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6.0 ENGINEERED SAFETY FEATURES

6.1 ENGINEERED SAFETY FEATURE MATERIALS

Materials used in the manufacture of engineered safety feature (ESF) components have been evaluated to ensure that material interactions will not occur that could potentially impair operation of the engineered safety features. Materials have been selected to withstand environmental conditions encountered during normal operation as well as during postulated accidents. Their compatibility with core and containment spray solutions has been considered and the effects of radiolytic decomposition products have been evaluated.

Coatings used on exterior surfaces within the primary containment are suitable for the environmental conditions expected. Nonmetallic thermal insulation utilized is required to have the proper ratio of leachable sodium plus silicate ions to leachable chloride plus fluoride ions to minimize the possibility of stress corrosion cracking.

6.1.1 METALLIC MATERIALS

6.1.1.1 Materials Selection and Fabrication

6.1.1.1.1 Material Specifications

Principal pressure retaining materials and the material specifications for the reactor coolant pressure boundary components are listed in <Table 5.2-5>. <Table 6.1-1> lists the principal pressure retaining materials and the material specifications for the engineered safety features components of the plant. All materials have been provided with ASME Section III material tests as required by the applicable component safety classifications. In addition, all Safety Class 2 carbon steel

pipings and valves installed in the main steam and feedwater systems meet the fracture toughness requirements specified in Subsection NB-2300, Section III of the ASME Code.

6.1.1.1.2 Engineered Safety Features Construction Material

<Section 5.2.3> discusses compatibility of the reactor coolant with materials exposed to the reactor coolant. These same materials are found in the engineered safety features components.

All ESF construction materials are resistant to stress corrosion in the presence of BWR coolant and containment spray. Conservative corrosion allowances are provided for all exposed surfaces of carbon steel. General corrosion of all other materials is negligible. Demineralized water, with no additives, is used as the core cooling water and for containment spray. Following a LOCA, this high purity water will have no detrimental effect on any of the ESF materials. All of the materials listed in <Table 6.1-1> are compatible.

6.1.1.1.3 Integrity of Engineered Safety Feature Components During Manufacture and Construction

6.1.1.1.3.1 Control of Sensitized Stainless Steel

Conformance with <Regulatory Guide 1.44> and <Regulatory Guide 1.31> is discussed in <Section 1.8> and <Section 5.2.3>. The following controls were used to avoid severe sensitization of balance-of-plant (BOP) piping and to comply with the intent of <Regulatory Guide 1.44> and <NUREG-0313>:

a. Corrosion Resistant Material

All safety-related Types 304, 316L and 347 modified stainless steel material used in the reactor coolant pressure boundary and ESF

systems were obtained in the solution annealed condition. Piping subject to hot bending is solution heat treated with all cold straightening limited to the two percent strain rule as specified by the ASME Section III Code. All BOP piping exposed to reactor coolant uses socket seal welded fittings with controlled heat input and welding interpass temperatures are limited to 350°F. Both the design of the weld and the weld procedure reduce the susceptibility of the material to crack sensitization.

b. Other Methods

The methods employed include:

1. Piping exposed to reactor coolant, within the scope of BOP piping, is included in four systems: reactor recirculation, nuclear boiler, standby liquid control, and control rod drive. The largest size piping used in these systems is 1-1/2 inch diameter Schedule 40 or thicker. Over 99 percent of the piping is one inch in diameter or less. This piping is joined by socket welded fittings. The design of the socket fittings, using fillet seal welds, limits the heat input into the pipe. When combined with a limited voltage, amperage and travel speed, this prevents sensitization of the Type 304 austenetic stainless steel piping used. The gas tungsten arc welds used in accordance with strict QA approved procedures result in a maximum of 45,000 joules per inch; shielded metal arc welds result in 56,000 joules per inch for all socket welded stainless steel pipe. This limited heat input control, when combined with the heat sink effect of the socket seal weld connection design, provides maximum protection from sensitization of the metal adjacent to the seal weld.
2. Using socket seal welds reduces the possibility of the welded portion of the pipe being exposed to the reactor coolant.

This also eliminates the necessity of interior grinding since the interior surface of the pipe is not disturbed.

3. Using weld heat input control to limit the material heat flux to a value that avoids the conditions that cause excessive sensitization.
4. Limiting weld interpass temperatures to 350°F and the weld weave pattern to four times the core wire diameter to control the heat buildup that contributes to excessive sensitization.

6.1.1.1.3.2 Cleaning and Contamination Protection Procedures

Specifications for ESF piping and components specify requirements for cleanliness and contamination protection during fabrication, shipment and storage as recommended by <Regulatory Guide 1.44>. Onsite and preoperational cleaning of ESF components is in accordance with the recommendations of <Regulatory Guide 1.37>. General compliance to regulatory guides is discussed in <Section 1.8>.

Exposure to contaminants capable of causing stress corrosion cracking of austenitic stainless steel components was avoided by carefully controlling all cleaning and processing materials which contact the stainless steel during manufacture and construction. Special care was exercised to ensure removal of surface contaminants prior to any heating operations. Water quality for cleaning, rinsing, flushing, and testing was controlled and monitored. Suitable packaging and protection was provided for components to maintain cleanliness during shipping and storage.

6.1.1.1.3.3 Cold Worked Stainless Steel

Austenitic stainless steel with a yield strength greater than 90,000 psi is not used in engineered safety features systems.

6.1.1.1.3.4 Nonmetallic Insulation

Nonmetallic thermal insulation materials in ESF systems are in accordance with the staff positions of <Regulatory Guide 1.36>. They have the proper ratio of leachable sodium plus silicate ions, to leachable chloride plus fluoride ions. A detailed discussion of the nonmetallic thermal insulation used inside containment is presented in <Section 6.1.2>.

6.1.1.1.4 Weld Fabrication and Assembly of Stainless Steel ESF Components

All ESF system components and piping have been constructed in accordance with the staff positions of <Regulatory Guide 1.31> or the interim positions specified in NRC Branch Technical Position MTEB 5-1, "Control of Stainless Steel Welding." General compliance or alternate approach assessment for <Regulatory Guide 1.31> is discussed in <Section 1.8> and <Section 5.2.3>.

6.1.1.1.5 Weld Fabrication and Assembly of Ferritic Steel ESF Components

All ESF system components and piping have been constructed in accordance with <Regulatory Guide 1.50> and <Regulatory Guide 1.71> as delineated in <Table 1.8-1>. General compliance may also be found in <Section 1.8> and <Section 5.2.3.3>. Moisture control on low hydrogen welding materials conforms to the requirements of Sections II and III of the Code. Compliance to the Code is also addressed in <Section 5.2.3.3>.

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

Demineralized water, with no additives, is employed in the core cooling water and containment sprays. No detrimental effects will occur on any

of the ESF materials from this high purity water. In addition, following an accident, the containment and drywell atmospheres are maintained below 4 percent (by volume) hydrogen in accordance with <Regulatory Guide 1.7> <Section 6.2.5>.

No soluble acids or bases are stored within containment, except for the 5,150 gallon capacity borax-boric acid solution storage tank. This volume of borax-boric acid will be injected into the reactor if a failure of the control rod drive system occurs.

Water used in the engineered safety features system is controlled to provide assurance against stress corrosion cracking of unstabilized austenitic stainless steel components. Water used for emergency core cooling systems and spray systems is controlled to ensure the following limits:

- a. Conductivity = 3 to <10 $\mu\text{mhos/cm}$ at 25°C
- b. Chloride (Cl^-) <0.50 ppm
- c. pH = 5.3 to 8.6 at 25°C

Water used in the ESF systems is stored in the suppression pool and the condensate storage tank. Water quality is maintained by deep bed demineralizers or filter demineralizers.

The coating systems used inside containment comply with the staff positions of <Regulatory Guide 1.54> and have undergone a qualification program to verify integrity following a LOCA. All coating materials (exceptions are noted in <Section 6.1.2>) used inside containment have been successfully tested for irradiation decontamination and DBA in accordance with the application specifications (Reference 1), (Reference 2), and (Reference 3).

The nonmetallic insulating system used inside containment (Owens-Corning "Nu'K'on" Fiberglas blanket insulation) has also undergone a qualification program (Reference 4) to verify its performance following a LOCA.

6.1.2 ORGANIC MATERIALS

Many protective coatings that are common in industrial use can deteriorate in a postaccident environment and contribute substantial quantities of foreign solids and residue to the reactor building sump. Therefore, protective coatings used inside the reactor building have demonstrated the ability to withstand postaccident conditions by satisfying all the criteria listed in ANSI N101.2 with minor exceptions as identified in <Table 6.1-2>. Also included in this qualification is the epoxy caulking material used to seal weld discontinuities, such as porosity and laminations prior to final application of the coating system.

The suitability of the reactor building coating systems to withstand the design-basis accident (DBA) has been evaluated. Coatings have been applied in accordance with manufacturer's recommendations. In addition, the guidance of <Regulatory Guide 1.54> is followed.

Organic coating materials for inside the reactor building are listed in <Table 6.1-2>. Stainless steels will not be placed in contact with organic coatings or cleaning materials that could contribute to stress corrosion cracking. These materials are compounds containing unacceptable levels of leachable chlorides, fluorides, lead, zinc, copper, sulfur, or mercury.

Various nonmetallic materials are used as follows: in bearings; ethylene propylene, silicone or butyl rubber for O-rings; wire wound asbestos for gaskets; and lubricants with less than 200 ppm leachable

chlorides. Cross-linked polyethylene or ethylene propylene rubber is used for electrical cable insulation and chlorosulfonated polyethylene is used for cable jacketing. The cabling was designed to withstand radiation dose. There is approximately 330,000 ft of electrical cabling inside containment which results in less than 40,000 lbs of these materials. Total exposed surface area of these materials is conservatively estimated to be 86,400 ft² based on an equivalent cable diameter of 1.0 inch. In addition, closed cell polyethylene is provided on chilled water lines in the wetwell areas in containment. The amount is less than 130 lbs and approximately four gallons of contact cement is used to apply the insulation to carbon steel pipe. The cement includes a chlorinated compound which is not leachable under either operating or LOCA temperatures. Any plastics or elastomers used in a high radiation area are evaluated to determine service deterioration in accordance with ANSI N4.1.

Penetrants used in liquid penetrant testing contain not more than one percent total sulfur and one percent total halogens (Reference 5).

The only significant organic materials used on equipment supplied by General Electric are the protective coatings used on some carbon steel components. These coatings are specified to meet the requirements of <Regulatory Guide 1.54> and are qualified using the standard ANSI tests. However, because of the impracticability of using these special coatings on all equipment, certain small size equipment (e.g., electronic/electrical trim, covers, face plates, valve handles, etc.) may be coated with unqualified organic coatings. In addition, certain touch-up and repair applications (for which valid DBA tests per ANSI N101.2 are not available) were used during construction to repair previously applied qualified coatings. The total coated area for these surfaces and this equipment is approximately 9,520 sq ft.

Heat insulation used within the containment (Owens Corning "Nu'K'on") is 95-100 percent inorganic. Exterior cloth and Fiberglas insulating wool

are the major components of the insulation. Together they represent over 95 percent of the total mass of the insulation.

The insulation is comprised of a quilted, light density, semi-rigid fibrous glass (pad) material, encapsulated in woven glass (cloth) to form a composite blanket. The blankets use Velcro for ease of installation and removal. The Velcro is made from two components: Nomex nylon for the base mat and loops, and stainless steel for the hooks.

Insulation is encapsulated with rolled and formed 22 gauge (304) stainless steel jacketing, combining quick release stainless steel latches and closure handles.

For anti-sweat insulation a closed cell polyethylene foam is used in limited quantity (less than 130 lbs) in areas of the wetwell above the Elevation 620'-6" floor.

6.1.3 REFERENCES FOR SECTION 6.1

1. Bechtel Corporation, "Standard Specification Coatings for Nuclear Power Plants," Specification Numbers CP-951, CP-952, CP-956.
2. American National Standards Institute, "Protective Coatings (Paints) for Light Water Nuclear Reactor Containment Facilities," ANSI N101.2, 1972.
3. American National Standards Institute, "Protective Coatings (Paints) for the Nuclear Industry," ANSI N5.12, 1974.
4. Topical Report OCF-1, "Nuclear Containment Insulation System," dated August 1977.

5. American Society of Mechanical Engineers, Section V, Article 6,
"Liquid Penetrant Examination," 1974 Edition With Addenda, up to
and including Winter 1975.

TABLE 6.1-1

PRINCIPAL ENGINEERED SAFETY FEATURES COMPONENTS
PRESSURE RETAINING MATERIALS

<u>Principal Component</u>	<u>Form</u>	<u>Material</u>	<u>ASME Specification</u>
RHR Heat Exchanger:			
Shell, head and channel	Plate	Carbon steel	SA-516, GR 70
Tube sheet	Plate	Carbon steel	SA-516, GR 70
Nozzles	Forging	Carbon steel	SA-105
Flanges	Forging	Carbon steel	SA-105
Tubes	Tubing	Stainless steel	SA-249, Type 304L
Bolts	Bar	Low alloy steel	SA-193, GR B7
Nuts	Forging	Low alloy steel	SA-194, GR 7
RHR, HPCS and LPCS Pumps:			
Bowl assembly	Casting	Cast steel	A-216, GR WCB (ASTM)
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	Plate	Carbon steel	SA-516, GR 70
	Pipe	Carbon steel	SA-106, GR B
Shaft	Bar	Stainless steel	A-276, Type 410 (ASTM)
Impeller	Casting	Stainless steel	A-351, GR CA6NM (ASTM)
Studs	Bolting	Alloy steel	SA-193, GR B7
Nuts	Nut	Low alloy steel	SA-194, GR 7
Cyclone separator body and cover	Bar	Stainless steel	SA-479
HPCS Valves:			
Body, bonnet and disc	Casting	Cast steel	SA-216, WCB
Alternate disc	Forging	Carbon steel	SA-105
Stem	Bar	Stainless steel	A-479 or A276, Type 410 (ASTM)
Studs	Bar	Alloy steel	SA-193, GR B7
Nuts	Bar	Steel	SA-194, GR 2H

TABLE 6.1-1 (Continued)

<u>Principal Component</u>	<u>Form</u>	<u>Material</u>	<u>ASME Specification</u>
Standby Liquid Control Pump:			
Fluid cylinder	Forging	Stainless steel	SA-182, F304
Cylinder head, valve, cover, and stuffing box flange plate	Plate	Stainless steel	SA-240, Type 304
Cylinder head extension, valve stop, and stuffing box	Bar	Stainless steel	SA-479, Type 304
Stuffing box gland and plungers	Forging	Nickel alloy	SA-564, Type 630
Studs	Bar	Alloy steel	SA-193, GR B7
Nuts	Forging	Alloy steel	SA-194, GR 7
Standby Liquid Control Storage Tank:			
Tank	Plate	Stainless steel	SA-240, Type 304
Fittings	Forgings	Stainless steel	SA-182, Type F304
Pipe	Pipe	Stainless steel	SA-312, Type 304
Welds	Electrodes	Stainless steel	SFA 5.4 and 5.9, Types 308, 308L, 316, 316L
Control Rod Velocity Limiter	Casting	Stainless steel	A-351, GR CF8 (ASTM)
<u>Other Components</u>			
Piping:			
	Pipe	Stainless steel	SA-312, Type 304
		Stainless steel	SA-312, Type 316L
		Stainless steel	SA-358, Type 304, C1-2
		Stainless steel	SA-376, Type 304
		Stainless steel	SA-376, Type 347 Modified
		Stainless steel	SA-403, GR WP304
		Stainless steel	SA-403, GR WP304H

TABLE 6.1-1 (Continued)

<u>Principal Component</u>	<u>Form</u>	<u>Material</u>	<u>ASME Specification</u>
		Carbon steel	SA-155, GR KCF70, C1-2
		Carbon steel	SA-155, GRKCF70, C1-1
		Carbon steel	SA-234, GR WPB
		Carbon steel	SA-106, GR B
		Alloy steel	SA-335, GR P11
		Alloy steel	SA-335, GR P22
		Alloy steel	SB-464
	Forgings	Stainless steel	SA-182, GR F304
		Stainless steel	SA-182, GR F316
		Carbon steel	SA-105
		Alloy steel	SA-182, GR F12
		Alloy steel	SA-182, GR F22
		Alloy steel	SA-182, GR F22
			C1-3
	Plate	Stainless steel	SA-240, Type 304
	Bolting	Alloy steel	SA-193, GR B7
	Nuts	Carbon steel	SA-194, GR 2H
Valves:			
	Castings	Stainless steel	SA-351, GR CF8M
		Stainless steel	SA-351, GR CF8
		Carbon steel	SA-216, GR WCB
		Stainless steel	SA-351, GR CF3M
		Alloy steel	SA-217, GR WC9
	Forgings	Stainless steel	SA-182, GR F316
		Stainless steel	SA-182, GR F304
		Alloy steel	SA-182, GR F11
		Alloy steel	SA-182, GR F12
		Alloy steel	SB-462
		Carbon steel	SA-105
		Alloy steel	SA-182, GR F22
		Alloy steel	SA-182, GR F22
			C1-3
	Plate	Carbon steel	SA-516, GR 70
	Bolts	Stainless steel	SA-564, Type 630
		Alloy steel	SA-193, GR BT
	Nuts	Stainless steel	SA-194, GR 8M
		Carbon steel	SA-194, GR 2H
	Bar	Stainless steel	SA-479, Type 316L
		Stainless steel	SA-564, Type 630 (per Code Case 1773)

TABLE 6.1-1 (Continued)

<u>Principal Component</u>	<u>Form</u>	<u>Material</u>	<u>ASME Specification</u>
Pumps:			
	Castings	Stainless steel	SA-351, GR CF8M
		Carbon steel	SA-216, GR WCB
	Plate	Stainless steel	SA-240, Type 304
		Carbon steel	SA-515, GR 70
	Pipe	Carbon steel	SA-106, GR B
		Carbon steel	SA-155, GR C55
	Bolts	Alloy steel	SA-193, GR B7
		Alloy steel	SA-193, GR B8
Vessels:			
	Plate	Stainless steel	SA-240, Type 304L
		Carbon steel	SA-283, GR C
	Pipe	Stainless steel	SA-312, Type 304L
Heat Exchangers:			
	Plate	Stainless steel	SA-240, GR 304L
		Carbon steel	SA-285, GR C
			SA-306, GR 60
			SA-515, GR 70
			SA-516, GR 70
	Pipe	Carbon steel	SA-53, GR B
	Forgings	Carbon steel	SA-181, GR I
	Extrusions (tubes)	Stainless steel	SA-249, GR 304L
		Copper	SB-359
	Bolts	Alloy steel	SA-193, GR B7
Nuts	Carbon steel	SA-194, GR 2H	

TABLE 6.1-2

ORGANIC COATING MATERIALS INSIDE REACTOR BUILDING

	Area	DFT _{max}	Materials	Material Weight	Total Weight
<u>Surface</u>	<u>(ft²)</u>	<u>(mils)</u>	<u>Type</u>	<u>(oz./mil-ft²)</u>	<u>(lbs)</u>
Steel ⁽¹⁾	842,954	8	Polyamide - cured epoxy	0.15	63,222
Concrete	92,485	20	Polyamide - cured epoxy	0.15	17,341
				Total	80,563
Steel ⁽²⁾	8,939	8	Water base and polyamide- cured epoxies	0.15	671
Steel ⁽²⁾	581	-	Powder coat	-	18

NOTES:

⁽¹⁾ Includes pipe, structural steel, plate, and equipment (approx.).

⁽²⁾ Applicable DBA tests are not available.

6.2 CONTAINMENT SYSTEMS

6.2.1 CONTAINMENT FUNCTIONAL DESIGN

6.2.1.1 Containment Structure

6.2.1.1.1 Design Bases

The pressure suppression containment system is designed to have the following functional capabilities:

- a. The containment and drywell have the capability to maintain functional integrity during and following peak transient pressures and temperatures which would occur following any postulated loss-of-coolant accident (LOCA). The LOCA includes the worst single failure (which leads to maximum containment and drywell pressure and temperature) and is further postulated to occur simultaneously with loss of offsite power and a safe shutdown earthquake (SSE). A detailed discussion of LOCA events is presented in <Section 6.2.1.1.3.3>. A detailed discussion of mass and energy released is presented in <Section 6.2.1.3>.
- b. The containment, in combination with other accident mitigation systems, limits fission product leakage during and following the postulated design basis accident to values less than leakage rates that would result in offsite doses greater than those stated in <10 CFR 50.67>.
- c. The containment system and drywell can withstand coincident fluid jet forces associated with the flow from the postulated rupture of any pipe within the containment or drywell.

- d. The containment design permits removal of fuel assemblies from the reactor core after the postulated LOCA.
- e. The containment system is protected from, or is designed to withstand, missiles from internal sources and excessive motion of pipes which could directly or indirectly endanger the integrity of the containment.
- f. The containment system provides means to channel the flow from postulated pipe ruptures in the drywell to the suppression pool.
- g. The containment system is designed to allow for periodically performing tests at the peak pressure calculated to result from the postulated design basis accident to confirm the leaktight integrity of the containment and containment penetrations.

6.2.1.1.2 Design Features

General layout drawings of the containment structure are provided by <Figure 1.2-3>, <Figure 1.2-4>, <Figure 1.2-5>, <Figure 1.2-6>, <Figure 1.2-7>, <Figure 1.2-8>, <Figure 1.2-9>, <Figure 1.2-10>, <Figure 1.2-11>, and <Figure 1.2-13>.

Design provisions for protection of the containment structure against internally and externally generated missiles are discussed in <Section 3.5>. Protection against pipe rupture is discussed in <Section 3.6>.

Codes, standards and guides applicable to the design of the containment and internal structures are addressed in <Section 3.8.1>, <Section 3.8.2> and <Section 3.8.3>.

The tests that demonstrate the functional capability of structural systems and components are discussed in <Section 6.2.1.6>.

The functional capability and frequency of operation of systems which maintain containment and subcompartment atmospheric conditions within limits during normal plant operation are discussed in <Section 9.4.6>.

Design provisions for protection of the containment structure against loss of integrity under external pressure loading conditions resulting from inadvertent operation of heat removal systems that could result in significant external structural loadings are described in <Sections 3.8.2> and <Section 6.2.1.1.4>.

6.2.1.1.3 Design Evaluation

6.2.1.1.3.1 Summary Evaluation

Key design parameters and the maximum calculated accident parameters for the pressure suppression containment are presented by <Table 6.2-1>. 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, no single design basis accident (DBA) for this containment system exists.

It is assumed for analytical purposes that the primary system and containment are initially at the maximum normal operating conditions. (Reference 1), (Reference 2), (Reference 3), and (Reference 4) describe relevant experimental verification of analytical models used to evaluate the containment system response.

6.2.1.1.3.2 Containment Design Parameters

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

<Table 6.2-3> provides a listing of performance parameters for the related engineered safety feature systems which supplement the design conditions presented by <Table 6.2-2> for containment cooling purposes during post-blowdown, long term accident operation. Performance parameters listed include those applicable to full capacity operation and to those conservatively reduced capacities assumed for containment analyses.

6.2.1.1.3.3 Accident Response Analysis

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

- a. Instantaneous guillotine rupture of a recirculation line.
- b. Instantaneous guillotine rupture of a main steam line.
- c. Rupture of an intermediate size liquid line.
- d. Rupture of a small size steam line.

Energy release resulting from these accidents is addressed in <Section 6.2.1.3>. The accident response analyses are discussed in detail in (Reference 31).

6.2.1.1.3.3.1 Recirculation Line Break

Immediately following rupture of a recirculation line, flow from both sides of the break is the maximum value limited by critical flow considerations. The total effective flow area is shown by <Figure 6.2-1>. On the side of the break adjacent to the suction nozzle, flow corresponds to critical flow in the pipe cross section. On the side adjacent to the injection nozzle, flow corresponds to critical

flow at the ten jet pump nozzles associated with the broken loop. In addition, the cleanup line crosstie adds to the critical flow area. <Table 6.2-4> summarizes the break areas. A penalty is added to the Peak Clad Temperature (PCT) to account for a flow path that is not in the base recirculation failure LOCA analysis. This flow path is from the bottom head drain of the Reactor Pressure Vessel to the recirculation loop and only impacts the recirculation line failure LOCA analysis. The penalty is added to the PCT values found in <Table 6.3-4> and <Table 15B.6.3-1>.

6.2.1.1.3.3.1.1 Assumptions for Reactor Blowdown

The response of the reactor coolant system during the blowdown period of the accident is analyzed using the following assumptions:

- a. Initial conditions for the recirculation line break accident are such that system energy is maximized and system mass is minimized. These conditions are as follows:
 1. The reactor is operating at 102 percent of rated power. This maximizes postaccident decay heat.
 2. Service water temperature is the maximum normal.
 3. Suppression pool mass is at the low water level with a maximum (positive) drywell-to-containment differential pressure (dP) for the long term analysis, and at the high water level with a minimum (negative) drywell-to-containment dP for the short term analysis.
 4. Suppression pool temperature is the maximum normal.

- b. The recirculation line is considered to be severed instantly. This results in the most rapid coolant loss and depressurization of the reactor pressure vessel. Coolant is discharged from both ends of the break.
- c. Reactor power generation ceases at the time of accident initiation as a result of void formation in the core region. Scram occurs less than one second after receipt of the high drywell pressure signal. The difference between shutdown times is negligible.
- d. Reactor pressure vessel depressurization flow rates are calculated using Moody's critical flow model (Reference 3), assuming "liquid only" outflow since this assumption maximizes the energy release to the drywell. "Liquid only" outflow implies that all vapor formed in the reactor pressure vessel by bulk flashing rises to the surface rather than being entrained in the existing flow. In reality, some of the vapor would be entrained in the break flow which would significantly reduce reactor pressure vessel discharge flow rates. Further, Moody's critical flow model, which assumes annular, isentropic flow, thermodynamic phase equilibrium, and maximized slip ratio, accurately predicts reactor pressure vessel outflows through small orifices. Actual rates through larger flow areas, however, are less than the model indicates because of the effects of a nearly homogeneous two phase flow pattern and phase nonequilibrium. These effects are conservatively neglected in the analysis.
- e. Core decay heat and sensible heat released in cooling the fuel to initial average coolant temperature are included in the reactor pressure vessel depressurization calculation. The rate of energy release is calculated using a conservatively high heat transfer coefficient throughout the depressurization period. The resulting high energy release rate causes the reactor pressure vessel to maintain nearly rated pressure for approximately 20 seconds. The

high reactor pressure vessel pressure increases the calculated blowdown flow rates. This, again, is conservative for analytical purposes. The sensible energy of the fuel stored at temperatures below the initial average coolant temperature is released to the reactor pressure vessel fluid along with the stored energy in the vessel and internals as vessel fluid temperatures decrease during the remainder of the transient calculation.

- f. The main steam isolation valves start closing at 0.5 seconds after the accident. These valves are fully closed in the shortest possible time of three seconds following closure initiation. Actually, the closure signal for the main steam isolation valves occurs as a result of low reactor water level. Therefore, the valves do not receive a signal to close for more than four seconds and the closing time may be as long as five seconds. By assuming rapid closure of the main steam isolation valves, the reactor pressure vessel is maintained at a high pressure which maximizes the calculated discharge of high energy water in the drywell.
- g. A complete loss of offsite power occurs simultaneously with the pipe break. This condition results in the loss of power conversion system equipment and also requires that all vital systems for long term cooling be supported by onsite power supplies.

6.2.1.1.3.3.1.2 Assumptions for Containment Pressurization

The pressure response of the containment during the first 10 seconds of the accident is analyzed using the following assumptions:

- a. Thermodynamic equilibrium exists in the drywell and containment. Since highly turbulent conditions are expected due to the blowdown flow, the analysis assumes complete mixing.

- b. Fluid flowing through the drywell to suppression pool vents is formed from a homogeneous mixture of the fluid in the drywell. Use of this assumption results in complete carryover of the drywell air and a higher positive flow rate of liquid droplets which conservatively maximizes vent pressure losses.
- c. Fluid flow in the drywell to suppression pool vents is compressible, except for the liquid phase.
- d. No heat loss occurs from the gases inside containment. Actually, condensation of some steam on the drywell surfaces would occur.

6.2.1.1.3.3.1.3 Assumptions for Long Term Cooling

Following the initial blowdown period, the emergency core cooling system (ECCS), discussed in <Section 6.3>, provides water for core flooding, containment spray and long term decay heat removal. The containment pressure and temperature response during this period is analyzed using the following assumptions:

- a. The low pressure core injection (LPCI) pumps are used to flood the core prior to 600 seconds after the accident. The high pressure core spray (HPCS) is available for the entire accident.
- b. The effects on suppression pool temperature decay energy, stored energy, sensible energy, energy added by ECCS pumps, and energy from the zirconium water reaction are considered.
- c. The suppression pool and the structures inside the containment are the only heat sinks available in the containment system. After a certain period, makeup water from the upper containment pool is included.

- d. After approximately 1,980 seconds, the residual heat removal (RHR) heat exchangers are activated to remove energy from the containment by means of recirculation cooling of the suppression pool with the emergency service water system. It is conservatively assumed that containment spray is not used.

Performance of the ECCS equipment during the long term cooling period is evaluated for each of the following cases of interest:

- a. Case A: Offsite Power Available, All ECCS Equipment Operating.
- b. Case B: Loss of Offsite Power, Minimum Diesel Power Available for ECCS.

6.2.1.1.3.3.1.4 Initial Conditions for Accident Analysis

<Table 6.2-4> provides the initial conditions and numerical values assumed for the recirculation line break accident, as well as the sources of energy considered prior to the postulated pipe rupture. The assumed conditions for the reactor blowdown are also provided.

<Table 6.2-5> presents the initial reactor coolant system and containment conditions used in all the accident response evaluations. This tabulation includes parameters for the reactor, drywell and containment.

Mass and energy release sources and rates for the containment response analyses are addressed in <Section 6.2.1.3>.

6.2.1.1.3.3.1.5 Short Term Accident Response

The calculated containment pressure and temperature responses for the recirculation line break are shown by <Figure 6.2-2> and <Figure 6.2-3>, respectively. Following the break, drywell pressure increases rapidly

due to the injection of the break flow. The peak drywell pressure occurs during the vent clearing phase of the transient as suppression pool water is being cleared from the vents. Following vent clearing, drywell pressure decreases as break flow decreases.

The containment is pressurized early in the transient by the carryover of noncondensibles from the drywell. As the transient continues, break flow is injected into the suppression pool and the temperature of the suppression pool water increases, causing containment pressure to increase. At the end of the blowdown, drywell pressure stabilizes at a slightly higher pressure than the containment, the difference being equal to the hydrostatic head of vent submergence. During the reactor pressure vessel depressurization phase most of the noncondensable gases initially in the drywell are forced into containment. However, following depressurization the noncondensibles are redistributed between the drywell and containment through the vacuum breaker system. This redistribution occurs as steam in the drywell is condensed by the relatively cool ECCS water which begins to cascade from the break causing drywell pressure to decrease (drywell vacuum breaker operation for this condition is not an ESF function).

The ECCS supplies sufficient core cooling water to control core heatup and limit metal-water reaction to less than one percent. After the reactor pressure vessel is flooded to the height of the jet pump nozzles, 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 reactor pressure vessel through the broken recirculation line in the form of hot water. This hot water flows into the suppression pool through the drywell to suppression pool vent system.

<Table 6.2-6> lists the peak pressure, temperature and time parameters for the recirculation line break as predicted for the conditions presented by <Table 6.2-4> and <Table 6.2-5>, and corresponds with

<Figure 6.2-2> and <Figure 6.2-3>. <Figure 6.2-2> includes the time dependent response of the drywell differential pressure.

During the blowdown period of the LOCA the pressure suppression vent system 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. <Figure 6.2-5> provides the mass flow through the vent system versus time relationship for this accident.

6.2.1.1.3.3.1.6 Long Term Accident Responses

To assess the adequacy of the containment following the initial blowdown transient, an analysis of the long term temperature and pressure response following the accident was performed. The assumptions used in this analysis are those discussed in <Section 6.2.1.1.3.3.1.3> for the two cases of interest.

a. Case A: Offsite Power Available, All ECCS Equipment Operating (3729 MWt)

This case assumes that offsite ac power is available to operate all cooling systems. During the first 1,800 seconds following the pipe break HPCS, low pressure core spray (LPCS), and all LPCI pumps are assumed to be operating. All flow is injected directly into the reactor vessel.

After 1,980 seconds both RHR heat exchangers are activated to remove energy from the containment. In this mode of operation LPCI flow is routed through both RHR heat exchangers where the fluid is cooled before being returned to the suppression pool.

The containment pressure response under recirculation line break is bounded by the maximum ECCS curve of <Figure 6.2-6>. Drywell and

suppression pool temperature responses for Case A are bounded by the maximum ECCS curves of <Figure 6.2-7> and shown on <Figure 6.2-8> respectively. After the initial blowdown and subsequent depressurization, core decay heat results in a gradual pressure and temperature rise in containment. When the energy removal rate of the RHR System equals the energy addition rate from decay heat, containment pressure and temperature reach a second peak value and then decrease gradually. <Table 6.2-7> summarizes equipment operation, the peak long term containment pressure and the peak suppression pool temperature.

b. Case B1: Loss of Offsite Power, Minimum ECCS Equipment Operating (3729 MWt)

This case assumes that no offsite power is available following the accident and that only minimum diesel generator power is available. After 1,980 seconds LPCI flow through one RHR heat exchanger is returned to the suppression pool. The containment pressure response under this set of conditions is bounded by the minimum ECCS curve of <Figure 6.2-6>. Drywell and suppression pool temperature responses for Case B are bounded by the minimum ECCS curves of <Figure 6.2-7> and shown on <Figure 6.2-8> respectively. A summary for this case is presented by <Table 6.2-7>.

<Figure 6.2-9> shows the rate at which the RHR heat exchanger removes heat from the suppression pool following a LOCA (<Section 6.2.2> describes the containment cooling mode of RHR System operation). The heat removal rate is shown by both Case A and Case B. The first case assumes that all ECCS equipment is available, including both RHR heat exchangers and the associated emergency service water pumps. The second case assumes the very degraded minimum cooling condition that limits heat removal capacity to one heat exchanger. For both cases it is conservatively assumed that all the time of the accident the

emergency service water is at the maximum design temperature as defined by <Table 6.2-3>.

- c. Case B2: Loss of Offsite Power, Minimum ECCS Equipment Operating (3833 MWt)

Case B1 (above) is reanalyzed at 102% core thermal power. <Figure 6.2-6a>, <Figure 6.2-7a>, and <Figure 6.2-8a> show the corresponding results of that evaluation.

6.2.1.1.3.3.1.7 Energy Balance during Accident

The following energy sources and sinks are required to establish an energy distribution in containment as a function of time (short term, long term) for this accident:

- a. Blowdown energy release rates.
- b. Decay heat rate and fuel relaxation sensible energy.
- c. Sensible heat rate (vessel and internals).
- d. Pump heat rate.
- e. Rate of removal of heat from the suppression pool <Figure 6.2-9>.
- f. Metal-water reaction heat rate.
- g. Rate of heat transfer to structural heat sinks.

A further discussion of Items a. through d. and f. above, is provided in <Section 6.2.1.3>.

6.2.1.1.3.3.1.8 Chronology of Accident Events

A complete description of the containment response to the recirculation line break is presented by <Section 6.2.1.1.3.3.1.5>, <Section 6.2.1.1.3.3.1.6>, and <Section 6.2.1.1.3.3.1.7>. Results for this accident are illustrated by <Figure 6.2-2>, <Figure 6.2-3>, <Figure 6.2-4>, and <Figure 6.2-5>, <Figure 6.2-8>, <Figure 6.2-8a>, and <Figure 6.2-9> and bounded by <Figure 6.2-6>, <Figure 6.2-6a>, <Figure 6.2-7>, and <Figure 6.2-7a>.

6.2.1.1.3.3.2 Main Steam Line Break

The postulated sudden rupture of a main steam line between the reactor pressure vessel and the flow limiter results in the maximum rate of primary system fluid flow and energy transfer to the drywell. This, in turn, results in the maximum drywell differential pressure. Steam flow immediately following rupture of a main steam line between the reactor pressure vessel and the break accelerates to the maximum allowed by the critical flow considerations. On the side adjacent to the reactor pressure vessel the flow corresponds to critical flow in the main steam line break area. Blowdown through the other side of the break occurs because the main steam lines are all interconnected by the bypass header at a point upstream of the turbine. This interconnection allows primary system fluid to flow from the three unbroken steam lines through the header back into the drywell through the broken line. Flow is limited by critical flow in the steam line flow restrictor. The total effective flow area for the main steam line break is given by <Figure 6.2-10>.

6.2.1.1.3.3.2.1 Assumptions for Reactor Blowdown

The response of the reactor coolant system during the blowdown period of the accident is analyzed using the assumptions listed in <Section 6.2.1.1.3.3.1.1> for the recirculation line break, with the following exceptions:

- a. Reactor pressure vessel depressurization flow rates are calculated using Moody's critical flow model (Reference 3). During the first second of blowdown the flow consists of saturated steam.

Immediately following the break the total steam flow rate leaving the reactor pressure vessel exceeds the rate of steam generation in the core. This causes an initial depressurization of the reactor pressure vessel. Void formation in the water within the reactor pressure vessel causes a rapid rise in vessel water level. It is conservatively assumed that water level reaches the vessel steam nozzles one second after the break occurs. The water level rise time of one second is the minimum that could occur under any reactor operating condition. After one second, a two-phase mixture is discharged from the break.

- b. The main steam isolation valves start to close at 0.5 seconds after the accident and are fully closed in the maximum time of five seconds after initiation of closure. By assuming slow closure of these valves, a large effective break area is maintained for a longer period of time. Peak drywell pressure occurs before the reduction in effective break area and is, therefore, insensitive to any additional delay in closure of the main steam isolation valves.

6.2.1.1.3.3.2.2 Assumptions for Containment Pressurization

The pressure response of the containment during the blowdown period of the accident is analyzed using the assumptions listed in <Section 6.2.1.1.3.3.1.2>.

6.2.1.1.3.3.2.3 Assumptions for Long Term Cooling

The containment pressure and temperature response during the period following blowdown is analyzed using the assumptions listed in <Section 6.2.1.1.3.3.1.3>.

6.2.1.1.3.3.2.4 Initial Conditions for Accident Analyses

<Table 6.2-4> lists the initial conditions and numerical values assumed for the main steam line break accident, as well as the sources of energy considered prior to the postulated pipe rupture. Assumed conditions for the reactor blowdown are also provided.

<Table 6.2-5> lists the initial reactor coolant system and containment conditions used in all the accident response evaluations. This tabulation includes parameters for the reactor, drywell and containment.

Mass and energy release sources and rates for the containment response analyses are presented in <Section 6.2.1.3>.

6.2.1.1.3.3.2.5 Short Term Accident Response

<Figure 6.2-11> and <Figure 6.2-12> show the pressure and temperature responses of the drywell and suppression pool during the first 10 seconds following the main steam line break accident. <Figure 6.2-13> shows the response of the drywell differential pressure and <Figure 6.2-14> shows the vent mass flow versus time.

The drywell atmosphere temperature approaches a peak after approximately one second of primary system blowdown. At that time, water level in the reactor pressure 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. The peak differential pressure occurs shortly after the vent clearing transient. As the

blowdown proceeds, the primary system pressure and fluid inventory decrease, resulting in reduced break flow rates. As a consequence the flow rate in the vent system and the differential pressure between the drywell and suppression pool begin to decrease.

<Table 6.2-6> presents the peak pressures, peak temperatures and times of this accident in comparison with similar parameters for the recirculation line break.

6.2.1.1.3.3.2.6 Long Term Accident Responses

After the primary system pressure has decreased to the drywell pressure, the blowdown is over. At this time the drywell contains saturated steam and drywell and containment pressures stabilize. The pressure difference corresponds to the hydrostatic pressure of vent submergence.

The drywell and suppression pool remain in this equilibrium condition until the reactor pressure vessel is reflooded. 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. At this point the ECCS flow spills into the drywell. The water spillage condenses the steam in the drywell, reducing drywell pressure. As drywell pressure drops below containment pressure the noncondensable gases from containment flow into the drywell until pressures in the two regions are equalized.

The long term containment pressure and temperature responses following the accident are identical to those described in <Section 6.2.1.1.3.3.1.6> for the recirculation line break. The results are shown on or bounded by <Figure 6.2-6>, <Figure 6.2-7>, <Figure 6.2-7a>, <Figure 6.2-8>, <Figure 6.2-8a>, and <Figure 6.2-9>. <Table 6.2-7> summarizes cooling equipment operation, peak long term containment pressure and peak suppression pool temperature at 3729 MWt. <Table 6.2-7a> summarizes the results at 3833 MWt.

6.2.1.1.3.3.2.7 Energy Balance during Accident

The following energy sources and sinks are required to establish an energy distribution in containment as a function of time (short term, long term) for this accident:

- a. Blowdown energy release rates.
- b. Decay heat and fuel relaxation sensible energy.
- c. Sensible heat rate (reactor pressure vessel and internals).
- d. Pump heat rate.
- e. Rate of removal of heat from the suppression pool <Figure 6.2-9>.
- f. Metal-water reaction heat rate.
- g. Rate of heat transfer to structural heat sinks.

A further discussion of Items a. through d. and f. above, is provided in <Section 6.2.1.3>. A complete energy balance for the main steam line break accident is provided by <Table 6.2-8> for the reactor system, containment and containment cooling systems at time zero, time of peak drywell pressure, time of end of reactor blowdown, and time of long term peak pressure in containment.

6.2.1.1.3.3.2.8 Chronology of Accident Events

A complete description of the containment response to the main steam line break is presented in <Section 6.2.1.1.3.3.2.5>, <Section 6.2.1.1.3.3.2.6>, and <Section 6.2.1.1.3.3.2.7>. Results for this accident analysis are shown by <Figure 6.2-6>, <Figure 6.2-6a>, <Figure 6.2-7>, <Figure 6.2-7a>, and <Figure 6.2-10>, <Figure 6.2-11>.

<Figure 6.2-12>, <Figure 6.2-13>, and <Figure 6.2-14>. A chronological sequence of events for this accident from time zero is provided by <Table 6.2-9>. This chronology is based on a power level of 3729 MWt.

6.2.1.1.3.3.3 Hot Standby Accident Analysis

This accident was not reanalyzed for power uprate since it was not the limiting case. The analysis presented below is based on a power of 3729 MWt.

Both the short term and long term response of the containment system have been evaluated, assuming that the reactor has been operating in the hot standby mode prior to the LOCA.

The peak drywell pressure following a main steam line break is dependent upon the rise time of the reactor pressure vessel water level since this determines the time at which two phase blowdown begins. A level rise time of one second is a conservative bounding condition for a main steam line break at a reduced reactor power level. However, since a one second level rise time was conservatively assumed for the LOCA at 104.2 percent of rated power, the peak drywell pressure following a blowdown at hot standby will be no higher than shown by <Figure 6.2-11>.

In the event of a recirculation line break, the short term blowdown flow rate is essentially independent of reactor power level if the same initial reactor pressure is assumed for all power levels. In practice the lower reactor pressures associated with reduced reactor power result in lower blowdown flow rates and in peak drywell pressures lower than the value presented by <Figure 6.2-2>. The short term drywell response to either a main steam line or recirculation line break is insensitive to suppression pool water temperature. This insensitivity is due to domination of the transient by the rate at which energy is transferred to the drywell and the rate at which vent clearing can be accomplished. Neither is sensitive to suppression pool water temperature.

The long term suppression pool and containment transient is only affected very slightly by a period of hot standby operation prior to blowdown. <Figure 6.2-15> presents a comparison of suppression pool temperature transients following a blowdown under the following conditions:

- a. Maximum normal suppression pool water temperature and 104.2 percent of rated power.
- b. Approximately 700 seconds of hot standby operation.

Since in both cases containment cooling (RHR System) is initiated 1,980 seconds after the start of the event, the peak long term suppression pool temperature is about the same, as can be seen from <Figure 6.2-15>.

6.2.1.1.3.3.4 Intermediate Size Breaks

This accident was not reanalyzed for power uprate since it was not the limiting case. The analysis presented below is based on a power of 3729 MWt.

The classification, intermediate size breaks, includes those breaks, the blowdown from which results in reactor depressurization and operation of the ECCS. This section describes the consequences to the containment of a 0.68 ft² break below the reactor pressure vessel water level. This break was chosen as being representative of the intermediate size break range. Such breaks can involve either reactor steam or liquid blowdown.

Following the occurrence of the 0.68 ft² break, drywell pressure increases at approximately one psi/sec. This transient is sufficiently slow that the dynamic effect of the water in the vents is negligible and the vents clear when the drywell to containment differential pressure is equal to the vent submergence hydrostatic pressure.

<Figure 6.2-16> and <Figure 6.2-17> show the drywell and containment pressure and temperature response, respectively. The ECCS response is discussed in <Section 6.3>. Approximately five seconds after the 0.68 ft² 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 containment free space. The continual purging of drywell air to the containment results in a gradual pressurization of both containment and drywell. Containment pressure continues to gradually increase due to the long term suppression pool heatup.

The ECCS is initiated as a result of the 0.68 ft² break and provides emergency cooling of the core. Operation of the ECCS is such that the reactor is depressurized in approximately 600 seconds, terminating the blowdown phase of the transient.

In addition, the suppression pool temperature at the end of blowdown is the same as that of the main steam line break because essentially the same amount of primary system energy is released during the blowdown. After reactor pressure vessel depressurization and reflood, water from the ECCS begins to flow out the break. This flow condenses the drywell steam and eventually causes the drywell and containment pressures to equalize in the same manner as described for a main steam line break.

The subsequent long term suppression pool and containment heatup transient that follows is essentially the same as for the main steam line break.

6.2.1.1.3.3.5 Small Size Breaks

This accident was not reanalyzed for power uprate since it was not the limiting case. The analysis presented below is based on a power of 3729 MWt.

6.2.1.1.3.3.5.1 Reactor System Blowdown Considerations

This section discusses the containment transient associated with small primary system blowdowns. The sizes of primary system ruptures in this category are those blowdowns that do not result in reactor pressure vessel depressurization due either to loss of reactor coolant or automatic operation of ECCS equipment. Following the occurrence of a break of this size, it is assumed that the reactor operators initiate an orderly plant shutdown and depressurization of the reactor system. The thermodynamic process associated with blowdown of primary system fluid is one of constant enthalpy. If the primary system break is below the water level, blowdown flow consists of reactor water. Blowdown from reactor pressure to drywell pressure results in flashing of approximately one third of the blowdown water to steam. Two thirds of the water remains liquid. Both phases are at saturation conditions corresponding to the drywell pressure. Thus, if the drywell is at atmospheric pressure, for example, the steam and liquid associated with a liquid blowdown would be at a temperature of 212°F.

If the primary system rupture is so located that the blowdown flow consists only of reactor steam, the resultant steam temperature in drywell is significantly higher than the temperature associated with liquid blowdown. This is because the constant enthalpy depressurization of high pressure, saturated steam results in superheated conditions. For example, decompression of saturated steam at 1,000 psia to atmospheric pressure results in superheated steam at 298°F (86°F of superheat).

A small reactor steam leak resulting in superheated steam imposes the most severe temperature conditions on the drywell structures and on the safety equipment in the drywell. For larger steam line breaks, the superheat temperature is nearly the same as for small breaks but the duration of the high temperature condition is less for the larger break. This is a result of the more rapid depressurization of the reactor

pressure vessel through the larger breaks. Depressurization is slower for the orderly shutdown assumed to terminate the small break.

6.2.1.1.3.3.5.2 Containment Response

For drywell design considerations, the following sequence of events is assumed to occur. With the reactor and containment operating under normal maximum 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 initiates the containment isolation logic. Drywell pressure continues to increase at a rate dependent upon the size of the steam leak. The pressure increase lowers the water level in the annulus until the level begins to clear the vents. At this time, air and steam start to enter the suppression pool. The steam is condensed and the air carries over to the containment free space. The air carryover results in a gradual pressurization of the containment at a rate dependent upon the size of the steam leak. Once all of the drywell air is carried over into the containment, short term containment pressurization ceases and the system reaches an equilibrium condition. The drywell contains only superheated steam and continued blowdown of reactor steam is condensed in the suppression pool. Suppression pool temperature continues to rise until the RHR heat exchanger heat removal rate equals the decay heat release rate.

6.2.1.1.3.3.5.3 Recovery Operations

The reactor operators are alerted to the small size break incident by the high drywell pressure signal and reactor scram. For purposes of evaluating the duration of the superheat condition in the drywell, it is assumed that the response of the operators is to shut down the reactor in an orderly manner, using the main condenser, while limiting the reactor cooldown rate to 100°F/hr. This results in depressurization of the primary system within six hours. At this time blowdown flow to the

drywell ceases and the superheat condition is terminated. If the operators elect to cool down and depressurize the primary system more rapidly than at 100°F/hr, the duration of the drywell superheat condition is shorter.

6.2.1.1.3.3.5.4 Drywell Design Temperature Considerations

For drywell design purposes it is assumed that there is a blowdown of reactor steam for the six hour cooldown period. The corresponding design temperature is determined by finding the combination of primary system pressure and drywell pressure that produces the maximum superheat temperature. The maximum drywell steam temperature occurs when the primary system is at approximately 450 psig and the drywell pressure is maximum. Thus, for design purposes, it is assumed that the drywell is at 15 psig. This results in temperatures of 330°F for the first three hours of the cooldown period, and 310°F for the next three hours.

6.2.1.1.3.4 Accident Analysis Models

6.2.1.1.3.4.1 Short Term Pressurization Model

The analytical models, assumptions and methods used by GE to evaluate the containment response during the reactor pressure vessel blowdown phase of a LOCA are described in (Reference 1) and (Reference 2).

6.2.1.1.3.4.2 Long Term Cooling Model

Once the reactor pressure vessel blowdown phase of the LOCA is over, a fairly simple model of the drywell and containment is used. During the long term post-blowdown transient, the RHR containment cooling system flow path is a closed system and the suppression pool mass is constant. The cooling loop model used for analysis is schematically illustrated by <Figure 6.2-18>.

The analytical models, assumptions and methods used by GE to evaluate the containment response during the long term cooling phase of a LOCA are described in (Reference 2).

6.2.1.1.3.5 High Energy Line Rupture Inside Containment

Some primary system pipes are routed from the drywell through the containment to the auxiliary building (main steam lines for example). If such pipes were unguarded, rupture within containment would result in direct release of primary system fluid to the containment atmosphere. The pressure suppression features of the containment would thus be bypassed and the potential would exist for a pipe rupture to produce significant containment pressures.

Because of the potential for overpressurizing containment, all reactor coolant pressure boundary pipes of a size which would result in containment overpressurization and which pass through the containment except LPCI, HPCS and LPCS pipes, are provided with guard pipes that vent to the drywell. Thus, in the event of rupture of a reactor coolant pressure boundary pipe, flow passes through the suppression pool vent system and the steam is condensed.

The LPCI, HPCS and LPCS pipes have check valves inboard of the drywell penetration that prevent blowdown to the containment.

The traversing incore probe, control rod drive insert and withdraw and instrument lines could discharge primary coolant to the containment in the event of a rupture. The unisolatable instrument line rupture results in the maximum discharge of primary coolant to containment. This accident is discussed in <Chapter 15>. Each instrument line includes a 1/4 inch diameter flow restricting orifice to limit the containment pressure increase to values well below the design pressure.

The major components of the reactor water cleanup system are located within containment and system piping could also discharge primary coolant to the containment in the event of rupture. The system suction line penetrates the drywell and is provided with a guard pipe. System components located inside containment are provided with break detection and isolation systems that limit the total blowdown fluid flow to containment to acceptable values.

6.2.1.1.4 Negative Pressure Design Evaluation

6.2.1.1.4.1 Evaluation of Drywell Negative Differential Pressure

Following the blowdown phase of a LOCA, air initially contained in the drywell has been purged to the containment and the drywell is full of steam. During this period the ECCS is injecting cooling water from the suppression pool into the reactor pressure vessel. When the reactor pressure vessel is flooded to the level of the break, water begins spilling into the drywell, condensing the steam and causing rapid depressurization of the drywell. A bounding calculation of the peak drywell negative differential pressure is based upon the following conservative assumptions:

- a. All air has been purged out of the drywell.
- b. Drywell vacuum breakers do not open.
- c. The suppression pool is at the design temperature of 185°F.
- d. The containment is at suppression pool temperature and 100 percent relative humidity.
- e. Steam in the drywell is cooled to suppression pool temperature.

Using the above assumptions, the final drywell pressure is equal to saturation pressure at 185°F or:

$$P_d = 8.38 \text{ psia}$$

Based upon the initial conditions listed in <Table 6.2-5>, the initial air masses in the drywell and containment are:

$$M_d = 19,193 \text{ lb}_m \quad (145^\circ\text{F at drywell})$$

$$M_c = 78,106 \text{ lb}_m$$

Using the assumptions in Items a. through e., above, the final containment pressure for the purposes of determining the drywell negative differential pressure is calculated as the summation of the partial pressures of air and vapor:

$$P_c = 28.48 \text{ psia}$$

Thus, the bounding negative pressure load across the drywell wall is:

$$\Delta P_d = P_d - P_c$$

$$\Delta P_d = -20.1 \text{ psid}$$

6.2.1.1.4.2 Evaluation of Containment Negative Pressure

If the containment spray system is activated during any period of time other than when it is designed to be operated, it is possible that a vacuum could be created inside the containment vessel. If the containment spray system is accidentally activated, an excessive vacuum is prevented from developing by means of vacuum breakers provided for this purpose.

Containment vacuum relief capability is necessary only to maintain containment integrity should the containment spray system be operated in such a way as to tend to create a vacuum inside containment. Although the containment spray system is adequately protected against

inadvertent, unintentional or incorrect operation by interlocks and administrative procedures <Section 6.2.2>, two hypothetical situations are considered, assuming these protective measures are bypassed in some manner and the spray is started:

- a. For the first situation, pressure and temperature conditions for the containment atmosphere are based on the following sequence of events:
 1. The atmosphere inside containment is at normal pressure and temperature.
 2. A 6-inch reactor water cleanup (RWCU) line break occurs inside containment.
 3. Isolation of the RWCU line is complete at 40 seconds after the accident. At this time, the containment pressure is 5 psig, temperature is 157°F, and the containment has been isolated.
 4. The vacuum breakers between the containment and drywell <Section 3.8.3> open to equalize the pressure of the containment and drywell. During this pressure equalization period, a portion of the containment air is swept into the drywell through the drywell vacuum breakers and remains there when the vacuum breakers close.

The maximum amount of air drawn into the drywell from containment is simply the difference between the amount of air in containment during normal operation and the amount of air in containment after the RWCU line isolation, assuming a slightly conservative relative humidity of 100 percent.

	<u>Normal Operating</u> <u>Conditions</u>	<u>RWCU Line</u> <u>Isolation</u>
Pressure, psig	0.0	5.0
Temperature, °F	90.0	157.0
Relative Humidity, %	50.0	100.0
Mass of air, lb _m	82,200.0	78,100.0

5. The containment spray system is activated at a flow rate of 10,500 gpm and a temperature of 60°F (minimum suppression pool temperature). Note: System flow rates of 6,000 gpm per spray loop (12,000 gpm total) are possible. A qualitative analysis indicated that the containment negative pressure analysis remains valid when considering an 87% spray efficiency for this event.
6. The containment is depressurized as a result of steam being condensed by the containment spray; the spray droplets are assumed to have an efficiency of 100 percent. Three cases are considered in sizing the containment vacuum breakers:
 - (a) Condensation rate including heat transfer structures with initial temperature of 80°F.
 - (b) Condensation rate including heat transfer structures with initial temperature of 90°F.
 - (c) The effects of the internal surfaces on the condensation rate are not considered. This is the limiting case.

The resulting pressure-temperature history within containment is calculated using the digital computer code CONTEMPT (Reference 5).

7. <Figure 6.2-19> and <Figure 6.2-20> present the results of the analyses for Cases a and b, respectively. Curves are plotted on these figures showing the containment vapor pressure and temperature, the total vapor mass, and the surface temperature of the internal concrete and steel liner as a function of time after RWCU isolation.

<Figure 6.2-21> presents the results for Case c.

As indicated by comparing these three figures the net effect of considering the internal surfaces is a lesser vacuum condition than when the effects of the internal surfaces are not considered. The peak vacuum calculated is 0.70 psig.

- b. For the second situation, it is assumed that the containment depressurization is a result of accidental initiation of the containment spray system during normal plant operation. The conditions present in the containment at the time of spray initiation are chosen to provide the most conservative results:

1. Maximum temperature in containment during normal operation: 105°F.
2. Minimum relative humidity in containment during normal operation: 30 percent.
3. Minimum spray water temperature: 60°F.

The source for the containment spray system water is the suppression pool. The normal water temperature in the pool is 90°F or equal to the normal operating temperature in the containment vessel outside the drywell. The minimum temperature of 60°F for the suppression pool is based on the minimum ambient air temperature (60°F) in the drywell and

containment vessel which could occur only under shutdown conditions. As described in <Section 9.4.6>, the reactor building ventilation system is designed to assure that the temperature of the containment atmosphere never falls below 60°F.

4. Spray system flow rate: 10,500 gpm Note: System flow rates of 6,000 gpm per spray loop (12,000 total) are possible. A qualitative analysis indicated that the containment negative pressure analysis remains valid when considering a flow rate of 12,000 gpm combined with a 95°F initial containment temperature (limited by the Technical Specifications) and assuming an 87% spray efficiency for this event.
5. During the evaporative cooling phase of the transient, the water drops sprayed into the containment absorb heat from the containment atmosphere and evaporate to contribute to saturating the containment atmosphere. This process is generally very rapid for spray rates typical of containment spray systems, and therefore, the time to complete the evaporative cooling process is important to compare to the vacuum breaker response time. In the calculations presented here the following assumptions are made:
 - (a) Spray efficiency is 100 percent.
 - (b) All of the spray water entering the containment is immediately vaporized and forms a homogeneous mixture with the containment atmosphere. Because of this conservative assumption, no detailed analytical heat/mass transfer modeling of the spray droplets is required.
 - (c) No heat is transferred back into the containment atmosphere from the structures during the transient.

- (d) The mass flow through two 24-inch diameter containment vacuum relief lines as a function of pressure differential is shown on <Figure 6.2-22>.

The equations used in the analyses are derived from the conservation of mass and energy and the equations of state of water and air. Utilizing the above assumptions the equations can be written as:

- (a) Conservation of mass:

$$\frac{dm_g}{dt} = \dot{m}_{fg}$$

$$\frac{dm_f}{dt} = \dot{m}_s - \dot{m}_{fg}$$

The assumption of instantaneous evaporation of the spray water defines the following additional relationship:

$$\dot{m}_{fg} = \dot{m}_s$$

and therefore,

$$\frac{dm_f}{dt} = \dot{m}_s - \dot{m}_{fg} = 0$$

- (b) Conservation of energy:

From these equations it can be seen that the equation for the conservation-of-energy includes the effect of evaporative cooling. This equation is given as:

$$\frac{d}{dt} [m_a u_a + m_g u_g + m_f u_f] = \dot{m}_s h_f = \dot{m}_{fg} h_f$$

For any addition of low enthalpy spray water, the heat required for evaporation lowers the total energy (temperature) of the vapor region, thereby lowering the partial pressure of the air (constant mass with decreasing temperature) while increasing the partial pressure of the vapor (result of increasing mass of vapor being more dominant than decreasing temperature).

(c) Equation of state:

$$p_a V = m_a R_a T$$

$$p_g V = m_g R_g T$$

Where:

m_a = Mass of air, lb_m

m_f = Mass of liquid water, lb_m

m_g = Mass of vapor, lb_m

\dot{m}_s = Rate of spray water, lb_m/sec

\dot{m}_{fg} = Rate of evaporation, lb_m/sec

u_a = Internal energy of air, Btu/lb_m

u_f = Internal energy of liquid water, Btu/lb_m

u_g = Internal energy of vapor, Btu/lb_m

h_f = Enthalpy of spray water, Btu/lb_m

p_a = Partial pressure of air, psi

p_g = Partial pressure of vapor, psi

R_a = Gas constant of air, $\text{ft}\cdot\text{lb}_f/^\circ\text{R}\cdot\text{lb}_m$

R_g = Gas constant of vapor, $\text{ft}\cdot\text{lb}_f/^\circ\text{R}\cdot\text{lb}_m$

T = Temperature of the containment atmosphere, $^\circ\text{R}$

V = Volume of the containment, ft^3

The Hamming's modified predictor - corrector method is used to obtain an approximate solution of the equations listed above (Reference 6).

6. After the evaporative cooling phase of the analyses is complete, the pressure - temperature history of the containment in the saturated condition is calculated utilizing the digital computer code CONTEMPT (Reference 5).

<Figure 6.2-23> shows the pressure and temperature inside containment as a function of time after start of the containment spray system. As indicated on this curve the resulting peak vacuum calculated is 0.72 psig.

In addition, <Figure 6.2-24> shows a curve of initial containment temperature versus relative humidity which results in a peak calculated vacuum of approximately 0.7 psig. This curve is used in establishing plant technical specifications.

As a result of these analyses, a value of 0.8 psi is selected as the design negative pressure differential between the outside atmosphere and containment atmosphere (acting inward on the containment vessel).

The containment vacuum relief system is available for any postulated situation requiring relief of a vacuum inside containment. This would include the rupture of a RWCU line. For this event, the containment vacuum relief isolation valves would initially be closed by high drywell pressure. After initiation of the spray system differential pressure switches in the containment override the containment isolation signal and open the vacuum relief isolation valves when the outside to containment atmosphere differential pressure has decreased to 0.0 psid. Vacuum breaker opening for other than the inadvertent containment spray actuation scenarios presented in this section is considered a non-ESF actuation. Conditions which result in changes to the containment

atmosphere, (such as containment venting and cooling, suppression pool level changes and weather changes), and subsequently result in containment vacuum breaker operation, are considered normal and expected, and are bounded by the limiting inadvertent containment spray situations. <Section 7.3.1> gives details of the instrumentation and logic for this system.

6.2.1.1.4.2.1 Vacuum Breaker Design

Two 24-inch nominal diameter vacuum relief lines are provided to obtain the vacuum relief cross-sectional area required to prevent the negative pressure inside containment from exceeding the design value of 0.8 psi during the hypothetical situations presented in this section. Two additional 24-inch nominal diameter vacuum relief lines are provided for redundancy.

Each vacuum relief line has a 24-inch nominal diameter, free swinging, simple check valve inside containment. This check valve serves as both the vacuum breaker device and the inner isolation valve for the vacuum relief line. Outside containment, each vacuum relief line has a 24-inch nominal diameter motor-operated butterfly valve to serve as the outer isolation valve.

A combination of any two vacuum relief lines provides a 10 percent margin between the maximum calculated negative pressure and the design negative pressure. If all four vacuum relief lines are assumed operable (i.e., no failures) this margin is increased to 80 percent. These margins are considered satisfactory for the following reasons:

- a. When a design negative pressure differential is selected and an

A/\sqrt{K} ratio is calculated, sizing of the vacuum breakers depends entirely upon the K value used. Precise, tested, k values are available from vacuum relief valve manufacturers. In addition, a

10 percent margin on friction losses is standard in sizing piping systems.

- b. The hypothetical situations used in the analyses are very conservative in that they assume during each of the occurrences that the suppression pool is at its minimum design temperature while the containment atmosphere is at its peak temperature.

6.2.1.1.4.2.2 Testing and Inspection

Periodically, each check valve is exercised to ensure that it is in proper working condition. For this purpose, each check valve has an air cylinder for power operated exercising. Testing and inspection of the containment isolation function of the check valves and the outer isolation valves are discussed in <Section 6.2.4.4> and <Section 6.2.1.6>.

6.2.1.1.4.2.3 Instrumentation Applications

The vacuum breaker check valves are normally closed. They begin to open under a pressure differential of 0.1 psid and close by gravity. The motor-operated outer isolation valves are controlled remote-manually from the control room or automatically, as discussed in <Section 6.2.4> and <Section 7.3.1>, and are open during normal plant operation. Position indicators on the vacuum breaker check valves annunciate in the control room if any of these check valves stick open during normal plant operation. If this occurs, the outer isolation valves are closed manually from the control room.

6.2.1.1.5 Steam Bypass of the Suppression Pool

6.2.1.1.5.1 Introduction

The concept of the pressure suppression reactor containment is that any steam released from the primary system is condensed by the suppression pool and does not have an opportunity to produce a significant pressurization effect on the containment. This is accomplished by channeling the steam into the suppression pool through a vent system. This arrangement forces steam released from the primary system to be condensed in the suppression pool. Should a leakage path exist between the drywell and containment, the leaking steam would result in pressurization of the containment. To mitigate the consequences of any steam bypassing the suppression pool, a high containment pressure signal automatically initiates the containment spray system any time after LOCA plus ten minutes. Realignment logic and interlocks affecting operation of containment spray are discussed in <Section 7.3>.

<Section 6.2.1.1.5.2>, <Section 6.2.1.1.5.3>, and <Section 6.2.1.1.5.4> present the results of calculations performed to determine the allowable leakage capacity between the drywell and containment.

6.2.1.1.5.2 Criteria

The allowable bypass leakage is defined as the amount of steam which could bypass the suppression pool without exceeding the design containment pressure. In calculating this value, a stratified atmosphere model is used to ensure conservatism.

6.2.1.1.5.3 Analysis

The allowable drywell leakage capacity has been evaluated for the complete spectrum of credible primary system rupture areas. This leakage capacity is expressed in terms of the parameter:

$$A / \sqrt{K}$$

Where:

A = Flow area of leakage path, ft².

K = Geometric and friction loss coefficient.

The parameter A / \sqrt{K} is dependent only upon the geometry of drywell leakage paths and is a convenient numerical definition of the overall drywell leakage capacity. It results from a consideration of the flow process in the leakage paths. Assuming steady-state, incompressible fluid flow theory to be applicable to the leakage flow, the pressure loss between the drywell and containment can be written as follows:

$$P_d - P_c = K \left[\frac{V^2}{2g_c} \right] \left[\frac{1}{144v} \right]$$

Where:

P_d = Drywell pressure, psia.

P_c = Containment pressure, psia.

K = Total loss coefficient of the flow path between the drywell and containment. 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 considerations, can be considered constant.

Where: (Continued)

V = Velocity of flow, ft/sec.

g_c = Gravitational constant, lb_m-ft/lb_f-sec².

v = Specific volume of fluid flowing in the leakage path, ft³/lb_m.

Given the equation for flow rate of $\dot{m}=AV/v$. If the leakage path flow rate is \dot{m} lb_m/sec and the flow area is A ft², the above equations can be rewritten to give:

$$\dot{m} = \frac{A}{\sqrt{K}} \sqrt{2g_c (P_d - P_c) \left(\frac{144}{v} \right)}$$

Thus for a given drywell to containment pressure differential, the leakage flow (capacity) is dependent only upon A/\sqrt{K} .

6.2.1.1.5.4 Bypass Capability with Containment Spray and Heat Sinks

An analysis has been performed which evaluates the bypass capability of the containment for small primary system breaks, considering containment spray and containment heat sinks as means for mitigating the effects of bypass leakage.

The flow rate of one containment spray loop is 5,250 gpm. This flow is assumed to start not sooner than ten minutes after the accident. The suppression pool water passes through the RHR heat exchanger and is injected into the upper containment region. The spray rapidly condenses the stratified steam and, therefore, creates a homogeneous air-steam mixture in containment. The available containment heat sinks, listed in <Table 6.2-10>, were considered with variable convective heat transfer coefficients, based upon the local instantaneous air-steam ratio. The cooldown rate was assumed to be 100°F/hr and the maximum design service

water temperature <Table 6.2-3> was used. The cooldown rate corresponds to the maximum rate which does not thermally cycle the reactor pressure vessel.

This analysis results in an allowable drywell leakage capability of A/\sqrt{K} of 1.68 ft². This allowable drywell leakage capability (1.68 ft²) is calculated to ensure that the containment design pressure of 15 psig (29.7 psia) will not be exceeded (Reference 37). A similar pressure transient for an A/\sqrt{K} of 1.0 ft² for a Mark III design (Reference 38) is provided in <Figure 6.2-25>.

Assumptions for allowable bypass calculations using heat sinks are as follows:

- a. Following occurrence of a pipe line break within the drywell, the air is purged through the vents into containment.
- b. Prior to containment spray operation, the bypassed steam is assumed to stratify in the upper containment.
- c. Air in containment is compressed by incoming steam.
- d. Containment spray is activated 180 seconds after containment pressure reaches 9 psig or at LOCA plus 13 minutes, whichever occurs later.
- e. Efficiency of containment spray is based upon the local steam to air ratio.
- f. Following spray activation, the air and steam in containment become mixed.

- g. Heat is transferred to exposed concrete and steel containment. The Uchida convective heat transfer coefficients used are based upon the local steam to air ratio.
- h. No energy is assumed to leave the containment, except through the RHR heat exchanger.

The following analysis provides an illustration of the methods used to calculate steam condensing capability under typical post-LOCA conditions. The condensation capability is calculated using the following equation:

$$\dot{M}_c = \dot{M}_s \left[\frac{N_s (T_c - T_s)}{H_{fg}} \right] C_p$$

Where:

- \dot{M}_c = Steam condensation rate, lb_m/sec.
- \dot{M}_s = Spray flow rate, lb_m/sec (degraded flow of one RHR pump).
- N_s = Spray efficiency.
- T_c = Containment temperature, °F.
- T_s = Spray temperature at the nozzles, °F.
- H_{fg} = Latent heat of vaporization, Btu/lb_m.
- C_p = Constant pressure specific heat of water, Btu/°F-lb_m.

The spray water temperature is calculated from:

$$T_s = T_p \frac{KHX}{\dot{M}} \left[\frac{T_p - T_{sw}}{C_p} \right]$$

Where:

- T_p = Suppression pool temperature, °F.
- KHX = Heat exchanger effectiveness, Btu/sec - °F (degraded).
- T_{sw} = Service water temperature °F.

Containment spray has a significant effect on the allowable bypass capacity. Use of spray increases the maximum allowable bypass rate by an order of magnitude and represents an effective backup means of condensing bypass steam.

A study was performed by GE NEDO-10977 (Reference 4), of potential cracking of the reinforced concrete drywell due to shrinkage, thermal gradients, seismic events, small breaks, LOCAs, and combinations of these has been performed and indicates no significant cracking of the drywell walls. The drywell liner is not required to maintain a leak tight barrier and the GE generic concrete crack analyses demonstrates that the maximum allowable bypass leakage will be maintained. The study also concluded that there was no through cracking due to a large break DBA in a main steam or recirculation line plus SSE.

The conservatisms in the crack study as supplemented by CEI/GAI calculations, reinforcement ratios and penetration reinforcement practices provide a conservative bound on estimated crack widths and drywell bypass leakage flow paths. Under an SBA, no bypass leakage flow paths are predicted through the drywell concrete. Under a DBA, even with the extremely conservative assumption of concurrent peak drywell pressures, maximum SSE earthquake forces, concurrent SRV actuations, and precracks at all construction joints, the maximum predicted bypass leakage flow path is only 0.35 ft² as compared to an allowable 5 ft².

6.2.1.1.6 Suppression Pool Dynamic Loads

Following a design basis LOCA in the drywell, the drywell atmosphere is rapidly compressed due to blowdown mass and energy addition to the drywell volume. This compression is transmitted to the water in the weir annulus in the form of a compressive wave which propagates through the horizontal vent system into the suppression pool.

Upon pressurization of the drywell, water in the weir annulus is depressed and forced out through the horizontal vent system into the suppression pool. This movement of pool water can result in a vent clearing reaction force on the weir wall and a water jet impingement force on the containment wall.

Following vent clearing, the air-steam-water mixture flows from the drywell through the vents and is injected into the suppression pool. A vent flow reaction load is imparted to the weir wall and a vent flow differential pressure loads the drywell wall.

During vent flow the steam component of the flow mixture condenses in the suppression pool while the air, since it is noncondensable, is released to the suppression pool in the form of high pressure air bubbles. Initial air bubble loads are experienced by all suppression pool retaining and submerged structures.

The continued addition and expansion of air within the suppression pool causes pool volume to swell resulting in acceleration of the pool surface vertically upward. This response of the suppression pool is referred to as bulk pool swell since the air is confined beneath the pool and is driving a solid ligament of water. Bulk pool swell air bubble and flow drag loads are imparted to the drywell and containment walls and to structures, components, etc., which may be located at low elevations above the normal pool surface. Bulk pool swell impact loads also result for low elevation structures and components.

Due to the effects of buoyancy, air bubbles rise faster than the suppression pool water mass and eventually break through the swollen surface and relieve the driving force beneath the pool. This breakup of the water ligament leads to the upward expulsion of a two-phase mixture of air and water and is referred to as pool swell in the froth mode. Structures which are located at higher elevations above the initial pool surface, i.e., the hydraulic control unit (HCU) floors, experience a pool swell froth impingement load due to impact of two phase flow.

In the annular region between the drywell and containment walls where pool swell occurs, a flow restriction exists at the HCU floor level, approximately 27' above the normal pool surface. The volume between the normal suppression pool surface and the HCU level is referred to as the

wetwell. During pool swell in the froth mode, passage of the air-water mixture through this restriction generates a two-phase flow pressure drop and produces a wetwell pressurization load on the HCU floors. Froth flow continues until the fluid kinetic energy is expended, followed by fallback of the water to the initial suppression pool level. Water fallback loads are experienced by all previously mentioned structures and equipment in the wetwell.

Following the initial pool swell event, the suppression system settles into a generally coherent phase during which vent flow rates are maintained from the drywell to the suppression pool. A resultant effect is the occurrence of vent flow steam condensation loads on pool retaining structures. As the reactor coolant system inventory of mass and energy is depleted near the end of blowdown, venting rates to the suppression pool diminish, allowing recovering of each row of horizontal vents. During phases of low vent mass flux, the suppression system behaves in an oscillatory manner, referred to as chugging. This periodic clearing and subsequent recovering of vents occurs since the vent flow cannot sustain bulk steam condensation at the vent exit. The resultant local fluctuations in pressure and water level generate chugging oscillation loads, predominantly on the drywell and weir wall.

Pool dynamic loading associated with relief valve operations has been identified in addition to loading associated with the design basis LOCA. Pressure waves are generated within the suppression pool when, upon first opening, relief valves discharge air and steam into the pool water. This phenomenon yields steam vent clearing loads which are imparted to pool retaining and submerged structures. The design basis loads for the containment system due to pool swell and safety/relief valve actuation discussed in <Appendix 3A>.

6.2.1.1.7 Containment Environment Control

The functional capability of the normal containment ventilation systems to maintain the temperature, pressure and humidity in the containment and subcompartments within prescribed limits (maximum allowable containment conditions) are discussed in <Section 9.4.6>.

6.2.1.1.8 Postaccident Monitoring

The containment atmosphere monitoring system provides highly reliable instrumentation for detecting and possibly predicting abnormal occurrences in containment and for monitoring following postulated accidents. Temperature and pressure sensors are furnished throughout the containment and drywell and are equipped with adjustable alarm features. Instrumentation channels are of high quality and accuracy so that precise monitoring information is available to the operator. The suppression pool is similarly instrumented for purposes of temperature monitoring.

Monitoring system components are considered to be safety-related and are qualified in accordance with IEEE Standards 323 (Reference 7) and 344 (Reference 8). Redundant channels are provided and independence is maintained in accordance with the criteria of IEEE Standards 279 (Reference 9) and 384 (Reference 10). Redundant Train A and Train B components are used.

Precision trip/calibration units continuously monitor each channel to ensure accurate alarming and inplace calibration at any time.

Recording of all containment atmosphere monitoring safety-related channels is accomplished at the postaccident monitoring panel in the control room.

Further details concerning the containment atmosphere monitoring system are presented in <Section 7.6>.

6.2.1.2 Containment Subcompartments

6.2.1.2.1 Design Bases

An analysis of the containment subcompartments containing high energy piping, in which breaks are postulated to occur, has been performed. The breaks selected for a pressurization analysis are those which release the most energy into the subcompartments.

All of the breaks were assumed to be circumferential. Appropriate restraints were used where applicable to limit the mass release while still producing conservative results.

Final design margins above the peak calculated pressure are given in the appropriate tables for each subcompartment analyzed.

6.2.1.2.2 Design Features

6.2.1.2.2.1 Analyses Performed

a. Reactor Annulus Pressurization

The reactor annulus was analyzed for recirculation line breaks and for a feedwater line break. Obstructions considered included feedwater piping, recirculation piping, instrument lines, inservice inspection pins, and insulation.

b. Drywell Head Region Pressurization

A six inch reactor core isolation cooling (RCIC) spray line enters the head region of the drywell and is attached to the top of the reactor pressure vessel <Figure 6.2-26>. When this line breaks, the reactor pressure vessel blows down into the head region which

is vented to the drywell through six 20-inch holes in the bulkhead member. The six inch RCIC line is not assumed to discharge any liquid since the RCIC outboard isolation valve is closed.

c. Reactor Water Cleanup Rooms Pressurization

Four reactor water cleanup rooms, listed below, were analyzed:

1. Heat exchanger room.
2. Filter demineralizer valve room.
3. Filter demineralizer room.
4. Drain valve nest room.

Plan and elevation drawings of the above rooms are provided by <Figure 6.2-27>.

d. Steam Tunnel Pressurization

The steam tunnel was analyzed for the same break as were the reactor water cleanup rooms. This break was used since the main steam lines are enclosed in guard pipes inside containment. Plan and elevation drawings showing the steam tunnel and the break location are provided by <Figure 6.2-27>.

e. Bulkhead Plate ΔP Analysis

An analysis has been performed to determine the peak pressure differential across the drywell bulkhead plate resulting from the postulated rupture of a main steam line at the reactor vessel nozzle. <Figure 6.2-51a> presents a schematic of the model used for the pressurization analysis.

6.2.1.2.2.2 Subcompartment Volumes

Volumes of the subcompartments are discussed in <Section 6.2.1.2.3>.

6.2.1.2.2.3 Vent Areas

Vent areas are discussed in <Section 6.2.1.2.3>.

6.2.1.2.3 Design Evaluation

The computer code COMPARE (Reference 11) was used in all subcompartment analyses, except for the analysis of the reactor annulus. The computer code RELAP4/MOD3 (Reference 12) was used in the analysis of the reactor annulus.

Initial conditions, along with nodal volumes, calculated peak pressure differentials and design pressure differentials, are presented by <Table 6.2-12>, <Table 6.2-13>, <Table 6.2-14>, <Table 6.2-15>, and <Table 6.2-16>.

Descriptions of break locations are presented in <Section 3.6> and below:

a. Reactor Annulus

Breaks were considered as follows:

1. The recirculation suction line break is assumed to occur at the nozzle of the reactor pressure vessel. A flow diverter with a 1 inch gap is constructed around the recirculation line as shown by <Figure 6.2-28>. This limits the amount of fluid discharged from the nozzle into the reactor annulus.

2. The recirculation discharge line break is a guillotine break which occurs within the reactor annulus.
3. The feedwater line break is a guillotine break which occurs in the reactor annulus area.

b. Drywell Head Region

The break in the six inch RCIC spray line is at the top of the reactor pressure vessel where the line enters the vessel.

c. Reactor Water Cleanup Rooms

All four reactor water cleanup rooms were conservatively analyzed using the blowdown flow for the same break. The inlet to the first regenerative heat exchanger is the assumed rupture point.

d. Steam Tunnel

The break used in the steam tunnel analysis is the same as that used for the reactor water cleanup rooms.

e. Bulkhead Plate ΔP Analysis

Nodal boundaries are based on actual flow restrictions ensuring that no substantial pressure gradient exists within a node. Nodalization information is as follows:

a. Reactor Annulus

The nodalization schemes of the reactor annulus were developed to provide detailed information, especially in the region surrounding the break itself, concerning the spatial pressure variation within the reactor annulus. These nodalization schemes for the

recirculation and feedwater line breaks are similar in detail to those used in reactor annulus analyses for other plants. The nodal schematic for the recirculation suction line break, which takes credit for symmetry is provided in <Figure 6.2-29>, where Node 25 is the break node and Node 24 is the drywell. The spatial pressure variation due to the mass and energy release from a recirculation discharge line using the recirculation suction model was also analyzed. The nodal schematic for the feedwater line break is provided in <Figure 6.2-30>, where Node 29 is the break node and Node 50 is the drywell.

b. Drywell Head Region

The drywell head region model is a two node model. Node 1 is the head region and Node 2 is the drywell.

c. Reactor Water Cleanup Rooms

Four reactor water cleanup rooms are modeled as follows:

1. The heat exchanger room is modeled using three nodes: the heat exchanger room, corridor and containment.
2. The filter demineralizer valve room is modeled using three nodes: the filter demineralizer valve room, corridor and containment.
3. The filter demineralizer room model is a two node model: the filter demineralizer room and containment.
4. The drain valve nest room is modeled using three nodes: the drain valve nest room, corridor and containment.

d. Steam Tunnel

The steam tunnel model is a two node model: Node 1 is the steam tunnel, Node 2 is the containment.

e. Bulkhead Plate ΔP Analysis

The COMPARE model used for the Bulkhead Plate ΔP Analysis is given in <Figure 6.2-51a>. Control Volumes 1 and 2 are the break nodes and Control Volume 5 models the drywell head region. The nodal descriptions are given in <Table 6.2-16a> and the flow path data is given in <Table 6.2-24a>.

Graphs of pressure differential with respect to time are provided for the subcompartments by the figures indicated:

a. Reactor annulus:

1. Recirculation line breaks, <Figure 6.2-31>, <Figure 6.2-32>, <Figure 6.2-33>, <Figure 6.2-34>, <Figure 6.2-35>, <Figure 6.2-36>, and <Figure 6.2-37>.
2. Feedwater line break, <Figure 6.2-38>, <Figure 6.2-39>, <Figure 6.2-40>, <Figure 6.2-41>, <Figure 6.2-42>, <Figure 6.2-43>, <Figure 6.2-44>, and <Figure 6.2-45>.

b. Drywell head region, <Figure 6.2-46>.

c. Reactor water cleanup rooms, <Figure 6.2-47>, <Figure 6.2-48>, <Figure 6.2-49>, and <Figure 6.2-50>.

d. Steam tunnel, <Figure 6.2-51>.

e. Bulkhead Plate ΔP Analysis, <Figure 6.2-51b> and <Figure 6.2-51c>.

Mass and energy releases used in the analyses are presented in the tables indicated or as stated below. All mass and energy as presented in the following tables is assumed to enter the break node.

a. Reactor annulus:

1. Recirculation suction line break, <Table 6.2-17>.
2. Recirculation discharge line break, <Table 6.2-17>.
3. Feedwater line break, <Table 6.2-18>.

b. Drywell head region: flow rate, 390 lb_m/sec; enthalpy, 1,190 Btu/lb_m.

c. Reactor water cleanup rooms, <Table 6.2-19>.

d. Steam tunnel inside reactor building. The RWCU line is the source of pipe rupture (main steam lines are guard piped), <Table 6.2-19>.

e. Bulkhead plate ΔP analysis - rupture of a main steam line in the drywell, <Table 6.2-26>.

<Table 6.2-20>, <Table 6.2-21>, <Table 6.2-22>, <Table 6.2-23>, <Table 6.2-24>, and <Table 6.2-24a> provide flow path information.

a. Reactor Annulus, <Table 6.2-20> and <Table 6.2-21>.

The compressible flow model as provided in (Reference 12) was used.

Special consideration has been given to the effect of the insulation attached to the biological shield wall following an accident within the reactor annulus.

The following assumptions have been used in determining nodal volumes and vent flow areas:

1. The annulus area below the reactor vessel skirt is assumed to remain isolated during the break transient.
2. Air seals which rupture at 15.0 psid separate the reactor annulus and the drywell.
3. Nodal volumes and initial flow area were evaluated, assuming that the insulation remained in place.
4. At the start of the accident all insulation within 120° of the break centerline is assumed to tear away from the biological shield wall and to catch on annulus piping and inservice inspection equipment. A study was made considering three different distribution schemes to represent how the insulation panels might catch on annulus obstructions. They are:
 - (a) Insulation panels are distributed evenly from 0° to 150° from the break centerline.
 - (b) Insulation panels are distributed evenly from 0° to 120° from the break centerline.
 - (c) Insulation panels are assumed to preferentially catch on obstructions between 30° and 120°.

Results of this study indicate that the even distribution of insulation panels over the 0°-120° range leads to a greater spatial pressure variation within the reactor annulus.

5. Insulation panels are assumed not to collapse for purposes of calculating vent area reduction, with the exception of those panels caught on obstructions which border the break node; these panels are assumed to collapse only in the direction of direct flow.
6. Flow blockage which occurs within the boundaries of a node is projected to the surrounding nodal junctions.

b. Drywell head region, <Table 6.2-22>

The flow models, as provided in the COMPARE (Reference 11) computer code, have been used.

Loss coefficients were calculated using the methods of (Reference 25) adjusted to the flow path minimum area.

c. Reactor water cleanup rooms, <Table 6.2-23>.

The flow models, as provided in the COMPARE computer code, have been used.

Loss coefficients were calculated using the methods of (Reference 25) adjusted to the flow path minimum area.

d. Steam tunnel, <Table 6.2-24>

The flow models, as provided in the COMPARE computer code, have been used.

Loss coefficients were calculated using the methods of (Reference 25) adjusted to the flow path minimum area.

e. Bulkhead Plate ΔP Analysis, <Table 6.2-24a>

The flow models, as provided in the COMPARE computer code have been used. Loss coefficients were calculated using the methods of (Reference 25) adjusted to flow path minimum area.

Reactor vessel insulation is assumed to be blown away and to block the opening to the drywell head region located in Control Volume No. 1.

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

This section presents information concerning the transient energy release rates from the reactor primary system to the containment system following a LOCA. Where the emergency core cooling systems enter into the determination of energy released to the containment, the single failure criteria have been applied to maximize the release.

6.2.1.3.1 Mass and Energy Release Data

<Table 6.2-25> provides the mass and enthalpy release data for the recirculation line break. Blowdown steam and liquid flow rates approach zero in approximately 376 seconds and do not change significantly during the remainder of the 24-hour period following the accident.

<Figure 6.2-52> shows the blowdown flow rates for the recirculation line break. These data were used in the containment pressure-temperature transient analyses discussed in <Section 6.2.1.1.3.3.1>.

<Table 6.2-26> provides the mass and enthalpy release data for the main steam line break. Blowdown steam and liquid flow rates approach zero in approximately 360 seconds and do not change significantly during the remainder of the 24-hour period following the accident. <Figure 6.2-53> shows the reactor pressure vessel blowdown flow rates for the main steam

line break as a function of time after the postulated rupture. This information was used in the containment response analyses discussed in <Section 6.2.1.1.3.3.2>.

6.2.1.3.2 Energy Sources

Reactor coolant system conditions prior to the line break are listed in <Table 6.2-4> and <Table 6.2-5>. Reactor blowdown calculations for containment response analyses are based upon these conditions during a LOCA.

The energy released to containment during a LOCA is comprised of the following:

- a. Stored energy in the reactor system.
- b. Energy generated by fission product decay.
- c. Energy from fuel relaxation.
- d. Sensible energy stored in reactor structures.
- e. Energy being added by the ECCS pumps.
- f. Metal-water reaction energy.

All of the above energies, except pump heat energy, are discussed or referenced in this section. The pump heat rate used in evaluating the containment response to the LOCA is selected as an input of 5,267 Btu/sec to the system with all ECCS equipment operating and 3,393 Btu/sec for minimum ECCS equipment operating.

Following each postulated accident event, the stored energy in the reactor system and the energy generated by fission product decay is

released. The rate of release of core decay heat for the evaluation of the containment response to a LOCA is presented by <Table 6.2-27> and <Table 6.2-27a> as a function of time after accident initiation.

The analysis at 3729 MWt <Table 6.2-27> used decay heat based on the ANS 5.0 (1973 standard) +20/10 (add 20% for the first 1000 seconds and 10% thereafter). The long-term analysis at 3833 MWt <Table 6.2-27a> used the more realistic ANS/ANSI 5.1 (1979 standard) with 2 sigma uncertainty adders.

Following a LOCA, the sensible energy stored in the reactor primary system metal is transferred to the recirculating ECCS water, and thus, contributes to suppression pool and containment heatup. <Figure 6.2-54> shows the variation of the sensible heat content of the reactor pressure vessel and internal structures during a main steam line break accident, based upon the temperature transient responses.

6.2.1.3.3 Reactor Blowdown Model Description

The reactor primary system blowdown flow rates were evaluated using the model described in (Reference 1).

6.2.1.3.4 Effects of Metal-Water Reaction

The containment systems are designed to accommodate the effects of metal-water reactions and other chemical reactions which may occur following a LOCA. The amount of metal water reaction which can be accommodated is consistent with the performance objectives of the ECCS. <Section 6.2.5.3> provides a discussion of the generation of hydrogen within containment by metal-water reaction. In evaluating the containment response, 16,221 Btu/sec of heat from metal-water reaction is included for the first 120 seconds. The containment response is

insensitive to the reaction time, even for the extremely conservative case where all the energy is included prior to the occurrence of peak drywell pressure.

6.2.1.3.5 Thermal Hydraulic Data for Reactor Analysis

Sufficient data to perform thermodynamic evaluations of the containment have been provided in <Section 6.2.1.1.3.3> and associated tables, in particular <Table 6.2-5>.

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

This section is not applicable to PNPP.

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

This section is not applicable to PNPP.

6.2.1.6 Testing and Inspection

Containment testing and inspection requirements are discussed in <Section 3.8.1>, <Section 3.8.2>, <Section 3.8.3>, and <Section 6.2.6>. No other special tests of either the drywell or containment structure are planned. Testing and inspection of other engineered safety features inside containment that interface with the containment structures are discussed in those sections which address the specific systems.

6.2.1.7 Instrumentation Requirements

The containment atmosphere monitoring system provides the operator with precise alarming, indicating and recording of the drywell and containment atmospheric conditions and suppression pool temperature

before, during and after a design basis accident. Additional details concerning the containment atmosphere monitoring system are provided in <Section 7.6>.

The drywell and containment vacuum relief system continuously monitors the drywell/containment and containment/atmosphere differential pressures and initiates automatic valve actuation as required. Redundant instrumentation is provided. Additional details concerning this system are presented in <Section 7.3>.

The combustible gas control system provides redundant hydrogen analyzers to monitor postaccident containment hydrogen concentration. Additional details concerning the hydrogen analyzers are provided in <Section 6.2.5> and <Section 7.3>.

The plant and area radiation monitoring system provides indication of containment ventilation exhaust and area radiation levels. Alarms are also provided. Details concerning radiation monitoring are provided in <Section 11.5> and <Section 12.3.4>.

6.2.2 CONTAINMENT HEAT REMOVAL SYSTEM

6.2.2.1 Design Bases

The containment heat removal system, consisting of the suppression pool cooling and containment spray systems, is an integral part of the RHR system. The purpose of this system is to prevent excessive containment temperatures and pressures, thus maintaining containment integrity following a LOCA, and to provide a mechanism to remove post-LOCA airborne fission product activity. To fulfill this purpose, the containment heat removal system meets the following safety design bases:

- a. The system limits the long term bulk temperature of the suppression pool to 185°F without spray operation when considering the energy

additions to the containment following a LOCA. These energy additions, as a function of time, are provided in <Section 6.2.1.3>.

- b. The single failure criteria applies to the system.
- c. The system is designed to safety grade requirements including the capability to perform its function following a safe shutdown earthquake.
- d. The system maintains operation during those environmental conditions imposed by the LOCA.
- e. Each active component of the system is testable during normal operation of the nuclear power plant.

6.2.2.2 System Design

The containment heat removal system is an integral part of the RHR system. Water is drawn from the suppression pool, pumped through one or both pairs of RHR heat exchangers and delivered to the suppression pool or to the containment spray header. The return flow penetrates containment through 18 inch lines at Elevation 603'-6" which is 10'-6" above normal water level. After penetrating containment, the return line descends vertically to Elevation 588'-6" where it turns and discharges horizontally into the pool. Physical model testing of the suction and discharge arrangement was performed to assure adequate mixing of the return water with the total suppression pool inventory. The large passive strainer installed in response to <NRC Bulletin 96-03> changed the suction configuration which was used in the physical model testing, the discharge configuration was unchanged. Analysis shows that pool mixing is not adversely affected by the new suction configuration. Because the large passive strainer is located on the floor of the suppression pool, any possible short circuiting of the RHR discharge to

its suction is not likely. Additionally, the large passive strainer design has continuous suction capability for the full 360 degrees around the suppression pool. During routine plant operation without significant quantities of debris present in the suppression pool, the majority of flow through the strainer to the RHR pumps will be from the strainer segments nearest the pump suction connection. However, debris buildup on the strainer post-LOCA will result in a more uniform flow distribution through the strainer along the entire length of the strainer (this has been confirmed by test). Spreading the source of suction over the entire strainer surface enhances mixing. Plan and section views are shown on <Figure 6.2-55> and <Figure 6.2-56>.

Water from the emergency service water system is pumped through the heat exchanger tube side to exchange heat with the processed water. Two cooling loops are provided, each being mechanically and electrically separate from the other to achieve redundancy. A process and instrumentation diagram is provided in <Section 5.4>. The process diagram, including the process data, is provided in <Section 5.4> for all design operating modes and conditions.

All portions of the containment heat removal system are designed to withstand operating loads and loads resulting from natural phenomena. All operating components can be tested during normal plant operation so that reliability can be assured. Construction codes and standards are discussed in <Section 5.4.7>.

The containment spray subsystem is started manually or automatically. The LPCI mode is automatically initiated from ECCS signals and the RHR system realigned for containment cooling by the plant operator after the reactor vessel water level has been recovered <Section 6.2.1>.

Suppression pool cooling is initiated in loop A or B by manually starting the emergency service water pump, closing the heat exchanger

bypass valve, opening the service water valve at the heat exchangers, closing the LPCI injection valve, and opening the pool return valve.

If a single failure occurred and the action which the plant operator is taking does not result in system initiation, then the operator will place the other totally redundant system into operation by following the same initiation procedure. If the operator chooses to utilize the containment spray, he must close the LPCI injection valves and open the spray valves. The containment spray mode is also initiated automatically on high containment pressure. Actual spray is delayed for 10 minutes and is initiated then only if LOCA and high drywell pressure signals are present. Automatic initiation is provided to protect the containment in the event of suppression pool bypass leakage as is described in <Section 6.2.1.1.5.4>.

If containment pressure is below the "high containment pressure" setpoint, actuation of the containment spray system can be performed manually from the control room.

Preoperational tests are performed to verify individual component operation, individual logic element operation and system operation up to the containment spray header. A sample of the nozzles are bench tested for flow rate versus pressure drop to evaluate the original hydraulic calculations. Finally, the headers are tested by air and visually inspected to verify that all nozzles are clear. Refer to <Section 5.4.7.4> for further discussion of preoperational testing.

Each ECCS pump takes suction directly from the suppression pool which does not have a sump. To prevent foreign objects in the suppression pool from entering the ECCS flow path, a strainer is located on the ECCS suction lines in the suppression pool.

The suction piping strainer from the suppression pool is specified with 3/32 (0.094") inch mesh openings capable of screening all foreign

particles which are of sufficient size to clog the RHR pump seal cyclone separators which have 1/8 inch orifices. The HPCS portion of the strainer uses the same mesh opening criteria as does the remainder of the strainer, however, an opening of 0.125 inch is allowed, which is sufficient to protect the HPCS pump's cyclone separator which has a 0.136 inch orifice. The large passive strainer is designed with sufficient strainer surface area to provide very low fluid approach velocities under all postulated debris loading conditions so that in the event the strainer becomes fully loaded (i.e., maximum postulated loading resulting from LOCA-generated and pre-LOCA debris materials), the minimum required NPSH is still provided to the RHR pumps during LPCI and suppression pool cooling modes. At a design flow rate of 7,800 gpm, the large strainer has a maximum pressure drop of 6.25 feet (HPCS) with the strainer fully loaded.

The strainer completely circumscribes the suppression pool and is fabricated from stainless steel plate and perforated (3/32" holes) stainless steel plate. The strainer is semi-circular in cross-section with two separate flow channels separated by an open central channel. The three ECCS divisions which connect to the strainer are physically separated through the use of internal divider plates in the flow channels; two divider plates are installed at each divisional interface location. The strainer rests on the floor of the suppression pool and the top of the strainer is approximately 3'-2" above the suppression pool floor. <Figure 6.2-83> shows the general layout of the strainer.

The mechanism for transport of insulation from the drywell into the containment suppression pool following an accident involves a series of occurrences, as discussed below.

The following types of insulation are used for piping and equipment within the containment:

- a. Metal-reflective insulation for the reactor pressure vessel.

- b. Metal-jacketed fiberglass blanket (Nu"K"on) for all hot piping and equipment. For those portions of hot piping and equipment where system configuration does not permit metal jacketing, only fiberglass blankets are installed.
- c. Polyethylene closed cell foam insulation on chilled water piping.

Metal-reflective insulation is installed in sections with overlapping edges and quick release latches with keepers.

Metal-jacketed fiberglass blanket insulation is installed in two-foot sections with Velcro fasteners on the longitudinal seam for ease of installation and removal. The blanket jacketing consists of a separate stainless steel sheath combining quick release latches and closure handles.

<Regulatory Guide 1.82> (Position 2.3.1.1) states that, consistent with the requirements of <10 CFR 50.46>, debris generation should be calculated for a number of postulated LOCAs of different sizes, locations, and other properties sufficient to provide assurance that the most severe postulated LOCAs are calculated. <Regulatory Guide 1.82> (Position 2.3.1.2) states an acceptable method for determining the shape of the zone of influence (ZOI) of a break is described in <NUREG/CR-6224>. The volume contained within the zone of influence should be used to estimate the amount of debris generated by a postulated break.

PNPP utilized NEDO-32686, Utility Resolution Guidance for ECCS Suction Strainer Blockage, <Section 3.2.1.2.3> (method 2) (Reference 34), to determine the size of a spherical ZOI. PNPP selected worst case break configuration based on the latest available information contained in (Reference 34). Specific pipe break locations were not used to identify the most severe zone of influence by a LOCA. Instead, a graphical model of the drywell was used to locate the center of the most severe

(largest) spherical zone of influence on the centerline of large bore drywell piping. A number of locations were compared for the greatest volume of insulation affected. The spherical zone size was determined independent of piping location, based on maximum pipe size (main steam piping diameter), and most severe offset and separation. The zone takes no credit for any fraction of debris that is considered non-transportable (i.e., all debris is 100% transportable).

(Reference 34) (Section 3.2.1.2.3, method 1) was used to calculate the volume of fibrous insulation available for transport. The ZOI calculation produced a quantity of fiber debris that was less than the installed fibrous insulation. The strainer design is based on the full drywell debris load and, therefore, is unaffected by changes in ZOI methodology, and conservative.

The quantity of "other" LOCA-generated debris (debris resulting from painted surfaces, fibrous, cloth, plastic, or particulate materials within the zone of influence that may produce debris) is based on the recommendations contained in the (Reference 34). The spherical ZOI was used to assess the mass of "other" drywell debris, and to demonstrate the safety margin available with the strainer design. The debris quantities used in the strainer design are considered very conservative.

PNPP has only considered fiber insulation quantities with respect to transportable insulation materials from the drywell. Stainless steel NUKON jacketing and reactor vessel metallic insulating materials and components at PNPP have been considered to be of sufficient density that they would sink to the bottom of the drywell or accumulate within the reactor vessel bioshield wall. Based on the very low approach velocity of the strainer, the design is not sensitive to any volume of reflective metal insulation generated as the result of a LOCA. Hence, the effects of pipe breaks inside the bioshield are bounded by the pipe breaks outside the bioshield which generate significantly greater amounts of fibrous insulation debris.

Due to low approach velocities associated with the large, passive strainer, metal tags and metallic insulation materials were assumed to settle in the Suppression Pool. PNPP takes no credit for settling of fiber insulation material, corrosion products/sludge, paint or coating debris, and plastic debris materials at the onset of a LOCA; however, significant debris settling was observed in a test program utilizing a 1/4-scale model of the strainer. The settling observed in the testing program is considered prototypical of the settling which would occur in the suppression pool after chugging and condensation oscillation have ceased.

With respect to debris transport, 100% of the drywell fiber insulation is considered to be transported to the suppression pool, without credit for any holdup time in the drywell. Similarly, the other amounts of debris are also assumed to be completely transported to the pool. It is assumed that the hydrodynamic actions during the first few minutes of a LOCA are sufficient to completely mix all debris which enters the pool from the drywell and to fully disperse all pre-existing debris resident in the suppression pool prior to LOCA occurrence. PNPP utilized the methods prescribed in the (Reference 34) and plant specific data to establish the limits for pre-LOCA debris in the suppression pool.

Prior to the initiation of suppression pool cooling, and once the suppression pool has settled from the initial hydrodynamic disturbances, the debris will either settle, be drawn to the strainer, or both. Since the 0.02 fps design approach velocity of the strainer is approximately the same as fiber settling velocities, fiber will be drawn to the strainer locally or else settle, with minimal tangential movement of the debris. This is potentially true to an even greater extent with the denser, particulate debris types such as ferrous materials, paint chips, etc. since their settling velocities are higher than fiber. However, these materials are also basically attracted to the strainer because of their dispersion within the fiber debris. Particulate materials which

are initially resident in the pool water nearest the strainer may also be expected to be preferentially drawn to the strainer mesh surface in lieu of settling, to be trapped by the fiber material which forms there.

The strainer design is such that debris will tend to collect first on the surface near the source of suction. As the debris bed thickness increases, the head loss will tend to increase through that portion of the strainer, and as a result the primary debris accumulation points will tend to migrate along the strainer, i.e., the strainer will be self-regulating with regards to debris accumulation and head loss. This process would continue until all debris has been captured by the strainer or has settled in the pool. If less than the maximum quantity of debris is generated, portions of the strainer may remain uncovered. Therefore, rate of accumulation of debris on the strainer is of no consequence.

The large toroidal passive strainer has been designed in accordance with <Regulatory Guide 1.82>, Revision 2. The suction strainer has been designed to preclude the potential for loss of NPSH caused by debris blockage during the period that the ECCS is required to maintain long-term cooling. The large toroidal passive strainer results in a very low approach velocity for water entering the strainer. Debris collected on the strainer surface is not expected to compact significantly (due to the very low approach velocity), resulting in minimal head loss. The testing of a 1/4-scale model of the strainer design confirmed the performance of the strainer and the behavior of the postulated debris bed as a function of time after the postulated LOCA. Because the debris bed will not be significantly compacted, flow will continue to pass through the debris (and the strainer) and thus the overall differential pressure will remain low. Maintaining a low differential pressure will ensure adequate NPSH for the ECCS pumps.

A complete DBA analysis of the interaction of the Nu"K"on insulation with the ECCS systems is presented in Owens-Corning Topical Report OCF-1.

Subsequent to the installation of the strainer and 1/4-scale testing, additional 1/4 scale testing was performed using a single strainer module, identified as the "small-scale test." The small-scale test facility was constructed mainly to respond to questions regarding post-LOCA coating failures. The small-scale tests included similarly conservative post-LOCA debris inventories as the initial 1/4 scale testing, plus significant fractions of drywell and containment primer and top coat, and additional debris. A simple flow model approach was used to correlate the small-scale test results to the initial 1/4-scale test configuration to conservatively bound the full-scale strainer head losses.

The chilled water anti-sweat insulation is only located above the floor grating on the wetwell Elevation 620'-6". If pieces were to reach the pool below, the pieces would float on the pool surface since the material has negligible water absorption capacity and weighs 2 lbs/ft³.

6.2.2.3 Design Evaluation

In the event of the postulated LOCA, the short term energy release from the reactor primary system will be dumped to the suppression pool. This will cause a pool temperature rise of approximately 45°F. Subsequent to the accident, fission product decay heat will result in a continuing energy input to the pool. The containment cooling system will remove this energy which is released into the primary containment system, thus resulting in acceptable suppression pool temperatures and containment pressures.

To evaluate the adequacy of the RHR system, the following sequence of events is assumed to occur:

- a. The reactor initially operating at 104.2 percent of original rated power (3729 MWt), a LOCA occurs.
- b. A loss of offsite power occurs and one emergency diesel fails to start and remains out-of-service during the entire transient. This is the worst single failure.
- c. Only three ECCS pumps are activated and operated as a result of Item b., above.

- d. After 30 minutes the plant operators activate one RHR heat exchanger in order to start containment heat removal. Once containment cooling has been established, no further operator actions are required.

NPSH requirements of <Regulatory Guide 1.1> are found in <Section 6.3.2.2>.

6.2.2.3.1 Summary of Containment Heat Removal Analysis

When calculating the long term, post-LOCA pool temperature transient, it is assumed that the initial suppression pool temperature and the emergency service water temperature are at their maximum values. This assumption maximizes the heat sink temperature to which the containment heat is rejected and thus maximizes the containment temperature. In addition, the RHR heat exchanger is assumed to be in a fouled condition at the time the accident occurs. This conservatively minimizes the heat exchanger heat removal capacity. The resultant suppression pool temperature transient is described in <Section 6.2.1.1.3.3.1> and is shown in <Figure 6.2-7> and <Figure 6.2-8>. Even with the degraded conditions outlined above, the maximum temperature is only 182.7°F, which is below the design limit specified in <Section 6.2.2.1>.

It should be noted that when evaluating this long term suppression pool transient, all heat sources in the containment are considered with no credit taken for any heat losses other than through the RHR heat exchanger. These heat sources are discussed in <Section 6.2.1.3>. <Figure 6.2-9> shows the actual heat removal rate of the RHR heat exchanger.

The conservative evaluation procedure described above clearly demonstrates that the RHR system in the suppression pool cooling mode limits the post-LOCA containment temperature transient.

6.2.2.4 Tests and Inspections

The preoperational test program of the containment cooling system is described in <Section 6.2.2.2>. Preoperational testing of the RHR system is discussed in <Section 5.4.7.4>.

6.2.2.5 Instrumentation Requirements

The details of the instrumentation are provided in <Section 7.3>. The suppression pool cooling mode of the RHR system is manually initiated from the control room.

6.2.3 SECONDARY CONTAINMENT FUNCTIONAL DESIGN

6.2.3.1 Design Bases

The secondary containment system includes the shield building and the annulus exhaust gas treatment system (AEGTS). Details of the AEGTS are given in <Section 6.5.3>. The following are the design bases for the shield building:

- a. The shield building is designed to collect the fission product leakage from the primary containment during and following a postulated design basis LOCA and delay it until it can be released to the environs after processing through the annulus exhaust gas treatment system such that the resultant offsite doses are less than the values set forth in <10 CFR 50.67>.
- b. The shield building is designed to withstand the peak transient pressures and temperatures which could occur due to the postulated design basis accident.

- c. The shield building is designed as a Seismic Category I structure.
- d. The shield building is maintained at a slight negative pressure relative to atmospheric pressure so any leakage post-LOCA through the shield building or the containment vessel is into this space.
- e. The design loads on the shield building are discussed in <Section 3.8.1>.
- f. The leak tightness of the shield building is continually verified by maintaining the annulus at a minimum vacuum of 0.66 inch water gauge. This constitutes a continuous testing program. Inspection of the secondary containment structure will not be necessary as long as a vacuum can be maintained through normal operation of plant equipment.

6.2.3.2 System Design

The shield building is a cylindrical reinforced concrete structure with a spherical dome enclosing the containment vessel. The internal diameter is 130 feet and the outside diameter is 136 feet. There is an annulus width of five feet between the containment vessel and the inside of the shield building. <Figure 1.2-3>, <Figure 1.2-4>, <Figure 1.2-5>, <Figure 1.2-6>, <Figure 1.2-7>, <Figure 1.2-8>, <Figure 1.2-9>, <Figure 1.2-10>, <Figure 1.2-11>, <Figure 1.2-12>, and <Figure 1.2-13> show plan and elevation views of the shield building.

There are two doors allowing access to the annulus area, both of which are normally locked. They are provided with position indicators and alarms which annunciate in the control room.

A tabulation of the design and performance data for the shield building is provided in <Table 6.2-28>.

The performance objective of the shield building is to collect and retain any fission product leakage from the containment vessel during and following a design basis accident and, in conjunction with the annulus exhaust gas treatment system, process and release the fission products to the environs in a controlled manner. This release is accomplished such that the resultant doses to the general public and the control room are within the values given in <10 CFR 50.67>.

The principal construction codes, standards and guides used in the design of the shield building are described in <Section 3.8.1>. In order to minimize the amount of radioactive material that leaks to the secondary containment following a design basis accident, primary containment penetrations are provided with redundant, ASME Code, Section III, Class 2, Seismic Category I isolation valves, one inside of the primary containment and one outside of the shield building, or some other acceptable configuration such as a closed system outside of containment. The piping out to the outboard containment isolation valve or in the closed system is also ASME Code, Section III, Class 2. This isolation valve arrangement functions to minimize "through-line" leakage, which is limited by leak rate testing as described below. The containment isolation system is discussed in more detail in <Section 6.2.4>. The containment and reactor vessel isolation control system is discussed in more detail in <Section 7.3.1>.

The containment boundary and all penetrations except for penetrations with guard pipes terminate in the annulus. Therefore, containment shell leakage and penetration leakage are considered to be totally directed to the annulus. The sources listed in <Table 6.2-33> are a summary of potential leakage paths that could bypass the AEGTS. The containment design basis accident leakage is 0.2 percent by weight of the contained atmosphere in 24 hours. The maximum test leakage rate permitted from the sources listed in <Table 6.2-33> is 10.08 percent of the total

containment leakage. This value will be the technical specification commitment for leakage bypassing the AEGTS as listed in the Technical Specifications. In order to verify that the total amount of potential bypass leakage will be within this limit, a testing and evaluation program will be conducted on isolation valves, personnel airlocks and guard pipes as described in <Section 6.2.4.3.1>.

The expected leakage rates per valve have been calculated and are shown on <Table 6.2-33> for the potential bypass leakage paths. In these calculations, it was assumed that the onsite leakage limit per valve will be the same as the shop test limits given in the valve specifications.

The air-filled lines penetrating primary containment that are not entirely contained in the secondary containment, and are potential bypass leakage paths, are identified in <Table 6.2-33>. The air supply lines to the ADS accumulators are not considered bypass leakage paths since they are safety-related lines that remain pressurized post-LOCA (at a pressure exceeding containment vessel design pressure).

6.2.3.3 Design Evaluation

All high energy lines which penetrate the containment wall and the shield wall are protected by guard pipes, with the exception of the control rod drive hydraulic supply <Table 6.2-33>. The CRD supply is a 2-1/2 inch water line with a normal operating pressure of 1,850 psi.

The justifying logic allowing postulation of pipe rupture locations in other high energy lines penetrating the containment which require guard pipe protection does not apply to this line. Guard pipes are used where the energy release rate of the postulated accident is such that the resulting pressurization must be confined to the drywell rather than

allowed to pressurize the containment volume directly. The energy release rate of an 1,850 psi fluid at a maximum of 140°F does not require that it be diverted from either the containment volume or from the annulus volume. Release of such a fluid within the containment volume will not prevent the AEGTS from performing its function, since pressurization of the volume does not result.

In the case of high energy lines whose postulated rupture would jeopardize containment design pressure capacity, the pipe break location criteria in MEB 3-1 were suspended and additional breaks postulated which could pose a hazard. Since the consequences of a 2-1/2 inch CRD line rupture are not significant with respect to hazard, the criteria of MEB 3-1 apply and no rupture is postulated to occur within the penetration. Due to line size being under 4 inches nominal, no longitudinal splits are considered to occur, only circumferential ruptures.

The postulation of a rupture in the 2-1/2 inch line at either end of the penetration will not jeopardize the integrity of the penetration. A break inside the containment structure will only load the penetration axially. Any break outside of the shield building will not load the penetration at all since no flow issues from the containment side of such a rupture, but from the pump side only.

An analysis of the pressure response profile in the shield building annulus following the design basis was performed using a modified version of the CONTEMPT-LT (Reference 5) computer code.

The following methods and assumptions were used in calculating the pressure response:

- a. Annulus design parameters given in <Table 6.2-28>.

- b. During a LOCA, the containment vessel will expand due to the pressure and temperature change within containment. It is included in the analysis by assuming that the volume expansion takes place as a function of Pressure and Temperature as shown in <Figure 6.2-59a> and <Figure 6.2-59b>.
- c. The annulus exhaust fan is assumed to purge at a rate of 2,000 cfm to the environment.
- d. The containment vessel and shield building were the only heat conducting structures considered in the analysis. Conservative values are summarized as follows:

<u>Description</u>	<u>Material</u>	<u>Surface Area</u> <u>(ft²)</u>	<u>Thickness</u> <u>(ft)</u>
Containment Vessel	Carbon Steel	74,912	0.125
Shield Building	Concrete	74,912 ⁽¹⁾	3.00

1. The surface area of the containment vessel and shield building are assumed to be the same in the computer analysis. The increased diameter of the annulus and concrete surface area of the containment fix at the annulus base are not included as this is conservative.
- e. The total heat transfer coefficient, h_t , between the containment atmosphere and the steel containment vessel is given by:

$$h_t = h_c + h_r$$

Where:

$$h_c = 0.3 (\Delta T)^{0.25}, \text{ as given by (Reference 13).}$$

$$h_c = \text{Convective film coefficient, Btu/hr-ft}^2\text{-}^\circ\text{F}_{\text{absolute}}$$

ΔT = Containment atmosphere temperature - initial containment vessel temperature, $^\circ\text{F}$. The containment temperature used is given by (Reference 26). The steel vessel temperature is assumed to be 90°F . The actual ΔT used is given in <Table 6.2-31>.

$$h_r = \epsilon_1 \epsilon_2 S \theta \frac{T_1^4 - T_2^4}{T_1 - T_2}$$

Where:

$$h_r = \text{Radiation coefficient, Btu/hr-ft}^2\text{-}^\circ\text{F}_{\text{absolute}}$$

$$\epsilon_1 = \text{Emissivity of steel} = 0.7$$

$$\epsilon_2 = \text{Emissivity of air} = 1.0$$

$$\theta = \text{Stefan-Boltzmann constant, } 0.1714 \times 10^{-8} \text{ Btu/hr-ft}^2\text{-}^\circ\text{F}_{\text{absolute}}$$

$$T_1 = \text{Higher temperature, } ^\circ\text{F}_{\text{absolute}}$$

$$T_2 = \text{Lower temperature, } ^\circ\text{F}_{\text{absolute}}$$

$$S = \text{Shape factor} = 1.0$$

The radiation coefficients are given in <Table 6.2-31> and assume the same relative temperatures used in calculating the convective film coefficient.

The total heat transfer coefficients as a function of time between the containment atmosphere and the steel containment vessel are listed in <Table 6.2-31>.

The total heat transfer coefficient for the steel vessel to the annulus atmosphere and for the annulus atmosphere to the shield building is taken to be $1.2 \text{ Btu/hr-ft}^2\text{-}^{\circ}\text{F}_{\text{absolute}}$.

- f. A schematic diagram of the model input into CONTEMPT is presented in <Figure 6.2-57>.

The results of the annulus pressure response are presented in <Figure 6.2-58>. As shown on this figure the annulus exhaust fans are adequate to maintain the annulus pressure below atmospheric pressure following a DBA-LOCA.

6.2.3.4 Tests and Inspections

The program for test and inspection of the containment isolation system is described in detail in <Section 6.2.4> and <Section 6.2.6>. The program for the annulus exhaust gas treatment system is described in <Section 6.5.3>.

6.2.3.5 Instrumentation Requirements

Design details and logic of the instrumentation for the AEGTS are given in <Section 7.3.1>.

6.2.4 CONTAINMENT ISOLATION SYSTEM

6.2.4.1 Design Bases

The design objective for the containment isolation systems is to allow normal or emergency passage of fluids through the containment boundary while preserving the ability of the boundary to prevent or limit the escape of fission products that may result from postulated accidents so that site boundary dose guidelines specified by <10 CFR 50.67> are not exceeded. This objective is achieved by provisions for automatic isolation of appropriate lines that penetrate the containment boundary.

The containment isolation systems are automatically actuated with input from the following signals:

- a. Low reactor pressure vessel water level (3 setpoints).
- b. High drywell pressure.
- c. High main steam line tunnel high ambient or differential temperature.
- d. Main steam line high area temperature, turbine building.
- e. (Deleted)
- f. High main steam line flow.
- g. Low main steam line pressure at turbine inlet.
- h. High radiation in containment and drywell purge exhaust.

- i. High reactor water cleanup system ambient or differential temperature.
- j. High temperature at outlet of reactor water cleanup system non-regenerative heat exchanger.
- k. High differential flow in reactor water cleanup system.
- l. Low line pressure in the steam supply line to the reactor core isolation cooling system.
- m. High steam flow in steam line to reactor core isolation cooling turbines.
- n. High residual heat removal system ambient or differential temperature.
- o. High reactor core isolation cooling system ambient.
- p. High pressure at the reactor core isolation cooling turbine exhaust diaphragm.
- q. High steam flow in the steam supply line to the reactor core isolation cooling system.
- r. High reactor vessel pressure (one signal for RHR valves).
- s. Low main condenser vacuum.
- t. Containment to atmosphere differential pressure greater than 0.0 psid.
- u. Standby liquid control system activated.

- v. Closure of reactor core isolation cooling turbine trip and steam shutoff valves.
- w. High drywell atmospheric radiation.
- x. (Deleted)
- y. (Deleted)
- z. (Deleted)
- aa. (Deleted)
- bb. Containment spray initiated.
- cc. High reactor pressure vessel water level.
- dd. Injection flow rate above setpoint (minimum flow valve control) for low and high pressure core sprays, residual heat removal and reactor core isolation cooling systems.
- ee. Discharge pressure below setpoint (minimum flow valve control) for high pressure core spray system.

The containment isolation systems can also be manually actuated from other locations as specified in <Table 6.2-32>. After initiation of containment isolation, either automatically or manually, the function goes to completion.

Upon receipt of signals indicating low water level in the reactor vessel or high drywell pressure, containment isolation valves in systems not required for emergency shutdown are closed except for the main steam

isolation, main steam line drain, and reactor water cleanup system isolation valves which do not close on high drywell pressure. These signals also activate systems associated with emergency core cooling.

Those fluid system penetrations that support systems not required for emergency operation are closed by the containment isolation systems. Fluid system penetrations that support engineered safety feature (ESF) systems have remotely operated isolation valves that may be closed from the control room if required.

The isolation criteria for determination of quantity, type and location of containment isolation valves for a particular system generally conform to the requirements of General Design Criteria (GDC) 54, 55, 56, and 57, and comply with the recommendations of <Regulatory Guide 1.11>.

Redundancy and physical separation are required in the electrical and mechanical design of the systems to ensure that no single failure in the containment isolation systems prevents performance of the intended functions. Protection of system components from missiles and from the effects of postulated high and moderate energy line breaks is also a design consideration.

Instrument lines that penetrate the containment boundary conform to the requirements of GDC 55 and 56, and comply with the recommendations of <Regulatory Guide 1.11>.

Containment isolation valves and associated piping and penetrations satisfy the requirements of the ASME Code, Section III, Class 1 or Class 2, as applicable. These components are also Seismic Category I. Classification of systems and equipment is presented in <Table 3.2-1>.

Upon loss of instrument air, air operated containment isolation valves fail in the position required for containment isolation. Closure times and leak tightness of containment isolation valves are sufficient to

ensure that the site boundary dose guidelines specified by <10 CFR 50.67> are not exceeded following a postulated accident. The "B" Train of the safety-related instrument air system provides postaccident makeup to the outboard MSIV accumulators to assure leak tightness of the outboard MSIVs. A capability for rapid closure of all lines provides a containment barrier within the lines that is sufficient to maintain leakage within permissible limits.

6.2.4.2 System Design

6.2.4.2.1 General

A summary of containment isolation valves is presented in <Table 6.2-32>. Each valve is described, including penetration number, applicable GDC or regulatory guide, fluid system, fluid, line size, valve arrangement, location, valve type, actuation mode, valve position, initiating signal, power source, etc. <Figure 6.2-60> illustrates the various containment isolation valve arrangements.

Justification for containment isolation provisions which differ from the requirements of GDCs is presented in <Section 6.2.4.2.2>.

Containment isolation valve closure times <Table 6.2-32> are established to prevent radiological effects from exceeding the guidelines specified by <10 CFR 50.67>. A discussion of valve closure times, for those valves through which a direct path from containment to the environment could exist, is provided in <Chapter 15>. Containment isolation for such lines is accomplished in accordance with NRC Branch Technical Position CSB 6-4, "Containment Purging During Normal Plant Operations." Additional discussion of Branch Technical Position CSB 6-4 is presented in <Section 6.2.4.2.3>.

Instrument lines which penetrate containment comply with the recommendations of <Regulatory Guide 1.11>. A power operated remotely controlled isolation valve is provided just outside containment. Instrument lines which penetrate both containment and drywell have a similar arrangement. A power operated isolation valve is provided just outside containment. Details of these arrangements are illustrated on <Figure 6.2-60>.

Inside containment each instrument line is either sized to adequately restrict flow or is provided with a 0.25 inch orifice as close to the beginning of the line as possible. In some cases a power operated valve is provided inside containment in lieu of a flow restriction.

Quality standards for containment isolation systems follow the recommendations of <Regulatory Guide 1.26>. Additional discussion of <Regulatory Guide 1.26> is presented within <Section 3.2.2>. Seismic classification is in accordance with the recommendations of <Regulatory Guide 1.29>. Additional discussion of <Regulatory Guide 1.29> is presented in <Section 3.2.1>. <Table 3.2-1> addresses classification of specific systems and equipment.

Containment isolation valves, actuators and controls are protected against damage from missiles, jet impingement and pipe whip. Potential sources of missiles and jet forces and locations of postulated pipe breaks are evaluated. Where potential hazards exist, protection is provided by physical separation, missile or jet shields, etc. These valves are located either inside containment, inside the intermediate building or inside the auxiliary building. Both of these structures are designed to satisfy Seismic Category I requirements and the turbine missile design criteria stated in <Section 3.5>.

To prevent debris from entering lines and obstructing valve closure, screens are provided on the open ends of lines through which a direct path from containment to the environment could exist.

Isolation valves are designed to be operable under the most adverse environmental conditions, such as maximum differential pressures, extreme seismic occurrences, steam laden atmosphere, high temperature, and high relative humidity. Normal and accident environmental conditions, for which the containment isolation valves and associated electrical equipment are designed, are discussed in <Section 3.11>. Also, where necessary, a dynamic system analysis to determine the effects of rapid valve closure under operating conditions, is required by the design specifications for piping systems associated with containment isolation valves. The valve operability assurance program for active, safety-related valves is discussed in <Section 3.9>.

Electrical redundancy is provided for power operated valves. Power for the operation of two isolation valves in a line (inside and outside containment) is supplied from two redundant, independent power sources without cross ties. In general, isolation valves outside containment are powered from the Division 1 power supply while isolation valves within containment are powered from the Division 2 power supply. Both Division 1 and Division 2 valves are generally powered from ac power sources. Loss of power to each motor-operated valve is annunciated.

Provisions for detecting leakage from remote-manually controlled systems are discussed in <Section 5.2.5>. Detection of leakage from containment is discussed in <Section 6.2.6>.

The fraction of the total containment leakage following a design basis accident/LOCA that could bypass the containment annulus exhaust gas treatment system is limited to the leakage from sources which constitute open systems or nonsafety-related systems. Safety class systems which are open systems are considered. For example, for determination of potential bypass leakage pathways it is assumed, for nonsafety-related systems (other than the feedwater lines, for approximately one hour following a LOCA), that the only portion of such a system remaining

intact would be the containment isolation valves and the piping between these valves. Additional leakage sources considered are the containment penetrations with guard pipes.

The containment boundary is surrounded by the annulus. Containment penetrations, except those with guard pipes, terminate in the annulus. Therefore, containment shell leakage and penetration leakage are totally directed into the annulus.

Containment design basis accident leakage is 0.2 percent, by weight, of the contained atmosphere in 24 hours. The maximum allowable combined test leakage rate from potential sources listed in <Table 6.2-33> is 10.08 percent of the total containment leakage. This value is the technical specification commitment for leakage bypassing the containment annulus exhaust gas treatment system.

To verify that the total amount of potential bypass leakage is within the established limit, the following test and evaluation program will be conducted in accordance with the Containment Leakage Rate Testing Program:

a. Isolation Valves

Since it is assumed for determination of potential bypass leakage pathways that nonsafety-related systems outside the containment isolation valves will not remain intact, (other than the feedwater lines, for approximately one hour following a LOCA), containment atmosphere must terminate at the outer containment isolation valve seat. The same effect is possible for open ended safety class systems. To assure that this potential source of bypass leakage is checked, isolation valves listed in <Table 6.2-40> are included in the periodic "Type C" test program discussed in <Section 6.2.6>.

The test method is also described within this section. By measuring the time related pressure decay or by directly measuring the leakage flow rate, each valve is quantitatively evaluated for leak tightness.

b. Guard Pipes

The basic configuration for guard pipes is to have one end open to the drywell and the other end welded closed to the process pipe outside the shield building. The primary potential leakage path is at the outer end closure welds. Secondary leakage paths may exist through discontinuities in the guard pipe material. Leakage testing is performed concurrently with the initial containment integrated leak rate test. Guard pipe leakage is detected by application of a soap bubble solution to 100 percent of the exposed guard pipe surface outside the shield building, including all weld connections. Any leak detected as a result of bubble formation will be eliminated (i.e., no detectable leakage) by performing the appropriate repair procedure. This test will be repeated each time a "Type A" test (containment integrated leak rate test) is performed <Section 6.2.6>.

c. Personnel Air Locks

Containment atmosphere could potentially bypass the annulus exhaust gas treatment system by leaking past the double seals on each door of the personnel airlocks. The personnel air lock leakage control system has been designed to eliminate this potential bypass leakage past the double seals on the outer doors <Section 6.2.6>. In addition, a test of the leak tightness of the volume between the double seals is performed frequently. By using the pressure decay method or leakage flow rate method, the actual leakage rate through the seals can be determined.

Results of "Type B" and "Type C" testing in accordance with the Containment Leakage Rate Testing Program are used to determine whether the 10.08 percent limit has been met.

Tests or analyses are also performed to demonstrate that the purge isolation valves function as specified. These are as follows:

- a. Necessary prototype tests and/or analyses to satisfy the recommendations of <Regulatory Guide 1.48> relative to demonstrating the operability of the equipment under the specified loading combination.
- b. Hydrostatic test of the valve body in accordance with the ASME Code, Section III.
- c. Leak test of the valve assembly at maximum operating pressure in accordance with applicable requirements of <10 CFR 50, Appendix J> (Option B).
- d. Valve operator performance test for closure speed under maximum operating pressure.

Systems penetrating containment can be isolated by remote-manual action as indicated in <Table 6.2-32>. Discussions of individual system isolations are presented in the sections which address the specific systems.

In the event of valve power failure, most motor-operated containment isolation valves remain "as is." Air operated valves close upon loss of air.

Main steam line isolation valves are spring loaded, pneumatic, piston operated globe valves, designed to fail closed upon loss of air pressure or loss of power to the solenoid operated pilot valves. Each main steam

isolation valve is served by two independent pilot valves, each of which is powered from an independent source. In addition, each main steam isolation valve is equipped with an air accumulator to assist in valve closure in the event of loss of air, loss of electrical power to the pilot valves and/or failure of the valve spring. The separate and independent action of either air pressure or spring force is capable of closing a main steam isolation valve.

Pneumatic and motor-operated containment isolation valves have status lights on the control switch in the control room to indicate open and closed position. A subset of these valves also have position indicator lights on the control room isolation status panel. Position of manual isolation valves is maintained by means of locking devices and/or administrative controls.

The containment isolation valves are designed in accordance with the requirements of Section III of the ASME Code.

6.2.4.2.2 Justification of Differences from General Design Criteria

The GDCs were not established specifically for BWR plants; rather, these criteria are intended to guide the design of all water cooled nuclear power plants. As a result, the GDCs are generic in nature and subject to a variety of interpretations. For this reason some cases exist with no "one-to-one" correspondence between the applicability of an individual GDC and plant design. In such cases, GE has developed a design that meets the intent of the criteria.

The isolation criteria within the GDCs contain clauses, such as, "unless it can be demonstrated ... on some other defined bases," which allows for an alternate design with reliability and performance capabilities that reflect the importance to safety of isolating the piping systems. Such alternates are described in <Section 6.2.4.2.2.1>, <Section 6.2.4.2.2.2>, and <Section 6.2.4.2.2.3>. The final measure by

which GE is assured that the BWR design is in agreement with the GDCs is receipt of the Advisory Committee on Reactor Safeguards (ACRS) letters permitting construction and operation of previous plants with comparable valving arrangements.

6.2.4.2.2.1 Justification with Respect to General Design
Criterion 55

The reactor coolant pressure boundary, as defined in <10 CFR 50.2(v)>, consists of the following: reactor pressure vessel; pressure retaining appurtenances attached to the vessel; valves, and pipes which extend from the reactor pressure vessel to, and including, the outermost isolation valve. The lines of the reactor coolant pressure boundary which penetrate containment are capable of isolating the containment, thereby precluding any significant release of radioactivity. Similarly, for lines which do not penetrate containment, but which do comprise a portion of the reactor coolant pressure boundary, the design ensures that isolation of the reactor coolant pressure boundary can be achieved. Items a, b and c, below, address influent lines, effluent lines and conclusions, respectively, with regard to GDC 55.

a. Influent Lines

Influent lines which penetrate containment and the drywell directly to the reactor coolant pressure boundary are equipped with at least two isolation valves. One valve is inside the drywell; the second is as close as possible to the external side of containment. These isolation valves protect the environment. Where needed, protection of the containment in the event of pipe rupture outside the drywell but within containment is further ensured by extension of the drywell by use of guard pipes. These guard pipes, together with the isolation valves, assure protection in the event of an active failure between drywell and containment. <Table 6.2-34> lists

those influent lines that comprise part of the reactor coolant pressure boundary and penetrate containment. The purpose of this table is to summarize the design of each line with respect to the requirements of GDC 55. Items 1 through 8, below, demonstrate that, although a word-for-word comparison with GDC 55 is not always practical, it is possible to demonstrate adequate isolation provisions on some other defined basis.

1. Feedwater Penetrations (P121 and P414)

Feedwater lines are part of the reactor coolant pressure boundary since they penetrate both the containment and drywell and connect to the reactor pressure vessel. Each line includes three isolation valves and is enclosed in a guard pipe.

The isolation valve inside the drywell is a control closure anti-water hammer check valve. The first isolation valve outside containment is also a control closure check valve and is located as close as possible to the outside containment wall. The outermost valve is a motor-operated gate valve. The two control closure check valves are designed and tested to close with no reverse flow.

Extension of the drywell by means of the guard pipe protects the containment from overpressurization in the event of a feedwater line break between the drywell and containment walls. The internal design temperature and pressure for the guard pipes which enclose the feedwater lines are the same as the design values specified for the enclosed feedwater lines. Should a break occur in a feedwater line, the control closure check valves prevent significant loss of reactor coolant inventory and provide immediate isolation. These check valves are tested in accordance with the

Inservice Testing Program, to verify this closure function. The outermost motor-operated valve does not close automatically upon occurrence of a protection system signal since, during a LOCA accident, maintenance of reactor coolant makeup from all sources, including nonsafety sources, is desirable. This valve, however, can be remotely closed from the control room to provide long term, high integrity leakage protection when, in the judgment of the operator, continued makeup from the feedwater system is no longer necessary. Power to these motor-operated valves can be provided from an alternate division under administrative controls, if necessary, following a LOCA. In addition, after feedwater flow terminates, the operator will initiate the feedwater leakage control system <Section 6.9> to provide a positive water seal on the seat, stem and bonnet of the motor-operated valve in each line. The check valves, coupled with the single motor-operated high integrity leakage protection gate valve on each line, provides an acceptable configuration for the feedwater lines.

A branch line connects to the feedwater line outboard of the second feedwater check valve, which is outboard of the containment. This branch line provides the pathway for RWCU water and RHR shutdown cooling water to return to the reactor vessel. For the RHR shutdown cooling return line, a safety-related globe valve is treated as a high integrity containment isolation valve, similar to the feedwater gate valves. The RHR system "outboard" of the globe valve is also treated as a closed system outside of containment, to control any leakage. For the RWCU return line, the piping "outboard" of the RWCU branch line check valve leads directly back to containment penetration P132, and is ASME Code Class 2, Seismic Category I, protected from pipe whip, missiles and jet forces, and analyzed for "break exclusion". This line is also

tested like a closed system outside of containment, i.e., per the Primary Coolant Sources Outside Containment Program (see <Table 6.2-40> for testing details on containment penetrations). The design of these branch lines also provides an acceptable configuration.

2. High Pressure Core Spray Line (P410)

The high pressure core spray line penetrates both the containment and the drywell and connects to the reactor pressure vessel. Isolation is provided by a hydraulically testable check valve inside the drywell and a motor-operated gate valve as close as possible to the outside of the containment wall. This gate valve maintains long term leakage control. Position indication for the hydraulically testable check valve is provided in the control room. The gate valve is automatically and remote-manually operated. A guard pipe is not necessary since the high pressure core spray fluid is at an energy level during system operation that containment overpressurization cannot result should the line break between the containment and the drywell.

3. Low Pressure Core Spray and Low Pressure Coolant Injection Lines (P112, P113, P412, and P411)

Isolation of the low pressure core spray and low pressure coolant injection system lines is accomplished by use of a hydraulically testable check valve and a motor-operated gate valve. The check valve is located as close as possible to the reactor vessel and is normally closed. This valve protects against containment overpressurization in the event of a line break between the check valve and the containment wall by preventing high energy reactor coolant from entering containment. The gate valves for the low pressure core spray and low pressure coolant injection line C are located outside

containment and are automatically and remote-manually operated and normally closed. The gate valves for the low pressure coolant injection lines A and B are inside containment and function similarly. All gate valves are automatically opened at the appropriate time to ensure that acceptable fuel design limits are not exceeded during a LOCA. A guard pipe is not necessary since the system fluid during operation is at such a low energy level to preclude the possibility of containment overpressurization should a break occur.

4. Control Rod Drive System Lines (P204)

The control rod drive system, located between the reactor pressure vessel and containment, includes two types of influent lines: the supply line that penetrates containment, and the insert and withdraw lines that penetrate the drywell.

Isolation of the supply line is accomplished by a check valve inside containment and a remote-manually actuated motor-operated block valve as close as possible to the outside of the containment wall.

The insert and withdraw lines are not part of the reactor coolant pressure boundary since these lines do not communicate directly to reactor coolant. The basis upon which these lines are designed is commensurate with the importance to safety of maintaining the pressure integrity of these lines. The classification of these lines is Quality Group B and they are designed in accordance with the ASME Code, Section III, Class 2.

In the design of the control rod drive system, it has been accepted practice to omit automatic valves for isolation purposes since inclusion of such a valve would introduce a

possible failure mechanism into the shutdown (scram) function. Manual shutoff valves are provided for isolation. In the event of a break in these lines, the manual valves provide isolation capability. In addition, a ball check valve in the control rod drive flange housing automatically seals the insert line in the event of a break. Containment overpressurization will not result from a line break in containment since these lines contain small volumes of fluids, resulting in relatively small blowdown masses.

As shown in <Figure 6.2-60 (1)>, Arrangement No. 4, the recirculation pump seal water supply line is connected to the control rod drive system downstream of the inboard containment isolation valve and is provided with check valve isolation both inside and outside the drywell.

5. Residual Heat Removal Head Spray and Reactor Core Isolation Cooling Lines (P123)

The residual heat removal head spray and reactor core isolation cooling lines join outside containment to form a common line which penetrates both the containment and the drywell and connects to the reactor pressure vessel. A hydraulically testable check valve is provided inside the drywell as close as possible to the reactor pressure vessel. Two valves, a check valve and a remote-manually actuated, motor-operated valve, are located outside containment in each line. The line is also enclosed in a guard pipe.

The hydraulically testable check valve inside the drywell is normally closed. The check valve(s) outside containment ensure immediate containment isolation in the event of a line break. The motor-operated valve in each line is remote-manually actuated to provide long term leakage control.

The guard pipe provides protection against containment overpressurization in the event of a line break between the drywell and containment walls. Should the check valve inside the drywell fail coincident with a line break, the guard pipe would direct the released fluid into the drywell.

Position indication lights are provided in the control room for the hydraulically testable check valves inside the drywell.

6. Standby Liquid Control Line (NA)

The standby liquid control system is located between the containment and the drywell. The standby liquid control line penetrates the drywell and connects to the reactor pressure vessel. Isolation is provided by a check valve inside the drywell and a check valve and explosive valve outside the drywell. The explosive valve provides an absolute seal for long term leakage control, as well as preventing leakage of boron solution into the reactor pressure vessel during normal reactor operation. Since the standby liquid control line is normally an isolated, nonflowing line, rupture is extremely improbable. However, should a break occur subsequent to actuation of the explosive valve, the check valves ensure isolation.

7. Residual Heat Removal Shutdown Cooling Return Lines (P121/P112 and P414/P410)

The residual heat removal shutdown cooling return lines discharge into the feedwater line between the control closure check valve and the motor-operated gate valve outside of containment. A check valve and a normally closed,

motor-operated, remote-manually actuated globe valve provide for isolation of the residual heat removal shutdown cooling return lines. See item 1 above for additional discussion.

8. Reactor Water Cleanup System Line (P419/P432)

The discharge line from the reactor water cleanup pumps penetrates containment and serves the reactor water cleanup regenerative heat exchangers inside containment.

Automatically actuated motor-operated gate valves, one inside, one outside containment, provide for isolation.

b. Effluent Lines

Effluent lines that form part of the reactor coolant pressure boundary and penetrate containment and/or the drywell are equipped with at least two isolation valves. One valve is inside the drywell, the other outside, but as close as possible to the containment. Where needed, the containment is protected, in the event of a pipe rupture outside of the drywell but inside containment, by guard pipes which enclose the process lines, forming an extension of the drywell. This combination of isolation valves and guard pipes assures protection in the event of a failure between drywell and containment walls.

<Table 6.2-35> lists those effluent lines that comprise part of the reactor coolant pressure boundary and that penetrate containment and/or the drywell. Items 1 through 4, below, address specifics of these lines.

1. Steam Lines (P124/P116, P416/P414, P122/P115, P415/P415, P422, and P423)

Steam lines include main steam, main steam drain, and reactor core isolation cooling steam lines.

The main steam lines from the reactor pressure vessel to the turbine penetrate both drywell and containment. Main steam line drains (one for each main steam line) in the drywell are headered together to form one line which penetrates both drywell and containment. Isolation for the main steam lines and main steam drain line is provided by automatically actuated block valves, one inside the drywell and one outside containment.

The steam supply line for the reactor core isolation cooling turbine branches from the main steam line inside the drywell. Isolation for this line is provided by one normally open gate valve and one normally closed globe valve inside the drywell, and one normally open gate valve outside containment. These motor-operated valves are capable of automatic and remote-manual actuation.

Use of guard pipes to enclose these steam lines prevents containment overpressurization in the event of line break between the drywell and containment walls. The internal design temperature and pressure for the guard pipes which enclose these steam lines are the same as the design values specified for the enclosed lines.

2. Reactor Water Cleanup Lines (P131/P132, P132, and P424)

The reactor water cleanup pumps are located outside containment; the heat exchangers and filter demineralizers, are located inside containment, but outside the drywell. The reactor water cleanup pump suction line from the reactor recirculation system lines and the reactor bottom head penetrates the drywell and containment. Two automatically actuated, motor-operated valves provide for isolation of this

line. One valve is just inside the drywell, the other is outside containment. A guard pipe encloses the line between the drywell and containment walls.

The reactor water cleanup pump discharge line to the heat exchangers and filter demineralizers penetrates containment. Two automatically actuated, motor-operated valves (one inside and one outside containment) provide for isolation of this line.

A blowdown line from the filter demineralizers penetrates containment and divides to form separate lines to the condenser and radwaste system. Automatically actuated, motor-operated valves, one inside and one outside containment, provide for isolation of this line.

The return line from the filter demineralizers penetrates containment and connects to the feedwater line between the outboard feedwater gate valve and the outboard (feedwater) check valve. Two automatically actuated, motor-operated gate valves provide for isolation of this line. One valve is inside, the other outside of containment.

3. Residual Heat Removal Shutdown Cooling Line (P421)

The residual heat removal shutdown cooling line branches from the B reactor recirculation loop and penetrates both the drywell and containment. A check valve, in parallel with a normally closed, remote-manually actuated, motor-operated valve isolates the line inside the drywell; a normally closed, remote-manually actuated, motor-operated valve isolates the line outside containment. A guard pipe encloses this line from the drywell wall to the containment wall to protect against containment overpressurization in the event of a line break.

4. Recirculation System Sample Line (NA)

A sample line from the recirculation system penetrates the drywell. This line is 3/4 inches in diameter and is designed in accordance with the requirements of the ASME Code, Section III, Class 2. A sample probe with a 1/8 inch diameter hole is located inside one recirculation discharge line within the drywell. In the event of a line break, this probe acts as a restricting orifice and limits escaping fluid flow. Two air operated valves which fail closed are provided for isolation of this line. Both sample isolation valves are located outside the drywell.

c. Conclusions Concerning General Design Criterion 55

To assure protection against the consequences of accidents involving the release of radioactive material, piping which forms portions of the reactor coolant pressure boundary has been shown to provide adequate isolation capability on a case-by-case basis. In all cases, a minimum of two barriers is shown to protect against release of radioactive materials. Where necessary to protect the containment against overpressure, guard pipes are provided which enclose the process pipes between the drywell and containment walls.

In addition to satisfying the requirements of GDC 55, the pressure retaining components which comprise the reactor coolant pressure boundary are designed to satisfy other appropriate requirements which minimize the probability or consequences of an accident rupture. Quality requirements for these components ensure that they are designed, fabricated and tested to the highest reactor plant component standards. The classification of components which comprise the reactor coolant pressure boundary is Quality Group A; these components are designed in accordance with the ASME Code,

Section III, Class 1. Additional information concerning classification is presented by <Table 3.2-1>. The containment and reactor vessel isolation control system is addressed in <Section 7.3>.

6.2.4.2.2.2 Justification with Respect to General Design
Criterion 56

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

<Table 6.2-36> lists those lines that penetrate primary containment and connect to the drywell and suppression chamber. The purpose of this table is to summarize the design of each listed line with respect to the requirements of GDC 56. Although a word-for-word comparison with GDC 56 is, in some cases, not practical, it is possible to demonstrate adequate isolation provisions on some other defined basis. It should be noted that this criterion does not reflect consideration of the BWR suppression pool design, since those lines which connect to the suppression pool would require placement of inside containment isolation valve under water. All of the lines which connect to the suppression pool are to or from the individual watertight ECCS pump rooms. Items a, b, c, d, e, and f, below, address influent lines to the suppression pool, effluent lines from the suppression pool, influent and effluent lines from the drywell and suppression pool free volume, airlock leakage control, inclined fuel transfer tube, and conclusions with respect to GDC 56, respectively.

a. Influent Lines to the Suppression Pool

1. Low Pressure Core Spray, High Pressure Core Spray, and Residual Heat Removal Test and Pump Minimum Flow Bypass Lines, Residual Heat Removal Heat Exchanger dump to Suppression Pool, and Suppression Pool Cleanup Return. (P105 and P409, and P407 and P408)

The low pressure core spray, high pressure core spray and residual heat removal test lines have isolation capability commensurate with the importance to safety of isolating these lines. Each line has a normally closed, motor-operated valve located outside containment. Containment isolation requirements are satisfied on the basis that the test lines are normally closed, low pressure lines, constructed to the same quality standards as the containment. Furthermore, the consequences of a break in one of these lines result in no significant effect on safety. All of these lines terminate below the minimum suppression pool drawdown level, except for lines through Penetrations P408 (RHR) and P409 (HPCS). These two lines terminate above the minimum suppression pool draw-down level. Penetration P409 has a vacuum breaker located above normal suppression pool level to mitigate water hammer effects.

The test return lines are also used for suppression pool return flow during other modes of operation. This reduces the number of penetrations, minimizing the potential pathways for radioactive material release. Typically, pump minimum flow bypass lines join the test return lines downstream of the test return isolation valve. The bypass lines are isolated by motor-operated valves and a restricting orifice is provided downstream of the valves.

The suppression pool cleanup return line via P105/P106 is used for suppression pool return flow during periods of suppression pool cleaning and mixing/cooling. This system is isolated with two motor-operated valves located outside containment. Containment isolation requirements are satisfied on the basis that the return line isolation valve design satisfied GDC 56 (Reference 36), and it is constructed to the same quality standards as the containment. These valves are normally closed when SPCU is not in operation. The consequence of a break in this line results in no significant effect on safety. Two motor-operated valves with separate divisional power are required to meet GDC 56 for influent lines connected directly to the suppression pool which is not a closed system outside containment.

2. Reactor Core Isolation Cooling Pump Minimum Flow Bypass,
(P104)

The reactor core isolation cooling pump minimum flow bypass line discharges into the suppression pool and terminates above the pool minimum drawdown level. A motor-operated, remote-manually actuated globe valve isolates the line outside containment. The valve is located as close to containment as possible.

3. Residual Heat Removal Heat Exchanger Vent and Relief Valve
Discharge Lines (P118, P107, P431, and P429)

Residual heat removal heat exchanger vent lines discharge to the suppression chamber through P118 and P431. Two normally closed, remote-manually actuated, motor-operated valves outside containment, and a check valve, located between containment and the drywell, provide isolation for P118.

Penetration P431 has one normally closed, remote-manually actuated, motor-operated valve and one normally open, automatic closing, remote-manually actuated, motor-operated valve outside containment, and a check valve, located between containment and the drywell.

Relief valve discharge lines from the residual heat removal heat exchangers and various emergency core cooling system suction and discharge lines discharge to the suppression pool. These vent lines are isolated by the relief valves. The addition of block valves would defeat the purpose of the relief valves. The relief valves set pressure is greater than 1.5 times containment design pressure.

A postaccident sampling system sample return line also ties into the relief valve discharge header and is isolated by two locked closed, remote-manually actuated solenoid valves outside containment.

- b. Effluent Lines from the Suppression Pool (P103 and P401 and P101; P102 and P402 and P403)

The low pressure core spray, high pressure core spray, reactor core isolation cooling, and residual heat removal suction lines are equipped with remote-manually actuated, motor-operated gate valves outside containment. These valves provide the ability to isolate in the event of a line break and also provide long term leakage control.

In addition, suction piping from the suppression pool is considered an extension of containment since this piping must be available for long term use following a design basis LOCA. Therefore, this piping is designed to the same quality standards as the containment. Thus, the need for isolation is obviated to some

degree by providing a high-quality system and by the fact that the piping runs to the water-tight ECCS pump rooms. Also, the emergency core cooling system discharge line fill system (emergency core cooling system waterleg pumps) takes suction from the respective emergency core cooling system pump effluent line from the suppression pool downstream of the isolation valve. The emergency core cooling system discharge line fill system suction line includes a manual valve provided for operational purposes. This system is isolated from containment by the respective emergency core cooling system pump suction valve from the suppression pool <Table 6.2-32>.

Also, each ECCS pump room is provided with leak detection capabilities as discussed in <Section 9.3.3>. 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 <Section 6.3.2.6>. No seals or gaskets are installed between the containment penetration and the isolation valve. The only potential path for leakage of suppression pool water into the ECCS pump rooms is through the pumps' suction lines, since these are the only lines penetrating the containment at an elevation below the suppression pool water level. See <Section 3.6.2.3.5.2> for discussion of unisolable/isolable leaks during non-accident (normal operation) and accident conditions.

Therefore, the need to size the ECCS pump rooms 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 level is not required due to the leak detection and isolation capabilities incorporated in the design. The potential reduction in suppression pool water inventory before detection and isolation

of a leaking seal or gasket in the pump room would be insignificant. Suppression pool makeup water during normal plant conditions is from the condensate water storage tank.

c. Influent and Effluent Lines from Drywell and Suppression Pool Free Volume

1. Combustible Gas Control and Post-LOCA Atmosphere Sampling Lines (P302 and P318, P423 and P425)

The combustible gas control system backup purge line which penetrates containment includes two normally closed, remote-manually actuated valves, one inside and one outside containment. The post-LOCA hydrogen sampling system lines which penetrate containment and connect to the drywell and suppression chamber air volume are equipped with a normally closed, solenoid operated isolation valve outside containment, and a 0.25 inch orifice inside containment. The design provides assurance of isolation of these lines.

The piping outside containment is a closed loop and is considered an extension of containment since it must be available for long term use following a design basis LOCA. Therefore, it is designed to the same quality standards as containment.

2. Containment Purge and Exhaust Lines (V313 and V314)

The containment purge and exhaust lines are each equipped with four automatically actuated isolation valves. One valve is outside containment, one valve is in a large normally closed branch line inside containment, and two in series valves are in a smaller intermittently used branch line.

3. Reactor Core Isolation Cooling Turbine Exhaust Line and
Turbine Exhaust Vacuum Relief Line (P106 and P115)

The RCIC Turbine Exhaust Line (P106) and the associated vacuum break line (P115) share common containment isolation valves:

The first containment isolation valves for the penetrations are: E51F068 (automatic) and E12F102 (locked closed). The second containment isolation valve downstream of E51F068 is nozzle check valve E51F040 (not a simple check).

Downstream of E12F102 is the RHR Relief Lines to the Suppression Pool (P107 and P429). The first isolation valves/barriers for the RHR Relief Lines (P107 and P429) are also the second isolation valves/barriers for the RCIC Turbine Exhaust Line and the Vacuum Break Line (P106 and P115). These valves/barriers are: E12F005, E12F025A, E12F025B, E12F025C, E12F055A, E12F055B, E21F018, N27F751, P87F264 and spectacle flanges 1E12D015A and 1E12D015B.

d. Containment Airlock Leakage Control System (P305/P312)

These lines serve a leakage control function by processing leakage past an inflatable seal. They are designed such that the open position of the valves provides a position of greater safety. Filtration of leakage is provided by means of an engineered safety feature system (AEGTS).

e. Inclined Fuel Transfer Tube (P205)

When the Inclined Fuel Transfer System (IFTS) blind flange is removed in Operating Mode 1, 2, or 3, the containment boundary inside the containment building is made by the remaining portion of the transfer tube containment isolation assembly, containment

bellows, and steel containment penetration, and outside the containment building by the transfer tube, drain line, drain valve, and local leak rate test valve (Reference 35).

f. Conclusions Concerning General Design Criterion 56

To assure protection against the consequences of accidents involving the release of significant amounts of radioactive material, piping that penetrates containment has been shown to provide adequate isolation capability on a case-by-case basis in accordance with GDC 56.

In addition to satisfying the isolation requirements specified by GDC 56, the pressure retaining components of these systems are designed, fabricated and tested in accordance with the requirements of the ASME Code, Section III. In some cases, provision of a high quality system obviates the need for isolation valves due to the diminished probability of a rupture in such a system. Additional information concerning classification is presented by <Table 3.2-1>. The containment and reactor vessel isolation control system is addressed in <Section 7.3>.

6.2.4.2.2.3 (Deleted)

6.2.4.2.3 Consideration of NRC Branch Technical Position CSB 6-4, "Containment Purging during Normal Plant Operations"

The containment vessel and drywell purge system is designed to achieve the objectives stated in Branch Technical Position CSB 6-4. Purge system containment isolation valves are capable of isolating containment within five seconds. The containment purge system is described in <Section 9.4.6>.

Radiological consequences for the initial cycle of a postulated LOCA during containment purge system operation have been evaluated in accordance with Branch Technical Position CSB 6-4. Since drywell purge system (approximately 25,000 cfm capacity) operation is restricted to reactor cold shutdown conditions and refueling operations, only the containment purge system (5,000 cfm capacity) was assumed to be operating at the start of the LOCA. The calculated site boundary doses are 0.9 rem to the thyroid and 162 mrem whole body. These doses are a small fraction of the <10 CFR 100> guideline values. This analysis is separate from the radiological dose analysis performed in <Chapter 15> for the limiting design basis LOCA and is not impacted by implementation of the alternative accident source term.

Major assumptions used in the dose analysis are as follows (initial cycle):

- a. A double ended guillotine break of the recirculation line was assumed to occur instantaneously. This accident was chosen because it represents the worst break and, consequently, the highest doses.
- b. Purge system isolation valve closure will isolate containment within five seconds (includes valve closure time of four seconds and an additional maximum time of one second for conservatism). During this period reactor coolant blowdown was conservatively estimated to be 109,766 pounds <Table 6.2-25>.
- c. Forty percent of the blowdown was assumed to flash to steam. It was conservatively assumed that the entire iodine activity in the flashed fraction of the total blowdown was instantaneously released to the containment atmosphere at the instant the accident occurred. Plate out of iodine was ignored. Retention of iodines in the suppression pool was also ignored although, actually, the flashed activity would first be dumped into the suppression pool and would then slowly evolve into containment.

- d. Specific activity in the reactor coolant was conservatively assumed to be 6.56 $\mu\text{Ci/g}$ of I-131 and 34.9 $\mu\text{Ci/g}$ of Xe-133, with other isotopes in proportionate quantities. This corresponds to spike conditions.
- e. Turbulence resulting from the high blowdown rates and operation of fan coolers in containment was assumed to ensure good mixing in the entire containment volume.
- f. Containment air was assumed to be released through two 18 inch purge lines, one supply and one exhaust, for five seconds. Constant flow rates through the open purge lines corresponding to the maximum containment pressure of approximately 3.0 psig during the release period <Figure 6.2-2> were used to determine a total flow to the environment of 1,020 pounds. This value is conservative since it ignores lower flow rates due to lower containment pressures and partial closure of the purge isolation valves at times prior to five seconds.
- g. No credit was allowed for iodine removal by the 99 percent efficient charcoal adsorbers in the containment purge exhaust lines.
- h. Site boundary χ/Q <Table 15.6-13> was used in the dose calculation. |

6.2.4.3 Design Evaluation

6.2.4.3.1 General Evaluation

To ensure the accomplishment of the design objective stated in <Section 6.2.4.1>, redundancy is provided in all design aspects of the containment isolation systems. Mechanical components are redundant and each isolation valve is protected, by separation and/or adequate barriers, against the consequences of potential missiles. Also, system

design specifications require each containment isolation valve to be operable under the most severe operating conditions to which it may be exposed. A program of testing is in place to ensure valve operability and leak tightness. Isolation valve arrangements provide backup in the event of accident and satisfy the requirements of GDC 54, 55, 56, and 57, and follow the recommendations of <Regulatory Guide 1.11>. Electrical redundancy is provided by valve arrangements which eliminate dependence upon one power source to achieve isolation. Electrical cables for isolation valves in the same line are routed separately. Cables are selected with consideration of the specific environmental conditions to which they may be subjected, such as magnetic fields, high radiation, high temperature, and high relative humidity. The containment isolation valve arrangements, with appropriate instrumentation, are illustrated by <Figure 6.2-60>. Modes of valve actuation are also redundant. The primary mode is automatic; the secondary mode is remote-manual. No active failure of a single valve or other component can prevent containment isolation.

All nonpowered isolation valves are administratively controlled and/or locked to ensure that position is known and maintained. The position of all power operated isolation valves is indicated in the control room. Instrumentation and controls associated with the containment isolation systems are discussed in <Chapter 7>.

6.2.4.3.2 Failure Mode and Effects Analyses

A single failure can be defined as a failure of some component in any safety system which results in a loss or degradation of the capability of the system to perform the safety function. Active components are defined as components that must perform a mechanical motion in the process of accomplishing a system safety function. <10 CFR 50, Appendix A> requires that electrical systems also be designed against

passive single failures, as well as active single failures. <Chapter 3> describes the implementation of these requirements, as well as GDCs 17, 21, 35, 41, 44, 54, 55, 56, and 57.

In single failure analysis of electrical systems, no distinction is made between mechanically active or passive components. All fluid system 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 satisfy the single failure criterion for both mechanical active and passive fluid system components that are required to perform a safety action.

6.2.4.4 Tests and Inspections

The containment isolation systems are scheduled to undergo periodic testing during reactor operation. Functional capabilities of power operated isolation valves are tested remote-manually from the control room. By observing position indicators and changes in the operation of the affected system, the operability of a particular isolation valve is verified.

Hydraulically testable check valves are used in certain systems, such as in the low pressure core spray, high pressure core spray and residual heat removal influent lines. These valves, located inside containment, are tested from a local panel to ensure functional capability when required to operate.

Leakage testing is addressed in <Section 6.2.4.2.1> and <Section 6.2.6>.

Instrument isolation valves inside and outside containment can be exercised from the control room and can be locally tested. An inservice inspection program for valves forming parts of the reactor coolant pressure boundary is described in <Section 5.2.4>.

6.2.5 COMBUSTIBLE GAS CONTROL IN CONTAINMENT

The control of combustible gas following a LOCA will be accomplished by mixing volumes of relatively high combustible gas concentration with those of low concentration. Prior to the time when the amount of combustible gas reaches critical mixture, electrical hydrogen recombiners are placed in operation. This controls any additional gas produced and subsequently reduces the hydrogen gas inventory. To aid in the long-term cleanup, the containment atmosphere can be purged through the annulus exhaust gas treatment system.

6.2.5.1 Design Bases

6.2.5.1.1 Safety Design Bases

The safety design bases are:

- a. To evaluate the hydrogen concentration as a function of time following the hypothetical LOCA, the hydrogen generation from the metal-water reaction and core and sump radiolysis is based on parameters found in <Regulatory Guide 1.7>.
- b. Hydrogen generated from the metal-water reaction, radiolysis and the corrosion of aluminum and zinc is assumed to evolve to the drywell atmosphere and form a homogeneous mixture. Several natural forces support this assumption. These natural forces include molecular diffusion and natural convection. Natural convection is promoted by temperature gradients existing in the drywell and the

cascading effect of the ECCS water exiting through the break. These forces offset the natural buoyancy force of hydrogen and promote mixing in the drywell. Mixing is promoted in the containment by these same natural forces. In addition, the initiation of the containment sprays will create turbulence in the containment and enhance mixing.

- c. The system design complies with all applicable requirements of <10 CFR 50, Appendix A> (Criterion 41) and <Regulatory Guide 1.7>.
- d. The system is capable of sampling and measuring the hydrogen concentration in both the drywell and containment vessel, and provide remote indication and alarms in the control room.
- e. The system is capable of mixing areas of high hydrogen concentration with areas of low hydrogen concentration to control combustible gas concentration in the drywell and containment vessel without reliance on purging.
- f. To control the long term buildup of hydrogen in the containment, recombiners are provided. There are two 100 percent capacity recombiners.
- g. Capability to purge the containment vessel and drywell atmospheres through a fission product removal system is also provided and available for long-term cleanup.
- h. The mixing subsystem and the electrical hydrogen recombiners meet the quality assurance, redundant instrumentation and power availability requirements assigned to an engineered safety feature system <Table 3.2-1>.

- i. All components in the mixing and hydrogen control systems are of Seismic Category I design and are capable of withstanding the temperature and pressure transients resulting from a LOCA. They can also withstand the humidity conditions and radiation environment in which the combustible gas control system components are located <Section 3.11>.
- j. Protection from postulated missiles and pipe whip is provided as required to ensure proper system operation. All active components of the drywell purge and hydrogen control systems are located in the containment outside the drywell. The major system components and associated performance data are listed in <Table 6.2-37>.
- k. Since operation of only one of the two independent combustible gas control systems is required, a single failure will not prevent the system from fulfilling its design function. The combustible gas control system failure analysis is presented in <Table 6.2-38>.
- l. The capability to periodically inspect and test systems and system components is discussed in <Section 6.2.5.4>.
- m. The hydrogen recombiners are freestanding units located in the containment. Therefore, the protection of personnel from radiation in the vicinity of the recombiners is not required.
- n. In accordance with <Regulatory Guide 1.7>, the concentration of hydrogen is the controlling factor for the combustible gas control system since the oxygen concentration is greater than five percent by volume.
- o. The combustible gas control system is placed in operation in accordance with Emergency Operating Procedures (EOPs).

6.2.5.2 System Design

A tabulation of the design and performance data for each system component is located in <Table 6.2-37>. A detailed discussion of instrumentation features is provided in <Section 7.3>.

6.2.5.2.1 Hydrogen Analysis Subsystem

The hydrogen analyzer is a thermal conductivity device that measures the percentage of hydrogen by volume by detecting changes in thermal conductivity. These changes are found by first measuring the thermal conductivity of a sample. The sample is then passed through a catalytic reactor which causes the hydrogen to react and be removed as water. The conductivity of the sample is measured again. The difference between the first and second conductivity measurement is the amount of hydrogen initially present in the sample. These analyses are designed to provide an accuracy of +10 percent to -1 percent of atomic hydrogen concentration. Tests have been conducted to qualify the analyzer in accordance with IEEE 323 (Reference 7), 334 (Reference 14) and 344 (Reference 8). Refer to <Section 3.10> and <Section 3.11>.

Since hydrogen is much lighter than air it diffuses rapidly and tends to form a uniform mixture. Because of this, it is unlikely that areas of higher concentrations could form. Any such concentrations would tend to form first at the high points, but only if the hydrogen release point were near the ceiling of an enclosed area and no circulation of atmosphere were to occur (Reference 15). These conditions will not be present after the LOCA since pressure and temperature transients due to the release of steam to the containment and drywell, as well as operation of the ECCS systems, will cause considerable turbulence throughout the area.

Since this turbulence will result in a relatively homogeneous mixture of hydrogen, the hydrogen sample locations are representative of the

containment and drywell atmosphere. Redundant sample lines from the space between the reactor vessel head and the drywell dome, the top of the drywell area and the containment dome are connected to redundant hydrogen analyzers.

Piping and instrumentation for the hydrogen analysis subsystem is presented in <Section 7.3>. Design of the subsystem is Safety Class 2, Seismic Category I.

The analyzer panels are located in the auxiliary building at Elevation 620'-0" and in the intermediate building at Elevation 654'-6" <Figure 1.2-5> and <Figure 1.2-7>. The hydrogen analysis subsystem operates independently of any other subsystem in the containment combustible gas control system.

Shielding and remotely operated valves are provided for the sample station to limit radiation exposure of plant personnel. The two redundant hydrogen analyzers and sample systems ensure that no single failure will prevent continuous monitoring of the hydrogen concentrations in the drywell and containment following a LOCA.

The analyzers and sample systems are manually initiated by the operator from the control room following a LOCA. The required presence of the operator in the control room and the relatively slow buildup of hydrogen concentrations in containment make delayed startup of these analyzers acceptable. Delaying initiation 15 minutes to 1 hour after a LOCA also subjects the analyzer to less severe sample conditions than the maximum LOCA conditions, for which they are designed, thereby increasing the probability of their successful operation. <Figure 6.2-61> indicates that hydrogen concentrations approach three percent in approximately 17 hours after a LOCA, providing more than sufficient time to initiate manual action. However, since the Three Mile Island event,

<NUREG-0737>, <Regulatory Guide 1.97>, and <10 CFR 50.44> have established additional requirements on the use and performance of the hydrogen analyzers <Appendix 1A, Item II.F.1.6>.

6.2.5.2.2 Hydrogen Mixing Subsystem

Initial control of the hydrogen concentration following a LOCA will be accomplished by mixing volumes of potentially high and low hydrogen concentrations. Mixing is accomplished by means of redundant, 500 scfm, centrifugal air compressors which take suction from the containment volume just below the service floor (Elevation 689'-6") and at the containment dome and discharge into the drywell. This pressurizes the drywell sufficiently to provide increase in drywell bypass leakage and uncover the upper row of suppression pool vents. The Mixing Compressor After Coolers are cooled using the RHR system.

The drywell allowable bypass leakage is much larger than the capacity of the compressors. There is a possibility that the drywell will not be pressurized sufficiently to uncover the upper row of suppression pool vents.

If the system can be pressurized to uncover the upper row of suppression pool vents, the best mixing is obtained by this method. This is due to the fact that the drywell volume is dispersed to the containment uniformly from around the entire circumference of the drywell at the lowest possible point, while the return to the drywell is from areas just below the service floor and the containment dome. If drywell bypass leakage accounts for the entire flow from the drywell, mixing should still be adequate due to the location of the two suction intakes just below the service floor and at the containment dome. This arrangement precludes any possibility of short circuiting the mixing subsystem.

Piping and instrumentation from the mixing subsystem are shown in <Figure 6.2-62>. Locations of the mixing compressors are shown in <Figure 1.2-8>.

Physical separation of components in the hydrogen mixing system assures proper operation despite pipe whip, missiles and jet impingement. There are two systems supplying air to the drywell. Supply piping and compressors are on adjacent quadrants outside the drywell. Hence, no single pipe break, missile or jet can disable both systems. This arrangement meets the criteria of <Regulatory Guide 1.46>. The piping is designed for the maximum differential pressure loads and is seismically supported. No ductwork is used in the system. This subsystem is designed as Safety Class 2, Seismic Category I. A prototype drywell purge compressor has been tested under LOCA conditions to ensure operation following a LOCA. The seismic qualification procedure for mechanical and electrical equipment is addressed in <Section 3.10>, and environmental qualification is described in <Section 3.11>.

6.2.5.2.3 Hydrogen Recombination System

The hydrogen control system is fully redundant and consists of two 100 percent capacity hydrogen recombiners. <Figure 6.2-61> depicts hydrogen concentration in the drywell and containment as a function of time.

- a. Each recombiner subsystem consists of a control panel and a power supply cabinet located in the control complex. The recombiner is located in containment, outside the drywell, at floor Elevation 664'-7". Air flows by natural convection through the unit. The recombiner is a completely passive device.

- b. The power supply cabinet contains an isolation transformer plus a controller to regulate the power to the recombiner. The controls for the power supply are located in the separate control panel and are manually actuated.
- c. Each hydrogen recombiner consists of the following design features:
 - 1. A preheater section consisting of a shroud placed around the central heaters to take advantage of heat conduction through the central walls for preheating incoming air.
 - 2. An orifice plate to regulate the rate of air flow through the unit.
 - 3. A heater section consisting of five banks of metal sheathed electric resistance heaters to heat the air flowing through it to hydrogen-oxygen recombination temperatures. Each bank contains 60 individual U-type heating elements.
 - 4. A mixing chamber which mixes and dilutes the hot effluent with containment air to lower the temperature of the discharge stream.
 - 5. An outer enclosure to protect the unit from impingement by containment spray.
 - 6. Except for electrical power, there is no need of any plant support service.
- d. Containment atmosphere is heated within the recombiner in a vertical duct causing it to rise by natural convection. As it rises, replacement air is drawn through intake louvers downward through a preheater section which will temper the air and lower its relative humidity. The preheated air then flows through an orifice

plate, sized to maintain a 100 scfm flow rate, to the heater section. The air flow is heated to a temperature above 1,150°F, the reaction temperature for the hydrogen-oxygen reaction, and any free hydrogen present reacts with atmospheric oxygen to form water vapor. After passing through the heater section, the flow enters a mixing section which is a louvered chamber where the hot gases are mixed and cooled with containment atmosphere before the gases are discharged directly into the containment. The air discharge louvers are located on three sides of the recombiner. To avoid short circuiting of previously processed air, no discharge louvers are located on the intake side of the recombiner.

Tests have verified that the hydrogen-oxygen recombination is not a catalytic surface effect associated with the heaters, but occurs due to the increased temperature of the process gases. As the phenomenon is not a catalytic effect, saturation of the unit cannot occur. Results of testing a prototype and production electric hydrogen recombiner are given in (Reference 16), (Reference 17), (Reference 18), (Reference 19), and (Reference 22). No differences exist between the recombiner system on which the qualification tests were conducted and the recombiner system which was supplied for Perry. For environmental qualification see <Section 3.11>. The system is designed to Class 1E, Seismic Category I requirements. The system is shown on <Figure 6.2-63>.

6.2.5.2.4 Purge Subsystem

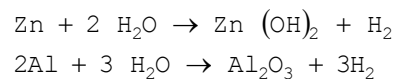
A purge subsystem is provided to aid in the long-term cleanup of combustible gas from containment and drywell provided that the site release rate is expected to remain below the plant limits. The purge subsystem is also used as a means to provide pressure relief from the drywell during startup and normal plant operation. Purging is accomplished by passing air from the drywell through the annulus exhaust gas treatment system (AEGTS) filters before being discharged to the outside atmosphere. The combustible gas purge subsystem consists of a

flow control valve, two isolation valves, and interconnecting pipe. The isolation valves, one inside and one outside containment, are motor-operated to isolate the containment vessel during a LOCA. The flow control valve is in the fail open position to maximize flow for drywell pressure control. Those portions of the purge subsystem associated with containment isolation are designed to meet Safety Class 2, Seismic Category I requirements. The system is shown on <Figure 6.2-62>.

6.2.5.3 Design Evaluation

The design evaluation of the combustible gas control system follows the guidelines of NRC Branch Technical Position CSB 6-2, "Control of Combustible Gas Concentrations in Containment Following a Loss-of-Coolant Accident," and is as follows:

- a. Generation of hydrogen from the corrosion of zinc and aluminum exposed to neutral water occurs in the following manner:



- b. The rate of hydrogen generated from these corrosion equations is defined as follows:

1. Zinc:

$$R = 356 e^{\frac{-7340}{T}}$$

[This equation is derived from experimental data as defined in (Reference 22).]

Where:

T = Temperature in K.

R = The hydrogen generation rate in lb-mole/ft²-hr

2. Aluminum:

$$R = 1.79 \times 10^{-7} \text{ (Reference 23)}$$

Where:

R = The hydrogen generation rate in lb-mole/ft²-hr

This rate is valid throughout the temperature range to which the aluminum is exposed during an accident.

- c. The corrodible surface areas inside the containment and drywell are maintained within the evaluated values listed below:

1. Drywell:

Area

Galvanized Steel	70,000 ft ²
------------------	------------------------

2. Containment:

Galvanized Steel	150,000 ft ²
------------------	-------------------------

Aluminum surface areas are not shown, since engineering analysis shows that the production of hydrogen from the corrosion of aluminum is negligible compared to other hydrogen generation sources. Additions of aluminum to the plant are evaluated for impact on hydrogen control analyses.

- d. The mass of the Zircaloy fuel cladding to a depth of 0.23 mils as specified in <Regulatory Guide 1.7> is 663 pounds.

- e. The integrated production of hydrogen for the containment and drywell due to radiolysis of water and corrosion of zinc is shown in <Figure 6.2-64>. The production of hydrogen from the corrosion of aluminum is negligible compared to the other sources.
- f. Negligible hydrogen and oxygen is contained in the reactor coolant system with respect to the other components of hydrogen generation following a LOCA.
- g. The total fission product decay power as a fraction of operating power versus time is given in <Table 6.2-27>.
- h. The fission product distribution model is discussed in <Section 12.2>.
- i. The integrated production of hydrogen in the drywell due to the zirconium-water reaction is 12.6 lb-moles for the first two minutes. There is no zirconium-water reaction after this time.
- j. The hydrogen concentration in containment plotted as a function of time is shown in <Figure 6.2-61> for a design basis LOCA.
- k. Two methods of hydrogen control are available. They are hydrogen mixing and hydrogen recombination. <Table 6.2-39> shows the design and operating parameters.
- l. Since hydrogen is much lighter than air, it diffuses rapidly, tending to form a uniform mixture. Because of this, it is unlikely that areas of higher concentrations could form; however, any such concentrations would tend to form at the high points of containment, where the containment spray lines and mixing system suction lines are located. The initiation of these systems would create turbulence and enhance mixing.

- m. The hydrogen mixing system piping is designed for the maximum differential pressure loads and is seismically supported. No ductwork is used in the system.

6.2.5.4 Testing and Inspection

The combustible gas control system is visually inspected in accordance with ASME Code Section XI. Compressors are tested by turning them on from the control room and measuring flow. Isolation valves are exercised periodically and position checked on indicators in the control room.

Testing of the hydrogen analyzers is done periodically by injecting a calibration gas into the sample lines and comparing the known concentration with the analyzer readout.

The hydrogen recombiners are tested in accordance with the Technical Specifications to check the calibration of the unit and proper operation of the heaters by energizing the unit and allowing temperatures to stabilize at the operating conditions.

Preoperational tests of the combustible gas control system are conducted during the final stages of plant construction prior to initial startup <Section 14.2>. These tests ensure correct functioning of all controls, instrumentation, compressors, recombiners, piping, and valves. System reference characteristics, such as pressure differentials and flow rates, are documented during the preoperational tests and are used as base points for measurements in subsequent operational tests.

In addition, inservice inspection of all ASME, Section III, Class 3 components is done in accordance with <Section 6.6>.

6.2.5.5 Instrumentation Requirements

Operation of the combustible gas control system is performed manually. On-off status of compressors and position of valves are indicated in the control room. Hydrogen concentration recorders and alarms, low flow, system bypassed alarms, and control switches are located in the control room.

The hydrogen analyzer recorder/switching station is located in the control room. The analyzer switching station allows the operator to manually select one of the four sample areas previously discussed. Both Hydrogen Analyzers annunciate high hydrogen concentration by volume at two levels. The setpoints have been selected to assist the Control Room operators in performance of hydrogen control functions. The high alarm is set to alert the operator that the hydrogen concentrations are increasing and that the minimum design concentration for operation of the hydrogen recombiners has been reached. The high-high alarm is set to alert the operator that hydrogen concentrations are continuing to increase and that several actions including operation of the Combustible Gas Mixing System is required prior to reaching four percent hydrogen concentration.

All lines in the system that connect the drywell with the containment vessel have isolation valves which close automatically on LOCA signal. Manual initiation and test operation are overridden by the LOCA signal. During normal plant operation, these isolation valves are closed. The hydrogen recombiners do not require any instrumentation inside the drywell or containment for proper operation after a LOCA. A thermocouple readout instrument is provided in the control complex for convenience in test and periodic checkout of the recombiner. A controller is operated from the control complex to regulate the power supply to the recombiner. Proper recombiners operation after an accident is determined by monitoring a watt-meter in the control complex.

Design details of the combustible gas control system instrumentation and controls are discussed in <Section 7.3>.

6.2.6 CONTAINMENT LEAKAGE TESTING

This section presents the testing program for determination of the primary reactor containment integrated leakage rate (Type A tests), primary containment penetration leakage rates (Type B tests) and primary containment isolation valve leakage rates (Type C tests) that complies with <10 CFR 50, Appendix J>. Testing requirements for piping penetration barriers and valves have been established using the intent of <10 CFR 50, Appendix J> (Option B). Exceptions taken to <10 CFR 50, Appendix J> for Type A, B or C tests are described and justified in <Table 6.2-40> and <Table 6.2-41>. This section also presents the testing program for determination of the drywell integrated leakage rate. The integrated leak rate test system is shown in <Figure 6.2-65>.

The primary containment leakage rates shall be demonstrated at the test schedule specified in accordance with the Containment Leakage Rate Testing Program.

Periodic Type A, B and C tests are performed to assure that leakage through the primary reactor containment and systems and components that penetrate the primary containment do not exceed allowable leakage rate values as specified in technical specifications. These periodic tests also ensure that proper maintenance and repairs are performed during the service life of the plant.

6.2.6.1 Primary Reactor Containment Integrated Leakage Rate Test

During the construction phase, localized leakage testing is employed <Section 3.8> to detect leaks which may affect containment integrity or the results of the initial integrated leak rate test.

Structural integrity test of the containment structure for strength requirements, described in <Section 3.8.1> and <Section 3.8.2>, must be satisfactorily completed prior to performance of the preoperational integrated leakage rate tests.

Upon completion of construction of the primary reactor containment including installation of all portions of mechanical, fluid, electrical, and instrumentation systems penetrating containment or associated with containment integrity, and upon satisfactory completion of the structural integrity test for strength described above, the preoperational containment integrated leakage rate test is performed to verify that the actual containment leakage rate does not exceed the design limits.

The preoperational integrated leakage rate tests are performed at the design basis accident pressure (P_a) to determine the measured leakage rate (L_{am}). (Operational integrated leakage rate tests are performed in accordance with the Technical Specifications).

The leakage rate, L_{am} , at test pressure, P_a , shall be less than 0.75 of L_a . Other pertinent test data, including test pressures, test duration and definition of terms are presented in <Table 6.2-41>.

Type A testing will be performed in accordance with the Containment Leakage Rate Testing Program <Section 6.2.6.6>.

The quantity and types of sensors associated with the primary containment integrated leakage rate instrumentation are listed in <Table 6.2-42>.

6.2.6.2 Containment Penetration Leakage Rate Test

Containment penetrations whose design incorporates resilient seals, gaskets or sealant compounds; air locks; equipment and access hatch

seals; and electrical penetrations receive preoperational and periodic Type B leakage rate tests in accordance with <10 CFR 50, Appendix J> (Option B). A list of all containment penetrations subject to Type B tests is provided in <Table 6.2-40>.

Containment personnel air lock door seals will receive periodic Type B tests.

All Type B tests are performed in accordance with the Containment Leakage Rate Testing Program <Section 6.2.6.6>.

a. Personnel Air Lock Leakage Control System

The personnel air lock leakage control system controls leakage by placing a vacuum on the space between the pair of door seals on the outer doors of the personnel air locks. The vacuum between the door seals is created by routing a small line into the containment annulus. The containment annulus is kept at a slight negative pressure by the annulus exhaust gas treatment system. Any bypass leakage past the personnel air lock outer door seals is routed back into the containment annulus where it is sent to the annulus exhaust gas treatment system. This system is classified as Safety Class 2 and is designed to survive and function through a single active failure.

6.2.6.3 Containment Isolation Valve Leakage Rate Tests

Those containment isolation valves which are Type C tested in accordance with <10 CFR 50, Appendix J> (Option B) are listed in <Table 6.2-40>.

See <Section 6.2.6.6> for the Containment Leakage Rate Testing Program.

6.2.6.4 Scheduling of Periodic Tests

The periodic leakage rate test schedules for Type A, B and C tests are given in the Containment Leakage Rate Testing Program.

Type B and C tests may be conducted at any time during normal plant operations or during shutdown periods as long as the time interval between tests for any individual Type B or C test does not exceed the maximum allowable interval specified in the Containment Leakage Rate Testing Program. Each time a Type B or C test is completed the overall total leakage rate for all required Type B and C tests is updated to reflect the most recent test results.

6.2.6.5 Special Testing Requirements

6.2.6.5.1 Drywell Leakage Rate Test

Following the drywell structural integrity test described in <Section 3.8.3>, the drywell leakage rate is measured at drywell design pressure. A drywell leakage rate test is also performed at reduced pressure. Subsequently, periodic drywell leakage rate tests are performed at an initial differential pressure of 2.5 psi. The reduced pressure tests verify that drywell to containment leakage does not exceed the allowable suppression pool bypass limits specified in technical specifications. The combination of the design pressure and reduced pressure leakage rate tests also verifies that the drywell will perform adequately for the full range of postulated primary system break sizes. The allowable drywell leakage rate, L_d , is 10 percent of the allowable bypass leakage for the small break accident with containment spray. The allowable bypass leakage is that leakage corresponding to a drywell $A/\sqrt{K} = 1.68 \text{ ft}^2$ <Section 6.2.1.1.5>. L_a is then the leakage rate corresponding to 0.168 ft^2 .

Drywell leakage rate tests are performed with the drywell isolation valves and system lineups in their postaccident position with the following exceptions:

- a. Emergency core cooling systems (ECCS) injection line valves are closed to prevent venting the drywell to the suppression pool.
- b. Air supply line isolation valves are closed to prevent the introduction of nonmetered air into the drywell.
- c. Control rod drive hydraulic system water supply valves remain open to maintain water flow through the control rod drives. The water flow prevents accumulation of crud in the control rod drives. Also, reactor water cleanup system valves are left open. The reactor water cleanup system is used to maintain reactor vessel level.
- d. Root valves for drywell pressure transmitter instrumentation interlocked with ECCS are closed to prevent ECCS actuation.

Any paths for equalizing drywell and containment pressure open during the Type A test are isolated. The containment air space external to the drywell is vented to atmosphere or to the annulus.

Preoperational drywell tests are performed as late as is practical in the construction sequence, but prior to initial operation.

For high pressure tests, the upper containment pool is filled to normal water level, the horizontal vents are plugged in order to achieve design pressure, and the suppression pool is filled.

After preoperational testing is satisfactorily completed, the drywell vent plugs can be removed and the suppression pool refilled. All subsequent periodic tests are conducted at a reduced pressure (less than that required to bubble drywell air through the top row of vents).

For all of the above tests, the drywell atmosphere is allowed to stabilize for one hour after attaining test pressure. When the steady-state conditions are achieved the drywell leakage rate is determined by metering the makeup air flow required to maintain the constant test pressure. Acceptance criteria are specified in technical specifications. Periodic drywell leakage rate tests are performed at the intervals specified in technical specifications.

The maximum allowable leakage rate into the annulus and the means to verify that the rate has not exceeded the bypass leakage rate is discussed in <Section 6.2.4>.

6.2.6.6 Primary Containment Leakage Rate Testing Program

The Primary Containment Leakage Rate Testing Program implements the leakage rate testing of the primary containment as required by <10 CFR 50.54(o)> and <10 CFR 50, Appendix J> (Option B) as modified by approved exemptions. This option allows the testing frequency for Type A, B, and C tests to be extended based on the leakage rate performance history of the components in the <10 CFR 50, Appendix J> program.

This Program is in accordance with the guidelines contained in NEI Topical Report NEI 94-01, Revision 3-A, with conditions and limitations in NEI 94-01, Revision 2-A, with the exceptions as identified in the Program and in USAR <Table 1.8-1>.

6.2.7 SUPPRESSION POOL MAKEUP SYSTEM

Following a LOCA, the suppression pool makeup system provides water from the upper containment pool to the suppression pool by gravity flow. The quantity of water provided is sufficient to account for all conceivable postaccident entrapment volumes (i.e., places where water can be stored while maintaining long term drywell vent water coverage).

6.2.7.1 Design Bases

The following criteria were used in the design of the suppression pool makeup system:

- a. The system is redundant with two 100 percent capacity lines. The redundant lines are physically separated and all electrical power and control is separated into two divisions in accordance with IEEE Standard 279.
- b. The system is Safety Class 2, Seismic Category I.
- c. The minimum long term postaccident suppression pool water coverage over the top of the top drywell vent is 2'-0".
- d. The minimum normal operation low water level (LWL) suppression pool height above the top drywell vent centerline is 6'-9.5".
- e. The maximum normal operation high water level (HWL) suppression pool height above the top drywell vent centerline is 7'-6".
- f. The suppression pool volume, between normal LWL and the minimum postaccident pool level, plus the makeup volume from the upper pool is adequate to supply all possible postaccident entrapment volumes for suppression pool water, and keep the suppression pool at an acceptable water level.

- g. The postaccident entrapment volumes causing suppression pool level drawdown include:
 - 1. The free volume inside and below the top of the drywell weir wall.
 - 2. The added water volume needed to fill the vessel from a condition of normal power operation to a postaccident complete fill of the vessel including top dome.
 - 3. Volume in the steam lines out to the first MSIV for three lines and out to the second MSIV on one line.
 - 4. An allowance for containment spray holdup on equipment and structural surfaces.
- h. No credit for feedwater or HPIS injection from the condensate storage tank is taken in calculating minimum postaccident suppression pool level.
- i. Piping components which would be wetted in the event of a drywell flooding transient (from inadvertent dump of the upper pool (SPMU) with the suppression pool at maximum operating level and under negative drywell pressure conditions) have been analyzed and would not result in adverse consequences to these components.
- j. The minimum normal operation suppression pool volume at LWL is adequate to act as a short term energy sink without taking credit for upper pool dump.
- k. The long term containment pressure and suppression pool temperature takes credit for the volume added postaccident from the upper containment pool.

1. The system gravity dump time through one of the two redundant lines is less than or equal to the minimum pump time; pump time is determined by dividing pumping volume (upper pool makeup volume plus volume in the suppression pool stored between LLWL and minimum top vent coverage) by the total maximum runout flow rate from all five ECCS pumps.

6.2.7.2 System Design

The piping system consists of two lines which penetrate the separator storage section of the upper containment pool through the side walls. One line is located on either side of the separator pool. From there, each line is routed down to the suppression pool on opposite sides of the steam tunnel. The elevation of the separator pool penetrations will limit the volume of water which can be dumped to the lower pool. This volume limitation along with adequate weir wall freeboard ensures that no excessive drywell flooding over the weir wall will occur for inadvertent opening of the valves on the suppression pool makeup lines.

The volume of the upper containment pool, which is available for suppression pool makeup, is equivalent to the volume of an 8.25-foot thick slice across the entire upper pool surface area, plus the separator pool volume between the top of the separator wall and the makeup system penetration to the upper pool. This requires that the refueling gate between the reactor well and the dryer storage pool be removed during power operation. The fuel transfer pool gate may be in place, removed, or placed in its stored position during power operation when the 360 Platform Troughs are not submerged in the Upper Containment Pool, since the volume of water in the fuel transfer pool is not then required for SPMU (Reference 32). However, the fuel transfer pool gate must be removed, or placed in its stored position when the 360 Platform Troughs are installed in the Upper Containment Pool during power operations, since the volume of water in the fuel transfer pool is then required for SPMU due to the inventory loss from the installation of the troughs.

The volume of water available for dump from the upper containment pool, when combined with the suppression pool, is adequate to supply all postaccident entrapment volumes and keep the suppression pool at an acceptable level (2 feet above the top of the horizontal drywell vents) to condense steam exiting the vents, and to ensure continuous coverage of the RHR A/B Test Return Line postaccident. The need to provide for this sufficient volume of "entrapment" water results in the operating limit for the minimum suppression pool level being 17'-9.5", plus a suppression pool level adjustment factor when a positive differential pressure exists between the drywell and the containment. This adjustment factor varies between 0.0" at 0.0 psid and 4.6" at 2.0 psid (the 2.0 psid value results in a minimum suppression pool level of approximately 18'-2"). This requirement for applying a limit of 17'-9.5" plus the level adjustment factor for positive differential pressures is quite different than the input assumptions for the containment response analyses, which assumed a minimum suppression pool water level of 17'-6" concurrent with the maximum drywell-to-containment operating limit of 2.0 psid <Table 6.2-5>. Although the containment response analysis showed that the smaller water volume was sufficient for containment cooling purposes, that smaller volume did not meet the entrapment volume design basis for the SPMU system.

The upper containment pool water level is required to be maintained during Operational Conditions 1, 2 and 3 at a level of 22'-9" above the RPV flange. The upper pool level may be reduced to 22'-5", as long as the suppression pool water level is raised 2.20 inches above the minimum allowable suppression pool water level of 17'-9.5" to compensate. Raising the minimum required suppression pool water level provides the same effective volume of water (by transferring a portion of the upper pool dump volume to the suppression pool) and ensures that after a suppression pool make-up system dump, adequate water coverage over the uppermost drywell horizontal vents and the long-term energy sink capability of the suppression pool are maintained (Reference 32).

In addition, when the inclined fuel transfer system blind flange is removed in Operational Conditions 1, 2, or 3, the upper containment pool water level is required to be maintained at a minimum level of 22'-9" above the reactor vessel flange and the suppression pool water level is required to be maintained at a minimum level of 17'-11.7". This is to ensure adequate suppression pool make up inventory in the event of an accident (Reference 39).

Also, just prior to a refueling outage, the reactor well to steam dryer pool gate may be installed during Operational Conditions 1, 2, or 3. The upper containment pool water level is required to be maintained at a minimum level of 23' above the reactor vessel flange and the suppression pool water is required to be maintained at a minimum level of 18'-3.2". During Operational Condition 3 (when entering a refueling outage with the reactor pressure reduced to or less than 230 psig and the reactor subcritical for at least 2 hours), the reactor well (cavity) portion of the upper containment pool may be drained to facilitate decontamination of the reactor well. The suppression pool water level will need to be increased and maintained with a prescribed band. The applicable water level bands will ensure adequate suppression pool makeup inventory in the event of an accident (Reference 41).

The fuel transfer pool gate may be in place, removed, or placed in its stored position when the 360 Platform Troughs are removed, since the volume of water in the fuel transfer pool is not required for SPMU. The fuel transfer pool gate must be removed, or placed in its stored position when the 360 Platform Troughs are installed, since the volume of water in the fuel transfer pool is required for SPMU due to the inventory loss from the installation of the troughs.

Each suppression pool makeup line has two normally closed valves in series. The valves on one line are on the same electrical division. Valves on the other line are on a different electrical division. All valves are powered from an onsite emergency power source which has divisional separation and redundancy.

The upper pool is dumped by gravity flow after opening the two normally closed series valves in each line. The valves on both lines receive divisionally separate but simultaneous signals to open. The open signal for each division is derived from either of two suppression pool level sensors. There are a total of four level sensors, two per division. There is also a series permissive signal permitting valve opening only when the LOCA signal exists. This LOCA signal is the same signal which actuates the ECCS pumps. This combination provides high reliability that the upper containment pool will be dumped when required but not dumped inadvertently by spurious signals.

The dump of the upper pool on low-low suppression pool level ensures adequate water volume to keep the suppression pool vents covered for all break sizes. In addition to the suppression pool level dump signal, the upper pool will also be dumped automatically from a timer set for LOCA + 30 minutes. This upper pool dump at 30 minutes postaccident insures that adequate heat sink is available long term regardless of break size or energy dump sequence.

The suppression pool makeup system will always operate following a DBA-LOCA as long as only one simultaneous equipment failure or operator error is postulated to occur. The suppression pool makeup system is specifically designed with redundant piping, valves and instrumentation to preclude failure to operate if a DBA-LOCA and any single equipment failure or operator error occurs <Figure 6.2-67>. Each of the two dump systems is independent and safety class.

The two series valves on each of two makeup system dump lines are located above the top of the drywell and outside the range of pool swell effect. The end of each line terminates in a configuration which provides an unobstructed free fall to the suppression pool. Pool level transmitters are located in the auxiliary building, protected from suppression pool dynamic effects.

6.2.7.3 Design Evaluation

6.2.7.3.1 Initiation

The opening of the makeup system valves is signaled by a series combination of low-low suppression pool level and a LOCA signal permissive (further discussion in <Section 6.2.7.2>). The low-low level signal is 16 inches below the normal LWL. Since maximum ECCS pump flow lowers the suppression pool at a rate of approximately 0.88 feet per minute, there is a minimum 1-1/2 minute delay between start of ECCS flow and dumping of the upper pool. The delay is actually longer than this because some vessel inventory mass is added to the suppression pool during blowdown steam condensation.

The makeup system dump valves can also be signaled to open by a LOCA signal in series with a 30 minute timer where the timer itself is started by the LOCA signal. This path of initiation logic is independent of suppression pool level and is specifically directed towards ensuring that the combined upper pool and suppression pool volumes are available as a heat sink for "small" breaks which do not lower the suppression pool to the LLWL trip, but continue to dump vessel blowdown energy into the pool. The minimum suppression pool volume, without upper pool dump, is adequate to meet all heat sink requirements for any combination sequence of vessel blowdown energy and decay heat energy out to 30 minutes.

A pool dump initiated from the LOCA +30 minute timer could result in higher vent submergence than the initial maximum of 7'-6". No problem exists in terms of pool swell since all the air would have been purged out of the drywell by the small break flow and only a small steam suppression pool vent flow would persist out to 30 minutes. Note that action of the drywell vacuum breakers which might reintroduce air into the drywell prior to 30 minutes postaccident will occur only after complete vessel depressurization and drywell steam condensation on the

"cold" ECCS break overflow of a relatively large break. The hypothesized high vent submergence would also have no effect on peak drywell pressure since the high submergence would only occur during small break flow events and after suppression pool vent clearing had already been established.

6.2.7.3.2 Flow

The suppression pool makeup volume is dumped in less than 10 minutes through one of two lines. The valves on the suppression pool makeup lines are fully open within 35 seconds of opening signal application.

6.2.7.3.3 Inadvertent Dump

The design of the opening signal circuitry for the suppression pool makeup valves assures high probability that no inadvertent dump will occur. The suppression pool level signal (LLWL) to open the valves is in series with a permissive which only allows the open signal to pass through when a LOCA signal exists on that division. Only a simultaneous signal of either suppression pool LLWL combined with LOCA, or LOCA with a 30-minute time delay will open both valves to allow gravity drain of the upper pool to the suppression pool. Even manual action is incapable of opening both series valves unless the administratively key locked dump test permissive switch is in the test mode or a LOCA permissive signal exists. The LOCA signal plus the timer signal after 30 minutes would dump the upper pool. However, the LOCA signal itself is a one out of two twice combination of high drywell pressure and low vessel water level <Figure 6.2-68> and a double failure is required to give a spurious LOCA signal.

Four level sensors indicate suppression pool water level with two sensors per electrical division. The two level sensors in one division are paralleled so that either sensor will initiate suppression pool makeup flow (pending LOCA permissive) from the makeup line whose series

valves are on the same electrical division as the level sensors. Level sensors on one electrical division cannot initiate flow from the makeup line where valves are in a separate electrical division.

There is a remote possibility that a single failure of a suppression pool level sensor and a concurrent LOCA event could initiate suppression pool makeup flow from one line such that the makeup flow started at the instant of LOCA. The flow from one makeup line will raise the suppression pool level at a rate of 0.538 feet per minute following full opening of the valves which normally prevent flow.

For a large break DBA, the peak drywell pressure occurs at about one second after the break with the pressure being reduced to the steady flow submergence of the top vent by about 30 seconds. Any pool swell associated loading would occur during the first few seconds while drywell air purge is taking place. Thus, the structural loading which would occur following a DBA would occur prior to any significant flow of water from a makeup line which was erroneously signaled to open at the same instant as the DBA.

The peak structural loadings associated with breaks smaller than the DBA are all less than the DBA case and only slightly extended in time. The drywell pressure for all size breaks is reduced to steady flow top vent submergence by one minute after the break.

The conclusion is that no increase in maximum structural loading due to a LOCA when an erroneous signal to initiate suppression pool makeup flow occurs at the instant of LOCA.

An inadvertent dump of the upper pool during any period of plant operation with a pressurized vessel does not represent, in and of itself, any hazard to the public, the plant operating personnel or any plant equipment. The drywell weir wall has sufficient freeboard height between the suppression pool surface and the top of the weir wall to

store most of the upper pool makeup volume on top of the normal suppression pool HWL with limited flooding over the weir wall into the drywell under negative drywell pressure conditions. The piping components which would be affected in this event have been analyzed for the floodings affect, and this event could not initiate a LOCA. The dumped upper pool makeup volume can be transferred back to the upper pool via several flow paths, thus restoring the initial suppression pool water level.

No fuel is stored in the upper pool during plant operations, therefore, depth of water shielding over fuel is not a concern for the inadvertent dump of the upper pool during plant operation. The Suppression Pool Makeup system valve initiation logic is designed with a keylocked system mode selector switch so that neither automatic nor manual action can open the suppression pool makeup valves while the plant is in the refueling mode.

6.2.7.4 Tests and Inspections

The suppression pool makeup valves will be manually tested periodically, one at a time, during plant power operation. An interlock prevents this manual testing unless the other valve in series on the same line is closed. The test will verify that the valve will open and close. Also, instruments will be periodically tested and inspected.

Preoperational testing will include a complete flow test of the system including a timed dump of the required makeup volume. Similar flow testing could be performed at any plant shutdown outage; however, the need for such testing occurs only a few times in the plant lifetime.

6.2.7.5 Instrumentation Requirements

Suppression pool water level sensors provide a signal to open the suppression pool makeup system valves. Four level instruments, two in each division, are provided. These instruments are the same analog instruments which measure normal water level variation with an extended range for LLWL <Section 7.3.1.>.

A level indication for the upper pool is also provided to alert the plant operating personnel if the level drops below that needed for the makeup volume. Level in the upper pool is normally maintained by a continuous overflow of level control weirs. The level is expected to stay constant during plant power operation, except for periods when level is purposely reduced, such as discussed in <Section 6.2.7.2>.

The upper pool and suppression pool temperature are monitored to ensure that the temperature does not exceed technical specification values. This ensures adequate heat sink capability of the suppression pool water, both short and long term.

A functional control diagram for the suppression pool makeup system is presented in <Section 7.3>.

6.2.8 HYDROGEN CONTROL SYSTEM

The accident at Three Mile Island - Unit 2, in 1979, involved a metal-water reaction in the core resulting in the release of amounts of hydrogen beyond the NRC's design basis accident requirements. Following the TMI accident, the NRC required a number of design improvements to all light water reactors. These TMI-related design improvements were for the purpose of further reducing the likelihood and effects of degraded core accidents, beyond the inherent design capability of the plants. Pursuant to these TMI-related items, the NRC staff required that plants with Mark III containments improve the hydrogen control capability by providing a system to control hydrogen generated from a postulated 75% metal-water reaction.

The final hydrogen control rule to address improved hydrogen control capability was published as an amendment to <10 CFR 50.44> on January 25, 1985, at 50 Federal Register 3498. For the PNPP Mark III containment design, the new rule requires a hydrogen control system capable of handling, without loss of containment structural integrity, an amount of hydrogen equivalent to that generated from a 75 percent metal-water reaction of the active fuel cladding during recoverable degraded core events. In addition, systems and components necessary to establish and maintain safe shutdown and containment integrity must be capable of performing their functions during and after exposure to the environmental conditions created by the burning of hydrogen.

The hydrogen control system, designed to control large amounts of hydrogen, is a distributed igniter system. The system burns hydrogen at low concentrations, thereby maintaining the concentration of hydrogen below levels which could potentially threaten containment integrity or equipment survivability.

6.2.8.1 Design Bases

6.2.8.1.1 Safety Design Bases

The safety design bases are:

- a. The hydrogen control system (HCS) will burn hydrogen at low volumetric concentrations. The igniters are designed to ensure controlled burning of hydrogen equivalent to that generated from a 75% metal-water reaction of the active fuel cladding (2,475 pounds hydrogen mass) as a result of a postulated recoverable degraded core accident.
- b. The HCS will be placed in operation upon reactor vessel water level decreasing to the top of active fuel (TAF) or upon hydrogen concentration in the drywell or containment reaching minimum detectable level. The HCS is secured automatically if power to the igniter is lost and manually when hydrogen concentrations inside the drywell or containment cannot be determined to be below predetermined limits.
- c. The HCS is designed with suitable redundancy to ensure that no single active component failure, including power supply failures, will prevent functioning of the system.
- d. Once activated, the igniters are designed to reliably ignite hydrogen for up to seven days.
- e. The HCS is designed as a safety grade system, and is capable of operating for the duration of the hydrogen generation event.
- f. The HCS igniter assemblies are classified and designed as electrical Class 1E and Seismic Category I. Each igniter is

powered from a Class 1E power supply which can be powered from the associated division diesel generator.

- g. All components in the HIS are capable of withstanding the temperature and pressure transients resulting from a LOCA. They can also withstand the humidity conditions and radiation environment in which the components are located. Environmental qualification of the hydrogen igniter assemblies is addressed in <Section 3.11>, and dynamic qualification is addressed in <Section 3.10>.
- h. HCS components located within the containment are demonstrated to survive the expected hydrogen combustion thermal environment where they are located.
- i. Igniter locations were based on criteria that considered potential hydrogen release locations, appropriate spacing in open areas, redundancy and potential for pocketing in enclosed regions. Based on these criteria, igniters are located in a ring above the suppression pool and throughout the containment and drywell. The igniters are located approximately every 30 feet with alternating divisional power supplies, such that a distance of approximately 60 feet may exist if only one emergency power division is available. In some cases, a distance of 35 feet and 70 feet may exist if adequate supports are not available or interference exists in the area of interest. These criteria were used except in the open area above the refueling floor in the upper containment and in the reflood region in the drywell. Two igniters, one from each power division, are located in enclosed areas within containment which would be subject to possible hydrogen pocketing.

6.2.8.2 System Design

A tabulation of the design and performance data for each system component is located in <Table 6.2-43>. A detailed discussion of instrumentation features is provided in <Section 7.6>.

The hydrogen control system (HCS) consists of 102 igniters distributed throughout the drywell and containment. The igniters are designed to ensure controlled burning of hydrogen produced during a hydrogen generation event.

The igniter is a thermal ignition device that when activated (by an electric current) produces a resistance at the element (or tip) and an increase in temperature of at least 1,700°F. This tip temperature is sufficient to cause combustion of the surrounding hydrogen gases at relatively low concentrations. The 102 igniters are glow plugs of the type commonly used in diesel engines.

The igniter is mounted in an igniter assembly or housing with only the top (glow plug) exposed. The housing is constructed of stainless steel and contains a transformer with multiple taps to step down the voltage to each igniter from 120 volt ac, a terminal block for connection of internal wiring, and all of the associated electrical wiring required to make the assembly functional. The housing (assembly) is designed with a spray shield which extends over the glow plug tip, to protect against a reduction in tip temperature caused by the direct impingement of containment spray. A junction box is attached to the exterior of the assembly and contains the cable termination. Gasketing material and sealant are provided to ensure leak-tightness of the igniter enclosure and junction box.

The 120 volt ac, 60 Hz, Class 1E power is supplied to the ignitor assemblies through hydrogen ignitor isolation panels. These isolation panels receive their power from Class 1E 480V motor control

centers (MCC) through 15 kVA transformers, rated 480-208/120 volt ac, 60 Hz, 3-phases with grounded neutrals, and a fuse panel. The fuse panel consists of a 40 ampere and 45 ampere fuse in series for each line to the hydrogen ignitor isolation panels. Each transformer is fed from a Class 1E MCC which is capable of being powered from one of the standby diesel generators. The 102 igniters are divided into six groups of approximately equal number, three groups in Division 1 and three groups in Division 2. Each group is powered from a separate hydrogen ignitor isolation panel. Disconnect devices are provided for maintenance and are normally closed (NC) so the igniters can be energized by operating the control room handswitches.

The HCS is manually energized by means of two handswitches located in the control room. There is one handswitch for the three Division 1 groups and one for the three Division 2 groups.

The instrumentation and controls for the HCS are discussed in <Section 7.6>.

The hydrogen igniters are arranged in several rings at different elevations within the drywell and containment. At least two igniters are located in each enclosed volume or area within the containment that are subject to possible hydrogen pocketing; each of the two igniters is powered from a separate power division to maintain divisional redundancy. <Figure 6.2-82> shows the locations of the igniter assemblies within containment.

6.2.8.3 Design Evaluation

A preliminary evaluation of the PNPP hydrogen control system has been completed. This evaluation meets and exceeds the requirements of <10 CFR 50.44> for preliminary analysis. The details of the analysis are provided in (Reference 29). The program included definition of hydrogen generation events, determination of hydrogen and steam release

histories for recoverable degraded cores, definition of hydrogen combustion thermal environments, and preliminary analyses of the survivability of essential equipment when exposed to the thermal and pressure environment inside containment during hydrogen combustion. An additional program of test and analysis has been conducted to support the final evaluation which was submitted to the NRC in accordance with <10 CFR 50.44>.

The preliminary evaluation is based on the analyses of two degraded core accident scenarios. One scenario involved a stuck-open relief valve (SORV) accompanied by a failure of ECCS. The second scenario involved a small steam line break in the drywell (DWB). These two scenarios were selected to define temperature and pressure responses in the containment and drywell. The SORV transient scenario was considered to be the more probable event and to create a more thermally limiting environment in the containment. The drywell break scenario was chosen only because of the potential for hydrogen combustion in the drywell and therefore a limiting thermal environment in the drywell.

For the two cases analyzed (using the CLASIX-3 code), the peak calculated containment pressure was approximately 21 psig and brief duration temperature peaks ranged from 643°F in the drywell to 760°F in the containment to 1,762°F in the wetwell. The results are provided in <Table 6.2-46>.

The peak calculated containment pressure is significantly below the containment pressure vessel capability <Section 3.8.2.3.6>.

A preliminary evaluation was performed to identify equipment inside containment that is required to survive a hydrogen burn. A list of equipment required to survive a hydrogen burn was established using the following criteria:

- a. Equipment and systems which must function to mitigate the consequences of the event.
- b. Equipment and structures required to maintain the integrity of the containment pressure boundary.
- c. Systems and components required to maintain the core in a safe shutdown condition.
- d. Instrumentation and systems which are used to monitor the course of the event and provide guidance to the operator for initiating actions in accordance with the emergency procedures.
- e. Components whose failure could preclude the ability of the above systems to fulfill their intended functions.

The list of equipment required to survive a hydrogen burn is found in <Table 6.2-44> and <Table 6.2-45>.

Preliminary equipment survivability was demonstrated by showing that the equipment survivability analyses performed for the Grand Gulf Nuclear Station (GGNS) are applicable to PNPP. The GGNS analyses had been previously reviewed and approved by the NRC.

Equipment thermal survivability was demonstrated by comparing the temperature profiles, thermal response analysis and equipment list of PNPP to those of GGNS. The temperature profiles of the PNPP and GGNS CLASIX-3 containment response analyses are comparable with the exception of minor differences. Further analysis of the thermal response of the

hydrogen igniter assembly showed that the PNPP CLASIX-3 temperature profile produced a lower maximum equipment temperature than the GGNS CLASIX-3 temperature profile. It was also shown that the list of equipment identified for PNPP was similar to that of GGNS. Therefore, it was concluded that the GGNS analyses are applicable to PNPP and that PNPP equipment will survive the temperatures produced during a hydrogen burn.

Equipment pressure survivability was demonstrated by comparing the qualification or design pressure to calculated peak pressures. The qualification or design pressures of required equipment bound the calculated peak pressures <Table 6.2-46> in all cases.

6.2.8.4 Testing and Inspection

The hydrogen control system (HCS) is tested by energizing each igniter assembly, verifying a surface temperature of at least 1,700°F for each accessible igniter and verifying by measurement, sufficient current/voltage to develop a surface temperature of 1,700°F for each of the remaining igniters.

Surveillance testing is periodically done by energizing all the igniter assemblies and performing a current and voltage measurement of each circuit to identify any inoperable igniters (see Technical Specifications).

6.2.8.5 Instrumentation Requirements

Operation of the hydrogen control subsystems is performed manually. The controls are in the control room. The igniters are energized by means of two handswitches. There is one handswitch for the three Division 1 groups and one for the three Division 2 groups. The switch positions

are OFF-NORM-ON with red-green indication lights. Input is provided to the hydrogen control system out-of-service annunciator in the control room on loss of control or motive power.

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31. Young, C. T., "Perry Technical Specifications Improvement-Containment Response Analysis," General Electric Company, NEDC-31940, March 1991.
32. Letter, M. D. Lyster (Centerior) to USNRC, "Revisions to the Suppression Pool Makeup Specification to Operate with a Reduced Upper Containment Pool Water Level Provided the Suppression Pool Water is Raised to Compensate," PY-CEI/NRR-1537L, November 16, 1992.
33. (Deleted)
34. Utility Resolution Guidance for ECCS Suction Strainer Blockage, NEDO-32686, November 1996.
35. License Amendment No. 100 dated February 24, 1999, D. Pickett (NRC) to L. Myers (FENOC).
36. License Amendment No. 21 dated May 23, 1989, T. Colburn (NRC) to A. Kaplan (CEI).
37. GE Data Book 22A3759AL Rev. 2, March 29, 1991, "Containment and NSSS Interface."
38. NUREG-0800 SRP 6.2.1.1.c, Pressure-Suppression Type BWR Containment.
39. License Amendment No. 123 dated March 7, 2003, S. Sands (NRC) to W. Kanda (FENOC).

40. (Deleted)

41. License Amendment No. 174 dated February 16, 2017.

TABLE 6.2-1

KEY DESIGN AND MAXIMUM ACCIDENT PARAMETERS FOR
PRESSURE SUPPRESSION CONTAINMENT

<u>Parameter</u>	<u>Design Value</u>	<u>Maximum Calculated Accident Value</u>
Containment Pressure, psig	15	6.4
Suppression Pool Temperature, °F	185	181.3
Drywell Pressure, psig	30	23.45
Drywell Temperature, °F	330	329.0

TABLE 6.2-2

CONTAINMENT DESIGN PARAMETERS

	<u>Drywell</u>	<u>Containment</u>
Drywell and Containment		
Negative Design Pressure, psig	-21.0	-0.8
Positive Design Pressure, psig	30	15
Design Temperature, °F	330	185
Maximum Allowable Leak Rate	$A/\sqrt{K} = 1.68 \text{ ft}^2$	0.2%/day
Suppression Pool Depth, ft		
Low Level	17.79	17.79
High Level	18.5	18.5
Vent System		
Number of Vents		120
Nominal Vent Diameter, ft		2.29
Total Vent Area, ft ²		495
Vent Centerline Submergence (low level), ft		
Top Row		6.5
Middle Row		11.0
Bottom Row		15.5
Vent Loss Coefficient (varies with number of vents open)		2.5-3.5

TABLE 6.2-3

ENGINEERED SAFETY FEATURE SYSTEMS
PERFORMANCE PARAMETERS FOR CONTAINMENT RESPONSE ANALYSES

	<u>Full Capacity</u>	<u>Containment Analysis Value</u>	
		<u>Case A</u>	<u>Case B</u>
<u>Containment Spray</u>			
Number of RHR Pumps	2	0	0
Number of Lines	2	0	0
Number of Heat Exchangers	2	0	0
Flow Rate, gpm/pump	5,250	0	0
<u>Containment Cooling System</u>			
Number of RHR Pumps	2	2	1
Pump Capacity, gpm/pump	7,100	7,100	7,100
RHR Heat Exchangers			
Type	Inverted U-tube, single pass shell, multipress tube, vertical mounting		
Number	2	2	1
Heat Transfer Area, ft ² /unit	14,850	-	-
Overall Heat Transfer Coefficient, Btu/hr-ft ² -°F/unit	200	-	-
Emergency Service Water			
Temperature, °F			
Minimum Design	32	-	-
Maximum Design	85	85	85

TABLE 6.2-3 (Continued)

	<u>Full Capacity</u>	<u>Containment Analysis Value</u>	
		<u>Case A</u>	<u>Case B</u>
Containment Heat Removal Capability (using 85°F emergency service water and 185°F pool temperature), Btu/hr/unit	158.4E+06	-	-

TABLE 6.2-4

ACCIDENT ASSUMPTIONS AND INITIAL CONDITIONS FOR
CONTAINMENT RESPONSE ANALYSES
 (at 3833 MWt)

Components of Effective Break Area(Recirculation Line Break), ft²

Recirculation Line	2.127
Cleanup Line	0.080
Jet Pumps	0.468

Primary Steam Energy Distribution⁽¹⁾, 10⁶ Btu

Steam Energy	23.6
Liquid Energy	299.6
Sensible Energy	
Reactor Vessel	89.62
Reactor Internals (less core)	51.59
Primary System Piping	35.55
Fuel ⁽²⁾	7.3

Other Assumptions Used in Analysis

Main Steam Closure Time, sec	
Recirculation Break	3.5
Main Steam Line Break	5.5
Scram Time, sec	<1

NOTES:⁽¹⁾ All energy values, except fuel, are based upon a 32°F datum.⁽²⁾ Fuel energy is based upon a datum of 285°F.

TABLE 6.2-5

INITIAL CONDITIONS EMPLOYED IN
CONTAINMENT RESPONSE ANALYSES
(at 3833 MWt)

Drywell Free Volume, ft ³	279,528
Containment Free Volume, ft ³	1.156 x 10 ⁶
Drawdown makeup system capacity, ft ³	32,573
Temperature of makeup water, °F	110
Suppression Pool Volume for Long Term Response Calculations, ft ³ (assumes initial pool level of 17.79 feet when drywell-to-containment dP is 2.0 psig, and includes the upper pool dump volume.)	144,292
Suppression Pool Volume for Short Term Response Calculations, ft ³ (assumes initial pool level of 18.5 feet when drywell-to-containment dP is -0.5 psig.)	118,131
Horizontal Vent I.D., inches	27.5
Total Number of Vents	120
Weir Annulus Width, inches	27.25
Submergence of Vent Centerline - 1st vent, ft	7.5 (HWL-STR) 6.5 (LWL-LTR)
Submergence of Vent Centerline - 2nd vent, ft	12.0-STR 11.0-LTR
Submergence of Vent Centerline - 3rd vent, ft	16.5-STR 15.5-LTR
Net Vent Area, ft ²	495
Drywell Air Temperature, °F	60-145 ⁽²⁾ (145 for long-term response; 135 for short-term response)

TABLE 6.2-5 (Continued)

Drywell Pressure, psig	-0.5 to 2.0 ⁽²⁾ (2.0 for long-term response; 0.5 for short-term response)
Drywell Humidity, %	normal; 40-50; (40) ⁽²⁾
Containment Air Temperature, °F	normal; 95 (104) ⁽²⁾
Containment Pressure, psig	-0.1 to 1.0 ⁽²⁾ (0 for long-term response; 1 for short-term response)
Containment Humidity, %	normal; 50-60; (50) ⁽²⁾
<u>Reactor Primary System Data</u>	
Reactor Power, MWt	3,833
Vessel Steam Output, lb _m /hr	16,689,000
Steam Dome Pressure, psia	1,060
Total Reactor Fluid Inventory, lb _m	5.641E5
Total Mass of Vessel and Internal Structures, lb (not including fuel)	2.764E6
Total Mass of UO ₂ in core, lb _m	3.416 x 10 ⁵
Total Mass of Active Zircaloy in core, lb _m	80,694
Main Steam Line Isolation Valve Closure Period, sec	4 ± 1 ⁽¹⁾

NOTES:

- ⁽¹⁾ For mass release calculations, 0.5 seconds must be added to account for instrumentation time needed to initiate main steam line isolation valve closure.
- ⁽²⁾ The analyses are valid for the values and ranges shown. Values actually used in the analyses are shown in parentheses.

TABLE 6.2-6

SUMMARY OF SHORT TERM CONTAINMENT RESPONSES TO
RECIRCULATION LINE AND MAIN STEAM LINE BREAKS
(up to 30 seconds)

	<u>Recirculation</u> <u>Line Break</u>	<u>Main Steam</u> <u>Line Break</u>
Peak Drywell Pressure, psig	22.84	23.45
Peak Drywell Differential Pressure, psid	21.07	21.35
Time of Peak Pressure, sec	1.79	1.80
Peak Drywell Temperature, °F	308.3	329.0
Peak Wetwell Pressure, psig	6.6	11.4
Time of Peak Wetwell Pressure, sec	3.0	3.0
Calculated Drywell Margin, %	23.9	21.8

TABLE 6.2-7

SUMMARY OF LONG TERM CONTAINMENT RESPONSES TO
RECIRCULATION LINE OR MAIN STEAM LINE BREAK
 (at 3729 MWt)

	<u>Case A</u> ⁽¹⁾	<u>Case B1</u> ⁽²⁾
Peak Containment Pressure, psig	5.8	7.8
Time of Peak Containment Pressure, sec	67,953	99,853
Peak Suppression Pool Temperature, °F	178.6	182.7
Calculated Containment Margin, %	61.4	48.3
HPCS Flow Rate, gpm	6,000	6,000
LPCS Flow Rate, gpm	6,000	6,000
RHRS Flow Rate, gpm	14,200 ⁽³⁾ 7,100 ⁽⁴⁾	7,100 ⁽³⁾ 7,100 ⁽⁴⁾

NOTES:

- ⁽¹⁾ Case A - Offsite Power Available, All ECCS Equipment Operating.
⁽²⁾ Case B1 - Loss-of-Offsite Power, Minimum ECCS Equipment Operating.
⁽³⁾ Suppression Pool Cooling Flow; None for First 30 minutes.
⁽⁴⁾ LPCI Mode of RHR.

TABLE 6.2-7a

SUMMARY OF LONG TERM CONTAINMENT RESPONSES TO
RECIRCULATION LINE OR MAIN STEAM LINE BREAK
 (at 3833 MWt)

	<u>Case B2⁽¹⁾</u>
Peak Containment Pressure, psig	6.4
Time of Peak Containment Pressure, sec	31,308
Peak Suppression Pool Temperature, °F	181.3
Calculated Containment Margin, %	56.7
HPCS Flow Rate, gpm	7,000
LPCS Flow Rate, gpm	7,000
RHRS Flow Rate, gpm	7,100 ⁽²⁾ 7,100 ⁽³⁾

NOTES:

- ⁽¹⁾ Case B2 - Loss-of-Offsite Power, Minimum ECCS Equipment Operating.
⁽²⁾ Suppression Pool Cooling Flow; None for First 30 minutes.
⁽³⁾ LPCI Mode of RHR.

TABLE 6.2-8

ENERGY BALANCE FOR
MAIN STEAM LINE BREAK ACCIDENT
(at 3729 MWt)

	Energy (Btu)			
	Initial (time zero)	Drywell Peak Pressure	End of Blowdown	Long Term Peak Wetwell Pressure
Reactor Coolant	3.2E+08	2.9E8	1.8E8	1.7E8
Fuel and Cladding				
Fuel	7.3E6	7.2E+06	0	0
Cladding	3.4E+06	3.4E+06	3.1E6	1.4E6
Core Internals	8.4E7	8.4E7	7.4E7	3.5E7
Reactor Vessel Metal	9.0E7	9.0E7	8.4E7	4.6E7
Reactor Coolant System Piping, Pumps, and Valves	Included in "Core Internals", above.			
Blowdown Enthalpy				
Liquid	0	0	0	0
Steam	0	3.4E7	4.1E8	4.1E9
Decay Heat	0	1.4E6	8.1E7	3.2E9
Metal-Water Reaction Heat	0	6.1E4	1.9E6	1.9E6
Drywell Structures	4.3E7	4.3E7	4.5E7	8.2E7
Drywell Air	2.0E6	4.5E5	0	0
Drywell Steam	1.1E6	1.4E7	1.8E7	2.2E7
Containment Air	7.6E+06	9.7E6	1.0E+07	1.0E7
Containment Steam	1.9E6	2.5E+06	2.7E6	9.8E6
Suppression Pool Water	4.0E8	4.3E8	7.7E8	1.2E9

TABLE 6.2-8 (Continued)

	Energy (Btu)			
	Initial (time zero)	Drywell Peak Pressure	End of Blowdown	Long Term Peak Wetwell Pressure
Upper Pool Dump Inventory	1.6	1.6	1.6	0
Energy Transferred by Heat Exchangers	0	0	0	3.9E+08
Passive Heat Sinks ⁽¹⁾ (in Containment Airspace and Suppression Pool)	6.4E7	6.4E7	6.4E7	8.4E7

NOTE:

- ⁽¹⁾ A new strainer design, which has the strainer resting on the floor of the suppression pool, replaces the individual strainers for the ECCS and RCIC system pumps in response to <NRC Bulletin 96-03>. The new strainer adds ~426 ft³ of steel to the suppression pool, however, the strainer displaces ~426 ft³ of suppression pool water. Analysis has shown that the addition of the strainer's steel and displacement of the water results in a negligible effect on the passive heat sinks available in the suppression pool and does not invalidate existing analyses.

TABLE 6.2-9

ACCIDENT CHRONOLOGY FOR
MAIN STEAM LINE BREAK ACCIDENT
 (at 3729 MWt)

<u>Event</u>	<u>Time (sec)</u>	
	<u>All ECCS in Operation</u>	<u>Minimum ECCS Available</u>
First Row Vent Cleared	0.934	0.934
Second Row Vent Cleared	1.171	1.171
Third Row Vent Cleared	1.550	1.550
Drywell Reaches Peak Pressure	3.733	3.733
Maximum Positive Differential Pressure Occurs	1.28	1.28
Third Row Vent Recovered	28.4	28.4
Initiation of ECCS Operation	30	30
Second Row Vent Recovered	47.8	47.8
End of Blowdown	319	360
First Row Vent Recovered	344	344
Initiation of RHR Heat Exchanger Operation	1,980	1,980
Containment Peak Pressure Reached	67,953	99,853

TABLE 6.2-10

AVAILABLE CONTAINMENT HEAT SINKS

<u>Item</u>	<u>Volume (ft³)</u>	<u>Surface Area (ft²)</u>	<u>Material</u>
Drywell Structures	4,556	54,891	Concrete
Containment Shell	9,122	72,975	Steel
Miscellaneous Steel Structures and Equipment	7,512 ⁽¹⁾	169,189	Steel
Miscellaneous Concrete Structures	38,836	21,360	Concrete

NOTE:

- ⁽¹⁾ A new strainer design, which has the strainer resting on the floor of the suppression pool, replaces the individual strainers for the ECCS and RCIC system pumps in response to <NRC Bulletin 96-03>. The new strainer adds ~426 ft³ of steel to the suppression pool, however, the strainer displaces ~426 ft³ of suppression pool water. Analysis has shown that the addition of the strainer's steel and displacement of the water results in a negligible effect on the passive heat sinks available in the suppression pool and does not invalidate existing analyses.

<TABLE 6.2-11>

DELETED

TABLE 6.2-12

REACTOR ANNULUS

a. Recirculation Suction Line

Control Volume No.	Description	Height (ft)	Volume (ft ³)	Volume Flow Area (ft ²)	Bottom Elevation (ft)	Initial Conditions			Calculated Peak $\Delta P^{(1) (2)}$ (psid)	Biological Shield Wall Design Pressure ⁽²⁾ (psid)
						Temperature (°F)	Pressure (psia)	Quality		
1	Volume 1 - Level 1	5.81	91.56	16.80	6.83	212.0	14.696	0.926	22.06	28.78
2	Volume 2 - Level 1	8.71	138.46	16.80	6.83	212.0	14.696	0.926	9.44	12.59
3	Volume 3 - Level 1	8.71	142.96	16.80	6.83	212.0	14.696	0.926	5.66	8.81
4	Volume 4 - Level 1	8.71	205.13	25.20	6.83	212.0	14.696	0.926	2.06	3.43
5	Volume 5 - Level 1	8.71	212.77	25.20	6.83	212.0	14.696	0.926	0.40	0.71
6	Volume 6 - Level 2	5.81	82.56	16.80	18.44	212.0	14.696	0.926	8.13	10.37
7	Volume 7 - Level 2	8.71	128.13	16.80	15.54	212.0	14.696	0.926	4.44	5.71
8	Volume 8 - Level 2	8.71	132.63	16.80	15.54	212.0	14.696	0.926	3.65	5.38
9	Volume 9 - Level 2	8.71	195.00	25.20	15.54	212.0	14.696	0.926	1.23	2.07
10	Volume 10 - Level 2	8.71	210.31	25.20	15.54	212.0	14.696	0.926	0.15	0.31
11	Volume 11 - Level 3	6.875	100.64	16.80	24.25	212.0	14.696	0.926	6.02	8.34
12	Volume 12 - Level 3	6.875	112.45	16.80	24.25	212.0	14.696	0.926	3.39	5.01
13	Volume 13 - Level 3	6.875	112.45	16.80	24.25	212.0	14.696	0.926	2.61	4.14
14	Volume 14 - Level 3	6.875	166.72	25.20	24.25	212.0	14.696	0.926	0.68	1.16
15	Volume 15 - Level 3	6.875	174.59	25.20	24.25	212.0	14.696	0.926	0.09	0.21
16	Volume 16 - Level 4	6.925	147.00	25.20	31.125	212.0	14.696	0.926	3.13	4.94
17	Volume 17 - Level 4	6.925	158.88	25.20	31.125	212.0	14.696	0.926	1.63	2.60
18	Volume 18 - Level 4	6.925	166.76	25.20	31.125	212.0	14.696	0.926	0.36	0.63
19	Volume 19 - Level 4	6.925	174.57	25.20	31.125	212.0	14.696	0.926	0.03	0.10
20	Volume 20 - Level 4	13.66	297.69	25.20	38.05	212.0	14.696	0.926	0.00	0.04
21	Volume 21 - Level 4	13.66	325.41	25.20	38.05	212.0	14.696	0.926	0.10	0.22
22	Volume 22 - Level 4	13.66	316.20	25.20	38.05	212.0	14.696	0.926	0.00	0.04
23	Volume 23 - Level 4	13.66	344.37	25.20	38.05	212.0	14.696	0.926	0.00	0.00
24	Volume 24 - Drywell	90.00	139,000.0	1544.0	0.0	212.0	14.696	0.52	-	-
25	Volume 25 - Blowdown Node	5.80	71.48	16.80	12.64	212.0	14.696	0.926	17.10	22.90
26	Volume 26 - Level 5	6.50	188.25	25.20	51.71	212.0	14.696	0.926	0.00	0.01
27	Volume 27 - Level 5	6.50	178.42	25.20	51.71	212.0	14.696	0.926	0.00	0.03
28	Volume 28 - Level 5	6.50	178.42	25.20	51.71	212.0	14.696	0.926	0.00	0.00
29	Volume 29 - Level 5	6.50	188.25	25.20	51.75	212.0	14.696	0.926	0.00	0.00

TABLE 6.2-12 (Continued)

b. Recirculation Discharge Line

Control Volume No.	Description	Height (ft)	Volume (ft ³)	Volume Flow Area (ft ²)	Bottom Elevation (ft)	Initial Conditions			Calculated Peak $\Delta P^{(1) (2)}$ (psid)	Biological Shield Wall Design Pressure ⁽²⁾ (psid)
						Temperature (°F)	Pressure (psia)	Quality		
1	Volume 1 - Level 1	5.81	91.56	16.80	6.83	212.0	14.696	0.926	31.49	44.09
2	Volume 2 - Level 1	8.71	138.46	16.80	6.83	212.0	14.696	0.926	11.80	16.51
3	Volume 3 - Level 1	8.71	142.96	16.80	6.83	212.0	14.696	0.926	8.19	11.47
4	Volume 4 - Level 1	8.71	205.13	25.20	6.83	212.0	14.696	0.926	3.03	4.23
5	Volume 5 - Level 1	8.71	212.77	25.20	6.83	212.0	14.696	0.926	0.65	0.91
6	Volume 6 - Level 2	5.81	82.56	16.80	18.44	212.0	14.696	0.926	12.79	17.89
7	Volume 7 - Level 2	8.71	128.13	16.80	15.54	212.0	14.696	0.926	5.90	8.26
8	Volume 8 - Level 2	8.71	132.63	16.80	15.54	212.0	14.696	0.926	4.27	5.96
9	Volume 9 - Level 2	8.71	195.00	25.20	15.54	212.0	14.696	0.926	1.74	2.44
10	Volume 10 - Level 2	8.71	210.31	25.20	15.54	212.0	14.696	0.926	0.28	0.39
11	Volume 11 - Level 3	6.875	100.64	16.80	24.25	212.0	14.696	0.926	8.98	12.57
12	Volume 12 - Level 3	6.875	112.45	16.80	24.25	212.0	14.696	0.926	4.74	6.64
13	Volume 13 - Level 3	6.875	112.45	16.80	24.25	212.0	14.696	0.926	3.55	4.96
14	Volume 14 - Level 3	6.875	166.72	25.20	24.25	212.0	14.696	0.926	1.05	1.47
15	Volume 15 - Level 3	6.875	174.59	25.20	24.25	212.0	14.696	0.926	0.20	0.28
16	Volume 16 - Level 4	6.925	147.00	25.20	31.125	212.0	14.696	0.926	4.41	6.16
17	Volume 17 - Level 4	6.925	158.88	25.20	31.125	212.0	14.696	0.926	2.26	3.16
18	Volume 18 - Level 4	6.925	166.76	25.20	31.125	212.0	14.696	0.926	0.56	0.78
19	Volume 19 - Level 4	6.925	174.57	25.20	31.125	212.0	14.696	0.926	0.10	0.13
20	Volume 20 - Level 4	13.66	297.69	25.20	38.05	212.0	14.696	0.926	0.04	0.06
21	Volume 21 - Level 4	13.66	325.41	25.20	38.05	212.0	14.696	0.926	0.20	0.28
22	Volume 22 - Level 4	13.66	316.20	25.20	38.05	212.0	14.696	0.926	0.04	0.06
23	Volume 23 - Level 4	13.66	344.37	25.20	38.05	212.0	14.696	0.926	0.00	0.00
24	Volume 24 - Drywell	90.00	139,000.0	1544.0	0.0	212.0	14.696	0.52	-	-
25	Volume 25 - Blowdown Node	5.80	71.48	16.80	12.64	212.0	14.696	0.926	28.19	39.47
26	Volume 26 - Level 5	6.50	188.25	25.20	51.71	212.0	14.696	0.926	0.02	0.03
27	Volume 27 - Level 5	6.50	178.42	25.20	51.71	212.0	14.696	0.926	0.03	0.04
28	Volume 28 - Level 5	6.50	178.42	25.20	51.71	212.0	14.696	0.926	0.00	0.00
29	Volume 29 - Level 5	6.50	188.25	25.20	51.75	212.0	14.696	0.926	0.00	0.00

NOTES:

(1) With respect to drywell.

(2) Biological shield wall design pressures are for time-step 0.024 seconds. A study was performed to determine which pressure loading would cause maximum shear and moment at the base of the structure. The controlling time-step pressures (0.024 sec for recirculation discharge break) were then used as the design loading case.

TABLE 6.2-13

REACTOR ANNULUS
FEEDWATER LINE BREAK

Control Volume	No.	Description	Height (ft)	Volume (ft ³)	Volume Flow Area (ft ²)	Bottom Elevation (ft)	Initial Conditions			Calculated Peak $\Delta P^{(1) (2)}$ (psid)	Biological Shield Wall Design Pressure ⁽²⁾ (psid)
							Temperature (°F)	Pressure (psia)	Quality		
	1	Volume 1 - Level 1	9.44	137.31	16.93	5.5	212.0	14.696	0.926	14.04	19.66
	2	Volume 2 - Level 1	9.44	205.27	25.39	5.5	212.0	14.696	0.926	14.03	19.64
	3	Volume 3 - Level 1	9.44	272.93	33.86	5.5	212.0	14.696	0.926	14.05	19.66
	4	Volume 4 - Level 1	9.44	280.55	33.86	5.5	212.0	14.696	0.926	14.29	20.01
	5	Volume 5 - Level 1	9.44	284.02	33.86	5.5	212.0	14.696	0.926	14.11	19.75
	6	Volume 6 - Level 1	9.44	282.10	33.86	5.5	212.0	14.696	0.926	14.00	19.60
	7	Volume 7 - Level 1	9.44	179.28	25.39	5.5	212.0	14.696	0.926	14.04	19.66
	8	Volume 8 - Level 2	9.44	121.97	16.93	14.94	212.0	14.696	0.926	17.49	24.49
	9	Volume 9 - Level 2	9.44	200.65	25.39	14.94	212.0	14.696	0.926	16.81	23.53
	10	Volume 10 - Level 2	9.44	272.93	33.86	14.94	212.0	14.696	0.926	15.65	21.91
	11	Volume 11 - Level 2	9.44	293.93	33.86	14.94	212.0	14.696	0.926	14.27	19.98
	12	Volume 12 - Level 2	9.44	292.15	33.86	14.94	212.0	14.696	0.926	14.38	20.12
	13	Volume 13 - Level 2	9.44	266.93	33.86	14.94	212.0	14.696	0.926	15.07	21.10
	14	Volume 14 - Level 2	9.44	189.49	25.39	14.94	212.0	14.696	0.926	16.50	23.10
	15	Volume 15 - Level 3	6.50	94.81	16.93	24.38	212.0	14.696	0.926	16.35	22.89
	16	Volume 16 - Level 3	6.50	149.73	25.39	24.38	212.0	14.696	0.926	17.19	24.07
	17	Volume 17 - Level 3	6.50	207.92	33.86	24.38	212.0	14.696	0.926	15.55	21.77
	18	Volume 18 - Level 3	6.50	219.92	33.86	24.38	212.0	14.696	0.926	13.60	19.04
	19	Volume 19 - Level 3	6.50	219.63	33.86	24.38	212.0	14.696	0.926	14.55	20.37
	20	Volume 20 - Level 3	6.50	201.92	33.86	24.38	212.0	14.696	0.926	15.34	21.46
	21	Volume 21 - Level 3	6.50	146.89	25.39	24.38	212.0	14.696	0.926	16.52	23.13
	22	Volume 22 - Level 4	6.50	99.81	16.83	30.88	212.0	14.696	0.926	17.37	24.32
	23	Volume 23 - Level 4	6.50	149.87	30.88	30.88	212.0	14.696	0.926	16.97	23.76
	24	Volume 24 - Level 4	6.50	202.63	33.86	30.88	212.0	14.696	0.926	15.14	21.18
	25	Volume 25 - Level 4	6.50	220.09	33.86	30.88	212.0	14.696	0.926	9.83	13.76
	26	Volume 26 - Level 4	6.50	220.09	33.86	30.88	212.0	14.696	0.926	14.83	20.75
	27	Volume 27 - Level 4	6.50	220.63	33.86	30.88	212.0	14.696	0.926	15.23	21.32
	28	Volume 28 - Level 4	6.50	154.83	35.39	30.88	212.0	14.696	0.926	13.57	19.00
	29	Volume 29 - Level 5	6.50	93.88	17.33	37.38	212.0	14.696	0.926	69.59	97.43
	30	Volume 30 - Level 5	6.50	120.07	17.33	37.38	212.0	14.696	0.926	159.01	222.61
	31	Volume 31 - Level 5	6.50	194.00	17.33	37.38	212.0	14.696	0.926	16.14	22.60
	32	Volume 32 - Level 5	6.50	211.46	17.33	37.38	212.0	14.696	0.926	0.58	0.81
	33	Volume 33 - Level 5	6.50	209.67	17.33	37.38	212.0	14.696	0.926	8.56	11.98
	34	Volume 34 - Level 5	6.50	182.96	17.33	37.38	212.0	14.696	0.926	12.88	18.03
	35	Volume 35 - Level 5	6.50	134.20	17.33	37.38	212.0	14.696	0.926	21.52	30.13
	36	Volume 36 - Level 6	6.50	102.43	17.33	43.88	212.0	14.696	0.926	15.22	21.31
	37	Volume 37 - Level 6	6.50	157.45	17.33	43.88	212.0	14.696	0.926	12.95	18.13
	38	Volume 38 - Level 6	6.50	212.51	17.33	43.88	212.0	14.696	0.926	1.71	2.39

TABLE 6.2-13 (Continued)

Control Volume No.	Description	Height (ft)	Volume (ft ³)	Volume Flow Area (ft ²)	Bottom Elevation (ft)	Initial Conditions			Calculated Peak $\Delta P^{(1) (2)}$ (psid)	Biological Shield Wall Design Pressure ⁽²⁾ (psid)
						Temperature (°F)	Pressure (psia)	Quality		
39	Volume 39 - Level 6	6.50	220.01	17.33	43.88	212.0	14.696	0.926	0.00	0.00
40	Volume 40 - Level 6	6.50	219.86	17.33	43.88	212.0	14.696	0.926	4.68	6.55
41	Volume 41 - Level 6	6.50	212.51	17.33	43.88	212.0	14.696	0.926	15.49	21.69
42	Volume 42 - Level 6	6.50	155.98	17.33	43.88	212.0	14.696	0.926	21.30	29.82
43	Volume 43 - Level 7	6.50	102.43	16.93	50.38	212.0	14.696	0.926	0.02	0.0
44	Volume 44 - Level 7	6.50	142.39	25.39	50.38	212.0	14.696	0.926	0.00	0.0
45	Volume 45 - Level 7	6.50	204.82	33.86	50.38	212.0	14.696	0.926	0.00	0.0
46	Volume 46 - Level 7	6.50	212.32	33.86	50.38	212.0	14.696	0.926	0.40	0.56
47	Volume 47 - Level 7	6.50	219.86	33.86	50.38	212.0	14.696	0.926	0.12	0.17
48	Volume 48 - Level 7	6.50	186.82	33.86	50.38	212.0	14.696	0.926	0.00	0.0
49	Volume 49 - Level 7	6.50	148.79	25.39	50.38	212.0	14.696	0.926	0.00	0.0
50	Volume 50 - Drywell	90.0	278,000	3089.0	0.0	212.0	14.696	0.52	-	-

NOTES:

⁽¹⁾ With respect to drywell.

⁽²⁾ Biological shield wall design pressures are for time-step 0.500 seconds. A study was performed to determine which time-step pressures would cause maximum shear and moment at the base of the structure. The controlling time-step pressures (at 0.500 sec for feedwater break) were then used as the design loading case.

TABLE 6.2-14

DRYWELL HEAD REGION ANALYSIS

<u>Control Volume No.</u>	<u>Description</u>	<u>Volume (ft³)</u>	<u>Initial Conditions</u>			<u>Calculated Peak $\Delta P^{(1)}$ (psid)</u>	<u>Design Pressure (psid)</u>
			<u>Temperature (°F)</u>	<u>Pressure (psia)</u>	<u>Rel. Hum. (%)</u>		
1	Drywell Head	7,610	150	14.696	100	6.6	9.2
2	Drywell	2.70E+05	150	14.696	0	-	-

NOTE:

⁽¹⁾ Calculated peak pressure differentials are with respect to containment.

TABLE 6.2-15

STEAM TUNNEL

Control Volume No.	Description	Volume (ft ³)	Initial Conditions			Calculated Peak $\Delta P^{(1)}$ (psid)	Design Pressure (psid)	
			Temperature (°F)	Pressure (psia)	Rel. Hum. (%)			
1	Steam Tunnel Inside Contain- ment	8,950	144	14.696	100	0.88	1.6	
2	Containment	1.156E+06	120	14.696	100	-	-	

NOTE:

⁽¹⁾ Calculated peak pressure differentials are with respect to containment.

TABLE 6.2-16

REACTOR WATER CLEANUP ROOMS ANALYSIS

<u>Control Volume No.</u>	<u>Description</u>	<u>Initial Conditions</u>				<u>Calculated Peak $\Delta P^{(1)}$ (psid)</u>	<u>Design Pressure (psid)</u>	
		<u>Volume (ft³)</u>	<u>Temperature (°F)</u>	<u>Pressure (psia)</u>	<u>Rel. Hum. (%)</u>			
1	RWCU Heat Exchanger Room	12,600	120	14.696	100.0	8.05	10.9	
2	Corridor	412	120	14.696	100.0	-	-	
3	Containment	1.156+6	120	14.696	100.0	-	-	
1	RWCU Filter Demineralizer Valve Room	6,580	120	14.696	100.0	7.18	9.3	
2	Corridor	302	120	14.696	100.0	-	-	
3	Containment	1.156+6	120	14.696	100.0	-	-	
1	RWCU Filter Demineralizer Room	3,430	120	14.696	100.0	4.89	6.3	
2	Containment	1.156+6	120	14.696	100.0	-	-	
1	RWCU Drain Valve Nest Room	2,915	120	14.696	100.0	8.33	10.8	
2	Corridor	179	120	14.696	100.0	-	-	
3	Containment	1.156+6	120	14.696	100.0	-	-	

NOTE:

⁽¹⁾ Calculated peak pressure differentials are with respect to containment.

TABLE 6.2-16a

DRYWELL BULKHEAD PLATE ΔP ANALYSIS
VOLUME DESCRIPTION

<u>Control Volume No.</u>	<u>Description</u>	<u>Volume (ft³)</u>	<u>Initial Conditions</u>			<u>Calculated Peak $\Delta P^{(1)}$ (psid)</u>	<u>Design Pressure (psid)</u>
			<u>Temperature (°F)</u>	<u>Pressure (psia)</u>	<u>Rel. Hum. (%)</u>		
1	252° to 288°	365	145	14.696	50	28.0	30.0
2	217° to 252°	364	145	14.696	50	27.8	30.0
3	288° to 72°	1,499	145	14.696	50	4.0	30.0
4	72° to 217°	1,491	145	14.696	50	4.0	30.0
5	Head Region	7,610	145	14.696	50		
6	Inside Bio-Wall	10,367	145	14.696	50		
7	Remainder of Drywell	255,990	145	14.696	50		

NOTE:

⁽¹⁾ Differential pressure across bulkhead plate.

TABLE 6.2-17

MASS AND ENERGY RELEASES

a. From Recirculation Suction Line Break With One Inch Flow Diverter

<u>Time (sec)</u>	<u>Into Annulus (lbm/sec)</u>	<u>Into Drywell (lbm/sec)</u>	<u>Enthalpy (Btu/lbm)</u>
0	2,642.6	1,570.2	533
1.0	2,638.2	1,567.6	533
2.0	2,635.2	1,568.2	533

b. Into Annulus from Recirculation Discharge Break

<u>Time (Sec)</u>	<u>Mass Flow Rate (lbm/sec)</u>	<u>Enthalpy (Btu/lbm)</u>
0.0	3,802.5	551.5
2.0	3,802.5	551.5

TABLE 6.2-18

MASS AND ENERGY INTO REACTOR ANNULUS DUE
TO FEEDWATER BREAK USED FOR ANALYSIS

<u>Time (Sec)</u>	<u>Mass (lbm/sec)</u>	<u>Enthalphy (Btu/lbm)</u>
0.0	5,515.0	497.4
0.00075	11,575.0	497.4
0.00175	9,833.3	497.4
0.00875	9,050.0	497.4
0.01105	9,270.0	497.4
0.0401	11,023.3	497.4
0.0521	12,421.7	497.4
0.0761	12,421.7	497.4
0.1001	13,423.3	497.4
0.1961	12,613.3	497.4
0.4941	13,596.7	497.4
0.7376	13,880.0	497.4
1.0	13,240.0	497.4
2.0	13,240.0	497.4

TABLE 6.2-19

REACTOR WATER CLEANUP SYSTEMDOUBLE ENDED PIPE BREAK⁽¹⁾ LIQUID BLOWDOWN

<u>Time (sec)</u>	<u>Total Flow Rate (lbm/sec)</u>	<u>Enthalpy (Btu/lbm)</u>
0	582	528.53
1.34	1163	528.53
18	1163	528.53
20	1018	528.35
21	945	528.35
22	872	528.35
23	800	528.35
24	727	528.35
25	654	528.35
26	582	528.35
28	436	528.35
30	291	528.35
32	145	528.35
34	0	-

NOTE:

⁽¹⁾ Suction line pipe size is 6-inch Schedule 80; discharge line pipe size is 4-inch Schedule 80.

TABLE 6.2-20

REACTOR ANNULUS RECIRCULATION LINE BREAK

Junction No.	From CV	To CV	Elevation (ft)	Minimum Flow Area (ft ²)	Inertia (L/A) (ft ⁻¹)	Head Loss, K				Total ⁽¹⁾
						Bends	Friction (fL/D)	Expansion	Contrac- tion	
1	1	2	9.74	12.03	0.558	0.15	0.0235	-	-	0.20
2	1	25	12.64	14.23	0.544	-	0.0226	-	-	0.05
3	2	3	11.19	20.45	0.324	0.15	0.0235	-	-	0.20
4	2	7	15.59	0.10	0.639	-	0.0226	-	-	0.05
5	2	25	14.09	0.01	0.820	0.15	0.0235	-	-	0.20
6	3	4	11.19	14.89	0.580	0.15	0.0235	-	-	0.20
7	3	8	15.54	3.57	0.592	-	0.0226	-	-	0.05
8	4	5	11.19	18.09	0.461	0.15	0.0235	-	-	0.20
9	4	9	15.54	0.10	0.457	-	0.0226	-	-	0.05
10	5	10	15.54	19.64	0.343	-	0.0226	-	-	0.05
11	6	7	21.35	9.88	0.629	0.15	0.0235	-	-	0.20
12	6	11	24.25	11.64	0.710	-	0.0226	-	-	0.05
13	6	25	18.44	11.64	0.706	-	0.0226	-	-	0.05
14	7	8	19.90	7.54	0.392	0.15	0.0235	-	-	0.20
15	7	12	24.25	14.38	0.695	-	0.0226	-	-	0.05
16	7	25	16.99	0.01	0.820	0.15	0.0235	-	-	0.20
17	8	9	19.90	9.52	0.673	0.15	0.0235	-	-	0.20
18	8	13	24.25	14.38	0.695	-	0.0226	-	-	0.05
19	9	10	19.90	10.74	0.925	0.15	0.0235	-	-	0.20
20	9	14	24.25	20.30	0.485	-	0.0226	-	-	0.05
21	10	15	24.25	24.79	0.307	-	0.0226	-	-	0.05
22	11	12	27.69	15.03	0.518	0.15	0.0235	-	-	0.20
23	11	16	31.125	3.14	0.588	-	0.0226	-	-	0.05
24	12	13	27.69	11.74	0.427	0.15	0.0235	-	-	0.20
25	12	16	31.125	5.41	0.406	-	0.0226	-	-	0.05
26	12	17	31.125	0.62	0.390	-	0.0226	-	-	0.05
27	13	14	27.69	15.03	0.776	0.15	0.0235	-	-	0.20
28	13	17	31.125	12.14	0.390	-	0.0226	-	-	0.05

TABLE 6.2-20 (Continued)

Junction No.	From CV	To CV	Elevation (ft)	Minimum Flow Area (ft ²)	Inertia (L/A) (ft ⁻¹)	Head Loss, K				
						Bends	Friction (fL/D)	Expansion	Contraction	Total ⁽¹⁾
29	14	15	27.69	15.33	0.519	0.15	0.0235	-	-	0.20
30	14	18	31.125	15.81	0.310	-	0.0226	-	-	0.05
31	15	19	31.125	24.35	0.339	-	0.0226	-	-	0.05
32	16	17	34.59	3.04	0.618	0.15	0.0235	-	-	0.20
33	16	20	38.05	0.10	1.328	-	0.0226	-	-	0.05
34	17	18	34.59	17.42	0.774	0.15	0.0235	-	-	0.20
35	17	21	38.05	12.24	0.882	-	0.0226	-	-	0.05
36	18	19	34.59	14.53	0.580	0.15	0.0235	-	-	0.20
37	18	22	38.05	8.58	0.970	-	0.0226	-	-	0.05
38	19	23	38.05	20.06	0.933	-	0.0226	-	-	0.05
39	20	21	44.885	19.71	0.494	0.15	0.0235	-	-	0.20
40	20	26	51.71	6.73	0.397	-	0.0226	-	-	0.05
41	21	22	44.885	16.73	0.454	0.15	0.0235	-	-	0.20
42	21	27	51.71	12.39	0.565	-	0.0226	-	-	0.05
43	22	23	44.885	26.29	0.473	0.15	0.0235	-	-	0.20
44	22	28	51.71	5.36	0.565	-	0.0226	-	-	0.05
45	23	29	51.71	15.80	0.397	-	0.0226	-	-	0.05
46	26	27	54.96	17.33	0.549	0.15	0.0235	-	-	0.20
47 ⁽²⁾	26	24	54.96	70.41	0.036	-	-	1.0	-	1.00
48	27	28	54.96	5.77	1.209	0.15	0.0235	-	-	0.20
49 ⁽²⁾	27	24	54.96	66.72	0.037	-	-	1.0	-	1.00
50	28	29	54.96	17.33	0.549	0.15	0.0235	-	-	0.20
51 ⁽²⁾	28	24	54.96	66.72	0.037	-	-	1.0	-	1.00
52 ⁽²⁾	29	24	54.96	70.41	0.036	-	-	1.0	-	1.00
53	0	25	15.44	1.00	0.0	-	-	-	-	0.0
54	0	24	15.54	1.00	0.0	-	-	-	-	0.0

NOTES:

⁽¹⁾ All horizontal flow paths are assumed to have a K of 0.20; all vertical flow paths are assumed to have a K of 0.05.

⁽²⁾ Air seals rupture at 15 psid.

TABLE 6.2-21

REACTOR ANNULUS
FEEDWATER LINE BREAK

Junction No.	From CV	To CV	Elevation (ft)	Minimum Flow Area (ft ²)	Inertia (L/A) (ft ⁻¹)	Head Loss, K				
						Bends	Friction (fL/D)	Expansion	Contraction	Total ⁽¹⁾
1	1	2	10.22	18.43	0.701	0.084	0.019	-	-	0.20
2	1	7	10.22	13.94	0.777	0.084	0.019	-	-	0.20
3	1	8	14.94	0.01	2.125	-	0.0244	-	-	0.05
4	2	3	10.22	16.01	0.875	0.112	0.026	-	-	0.20
5	2	9	14.94	0.01	0.820	-	0.0199	-	-	0.05
6	3	4	10.22	14.49	1.732	0.112	0.029	-	-	0.20
7	3	10	14.94	0.01	0.593	-	0.0173	-	-	0.05
8	4	5	10.22	19.97	1.258	0.112	0.029	-	-	0.20
9	4	11	14.94	12.26	0.592	-	0.0173	-	-	0.05
10	5	6	10.22	19.67	1.387	0.112	0.029	-	-	0.20
11	5	12	14.94	14.43	0.675	-	0.0173	-	-	0.05
12	6	7	10.22	14.50	1.132	0.112	0.026	-	-	0.20
13	6	13	14.94	0.01	0.597	-	0.0173	-	-	0.05
14	7	14	14.94	0.01	1.086	-	0.0199	-	-	0.05
15	8	9	19.66	8.01	0.968	0.084	0.019	-	-	0.20
16	8	14	19.66	7.36	0.875	0.084	0.019	-	-	0.20
17	8	15	24.38	13.27	0.471	-	0.0206	-	-	0.05
18	9	10	19.66	8.68	1.241	0.112	0.026	-	-	0.20
19	9	16	24.38	22.39	0.341	-	0.0168	-	-	0.05
20	10	11	19.66	8.68	1.870	0.112	0.029	-	-	0.20
21	10	17	24.38	30.86	0.271	-	0.0146	-	-	0.05
22	11	12	19.66	18.02	1.628	0.112	0.029	-	-	0.20
23	11	18	24.38	33.53	0.235	-	0.0146	-	-	0.05
24	12	13	19.66	8.01	2.248	0.112	0.029	-	-	0.20
25	12	19	24.38	33.20	0.235	-	0.0146	-	-	0.05
26	13	14	19.66	7.35	1.581	0.112	0.026	-	-	0.20
27	13	20	24.38	25.85	0.275	-	0.0146	-	-	0.05
28	14	21	24.38	16.72	0.374	-	0.0168	-	-	0.05

TABLE 6.2-21 (Continued)

Junction No.	From CV	To CV	Elevation (ft)	Minimum Flow Area (ft ²)	Inertia (L/A) (ft ⁻¹)	Head Loss, K				
						Bends	Friction (fL/D)	Expansion	Contraction	Total ⁽¹⁾
29	15	16	27.63	5.93	0.679	0.084	0.023	-	-	0.20
30	15	21	27.63	9.02	0.511	0.084	0.023	-	-	0.20
31	15	22	30.88	7.92	0.689	-	0.0168	-	-	0.05
32	16	17	27.63	5.93	1.310	0.112	0.031	-	-	0.20
33	16	23	30.88	11.98	0.614	-	0.0137	-	-	0.05
34	17	18	27.63	9.35	3.144	0.112	0.035	-	-	0.20
35	17	24	30.88	21.08	0.253	-	0.0119	-	-	0.05
36	18	19	27.63	15.89	1.553	0.112	0.035	-	-	0.20
37	18	25	30.88	33.19	0.243	-	0.0119	-	-	0.05
38	19	20	27.63	7.91	10.548	0.112	0.035	-	-	0.20
39	19	26	30.88	31.64	0.355	-	0.0119	-	-	0.05
40	20	21	27.63	0.71	7.449	0.112	0.031	-	-	0.20
41	20	27	30.88	22.74	0.260	-	0.0119	-	-	0.05
42	21	28	30.88	16.65	0.358	-	0.0137	-	-	0.05
43	22	23	34.13	11.50	0.941	0.084	0.023	-	-	0.20
44	22	28	34.13	6.64	0.666	0.084	0.023	-	-	0.20
45	22	29	37.38	8.59	0.828	-	0.0168	-	-	0.05
46	23	24	34.13	6.71	2.192	0.112	0.031	-	-	0.20
47	23	30	37.38	0.01	0.655	-	0.0137	-	-	0.05
48	24	25	34.13	12.55	0.796	0.112	0.035	-	-	0.20
49	24	31	37.38	14.70	0.660	-	0.0119	-	-	0.05
50	25	26	34.13	17.33	0.796	0.112	0.035	-	-	0.20
51	25	32	37.38	28.41	0.493	-	0.0119	-	-	0.05
52	26	27	34.13	7.77	1.259	0.112	0.035	-	-	0.20
53	26	33	37.38	27.41	0.565	-	0.0119	-	-	0.05
54	27	28	34.13	17.33	0.703	0.112	0.031	-	-	0.20
55	27	34	37.38	11.97	0.703	-	0.0119	-	-	0.05
56	28	35	37.38	11.48	0.782	-	0.0137	-	-	0.05
57	29	30	40.63	5.87	0.701	0.084	0.023	-	-	0.20
58	29	35	40.63	12.55	0.943	0.084	0.023	-	-	0.20

TABLE 6.2-21 (Continued)

Junction No.	From CV	To CV	Elevation (ft)	Minimum Flow Area (ft ²)	Inertia (L/A) (ft ⁻¹)	Head Loss, K				
						Bends	Friction (fL/D)	Expansion	Contraction	Total ⁽¹⁾
59	29	36	43.88	13.37	0.459	-	0.0168	-	-	0.05
60	30	31	40.63	0.01	1.382	0.112	0.031	-	-	0.20
61	30	37	43.88	1.34	0.406	-	0.0137	-	-	0.05
62	31	32	40.63	4.00	1.793	0.112	0.035	-	-	0.20
63	31	38	43.88	21.49	0.422	-	0.0119	-	-	0.05
64	32	33	40.63	13.00	1.623	0.112	0.035	-	-	0.20
65	32	39	43.88	28.41	0.422	-	0.0119	-	-	0.05
66	33	34	40.63	2.33	2.052	0.112	0.035	-	-	0.20
67	33	40	43.88	27.85	0.482	-	0.0119	-	-	0.05
68	34	35	40.63	0.99	1.602	0.112	0.031	-	-	0.20
69	34	41	43.88	18.76	0.433	-	0.0119	-	-	0.05
70	35	42	43.88	15.88	0.419	-	0.0137	-	-	0.05
71	36	37	47.13	11.90	0.663	0.084	0.023	-	-	0.20
72	36	42	47.13	15.44	0.511	0.084	0.023	-	-	0.20
73	36	43	50.38	10.53	0.479	-	0.0168	-	-	0.05
74	37	38	47.13	11.90	1.198	0.112	0.031	-	-	0.20
75	37	44	50.38	8.31	0.549	-	0.0137	-	-	0.05
76	38	39	47.13	13.33	1.149	0.112	0.035	-	-	0.20
77	38	45	50.38	23.75	0.234	-	0.0119	-	-	0.05
78	39	40	47.13	16.67	1.543	0.112	0.035	-	-	0.20
79	39	46	50.38	30.41	0.338	-	0.0119	-	-	0.05
80	40	41	47.13	15.00	1.641	0.112	0.035	-	-	0.20
81	40	47	50.38	31.19	0.312	-	0.0119	-	-	0.05
82	41	42	47.13	13.55	1.056	0.112	0.031	-	-	0.20
83	41	48	50.38	19.75	1.222	-	0.0119	-	-	0.05
84	42	49	50.38	15.06	0.322	-	0.0137	-	-	0.05
85	43	44	53.63	1.66	1.345	0.084	0.023	-	-	0.20
86	43	49	53.63	11.37	0.511	0.084	0.023	-	-	0.20
87 ⁽²⁾	43	50	53.63	38.17	0.054	-	-	1.00	-	1.00
88	44	45	53.63	1.66	1.172	0.112	0.031	-	-	0.20

TABLE 6.2-21 (Continued)

Junction No.	From CV	To CV	Elevation (ft)	Minimum Flow Area (ft ²)	Inertia (L/A) (ft ⁻¹)	Head Loss, K				
						Bends	Friction (fL/D)	Expansion	Contraction	Total ⁽¹⁾
89 ⁽²⁾	44	50	53.63	53.54	0.043	-	-	1.00	-	1.00
90	45	46	53.63	2.62	2.036	0.112	0.035	-	-	0.20
91 ⁽²⁾	45	50	53.63	83.11	0.033	-	-	1.00	-	1.00
92	46	47	53.63	14.11	1.543	0.112	0.035	-	-	0.20
93 ⁽²⁾	46	50	53.63	88.59	0.032	-	-	1.00	-	1.00
94	47	48	53.63	10.92	2.386	0.112	0.035	-	-	0.20
95 ⁽²⁾	47	50	53.63	91.34	0.031	-	-	1.00	-	1.00
96	48	49	53.63	0.01	3.555	0.112	0.031	-	-	0.20
97 ⁽²⁾	48	50	53.63	72.09	0.035	-	-	1.00	-	1.00
98 ⁽²⁾	49	50	53.63	56.70	0.041	-	-	1.00	-	1.00
99	0	29	40.69	1.00	0.0	-	-	-	-	0.0

NOTES:

⁽¹⁾ All horizontal flow paths are assumed to have a K of 0.20.

All vertical flow paths are assumed to have a K of 0.05.

⁽²⁾ Air seals rupture at 15.0 psid.

TABLE 6.2-22

DRYWELL HEAD FLOW PATH DATA

Junc- tion No.	From CV	To CV	Minimum Flow Area (ft ²)	Inertia L/A (ft ⁻¹)	Entrance Head Losses, K ⁽¹⁾					Exit Head Losses, K ⁽¹⁾				
					Bends	Fric- tion (f L/D)	Expan- sion	Contrac- tion	Total	Bends	Fric- tion (f L/D)	Expan- sion	Contrac- tion	Total
1	1	2	13.1	0.77	-	-	-	0.5	0.5	-	0.02	1.0	-	1.02

NOTE:

⁽¹⁾ Head loss terms are with respect to minimum flow area.

TABLE 6.2-23

REACTOR WATER CLEANUP ROOMS FLOW PATH DATA

Junc- tion No. ⁽²⁾	From CV	To CV	Minimum Flow Area (ft ²)	Inertia L/A (ft ⁻¹)	Entrance Head Losses, K ⁽¹⁾					Exit Head Losses, K ⁽¹⁾				
					Bends	Fric- tion (f L/D)	Expan- sion	Contrac- tion	Total	Bends	Fric- tion (f L/D)	Expan- sion	Contrac- tion	Total
a. 1	1	2	16.25	0.444	0.000	0.000	0.000	0.500	0.500	0.497	0.021	0.080	0.000	0.598
2	2	3	16.93	0.356	0.683	0.023	0.000	0.225	0.931	0.000	0.000	1.000	0.000	1.000
b. 1	1	2	16.25	0.247	0.000	0.000	0.000	0.500	0.500	0.390	0.007	0.120	0.000	0.517
2	2	3	24.43	0.175	0.000	0.016	0.000	0.000	0.016	0.882	0.000	1.000	0.150	2.032
c. 1	1	2	17.7	0.258	0.300	0.029	0.000	0.500	0.829	0.000	0.000	1.000	0.000	1.000
d. 1	1	2	16.93	0.717	0.780	0.025	0.000	0.189	0.994	0.000	0.012	0.000	0.195	0.207
2	2	3	16.25	0.191	0.719	0.012	0.000	0.000	0.731	0.000	0.000	1.000	0.051	1.051

NOTES:

⁽¹⁾ Head loss terms are with respect to minimum flow area.

⁽²⁾ RWCU rooms are as follows:

- a. Heat Exchanger Room
- b. Filter Demineralizer Valve Room
- c. Filter Demineralizer Room
- d. Drain Valve Nest Room

TABLE 6.2-24

STEAM TUNNEL FLOW PATH DATA

Junc- tion No.	From CV	To CV	Minimum Flow Area (ft ²)	Inertia L/A (ft ⁻¹)	Entrance Head Losses, K ⁽¹⁾					Exit Head Losses, K ⁽¹⁾				
					Bends	Fric- tion (f L/D)	Expan- sion	Contrac- tion	Total	Bends	Fric- tion (f L/D)	Expan- sion	Contrac- tion	Total
1	1	2	16.76	0.92	1.53	0.06	-	0.51	2.1	-	-	1.0	-	1.0
2	1	2	25.8	0.24	-	-	-	0.5	0.5	-	0.16	1.0	-	1.16

NOTE:

⁽¹⁾ Head loss terms are with respect to minimum flow area.

TABLE 6.2-24a

DRYWELL BULKHEAD PLATE ΔP ANALYSIS FLOW PATH DESCRIPTION

Junction No.	From CV	To CV	Minimum Flow Area (ft ²)	Inertia L/A (ft ⁻¹)	Forward Direction		Reverse Direction	
					Entrance Loss (K)	Exit Loss (K)	Entrance Loss (K)	Exit Loss (K)
1	1	2	27.8	0.234	0.173	0.086	0.173	0.086
2	1	3	27.8	0.550	0.202	0.086	0.202	0.086
3	1	6	14.5	0.992	0.365	0.053	0.152	0.232
4	1	7	37.9	0.126	0.007	0.401	0.324	0.0
5	2	4	39.4	0.536	0.091	0.0	0.091	0.0
6	2	6	19.55	0.983	0.365	0.053	0.155	0.229
7	2	7	38.4	0.128	0.007	0.398	0.323	0.0
8	3	4	27.8	0.887	0.339	0.176	0.339	0.176
9	3	5	5.17	0.285	0.493	0.923	0.501	0.893
10	3	6	97.0	0.244	0.286	0.0	0.076	0.177
11	3	7	197.8	0.031	0.007	0.421	0.331	0.0
12	4	5	7.4	0.344	0.494	0.939	0.499	0.915
13	4	6	97.0	0.244	0.304	0.006	0.094	0.183
14	4	7	192.4	0.031	0.416	0.007	0.330	0.0

TABLE 6.2-25

REACTOR BLOWDOWN DATA FOR RECIRCULATION SUCTION LINE BREAK
(at 3729 MWt)

<u>Time</u> <u>(sec)</u>	<u>Reactor</u> <u>Vessel</u> <u>Pressure</u> <u>(psia)</u>	<u>Liquid</u> <u>Flow</u> <u>(lbs/sec)</u>	<u>Liquid</u> <u>Enthalpy</u> <u>(Btu/lb)</u>	<u>Steam</u> <u>Flow</u> <u>(lbs/sec)</u>	<u>Steam</u> <u>Enthalpy</u> <u>(Btu/lb)</u>
0	1060.	19380.	530.0		
0.003	1060.	29070.	530.0		
1.00	1050.	28460.	530.0	---	---
2.00	1039.	24220.	530.0	---	---
2.01	1039.	22202.	549.0	---	---
2.50	1036.	21990.	548.0	---	---
3.06	1036.	21990.	548.0	---	---
6.06	1062.	22240.	551.9	---	---
8.00	1075.	22370.	553.9	---	---
10.87	1081.	22430.	554.8	---	---
19.29	1021.	21850.	545.8	---	---
19.30	1020.	11490.	545.8	3025.	1192.1
24.66	767.	9192.	503.9	2457.	1200.3
26.04	712.	8705.	493.8	2318.	1201.6
30.73	550.	7312.	461.0	1856.	1204.3
35.73	413.	6398.	427.6	1399.	1204.7
40.10	330.	5807.	403.7	1106.	1203.6
50.45	193.	5006.	352.4	598.3	1197.9
60.45	126.	4815.	316.7	330.3	1191.3
70.95	95.8	4842.	295.4	204.8	1186.4
100.65	64.8	4736.	267.4	82.2	1179.1
149.65	54.7	4756.	256.1	36.9	1175.8
200.65	54.8	4892.	256.2	23.5	1165.9
300.25	40.9	4168.	237.5	7.6	1170.2
375.91	31.2	65.0	221.2	0.06	1164.9
375.94	31.2	---	---	---	---

TABLE 6.2-26

REACTOR BLOWDOWN DATA FOR MAIN STEAM
LINE BREAK - Minimum ECCS
 (at 3729 MWt)

<u>Time</u> <u>(sec)</u>	<u>Reactor</u> <u>Vessel</u> <u>Pressure</u> <u>(psia)</u>	<u>Liquid</u> <u>Flow</u> <u>(lbs/sec)</u>	<u>Liquid</u> <u>Enthalpy</u> <u>(Btu/lb)</u>	<u>Steam</u> <u>Flow</u> <u>(lbs/sec)</u>	<u>Steam</u> <u>Enthalpy</u> <u>(Btu/lb)</u>
0	1060.	0	551.6	9850.	1190.7
0.05	1056.	0	551.0	11440.	1190.8
0.22	1047.	0	549.7	8259.	1191.1
0.999	1018.	0	545.4	8025.	1192.3
1.000	1018.	27000.	545.3	1094.	1192.3
2.01	1017.	26670.	545.2	1200.	1192.3
4.01	1011.	25890.	544.2	1419.	1192.5
6.00	1007.	19730.	543.6	1273.	1192.7
8.03	1001.	19130.	542.7	1425.	1192.9
10.03	985.	18430.	540.2	1560.	1193.5
15.03	908.	16340.	528.0	1805.	1196.1
20.03	789.	14040.	507.8	1880.	1199.7
25.03	658.	11810.	492.7	1794.	1201.7
30.03	542.	9877.	492.3	1634.	1201.8
50.06	202.	5769.	356.5	652.6	1198.5
70.06	96.1	5132.	295.7	230.6	1186.5
90.15	72.0	5083.	274.7	125.7	1181.1
100.9	62.5	4984.	265.0	91.0	1178.4
119.9	56.2	4980.	257.8	62.9	1176.4
140.4	53.8	5022.	254.9	47.7	1175.5
200.4	53.8	5213.	255.0	29.1	1175.5
300.3	40.2	4450.	236.5	10.2	1170.9
350.9	30.9	1207.	220.7	1.6	1164.7
359.9	30.2	112.3	219.3	0.14	1164.3

TABLE 6.2-27

CORE DECAY HEAT FOLLOWING LOSS-OF-COOLANT
ACCIDENT FOR CONTAINMENT ANALYSIS
 (at 3729 MWt)

<u>Time (sec)</u>	<u>Normalized Core Heat⁽¹⁾</u>
0	1.084
2.0	0.5574
6.0	0.5491
10.0	0.3856
20.0	0.125
60.0	0.04692
100.0	0.0426
120.0	0.041
200.0	0.0358
800.0	0.026
1,000.0	0.0245
8,000.0	0.013
20,000.0	0.0101
60,000.0	0.00739
200,000.0	0.00512
600,000.0	0.0036
2,000,000.0	0.00237

NOTE:

⁽¹⁾ Normalized to 3729 MWt; includes fuel relaxation energy.

TABLE 6.2-27a

CORE DECAY HEAT FOLLOWING LOSS-OF-COOLANT
ACCIDENT FOR LONG-TERM CONTAINMENT ANALYSIS
 (at 3833 MWt)

<u>Time (sec)</u>	<u>Core Heat⁽¹⁾</u>
0	1.0000
2.0	0.1554
4.0	0.07437
10.0	0.05306
20.0	0.04611
60.0	0.03741
100.0	0.03373
150.0	0.03124
200.0	0.02960
800.0	0.02249
1,000.0	0.02130
8,000.0	0.01162
20,000.0	0.009216
60,000.0	0.007025
100,000.0	0.006116

NOTE:

- ⁽¹⁾ Using decay heat standard ANSI/ANS 5.1-1979 and two sigma. Does not include heat from metal-water reactions or sensible heat stored in fuel and structure.

TABLE 6.2-28

ADDITIONAL INFORMATION TO BE PROVIDED FOR
DUAL CONTAINMENT PLANTS

I. Secondary Containment Design

A.	Free volume @ 106°F, ft ³	392,548	
B.	Pressure, inches water gauge	-0.25	
C.	Leak rate at postaccident pressure, %/day	100	
D.	Exhaust fans		
1.	Number	2	
2.	Type	Centrifugal	
E.	Filters		
1.	Number	2	
2.	Type	Charcoal filter train consisting of a demister, roughing filter, electric heating coil, HEPA filters, charcoal filters (4 inches deep), and HEPA filter.	

II. Transient Analysis

A.	Initial annulus conditions		
1.	Pressure, psia	14.682	
2.	Temperature, °F	106	
3.	Outside air temperature, °F	95	
4.	Thickness of secondary containment wall, inches	36	
5.	Thickness of primary containment wall, inches	1.5	

TABLE 6.2-28 (Continued)

B. Thermal characteristics

1.	Primary containment wall	
a.	Coefficient of linear expansion, in/in-°F at 185°F	6.34×10^{-6}
b.	Modulus of elasticity, psi at 185°F	27.7×10^6
c.	Thermal conductivity, Btu/hr-ft-°F	26
d.	Thermal capacitance, Btu/ft ³ -°F	51
2.	Secondary containment wall	
a.	Thermal conductivity, Btu/hr-ft-°F	0.8
b.	Thermal capacitance, Btu/ft ³ -°F	30
3.	Heat transfer coefficients	
a.	Primary containment atmosphere to primary containment wall, Btu/hr-ft ² -°F _{absolute}	<Table 6.2-31>
b.	Primary containment wall to secondary containment atmosphere, Btu/hr-ft ² -°F _{absolute}	1.2
c.	Secondary containment wall to secondary containment atmosphere, Btu/hr-ft ² -°F _{absolute}	1.2

TABLE 6.2-29

DELETED

|

TABLE 6.2-30

DELETED

|

TABLE 6.2-31

OVERALL HEAT TRANSFER COEFFICIENTS
AS A FUNCTION OF TIME⁽¹⁾

<u>Time</u> <u>(hours)</u>	<u>ΔT</u> <u>(°F)</u>	<u>h_c</u> <u>Btu/hr-ft²-°F</u>	<u>h_r</u> <u>Btu/hr-ft²-°F</u>	<u>h_T</u> <u>Btu/hr-ft²-°F</u>
0.0	0.0	0.0	0.0	0.0
2.788E-04	5.0	0.449	0.808	1.257
5.556E-04	10.0	0.533	0.819	1.372
8.333E-04	15.0	0.590	0.830	1.420
1.389E-03	25.0	0.671	0.853	1.524
1.667E-03	30.0	0.702	0.865	1.567
2.778E-03	33.1	0.720	0.872	1.592
4.167E-03	36.9	0.739	0.881	1.620
5.556E-03	40.8	0.758	0.890	1.648
8.333E-03	48.5	0.792	0.909	1.701
1.250E-02	60.0	0.835	0.937	1.772
1.389E-02	60.3	0.836	0.938	1.774
2.778E-02	63.5	0.847	0.946	1.793
5.556E-02	69.8	0.867	0.962	1.829
8.333E-02	76.1	0.886	0.978	1.864
1.111E-01	82.4	0.904	0.995	1.899
1.389E-01	88.7	0.921	1.012	1.933
1.677E-01	95.0	0.937	1.028	1.965
2.778E-01	95.0	0.937	1.028	1.965
24.0	95.0	0.937	1.028	1.965

NOTE:

- ⁽¹⁾ Values support original analysis. Changes to input parameters affecting these values are reviewed to ensure analysis results remain valid.

TABLE 6.2-32

CONTAINMENT ISOLATION VALVE SUMMARY^{(1) (2)}

Penetration No. ⁽³⁾	GDC/ Reg. Guide	System Number	Fluid	Line Size (in)	Essential System ⁽⁴⁾	Figure 6.2-60 Arr. No.	System and Valve Number	Loc ⁽⁵⁾	Type C Test	Pipe Length ⁽⁶⁾	Valve		Actuation Mode		Valve Position				Isolation Signal ⁽⁸⁾	Closure Time (sec) ⁽⁹⁾	Power Source 1E Bus ⁽¹⁰⁾	Norm. Flow Dir.
											Type	Oper.	Pri.	Sec.	Norm	Shut down	Post Acc.	Power Fail ⁽⁷⁾				
P101	GDC56	RCIC Pump Suction	Water	6	Yes ⁽²³⁾	21	E51F031	O	No ⁽²⁷⁾	20'-8"	Gate	EM	E	M	CL	CL	OP or CL	AI	RM _s , T, J, K, M, F, V, Q	30	1	Out
P102	GDC56, RG1.11	RHR A Pump Suction Suppression Pool Makeup	Water	24	Yes	13	E12F004A	O	No ⁽²⁷⁾	18'-4"	Gate	EM	E	M	OP	CL	OP	AI	RM _s	Std.	1	Out
			Water	3/4	Yes	55	G43F050A	O	No	<10"	Globe	S	E	-	OP	OP	OP	AI	RM _c	<3	1	-
P103	GDC56	LPCS Pump Suction	Water	24	Yes	6	E21F001	O	No ⁽²⁷⁾	14'-7"	Gate	EM	E	M	OP	OP	OP	AI	RM _c	Std.	1	Out
P104	GDC56	RCIC Pump Discharge Min. Flow to Sup. Pool	Water	2	Yes ⁽²³⁾	18	E51F019	O	Yes	NA	Globe	EM	E	M	CL	CL	OP or CL	AI	RM _s , BB, FF, EE	8	1	In
P105	GDC56	RHR A Min. Flow & Test; LPCS Pump Min. Flow & Test; RHR Heat Exchanger Dump to Suppression Pool Suppression Pool Cleanup Return	Water	18	Yes	15	E12F024A	O	No ⁽²⁷⁾	15'-9"	Globe	EM	E	M	CL	CL	OP or CL	AI	C, G, RM _s , AA	100	1	In
			Water	4	Yes	15	E12F011A	O	No ⁽²⁷⁾	18'-9"	Globe	EM	E	M	CL	CL	CL	AI	C, G, RM _s	Std.	1	In
			Water	12	Yes	15	E21F012	O	No ⁽²⁷⁾	21'-9"	Globe	EM	E	M	CL	CL	CL	AI	C, G, RM _c	Std.	1	In
			Water	4	Yes	15	E21F011	O	No ⁽²⁷⁾	24'-0"	Gate	EM	E	M	OP	OP	OP or CL	AI	RM _c , BB	23.5	1	In
			Water	6	Yes	15	E12F064A	O	No ⁽²⁷⁾	77'-7"	Gate	EM	E	M	OP	OP	OP or CL	AI	RM _s , BB	15	1	In
			Water	6	No	56	E12F609	O	No ⁽²⁷⁾		Gate	EM	E	M	CL	CL	CL	AI	B, G, RM _c	Std.	1	In
			Water	6	No	56	E12F610	O	No ⁽²⁷⁾		Gate	EM	E	M	CL	CL	CL	AI	B, G, RM _c	Std.	2	In
			Cont. Atmos.	18	Yes	15	E12D003A	I	No ⁽¹⁶⁾	NA	-	-	-	-	-	-	-	-	-	-	-	-
P106	GDC56	RCIC Turbine Exhaust	Cont. Atmos.	12	Yes ⁽²³⁾	10 (b)	E51F068	O	Yes	13'-0"	Gate	EM	E	M	OP	OP	CL	AI	RM _s , J, G	65	1	In
			Cont. Atmos.	1-1/2	Yes	10 (a)	E12F102	O	Yes	48'-8"	Globe	M	M	-	LC	LC	LC	-	-	-	-	-
			Cont. Atmos.	10	Yes ⁽²³⁾	10 (b)	E51F040	O	Yes	NA	Noz.Chk	-	M	-	CL	CL	OP or CL	-	-	-	-	In
			Water	2	Yes	37	E12F025A	O	Yes	NA	Rel	P	P	-	CL	CL	CL	AI	-	-	-	In
			Water	2	Yes	57	E12F025B	O	Yes	NA	Rel	P	P	-	CL	CL	CL	AI	-	-	-	In
			Water	2	Yes	57	E12F025C	O	Yes	NA	Rel	P	P	-	CL	CL	CL	AI	-	-	-	In
			Water	6	Yes	37	E12F055A	O	Yes	NA	Rel	P	P	-	CL	CL	CL	AI	-	-	-	In
			Water	6	Yes	57	E12F055B	O	Yes	NA	Rel	P	P	-	CL	CL	CL	AI	-	-	-	In
			Water	1	Yes	57	E12F005	O	Yes ⁽¹⁶⁾	NA	Rel	P	P	-	CL	CL	CL	AI	-	-	-	In
			Water	2	Yes	37	E21F018	O	Yes	NA	Rel	P	P	-	CL	CL	CL	AI	-	-	-	In
			Water	1	Yes ⁽²³⁾	37	N27F751	O	Yes	NA	Globe	M	M	-	LC	LC	LC	AI	-	-	-	In
			Water	1/2	No	57	P87F264	O	Yes	NA	Globe	S	E	-	CL	CL	OP or CL	FC	RM _p	<3	-	In
			Cont. Atmos.	1	Yes ⁽²³⁾	37	E12D015A	O	No ⁽¹⁶⁾	NA	-	-	-	-	-	-	-	-	-	-	-	In
			Cont. Atmos.	1	Yes ⁽²³⁾	57	E12D015B	O	No ⁽¹⁶⁾	NA	-	-	-	-	-	-	-	-	-	-	-	In
P107	GDC56	RHR A Loop Relief Line to Sup. Pool	Water	2	Yes	37	E12F025A	O	Yes	29'-6"	Rel	P	P	-	CL	CL	CL	AI	-	-	-	In
			Water	2	Yes	57	E12F025B	O	Yes	45'-6"	Rel	P	P	-	CL	CL	CL	AI	-	-	-	In
			Water	2	Yes	57	E12F025C	O	Yes	39'-6"	Rel	P	P	-	CL	CL	CL	AI	-	-	-	In
			Water	6	Yes	37	E12F055A	O	Yes	77'-3"	Rel	P	P	-	CL	CL	CL	AI	-	-	-	In
			Water	6	Yes	57	E12F055B	O	Yes	92'-6"	Rel	P	P	-	CL	CL	CL	AI	-	-	-	In
			Water	1	Yes	57	E12F005	O	Yes	91'-9"	Rel	P	P	-	CL	CL	CL	AI	-	-	-	In
			Cont. Atmos.	1	Yes ⁽²³⁾	37	N27D0027	O	No ⁽¹⁶⁾		-	-	-	-	-	-	-	-	-	-	-	In
			Water	2	Yes	37	E21F018	O	Yes	73'-1"	Rel	P	P	-	CL	CL	CL	AI	-	-	-	In
			Water	1	Yes ⁽²³⁾	37	N27F751	O	Yes	NA	Globe	M	M	-	LC	LC	LC	AI	-	-	-	In
			Cont. Atmos.	12	Yes ⁽²³⁾	10 (b)	E51F068	O	Yes	NA	Gate	EM	E	M	OP	OP	CL	AI	RM _s , J, G	Std.	1	In
			Water	1/2	No	57	P87F083	O	Yes	12'	Globe	S	E	-	CL	CL	OP or CL	FC	RM _p	<3	-	In
			Water	1/2	No	57	P87F264	O	Yes	NA	Globe	S	E	-	CL	CL	OP or CL	FC	RM _p	<3	-	In
			Cont. Atmos., Water, Steam	1-1/2	Yes	10 (a)	E12F102	O	Yes	135'-1"	Globe	M	M	-	LC	LC	LC	-	-	-	-	-
			Cont. Atmos.	1	Yes ⁽²³⁾	37	E12D015A	O	No ⁽¹⁶⁾	NA	-	-	-	-	-	-	-	-	-	-	-	In
			Cont. Atmos.	1	Yes ⁽²³⁾	57	E12D015B	O	No ⁽¹⁶⁾	NA	-	-	-	-	-	-	-	-	-	-	-	In
P108	GDC56	Condensate Supply	Cond.	12	Yes ⁽²³⁾	27 (a)	P11F060	O	Yes	10'-9"	B'fly	EM	E	M	OP	OP	CL	AI	B, G, RM _c	35	1	In
				12	Yes ⁽²³⁾	27 (a)	P11F545	I	Yes	NA	Chk	P	P	-	CL	CL	CL	-	Rev. Flow	-	-	In
P109	GDC56	Containment Leak Rate Test Connection Blowdown Line	Cont. Atmos.	8	No	35 (b)	Spect. Flange	O	No ⁽¹⁶⁾	12'-9-1/2"	-	-	-	-	-	-	-	-	-	-	-	Out
			Cont. Atmos.	6	No	35 (b)	Spect. Flange	I	No ⁽¹⁶⁾	NA	-	-	-	-	-	-	-	-	-	-	-	Out
			Cont. Atmos.	8	No	35 (b)	Blind Flange	I	No	NA	-	-	-	-	-	-	-	-	-	-	-	Out

TABLE 6.2-32 (Continued)

Penetration No. ⁽³⁾	GDC/ Reg. Guide	System Number	Fluid	Line Size (in)	Essential System ⁽⁴⁾	Figure 6.2-60 Arr. No.	System and Valve Number	Loc ⁽⁵⁾	Type C Test	Pipe Length ⁽⁶⁾	Valve		Actuation Mode		Valve Position			Isolation Signal ⁽⁸⁾	Closure Time (sec) ⁽⁹⁾	Power Source 1E Bus ⁽¹⁰⁾	Norm. Flow Dir.		
											Type	Oper.	Pri.	Sec.	Norm	Shut down	Post Acc.					Power Fail ⁽⁷⁾	
P111	GDC56	Condensate Return (Containment Pool Drain Line)	Water	10	No	27 (b)	P11F080	O	Yes	12'-0"	B' fly	EM	E	M	CL	CL	CL	AI	B,G,RM _c	30	1	Out	
			Water	10	No	27 (b)	P11F090	I	Yes	NA	B' fly	EM	E	M	CL	CL	CL	AI	B,G,RM _c	35	2	Out	
			Water	1/2	No	27 (b)	P11F629	I	Yes	NA	Rel	P	P	-	CL	CL	OP or CL	-	-	-	-		
P112	GDC55	LPCS Pump Discharge to Reactor Pressure Vessel	Water	12	Yes	7	E21F005	O	No ⁽²⁷⁾	29'-0"	Gate	EM	E	M	CL	CL	OP	AI	RM _c 32 ⁽¹⁹⁾	1	In		
			Water	12	Yes	7	E21F006	I	No ⁽²⁷⁾	NA	Chk	P	See Note ⁽¹¹⁾ -		CL	CL	OP	-	Rev. Flow	-	-	In	
P113	GDC55	LPCI A to Reactor and RHR to Containment Spray and Containment Pool Cooling	Water	12	Yes	11	E12F027A	O	No ⁽²⁷⁾	25'-9"	Gate	EM	E	M	OP	OP	OP	AI	RM _s	Std.	1	In	
			Water	12	Yes	11	E12F042A	I	No ⁽²⁷⁾	NA	Gate	EM	E	M	CL	CL	OP	AI	RM _s ,AA	32 ⁽¹⁹⁾	1	In	
			Water	12	Yes	11	E12F037A	I	No ⁽²⁷⁾	NA	Globe	EM	E	M	CL	OP	OP or CL	AI	A,U,M,RM _s	Std.	1	In	
				12	Yes	11	E12F028A	I	No ⁽²⁷⁾	NA	Gate	EM	E	M	CL	CL	OP	AI	RM _s	<90	1	In	
P114	GDC56	Containment Vacuum Relief	Atmos.	24	Yes	19	M17F015	O	Yes	2'-6"	B' fly	EM	E	M	OP	OP	CL	AI	B,G,X,RM _c	5	1	In	
			Atmos.	24	Yes	19	M17F010	I	Yes	NA	Chk	P	V	-	CL	CL	CL	-	Rev. Flow	-	-	In	
P115	GDC56	RCIC Turbine Exhaust Vacuum Relief	Cont. Atmos.	1-1/2	Yes	10 (a)	E12F102	O	Yes	33'-3"	Globe	M	M	-	LC	LC	LC	-	-	-	-	-	
			Water	2	Yes	37	E12F025A	O	Yes	NA	Rel	P	P	-	CL	CL	CL	AI	-	-	-	In	
			Water	2	Yes	57	E12F025B	O	Yes	NA	Rel	P	P	-	CL	CL	CL	AI	-	-	-	In	
			Water	2	Yes	57	E12F025C	O	Yes	NA	Rel	P	P	-	CL	CL	CL	AI	-	-	-	In	
			Water	6	Yes	37	E12F055A	O	Yes	NA	Rel	P	P	-	CL	CL	CL	AI	-	-	-	In	
			Water	6	Yes	57	E12F055B	O	Yes	NA	Rel	P	P	-	CL	CL	CL	AI	-	-	-	In	
			Water	1	Yes	57	E12F005	O	Yes	NA	Rel	P	P	-	CL	CL	CL	AI	-	-	-	In	
			Cont. Atmos.	1	Yes ⁽²³⁾	37	E12D015A	O	No ⁽¹⁶⁾	NA	-	-	-	-	-	-	-	-	-	-	-	In	
			Cont. Atmos.	1	Yes ⁽²³⁾	57	E12D015B	O	No ⁽¹⁶⁾	NA	-	-	-	-	-	-	-	-	-	-	-	-	In
			Water	2	Yes	37	E21F018	O	Yes	NA	Rel	P	P	-	CL	CL	CL	AI	-	-	-	In	
			Water	1	Yes ⁽²³⁾	37	N27F751	O	Yes	NA	Globe	M	M	-	LC	LC	LC	AI	-	-	-	In	
			Water	1/2	No	57	P87F264	O	Yes	NA	Globe	S	E	-	CL	CL	OP or CL	FC	RM _p	<3	-	In	
			Cont. Atmos.	12	Yes ⁽²³⁾	10 (b)	E51F068	O	Yes	18'-0"	Gate	EM	E	M	OP	OP	CL	AI	RM _s ,J,G	Std.	1	In	
			Cont. Atmos.	10	Yes ⁽²³⁾	10 (b)	E51F040	O	Yes	NA	Noz. Chk	EM	E	M	CL	CL	OP or CL	AI	-	-	-	-	In
P116	GDC57	Air Supply to ADS Accumulator	Air	1	Yes	33	P57F524B	I	Yes	NA	Chk	P	P	-	CL	CL		-	Rev. Flow	-	-	In	
			Air	1	Yes	33	P57F015B	O	Yes	10'	Globe	EM	E	M	OP	OP	OP	AI	RM _c	Std.	2	In	
P117	GDC56	Nitrogen Supply to Control Rod Drive	Nitrogen	2	No	41	P86F002	O	Yes	<20'	Globe	EM	E	M	CL	CL	CL	AI	B,G,RM _c	Std.	1	In	
			Nitrogen	2	No	41	P86F528	I	Yes	NA	Chk	P	P	-	CL	CL	CL	-	Rev. Flow	-	-	In	
P118	GDC56	RHR Heat Exchanger Vent to Suppression Pool	Noncondens.	1	Yes	45	E12F073A	O	Yes	13'-9"	Globe	EM	E	M	CL	CL	CL	-	RM _c	Std.	1	In	
				1	Yes	45	E12F558A	I	Yes	NA	Chk	P	P	-	CL	CL	CL	-	Rev. Flow	-	-	In	
P119	GDC56	Containment Leak Rate Test PI	Cont. Atmos.	1/2	No	35	Spect. Flange	O	No ⁽¹⁶⁾	19'-9-3/16"	-	-	-	-	-	-	-	-	-	-	-	Out	
			Cont. Atmos.	1/2	No	35	Spect. Flange	I	No ⁽¹⁶⁾	NA	-	-	-	-	-	-	-	-	-	-	-	-	Out
P120	GDC56	Containment Leak Rate - Pressurization Line	Air	8	No	35 (b)	Spect. Flange	O	No ⁽¹⁶⁾	12'-9-1/2"	-	-	-	-	-	-	-	-	-	-	-	In	
			Air	6	No	35 (b)	Spect. Flange	I	No ⁽¹⁶⁾	NA	-	-	-	-	-	-	-	-	-	-	-	In	
			Air	8	No	35 (b)	Blind Flange	I	No ⁽¹⁶⁾	NA	-	-	-	-	-	-	-	-	-	-	-	In	
P121	GDC55	Feedwater A, RHR, and RWCU Return to Reactor Pressure Vessel	Water	20	Yes ⁽²³⁾	2	B21F065A	O	Yes	26'-9-5/8"	Gate	EM	E	M	OP	CL	OP or CL	AI	RM _c	Std.	1 ⁽²⁰⁾	In	
			Water	20	Yes ⁽²³⁾	2	B21F032A	O	No ⁽²¹⁾	NA	Chk	P	P	-	OP	CL	OP or CL	-	Rev. Flow	-	-	In	
			Water	20	Yes ⁽²³⁾	2	N27F559A	I	No ⁽²¹⁾	NA	Chk	P	P	-	OP	CL	OP or CL	-	Rev. Flow	-	-	In	
			Water	12	Yes	2	E12F053A	O	Yes	39'-7-2/8"	Globe	EM	E	M	CL	OP	CL	AI	RM _c ,A,U	33	1	In	
P122	GDC55	Main Steam Line C	Steam	26	Yes ⁽²³⁾	1 (a)	B21F028C	O	Yes	17'-2-3/4"	Globe	A	A	SP	OP	CL	CL	FC	C,E,F,S,N,P,RM _c	See Note ⁽¹³⁾	-	Out	
			Steam	26	Yes ⁽²³⁾	1 (a)	B21F022C	I	Yes	NA	Globe	A	A	SP	OP	CL	CL	FC	C,E,F,S,N,P,RM _c	See Note ⁽¹³⁾	-	Out	
P123	GDC55	RCIC Pump Discharge and RHR Head Spray	Water	6	Yes ⁽²³⁾	5	E51F066	I	No ⁽²⁷⁾	NA	Chk	P	See Note ⁽¹¹⁾ -		CL	CL	OP or CL	-	Rev. Flow	-	-	In	
			Water	6	Yes ⁽²³⁾	5	E51F013	O	No ⁽²⁷⁾	43'-6"	Gate	EM	E	M	CL	CL	OP or CL	AI	RM _s ,EE,FF	15	1	In	
			Water	6	Yes	5	E12F023	O	No ⁽²⁷⁾	50'-5"	Globe	EM	E	M	CL	CL	OP or CL	AI	A,M,U,RM _s	Std.	1	In	
P124	GDC55	Main Steam Line A	Steam	26	Yes ⁽²³⁾	1 (a)	B21F028A	O	Yes	16'-5-7/8"	Globe	A	A	SP	OP	CL	CL	FC	C,E,F,S,N,P,RM _c	See Note ⁽¹³⁾	-	Out	
			Steam	26	Yes ⁽²³⁾	1 (a)	B21F022A	I	Yes	NA	Globe	A	A	SP	OP	CL	CL	FC	C,E,F,S,N,P,RM _c	See Note ⁽¹³⁾	-	Out	

TABLE 6.2-32 (Continued)

Penetration No. ⁽³⁾	GDC/ Reg. Guide	System Number	Fluid	Line Size (in)	Essential System ⁽⁴⁾	Figure 6.2-60 Arr. No.	System and Valve Number	Loc ⁽⁵⁾	Type C Test	Pipe Length ⁽⁶⁾	Valve		Actuation Mode		Valve Position			Isolation Signal ⁽⁸⁾	Closure Time (sec) ⁽⁹⁾	Power Source 1E Bus ⁽¹⁰⁾	Norm. Flow Dir.	
											Type	Oper.	Pri.	Sec.	Norm	Shut down	Post Acc.					Power Fail ⁽⁷⁾
P131	GDC55	RWCU Pump Suction	Water	6	Yes ⁽²³⁾	49	G33F001	I	Yes	NA	Gate	EM	E	M	OP	OP	CL	AI	L,B,F,H,Y,RM _c	20	2	Out
			Water	6	Yes ⁽²³⁾	49	G33F004	O	Yes	14'-0"	Gate	EM	E	M	OP	OP	CL	AI	L,B,F,H,W,Y, RM _c ,RM _f	20	1	Out
P132	GDC55	RWCU Line from Regenerative Heat Exchanger to Feedwater	Water	6	Yes ⁽²³⁾	44	G33F040	I	Yes	NA	Gate	EM	E	M	OP	OP	CL	AI	L,B,F,H,RM _c	27	2	Out
			Water	6	Yes ⁽²³⁾	44	G33F039	O	Yes	10'-9"	Gate	EM	E	M	OP	OP	CL	AI	B,F,H,L,RM _c	27	1	Out
P201	GDC56	Drywell Atmosphere Radiation Monitor Line	Drywell Atmos.	1	No	52	D17F079A	O	Yes	<10'	Globe	S	E	-	OP	OP	CL	AI	B,G,RM _c	<3	1	In
			Drywell Atmos.	1	No	52	D17F079B	I	Yes	<10'	Globe	S	E	-	OP	OP	CL	AI	B,G,RM _c	<3	2	In
			Drywell Atmos.	1	No	52	D17F071A	O	Yes	<10'	Ball	EM	E	M	OP	OP	CL	FC	B,G,RM _c	<3	1	Out
			Drywell Atmos.	1	No	52	D17F071B	I	Yes	<10'	Ball	EM	E	M	OP	OP	CL	FC	B,G,RM _c	<3	2	Out
P203	GDC56	Fuel Pool Cooling Supply	Water	8	No	26(a)	G41F100	O	Yes	10'-9"	B'fly	EM	E	M	OP	OP	CL	AI	B,G,RM _c	35	1	In
			Water	8	No	26(a)	G41F522	I	Yes	NA	Chk	P	P	-	OP	OP	CL	-	Rev. Flow	-	-	In
P204	GDC55	Control Rod Drive to Reactor Pressure Vessel	Condensate	2-1/2	Yes ⁽²³⁾	3	C11F083	O	Yes	18'-0"	Gate	EM	E	M	OP	OP	CL	AI	RM _c	Std.	1	In
			Condensate	2-1/2	Yes ⁽²³⁾	3	C11F122	I	Yes	NA	Chk	P	P	-	OP	OP	CL	-	Rev. Flow	-	-	In
P205	GDC56	Fuel Transfer Tube	Water	24	No	36	Double gasket See Note ⁽¹⁴⁾	I	No See Note ⁽¹⁴⁾	NA	-	-	-	-	-	-	-	-	-	-	-	-
P208	GDC56	Containment Vacuum Relief	Atmos.	24	Yes	19	M17F025	O	Yes	2'-6"	B'fly	EM	E	M	OP	OP	CL	AI	B,G,X,RM _c	5	1	In
			Atmos.	24	Yes	19	M17F020	I	Yes	NA	Chk	P	V	-	CL	CL	CL	-	Rev. Flow	-	-	In
P210	GDC56	Carbon Dioxide to Fire Protection System	CO ₂	4	No	42	P54F340	O	Yes	12'-6"	Gate	EM	E	M	CL	CL	CL	AI	B,G,RM _c	20	1	In
			CO ₂	4	No	42	P54F1098	I	Yes	NA	Chk	P	P	-	CL	CL	CL	-	Rev. Flow	-	-	In
P301	GDC56	Fuel Pool Cooling Return	Water	10	No	26(b)	G41F145	O	Yes	13'-0"	B'fly	EM	E	M	OP	OP	CL	AI	B,G,RM _c	35	1	Out
			Water	10	No	26(b)	G41F140	I	Yes	NA	B'fly	EM	E	M	OP	OP	CL	AI	B,G,RM _c	35	2	Out
			Water	3/4	No	26(b)	G41F801	I	Yes	NA	Check	P	P	-	CL	CL	OP or CL	-	Rev. Flow	-	-	-
P302	GDC56	Backup Hydrogen Purge System	Drywell Atmos.	2	Yes	39(b)	M51F110	O	Yes	18'-0"	Globe	EM	E	M	CL	OP or CL	CL	AI	C,G,GG,RM _c	Std.	1	Out
			Drywell Atmos.	2	Yes	39(b)	M51F090	I	Yes	NA	Globe	EM	E	M	CL	OP	CL	AI	C,G,GG,RM _c	Std.	2	Out
P304	GDC56	Air Supply to ADS Accumulators	Air	1	Yes	33	P57F015A	O	Yes	10'-6"	Globe	EM	E	M	OP	OP	OP	AI	RM _c ,RM _f	Std.	1	In
			Air	1	Yes	33	P57F524A	I	Yes	NA	Chk	P	P	-	CL	CL	CL	-	-	-	-	In
P305	GDC56	Lower Personnel Airlock	Air	3/4	Yes ⁽²³⁾	56	P53F010	O	Yes	<10'	Globe	S	E	-	CL	CL	CL	AI	B,G,RM _c	<3	2	In
			Air	3/4	Yes ⁽²³⁾	56	P53F015	O ⁽²²⁾	Yes	<10'	Globe	S	E	-	CL	CL	CL	AI	B,G,RM _c	<3	2	In
			Air	3/4	Yes ⁽²³⁾	56	P53F070	O	Yes	<10'	Globe	S	E	-	OP	OP	CL	FC	B,G,RM _c	<3	1	In
			Air	3/4	Yes ⁽²³⁾	56	P53F030	O	Yes	<10"	Globe	S	E	-	OP	OP	OP	AI	RM _c ,B,G	<3	1	Out
			Air	3/4	No	56	P53F035	O	Yes	<10'	Globe	S	E	-	CL	CL	CL	AI	RM _c ,B,G	<3	1	Out
			Air	3/4	Yes ⁽²³⁾	56	P52F160	O	Yes	<10'	Globe	S	E	-	OP	OP	CL	FC	RM _c ,B,G	<3	1	In
			Air	3/4	Yes ⁽²³⁾	56	Inflatable gaskets	I	No ⁽¹⁶⁾	NA	-	-	-	-	-	-	-	-	-	-	-	-
			Air	1/2	No	-	P53F556	I	Yes	<10'	Needle	M	M	-	OP	OP	OP	AI	-	-	-	-
			Air	1	No	-	P53F536	O ⁽²²⁾	Yes	<10'	Globe	M	M	-	LC	LC	LC	AI	-	-	-	-
			Air	1	No	-	P53F570	O	Yes	<10'	Globe	M	M	-	LC	LC	LC	AI	-	-	-	-
			Air	1	No	-	P53F579A	O	Yes	<10'	Ball	M	M	-	CL	CL	CL	AI	-	-	-	-
			Air	1	No	-	P53F579B	I ⁽²²⁾	Yes	<10'	Ball	M	M	-	CL	CL	CL	AI	-	-	-	-
			Air	1	No	-	P53F580A	O ⁽²²⁾	Yes	<10'	Ball	M	M	-	CL	CL	CL	AI	-	-	-	-
			Air	1	No	-	P53F580B	I	Yes	<10'	Ball	M	M	-	CL	CL	CL	AI	-	-	-	-
			Air	1	No	-	P53F581	I ⁽²²⁾	Yes	<10'	Relief	-	-	-	CL	CL	CL	AI	-	-	-	In
			Air	1	No	-	P53F582	I	Yes	<10'	Relief	-	-	-	CL	CL	CL	AI	-	-	-	In
P306	GDC57	Instrument Air	Air	2	Yes ⁽²³⁾	34	P52F200	O	Yes	24'-6"	Globe	EM	E	M	OP	OP	CL	AI	C,G,RM _c	Std.	1	In
			Air	1-1/2	Yes ⁽²³⁾	34	P52F550	I	Yes	NA	Chk	P	P	-	OP	OP	CL	-	Rev. Flow	-	-	In

TABLE 6.2-32 (Continued)

Penetration No. ⁽³⁾	GDC/ Reg. Guide	System Number	Fluid	Line Size (in)	Essential System ⁽⁴⁾	Figure 6.2-60 Arr. No.	System and Valve Number	Loc ⁽⁵⁾	Type C Test	Pipe Length ⁽⁶⁾	Valve		Actuation Mode		Valve Position			Isolation Signal ⁽⁸⁾	Closure Time(sec) ⁽⁹⁾	Power Source 1E Bus ⁽¹⁰⁾	Norm. Flow Dir.	
											Type	Oper.	Pri.	Sec.	Norm	Shut down	Post Acc.					Power Fail ⁽⁷⁾
P308	GDC56	Service Air	Air	2-1/2	No	32	P51F150	O	Yes	11'-3"	Globe	A	A	M	OP	OP	CL	FC	B,G,RM _c	15	-	In
			Air	2-1/2	No	32	P51F530 ⁽²⁶⁾	I	Yes	NA	Chk	P	P	-	CL	CL	CL	-	Rev. Flow	-	-	In
P309	GDC56	Demineralized Water	Demin. Wtr.	3	No	29	P22F010	O	Yes	17'-3"	Gate	EM	E	M	OP	OP	CL	AI	B,G,RM _c	22 Max.	1	In
			Demin. Wtr.	3	No	29	P22F577	I	Yes	NA	Chk	P	P	-	CL	CL	CL	-	Rev. Flow	-	-	In
P310	GDC56	Nuclear Closed Cooling Water Supply	Water	12	No	25 (a)	P43F055	O	Yes	8'-6"	B'fly	EM	E	M	OP	OP	CL	AI	C,G,RM _c	35	1	In
			Water	12	No	25 (a)	P43F721	I	Yes	NA	Chk	P	P	-	OP	OP	CL	-	Rev. Flow	-	-	In
P311	GDC56	Nuclear Closed Cooling Water Return	Water	12	No	25 (b)	P43F140	O	Yes	8'-6"	B'fly	EM	E	M	OP	OP	CL	AI	C,G,RM _c	30	1	Out
			Water	12	No	25 (b)	P43F215	I	Yes	NA	B'fly	EM	E	M	OP	OP	CL	AI	C,G,RM _c	35	2	Out
			Water	1/4	No	25 (b)	1P43F851	I	Yes	NA	Relief	P	P	-	CL	CL	OP or CL	-	-	-	-	-
P312	GDC56	Upper Personnel Airlock	Air	3/4	Yes ⁽²³⁾	56	P53F020	O	Yes	<10'	Globe	S	E	-	CL	CL	CL	AI	B,G,RM _c	<3	2	In
			Air	3/4	Yes ⁽²³⁾	56	P53F025	O ⁽²²⁾	Yes	<10'	Globe	S	E	-	CL	CL	CL	AI	B,G,RM _c	<3	2	In
			Air	3/4	Yes ⁽²³⁾	56	P53F075	O	Yes	<10'	Globe	S	E	-	OP	OP	CL	FC	B,G,RM _c	<3	1	In
			Air	3/4	Yes ⁽²³⁾	56	P53F040	O	Yes	<10'	Globe	S	E	-	OP	OP	OP	AI	RM _c ,B,G	<3	1	Out
			Air	3/4	No	56	P53F045	O	Yes	<10'	Globe	S	E	-	CL	CL	CL	AI	RM _c ,B,G	<3	1	Out
			Air	3/4	Yes ⁽²³⁾	56	P52F170	O	Yes	<10'	Globe	S	E	-	OP	OP	CL	FC	B,G,RM _c	<3	1	In
			Air	3/4	Yes ⁽²³⁾	56	Inflatable gaskets	I	No ⁽¹⁶⁾	NA	-	-	-	-	-	-	-	-	-	-	-	-
			Air	1/2	No	-	P53F561	I	Yes	<10'	Needle	M	M	-	OP	OP	OP	AI	-	-	-	-
			Air	1	No	-	P53F541	O ⁽²²⁾	Yes	<10'	Globe	M	M	-	LC	LC	LC	AI	-	-	-	-
			Air	1	No	-	P53F571	O	Yes	<10'	Globe	M	M	-	LC	LC	LC	AI	-	-	-	-
			Air	1	No	-	P53F593A	O	Yes	<10'	Ball	M	M	-	CL	CL	CL	AI	-	-	-	-
			Air	1	No	-	P53F593B	I ⁽²²⁾	Yes	<10'	Ball	M	M	-	CL	CL	CL	AI	-	-	-	-
			Air	1	No	-	P53F594A	O ⁽²²⁾	Yes	<10'	Ball	M	M	-	CL	CL	CL	AI	-	-	-	-
			Air	1	No	-	P53F594B	I	Yes	<10'	Ball	M	M	-	CL	CL	CL	AI	-	-	-	-
			Air	1	No	-	P53F595	I ⁽²²⁾	Yes	<10'	Relief	-	-	-	CL	CL	CL	AI	-	-	-	In
			Air	1	No	-	P53F596	I	Yes	<10'	Relief	-	-	-	CL	CL	CL	AI	-	-	-	In
V313	GDC56	Containment Purge Supply	Cont. Atmos.	42	No	30 (a)	M14F040	O	Yes	2'-0"	B'fly	A	A	-	CL	OP	CL	FC	B,G,RM _c ,Z	4	-	In
			Cont. Atmos.	42	No	30 (a)	M14F045	I	Yes	NA	B'fly	A	A	-	CL	OP	CL	FC	B,G,RM _c ,Z	4	-	In
			Cont. Atmos.	18	No	30 (a)	M14F190	I	Yes	NA	B'fly	A	A	-	CL	OP	CL	FC	B,G,RM _c ,Z	4	-	In
			Cont. Atmos.	18	No	30 (a)	M14F195	I	Yes	NA	B'fly	A	A	-	CL	OP	CL	FC	B,G,RM _c ,Z	4	-	In
V314	GDC56	Containment Purge Exhaust	Cont. Atmos.	42	No	30 (b)	M14F090	O	Yes	2'-0"	B'fly	A	A	-	CL	OP	CL	FC	B,G,RM _c ,Z	4	-	Out
			Cont. Atmos.	42	No	30 (b)	M14F085	I	Yes	NA	B'fly	A	A	-	CL	OP	CL	FC	B,G,RM _c ,Z	4	-	Out
			Cont. Atmos.	18	No	30 (b)	M14F200	I	Yes	NA	B'fly	A	A	-	CL	OP	CL	FC	B,G,RM _c ,Z	4	-	Out
			Cont. Atmos.	18	No	30 (b)	M14F205	I	Yes	NA	B'fly	A	A	-	CL	OP	CL	FC	B,G,RM _c ,Z	4	-	Out
P315	GDC56	SLC Tank Refill	Sodium Pentaborate Solution	2	Yes ⁽²³⁾	60	C41F518	O	Yes	29'	Globe	M	M	-	LC	LC	LC	-	-	-	In	
						C41F520	I	Yes	NA	Chk	P	P	-	CL	CL	CL	-	Rev. Flow	-	-	In	
P317	GDC56	Containment Atmosphere Radiation Monitor Line	Cont. Atmos.	1	No	54	D17F089A	O	Yes	<10'	Globe	S	E	-	OP	OP	CL	AI	B,G,RM _c	<3	1	In
			Cont. Atmos.	1	No	54	D17F089B	I	Yes	NA	Globe	S	E	-	OP	OP	CL	AI	B,G,RM _c	<3	2	In
			Cont. Atmos.	1	No	54	D17F081A	O	Yes	<10'	Ball	EM	E	M	OP	OP	CL	FC	B,G,RM _c	<3	1	Out
			Cont. Atmos.	1	No	54	D17F081B	I	Yes	NA	Ball	EM	E	M	OP	OP	CL	FC	B,G,RM _c	<3	2	Out
	GDC56	Containment Leak Rate	Cont. Atmos.	1/2	No	35	Spect. Flange	O	No ⁽¹⁶⁾	12'-2-1/4"	-	-	-	-	-	-	-	-	-	-	-	Out
			Cont. Atmos.	1/2	No	35	Pipe Cap	I	No ⁽¹⁶⁾	NA	-	-	-	-	-	-	-	-	-	-	-	Out
			Cont. Atmos.	3/4	No	35	Spect. Flange	O	No ⁽¹⁶⁾	12'-2-1/4"	-	-	-	-	-	-	-	-	-	-	-	Out
			Cont. Atmos.	3/4	No	35	Pipe Cap	I	No ⁽¹⁶⁾	NA	-	-	-	-	-	-	-	-	-	-	-	Out
			Cont. Atmos.	3/4	No	35	E61F550	I	Yes	NA	Globe	M	M	-	LC	LC	LC	-	-	-	-	Out
			Cont. Atmos.	1/2	No	35	E61F549	I	Yes	NA	Globe	M	M	-	LC	LC or OP	LC	-	-	-	-	Out

TABLE 6.2-32 (Continued)

Penetration No. ⁽³⁾	GDC/ Reg. Guide	System Number	Fluid	Line Size (in)	Essential System ⁽⁴⁾	Figure 6.2-60 Arr. No.	System and Valve Number	Loc ⁽⁵⁾	Type C Test	Pipe Length ⁽⁶⁾	Valve		Actuation Mode		Valve Position				Isolation Signal ⁽⁸⁾	Closure Time (sec) ⁽⁹⁾	Power Source 1E Bus ⁽¹⁰⁾	Norm. Flow Dir.
											Type	Oper.	Pri.	Sec.	Norm	Shut down	Post Acc.	Power Fail ⁽⁷⁾				
P318	RG1.11	Combustible Gas Control Postaccident Hydrogen Analyzer	Cont. Atmos.	1/2	Yes	58(b)	M51F210B	O	Yes	<10'	Globe	S	E	-	CL	CL	OP	AI	RM _c	<3	2	Out
			Drywell Atmos.	1/2	Yes	58(b)	M51F220B	O	Yes	F10'	Globe	S	E	-	CL	CL	OP	AI	RM _c	<3	2	Out
			Drywell Atmos.	1/2	Yes	58(b)	M51F230B	O	Yes	<10'	Globe	S	E	-	CL	CL	OP	AI	RM _c	<3	2	Out
			Sup. Pool Atmos. Analyzer	1/2	Yes	58(b)	M51F240B	O	Yes	<10'	Globe	S	E	-	CL	CL	OP	AI	RM _c	<3	2	Out
			Exhaust Sup. Pool	1/2	Yes	58(b)	M51F250B	O	Yes	<10'	Globe	S	E	-	CL	CL	OP	FC	RM _c	<3	2	In
			Atmos. Gas Sample	1/2	No	58(b)	P87F071	O	Yes	<15	Globe	S	E	-	CL	CL	OP or CL	FC	RM _p	<3	-	Out
			Return	1/2	No	58(b)	P87F065	O	Yes	<15	Globe	S	E	-	CL	CL	OP or CL	FC	RM _p	<3	-	In
			Drywell Atmos.	1/2	No	58(b)	P87F077	O	Yes	<15	Globe	S	E	-	CL	CL	OP or CL	FC	RM _p	<3	-	Out
			Cont. Atmos.	1/2	No	58(b)	P87F074	O	Yes	<15	Globe	S	E	-	CL	CL	OP or CL	FC	RM _p	<3	-	Out
P319	RG1.11	Containment Atmosphere Monitoring	Cont. Atmos.	3/4	Yes ⁽²³⁾	38(b)	D23F030B	O	No	<10'	Globe	S	E	-	OP	OP	OP	AI	RM _c	<3	2	-
			Cont. Atmos.	3/4	Yes ⁽²³⁾	38(b)	1/4 orifice	I	No	NA	-	-	-	-	-	-	-	-	-	-	-	-
			Drywell Atmos.	3/4	Yes ⁽²³⁾	38(a)	D23F040B	O	No	<10'	Globe	S	E	-	OP	OP	OP	AI	RM _c	<3	2	-
			Drywell Atmos.	3/4	Yes ⁽²³⁾	38(a)	1/4" orifice	I	No	NA	-	-	-	-	-	-	-	-	-	-	-	-
	GDC56	Containment Leak Rate	Cont. Atmos.	3/4	No	35	Spect. Flange	O	No ⁽¹⁶⁾	14'-11"	-	-	-	-	-	-	-	-	-	-	-	Out
			Cont. Atmos.	3/4	No	35	E61F551	I	Yes	NA	Globe	M	M	-	LC	LC	LC	-	-	-	-	Out
																or OP						
			Cont. Atmos.	3/4	No	35	Pipe Cap	I	No ⁽¹⁶⁾	NA	-	-	-	-	-	-	-	-	-	-	-	Out
			Cont. Atmos.	1/2	No	35	Spect. Flange	O	No ⁽¹⁶⁾	15'-11"	-	-	-	-	-	-	-	-	-	-	-	Out
Cont. Atmos.	1/2	No	35	E61F552	I	Yes	NA	Globe	M	M	-	LC	LC	LC	-	-	-	-	Out			
														or OP								
Cont. Atmos.	1/2	No	35	Pipe Cap	I	No ⁽¹⁶⁾	NA	-	-	-	-	-	-	-	-	-	-	-	-	Out		
P320	GDC56	Containment Vacuum Relief	Cont. Atmos.	3/4	Yes	59	M17F065	O	No	<10'	Globe	S	E	-	OP	OP	OP	AI	RM _c	<3	2	-
			Cont. Atmos.	3/4	Yes	59	1/4" orifice	I	No	NA	-	-	-	-	-	-	-	-	-	-	-	-
	RG1.11	Containment Atmosphere Monitoring	Cont. Atmos.	3/4	Yes ⁽²³⁾	38(b)	D23F020B	O	No	<10'	Globe	S	E	-	OP	OP	OP	AI	RM _c	<3	2	-
			Cont. Atmos.	3/4	Yes ⁽²³⁾	38(b)	1/4" orifice	I	No	NA	-	-	-	-	-	-	-	-	-	-	-	-
			Cont. Atmos.	3/4	Yes ⁽²³⁾	38(b)	D23F010B	O	No	<10'	Globe	S	E	-	OP	OP	OP	AI	RM _c	<3	2	-
			Cont. Atmos.	3/4	Yes ⁽²³⁾	38(b)	1/4" orifice	I	No	NA	-	-	-	-	-	-	-	-	-	-	-	-
P401	GDC56	HPCS Pump Suction	Water	24	Yes	8	E22F015	O	No ⁽²⁷⁾	18'-3"	Gate	EM	E	M	CL	CL	OP or CL	AI	RM _c	24	3	Out
	RG1.11	Suppression Pool Makeup	Water	3/4	Yes	55	G43F060	O	No	<10'	Globe	S	E	-	OP	OP	OP	AI	RM _c	<3	-	-
		Postaccident Sampling (PASS)	Water	3/4	No	55	P87F037	O	Yes	<15'	Globe	S	E	-	CL	CL	OP or CL	FC	RM _p	<3	-	Out
P402	GDC56	RHR B Pump Suction	Water	24	Yes	13	E12F004B	O	No ⁽²⁷⁾	17'-8/16"	Gate	EM	E	M	OP	CL	OP	AI	RM _c	Std.	2	Out
	RG1.11	Suppression Pool Makeup	Water	3/4	Yes	55	G43F050B	O	No	<10'	Globe	S	E	-	OP	OP	OP	AI	RM _c	<3	2	-
P403	GDC56	RHR C Pump Suction	Water	24	Yes	13	E12F105	O	No ⁽²⁷⁾	20'-11/16"	Gate	EM	E	M	OP	CL	OP	AI	RM _c	Std.	2	Out
P404	GDC56	Chilled Water Supply	Water	6	Yes	28(a)	P50F060	O	Yes	9'-6"	B'fly	EM	E	M	OP	OP	CL	AI	B,G, RM _c	30	1	In
			Water	6	Yes	28(a)	P50F539	I	Yes	NA	Chk	P	P	-	OP	OP	CL	-	Rev. Flow	-	-	In
P405	GDC56	Chilled Water Return	Water	6	Yes	28(b)	P50F150	O	Yes	10'-3"	B'fly	EM	E	M	OP	OP	CL	AI	B,G, RM _c	35	1	Out
			Water	6	Yes	28(b)	P50F140	I	Yes	NA	B'fly	EM	E	M	OP	OP	CL	AI	B,G, RM _c	35	2	Out
			Water	1/4	No	28(b)	P50F606	I	Yes	NA	Relief	P	P	-	CL	CL	OP or CL	-	-	-	-	-
P406	GDC56	Fire Protection Water	Nitrogen ⁽²⁴⁾	4	No	51	P54F726	O	Yes	13'-3"	Gate	M	M	-	LC	LC	LC	-	-	-	-	In
			Nitrogen ⁽²⁴⁾	4	No	51	P54F727	I	Yes	NA	Gate	M	M	-	LC	LC	LC	-	-	-	-	In
P407	GDC56	RHR B Test and Pump Minimum Flow Line, RHR Heat Exchanger Dump to Suppression Pool	Water	18	Yes	17	E12F024B	O	No ⁽²⁷⁾	20'-3"	Globe	EM	E	M	CL	CL	OP or CL	AI	C,G, RM _c , AA	100	2	In
			Water	4	Yes	17	E12F011B	O	No ⁽²⁷⁾	15'-9"	Globe	EM	E	M	CL	CL	CL	AI	C,G, RM _c	Std.	2	In
			Water	6	Yes	17	E12F064B	O	No ⁽²⁷⁾	78'-0"	Gate	EM	E	M	OP	OP	OP or CL	AI	RM _c , BB	15	2	In
			Cont. Atmos.	18	Yes	17	E12D003B	I	No ⁽¹⁶⁾	NA	-	-	-	-	-	-	-	-	-	-	-	-

TABLE 6.2-32 (Continued)

Penetration No. ⁽³⁾	GDC/ Reg. Guide	System Number	Fluid	Line Size (in)	Essential System ⁽⁴⁾	Figure 6.2-60 Arr. No.	System and Valve Number	Loc ⁽⁵⁾	Type C Test	Pipe Length ⁽⁶⁾	Valve		Actuation Mode		Valve Position			Isolation Signal ⁽⁸⁾	Closure Time (sec) ⁽⁹⁾	Power Source 1E Bus ⁽¹⁰⁾	Norm. Flow Dir.	
											Type	Oper.	Pri.	Sec.	Norm	Shut down	Post Acc.					Power Fail ⁽⁷⁾
P408	GRDC56	RHR C Test and Pump Minimun Flow Line	Water	18	Yes	23	E12F021	O	Yes	18'-7-1/2"	Globe	EM	E	M	CL	CL	CL	AI	C,G,RM _c	90	2	In
			Water	6	Yes	22	E12F064C	O	Yes	97'-0"	Gate	EM	E	M	OP	OP	OP or CL	AI	RM _c ,BB	15	2	In
P409	GRD56	HPCS Minimum Flow Line and Test Line to Suppression Pool	Water	12	Yes	24	E22F023	O	Yes	27'-9"	Globe	EM	E	M	CL	CL	CL	AI	B,G,RM _c	Std.	3	In
			Water	12	Yes	24	E22F012	O	Yes	17'-0"	Gate	EM	E	M	CL	CL	OP or CL	AI	RM _c ,BB,CC	8	3	In
			Water	2	Yes	24	E22F035	O	Yes	29'-8-1/2"	Relief	P	P	-	CL	CL	CL	-	-	-	-	In
P410	GDC55	HPCS to Reactor Pressure Vessel	Water	12	Yes	9	E22F004	O	No ⁽²⁷⁾	22'-0"	Gate	EM	E	M	CL	CL	OP	AI	RM _c ,DD	16 ⁽¹⁹⁾	3	In
			Water	12	Yes	9	E22F005	I	No ⁽²⁷⁾	NA	Chk	P	See Note ⁽¹¹⁾	-	CL	CL	OP	-	Rev. Flow	-	-	In
P411	GDC55	LPCI C to Reactor	Water	12	Yes	12	E12F042C	O	No ⁽²⁷⁾	22'-9"	Gate	EM	E	M	CL	CL	OP	AI	RM _c	32 ⁽¹⁹⁾	2	In
			Water	12	Yes	12	E12F041C	I	No ⁽²⁷⁾	NA	Chk	P	See Note ⁽¹¹⁾	-	CL	CL	OP	-	Rev. Flow	-	-	In
P412	GDC55	LPCI B to Reactor and RHR to Containment Spray and Containment Pool Cooling	Water	12	Yes	11	E12F027B	O	No ⁽²⁷⁾	19'-3"	Gate	EM	E	M	OP	OP	OP	AI	RM _c	Std.	2	In
			Water	12	Yes	11	E12F042B	I	No ⁽²⁷⁾	NA	Gate	EM	E	M	CL	CL	OP	AI	RM _c ,AA	32 ⁽¹⁹⁾	2	In
			Water	12	Yes	11	E12F037B	I	No ⁽²⁷⁾	NA	Globe	EM	E	M	CL	OP	OP or CL	AI	A,M,U,RM _c	Std.	2	In
			Water	12	Yes	11	E12F028B	I	No ⁽²⁷⁾	NA	Gate	EM	E	M	CL	CL	OP	AI	RM _c	<90 ⁽¹⁹⁾	2	In
P413	GDC56	PASS	Water	3/4	No	61	P87F049	I	Yes	NA	Globe	S	E	-	CL	CL	OP or CL	FC	RM _p	<3	-	Out
			Water	3/4	No	61	P87F055	O	Yes	<10'	Globe	S	E	-	CL	CL	OP or CL	FC	RM _p	<3	-	Out
			Air ⁽²⁴⁾	3/4	No	61	P87F046	I	Yes	NA	Globe	S	E	-	CL	CL	OP or CL	FC	RM _p	<3	-	Out
			Air ⁽²⁴⁾	3/4	No	61	P87F052	O	Yes	<10'	Globe	S	E	-	CL	CL	OP or CL	FC	RM _p	<3	-	Out
			Water	1/4	No	61	P87F277	I	Yes	NA	Rel	P	P	-	CL	CL	OP or CL	-	-	-	-	Out
P414	GDC55	Feedwater B, RHR and RWCU Return to Reactor Pressure Vessel	Water	20	Yes ⁽²³⁾	2	B21F065B	O	Yes	26'-9-5/8"	Gate	EM	E	M	OP	CL	OP or CL	AI	RM _c	Std.	1 ⁽²⁰⁾	In
			Water	20	Yes ⁽²³⁾	2	B21F032B	O	No ⁽²¹⁾	NA	Chk	P	P	-	OP	CL	OP or CL	-	Rev. Flow	-	-	In
			Water	20	Yes ⁽²³⁾	2	N27F559B	I	No ⁽²¹⁾	NA	Chk	P	P	-	OP	CL	OP or CL	-	Rev. Flow	-	-	In
			Water	12	Yes	2	E12F053B	O	Yes	39'-7-3/8"	Globe	EM	E	M	CL	OP	CL	AI	RM _c ,A,U	33	2	In
P415	GDC55	Main Steam Line D	Steam	26	Yes ⁽²³⁾	1 (a)	B21F028D	O	Yes	16'-5-7/8"	Globe	A	A	SP	OP	CL	CL	FC	C,E,F,S,N,P,RM _c	See Note ⁽¹³⁾	-	Out
			Steam	26	Yes ⁽²³⁾	1 (a)	B21F022D	I	Yes	NA	Globe	A	A	SP	OP	CL	CL	FC	C,E,F,S,N,P,RM _c	See Note ⁽¹³⁾	-	Out
P416	GDC55	Main Steam Line B	Steam	26	Yes ⁽²³⁾	1 (a)	B21F028B	O	Yes	17'-4-3/4"	Globe	A	A	SP	OP	CL	CL	FC	C,E,F,S,N,P,RM _c	See Note ⁽¹³⁾	-	Out
			Steam	26	Yes ⁽²³⁾	1 (a)	B21F022B	I	Yes	NA	Globe	A	A	SP	OP	CL	CL	FC	C,E,F,S,N,P,RM _c	See Note ⁽¹³⁾	-	Out
P417	GDC56	Drywell and Containment Equipment Drain Sump to Radwaste	Water	3	No	40	G61F080	O	Yes	26'-6"	Gate	EM	E	M	OP	OP	CL	AI	B,G,RM _c	22	1	Out
			Water	3	No	40	G61F075	I	Yes	NA	Gate	EM	E	M	OP	OP	CL	AI	B,G,RM _c	22	2	Out
			Water	3/4	No	40	G61F0655	I	Yes	NA	Check	P	P	-	CL	CL	OP or CL	-	Rev. Flow	-	-	-
P418	GDC56	Drywell and Containment Floor Drain Sump to Radwaste	Water	3	No	43	G61F170	O	Yes	28'-3"	Gate	EM	E	M	OP	OP	CL	AI	B,G,RM _c	22	1	Out
			Water	3	No	43	G61F165	I	Yes	NA	Gate	EM	E	M	OP	OP	CL	AI	B,G,RM _c	22	2	Out
			Water	3/4	No	43	G61F657	I	Yes	NA	Check	P	P	-	CL	CL	OP or CL	-	Rev. Flow	-	-	-
P419	GDC55	RWCU Pump Discharge	Water	4	No	48	G33F054	O	Yes	10'-6"	Gate	EM	E	M	OP	OP	CL	AI	L,B,F,H,RM _c	15.5 ⁽²⁵⁾	1	In
			Water	4	No	48	G33F053	I	Yes	NA	Gate	EM	E	M	OP	OP	CL	AI	L,B,F,H,RM _c	15.5 ⁽²⁵⁾	2	In
P420	GDC56	RWCU Backwash Transfer Pump to Radwaste	Water	4	No	46	G50F277	O	Yes	12'-0"	Gate	EM	E	M	OP	OP	CL	AI	B,G,RM _c	Std.	1	Out
			Water	4	No	46	G50F272	I	Yes	NA	Gate	EM	E	M	OP	OP	CL	AI	B,G,RM _c	Std.	2	Out
			Water	1/2	No	46	G50F823	I	Yes	NA	Relief	P	P	-	CL	CL	OP or CL	-	-	-	-	-
P421	GDC55	RHR Reactor Shutdown Cooling Suction	Water	20	Yes	20	E12F008	O	No ⁽²⁷⁾	14'-0"	Gate	EM	E	M	CL	OP	CL	AI	U,A,M,RM _s	<33	1	Out
			Water	20	Yes	20	E12F009	I	No ⁽²⁷⁾	NA	Gate	EM	E	M	CL	OP	CL	AI	U,A,M,RM _s	<33	2	Out
			Water	3/4	Yes	20	E12F550	I	No ⁽²⁷⁾	NA	Chk	P	P	-	CL	CL	CL	-	Rev. Flow	-	-	Out
P422	GDC55	RCIC Steam Supply	Steam	10	Yes ⁽²³⁾	1 (c)	E51F063	I	Yes	NA	Gate	EM	E	M	OP	CL	OP or CL	AI	J,F,K,M,T,RM _s , V,Q	20	2	Out
			Steam	10	Yes ⁽²³⁾	1 (c)	E51F064	O	Yes	13'-2"	Gate	EM	E	M	OP	CL	OP or CL	AI	J,F,K,M,T,RM _s , V,Q	20	1	Out
			Steam	1	Yes ⁽²³⁾	1 (c)	E51F076	I	Yes	NA	Globe	EM	E	M	CL	CL	CL	AI	J,F,K,M,T,RM _s , V,Q	Std.	2	Out

TABLE 6.2-32 (Continued)

Penetration No. ⁽³⁾	GDC/ Reg. Guide	System Number	Fluid	Line Size (in)	Essential System ⁽⁴⁾	Figure 6.2-60 Arr. No.	System and Valve Number	Loc ⁽⁵⁾	Type C Test	Pipe Length ⁽⁶⁾	Valve		Actuation Mode		Valve Position			Isolation Signal ⁽⁸⁾	Closure Time (sec) ⁽⁹⁾	Power Source 1E Bus ⁽¹⁰⁾	Norm. Flow Dir.		
											Type	Oper.	Pri.	Sec.	Norm	Shut down	Post Acc.					Power Fail ⁽⁷⁾	
P423	GDC55	Main Steam Line Drain	Water	3	No	1 (b)	B21F019	O	Yes	13'-0"	Gate	EM	E	M	OP	CL	CL	AI	C, E, F, S, N, P, RM _c , RM _f	25	1	Out	
			Water	3	No	1 (b)	B21F016	I	Yes	NA	Gate	EM	E	M	OP	CL	CL	AI	C, E, F, S, N, P, RM _c	25	2	Out	
P424	GDC55	RWCU to Main Condenser and Radwaste	Water	4	No	14	G33F034	O	Yes	8'-6"	Gate	EM	E	M	CL	CL	CL	AI	L, B, F, H, RM _c	20 ⁽²⁵⁾	1	Out	
			Water	4	No	14	G33F028	I	Yes	NA	Gate	EM	E	M	CL	CL	CL	AI	L, B, F, H, RM _c	20 ⁽²⁵⁾	2	Out	
			Water	1/4	No	14	G33F0646	I	Yes	NA	Rel	P	P	-	CL	CL	OP or CL	-	-	-	-		
P425	RG1.11	Combustible Gas Control Postaccident Hydrogen Analyzer Drywell Atmos. Sup. Pool Atmos. Analyzer Exhaust	Cont. Atmos. Drywell	1/2	Yes	58 (a)	M51F210A	O	Yes	<10'	Globe	S	E	-	CL	CL	OP	AI	RM _c	3	1	Out	
			Atmos.	1/2	Yes	58 (a)	M51F220A	O	Yes	<10'	Globe	S	E	-	CL	CL	OP	AI	RM _c	3	1	Out	
				1/2	Yes	58 (a)	M51F230A	O	Yes	<10'	Globe	S	E	-	CL	CL	OP	AI	RM _c	3	1	Out	
				1/2	Yes	58 (a)	M51F240A	O	Yes	<10'	Globe	S	E	-	CL	CL	OP	AI	RM _c	3	1	Out	
	RG1.11	Containment Atmosphere Monitoring	Cont. Atmos.	3/4	Yes ⁽²³⁾	38 (b)	D23F050	O	No	<10'	Globe	S	E	-	OP	OP	OP	AI	RM _c	3	3	-	
			Cont. Atmos.	3/4	Yes ⁽²³⁾	38 (b)	1/4" orifice	I	No	NA	-	-	-	-	-	-	-	-	-	-	-		
P428	GDC56	Containment Vacuum Relief	Atmos.	24	Yes	19	M17F035	O	Yes	2'-6"	B' fly	EM	E	M	OP	OP	CL	AI	B, G, X, RM _c	5	2	In	
			Atmos.	24	Yes	19	M17F030	I	Yes	NA	Chk	P	V	-	CL	CL	CL	-	Rev. Flow	-	-	In	
P429	GDC56	RHR Relief Line to Suppression Pool	Water	2	Yes	37	E12F025A	O	Yes	29'-6"	Rel	P	P	-	CL	CL	CL	AI	-	-	-	In	
			Water	2	Yes	57	E12F025B	O	Yes	45'-6"	Rel	P	P	-	CL	CL	CL	AI	-	-	-	In	
			Water	2	Yes	57	E12F025C	O	Yes	39'-6"	Rel	P	P	-	CL	CL	CL	AI	-	-	-	In	
			Water	6	Yes	37	E12F055A	O	Yes	77'-3"	Rel	P	P	-	CL	CL	CL	AI	-	-	-	In	
			Water	6	Yes	57	E12F055B	O	Yes	92'-6"	Rel	P	P	-	CL	CL	CL	AI	-	-	-	In	
			Water	1	Yes	57	E12F005	O	Yes	91'-9"	Rel	P	P	-	CL	CL	CL	AI	-	-	-	In	
			Cont. Atmos.	1	Yes ⁽²³⁾	37	N27D006	O	No ⁽¹⁶⁾	NA	-	-	-	-	-	-	-	-	-	-	-	In	
			Water	2	Yes	37	E21F018	O	Yes	73'-1"	Rel	P	P	-	CL	CL	CL	AI	-	-	-	In	
			Water	1	Yes ⁽²³⁾	37	N27F751	O	Yes	NA	Globe	M	M	-	LC	LC	LC	AI	-	-	-	In	
			Cont. Atmos.	12	Yes ⁽²³⁾	10 (b)	E51F068	O	Yes	NA	Gate	EM	E	M	OP	OP	CL	AI	RM _s , J, G	Std.	1	In	
			Water	1/2	No	57	P87F083	O	Yes	12'	Globe	S	E	-	CL	CL	OP or CL	FC	RM _p	<3	-	In	
			Water	1/2	No	57	P87F264	O	Yes	NA	Globe	S	E	-	CL	CL	OP or CL	FC	RM _p	<3	-	In	
			Cont. Atmos.																				
			Water, Steam	1-1/2	Yes	10 (a)	E12F102	O	Yes	221'-8"	Globe	M	M	-	LC	LC	LC	-	-	-	-	-	-
			Cont. Atmos.	1	Yes ⁽²³⁾	37	E12D015A	O	No ⁽¹⁶⁾	NA	-	-	-	-	-	-	-	-	-	-	-	-	In
			Cont. Atmos.	1	Yes ⁽²³⁾	57	E12D015B	O	No ⁽¹⁶⁾	NA	-	-	-	-	-	-	-	-	-	-	-	-	In
P431	GDC56	RHR Heat Exchanger Vent to Suppression Pool SDC Header Leak-off	Noncondens., Steam	1	Yes	45	E12F073B	O	Yes	12'-9"	Globe	EM	E	M	OP	CL	CL	AI	RM _c , A, G, M	Std.	2	In	
				1	Yes	45	E12F558B	I	Yes	NA	Chk	P	P	-	CL	CL	CL	-	Rev. Flow	-	-	In	
P433	RG1.11	Containment Atmosphere Monitoring System	Cont. Atmos.	3/4	Yes ⁽²³⁾	38 (b)	D23F030A	O	No	<10'	Globe	S	E	-	OP	OP	OP	AI	RM _c	<3	1	-	
			Cont. Atmos.	3/4	Yes ⁽²³⁾	38 (b)	1/4" orifice	I	No	NA	-	-	-	-	-	-	-	-	-	-	-		
			Drywell Atmos.	3/4	Yes ⁽²³⁾	38 (a)	D23F040A	O	No	<10'	Globe	S	E	-	OP	OP	OP	AI	RM _c	<3	1	-	
			Drywell Atmos.	3/4	Yes ⁽²³⁾	38 (a)	1/4" orifice	I	No	NA	-	-	-	-	-	-	-	-	-	-	-		
P434	RG1.11	Containment Atmosphere Monitoring	Cont. Atmos.	3/4	Yes ⁽²³⁾	38 (b)	D23F010A	O	No	<10'	Globe	S	E	-	OP	OP	OP	AI	RM _c	<3	1	-	
			Cont. Atmos.	3/4	Yes ⁽²³⁾	38 (b)	1/4" orifice	I	NO	NA	-	-	-	-	-	-	-	-	-	-	-		
			Cont. Atmos.	3/4	Yes ⁽²³⁾	38 (b)	D23F020A	O	No	<10'	Globe	S	E	-	OP	OP	OP	AI	RM _c	<3	1	-	
			Cont. Atmos.	3/4	Yes ⁽²³⁾	38 (a)	1/4" orifice	I	No	NA	-	-	-	-	-	-	-	-	-	-	-		
	GDC56	Containment Vaccum Relief	Cont. Atmos.	3/4	Yes ⁽²³⁾	59	M17F055	O	No	<10'	Globe	S	E	-	OP	OP	OP	AI	RM _c	<3	1	-	
			Cont. Atmos.	3/4	Yes ⁽²³⁾	59	1/4" orifice	I	No	NA	-	-	-	-	-	-	-	-	-	-	-		
P436	GDC56	Containment Vaccum Relief	Atmos.	24	Yes	19	M17F045	O	Yes	2'-6"	B' fly	EM	E	M	OP	OP	CL	AI	B, G, X, RM _c	5	2	In	
			Atmos.	24	Yes	19	M17F040	I	Yes	NA	Chk	P	V	-	CL	CL	CL	NA	Rev. Flow	-	-	In	

TABLE 6.2-32 (Continued)

NOTES:

- (1) Through line leakage classification is discussed in <Section 6.2.3>.
- (2) Abbreviations used are as follows:

A	- Air	LPCS	- Low Pressure Core Spray System
ADS	- Automatic Depressurization System	M	- Manual
AI	- As is	NA	- Not applicable
B'fly	- Butterfly valve	OP	- Open
Chk	- Check valve	P	- Process fluid
Cl	- Closed	RCIC	- Reactor Core Isolation Cooling System
E	- Electric	Rel	- Relief Valve
EH	- Electrohydraulic	RHR	- Residual Heat Removal System
EM	- Electric Motor	RWCU	- Reactor Water Cleanup System
FC	- Fail Closed	S	- Solenoid
H	- Hydraulic	SLC	- Standby Liquid Control System
HPCS	- High Pressure Core Spray System	SP	- Spring
LC	- Locked or Sealed in Closed Position	V	- Vacuum in containment
LPCI	- Low Pressure Coolant Injection System		

- (3) Penetrations not listed are spares and are capped, except penetration P202 which is the equipment hatch.
- (4) Essential systems are engineered safety feature systems which are required for shutdown.
- (5) Location inside (I) or outside (O) of containment.
- (6) Length of pipe from containment to outermost isolation valve.
- (7) All motor-operated isolation valves remain in last position upon failure of valve power. All air operated valves close upon loss of motive air.
- (8) Remote-manual (RM) valves can be opened or closed by remote-manual switch operation during any mode of reactor operation, except when an automatic signal is present. All remote-manual valves have position indicator lights at the remote-manual switch, and a subset of these valves have an additional set of position indicating lights at the control room isolation status panel. Isolation signals are defined as follows:

Signal	Description
A	Reactor vessel low water level - Level 3. (A scram occurs at this level. This is the highest of the three isolation low water level signals.)
B	Reactor vessel low water level - Level 2. (This is the second of the three low water level signals. The reactor core isolation cooling and high pressure core spray systems are activated at this level.)
C	Reactor vessel low water level - Level 1. (This is the lowest of the three water level signals. Main steam line isolation occurs at this level. The low pressure core spray and low pressure coolant injection systems are also activated at this level.)
D	Spare.
E	Line break - main steam line (steam line high steam flow).
F	Line break - main steam line (main steam line tunnel high space ambient high temperature).
G	High drywell pressure.
H	Line break in reactor water cleanup system (high ambient high temperature).
J	Line break in the steam supply line to reactor core isolation cooling system (low steam line pressure).
K	Line break in reactor core isolation cooling system steam line to turbine (high steam flow).
L	High differential flow in the reactor water cleanup system.
M	Line break in residual heat removal system (high ambient high temperature).
N	Low main condenser vacuum.
P	Low main steam line pressure at inlet to turbine (RUN mode, only).
Q	Line break in reactor core isolation cooling system (high ambient).
S	High main steam line temperature, turbine building.

TABLE 6.2-32 (Continued)

NOTES: (Continued)

<u>Signal</u>	<u>Description</u>
T	High pressure reactor isolation cooling turbine exhaust diaphragm.
U	High reactor vessel pressure - close residual heat removal - shutdown cooling valves and head cooling valves.
V	Line break in the steam supply line to reactor core isolation cooling system (high steam flow).
W	High temperature at outlet of cleanup system nonregenerative heat exchanger.
X	Containment to atmosphere differential pressure greater than 0.0 psid.
Y	Standby liquid control system actuated.
Z	High radiation, containment and drywell ventilation exhaust.
RM _c	Remote-manual switch from control room. (All automatically actuated containment isolation valves are capable of remote operation from the control room.)
RM _f	Remote-manual switch from motor control center. (Provided for control room isolation and remote shutdown of the valve. In case a fire occurs inside the control room per <10 CFR 50, Appendix R>, Method A.
RM _s	Remote-manual switch from shutdown panel. (Provided in addition to RM _c , noted above, on selected valves as indicated.)
RM _p	Remote-manual switch from Postaccident Sampling System panel. Switches are keylocked closed.
AA	Containment spray initiated.
BB	Injection flow above setpoint (minimum flow valve control).
CC	Discharge pressure below setpoint (minimum flow valve control).
DD	Reactor vessel high water level - Level 8.
EE	E51F510 closes.
FF	E51F045 closes.
GG	High drywell atmospheric radiation

⁽⁹⁾ Standard (Std) closure time, based upon nominal pipe diameter, is approximately 12 inches/minute for gate valves and approximately 4 inches/minute for globe valves. The standard closure time for butterfly valves is 30 to 60 seconds.

⁽¹⁰⁾ AC motor-operated valves required for isolation functions are powered from the ac standby power buses. DC operated isolation valves are powered from the batteries.

⁽¹¹⁾ Testable check valves are designed for remote opening with zero differential pressure across the valve seat. The valves close under reverse flow conditions, even if the test switch is positioned to open. The valves open when pump pressure exceeds reactor pressure, even if the test switch is positioned to close.

⁽¹²⁾ (Deleted)

⁽¹³⁾ Main steam line isolation valves require that both solenoid pilots be de-energized to close. Accumulator air pressure plus spring act to close valves when both pilots are de-energized. Voltage failure at only one pilot does not cause valve closure. These valves are designed to close fully in 2.5 to 5 seconds <Section 5.4.5.3>.

⁽¹⁴⁾ During reactor operation, a blind flange is installed on the outboard end of the transfer tube as the containment boundary. The IFTS containment isolation assembly receives a Type B test. When the blind flange is removed in Mode 1, 2, or 3, the containment boundary inside the containment building is made by the remaining portion of the transfer tube containment isolation assembly, containment bellows, and steel containment penetration, and outside the containment building by the transfer tube, drain line, drain valve, and local leak rate test valve. The portion of the tube outside of the blind flange receives a Type C test out to the drain line valve 1F42-F003. Opening of the drain line valve during Mode 1, 2, or 3 with the blind flange removed is addressed by administrative controls. These require a designated individual to take manual action to close the valve in the event of an accident.

⁽¹⁵⁾ (Deleted)

⁽¹⁶⁾ This receives a Type B test.

⁽¹⁷⁾ (Deleted)

⁽¹⁸⁾ (Deleted)

⁽¹⁹⁾ The ECCS response time requirement for these injection valves is met with the valves partially opened. Times shown in this table are for full (100%) valve strokes.

⁽²⁰⁾ Division 3 power is also available by postaccident operator manual action (procedurally controlled).

⁽²¹⁾ The feedwater check valves utilize an alternate non-type water leak testing methodology per the Inservice Testing Program to verify their proper closure.

⁽²²⁾ "Location" (see Note 5) of these valves inside (I) or outside (O) of containment varies depending on whether the inner or outer airlock door is open and/or inoperable.

⁽²³⁾ Non-ESF systems that would be desirable to use to mitigate the consequence of an accident.

⁽²⁴⁾ Applicable in Operational Modes 1, 2 and 3.

⁽²⁵⁾ Time delay associated with these valves. Maximum time to close (stroke time and delay) is 60 seconds or less.

⁽²⁶⁾ P51F0530 is a 2.0" check valve installed in the 2.5" line via reducers.

⁽²⁷⁾ This valve has been excluded from Type C testing since it does not constitute a potential primary containment atmospheric pathway during and following a Design Basis Accident (DBA).

TABLE 6.2-33

POTENTIAL SECONDARY CONTAINMENT BYPASS LEAKAGE PATHS^{(1) (2) (4) (5) (9) (10) (14)}

Primary Containment Penetration No.	Description	Valve No.	Valve Type	Loc.	Line Size (in.)	Bypass Leakage Barrier ⁽³⁾	Expected Air Leakage Rate (SCCM)
P106	RCIC Turbine	E51-F0068	Gate	O	12.00	10 (b)	190.0
	Exhaust, RCIC	E51-F0040	Noz.Chk.	O	10.00	10 (b)	2.0
	Turbine Exhaust						
	Vacuum Relief, and	E12-F102	Glb.	O	1.5	10 (a)	23.8
	RHR A&B Relief						
	Valve Discharge to	N27-F751	Glb.	O	1.0	37	0.08
	Suppression Pool	P87-F083	Glb.	O	0.50	57	0.0
		P87-F264	Glb.	O	0.50	57	0.0
		See Note ⁽¹⁵⁾					
P108	Condensate Supply	P11-F060	Btf.	O	12.00	27 (a)	0.63
		P11-F545	Chk.	I	12.00	27 (a)	384.48
P109	ILRT Blowdown Line	See Note ⁽⁶⁾	-	-	8.00	35	0.00
P111	Condensate Return	P11-F080	Btf.	O	10.00	27 (b)	0.52
		P11-F090	Btf.	I	10.00	27 (b)	0.52
		P11-F629	Relief	I	0.50	27 (b)	See Note ⁽¹⁷⁾
P114	Containment Vacuum Relief	M17-F015	Btf.	O	24.00	19	3.60
		M17-F010	Chk.	I	24.00	19	3.40
			O-rings	I	24.00	19	See Note ⁽¹³⁾
P115	See P106						
P117	Nitrogen Supply to CRD	P86-F002	Glb.	O	2.00	41	64.88
		P86-F528	Chk.	I	2.00	41	64.88
P119	ILRT Pressure Indicating Line	See Note ⁽⁶⁾	-	-	0.50	35	0.00

TABLE 6.2-33 (Continued)

<u>Primary Containment Penetration No.</u>	<u>Description</u>	<u>Valve No.</u>	<u>Valve Type</u>	<u>Loc.</u>	<u>Line Size (in.)</u>	<u>Bypass Leakage Barrier⁽³⁾</u>	<u>Expected Air Leakage Rate (SCCM)</u>
P120	ILRT Pressurization Line	See Note ⁽⁶⁾	-	-	8.00	35 (b)	0.00
P422	RCIC Turbine Supply	E51-F0063	Gate	I	10.00	1 (c)	158.3
		E51-F0064	Gate	O	10.00	1 (c)	158.3
		E51-F0076	Glb.	I	1.00	1 (c)	15.8
P131	RWCU Pump Suction	G33-F001	Gate	I	6.00	49	192.24
		G33-F004	Gate	O	6.00	49	192.24
P201	Drywell Atm. Rad. Monitor Line	D17-F079A	Glb.	O	1.00	52	0.58
		D17-F079B	Glb.	I	1.00	52	0.58
		D17-F071A	Ball	O	1.00	52	0.06
		D17-F071B	Ball	I	1.00	52	0.06
P203	Fuel Pool Cooling Supply	G41-F100	Btf.	O	8.00	26 (a)	0.42
		G41-F522	Chk.	I	8.00	26 (a)	256.32
P204	CRD Supply	C11-F083	Gate	O	2.50	4	80.10
		C11-F122	Chk.	I	2.50	4	80.10
P205	Fuel Transfer Tube	See Note ^{(6) (16)}	-	-	-	36	0.00
		F42-F003	Ball	O	4.00	36	0.00
P208	Containment Vacuum Relief	M17-F025	Btf.	O	24.00	19	3.60
		M17-F020	Chk.	I	24.00	19	3.40
		O-rings		I	24.00	19	See Note ⁽¹³⁾
P210	CO ² to Fire Protection System	P54-F340	Gate	O	4.00	42	128.16
		P54-F1098	Chk.	I	4.00	42	128.16

TABLE 6.2-33 (Continued)

Primary Containment Penetration No.	Description	Valve No.	Valve Type	Loc.	Line Size (in.)	Bypass Leakage Barrier ⁽³⁾	Expected Air Leakage Rate (SCCM)
P301	Fuel Pool Cooling Return	G41-F145	Btf.	O	10.00	26 (b)	0.52
		G41-F140	Btf.	I	10.00	26 (b)	0.52
		G41-F801	Chk.	I	0.75	26 (b)	See Note ⁽¹⁷⁾
P305	Containment Personnel Air Locks ⁽⁸⁾	P53-F536/ F570	Glb.	O ⁽¹⁸⁾ / O	1.00	56	32.04
		P53-F579A	Ball	O	1.00	56	—
		P53-F579B	Ball	I ⁽¹⁸⁾	1.00	56	—
		P53-F580A	Ball	O ⁽¹⁸⁾	1.00	56	—
		P53-F580B	Ball	I	1.00	56	—
		P53-F581	Relief	I ⁽¹⁸⁾	1.00	56	—
		P53-F582	Relief	I	1.00	56	—
		P53-F010	Glb.	O	0.75	—	—
		P53-F015	Glb.	O ⁽¹⁸⁾	0.75	—	—
		P53-F070	Glb.	O	0.75	—	—
		P53-F556	Needle	I	0.5	—	—
P306	Instrument Air	P52-F200	Glb.	O	2.00	34	64.88
		P52-F550	Chk.	I	1.50	34	15.00
P308	Service Air	P51-F150	Glb.	O	2.50	32	0.13
		P51-F530 ⁽¹⁹⁾	Chk.	I	2.50	32	80.10
P309	Demineralized Water	P22-F010	Gate	O	3.00	29	96.12
		P22-F577	Chk.	I	3.00	29	96.12
P310	NCC Water Supply	P43-F055	Btf.	O	12.00	25 (a)	0.63
		P43-F721	Chk.	I	12.00	25 (a)	384.48

TABLE 6.2-33 (Continued)

Primary Containment Penetration No.	Description	Valve No.	Valve Type	Loc.	Line Size (in.)	Bypass Leakage Barrier ⁽³⁾	Expected Air Leakage Rate (SCCM)
P311	NCC Water Return	P43-F140	Btf.	O	12.00	25 (b)	0.63
		P43-F215	Btf.	I	12.00	25 (b)	0.63
		P43-F851	Relief	I	0.25	25 (b)	See Note ⁽¹⁷⁾
P312	Containment Personnel Air Locks ⁽⁸⁾	P53-F541/ F571	Glb.	O ⁽¹⁸⁾ / O	1.00	56	32.04
		P53-F593A	Ball	O	1.00	56	—
		P53-F593B	Ball	I ⁽¹⁸⁾	1.00	56	—
		P53-F594A	Ball	O ⁽¹⁸⁾	1.00	56	—
		P53-F594B	Ball	I	1.00	56	—
		P53-F595	Relief	I ⁽¹⁸⁾	1.00	56	—
		P53-F596	Relief	I	1.00	56	—
		P53-F020	Glb.	O	0.75	—	—
		P53-F025	Glb.	O ⁽¹⁸⁾	0.75	—	—
		P53-F075	Glb.	O	0.75	—	—
		P53-F561	Needle	I	0.5	—	—
V313	Containment Purge Supply	M14-F040	Btf.	O	42.00	30 (a)	6.30
		M14-F045	Btf.	I	42.00	30 (a)	6.30
		M14-F190	Btf.	I	18.00	30 (a)	2.70
		M14-F195	Btf.	I	18.00	30 (a)	2.70
V314	Containment Purge Exhaust	M14-F090	Btf.	O	42.00	30 (b)	6.30
		M14-F085	Btf.	I	42.00	30 (b)	6.30
		M14-F200	Btf.	I	18.00	30 (b)	2.70
		M14-F205	Btf.	I	18.00	30 (b)	2.70
P315	Standby Liquid Control	C41-F518	Glb.	O	2.00	60	0.16
		C41-F520	Chk.	I	2.00	60	0.16

TABLE 6.2-33 (Continued)

Primary Containment Penetration No.	Description	Valve No.	Valve Type	Loc.	Line Size (in.)	Bypass Leakage Barrier ⁽³⁾	Expected Air Leakage Rate (SCCM)	
P317	Containment Radia- tion Monitoring Supply and Return	D17-F089A	Glb.	O	1.00	54	0.00	
		D17-F089B	Glb.	I	1.00	54	0.00	
		D17-F081A	Ball	O	1.00	54	0.00	
		D17-F081B	Ball	I	1.00	54	0.00	
	ILRT Instrumentation							
	See Note ⁽⁶⁾							
			E61-F549	Glb.	I	0.50	16.02	
			E61-F550	Glb.	I	0.75	24.03	
	P318	Postaccident Sampling	P87-F065	Sol. Glb.	O	0.50	58 (b)	0.43
			P87-F071	Sol. Glb.	O	0.50	58 (b)	0.43
P87-F074			Sol. Glb.	O	0.50	58 (b)	0.43	
P87-F077			Sol. Glb.	O	0.50	58 (b)	0.43	
P319	ILRT Instrumentation	E61-F551	Glb.	I	0.75		24.03	
		E61-F552	Glb.	I	0.50		16.02	
		See Note ⁽⁶⁾						
P401	Postaccident Sampling	P87-F037	Sol. Glb.	O	0.75	55	See Note ⁽¹²⁾	
P404	Chilled Water Supply	P50-F060	Btf.	O	6.00	28 (a)	0.32	
		P50-F539	Chk.	I	6.00	28 (a)	192.24	
P405	Chilled Water Return	P50-F150	Btf.	O	6.00	28 (b)	0.32	
		P50-F140	Btf.	I	6.00	28 (b)	0.32	
		P50-F606	Relief	I	0.25	28 (b)	See Note ⁽¹⁷⁾	
P406	Fire Protection Water	P54-726	Gate	O	4.00	51	128.16	
		P54-727	Gate	I	4.00	51	128.16	

TABLE 6.2-33 (Continued)

Primary Containment Penetration No.	Description	Valve No.	Valve Type	Loc.	Line Size (in.)	Bypass Leakage Barrier ⁽³⁾	Expected Air Leakage Rate (SCCM)
P413	Postaccident Sampling	P87-F049	Sol. Glb.	I	0.75	61	0.43
		P87-F055	Sol. Glb.	O	0.75	61	0.43
		P87-F046	Sol. Glb.	I	0.75	61	0.43
		P87-F052	Sol. Glb.	O	0.75	61	0.43
		P87-F277	Relief	I	0.25	61	0.0
P417	Drywell and Containment Equip Sump to Radwaste	G61-F080	Gate	O	3.00	40	96.12
		G61-F075	Gate	I	3.00	40	96.12
		G61-F0655	Check	I	0.75	40	0.0
P418	Drywell and Containment Floor Drain Sump to Radwaste	G61-F170	Gate	O	3.00	43	96.12
		G61-F165	Gate	I	3.00	43	96.12
		G61-F657	Chk.	I	0.75	43	See Note ⁽¹⁷⁾
P419	RWCU Pump Discharge	G33-F054	Gate	O	4.00	48	128.16
		G33-F053	Gate	I	4.00	48	128.16
P420	RWCU Backwash Transfer Pump to Radwaste	G50-F277	Gate	O	4.00	46	128.16
		G50-F272	Gate	I	4.00	46	128.16
		G50-F823	Relief	I	0.50	46	See Note ⁽¹⁷⁾
P423	Main Steam Line Drain	B21-F019	Gate	O	3.00	1 (b)	90.78
		B21-F016	Gate	I	3.00	1 (b)	90.78
P424	RWCU to Main Condenser and Radwaste	G33-F034	Gate	O	4.00	14	128.16
		G33-F028	Gate	I	4.00	14	128.16
		G33-F0646	Relief	I	0.25	14	0.0

TABLE 6.2-33 (Continued)

<u>Primary Containment Penetration No.</u>	<u>Description</u>	<u>Valve No.</u>	<u>Valve Type</u>	<u>Loc.</u>	<u>Line Size (in.)</u>	<u>Bypass Leakage Barrier⁽³⁾</u>	<u>Expected Air Leakage Rate (SCCM)</u>
P428	Containment Vacuum Relief	M17-F035	Btf.	O	24.00	19	3.60
		M17-F030	Chk.	I	24.00	19	3.40
		O-rings		I	24.00	19	See Note ⁽¹³⁾
P429	RHR A&B Relief Valve Discharge to Suppression Pool, RCIC Turbine Exhaust and RCIC Pump Turbine Exhaust Vacuum Relief						
		E12-F102	Glb.	O	1.5	10 (a)	23.8
		E51-F068	Gate	O	12.00	10 (b)	190.0
		N27-F751	Glb.	O	1.0	37	0.08
		P87-F083	Glb.	O	0.50	57	0.0
		P87-F264	Glb.	O	0.50	57	0.0
		See Note ⁽¹⁵⁾					
P436	Containment Vacuum Relief	M17-F045	Btf.	O	24.00	19	3.60
		M17-F040	Chk.	I	24.00	19	3.40
		O-rings		I	24.00	19	See Note ⁽¹³⁾

TABLE 6.2-33 (Continued)

	Size of Line Penetrating Primary <u>Containment (in.)</u>	<u>Termination Region</u>	<u>Bypass Leakage Barrier</u>	<u>Expected Leakage Rate (SCCM)</u>
<u>Lines Enclosed by Guard Pipes</u>				
Main Steam	38	Environment	Welded Conn.	0
Main Steam Drains	14	Environment	Welded Conn.	0
Feedwater	32	Environment	Welded Conn.	0
Reactor Core Isolation Cooling Steam Supply	22	Environment	Welded Conn.	0
Reactor Core Isolation Cooling Pump Discharge and Residual Heat Removal Head Spray	18	Environment	Welded Conn.	0
Reactor Water Cleanup Pump Suction	18	Environment	Welded Conn.	0
Low Pressure Core Spray Pump Suction from Suppression Pool	36	Environment	Welded Conn.	0
High Pressure Core Spray Pump Suction from Suppression Pool	36	Environment	Welded Conn.	0
Residual Heat Removal A, B and C Suction from Suppression Pool	36	Environment	Welded Conn.	0
Reactor Core Isolation Cooling Suction from Suppression Pool	18	Environment	Welded Conn.	0
Residual Heat Removal Shutdown Supply Suction	32	Environment	Welded Conn.	0
Reactor Water Cleanup Pump Discharge	16	Environment	Welded Conn.	0
Reactor Water Cleanup from Regenerative Heat Exchanger to Feedwater	16	Environment	Welded Conn.	0
<u>Personnel Airlocks</u> ⁽⁸⁾	9'-7"	Environment	Resilient Seals	0

TABLE 6.2-33 (Continued)

NOTES:

- (1) A Technical Specification commitment of 10.08 percent of total design containment leakage is made for the maximum test leakage bypassing the containment annulus exhaust gas treatment system (for leakage sources in lines penetrating primary containment and personnel airlocks).
- (2) Expected bypass water leakage sources from components that circulate core cooling water following LOCA:
- a. Pumps - expected total leakage of 5 gal./day from residual heat removal, high pressure core spray, and low pressure core spray pumps.
 - b. Valves - expected total valve stem leakage of 300 cc/hr from systems handling reactor fluid outside containment.
 - c. FW-LCS, ECCS and RCIC Branch Lines - expected total leakage of 1.663 gal./hr.
- (3) Bypass leakage barrier arrangement is shown on the designated detail of <Figure 6.2-60>.
- (4) Closed Systems Outside Containment

Piping systems which penetrate containment and are closed outside containment are tested for potential water leakage under the guidance of <NUREG-0737> Item III.D.1.1. The expected bypass water leakage from these systems is identified in Note 2 above. The limit on bypass water leakage is identified in <Section 15.6.5.5.1.2.b>. Valves in closed systems outside containment, which are potential air leakage sources, are identified in the text of <Table 6.2-33>.

The redundant containment isolation provisions for each penetration consist of an isolation valve and a closed system outside containment which are in compliance with <10 CFR 50, Appendix A>, Criteria 54, and with GDCs 55 and 56, utilizing the "other defined basis" provision. The closed system is missile protected, Seismic Category I, Safety Class 2, and has a temperature and pressure rating in excess of that for containment.

TABLE 6.2-33 (Continued)

NOTES: (Continued)

The following penetrations lead to closed systems outside containment:

P101, P102, P103, P104, P105, P107, P112, P113, P118, P121^(a), P123, P132, P401, P402, P403, P407, P408, P409, P410, P411, P412, P414^(a), P421, P429, P431.

NOTE:

^(a) These feedwater penetrations have a branch line leading to the RHR Shutdown Cooling Return Line and the RWCU Return Line:

- The RHR Shutdown Cooling Return Line leads to RHR, which is considered a closed system outside containment. The branch line leading to RHR is examined for external leakage from mechanical joints (e.g., valve stem or bonnet, seals or gaskets).
- The RWCU Return Line leads back into the containment, and is tested like a closed system outside containment, with the specific acceptance criteria that leakage exterior to the piping will be eliminated.
- See <Table 6.2-40> for testing details on containment penetrations.

⁽⁵⁾ Instrument Lines

Instrument lines penetrating containment are assumed to allow zero bypass leakage. Valves in instrument lines penetrating containment are open post-LOCA in order to fulfill the instruments' functions. The instruments are designed to allow zero bypass leakage. Penetrations containing instrument lines appear below:

P102, P318, P320, P401, P402, P425, P433, P434

TABLE 6.2-33 (Continued)

NOTES: (Continued)

(6) Spectacle or Blind Flanges

Penetrations which contain lines isolated by spectacle or blind flanges are assumed to allow zero bypass leakage. Spectacle and blind flanges are local leak rate tested. Penetrations whose lines are isolated by spectacle or blind flanges appear below:

Unit 1: P109, P119, P120, P205, P317, P319

Unit 2: P428, P429, P427, P304

(7) Leakage Control System

The feedwater system has a dedicated leakage control system <Section 6.9> which provides seal water to the bonnet, stem, and seat of the outboard gate valves (B21-F065A/B). Water leakage from the Feedwater Leakage Control System (FWLCS) piping, and from the bonnet, stem, and seat of the outboard gate valves, are controlled under the Primary Coolant Sources Outside Containment Program, Technical Specification 5.5.2. Outboard gate valve bonnet and stem leakage identified during the system walkdown at pressures >1000 psig will be eliminated. The gate valves will be closed during this check. Gate valve seat water leakage measured at $\geq 1.1 P_a$ will be limited by the Program, which restricts allowable leakages to half of that assumed in the dose calculations <Section 15.6.5.5.1.2.b>. This water leakage is not added into the secondary containment bypass air leakage totals. The FWLCS seals the feedwater lines going through the following penetrations:

Unit 1: P121, P414

Unit 2: P112, P410

The leakage control systems meet single failure criteria, are missile protected, Seismic Category I, Safety Class 2, and have temperature and pressure ratings in excess of that for the containment.

TABLE 6.2-33 (Continued)

NOTES: (Continued)

- ⁽⁸⁾ The personnel airlock door seals are not considered a bypass leakage path when the outer door is operable because all leakage through the outer door seals is routed back into the annulus area between the annulus shield wall and containment where it is treated by the Annulus Exhaust Gas Treatment System.

All direct leakage paths associated with the barrel/doors which may bypass the outer door seals under the various modes of operation/availability are assigned as bypass.

P305, P312

- ⁽⁹⁾ The air supply lines to the ADS accumulators and personnel air lock door seal accumulators are not considered bypass leakage paths since they are safety-related lines that remain pressurized post-LOCA (at a pressure exceeding containment vessel design pressure.):

P116, P304, P305, P312

- ⁽¹⁰⁾ The following penetrations contain lines from more than one system, and therefore, may appear more than once in this table and in its notes:

P102, P305, P312, P317, P319, P320, P401, P402, P425, P434

- ⁽¹¹⁾ (Deleted)

- ⁽¹²⁾ If valves leak, they will leak water. Expected leakage: none.

TABLE 6.2-33 (Continued)

NOTES: (Continued)

- ⁽¹³⁾ It is necessary to break O-ring seals to test the isolation valves for this penetration. After the valve test, the O-rings will be reinstalled and Type B tests performed. If the O-rings leak, they will be fixed to eliminate leakage.
- ⁽¹⁴⁾ The Alternate Hydrogen Purge System line (P302) is not a bypass leakage path because it ties into the Annulus Exhaust Gas Treatment System.
- ⁽¹⁵⁾ Leakage from test spacers E12D015A & E12D015B are considered bypass leakage.
- ⁽¹⁶⁾ The IFTS penetration (P205) is not normally considered a bypass leakage path because all leakage through the bellows and bellows assembly flange joints, is routed to the annulus area between the shield wall and containment where it is treated by the annulus exhaust gas treatment system. Any leakage into the IFTS tube is still considered a potential secondary containment bypass leakage path. With the blind flange removed, the bypass leakage path is the IFTS tube and the drain line and drain valve, 1F42-F003.
- ⁽¹⁷⁾ Leakage limits established per the ASME OM Code IST Program.
- ⁽¹⁸⁾ "Location" (see column title) of these valves inside (I) and outside (O) of containment varies depending on whether the inner or outer air lock door is open and/or inoperable.
- ⁽¹⁹⁾ P51-F0530 is a 2.0" valve installed in the 2.5" line via reducers.

REACTOR COOLANT PRESSURE BOUNDARY INFLUENT LINES THAT PENETRATE CONTAINMENT/DRYWELL

NOTES :

GP - Guard pipe
MOV - Motor-operated valve
TCV - Testable check valve

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TABLE 6.2-35

REACTOR COOLANT PRESSURE BOUNDARY EFFLUENT LINES THAT PENETRATE CONTAINMENT/DRYWELL

<u>Effluent Line</u>	<u>Isolation Devices(s) ⁽¹⁾</u>			<u>Reference Section</u>
	<u>Inside Drywell</u>	<u>Between Drywell and Containment</u>	<u>Outside Containment</u>	
Main Steam Lines	AOV	GP	AOV	<Section 6.2.4.2.2.1.b.1>
Main Steam Drains	MOV	GP	MOV	<Section 6.2.4.2.2.1.b.1>
Reactor Core Isolation Cooling Steam Supply	MOV	GP	MOV	<Section 6.2.4.2.2.1.b.1>
Reactor Water Cleanup to Pumps	MOV	GP	MOV	<Section 6.2.4.2.2.1.b.2>
Reactor Water Cleanup to Condenser and Radwaste	-	MOV	MOV	<Section 6.2.4.2.2.1.b.2>
Reactor Water Cleanup from Regenerative Heat Exchangers to Feedwater	-	MOV	MOV	<Section 6.2.4.2.2.1.b.2>
Residual Heat Removal Shutdown Cooling Line	MOV,CV	GP	MOV	<Section 6.2.4.2.2.1.b.3>
Recirculation System Sample Line	-	AOV(2)	-	<Section 6.2.4.2.2.1.b.4>

NOTE:

⁽¹⁾ Isolation devices:

AOV - Air operated valve
GP - Guard pipe

MOV - Motor-operated valve
CV - Check valve

TABLE 6.2-36

PRIMARY CONTAINMENT ISOLATION FOR LINES THAT PENETRATE CONTAINMENT
AND CONNECT TO THE SUPPRESSION POOL

<u>Line</u>	<u>Isolation Device(s) ⁽¹⁾ Outside Containment</u>	<u>Reference Section</u>
<u>Influent Lines</u>		
Low Pressure Core Spray Minimum Flow Line	MOV	<Section 6.2.4.2.2.2.a.1>
Low Pressure Core Spray Test Line	MOV	<Section 6.2.4.2.2.2.a.1>
High Pressure Core Spray Minimum Flow Line	MOV	<Section 6.2.4.2.2.2.a.1>
High Pressure Core Spray Test Line	MOV	<Section 6.2.4.2.2.2.a.1>
Residual Heat Removal Suppression Pool Cooling and Test Line	MOV	<Section 6.2.4.2.2.2.a.1>
Residual Heat Removal Heat Exchanger Dump to Suppression Pool	MOV	<Section 6.2.4.2.2.2.a.1>
Residual Heat Removal Minimum Flow Lines	MOV	<Section 6.2.4.2.2.2.a.1>
Reactor Core Isolation Cooling Minimum Flow Line	MOV	<Section 6.2.4.2.2.2.a.2>
Reactor Core Isolation Cooling Turbine Exhaust Line	MOV, CV, RV	<Section 6.2.4.2.2.2.c.3>
Residual Heat Removal Heat Exchanger Vent to Suppression Pool	MOV	<Section 6.2.4.2.2.2.a.3>
Relief Valve Discharge Lines	RV, MOV	<Section 6.2.4.2.2.2.a.3>
<u>Effluent Lines</u>		
Low Pressure Core Spray Suction Line	MOV	<Section 6.2.4.2.2.2.b>

TABLE 6.2-36 (Continued)

<u>Line</u>	Isolation Device(s) ⁽¹⁾ Outside <u>Containment</u>	<u>Reference Section</u>
<u>Influent Lines</u> (Continued)		
High Pressure Core Spray Alternate Suction Line	MOV	<Section 6.2.4.2.2.2.b>
Reactor Core Isolation Cooling System Alternate Suction Line	MOV	<Section 6.2.4.2.2.2.b>
Residual Heat Removal Suction Line	MOV	<Section 6.2.4.2.2.2.b>

NOTE:

⁽¹⁾ Isolation devices:

CV - Check valve

MOV - Motor-operated valve

RV - Relief valve

TABLE 6.2-37

COMBUSTIBLE GAS CONTROL SYSTEM
EQUIPMENT DESIGN AND PERFORMANCE DATA

a. Combustible Gas Mixing Units

1.	Compressor	Centrifugal
	Max inlet pressure, psia	23.3
	Max discharge pressure, psia	29.13
	Max inlet temperature, °F	185
	Max discharge temperature, °F	238
	Relative humidity (inlet), %	100
	Capacity, scfm (Minimum)	500
	Power requirement, BHP (Maximum)	60
2.	Heat Exchanger (After Cooler) - Isolated from cooling water flow rate via closure of 1M51-F591A(B) valve.	
	Cooling Water	RHR System
	Cooling Water Pressure (tube side), psig	160
	Air Temperature in/out, °F	238/238
	Cooling Water Temp. in/out, °F	140/140
3.	Material	
	Compressor	
	Casing	cast steel
	Shroud	aluminum
	Impeller	17-4 ph S.S.
	Heat Exchanger	
	Tube	304 S.S.
4.	Manufacturer	Turbonetics

TABLE 6.2-37 (Continued)

b. Combustible Gas Purging Drywell
Isolation Valves

Type	globe
Body	SA-216, GR, WCB
Stem	SA-564
Disc	SA-351, GR CF8M
Seating Surface	Stellite No. 6

c. Hydrogen Recombiner

1. Material

Outer Structure	Type 300 series S.S.
Inner Structure	Incoloy 800
Heater Element Sheath	Incoloy 800
Base Skid	Type 300 series S.S.

2. Power

Maximum, kW	75
Nominal, kW	50

3. Capacity, scfm (min.)

100

4. Temperatures

Gas in, °F	60 to 185
Outlet of heater section, °F	1,150 to 1,450
Exhaust, °F	~50 above ambient

TABLE 6.2-37 (Continued)

5.	Heaters	
	Number	5 banks
	Max. heat flux, watts/in ²	5.8
	Max. sheath temperature, °F	1,550
6.	Manufacturer	Westinghouse
d.	Piping	
	Material	Carbon Steel

TABLE 6.2-38

COMBUSTIBLE GAS CONTROL SYSTEM
FAILURE ANALYSIS

<u>Component</u>	<u>Malfunction</u>	<u>Comments</u>
Drywell purge compressors	Failure of compressor resulting in inability to purge the drywell of combustible gases	Should operating subsystem compressor fail, will be manually shutdown. Redundant subsystem will maintain system capabilities.
Valve control power	Failure of valve control power resulting in inability to open valves	Should valve control power fail, valves will remain shut: redundant subsystem will maintain system capabilities.
Hydrogen recombiner	Failure of recombiner to operate resulting in loss of hydrogen removal capability	Should recombiner fail, redundant recombiner will maintain system capabilities.
Hydrogen analyzer	Failure of analyzer or sample system	Should the analyzer or sample system fail, redundant analysis or sample system will maintain system capabilities.

TABLE 6.2-39
DESIGN AND OPERATING PARAMETERS FOR
HYDROGEN CONTROL

<u>System</u>	<u>System Design</u>	<u>System Used For Analysis</u>
Drywell/ Containment Atmosphere Mixing	Two-500 scfm compressors	One-500 scfm compressor
Hydrogen Recombiner	Two-100 scfm recombiner	One-100 scfm recombiner

TABLE 6.2-40

PRIMARY REACTOR CONTAINMENT PENETRATION AND CONTAINMENT ISOLATION VALVE
LEAKAGE RATE TEST LIST

Penetration No.	Description	Penetration Test	Inboard Barrier Test	Inboard Containment Isolation Barrier Description/ Valve No.	References	Outboard Barrier Test	Outboard Containment Isolation Barrier Description/ Valve No.	References
P202	Equipment hatch	B		Double O-ring	See Note ⁽¹⁾	-	-	-
P305	Lower Personnel Airlock Barrel	B		Inner door Inflatable gaskets	See Note ⁽²⁾ See Note ⁽¹⁾		Outer door Inflatable gaskets	See Note ⁽²⁾ See Note ⁽¹⁾
			C	P53F556		C	P53F536/F570	See Note ^{(3) (32)}
			C	P53F579B	See Note ⁽³²⁾	C	P53F015/F070	See Note ⁽³²⁾ /-
			C	P53F580B		C	P53F035	See Note ⁽⁷⁾
			C	P53F581	See Note ⁽³²⁾	C	P52F160	-
			C	P53F582		C	P53F010	-
						C	P53F030	-
						C	P53F579A	-
						C	P53F580A	See Note ⁽³²⁾
P312	Upper Personnel Airlock Barrel	B		Inner door Inflatable gaskets	See Note ⁽²⁾ See Note ⁽¹⁾		Outer door Inflatable gaskets	See Note ⁽²⁾ See Note ⁽¹⁾
						C	P53F541/F571	See Note ^{(3) (32)}
			C	P53F561		C	P53F025/F075	See Note ⁽³²⁾ /-
			C	P53F593B	See Note ⁽³²⁾	C	P53F045	See Note ⁽⁷⁾
			C	P53F594B		C	P52F170	-
			C	P53F595	See Note ⁽³²⁾	C	P53F020	-
			C	P53F596		C	P53F040	-
						C	P53F593A	-
						C	P53F594A	See Note ⁽³²⁾
P205	Fuel transfer tube	B		Double gasket	See Note ^{(1) (12)}	-	-	-
						C	F42F003	See Note ⁽³⁰⁾
P124	Main steam line A	B	C	B21F022A	See Note ^{(3) (4) (12) (13)}	C	B21F028A	See Note ^{(4) (13)}
P416	Main steam line B	B	C	B21F022B	See Note ^{(3) (4) (12) (13)}	C	B21F028B	See Note ^{(4) (13)}
P122	Main steam line C	B	C	B21F022C	See Note ^{(3) (4) (12) (13)}	C	B21F028C	See Note ^{(4) (13)}
P415	Main steam line D	B	C	B21F022D	See Note ^{(3) (4) (12) (13)}	C	B21F028D	See Note ^{(4) (13)}

TABLE 6.2-40 (Continued)

Penetration No.	Description	Penetration Test	Inboard Barrier Test	Inboard Containment Isolation Barrier Description/ Valve No.	References	Outboard Barrier Test	Outboard Containment Isolation Barrier Description/ Valve No.	References
P121	Feedwater A, RHR, RWCU Return to Reactor Pressure Vessel	B	-	N27F559A	See Note ^{(5) (13)}	- C C -	B21F032A B21F065A E12F053A RWCU Closed System	See Note ^{(5) (13)} See Note ^{(5) (13)} See Note ^{(7) (13) (31)} See Note ^{(13) (31)}
P414	Feedwater B, RHR, RWCU Return to Reactor Pressure Vessel	B	-	N27F559B	See Note ^{(5) (13)}	- C C -	B21F032B B21F065B E12F053B RWCU Closed System	See Note ^{(5) (13)} See Note ^{(5) (13)} See Note ^{(7) (13) (31)} See Note ^{(13) (31)}
P102	RHR pump A suction	-	-	E12F004A	See Note ^{(9) (13) (17) (34)}	- -	Closed system -	See Note ⁽⁷⁾
P402	RHR pump B suction	-	-	E12F004B	See Note ^{(9) (13) (17) (34)}	- -	Closed system -	See Note ⁽⁷⁾
P403	RHR pump C suction	-	-	E12F105	See Note ^{(9) (13) (17) (34)}	- -	Closed system -	See Note ⁽⁷⁾
P421	RHR shutdown cooling suction	B	- -	E12F009 E12F550	See Note ^{(12) (18) (34)} See Note ^{(18) (34)}	-	E12F008	See Note ^{(18) (34)}
P105	RHR A min. flow & test; LPCS pump min. flow & test; RHR Heat Exchanger Dump to Suppression Pool; SPCU Return Line	-	- - - - B - -	E12F011A E12F064A E12F024A E12F609 E12D003A E21F011 E21F012	See Note ^{(9) (13) (16) (34)} See Note ^{(9) (13) (16) (34)} See Note ^{(9) (13) (16) (34)} See Note ^{(9) (13) (34)} - See Note ^{(9) (13) (16) (34)} See Note ^{(9) (13) (16) (34)}	- - - - - - -	Closed system (RHR) Closed system (RHR) Closed system (RHR) E12F610 - Closed system (LPCS) Closed system (LPCS)	See Note ⁽⁷⁾ See Note ⁽⁷⁾ See Note ⁽⁷⁾ See Note ⁽³⁴⁾ - See Note ⁽⁷⁾ See Note ⁽⁷⁾
P407	RHR B test and min. flow; RHR Heat Exchanger Dump to Suppression Pool	-	- - - B	E12F011B E12F064B E12F024B E12D003B	See Note ^{(9) (13) (16) (34)} See Note ^{(9) (13) (16) (34)} See Note ^{(9) (13) (16) (34)} -	- - - - - -	Closed system (RHR) Closed system (RHR) Closed system (RHR) - - -	See Note ⁽⁷⁾ See Note ⁽⁷⁾ See Note ⁽⁷⁾ - - -

TABLE 6.2-40 (Continued)

Penetration No.	Description	Pene- tration Test	Inboard Barrier Test	Inboard Containment Isolation Barrier	References	Outboard Barrier Test	Outboard Containment Isolation Barrier	References
				Barrier Description/ Valve No.			Barrier Description/ Valve No.	
P408	RHR C pump test, and minimum flow line	-	C	E12F064C	See Note ^{(6) (13) (16) (35)}	-	Closed system (RHR)	See Note ⁽⁷⁾
			C	E12F021	See Note ^{(13) (16)}	-	Closed system (RHR)	See Note ⁽⁷⁾
						-	-	
						-	-	
						-	-	
P107	RHR A relief valve discharge to suppression pool	-	C	E12F025B	See Note ^{(8) (13)}	-	Closed system	See Note ⁽⁷⁾
			C	E12F025C	See Note ^{(8) (13)}	-	Closed system	See Note ⁽⁷⁾
			C	E12F005	See Note ^{(8) (13)}	-	Closed system	See Note ⁽⁷⁾
			C	E12F055B	See Note ^{(8) (13)}	-	Closed system	See Note ⁽⁷⁾
						-	-	
			C	E12F055A	See Note ^{(8) (13)}	-	Closed system	See Note ⁽⁷⁾
			C	E12F025A	See Note ^{(8) (13)}	-	Closed system	See Note ⁽⁷⁾
			B	E12D015A	-	-	-	
			B	E12D015B	-	-	-	
						-	-	
						-	-	
			C	E12F102	See Note ^{(3) (13) (26)}	C	E51F068	-
			C			-	-	
			C	E21F018	See Note ^{(8) (13)}	-	Closed system	See Note ⁽⁷⁾
			C	N27F751	See Note ^{(3) (13)}	B	N27D006	See Note ⁽¹³⁾
			C	P87F264	See Note ⁽³⁾	C	P87F083	See Note ⁽³⁾
P429	RHR B relief valve discharge to suppression pool	-	-	Same as penetration P107			Same as penetration P107	
P422	Steam supply to RCIC turbine	B	C	E51F063	See Note ^{(3) (12)}	C	E51F064	-
			C	E51F076	See Note ⁽³⁾	C		
P118	RHR heat exchanger vents to suppres- sion vessel	-	C C	E12F558A	See Note ⁽¹³⁾	C	E12F073A	See Note ⁽¹³⁾
P431	RHR heat exchanger vents to suppres- sion vessel	-	C C	E12F558B	See Note ⁽¹³⁾	C	E12F073B	See Note ⁽¹³⁾

TABLE 6.2-40 (Continued)

Penetration No.	Description	Pene- tration Test	Inboard Barrier Test	Inboard Containment Isolation Barrier Barrier Description/ Valve No.	References	Outboard Barrier Test	Outboard Containment Isolation Barrier Barrier Description/ Valve No.	References
P123	RCIC and RHR to RPV head spray	B	-	E51F066	See Note ^{(12) (13) (34)}	- -	E51F013 E12F023	See Note ^{(13) (34)} See Note ^{(3) (13) (34)}
P113	LPCI A to reactor and RHR to cont. spray and cont. pool cooling	-	- - -	E12F042A E12F037A E12F028A	See Note ^{(13) (34)} See Note ^{(3) (13) (34)} See Note ^{(13) (34)}	-	E12F027A	See Note ^{(13) (34)}
P412	LPCI B to reactor and RHR to cont. spray and cont. pool cooling	-	- - -	E12F042B E12F037B E12F028B	See Note ^{(13) (34)} See Note ^{(3) (13) (34)} See Note ^{(13) (34)}	-	E12F027B	See Note ^{(13) (34)}
P411	LPCI C to Reactor	-	-	E12F041C	See Note ^{(13) (34)}	- -	E12F042C	See Note ^{(13) (34)}
P104	RCIC pump minimum flow line to suppression pool	-	C	E51F019	See Note ^{(3) (13) (16)}	- -	- Closed system	- See Note ⁽⁷⁾
P101	RCIC pump suction	-	-	E51F031	See Note ^{(9) (13) (17) (34)}	- -	Closed system -	See Note ⁽⁷⁾ -
P106	RCIC turbine exhaust	- -	C C	E51F068 E12F102	See Note ^{(13) (26)} See Note ^{(3) (13) (26)}	C C C C C C B B C C C C	E12F025B E12F025C E12F005 E12F055B E12F055A E12F025A E12D015A E12D015B E21F018 N27F751 P87F264 E51F0040	See Note ^{(8) (13) (26)} See Note ^{(8) (13) (26)} See Note ^{(8) (13) (26)} See Note ^{(8) (13) (26)} See Note ^{(8) (13) (26)} See Note ^{(8) (13) (26)} See Note ⁽²⁶⁾ See Note ⁽²⁶⁾ See Note ^{(8) (13) (26)} See Note ^{(8) (13) (26)} See Note ^{(3) (26)} See Note ⁽²⁶⁾

TABLE 6.2-40 (Continued)

Penetration No.	Description	Pene- tration Test	Inboard Barrier Test	Inboard Containment Isolation Barrier Barrier Description/ Valve No.	References	Outboard Barrier Test	Outboard Containment Isolation Barrier Barrier Description/ Valve No.	References
P115	RCIC pump turbine exhaust vacuum relief	- -	C C	E51F068 E12F102	See Note ⁽¹³⁾ (26) See Note ⁽³⁾ (13) (26)	C C C C C C B B C C C C	E12F025B E12F025C E12F005 E12F055B E12F055A E12F025A E12D015A E12D015B E21F018 N27F751 P87F264 E51F0040	See Note ⁽⁸⁾ (13) (26) See Note ⁽⁸⁾ (13) (26) See Note ⁽⁸⁾ (13) (26) See Note ⁽⁸⁾ (13) (26) See Note ⁽⁸⁾ (13) (26) See Note ⁽⁸⁾ (13) (26) See Note ⁽²⁶⁾ See Note ⁽²⁶⁾ See Note ⁽⁸⁾ (13) (26) See Note ⁽³⁾ (13) (26) See Note ⁽³⁾ (26) See Note ⁽²⁶⁾
P401	HPCS pump suction	-	-	E22F015	See Note ⁽⁹⁾ (13) (17) (34)	-	Closed system	See Note ⁽⁷⁾
P410	HPCS pump discharge to RPV	-	-	E22F005	See Note ⁽¹³⁾ (34)	-	E22F004	See Note ⁽¹³⁾ (34)
P409	HPCS min. flow and test line to suppression pool	-	C C C	E22F012 E22F035 E22F023	See Note ⁽¹³⁾ (16) See Note ⁽⁸⁾ (13) (16) See Note ⁽¹³⁾ (16)	- - -	Closed system Closed system Closed system	See Note ⁽⁷⁾ See Note ⁽⁷⁾ See Note ⁽⁷⁾
P103	LPCS pump suction	-	-	E21F001	See Note ⁽⁹⁾ (13) (17) (34)	C	Closed system	See Note ⁽⁷⁾
P112	LPCS pump discharge to RPV	-	-	E21F006	See Note ⁽¹³⁾ (34)	-	E21F005	See Note ⁽¹³⁾ (34)
P423	Main steam line drain	B	C	B21F016	See Note ⁽³⁾ (12)	C	B21F019	-
P131	RWCU pump suction	B	C	G33F001	See Note ⁽¹²⁾ (13)	C	G33F004	See Note ⁽¹³⁾
P419	RWCU pump discharge	-	C	G33F053	-	C	G33F054	-
P132	RWCU return to feedwater	-	C	G33F040	-	C	G33F039	-

TABLE 6.2-40 (Continued)

Penetration No.	Description	Pene- tration Test	Inboard Barrier Test	Inboard Containment Isolation Barrier Barrier Description/ Valve No.	References	Outboard Barrier Test	Outboard Containment Isolation Barrier Barrier Description/ Valve No.	References	
P424	RWCU to main condenser and radwaste	-	C C	G33F028 G33F0646	- See Note ⁽⁸⁾	C	G33F034	-	
P203	Fuel pool cooling and cleanup supply	-	C	G41F522	-	C	G41F100	-	
P301	Fuel pool cooling and cleanup system return	-	C C	G41F140 G41F801	- -	C	G41F145	-	
V313	Containment purge supply	-	C C C	M14F045 M14F190 M14F195	See Note ⁽³⁾ See Note ⁽³⁾ See Note ^{(3) (28)}	C	M14F040	-	
V314	Containment purge exhaust	-	C C C	M14F200 M14F085 M14F205	See Note ⁽³⁾ See Note ⁽³⁾ See Note ^{(3) (28)}	C	M14F090	-	
P404	Chilled water supply	-	C	P50F539	See Note ⁽¹³⁾	C	P50F060	See Note ⁽¹³⁾	
P405	Chilled water return	-	C C	P50F140 P50F606	See Note ⁽¹³⁾ See Note ⁽⁸⁾	C	P50F150	See Note ⁽¹³⁾	
P417	Drywell and con- tainment equipment drain sump pump discharge	-	C C	G61F075 G61F0655	See Note ⁽³⁾	C	G61F080	See Note ⁽³⁾	
P418	Drywell and con- tainment floor drain sump pump discharge	-	C C	G61F165 G61F657	See Note ⁽³⁾ -	C	G61F170	See Note ⁽³⁾	
P309	Demineralized water supply to containment	-	C	P22F577	-	C	P22F010	-	

TABLE 6.2-40 (Continued)

Penetration No.	Description	Pene- tration Test	Inboard Barrier Test	Inboard Containment Isolation Barrier Barrier Description/ Valve No.	References	Outboard Barrier Test	Outboard Containment Isolation Barrier Barrier Description/ Valve No.	References
P420	RWCU backwash transfer pump discharge to radwaste	-	C C	G50F272 G50F823	- See Note ⁽⁸⁾	C	G50F277	-
P310	Nuclear closed cooling water supply	-	C	P43F721	See Note ⁽¹³⁾	C	P43F055	See Note ⁽¹³⁾
P311	Nuclear closed cooling water return	-	C C	P43F215 P43F851	See Note ⁽¹³⁾ See Note ⁽⁸⁾	C	P43F140	See Note ⁽¹³⁾
P108	Condensate supply	-	C	P11F545	-	C	P11F060	-
P111	Condensate return	-	C C	P11F090 P11F629	- -	C	P11F080	-
P116	Air supply to ADS accumulators	-	C	P57F524B	See Note ⁽¹⁵⁾	C	P57F015B	See Note ⁽¹⁵⁾
P304	Air supply to ADS accumulators	-	C	P57F524A	See Note ⁽¹⁵⁾	C	P57F015A	See Note ⁽¹⁵⁾
P308	Service air	-	C	P51F530	See Note ⁽³³⁾	C	P51F150	-
P306	Instrument Air	-	C	P52F550	-	C	P52F200	-
P114	Containment Vacuum Relief	-	C	M17F010	-	C	M17F015	-
P208	Containment Vacuum Relief	-	C	M17F020	-	C	M17F025	-
P428	Containment Vacuum Relief	-	C	M17F030	-	C	M17F035	-
P436	Containment Vacuum Relief	-	C	M17F040	-	C	M17F045	-
P434	Containment Vacuum Relief	-	A	M17F055 and ¼" orifice	See Note ⁽¹⁴⁾	-	-	-

TABLE 6.2-40 (Continued)

Penetration No.	Description	Pene- tration Test	Inboard Barrier Test	Inboard Containment Isolation Barrier Barrier Description/ Valve No.	References	Outboard Barrier Test	Outboard Containment Isolation Barrier Barrier Description/ Valve No.	References	
P320	Containment Vacuum Relief	-	A	M17F065 and ¼" orifice	See Note ⁽¹⁴⁾	-	-	-	
P406	Fire Protection Water	-	C	P54F727	-	C	P54F726	-	
P210	CO ₂ to Fire Protection	-	C	P54F1098	-	C	P54F340	-	
P302	Backup H ₂ Purge	-	C	M51F090	-	C	M51F110	-	
P425	Post-LOCA H ₂ Analyzer	-	C	M51F210A	See Note ^{(3) (14) (21) (23)}	-	-	-	
			C	M51F220A	See Note ^{(3) (14) (22) (23)}				
			C	M51F230A	See Note ^{(3) (14) (22) (23)}				
			C	M51F240A	See Note ^{(3) (14) (21) (23)}				
			C	M51F250A	See Note ^{(3) (14) (21) (23)}				
P315	SLC Tank Refill	-	C	C41F520	-	C	C41F518	-	
P318	Post-LOCA H ₂ Analyzer	-	C	M51F210B	See Note ^{(3) (14) (21) (23)}	-	-	-	
			C	M51F220B	See Note ^{(3) (14) (22) (23)}				
			C	M51F230B	See Note ^{(3) (14) (22) (23)}				
			C	M51F240B	See Note ^{(3) (14) (21) (23)}				
			C	M51F250B	See Note ^{(3) (14) (21) (23)}				
		-	C	P87F071	See Note ^{(3) (14) (23)}				
			C	P87F065	See Note ^{(3) (14) (23)}				
			C	P87F077	See Note ^{(3) (14) (23)}				
P109	ILRT Vent	-	C	P87F074	See Note ^{(3) (14) (23)}	B	E61D003	-	
			B	E61D017 Blind flange	-				
P120	ILRT Pressurization Line	-	B	E61D016 Blind flange	-	B	E61D001	-	
P119	ILRT Instrumentation	-	B	E61D014	-	B	E61D015	-	

TABLE 6.2-40 (Continued)

Penetration No.	Description	Pene- tration Test	Inboard Barrier Test	Inboard Containment Isolation Barrier Barrier Description/ Valve No.	References	Outboard Barrier Test	Outboard Containment Isolation Barrier Barrier Description/ Valve No.	References	
P317	ILRT Instrumentation	-	C B C B	E61F549 Pipe cap E61F550 Pipe cap	See Note ^{(3) (13) (23)} See Note ^{(13) (23)} See Note ^{(3) (13) (23)} See Note ^{(13) (23)}	B B	E61D007 E61D006	See Note ^{(13) (23)} See Note ^{(13) (23)}	
P319	ILRT Instrumentation	-	C B C B	E61F551 Pipe cap E61F552 Pipe cap	See Note ^{(3) (13) (23)} See Note ^{(13) (23)} See Note ^{(3) (13) (23)} See Note ^{(13) (23)}	B B	E61D005 E61D004	See Note ^{(13) (23)} See Note ^{(13) (23)}	
P117	N ₂ Supply to CRD	-	C	P86F528	-	C	P86F002	-	
P204	CRD to Reactor Pressure Vessel	-	C	C11F122	See Note ⁽¹³⁾	C	C11F083	See Note ⁽¹³⁾	
P201	Drywell Atmosphere Radiation Monitor	-	C C C	D17F071B D17F079B D17F079A	See Note ^{(3) (23)} See Note ⁽²³⁾ -	C	D17F071A	See Note ⁽³⁾	
P413	PASS	-	C C C	P87F049 P87F046 P87F277	See Note ^{(3) (23)} See Note ^{(3) (23)} See Note ^{(8) (23)}	C C	P87F055 P87F052	See Note ^{(3) (23)} See Note ^{(3) (23)}	
P317	Containment Atmos- phere Radiation Monitor	- -	C C	D17F081B D17F089B	See Note ^{(3) (23)} See Note ⁽²³⁾	C C	D17F081A D17F089A	See Note ⁽³⁾ -	
P319	Containment Atmos- phere Monitoring	-	A A	¼" orifice and D23F030B ¼" orifice and D23F040B	See Note ^{(14) (23)} See Note ^{(14) (23)}	-	-	-	
P320	Containment Atmos- phere Monitoring	-	A A	¼" orifice and D23F020B ¼" orifice and D23F010B	See Note ^{(14) (23)} See Note ^{(14) (23)}	-	-	-	
P425	Containment Atmos- phere Monitoring	-	A	¼" orifice and D23F050	See Note ⁽¹⁴⁾	-	-	-	

TABLE 6.2-40 (Continued)

Penetration No.	Description	Penetration Test	Inboard Barrier Test	Inboard Containment Isolation Barrier Description/ Valve No.	References	Outboard Barrier Test	Outboard Containment Isolation Barrier Description/ Valve No.	References
P433	Containment Atmosphere Monitoring	-	A	¼" orifice and D23F030A	See Note ⁽¹⁴⁾ (23)	-	-	-
			A	¼" orifice and D23F040A	See Note ⁽¹⁴⁾ (23)			
P434	Containment Atmosphere Monitoring	-	A	¼" orifice and D23F010A	See Note ⁽¹⁴⁾ (23)	-	-	-
			A	¼" orifice and D23F020A	See Note ⁽¹⁴⁾ (23)			
P102	Suppression Pool Makeup	-	A C C	G43F050A	See Note ⁽¹⁴⁾ (23)	-	-	-
P402	Suppression Pool Makeup	-	A C C	G43F050B	See Note ⁽¹⁴⁾ (23)	-	-	-
P401	Suppression Pool Makeup	-	A C C C C	G43F060	See Note ⁽¹⁴⁾ (23)	-	-	-
				P87F037	See Note ⁽³⁾ (9) (14) (23) (35)			
Various	Spares	-	A	Capped	-	-	-	-
Various	Electrical	-	B	Double O-rings	See Note ⁽¹¹⁾	-	-	-

NOTES:

- ⁽¹⁾ Penetration is sealed by a blind flange or door with double O-ring or double gasket seals. The gaskets or O-rings are leak checked by pressurizing the space between them.
- ⁽²⁾ Personnel air lock volume is pressurized to pressure (Pa) as given in technical specifications. During the airlock test, tie downs are installed on the inner door since it is not designed to withstand the full differential pressure across the door in the reverse direction. Pressurizing the air lock barrel also tests the air lock mechanical and electrical penetrations.
- ⁽³⁾ Leak testing direction for valves having a functional differential pressure of 15 psi or less will be in accordance with the ASME Code for Operation and Maintenance of Nuclear Power Plants (ASME OM Code).

TABLE 6.2-40 (Continued)

NOTES: (Continued)

- ⁽⁴⁾ MSIV seat leakage rate shall not exceed 100 scfh for any main steam line with maximum total leakage for all four main steam lines less than or equal to 250 scfh.

MSIV leakage is not included in the 0.60 L_a Type B and C test totals (Reference 33).

- ⁽⁵⁾ The <10 CFR 50, Appendix J> Type C test for the feedwater lines is provided by the Type C hydrostatic tests performed on the long term, high integrity leakage protection valves, i.e., the motor-operated gate valves (B21-F065A/B). Water leakage from the Feedwater Leakage Control System (FWLCS) piping, and from the bonnet, stem and seat of the motor-operated gate valves, are controlled under the Primary Coolant Sources Outside Containment Program, Technical Specification 5.5.2. Outboard gate valve bonnet and stem leakage identified during the system walkdown at pressure >1000 psig will be eliminated. The gate valves will be closed during this check. Gate valve seat water leakage measured at $\geq 1.1 P_a$ will be limited by the program, which restricts allowable leakages to half of that assumed in the dose calculations <Section 15.6.5.5.1.2.b>. This water leakage is not redundantly added into the Type C 0.60 L_a totals, secondary containment bypass air leakage totals or the hydrostatic test program totals. The feedwater check valves (N27-F559A/B and B21-F032A/B) utilize an alternate non-Type C water leak testing methodology per the Inservice Testing Program, to verify proper closure (see <Section 15.6.6.5.2.4>).

- ⁽⁶⁾ System remains water filled post-LOCA. Isolation valve tested with water to a pressure not less than 1.10 P_a. Isolation valve leakage not included in 0.60 L_a Type B and C test totals.

- ⁽⁷⁾ The redundant containment isolation provisions for this penetration consist of an isolation valve and a closed system outside containment which is in compliance with <10 CFR 50, Appendix A>, GDC 54, and with GDCs 55 and 56, utilizing the "other defined basis" provision. A single active failure can be accommodated. The closed system is missile protected, Seismic Category I, Safety Class 2, and has a temperature and pressure rating in excess of that for the containment. Closed system integrity is maintained and verified during system leak tests <NUREG-0737>, Item III.D.1.1.

However, the portions of the Personnel Airlock Leakage Control System designated as closed systems to secondary containment (see P305 and P312) are not given closed system leak tests for the following reason. The closed system piping is routed to the annulus. The Annulus Exhaust Gas Treatment System maintains the annulus at a vacuum. Any post-LOCA containment atmosphere in the airlock leakage control lines will flow into the annulus. As a result, no containment atmosphere will bypass the annulus. Therefore, closed system tests at P_a on the airlock leakage control lines are unnecessary.

- ⁽⁸⁾ Relief valve.

- ⁽⁹⁾ System is sealed from the primary containment atmosphere because its line terminates below the water level of the suppression pool. Leakage is not included in 0.60 L_a Type B and C test totals.

- ⁽¹⁰⁾ (Deleted)

- ⁽¹¹⁾ Modular type electrical penetration with header plate bolted to penetration nozzle. Double O-ring seals with test connection is provided at interface.

- ⁽¹²⁾ Penetration design utilizes a double bellows for containment isolation. The bellows is leak checked by pressurizing the space between the inner and outer bellows. The fuel transfer tube bellows is sealed on both ends with double gasketed flange joints. These joints are leak checked by pressurizing the space between the double gaskets. The fuel transfer tube has a bellows assembly installed in the annulus to permit confirmatory leak testing of the fuel transfer tube bellows. The fuel transfer tube bellows is leak checked by pressurizing the annular space between the IFTS tube and the bellows assembly via a test connection on the bellows assembly.

- ⁽¹³⁾ System is not vented and drained for Type A test. This allowance for main steam lines is provided by (Reference 33).

TABLE 6.2-40 (Continued)

NOTES: (Continued)

- ⁽¹⁴⁾ Isolation valving for instrument lines which penetrate the containment conform to the requirements of <Regulatory Guide 1.11>. The ISI program will provide assurance of the operability and integrity of the isolation provisions. Type C testing will not be performed on the instrument line isolation valves. The instrument lines will be within the boundaries of the Type A test, open to the media (containment atmosphere or suppression pool water) to which they will be exposed under postulated accident conditions. Three exceptions to the above are Penetrations P401, P318 and P425. Isolation valves for the three penetrations include H₂ Analyzer and Postaccident Sampling System Valves. The valves are normally closed post-LOCA, open only intermittently, and will therefore receive Type C tests.
- ⁽¹⁵⁾ System remains pressurized with air post-LOCA.
- ⁽¹⁶⁾ These lines are always filled with water on the outboard side of the containment, thereby forming a water seal. They are maintained at a pressure that is always higher than primary containment pressure by jockey pumps or hydrostatic head, thus precluding any outleakage from primary containment. However, even if outleakage did occur, it would be into an ESF system which forms a closed loop outside primary containment. Thus, any leakage from primary containment would return to primary containment through this closed loop.
- A system leakage test <NUREG-0737>, Item III.D.1.1. will be performed as described below to ensure the leak-tightness of the ECCS and RCIC systems. The systems will be pressurized with water and a leakage rate for the entire system will then be determined and compared to an acceptance limit.
- ⁽¹⁷⁾ The ECCS and RCIC suction lines are normally filled with water on both the inboard and outboard side of containment, thereby forming a water seal to the containment environment. The valves are open during post-LOCA conditions to supply a water source for the ECCS pumps. Since a break in an ECCS line need not be considered in conjunction with a DBA, the only possible situation requiring one of these valves to be closed during a DBA is an unacceptable leakage in an emergency core cooling system. However, because these ECCS systems are constantly monitored for excessive leakage, this is not a credible event.
- ⁽¹⁸⁾ During Type A test, this penetration is operational to provide shutdown cooling via the RHR system.
- ⁽¹⁹⁾ (Deleted)
- ⁽²⁰⁾ (Deleted)
- ⁽²¹⁾ Post LOCA H₂ Analyzer - Containment
- ⁽²²⁾ Post LOCA H₂ Analyzer - Drywell
- ⁽²³⁾ Penetration contains more than one line
- ⁽²⁴⁾ (Deleted)
- ⁽²⁵⁾ (Deleted)

TABLE 6.2-40 (Continued)

NOTES: (Continued)

- ⁽²⁶⁾ Valve E12F102 serves as a first isolation barrier for P107 and P429. Valve E51F068 serves as a second isolation barrier for P107 and P429.
- The RCIC Turbine Exhaust Line (P106) and the associated vacuum break line (P115) share common containment isolation valves:
- The first containment isolation for the penetrations are: E51F068 (automatic) and E12F102 (locked closed). The second containment isolation valve downstream of E51F068 is nozzle check valve E51F040 (not a simple check).
- Downstream of E12F0102 is the RHR Relief Lines to the Suppression Pool (P107 and P429). The first isolation valves/barriers for the RHR Relief Lines (P107 and P429) are also the second isolation valves/barriers for the RCIC Turbine Exhaust Line and the Vacuum Break Line (P106 and P115). These valves/barriers are: E12F005, E12F025A, E12F025B, E12F025C, E12F055A, E12F055B, E21F018, N27F751, P87F264 and spectacle flanges 1E12D015A and 1E12D015B.
- ⁽²⁷⁾ (Deleted)
- ⁽²⁸⁾ The containment purge 18-inch supply and exhaust lines are provided with double inboard isolation valves. For each 18-inch line, both inboard valves are Type C tested and the highest leakage is designated as the inboard barrier leakage. Leakage through the test connection between the 18-inch isolation valves is summed with the innermost 18-inch valve leakage.
- ⁽²⁹⁾ (Deleted)
- ⁽³⁰⁾ The portion of the large transfer tube piping outboard of the blind flange and down to the drain line valve 1F42-F003 (the portion of the tube which becomes exposed to containment air during the draining portion of the IFTS operation) will also be part of the leakage rate test boundary and will be tested with air.
- ⁽³¹⁾ The RWCU and RHR lines return water to the Reactor Vessel via the feedwater lines. The piping "outboard" of the containment is ASME Code Class 2, Seismic Category I, protected from pipe whip, missiles and jet forces, and analyzed for "break exclusion". The branch line leading to the RHR closed system outside containment is examined for external leakage from mechanical joints (e.g., valve stem or bonnet, seals, and gaskets). The RWCU branch line leads directly back to containment penetration P132, and contains only mechanical joints, including the packing on the outboard containment isolation valve (G33-F039) for penetration P132 (see the P132 entry). This outboard valve (G33-F039), including the stem and bonnet, is already part of the air leak rate test program. The remainder of the RWCU piping between the feedwater line and penetration P132 will be added to the Technical Specification 5.5.2 Primary Coolant Sources Outside Containment Program, with a specific leakage acceptance limit of zero (0) external water leakage when tested at > 1000 psig.
- ⁽³²⁾ "Barrier Description" of these valves inside (I) or outside (O) of containment varies depending on whether the inner or outer airlock door is open and/or inoperable.
- ⁽³³⁾ P51F0530 is a 2.0" valve installed in the 2.5" line via reducers.
- ⁽³⁴⁾ This valve has been excluded from Type C testing since it does not constitute a potential primary containment atmospheric pathway during and following a Design Basis Accident (DBA).
- ⁽³⁵⁾ This system will be pressurized with water to a minimum pressure of 1.10 Pa and a leakage rate for the entire system will be compared to an acceptance limit.

TABLE 6.2-41

CONTAINMENT LEAK RATE TEST DATAType A Test Definitions

A.	Peak Test Pressure	$P_a = 7.80$ psig	
	The calculated peak containment internal pressure related to the design basis loss-of-coolant accident (LOCA).		
B.	Maximum Allowable Leakage Rate at P_a	$L_a = 0.2\%$ by weight of the contained atmosphere in 24 hrs.	
C.	Measured Leakage Rate at P_a	L_{am}	
	Estimate of the leakage rate, derived as a function of the least squares slope and intercept, for the Type A test at pressure P_a obtained from testing the primary containment by simulating some of the conditions that would exist under design-basis accident conditions (e.g., vented, drained, flooded, or pressurized).		
D.	Imposed Leakage Rate	L_o	
	The known leakage rate superimposed on the containment during the verification test.		
E.	Composite Leakage Rate	L_c	
	The composite leakage rate is the algebraic sum of the calculated leakage rate L_a and the superimposed leakage rate L_o .		

TABLE 6.2-41 (Continued)

Type A Test Definitions (Continued)

F. Test Duration

1. After the containment atmosphere has stabilized, the integrated leakage rate test period begins. The duration of the test period is sufficient to enable adequate data to be accumulated and statistically analyzed so that leakage rate and upper confidence limit can be accurately determined.
2. A Type A test shall last a minimum of eight hours after stabilization and shall have a total of not less than 30 sets of data points at approximately equal time intervals.
3. A Type A test cannot be successfully terminated until the acceptance criteria of <Chapter 16> are met.

G.	Containment Atmosphere Temperature Limits During Type A Test	40°-120°F
H.	Containment Free Air Volume	1,441,900 ft ³ (includes drywell free air volume)
I.	Drywell Free Air Volume	276,500 ft ³

TABLE 6.2-42

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TABLE 6.2-43

HYDROGEN CONTROL SYSTEM
EQUIPMENT DESIGN AND PERFORMANCE DATA

1.	Number of Igniters	
	- Drywell	17
	- Wetwell	12
	- Enclosed Areas	22
	- Containment	<u>51</u>
	Total	102
2.	Igniter Location Criteria (except drywell below weir wall and containment above refueling floor)	1 ESF Division operable: 60 foot spacing not to exceed 70 feet 2 ESF Divisions operable 30 foot spacing not to exceed 35 feet
3.	Igniter Assembly Manufacturer	Power Systems Division of Morris Knudson
4.	Igniter Operating Temperature	1,700°F @ 12 volt ac
5.	Igniter Transformer	0.2 KVA Dongan Model 52-20-472
6.	Igniter Qualification Temperature	345°F for 3 hours
7.	Igniter Qualification Pressure	33 psig
8.	System Operation	Manually via 2 control room handswitches (1 switch per division)
9.	Power Supply	120 Vac +/- 10% from ESF Power Panels powered off of motor control centers from ESF buses (onsite and offsite AC power supplies)

TABLE 6.2-44

CONTAINMENT EQUIPMENT SURVIVABILITY LIST

<u>Equipment Identification Number</u>	<u>Equipment Description</u>	<u>Function</u>	<u>Elevation</u>	<u>Azimuth</u>	<u>Rx Centerline Distance</u>	<u>Qualification</u>	
						<u>Temp (F)</u>	<u>Pressure (psig)</u>
1B21N044C	Pressure Transmitter	Rx Vessel Level (Fuel Zone)	622'-4"	120°	43'-3"	318	73
1B21N044D	Pressure Transmitter	Rx Vessel Level (Fuel Zone)	634'-8"	305°	47'-6-1/2"	318	73
1B21N091A	Pressure Transmitter	Rx Vessel Level (Wide Range)	622'-7"	53°	52'-0"	318	73
1B21N091B	Pressure Transmitter	Rx Vessel Level (Wide Range)	622'-1"	225°	43'-6"	318	73
1B21N062A	Pressure Transmitter	Rx Vessel Pressure	624'-6"	53°	53'-2"	318	73
1B21N062B	Pressure Transmitter	Rx Vessel Pressure	622'-1"	230°	43'-6"	318	73
1D23N130A	Containment RTD	Containment Temperature Monitoring	720'-6"	280°	60'	340 ⁽¹⁾	70
1D23N130B	Containment RTD	Containment Temperature Monitoring	720'-6"	100°	60'	340 ⁽¹⁾	70
1D23N140A	Containment RTD	Containment Temperature Monitoring	689'-4"	45°	58'	340 ⁽¹⁾	70
1D23N140B	Containment RTD	Containment Temperature Monitoring	689'-4"	210°	58'	340 ⁽¹⁾	70
1D23N150A	Containment RTD	Containment Temperature Monitoring	647'-0"	54°	56'	340 ⁽¹⁾	70
1D23N150B	Containment RTD	Containment Temperature Monitoring	645'-6"	251°	56'	340 ⁽¹⁾	70
1D23N160A	Containment RTD	Containment Temperature Monitoring	613'-0"	69°	56'	340 ⁽¹⁾	70

TABLE 6.2-44 (Continued)

Equipment Identification Number	Equipment Description	Function	Elevation	Azimuth	Rx Centerline Distance	Qualification	
						Temp (F)	Pressure (psig)
1D23N160B	Containment RTD	Containment Temperature Monitoring	613'-0"	251°	56'	340 ⁽¹⁾	70
1E12F028A	Containment Spray Inboard Isolation Valve (MO)	Containment Spray	643'-6"	37°	55'-6"	340	105
1E12F028B	Containment Spray Inboard Isolation Valve (MO)	Containment Spray	643'-9"	335°	56'-3"	340	105
1E12F042A	LPCI Inboard Isolation Valve (MO)	Low Pressure Coolant Injection	628'-0"	41°	44'-0"	340	105
1E12F042B	LPCI Inboard Isolation Valve (MO)	Low Pressure Coolant Injection	628'-0"	315°	55'-6"	340	105
1E12F537A	Containment Spray Valve (MO)	Containment Spray	692'-0"	40°	58'-0"	340	105
1E12F537B	Containment Spray Valve (MO)	Containment Spray	692'-0"	320°	58'-0"	340	105
1G41F140	FPC Inboard Isolation Valve (MO)	Containment Venting	629'-5"	228°	59'	340	105
1M17F010	Containment Vacuum Relief System	Containment Vacuum Relief Check Valve	693'-0"	58°	60'	250	30
1M17F020	Containment Vacuum Relief System	Containment Vacuum Relief Check Valve	693'-0"	150°	60'	250	30
1M17F030	Containment Vacuum Relief System	Containment Vacuum Relief Check Valve	693'-0"	302°	60'	250	30
1M17F040	Containment Vacuum Relief System	Containment Vacuum Relief Check Valve	693'-0"	315°	60'	250	30

TABLE 6.2-44 (Continued)

Equipment Identification Number	Equipment Description	Function	Elevation	Azimuth	Rx Centerline Distance	Qualification	
						Temp (F)	Pressure (psig)
1M51C001A	Hydrogen Mixing Compressor and Motor	Hydrogen Mixing	664'-7"	300°	24'	192	80
1M51C001B	Hydrogen Mixing Compressor and Motor	Hydrogen Mixing	664'-7"	245°	25'	192	80
1M51D001A	Hydrogen Recombiner	Removal of Hydrogen by Hydrogen and Oxygen Recombination	664'-7"	304°	31'	1,700 - 1,750 (Heater Element)	45.3
1M51D001B	Hydrogen Recombiner	Removal of Hydrogen by Hydrogen and Oxygen Recombination	664'-7"	236°	30'	1,700 - 1,750 (Heater Element)	45.3
1M51F010A	Hydrogen Mixing Compressor Isolation Valve (MO)	Drywell Isolation	667'-7"	309°	25'-6"	340	105
1M51F010B	Hydrogen Mixing Compressor Isolation Valve (MO)	Drywell Isolation	667'-7"	245°	22'	340	105
1M51F020A	Combustible Gas Control System Supply Valve (MO)	Compressor Cooling Water	666'-0"	300°	32'	340	105
1M51F020B	Combustible Gas Control System Supply Valve (MO)	Compressor Cooling Water	666'-0"	255°	30'	340	105
1M51F501A	Hydrogen Mixing Compressor Check Valve	Check Valve for Drywell Purge Compressor	666'-0"	305°	27'	350	300
1M51F501B	Hydrogen Mixing Compressor Check Valve	Check Valve for Drywell Purge Compressor	665'-3"	250°	23'	350	300

TABLE 6.2-44 (Continued)

Equipment Identification Number	Equipment Description	Function	Elevation	Azimuth	Rx Centerline Distance	Qualification	
						Temp (F)	Pressure (psig)
1M56S001	Hydrogen Ignition System	Hydrogen Ignition	611'-6"	352°	48'-6"	345	33
1M56S002	Hydrogen Ignition System	Hydrogen Ignition	612'-6"	4°	49'-0"	345	33
1M56S003	Hydrogen Ignition System	Hydrogen Ignition	619'-6"	69°	54'-6"	345	33
1M56S004	Hydrogen Ignition System	Hydrogen Ignition	619'-6"	89°	54'-3"	345	33
1M56S005	Hydrogen Ignition System	Hydrogen Ignition	662'-6"	34°	57'-0"	345	33
1M56S006	Hydrogen Ignition System	Hydrogen Ignition	687'-4"	36°	52'-3"	345	33
1M56S023	Hydrogen Ignition System	Hydrogen Ignition	619'-6"	54°	54'-0"	345	33
1M56S024	Hydrogen Ignition System	Hydrogen Ignition	619'-6"	116°	52'-0"	345	33
1M56S025	Hydrogen Ignition System	Hydrogen Ignition	619'-6"	150°	51'-7"	345	33
1M56S026	Hydrogen Ignition System	Hydrogen Ignition	619'-6"	185°	51'-8"	345	33
1M56S027	Hydrogen Ignition System	Hydrogen Ignition	619'-6"	221°	51'-9"	345	33
1M56S028	Hydrogen Ignition System	Hydrogen Ignition	619'-6"	251°	53'-8"	345	33
1M56S029	Hydrogen Ignition System	Hydrogen Ignition	619'-6"	289°	53'-4"	345	33
1M56S030	Hydrogen Ignition System	Hydrogen Ignition	619'-6"	324°	53'-3"	345	33

TABLE 6.2-44 (Continued)

Equipment Identification Number	Equipment Description	Function	Elevation	Azimuth	Rx Centerline Distance	Qualification	
						Temp (F)	Pressure (psig)
1M56S031	Hydrogen Ignition System	Hydrogen Ignition	638'-0"	359°	41'-6"	345	33
1M56S032	Hydrogen Ignition System	Hydrogen Ignition	640'-0"	155°	46'-0"	345	33
1M56S033	Hydrogen Ignition System	Hydrogen Ignition	640'-0"	187°	46'-0"	345	33
1M56S034	Hydrogen Ignition System	Hydrogen Ignition	640'-0"	324°	53'-6"	345	33
1M56S035	Hydrogen Ignition System	Hydrogen Ignition	640'-4-3/4"	61°	51'-6"	345	33
1M56S036	Hydrogen Ignition System	Hydrogen Ignition	640'-5-1/2"	118°	51'-6"	345	33
1M56S037	Hydrogen Ignition System	Hydrogen Ignition	640'-5"	227°	46'-0"	345	33
1M56S038	Hydrogen Ignition System	Hydrogen Ignition	639'-4"	261°	54'-0"	345	33
1M56S039	Hydrogen Ignition System	Hydrogen Ignition	651'-1"	286°	41'-6"	345	33
1M56S040	Hydrogen Ignition System	Hydrogen Ignition	647'-4"	2°	41'-6"	345	33
1M56S041	Hydrogen Ignition System	Hydrogen Ignition	650'-6-3/4"	41°	50'-6"	345	33
1M56S042	Hydrogen Ignition System	Hydrogen Ignition	650'-6"	87°	49'-0"	345	33
1M56S043	Hydrogen Ignition System	Hydrogen Ignition	651'-0"	101°	49'-0"	345	33
1M56S044	Hydrogen Ignition System	Hydrogen Ignition	660'-0"	86°	44'-6"	345	33

TABLE 6.2-44 (Continued)

Equipment Identification Number	Equipment Description	Function	Elevation	Azimuth	Rx Centerline Distance	Qualification	
						Temp (F)	Pressure (psig)
1M56S045	Hydrogen Ignition System	Hydrogen Ignition	660'-6"	95°	48'-6"	345	33
1M56S046	Hydrogen Ignition System	Hydrogen Ignition	664'-0"	54°	51'-0"	345	33
1M56S047	Hydrogen Ignition System	Hydrogen Ignition	665'-0"	114°	52'-0"	345	33
1M56S048	Hydrogen Ignition System	Hydrogen Ignition	662'-6"	147°	53'-0"	345	33
1M56S049	Hydrogen Ignition System	Hydrogen Ignition	662'-7-3/4"	218°	51'-0"	345	33
1M56S050	Hydrogen Ignition System	Hydrogen Ignition	664'-7"	251°	49'-6"	345	33
1M56S051	Hydrogen Ignition System	Hydrogen Ignition	661'-6"	289°	50'-0"	345	33
1M56S052	Hydrogen Ignition System	Hydrogen Ignition	661'-6"	324°	49'-6"	345	33
1M56S053	Hydrogen Ignition System	Hydrogen Ignition	669'-6"	0°	54'-6"	345	33
1M56S054	Hydrogen Ignition System	Hydrogen Ignition	684'-9"	355°	52'-6"	345	33
1M56S055	Hydrogen Ignition System	Hydrogen Ignition	686'-0"	75°	48'-0"	345	33
1M56S056	Hydrogen Ignition System	Hydrogen Ignition	686'-0"	85°	47'-0"	345	33
1M56S057	Hydrogen Ignition System	Hydrogen Ignition	686'-0"	95°	47'-0"	345	33
1M56S058	Hydrogen Ignition System	Hydrogen Ignition	686'-0"	105°	48'-0"	345	33

TABLE 6.2-44 (Continued)

Equipment Identification Number	Equipment Description	Function	Elevation	Azimuth	Rx Centerline Distance	Qualification	
						Temp (F)	Pressure (psig)
1M56S059	Hydrogen Ignition System	Hydrogen Ignition	686'-0"	75°	35'-0"	345	33
1M56S060	Hydrogen Ignition System	Hydrogen Ignition	686'-0"	105°	35'-0"	345	33
1M56S061	Hydrogen Ignition System	Hydrogen Ignition	689'-6"	45°	48'-0"	345	33
1M56S062	Hydrogen Ignition System	Hydrogen Ignition	689'-6"	133°	41'-0"	345	33
1M56S063	Hydrogen Ignition System	Hydrogen Ignition	689'-6"	229°	48'-0"	345	33
1M56S064	Hydrogen Ignition System	Hydrogen Ignition	689'-6"	252°	43'-6"	345	33
1M56S065	Hydrogen Ignition System	Hydrogen Ignition	689'-6"	289°	43'-0"	345	33
1M56S066	Hydrogen Ignition System	Hydrogen Ignition	689'-6"	310°	48'-6"	345	33
1M56S067	Hydrogen Ignition System	Hydrogen Ignition	715'-6"	359°	58'-9"	345	33
1M56S068	Hydrogen Ignition System	Hydrogen Ignition	715'-6"	27°	58'-9"	345	33
1M56S069	Hydrogen Ignition System	Hydrogen Ignition	715'-6"	62°	58'-9"	345	33
1M56S070	Hydrogen Ignition System	Hydrogen Ignition	715'-6"	87°	58'-9"	345	33
1M56S071	Hydrogen Ignition System	Hydrogen Ignition	715'-6"	119°	58'-9"	345	33
1M56S072	Hydrogen Ignition System	Hydrogen Ignition	715'-6"	151°	58'-9"	345	33

TABLE 6.2-44 (Continued)

Equipment Identification Number	Equipment Description	Function	Elevation	Azimuth	Rx Centerline Distance	Qualification	
						Temp (F)	Pressure (psig)
1M56S073	Hydrogen Ignition System	Hydrogen Ignition	715'-6"	178°	58'-9"	345	33
1M56S074	Hydrogen Ignition System	Hydrogen Ignition	715'-6"	209°	58'-9"	345	33
1M56S075	Hydrogen Ignition System	Hydrogen Ignition	715'-6"	241°	58'-9"	345	33
1M56S076	Hydrogen Ignition System	Hydrogen Ignition	715'-6"	273°	58'-9"	345	33
1M56S077	Hydrogen Ignition System	Hydrogen Ignition	715'-6"	300°	58'-9"	345	33
1M56S078	Hydrogen Ignition System	Hydrogen Ignition	715'-6"	331°	58'-9"	345	33
1M56S079	Hydrogen Ignition System	Hydrogen Ignition	745'-6"	359°	48'-0"	345	33
1M56S080	Hydrogen Ignition System	Hydrogen Ignition	745'-6"	34°	48'-0"	345	33
1M56S081	Hydrogen Ignition System	Hydrogen Ignition	745'-6"	72°	48'-0"	345	33
1M56S082	Hydrogen Ignition System	Hydrogen Ignition	745'-6"	102°	48'-0"	345	33
1M56S083	Hydrogen Ignition System	Hydrogen Ignition	745'-6"	143°	48'-0"	345	33
1M56S084	Hydrogen Ignition System	Hydrogen Ignition	745'-6"	180°	48'-0"	345	33
1M56S085	Hydrogen Ignition System	Hydrogen Ignition	745'-6"	216°	48'-0"	345	33
1M56S086	Hydrogen Ignition System	Hydrogen Ignition	745'-6"	252°	48'-0"	345	33

TABLE 6.2-44 (Continued)

Equipment Identification Number	Equipment Description	Function	Elevation	Azimuth	Rx Centerline Distance	Qualification	
						Temp (F)	Pressure (psig)
1M56S087	Hydrogen Ignition System	Hydrogen Ignition	745'-6"	287°	48'-0"	345	33
1M56S088	Hydrogen Ignition System	Hydrogen Ignition	745'-6"	324°	48'-0"	345	33
1M56S089	Hydrogen Ignition System	Hydrogen Ignition	757'-0"	0°	1'-0"	345	33
1M56S090	Hydrogen Ignition System	Hydrogen Ignition	757'-0"	180°	1'-0"	345	33
1M56S091	Hydrogen Ignition System	Hydrogen Ignition	645'-7"	168°	60'-0"	345	33
1M56S092	Hydrogen Ignition System	Hydrogen Ignition	645'-0"	172°	58'-0"	345	33
1M56S093	Hydrogen Ignition System	Hydrogen Ignition	613'-4"	7°	44'-0"	345	33
1M56S094	Hydrogen Ignition System	Hydrogen Ignition	612'-5"	13°	42'-8"	345	33
1M56S095	Hydrogen Ignition System	Hydrogen Ignition	612'-6"	344°	42'-6"	345	33
1M56S096	Hydrogen Ignition System	Hydrogen Ignition	612'-3"	351°	43'-6"	345	33
1M56S097	Hydrogen Ignition System	Hydrogen Ignition	638'-8"	289°	49'-6"	345	33
1M56S098	Hydrogen Ignition System	Hydrogen Ignition	685'-6"	342°	53'-0"	345	33
1M56S099	Hydrogen Ignition System	Hydrogen Ignition	685'-6"	17°	50'-6"	345	33
1M56S100	Hydrogen Ignition System	Hydrogen Ignition	686'-0"	75°	25'-0"	345	33

TABLE 6.2-44 (Continued)

Equipment Identification Number	Equipment Description	Function	Elevation	Azimuth	Rx Centerline Distance	Qualification	
						Temp (F)	Pressure (psig)
1M56S101	Hydrogen Ignition System	Hydrogen Ignition	686'-0"	105°	25'-0"	345	33
1R72S001	Electrical Penetrations	Containment Boundary	659'-0"	221°	60'	340	108
1R72S002	Electrical Penetrations	Containment Boundary	659'-0"	228°	60'	340	108
1R72S003	Electrical Penetrations	Containment Boundary	656'-3"	221°	60'	340	108
1R72S004	Electrical Penetrations	Containment Boundary	657'-1-1/2"	248°	60'	340	108
1R72S005	Electrical Penetrations	Containment Boundary	656'-3"	228°	60'	340	108
1R72S006	Electrical Penetrations	Containment Boundary	657'-1-1/2"	242°	60'	340	108
1R72S007	Electrical Penetrations	Containment Boundary	651'-6"	221°	60'	340	108
1R72S008	Electrical Penetrations	Containment Boundary	649'-9"	221°	60'	340	108
1R72S009	Electrical Penetrations	Containment Boundary	651'-6"	248°	60'	340	108
1R72S010	Electrical Penetrations	Containment Boundary	649'-9"	248°	60'	340	108
1R72S011	Electrical Penetrations	Containment Boundary	657'-1-1/2"	235°	60'	340	108
1R72S012	Electrical Penetrations	Containment Boundary	651'-6"	228°	60'	340	108
1R72S013	Electrical Penetrations	Containment Boundary	649'-9"	228°	60'	340	108

TABLE 6.2-44 (Continued)

Equipment Identification Number	Equipment Description	Function	Elevation	Azimuth	Rx Centerline Distance	Qualification	
						Temp (F)	Pressure (psig)
1R72S014	Electrical Penetrations	Containment Boundary	651'-6"	242°	60'	340	108
1R72S015	Electrical Penetrations	Containment Boundary	649'-9"	242°	60'	340	108
1R72S016	Electrical Penetrations	Containment Boundary	643'-3"	221°	60'	340	108
1R72S017	Electrical Penetrations	Containment Boundary	641'-6"	221°	60'	340	108
1R72S018	Electrical Penetrations	Containment Boundary	643'-3"	228°	60'	340	108
1R72S019	Electrical Penetrations	Containment Boundary	641'-6"	228°	60'	340	108
1R72S020	Electrical Penetrations	Containment Boundary	643'-3"	248°	60'	340	108
1R72S021	Electrical Penetrations	Containment Boundary	641'-6"	241°	60'	340	108
1R72S022	Electrical Penetrations	Containment Boundary	643'-3"	242°	60'	340	108
1R72S023	Electrical Penetrations	Containment Boundary	641'-6"	248°	60'	340	108
1R72S024	Electrical Penetrations	Containment Boundary	643'-3"	235°	60'	340	108
1R72S025	Electrical Penetrations	Containment Boundary	651'-6"	235°	60'	340	108
1R72S026	Electrical Penetrations	Containment Boundary	638'-4"	221°	60'	340	108
1R72S027	Electrical Penetrations	Containment Boundary	638'-4"	228°	60'	340	108

TABLE 6.2-44 (Continued)

Equipment Identification Number	Equipment Description	Function	Elevation	Azimuth	Rx Centerline Distance	Qualification	
						Temp (F)	Pressure (psig)
1R72S028	Electrical Penetrations	Containment Boundary	641'-6"	223°	60'	340	108
1R72S029	Electrical Penetrations	Containment Boundary	656'-3"	223°	60'	340	108
1R72S030	Electrical Penetrations	Containment Boundary	643'-3"	223°	60'	340	108
1R72S031	Electrical Penetrations	Containment Boundary	649'-9"	223°	60'	340	108
1R72S032	Electrical Penetrations	Containment Boundary	657'-1-1/2"	241°	60'	340	108
1R72S033	Electrical Penetrations	Containment Boundary	649'-9"	235°	60'	340	108
1R72S035	Electrical Penetrations	Containment Boundary	641'-6"	242°	60'	340	108
1R72S036	Electrical Penetrations	Containment Boundary	649'-9"	241°	60'	340	108
1R72S038	Electrical Penetrations	Containment Boundary	651'-6"	241°	60'	340	108
	Upper Personnel Airlock Seal		692'-10"	225°	60'	330	60
	Lower Personnel Airlock Seal		599'-9"	241°	60'	330	60
	Equipment Hatch Seal		620'-6"	133°	60'	330	60
	Control Cable and Small Power Cable		Various Locations			346	113
	Instrument Cable		Various Locations			346 ⁽²⁾	113

NOTES:

⁽¹⁾ RTD qualified to 485°F for 30 seconds and then 340°F for 3 hours.

⁽²⁾ Instrument cable qualified to 385°F for 12 minutes and then 346°F for 3 hours.

TABLE 6.2-45

DRYWELL EQUIPMENT SURVIVABILITY LIST

Equipment Identification Number	Equipment Description	Function	Elevation	Azimuth	Rx Centerline Distance	Qualification		Manufacturer	Model
						Temp (F)	Pressure (psig)		
1B21F041A	Automatic Depressurization System Valve	RPV Pressure Relief/ADS	636'-5"	51°	20'	355	60	Dikkers	G471-6/125.04
1B21F041B	Automatic Depressurization System Valve	RPV Pressure Relief/ADS	636'-5"	277°	26"	355	60	Dikkers	G471-6/125.04
1B21F041E	Automatic Depressurization System Valve	RPV Pressure Relief/ADS	636'-5"	31°	21'	355	60	Dikkers	G471-6/125.04
1B21F041F	Automatic Depressurization System Valve	RPV Pressure Relief/ADS	636'-5"	289°	26'	355	60	Dikkers	G471-6/125.04
1B21F047D	Automatic Depressurization System Valve	RPV Pressure Relief/ADS	636'-5"	308°	20'	355	60	Dikkers	G471-6/125.04
1B21F047H	Automatic Depressurization System Valve	RPV Pressure Relief/ADS	636'-5"	322°	21'	355	60	Dikkers	G471-6/125.04
1B21F051C	Automatic Depressurization System Valve	RPV Pressure Relief/ADS	636'-5"	88°	25'	355	60	Dikkers	G471-6/125.04
1B21F051G	Automatic Depressurization System Valve	RPV Pressure Relief/ADS	636'-5"	71°	26'	355	60	Dikkers	G471-6/125.04
1B21F410A	Automatic Depressurization System Valve Solenoid	RPV Pressure Relief/ADS	Location Same as ADS Valve			355	50	Seitz	6A39
1B21F410B	Automatic Depressurization System Valve Solenoid	RPV Pressure Relief/ADS	Location Same as ADS Valve			355	50	Seitz	6A39
1B21F411A	Automatic Depressurization System Valve Solenoid	RPV Pressure Relief/ADS	Location Same as ADS Valve			355	50	Seitz	6A39
1B21F411B	Automatic Depressurization System Valve Solenoid	RPV Pressure Relief/ADS	Location Same as ADS Valve			355	50	Seitz	6A39

TABLE 6.2-45 (Continued)

Equipment Identification Number	Equipment Description	Function	Elevation	Azimuth	Rx Centerline Distance	Qualification		Manufacturer	Model
						Temp (F)	Pressure (psig)		
1B21F414A	Automatic Depressurization System Valve Solenoid	RPV Pressure Relief/ADS	Location Same as ADS Valve			355	50	Seitz	6A39
1B21F414B	Automatic Depressurization System Valve Solenoid	RPV Pressure Relief/ADS	Location Same as ADS Valve			355	50	Seitz	6A39
1B21F415A	Automatic Depressurization System Valve Solenoid	RPV Pressure Relief/ADS	Location Same as ADS Valve			355	50	Seitz	6A39
1B21F415B	Automatic Depressurization System Valve Solenoid	RPV Pressure Relief/ADS	Location Same as ADS Valve			355	50	Seitz	6A39
1B21F422A	Automatic Depressurization System Valve Solenoid	RPV Pressure Relief/ADS	Location Same as ADS Valve			355	50	Seitz	6A39
1B21F422B	Automatic Depressurization System Valve Solenoid	RPV Pressure Relief/ADS	Location Same as ADS Valve			355	50	Seitz	6A39
1B21F425A	Automatic Depressurization System Valve Solenoid	RPV Pressure Relief/ADS	Location Same as ADS Valve			355	50	Seitz	6A39
1B21F425B	Automatic Depressurization System Valve Solenoid	RPV Pressure Relief/ADS	Location Same as ADS Valve			355	50	Seitz	6A39
1B21F442A	Automatic Depressurization System Valve Solenoid	RPV Pressure Relief/ADS	Location Same as ADS Valve			355	50	Seitz	6A39
1B21F442B	Automatic Depressurization System Valve Solenoid	RPV Pressure Relief/ADS	Location Same as ADS Valve			355	50	Seitz	6A39
1B21F444A	Automatic Depressurization System Valve Solenoid	RPV Pressure Relief/ADS	Location Same as ADS Valve			355	50	Seitz	6A39

TABLE 6.2-45 (Continued)

Equipment Identification Number	Equipment Description	Function	Elevation	Azimuth	Rx Centerline Distance	Qualification		Manufacturer	Model
						Temp (F)	Pressure (psig)		
1B21F444B	Automatic Depressuri- zation System Valve Solenoid	RPV Pressure Relief/ADS	Location Same as ADS Valve			355	50	Seitz	6A39
1D23N100A	Drywell RTD	Drywell Temperature Monitoring	653'-8"	315°	17'	340 ⁽¹⁾	70	Weed	611
1D23N100B	Drywell RTD	Drywell Temperature Monitoring	653'-8"	135°	16'	340 ⁽¹⁾	70	Weed	611
1D23N110A	Drywell RTD	Drywell Temperature Monitoring	634'-0"	308°	36'-6"	340 ⁽¹⁾	70	Weed	611
1D23N110B	Drywell RTD	Drywell Temperature Monitoring	634'-0"	145°	36'-6"	340 ⁽¹⁾	70	Weed	611
1D23N120A	Drywell RTD	Drywell Temperature Monitoring	605'-0"	308°	36'-6"	340 ⁽¹⁾	70	Weed	611
1D23N120B	Drywell RTD	Drywell Temperature Monitoring	604'-6"	150°	36'-6"	340 ⁽¹⁾	70	Weed	611
1M56S008	Hydrogen Ignition System	Hydrogen Ignition	629'-1-1/2"	12°	36'-6"	345	33	Power Systems	6043
1M56S009	Hydrogen Ignition System	Hydrogen Ignition	637'-0"	41°	36'-6"	345	33	Power Systems	6043
1M56S010	Hydrogen Ignition System	Hydrogen Ignition	636'-3-1/2"	90°	36'-6"	345	33	Power Systems	6043
1M56S011	Hydrogen Ignition System	Hydrogen Ignition	636'-7"	137°	36'-6"	345	33	Power Systems	6043
1M56S012	Hydrogen Ignition System	Hydrogen Ignition	632'-3"	180°	36'-6"	345	33	Power Systems	6043
1M56S013	Hydrogen Ignition System	Hydrogen Ignition	631'-5"	221°	36'-6"	345	33	Power Systems	6043
1M56S014	Hydrogen Ignition System	Hydrogen Ignition	636'-10"	273°	36'-6"	345	33	Power Systems	6043

TABLE 6.2-45 (Continued)

Equipment Identification Number	Equipment Description	Function	Elevation	Azimuth	Rx Centerline Distance	Qualification		Manufacturer	Model
						Temp (F)	Pressure (psig)		
1M56S015	Hydrogen Ignition System	Hydrogen Ignition	630'-9-1/2"	322°	36'-6"	345	33	Power Systems	6043
1M56S016	Hydrogen Ignition System	Hydrogen Ignition	660'-0"	0°	31'-6"	345	33	Power Systems	6043
1M56S017	Hydrogen Ignition System	Hydrogen Ignition	659'-8"	57°	29'-6"	345	33	Power Systems	6043
1M56S018	Hydrogen Ignition System	Hydrogen Ignition	659'-8"	114°	30'-0"	345	33	Power Systems	6043
1M56S019	Hydrogen Ignition System	Hydrogen Ignition	659'-8"	172°	30'-0"	345	33	Power Systems	6043
1M56S020	Hydrogen Ignition System	Hydrogen Ignition	659'-8"	225°	28'-0"	345	33	Power Systems	6043
1M56S021	Hydrogen Ignition System	Hydrogen Ignition	660'-0"	280°	30'-0"	345	33	Power Systems	6043
1M56S022	Hydrogen Ignition System	Hydrogen Ignition	660'-0"	317°	31'-0"	345	33	Power Systems	6043
1M56S102	Hydrogen Ignition System	Hydrogen Ignition	670'-0"	350°	13'-0"	345	33	Power Systems	6043
1M56S103	Hydrogen Ignition System	Hydrogen Ignition	670'-0"	4°	13'-0"	345	33	Power Systems	6043
	Control Cable and Small Power Cable Instrument Cable		Drywell (Various Locations) (Various Locations)			346	113	Rockbestos	EKB-2
						346 ⁽²⁾	113	Brand-Rex	EKC-1
	Drywell Personnel Airlock Seal	Containment Boundary	599'-9"	105°	36'-6"	330	60	W.J. Wooley	

TABLE 6.2-45 (Continued)

Equipment Identification Number	Equipment Description	Function	Elevation	Azimuth	Rx Centerline Distance	Qualification		Manufacturer	Model
						Temp (F)	Pressure (psig)		
	Drywell Equipment Hatch Seal	Containment Boundary	599'-9"	227°	36'-6"	330		Newport News Industrial Corp.	

NOTES:

- ⁽¹⁾ RTD qualified to 485°F for 30 seconds and then 340°F for 3 hours.
- ⁽²⁾ Instrument cable qualified to 385°F for 12 minutes and then 346°F for 3 hours.

TABLE 6.2-46

PERRY CLASIX-3 RESULTS

		<u>SORV</u> ⁽²⁾	<u>DWB</u> ⁽³⁾
Number of burns	DW ⁽¹⁾	0	0 [1]
	WW	32	30 [8]
	CT	2	0 [1]
Total H ₂ Burned (lbm)	DW	0	0 [117]
	WW	1,220	1,361 [472]
	CT	791	0 [340]
H ₂ Remaining (lbm)	DW	15	692 [203]
	WW	293	151 [41]
	CT	294	409 [81]
Peak Temp. (°F)	DW	191 (154)	251 [643]
	WW	1,762 (1,364)	1,201 [1,763]
	CT	760 (236)	192 [587]
Peak Pressure (psig)	DW	15.9 (10.7)	13.8 [17.3]
	WW	21.1 (12.6)	12.2 [19.4]
	CT	21.2 (9.9)	10.9 [19.4]

NOTES:

⁽¹⁾ Drywell, wetwell and containment are abbreviated as DW, WW and CT.

⁽²⁾ () Maxima due to wetwell burns.

⁽³⁾ [] Values due to extension past end of hydrogen release.

6.3 EMERGENCY CORE COOLING SYSTEMS

6.3.1 DESIGN BASES AND SUMMARY DESCRIPTION

<Section 6.3.1> provides the design bases for the emergency core cooling systems (ECCS), a summary description of the several systems as an introduction to the more detailed design descriptions provided in <Section 6.3.2>, and the performance analysis provided in <Section 6.3.3>. <Appendix 15B>, Reload Safety Analysis provides the results of a performance analysis for the new (reload) fuel designs.

6.3.1.1 Design Bases

6.3.1.1.1 Performance and Functional Requirements

The ECCS is designed to provide protection against postulated loss-of-coolant accidents (LOCA) caused by ruptures in primary system piping. The functional requirements (for example, coolant delivery rates) specified in detail in <Table 6.3-1> are such that the system performance under all LOCA conditions postulated in the design satisfies the requirements of <10 CFR 50.46>, "Acceptance Criteria for Emergency Core Cooling System for Light Water Cooled Nuclear Power Reactors". These requirements, the most important of which is that the post-LOCA peak cladding temperature be limited to 2,200°F, are summarized in <Section 6.3.3.2>. In addition, the ECCS is designed to following requirements:

- a. Protection is provided for any primary system line break up to and including the double-ended break of the largest line.
- b. Two independent phenomenological cooling methods (flooding and spraying) are provided to cool the core.

- c. One high pressure cooling system is provided which is capable of maintaining water level above the top of the core and eliminating the need for automatic depressurization system actuation for line breaks of less than 1 inch nominal diameter.
- d. No operator action is required until 10 minutes after an accident to allow for operator assessment and decision.
- e. The ECCS is designed to satisfy all criteria specified in this section for any normal mode of reactor operation.
- f. A sufficient water source and the necessary piping, pumps and other hardware are provided so that the containment and reactor core can be flooded for possible core heat removal following a LOCA.

6.3.1.1.2 Reliability Requirements

The following reliability requirements apply:

- a. The ECCS conforms to licensing requirements, and good design practices of isolation, separation and common mode failure considerations.
- b. To meet the above requirements, the ECCS network shall have built-in redundancy so that adequate cooling can be provided, even in the event of specified failures. As a minimum, the following equipment shall make up the ECCS:
 - 1. 1 High Pressure Core Spray (HPCS)
 - 2. 1 Low Pressure Core Spray (LPCS)
 - 3. 3 Low Pressure Coolant Injection (LPCI) Loops
 - 4. 1 Automatic Depressurization System (ADS)

- c. The system is designed so that a single active or passive component failure, including power buses, electrical and mechanical parts, cabinets and wiring will not disable the ADS.
- d. In the event of a break in a pipe that is not a part of the ECCS, no single active component failure in the ECCS shall prevent automatic initiation and successful operation of less than the following combination of ECCS equipment:
 - 1. 3 LPCI loops, the LPCS and the ADS (i.e., HPCS failure); or
 - 2. 2 LPCI loops, the HPCS and the ADS (i.e., failure of diesel generator supplying LPCS/LPCI); or
 - 3. 1 LPCI loop, the LPCS, the HPCS, and ADS (i.e., "diesel generator" failure).
- e. In the event of a break in a pipe that is a part of the ECCS, no single active component failure in the ECCS shall prevent automatic initiation and successful operation of less than the following combination of ECCS equipment:
 - 1. 2 LPCI loops and the ADS; or
 - 2. 1 LPCI loop, the LPCS and the ADS; or
 - 3. 1 LPCI loop, the HPCS and the ADS; or
 - 4. The LPCS, the HPCS and ADS.

These are the minimum ECCS combinations which result after assuming any failure (from Item 4, above) and assuming that the ECCS line break disables the affected system.

- f. Long term (10 minutes after the initiation signal) cooling requirements call for the removal of decay heat via the emergency service water system. In addition to the break which initiated the loss-of-coolant event, the system is able to sustain one failure, either active or passive and still have at least one low pressure ECCS pump operating with a heat exchanger and 100 percent emergency service water flow.
- g. Offsite power is the preferred source of power for the ECCS network and every reasonable precaution must be made to assure its high availability. However, onsite emergency power shall be provided with sufficient diversity and capacity so that all the above requirements can be met even if offsite power is not available.
- h. The onsite diesel fuel reserve shall be in accordance with IEEE Standard 308-1974 criteria.
- i. Diesel-load configuration shall be as follows:
 - 1. 1 LPCI loop (with heat exchanger) and the LPCS connected to a single diesel generator.
 - 2. 2 additional LPCI loops (1 loop with heat exchanger) connected to a single diesel generator.
 - 3. The HPCS connected to a single diesel generator.
- j. Systems which interface with, but are not part of, the ECCS are designed and operated such that failure(s) in the interfacing systems shall not propagate to and/or affect the performance of the ECCS.

- k. Non-ECCS systems interfacing with the ECCS buses shall automatically be shed from and/or be inhibited from the ECCS buses when a LOCA signal exists and offsite ac power is not available.
- l. No more than one storage battery shall be connectable to a dc power bus.
- m. Each system of the ECCS including flow rate and sensing networks is capable of being tested during shutdown. All active components are capable of being tested during plant operation, including logic required to automatically initiate component action.
- n. Provisions for testing the ECCS network components (electronic, mechanical, hydraulic, and pneumatic, as applicable) are installed in such a manner that they are an integral and nonseparable part of the design.
- o. The ECCS is designed to withstand the passive failure of valve stem packings and pump seals following a LOCA.

6.3.1.1.3 ECCS Requirements for Protection from Physical Damage

The emergency core cooling system piping and components are protected against damage from movement, from thermal stresses, from the effects of the LOCA, and the safe shutdown earthquake (SSE).

The ECCS is protected against the effects of pipe whip which might result from piping failures up to and including the design basis event LOCA. This protection is provided by separation, pipe whip restraints or energy absorbing materials if required. One of these three methods is applied to provide protection against damage to piping and components of the ECCS which otherwise could result in a reduction of ECCS effectiveness to an unacceptable level.

The ECCS piping and components located outside the reactor building are protected from internally and externally generated missiles by the reinforced concrete structure of the auxiliary building ECCS pump rooms. In addition, the watertight construction of the ECCS pump rooms when required protects against mass flooding of redundant ECCS pumps.

Mechanical separation outside the drywell is achieved as follows:

- a. The ECCS is separated into three functional groups:
 1. HPCS
 2. LPCS + 1 LPCI + 100% emergency service water and heat exchanger
 3. 2 LPCI pumps + 100% emergency service water and heat exchanger
- b. The equipment in each group is separated from that in the other two groups. In addition, the HPCS and RCIC (which is not part of the ECCS) are separated.
- c. Separation barriers are constructed between the functional groups as required to assure that environmental disturbances such as fire, pipe rupture, falling objects, etc., affecting one functional group will not affect the remaining groups. In addition, separation barriers are provided as required to assure that such disturbances do not affect both the RCIC and the HPCS.

6.3.1.1.4 ECCS Environmental Design Basis

Each emergency core cooling system, and the RCIC system, has a safety-related injection/isolation testable check valve located in piping within the drywell. In addition, the RCIC system has an

isolation valve in the drywell portion of its steam supply piping. All valves are located above the highest water level expected in the drywell during any accident. The valves are qualified for the following environmental conditions:

- a. Normal and upset plant operating ambient temperatures, relative humidities and pressures for each area of the drywell.
- b. Envelope-of-accident conditions for temperature, relative humidity and pressure within the drywell for various time periods following the accident.
- c. Normal and envelope-of-accident radiation environment (gamma and neutron).

The portions of ECCS and RCIC piping and equipment located between the drywell and the secondary containment are qualified for the following environmental conditions:

- a. Normal and upset plant operating ambient temperatures, relative humidities and pressures.
- b. Envelope-of-accident conditions for temperature, relative humidity and pressure for various time periods following the accident.
- c. Normal and envelope-of-accident radiation environment (gamma and neutron).

6.3.1.2 Summary Descriptions of ECCS

The ECCS injection network comprises a high pressure core spray (HPCS) system, a low pressure core spray (LPCS) system and the low pressure coolant injection (LPCI) mode of the residual heat removal (RHR) System.

These systems are briefly described here as an introduction to the more detailed system design descriptions provided in <Section 6.3.2>. The automatic depressurization system (ADS) which assists the injection network under certain conditions is also briefly described. Boiling water reactors which employ the same ECCS design are listed in <Table 1.3-3>.

6.3.1.2.1 High Pressure Core Spray

The HPCS pumps water through a peripheral spray ring sparger mounted above the reactor core. Coolant is supplied over the entire range of system operation pressures. The primary purpose of HPCS is to maintain reactor vessel inventory after small breaks which do not depressurize the reactor vessel. HPCS also provides spray cooling heat transfer during breaks in which core uncover is calculated.

6.3.1.2.2 Low Pressure Core Spray

The LPCS is an independent loop similar to the HPCS, the primary difference being the LPCS delivers water over the core at relatively low reactor pressures. The primary purpose of the LPCS is to provide inventory makeup and spray cooling during large breaks in which the core is calculated to uncover. Following ADS initiation, LPCS provides inventory makeup following a small break.

6.3.1.2.3 Low Pressure Coolant Injection

LPCI is an operating mode of the residual heat removal system. Three pumps deliver water from the suppression pool to the bypass region inside the shroud through three separate reactor vessel penetrations to provide inventory makeup following large pipe breaks. Following ADS initiation, LPCI provides inventory makeup following a small break.

6.3.1.2.4 Automatic Depressurization System

The ADS uses a number of the reactor safety/relief valves to reduce reactor pressure during small breaks in the event of a postulated HPCS failure. When the vessel pressure is reduced to within the capacity of the low pressure systems (LPCS and LPCI), these systems provide inventory makeup so that acceptable postaccident temperatures are maintained.

6.3.2 SYSTEM DESIGN

A more detailed description of the individual systems including individual design characteristics of the systems are covered in detail in <Section 6.3.2.1>, <Section 6.3.2.2>, <Section 6.3.2.3>, and <Section 6.3.2.4>. <Table 6.3-7> provides a list of significant ECCS design parameters along with their design bases. The following discussion provides details of the combined systems; in particular, those design features and characteristics which are common to all systems are discussed.

6.3.2.1 Schematic Piping and Instrumentation Diagrams

The piping and instrumentation diagrams (P&IDs) for the ECCS are identified in <Section 6.3.2.2>. The process diagrams which identify the various operating modes of each system are also identified in <Section 6.3.2.2>.

6.3.2.2 Equipment and Component Descriptions

The starting signal for the ECCS comes from at least two independent and redundant sensors of drywell pressure and low reactor water level. The ECCS is actuated automatically and requires no operator action during the first 10 minutes following the accident. A time sequence for starting of the systems is provided in <Table 6.3-2>.

Electric power for operation of the ECCS is from the regular ac power sources. Upon loss of the regular power, operation is from onsite standby ac power sources. Standby sources have sufficient diversity and capacity so that all ECCS requirements are satisfied. The HPCS is powered from one ac supply bus. The LPCS and one LPCI loop are powered from a second ac supply bus and the two remaining LPCI loops are powered from a third and separate ac supply bus. The HPCS has its own diesel generator as its alternate power supply. The LPCS and one LPCI loop switch to one site backup power supply and the other two LPCI loops switch to a second site backup power supply. <Section 8.3> contains a more detailed description of the power supplies for the ECCS.

- a. <Regulatory Guide 1.1>, Net Positive Suction Head for Emergency Core Cooling and Containment Heat Removal System Pumps.

General Compliance or Alternate Approach Assessment

This guide prohibits design reliance on pressure and/or temperature transients expected during a LOCA for assuring adequate NPSH. The guidelines of this Regulatory Guide are applicable to the HPCS, LPCS, and LPCI pumps.

The BWR design conservatively assumes 0 psig containment pressure and maximum expected temperatures of the pumped fluids. Thus, no reliance is placed on pressure and/or temperature transients to assure adequate NPSH.

Requirements for NPSH are available at the centerline of the pump suction nozzles for each pump and are given in <Figure 6.3-1> (HPCS), <Figure 6.3-2> (LPCS), <Figure 6.3-3> (LPCI). Pump characteristic curves are given in <Figure 6.3-4> (HPCS), <Figure 6.3-5> (LPCS) and <Figure 6.3-6> (LPCI).

- b. <Regulatory Guide 1.82>, Water Sources for Long-Term Recirculation Cooling Following a Loss-of-Coolant Accident

General Compliance or Alternate Approach Assessment

The design of the large toroidal passive ECCS suction strainer was evaluated against the regulatory positions contained in <Regulatory Guide 1.82>, Revision 2.

The suction strainer is designed to preclude the potential for loss of NPSH caused by debris blockage during the period that the ECCS is required to maintain long-term cooling. The large toroidal passive strainer results in a very low approach velocity for water entering the strainer. Debris collected on the strainer surface is not expected to compact significantly (due to the very low approach velocity), resulting in minimal head loss. A 1/4-scale model of the strainer design was tested to confirm the performance of the strainer and the behavior of the postulated debris bed as a function of time after the postulated LOCA. Because the debris bed will not be significantly compacted, flow will continue to pass through the debris (and the strainer) and thus the overall differential pressure will remain low. Maintaining a low differential pressure will ensure adequate NPSH for the ECCS pumps.

The size of the openings in the suppression pool suction strainer material has been chosen based on the minimum restrictions found in systems served by the suppression pool.

The ECCS pump suction is designed to prevent degradation of pump performance through air ingestion and other adverse hydraulic effects. All of the suction piping remains below the surface of the suppression pool, and due to the very low approach velocity design of the strainer and the depth of the strainer in the pool,

vortexing will not be present. The strainer is located at the bottom of the suppression pool, below the elevation of the SRV quencher arms. Because of the physical size of the strainer, some encroachment into the recommended exclusion zone around the quenchers occurs. However, the strainer design is such that any air that may enter the strainer will be released through the strainer mesh before traveling to the pump suction plenums; that is, air entrainment in the strainer will be minimized.

The strainer does not involve any modification in the arrangement of drains from upper floors in the containment. In addition, there are no floor or equipment drains from the containment or reactor building that drain directly into the suppression pool with the exception of one normally closed 3" diameter refueling bellows drain line, two normally closed small bore Control Rod Drive system scram discharge volume drain lines, and one normally closed 6" diameter Condensate and Refueling Water system drain line. Because of the water quality and cleanliness level of the systems these lines are considered to contribute very little or no debris to the suppression pool. The Suppression Pool Makeup dump lines discharge upper containment pool water into the suppression pool following a LOCA. The upper containment pools are maintained in a clean condition by operation of the Fuel Pool Cleanup system and through the foreign material exclusion program. Therefore, debris from these lines will be limited to any corrosion products present in the dump lines carried along by the dump flow. In addition, the strainer is located at the bottom of the suppression pool such that it is highly unlikely that any debris from drains from the upper regions of the containment could impinge directly on the suction strainer.

The suction strainer is designed such that its support structure will protect it from the effects of large debris. The strainer is

designed so that it is capable of withstanding LOCA-induced hydrodynamic loads. PNPP utilizes GESSAR II methods combined with acoustic methodology which demonstrates that the strainer can withstand the LOCA-induced hydrodynamic loads. Missile protection was evaluated and determined to be of no concern based on the location of the strainer and the postulated missile sources for Perry. The maximum force that the perforated plate material could withstand without breach has been defined, such that appropriate controls are in place on handling and use of potentially damaging large objects taken in containment.

The suction strainer was evaluated and shown to be able to withstand loads associated with design basis seismic events without loss of structural integrity. In addition, the design incorporates provisions so that bolts do not lose torque during any vibratory motion, and incorporates restraints to preclude radial and axial movement of the strainer.

Low carbon 304 stainless steel is used as the primary material for ECCS suction strainer to prevent corrosive degradation during periods of inactivity and normal operation.

Perry has established a containment cleanliness program for the control of foreign materials, and other programs to minimize the potential for strainer fouling from operations generated debris.

Perry takes no credit for LOCA generated debris hold up in the drywell, and the design does not include debris interceptors of any kind.

The strainer design requires no operator actions in response to debris accumulations or to otherwise assure availability of adequate NPSH for the ECCS pumps. Perry has control room

indication for the RHR A, RHR B, and HPCS pumps' suction pressure, and control room annunciation for low suction pressure for RHR A, RHR B and HPCS. ECCS pump suction pressure is addressed in the Emergency Operating Procedures (EOPs). Accordingly, no additional safety-related instrumentation is required.

The design of the ECCS suction strainer is passive.

Perry conducts comprehensive inspections during refueling outages to evaluate the cleanliness of the suppression pool. The strainer is periodically monitored by visual inspection for evidence of structural degradation or debris fouling. The frequency of suppression pool inspection and cleaning activities is determined based on plant specific debris collection data.

The large toroidal passive strainer does not require operator actions to prevent the accumulation of debris on the strainer or to mitigate the consequences of debris accumulation. The design of the strainer provides sufficient area to accommodate the maximum quantity of debris that is expected to be produced following a design basis LOCA combined with postulated in situ debris quantities. The Emergency Operating Procedures (EOPs) contain guidance to the operator on the use of alternate water sources to provide a diverse means of providing long-term cooling to the core.

See USAR <Section 6.2.2.2> for a discussion of the ECCS suction strainer compliance with <Regulatory Guide 1.82> as it pertains to debris generation and transport.

NPSH available to the ECCS pumps has been determined in accordance with <Regulatory Guide 1.1>. Pressure drop across the suction strainer is based on results from testing and conservative analysis. The vapor pressure for suppression pool water used in

NPSH calculations for events where significant debris generation is expected is based on a suppression pool bulk water temperature of 185°F, which is the maximum design temperature of the containment. Analyses show maximum suppression pool temperatures to be less than the containment design temperature of 185°F. Containment pressure is assumed to be atmospheric in accordance with <Regulatory Guide 1.1> requirements.

Tests have quantified head loss caused by debris blockage on the strainer. Head loss measured during the testing accounts for the possible filtration of particulates by the debris bed. Tests were conducted to determine the performance characteristics of the passive strainer for the quantities and types of debris predicted following postulated accidents.

6.3.2.2.1 High Pressure Core Spray (HPCS) System

The high pressure core spray (HPCS) system consists of a single motor-driven centrifugal pump located outside the primary containment, a spray sparger in the reactor vessel located above the core (separate from the LPCS sparger) and associated system piping, valves, controls, and instrumentation. The system is designed to operate from normal offsite auxiliary power or from a standby diesel generator supply if offsite power is not available. The piping and instrumentation diagram, <Figure 6.3-7> for the HPCS, shows the system components and their arrangement. The HPCS system process diagram, <Figure 6.3-1>, shows the design operating modes of the system. A simplified system flow diagram showing system injection into the reactor vessel is in <Figure 6.3-1>. The HPCS pump curves are included in <Figure 6.3-75>.

The principal active HPCS equipment is located outside the primary containment. Suction piping is provided from the condensate storage tank and the suppression pool. In the event that the condensate storage

water supply becomes exhausted or is not available, automatic switchover to the suppression pool water source will ensure a closed cooling water supply for continuous operation of the HPCS system. HPCS pump suction is also automatically transferred to the suppression pool if the suppression pool water level exceeds a prescribed value. The condensate storage tank reserves water just for use by the HPCS and RCIC. When the Control Room is notified of the issuance of a tornado warning for the vicinity of the plant, or if a tornado is sighted in the immediate vicinity of the plant, administrative controls require the HPCS suction to be aligned to the tornado missile protected suppression pool.

After the HPCS injection piping enters the vessel, it divides and enters the shroud at two points near the top of the shroud. A semicircular sparger is attached to each outlet. Nozzles are spaced around the spargers to spray the water radially over the core and into the fuel assemblies.

The HPCS discharge line to the reactor is provided with two isolation valves. One of these valves is a hydraulically testable check valve located inside the drywell as close as practical to the reactor vessel. HPCS injection flow causes this valve to open during LOCA conditions (i.e., no power is required for valve actuation during LOCA). If the HPCS line should break outside the containment, the check valve in the line inside the drywell will prevent loss of reactor water outside the containment. The other isolation valve (which is also referred to as the HPCS injection valve) is a motor-operated gate valve located outside the primary containment as close as practical to HPCS discharge line penetration into the containment. This valve is capable of opening with the maximum differential pressure across the valve expected for any system operating mode including HPCS pump shutoff head. The valve will reach the position required to deliver rated flow within 29 seconds following the initiation of a LOCA signal per the design basis analysis. This valve is normally

closed to back up the inside testable check valve for containment integrity purposes. A test line is provided between the two valves. The test connection line is normally closed with two valves to assure containment integrity.

Remote controls for operating the motor-operated components and diesel generator are provided in the main control room. The controls and instrumentation of the HPCS system are described, illustrated and evaluated in detail in <Chapter 7>.

The location and type of the manual valves in the HPCS system are detailed in <Table 6.3-7> <Figure 6.3-7>. Design considerations are given to protect the system's safety functions from an undetected, incorrect positioning of any of these manual valves. Administrative controls likewise serve to minimize the possibility of such errors. These design/operations features are outlined in <Table 6.3-8> <Section 6.3.2.8>.

The system is designed to pump water into the reactor vessel over a wide range of pressures. For small breaks that do not result in rapid reactor depressurization, the system maintains reactor water level and depressurizes the vessel. For large breaks the HPCS system cools the core by a spray.

If a LOCA should occur, a low water level signal or a high drywell pressure signal initiates the HPCS and its support equipment. The system can also be placed in operation manually.

The HPCS system is capable of delivering rated flow into the reactor vessel within 29 seconds following receipt of an automatic initiation signal.

When a high water level in the reactor vessel is signaled, the HPCS is automatically stopped by a signal to the injection valve to close.

The HPCS system also serves as a backup to the RCIC system in the event the reactor becomes isolated from the main condenser during operation and feedwater flow is lost.

If normal auxiliary power is not available, the HPCS pump motor is driven by its own onsite power source. The HPCS standby power source is discussed in <Section 8.3>.

The HPCS pump head flow characteristic used in LOCA analyses is shown in <Figure 6.3-4>. When the system is started, initial flow rate is established by primary system pressure. As vessel pressure decreases, flow will increase. When vessel pressure reaches 200 psid (differential pressure between the reactor vessel and the suction source) the system reaches rated core spray flow. The HPCS motor size is based on peak horsepower requirements.

The elevation of the HPCS pump (suction nozzle centerline at Elevation 571'-7") is sufficiently below the water level of both the condensate storage tank and the suppression pool to provide a flooded pump suction.

The minimum net positive suction head (NPSH) requirement specified by the manufacturer at the maximum allowed runout flow of 7,655 gpm is 4.5 feet at a location three feet above the pump mounting flange for pump suction from both the suppression pool and condensate storage tank.

NPSH requirements are met by providing adequate suction head and suction line size. The available NPSH, calculated in accordance with <Regulatory Guide 1.1>, is based on the following design conditions:

- a. Pump maximum runout flow from the suppression pool and the condensate storage tank.
- b. Containment at atmospheric pressure. Condensate storage tank (CST) at atmospheric pressure.

- c. Maximum suppression pool temperature of 185°F. Maximum CST temperature of 125°F.
- d. CST at automatic transfer water level of Elevation 626'-8". Suppression pool at minimum drawdown water level of Elevation 589'-0".
- e. HPCS pump suction piping arrangement as shown on <Figure 6.3-7>.
- f. Suction strainer fully loaded (i.e., maximum postulated loading resulting from LOCA-generated and pre-LOCA debris materials at bounding conditions) with an approximate 6.25 ft pressure drop.

The available NPSH at the time immediately preceding automatic transfer from the condensate storage tank to the suppression pool is approximately 67 feet, based on the above design conditions.

The available NPSH with suction taken from the suppression pool is approximately 19 feet, at 3 feet above the pump mounting flange, based on the above design conditions.

If the SPCU system is in operation during HPCS initiation, there is a brief period of time that flow could be to both the HPCS pump and to the SPCU system (occurs due to the overlap in SPCU system isolation valve closure time and the HPCS initiation time). The NPSH requirements of the HPCS pump during this brief period of concurrent operation are satisfied when calculated in accordance with <Regulatory Guide 1.1>.

The final design calculations, based on the <Regulatory Guide 1.1> position, indicate an available NPSH for the HPCS system sufficient to ensure pump performance capable of accomplishing the required safety functions in both modes.

For preoperational testing the HPCS pump is provided with a test line back to the condensate storage tank. During preoperational testing the HPCS pump was tested for flow capacity and suction pressure, since the condensate storage tank is pumped down to the low level transfer point. From this test the flow capacity and NPSH were compared to the vendor data, and visual observations were made for vortexing.

A motor-operated valve is provided in the suction line from the suppression pool. The valve is located as close to the suppression pool penetration as practical. This valve is used to isolate the suppression pool water source when the HPCS system suction is from the condensate storage system and to isolate the system from the suppression pool in the event a leak develops in the HPCS system.

The HPCS pump characteristics, head, flow, horsepower, and required NPSH are shown in <Figure 6.3-75>.

The design pressure and temperature of the system components are established based on the ASME Section III Boiler and Pressure Vessel Code. The design pressures and temperatures, at various points in the system can be obtained from the miscellaneous information blocks on the HPCS process diagram, <Figure 6.3-1>.

A check valve, flow element and restricting orifice are provided in the HPCS discharge line from the pump to the injection valve. The check valve is located below the minimum suppression pool water level and is provided so the piping downstream of the valve can be maintained full of water by the discharge line fill system <Section 6.3.2.2.5>. The flow element is provided to measure system flow rate during LOCA and test conditions and for automatic control of the minimum low flow bypass gate valve. The measured flow is indicated in the main control room.

The restricting orifice is sized during preoperational testing of the system to limit system flow to acceptable values as described on the HPCS system process diagram, <Figure 6.3-1>.

A low flow bypass line with a motor-operated gate valve connects to the HPCS discharge line upstream of the check valve on the pump discharge line. The line bypasses water to the suppression pool via the 12-inch full-flow test return line to prevent pump damage due to overheating when other discharge line valves are closed. The valve automatically closes when flow in the main discharge line is sufficient to provide the required pump cooling. Water hammer effects due to opening of this line into the 12-inch full-flow test return line are prevented by a vacuum breaker inside containment, which ensures the test return line is filled with air.

The HPCS system incorporates relief valves to protect the components and piping from inadvertent overpressure conditions. One relief valve, set to relieve at 1,560 psig, is located on the discharge side of the pump downstream of the check valve to relieve thermally expanded fluid. A second relief valve is located on the suction side of the pump and is set at 100 psig with a capacity of >10 gpm - 10% accumulation. To relieve thermally expanded fluid between the two isolation valves of the test line to the condensate storage tank, a bypass line and check valve are installed around the inboard isolation valve (F010). Relief valve F035 of the HPCS system discharges to the 12-inch full-flow test return line, which terminates below the surface of the suppression pool. For this valve, the dynamic loads such as thrust and momentum caused by relief valve opening are calculated for the discharge piping and include such effects as backpressure caused by submergence of the discharge piping in the suppression pool. The dynamic loads have been included in the piping stress analysis.

The discharge line is supported as Seismic Category I piping, which includes waterhammer effects in the dynamic analysis of the piping. The discharge line is supported as deemed necessary by the piping analysis to preclude any adverse effects from waterhammer.

The HPCS components and piping are positioned to avoid damage from the physical effects of design basis accidents, such as pipe whip, missiles, high temperature, pressure, and humidity.

The HPCS equipment and support structures are designed in accordance with Seismic Category I criteria <Section 3.2>. The system is assumed to be filled with water for seismic analysis.

Provisions are included in the HPCS system which permits the HPCS system to be tested. These provisions are:

- a. All active HPCS components are testable during normal plant operation.
- b. A full flow test line is provided to route water from and to the condensate storage tank without entering the reactor pressure vessel. The suction line from the condensate tank also provides reactor grade water to fully test the HPCS including injection into the RPV during shutdown.
- c. A full flow test line is provided to route water from and to the suppression pool without entering the reactor pressure vessel.
- d. Instrumentation is provided to indicate system performance during normal test operations.
- e. All motor-operated valves are capable of either local or remote-manual operation for test purposes.

- f. System relief valves are removable for bench-testing during plant shutdown.

6.3.2.2.2 Automatic Depressurization System (ADS)

If the RCIC and HPCS cannot maintain the reactor water level, the automatic depressurization system, which is independent of any other ECCS, reduces the reactor pressure so that flow from LPCI and LPCS systems enters the reactor vessel in time to cool the core and limit fuel cladding temperature.

The automatic depressurization system employs the nuclear system pressure relief valves to relieve high pressure steam to the suppression pool. The design, number, location, description, operational characteristics, and evaluation of the pressure relief valves are discussed in detail in <Section 5.2.2>. The operation of the ADS is discussed in <Section 7.3.1>.

6.3.2.2.3 Low Pressure Core Spray (LPCS) System

The low pressure core spray system consists of: a centrifugal pump that can be powered by normal auxiliary power or the standby ac power system; a spray sparger in the reactor vessel above the core (separate from the HPCS sparger); piping and valves to convey water from the suppression pool to the sparger; and associated controls and instrumentation. <Figure 6.3-8>, the LPCS system P&ID, presents the system components and their arrangement. The LPCS system process diagram, <Figure 6.3-2>, shows the design operating modes of the system. <Figure 6.3-2> includes a simplified system flow diagram showing injection into the reactor vessel by the LPCS system. The LPCS pump curves are included in <Figure 6.3-76>.

When low water level in the reactor vessel or high pressure in the drywell is sensed, the low pressure core spray system automatically

starts. When reactor pressure is below LPCS system design pressure, the injection valve opens and the system sprays water into the top of the fuel assemblies to cool the core. The LPCS injection piping enters the vessel, divides and enters the core shroud at two points near the top of the shroud. A semicircular sparger is attached to each outlet. Nozzles are spaced around the sparger to spray the water radially over the core and into the fuel assemblies.

The LPCS is designed to provide cooling to the reactor core only when the reactor vessel pressure is low, as is the case for large LOCA break sizes. However, when the LPCS operates in conjunction with the ADS the effective core cooling capability of the LPCS is extended to all break sizes because the ADS will rapidly reduce the reactor vessel pressure to within the LPCS operating range. The system head flow characteristic assumed for LOCA analyses is shown in <Figure 6.3-5>.

The low pressure core spray pump and all motor-operated valves can be operated individually by manual switches located in the control room. Operating indication is provided in the control room by a flowmeter and valve indicator lights.

The location and type of the manual valves in the LPCS system are detailed in <Table 6.3-9> <Figure 6.3-8>. Design considerations are given to protect the system's safety functions from an undetected, incorrect positioning of any of these manual valves. Administrative controls likewise serve to minimize the possibility of such errors. These design/operations features are outlined in <Table 6.3-9> <Section 6.3.2.8>.

To assure continuity of core cooling, signals to isolate the containment do not operate any low pressure core spray system valves.

The LPCS discharge line to the reactor is provided with two isolation valves. One of these valves is a hydraulically testable check valve

located inside the drywell as close as practical to the reactor vessel. LPCS injection flow causes this valve to open during LOCA conditions (i.e., no power is required for valve actuation during LOCA). If the LPCS line should break outside the containment the check valve in the line inside the drywell will prevent loss of reactor water outside the containment.

The other isolation valve (which is also referred to as the LPCS injection valve) is a motor-operated gate valve located outside the primary containment as close as practical to LPCS discharge line penetration into the containment. This valve is capable of opening with the maximum differential across the valve expected for any system operating mode. The valve is capable of opening against a differential pressure equal to normal reactor pressure minus the minimum LPCS system shutoff pressure. The valve will reach a position required to deliver rated flow within 32 seconds following receipt of the pressure permissive following a maximum recirculation line break accident. This valve is normally closed to back up the inside testable check valve for containment integrity purposes. A test line is provided between the two valves. The test connection line is normally closed with two valves to assure containment integrity.

The LPCS system components and piping are arranged to avoid unacceptable damage from the physical effect of design-basis accidents, such as pipe whip, missiles, high temperature, pressure, and humidity.

All principal active LPCS equipment is located outside the primary containment.

A check valve, flow element and restricting orifice are provided in the LPCS discharge line from the pump to the injection valve. The check valve is located below the minimum suppression pool water level and is provided so the piping downstream of the valve can be maintained full of

water by the discharge line fill system <Section 6.3.2.2.5>. The flow element is provided to measure system flow rate during LOCA and test conditions and for automatic control of the minimum low flow bypass gate valve. The measured flow is indicated in the main control room. The restricting orifice is sized during preoperation test of the system to limit system flow to acceptable values as shown on <Figure 6.3-2>.

The LPCS pump (pump performance test results) characteristics, head, flow, horsepower, and required NPSH are shown in <Figure 6.3-5> and <Figure 6.3-76>.

A low flow bypass line with a motor-operated gate valve connects to the LPCS discharge line upstream of the check valve on the pump discharge line. The line bypasses water to the suppression pool to prevent pump damage due to overheating when other discharge line valves are closed or reactor pressure is greater than the LPCS system discharge pressure following system initiation. The valve automatically closes when flow in the main discharge line is sufficient to provide required pump cooling.

LPCS flow passes through a motor-operated pump suction valve that is normally open. This valve can be closed by a remote-manual switch (located in the control room) to isolate the LPCS system from the suppression pool should a leak develop in the system. This valve is located in the core spray pump suction line as close to the suppression pool as practical. A closed loop is established for the spray water escaping from the break.

The design pressure and temperature of the system components are established based on the ASME Section III boiler and pressure vessel code. The design pressures and temperatures at various points in the system can be obtained from the miscellaneous information blocks on the LPCS process diagram <Figure 6.3-2>.

The LPCS pump is located in the auxiliary building sufficiently below the water level in the suppression pool to assure a flooded pump suction and to meet pump NPSH requirements are met with the containment at atmospheric pressure and the suction strainer fully loaded (i.e., maximum postulated loading resulting from LOCA-generated and pre-LOCA debris materials). A pressure gauge is provided to indicate the suction head. The available NPSH, calculated in accordance with <Regulatory Guide 1.1>, is based on the following design conditions:

- a. Pump design maximum runout flow of 7,800 gpm
- b. Atmospheric containment pressure
- c. Maximum suppression pool water temperature of 185°F
- d. Suppression pool minimum design water level at Elevation 589'-0"
- e. LPCS pump suction nozzle centerline at Elevation 571'-7"
- f. Suction strainer fully loaded (3.2 feet).

The minimum NPSH requirement is, as specified by the manufacturer, 5 feet at a point 2 feet above the top of the pump mounting flange. This reference point is two feet, 1-1/4 inches above the pump suction nozzle. The calculated minimum available NPSH for the LPCS pump is 28.4 feet. The LPCS pump characteristics are shown on <Figure 6.3-76>. The LPCS system incorporates relief valves to prevent the components and piping from inadvertent overpressure conditions. One relief valve, located on the pump discharge, is set at approximately 600 psig with a capacity of 100 gpm at 10 percent accumulation. The second relief valve is located on the suction side of the pump and is set for 100 psig at a capacity of 10 gpm at 10 percent accumulation. Relief Valve F018 of the LPCS system has a discharge line terminating below the surface of the suppression pool. For this valve, the dynamic loads such as thrust and

momentum caused by relief valve opening are calculated for the discharge piping and include such effects as backpressure caused by submergency of the discharge piping in the suppression pool. The dynamic loads have been included in the piping stress analysis.

The discharge line is supported as Seismic Category I piping, which includes waterhammer effects in the dynamic analysis of the piping. The discharge line is supported as deemed necessary by the piping analysis to preclude any adverse effects from waterhammer.

The LPCS system piping and support structures are designed in accordance with Seismic Category I criteria <Section 3.2>. The system is assumed to be filled with water for seismic analysis.

Provisions are included in the LPCS system which permits the LPCS system to be tested. These provisions are:

- a. All active LPCS components are testable during normal plant operation.
- b. A full flow test line is provided to route water from and to the suppression pool without entering the reactor pressure vessel.
- c. A suction test line supplying reactor grade water is provided to test pump discharge into the reactor pressure vessel during normal plant shutdown.
- d. Instrumentation is provided to indicate system performance during normal and test operations.
- e. All motor-operated valves and check valves are capable of operation for test purposes.

- f. Relief valves are removable for bench testing during plant shutdown.

6.3.2.2.4 Low Pressure Coolant Injection (LPCI) System

The low pressure coolant injection subsystem is an operating mode of the RHR system. The LPCI system is automatically actuated by low water level in the reactor or high pressure in the drywell and uses the three RHR motor-driven pumps to draw suction from the suppression pool and inject cooling water flow into the reactor core and accomplish cooling of the core by flooding. Each loop has its own suction and discharge piping and separate vessel nozzle which connects with the core shroud to deliver flooding water on top of the core. The system is a high volume core flooding system.

The LPCI system, like the LPCS system, is designed to provide cooling to the reactor core only when the reactor vessel pressure is low, as is the case for large LOCA break sizes. However, when the LPCI operates in conjunction with the ADS then the effective core cooling capability of the LPCI is extended to all break sizes because the ADS will rapidly reduce the reactor vessel pressure to the LPCI operating range. The head flow characteristics assumed in the LOCA analyses for the LPCI pumps are shown in <Figure 6.3-6>. The LPCI pump curves are included in <Figure 6.3-77>.

<Figure 6.3-3> shows a process diagram and process data for the RHR system, including LPCI. The RHR pumps receive power from ac power buses having standby power source backup supply. Two RHR pump motors and the associated automatic motor-operated valves receive ac power from one bus, while the LPCS pump and the other RHR pump motor and valves receive power from another bus <Section 8.3>.

The pump, piping, controls, and instrumentation of the LPCI loops are separated and protected so that any single physical event, or missiles generated by rupture of any pipe in any system within the drywell, cannot make all loops inoperable.

To assure continuity of core cooling, signals to isolate the primary containment do not operate any RHR system valves which interfere with the LPCI mode of operation.

Each LPCI discharge line to the reactor is provided with two isolation valves. The valve inside the drywell is a testable check valve and the valve outside the drywell is a motor-operated gate valve. No power is required to operate the check valve inside of the drywell; rather, it opens as a result of LPCI injection flow. If a break were to occur outboard of the check valve it would close to isolate the reactor from the line break.

The motor-operated valve outside of the drywell is called the LPCI injection valve and is located as close as practical to the drywell wall. It is capable of opening against the maximum differential expected for the LPCI modes (i.e., normal reactor pressure minus the upstream pressure with the RHR pump running at minimum flow).

The valve will reach a position required to deliver rated flow within 32 seconds following a maximum recirculation line break accident and receipt of the pressure permissive.

The process diagram <Figure 6.3-3> and P&ID <Figure 5.4-13> indicate a great many flow paths are available other than the LPCI injection line. However, the low water level or high drywell pressure signals which automatically initiate the LPCI mode are also used to isolate all other modes of operation and revert other system valves to the LPCI lineup.

The heat exchanger bypass valves will receive a signal to automatically open 110 seconds after a LOCA in order to provide rated LPCI flow to the reactor pressure vessel. During standby LPCI lineup, these valves are positioned full open and hence are not affected by the 110 second delay to auto open. During operation in the suppression pool cooling or test return mode, the 110 second time delay will allow the suppression pool test return valve to close under nominal flow rate conditions prior to auto opening the heat exchanger bypass valves.

The heat exchanger outlet valves also receive a signal to automatically open 110 seconds after a LOCA and are controlled in a manner similar to the heat exchanger bypass valves. The heat exchanger inlet valves receive no automatic signals to open, however, these valves are administratively maintained in a full open position to provide flow to the heat exchangers and ensure LPCI mode operability.

Design considerations have been given to protect the LPCI mode safety functions from an undetected, incorrect positioning of the manual valves in the system. Administrative controls likewise serve to minimize the possibility of such errors. The location and type of these valves as well as the design/operations features are outlined in <Table 6.3-10> and <Section 6.3.2.8>.

A check valve in the pump discharge line is used together with a discharge line fill system <Section 6.3.2.2.5> to prevent water hammer resulting from pump start against a potential shutoff condition. A flow element in the pump discharge line is used to provide a measure of system flow and to originate automatic signals for control of the pump minimum flow valve. The minimum flow valve permits a small flow to the suppression pool in the event no discharge valve is open, or in the case of a LOCA, the vessel pressure is higher than pump shutoff head.

Using the suppression pool as the source of water for the LPCI establishes a closed loop for recirculation of LPCI water escaping from the break.

The design pressures and temperatures at various points in the system, during each of the several modes of operation of the RHR subsystems, can be obtained from the miscellaneous information blocks on the LPCI process diagram, <Figure 6.3-3>.

LPCI pumps and equipment are described in detail in <Section 5.4.7>, which also describes the other functions served by the same pumps if not needed for the LPCI function.

The heat exchangers are discussed in <Section 6.2.2>. The portions of the RHR required for accident protection including support structures are designed in accordance with Seismic Category I criteria <Section 3.2>. The LPCI pump characteristics are shown in <Figure 6.3-6>.

The LPCI system incorporates a relief valve on each of the pump discharge lines which protects the components and piping from inadvertent overpressure conditions. These valves, E12-F025A, B and C, are set to relieve pressure at 485 psig, which is below the system design pressure of 500 psig. The following relief valves in the RHR system have discharge lines terminating below the surface of the suppression pool:

- a. F055A & B
- b. F025A, B & C
- c. F005

For these valves, the dynamic loads such as thrust and momentum caused by relief valve opening are calculated for the discharge piping and

include such effects as backpressure caused by submergence of the discharge piping in the suppression pool. The dynamic loads have been included in the piping stress analysis.

The discharge lines are supported as Seismic Category I piping, which includes waterhammer effects in the dynamic analysis of the piping. The discharge lines are supported as deemed necessary by the piping analysis to preclude any adverse effects from waterhammer.

Provisions are included in the LPCI system to permit testing of the system. These provisions are:

- a. All active LPCI components are designed to be testable during normal plant operation.
- b. A discharge test line is provided for the three pump loops to route suppression pool water back to the suppression pool without entering the reactor pressure vessel.
- c. A suction test line, supplying reactor grade water, is provided to test loop "C" discharge into the reactor pressure vessel during normal plant shutdown.
- d. Instrumentation is provided to indicate system performance during normal and test operations.
- e. All motor-operated valves and check valves are capable of manual operation for test purposes.
- f. Shutdown lines taking suction from the recirculation system are provided for loops "A" and "B" to test pump discharge into the reactor pressure vessel after normal plant shutdown and to provide for shutdown cooling.

- g. All relief valves are removable for bench-testing during plant shutdown.

6.3.2.2.5 ECCS Discharge Line Fill System

A requirement of the core cooling systems is that cooling water flow to the reactor vessel be initiated rapidly when the system is called on to perform its function. This quick-start system characteristic is provided by quick-opening valves, quick-start pumps and standby ac power source. The lag between the signal to start the pump and the initiation of flow into the RPV can be minimized by keeping the core cooling pump discharge lines full. Additionally, if these lines were empty when the systems were called for, the large momentum forces associated with accelerating fluid into a dry pipe could cause physical damage to the piping. Therefore, the ECCS discharge line fill system is designed to maintain the pump discharge lines in a filled condition. The systems are filled and vented to remove any potentially damaging air or non-condensables.

Since the ECCS discharge lines are elevated above the suppression pool, check or stop-check valves are provided near the pumps to prevent back flow from emptying the lines into the suppression pool. Past experience has shown that these valves will leak slightly, producing a small back flow that will eventually empty the discharge piping. To ensure that this leakage from the discharge lines is replaced and the lines are always kept filled, a water leg pump is provided for each of the three ECCS divisions. The power supply to these pumps is classified as essential when the main ECCS pumps are deactivated.

The fill system, typical for each of the three ECCS divisions, consists of a jockey pump that takes suction from the corresponding ECCS division's pump suction line(s) from the suppression pool and discharges downstream of the check valves on the ECCS pump discharge line. The P&ID's for the fill systems are shown on <Figure 5.4-13>, <Figure 6.3-7>, and <Figure 6.3-8>.

Each jockey pump will ensure that the discharge lines are full up to the isolation valves considering conservative estimates of expected leakage through the boundary valves. A typical performance curve for the jockey pumps, defined by parameters of 32.5 psi and 40 gpm, is given in <Figure 6.3-74>. For each ECCS division, the minimum keep-fill pressure and flow requirements have been determined, assuming conservative estimates of leakage from the systems. A significant difference exists between these minimum keep-fill flow and pressure requirements and the flows and pressures to which the pumps are permitted to degrade under the ASME Code required pump testing program. This difference is adequate to meet an unexpected increase in leakage due to equipment deterioration, etc. To prevent overheating of the jockey pumps if the discharge lines valves do not leak, a low-flow bypass line is provided to continuously circulate water back to the ECCS pump suction lines.

Initial filling of the piping systems is accomplished using the combination of jockey pumps, condensate water supply lines (located a minimum distance from filled system boundary valves), maintenance drains, vents, and test connections that are available, as shown on the P&ID's. All potentially damaging air is eliminated from the ECCS pump discharge lines when the fill system is placed into service by opening vents at all piping high points until water begins to flow from the vents. A high point venting procedure is repeated, after initial fill of the system, any time the jockey pump is stopped and restarted, and following any indication of low discharge line pressure. Pressure instrumentation provided on the jockey pump's discharge line initiates an alarm in the control room when pressure in the discharge line is less than the hydrostatic head required to maintain the line full. Indication is also provided in the control room as to when the jockey pumps are operating.

If maintenance on a particular ECCS loop requires draining, the other loops in the system are protected by the isolation of the loop being drained by that lines isolation valves.

The fill system, in accordance with surveillance procedures, is tested to ensure that the discharge lines are full as required. Calibration and functional testing of the pressure switches and associated alarms are performed to ensure their continued ability to perform their desired function.

A small amount of dissolved gas may come out of solution during the interval between surveillance tests and accumulate at the high point vent(s) even though the jockey pump system is functioning properly. This is considered to be a normal phenomenon that will not compromise the system integrity.

6.3.2.2.6 Gas Management

Both the suction and discharge piping in plant systems addressed in responses to Nuclear Regulatory Commission (NRC) Generic Letter (GL) 2008-01, "Managing Gas Accumulation in Emergency Core Cooling, Decay Heat Removal, and Containment Spray Systems," have the potential to develop pockets of entrained gases. "Gas" as used herein includes air, nitrogen, hydrogen, water vapor, or any other void that results in a non-water-solid condition. Maintaining pump suction and discharge piping sufficiently filled with water is necessary to ensure the systems will perform properly and will inject the flow assumed in the safety analyses into the Reactor Coolant System (RCS) or containment upon demand. This will also prevent damage from pump cavitation or water hammer, and pumping of unacceptable quantities of noncondensable gas into the reactor vessel following an injection signal or during operation of Residual Heat Removal (RHR) shutdown cooling. Consistent with the responses to <GL 2008-01> for the Perry Nuclear Power Plant (Reference 21), (Reference 22), (Reference 23), (Reference 24), and NRC closure of the Generic Letter (Reference 25), gas management calculations and procedures address the following elements:

- a. Identification of the systems/subsystems for which gas accumulation is managed;

- b. Identification of locations susceptible to gas accumulation that should be monitored in suction and discharge piping, and the type of monitoring performed;
- c. Identification of gas void performance-based monitoring frequencies and modifications to the frequencies;
- d. Determination of appropriate corrective actions for identified gas accumulations;
- e. Identification of void acceptance criteria; and
- f. Identification of methodology used to determine the void acceptance criteria.

6.3.2.3 Applicable Codes and Classifications

The applicable codes and classification of the ECCS are specified in <Section 3.2>. All piping systems and components (pumps, valves, etc.) for the ECCS comply with applicable codes, addenda, code cases, and errata in effect at the time the equipment is procured. The piping and components of each ECCS within the containment and out to and including the pressure retaining injection valve are Safety Class 1. The remaining piping and components are Safety Class 2, 3 or non-code as indicated in <Section 3.2>, and as indicated on the individual system P&ID. The equipment and piping of the ECCS are designed to the requirements of Seismic Category I. This seismic designation applies to all structures and equipments essential to the core cooling function. IEEE codes applicable to the controls and power supplies are specified in <Section 7.1>.

6.3.2.4 Material Specifications and Compatibility

Materials specifications and compatibility for the ECCS are presented in <Section 6.1> and <Section 3.2>. Nonmetallic materials such as lubricants, seals, packings, paints, and primers, insulation, as well as

metallic materials, etc., are selected as a result of an engineering review and evaluation for compatibility with other materials in the system and the surroundings with concern for chemical, radiolytic, mechanical, and nuclear effects. Materials used are reviewed and evaluated with regard to radiolytic and pyrolytic decomposition and attendant effects on safe operation of the ECCS.

6.3.2.5 System Reliability

A single failure analysis shows that no single failure prevents the starting of the ECCS when required, or the delivery of coolant to the reactor vessel. No individual system of the ECCS is single failure proof with the exception of the ADS; hence, it is expected that single failures will disable individual systems of the ECCS. The most severe effects of single failures with respect to loss of equipment occur if the LOCA occurs in combination with an ECCS pipe break coincident with a loss of offsite power. The consequences of the most severe single failures are shown in <Table 6.3-3>.

6.3.2.6 Protection Provisions

Protection provisions are included in the design of the ECCS. Protection is afforded against missiles, pipe whip and flooding. Also accounted for in the design are thermal stresses, loadings from a LOCA and seismic effects.

The ECCS piping and components located outside the drywell are protected from internally and externally generated missiles by the reinforced concrete structure of the ECCS pump rooms.

The ECCS is capable of withstanding the passive failure of valve stem packings and pump seals following a LOCA. The water tight construction of these ECCS pump rooms would contain any system leakage within the

affected ECCS room, thereby preventing common mode flooding of the ECCS rooms, as described in <Section 9.3.3>. The maximum leakage due to a failure of this nature would be 23 gpm or less from an HPCS, LPCS or RHR pump seal failure. Valve stem leakage would be significantly less than this leakage. Each ECCS pump room is provided with leak detection capabilities <Section 9.3.3> and <Section 6.2.4.2.2.2> which would identify the faulted ECCS with an alarm system in the control room. The operator would then remotely isolate the influent and effluent containment isolation valves for the affected system. Based on operator action 30 minutes after the alarm, a total of 690 gallons would be spilled in the affected ECCS room. This reduction in suppression pool water inventory is insignificant and would not reduce the suppression pool water level below the minimum drawdown level. The design of the ECCS suction valves, which are flexible wedge gate valves, is such that the valve seal (packing) is isolated from the system when the valve is closed. See <Section 3.6.2.3.5.2> for discussion of unisolable/isolable leaks during non-accident (normal operation) and accident conditions.

The ECCS is protected against the effects of pipe whip, which might result from piping failures up to and including the design basis event LOCA. This protection is provided by separation, pipe whip restraints and energy absorbing materials. These three methods are applied to provide protection against damage to piping and components of the ECCS which otherwise could result in a reduction of ECCS effectiveness to an unacceptable level. See <Section 3.6> for criteria on pipe whip.

The component supports which protect against damage from movement and from seismic events are discussed in <Section 5.4.14>. The methods used to provide assurance that thermal stresses do not cause damage to the ECCS are described in <Section 3.9>.

6.3.2.7 Provisions for Performance Testing

Periodic system and component testing provisions for the ECCS are described in <Section 6.3.2.2> as part of the individual system descriptions.

6.3.2.8 Manual Actions

The ECCS is actuated automatically and requires no operator action during the first 10 minutes following the accident. During the long term cooling period (after 10 minutes), the operator will take action as specified in <Section 6.2.2> to place the containment cooling system into operation.

The operator has multiple instrumentation available in the control room to assist him in assessing the post-LOCA conditions. This instrumentation provides reactor vessel pressures, water levels, containment pressure, temperature, and radiation levels as well as indicating the operation of the ECCS. ECC system flow indication is the primary parameter available to assess proper operation of the system. Other indications such as position of valves, status of circuit breakers and essential power bus voltage are also available to assist him in determining system operating status. The electrical and instrumentation complement to the ECCS is discussed in detail in <Section 7.3>. Other available instrumentation is listed in the P&IDs for the individual systems. Much of the monitoring instrumentation available to the operator is discussed in more detail in <Chapter 5> and <Section 7.3>.

6.3.3 PERFORMANCE EVALUATION

This section provides the results of the ECCS Loss-of-Coolant Accident (LOCA) evaluation. Note, the initial core through Cycle 7 used the SAFE/REFLOOD/CHASTE methodology (Reference 1). Cycle 8 and

subsequent operating cycles are based on the SAFER/GESTR-LOCA methodology (Reference 4). (Reference 5) documents the analysis based on the SAFER/GESTR-LOCA methodology in support of cycle 8 at a core power level of 3,579 MWt for a new licensing basis. The licensing results have been revised and included in (Reference 18). The limiting case is reanalyzed for cycle 8 at a core power level of 3,758 MWt and results are documented in (Reference 18).

The SAFER/GESTR-LOCA methodology evaluates the short-term and long-term reactor vessel blowdown response to a pipe rupture, the subsequent fuel heatup, and core reflooding by the ECCS. The reactor vessel pressure and water levels, ECCS performance, and other primary thermal-hydraulic phenomena are predicted as a function of time. These predictions are then used to determine the fuel cladding temperature under various accident conditions.

The SAFER/GESTR-LOCA application methodology consists of three main parts. First, potentially limiting LOCA cases are determined by applying realistic (nominal) analytical models across the entire break spectrum of postulated breaks. Second, limiting LOCA cases are analyzed with an Appendix K model (inputs and assumptions) which incorporates the required features of <10 CFR 50, Appendix K>. For the most limiting cases, a Licensing Basis Peak Cladding Temperature (PCT) is calculated based on the nominal PCT with an adder to account statistically for the differences between the nominal and <10 CFR 50, Appendix K> assumptions. Finally, a statistically derived Upper Bound PCT is calculated to demonstrate the conservatism of the Licensing Basis PCT. The Licensing Basis PCT conforms to all the requirements of <10 CFR 50.46> and <10 CFR 50, Appendix K>.

The performance of the ECCS with the new (reload) fuel types is presented in <Appendix 15B>, Reload Safety Analysis.

The ECCS performance is evaluated for the entire spectrum of break sizes for postulated LOCA's. The accidents, as listed in <Chapter 15>, for which ECCS operation is required are:

- a. Feedwater line break
- b. Steam system piping break outside of containment
- c. Loss-of-coolant accidents

<Chapter 15> provides the radiological consequences of the above listed events.

6.3.3.1 ECCS Bases for Technical Specifications

The maximum average planar linear heat generation rates (MAPLHGRs) calculated in this performance analysis provide the basis for technical specifications designed to ensure conformance with the acceptance criteria of <10 CFR 50.46>. Minimum ECCS functional requirements are specified in <Section 6.3.3.4> and <Section 6.3.3.5> and testing requirements are discussed in <Section 6.3.4>. Limits on minimum suppression pool water level are discussed in <Section 6.2>.

6.3.3.2 Acceptance Criteria for ECCS Performance

The applicable acceptance criteria, extracted from <10 CFR 50.46>, "Acceptance Criteria for Emergency Core Cooling Systems for Light-Water-Cooled Nuclear Power Reactors," are listed below.

Criterion 1 - Peak Cladding Temperature: The calculated maximum fuel element cladding temperature shall not exceed 2200°F.

Criterion 2 - Maximum Cladding Oxidation: The calculated total local oxidation shall not exceed 0.17 times the total cladding thickness before oxidation.

Criterion 3 - Maximum Hydrogen Generation: The calculated total amount of hydrogen generated from the chemical reaction of the cladding with water or steam shall not exceed 0.01 times the hypothetical amount that would be generated if all the metal in the cladding cylinder surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react.

Criterion 4 - Coolable Geometry: Calculated changes in core geometry shall be such that the core remains amenable to cooling.

Criterion 5 - Long-Term Cooling: After any calculated successful initial operation of the ECCS, the calculated core temperature shall be maintained at an acceptably low value and decay heat shall be removed for the extended period of time required by the long-lived radioactivity remaining in the core.

The conformance with Criteria 1 through 3 is presented in (Reference 5) and summarized in <Section 6.3.3.7.3>. As discussed in (Reference 1), Section III.A, and presented in (Reference 5), conformance to Criterion 4 is demonstrated by conformance to Criteria 1 and 2. Conformance to Criterion 5 is demonstrated generically for General Electric BWRs in (Reference 1), Section III.A. (Reference 5). This remains unchanged by application of SAFER/GESTR-LOCA methodology.

6.3.3.3 Single Failure Considerations

The functional consequences of potential operator errors and single failures (including those which might cause any manually controlled electrically operated valve in the ECCS to move to a position which

could adversely affect the ECCS), and the potential for submergence of valve motors in the ECCS are discussed in <Section 6.3.2>. There it was shown that all potential single failures are no more severe than one of the single failures identified in <Table 6.3-3>. Therefore, it is only necessary to consider each of these single failures in the performance analyses. For large breaks, failure of one of the diesel standby generators is in general the most severe failure. For small breaks, the HPCS is the most severe failure.

6.3.3.4 System Performance During the Accident

In general, the system response to an accident can be described as:

- a. receiving an initiation signal,
- b. a small lag time (to open all valves and have the pumps up to rated speed) and
- c. the ECCS flow entering the vessel.

Key ECCS initiating signals and time delays for all the ECC systems are provided in <Table 6.3-1>. The minimization of the delay from the receipt of signal until the ECCS pumps have reached rated speed is limited by the physical constraints on accelerating the diesel generators and pumps. The delay time due to valve motion in the case of high pressure system provides a suitably conservative allowance for valves available for this application. In the case of the low pressure system, the time delay for valve motion is such that the pumps are at rated speed prior to the time the vessel pressure reaches the pump shutoff pressure.

The flow delivery rates analyzed in <Section 6.3.3> can be determined from the head-flow curves in <Figure 6.3-4>, <Figure 6.3-5>, and

<Figure 6.3-6> of <Section 6.3.2>. The ECCS leakage flow is accounted for in the analysis. The ECCS piping inside the vessel (between the vessel wall and shroud) has various leakage paths through slip joints and vent holes. Some of the ECCS water injected into the vessel is lost through these leakage paths into the downcomer region before reaching the region inside the shroud. The standard system leakage rates assumed are 80 gpm per LPCI pump and 100 gpm for HPCS and for LPCS.

For the low pressure ECCS, there are two logic paths in SAFER that determine the time of flow injection. The first path is the system startup time and is governed by the diesel start time, electrical load sequencing, pump start time, and injection valve stroke time. The second path is governed by valve pressure permissive and injection valve stroke time. The limiting (longest) delays for the system startup and pressure permissive paths are determined and used for the analysis. In specific, for low pressure ECCS, no injection flow is assumed until 32 seconds following receipt of the low pressure injection valve pressure permissive. For high pressure ECCS, a 29 second system delay time from the accident initiation signal is assumed, encompassing the diesel start time plus time for the injection valve to open.

Simplified piping and instrumentation and functional control diagrams for the ECCS are provided in <Section 6.3.2>. The operational sequence of ECCS for the DBA is presented in <Table 6.3-2>.

Operator action is not required, except as a monitoring function, during the short term cooling period following the LOCA. During the long term cooling period, the operator will take action as specified in <Section 6.2.2> to place the containment cooling system into operation.

6.3.3.5 Use of Dual Function Components for ECCS

With the exception of the LPCI system, the systems of the ECCS are designed to accomplish only one function: to cool the reactor core

following a loss of reactor coolant. To this extent, components or portions of these systems (except for pressure relief) are not required for operation of other systems which have emergency core cooling functions, or vice versa. Because either the ADS initiating signal or the overpressure signal opens the safety/relief valve, no conflict exists.

The LPCI subsystem, however, uses the RHR pumps and some of the RHR valves and piping. When the reactor water level is low, the LPCI subsystem has priority through the valve control logic over the other RHR subsystems for containment cooling or shutdown cooling. Immediately following a LOCA, the RHR system is directed to the LPCI mode.

6.3.3.6 Limits on ECCS System Parameters

The limits on the ECC system parameters are discussed in <Section 6.3.3.1> and <Section 6.3.3.7.1>.

Any number of components in any given system may be out-of-service, up to and including the entire system. The maximum allowable out-of-service time is a function of the level of redundancy and the specified test intervals as generally discussed in <Appendix 15A.5>.

6.3.3.7 ECCS Analyses for LOCA

6.3.3.7.1 LOCA Analysis Procedures and Input Variables

The methodology used for the SAFER/GESTR-LOCA analysis is documented in (Reference 4). A summary of the set of codes used in this analysis is given below.

The significant input parameters are contained in <Table 6.3-1> and on <Figure 6.3-10>.

The Shroud Head Stud Assembly Mechanism (SHSAM) leakage flow, ECCS leakage flow and the flow path of vessel Bottom Head Drain (BHD) connection to the broken recirculation line are accounted for in the analysis.

Short-Term Thermal-Hydraulic Model (LAMB)

The LAMB model (Reference 1) analyzes the short-term blowdown phenomena for postulated large pipe breaks in which nucleate boiling is lost before the water level drops sufficiently to uncover the active fuel. The LAMB output (most importantly, core flow as a function of time) is used in the SCAT/TASC model for calculating blowdown heat transfer and fuel dryout time.

Transient Critical Power Model (SCAT/TASC)

The SCAT model (Reference 1) completes the transient short-term thermal-hydraulic calculation for large recirculation line breaks. The time and location of boiling transition are predicted during the period of recirculation pump coastdown. When the core inlet flow is low, SCAT also predicts the resulting bundle dryout time and location. The calculated fuel dryout time is an input to the long-term thermal-hydraulic transient model, SAFER. For GE11 and later fuel, an improved SCAT model (designated "TASC") is used to predict the time and location of boiling transition and dryout time. This model explicitly models the axially varying flow areas and heat transfer surface resulting from the GE11 part length fuel rods, and incorporates the critical power correlation for GE11 and GE12 (Reference 6) and (Reference 7).

Thermal-Mechanical Model (GESTR-LOCA)

The GESTR-LOCA (Reference 8) model provides the parameters to initialize the fuel stored energy and fuel rod fission gas inventory at the onset

of a postulated LOCA for input to SAFER. GESTR-LOCA also establishes the initial transient pellet-cladding gap conductance for input to both SAFER and SCAT/TASC.

Long Term Thermal-Hydraulic Model (SAFER)

The SAFER model (Reference 9), (Reference 10), (Reference 11), (Reference 12), and (Reference 13) calculates the long-term system response of the reactor over a complete spectrum of hypothetical break sizes and locations. SAFER is compatible with the GESTR-LOCA fuel rod model for gap conductance and fission gas release. SAFER calculates the core and vessel water levels, system pressure response, ECCS performance, and other primary thermal-hydraulic phenomena occurring in the reactor as a function of time. SAFER realistically models all regimes of heat transfer that occur inside the core, and provides the PCT and the heat transfer coefficients (which determine the severity of the temperature change) as a function of time. Part length fuel rods, found in GE11 and later fuel types, are modeled as full-length rods, which conservatively overestimates the hot bundle power.

6.3.3.7.2 Accident Description

A detailed description of the LOCA methodology is provided in (Reference 4). The analysis is documented in (Reference 18). For convenience, a short description of the major events during the design basis accident (DBA) is included here.

Immediately after the postulated double-ended recirculation line break, vessel pressure and core flow begin to decrease. The initial pressure response is governed by the closure of the main steam isolation valves and the relative values of energy added to the system by decay heat and energy removed from the system by the initial blowdown of fluid from the downcomer. The initial core flow decrease is rapid because the recirculation pump in the broken loop ceases to pump almost immediately

because it has lost suction. The pump in the intact loop coasts down relatively slowly. This pump coastdown governs the core flow response for the next several seconds. When the jet pump suction uncovers, calculated core flow decreases to near zero. When the recirculation pump suction nozzle uncovers, the energy release rate from the break increases significantly and the pressure begins to decay more rapidly. As a result of the increased rate of vessel pressure loss, the initially subcooled water in the lower plenum saturates and flashes up through the core, increasing the core flow. This lower plenum flashing continues at a reduced rate for the next several seconds.

The core water level is restored during the rapid depressurization following uncover of the break, and then begins dropping again due to the mass loss out of the break. Once the pressure drops below the shutoff head of the low pressure systems they begin injecting water into the vessel and rapidly restore the water level in the core. In the analysis, no credit is taken for flow until the injection valve is fully open and the reactor pressure is below the system shutoff head. The uncover duration is shorter because the injection of LPCI water into the bypass region makes it possible to reflood the core before the lower plenum subcools.

The calculated fuel cladding temperature results show that the first peak heatup occurs during the early boiling transition (early dryout) period, which is due to the initial stored energy or gap conductance. The second peak heatup is caused by core uncover which is slower, being governed by decay heat and heat transfer rate. Finally the heatup is terminated when the core is recovered by the accumulation of ECCS water.

6.3.3.7.3 Break Spectrum Calculations - Recirculation Line Breaks

Several break sizes are analyzed to determine the limiting single failure <Section 6.3.3.3> using nominal assumptions documented in (Reference 5) and (Reference 18) and the inputs discussed in

<Section 6.3.3.7.1>. The drain line flow path between the vessel bottom head and the broken recirculation line is considered in the analysis. The analysis reported in (Reference 18) established the shape of the PCT versus break area curve (break spectrum) to determine the limiting break. This ensures the limiting combination of break size, location, and single failure had been identified, and is similar to that determined in the generic evaluation (Reference 4). The trend of PCT with break size is consistent with the trend observed in the generic break spectrum.

Once the Design Basis Accident (DBA) which includes the limiting single failure has been determined using nominal assumptions, the event is then analyzed using the <10 CFR 50, Appendix K> methodology. The <10 CFR 50, Appendix K> calculations validate the limiting conditions and provides inputs into the calculation of the licensing and upper bound PCTs.

The Perry analysis demonstrates that the DBA is the recirculation suction line break with a coincident HPCS Diesel Generator (DG) failure. This event results in the same first peak PCT as for LPCI DG and LPCS DG failures using the nominal assumptions. However, the DBA suction break with HPCS DG failure results in a more severe second peak PCT than for the other two single failures for the plant using the nominal assumptions.

The nominal calculations <Figure 6.3-9> show that for large breaks (i.e., $\geq 1.0 \text{ ft}^2$) the calculated PCTs remain flat and decrease with decreasing break size from the DBA to the 1.0 ft^2 range due to the first peak PCT limiting resulted from early boiling transition (dryout). In the small break range (i.e., $< 1.0 \text{ ft}^2$), where the depressurization of the reactor depends on the Automatic Depressurization System (ADS), the calculated PCT increases first with decreasing break size and then decreases again. For small breaks, which do not experience early dryout, the calculated PCT occurs during the core uncover. A 0.1 ft^2 suction line break is limiting as shown in the representative small break spectrum.

The <10 CFR 50, Appendix K> calculations demonstrated that the DBA recirculation suction line break with HPCS DG failure remains the limiting case.

A summary of the results of the break spectrum calculations is shown in tabular form in <Table 6.3-4> and graphically in <Figure 6.3-9>.

6.3.3.7.4 Calculations for Non-Recirculation Line Breaks

Non-recirculation line breaks are analyzed for the limiting fuel type, using nominal assumptions (Reference 4). The analysis considers the HPCS line break, steam line break inside and outside the containment, feedwater line break and LPCI line break. For all of these events the calculated PCTs do not exceed the steady state value during normal operation, and thus are bounded by the recirculation line breaks results.

6.3.3.7.5 Compliance Evaluations

Conformance to the acceptance criteria of <10 CFR 50.46> and <10 CFR 50, Appendix K> is demonstrated by the Licensing Basis PCT. The Licensing Basis PCTs for the plant specific fuel types are calculated for the DBA suction line break with the limiting HPCS DG failure using the results presented in (Reference 18) and the methodology documented in (Reference 4), (Reference 5), and (Reference 18). An adder is added to the nominal PCT to generate the Licensing Basis PCT. The adder incorporates features required by <10 CFR 50, Appendix K> which are not already included in the nominal calculations. Plant variable uncertainties included backflow leakage, ECCS water temperature, ECCS initiation signal, stored energy, gap pressure, and ADS time delay are considered in the plant specific adder. The Licensing Basis PCT is required not to exceed 2200°F.

Finally, the conservatism of the Licensing Basis PCT is demonstrated by comparison to a statistically derived Upper Bound PCT. The statistical based Upper Bound PCT is a function of the limiting case nominal PCT, modeling bias and plant specific variable uncertainties. The evaluation method in (Reference 4) is used to determine the uncertainty term which involves stored energy, decay heat, PLHGR, break flow and the initial MCPR. The Upper Bound PCT is then conservatively determined by adjusting the plant specific difference between the nominal and the derived PCT. The Upper Bound PCT is required to be less than the Licensing Basis PCT.

<Table 6.3-5> provides the Licensing Basis PCT results for the limiting DBA recirculation line suction break.

6.3.3.7.6 Alternate Operating Mode Considerations

The SAFER/GESTR-LOCA analysis was applied to three alternate operating modes: Maximum Extended Operating Domain (MEOD), feedwater temperature reduction, and single loop operation. Details of these operating modes are contained in USAR <Appendix 15D>, <Appendix 15E>, and <Appendix 15F>, respectively. SAFER/GESTR-LOCA results are fuel type specific and, as such, are contained in USAR <Appendix 15B>, Reload Analysis.

6.3.3.8 Conclusions

The Perry SAFER/GESTR-LOCA analysis results provided in <Section 6.3.3.7.3> (Reference 18) demonstrate that a sufficient number of plant-specific PCT points have been evaluated to establish the shape of both the nominal and Appendix K PCT versus break size curves. The analysis also demonstrates that the limiting Licensing Basis PCT occurs for the recirculation suction line DBA.

Compliance with the applicable acceptance criteria provided in <Section 6.3.3.2> is provided in <Table 6.3-5>. With the verification that the Licensing Basis PCT is greater than the Upper Bound PCT, the level of safety and conservatism of this analysis meets the NRC approved acceptance criteria of <10 CFR 50.46> and <10 CFR 50, Appendix K>.

6.3.4 TESTS AND INSPECTIONS

6.3.4.1 ECCS Performance Tests

All systems of the ECCS are tested for their operational ECCS function during the preoperational and/or startup test program. Each component is tested for power source, range, direction of rotation, set point, limit switch setting, torque switch setting, etc. Each pump is tested for flow capacity for comparison with vendor data (this test is also used to verify flow measuring capability). The flow tests involve the same suction and discharge source, i.e., suppression pool or condensate storage tank.

All logic elements are tested individually and then as a system to verify complete system response to emergency signals including the ability of valves to revert to the ECCS alignment from other positions.

Finally, the entire system is tested for response time and flow capacity taking suction from its normal source and delivering flow into the reactor vessel. This last series of tests is performed with power supplied from both offsite power and onsite emergency power.

See <Chapter 14> for a detailed discussion of preoperational testing for these systems.

6.3.4.2 Reliability Tests and Inspections

The average reliability of a standby (non-operating) safety system is a function of the duration of the interval between periodic functional tests. The factors considered in determining the periodic test interval of the ECCS are: the desired system availability (average reliability), the number of redundant functional system success paths, the failure rates of the individual components in the system, and the schedule of periodic tests (simultaneous versus uniformly staggered versus randomly staggered). For the ECCS the above factors were used to determine safe test intervals utilizing the methods described in (Reference 2).

All of the active components of the HPCS system, LPCS system and LPCI systems are designed so that they may be tested during normal plant operation. Full flow test capability is provided by a test line back to the suction source. The full flow test is used to verify the capacity of each ECCS pump loop while the plant remains undisturbed in the power generation mode. In addition, each individual valve may be tested during normal plant operation. Input jacks are provided so that each ECCS loop can be tested for response time during racking out of the injection valve breaker.

All of the active components of the ADS system except the safety/relief valves and their associated solenoid valves are designed so they may be tested during normal plant operation. The safety/relief valves and associated solenoid valves are all tested on a Technical Specification specified frequency during plant startup following a refueling outage. Safety/relief valves and their associated solenoid valves which have been overhauled during a plant outage are tested during the startup following that outage.

Testing of the initiating instrumentation and controls portion of the ECCS is discussed in <Section 7.3.1>. The emergency power system, which supplies electrical power to the ECCS in the event that offsite power is unavailable, is tested as described in <Section 8.3.1>. The frequency of testing is specified in technical specifications. Visual inspections of all the ECCS components located outside the drywell can be made at any time during power operation. Components inside the drywell can be visually inspected only during periods of access to the drywell. When the reactor vessel is open, the spargers and other internals can be inspected.

6.3.4.2.1 HPCS Testing

The HPCS can be tested at full flow with condensate storage tank water at any time during plant operation except when the reactor vessel water level is low, when the condensate level in the condensate storage tank is below the reserve level or when the valves from the suppression pool to the pump are open. If an initiation signal occurs while the HPCS is being tested, the system returns automatically to the operating mode.

The two motor-operated valves in the test line to the condensate storage system are interlocked closed when the suction valve from the suppression pool is open.

The functional test of the HPCS pump is performed at a single pressure and flow calculated to satisfy all design bases flows and pressures. The test is performed by pumping water from either the condensate storage tank, through the full flow test return line, and back to the condensate storage tank or from the suppression pool to the suppression pool.

The suction valve from the unused source and the discharge valve to the reactor remain closed. These valves are tested separately to ensure their operability.

The HPCS test conditions are tabulated on the HPCS process flow diagram, Figure 6.3-1.

The HPCS pre-operational testing was conducted in all modes of operation and included automatic transfer of pump suction from the condensate storage tank to the suppression pool for both modes of initiation (high suppression pool level and low condensate storage tank level). The pump head flow characteristics and NPSH were checked for consistency and design specifications for the various modes of operation. See Section 14.2.12.1.15 for details.

6.3.4.2.2 ADS Testing

The ADS valves are fully tested during the time when the reactor is at reduced pressure prior to or following a refueling outage. This testing includes simulated automatic actuation of the system throughout its emergency operating sequence. Each individual ADS valve is manually actuated.

During plant operation the ADS system can be checked as discussed in <Section 7.3.1>.

6.3.4.2.3 LPCS Testing

The LPCS pump and valves are tested periodically during reactor operation. With the injection valve closed and the return line open to the suppression pool, full flowing pump capability is demonstrated. The injection valve and the check valve are tested in a manner similar to that used for the LPCI valves. The system test conditions during reactor shutdown are shown on the LPCS system process diagram, <Figure 6.3-2>.

6.3.4.2.4 LPCI Testing

Each LPCI loop can be tested during reactor operation. The test conditions are tabulated in <Figure 6.3-3>. During plant operation, this test does not inject cold water into the reactor because the injection valve is not opened. The injection line portion is tested during the shutdown cooling mode of operation. This prevents unnecessary thermal stresses.

To test an LPCI pump at rated flow, the test line valve to the suppression pool is opened, the pump suction valve from the suppression pool is opened (this valve is normally open) and the pumps are started using the remote-manual switches in the control room. Correct operation is determined by observing the instruments in the control room.

If an initiation signal occurs during the test, the LPCI system returns to the operating mode. The valves in the test bypass lines are closed automatically to assure that the LPCI pump discharge is correctly routed to the vessel.

6.3.4.2.5 ECCS Check Valves

ECCS check valves in the discharge side of the RHR/LPCI, HPCS and LPCS systems perform an isolation function that they protect low pressure systems from full reactor pressure. These ECCS check valves, as identified below, will be classified Category AC per the ASME OM Code with leak testing for this class of valve being performed to code specifications.

<u>System</u>	<u>Valve Identification</u>	<u>USAR Figure</u>
RHR/LPCI	F041A, B, C	<Figure 5.4-13>
HPCS	F005	<Figure 6.3-7>
LPCS	F006	<Figure 6.3-8>

The necessary test provisions (test connections, block valves, etc.) to leak test each valve have been incorporated in the design, as shown on <Figure 5.4-13>, <Figure 6.3-7>, and <Figure 6.3-8>.

Leak testing will be performed at each refueling after the valves have been exercised.

The required leak test schedule and leak detection criteria are provided in technical specifications.

6.3.5 INSTRUMENTATION REQUIREMENTS

Design details including redundancy and logic of the ECCS instrumentation are discussed in <Section 7.3>.

All instrumentation required for automatic and manual initiation of the HPCS, LPCS, LPCI, and ADS is discussed in <Chapter 7> and is designed to meet the requirements of IEEE Standard 279 and other applicable regulatory requirements. The HPCS, LPCS, LPCI, and ADS can be manually initiated from the control room.

The HPCS, LPCS and LPCI are automatically initiated on low reactor water level or high drywell pressure (see <Table 6.3-1> for specific initiation levels for each system). The ADS is automatically actuated by sensed variables for reactor vessel low water level plus indication that at least one LPCI or LPCS pump is operating. The HPCS, LPCS and LPCI automatically return from system flow test modes to the emergency core cooling mode of operation following receipt of an automatic initiation signal. The LPCS and LPCI system injection into the RPV begin when reactor pressure decreases to system discharge shutoff pressure.

HPCS injection begins as soon as the HPCS pump is up to speed and the injection valve is open since the HPCS is capable of injecting water into the RPV over a pressure range from 1,200 psid (differential pressure between RPV and pump suction source) to 0 psid.

6.3.6 REFERENCES FOR SECTION 6.3

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3. General Electric Company "General Electric Standard Application for Reactor Fuel," including the United States Supplement, NEDE-24011-P-A and NEDE-24011-P-A-US (latest approved revision).
4. "The GESTR-LOCA and SAFER Models for the Evaluation of the Loss-of-Coolant Accident, Volume III, SAFER/GESTR Application Methodology" NEDE-23785-PA, General Electric Company, Revision 1, October 1984.
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TABLE 6.3-1

PARAMETERS USED IN PERRY SAFER/GESTR-LOCA ANALYSIS

A. PLANT PARAMETERS

Plant Parameters	Nominal	<10 CFR 50, Appendix K>
Core Thermal Power (MWt) ⁽¹⁾	3758	3833
Corresponding Power (% of 3758 MWt)	100	102
Vessel Steam Output (Mlb/hr)	16.30	16.63
Vessel Steam Output (% rated)	100	102
Core Flow (Mlb/hr)	104.0	104.0
Core Flow (% rated)	100	100
Vessel Steam Dome Pressure (psia)	1040	1060
Maximum Recirculation Suction Line Break Area (ft ²)	2.73	2.73
Bottom Head Drain Line Break Area (ft ²)	0.0155	0.0155

B. ECCS PARAMETERS

1. Low Pressure Coolant Injection (LPCI) System

Variable	Units	Analysis Value
a. Maximum vessel pressure at which pumps can inject flow	psid (vessel to drywell)	225
b. Minimum rated flow (into shroud)		
• Vessel pressure at which below listed flow rates are quoted	psid (vessel to drywell)	20
• One (1) LPCI pump injecting into shroud	gpm	6500
• Two (2) LPCI pumps injecting into shroud	gpm	13000
• Three (3) LPCI pumps injecting into shroud	gpm	19500

TABLE 6.3-1 (Continued)

B. ECCS PARAMETERS (Continued)

	Variable	Units	Analysis Value
c.	Run-out flow at 0 psid (vessel to drywell)		
	• One (1) LPCI pump injecting into shroud	gpm	7000
	• Two (2) LPCI pumps injecting into shroud	gpm	14000
	• Three (3) LPCI pumps injecting into shroud	gpm	21000
d.	Initiating signals		
	• Low-low water level, or	inches above vessel "zero"	373.4
	• High drywell pressure	psig	2.0
e.	Vessel pressure at which injection valve may open	psia	450
f.	Time from initiating signal (Item 1.d) to pump at rated speed and capable of rated flow with emergency (diesel) power including sequencing delays	sec	27
g.	Time from initiating signal (Item 1.d) to power at injection valves	sec	10
h.	Injection valve stroke time-opening	sec	32
2.	Low Pressure Core Spray (LPCS) System		
	Variable	Units	Analysis Value
a.	Maximum vessel pressure at which pumps can inject flow	psid (vessel to drywell)	289

TABLE 6.3-1 (Continued)

2. Low Pressure Core Spray (LPCS) System (Continued)

	Variable	Units	Analysis Value
b.	Minimum rated flow at vessel-to-drywell pressure (into shroud)	gpm psid	6000 122
c.	Run-out flow at 0 psid (vessel to drywell)	gpm	6600
d.	Initiating signals	inches above vessel "zero"	373.4
	• Low-low water level, or		
	• High drywell pressure	psig	2.0
e.	Run-out flow at 0 psid (vessel to drywell)	gpm	6600
f.	Vessel pressure at which injection valve may open	psig	450
g.	Injection valve stroke time-opening	sec	32
h.	Time from initiating signal (Item 2.d) to power at injection valves	sec	10
i.	Time from initiating signal (Item 2.d) to pump at rated speed with emergency (diesel) power including sequencing delays	sec	27

3. High Pressure Core Spray (HPCS) System

	Variable	Units	Analysis Value
a.	Vessel pressure at which flow may commence	psid	1200
b.	Minimum rated flow and vessel pressure	gpm/psid (vessel to source of suction)	517/1200 1550/1147 6000/200

TABLE 6.3-1 (Continued)

3. High Pressure Core Spray (HPCS) System (Continued)

	Variable	Units	Analysis Value
c.	Run-out flow at 0 psid (vessel to source of suction)	gpm	6000
d.	Initiating signals	inches above vessel "zero"	485.4
	• Low-low water level, or		
	• High drywell pressure	psig	2.0
e.	Maximum allowable time delay from initiating signal until rated flow is available through the injection valve	seconds	29

4. Automatic Depressurization System (ADS)

	Variable	Units	Analysis Value
a.	Total number of valves with ADS function available		8
b.	Number of ADS valves assumed in the analysis		8
c.	Pressure at which below listed capacity is quoted	psig	1080
d.	Minimum flow capacity at pressure given in 4.c with all available ADS valves open	lb/hr	6.724×10^6
e.	Initiating signals	inches above vessel "zero"	373.4
	• Low-low water level, and		
	• ADS Timer Delay from initiating signal completed to the time valves are open	sec	120

TABLE 6.3-1 (Continued)

C. FUEL PARAMETERS

Fuel Parameter		Analysis Value GE11
PLHGR (kW/ft)	- <10 CFR 50, Appendix K>	14.4x1.02
	- Nominal	13.8
MAPLHGR (kW/ft)	- <10 CFR 50, Appendix K>	13.4x1.02
	- Nominal	12.9
Initial Operating MCPR	- <10 CFR 50, Appendix K>	1.16
	- Nominal	1.20
Axial Peaking Factor		1.4
Number of Fuel Rods per Bundle		74

NOTE:

- ⁽¹⁾ The nominal core thermal power (3758 MWt) corresponds to the current licensed rated power value. A conservative core thermal power of 3833 MWt which corresponds to 102% of the rated power was used for the <10 CFR 50, Appendix K> evaluation.

TABLE 6.3-2

OPERATIONAL SEQUENCE OF EMERGENCY CORE COOLING SYSTEMS FOR
DESIGN BASIS ACCIDENT^{(1) (2)}

<u>Time (sec)</u>	<u>Events</u>
0	Design basis loss-of-coolant accident assumed to start; normal auxiliary power assumed to be lost.
~0	Drywell high pressure and reactor low water level reached. All diesel generators signaled to start; scram; HPCS, LPCS, and LPCI signaled to start on high drywell pressure.
~4	Reactor low-low water level reached. HPCS receives second signal to start.
~6	Reactor low-low-low water level reached. Second signal to start LPCI and LPCS; auto-depressurization sequence begins; main steam isolation valve signaled to close.
<10	Division 1 and 2 diesel generators ready to load; begin energizing LPCI and LPCS pump motors.
~28	Pressure permissive for LPCI and LPCS injection valve reached.
~29	HPCS pump at rated speed and injection valve open sufficient to pass required flow.
~60	LPCI and LPCS pumps at rated speed, and LPCI and LPCS injection valves open sufficient to pass required flow.
~150	Core reflooded assuming worst single failure; heatup terminated.
>10 min	Operator shifts to containment cooling.

NOTES:

⁽¹⁾ For the purpose of all but the next to last entry on this table, all ECCS equipment is assumed to function as designed. Performance analysis calculations consider the effects of single equipment failures <Section 6.3.2.5> and <Section 6.3.3.3>. The recirculation suction line break DBA with limiting HPCS DG failure case using <10 CFR 50, Appendix K> assumptions is used.

⁽²⁾ Credit is taken in LOCA analyses for ECCS starts on high drywell pressure signal.

TABLE 6.3-3

PERRY SINGLE FAILURE EVALUATION

The table below shows the various combinations of Automatic Depressurization System (ADS), High Pressure Core Spray (HPCS) System, Low Pressure Coolant Injection (LPCI) System and Low Pressure Core Spray (LPCS) System which might be operable in an assumed design basis accident situation. In performing the ECCS performance analysis with SAFER/GESTR-LOCA, it is assumed that no postulated single active component will result in less than certain minimum combinations of systems remaining operable.

The following single, active failures will be considered in the ECCS performance evaluation for recirculation suction line break:

Assumed Failure ⁽¹⁾	Systems Remaining ⁽²⁾
LPCI Emergency Diesel Generator (D/G)	All ADS, HPCS, LPCS, 1 LPCI
LPCS Emergency DG	All ADS, HPCS, 2 LPCI
HPCS Emergency DG	All ADS, LPCS, 3 LPCI
One ADS Valve	All ADS minus one, HPCS, LPCS, 3 LPCI

NOTES:

- ⁽¹⁾ Other postulated failures are not specifically considered because they all result in at least as much ECCS capacity as one of the above assumed failures.
- ⁽²⁾ Systems remaining, as identified in this table, are applicable to all non-ECCS line breaks. For a LOCA from an ECCS line break, the systems remaining are those listed, less the ECCS system in which the break is assumed.

TABLE 6.3-4

SUMMARY OF RECRICULATION LINE BREAK RESULTS
FOR PERRY SAFER/GESTR ANALYSIS FOR CYCLE 8⁽¹⁾

Break Size & Recirc Break Location	Single Failure ⁽²⁾	GE11 ⁽³⁾ 1st/2nd peak PCT (°F)
Nominal:		
DBA, Suction	HPCS DG	907/809
1.0 ft ² , Suction	HPCS DG	891/590
0.1 ft ²	HPCS DG	650
<10 CFR 50, Appendix K>:		
DBA, Suction	HPCS DG	1130/1328
1.0 ft ²	HPCS DG	1144/1021
0.1 ft ²	HPCS DG	835

NOTES:

- ⁽¹⁾ Table does not include values associated with MEOD, feedwater temperature reductions, or single loop operations.
- ⁽²⁾ Only the results for the most limiting single failure are presented in this Table.
- ⁽³⁾ The calculated FCT results for the limiting fuel (GE11) would be bounding for GE10 and GE12 fuels.

TABLE 6.3-5

SAFER/GESTR-LOCA RESULTS FOR PERRY⁽⁴⁾

1.	Limiting Fuel Type (Cycle 8)	GE11 Fuel	
2.	Limiting Break	DBA Suction	Acceptance Criteria
3.	Limiting Failure	HPCS DG	-
4.	Peak Cladding Temperature (Licensing Basis)	< 1370 ⁽³⁾	≤ 2200 ⁽¹⁾
5.	Upper Bound PCT	< 1250 ⁽³⁾	≤ 1600°F ⁽²⁾
6.	Maximum Local Oxidation	< 0.2%	≤ 17% ⁽¹⁾
7.	Core-Wide Metal-Water Reaction	< 0.1%	≤ 1,0% ⁽¹⁾

NOTES:

⁽¹⁾ Acceptance criteria is from <10 CFR 50.46>.

⁽²⁾ Acceptance criteria is from NRC SER written for SAFER/GESTR-LOCA methodology.

⁽³⁾ SAFER/GESTR-LOCA Licensing Basis PCT for MEOD. The licensing result for the limiting fuel (GE11) would be bounding for GE10 and GE12 fuels.

⁽⁴⁾ <Appendix 15B>, Reload Safety Analysis provides the results of a performance analysis for the new (reload) fuel designs.

TABLE 6.3-6

KEY TO FIGURES

<ul style="list-style-type: none"> - Recirculation Break - Break Size - Location - Failure - Assumption 	Limiting Large Break	Intermediate Break	Typical Small Break
	DBA	1.0 ft ²	0.1 ft ²
	Suction	Suction	Suction
	HPCS DG	HPCS DG	HPCS DG
	Nominal	Nominal	Nominal
Water Level in Hot & Average Channels	<Figure 6.3-11 (1)>	<Figure 6.3-12 (1)>	<Figure 6.3-13 (1)>
Reactor Vessel Pressure	<Figure 6.3-11 (2)>	<Figure 6.3-12 (2)>	<Figure 6.3-13 (2)>
Peak Cladding Temperature ⁽¹⁾	<Figure 6.3-11 (6)>	<Figure 6.3-12 (6)>	<Figure 6.3-13 (6)>
Heat Transfer Coefficient ⁽¹⁾	<Figure 6.3-11 (7)>	<Figure 6.3-12 (7)>	<Figure 6.3-13 (7)>
ECCS Flow	<Figure 6.3-11 (5)>	<Figure 6.3-12 (5)>	<Figure 6.3-13 (5)>
Core Inlet Flow	<Figure 6.3-11 (10)>		
Minimum Critical Power Ratio	<Figure 6.3-11 (11)>		

TABLE 6.3-6 (Continued)

- Recirculation Break	Limiting Large Break	Intermediate Break	Typical Small Break
- Break Size	DBA	1.0 ft ²	0.1 ft ²
- Location	Suction	Suction	Suction
- Failure	HPCS DG	HPCS DG	HPCS DG
- Assumption	<10 CFR 50, Appendix K>	<10 CFR 50, Appendix K>	<10 CFR 50, Appendix K>
Water Level in Hot & Average Channels	<Figure 6.3-14 (1)>	<Figure 6.3-15 (1)>	<Figure 6.3-16 (1)>
Reactor Vessel Pressure	<Figure 6.3-14 (2)>	<Figure 6.3-15 (2)>	<Figure 6.3-16 (2)>
Peak Cladding Temperature ⁽¹⁾	<Figure 6.3-14 (6)>	<Figure 6.3-15 (6)>	<Figure 6.3-16 (6)>
Heat Transfer Coefficient ⁽¹⁾	<Figure 6.3-14 (7)>	<Figure 6.3-15 (7)>	<Figure 6.3-16 (7)>
ECCS Flow	<Figure 6.3-14 (5)>	<Figure 6.3-15 (5)>	<Figure 6.3-16 (5)>
Core Inlet Flow	<Figure 6.3-14 (10)>		
Minimum Critical Power Ratio	<Figure 6.3-14 (11)>		

NOTE:

⁽¹⁾ Plots are for GE11 fuel only.

TABLE 6.3-7

ECCS DESIGN PARAMETERS

<u>System</u>	<u>Parameter</u>	<u>Design Value</u>	<u>Basis</u>
LPCS	Pool suction line design pressure	100 psig	Nominal value, suction from RPV (shutdown test)
RHR	Pool suction line design pressure	100 psig	Nominal value, suction from RPV (shutdown test)
HPCS	Design pressure for suction from condensate storage	100 psig	Nominal value, suction from condensate tank
RHR	Shutdown suction line pressure	200 psig	Max vessel cut in pressure + max. vessel water level above pump
LPCS	Pump discharge line pressure	600 psig ⁽¹⁾	Shutoff head + max. suction pressure
RHR (LPCI)	Pump discharge line pressure	500 psig ⁽¹⁾	Shutoff head + max. suction pressure
HPCS	Pump discharge line pressure	1,575 psig	Shutoff head + max. suction pressure
LPCS	Pump suction & discharge temp.	185°F	Maximum expected temperature of pumped fluid (i.e., maximum containment design temperature) <Regulatory Guide 1.1>
HPCS	Pump suction & discharge temp.	185°F	Maximum expected temperature of pumped fluid (i.e., maximum containment design temperature) <Regulatory Guide 1.1>

TABLE 6.3-7 (Continued)

<u>System</u>	<u>Parameter</u>	<u>Design Value</u>	<u>Basis</u>
RHR (LPCI)	Pool suction	185°F	Maximum expected temperature of pumped fluid (i.e., maximum containment design temperature) <Regulatory Guide 1.1>
RHR	Shutdown line temperature	358°F	Max shutdown suction temperature (saturation @ 135 psig)
LPCS	Rated flow	6,000 gpm @122 psid (over drywell)	<Table 6.3-1>
RHR (LPCI)	Rated flow	6,500 gpm/ loop - 3 loops @20 psid (over drywell)	<Table 6.3-1>
HPCS	Rated Flow	6,000 gpm @200 psid	<Table 6.3-1> (value selected to provide adequate core cooling, all design basis events)
LPCS	RPV pressure at beginning flow	289 psid (over drywell)	<Table 6.3-1> (value selected to provide adequate core cooling all design basis events)
RHR (LPCI)	RPV pressure at beginning flow	225 psid (over drywell)	<Table 6.3-1> (value selected to provide adequate core cooling for all design basis events)

TABLE 6.3-7 (Continued)

<u>System</u>	<u>Parameter</u>	<u>Design Value</u>	<u>Basis</u>
HPCS	RPV pressure	1,200 psid	<Table 6.3-1> (value selected to provide adequate core cooling for all design basis events)
LPCS	Time to rated speed	27 sec	<Table 6.3-1> (value selected to provide adequate core cooling for all design basis events)
RHR (LPCI) pump	Time to rated speed	27 sec	<Table 6.3-1> (value selected to provide adequate core cooling for all design basis events)
HPCS Pump	Time to rated speed	27 sec	<Table 6.3-1> (value selected to provide adequate core cooling for all design basis events)
LPCS	Injection valve open sufficient to pass rated flow	32 sec ⁽²⁾ (after pressure permissive)	<Table 6.3-1> (value selected to provide adequate core cooling for all design basis events)
RHR (LPCI)	Injection valve open sufficient to pass rated flow	32 sec ⁽²⁾ (after pressure permissive)	<Table 6.3-1> (value selected to provide adequate core cooling for all design basis events)
HPCS	Injection valve open sufficient to pass rated flow	29 sec ⁽³⁾	<Table 6.3-1> (value selected to provide adequate core cooling for all design basis events)

TABLE 6.3-7 (Continued)

NOTES:

- (1) Discharge piping design rating is based on ASME code Section III criteria; applicable pressure/temperature rating is 720 psig (@100°F) to 625 psig (@500°F).
- (2) Calculated valve stroke time is 32 seconds. Required flow will be established prior to the valve reaching full open, providing margin for considerations such as diesel generator operation at minimum allowable frequency and voltage.
- (3) Calculated injection valve stroke time is 29 seconds including the time to start the emergency diesel generator. Required flow will be established prior to reaching full open, providing margin for considerations such as diesel generator operation at minimum allowable frequency and voltage. Design basis analysis assumes HPCS injection begins at 29 seconds.

TABLE 6.3-8

MANUAL VALVES IN HPCS SYSTEM⁽¹⁾

<u>Valve No.</u>	<u>Type</u>	<u>Location</u>	<u>Service</u>	<u>Function</u>	<u>Methods for Minimizing Positioning Error⁽²⁾</u>
F036	12" Gate	Drywell	Main Process Line	Main Process Line Block Valve	NO; position indications light (control room mounted)
F026	6" Gate	Aux. Bldg.	Flushing	Backflush Line By- passing Check Valve (F024)	NC; closed
F031	6" Gate	Aux. Bldg.	Flushing	Flushing Water Supply line to HPCS pump Discharge	NC; backed by check valve F003
F034	2" Globe	Aux. Bldg.	Jockey Pump Lines	Jockey Pump suction isolation valve	NO; during HPCS operation, position is not critical; closed
F033	1" Globe	Aux. Bldg.	Jockey Pump Lines	Jockey Pump minimum flow line to HPCS pump suction	NO; during HPCS operation, position is not critical
F006	1-1/2" Stop Check	Aux. Bldg.	Jockey Pump Lines	HPCS Jockey Pump Discharge isolation valve	Open - Throttled; HPCS operation would not be affected by this valves position (closed)
F019	6" Gate	Aux. Bldg.	Flushing and Servicing	HPCS Pump suction line Drain to Radwaste System	NC

TABLE 6.3-8 (Continued)

NOTES:

⁽¹⁾ Piping low point drains, high point vents and test connections are provided with dual isolation.

⁽²⁾ NO = Normally open

NC = Normally closed

Backed by . . ., = Double valve arrangement precluding impact on system operation without two position errors and/or a non-manual valve failure.

Closed = Indicates valve is in line that forms closed loop with piping that, without a double positioning error, would have no effect on system functioning.

TABLE 6.3-9

MANUAL VALVES IN LPCS SYSTEM⁽¹⁾

<u>Valve No.</u>	<u>Type</u>	<u>Location</u>	<u>Service</u>	<u>Function</u>	<u>Methods for Minimizing Positioning Error⁽²⁾</u>
F007	12" Gate	Drywell	Main Process Line	Main Process line block valve	NO; position indicating light (control room mounted)
F025	6" Gate	Aux. Bldg.	Flushing	Flushing water supply to LPCS pump discharge piping to vessel	NC; backed by a blind flange
F004	6" Gate	Aux. Bldg.	Flushing	Back flush line bypassing check valve F003	NC; Closed
F008	6" Gate	Aux. Bldg.	Flushing and Servicing	LPCS pump suction line drain to radwaste system	NC; Closed
F032	2" Globe	Aux. Bldg.	Jockey Pump Lines	LPCS Jockey pump suction isolation valve	NO; LPCS operation would not be affected by this valves position (closed)
F035	3/4" Globe	Aux. Bldg.	Jockey Pump Lines	Jockey Pump minimum flow to LPCS suction lines	NO; LPCS operation would not be affected by this valves position (closed)
F034	1-1/2" Globe Stop Check	Aux. Bldg.	Jockey Pump Lines	LPCS Jockey pump Discharge isolation valve	Open-Throttled; LPCS operation would not be affected by this valves position (closed)

TABLE 6.3-9 (Continued)

NOTES:

⁽¹⁾ Piping low point drains, high point vents and test connections are provided with dual isolation.

⁽²⁾ NO = Normally open

NC = Normally closed

Backed by . . ., = Double valve arrangement precluding impact on system operation without two position errors and/or a non-manual valve failure.

Closed = Indicates valve is in line that forms closed loop with piping that, without a double positioning error, would have no effect on system functioning.

TABLE 6.3-10

MANUAL VALVES IN LPCI (RHR) SYSTEM⁽¹⁾

<u>Valve No.</u>	<u>Type</u>	<u>Location</u>	<u>Service</u>	<u>Function</u>	<u>Methods for Minimizing Positioning Error⁽²⁾</u>
F029 A,B,C	18" Gate	Aux. Bldg.	Main Process Line	Block Valves on RHR Pumps discharge lines	NO; See Note ⁽³⁾
F039 A,B,C	12" Gate	Drywell	Main Process Line	Process Line Block Valve for leak testing check valves A,B,C	NO; position indicating light (control room mounted)
F066 A,B	10" Gate	Aux. Bldg.	Fuel Pool Cooling	From spent fuel pool to RHR pumps A, B suction	NC; position monitoring switches
F067	18" Gate	Aux. Bldg.	Main Process Line	Shutdown suction to Pump C	NC
F099 A,B	10" Gate	Aux. Bldg.	Fuel Pool Cooling	Return to spent fuel after heat exchanger pass	NC
F552 A	10" Stop Check	Aux. Bldg.	Flushing	RHR Line flushing	NO
F552B	10" Stop Check	Aux. Bldg.	Flushing	RHR Line flushing	Locked Closed
F018 A,B,C	6" Gate	Aux. Bldg.	Minimum flow and standby	Minimum flow line to suppression pool from RHR pumps A,B,C discharges	Locked in throttled position

TABLE 6.3-10 (Continued)

<u>Valve No.</u>	<u>Type</u>	<u>Location</u>	<u>Service</u>	<u>Function</u>	<u>Methods for Minimizing Positioning Error⁽²⁾</u>
F047A,B	18" Gate	Aux. Bldg.	Main Process Line	RHR Heat exchanger isol. valve	NO; position indicating light (control room mounted) See Note ⁽⁴⁾
F071 A,B	8" Gate	Aux. Bldg.	Flushing	RHR A,B Pump suction line drains to radwaste system	NC
F072 A,B	8" Gate	Aux. Bldg.	Flushing	RHR A,B Pump discharge Line drains to Radwaste system	NC
F511 A,B	8" Gate	Aux. Bldg.	Flushing	RHR A,B,C pump drains to radwaste system	LO; backed by NC valves F071 A,B,C and F072 A,B,C
F010	20" Gate	Drywell	Main Process Line	Decay Heat Removal	NO; position indicating light (control room mounted)
F071C	8" Gate	Aux. Bldg.	Flushing	RHR C pump suction line drains to radwaste system	NC; backed by NC valve F072C
F072C	8" Gate	Aux. Bldg.	Flushing	RHR C pump discharge line drains to Radwaste system	NC; backed by NC valves F071C and F511B
F082	2" Globe	Aux. Bldg.	Jockey Pump Lines	RHR loop B,C Jockey pump suction isolation	NO; during LPCI operation, position is not critical; closed
F085 A,B,C	1-1/2" Globe Stop Check	Aux. Bldg.	Jockey Pump Lines	RHR loop A,B,C Jockey pump discharge isolation	NO

TABLE 6.3-10 (Continued)

<u>Valve No.</u>	<u>Type</u>	<u>Location</u>	<u>Service</u>	<u>Function</u>	<u>Methods for Minimizing Positioning Error⁽²⁾</u>
F502 A,B, C,D	6" Gate	Aux. Bldg.	Flushing	RHR Heat exchanger line Flushing	NC; backed by F503 A,B,C,D.
F503 A,B, C,D	6" Gate	Aux. Bldg.	Flushing	RHR Heat exchanger line Flushing	NC; backed by F502, A,B,C,D.
F504	8" Gate	Aux. Bldg.	Flushing	RHR Loop B pump discharge flushing line	NC; backed by F104
F102	1-1/2" Globe	Aux. Bldg.	Main Process Line	RCIC Turb. Exh. Vac. RLF to RHR	Locked Closed; backed by F078
F0621	12" Gate	Aux. Bldg.	Main Process Line	Provide isolation of RHR Loop A from ADHR system for maintenance	NO; Lock open during Modes 1, 2, and 3
F0625A	4" Ball	Aux. Bldg.	Flushing	RHR A Chemical Decon.	NC; backed by blind flange

NOTES:

⁽¹⁾ Piping low point drains, high point vents and test connections are provided with dual isolation.

⁽²⁾ NO = Normally open

NC = Normally closed

Backed by . . ., = Double valve arrangement precluding impact on system operation without two position errors and/or a non-manual valve failure.

Closed = Indicates valve is in line that forms closed loop with piping that, without a double positioning error, would have no effect on system functioning.

⁽³⁾ The incorrect positioning of F029 A,B,C would also be detected during normal plant operations when LPCI flow capacity verification tests are periodically conducted.

⁽⁴⁾ Electrically operated valve with no automatic open function administratively controlled to ensure it is open.

6.4 HABITABILITY SYSTEMS

Control room systems are designed in accordance with the design bases described in <Section 6.4.1> so that habitability of the control room can be maintained under normal and accident conditions. The general guidance contained in General Design Criterion 19 of <10 CFR 50, Appendix A>, and the specific guidance contained in <Regulatory Guide 1.78> is reflected throughout this section.

6.4.1 DESIGN BASES

The design bases for control room habitability systems are as follows:

a. Control Room Envelope

The control room envelope includes most areas located on Elevation 654'-6" of the control complex, with the exception of the elevator area, the stairwell, and the mechanical and electrical chase areas. Housed within this control room envelope are the monitoring equipment, instrumentation and control panels required for safe operation and shutdown of the plant. The control room envelope is provided with fire protection equipment, adequate lighting, communications equipment, kitchen, sanitary, administrative and storage facilities, and spaces necessary to perform the normal plant operations required to maintain the plant in a safe condition following an accident. The control room envelope ambient atmosphere is normally maintained at the conditions presented in <Figure 3.11-17>.

b. Period of Habitability

The control room envelope is equipped to sustain seven people for a period of seven days following an accident.

c. Capacity

The normal occupancy level of the control room is six people following an accident.

d. Food, Water, Medical Supplies, and Sanitary Facilities

First aid equipment, food and water are provided to sustain seven people for seven days following an accident. Chemical toilet facilities are provided for use in the event that normal sanitary facilities become inoperative.

e. Radiation Protection

Radiation protection, as required by <10 CFR 50.67>, is provided by shield walls on the four exposures, shield slabs at floor and ceiling, radiation monitoring equipment, and emergency filtering systems. The control room atmosphere is monitored for radiation. When required, the control room atmosphere can be recirculated through the emergency filter system to remove contaminants. This filter system consists of roughing, high efficiency particulate air (HEPA) and charcoal filters. Assumptions and analyses regarding sources and amounts of radioactivity which may surround or leak into the control room and related shielding requirements are discussed in <Chapter 12> and <Chapter 15>. The radiation monitoring system is discussed in <Section 12.3.4>.

f. Noxious Gas Protection

Smoke detectors located in the control room air supply duct and in the emergency filter system discharge duct actuate alarms to indicate the presence of smoke in these locations. Additionally,

the control room can be purged with outside air if required. Conformance with the guidelines given in <Regulatory Guide 1.78> is discussed in <Section 2.2.3>.

g. Toxic Gas Protection

Based on the data in <Section 2.2.2>, the control room HVAC system is provided with oxygen monitors which alarm upon detecting a loss of oxygen in the air due to carbon dioxide buildup. No other toxic gas detectors or interlocks are considered necessary for ensuring control room habitability <Section 2.2.3.1.2.1>.

h. Respiratory Protection

Breathing protection apparatus are provided for control room occupants. Breathable air is provided by 10 compressed air cylinders located in the service building at Elevation 620'-6". This air system supplies six breathable air stations with five connections on each station located in the control room. Self-contained breathing apparatus (SCBA) are provided for control room occupants. These SCBA can be supplied with air from the six breathable air stations in the control room or their own one hour compressed air bottle. The capacity of the 10 compressed air cylinders is sufficient to supply seven men for approximately six hours, plus each man has a one hour supply in his SCBA bottle. The SCBA bottles and the large air cylinders can be filled by a breathable air compressor located onsite or via offsite services.

i. Habitability System Operation during Emergencies

Operation of the habitability system during emergencies is discussed in <Section 6.4.3>.

j. Emergency Monitors and Control Equipment

Emergency monitors and control equipment are discussed in the PNPP Emergency Plan.

6.4.2 SYSTEM DESIGN

6.4.2.1 Definition of Control Room Envelope

The control room envelope comprises those areas to which the control room operator could require access during an emergency. It includes the following:

- a. Control room main control board and monitoring panel area - continuous occupancy required.
- b. Chart and storage room - infrequent access required.
- c. Conference Room - infrequent access required.
- d. Kitchen facility - infrequent access required.
- e. Toilet room - infrequent access required.
- f. Corridors/Hallway - infrequent access required.

The envelope includes those components that provide a boundary between the environment inside the control room and surrounding atmosphere. The components include the control room ventilation system, structural penetrations (electrical and mechanical), access doors (plenum, duct, personnel), door seals, as well as the walls, ceiling, and floor of the 654' elevation of the control complex and duct chase. <Figure 6.4-1> and <Figure 6.4-2>

The equipment to which the control room operator could require access during an emergency is listed in <Table 6.4-1>.

6.4.2.2 Ventilation System Design

The heating, ventilating and air conditioning (HVAC) and the emergency filtration systems for the control room are shown schematically in

<Figure 6.4-1 (1)>. This figure illustrates components, ducts, dampers, instrumentation, and normal and emergency air flow rates, and consists of the following:

- a. Control room HVAC system
- b. Control room emergency recirculation system

The components are not subject to the effects of catastrophic weather, internal or external missiles. Pipe whip, jet impingement and flooding effects are discussed in <Section 3.6> and are insufficient to cause a loss of system redundancy.

<Figure 6.4-2> presents a layout drawing of the control room, indicating doors, corridors, stairwells, and shielded walls.

The location of potential radioactive gas releases and their effect upon control room operation, and the monitoring instrumentation and controls located therein are discussed in <Chapter 11>.

6.4.2.2.1 Control Room HVAC System

The function of this system is to provide cooling, heating, ventilation, and, when required, smoke removal for the control room equipment areas and office during normal plant operation, plant shutdown, loss of offsite power, and during periods of emergency (loss-of-coolant accident or high radiation conditions).

This system operates continuously to supply 45,000 cfm of conditioned air, including 6,000 cfm outside air for ventilation, to the control room to dissipate the internal heat load generated and maintain the control room ambient air at the conditions presented in <Figure 3.11-17>.

During normal plant operation, the system supply (M25-C001A or B) and return fans (M25-C002A or B) run continuously, the outside air intake dampers (M25-F010A or B, M25-F020B or A), and the return damper (M25-F110A or B) are open, and the exhaust air damper (M25-F130A or B) is closed. Either the "A" set or the "B" set of supply and return components is in operation with the idle redundant equipment as backup. The emergency recirculation system is idle and closed off from the HVAC system by its closed discharge (M26-F040A or B) damper; see <Section 6.4.2.2.2> for further details on the operation of this system. Normally, 4,900 cfm is assumed to exfiltrate from the control room through normal openings, thereby ensuring a positive pressure inside the room.

Electric heating coils (with SCR controllers) in the branch supply ducts to the various zones in the control room are provided to control the final ambient air temperature in each zone. An electronic thermostat in each zone is used to provide a signal to the SCR controller which will control the heating coil, depending upon final room temperature.

Humidification is also provided by the control and computer rooms humidification system. An electronic humidity controller, with local indication, is located in the general area of the control room to modulate the electric motor-operated valve on the humidifier (M29-B002A, B). For further details in the operation of the humidification system see <Section 9.4.12>.

In the event of a LOOP condition the radiation monitors go to the fail safe condition, thereby ensuring that the running train shifts to the emergency recirculation mode and the standby train starts in the emergency recirculation mode so that both sets "A" and "B" are running. Power is supplied by the standby diesel generators. The operator may shut down one of the trains after it has been established that both are operating satisfactorily.

The instrument air system which supplies control air to the pneumatic dampers is not connected to the emergency bus. In the event of loss of offsite power (without LOCA) coupled with subsequent loss of control air, the pneumatic dampers assume their failed position as shown in <Figure 6.4-1 (1)>. This places the system in the recirculation mode with back flow prevented by check damper (M25-F551A, B).

In the event of high smoke condition, except during a LOOP/LOCA condition, the smoke can be purged by manually setting the mode selector switch in the "smoke clear" mode.

The main components of this system are located in the control complex at Elevation 679'-6" and consist of two redundant supply plenums (M25-B001A and B001B), two redundant supply fans (M25-C001A, C001B) and two redundant return fans (M25-C002A, C002B), each rated at 100 percent of the total required capacity.

Each supply plenum includes roughing filters and chilled water coils. Each outside air intake duct is provided with redundant dampers (M25-F010A or F010B and M25-F020A or F020B) in series to reduce the outside air inleakage when the system is in the emergency recirculation mode. A check damper (M25-F510A or F510B) is provided in the discharge duct of each supply fan. In addition, manually operated balancing dampers for balancing, and fire dampers for fire protection, are provided.

Each return unit consists of a centrifugal fan (M25-C002A or C002B), an exhaust air isolation damper (M25-F130A or F130B), a return air isolation damper (M25-F110A or F110B), manually operated balancing dampers, and fire dampers. The exhaust isolation damper (M25-F130A or F130B) is normally closed and the return isolation damper (M25-F110A or F110B) is normally open.

The fans, filter elements, coils, and dampers are of standard industrial design manufactured in accordance with the Quality Assurance (QA) requirements of Safety Class 3, Seismic Category I items. The filter racks and plenums are specially designed to satisfy system space requirements and also meet the above QA requirements. However, the electric duct heating coils and the humidifiers are nonsafety-related items. Design information for the major components in this system is listed in <Table 6.4-2>.

6.4.2.2.2 Control Room Emergency Recirculation System

The Perry Nuclear Power Plant utilizes the Alternate Source Term analyses approved by the Nuclear Regulatory Commission through License Amendments 122, 166, and 180. The control room emergency recirculation system is tested to ensure control room habitability following a design basis loss of coolant accident. The electric heaters described below are not required for the control room emergency recirculation system to perform its specified safety function based on the Alternate Source Term analyses. This system provides the necessary supplementary particulate and halogen filtration of the air supplied to the control room areas and offices during emergency periods and other abnormal conditions for personnel protection.

This system is automatically activated by an emergency signal (such as LOCA), or high radiation, or by manually setting the mode selector switch to the emergency recirculation mode. While the system is automatically activated, no credit is taken during the first 30 minutes following a design basis LOCA. As a result of a LOOP condition the system will automatically activate in the emergency recirculation mode when power is restored to the emergency busses. The operator may shut down one of the trains after it has been established that both are operating satisfactorily. In addition, the receipt of the emergency signal causes dampers in the control room HVAC system to be automatically positioned to the emergency recirculation mode of operation (see Note 4 of <Figure 6.4-1 (2)>, for the damper

positions).The vortex damper operator (M25-F260A, B) of the supply fan (M25-C001A, B) is then automatically de-energized allowing the operator spring to partially close the variable inlet vanes to reduce the supply air flow to 30,000 cfm. This flow reduction is required so that the supply fan and recirculation fan flow rates are compatible. Return fan (M25-C002A or B) is also deactivated and the electric heating coil in the charcoal filter train is automatically energized upon receipt of an emergency signal. The electric heaters are not credited in design basis accident analyses for humidity control.

The emergency recirculation system causes the supply air to be filtered through the charcoal filter train (M26-D001A, B) before being distributed to the control room. This system is idle during normal plant operation. During periods of loss of offsite power, emergency power will be supplied by the standby diesel generators.

The degree to which the recommendations of <Regulatory Guide 1.52> are followed is given in <Table 6.5-1>.

The main components of this system are located in the control complex at Elevation 679'-6" and consist of two 100 percent capacity filter trains. Each filter train includes the following sequential components: demisters, roughing filters, electric heating coil, HEPA prefilters, charcoal filters, HEPA after filters, centrifugal fan, isolation damper, and check damper.

The fans, filter elements and dampers are of standard industrial design, manufactured in accordance with Quality Assurance (QA) requirements of Safety Class 3, Seismic Category I items. The filter racks, frames and housing are specially designed to satisfy the system space requirements and also meet the above QA requirements.

Design information for the major components in this system is listed in <Table 6.4-3>.

6.4.2.3 Leak Tightness

The control room system is designed so that, when operating in a normal mode (admitting outside air), the system automatically maintains a positive differential pressure between the control room and the outside and thus, between the control room and adjacent spaces. During an emergency, when the system operates in the recirculation mode (no designed admittance of outside air), no attempt is made to pressurize the control room. In the recirculation mode, the potential paths of air infiltration to the control room include (1) outside air dampers and relief air dampers, (2) openings around supply and return ducts in the control room walls and in duct chase floors, (3) openings for electrical conduit and cables in the control room and chase walls and floors, (4) doors, and (5) piping.

A review of these paths, summarized as follows, indicates that infiltration through these paths during the recirculation mode is minimal:

- a. The outside air is sealed from the control room by two 40 inch wide by 48 inch high dampers in series, both gasketed and arranged to close in the recirculation mode and to fail closed upon loss of control air or power. While either damper can be used for isolation, the outermost isolation damper from the control room provides the boundary of the envelope. The maximum leakage is 29 cfm through a single 42-inch by 48-inch outside air damper in the recirculation mode, at an estimated pressure differential across the damper of 0.25 inches of water. Maximum leakage through two closed dampers in series is approximately 20 cfm. This estimate conservatively assumes no pressurization in the control room. If the control room were pressurized, the pressure differential across the dampers would decrease. The duct pressure loss calculations for the control room system indicate that, during operation in the recirculation mode and with control room ambient.

pressure assumed to be 0 psig, the pressure at the inside of the outside air inlet damper would be 0.25 inches of water.

- b. Exhaust isolation is provided by a normally closed bubble tight damper and a normally open damper that closes on recirculation. During recirculation, a check damper also provides isolation in the exhaust duct.

- c. Openings around supply and return ducts in the walls and floors of the control room and chases are sealed with expanded silicone foam with a fire resistance rating of three hours. These openings are air tight.
- d. Openings for electrical conduit and cables in the walls and floors of the control room and chases are sealed in the same manner as the openings noted in Item c, above.
- e. Doors from the control room to the chase area and stairwells are three hour fire rated doors with closures. The maximum anticipated total air gap around each door is 0.27 ft².
- f. Piping to plumbing fixtures, drains and potable water leaving the control room are sealed in the manner noted in Item c, above.

None of the control room doors lead directly to the outside. All doors lead to closed chase spaces, closed stairwells or closed corridor space. Thus, neither outside wind conditions nor other ventilation system cause infiltration or leakage into the control room.

Various plant activities may require a temporary degradation of the control room boundary and consequently, increase the inleakage. This is discussed in more detail in <Section 6.4.4.1>.

6.4.2.4 Interaction With Other Zones and Pressure Containing Equipment

The control room ventilation system does not communicate with other building areas where potential for radioactivity exists.

There are no pressure containing pipes or equipment containing hazardous chemicals in the control room chases.

The normal operating mode of the control room ventilation system maintains a positive pressure differential between the control room and adjacent spaces. The likelihood of infiltration is therefore further reduced. A radiation monitor sensing control room atmospheric radioactivity immediately causes the ventilation system to shift into the recirculation mode upon detection of high gaseous activity. Thus, the potential for entry of outside airborne material into the control room is reduced. The control room ventilation system can also be used to purge the control room with outside air at a constant flow rate of 30,000 cfm.

6.4.2.5 Shielding Design

The control room shielding design is discussed in <Section 12.1>.

6.4.3 SYSTEM OPERATIONAL PROCEDURES

The control room is served by redundant normal and redundant emergency HVAC systems. The emergency systems provide for operation in the recirculation mode, the smoke clearing mode during loss of offsite power and for activation of the carbon dioxide fire protection system and charcoal water spray system.

The normal control room HVAC system operates continuously to provide heating, ventilating and cooling to various equipment and personnel areas in the control room.

The control room emergency recirculation system is idle except when activated by an emergency signal or when the mode selector switch is manually set to "EMER. RECIRC." During periods of emergency, such as a

LOCA or high radiation, the dampers automatically are positioned for the recirculation mode. Also, following this type of emergency the return fan (M25-C002A or B) stops, both emergency recirculation fans (M26-C001A & B) and the control room HVAC supply fans (M25-C001A & B) start simultaneously, and subsequently, one fan train may be manually stopped.

The control room has a smoke removal mode of operation used to purge the control room with outside air when smoke is detected. This is accomplished by remote-manually positioning the mode selector switch located on panel H13-P904 to "SMOKE-CLEAR." Operation of this switch positions the related dampers to the purge mode and automatically positions the variable inlet vanes of the supply and return fans at 30,000 cfm flow.

The position of system dampers during the normal, emergency recirculation and smoke clearing mode is indicated by Note 4 of <Figure 6.4-1 (2)>.

If loss of offsite power occurs (without LOCA), emergency power is provided by the standby diesel generators and the system operates with dampers in the emergency recirculation mode. At this time, both supply fans (M25-C001A & B) and both emergency recirculation fans (M26-C001A & B) start automatically.

If excess temperature occurs in emergency filter plenums (as indicated by a high-high temperature alarm), the fire protection water spray system in the filter plenum can be activated manually. This action also energizes the solenoid valve (M26-F081A or B) and opens the drain valve (M26-F080A or B). Deactivating the spray deluge valve de-energizes the solenoid valve (M26-F081A or B); however, the drain valve remains open until it is manually closed by a manual override lever.

6.4.4 DESIGN EVALUATION

Each of the operating systems which ensures control room habitability is discussed in detail in other sections. These systems, and the section in which they are discussed, are as follows:

- a. Control room ventilation system, <Section 6.4.2>.
- b. Fire protection system, <Section 9.5.1>.
- c. Communications system, <Section 9.5.2>.
- d. Lighting system, <Section 9.5.3>.
- e. Offsite power system, <Section 8.2>.
- f. Onsite power system, <Section 8.3>.
- g. Radiation monitoring system, <Section 12.3.4>.

A summary evaluation of control room habitability based on selected considerations is presented in <Section 6.4.4.1>, <Section 6.4.4.2>, <Section 6.4.4.3>, <Section 6.4.4.4>, <Section 6.4.4.5>, and <Section 6.4.4.6>.

6.4.4.1 Radiological Protection

The evaluation of radiological exposures to control room operators following a design basis loss-of-coolant accident is presented in <Section 15.6.5>.

The analysis in <Section 15.6.5> assumes a constant unfiltered control room inleakage value of 1375 cfm after 30 minutes and for the duration of the LOCA accident.

Normally, the control room boundary inleakage is maintained at a value consistent with pre-operational testing such that the actual inleakage is substantially less than 1375 cfm.

Throughout the life of the plant, various plant activities may need to be performed which temporarily degrade the control room boundary such that the unfiltered inleakage significantly exceeds 1375 cfm. The design basis radiological calculations for a postulated LOCA assume an unfiltered inleakage of 6600 cfm for the first 30 minutes, which shows that it is acceptable to delay the restoration of the control room boundary, provided that once it is restored, the actual unfiltered inleakage would be reduced to at or below 1375 cfm for the remainder of the accident. This allows for temporary degradations of the boundary to occur without impacting overall accident dose to the control room operators. Administrative controls are utilized during planned degradations to ensure the boundary can be rapidly restored to within the bounding parameters of the analysis.

6.4.4.2 Toxic Gas Protection

No toxic materials which could interfere with control room occupancy are stored in the plant. Sodium hypo-chlorite, rather than chlorine, is used as a biocide. No chlorine is stored on site. The potential effects of offsite and onsite hazardous materials are discussed in <Section 2.2.2> and <Section 2.2.3>. Protection against offsite toxic gases are detailed in <Section 6.4.1.g>.

6.4.4.3 Control Room Emergency Recirculation System

The general arrangement and control of the control room emergency recirculation system is as described in <Section 6.4.2.2.2>. Detailed information concerning the emergency filter is presented in <Section 6.5.1>. The equipment is shielded, housed in a Seismic Category I structure, separated, redundant, and powered from the

Class 1E electrical system. It is equipped with filters designed in accordance with the requirements of <Regulatory Guide 1.52> <Section 6.5.1>.

No single failure results in loss of system function.

In case of loss of offsite power, redundant emergency power will be provided by the standby diesel generators and the fans will be automatically started.

The electric heating coils for the emergency recirculation system are connected to the standby diesel generators to allow operation of the coils during loss of offsite power. The power supply to each filter train heating coil is from a separate safety division bus. The electric heating coils are not credited in design basis accident analysis for humidity control.

Single component failure analysis of the control room emergency filter system is discussed in <Table 6.4-4>.

Implementation of scheduled field testing, inspection and maintenance programs ensure that each filtering system performs in accordance with design requirements.

6.4.4.4 Control of the Control Room Thermal Environment

The control room air handling system operates during normal and emergency periods to maintain an environment suitable for personnel and equipment. The conditions maintained and general system description are presented in <Section 6.4.2.2.1>. The system satisfies the single failure criteria by providing redundant, separated and shielded air handling and cooling water systems (see <Section 9.4.9> for details on the cooling water system). Thus, the integrity and operability of this system during normal and emergency periods is ensured.

Each portion of the system is sized for its maximum anticipated internal cooling load, considering outdoor summer design conditions expected for no more than 2.5 percent of the year. Additional margin is provided in equipment selection to allow for degradation of coil surfaces, films and changes in system air flow during operation. Scheduled maintenance procedures assure that each system performs in accordance with design requirements.

6.4.4.5 Fire Protection

Protection against fire hazards is provided by fire hose cabinets, CO₂ extinguishers and dry chemicals in the control room and adjacent areas. These fire suppression devices are provided in accordance with requirements of the National Fire Codes of the National Fire Protection Association (NFPA), the requirements of American Nuclear Insurers (ANI), the applicable regulations of the State of Ohio, and the requirements of the Occupational Safety and Health Administration (OSHA). The state of readiness of the fire protection system is maintained by enforcing a program of listing, inspection and maintenance. Further evaluation of the fire protection system is presented in <Section 9.5.1>.

6.4.4.6 Food, Water and Sanitation

A seven day supply of food is provided in the control room for emergency use. If the emergency requires confinement for periods longer than seven days, additional food will be brought onsite in protected containers. Site accessibility will be determined by the Radiation Protection Manager.

Potable water is normally available from the potable water system described in <Section 9.2.4>. Should this system become unavailable during an emergency period, stored, bottled water is available in an area immediately adjacent to the control room. Additional bottled water could be brought into the control room, if required.

Normal sanitation facilities are available as described in <Section 9.2.4>. Chemical toilets are also available in these areas for use in the event that the normal facilities become inoperative following an emergency.

6.4.5 TESTING AND INSPECTION

The equipment which maintains control room habitability includes the emergency filter system, the control room air handling system and the chilled water system.

Components of these systems are subjected to documented preoperational test procedures to verify proper wiring, system integrity and leak tightness, proper function of system components and control devices under normal and emergency conditions, and to establish system air and water balance in accordance with design requirements. Those not in accordance with design requirements are evaluated or repaired/replaced prior to final acceptance.

The main components of the control room HVAC system, control room emergency recirculation system and the control complex chilled water system are readily accessible for inspection, testing and maintenance during normal plant operation or shutdown. Redundancy in the system enables inspection, maintenance and testing to be performed without interrupting the normal operation of the systems.

Periodic tests will be performed on the control room emergency filter system. These tests will include measurement of differential pressure across the filter units and determination of filter efficiency to demonstrate that aging, weathering or poisoning of the filters has not significantly degraded the adsorption material in the charcoal and HEPA filters.

After system startup, test and balance procedures have been completed, the system will be periodically and routinely tested, checked and/or inspected as follows:

- a. Inspections are made for signs of corrosion, metal fatigue, excess vibration, and tightness of isolation dampers.
- b. Filter pressure drops are checked and recorded.
- c. Water and air flow rates in main pipes or main ducts are checked and verified against the design flow rates.
- d. Functions of dampers, valves or control devices necessary for component isolation or changeover from normal to emergency mode are verified.
- e. Charcoal filter canisters are laboratory tested in accordance with the recommendations of <Regulatory Guide 1.52>.
- f. Bearings are lubricated.
- g. The redundant components of the system are switched from the standby mode to the operating mode.
- h. As part of the Control Room Envelope Habitability Program, the unfiltered air in-leakage into the control room envelope will be measured in accordance with the recommendations of <Regulatory Guide 1.197>.
- i. An assessment of the Control Room Envelope Habitability Program will be performed in accordance with the frequencies specified in <Regulatory Guide 1.197>.

6.4.6 INSTRUMENTATION REQUIREMENTS

Habitability systems instrumentation and control equipment provides for control and monitoring of performance and status during system operation and testing. The instrumentation and control provisions for each of the systems used to ensure control room habitability are discussed in other sections. A list of these systems and the sections in which they are discussed is presented in <Section 6.4.4>. A summary discussion is presented in <Section 6.4.6.1>, <Section 6.4.6.2>, <Section 6.4.6.3>, and <Section 6.4.6.4>.

6.4.6.1 Control Room HVAC System

Operation of this system is initiated manually from the Unit 1 control room, such that either the "A" or the "B" set of supply and return components is operating, with the redundant set of components idle as backup.

The control room HVAC system (exclusive of the emergency recirculation fans, ducts and filters) is provided with controls for temperature, flow and humidity. This system is provided with alarms, status lights and indicators in the Unit 1 control room to provide the operator with sufficient information to determine the status and operation of the system from the control room. The system is also equipped with sensors and detectors to detect, alarm and monitor smoke and radiation. The details of the instrumentation and controls for this system are presented in <Section 7.3.1>.

6.4.6.2 Control Room Emergency Recirculation System

Operation of this system is initiated automatically upon receipt of an emergency signal or manually via the mode selector switch. During emergency recirculation mode of operation, one of the two fans (the one related to the active control room HVAC system) operates continuously.

The control room emergency recirculation system is provided with controls for automatic or manual initiation. This system has sensors, instruments and indicators to monitor flow, pressure drop across filters, humidity, and charcoal filter temperature. Alarms, status lights and indicators located in the Unit 1 control room provide the operator sufficient information to determine system status and operation. The system instrumentation is designed essentially to conform with <Regulatory Guide 1.52>. No recorders are provided since this system is intended for infrequent use (periodic testing and emergency periods only), and operability and reliability difficulties

become a factor for recorders that are infrequently used. Complete details of the system instrumentation and controls are presented in <Section 7.3.1>.

6.4.6.3 Lighting System

Actuation of the control room emergency dc lighting system occurs automatically if the ac lighting system power is lost.

6.4.6.4 Offsite and Onsite Power Systems

Operator control and monitoring of the status of onsite power, as well as offsite power feeds, are performed from control panels in the control room. Automatic actuation of each onsite ac emergency diesel generator occurs when an undervoltage condition is sensed on the associated bus.

TABLE 6.4-1

EQUIPMENT WHICH COULD REQUIRE CONTROL ROOM OPERATOR
ACCESS DURING AN EMERGENCY

<u>Item or Equipment</u>	<u>Location Within Control Room Envelope</u>
Control and Monitoring Panels	Identified on <Figure 6.4-2> and <Figure 9A-19>
Portable Radiation Measuring Instruments	Hallway outside control room
Emergency Procedures, Manuals and Drawings	Operator's work station
Self-Contained Breathing Apparatus	Store room
Communications Equipment	Operator's work station
Fire Extinguishing Equipment	Identified on <Figure 9A-19>
Food Supplies	Kitchen

TABLE 6.4-2

DESIGN DATA FOR CONTROL ROOM HVAC SYSTEM MAJOR COMPONENTS

A. PLENUMS

No. of plenums	2 (each 100%)
Manufacturer	AAF

B. SUPPLY FAN

No. of fans	2 (each 100%)
Manufacturer	Westinghouse
Fan type	Centrifugal, SWSI
Fan size (wheel diameter), in.	49
Arrangement	No. 3
Discharge position	Upblast
Air quantity required, cfm	45,000 per fan
Static pressure required, in. w.g.	5 per fan
Motor horsepower, hp	60 per fan
Motor location	Direct driven
Motor speed, rpm	860
Motor electrical characteristics	460V, 3 phase, 60 Hertz

C. RETURN FAN

No. of fans	2 (each 100%)
Manufacturer	Westinghouse
Fan type	Centrifugal, SWSI
Fan size (wheel diameter), in.	49
Arrangement	No. 3
Discharge position	Upblast

TABLE 6.4-2 (Continued)

Air quantity required, cfm	39,000 per fan
Static pressure required, in. w.g.	3 per fan
Motor horsepower, hp	60 per fan
Motor location	Direct Driven
Motor speed, rpm	900
Motor electrical characteristics	460V, 3 phase, 60 Hertz

D. COOLING COILS

No. of coil bank	1 per plenum
Type of coil	Horizontal finned tubes air-water counter flow
Cooling capacity, tons	155 per coil bank
Entering water temperature, °F	45
Leaving water temperature, °F	54.30
Chilled water flow rate (max.), gpm	400 per coil bank
Max. coil face velocity, fpm	600 per coil bank

E. DUCT REHEAT COILS

No. of coils	9 for Unit 1 & Unit 2
Type	Electric
Type of Controller	SCR
Quantity and kW rating	2 - 2.5 kW 1 - 3 kW 4 - 30 kW 2 - 35 kW
Electrical characteristics	480V, 3 phase, 60 Hertz

TABLE 6.4-2 (Continued)

F. ROUGHING FILTERS

No. of filter banks	1 per plenum
Manufacturer and Model	AAF, Varicel 9-2424-12 (or equal)
Material	Glass fiber with aluminum separators
No. of cells per bank	25
Rated flow per cell, cfm	2,000
Efficiency, %	90 (per ASHRAE 52-68)
Max resistance, in. w.g.	0.55 (clean)

TABLE 6.4-3

DESIGN DATA FOR CONTROL ROOM EMERGENCY RECIRCULATION SYSTEMMAJOR COMPONENTS

A. PLENUMS

No. of plenums	2 (each 100%)
Manufacturer	CVI

B. FAN

No. of fans	1 - 100% capacity per plenum
Manufacturer	Westinghouse
Fan type	Centrifugal, SWSI
Fan size (wheel diameter), in.	36-1/2
Arrangement	No. 8
Discharge position	Upblast
Motor location	Direct Driven
Motor speed, rpm	1,800 (Nominal)
Air quantity required, cfm	30,000 per fan
Static pressure required, in. w.g.	8 per fan
Motor horsepower, hp	100 per fan
Motor electrical characteristics	460V, 3 phase, 60 Hertz

C. FILTERS

1. Demisters

Manufacturer and Model	ACS 101-55
No. of demister bank	1 per plenum
No. of cells/bank	21

TABLE 6.4-3 (Continued)

Rated flow per cell, cfm	1,600
Max. resistance at rated flow, in. w.g.	0.97 (clean)

2. Roughing Filters

No. of filter banks	1 per plenum
Manufacturer and Model	Flanders, 00A-0-02-03NL (or equal)
No. of cells/bank	21
Rated flow per cell, cfm	2,000
Material	Glass fiber without separators
Efficiency, %	90 (per ASHRAE 52-68)
Max. resistance at rated flow, in. w.g.	0.49 (clean)

3. HEPA Prefilters and After-filters

No. of filter banks	2 per plenum
Manufacturer and Model	Flanders, 007-0-02-03-NU (or equal)
No. of cells per bank	21
Rated flow per cell, cfm	1,500
Material	Continuous pleated web of glass fiber without separators
Efficiency, %	99.97 for particles 0.3 microns and larger
Max. resistance at rated flow, in. w.g.	1.2 (clean)

TABLE 6.4-3 (Continued)

4. Charcoal Filters

Manufacturer and Model	CVI, HECA Module
No. of filter banks/plenum	1
No. of beds per banks	15 - 2" thick beds
Rated flow per bed, cfm	2,000
Material	Activated coconut charcoal impregnated with KI
Efficiency, %	99.9 on elemental iodine 97 on methyl iodide at 86°F and 95 percent RH
Max. resistance at rated flow, in. w.g.	1.4 (clean)
Charcoal ignition temperature °F	626

5. Electric Heating Coil

No. of coil banks	2 per plenum
Heating capacity, kW	50 each
Electrical characteristics	480V, 3 phase, 60 Hertz

TABLE 6.4-4

SINGLE FAILURE ANALYSIS

A. CONTROL ROOM EMERGENCY FILTER SYSTEM

<u>Component</u>	<u>Malfunction</u>	<u>Comments</u>
Fan	Failure of operating fan resulting in loss of air flow	Two 100 percent capacity fans are provided. If the operating fan fails, resultant loss of air flow in the duct actuates the alarm in the control room and the standby fan will be started manually by the operator.
HEPA filters	Release of DOP resulting in contamination of charcoal filters	After initial installation of the HEPA filters and periodically thereafter, the filter bank leak integrity will be determined by using DOP. On the basis of testing each HEPA filter element individually, a maximum of approximately 2.82 micrograms of DOP will be retained in each HEPA filter element. The control room emergency filter system has one HEPA filter element upstream of the charcoal filter and one HEPA filter element downstream of the charcoal filter. It is possible that the DOP can be released

TABLE 6.4-4 (Continued)

<u>Component</u>	<u>Malfunction</u>	<u>Comments</u>
		<p>from the HEPA filter as the filter temperature increases. However, the charcoal adsorber being a poor particulate filter would retain a negligible quantity of DOP released in this manner.</p> <p>Combining I-131 with the retained DOP in the HEPA filter results in an insignificant amount of methyl iodine formed. The system has enough charcoal capacity to adsorb the maximum loading of both radioactive and non-radioactive isotopes of iodine and bromine. Also, the temperature of this system is substantially below the 410°F flash point of DOP.</p>
Charcoal filters	High temperature in the charcoal beds	Two 100 percent capacity filter trains are provided in this system. In addition, temperature indicators and switches are provided in each charcoal cell bed to alarm in the control room and give an indication (readout) on rising

TABLE 6.4-4 (Continued)

<u>Component</u>	<u>Malfunction</u>	<u>Comments</u>
		<p>charcoal temperature. Also, since the air flowing through the charcoal is at a low temperature (75°-80°F), this air will keep the charcoal filter temperature from rising to the critical value (250°-300°F desorption temperature).</p> <p>However, should the temperature reach 250°F, an alarm is set in the control room to signal the operator. Actions can then be initiated to activate the charcoal water spray system and prevent the charcoal beds from desorbing. Also at this time, the standby charcoal filter train will be activated so that operation of the emergency recirculation system is not interrupted.</p>
Filter train	Failure resulting in high differential pressure across HEPA or charcoal filters	High differential pressure across filters results in low air flow which is alarmed in the control room. Operator may switch to the redundant train.

TABLE 6.4-4 (Continued)

<u>Component</u>	<u>Malfunction</u>	<u>Comments</u>
	Failure resulting in high temperatures in the charcoal bed	High temperature is annunciated in the control room. High-high temperature is alarmed in the control room to signal operator. Actions can then be taken to initiate the deluge system. Initiation of the deluge system is indicated in the control room and the operator may switch to the redundant train.
Dampers	Loss of control air	The dampers are spring assisted to position them in the safety or emergency mode of operation on loss of control air.
Ducting	Duct failure resulting in restriction or loss of air flow	This system is designed with separate and redundant ducting and dampers.
B. CONTROL ROOM HVAC SYSTEM		
Fan	Failure of operating fan resulting in loss of air flow.	Two 100 percent capacity fans are provided. If the operating fan fails, resultant loss of air flow in the duct actuates the alarm in the control room and the standby fan will be started manually by the operator.

TABLE 6.4-4 (Continued)

<u>Component</u>	<u>Malfunction</u>	<u>Comments</u>
Supply Plenum	Failure resulting in high differential pressure across roughing filter or cooling coil.	High differential pressure across filter or coil results in low air flow which is alarmed in the control room. Operator may switch to the redundant train.
	Failure resulting in high control room temperature due to loss of chilled water flow through cooling coil.	High control room temperature is monitored in control room. Operator may manually start standby control water flow complex chiller and pump and redundant air handling train.
Humidifier	Failure resulting in low or high humidity in control room.	Humidity is indicated in control room. Humidifier and associated air handling train may be started by the operator upon indication of high or low control room humidity.
Dampers	Loss of control air or electronic signal.	The dampers are spring assisted to position them in the safety or emergency mode of operation on loss of control signal.
Ducting	Duct failure resulting in restriction or loss of air flow.	This system is designed with separate and redundant ducting and dampers.

6.5 FISSION PRODUCT REMOVAL AND CONTROL SYSTEMS

6.5.1 ENGINEERED SAFETY FEATURES (ESF) FILTER SYSTEMS

The control room emergency recirculation system (CRERS), the exhaust subsystem of Fuel Handling Area Ventilation System (FHAVS) known as the Fuel Handling Area Exhaust Subsystem (FHAES), and the annulus exhaust gas treatment system (AEGTS) are the ESF filter systems that reduce the concentration of airborne radioactive contaminants following a design basis accident (DBA).

6.5.1.1 Design Bases

Design bases for the charcoal adsorber plenums of the CRERS, FHAES and the AEGTS are as follows:

a. Design Criteria

The CRERS, FHAES and AEGTS are safety-related. System design conforms with the requirements of General Design Criteria (GDC) 1, 2, 3, 4, 19, 60, and 61 of <10 CFR 50, Appendix A>. To satisfy the requirements of these GDCs, the guidance presented in <Regulatory Guide 1.3>, <Regulatory Guide 1.13>, <Regulatory Guide 1.26>, <Regulatory Guide 1.29>, <Regulatory Guide 1.47>, and <Regulatory Guide 1.52> has been considered in the design of these systems.

b. Need for Filtration

The remote possibility of airborne radioactive contaminants entering the control room following a LOCA and the requirements of <10 CFR 50.67> establish the need for the CRERS for filtration of control room air. <10 CFR 50.67> requires, in part, that adequate radiation protection be provided to permit access to, and occupancy of, the control room under accident conditions for the duration of

the accident without radiation exposure to personnel in excess of 5 rem TEDE.

The remote possibility of release of airborne radioactive contaminants due to a fuel handling accident, the requirements of GDC 61, and the recommendations of <Regulatory Guide 1.13> establish the need for the FHAES to accomplish fuel pool area air filtration. GDC 61 requires, in part, that fuel storage and handling, and radioactive waste and other systems that may contain radioactivity be designed to ensure adequate safety under normal and postulated accident conditions and that appropriate filtering systems be provided. However, no accident dose calculations credit the FHAES <Section 15.7.4> and <Section 15.7.6>.

The AEGTS is provided to reduce the radiological consequences of fission product releases in the containment from a LOCA, or fuel handling accident involving recently irradiated fuel, although credit is no longer taken for AEGTS filtration in the FHA dose calculation <Section 15.7.6>. AEGTS collects and filters the leakage from containment. Also, the AEGTS is designed to maintain a negative pressure in the annulus relative to the outside which minimizes ground level release of airborne radioactivity due to containment exfiltration during normal and postaccident conditions.

c. Component System Sizing

Two 100 percent capacity filter units are provided for the CRERS. Air flow rate for the CRERS is 30,000 cfm per plenum. Based on this assumed air flow rate and the assumed charcoal adsorber efficiencies and factors discussed in <Section 15.6>, the overall dose to the operators following an accident has been shown to satisfy the requirements of <10 CFR 50.67>.

Three 50 percent capacity filter units are provided for the FHAES. The FHAES provides exhaust flow from the fuel handling area, the fuel pool cooling equipment rooms, the control rod drive maintenance area, and the control rod drive pump areas. Flow is 30,000 cfm. Of this quantity, 15,300 cfm is exhausted directly from the fuel pool area. This air flow rate is based on flow patterns that should entrain contaminants escaping from the fuel pool area.

Two 100 percent capacity AEGTS filter units are provided. Air flow rate for the AEGTS is 2,000 cfm per plenum. Based on this flow rate, the negative pressure in the annulus is maintained at -0.25 inches of water gauge minimum continuously.

Components of these filter systems have been sized to handle system air flow based on the recommendations of <Regulatory Guide 1.52>, ERDA 76-21 and general engineering practice.

d. Fission Product Removal Capability

The fission product removal capability of the activated charcoal adsorber material used in the CRERS, FHAES and AEGTS is based on the recommendations of <Regulatory Guide 1.52>.

The decontamination efficiency of the AEGTS charcoal adsorber is 99 percent for both elemental iodine and organic species of iodine. However, none of the radiological consequence analyses credit any removal of elemental and organic iodines by the charcoal filters in the AEGTS. The AEGTS charcoal adsorber bed is 4 inches deep with annulus exhaust air maintained at less than 95 percent relative humidity.

The decontamination efficiencies of the CRERS and FHAES charcoal adsorbers are 99 percent for elemental iodine and 95 percent for

organic species of iodine. For the CRERS, the design basis LOCA <Section 15.6.5.5.1.9> credits an 80 percent removal efficiency of elemental and organic iodines by the charcoal filters in the CRERS. The Steam System Piping Break Outside Containment <Section 15.6.4>, Control Rod Drop Accident <Section 15.4.9>, and the Fuel Handling Accident <Section 15.7.4> and <Section 15.7.6>, do not take credit for the charcoal filters in the CRERS. For the FHAES, the alternative source term FHA analysis took no credit for the charcoal adsorbers. The CRERS and FHAES charcoal adsorber beds are 2 inches deep. Exhaust air for both plenums is maintained at less than 95 percent relative humidity.

The HEPA filter efficiency of all the plenums is 99.97 percent on particles 0.3 microns and larger. However, the LOCA analysis only credits the HEPA filters in the AEGTS and CRERS at an efficiency of 99 percent. The other design basis radiological calculations do not take credit for the HEPA filters in the AEGTS, CRERS, or FHAES.

Additional bases for the design of the CRERS, FHAES and AEGTS are presented in <Section 6.4>, <Section 9.4.2>, and <Section 6.5.3> respectively.

6.5.1.2 System Design

The design features of the CRERS, FHAES and AEGTS are compared to the recommendations of <Regulatory Guide 1.52> in <Table 6.5-1>, <Table 6.5-2>, and <Table 6.5-3> respectively.

Design of the activated charcoal adsorber plenums used in the CRERS, FHAES and AEGTS follows the guidelines of <Regulatory Guide 1.52> and ERDA 76-21.

Each charcoal adsorber plenum contains the following:

- a. Demisters to remove large particles and water droplets (about 1 micron diameter).
- b. Roughing filters to remove large particles (about 1 micron).
- c. HEPA filters to remove small particles (0.3 to 1 micron), including fission product aerosols (particulates).
- d. Electric heater coils.

- e. Gasketless activated charcoal adsorber beds to remove gaseous elemental and organic iodines.
- f. HEPA filters downstream of the charcoal beds to remove charcoal particles that may be entrained in the air stream.
- g. A fan external to the plenum.
- h. Instrumentation.
- i. Test ports.
- j. Water deluge system for fire protection.

Plenum housings and filter support frames are shop fabricated. Potential leakage and bypass paths are closed by seal welding. No caulking or sealant is used. Housings are fabricated of carbon steel sheet. Filter support frames are of unpainted stainless steel.

Roughing and HEPA filters are mounted in frames in accordance with the recommendations of ERDA 76-21.

The activated charcoal adsorber is bulk loaded into the permanently installed, gasketless adsorber section which is seal welded to the housing and support frames of the plenum. Tray type activated charcoal adsorber units are not used.

Spent charcoal adsorber material is vacuumed from the bottom or top of the plenum and is loaded into 55 gallon drums for shipment off site. New charcoal adsorber material is added at the top of the adsorber section. Personnel are not directly exposed to potentially contaminated adsorber material during the changing procedure.

The AEGTS and CRERS Roughing and HEPA filters are replaced when pressure drop across a filter exceeds the technical specification value. Pressure drop is measured by permanently installed differential pressure indicators. Charcoal adsorber material is changed when laboratory test results from representative samples show that the adsorber fails to satisfy the requirements of the Ventilation Filter Test Program.

Essential services, such as power and electrical control cables associated with ESF filter systems, are protected as described in <Section 8.3.1.4>.

The charcoal adsorber portion of each filter train is provided with a high temperature detection and water spray system to allow flooding of the charcoal bed in the unlikely event of high temperature in the charcoal (to preclude the possibility of iodine desorption).

6.5.1.3 Design Evaluation

Design and safety evaluations of the CRERS, FHAES and AEGTS are presented in <Section 6.4>, <Section 9.4.2>, and <Section 6.5.3> respectively.

The charcoal adsorber plenums are not exposed to conditions that can impair plenum efficiency.

The FHAES and AEGTS are normally operated continuously during plant operation.

The CRERS is operated for at least 15 minutes each month as recommended by <Regulatory Guide 1.52>; during this operation of CRERS, the exhaust air is free of radioactive contaminants. Air exhausted or circulated through the charcoal adsorber plenums is not expected to contain enough

radioactive contaminants following a DBA to develop decay heat that could ignite the charcoal adsorber material.

The charcoal adsorber plenums are redundant, physically separated and powered from separate Class 1E electrical systems.

In the event smoke from a fire is exhausted through any charcoal filter, the filter will be tested for any degradation in charcoal performance as a result of the smoke. This testing will be performed in accordance with the Ventilation Filter Testing Program. If the testing indicates that degradation has occurred beyond acceptable limits, the charcoal will be replaced. For charcoal filters in systems needed to mitigate the consequences of a LOCA, the filters will be tested and the charcoal replaced, if required, within a period specified in the technical specifications.

6.5.1.4 Tests and Inspections

Tests and inspections of the CRERS, FHAES and AEGTS charcoal adsorber plenums are performed prior to startup and on a periodic basis thereafter. Other tests and inspections of these filter systems are discussed in <Section 6.4>, <Section 9.4.2>, and <Section 6.5.3> respectively.

6.5.1.4.1 Filter and Charcoal Adsorber Tests

HEPA filters are individually tested by an appropriate filter test facility at 100 percent and 20 percent of rated flow, in accordance with the recommendation of <Regulatory Guide 1.52>. Original or replacement HEPA filters used in the CRERS, FHAES and AEGTS are tested as indicated above.

Each batch of charcoal adsorber material satisfies the "acceptable results" recommended by <Regulatory Guide 1.52>. Since the charcoal

adsorber material is expected to be replaced several times during the 40 year life of the plant, test methods used may change. Therefore, only "acceptable results" of the tests are specified.

6.5.1.4.2 Inplace Testing

After all the air cleaning units for CRERS, FHAES and AEGTS are installed, preoperational and inplace tests are performed as described in <Chapter 14> and <Table 1.8-1>, with respect to <Regulatory Guide 1.52>.

6.5.1.4.3 Operation

Periodic testing and inspection of CRERS, FHAES and AEGTS shall be performed in accordance with the Ventilation Filter Testing Program described in the plant technical specifications.

6.5.1.5 Instrumentation Requirements

Instrumentation and actuation requirements for the CRERS, AEGTS and FHAES are discussed in <Section 6.4.6.2>, <Section 7.3.1.1.9>, and <Section 7.3.1.1.16> respectively.

Each of the activated charcoal adsorber plenums has locally mounted differential pressure indicators to show the pressure drop across each filter bank in a plenum. These indicators permit the operator to determine when prefilters and HEPA filters should be changed.

Each charcoal adsorber bed is provided with a permanently installed temperature sensing device and temperature switches which actuate alarms upon detection of a bed temperature of 225°F and 250°F. The water deluge fire protection system is manually actuated upon subsequent verification of a plenum fire.

6.5.1.6 Materials

Estimated quantities of materials used in the activated charcoal adsorber plenums for CRERS, FHAES and AEGTS are listed in <Table 6.5-4>, <Table 6.5-5>, and <Table 6.5-6> respectively. The governing specifications for the various materials are also listed and provide information regarding chemical composition of materials used.

There are no radiolytic or pyrolytic decomposition products from the ESF filter systems. Actuation of the activated charcoal adsorber plenum water deluge fire protection systems will extinguish a charcoal fire before pyrolytic decomposition products are formed. None of these systems are located in areas where gamma radiation sources are sufficiently strong to cause radiolytic decomposition products. Therefore, decomposition products do not affect any engineered safety features.

6.5.2 CONTAINMENT SPRAY SYSTEM

6.5.2.1 Design Bases

- a. The containment spray system (CSS) is a part of the residual heat removal (RHR) system.
- b. The CSS provides containment cooling following a loss-of-coolant accident, in addition to being a fission product removal mechanism. Refer to <Section 6.2.2> for the heat removal function of the CSS.
- c. The CSS consists of two completely redundant and independent loops. (Loops "A" & "B")
- d. The CSS is designed to remain operable in the containment accident environment, which is discussed in <Section 3.11>.

- e. The CSS is designed such that a single failure of any active component will not degrade the ability of the system to fulfill design objectives. Each loop of the CSS receives power from a separate emergency diesel generator, in the event that offsite power is unavailable during an accident. The two loops are physically separate from each other, so that a failure in one loop will not result in failure of the other loop due to fire, flooding, pipe breaks, or missiles.
- f. The CSS is designed to Seismic Category I requirements. System components as appropriate are designed to meet ASME Code Section III, Class 2 requirements.
- g. The CSS is designed to permit periodic testing as described in <Section 6.2.2> and <Section 6.5.2.4>.

6.5.2.2 System Design

The source of water supply for the CSS during all phases of system operation is the suppression pool. The water is pumped by the RHR pumps through the RHR heat exchangers to the containment spray headers. The CSS is shown on <Figure 5.4-14 (3)>.

Each loop of the containment spray system consists of three concentric spray ring headers with equally spaced spray nozzles. Loop "A" consists of 104 nozzles on the top ring, 113 nozzles on the middle rings and 129 nozzles on the bottom ring. Loop "B" consists of 102 nozzles on the top, 113 nozzles on the middle and 129 nozzles on the bottom ring. The system diagram for the containment spray system is shown in <Figure 6.5-3>.

The spray nozzles used are SPRACO 1713A hollow-cone ramp bottom nozzles each of which is capable of a flow of 15.2 gpm with a pressure drop of 40 psid. These nozzles have an approximately 3/8-inch spray orifice and

are not subject to clogging by particles less than 1/4 inch in maximum dimension. Each nozzle header is independently oriented to ensure efficient coverage of the containment volume.

The minimum water supply flow rate to the containment spray system is 5,250 gpm.

There are no spray additives for the CSS (other than the pH buffering chemical, boron solution, from the standby liquid control system, which is injected into the reactor vessel and suppression pool following a design basis LOCA). The CSS will automatically initiate after 10 minutes of a LOCA signal if containment pressure exceeds the high pressure setpoint. If containment pressure is less than high pressure setpoint, the control room operator can actuate the system manually.

The sprayed and unsprayed volumes and regions of the containment, with their associated mixing rates, are discussed in <Section 15.6.5>. The CSS takes no credit for ventilation.

6.5.2.3 Design Evaluation

The containment spray mode of the RHR system is safety-related and is designed to operate following the postulated design basis loss-of-coolant accident. A high degree of system reliability is maintained through system quality control, by general equipment arrangement to provide access for inspection and maintenance and by periodic testing. A single failure analysis of the RHR system is given in <Section 6.2.2>.

Because of the large volume of the containment atmosphere swept by the sprays, the spray mode serves as a removal mechanism for fission products postulated to be dispersed in the containment atmosphere following an accident. Radioiodine in its various forms is the fission

product of primary concern in the evaluation of a loss-of-coolant accident. The major benefit of the containment spray is its capacity to collect and remove particulate (aerosol) and elemental iodines from the containment atmosphere and thus reduce their release to the environment. Offsite and control room operator doses are a function of both the rate of removal and the final equilibrium decontamination factor. The dose calculation assumes (non-mechanistically) that the containment spray will operate for up to 24 hours. However, the dose calculations also expand on this assumption, noting the following:

- 1) The dose calculations assume the sprays are run for the first 24 hours, then are suspended. This is the most important time period for scrubbing of radiation down into the suppression pool. However, in an actual event, spray use would not necessarily be suspended at 24 hours, if appropriate conditions for their use still existed. Therefore, the phrase "up to" is not intended to be interpreted to stop using sprays after 24 hours.
- 2) The phrase "up to" is intended to mean that in an actual event, the sprays will be run when it is appropriate, and not necessarily the entire time during the first 24 hours of a LOCA. This does not invalidate the assumptions in the dose calculations. The accident guidance to operators must be written to be symptom based, rather than event based. Most postulated LOCAs will not result in large radiation releases. Therefore, it would not be appropriate to run containment sprays for 24 hours following such an event. Another critical factor in spray use is containment pressure. Use of the sprays will work to reduce containment pressures, due to steam condensation and the containment heat removal function that they provide. In the majority of cases, if a high radiation signal is present from the containment radiation monitor and pressures are elevated in containment, the sprays would be run. However, if containment pressure gets reduced to near zero and use of the sprays is terminated by the operators, this does not have an

adverse impact on offsite doses (or the dose calculations) since the driving pressure for containment and MSIV leakage has been eliminated. The dose calcs assume that the maximum allowable leakage (La) corresponding to the peak postaccident pressure (Pa) remains during the entire 24 hour period, so if containment pressure actually gets reduced to substantially less than Pa, a reduction in leakage and the resultant offsite doses will follow.

6.5.2.3.1 Iodine Removal Performance Evaluation

The elemental, organic, and particulate (aerosol) iodine removal analysis is based on the assumptions presented below and in <Table 6.5-9>.

The analysis uses the flow associated with only one RHR pump operating in the containment spray mode. It is conservatively assumed that the containment spray system directly sprays approximately 41 percent of the total containment free volume (excluding the drywell). <Section 15.6.5> provides a description of the volumes and flow paths used in the analyses.

Elemental iodine removal is credited in the drywell and containment volumes. Airborne elemental iodine is removed by deposition to the walls in the drywell and containment. As reported in <Section 5.1.2> of <NUREG/CR-0009> (Reference 1), this process is driven by the temperature differences between the surfaces and the atmosphere.

The calculated elemental iodine deposition removal factors for the sprayed and unsprayed containment and the drywell are given in <Table 6.5-11>. These elemental iodine removal factors are applied until a decontamination factor (DF) of up to 200 is obtained.

It has been conservatively assumed in these evaluations of spray removal effectiveness that organic iodine forms are not removed by the sprays.

The model for particulate (aerosol) iodine removal is built into the RADTRAD code. Input parameters are presented in <Table 6.5-9>. The particulate removal coefficient is reduced by a factor of 10 after the aerosol mass has been depleted by a factor of 50. Particulate removal is assumed to end at 24 hours when the sprays are assumed to be stopped.

6.5.2.3.2 Evaluation of Analytical Assumptions

6.5.2.3.2.1 Iodine Retention by Spray Solution

The equilibrium between the concentrations of iodine in the liquid and vapor phases is given by the partition coefficient, H , which is a

function of iodine concentration, pH and temperature. In accordance with (Reference 3) re-evolution of iodine does not have to be considered, (i.e., H will be very large) as long as the pH of the suppression pool is maintained greater than or equal to 7.0 postaccident.

6.5.2.3.2.2 Elemental Iodine and Particulate Removal Constant

(Deleted)

6.5.2.4 Tests and Inspections

The CSS spray nozzles will be verified unobstructed following maintenance which could result in nozzle blockage. The test may be performed using an inspection of the nozzle or an air or smoke flow test. Further testing and inspection of the CSS is described in <Section 6.2.2>.

6.5.2.5 Instrumentation Requirements

Instrumentation requirements for the CSS are discussed in <Section 7.3.1.1.4>.

6.5.2.6 Materials

No spray additives are used in the CSS other than the pH buffering chemical (boron solution) from the standby liquid control system, which is injected into the reactor vessel and suppression pool following a design basis LOCA.

6.5.3 FISSION PRODUCT CONTROL SYSTEMS

6.5.3.1 Primary Containment

The primary containment vessel is a hybrid pressure retaining structure composed of a steel cylinder and ellipsoidal dome secured to a steel lined reinforced concrete foundation mat. The containment vessel is designed to contain radioactive material that might be released during an accident and to ensure leak tightness during normal operating and

accident conditions. Details of the primary containment vessel design are discussed in <Section 3.8>, and a section layout of containment vessel is shown on <Figure 3.8-1>. The primary containment pressure suppression concept uses the General Electric Mark III design.

The primary containment vessel steel shell, mechanical penetrations, isolation valves, hatches, and locks will limit release of radioactive materials (subsequent to postulated accidents) such that the resulting offsite doses are less than the licensing basis limits. Primary containment parameters affecting fission product release accident analyses are given in <Table 6.5-7>.

Long term primary containment pressure response to the design basis accident is discussed in <Section 6.2.1>.

Redundant, safety-related hydrogen recombiners are provided in the primary containment as the primary means of controlling postaccident hydrogen concentrations. A hydrogen purge system is provided for alternate hydrogen control. Details of the postaccident hydrogen control system are discussed in <Section 6.2.5>.

During normal operation, the primary containment may be intermittently vented through a charcoal filter train. These normal primary containment ventilation supply and exhaust system penetrations are automatically isolated in response to LOCA signals (high drywell pressure and low reactor water level) and high radiation signals.

The penetrations are provided with redundant, Seismic Category I, air operated, fail-closed, ASME Code Section III, Class 2 butterfly valves which assure prompt and tight closure of the openings. <Section 9.4.6> presents a detailed discussion of the containment purge system.

6.5.3.2 Secondary Containment

The secondary containment (shield building) is a reinforced concrete structure consisting of a flat foundation mat, a cylindrical wall and a shallow dome. The secondary containment boundary consists of the volume between the shield building and the primary containment vessel steel shell. Details of the shield building structural design are discussed in <Section 3.8> and the boundary region is shown in <Figure 3.8-1>.

The annulus exhaust gas treatment system (AEGTS) processes the ambient air in the annular space between the shield building and the primary containment vessel to limit the release to the environment of radioisotopes which may leak from the primary containment under accident conditions. The AEGTS is a recirculation type system with split flow. Some of the filtered air extracted from the annulus space is recirculated and some is discharged to the unit vent. An analysis indicating the effectiveness of this system in controlling contamination releases is presented in <Section 15.6.5>.

6.5.3.2.1 Design Bases

Design bases for the annulus exhaust gas treatment system are as follows:

- a. The AEGTS is classified as Safety Class 2, Seismic Category I. The design of this system complies with the requirements of General Design Criteria (GDC) 1, 2, 3, 4, and 5 of <10 CFR 50, Appendix A>, and <10 CFR 50, Appendix B> concerning ground level accident releases. The recommendations of <Regulatory Guide 1.26>, <Regulatory Guide 1.29>, <Regulatory Guide 1.47>, <Regulatory Guide 1.52>, <Regulatory Guide 1.53>, and <Regulatory Guide 3.2>, National Fire Protection Association (NFPA) 90A, and Branch Technical Position APCS 9.5-1 have also been considered in the system design and equipment procurement.

b. The AEGTS is:

1. Required to function during normal, shutdown and refueling operations, loss of offsite power periods and following a LOCA.
2. Started manually from the control room with the redundant system started automatically when low system air flow is indicated.
3. Designed to continuously maintain a negative pressure differential of 0.25 inches of water gauge minimum between the containment vessel annulus ambient and the outside (sensed in the auxiliary, intermediate and fuel handling buildings). The design parameters of this system include the leakage through the shield building (100 percent of the annulus volume per day), leakage from the containment vessel following an accident (0.2 percent of the containment volume per day), increased annulus temperature following an accident, increased annulus air pressure resulting from the containment vessel expansion following an accident, and the discharge from the hydrogen purge subsystem.
4. Designed to direct the exhaust flow from the annulus through a charcoal filter plenum to ensure that the release of radioactivity to the environment is below permissible discharge limits.
5. Designed so that the exhaust inlet points from the annulus are remotely located from the return air to the annulus, thus promoting mixing of the annulus air.
6. Continuously monitored in the control room to indicate system operating status, system malfunction, high radiation in fan

discharges, high smoke in fan discharges, temperatures in the charcoal plenums, and annulus to outside ambient pressure differential (sensed in the auxiliary, intermediate and fuel handling buildings).

7. Provided with redundant and separated equipment, control and power supplies so that a single active component failure will not prevent satisfactory system operation.
8. Designed with system operating components located in equipment areas not affected by internally generated missiles, pipe whip or jet impingement resulting from breaks in high or moderate energy piping.

6.5.3.2.2 System Description

The AEGTS is shown on <Figure 6.5-1>. This system functions continuously during normal, shutdown and refueling operations, during loss of offsite power periods and following a LOCA to maintain a negative pressure differential between the containment vessel annulus ambient and the outside.

The system includes two 100 percent capacity filter plenums (M15-D001A, B), two 100 percent capacity centrifugal fans (M15-C001A, B), and redundant supply, recirculation and exhaust ductwork systems. The plenums and fans are located in the intermediate building at Elevation 620'-6". The system is provided with "A" equipment and "B" equipment, located in separate rooms. The duct distribution systems in the annulus are arranged so that the annulus ambient air is continuously circulated and mixed throughout the annulus space and then directed to the charcoal filter plenum. Air is extracted at one location and returned at another remote location to ensure adequate mixing of the recirculated air with the annulus volume. Also, adequate distribution ductwork is provided. The main exhaust header draws air

from various regions of the annulus and the recirculation header returns the recirculated air at a different region. The exhaust header and the recirculation headers are located at appropriate distance from each other to assure that adequate mixing occurs in the annulus <Figure 6.5-2>. A portion of this filtered air is then discharged to the unit vent and the remaining portion is returned to the annulus. The continuous circulation and mixing ensures that any inleakage to the annulus ambient will be mixed with this ambient before being filtered and discharged. The amount of air discharged to the plant vent and the amount returned to the annulus is automatically controlled so that the negative differential pressure is maintained.

The total extraction rate from the annulus is 2,000 cfm and the design maximum discharge rate to the atmosphere is 2,000 cfm. During normal operation the expected discharge to the unit vent is approximately 700 cfm, based on test data leakage through the shield building into the annulus (100 percent of annulus volume per day) at a negative pressure differential of 0.66 inches of water gauge. The 0.66 inches of water gauge pressure differential is provided to maintain the 0.25 inches of water gauge minimum pressure differential required due to instrument location, to meet plant post-LOCA conditions, and to adjust for all environmental conditions. During an accident, the maximum expected discharge rate is approximately 1,000 cfm. This takes into consideration a postulated leakage from the containment vessel to the annulus of 1.6 cfm (equivalent to 0.2 percent of containment volume per day at a containment pressure of 15 psi), the air expansion due to increase in temperature during the accident, and the increase in annulus air pressure due to the expansion of the containment steel shell. However, the discharge and recirculation dampers will modulate to vary the discharge flow from the expected maximum (approximately 1,000 cfm) to the design maximum (2,000 cfm), as required, to achieve and maintain a negative pressure differential of 0.66 inches of water gauge between the annulus and the atmosphere (sensed in the auxiliary, intermediate and fuel handling buildings).

Two pressure differential transmitters, each located 180° apart, are provided for each filter plenum and fan. These transmitters are used to monitor the negative pressure differential in the annulus and send appropriate signals to the differential pressure controllers and recorders located in the control room. The differential pressure controller will modulate the discharge and recirculation dampers accordingly.

Regular ac offsite power sources are provided for this system. If loss of offsite power occurs during a LOCA, redundant emergency power is available from the diesel generators. Inactive circulating and filtering components are isolated from the annulus space and the unit by check dampers, and by automatically controlled duct dampers. The redundant system automatically starts upon indication of low air flow from the operating system.

Design information for the annulus exhaust gas treatment system components is listed in <Table 6.5-8>.

6.5.3.2.3 Safety Evaluation

The AEGTS maintains a negative pressure differential between the containment vessel annulus and the outside so that leakage from the containment vessel will be detained in the annular space, mixed with the annulus space air, diluted with air leakage into the annular space and filtered before release to the unit vent.

The mixing and dilution of the annulus air is ensured by the supply and exhaust duct arrangement which includes distribution ducts with multiple supply and exhaust outlets along the circumference of the annulus space. The supply distribution ducts are near the bottom of the annulus (approx. Elevation 599'-0") and the exhaust distribution ducts are near the top of the annulus (approx. Elevation 742'-0"). The annulus width is five feet. This resultant volume is the space in which containment

vessel leakage is mixed with annulus space air before being directed to the filter plenum.

The maximum exhaust flow rate from the annulus space directed to the filter is 2,000 cfm and the maximum expected amount of the filtered air released to the unit vent is 1,000 cfm. The remaining 1,000 cfm is circulated back to the annulus space for subsequent refiltering. This arrangement of duct systems, and the high proportion of recirculated and refiltered air, assures maximum mixing of leakage and maximum filtering before release to the unit vent.

Minimum release of contamination after filtering is further assured by the filter plenum components which include a roughing filter, a demister, an electric heating coil for lowering of relative humidity, HEPA prefilters, 4-inch deep charcoal filters, and HEPA after-filters. When the system is operating at normal meteorological conditions (wind velocity up to 30 mph), the annulus is maintained at a negative pressure differential of 0.25 inches water gauge minimum relative to the outside by modulating the exhaust and recirculation dampers (sensed in the auxiliary, intermediate and fuel handling buildings).

Operation of the system is monitored with alarms in the control room for system malfunctions including high temperature in ducts and plenums, high smoke, high radiation, loss of negative differential pressure, and high relative humidity of the exhaust air.

Malfunctioning system components which affect system operation are stopped and isolated and redundant components are automatically placed in service. This assures maintenance of the negative differential pressure in the annulus at all times.

The main components of the system are located in separate rooms. No unguarded high energy lines pass through the annulus space or in the AEGTS equipment areas.

Redundancy in this system provides capability to maintain a negative differential pressure and to remove contaminants when considering failure of an active or passive component.

If decay heat in the charcoal filter section raised the charcoal temperature to 250°F, a control room alarm would alert the operator so that the charcoal filter deluge system could be manually activated upon verification of a plenum fire.

6.5.3.2.4 Inspection and Testing Requirements

The various components of the AEGTS are accessible for inspection and testing during normal plant operation. The ability to isolate an idle redundant component enables inspection maintenance and testing to be performed while the system is in normal operation. Periodic tests, as recommended by <Regulatory Guide 1.52>, will be performed on the systems. These tests will include measurement of differential pressure across the filter banks and field and laboratory determination of filter leakage and efficiency. This will demonstrate that aging, weathering or poisoning of the filters has not significantly degraded the filter mounting or adsorptive material in the charcoal banks.

These tests will also include verification of the functional performance of fans, dampers, controls, and other safety devices to ensure that these components perform their function reliably.

6.5.3.2.5 Instruments, Controls and Protective Devices

Operation of the AEGTS is initiated by manually starting (from the control room) one of the two exhaust fans which then operates continuously. The details of the instrumentations and controls for this system are discussed in <Section 7.3.1>.

6.5.4 ICE CONDENSER AS A FISSION PRODUCT CLEANUP SYSTEM

This section is not applicable to PNPP.

6.5.5 REFERENCES FOR SECTION 6.5

1. Postma, A. K.; Sherry, R. R.; Tam, P. S.; "Technological Bases for Models of Spray Washout of Airborne Contaminants in Containment Vessels," <NUREG/CR-0009>, October 1978.
2. ANSI/ANS-56.3-1979, "American National Standard for PWR and BWR Containment Spray System Design Criteria."
3. Electrical Power Research Institute, "Generic Framework for Application of Revised Accident Source Terms to Operating Plants," TR-105909, Interim Report, November, 1995.
4. Regulatory Guide 1.52, "Design, Inspection, and Testing Criteria for Air Filtration and Adsorption Units of Post-Accident Engineered-Safety-Feature Atmosphere Cleanup Systems in LightWater-Cooled Nuclear Power Plants", Revision 2, March 1978.
5. NUREG-0800, "Standard Review Plan (SRP) 6.5.2, Containment Spray As A Fission Product Cleanup System," Revision 4, March 2007.
6. Calculation 3.2.15.16, "Design Basis LOCA Dose Evaluation Using Alternate Source Terms," Revision 1, July 2014.

TABLE 6.5-1

COMPARISON OF CONTROL ROOM EMERGENCY RECIRCULATION^{(1) (2)}
SYSTEM WITH <REGULATORY GUIDE 1.52> POSITIONS

<u>Regulatory Position</u>	<u>System Design Feature</u>
1.a	The design conforms with this position.
1.b	The design conforms with this position.
1.c	The design conforms with this position.
1.d	The design conforms with this position.
1.e	The design conforms with this position.
2.a	The design conforms with this position.
2.b	The design conforms with this position.
2.c	The design conforms with this position.
2.d	The filter units are not exposed to pressure surges from the postulated DBA.
2.e	The design conforms with this position.
2.f	The design conforms with this position.
2.g	Pressure drops and flow rates are not monitored in the control room.
2.h	The design conforms with this position.
2.i	The design conforms with this position.
2.j	The design conforms with the intent of this position. See <Section 12.1> for a discussion of conformance with <Regulatory Guide 8.8>.
2.k	The design conforms with this position.
2.l	The design conforms with this position. Duct and housing leak tests will be performed in accordance with Section 6 of ANSI N510-1980 instead of ANSI N510-1975.
3.a	The design conforms with this position.
3.b	The design conforms with this position. Heater sizing conforms with Section 5.5 of ANSI N509-1980 rather than ANSI N509-1976.

TABLE 6.5-1 (Continued)

<u>Regulatory Position</u>	<u>System Design Feature</u>
3.c	The design conforms with this position.
3.d	The design conforms with this position.
3.e	The design conforms with this position.
3.f	The design conforms with this position.
3.g	The design conforms with this position.
3.h	The design conforms with the intent of the recommendations of Section 4.5.8 of ERDA 76-21.
3.i	The design conforms with this position. The impregnated activated carbon shall meet the qualification and batch test results summarized in Table 5-1 of ANSI N509-1980, which meets or exceeds the requirements of Table 5-1 of ANSI N509-1976.
3.j	The design conforms with this position. The adsorbent shall meet the requirements of Table 5-1 of ANSI N509-1980, which meets or exceeds the requirements of Table 5-1 of ANSI 509-1976.
3.k	The design of the charcoal adsorber section considers possible radioactivity-induced fires. Water spray deluge fire protection system is provided.
3.l	The design conforms with the intent of the recommendations of Sections 5.7 and 5.8 of ANSI N509-1976.
3.m	The design conforms with this position.
3.n	The design conforms with the recommendations of Section 5.10 of ANSI N509-1976.
3.o	The design conforms with this position.

TABLE 6.5-1 (Continued)

<u>Regulatory Position</u>	<u>System Design Feature</u>
3.p	The design conforms with the intent of the recommendations of Section 5.9 of ANSI N509-1976.
4.a	The design conforms with the intent of the recommendations of Section 2.3.8 of ERDA 76-21 and Section 4.7 of ANSI N509-1976.
4.b	The maximum length of component plus 2'-6", approximately, is provided due to space limitation imposed by equipment room size. It was determined that this is adequate for the replacement of the roughing and HEPA filters and is consistent with the manufacturer's recommendations.
4.c	The design conforms with this position.
4.d	Refer to Regulatory Position 6.a.
4.e	Preoperational Phase Testing meets the intent of this position. The testing will be performed while active construction is in progress on the project but sufficiently complete to assure that the installed HEPA filters and charcoal are not subjected to airflow that would invalidate inplace testing.

TABLE 6.5-1 (Continued)

<u>Regulatory Position</u>	<u>System Design Feature</u>
6.a	Testing procedure will operate the train for at least 15 minutes each month. The heaters will not be tested due to revised charcoal adsorber testing requirements consistent with ventilation systems without heaters.
6.b	Testing conforms to this position.
6.c	Testing procedures will meet the intent of this position. An exception has been made to use ANSI N510-1980 in lieu of ASME N511-2007 to align with Plant Technical Specifications.
6.d	Testing conforms to this position.
6.e	Testing conforms to this position.
6.f	Testing procedures will meet the intent of this position. An exception has been made to use ANSI N510-1980 in lieu of ASME N511-2007 to align with Plant Technical Specifications.
6.g	Testing conforms to this position.
6.h	Testing conforms to this position.
6.i	Testing conforms to this position.
7.a	Testing meets the intent of this position.
7.b	Sampling and analysis is performed at least once per 18 months, or following painting, fire or chemical release in any ventilation zone communicating with the subsystem. Laboratory testing is performed in accordance with ASTM-D3803-1989.

TABLE 6.5-1 (Continued)

<u>Regulatory Position</u>	<u>System Design Feature</u>
7.c	The activated carbon methyl iodide penetration has an acceptance criteria of less than 10%. The LOCA analysis is the only accident analysis that credits the control room emergency recirculation system charcoal adsorbers. The LOCA analysis utilizes an 80 percent elemental and organic iodine removal efficiency with a safety factor of two.
7.d	Testing conforms to this position.

TABLE 6.5-1 (Continued)

Regulatory Position

System Design Feature

NOTE:

- (1) As discussed in <Section 15.6.5.5.1.9>, the design basis LOCA only credits an 80 percent removal efficiency of elemental and organic iodines by the charcoal filters in the CRERS. The steam system piping break outside containment <Section 15.6.4>, Control Rod Drop Accident <Section 15.4.9>, and the Fuel Handling Accident <Section 15.7.4> and <Section 15.7.6>, do not take credit for the charcoal filters in the CRERS.
- (2) Regulatory Positions 1.a through 4.e evaluate System Design Features against Revision 2 of <Regulatory Guide 1.52>, which detailed Environmental Design Criteria, System Design Criteria, Component Design Criteria and Qualification Testing, and Maintenance. Regulatory Positions 6.a through 7.d evaluate System Design Features against Revision 4 of <Regulatory Guide 1.52>, which detail In-Place Testing Criteria and Laboratory Testing Criteria for Activated Carbon. Revision 4 of <Regulatory Guide 1.52> has been implemented by License Amendment 180.

TABLE 6.5-2

COMPARISON OF FUEL HANDLING AREA EXHAUST⁽¹⁾
SUBSYSTEM WITH <REGULATORY GUIDE 1.52> POSITIONS

<u>Regulatory Position</u>	<u>System Design Feature</u>
1.a	The design conforms with this position.
1.b	The design conforms with this position.
1.c	The design conforms with this position.
1.d	The design conforms with this position.
1.e	The design conforms with this position.
2.a	The design conforms with this position.
2.b	The design conforms with this position.
2.c	The design conforms with this position.
2.d	The filter units are not exposed to pressure surges from the postulated DBA.
2.e	The design conforms with this position.
2.f	The design conforms with this position.
2.g	Pressure drops and flow rates are not monitored in the control room.
2.h	The design conforms with this position.
2.i	The design conforms with this position.
2.j	The design conforms with the intent of this position. See <Section 12.1> for a discussion of conformance with <Regulatory Guide 8.8>.
2.k	The design conforms with this position.
2.l	The design conforms with this position. Duct and housing leak tests will be performed in accordance with Section 6 of ANSI N510-1980 instead of ANSI N510-1975
3.a	The design conforms with this position.
3.b	The design conforms with this position.
3.c	The design conforms with this position.

TABLE 6.5-2 (Continued)

<u>Regulatory Position</u>	<u>System Design Feature</u>
3.d	The design conforms with this position.
3.e	The design conforms with this position.
3.f	The design conforms with this position.
3.g	The design conforms with this position.
3.h	The design conforms with the intent of the recommendations of Section 4.5.8 of ERDA 76-21.
3.i	The design conforms with this position. The impregnated activated carbon shall meet the qualification and batch test results summarized in Table 5-1 of ANSI N509-1980, which meets or exceeds the requirements of Table 5-1 of ANSI N509-1976.
3.j	The design conforms with this position. The adsorbent shall meet the requirements of Table 5-1 of ANSI N509-1980, which meets or exceeds the requirements of Table 5-1 of ANSI N509-1976.
3.k	The design of the charcoal adsorber section considers possible radioactivity-induced fires. Water spray deluge fire protection system is provided.
3.l	The design conforms with the intent of the recommendations of Sections 5.7 and 5.8 of ANSI N509-1976.
3.m	The design conforms with this position.
3.n	The design conforms with the recommendations of Section 5.10 of ANSI N509-1976.
3.o	The design conforms with this position.

TABLE 6.5-2 (Continued)

<u>Regulatory Position</u>	<u>System Design Feature</u>
3.p	The design conforms with the intent of the recommendations of Section 5.9 of ANSI N509-1976.
4.a	The design conforms with the intent of the recommendations of Section 2.3.8 of ERDA 76-21 and Section 4.7 of ANSI N509-1976.
4.b	The maximum length of component plus 2'-6", approximately, is provided due to space limitation imposed by equipment room size. It was determined that this is adequate for the replacement of the roughing and HEPA filters.
4.c	The design conforms with this position and is consistent with the manufacturer's recommendations.
4.d	Refer to Regulatory Position 6.a.
4.e	Preoperational Phase Testing meets the intent of this position. The testing will be performed while active construction is in progress on the project but sufficiently complete to assure that the installed HEPA filters and charcoal are not subjected to airflow that would invalidate inplace testing.

TABLE 6.5-2 (Continued)

<u>Regulatory Position</u>	<u>System Design Feature</u>
6.a	An exception has been taken to this position. The Fuel Handling Building and FHBES are not credited in mitigating the consequences of a fuel handling accident in the Fuel Handling Building, thus there is no need for testing.
6.b	Testing conforms to this position.
6.c	Testing procedures will meet the intent of this position. An exception has been made to use ANSI N510-1980 in lieu of ASME N511-2007 to align with Plant Technical Specifications.
6.d	Testing conforms to this position.
6.e	Testing conforms to this position.
6.f	Testing procedures will meet the intent of this position. An exception has been made to use ANSI N510-1980 in lieu of ASME N511-2007 to align with Plant Technical Specifications.
6.g	Testing conforms to this position.
6.h	Testing conforms to this position.
6.i	Testing conforms to this position.
7.a	Testing meets the intent of this position.
7.b	Sampling and analysis is performed at least once per 18 months, or following painting, fire or chemical release in any ventilation zone communicating with the subsystem. Laboratory testing is performed in accordance with ASTM-D3803-1989.

TABLE 6.5-2 (Continued)

<u>Regulatory Position</u>	<u>System Design Feature</u>
7.c	Testing procedures meet the intent of this position. Acceptance criteria is penetration less than 10 percent. Laboratory testing is performed in accordance with ASTM-D3808-1989.
7.d	Testing conforms to this position.

NOTE:

- ⁽¹⁾ Regulatory Positions 1.a through 4.e evaluate System Design Features against Revision 2 of <Regulatory Guide 1.52>, which detailed Environmental Design Criteria, System Design Criteria, Component Design Criteria and Qualification Testing, and Maintenance. Regulatory Positions 6.a through 7.d evaluate System Design Features against Revision 4 of <Regulatory Guide 1.52>, which detail In-Place Testing Criteria and Laboratory Testing Criteria for Activated Carbon. Revision 4 of <Regulatory Guide 1.52> has been implemented by License Amendment 180.

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TABLE 6.5-3

COMPARISON OF ANNULUS EXHAUST GAS TREATMENT^{(1) (2)}
SYSTEM WITH <REGULATORY GUIDE 1.52> POSITIONS

<u>Regulatory Position</u>	<u>System Design Feature</u>
1.a	The design conforms with this position.
1.b	The design conforms with this position.
1.c	The design conforms with this position.
1.d	The design conforms with this position.
1.e	The design conforms with this position.
2.a	The design conforms with this position.
2.b	The design conforms with this position.
2.c	The design conforms with this position.
2.d	The filter units are not exposed to pressure surges from the postulated DBA.
2.e	The design conforms with this position.
2.f	The design conforms with this position.
2.g	Pressure drops and flow rates are not monitored in the control room.
2.h	The design conforms with this position.
2.i	The design conforms with this position.
2.j	The design conforms with the intent of this position. See <Section 12.1> for a discussion of conformance with <Regulatory Guide 8.8>.
2.k	The design conforms with this position.
2.l	The design conforms with this position. Duct and housing leak tests will be performed in accordance with Section 6 of ANSI N510-1980 instead of ANSI N510-1975
3.a	The design conforms with this position.
3.b	The design conforms with this position.
3.c	The design conforms with this position.

TABLE 6.5-3 (Continued)

<u>Regulatory Position</u>	<u>System Design Feature</u>
3.d	The design conforms with this position.
3.e	The design conforms with this position.
3.f	The design conforms with this position.
3.g	The design conforms with this position.
3.h	The design conforms with the intent of the recommendations of Section 4.5.8 of ERDA 76-21.
3.i	The design conforms with this position. The impregnated activated carbon shall meet the qualification and batch test results summarized in Table 5-1 of ANSI N509-1980, which meets or exceeds the requirements of Table 5-1 of ANSI N509-1976.
3.j	The design conforms with this position. The adsorbent shall meet the requirements of Table 5-1 of ANSI N509-1980, which meets or exceeds the requirements of Table 5-1 of ANSI N509-1976.
3.k	The design of the charcoal adsorber section considers possible radioactivity-induced fires. Water spray deluge fire protection system is provided.
3.l	The design conforms with the intent of the recommendations of Sections 5.7 and 5.8 of ANSI N509-1976.
3.m	The design conforms with this position.
3.n	The design conforms with the recommendations of Section 5.10 of ANSI N509-1976.
3.o	The design conforms with this position.

TABLE 6.5-3 (Continued)

<u>Regulatory Position</u>	<u>System Design Feature</u>
3.p	The design conforms with the intent of the recommendations of Section 5.9 of ANSI N509-1976.
4.a	The design conforms with the intent of the recommendations of Section 2.3.8 of ERDA 76-21 and Section 4.7 of ANSI N509-1976.
4.b	The maximum length of component plus 2'-6", approximately, is provided due to space limitation imposed by equipment room size. It was determined that this is adequate for the replacement of the roughing and HEPA filters and is consistent with the manufacturer's recommendations.
4.c	The design conforms with this position.
4.d	Refer to Regulatory Position 6.a.
4.e	Preoperational Phase Testing meets the intent of this position. The testing will be performed while active construction is in progress on the project but sufficiently complete to assure that the installed HEPA filters and charcoal are not subjected to airflow that would invalidate inplace testing.

TABLE 6.5-3 (Continued)

<u>Regulatory Position</u>	<u>System Design Feature</u>
6.a	Testing procedure will operate the train for at least 15 minutes each month. The heaters will not be tested due to revised charcoal adsorber testing requirements consistent with ventilation systems without heaters.
6.b	Testing conforms to this position.
6.c	Testing procedures will meet the intent of this position. An exception has been made to use ANSI N510-1980 in lieu of ASME N511-2007 to align with Plant Technical Specifications.
6.d	Testing conforms to this position.
6.e	Testing conforms to this position.
6.f	An exception has been taken to this position. None of the accident analyses credit iodine removal by charcoal adsorbers, thus there is no need for testing.
6.g	An exception has been taken to this position. None of the accident analyses credit iodine removal by charcoal adsorbers, thus there is no need for testing.
6.h	An exception has been taken to this position. None of the accident analyses credit iodine removal by charcoal adsorbers, thus there is no need for testing.
6.i	Testing conforms to this position.

TABLE 6.5-3 (Continued)

<u>Regulatory Position</u>	<u>System Design Feature</u>
7.a	Testing meets the intent of this position.
7.b	Sampling and analysis is performed at least once per 18 months, or following painting, fire or chemical release in any ventilation zone communicating with the subsystem. Laboratory testing is performed in accordance with ASTM-D3803-1989.
7.c	Testing procedures meet the intent of this position. Acceptance Criteria meets or exceeds the standards in <Regulatory Guide 1.52> Table 2. Laboratory testing is performed in accordance with ASTM-D3803-1989.
7.d	Testing conforms to this position.

TABLE 6.5-3 (Continued)

Regulatory Position

System Design Feature

NOTE:

- (1) As discussed in <Chapter 15>, none of the design basis events credit any removal of elemental and organic iodines by the charcoal filters in the AEGTS.
- (2) Regulatory Positions 1.a through 4.e evaluate System Design Features against Revision 2 of <Regulatory Guide 1.52>, which detailed Environmental Design Criteria, System Design Criteria, Component Design Criteria and Qualification Testing, and Maintenance. Regulatory Positions 6.a through 7.d evaluate System Design Features against Revision 4 of <Regulatory Guide 1.52>, which detail In-Place Testing Criteria and Laboratory Testing Criteria for Activated Carbon. Revision 4 of <Regulatory Guide 1.52> has been implemented by License Amendment 180.

TABLE 6.5-4

CONTROL ROOM EMERGENCY RECIRCULATION SYSTEM
MATERIALS LIST (DESIGN DATA)

Filter Unit Housing

Number of filter units	2 for Unit 1 & Unit 2
Manufacturer	CVI-Pennwalt
Filter holding frame	Stainless steel, ASTM A 479, Type 304

Demisters

Number, per filter unit	21
Manufacturer & model no.	ACS, 101-55
General standards	MSAR-71-45
Frame Material	Stainless steel, ASTM A 479, Type 304
Media	Stainless steel and fiberglass woven mesh

Roughing Filters

Number, per filter unit	21
Manufacturer	Flanders (or equal)
Model number	00A-0-02-03NL
General standards	UL 900 Class 1; UL 586
Frame material	14-gauge Type 409 Stainless Steel
Filter material	95 percent boron silicate fiberglass, 5 percent organic material (MIL-F-51079B)

TABLE 6.5-4 (Continued)

Adhesives	Fire retardant polyurethane and rubber based adhesives
Gaskets	Sponge Neoprene (SCE-43)
Weight, per filter, lbs	
Steel, approx.	25
Glass and Miscellaneous Material, approx.	10
Total	35
<u>HEPA Filters Upstream and Downstream</u>	
Number, per filter unit	42
Manufacturer	Flanders (or equal)
Model number	007-0-02-03NU
General standards	MIL-F-51068D; UL 900 Class 1; UL 586
Frame material	14-gauge Type 409 Stainless Steel
Filter material	95 percent boron silicate fiberglass, 5 percent organic material (MIL-F-51079)
Adhesives	Fire retardant solid urethane
Gaskets	Sponge Neoprene (SCE-43) (ASTM D1056)
Weight, per filter, lb	
Steel, approx.	25
Glass and miscellaneous material, approx.	15
Total	40

TABLE 6.5-4 (Continued)

Activated Charcoal Adsorber

Manufacturer	CVI-Pennwalt
Type of media	New, activated coconut shell charcoal
Impregnant	Potassium iodide (KI) and elemental iodine type
Weight of carbon, lb	5,700
Adsorber enclosure	Stainless steel, ASTM A240, Type 304

Electric Heating Coil

Manufacturer	CVI-Pennwalt
Frame material	Stainless steel
Heating element	Stainless steel sheathed elements

TABLE 6.5-5

FUEL HANDLING AREA EXHAUST SUBSYSTEM
MATERIALS LIST (DESIGN DATA)

Filter Unit Housing

Number of filter units	3
Manufacturer	CVI-Pennwalt
Filter holding frame	Stainless steel, ASTM A479, Type 304

Demisters

Number per filter unit	12
Manufacturer & model no.	ACS, 101-55
General standards	MSAR-71-45
Frame material	Stainless steel, ASTM A479, Type 304
Media	Stainless steel and fiberglass woven mesh

Roughing Filters

Number, per filter unit	12
Manufacturer	Flanders (or equal)
Model number	00A-0-02-03NL
General standards	UL 900 Class 1; UL 586
Frame material	14-gauge Type 409 Stainless Steel
Filter material	95 percent boron silicate fiberglass, 5 percent organic material
Adhesives	Fire retardant polyurethane and rubber based adhesives

TABLE 6.5-5 (Continued)

Gaskets	Sponge Neoprene (SCE-43)
Weight, per filter, lbs	
Steel, approx.	25
Glass and miscellaneous material, approx.	10
Total	35
<u>HEPA Filters Upstream and Downstream</u>	
Number, per filter unit	24
Manufacturer	Flanders (or equal)
Model number	007-0-02-03NU
General standards	MIL-F-51068D; UL 900 Class 1; UL 586
Frame material	14-gauge Type 409 Stainless Steel
Filter material	95 percent boron silicate fiberglass, 5 percent organic material (MIL-F-51079)
Adhesives	Fire retardant solid urethane
Gaskets	Sponge Neoprene (SCE-43) (ASTM D1056)
Weight, per filter, lb	
Steel, approx.	25
Glass and miscellaneous material, approx.	15
Total	40

TABLE 6.5-5 (Continued)

Activated Charcoal Adsorber

Manufacturer	CVI-Pennwalt
Type of media	New, activated coconut shell charcoal
Impregnant	Potassium iodide (KI) and elemental iodine type
Weight of carbon, lb	3,040
Adsorber enclosure	Stainless steel, ASTM A240, Type 304

Electric Heating Coil

Manufacturer	CVI-Pennwalt
Frame material	Stainless steel
Heating element	Inconel steel sheathed elements

TABLE 6.5-6

ANNULUS EXHAUST GAS TREATMENT SYSTEM
MATERIALS LIST (DESIGN DATA)

Filter Unit Housing

Number of filter units	2
Manufacturer	CVI-Pennwalt
Filter holding frame	Stainless steel, ASTM A479, Type 304

Demisters

Number, per filter unit	2
Manufacturer & model no.	ACS, 101-55
General standards	MSAR-71-45
Frame material	Stainless steel, ASTM A479, Type 304
Media	Stainless steel and fiberglass woven mesh

Roughing Filters

Number, per filter unit	2
Manufacturer	Flanders (or equal)
Model number	00A-0-02-03NL
General standards	UL 900 Class 1; UL 586
Frame material	14-gauge Type 409 Stainless Steel
Filter material	95 percent boron silicate fiberglass, 5 percent organic material

TABLE 6.5-6 (Continued)

Adhesives	Fire retardant polyurethane and rubber based adhesives
Gaskets	Sponge Neoprene (SCE-43)
Weight, per filter, lbs	
Steel, approx.	25
Glass and miscellaneous material, approx.	10
Total	35
<u>HEPA Filters Upstream and Downstream</u>	
Number, per filter unit	4
Manufacturer	Flanders (or equal)
Model number	007-0-02-03NU
General standards	MIL-F-51068D; UL 900 Class 1; UL 586
Frame material	14-gauge Type 409 Stainless Steel
Filter material	95 percent boron silicate fiberglass, 5 percent organic material (MIL-F-51079)
Adhesives	Fire retardant solid urethane
Gaskets	Sponge Neoprene (SCE-43) (ASTM D1056)
Weight, per filter, lb	
Steel, approx.	25
Glass and miscellaneous material, approx.	15
Total	40

TABLE 6.5-6 (Continued)

Activated Charcoal Adsorber

Manufacturer	CVI-Pennwalt
Type of Media	New, activated coconut shell charcoal
Impregnant	Potassium iodide (KI) and elemental iodine type
Weight of carbon, lb	804
Adsorber enclosure	Stainless steel, ASTM A240, Type 304

Electric Heating Coil

Manufacturer	CVI-Pennwalt
Frame material	Stainless steel
Heating element	Inconel steel sheathed elements

TABLE 6.5-7

PRIMARY CONTAINMENT OPERATION FOLLOWING A DESIGN BASIS ACCIDENT

Type of structure	Steel cylinder with ellipsoidal dome secured to a steel lined reinforced concrete foundation mat.
Internal fission product removal systems	None
Free volume of primary containment, ft ³	1.16 x 10 ⁶ (excluding drywell)
Hydrogen purge system operation	<Section 6.2.5>
Containment leakage rate, Vol%/day	0.20
Effectiveness of internal fission product removal systems	Not applicable

TABLE 6.5-8

DESIGN DATA FOR ANNULUS EXHAUST GAS TREATMENT SYSTEM COMPONENTSPlenums (M15-D001A, 1B)

No. of plenums	2
Manufacturer	CVI Corporation

Demisters

Manufacturer and model	ACS, 101-55
No. of banks per plenum	1
No. of cells per bank	2
Material	Stainless steel mesh
Rated flow per cell, cfm	1,600
Max. resistance at rated flow, in. w.g.	0.97 (clean)

Roughing Filters

Manufacturer and model	Flanders, 00A-0-02-03NL (or equal)
No. of filter banks per plenum	1
No. of cells per bank	2
Rated flow per cell, cfm	2,000
Material	Glass fiber without separators
Efficiency, %	90 (per ASHRAE 52-68)
Max. resistance at rated flow, in. w.g.	0.49 (clean)

TABLE 6.5-8 (Continued)

HEPA Filters

Manufacturer and model	Flanders, 007-0-02-03NU (or equal)
No. of filter banks per plenum	2
No. of cells per bank	2
Rated flow per cell, cfm	1,500
Material	Glass fiber without separators
Efficiency, %	99.97 (on particles 0.3 micron or larger)
Max. resistance at rated flow, in w.g.	1.2 (clean)

Charcoal Filters

Manufacturer and Model	CVI, HECA module
No. of charcoal beds per plenum	2 (4 inches thick)
Rated flow per bed, cfm	1,000
Material	Activated coconut charcoal impregnated with KI
Efficiency, %	99.9 on elemental Iodine 97 on methyl iodide at 86°F and 95 percent RH
Maximum resistance at rated flow, in w.g.	2.5 (clean)

Heating Coil

Manufacturer	CVI-Pennwalt
No. of coils per plenum	1

TABLE 6.5-8 (Continued)

Heating Coil (Continued)

Heating capacity per coil, kW	20
Electrical characteristics	480V, 3 phase, 60 Hertz

Fans (M15-C001A, 1B)

No. required	2
Manufacturer	Westinghouse
Fan type	Centrifugal SISW
Fan size (wheel diameter), in.	15-1/2
Arrangement	No. 8
Discharge position	Upblast
Air quantity required per fan, cfm	2,000
Static pressure required, in w.g.	11
Fan motor horsepower	15
Motor electrical characteristics	460V, 3 phase, 60 Hertz

TABLE 6.5-9

INPUT PARAMETERS FOR THE SPRAY REMOVAL ANALYSIS

Containment (excluding drywell) net free volume, ft ³	1.1654 x 10 ⁶
Sprayed containment volume, F ³	481174
Unsprayed containment volume, F ³	684226
Mean spray fall height, ft	54.05
Number of spray pumps operating	1
Spray flow rate, gpm	5,250
Spray solution pH	7.0
Q, Spray Flux, cfm/ft ²	0.0621
Alpha, unsprayed/sprayed volume	1.422
Pct, uncertainty percentile	10

<TABLE 6.5-10>

DELETED

TABLE 6.5-11

ELEMENTAL IODINE DEPOSITION REMOVAL FACTORS

	Volume (ft ³)	Wall Area (ft ²)	Removal Factor (hr ⁻¹)
Drywell	276,500	15,000	0.878
Sprayed Containment	481,174	29,000	0.975
Unsprayed Containment	684,226	61,000	1.443

6.6 INSERVICE INSPECTION OF CLASS 2 AND 3 COMPONENTS

6.6.1 COMPONENTS SUBJECT TO EXAMINATION

All Quality Group B and Quality Group C components will be examined in accordance with Section XI of the ASME Code. The edition and addenda will be in accordance with <10 CFR 50.55a> as indicated in the Inservice Examination Program.

The Inservice Examination Program covers Class 2 and 3 systems and components as described in Section XI. Exceptions for those portions of systems that cannot be examined to fully meet the requirements of Section XI, if any, are fully identified and the reasons for the exceptions given in the program. The program also defines a schedule for examinations.

6.6.2 ACCESSIBILITY

The design and arrangement of Class 2 system components provide adequate clearances to conduct the required examinations at the code required inspection interval, and the design and arrangement of Class 3 system components also provides adequate clearances.

6.6.3 EXAMINATION TECHNIQUES AND PROCEDURES

The Inservice Examination Program describes the scope of the examinations and includes isometric drawings and component sketches. The drawings show weld locations in the various piping systems and on components. Boundary diagrams and classification tables are incorporated into the program to delineate systems boundaries. The program specifies the type of examinations to be performed and the total extent of the examination coverage for each system component.

Detailed procedures for volumetric (ultrasonic), surface penetrant and visual examinations are used in support of the program. Accompanying drawings include diagrams of calibration blocks and unique designations for each block to be used in the examination procedure.

6.6.4 INSPECTION INTERVALS

An inspection schedule for Class 2 system components is developed in accordance with the guidance of Section XI, Subarticle IWC-2400, and a schedule for Class 3 system components is developed according to Subarticle IWD-2400.

6.6.5 EXAMINATION CATEGORIES AND REQUIREMENTS

The inservice examination categories and requirements for Class 2 components are in agreement with Section XI, Subarticle IWC-2500. Inservice examination categories and requirements for Class 3 components are in agreement with Section XI, Subarticle IWD-2500.

6.6.6 EVALUATION OF EXAMINATION RESULTS

The evaluation of Class 2 component examination results will comply with the requirements of Article IWC-3000 of Section XI. The method to be used in the evaluation of examination results for Class 3 components will comply with the requirements of Article IWD-3000 of Section XI. The repair procedures for Class 2 components will comply with the requirements of Article IWA-4000 of Section XI. The procedures to be used for repair of Class 3 components will be in agreement with Article IWA-4000 of Section XI.

6.6.7 SYSTEM PRESSURE TESTS

The program for Class 2 system pressure testing will comply with the criteria of Code Section XI, Article IWC-5000. The program for Class 3 system pressure tests will comply with the criteria of Article IWD-5000.

6.6.8 AUGMENTED INSERVICE INSPECTION TO PROTECT AGAINST POSTULATED PIPING FAILURES

An Augmented Inservice Examination Program for high energy piping in containment penetration break exclusion regions <Section 3.6.2.1.7> is included in the Inservice Examination Program. It follows the same general outline as the Inservice Examination Program and includes areas subject to examination, method of examination, and extent and frequency of examinations.

6.7 MAIN STEAM LINE ISOLATION VALVE LEAKAGE CONTROL SYSTEM

The main steam line isolation valve leakage control system (MSIVLCS) has been eliminated and is abandoned in place.

6.8 SAFETY-RELATED INSTRUMENT AIR SYSTEM

6.8.1 DESIGN BASES

The function of the safety-related instrument air system is to continuously supply clean, dry, oil-free air for the initial charge and recharging of the automatic depressurization system (ADS) safety relief valve accumulators when the depressurization function of the safety relief valves is used. The "B" Train safety-related instrument air system also provides postaccident makeup to the outboard MSIV air accumulators. Air receiver tanks are sized by volume to provide a sufficient quantity of air for recharging the ADS accumulators, outboard MSIV accumulators and the low-low set relief valve accumulator under accident conditions. In addition, the tanks contain a sufficient volume of air to provide makeup for system leakage for a period of seven days after an accident occurs. After this initial seven-day period, the system can be recharged with the air compressors or commercially available compressed air bottles, thus assuring the required 100 days of operation of the ADS valves and a continued pneumatic source for the outboard MSIVs. As long as the ADS valves remain open the reactor will not repressurize and low pressure ECCS pumps will be able to maintain cooling flow to the reactor.

6.8.2 SYSTEM DESIGN

The safety-related instrument air system is shown on <Figure 6.8-1>. The system is designed to provide clean, dry air continuously at 160 to 170 psig to the ADS and low-low set relief valve accumulators and at an approximate setting of 85 psig (45 psig minimum) for the outboard MSIV accumulators. The system stores air at 160 to 170 psig in receiver tanks downstream of the purifier package.

One reciprocating type air compressor and air dryer is provided. The compressor supplies nominal 21.3 scfm and is automatically operated by a pressure switch on the compressor. The compressor automatically starts and stops to maintain a tank pressure of approximately 160 to 170 psig. The compressor unloads automatically after it shuts off.

The system has a connection for recharging the receiver tanks or safety relief valve accumulators directly to supplement compressor recharging in emergency and normal situations when the compressor is out-of-service.

One large, low pressure air receiver tank is provided for safety-related air storage in each of two supply lines for the ADS and low-low set relief valve accumulators.

The "B" Train safety-related instrument air system also provides post-accident makeup to the outboard MSIV air accumulators.

Two small air receiver tanks are also provided in each supply line to supply nonsafety air storage to minimize compressor cycling. The six tanks are of welded steel construction and are designed in accordance with ASME Section III, Division 1, of the Boiler and Pressure Vessel Code. However, only the large air receiver tanks are designated as Safety Class 3; the small air receiver tanks are designated as Nonsafety Class.

Air receiver tank pressure in the large air receiver tanks is sensed and transmitted to the control room. An alarm is sounded in the control room if the receiver tank pressure in the large air receiver tanks decreases to 155 psig.

Each large tank is of the vertical, cylindrical type, measuring 10 feet in diameter and 24 feet in height. The smaller tanks are 18 inches in diameter and six feet in height. The tanks are equipped with relief and drain valves. The relief valves are set to relieve at 180 psig for the larger tank and 2,750 psig for the smaller tanks.

Motor-operated containment and drywell isolation valves are also provided for the safety-related instrument air system.

6.8.3 DESIGN EVALUATION

The air system is Safety Class 2 and 3, except for the section upstream of the dual isolation check valves which are upstream of the large safety-related air receiver tanks; this section is nonsafety <Figure 6.8-1>.

The system has one compressor for supplying air to the air receiver tanks, which in turn supply air to the ADS, outboard MSIVs and low-low set relief valve accumulators.

Physically separate air lines are employed to distribute air at 160 to 170 psig to ADS and low-low set relief valve accumulators and at an approximate setting of 85 psig (45 psig minimum) for the outboard MSIV Accumulators. Each of the two physically separate air lines supplies eight ADS valve accumulators. The "A" Train air storage tank also supplies the low-low set relief valve accumulator. The "B" Train also supplies the outboard MSIV accumulators. Check valves are provided

between each division's large safety-related receiver tank and the compressor to assure no backflow from the receiver tanks to the compressor.

The storage capacity (1,350 cubic feet water volume) for each division's safety-related receiver tank is based on the requirements established by the General Electric Company for the operation of the safety relief valves as well as sufficient storage capacity to allow for leakage from ADS/SRV valves and the outboard MSIVs over seven days. This capacity is based on a minimum tank pressure of 150 psig.

The selection of piping and valves in the safety-related Class 2 and 3 portion of the system is based on a design pressure of 200 psig. Relief valves are provided on the air receiver tanks to ensure that the system does not exceed the design pressure. The valves are set to relieve at 180 psig. The design pressure for portions of the system not protected by a relief valve has been increased to 210 psig to address thermal overpressure concerns.

A relief valve set at 150 psig is installed downstream of pressure regulator 1P57-F002B.

6.8.4 TESTS AND INSPECTIONS

Operator action during normal operation consists of periodically checking that system pressure is maintained within the correct range, and servicing the compressor and air dryer as necessary. This includes checking the compressor and air dryer visual indicators which signal when filter replacement is necessary. When the alarm in the control room indicates low receiver tank pressure, the air compressor is manually operated and runs until the system pressure is returned to the operating range.

Scheduled checks will be made to assure that the air receiver tanks have retained their pressure integrity.

A scheduled program of testing and inspection will be maintained to ensure that all system components and their control systems are in good working condition. Air quality will be tested on a yearly basis downstream of the air dryer for dewpoint and particulate contamination. The required air quality to safety-related components supplied from this system is: zero particulates larger than 40 microns and a dewpoint less than 0°F for ADS valves and MSIVs.

All instrumentation, control and alarm devices will be tested and calibrated at regular intervals. Manufacturer recommendations will be observed for inspection, and preventive and inservice maintenance.

6.8.5 INSTRUMENTATION REQUIREMENTS

Instrumentation provided on the compressor control panel includes a high air temperature switch, and a discharge pressure switch.

Pressure indication is provided locally and in the control room. An alarm is sounded in the control room when the safety-related receiver tank pressure decreases to 155 psig.

Manual switches with status lights are provided in the control room for closing motor-operated containment and drywell isolation valves if a LOCA condition occurs. No automatic closure signal is provided.

Relief valves are provided upstream of the double check valves on the air receiver tanks and downstream of the MSIV source line pressure regulator to ensure that the system does not exceed the design pressure of each section of the system.

6.9 FEEDWATER LEAKAGE CONTROL SYSTEM

6.9.1 DESIGN BASES

- a. The feedwater leakage control system (FWLC) is designed in accordance with Seismic Category I and quality group classification requirements to comply with <Regulatory Guide 1.26> and <Regulatory Guide 1.29>. The system meets the intent of <Regulatory Guide 1.96>, where applicable <Table 3.2-1>.
- b. The FWLC system is designed with sufficient redundancy, separation, reliability, and capacity as a safety-related system consistent with the need to maintain containment integrity for as long as postulated LOCA conditions require.
- c. The FWLC system is capable of performing its intended safety function following a loss of all offsite power coincident with the postulated design basis LOCA.
- d. The FWLC system is designed with sufficient capacity and capability to prevent leakage through the feedwater lines consistent with containment integrity under the conditions associated with the postulated design basis LOCA.
- e. The FWLC system is provided with interlocks actuated from appropriately designed safety systems or circuits to prevent inadvertent system operation.
- f. The FWLC system is designed to permit testing of the operability of controls and actuating devices as well as the complete functioning of the system during plant shutdowns.

- g. The FWLC system is designed so that effects resulting from a system single active component failure will not affect the integrity of the feedwater lines or the operability of containment isolation valves.
- h. The FWLC system is protected from the effects of internally generated missiles, pipe break failures and adverse environments associated with a LOCA.
- i. The normal power supply to the feedwater penetration motor-operated gate valves is from Division 1. The licensing basis for the feedwater penetrations is that the gate valves are successfully closed by the control room operator. Power is also available to these gate valves from Division 3 (alternate power source connected per plant procedures). This improves the reliability of the penetration in the event of a total loss of both the normal and emergency AC power from Division 1. Physical and electrical separation between Division 1 and Division 3 will be maintained during normal operation by employing two features:
 - 1. Normally open, fused disconnect switches at both ends of the circuit, and
 - 2. Fuses normally stored out of the circuit.

6.9.2 SYSTEM DESCRIPTION

The FWLC consists of piping, valves and instrumentation as shown in <Figure 6.9-1>. The system components are designed to the requirements of <Table 3.2-1>, Item XLIX.

The FWLC system consists of two subsystems designed to eliminate through-line leakage in the feedwater piping by providing a positive seal for the stem, bonnet and seat of the outboard isolation valve on

each line. The Division 2 subsystem uses the residual heat removal (RHR) waterleg pump and the Division 1 subsystem uses the low pressure core spray (LPCS) waterleg pump to supply sealing water through the bonnets of the MOVs. Following closure of the MOVs, the sealing water seals the stems, bonnets and seats, and isolates the feedwater lines.

Following a LOCA, the FWLC system is manually initiated from the control room. The operator first verifies feedwater unavailability through low feedwater pressure (approximately 30 psig), then closes the outboard containment isolation (motor-operated gate) valves with the keylock switches, and opens one of the motor-operated FWLC system valves from the control room. The suppression pool sealing water from one of the waterleg pumps (or both if they are available) is routed to both MOVs.

Since the source of sealing water is the suppression pool, a 30-day water supply is ensured. Operation of the FWLC system will not affect the function of the suppression pool since the allowed valve leakage outside of the containment is very small, and inleakage to containment would eventually return to the pool when the drywell is flooded back over the weir wall.

When the FWLC system is initiated manually following a LOCA, there should be no demand for keep-fill water in the RHR and LPCS systems since these systems will be operating. Therefore, the waterleg pump should be totally dedicated to provide sealing water to the FWLC system. A single waterleg pump has the capacity to provide the necessary sealing water to the FWLC system.

The feedwater system will not be completely drained since the system will be intact and operating initially post-LOCA.

The feedwater design includes a backup flow path through a motor driven pump. When the turbine driven feed pumps lose driving steam and trip on

vessel Level 2 post-LOCA, flow is automatically diverted through the motor driven pump. The motor driven feed pump and/or the feedwater booster pumps will continue to pump water into containment post-LOCA.

The pumps will continue to operate for about 10 minutes before the feedwater booster pumps trip on low water level. During this time, no extraction heating is available and cold water from the condenser hotwell is being pumped into the vessel which cools down the feedwater and the piping. When feedwater flow is finally stopped, feedwater flashing is not expected to occur. Therefore, a significant voiding of the piping is not expected.

In the case where the feedwater lines do not remain completely water filled, the feedwater system can be operated to ensure positive pressurization up to the motor-operated gate valve, and thus leakage is into the reactor vessel (or drywell if the LOCA is a feedwater line break inside the drywell). The FWLC system will be initiated and begin to fill the volume in the bonnet and seat of the motor-operated gate valve. When the volumes fill up, the pressure will increase and the water seal will be established.

If a divisional failure is assumed, the redundant waterleg pump would still be available for providing a water seal on the feedwater line. Under these conditions, calculations show it would take less than 9 minutes to fill and maintain a feedwater water seal.

Based on the conservative assumptions listed above, the FWLC system will provide an adequate seal within approximately one hour following a LOCA. If a loss of offsite power is assumed at this time, one or both of the FWLC subsystems will maintain the volume of water on the stem, bonnet and seat of the outboard motor-operated gate valves, thus maintaining the feedwater line isolation. During this first hour, operation of the feedwater system will maintain a system pressure higher than the drywell pressure, thus ensuring water leakage into the vessel.

In the event that the feedwater system becomes inoperable during the rapid vessel depressurization following a LOCA, the water in the feedwater piping will begin to flash into the drywell. It is expected that a water seal would remain for a sufficient length of time following the accident until the operator remotely isolates the motor-operated valve. Thus, a water seal would exist in the piping beyond (outboard) of the motor-operated valve. Initiation of the FWLCS to the bonnet, stem and seats of the motor-operated valve will then provide the water seal for the remainder of the 30 days.

6.9.3 DESIGN EVALUATION

The FWLC system is designed to prevent the release of radioactivity through the feedwater line isolation valves by providing a supplemental water seal following a loss of all offsite power coincident with the postulated design basis loss-of-coolant accident. The two redundant subsystems are physically separated to the maximum amount practical in order to minimize the exposure of the system components to missiles and to the effects of pipe whip or jet impingement from high energy line breaks. The common piping segments are physically protected from these postulated effects.

The FWLC system is Seismic Category I and is capable of performing its intended function following an active component failure. Each subsystem is powered from a different division of the ESF power supply. The possibility of a single active failure of one of the feedwater system motor-operated valves (which are a part of the feedwater system rather than the Feedwater Leakage Control System) is addressed in <Table 6.9-1>.

Double series isolation valves are provided in the FWLCS supply lines to ensure that no single active failure will affect the integrity of the feedwater lines.

Non-seismic systems and components in the area of the FWLC system have been analyzed for the effects of their failure. Additional supports or protection by barriers is provided to assure the FWLC system is not jeopardized by non-seismic failures during an earthquake. A single failure analysis of the FWLC system is contained in <Table 6.9-1>.

6.9.4 TESTS AND INSPECTIONS

The FWLC system is hydrostatically tested prior to startup. The complete functioning of the system, including operability of controls and actuating devices, can be tested during periods of plant shutdown, but in no case at intervals greater than two years.

6.9.5 INSTRUMENTATION REQUIREMENTS

Each FWLC subsystem is manually initiated from the control room following a postulated LOCA. Independent pressure instrumentation is provided for each FWLC subsystem in order to prevent operation while the feedwater lines are pressurized. Division 1 of the FWLC system is interlocked to preclude initiation unless the motor-operated shutoff valves are closed. The operation of Division 2 is controlled by plant procedures/instructions.

TABLE 6.9-1

SINGLE FAILURE ANALYSIS OF FEEDWATER LEAKAGE CONTROL SYSTEM

<u>COMPONENT/EQUIPMENT</u>	<u>MALFUNCTION</u>	<u>CONSEQUENCES</u>
RHR waterleg pump LPCS waterleg pump	Either pump fails to operate	One subsystem is inoperative. System requirements met by redundant pump and associated subsystem.
Motor-operated valve on either waterleg pump discharge line	Either valve fails to open	One subsystem is inoperative. System requirements met by redundant subsystem.
Division 1 AC electrical power	Total loss of both the normal and emergency Division 1 AC electrical power sources	Division 3 electrical power is made available to the feedwater system motor-operated gate valves by manual operator action per plant procedure. The Division 2 FWLC subsystem is then utilized to provide the water seal on the MOV.
Feedwater shutoff valve	Feedwater shutoff valve fails to close	The Feedwater Leakage Control System is vulnerable to this extremely unlikely event. However, the licensing basis for the feedwater penetrations is that a water seal in the feedwater piping outboard of the shutoff valves would remain for a sufficient length of time following the accident until the control room operator successfully isolates the motor-operated valves. Therefore,

TABLE 6.9-1 (Continued)

<u>COMPONENT/EQUIPMENT</u>	<u>MALFUNCTION</u>	<u>CONSEQUENCES</u>
		these valves are assumed to work. Also, each feedwater line contains two check valves which are classified as containment isolation valves on this penetration, and they provide a redundant containment isolation function. The proper closure of the check valves is verified per the Inservice Testing Program.