammed in the tube run such that it could not be retracted, this information would be supplied to the operator who would, in turn, investigate the situation to determine if the shear valve should be operated.

The Unit 2 TIP System purge line to the Indexing Mechanisms contains two check valves in series for containment isolation. Both check valves are outside containment.

Lines such as the closed cooling water lines, which neither connect to the reactor primary system nor are open into the primary containment, are provided with at

least one remotely operable valve located outside the primary containment, or a check valve on the influent line outside the containment.

Instrumentation piping connecting to the reactor primary system which leaves the primary containment is dead ended at instruments located in the Reactor Building except for the reactor recirculation sample line for the PASS (see Section 10.21). These lines are provided with manual isolation valves and an excess flow check valve. The reactor recirculation sample line for the PASS is taken from a jet pump instrument line downstream of the excess flow check valve with integral restricting orifice. This small (1/2-inch, schedule 80) line is normally isolated near the tie-in point on the jet pump instrument line by a remote manual solenoid valve controlled from the main control room. This solenoid valve would only be open during periodic testing of the PASS or during PASS sampling operations, post-accident, when the reactor is at high pressures. For large break LOCA's where reactor vessel pressure may not be sufficient to provide sufficient head to obtain a sample from this tie-in, the PASS connection on the RHR system (see Section 10.21) would be used. Although not performing a strict containment isolation function, this solenoid valve will be local leak rate tested since the sample line can communicate with the environment through the PASS sample panel located in the turbine building.

Instrumentation piping, which opens into the drywell and pressure suppression chamber and whose external branches terminate in dead end service and are capable of withstanding drywell design conditions, utilize one locally operated block valve.

The Containment Atmospheric Dilution inlet lines to the drywell and pressure suppression chamber contain a solenoid operated valve and a check valve. Both valves are located outside primary containment.

All isolation valves (except non-testable check valves, RHR/LPCI System I and System II, and Core Spray System I and System II inboard isolation check valves and H_2O_2 CAM sampling isolation valves) are provided with limit switches which are used to indicate in the control room that the valves are either open or closed. For the isolation valves in the sampling and sample return lines in the H_2O_2 CAM System, the valve position is identified in the control room by indication of energization of the solenoid valves.

Note:
For Units 1, 2, and 3, the RHR check valve actuator, controls and indication functions have been deleted.

All power actuated isolation valves are capable of being actuated from the control room. The ECCS inboard isolation check valves can only be actuated during cold shutdown (MODE 4 or MODE 5) when control air is connected.

5.2.3.6. Primary Containment Venting and Vacuum Relief (Figures 5.2-2a - Sheets 1, 2, and 3)

The drywell is maintained at approximately 1.3 psid by the delta-P compressor and the suppression pool is maintained slightly above atmospheric during normal plant operation in order to ensure that containment pressures are maintained within design limits. The containment is periodically vented to eliminate pressure fluctuations caused by temperature changes during various operating modes. This is accomplished through vent connections which normally discharge to the SBGT system. When the reactor is at a temperature greater than 212°F (Modes 1, 2, or 3) the inboard vent valves are normally open and the outboard vent valve is normally closed. The pressure suppression chamber is vented separately to SBGT.

The purge valves leading to ductwork on Units 1, 2, and 3 are designed to isolate within 2.5 seconds to allow purging during operation, as limited by the technical specifications. (See FSAR, paragraph 5.3.3.6.3) The closure time for valve 76-24 is equal to or less than five seconds. This valve is in a hard-piped system, and a five-second closure time meets NRC Branch Technical Position CSB 6-4.

Protective screens have been installed before the first isolation valve outside primary containment in the containment purge piping to comply with CSB 6-4, to protect the valves from debris which might affect isolation.

Automatic vacuum relief devices are used to prevent the primary containment from exceeding the external design pressure. The drywell vacuum relief valves draw air from the pressure suppression chamber. The pressure suppression chamber vacuum relief device draws air from the Reactor Building.

The pressure suppression chamber vacuum relief system consists of two vacuum breakers in series in each of two lines to atmosphere. One valve is air-operated and is actuated by a differential pressure signal; upon loss of electrical power or air, the valve will fail in the open position. The second valve is self-actuating. The combined pressure drop at rated flow through both valves does not exceed the difference between pressure suppression chamber design external pressure and maximum atmospheric pressure.

The force required to open each self-actuating vacuum breaker is periodically determined in accordance with Technical Specification 3.6.1.5 surveillance requirements. A visual inspection of the self-actuating vacuum breakers is also performed as part of the surveillance procedure.

The drywell vacuum breakers, which connect the pressure suppression chamber and drywell, are sized on the basis of the Bodega Bay pressure suppression

system tests.[1] The vacuum breaker flow area is proportional to the flow area of the vents connecting the drywell and pressure suppression pool. Their chief purpose is to prevent excessive water level increase in that portion of the vent discharge lines that is submerged in pressure suppression pool water, which results in a pressure differential across the vent pipe. The Bodega Bay tests regarding 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 (twelve 18-inch valves) selected on this test basis is more than adequate to limit the pressure differential between the pressure suppression chamber and drywell during post-accident drywell cooling operations to a value which is within suppression system design values.

The drywell vacuum breaker valves FCV-64-28A through 28M, are located in the pressure suppression chamber and use silicone rubber as the seat material. These valves function to prevent steam from bypassing the pressure suppression chamber, thus they assure pressure suppression. The silicone valve seats are functional for the temperature and exposure dose for several hours. The material approaches its radiation damage limit at 10^7 rad after several hours; however, by this time the valves have performed their function of assuring pressure suppression.

5.2.3.7 Primary Containment Normal Heating, Ventilation, and Air Conditioning Systems

Maintaining each drywell average ambient temperature ≤ 150°F during normal plant operation assures that the insulation on motors, isolation valves, operators and sensors, instrument cable, electrical cable and gasket materials or sealants used at the penetrations has a sustained life without deterioration.

Each drywell is cooled during normal operation of the unit by a closed loop ventilation system designed to hold the average temperature in the drywell to ≤ 150°F. (See Subsection 10.6, "Reactor Building Closed Cooling Water System".) The atmosphere is circulated in the drywell normally by eight fans assembled in two groups of five, with one spare in each group (ten fans total). Spare fans may be placed in service at operator discretion to provide additional margin. Each fan has an individual cooling coil associated with it. Water from the Reactor Building Closed Cooling Water system is employed to remove heat from the coolers.

The drywell blowers and Reactor Building Closed Cooling Water System (RBCCWS) are kept in service upon extended loss of offsite power. This serves

1 Bodega Bay Preliminary Hazards Summary Report, Appendix 1, Docket 50-205, December 28, 1962

to prevent high drywell temperatures and associated equipment damage in units that have not sustained a loss-of-coolant accident (LOCA). Continued operation of the RBCCWS is desirable because this system provides the preferred method of cooling the drywell; however, the RBCCWS is not essential to safety. In the event of failure of the RBCCWS and drywell blowers such that the drywell temperature in non-LOCA units could exceed the design value, the operator has sufficient time to manually initiate drywell cooling using torus water, the RHR pumps, and containment spray headers.

The loss of offsite power leads to the drywell blowers being unavailable for a short term. The worst case unavailability is when a 480-V load shed signal is received where restarting of the drywell cooling is delayed until the diesel generator loading will accept loading of the blowers. This leads to the conservative estimates of drywell temperature and pressure for the short term transients in the drywell under normal operating conditions for the non-accident unit (e.g., all equipment operates as expected). Emergency power reactivates the cooling which immediately terminates the temperature and pressure rises.

For Units 1 and 2, six of the blowers are automatically returned in a staggered sequence from 40 to 90 seconds. While maintaining diesel generator load limits, as required, additional blowers are manually returned to service. The drywell pressure increases above 2.45 psig (drywell high-pressure setpoint) for a short period of time (satisfying half of the logic for an accident signal); however, the full accident signal is not created because normal reactor pressure is above the 450 psig low reactor pressure limit. The drywell air-space temperature remains below maximum containment design temperature limit.

For Unit 3, eight of the drywell blowers are automatically returned after 40 seconds. The drywell pressure remains below the 2.45 psig limit above which satisfies half of the logic for an accident signal (the other half is low reactor pressure below 450 psig), and the drywell air-space temperature remains below the maximum containment design temperature.

In consideration of single failures, adequate drywell cooling may not be available to prevent a high drywell pressure signal from being received. In order to prevent a full accident signal from being received when reactor pressure reduces to the low reactor pressure setpoint on the units that have not sustained a LOCA, operator action can be taken to bypass the high drywell pressure signal. In this scenario, HPCI is initiated upon receiving the high drywell pressure signal and maintains reactor level. The drywell air-space temperature remains below maximum containment design temperature limit (Subsection 5.2.3).

Separate fans located outside the drywell are used to purge the drywell prior to entering this area for maintenance work. (See Subsection 5.3, "Secondary Containment System.") The Primary Containment Ventilation System is shown in Figures 5.2-2a - Sheets 1, 2, and 3.

5.2.3.8 Containment Inerting System

Following each startup (within 24-hours after thermal power is > 15% rated thermal power), the primary containment is purged of air with pure nitrogen until the atmosphere contains less than 4 percent oxygen. The Containment Inerting System is used during the initial purging of the primary containment and provides a supply of makeup nitrogen. The system consists of two liquid nitrogen storage tanks with two makeup vaporizers, a common purge vaporizer, pressure reducing valves and controllers, instrumentation valves and piping as shown in Figures 5.2-6a sheets 1, 3, 4, and 6.

Nitrogen is supplied from the common onsite storage tanks through the common purge vaporizer or makeup vaporizers where the liquid nitrogen is converted to the gaseous state. The gaseous nitrogen then flows through the purge or makeup pressure-reducing valves and flow meters into each containment pressure suppression chamber or drywell, where it mixes with the air. A safety valve in the nitrogen supply system prevents overpressurization of the containment.

The purge supply piping is configured such that it is possible to establish a large bypass path from the drywell to the pressure suppression chamber. If this path is established, then the pressure suppression function of the primary containment could be compromised. Administrative controls prevent the simultaneous purging or inerting of the drywell and the pressure suppression chamber except when the unit is at cold shutdown (MODE 4 or MODE 5).

The drywell ventilation blowers are normally operated during the purge operation to maximize mixing of the nitrogen and air. Gases purged from the containment are vented either through the Reactor Building Ventilation System (containment purge) or the Standby Gas Treatment System.

The Reactor Building exhaust route will be used for radioactivity releases of low concentration. When releasing drywell atmosphere through the building exhaust, the radioactivity release is filtered by both HEPA and charcoal adsorbers, monitored, and recorded by the plant ventilation exhaust radiation monitoring system (FSAR paragraph 7.12.6). Radioactivity in the exhaust would also be detected by the Reactor Building Ventilation Radiation Monitoring System (FSAR paragraph 7.12.5). High radioactivity would result in primary and secondary containment isolation and automatic startup of the Standby Gas Treatment System. The stack exhaust route will result in the effluent being processed by the

HEPA filters and charcoal adsorbers in the Standby Gas Treatment System. The processed stream is then monitored by the main stack radiation system (FSAR paragraph 7.12.3) and released through the plant stack.

Purging continues until the oxygen content of the containment atmosphere is less than 4 percent as measured by the oxygen analyzer (Figures 5.2-6a sheets 2, 5, and 7). This takes approximately 4 hours and requires 3 to 5 containment atmosphere volumetric changes.

The inerting system also continues to supply makeup gas, required by temperature changes and leakage, during planned operations. This makeup includes gas vented from pneumatic equipment inside the drywell that utilizes nitrogen supplied from the Drywell Control Air System. The primary containment is held at a slight positive pressure by the inerting system as a means of leak-rate monitoring. The atmosphere is monitored and results are recorded in the Main Control Room. Both the purge and makeup operations of the inerting system are controlled from the control room. A high oxygen concentration alarm is located in the control room.

5.2.3.9 Drywell-Pressure Suppression Chamber Pressure Differential System

Each unit has a system to maintain a controlled pressure differential between the drywell and the pressure suppression chamber. This system consists of a compressor connected into the primary containment purge line to form a loop connecting the drywell and the pressure suppression chamber. Nitrogen is pumped from the pressure suppression chamber to the drywell to create the pressure differential. The compressor is an Ingersoll-Rand, non-lubricated type capable of providing 136 SCFM at a maximum discharge pressure of 125 psig. Details of the connections into the primary containment purge line showing the associated isolation valves and location of the compressor within the system are shown on Figures 5.3-3a, 5.3-3c, and 5.3-3d.

The system is set to establish an operating pressure difference between the drywell and the pressure suppression chamber in the range of 1.1 to 1.35 psi, with the drywell at the higher pressure. Pressure differential control is operated by either of two independent channels. A 0-2 psid pressure transmitter provides the determination of system pressure differential. Water level control within the pressure suppression chamber is also conducted with either of two transmitters and indicators.

The system is not a safety system and therefore is automatically isolated in the event of a LOCA. The purpose of the drywell-pressure suppression chamber pressure differential system is to reduce the thermo-hydrodynamic loads imposed on the pressure suppression chamber during a blowdown following a LOCA.

5.2.4 Safety Evaluation

5.2.4.1 General

The primary containment and its associated safeguards systems are designed to accomplish four principal functions, namely:

a. To accommodate the transient pressures and temperatures associated with the equipment failures within the containment,

b. To accommodate and mitigate the effects of potential metal-water reaction subsequent to postulated accidents involving loss of coolant,

c. To provide a high integrity barrier against leakage of any fission products associated with these equipment failures, and

d. To provide containment protection against damaging effects of missiles.

These factors are considered in the following evaluation of the integrated Primary Containment System.

5.2.4.2 Primary Containment Characteristics During Reactor Blowdown

In order to establish a design basis for the pressure suppression containment with regard to pressure rating and steam condensing capability, the maximum rupture size of the reactor primary system must be defined. For this design, an instantaneous, circumferential rupture of one 28-inch recirculation line has been selected as a basis for determining the maximum gross drywell pressure and the condensing capability of the pressure suppression system. The selection of an equipment failure of this size for the design basis is entirely arbitrary, since circumferential failure of a recirculation pipe is considered to be of such low probability as to be considered incredible. Nevertheless, for design purposes these failure conditions have been selected to establish the containment parameters, but the failure modes and the magnitude of failures are assessed as being incredible.

The design pressure is established on the basis of the Bodega Bay pressure suppression tests. The design pressure is primarily a function of the postulated rupture area, the drywell to pressure suppression chamber vent area and configuration, vent submergence below the water level in the pressure suppression pool, and the final equilibrium pressure in the pressure suppression chamber.

In establishing the containment design, circumferential pipe ruptures are assumed with sufficient distance separation to allow full potential flow from each end of the pipe. For pre-uprate conditions, normal pipeline flow losses are not considered in establishing rupture flow rates.

The containment design parameters listed in Table 5.2-1 are concerned primarily with the effects on the primary containment caused by the blowdown immediately following the postulated double-ended rupture of the recirculation piping.

The parameters having the greatest effect on drywell design pressure are the ratio of pipe break area to total vent area, the vent submergence below the water level in the pressure suppression pool, initial system pressure, and the equilibrium pressure in the pressure suppression chamber before the postulated rupture.

Sufficient water is provided in the pressure suppression pool to accommodate the initial energy which can be transiently released into the drywell from the postulated pipe failure. The pressure suppression chamber is sized to contain this water, plus the water displaced from the reactor primary system together with the free air initially contained in the drywell.

The difference in the key parameters is either in the conservative direction or results in second order effects on the peak pressure, leading to the conclusion that the design will result in significantly lower pressure peaks than those measured.

The primary containment response analysis to the design basis loss-of-coolant accident is presented in Section 14.0, "Plant Safety Analysis."

The break area assumed (for the purpose of calculating the containment peak transient pressure and establishing the break vent area ratio) was 4.2 ft^2. This is equivalent to the total area of 10 jet pump injection nozzles, and the recirculation suction line in the broken loop. This area gives a break-to-vent area ratio which is within the range tested during the Bodega Bay series of pressure suppression tests. In calculating the peak pressures for pre-uprate conditions, no credit has been taken for pipe friction, the pump, and flow nozzle which will significantly reduce the flow. Under all operating conditions (Unit 2 only), one valve in the equalizing line between the two reactor recirculation pump discharge pipes shall be open and the other valve shall be closed. The reactor recirculation equalizing line and associated valves have been removed for Units 1 and 3.

5.2.4.3 Primary Containment Characteristics after Reactor Blowdown

After the blowdown of the primary coolant into the drywell immediately following the recirculation line break, the temperature of the pressure suppression chamber

water approaches 152ºF. The maximum primary containment system pressure is 49.1 psig. Most of the non-condensable gases would be transported to the pressure suppression chamber during the blowdown. However, soon after initiation of the drywell spray, they would redistribute between the drywell and pressure suppression chamber via the vacuum breaker system as the spray reduces drywell pressure.

Redundant drywell instrumentation is provided in the control room for the operator. Drywell temperatures are printed out, and the readings of any two separated sensors can be used to determine the temperature of the drywell atmosphere to alleviate the problem of local variations. High drywell temperature (160°F, Process Limit) is annunciated in the control room. Drywell pressure is indicated and recorded in the control room and torus pressure (about 2 psi lower reading, but considered redundant indication) is indicated in the control room. High drywell pressure (20 PSIG, Nominal Setpoint) is annunciated in the control room. Annunciation is also provided if either of two criteria is exceeded: (1) high drywell pressure (35 PSIG, Process Limit), or (2) high drywell pressure (2.5 PSIG, Process Limit) exists for approximately 30 minutes coincident with high drywell temperature (281°F, Process Limit). This information is provided to the operator, through both indication and alarm, for manual initiation of containment spray in accordance with the emergency operating instructions. Therefore, the containment temperature will be limited to its design value for all loss of coolant accidents.

The Unit 1, Unit 2, and Unit 3 Torus Water Level Monitoring Instruments LT-64-159A and LT-64-159B measure torus water level from 2 feet above the bottom of the torus (below the lowest ECCS suction) to 5 feet above normal level. The torus level monitors are to be used to supply information to the operator for use in determining the nature of an accident and for better understanding of post-accident conditions for Unit 1, Unit 2, and Unit 3.

The Core Spray System removes the decay heat and stored heat from the core, thereby minimizing core heatup and any metal-water reaction. The core heat would be removed from the reactor vessel through the broken recirculation line in the form of hot liquid. This hot liquid would combine with liquid from the drywell spray and flow into the pressure suppression chamber via the drywell to pressure suppression chamber connecting vents.

Steam flow would be negligible. The energy transported to the pressure suppression chamber water would be removed from the primary containment system by the RHR system heat exchangers in the containment cooling mode.

In order to assess the primary containment response after the blowdown, primary containment analysis was performed. The result of these analyses are presented in Section 14.12, “Analysis of the Primary Containment Response.”

5.2.4.4 Primary Containment Capability

The pressure of the primary containment system depends on both the system temperatures and the amount of non-condensable gases. Thus, the capability of the system to house resulting gases from metal-water reaction varies with the rate and extent of the reaction.

Capability is defined as the maximum percent of fuel channels and fuel cladding material which can enter into a metal-water reaction during a specified duration without the design pressure of the containment structure being exceeded. The analysis of the postulated loss-of-coolant accident, discussed in Section 14.0 and Section 6.0, shows that the operation of either of the two Core Spray Systems will maintain continuity of core cooling such that the extent of the resultant metal-water reaction would be less than 0.1 percent. However, to evaluate the containment system design capability, various percentages of metal-water reaction were assumed to take place over various durations of time. This analysis presents an arbitrary method of measuring system capability without requiring prediction of the detailed events in a particular accident condition. The results are presented in Section 14.0.

5.2.4.5 Primary Containment Leakage Analysis

The primary containment for each unit is constructed in such a manner that it can be verified initially that, at the maximum pressure resulting from the design basis accident, the leakage rate is not in excess of 2.0 percent per day of the free volume of the primary containment (L_a). Two tests were performed. The initial test was performed at a reduced pressure of 25 psig (P_t) to determine the leakage rate (L_{tm}). The second test was performed at 49.6 psig (P_a) to measure the leakage rate (L_{am}). The leakage characteristics yielded by measurements L_{tm} and L_{am} were used to establish the maximum allowable reduced pressure leakage rate (L_t). To verify primary containment integrity throughout the service life of the unit, periodic leakage rate tests will be performed. Details of the leakage rate tests are provided in the Primary Containment Leakage Rate Testing Program as referenced by Technical Specification, Section 5.5.12.

5.2.4.6 Missile and Pipe Whip Prevention

In the design of the primary containment and of the components therein, special consideration has been given to missile and pipe whip prevention under the assumed accident conditions. The following summarizes the pertinent design considerations.

The containment penetrations and isolation valves are protected from pipe whip by anchors located at or near the isolation valves.

Large pipes that penetrate the containment are designed so that, if necessary, they have anchors or limit stops located outside the containment to limit the movement of the pipe. These stops are designed to withstand the jet forces associated with the postulated break of the pipe and thus maintain the integrity of the containment.

The space between the containment vessel and the concrete is controlled, so that in areas which are not backed up by concrete and are subjected to jet forces the integrity of the containment will not be violated. Concrete backing is not available for the vent openings to the pressure suppression chamber and jet deflectors are put across these openings for jet protection.

The quality control of the fabrication of the pipe, the inspection of the pipe, and the conservative design of the pipe is given a high degree of attention. This approach to prevent pipe failure is substantiated by the long history in the utility industry,

during which time no such circumferential pipe failures have been recorded for the piping materials to be used for this plant.

If a pipe leak should occur, means for detecting even small leaks are available in the design so that proper action could be taken before they could develop into an appreciable break. Therefore, based upon the conservative piping design utilizing proven engineering design practice, the proper choice of piping materials, the use of conservative quality control standards and procedures for piping fabrication and installation, and extensive studies of modes of pipe failure, it is concluded that pipes will not break in such a manner as to bring about movement of the pipes sufficient to damage the primary containment vessel. Nevertheless, the recirculation lines within the primary containment are provided with a system of pipe restraints designed to limit excessive motion associated with pipe split or circumferential break. The design utilizes a number of supports and limit stops which permit thermal expansion of the pipe. Both types of breaks, the circumferential break or the longitudinal split, are considered in the support and limit stop arrangement.

Even though the emphasis has been placed on the prevention of the occurrence of a pipe whip, special care is also taken in component arrangements to see that equipment associated with Engineered Safety Systems, such as the core spray and the LPCI, are segregated in such a manner that the failure of one cannot cause the failure of the other. Both core spray lines enter the upper cylindrical portion of the drywell and are connected to the pressure vessel nozzles in an arrangement that precludes whip in one line from affecting the other line. In addition, the arrangement prevents whip of one recirculation line from affecting both core spray lines. Each LPCI injection loop injects coolant through separate portions of the recirculation system 180° apart. Each containment spray header and support structure is designed to withstand a load equivalent to the jet forces associated with a break of the largest pipe within the drywell. The containment spray headers are physically separated by 25 feet. With the exception of the incore monitoring detectors, sensors associated with the Reactor Protection System, including the drywell pressure detectors, are located external to the drywell and concrete structure, and are thus protected; however, the incore monitoring detectors are physically separated in the drywell, as are the sensing lines to the other aforementioned instruments. The redundant channels of reactor level and pressure sensing lines are located in the cylindrical section of the drywell 180° apart for maximum physical separation. The redundant channels of sensing lines are physically separated to the maximum possible extent and exit the drywell approximately 180° apart. Additionally, the control rod drive mechanisms are located in a concrete vault that provides protection.

In addition, the energy-absorbing material is added to the interior of the drywell surface to the maximum possible extent, in those areas potentially subject to damage from a circumferential break at a weld joint in mainsteam, feedwater, and

RHR piping, and limited only by existing installation. The extent of this coverage is shown in Figure 5.2-6e.

The energy-absorbing system absorbs the initial impact of the pipe section and distributes the force over a portion of the primary containment shell and biological shield wall concrete. The material is manufactured by the H. H. Robertson Company and is modified type "DFK" siding with 1/4-inch steel plate spot welded to each face. This composite material has been referred to as "Tornado Siding." The panel protection capability was tested by a dynamic method of striking the panels with a wood plank (4-inches x 12-inches at 105 lb), measuring its velocity at impact, and calculating the average energy absorption capacity. This total average energy was calculated to be 930,000 ft-lb/sq ft. Therefore, this siding is capable of absorbing approximately 1×10^6 ft-lb of kinetic energy/ft^2.

The siding (24-inches x 24-inches panels) is attached to (1/2-inch studs, welded) the steel containment pressure vessel. The use of small panels permits the material to follow the contour of the vessel. The material, or the design intent of this method of attachment, should not restrict access to piping welds or component welds for in-service inspection. The energy-absorbing system has a negligible effect on the free containment volume and no effect on the accident analysis.

Although it has been concluded that, with the application of conservative piping design and proven engineering practices, pipes will not break in such a manner as to bring about movement of pipes sufficient to damage the primary containment vessel, the design of the containment and piping systems does consider the possibility of missiles being generated from the failure of flanged joints, such as valve bonnets, valve stems, and recirculation pumps, and from instrumentation such as thermowells. The design philosophy is that there be no missiles which will penetrate the containment. This is accomplished in practice through the specific design of the containment and contained systems, which takes into account the potential for generation of missiles and minimizes the possibility of containment violation. In considering potential missile sources of this nature, none have been found against which further design action is required.

The most positive manner to achieve missile prevention is through basic equipment arrangement such that, if failure should occur, the direction of flight of the missile is away from the containment vessel. The arrangement of plant components takes this possibility into account even though such missiles may not have enough energy to penetrate the containment.

Analyses have led to the conclusion that if instruments, ejected thermowells, etc., should become missiles, they would not have sufficient energy to penetrate the containment. It has also been concluded that large, massive rotating

components, such as the reactor recirculating pump motors, would not have sufficient energy to move this mass to the containment wall.

The coolant flow from a double-ended break of the recirculation piping could cause a recirculation pump and motor to overspeed. Due to this overspeed possibility, a study was conducted by General Electric to assess the missile possibilities that could result from pipe breaks at various locations. The results of this study have been submitted to the AEC by General Electric in Licensing Topical Report NEDO-10677, Analysis of Recirculation Pump Overspeed in a Typical General Electric Boiling Water Reactor (October 1972). This report recommends the installation of a decoupling device between the pump and motor to prevent destructive motor overspeed. The GE document “Analysis of Recirculation Pump Under Accident Conditions” dated January 14, 1977, demonstrated that there is no need for protective equipment on the recirculation pumps in GE BWRs. Decoupling devices were not installed in the recirculation pumps at BFN.

A probability study was initiated by GE to see if additional restraints to maintain pipe alignment after the pipe break in order to contain the pump missiles were warranted. A summary of this study, "Probabilistic Analysis of the Effects of Missiles Formed in the Recirculation System Following Postulated Pipe Rupture," has been submitted as part of Docket No. 50-333 in FSAR Supplement 20 to the James A. FitzPatrick Nuclear Power Plant license application.

This study concluded that:

1. The cost would be prohibitive because of construction delays,

2. The combined probability of a LOCA and damage to the containment from a recirculation pump missile is sufficiently low to be identified as a Class 9 accident, and, therefore, no specific action to prevent its occurrence is required,

3. The probability of releasing additional radioactive material is slightly increased when recirculation pump missiles are considered, but the overall probability of a radioactive release is still extremely small, and

4. Incorporation of additional restraints would not provide substantially greater protection for the health and safety of the public, whereas the cost is disproportionally increased for the concomitant minimal increase in overall safety.

The assumptions in the analysis and the piping layout inside the containment are such that this study is conservative for the Browns Ferry Nuclear Plant. Thus, this probabilistic study is applicable without modifications. A similar study for missile

shields, to protect the containment and piping inside the containment, was not conducted because it would not differ substantially from this study. It would reach the same general conclusions previously reached for additional piping restraints. Therefore, neither additional restraints nor missile shields are warranted in the Browns Ferry Nuclear Plant.

Also, the pump impellers and motor rotors, upon failure, would be contained within their housings and would not generate missiles. There is the potential for valve bonnets to become missiles based on the assumption of failure of all bonnet bolts. This requires instantaneous, clean severance of all bolts, without any overturning motion. The damage potential is dependent upon the size of the valve and system in which the valve is located. Therefore, valve arrangement is important and is taken into consideration in the overall plant design.

In addition to the care with which equipment is oriented with regard to potential missile generation, special care is taken in component arrangements to see that equipment associated with Engineered Safety Systems, such as the core spray and the containment spray, is segregated in such a manner that the failure of one cannot cause the failure of the other, or that the failure of any component which would bring about the need for these Engineered Safeguard Systems would not render the safeguard system inoperable. In addition, each containment spray header and support structure is designed to withstand a load equivalent to the jet forces associated with a break of the largest pipe within the drywell. With the exception of the incore monitoring detectors, sensors associated with the Reactor Protection System, including the drywell pressure detectors, are located external to the drywell and concrete structure, and are thus protected. Additionally, the control rod drive mechanisms are located in a concrete vault that provides protection. The pressure suppression chamber has no source of internal or external missile generation, and the vent pipes joining it with the drywell are protected by the jet deflectors. The vent discharge headers and piping are designed to withstand the jet reaction force caused by flow discharge into the pressure suppression pool. Redundant subsections of vital systems are physically separated within the primary containment to minimize the probability that more than one redundant subsection could be damaged.

The drywell is completely enclosed in a reinforced concrete structure having a thickness of 4 to 6 feet. This concrete structure, in addition to serving as the basic biological shielding for the reactor system, also provides a major mechanical barrier for the protection of the drywell and reactor system against potential missiles generated external to the primary containment, including pipeline failures of the main steam, feedwater, HPCI steam, RCIC steam, and reactor water cleanup lines.

There are two segmented shield plugs above the drywell, each 3-ft thick, and separated vertically by 1 inch. The top corner, edges, and bottom corner of the

plugs are formed by 1/2-inch steel plate anchored by headed concrete anchors. Each plug is supported along the periphery by a ledge on the drywell concrete structure. The ledges are formed by the stainless steel plate which lines the reactor well cavity. Headed concrete anchors are used to anchor the plate at the ledges.

The segmented, reinforced concrete shield plug above the drywell has been analyzed for the impact of those missiles which have been postulated to reach the refueling floor level. These missiles are identified in GE Topical Report GE APED-5696, "Tornado Protection for the Spent Fuel Storage Pool." November 1968. The method of analysis is that given for structural design for impulsive loads, Chapter 3, Section IV, Department of the Army Technical Manual, TM 5-855-1, Fundamentals of Protective Design, July 1965.

The analysis shows that one 3-ft thick plug is capable of resisting the missile having the greatest impact energy. None of the postulated missiles will penetrate the plug to a depth that will cause scabbing of the underside.

To provide for possible movement as a result of operating and accident loads, 1/2-inch clearance is provided between the shield plugs and the drywell concrete structure. Frictional forces can be developed at the plug support points on the steel plate surfaces which form the edges of the plugs and the supporting ledges. The plugs and the ledges are capable of resisting the forces developed.

5.2.4.7 Penetrations

In order to minimize post-accident containment leakage, the containment penetrations are designed to withstand the normal environmental conditions which may prevail during plant operation, and to retain their integrity during the following postulated accidents.

Pipelines which penetrate the containment shell, and which are capable of exerting a reaction force due to line thermal expansion or containment movement which cannot be restrained by the containment shell, are provided with bellows expansion seals, appropriate guards, limit stops, or anchors as required to maintain stresses within allowable design limits. These design features are utilized to ensure integrity of the penetration during plant operation and during accident conditions.

Pipelines which penetrate the containment where the reactive forces can be restrained by the containment shell are provided with full strength attachment welds between the pipe and the containment shell. These penetrations are designed for long-term integrity without the use of a bellows seal. Electrical penetration assemblies require special design consideration to achieve zero

leakage because of the design restriction imposed by creepage characteristics of electrical insulation.

TIP guide tubes and their penetrations also require special design considerations, due to the fact that they are the means of passage from the interior of the reactor vessel to the Reactor Building for the TIP fission chamber and its electromechanical drive cable. The TIP guide tubes within the RPV are designed to ASME Boiler and Pressure Vessel Code, Section III. The drywell penetrations have double-seal testable flange, with the guide tube brazed to the flange as described in paragraph 5.2.3.4.4.

A personnel access lock is provided with interlocked double doors, so that access may be made to the containment while the reactor primary system is pressurized. Double doors are provided to assure that containment integrity is maintained while access is being made.

Equipment access hatches are sealed in place, using flexible double seals to assure leak-tightness. These openings are closed at all times when containment is required.

The containment shell, electrical penetrations, and piping penetrations are metallic components (with a ceramic filler, or equivalent, in the electrical penetrations) that are designed to pressure vessel standards; thus, no degradation will occur from temperature, pressure, or radiation damage.

Inspection and surveillance provide additional assurance of integrity and functional performance of the penetrations. For this reason, provisions are made to leak-test electrical penetrations, the personnel access lock, the access hatches, and those pipe penetrations having bellows seals. This can be accomplished without pressurizing the entire containment system. Provisions are made in the design of the integrated containment system to monitor for gross leakage of the primary containment to demonstrate that all penetrations are sealed during plant operation. This function is performed by the containment atmospheric control system, as described in paragraph 5.2.3.8.

5.2.4.8 Isolation Valves

One of the basic purposes of the Primary Containment System is to provide a minimum of one protective barrier between the reactor core and the environmental surroundings subsequent to an accident involving failure of the piping components of the reactor primary system. To fulfill its role as an insurance barrier, the primary containment is designed to remain intact before, during, and subsequent to any design basis accident of the process system installed either inside or outside the primary containment. The process system and the primary containment are considered as separate systems, but where process lines

penetrate the containment, the penetration design achieves the same integrity as the primary containment structure itself. The process line isolation valves are designed to achieve the containment function inside the process lines when required.

Since a rupture of a large line penetrating the containment and connecting to the reactor coolant system may be postulated to take place at the containment boundary, the isolation valve for that line is required to be located within the containment. This inboard valve in each line is required to be closed automatically on various indications of reactor coolant loss. A certain degree of additional reliability is added if a second valve, located outboard on the containment and as close as practical to it, is included. This second valve also closes automatically if the inboard valve is normally open during reactor operation. If a failure involves one valve, the second valve is available to function as the containment barrier. By physically separating the two valves, there is less likelihood that a failure of one valve would cause a failure of the second. The two valves in series are provided with independent power sources.

The ability of the steam line penetration and the associated steam line isolation valves to fulfill the containment safety design basis (paragraph 5.2.2), under several postulated conditions of the steam line, is shown below by consideration of various assumed steam line break locations.

a. The failure occurs within 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 a loss-of-coolant accident, except that the pressure transient is less severe since the blowdown rate is slower. Both isolation valves close upon receipt of the signal indicating low water level in the reactor vessel. 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 within the drywell and renders the inner isolation valve inoperable. Again, the reactor steam will blow down into the primary containment. The outer isolation valve will close upon receipt of the low water level signal, and the reactor becomes isolated within the primary containment, as above.

c. The failure occurs downstream of the inner isolation valve either within the drywell or within the guard pipe.

   Both isolation valves will close upon receipt of a signal indicating low water level in the reactor vessel. The guard pipe is designed to accommodate

such a failure without damage to the drywell penetration bellows, and the design of the pipeline supports protects its welded juncture to the drywell vessel. Thus, the reactor vessel is isolated within the primary containment by means of the inner isolation valve, and the primary containment integrity is maintained 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 outer isolation valve and the turbine.

The steam will blow down directly into the pipe tunnel or the Turbine Building. Steam releases into the tunnel are detected by temperature sensors. When these sensors detect a high temperature condition in the steam tunnel, they initiate main steam isolation. This action isolates the reactor, completes the containment integrity, and places two barriers in series between the reactor core and the outside environment. Pipe supports prevent containment damage. The offsite consequences of this failure are presented in the accident analysis as discussed in Section 14.0, "Plant Safety Analysis."

It should be noted also that the turbine stop valves, located in the steam lines just ahead of the turbine, will provide a backup containment barrier, in addition to the outer isolation valves, for such breaks as a, b, and c as discussed above.

The exceptions to the arrangement of isolation valves described above (1 inboard, 1 outboard), for lines connecting directly to the containment or reactor primary system, are made only in the cases where it leads to a less desirable situation because of required operation or maintenance of the system in which the valves are located. In the cases where, for example, the two isolation valves are located outside the containment, special attention is given to assure that the piping to the isolation valves has an integrity at least equal to the containment.

The TIP system isolation valves are normally closed. When the TIP system cable is inserted, the valve of the selected tube opens automatically and the chamber and cable are inserted. Insertion, calibration, and retraction of the chamber and cable require approximately 5 minutes. Retraction requires a maximum of 1-1/2 minutes. If closure of the valve is required during calibration, the isolation signal causes the cable to be retracted and the valve to close automatically on completion of cable withdrawal. A manually actuated shear valve is also provided in the event the cable cannot be withdrawn. Reinsertion of the TIP probe upon clearing of the Group 8 isolation signal requires manual operator intervention to reset the insertion logic.

It is not 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 which are open to the drywell, and some effluent process lines. However, under these conditions, it is essential that containment and Core Standby Cooling Systems be operable. For this reason, specific signals are utilized for isolation of the various process and safeguards systems (see Subsection 7.3).

Isolation valves must be closed before significant amounts of fission products are released from the reactor core under design basis accident conditions. Because the amount of radioactive materials 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 below the top of the core.

All of the primary containment system isolation valves, shown on FSAR Figures 5.2-2a sheets 1, 2, and 3, utilize either elastomer or metal seats. All the valves with elastomer seats are located outside the primary containment. The elastomer valve seats are designed to withstand the temperature and exposure dose for their locations. The elastomer seat isolation valves will maintain their structural integrity and leak tightness following a DBA. The isolation valves which utilize metal seats will maintain their structural integrity and leak tightness following a DBA.

Valves, sensors, and other automatic devices essential to the isolation of the containment are provided with means for periodically testing the functional performance of the equipment. Such tests are necessary to provide reasonable assurance that the containment isolation devices perform as required when called upon to do so.

5.2.4.9 Containment Inerting System

In the event of a loss-of-coolant accident, the Core Standby Cooling Systems prevent generation of significant quantities of hydrogen capable of being ignited. Maintaining the oxygen content of the primary containment atmosphere at less than 4 percent ensures no combustion of the hydrogen and oxygen, thus assuring containment integrity.

5.2.5 Inspection and Testing

The following represents areas of surveillance and testing that are provided for the various systems or components of the primary containments as they apply during construction or plant operation. Access is provided to conduct periodic in-service examinations of the primary containment boundary in accordance with the

applicable subsections of ASME Section XI Code. The area and type of examinations are contained in plant procedures.

5.2.5.1 Primary Containment Integrity and Leak Tightness

Fabrication procedures, nondestructive testing, and sample coupon tests are in accordance with the ASME Boiler and Pressure Vessel Code, Section III, Subsection B. Provisions were made to test the integrity of the primary containment systems during construction phases. These tests included a pneumatic test of the drywell and pressure suppression chamber at 1.25 times their design pressure in accordance with code requirements.

After installation of new penetrations in the drywell or suppression chamber, leakage tests will be conducted in accordance with applicable codes. Periodic integrated leakage rate tests are performed as required by the plant technical specifications. Since both the drywell and pressure suppression chamber have the same design pressure, it is possible to test the entire primary containment at the same pressure and without the necessity of providing temporary closures to isolate the pressure suppression chamber from the drywell. Penetrations welded directly to the primary containment are tested with the complete containment vessel. Inspections during these tests, periodic in-service inspections, and tests throughout plant life ensure early detection and repair of any leaks or other deterioration of the primary containment.

5.2.5.2 Penetrations

With the exception of the pipe penetrations which are welded directly to the primary containment shell, it is possible to leak-test individual containment penetrations without pressurizing the entire containment system. For those with double seals, testing may be accomplished by pressurizing the penetration between the double seals utilizing the pressure tap. Leak detection may then be accomplished by use of a soap solution, pressure decay, displacement, mass flow method, or volume flow method.

Pipe penetrations which must accommodate thermal movement are provided with double expansion bellows. The bellows expansion joints are designed for the containment system design pressure, and the inner bellows can be checked for leak tightness when the containment system is pressurized. In addition, these joints are provided with a test tap so that the space between the bellows can be pressurized to the calculated peak accident pressure to permit testing the individual penetrations for leakage.

The drywell to pressure suppression chamber vent pipe expansion bellows are not designed to be separately tested to verify leak tightness. The leak tightness of the drywell to pressure suppression chamber vent pipe expansion bellows is verified

during the periodic integrated leak-rate tests of the primary containment. These expansion bellows are an integral part of the primary containment vessel and, therefore, are not considered as penetrations per se. These bellows were designed as Class B vessels in accordance with the 1965 edition of the ASME Boiler and Pressure Vessel Code, Section III, Nuclear Vessels, and Code Cases 1177 and 1330. The manufacturing process began with a 0.078-inch thick plate made of Type 304 stainless steel that was formed into tubing, 84 inches in diameter and 24 inches in length, with two longitudinal seam welds. These welds were given a 100-percent radiography test prior to forming the convolutions, and a magnetic particle examination was given to all the butt welds used in the assembly process. After fabrication, cover plates were added to each bellows in order to conduct leak-tightness tests with Freon-12 and hydrostatic tests with 95 psig externally applied pressure. These tests and examinations verified the structural integrity of the bellows before the cover plates were removed in preparation for installation of the bellows in the Browns Ferry plant. An integrated leak-test was conducted on the completed containment (with bellows installed) that verified the initial structural integrity of the "as-built" primary containment. Periodic integrated leak testing of these bellows is sufficient since their normal deflections are only a small fraction of the design values and, also, the number of flexure cycles is only a small fraction of the limiting value. Thus, the likelihood of these bellows developing a leak is extremely small, and the test frequency proposed is the same as for the containment vessel proper. Sketches of the drywell vent pipe expansion bellows are provided in Figures 5.2-6f and -6g. The pneumatic test pressure is not less than 49.1 psig. The design pressure is 56 psig, and the manufacturer's test pressure is 70 psig.

Electrical penetrations are also provided with double seals and are separately testable. The test taps and the seals are so located that the tests of the electrical penetrations can be conducted without entering or pressurizing the drywell or pressure suppression chamber.

All containment closures which are fitted with resilient seals or bellows are separately testable to verify leak tightness. The covers on flanged closures, such as the equipment access hatch cover, the drywell head, access hatches and personnel air lock compartment are provided with double seals and with a test tap which will allow pressurizing the space between the seals without pressurizing the entire containment system. In addition, provision is made so that the space between the airlock doors can be pressurized to full drywell design pressure. The double O-ring seal type penetrations are designed to allow testing between the seals up to the containment design pressure.

5.2.5.3 Isolation Valves

The test capabilities which are incorporated in the primary containment system to permit leak detection testing of containment isolation valves as specified in Table 5.2-2 are separated into two categories.

The first category consists of those pipelines which open into the containment and do not terminate in closed loops outside the containment but contain two isolation valves in series.

Test taps are provided between the two valves which permit leakage monitoring of the first valve when the containment is pressurized. The test tap can also be used to pressurize between the two valves to permit leakage testing of both valves simultaneously. The valves, associated sensors, and equipment which will be subjected to containment pressures during the periodic leakage test are designed to withstand containment design pressure without failure or loss of functional performance. The functional performance of these devices has been verified by demonstration either during the leakage tests or subsequent to the test but prior to startup.

The second category consists of those pipelines which connect to the reactor system and contain two isolation valves in series. A leak-off line is provided between the two valves, and a drain line is provided downstream of the outboard valve. This arrangement permits monitoring of leakage on the inboard and outboard valves during reactor system hydrostatic tests, which can be conducted at pressures exceeding the reactor system operating pressure.

Surveillance requirements for primary containment isolation valves are given in Subsection 3.6.1.3 of the Technical Specifications.

5.2.5.4 Containment Inerting System

The Containment Inerting System is proven operable by its use during normal plant operations. Portions of the system normally closed to flow can be tested to ensure their operability and the integrity of the system.

5.2.6 Combustible Gas Control in Primary Containment

In normal operation, the primary containment atmosphere is maintained at less than 4.0 percent oxygen by volume, with the balance in nitrogen. Following a loss-of-coolant accident, hydrogen is evolved within the containment from metal-water reactions, and hydrogen and oxygen are produced by radiolysis of water. These are the only significant sources of hydrogen and oxygen. If the concentrations of hydrogen and oxygen were not controlled, a combustible gas

mixture could be produced. To ensure that a combustible gas mixture does not form, the oxygen concentration is kept below 5 percent by volume, or the hydrogen concentration is kept below 4 percent by volume.

The concentration of combustible gases in containment following a loss-of-coolant accident is controlled by a Containment Atmosphere Dilution (CAD) system. This system is capable of keeping the concentration of oxygen in the containment atmosphere below 5 percent when hydrogen and oxygen generation rates specified in Safety Guide 7, Control of Combustible Gas Concentrations in Containment Following a Loss-of-Coolant Accident, are assumed. In the event that post-accident monitoring showed that hydrogen and oxygen generation rates were substantially below those specified in the guide, the system could be operated to maintain either the hydrogen concentration below 4 percent or the oxygen concentration below 5 percent.

5.2.6.1 Design Basis

a. The CAD system shall be a shared system, capable of supplying nitrogen to the primary containment of Unit 1, Unit 2, and Unit 3.

b. The CAD system shall be capable of supplying nitrogen at a rate sufficient to maintain the oxygen concentrations of both the drywell and pressure suppression chamber atmospheres below 5.0 percent by volume, based on hydrogen and oxygen generation rates as set forth in AEC Safety Guide 7, Control of Combustible Gas Concentrations in Containment Following a Loss-of-Coolant Accident.

c. The containment atmosphere dilution portion of the CAD system shall be designed as an engineered safety feature. The system shall be operated remote-manually from the Main Control Room and shall be designed for possible startup 10 hours after a LOCA.

d. The Containment Air Monitoring (CAM) system provides a monitor for measuring oxygen and hydrogen concentrations inside primary containment. Equipment is also provided for sensor calibration.

e. The CAD system shall include means for releasing gas from either the drywell or the pressure suppression chamber in a controlled manner. The gas shall be released via the Standby Gas Treatment System. The containment vent portion of the CAD system is not an engineered safety feature since the venting function is not required to maintain the containment oxygen concentration below the 5 percent limit. However, redundant vent paths to the Standby Gas Treatment System are provided. Under severe accident conditions with loss of long term decay heat

removal capability, the HCVS may be used in lieu of CAD system venting to control primary containment pressure. The HCVS shall release gas from the pressure suppression chamber directly to the independent release points above the Unit 1, 2, and 3 Reactor Buildings.

f. Nitrogen storage capacity sufficient for 7 days of post-LOCA operation shall be provided.

g. The containment pressure shall not exceed 30 psig as a result of CAD system operation.

h. The CAD system shall provide an emergency source of nitrogen to the Automatic Depressurization System (ADS) main steam relief valve accumulators. This capability meets the requirements of NUREG-0737, Item II.K.3.28. The CAD system shall provide backup supply of nitrogen to operate the HCVS valves in the event control air is not available.

5.2.6.2 CAD System Design

The CAD system design is similar among units. A flow diagram of the Unit 1, Unit 2, and Unit 3 CAD system is shown in Figures 5.2-7 sheets 1, 2, and 3 and a control diagram is shown in Figures 5.2-8 sheets 1, 2 and 3. The system includes nitrogen supply facilities and gas release facilities.

In the event of a Beyond Design Basis External Event (BDBEE) to meet diverse and flexible coping strategies (FLEX) requirements, the Units 1, 2, and 3 CAD system may be supplied pressurized nitrogen through the existing permanent test connection isolation valve.

The CAD system nitrogen supply facilities include two trains, each of which is capable of supplying nitrogen through separate piping systems to the drywell and suppression chamber. Each train includes a liquid nitrogen supply tank, an ambient vaporizer, an electric heater, a manifold with branches to each primary containment, and pressure, flow, and temperature controls.

The nitrogen storage tanks have a total volume of 4000 gallons each. The tanks are filled to approximately 3400 gallons maximum level leaving a gas volume above the liquid. The required technical specification level of 2615 gallons for Unit 2 and 2615 gallons for Units 1 and 3 is adequate for the first 7 days of CAD

operation. The gas above the liquid in the tank is maintained at a minimum pressure of 100 psig. The nitrogen vaporizers use the ambient atmosphere as the heat source. Each is capable of producing 100 SCFM of gas when the atmospheric temperature is -20°F. The gas temperature is about 20° below ambient temperature.

Unqualified electric heaters are provided for use during cold weather to warm the gas. This is done to avoid brittle fracture problems in the carbon steel lines to which the stainless steel nitrogen supply piping connects at the drywell and pressure suppression chamber. The heaters have a capacity of 3 kilowatts each and are thermostatically controlled to maintain an effluent temperature between 50°F and 60°F. The heater in train A is supplied from reactor MOV board 1C, and the heater in train B is supplied from reactor MOV board 3B.

With nitrogen-operated valve 0-FCV-84-5 (train A) or 0-FCV-84-16 (train B) open, nitrogen is admitted to the drywell by opening solenoid-operated valve FSV-84-8A or FSV-84-8D, and to the pressure suppression chamber by opening solenoid-operated valve FSV-84-8B or FSV-84-8C. An isolation bypass switch for FSV-84-8A & 8D and 8B & 8C is provided to permit possible operation of the CAD system 16 hours after a LOCA. Manual valves 84-37 and 84-38 are preset to limit the flow to 100 CFM.

The two nitrogen storage tanks and their associated vaporizers are located outdoors at least 150 feet south of the Reactor Building. The A tank is positioned in front of the far west end of the Unit 2 Reactor Building, and the B tank is positioned past the east end of the Reactor Building and the Unit 3 Diesel Generator Building. The tanks and vaporizers are located so as to permit free circulation of air around the vaporizers.

Nitrogen supply lines run around the south side of the Reactor Building, and branch lines to each unit enter the building through six large steel and concrete pipe tunnels. Inside the building, each of the six branch lines is brought to a location near primary containment where it branches into two lines, one of which goes to the drywell and the other to the pressure suppression chamber. The two

branches serving the drywell are joined and connect into the drywell air purge line (refer to Figures 5.2-2a sheets 1, 2, and 3) at a point between the drywell and the first isolation valve. The two branches serving the pressure suppression chamber are joined and connect into the pressure suppression chamber air purge line (refer to Figures 5.2-2a - Sheets 1, 2 and 3) at a point between the pressure suppression chamber and the first isolation valve.

Thermocouple TE-84-32 is inserted into the nitrogen supply line that connects into the drywell air purge lines. Thermocouple TE-84-31 is inserted into the nitrogen supply line that connects into the pressure suppression chamber air purge line. In the event that the temperature sensed by any of these six thermocouples drops to 45°F, an alarm is actuated in the Main Control Room. This alarm (light) alerts the operator that the nitrogen supply valve which is open should be closed.

Dual paths are provided for releasing gas from the drywell or the pressure suppression chamber. Each path includes a butterfly valve, a throttle valve, a pressure switch, and a flow element. The butterfly valves and throttle valve FCV-84-20 automatically close on a containment isolation signal. Key operated switches are provided for bypassing the containment isolation signals for the butterfly valves when the reactor mode switch is not in the RUN position (MODE 1). Except during surveillance testing or a LOCA, throttle valve FCV-84-19 remains closed.

Depending on the path selected, the appropriate butterfly valve in the vent system is opened and the CAD system flow control valve FCV-84-19 or FCV-84-20 is opened. The flow rate is regulated at 100 CFM by the flow control valve, which, in turn, is controlled by the associated flow transmitter, FT-84-19 or FT-84-20. Pressure switch PS-84-21 or PS-84-22 closes the flow control valve FCV-84-19 or FCV-84-20.

Oxygen and hydrogen monitoring systems are described elsewhere in this section under the heading of "CAM System".

The CAD system, including nitrogen storage tanks, vaporizers, piping, and valves, is an Engineered Safeguards System and is designed to meet seismic Class I requirements. The system is designed in accordance with the following:

a. United States Atomic Energy Commission (USAEC), Safety Guides for Water-Cooled Nuclear Power Plants, revised March 10, 1971, Safety Guide No. 7, "Control of Combustible Gas Concentration in Containment Following a Loss-of-Coolant Accident."

b. USA Standard Code for Pressure Piping, Power Piping, USAS B31.1.0, 1967 edition, as published by the American Society of Mechanical Engineers, as supplemented by the requirements of the applicable GE

specifications, which are implemented in lieu of the outdated B31 Nuclear Code Cases - N2, N7, N9, and N10. The installation is in accordance with existing plant construction specifications for the applicable TVA piping classification.

c. Institute of Electrical and Electronics Engineers (IEEE), IEEE 279-1971, Nuclear Power Plant Protection Systems.

The nitrogen storage tanks and vaporizers are constructed of stainless steel and aluminum alloy. Nitrogen supply valves and piping are of stainless steel. Carbon steel piping and valves are used in the gas release lines.

The minimum design pressure for the CAD system piping is 150 psig. Portions of the system are designed to withstand higher pressures.

Valves in the system which are not hand-operated employ several means for actuation and are operable from the Main Control Room. Flow control valves 0-FCV-84-5 and 0-FCV-84-16 are nitrogen operated. Valves FSV-84-8A through FSV-84-8D are solenoid operated. Valves FCV-84-19 and FCV-84-20 are operated with instrument air or with nitrogen. The containment isolation valves used in CAD system operation (FCV-64-29, FCV-64-31, FCV-64-32, and FCV-64-34 on Figures 5.2-8 sheets 1, 2, and 3) are air- or nitrogen-operated valves which fail closed on loss of air and nitrogen. Since the station air supply might not be available in a post-LOCA situation, nitrogen from the CAD system supply manifolds is used as a backup gas for actuating the valves. Nitrogen from the train A manifold supplies valves FCV-64-29, FCV-64-32, and FCV-84-19, while nitrogen from the train B manifold supplies valves FCV-64-31, FCV-64-34, and FCV-84-20.

Nitrogen from the CAD system train A manifold also is used as a backup gas for actuating the torus vacuum breaker valves FCV-64-20 and FCV-64-21. No special provisions are made for mixing the added nitrogen with the containment atmosphere. The CAD concept is based on maintaining the oxygen concentration below the Safety Guide 7 limit of 5 percent; thus, the only concern from a mixing viewpoint is the potential degree of non-uniformity in oxygen concentration that would occur in the containment. There are three mixing forces existing in the containment after a loss-of-coolant accident: diffusion, natural convection, and forced convection. Forced convection is the most difficult mixing force to quantitatively evaluate, and detailed calculations of its effects on concentration gradients have not been done. However, detailed calculations have been done on the other two mixing forces, that is, diffusion and natural convection. The details of this analysis were presented in Amendment 2 of the Duane Arnold Energy Center FSAR in response to question G1.1(d). The referenced calculations showed that the maximum oxygen concentration deviation would be 2 percent from the average at the surface of the pressure suppression pool using

conservative assumptions relative to the natural convection driving force. Less conservative assumptions for natural convection would result in a maximum concentration deviation of only 0.3 percent. In other words, given an average oxygen concentration of 5 percent, the maximum concentration at the pressure suppression pool surface would be 5.10 percent, or, less conservatively, 5.015 percent. Based on the results of this analysis, it has been concluded that the assumption of a uniform oxygen concentration in the containment is reasonable.

To promote mixing, containment sprays can be operated intermittently.

CAM System

A Containment Air Monitoring (CAM) system is provided (one per unit) to monitor both hydrogen and oxygen in the drywell and torus. Measurement capability is provided over the range of 0-100 percent hydrogen concentration in a containment pressure range of -2.7 psig to 56 psig. Indication of hydrogen concentration in the containment atmosphere is provided in the control room.

The CAM systems consist of a sampling loop comprised of piping, isolation valves and a sample return pump; an analyzer located outside containment; and control room readout and control equipment. All piping and valves from the containment to the analyzer cabinets are ASME III, Class 2. All equipment necessary for post-LOCA operation is seismic Class I. Piping for Unit 1, from the containment isolation valve to the analyzer is ASME III, Class 3 and Seismic Class II. The cabinets containing the sensors are supplied from a 120 VAC I&C bus and the pumps are powered from a 480 VAC MOV board. The solenoid operated isolation valves can be remotely operated from the control room by a key-locked switch which overrides the primary containment isolation signal. The CAM systems and associated electrical equipment can be supplied from non-divisional power. The containment isolation valves must be supplied from 1E power. The valves will fail closed on loss of power. The pumps used to pull an air sample are bellows type (Units 2 and 3) and diaphragm type (Unit 1) pumps capable of pulling a sample from either the drywell or the torus in less than two minutes. The samples are exhausted back into the torus.

The temperature and humidity of the sample gas are controlled to ensure reliable and accurate readings. Less than two minutes is required to pull a sample from either the drywell or the torus. The piping and pressure retaining components are designed to withstand at least 56 psig. The system is also capable of operating when the drywell or the torus is under a vacuum down to -2.7 psig. The analyzers are Reg. Guide 1.97, Category 3 and are not required to withstand post-accident radiation exposure, nor are they engineered safeguards systems.

CAD System Operation

The CAD system is operated manually. Following a LOCA, records will be kept of hydrogen and oxygen concentrations and pressures in the drywell and pressure suppression chamber, and calculations will be made of the production rates of hydrogen and oxygen in each of these volumes. Nitrogen additions will be made periodically, as needed, to keep the oxygen content below 5 percent in each volume. Additions will be made separately to the drywell and the pressure suppression chamber. The amount of nitrogen to be added may be determined by the following equation:

$$V_N = V_i\left(\frac{c_i}{c_f} - 1\right)$$

where V_N = volume of nitrogen to be added, SCF
V_i = volume of gas in drywell or pressure suppression chamber before nitrogen addition, SCF

C_i = initial concentration of oxygen or hydrogen
C_f = desired final concentration of oxygen or hydrogen.

If hydrogen and oxygen production rates approach those assumed in Safety Guide 7, the containment pressure will increase and may reach the predetermined limit of 30 psig. Before this pressure is reached, containment venting will be initiated.

Gas releases will be made periodically and will be made separately from the drywell and pressure suppression chamber. Releases will be made during periods when meteorological conditions are most favorable. Gas will be released at a rate of about l00 CFM until the desired volume has been released. Releases are continued until the containment pressure has been reduced to atmospheric. Nitrogen additions will be continued during the period in which the containment pressure is being reduced to atmospheric. Additions and releases will be made at different times.

The operator manually controls nitrogen venting time and frequency. Changes in containment pressure are slow. To reduce containment pressure by one psi, for example, 19,000 SCF of gas would be released. At 100 CFM, the release time would be about 190 minutes.

The operator will have available to him information on the pressure, temperature, hydrogen content, oxygen content, radioactivity in the containment atmosphere,

and amount of nitrogen added for both the drywell and pressure suppression chamber. Meteorological information will be available also. Using this information, an operator can safely follow the venting procedure without exceeding the 10 CFR 50.67 limits following a LOCA.

5.2.6.3 Design Evaluation

The post LOCA containment gas concentration for hydrogen and oxygen concentrations versus time curves for the CAD System are described below. This analysis was based on the following assumptions:

- Extended Power Uprate (EPU) condition
- Safety Guide 7 design basis requirements
- No containment leakage
- Hydrogen gas generated from the metal water reaction is included as a diluent in the calculation of nitrogen capacity to maintain oxygen concentration below 5% by volume
- Initial drywell and wetwell oxygen concentration is 4%
- Post LOCA containment temperature versus time profile used in the analysis Is conservative with respect to the EPU post LOCA containment temperatre versus time profile. Increase in post LOCA containment temperature due to the potential increase of decay heat from the effects of additional actinides and activation products (SIL-636[2]) is beneficial to the CAD system and, therefore, bounded by the temperature profile used.
- Decay heat data are based on finite exponential series expressions defined in the U.S. NRC Standard Review Plan 6.2.5, Appendix A. These fuel-independent generic decay heat data are conservative and, as stated in the SRP, over-predict the standard curve specified by the American Nuclear Society Standard ANS 5.1-1979 by 20% between the decay times of 400 and $4x10^7$ seconds. These conservative decay heat data bound best-estimate decay heat curves applicable for the Browns Ferry fuel design, including the effects of additional actinides and activation products (SIL-636).

Two post LOCA conditions were evaluated:

1. Design Basis Case - conditions are one RHR loop operating, 95°F cooling water, 95°F initial suppression pool temperature and no containment sprays operating. The design basis case determines the nitrogen storage capacity required for the CAD system

2 GE Nuclear Energy Service Information Letter SIL-636 Rev.1, “Additional Terms included in Reactor Decay Heat Calculations.” (June 2001).

2. Bounding Case - cold conditions are one RHR loop in service, 40°F cooling water, 60°F initial suppression pool temperature, and no containment sprays operating. These conditions minimize the beneficial dilution effects of water vapor in the containment. The bounding case demonstrates that adequate operator response time and time to replenish nitrogen storage capacity exist for the full range of plant conditions.

AREVA Design Evaluation

A full core load of ATRIUM 10 fuel was assumed for the calculation of the active cladding mass. The results for the design basis case, under EPU conditions, are reflected by the figures summarized below:

Figure 5.2-13 provides the hydrogen and oxygen concentrations in the drywell and pressure suppression chamber following a LOCA without dilution for ATRIUM 10 fuel.

Figure 5.2-14 provides the hydrogen and oxygen concentrations in the drywell with dilution for ATRIUM 10 fuel.

Figure 5.2-15 provides the hydrogen and oxygen concentrations in the pressure suppression chamber with dilution for ATRIUM 10 fuel.

Figure 5.2-16 provides the maximum nitrogen required for dilution for ATRIUM 10 fuel. This shows that a maximum of about 200,000 SCF of nitrogen would be required in the first seven days following a LOCA. 2615 gallons of liquid nitrogen is equivalent to 200,000 SCF of gaseous nitrogen. In normal operation, the tanks will be refilled to maintain a minimum of 2615 gallons of nitrogen as required by technical specifications.

Figure 5.2-17 provides the maximum containment pressure following LOCA with dilution for ATRIUM 10 fuel.

For the design basis case after the design basis LOCA during the first 42 hours, water vapor in the drywell atmosphere and hydrogen produced by the metal-water reaction are sufficient to keep the oxygen concentration in the drywell below five percent. In the pressure suppression chamber, the oxygen concentration remains below five percent for the first 30 hours. Figures 5.2-14 and 5.2-15 include the effect of dilution by water vapor as well as hydrogen. The water vapor content is based on assumed post-LOCA conditions, which give maximum containment pressure.

For the bounding case, nitrogen injection is not expected to be required for 27 hours in the drywell and 12 hours in the suppression pool. These initiation times meet the minimum 10 hour design basis and do not challenge the operators capability to begin injection at the appropriate time. Since the CAD nitrogen storage capacity is defined by the design basis case, CAD would need to be replenished in 6.5 days at the bounding case conditions. These results confirm acceptable CAD capability for the full range of plant conditions.

ATRIUM-10 and ATRIUM-10XM fuel assemblies were assessed per the following Reference:

> FS1-0019597, Revision 1, Browns Ferry Units 1, 2, and 3 Extended Power Uprate Task Report, Task 411: Combustible Gas Control in Containment.

The fuel designs were found to meet Technical Specification Requirements and Bases assumptions. ATRIUM-10 results remain bounding.

General Design Evaluation

The CAD System operation during a LOCA can be carried out without exceeding 10 CFR 50.67 doses without exceeding a containment pressure of 30 psig. The dose calculation assumptions associated with fission product release to the environs by secondary containment are contained in FSAR, Chapter 14, Subsection 14.6.3.6.

With the provision of a redundant CAD System vent mechanisms as a backup containment purge system is not required. The nitrogen supply can be easily replenished from multiple facilities within seven days. There are multiple liquid nitrogen distribution facilities that are located within a one-day travel distance from Browns Ferry.

All enclosed areas and compartments of the Reactor Building have been examined with respect to possible hazards resulting from leakage from the containment. This examination showed that leakage to open areas of the Reactor Building will not produce hazardous gas mixtures. Adequate mixing occurs as a result of the high diffusion rate of hydrogen and some convection. Leakage into the space between the drywell and the surrounding concrete will not present a hazard because there is no ignition source. In addition, there is no ignition source

in the space above the drywell head. Similarly, there are no ignition sources in the clearance spaces between the shield plugs for the equipment access hatches, and the biological shield around the drywell.

Two areas requiring consideration are the pressure suppression chamber room and the personnel access room. Of the two areas, the personnel access room is the worst case. The pressure suppression chamber room has several openings that will allow convective mixing with other Reactor Building areas.

The total rate of leakage from the primary containment is based on the design criterion of 2 percent of the drywell volume per day at peak accident pressure.

From Figure 5.2-14, the maximum concentration of hydrogen in the containment atmosphere with no leakage (for ATRIUM-10 fuel) is about 14 volume percent and occurs about 48 hours after the LOCA.

It is postulated that a substantial portion of the total drywell leakage may occur through the two equipment access hatches and the personnel airlock. The personnel airlock is 7 feet in diameter and is equipped with double seals. Each equipment hatch is 12-feet 10-inches in diameter.

It is assumed that the rate of in-leakage to the personnel access room is 1/6 of the total leakage from the drywell. This is very conservative, since leakage must occur through two doors to reach the access room.

A convection flow path (see Figures 5.2-19, 5.2-20, and 5.2-21) exists from the pressure suppression chamber area to the personnel access room through floor sleeves. The return path for the cooler Reactor Building air is through the stairwell to the lower Reactor Building area (see Figure 5.2-21). Calculations show that a differential temperature of 20°F will produce a flow of 10 SCFM through the convective path described.

It can be concluded from this analysis that no hazard is produced as a result of leakage from the containment to enclosed areas of the Reactor Building. Further, the Standby Gas Treatment System will remove the hydrogen from the Reactor Building so that the hydrogen concentration in the building will not reach a hazardous level. The volume of the secondary containment is just below 9 million cubic feet and the Standby Gas Treatment System allowable surveillance in-leakage is given in Section 5.3.3.7. Based on this flow rate, the building volume will be changed at least once every day.

It is not anticipated that the integrity of the primary containment will deteriorate to a point where excessive leakage will occur following a design basis accident.

Capability is provided to release gas from primary containment through the Standby Gas Treatment System using CAD system vent valves should primary containment integrity be challenged from overpressure following a design basis accident. Under severe accident conditions with loss of long term decay heat removal capability, the HCVS may be used in lieu of CAD system venting to control primary containment pressure.

5.2.6.4 Testing and Inspections

Preoperational tests of the completed installation were conducted to establish that individual components perform as required.

Following interconnection with the individual units, each train of the nitrogen supply portion of the CAD system was operated to supply nitrogen to the primary containment. Manual valves 84-37 and 84-38 in the gas supply manifold were adjusted to limit flow to 100 CFM. The flow control valves in the gas release paths (FCV-84-19 and FCV-84-20) were adjusted to limit flow to 100 CFM, using air supplied through the test connections.

5.2.7 (Deleted)

5.2.7.2 (Deleted)

5.2.7.3 (Deleted)

5.2.8 Hardened Containment Venting System

5.2.8.1 Introduction

The consequences of several beyond design basis accident scenarios are more severe than the accidents previously considered herein. The primary containment pressure during these accidents is estimated to exceed its design capacity. Thus, the primary containment fails, potentially to the environment as well. The HCVS provides an emergency primary containment vent path to prevent, or at least slow down, the buildup of potentially damaging pressure within the primary containment.

5.2.8.2 System Description

The HCVS provides a direct vent path from the torus (wetwell) to independent release points above the Unit 1, 2, and 3 Reactor Buildings at an elevation of 741'-6". The vent flow path exits the torus via the existing 20" pressure suppression chamber supply, which will be isolated during venting by the existing primary containment isolation valve FCV-64-19. The Unit 1, 2, and 3 HCVS path consists of torus penetration X-205, the 20" pressure suppression chamber supply piping downstream of valve FCV-64-20, and a 14" line that exits the Reactor Building. The 14" Sch 30 pipe transitions to 14" Sch 40 exterior to the Reactor Building in an underground valve pit before turning and routing vertically up the exterior of the Reactor Building. At the 664' elevation the pipe turns to enter the refuel floor area and then continuing to the vent termination point at elevation 741'-6" above the Reactor Building roof. Two 14" pneumatically operated butterfly valves (FCV-64-221 and FCV-64-222) provide primary containment isolation and can be remotely operated from the Main Control Room. The HCVS has a design pressure and temperature of 62 psig and 350°F and a maximum operating pressure of 56 psig. Electrical power to the HCVS isolation valves is from RMOV boards which remain powered during an Extended Loss of AC Power (ELAP) event. Power is available via manual transfer for each isolation valve to a dedicated HCVS battery system which ensures operability during the first 24 hours of an ELAP. The vent pipe is safety-related TVA piping Class D (similar to ASME III Class 2) up to and inclusive of the outboard containment isolation valve. The downstream piping is non-safety related and is TVA piping Class P (similar to ASME III Class 3) up to the valve pit flange location, downstream the piping is TVA piping Class M up to the vent release point. A pneumatic supply from System 032 (Control Air) with System 084 (Containment Air Dilution) backup serves each operator for primary containment isolation valves FCV-064-0221 and FCV-064-0222. The CAD supply and associated control air piping is safety-related and is supported Seismic Class I to ensure its availability. A backup nitrogen system is provided to ensure continued operation during the first 24 hours during

an ELAP event resulting in a loss of Control Air or CAD. During normal plant operation, the HCVS containment isolation valves will remain closed. In response to a severe accident (long-term loss of decay heat removal), plant management could direct the control room operators to employ the HCVS to relieve excessive pressure within the containment. In this case, the operator will follow a written Emergency Operating Procedure for HCVS operation.

5.2.8.3 Radiological Consequences of HCVS Use

The exhaust gases released by the HCVS following a beyond design basis accident would have initially been "washed" by the pressure suppression pool water which would reduce the particulate released. These exhaust gases are vented to a release point above the Unit 1, 2, and 3 Reactor Buildings.