CHAPTER 6

ENGINEERED SAFETY FEATURES

6.1 ENGINEERED SAFETY FEATURE MATERIALS

Materials used in Nine Mile Point Nuclear Station - Unit 2 (Unit 2) engineered safety feature (ESF) components have been selected on the basis of engineering review and evaluation to ensure that material interactions that could impair operation do not occur. Materials have been selected to withstand environmental conditions encountered during both normal operations and postulated accidents without adverse effects on ESF service, performance, or operation.

6.1.1 Metallic Materials

Most metallic materials used in ESF systems comply with the material specifications of ASME Boiler and Pressure Vessel Code Section II. In cases where it is impossible to adhere to ASME specifications, metallic materials have been selected in compliance with other standards (e.g., ASTM).

6.1.1.1 Materials Selection and Fabrication

6.1.1.1.1 Specifications for Principal ESF Pressure-Retaining Materials

Principal pressure-retaining materials and appropriate material specifications for reactor coolant pressure boundary (RCPB) components are given in Table 5.2-5. Tables 6.1-1 and 6.1-2 list the principal pressure-retaining materials and appropriate materials specifications for plant ESF components.

6.1.1.1.2 ESF Construction Material

All ESF materials are resistant to intergranular stress corrosion cracking (IGSCC) in the environment of the boiling water reactor (BWR) coolant. Piping for IGSCC service sensitive systems is constructed from carbon steel or solution heat-treated, low-carbon (0.035 percent maximum) stainless steel (L grades) except for the shear plug in the standby liquid control system (SLCS) which is Type 304 stainless steel. Periodic testing of this valve requires parts replacement during every other refueling outage to ensure its integrity.

A conservative corrosion allowance of 0.08 in minimum is provided for all surfaces of carbon and low-alloy steel piping exposed to reactor water. General corrosion on stainless steel is negligible. Demineralized water containing soluble zinc is used, as described in Section 9.5.11.1.2, as reactor water. Following a loss-of-coolant accident (LOCA), demineralized water and soluble zinc have no detrimental effect on ESF materials.

6.1.1.1.3 Integrity of ESF Components During Manufacturing and Construction

Control of Sensitized Stainless Steel

Controls applied to nuclear steam supply system (NSSS)-supplied ESF components to avoid severe sensitization are the same as controls applied to RCPB components (Section 5.2.3.4). An assessment of compliance with Regulatory Guide (RG) 1.44 for the NSSS-supplied ESF is provided in Section 5.2.3.4.

All non-NSSS supplied ESF components comply with RG 1.44 recommendations. Low carbon stainless steel grades (L grades) are used in systems with operating temperatures in excess of 200°F. Use of non-L grade material is restricted to low temperature piping (200°F or less). An intergranular corrosion test on the base metal heat-affected zones (HAZ) of weldments is not performed in the latter case. Refer to Section 1.8 for the Unit 2 position on RG 1.44.

Cleaning and Contamination Protection Procedures

Specifications for NSSS-supplied ESF piping and component cleanliness and contamination protection during fabrication, shipment, and storage are discussed in Section 5.2.3.4.1.

All non-NSS specifications are provided for ESF piping and component cleanliness and contamination protection during fabrication, shipment, and storage in the construction phase in accordance with RG 1.37, 1.38, and 1.44.

During fabrication, contamination of austenitic stainless steel by compounds that may alter physical or metallurgical structure and/or properties is controlled or avoided. Painting stainless steel is not permitted unless a case-by-case review demonstrates that contamination is controlled or avoided. Mechanical cleaning (grinding, wire brushing, etc.) is done with tools not used on other materials.

Internal surfaces of completed components are cleaned to the appropriate level of cleanness defined by ANSI N45.2.1. Water quality for final flushes of fluid systems is equivalent to the quality of the operating system water except for oxygen content.

Onsite and preoperational cleaning of ESF components is in accordance with the Unit 2 position on RG 1.37 described in Section 1.8.

Cold Worked Stainless Steel

Austenitic stainless steel with a yield strength of greater than 90,000 psi is not used in ESF systems. In the field, cold bending is allowed only on 2-in and smaller pipe.

Thermal Insulation

Two types of external insulation are used on Unit 2. Stainless steel reflective metal insulation does not contribute to any surface contamination and has no4 effect on plant materials. Nonmetallic insulation materials in ESF systems comply with RG 1.36 and have the proper ratio of leachable sodium plus silicate ions to leachable chloride plus fluoride ions. See Section 1.8 for the Unit 2 position on RG 1.36.

6.1.1.1.4 Weld Fabrication and Assembly of Stainless Steel ESF Components (Non-NSSS Supplied Components)

Recommendations of RG 1.31 for stainless steel filler metal have been followed. RG 1.50 is applied for preheat temperature control requirements and welding procedure qualifications used for welding low-alloy steel. Where it is impractical to maintain preheat until postweld heat treatment, a temperature of 300°F or applicable preheat temperature, whichever is higher, is maintained for 2 hr/in of thickness to ensure hydrogen removal.

Recommendations of RG 1.71 for welder qualification for areas of limited accessibility have been followed. In lieu of regulatory guide requirements for welder qualification, additional volumetric weld inspection may be used. Refer to Section 1.8 for the Unit 2 positions on RG 1.31, 1.50, and 1.71.

Control of welding for NSSS-supplied components is discussed in Section 5.2.3.3.

6.1.1.2 Composition, Compatibility, and Stability of Containment and Core Spray Coolants

Containment spray and core cooling water for the ESF systems are supplied from the condensate storage tanks (CST) or suppression pool. The quality of water stored in the CSTs is maintained as described in Section 9.2.6.1.2.

The suppression pool is initially filled with high-purity water of the quality described in Section 9.2.6.1.2 from either the condensate storage or demineralized water makeup system. Chloride concentration in the suppression pool water is originally established at less than 0.5 ppm Cl. Grab sample analyses are performed, as needed, on suppression pool water (Section 9.3.2). If required, provision has been made for periodic filtration and demineralization by processing through the radwaste treatment system. No detrimental effects occur on ESF materials from this demineralized water.

Hydrogen generation resulting from the corrosion of materials by the containment spray during a design basis accident (DBA) is discussed in Section 6.2.5.

6.1.2 Organic Materials

Organic materials used within the primary containment include protective coatings, as identified below, and other materials, as identified in Table 6.1-3.

Hydrogen generation by protective coatings and other organic materials, and combustible gas control are discussed in Section 6.2.5.

6.1.2.1 Protective Coatings in the Suppression Pool

The only use of protective coatings in the suppression pool is on certain valve actuator enclosures, as listed in Table 6.1-3.

All other materials exposed to the suppression pool atmosphere, including the primary containment liner, floor liner, pedestal liner, downcomers, piping, and valves, are stainless steel or other corrosion-resistant alloys.

6.1.2.2 Protective Coatings in the Drywell

The majority of the exposed surfaces within the drywell, i.e., primary containment liner, drywell head, biological shield wall, structural steel cranes, pipe rupture restraints, pipe supports, piping, and concrete, are coated with materials qualified in accordance with ANSI N101.2 and applied in accordance with RG 1.54 as addressed in Table 1.8-1. The coating systems used on metallic surfaces are either an inorganic zinc primer with or without a catalyzed epoxy enamel topcoat, or a catalyzed epoxy enamel primer with or without a catalyzed epoxy enamel topcoat. A catalyzed epoxy surface covering with a catalyzed epoxy enamel topcoat is used on concrete surfaces.

The total estimated area of any unqualified protective coating used on surfaces within the drywell is given in Table 6.1-3. These surfaces include, but are not limited to, items such as valve bodies, handwheels, electrical and control panels, loudspeakers, emergency light cases, etc. The amount of other organic materials used, such as cable insulation, is also included in the table.

Untopcoated Carbo-Zinc 11 (CZ-11), an inorganic zinc primer, has been tested and DBA qualified for the thickness range of 2.0 to 3.3 mils (reference Carboline Test Program No. 02294, dated May 1, 1985). Similar tests were conducted by Oak Ridge National Laboratory in 1982 (reference Test No. ORNL A9675, 10-13-2, dated October 21, 1982). Although the CZ-11 does not possess the same physical characteristics of hardness as the material represented by the ORNL test, the mode of degradation in both tests was by granulation. The resulting very fine particles, less than 20 microns in size, have been evaluated in strainer head loss tests, paint debris evaluations and calculations in accordance with the guidance provided in RG 1.82 Revision 2, and the Boiling Water

Reactor Owners' Group (BWROG) Utility Resolution Guidance (NEDO-32686 Revision 0). The results of these tests, evaluations and calculations were utilized to conservatively size the emergency core cooling system (ECCS) suction strainers. There also exists in the primary containment untopcoated CZ-11 in the thickness range of 3.3 to 6.0 mils that has not been specifically tested. Extrapolating the results of the tests performed on other thicknesses of the same material and on the same thicknesses of comparable material, the untopcoated CZ-11 for thicknesses greater than 3.3 mils will behave in a similar manner in that degradation, if it occurs, will be by granulation. However, since material of this thickness range has not been specifically qualified, it is included in Table 6.1-3.

The total amount of all unqualified protective coatings, if assumed to create debris under DBA conditions, is determined not to be a safety problem since the sizing of the ECCS strainers assumed failure and 100-percent transport to the suppression pool of all unqualified coatings included in Table 6.1-3, plus that within the break zone of influence (ZOI) during a LOCA. The coating debris, along with other plant-specific debris (i.e., fibrous insulation, dust, dirt and sludge), was evaluated and utilized to conservatively size the ECCS strainers in accordance with the guidance and requirements of RG 1.82 Revision 2 and the BWROG Utility Resolution Guidance (NEDO-32686 Revision 0).

TABLE 6.1-1
(Sheet 1 of 2)
PRINCIPAL PRESSURE-RETAINING MATERIAL FOR ESF COMPONENTS
(Non-NSSS Scope of Supply)

Component	Form	Material	Specification (ASME/ASTM)
Containment			
Drywell liner			See Section 3.8.1.6.2 for material specifications
Suppression chamber liner			
Drywell head			
Access openings			
Penetrations			
Suppression vent downcomers	Pipe	Stainless steel	SA-312 Type 304
Vacuum relief valves	Forging	Stainless steel	SA-182 F316L
Reactor pedestal liner	Plate	Stainless steel	SA-240 Type 304L
DBA Hydrogen Recombiner System			
Piping-external	Pipe	Carbon steel	SA-106 Gr. B
	Pipe	Stainless steel	SA-312 Type 304
Fittings-external	Forging	Carbon steel	SA-105
	Forging	Carbon steel	SA-234 WPB
	Forging	Stainless steel	SA-182 F304
	Forging	Stainless steel	SA-403 WP304 or WP304W
Valves-external	Forging	Carbon steel	SA-105
	Forging	Stainless steel	SA-182 F316
Bolts	Bar	Low-alloy steel	SA-193 Gr. B7
	Bar	Stainless steel	SA-193 Gr. B6
Nuts	Forging	Carbon steel	SA-194 Gr. 2H
	Forging	Stainless steel	SA-194 Gr. 6
Strainers			
Body	Casting	Stainless steel	SA-351 Gr. CF8
Flange	Forging	Stainless steel	SA-182 F304
Recombiner Unit			
Heat exchanger			
Piping	Pipe	Stainless steel	SA-312 Type 316
Fittings	Forgings	Stainless steel	SA-182 Type 316
Reaction chamber			
Piping	Pipe	Stainless steel	SA-312 Type 304
			SA-358 Type 304
			SA-376 Type 304
Piping	Pipe	Carbon steel	SA-333 Gr. 6
	Pipe	Carbon steel	SA-106 Gr. B
Valves	Forging	Carbon steel	SA-105

TABLE 6.1-1
(Sheet 2 of 2)

Component	Form	Material	Specification (ASME/ASTM)
ECCS			
Piping	Pipe	Carbon steel	SA-106 Gr. B
	Pipe	Stainless steel	SA-358 Type 304 Cl. 1 or Type 316
	Pipe	Stainless steel	SA-376 Type 304 or SA-312 Type 304
Fittings	Forging	Carbon steel	SA-105
		Carbon steel	SA-234 WPB
		Stainless steel	SA-182 F304
		Stainless steel	SA-403 WP304 or WP304W
Valves	Forging	Carbon steel	SA-105
	Castings	Carbon steel	SA-216 Gr. WCB
	Plates	Carbon steel	SA-515 Gr. 70
	Forgings	Stainless steel	SA-182 F316 or F304
	Castings	Stainless steel	SA-351 Gr. CF8
Standby Liquid Control System			
Injection line	Pipe	Stainless steel	SA-312 Type 316L
Valves: ASME Safety Class I	Forgings	Stainless steel	SA182-F316L
ASME Safety Class II	Forgings	Stainless steel	SA182-F316

TABLE 6.1-2
(Sheet 1 of 2)
PRINCIPAL ESF COMPONENT MATERIALS
(NSSS Scope of Supply)

Component	Form	Material	Specification (ASTM/ASME)
RHR Heat Exchanger			
Shell, head and channel	Plate	Carbon steel	SA-516 Gr. 70
Tubesheet	Plate	Carbon steel	SA-516 Gr. 70
Tubesheet - cladding on channel side	Cladding	Weld deposit	Type 309L or 309 for first layer, 308L for subsequent layers
Nozzles - shell inlet and outlet	Forgings	Carbon steel	SA-350-LF2
Nozzles - channel inlet and outlet	Forgings	Carbon steel	SA-350-LF2
Flanges - shell side	Forgings	Carbon steel	SA-350-LF2
Flanges - channel side	Forgings	Carbon steel	SA-350-LF2
Tubes	Tubing	Stainless steel	SA-249 T-304L
Studs	Bar	Low-alloy steel	SA-193 Gr. B7
Nuts	Bar	Low-alloy steel	SA-194 Gr. 7
RHR, HPCS, and LPCS Pumps			
Bowl assembly	Casting	Carbon steel	SA-216 Gr. WCB or A-216 Gr. WCB
Discharge head shell	Plate	Carbon steel	SA-516 Gr. 70
Discharge head cover	Forging	Carbon steel	SA-105
Suction barrel shell and dished head	Plate	Carbon steel	SA-516 Gr. 70
Flanges	Forging	Carbon steel	SA-105
Pipe (RHR, LPCS pumps)	Pipe	Carbon steel	SA-106 Gr. B
Pipe (HPCS pump)	Pipe	Carbon steel	SA-106 Gr. B
Shaft	Bar	Stainless steel	A-276 Type 410 Cond. H
Impeller	Casting	Stainless steel	A-351 Gr. CA6NM
Studs	Bar	Low-alloy steel	SA-193 Gr. B7
Nuts	Forgings	Low-alloy steel	SA-194 Gr. 7
Cyclone separator body and cover	Bar	Stainless steel	SA-479 Type 304
HPCS Valves			
Body, bonnet, and disc	Casting	Carbon steel	SA-216 WCB
Stem	Bar	Stainless steel	A-564 Type 630-1075 for valve F012 and F004; A-479 Type 410 or A-276 Type 410 for valve F023; A-479 Type 410 or A-564 Type 630-1075 for valves F001, F010, F011 and F015
Studs	Bar	Alloy steel	SA-193 Gr. B7
Nuts	Forgings	Carbon steel	SA-194 Gr. 2H
Control Rod Velocity Limiter	Casting	Stainless steel	A351 Gr. CF8 or CF3
Main Steam Flow Restrictor	Casting Casting	Stainless steel Carbon steel	SA351 Gr. CF8 (upstream insert) SA216 Gr. WCB (downstream insert)

TABLE 6.1-2
(Sheet 2 of 2)

Component	Form	Material	Specification (ASTM/ASME)
Main Steam Safety Relief Valves			
Body and bonnet	Casting	Carbon steel	SA-352, Gr. LCB
Disc	Casting	Stainless steel	SA-351, Gr. CF3A
Stem	Bar	Stainless steel	A-564 Type 630

TABLE 6.1-3
(Sheet 1 of 1)
UNQUALIFIED PROTECTIVE COATINGS AND ORGANIC MATERIALS
USED INSIDE THE PRIMARY CONTAINMENT

	Material	Quantity
Protective Coatings		
Inside Drywell		
Portions of the liner and supports and also certain misc. equipment	Inorganic zinc	6,700 ft^2 @3.3 to 6.0 mil DFT
	Epoxy based	8,960 ft^2 @8 mil DFT
	Alkyd based	300 ft^2 @3 mil DFT
	Modified phenolic	460 ft^2 @5 mil DFT
Recirculation pumps	Alkyd based	1,100 ft^2 @5 mil DFT
		220 ft^2 @3 mil DFT
Inside Suppression Pool		
Valve actuator enclosures	Alkyd based	50 ft^2 @3 mil DFT
Other Organic Materials		
Cable Insulation	Covered	Uncovered
Ethylene propylene rubber	1,920 lb @23 ft^3	1,280 lb @16 ft^3
Hypalon	8,890 lb @92 ft^3	2,320 lb @23.3 ft^3
Cross-linked polyethylene	7,225 lb @82.9 ft^3	100 lb @1.1 ft^3
Polypropylene	809 lb @8 ft^3	0
Motor electrical insulation[(1)]	None	1,390 lb
Shimming material	Devcon plastic steel B (catalyzed epoxy with 80% steel)	300 lb

(1) Approximate weight of recirculation drive motor stator insulation, wedges, and detectors.

6.2 CONTAINMENT SYSTEMS

6.2.1 Containment Functional Design

This section establishes the design basis for the primary containment structure and provides the major design features and evaluation of the primary containment to perform the intended safety functions during all normal and postulated accident conditions throughout the operating life of the plant.

6.2.1.1 Containment Structure

The primary containment structure of Unit 2 consists of the drywell, the pressure suppression chamber which stores a large volume of water, and the drywell floor which separates the drywell and suppression chamber. The drywell is a steel-lined reinforced concrete vessel in the shape of a frustum of a cone, closed by a dome with a torispherical head. The pressure suppression chamber is a cylindrical stainless steel clad steel-lined reinforced concrete vessel located below the drywell. The primary containment structure houses the reactor vessel, the reactor recirculation system, and other branch connections of the RCPB.

6.2.1.1.1 Design Bases

The primary containment structure, including subcompartments (Section 6.2.1.2), meets the following functional design bases.

<u>Containment Vessel Design</u>

The containment vessel design bases are:

1. The primary containment has the capability to maintain its functional integrity during and following the peak transient pressure and temperatures that would occur following any postulated LOCA. The LOCA includes the worst single failure (which leads to the maximum primary containment pressure and temperature) and is further postulated to occur simultaneously with loss of offsite power (LOOP) and a safe shutdown earthquake (SSE).

2. The primary containment structure also withstands the peak environmental transient pressures and temperatures associated with the postulated spectrum of line breaks.

3. The primary containment has the capability to withstand jet forces associated with the flow from the postulated rupture of any pipe within it.

4. The primary containment system is protected from or designed to withstand missiles from internal sources

and excessive motion of pipes that could directly or indirectly endanger the integrity of the containment.

5. The primary containment is designed for the hydrodynamic loads in the Design Assessment Report for Hydrodynamic Loads (DAR, Appendix 6A).

Containment Subcompartment Design

The effects of primary containment subcompartment pressurization (Section 6.2.1.2) on the postulated pipe ruptures have been evaluated.

Drywell Internal Pressure

The DBA postulated for the calculation of the maximum pressure acting on the drywell walls is a double-ended rupture (DER) of a 24-in recirculation suction line. This event is more severe than a DER of the 26-in main steam line. The peak calculated drywell pressure following the recirculation pump suction line DER is shown in Table 6.2-4.

Suppression Chamber Internal Pressure

The DBA for the suppression chamber is a DER of a 24-in recirculation suction line. The peak pressure occurs shortly after the initial reactor blowdown. The maximum calculated suppression chamber pressure is shown in Table 6.2-4.

Drywell Floor Differential Pressure

1. Downward ΔP

   The DBA for the downward floor differential pressure is a DER of a 24-in recirculation suction line. The maximum differential pressure (drywell to suppression chamber) occurs at the time of downcomer vent clearing when steam and air begin to flow to the suppression chamber. The maximum calculated downward differential pressure on the floor is shown in Table 6.2-4. A significant margin remains with respect to the design downward differential pressure of 25 psid. In addition to this, the drywell floor is structurally designed to withstand a pressure load equal to 1.5 times the design pressure (i.e., 25 psid), thereby widening the safety margin.

2. Upward ΔP

   The DBA for the upward floor differential pressure is a small steam line break with rapid drywell depressurization caused by initiation of the drywell

sprays considering the worst-case minimum spray water temperature.

In this case all the drywell air is transferred to the suppression chamber prior to spray initiation. It is assumed that the Operator actuates drywell sprays 30 min after the drywell pressure reaches 30 psig and a rapid depressurization of the drywell is initiated.

The drywell depressurization and upward ΔP are limited by the return flow of air from suppression chamber to the drywell through the vacuum breakers.

The following major assumptions have been made to maximize the upward floor ΔP.

1. Initial suppression pool temperature is 40°F.

2. The service water temperature is assumed to be 32°F.

3. Containment spray flow to the suppression chamber is ignored.

4. Both loops of drywell sprays are actuated with 100 percent thermal effectiveness.

The maximum upward loading on the drywell floor is 4.70 psid or less for the worst-case transient assuming that only three out of the four vacuum breaker flow paths are available. The design upward ΔP is 10 psid which provides a substantial margin relative to the maximum calculated value. In addition to this, the drywell floor is structurally designed to withstand a differential pressure load equal to 1.5 times the design pressure (i.e., 10 psid), thereby widening the safety margin.

<u>Drywell Atmosphere Temperature</u>

The DBA for the drywell atmosphere temperature is a postulated small steam line break. The maximum drywell atmospheric temperature is 340°F and the maximum drywell liner temperature corresponding to the saturation pressure equivalent to the design pressure of 45 psig is 293°F.

<u>Suppression Chamber Temperature</u>

The DBA for the suppression chamber and the suppression pool temperature is the DER of a 24-in recirculation suction line. The design basis temperature of 212°F occurs between 1 and 15 hr after the break depending on the number of residual heat removal (RHR) pumps and heat exchangers used to remove heat from the system.

Mass and Energy Release for the Primary Containment Design

The maximum postulated release of mass and energy to the primary containment is based upon the instantaneous DER of a 24-in reactor recirculation pump suction line or 26-in main steam line. The effects of metal-water reaction and other chemical reactions following the DBA can be accommodated in the primary containment design. Further description of mass and energy release data is provided in Section 6.2.1.3.

Energy Removal from the Containment

Energy is removed from the primary containment after a LOCA by circulating the suppression pool water through the RHR system heat exchangers where heat is removed by the service water (SWP) system. The primary containment spray mode of the RHR system can also be used to condense steam and reduce the pressure and temperature in the primary containment following a LOCA. Section 6.2.2 describes these ESFs in more detail. Energy is also absorbed from the primary containment atmosphere by way of passive heat sinks.

Pressure Suppression Feature

The Unit 2 primary containment conforms to the fundamental principles of a Mark II pressure suppression system. The water stored in the suppression pool is capable of condensing the steam displaced into the pool through the downcomer vents, and the amount of water is sufficient that no Operator action is required for at least 10 min immediately following initiation of a LOCA. In addition, the design allows any significant amounts of water from pipe breaks within the primary containment to drain back to the suppression pool. This closed loop ensures a continuous, adequate supply of water for core cooling.

Primary Containment Leakage Design

The primary containment system, in conjunction with other engineered safeguards, has the capability to limit leakage during the postulated design basis LOCA so that offsite doses do not exceed the criteria set forth in 10CFR50.67.

Hydrostatic Loading Design

The primary containment design permits filling the primary containment with water to the level of the reactor vessel flange.

Primary Containment Leakage Testability

The primary containment system is designed to allow for periodically conducting tests in accordance with 10CFR50 Appendix J to confirm the leak-tight integrity of the primary containment and its penetrations.

Suppression Pool Hydrodynamic Loads

The amplitudes and frequency of the dynamic forcing functions that result from hydrodynamic loads due to steam and air discharges into the suppression pool are discussed in the Unit 2 DAR for Hydrodynamic Loads (Appendix 6A).

6.2.1.1.2 Design Features

The primary containment design features include the drywell, the pressure suppression chamber, downcomers between the drywell and the suppression chamber, isolation valves, vacuum breakers, and the RHR system for containment heat removal. The primary containment is a steel-lined, reinforced concrete pressure vessel that is closed at the top by the drywell head assembly. The steel drywell head assembly forms a gas-tight enclosure (Section 3.8.1). The analysis of the steel membrane is included in Section 3.8.1. The design, description, and analysis of the reinforced concrete primary containment structure are included in Section 3.8.1. The principal design parameters of the primary containment, suppression pool, and vent downcomers are listed in Table 6.2-3.

Drywell

The drywell is a steel-lined reinforced concrete vessel in the shape of a truncated cone having a base diameter of approximately 91 ft and a top diameter of approximately 34 ft. The floor of the drywell serves both as a pressure barrier between the drywell and the suppression chamber and as the support structure for the reactor pedestal and downcomers.

The drywell houses the reactor and associated equipment. The primary function of the drywell is to contain the radioactivity and withstand pressures and temperatures resulting from a breach of the RCPB, up to and including an instantaneous circumferential break of a single reactor recirculation pump suction pipe, and to provide a holdup time for decay of any radioactive material released. The drywell is designed to resist the forces of an internal design pressure of 45 psig in combination with thermal, seismic, and other forces as outlined in Chapter 3.

Pressure Suppression Chamber

The pressure suppression chamber is a stainless steel clad steel-lined, reinforced concrete vessel in the shape of a cylinder, having an inside diameter of 91 ft. The foundation mat, to which the vessel is anchored, is lined with steel plates within the inside diameter of the cylinder. The steel plates are welded to each other and to steel embedments to maintain the primary containment function of a gas-tight enclosure.

Pressure Suppression Pool

The pressure suppression pool, which is contained in the pressure suppression chamber, stores sufficient water to condense the steam released from blowdown of the reactor coolant system (RCS) after a LOCA or from safety/relief valve (SRV) discharge during accident or normal operational transients. Steam is transferred to the pressure suppression pool by the downcomers and the discharge piping of the SRVs.

Approximately 150,000 cu ft of water is contained within the suppression chamber. The suppression pool serves both as a heat sink for transients and accidents and as a reservoir of water for the core standby cooling systems. It is the primary source of water for the low-pressure core spray (LPCS) and low-pressure coolant injection (LPCI) systems, provides a safety-related source of water for the reactor core isolation cooling (RCIC) and high-pressure core spray (HPCS) systems. The water level and temperature of the pressure suppression pool are continuously monitored in the control room.

Downcomers

The downcomers consist of 121 pipes open to the drywell and submerged 9.5 ft below the low water level (operating minimum) of the suppression pool, providing a flow path for uncondensed steam into the pool. The internal diameter of each downcomer is 23.25 in. The downcomers project 3 to 6 in above the drywell sloping (or sloped) floor so that small quantities of water leakage flow past the downcomers and are collected in the drywell floor drain system. Each downcomer opening is shielded by a 2 1/4-in thick steel deflector plate to prevent overloading any single vent pipe by direct flow from a pipe break to that particular vent. The deflector plate also minimizes the potential for downcomer blockage by debris.

Vacuum Breakers

Vacuum breakers provide a return flow path from the suppression chamber gas space to the drywell. The vacuum breakers are designed to limit the negative differential pressure between the drywell and the suppression chamber to less than the design value (10 psid). Each vacuum breaker flow path has two relief valves in series to ensure a leak-tight boundary under positive drywell-to-suppression chamber differential pressure conditions. Three flow paths are required for the vacuum breaker design basis. One additional flow path is provided to accommodate the postulated single failure of one vacuum breaker.

The vacuum breakers are located inside the primary containment drywell and do not, therefore, form an extension of the primary containment boundary. These valves are mounted in piping that connects the drywell and suppression chamber. This location removes the vacuum breakers from the direct effects of chugging transients. Each vacuum breaker is designed to withstand a valve disc velocity of 20.8 radians/sec opening and 22.4 radians/sec

closing. The opening and closing velocities for pool swell phenomena have been determined and analyzed by Continuum Dynamics, Inc., for Commonwealth Edison Company's Lasalle Units 1 and 2. Testing was also performed. The Unit 2 vacuum breakers are of the same size and of similar design as the Lasalle valves. The tests were performed on a "modified" and an "original" as-installed valve configuration with the following results which were compared to Unit 2 opening and closing velocities.

	Opening Velocity	Closing Velocity
Lasalle		
Required	18.5 rad/sec	25.5 rad/sec
"Original" valve test	19.8 rad/sec	25.0 rad/sec
"Modified" valve test	20.2 rad/sec	25.8 rad/sec
Unit 2	20.8 rad/sec	22.4 rad/sec

The "original" configuration showed deformations occurred and the pallet did not seat uniformly. A bypass flow area of 10.8 sq in was measured compared to an allowable of 21.8 sq in. No other damage was observed and it was concluded that the valves would be acceptable for the intended function.

The modified valve showed no damage.

The modifications have been included in the Unit 2 valves and consist of:

1. Addition of an impact load distribution device.
2. Material upgrade of the eccentric shaft and hinge plate.
3. Hinge plate thickness increase.
4. Addition of a bumper stop to soften the opening velocity impact.

The closing test velocities exceeded the Unit 2 velocity for both tests, and the "modified" valve test was within 3 percent of the Unit 2 opening velocity.

Analysis for the modified vacuum breakers indicated that margins of safety of the disc were 1.3 rad/sec for closing and 2.7 rad/sec for opening velocity. Since this margin exists and the test velocity was within 3 percent of the Unit 2 opening, the Unit 2 valves are shown to be adequate and need not be tested.

The vacuum breakers have the capability for remote manual testing. This design provides assurance of limiting the

differential pressure between the drywell and suppression chamber and ensures proper valve operation and testing during normal plant operation. A monthly operability test of the drywell-to-suppression pool vacuum breakers is conducted by cycling each vacuum breaker through at least one complete cycle of full travel. Verification of actual position is normally determined by the full open position switch and the full shut position limit switches.

No vacuum relief valves are provided between the drywell and the reactor building atmosphere. The primary containment structure can accommodate subatmospheric pressure of approximately 10 psia at maximum operating water level.

6.2.1.1.3 Design Evaluation

The original licensed thermal power (OLTP), pre-uprate large line break analyses described in this section were performed at 105 percent of the original rated steam flow, corresponding to 104.3 percent of the original rated thermal power (3,323 MWt). These analyses provide the basis for identifying the limiting large break and for establishing the containment leak rate test pressure, and also evaluate the sensitivity of the containment response to variations in initial conditions and analysis assumptions which are also applicable to EPU and MELLLA+ on a relative basis.

For EPU, the limiting DBA LOCA (a double-ended guillotine break of a recirculation suction line) for containment pressure is analyzed to demonstrate that power uprate operation does not result in exceeding the containment design limits and P_a (peak calculated containment internal pressure) established in NMP2 Technical Specification 5.5.12.b of 39.75 psig. For operation in MELLLA+ domain, analysis demonstrates continued compliance with this limit. This limiting case analysis is addressed in Section 6.2.1.1.6 for EPU and 6.2.1.1.7 for MELLLA+.

The key design parameters and maximum calculated parameters are provided in Tables 6.2-3 through 6.2-5. These design and maximum calculated accident parameters are not determined from a single accident event but from an envelope of accident conditions. As a result, there is no single DBA for this containment system.

A maximum drywell and suppression chamber pressure occurs near the end of the blowdown phase of a LOCA. Approximately the same peak pressure occurs for the break of either a recirculation line or a main steam line. Both accidents are evaluated.

The most severe drywell temperature condition is predicted for a small steam break above the reactor water level. To demonstrate that breaks smaller than the rupture of the largest primary system pipe will not exceed the primary containment design pressure, the primary containment system responses to an intermediate size liquid break and a small size steam break are

evaluated. The results show that the primary containment design pressure is not exceeded for intermediate and small break sizes.

Table 6.2-6 lists the design performance parameters of the related ESF systems for primary containment heat removal during post-accident operation and those reduced capacities assumed for primary containment analyses.

Additional containment evaluation is provided in Appendix 15B for operation under single recirculation loop conditions.

Accident Response Analysis

The response of the pressure suppression containment system to postulated pipe breaks in the drywell is analyzed with the LOCTVS computer program. This program models the reactor system, drywell, downcomer vent system, suppression chamber, and active and passive heat removal systems.

The primary results of the LOCTVS accident analyses consist of the transient pressure and temperature response of the drywell and suppression chamber, and the transient temperature response of the suppression pool. The analyses assume instantaneous pipe ruptures concurrent with the LOOP. The worst-case single active component failure is the failure of one of the two electrical divisions (Division I or II) resulting in minimum containment heat removal capability. Failure of either Division I or II results in nearly identical peak pressure and temperature.

Break Spectrum

To ensure that the primary containment system design parameters are not exceeded for pipe breaks smaller than the DER of the largest primary system pipe, a spectrum of break sizes is analyzed. The following breaks are considered:

1. An instantaneous guillotine DER of a recirculation pump suction line.

2. An instantaneous guillotine DER of a main steam line.

3. An intermediate size liquid line rupture.

4. A small size steam line rupture.

Energy release from these accidents is reported in Section 6.2.1.3.

Recirculation Line Rupture

The instantaneous guillotine DER rupture of a recirculation suction line results in the discharge of maximum flow rate of primary system fluid and energy into the drywell. Figure 6.2-1 shows the location of the break. Immediately following the

rupture, the flow out of both sides of the break will be limited to the maximum allowed by critical flow considerations. Figure 6.2-1 shows a schematic view of the flow paths to the break. In the side adjacent to the pump suction nozzle, the flow will correspond to critical flow in the pipe cross section of 2.598 sq ft. Reverse flow in the discharge side of the recirculation pump will correspond to critical flow at the 50 jet pump nozzles (5 nozzles per jet pump) associated with the broken loop, providing an effective break area of 0.461 sq ft. In addition, a 4-in cleanup line crosstie adds 0.088 sq ft to the critical flow area, yielding a total of 3.147 sq ft (Figure 6.2-1).

Assumptions for Reactor Blowdown The following assumptions are made with respect to the blowdown from the RCPB:

1. The reactor is operating at 105 percent of the original rated steam flow (corresponding to a core thermal power of 3,467 MWt) at the time of the recirculation line break. The initial conditions are selected to maximize the parameter of interest, that is, primary containment pressure.

2. The instantaneous DER of the recirculation suction line is considered. This results in the most rapid coolant loss and depressurization, with coolant being discharged from both ends of the break.

3. The vessel depressurization flow rates are calculated using Moody's critical flow model assuming liquid-only outflow, since this assumption maximizes the energy release to the drywell[1,2]. Liquid-only outflow implies that all vapor formed in the reactor pressure vessel (RPV) by flashing rises to the surface rather than being entrained in the blowdown. In reality, some of the vapor would be entrained in the break flow which would significantly reduce the RPV discharge flow rates.

4. The core decay heat and the sensible heat (after the accident) released in cooling the fuel to 551°F are included in the RPV depressurization calculation. The rate of energy release is calculated using a conservatively high heat transfer coefficient through the depressurization period. By maximizing the assumed energy release rate, the RPV is maintained at nearly rated pressure (within 10 percent) for approximately 16 sec. This high RPV pressure increases the calculated blowdown flow rates, which is again conservative for containment analysis purposes. The stored energy of the fuel at temperatures below 551°F is released to the vessel fluid along with the stored energy in the vessel and internals, as vessel fluid temperatures decrease below 551°F during the remainder of the transient calculation.

5. The main steam isolation valves (MSIVs) start closing at 0.5 sec after the accident. They are assumed to fully close in the shortest possible time of 3 sec following closure initiation. The closure signal for the MSIVs may occur from low reactor water level, so the valves may not receive a signal to close for more than 4 sec, and the closing time may be as long as 5 sec. By assuming rapid closure of these valves, the RPV is maintained at a high pressure which maximizes the calculated discharge of high-energy water into the drywell.

6. The reactor feedwater flow continues until all the high-energy feedwater (above 200°F) is depleted. This results in slightly lower pressure initially as compared to a case with no feedwater; however, subsequent peak (maximum peak) pressure is higher.

   The recirculation line blowdown mass flow rate, the corresponding enthalpy, and the RCPB pressure with and without the addition of feedwater are summarized in Tables 6.2-7 and 6.2-8, respectively.

<u>Assumptions for Primary Containment Pressurization</u> The pressure response of the primary containment during the blowdown period of the accident is analyzed using the following assumptions:

1. Thermodynamic equilibrium exists in the drywell and suppression chamber. The analysis assumes complete mixing of the effluents in the drywell and suppression chamber.

2. Downcomer flow consists of the homogeneous mixture of the fluid in the drywell. The use of this assumption results in liquid carryover into the drywell downcomer vents.

3. The fluid flow in the drywell-to-suppression chamber downcomer is compressible except for the liquid phase.

<u>Assumptions for Long-Term Cooling</u> Following the blowdown period, the emergency core cooling systems (ECCS) (Section 6.3) provide water for core flooding and long-term decay heat removal. The primary containment pressure and temperature response during this period are analyzed using the following assumptions:

1. Two RHR pumps in LPCI mode are used to flood the core prior to 10 min after the accident. The HPCS pump is assumed to be available for the entire transient.

2. After 10 min, one of the RHR pumps in LPCI mode may be transferred to the containment spray or pool cooling mode. An additional 10 min (total of 20 min) has been

provided to the Operator to turn on the sprays. This is a manual operation. Note that Case C, in Item 5 below, addresses the accident wherein no credit is taken for containment sprays.

3. The effect of decay energy, stored energy, and energy from the metal-water reaction are considered.

4. After approximately 20 min, the RHR heat exchanger is activated to remove heat from the primary containment via the suppression pool.

5. The performance of the ECCS equipment during the long-term cooling period is evaluated for each of the following three cases of interest:

   Case A — Offsite power available; all ECCS equipment and containment spray operating

   Case B — LOOP; minimum diesel power available for ECCS and containment spray mode (only Electrical Division II available)

   Case C — LOOP; minimum diesel power available for ECCS and pool cooling mode (only Electrical Division II available)

Initial Conditions for Accident Analysis The accident response analyses assume that the drywell, wetwell, and suppression pool are initially at the maximum normal operating conditions. Table 6.2-9 provides the initial RCPB and primary containment conditions used in the accident response evaluations.

Energy Sources All major sources of energy are considered in the calculation of the mass and energy release to the primary containment from each postulated pipe break. The sources of available energy include:

1. Stored energy in the RCPB.

2. Fission product and heavy element decay heat.

3. Fission power coastdown heat.

4. Stored energy in the RCPB metal piping, structures, and core.

5. Metal water reaction energy.

6. ECCS pump heat.

These data are provided in Tables 6.2-10 through 6.2-14.

<u>Short-Term Accident Response</u> The calculated primary containment pressure and temperature responses for the recirculation line break are shown on Figures 6.2-2 through 6.2-5 and 6.2-6 through 6.2-9, respectively. The calculated peak drywell pressure is 39.75 psig, which is below the primary containment design pressure of 45 psig.

The suppression chamber is initially pressurized by the carryover of noncondensables from the drywell. As the transient continues, steam flows through the vent system and the temperature of the suppression pool water increases, causing an increase in the suppression chamber pressure and a corresponding increase in the drywell pressure. The peak drywell pressure occurs near the end of the blowdown and then the drywell pressure stabilizes at a higher pressure than that of the suppression chamber, the difference being equal to the height of water to the downcomer submergence. During the RPV depressurization phase, most of the noncondensable gases initially in the drywell are forced into the suppression chamber. However, following the depressurization the noncondensables will redistribute between the drywell and suppression chamber via the vacuum breaker system. This redistribution takes place as drywell pressure is decreased by the steam condensation process occurring in the drywell.

The ECCS supplies sufficient core cooling water to control core heatup and limit metal-water reaction to less than 1 percent pursuant to 10CFR50.46. The excess flow discharges through the recirculation line break into the drywell. This flow of water (steam flow is negligible) transports the core decay heat out of the RPV, through the broken recirculation line, in the form of hot water which flows into the suppression pool via the suppression chamber downcomer. This flow, in addition to heat losses to the drywell walls, provides a heat sink for the drywell atmosphere. The resulting depressurization of the drywell causes the vacuum breakers to open, allowing the noncondensables in the suppression chamber to redistribute into the drywell. Table 6.2-4 provides the peak pressure, temperature, and time parameters for the recirculation line break as predicted for the conditions of Tables 6.2-3, 6.2-6, and 6.2-9 corresponding with Figures 6.2-4 and 6.2-8. The peak calculated drywell floor differential pressure is 16.89 psid.

During the blowdown period of a LOCA, the pressure suppression downcomer conducts the flow of the steam-water-gas mixture in the drywell to the suppression pool for condensation of the steam. The pressure differential between the drywell and suppression pool controls this flow versus time. Figure 6.2-10 provides the mass flow versus time relationship through the vent system for this accident.

<u>Long-Term Accident Response</u> To assess the adequacy of the primary containment following the initial blowdown transient, an analysis was made of the long-term temperature and pressure

response following the accident. The analysis assumptions are those discussed previously for the three cases of interest.

Case A - All ECCS Equipment Operating (with Containment Spray)
This case assumes that offsite ac power is available to operate all ECCS equipment. During the first 10 min following the pipe break, the HPCS, LPCS, and all three RHR pumps in LPCI mode are assumed operating. All flow is injected directly into the reactor vessel. The Operator will turn on the sprays within 20 min following the signal that drywell pressure has reached 30 psig since, in the case of a large break, the spray system is not available for the first 10 min of the transient. Therefore, the Operator will switch two RHR pumps from the LPCI mode to containment spray mode to remove the energy and airborne fission products from the primary containment after 10 min and complete this action within the next 10 min. During this mode of operation the flow from two of the RHR pumps is routed through the RHR heat exchangers, where it is cooled before being discharged into the primary containment spray headers.

The primary containment pressure response to this set of conditions is shown on Figure 6.2-5. The corresponding drywell and suppression pool temperature responses are shown on Figures 6.2-6 and 6.2-11. After the subsequent depressurization due to LPCS and LPCI core spray, energy addition due to core decay heat results in a gradual pressure and temperature rise in the primary containment. When the energy removal rate of the RHR exceeds the energy addition rate from the decay heat, the primary containment pressure and temperature reach a peak and decrease gradually.

Case B - Division I Unavailable (with Containment Spray Mode)
This case assumes electrical Divisions II and III are available. For the first 10 min following the accident, one HPCS and two RHR pumps in LPCI mode are used to cool the core. Containment spray is operating and injecting into the drywell within 20 min after the spray initiation signal following the accident. During this mode of operation the RHR pump through one RHR heat exchanger is discharged through the primary containment spray nozzles. Under this condition, one HPCS pump and one RHR pump in LPCI mode continue to cool the core after 10 min while the other RHR pump operates in containment spray mode after 20 min. The containment response to this set of conditions is shown on Figure 6.2-3. The corresponding drywell and suppression pool temperature responses are shown on Figures 6.2-7 and 6.2-11.

Case C - Division I Unavailable (Pool Cooling Mode) This case assumes electrical Divisions II and III are available. For the first 10 min following the accident, one HPCS and two RHR pumps in the LPCI mode are used to cool the core.

After 10 min, one of the RHR pumps in LPCI mode is transferred to pool cooling mode to remove energy from the suppression pool. Under this condition, one HPCS pump and one RHR pump in LPCI mode continue to flood the reactor vessel after 10 min. The primary

containment pressure response to this set of conditions is shown on Figures 6.2-2 (without feedwater) and 6.2-4 (with feedwater). The corresponding drywell and suppression pool temperature responses are shown on Figures 6.2-8, 6.2-9, and 6.2-11.

When comparing the primary containment spray Case B with the pool cooling Case C, the same duty on the RHR heat exchangers is obtained since the suppression pool temperature response is approximately the same as that shown on Figure 6.2-11. Thus, the same amount of energy is removed from the pool whether the exit flow from the RHR heat exchanger is injected back into the suppression pool or into the drywell as spray. Primary containment pressure is identical for the first 10 min; therefore, the peak pressures for Cases B and C are the same and the pressure is less than the primary containment design pressure of 45 psig.

Figure 6.2-12 shows the rate at which the RHR system heat exchanger removes the heat from the suppression pool following a LOCA. The heat removal rate is shown for the three cases of interest. The first assumes that all the ECCS equipment is available, including both RHR heat exchangers and the necessary service water pumps. The second case is for the very degraded minimum cooling condition that would limit the heat removal capability to that of one RHR heat exchanger. For all cases, it was assumed that at the time of the accident the service water temperature was 77°F and RHR heat exchanger K-factor about 80 percent of design. This is the maximum that is expected at this site and is unlikely to exist for more than a limited period in the summer. The maximum service water temperature was subsequently increased to 84°F. The impact of 84°F and power uprate is discussed in Section 6.2.1.1.6.

<u>Energy Balance During Accident</u> An energy balance for this accident showing the energy distribution at various times is given in Table 6.2-15. For the purpose of performing the energy balance, the system boundary is defined to be the primary containment. Everything within the system is grouped according to whether or not it is primarily a heat source or a heat sink, although many items behave as both during the course of the transient. Entries opposite these items represent stored internal energy at a particular time. The reference temperature for stored heat is 32°F except for heat sinks where the reference temperature is their initial temperature. For drywell heat sinks, this is 135°F; for wetwell heat sinks, 90°F. The following energy sources and sinks are considered:

1. Blowdown energy release rates.

2. Decay heat rate and fuel relaxation energy.

3. Sensible heat rate.

4. Pump heat rate.

5. Heat removal rate from the suppression pool.

6. Passive heat sinks (modeling of the passive heat sinks is fully described in Tables 6.2-1 and 6.2-2).

For the case of a DER of the recirculation line, the energy balance is performed for the reactor system, the primary containment system, and the containment heat removal systems at time zero, at the time of peak drywell pressure, at the end of reactor blowdown, and at the time of the long-term peak pressure reached in the primary containment.

Chronology of Accident Events The complete description of the primary containment response to the design basis recirculation line break is given above. A chronological sequence of events for this accident from time zero is provided in Table 6.2-16.

Main Steam Line Break

The instantaneous DER of a main steam line between the RPV and the flow restrictor results in the maximum discharge of primary system fluid and energy to the drywell. The flow on both sides of the break will accelerate to the maximum allowed by critical flow considerations. The critical flow rate is determined by using the Moody flow model with the conservative assumption of zero friction. The flow from the reactor vessel side of the break is critical in the 3.05-sq ft area main steam line nozzle. Blowdown through the other end of the break occurs because the main steam lines are interconnected upstream of the turbine by the main steam header. This interconnection allows primary system fluid to flow to the drywell via the broken line. Flow will be limited by the critical flow in the 0.913-sq ft steam line flow restrictor. The total effective flow area is thus 3.963 sq ft, which is the sum of the steam line cross-sectional area and the flow restrictor area. Section 6.2.1.3 provides information on the mass and energy release rates.

The decrease in steam pressure at the turbine inlet initiates closure of the MSIVs within approximately 200 msec after the break occurs. Also, MSIV closure signals are generated as the differential pressure across the main steam line flow restrictors increases above isolation setpoints. The instruments sensing flow restrictor differential pressure generate isolation signals within approximately 500 msec after the break occurs.

After 4 sec, the MSIVs in the broken line have closed sufficiently so that the MSIV flow area equals the flow restrictor area. At that time, the critical flow location changes from the flow restrictor to the MSIVs. Subsequent closure of the MSIVs in the broken line terminates flow from the flow restrictor side of the break at 5.5 sec after the postulated failure of the main steam line. Figures 6.2-13 and 6.2-14 show

the break schematic and total effective break area versus time, respectively. The closing time of the MSIVs is between 3 and 5 sec.

Immediately following the break, the total flow rate of steam leaving the vessel exceeds the steam generation rate. This steam flow to steam generation mismatch causes an initial depressurization of the reactor vessel, and the resultant formation of steam bubbles within the reactor vessel liquid causes a rapid rise in water level. When the froth level reaches the vessel steam nozzles and the top of the steam dryers, flow out of the break changes from steam to a two-phase mixture. The two-phase critical flow rates are determined from the Moody model with the known values of vessel pressure and mixture enthalpy. During the first second of the blowdown, the blowdown flow will consist of saturated reactor steam. This initial period of all steam discharge results in a drywell atmosphere temperature condition of approximately 310°F. Figures 6.2-15 through 6.2-18 and 6.2-19 through 6.2-22 show the pressure and temperature response of the drywell and suppression chamber during the primary system blowdown phase of the accident. Suppression pool temperature response is shown on Figure 6.2-23.

Figure 6.2-21 shows that the drywell atmosphere temperature approaches approximately 310°F after 1 sec of primary system blowdown. At that time, the water level in the vessel reaches the steam line nozzle elevation and the blowdown flow changes to a two-phase mixture. This increased flow causes a more rapid drywell pressure rise. However, the peak drywell floor differential pressure is 14.90 psi, which occurs shortly after the downcomer vent clearing transient. As the blowdown proceeds, the primary system pressure and fluid inventory decrease, resulting in reduced break flow rates.

Approximately 75 sec after the start of the accident, the primary system pressure has dropped significantly. At this time the drywell atmosphere contains only steam. The blowdown continues because the RPV is still being pressurized from the hot feedwater addition until approximately 250 sec, at which time the peak drywell pressure occurs. Following this, drywell pressure is equal to the reactor vessel pressure and initial blowdown is over. Passive heat sinks continue to remove the energy and the drywell and suppression chamber pressure gradually reduce.

Table 6.2-4 presents the peak pressures, peak temperatures, and times of this accident with feedwater addition as compared to the recirculation line break.

The drywell and suppression chamber remain in this equilibrium condition until the RPV refloods. During this period, the ECCS pumps inject cooling water from the suppression pool into the reactor. This injection of water eventually floods the reactor vessel to the level of the steam line nozzles, and at this time, the ECCS flow spills into the drywell and thus reduces the

drywell pressure. As soon as the drywell pressure drops below the suppression chamber pressure, the drywell vacuum breakers open and noncondensable gases from the suppression chamber flow back into the drywell. This process continues until the pressure in the two regions equalizes.

Intermediate Breaks

This classification covers those breaks for which the blowdown results in reactor depressurization and operation of the ECCS. This section describes the consequences to the primary containment of a 0.1-sq ft liquid break. This break area was chosen as being representative of the intermediate size break area range for both steam and liquid breaks.

Following an intermediate size break, the drywell pressure increases at a sufficiently slow rate that the dynamic effect of downcomer vent clearing is negligible and the downcomer clear when the drywell-to-wetwell differential pressure is equal to the downcomer submergence pressure. For Unit 2 primary containment design, the maximum distance between the pool surface and the bottom of the downcomer is 11 ft; thus the water level in the downcomers will reach this point when the drywell-to-containment pressure differential reaches 4.77 psid.

The ECCS is initiated by a LOCA signal from the 0.1-sq ft break and provides emergency cooling of the core. The operation of these systems is such that the reactor is depressurized in approximately 600 sec. This terminates the blowdown phase of the transient. The ECCS response is discussed in Section 6.3.

Approximately 5 sec after the break occurs, air, steam, and water start to flow from the drywell to the suppression pool; the steam is condensed and the air enters the suppression chamber. The continual purging of drywell air to the suppression chamber results in a gradual pressurization of both the wetwell and drywell over a period of approximately 1800 sec. The peak pressure will not exceed 39.75 psig. Operator action is assumed at 30 min (1800 sec) to initiate containment spray. This results in a rapid drop in containment pressure.

In addition, the suppression pool temperature is the same as following the recirculation suction line rupture (DBA) because essentially the same amount of primary system energy is released during the blowdown. After reactor depressurization, the flow through the break changes to suppression pool water that is being injected into the RPV by the ECCS. This flow condenses the drywell steam and eventually causes the drywell and suppression chamber pressures to equalize in the same manner as following a recirculation suction line rupture (DBA). The subsequent long-term suppression pool and primary containment heatup transient that follows is essentially the same as for the recirculation suction line break (DBA). From this analysis, it is concluded that the consequences of an intermediate size break

are less severe than those from a recirculation suction line rupture (DBA).

Small Breaks

This section discusses the primary containment transient response associated with small breaks. The RCPB ruptures in this category are those blowdowns that will not result in reactor depressurization either due to loss of reactor coolant or automatic operation of the ECCS equipment.

Reactor System Blowdown Considerations Following the occurrence of a break of this size, it is assumed that the Reactor Operators will initiate an orderly plant shutdown and depressurization of the reactor system. The thermodynamic process associated with the blowdown of primary system reactor coolant is one of constant enthalpy. If the primary system break is below the water level, the blowdown flow will consist of reactor water. Blowdown from reactor pressure to the drywell pressure will flash approximately one-third of the water to steam and two-thirds will remain as liquid. Both phases will be at saturation conditions corresponding to the drywell pressure. Thus, if the drywell is at atmospheric pressure, the steam and liquid associated with a liquid blowdown will be at 212°F. Similarly, if the primary containment is assumed to be at its design pressure of 45 psig, the reactor coolant will blow down to approximately 293°F steam and water.

If the primary system rupture is located so that the blowdown flow consists of reactor steam only, the resultant steam temperature in the containment is significantly higher than the temperature associated with liquid blowdown. This is because the enthalpy of high-energy saturated steam is nearly twice that of the saturated liquid.

Based upon this thermodynamic process, it is concluded that a small reactor steam leak will impose the most severe temperature conditions on the drywell structures and the safety-related equipment in the drywell. For larger steam line breaks, the temperature is nearly the same as for small breaks, but the duration of the high temperature condition is shorter. This is because the larger steam breaks will depressurize the reactor more rapidly than the orderly reactor shutdown that is assumed to terminate the small break.

Containment Response For drywell design consideration, the following sequence of events is assumed to occur. With the reactor and primary containment operating at the maximum normal conditions, a small break occurs that allows blowdown of reactor steam to the drywell. The resulting pressure increase in the drywell leads to a high drywell pressure signal that scrams the reactor and activates the primary containment isolation system. The drywell pressure continues to increase at a rate dependent on the size of the steam leak. This pressure increase lowers the

water level in the vents until the level reaches the bottom of the vents. At this time, air and steam start to enter the suppression pool. The steam is condensed and the air is carried over to the suppression chamber. The air carryover results in a gradual pressurization of the suppression chamber at a rate dependent on the size of the steam leak. Once all the drywell air is carried over to the suppression chamber, the primary containment pressure very slowly increases due to suppression pool heatup.

For a 0.1-sq ft liquid break, the drywell and wetwell pressurize over a significantly longer period than as described for the large break (approximately 7200 sec <39.75 psig). Operator action is assumed at less than 120 min (7200 sec) to initiate containment spray. This results in a rapid drop in containment pressure.

Recovery Operation The Reactor Operators are alerted to the incident by the high drywell pressure signal and the reactor scram. It is assumed that their response will be to cool down the reactor in an orderly manner using the RHR heat exchangers or main condenser and limiting the reactor cooldown rate to 100°F/hr.

It is assumed that the Operator initiates the sprays and they become effective for reduction of containment pressure 20 min after the drywell reaches 30 psig. Vacuum breakers open and the pressure between the drywell and wetwell equalizes. When the suppression pool temperature reaches 185°F the Operator transfers the one RHR pump from the containment spray mode to the reactor shutdown cooling mode. An additional 16 min is provided for the Operators to complete this action, during which time it is assumed that no cooling takes place in the RHR heat exchanger.

The reactor primary system is completely depressurized in 6 hr. At this time, the blowdown flow to the drywell ceases and the primary containment pressure and temperature begin to reduce.

Drywell Environmental Design Temperature Considerations For drywell design purposes, it is assumed that there is a blowdown of reactor steam for the 6-hr cooldown period and the passive heat sinks are neglected. The corresponding design temperature is determined by finding the combination of RCS pressure and drywell pressure that produces the maximum calculated atmospheric temperature. The maximum drywell temperature of 325.8°F occurs when the RCS is at approximately 470 psia and the drywell pressure is 50.2 psia. For the drywell structural design the maximum calculated temperature is 281.3°F, which is the saturation temperature corresponding to 50.2 psia drywell pressure.

Steam Bypass

In a pressure suppression type containment, steam released from the primary system following a postulated LOCA is condensed by the suppression pool and does not have an opportunity to produce a significant pressurization effect on the primary containment. This is accomplished by directing the steam into the suppression pool through connecting downcomer vents. This arrangement forces any steam released from the RCPB to be condensed in the suppression pool.

In the highly unlikely event of a reactor depressurization to the drywell accompanied by a simultaneous open bypass path between the drywell and suppression chamber, the leaking steam would significantly pressurize the containment. Therefore, the allowable bypass leakage is defined as the amount of steam that could bypass the suppression pool without exceeding the primary containment design pressure of 45 psig. The allowable bypass leakage is a function of the nature of the leakage path, the duration of the pressure differential across the leakage path, and the rate of condensation of leakage steam inside the suppression chamber.

The bypass area is defined in this analysis as the total flow area for leakage between the drywell and suppression chamber. Following a reactor system break, the air-steam mixture within the drywell passes through various leakage paths into the suppression chamber and causes pressurization of the suppression chamber. However, this analysis conservatively assumes that only steam leaks through the bypass area which results in the lowest bound on the allowable bypass area.

Possible leak paths are as follows:

1. Drywell floor seams.
2. Downcomer and SRV piping and penetrations.
3. Vacuum breakers.
4. Instrumentation test vents and air piping.
5. Floor and equipment drain piping and penetrations.
6. Concrete floor.

The allowable drywell bypass leakage capacity is expressed in terms of the parameter $A/\sqrt{K}$.

Where:

A = Flow area of leakage path, sq ft

K = Geometric and friction loss coefficient

This parameter is dependent only on the geometry of drywell leakage paths and is a convenient numerical definition of the overall drywell bypass leakage capacity. It results from a consideration of the flow process in the leakage paths. Assuming the steady-state noncompressible fluid flow theory (Bernoulli equation) to be applicable to the leakage flow, the pressure loss between the drywell and wetwell can be written:

$$PD - PW = \frac{KV^2}{2g_c v}\ 1/144\ psid \qquad (6.2\text{-}1)$$

Where:

PD = Drywell pressure, psi

PW = Wetwell pressure, psi

K = Total geometric and friction loss coefficient of the flow between the drywell and wetwell. These losses include entrance, exit, discontinuities, and friction. The latter is somewhat dependent upon the Reynolds number of the fluid flow but for drywell leakage consideration, it is considered constant.

V = Velocity of flow, ft/sec

g_c = Conversion factor, 32.2 lbm-ft/lbf-sec^2

v = Specific volume of the fluid flowing in the leakage path, ft^3/lbm

If the bypass leakage path flow rate is M (lbm/sec) and the flow area is A (sq ft), the above equation can be rewritten to give:

$$\dot{M} = \frac{A}{\sqrt{K}} \sqrt{2\,g_c\ (PD\ -\ PW)\ 144\ /\ v}\ lbm\,/\,\mathrm{sec} \qquad (6.2\text{-}2)$$

Thus, for a given drywell-to-wetwell pressure differential, the leakage flow (capacity) is dependent on $A/\sqrt{K}$ and the specific volume of the fluid flowing in the leakage path (which depends on the drywell internal pressure).

The purpose of the steam bypass analysis is to determine the leakage rate (in terms of bypass leakage capacity, $A/\sqrt{K}$) that would result in drywell pressurization to design pressure for the complete spectrum of line break sizes. The results of this analysis are summarized on Figure 6.2-28. This figure shows that allowable bypass leakage capacity ranges from approximately 0.061 sq ft for a large break to 0.069 sq ft for small steam line breaks.

The size of RCPB break determines the magnitude and duration of the pressure differential across the drywell leakage paths. When a very large break occurs, such as the DER of a main steam line, the high mass and energy flow from the RCPB pressurizes the drywell, generating a high pressure differential across the assumed leakage paths and producing high leakage flow rates. This short duration of reactor blowdown gives a large allowable bypass leakage capacity. When blowdown is over, the pressure differential across the leakage path dissipates and leakage flow and primary containment pressurization cease.

Small and intermediate breaks, on the other hand, result in slow RCPB depressurization. The reactor is scrammed due to the high drywell pressure resulting from the energy and mass released from the RCPB break.

During this period, the blowdown flow from the RCS forces the drywell air into the wetwell. The blowdown steam is condensed in the suppression pool, after the water level in the downcomer vents is depressed from the incoming steam and air. This results in an essentially continuous pressure differential between the drywell and suppression chamber of at least 4.76 psid. The allowable bypass leakage capacity for these conditions is an $A/\sqrt{K}$ of $>$0.05 sq ft.

The bypass leakage analysis is performed assuming that only steam leaks through the bypass paths. This assumption conservatively minimizes the allowable bypass leakage capacity by maximizing the primary containment pressurization from the assumed leakage. The results shown on Figure 6.2-28 are also based on the following assumptions and parameters shown in Table 6.2-52.

1. The pipe break, LOOP, and failure of Division II diesel generator occur at time zero.

2. Reactor coolant makeup is provided by high and low pressure ECCS pumps depending upon the reactor pressure and water level. Feedwater is unavailable due to the LOOP and unit trip.

3. To maximize steam flow from the reactor vessel through the pipe break, it is assumed that the Operator throttles the ECCS flows at 10 min after the accident. It is also assumed that the Operator does not initiate controlled reactor cooldown or actuate the automatic depressurization system (ADS).

4. For small breaks which do not depressurize the reactor, it is assumed that reactor pressure is maintained by automatic operation of the SRVs in the power-actuated relief mode according to the relief setpoints defined in Table 5.2-2.

5. The containment spray mode of the RHR system is assumed to operate at 30 min after the accident. The system flow details are shown in Table 6.2-52.

6. It is assumed that no heat or mass transfer takes place between the pool surface and the suppression chamber atmosphere.

7. Passive heat sinks, summarized in Tables 6.2-1 and 6.2-2, absorb energy from the drywell and suppression chamber. The UCHIDA heat transfer correlation is used and condensate film revaporization is limited to 8 percent.

The Unit 2 analysis is based on the manual initiation of containment sprays instead of automatic sprays as required by the Standard Review Plan (NUREG-0800). The containment spray system is QA Category I and classified as an ESF.

For the worst-case event of steam bypass, it has been determined that the Operator has 30 min following the accident to establish spray flow with one loop of the RHR system. The manual initiation of the containment spray can be accomplished in approximately 2 to 4 min considering the valve stroke times involved.

The containment transient (before and after Operator action) for a typical steam bypass event is shown on Figure 6.2-28A.

The rate of heat transfer to the passive heat sinks is dictated by the temperature difference between the containment atmosphere and the sink surface and by the nitrogen-to-steam mass ratio of the atmosphere. The surface heat transfer coefficient is determined for each heat sink as a function of the appropriate nitrogen-to-steam mass ratio from the UCHIDA correlation, which yields coefficients ranging from the minimum value of 2.0 Btu/hr-ft^2 °F to the maximum value of 280 Btu/hr-ft^2 °F. When the surface temperature of the heat sink (Ts) is greater than the saturation temperature of the atmosphere (Tg), the minimum UCHIDA coefficient is used. When Ts is less than Tg, the UCHIDA condensing heat transfer rate is compared to the convective heat transfer rate and the larger of the two is selected. The condensing heat transfer rate is given by:

$$q\ cond = huA\ (Tg - Ts)$$

The convective heat transfer rate is given by:

$$q\ conv = hcA\ (Ta - Ts)$$

Where:

hu = UCHIDA condensing heat transfer coefficient

hc = 2.0 Btu/hr-ft^2 °F for convection

A = Heat sink surface area

Ta = Drywell or suppression chamber atmosphere temperature

Tg = Saturation temperature of the drywell or suppression chamber atmosphere

Ts = Heat sink surface temperature

Under superheated drywell or suppression chamber atmosphere conditions, heat sink condensate is subject to revaporization due to convective heat gain from the superheated atmosphere. Partial revaporization limited to 8 percent of the condensate mass is assumed in this analysis.

The heat transfer coefficients for the primary containment wall (heat sinks 7 and 14 of Table 6.2-2) are shown on Figure 6.2-28B.

The spray drop thermal efficiency of 90 percent has been assumed in the steam bypass analysis. However, a sensitivity study with spray thermal efficiency reduced to 50 percent demonstrates that there is no limitation on bypass capability due to this assumption. Figure 6.2-28C shows that the drywell and suppression chamber peak pressures occur at the time of spray initiation with either 50 percent or 90 percent spray thermal efficiency.

A sensitivity study considering heat transfer from the drywell atmosphere into the suppression chamber atmosphere through the steel downcomer pipes is included in the bypass analysis. However, at the time of spray initiation (1800 sec), the downcomer metal is at 291°F compared to the spray water temperature of 113°F. Therefore, about 7.55 million Btus of sensible heat is stored in the downcomer pipes. In the worst case, this energy could vaporize the suppression chamber spray water for about 4 min after spray actuation. The sensitivity study, including this additional energy source, shows that there is no additional limitation on steam bypass capability due to this heat source.

The steam bypass analytical model assumes that nitrogen purged from the drywell due to steam blowdown is forced through the downcomer vent system and enters the suppression chamber atmosphere as dry nitrogen at the suppression pool temperature. For the limiting steam bypass case, the majority of the drywell nitrogen mass is purged to the suppression chamber approximately 300 sec after the pipe break. In this time interval, pool temperature increases from 90°F to 99°F. A sensitivity study considering nitrogen saturated with vapor at pool temperature indicates that the dry nitrogen carryover assumption has negligible effect on the steam bypass capability. With saturated

nitrogen carryover, peak containment pressure at 1,800 sec increased by approximately 0.28 psi.

Energy absorbed (integrated energy) by the containment heat sinks and sprays at various times is provided in Table 6.2-27A. The energy and mass balance is shown in Table 6.2-53.

Unit 2 utilizes all-welded construction to prevent bypass leakage between the drywell and the suppression pool air space.

To ensure that drywell bypass leakage conforms to the design basis, a leak test is conducted at the drywell-to-suppression chamber design pressure differential of 25 psid. The acceptance criterion for this test is that the measured leakage must be less than 10 percent of the bypass leakage capacity based on an $A/\sqrt{K}$ of 0.054 sq ft, in accordance with Technical Specification SR3.6.1.1.2.

At the first refueling outage a leak test was conducted to verify bypass leakage. The schedule for subsequent bypass leakage tests is in accordance with the Technical Specifications.

In accordance with 10CFR50 Appendix J, a visual inspection shall be conducted of the exposed accessible interior and exterior surfaces of the suppression chamber (including each vacuum relief valve and associated piping).

<u>Negative Primary Containment Differential Pressure</u>

The maximum negative differential pressure for the primary containment results from the assumed inadvertent actuation of the containment spray system during normal plant operation with minimum spray water temperature and minimum air mass inside the containment. The term air mass is used for simplicity and refers to the nitrogen inerted atmosphere.

Containment depressurization following a postulated pipe break in the drywell is less severe than inadvertent spray actuation because the suppression pool temperature, which dictates the final containment temperature and pressure, is increased due to blowdown steam condensation in the pool. In addition, the air mass inside the Mark II primary containment is constant during normal operation and the drywell floor vacuum breakers allow air to return to the drywell during post-LOCA drywell depressurization to limit the upward pressure differential on the drywell floor.

The following assumptions and analysis describe the worst-case negative containment pressure differential resulting from inadvertent spray actuation.

1. Air in the drywell and suppression chamber is assumed to be an ideal gas.

2. Vacuum breakers between the suppression chamber and drywell start to open at a differential pressure of 0.25 psid resulting in equalization of the pressure in the two compartments.

3. Initial conditions of minimum pressure (14.2 psia), maximum temperature (drywell at 150°F, wetwell at 122°F), and maximum relative humidity (drywell at 100%) are assumed to minimize the containment air mass.

4. The suppression pool is at the minimum temperature of 70°F and suppression chamber dew point of 70°F.

5. The service water which cools the RHR (containment spray) heat exchanger is assumed to be at the minimum temperature of 32°F.

6. The minimum spray temperature is determined as follows:

$$T_{sp} = T_p - \frac{K\left(T_p - T_{sw}\right)}{m_{sp}\ C_p} \qquad (6.2\text{-}3)$$

Where:

T_{sp} = Spray temperature

T_p = Pool temperature (70°F)

T_{sw} = Service water temperature (32°F)

m_{sp} = Spray flow rate (3.7×10^6 lbm/hr)

C_p = Specific heat of spray (1 Btu/lbm°F)

K = Heat exchanger performance factor (1,373,800 Btu/hr°F) (unfouled)

Therefore:

T_{sp} = 56°F

7. The drywell and suppression chamber pressure after spray actuation is determined assuming final temperature equal to the spray temperature.

8. The suppression chamber pressure after spray actuation is determined by adding the pressure differential across the vacuum breakers to the drywell pressure.

9. The final minimum containment pressure, considering pressure equalization between the drywell and wetwell due to the vacuum breakers, is calculated as follows:

$$P_{DW}^{F} = \frac{\left(\frac{T_{WW}^{F}}{T_{DW}^{I}}\right)\left(\frac{V_{DW}}{V_{WW}}\right) P_{DW,N}^{I} + \left(\frac{T_{WW}^{F}}{T_{DW}^{I}}\right) P_{WW,N}^{I} - \Delta P}{1 + \frac{V_{DW}}{V_{WW}}} + P_{VAPOR}^{F} \qquad (6.2\text{-}8)$$

Where:

N = Noncondensible

F = Final

I = Initial

WW = Suppression chamber

DW = Drywell

T = Temperature

P = Pressure

V_{DW} = Free volume of the drywell

V_{WW} = Free volume of the suppression chamber

ΔP = Vacuum breaker differential pressure between drywell and suppression chamber

and finally:

$P_{DW,MIN}$ = 10.18 psia

$P_{WW,MIN}$ = 10.68 psia

Assuming atmospheric pressure of 14.7 psia in the reactor building (outside containment), the maximum negative containment differential pressure is:

Maximum negative ΔP = 10.18 - 14.7
= -4.5 psid

Thus, an extremely conservative end-point analysis of this event predicts a maximum negative pressure differential of -4.5 psid compared to the design negative pressure differential of -4.7 psid.

Suppression Pool Dynamic Loads

The manner in which suppression pool dynamic loads resulting from postulated LOCAs, transients, and seismic events have been integrated into the Unit 2 design is described in the Unit 2 DAR for Hydrodynamic Loads (Appendix 6A).

Asymmetric Loading Conditions

The manner in which potential asymmetric loads were considered for Unit 2 is fully described in the DAR for Hydrodynamic Loads (Appendix 6A).

Containment Ventilation System

The primary containment ventilation system is discussed in Section 9.4.

Post-accident Monitoring

A description of the post-accident monitoring system is provided in Section 7.5.2.

Analytical Models

LOCA Containment Response Model for Large Breaks For the original, pre-uprate analysis, the pressure and temperature response of the primary containment and the suppression pool temperature response following a LOCA in the RCPB are determined as functions of time with the LOCTVS computer program. The LOCTVS program performs numerical integrations principally of the mass and energy conservation equations, and also solves the momentum conservation equation as required to determine flow rates between nodes. LOCTVS simulates behavior of the pressure suppression containment system, the RCS, the primary containment heat sources and sinks, and the containment heat removal systems. SWECO 8101, a topical report, provides a detailed description of the analytical models used and the assumptions employed[1]. The analytical models and assumptions incorporated in the LOCTVS program to predict conservatively the response of the pressure suppression containment system are summarized in the following paragraphs.

The drywell is modeled as a constant volume system, initially containing a homogeneous mixture of air and water vapor. The total energy and mass of air, water vapor, and water inside the drywell are determined at all times by numerical integration of the appropriate flows. The flows included in the LOCTVS drywell model are the reactor blowdown, the vent flow, and the heat transfer to the primary containment structures.

Blowdown from the RCPB causes the drywell pressure to increase and the level of water in the downcomer to be depressed until these are cleared. A mixture of air, steam, and water is then forced through the downcomers into the suppression pool. The

water and condensed steam are added to the suppression pool inventory while the air from the drywell is added to the suppression chamber, increasing the suppression chamber pressure and the pressure at the downcomer exit. The drywell pressure is dictated by the downcomer exit pressure, the dynamic pressure losses associated with a given downcomer flow, the drywell free volume, and the blowdown flow rate. For calculation of flow through the downcomers into the suppression pool, a homogeneous flow model is employed. This model, in combination with the frictionless Moody flow model used for blowdown calculations, provides a computed flow rate into the drywell that is large and a flow rate out of the drywell that is small; thus, the rate of storage in the drywell is large and the pressure calculated is conservatively high.

The thermodynamic state of the drywell atmosphere following a recirculation suction line break (DBA) is assumed to be always saturated. The break effluent is homogeneously mixed with the atmosphere and undergoes a pressure flash, with subsequent dropout of the decompressed and unflashed liquid.

In determining the state of the drywell (i.e., pressure and temperature) for a steam line break, the steam mass and enthalpy are added to the drywell atmosphere. Initially, the blowdown consists of saturated steam, but as the reactor vessel pressure drops the water level in the vessel swells. Eventually the froth level reaches the top of the steam dryers, and the blowdown is assumed to change from steam to a two-phase mixture.

At the time the two-phase blowdown starts, the drywell contains superheated steam and air. If instantaneous and complete mixing between the steam, air, and two-phase mixture is assumed, drywell temperature and pressure decrease. This decrease results from heat being extracted from the high temperature steam and air to evaporate the liquid phase of the blowdown. For the sake of conservatism, the assumption of instantaneous homogeneous mixing is modified for the period immediately following the start of two-phase flow until after peak drywell pressure and temperature occur (except for downcomer flow calculations where, again to be conservative, a homogeneous drywell mixture is always assumed).

The modification of the homogeneous mixing assumption is as follows. The steam portion of the blowdown goes directly to the drywell atmosphere. The liquid portion undergoes a pressure flash, where the flashed steam goes directly to the drywell atmosphere and the decompressed liquid falls to the drywell floor. Saturated steam resulting directly from the blowdown and the flashing process mixes uniformly with the existing drywell air and steam.

The mode of heat transfer to the drywell and primary containment heat sinks is determined from the individual heat sink surface temperature and the saturation temperature of the atmosphere adjoining the heat sink surface. The criteria used to determine

the heat transfer mode are described in Section 8.3 of the LOCTVS topical report[1]. The condensing heat transfer coefficient is determined as a function of the air-steam mass ratio of the adjoining volume using the UCHIDA correlation. Accordingly, the heat transfer coefficient used by LOCTVS varies from a maximum value of 280 Btu/hr-sq ft-°F to a minimum value of 2 Btu/hr-sq ft-°F, as shown on Figure 6.2-29.

LOCTVS incorporates models for downcomer clearing, downcomer flow, suppression pool swell, and suppression chamber pressurization. These models are similar to those of the GE Mark II analytical model described in NEDM-10320. The Mark II version of LOCTVS has shown excellent agreement with the analytical results presented in NEDM-10320. LOCTVS results have also been favorably compared to the Pressure Suppression Test Facility (PSTF) test results.

In this comparison the PSTF geometry and conditions were simulated using LOCTVS for two of the test cases - one with liquid blowdown and one with steam. The blowdown was released in the drywell, thereby producing high-energy pipe break conditions which cause vent clearing and pool swell in the suppression chamber.

The blowdown was calculated based on Moody flow through a venturi of given diameter. This simulated the test condition which actually controlled blowdown using a critical flow venturi. Other test parameters were duplicated such as vent submergence, vent pipe diameter, and drywell/suppression pool initial temperature. The LOCTVS model was then executed for one complete cycle of pool swell and fallback. The LOCTVS results conservatively envelope the PSTF test data as shown on Figures 6.2-29A through 6.2-29J.

In general, primary containment peak pressure is sensitive only to those variables that affect the long-term analysis, such as initial suppression pool temperature, decay heat rate, RHR heat exchanger heat transfer coefficient, and passive heat sink area (Table 6.2-2). Drywell floor peak pressure differential is sensitive only to those variables that affect the short-term analysis, such as downcomer area, submergence, air carryover rate, blowdown flow rate, and RPV water level swell time. The sensitivity analysis results are provided in Tables 6.2-17 through 6.2-27.

<u>Models for Analysis of Intermediate and Small Breaks</u> The CONSBA computer program has been utilized for the analysis of intermediate and small breaks. This program calculates the thermodynamic response of BWR primary containment and uses a finite difference technique based on input-specified time steps to solve the transient equations. In each time step, the program determines the mass and energy flow across all control volumes and performs state calculations for the reactor vessel,

suppression pool, drywell, suppression chamber atmosphere, and water on the drywell floor assuming equilibrium conditions.

CONSBA has the capability to follow an input-specified reactor cooldown rate by controlling the opening and closing of valves in the SRV system. Five groups of manually-actuated SRVs are included in the model and available to discharge steam from the reactor vessel to the suppression pool, thus allowing the plant Operators to cool down the reactor vessel. At the user's option the program will calculate the integrated mass and energy flow of each compartment and system and print a mass and energy balance.

Ten groups of automatically-actuated SRVs are included in the model and are available to discharge steam from the reactor vessel to the suppression pool, thus relieving the pressure in the reactor vessel. Valves open when reactor vessel pressure reaches a specified setpoint and remain open until the pressure drops below a specified setpoint. The flow through the SRV is assumed to be frictionless Moody flow at all times and the assumed enthalpy is that of the reactor vessel steam.

The reactor coolant is assumed to be in a saturated equilibrium condition. In the case of a pipe break in the drywell, either steam or water will be discharged from the reactor vessel depending on the location of the break and water level in the reactor vessel. Frictionless Moody critical flow tables are interpolated to determine the blowdown mass flux as a function of reactor vessel pressure and for either reactor coolant saturated liquid or steam enthalpy, depending on the type of the break specified. Blowdown flow becomes zero when reactor vessel pressure drops below the drywell pressure.

The drywell atmosphere is either a homogeneous mixture of air and steam or all steam. The suppression chamber atmosphere is always a homogeneous mixture of air and steam.

Downcomers and vacuum breakers are modeled in the program to relieve pressure buildup in the drywell or suppression chamber. Since this program is written for small or intermediate breaks, the pressure buildup in the drywell is expected to be significantly slower than the large breaks. The dynamic clearing at the downcomer is not analyzed in the program. The hydrostatic pressure at the downcomer discharge is calculated and compared with the pressure differential between the drywell and suppression chamber to determine if there is any vent flow. Flow through the downcomer is a homogeneous mixture of air and steam.

Water in excess of an input-specified maximum limit on the drywell floor is assumed to overflow through the downcomer system and is added to the suppression pool inventory in the next time step. The suppression pool and water accumulated on the drywell floor are assumed to be saturated water. The vacuum breakers will be open if the pressure differential between the suppression chamber and drywell is greater than the input vacuum breaker

pressure setpoint. Vacuum breaker flow is assumed to be a homogeneous steam and air mixture.

Heat sinks are modeled in CONSBA to determine the amount of energy absorbed by various structures of the primary containment. Concrete and steel structures (passive heat sinks) may absorb energy from the local environment after an accident due to elevated temperature in the drywell, suppression chamber, and suppression pool.

CONSBA models the ECCS and heat removal through the RHR heat exchangers. Unit 2 has two RHR heat exchanger loops. Both loops can be used for pool cooling; only one can be used for reactor shutdown cooling at a given time. Pool cooling mode is achieved by pumping suppression pool water through the heat exchanger and discharging it back to the pool. Shutdown cooling is achieved by recirculating the reactor vessel coolant through the RHR heat exchanger.

CONSBA also determines the steam or air/steam leakage rate from the drywell to the suppression chamber atmosphere bypassing the suppression pool. The Darcy equation is used to calculate this leakage.

6.2.1.1.4 Sensitivity of Suppression Chamber Air Space Temperature Increase on LOCAs

The large break LOCA analysis described in Section 6.2.1.1.3 is done using 90°F suppression chamber air space temperature. The maximum allowed temperature is 122°F. This change results in a reduction of initial suppression chamber air mass and a small decrease in the maximum calculated drywell pressure. Thus, use of 90°F is conservative.

6.2.1.1.5 Section Deleted

6.2.1.1.6 Impact of Extended Power Uprate on Containment Response Analysis

Section 6.2.1.1.3 of the NMP2 USAR provides the original licensed thermal power containment responses to various postulated accidents that validate the design basis for the containment. This section addresses the impact of EPU to 3988 MWt (Reference 16). Operation at EPU changes some of the conditions for the containment analyses. For example, the short-term DBA-LOCA containment response during the blowdown is governed by the blowdown flow rate. This blowdown flow rate is dependent on the reactor initial thermal-hydraulic conditions, such as vessel pressure and the mass and energy of the vessel fluid inventory, which change slightly with EPU. Also, the long-term heatup of the suppression pool following a LOCA or a transient is governed by the ability of the RHR to remove decay heat. Because the decay heat depends on the initial reactor power level, the

long-term containment response is affected by EPU. The containment response was reanalyzed only for limiting cases to demonstrate the plant’s capability to operate with a rated power increase to 3988 MWt. The EPU analyses of containment pressure and temperature responses are described below. These analyses were performed at 102 percent of EPU rated thermal power (RTP) level.

Recirculation Line Break - Short-Term Accident Response

The short-term analysis is directed primarily at determining the containment pressure response during the initial blowdown of the reactor vessel inventory to the containment following a large break inside the drywell. The short-term containment response analyses were performed for the limiting DBA LOCA that assumes a double-ended guillotine break of a recirculation suction line to demonstrate that EPU does not result in exceeding the containment design limits. The short-term analysis covers the blowdown period during which the maximum differential pressure between the drywell and wetwell occur. The peak calculated drywell-to-wetwell pressure also remains within its design value (Table 6.2-4).

Due to the NMP2 containment configuration with respect to crucial parameters such as downcomer vent area to break area, the NMP2 initial peak drywell pressure excursion during the initial blowdown period of a DBA LOCA would not produce the most limiting containment pressure conditions. As a result, the potential that the drywell pressure will exceed this initial spike at some time later in the transient when the entire content of the drywell air has been transferred into the wetwell was examined. Therefore, in addition to the typically analyzed suite of short-term cases run for a limited duration of about 40 sec, additional extended-short-term cases were analyzed that include a longer transient run extended to over 250 sec of transient time to ensure that the peak pressure condition is analyzed.

The M3CPT code was used to model the short-term containment pressure and temperature response.

Recirculation Line Break – Extended-Short-Term Accident Response

For NMP2, the extended-short-term analysis remained limiting for peak pressure at EPU. Table 6.2-4 includes comparisons of the pressure values calculated for EPU to the design pressures and to pressure values from previous calculations. The EPU containment peak analysis is performed using two sets of containment initial conditions. For normal plant operation at power, the drywell temperature of 105°F, drywell relative humidity of 40 percent, and suppression pool temperature of 90°F were confirmed to bound normal operating conditions based on the statistical analysis. The maximum calculated containment pressure for EPU remains within the P_a of 39.75 psig. In addition, an analysis was

performed to ensure that the drywell accident pressure would not exceed 45 psig design pressure based on the worst combination of initial conditions (drywell temperature of 70°F, drywell relative humidity of 20 percent, and suppression pool temperature of 90°F). The extended-short-term containment pressure and temperature response is shown on Figures 6.2-4a and 6.2-8a, respectively.

The M3CPT code is no longer utilized to determine the second pressure peak (i.e., extended-short-term containment analysis); however, the M3CPT code is utilized to provide the mass and energy input to GOTHIC code. The GOTHIC code (version 7.2b) is utilized to determine the peak containment pressure for the second peak.

<u>Recirculation Line Break – Long-Term Accident Response</u>

The analysis of the DBA LOCA was performed at 102 percent of EPU RTP. Based on the original pre-uprate sensitivity study described in Section 6.2.1.1.3, Case C (with feedwater) is re-analyzed for EPU. The long-term containment response is shown on Figures 6.2-4b and 6.2-8b. The calculated peak values for LOCA bulk pool temperature for the original pre-uprate and the EPU RTP case are compared in Table 6.2-4. The EPU analyses were performed using NMP2-specific decay heat table based on ANS/ANSI 5.1-1979 with 2-sigma adders with additional actinides and activation products per GE SIL 636 (Reference 11). The analysis assumed the single failure of one of the two RHR heat exchangers. The EPU analysis also used more realistic RHR heat exchanger performance assumptions to limit the predicted suppression pool temperature to its original pre-uprate value. The resulting calculated peak bulk suppression pool temperature is 207°F. This temperature is well within the suppression pool temperature design of 212°F.

The wetwell gas space peak temperature response was calculated assuming a heat transfer model that promotes thermal equilibrium between the pool and wetwell gas space. Table 6.2-4 shows the calculated bulk pool temperature of 207°F for the DBA LOCA at EPU. The wetwell gas space temperature is also 207°F for EPU. The wetwell gas temperatures are less than the wetwell design temperature of 270°F.

A chronological sequence of events for the EPU accident is provided in Table 6.2-16a.

The SHEX code was used to model the long-term containment pressure and temperature response. The key models in SHEX are based on models described in Reference 7.

<u>Main Steam Line Break</u>

Based on the review of the original pre-uprate containment analysis described in Section 6.2.1.1.3, it is concluded that the DER of the main steam line is not the limiting break for the containment design basis. Therefore, the containment accident analysis was not performed for the main steam line break.

Intermediate Breaks

Based on the review of the original pre-uprate containment analysis described in Section 6.2.1.1.3, it is concluded that the intermediate break is not the limiting break for the containment design basis. Therefore, the containment accident analysis was not performed for the intermediate break.

Small Breaks

Based on the review of the original pre-uprate containment analysis described in Section 6.2.1.1.3, it is concluded that the small breaks are not the limiting breaks for the containment design basis with the exception of the drywell environment temperature. Therefore, the containment accident analysis was not performed for the small breaks. The impact on the air temperature is discussed below.

Containment Gas Temperature Response

The drywell design temperature (340°F) has been determined based on a bounding analysis of the superheated gas temperature which can be caused by a blowdown of steam to the drywell during a small break LOCA. This analysis conservatively determined a bounding combination of vessel pressure and drywell pressure that produces a maximum calculated drywell temperature. The USAR reported that expansion of reactor steam under these conditions would result in a calculated peak drywell temperature of 325.8°F. These bounding conditions, which are described in Section 6.2.1.1.3, are derived independent of the initial reactor power. Therefore, EPU has no effect on the peak drywell temperature.

The wetwell gas space peak temperature response was calculated assuming a heat transfer model that promotes thermal equilibrium between the pool and wetwell gas space. Table 6.2-4 shows the calculated bulk pool temperature of 207°F for the DBA LOCA at EPU. The wetwell gas space temperature is also 207°F for EPU. The wetwell gas temperatures are less than the wetwell design temperature of 270°F.

Steam Bypass

The original pre-uprate steam bypass analysis assumes an initiation time for containment sprays of 30 minutes. Use of this initiation time at EPU conditions results in a primary containment pressure increase that is within the primary containment design; however, the margin to the design pressure is

significantly reduced. The current licensing basis for the initiation of containment sprays for similar events provided in Section 6.2.1.1.3 (Assumptions for Long-Term Cooling) and in the Alternative Radiological Source Term Safety Evaluation (Reference 20) is 20 min. In addition, the OLTP steam bypass analysis assumption #6 was changed to credit the heat and mass transfer between the pool surface and the suppression chamber atmosphere consistent with the GEH steam bypass analysis method. For this reason, the primary containment steam bypass analysis for EPU was performed assuming a containment spray initiation time of 20 min and resulted in the restoration of the margin to the design pressure.

Negative Primary Containment Differential Pressure

The original pre-uprate analysis and assumptions/inputs described in Section 6.2.1.1.3 are not affected by the EPU.

Suppression Pool (Containment) Dynamic Loads

The impact of the EPU is discussed in Sections 6A.3.1 and 6A.4.1.

Asymmetric Loading Conditions

See Section 6.2.1.2.5.

Analytical Models

The analytical models for the containment analysis remains the same as those described in Section 6.2.1.1.3 with the exception of large break accidents.

The M3CPT code in lieu of LOCTVS was used to model the short-term containment pressure and temperature response and suppression pool dynamic loading evaluation. The modeling used in the M3CPT analyses is described in References 6 and 7. References 6 and 7 describe the basic containment analytical models used in GEH codes. Reference 10 describes the more detailed RPV model (LAMB) used for determining the vessel break flow in the containment analyses for EPU. The use of the LAMB blowdown flow in M3CPT was identified in ELTR1 by reference to the LAMB code qualification in Reference 10.

The GOTHIC code (version 7.2b) in lieu of LOCTVS is utilized to determine the peak containment pressure for the extended-short-term containment peak pressure to demonstrate that the DBA LOCA peak pressure is within the P_a and design pressure. The M3CPT code is no longer utilized to determine the second pressure peak (i.e., extended-short-term containment analysis); however, the M3CPT code is utilized to provide the mass and energy input to GOTHIC code.

The SHEX code in lieu of LOCTVS was used to model the long-term containment pressure and temperature response. The key models in

SHEX are based on models described in Reference 7. The SHEX code is used for the long-term bulk pool temperature response and the limiting alternate shutdown activity in Section 15.2.

6.2.1.1.7 Impact of Maximum Extended Load Line Limit-Plus (MELLLA+) Operation on Containment Response Analysis

Section 6.2.1.1.3 of the NMP2 USAR provides the original licensed thermal power containment responses to various postulated accidents that validate the design basis for the containment. Section 6.2.1.1.6 presents analysis performed to support extended power uprate (EPU) of the plant to 3988 MWt (Reference 16). This section addresses subsequent evaluation of containment response to support allowable operation in the extended operation domain to the Maximum Extended Load Line Limit-Plus (or, MELLLA+), based on lower allowable core flow and the EPU power conditions.

Operation on the MELLLA+ operating domain has the potential to affect the break flow mass and energy for a double-ended break of a recirculation suction line (DBA-LOCA) with possible impact on containment peak pressures. The effect on break flow mass and energy is dependent on reactor initial thermal-hydraulic loads, as well. A generic disposition has been developed as presented in the MELLLA+ Licensed Topical Report (Reference 21) which confirms that no change would be expected for the Long-Term Suppression Pool Cooling Temperature Response since sensible and decay heat do not increase in the MELLLA+ operating domain compared to the EPU basis.

<u>Recirculation Line Break-Short-Term Accident Response</u>

Analysis has been performed, using the same methods as described in Section 6.2.1.1.6 for the short-term accident response for peak pressure within the MELLLA+ domain and demonstrated that results on all key parameters such as peak drywell pressure remain bounded as compared to results obtained for Nine Mile Point Unit 2 under the assumption of EPU operation and full flow condition.

<u>Recirculation Line Break-Extended Short-Term Accident Response</u>

The extended short-term analysis discussed in Section 6.2.1.1.6 was evaluated and determined to be insensitive to changes occasioned by operation in the MELLLA+ operating domain. For this longer-term effect, the minor variability in the initial vessel inventory energy associated with the various points in the operating range would have negligible impact in comparison to the overall mass and energy contributions. Further, with confirmatory analysis demonstrating the bounding character of the recirculation line break for the short-term accident response (above), it was concluded that no re-analysis would be warrented to confirm the acceptability of this extended short-term response. It would remain bounded by the EPU results of Section 6.2.1.1.6.

Suppression Pool (Containment) Dynamic Loads

The impact of the MELLLA+ domain on hydraulic loads is discussed in Sections 6A.3.1 and 6A.4.1.

6.2.1.2 Containment Subcompartments

6.2.1.2.1 Design Bases

The drywell subcompartments are designed in accordance with the following criteria:

1. A pressure response analysis is given for each primary containment subcompartment containing high-energy piping in which breaks are postulated. The definition of high-energy piping and the criteria for postulating breaks are outlined in Section 3.6A.

   The break selected for the design evaluation produced, by virtue of its size and location, the greatest release of blowdown mass and energy into the subcompartment, during normal operation and hot standby condition. The breaks used in the design evaluations are listed in Section 6.2.1.2.3.

2. All circumferential breaks are considered to be fully double ended and no credit is taken for limiting blowdown generation due to pipe restraint locations. The effective cross-sectional flow area of the pipe is used in the jet discharge evaluation for breaks.

3. The suppression chamber and suppression pool are assumed not available to relieve pressure from the drywell region.

4. No heat sink credit is taken.

5. The design pressure differentials for all subcompartments are higher than the calculated peak pressure differentials resulting from the postulated pipe breaks.

6. No credit is taken for blowout panels in the subcompartment analyses.

6.2.1.2.2 Design Features

For the most part, the drywell is a large continuous volume interrupted at various locations by piping, grating, ventilation ducting, etc. Two volumes within the drywell classified as subcompartments are:

1. Reactor Pressure Vessel (RPV) - Biological Shield Wall (BSW) Annulus The 1 ft 8 1/2-in thick cylindrical primary shield wall surrounds the RPV. It has an outside radius of 15 ft 9 1/4 in and extends from the reactor pedestal elevation to el 314 ft 1 1/2 in. Breaks in the recirculation water discharge and suction piping, LPCI piping, LPCS piping, and feedwater piping are analyzed. Venting occurs through the top of this annular region to the drywell and also through the flow diverter doors on the recirculation suction lines for the case of a DER of a recirculation suction line.

2. Drywell Head The drywell head surrounds the RPV head. The detachable portion connects to the refueling bulkhead (Figure 6.2-30) at el 329 ft 7 1/8 in. The vent area supplied through the refueling bulkhead consists of two ventilation exhaust openings at azimuths 105 and 285 deg and four annular vent areas at azimuths 45, 165, 225, and 345 deg (Figure 6.2-30A). All vent areas are normally open and are closed only during refueling.

   The ductwork that extends up to or penetrates the refueling bulkhead openings is not considered available vent area. However, the open annular areas around the ductwork are considered available as open vent area.

   Breaks are postulated in the RCIC head spray line and the recirculation suction piping.

   Drawings depicting piping, equipment, and compartment/venting locations are provided in Section 3.6A. The volumes and vent areas are discussed in Section 6.2.1.2.3. The subcompartments described do not incorporate blowout panels. No credit is taken for vent areas that become available after the pipe break occurs.

6.2.1.2.3 Design Evaluation

The evaluations described in this section are based on original rated design conditions. The impact of power uprate is addressed in Section 6.2.1.2.5.

The breaks utilized in the design evaluation of the primary containment subcompartments are listed in Table 6.2-28. The tables and figures that contain the nodal parameters and results for each analysis are also listed in Table 6.2-28.

The primary containment subcompartment design evaluations were performed with the SWEC THREED computer code. THREED considers two-phase, two-component (steam-water-air) flow through the vents and accounts for the fluid inertia effects. A description of the THREED analytical model is provided in Appendix 6B.

The blowdown mass and energy releases for each of the breaks are provided in the tables that are cross-referenced in Table 6.2-28. For all cases, the blowdown data are based upon conservative methodology developed by GE using the Moody steady-slip flow model with subcooling[3].

The assumed initial conditions for the subcompartment volumes are conservatively chosen to maximize transient differential pressure responses. The initial conditions are given in the subcompartment nodal description tables. The minimum relative humidity of 20 percent assumed for several of the subcompartment analyses is based on meteorological data. The review of 2 yr of these data indicates that the relative humidity of the environment was at or above 20 percent for 99.85 percent of the time. However, a sensitivity run, assuming zero percent relative humidity, for the case of a 12-in feedwater line break in the RPV-BSW annulus, showed a pressure increase of only 0.2 psi in the break node, and a maximum pressure increase of 0.85 psi adjacent nodes.

The piping systems assumed to rupture in the subcompartments are identified in Table 6.2-28. Break locations are discussed in Section 3.6A.

The description of and justification for the subsonic and sonic flow models, and the degree of entrainment used in vent flow calculations are given in Appendix 6B.

The subcompartment nodalization schemes are tabulated and provided. The nodalization schemes are selected to maximize pressure differentials across node boundaries. Restrictions resulting from structural components or equipment placement were selected as nodal boundaries for the flow models.

<u>RPV-BSW Annulus Models</u>

RPV-BSW annulus volumes modeled are shown on Figures 6.2-52, 6.2-58, 6.2-61, 6.2-63, and 6.2-65. A 20-node model was used for the recirculation suction line break, the recirculation discharge line break, and the LPCI line break, and a 21-node model was used for the feedwater line break and the LPCS line break. Only half of the annulus is modeled in each case due to symmetry about the location of the rupture. Nodes close to the break are made comparatively small with respect to the other nodes since the pressure gradient is greater close to the break node.

Flow diverters are incorporated in the BSW penetration sleeves for the two recirculation water suction lines. These diverters help to minimize the asymmetric loads resulting from the pressurization of the RPV-BSW annulus. Flow diverters are not used for the recirculation water discharge lines, feedwater lines, LPCI lines, or the LPCS line.

No credit is taken for any penetrations through the BSW, other than flow diverters described earlier, that might allow additional venting out of the annulus. All mass flow from the RPV-BSW annulus is required to vent around the stabilizers on top of the BSW. When the blowdown occurs, pressure differentials develop across the BSW. When the mass flow leaves the top of the annulus and begins to enter the drywell, the pressure differentials decrease to values well below the peaks. Vent paths where choked flow occurs are indicated in the vent path description tables for each analysis. During the blowdown the RPV insulation panels are assumed to displace toward the BSW. The insulation is considered to be incompressible.

Flow around pipes and nozzles is modeled as flow encountering sudden contractions and expansions in the flow area. Pipes obstructing flow are projected onto the next junction.

Drywell Head Analysis

The drywell head analysis considers pipe breaks that produce upward and downward loadings on the refueling bulkhead. Downward pressure on the refueling bulkhead is determined by the RCIC head spray line case. Upward pressure values on the refueling bulkhead are based on a recirculation suction line break. The ductwork is assumed to stay in place for the duration of the analysis. No flow is considered through the ducts. The volume between the RPV head and the surrounding insulation is considered available during the transient and the insulation is assumed incompressible.

Flow Coefficients

The flow coefficient (C) for a particular geometry is determined as a function of the equivalent head loss coefficient (K_{eff}) for that flow system as follows:

$$C = \frac{1}{\sqrt{K_{eff}}} \qquad (6.2\text{-}11)$$

The value of K_{eff} is simply the sum of the head losses for separate parts of the system. These head losses are defined as follows:

1. Entrance Loss or Contraction Determined as a function of the ratio of the upstream cross-sectional area to the cross-sectional area of the contraction.

2. Exit Loss or Expansion Determined as a function of the ratio of the cross-sectional area upstream of the expansion to the cross-sectional area of the expansion.

3. Resistance of Bends to Flow of Fluid Determined by the angle and length of the bend.

4. Friction Loss Although generally very small, it is calculated as an fl/d term, where f is the nondimensional friction coefficient, l is the junction length, and d is the cross-sectional hydraulic diameter.

5. Form Losses They are due to objects in the flow path, such as grating, and included in the vent path description tables with the friction losses. The formulas in APED-4378[2] are used to calculate the form losses.

The previously-listed losses are defined specifically in APED-4378[2] and NEDO-24548[3]. Values for the respective components are listed in the vent path description table for each break analyzed.

The pressure response graphs of nodes within each subcompartment model are provided with the figure numbers listed in Table 6.2-28.

Sensitivity Studies

Force and moment time-histories are calculated for all five postulated pipe breaks in the RPV-BSW annulus. The feedwater line DER produces the greatest moment on the BSW. The recirculation inlet DER produces the greatest forces on the BSW along the break axis.

A force and moment sensitivity study has been done on the feedwater line DER nodalization model. This nodalization model has been expanded from 21 to 37 nodes to determine the effect additional nodes have on the force and moment values. The 37-node model is shown on Figure 6.2-55. Nodal and vent path descriptions are shown on Tables 6.2-38 and 6.2-39, Sheets 1 through 6. The blowdown mass and energy release for the 21- and 37-node models are identical (Table 6.2-37).

Peak forces and moments in the 37-node model are approximately 9 percent less than those calculated in the 21-node model. This is due to the difference in the configuration of the two models. The 21-node model projects the areas of pipes within a node onto the nodal boundaries. This produces smaller and more conservative junction areas and restricts flow to a greater extent. In the 37-node model, finer meshing reduces the need to project pipe areas.

Force and moment data and results are provided with the figure and table numbers of Table 6.2-46.

The 37-node feedwater line DER model produces a peak nodal pressure differential of 34.83 psid versus 28.62 psid for the 21-node model. Nodal pressure and pressure differential responses are shown on Figures 6.2-56, Sheets 1 through 10, and Figure 6.2-57.

No sensitivity study has been done for the drywell head subcompartment because it is an open hemispherical volume.

6.2.1.2.4 Asymmetric LOCA Loads

The guidance of NUREG-0609 is used to analyze asymmetric LOCA loads. The following is a brief description of the methodology:

1. Pressure-Time Histories

   The pressure-time histories in the annulus region between the RPV and shield wall are generated from feedwater, LPCI, LPCS, and recirculation line breaks. The SWEC THREED code, which uses nodalized mass and energy balance, is applied in this analysis.

2. Concentrated Force-Time Histories

   The forcing function of jet impingement on the shield wall is obtained from the break flow transient caused by these line breaks. Forcing functions of jet reaction on the RPV, jet impingement on the RPV, and the pipe whip restraint load on restraint anchors are obtained from these line breaks.

3. Integrated Dynamic Analyses

   Beam models are used with the pressure-time and concentrated force-time histories to determine the effects on the shield wall, pedestal, vessel support skirt, core support and internals, and control rod drives (CRDs). These dynamic analyses yield displacements, accelerations, forces, stresses, and moments.

4. Attached Piping Analysis

   Acceleration-time history from the integrated dynamic analysis is used to generate response spectra for the stress analysis of the attached piping. This analysis covers ECCS lines, primary coolant piping, and associated piping supports.

5. Load Combination for Vessel and Piping

   Asymmetric LOCA loads, combined with SSE by the square root of the sum of the squares (SRSS) methodology, are treated as a faulted condition for evaluation versus

the ASME Code. Load combinations are given in Tables 3.9A-2 and 3.9B-2.

6.2.1.2.5 Impact of Extended Power Uprate on Subcompartment Pressurization and Annulus Pressurization Evaluations

An annular structure of reinforced concrete is located inside the drywell around the RPV in order to provide thermal and radiation shielding, and is called the BSW. The BSW is designed to withstand the differential pressure that would develop across the wall as a result of a high-pressure pipe break within the annulus (i.e., between the RPV and the BSW). A pipe break in this region results in a combination of four dynamic loads, collectively referred to as annulus pressurization (AP) loads. These four dynamic loads consist of: 1) the asymmetric pressurization of the annular area between the BSW and RPV, 2) the jet reaction resulting from the break flow through the vessel nozzle, 3) the jet impingement on the vessel of the break flow from the broken pipe, and 4) the impact load absorbed by the pipe whip restraint. These loads are a function of the break size, location, fluid thermal-hydraulic conditions, and the annular vent area to the rest of the drywell.

The second subcompartment region analyzed is the volume between the RPV head and the drywell head. Because of the limited vent area between the drywell head compartment and the drywell, a pipe break in either region can cause a differential pressure load across the drywell head refueling bulkhead plate.

<u>AP Load Evaluation</u>

The review of the impact of EPU conditions on the AP load identified several non-conservative assumptions including GEH Safety Communication SC09-01. These issues were related to the original design basis for mass energy release, the dynamic structural response for off-rated conditions, and the limiting mass energy release. In order to reconcile these non-conservative assumptions, a review of the AP pressure time histories was performed for the large piping segment breaks within the annulus for effects including the structural dynamic response of the reactor vessel, reactor vessel internals, attachments to the vessel, and attachments to the BSW. The breaks included the reactor recirculation discharge, feedwater, and LPCI for several power flow points along the maximum extended load line limit analysis (MELLLA) boundary from minimum flow through maximum EPU power. This review determined that the impact of EPU operation on the resulting structural forces and accelerations was an increase in the range of 0 percent to 8 percent when compared to those for OLTP/CLTP (current licensed thermal power) operation. However, the result of the combined changes of the non-conservative assumptions and EPU is an increase in the AP load structural forces and accelerations in the range of 0 percent to 45 percent for most of the components and structures evaluated, with the increases for a few components

in the range of 63 percent to 133 percent. The results from the updated dynamic analyses, including impacts from both EPU and the non-conservative assumptions, were compared against those used as input to the component structural analyses of record. The effect of the increase in AP loads on the total component stresses is reduced when the AP loads are combined with the SSE seismic loads by the square root of the sum of the squares in the faulted load combination. The SSE seismic loads in the load combination are not affected by EPU. The results of these evaluations show that all reactor vessel and internals and associated vessel attachments and supports remain within design basis faulted allowable limits.

The containment structures, systems and components (SSC) important to safety will continue to be protected from the dynamic effects resulting from pipe breaks and that the subcompartments will continue to have sufficient margins to prevent fracture of the structure due to pressure difference across the walls following implementation of the proposed EPU.

Subcompartment Pressurization Evaluation

The pressure loading on the drywell head refueling bulkhead plate due to a postulated break in the RCIC head spray line in the drywell head subcompartment is not affected by EPU because the steam dome pressure is constant with the CLTP. However, the postulated recirculation suction line break in the drywell affects the upward pressure loading on the bulkhead plate. The fluid enthalpy at the break location is not significantly affected (less than 1 percent), while the break location pressure is essentially the same as at CLTP. Consequently, the mass and energy release from the break is not significantly affected. The upward pressure loading on the plate has more than 10 percent margin to the allowable stresses. Therefore, the drywell head refueling bulkhead plate design remains adequate.

As discussed earlier, the differential pressure loading on the BSW is not significantly affected by the EPU. The peak BSW asymmetric pressure load resulting from the limiting recirculation pump discharge line break at CLTP and at EPU conditions remains below the BSW design differential pressure. The original BSW design used conservative asymmetric and uniform pressure loads.

Annulus Pressurization Load Increase - Piping, Components and Supports

During the review of the impact of the EPU conditions on the AP load break energy, several non-conservative assumptions, including GEH Safety Communication SC09-01, were discovered related to the original design basis for mass energy release and the limiting mass energy release. As a result, new AP acceleration response spectra were developed for use in the evaluation of piping, components and supports. The new AP

acceleration response spectra, at some nodal locations, demonstrated shifts in frequency, and/or increased accelerations over the CLTP AP acceleration response spectra.

Annulus pressurization between the RPV and BSW most directly affects piping that is inside the drywell, attached to the RPV, and/or supported from the BSW. RPV attached (RCPB) piping (ASME Class 1) systems include FWS, MSS, CSH, CSL, ICS, RCS, RHS, SLCS and WCS. Of these, only the FWS and MSS systems experienced changes in flows, pressures or temperatures as a result of EPU. RCS (including the attached RHS shutdown cooling piping), CSL, WCS and ICS head spray piping inside primary containment, while not affected by EPU (no increase in flow, pressure or temperature), was also evaluated for the increased AP load. These systems were selected based on the locations with the minimum margin to design allowable at the vessel nozzle and include all the Class 1 systems discussed in Section 6A.9.1.1.4. The quantitative analysis determined that the piping, components and supports for these systems are within the design basis required margins. Based on this analysis, the other Class 1, 2 and 3 systems inside the drywell were assessed qualitatively to validate that all design margins are maintained at EPU conditions.

In addition to piping inside the drywell, the increased AP acceleration response spectra were used to evaluate: 1) piping inside the wetwell (primary containment below the drywell floor), and 2) piping outside, but connecting to primary containment penetrations. Even with the increased AP loads, other hydrodynamic loads remain governing or dominant in the determination of limiting faulted condition loads and stresses. Thus, the limiting faulted condition loads and stresses at these locations are unaffected by the increased AP loads.

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

The OLTP containment analyses use the LOCTVS computer program to calculate the mass and energy released following postulated large break LOCAs, as described in the following subsections. LOCTVS also calculates the containment system response as described in Section 6.2.1.1.3. For a description of the mass and energy release models incorporated by LOCTVS, see SWECO 8101[1].

Methods used for EPU and MELLLA+ containment response analyses are addressed in Section 6.2.1.1.6 and 6.2.1.1.7, respectively.

6.2.1.3.1 Mass and Energy Release Data

<u>OLTP</u> Table 6.2-7 provides the mass and enthalpy release data for the recirculation suction line DER. Figures 6.2-34 and 6.2-35 show the blowdown flow rates and enthalpy for the recirculation line break. Table 6.2-50 provides the mass and enthalpy release data for the main steam line break. Figures 6.2-36 and 6.2-37

show the vessel blowdown flow rates and enthalpy for the main steam line break as a function of time after the postulated rupture.

EPU Figures 6.2-34a and 6.2-35a show the blowdown flow rates and enthalpy for the limiting recirculation line break.

MELLLA+ As confirmed in Section 6.2.1.1.7, the blowdown flow rates and enthalpy from conditions within the MELLLA+ operating domain are bounded by the results presented for the EPU condition.

6.2.1.3.2 Energy Sources

The RCPB conditions prior to the DER of recirculation suction line are presented in Table 6.2-9 for OLTP and Table 6.2-9a for EPU analyses. Reactor blowdown calculations for containment response analyses are based on these conditions during a LOCA.

OLTP Following each postulated accident, the stored energy in the RCPB and the energy generated by fission product decay will be released. The rate of release of core decay heat for the evaluation of the primary containment response to a LOCA is provided in Table 6.2-10. Following a LOCA, the sensible energy stored in the RCPB metal will be transferred to the recirculating ECCS water and will thus contribute to suppression pool and primary containment heatup. Table 6.2-12 shows the reactor system metal and core-stored energy release rate to the reactor coolant following a recirculation line break.

EPU The EPU analysis includes the same energy sources used by OLTP analysis. The decay heat for short-term and extended-short-term is based on ANS 5-1971 + 20 percent, while long-term analysis for peak suppression pool analysis is based on ANS 5.1-1979 + 2 sigma. The combined heat release to coolant is shown in Table 6.2-10a.

MELLLA+ The MELLLA+ analysis is based on consistent energy sources as the EPU analysis.

6.2.1.3.3 Effects of Metal-Water Reaction

OLTP The primary containment systems are designed to accommodate the effects of metal-water reactions and other chemical reactions following a postulated DBA. The amount of metal-water reaction considered in the LOCTVS blowdown data is 0.7 percent which is consistent with the performance objectives of the ECCS assumed to occur in the first 2 min following the pipe break (Table 6.2-13). The above metal-water reaction value is 5 times that calculated in accordance with 10CFR50 Appendix K.

EPU The calculated metal-water reaction is <0.1 percent.

MELLLA+ The calculated metal-water reaction for MELLLA+ is compared to the EPU result.

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

This is not applicable.

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

This is not applicable.

6.2.1.6 Testing and Inspection

Containment testing and inspection programs to verify the structural adequacy of the primary containment are described in Section 3.8.1.7. The tests for verifying that the containment and drywell leakage rates are within allowable limits are described in Section 6.2.6.

6.2.1.7 Instrumentation Requirements

Description

Safety-related instruments and controls are provided for automatic and manual control of the containment atmosphere monitoring system (CMS). The controls and monitors described below are located in the main control room. The control logic is shown on Figure 6.2-38.

Additional requirements for containment atmosphere monitoring will be provided in accordance with Section 1.10, Task II.F.1.

Operation

The CMS monitors and/or samples the following parameters:

1. Primary containment pressure.

2. Primary containment atmosphere and suppression pool water temperatures.

3. Primary containment hydrogen and oxygen.

4. Primary containment particulate radiation.

5. Primary containment gaseous radiation.

6. Primary containment radioactive iodine (sample collection only).

7. Suppression chamber pressure.

8. Suppression pool level.

The CMS can continuously sample the hydrogen/oxygen content of the primary containment by drawing samples from five different areas: three from the drywell and two from the suppression chamber. The sampling point can be automatically switched by a cycle timer for the drywell samples or manually selected for both the drywell and suppression chamber areas. Sampling consists of continuous drawing, analyzing, and returning the sample to the area from which it was drawn.

The hydrogen-oxygen analyzer is controlled manually. An interlock is provided to automatically stop the analyzer, or prevent it from starting, when either of the hydrogen-oxygen analyzer instrument isolation valves is closed or when a LOCA or manual isolation signal is present. The CMS primary containment isolation valves close automatically on receipt of manual isolation or LOCA signal. The valves can also be controlled manually. The LOCA signal can be overridden by a keylock switch. The CMS drywell inboard sampling valves are controlled automatically by the cycle timer and the sample path selector. The valves can also be operated manually.

<u>Monitoring</u>

Indicators are provided for:

1. Primary containment hydrogen-oxygen concentration.
2. Drywell pressure (normal and wide ranges).
3. Suppression chamber pressure (normal and wide ranges).
4. Suppression pool level (normal and wide ranges).
5. Drywell temperature.
6. Suppression chamber temperature.
7. Suppression pool water temperature.
8. Drywell atmosphere train leakage gaseous radiation level.
9. Drywell atmosphere train leakage particulate radiation level.

Recorders are provided for:

1. Primary containment hydrogen-oxygen concentration.
2. Drywell pressure (normal and wide ranges).

3. Suppression chamber pressure.

4. Suppression pool level.

5. Drywell temperature.

6. Suppression chamber temperature.

7. Suppression pool water temperature.

8. Drywell atmosphere train leakage gaseous radiation level.

9. Drywell atmosphere train leakage particulate radiation level.

Alarms are provided for each CMS division:

1. Primary containment hydrogen-oxygen concentration high.

2. Drywell pressure high.

3. Drywell temperature high.

4. CMS primary containment isolation valve inoperable.

5. CMS LOCA override.

6. Suppression pool water temperature high.

7. Process airborne radiation monitor activated.

6.2.2 Containment Heat Removal System

The containment heat removal function is accomplished by either the suppression pool cooling or the containment spray mode of the RHR system. Containment spray consists of two independent spray headers in the drywell and one common spray header in the suppression chamber. Containment spray water is discharged through spray nozzles by the RHR pumps and cooled by the RHR heat exchangers.

While credit is taken for containment spray fission product removal following a large break accident to support the alternative source term (AST) methodology, containment spray heat removal is not required for this event. However, the sprays are necessary to limit the primary containment pressure following a small or intermediate steam line break considering steam bypass factor of at least 0.05 sq ft $A/\sqrt{K}$.

Systems that control fission products in order to reduce the concentration and quantity of fission products released to the environment are discussed in Section 6.5.

6.2.2.1 Design Bases

Systems utilized for post-accident containment heat removal meet the following safety design bases:

1. Systems are designed to limit the long-term bulk temperature of the suppression pool water to 212°F when considering various sources of energy addition to the containment following a LOCA (Section 6.2.1.1).

2. Systems are designed so that no single failure results in loss of the safety function.

3. Systems are qualified for the environmental conditions imposed by a LOCA (Section 3.11).

4. Systems are designed to safety-related requirements including the capability to perform their functions following a SSE. Systems are designed as Category I and Safety Class 2.

5. Systems are designed to withstand dynamic loads resulting from suppression pool hydrodynamic conditions. These various hydrodynamic loads are fully described in the Unit 2 DAR for Hydrodynamic Loads (Appendix 6A).

6. Each active component of the systems is capable of being periodically inspected and tested during plant operation.

6.2.2.2 System Design

When the RHR system is in the containment spray mode, the RHR pumps draw water from the suppression pool, pass it through the RHR heat exchangers, and inject it into the spray headers. When the RHR system is in the pool cooling mode, the RHR pumps draw water from the suppression pool, pass it through the RHR heat exchangers, and return it to the suppression pool. Cooling water from the service water system is pumped through the heat exchanger tubes to exchange heat with the suppression pool water. Two cooling loops are provided, each mechanically and electrically separate from the other to achieve redundancy. The process diagram, including process data (Figure 5.4-14), covers all operating modes and conditions.

RHR heat exchanger details are listed in Table 6.2-6. Figure 6.2-39 shows the schematic of the containment spray system. Codes and standards used for design purposes are addressed in Section 5.4.7. Environmental qualification of the containment heat removal systems is discussed in Section 3.11.

The RHR pumps are automatically initiated in the LPCI mode from diverse signals as described in Section 7.3.1.1.1.4. The LOCA analysis assumed uninterrupted LPCI flow to the reactor core during the first 10 min of the accident. In accordance with emergency operating procedures (EOPs), a RHR pump may be realigned from the LPCI mode to the containment spray or suppression pool cooling mode of operation independent of the 10 min elapsed time criteria assumed in the LOCA analysis. The approach utilized in the EOPs is acceptable as these procedures contain adequate cautions to deter the Operator from premature flow diversion of RHR in the LPCI mode of operation. The guidance provided in EOPs assures that the peak clad temperature (PCT) identified by the Updated Safety Analysis Report (USAR) LOCA analyses is not increased due to Operator realignment of the RHR system during an accident. Therefore, EOPs may result in diversion of RHR flow from the core cooling function prior to 10 min into the accident; however, such action will not increase the PCT of the fuel. The Operator has 20 min after a LOCA to establish the containment spray cooling mode.

In the event that a single failure occurs and the procedure the Operator is following does not result in system initiation, the Operator places the other totally redundant loop into operation by following the same initiation procedure. The Operator would require about 5 min to change over from the LPCI mode of the RHR system to the containment spray cooling mode of the RHR system. The RHR heat exchangers, the service water pump, and the valves are remote manually operated. The RHR pool cooling suction and discharge arrangement is shown on Figure 6.2-87. All suction and discharge points are located near the pool wall with suction taken approximately 14 ft above the pool floor and discharge at 22 ft above the floor, vertically downward. Loop A suction and discharge points are separated circumferentially by 140 deg of arch length (80-ft chord length) and loop B suction and discharge are also approximately 140 deg apart. This particular arrangement will promote the natural circulation and mixing between the discharged water and the suppression pool water. This arrangement precludes suction and discharge interaction when only one RHR loop is operating in the pool cooling mode. It may be noted that one loop operating is the worst case for maximum suppression pool temperature.

Each RHR pump takes suction directly from the suppression pool through a strainer that prevents foreign objects and debris in the suppression pool from entering the ECCS and spray system flow path. The RHR suction strainers are approximately 8 ft below the minimum drawdown water level of the suppression pool. The ECCS suction strainers, including the RHR suction strainers, are all designed to prevent the passage of any particle larger than 0.104 in in diameter (Figure 6.2-39a). This particle size is smaller than any portion of the ECCS flow path, as shown in the following dimensions:

ECCS pump seal water orifices	1/8-in (0.125") diameter

Drywell spray nozzle orifices	1 1/64-in (1.0156") diameter
Suppression pool spray nozzle orifices	23/64-in (0.359") diameter
Minimum core spray internal diameter	5/8-in (0.625")
Minimum fuel channel spacing	4/10-in (0.400")

An evaluation of the effects of debris on ECCS performance was performed.[10] This evaluation demonstrated the characterization and potential effects of debris passing through strainers on ECCS spray nozzles, pump seal water orifices, RHR heat exchangers, and fuel channel spacing. The conclusions of this evaluation were as follows.

Rust, paint, and fiberglass debris that pass through suppression pool strainers are subjected to ECCS flow rates and turbulence that cause disintegration into particles smaller than strainer hole sizes.

Loose strands of fiber that may pass through strainers in large concentrations (fiber blitz) could plug pump seal cooling water orifices. The consequences of a plugged orifice are high seal temperature, and eventually poor seal life leading to seal leakage. The maximum seal leakage flow would be 23 gpm. For Unit 2, this leak rate would be insignificant with respect to total system flow.

Iron oxide passing through strainers could contribute to accelerated seal wear due to abrasion, but would be gradual and require months of wear to significantly degrade performance.

The evaluation concludes that adequate core cooling during a LOCA will not be compromised by the presence of rust, epoxy paint chips, sand, iron oxide sludge, and fibrous debris in the ECCS system or reactor core. The potential failure of ECCS pumps, inadequate cooling capacity from the RHR heat exchangers, plugging of the core spray header nozzles, plugging of the containment spray nozzles, corrosion or chemical reaction with other reactor materials, or fuel bundle flow blockage will be precluded.

To ensure that the system function is maintained, the strainers are oversized to minimize pressure drop and flow velocities if the strainer should remove suspended debris and become partially clogged. The inlet flow (perforated plate) area of the strainers is approximately 100 times the suction line flow area.

The RHR suction strainers are designed to withstand any loads during suppression pool transients, and are designed to withstand an accident service pressure of 45 psi. Design mechanical loads consisted of the following:

1. Direct Drag Loads (SRV bubble, LOCA bubble, condensation, oscillation, chugging, and seismic sloshing)

2. Deadweight

3. Operating Pressure

4. Head of Static Fluid

5. Earthquake plus SRV plus LOCA vibratory loading

6. Pipe loads

The equipment has been designed and qualified by analyses to be capable of continued operation under the simultaneous application of all normal, operating, and seismic loads caused by operating basis earthquake (OBE), SSE, SRV, discharge loads, LOCA loads, and SS loads caused by the water motion inside the suppression pool and the loads due to differential pressure (H) across the surfaces of the equipment. Equipment adequacy was demonstrated for each of the load combinations described in Table 3.9A-6.

Net Positive Suction Head

The NPSH details are provided in Section 6.3.2.2.

Insulation

Types of insulation used for piping and equipment within the drywell and suppression chamber are discussed in the following paragraphs.

For piping and equipment located within the drywell that require insulation to minimize heat loss, primarily metal-reflective-type insulation is used.

Metal-reflective insulation is an all-metal construction-type insulation that has a stainless steel inside and outside jacket which encapsulates multiple layers of stainless steel insulation material. Metal-reflective insulation is installed in sections with overlapping edges and quick-release latches with keepers.

Two other types of insulation are used inside the drywell for special and limited application: Min-k and Temp-Mat insulation. Min-k is a powder-type insulation used where space is limited and is encapsulated in stainless steel so as to be watertight. Temp-Mat is a borated, spun glass, blanket-type insulation used where it is necessary to lower the neutron flux (i.e., at the primary shield wall penetration), and is also encapsulated in stainless steel (see Table 6.2-64).

No antisweat insulations are used within the primary containment.

Fibrous insulation in the ZOI for a worst-case DBA LOCA is assumed to become debris. The worst-case DBA LOCA for generating fibrous insulation debris is a double-ended recirculation line break outside the bioshield. This break is assumed to destroy all Min-K insulation in the ZOI. ZOI calculations were performed in accordance with the BWROG Utility Resolution Guidance for ECCS Suction Strainer Blockage (BWROG 96125 and NEDO-32686).

All of this insulation was assumed to be 100-percent transportable to the suppression pool, with the exception of the outer covering. The outer covering is not transportable since air jet tests demonstrate the covering is not destroyed, and all Min-K insulation is above drywell gratings at el 261' and 249'. The fibrous insulation and other plant-specific debris were evaluated and utilized to conservatively size the ECCS strainers in accordance with the guidance of RG 1.82 Revision 2 and NEDO-32686 Revision 0.

6.2.2.3 Design Evaluation

<u>Pre-EPU Analysis</u>

The DBA for the containment spray system is failure of a steam line having a break area equal to 0.3 sq ft with suppression pool steam bypass of 0.05 sq ft $A/\sqrt{K}$. In the long term, this accident is similar to the DER of a recirculation suction or a main steam line. If such an event occurred, the short-term (prior to the actuation of the RHR heat exchangers) energy released from the RCS would be absorbed by the suppression pool, and the suppression pool temperature would increase. In the long term, fission product decay heat would continue to be absorbed by the pool. Unless this energy is removed from the suppression pool, a high containment pressure/temperature would result.

The containment cooling mode of the RHR system with the containment sprays is used to remove heat from the suppression pool and to limit the long-term, post-LOCA primary containment internal pressure and the suppression pool temperature to less than 45 psig and 212°F, respectively.

To evaluate the adequacy of the containment heat removal system, the following sequence of events is assumed to occur:

1. With the reactor initially at 102 percent of rated thermal power, a steam line failure wit°h a rupture area of 0.3 sq ft occurs. The bypass of steam is assumed, with an area of 0.05 sq ft $A/\sqrt{K}\ factor$ through the drywell floor.

2. LOOP occurs and one standby diesel generator fails to start and remains out of service during the entire transient. This is the most limiting single failure, failure of the Division II electrical system.

3. Only three ECCS pumps are functional following the postulated LOOP and one standby diesel generator failure. Thus, one HPCS pump, one LPCS pump, and one RHR pump are available for emergency core cooling and containment heat removal functions.

4. Within 30 min after the accident, the plant Operator actuates the RHR heat exchanger and the containment spray. This involves transferring the RHR pump from LPCI to the containment spray mode and starting another service water pump.

   Once the containment heat removal function is established, no further Operator action is required.

When calculating the long-term, post-LOCA primary containment pressure and suppression pool temperature, it is assumed that the service water temperature is at a maximum value of 82°F*. In addition, the RHR heat exchanger is assumed to be in a fully fouled condition during the transient (tubeside fouling, 0.001 hr-°F-sq ft/Btu and shellside fouling, 0.0005 hr-°F-sq ft/Btu). Both of these assumptions are conservative and maximize the containment pressure and the suppression pool temperature response.

<u>EPU Analysis</u>

Refer to Section 6.2.1.1.6 (Steam Bypass).

6.2.2.3.1 Containment Sprays

6.2.2.3.1.1 Design Bases

The containment spray system is capable of quickly reducing containment pressure during the post-accident period of a LOCA through condensation of steam in the drywell and through cooling of the noncondensable gases in the free volume above the suppression pool. The sprays also remove airborne fission products following a LOCA accompanied by significant fuel damage, in support of the AST methodology. The design of the containment spray system is in accordance with Category I and Safety Class 2 requirements.

6.2.2.3.1.2 System Design

The containment spray system consists of two subsystems, the drywell spray and the suppression chamber spray. The drywell spray consists of two independent loops and spray headers (Figure6.2-39). The suppression chamber spray consists of one spray header supplied from two otherwise independent loops. Since the water source for all containment sprays is the suppression pool, the system is a closed loop. Following a LOCA

accompanied by significant fuel damage, the spray water will be maintained at a pH ≥ 7.0 by injection of the SLCS sodium pentaborate solution into the reactor/containment system in support of the AST methodology. The spray water is cooled by the RHR heat exchangers. The calculated minimum flows for the drywell and suppression chamber sprays are shown in Table 6.2-52. Containment spray is an operational mode of the RHR system (Section 5.4.7).

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* As a result of an increase in maximum service water design temperature to 84°F, the containment response has been re-evaluated. The resulting containment temperatures and pressures remain bounded by the current analysis.

The containment spray isolation valves are electrically interlocked to allow actuation of the drywell spray only when: 1) there is a LOCA signal or a system-level LPCI manual initiation signal, and 2) there is a high drywell pressure signal present. A second electrical interlock prevents actuation of either the drywell or suppression chamber spray lines until the corresponding LPCI injection valve is shut.

The containment spray system is safety related and, in case of LOOP, supplied with a redundant onsite standby power source.

The system is designed to operate under the conditions indicated in Table 6.2-6.

The LOCA analysis assumed uninterrupted LPCI flow to the reactor core during the first 10 min of the accident. In accordance with EOPs, a RHR pump may be realigned from the LPCI mode to the containment spray or suppression pool cooling mode of operation independent of the 10 min elapsed time criteria assumed in the LOCA analysis. The approach utilized in the EOPs is acceptable as these procedures contain adequate cautions to deter the Operator from premature flow diversion of RHR in the LPCI mode of operation. The guidance provided in EOPs assures that the PCT identified by the USAR LOCA analyses is not increased due to Operator realignment of the RHR system during an accident. Therefore, EOPs may result in diversion of RHR flow from the core cooling function prior to 10 min into the accident; however, such action will not increase the PCT of the fuel.

Distribution of spray in the air space is made as complete and uniform as practical with minimal direct impingement on wall and component surfaces. The sizes, types, number, and location of the spray nozzles are suitable for delivering the required quantity of water in the proper spray pattern and particle size shown on Figure 6.2-91. The suppression chamber spray header and spray nozzle locations are shown on Figure 6.2-92. The lower and upper drywell spray header locations are shown on Figure 6.2-93.

The spray nozzle locations for the two drywell spray headers are shown on Figure 6.2-94.

The expected spray pattern of the spray nozzles is hollow cone.

Figures 6.2-40 through 6.2-42 show expected spray coverage several feet below the spray nozzles. Figure 6.2-43 shows the extent of the volume coverage by the sprays. Spray drops are expected to reach thermal equilibrium with the primary containment atmosphere. Table 6.2-51 shows the containment spray system parameters.

The spray system is designed to provide 94 percent of the system flow to the drywell spray header and 6 percent of the flow to the wetwell header with one RHR loop in the spray mode. Following a LOCA, the drywell and wetwell would be pressurized and the downcomer vents would be hot due to steam flow to the suppression pool. If the sprays are initiated in this condition, the drywell and wetwell pressures would drop, as indicated on Figure 6.2-28A, assuming 90-percent spray thermal effectiveness. Reduced wetwell spray effectiveness resulting from spray impingement on the hot downcomer pipes would tend to increase the wetwell pressure.

However, this pressure increase will be limited by the vacuum breakers which will open to permit any uncondensed steam in the wetwell to flow into the drywell and condense. For these reasons, the Unit 2 LOCA steam bypass analysis is not sensitive to reduced wetwell spray effectiveness.

6.2.2.3.1.3 Design Evaluation

Due to the redundancy and separation of the containment spray loops, containment spray is available to rapidly reduce containment pressure and reduce airborne fission products during the post-accident period. The long-term containment pressure response is shown on Figures 6.2-3 and 6.2-4 (recirculation pump suction line break) and Figures 6.2-16 and 6.2-17 (main steam line break) for the spray and no-spray cases. Even with minimum ECCS operation and no containment spray, the post-accident containment pressure remains significantly below the containment design value of 45 psig.

Airborne fission product removal in support of AST methodology is in accordance with the requirements of RG 1.183. Additional details are provided in Section 15.6.5.

6.2.2.3.2 NPSH Availability

The available NPSH for the RHR pumps is calculated based on the regulatory position of RG 1.1 with plant-specific debris mix. The details are provided in Section 6.3.2.2.

6.2.2.3.3 Heat Removal

Analysis of the containment heat removal capability is performed for large and intermediate breaks and is discussed in Section 6.2.1.1. The resultant primary containment pressure and temperature responses are described in Section 6.2.1.1.

Even with the very degraded conditions of heat exchanger and service water temperature previously outlined, the peak primary containment pressure does not exceed the containment design pressure of 45 psig, and the peak suppression pool water temperature does not exceed the design suppression pool water temperature of 212°F.

A system-level/qualitative-type plant failure modes and effects analysis (FMEA) of the RHR system is provided in Appendix 15A, Plant Nuclear Safety Operational Analysis (NSOA). Originally, the FMEA of the balance-of-plant (BOP) instrumentation and control components of the RHR system (suppression pool cooling mode and containment spray cooling mode) was contained in the Unit 2 FMEA document, which is historical. FMEAs for plant systems are now performed and controlled by the design process.

Figures and tables showing the calculated performance (pre-EPU) of the key variables as a function of time following the occurrence of a DBA, assuming minimum ESFs available, are summarized as follows:

1. Accident parameters used in the analysis (Table 6.2-52 for pre-EPU and Table 6.2-52a for EPU).

2. Primary containment pressure for the steam line break area of 0.3 sq ft (Figure 6.2-28a).

3. Suppression pool water temperature for the steam line break area of 0.3 sq ft (Figure 6.2-45).

4. Integrated energy content of the water containment and the suppression pool (Table 6.2-53).

5. Integrated energy absorbed by the passive heat sinks and removed by the RHR heat exchanger (Table 6.2-53).

6. Heat removal rate of the RHR heat exchanger for the steam line break area of 0.3 sq ft (Figure 6.2-46).

7. Suppression pool water temperature for DER of recirculation suction line (Figure 6.2-11).

The very conservative evaluation procedure previously described clearly demonstrates that the RHR system in the containment cooling mode can meet its design objective of safely terminating the post-accident primary containment temperature transient.

6.2.2.4 Tests and Inspections

Preoperational and operational testing and periodic inspection of containment heat removal system components are described in Sections 5.4.7.4, 6.3.2.7, 14.2, and the Technical Specifications. Logic is provided to prevent normal testing of one drywell spray isolation valve when the other valve in the same loop is open.

6.2.2.5 Instrumentation Requirements

The RHR containment spray cooling mode and the RHR suppression pool cooling mode of the RHR system are manually initiated from the main control room. Details of the instrumentation are provided in Sections 7.3.1.1.3 and 7.3.1.1.4, respectively.

6.2.3 Secondary Containment Functional Design

The secondary containment, consisting of the reactor building and auxiliary bay structures, completely surrounds the primary containment. The secondary containment is maintained at a negative pressure of at least 0.25 in W.G. with respect to the surrounding outside atmosphere. The secondary containment provides a means of controlling fission product leakage to the environment. This section discusses the reactor building design, the role of the reactor building ventilation system, and the standby gas treatment system (SGTS) which is used to depressurize and clean the reactor building atmosphere. The SGTS is discussed in detail in Section 6.5.1.

6.2.3.1 Design Bases

The reactor building structure completely encloses the reactor and the primary containment.

The reactor building structure, in conjunction with the SGTS and portions of the reactor building ventilation system, provides the means of controlling and minimizing leakage from the primary containment to the outside atmosphere during a LOCA and from the refueling facilities (including the spent fuel pool) during a postulated refueling accident.

The normal reactor building ventilation system (HVRS) is designed to automatically shut down and isolate and to automatically start the SGTS and safety-related unit coolers on receipt of any of the following signals that indicate either a LOCA or a refueling accident:

1. High drywell pressure.

2. Reactor vessel low water level.

3. High radiation level in exhaust ducts above or below the refueling floor.

The secondary containment pressure control function utilizes the HVRS (normal operation) and the SGTS (emergency operation) instrumentation and controls to maintain a negative pressure of at least 0.25 in W.G. with respect to the atmosphere. This ensures that while the systems are operating, any leakage is into the reactor building. All reactor building air is either exhausted through the exhaust air plenum, where it is constantly monitored, or discharged through the filtration units of the SGTS.

The reactor building isolation signals, isolation dampers for the HVRS, and the SGTS are all designed to Category I and Class 1E requirements. The design basis for the SGTS is given in Section 6.5.1.

The reactor building structure houses the refueling and reactor servicing equipment, the new and spent fuel storage facilities, and other reactor auxiliary or service equipment, including the RCIC system, reactor water cleanup (RWCU) demineralizer system, SLCS system, CRD system equipment, HPCS, and electrical equipment components. The reactor building auxiliary bays house the LPCS system, the RHR system heat exchangers and pumps, the reactor building closed loop cooling water (RBCLCW) system heat exchangers, and electrical equipment components. The reactor building structure protects the equipment from the SSE, design basis tornado (DBT) and tornado-generated missiles, and design basis wind. The reactor building structure is designed to meet the following design bases:

1. The reactor building is designed to meet Category I requirements.

2. The reactor building is designed and constructed in accordance with the structural design criteria presented in Section 3.8.

3. The reactor building design provides for low in-leakage and out-leakage during reactor operation.

4. The reactor building is designed to withstand applied wind pressures resulting from the design basis wind velocity (Section 3.3).

5. The reactor building is designed to withstand pipe whip loads plus jet impingement or jet reaction loads due to high-energy pipe breaks outside primary containment (Section 3.6).

6. The reactor building design allows for periodic inspections and functional tests of the penetrations.

7. The reactor building is designed to withstand tornado-generated missiles (Section 3.5).

8. The reactor building is designed for all probable combinations of the design basis wind, DBT velocities, and associated differences of pressure within the structure and atmospheric pressure outside the structure. The reactor building design incorporates the pressure loading from various conditions (i.e., events) that may be encountered during plant operation. These events include:

   a. Negative internal pressure of at least 0.25 in of water under which the reactor building is normally required to operate.

   b. Negative internal pressure of 0.6 in of water (0.022 psig) relative to the outside atmosphere, which exceeds the 0.25 in of water in Item a, to account for any uncertainty in pressure measurement and to account for any negative pressures actually developed by the SGTS, or due to outside ambient/secondary containment temperature extremes, or by unknown causes.

   c. Positive pressure of 6.92 in of water (0.25 psig) relative to the outside atmosphere, to account for: any positive pressure transient that the reactor building may experience following a postulated pipe break in the reactor building, any outward positive differential pressures created by wind loads, and any uncertainty in pressure measurements.

9. All entrances to the reactor building are through double door airlock systems. Table 6.2-62 provides a listing of these entrances and their locations.

   Use of these entrances is under administrative control to maintain a negative reactor building differential pressure, thus ensuring secondary containment integrity.

   All of these entrances are equipped to provide local position indication.

   All entrances leading into secondary containment from uncontrolled areas are equipped with security alarms that annunciate in an alarm station continuously occupied by Security personnel. The control room has continuous communications capability with the alarm station. Accordingly, if Control Room Operators need to ascertain status of secondary containment entrances, they may readily obtain it from Security personnel in the alarm station.

6.2.3.2 System Design

Refer to Figures 1.2-6 through 1.2-12 for general arrangement drawings of the reactor building showing plan and elevation views of the boundary of the structure. Also refer to Figures 3.8-1 and 3.8-2. Refer to Table 6.2-54 for the design and performance data for the secondary containment structure.

The reactor building design criteria are described in Section 3.8.4. Refer to Section 3.8.4 for identification of the codes, standards, and guides applied in the design of the reactor building structure.

6.2.3.2.1 Reactor Building Ventilation System

Normal ventilation for the reactor building is described in Section 9.4.2.

6.2.3.2.2 Postaccident Design Provisions

The major design provisions that prevent post-accident primary containment leakage from bypassing the SGTS (except for those lines identified as potential bypass leakage paths in Tables 6.2-55a, b, c, and d) are the secondary containment pressure control instrumentation of the SGTS, the reactor building ventilation isolation system, the isolation signals listed below, and the standby power system.

A portion of HVRS is required to operate during accident conditions. Part of the system is automatically shut down, the HVRS emergency recirculation dampers are aligned, and the SGTS starts in the event of any of the following isolation signals:

1. Reactor vessel low water level.

2. High drywell pressure.

3. High radiation level in exhaust ducts above or below the refueling floor.

The SGTS can also be started manually.

All ventilation system penetrations of the reactor building (except those of the SGTS) are fitted with two fail-closed, air-operated dampers in series. All dampers automatically close on any one of the aforementioned isolation signals.

Penetrations of the reactor building are designed with leakage characteristics that ensure that the leakage requirements of the entire building are not exceeded.

Railcar entrance to the reactor building railroad airlock is through an interlocking double door airlock system. The railroad airlock is completely within and along the northeast side of the reactor building at el 261 ft. One of the interlocked doors is

the exterior railcar door at the north end of the railroad airlock, and the other is the interior railcar door at the south end of the railroad airlock. A smaller door for personnel ingress and egress is incorporated into the design of the interior railcar door. All three doors must be closed before any one of them can be opened.

The reactor building pressure control function automatically maintains a subatmospheric pressure of at least 0.25 in W.G. by monitoring the differential pressure between the reactor building interior and the external atmospheric pressure. The differential pressure is monitored by a differential pressure transmitter. The signal that indicates the differential pressure also controls the position of the recirculation dampers in the HVRS supply fan units. In the event of reactor building isolation, the reactor building pressure control instrumentation regulates the reactor building pressure by controlling the SGTS recirculation flow.

The reactor building pressure control instrumentation is designed to eliminate fluctuations in reactor building pressure caused by such factors as wind gusts. Reactor building pressure is indicated and recorded and loss of negative pressure is alarmed in the main control room.

6.2.3.2.3 Bypass Leakage Paths

Table 6.2-56 presents a tabulation of all primary containment process piping penetrations including the potential reactor building bypass leakage paths. The potential bypass leakage paths are routed through the reactor building and terminate in the radwaste, standby gas treatment, turbine generator buildings, or yard. No guard pipes are used on penetrations and, therefore, guard pipes cannot constitute a bypass leakage path. All process lines that rely on a closed system within the primary containment as a leakage boundary terminate within the reactor building; therefore, these lines are not considered potential bypass leakage paths.

Bypass leakage is included in the radiological evaluation of design basis events. This is discussed in Section 15.6.5.5. Tables 6.2-55a, b, c, and d show the bypass leakage paths considered. They include four main steam lines, two main steam drain lines, one RWCU line, one feedwater line, four post-accident sampling lines (until such time as a modification eliminates the bypass leakage paths), six primary containment purge lines, four drywell floor and equipment vent and drain lines, and six nitrogen/instrumentation lines.

All leakage is conservatively assumed to be across isolation valve seats and to remain within the system piping until released to the environment. Any leakage escaping across outboard isolation valve stem packing would be released to the secondary containment or main steam tunnel. Any leakage into the secondary containment would be processed by the SGTS. Contaminants leaked

into the main steam tunnel will be transported to the environment more slowly due to the much larger cross-sectional area of the tunnel and the resulting slower average velocities.

No credit is taken for a reduction in bypass leakage due to water inboard of or trapped between isolation valves. The isolation valves are assumed to leak containment atmosphere instantaneously following the accident. No credit is taken for the time required to initially pressurize the volume between the isolation valves. Leakage transport time to the environment is based on 1/2 of the available horizontal and vertically downward flow piping located between the isolation valve and the environment.

Further conservatism is added to the analysis by the assumption that all isolation valves in these paths, except the MSIV and feedwater check valves, leak at a rate equal to the maximum permissible recommended acceptance level of 7.5 scf/day per inch of nominal valve diameter at functional pressure, based on the 1983 Edition, Summer 1983 Addenda, of ASME Section XI, IWV-3426.

The MSIVs are assumed to leak at 24 scfh, nearly ten times the valve design limit. Leakage across check valves, except the feedwater check valves, is assumed to be at twice the recommended rate of 7.5 scf/day per inch of nominal valve diameter, based on ASME Section XI, 1983 Edition, Summer 1983 Addenda, Subsection IWV-3426. Leakage across the feedwater valves is assumed to be 12 scfh.

Several process lines eliminate bypass leakage by the use of water seals. These are discussed below and include condensate makeup and drawoff (CNS), RCIC, and HPCS. Feedwater system (FWS) is also discussed below, but no credit for water seal is applied for that system. A typical loop seal is shown on Figure 6.2-88.

<u>CNS</u>

While not directly connected to the primary containment, the CNS system is used as the alternate fill source to the RHR, HPCS, LPCS, and RWCU systems. Each condensate fill connection to these systems is isolated by means of a normally closed globe valve. The main supply line into the secondary containment contains a check valve at the low point which, in case of a pipe break outside the containment, is sealed by a 70-ft leg of water. Although the CNS system is not of seismic design, any line break within the reactor building would provide a preferential flow path, for containment atmosphere leakage, into the reactor building atmosphere. Under this condition gaseous leakage would be collected by the SGTS and thus not be classified as bypass leakage.

<u>RCIC</u>

The RCIC path from the primary containment to the condensate storage building is protected from bypass leakage. When RCIC is

taking suction from the CST (2CNS-TK1A), the tank static head pressure maintains a 23-psig water seal at valve 2ICS*V28 and/or 2ICS*MOV136 (Figure 6.2-81). Also, the piping arrangement as shown on Figure 6.2-81 provides a loop seal with a high point at 2ICS*MOV136. Thus, any containment atmosphere leakage through this valve during the period that containment pressures exceed water seal pressure would be trapped at this high point. If a LOCA and a SSE take place simultaneously and a condensate line break occurs, 2ICS*MOV129 on the condensate tank line will shut automatically, creating an additional barrier to bypass leakage.

<u>HPCS</u>

The arrangement of the HPCS suction line from CST 2CNS-TK1B provides enough static head pressure to keep a 75-ft (32-psig) water seal at the line low point (valve 2CSH*MOV101) on Figure 6.2-83. Further, the piping arrangement, as shown on Figure 6.2-83, provides two intermediate loop seals with high points at valves 2CSH*MOV118 and 2CSH*V59, ensuring that any containment atmosphere leakage occurring during the 20 min that containment pressures exceed water seal pressure would be trapped between these high points. If a LOCA and a SSE take place simultaneously and a condensate line break occurs, 2CSH*MOV101 on the CST line will shut automatically, creating an additional barrier against bypass leakage.

<u>FWS</u>

For LOCAs not involving a feedwater line break, sufficient water exists in the vertical feedwater piping between the containment penetration and the reactor vessel to prevent bypass leakage for at least 30 days after the accident. See Figure 6.2-84.

For a break in feedwater piping inside containment, bypass leakage through this piping is included in the analysis of Section 15.6. However, as discussed below, a water seal restored after the break will effectively prevent escape of containment atmosphere to the environment after 10 min.

In considering a break in the feedwater piping within the primary containment, credit can be given to the piping arrangement which provides low stress levels along with pipe whip restraints. Consequently, it can be stated that the containment penetration is a break exclusion area. Assuming a break in the feedwater line at the end of the break exclusion region inside the primary containment (see Section 3.6A), sufficient water will remain in the line, even after flashing due to initial depressurization, to maintain a vertical water seal on the feedwater isolation valves (Figure 6.2-84). Water losses due to long-term containment pressure reduction and the associated water vaporization, and the back-leakage through the two isolation check valves for 30 days, will be replenished by reactor water leaking from the break. Within 10 min after the break, the ECCS injection water will reflood the reactor to above the level of the feedwater sparger.

At that point water would flood back into the feedwater piping and then into the intact containment penetration piping (Figure 6.2-84). This would more than make up for any losses due to leakage out the containment isolation valves. Thus a continuous water seal is provided to prevent any bypass leakage through the feedwater lines after the initial 10-min refilling period. Notwithstanding the above, bypass leakage through a ruptured feedwater line is included in the radiological analysis for the entire 30-day period to ensure conservative analysis results.

In addition to the two isolation check valves, each feedwater line has a remote-manual gate valve outboard of the isolation valves that may be shut subsequent to a LOCA anytime the Operators determine that feedwater flow is unnecessary or unavailable. The gate valve provides further back-leakage control. However, this valve is assumed to remain open for the purpose of evaluating bypass leakage.

6.2.3.2.4 Bypass Leakage Rates

Bypass leakage rates as a function of time after the postulated LOCA are predicted for each path by two methods, assuming isothermal flow and isentropic flow. Tables 6.2-55a and 55c list the bypass paths considered and their contributions to the total bypass leakage, assuming isothermal flow determined with the following equation:

$$\dot{m} = \frac{A}{\sqrt{K}} \left\{ \frac{g_c (P_u^2 - P_D^2)}{R T_u} \right\}^{1/2} \qquad (6.2\text{-}12)$$

Where:

- P_u = Upstream absolute pressure (post-LOCA pressure/temperature profile per Section 6.2.1)
- P_D = Downstream absolute pressure
- T_u = Upstream absolute temperature
- R = Gas constant
- K = Resistance coefficient
- A = Flow area
- m = Mass flow rate
- g_c = Conversion constant

To quantify the sensitivity of the bypass leakage analysis to the flow model assumption, the bypass calculation was repeated considering the leakage flow to be characterized as isentropic

flow through an orifice. Tables 6.2-55b and 55d summarize the isentropic flow results determined with the following equation:

$$\dot{m} = A\left\{2\,g_c\left(\frac{\gamma}{\gamma - 1}\right)\left(\frac{P^2{}_u}{RT_u}\right)\left(\frac{P_D}{P_u}\right)\frac{2}{\gamma}\left[1 - \left(\frac{P_D}{P_U}\right)\frac{\gamma_{-1}}{\gamma}\right]\right\}^{1/2} \qquad (6.2\text{-}13)$$

Where:

P_u = Upstream absolute pressure (post-LOCA pressure/temperature profile per Section 6.2.1)

P_D = Downstream absolute pressure

T_u = Upstream absolute temperature

R = Gas constant

γ = Specific heat ratio

g_c = Conversion constant

A = Orifice flow area (to be determined from the Technical Specification of allowable leak rate)

$\dot{m}$ = Mass flow rate

The isentropic flow is generally 5 to 35 percent higher than the isothermal flow depending upon the number of valves and time after LOCA.

In each case the fractional flow rate is evaluated using the following equation:

$$f = \frac{\dot{m}}{\rho V} \qquad (6.2\text{-}14)$$

Where:

$\dot{m}$ = Mass flow rate

ρ = Density of containment air and steam mixture (P/RT)

V = Containment volume

f = Fractional flow rate

The containment bypass leak rate for various paths is calculated based on two closed valves in series or one closed and one open valve depending upon the direct consequence of the postulated

failure of one emergency diesel generator or the failure of one MSIV to close.

Two single failure scenarios are included in the analysis to predict the bypass leakage rates. One scenario is the postulated failure of one emergency diesel generator, combined with LOOP which results in loss of one division of electrical power. This scenario results in all motor-operator containment isolation valves on that division failing as is (see Table 6.2-56), thereby reducing the restrictions to bypass leakage. The containment bypass leakage rates for various paths are calculated based on two closed valves in series, or one closed and one open valve depending upon the direct consequence of the postulated failure of one emergency diesel generator. The other single failure scenario is the failure of one MSIV to close. Both single failure scenarios are included in Tables 6.2-55a, b, c, and d.

6.2.3.2.5 Iodine Plateout Considerations

The radiological consequences arising from bypass leakage are provided in Section 15.6.5. The analyses include credits for elemental iodine deposition on the walls of the piping between the isolation valve and the release point. Details of the iodine deposition analysis can be found in Section 15.6.5.5.3.

6.2.3.2.6 Activity Transport Delay Considerations

The leakage of activity from the primary containment to the environment is through a portion of piping downstream of the outer isolation valve. Because of the very low leakage rates, there is a considerable transport delay time between the outer isolation valve and the release point. Therefore, the analyses include the credit of the delay time in the dose calculations. This is further explained in Section 15.6.5.5.3.

6.2.3.3 Design Evaluation

6.2.3.3.1 LOCA Temperature and Pressure Transient

During normal plant operation the reactor building and auxiliary bays are maintained at a negative pressure of at least 0.25 in W.G., relative to the outside atmosphere by the HVRS described in Section 9.4. In the event of a LOCA, the HVRS is isolated and the SGTS is initiated upon receipt of any of the three signals listed in Section 6.2.3.2.2. Details of the SGTS are provided in Section 6.5.1.

6.2.3.3.1.1 Summary and Conclusions

The following summarizes the drawdown analysis:

1. For a 2,670 acfm in-leakage at 0.25 in W.G. pressure differential between secondary containment and the environment, secondary containment temperature of

105°F, outside air temperature of -20°F, and a temperature differential between the secondary containment and the service water in the range of 0 to 10°F, the time required to reestablish -0.25 in W.G. in the secondary containment following a LOCA (i.e., drawdown time) is estimated to be less than 1 hr. The radiological dose evaluation uses 1-hr drawdown time for conservatism.

2. The SGTS has adequate capacity to maintain the secondary containment at at least 0.25 in W.G. vacuum for an extended period of time.

The difference between secondary containment in-leakage and SGTS exhaust flow is the net SGTS exhaust rate, and is referred to as differential flow.

During the winter months, the colder air temperature is expected to increase the air in-leakage resulting in a lower effective differential flow. An increased differential temperature is required to offset the effect of this lower differential flow. Figure 6.2-77 illustrates the relationship between external temperature and the required differential temperature for the in-leakage rate specified in Item 1.

6.2.3.3.1.2 Calculation Approach

The secondary containment drawdown time is dependent upon the following parameters:

1. Secondary containment environmental conditions (pressure, temperature, relative humidity) prior to LOCA.

2. Secondary containment in-leakage.

3. SGTS exhaust capacity and startup delay.

4. Building unit cooler and central cooling system capacity and startup delay.

5. Service water temperature.

6. Outside environmental conditions (pressure, temperature, relative humidity).

7. Heat generation and evaporation.

8. Building size.

Due to seasonal variation, secondary containment temperature, service water temperature, outside air temperature, and secondary containment and outside relative humidities are expected to vary

within certain limits. Therefore, the drawdown time may vary depending upon these parameters for a given plant condition. The drawdown time is influenced by the differential temperature between the secondary containment and the service water, as well as the differential flow between the secondary in-leakage rate and the SGTS exhaust. A smaller differential temperature results in low heat removal rate which increases the drawdown time. A smaller differential flow results in a lower effective SGTS exhaust rate which also increases the drawdown time. The secondary containment temperature has minor effect on the drawdown time for a constant differential temperature between the secondary containment air and the service water.

The outside air temperature change has little direct effect on the drawdown time. However, outside air temperature affects the in-leakage rate due to a change of air density relative to elevation. For example, in winter months a large temperature differential between the secondary containment and outside air creates a larger differential pressure at ground level than at the roof. Thus, the secondary containment in-leakage increases. Therefore, the outside air temperature indirectly increases the drawdown time.

Depending upon its value, the relative humidity inside the containment can have a significant impact on the drawdown time. Since, during normal operation, ventilation system causes about two air changes per hour in the building, the relative humidity inside the containment is essentially dependent on the outside relative humidity. The evaporation from the spent fuel pool does not have a significant impact on relative humidity.

The drawdown time estimation is done using THREED computer code. The secondary containment is divided into multiple subvolumes based on localized heat loads and cooling rates. The heat load estimation considers the operation of equipment prior to LOCA, initiation of equipment following LOCA, and residual heat of equipment which trips following LOCA. The heat removal of the cooling systems is based on the differential temperature between the building and the cooling water. The analysis also considers the worst-case single failure of 600 V Division II bus failure without LOOP and appropriate system startup delays.

6.2.3.3.1.3 Assumptions

Some of the assumptions applied to this analysis were:

1. A LOCA and the loss of the Division II 600-V bus are assumed to occur simultaneously. No LOOP is postulated to occur. This combination results in the highest post-LOCA heat load with the lowest cooling capacity within the secondary containment.

2. Adiabatic boundary conditions are assumed for the surface of the reactor building and auxiliary bay structure exposed to the outside air temperature.

3. The initial secondary containment environmental conditions are atmospheric pressure. ECCS rooms modeled in the calculation are conservatively assumed to have an initial air temperature equal to the service water temperature.

4. The initial relative humidity inside the secondary containment is 75 percent or less. The maximum 1-hourly dew point at the site during October through April is 63°F. During this time frame, the minimum reactor building temperature is 70°F or more. When the 63°F dew point air is heated to the containment temperature of 70°F, it results in a relative humidity of about 75 percent. During summer months, the maximum 1-hourly dew point is 73°F. The reactor building temperature during the summer, with outside temperature greater than 73°F, is expected to be approximately 85°F. The 73°F dew point air, when heated to 85°F, results in relative humidity of about 67 percent.

5. The maximum spent fuel heat load calculated for the plant design basis analysis is used.

6. Evaporation and convection effects from spent fuel pool are included in drawdown time estimation. No other source of latent heat is considered.

7. It is assumed that the Operator does not divert the SGTS fan capacity for decay heat cooling of the other loop for a duration of at least 5 hr after LOCA. The SGTS begins operation 90 sec after the LOCA occurs, at a nominal flow rate of 3,720 cfm.

8. The compressive effect of primary containment expansion is assumed to be insignificant.

9. The primary containment wall heat gain is assumed to increase linearly from the normal heat gain at time zero to the full emergency heat gain at 24 hr to account for the heat storage capacity of the 5.25-ft thick concrete wall.

10. The ECCS piping heat gain is based on the calculated pool temperature transient following a LOCA and a reactor building temperature of 70°F.

11. The sensible heat removal rate of each operating unit cooler is determined based on reactor building

temperature and relative humidity. The unit coolers are unavailable during the first 90 sec of the LOCA. A 40-percent degradation is assumed for all drawdown-related unit coolers, with the exception of 2HVR*413A and B which are assumed to be degraded 20 percent.

12. The nonessential lighting heat load is assumed to be available.

13. Secondary containment temperature range is 70°F to 105°°F.

14. One spent fuel pool heat exchanger and two component cooling heat exchangers are in operation for cooling the spent fuel pool.

6.2.3.3.2 High-Energy Line Break Evaluation

All high-energy lines within the reactor building and the analysis of line rupture for any of these lines are discussed in Sections 3.6.1 and 3.6.2.

6.2.3.4 Test and Inspection

Tests and inspections of the HVRS and the SGTS will be performed prior to initial fuel load and periodically thereafter in accordance with Technical Specification requirements.

To demonstrate that the SGTS will accomplish its design objectives under DBA conditions, the Technical Specifications require that the system must be tested every 24 months ±25 percent. Such testing will be done only when the outside air temperature is less than secondary containment air temperature to ensure compliance with the drawdown test acceptance criteria analysis assumptions.

Four curves are developed to determine the SGTS performance and the secondary containment leak-tightness (see Figures 6.2-95a, 6.2-95b, 6.2-95c and 6.2-95d).

Technical Specification surveillance 3.6.4.1.4, commonly referred to as the secondary containment drawdown test, is performed to ensure that the SGTS is capable of establishing an acceptable negative pressure within the time constraints imposed by the surveillance test drawdown analysis. This analysis does not model LOCA heat loads and unit cooler performance since a LOCA is not simulated during the test.

In addition, a SGTS subsystem is operated for 1 hr to ensure that measured building in-leakage is within the requirements stipulated in Technical Specification surveillance 3.6.4.1.5.

The performance of the SGTS is verified by measuring the time it takes to achieve a -0.25 in W.G. pressure from initiation of the test signal. This time is the drawdown time. The SGTS is run for 1 hr to ensure building stability and then the flow meter reading is recorded in order to determine if building in-leakage requirements are met. The purpose of the four figures (6.2-95a, 6.2-95b, 6.2-95c and 6.2-95d) is to allow comparison of the test results against the design basis without recalculating the LOCA drawdown analysis using test condition parameter values each time a test is run.

To use these figures, the following are required from plant testing:

1. Q - SGTS flow meter reading (scfm),

2. T_{out} - outside air temperature at the time of test (F),

3. dp - reactor building pressure at equilibrium from test ("WG),

4. T_{FLT} - charcoal filter outlet temperature at the time of test (F), and

5. T_{rb} - reactor building (secondary containment) temperature at the time of test (F).

The above 5 parameters are used with the 4 figures in the following manner:

1. From Figure 6.2-95a, look up F1 (correction factor) for the appropriate dp and $\Delta T = T_{rb} - T_{out}$,

2. From Figure 6.2-95b, look up F2 (correction factor) for the appropriate T_{out} and T_{FLT}

   Qactual = in-leakage rate (acfm of outside air) into building at -0.25 in W.G. at roof elevation during the test, or

   Qactual = F1 * F2 * Q,

3. From Figure 6.2-95c, look up acceptable drawdown time for the appropriate Qactual

   Acceptance Criteria: drawdown time test result is less than required drawdown time from Figure 6.2-95c,

4. From Figure 6.2-95d, look up F3 (correction factor) for the appropriate T_{out} and T_{rb}

Acceptance Criteria: F3 * Qactual calculated from above is less than 2,670 acfm at -20°F outside air, 105°F secondary containment temperature, and 1/4" WG dp at the roof.

The F1 correction factor converts the test condition flow rate to equivalent in-leakage rate at -0.25 in W.G. The F2 correction factor converts a raw meter reading to building in-leakage. The F2 correction factor also accounts for actual gas conditions at the flow orifices. The F3 correction factor converts the test in-leakage @ 1/4" W.G. to in-leakage @ 1/4" W.G. and the reference temperatures, 105°F secondary containment and -20°F outside air.

The drawdown time limit for each drawdown surveillance test must be determined based on the in-leakage measured during the test (corrected to account for test pressures and temperatures) to confirm the SGTS is capable of performing its intended function in accordance with the design basis.

Figure 6.2-95d provides a family of curves for various secondary containment temperatures (T_{rb}). To determine the in-leakage, the closest lower secondary containment temperature is used. The in-leakage value determined using F1, F2, and F3 from Figures 6.2-95a, 6.2-95b, and 6.2-95d must be less than or equal to 2,670 acfm.

The SGTS fans are rated for a design flow of 4,000 cfm. Under post-LOCA drawdown operation, a nominal flow of 3,720 cfm is drawn from secondary containment through the on-line filter train (see Figure 9.4-8L). Under post-accident conditions, decay heat cooling can be provided by manually activating the decay heat removal valves.

Secondary containment equipment hatches and access doors which are required to be closed during accident conditions are described in Technical Requirements Manual (TRM) Section 3.6.4.1.

Secondary containment isolation manual valves and blind flanges required to be closed during accident conditions are described in TRM Section 3.6.4.2.

Secondary containment automatic isolation valves, and their associated isolation times to satisfy Technical Specifications, are also described in TRM Section 3.6.4.2.

6.2.3.5 Instrumentation Requirements

A reactor building negative air pressure of at least 0.25 in W.G. is automatically maintained under normal operating conditions by the HVRS. Normally, modulating air dampers automatically

recirculate supply air to maintain negative pressure in the reactor building. During accident conditions (LOCA), isolation dampers in the air supply and air exhaust ducts will close automatically; the supply and exhaust air fans will stop, and an emergency recirculation air unit cooler and other safety-related unit coolers will start automatically to recirculate air through the reactor building. The SGTS will be automatically initiated and used to filter and maintain the required negative reactor building differential air pressure.

A detailed description of the reactor building heating, ventilating and air conditioning (HVAC) system is provided in Section 9.4.2. Refer to Section 7.3 for a description of instrumentation for the SGTS. Functional design details and logic are described in Section 7.3.1.1.5. Refer to Section 7.3.1.1.2 for a description of the instrumentation, functional design details, and logic for the primary containment and reactor vessel isolation control system (PCRVICS). Primary containment penetration lines and isolation signals applied to each are provided in Table 6.2-56.

6.2.4 Primary Containment Isolation System

6.2.4.1 Design Bases

6.2.4.1.1 Safety Design Bases

The safety design bases for the primary containment isolation system are:

1. Primary containment isolation valves provide for the necessary isolation of the primary containment in the event of accidents or other conditions to preclude radioactive releases from primary containment.

2. Capability for rapid closure or isolation of all pipes or ducts that penetrate the primary containment is provided so that leakage is maintained within permissible limits.

3. The design of isolation valving for lines penetrating the primary containment follows the requirements of General Design Criteria (GDC) 54 through 57 as noted in Table 6.2-56.

4. Isolation valving for instrument lines that penetrate the primary containment conforms to the requirements of RG 1.11.

5. Isolation valves, actuators, and controls are protected against loss of safety function from missiles.

6. The design of the primary containment isolation valves and associated piping and penetrations meets Category I requirements.

7. Primary containment isolation valves and associated piping and penetrations meet the requirements of the ASME Boiler and Pressure Vessel Code, Section III, Safety Classes 1 or 2, as applicable.

8. Primary containment isolation valve closure speeds limit radiological effects from exceeding the criteria established by 10CFR50.67.

The primary objective of the primary containment isolation system is to provide protection against release of radioactive materials to the environment from accidents occurring to the RCPB or lines connecting to the RCPB or penetrating primary containment. This is accomplished by automatic isolation valve closure of appropriate lines that penetrate the primary containment system. Actuation of the primary containment isolation system is automatically initiated at specific limits defined for reactor plant operation and, after the isolation function is initiated, it goes to completion.

The primary containment isolation system, in general, closes those fluid lines penetrating containment that support systems not required for emergency operation. Those fluid lines penetrating the primary containment which support ESF systems have remote manual isolation valves that can be closed from the control room, if required.

Redundancy and physical separation are required in the electrical and mechanical design to ensure that no single failure in the primary containment isolation system prevents the system from performing its intended functions.

The seismic and safety classification of equipment, systems, and penetration piping, up to and including the first isolation valve, is shown in Table 3.2-1. All piping is seismic Category I and either Safety Class 1 or 2. Conformance to SRP Section 3.6.2 requirements is discussed in Section 3.6A.2.1.5. Leakage detection capabilities for leakage external to the primary containment are discussed in Section 5.2.5. Actuation of the primary containment isolation system is initiated by various signals as listed in Table 6.2-56.

Primary containment isolation valves are designed to minimize leakage from shaft and/or bonnet seals. Additionally, periodic inspection, testing, and maintenance procedures under normal operating conditions are intended to minimize the potential for leakage under off-normal conditions. Radiological consequences of potential leakage from ESF systems are addressed in Section 15.6.5.5.3 and Table 15.6-13.

Criteria for the design of the PCRVICS are listed in Section 7.3.1.1.2. The bases for assigning certain signals for primary containment isolation are also listed and explained in Section 7.3.1.1.2.

Instrument lines that penetrate the primary containment conform to RG 1.11 and GDC 55 and 56.

6.2.4.2 System Design

The general criteria governing the design of the primary containment isolation system are provided in Sections 3.1.2 and 6.2.4.1. Table 6.2-56 summarizes the primary containment penetrations and contains information as to:

1. Status open or closed under normal operating conditions and accident situations.

2. Primary and secondary modes of actuation for the isolation valves.

3. Parameters sensed to initiate isolation valve closure.

4. Closure time for principal isolation valves to secure primary containment isolation.

5. Applicable general design criteria.

P&IDs which show the primary containment penetrations along with their associated piping, branch connections, piping safety class, and pressure boundary piping for each system are provided in each specific system description section.

Protection is provided for isolation valves, actuators, and controls against damage from missiles. All potential sources of missiles are evaluated. Where possible hazards exist, protection is afforded by separation, missile shields, or location. See Section 3.5 for a discussion of evaluation techniques.

Isolation valves are designed to be operable under adverse environmental conditions (Section 3.11) such as maximum differential pressures, extreme seismic occurrences, high temperature, and high humidity. Overpressurization protection is provided on penetrations susceptible to overpressure of an isolated segment due to post-accident primary containment temperature.

Redundancy and physical separation are provided in the electrical and/or mechanical design to ensure that no single failure in the primary containment isolation system prevents the system from performing its intended functions. Where a penetration is part of a redundant train in an ESF system, isolation valves for that train receive power from a single electrical division. This is

necessary so that single failure of an electrical division cannot disable both trains of the ESF system.

The MSIVs are globe valves designed to fail closed on loss of power.

The MSIVs shall be Type C tested in accordance with 10CFR50 Appendix J. The test medium shall be air/nitrogen and the test pressure shall be 40.00 psig.

The design and operation of the MSIVs is described in Section 5.4.5.

It should be noted that all motor-operated isolation valves remain in the as-is position upon failure of valve power. On the other hand, all air-operated valves (AOV) (not applicable to air-testable check valves) close on loss of air.

The swing-check valves located on the drywell-to-wetwell vacuum breaker lines cycle open and closed based on the difference in atmospheric pressure between the drywell and wetwell. A pneumatic source is not required in order for these valves to perform their safety function; however, a pneumatic supply is available for test purposes.

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

Outside isolation valves are located as close as practical to the primary containment. Except as listed below, outside isolation valves are within 10 ft of the containment wall.

Valve Number	Penetration Number	Pipe Length From Outside Containment
2RHS*MOV1B	Z-5B	20'-9"
2RHS*MOV33A	Z-7A	18'-3"
2CSH*MOV111	Z-13	50'-0"
2CSH*MOV105	Z-13	45'-6"
2DFR*MOV139	Z-43	20'-10"
2CPS*SOV119	Z-59	14'-6"
2MSS*MOV208	Z-1A-1D	36'-0"
2RHS*MOV30A	Z-6B	10'-6"
2RHS*MOV30B	Z-6A	19'-3"
2RCS*V59A	Z-38A	33'-0"
2RCS*V59B	Z-38B	31'-0"
2CMS*SOV26C	Z-61B	15'-0"
2CMS*SOV35A	Z-61C	18'-3"
2ICS*MOV148	Z-90	23'-10"
2ICS*MOV164	Z-90	29'-11"
2WCS*MOV200	Z-4A	57'-8"
2WCS*MOV200	Z-4B	65'-8"

2RHS*V192	Z-90	26'-6"
2CPS*AOV111	Z-51	11'-10"

For the above outboard isolation valves, locations have been established as close as practical to the containment while also satisfying pipe supporting and flexibility requirements, clearing other equipment in the area, and providing access for maintenance, testing, and in-service inspection (ISI). As discussed in Sections 3.5, 3.6, and Appendix 3C, the lines between the containment and the isolation valves are analyzed to ensure that their integrity is maintained against the effects of missiles, pipe whip, and jet impingement loads.

The penetrations shown on Table 6.2-63 involve relief valve discharge headers which combine inputs from several sources into one pipe penetrating to the primary containment. In this manner, they reduce the amount of piping and number of containment penetrations needed to satisfy system process requirements. In Table 6.2-63, piping lengths from outside the containment are separated into two lengths; first, from the containment isolation valve to the common piping header to the containment penetration. In all cases, relief and safety valves which serve as outside containment isolation valves have been located as close as practical to the containment, considering available piping arrangements and the requirement of ASME Code Section III, Subsection NC-7100, that relieving devices be located as close as practical to the major source of overpressure.

The 12-in lines passing through containment penetrations Z-88A and Z-88B originate from RHS*SV34A and B and RHS*SV62A and B. The layout of the 12-in lines is dictated by the required location of SV34 and SV62, which are positioned on the inlet piping to the RHR heat exchangers. These 12-in headers are protected from the effects of rapid depressurization, caused by condensation of steam vented by SV34 and SV62, by vacuum breakers 2RHS*RVV35A and B and 2RHS*RVV36A and B. These 12-in headers vent the 1-in RHR heat exchanger vent lines (RHS*MOV26A, B and 27A, B), the 3/4-in relief valve discharge vacuum breaker lines (RHS*V19, V20, V117, and V118), and the 1-in RHR heat exchanger relief valve discharge lines (RHS*RV56A, B). The connection of these lines to the 12-in headers does not significantly increase the containment volume beyond that represented by the actual 12-in headers themselves, while it does significantly reduce the number of containment penetrations and the amount of piping required for the several flow paths returning to the suppression pool. Thus, this arrangement locates outside isolation valves as close as practical to the primary containment. As discussed above, all containment penetration piping is analyzed to ensure integrity for its entire length against the effects of missiles, pipe whip, and jet impingement loads.

Whenever relief or safety valves are used as containment isolation valves, their set pressure is at least 1.5 times the design pressure of the primary containment. The one exception to

this is valve 2CSL*RV123, with a set pressure of 46 psig. For this valve, only the discharge side is part of the primary containment boundary. Thus, containment pressure during an accident is applied to the discharge side of the valve.

6.2.4.3 Design Evaluation

6.2.4.3.1 Introduction

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

Mechanical components, such as isolation valve arrangements, are redundant to provide backup in the event of accident conditions. The arrangements with appropriate instrumentation are described in Table 6.2-56. The isolation valves have redundancy in mode initiation. Generally, the primary mode is automatic and the secondary mode is remote manual. A program of testing (Section 6.2.4.4) is maintained to ensure valve operability and leak-tightness.

The design specifications require each isolation valve to be operable under the most severe operating conditions that it might experience. Each isolation valve is afforded protection, by separation and/or adequate barriers, from the consequences of potential missiles.

Electrical redundancy is provided in isolation valve arrangements; this eliminates dependency on one power source to attain isolation. Electrical cables for isolation valves in the same line have been routed separately. Cable selection is based on the specific environment (Section 3.11) to which they may be subjected, such as high radiation, high temperature, and high humidity.

Administrative control provisions ensure that the position of all nonpowered isolation valves is maintained and known. The position for all power-operated valves is indicated in the main control room. Discussion of instrumentation and controls for the isolation valves is included in Chapter 7.

Containment isolation considerations in the event of a Station Blackout (SBO) are addressed in Section 8.3.1.5.

6.2.4.3.2 Evaluation Against General Design Criteria

<u>Evaluation Against Criterion 55</u>

Criterion 55 requires that lines which are part of the RCPB and penetrate the primary containment must have two isolation valves, one inside the primary containment and one outside, unless it can be demonstrated that the primary containment isolation provisions for a specific class of lines are acceptable on some other basis.

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

Influent Lines Influent lines that penetrate the primary containment and connect directly to the RCPB are equipped with at least two isolation valves, one inside the drywell and the other as close to the external side of the primary containment as practical, or by one isolation valve and a closed system outside primary containment. Protection of the environment is provided by these isolation boundaries, as discussed below. Although only one valve and a closed system outside containment are credited for ECCS and RCIC system influent lines and the RHR return lines, these lines still include a second isolation valve. To reduce radiation exposure to personnel, only one of the two valves in each penetration is local leak rate tested per 10CFR50 Appendix J and credited for containment leakage control. Penetrations that do not constitute a potential primary containment atmospheric pathway during and following a DBA do not require Appendix J Type C testing. Pressure Isolation Valve Leakage testing of the outside containment isolation valve in water filled systems may be substituted for Appendix J Type C testing to ensure the penetration will remain filled with water during and following a DBA and therefore does not constitute a potential primary containment atmospheric pathway.

Table 6.2-56 contains those influent lines that compose the RCPB and penetrate the primary containment. Although a word-for-word comparison with Criterion 55 is not practical in some cases, adequate isolation provisions are demonstrated on a well-defined basis.

1. Feedwater Lines The feedwater lines are part of the RCPB as they penetrate the drywell to connect with the RPV. The isolation valve inside the drywell is a Y-pattern check valve, located as close as practical to the primary containment wall. Outside the primary containment is a testable Y-pattern check valve located as close as practical to the primary containment wall. This valve fails closed upon the loss or reversal of fluid flow. Away from the primary containment is a motor-operated gate valve. Should a break occur in the

feedwater line, the check valves prevent significant loss of reactor coolant inventory and offer immediate isolation. However, in case a LOCA occurs without a seismic event, the design allows the condensate and feedwater pumps to supply feedwater to the vessel. For this reason, the outermost gate valve does not automatically isolate upon signal from the protection system. The gate valve does meet the same environmental and seismic qualifications as the outboard isolation valve. The valve can be remotely closed from the control room to provide long-term leakage protection upon Operator judgment that feedwater makeup is unavailable or unnecessary. No credit is taken for feedwater flow in assessing core and containment response to a LOCA.

An analysis was conducted to evaluate the ability of feedwater check valves 2FWS*V23 and 2FWS*F012 to withstand rapid closure following a postulated pipe break outside the containment. Since long-term leakage protection is provided by MOV21, the acceptance criterion is that gross leak rates do not occur because of disk rupture, serious fracture of the seat/disk interface, or misalignment of the disk with respect to the seat from this faulted event.

Following the guidelines of Appendix F of the ASME III Code, inelastic systems and component analyses were conducted using the nonlinear transient option of the ANSYS (Appendix 3A.25) program. Seismic and dead loads were not considered because of their insignificant magnitude compared to impact. The nonlinear stress/strain relationship was approximated by a bilinear curve adjusted for temperature and strain rate effects.

Stresses in the rock shaft, tail link, seat, and disk were below ASME III Class 1 allowables for faulted conditions. It is concluded that both valves will remain intact and that any leakage will be within the makeup capability of the HPCS or the RCIC system, following rupture of the feedwater piping outside the containment.

2. <u>HPCS Line</u> The HPCS line penetrates the drywell to inject directly into the RPV. Isolation is provided by a manual testable check valve located inside the drywell, and a remote manually-actuated gate valve located as close as practical to the exterior wall of the primary containment. Long-term leakage control is maintained by the combination of one of the two valves or Pressure Isolation Valve Leakage testing of the outside containment isolation valve ensuring that the penetration will remain filled with water during and

following a DBA and therefore does not constitute a potential primary containment atmospheric pathway and the closed system barrier. If a LOCA occurred, the outside gate valve would receive an automatic signal to open.

3. <u>LPCI/RHS Lines</u> The LPCI injection lines contain remote manually-operated gate valves and manual testable check valves (2RHS*V16A/B/C). Long-term leakage control is maintained by the combination of one of the two valves and the closed system outside containment. Both types of valves are normally closed with the gate valves receiving an automatic signal to open at the appropriate time to assure that acceptable fuel design limits are not exceeded in the event of a LOCA. The check valves are located as close as practical to the RPV. The normally closed check valves protect against primary containment overpressurization in the event of pipe rupture between the check valve and primary containment wall by preventing high-energy reactor water from entering the primary containment. Once the system is in operation, the low energy of the influent fluid (220°F maximum) excludes any possibility of containment overpressurization should a break occur.

   <u>RHR Shutdown Cooling Return Lines and Associated Bypass Line</u> These penetrations contain an automatic globe valve outside containment and a check valve and an automatic valve (bypass) in parallel inside containment. Both types of valves are normally closed with the automatic valves receiving a primary containment isolation signal. Satisfaction of the isolation criteria is accomplished by use of either the outside automatic valve or the inside valves or Pressure Isolation Valve Leakage testing of the outside containment isolation valve ensuring that the penetration will remain filled with water during and following a DBA and therefore does not constitute a potential primary containment atmospheric pathway, and a closed system outside containment.

3a. <u>LPCS Line</u> The LPCS injection line contains a remote manually-operated gate valve and a check valve. Long-term leakage control is maintained by the combination of one of the two valves or Pressure Isolation Valve Leakage testing of the outside containment isolation valve ensuring that the penetration will remain filled with water during and following a DBA and therefore does not constitute a potential primary containment atmospheric pathway and the closed system outside containment. Both types of valves are normally closed with the gate valve receiving an automatic signal to open at the appropriate time to assure that acceptable fuel design

limits are not exceeded in the event of a LOCA. The check valve is located as close as practical to the RPV. The normally closed check valve protects against primary containment overpressurization in the event of pipe rupture between the check valve and primary containment wall by preventing high-energy reactor water from entering the primary containment. Once the system is in operation, the low energy of the influent fluid (220°F maximum) excludes any possibility of containment overpressurization should a break occur.

4. CRD Lines The CRD system insert and withdraw lines penetrate the drywell; however, these lines are not part of the RCPB since they do not directly communicate with the reactor coolant. The classification of these lines is Quality Group B, and they are, therefore, designed in accordance with ASME Section III, Safety Class 2. The basis to which the CRD insert and withdraw lines are designed is commensurate with the safety importance of maintaining pressure integrity of these lines. See Note 17 of Table 6.2-56 for further discussion.

5. RCIC Line The head spray line penetrates the drywell and discharges directly into the RPV. The check valve inside the drywell is normally closed and has position indication lights in the main control room to verify its position. The check valve is located as close as practical to the RPV. Two types of valves, a check valve and a remote manual block valve, are located outside the containment. The check valve assures immediate isolation of the primary containment in the event of a line break. Additional influent lines are connected to this penetration outside containment, each of which has either a normally closed valve or a lock closed manual valve. Long-term leakage control is maintained by either the inboard check valve or one of the outboard valves in each parallel line (not including the outboard check valve), in combination with a closed system outside containment.

6. SLCS Lines The SLCS line penetrates the drywell and connects to the RPV. In addition to a simple check valve inside the drywell, a check valve and explosive actuated valves are located outside the drywell. Since the SLCS line is a normally closed, nonflowing line, the possibility of rupture of this line is extremely remote. The explosive actuated valves function as third isolation valves. These valves provide an absolute seal for long-term leakage control and prevent leakage of sodium pentaborate into the RPV during normal reactor operation.

7. <u>RWCU System</u> The RWCU pumps, heat exchangers, and filter demineralizers are located outside the drywell. The return line from the filter demineralizers connects to the feedwater line outside the primary containment between the block valve and the outboard primary containment feedwater check valve. Isolation of this line is provided by the feedwater system check valves inside and outside the primary containment. A motor-operated globe valve is provided in the RWCU return line as a third isolation valve. The valve can be remotely closed from the control room to provide long-term leakage protection.

   Should a break occur in the RWCU return line, the feedwater system isolation check valves would prevent significant loss of inventory and offer immediate isolation, while the RWCU return line isolation valve would provide long-term leakage control.

8. <u>Recirculation Pump Seal Water Supply Line</u> The recirculation pump seal water line extends from the recirculation pump through the drywell and connects to the CRD supply line outside the primary containment. The seal water line forms a part of the RCPB. The recirculation pump seal water line is 3/4 in in diameter and is Class B from the recirculation pump through the second check valve (located outside and as close as practical to the primary containment). From this valve to the CRD connection the line is Class D. In a postulated failure of this line, the flow rate through the broken line has been calculated to be substantially less than that permitted for a broken instrument line.

Continued recirculation pump seal purge is required whenever reactor coolant temperature is above 200°F and the pump is not isolated. Three check valves in series, two outside the primary containment, are used to provide containment isolation while permitting seal purge, if available. This design will prevent seal damage during containment isolation events. Therefore, automatic isolation valves are not desirable.

The seal purge lines are continually pressurized (and therefore leak tested) above reactor pressure. Thus, any leakage from these lines would be detected either through the floor drain system monitors or by routine surveillance by plant Operators. In addition, the seal purge pressure is continually monitored by pressure transmitters with control room indication. Therefore, the integrity of these lines is continuously verified.

<u>Effluent Lines</u> Effluent lines that form part of the RCPB and penetrate primary containment are equipped with at least two isolation valves, one inside the drywell and the other outside,

located as close to the primary containment as practical, or by one isolation valve and a closed system outside primary containment. Protection of the environment is provided by these isolation boundaries as discussed below. Although only one valve and a closed system outside containment are credited for the RHR shutdown cooling supply line, it still includes a second isolation valve. To reduce radiation exposure to personnel, only one of the two isolation valves is local leak rate tested per 10CFR50 Appendix J and credited for containment leakage control. Penetrations that do not constitute a potential primary containment atmospheric pathway during and following a DBA do not require Appendix J Type C testing. Pressure Isolation Valve Leakage testing of the outside containment isolation valve in water filled systems may be substituted for Appendix J Type C testing to ensure the penetration will remain filled with water during and following a DBA and therefore does not constitute a potential primary containment atmospheric pathway. Table 6.2-56 also contains those effluent lines that compose the RCPB and penetrate the primary containment. Although word-for-word comparison with Criterion 55 is not practical in some cases, adequate isolation provisions are demonstrated on a well-defined basis.

1. <u>Main Steam, Main Steam Drain Lines, and RCIC/RHR Steam Supply Lines</u> The main steam lines extend from the RPV to the main turbine and condenser system, and penetrate the primary containment. The main steam drain lines also penetrate the containment. The RHR steam supply line/RCIC turbine steam line connect to the main steam line inside the drywell and penetrate the primary containment. Isolation is provided by automatically-actuated block valves inside the primary containment for the RHR steam supply line/RCIC turbine steam line.

2. <u>Recirculation System Sample Lines</u> A sample line from the recirculation system penetrates the drywell. The sample line is 3/4 in in diameter and is designed to ASME Section III, Safety Class 2. Two solenoid-operated valves (SOVs) which fail closed are provided, one inside and one outside located as close to the primary containment as practical.

3. <u>RHR Shutdown Cooling Supply Line</u> The RHR shutdown cooling supply line contains an automatic gate valve inside and outside containment. Both valves are normally closed during power operation and receive a primary containment isolation signal. Satisfaction of isolation criteria is accomplished by use of either of the automatic valves or Pressure Isolation Valve Leakage testing of the outside containment isolation valve ensuring that the penetration will remain filled with water during and following a DBA and therefore does not constitute a potential primary containment

atmospheric pathway and a closed system outside containment.

As discussed above, the following penetrations contain two isolation valves, but rely on a single isolation valve and a closed system outside the containment for long-term leakage control.

Penetration No.	Description
Z-10A	RHS Shutdown Return Loop A to Reactor Recirc Loop A / RHS Shutdown Cooling Return Line Inboard Valve Bypass Line
Z-10B	RHS Shutdown Return Loop B to Reactor Recirc Loop B / RHS Shutdown Cooling Return Line Inboard Valve Bypass Line
Z-9A	RHS/LPCI Loop A to RPV
Z-9B	RHS/LPCI Loop B to RPV
Z-9C	RHS/LPCI Loop C to RPV
Z-11	RHS Shutdown Supply from Reactor Recirc
Z-14	CSH to RPV
Z-16	CSL to RPV
Z-22	ICS to RPV / RHR Reactor Head Spray

System piping and valves outside the containment which are a part of the closed system boundary are of Category I, Safety Class 2 design; do not communicate with the outside atmosphere; are protected from the effects of HELB and missiles; are protected from overpressure from thermal expansion due to contained fluid; and have been evaluated to ensure they can withstand design temperature and pressure ratings at least equal to that for the containment. In addition, due to the stringent design requirements placed on these lines because of their importance in the ECCS and isolation cooling functions, they are fully designed to withstand LOCA transients and environment. Finally, the penetrations are tested in accordance with 10CFR50 Appendix J, and the LLRT results of the tested valves are included in the Type B & C leakage summary or Pressure Isolation Valve Leakage testing ensuring that the penetration will remain filled with water during and following a DBA and therefore does not constitute a potential primary containment atmospheric pathway. Branch lines from the closed systems are valved closed and procedurally controlled. In case of an active failure of the isolation valve, primary containment leakage would be contained within the closed system boundary.

Conclusion on Criterion 55

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

In addition to meeting the isolation requirements stated in Criterion 55, the pressure-retaining components that compose the RCPB are designed to minimize the probability or consequences of an accidental pipe rupture. The quality requirements for these components ensure that they are designed, fabricated, and tested to the highest quality standards of all reactor plant components. These components are designed in accordance with the ASME Boiler and Pressure Vessel Code, Section III, Safety Class 1, except for piping and valves smaller than 1 in which are Safety Class 2.

It is concluded that the design of the piping system that composes the RCPB and penetrates containment satisfies Criterion 55. For further discussion, see the following:

1. Quality Group Classification, Table 3.2-1.

2. Containment and Reactor Vessel Isolation Control System, Section 7.3.1.1.2.

Evaluation Against Criterion 56

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

Table 6.2-56 includes those lines that penetrate the primary containment and connect to the drywell and suppression chamber. Although a word-for-word comparison with Criterion 56 is not practical in some cases, adequate isolation provisions are demonstrated on a well-defined basis.

Influent Lines to Suppression Pool

1. LPCS, HPCS, and RHR Test Lines The LPCS, HPCS, and RHR test lines have test isolation capabilities commensurate with the importance to safety of isolating these lines. The HPCS line has a normally closed remote manually-actuated motor-operated valve (MOV) located outside the primary containment. Primary containment isolation requirements are met on the basis that the test lines are normally closed, low-pressure

lines constructed to the same quality standards as the primary containment. Furthermore, the consequences of a break in these lines result in no significant safety consideration. The LPCS and RHR test return line has a normally open remote manually-actuated isolation valve.

The test return lines are also used for suppression chamber return flow during other modes of operation. In this manner the number of penetrations are reduced, minimizing the potential pathways for radioactive material release. Typically, pump minimum flow bypass lines join the respective test return lines upstream of the test return isolation valve. The minimum flow bypass lines are isolated by MOVs.

2. <u>RCIC Turbine Exhaust and RCIC Pump Minimum Flow Bypass</u> These lines, which penetrate the containment and discharge to the suppression pool, are equipped with motor-operated, remote manually-actuated isolation valves located as close to the primary containment as possible. In addition, there is a simple check valve upstream of the isolation valve that provides positive actuation for immediate isolation in the event of a break upstream of this valve. The gate valve in the RCIC turbine exhaust is normally open and interlocked to prevent opening of the inlet steam valve to the turbine while the turbine exhaust valve is not in a full open position. The RCIC pump minimum flow bypass line is isolated by a normally closed globe valve with a check valve installed upstream.

3. <u>RHR Heat Exchanger Vent Lines</u> The RHR heat exchanger vent lines discharge to the RHR safety relief valve discharge line (SRVDL) which in turn discharges to the suppression pool. The vent lines are isolated from the relief valve discharge line by two remote-controlled motor-operated globe valves. These isolation valves are normally closed and valve positions are indicated in the main control room to provide the Operator with the indication of valve status.

4. <u>RHR Relief Valve Discharge Lines</u> The RHR relief valve discharge to the suppression pool has no valve other than the relief valve. This relief valve will not be opened during normal operation and therefore can be considered as normally closed and adequate under the same criteria as the suppression chamber spray line.

5. <u>RCIC Turbine Exhaust Vacuum Breaker System Lines</u> This line has two automatic MOVs and two check valves. This line runs between the suppression pool air space and the RCIC turbine exhaust line downstream of the exhaust line check valve. Positive isolation is automatic via a combination of low reactor steam pressure and high

drywell pressure. The automatic isolation signals energize a 70-sec time delay to slow automatic closure of MOVs 2ICS*MOV148 and 2ICS*MOV164. The time delay is provided to ensure sufficient time to fully condense the exhaust steam in the RCIC turbine exhaust line. The vacuum breaker complex is placed outside the primary containment where there is a more desirable environment. In addition, the valves are readily accessible for maintenance and testing.

<u>Effluent Lines from Suppression Chamber</u> The RHR, RCIC, LPCS, and HPCS suction lines contain motor-operated, remote manually-actuated valves that ensure isolation of these lines in the event of a break. These valves also provide long-term leakage control. In addition, the suction piping from the suppression chamber is considered an extension of containment since it must be available for long-term usage following a design basis LOCA and, as such, is designed to the same quality standards as the primary containment. Thus, the need for isolation is conditional. The ECCS discharge line fill system (ECCS water leg pumps) takes suction from the respective ECCS pump effluent line from the suppression pool downstream of the isolation valve. The ECCS discharge line fill system suction line has a manual valve for operational purposes. This system is isolated from the primary containment by the respective ECCS pump suction valve from the suppression pool, as listed in Table 6.2-56.

Each ECCS pump room is provided with leak detection capabilities, as discussed in Section 9.3.3.3. If leakage from a seal or gasket is detected in one of the pump rooms during normal plant conditions, the remotely-operated valve installed in the pump suction line would be closed, thereby isolating the leaking component from the suppression pool water. Between the isolation valve and the penetration there is a manual valve in the low-pressure core spray (CSL) and RHR systems. These valves are welded on the suppression pool side and gasketed on the pump side. The suction line from the penetration to the ECCS pump rooms is inside flood troughs which carry all leakage to the pump rooms. Leakage from this valve gasket would be minimal. However, if the leakage became significant, the water would be detected in the ECCS pump room by the leak detection system (LDS). After detection, the manual valve could be closed and the gasket repaired prior to any significant loss of suppression pool water.

The only potential path for leakage of suppression pool water into the ECCS pump rooms is through the pump suction lines, as these are the only lines that penetrate the containment at an elevation below the suppression pool water level.

The need to size the ECCS pump room so that the volume of suppression pool water needed to fill the ECCS pump room would not reduce the suppression pool level below the minimum drawdown

line is not required. This is due to the leak detection, isolation, and repair capabilities incorporated into the design. The potential reduction in suppression pool water inventory before detection, isolation, and repair of a leaking gasket in the pump room would be insignificant. Suppression pool makeup during normal plant conditions is from the CST.

The elevations of the ECCS pump suction centerlines and the suppression pool minimum drawdown level are 195'-0" and 197'-8", respectively.

Influent and Effluent Lines from Drywell and Suppression Chamber Free Volume

1. Primary Containment Purge Lines The drywell and suppression chamber purge lines have isolation capabilities commensurate with the importance to safety of isolating these lines. The drywell purge lines and one of the suppression chamber purge lines have two normally closed/fail closed valves - one located inside (nitrogen operated) and one located outside (air operated) the primary containment. The other suppression chamber purge line has two normally closed/fail closed valves, both located outside primary containment (see Section 3.1.2.56). The inboard end of each 12-in and 14-in valve located inside the primary containment is provided with a QA Category I debris screen to prevent entrainment of foreign matter in the valve seat. The isolation valves are interlocked to preclude opening of the valves while a primary containment isolation signal exists (Table 6.2-56). The radiological consequences of a LOCA occurring during containment purge system (CPS) operation (isolation valves wide open) with the SGTS in the pressure control mode are discussed in Section 15.6.5.

2. Primary Containment Atmosphere Monitoring System Sampling Lines The primary containment atmosphere monitoring system consists of radiation and hydrogen/oxygen monitoring lines. Each line, suction, and discharge penetrates the primary containment and continuously monitors the radiation level and hydrogen/oxygen concentration during normal operation. These lines are equipped with two solenoid-operated isolation valves, one inside the primary containment and the other outside, located as close as possible to the primary containment. The hydrogen/oxygen monitoring lines are also used to continuously monitor the primary containment air during the post-LOCA period. Each isolation valve receives isolation signals. The isolation valves for hydrogen/oxygen monitoring lines are provided with individual keylock switches to override the isolation signal and initiate system operation during the post-LOCA period.

3. Suppression Chamber Spray Lines The suppression chamber spray lines penetrate the primary containment to remove energy by condensing steam and cooling noncondensable gases in the suppression chamber. Each line is equipped with a normally closed MOV located outside and as close as possible to the primary containment. This normally closed valve receives an automatic isolation signal. Primary containment isolation requirements are met on the basis that the spray header injection lines are normally closed, and the lines are constructed to the same quality standards as the primary containment.

4. Drywell-to-Wetwell (DW-WW) Vacuum Relief Lines The four DW-WW vacuum relief penetrations are each equipped with two positive closing swing check valves. The air operator on the swing check valve is used only for testing. Swing check valve operation is initiated by the pressure difference between the drywell and wetwell.

5. Reactor Building Closed Loop Cooling Water The RBCLCW lines penetrate the drywell to provide cooling water to the recirculation pumps and motors and drywell unit coolers. Each line that penetrates the primary containment has an automatic isolation valve inside and outside the containment. These valves also have remote manual operation capability from the control room.

6. Containment Spray Lines The containment (drywell) spray lines penetrate the primary containment to remove energy by condensing steam in the drywell. Each line is equipped with two normally closed MOVs located outside and as close as possible to the primary containment. Primary containment isolation requirements are satisfied by use of either valve

   outside containment and a closed system outside containment.

As discussed above, the following penetrations rely on a single isolation valve and a closed system outside the containment.

Note that the containment (drywell) spray lines (penetrations Z-8A and Z-8B) have two normally closed remote manual valves. Only one of the two valves in each penetration is local leak rate tested per 10CFR50 Appendix J and credited for containment leakage control to reduce radiation exposure to personnel.

Penetration No.	Description
Z-5A, B, and C	RHS pump suction from suppression pool

Z-6A and B	RHS test return line to suppression pool
Z-7A and B	RHS containment spray to suppression pool
Z-8A and B	RHS containment spray to drywell
Z-12	HPCS pump suction from suppression pool
Z-13	HPCS test return and minimum flow bypass to suppression pool
Z-15	LPCS pump suction from suppression pool

Penetration No.	Description
Z-17	RCIC suction from the suppression pool
Z-18	RCIC minimum flow to suppression pool
Z-19	RCIC turbine exhaust
Z-73	RHS relief valve discharge to suppression pool
Z-88A and B	RHS safety valve discharge to suppression pool
Z-98A and B	RHS relief valve discharge to suppression pool

System piping and valves outside the containment, which are a part of the closed system boundary, are of Category I, Safety Class 2 design; are protected from missiles; and have design temperature and pressure ratings at least equal to that for the containment. Branch lines from the closed system are valved closed and procedurally controlled. In case of an active failure of the isolation valve, primary containment leakage would be contained within the closed system boundary. System reliability is enhanced by the simplicity of design obtained with the use of a single isolation valve outside the containment by reducing the number of possible active failures. Those lines which connect to the suppression pool do not have an isolation valve located inside the primary containment, as this would necessitate placement of the valve underwater. In effect, this would introduce an unreliable design element into a system requiring high reliability.

Conclusion on Criterion 56

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

isolation requirements, the pressure-retaining components of these systems are designed to the same quality standards as the primary containment.

Evaluation Against Criterion 57

Lines penetrating the primary containment for which neither Criterion 55 nor Criterion 56 governs compose the closed system isolation valve group. Influent and effluent lines of this group are isolated by automatic or remote manual isolation valves located as close as possible to the primary containment boundary. The reactor recirculation system hydraulic control lines to the flow control valve contain an isolation valve located outside the drywell that closes automatically upon receipt of its isolation signal. The hydraulic lines and their isolation valves are discussed in Note 26 of Table 6.2-56.

Evaluation Against Regulatory Guide 1.11

Instrument lines that penetrate the primary containment from the RCPB are equipped with a restricting orifice or equivalent reduced orifice valve located inside the drywell and an excess flow check valve located outside and as close as practical to the primary containment, in accordance with RG 1.11. Those instrument lines that do not connect to the RCPB are equipped with isolation valves whose status is indicated in the control room in accordance with RG 1.11.

Traversing in-core probe (TIP) subsystem guide tubes are classified as instrument lines in accordance with RG 1.11. The justification for this classification is provided in GE NEDC-22253, BWROG Evaluation of Containment Isolation Concerns, October 1982. The TIP guide tubes have an isolation valve that closes remote manually after the TIP cable and fission chamber have been retracted. An additional or backup isolation shear valve is included in series with this isolation valve. Both valves are located outside the drywell. The TIP system and isolation provisions are discussed in Note 19 of Table 6.2-56.

6.2.4.3.3 Failure Modes and Effects Analysis

A single failure can be defined as a failure of some component in any safety system that results in a loss or degradation of the system's capability to perform its safety function. Active components are defined in RG 1.48 as components that must perform a mechanical motion during the course of accomplishing a system safety function. Appendix A to 10CFR50 requires that electrical systems be designed against passive single failures as well as active single failures. Chapter 3 describes the implementation of these standards as well as GDC 17, 21, 35, 41, 44, 54, 55, and 56.

In single-failure analysis of electrical systems, no distinction is made between mechanically active or passive components; all

fluid system electrically-operated components such as valves are considered electrically active whether or not mechanical action is required. Electrical systems as well as mechanical systems are designed to meet the single-failure criterion for both mechanically active and passive fluid system components regardless of whether the component is required to perform a safety action or not. Even though a component such as an electrically-operated valve is not designed to receive a signal to change state (open or closed) in a safety scheme, it is assumed as a single failure that the system component changes state or fails. Electrically-operated valves include valves such as solenoid valves and solenoid-actuated AOVs or valves that are directly operated by an electrical motor. In addition, all electrically-operated valves that are automatically actuated can also be manually actuated from the main control room. Therefore, a single failure in any electrical system is analyzed regardless of whether the loss of a safety function is caused by a component failing to perform a requisite mechanical motion or a component performing an unnecessary mechanical motion.

6.2.4.3.4 Operator Actions

A trip of an isolation control system channel is annunciated in the main control room so that the Operator is immediately informed. All motor-operated and air-operated isolation valves have open-close status lights. The following general information is presented to the Operator by the isolation system:

1. Annunciation of each process variable that has reached a trip point.

2. Computer readout of trips on main steam line tunnel temperature or main steam line excess flow.

3. Control power failure annunciation for each channel.

4. Annunciation of steam leaks in each of the systems monitored (main steam, RWCU, and RHR).

The leakage detection system detects possible leakage from lines inside/outside containment and provides the Operator in the main control room with information required to isolate fluid systems equipped with remote manual isolation valves. Parameters used to detect leakage are high radiation, high area temperature, high sump level, and RPV level and pressure as discussed in Sections 5.2.5.1.3, 7.6.1.3, and 12.3.4.1. System parameters such as flow, pressure, and temperature are indicated and/or alarmed in the main control room. These enable the Operator to detect degraded system performance attributable to system leakage and take appropriate action to isolate systems that are potential leakage paths.

This information will enable the Operator to decide if he needs to operate a remote manual valve in the event of a LOCA.

6.2.4.4 Tests and Inspections

The primary containment isolation system is scheduled to undergo periodic testing during reactor operation. The functional capabilities of power-operated isolation valves are tested manually from the main control room. By observing position indicators and changes in the affected system operation, the closing ability of a particular isolation valve is demonstrated. A discussion of testing and inspection pertaining to primary containment isolation valves is provided in Section 6.2.6, TRM Section 3.6.1, and the Technical Specifications Bases. Table 6.2-56 lists all primary containment isolation valves.

Instruments will be periodically tested and inspected. Test and/or calibration points will be supplied with each instrument.

Excess flow check valves will be periodically tested by opening a test drain valve downstream of the excess flow check valve and verifying proper operation. Preoperational testing is discussed in Section 14.2.12.

Containment isolation valve leak testing is discussed in Section 6.2.6.

Leakage testing of the closed ESF systems outside containment is performed, as required, in accordance with the inservice testing (IST) program, the inservice pressure testing (ISPT) program, 10CFR50.55a, and the applicable Code, as discussed in Sections 6.6 and 3.9A.6. Any airborne radioactivity resulting from leakage from these ESF systems following a LOCA is processed through the SGTS prior to discharge to the environment. The offsite doses from this source are small. This contribution has been accounted for in the radiological assessment of the site. Section 15.6.5.5.3 and Table 15.6-13 discuss the methodology and assumptions used in determining the radiological consequences of leakage from the primary containment and from ESF systems following a LOCA.

Additional requirements for the PCRVICS will be provided in accordance with Section 1-10, Task II.E.4.2.

6.2.5 Combustible Gas Control in Containment

To assure that the primary containment integrity is not endangered by generation of combustible gases following a postulated LOCA, the primary containment (drywell and suppression chamber) will be inerted with nitrogen (Section 1.10). Systems for controlling the relative concentrations of oxygen and hydrogen are provided within the plant. The system includes subsystems for mixing the primary containment atmosphere, monitoring oxygen and hydrogen concentrations, and reducing oxygen and hydrogen concentrations without relying on primary containment purging to the environment. The primary containment

purge system is available to aid in post-accident cleanup operations.

6.2.5.1 Design Bases

The following design bases were used for the combustible gas control system (CGCS) design:

1. The CGCS is designed to limit the oxygen or hydrogen concentration to 5 volume percent within the primary containment following a LOCA.

2. A recombiner mixes the drywell atmosphere and the suppression chamber atmosphere. Prior to initiation of the recombiner, the drywell and the suppression chamber will be mixed uniformly due to natural convection and molecular diffusion. Mixing will be further promoted by operation of the containment sprays. The Operator actuates the containment sprays within 30 min after the LOCA. The criteria for the operation of containment sprays is specified in Section 6.2.1.1.

3. The recombiners will be started manually by the Operator prior to either the hydrogen concentration reaching 4 volume percent or oxygen concentration reaching 4.5 volume percent in the drywell or the suppression chamber. An alarm is provided to alert the Operator to these conditions.

4. Two identical Category I recombiners are provided to limit oxygen or hydrogen concentration. Operation of either recombiner will limit combustible gas concentration to a safe value.

5. The components of the CGCS are protected from missiles and pipe whip to assure proper operation under accident conditions as required for safety-related systems. The recombiners and monitors are located outside the primary containment.

6. The components of the CGCS are designed as Category I and Safety Class 2.

7. All components that are subjected to primary containment atmosphere will be capable of withstanding the humidity, temperature, pressure, and radiation conditions in the containment following a LOCA.

8. The CGCS can be inspected or tested during normal plant conditions.

9. The recombiners are located in the reactor building.

10. The primary containment purge system is provided to aid in the post-accident cleanup operation. The primary containment atmosphere can be purged through the SGTS to the outside environment. Nitrogen makeup will also be available during the purging operation.

6.2.5.2 System Design

The CGCS provides effective control over hydrogen and oxygen generated following a LOCA. The system consists of the following features:

1. Atmospheric mixing is achieved by natural processes. Mixing could be enhanced by operation of the containment sprays, which are used to depressurize the primary containment. The criteria for the manual operation of the containment sprays are provided in Section 6.2.1.1.

2. One of the two 100-percent capacity hydrogen recombiners is manually initiated after a LOCA to preclude the oxygen and hydrogen concentration from exceeding 5 volume percent.

3. The primary containment nitrogen inerting system establishes and maintains an oxygen-deficient atmosphere ($\leq$4 volume percent) in the primary containment during normal operation.

4. The redundant hydrogen and oxygen analyzer system measures hydrogen and oxygen in the drywell and suppression chamber during LOCA conditions. Hydrogen and oxygen concentrations are displayed in the main control room. Piping and instrumentation for this system are shown on Figures 6.2-71a and 6.2-71b. Safety-related display instrumentation for containment monitoring is listed in Table 7.5-1.

5. All controls for operating the hydrogen recombiner system (HCS) are located in the main control room. Controls for the containment monitoring system (CMS) are in the control room and north (south) auxiliary bays.

6. A tabulation of the design and performance data for each system component is listed in Table 6.2-57.

7. Environmental qualification information for safety-related equipment is given in Section 3.11.

8. Electrical requirements for equipment associated with this system are in accordance with IEEE Class 1E standard.

The CGCS is considered an extension of the primary containment in post-LOCA conditions and consequently will be included within the boundary of the Type A test (Section 6.2.6). The DBA HCS system meets the criteria of Standard Review Plan (SRP) 6.2.3 for closed loop systems as follows:

1. Containment atmosphere does not directly communicate with the environment following a LOCA.

2. Designed in accordance with Quality Group B standards.

3. Meets Category I design requirements.

4. Is designed to primary containment pressure and temperature design conditions as applicable.

5. Is designed for protection against pipe whip, missiles, and jet forces.

6. Is tested for leakage.

6.2.5.2.1 Atmospheric Mixing

The function of post-LOCA mixing in the drywell and suppression chamber is performed by the primary containment spray system, recombiner system, and natural processes. At approximately 30 min following the postulated accident, the redundant containment spray systems in the drywell and suppression chamber can be initiated to depressurize the containment. The turbulence induced by the spray ensures a well-mixed primary containment atmosphere. In addition to the spray system, the blowdown of steam and water through the broken pipe creates a large degree of turbulence and promotes mixing of the entrained hydrogen and oxygen with the primary containment atmosphere. The natural convection currents arising from temperature differences between the atmosphere and primary containment walls, and diffusion will promote a well-mixed atmosphere and prevent hydrogen and oxygen stratification. With the above mixing capabilities, there is minimal potential for a nonuniform hydrogen and oxygen concentration within the primary containment.

The atmosphere between the drywell and suppression chamber will be mixed during the depressurization phase of the LOCA. When activated the recombiner units will also serve to effect mixing between these two compartments. The recombiner will draw air from the drywell and/or the suppression chamber and discharge to the suppression chamber. This will in turn cause the atmosphere from the suppression chamber to circulate into the drywell via the vacuum breaker lines.

There are three interior subcompartments where gases may not achieve thorough mixing with the bulk of the primary containment atmosphere. The drywell head area, used for reactor vessel

refueling purposes, is one such subcompartment. There are no sources of oxygen in this area; therefore, local oxygen concentration is not expected to be greater than the bulk oxygen concentration. The other two subcompartments are the CRD area in the drywell and the volume enclosed by the pedestal wall in the suppression chamber. Due to the large open area between these two subcompartments and the bulk atmosphere, significant concentration gradients are unlikely.

6.2.5.2.2 Hydrogen Recombiner System

As per NRC revision of 10CFR50.44, which eliminated the design basis LOCA hydrogen release, along with NRC-approved License Amendment 124, which removed the hydrogen recombiner requirements from Technical Specifications, the hydrogen recombiners are no longer required for DBA LOCA hydrogen concentration control. The hydrogen recombiners do, however, remain necessary to ensure adequate atmospheric mixing in the primary containment during and following DBA LOCA.

The long-term control of hydrogen and oxygen is achieved by means of two identical 150-scfm thermal hydrogen recombiners, located in the reactor building and controlled from the main control room. The recombiner system removes gas from the drywell or suppression chamber, recombines the hydrogen with oxygen, and returns the gas mixture along with the condensate to the suppression chamber. Flow from the suppression chamber atmosphere to the drywell through the vacuum breakers prevents the suppression chamber pressure from exceeding the drywell pressure by more than 0.25 psi.

Operation of any one recombiner will provide effective control over combustible gases within primary containment. Figure 6.2-72a and b shows the piping and instrumentation diagram (P&ID) of the recombiner system. The manufacturer of the hydrogen recombiner is the Atomics International Division, Energy Systems Group of Rockwell International.

The recombiner unit is skid mounted and is an integral package. All pressure-containing equipment including piping between components is considered an extension of the containment and, therefore, is designed to ASME Section III, Safety Class 2 requirements. The skid and the equipment mounted on it are designed to meet Category I requirements.

The recombiner unit consists of a blower, electric heater, reaction chamber, and water spray cooler. The reaction chamber is capable of processing 150 scfm of gas containing up to either 2 1/2 volume percent of oxygen and unlimited excess hydrogen or 5 volume percent of hydrogen with excess oxygen. Under these conditions, recombination efficiency is virtually 100 percent. The recombiner is not designed to operate when hydrogen concentration exceeds 5 volume percent with excess oxygen.

The recombination process takes place within the recombiner as a result of high temperature. The resulting water vapor is then cooled along with other gases and returned to the suppression chamber.

The recombiner unit, which requires a 1 1/2-hr warmup period, is initiated manually from the control room prior to either hydrogen concentration reaching 4 volume percent or oxygen concentration reaching 4.5 volume percent. This occurs for the hydrogen concentration, approximately 2 and 1/2 days after the DBA. Once placed in operation, the system continues to operate until it is manually shut down when an adequate margin below the hydrogen or oxygen concentration design limit is reached.

The operation of the system can be tested from the control room. The test consists of energizing the blower and heaters and observing system operation to see if components are performing properly. Flow and pressure measurement devices are periodically calibrated.

Cooling water required for operation of the system after a LOCA is taken from the SWP system after passing through 100-mesh size, Y-type strainers. Demineralized water from the makeup water system is used for functional testing of the recombiner units. The cooling water is used to cool the water vapor and the residual gases leaving the recombiner prior to returning them to the primary containment.

6.2.5.2.3 Primary Containment Nitrogen Inerting System

Oxygen control within primary containment during normal plant operation is achieved by means of the nitrogen inerting system. During normal plant operation, oxygen concentration is maintained at or below 4 volume percent using this system.

The system is designed to supply nitrogen to the primary containment for initial inerting and for makeup during normal operation.

6.2.5.2.4 Primary Containment Purge

Primary containment purge capability is provided in accordance with RG 1.7 and as an aid in cleanup following an accident. This function is fulfilled by the combined operation of the CPS and the SGTS.

During normal plant operation, the CPS also functions, in conjunction with the nitrogen inerting system (GSN) and the SGTS, to maintain primary containment pressure at about 0.75 psig and oxygen concentration at or below 4 percent by volume. This is accomplished by injecting the required quantity of nitrogen into the primary containment through the CPS and/or extracting the required volume of gas through the CPS exhaust. The exhaust flow is routed through piping to the SGTS, where it passes through the

SGTS filters and a radiation monitor before being released from the plant stack to the environment. All CPS primary containment isolation valves are automatically closed after 15 sec when a high radiation level is detected in the exhaust flow. This time delay of 15 sec prevents automatic closure of CPS primary containment isolation valves due to spurious power transients.

The CPS P&ID (Figure 9.4-8) shows the piping and instrumentation used in this mode of operation.

6.2.5.2.5 Hydrogen and Oxygen Monitoring System

The hydrogen and oxygen concentrations are monitored by the two fully-independent hydrogen/oxygen analyzer trains. The redundant system design ensures that the volumes are sampled in the event of the functional failure of one of the analyzer trains. The location of oxygen and hydrogen sample points within the drywell and the suppression chamber are provided in Table 6.2-59A. These sampling points are distributed vertically and radially throughout the drywell and suppression chamber. Structures and equipment within the region of the sampling points are listed in Table 6.2-59B. Following an accident, this system will be activated manually to monitor combustible gas concentrations. After activation, the system completes an initial warmup cycle and then continuously monitors the primary containment hydrogen and oxygen concentration by drawing samples from five different areas: three from the drywell and two from the suppression chamber. For the drywell samples, the sample source and the return points are selected by a sequencing timer that controls the opening and closing of SOVs in the sample and return lines. All the samples drawn are returned to their origins. When the sequencing timer is utilized, each sample valve in the drywell remains open for 20 min.

For the suppression chamber, the sample source and the return point are selected manually. The sample is drawn from both areas simultaneously, combined, and then analyzed by the hydrogen/oxygen analyzer and returned to the suppression chamber.

The required accuracy of the hydrogen and oxygen analyzer is $\pm$5 percent of full scale, and the 90-percent response time to sample the concentration is less than 60 sec. Both hydrogen and oxygen analyzers are supplied with two range readouts. The hydrogen and oxygen analyzers have 0-10, 0-25, and 0-30 percent ranges.

6.2.5.3 Design Evaluation

The Unit 2 primary containment atmosphere will be inerted with nitrogen during normal operation of the plant. Oxygen concentration within the primary containment will be maintained at or below 4 volume percent (based on noncondensable gases). Following an accident, oxygen and hydrogen concentrations will be controlled by means of the recombiner system.

In evaluating the CGCS design, it is necessary to consider:

1. Oxygen and hydrogen sources in a post-accident environment.

2. Distribution of oxygen and hydrogen in the drywell and the suppression chamber.

3. Primary containment pressure and temperature during the containment cooldown phase of the accident.

6.2.5.3.1 Sources of Oxygen and Hydrogen

Short-Term Hydrogen Generation

In the period immediately after the LOCA, hydrogen is generated by both radiolysis and metal-water reaction. However, the short-term contribution from radiolysis is insignificant compared to that of the metal-water reaction. The metal-water reaction of steam with the zirconium fuel cladding which produces hydrogen is:

$$Zr + 2H_2O \rightarrow ZrO_2 + 2H_2 \qquad (6.2\text{-}14)$$

Based on LOCA calculational procedures and analysis of ECCS performance in conformance with 10CFR50.46 and Appendix K of 10CFR50, the extent of the chemical reaction is estimated to be <0.1 percent of the fuel cladding material. The metal-water reaction generated hydrogen, based on a core-wide penetration of 0.00023 in, results in a metal-water reaction greater than five times the calculated value of 0.1 percent (0.5 percent). Therefore, 0.00023 in cladding is assumed to react with water to produce hydrogen in accordance with RG 1.7. The duration of this reaction is assumed to be 120 sec with a constant reaction rate. The resulting hydrogen is assumed to be distributed between the drywell and the wetwell. Figures 6.2-72D and 6.2-72E show hydrogen generation rates and integrated values as a function of time following the accident.

Short-Term Oxygen Source

The only source of air addition to primary containment is the operation of relief valves inside the primary containment. These relief valves are part of the breathing and service air systems, and are normally isolated during reactor operation. Due to high temperature following a LOCA inside primary containment, a portion of these systems (inside primary containment) becomes pressurized and relieves pressure by expelling about 126 std cu ft of air into the primary containment.

The primary containment does not have any provision for storing portable air packs for breathing. The operating procedures would have appropriate controls for the use of portable air packs.

The ADS valves are nitrogen operated; therefore, operation of these valves will not result in addition of oxygen in the primary containment. The short-term oxygen source has not been considered in the oxygen concentration evaluation, as it is very small.

Long-Term Hydrogen/Oxygen Generation

Hydrogen and oxygen are produced by decomposition of water due to absorption of the fission product decay energy immediately after a LOCA. Generation of hydrogen and oxygen due to radiolysis of core cooling water is an important factor in determining the long-term gas mixture composition within the primary containment. A fission product distribution model, as outlined in RG 1.7, is used to calculate hydrogen/oxygen generation rates. The in-core radiolysis (due to core gammas) contributes hydrogen and oxygen to the drywell, and radiolysis due to fission products contributes hydrogen and oxygen directly to the suppression chamber and the drywell atmospheres. The division of hydrogen and oxygen between the suppression chamber and the drywell depends upon the fraction of water holdup on the drywell floor and water in the reactor vessel.

Hydrogen can also be formed by corrosion of metals and decomposition of organic materials in the primary containment. The significant portion of this source is from the corrosion of zinc, which is included in the analysis. The temperature-dependent hydrogen production rate is based on NUREG/CR-2812[4]. The temperature-dependent hydrogen generation rate for demineralized water is shown in Table 6.2-59C. The galvanized steel and zinc primer surface areas exposed to sprays are shown in Table 6.2-59D. The surface area used in the analysis is approximately 15 percent higher than tabulated values. The corrosion of aluminum in demineralized water is very small. The Griess and Creek[5] test data suggest the hydrogen production rate to be between 4.76×10^{-5} to 3.23×10^{-3} std cu ft of H_2 per sq ft/hr. Assuming that the corrosion in the Griess and Creek test is mainly due to 285°F and 212°F water temperature, the average rate is 4×10^{-4} std cu ft of H_2 per sq ft/hr. Considering the aluminum surface area directly exposed to the spray environment and the above H_2 generation rate, a total of 250 scf of hydrogen would be evolved within 20 days following a LOCA. Since this source is small compared to other sources of hydrogen, aluminum corrosion and associated hydrogen production is ignored in the analysis.

Figures 6.2-72D and 6.2-72E show hydrogen generation rates and integrated values. The quantity of hydrogen initially contained within the RCS is negligible; hence, it is neglected.

6.2.5.3.2 Accident Description

Following the postulated recirculation suction line DER, the metal-water reaction begins in the core region and produces hydrogen immediately. The reaction is assumed to last 2 min. The radiolysis of coolant in the core region, water on the drywell floor, and suppression pool water begins immediately. The hydrogen and oxygen thus generated evolve to the drywell and suppression chamber atmospheres.

The combustible gases in the drywell and the suppression chamber would approach the flammability limit, if uncontrolled, after 6.7 days. Prior to this, pressure and temperature within the primary containment are shown by analysis (Section 6.2.1) to have dropped to a level that will permit operation of the recombiner. The recombiner system is manually activated when oxygen or hydrogen concentration reaches the limits described in Section 6.2.5.1. The recombiner system takes suction from the primary containment atmosphere, recombines the hydrogen and oxygen to form water vapor, and returns the exhaust to the suppression chamber. This results in a small pressure buildup in the suppression chamber that causes the opening of the vacuum breaker valves between the drywell and suppression chamber. As a result, the flow of the gas mixture from the suppression chamber to the drywell is established. This arrangement of recombiner suction and discharge promotes mixing of the two volumes in the primary containment.

6.2.5.3.3 Analysis

Based on the preceding hydrogen and oxygen generation sources and the accident description, the oxygen and hydrogen concentration in the drywell and suppression chamber is obtained as a function of time. However, the analysis conservatively assumes that the recombiner system is manually activated prior to either the hydrogen concentration reaching 4 volume percent or the oxygen concentration reaching 4.5 volume percent. To calculate the redistribution of the hydrogen and oxygen between the drywell and suppression chamber, a two-region computer model of the primary containment system is used. This model takes into consideration hydrogen and oxygen generation from the metal-water reaction and radiolysis. The calculation determines the inventory, partial pressure, and mole fraction of each atmospheric constituent in both regions as a function of time.

Tables 6.2-58, 6.2-59, 6.2-59C, and 6.2-59D present the parameters used in the analysis of the oxygen and hydrogen buildup within the primary containment. The minimum recombiner flow necessary to control the formation of combustible gases is 120 scfm. Although the recombiner has a design processing capacity of 150 scfm, the analysis to determine post-accident hydrogen and oxygen concentrations within primary containment

uses the 120 scfm flow. The hydrogen and oxygen concentration transient plots are shown on Figures 6.2-72H and 6.2-72I.

Operation of the recombiner at 150 scfm, as opposed to 120 scfm, would reduce at a faster rate the post-accident concentrations of combustible gases in primary containment.

6.2.5.3.4 Failure Modes and Effects Analysis

Originally, the FMEA for the CGCS was contained in the Unit 2 FMEA document, which is historical. FMEAs for plant systems are now performed and controlled by the design process.

6.2.5.4 Tests and Inspections

Each active component of the CGCS is testable during normal reactor power operation. This system will be tested periodically to assure that it will operate correctly whenever required. Preoperational tests of the CGCS are conducted during the final stages of plant construction prior to initial startup. These tests assure correct functioning of all controls, instrumentation, recombiners, piping, and valves. System reference characteristics such as pressure differentials and flow rates are documented during the preoperational tests and will be used as base points for measurement in subsequent operational tests.

During normal operation, the recombiner system piping, valves, instrumentation, wiring, and other components can be inspected visually at any time since, with the exception of the internal isolation valves, they are outside the primary containment. Further information may be found in Chapter 14.

6.2.5.5 Instrumentation Requirements

<u>Description</u>

Safety-related instruments and controls are provided for automatic and manual control of the hydrogen recombiners. The controls and monitors described below are located in the main control room. The control logic is shown on Figure 6.2-72K.

Instrumentation requirements for the CPS and the SGTS portions of the CGCS are described in Sections 9.4.2.5 and 6.5.1.5, respectively.

<u>Operation</u>

The hydrogen recombiner inlet and outlet isolation valves close automatically on a LOCA or manual isolation signal and can be opened manually during a LOCA by means of the associated hydrogen recombiner LOCA override keylock switch.

The redundant cooling water block valves located in the water supply lines are manually operated. These valves are interlocked with the recombiner discharge line containment isolation valves so that they cannot be opened unless the isolation valves are already open. They will also automatically close if the isolation valves are closed.

The strainer blowdown drain valves are interlocked with the redundant cooling water block valves. In the automatic mode, the blowdown drain valves close when the associated block valve is opened and will open when the block valves close. The blowdown valves can also be closed manually and opened manually.

Recombiner cooling water inlet valves close automatically when the associated recombiner unit is turned off. The air inlet valves are manually closed after the recombiner unit is turned off and they will stop when the associated control switch is released.

Recombiner gas heaters and the gas blower are turned on manually, after which the reaction chamber temperature is automatically controlled by the SCR controller. Temperatures are set at manual/automatic stations. Interlocks prevent operation of the recombiner when its cooling water inlet and block valves are less than fully open, when through gas flow is low, when heater gas inlet or outlet temperature is high, or when high temperature or pressure conditions prevail. Gas blowers are turned off under recombiner high temperature or pressure conditions.

Monitoring

Indicators are provided for each recombiner for each of the following parameters:

1. Blower inlet temperature.
2. Heater wall temperature.
3. Reaction chamber shell temperature.
4. Return gas temperature.
5. Reaction chamber temperature.
6. Inlet temperature.
7. Heater inlet gas temperature.
8. Heater outlet gas temperature.
9. Inlet pressure.
10. Through gas flow.

Alarms are provided for each of the following conditions:

1. Hydrogen recombiner system inoperable.
2. Recombiner blower inlet temperature high.
3. Recombiner heater wall temperature high.
4. Recombiner heater wall temperature high/high.
5. Recombiner reaction chamber shell temperature high.
6. Recombiner reaction chamber shell temperature high/high.
7. Recombiner return gas temperature high.
8. Recombiner inlet pressure high.
9. Recombiner reaction chamber temperature high.
10. Recombiner reaction chamber temperature low.
11. Recombiner through gas flow low.
12. Recombiner heater gas inlet temperature high.
13. Recombiner heater gas outlet temperature high.
14. Primary containment isolation valve LOCA override.
15. Recombiner containment isolation valves motor overload.

6.2.6 Containment Leakage Testing

This section presents the proposed testing program for the primary containment, containment penetrations, and containment isolation barriers that comply with the requirements of the general design criteria and Appendix J, Option B, to 10CFR50. As described in Table 1.8-1 of the USAR, the Appendix J Testing Program Plan, and the Technical Specifications, implementation of the testing program is in accordance with NEI 94-01, Revision 2-A, rather than RG 1.163 (September 1995). Each of the tests described in this section will be performed during the startup and test program and as periodic tests.

6.2.6.1 Containment Integrated Leakage Rate Test (ILRT) (Type A Test)

Following the completion of the construction, repair, inspection, and testing of welded joints, penetrations, and mechanical closures, including the satisfactory completion of the structural integrity tests as described in Section 3.8.1, a preoperational primary containment leakage rate test was performed to verify

that the actual containment leak rate does not exceed the design limits. To ensure a successful ILRT, local leakage tests (Type B and C tests) were performed on penetrations and isolation valves, and repairs were made, if necessary, to ensure that leakage through the containment isolation barriers did not exceed the limits established by Technical Specifications or the Owner.

A general inspection of the accessible interior and exterior surfaces of the primary containment and components will be performed prior to any Type A test. Any structural deteriorations requiring repair prior to performing Type A tests and the corrective actions taken will be included in the test report. Leakage rates of equipment to be Type B or C tested shall include "as-found" (pre-repair) and "as-left" (post-repair) condition, with the exception of the test performed prior to the initial Type A test.

An ILRT is performed on the entire primary containment to determine that the total leakage through all primary containment isolation barriers does not exceed the design leakage rate of 1.1 percent/day at the primary containment DBA LOCA pressure (P_a). The pertinent test data, including test pressures and acceptance criteria, are presented in Table 6.2-60.

Systems penetrating containment that may not be vented to the primary containment atmosphere during the ILRT, but whose containment isolation valves are Type C tested, are listed below.

	System	Exception Justification
1.	Reactor building closed loop cooling water (RBCLCW)	1
2.	Low-pressure core injection subsystem of residual heat removal (LPCI)	1, 2
3.	High-pressure core spray (HPCS)	1, 2
4.	Low-pressure core spray (LPCS)	1, 2
5.	Reactor core isolation cooling (RCIC)	2
6.	Feedwater (FWS)	2
7.	Standby liquid control (SLC)	2
8.	Reactor coolant recirculation pump seal injection	1
9.	Reactor water cleanup (RWCU)	1
10.	Control rod drive	1, 2

11.	Main steam	1
12.	Shutdown cooling subsystem of residual heat removal	1, 2
13.	Containment spray subsystem of residual heat removal	2
14.	Suppression pool cooling subsystem of residual heat removal	2

Exception Justification

1. Systems that are required for proper conduct of the Type A test or to maintain the plant in a safe condition during the test shall be operable in their normal mode and need not be vented.

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

The Type A ILRT is normally performed at the end of a refueling outage. At this time, the RPV head is installed and tensioned. In order to maintain the minimum vessel flange and head flange temperature required by the Technical Specifications, the reactor vessel water level may be raised to the flange level. Thus, the main steam lines may be flooded during the time the ILRT is performed, and may not be vented to the primary containment. The CRD and hydraulic control for the reactor recirculation flow control valves will not be vented during the ILRT as justified by Notes 17 and 26 of Table 6.2-56, respectively.

During the Type A test, the RPS system will be energized (i.e., SCRAM reset). Scram discharge volume (SDV) vent and drain valves (2RDS*AOV123, 124, 130 and 132) will be Type C tested and leakages will be added to Type A results for overall containment leakage.

The preoperational (initial) Type A test was performed in accordance with 10CFR50 Appendix J, ANS-N45.4/ANSI-56.8-1981. This method employs:

4 hr (min) stabilization period

24 hr (min) ILRT test period (utilizing total time analysis of BN-TOP-1 and mass point analysis technique of ANSI-56.8)

1 to 4 hr (min) verification period

The 24-hr Type A test provided the baseline for post-operational tests.

Subsequent refueling outage Type A tests can be performed using total time analysis which provides the bases for a Type A test duration of 6 hr (min) with 20 data sets (min) (BN-TOP-1 referenced).

With the implementation of Option B, "Performance-Based Requirements of Appendix J to 10CFR50," for Type A, B and C leakage rate testing, the Type A test may be performed using a mass point analysis for a test duration of 8-24 hr with 30 data sets (min) in accordance with ANSI/ANS-56.8-2002, or a total time analysis for a test duration of 6 hr (min) with 20 data sets (min) in accordance with BN-TOP-1.

The test method utilized is the absolute method utilizing the total time and/or mass point analysis techniques applicable to the test duration. Values of primary containment atmosphere dry-bulb temperature, dew point temperature (vapor pressure), and pressure are used in the leakage rate calculations.

The primary containment leakage monitoring system (LMS) provides means for monitoring the primary containment pressure during ILRT. Two independent pressure-sensing lines, each equipped with a quartz digital-type absolute pressure manometer, are provided in LMS system. The quartz manometers are required for the Type A test only. To protect these devices from damage, the quartz manometers will be isolated, disconnected, and stored when not in use. A third quartz manometer is provided as a spare (Figure 6.2-73). Two independent temporary quartz digital-type absolute pressure gauges may be used in place of the instruments installed in LMS.

Eighteen temperature elements and six humidity analyzers are provided in the CMS system to monitor dry-bulb and dew point temperatures, respectively (Figures 6.2-71a and 6.2-71b). Additional instrumentation may be installed as required to complete the test. Temporary temperature elements and temporary humidity analyzers may be used in place of the instruments installed in LMS.

The test procedure, test equipment and facilities, period of testing, and verification of leak test accuracy follow the recommendations of BN-TOP-1 or ANSI/ANS-56.8-2002, as described in the 10CFR50 Appendix J Testing Program Plan.

Acceptance criteria and test intervals Type A, B, and C tests will be in conformance with 10CFR50 Appendix J, Option B.

6.2.6.2 Containment Penetration Leakage Rate Tests (Type B Tests)

Containment penetrations whose design incorporates resilient seals, gaskets, or sealant compounds; airlock door seals, equipment and access hatches with resilient seals or gaskets; and other such penetrations received a preoperational leak test in

accordance with Appendix J to 10CFR50. Periodic leak tests will be in accordance with 10CFR50 Appendix J, Option B.

The following penetrations will be tested to Type B criteria:

1. Equipment hatch (Figure 3.8-6).
2. Personnel airlock/equipment hatch (Figure 3.8-6).
3. Suppression pool access hatch (Figure 3.8-7).
4. CRD removal hatch (Figure 3.8-7).
5. Drywell head (Figure 3.8-8).
6. Electrical penetrations (Figure 3.8-9).
7. TIP 2NMT*Z31A,B,C,D & E penetration flanges (Figure 6.2-75b).
8. Midspan single O-ring flange in 2NMT*Z31B (Figure 6.2-75a).
9. Escape airlock (Figure 3.8-6).
10. Blind flanges on piping penetrations Z46C and Z74.
11. TIP N_2 purge penetration (Figure 6.2-75).
12. TIP 2NMT*Z31A,B,C,D & E penetration bellows.

The makeup pressure test will be utilized to determine primary containment penetration leak rates. In this test, leakage is measured by pressurization between the double seal and measurement of the makeup flow required to maintain the test pressure.

Penetrations that rely on welds for sealing will not be Type B tested but they will be subject to the ILRT conditions. Leakage from these penetrations will be included in the overall leakage rate measured during the ILRT. Test pressures are given in Table 6.2-60.

The primary containment airlocks shall be tested at P_a in accordance with 10CFR50 Appendix J, Option B.

The acceptance criteria for the preoperational primary containment penetration leakage rate test are in compliance with the criteria given in Appendix J to 10CFR50. The periodic testing acceptance criteria are established in Technical Specifications.

6.2.6.3 Primary Containment Isolation Valve Leakage Rate Tests (Type C Tests)

Primary containment isolation valve leakage rate tests will be performed by local pressurization in accordance with the requirements of Appendix J to 10CFR50, Option B. The pressure will be applied in the same direction as it would be applied when the valve is required to perform its safety function, unless it can be determined and documented that the results from the test for a pressure applied in the opposite direction will provide equivalent or more conservative results. Table 6.2-65 lists all plant primary containment isolation valves which may be reverse tested, and provides the justification necessary to ensure that the reverse test is as conservative as testing the valve in the same direction as post-accident flow.

Containment isolation valves will be Type C tested in accordance with Appendix J, Option B. Each valve to be tested will be closed by normal operation, without any preliminary exercising or adjustment. Table 6.2-56 lists all primary containment isolation valves on pipelines penetrating the primary containment.

Primary containment isolation valves with bonnet vent pathways installed to prevent pressure locking may be tested in accordance with Appendix J by pressurizing the main line and the bonnet vent pathway simultaneously since both pathways will be open during an accident. This method will pressurize the valve between the seats as well as upstream of the disc.

It should be noted that test, vent, and drain (TVD) connections within the boundaries of containment isolation valves are not Type C tested since they are small lines, usually 3/4 in or smaller; they are normally only open for the Type C testing or ILRT; they are part of a double isolation barrier for redundancy, consisting of either the main process inboard isolation valve and a single TVD valve for those TVD connections outside the main process inboard isolation valve, or double TVD valves for those TVD connections inside the main process inboard isolation valve; and they are designed as Quality Assurance (QA) Category I, Safety Class 2. In addition, their use is controlled by administrative procedures.

The test pressures and acceptance criteria for the primary containment isolation valve leakage rate tests are given in Table 6.2-60.

6.2.6.4 Additional Requirements

The combined leakage rate for all penetrations and valves subject to Type B and C tests will be in accordance with 10CFR50 Appendix J, Option B.

6.2.6.5 Scheduling and Reporting of Periodic Tests

The periodic leakage test schedule is given in Technical Specifications.

6.2.6.6 Special Testing Requirements

The reactor building will be tested as required by Technical Specifications.

Preoperational high- and low-pressure suppression pool bypass leakage tests will be performed once prior to fuel load to determine the bypass leakage from the drywell into the suppression chamber. The high-pressure test will be initiated at the pressure differential of 25 psi +0.5, -0.0 between the drywell and suppression chamber. The low-pressure test will be initiated at 3 psid. In each case, the drywell pressure will be monitored at regular time intervals and compared with expected pressure decay for $A/\sqrt{K}$ of 0.0054 sq ft.

The test pressures and acceptance criteria for the drywell bypass leakage tests are given in Table 6.2-61.

6.2.7 References

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5. BNL-NUREG-24532 (Informal Report), Hydrogen Release Rates from Corrosion of Zinc and Aluminum, May 1978.

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17. Deleted.

18. NEDC-32523P-A, "Generic Evaluations of General Electric Boiling Water Reactor Extended Power Uprate," Class III, February 2000, including Supplement 1.

19. NEDC-33004P-A, "Constant Pressure Power Uprate," Revision 4, Class III, July 2003.

20. NRC Letter, "Nine Mile Point Nuclear Station, Unit No. 2 – Issuance of Amendment Re: Implementation of Alternative Radiological Source Term (TAC No. MD5758), dated May 29, 2008.

21. NEDC-33006P-A, "Licensing Topical Report, General Electric Boiling Water Reactor, Maximum Extended Load Line Limit Analysis Plus," Revision 3, Class III, June 2009.

TABLE 6.2-1
(Sheet 1 of 1)
THERMOPHYSICAL PROPERTIES OF PASSIVE HEAT SINKS

Material	Density (lbm/cft)	Specific Heat (Btu/lbm-°F)	Thermal Conductivity (Btu/hr-ft-°F)
Carbon steel	490	0.11	26.0
Stainless steel	488	0.11	9.40
Concrete	144	0.20	0.54

TABLE 6.2-2
(Sheet 1 of 2)
MODELING OF PASSIVE HEAT SINKS

Heat Sink No.	Description	Exposed Surface Area of One Side (ft^2)	Material(s) (from left to right)	Thickness (ft)	Surface Exposure (left/right)
1	Drywell head	2,395	Carbon steel	0.09375	Drywell/reactor building
2	Refueling bulkhead	372	Stainless steel Carbon steel	0.02083 0.08333	Drywell/drywell
3	Star truss, ring girder, and stabilizers	890	Carbon steel	0.1458	Insulated/drywell
4	Grating and structural steel in drywell	26,800	Carbon steel	0.04916	Drywell/drywell
5	Biological shield wall	4,160	Concrete Carbon steel	0.7292 0.125	Insulated/drywell
6	Reactor pedestal in drywell	1,450	Concrete	2.56	Insulated/drywell
7	Drywell/wall	17,100	Carbon steel Concrete	0.03125 5.22	Drywell/reactor building
8	Drywell floor suppression chamber ceiling	5,700	Concrete Carbon steel Concrete	0.4008 0.01562 4.00	Drywell/suppression chamber
9	Reactor pedestal in suppression chamber above pool	2,134	Stainless steel Concrete Stainless steel	0.0208 4.00 0.0208	Suppression chamber/suppression chamber
10	Reactor pedestal in suppression chamber in pool region	1,430	Stainless steel Concrete Stainless steel	0.0208 4.00 0.0208	Pool region/pool region
11	Steel in suppression chamber	6,650	Stainless steel	0.03033	Suppression chamber/suppression chamber
12	Downcomers exposed to suppression chamber	27,684	Stainless steel	0.03125	Drywell/suppression chamber
13	Concrete exposed to suppression chamber airspace	3,130	Concrete	1.016	Suppression chamber/suppression chamber

TABLE 6.2-2 (Cont'd.)
(Sheet 2 of 2)

Heat Sink No.	Description	Exposed Surface Area of One Side (ft^2)	Material(s) (from left to right)	Thickness (ft)	Surface Exposure (left/right)
14	Suppression chamber wall	10,363	Stainless steel Concrete	0.03125 5.22	Suppression chamber/reactor building
15	Suppression pool wall	6,706	Stainless steel Concrete	0.03125 5.22	Pool region/reactor building
16	Containment mat	6,070	Stainless steel Concrete Carbon steel Concrete	0.01042 1.00 0.02083 5.177	Pool region/insulated

TABLE 6.2-3
(Sheet 1 of 3)
CONTAINMENT DESIGN PARAMETERS

Design Parameters	
Drywell	
1. Internal design pressure, psig	45
2. External design pressure, psig	4.7
3. Drywell floor design differential pressure, psid	
a. Downward	25
b. Upward	10
4. Design environmental temperature, °F	340
5. Design structural temperature, °F	293
6. Design leak rate, % by weight of the containment air/day at 45 psig and 293°F	1.1
7. Drywell net free volume (including downcomers), ft^3	303,418[(1)]
Suppression Chamber	
1. Internal design pressure, psig	45
2. External design pressure, psig	4.7
3. Drywell floor design differential pressure, psid	
a. Downward	25
b. Upward	10
4. Design environmental temperature, °F	270
5. Design structural temperature, °F	212
6. Design leak rate, % by weight of the containment air/day at 45 psig and 293°F	1.1
7. Suppression chamber free volume, ft^3	
a. Minimum at HWL	192,028[(2)]
b. Maximum at LWL	201,322[(2)]
Suppression Pool	
1. Suppression pool volume, ft^3	
a. Minimum at LWL	145,495[(3)]
b. Maximum at HWL	154,794[(3)]
2. Suppression pool surface area, ft^2	5,813[(4)]
3. Suppression pool depth, ft	
a. Minimum at LWL	23.47
b. Maximum at HWL	24.97

TABLE 6.2-3
(Sheet 2 of 3)

Downcomers

1.	Number of downcomers	121[(4)]
2.	Downcomer diameter, ID, in	23.25
3.	Net free downcomer area, ft^2	362.64[(4)]
4.	Downcomer submergence, ft	
	a. Minimum	9.5
	b. Maximum	11
5.	Downcomer system loss coefficient (entrance + pipe friction)	1.37

Vacuum Breakers

1.	No. of vacuum breaker sets	4
2.	Opening pressure, suppression chamber to drywell, psid	0.25
3.	Pipe, ID, in	23.25
4.	Overall loss coefficient of each assembly	12

KEY: HWL = High water level
LWL = Low water level

(1) The original LOCA analysis results are based on 303,418 cu ft of drywell volume as shown. The calculated net free volume of the drywell is 306,200 cu ft. This difference has a negligible effect on drywell peak pressure (see Table 6.2-17). The EPU/MELLLA+ analyses (Sections 6.2.1.1.6 and 6.2.1.1.7) use 306,200 cu ft.

(2) The original LOCA analysis results are based on suppression chamber free volumes of 192,028 cu ft minimum at HWL and 201,322 cu ft maximum at LWL as shown. The calculated free volumes of the suppression chamber are 190,600 cu ft minimum at HWL and 199,800 cu ft maximum at LWL. These differences have a negligible effect on drywell peak pressure. The EPU/MELLLA+ analyses (Sections 6.2.1.1.6 and 6.2.1.1.7) use the calculated volumes (190,600 cu ft at HWL; 199,800 cu ft at LWL).

(3) The original LOCA analysis results are based on suppression pool volumes of 145,495 cu ft minimum at LWL and 154,794 cu ft maximum at HWL as shown. The calculated volumes of the suppression pool are 145,200 cu ft minimum at LWL and 154,400 cu ft maximum at HWL as shown. These differences have a negligible effect on drywell peak pressure. The EPU/MELLLA+ analyses (Sections 6.2.1.1.6 and 6.2.1.1.7) use the calculated volumes (145,200 cu ft at LWL; 154,400 cu ft at HWL).

(4) The original LOCA analysis results are based on 123 downcomers. Two downcomers were eliminated (121 remain

TABLE 6.2-3
(Sheet 3 of 3)

functional) in a subsequent design modification to accommodate quenchers on the RHR heat exchanger relief valve discharge lines. The original LOCA analysis results also are based on a suppression pool surface area of 5,813 sq ft and a net free downcomer area of 362.64 sq ft as shown. These values are consistent with the 123 downcomers included in the LOCA analysis. The area values that are consistent with the 121 functional downcomers are 5,800 sq ft of suppression pool surface area and 357 sq ft of net free downcomer area. The difference between the areas corresponding to 121 versus 123 downcomers is less than 2 percent, and has a negligible effect on drywell peak pressure. The EPU analyses (Section 6.2.1.1.6) use the parameter values based on 121 functional downcomers.

TABLE 6.2-4
(Sheet 1 of 1)
RESULTS OF LARGE BREAK ACCIDENT ANALYSIS (CASE C[3])

	OLTP Analysis		Extended Power Uprate Analysis
	Recirculation Line Break	Steam Line Break	Recirculation Line Break
1. Peak drywell pressure, psig	39.75	38.16	39.5
2. Time of peak drywell pressure, sec	252.60	245.60	147
3. Peak drywell floor differential downward pressure, psid	16.89	14.90	18.52
4. Time of peak drywell floor differential, sec	0.95	0.95	1.07
5. Peak suppression chamber pressure, psig	33.98	31.87	33.8
6. Time of peak suppression chamber pressure, sec	444.09	247.60	148
7. Peak drywell temperature, °F	286.21	303.24	286.1
8. Peak suppression chamber temperature, °F	206.70	207.17	207
9. Peak bulk pool temperature, °F	206.84	207.52	207

* As a result of the 24-month fuel cycle, the reactor core decay heat has increased by a small amount. The resulting suppression pool temperature is less than the design limit.

(1) Additional analysis to reconcile the Technical Specification pump curves against the assumed pump curves used as input for the power uprate analysis shows a maximum containment pressure of <39.75 psig.

(2) As a result of an increase in maximum service water temperature to 84°F, the peak bulk pool temperature will increase by a small amount from that determined in the LOCA analysis. The resulting temperature remains less than the design limit.

(3) The EPU analysis assumes the failure of one RHR heat exchanger with the continued availability of all ECCS pumps with their contribution to pump heat.

TABLE 6.2-5
(Sheet 1 of 1)
LONG-TERM PRIMARY CONTAINMENT RESPONSE SUMMARY

Recirculation Suction Line DER
(OLTP Analysis)

Case	ECCS Pumps HPCS/LPCI/ LPCS	Containment Cooling Pumps Cont. Spray/ Pool Cooling	Long-Term Peak Pressure (psig)	Peak Bulk Pool Temperature (°F)
A	1/1/1	2/0	9.35	175.53
B	1/1/0	1/0	16.91	206.51
C	1/1/0	0/1	19.32	206.84

TABLE 6.2-5a
(Sheet 1 of 1)
LONG-TERM PRIMARY CONTAINMENT RESPONSE SUMMARY

Recirculation Suction Line DER
(EPU Analysis)

Case	ECCS Pumps HPCS/LPCI/ LPCS	Containment Cooling Pumps Cont. Spray/ Pool Cooling	Long-Term Peak Pressure (psig)	Peak Bulk Pool Temperature (°F)
C	1/1/0	0/1	18.3	207

TABLE 6.2-6
(Sheet 1 of 2)
ENGINEERED SAFETY FEATURE SYSTEMS
INFORMATION FOR CONTAINMENT RESPONSE ANALYSES

For Large Break Accidents

	Full Capacity	OLTP Analysis			EPU Analysis[5]
		Case A	Case B	Case C	
Drywell Spray (RHR System)					
1. Number of pumps	2	2	1	0	0
2. Capacity per pump	6,870 gpm	See Figure 6.2-47			
3. Number of headers	2	2	1	0	0
4. Flow distribution, %	94	94	94	0	0
5. Spray thermal efficiency, %	100	90	90	0	0
Suppression Chamber Spray (RHR System)					
1. Number of pumps	2	2	1	0	0
2. Capacity per pump	6,870 gpm	See Figure 6.2-47			
3. Number of headers	1	1	1	0	0
4. Flow distribution, %	6	6	6	0	0
5. Spray thermal efficiency, %	100	0	0	0	0
RHR Heat Exchanger					
1. Number of heat exchangers	2	2	1	1	1
2. Type of heat exchanger	Shell/tube	Shell/tube	Shell/tube	Shell/tube	Shell/tube
3. K-factor, Btu/sec-°F (pool cooling)	270*	199	199	199	265
4. Design service water temperature, °F					
a. Minimum	32				
b. Maximum	84	77	77	77	84

* Minimum required per RHR heat exchanger performance testing program.

TABLE 6.2-6
(Sheet 2 of 2)

	Full Capacity	OLTP Analysis[2]			EPU Analysis[5]
		Case A	Case B	Case C	
Emergency Core Cooling Systems (ECCS)					
1. High-pressure core spray (HPCS) system					
a. Number of pumps	1	1	1	1	1
b. Capacity of pump	Fig. 6.3-3a	See Figure 6.2-48			≤6,250 gpm
2. Low-pressure core spray (LPCS) system					
a. Number of pumps	1	1	0	0	1
b. Capacity of LPCS pump	Fig. 6.3-4a	See Figure 6.2-49			≤6,600 gpm
3. Low-pressure coolant injection (LPCI) mode					
a. Number of pumps	3	3	2	2	3
b. Capacity of each LPCI (RHR) pump	Fig. 6.3-5a	See Figure 6.2-50			≤6,660 gpm
4. Automatic depressurization system					
a. Total number of safety/relief valves	18	0	0	0	0
b. Number actuated on ADS	7	0	0	0	0

* Deleted.
** Deleted.
*** Deleted.
(1) Deleted.
(2) Ten min after LOCA, LPCI mode is switched to either containment spray mode (Cases A, B) or pool cooling mode (Case C), leaving one RHR pump in LPCI mode for long-term coolant injection.
(3) Deleted.
(4) Deleted.
(5) The EPU analysis assumes the failure of one RHR heat exchanger with the continued availability of all ECCS pumps with their contribution to pump heat.

TABLE 6.2-7
(Sheet 1 of 1)
MASS AND ENERGY RELEASE DATA – OLTP ANALYSIS[6]

Recirculation Suction Line DER With Feedwater (Case C)

Pipe ID, in	21.564
Effective break area	See Figure 6.2-51
Blowdown code	LOCTVS

Blowdown Table

Time After Accident (sec)	Blowdown (lb/sec)	Enthalpy (Btu/lb)	Reactor Vessel Pressure (psia)
0.01	31,625	550.7	1,055
0.1	31,617	550.6	1,054
0.5	31,502	549.8	1,049
1.0	31,357	548.9	1,043
2.0	23,780	547.8	1,035
4.0	23,868	548.4	1,040
8.0	23,743	547.5	1,034
16.0	22,804	534.3	948
23.3[1]	21,515	516.1	837
23.4[1]	14,263	635.3	835
30.0[2]	11,301	600.7	604
50.0	6,419	462.5	230
75.1	5,662	341.2	105
100.1	5,008	300.0	77
200.1	6,283	268.1	66
252.6[3]	5,181	257.9	56
257.6[4]	0	-	-
282.6[5]	0	-	-
283.6	3,291	244.8	-
500.1	3,329	194.4	19
1,000.0	2,201	176.0	-
10,000.0	2,183	171.6	-
100,000.0	2,169	179.7	-

(1) Blowdown changes from liquid to two-phase.
(2) ECC pumps start.
(3) Peak drywell pressure.
(4) Blowdown ends.
(5) Water level recovers to jet pump nozzle level.
(6) EPU blowdown rates and enthalpy are shown on Figures 6.2-34a and 6.2-35a.

TABLE 6.2-8
(Sheet 1 of 1)
MASS AND ENERGY RELEASE DATA - OLTP ANALYSIS

Recirculation Suction Line DER Without Feedwater (Case C)

Time (sec)	Blowdown Rate (lb/sec)	Enthalpy (Btu/lb)	Reactor Vessel Pressure (psia)
0.01	31,625	550.7	1,055
0.1	31,635	550.7	1,055
0.5	31,595	550.5	1,053
1.0	31,555	550.2	1,051
2.0	24,098	550.3	1,052
4.0	24,416	553.5	1,074
8.0	24,788	557.9	1,103
17.8(1)	24,322	552.5	1,067
17.9(1)	15,581	695.8	1,066
30.0(2)	7,692	724.1	592
60.3(3)	2,425	450.6	94
94.5(4)	0	-	-
200.6(5)	0	-	-
201.6	3,338	232.2	-
500.1	3,349	173.3	-
1,000.1	2,213	158.9	-
10,000	2,192	161.0	-
57,750(6)	2,167	183.7	-
100,000	2,171	177.7	-

(1) Blowdown changes from liquid to two-phase.
(2) ECCS pump start.
(3) Peak drywell pressure.
(4) Blowdown ends.
(5) Water level recovers to jet pump nozzle level.
(6) Peak pool temperature.

TABLE 6.2-9
(Sheet 1 of 1)
INITIAL CONDITIONS FOR CONTAINMENT RESPONSE ANALYSIS
(OLTP Analysis)

<u>Reactor Coolant</u>	
Reactor power, MWt	
Original analysis (corresponding to 105% steam flow)	3467
Average coolant pressure, psig	1040
Average coolant temperature, °F	551
Mass of reactor coolant system - liquid, lb	644,850
Mass of reactor coolant system - steam, lb	24,324
Volume of liquid in reactor vessel, ft^3	12,570
Volume of liquid in recirculation loops, ft^3	770
Volume of liquid in feedwater pipes, ft^3	520
Volume of steam in reactor vessel, ft^3	8564
Volume of steam in steam lines, ft^3	1510
<u>Drywell</u>	
Pressure, psig	0.75
Temperature, °F	135
Relative humidity, %	40
<u>Suppression Chamber</u>	
Pressure, psig	0.75
Air temperature, °F	90[1]
Pool water temperature, °F	90
Relative humidity, %	100
Water volume, ft^3 (HWL)	154,400
Downcomer submergence, maximum, ft	11

[1] The initial suppression chamber air temperature used in the containment response analysis is 90°F. The suppression chamber air space temperature could be as high as 122°F. Use of 122°F would result in lower peak pressures.

TABLE 6.2-9a
(Sheet 1 of 1)
INITIAL CONDITIONS FOR CONTAINMENT RESPONSE ANALYSIS
(EPU Analysis)

<u>Reactor Coolant</u>	
Reactor power, MWt (102% of rated power)	4068
Average coolant pressure, psig	1040
Average coolant temperature, °F	551
Mass of reactor coolant system - liquid, lb	See Ref. 16
Mass of reactor coolant system - steam, lb	See Ref. 16
Volume of liquid in reactor vessel, ft^3	11,799
Volume of liquid in recirculation loops, ft^3	770
Volume of liquid in feedwater pipes, ft^3	See Ref. 16
Volume of steam in reactor vessel, ft^3	See Ref. 16
Volume of steam in steam lines, ft^3	1663
<u>Drywell</u>	
Pressure, psig	0.75
Temperature, °F	105
Relative humidity, %	40
<u>Suppression Chamber</u>	
Pressure, psig	0.75
Air temperature, °F	90
Pool water temperature, °F	90
Relative humidity, %	100
Water volume, ft^3 (HWL)	154,400 for peak drywell pressure 145,200 for peak pool temperature
Downcomer submergence, maximum, ft	11

TABLE 6.2-10
(Sheet 1 of 1)
DECAY HEAT RATE AFTER SCRAM - OLTP ANALYSIS

Time After Accident (sec)	Decay Heat (Btu/sec)	Time After Accident (sec)	Decay Heat (Btu/sec)
0.0	295,822	4,800	48,252
0.01	286,060	6,000	44,998
0.2	278,533	7,200	42,730
0.5	269,165	9,000	40,330
0.7	264,136	10,800	38,555
1.0	257,957	12,600	37,142
3.0	235,310	14,400	35,893
5.0	224,036	16,200	34,775
7.0	216,311	18,000	33,757
10.0	207,437	19,800	32,833
20.0	187,485	21,600	31,972
30.0	177,427	23,400	31,173
60.0	156,818	36,000	26,972
90.0	146,202	54,000	23,564
180.0	125,133	72,000	21,618
390.0	102,124	90,000	20,326
600.0	92,822	259,200	14,410
1,500.0	70,438	432,000	11,494
2,400.0	61,202	604,800	9,880
3,600.0	53,215	1,000,000	8,119

TABLE 6.2-10a
(Sheet 1 of 1)
DECAY HEAT RATE AFTER SCRAM – EPU LONG-TERM ANALYSIS

Time After Accident (sec)	Decay[1] Heat (Btu/sec)	Time After Accident (sec)	Decay[1] Heat (Btu/sec)
0.0	4,040,000	7,203	44,100
0.01	4,030,000	14,426	38,000
0.51	3,230,000	36,058	30,700
1.0	2,440,000	72,051	25,400
5.1	2,250,000	259,930	17,800
11.0	1,530,000	604,088	12,500
30.0	546,000	1,000,000	10,200
60.0	329,000		
119.8	311,000		
120.2	124,000		
180.3	115,000		
390.7	98,500		
600.9	90,000		
2,412.1	62,800		
3,601.0	55,100		

(1) The data represent combined heat (fission power, decay heat, fuel and hot metal sensible heat, and metal-water reaction heat) to reactor coolant.

TABLE 6.2-11
(Sheet 1 of 1)
FISSION POWER COASTDOWN HEAT TO COOLANT - OLTP ANALYSIS

Time After Accident (sec)	Heat Rate to Coolant (Btu/sec)
0.0	3,286,909
0.1	2,988,786
0.2	2,809,649
0.4	2,219,321
0.6	1,719,053
1.0	908,173
2.0	317,515
5.0	27,939
7.0	11,504
70.0	0

TABLE 6.2-12
(Sheet 1 of 1)
HEAT TO COOLANT FROM FUEL AND HOT METALS - OLTP ANALYSIS

Recirculation Suction Line DER With Feedwater (Case C)

Time After Accident (sec)	Heat Release Rate of Fuel (Btu/sec)	Heat Release Rate of Metals (Btu/sec)	Total Release Heat to Coolant (Btu/sec)
0.0	0	0	0
0.255	775,041	8,011	783,052
0.75	1,719,046	13,527	1,732,573
1.0	1,926,367	14,575	1,940,942
1.5	2,168,014	15,570	2,183,584
2.0	2,181,091	14,576	2,195,667
2.5	2,092,005	12,367	2,104,372
5.5	1,419,480	5,585	1,425,065
13.0	465,232	37,966	503,198
18.0	226,854	63,901	290,755
23.0	142,700	101,709	244,409
28.0	133,846	161,817	295,663
33.0	143,354	222,290	365,644
50.5	115,729	324,300	440,029
105.25	16,273	296,096	312,369
202.1	2,321	238,320	240,641
404.1	6,233	122,560	128,793
808.1	749	29,400	30,149
1,608.1	253	10,500	10,753
3,040.1	31	3,700	3,731

TABLE 6.2-13
(Sheet 1 of 1)
METAL-WATER REACTION HEAT RATE - OLTP ANALYSIS

Time After Accident (sec)	Heat Rate (Btu/sec)
0	15,290
10	15,290
20	15,290
30	15,290
40	15,290
50	15,290
60	15,290
70	15,290
120	15,290
120.0001	0
1×10^6	0

TABLE 6.2-14
(Sheet 1 of 1)
ECCS PUMP HEAT TO COOLANT (CASE B AND C) - OLTP ANALYSIS

Time After Accident (sec)	Heat Rate (Btu/sec)
0.0	0
30.0	0
30.0001	3,583
1,000,000.0	3,583

TABLE 6.2-14a
(Sheet 1 of 1)
ECCS PUMP HEAT TO COOLANT (CASE C) – EPU LONG-TERM ANALYSIS

Time After Accident (sec)	Heat Rate (Btu/sec)
0.0	0
53.1	0
53.2	2,160
66.4	2,160
60.5	0
60.5001	5,337
1,000,000.0	5,337

TABLE 6.2-15
(Sheet 1 of 2)
ENERGY BALANCE - OLTP ANALYSIS

Recirculation Suction Line DER with Feedwater (Case C)

(Energy Distribution in Millions of Btus)

	Seconds				
	0.0	23.40	252.60	257.59	283.09
Heat Sources					
1. Reactor coolant					
a. Steam	28.17392	43.18941	2.73270	2.65715	2.10555
b. Liquid	360.79565	93.63072	89.80835	87.45774	104.80822
2. Stored heat					
a. Core	32.99840	14.90545	7.54116	7.48245	7.29455
b. Vessel	93.40133	93.33778	90.60350	90.54989	90.27790
c. Internals	84.50320	83.98373	63.30014	62.96731	61.36406
Heat Sinks					
3. Drywell air	0.35326	0.00787	0.00835	0.00834	0.00830
4. Drywell water vapor	0.91843	37.28526	41.87342	41.70043	40.96413
5. Drywell water on floor	0.0	21.92422	33.84970	33.92318	33.98570
6. Drywell heat sinks	0.0	19.18586	46.11338	46.27274	46.88397
7. Suppression chamber air	0.13900	0.45202	0.63234	0.63434	0.63450
8. Suppression chamber water vapor	0.42691	0.70645	1.42400	1.43450	1.44707
9. Suppression chamber pool	557.57594	781.70266	1,162.13325	1,166.06208	1,157.04832
10. Suppression chamber heat sinks	0.0	-0.01084	0.05203	0.05854	0.09130
11. Pool heat sinks	0.0	0.10830	1.40890	1.42939	1.52079
Heat Inputs					
12. Coastdown heat	0.0	3.36247	3.56069	3.56069	3.56069
13. Fission product decay heat	0.0	4.71767	37.34923	37.95238	40.98582
14. Zirconium-water reaction	0.0	0.35745	1.82691	1.82691	1.82691
15. Feedwater	0.0	28.91758	302.38182	302.38182	302.38182
Heat Outputs					
16. Steam line	0.0	9.93873	9.93873	9.93873	9.93873
17. Containment air coolers	0.0	0.0	0.0	0.0	0.0
18. RHR heat exchanger	0.0	0.0	0.0	0.0	0.0

TABLE 6.2-15
(Sheet 2 of 2)

	Seconds				
	50,000	60,000	70,000	80,000	108,950
Heat Sources					
1. Reactor coolant					
a. Steam	0.72618	0.71491	0.69667	0.67408	0.60624
b. Liquid	82.77200	82.42693	81.86074	81.14117	78.84782
2. Stored heat					
a. Core	5.23103	5.20198	5.15952	5.10797	4.94937
b. Vessel	74.92344	74.87634	74.79899	74.70077	74.38867
c. Internals	0.06397	39.95067	39.76467	39.52845	38.77786
Heat Sinks					
3. Drywell air	0.65211	0.65337	0.65065	0.64656	0.63294
4. Drywell water vapor	13.86156	13.65837	13.32184	12.89964	11.62191
5. Drywell water on floor	25.89330	25.79090	25.61790	25.39490	24.67829
6. Drywell heat sinks	109.19730	117.74306	125.39995	132.28088	149.21547
7. Suppression chamber air	0.42229	0.41801	0.41396	0.40877	0.39144
8. Suppression chamber water vapor	6.03021	6.00779	5.90729	5.75536	5.23886
9. Suppression chamber pool	1,873.33376	1,870.69312	1,860.69914	1,845.61741	1,792.38374
10. Suppression chamber heat sinks	38.29997	41.53898	44.03275	45.93566	49.24830
11. Pool heat sinks	31.61547	34.84482	37.66758	40.15445	45.96347
Heat Inputs					
12. Coastdown heat	3.56069	3.56069	3.56069	3.56069	3.56069
13. Fission product decay heat	1,926.79219	2,197.83475	2,457.52243	2,707.55046	3,396.18765
14. Zirconium-water reaction	1.82691	1.82691	1.82691	1.82691	1.82691
15. Feedwater	302.38182	302.38182	302.38182	302.38182	302.38182
Heat Outputs					
16. Steam line	9.93873	9.93873	9.93873	9.93873	9.93873
17. Containment air coolers	0.0	0.0	0.0	0.0	0.0
18. RHR heat exchanger	1,162.72256	1,423.01747	1,682.08051	1,938.75712	2,663.28858

TABLE 6.2-16
(Sheet 1 of 1)
ACCIDENT CHRONOLOGY - OLTP ANALYSIS

Case C (With Feedwater)

	Recirculation Line Break (sec)	Steam Line Break (sec)
1. Downcomers clear	0.85	0.80
2. Froth level reaches top of steam dryer	-	1.11
3. Froth level reaches break location	23.40	-
4. ECC pumps start	30.00	30.00
5. Feedwater stops	244.40	244.40
6. Peak drywell pressure	252.60	245.60
7. End of blowdown	257.10	247.10
8. Reactor pressure vessel is reflooded	283.10	338.60
9. Vacuum breaker opens	444.10	548.10
10. RHR heat exchanger starts (pool cooling)	1,200.0	1,200.0
11. Pool reaches peak temperature	51,550.10	50,800.0

TABLE 6.2-16a
(Sheet 1 of 1)
ACCIDENT CHRONOLOGY - EPU ANALYSIS

Case C (With Feedwater)

	Recirculation Line Break (sec)
1. Downcomers clear	1.0
2. Froth level reaches top of steam dryer	-
3. Froth level reaches break location	21.2
4. ECC pumps start	53-66
5. Feedwater stops	145
6. Peak drywell pressure	147
7. End of blowdown	156
8. Reactor pressure vessel is reflooded	177
9. Vacuum breaker opens	258
10. RHR heat exchanger starts (pool cooling)	1,800
11. Pool reaches peak temperature	43,068

TABLE 6.2-17
(Sheet 1 of 1)
EFFECT OF DRYWELL VOLUME ON PEAK DRYWELL PRESSURE - OLTP ANALYSIS

Recirculation Suction Line DER (Case C)

Drywell Volume (%)	Peak Pressure (psig)
90	37.49
100	39.75
110	41.90

TABLE 6.2-18
(Sheet 1 of 1)
EFFECT OF FEEDWATER ON PEAK DRYWELL PRESSURE -
OLTP ANALYSIS

Recirculation Suction Line DER (Case C)

Break Type	Feedwater Status	Peak Pressure (psig)
Recirculation	Added	39.75
Recirculation	Not added	33.67
Steam	Added	38.16
Steam	Not added	33.98

TABLE 6.2-19
(Sheet 1 of 1)
EFFECT OF AIR CARRYOVER FROM DRYWELL
ON PEAK DRYWELL PRESSURE - OLTP ANALYSIS

Recirculation Suction Line DER (Case C)

Air Carryover Status	Peak Pressure (psig)
No air carryover	17.41
Air carryover	39.75

TABLE 6.2-20
(Sheet 1 of 1)
EFFECT OF DOWNCOMER LOSS FACTOR ON
PEAK DRYWELL PRESSURE - OLTP ANALYSIS

Recirculation Suction Line DER (Case C)

Flow Loss Coefficient of Nominal (%)	Peak Pressure (psig)
90	39.74
100	39.75
110	39.74

TABLE 6.2-20a
(Sheet 1 of 1)
EFFECT OF NO. OF DOWNCOMERS ON
PEAK DRYWELL PRESSURE - OLTP ANALYSIS

Recirculation Suction Line DER (Case C)

No. of Downcomers	Peak Pressure (psig)
123	39.75
121	39.86

TABLE 6.2-21
(Sheet 1 of 1)
EFFECT OF STEAM BYPASS ON PEAK
DRYWELL PRESSURE - OLTP ANALYSIS

Main Steam Line DER (Case C)

Steam Bypass $A/\sqrt{K}$ Factor (ft^2)	Peak Pressure (psig)
0.00	38.16
0.05	38.17

TABLE 6.2-22
(Sheet 1 of 1)
EFFECT OF DOWNCOMER SUBMERGENCE
ON PEAK DRYWELL PRESSURE - OLTP ANALYSIS

Recirculation Suction Line DER (Case C)

Downcomer Submergence (ft)	Peak Pressure (psig)
10	39.21
11	39.75

TABLE 6.2-23
(Sheet 1 of 1)
EFFECT OF SUPPRESSION CHAMBER VOLUME
ON PEAK DRYWELL PRESSURE - OLTP ANALYSIS

Recirculation Suction Line DER (Case C)

Suppression Chamber Air Volume (%)	Peak Pressure (psig)
90	42.86
100	39.75
110	37.07

TABLE 6.2-24
(Sheet 1 of 1)
EFFECT OF INITIAL HUMIDITY ON
PEAK DRYWELL PRESSURE - OLTP ANALYSIS

Recirculation Suction Line DER (Case C)

Relative Humidity (%)	Peak Pressure (psig)
40	39.75
100	37.32

TABLE 6.2-25
(Sheet 1 of 1)
EFFECT OF INITIAL DRYWELL TEMPERATURE ON
PEAK DRYWELL PRESSURE - OLTP ANALYSIS

Recirculation Suction Line DER (Case C)

Drywell Temperature (°F)	Peak Pressure (psig)
120	40.79
135	39.75
150	38.48

TABLE 6.2-26
(Sheet 1 of 1)
EFFECT OF INITIAL POOL AND SUPPRESSION CHAMBER TEMPERATURE ON PEAK DRYWELL PRESSURE - OLTP ANALYSIS

Recirculation Suction Line DER (Case C)

Pool Temperature (°F)	Peak Pressure (psig)
80	38.58
90	39.75
100	40.95

TABLE 6.2-27
(Sheet 1 of 1)
EFFECT OF BREAK AREA ON PEAK DRYWELL PRESSURE -
OLTP ANALYSIS

Recirculation Suction Line DER (Case C)

Break Area (ft^2)	Peak Pressure (psig)
2.65	39.59
2.95	39.75
3.25	39.83

TABLE 6.2-27A
(Sheet 1 of 1)
STEAM BYPASS ANALYSIS HEAT BALANCE SUMMARY
CONTAINMENT HEAT REMOVAL SUMMARY

	0.30 ft^2 Steam Break with 0.05 ft^2 $A/\sqrt{K}$ Steam Bypass (1x10^6 Btu)				
Time (sec)	0.0	900.0	1800.0	21,003.0	36,000.0
Energy Removed					
Drywell heat sinks	0.0	42.0	48.4	51.7	85.5
Wetwell heat sinks	0.0	1.2	3.7	-0.2	6.4
Pool heat sinks	0.0	0.8	1.7	15.1	21.8
Drywell and wetwell spray heat exchanger	0.0	0.0	0.0	374.5	429.7

TABLE 6.2-28
(Sheet 1 of 1)
PRIMARY CONTAINMENT SUBCOMPARTMENT ANALYSIS SUMMARY

			Tables			Figures		
Subcompartment	Model	Design Basis Line Break	Nodal Description	Vent Path Description	Blowdown Data	Nodalization Diagram	Nodal Pressures	Nodal Pressure Differentials
Drywell head	2-node	RCIC head spray	6.2-29	6.2-30	6.2-31	6.2-31	6.2-31A	6.2-31B
Drywell head	2-node	Recirculation suction	6.2-32	6.2-33	6.2-34	6.2-32	6.2-33A	6.2-33B
RPV-BSW annulus	21-node	Feedwater	6.2-35	6.2-36	6.2-37	6.2-52	6.2-53	6.2-54
RPV-BSW annulus	37-node	Feedwater	6.2-38	6.2-39	6.2-37	6.2-55	6.2-56	6.2-57
RPV-BSW annulus	20-node	LPCI	6.2-40	6.2-41	6.2-42	6.2-58	6.2-59	6.2-60
RPV-BSW annulus	21-node	LPCS	6.2-43	6.2-43A	6.2-43B	6.2-61	6.2-62	6.2-62A
RPV-BSW annulus	20-node	Recirculation inlet	6.2-44	6.2-44A	6.2-44B	6.2-63	6.2-64	6.2-64A
RPV-BSW annulus	20-node	Recirculation suction	6.2-45	6.2-45A	6.2-45B	6.2-65	6.2-66	6.2-66A

TABLE 6.2-29
(Sheet 1 of 1)
SUBCOMPARTMENT NODAL DESCRIPTION

6-Inch RCIC Head Spray Line Break
Drywell Head Subcompartment

Volume No.	Volume (ft^3)	Initial Conditions: Temp. (°F)	Initial Conditions: Pressure (psia)	Initial Conditions: Humidity (%)	DBA Break Conditions: % Break in Vol.	DBA Break Conditions: Break Line	DBA Break Conditions: Break Area (ft^2)	DBA Break Conditions: Break Type	Calculated Peak Pressure Difference[(1)] (psid)	Design Peak Pressure Difference[(1)] (psid)	Design Margin[(2)] (%)
1	8,700	150	14.2	0	100	RCIC	0.181	DER	7.65	10.66	28.2
2	293,570	150	14.2	0	0				0.00	-	-

(1) Peak pressure difference [(Pl-P2) peak] is shown on Figure 6.2-31B.
(2) Design margin: 1-(calculated Δ peak/design Δ peak).

TABLE 6.2-30
(Sheet 1 of 1)
SUBCOMPARTMENT VENT PATH DESCRIPTION

6-Inch RCIC Head Spray Line Break
Drywell Head Subcompartment

Vent Path No.	From Volume Node No.	To Volume Node No.	Description of Vent Path Flow (Choked/Unchoked)	Vent Area (ft^2)	L/A (ft/ft^2)	Head Loss Coefficient				
						Friction	Turning	Expansion	Contraction	Total
1	1	2	Unchoked	2.405	0.6789	0.0017	1.281	0.9936	0.4975	2.77
2	1	2	Unchoked	1.536	0.7181	0.0027	1.245	0.9958	0.4984	2.74
3	1	2	Unchoked	2.582	0.6742	0.0060	1.330	0.9931	0.4973	2.63
4	1	2	Unchoked	1.536	0.7181	0.0027	1.245	0.9958	0.4984	2.74
5	1	2	Unchoked	2.405	0.6789	0.0017	1.281	0.9936	0.4975	2.77
6	1	2	Unchoked	1.536	0.7181	0.0027	1.245	0.9958	0.4984	2.74

TABLE 6.2-31
(Sheet 1 of 1)
BLOWDOWN DATA

6-Inch RCIC Head Spray Line Break
Drywell Head Subcompartment

Time (sec)	Blowdown Mass Flow Rate (lbm/sec)	Blowdown Enthalpy (Btu/lbm)	Blowdown Energy Release Rate (Btu/sec)
0.0	407.3	1,191.0	4.85×10^{5}
2.5	407.3	1,191.0	4.85×10^{5}

NOTE: For this case, the mass and energy release is assumed constant until after the occurrence of the peak pressure difference between Nodes 1 and 2.

TABLE 6.2-32
(Sheet 1 of 1)
SUBCOMPARTMENT NODAL DESCRIPTION

24-Inch Recirculation Suction Line Break
Drywell Head Subcompartment

Volume No.	Volume (ft^3)	Initial Conditions			DBA Break Conditions				Calculated Peak Pressure Difference[1] (psid)	Design Peak Pressure Difference[1] (psid)	Design Margin[2] (%)
		Temp. (°F)	Pressure (psia)	Humidity (%)	% Break in Vol.	Break Line	Area (ft^2)	Break Type			
1	8,700	150	14.2	0	0				0.006.36	-	-
2	293,570	150	14.2	0	100	Recirc. Suction	2.60	DER		7.22	11.9

(1) Peak pressure difference [(P2-P1) peak] is shown on Figure 6.2-33B.
(2) Design margin: 1-(calculated Δ peak/design Δ peak).

TABLE 6.2-33
(Sheet 1 of 1)
SUBCOMPARTMENT VENT PATH DESCRIPTION

24-Inch Recirculation Suction Line Break
Drywell Head Subcompartment

Vent Path No.	From Volume Node No.	To Volume Node No.	Description of Vent Path Flow (Choked/Unchoked)	Vent Area (ft^2)	L/A (ft/ft^2)	Head Loss Coefficient				
						Friction	Turning	Expansion	Contraction	Total
1	2	1	Unchoked	2.405	0.5649	0.0017	1.278	0.9871	0.4984	2.76
2	2	1	Unchoked	1.536	0.6041	0.0027	1.244	0.9918	0.4989	2.74
3	2	1	Unchoked	2.582	0.5602	0.0060	1.130	0.9862	0.4983	2.62
4	2	1	Unchoked	1.536	0.6041	0.0027	1.244	0.9918	0.4989	2.74
5	2	1	Unchoked	2.405	0.5649	0.0017	1.278	0.9871	0.4984	2.76
6	2	1	Unchoked	1.536	0.6041	0.0027	1.244	0.9918	0.4989	2.74

TABLE 6.2-34
(Sheet 1 of 1)
BLOWDOWN DATA

24-Inch Recirculation Suction Line Break
Drywell Head Subcompartment

Time (sec)	Blowdown Mass Flow Rate (lbm/sec)	Blowdown Enthalpy (Btu/lbm)	Blowdown Energy Release Rate (Btu/sec)
0.00	35,160	531.7	1.869×10^{7}
2.50	35,160	531.7	1.869×10^{7}

TABLE 6.2-35
(Sheet 1 of 1)
SUBCOMPARTMENT NODAL DESCRIPTION

12-Inch Feedwater Line Break
21 Node Model
RPV-BSW Annulus

Volume No.	Volume (ft^3)	Initial Conditions			DBA Break Conditions				Calculated Peak Pressure Difference[1] (psid)	Design Peak Pressure Difference[1] (psid)	Design Margin[2] (%)
		Temp. (°F)	Pressure (psia)	Humidity (%)	% Break in Vol.	Break Line	Area (ft^2)	Break Type			
1	273.8	150.00	14.20	20.00	0				14.06	70.60	80.1
2	273.8	150.00	14.20	20.00	0				14.05	70.60	80.1
3	276.3	150.00	14.20	20.00	0				14.03	70.6070.60	80.1
4	275.8	150.00	14.2014.	20.00	0				14.00	70.60	80.2
5	276.3	150.00	20	20.00	0				13.98	70.6070.60	80.2
6	276.3	150.00	14.20	20.00	0				13.97	70.60	80.2
7	264.0	150.00	14.20	20.00	0				14.12	70.6070.60	80.0
8	264.0	150.00	14.20	20.00	0				14.05	70.60	80.1
9	264.7	150.00	14.20	20.00	0				13.96	70.6070.60	80.2
10	266.6	150.00	14.2014.	20.00	0				13.89	70.60	80.3
11	266.6	150.00	20	20.00	0				13.87	70.6070.60	80.4
12	264.7	150.00	14.2014.	20.00	0				13.87	70.6070.6070.	80.4
13	118.7	150.00	20	20.00	0				16.22	60	77.0
14	251.7	150.00	14.20	20.00	0				14.06		80.1
15	248.5	150.00	14.20	20.00	0				13.86		80.4
16	250.4	150.00	14.2014.	20.00	0				13.76	70.60	80.5
17	251.8	150.00	20	20.00	0				13.72	-	80.6
18	249.9	150.00	14.2014.	20.00	0				13.71		80.6
19	98.9	150.00	20	20.00	100	Feedwater	(See Table 6.2-37)	DER	28.62		59.5
20	34.2	150.00	14.20	20.00	0				27.02		61.7
21	144750.0	150.00	14.20	20.00	0				0.00		-

(1) Peak pressure difference [(P_i-P_{21}) peak] is shown on Figure 6.2-54.
(2) Design margin: 1-(calculated Δ peak/design Δ peak).

TABLE 6.2-36
(Sheet 1 of 1)
SUBCOMPARTMENT VENT PATH DESCRIPTION

12-Inch Feedwater Line Break
21-Node Model RPV-BSW Annulus

Vent Path No.	From Volume Node No.	To Volume Node No.	Description of Vent Path Flow (Choked/Unchoked)	Vent Area (ft^2)	L/A (ft/ft^2)	Head Loss Coefficient				
						Friction	Turning	Expansion	Contraction	Total
1	1	2	Unchoked	39.25	0.166	0.0275	0.068	0.006	0.037	0.1385
2	1	7	Unchoked	11.96	1.359	0.074	-	0.084	0.145	0.3030
3	2	3	Unchoked	40.74	0.160	0.0275	0.068	0.002	0.020	0.1175
4	2	8	Unchoked	11.96	1.359	0.074	-	0.084	0.145	0.3030
5	3	4	Unchoked	40.74	0.160	0.0275	0.068	0.002	0.020	0.1175
6	3	9	Unchoked	13.45	1.208	0.074	-	0.040	0.100	0.2140
7	4	5	Unchoked	39.55	0.165	0.0275	0.068	0.005	0.034	0.1345
8	4	10	Unchoked	12.26	1.325	0.074	-	0.076	0.134	0.2840
9	5	6	Unchoked	40.74	0.160	0.0275	0.068	0.002	0.020	0.1175
10	5	11	Unchoked	13.45	1.208	0.074	-	0.040	0.100	0.2140
11	6	12	Unchoked	13.45	1.208	0.074	-	0.040	0.100	0.2140
12	7	8	Unchoked	37.76	0.173	0.0255	0.068	0.006	0.039	0.1385
13	7	13	Choked	16.20	0.710	0.071	-	0.001	0.019	0.0910
14	8	9	Unchoked	38.62	0.169	0.0255	0.068	0.003	0.028	0.1245
15	8	14	Unchoked	16.20	0.958	0.071	-	0.001	0.019	0.0910
16	9	10	Unchoked	37.56	0.174	0.0255	0.068	0.005	0.041	0.1395
17	9	15	Unchoked	13.45	1.154	0.071	-	0.040	0.100	0.2110
18	10	11	Unchoked	39.25	0.166	0.0255	0.068	0.002	0.021	0.1165
19	10	16	Unchoked	16.83	0.922	0.071	-	-	-	0.0710
20	11	12	Unchoked	39.25	0.166	0.0255	0.068	0.002	0.021	0.1165
21	11	17	Unchoked	16.83	0.922	0.071	-	-	-	0.0710
22	12	18	Unchoked	13.45	1.154	0.071	-	0.040	0.100	0.2110
23	13	14	Unchoked	17.18	0.380	0.0255	0.048	0.309	0.026	0.4085
24	13	19	Choked	16.83	0.388	0.031	-	-	-	0.0310
25	19	14	Choked	14.76	0.663	0.0255	0.048	0.383	0.020	0.4765
26	19	20	Choked	16.20	0.248	0.027	-	0.001	0.019	0.0470
27	20	14	Choked	5.19	2.081	0.0255	0.124	0.750	-	0.8995
28	20	21	Choked	9.00	0.225	0.009	-	0.876	0.233	1.1175
29	14	15	Choked	35.31	0.185	0.0255	0.066	0.008	0.044	0.1435
30	14	21	Choked	9.00	1.676	0.067	-	0.876	0.233	1.1760
31	15	16	Unchoked	34.15	0.191	0.0255	0.066	0.014	0.059	0.1645
32	15	21	Choked	7.56	1.995	0.067	-	0.895	0.275	1.2370
33	16	17	Unchoked	37.00	0.176	0.0255	0.066	0.002	0.022	0.1155
34	16	21	Choked	7.56	1.995	0.067	-	0.895	0.275	1.2370
35	17	18	Unchoked	38.69	0.169	0.0255	0.066	-	-	0.0915
36	17	21	Choked	9.00	1.676	0.067	-	0.876	0.233	1.1760
37	18	21	Choked	9.00	1.676	0.067	-	0.876	0.233	1.1760

TABLE 6.2-37
(Sheet 1 of 1)
BLOWDOWN DATA

12-Inch Feedwater Line Break
21-Node and 37-Node Models
RPV-BSW Annulus

Time (sec)	Blowdown Mass Flow Rate (lbm/sec)	Blowdown Enthalpy (Btu/lbm)	Blowdown Energy Release Rate (Btu/sec)*	Total Effective Break Area (ft^2)
0.00000	6,650.5	402.0	2.674×10^6	1.412
0.01530	6,650.5	402.0	2.674×10^6	1.412
0.01531	4,718.0	402.0	1.897×10^6	1.030
0.02066	4,718.0	402.0	1.897×10^6	1.030
0.02067	7,981.0	402.0	3.209×10^6	1.030
1.00000	7,981.0	402.0	3.209×10^6	1.030

* Due to symmetry in the nodalization, the tabulated blowdown represents one half of the total blowdown.

TABLE 6.2-38
(Sheet 1 of 2)
SUBCOMPARTMENT NODAL DESCRIPTION

12-Inch Feedwater Line Break
37 Node Model
RPV-BSW Annulus

Volume No.	Volume (ft^3)	Initial Conditions			DBA Break Conditions				Calculated Peak Pressure Difference(1) (psid)	Design Peak Pressure Difference(1) (psid)	Design Margin(2) (%)
		Temp. (°F)	Pressure (psia)	Humidity (%)	% Break in Vol.	Break Line	Area (ft^2)	Break Type			
1	261.65	150.00	14.20	20.00	0				12.78	70.60	81.9
2	261.65	150.00	14.20	20.00	0				12.77	70.60	81.9
3	276.05	150.00	14.20	20.00	0				12.75	70.60	81.9
4	19.22	150.00	14.20	20.00	0				12.68	70.60	82.0
5	118.56	150.00	14.20	20.00	0				12.73	70.60	82.0
6	19.22	150.00	14.20	20.00	0				12.68	70.60	82.0
7	118.56	150.00	14.20	20.00	0				12.72	70.60	82.0
8	276.05	150.00	14.20	20.00	0				12.70	70.60	82.0
9	276.05	150.00	14.20	20.00	0				12.68	70.60	82.0
10	89.38	150.00	14.20	20.00	0				12.87	70.60	81.8
11	177.28	150.00	14.20	20.00	0				12.87	70.60	81.8
12	177.28	150.00	14.20	20.00	0				12.83	70.60	81.8
13	89.38	150.00	14.20	20.00	0				12.81	70.60	81.9
14	132.23	150.00	14.20	20.00	0				12.77	70.60	81.9
15	132.23	150.00	14.20	20.00	0				12.72	70.60	82.0
16	414.35	150.00	14.20	20.00	0				12.56	70.60	82.2
17	434.68	150.00	14.20	20.00	0				12.53	70.60	82.3
18	132.23	150.00	14.20	20.00	0				12.54	70.60	82.2
19	132.23	150.00	14.20	20.00	0				12.54	70.60	82.2
20	38.51	150.00	14.20	20.00	0				14.48	70.60	79.5
21	76.98	150.00	14.20	20.00	0				13.03	70.60	81.5
22	76.98	150.00	14.20	20.00	0				12.86	70.60	81.8
23	38.51	150.00	14.20	20.00	0				12.81	70.60	81.9
24	72.84	150.00	14.20	20.00	0				12.73	70.60	82.0
25	73.13	150.00	14.20	20.00	0				12.67	70.60	82.1
26	83.64	150.00	14.20	20.00	0				12.46	70.60	82.4
27	82.66	150.00	14.20	20.00	0				12.46	70.60	82.4
28	34.19	150.00	14.20	20.00	0				34.58	70.60	51.0
29	62.00	150.00	14.20	20.00	100	Feedwater	(See Table 6.2-37)	DER	34.83	70.60	50.7
30	17.08	150.00	14.20	20.00	0				19.46	70.60	72.4
31	31.89	150.00	14.20	20.00	0				23.02	70.60	67.4
32	83.63	150.00	14.20	20.00	0				12.82	70.60	81.8
33	61.73	150.00	14.20	20.00	0				12.86	70.60	81.8

TABLE 6.2-38
(Sheet 2 of 2)

Volume No.	Volume (ft^3)	Initial Conditions: Temp. (°F)	Initial Conditions: Pressure (psia)	Initial Conditions: Humidity (%)	DBA Break Conditions: % Break in Vol.	DBA Break Conditions: Break Line	DBA Break Conditions: Area (ft^2)	DBA Break Conditions: Break Type	Calculated Peak Pressure Difference[(1)] (psid)	Design Peak Pressure Difference[(1)] (psid)	Design Margin[(2)] (%)
34	102.56	150.00	14.20	20.00	0				12.25	70.60	82.6
35	102.56	150.00	14.20	20.00	0				12.2012.16	70.60	82.7
36	167.26	150.00	14.20	20.00	0				0.00	70.60	82.8
37	144750.00	150.00	14.20	20.00	0					-	-

(1) Peak pressure difference [(Pi-P_{37}) peak] is shown on Figure 6.2-51.
(2) Design margin: 1-(calculated Δ peak/design Δ peak).

TABLE 6.2-39
(Sheet 1 of 6)
SUBCOMPARTMENT VENT PATH DESCRIPTION

12-Inch Feedwater Line Break
37 Node Model
RPV-BSW Annulus

Vent Path No.	From Volume Node No.	To Volume Node No.	Description of Vent Path Flow (Choked/Unchoked)	Vent Area (ft^2)	L/A (ft^{-1})	Head Loss Coefficient				
						Friction	Turning	Expansion	Contraction	Total
1	1 2	2 1	Unchoked Unchoked	37.378 37.378	0.174 0.174	0.0276 0.0276	0.0647 0.0647	0.0061 0.0061	0.0392 0.0392	0.138 0.138
2	1 10	10 1	Unchoked Unchoked	3.919 3.919	4.076 4.076	0.0712 0.0712	- -	0.0906 0.5883	0.3835 0.1505	0.545 0.810
3	1 11	11 1	Unchoked Unchoked	8.036 8.036	1.988 1.988	0.0712 0.0712	- -	0.0802 0.2727	0.2611 0.1416	0.413 0.486
4	2 3	3 2	Unchoked Unchoked	38.867 38.867	0.168 0.168	0.0276 0.0276	0.0647 0.0647	0.0068 0.0017	0.0208 0.0413	0.120 0.135
5	2 12	12 2	Unchoked Unchoked	8.036 8.036	1.988 1.988	0.0712 0.0712	- -	0.0802 0.2727	0.2611 0.1416	0.413 0.486
6	2 13	13 2	Unchoked Unchoked	3.919 3.919	4.076 4.076	0.0712 0.0712	- -	0.0906 0.5883	0.3835 0.1505	0.545 0.810
7	3 4	4 3	Unchoked Unchoked	4.508 4.508	1.084 1.084	0.0191 0.0191	0.0971 0.0971	0.0742 0.7985	0.4468 0.1362	0.637 1.051
8	3 5	5 3	Unchoked Unchoked	36.174 36.174	0.135 0.135	0.0209 0.0209	0.0458 0.0458	- -	- -	0.067 0.067
9	3 14	14 3	Unchoked Unchoked	6.722 6.722	2.417 2.417	0.0724 0.0724	- -	0.0402 0.3604	0.3002 0.1003	0.413 0.533
10	3 15	15 3	Unchoked Unchoked	6.722 6.722	2.417 2.417	0.0724 0.0724	- -	0.0402 0.3604	0.3002 0.1003	0.413 0.533
11	4 5	5 4	Unchoked Unchoked	7.813 7.813	1.058 1.058	0.0368 0.0368	- -	0.0050 0.0050	0.0354 0.0354	0.077 0.077
12	4 6	6 4	Unchoked Unchoked	5.599 5.599	0.582 0.582	0.0127 0.0127	0.0657 0.0657	0.0093 0.0093	0.0481 0.0481	0.136 0.136

TABLE 6.2-39
(Sheet 2 of 6)

Vent Path No.	From Volume Node No.	To Volume Node No.	Description of Vent Path Flow (Choked/Unchoked)	Vent Area (ft^2)	L/A (ft^{-1})	Head Loss Coefficient				
						Friction	Turning	Expansion	Contraction	Total
13	4 16	16 4	Unchoked Unchoked	6.722 6.722	2.028 2.028	0.0608 0.0608	- -	0.3604 0.0402	0.1003 0.3002	0.522 0.401
14	5 7	7 5	Unchoked Unchoked	35.578 35.578	0.092 0.092	0.0139 0.0139	0.0309 0.0309	0.0003 0.0003	0.0082 0.0082	0.053 0.053
15	6 7	7 6	Unchoked Unchoked	7.813 7.813	1.058 1.058	0.0368 0.0368	- -	0.0050 0.0050	0.0354 0.0354	0.077 0.077
16	6 8	8 6	Unchoked Unchoked	4.508 4.508	1.084 1.084	0.0191 0.0191	0.0971 0.0971	0.7985 0.0742	0.1382 0.4468	1.051 0.637
17	6 16	16 6	Unchoked Unchoked	6.722 6.722	2.028 2.028	0.0608 0.0608	- -	0.3604 0.0402	0.1003 0.3002	0.522 0.401
18	7 8	8 7	Unchoked Unchoked	36.174 36.174	0.135 0.135	0.0209 0.0209	0.0458 0.0458	- -	- -	0.067 0.067
19	8 9	9 8	Unchoked Unchoked	40.682 40.682	0.160 0.160	0.0275 0.0275	0.0659 0.0659	0.0016 0.0016	0.0199 0.0199	0.115 0.115
20	8 17	17 8	Unchoked Unchoked	13.444 13.444	1.583 1.583	0.0948 0.0948	- -	0.0402 0.0402	0.1003 0.1003	0.235 0.235
21	9 18	18 9	Unchoked Unchoked	6.722 6.722	2.417 2.417	0.0724 0.0724	- -	0.0402 0.3604	0.3002 0.1003	0.413 0.533
22	9 19	19 9	Unchoked Unchoked	6.722 6.722	2.417 2.417	0.0724 0.0724	- -	0.0402 0.3604	0.3002 0.1003	0.413 0.533
23	10 11	11 10	Unchoked Unchoked	41.014 41.014	0.079 0.079	0.0127 0.0127	0.0351 0.0351	0.0001 0.0001	0.0038 0.0038	0.052 0.052
24	10 20	20 10	Unchoked Choked	5.292 5.292	2.173 2.173	0.0512 0.0512	- -	0.0031 0.0031	0.0280 0.0280	0.082 0.082
25	11 12	12 11	Unchoked Unchoked	38.152 38.152	0.114 0.114	0.0170 0.0170	0.0464 0.0464	0.0059 0.0059	0.0384 0.0384	0.108 0.108
26	11 21	21 11	Unchoked Choked	10.898 10.898	1.055 1.055	0.0512 0.0512	- -	0.0008 0.0008	0.0140 0.0140	0.066 0.066

TABLE 6.2-39
(Sheet 3 of 6)

Vent Path No.	From Volume Node No.	To Volume Node No.	Description of Vent Path Flow (Choked/Unchoked)	Vent Area (ft^2)	L/A (ft^{-1})	Head Loss Coefficient				
						Friction	Turning	Expansion	Contraction	Total
27	12 13	13 12	Unchoked Unchoked	41.014 41.014	0.079 0.079	0.0127 0.0127	0.0351 0.0351	0.0001 0.0001	0.0038 0.0038	0.052 0.052
28	12 22	22 12	Unchoked Unchoked	10.898 10.898	1.055 1.055	0.0512 0.0512	- -	0.0008 0.0008	0.0140 0.0140	0.066 0.066
29	13 14	14 13	Unchoked Unchoked	39.641 39.641	0.068 0.068	0.0106 0.0106	0.0294 0.0294	0.0009 0.0017	0.0204 0.0154	0.061 0.057
30	13 23	23 13	Unchoked Unchoked	5.292 5.292	2.173 2.173	0.0512 0.0512	- -	0.0031 0.0031	0.0280 0.0280	0.082 0.082
31	14 15	15 14	Unchoked Unchoked	39.213 39.213	0.083 0.083	0.0127 0.0127	0.0350 0.0350	0.0017 0.0017	0.0206 0.0206	0.070 0.070
32	14 24	24 14	Unchoked Unchoked	6.722 6.722	1.849 1.849	0.0554 0.0554	- -	0.0402 0.0402	0.1003 0.1003	0.196 0.196
33	15 16	16 15	Unchoked Unchoked	39.213 39.213	0.125 0.125	0.0191 0.0191	0.0517 0.0517	0.1478 0.0017	0.0206 0.1922	0.239 0.265
34	15 25	25 15	Unchoked Unchoked	6.722 6.722	1.849 1.849	0.0554 0.0554	- -	0.0402 0.0402	0.1003 0.1003	0.196 0.196
35	16 17	17 16	Unchoked Unchoked	60.327 60.327	0.108 0.108	0.0255 0.0255	0.0780 0.0780	0.0091 0.0028	0.0265 0.0477	0.139 0.154
36	16 35	35 16	Unchoked Unchoked	13.708 13.708	1.133 1.133	0.0692 0.0692	- -	0.0342 0.0342	0.0925 0.0925	0.196 0.196
37	17 18	18 17	Unchoked Unchoked	39.213 39.213	0.125 0.125	0.0191 0.0191	0.0517 0.0517	0.0017 0.1698	0.2060 0.0206	0.279 0.261
38	17 26	26 17	Unchoked Unchoked	25.791 25.791	0.189 0.189	0.0191 0.0191	0.0397 0.0397	- -	- -	0.059 0.059
39	17 36	36 17	Unchoked Unchoked	15.131 15.131	1.026 1.026	0.0692 0.0692	- -	0.3027 0.0101	0.0502 0.2751	0.422 0.354
40	18 19	19 18	Unchoked Unchoked	39.213 39.213	0.083 0.083	0.0127 0.0127	0.0350 0.0350	0.0017 0.0017	0.0206 0.0206	0.070 0.070

TABLE 6.2-39
(Sheet 4 of 6)

Vent Path No.	From Volume Node No.	To Volume Node No.	Description of Vent Path Flow (Choked/Unchoked)	Vent Area (ft^2)	L/A (ft^{-1})	Head Loss Coefficient				
						Friction	Turning	Expansion	Contraction	Total
41	18	26	Unchoked	6.722	1.936	0.0580	-	0.0402	0.1003	0.199
	26	18	Unchoked	6.722	1.936	0.0580	-	0.0402	0.1003	0.199
42	19	27	Unchoked	6.722	1.936	0.0580	-	0.0402	0.1003	0.199
	27	19	Unchoked	6.722	1.936	0.0580	-	0.0402	0.1003	0.199
43	20	21	Unchoked	17.307	0.188	0.0127	0.0261	0.0003	0.0089	0.048
	21	20	Unchoked	17.307	0.188	0.0127	0.0261	0.0003	0.0089	0.048
44	20	29	Unchoked	5.602	1.123	0.0280	-	-	-	0.028
	29	20	Choked	5.602	1.123	0.0280	-	-	-	0.028
45	21	22	Choked	17.158	0.253	0.0170	0.0345	0.0007	0.0131	0.065
	22	21	Unchoked	17.158	0.253	0.0170	0.0345	0.0007	0.0131	0.065
46	21	29	Unchoked	5.606	1.122	0.0280	-	-	-	0.028
	29	21	Choked	5.606	1.122	0.0280	-	-	-	0.028
47	21	31	Unchoked	5.143	1.223	0.0280	-	0.0068	0.2706	0.305
	31	21	Choked	5.143	1.223	0.0280	-	0.2930	0.0413	0.362
48	22	23	Unchoked	17.307	0.188	0.0127	0.0261	0.0003	0.0089	0.048
	23	22	Unchoked	17.307	0.188	0.0127	0.0261	0.0003	0.0089	0.048
49	22	33	Unchoked	10.749	0.493	0.0236	-	0.1302	0.0206	0.174
	33	22	Unchoked	10.749	0.493	0.0236	-	0.0017	0.1804	0.206
50	23	14	Choked	1.387	1.957	0.0106	0.0974	-	-	0.108
	14	23	Unchoked	1.387	1.957	0.0106	0.0974	-	-	0.108
51	23	24	Choked	16.234	0.167	0.0106	0.0224	-	-	0.033
	24	23	Unchoked	16.234	0.167	0.0106	0.0224	-	-	0.033
52	23	33	Unchoked	5.606	0.946	0.0236	-	-	-	0.024
	33	23	Choked	5.606	0.946	0.0236	-	-	-	0.024
53	24	25	Unchoked	21.113	0.154	0.0127	0.0260	0.0055	0.0370	0.081
	25	24	Unchoked	21.113	0.154	0.0127	0.0260	0.0055	0.0370	0.081
54	24	34	Unchoked	6.722	1.123	0.0336	-	0.3604	0.1003	0.494
	34	24	Unchoked	6.722	1.123	0.0336	-	0.0402	0.3002	0.374

TABLE 6.2-39
(Sheet 5 of 6)

Vent Path No.	From Volume Node No.	To Volume Node No.	Description of Vent Path Flow (Choked/Unchoked)	Vent Area (ft^2)	L/A (ft^{-1})	Head Loss Coefficient				
						Friction	Turning	Expansion	Contraction	Total
55	25 16	16 25	Unchoked Unchoked	21.377 21.377	0.229 0.229	0.0191 0.0191	0.0384 0.0384	0.4414 0.0039	0.0312 0.3322	0.530 0.394
56	25 34	34 25	Unchoked Unchoked	6.986 6.986	1.080 1.080	0.0336 0.0336	- -	0.3418 0.0286	0.0846 0.2923	0.460 0.355
57	26 27	27 26	Unchoked Unchoked	24.104 24.104	0.135 0.135	0.0127 0.0127	0.0269 0.0269	0.0043 0.0043	0.0327 0.0327	0.077 0.077
58	26 36	36 26	Choked Unchoked	8.409 8.409	0.897 0.897	0.0336 0.0336	- -	- -	- -	0.034 0.034
59	27 36	36 27	Unchoked Unchoked	6.722 6.722	1.123 1.123	0.0336 0.0336	- -	0.6402 0.0402	0.1003 0.4001	0.774 0.474
60	28 29	29 28	Unchoked Choked	10.898 10.898	0.402 0.402	0.0195 0.0195	- -	0.0008 0.0008	0.0140 0.0140	0.034 0.034
61	28 30	30 28	Choked Unchoked	7.508 7.508	0.434 0.434	0.0127 0.0127	0.0551 0.0551	0.0016 0.0016	0.0201 0.0201	0.090 0.090
62	28 37	37 28	Choked Unchoked	3.372 3.372	0.905 0.905	0.0068 0.0068	- -	0.9289 0.4890	0.3496 0.4819	1.285 0.978
63	29 31	31 29	Choked Unchoked	14.316 14.316	0.228 0.228	0.0127 0.0127	0.0276 0.0276	0.0005 0.0005	0.0107 0.0107	0.052 0.052
64	30 31	31 30	Unchoked Unchoked	5.292 5.292	0.828 0.828	0.0195 0.0195	- -	0.0031 0.0031	0.0280 0.0280	0.051 0.051
65	30 32	32 30	Choked Unchoked	7.822 7.822	0.555 0.555	0.0170 0.0170	0.0729 0.0729	- -	- -	0.090 0.090
66	30 37	37 30	Choked Unchoked	5.606 5.606	0.544 0.544	0.0068 0.0068	- -	0.7739 -	- 0.4398	0.781 0.447
67	31 32	32 31	Choked Unchoked	5.072 5.072	0.856 0.856	0.0170 0.0170	0.0981 0.0981	- -	- -	0.115 0.115
68	31 33	33 31	Choked Unchoked	9.094 9.094	0.478 0.478	0.0170 0.0170	0.0588 0.0588	0.0023 0.1432	0.1892 0.0242	0.267 0.243

TABLE 6.2-39
(Sheet 6 of 6)

Vent Path No.	From Volume Node No.	To Volume Node No.	Description of Vent Path Flow (Choked/Unchoked)	Vent Area (ft^2)	L/A (ft^{-1})	Head Loss Coefficient				
						Friction	Turning	Expansion	Contraction	Total
69	32	33	Unchoked	15.131	0.289	0.0195	-	0.0101	0.0502	0.080
	33	32	Unchoked	15.131	0.289	0.0195	-	0.0101	0.0502	0.080
70	32	34	Choked	11.207	0.581	0.0255	0.0567	0.0867	0.0654	0.234
	34	32	Unchoked	11.207	0.581	0.0255	0.0567	0.0171	0.1472	0.247
71	32	37	Choked	8.978	0.560	0.0112	-	0.8757	0.2331	1.120
	37	32	Unchoked	8.978	0.560	0.0112	-	0.2173	0.4679	0.696
72	33	24	Unchoked	7.870	0.621	0.0191	0.0658	0.4288	0.0883	0.602
	24	33	Unchoked	7.870	0.621	0.0191	0.0658	0.0312	0.3274	0.444
73	34	35	Unchoked	14.462	0.451	0.0255	0.0520	0.0080	0.0448	0.130
	35	34	Unchoked	14.462	0.451	0.0255	0.0520	0.0080	0.0448	0.130
74	34	37	Choked	8.978	0.690	0.0138	-	0.8757	0.2331	1.123
	37	34	Unchoked	8.978	0.690	0.0138	-	0.2173	0.4679	0.699
75	35	36	Unchoked	11.207	0.872	0.0382	0.0801	0.0171	0.1472	0.283
	36	35	Unchoked	11.207	0.872	0.0382	0.0801	0.0867	0.0654	0.270
76	35	37	Choked	8.978	0.690	0.0138	-	0.8757	0.2331	1.123
	37	35	Unchoked	8.978	0.690	0.0138	-	0.2173	0.4679	0.699
77	36	37	Choked	17.956	0.280	0.0112	-	0.8757	0.2331	1.120
	37	36	Unchoked	17.956	0.280	0.0112	-	0.2173	0.4679	0.696

TABLE 6.2-40
(Sheet 1 of 1)
SUBCOMPARTMENT NODAL DESCRIPTION

12-Inch Low-Pressure Coolant Injection Line Break
RPV-BSW Annulus

Volume No.	Volume (ft^3)	Initial Conditions Temp. (°F)	Initial Conditions Pressure (psia)	Initial Conditions Humidity (%)	DBA Break Conditions % Break in Vol.	DBA Break Conditions Break Line	DBA Break Conditions Break Area (ft^2)	DBA Break Conditions Break Type	Calculated Peak Pressure Difference[1] (psid)	Design Peak Pressure Difference[1] (psid)	Design Margin[2] (%)
1	273.71	550	14.2	0	0				5.93	70.60	91.6
2	254.40	550	14.2	0	0				5.72	70.60	91.9
3	273.71	550	14.2	0	0				5.35	70.60	92.4
4	273.12	550	14.2	0	0				4.73	70.60	93.3
5	273.12	550	14.2	0	0				6.25	70.60	91.1
6	273.71	500[3]	14.2	0	0				6.13	70.60	91.3
7	230.79	550	14.2	0	0				7.06	70.60	90.0
8	258.18	550	14.2	0	0				6.60	70.60	90.7
9	262.85	550	14.2	0	0				7.38	70.60	89.5
10	263.01	550	14.2	0	0				4.95	70.60	93.0
11	264.19	550	14.2	0	0				4.13	70.60	94.2
12	264.19	550	14.2	0	0				3.89	70.60	94.5
13	213.81	550	14.2	0	0				9.18	70.60	87.0
14	251.78	550	14.2	0	0				6.17	70.60	91.3
15	246.02	550	14.2	0	0				7.94	70.60	88.8
16	250.76	550	14.2	0	0				5.80	70.60	91.8
17	247.21	550	14.2	0	0				4.92	70.60	93.0
18	248.58	550	14.2	0	0				-4.01	70.60	94.3
19	137621.0	150	14.2	0	0				0.00	-	-
20	64.28	550	14.2	0	100	LPCI	(See Table 6.2-42)	DER	32.46	70.60	54.0

(1) Peak pressure difference [(Pi-P19) peak] is shown on Figure 6.2-60.
(2) Design margin: 1-absolute value of (calculated Δ peak/design Δ peak).
(3) Initial temperature of Node 6 set at 500°F for complete execution of the computer program.

TABLE 6.2-41
(Sheet 1 of 1)
SUBCOMPARTMENT VENT PATH DESCRIPTION

12-Inch Low-Pressure Coolant Injection Line Break
RPV-BSW Annulus

Vent Path No.	From Volume Node No.	To Volume Node No.	Description of Vent Path Flow (Choked/Unchoked)	Vent Area (ft^2)	L/A (ft/ft^2)	Head Loss Coefficient				
						Friction	Turning	Expansion	Contraction	Total
1	1	2	Unchoked	40.489	0.161	0.027	0.066	0.002	0.022	0.117
2	1	7	Unchoked	12.980	1.174	0.068	-	0.050	0.112	0.230
3	2	3	Unchoked	35.040	0.186	0.027	0.066	0.030	0.086	0.209
4	2	8	Unchoked	5.811	2.796	0.073	-	0.425	0.326	0.824
5	3	4	Unchoked	40.489	0.161	0.027	0.066	0.002	0.022	0.117
6	3	9	Unchoked	12.980	1.252	0.073	-	0.050	0.112	0.235
7	4	5	Unchoked	39.176	0.166	0.027	0.066	0.006	0.038	0.137
8	4	10	Unchoked	12.324	1.318	0.073	-	0.069	0.131	0.273
9	5	6	Unchoked	40.489	0.161	0.027	0.066	0.002	0.022	0.117
10	5	11	Unchoked	12.324	1.318	0.073	-	0.069	0.131	0.273
11	6	12	Unchoked	12.980	1.252	0.073	-	0.050	0.112	0.235
12	7	8	Unchoked	33.905	0.192	0.026	0.062	0.029	0.026	0.143
13	7	20	Choked	16.009	0.561	0.040	-	0.002	0.021	0.063
14	8	9	Unchoked	36.615	0.178	0.026	0.068	0.011	0.052	0.157
15	8	14	Unchoked	16.702	0.929	0.070	-	-	-	0.070
16	9	10	Unchoked	37.168	0.175	0.026	0.068	0.008	0.046	0.148
17	9	15	Unchoked	14.148	1.097	0.070	-	0.023	0.076	0.169
18	10	11	Unchoked	39.029	0.167	0.026	0.068	0.002	0.023	0.119
19	10	16	Unchoked	14.841	1.046	0.070	-	0.012	0.056	0.138
20	11	12	Unchoked	39.029	0.167	0.026	0.068	0.002	0.023	0.119
21	11	17	Unchoked	16.702	0.929	0.070	-	-	-	0.070
22	12	18	Unchoked	16.702	0.929	0.070	-	-	-	0.070
23	13	14	Unchoked	29.109	0.224	0.026	0.060	0.061	0.066	0.213
24	13	19	Unchoked	4.347	3.205	0.062	-	0.937	0.370	1.369
25	14	15	Unchoked	38.648	0.169	0.026	0.066	-	-	0.092
26	14	19	Unchoked	16.009	0.933	0.067	-	0.778	0.021	0.866
27	15	16	Unchoked	33.065	0.197	0.026	0.066	0.021	0.072	0.185
28	15	19	Unchoked	4.347	3.435	0.067	-	0.937	0.370	1.374
29	16	17	Unchoked	38.648	0.169	0.026	0.066	-	-	0.092
30	16	19	Unchoked	16.702	0.894	0.067	-	0.769	-	0.836
31	17	18	Unchoked	31.789	0.205	0.026	0.066	0.032	0.089	0.213
32	17	19	Unchoked	4.347	3.435	0.067	-	0.937	0.370	1.374
33	18	19	Unchoked	13.565	1.101	0.067	-	0.810	0.094	0.971
34	20	8	Choked	4.431	1.471	0.026	0.130	0.796	0.068	1.020
35	20	13	Choked	16.702	0.511	0.038	-	-	-	0.038
36	20	14	Choked	5.124	1.272	0.026	0.130	0.753	-	0.909

TABLE 6.2-42
(Sheet 1 of 1)
BLOWDOWN DATA

12-Inch Low-Pressure Coolant Injection Line Break
RPV-BSW Annulus

Time (sec)	Blowdown Mass Flow Rate (lbm/sec)*	Blowdown Enthalpy (Btu/lbm)	Blowdown Energy Release Rate (Btu/sec)*	Total Effective Break Area (ft^2)
0.000000	3,090.2	531.8	1.643×10^6	1.429
0.001790	3,090.2	531.8	1.643×10^6	1.429
0.001791	3,749.8	531.8	1.994×10^6	1.220
0.020000	3,749.8	531.8	1.994×10^6	1.220
0.020001	2,223.0	531.8	1.182×10^6	0.514
3.000000	2,223.0	531.8	1.182×10^6	0.514

* Due to symmetry in the nodalization, the tabulated blowdown represents one-half of the total blowdown.

TABLE 6.2-43
(Sheet 1 of 1)
SUBCOMPARTMENT NODAL DESCRIPTION

10-Inch Low-Pressure Core Spray Line Break
RPV-BSW Annulus

		Initial Conditions			DBA Break Conditions						
Volume No.	Volume (ft^3)	Temp. (°F)	Pressure (psia)	Humidity (%)	% Break in Vol.	Break Line	Break Area (ft^2)	Break Type	Calculated Peak Pressure Difference[1] (psid)	Design Peak Pressure Difference[1] (psid)	Design Margin[2] (%)
1	273.12	550	14.2	0	0				6.97	70.60	90.1
2	273.71	550	14.2	0	0				6.35	70.60	91.0
3	273.71	550	14.2	0	0				6.13	70.60	91.3
4	264.06	550	14.2	0	0				6.97	70.60	90.1
5	264.06	550	14.2	0	0				6.70	70.60	90.5
6	273.71	550	14.2	0	0				7.25	70.60	89.7
7	264.19	550	14.2	0	0				7.13	70.60	89.9
8	264.19	550	14.2	0	0				8.56	70.6070.60	87.9
9	261.83	550	14.2	0	0				9.94	70.60	85.9
10	261.02	550	14.2	0	0				7.58	70.60	89.3
11	261.02	550	14.2	0	0				7.82	70.60	88.9
12	261.83	550	14.2	0	0				6.58	70.60	90.7
13	70.06	550	14.2	0	0				20.66	70.60	70.7
14	249.57	550	14.2	0	0				11.04	70.60	84.4
15	247.21	550	14.2	0	0				11.11	70.6070.60	84.3
16	249.41	550	14.2	0	0				8.24	70.60	88.3
17	249.57	550	14.2	0	0				6.68		90.5
18	247.21	550	14.2	0	0				5.88	70.60	91.7
19	137621.0	150	14.2	0	0				0.00		
20	62.76	550	14.2	0	100	LPCS	(See Table 6.2-43B)	DER	26.80	 70.60	62.0
21	115.07	550	14.2	0	0				14.21		79.9

(1) Peak pressure difference [(Pi-P19) peak] is shown on Figure 6.2-62A.
(2) Design margin: 1-(calculated peak/design

TABLE 6.2-43A
(Sheet 1 of 1)
SUBCOMPARTMENT VENT PATH DESCRIPTION

10-Inch Low-Pressure Core Spray Line Break
RPV-BSW Annulus

Vent Path No.	From Volume Node No.	To Volume Node No.	Description of Vent Path Flow (Choked/Unchoked)	Vent Area (ft^2)	L/A (ft/ft^2)	Head Loss Coefficient				
						Friction	Turning	Expansion	Contraction	Total
1	1	2	Unchoked	39.176	0.166	0.027	0.066	0.006	0.038	0.137
2	1	7	Unchoked	11.667	1.392	0.073	-	0.091	0.151	0.315
3	2	3	Unchoked	40.489	0.161	0.027	0.066	0.002	0.022	0.117
4	2	8	Unchoked	12.980	1.252	0.073	-	0.050	0.112	0.235
5	3	4	Unchoked	40.489	0.161	0.027	0.066	0.002	0.022	0.117
6	3	9	Unchoked	12.980	1.252	0.073	-	0.050	0.112	0.235
7	4	5	Unchoked	37.129	0.176	0.027	0.066	0.015	0.062	0.170
8	4	10	Unchoked	9.620	1.689	0.073	-	0.180	0.212	0.465
9	5	6	Unchoked	40.489	0.161	0.027	0.066	0.002	0.022	0.117
10	5	11	Unchoked	9.620	1.689	0.073	-	0.180	0.212	0.465
11	6	12	Unchoked	12.980	1.252	0.073	-	0.050	0.112	0.235
12	7	8	Unchoked	39.029	0.167	0.026	0.068	0.002	0.023	0.119
13	7	21	Choked	16.702	0.684	0.051	-	-	-	0.051
14	8	9	Unchoked	39.029	0.167	0.026	0.068	0.002	0.023	0.119
15	8	14	Unchoked	16.702	0.929	0.070	-	-	-	0.070
16	9	10	Unchoked	36.475	0.179	0.026	0.068	0.012	0.054	0.160
17	9	15	Unchoked	12.980	1.196	0.070	-	0.050	0.112	0.232
18	10	11	Unchoked	35.669	0.183	0.026	0.068	0.016	0.064	0.174
19	10	16	Unchoked	16.009	0.970	0.070	-	0.002	0.021	0.093
20	11	12	Unchoked	36.475	0.179	0.026	0.068	0.012	0.054	0.160
21	11	17	Unchoked	16.009	0.970	0.070	-	0.002	0.021	0.093
22	12	18	Unchoked	12.980	1.196	0.070	-	0.050	0.112	0.232
23	13	14	Choked	10.748	0.606	0.026	0.080	0.521	-	0.627
24	13	19	Choked	10.524	0.902	0.042	-	0.850	0.185	1.077
25	14	15	Unchoked	38.648	0.169	0.026	0.066	-	-	0.092
26	14	19	Unchoked	10.524	1.419	0.067	-	0.850	0.185	1.102
27	15	16	Unchoked	33.065	0.197	0.026	0.066	0.021	0.072	0.185
28	15	19	Unchoked	10.524	1.419	0.067	-	0.850	0.185	1.102
29	16	17	Unchoked	37.955	0.172	0.026	0.066	0.0003	0.009	0.1013
30	16	19	Choked	9.831	1.519	0.067	-	0.861	0.206	1.134
31	17	18	Unchoked	33.065	0.197	0.026	0.066	0.021	0.072	0.185
32	17	19	Unchoked	10.524	1.419	0.067	-	0.850	0.185	1.102
33	18	19	Unchoked	10.524	1.419	0.067	-	0.850	0.185	1.102
34	20	13	Choked	14.841	0.276	0.018	-	0.012	0.056	0.086
35	20	14	Choked	6.526	0.999	0.026	0.082	0.690	0.182	0.980
36	21	14	Choked	17.652	0.369	0.026	0.052	0.295	-	0.373
37	21	20	Choked	16.702	0.326	0.024	-	-	-	0.024

TABLE 6.2-43B
(Sheet 1 of 1)
BLOWDOWN DATA

10-Inch Low-Pressure Core Spray Line Break
RPV-BSW Annulus

Time (sec)	Blowdown Mass Flow Rate (lbm/sec)*	Blowdown Enthalpy (Btu/lbm)	Blowdown Energy Release Rate (Btu/sec)*	Total Effective Break Area (ft^2)
0.000000	2,184.2	531.8	1.162×10^6	1.010
0.002260	2,184.2	531.8	1.162×10^6	1.010
0.002261	2,830.7	531.8	1.505×10^6	0.904
0.002940	2,830.7	531.8	1.505×10^6	0.904
0.002941	3,909.8	531.8	2.079×10^6	0.904
0.117940	3,909.8	531.8	2.079×10^6	0.904
0.117941	1,751.6	531.8	9.315×10^5	0.405
3.000000	1,751.6	531.8	9.315×10^5	0.405

* Due to symmetry in the nodalization, the tabulated blowdown represents one half of the total blowdown.

TABLE 6.2-44
(Sheet 1 of 1)
SUBCOMPARTMENT NODAL DESCRIPTION

12-Inch Recirculation Inlet Line Break
RPV-BSW Annulus

Volume No.	Volume (ft^3)	Initial Conditions			DBA Break Conditions				Calculated Peak Pressure Difference[1] (psid)	Design Peak Pressure Difference[1] (psid)	Design Margin[2] (%)
		Temp. (°F)	Pressure (psia)	Humidity (%)	% Break in Vol.	Break Line	Break Area (ft^2)	Break Type			
1	242.09	550	14.2	0	0				11.69	70.60	83.4
2	273.71	550	14.2	0	0				10.21	70.60	85.5
3	273.71	550	14.2	0	0				9.64	70.60	86.3
4	264.06	550	14.2	0	0				9.57	70.60	86.4
5	264.06	550	14.2	0	0				11.93	70.60	83.1
6	273.71	550	14.2	0	0				11.58	70.60	83.6
7	233.16	550	14.2	0	0				9.94	70.60	85.9
8	264.19	550	14.2	0	0				9.05	70.60	87.2
9	261.83	550	14.2	0	0				9.68	70.60	86.3
10	261.02	550	14.2	0	0				10.13	70.60	85.7
11	261.02	550	14.2	0	0				11.44	70.60	83.8
12	261.83	550	14.2	0	0				9.17	70.60	87.0
13	247.89	550	14.2	0	0				8.03	70.60	88.6
14	249.57	550	14.2	0	0				7.97	70.60	88.7
15	247.21	550	14.2	0	0				7.92	70.60	88.8
16	249.41	550	14.2	0	0				7.32	70.60	89.6
17	249.57	550	14.2	0	0				7.41	70.60	89.5
18	247.21	550	14.2	0	0				7.48	70.60	89.4
19	137621.0	150	14.2	0	0				0.00	-	-
20	62.08	550	14.2	0	100	Recirc. inlet	(See Table 6.2-44B)	DER	29.92	70.60	57.6

(1) Peak pressure difference [(Pi-P19) peak] is shown on Figure 6.64A.
(2) Design margin: 1-(calculated □ peak/design □peak).

TABLE 6.2-44A
(Sheet 1 of 1)
SUBCOMPARTMENT VENT PATH DESCRIPTION

12-Inch Recirculation Inlet Line Break
RPV-BSW Annulus

Vent Path No.	From Volume Node No.	To Volume Node No.	Description of Vent Path Flow (Choked/Unchoked)	Vent Area (ft^2)	L/A (ft/ft^2)	Head Loss Coefficient				
						Friction	Turning	Expansion	Contraction	Total
1	1	2	Unchoked	35.913	0.181	0.028	0.062	0.023	0.018	0.131
2	1	20	Choked	15.389	0.602	0.042	-	0.006	0.040	0.088
3	2	3	Unchoked	40.489	0.161	0.027	0.066	0.002	0.022	0.117
4	2	8	Unchoked	12.980	1.252	0.073	-	0.050	0.112	0.235
5	3	4	Unchoked	40.489	0.161	0.027	0.066	0.002	0.022	0.117
6	3	9	Unchoked	12.980	1.252	0.073	-	0.050	0.112	0.235
7	4	5	Unchoked	37.129	0.176	0.027	0.066	0.015	0.062	0.170
8	4	10	Unchoked	9.620	1.689	0.073	-	0.180	0.212	0.465
9	5	6	Unchoked	40.489	0.161	0.027	0.066	0.002	0.022	0.117
10	5	11	Unchoked	9.620	1.689	0.073	-	0.180	0.212	0.465
11	6	12	Unchoked	12.980	1.252	0.073	-	0.050	0.112	0.235
12	7	8	Unchoked	35.791	0.182	0.026	0.062	0.016	-	0.104
13	7	13	Unchoked	16.702	0.869	0.065	-	-	-	0.065
14	8	9	Unchoked	39.029	0.167	0.026	0.068	0.002	0.023	0.119
15	8	14	Unchoked	16.702	0.929	0.070	-	-	-	0.070
16	9	10	Unchoked	36.475	0.179	0.026	0.068	0.012	0.054	0.160
17	9	15	Unchoked	12.980	1.196	0.070	-	0.050	0.112	0.232
18	10	11	Unchoked	35.669	0.183	0.026	0.068	0.016	0.064	0.174
19	10	16	Unchoked	16.009	0.970	0.070	-	0.002	0.021	0.093
20	11	12	Unchoked	36.475	0.179	0.026	0.068	0.012	0.054	0.160
21	11	17	Unchoked	16.009	0.970	0.070	-	0.002	0.021	0.093
22	12	18	Unchoked	12.980	1.196	0.070	-	0.050	0.112	0.232
23	13	14	Unchoked	34.926	0.187	0.026	0.066	0.009	0.048	0.149
24	13	19	Choked	8.956	1.667	0.067	-	0.872	0.232	1.171
25	14	15	Unchoked	38.648	0.169	0.026	0.066	-	-	0.092
26	14	19	Choked	10.524	1.419	0.067	-	0.850	0.185	1.102
27	15	16	Unchoked	33.065	0.197	0.026	0.066	0.021	0.072	0.185
28	15	19	Choked	10.524	1.419	0.067	-	0.850	0.185	1.102
29	16	17	Unchoked	37.955	0.172	0.026	0.066	0.0003	0.009	0.1013
30	16	19	Choked	9.831	1.519	0.067	-	0.861	0.206	1.134
31	17	18	Unchoked	33.065	0.197	0.026	0.066	0.021	0.072	0.185
32	17	19	Choked	10.524	1.419	0.067	-	0.850	0.185	1.102
33	18	19	Choked	10.524	1.419	0.067	-	0.850	0.185	1.102
34	20	2	Choked	3.263	1.998	0.026	0.130	0.852	0.182	1.190
35	20	7	Choked	16.702	0.538	0.040	-	-	-	0.040
36	20	8	Choked	3.263	1.998	0.026	0.130	0.846	0.182	1.184

TABLE 6.2-44B
(Sheet 1 of 1)
BLOWDOWN DATA

12-Inch Recirculation Inlet Line Break
RPV-BSW Annulus

Time (sec)	Blowdown Mass Flow Rate (lbm/sec)*	Blowdown Enthalpy (Btu/lbm)	Blowdown Energy Release Rate (Btu/sec)*	Total Effective Break Area (ft^2)
0.000000	3,124.8	532.8	1.665×10^6	1.4450
0.009580	3,124.8	532.8	1.665×10^6	1.4450
0.009581	4,686.2	532.8	2.497×10^6	1.4450
0.170000	4,686.2	532.8	2.497×10^6	1.4450
0.170001	3,358.4	532.8	1.789×10^6	0.7765
3.000000	3,358.4	532.8	1.789×10^6	0.7765

* Due to symmetry in the nodalization, the tabulated blowdown represents one half of the total blowdown.

TABLE 6.2-45
(Sheet 1 of 1)
SUBCOMPARTMENT NODAL DESCRIPTION

24-Inch Recirculation Suction Break
RPV-BSW Annulus

Volume No.	Volume (ft^3)	Initial Conditions: Temp. (°F)	Initial Conditions: Pressure (psia)	Initial Conditions: Humidity (%)	DBA Break Conditions: % Break in Vol.	DBA Break Conditions: Break Line	DBA Break Conditions: Area (ft^2)	DBA Break Conditions: Break Type	Calculated Peak Pressure Difference[1] (psid)	Design Peak Pressure Difference[1] (psid)	Design Margin (%)
1	227.9	150	14.2	20	0				8.26	70.60	88.3
2	276.4	150	14.2	20	0				6.05	70.60	91.4
3	276.4	150	14.2	20	0				5.88	70.60	91.7
4	276.4	150	14.2	20	0				5.86	70.60	91.7
5	276.4	150	14.2	20	0				6.71	70.60	90.5
6	273.9	150	14.2	20	0				7.85	70.60	88.9
7	218.1	150	14.2	20	0				7.53	70.60	89.3
8	264.8	150	14.2	20	0				4.71	70.60	93.3
9	266.7	150	14.2	20	0				4.79	70.60	93.3
10	266.7	150	14.2	20	0				4.68	70.60	93.4
11	266.7	150	14.2	20	0				4.91	70.60	93.0
12	264.1	150	14.2	20	0				4.89	70.60	93.1
13	251.8	150	14.2	20	0				4.53	70.60	93.6
14	248.6	150	14.2	20	0				3.63	70.60	94.9
15	250.5	150	14.2	20	0				3.67	70.60	94.8
16	251.9	150	14.2	20	0				3.33	70.60	95.3
17	250.0	150	14.2	20	0				4.35	70.60	93.8
18	251.8	150	14.2	20	0				5.18	70.60	92.7
19	44,750.0	150	14.2	20	85	Recirc. Suction	(See Table 6.2-45B)		0.00	-	-
20	92.06	150	14.2	20	15				18.74	70.60	73.5

(1) Peak pressure difference [(Pi-P19) peak] is shown on Figure 6.2-66A.
(2) Design margin: 1-(Calculated Δ peak/design Δ peak).

TABLE 6.2-45A
(Sheet 1 of 1)
SUBCOMPARTMENT VENT PATH DESCRIPTION

24-Inch Recirculation Suction Line Break
RPV-BSW Annulus

Vent Path No.	From Volume Node No.	To Volume Node No.	Description of Vent Path Flow (Choked/Unchoked)	Vent Area (ft^2)	L/A (ft/ft^2)	Head Loss Coefficient				
						Friction	Turning	Expansion	Contraction	Total
1	20	1	Choked	16.83	0.581	0.060	-	-	-	0.0600
2	20	2	Choked	6.01	0.915	0.0255	0.098	0.74	0.110	0.9735
3	20	8	Choked	6.01	0.898	0.0255	0.098	0.73	0.110	0.9635
4	20	7	Choked	16.83	0.571	0.027	-	-	-	0.0270
5	1	2	Unchoked	34.73	0.188	0.0279	0.062	-	-	0.0899
6	2	3	Unchoked	40.74	0.160	0.0275	0.068	0.002	0.020	0.1175
7	2	8	Unchoked	13.45	1.208	0.074	-	0.0403	0.100	0.2143
8	3	4	Unchoked	39.55	0.165	0.0275	0.068	0.0046	0.034	0.1341
9	3	9	Unchoked	12.26	1.325	0.074	-	0.0737	0.136	0.2837
10	4	5	Unchoked	40.74	0.160	0.0275	0.068	0.002	0.020	0.1175
11	4	10	Unchoked	13.45	1.208	0.074	-	0.0403	0.100	0.2143
12	5	6	Unchoked	40.74	0.160	0.0275	0.068	0.002	0.020	0.1175
13	5	11	Unchoked	13.45	1.208	0.074	-	0.0403	0.100	0.2143
14	6	12	Unchoked	11.96	1.359	0.074	-	0.084	0.140	0.2980
15	7	8	Unchoked	32.61	0.200	0.0255	0.098	-	-	0.1235
16	7	13	Unchoked	16.20	0.865	0.058	-	0.002	0.020	0.0800
17	8	9	Unchoked	37.56	0.174	0.0255	0.068	0.007	0.040	0.1405
18	8	14	Unchoked	13.45	1.154	0.071	-	0.0403	0.010	0.2113
19	9	10	Unchoked	39.25	0.166	0.0255	0.068	0.002	0.020	0.1155
20	9	15	Unchoked	16.83	0.922	0.071	-	-	-	0.0710
21	10	11	Unchoked	39.25	0.166	0.0255	0.068	0.002	0.020	0.1155
22	10	16	Unchoked	16.83	0.922	0.071	-	-	-	0.0710
23	11	12	Unchoked	37.56	0.174	0.0255	0.068	0.007	0.040	0.1405
24	11	17	Unchoked	13.45	1.154	0.071	-	0.0403	0.100	0.2113
25	12	18	Unchoked	16.20	0.958	0.071	-	0.0014	0.020	0.0924
26	13	14	Unchoked	35.31	0.185	0.0255	0.066	0.0077	0.040	0.1392
27	13	19	Unchoked	9.00	1.676	0.067	-	0.876	0.233	1.1760
28	14	15	Unchoked	34.15	0.194	0.0255	0.066	0.0140	0.060	0.1655
29	14	19	Unchoked	7.57	1.991	0.067	-	0.895	0.275	1.2370
30	15	16	Unchoked	35.31	0.185	0.0255	0.066	0.0077	0.040	0.1392
31	15	19	Unchoked	7.57	1.991	0.067	-	0.895	0.275	1.2370
32	16	17	Unchoked	38.69	0.169	0.0255	0.066	-	-	0.0915
33	16	19	Unchoked	9.00	1.676	0.067	-	0.876	0.233	1.1760
34	17	18	Unchoked	33.62	0.194	0.0255	0.066	0.0173	0.070	0.1788
35	17	19	Unchoked	9.00	1.676	0.067	-	0.876	0.233	1.1760
36	18	19	Unchoked	8.375	1.801	0.067	-	0.884	0.251	1.2020

TABLE 6.2-45B
(Sheet 1 of 1)
BLOWDOWN DATA

24-Inch Recirculation Suction Line Break
RPV-BSW Annulus

Time (sec)	Blowdown Mass Flow Rate (lbm/sec)*	Blowdown Enthalpy (Btu/lbm)	Blowdown Energy Release Rate (Btu/sec)*	Total Effective Break Area (ft^2)
0.0000	17,247	529.0	9.124×10^6	5.072
1.5400	17,247	529.0	9.124×10^6	5.072
1.5401	13,108	529.0	6.934×10^6	2.891
2.0000	13,108	529.0	6.934×10^6	2.891

* Due to symmetry in the nodalization, the tabulated blowdown represents one half of the total blowdown. Of the tabulated blowdown, 85 percent is directed to Node 19 and 15 percent to Node 20 by the flow diverter.

TABLE 6.2-46
(Sheet 1 of 1)
FORCE AND MOMENT SENSITIVITY STUDY SUMMARY

Subcompartment	Model	Design Basis Line Description	Tables	Figures	
			Data Sheet	Forces	Moments
RPV-BSW annulus	21-node	Feedwater	6.2-48	6.2-68	6.2-69
RPV-BSW annulus	37-node	Feedwater	6.2-49	6.2-68A	6.2-69A

NOTES:

1. Maximum forces and moments are listed in Table 6.2-47.
2. The annulus pressurization geometry for force and moment calculations is shown on Figure 6.2-67.

TABLE 6.2-47
(Sheet 1 of 1)
MAXIMUM FORCES AND MOMENTS ON THE BSW
FEEDWATER LINE BREAKS
RPV-BSW ANNULUS

	21 Node Model	37 Node Model
X-Direction Force:	509.90 kips at 0.03100 sec	485.29 kips at 0.03100 sec
Z-Direction Force:	-2701.84 kips at 0.64000 sec	-2461.74 kips at 0.62000 sec
Resultant Force:	2703.10 kips at 0.64000 sec	2466.08 kips at 0.61000 sec
X-Direction Moment:	-64496.49 kip-ft at 0.62000 sec	-58806.14 kip-ft at 0.61000 sec
Z-Direction Moment:	-18297.80 kip-ft at 0.02700 sec	-17485.39 kip-ft at 0.02700 sec
Resultant Moment:	64588.45 kip-ft at 0.62000 sec	59114.79 kip-ft at 0.60000 sec

NOTES:

1. There are no forces nor moments with respect to the y-axis as defined on Figure 6.2-67.
2. These forces and moments are based on a half-annular model.

TABLE 6.2-48
(Sheet 1 of 1)
PROJECTED AREAS AND MOMENT ARMS FOR FORCE AND MOMENT CALCULATIONS
12-INCH FEEDWATER LINE BREAK
21 NODE MODEL
RPV-BSW ANNULUS

Node No.	Nodal Height (ft)	Boundary Azimuths(°)	Span (°)	Projected X-Area (sq ft)	Projected Z-Area (sq ft)	Moment Arm (ft)
1	16.54	180.00 - 210.00	30.00	116.29	-31.16	8.269
2	16.54	150.00 - 180.00	30.00	85.13	-85.13	8.269
3	16.54	120.00 - 150.00	30.00	31.16	-116.29	8.269
4	16.54	90.00 - 120.00	30.00	-31.16	-116.29	8.269
5	16.54	60.00 - 90.00	30.00	-85.13	-85.13	8.269
6	16.54	30.00 - 60.00	30.00	-116.29	-31.16	8.269
7	15.96	180.00 - 210.00	30.00	112.22	-30.07	24.518
8	15.96	150.00 - 180.00	30.00	82.15	-82.15	24.518
9	15.96	120.00 - 150.00	30.00	30.07	-112.22	24.518
10	15.96	90.00 - 120.00	30.00	-30.07	-112.22	24.518
11	15.96	60.00 - 90.00	30.00	-82.15	-82.15	24.518
12	15.96	30.00 - 60.00	30.00	-112.22	-30.07	24.518
13	7.06	180.00 - 210.00	30.00	49.64	-13.30	36.028
14	15.08	150.00 - 180.00	30.00	77.65	-77.65	40.041
15	15.08	120.00 - 150.00	30.00	28.42	-106.07	40.041
16	15.08	90.00 - 120.00	30.00	-28.42	-106.07	40.041
17	15.08	60.00 - 90.00	30.00	-77.65	-77.65	40.041
18	15.08	30.00 - 60.00	30.00	-106.07	-28.42	40.041
19	6.00	180.00 - 210.00	30.00	42.19	-11.30	42.558
20	2.02	180.00 - 210.00	30.00	14.24	-3.82	46.571
21	0.0	0.00 - 0.00	0.00	0.00	0.00	0.00

TABLE 6.2-49
(Sheet 1 of 1)
PROJECTED AREAS AND MOMENT ARMS FOR FORCE AND MOMENT CALCULATIONS
12-INCH FEEDWATER LINE BREAK
37 NODE MODEL
RPV-BSW ANNULUS

Node No.	Nodal Height (ft)	Boundary Azimuths(°)	Span(°)	Projected X-Area (sq ft)	Projected Z-Area (sq ft)	Moment Arm (ft)
1	15.82	180.00 - 210.00	30.00	111.26	-29.81	7.911
2	15.82	150.00 - 180.00	30.00	81.45	-81.45	7.911
3	16.53	120.00 - 150.00	30.00	31.15	-116.24	8.265
4	2.42	105.00 - 120.00	15.00	-1.16	-8.80	15.323
5	14.11	105.00 - 120.00	15.00	-6.76	-51.37	7.057
6	2.42	90.00 - 105.00	15.00	-3.40	-8.20	15.323
7	14.11	90.00 - 105.00	15.00	-19.83	-47.87	7.057
8	16.53	60.00 - 90.00	30.00	-85.09	-85.09	8.265
9	16.53	30.00 - 60.00	30.00	-116.24	-31.15	8.265
10	16.13	200.00 - 210.00	10.00	39.38	-3.45	23.885
11	16.13	180.00 - 200.00	20.00	74.01	-26.94	23.885
12	16.13	160.00 - 180.00	20.00	60.33	-50.62	23.885
13	16.13	150.00 - 160.00	10.00	22.67	-32.38	23.885
14	15.96	135.00 - 150.00	15.00	22.42	-54.13	24.510
15	15.96	120.00 - 135.00	15.00	7.65	-58.08	24.510
16	24.85	90.00 - 120.00	30.00	-46.83	-174.76	28.958
17	26.02	60.00 - 90.00	30.00	-133.94	-133.94	29.542
18	15.96	45.00 - 60.00	15.00	-54.13	-22.42	24.510
19	15.96	30.00 - 45.00	15.00	-58.08	-7.65	24.510
20	6.88	200.00 - 210.00	10.00	16.79	-1.47	35.385
21	6.88	180.00 - 200.00	20.00	31.55	-11.48	35.385
22	6.88	160.00 - 180.00	20.00	25.72	-21.58	35.385
23	6.88	150.00 - 160.00	10.00	9.67	-13.81	35.385
24	8.90	135.00 - 150.00	15.00	12.50	-30.17	36.937
25	8.90	120.00 - 135.00	15.00	4.26	-32.38	36.937
26	10.06	45.00 - 60.00	15.00	-34.13	-14.14	37.521
27	10.06	30.00 - 45.00	15.00	-36.63	-4.82	37.521
28	3.05	190.00 - 210.00	20.00	14.68	-2.59	46.057
29	5.71	190.00 - 210.00	20.00	27.45	-4.84	41.677
30	3.05	180.00 - 190.00	10.00	6.78	-3.16	46.057
31	5.71	180.00 - 190.00	10.00	12.68	-5.91	41.677
32	5.03	150.00 - 180.00	30.00	25.90	-25.90	45.067
33	3.73	150.00 - 180.00	30.00	19.19	-19.19	40.688
34	6.20	120.00 - 150.00	30.00	11.68	-43.58	44.404
35	6.20	90.00 - 120.00	30.00	-11.68	-43.58	44.484
36	5.03	30.00 - 90.00	60.00	-61.27	-35.38	45.067
37	0.00	0.00 - 0.00	0.00	0.00	0.00	0.00

TABLE 6.2-50
(Sheet 1 of 1)
MASS AND ENERGY RELEASE DATA - OLTP ANALYSIS

Main Steam Line DER With Feedwater (Case C)

Pipe ID (in)	23.362
Effective break area	Figure 6.2-14
Blowdown code	LOCTVS

Blowdown Table

Time (sec)	Blowdown Rate (lb/sec)	Enthalpy (Btu/lb)	Reactor Vessel Pressure (psia)
0.01	11,564	1,189.6	1,055
0.12[1]	11,482	1,189.9	1,048
0.13	8,606	1,189.9	1,047
0.50	8,444	1,190.7	1,028
1.11[2]	8,182	1,191.8	999
1.50	27,125	581.5	987
2.00	27,345	576.8	981
5.00	22,450	575.1	949
10.00	18,889	581.8	897
20.00	15,426	572.2	714
30.00[5]	12,187	548.6	534
50.00	7,764	471.6	273
100.10	4,055	353.8	87
200.10	5,540	264.1	59
245.6[6]	1,652	380.4	-
247.6[3]	0	-	-
338.1[4]	0	-	-
338.6	2,387	231.2	-
1,000.1	2,197	180.7	-
10,000	2,184	171.0	-

(1) Inventory period ends.
(2) Froth level reaches top of steam dryer.
(3) Blowdown ends.
(4) Water level recovers to steam line elevation.
(5) ECCS pump starts.
(6) Peak drywell pressure.

TABLE 6.2-51
(Sheet 1 of 1)
CONTAINMENT SPRAY PARAMETERS

<u>Drywell</u>		
1.	No. of independent loops	2
2.	No. of spray nozzles/loop	64/Loop A 59/Loop B
3.	Nozzle manufacturer	SPRACO
4.	Nozzle No. (SPRACO)	47-1815-26
5.	Flow rate and pressure drop	
	a. Loop A	104 gpm @ 42 psi*
	b. Loop B	111 gpm @ 47 psi*
6.	Drop size (Sauter mean)	959 microns
7.	Spray drop efficiency	100%
<u>Suppression Chamber</u>		
1.	No. of loops	1 (common to both RHR pumps)
2.	No. of spray nozzles	35
3.	Nozzle manufacturer	SPRACO
4.	Nozzle No. (SPRACO)	47-0516-14
5.	Flow rate and pressure drop	11 gpm @ 30 psi*
6.	Drop size (Sauter mean)	773 microns
7.	Spray drop efficiency	100%

* Values shown are from nozzle test data. The hydraulic analysis for the containment spray system flow rates used a resistance coefficient (k) which is 56 percent greater than the k given by this test data.

TABLE 6.2-52
(Sheet 1 of 1)
ACCIDENT ANALYSIS PARAMETERS USED
FOR DBA OF CONTAINMENT HEAT REMOVAL
(PRE-EPU ANALYSIS)

1.	Design basis accident (for containment sprays)	Steam line break area of 0.3 ft^2
2.	Steam bypass factor	0.05 ft^2 ($A/\sqrt{K}$ factor)
3.	Containment spray initiation	a. Manual action b. Spray operation within 30 min[1] after break
4.	Containment parameters	Tables 6.2-1, 6.2-2, and 6.2-3
5.	Spray rate, gpm	
	Drywell	6,672
	Suppression chamber	428
6.	Heat exchanger K factor	239 Btu/sec/°F
7.	No. of downcomers	121
8.	Spray drop efficiency	90%

(1) EPU evaluation is based on spray initiation time of 20 min.

TABLE 6.2-53
(Sheet 1 of 2)
ENERGY/MASS BALANCE (STEAM BYPASS ANALYSIS)

Steam Line Break: 0.3 Ft^2

Control Volume: Suppression Pool, Reactor Vessel, Drywell Atmosphere, Suppression Chamber Atmosphere, and Liquid on Drywell Floor
(Pre-EPU Analysis)[(1)]

	Seconds				
	0	900	1,800	21,003	36,000
Total mass, lbm	10293942.4	10293315.4	10301145.4	10468211.5	10598685.4
Total internal energy, Btu	940335705.1	1071191462.1	1147890247.9	1668854263.1	1699385105.3
Integrated Flow and Energy into Control Volume					
Coastdown heat/decay heat	0	125748728.9	194404337.9	990921439.2	1442048824.8
Feedwater metal heat	0	56490542.1	71624743.2	110429368.9	144504489.9
Feedwater mass	0	0.0	0.0	0.0	0.0
Feedwater energy	0	0.0	0.0	0.0	0.0
CRD-Mass	0	7830.0	15660.0	182726.1	313200.0
CRD-Energy	0	845640.0	1691280.0	19734418.8	33825600.0
Pump heat	0	1811040.0	3754975.0	58733164.0	100589986.0
Total mass into control volume	0	7830.0	15660.0	182726.1	313200.0
Total energy into control volume	0	184895951.0	271475336.1	1179818391.0	1720968900.7
Integrated Flow and Energy Out of Control Volume					
RHR heat exchanger shutdown cooling mode	0	0	0	0	408440154.4
RHR heat exchanger pool cooling mode	0	0	0	0	0
Main steam mass	0	8457.0	8457.0	8457.0	8457.0
Main steam energy	0	10073333.7	10073333.7	10073333.7	10073333.7
Drywell and suppression chamber spray heat exchanger (RHR)	0	0.0	48051.8	374541850.6	429671679.6
Drywell heat sinks	0	41993245.3	48403648.3	51707053.5	85507110.0
Suppression chamber heat sinks	0	1175973.7	3712435.5	-162200.5	6379837.1
P1 heat sinks	0	797641.3	1683323.9	15139795.5	21847385.7
Total mass out of control volume	0	8457.0	8457.0	8457.0	8457.0
Total energy out of control volume	0	54040194.0	63920793.3	451299833.0	961919500.4

TABLE 6.2-53
(Sheet 2 of 2)

	Seconds				
	0	900	1,800	21,003	36,000
Integrated Flow and Energy Out of Control Volume (cont'd.)					
Total offset in control volume mass balance evaluated from 0 sec		0	0	0	0
Total offset in control volume energy balance evaluated from 0 sec		0	0	0	0

(1) The EPU evaluation is based on 20 min spray initiation time, whereas pre-EPU analysis used 30 min spray initiation time.

TABLE 6.2-54
(Sheet 1 of 1)
SECONDARY CONTAINMENT DATA

1. Total Volume	4.01 x 10^6 cu ft
Reactor building refueling area	1,641,600 cu ft
Reactor building volume below refueling area	1,970,000 cu ft
HPCS room	16,625 cu ft
LPCS room	20,750 cu ft
RHR pump room A	20,980 cu ft
RHR pump room B	20,830 cu ft
RHR pump room C	20,830 cu ft
RHR heat exchanger room A	26,600 cu ft
RHR heat exchanger room B	27,360 cu ft
North auxiliary bay	201,000 cu ft
South auxiliary bay	164,000 cu ft
2. Area Environmental Conditions	See Table 9.4-1
3. SGTS System (see Figure 6.2-78)	
SGTS flow without recirculation	3,720 cfm
SGTS flow with recirculation	4,000 cfm
4. In-leakage Rate (maximum)	2,670 cfm of outside air at -0.25 in W.G. ΔP at the roof, when outside air is -20°F and secondary containment at 105°F
5. Cooling System Data	See Table 9.4-3* (Historical)

* The heat removal rates are for 104°F inlet air temperature, 77°F inlet cooling water temperature and dry air condition.

TABLE 6.2-55
(Sheet 1 of 1)

THIS TABLE HAS BEEN DELETED

TABLE 6.2-55a
(Sheet 1 of 2)
EVALUATION OF POTENTIAL BYPASS LEAKAGE PATHS (ISOTHERMAL FLOW MODEL) - LOSS OF ONE DIESEL GENERATOR

Line Description	Termination Region	Bypass Leakage Barrier	Leak Rate[3]		Containment Bypass Leak Rate (Fraction/Day)[5]				
			Tech. Spec. SCFH[1,4]	Fraction/ Day[2]	0-2 hr	0-8 hr	8-24 hr	1-4 day	4-30 day
Main steam line	Turbine Building	2-26" valves in each line	24	0.520×10^{-3}	See Note 8				
Main steam drain line (inboard)	Turbine Building	1-6" valve	1.875	0.435×10^{-4}	0.633×10^{-4}	0.572×10^{-4}	0.555×10^{-4}	0.497×10^{-4}	0.344×10^{-4}
Main steam drain line (outboard)	Turbine Building	1-2" valve	0.625	0.145×10^{-4}	0.211×10^{-4}	0.191×10^{-4}	0.185×10^{-4}	0.166×10^{-4}	0.115×10^{-4}
4 Post-accident sampling lines	Radwaste Tunnel	1-3/4" valve in each line	0.2344	0.543×10^{-5}	0.317×10^{-4}	0.286×10^{-4}	0.277×10^{-4}	0.249×10^{-4}	0.172×10^{-4}
Drywell equipment drain line	Radwaste Tunnel	1-4" valve	1.25	0.290×10^{-4}	0.422×10^{-4}	0.381×10^{-4}	0.370×10^{-4}	0.331×10^{-4}	0.229×10^{-4}
Drywell equipment vent line	Radwaste Tunnel	1-2" valve	0.625	0.145×10^{-4}	0.211×10^{-4}	0.191×10^{-4}	0.185×10^{-4}	0.166×10^{-4}	0.115×10^{-4}
Drywell floor drain line	Radwaste Tunnel	1-6" valve	1.875	0.435×10^{-4}	0.633×10^{-4}	0.572×10^{-4}	0.555×10^{-4}	0.497×10^{-4}	0.344×10^{-4}
Drywell floor vent line	Radwaste Tunnel	1-3" valve	0.9375	0.217×10^{-4}	0.317×10^{-4}	0.286×10^{-4}	0.277×10^{-4}	0.249×10^{-4}	0.172×10^{-4}
RWCU line	Turbine Building	1-8" valve	2.5	0.579×10^{-4}	0.845×10^{-4}	0.763×10^{-4}	0.739×10^{-4}	0.663×10^{-4}	0.459×10^{-4}
Feedwater line	Turbine Building	2-24" check valves	12	0.278×10^{-3}	0.270×10^{-3}	0.246×10^{-3}	0.239×10^{-3}	0.216×10^{-3}	0.153×10^{-3}
CPS supply line to drywell	Standby Gas Trtmt Area	2-14" valves	4.38	0.102×10^{-3}	0.985×10^{-4}	0.898×10^{-4}	0.873×10^{-4}	0.789×10^{-4}	0.557×10^{-4}
CPS supply line to drywell	Standby Gas Trtmt Area	2-2" valves	0.625	0.145×10^{-4}	0.141×10^{-4}	0.128×10^{-4}	0.125×10^{-4}	0.113×10^{-4}	0.795×10^{-5}
CPS supply line to supp. chamber[6]	Standby Gas Trtmt Area	2-12" valves	3.75	0.523×10^{-4}	0.507×10^{-4}	0.462×10^{-4}	0.450×10^{-4}	0.406×10^{-4}	0.287×10^{-4}

TABLE 6.2-55a
(Sheet 2 of 2)

Line Description	Termination Region	Bypass Leakage Barrier	Leak Rate[3] Tech. Spec. SCFH[1,4]	Leak Rate[3] Fraction/Day[2]	Containment Bypass Leak Rate (Fraction/Day)[5] 0-2 hr	0-8 hr	8-24 hr	1-4 day	4-30 day
CPS supply line to supp. chamber[6]	Standby Gas Treatment Area	2-2" valves	0.625	0.871×10^{-5}	0.845×10^{-5}	0.770×10^{-5}	0.749×10^{-5}	0.677×10^{-5}	0.478×10^{-5}
Inst. air to SRV accumulators	Yard	1-1 1/2" SOV	Combined Leakage 3.6[7]	0.834×10^{-4}	0.122×10^{-3}	0.110×10^{-3}	0.106×10^{-3}	0.954×10^{-4}	0.660×10^{-4}
Inst. air to drywell	Yard	1-1 1/2" SOV							
Inst. air to drywell	Yard	1-1 1/2" SOV							
Inst. air to CPS valve in supp. Chamber[9]	Yard	1-1 1/2" check valve							
Inst. air to CPS valve in supp. chamber	Yard	1-1 1/2" check valve							
N_2 purge to TIP index mechanism	Yard	1-1/2" check valve							
Inst. air to ADS accumulators	Yard	1-1 1/2" check valve	0.9375	0.217×10^{-4}	0.317×10^{-4}	0.286×10^{-4}	0.277×10^{-4}	0.249×10^{-4}	0.172×10^{-4}
Inst. air to ADS accumulators	Yard	1-1 1/2" check valve	0.9375	0.217×10^{-4}	0.317×10^{-4}	0.286×10^{-4}	0.277×10^{-4}	0.249×10^{-4}	0.172×10^{-4}

(1) Std Conditions: 14.7 psia and 68°F.
(2) Fraction/Day is defined as fraction of drywell volume leakage/day per line under test conditions.
(3) Test Conditions: Air medium; 40 psig and 80°F.
(4) The leak rate is based on ASME Section XI, 1983 Edition through Summer 1983 Addenda (Subsection IWV-3426) applied to each valve, except for main steam and feedwater lines.
(5) Fraction/Day is defined as fraction of drywell volume leakage/day under LOCA conditions.
(6) Leak rate is defined as a fraction of entire primary containment volume under LOCA conditions.
(7) All these paths terminate at 2GSN-TK2 within the reactor building, and only one line goes out of the reactor building.
(8) Technical Specification leak rate for MSIVs is 24 SCFH, but fraction/day leak rates are not computed for isothermal conditions. The isentropic flow case (see Table 6.2-55b) is most conservative and is used in the radiological analysis.
(9) The instrument air bypass leakage path, through containment penetration Z-92, containing valves 2CPS*SOV133 and 2CPS*V51, has been eliminated. However, the leakage rate through this path has been retained in the analysis, as the results are more conservative.

TABLE 6.2-55b
(Sheet 1 of 2)
EVALUATION OF POTENTIAL BYPASS LEAKAGE PATHS (ISENTROPIC FLOW MODEL) - LOSS OF ONE DIESEL GENERATOR

Line Description	Termination Region	Bypass Leakage Barrier	Leak Rate[3]		Containment Bypass Leak Rate (Fraction/Day)[5][8]				
			Tech. Spec. SCFH[1,4]	Fraction/ Day[2]	0-2 hr	0-8 hr	8-24 hr	1-4 day	4-30 day
Main steam line	Turbine Bldg	2-26" valves in each line	24	0.520×10^{-3}	0.606×10^{-3}	0.564×10^{-3}	0.563×10^{-3}	0.479×10^{-3}	0.364×10^{-3}
Main steam drain line (inboard)	Turbine Bldg	1-6" valve	1.875	0.435×10^{-4}	0.663×10^{-4}	0.615×10^{-4}	0.597×10^{-4}	0.564×10^{-4}	0.442×10^{-4}
Main steam drain line (outboard)	Turbine Bldg	1-2" valve	0.625	0.145×10^{-4}	0.221×10^{-4}	0.205×10^{-4}	0.199×10^{-4}	0.188×10^{-4}	0.147×10^{-4}
4 Postaccident sampling lines	Radwaste Tunnel	1-3/4" valve in each line	0.2344	0.543×10^{-5}	0.331×10^{-4}	0.307×10^{-4}	0.298×10^{-4}	0.282×10^{-4}	0.221×10^{-4}
Drywell equipment drain line	Radwaste Tunnel	1-4" valve	1.25	0.290×10^{-4}	0.442×10^{-4}	0.410×10^{-4}	0.398×10^{-4}	0.376×10^{-4}	0.294×10^{-4}
Drywell equipment vent line	Radwaste Tunnel	1-2" valve	0.625	0.145×10^{-4}	0.221×10^{-4}	0.205×10^{-4}	0.199×10^{-4}	0.188×10^{-4}	0.147×10^{-4}
Drywell floor drain line	Radwaste Tunnel	1-6" valve	1.875	0.435×10^{-4}	0.663×10^{-4}	0.615×10^{-4}	0.597×10^{-4}	0.564×10^{-4}	0.442×10^{-4}
Drywell floor vent line	Radwaste Tunnel	1-3" valve	0.9375	0.217×10^{-4}	0.331×10^{-4}	0.307×10^{-4}	0.298×10^{-4}	0.282×10^{-4}	0.221×10^{-4}
RWCU line	Turbine Bldg	1-8" valve	2.5	0.579×10^{-4}	0.883×10^{-4}	0.820×10^{-4}	0.796×10^{-4}	0.751×10^{-4}	0.589×10^{-4}
Feedwater line	Turbine Bldg	2-24" check valves	12	0.278×10^{-3}	0.326×10^{-3}	0.303×10^{-3}	0.296×10^{-3}	0.275×10^{-3}	0.206×10^{-3}
CPS supply line to drywell	Standby Gas Trtmt Area	2-14" valves	4.38	0.102×10^{-3}	0.119×10^{-3}	0.111×10^{-3}	0.108×10^{-3}	0.100×10^{-3}	0.751×10^{-4}
CPS supply line to drywell	Standby Gas Trtmt Area	2-2" valves	0.625	0.145×10^{-4}	0.170×10^{-4}	0.158×10^{-4}	0.154×10^{-4}	0.143×10^{-4}	0.107×10^{-4}
CPS supply line[6] to supp. chamber	Standby Gas Trtmt Area	2-12" valves	3.75	0.523×10^{-4}	0.612×10^{-4}	0.570×10^{-4}	0.556×10^{-4}	0.517×10^{-4}	0.387×10^{-4}
CPS supply line[6] to supp. chamber	Standby Gas Trtmt Area	2-2" valves	0.625	0.871×10^{-5}	0.102×10^{-4}	0.950×10^{-5}	0.926×10^{-5}	0.861×10^{-5}	0.644×10^{-5}

TABLE 6.2-55b
(Sheet 2 of 2)

Line Description	Termination Region	Bypass Leakage Barrier	Leak Rate[3]		Containment Bypass Leak Rate (Fraction/Day)[5][8]				
			Tech. Spec. SCFH[1,4]	Fraction/ Day[2]	0-2 hr	0-8 hr	8-24 hr	1-4 day	4-30 day
Inst. air to SRV accumulators	Yard	1-1 1/2" SOV	Combined Leakage 3.6[7]	0.834×10^{-4}	0.127×10^{-3}	0.118×10^{-3}	0.115×10^{-3}	0.108×10^{-3}	0.848×10^{-4}
Inst. air to drywell	Yard	1-1 1/2" SOV							
Inst. air to drywell	Yard	1-1 1/2" SOV							
Inst. air to CPS valve in supp. chamber[9]	Yard	1-1 1/2" check valve							
Inst. air to CPS valve in supp. chamber	Yard	1-1 1/2" check valve							
N_2 purge to TIP index mechanism	Yard	1-1/2" check valve							
Inst. air to ADS accumulators	Yard	1-1 1/2" check valve	0.9375	0.217×10^{-4}	0.331×10^{-4}	0.307×10^{-4}	0.298×10^{-4}	0.282×10^{-4}	0.221×10^{-4}
Inst. air to ADS accumulators	Yard	1-1 1/2" check valve	0.9375	0.217×10^{-4}	0.331×10^{-4}	0.307×10^{-4}	0.298×10^{-4}	0.282×10^{-4}	0.221×10^{-4}

(1) Std Conditions: 14.7 psia and 68°F.
(2) Fraction/Day is defined as fraction of drywell volume leakage/day per line under test conditions.
(3) Test Conditions: Air medium; 40 psig and 80°F.
(4) The leak rate is based on ASME Section XI, 1983 Edition through Summer 1983 Addenda (Subsection IWV-3426) applied to each valve, except for main steam lines and feedwater lines.
(5) Fraction/Day is defined as fraction of drywell volume leakage/day under LOCA conditions.
(6) Leak rate is defined as a fraction of entire primary containment volume under LOCA conditions.
(7) All these paths terminate at 2GSN-TK2 within the reactor building, and only one line goes out of the reactor building.
(8) Alternate MSIV leak rates below those indicated may be applied.
(9) The instrument air bypass leakage path, through containment penetration Z-92, containing valves 2CPS*SOV133 and 2CPS*V51, has been eliminated. However, the leakage rate through this path has been retained in the analysis, as the results are more conservative.

TABLE 6.2-55c
(Sheet 1 of 2)
EVALUATION OF POTENTIAL BYPASS LEAKAGE PATHS (ISOTHERMAL FLOW MODEL - MSIV FAILURE)

Line Description	Termination Region	Bypass Leakage Barrier	Leak Rate[3]		Containment Bypass Leak Rate (Fraction/Day)[5]				
			Tech. Spec. SCFH[1,4]	Fraction/ Day[2]	0-2 hr	0-8 hr	8-24 hr	1-4 day	4-30 day
Main steam line	Turbine Bldg	1-26"	24	0.520×10^{-3}	SEE NOTE 8				
Main steam drain line (inboard)	Turbine Bldg	2-6" valve	1.875	0.435×10^{-4}	0.422×10^{-4}	0.384×10^{-4}	0.374×10^{-4}	0.338×10^{-4}	0.239×10^{-4}
Main steam drain line (outboard)	Turbine Bldg	1-2" valve	0.625	0.145×10^{-4}	0.211×10^{-4}	0.191×10^{-4}	0.185×10^{-4}	0.166×10^{-4}	0.115×10^{-4}
4 Post-accident sampling lines	Radwaste Tunnel	2-3/4" valve in each line	0.2344	0.543×10^{-5}	0.211×10^{-4}	0.192×10^{-4}	0.187×10^{-4}	0.169×10^{-4}	0.119×10^{-4}
Drywell equipment drain line	Radwaste Tunnel	2-4" valve	1.25	0.290×10^{-4}	0.281×10^{-4}	0.256×10^{-4}	0.249×10^{-4}	0.225×10^{-4}	0.159×10^{-4}
Drywell equipment vent line	Radwaste Tunnel	2-2" valve	0.625	0.145×10^{-4}	0.141×10^{-4}	0.128×10^{-4}	0.125×10^{-4}	0.113×10^{-4}	0.795×10^{-5}
Drywell floor drain line	Radwaste Tunnel	2-6" valve	1.875	0.435×10^{-4}	0.422×10^{-4}	0.384×10^{-4}	0.374×10^{-4}	0.338×10^{-4}	0.239×10^{-4}
Drywell floor vent line	Radwaste Tunnel	2-3" valve	0.9375	0.217×10^{-4}	0.211×10^{-4}	0.192×10^{-4}	0.187×10^{-4}	0.169×10^{-4}	0.119×10^{-4}
RWCU line	Turbine Bldg	2-8" valve	2.5	0.579×10^{-4}	0.562×10^{-4}	0.512×10^{-4}	0.498×10^{-4}	0.450×10^{-4}	0.318×10^{-4}
Feedwater line	Turbine Bldg	2-24" check valves	12	0.278×10^{-3}	0.270×10^{-3}	0.246×10^{-3}	0.239×10^{-3}	0.216×10^{-3}	0.153×10^{-3}
CPS supply line to drywell	Standby Gas Trtmt Area	2-14" valves	4.38	0.102×10^{-3}	0.985×10^{-4}	0.898×10^{-4}	0.873×10^{-4}	0.789×10^{-4}	0.557×10^{-4}
CPS supply line to drywell	Standby Gas Trtmt Area	2-2" valves	0.625	0.145×10^{-4}	0.141×10^{-4}	0.128×10^{-4}	0.125×10^{-4}	0.113×10^{-4}	0.795×10^{-5}
CPS supply line to supp. chamber[6]	Standby Gas Trtmt Area	2-12" valves	3.75	0.523×10^{-4}	0.507×10^{-4}	0.462×10^{-4}	0.450×10^{-4}	0.406×10^{-4}	0.287×10^{-4}
CPS supply line[6] to supp. chamber	Standby Gas Trtmt Area	2-2" valves	0.625	0.871×10^{-5}	0.845×10^{-5}	0.770×10^{-5}	0.749×10^{-5}	0.677×10^{-5}	0.478×10^{-5}

TABLE 6.2-55c
(Sheet 2 of 2)

Line Description	Termination Region	Bypass Leakage Barrier	Leak Rate[3]		Containment Bypass Leak Rate (Fraction/Day)[5]				
			Tech. Spec. SCFH[1,4]	Fraction/ Day[2]	0-2 hr	0-8 hr	8-24 hr	1-4 day	4-30 day
Inst. air to SRV accumulators	Yard	2-1 1/2″ SOV	Combined Leakage 3.6[7]	0.834×10^{-4}	0.810×10^{-4}	0.738×10^{-4}	0.717×10^{-4}	0.649×10^{-4}	0.458×10^{-4}
Inst. air to drywell	Yard	2-1 1/2″ SOV							
Inst. air to drywell	Yard	2-1 1/2″ SOV							
Inst. air to CPS valve in supp. chamber[9]	Yard	1-1″ valve & 1-1 1/2″ check valve							
Inst. air to CPS valve in supp. chamber	Yard	1-1″ valve & 1-1 1/2″ check valve							
N_2 purge to TIP index mechanism	Yard	1-1″ valve & 1-1 1/2″ check valve							
Inst. air to ADS accumulators	Yard	1-1 1/2″ valve & 1-1 1/2″ check valve	0.9375	0.217×10^{-4}	0.211×10^{-4}	0.192×10^{-4}	0.187×10^{-4}	0.169×10^{-4}	0.119×10^{-4}
Inst. air to ADS accumulators	Yard	1-1 1/2″ valve & 1-1 1/2″ check valve	0.9375	0.217×10^{-4}	0.211×10^{-4}	0.192×10^{-4}	0.187×10^{-4}	0.169×10^{-4}	0.119×10^{-4}

(1) Std. conditions: 14.7 psia and 68°F.
(2) Fraction/Day is defined as fraction of drywell volume leakage/day per line under test conditions.
(3) Test Conditions: Air medium; 40 psig and 80°F.
(4) The leak rate is based on ASME Section XI, 1983 Edition through Summer 1983 Addenda (Subsection IWV-3426) applied to each valve, except for main steam and feedwater lines.
(5) Fraction/Day is defined as fraction of drywell volume leakage/day under LOCA conditions.
(6) Leak rate is defined as a fraction of entire primary containment volume under LOCA conditions.
(7) All these paths terminate at 2GSN-TK2 within the reactor building, and only one line goes out of the reactor building.
(8) Technical Specification leak rate for MSIVs is 24 SCFH, but Fraction/Day leak rates are not computed for isothermal conditions. The isentropic flow case (see Table 6.2-55d) is most conservative and is used in the radiological analysis.
(9) The instrument air bypass leakage path, through containment penetration Z-92, containing valves 2CPS*SOV133 and 2CPS*V51, has been eliminated. However, the leakage rate through this path has been retained in the analysis, as the results are more conservative.

TABLE 6.2-55d
(Sheet 1 of 2)
EVALUATION OF POTENTIAL BYPASS LEAKAGE PATHS (ISENTROPIC FLOW MODEL - MSIV FAILURE)

Line Description	Termination Region	Bypass Leakage Barrier	Leak Rate[3]		Containment Bypass Leak Rate (Fraction/Day)[5][8]				
			Tech. Spec. SCFH[1,4]	Fraction/ Day[2]	0-2 hr	0-8 hr	8-24 hr	1-4 day	4-30 day
Main steam line	Turbine Bldg	1-26"	24	0.520×10^{-3}	0.788×10^{-3}	0.730×10^{-3}	0.710×10^{-3}	0.648×10^{-3}	0.505×10^{-3}
Main steam drain line (inboard)	Turbine Bldg	2-6" valve	1.875	0.435×10^{-4}	0.509×10^{-4}	0.474×10^{-4}	0.462×10^{-4}	0.430×10^{-4}	0.321×10^{-4}
Main steam drain line (outboard)	Turbine Bldg	1-2" valve	0.625	0.145×10^{-4}	0.221×10^{-4}	0.205×10^{-4}	0.199×10^{-4}	0.188×10^{-4}	0.147×10^{-4}
4 Postaccident sampling lines	Radwaste Tunnel	2-3/4" valve in each line	0.2344	0.543×10^{-5}	0.254×10^{-4}	0.237×10^{-4}	0.231×10^{-4}	0.215×10^{-4}	0.161×10^{-4}
Drywell equipment drain line	Radwaste Tunnel	2-4" valve	1.25	0.290×10^{-4}	0.339×10^{-4}	0.316×10^{-4}	0.308×10^{-4}	0.286×10^{-4}	0.214×10^{-4}
Drywell equipment vent line	Radwaste Tunnel	2-2" valve	0.625	0.145×10^{-4}	0.170×10^{-4}	0.158×10^{-4}	0.154×10^{-4}	0.143×10^{-4}	0.107×10^{-4}
Drywell floor drain line	Radwaste Tunnel	2-6" valve	1.875	0.435×10^{-4}	0.509×10^{-4}	0.474×10^{-4}	0.462×10^{-4}	0.430×10^{-4}	0.321×10^{-4}
Drywell floor vent line	Radwaste Tunnel	2-3" valve	0.9375	0.217×10^{-4}	0.254×10^{-4}	0.237×10^{-4}	0.231×10^{-4}	0.215×10^{-4}	0.161×10^{-4}
RWCU line	Turbine Bldg	2-8" valve	2.5	0.579×10^{-4}	0.678×10^{-4}	0.632×10^{-4}	0.616×10^{-4}	0.573×10^{-4}	0.429×10^{-4}
Feedwater line	Turbine Bldg	2-24" check valves	12	0.278×10^{-3}	0.326×10^{-3}	0.303×10^{-3}	0.296×10^{-3}	0.275×10^{-3}	0.206×10^{-3}
CPS supply line to drywell	Standby Gas Trtmt Area	2-14" valves	4.38	0.102×10^{-3}	0.119×10^{-3}	0.111×10^{-3}	0.108×10^{-3}	0.100×10^{-3}	0.751×10^{-4}
CPS supply line to drywell	Standby Gas Trtmt Area	2-2" valves	0.625	0.145×10^{-4}	0.170×10^{-4}	0.158×10^{-4}	0.154×10^{-4}	0.143×10^{-4}	0.107×10^{-4}
CPS supply line to supp. chamber[6]	Standby Gas Trtmt Area	2-12" valves	3.75	0.523×10^{-4}	0.612×10^{-4}	0.570×10^{-4}	0.556×10^{-4}	0.517×10^{-4}	0.387×10^{-4}
CPS supply line to supp. chamber[6]	Standby Gas Trtmt Area	2-2" valves	0.625	0.871×10^{-5}	0.102×10^{-4}	0.950×10^{-5}	0.926×10^{-5}	0.861×10^{-5}	0.644×10^{-5}

TABLE 6.2-55d (Cont'd.)
(Sheet 2 of 2)

Line Description	Termination Region	Bypass Leakage Barrier	Leak Rate[3] Tech. Spec. SCFH[1,4]	Leak Rate[3] Fraction/ Day[2]	Containment Bypass Leak Rate (Fraction/Day)[5][8] 0-2 hr	0-8 hr	8-24 hr	1-4 day	4-30 day
Inst. air to SRV accumulators	Yard	2-1 1/2" SOV	Combined Leakage 3.6[7]	0.834×10^{-4}	0.977×10^{-4}	0.909×10^{-4}	0.887×10^{-4}	0.825×10^{-4}	0.617×10^{-4}
Inst. air to drywell	Yard	2-1 1/2" SOV							
Inst air to drywell	Yard	2-1 1/2" SOV							
Inst. air to CPS valve in supp. chamber[9]	Yard	1-1" valve & 1-1 1/2" check valve							
Inst. air to CPS valve in supp. chamber	Yard	1-1" valve & 1-1 1/2" check valve							
N_2 purge to TIP index mechanism	Yard	1-1" valve & 1-1 1/2" check valve							
Inst. air to ADS accumulators	Yard	1-1 1/2" valve & 1-1 1/2" check valve	0.9375	0.217×10^{-4}	0.254×10^{-4}	0.237×10^{-4}	0.231×10^{-4}	0.215×10^{-4}	0.161×10^{-4}
Inst. air to ADS accumulators	Yard	1-1 1/2" valve & 1-1 1/2" check valve	0.9375	0.217×10^{-4}	0.254×10^{-4}	0.237×10^{-4}	0.231×10^{-4}	0.215×10^{-4}	0.161×10^{-4}

(1) Std. Conditions: 14.7 psia and 68°F.
(2) Fraction/Day is defined as fraction of drywell volume leakage/day per line under test conditions.
(3) Test Conditions: Air medium; 40 psig and 80°F.
(4) The leak rate is based on ASME Section XI, 1983 Edition through Summer 1983 Addenda (Subsection IWV-3426) applied to each valve, except for main steam lin5es and feedwater lines.
(5) Fraction/Day is defined as fraction of drywell volume leakage/day under LOCA conditions.
(6) Leak rate is defined as a fraction of entire primary containment volume under LOCA conditions.
(7) All these paths terminate at 2GSN-TK2 within the reactor building, and only one line goes out of the reactor building.
(8) Alternate MSIV leak rates below those indicated may be applied. (9) The instrument air bypass leakage path, through containment penetration Z-92, containing valves 2CPS*SOV133 and 2CPS*V51, has been eliminated. However, the leakage rate through this path has been retained in the analysis, as the results are more conservative.

TABLE 6.2-56
(Sheet 1 of 22)
CONTAINMENT ISOLATION PROVISIONS FOR
FLUID LINES

Pene. No.	System Designation	GDC or Reg. Guide	ESF Sys.	Fluid	Size (in)	FSAR Arrange. Figure[1]	Location of Valve Inside/ Outside Primary Containment	Length of Pipe - Cont. to Outside Isolation Valve	Type Test[1]	Poten. Bypass Lkg. Path	Number: SWEC	Number: GE	Valve[9]: Type	Oper.	Actuator Mode: Primary	Actuator Mode: Secondary	Position: Norm.[3]	Position: Shtdwn	Position: Post-Acc.	Position: Power Fail.[10]	Isolation Signal[4]	Closure Time[5,6]	Power Src.[7]	Notes
Z-1A	Main Steam Line A	55	No	Steam	26	10.1-3e, 10.1-3f	Inside Outside	 4'-11"	C C	Yes	2MSS*AOV6A 2MSS*AOV7A	B22-F022A B22-F028A	Globe Globe	AOV AOV	Air to open; spring or air to close	N/A	Open	Closed	Closed	Closed	X,D,Z, E,P,T,R, RM,AA	3 to 5 sec	N/A	8
	Main Steam Line A Drain Line				2		Outside	36'-0"	C		2MSS*MOV208	B22-F068	Globe	MOV	Elec.	Manual	Closed	Open	Closed	FAI	Z,X,D, E,AA,P,T, R,RM	12 sec	Div I	2
Z-1B	Main Steam Line B	55	No	Steam	26	10.1-3e, 10.1-3f	Inside Outside	 4'-11"	C C	Yes	2MSS*AOV6B 2MSS*AOV7B	B22-F022B B22-F028B	Globe Globe	AOV AOV	Air to open; spring or air to close	N/A	Open	Closed	Closed	Closed	X,D,Z, E,P,R,T, RM,AA	3 to 5 sec	N/A N/A	8
	Main Steam Line B Drain Line				2		Outside	36'-0"	C		2MSS*MOV208	B22-F068	Globe	MOV	Elec.	Manual	Closed	Open	Closed	FAI	Z,X,D, E,AA,P,R, T,RM	12 sec	Div I	2
Z-1C	Main Steam Line C	55	No	Steam	26	10.1-3e, 10.1-3f	Inside Outside	 4'-11"	C C	Yes	2MSS*AOV6C 2MSS*AOV7C	B22-F022C B22-F028C	Globe Globe	AOV AOV	Air to open; spring or air to close	N/A	Open	Closed	Closed	Closed	X,D,Z, E,P,T,R, RM,AA	3 to 5 sec	N/A	8
	Main Steam Line C Drain Line				2		Outside	36'-0"	C		2MSS*MOV208	B22-F068	Globe	MOV	Elec.	Manual	Closed	Open	Closed	FAI	Z,X,D, E,AA,P,R, T,RM	12 sec	Div I	2
Z-1D	Main Steam Line D	55	No	Steam	26	10.1-3e, 10.1-3f	Inside Outside	 5'-0"	C C	Yes	2MSS*AOV6D 2MSS*AOV7D	B22-F022D B22-F028D	Globe Globe	AOV AOV	Air to open; spring or air to close	N/A	Open	Closed	Closed	Closed	X,D,Z, E,P,T,R, RM,AA	3 to 5 sec	N/A	8
	Main Steam Line D Drain Line				2		Outside	36'-0"	C		2MSS*MOV208	B22-F068	Globe	MOV	Elec.	Manual	Closed	Open	Closed	FAI	Z,X,D, E,P,T,R, RM,AA	12 sec	Div I	2
Z-2	Main Steam Drain Line	55	No	Steam	6	10.1-3e, 10.1-3f	Inside Outside	 1'-2"	C C	Yes	2MSS*MOV111 2MSS*MOV112	B22-F016 B22-F019	Globe Globe	MOV MOV	Elec. Elec.	Manual Manual	Closed Closed	Open Open	Closed Closed	FAI FAI	Z,X,D, E,AA,P,T, R,RM Z,X,C,D, E,P,T,R, RM,AA	40 sec 40 sec	Div II Div I	
Z-3	Spare				2				A															
Z-4A	Feedwater Line A to RPV	55	No	Water	24	10.1-6b, 5.4-16b	Outside Inside Outside	2'-0" 21'-2"	C C C	Yes	2FWS*V23A 2FWS*V12A 2FWS*MOV21A	B22-F032A B22-F010A B22-F065A	Swing Check Swing Check Gate	N/A N/A MOV	Process Process Elec.	N/A N/A Manual	Open Open Open	Closed Closed Closed	Closed Closed Closed	N/A N/A FAI	Reverse flow Reverse flow RM	The time it takes for one valve volume to pass thru the valve N/A	N/A Div I	32,38
				Water	8	10.1-6b, 5.4-16b	Outside	57'-8"	C		2WCS*MOV200	G33-F040	Globe	MOV	Elec.	Manual	Open	Open	Closed	FAI	RM	N/A	Div I	
Z-4B	Feedwater Line B to RPV	55	No	Water	24	10.1-6b, 5.4-16b	Inside Outside Outside	 2'-0" 21'-2"	C C C	Yes	2FWS*V12B 2FWS*V23B 2FWS*MOV21B	B22-F010B B22-F032B B22-F065B	Swing Check Swing Check Gate	N/A N/A MOV	Process Process Elec.	N/A N/A Manual	Open Open Open	Closed Closed Closed	Closed Closed Closed	N/A N/A FAI	Reverse flow Reverse flow RM	The time it takes for one valve volume to pass thru the valve N/A	N/A Div I	32,38
				Water	8	10.1-6b, 5.4-16b	Outside	65'-8"	C		2WCS*MOV200	G33-F040	Globe	MOV	Elec.	Manual	Open	Open	Closed	FAI	RM	N/A	Div I	

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											Number		Valve[9]											
															Actuator Mode		Position							
Pene. No.	System Designation	GDC or Reg. Guide	ESF Sys.	Fluid	Size (in)	FSAR Arrange. Figure[1]	Location of Valve Inside/ Outside Primary Containment	Length of Pipe - Cont. to Outside Isolation Valve	Type Test[1]	Poten. Bypass Lkg. Path	SWEC	GE	Type	Oper.	Primary	Secondary	Norm.[3]	Shtdwn	Post-Acc.	Power Fail.[10]	Isolation Signal[4]	Closure Time[5,6]	Power Src.[7]	Notes
Z-5A	RHS Pump A Suction from Supp. Pool	56	Yes	Water	24	5.4-13c	Outside	8'-1"	N/A	No[29]	2RHS*MOV1A	E12-F004A	Tri-cent. Btr-fly	MOV	Elec.	Manual	Open	Closed	Open	FAI	RM	45	Div I	13,39
Z-5B	RHS Pump B Suction from Supp. Pool	56	Yes	Water	24	5.4-13f	Outside	20'-9"	N/A	No[29]	2RHS*MOV1B	E12-F004B	Tri-cent. Btr-fly	MOV	Elec.	Manual	Open	Closed	Open	FAI	RM	45	Div II	13,39
Z-5C	RHS Pump C Suction from Supp. Pool	56	Yes	Water	24	5.4-13g	Outside	9'-2"	N/A	No[29]	2RHS*MOV1C	E12-F004C	Tri-cent. Btr-fly	MOV	Elec.	Manual	Open	Closed	Open	FAI	RM	45	Div II	13,39
Z-6A	RHS Test Line Loop B to Supp. Pool	56	Yes	Water	18	5.4-13c	Outside	19'-3"	C	No[29]	2RHS*MOV30B	E12-F105B	Tri-cent. Btr-fly	MOV	Elec.	Manual	Open	Closed	Open	FAI	RM	85	Div II	15
Z-6B	RHS Test Line Loop A to Supp. Pool	56	Yes	Water	18	5.4-13c	Outside	10'-6"	C	No[29]	2RHS*MOV30A	E12-F105A	Tri-cent. Btr-fly	MOV	Elec.	Manual	Open	Closed	Open	FAI	RM	85	Div I	15
Z-7A	RHS Containment Spray Loop A to Supp. Pool	56	Yes	Water	4	5.4-13c	Outside	19'-8"	C	No[29]	2RHS*MOV33A	E12-F027A	Globe	MOV	Elec.	Manual	Closed	Closed	Open	FAI	X*,F*,RM	15	Div I	14,15, 41
Z-7B	RHS Containment Spray Loop B to Supp. Pool	56	Yes	Water	4	5.4-13c	Outside	4'-6"	C	No[29]	2RHS*MOV33B	E12-F027B	Globe	MOV	Elec.	Manual	Closed	Closed	Open	FAI	X*,F*,RM	15	Div II	14,15, 41
Z-8A	RHS Containment Spray Loop A to Drywell	56	Yes	Water	16	5.4-13a	Outside Outside	2'-0" 11'-2"	C C	No[29]	2RHS*MOV25A 2RHS*MOV15A	E12-F017A E12-F016A	Gate Gate	MOV MOV	Elec. Elec.	Manual Manual	Closed Closed	Closed Closed	Open Open	FAI FAI	RM RM	77 90	Div I Div I	13,15, 41
Z-8B	RHS Containment Spray Loop B to Drywell	56	Yes	Water	16	5.4-13b	Outside Outside	0'-0" 5'-10"	C C	No[29]	2RHS*MOV25B 2RHS*MOV15B	E12-F017B E12-F016B	Gate Gate	MOV MOV	Elec. Elec.	Manual Manual	Closed Closed	Closed Closed	Open Open	FAI FAI	RM RM	77 90	Div II Div II	13,15, 41
Z-9A	RHS/LPCI Loop A to RPV	55	Yes	Water	12	5.4-13a	Outside Inside	0'-11"	C C	No[29]	2RHS*MOV24A 2RHS*V16A	E12-F042A E12-F041A	Gate Check	MOV N/A	Elec. Process	Manual Manual	Closed Closed	Closed Closed	Open Open	FAI N/A	RM Reverse flow	25 N/A	Div I N/A	13,15, 40,41
Z-9B	RHS/LPCI Loop B to RPV	55	Yes	Water	12	5.4-13a, 5.4-13b	Outside Inside	6'-6"	C C	No[29]	2RHS*MOV24B 2RHS*V16B	E12-F042B E12-F041B	Gate Check	MOV N/A	Elec. Process	Manual Manual	Closed Closed	Closed Closed	Open Open	FAI N/A	RM Reverse flow	25 N/A	Div II N/A	13,15, 40,41
Z-9C	RHS/LPCI Loop C to RPV	55	Yes	Water	12	5.4-13a, 5.4-13b	Outside Inside	0'-11"	C C	No[29]	2RHS*MOV24C 2RHS*V16C	E12-F042C E12-F041C	Gate Check	MOV N/A	Elec. Process	Manual Manual	Closed Closed	Closed Closed	Open Open	FAI N/A	RM Reverse flow	25 N/A	Div II N/A	13,15, 40,41
Z-10A	RHS Shutdown Return Loop A to Reactor Recirc Loop A	55	No	Water	12	5.4-13a	Outside Inside	6'-0"	C C	No[29]	2RHS*MOV40A 2RHS*V39A	E12-F053A E12-F050A	Globe Check	MOV N/A	Elec. Process	Manual Manual (test only)	Closed Closed	Open Open	Closed Closed	FAI N/A	A,L,M,Z, RM,CC,DD Reverse flow	25 N/A	Div I N/A	41, 45
	RHS Shutdown Cooling Return Line Inboard Valve Bypass Line	55	No	Water	2	5.4-13a	Inside		C		2RHS*MOV67A	E12-F099A	Globe	MOV	Elec.	Manual	Closed	Closed	Closed	FAI	A,L,M,Z, RM,CC,DD	9	Div I	41, 45

Pene. No.	System Designation	GDC or Reg. Guide	ESF Sys.	Fluid	Size (in)	FSAR Arrange. Figure[1]	Location of Valve Inside/ Outside Primary Containment	Length of Pipe - Cont. to Outside Isolation Valve	Type Test[1]	Poten. Bypass Lkg. Path	Number: SWEC	Number: GE	Valve[9]: Type	Valve: Oper	Actuator Mode: Primary	Actuator Mode: Secondary	Position: Norm.[3]	Position: Shtdwn	Position: Post-Acc.	Position: Power Fail.[10]	Isolation Signal[4]	Closure Time[5,6]	Power Src.[7]	Notes
Z-10B	RHS Shutdown Return Loop B to Reactor Recirc Loop B	55	No	Water	12	5.4-13a, 5.4-13b	Outside	1'-1"	C	No[29]	2RHS*MOV40B	E12-F053B	Globe	MOV	Elec.	Manual	Closed	Open	Closed	FAI	A,L,M,Z, RM,CC,DD	25	Div II	41, 45
							Inside		C		2RHS*V39B	E12-F050B	Check	N/A	Process	Manual (test only)	Closed	Open	Closed	N/A	Reverse flow	N/A	N/A	
	RHS Shutdown Cooling Return Line Inboard Valve Bypass Line	55	No	Water	2	5.4-13a, 5.4-13b	Inside		C		2RHS*MOV67B	E12-F099B	Globe	MOV	Elec.	Manual	Closed	Closed	Closed	FAI	A,L,M,Z, RM,CC,DD	9	Div II	41, 45
Z-11	RHS Shutdown Supply from Reactor Recirc	55	No	Water	20	5.4-13a	Outside	1'-2"	C	No[29]	2RHS*MOV113	E12-F008	Gate	MOV	Elec.	Manual	Closed	Open	Closed	FAI	A,L,M,Z, RM,CC,DD	27	Div I	41, 45
							Inside		C		2RHS*MOV112	E12-F009	Gate	MOV	Elec.	Manual	Closed	Open	Closed	FAI	A,L,M,Z, RM,CC,DD	27	Div II	
							Inside		C		2RHS*RV152	E12-F231	Rlf.	N/A	Auto	N/A	Closed	Closed	Closed	N/A	N/A	N/A	N/A	
Z-12	CSH Suction from Suppression Pool	56	Yes	Water	20	6.3-6a	Outside	2'-5"	N/A	Yes[30]	2CSH*MOV118	E22-F015	Gate	MOV	Elec.	Manual	Closed	Closed	Open	FAI	RM	18	Div III	13,39
Z-13	CSH Test Return to Suppression	56	Yes	Water	12	6.3-6a, 6.3-6b	Outside	50'-0"	C	No[29]	2CSH*MOV111	E22-F023	Globe	MOV	Elec.	Manual	Closed	Closed	Closed	FAI	B*,F*,RM	60	Div III	
	HPCS Min Flow Bypass		Yes	Water	4	6.3-6a, 6.3-6b	Outside	45'-6"	C		2CSH*MOV105	E22-F012	Gate	MOV	Elec.	Manual	Closed	Closed	Closed	FAI	RM	7	Div III	
Z-14	CSH to RPV	55	Yes	Water	12	6.3-6a	Inside		C	No[29]	2CSH*V108	E22-F005	Check	N/A	Process	Manual	Closed	Closed	Open	N/A	Reverse flow	N/A	N/A	13,40, 41, 45
							Outside	1'-0"	C		2CSH*MOV107	E22-F004	Gate	MOV	Elec.	Manual	Closed	Closed	Open	FAI	RM	12	Div III	
Z-15	CSL Suction from Suppression Pool	56	Yes	Water	20	6.3-7a	Outside	7'-8"	N/A	No[29]	2CSL*MOV112	E21-F074	Btr-fly	MOV	Elec.	Manual	Open	Open	Open	FAI	RM	90	Div I	13,39
Z-16	CSL to RPV	55	Yes	Water	12	6.3-7a	Inside		C	No[29]	2CSL*V101	E21-F006	Check	N/A	Process	Manual	Closed	Closed	Open	N/A	Reverse flow	N/A	N/A	13,40, 41, 45
							Outside	1'-0"	C		2CSL*MOV104	E21-F005	Gate	MOV	Elec.	Manual	Closed	Closed	Open	FAI	RM	16	Div I	
Z-17	ICS Suction from Suppression Pool	56	No	Water	6	5.4-9a	Outside	0'-6"	N/A	Yes[30]	2ICS*MOV136	E51-F031	Gate	MOV	Elec.	Manual	Closed	Closed	Open	FAI	RM	19	125 VDC	39
Z-18	ICS Minimum Flow to Suppression Pool	56	No	Water	2	5.4-9a	Outside	2'-4"	C	No[29]	2ICS*MOV143	E51-F019	Globe	MOV	Elec.	Manual	Closed	Closed	Closed	FAI	RM	10	125 VDC	
Z-19	ICS Turbine Exhaust to Suppression Pool	56	No	Steam	12	5.4-9a	Outside	1'-6"	C	No[29]	2ICS*MOV122	E51-F068	Gate	MOV	Elec.	Manual	Open	Open	Open	FAI	RM	80	125 VDC	16
Z-20	Spare		No		3/4				A															
Z-21A	Steam to ICS Turbine and RHS Heat Exchangers	55	No	Steam	10	5.4-9a	Outside	0'-9"	C	No[29]	2ICS*MOV121	E51-F064	Gate	MOV	Elec.	Manual	Open	Closed	Open	FAI	BB,M,CC, DD,H,K, RM,Z	25	Div I	
							Inside		C		2ICS*MOV128	E51-F063	Gate	MOV	Elec.	Manual	Open	Closed	Open	FAI	BB,M,CC, DD,H,K,RM	25	Div II	
	ICS Turbine Steam Supply Bypass to Inboard Isolation Valve		No	Steam	1	5.4-9a	Inside		C	No[29]	2ICS*MOV170	E51-F076	Globe	MOV	Elec.	Manual	Closed	Closed	Closed	FAI	BB,M,CC, DD,H,K,RM	9	Div II	
Z-21B	Spare		No		4				A															

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Pene. No.	System Designation	GDC or Reg. Guide	ESF Sys.	Fluid	Size (in)	FSAR Arrange. Figure[1]	Location of Valve Inside/ Outside Primary Containment	Length of Pipe - Cont. to Outside Isolation Valve	Type Test[1]	Poten. Bypass Lkg. Path	Number		Valve[9]											
															Actuator Mode		Position							
											SWEC	GE	Type	Oper.	Primary	Secondary	Norm.[3]	Shtdwn	Post-Acc.	Power Fail.[10]	Isolation Signal[4]	Closure Time[5,6]	Power Src.[7]	Notes
Z-22	ICS to RPV	55	No	Water	6	5.4-9c, 5.4-13b	Outside	0'-6"	C	No[29]	2ICS*V156	-	Check	N/A	Process	N/A	Closed	Open	Open	Closed	Reverse flow	N/A	N/A	41
							Inside		C		2ICS*V157	-	Check	N/A	Process	N/A	Closed	Open	Open	Closed	Reverse flow	N/A	N/A	
							Outside	4'-3"	C		2ICS*MOV126	E51-F013	Gate	MOV	Elec.	Manual	Closed	Closed	Open	FAI	RM	20	Div I	
					3/4		Outside	5'-11"	C	No	2ICS*V288	-	Globe	Man.	Manual	N/A	Closed	Closed	Closed	N/A	LC	N/A		
	RHR Reactor Head Spray			Water	6	5.4-9c, 5.4-13b	Outside	29'-5"	C		2RHS*MOV104	E12-F023	Globe	MOV	Elec.	Manual	Closed	Open	Closed	FAI	A,L,M,CC, RM,DD,Z	32	Div I	41, 43
Z-23	WCS Supply from RCS & RPV	55	No	Water	8	5.4-16a	Inside		C	Yes	2WCS*MOV102	G33-F001	Globe	MOV	Elec.	Manual	Open	Open	Closed	FAI	B,J,U,S, RM,DD,Z	13	Div II	
				Water	8		Outside	1'-3"	C		2WCS*MOV112	G33-F004	Globe	MOV	Elec.	Manual	Open	Open	Closed	FAI	B,J,S,U, W,Z,RM,DD	12	Div I	
Z-24	Spare		No		3				A															
Z-25	RDS Lines to RPV		Yes																					
	53 Insert			Water	1	N/A	Outside	125'-0"		No[29]							See Note 17							
	53 With-drawal				3/4		Outside	125'-0”																
Z-26	RDS Lines to RPV		Yes																					
	39 Insert			Water	1	N/A	Outside	125'-0"		No[29]							See Note 17							
	39 With-drawal				3/4		Outside	125'-0"																
Z-27	RDS Lines to RPV		Yes																					
	54 Insert			Water	1	N/A	Outside	125'-0"		No[29]							See Note 17							
	54 With-drawal				3/4		Outside	125'-0"																
Z-28	RDS Lines to RPV		Yes																					
	39 Insert			Water	1	N/A	Outside	125'-0"		No[29]							See Note 17							
	39 With-drawal				3/4		Outside	125'-0"																
Z-29	SLCS to RPV	55	No	Boron Solu-tion	1 1/2	9.3-17a	Inside		C	No[31]	2SLS*V10	C41-F007	Check	N/A	Process	N/A	Closed	Closed	Closed	N/A	Reverse flow	N/A	N/A	
							Outside	2'-10"	C		2SLS*MOV5A	C41-F006A	Stop Check Globe	MOV	Elec.	Manual	Closed	Closed	Closed	FAI	Reverse flow, RM	N/A	N/A	36
							Outside	3'-10"	C		2SLS*MOV5B	C41-F006B	Stop Check Globe	MOV	Elec.	Manual	Closed	Closed	Closed	FAI	Reverse Flow, RM	N/A	N/A	36
Z-30A	Spare		No		3				A															
Z-30B	Spare		No		3				A															
Z-31A	TIP Drive Guide Tube to RPV	1.11	No	Note 19	1 1/2	6.2-70	Outside	2'-4"	C	No[31]	2NMS*SOV1A	C51-J004A	Ball	SOV	Elec.	N/A	Closed	Closed	Closed	Closed	B,F,RM,Z	N/A	120 VAC	18,19, 28,34
							Outside				2NMS*VEX1A	C51-J004A	Shear	N/A	N/A	N/A	Open	Open	Open	Open	RM	N/A	125 VDC	
Z-31B	TIP Drive Guide Tube to RPV	1.11	No	Note 19	1 1/2	6.2-70	Outside	5'-4"	C	No[31]	2NMS*SOV1B	C51-J004B	Ball	SOV	Elec.	N/A	Closed	Closed	Closed	Closed	B,F,RM,Z	N/A	120 VAC	18,19, 28,34
							Outside				2NMS*VEX1B	C51-J004B	Shear	N/A	N/A	N/A	Open	Open	Open	Open	RM	N/A	125 VDC	
Z-31C	TIP Drive Guide Tube to RPV	1.11	No	Note 19	1 1/2	6.2-70	Outside	2'-4"	C	No[31]	2NMS*SOV1C	C51-J004C	Ball	SOV	Elec.	N/A	Closed	Closed	Closed	Closed	B,F,RM,Z	N/A	120 VAC	18,19, 28,34
							Outside				2NMS*VEX1C	C51-J004C	Shear	N/A	N/A	N/A	Open	Open	Open	Open	RM	N/A	125 VDC	
Z-31D	TIP Drive Guide Tube to RPV	1.11	No	Note 19	1 1/2	6.2-70	Outside	2'-4"	C	No[31]	2NMS*SOV1D	C51-J004D	Ball	SOV	Elec.	N/A	Closed	Closed	Closed	Closed	B,F,RM,Z	N/A	120 VAC	18,19, 28,34
							Outside				2NMS*VEX1D	C51-J004D	Shear	N/A	N/A	N/A	Open	Open	Open	Open	RM	N/A	125 VDC	
Z-31E	TIP Drive Guide Tube to RPV	1.11	No	Note 19	1 1/2	6.2-70	Outside	2'-7"	C	No[31]	2NMS*SOV1E	C51-J004E	Ball	SOV	Elec.	N/A	Closed	Closed	Closed	Closed	B,F,RM,Z	N/A	120 VAC	18,19, 28,34
							Outside				2NMS*VEX1E	C51-J004E	Shear	N/A	N/A	N/A	Open	Open	Open	Open	RM	N/A	125 VDC	

TABLE 6.2-56
(Sheet 5 of 22)

Pene. No.	System Designation	GDC or Reg. Guide	ESF Sys.	Fluid	Size (in)	FSAR Arrange. Figure[1]	Location of Valve Inside/ Outside Primary Containment	Length of Pipe - Cont. to Outside Isolation Valve	Type Test[1]	Poten. Bypass Lkg. Path	Number SWEC	Number GE	Valve[9] Type	Oper.	Actuator Mode Primary	Actuator Mode Secondary	Position Norm.[3]	Position Shtdwn	Position Post-Acc.	Position Power Fail.[10]	Isolation Signal[4]	Closure Time[5,6]	Power Src.[7]	Notes
Z-32	N_2 Purge to TIP Index Mechanism	56	No	N_2	1 1/2	9.3-20b	Outside	7'-6"	C	Yes	2GSN*SOV166	-	Globe	SOV	Elec.	N/A	Open	Closed	Closed	Closed	B,F,RM,Z	5	120 VAC	
							Inside	-	C		2GSN*V170	-	Check	N/A	Process	N/A	Open	Closed	Closed	N/A	Reverse flow	N/A	N/A	

TABLE 6.2-56
(Sheet 6 of 22)

Pene. No.	System Designation	GDC or Reg. Guide	ESF Sys.	Fluid	Size (in)	FSAR Arrange. Figure[1]	Location of Valve Inside/ Outside Primary Containment	Length of Pipe - Cont. to Outside Isolation Valve	Type Test[1]	Poten. Bypass Lkg. Path	Number: SWEC	Number: GE	Valve[9]: Type	Valve: Oper.	Actuator Mode: Primary	Actuator Mode: Secondary	Position: Norm.[3]	Position: Shtdwn	Position: Post-Acc.	Position: Power Fail.[10]	Isolation Signal[4]	Closure Time[5,6]	Power Src.[7]	Notes
Z-33A	CCP Supply to RCS Pump A	56	No	Water	4	9.2-3d, 9.2-1m	Inside Outside Inside	7'-0"	C C C	No[31]	2CCP*MOV94A 2CCP*MOV17A 2CCP*RV1019A	- - -	Gate Gate Rlf.	MOV MOV N/A	Elec. Elec. Auto	Manual Manual N/A	Open Open Closed	Open Open Closed	Closed Closed Closed	FAI FAI N/A	B,F,RM,Z B,F,RM,Z N/A	26 26 N/A	Div II Div I N/A	6
Z-33B	CCP to RCS Pump B	56	No	Water	4	9.2-3b, 9.2-1f	Inside Outside Inside	7'-0"	C C C	No[31]	2CCP*MOV94B 2CCP*MOV17B 2CCP*RV170	- - -	Gate Gate Rlf.	MOV MOV N/A	Elec. Elec. Auto	Manual Manual N/A	Open Open Closed	Open Open Closed	Closed Closed Closed	FAI FAI N/A	B,F,RM,Z B,F,RM,Z N/A	26 26 N/A	Div II Div I N/A	
Z-34A	CCP Return from RCS Pump A	56	No	Water	4	9.2-3d	Inside Outside Inside	7'-0"	C C C	No[31]	2CCP*MOV16A 2CCP*MOV15A 2CCP*RV1020A	- - -	Gate Gate Rlf.	MOV MOV N/A	Elec. Elec. Auto	Manual Manual N/A	Open Open Closed	Open Open Closed	Closed Closed Closed	FAI FAI N/A	B,F,RM,Z B,F,RM,Z N/A	26 26 N/A	Div II Div I N/A	
Z-34B	CCP Return from RCS Pump B	56	No	Water	4	9.2-3a	Inside Outside Inside	7'-0"	C C C	No[31]	2CCP*MOV16B 2CCP*MOV15B 2CCP*RV171	- - -	Gate Gate Rlf.	MOV MOV N/A	Elec. Elec. Auto	Manual Manual N/A	Open Open Closed	Open Open Closed	Closed Closed Closed	FAI FAI N/A	B,F,RM,Z B,F,RM,Z N/A	26 26 N/A	Div II Div I N/A	
Z-35	Spare				4				A															
Z-36	Service Air to Drywell	56	No	Air	2	9.3-1j	Outside Inside	0'-7"	C C	No[31]	2SAS*HCV161 2SAS*HCV163	- -	Globe Globe	Man. Man.	Manual Manual	N/A N/A	Closed Closed	Open Open	Closed Closed	N/A N/A	LMC,LC LMC,LC	N/A N/A	N/A N/A	
Z-37	Breathing Air to Drywell	56	No	Air	2	9.3-3e	Outside Inside	0'-7"	C C	No[31]	2AAS*HCV134 2AAS*HCV136	- -	Globe Globe	Man. Man.	Manual Manual	N/A N/A	Closed Closed	Open Open	Closed Closed	N/A N/A	LMC,LC LMC,LC	N/A N/A	N/A N/A	
Z-38A	RDS to Recirc Pump A Seal	55	No	Water	3/4	5.4-2b	Inside Outside Outside	 0'-0" 33'-0"	C C C	No[29]	2RCS*V60A 2RCS*V90A 2RCS*V59A	B35-F013A B35-F009A B35-F017A	Check Check Check	N/A N/A N/A	Process Process Process	N/A N/A N/A	Open Open Open	Closed Closed Closed	Closed Closed Closed	N/A N/A N/A	Reverse flow Reverse flow Reverse flow	N/A N/A N/A	N/A	
Z-38B	RDS to Recirc Pump B Seal	55	No	Water	3/4	5.4-2c	Inside Outside Outside	 0'-0" 31'-0"	C C C	No[29]	2RCS*V60B 2RCS*V90B 2RCS*V59B	B35-F013B B35-F009B B35-F017B	Check Check Check	N/A N/A N/A	Process Process Process	N/A N/A N/A	Open Open Open	Closed Closed Closed	Closed Closed Closed	N/A N/A N/A	Reverse flow Reverse flow Reverse flow	N/A N/A N/A	N/A	
Z-39	Drywell Floor Drain Line	56	No	Water	6	9.3-9e	Inside Outside Inside	1'-6"	C C C	Yes	2DFR*MOV121 2DFR*MOV120 2DFR*RV228	- - -	Gate Gate Rlf.	MOV MOV N/A	Elec. Elec. Auto	Manual Manual N/A	Open Open Closed	Closed Closed Closed	Closed Closed Closed	FAI FAI N/A	B,F,RM,Z B,F,RM,Z N/A	36 28 N/A	Div II Div I N/A	
Z-40	Equipment Drains from Drywell	56	No	Water	4	9.3-9f	Inside Outside Inside	4'-2"	C C C	Yes	2DER*MOV119 2DER*MOV120 2DER*RV344	- - -	Gate Gate Rlf.	MOV MOV N/A	Elec. Elec. Auto	Manual Manual N/A	Open Open Closed	Closed Closed Closed	Closed Closed Closed	FAI FAI N/A	B,F,RM,Z B,F,RM,Z N/A	22 22 N/A	Div II Div I N/A	
Z-41	Reactor Coolant Recirc to Sample Cooler/CAVS	55	No	Water	3/4	5.4-2b	Inside Outside	0'-0"	C C	No[31]	2RCS*SOV104 2RCS*SOV105	B35-F019 B35-F020	Globe Globe	SOV SOV	Elec. Elec.	N/A N/A	Open Open	Closed Closed	Closed Closed	Closed Closed	B,RM,Z B,RM,Z	5 5	Div II Div I	
Z-42A	Fire Protection for Reactor Recirc Pump	56	No	Air	2	9.5-1g	Inside Outside	3'-0"	C C	No[31]	2FPW*SOV219 2FPW*SOV218	-	Globe Globe	SOV SOV	N/A N/A	N/A N/A	Closed Closed	Closed Closed	Closed Closed	N/A N/A	N/A N/A	N/A N/A	N/A N/A	35 35
Z-42B	Fire Protection Water for Reactor Recirc Pump	56	No	Air	2	9.5-1g	Inside Outside	3'-0"	C C	No[31]	2FPW*SOV221 2FPW*SOV220	- -	Globe Globe	SOV SOV	N/A N/A	N/A N/A	Closed Closed	Closed Closed	Closed Closed	N/A N/A	N/A N/A	N/A N/A	N/A N/A	35 35
Z-43	Drywell Floor Drain Tank Vent	56	No	Air	3	9.3-9c, 9.3-9e	Inside Outside	20'-10"	C C	Yes	2DFR*MOV140 2DFR*MOV139	- -	Gate Gate	MOV MOV	Elec. Elec.	Manual Manual	Open Open	Closed Closed	Closed Closed	FAI FAI	B,F,RM,Z B,F,RM,Z	13 13	Div II Div I	
Z-44A	Capped Spare				3				A															
Z-44B	Capped Spare				3				A															
Z-44C	Capped Spare				3				A															
Z-44D	Capped Spare				3				A															
Z-44E	Service Air to Drywell	56	No	Air	2	9.3-1j	Outside Inside	0'-5"	C C	No[31]	2SAS*HCV160 2SAS*HCV162	- -	Globe Globe	Man. Man.	Manual Manual	N/A N/A	Closed Closed	Open Open	Closed Closed	N/A N/A	LMC,LC LMC,LC	N/A N/A	N/A N/A	
Z-44F	Breathing Air to Drywell	56	No	Air	2	9.3-3e	Outside Inside	0'-5"	C C	No[31]	2AAS*HCV135 2AAS*HCV137	- -	Globe Globe	Man. Man.	Manual Manual	N/A N/A	Closed Closed	Open Open	Closed Closed	N/A N/A	LMC,LC LMC,LC	N/A N/A	N/A N/A	

TABLE 6.2-56
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Pene. No.	System Designation	GDC or Reg. Guide	ESF Sys.	Fluid	Size (in)	FSAR Arrange. Figure[1]	Location of Valve Inside/ Outside Primary Containment	Length of Pipe - Cont. to Outside Isolation Valve	Type Test[1]	Poten. Bypass Lkg. Path	Number		Valve[9]											
															Actuator Mode		Position							
											SWEC	GE	Type	Oper.	Primary	Secondary	Norm.[3]	Shtdwn	Post-Acc.	Power Fail.[10]	Isolation Signal[4]	Closure Time[5,6]	Power Src.[7]	Notes
Z-45	Equipment Drain Tank (2DER-TK1) Vent to Drywell	56	No	Air	2	9.3-9f	Inside Outside	0’-0”	C C	Yes	2DER*MOV130 2DER*MOV131	- -	Globe Globe	MOV MOV	Elec. Elec.	Manual Manual	Open Open	Closed Closed	Closed Closed	FAI FAI	B,F,RM,Z B,F,RM,Z	9 9	Div II Div I	
Z-46A	CCP Supply to Drywell Space Cooler	56	No	Water	8	9.2-3c	Inside Outside Inside	7'-0"	C C C	No[31]	2CCP*MOV273 2CCP*MOV265 2CCP*RV1021A	- - -	Gate Gate Rlf.	MOV MOV N/A	Elec. Elec. Auto	Manual Manual N/A	Open Open Closed	Open Open Closed	Closed Closed Closed	FAI FAI N/A	B,F,RM,Z B,F,RM,Z N/A	36 38 N/A	Div II Div I N/A	
Z-46B	Capped Spare				4				A															
Z-46C	Fire Protection Water for Containment Hose Reel Standpipe						See Note 20		B	No[31]														
Z-46D	Capped Spare				4				A															
Z-47	CCP Return from Drywell Space Cooler	56	No	Water	8	9.2-3c	Inside Outside Inside	7'-3"	C C C	No[31]	2CCP*MOV122 2CCP*MOV124 2CCP*RV1022A	- - -	Gate Gate Rlf.	MOV MOV N/A	Elec. Elec. Auto	Manual Manual N/A	Open Open Closed	Open Open Closed	Closed Closed Closed	FAI FAI N/A	B,F,RM,Z B,F,RM,Z N/A	38 36 N/A	Div II Div I N/A	
Z-48	Purge Exhaust from Drywell	56	No	Air	14	9.4-8k	Inside Outside	- 7'-4"	C C	No[31]	2CPS*AOV108 2CPS*AOV110	- -	Btr-fly Btr fly	AOV AOV	Pneumatic Pneumatic	Manual Manual	Closed Closed	Closed Closed	Closed Closed	Closed Closed	B,F,Y,RM, Z B,F,Y,RM, Z	5 5	Div II Div I	21
Z-49	Purge Inlet to Drywell	56	No	Air/ N_2	14	9.4-8k	Inside Outside	- 4'-0"	C C	Yes	2CPS*AOV106 2CPS*AOV104	- -	Btr-fly Btr-fly	AOV AOV	Pneumatic Pneumatic	Manual Manual	Closed Closed	Closed Closed	Closed Closed	Closed Closed	B,F,Y,RM, Z B,F,Y,RM, Z	5 5	Div II Div I	21
Z-50	Purge Inlet to Wetwell	56	No	Air/ N_2	12	9.4-8k	Inside Outside	- 4'-3"	C C	Yes	2CPS*AOV107 2CPS*AOV105	- -	Btr-fly Btr-fly	AOV AOV	Pneumatic Pneumatic	Manual Manual	Closed Closed	Closed Closed	Closed Closed	Closed Closed	B,F,Y,RM, Z B,F,Y,RM, Z	5 5	Div II Div I	21
Z-51	Purge Exhaust from Wetwell	56 See Note 44	No	Air	12	9.4-8k	Outside Outside	6'-0" 11'-10"	B,C C	No[31]	2CPS*AOV109 2CPS*AOV111	- -	Btr-fly Btr-fly	AOV AOV	Pneumatic Pneumatic	Manual Manual	Closed Closed	Closed Closed	Closed Closed	Closed Closed	B,F,Y,RM, Z B,F,Y,RM, Z	5 5	Div II Div I	21
Z-52A	Capped Spare				1				A															
Z-52B	Capped Spare				1				A															
Z-53A	Instrument Air to ADS Valve Accumulators	56	No	N_2	1 1/2	9.3-1d	Outside Inside	1'-0"	C C	Yes	2IAS*SOV164 2IAS*V448		Globe Check	SOV N/A	Elec. Process	N/A N/A	Open Open	Open Open	Open Open	Closed N/A	B,F,RM,Z Reverse flow	5 N/A	Div I N/A	
Z-53B	Instrument Air to ADS Valve Accumulators	56	No	N_2	1 1/2	9.3-1f	Outside Inside	1'-0"	C C	Yes	2IAS*SOV165 2IAS*V449	- -	Globe Check	SOV N/A	Elec. Process	N/A N/A	Open Open	Open Open	Open Open	Closed N/A	B,F,RM,Z Reverse flow	5 N/A	Div II N/A	
Z-53C	Instrument Air to MSRV Accumulator Tank	56	No	N_2	1 1/2	9.3-1d	Outside Inside	1'-0"	C C	Yes	2IAS*SOV166 2IAS*SOV184	- -	Globe Globe	SOV SOV	Elec. Elec.	N/A N/A	Open Open	Open Open	Closed Closed	Closed Closed	B,F,RM,Z B,F,RM,Z	5 5	Div I Div II	
Z-54A	Capped Spare				3				A															
Z-55A	Hydrogen Recombiner 1A Supply to Wetwell	56	Yes	Air	3	6.2-72a	Inside Outside	2'-0"	A,C A,C	No[31]	2HCS*MOV4A 2HCS*MOV1A	- -	Gate Gate	MOV MOV	Elec. Elec.	Manual Manual	Closed Closed	Closed Closed	Open Open	FAI FAI	B,F,RM,Z B,F,RM,Z	19 19	Div I Div I	12,22
Z-55B	Hydrogen Recombiner 1B Supply to Wetwell	56	Yes	Air	3	6.2-72a	Inside Outside	2'-0"	A,C A,C	No[31]	2HCS*MOV4B 2HCS*MOV1B	- -	Gate Gate	MOV MOV	Elec. Elec.	Manual Manual	Closed Closed	Closed Closed	Open Open	FAI FAI	B,F,RM,Z B,F,RM,Z	19 19	Div II Div II	12,22
Z-56A	Hydrogen Recombiner 1A Return from Drywell	56	Yes	Air	3	6.2-72a	Inside Outside	2'-0"	A,C A,C	No[31]	2HCS*MOV6A 2HCS*MOV3A	- -	Gate Gate	MOV MOV	Elec. Elec.	Manual Manual	Closed Closed	Closed Closed	Open Open	FAI FAI	B,F,RM,Z B,F,RM,Z	19 19	Div I Div I	12,22

TABLE 6.2-56
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Pene. No.	System Designation	GDC or Reg. Guide	ESF Sys.	Fluid	Size (in)	FSAR Arrange. Figure[1]	Location of Valve Inside/ Outside Primary Containment	Length of Pipe - Cont. to Outside Isolation Valve	Type Test[1]	Poten. Bypass Lkg. Path	Number		Valve[9]												
															Actuator Mode		Position							
											SWEC	GE	Type	Oper.	Primary	Secondary	Norm.[3]	Shtdwn	Post-Acc.	Power Fail.[10]	Isolation Signal[4]	Closure Time[5,6]	Power Src.[7]	Notes
Z-56B	Hydrogen Recombiner 1B Return from Drywell	56	Yes	Air	3	6.2-72a	Inside Outside	 2'-0"	A,C A,C	No[31]	2HCS*MOV6B 2HCS*MOV3B	- -	Gate Gate	MOV MOV	Elec. Elec.	Manual Manual	Closed Closed	Closed Closed	Open Open	FAI FAI	B,F,RM,Z B,F,RM,Z	19 19	Div II Div II	12,22
Z-57A	Hydrogen Recombiner 1A Return from Wetwell	56	Yes	Air	3	6.2-72a	Inside Outside	 2'-0"	A,C A,C	No[31]	2HCS*MOV5A 2HCS*MOV2A	- -	Globe Globe	MOV MOV	Elec. Elec.	Manual Manual	Closed Closed	Closed Closed	Open Open	FAI FAI	B,F,RM,Z B,F,RM,Z	19 19	Div I Div I	12,22
Z-57B	Hydrogen Recombiner 1B Return from Wetwell	56	Yes	Air	3	6.2-72a	Inside Outside	 2'-0"	A,C A,C	No[31]	2HCS*MOV5B 2HCS*MOV2B	- -	Globe Globe	MOV MOV	Elec. Elec.	Manual Manual	Closed Closed	Closed Closed	Open Open	FAI FAI	B,F,RM,Z B,F,RM,Z	19 19	Div II Div II	12,22
Z-58	Containment Purge to Drywell	56	No	Air	2	9.4-8k	Inside Outside	 3'-4"	C C	Yes	2CPS*SOV122 2CPS*SOV120	- -	Globe Globe	SOV SOV	Elec. Elec.	N/A N/A	Closed Closed	Closed Closed	Closed Closed	Closed Closed	B,F,Y,RM,Z B,F,Y,RM,Z	2 2	Div II Div I	
Z-59	Containment Purge to Wetwell	56	No	Air	2	9.4-8k	Inside Outside	 14'-6"	C C	Yes	2CPS*SOV121 2CPS*SOV119	- -	Globe Globe	SOV SOV	Elec. Elec.	N/A N/A	Closed Closed	Closed Closed	Closed Closed	Closed Closed	B,F,Y,RM,Z B,F,Y,RM,Z	2 2	Div II Div I	
Z-60A	CMS from Drywell	56	No	Air	3/4	6.2-71a	Inside Outside	 1'-2"	C C	No[31]	2CMS*SOV61A 2CMS*SOV60A	- -	Globe Globe	SOV SOV	Elec. Elec.	N/A N/A	Open Open	Closed Closed	Closed Open	Closed Closed	B,F,RM,Z B,F,RM,Z	5 5	Div II Div I	12
Z-60B	CMS from Drywell	56	Yes	Air	3/4	6.2-71a	Inside Outside	 1'-2"	C C	Yes[33]	2CMS*SOV24A 2CMS*SOV24C	- -	Globe Globe	SOV SOV	Elec. Elec.	N/A N/A	Open Open	Closed Closed	Closed Closed	Closed Closed	B,F,RM,Z B,F,RM,Z	5 5	Div I Div I	12
Z-60C	CMS to Drywell	56	No	Air	3/4	6.2-71a	Inside Outside	 0'-3"	C C	No[31]	2CMS*SOV63A 2CMS*SOV62A	- -	Globe Globe	SOV SOV	Elec. Elec.	N/A N/A	Open Open	Closed Closed	Closed Closed	Closed Closed	B,F,RM,Z B,F,RM,Z	5 5	Div II Div I	12
Z-60D	CMS to Drywell	56	Yes	Air	3/4	6.2-71a	Inside Outside	 0'-4"	C C	Yes[33]	2CMS*SOV33A 2CMS*SOV32A	- -	Globe Globe	SOV SOV	Elec. Elec.	N/A N/A	Open Open	Closed Closed	Closed Closed	Closed Closed	B,F,RM,Z B,F,RM,Z	5 5	Div I Div I	12
Z-60E	CMS from Drywell	56	No	Air	3/4	6.2-71a	Inside Outside	 0'-7"	C C	No[31]	2CMS*SOV61B 2CMS*SOV60B	- -	Globe Globe	SOV SOV	Elec. Elec.	N/A N/A	Open Open	Closed Closed	Closed Closed	Closed Closed	B,F,RM,Z B,F,RM,Z	5 5	Div II Div I	12
Z-60F	CMS from Drywell	56	Yes	Air	3/4	6.2-71a	Inside Outside	 0'-7"	C C	Yes[33]	2CMS*SOV24B 2CMS*SOV24D	- -	Globe Globe	SOV SOV	Elec. Elec.	N/A N/A	Open Open	Closed Closed	Closed Closed	Closed Closed	B,F,RM,Z B,F,RM,Z	5 5	Div II Div II	12
Z-60G	CMS to Drywell	56	No	Air	3/4	6.2-71a	Inside Outside	 0'-7"	C C	No[31]	2CMS*SOV63B 2CMS*SOV62B	- -	Globe Globe	SOV SOV	Elec. Elec.	N/A N/A	Open Open	Closed Closed	Closed Closed	Closed Closed	B,F,RM,Z B,F,RM,Z	5 5	Div II Div I	12
Z-60H	CMS to Drywell	56	Yes	Air	3/4	6.2-71a	Inside Outside	 1'-0"	C C	Yes[33]	2CMS*SOV33B 2CMS*SOV32B	- -	Globe Globe	SOV SOV	Elec. Elec.	N/A N/A	Open Open	Closed Closed	Closed Closed	Closed Closed	B,F,RM,Z B,F,RM,Z	5 5	Div II Div II	12
Z-61A	Capped Spare				3/4				A															
Z-61B	CMS from Wetwell	56	Yes	Air	3/8	6.2-71b	Inside Outside	 15'-0"	C C	No[31]	2CMS*SOV26A 2CMS*SOV26C	- -	Globe Globe	SOV SOV	Elec. Elec.	N/A N/A	Open Closed	Closed Closed	Closed Closed	Closed Closed	B,F,RM,Z B,F,RM,Z	5 5	Div I Div I	12
Z-61C	CMS to Wetwell	56	Yes	Air	3/4	6.2-71b	Inside Outside	 18'-3"	C C	No[31]	2CMS*SOV34A 2CMS*SOV35A	- -	Globe Globe	SOV SOV	Elec. Elec.	N/A N/A	Open Closed	Closed Closed	Closed Closed	Closed Closed	B,F,RM,Z B,F,RM,Z	5 5	Div I Div I	12
Z-61D	Capped Spare				3/4				A															
Z-61E	CMS from Wetwell	56	Yes	Air	3/4	6.2-71b	Inside Outside	 0'-4"	C C	No[31]	2CMS*SOV26B 2CMS*SOV26D	- -	Globe Globe	SOV SOV	Elec. Elec.	N/A N/A	Open Closed	Closed Closed	Closed Open	Closed Closed	B,F,RM,Z B,F,RM,Z	5 5	Div II Div II	12
Z-61F	CMS to Wetwell	56	Yes	Air	3/4	6.2-71b	Inside Outside	 0'-4"	C C	No[31]	2CMS*SOV34B 2CMS*SOV35B	- -	Globe Globe	SOV SOV	Elec. Elec.	N/A N/A	Open Closed	Closed Closed	Closed Closed	Closed Closed	B,F,RM,Z B,F,RM,Z	5 5	Div II Div II	
Z-67	Spare				10				A															
Z-68	Capped Spare				10				A															
Z-69	Spare				6				A															
Z-70	Capped Spare				6				A															
Z-71	Spare				3				A															
Z-72	Capped Spare				14				A															
Z-73	RHS Relief Valve Discharge to Supp. Pool	56	No	Water	6	5.4-13b, 5.4-13c, 5.4-13d	Outside Outside	48'-6"	N/A N/A	No[29] No[29]	2RHS*RV108 2RHS*RV20C	E12-F036 E12-F025C	RV RV	N/A N/A	N/A N/A	N/A N/A	N/A N/A	N/A N/A	N/A N/A	N/A N/A	None None	N/A N/A	N/A N/A	25 25

TABLE 6.2-56

Pene. No.	System Designation	GDC or Reg. Guide	ESF Sys.	Fluid	Size (in)	FSAR Arrange. Figure[1]	Location of Valve Inside/ Outside Primary Containment	Length of Pipe - Cont. to Outside Isolation Valve	Type Test[1]	Poten. Bypass Lkg. Path	Number		Valve[9]											
															Actuator Mode		Position							
											SWEC	GE	Type	Oper.	Primary	Secondary	Norm.[3]	Shtdwn	Post-Acc.	Power Fail.[10]	Isolation Signal[4]	Closure Time[5,6]	Power Src.[7]	Notes
Z-74	Flanged Spare				6				B															
Z-75	Capped Spare				3				A															
Z-76	Capped Spare				3				A															
Z-77	Capped Spare				1 1/2				A															
Z-78	Capped Spare				1 1/2				A															
Z-79	Capped Spare				1 1/2				A															
Z-80	Spent Fuel Pool Cooling	56	No	Water	1 1/2	9.1-5c	Outside Inside	1'-6"	C C	No[31]	2SFC*V203 2SFC*V204	- -	Globe Globe	Man. Man.	Manual Manual	N/A N/A	Closed Closed	Closed Closed	Closed Closed	N/A N/A	LC LC	N/A N/A	N/A N/A	
Z-81	Capped Spare				1 1/2				A															
Z-82	Capped Spare				1				A															
Z-83	Capped Spare				1				A															
Z-85	Capped Spare				1				A															
Z-86	Capped Spare				1				A															
Z-87	Capped Spare				1				A															
Z-88A	RHS Safety Valve Discharge to Supp. Pool	56	Yes	Steam	12	5.4-13c, 5.4-13d	Outside	116'-2"	N/A	No[29]							See Note 23							39
Z-88B	RHR Safety Valve Discharge to Supp. Pool	56	Yes	Steam	12	5.4-13c, 5.4-13e	Outside	106'-3"	N/A	No[29]							See Note 24							39
Z-89A	LMS from Drywell	56	No	Air	3/4	6.2-73a	Inside Outside	 0'-2"	C C	No[31]	2LMS*SOV152 2LMS*SOV153	- -	Globe Globe	SOV SOV	Elec. Elec.	N/A N/A	Closed Closed	Closed Closed	Closed Closed	Closed Closed	B,F,RM,Z B,F,RM,Z	5 5	Div II Div I	
Z-89B	Capped Spare				3/4				A															
Z-89C	LMS from Wetwell	56	No	Air	3/4	6.2-73a	Inside Outside	 0'-2"	C C	No[31]	2LMS*SOV156 2LMS*SOV157	- -	Globe Globe	SOV SOV	Elec. Elec.	N/A N/A	Closed Closed	Closed Closed	Closed Closed	Closed Closed	B,F,RM,Z B,F,RM,Z	5 5	Div II Div I	
Z-89D	Capped Spare				3/4				A															
Z-90	ICS Vacuum Breaker	56	No	Air	1 1/2 3/4	5.4-9a, 5.4-13e	Outside Outside Outside	23'-10" 29'-11" 26'-6"	C C C	No[29]	2ICS*MOV148 2ICS*MOV164 2RHS*V192	E51-F086 E51-F080 E51-F102	Globe Globe Globe	MOV MOV Man.	Elec. Elec. Manual	Manual Manual N/A	Open Open Closed	Closed Closed Closed	Open Open Closed	FAI FAI N/A	F&H,RM F&H,RM LC	14 14 N/A	Div II Div I N/A	42 42
Z-91A	Instrument Air to Drywell	56	No	N_2	1 1/2	9.3-1g	Outside Inside	1'-0"	C C	Yes	2IAS*SOV167 2IAS*SOV185	- -	Globe Globe	SOV SOV	Elec. Elec.	N/A N/A	Closed Closed	Closed Closed	Closed Closed	Closed Closed	B,F,RM,Z B,F,RM,Z	5 5	Div I Div II	
Z-91B	Instrument Air to Drywell	56	No	N_2	1 1/2	9.3-1g	Outside Inside	1'-0"	C C	Yes	2IAS*SOV168 2IAS*SOV180	- -	Globe Globe	SOV SOV	Elec. Elec.	N/A N/A	Closed Closed	Closed Closed	Closed Closed	Closed Closed	B,F,RM,Z B,F,RM,Z	5 5	Div I Div II	
Z-91C	Capped Spare				1 1/2				A															
Z-91D	Capped Spare				1 1/2				A															
Z-92	Capped Spare				1	9.4-8k			A															
Z-96	N_2 Supply to Actuators for 2CPS*AOV107	56	No	Air/ N_2	1	9.4-8k	Outside Inside	19'-4" -	C C	Yes	2CPS*SOV132 2CPS*V50	- -	Globe Check	SOV N/A	Elec. Process	N/A N/A	Closed Closed	Closed Closed	Closed Closed	Closed N/A	B,F,Y,RM, Z Reverse flow	5 N/A	Div II N/A	
Z-98A	RHR Relief Valve Discharge to Supp. Pool	56	Yes	Water	3	5.4-13c, 5.4-13f, 6.3-7a	Outside	207'-6"	N/A	No[29]	2CSL*RV123 2CSL*RV105 2RHS*RV61A 2RHS*RV110 2RHS*RV139 2RHS*RV20A	E21-F031 E21-F018 E12-F088A E12-F005 E12-F030 E12-F025A	Rlf Vlvs	N/A	N/A	N/A	N/A	N/A	N/A	N/A	None	N/A	N/A	25

TABLE 6.2-56
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Pene. No.	System Designation	GDC or Reg. Guide	ESF Sys.	Fluid	Size (in)	FSAR Arrange. Figure[1]	Location of Valve Inside/ Outside Primary Containment	Length of Pipe - Cont. to Outside Isolation Valve	Type Test[1]	Poten. Bypass Lkg. Path	Number		Valve[9]										
															Actuator Mode		Position							
											SWEC	GE	Type	Oper.	Primary	Secondary	Norm.[3]	Shtdwn	Post-Acc.	Power Fail.[10]	Isolation Signal[4]	Closure Time[5,6]	Power Src.[7]	Notes
Z-98B	RHR Relief Valve Discharge to Supp. Pool	56	Yes	Water	3	5.4-13b, 5.4-13c, 5.4-13f, 5.4-13g, 6.3-6b	Outside	89'-8"	N/A	No[29]	2CSH*RV114 2CSH*RV113 2RHS*RV61B 2RHS*RV61C 2RHS*RV20B	E22-F035 E22-F014 E12-F088B E12-F088C E12-F025B	Rlf Vlvs	N/A	N/A	N/A	N/A	N/A	N/A	N/A	None	N/A	N/A	25
Z-99A	Hydraulic Unit from Recirc FCV HYV 17A (Drain Line)	56	No	Hydr.	3/4	5.4-2a	Outside Inside	0'-0" 0'-0"	N/A	No[31]	2RCS*SOV68A 2RCS*SOV82A	-	Globe Globe	SOV SOV	Elec. Elec.	N/A N/A	Open Open	Closed Closed	Closed Closed	Closed Closed	B,F,RM,Z B,F,RM,Z	10 10	Div I Div II	26
Z-99B	Hydraulic Unit to Recirc FCV HYV 17A (Open Line)	56	No	Hydr.	1	5.4-2a	Outside Inside	0'-0" 0'-0"	N/A	No[31]	2RCS*SOV67A 2RCS*SOV81A	-	Globe Globe	SOV SOV	Elec. Elec.	N/A N/A	Open Open	Closed Closed	Closed Closed	Closed Closed	B,F,RM,Z B,F,RM,Z	10 10	Div I Div II	26
Z-99C	Hydraulic Unit to Recirc FCV HYV 17A (Pilot Line)	56	No	Hydr.	1	5.4-2a	Outside Inside	0'-0" 0'-0"	N/A	No[31]	2RCS*SOV66A 2RCS*SOV80A	-	Globe Globe	SOV SOV	Elec. Elec.	N/A N/A	Open Open	Closed Closed	Closed Closed	Closed Closed	B,F,RM,Z B,F,RM,Z	10 10	Div I Div II	26
Z-99D	Hydraulic Unit to Recirc FCV HYV 17A (Closed Line)	56	No	Hydr.	1	5.4-2a	Outside Inside	0'-0" 0'-0"	N/A	No[31]	2RCS*SOV65A 2RCS*SOV79A	-	Globe Globe	SOV SOV	Elec. Elec.	N/A N/A	Open Open	Closed Closed	Closed Closed	Closed Closed	B,F,RM,Z B,F,RM,Z	10 10	Div I Div II	26
Z-100A	Hydraulic Unit to Recirc FCV HYV 17B (Drain Line)	56	No	Hydr.	3/4	5.4-2a	Outside Inside	0'-0" 0'-0"	N/A	No[31]	2RCS*SOV68B 2RCS*SOV82B	-	Globe Globe	SOV SOV	Elec. Elec.	N/A N/A	Open Open	Closed Closed	Closed Closed	Closed Closed	B,F,RM,Z B,F,RM,Z	10 10	Div I Div II	26
Z-100B	Hydraulic Unit to Recirc FCV HYV 17B (Open Line)	56	No	Hydr.	1	5.4-2a	Outside Inside	0'-0" 0'-0"	N/A	No[31]	2RCS*SOV67B 2RCS*SOV81B	-	Globe Globe	SOV SOV	Elec. Elec.	N/A N/A	Open Open	Closed Closed	Closed Closed	Closed Closed	B,F,RM,Z B,F,RM,Z	10 10	Div I Div II	26
Z-100C	Hydraulic Unit to Recirc FCV HYV 17B (Pilot Line)	56	No	Hydr.	1	5.4-2a	Outside Inside	0'-0" 0'-0"	N/A	No[31]	2RCS*SOV66B 2RCS*SOV80B	-	Globe Globe	SOV SOV	Elec. Elec.	N/A N/A	Open Open	Closed Closed	Closed Closed	Closed Closed	B,F,RM,Z B,F,RM,Z	10 10	Div I Div II	26
Z-100D	Hydraulic Unit to Recirc FCV HYV 17B (Closed Line)	56	No	Hydr.	1	5.4-2a	Outside Inside	0'-0" 0'-0"	N/A	No[31]	2RCS*SOV65B 2RCS*SOV79B	-	Globe Globe	SOV SOV	Elec. Elec.	N/A N/A	Open Open	Closed Closed	Closed Closed	Closed Closed	B,F,RM,Z B,F,RM,Z	10 10	Div I Div II	26
-	All Instr. Lines from Reactor Vessel	RG 1.11	Note 37	Air/ Water	3/4	6.2-70a	Outside	<10'-0"	A	No[31]	EF check valves	-	EFV	N/A	Auto	N/A	Open	Open	Open	Open	Excess flow	N/A	N/A	27,37
-	All Instr. Lines Penetrating Primary Containment	RG 1.11	Note 37	Air/ Water	3/4	6.2-70a	Outside	<10'-0"	A	No[31]	EFV	-	EFV	N/A	Auto	N/A	Open	Open	Open	Open	Excess flow	N/A	N/A	27,37

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KEY TO ISOLATION SIGNALS:

A = Low reactor vessel water level 3

B = Low reactor vessel water level 2
+
D = High main steam line flow

E = High main steam line tunnel area ambient temperature

F = High drywell pressure

H = RCIC steam supply pressure low

J = High reactor water cleanup system equipment area ambient temperatures

K = Reactor core isolation cooling pipe routing area high temperature and RCIC equipment area high temperature, high steam line flow, high turbine exhaust diaphragm pressure

L = High reactor vessel pressure

M = High residual heat removal system equipment area ambient temperatures

P = Low main steam line turbine inlet pressure

R = Low main condenser vacuum

S = Standby liquid control system actuated

T = High main steam line tunnel differential temperature

U = High reactor water cleanup system differential flow

W = High reactor water cleanup system nonregenerative heat exchanger outlet temperature (not a primary containment isolation signal)

X = Low reactor vessel water level 1

Y = Standby gas treatment exhaust radiation high

Z = Manual isolation
LC = Locked closed
RM = Remote manual switch from control room

AA = Main steam line lead enclosure high ambient temperature

BB = RCIC/RHR steam line flow-high

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CC = Reactor building high ambient temperature

DD = Reactor building pipe chase high ambient temperature

* = Valve closes on system initiation signal; not part of primary containment isolation signal

LMC = Local manual control, locked closed, position indication in control room

NOTES:

(1) Isolation valve arrangements depicting leakage (Type C) test provisions are shown on the referenced figures. Further discussion on Type A and C testing is also provided in Section 6.2.6.

(2) Provisions have been made in the control room to secure closed the main steam drain line valves 2MSS*SOV97A, B, C, D. Power supply fuses for these valves are removed during normal plant operation except during startup and shutdown or during periods of operation with associated main steam line inboard isolation valve (2MSS*AOV6A, B, C, or D) closed.

(3) Normal status position of valve (open or closed) is the position during normal power operation of the reactor (see Normal Position column).

(4) Primary containment and reactor vessel isolation signals are indicated by letters. Isolation signals generated by the individual system process control signals or for remote manual closure based on information available to the operator are discussed in the referenced notes in the Isolation Signal column.

(5) The specified closure rates are based on the individual valve purchase specification or vendor test data, which meet or exceed containment isolation and/or system operational requirements. Reported times are in seconds. Maximum allowable times are listed in the Technical Requirements Manual (TRM).

(6) The standard minimum closing rate is 12 in/min of nominal valve diameter for gate valves and 4 in/min of valve stem travel for globe valves. For example, a 12-in gate valve will close in 1 min.

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(7) Ac motor-operated valves required for isolation functions are powered from the ac standby power buses. Dc-operated isolation valves are powered from safety-related station batteries.

(8) Main steam isolation valves require that both solenoid pilots be de-energized to close the valves. Accumulator air pressure plus spring force act together to close the valves when both pilots are de-energized. Voltage failure at only one pilot does not cause closure.

(9) All isolation valves are Category I.

(10) All motor-operated isolation valves remain in the as-is position upon failure of valve power (FAI = Fail as is). All air-operated valves close on motive air failure in the safe position.

(11) Testable check valves are designed for remote opening with zero differential pressure across the valve seat. The valves will close on reverse flow even though the test switches may be positioned for open. The valves open when pump pressure exceeds reactor pressure even though the test switch may be positioned for close.

(12) The hydrogen recombiner system and containment monitoring system will be included in Type A testing with the isolation valves open or a separate leak test will be conducted. The system leak rate will then be added to measure leakage rate (Lam) at the upper confidence limit (UCL) as a penalty. In addition, Type C testing of the primary containment isolation valves will be performed.

(13) These valves are the ECCS and drywell spray suction and discharge isolation valves. ECCS operation is essential during the LOCA period; therefore, there are no automatic isolation signals. A high level alarm in the appropriate reactor building sump indicates excessive ECCS leakage into the secondary containment.

(14) Suppression pool spray valves have interlocks that allow them to be manually reopened after automatic closure. This setup permits suppression pool spray for high drywell pressure conditions. When automatic signals are not present, these valves may be opened for test or operating convenience.

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(15) Due to redundancy within the ECCS, some subsystems may be secured during the long-term cooling period. In addition, RHR Loops A and B have several discharge paths (LPCI, drywell spray, suppression chamber spray, suppression pool cooling) which the operator may select during the 30-day post-LOCA period.

(16) The RCIC steam exhaust valve, 2ICS*MOV122, is normally open at all times. Should a leak occur, it would be detected and alarmed by the RCIC room high temperature leak detection system.

(17) Criterion 55 concerns lines of the RCPB that penetrate the primary reactor containment. The CRD insert and withdraw lines are not part of the RCPB. The classification of the insert and withdraw lines is Quality Group B and, therefore, they are designed in accordance with ASME Section III, Safety Class 2. The basis to which the CRD lines are designed is commensurate with the safety importance of isolating these lines. Since these lines are vital to the scram function, their operability is of utmost concern.

In the design of this system, it has been accepted practice to omit automatic valves for isolation purposes as this introduces a possible failure mechanism. As a means of providing positive actuation, manual shutoff valves are used. In the event of a break on these lines, the manual valves may be closed to ensure isolation. In addition, a ball check valve located in the insert line inside the CRD is designed to automatically seal this line in the event of a break.

(18) The operator's indication that remote-manual closure of the TIP shear valves is required is failure of the TIP ball valves to close.

(19) Under normal operating conditions, the TIP system guide tubes do not communicate with the containment atmosphere because the nitrogen purge system maintains the indexing mechanism at a slightly higher pressure than primary containment. The TIP system indexers are equipped with an inlet check valve, whose function is to equalize the internal pressure of the indexer when the drywell becomes pressurized. If a LOCA or containment pressurizing event were to occur, this inlet check valve would open allowing direct communication between the containment atmosphere and the TIP guide tubes. As a result, GDC 56 is the applicable NRC requirement for the TIP system isolation design. GDC 56 does, however, allow deviation from the aforementioned

isolation requirements when the design is justified on "some other defined basis." In accordance with GE NEDC-22253, BWR Owner's Group Evaluation of Containment Isolation Concerns, October 1982, the TIP tubing can be classified as “instrument lines” in accordance with RG 1.11. RG 1.11 describes a method acceptable to the NRC for implementation of GDC 56 for instrument lines. The aforementioned BWROG report provides the justification for this classification.

The safety features were reviewed by the NRC for BWR/4 (Duane Arnold), BWR/5 (Nine Mile Point Unit 2) and BWR/6 (GESTAR II), and it was concluded that the design of the containment isolation system meets the objectives and intent of the general design criteria.

Isolation is accomplished by a seismically-qualified solenoid-operated ball valve that is normally closed. To ensure isolation capability, an explosive shear valve is installed in each line. Upon receipt of a signal (manually initiated by the operator), this explosive valve will shear the TIP cable and seal the guide tube.

When the TIP system cable is inserted, the ball valve of the selected tube opens automatically so that the probe and cable can advance. A maximum of five valves may be opened at any one time to conduct calibration, and any one guide tube is used, at most, a few hours per year.

If closure of the line is required during calibration, a signal causes a cable to be retracted and the ball valve to close automatically after completion of cable withdrawal. If a TIP cable fails to withdraw or a ball valve fails to close, the explosive shear valve is actuated. The ball valve position is indicated in the control room.

The Unit 2 TIP system design specifications require that the maximum leakage rate of the ball and shear valves be in accordance with the Manufacturer's Standardization Society (hydrostatic testing of valves).

The TIP isolation valve and the shear valve both have a leak integrity requirement of 10^{-3} atm cc/sec for air-water combination and water alone. This leakage rate represents less than 10^{-3} cc/sec of fluid at the following conditions:

Air-water combinations: 0-125 psig and 300°F

Water: 1,250 psig and <450°F

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As stated above, the penetration is automatically closed following use. During normal operation the penetration will be open approximately 8 hr/month to obtain TIP information. If a failure occurred, such as inability to withdraw the TIP cable, the shear valve could be closed to isolate the penetrations. Installation requirements are that the guide tube/penetration flange/ball and shear valve composite assembly not leak at a rate greater than 10^{-4} atm cc/sec at 125 psig. Further leak testing of the shear valves is not recommended since destructive testing would be required.

The periodic surveillance testing of the shear valves will be performed in accordance with Technical Specifications.

(20) Removable spool piece that is removed during normal operation; it is installed when the plant is down and fire protection is needed inside the primary containment.

(21) Air-operated valves 2CPS*AOV104, 105, 106, 107, 108, 109, 110, and 111 are remote manually opened before personnel entry into the primary containment.

(22) System isolation valves are normally closed. The system is placed in operation only if the hydrogen monitors detect hydrogen buildup after a LOCA. The operator has flow indication, in the main control room, of gas entering the hydrogen recombiner units. Should the gas flow rate drop below 40 cfm for 55 sec, the units will trip automatically.

(23) This line consists of inputs from the valves listed below. The line discharges below the suppression pool water level and therefore is not exposed to the primary containment atmosphere.

2RHS*SV34A and 2RHS*SV62A - steam-condensing line safety valves are gagged closed and no longer operational.

2RHS*RV56A - RHR heat exchanger shell side relief valve.

2RHS*MOV26A and 2RHS*MOV27A - RHR heat exchanger vent line isolation valves. Valve position is indicated in the main control room to provide the operator confirmation of valve status.

2RHS*V20 and 2RHS*V19 - vacuum breaker line.

2RHS*RVV35A and 2RHS*RVV36A - vacuum breakers.

The above-listed relief, safety, and vacuum breaker valves are not included in Type C or Type A tests because they are isolated from the containment atmosphere by the suppression pool water level. Also, Type C testing of these valves would require testing provisions which conflict with the intent of criteria regarding containment design. In-place leak testing of these valves would require blank flanges in adjacent piping, since there are no block valves to provide a test boundary. Relief valve piping is intentionally designed without isolation capability to comply with the guidance of ASME III, Section NC-7000. The addition of flange connections would create another primary to secondary leakage path, as well as require Type B testing. Furthermore, individual leak testing of these valves would result in excessive radiation exposure to plant personnel. The valve piping must be dismantled, or the valves must be removed, and all the valves are located in potentially high radiation areas. This would be contrary to ALARA philosophy.

(24) This line consists of inputs from the valves listed below. The line discharges below the suppression pool water level and therefore is not exposed to the primary containment atmosphere.

2RHS*SV34B and 2RHS*SV62B - steam-condensing line safety valves are gagged closed and no longer operational.

2RHS*RV56B - RHR heat exchanger shell side relief valve.

2RHS*MOV26B and 2RHS*MOV27B - RHR heat exchanger vent line isolation valves. Valve position is indicated in the main control room to provide the Operator confirmation of valve status.

2RHS*V117 and 2RHS*V118 - vacuum breaker line.

2RHS*RVV35B and 2RHS*RVV36B - vacuum breakers.

The above-listed relief, safety, and vacuum breaker valves are not included in the Type A containment integrated leak rate test since they do not provide a potential containment atmosphere leak path. They are not included in Type C testing, based on the design considerations discussed in Note 23.

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(25) This line consists of inputs from the applicable valves listed below. The line discharges at el 195'-6", which is 2'-2" below the minimum water level in the suppression pool and, therefore, is not exposed to the primary containment atmosphere. All of the valves are relief valves which provide relief for high-/low-pressure interface leakage, except 2RHS*RV108, which provides relief for upstream level control failure.

For penetration Z73
2RHS*RV108
2RHS*RV20C

For penetration Z98A
2RHS*RV61A
2RHS*RV20A
2RHS*RV110
2CSL*RV123
2CSL*RV105
2RHS*RV139

For penetration Z98B
2RHS*RV61C
2RHS*RV61B
2CSH*RV114
2CSH*RV113
2RHS*RV20B

The above-listed relief valves are not subject to the Type A containment integrated leak rate test for external leakage since they do not provide a potential containment atmosphere leak path. They are not included in Type C testing, based on the design considerations discussed in Note 23. The potential for containment pressure lifting these relief valves under accident conditions is not a concern since the relief valves listed above, including those in Notes 23 and 24, have setpoints greater than 1.5 times the containment design pressure of 45 psig. The one exception to this is valve 2CSL*RV123 with a setpoint of 46 psig. For this valve, only the discharge side of the valve is part of the primary containment boundary. Thus, containment atmosphere would pressurize the discharge side of the valve.

(26) Penetrations Z-99A,B,C,D, and Z-100A,B,C,D contain lines for the hydraulic control of the reactor recirculation flow control valve. These lines contain hydraulic fluid used to position the reactor recirculation flow control valve, and are protected against the effects of pipe whip and jet impingement.

Integrity of the system is, essentially, constantly monitored since the system is under a constant operating pressure of 1850-1950 psig. Any leakage through this system would be noticed because operation would be erratic and because of indications provided on the hydraulic control unit. In addition, in order to perform Type C tests on these lines, the system would have to be disabled and drained of hydraulic fluid. This is considered to be detrimental to the proper operation of the system since possible damage could occur in establishing the test condition or restoring the system to normal. These lines and associated isolation valves should therefore be considered to be exempt from containment testing. A specific exemption will be forwarded under separate cover. The relatively slow closing time of these solenoids is due to the hydraulic fluid used in this system.

(27) Instrument lines that penetrate primary containment conform to RG 1.11. The lines that connect to the reactor pressure boundary include a restricting orifice or equivalent reduced orifice valve inside containment, are Category I, and may terminate in instruments that are Category I. The instrument lines also include manual isolation valves and excess flow check valves or equivalent. These penetrations will not be Type C tested since the integrity of the lines is continuously demonstrated during plant operations where subject to reactor operating pressure. In addition, all lines are subject to the Type A test pressure on a regular interval. Leak-tight integrity is also verified with completion of functional and calibration surveillance activities as well as by visual observations during Operator tours.

(28) Signal B or F cause automatic withdrawal of tip probe. When probe is withdrawn, the solenoid-operated ball valve automatically closes by mechanical action.

(29) This path does not constitute a bypass leakage path, because a closed piping system outside the primary containment provides a leakage boundary. The piping/components outside the primary containment qualify as a closed system for the following reasons:

a. The system leakage boundary leak path does not directly communicate with the environment following a LOCA.

b. The system leakage boundary piping/components are designed in accordance with Quality Group B standards as defined by RG 1.26.

c. The system leakage boundary is designed to meet seismic Category I design requirements.

d. The system leakage boundary is designed to at least the primary containment pressure and temperature design conditions.

e. The system leakage boundary is designed for protection against pipe whip, missiles, and jet forces in a manner similar to that for engineered safety features.

f. The system leakage boundary is tested for leakage, unless system integrity is demonstrated to be maintained during normal plant operations.

(30) This line/path is excluded from further consideration as a potential bypass leakage path, because a water seal is provided to prevent leakage from bypassing the secondary containment. There is sufficient fluid available to maintain the seal for at least 30 days following a LOCA (see Section 6.2.3.2.3 for seal details).

(31) This line/path is excluded from further consideration as a potential bypass leakage path because (per BTP CSB 6-3, Section A) leakage from the primary containment cannot circumvent the secondary containment boundary and escape directly to the environment; that is, leakage cannot bypass the leakage collection and filtration systems of the secondary containment. Filtration of leakage is assured, because either the piping terminates in the secondary containment or leakage is directly routed to the filtration systems.

(32) In addition to a swing check valve inside containment and an air assist check valve outside containment, similar to an Atwood-Morrill boiler feed check valve as described in Catalog 63, Section I, a third valve with high leak-tight integrity is provided in each line outside containment. The spring-loaded piston operator of an air assist check valve is held in open position by air pressure during normal operation. Fail-open solenoid valves are used to ease air pressure to permit the check valve piston operator to close. The high leak-tight integrity isolation valve will be remote manually operated from the control room when signals indicate loss of feedwater flow.

(33) Bypass leakage through these penetrations is via the post-accident sample system branch connections. Leakage volumes are accounted for as post-accident sampling system bypass leakage. The bypass leakage rate limit specified in Tables 6.2-55a, b, c, and d shall apply until such time as a modification eliminates the potential leakage paths.

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(34) The metal bellows at the ends of the TIP system drywell penetration flanges, in addition to the flanges themselves and the midspan flange in 2NMT*Z31B, will be subject to Type B testing.

(35) These valves have been abandoned in place and associated piping has been capped. Valves are not electrically connected.

(36) Valve is a simple check during normal plant operations to meet ATWS requirements. Explosive valves 2SLS*VEX3A/B, when intact, are considered blind flanges to prevent inadvertent forward flow through the SLS system into the reactor vessel. This ensures the check valves 2SLS*MOV5A/B remain in the closed position. After SLS injection, operator action can be taken to close the motor-operated stem of valves 2SLS*MOV5A/B to meet long-term containment isolation requirements.

(37) ESF designations for instrument lines penetrating the reactor vessel and primary containment are as follows:

Instrument lines associated with CSH, CSL, RHS, and CMS are ESF.

Instrument lines associated with MSS, ICS, DER, WCS, and RCS are not ESF.

(38) Air assist check valves are normally held open by flow. The testing feature can move the valve disc slightly into the flow stream, but is not capable of closing with full flow in the pipe. The valves will close upon cessation of flow in the line and the testing feature will not inhibit closure.

(39) These valves are not included in Type C or Type A tests because they are isolated from the containment atmosphere by the suppression pool water level and do not provide a potential containment atmospheric leak path.

(40) Manual testable check valves are designed to close with zero differential pressure across the valve seat. The valves will close on reverse flow. The valves open when pump pressure exceeds reactor pressure.

(41) Penetrations that are associated with closed loop systems outside containment (CLOC) only require Appendix J Type C testing of a single CIV inside or outside containment, provided a closed system outside containment is maintained. The closed system boundary includes the main process lines from the suppression pool or process penetration on the suction side of the associated pumps through the containment penetration on the discharge side, out to and

including the first normally closed isolation valves (i.e., drain valves, interface boundary valves, relief valves, etc.). Designation of the credited valve is identified in the Technical Requirements Manual (TRM) and controlled under the NMP2 10CFR50 Appendix J Testing Program.

Note that for Penetration Z22, 2ICS*V156 may not be credited as the only tested isolation valve in conjunction with a closed loop, since it is a simple check valve outside containment which does not meet the provisions of 10CFR50 Appendix A, GDC 55.

(42) The automatic isolation signals to RCIC turbine exhaust vacuum breaker line isolation valves 2ICS*MOV148 and 2ICS*MOV164 energize a 70-sec time delay to slow closure of these valves. The time delay is provided to ensure the exhaust steam in the turbine exhaust line has fully condensed prior to isolating the vacuum breaker line.

(43) Power supply breaker to 2RHS*MOV104 is locked open and administratively controlled during normal plant operation to prevent its spurious operation during and following an Appendix R fire event.

(44) The pipe extending into the suppression chamber through containment penetrations Z-52 is designed with both PCIVs outside of primary containment (see Figure 9.4-8k). While this does not meet 10 CFR 50 Appendix A, Criterion 56, this configuration is consistant with all of the Mark II containment nuclear plants in the United States and is consistant with that described in the NMP2 Safety Evaluation Report (Ref. NUREG -1047) and the current Standard Review Plan (SRP) for “Containment Isolation System” (SRP 6.4.2).

(45) Penetrations that are associated with a closed loop outside containment and do not constitute a potential primary containment atmospheric pathway during and following a DBA do not require Appendix J Type C testing as stated above in note (41). Pressure Isolation Valve Leakage testing of the outside containment isolation valve in water filled systems may be substituted provided a closed system outside containment is maintained and PIV testing confirms that the penetration will remain filled with water during and following a DBA and therefore does not constitute a potential primary containment atmospheric pathway.

The closed system boundary includes the main process lines from the suppression pool or process penetration on the suction side of the associated pumps through the containment penetration on the discharge side, out to and including the first normally closed isolation valves (i.e., drain valves, interface boundary valves, relief valves,

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etc.). The following designated closed system penetration Pressure Isolation Valves may be leak tested and credited for the Appendix J Program:

- 2RHS*MOV40A,B Shutdown Cooling Return Outside IVs (penetrations Z-10A/10B)
- 2RHS*MOV113 SDC Supply Outside IV (penetration Z-11)
- 2CSH*MOV107 CSH to RPV Outside IV (penetration Z-14)
- 2CSL*MOV104 CSL to RPV Outside IV (penetration Z-16)

TABLE 6.2-57
(Sheet 1 of 1)
COMBUSTIBLE GAS CONTROL SYSTEM COMPONENT DESCRIPTION

Recombiner System	
Type	Thermal
Quantity	2, each 100% capacity
Gas processing capacity of reaction chamber	150 scfm of gas with either 2 1/2% maximum O_2 with excess H_2 or 5% maximum H_2 with excess O_2
Maximum containment pressure for operation of recombiners	17.1 psig
Heater power requirement	90 kW
Recombiner suction	Drywell or suppression chamber
Recombiner discharge	Suppression chamber
Manufacturer	Rockwell International
Containment Purge System (SGTS Fan)	
Fan Type	Centrifugal
Quantity	2
Capacity	4,000 cfm each
Manufacturer	Buffalo Forge Co.
Monitoring System	
Number of loops	2
Number of sampling points per loop	5
Drywell	3
Suppression Chamber	2
Sampling time per point	20 min (Drywell samples only when sequence timer is used)
Operation	Continuous or standby
Instrument range (O_2)	0-10%, 0-25% and 0-30%
Instrument range (H_2)	0-10%, 0-25% and 0-30%
Instrument error (of total range)	±3% (maximum)
Manufacturer	Meggitt Safety Systems Inc.

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(Sheet 1 of 1)
GENERAL PARAMETERS USED IN CALCULATING POST-DBA OXYGEN/HYDROGEN CONCENTRATIONS

Fission product distribution model:	
Halogens	50% released from core
Noble gases	100% released from core
Other fission products from core and intimately mixed with the coolant	1% of solids released
Fraction of fission product energy absorbed by coolant:	
<u>Beta</u>	
Beta from fission products in fuel rods	0.0
Beta from fission products intimately mixed with suppression pool water	1.0
<u>Gamma</u>	
Gamma from fission products in fuel rods absorbed by coolant in core region	0.1
Gamma from fission products intimately mixed with suppression pool water	1.0
H_2 radiolytic generation rate	0.5 molecule/100 eV
O_2 radiolytic generation rate	0.25 molecule/100 eV
O_2 concentration limit	5 volume percent

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(Sheet 1 of 2)
PLANT PARAMETERS USED IN POST-DBA
COMBUSTIBLE GAS CONCENTRATION ANALYSIS

Reactor power	3,988 MW
Drywell free volume	306,200 ft^3
Suppression chamber free volume (at high pool water level)	190,600 ft^3
Initial drywell pressure	15.45 psia
Initial drywell temperature	150°F
Initial drywell relative humidity	100%
Initial suppression chamber pressure	15.45 psia
Initial suppression chamber temperature	122°F
Initial suppression chamber relative humidity	100%
Zircaloy reaction with steam	GE14=679 lbm GNF2=677 lbm
Duration of reaction	120 sec
Fraction of water in drywell and reactor vessel	5.8%
Downcomer submergence at high pool water level	11 ft
Vacuum breaker setpoint	0.25 psid
Initial O_2 concentration	4 volume percent
Recombiner capacity	120 scfm[(1)]
Recombination efficiency	~100%

TABLE 6.2-59 (Cont'd.)
(Sheet 2 of 2)

Temperature transient for primary containment (recirculation suction line DER)	Figure 6.2-7

(1) The minimum recombiner flow necessary to control the formation of combustible gases is 120 scfm. Although the recombiner has a design processing capacity of 150 scfm, the analysis to determine the post-accident hydrogen and oxygen concentrations within primary containment uses the minimum flow of 120 scfm. Operation of the recombiner at 150 scfm would improve (reduce) the post-accident concentrations of combustible gases in primary containment.

TABLE 6.2-59A
(Sheet 1 of 2)
HYDROGEN AND OXYGEN SAMPLING POINTS WITHIN
PRIMARY CONTAINMENT

Drywell

Sample Point	Loop	Elevation	Radial Location		Function
1	A	270'-0"	1'-6 9/16"N	17'-8 7/16"W	Supply
2	A	289'-3 7/16"	16'-1 1/4"N	2'-5 3/8"E	Supply
3	A	320'-5/16"	3'-8"N	20'-9 1/2"E	Supply
4	B	266'-3 1/4"	7'-0"S	24'-7"E	Supply
5	B	300'-6"	21'-4 3/4"S	11'-7"W	Supply
6	B	322'-7"	6'-2 3/4"S	16'-1 3/4"W	Supply

TABLE 6.2-59A (Cont'd.)
(Sheet 2 of 2)

Suppression Chamber

Sample Point	Loop	Elevation	Radial Location		Function
1	A	234'-2"	29'-10 1/2"N	26'-9 1/4"W	Supply
2	A	218'-7 1/2"	26'-3 1/2"N	27'-4"E	Supply
3	B	220'-6"	27'-5"S	30'-2 3/4"E	Supply
4	B	219'-3 1/2"	11'-9"S	30'-0"W	Supply

TABLE 6.2-59B
(Sheet 1 of 1)
STRUCTURES, PIPING, AND EQUIPMENT IN
VICINITY OF HYDROGEN AND OXYGEN SAMPLING POINTS

<u>Drywell</u>

Sample Point 1	-	BSW, RCS line, miscellaneous large bore pipe restraint structures, small bore lines, and conduits with supports.
Sample Point 2	-	BSW, miscellaneous restraint structures, and conduit piping with supports.
Sample Point 3	-	Primary containment wall, miscellaneous pipe restraint structures, and exhaust ducts with supports.
Sample Point 4	-	MSS SRV discharge piping, restraint beams, and miscellaneous restraint structures.
Sample Point 5	-	FWS, MSS line, MSS SRVs, miscellaneous supports and support steel.
Sample Point 6	-	Small bore lines and supports.

<u>Suppression Chamber</u>

Sample Points 1 through 4	-	MSS SRV discharge piping and drywell floor downcomers.

NOTE: The sample point locations are defined in Table 6.2-59A.

TABLE 6.2-59C
(Sheet 1 of 1)
CORROSION RATES

Material	Corrosion Rate (scf/ft^2-hr)	Applicable Temperature Range
Aluminum	4.0×10^{-4} (constant)	Up to 285°F
Zinc	$0.6764 \exp\left[\frac{-5113.25}{(460+T)}\right]$	119.12°F≤T≤224.06°F
	$2.8245 \ x \ 10^{11} \exp\left[\frac{-23416.67}{(460+T)}\right]$	224.06°F≤T≤334.22°F

TABLE 6.2-59D
(Sheet 1 of 1)
ALUMINUM AND ZINC INVENTORY EXPOSED TO SPRAYS

Material Type	Surface Area (ft^2)	Weight (lbm)
Aluminum	1,300	41,400
Galvanized steel	57,510*	5,410*
Zinc primer	99,487*	3,640*

* The hydrogen generation/control analysis considers 15 percent larger surface area and weight as a conservative allowance for the uncertainty of these inventories.

TABLE 6.2-60
(Sheet 1 of 1)
PRIMARY CONTAINMENT LEAKAGE TESTING

Test Type	Test Description	Calculated Peak Pressure[1] (Pa)(psig)	Leak Rates at Pa[3] (%/24 hr) Maximum Allowable (La)	Leak Rates at Pa[3] (%/24 hr) Design (Ld)	Test Pressure (Pa)(psig)
A	Integrated leak rate	39.75	1.1	1.1	39.75
B	Local penetration leak rate	39.75	[2]	[2]	39.75
C	Local containment isolation valve leak rate (air)	39.75	[2]	Determined by owner or specified by Technical Specifications	39.75
	OR				
	Local containment isolation valve leak rate (water)	39.75	N/A. Leakage will be limited to less than or equal to 1 gpm times the total number of containment isolation valves in hydrostatically tested lines		43.73

(1) The peak drywell pressure calculated for power uprate and EPU, including the MELLLA+ operating domain, using GE codes and models (Section 6.2.1.1.6 and Table 6.2-4) is less than the originally calculated value of 39.75 psig; therefore, test pressures continue to be based on a P_a value of 39.75 psig.

(2) The combined leakage rate of all penetrations and valves subject to Type B and C tests (except for MSIVs and valves which are hydrostatically tested) is less than 0.6 La or 0.66 percent by weight of the primary containment air/24 hr, as specified in Appendix J to 10CFR50.

(3) Fractional leak rates based on a total primary containment volume of 496,800 cu ft.

TABLE 6.2-61
(Sheet 1 of 1)
SUPPRESSION POOL STEAM BYPASS LEAKAGE TESTS

	High-Pressure Test	Low-Pressure Test
Frequency	Once prior to fuel load	Once prior to fuel load and every outage that a containment integrated leak rate test is performed
Differential pressure, psi	25 +0.5,-0.0	3
Test allowable bypass capacity[1] $(A/\sqrt{K})$, ft^2	0.0054	0.0054
Acceptable pressure decay region[2]	See Figure 6.2-89	See Figure 6.2-90

(1) 10 percent of the minimum drywell bypass leakage capability.

(2) The acceptable pressure decay region is calculated assuming a drywell temperature of 70°F. For drywell temperature greater than 70°F, these figures may be used, as they are conservative.

TABLE 6.2-62
(Sheet 1 of 1)
SECONDARY CONTAINMENT ACCESS DOORS

Door Pair No.	Building Elevation	Quadrant
R240-6*, R240-7	240'-0"	Southwest
R261-1, R261-2*	261'-0"	Southwest
R261-5, RS261-2	261'-0"	Southeast
RR261-2B*, SG261-3	261'-0"	Northeast
RR261-2A*, RR261-1	261'-0"	Northeast
NA262-1*, NA262-2	261'-0"	Northwest
SA262-2, SA262-3*	261'-0"	Southeast
SA262-1, SA262-4	261'-0"	Southeast
RS289-1, RS289-3	289'-0"	Southeast
RS306-1, RS306-2	306'-6"	Southeast
RS328-1, RS328-3	328'-10"	Southeast
RS353-1, RS353-3	353'-10"	Southeast
RS408-1, RS408-2	408'-0"	Southeast

NOTE: Door NA262-1 is disabled for nuclear security purposes.

* These doors are equipped with security alarms that annunciate in an alarm station continuously occupied by Security personnel, and with which the control room has continuous communications capability.

TABLE 6.2-63

CONTAINMENT PENETRATIONS WITH
RELIEF VALVE DISCHARGE HEADERS

Valve Number	Penetration Number	Pipe Length and Size Outside Containment (±1 ft)	
		Valve to Header	Header to Containment
2RHS*RV108	Z-73	4" Line, Length 45'-0"	6" Line, Length 4'-8"
2RHS*RV20C	Z-73	1" Line, Length 26'-4"	6" Line, Length 3'-2"
2RHS*SV34A	Z-88A	N/A	6" and 12" Line, Length 97'-7"
2RHS*SV62A	Z-88A	N/A	8" and 12" Line, Length 95'-2"
2RHS*RV56A	Z-88A	1" Line, Length 38'-3"	12" Line, Length 90'-4"
2RHS*MOV26A	Z-88A	1" Line, Length 25'-6"	12" Line, Length 91'-2"
2RHS*MOV27A	Z-88A	1" Line, Length 24'-0"	12" Line, Length 91'-2"
2RHS*V20	Z-88A	3/4" Line, Length 9'-4"	12" Line, Length 93'-1"
2RHS*V19	Z-88A	3/4" Line, Length 11'-4"	12" Line, Length 93'-1"
2RHS*RVV35A & 36A	Z-88A	10" Line, Length 3'-2" & 4'-4"	12" Line, Length 36'-0"
2RHS*SV34B	Z-88B	N/A	6" and 12" Line, Length 84'-6"
2RHS*SV62B	Z-88B	N/A	8" and 12" Line, Length 82'-2"
2RHS*RV56B	Z-88B	1" Line, Length 51'-10"	6" and 12" Line, Length 81'-0"
2RHS*MOV26B	Z-88B	1" Line, Length 32'-5"	6" and 12" Line, Length 81'-0"
2RHS*MOV27B	Z-88B	1" Line, Length 29'-11"	6" and 12" Line, Length 81'-0"
2RHS*V117	Z-88B	3/4" Line, Length 9'-10"	12" Line, Length 80'-0"

TABLE 6.2-63 (Cont'd.)

Valve Number	Penetration Number	Pipe Length and Size Outside Containment (±1 ft)	
		Valve to Header	Header to Containment
2RHS*V118	Z-88B	3/4" Line, Length 8'-10"	12" Line, Length 80'-0"
2RHS*RVV35B & 36B	Z-88B	10" Line, Length 3'-2" & 4'-4"	12" Line, Length 27'-5"
2RHS*RV20A	Z-98A	1" Line, Length 10'-6"	3" Line, Length 27'-6"
2RHS*RV61A	Z-98A	1" Line, Length 13'-5"	3" Line, Length 19'-1"
2RHS*RV110	Z-98A	1" Line, Length 26'-7"	3" Line, Length 36'-5"
2RHS*RV139	Z-98A	1" Line, Length 19'-7"	3" Line, Length 39'-7"
2CSL*RV105	Z-98A	1" Line, Length 183'-9"	3" Line, Length 38'-1"
2CSL*RV123	Z-98A	1" Line, Length 21'-3"	3" Line, Length 37'-1"
2RHS*RV20B	Z-98B	1" Line, Length 25'-0"	3" Line, Length 15'-9"
2CSH*RV113	Z-98B	1" Line, Length 108'-0"	3" Line, Length 16'-6"
2CSH*RV114	Z-98B	1" Line, Length 81'-5"	3" Line, Length 18'-0"
2RHS*RV61B	Z-98B	1" Line, Length 9'-6"	3" Line, Length 25'-6"
2RHS*RV61C	Z-98B	1" Line, Length 13'-6"	3" Line, Length 30'-6"

TABLE 6.2-64

TYPE AND QUANTITY OF
INSULATION USED IN DRYWELL

Material	Volume, ft^3	25% Margin, ft^3
Temp-Mat	69	86
Min-K	366	458

TABLE 6.2-65

REVERSE TESTED CONTAINMENT ISOLATION VALVES

Pene-tration No.	System	Valve I.D.	Valve Type	Justi-fication
Z8A	RHS	MOV25A	Flexible Disc Gate	1
Z8B	RHS	MOV25B	Flexible Disc Gate	1
Z12	CSH	MOV118	Flexible Disc Gate	1
Z18	ICS	MOV143	Globe	2
Z17	ICS	MOV136	Flexible Disc Gate	1
Z19	ICS	MOV122	Flexible Disc Gate	1
Z21A	ICS	MOV128	Flexible Disc Gate	1
Z48	CPS	AOV108	Butterfly	3
Z51	CPS	AOV109	Butterfly	3
Z50	CPS	AOV107	Butterfly	3
Z49	CPS	AOV106	Butterfly	3
Z55A	HCS	MOV4A	Flexible Disc Gate	1
Z55B	HCS	MOV4B	Flexible Disc Gate	1
Z56A	HCS	MOV6A	Flexible Disc Gate	1
Z57A	HCS	MOV5A	Globe	2
Z56B	HCS	MOV6B	Flexible Disc Gate	1
Z57B	HCS	MOV5B	Globe	2
Z58	CPS	SOV122	Globe	4
Z59	CPS	SOV121	Globe	4
Z60A	CMS	SOV61A	Globe	4
Z60C	CMS	SOV63A	Globe	4
Z60D	CMS	SOV33A	Globe	4
Z61C	CMS	SOV34A	Globe	4
Z60E	CMS	SOV61B	Globe	4
Z60G	CMS	SOV63B	Globe	4
Z60H	CMS	SOV33B	Globe	4
Z61F	CMS	SOV34B	Globe	4
Z11	RHS	RV152	Relief	5
Z33B	CCP	RV170	Relief	5
Z34B	CCP	RV171	Relief	5
Z33A	CCP	RV1019A	Relief	5
Z34A	CCP	RV1020A	Relief	5
Z46A	CCP	RV1021A	Relief	5
Z47	CCP	RV1022A	Relief	5
Z39	DFR	RV228	Relief	5
Z40	DER	RV344	Relief	5
Z1A	MSS	AOV6A	Globe	2
Z1B	MSS	AOV6B	Globe	2
Z1C	MSS	AOV6C	Globe	2
Z1D	MSS	AOV6D	Globe	2

TABLE 6.2-65 (Cont'd.)

Penetration No.	System	Valve I.D.	Valve Type	Justification
Z4A	FWS	MOV21A	Flexible Disc Gate	1
Z4B	FWS	MOV21B	Flexible Disc Gate	1

Justification Notes:

1. Flexible disc gate valves may be tested using a test connection (TC) between the disc seats. (This is an alternate method of reverse direction testing by pressurizing the area between the disc seats. This method is used in lieu of the forward flow test.) This is a conservative test since both LOCA and non-LOCA seat leakage is measured.

2. Globe valves are oriented to ensure LLRT test pressure tends to unseat the valve, whereas LOCA pressure will tend to seat the valve. This is conservative for testing.

3. Butterfly valves are reverse tested which will provide equivalent results since the seating area(s) and test pressure force(s) will be equal in either direction.

4. Solenoid valves are of the pressure-balanced bellows type or equivalent. By design, neither upstream nor downstream pressure can exert a force on the disc, and the spring force of the bellows is the only force tending to seat the valve disc. Reverse flow testing is therefore equivalent to testing in the same direction as post-accident flow.

5. Relief valves are nozzle-type, spring-actuated relief valves. The valves are orientated to ensure test pressure tends to unseat the valve, whereas LOCA pressure will tend to seat the valve. This is conservative for testing.