LaSalle UNITS 1 AND 2

UFSAR, REVISION 23

AND

FIRE PROTECTION REPORT (FPR), REVISION 8

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DRAWINGS CITED IN THIS CHAPTER*

DRAWING*	SUBJECT
M-89	P&ID Standby Gas Treatment System, Units 1 and 2
M-94	P&ID Low Pressure Core Spray (LPCS) System, Unit 1
M-95	P&ID High Pressure Core Spray (HPCS) System, Unit 1
M-100	P&ID Control Rod Drive Hydraulic Piping System, Unit 1
M-130	P&ID Containment Combustible Gas Control System
M-140	P&ID Low Pressure Core Spray (LPCS) System, Unit 2
M-141	P&ID High Pressure Core Spray (HPCS) System, Unit 2
M-146	P&ID Control Rod Drive Hydraulic Piping System, Unit 2
M-1443	P&ID Control Room Air Conditioning System
M-1468	P&ID Refrigerant Piping Control Room HVAC System
M-3443	HVAC C&I Details Control Room Air Conditioning System

* The listed drawings are included as "General References" only; i.e., refer to the drawings to obtain additional detail or to obtain background information. These drawings are not part of the UFSAR. They are controlled by the Controlled Documents Program.

CHAPTER 6.0 - ENGINEERED SAFETY FEATURES

The engineered safety features of LaSalle County Station are those systems whose actions are essential to a safety action required to mitigate the consequences of postulated accidents. The features can be divided into five general groups as follows: containment systems, emergency core cooling systems (ECCS), habitability systems, fission product removal and control systems and other systems. The LSCS engineered safety features, listed by their appropriate general grouping, are given below:

GROUP	SYSTEM
Containment Systems	
	Primary Containment
	Secondary Containment
	Containment Heat Removal System
	Combustible Gas Control System
	Containment Isolation System
Emergency Core Cooling System	
	High-Pressure Core Spray System (HPCS)
	Low-Pressure Core Spray System (LPCS)
	Low-Pressure Coolant Injection System (LPCI)
	Automatic Depressurization System (ADS)
Habitability Systems	
	Control Room HVAC
Fission Product Removal and Control Systems	
	Standby Gas Treatment System
	Emergency Make-Up Air Filter System

GROUP	SYSTEM
Other Systems	Main Steamline Isolation Valve Isolated Condenser Leakage Treatment Method

6.1 ENGINEERED SAFETY FEATURE MATERIALS

The materials utilized in the LSCS engineered safety feature systems have been selected on the basis of an engineering review and evaluation for compatibility with:

a. the normal and accident service conditions of the (engineered safety feature) ESF system,

b. the normal and accident environmental conditions associated with the ESF system,

c. the maximum expected normal and accident radiation levels to which the ESF will be subjected, and

d. other materials to preclude material interactions that could potentially impair the operation of the ESF systems.

The materials selected for the ESF systems are expected to function satisfactorily in their intended service without adverse effects on the service, performance or operation of any ESF.

6.1.1 Metallic Materials

In general, all metallic materials used in ESF systems comply with the material specifications of Section II of the ASME Boiler and Pressure Vessel Code. Pressure-retaining materials of the ESF systems comply with the stringent quality requirements of their applicable quality group classification and ASME B&PV Code, Section III classification. Adherence to these requirements assures materials of the highest quality for the ESF systems. In those cases where it is not possible to adhere to the ASME material specifications, metallic materials have been selected in compliance with other nationally recognized standards, e.g., ASTM, where practicable, or chosen in compliance with current industry practice.

6.1.1.1 Materials Selection and Fabrication

Metallic materials in ESF systems have, in general, been designed for a service life of 40 years, with due consideration of the effects of the service conditions upon the properties of the material, as required by Section III of the ASME B&PV Code, Article NC-2160.

Pressure retaining components of the ECCS have been designed with the following corrosion allowances, in compliance with the general requirement of Section III of the ASME B&PV Code, Article NC-3120:

a. Ferritic Materials

1. water service 0.08 inches

2. steam service 0.120 inches

b. Austenitic Materials 0.0024 inches

For ESF systems other than ECCS, appropriate corrosion allowances, considering the service conditions to which the material will be subjected, have been applied.

The metallic materials of the ESF systems have been evaluated for their compatibility with core and containment spray solutions. No radiolytic or pyrolytic decomposition of ESF material will occur during accident conditions, and the integrity of the containment or function of any other ESF will not be effected by the action of core or containment spray solutions.

Material specification for the principal pressure-retaining ferritic, austenitic, and nonferrous metals in each ESF component are listed in Table 6.1-1. Materials that would be exposed to the core cooling water and containment sprays in the event of a loss-of-coolant accident are identified in this table. Sensitization of austenitic stainless steel is prevented by the following actions:

a. Design specifications for austenitic stainless steel components require that the material be cleaned using halide free cleaning solutions and that special care be exercised in the fabrication, shipment, storage, and construction to avoid contaminants.

b. Design specifications call for ASME material, which is to be supplied in the solution annealed condition.

c. Design specifications prohibit the use of materials that have been exposed to sensitizing temperatures in the range of 800° F to 1500° F.

Cold-worked austenitic stainless steels with yield strengths greater than 90,000 psi are not utilized in ESF systems. Therefore, there are no compatibility problems with core cooling water or the containment sprays.

Metallic reflective thermal insulation is used exclusively inside the primary containment. Premoulded non-hydrophobic Microtherm MPS Insulation enclosed in a 24 gauge stainless steel jacket is installed on the Unit 2 RVWLIS piping, 2NB86A-3/4" and 2NB88A-3/4", and the main steam high-flow instrument piping, 2MSC6AD-3/4" inside primary containment. Premoulded non-hydrophobic Microtherm MPS insulation enclosed in

24 gauge stainless steel jacket is installed on Unit 1 RVWLIS piping 1NB09A-2", 1NB09B-1", 1NB88A-1", 1NB24A-2", and 1NB24B-1", and the main steam high-flow instrument piping, 1MSC6AK-3/4", inside primary containment. The aforementioned Microtherm Insulation is also installed on the Unit 1 main steam high-flow instrument piping, 1MSC6AK-3/4", inside primary containment.

ARMAFLEX insulation is installed on the chilled water system inside primary containment.

Outside containment, calcium silicate or an engineering approved alternative thermal insulation is utilized. Design specifications on the nonmetallic insulation require that it be in accordance with Regulatory Guide 1.36, in order to avoid the possibility of chloride induced stress corrosion cracking in austenitic stainless steel in contact with the insulation.

To avoid hot cracking (fissuring) during weld fabrication and assembly of austenitic stainless steel components of the ESF, the design specifications require the following:

a. Maximum delta ferrite content for wrought and duplex cast components is 5% - 15%.

b. Chemical analyses are performed on undiluted weld deposits, or alternately, on the wire, consumable insert, etc., to verify the delta ferrite content.

c. Delta ferrite content in weld metal is determined using magnetic measurement devices.

d. Maximum interpass temperature shall not exceed 350°F during welding.

e. Test results as discussed above are included in the qualification test report.

f. Weld materials meet the requirements of Section III.

g. Production welds are examined to verify that the specified delta-ferrite levels are met.

h. Welds not meeting these levels are unacceptable and must be removed.

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

The source of water for HPCS is the suppression pool. The LPCS and LPCI are supplied from the suppression pool only. Water quality is maintained at a high level of purity with the possible exception of potentially high soluble-iron metallic impurities. Additional discussion of the water qualities are given in Subsections 6.1.3, 9.2.7, and 9.2.11. Limited corrosion inhibitors or other additives (such as zinc and noble metals) are present in either source.

The containment spray utilizes the suppression pool as its source of supply. No radiolytic or pyrolytic decomposition of ESF materials are induced by the containment sprays. The containment sprays should not be a source of stress-corrosion cracking in austenitic stainless steel during a LOCA.

6.1.2 Organic Materials

Table 6.1-2 lists all the organic compounds that exist within the containment in significant amounts. All these materials in ESF components have been evaluated with regard to the expected service conditions, and have been found to have no adverse effects on service, performance, or operation.

The dry well liner and coated exposed metal surfaces inside containment are prime coated with an inorganic zinc compound that has been fully qualified in accordance with ANSI standards N101.2, N101.4, and N512 , with the exception of a small quantity (44 gallons) used on pipe hangers and snubber attachments and recirculating pump motors. Uncoated metal surfaces shall be evaluated for acceptability. No radiolytic or pyrolytic decomposition or interaction with other ESF materials will occur.

6.1.3 Postaccident Chemistry

The post-accident chemical environment inside the primary containment will consist of water from the suppression pool and the cycled condensate storage tank, i.e. water sources for the high pressure core spray, low pressure core spray, low pressure core injection, reactor core isolation cooling and containment spray. The suppression pool may contain trace amounts of corrosion inhibiting chemicals such as hydrogen, zinc and noble metals. Additionally, portions of the Reactor Building Closed Cooling Water (RBCCW) system and the Primary Containment Chilled Water (PCCW) system are inside the containment. Both systems contain limited

amounts of corrosion inhibitors, and have portions of their piping inside containment classified as Seismic Category 2. During a Design Basis Accident (DBA) either or both of these systems can fail and release the corrosion inhibitors to the suppression pool before isolation. Due to the limited quantity (trace amounts) of these chemicals in the secondary systems and the dilution factor as a result of a DBA, the water will be approximately neutral (pH = 7), and there will be no adverse affect to equipment, coatings or other materials during ECCS or RCIC operation.

TABLE 6.1-1
(SHEET 1 OF 5)

PRINCIPAL PRESSURE-RETAINING
MATERIAL FOR ESF COMPONENTS

I. Containment Systems	
A. Primary Containment	
1. Containment Walls	4500 psi Concrete
*2. Drywell Liner	SA-516, Grade 60
*3. Suppression Chamber Liner	SA-240, Type 304
*4. Drywell Head	SA-516, Grade 70
*5 Penetrations	
a. Drywell	
Penetration Sleeve	SA-333, Grade 1 or 6
	SA-516, Grade 70
Penetration Head Fitting	SA-516, Grade 60
	SA-240, Type 304
	SA-240, Type 316
	SA-350, Grade LF1
b. Suppression Chamber	
Penetration Sleeve	SA-240, Type 304
	SA-312, Grade TP 304
Penetration Head Fitting	SA-516, Grade 60
	SA-240, Type 304
	SA-350, Grade LF1
*6. Equipment Hatch	SA-516, Grade 70
*7. Personnel Access Hatch	
a. Drywell	SA-516, Grade 70
b. Suppression Chamber	SA-240, Type 304
*8. Suppression Vent Downcomers	SA-240, Type 304

Note: The materials of the process pipes associated with primary containment penetrations are addressed separately.

*Indicates that material may be subjected to containment spray or core cooling water in the event of a loss-of-coolant accident.

TABLE 6.1-1
(SHEET 2 OF 5)

*9. Vacuum Relief Piping	
a. Drywell to Suppression Chamber Penetration	SA-106, Grade B
b. Suppression Chamber Penetration	SA-312, Grade TP 304 (Seamless)
10. Vacuum Relief Valves	SA-105
*11. Pressure Retaining Bolts	
a. Drywell	SA-320, Grade L43
	SA-193, Grade B7
	SA-194, Grade 7
b. Suppression Chamber	SA-193, Class 2, Grade B8C, Type 347
	SA-194, Class 2, Grade 83, Type 347
B. Secondary Containment	
1. Ducts	A-526
2. Dampers	A-285, Grade B
	A-181, Grade 1
C. Containment Heat Removal System	
1. RHR Pumps	A-516, Grade 70
2. RHR Heat Exchanger	
a. Shell Side	SA-516, Grade 70
b. Tube Side	SA-249, Grade TP 304L
*3. Piping	SA-106, Grade B
*4. Valves	SA-216, Grade WCB or SA-105
*5. Pressure-Retaining Bolting	SA-193, Grade B7
*6. Welding Material	SFA-5.18E70S-3(F-6, A-1)
D. Containment Isolation System	
*1. Piping	SA-106, Grade B or SA-312, Grade TP 304

*Indicates that material may be subjected to containment spray or core cooling water in the event of a loss-of-coolant accident.

TABLE 6.1-1
(SHEET 3 OF 5)

*2. Valves	SA-216, Grade WCB or SA-105 or SA-182, Grade 316L or Grade F316 or SA-351, Grade C8FM or SA-351 Grade CF3
*3. Pressure-Retaining Bolting	SA-193, Grade B7
*4. Welding Material	SFA-5.18E70S-3 (F-6, A-1)
E. Combustible Gas Control System	
1. Piping	SA-106, Grade B
2. Valves	SA-216, Grade WCB
3. Recombiner	SA-358, Grade 304
4. Blower	
5. Pressure-Retaining Bolting	SA-193, Grade B7
6. Welding Material	SFA-5.18E70S-3 (F-6, A-1)
II. Emergency Core Cooling System	
A. High-Pressure Core Spray	
1. Pump	A-516, Grade 70
2. Piping	
*a. Inside Reactor Building	SA-106, Grade B
b. Outside Reactor Building	SA-409, Grade TP 304
*3. Valves	SA-216, Grade WCB or SA-105
*4. Pressure-Retaining Bolting	SA-193, Grade B7
*5. Welding Materials	SFA-5.18E70S-3 (F-6, A-1)
B. Low-Pressure Core Spray	
1. Pump	A-516, Grade 70
*2. Piping	SA-106, Grade B
*3. Valves	SA-216, Grade WCB or SA-105

*Indicates that material may be subjected to containment spray or core cooling water in the event of a loss-of-coolant accident.

TABLE 6.1-1
(SHEET 4 OF 5)

*4. Pressure-Retaining Bolting	SA-193, Grade B7
*5. Welding Materials	SFA-5.18E70S-3 (F-6, A-1)
A. Low-Pressure Coolant Injection	
1. RHR Pump	A-516, Grade 70
*2. Piping	SA-106, Grade B
*3. Valves	SA-216, Grade WCB or SA-105
*4. Pressure-Retaining Bolting	SA-193, Grade B7
*5. Welding Materials	SFA-5.18E70S-3 (F-6, A-1)
B Automatic Depressurization System	
*1. Piping	
a. Inlet	SA-155, Grade KCF70
b. Outlet	SA-106, Grade B
*2. Valves	
III. Habitability System	
A. Blowers	A-283, A-242
B. Dampers	A-285, Grade B
	A-181, Grade 1
C. Ducts	A-526
D. Housing	A-36
IV. Fission Product Removal and Control System	
A. Standby Gas Treatment System	
1. a. Piping (Downstream of Filter Unit)	SA-106, Grade B
b. Piping (Upstream of Filter Unit)	A-106, Grade B
2. Housing	A-36

*Indicates that material may be subjected to containment spray or core cooling water in the event of a loss-of-coolant accident.

TABLE 6.1-1
(SHEET 5 OF 5)

3. Valves	SA-216, Grade WCB or SA-105, or SA-516, Grade 7
4. Dampers	A-285, Grade B
	A-181, Grade 1
5. Blowers	A-283, A-242
6. Pressure-Retaining Bolting	
a. Pressure-Retaining Bolting (Downstream of Filter Unit)	SA-193, Grade B7
b. Pressure-Retaining Bolting (Upstream of Filter Unit)	A-193, Grade B7
7. Welding Materials	SFA-5.18E70S-3 (F-6,A-1)
B. Emergency Air Filter System	
1. Ducts	A-526
2. Dampers	A-285, Grade B
	A-181, Grade 1
3. Housing	A-36
4. Blower	A-283, A-242
V. Other Systems	
A. Main Steamline Isolation Valve Leakage Control System (Deleted)	

*Indicates that material may be subjected to containment spray or core cooling water in the event of a loss-of-coolant accident.

TABLE 6.1-2
(SHEET 1 OF 2)

ORGANIC MATERIALS WITHIN THE
PRIMARY CONTAINMENT

MATERIAL	USE	QUANTITY
Acrylomitrile Butadiene/PVC Foam Rubber	ARMAFLEX Insulation on the Chilled Water Piping	Throughout Drywell
Chlorosulfinated Polyethylene (Hypalon)	Low Voltage Electrical Power Cable Jacketing and Insulation Material	Throughout Drywell
Etylene Propylene Rubber (EPR)	Low Voltage Electrical Power Cable Jacketing and Insulation Material	Throughout Drywell
High Temperature Ethylene Propylene	Medium Voltage Electrical Power Cable Jacketing and Insulation Material	Throughout Drywell
Hypalon/Hypalon	Instrumentation Cable Insulation/Jacketing Material	Throughout Drywell
EPR/Hypalon	Instrumentation Cable Insulation/Jacketing Material	Throughout Drywell
Cross-Linked Polyolefin/Alkaneimide Polymer	Instrumentation Coaxial and Triaxial Insulation/ Jacketing Material	Throughout Drywell
Modified Phenolic	Coating for Exposed Carbon Steel Surfaces	16 ft^3
Modified Phenolic Surfacer	Coating for Exposed Concrete Surfaces	17 ft^3
Modified Phenolic Finish	Coating for Exposed Concrete Surfaces	5 ft^3

TABLE 6.1-2
(SHEET 2 OF 2)

MATERIAL	USE	QUANTITY
Alkyd Primer and Finish	Pipe hangers and Snubber Attachments and GE Recirculating Pump	44 gal.
Lube Oil	Reactor Recirculation Pump Motor (2 motors/unit)	145 gal per unit
Silicone Fluid (SF 1147, GE)	MSIV Hydraulic Fluid (4 valves within containment)	1 ½ gal. per valve
Non-separating high temperature grease	Drywell cooling area coolers	< 1 gal.
Fyrquel 220/or Fyrquel EHC (stauffer)	Recirculation Control Valve Hydraulic Fluid (2 valves)	118 gal. per valve
Silicone Fluid	Lisega Hydraulic Snubbers	< 1 ½ gal. per snubber
Fiberglass Reinforced Silicone Fabric	1 (2) RF01 and 1 (2) RE02 Sump Cover Mat	400 ft^2 per unit
Silicone Sealant	1 (2) RF01 and 1 (2) RE02 Sump Cover Mat	< 1 gal. per unit

6.2 CONTAINMENT SYSTEMS

6.2.1 Containment Functional Design

This section establishes the design bases for the primary containment structure, describes the major design features of the structure, and presents an evaluation of the capacity of the containment to perform its required safety function during all normal and postulated accident conditions described in this UFSAR.

6.2.1.1 Containment Structure

6.2.1.1.1 Design Bases

The primary containment structure has been designed to meet the following safety design bases:

a. Containment Vessel Design

   1. The containment structure has the capability to withstand the peak transient pressures and temperatures that could occur due to the postulated design-basis accident (DBA).

   2. The containment has the capability to maintain its functional integrity indefinitely after the postulated DBA.

   3. The containment structure also withstands the peak environmental transient pressures and temperatures associated with the postulated small line break inside the drywell.

   4. The containment structure has also been designed to withstand the coincident fluid jet forces associated with the flow from the postulated rupture of any pipe within the containment.

   5. The containment has also been designed to withstand the hydrodynamic forces associated with a DBA and safety-relief valve discharge, as described in the LaSalle Design Assessment Report. Design loading combinations are also described in the design assessment report: Design pressure and temperature, and the major containment design parameters are listed in Table 6.2-1.

b. Containment Subcompartment Design

   The internal structures of the containment have been designed to accommodate the peak transient pressures and temperatures

associated with the postulated design-basis accident (DBA). The effects of subcompartment pressurization for the postulated pipe ruptures have been evaluated. Subcompartment pressurization is more fully discussed in Subsection 6.2.1.2.

c. Containment Internals Design

The drywell floor has been designed to withstand a downward acting differential pressure of 25 psig in combination with the normal operating loads and safe shutdown earthquake (SSE). The drywell floor has also been designed to accommodate an upward acting deck differential pressure of 5 psig, in order to account for the wetwell pressure increase that could occur after a loss-of-coolant accident (LOCA).

d. Containment Design for Mass and Energy Release

1. The maximum postulated release of mass and energy to the containment is based upon the instantaneous circumferential rupture of a 24- inch reactor recirculation line or a 26-inch main steamline.

2. The effects of metal-water reactions and other chemical reactions following the DBA can be accommodated in the containment design.

e. Energy Removal Features

The RHR system, through the containment cooling mode, is utilized to remove energy from the containment following a LOCA by circulating the suppression pool water through a residual heat removal (RHR) heat exchanger for cooling, and returning the water to the pool through the low-pressure core injection (LPCI) in the reactor pressure vessel (RPV) or the suppression chamber spray header. The containment spray mode of the RHR system can also be utilized to condense steam and reduce the temperature in the drywell following a LOCA. A more detailed description is available in Subsection 6.2.2. The RHR containment cooling mode energy removal capability is not affected by a single failure in the system, since a completely redundant loop is available to perform this function. Two redundant loops of the containment spray system are also provided.

f. Pressure Reduction Features

The containment vent system directs the flow from postulated pipe ruptures to the pressure suppression pool, and distributes such flow uniformly throughout the pool, to condense the steam portion of the flow rapidly, and to limit the pressure differentials between the drywell and wetwell during various postaccident cooling modes.

g. Hydrostatic Loading Design

The containment design permits filling the containment system drywell with water to a level 1 foot below the refueling floor to permit removal of fuel assemblies during postaccident recovery.

h. Impact Loading Design

The containment system is protected against missiles from internal or external sources and excessive motion of pipes that could directly or indirectly jeopardize containment integrity.

i. Containment Leakage Design

The containment limits leakage during and following the postulated DBA to values less than leakage rates that would result in offsite doses greater than 10 CFR 100.

j. Containment Leakage Testability

It is possible to conduct periodical leakage tests as may be appropriate to confirm the integrity of the containment at calculated peak pressure resulting from the postulated DBA.

For the purposes of the containment structure design, the design-basis accident (DBA) is defined as a mechanical failure of the reactor primary system equivalent to the circumferential rupture of one of the recirculation lines. During the DBA, the long-term peak suppression pool temperature shall not exceed the design temperature.

6.2.1.1.2 Design Features

The primary containment is a concrete structure with the exception of the drywell head and access penetrations, which are fabricated from steel. The major components are shown in Figure 3.8-1. The concrete is designed to resist all loads associated with the design-basis accident.

The primary containment walls have a steel liner, which acts as a low leakage barrier for release of fission products.

The walls of the primary containment are posttensioned concrete; the base mat is conventional reinforced concrete. The dividing floor between the drywell and suppression chamber is conventional reinforced concrete and is supported on a cylindrical base at its center, on a series of concrete columns and from the containment wall at the periphery of the slab.

The drywell floor is rigidly connected to the primary containment wall. A full moment and shear connection is provided by dowels and shear lugs welded to the reinforced liner plate as shown in Figure 3.8-4. The thermal expansion is accounted for in the containment design; the resulting forces and moments are accommodated within the allowable stress limits.

The primary containment walls support the reactor building floor loads and, in addition, also serve as the biological shield. A detailed discussion of the structural design bases is given in Chapter 3.0. The codes, standards, and guides applied in the design of the containment structure and internal structures are identified in Chapter 3.0.

The walls of the primary containment structure are posttensioned, using the BBRV system of posttensioning utilizing parallel lay, unbonded type tendons. The tendons are fabricated from 90 one-quarter inch diameter, cold drawn, stress relieved, prestressing grade wire. Each tendon is encased in a conduit. The walls are prestressed both vertically and horizontally for floor elevations below 820 feet. The horizontal tendons are placed in a 240° system using three buttresses as anchorages with the tendons staggered so that two-thirds of the tendons at each buttress terminate at that buttress. For floor elevations above 820 feet, the horizontal tendons are placed in a 360° system using two buttresses as anchorages. Access to the tendon anchorages is maintained to allow for periodic inspection. For a typical layout of hoop tendons, see Figure 3.8-11. A typical layout of the vertical tendons is illustrated in Figure 3.8-11.

All liner joints have full penetration welds. The field welds have leaktightness testing capability by having a small steel channel section welded over each liner weld. Fittings are provided in the channel for leak testing of the liner welds under pressure. The actual containment leakage boundary during normal operation and accident conditions consists of the liner and liner joint butt welds when the leak test channel is vented to the containment atmosphere and the combined containment liner, liner joint butt welds, containment liner leak test channels, channel fillet welds and the leak test connections when the leak test channel test connection plugs are installed. The liner anchorage system considers the effects of temperature, negative pressure, prestressing, and stress transfer around penetrations.

Drywell

The drywell is a steel-lined posttensioned concrete vessel in the shape of a truncated cone having a base diameter of approximately 83 feet and a top diameter of 32 feet.

The floor of the drywell serves both as a pressure barrier between the drywell and suppression chamber and as the support structure for the reactor pedestal and downcomers. The drywell head is bolted at a steel ring girder attached to the top of the concrete containment wall and is sealed with a double seal. The double seal on the head flange provides a plenum for determining the leaktightness of the bolted connection. The base of the ring serves as the top anchorage for the vertical prestressing tendons and the top of the ring serves as anchorage for the drywell head.

The drywell houses the reactor and its associated auxiliary systems. The primary function of the drywell is to contain the effects of a design-basis recirculation line break and direct the steam released from a pipe break into the suppression chamber pool. The drywell is designed to resist the forces of an internal design pressure of 45 psig in combination with thermal, seismic, and other forces as outlined in Chapter 3.0.

The drywell is provided with a 12-foot diameter equipment hatch for removal of equipment for maintenance and an air lock for entry of personnel into the drywell. Under normal plant operations, the equipment hatch is kept sealed and is opened only when the plant is shut down for refueling and/or maintenance.

The equipment hatch is covered with a steel dished head bolted to the hatch opening frame which is welded to the steel liner. A double seal is utilized to ensure leaktightness when the hatch is subjected to either an internal or external pressure. The space between the double seal serves as a plenum for leak testing the hatch seal.

The personnel air lock is a cylindrical intake welded to the steel liner. The double doors are interlocked to maintain containment integrity during operation.

All welds that make up the vapor barrier have test channels to permit leak testing of the welds: When the leak test channel test connections are plugged, the leak test channel is part of the vapor barrier.

The primary containment ventilation system, as described in Subsection 9.4.9, is provided to maintain drywell temperatures at approximately 135° F during normal plant operation.

The primary containment vent and purge system, as described in Subsection 9.4.10, is designed to purge potentially radioactive gases from the drywell and suppression chamber prior to and during personnel access to the containment.

Containment penetration cooling is provided on high temperature penetrations through the primary containment wall by the reactor building closed cooling water system. The penetrations served by this system and the design basis for the cooling loads are described in Subsection 9.2.3.

Pressure Suppression Chamber and Vent System

The primary function of the suppression chamber is to provide a reservoir of water capable of condensing the steam flow from the drywell and collecting the noncondensable gases in the suppression chamber air space. The suppression chamber is a stainless steel-lined posttensioned concrete vessel in the shape of a cylinder, having an inside diameter of 86 feet 8 inches. The foundation mat serves as the base of the suppression chamber. The suppression chamber is designed for the same internal pressure as the drywell in combination with the thermal, seismic, and other forces. The liner design and testing are the same as covered previously within this subsection (6.2.1.1.1.2).

The entire suppression chamber is lined with stainless steel. The drywell floor support columns are also provided with a stainless steel liner on the outside surface.

Two 36-inch diameter openings are provided for access into the suppression chamber for inspection. Under normal plant operation, these access openings are kept sealed. They are opened only when the plant is shut down for refueling and/or maintenance. The access openings are located in the cylindrical walls of the chamber 14 feet 2 inches above the suppression pool water level. The access openings are closed using a bolted steel hatch cover. The hatch cover is designed with a double seal and test plenum to ensure leaktightness.

The suppression chamber vent system consists of 98 downcomer pipes open to the drywell and submerged 12 feet 4 inches below the low water level of the suppression pool, providing a flow path for uncondensed steam into the water. Each downcomer has a 23.5-inch internal diameter. The downcomers project 6 inches above the drywell floor to prevent flooding from a broken line. Each vent pipe opening is shielded by a 1-inch thick steel deflector plate to prevent overloading any single vent pipe by direct flow from a pipe break to that particular vent. The principal parameters for design of the primary containment, suppression pool, reactor building and the vent downcomers are listed in Table 6.2-1.

Vacuum Relief System

Vacuum relief valves are provided between the drywell and suppression chamber to prevent exceeding the drywell floor negative design pressure and backflooding of the suppression pool water into the drywell.

In the absence of vacuum relief valves, drywell flooding could occur following isolation of a blowdown in the drywell. Condensation of blowdown steam on the drywell walls and structures could result in a negative pressure differential between the drywell and suppression chamber.

The vacuum relief valves are designed to equalize the pressure between the drywell and wetwell air space regions so that the reverse pressure differential across the diaphragm floor will not exceed the design value of five pounds per square inch.

The vacuum relief valves (four assemblies) are outside the primary containment and form an extension of the primary containment boundary. The vacuum relief valves are mounted in special piping which connects the drywell and suppression chamber, and are evenly distributed around the suppression chamber air volume to prevent any possibility of localized pressure gradients from occurring due to geometry. In each vacuum breaker assembly, two local manual butterfly valves, one on each side of the vacuum breaker, are provided as system isolation valves should failure of the vacuum breaker occur.

The vacuum relief valves are instrumented with redundant position indication and are indicated in the main control room. The valves are provided with the capability for local manual testing. The position indication requirements for the vacuum relief valves are located in the Administrative Technical Requirements. (References 21, 22, and 23)

This design provides adequate assurance of limiting the differential pressure between the drywell and suppression chamber and assures proper valve operation and testing during normal plant operation.

No vacuum relief valves are provided between the drywell and the reactor building atmosphere. The concrete containment structure has the ability to accommodate subatmospheric pressures of approximately 5 psi absolute.

6.2.1.1.3 Design Evaluation

The key design parameters for the pressure suppression containment being provided for the LaSalle County Station (LSCS) are listed in Table 6.2-1.

These design parameters are not determined from a single accident event but from an envelope of accident conditions. As a result, there is no single design-basis accident (DBA) for this containment system.

6.2.1.1.3.1 Progression of Analysis Basis

The containment system was analyzed originally at 3434 MWt reactor power.

Subsequent analysis has been performed reflective of power uprate to 3559 MWt. However, not all analysis of the original set were performed again assuming the change in power.

Noting that recirculation line breaks produced limiting results, these were assessed at the uprated power, and cases for main steamline breaks were not re-analyzed; rather, in reporting the analysis results in subsequent sections, it is understood that instance of limiting results for compliance purposes are based on this latest 3559 MWt power and related analysis. Main steamline break results which were not replaced by subsequent re-analysis still appear. Also, in presentation of definition for loads, venting and other instances where effects of the power changes would not produce significant variance, original analysis results appear. These remain to provide context, understanding them to be presented for completeness and archival purpose, albeit on the former power level basis.

6.2.1.1.3.2 Analysis

A maximum drywell and suppression chamber pressure of 42.6 psig and 28.7 psig, respectively is predicted near the end of the blowdown phase of a loss-of-coolant accident (LOCA) transient for a hypothetical recirculation line break at rated power. Approximately the same peak pressure occurs for either the break of a recirculation line or a main steamline.

The most severe drywell temperature condition is predicted for a small primary system rupture above the reactor water level that results in the blowdown of reactor steam to the drywell. Based upon the thermodynamic conditions this would produce high temperature steam in the drywell.

In order to demonstrate that breaks smaller than the rupture of the largest primary system pipe will not exceed the containment design parameters, the blowdown phase of an intermediate size break is evaluated. Containment design conditions are not exceeded for this or the other break sizes.

All of the analyses assume that the primary system and containment are at the maximum normal operating conditions. References are provided that describe relevant experimental verification of the analytical models used to evaluate the containment response.

Table 6.2-1 provides a listing of the key design parameters of the LSCS primary containment system including the design characteristics of the drywell, suppression chamber and the pressure suppression vent system.

Table 6.2-2 provides the performance parameters of the related engineered safety feature systems which supplement the design conditions of Table 6.2-1 for containment cooling purposes during postaccident operation. Performance parameters given include those applicable to full capacity operation and to those reduced capacities employed for containment analyses.

6.2.1.1.3.3 Accident Response Analysis

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

a. an instantaneous guillotine rupture of a recirculation line,

b. an instantaneous guillotine rupture of a main steam-line,

c. an intermediate size liquid line rupture, and

d. a small size steamline rupture.

Energy release from these accidents is reported in Subsection 6.2.1.3.

The accident response analysis is based on the GE calculations. This is determined based on the containment response being dependent on the amount of energy in the system, the containment design, and the failure modes that allow the pressurization to occur rather than the fuel type. The amount of energy in the system is based on initial conditions and the assumed blowdown. As the blowdown assumed for the containment response analysis as shown in Table 6.2-18 bounds the blowdown predicted by the SPC LOCA methodology and results, less energy would be released to the containment using the SPC blowdown.

The limiting event, an instantaneous guillotine rupture of a recirculation line, was analyzed to perform the containment functional evaluation. The analysis was performed in accordance with the Generic Guidelines for General Electric Boiling Water Reactor Power Uprate, NEDC-31897P-A (Reference 24). This analysis employed essentially the same methodology as the base analysis shown in Attachment 6.C, while taking a more detailed modeling approach for the reactor vessel blowdown evaluation. The analysis results are included in Section 6.2.1.1.3.3.1. Detail of the base analysis is kept for historical reference in Attachment 6.C.

The current licensing basis analysis (Reference 31), performed at 3559 MWt (102% of 3489 MWt) bounds the MUR operating conditions at 3546 MWt (Reference 35).

6.2.1.1.3.3.1 <u>Recirculation Line Rupture</u>

The instantaneous guillotine rupture of a main recirculation line results in the maximum flow rate of primary system fluid and energy into the drywell as illustrated in Figure 6.2-1 by the diagram showing the location of a recirculation line break.

Immediately following the rupture, the flow out of both sides of the break will be limited to the maximum allowed by critical flow considerations. Figure 6.2-1 shows a schematic view of the flow paths to the break. Flow in the suction side of the recirculation pump will correspond to critical flow in the 2.565 square foot pipe cross section. Flow in the discharge side of the recirculation pump will correspond to critical flow at the ten jet pump nozzles associated with the broken loop, providing an effective break area of 0.468 ft^2. In addition, there is a 4- inch cleanup line crosstie that will add 0.080 ft^2 to the critical flow area, yielding a total of 3.113 ft^2.

<u>Assumptions for Reactor Blowdown</u>

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

a. At the time the recirculation pipe breaks, the reactor is operating at the most severe condition that maximizes the parameter of interest; that is, primary containment pressure.

b. The recirculation line is considered to be severed instantly. This results in the most rapid coolant loss and depressurization, with coolant being discharged from both ends of the break.

c. The reactor is shut down at the time of accident initiation because of void formation in the core region. Scram also occurs in less than 1 second from receipt of the high drywell pressure signal. The difference between shutdown at time zero and 1 second is negligible.

d. The vessel depressurization flow rates are calculated using Moody's critical flow model (Reference 1) assuming "liquid only" outflow, since this assumption maximizes the energy release to the containment: "Liquid only" outflow requires that all vapor formed in the RPV by bulk flashing rises to the surface rather than being entrained in the existing flow. Some of the vapor would be entrained and would significantly reduce the RPV discharge flow rates. Moody's critical flow model, which assumes annular, isentropic flow, thermodynamic flow, thermodynamic phase equilibrium, and maximized slip ratio, accurately predicts vessel outflows through small diameter orifices. However, actual rates through larger flow areas are less than the model indicates because of the effects of a near homogeneous two- phase flow pattern and phase nonequilibrium. This effect is in addition to the reduction caused by vapor entrainment, discussed previously.

e. The core decay heat and the sensible heat released in cooling the fuel to 545° F 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. By maximizing the assumed energy release rate, the RPV is maintained at nearly rated pressure for approximately 20 seconds. The high RPV pressure increases the calculated blowdown flowrates; this is conservative for containment analysis purposes. With the RPV fluid temperature remaining near 545° F, however, the calculated release of sensible energy stored below 545° F is negligible during the first 20 seconds. The sensible energy is released later, but does not affect the peak drywell pressure. The small effect of sensible energy release on the long-term suppression pool temperature is included.

f. The main steam isolation valves are assumed to start closing at 0.5 seconds after the accident. They are assumed to be fully closed in the shortest possible time of 3 seconds following closure initiation. Actually, the closure signal for the main steam isolation valves is expected to occur from low water level, so these valves may not receive a signal to close for more than 4 seconds, and the closing time could be as long as 5 seconds. By assuming rapid closure of these valves, the RPV is maintained at a high pressure, which maximizes the discharge of high energy steam and water into the primary containment: In addition, the rapid closure of the main steam isolation valves cuts off motive power to the steam-driven feedwater pumps.

g. Reactor feedwater flow is assumed to stop instantaneously at time zero. Since cooler feedwater flow tends to depressurize the RPV, thereby reducing the discharge of steam and water into the primary containment, this assumption is considered conservative and consistent with that of assumption f.

With respect to suppression pool temperature, this assumption has been supplemented with an additional evaluation to evaluate the suppression pool long term temperature response. For this evaluation, the feedwater is assumed to have been injected into the suppression pool, by the end of the recirculation piping break blowdown phase (at 600 seconds), in order to assess long term peak pool temperature. See paragraph entitled "Evaluation of Post-LOCA Feedwater Injection" in this section.

h. 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.

Assumptions for Containment Pressurization

The pressure response of the containment during the blowdown period of the accident is analyzed using the following assumptions:

a. Thermodynamic equilibrium exists in the drywell and suppression chamber. Since nearly complete mixing is achieved, the analysis assumes complete mixing, which is in the conservative direction.

b. The fluid flowing through the drywell-to-suppression chamber vents is formed from a homogeneous mixture of the fluid in the drywell. The use of this assumption results in complete liquid carry-over into the drywell vents.

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

d. No heat loss from the gases inside the primary containment is assumed. This adds extra conservatism to the analysis; that is, the analysis will tend to predict higher containment pressures than would actually result.

Assumptions for Long-Term Cooling

Following the blowdown period, the emergency core cooling systems (ECCS) discussed in Section 6.3 provide water for core flooding and long-term decay heat removal. The containment pressure and temperature response during this period are analyzed using the following assumptions:

a. The LPCI pumps are used to flood the core prior to 600 seconds after the accident. The high-pressure core spray (HPCS) is assumed available for the entire accident.

b. After 600 seconds, the LPCI pump flow may be diverted from the RPV to the containment spray. This is a manual operation. Actually, the containment spray need not be activated at all to keep the containment pressure below the containment design pressure. Prior to activation of the containment cooling mode (arbitrarily assumed at 600 seconds after the accident), all of the LPCI pump flow will be used only to flood the core.

c. The effect of decay energy, stored energy, and energy from the metal-water reaction on the suppression pool temperature are considered.

d. During the long-term containment response (after depressurization of the reactor vessel is complete) the suppression pool is assumed to be the only heat sink in the containment system.

e. After approximately 600 seconds, the RHR heat exchangers are activated to remove energy from the containment via recirculation cooling from the suppression pool with the RHR service water systems.

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

   Case A - Offsite Power Available

   All ECCS equipment and containment spray operating.

   Case B - Loss of Offsite Power

   Minimum diesel power available for ECCS and containment spray.

   Case C - Same as Case B (except no containment spray)

Initial Conditions for Accident Analyses

Table 6.2-3 provides the initial reactor coolant system and containment conditions used in all the accident response evaluations. The tabulation includes parameters for the reactor, the drywell, the suppression chamber and the vent system. A supplementary safety evaluation has also been performed, as discussed in Section 6.2.1.8, to evaluate an increase in the initial suppression pool temperature value to 105° F.

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.

The mass and energy release sources and rates for the containment response analyses are given in Subsection 6.2.1.3.

Short-Term Accident Response

The calculated containment pressure and temperature responses for the recirculation line break are shown in Figures 6.2-2 and 6.2-3 respectively. The calculated peak drywell pressure is 42.6 psig, which is 5.3% below the containment design pressure of 45 psig.

The suppression chamber is pressurized by the carryover of noncondensables from the drywell and by heatup of the suppression pool. As the vapor formed in the drywell is condensed in the suppression pool, the temperature of the suppression chamber water approaches 150° F and the suppression chamber pressure stabilizes at approximately 30 psig. The drywell pressure stabilizes at a slightly higher pressure, the difference being equal to the downcomer submergence. During the RPV depressurization phase, most of the noncondensable gases in the drywell initially are forced into the suppression chamber. However, following the depressurization the noncondensables will redistribute between the drywell and suppression chamber via the vacuum breaker system. This redistribution takes place as pressure is decreased by the steam condensation process occurring in the drywell.

The LPCI and LPCS systems supply sufficient core cooling water to control core heatup and limit metal-water reaction to less than 0.2%. After the RPV 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 RPV, through the broken recirculation line, in the form of hot water which flows into the suppression chamber via the drywell to suppression chamber vent system. This flow, in addition to heat losses to the drywell walls, provides a heat sink for the drywell atmosphere, causes a depressurization of the containment, and redistributes the noncondensables as the steam in the drywell is condensed.

Table 6.2-8 provides the peak pressure, temperature, and time parameters for the recirculation line break as predicted for the conditions of Table 6.2-1 and in correspondence with Figures 6.2-2 and 6.2-3. The transient peak calculated drywell floor (deck) differential pressure is 24.4 psid, which is 2.4% below the design sustained differential pressure of 25 psid.

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 versus time. Figure 6.2-4 provides the representative mass flow versus time relationship through the vent system for this accident (As explained in Section 6.2.1.1.3.1, this figure is representative and is based on original analysis, determined to be less sensitive to subsequent changes). A supplementary evaluation has been performed for the addition of feedwater to the suppression pool to assess the impact on long term pool temperature. This evaluation estimates that the peak short term pool temperature will increase by an additional 15.4° F. This results in a short term pool temperature (at 600 seconds) of approximately 166° F . For further discussion, see Section 6.2.1.1.3.3.1 in the paragraph titled, “Evaluation of Post-LOCA Feedwater Injection.”

<u>Long-Term Accident Responses</u>

In order to assess the adequacy of the containment following the initial blowdown transient, an analysis was made of the long-term temperature and pressure response following the accident. The analysis assumptions are those discussed previously for the three cases of interest. The initial pressure response of the containment (the first 600 seconds after the break) is the same for each case.

<u>Case A - All ECCS Equipment Operating (with containment spray</u>)

This case assumes that offsite a-c power is available to operate all cooling systems. During the first 600 seconds following the pipe break, the high-pressure core spray (HPCS), low-pressure core spray (LPCS), and all three LPCI pumps are assumed operating. All flow is injected directly into the reactor vessel.

After 600 seconds, both RHR heat exchangers are activated to remove energy from the containment. During this mode of operation the flow from two of the LPCI pumps is routed through the RHR heat exchanger, where it is cooled before being discharged into the containment spray header.

After the initial blowdown and subsequent depressurization due to core spray and LPCI core flooding, energy addition due to core decay heat results in a gradual pressure and temperature rise in the containment. When the energy removal rate of the RHR exceeds the energy addition rate from the decay heat, the containment

pressure and temperature reach a second peak value and decrease gradually. Table 6.2-5 summarizes the cooling equipment operation, the peak containment pressure following the initial blowdown peak, and the peak suppression pool temperature.

Case B - Loss of Offsite Power (with containment spray)

This case assumes no offsite power is available following the accident with only minimum diesel power. The containment spray is operating and injecting into the drywell after 600 seconds. During this mode of operation the LPCI flow through one RHR heat exchanger is discharged into the containment spray nozzles.

A summary of this case is given in Table 6.2-5.

Case C - Loss of Offsite Power (no containment spray)

This case assumes that no offsite power is available following the accident, with only minimum diesel power. For the first 600 seconds following the accident, one HPCS and two LPCI pumps are used to cool the core. After 600 seconds the spray may be manually activated to further reduce containment pressure if desired. This analysis assumes that the spray is not activated.

After 600 seconds, one RHR heat exchanger is activated to remove energy from the containment. During this mode of operation, one of the two LPCI pumps is shut down and the service water pumps to the RHR heat exchanger are activated. The LPCI flow is cooled by the RHR heat exchanger before being discharged into the reactor vessel.

A summary of this case is given in Table 6.2-5.

When comparing the "spray" Case B with the "no spray" Case C, the same duty on the RHR heat exchanger is obtained since the suppression pool temperature response is approximately the same. Thus, the same amount of energy is removed from the pool whether the exit flow from the RHR heat exchanger is injected into the reactor vessel or into the drywell as spray. However, the peak containment pressure is higher for the "no spray" case, but the pressure is still much less than the containment design pressure of 45 psig. (Subsection 6.2.2.3 describes the containment cooling mode of the RHR system.)

A supplemental evaluation has been performed for the purpose of evaluating the suppression pool long term temperature response. For this evaluation, the feedwater is assumed to have been injected into the suppression pool, by the end of the recirculation piping break blowdown phase (at time t = 600 seconds), in order to assess long term peak pool temperature. See paragraph entitled "Evaluation of Post-LOCA Feedwater Injection" in this section. Additionally, a slightly reduced RHR pump flow rate of 7200 gpm (versus 7450 gpm) has been evaluated, as discussed in Section 6.2.2.3.4. Both of these evaluations are evaluated for the DBA-LOCA in Reference 18. The results indicate an increase in the long term peak suppression pool temperature of approximately 8 F due to the feedwater injection and an approximately 1.5° F increase due to the lower RHR flow rate. The 200° F peak pool temperature given in Table 6.2-5 is not exceeded. Plant specific safety evaluations have been performed and have concluded that the existing DBA-LOCA analyses referenced above bounds these effects on the containment response.

Energy Balance During Accident

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

a. blowdown energy release rates,

b. decay heat rate and fuel relaxation energy,

c. sensible heat rate,

d. pump heat rate, and

e. heat removal rate from suppression pool.

Items a, b, and c are provided in Subsection 6.2.1.3. The pump heat rate value that has been used in the evaluation of the containment response to a LOCA for Case A is 4881 Btu/sec. A complete energy balance for the recirculation line break accident is given in Table 6.2-6 for the reactor system, the containment, and the containment cooling systems at time zero, at the time of peak drywell pressure, at the end of reactor blowdown, and at the time of the long-term second peak pressure reached in the containment.

The energy and mass balance have been annotated to include the effects of feedwater coastdown/injection on the long term peak suppression pool temperature. See paragraph entitled "Evaluation of Post-LOCA Feedwater Injection" in this section and footnote in Table 6.2-6.

Chronology of Accident Events

The complete description of the containment response to the design-basis recirculation line break has been given above. A chronological sequence of events for this accident from time zero is provided in Table 6.2-7.

The original and 1988 General Electric containment analysis (references 8 & 17), assumed feedwater flow stopped at the initiation of the LOCA. This assumption is conservative for an assessment on the peak cladding temperature (PCT) or containment pressure and temperature response. However, in order to make a more conservative analysis on the suppression pool predicted temperatures, the feedwater energy due to feedwater pump coastdown, or depressurization and resulting feedwater liquid carryover to the pool, should be taken into account in the suppression pool energy balance. A supplementary evaluation was performed to assess the impact on peak suppression pool temperature due to the addition of energy from the feedwater system. (Reference 18)

For this evaluation, the feedwater mass downstream of the 2nd Low Pressure Feedwater Heater is injected into the vessel. The feedwater upstream of this feedwater heater is at a temperature less than 212° F and would not be expected to be injected into the vessel during a DBA-LOCA. The mechanism for FW injection into the vessel during a LOCA with loss of onsite power is flashing of feedwater liquid when the vessel drops below the saturation pressure corresponding to the feedwater liquid temperature. Thus, only feedwater initially at a temperature above 212° F is assumed to flash and be injected into the vessel. This is conservative since vessel pressures are expected to remain higher than atmospheric pressure during the period when the peak pool temperature occurs. The latest revision of plant piping drawings were used as input to determine the feedwater volume.

Additionally, the sensible energy in the feedwater system metal is also added to the feedwater liquid injected into the vessel. It is conservatively assumed that the feedwater flowing into the vessel and coming into contact with hotter feedwater piping metal downstream, will instantaneously achieve thermal equilibrium with the hotter feedwater system metal. This maximizes the metal sensible energy transfer to the feedwater.

For the analysis, all feedwater mass and energy is injected to the vessel and subsequently transferred to the suppression pool by 600 seconds into the LOCA event. This is modeled by adding all the feedwater mass and energy input at time t = 600 seconds. Based on this previous discussion, this analysis provides a conservative estimate of the amount of energy addition to the pool due to feedwater injection.

The results indicate an increase in the long term peak suppression pool temperature of approximately 8° F (Reference 18). The 200° F peak pool temperature given in Table 6.2-5 is not exceeded.

Additional Analysis Basis Features

The bases for the analysis demonstrating current compliance for an instantaneous guillotine rupture of a recirculation line (Reference 25) employs the methodology as described in the foregoing sections of the UFSAR, generally, and also applies updated input and methods improvements in certain areas. For the short-term containment response analysis, the blowdown calculation was performed using the LAMB break flow model (Reference 26) for the recirculation line breaks, with flow rate and enthalpy determined at the current initial reactor power of 3559 MWt and initial pressure of 1025 psig. For the limiting long-term containment response, Case C, the basis analysis similarly has been performed at 3559 MWt, assuming the same availability of ECCS pumps and RHR heat exchangers as represented. The core decay heat is updated, based on the ANSI/ANS 5.1- 1979 decay heat model with a two-sigma uncertainty adder. (The decay heat calculations also include contributions from miscellaneous actinides and activation products consistent with the recommendation of GE SIL 636). Also, feedwater flow is modeled to be injected until all feedwater above 212°F is removed from the line to the RPV to maximize pool heat-up.

These results are presented for the limiting recirculation line break cases in Tables 6.2-5, 6.2-8, 6.2-18, 6.2-20, and are reflected in Tables showing input for the recirculation line breaks. Figures 6.2-2, 6.2-3, 6.2-5a, 6.2-5b, 6.2-6a, 6.2-6b, 6.2-7a, and 6.2-7b present the response for the recirculation line breaks incorporating these features.

An additional observation has been made and addressed with regard to the limiting single failure condition for the analysis of containment response assuming recirculation line breaks. A GE Safety Communication (SC06-01) was issued identifying an alternate single failure that can be more limiting with respect to peak suppression pool temperature during the Design Basis LOCA than reported in the existing license basis analysis. The licensing basis analysis (Reference 31) described here assumes the single failure of an emergency diesel generator. This implies one residual heat removal (RHR) division is lost; minimum emergency core cooling and RHR containment cooling pumps would be available, comparable to conditions as assumed for Case C, described above, with minimum suppression pool cooling. However, this assumed failure also minimizes the pump heat to the suppression pool. GE SC06-01 notes a potential alternate worst-case accident scenario with respect to suppression pool temperature may exist where the postulated single failure results in loss of one RHR division, but with all ECCS pumps remaining available. In this configuration, the pump heat to the suppression pool is maximized and can result in a higher peak suppression pool temperature. A supplemental analysis was performed in Reference 33 that determines the impact of the concerns of GE SC06-01. The analysis

uses the same inputs and assumptions in Reference 31 except for the following significant differences:

1. All ECCS pumps are assumed to be available and operate in accordance with their design requirements for reactor vessel coolant make-up.

2. A single active failure is assumed, which results in only one RHR heat exchanger being operable for containment cooling for the duration of the event.

3. A RHR service water temperature of 107°F and a RHR heat exchanger K-factor of 438 Btu/sec-°F have been evaluated in Reference 36.

The resultant maximum long-term post DBA-LOCA suppression pool temperature is 197°F. This result is favorable to the analysis result for recirculation line breaks described above.

Revised Pool Swell Analysis

Pool swell response profiles for a DBA LOCA have been reanalyzed (Reference 37) with changes in methodologies. The revised analysis also supersedes the information in section 3.3.1.3 of the LSCS Mark II Design Assessment Report, regarding Pool Swell Analytical Model (PSAM) that was used. This is applicable to both units. The approved methodology changes (Reference 39) are as follows:

1. The DBA LOCA mass and energy (M&E) release revised analysis uses GE Hitachi Nuclear Energy (GEH) TRACG computer code. During the time of interest (i.e.,O seconds to 2 seconds) following the DBA LOCA, the results show a small difference. The NRC has approved the application of TRACG for LOCA evaluations with direct applicability of the blowdown model. The analysis for M&E release covers the entire power/flow map including the effect of operation with feedwater temperature reduction (FWTR).

2. The revised analysis initially uses air flow and transitions to a realistic air/steam mixture for simulation of downcomer vent flow. During a DBA LOCA, as an initial condition, the lower portion of a downcomer vent has a water column and the remaining is filled with 100 percent air. Subsequent to the discharge of the water column, the previous analysis conservatively assumed 100 percent air flow through the downcomer vent during the transient. The revised analysis (subsequent to the discharge of the water column) is based on an initial 100 percent air flow through the downcomer vent which subsequently transitions to a realistic flow of an air/steam mixture with a constant

0.39 fraction of steam and 0.61fraction of air. This mixture ratio assumed in the analysis is conservative because it has conservatively higher fraction of air than the air fraction regarding the asymmetric LOCA pool boundary loads.

3. The previous analysis for the LOCA drywell pressure response which was used as an input to the suppression pool swell analysis, accounted for the effect of the downcomer vent back pressure on the vent flow. The revised analysis for the drywell pressure does not include the effect of the downcomer vent back pressure. NUREG-0487 accepted the use of the predicted drywell pressure based on the GEH containment models for input to the suppression pool swell model without accounting for LOCA bubble formation backpressure effects on vent flow. As described in NUREG-0487, this acceptance was based on a greater calculated pool swell response with the drywell pressure prediction from the GEH containment model relative to the calculated pool swell response obtained with measured test drywell pressure. Even though the revised analysis reduces the conservatism, the NRC staff found it acceptable. In NUREG-0487, Section 111.8.3.a.6, the analysis using the NEDM-10320 methodology, which does not account for the vent back pressure, is acceptable based on the comparison and test data.

4. Applying the drywell pressure response as an input, the suppression pool swell height is calculated as a function of time. The criteria given in NUREG-0487, Supplement 1for the maximum pool swell height is greater of the following (a) 1.5 times the vent submergence; (b) the elevation corresponding to the drywell floor uplift differential pressure used for design assessment. Option (a) was calculated to be greater and was used in the revised analysis to ensure conservatism and acceptability by the NRC.

6.2.1.1.3.3.2 Main Steamline Break

The main steamline break, which is not the limiting event with respect to the containment response, was not re-analyzed at current reactor power. Results from baseline analyses are described in this subsection, for information, and confirming the non-limiting nature of this break location.

The sequence of events immediately following the rupture of a main steamline between the reactor vessel and the flow limiter has been determined. The flow on both sides of the break will accelerate to the maximum allowed by critical flow considerations. In the side adjacent to the reactor vessel, the flow will correspond to critical flow in the 2.98-ft^2 steamline cross section. Blowdown through the other side of the break can occur because the steamlines are all interconnected at a point

upstream of the turbine by the bypass header. This interconnection allows primary system fluid to flow from the three unbroken steamlines, through the header and back into the drywell via the broken line. Flow will be limited by critical flow in the 0.94-ft^2 steamline flow restrictor. The total effective flow area is thus 3.92 ft^2, which is the sum of the steamline cross-sectional area and the flow restrictor area. Subsection 6.2.1.3 provides information on the mass and energy release rates.

Immediately following the break, the total steam flow rate leaving the vessel would be approximately 12,000 lb/sec, which exceeds the steam generation rate in the core of 4,500 lb/sec. This steam flow to steam generation mismatch causes an initial depressurization of the reactor vessel at a rate of 50 psi/sec. The void formation in the reactor vessel water causes a rapid rise in the water level, and it is conservatively assumed that the water level reaches the vessel steam nozzles 1 second after the break occurs. The water level rise time of 1 second is the minimum that could occur under any reactor operating condition. From that time on, a two-phase mixture would be discharged from the break. During the first second of the blowdown, the blowdown flow will consist of saturated reactor steam. This steam will enter the containment in a super-heated condition of approximately 330°F.

Figures 6.2-8 and 6.2-9 show the pressure and temperature response of the drywell and containment during the primary system blowdown phase of the accident.

Figure 6.2-9 shows that the drywell atmosphere temperature approaches 330°F after 1 second of primary system steam blowdown. At that time, the water level in the vessel will reach the steamline nozzle elevation and the blowdown flow will change to a two-phase mixture. This increased flow causes a more rapid drywell pressure rise. However, the peak differential pressure is 24.2 psid, which occurs shortly after the vent clearing transient. As the blowdown proceeds, the primary system pressure and fluid inventory will decrease and this will result in reduced break flow rates.

As a consequence, the flow rate in the vent system also starts to decrease, and this results in a decreasing differential pressure between the drywell and containment.

Table 6.2-8 presents the peak pressures, peak temperatures, and times of this accident as compared to the recirculation line break.

Approximately 50 seconds after the start of the accident, the primary system pressure will have dropped to the drywell pressure and the blowdown will be over. At this time the drywell will contain pure steam, and the drywell and suppression chamber pressures will stabilize at approximately 30 and 25 psig, respectively; the difference corresponds to the hydrostatic pressure at the lower end of the submerged vents.

The drywell and containment will remain in this equilibrium condition until the reactor pressure vessel refloods. During this period, the emergency core cooling pumps will be injecting cooling water from the suppression pool into the reactor. This injection of water will eventually flood the reactor vessel to the level of the steamline nozzles, and at this time, the ECCS flow will spill into the drywell. The water spillage will condense the steam in the drywell and thus reduce the drywell pressure. As soon as the drywell pressure drops below the suppression chamber pressure, the drywell vacuum breakers will open and noncondensable gases from the suppression chamber will flow back into the drywell. This process will continue until the pressures in the two regions equalize and stabilize at approximately 7.5 psig.

6.2.1.1.3.3.3 Intermediate Breaks

The intermediate-size break, which is not the limiting event with respect to the containment response, was not re-analyzed at current reactor power. Results from baseline analyses are described in this subsection, for information, and confirming the non-limiting nature of this break size.

The failure of a recirculation line results in the most severe pressure loading on the drywell structure. However, as part of the containment performance evaluation, the consequences of intermediate breaks are also analyzed. This classification covers those breaks for which the blowdown will result in reactor depressurization and operation of the ECCS. This subsection describes the consequences to the containments of a 0.1-ft^2 break below the RPV water level. This break area was chosen as being representative of the intermediate size break area range. These breaks can involve either reactor steam or liquid blowdown.

Following the 0.1-ft^2 break, the drywell pressure increases at approximately 1 psi/sec. This drywell pressure transient is sufficiently slow so that the dynamic effect of the water in the vents is negligible and the vents will clear when the drywell-to-wetwell differential pressure is equal to the vent submergence pressure. For the LSCS containment design, the maximum distance between the pool surface and the bottom of the vents is 12 feet 10 inches. Thus, the water level in the vents will reach this point when the drywell-to-containment pressure differential reaches 5.2 psid.

Figures 6.2-10 and 6.2-11 show the drywell and wetwell pressure and temperature response, respectively. The ECCS response is discussed in Section 6.3. Approximately 5 seconds after the 0.1-ft^2 break occurs, air, steam, and water will start to flow from the drywell to the suppression pool; the steam will be condensed and the air will enter the wetwell free space. After 5 seconds there will be a constant pressure differential of 5.2 psid between the drywell and wetwell. The continual purging of drywell air to the suppression chamber will result in a gradual pressurization of both the wetwell and drywell to about 22 and 27 psig, respectively. Some continuing containment pressurization will occur because of the gradual pool heatup.

The ECCS will be initiated by the 0.1-ft^2 break and will provide emergency cooling of the core. The operation of these systems is such that the reactor will be depressurized in approximately 600 seconds. This will terminate the blowdown phase of the transient. The drywell will be at approximately 27 psig and the suppression chamber at approximately 22 psig.

In addition, the suppression pool temperature will be the same as following the DBA because essentially the same amount of primary system energy would be released during the blowdown. After reactor depressurization, the flow through the break will change to suppression pool water that is being injected into the RPV by the ECCS. This flow will condense the drywell steam and will eventually cause the drywell and containment pressures to equalize in the same manner as following a recirculation line rupture.

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

From this description, it can be concluded that the consequences of an intermediate size break are less severe than those from a recirculation line rupture.

6.2.1.1.3.3.4 <u>Small Size Breaks</u>

The small-size break, which is not the limiting event with respect to the containment response, was not re-analyzed at current reactor power. Results from baseline analyses are described in this subsection, for information, and confirming the non-limiting nature of this break size.

Reactor System Blowdown Considerations

This subsection discusses the containment transient associated with small primary system blowdowns. The sizes of primary system ruptures in this category are those blowdowns that will not result in reactor depressurization due either to loss of reactor coolant or automatic operation of the ECCS equipment. Following the occurrence of a break of this size, it is assumed that the reactor operators will initiate an orderly plant shutdown and depressurization of the reactor system. The thermodynamic process associated with the blowdown of primary system fluid is one of constant enthalpy. If the primary system break is below the water level, the blowdown flow will consist of reactor water. Blowdown from reactor pressure to the drywell pressure will flash approximately one-third of this water to steam and two-thirds will remain as liquid. Both phases will be at saturation conditions corresponding to the drywell pressure. Thus, if the drywell is at atmospheric pressure, the steam and liquid associated with a liquid blowdown would be at 212° F. Similarly, if the containment is assumed to be at its design pressure, the reactor coolant will blow down to approximately 293° F steam and water.

If the primary system rupture is located so that the blowdown flow consists of reactor steam only, the resultant steam temperature in the containment is significantly higher than the temperature associated with liquid blowdown. This is because the enthalpy of high-energy saturated steam is nearly twice that of saturated liquid. The higher enthalpy will result in a superheat condition. For example, decompression of 1000-psia steam to atmospheric pressure will result in 298° F superheated steam (86° F of superheat).

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

Containment Response

For drywell design consideration, the following sequence of events is assumed to occur. With the reactor and containment operating at the maximum normal conditions, a small break occurs that allows blowdown of reactor steam to the drywell. The resulting pressure increase in the drywell will lead to a high drywell pressure signal that will scram the reactor and activate the containment isolation system. The drywell pressure will continue to increase at a rate dependent upon the size of the steam leak. This pressure increase will lower the water level in the vents until the level reaches the bottom of the vents. At this time, air and steam will start to enter the suppression pool. The steam will be condensed and the air will be carried over to the suppression chamber free space. The air carry-over will result in a gradual pressurization of the containment at a rate dependent upon the size of the steam leak. Once all the drywell air is carried over to the suppression chamber, pressurization of the containment will cease and the system will reach an equilibrium condition with the drywell pressure at 27 psig and the suppression chamber at approximately 22 psig. The drywell will contain only superheated steam, and continued blowdown of reactor steam will condense in the suppression pool.

Recovery Operations

[redacted]

Drywell Design Temperature Considerations

For drywell design purposes, it is assumed that there is a blowdown of reactor steam for the 6-hour cooldown period. The corresponding design temperature is determined by finding the combination of primary system pressure and containment pressure that produces the maximum superheat temperature. Thus for design purposes, this results in a temperature condition of 340° F.

6.2.1.1.3.4 Accident Analysis Models

The short-term pressurization analytical models, assumptions, and methods used by GE to evaluate the containment response during the reactor blowdown phase of a LOCA are described in References 2 and 3.

Once the RPV blowdown phase of the LOCA is over, a fairly simple model of the drywell and suppression chamber may be used. During the long-term, post-blowdown containment cooling mode, the ECCS flow path is a closed loop and the suppression pool mass will be constant. Schematically, the cooling model loop is shown in Figure 6.2-12. Since there is no storage other than in the suppression pool (the RPV is reflooded during the blowdown phase of the accident), the mass flowrates shown in the figure are equal, thus:

$$\dot{m}_{D_O} = \dot{m}_{S_O} = \dot{m}_{eccs} \; .$$

Analytical Assumptions

The key assumptions employed in the model are as follows:

a. The drywell and suppression chamber atmosphere are both saturated (100% relative humidity).

b. The drywell atmosphere temperature is equal to the temperature of the coolant spilling from the RPV, or to the spray temperature if the sprays are activated.

c. The suppression chamber atmosphere temperature is equal to the suppression pool temperature or to the spray temperature if the sprays are activated.

d. No credit is taken for heat losses from the primary containment or to the containment internal structures.

Energy Balance Considerations

The rate of change of energy in the suppression pool, E_p, is given by:

$$\frac{d}{dt}\left(E_p\right) = \frac{d}{dt}\left(M_{w_s} \cdot h_s\right)$$

$$= h_s \cdot \frac{d}{dt}\left(M_{w_s}\right) + M_{w_s} \cdot \frac{d}{dt}\left(h_s\right).$$

Since $\frac{d}{dt}(M_{w_s}) = 0$ (because there is no storage), and for water at the conditions that will exist in the containment:

$$\frac{d}{dt}\left(h_S\right) = C_p \cdot \frac{d}{dt}\left(T_S\right)$$

where:

C_p = 1.0 for the specific heat of pool water, Btu/ lb-°F
T_s = pool temperature, °F.

The pool energy balance yields:

$$M_{w_s} \; C_p \cdot \frac{d}{dt}(T_s) = \dot{m}_{D_o} \; h_D \; \dot{m}_{s_o} \; h_s \cdot$$

This equation can be rearranged to yield:

$$\frac{d}{dt}\,(T_S) = \frac{\dot{m}_{D_O}\; h_D - m_{S_O}\; h_S}{M_{W_S}} \cdot$$

An energy balance on the RHR heat exchanger yields

$$h_c = h_s - \frac{q_{H_x}}{\dot{m}_{s_o}} \qquad (6.2\text{-}3)$$

where:

h_c = enthalphy of ECCS flow entering the reactor, Btu/lb.

Similarly, an energy balance on the RPV will yield:

$$h_D = h_c - \frac{\dot{q}_D + \dot{q}_e}{\dot{m}_{eccs}} \cdot$$

Combining Equations 6.2-1, 6.2-2, 6.2-3, and 6.2-4 gives

$$\frac{d}{dt}(T_S) = \frac{\dot{q}_D + \dot{q}_e - \dot{q}_{H_X}}{\dot{M}_{W_S}} \cdot$$

This differential equation is integrated by finite difference techniques to yield the suppression pool temperature transient.

Containment Thermodynamic Conditions

Once the energy equations are solved, the drywell and suppression chamber atmospheric temperatures can be calculated.

For the case in which no containment spray is operating, the suppression chamber temperature, T_w, at any time will be equal to the current temperature of the pool, T_s, and the drywell temperature, T_d, will be equal to the temperature of the fluid leaving the RPV. Thus:

$$T_D = T_s + \frac{\dot{q}_D + \dot{q}_e - \dot{q}_{H_X}}{\dot{m}_{eccs}}$$

and $T_w = T_s$.

For the case in which the containment spray is assumed to be operating, both the drywell and suppression chamber atmosphere will be at the spray temperature, T_{sp}

where:

$$T_{sp} = T_s - \frac{\dot{q}_{H_X}}{\dot{m}_{eccs}}$$

and, $T_D = T_w = T_{sp}$.

Using the suppression chamber and drywell atmosphere temperatures, and assumption (a) (drywell and suppression chamber saturated), it is possible to solve for the containment total pressures, since:

$$P_D = P_{a_D} + P_{v_D} \qquad (6.2\text{-}6)$$

$$P_s = P_{a_s} + P_{v_s} \qquad (6.2\text{-}7)$$

where:

P_D	=	drywell total pressure, psia,
P_{a_D}	=	partial pressure of air in drywell, psia,
P_{v_D}	=	partial pressure of water vapor in drywell, psia,
P_s	=	suppression chamber total pressure, psia,
P_{a_s}	=	partial pressure of air in the suppression chamber, psia,

P_{v_s} = partial pressure of water vapor in the suppression chamber, psia,

and, from the Ideal Gas Law:

$$P_{a_D} = \frac{M_{a_D} RT_D}{V_D\, 144} \quad (6.2\text{-}8)$$

$$P_{a_S} = \frac{M_{a_S} RT_W}{V_S\, 144} \quad (6.2\text{-}9)$$

where:

M_{a_D} = mass of air in drywell, lb,

M_{as} = mass of air in the suppression chamber, lb,

R = gas constant ft-lbf/lb

V_D = drywell free volume, ft^3.

V_s = suppression chamber free volume, ft^3.

With known values of T_D and T_w , Equations 6.2-6, 6.2-7, 6.2-8 and 6.2-9 can be solved by transient analysis and iteration. This iteration procedure is also used to calculate the unknown quantities M_{a_D} and M_{as}.

Solution of Equations

The transient analysis is based on successive time step integration of the suppression pool temperature. When this integration has been performed and the value of T_s at the end of a time step has been calculated, a pressure balance is made. Using values of M_{a_D} and M_{as} from the end of the previous time step and the updated values of T_D and T_s, a check is made to see if P_s is greater than or equal to P_D using Equations 6.2-6, 6.2-7, 6.2-8, and 6.2-9. If P_s is greater than or equal to P_D, then the two values are made equal. The vacuum breakers between the drywell and suppression chamber are provided to ensure that P_s cannot be greater than P_D.

Hence, with

$$P_D = P_s$$

and knowing that:

$$M_{a_D} + M_{a_s} = \text{constant}; \qquad (6.2\text{-}10)$$

where the constant is the known total initial mass of air in the suppression chamber and drywell prior to the accident, Equations 6.2-6, 6.2-7, 6.2-8, and 6.2-9 can be solved for M_{a_s}, M_{a_D}, and P_s/P_D.

It is conservatively assumed that the total mass of air remains constant, which ignores any containment leakage that might occur during the transient.

If, as a result of the end-of-time-step pressure check,

$$P_S \leq P_D \leq P_S + \frac{H}{V_W},$$

where:

H = submergence of vents, ft, and

V_w = specific volume of fluid in vent, ft³/lb

then the pressure in the drywell is higher than the pressure in the suppression chamber but not sufficiently so to depress the water to the bottom of the vents and thus permit air to flow from the drywell to the suppression chamber. Under these circumstances, no air transfer is assumed to have occurred during the time step, and Equations 6.2-6, 6.2-7, 6.2-8, and 6.2-9 are solved using the updated temperatures with the same M_{a_s} and M_{a_D} values from the previous time step.

If the end-of-time step pressure check shows:

$$P_D \geq P_S + \frac{H}{V_W}$$

then the drywell pressure is set to the value:

$$P_D = P_S + \frac{H}{V}. \qquad (6.2\text{-}11)$$

This requires that the drywell pressure never exceed the suppression chamber pressure by more than the hydrostatic head associated with the submergence of the vents. To maintain this condition, some transfer of drywell air to the suppression chamber will be required. The amount of air transfer is calculated by using Equation 6.2-10 and combining Equations 6.2-6, 6.2-7, 6.2-8, 6.2-9 and 6.2-11 to give:

$$P_{v_D} + \frac{M_{a_D} R T_D}{144 V_D} = P_{v_s} + \frac{M_{a_s} R T_w}{144 V_s} + \frac{H}{v_w}$$

which can be solved for the unknown air masses. The total pressures can then be determined.

6.2.1.1.4 Negative Pressure Design Evaluation

Containment negative pressure has been addressed in Chapter 3.0 and in the Design Assessment Report.

6.2.1.1.5 Suppression Pool Bypass Effects

Protection Against Bypass Paths

The pressure boundary between drywell and suppression chamber including the vent pipes, vent header, and downcomers are fabricated, erected, and inspected by nondestructive examination methods in accordance with and to the acceptance standards of the ASME Code Section III, Subsection B, 1971 (Summer 1972 Addenda). This special construction, inspection and quality control ensures the integrity of this boundary. The design pressure and temperature for this boundary was established at 25 psid and 340° F, which is substantially greater than conditions during a DBA. Actual peak accident differential pressure and temperature across this boundary will be less than their design values during a LOCA. In addition a stainless steel liner has been provided between the drywell and the wetwell as described in Chapter 3.0.

All penetrations of this boundary except the vacuum breaker seats and suppression pool temperature monitoring probe penetrations and testing penetrations are welded. All penetrations are available for periodic visual inspection.

The following paragraphs describe the evaluation of the steam bypass event.

Reactor Blowdown Conditions and Operator Response

In the highly unlikely event of a reactor depressurization to the drywell accompanied by a simultaneous open bypass path between the drywell and suppression chamber, several postulated conditions may occur. For a given primary system break area, the maximum allowable leakage capacity can be determined when the containment pressure reaches the design pressure at the end of reactor blowdown. The most limiting conditions would occur for those primary system break sizes which do not cause rapid reactor depressurization. This corresponds to breaks of less than approximately 0.4 ft^2 which require some operator action to terminate the reactor blowdown.

Immediately after the postulated conditions given above for a small primary system break, there would be a fairly rapid rise in containment pressure as the noncondensable gases in the drywell are carried over to the suppression chamber. During this portion of the transient, it is assumed that the plant operators are unaware that a leakage path exists. Under normal circumstances, the maximum pressure that can occur in the suppression chamber is approximately 25 psig. This is the pressure that would result if all of the noncondensable gases initially in the containment are carried over to the suppression chamber free space. For the maximum allowable leakage calculations, it was assumed that the plant operators realize a leakage path exists only when the suppression chamber pressure reaches 30 psig. For conservatism, an additional 10-minute delay is assumed before any corrective action is taken to terminate the transient. The corrective action is also assumed to take 5 minutes to be effective. At that time, the containment pressure would be equal to the design pressure if the allowable leakage had occurred. The specific type of corrective action taken after 10 minutes is not accounted for in the analysis. The operators have several options available to them. If the source of the leakage is undefined, they could depressurize the primary system via either the main condenser or relief valves, or they could activate the containment sprays.

<u>Analytical Assumptions</u>

When calculating the allowable leakage capacities for a spectrum of break sizes, the following assumptions are made:

a. Flow through the postulated leakage path is pure steam. For a given leakage path, if the leakage flow consists of a mixture of liquid and vapor, the total leakage mass flowrate is higher, but the steam flowrate is less than for the case of pure steam leakage. Since the steam entering the suppression chamber free space results in the additional containment pressurization, this is a conservative assumption.

b. There is no condensation of the leakage flow on either the suppression pool surface or the containment and vent system structures. Since condensation acts to reduce the suppression chamber pressure, this is a conservative assumption. For an actual containment there will be condensation, especially for the larger primary system breaks where vigorous agitation at the pool surface will occur during blowdown.

<u>Analytical Results</u>

The LSCS containment has been analyzed to determine the allowable leakage between the drywell and suppression chamber.

Figure 6.2-13 shows the allowable leakage capacity $(A/\sqrt{K})$ as a function of primary system break area. A is the area of the leakage flow path and K is the total geometric loss coefficient associated with the leakage flow path.

The maximum allowable leakage capacity is at $(A/\sqrt{K})$ = .030 ft^2. Since a typical geometric loss factor would be 3 or greater, the maximum allowable leakage area would be .052 ft^2. This corresponds to a 3-inch line size.

Figure 6.2-13 is a composite of two curves.

If the break area is greater than approximately 0.4 ft^2, reactor depressurization will terminate the transient and allow higher leakage. However break areas less than 0.4 ft^2 result in continued reactor blowdown which limits the allowable leakage. Figure 6.2-14 shows the containment response associated with breaks larger than 0.4 ft^2. The containment pressure would reach design pressure at the end of reactor blowdown. Figure 6.2-15 shows the same response for a typical small break less than 0.4 ft^2. The containment pressure would reach design conditions, in this case, approximately 5 minutes after operator action.

6.2.1.1.6 Suppression Pool Dynamic Loads

The manner in which suppression pool dynamic loads resulting from postulated loss-of-coolant accidents, transients, and seismic events have been integrated into the LSCS design is completely described in the LaSalle Design Assessment Report, which was submitted with the FSAR as a reference document. The load histories, load combinations, and analyses are all presented in detail in this referenced report. A safety relief valve in-plant test was conducted on unit 1 as committed by Commonwealth Edison per NUREG-0519. A report entitled "Commonwealth Edison Proprietary LaSalle County I In-Plant S/RV Test Initial Evaluation Report" was submitted March 4, 1983 (C. W. Schroeder to A. Schwencer) and resubmitted October 14, 1983 (C.W. Schroeder to H.R. Denton). The document contains information and data demonstrating the adequacy of existing design basis hydrodynamic loads resulting from safety/relief valve actuation.

Supplementary evaluations have been performed, as discussed in Section 6.2.1.8, to verify that an increase in the initial suppression pool temperature (from 100° F to 105° F) would not significantly impact the dynamic loading scenarios associated with containment response to postulated LOCAs and SRV operation.

Containment Dynamic Loads were evaluated for the current licensed thermal power in Reference 25. The evaluation shows the LOCA and SRV loads remain within the defined limits.

6.2.1.1.7 Asymmetric Loading Conditions

The manner in which potential asymmetric loads were considered for LSCS is fully described in the Design Assessment Report. A description of the analytical models utilized for these analyses, as well as a description of the containment testing program, is also presented in this report.

6.2.1.1.8 Containment Ventilation System

The primary containment ventilation system is discussed in Section 9.4.

6.2.1.1.9 Postaccident Monitoring

A description of the postaccident monitoring system is provided in Section 7.5.

6.2.1.1.10 Drywell-to-Wetwell Vacuum Breaker Valves Evaluation for LOCA Loads

During the pool swell phase of a loss-of-coolant accident, air flows from the drywell through the vent pipes and the suppression pool into the suppression chamber air space resulting in a rise of the suppression pool surface and compression of the air space region above it. This transient wetwell air space pressurization may cause the vacuum breaker valves to experience high opening and closing impact velocities. To estimate the valve disc actuation velocities, the Mark II Owner's Group developed a vacuum breaker valve dynamic model described in NEDE-22178-P(1), "Mark II Containment Drywell-to-Wetwell Vacuum Breaker Models," August 1982, which describes the generic methodology used to calculate the response of the drywell-to-wetwell vacuum breaker to certain transients in the Mark II containment. The LaSalle plant, however, is unique in that it is the only domestic Mark II plant which has its vacuum breakers located outside containment. Because of this feature, the Mark II Owners Group model was modified to take credit for the pressure losses associated with the external piping and isolation valves which connect the vacuum breaker between the wetwell and drywell at LaSalle. In a letter dated December 28, 1982, CECo submitted a report to the NRC, CDI-82-33, "Reanalysis of the LaSalle Wetwell-to-Drywell Vacuum breakers under Pool Swell Loading Condition," December 1982, outlining the valve modeling improvement which have been made to take credit for the pressure losses associated with vacuum breaker piping. This report documents the reduction of the valve impact velocities during pool swell which are attributed to the use of a more realistic hydrodynamic torque on the valve disc. This analysis has been accepted by the NRC. However, because the hydrodynamic loads associated with a loss-of-coolant accident were not considered in the original design of the vacuum breaker, CECo decided to modify the vacuum breakers to improve performance and reliability, and to further increase the margin of safety. The modifications included material upgrade and/or dimensional changes to strengthen eccentric shaft, hinge arms, hinge plates, fasteners and a load distribution device to reduce the severity of the vacuum

breaker pallet opening impact loading. The modified design was tested under an applied mechanical force which produced an opening pallet impact velocity of 20.2 radians/second and a closing impact velocity of 25.8 radians/second. The predicted pallet impact velocities for LaSalle are an opening impact velocity of 16.6 radians/second and a closing impact velocity of 24.2 radians/second. After testing, the vacuum breaker leak rate was verified to be within the acceptable limit. The test results verified the operability and functional capability of the vacuum breaker well in excess of the predicted opening and closing impact velocities, and, thus, demonstrated that the modified LaSalle vacuum breakers will function properly under pool swell induced impact loadings with a considerable margin of safety.

6.2.1.1.11 Impact of Increased Initial Suppression Pool Temperature

Supplementary safety evaluations have been performed, as discussed in Section 6.2.1.8, to verify that an increase in the initial suppression pool temperature (from 100° F to 105° F) would not significantly impact the consequences of the various containment line break analyses.

6.2.1.2 Containment Subcompartments

For the most part, the drywell is a large continuous volume interrupted at various locations by piping, grating, ventilation ducting, etc. The only two volumes within the drywell which can be classified as subcompartments are the annular volume between the biological shield and the reactor pressure vessel, and the volume bounded by the drywell head and the reactor vessel head. These regions are referred to as the biological shield annulus and head cavity, respectively, and require special design consideration resulting from the postulation of line breaks in these volumes.

6.2.1.2.1 Design Bases

The methodology used to determine the containment subcompartment pressurization loads and the results pertaining to the pressurization loads documented herein are applicable to reactor operation at or below the current licensed thermal power (Reference 30).

Biological Shield Annulus

Pressure transients within the biological shield annulus are important for two considerations: (1) determination of the design conditions for the shield wall, and (2) determination of the tipping forces on the reactor pressure vessel. It is not a priori clear that one line break will yield the most severe conditions for both considerations. Therefore, consequences of two line breaks were studied:

(a) a complete circumferential failure of one of the two recirculation outlet lines at the safe end to pipe weld, and (b) a complete circumferential failure of one of the six feedwater lines at the safe end to pipe weld. While it was assumed that the recirculation line break with its high mass and energy blowdown rates yields most severe shield wall loads, the break of the feedwater line was added to determine the most severe conditions on the vessel. The pressure transients following either postulated break were used in determination of shield wall and pressure vessel design adequacy.

The performed pressurization analyses for the postulated recirculation line break and feedwater line break were based on the nodalization schemes depicted on Figures 6.2-16 and 6.2-17, respectively. Both nodalization schemes were given careful consideration to assure correct local and overall pressure responses.

Recirculation Line Break

The sudden injection of the subcooled liquid into the shield penetration (Node 35) and adjoining annulus initially causes a significant fraction of the liquid to flash to steam, pressurizing the penetrations and annulus. The responses of the penetration volume and adjoining subcompartments are shown on Figure 6.2-18. Within 10 milliseconds after the postulated break both flows out of the penetration have choked. Some 10 milliseconds later, both the penetration pressure and the pressure in the surrounding annulus node peak, reflecting subcooling and inventory effects addressed in the blowdown flow rates. Flow into the annulus initially proceeds in all directions, but soon swings preferentially upward in response to increasing pressures within the dead-ended skirt region. By 0.1 second into the transient, the pressures in and about the penetration have stabilized and shortly after (by 0.5 seconds), the differential pressures across the shield wall have begun to decrease (Figure 6.2-21). The differential pressure across the shield wall peaks at 115 psid in the region immediately around the penetration. Peak differential pressure across the shield door in the penetration, however, reaches 325 psid.

Feedwater Line Break

Pressurization effects of the postulated feedwater line break are much less pronounced than for the recirculation break. Much of the injected fluid finds its way up and out of the annulus and over the top of the shield wall and into the drywell. Nevertheless, the differential pressure across the shield wall surrounding the penetration peaks at 50 psid, while the differential pressure across the shield door in the penetration reaches 205 psid (Figure 6.2-22). By 0.5 second into the transient all the differential pressures across the shield wall have peaked and are decreasing (Figure 6.2-23).

The break area for the recirculation line break was assumed to be time dependent and limited by effects of pipe restraints (see Attachment 6A). The feedwater line break was assumed to provide instantaneous full size break area. Both break models included the effects of subcooled liquid inventory in the determination of mass and energy flux data.

No margins were applied to the calculated differential pressures for this final pressurization analysis.

Head Cavity

The head cavity area was analyzed for specific line breaks. They were: 1) a break of the recirculation outlet line within the drywell; and 2) a break of the main steamline within the drywell; and, 3) a simultaneous break of the head spray line and the RPV head vent line within the head cavity. These analyses were carried out to establish the pressure differentials that would exist across the refueling bulkhead plate as a result of these accident conditions. The break of the recirculation outlet line, the drywell DBA, was found to produce the highest pressure differential across the refueling bulkhead plate, a value of 9.0 psid upward. The simultaneous break of the head spray line and RPV head vent line caused a pressure differential of 7.0 psid downward. The main steamline data are not presented due to the fact that the recirculation line break produced the higher differential pressure value.

The break size, mass flow rate, and energy content for the recirculation line were defined in Subsection 6.2.1.1.3.1 and Table 6.2-18. The supporting assumptions for these data are also supplied in the same subsection. The break size, mass flow rate, and energy content for the head spray line were determined using Moody's flow through the 3.72-inch diameter head spray nozzle at reactor conditions with a multiplier of 1.0. Flow from the other side of the head spray line break was neglected. In addition, the simultaneous break of the RPV head vent line was considered because of the lack of whip restraints on the head spray line. The break size, mass flow rate, and energy content for the RPV head vent line were determined using Moody's flow at reactor conditions with a multiplier of 1.0. The RPV head vent line was postulated to rupture at the four-to-two inch reducer in the line located in the head cavity. The flow occurred at both ends of the break, one having a diameter of 4.0 inches and the other 2.0 inches.

No margin was applied to the results, since the analysis was done for the final design, and a margin is not required for that situation. However, a margin does exist, and this is indicated in Tables 6.2-11 and 6.2-12.

6.2.1.2.2 Design Features

Biological Shield Annulus

The biological shield annulus is an annular space 48.7 feet high and about 2 feet thick formed by the reactor pressure vessel and its skirt and the biological shield wall. The shield wall is provided with 32 penetrations to allow for routing for the lines connected to the vessel. The shield wall is also pierced to provide 2 HVAC openings and 2 reactor skirt access doors. The 3-1/2 inch thermal insulation divides the shield annulus, except for the lower skirt portion, into 2 almost equal annului. The inner steel shell of the annulus is spanned with vertical and horizontal

stiffeners which extend 5 inches into the annulus. Egress to the drywell at the top of the shield is partially blocked by the gusset plates supporting the reactor vessel stabilizers (Figures 3.8-23). The penetrations in the shield wall are designed with shield doors with a gap of approximately 3 inches between the doors and the thermal insulation on the penetrating lines. Figure 3.8-39 provides an exterior wall stretchout of the shield wall.

In the annulus pressurization analysis, it was assumed that following the postulated line break the vessel insulation within the annulus was instantaneously displaced to the shield wall. The vessel insulation support structure remains in its original configuration. Venting of the annulus into the drywell was possible through the annulus between the pipe and shield doors in the 32 nozzle penetrations in the shield wall and by means of an opening at the top of the shield wall above which the insulation was assumed to blow out instantaneously when the pressure across the insulation above the shield wall reaches 3 psid. Other possible vent paths such as HVAC openings, reactor skirt access doors, and insulation blowout panels were assumed to remain closed.

Head Cavity

Note: The current flow paths have been changed to include the two manholes between the head cavity and the drywell and the four ducted HVAC vents have been modified by the addition of discharge nozzles. The impact of this change has been evaluated and it has been determined that the analysis presented here is bounding.

The physical system, shown in Figure 3.8-1, was modeled as three node with two flow paths for this analysis. The head cavity, drywell, and wetwell are all described by single volumes. The model for the simultaneous break of the head spray and RPV head vent lines in the head cavity is shown in Figure 6.2-19, and that for the recirculation line break in the drywell in Figure 6.2-20. The pertinent data regarding the volumes and flow paths are given in Tables 6.2-11 through 6.2-14. There are eight HVAC vents in the refueling bulkhead plate: four sixteen-inch diameter supply vents, and four eighteen-inch diameter return vents. The return vents have ductwork attached to them. All of the HVAC (supply and return) were modeled for the postulated break in the head cavity since the pressure in the return vents with the ductwork would always be greater than the drywell pressure. However, only the supply vents were considered to allow flow for the breaks in the drywell. It was assumed that the HVAC return ductwork would be crushed by the fast rising drywell pressure. The downcomer vents between the drywell and wetwell were modeled as one flow path with a valve in the path set to open at 0.824 second for the recirculation line break. The 0.824 second was taken as a conservative estimate of the time normally required to clear the downcomer vents. At this time, the entire vent area becomes available for pressure relief of the drywell and head cavity region. The simultaneous head spray line and RPV head

vent line break is a much smaller break and results in a relatively slow pressurization of the drywell. A valve was again used in the flow path, but in this instance, the valve opening was dependent upon the drywell pressure exceeding the hydrostatic head at the downcomer exit. The opening differential pressure used was 5.2 psid which is equivalent to a 12-foot downcomer submergence. The flow was carried over directly into the wetwell air volume. No credit was taken for condensation. The flow through both flow paths was taken to be a completely homogeneous mixture.

6.2.1.2.3 Design Evaluation

Biological Shield Annulus

The RELAP 4 Mod 3 computer code was used to perform the analyses. The assumptions made in modeling the problem were in accordance with the applicable USNRC guidelines.

The mass and energy blowdown rates were determined according to the methods described in Attachment 6.A.

Initial conditions in the annulus and drywell are indicated in Tables 6.2-9 and 6.2-10.

In subsonic flow conditions, two flow models were used, as defined in RELAP 4 Mode 3: (a) compressible flow, single stream model was used for the path of major flow direction, and (b) incompressible flow without momentum flux model was used for flow paths other than the paths of the major flow direction. For sonic flow conditions the Moody or sonic choking model were specified with the multiplier 0.6 for the Moody choking model. Homogeneous flow was assumed for the vent mixture.

The biological shield annulus between the reactor pressure vessel and the shield wall was modeled differently for each of the two postulated line breaks. In either case, advantage was taken of the near symmetry of the annular space across the vertical plane passing through the centerline of the failed line.

Nodalization of the biological shield annulus was determined on the basis of natural geometric boundaries and the constraint that the pressure drop within a node be reasonably low as compared to pressure drop across the boundaries of the node. Nodal boundaries were suggested by the presence of the reinforcing steel, thermal insulation support structure and nozzles. Significant pressure drops near the break suggested smaller nodes (by and large limited with two successive obstructions) around the penetration than elsewhere (Figures 6.2-37 and 6.2-38). Therefore the assumption was made that since RELAP 4 allows input of loss coefficients only at the junctions between nodes, the junctions should be placed at points where major

pressure losses occur. Furthermore, it may be concluded that increasing the number of junctions (by making smaller nodes) beyond this point will yield no improvement in the accuracy of the results.

To test this hypothesis, a sensitivity study was performed on the sacrificial shield nodalization. Using the original nodalization (Figure 6.2-39) as a basis, an "equivalent" model was run which maintained the nodalization near the break but drastically reduced the number of nodes further from the break (Figure 6.2-40). This model demonstrated identical pressure response close to the break and only minor differences away from the break (Figures 6.2-41 and 6.2-42). This indicated that the nodalization far from the break was sufficiently refined in the original model and that the "equivalent" model could be used to simulate a response close to the break.

Two additional models were run. The first combined the nodes closest to the break into one large node (Figure 6.2-43). The pressure response was not consistent with the original runs (Figures 6.2-44 and 6.2-45). This indicated that a model which does not locate node boundaries at all flow restrictions close to the break is not acceptable. The last model substituted six nodes for the three original nodes, causing junctions to occur at locations which coincide with no actual flow restriction (Figure 6.2-46). This model showed a net increase of 5% in the force caused by the pressures in the area being investigated. An examination of the axial and circumferential pressure distributions showed only minor differences (Figures 6.2-47 and 6.2-48).

The sensitivity study indicates that the original nodalization provides an adequate description of the pressurization of the sacrificial shield annulus. An increase in the complexity of the RELAP 4 model would not result in a significant change in the results.

As previously indicated, half of the annulus was nodalized in case of either postulated line break; for the recirculation line break half-annulus consisted of 35 nodes and the half-drywell of 3 nodes (Table 6.2-9), while for the feedwater line break the half-annulus consisted of 29 nodes and the half-drywell of 3 nodes (Table 6.2-10). Volume of each node was calculated as a net volume, that is, the respective volume of the annulus including the volume of penetrations (if any) was corrected for the volume of the insulation and nozzles. The junctions, 85 and 69 for the recirculation line break and feedwater line break respectively, were assigned the smallest flow area anywhere between the centers of two volumes. All partial loss coefficients, k_j's, were derived from Reference 6. The total loss coefficient k_t was then determined by adding the weighted partial loss coefficients in series:

$$k_t = \sum_i K_i \left(\frac{A_t}{A_i} \right)^2$$

where A_t is the junction area and A_i is the area within the junction and pertaining to the partial loss coefficient k. When parallel paths, j, were combined, the following relations were utilized:

$$A_t = \sum_j A_j$$

$$K_t = \left[\sum_i \left(\frac{A_i}{A_t} \frac{1}{\sqrt{k_i}} \right) \right]^{-2}$$

Only similar junctions were combined in this manner (like 2 or more penetrations connecting drywell with the same volume of the annulus), other junctions were modeled separately.

Inertia coefficients were similarly calculated using simplified conservative approximations to the integrated junction characteristics. Thus, for the junctions with only minor variations, in cross-sectional flow area along the junction, the inertia, I, was approximated by:

$$I = \frac{1}{A_t} \sum_i L_i$$

where L_i is the distance along the junction where junction's cross-sectional area is A_i. In cases where there appear major variations in the cross-sectional flow area (constriction in the conduit) the inertia was estimated by:

$$I = \frac{L_1 - d}{A_1} + \frac{L_o + 2d}{A_o} + \frac{L_2 - d}{A_2}$$

where d is a "characteristic" diameter of the constriction of length L_o and with area A_o (for an orifice the characteristic diameter is taken to be the diameter of the orifice). L_1, A_1 and L_2, A_2 are the length and flow area of the conduit partitioned by the constriction. In special cases, where the constriction is not an ordinary orifice, a variation of the above relation was used to evaluate I.

Parallel paths were characterized by:

$$I = \left(\sum_j \frac{1}{I_j} \right)^{-1}$$

To further illustrate methods of determination of the junction characteristics, treatment of selected representative junctions will be shown in detail. The junctions are those for the recirculation line break nodalization scheme: 9, 47, 72.

Junction 9 connects the break volume (node 35), which consists of the half-annulus in the recirculation line penetration extended from the shield door to the reactor vessel, with the surrounding annular node (34). The minimum junction area was in this case within the break volume, half of the annular area formed by the recirculation line and the penetration wall was calculated to be 7.04 ft^2. In determining the loss coefficient for this junction, Diagram 11-9, Reference 6, was utilized. An upper limit value was set at 0.85 and considered the only loss for this junction.

The inertia coefficient, I, for the junction was calculated as a sum of two contributions: (a) inertia through the half-annulus of the penetration (0.23), and (b) an upper limit estimate of the inertia within the annulus, node 34 (0.07), totaling 0.30 ft^{-1}.

Junction 47 is a vertical junction connecting nodes 16 and 21. The junction area is the related annulus cross-section area reduced by two constrictions, stiffener and the thermal insulation support structure. Although the constrictions appear at different elevations (11 inches apart), they were assumed at the same elevation. This assumption leads to the junction area of 7.72 ft^2 (upstream volume flow area is 11.87 ft^2 and the flow area of the downstream volume is 12.36 ft^2). The loss coefficient was estimated using Diagram 4-9 of Reference 6, at 0.66 for flow area 7.72 ft^2. The total junction loss coefficient is therefore 0.67. The junction area is characterized by the radial width of 1.45 feet. This width was taken as the characteristic length, d, for the purposes of the inertia coefficient determination. Then, using a variation of the above described relation for I,

$$I = \frac{d}{A_o} + \frac{L - d}{A_2}$$

it was found that I = 0.45 ft^{-1}.

Junction 72 is an example of the vent path through the line penetration and connects annular node 28 with the containment node 37. The actual penetration is located on the boundary between nodes 28 and 29. For this reason, only half of the penetration was treated as the junction 72.

The minimum area of the junction is the cross-sectional area of the half of annulus between the shield door and penetration line. It was determined to be 9.71 ft^2. Half-penetration flow area was calculated at 5.33 ft^2. The inertia coefficient for this junction was determined on the basis of the above areas and the characteristic diameter as being the hydraulic diameter at the penetration exit (3.3 ft^{-1}). The loss coefficient for the junction was, however, determined for the whole penetration and it consisted of a friction loss (0.02 for A = 10.65 ft^2), turning losses at the nozzle and contraction-expansion losses at the shield doors. The turning losses were approximated with losses in the branch of a tee section as shown in Diagram 7-21, Reference 6, and estimated at 1.05 based on the penetration area 10.65 ft^2. The loss at the shield door was approximated with a loss due to a discharge from a straight conduit through a thick-walled orifice or grid, Diagram 11-28, Reference 6, and calculated at 1.69 based on the penetration exit area 1.424 ft^2. Then the total loss coefficient based on the area 1.424 ft^2 is 1.71, which is the loss coefficient of the junction.

A complete review of all volume and junction parameters as used in the analyses is given in Tables 6.2-9, 6.2-10, 6.2-24, and 6.2-25. Tables of junction characteristics include an indication whether the junction was choked during the analysis. The junctions closer to the break volume choked very early in the transient; an indication that the pressurization was hardly a function of either assigned loss coefficients or inertia coefficients.

Mass and energy blowdown rates used in the analysis are given in Tables 6.2-26 and 6.2-27.

Figure 6.2-18 depicts the calculated differential pressures across the biological shield wall (doors) for the postulated recirculation line break. Figures 6.2-49 and 6.2-50 show final pressure distribution in axial and circumferential direction, respectively also for the recirculation line break. Figures 6.2-22, 6.2-51, and 6.2-52 give the same information for the postulated feedwater line break.

Head Cavity

Note: The current flow paths have been changed to include the two manholes between the head cavity and the drywell and the four ducted HVAC vents have been modified by the addition of discharge nozzles. The impact of this change has been evaluated and it has been determined that the analysis presented here is bounding.

The computer code utilized for this investigation was RELAP4/Mod 5 (Reference 7) as received from the Argonne Code Center. A listing of the input for each case (Tables 6.2-15 and 6.2-16) is provided to demonstrate the options of the code that were utilized to obtain a solution. The mass and energy inputs were taken from Table 6.2-18 for the recirculation line break, and calculated based on Moody's flow model with a multiplier of 1.0 for the simultaneous head spray line and RPV head vent line break. The details regarding the data contained in Table 6.2-18 are given in Subsection 6.2.1.1.3.1. The basic assumptions utilized in the analysis are given below.

a. Thermodynamic equilibrium exists in each containment subcompartment. The containment option of the RELAP4/MOD5 computer code was utilized which allows for the flow of air, water vapor, and liquid between the nodes.

b. The constituents of the fluid flowing through the subcompartment vents are based on a homogeneous mixture of the fluid in the subcompartment. The consequences of this assumption result in complete liquid carry-over through subcompartment vents.

c. No heat loss from the gases inside the primary containment is assumed. This adds extra conservatism to the analysis, i.e., the analysis will tend to predict higher containment pressures than would actually exist.

d. Incompressible single-stream flow without momentum flux was used for all junctions.

e. The Moody model for critical flow was used when choking occurred in a junction.

f. The stagnation properties which include dynamic velocity effects were used to determine the flow rate in conjunction with the Moody model.

g. A contraction coefficient of 0.6 was implemented with the junction flow areas which reduces the flow and retains higher pressures closer to the break. In addition, a contraction coefficient of 1.0 was utilized for the fill junction which was used to simulate the break.

h. The reactor pressure vessel head insulation remains in place and retains its structural integrity during any postulated accident. This is conservative since the RPV head cavity volume is minimized which will result in higher pressures in the head cavity.

i. The manholes between the head cavity and the drywell are assumed to be closed. This reduces the flow area between the volumes increasing the differential pressure across the bulkhead.

j. All of the HVAC vents (supply and return) are modeled for the postulated break in the head cavity since the pressure in the return vents with the ductwork would always be greater than the drywell pressure. However, only the supply vents are considered to allow flow for the breaks in the drywell. It is assumed that the HVAC return ductwork would be crushed by the rising drywell pressure.

k. To simplify the input to RELAP4/MOD5, the flow area properties of the HVAC vents are combined into one equivalent vent.

l. The downcomers are represented by an equivalent single flow path with a flow area equal to the sum of the actual flow areas.

m. The modeling of downcomer clearing the initiation of flow into the wetwell was modeled in two ways. In the case of the recirculation line break, the downcomer clearing is extremely rapid. To accurately simulate this, the model would have to be rather complex due to the large inertial and frictional effects present in the downcomer. This complexity was avoided by making use of an accident chronology shown in Table 6.2-7 which found the vent clearing time to be 0.824 second. A valve was placed in the flow path and opened 0.824 second after the line break. The simultaneous head spray line and RPV head vent line break is a much smaller break and results in a relatively slow pressurization of the drywell. A valve was again used in the flow path, but in this instance, the valve opening was dependent upon the drywell pressure exceeding the hydrostatic head at the downcomer exit. The opening differential pressure used was 5.2 psid which is equivalent to a 12-foot downcomer submergence.

n. No significant depressurization of the reactor pressure vessel occurs during the postulated break.

o. The simultaneous pipe break of the head spray line and the RPV head vent line was considered because of the lack of whip restraints on the head spray line. The resultant whip of the head spray line is assumed to rupture the RPV head vent line. Neither the RCIC nor the RHR system is operating during the time of the head spray line break, i.e., the RHR-RCIC stop valve is assumed to be closed during the time of the accident. The RPV head vent line is connected at the RPV head and at the main steam header. Therefore, a break in this line results in a two direction blowdown, one side feeds directly from the RPV, and

other feeds from the main steamline. The head spray line has a limiting flow area at the head spray nozzle which has a diameter of 3.72 inches. The RPV head vent line is postulated to rupture at the 4-inch to 2-inch reducer in the line located in the head cavity. The steam flow occurs at both ends of the break, one having a diameter of 4.0 inches and the other 2.0 inches. The total flow area was determined to be 0.163 square feet. All of the flows are assumed to have the same RPV conditions which are a pressure of 1050.0 psia and an enthalpy of 1190.0 Btu/lbm. Utilizing Moody's choked flow tables from RELAP4/MOD5, a maximum flow of 2200.0 lbm/sec-ft^2 or 357.9 lbm/sec was calculated. This is used as a constant flow rate for the break in the head cavity.

p. The mass and energy release rates used for the recirculation line break are those given in Table 6.2-18. The break sizes are specified in Subsection 6.2.1.1.3.1.1 and the details regarding line size, break size, orifice size, etc., are given in Table 6.2-4.

q. RELAP4/MOD5 lacks the ability to model steam condensation in the suppression pool. This limitation has no effect on the results obtained prior to vent clearing but will result in an overestimation of the pressure rise in the wetwell after vent clearing. Since the maximum differential pressure across the refueling bulkhead occurs very shortly after downcomer vent clearing in the case of the recirculation line break, the effect is negligible. However, it is noted that the long-term pressure values are not realistic because of this modeling method. In the case of the break in the head cavity, flow through the downcomers does not begin until long after the peak differential pressure across the refueling bulkhead plate occurs.

r. The initial conditions are taken to be the normal operating conditions as given in Table 6.2-3 except with a relative humidity of 0.1%. In the head cavity and drywell the initial pressure is 15.45 psia, the initial temperature is 135° F and the relative humidty is 0.1%. In the wetwell the initial pressure is 15.45 psia, the initial temperature is 100° F and the relative humidity is 0.1%.

The node and flow path data specifics are given in Tables 6.2-11 and 6.2-12 for the simultaneous break of the head spray and RPV head vent lines and Tables 6.2-13 and 6.2-14 for the recirculation line break. The nodes and flow paths are graphically depicted in Figure 6.2-19 for the simultaneous break of the head spray line and RPV head vent line, and Figure 6.2-20 for the recirculation line break.

A description of the loss coefficient determination for the flow paths is provided. This problem has only two flow paths to consider. The first path connects the head

cavity to the drywell and consists of eight ports through the bulkhead plate. Four of these ports are the HVAC supply ports for the head cavity and do not have any ductwork attached to them. The remaining four ports are the HVAC return ducts from the head cavity and have ductwork attached to them. All of the HVAC vents (supply and return) were modeled for the postulated break in the head cavity since the pressure in the return vents with the ductwork would always be greater than the drywell pressure. The losses considered were the turning losses of the fluid around the RPV head from the break to the HVAC ports in the bulkhead. These losses are very small since the turning radius around the RPV head is so large. Therefore, this loss was neglected. The ports without the ductwork were considered as thick-edged orifices. This loss coefficient was determined using Diagram 4-14 of Reference 6 and was calculated to be 1.52. The ports with the ductwork consist of a 24-inch to 18-inch diameter reducer followed by ductwork which includes a series of elbows and one tee. The flow finally exits into the drywell through one of the tee branches. Diagrams 3-9, 6-1, and 7-25 of Reference 6 were used to calculate the loss coefficient and it was determined to be 4.62. Since the flow through the ports with and without ductwork is parallel, the losses were combined for parallel flow and the total loss coefficient was calculated, as described in Subsection 6.2.1.2.3, to be 2.62. The flow area for this case is the total of the minimum flow areas through each of the eight HVAC vents. The total flow area was determined to be 11.12 square feet. For the recirculation line break within the drywell, only the supply vents which are without ductwork were considered to allow for flow. It is assumed that the HVAC return ductwork would crush because the drywell pressure would be greater than the pressure in the ductwork. The loss coefficient for this case is calculated for the ports without the ductwork. The loss coefficient was determined as mentioned earlier and was calculated to be 1.52. The flow area for this case was determined to be 4.92 square feet.

The loss coefficient for the second flow path, through the downcomers, was taken from Table 6.2-1 and is 5.2. No attempt was made to model the inertial effects of the clearing transient. The path was treated as a valve that opened at a prespecified time of 0.824 second for the recirculation line break. For the simultaneous head spray line and RPV head vent line break, the path was treated as a valve that opened when the drywell pressure exceeded the hydrostatic head of 5.2 psid which is equivalent to a 12-foot downcomer submergence. The path model considers no inertial effects; this is a conservative approach, since it has the effect of making the pressure differentials across the bulkhead plate higher.

Figure 6.2-24 depicts the pressure histories of the head cavity and drywell for the break in the head cavity and the recirculation line break in the head cavity and the recirculation line break in the drywell. The pressure differential histories across the bulkhead plate for the break in the head cavity and the recirculation line break in the drywell are shown in Figure 6.2-25. The peak pressure differential for each break was found to be 9.0 psid upward for the recirculation line break and 7.0 psid downward for the simultaneous head spray line and RPV head vent line break. The

differential pressure history as shown for the simultaneous break of the head spray line and RPV head vent line shows two differential pressure peaks. The first differential pressure peak is due to the sudden pressurization of the head cavity and the second peak is due to the sudden opening of the downcomers at a pressure differential between the drywell and wetwell of 5.2 psid. This second peak is erroneous because no inertial effects were modelled in the downcomer flow path and therefore was not considered as the design downward differential pressure. The design pressure differential is 10.6 psid in both directions. This provides for a margin factor of approximately 1.2 at the final design stage.

6.2.1.2.4 Impact of Increased Initial Suppression Pool Temperature

Supplementary safety evaluations have been performed, as discussed in Section 6.2.1.8, to verify that an increase in the initial suppression pool temperature would not significantly impact the consequences of this accident scenario.

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

This section contains a description of the transient energy release rates from the reactor primary system to the containment system following a LOCA with minimum ESF performance. In general, a very conservative analytical approach is taken in that all possible sources of energy are accounted for, whereas the suppression pool is assumed to be the only available heat sink. No credit is taken for either the heat that will be stored in the suppression chamber and drywell structures, or the heat that will be transmitted through the containment and dissipated to the environment.

The analysis is performed with consistent methodology, including the RPV blowdown, as the short-term containment analysis, as noted in Subsection 6.2.1.1.3. The break flow rate and enthalpy used for the short-term containment response analysis at 3559 MWt are given in Table 6.2-18. For the analysis of the long-term containment response, one of the key input assumptions updated for the current analysis is that the core decay heat is based on the ANSI/ANS 5.1-1979 decay heat model with a two sigma uncertainty adder. The core decay heat values used in the analysis are provided in Table 6.2-20. The following subsections explain how the transient mass and release rates from the vessel to the containment were determined.

6.2.1.3.1 Mass and Energy Release Data

Table 6.2-18 provides the mass and enthalpy release data for the containment DBA, recirculation line break. Blowdown steam and liquid flow rates and their respective enthalpies are reported for a 24-hour period following the accident. Figures 6.2-26

and 6.2-27 show the blowdown flow rates for the recirculation lines break graphically. This data was employed in the DBA containment pressure-temperature transient analyses reported in Subsection 6.2.1.1.3.1.

Table 6.2-19 provides the mass and enthalpy release data for the main steamline break. Blowdown data is presented for a 24-hour period following the accident. Figure 6.2-28 shows the vessel blowdown flow rates for the main steamline break as a function of time after the postulated rupture. This information has been employed in the containment response analyses presented in Subsection 6.2.1.1.3.1.

6.2.1.3.2 Energy Sources

The reactor coolant system conditions prior to the design basis recirculation line break are presented in Tables 6.2-3 and 6.2-4. Reactor blowdown calculations for containment response analyses are based upon these conditions during a loss-of-coolant accident.

Following each postulated accident event, the stored energy in the reactor system and the energy generated by fission product decay will be released. The rate of release of core decay heat for the evaluation of the containment response to a LOCA is provided in Table 6.2-20 as a function of time after accident initiation. This data is based upon the ANS 5.1-1979 Decay Heat Standard, assumptions appropriate for a 24 month operating cycle, normalized to the current power level, and includes a two standard devisation (2 Sigma) confidence factor.

Following a LOCA, the sensible energy stored in the reactor primary system metal will be transferred to the recirculating ECCS water and will thus contribute to the suppression pool and containment heatup. Figure 6.2-29 shows representative temperature transients of the various primary system structures which contribute to this sensible energy transfer. Figure 6.2-30 shows representative variation of the sensible heat content of the reactor vessel and internal structures during a recirculation line break accident based upon the temperature transient responses.

6.2.1.3.3 Effects of Metal-Water Reaction

The containment systems shall accommodate the effects of metal-water reactions and other chemical reactions following a postulated DBA. The amount of metal-water reaction is limited to values consistent with the performance objectives of the emergency core cooling systems (ECCS).

6.2.1.3.4 Impact of Increased Initial Suppression Pool Temperature

Supplementary safety evaluations have been performed, as discussed in Section 6.2.1.8, to verify that an increase in the initial suppression pool temperature would not significantly impact the consequences of this accident scenario.

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

Not applicable.

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

Not applicable.

6.2.1.6 Testing and Inspection

Containment testing and inspection programs are fully described in Subsection 6.2.6 and in Chapter 14.0 of the FSAR. The requirements and bases for acceptability are outlined completely in the Technical Specifications.

6.2.1.7 Instrumentation Requirements

A complete description of the instrumentation employed for monitoring the containment conditions and actuating those systems and components having a safety function is presented in Chapter 7.0.

6.2.1.8 Evaluation of 105° F Suppression Pool Initial Temperature

Temperature limits on the suppression pool for Boiling Water Reactors (BWR) with Mark II containment were implemented to minimize the potential for high amplitude loads on the pool during accident events. However, some of the limits were implemented with excessive conservatism because the loading phenomena were not completely understood. This suppression pool temperature limit has therefore been historically chosen based on the maximum expected service water temperature. For LaSalle County Station Units 1 and 2, the licensing safety evaluations were based upon a 100° F suppression pool water temperature, which was equivalent to the Ultimate Heat Sink design temperature limit.

Hot weather in Illinois can cause the temperature of the ultimate heat sink to rise to the point where the suppression pool temperature limit of 100° F may be exceeded. However, the ultimate heat sink design limit will not be exceeded. To prevent an unnecessary plant shutdown during a period of high electrical demand, plant specific safety evaluations have been performed (References 10-20) to demonstrate that plant operation with higher suppression pool temperature is acceptable, i.e., the plant safety limits will still be met with the higher temperatures.

The suppression pool was designed to function as both a heat sink and an emergency water source during transient and accident events as discussed throughout section 6.2. Therefore, performance of the following evaluations were required to support a 5° F increase in the initial suppression pool temperature as LaSalle County Station Units 1 and 2:

a) Containment loads associated with SRV operation including air clearing loads and steam condensation loads.

b) Containment response associated with LOCA events including the peak pressure and temperature design limits, condensation capability, condensation oscillation loads (CO), and chugging loads.

c) Equipment performance for design basis events including the impact on the core cooling capability of the ECCS and the parameters which could impact the operability of the ECCS pumps (such as NPSH availability, etc.).

d) Equipment and ECCS performance for other non-LOCA events, e.g., ATWS.

For each of these cases the evaluation showed that the increase of the initial suppression pool temperature would have an insignificant impact on the existing design margin for the suppression pool and ECC systems. Peak local pool temperature will increase by 3° F at a 105° F initial pool bulk temperature for SRV related events.*

The results of this evaluation were submitted to the NRC (Reference 11), and an approved license amendment to change the maximum suppression pool temperature limit to 105° F was received (Reference 12). The Ultimate Heat Sink design temperature limit is changed to 107° F in Reference 36.

6.2.2 Containment Heat Removal System

The containment heat removal system function is accomplished by the containment cooling mode of the RHR system. The system is also equipped with spray headers in the drywell and suppression chamber areas. However, no credit was taken for these spray headers for either heat removal or fission product control following a LOCA.

6.2.2.1 Design Bases

The containment heat removal system, consisting of the suppression pool cooling system, is an integral part of the RHR system. It meets the following safety design bases:

a. The source of water for restoring RPV coolant inventory is located within the containment to establish a closed cooling-water path.

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

c. This system, in conjunction with the ECC systems, has such diversity and redundancy that no single failure can result in its inability to cool the core adequately (Subsection 6.3.1).

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* Peak bulk suppression pool temperature, in the case of LOCA events, is still approximately 10° F below the allowable values.

d. To ensure that the RHR containment cooling subsystem operates satisfactorily following a LOCA, each active component shall be testable during operation of the nuclear system.

6.2.2.2 System Design

The containment cooling subsystem is an integral part of the RHR system, as described in Subsection 5.4.7. The piping and instrumentation diagram is given in Drawing Nos. M-96 (sheets 1-4) and M-142 (sheets 1-4). Redundancy is achieved by having two complete containment cooling systems.

Consideration of the fouling of heat exchangers and the selection of temperatures for heat exchanger design are discussed in Subsection 5.4.7.

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 46° F. Subsequent to the accident, fission product decay heat will result in a continuing energy dump to the pool. Unless this energy is removed from the primary containment system, it will eventually result in unacceptable suppression pool temperatures and containment pressures. The containment cooling mode of the RHR system is used to remove heat from the suppression pool.

A supplementary evaluation has been performed for the addition of feedwater to the suppression pool to assess the impact on long term pool temperature. This evaluation estimates that the peak short term pool temperature will increase by an additional 15.4° F. This results in a short term pool temperature (at 600 seconds) of approximately 166° F. Further details are given in Section 6.2.1.1.3.1.1 in the paragraph titled, "Evaluation of Post-LOCA Feedwater Injection".

6.2.2.3.1 RHR Containment Cooling Mode

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

In order to evaluate the adequacy of the RHR system, the following limiting case is postulated:

a. Reactor initially at maximum power.

b. Isolation scram occurs.

c. Manual depressurization discharges heat to suppression pool.

d. Suppression pool cooling is established approximately 10 minutes after the technical specification limit for pool water temperature is reached.

A complete discussion of the suppression pool temperature transients is contained in Chapter 6 of the LSCS-DAR.

The suppression pool temperature transients have been analyzed based on an increased initial suppression pool temperature of 105° F as discussed in Section 6.2.1.8. The scenarios analyzed are based on those specified in NUREG-0783, Reference 15 provides the results of this analysis. For all analyzed cases the long term suppression pool temperature is less than 200° F.

6.2.2.3.2 Summary of Containment Cooling Analysis

When calculating the long term, post LOCA pool temperature transient, it is assumed that one RHR heat exchanger loop is not available, the suppression pool level initially is at the technical specification minimum, the suppression pool temperature initially is at the technical specification maximum, and the design RHR heat exchanger fouling factors are used. No credit is taken for heat loss to environs or to the pool structures.

It is concluded that even with the very conservative assumptions described above, the RHR system in the containment cooling mode can meet its design objective of safely terminating the limiting case temperature transient. A maximum suppression pool transient temperature of 200 degrees F has been supported by analysis.

6.2.2.3.3 Impact of Increased Initial Suppression Pool Temperature

Supplementary evaluations have been performed, as discussed in Section 6.2.1.8, to verify that an increase in the initial suppression pool temperature would not impact the ability of the RHR containment cooling system to meet its design objective.

6.2.2.3.4 Impact of Reduced RHR Suppression Pool Cooling Flow Rate

Additional supplementary evaluation has been completed which considers an RHR pump flow rate during the suppression pool cooling of 7200 gpm. As noted in Table 6.2-2, the previous analysis used a flow rate of 7450 gpm. Although the RHR pump is capable of such performance, the minimum required Technical Specification flow per specification SR 3.6.2.3.2 is only 7200 gpm. Since suppression pool cooling is only initiated after 600 seconds into the DBA-LOCA, the affect of this lower flow rate will be seen as slightly lower efficiency for the RHR heat exchanger

and a higher long term suppression pool temperature. The results of the Reference 18 General Electric analysis indicate an increase in the long term pool temperature of 1.5° F for the DBA-LOCA case.

For cases which involve SRV blowdown to the suppression pool the lower RHR pump flow rate was assessed in S&L Calculation 3C7-0181-003, Rev. 3 (Reference 15) and the effect on the peak suppression pool temperature was an increase of less than or equal to 1° F in the peak suppression pool temperature. For all cases examined, the highest peak pool temperature calculated is 195° F which is still less than 200° F peak temperature for all cases analyzed. Thus, complete steam condensation is assured with these elevated pool temperatures.

6.2.2.3.5 Impact of Power Uprate

The resultant post-LOCA maximum suppression pool temperature at 102% of uprated reactor thermal power is 196.1° F, as shown in Table 6.2-5. The resultant maximum long-term post DBA-LOCA suppression pool temperature with the concerns of SC06-01 addressed is 197°F as shown in Table 6.2-5. The maximum suppression pool temperature for NUREG-0783 events is 190.7° F as evaluated in Reference 31.

The suppression pool limit for events with SRV discharge is evaluated in References 25 and 27. In the NRC's Safety Evaluation of Reference 28 for the elimination of local suppression pool temperature limits for plants with T-Quenchers, an additional concern was raised on the potential transfer of non-condensed SRV steam plumes to ECCS suction strainers. An analysis was performed in Reference 29 that modeled the steam plume formation, determined the extent of steam plume projection, and verified that the plume can not enter ECCS suction strainers. However, the analysis determined the existence of a potential steam ingestion concern for the "K" SRV and the Reactor Core Isolation Cooling (RCIC) suction strainer, if the temperature of the suppression pool is above 200° F. Administrative controls have been implemented to caution the operators on the use of "K" SRV and RCIC simultaneously when the suppression pool temperature is above 200° F.

6.2.2.3.6 Sensitivity of Initiation Time of RHR Containment Cooling Mode

A one-time sensitivity analysis was performed to determine the impact on the peak suppression pool temperature, if the start of the RHR Containment Cooling Mode is delayed for longer than 10 minutes, following a DBA-LOCA. Manual operator action from the main control room is needed, in order for Suppression pool cooling to be initiated. These actions could require up to a few minutes to accomplish (accounting for valve stroke times, etc.). The impact on peak suppression pool temperature was studied if the start of suppression pool cooling is delayed from 10 minutes to 30 minutes.

The study utilized power uprate decay heat loads. The results of this study indicate there is a very small impact on peak suppression pool temperature. The 30 minute case results in an increase of 2.0 deg-F, which, when added to the current analysis peak of 197 deg-F, results in a postulated peak temperature of 199 deg-F. This peak temperature does not challenge the suppression pool design limits. The operator actions to re-align RHR are anticipated to require much less time than the additional 20 minutes of this analysis. The increase in peak suppression pool temperature is concluded to be negligible (i.e. less than 1 deg-F) for these anticipated starting times which are only a few minutes longer than 10 minutes.

6.2.2.4 Test and Inspections

The operational testing and the periodic inspection of components of the containment heat removal system are described in Subsection 5.4.7.4.

6.2.2.5 Instrumentation Requirements

Suppression pool cooling by the RHR system is manually initiated from the control room where sufficient instrumentation is provided for that purpose.

6.2.3 Secondary Containment Functional Design

The Secondary Containment consists of the Reactor Building, the equipment access structure, and a portion of the main steam tunnel and has a minimum free volume of 2,875,000 cubic feet.

The reactor building completely encloses the reactor and its primary containment. The structure provides secondary containment when the primary containment is closed and in service, and primary containment when the primary containment is open, as it is during the refueling period. The reactor building houses the refueling and reactor servicing equipment, the new and spent fuel storage facilities, and other reactor auxiliary or service equipment, including the reactor core isolation cooling system, reactor water cleanup demineralizer system, standby liquid control system, control rod drive system equipment, the emergency core cooling system, and electrical equipment components.

6.2.3.1 Design Bases

The functional capability of the ventilation system to maintain negative pressure in the secondary containment with respect to outdoors is discussed in Subsection 9.4.2.

6.2.3.2 System Design

The reactor building is designed and constructed in accordance with the design criteria outlined in Chapter 3.0. The reactor building exterior walls and superstructure up to the refueling floor are constructed of reinforced concrete.

Above the level of the refueling floor, the building structure is fabricated of structural steel members, insulated siding and a metal roof. Joints in the superstructure paneling are detailed to assure leaktightness. Penetrations of the reactor building are designed with leakage characteristics consistent with leakage requirements of the entire building. The reactor building is designed to limit the inleakage to 100% of the reactor building free volume per day at a negative interior pressure of 0.25 inch H_20 gauge, while operating the standby gas treatment system. The building structure above the refueling floor is also designed to contain a negative interior pressure of 0.25 inch H_20 gauge.

Personnel entrance to the reactor building is through an interlocking double door airlock. Rail car access openings in the reactor building at elevation 710 feet 6 inches provided with double doors to assure that building access will not interfere with maintaining integrity of the secondary containment.

Ventilation for the reactor building is provided by means of a once-through ventilation system. Outdoor air is filtered then evaporatively or chilled glycol cooled to *reduce the supply air dry bulb temperature to increase the sensible cooling capacity of this air. This air is then preheated as required to satisfy the plant operating conditions.

The equipment is arranged as follows: outside air inlet, filter, chilled glycol/heating coil evaporative *cooler (abandoned-in-place), resistive heating coils, and supply fans. Three 50% vane axial fans are provided, two of which normally operate and one which serves as a standby.

Supply air is distributed to the reactor building by means of a duct system to provide equipment cooling in various areas as required. Air is routed from clean areas to areas with progressively greater contamination potential. Pressure differential control dampers are used as required to maintain negative pressures in potentially contaminated cubicles. All exhaust air is routed through a return duct system to the exhaust fans.

All supply air delivered to the refueling floor level is exhausted from the periphery of the spent fuel and equipment storage pools and the reactor well. This air is routed directly to the main system exhaust duct. Three vane axial exhaust fans are provided, two of which normally operate and one of which serves as a standby. The discharge from the exhaust fans is routed to the plant vent where the air is discharged to the atmosphere. All exhaust air is monitored for radiation.

Normal ventilation systems are not required to operate during accident conditions and are automatically shut down whenever the standby gas treatment system starts. The equipment for this system is not powered from essential buses. To

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* Note: The evaporative coolers are abandoned-in-place.

maintain the integrity of the secondary containment, two isolation dampers are provided in the supply air duct between the supply fan discharge and the penetration through the secondary containment wall.

The secondary containment structure protects the equipment in the building from externally generated missiles. Piping systems within the secondary containment have been analyzed for high energy pipe breaks outside primary containment and pipe whip restraints are provided as required. The effects of jet impingment have also been analyzed and included in the design of the structure and pipe whip restraints. For more information on high energy pipe breaks outside primary containment see Appendix C.

The isolation features and isolation signals for secondary containment are discussed in Section 6.5, Chapter 7.0 and Subsection 9.4.2.

6.2.3.3 Design Evaluation

The design evaluation of secondary containment ventilation system and atmospheric cleanup system is given in Section 6.5 and Subsection 9.4.2.

6.2.3.4 Test and Inspections

The program for initial performance testing is outlined in the Technical Specifications. Periodic functional testing of the secondary containment and secondary containment isolation system is described in the Technical Specifications.

6.2.3.5 Instrumentation Requirements

The instrumentation to be employed for the monitoring and actuation of the standby gas treatment system is fully described in Chapter 7.0.

The instrumentation used for the monitoring and actuation of the ventilation and cleanup system is discussed in Subsections 7.3.8 and 7.6.1.2.

6.2.4 Containment Isolation System

The primary objective of the containment isolation system is to provide protection against the release of radioactive materials to the environment through the fluid system lines penetrating the containment. This objective is accomplished by ensuring that isolation barriers are provided in all fluid lines that penetrate primary containment, and that automatic closure of the appropriate isolation valves occurs.

6.2.4.1 Design Bases

The design requirements for containment isolation barriers are:

a. The capability of closure or isolation of pipes or ducts that penetrate the containment is provided to ensure a containment barrier sufficient to maintain leakage within permissible limits.

b. The arrangements of isolation valving and the criteria used to establish the isolation provisions conform to the requirements of General Design Criteria 54 through 57, as discussed in Section 3.1.

c. The design of all containment isolation valves and associated piping and penetrations is Seismic Category I.

d. Containment isolation valves and associated piping and penetrations meet the requirements of the ASME Boiler and Pressure Vessel Code, Section III, for Class 1 or 2 components, as applicable.

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

f. Containment isolation valves provide the necessary isolation of the containment in the event of accidents or other conditions to limit the untreated release of radioactive materials from the containment in excess of the design limits.

g. Appropriate isolation valves are automatically closed by the signals listed in Table 6.2-21. The criteria for assigning isolation signals to their associated isolation valves is described in Subsection 7.3.2. Once the isolation function is initiated, it goes to completion.

h. Redundancy and physical separation are required in the electrical and mechanical design to ensure that no single failure in the system prevents the system from performing its safety function.

The governing conditions under which containment isolation becomes mandatory are high drywell pressure or low water level in the reactor vessel. One or both of these signals initiate closure of isolation valves not required for emergency shutdown of the plant. These same signals also initiate the ECCS. The valves associated with an ECCS may be closed remote manually from the control room or close automatically, as appropriate.

Excess flow check valves are used as a means of automatic isolation on all static instrument sensing lines that penetrate the drywell containment and connect to

either the reactor pressure boundary or the drywell atmosphere. The valve is located downstream of the root valve and as close as practical to the outside surface of the containment. This valve is automatically closed to restrict flow in case of a sensing line break outside containment.

Backfill Injection lines have been added to the reference legs originating from Condensing Chambers 1(2) B21-D004A/B/C/D to comply with NRC Bulletin 93-03. These lines use two simple check valves in series to accomplish the outboard containment isolation function. It is acceptable to use the two simple check valves instead of one excess flow check valve for the backfill injection lines because these lines would not need the built-in bleed flow path in an excess flow check valve to reopen when appropriate. The 4 lbs./hr. CRD flow would reopen the check valves when it is available. If it is not available, it is not appropriate to reopen the check valves. This meets the Regulatory Guide 1.11 "... the valve should reopen automatically or be capable of being reopened readily under the conditions that prevail when reopening is appropriate. It should not be necessary to break a line to reopen a closed valve."

In addition, there is no instrument reading that will be significantly effected by the closure of these check valves.

Dead-end instrument sensing lines that are in communication with the reactor pressure boundary and penetrate the primary containment are equipped with 1/4 inch orifice as close to the process as possible inside the drywell.

6.2.4.2 System Design

Table 6.2-21 presents the design information regarding the containment isolation provisions for fluid system lines and instrument lines penetrating the containment. Containment isolation signals are identified in Table 6.2-21 and valve arrangements are represented in Figure 6.2-31.

The plant protection system signals that initiate closure of the containment isolation valves are listed in Table 7.3-2.

The isolation provisions follow the requirements of General Design Criteria 54, 55, 56, and 57. General Design Criteria 54 applies to all of the containment isolation valves. Compliance with General Design Criteria 55, 56, and 57 is described below. The justification for this design is also presented.

6.2.4.2.1 Evaluation Against General Design Criterion 55

Feedwater Line

Each feedwater line forming a part of the reactor coolant pressure boundary is provided with a swing type check valve on Unit 1 and a swing type check valve on Unit 2 inside the containment, and a nonslam type, air operated testable check valve outside the containment, as close as

practicable to the containment wall. In addition, a motor-operated gate valve is installed upstream of the outside isolation valve to provide long-term isolation capability.

During a postulated LOCA, it is desirable to maintain reactor coolant makeup from all available sources. Therefore, it would not improve safety to install a feedwater isolation valve that closed automatically on signals indicating a LOCA, and, thereby, eliminate a source of reactor makeup. The provision of the check valves, however, ensure the prevention of a significant loss of reactor coolant inventory and offer immediate isolation should a break occur in the feedwater line. For this reason, the outermost valve does not automatically isolate upon signal from the protection system. The valve is remote manually closed from the main control room to provide long-term leakage protection upon operator determination that continued makeup from the feedwater system is unavailable or unnecessary.

In addition, the outboard check valve is provided with a special actuator that performs the following functions:

a. The actuator is capable of partially moving the valve disc into the flow stream during normal plant operation in order to ensure that the valve is not bound in the open position. The actuator is not capable of fully closing the valve against flow, however, and there is no significant disruption of feedwater flow.

b. The actuator is capable of applying a seating force to the valve at low differential pressures and abnormal conditions. This improves the leaktightness capability of the valves. The actuator will be utilized during leak testing.

<u>ECCS Lines to the RPV</u>

The subject penetration(s) meet the alternate primary containment isolation criteria of NUREG 0800 “Standard Review Plan for the review of Safety Analysis Reports for Nuclear Power Plants” (SRP) instead of the explicit requirements of GDC 55.

The HPCS, LPCS, and LPCI lines penetrate the drywell and inject coolant directly into the reactor pressure vessel. Isolation is provided on each of these lines by a normally closed check valve inside the containment and a normally closed motor-operated gate valve located outside the containment, as close as practicable to the exterior wall of the containment. If a loss-of-coolant accident occurred, each of these valves would be required to open to supply coolant to the RPV. The motor-operated gate valves are automatically opened by their appropriate signals, and the check valves are opened by the coolant flow in the line. The opening capability of the check valve can be tested by monitoring flow through the valve into the reactor vessel.

Control Rod Drive Lines

The control rod drive system, has two types of lines to the RPV; the insert and withdraw lines that penetrate the drywell and connect to the control rod drive.

The control rod drive insert and withdraw lines can be isolated by the solenoid valves outside the primary containment. These lines that extend outside the primary containment are small, and terminate in a system that is designed to prevent out-leakage. Solenoid valves normally are closed, but open on rod movement and during reactor scram. In addition, a ball check valve located in the control rod drive flange housing automatically seals the insert line in the event of a break.

RHR and RCIC Head Spray Lines

The subject penetration(s) meet the alternative primary containment isolation criteria of NUREG 0800 "Standard Review Plan for the review of Safety Analysis Reports for Nuclear Power Plants" (SRP) instead of the explicit requirements of GDC 55.

The RHR and RCIC head spray lines meet outside the containment to form a common line which penetrates the drywell and discharges directly into the reactor pressure vessel. The testable check valve inside the drywell is normally closed. The testable check valve is located as close as practicable to the reactor pressure vessel. Three types of valves, a testable check valve, a normally closed motor-operated remote manual gate valve, and a normally closed motor-operated automatic globe valve, are located outside the containment. The check valve assures immediate isolation of the containment in the event of a line break. The globe valve on the RHR line receives an automatic isolation signal while the gate valve on the RCIC line is remote manually actuated to provide long-term leakage control.

Standby Liquid Control System Lines

The standby liquid control system line penetrates the drywell and connects to the reactor pressure vessel. In addition to a simple check valve inside the drywell, a check valve together with an explosive actuated valve are located outside the drywell. Since the standby liquid control line is a normally closed, nonflowing line, rupture of this line is extremely remote. The explosive actuated valve, though, functions as a third isolation valve. This valve provides an absolute seal for long-term leakage control as well as preventing leakage of sodium pentaborate into the reactor pressure vessel during normal reactor operation.

Reactor Water Cleanup System

The reactor water cleanup (RWCU) pumps, heat exchangers, and filter demineralizers are located outside the primary containment. The return line from the filter demineralizers connects to the feedwater line outside the containment between the outside containment feedwater check valve and the outboard motor-operated gate valve. Isolation of this line is provided by the feedwater system check

valve inside the containment, the feedwater check valve outside the containment, and a motor-operated gate valve which provides a long term isolation capability.

During the postulated loss-of-coolant accident, it is desirable to maintain reactor coolant makeup. For this reason, valves which automatically isolate upon signal are not included in the design of the system. Consequently, a third valve is required to provide long-term leakage control. Should a break occur in the reactor water cleanup return line, the check valves would prevent significant loss of inventory and offer immediate isolation, while the outermost isolation valve would provide long-term leakage control.

Recirculation Pump Seal Water Supply Line

The recirculation pump seal water line extends from the recirculation pump through the drywell and connects to the CRD supply line outside the primary containment. The seal water line forms a part of the reactor coolant pressure boundary, therefore the consequences of failing this line have been evaluated. This evaluation shows that the consequences of breaking this line is less severe than that of failing an instrument line. The recirculation pump seal water line is 3/4-inch Class B from the recirculation pump through the second check valve (located outside the containment). From this valve to the CRD connection the line is Class D. Should this line be postulated to fail and either one of the check valves is assumed not to close (single active failure), the flow rate through the broken line has been calculated to be substantially less than that permitted for a broken instrument line. Therefore, the two check valves in series provide sufficient isolation capability for postulated failure of this line.

RHR Shutdown Cooling Return Line

The subject penetration(s) meet the alternative primary containment isolation criteria of NUREG 0800 “Standard Review Plan for the review of Safety Analysis Reports for Nuclear Power Plants” (SRP) instead of the explicit requirements of GDC 55.

The shutdown cooling return lines are connected to the reactor recirculation pump discharge lines. The isolation valve arrangement on these lines is identical to that on the ECCS lines connected to the RPV. However, the motor-operated valve outside containment closes automatically upon receipt of an isolation signal.

RHR Shutdown Cooling Suction Line

The penetration (M-7) has been protected by a relief valve mounted between the inboard automatic isolation and the containment penetration. This relief valve was added in response to NRC Generic Letter GL 96-06 concerns for isolated line overpressurization during a LOCA.
Because the RHR Shutdown Cooling piping up to and including the outer containment penetration automatic isolation valve is part of the RCPB, the penetration configuration must meet GDC 55.

Reactor Recirculation System Sample Line

The Reactor Recirculation sample line is a 3/4" line that is an extension of the RCPB to the outboard isolation valve. The containment penetration (M-36) has an automatic isolation inside containment and an automatic isolation outside containment. A 3/4" bypass line with a check valve has been added around the inboard isolation valve in response to Generic Letter 96-06. The check valve will open to relieve penetration overpressurization following a LOCA. Manual valves between the check valve and the RR 24" process line will be maintained locked open, when required for overpressure protection, to assure a vent path for overpressure protection.

The two automatic valves and the inboard check valve meet the requirements of GDC 55.

6.2.4.2.2 Evaluation Against General Design Criterion 56

Primary Containment Chilled Water System

The Primary Containment Chilled Water System (PCCW) consists of two independent trains of cooling for the primary containment atmosphere. Each train penetrates the containment with a supply and return line. Each line has an inboard and an outboard automatic isolation valve. Each penetration (M-25, M-27, M-28, M-26) has been protected by a relief valve mounted between the inboard automatic isolation and the containment penetration. These relief valves were added in response to NRC Generic Letter GL 96-06 concerns for isolated line overpressurization during a LOCA.

The penetration configuration must meet GDC 56.

RCIC Turbine Exhaust Vacuum Breaker Line
Minimum Flow Bypass

The RCIC turbine exhaust line is provided with a vacuum breaker system to prevent condensation of the exhaust steam from inducing a vacuum in the line. The vacuum relief line connects the turbine exhaust line to the suppression chamber atmosphere. Two check valves in-series in the line prevent steam from exhausting to the vapor space above the pool, and two motor-operated globe valves, one on either side of the aforementioned check valves, provide remote manual isolation capability for the RCIC turbine exhaust vacuum breaker line.

Combustible Gas Control and Post-LOCA Atmosphere Sampling Lines

The post-LOCA sampling system lines which penetrate the containment and connect to the drywell and suppression chamber air volume are each equipped with

a single divisional fail-open, solenoid operated isolation valve located outside and as close to the containment as possible. The combustible gas control system lines which penetrate the containment are equipped with two normally closed motor-operated valves in series, located outside containment, remote manually actuated from the control room. These valves provide assurance of isolating these lines in the event of a break and also provide long-term leakage control. In addition, the piping is considered an extension of containment boundary since it must be available for long-term usage following a design basis loss-of-coolant accident, and, as such, is designed to the same quality standards as the primary containment. Thus, the need for isolation is conditional.

Containment Vent and Purge and Containment Drain Lines

The drywell and suppression chamber vent and purge and containment drain lines have test isolation capabilities commensurate with the importance to safety of isolating these lines. Each line has two normally closed, instrument air powered, air cylinder actuated valves located outside the primary containment. The air cylinders are operated by solenoid valves connected to the control logic. Containment isolation requirements are met on the basis that the purge and drain lines are normally closed, low-pressure lines constructed to the same quality standards as the containment and meet the Branch Technical Position CSB 6-4. These isolation valves are interlocked to preclude opening of the valves while a containment isolation signal exists. Furthermore, the consequences of a break in these lines result in no significant safety consideration.

Drywell and Suppression Chamber Air Sampling Lines

The air sampling lines are used for continuously drawing containment air during normal operation as part of the leak detection system. These lines are equipped with two normally open, solenoid operated, spring to close valves in series, located outside and as close as possible to the containment. This manner of routing the system piping reduces the number of containment penetrations and minimizes the potential pathways for radioactive material release. In addition, the piping upstream of the air sampling isolation valves is considered an extension of the containment since it must be available for long-term usage following a design basis loss-of-coolant accident. The piping is part of the post-LOCA atmosphere sampling system, and as such, is designed and fabricated to the same quality standards as the containment. Containment isolation requirements are met on the basis that these lines are low-pressure lines constructed to the same quality standards as the containment furthermore, the consequences of a break in these lines result in no significant safety consideration.

Service Air and Clean Condensate Supply Lines

The Service Air and Clean Condensate supply lines, which penetrate the containment, provide air and water service connectors inside the drywell during reactor shutdown and outages. These lines are equipped with two manually operated valves which are locked closed during reactor operations. In addition, each line is equipped with a spool piece which is removed and respective blank flanges installed during reactor operations. The valves and spool pieces are located outside of and as close as possible to the containment. This manner of routing the system piping reduces the number of containment penetrations. Since these lines are isolated during reactor operations, the potential pathways for radioactive material release is minimized. Furthermore, the consequences of a break in these lines result in no significant safety consideration.

Reactor Building Closed Cooling Water System

The Reactor Building Closed Cooling Water System (RBCCW) inside containment consists of a closed loop providing cooling for the reactor recirculation pump heat loads and penetration heat loads. The system penetrates the containment with a supply and return line. Each line has an inboard and an outboard automatic isolation valve. The containment isolation signals to these valves can be overridden by using key locked bypass switches. Each penetration (M-16, M-17) has been protected by a relief valve mounted between the inboard automatic isolation and the containment penetration. These relief valves were added in response to NRC Generic Letter GL 96-06 concerns for isolated line overpressurization during a LOCA.

The penetration configuration must meet GDC 56.

Primary Containment Chilled Water System

The Primary Containment Chilled Water System (PCCW) consists of two independent trains of cooling for the primary containment atmosphere. Each train penetrates the containment with a supply and return line. Each line has an inboard and an outboard automatic isolation valve. Each penetration (M-25, M-27, M-28, M-26) has been protected by a relief valve mounted between the inboard automatic isolation and the containment penetration. These relief valves were added in response to NRC Generic Letter GL 96-06 concerns for isolated line overpressurization during a LOCA.

The penetration configuration must meet GDC 56.

6.2.4.2.3 Evaluation Against General Design Criterion 57

Lines penetrating the primary containment for which neither Criterion 55 nor Criterion 56 govern comprise the closed system isolation valve group.

Influent and effluent lines of this group are isolated by automatic or remote manual isolation valves located as closely as possible to the containment boundary.

ECCS Pump Test Lines and Minimum Flow Bypass Lines

The LPCS, HPCS, and RHR pump test and minimum flow bypass lines have isolation capabilities. All the pump test lines are equipped with normally closed motor-operated globe valve outside the containment that is opened only during pump testing. The RHR pump test lines discharge below the surface of the suppression pool. Thus, the lines are not directly open to the containment atmosphere, since the pool acts to seal the discharge from the containment. The LPCS and HPCS lines discharge into the air space above the suppression pool surface. All the test lines are low-pressure lines, constructed to the same quality standards as the containment. All valves can be remote manually operated from the main control room, and close automatically on a system start signal.

The minimum flow bypass line on the HPCS has a normally closed motor-operated gate valve located outside the containment while the LPCS and RHR are minimum flow bypass lines equipped with a normally open motor-operated gate valve. A high speed valve is utilized to assure that pump minimum flow requirements are met. The LPCS and RHR valves are closed when adequate flow in the pump discharge lines is established. The minimum flow bypass lines connect into the associated pump test lines outside the containment. This reduces the number of penetrations through the primary containment, thus minimizing the potential pathways for radioactive material release.

RCIC Turbine Exhaust, Vacuum Pump Discharge and RCIC Pump Minimum Flow Bypass

The RCIC turbine exhaust and vacuum pump discharge lines which penetrate the containment and connect to the suppression chamber are equipped with a normally open, motor-operated, remote manually actuated valve located as close to the containment as possible. The RCIC turbine exhaust line motor-operated isolation valve is a gate valve and the RCIC vacuum pump discharge line moter-operated isolation valve is a globe valve. In addition, there is a simple check valve upstream of the motor-operated valve which provides positive actuation for immediate isolation in the event of a break upstream of this valve. The gate valve in the RCIC turbine exhaust is designed to be locked open in the control room and interlocked to preclude opening of the inlet steam valve to the turbine while the turbine exhaust valve is not in a full open position. The RCIC vacuum pump discharge line is also normally open but has no requirement for interlocking with the steam inlet valve to the turbine. The RCIC pump minimum flow bypass line is isolated by a normally closed motor-operated globe valve with a check valve installed upstream. This valve is controlled by sensors in the RCIC pump discharge line flow and pressure. The valve is also remote manually controlled from the main control room.

The RCIC turbine exhaust line is also provided with a vacuum breaker system to prevent condensation of the exhaust steam from inducing a vacuum in the line. The vacuum relief line connects the turbine exhaust line to the suppression chamber atmosphere.

Two check valves in-series in the line prevent steam from exhausting to the vapor space above the pool, and two motor-operated globe valves provide remote manual isolation capability for the vacuum breaker line.

ECCS and RCIC Safety/Relief Valves

The safety/relief valves which serve the RHR shutdown cooling line located outside primary containment, RHR Pumps A and C suction lines, RHR Pumps A, B, and C discharge lines, RHR Heat Exchanger drain lines to the RCIC System, LPCS and HPCS suction drain lines, RHR Pumps A and B suction drain lines and discharge drain lines, RHR Pump C discharge drain line, LPCS Pump suction and pump discharge lines, and the HPCS Pump suction line and water leg pump discharge line, discharge water into the air space above the suppression pool surface. The safety/relief valve on RHR Pump B suction line discharges water below the suppression pool surface. The safety/relief valves on the RHR Heat Exchangers Shell Side and the RCIC steam supply lines to the RHR Heat Exchangers discharge steam below the suppression pool surface. The safety/relief valves are normally closed and provide a containment barrier in the lines. The thermal expansion safety/relief valve on the Unit 1 HPCS pump discharge line discharges water to the reactor building equipment drains and is normally closed. The thermal expansion safety/relief valve on the Unit 2 HPCS pump discharge line discharges water to the Unit 2 HPCS Pump Room and is normally closed. The safety/relief valves on the RCIC Lube Oil Cooler Supply Line, the RCIC System Pump suction line, and the RCIC Barometric Condenser discharge water to the reactor building equipment drains and are normally closed. Block valves cannot be added to the safety/relief valve discharge lines because they would preclude proper operation of the safety/relief valves, and are prohibited by the piping codes.

ECCS and RCIC Pump Suction Lines

The RHR, RCIC, LPCS, and HPCS suction lines contain motor-operated, remote manually actuated, gate valves which provide assurance of isolating these lines in the event of a break. These valves also provide long-term leakage control. In addition, the suction piping from the suppression chamber is considered an extension of containment since it must be available for long-term usage following a design basis loss-of-coolant accident, and as such is designed to the same quality

standards as the containment. Thus, the need for isolation is conditional since the ECCS pumps take suction from the suppression pool in order to mitigate the consequences of LOCA. Therefore, their proper position for performing their safety fuction is open, not closed.

It should also be noted that the suction line of the ECCS pumps serves as the source of supply to the water leg pumps, which keep the ECCS discharge lines filled to avoid hydrodynamic effects on ECCS pump initiation. Isolating these water leg pumps from their supply source would degrade rather than improve the safe operation of the plant. However, the suction lines are provided with a motor-operated gate valve that can be remote manually closed from the control room, if required by a system line break or other highly unlikely event.

6.2.4.2.4 Miscellaneous

Compliance with regulatory guides is addressed in Appendix B.

The isolation valves have been designed against loss of function from missiles, jet forces, pipe whip, and earthquake. The containment isolation valves and valve operators have been designed to operate under normal plant and postulated accident conditions. The effects of radiation, humidity, pressure and temperature both inside and outside the containment, as defined in Chapter 3.0, have been accounted for in the valve design.

Containment isolation valves are provided with adequate mechanical redundancy to preclude common mode failures. The power supplies to the inboard isolation valves are provided from a separate electrical division than those that supply the outboard isolation valves. Therefore, a common mode failure in one electrical division would not prevent containment isolation. The vent and purge valves consist of Air Operated Valves and Motor Operated Valves. See Table 6.2-21 for specific valve characteristics.

A complete list of Primary Containment Isolation Valves is contained in Table 6.2-28.

A leak detection system has been provided to detect leakage for determining when to isolate the affected systems that require remote manual isolation. This leak detection system is described in Subsection 5.2.5.

The design provisions for testing the leakage rates of the containment isolation valves are shown in the valve arrangement drawings, Figure 6.2-31 as referenced in Table 6.2-21. The test connections indicated consist of a double-valved test line with provision for a pressure gauge attachment.

The design provision for testing the leakage rates of the containment isolation valves 2FC086 and 2FC115 is shown on valve arrangement drawing, Figure 6.2-31, Sheet 10C, Detail "AD". The test connection indicated consists of a single valve test line with a provision for a pressure gauge attachment.

6.2.4.3 Design Evaluation

The main objective of the containment isolation system is to provide protection by preventing releases to the environment of radioactive materials. Redundancy is provided in design aspects to satisfy the requirement that an active failure of a single valve or component does not prevent containment isolation: Mechanical components are redundant, as shown by the isolation valve arrangements.

Electrical redundancy is provided in isolation valve arrangements to eliminate dependence on a single power source to attain isolation. Electrical cables for isolation valves in the same process line have been routed separately. Cables have been selected based upon the specific environment to which they will be subjected.

Provisions ensure that the position of all nonpowered isolation valves is maintained. For all powered valves, the position is indicated in the main control room. A discussion of the instrumentation and controls associated with the isolation valves is given in Chapter 7.0.

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

6.2.4.4 Tests and Inspections

A discussion of the testing and inspection pertaining to isolation valves is provided in Subsection 6.2.6, the Technical Specifications, and Table 6.2-21.

6.2.5 Combustible Gas Control in Containment

In order to assure that the containment integrity is not endangered due to the generation of combustible gases following a postulated LOCA, systems for controlling the relative concentrations of such gases are provided within the plant. The system includes subsystems for mixing the containment atmosphere, monitoring hydrogen concentration, reducing combustible gas concentrations, and, as a backup, purging. The hydrogen recombining function of the hydrogen recombiners is abandoned in place.

6.2.5.1 Design Bases

The hydrogen recombining function of the hydrogen recombiners is abandoned in place. The valves that provide RHR cooling water to the hydrogen recombiners are also abandoned in place in the closed position. The blower and associated piping are not abandoned and remain operational to maintain the drywell mixing function. The design basis information for the hydrogen recombination function remains for historical reference.

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

a. A double-ended rupture of a main recirculation line results in the most rapid coolant loss and reactor depressurization, with the coolant being discharged from both ends of the break. The noncondensable gas initially in the drywell is forced into the suppression chamber during the RPV depressurization phase. This transfer process takes place through downcomers that connect the drywell and suppression chambers. The postulated metal-water reaction begins in the core region and is assumed to produce hydrogen immediately after the recirculation pipe breaks. The reaction would last 2 minutes during which 0.945% of the active Zircaloy fuel cladding has reacted. The radiolysis of the coolant in the core region, water sump on the drywell floor and suppression pool also is assumed to begin immediately. The hydrogen and oxygen thus generated will evolve to drywell and suppression chamber atmospheres.

b. The combustible gas control system has the capability for monitoring the hydrogen concentration in drywell and suppression chamber and alarming as the hydrogen concentration reaches 4%. It also has the capability of mixing the atmospheres of both drywell and suppression chamber. It also will control the combustible gas concentrations in the primary containment without reliance on purging and without the release of radioactive material to the environment.

c. The primary systems for combustible gas control, including measuring, meet the design, quality assurance, redundancy, energy source, and instrumentation requirements for an engineered safety feature system according to Appendix A of 10 CFR 50.

d. The combustible gas control system will be activated after a LOCA in time to assure that the hydrogen concentration does not exceed 4 volume percent of hydrogen in either the drywell or wetwell atmospheres. In addition, the LSCS containment is nitrogen inerted to

an oxygen concentration of 4% by volume. This is below the combustible limit of oxygen in hydrogen but still provides enough oxygen to react with all the hydrogen that would be produced by the metal water reaction.

e. One recombiner system is provided for each nuclear unit. Each recombiner is capable of being cross-connected to the other unit to provide 100% redundancy. The recombiners are located outside of the primary containment in an accessible area and, therefore, routine maintenance, testing and/or inspection can be performed during normal plant operation or shutdown conditions.

f. The components of the combustible gas control system are protected from missiles and pipe whip to assure proper operation under accident conditions as required for safety-related systems. The system has been designed to perform in the event of failure of any one of its active components.

g. The combustible gas control systems are designed as Seismic Category I devices. As previously mentioned, the units are capable of being cross-connected to provide redundancy and are further capable of withstanding the temperature and pressure transients resulting from a LOCA. All components that can be subjected to containment atmosphere are capable of withstanding the humidity, temperature, pressure, and radiation conditions in the containment following a LOCA.

h. The combustible gas control system is designed to remain operable in the postaccident environment in the reactor building. Components subjected to the reactor containment postaccident environment are likewise designed for those conditions.

i. The combustible gas control system recombiner units are located outside of the primary containment in an accessible area. They can be inspected or tested during normal plant operation or during shutdown conditions.

j. The hydrogen recombiner units are fixed units that are permanently installed; therefore, it is not necessary to have the ability to transport them.

k. The recombiner units are remotely started from the control room and the local control panel in the auxiliary electric equipment room. They are designed such that there are no local operating adjustments required on a unit operating in a post-LOCA environment. This fact eliminates the necessity of biological shielding.

6.2.5.2 System Design

The combustible gas control system consists of four subsystems: a mixing system, a hydrogen monitoring system, two hydrogen recombiners, and a purge system. The design features of these four systems are described in the following sections.

The hydrogen recombining function of the hydrogen recombiners is abandoned in place. The valves that provide RHR cooling water to the hydrogen recombiners are also abandoned in place in the closed position. The blower and associated piping are not abandoned and remain operational to maintain the drywell mixing function. The design basis information for the hydrogen recombination function remains for historical reference.

Hydrogen Mixing System

The function of the mixing subsystem is to ensure that local concentrations with greater than 4% hydrogen cannot occur within the primary containment following a LOCA.

The atmospheres of both drywell proper and suppression chamber area, each of which is a single compartment, are well mixed. The mixing is achieved by natural convection processes. Natural convection occurs as a result of the temperature difference between the bulk gas space in the vessel and the containment wall. The natural convective action is enhanced by the momentum of steam emitted from the point of rupture. There are two interior subcompartments where gases may not achieve thorough mixing with the bulk containment atmosphere. The drywell head area, which is for reactor vessel refueling purposes, is one such subcompartment. The other is the control rod drive area immediately below the reactor pressure vessel. The physical arrangements and/or location of the monitoring system and the hydrogen recombiner system are such that concentrations above the 4% limit of combustible gases will not occur.

The atmosphere between the drywell and suppression pools will be mixed during the depressurization phase of the LOCA. The hydrogen recombiner units will also serve to affect mixing between these two compartments. The hydrogen recombiner will take suction on the drywell and discharge to the suppression pool. This will in turn cause the atmosphere from the suppression pool to circulate into the drywell via the vacuum breaker lines.

The monitoring system will alert the operator of the concentration within these subcompartments and the positions of the effluent and suction points of the recombiner will preclude the building of concentrations above the limit in these areas as well as the drywell and wetwell proper.

Hydrogen Monitoring System

The hydrogen monitoring system forms a part of the primary containment monitoring system which is discussed in Subsection 7.5.2.

Hydrogen Recombiner System

The concentration of combustible gases in the primary containment (drywell and suppression pool areas) following a LOCA is controlled by the hydrogen recombiner system. The combustible gas control system contains one hydrogen recombiner per reactor unit. The hydrogen recombiner is located outside of the primary containment. The amount of Hydrogen in the effluent gas being returned to the wetwell shall not exceed 0.1% by volume. The system will process the primary containment atmosphere at a rate of at least 125 scfm using a blower to supply containment gases to the recombiner. The recombination process

takes place within the recombiner as a result of an exothermic reaction. The steam is then cooled and the resulting water and remaining gases are returned to the primary containment. Suction is taken from the drywell area, and the discharge is returned to the suppression pool area above water level.

The hydrogen recombiner unit is skid mounted and is an integral package. All pressure containing equipment including piping between components is considered as an extension of the containment and, therefore, is designed as ASME III Class 2. The skid and the equipment mounted on it are designed to meet Seismic Category I requirements. The hydrogen recombiner system is designed to accommodate conditions present in the containment (temperature and pressure) following a LOCA event. Piping and instrumentation for the system are shown in Drawing No. M-130. The hydrogen recombiner unit, which requires a 1-2 hour warmup period, is initiated manually from the control room and the local control panel in the aux. electric equipment room. It is initiated prior to primary containment hydrogen concentration reaching 3 volume percent which occurs approximately 5 hours after the accident. Based on the original core loading, the time at which containment hydrogen generation reaches 4 volume percent varies with fuel types located in the core. However, this is acceptable based on Design Basis described in Section 6.2.5.1.d. Once placed in operation, the system continues to operate until it is manually shut down when an adequate margin below the hydrogen concentration design limit is reached. The operation of the system can be tested from the control room or the auxiliary equipment room. The test consists of energizing the blower and heaters and observing system operation to see if components are performing properly. Flow and pressure measurement devices are periodically calibrated.

The hydrogen recombiner system is serviced by electrical power and cooling water systems, which are placed in operation concurrent with a loss-of-coolant accident. Cooling water required for the operation of the system is taken from the residual heat removal system. The cooling water is utilized to cool the water vapor and the residual gases leaving the recombiner prior to returning them to the containment. All hydrogen recombiner unit cooling water is returned to the suppression pool.

Each recombiner unit has the capability of serving either containment; therefore, there is 100% redundancy of all components and controls.

All functions and controls necessary to start the combustible gas control system are also located in the control room and in the auxiliary electric equipment room which is readily accessible from the control room.

6.2.5.3 Design Evaluation

The hydrogen recombining function of the hydrogen recombiners is abandoned in place. The valves that provide RHR cooling water to the hydrogen recombiners are also abandoned in place in the closed position. The blower and associated piping are not abandoned and remain operational to maintain the drywell mixing function. The design basis information for the hydrogen recombination function remains for historical reference.

6.2.5.3.1 General

In evaluating the combustible gas control system design, it was found necessary to consider:

a. hydrogen generated in the post-LOCA environment,

b. resultant drywell and containment concentrations, and

c. the functional requirements of the combustible gas control system.

The following analytical results are provided:

a. The beta, gamma, and beta plus gamma energy release rates plotted as functions of time (Figure 6.2-32).

b. The integrated beta, gamma and beta plus gamma energy release plotted as functions of time (Figure 6.2-33).

c. The integrated production of combustible gas within the containment (drywell and suppression chamber) plotted as a function of time for each source (i.e., metal-water reaction and radiolysis) (Figure 6.2-34).

d. The concentration of combustible gas in the drywell and suppression chamber plotted as a function of time, if uncontrolled (Figure 6.2-35). This curve establishes the basis for activation of the combustible gas control system.

e. The combustible gas concentration in the containment (drywell and suppression chamber) plotted as a function of time with (125 scfm) 100% recombiner capacity initiated at 5 hours after LOCA (Figure 6.2-36).

6.2.5.3.2 Sources of Hydrogen

Short-Term Hydrogen Generation

In the period immediately after the LOCA, hydrogen is generated by both radiolysis and metal-water reaction. However, in evaluating short-term hydrogen generation, the contribution from radiolysis is insignificant when compared to the hydrogen generated by the metal-water reaction. The only metal-water reaction considered to be significant is reaction of water with the zirconium fuel cladding which produces hydrogen by the following reaction:

$$Zr + 2H_2O \rightarrow ZrO_2 + 2H_2$$

Based on loss-of-coolant accident calculational procedures and the analyses of emergency core cooling system (ECCS) performance in conformance with 10 CFR 50.46 and Appendix K, the extent of the above chemical reaction is estimated to be 0.1% of the fuel cladding material. However, the metal-water reaction-generated hydrogen based on a core-wide penetration of 0.00023 inches for 764 bundles with each bundle containing 101 pounds of zirconium in the active fuel cladding, results in a 0.945% metal-water reaction. Therefore, 0.945% of fuel cladding, which is greater than five times the maximum amount calculated in accordance with 10 CFR 50.46, is assumed to react with water to produce hydrogen. The duration of this reaction is assumed to be 120 seconds with a constant reaction rate. The resulting hydrogen is assumed to be uniformly distributed in the drywell containment. This assumption is supported by the test data reported in BNWL 1592 of July 1971. Figure 6.2-34 presents the accumulated hydrogen generation as a result of this chemical reaction.

Long-Term Hydrogen Generation

Hydrogen is also produced by decomposition of water due to absorption of the fission product decay energy immediately after LOCA.

$$2H_2O \rightarrow 2H_2 + O_2$$

Generation of hydrogen and oxygen due to radiolysis of coolant water is an important factor in determining the long-term gas mixture composition within the containment compartments. Conservative assumptions were used to determine the fission product distribution model that applies after the accident and, therefore, the hydrogen generation rates. The incore radiolysis contributes hydrogen to the drywell, and radiolysis of the suppression pool water contributes hydrogen directly to the suppression chamber. Hydrogen is also discharged from the radiolysis of sump water on drywell floor into the drywell atmosphere. The total decay energy utilized in the analyses was based on American Nuclear Society Standard ANS 5.1-1979 multiplied by a factor of 1.2, conservatively assuming a 1000-day reactor

operating time at constant full power level to determine the fission product buildup. Halogen and noble gas inventories were determined from TID-14844.

Hydrogen can also be formed by corrosion of metals in the containment. The significant portion of this source is from the corrosion of zinc and aluminum. Since the spray system uses only demineralized water for the purpose of reducing temperature and pressure inside the drywell, the corrosion of aluminum and zinc will contribute a negligible amount of hydrogen to the containment atmosphere. Hydrogen is, during normal operation of the plant, dissolved in the primary system water. Figure 6.2-35 presents the accumulated hydrogen and oxygen generation from both chemical reaction and radiolysis decomposition of water.

6.2.5.3.3 Accident Description

A complete description of the post-LOCA conditions is found in Subsection 6.2.1 and Section 6.3.

Following the postulated LOCA, the postulated metal-water reaction begins in the core region and is assumed to produce hydrogen immediately after the recirculation pipe breaks. The reaction lasts 2 minutes during which 0.945% of the active zircaloy fuel cladding reacts. The radiolysis of the coolant in the core region, water sump on the drywell floor and suppression pool is assumed to begin immediately. The hydrogen and oxygen thus generated will evolve to drywell and suppression chamber atmospheres. The hydrogen concentration in the drywell would, after about 15 hours, approach the flammability limit if uncontrolled. The hydrogen recombiner system is manually activated before the hydrogen concentration reaches 3 volume percent. The recombiner system takes gases from the drywell atmosphere, recombines the hydrogen with oxygen to form water vapor, and returns the resulting cooled water and remaining gases to the suppression chamber. The pressure buildup in the suppression chamber due to the operation of recombiner system taking suction on the drywell and discharging to the suppression pool will cause the opening of the vacuum breaker valves between the drywell and suppression chamber. As a result, the flow of the gas mixture from the wetwell to the drywell will balance the negative pressure differential between two volumes and will also result in lower concentrations due to the influx of the wetwell gases.

6.2.5.3.4 Analysis

Based on the above hydrogen sources and the accident description, the hydrogen concentration in the drywell and suppression chamber is calculated as a function of time. In formulating the model of the Mark II containment for these calculations, a conservative assumption is made, namely the interchange of mass between the drywell and the suppression chamber through downcomers which takes place during blowdown process is neglected, that is, no hydrogen is removed from the drywell except through the recombiner system. This assumption is conservative, as

it results in a shorter time for the drywell hydrogen concentration to reach the flammability limit. Furthermore, the hydrogen and oxygen gases can flow back to the drywell from suppression chamber through vacuum breakers due to pressure increase in the suppression chamber by the operation of the recombiner system.

Table 6.2-22 gives all of the necessary parameters used to determine the amount of hydrogen generation in the LSCS analysis. The results of the analyses are presented in Figures 6.2-35 and 6.2-36. It was determined that the uncontrolled hydrogen concentration in the drywell eventually reaches 4% by volume (dry basis) approximately 15 hours after the LOCA. The suppression chamber hydrogen concentration was determined to be 3.0% by volume due to radiolytic hydrogen generation. Prior to the drywell concentration reaching 3% by volume, a recombiner system is activated. A single system is designed to keep the hydrogen concentration below 4% by volume at all times until radiolytic generation has ceased. The performance of the recombiner system, which is initiated 5 hours after LOCA, is shown in Figure 6.2-36. The hydrogen concentration is 3.0% by volume at the time of initiation. Thus, the use of a single 125 scfm recombiner system provides effective control of hydrogen concentration and, therefore, would prevent the formation of combustible gas mixture in both drywell and suppression chamber.

6.2.5.4 Testing and Inspections

Each active component of the combustible gas control system is testable during normal reactor power operation.

The combustible gas control systems and the containment purge system will be tested periodically to assure that they will operate correctly. Preoperational tests of the combustible gas control system are conducted during the final stages of plant construction prior to initial startup (Chapter 14.0). These tests assure correct functioning of all controls, instrumentation, recombiners, piping, and valves. System reference characteristics, such as pressure differentials and flow rates, are documented during the preoperational tests and are used as base points for measurements in subsequent operational tests.

6.2.5.5 Instrumentation Requirements

The instrumentation provisions for actuating the combustible gas control system and monitoring the system are described in Subsection 7.3.5.

6.2.6 Containment Leakage Testing

This section presents the testing program for the reactor containment, containment penetrations and containment isolation barriers that comply with the requirements of the General Design Criteria and Appendix J to 10 CFR 50. Each of the tests

described in this Subsection was performed as a preoperational and will be performed as a periodic test.

6.2.6.1 Containment Integrated Leakage Rate Test

Following the completion of the construction, repair, inspection, and testing of welded joints, penetrations, and mechanical closures including the satisfactory completion of the structural integrity tests as described in Subsection 3.8.1.7, a preoperational containment leakage rate test was performed to verify that the actual containment leak rate does not exceed the design limits. In order to ensure a successful integrated leak rate test, local leakage tests (Type B and C tests) were performed on penetrations and isolation valves, and repairs are made, if necessary, to ensure that leakage through the containment isolation barriers does not exceed the design limits.

An integrated leakage rate test is then performed on the entire containment in order to determine that the total leakage (exclusive of MSIV leakage) through containment isolation barriers does not exceed the maximum allowable leakage rate of 1.0% per day at the calculated peak containment internal pressure at 42.6 psig. The pertinent test data, including test pressures and acceptance criteria, is presented in Table 6.2-23.

Pretest requirements have been described in the preoperational test abstract included in Chapter 14.0 of the FSAR. As stated therein, power operated isolation valves will be closed by their actuators prior to the start of the integrated leakage rate test.

During the integrated leak rate test the containment systems are configured as follows;

a. Reactor building closed cooling water - lined up for normal operation; isolation valves closed and system filled.

b. Primary containment chilled water - lined up for normal operation; isolation valves closed and system filled.

c. Residual heat removal - One loop lined up in shutdown cooling mode. Other loops lined up in low-pressure coolant injection standby mode and isolated, containment and suppression pool spray flow paths isolated, full flow test lines isolated, reactor head cooling flow path isolated, minimum flow isolated, shutdown cooling discharge line isolated on standby system and condensate discharge from RHR heat exchangers shell side flow path isolated; system filled. May be lined up in normal standby injection mode.

d. Low-pressure core spray - system filled and isolated. May be lined up in normal standby injection mode.

e. High-pressure core spray - system filled and isolated. May be lined up in normal standby injection mode.

f. Reactor core isolation cooling - isolation valves closed; RCIC condensate filled and isolated. RCIC full flow test return line to suppression pool filled and isolated.

g. Reactor water cleanup - suction line filled and isolated; return line filled and isolated.

h. Standby liquid control - lines filled and isolated.

i. Control rod drive - system filled. Vented outboard of HCU directional control valves.

j. Reactor recirculation system - pumps off, system filled.

k. RPV and primary containment instrumentation - lines filled and vented to containment instrumentation to the RPV or drywell will be opened.

l. Neutron monitoring system (TIP) - TIPs will be fully withdrawn and the ball valves closed.

m. Floor and equipment drains - sumps pumped or drained down to low water level, isolation valves closed.

n. Clean condensate - drained and vented, isolation valves closed, spool piece removed and blind flange installed or filled and isolated and system leakage added to type A result.

o. Service air - vented, isolation valves closed, spool piece removed and blind flange installed.

p. Feedwater - filled and isolated.

q. Main steam - filled, isolation valves closed.

r. Containment monitoring - post-LOCA monitoring system open to containment, pumps off, valves open; drywell monitoring and sampling system isolated, pumps off.

s. Post-LOCA hydrogen control - lined up for unit operation, isolation valves open or isolated and system leakage added to type A result.

t. Primary containment instrument air - all accumulators vented, isolation valves closed.

u. Fuel Pool Cooling - Cycled Condensate to Refueling Bellows filled and isolated, Reactor Well Drain filled and isolated.

v. All accessible liner leak test channel plugs are verified installed.

The Type C leak rates for the following penetrations are added to the Type A test results on a Minimum-Path Basis:

a. reactor building closed cooling water,

b. primary containment chilled water,

c. RHR shutdown cooling suction,

d. reactor core isolation cooling steam supply,

e. reactor water cleanup suction,

f. reactor water sample,

g. floor and equipment drains,

h. inboard MSIV drain,

i. Feedwater Lines,

j. RCIC Full Flow Test Return Line to Suppression Pool.

k. Cycled Condensate to Refueling Bellows

l. Reactor Well Drain

Measures will be taken to ensure stabilization of the containment conditions prior to containment leakage rate testing.

The test method utilized is the absolute method, as described in ANSI/ANS 56.8-1994. The test procedure, test equipment and facilities, period of testing, and verification of leak test accuracy also follow the recommendations of ANSI/ANS 56.8-1994.

The acceptance criteria for the preoperational containment integrated leakage rate test are in compliance with the criteria given in Appendix J of 10 CFR 50. except as

noted below. Structural verification test acceptance criteria are described in Subsection 3.8.1.7.

The acceptance criteria for the periodic containment integrated leakage rate test are in compliance with the criteria given in 10CFR50 Appendix J Option B, NRC Reg Guide 1.163, NEI-94-01, Rev. 0, and ANSI/ANS 56.8-1994. The As-Found Type A test leakage must be less than the acceptance criterian of 1.0 La (Primary Containment overall leakage rate acceptance criterion). During the first unit startup following testing (prior to entering a mode where containment integrity is required) the As-Left Type A leakage rate shall not exceed 0.75 La.

6.2.6.2 Containment Penetration Leakage Rate Test

Containment penetrations whose design incorporates resilient seals, gaskets, or sealant compounds; air lock door seals, equipment and access doors with resilient seals or gaskets; and other such penetrations received a preoperational and will be periodically leak tested in accordance with Appendix J of 10 CFR 50 except as noted in the following paragraph.

The following penetrations were preoperationally and will be periodically tested to Type B criteria:

a. equipment access hatch,

b. personnel air lock, by (when containment integrity is required, the personnel airlock should be tested within 7 days after each containment access except when the airlock is being used for multiple entries, then at least once per 30 days, by verifying seal leakage to be less than or equal to 5 scfh when the gap between the door seals is pressurized to greater than or equal to 10 psig - exception to 10 CFR 50 Appendix J) overall air lock leakage rate is less than or equal to 0.05 La when tested at greater than or equal to Pa.

c. drywell head,

d. suppression chamber access hatches,

e. CRD removal hatch,

f. electrical penetrations,

g. TIP penetration flanges, SA flange and MC flange,

h. Drywell to suppression pool vacuum breaker and associated manual isolation valves flanges and actuator seals,

i. Vent and purge isolation valve flanges, and packing

j. HPCS minimum flow line branch line 1(2)HP20C-2" Blind flanges

k. RCIC spectacle flange 1(2)E51-D316 blind flange half when required. See Table 6.2-21 note 49.

l. ECCS Relief Valves Discharging to Suppression Pool Inbound (Containment Side) Flanges.

It should be noted that no pipe penetrations are provided with expansion bellows. The containment penetration is an anchor point in the system, and the thermal movements have been accounted for on this basis. Therefore, no leakage rate testing of expansion bellows penetration assemblies will be required.

Test methods utilized to determine containment penetration leak rates are described as follows:

a. <u>Equipment Access, CRD Removal, and Suppression Chamber Acess</u>

The equipment access hatch has been furnished with a double-gasketed flange and bolted dished door, as shown in Figure 3.8-34. The CRD removal and suppression chamber access hatches have been furnished with a double-gasketed flange and bolted door. Provision is made to test pressurize the space between the double gaskets of the door flanges and the doors.

b. <u>Personnel Air Lock</u>

The personnel lock is constructed as a double-door, latched, welded steel vessel, as shown in Figure 3.8-33. The space between the air doors can be pressurized to peak containment pressure through the test connections provided. Each of the doors are provided with a test connection for pressurizing between the seals.

In addition, all four shaft seal assemblies are provided with a test connection to allow for individual shaft seal leak test.

c. <u>Drywell Head</u>

A double-gasketed seal and test tap, as shown in Figure 3.8-5, is provided for leak rate testing of the drywell head.

d. <u>Electrical Penetrations</u>

Each electrical penetration, as represented in Figure 3.8-21 and listed in Table 3.8-1 (with an "E" penetration number), is provided with a pressure gauge to monitor leakage. The double-gasketed and O-ring seals are provided with a test connection for leak rate testing.

e. Tip Penetration Flanges, Clean Condensate (MC) and Service Air (SA) Penetrations

Each TIP MC or SA penetration flange is provided with a double-gasketed seal and a test connection for type B leak testing.

f. Drywell to Suppression Pool Vacuum Breakers

Each drywell to suppression pool vacuum breaker has two double-gasketed flanges and a manual actuator O-ring and shaft seal. These seals are provided with test connections for leak testing. The Vacuum Breaker line manual isolation valves have a double-gasketed flange on the inboard or containment side provided with test connections for leak testing. The outboard flanges on the manual isolation valves are leak tested by pressurizing the entire vacuum breaker line and performing soap bubble test on the outboard flange. The stem seal or packing of these valves will be tested either locally or by primary containment pressurization and subsequent soap bubble inspection.

g. Vent and Purge Isolation Valves

Each inboard vent and purge valve has a double-gasketed flanged seal on its containment side. These seals are provided with test connections for leak testing. The stem packing of these valves is also provided with a test connection for packing leak test. See also Table 6.2-21 Note 41.

h. HPCS Minimum Flow Line Blind Flanges

One double-gasketed blind flange is installed on each of the HPCS minimum flow line branch connections 1(2)HP20C-2". These flanges are provided with a test connection for type B leak testing.

i. RCIC Spectacle Flange 1(2)E51-D316

The installed blind flange half of spectacle flange 1(2)E51-D316 is tested by pressurizing with air the upstream RCIC full flow test return line to Condensate Storage Tank and then check for leaks at the flange upstream gasket joint. Done when required per Table 6.2-21 note 49.

j. ECCS Relief Valves' Containment Side Flanges are Type B tested by one of the following methods: Test Port/Testable Gasket; Primary Containment Pressurization and subsequent soap bubble inspection; Special Test Equipment mounted over the flange thus pressurizing against the gasket.

   Test pressures are given in Table 6.2-23.

The acceptance criteria for the preoperational containment penetration leakage rate test is in compliance with the criteria given in Appendix J of 10 CFR 50. The periodic test acceptance criteria is established in accordance with the LaSalle County Station Local Leak Rate Test Program, and also is in agreement with Appendix J Option B of 10 CFR 50, NRC Regulatory Guide 1.163, Nuclear Energy Institute NEI-94-01 Rev. 0, and ANSI/ANS-56.8-1994.

6.2.6.3 Containment Isolation Valve Leakage Rate Test

Those containment isolation valves that are to receive a Type C test are so indicated in Table 6.2-21.

Test taps for leakage rate testing have been provided on the lines associated with the containment isolation valves. These taps are indicated on the valve arrangement drawings associated with Table 6.2-21. The test method is as described in Appendix J of 10 CFR 50. Test pressures are shown in Table 6.2-23.

The acceptance criteria for the leakage rate testing is given in Table 6.2-23 and the Primary Containment Leak Rate Testing Program.

6.2.6.4 Scheduling and Reporting of Periodic Tests

The periodic leakage test schedule is given in the LaSalle County Station Leak Rate Test Program.

6.2.6.5 Special Testing Requirements

The secondary containment will be tested as required by the Technical Specifications.

6.2.7 References

1. F. J. Moody, "Maximum Two-Phase Vessel Blowdown from Pipes," Topical Report APED-4827, General Electric Company, 1965.

2. A. J. James, "The General Electric Pressure Suppression Containment Analytical Model, (NEDO-10320), April 1971.

3. A. J. James, "The General Electric Pressure Suppression Containment Analytical Model," April 1971, Supplement 1, (NEDO-10320), May 1971.

4. K. V. Moore and W. H. Ratting, "RELAP 4-A Computer Program for Transient Thermal-Hydraulic Analysis, "ANCR-1127, Aerojet Nuclear Company, December 1973.

5. F. J. Moody, "Maximum Rate of a Single Component, Two Phase Mixture," Journal of Heat Transfer, Transactions, American Society of Mechanical Engineers, Vol. 87, No. 1, February 1965.

6. I. E. Idelchik, Handbook of Hydraulic Resistance, AEC-TR-6630, 1966.

7. "RELAP 4/MOD5 A Computer Program for Transient Thermal- Hydraulic Analysis of Nuclear Reactors and Related Systems," ANCR-NUREG-1335, Aerojet Nuclear Company, September 1976.

8. NEI 94-01, Rev. 0, July 26, 1995, Nuclear Energy Institute Industry Guideline for Implementing Performance-Based Option of 10CFR Part 50 Appendix J.

9. ANSI/ANS 56.8-1994, American National Standard for Containment System Leakage Testing Requirements.

10. GE Document EAS-49-0888, "Justification of Continued Operation With Increased Suppression Pool Temperature at LaSalle County Station," Revision 1, August 1988. (Proprietary)

11. Technical Specification Submittal Letter Sections 3.6.2.1 and 4.6.2.1, dated 10-07-88.

12. Amendment 67 for Unit 1 (Facility Operating License NFP-11), and Amendment 49 for Unit 2 (Facility Operating License NFP-18), dated July 7, 1989.

13. Calc. L001799, Rev. 0, "Assessment of Containment Line Base Mat Reactor Pedestal, Downcomer Bracing, Drywell Floor & Suppression Pool Columns for Suppression Pool Temperature Increase."

14. Calc. L001800, Rev. 0, "Assessment of Containment Wall for Suppression Pool Temperature Increase"

15. Calc. L001810, Rev. 0, "Impact of Increase in the Suppression Pool Temperature at LaSalle on Design Basis Suppression Pool Dynamic Loads."

16. Deleted

17. Calc. 3C7-0181-003, Rev. 3, "Suppression Pool Temperature Transient Studies"

18. General Electric Letter Report GE-NE-B13-01920-013, January 1998, "Current Suppression Pool Water Temperatures Following a Design Basis Accident for LaSalle County Station Units 1 and 2"

19. General Electric Report EAS-083-1188, "Elimination of the High Suppression Pool Temperature Limit for LaSalle County Station Units 1 & 2", dated November 1988.

20. General Electric Letter Report GE-NE-T23-00762-00-01, July 1998, "Evaluation of Peak Suppression Pool Temperature with Assumption of Feedwater Coastdown and Reduced RHR Flow Rate During Long-Term Containment Cooling"

21. Letter from J. A. Benjamin (ComEd) to U. S. NRC, “Request for a Change to the Technical Specifications, 'Vacuum Relief System'” dated August 6, 1999.

22. Letter from J. A. Benjamin (ComEd) to U. S. NRC, “Supplemental Information to Request for a Change to the Technical Specifications to Vacuum Relief System” dated November 15, 1999.

23. Letter dated December 21, 1999 from D. M. Skay to O. D. Kingsley, “Issuance of Amendments, approved amendment 138 for LaSalle Unit 1 and amendment 122 for LaSalle Unit 2.”

24. Licensing Topical Report, “Generic Guidelines for General Electric Boiling Water Reactor Power Uprate,” NEDC-31897P-A, May 1992.

25. LaSalle County Station Power Uprate Project, Task 400, "Containment System Response,” GE-NE-A1300384-02-01R1, Revision 1, October 1999 (and Task Report Changes based on Steam Plume Analysis, GE-LPUP-332, dated 5/4/2000).

26. General Electric Company, “General Electric Company Analytical Model for Loss-of Coolant Analysis in Accordance with 10CFR50 Appendix K,” NEDO-20566A, September 1986.

27. ComEd letter to NRC, "Response to Request for Additional Information License Amendment Request for Power Uprate Operation," dated 3/31/2000.

28. General Electric Company, NEDO-30832, “Elimination of Limit on Local Suppression Pool Temperature for SRV Discharge with Quenchers,” Class I, December 1984, (NRC approved version with NRC Safety Evaluation Report issued as NEDO-30832-A, Class I, May 1995).

29. General Electric Analysis of LaSalle Steam Plume Ingestion Potential, NSA 00-116, dated 3/29/2000.

30. LaSalle County Station Power Uprate Project, Task 401, "Annulus Pressurization,” GE-NE-A1300384-06-01, Revision 0, June 1999.

31. Design Analysis No. L-002874, Rev. 1, “LaSalle County Station Power Uprate Project Task 400: Containment System Response (GE-NG-A1300384-02-01 R3) Revision 3”.

32. EC #334017, Rev. 0, “Increased Cooling Water Temperature Evaluation to a new Maximum Allowable of 104°F.”

33. Design Analysis L-003352, Rev. 0, "Evaluation for GE Safety Communication SC06-01 Containment System Response (GEH 0000-0069-6598-R0)."

34. Design Analysis L-003509, Revision 0, "Evaluation of Appendix R, Station Blackout, Containment and Source Terms for LaSalle MUR Power Uprate," July 2010.

35. Design Analysis L-003566, Revision 0, "T1000 Series – S/U Test and Generic Applicability," July 2010.

36. EC #388666, Rev. 0, "Revise Design Analyses for UHS Temperature of 107°F"

37. Design Analysis L-003903, Revision 001, “Pool Swell Response (GEH 0000-0163-8881-R0),” December 2017.

38. EC #397610, Rev. 0, “Revised Pool Swell Response”

39. Safety Evaluation by the Office of Nuclear Reactor Regulation Related to Amendment No. 225 to Renewed Facility Operation License No. NPF-11 and Amendment No. 211 to Renewed Facility Operating License No. NPF-18, Exelon Generation Company LLC, LaSalle County Station Units 1 and 2, Docket No. 50-373 and 50-374.

TABLE 6.2-1
(SHEET 1 OF 2)

CONTAINMENT DESIGN PARAMETERS

	DRYWELL	SUPPRESSION CHAMBER
A. Drywell and Suppression Chamber		
1. Internal design pressure, psig	45	45
2. External design pressure, psig	5	5
3. Drywell deck design differential pressure, psid		
a) Downward	25	25
b) Upward	5	5
4. Design temperature, °F	340	275
5. Drywell (including vents) net free volume, ft^3	229,538	
6. Design leak ratio, %/day @ 45 psig	0.5	0.5
7. Suppression chamber free volume, ft^3		
a) minimum		164,800
b) maximum		168,100
8. Suppression chamber water volume		
a) Minimum, ft^3		128,800
b) Maximum, ft^3		131,900
9. Pool cross-section area, ft^2		
a) Water surface (excluding pedestal and drywell floor support columns)		4999
b) Total		5899
10. Pool depth (normal), ft		26.5

TABLE 6.2-1
(SHEET 2 OF 2)

	DRYWELL	SUPPRESSION CHAMBER
B. Vent System		
1. Number of downcomers		98
2. Internal downcomer diameter, in.		23.5
3. Total vent area, ft^2*		295
4. Downcomer submergence*		12 ft 4 in. (maximum)
5. Downcomer loss factor*		5.2

* The actual limiting area is 232 ft^2 based on the opening size through the downcomer protective covers (top hats). The corresponding loss factor is 3.2. However, since the analysis requires that the entrance losses, pipe losses and exit losses be based on a single area, the higher loss factor of 5.2 was utilized, resulting in a higher pressure and, therefore, a more conservative analysis.

TABLE 6.2-2
(SHEET 1 OF 2)

ENGINEERED SAFETY SYSTEMS INFORMATION FOR CONTAINMENT RESPONSE ANALYSES

		CONTAINMENT ANALYSIS VALUE*		
	FULL CAPACITY	CASE A	CASE B	CASE C
A. Drywell Spray System (RHR system)				
1. Number of pumps	2	2	1	0
2. Number of lines	2	2	1	0
3. Number of headers	2	2	1	0
4. Spray flow rate, gpm/pump	6700	6700	6700	0
5. Spray thermal efficiency, %	---	---	---	---
B. Suppression Pool Spray (RHR system)				
1. Number of pumps	2	2	1	0
2. Number of lines	2	2	1	0
3. Number of headers	1	1	1	0
4. Spray flow rate, gpm/pump	450	450	450	0
5. Spray thermal efficiency, %	---	---	---	---
C. Containment Cooling System (RHR system)				
1. Number of pumps	2	2	1	1
2. Pump capacity, gpm/pump	7450**		7450	
3. Heat exchangers				
a. Type - inverted U-tube, single pass shell, multipass tubes, vertical mounting				
b. Number	2	2	1	1
c. Heat transfer area, ft^2 /unit	11,000	11,000	11,000	11,000
d. Overall heat transfer coefficient, Btu/hr - ft^2 - °F	215			

* Cases A, B, and C defined in Table 6.2-5.

** A supplementary evaluation has been performed for a slightly reduced RHR pump flow rate of 7200 gpm (suppression pool cooling mode); as discussed in Section 6.2.2.3.4 long term suppression pool temperature is not significantly impacted and the peak long term pool temperature does not exceed the 200°F maximum value given in Table 6.2-5.

TABLE 6.2-2
(SHEET 2 OF 2)

	FULL CAPACITY	CONTAINMENT ANALYSIS VALUE*		
		CASE A	CASE B	CASE C
e. Secondary coolant flow rate per exchanger, lb/hr	3.7×10^6	---	3.7×10^6	---
f. Design service water temperature (CSCS)				
Minimum, °F	32			
Maximum, °F	100	100	100	100
D. ECCS Systems:				
1. High-pressure core spray (HPCS)				
a. Number of pumps	1	1	1	1
b. Number of lines	1	1	1	1
c. Flow rate, gpm	6200	6200	6200	6200
2. Low-pressure core spray (LPCS)				
a. Number of pumps	1	1	0	0
b. Number of lines	1	1	0	0
c. Flow rate (rated), gpm/line	6250	6250	0	0
d. Number of headers	2	2	0	0
3. Low-pressure coolant injection (LCPI)				
a. Number of pumps	3	3	1	1
b. Number of lines	3	3	1	1
c. Flow rate, gpm/line	7067	7067	7067	7067
4. Residual heat removal (RHR)				
a. Pump flow rate:				
Shell side	7450**			
Tube side	7400			
b. Source of cooling water		RHR service water		
c. Flow begins, seconds		Manual, approximately 600 ***		
E. Automatic Depressurization System				
1. Total number of safety/relief valves	18			
2. Number actuated on ADS	7			

*** Refer to Section 6.2.2.3.6 for further discussion on the sensitivity of this time period.

* Cases A, B, and C defined in Table 6.2-5.

TABLE 6.2-3
(SHEET 1 OF 2)

INITIAL CONDITIONS EMPLOYED
IN CONTAINMENT RESPONSE ANALYSES

A. Reactor Coolant System (at 105% rated steam flow and at normal liquid levels)

1.	Reactor power level, MWt	3559
2.	Average coolant pressure, psig	1025
3.	Average coolant temperature, °F	550
4.	Mass of reactor coolant system liquid, lbm	676,700
5.	Mass of reactor coolant system steam, lbm	24,900
6.	Liquid plus steam energy, Btu	380×10^6
7.	Volume of water in vessel, ft^3	11,175
8.	Volume of steam in vessel, ft^3	9,640
9.	Volume of water in recirculation loops, ft^3	1,030
10.	Volume of steam in steamlines, ft^3	1,030
11.	Volume of water in feedwater line, ft^3	20,778*
12.	Volume of water in miscellaneous lines, ft^3	191
13.	Total reactor coolant volume, ft^3	22,712
14.	Stored water	
	a. Condensate storage tank, gal	350,000
	b. Fuel storage pool, ft^3	50,000

* Does not represent the feedwater volume used in post-LOCA feedwater coastdown/injection evaluation. This evaluation is discussed in detail in Section 6.2.1.1.3.1.1 in paragraph titled, "Evaluation of Post-LOCA Feedwater Injection".

TABLE 6.2-3
(SHEET 2 OF 2)

B. Containment	Drywell	Suppression Chamber
1. Pressure, psig	0.75	0.75
2. Inside temperature, °F	98	105
3. Outside temperature, °F	104	104
4. Relative humidity, %	20	100
5. Service water temperature (CSCS), °F (1)	100	100
6. Water volume, ft^3 (minimum)	---	128,800*
(maximum)		131,900*
7. Vent submergence, (maximum)	---	12.33 ft.
(minimum)		11.7 ft.

*Conservative value used in Reference 22.

(1) Evaluated for post-accident peak of 107°F in Reference 36.

TABLE 6.2-4

MASS AND ENERGY RELEASE DATA FOR ANALYSIS OF WATER POOL PRESSURE-SUPPRESSION CONTAINMENT ACCIDENTS

A. Effective accident break area (total), ft^2	3.113
Pipe ID, in.	21.686
B. Components of effective break area:	
1. Recirculation line area, ft^2	2.565
2. Cleanup line area, ft^2	0.080
3. Jet pumps area, ft^2	0.468
C. Break area/vent area ratio	0.010
D. Primary system energy distribution*	
1. Steam energy, 10^6 Btu	29.6
2. Liquid energy, 10^6 Btu	355.3
3. Sensible energy, 10^6 Btu	
a. Reactor vessel	106.1
b. Reactor internals (less core)	58.6
c. Primary system piping	34.6
d. Fuel**	25.2
E. Assumptions used in pressure transient analysis	
1. Feedwater valve closure time	Instantaneous See Note 1
2. MSIV closure time (sec)	3.5
3. Scram time (sec)	< 1
4. Liquid carryover, %	100
5. Turbine stop valve closure (sec)	0.2

* All energy values except fuel are based on a 32°F datum.
** Fuel energy is based on a datum of 285°F.

Note 1 This assumption has been supplemented for a conservative evaluation on the peak long term suppression pool temperature. This supplemental evaluation postulates the addition of feedwater mass and energy injected at time t=600 seconds after LOCA. Section 6.2.1.1.3.1.1 in the paragraph titled, "Evaluation of Post-LOCA Feedwater Injection" discusses this in further detail.

TABLE 6.2-5
(SHEET 1 OF 3)

LOSS OF COOLANT ACCIDENT LONG TERM
PRIMARY CONTAINMENT RESPONSE SUMMARY

CASE	LPCI AND/OR LPCS PUMPS	SERVICE WATER PUMPS	CONTAINMENT SPRAY (gal/min)	HPCS (gal/min)	LPCI AND/OR LPCS (gal/min)	PEAK POOL TEMPERATURE (°F) **	SECONDARY PEAK PRESSURE (psig)
A	3/1	4	14,134	6200	21,200/ 6,250	168.4	5.3
B	1/0	2	7,067	6200	7067/0	200	9.6
C	1/0	2	0	6200	7067/0	200++	14.2

** Supplementary evaluations have been performed, as discussed in Section 6.2.1.8, based on an increase in the initial suppression pool temperature (from 100°F to 105°F), the peak suppression pool bulk temperature is less than 200°F.

++ A supplementary evaluation, for the effect on long term peak pool temperature, has been performed for the addition of feedwater mass and energy at t=600 seconds and a reduced RHR pump flow in the suppression pool cooling mode (7200 gpm versus 7450 gpm). The 200°F peak pool temperature given above is not exceeded.

TABLE 6.2-5
(SHEET 2 OF 3)

LOSS OF COOLANT ACCIDENT LONG TERM

PRIMARY CONTAINMENT RESPONSE SUMMARY

CASE	LPCI AND/OR LPCS PUMPS	SERVICE WATER PUMPS	CONTAINMENT SPRAY (gal/min)	HPCS (gal/min)	LPCI AND/OR LPCS (gal/min)	PEAK POOL TEMPERATURE* (°F)	PRIMARY PEAK SUPPRESSION CHAMBER PRESSURE (PSIG)	SECONDARY SUPPRESSION CHAMBER PEAK PRESSURE (psig)
C (@ CLTP)	1/0	2	0	Note 1	Note 2**	196.1	27.9	12.4
C***	3/1	2	0	Note 1	Note 2	197	26.8	13.4

* See Figures 6.2-7a and 6.2-7b for long term containment responses vs. time.

** RHR flow through heat exchanger (Reference 20).

*** GE SC06-01 concerns addressed (Reference 33).

TABLE 6.2-5
(SHEET 3 OF 3)

LOSS OF COOLANT ACCIDENT LONG TERM

PRIMARY CONTAINMENT RESPONSE SUMMARY

Note 1: Flow varies with reactor/drywell differential pressure.
HPCS flow = 7175 gpm when reactor/drywell Dp = 0 psid.
HPCS flow = 620 gpm when reactor/drywell Dp = 2000 psid.
Refer to Reference 31 for details.

Note 2: Flow varies with reactor/drywell differential pressure.
LPCI flow = 8400 gpm when reactor/drywell Dp = 0 psid.
LPCI flow = 745 gpm when reactor/drywell Dp = 2000 psid.
LPCS flow = 7800 gpm when reactor/drywell Dp = 0 psid.
LPCS flow = 625 gpm when reactor/drywell Dp = 2000 psid.
Refer to Reference 31 for details.

TABLE 6.2-6

ENERGY BALANCE FOR DESIGN-BASIS RECIRCULATION LINE BREAK ACCIDENT

	PRIOR TO DBA (0 sec)	AT TIME OF PEAK PRESSURE DIFFERENCE (0.75 at Recirc.)	AT END OF BLOWDOWN (~53 sec)	AT TIME OF PEAK CONTAINMENT PRESSURE (~27009 sec - minimum ECCS available; ~7047 sec - all ECCS Available)	UNIT
1. Reactor coolant (vessel & pipe inventory)	414.0×10^6	400×10^6	11.8×10^6	45.6×10^6 /41.8×10^6	Btu
2. Fuel and cladding	34.0				
Fuel	34.8×10^6	32.3×10^6	12.8×10^6	4.07×10^6/3.72×10^6	Btu
Cladding	3.05×10^6	3.05×10^6	2.99×10^6	0.956×10^6 /0.904×10^6	Btu
3. Core internals, also reactor coolant piping pumps & valves	91.2×10^6	91.2×10^6	91.2×10^6	31.4×10^6/55.5×10^6	Btu
4. Reactor vessel metal	107.0×10^6	107.0×10^6	107.0×10^6	37×10^6/64.4×10^6	Btu
5. Reactor coolant piping, pumps and valves	Included in (3)				
6. Blowdown enthalpy	NA	546	NA	NA	Btu/lbm
7. Decay heat	0	$.402920 \times 10^6$	8.802×10^6	1020×10^6/383.0×10^6	Btu
8. Metal-water reaction heat	0	0	0.02×10^6	$.471 \times 10^6$/$.471 \times 10^6$	Btu
9. Drywell structures	Storage Capacitance Neglected			Btu	
10. Drywell air	1.52×10^6	1.73×10^6	0	1.77×10^6/158×10^6	Btu
11. Drywell steam	0.335×10^6	7.41×10^6	25.7×10^6	7.06×10^6/5.32×10^6	Btu
12. Containment air	1.17×10^6	1.17×10^6	2.77×10^6	1.41×10^6/1.49×10^6	Btu
13. Containment steam	0.522×10^6	0.522×10^6	1.29×10^6	5.57×10^6/2.86×10^6	Btu
14. Suppression pool water	887×10^6	887×10^6	1300×10^6	1770×10^6/1490×10^6	Btu
15. Heat transferred by heat exchangers	0	0	0	752×10^6/260×10^6	Btu

NOTE 1: Results of analysis for MS and recirc line breaks are approximately the same; however, the progress of the events is more rapid for the MS break than for the recirc.
Note 2: A supplementary evaluation, for the effect on long term peak pool temperature, has been performed for the addition of feedwater mass and energy injection at t=600 seconds, the total additional energy calculated due to the feedwater volume and the feedwater piping metal sensible heat is 2.07 x E08 Btu. (Ref. 18).

TABLE 6.2-7

ACCIDENT CHRONOLOGY
DESIGN-BASIS RECIRCULATION LINE BREAK ACCIDENT

	TIME (sec)	
	ALL ECCS IN OPERATION	MINIMUM ECCS AVAILABLE
Vents cleared	0.824	0.824
Drywell reaches peak pressure	20.14	20.14
Maximum positive differential pressure occurs	0.831	0.831
Initiation of the ECCS	30	30
End of blowdown	52.15	52.15
Vessel reflooded	( )	109.53
Introduction of RHR heat exchanger	(approx.) 600*	(approx.) 600*
Containment reaches peak secondary pressure	10,915	27,009

* Refer to Section 6.2.2.3.6 for further discussion on the sensitivity of this time period.

TABLE 6.2-8

SUMMARY OF ACCIDENT RESULTS FOR
CONTAINMENT RESPONSE TO RECIRCULATION LINE AND
STEAMLINE BREAKS

A. Accident Parameters	RECIRCULATION LINE BREAK *	STEAMLINE BREAK
1. Peak drywell pressure, psig	42.6	32
2. Peak drywell deck differential pressure, psid	24.4	17.5
3. Time(s) of peak pressures, sec	13	11
4. Peak drywell temperature, °F	289	320
5. Peak suppression chamber pressure, psig	28.7	25
6. Time of peak suppression chamber pressure, sec	>40	50
7. Peak suppression pool temperature during blowdown, °F	135**	100**
8. Peak suppression pool temperature, long term, °F	200++	
9. Calculated drywell margin, %	5.3	
10. Calculated suppression chamber margin, %	36.2	
11. Calculated deck differential pressure margin, %	2.4	

*See Figure 6.2-2 for plots of pressures vs time.
See Figure 6.2-3 for plots of temperatures vs time.

** As discussed in Section 6.2.1.8 supplementary evaluations have been satisfactorily completed with a 105°F initial suppression pool temperature.

++ See Notes in Table 6.2-5.

TABLE 6.2-9

SUBCMPARTMENT NODAL DESCRIPTION RECIRCULATION OUTLET LINE BREAK WITH SHIELDING DOORS

NODE NUMBER	DESCRIPTION	VOLUME (ft^3)	HEIGHT (ft)	FLOW CROSS-SECTIONAL AREA (ft)	BOTTOM ELEVATION (ft)	INITIAL CONDITIONS TEMP, (°F)	PRESS, (psia)	HUMID, *(%)	CALC. PEAK PRESS DIFF, (psid)
1	Lower Reactor Skirt	100.6	5.07	18.40	755.29	550.	15.45	0.1	47.9
2	Lower Reactor Skirt	100.6	5.07	18.40	755.29	550.	15.45	0.1	48.0
3	Lower Reactor Skirt	100.6	5.07	18.40	755.29	550.	15.45	0.1	47.4
4	Lower Reactor Skirt	150.9	5.07	23.36	755.29	550.	15.45	0.1	47.9
5	Lower Reactor Skirt	150.9	5.07	23.36	755.29	550.	15.45	0.1	48.1
6	Upper Reactor Skirt	121.0	7.47	20.98	760.36	550.	15.45	0.1	47.9
7	Upper Reactor Skirt	121.0	7.47	20.98	760.36	550.	15.45	0.1	48.0
8	Upper Reactor Skirt	121.0	7.47	20.98	760.36	550.	15.45	0.1	47.5
9	Upper Reactor Skirt	181.5	7.47	25.64	760.36	550.	15.45	0.1	48.1
10	Upper Reactor Skirt	181.5	7.47	25.64	760.36	550.	15.45	0.1	47.8
11	Lower Recirc. Noz. Sect.	39.87	6.92	10.02	767.83	550.	15.45	0.1	74.2
12	Lower Recirc. Noz. Sect.	54.28	4.90	10.50	767.83	550.	15.45	0.1	47.3
13	Lower Recirc. Noz. Sect.	61.94	4.90	10.50	767.83	550.	15.45	0.1	48.2
14	Lower Recirc. Noz. Sect.	81.43	4.90	13.47	767.83	550.	15.45	0.1	48.2
15	Lower Recirc. Noz. Sect.	80.54	4.90	13.47	767.83	550.	15.45	0.1	46.4
16	Upper Recirc. Noz. Sect.	26.77	2.67	8.43	774.75	550.	15.45	0.1	72.0
17	Upper Recirc. Noz. Sect.	52.18	4.69	10.30	772.73	550.	15.45	0.1	45.2
18	Upper Recirc. Noz. Sect.	52.18	4.69	10.30	772.73	550.	15.45	0.1	40.9
19	Upper Recirc. Noz. Sect.	78.28	4.69	13.27	772.73	550.	15.45	0.1	37.7
20	Upper Recirc. Noz. Sect.	77.39	4.69	13.27	772.73	550.	15.45	0.1	37.2
21	Mid-Section	67.48	6.41	12.44	777.42	550.	15.45	0.1	39.5
22	Mid-Section	67.48	6.41	12.44	777.42	550.	15.45	0.1	39.2
23	Mid-Section	67.48	6.41	12.44	777.42	550.	15.45	0.1	36.7
24	Mid-Section	101.2	6.41	15.52	777.42	550.	15.45	0.1	36.0
25	Mid-Section	101.2	6.41	15.52	777.42	550.	15.45	0.1	35.9
26	LPCI Noz. Sect.	171.1	9.59	18.61	783.83	550.	15.45	0.1	27.6
27	LPCI Noz. Sect.	155.8	9.59	18.61	783.83	550.	15.45	0.1	27.3
28	LPCI Noz. Sect.	155.8	9.59	18.61	783.83	550.	15.45	0.1	26.7
29	LPCI Noz. Sect.	171.1	9.59	18.61	783.83	550.	15.45	0.1	26.5
30	Feedwater Noz. Sect.	155.8	8.81	17.86	793.42	550.	15.45	0.1	19.3
31	Feedwater Noz. Sect.	153.4	8.81	17.86	793.42	550.	15.45	0.1	19.0
32	Feedwater Noz. Sect.	143.9	8.81	17.86	793.42	550.	15.45	0.1	18.9
33	Feedwater Noz. Sect.	164.1	8.81	17.86	793.42	550.	15.45	0.1	19.0
34	Annulus Receiver	19.76	6.92	10.02	767.83	550.	15.45	0.1	115.1
35	Break Node	19.52	4.92	7.04	769.56	550.	15.45	0.1	322.0
36	Upper Drywell	16315.	41.0	400.	793.42	135.	15.45	15.0	--
37	Mid-Drywell	11665.	12.1	965.	781.32	135.	15.45	15.0	--
38	Lower Drywell	82775.	44.7	1850.	736.62	135.	15.45	15.0	--

* Relative humidity.

TABLE 6.2-10

SUBCOMPARTMENT NODAL DESCRIPTION
FEEDWATER LINE BREAK WITH SHIELDING DOORS

NODE NUMBER	DESCRIPTION	VOLUME (ft^3)	HEIGHT (ft)	FLOW CROSS-SECTIONAL AREA (ft)	BOTTOM ELEVATION (ft)	INITIAL CONDITIONS			CALC. PEAK PRESS DIFF, (psid)
						TEMP, (°F)	PRESS, (psia)	HUMID, *(%)	
1	Lower Reactor Skirt	150.9	5.07	23.36	755.29	550.	15.45	0.1	14.0
2	Lower Reactor Skirt	150.9	5.07	23.36	755.29	550.	15.45	0.1	14.0
3	Lower Reactor Skirt	150.9	5.07	23.36	755.29	550.	15.45	0.1	14.0
4	Lower Reactor Skirt	150.9	5.07	23.36	755.29	550.	15.45	0.1	14.1
5	Upper Reactor Skirt	181.5	7.47	23.80	760.36	550.	15.45	0.1	14.0
6	Upper Reactor Skirt	181.5	7.47	23.80	760.36	550.	15.45	0.1	13.9
7	Upper Reactor Skirt	181.5	7.47	23.80	760.36	550.	15.45	0.1	14.0
8	Upper Reactor Skirt	181.5	7.47	23.80	760.36	550.	15.45	0.1	14.1
9	Recirc. Noz. Sect.	159.7	9.59	17.83	767.83	550.	15.45	0.1	14.4
10	Recirc. Noz. Sect.	157.9	9.59	17.83	767.83	550.	15.45	0.1	14.1
11	Recirc. Noz. Sect.	157.9	9.59	17.83	767.83	550.	15.45	0.1	13.6
12	Recirc. Noz. Sect.	167.4	9.59	17.83	767.83	550.	15.45	0.1	13.4
13	Mid-Section	67.48	6.41	12.44	777.42	550.	15.45	0.1	18.2
14	Mid-Section	67.48	6.41	12.44	777.42	550.	15.45	0.1	15.5
15	Mid-Section	67.48	6.41	12.44	777.42	550.	15.45	0.1	14.0
16	Mid-Section	101.2	6.41	15.79	777.42	550.	15.45	0.1	13.5
17	Mid-Section	101.2	6.41	15.79	777.42	550.	15.45	0.1	13.3
18	LPCI Noz. Sect.	100.8	9.59	15.52	783.83	550.	15.45	0.1	17.7
19	LPCI Noz. Sect.	110.0	9.59	15.52	783.83	550.	15.45	0.1	16.1
20	LPCI Noz. Sect.	116.1	9.59	15.52	783.83	550.	15.45	0.1	14.3
21	LPCI Noz. Sect.	171.1	9.59	18.61	783.83	550.	15.45	0.1	13.0
22	LPCI Noz. Sect.	155.8	9.59	18.61	783.83	550.	15.45	0.1	12.6
23	Annulus Receiver	45.22	10.58	13.39	793.42	550.	15.45	0.1	50.8
24	Feedwater Noz. Sect.	55.63	10.58	13.39	793.42	550.	15.45	0.1	36.9
25	Feedwater Noz. Sect.	116.2	10.58	16.48	793.42	550.	15.45	0.1	21.3
26	Feedwater Noz. Sect.	131.5	10.58	16.48	793.42	550.	15.45	0.1	11.5
27	Feedwater Noz. Sect.	176.7	10.58	19.57	793.42	550.	15.45	0.1	10.5
28	Feedwater Noz. Sect.	171.8	10.58	19.57	793.42	550.	15.45	0.1	10.3
29	Break Node	16.12	4.00	5.42	796.75	550.	15.45	0.1	201.6
30	Lower Drywell	16315.	41.00	400.	793.42	135.	15.45	15.0	--
31	Mid Drywell	11665.	12.10	965.	781.32	135.	15.45	15.0	--
32	Upper Drywell	82775.	44.70	1850.	736.62	135.	15.45	15.0	--

* Relative humidity.

TABLE 6.2-11

SUBCOMPARTMENT NODAL DESCRIPTION

SIMULTANEOUS BREAK OF THE HEAD SPRAY LINE AND RPV HEAD VENT LINE IN THE HEAD CAVITY

					INITIAL CONDITIONS				DBA BREAK CONDITIONS					
Volume No.	Description	Height, ft	Cross-Sectional Area, ft^2	Volume ft^3	Temp. °F	Press. psia	Humid. *%	Break Loc. Vol. No.	Break Line	Break Area, ft^2	Break Type	Calc. Peak Press Diff. psid	Design Peak Press Diff. psid	Design Margin %
1	Head Cavity	15.57	261.5	4072.	135.	15.45	0.1	1	1RI24B-6 + 1NB13A-4	.163	Double-ended guillotine break	7.0 nodes 1 to 2	10.6	150
2	Drywell	79.74	2315.0	184664.	135.	15.45	0.1							
3	Wetwell	33.87	5198.0	176085.	100**	15.45	0.1							

* Relative humidity
The peak differential pressure across the bulkhead plate ($P_{node\ 1} - P_{node\ 2}$) for this case = 7.0 psid
Design differential pressure across the bulkhead plate = 10.6 psid
** As discussed in Section 6.2.1.8 supplementary evaluations have been satisfactorily completed with a 105ºF initial suppression pool temperature.

TABLE 6.2-12

SUBCOMPARTMENT NODAL DESCRIPTION
RECIRCULATION LINE BREAK IN THE DRYWELL

					INITIAL CONDITIONS				DBA BREAK CONDITIONS					
Volume No.	Description	Height, ft	Cross-Sectional Area, ft^2	Volume ft^3	Temp. °F	Press. psia	Humid.* %	Break Loc. Vol. No.	Break Line	Break Area, ft^2	Break Type	Calc. Peak Press Diff. psid	Design Peak Press Diff. psid	Design Margin %
1	Head Cavity	15.57	261.5	4072.	135.	15.45	0.1							
2	Drywell	79.74	2315.0	177049.	135.	15.45	0.1	2	Recirculation line	3.216	Double-ended guillotine	9.0	10.6	118
3	Wetwell	33.87	5198.0	176085.	100**	15.45	0.1							

* Relative humidity

The peak differential pressure across the bulkhead plate ($P_{node1} – P_{node\ 2}$) for this case = -9.0 psid

The design differential pressure across the bulkhead plate = 10.6 psid

** As discussed in Section 6.2.1.8 supplementary evaluations have been satisfactorily completed with a 105°F initial suppression pool temperature.

TABLE 6.2-13

SUBCOMPARTMENT VENT PATH DESCRIPTION

SIMULTANEOUS BREAK OF THE HEAD SPRAY LINE AND
RPV HEAD VENT LINE IN THE HEAD CAVITY

VENT PATH NO	FROM VOL. NODE NO.	TO VOL. NODE NO.	DESCRIPTION OF VENT PATH FLOW CHOKED UNCHOKED	AREA** ft2	LENGTH** ft	HYDRAULIC DIAMETER ft	HEAD LOSS, K FRICTION K, ft/d	TURNING LOSS, K	EXPANSION, K	CONTRACTION, K	TOTAL
1	1	2	HVAC vents through bulkhead plate choked		6.12	3.76	-	-	-	-	2.62
2*	2	3	98-24 inch downcomers unchoked	295.00	70.8	19.38	-	-	-	-	1.90
3	0	1	Break of head spray line & RPV head vent line in head cavity fill	0.163	0.0	0.46	-	-	-	-	0.00

* Opened on a differential pressure of 5.2 psid

** Length/Area is the inertial term input directly into RELAP4 / MOD5

TABLE 6.2-14

SUBCOMPARTMENT VENT PATH DESCRIPTION

RECIRCULATION LINE BREAK IN THE DRYWELL

VENT PATH NO	FROM VOL. NODE NO.	TO VOL. NODE NO.	DESCRIPTION OF VENT PATH FLOW CHOKED UNCHOKED	AREA** ft2	LENGTH** ft	HYDRAULIC DIAMETER ft	HEAD LOSS, K: FRICTION K, ft/d	HEAD LOSS, K: TURNING LOSS, K	HEAD LOSS, K: EXPANSION, K	HEAD LOSS, K: CONTRACTION, K	HEAD LOSS, K: TOTAL
1	1	2	HVAC vents without ductwork through bulkhead plate unchoked	11.12	6.12	3.76	-	-	-	-	2.62
2*	2	3	98-24 inch downcomers unchoked	295.00	70.8	19.38	-	-	-	-	1.90
3	0	2	Recirculation line break in drywell fill	1.00	0.00	0.46	-	-	-	-	0.00

* Opened on a differential pressure of 5.2 psid
** Length/Area is the inertial term input directly into RELAP4 / MOD5

TABLE 6.2-15

SIMULTANEOUS BREAK OF THE HEAD SPRAY LINE AND RPV HEAD VENT LINE IN THE HEAD CAVITY INPUT DATA*

```
= LA SALLE - HEAD CAVITY PRESSURIZATION - 3C7-0476-003 REV 0  4266-00
* PROBLEM DIMENSIONS
010001   -2  9  5  3  3  0  0  3  0  1  0  1  0  0  0  0  0  3
* PROBLEM CONSTANTS
010002   0.0  1.0
* EDIT VARIABLES
020000   AP 1  AP 2  AP 3  JW 2  AH 1  JW 1  TD 1  FD 1  AD 1
* TIME STEP DATA
030010   1   50   0   0   0.01        0.00005   2.0
030020   1   50   0   0   0.002       0.00005   3.5
030030   1   50   0   0   0.0005      0.00005   3.9
030040   1   50   0   0   0.01        0.00005   8.0
030050   1   50   0   0   0.1         0.00005  30.0
* TRIP CONTROLS
040010   1   1   0   0   20.0   0.0
040020   2   4   2   3    5.2   0.0
040030   3   1   0   0    0.0   0.0
* VOLUME DATA
050011   0 0 15.45 135. .001   4072. 15.57 0. 0  261.5 18.3 819.73 0
050021   0 0 15.45 135. .001 184664. 79.74 0. 0 2315.0 54.3 740.00 0
050031   0 0 15.45 100. .001 176085, 33.87 0. 0 5198.0 81.4 706.14 0
* JUNCTION DATA
080011   1 2 0 0 0. 11.12 819.73 .55 2.62 2.62 0 1 0 3 0. .6 -1 0 0.
080021   2 3 0 1   0.0 295.000   740.00 .24 1.9 1.9 0 1 0 3 0. .6 -1 0 0.
080031   0 1 1 0 357.9   0.16267 827.52 .00 0.0 0.0 0 1 0 3 0. 1. -1 0 0.
* VALVE DATA CARDS
110010   -2   0   0   0.   0.   0.   0.
* FILL TABLE CARDS
130100   3   1   2   3   'LBS/SEC'   550.   1.   0.
130101   0    2200.   30.   2200.
```

* RELAP4/MOD5 computer code utilized for analysis.

TABLE 6.2-16

RECIRCULATION LINE BREAK INPUT DATA*

```
= LA SALLE - HEAD CAVITY PRESSURIZATION - 3C7-0476-003 REV 0  4266-00
* RECIRCULATION LINE BREAK
* 4 HVAC INLET VENTS AVAILABLE FOR FLOW INTO HEAD CAVITY
* PROBLEM DIMENSIONS
010001   -2  9  2  3  3  0  0  3  0  1  0  1  0  0  0  0  0  3
* PROBLEM CONSTANTS
010002   0.0  1.0
* EDIT VARIABLES
020000   AP 1  AP 2  AP 3  JW 2  AH 2  JW 1  TD 1  FD 1  TD 2
* TIME STEP DATA
030010   1   50   0   0   0.005      0.00005   2.0
030020   1   50   0   0   0.01       0.00005  30.0
* TRIP CONTROLS
040010   1   1   0   0   10.0   0.0
040020   2   1   0   0    0.824 0.0
040030   3   1   0   0    0.0   0.0
* VOLUME DATA
050011   0 0 15.45 135.  .001   4072.  15.57 0. 0  261.5 18.3 819.73 0
050021   0 0 15.45 135.  .001 177049.  79.74 0. 0 2315.0 54.3 740.00 0
050031   0 0 15.45 100.  .001 176085.  33.87 0. 0 5198.0 81.4 706.14 0
* JUNCTION DATA
080011   1 2 0 0 0. 4.92 819.73 .83 1.52 1.52 0 1 0 3 0. .6 -1 0 0.
080021   2 3 0 1   0.0 295.000   740.00 .24 1.9 1.9 0 1 0 3 0. .6 -1 0 0.
080031   0 2 1 0 25690.  1. 770.  .00 0.0 0.0 0 1 0 3 0. 1. -1 0 0.
* VALVE DATA CARDS
110010   -2   0   0   0.   0.   0.   0.
* FILL TABLE CARDS
130100   3   4   9   1   'LBS/SEC'
*           TIME         FLOW        ENTHALPY
130101      0.0000      22710.0       532.0
130102      0.0016      22710.0       532.0
130103      0.0017      34060.0       532.0
130104      1.5500      34060.0       532.0
130105      1.5600      27550.0       532.0
130106      1.7500      27550.0       532.0
130107      1.7600      24840.0       547.0
130108      1.9800      24840.0       547.0
130109     10.1100      24320.0       538.0
.
```

* RELAP4/MOD5 computer code utilized for analysis.

TABLE 6.2-17

MAIN STEAMLINE BREAK INPUT DATA

```
LISTING OF INPUT DATA FOR CASE  1

=DATA SET 071576-2RLASALLE STUDY 3C7-0476-003  09.8.026-3.0 RELAP4 - MAIN STEAM
* PROBLEM DIMENSIONS
010001    -2  9  2  3  3  0  0  7  0  1  0  5  0  0  0  0  0            * PROB-DIM
* PROBLEM CONSTANTS
010002     0.0  1.0
*EDIT VARIABLES
020000     AP 1   AP 2   AP 3   JW 2   AH 2   JW 1   TD 1   FD 1   TD 2
* TIME STEPS
030010     1  50  0  0  0.005  0.00005  2.0
030020     1  50  0  0  0.01   0.00005  10.0
* TRIP CONTROLS
040010     1  1  0  0  10.0  0.0
040020     2  1  0  0  0.75  0.0
040030     3  1  0  0  0.0  0.0
* VOLUME DATA CARDS  -- 3.7 - P89
050011     0  0  15.45  -1. 0.556  4072.  15.57  15.57  0  261.5  0.  819.73
050021     0  0  15.45  -1.  0.556  177049.  79.74  79.74  0  2315.  0.  740.00
050031     0  0  15.45  -1.  0.524  176085.  33.87  33.87  0  5198.  0.  706.14
* JUNCTION DATA CARDS 08XXXY -- 3.10 - P91
080011     2  1  0  0  0.  6.213  819.73  0.84  1.56  0.  1 0 0 0 0.  0.6  1 0
080021     2  3  0   1  0.  295.  740.00  0.24  1.9  0.  1 0 0 0 0.  0.6  1 0
080031     0 2 1 0 8646.0  1.  770.  0.  0.  0.  0 0 0 0  0.  0.  0 0   * M-STEAM
080041     0  2  2  0  0.  1.  770.  0.  0.  0.  0 0 0 0  0.  0.  0 0   * M-STEAM
080051     0  2  3  0  0.  1.  770.  0.  0.  0.  0 0 0 0  0.  0.  0 0   * M-STEAM
080061     0  2  4  0  0.  1.  770.  0.  0.  0.  0 0 0 0  0.  0.  0 0   * M-STEAM
080071     0  2  5  0  0.  1.  770.  0.  0.  0.  0 0 0 0  0.  0.  0 0   * M-STEAM
* VALVE DATA CARDS 11XXX0  -- 3.16 P97
110010     -2  0.  0.  0.  0.
* FILL TABLE DATA CARDS - 13XXYY  -- 3.18 P98
130100     4  3  0  0  1.0  547.75    0.0  8646.  1.0  8646.
130101     1.01  0.0  10.0  0.0
130200     6  3  0  0  1.0  547.43  0.0  0.0   1.0  0.0
130201       1.01  920.2   4.39  1319.0  4.4  0.0  10.0  0.0
130300     4  3  0  0  1.0  545.55  0.0  0.0    4.39  0.0
130301     4.40  1319.0   10.14  2051.0
130400     6  3  0  0  0.0  547.67   0.0  0.0  0.99  0.0
130401     1.0  28390.0  4.39  27460.0  4.4  0.0  10.0  0.0
130500     4  3  0  0  0.0  547.08  0.0  0.0  4.39  0.0
130501     4.40  27460.0   10.14  24430.0
.
```

TABLE 6.2-18

REACTOR BLOWDOWN FOR RECIRCULATION LINE BREAK

(SHEET 1 OF 2)

Time (sec)	Break Flow Rate (lbm/sec)	Break Flow Enthalpy (BTU/lbm)
0	0	519
0.111328	3.6730 $x10^4$	516.7
1.017578	3.2610 $x10^4$	519.8
2.267578	2.6310 $x10^4$	524.2
3.517578	2.4950 $x10^4$	528.8
4.392578	2.5160 $x10^4$	532.3
5.330078	2.5590 $x10^4$	535.9
5.923828	2.5780 $x10^4$	538.9
6.095703	2.5790 $x10^4$	542.8
6.564453	2.5610 $x10^4$	549.9
7.939453	2.5810 $x10^4$	553.9
8.908203	2.6050 $x10^4$	561.1
9.220703	2.5660 $x10^4$	568.3
9.533203	2.5300 $x10^4$	568.4
9.845703	2.5160 $x10^4$	568.5
10.1582	2.5090 $x10^4$	568.8
10.4707	2.5030 $x10^4$	569.1
10.7832	2.4970 $x10^4$	569.1
11.0957	2.4900 $x10^4$	569.2
11.4082	2.4800 $x10^4$	567.9
11.7207	2.4690 $x10^4$	565.1
11.94727	2.4550 $x10^4$	563.5
12.0332	2.1910 $x10^4$	640
12.2832	1.2200 $x10^4$	852.3
12.48633	9.9600 $x10^3$	905.1
12.72217	9.5870 $x10^3$	901.3

TABLE 6.2-18

REACTOR BLOWDOWN FOR RECIRCULATION LINE BREAK

(SHEET 2 OF 2)

Time (sec)	Break Flow Rate (lbm/sec)	Break Flow Enthalpy (BTU/lbm)
12.72949	9.5630 $\times 10^3$	902.7
12.74854	9.4940 $\times 10^3$	906.6
13.17041	9.1480 $\times 10^3$	902
13.48291	9.6230 $\times 10^3$	850.5
13.98291	9.6930 $\times 10^3$	838.9
14.60791	9.8810 $\times 10^3$	816.2
15.23291	1.0130 $\times 10^4$	791.2
16.17041	1.0480 $\times 10^4$	758.1
17.42041	1.0630 $\times 10^4$	729.5
18.54541	1.0640 $\times 10^4$	709
19.85791	1.0710 $\times 10^4$	687.4
21.73291	1.0180 $\times 10^4$	673.4
23.48291	9.6530 $\times 10^3$	662.8
25.10791	8.9820 $\times 10^3$	659.9
27.42041	7.9910 $\times 10^3$	656.3
29.29541	7.2020 $\times 10^3$	657.3
31.29541	6.3230 $\times 10^3$	659.3
32.67041	5.7390 $\times 10^3$	657.7

TABLE 6.2-19

REACTOR BLOWDOWN DATA FOR MAIN STEAMLINE BREAK

TIME (sec)	STEAM FLOW (lb/sec)	LIQUID FLOW (lb/sec)	STEAM ENTHALPY (Btu/lb)	LIQUID ENTHALPY (Btu/lb)
0.0	11770	0	1190.9	550.9
0.19	11600	0	1191.3	549.1
0.194	8577	0	1191.3	549.1
0.999	8369	0	1192.3	545.3
1.0	899	28450	1192.3	545.3
4.0	1169	27230	1193.4	540.8
10.1	1248	19050	1195.9	529.2
20.38	1730	14680	1200.6	501.3
30.13	1874	9762	1204.2	462.4
40.0	1545	4932	1204.0	409.6
50.0	552	3058	1192.4	322.0
55.32	8.4	253	1173.4	247.9
55.44	0	0	1173.0	246.7

Note: This table is extracted from original analysis and is presented here as historical and representative of comparable response as would be expected for current analysis for this non-limiting break location (See Section 6.2.1.1.3.1).

TABLE 6.2-20

CORE DECAY HEAT FOLLOWING LOCA

FOR CONTAINMENT ANALYSIS

TIME (Seconds)	NORMALIZED CORE HEAT*
0.0	1.0
1.0	0.589
4.0	0.577
10.0	0.377
20.0	0.117
40.0	0.0466
60.0	0.0421
80.0	0.0399
120.0	0.0375
1,000.0	0.0211
10,000.0	0.0108
20,000.0	0.00903
40,000.0	0.00762
80,000.0	0.00634

*Normalized Power = 3559 MWt
Includes fission energy, decay energy, fuel relaxation energy, and metal-water reaction energy

Summary of Lines Penetrating the Primary Containment

CONTAINMENT PENETRATION NUMBER	NRC GDC	LINE ISOLATED	FLUID CONTAINED	LINE SIZE (in)	ESF SYSTEM (NOTE 21)	THROUGH LINE LEAKAGE CLASSIFICATION (NOTE 14,15)	VALVE ARRANGEMENT FIGURE 6.2-31	VALVE NUMBER	LOCATION WITH RESPECT TO CONTAINMENT	TYPE C TEST	LENGTH OF PIPE FROM CONTAINMENT TO OUTERMOST VALVE (ft)
M-1 TO M-4	55	Main Steam (includes drain line)	Steam	26 26 1 1/2	No No No	A (b) A (b) A (b)	Detail (a)	1&2B21-F022A,B,C,D 1&2B21-F028A,B,C,D 1&2B21-F067A,B,C,D	Inside Outside Outside	Yes (Note 30) Yes (Note 30) Yes (Note 30)	N/A 11 N/A
M-5 & M-6	55	Reactor Feed (includes connection to RWC)	Condensate	24 24 24 4	No No No No	AC (b) AC (b) A (b) A (b)	Detail (b)	1&B21-F010A,B 1&2B21-F032A,B 1&2B21-F065A,B 1&2G33-F040	Inside Outside Outside Outside	Yes Yes Yes Yes	N/A N/A 43 N/A
M-7	55	RHRS/Shutdown Suction	Reactor Water	20 20 3/4	No No No	A (b) A (b) A(b)	 Detail (ah)	1&2E12-F009 1&2E12-F008 1&2E12-F460	Inside Outside Inside	No (Note 63) No (Note 63) No (Note 63)	N/A 8 N/A
M-8 & M-9	55 (Note 28)	RHRS/Shutdown Return	Reactor Water	12 12 2	No No No	AC (a) A (b) A (a)	Detail (d)	1&2E12-F050A,B 1&2E12-F053A,B 1&2E12-F099A,B	Inside Outside Inside	No (Note 63) No (Note 63) No (Note 63)	N/A 3 N/A
M-10	55 (Note 28)	LPCS Injection	Suppression Pool Water	12 12	Yes Yes	AC (a) A (b)	Detail (AJ)	1&2E21-F006 1&2E21-F005	Inside Outside	No (Note 63) No (Note 63)	N/A 3
M-11	55 (Note 28)	HPCS Injection	Suppression Pool Water	12 12	Yes Yes	AC (a) A (b)	Detail (AJ)	1&2E22-F005 1&2E22-F004	Inside Outside	No (Note 63) No (Note 63)	N/A 3
M-12 to M-14	55 (Note 28)	RHR/LPCI Injection	Suppression Pool Water	12 12	Yes Yes	AC (a) A (b)	Detail (AJ)	1&2E12-F041A,B,C 1&2E12-F042A,B,C	Inside Outside	No (Note 63) No (Note 63)	N/A 7
M-15	55	Steam to RCIC System (Includes Rhr Supply)	Steam	10 1 10 4	Yes Yes No Yes	A (b) A (b) A (b) A (b)	Detail (e)	1&2E51-F063 1&2E51-F076 1(2)E51-D324 1&2E51-F008	Inside Inside Outside Outside	Yes Yes Yes Yes	N/A N/A 13 max. N/A
M-16	56	Cooling Water Supply	Demineralized Water	6 6 3/4	No No No	A (b) A (b) A(b)	Detail (f)	1&2WR029 1&2WR179 1&2WR225	Outside Inside Inside	Yes Yes Yes	4 N/A N/A
M-17	56	Cooling Water Return	Demineralized Water	6 6 3/4	No No No	A (b) A (b) A(b)	Detail (f)	1&2WR040 1&2WR180 1&2WR226	Outside Inside Inside	Yes Yes Yes	5 N/A N/A
M-18 & M-19	56	RHRS/Containment Spray	Suppression Pool Water	16 16 6	No No No	A (b) A (b) A (b)	Detail (g) Unit 1/ Detail (AM) Unit 2	1&2E12-F017A,B 1&2E12-F016A,B 2E12-F480A,B	Outside Outside Outside	Yes No (Note 63) Yes	N/A 11 11
M-20	56	Drywell Purge	Air	26 26 1 1/2 1 1/2 8	No No No No No	A (b) A (b) A (b) A (b) A (b)	Detail (s) Detail (s) Detail (s) Detail (s)	1&2VQ030 1&2VQ029 1&2VQ047 1&2VQ048 1&2VQ042	Outside Outside Outside Outside Outside	Yes Yes Yes Yes Yes	N/A 10 N/A 10 max. 10 max.

CONTAINMENT PENETRATION NUMBER	VALVE TYPE	ASME SECTION III CODE CLASS	PRIMARY METHOD OF ACTUATION	SECONDARY METHOD OF ACTUATION	NORMAL VALVE POSITION	SHUTDOWN VALVE POSITION	POST ACCIDENT POSITION	POWER FAILURE VALVE POSITION (6)	ISOLATION SIGNAL	VALVE CLOSURE TIME (7)	POWER SOURCE	REMARKS
M-1 to M-4	AO Globe AO Globe MO Gate	1 1 1	Auto Auto Auto	RM RM RM	O O O	C C C	C C C	C C As is	C,D,E,H,P,RM C,D,E,H,P,RM C,D,E,H,P,RM	3 to 5 3 to 5 Standard	ESS 2 ESS 1 ESS 1	Note (1,20) Note (1) Note (48)
M-5 to M-6	Swing Check U1/Swing Check U2 AO No Slam-Check MO Gate MO Gate	1 1 2 2	Process Process RM RM	NA RM M M	O O O O	C C C O	C C C C	NA NA As is As is	Rev. Flow B,F,Rev. Flow RM(Note 34) RM(Note 34)	Instantaneous Instantaneous Standard Standard	NA ESS 2 ESS 1 ESS 1	Note (17) Note (20, 53, 54)
M-7	MO Gate MO Gate Relief	1 1 2	Auto Auto Process	RM RM N/A	C C C	O O C	C C C	As is As is C	A,D,U,RM A,D,U,RM N/A	40 sec 40 sec Instantaneous	ESS 2 ESS 1 N/A	Note (51) 1E12-F008
M-8 & M-9	No Slam-Check MO Globe MO Globe	1 1 1	Process Auto Auto	NA RM RM	C C C	O O O	C C C	NA As is As is	Rev. Flow A,D,U,RM A,D,F,U,RM	Instantaneous 29 sec Standard	ESSA 2 ESS 1 ESS 1	Note (3)
M-10	No Slam-Check MO Gate	1 1	Process Auto	NA RM	C C	C C	O O	NA As is	Rev. Flow RM (Notes 31, 36)	Instantaneous Standard	ESS 1 ESS 1	Note (3) Note (51)
M-11	No Slam-Gate MO Gate	1 1	Process Auto	NA RM	C C	C C	O O	NA As is	Rev. Flow RM (Notes 31, 36)	Instantaneous Standard	ESS 3 ESS 3	Note (3) Note (51)
M-12 to M-14	No Slam-Gate MO Gate	1 1	Process Auto	NA RM	C C	C C	O O	NA As is	Rev. Flow RM (Notes 31, 36)	Instantaneous Standard	Note (22) Note (22)	Note (3) Note (51)
M-15	MO Gate MO Globe NA MO Gate	1 1 1 1	Auto Auto NA Auto	RM RM NA RM	O C C O	O C C O	O O C C	As is As is NA As is	D,RM D,RM NA D,RM	15 sec Standard NA Standard	ESS 2 ESS 2 NA ESS 1	Note (20) Note (20) Note (60)
M-16	MO Gate MO Gate Relief	2 2 2	Auto Auto Process	RM RM N/A	O O C	O O C	C C C	As is As is C	B,F,RM B,F,RM N/A	Standard Standard N/A	ESS 1 ESS 2 N/A	
M-17	MO Gate MO Gate Relief	2 2 2	Auto Auto Process	RM RM N/A	O O C	O O C	C C C	As is As is C	B,F,RM B,F,RM N/A	Standard Standard N/A	ESS 1 ESS 2 N/A	
M-18 & M-19	MO Gate MO Gate Gate	2 2 2	Auto Auto M	RM RM N/A	C C C	C C C	C C C	As is As is N/A	G,RM G,RM N/A	Standard Standard N/A	Note (22) Note (22) N/A	Note (2,20,52,54) Note (2, 51, 52) Note (54)
M-20	AO Butterfly AO Butterfly MO Globe MO Globe AO Butterfly	2 2 2 2 2	Auto Auto Auto Auto Auto	RM RM RM RM RM	C C O O C	C C C C O	C C C C C	C C As is As is C	B,F,Y,Z,RM B,F,Y,Z,RM B,F,Y,Z,RM B,F,Y,Z,RM B,F,Y,Z,RM	10 sec 10 sec 23 sec 23 sec 10 sec	ESS 2 ESS 1 ESS 2 ESS 1 ESS 1	Note (8,20,41,46,50,54) Note (8,46) Note (20,54)) Note (46)

CONTAINMENT PENETRATION NUMBER	NRC GDC	LINE ISOLATED	FLUID CONTAINED	LINE SIZE (in)	ESF SYSTEM (NOTE 21)	THROUGH LINE LEAKAGE CLASSIFICATION (NOTE 14,15)	VALVE ARRANGEMENT FIGURE 6.2-31	VALVE NUMBER	LOCATION WITH RESPECT TO CONTAINMENT	TYPE C TEST	LENGTH OF PIPE FROM CONTAINMENT TO OUTERMOST VALVE (ft)
M-21	56 56 (Note 32)	Vent from Drywell Drywell Pressure	Air Air	26 2 26 2 3/4	No No No No No	A(b) A(b) A(b) A(b) C	Detail (h) Detail (w)	1&2VQ034 1&2VQ035 1&2VQ036 1&2VQ068 1&2CM102	Outside Outside Outside Outside Outside	Yes Yes Yes Yes No	N/A N/A 23 max N/A 10 max.
M-21	55 (Note 33)	RPV Level and Pressure	Reactor Water	3/4	Yes	C	Detail (AB)	1B21-F571	Outside	No	10 max.
M-22	55	Main Stream Drains	Stream-Water Mixture	3 3	No No	A(b) A(b)	Detail (c)	1&2B21-F016 1&2B21-F019	Inside Outside	Yes Yes	N/A 6
M-23		Spare (Unit 1)									
M-23	56	Combustible Gas Control Drywell Suction	AIR/Vapor Mixture	4 4	Yes Yes	A(b) A(b)	Detail (g)	2HG001B 2HG002B	Outside Outside	Yes Yes	N/A 10
M-24		Spare									
M-25 & M-26	56	Chilled Water Supply	Demineralized Water	8 8 3/4	No No No	A(b) A(b) A(b)	Detail (AF)	1&2VP063A,B 1&2VP113A,B 1&2VP198A,B	Outside Inside Inside	Yes Yes Yes	6 N/A N/A
M-27 & M-28	55	Chilled Water Return	Demineralized Water	8 8 3/4	No No No	A(b) A(b) A(b)	Detail (AF)	1&2VP053A,B 1&2VP114A,B 1&2VP197A,B	Outside Inside Inside	Yes Yes Yes	6 N/A N/A

CONTAINMENT PENETRATION NUMBER	VALVE TYPE	ASME SECTION III CODE CLASS	PRIMARY METHOD OF ACTUATION	SECONDARY METHOD OF ACTUATION	NORMAL VALVE POSITION	SHUTDOWN VALVE POSITION	POST ACCIDENT POSITION	POWER FAILURE VALVE POSITION (6)	ISOLATION SIGNAL	VALVE CLOSURE TIME (7)	POWER SOURCE	REMARKS
M-21	AO Butterfly MO Globe AO Butterfly MO Globe Excess Flow Check	2 2 2 2 2	Auto Auto Auto Auto Process	RM RM RM RM N/A	C C C C O	C C C C O	C C C C O	C As is C As is N/A	F,B,Y,Z,RM F,B,Y,Z,RM F,B,Y,Z,RM F,B,Y,Z,RM F,B,Y,Z,RM	10 Sec 5 Sec 10 Sec 5 Sec Instantaneous	ESS 2 ESS 2 ESS 1 ESS 1 NA	Note (8,20,41,46,54) Note (8,20) Note (8,46) Note (8)
M-21	EFCV	2	Process	NA	O	O	O	NA	Flow	Instantaneous	NA	Note (23,33)
M-22	MO Gate MO Gate	1 1	Auto Auto	RM RM	O O	C C	C C	As is As is	C,D,E,H,P,RM C,D,E,H,P,RM	Standard Standard	ESS 2 ESS 1	Note (20),(51) Note (51)
M-23												
M-23	MO Gate MO Globe	2 2	RM RM	M M	C C	C C	O O	As is As is	RM(Note 37) RM(Note 37)	Standard Standard	Note (23) Note (23)	Note (20,54)
M-24												
M-25 TO M-26	MO Gate MO Butterfly Relief	2 2 2	Auto Auto Process	RM RM N/A	O O C	O O C	C C C	As is As is N/A	B,F,RM B,F,RM Process	Standard Standard N/A	ESS 1 ESS 2 N/A	 Note (20) Note (20)
M-27 & M-28	MO Gate MO Butterfly Relief	2 2 2	Auto Auto Process	RM RM N/A	O O C	O O C	C C C	As is As is N/A	B,F,RM B,F,RM Process	Standard Standard N/A	ESS 1 ESS 2 N/A	 Note (20) Note (20)

CONTAINMENT PENETRATION NUMBER	NRC GDC	LINE ISOLATED	FLUID CONTAINED	LINE SIZE (in)	ESF SYSTEM (NOTE 21)	THROUGH LINE LEAKAGE CLASSIFICATION (NOTE 14,15)	VALVE ARRANGEMENT FIGURE 6.2-31	VALVE NUMBER	LOCATION WITH RESPECT TO CONTAINMENT	TYPE C TEST	LENGTH OF PIPE FROM CONTAINMENT TO OUTERMOST VALVE (ft)
M-29	55 (Note 28)	RCIC RPV Head Spray (Includes RHR Head Spray)	Condensate	6 6 6 6	Yes Yes Yes Yes	AC(a) AC(a) A(b) A(b)	Detail (i)	1 &2E51-F066 1 &2E51-F065 1 &2E51-F013 1 &2E12-F023	Inside Outside Outside Outside	No (Note 28) No (Note 28) No (Note 63) No (Note 63)	N/A N/A 20 Max (Unit 1) 10 Max (Unit 2) N/A
M-30	55	Reactor Cleanup	Reactor Water	6 6	No No	A(b) A(b)	Detail (t)	1 &2G33-F001 1 &2G33-F004	Inside Outside	Yes Yes	N/A 5
M-31& M-32	NA (Note 45)	Containment High Rad Detector									
M-33	56	Combustible Gas Control Drywell Suction	Air/Vapor Mixture	4 4	Yes Yes	A(b) A(b)	Detail (g)	1HG001B 1HG002B	Outside Outside	Yes Yes	N/A 10
M-33	Spare (Unit 2)										
M-34	55	Standby Liquid Control	Sodium Pentaborate Solution	1 1/2 1 1/2 1 1/2	No No No	AC(a) C AD(a)	Detail (u)	1&2C41-F007 1&2C41-F006 1&2C41-F004A,B	Inside Outside Outside	No (Note 62) No No (Note 62)	N/A N/A 100
M-35	Spare										
M-36	55	Recirc. Loop Sampling	Reactor Water	3/4 3/4 3/4	No No No	A(b) A(b) A(b)	Detail (ae)	1&2B33-F019 1&2B33-F395 1&2B33-F020	Inside Inside Outside	Yes Yes Yes	N/A N/A 10 Max
M-37	56	Clean Condensate	Condensate	3 3	No No	A(b) A(b)	Detail (ai)	1&2MC033 1&2MC027	Outside Outside	No (Note 43) No (Note 43)	N/A 4
M-38	56	Service Air	Air	3 3	No No	A(b) A(b)	Detail (v)	1&2SA046 1&2SA042	Outside Outside	No (Note 43) No (Note 43)	N/A 4
M-39	Spare										
M-40A,B,C,D	55 (Note 24)	CRD Insert	Condensate	1	No	A	Note (24)	1&2C11-D001-120 1&2C11-D001-123	Outside Outside	No No	45 Max
M-41A,B,C,D	55 (Note 24)	CRD Withdrawal	Condensate	3/4	No	A	Note (24)	1&2C11-D001-121 1&2C11-D001-122	Outside Outside	No No	45 Max
M-42 to M-46	54	TIP Drive	NA	3/8	No	NA	Note (18)	1&2C51-J004	Outside	Yes Note (18)	2
M-47	54	Air Supply	Air	3/4	No	A(b)		1&2IN031	Outside	Yes	
M-48	Spare										

CONTAINMENT PENETRATION NUMBER	VALVE TYPE	ASME SECTION III CODE CLASS	PRIMARY METHOD OF ACTUATION	SECONDARY METHOD OF ACTUATION	NORMAL VALVE POSITION	SHUTDOWN VALVE POSITION	POST ACCIDENT POSITION	POWER FAILURE VALVE POSITION (6)	ISOLATION SIGNAL	VALVE CLOSURE TIME (7)	POWER SOURCE	REMARKS
M-29	No Slam-Check No Slam-Check MO Gate MO Globe	1 1 1 1	Process Process Auto Auto	NA NA RM RM	C C C C	C C C C	C C C C	NA NA As is As is	Rev. Flow Rev. Flow RM (Note 31) A,D,U,RM(Note 31)	Instantaneous Instantaneous 15 Sec Standard	ESS 1 ESS 1 ESS 1 ESS 1	Note (3) Note (3) Note (51)
M-30	MO Gate MO Gate	1 1	Auto Auto	RM RM	O O	O O	C C	As is As is	B,J,RM B,J,RM	≤ 10 sec ≤ 10 sec	ESS 2 ESS 1	Note (61)
M-31 & M-32												
M-33	MO Gate MO Globe	2 2	RM RM	M M	C C	C C	O O	As is As is	RM(Note 37) RM(Note 37)	Standard Standard	Note (23) Note (23)	Note (20,54)
M-33												
M-34	No Slam-Check No Slam-Check Explosive	1 1 1	Process Process RM	NA NA NA	C C C	C C C	C C C	NA NA	Rev. Flow Rev. Flow NA	-- -- NA	NA NA	
M-35												
M-36	AO Globe Check AO Globe	2 2 2	Auto Process Auto	RM N/A RM	O C O	O C O	C C C	Closed N/A Closed	B,C,RM Reverse Flow B,C,RM	Standard Instantaneous Standard	ESS 2 N/A ESS 1	Note (9,42) Note (9,42)
M-37	Gate Gate	2 2	M M	NA NA	C C	C C	C C	NA NA	NA NA	NA NA	NA NA	Note (43) Note (43)
M-38	Gate Gate	2 2	M M	NA NA	C C	C C	C C	NA NA	NA NA	NA MA	NA NA	Note (43) Note (43)
M-39												
M-40 A, B, C, D	SO Gate SO Gate	Note (27) Note (27)	Auto Auto	RM RM	C C	C C	C C	As is As is	A,RM A,RM	Instantaneous Instantaneous		Typical of 185 Typical of 185
M-41 A, B, C, D	SO Gate SO Gate	Note (27) Note (27)	Auto Auto	RM RM	C C	C C	C C	As is As is	A,RM A,RM	Instantaneous Instantaneous		Typical of 185 Typical of 185
M-42 to M-46	Solenoid Ball	2	Auto	RM	C	C	C	C	A,F,RM (note 31)	NA	NA	
M-47	SO Globe	2	Auto	RM	O	O	C	C	B,F,RM	5 sec	ESS 2	
M-48	Spare											

CONTAINMENT PENETRATION NUMBER	NRC GDC	LINE ISOLATED	FLUID CONTAINED	LINE SIZE (in)	ESF SYSTEM (NOTE 21)	THROUGH LINE LEAKAGE CLASSIFICATION (NOTE 14,15)	VALVE ARRANGEMENT FIGURE 6.2-31	VALVE NUMBER	LOCATION WITH RESPECT TO CONTAINMENT	TYPE C TEST	LENGTH OF PIPE FROM CONTAINMENT TO OUTERMOST VALVE (ft)
M-49 & M-50	56	Recric. Flow Control Valve Hydraulic Piping	Hydraulic Fluid (Fyrquel)	3/4	No	Note (19)	Detail (c)	1&2B33-F338A,B	Inside	No (Note 35)	N/A
				3/4	No	Note (19)	Detail (c)	1&2B33-F339A,B	Outside	No (Note 35)	
				1/2	No	Note (19)	Detail (c)	1&2B33-F340A,B	Inside	No (Note 35)	N/A
				1/2	No	Note (19)	Detail (c)	1&2B33-F341A,B	Outside	No (Note 35)	
				1/2	No	Note (19)	Detail (c)	1&2B33-F342A,B	Inside	No (Note 35)	N/A
				1/2	No	Note (19)	Detail (c)	1&2B33-F343A,B	Outside	No (Note 35)	
				3/4	No	Note (19)	Detail (c)	1&2B33-F344A,B	Inside	No (Note 35)	N/A
				3/4	No	Note (19)	Detail (c)	1&2B33-F345A,B	Outside	No (Note 35)	
M-51	Spare										
M-52	55 (Note 33)	RPV Level	Reactor Water	3/4	Yes	C	Detail (AB)	2B21-F570	Outside	No (Note 33)	10 Max
M-53	56	Combustible Gas Control Drywell Suction	Air/Vapor Mixture	4	Yes	A(b)	Detail (g)	1&21HG001A	Outside	Yes	N/A
				4	Yes	A(b)		1&21HG002A	Outside	Yes	10
M-54 (Unit 1)	Spare										
M-54 (Unit 2)	56	Air Dryer Blowdown	Air	3	No	A(b)	Detail (g)	2IN074	Outside	Yes	
				3	No	A(b)		2IN075	Outside	Yes	
	56	Drywell Pneumatic Comp Discharge	Air	2	No	AC(b)	Detail (AL)	2IN018	Outside	Yes	N/A
				2	No	A(b)		2IN017	Outside	Yes	5
	56	Drywell Pneumatic Comp Suction	Air	2 1/2	No	A(b)	Detail (g)	2IN001A	Outside	Yes	N/A
				2 1/2	No	A(b)		2IN001B	Outside	Yes	5

CONTAINMENT PENETRATION NUMBER	VALVE TYPE	ASME SECTION III CODE CLASS	PRIMARY METHOD OF ACTUATION	SECONDARY METHOD OF ACTUATION	NORMAL VALVE POSITION	SHUTDOWN VALVE POSITION	POST ACCIDENT POSITION	POWER FAILURE VALVE POSITION(6)	ISOLATION SIGNAL	VALVE CLOSURE TIME (7)	POWER SOURCE	REMARKS
M-49 & M-50	SO Globe	2	Auto	RME	O	O	C	C	B,F,RME	Instantan.	ESS 2	Note (35)
	SO Globe	2	Auto	RME	O	O	C	C	B,F,RME	Instantan.	ESS 1	Note (35)
	SO Globe	2	Auto	RME	O	O	C	C	B,F,RME	Instantan.	ESS 2	Note (35)
	SO Globe	2	Auto	RME	O	O	C	C	B,F,RME	Instantan.	ESS 1	Note (35)
	SO Globe	2	Auto	RME	O	O	C	C	B,F,RME	Instantan.	ESS 2	Note (35)
	SO Globe	2	Auto	RME	O	O	C	C	B,F,RME	Instantan.	ESS 1	Note (35)
	SO Globe	2	Auto	RME	O	O	C	C	B,F,RME	Instantan.	ESS 2	Note (35)
	SO Globe	2	Auto	RME	O	O	C	C	B,F,RME	Instantan.	ESS 1	Note (35)
M-51												
M-52	EFCV	2	Process	NA	O	O	O	NA	Flow	Instantan.	NA	
M-53	MO Gate	2	RM	M	C	C	O	As is	RM (Note 37)	Standard	Note (23)	Note (20,54)
	MO Globe	2	RM	M	C	C	O	As is	RM (Note 37)	Standard	Note (23)	
M-54 (Unit 1)												
M-54 (Unit 2)	AO Globe	2	Auto	M	O	O	C	C	F,H,RM	Standard	ESS 2	
	AO Globe	2	Auto	M	O	O	C	C	F,H,RM	Standard	ESS 1	
	No Slam-Check	2	Process	NA	O	O	C	NA	NA	Instantan.		
	AO Globe	2	Auto	M	O	O	C	C	F,H,RM	Standard	ESS 2	Note (28)
	AO Globe	2	Auto	RM	O	O	C	C	F,H,RM	Standard	ESS 2	Note (20)
	AO Globe	2	Auto	RM	O	O	C	C	F,H,RM	Standard	ESS 1	

CONTAINMENT PENETRATION NUMBER	NRC GDC	LINE ISOLATED	FLUID CONTAINED	LINE SIZE (in)	ESF SYSTEM (NOTE 21)	THROUGH LINE LEAKAGE CLASSIFICATION (NOTE 14,15)	VALVE ARRANGEMENT FIGURE 6.2-31	VALVE NUMBER	LOCATION WITH RESPECT TO CONTAINMENT	TYPE C TEST	LENGTH OF PIPE FROM CONTAINMENT TO OUTERMOST VALVE (ft)
M-55	57	ADS Pneumatic Supply	Nitrogen or Air	1	Yes	B	Detail (j)	1 & 2IN100	Outside	No (Note 38)	5
M-56	55 (Note 33)	Reactor Water Level	Reactor Water	3/4	Yes	C	Detail (w)	1 &2B21-F372	Outside	No (Note 33)	10 Max
M-57	Spare										
M-58	Deleted										
M-59	56 (Note 58)	Clean Condensate to Refueling Bellows	Condensate	2 2	No No	A(b) A(b)	Detail (v)	1&2FC113 1&2FC114	Outside Outside	Yes Yes	N/A 5
M-59	55 (Note 33)	RPV Level and Pressure	Reactor Water	3/4	Yes	C	Detail (AB)	1B21-F570	Outside	No	10 Max
M-60 (Unit 1)	56	Drywell Pneumatic Compressor Discharge	Air	2 2 3 3	No No No No	AC(b) A(b) A(b) A(b)	Detail (AL) Detail (g)	1IN018 1IN017 1IN074 1IN075	Outside Outside Outside Outside	Yes Yes Yes Yes	N/A 5
M-60 (Unit 2)	57	ADS Pneumatic Supply	Nitrogen or Air	1	Yes	B	Detail (j)	2IN101	Outside	No (Note 38)	5
M-61 (Unit 1)	57	ADS Pneumatic Supply	Nitrogen or Air	1	Yes	B	Detail (j)	1IN101	Outside	No (Note 38)	5
M-61 (Unit 2)	Spare										
M-62 (Unit 1)	56	Drywell Pneumatic Comp Discharge	Air	2 1/2 2 1/2	No No	A(b) A(b)	Detail (g)	1IN001A 1IN001B	Outside Outside	Yes Yes	N/A 5
M-62 (Unit 2)	Spare										
M-63 & M-64	55	Recirc. Pump Seal Injection Supply	Condensate	3/4 3/4	No No	A(a) A(a)	Detail (h) Note (25)	1&2B33-F013A,B 1&2B33-F017A,B	Inside Outside	Yes (Note 25) Yes (Note 25)	 N/A
M-65	56 (Note 58)	Reactor Well Bulkhead Drain	Water	10 10	No No	A(b) A(b)	Detail (V) (Unit 1 only) Detail (AD) (Unit 2 only)	1&2FC115 1&2FC086	Outside Outside	Yes Yes	N/A 5
M-65 (Unit 2)	55 (Note 33)	RPV Level	Reactor Water	3/4	Yes	C	Detail (AB)	2B21-F571	Outside	No (Note 33)	10 max

CONTAINMENT PENETRATION NUMBER	VALVE TYPE	ASME SECTION III CLASS	PRIMARY METHOD OF ACTUATION	SECONDARY METHOD OF ACTUATION	NORMAL VALVE POSITION	SHUTDOWN VALVE POSITION	POST ACCIDENT POSITION	POWER FAILURE VALVE POSITION (6)	ISOLATION SIGNAL	VALVE CLOSURE TIME (7)	POWER SOURCE	REMARKS
M-55	SO Globe	2	RM	M	O	O	O	O	NA	Instantaneous	ESS 2	
M-56	Excess Flow check	2	Process	NA	O	O	O	NA	Flow	Instantaneous	NA	
M-57												
M-58												
M-59	Globe Globe	2 2	M M	NA NA	L.C. L.C.	C C	C C	NA NA	NA NA	NA NA	NA NA	Note (20,54) Note (20)
M-59	EFCV	2	Process	NA	O	O	O	NA	Flow	Instantaneous	NA	Note(23,33)
M-60 (Unit 1)	Check AO Globe AO Globe AO Globe	2 2 2 2	Process Auto Auto Auto	NA M M M	O O O O	O O O O	C C C C	NA C C C	NA B,F,RM B,F,RM B,F,RM	Instantaneous Standard Standard Standard	 ESS 2 ESS 2 ESS 1	Note (28) Note (28)
M-60 (Unit 2)	SO Globe	2	RM	M	O	O	FO	FO	NA	Instantaneous	ESS 2	
M-61 (Unit 2)	SO Globe	2	RM	M	O	O	FO	FO	NA	Instantaneous	ESS 2	
M-61 (Unit 2)												
M-62 (Unit 1)	AO Globe AO Globe	2 2	Auto Auto	RM RM	O O	O O	C C	C C	B,F,RM B,F,RM	Standard Standard	ESS 2 ESS 1	Note (20)
M-62 (Unit 2)												
M-63 & M-64	No Slam-Check No Slam-Check	2 2	Process Process	NA NA	O O	O O	C C	NA NA	Reverse Flow Reverse Flow	Instantaneous Instantaneous	NA NA	
M-65	Gate Gate	2 2	M M	NA NA	C C	C C	C C	NA NA	NA NA	NA NA	NA NA	Note (20 ,54) (Note 20 Unit 1 only)
M-65 (Unit 2)	EFCV	2	Process	NA	O	O	O	NA	Flow	Instantaneous	NA	

CONTAINMENT PENETRATION NUMBER	NRC GDC	LINE ISOLATED	FLUID CONTAINED	LINE SIZE (in)	ESF SYSTEM (NOTE 21)	THROUGH LINE LEAKAGE CLASSIFICATION (NOTE 14, 15)	VALVE ARRANGEMENT FIGURE 6.2-31	VALVE NUMBER	LOCATION WITH RESPECT TO CONTAINMENT	TYPE C TEST	LENGTH OF PIPE FROM CONTAINMENT TO OUTERMOST VALVE (ft)
M-66	56	Suppression Chamber Purge Line	Air	26 26 1 1/2 1 1/2 8	No No No No No	A (b) A (b) A (b) A (b) A (b)	Detail (s) Detail (s) Detail (s) Detail (s)	1&2VQ027 1&2VQ026 1&2VQ050 1&2VQ051 1&2VQ043	Outside Outside Outside Outside Outside	Yes Yes Yes Yes Yes	N/A 8 7 Max.
M-67	56	Suppression Chamber Vent Line	Air	26 26 2	No No No	A (b) A (b) A (b)	Detail (h)	1&2VQ031 1&2VQ040 1&2VQ032	Outside Outside Outside	Yes Yes Yes	N/A 17 N/A
M-68	56 (Note 28)	LPCS Suction from Suppression Pool	Suppression Pool Water	24	Yes	B	Detail (m)	1&2E21-F001	Outside	No (Note 39)	2
M-69	56 (Note 28)	HPCS Suction from Suppression Pool	Suppression Pool Water	24	Yes	B	Detail (m)	1&2E22-F015	Outside	No (Note 39)	5
M-70	56 (Note 28) 56 (Note 32)	RHR (LPCI) Suction From Supp. Pool Supp. Pool Water Level	Suppression Pool Water Supp. Pool /water	24 3/4	Yes No	B C	Detail (m) Detail (w)	1&2E12-F004A 1&2CM002	Outside Outside	No (Note 39) No (Note 32)	2 10 Max.
M-71	56 (Note 28) 56 (Note 32)	RHR (LPCI) Suction From Supp. Pool Supp. Pool Water Level	Suppression Pool Water Supp. Pool Water	24 3/4	Yes No	B C	Detail (m) Detail (w)	1&2E12-F004C 1&2CM010	Outside Outside	No (Note 39) No (Note 32)	2 10 Max.

CONTAINMENT PENETRATION NUMBER	VALVE TYPE	ASME SECTION III CLASS	PRIMARY METHOD OF ACTUATION	SECONDARY METHOD OF ACTUATION	NORMAL VALVE POSITION	SHUTDOWN VALVE POSITION	POST ACCIDENT POSITION	POWERFAILURE VALVE POSITION (6)	ISOLATION SIGNAL	VALVE CLOSURE TIME (7)	POWER SOURCE	REMARKS
M-66	AO Butterfly AO Butterfly MO Globe MO Globe AO Butterfly	2 2 2 2 2	Auto Auto Auto Auto Auto	RM RM RM RM RM	C C O O C	C C C C O	C C C C C	C C As is As is C	F,B,Y,Z,RM F,B,Y,Z,RM F,B,Y,Z,RM F,B,Y,Z,RM F,B,Y,Z,RM	10 sec. 10 sec. 23 sec. 23 sec. 10 sec.	ESS 2 ESS 1 ESS 1 ESS 1 ESS 1	Note(8,20,46,54) Note(8,46) Note(20, 54) Note (46)
M-67	AO Butterfly AO Butterfly MO Globe	2 2 2	Auto Auto Auto	RM RM RM	C C C	C C C	C C C	C C As is	F,B,Y,Z,RM F,B,Y,Z,RM F,B,Y,Z,RM	10 sec. 10 sec. Standard	ESS 2 ESS 1 ESS 2	Note (8,20,41,46, 54) Note (8, 46) Note (8,20)
M-68	MO Gate	2	RM	M	O	O	O	As is	RM (Note 36)	Standard	ESS 1	Note (20)
M-69	MO Gate	2	Auto	RM	O	O	O	As is	RM (Note 36)	Standard	ESS 3	Note (20)
M-70	MO Gate EFCV	2 2	RM Process	M NA	O O	O O	O O	As is NA	RM (Note 36) Flow	Standard Instantan.	Note (22) NA	Note (20)
M-71	MO Gate EFCV.	2 2	RM Process	M NA	O O	O O	O O	As is NA	RM (Note 36) Flow	Standard Instantan.	Note (22) Na	Note (20)

CONTAINMENT PENETRATION NUMBER	NRC GDC	LINE ISOLATED	FLUID CONTAINED	LINE SIZE (in)	ESF SYSTEM (NOTE 21)	THROUGH LINE LEAKAGE CLASSIFICATION (NOTE 14,15)	VALVE ARRANGEMENT FIGURE 6.2-31	VALVE NUMBER	LOCATION WITH RESPECT TO CONTAINMENT	TYPE C TEST	LENGTH OF PIPE FROM CONTAINMENT TO OUTERMOST VALVE (ft)
M-72	56 (Note 28)	RHR (LPCI) Suction From Supp. Pool	Suppression Pool Water	24	Yes	B	Detail (m)	1&2E12-F004B	Outside	No (Note 39)	2
M-73 & M-74	56	RHR to Suppression Pool Spray Header	Suppression Pool Water	4	No	B	Detail (z)	1&2E12-F027A,B	Outside	No (Note 29)	23
M-75	56 (Note 28)	RCIC Pump Suction From Suppression Pool	Suppression Pool Water	8	Yes	B	Detail (m)	1&2E51-F031	Outside	No (Note 39)	2
M-76	56 (Note 28)	RCIC Turbine Exhaust	Steam	10	Yes Yes	A (b) A (b)	Detail (o)	1&2E51-F068 1&2E51-F040	Outside Outside	Yes Yes	3 N/A
M-77	56 (Note 28) 56 (Note 28)	LPCS Test Line LPCS Min. Flow Line RHR Suction RV RCIC Full Flow Test Return to Supp. Pool	Suppression Pool Water Suppression Pool Water	14 4 2 4	Yes Yes Yes Yes	B B B B	Detail (AA)	1&2E21-F012 1&2E21-F011 1&2E12-F088A 1(2)E51-F362 1(2)E51-F363 1(2)E51-F022 1(2)E51-F059	Outside Outside Outside Outside Outside Outside Outside	No (Note 29) No (Note 29) No (Note 29) Yes (Note 49) Yes (Note 49) Yes (Note 63) Yes (Note 63)	225 max. 215 max. 230 max.
M-78	Spare										
M-79 & M-84	56 (Note 28)	RHR Min. Flow Line RHR Test Line	Supp. Pool Water	18 18 14 8 4 2	Yes Yes Yes Yes Yes Yes	B B B B B C	Detail (q),(AG)	1&2E12-F024A,B 1&2E12-F021 1&2E12-F302 1&2E12-F064A,B,C 1&2E12-F011A,B 1&2E12-F088B	Outside Outside Outside Outside Outside Outside	No (Note 29) No (Note 29) No (Note 29) No (Note 29) No (Note 29) No (Note 29)	300 Max.
M-80	56 (Note 28)	RCIC Pump Min. Flow Line	Condensate	2	Yes	B	Detail (r)	1&2E51-F019	Outside	No (Note 29)	40
M-81	56 (Note 28)	RCIC Vacuum Pump Discharge	Condensate	1 1/4 1 1/4	No No	A (b) A (b)	Detail (r)	1&2E51-F069 1&2E51-F028	Outside Outside	Yes Yes	3 N/A
M-82	56 (Note 28)	HPCS Test Line HPCS Min Flow Line	Condensate	14 4	Yes Yes	B B	Detail (l)	1&2E22-F023 1&2E22-F012	Outside Outside	No (Note 29) No (Note 29)	29 Max.
M-83 & M-93	56 (Note 28)	LPCS Safety/Relief Valve Discharge	Suppression Pool	4 2	Yes Yes	C C	Detail (AK)	1&2E21-F018 1&2E21-F031	Outside Outside	No (Note 29) No (Note 29)	125 Max.
M-85 M-86 M-87 M-90 M-91 M-99	56 (Note 28)	RHR Safety/Relief Valve Discharge	Suppression Pool Water	2 2 2 2 2	Yes Yes Yes Yes Yes	C C C C C	Detail (AK)	1&2E12-F025A 1&2E12-F025B 1&2E12-F025C 1&2E12-F088C 1&2E12-F030 1&2E12-F005	Outside Outside Outside Outside Outside Outside	No (Note 29) No (Note 29) No (Note 29) No (Note 29) No (Note 29) No (Note 29)	69 Max

CONTAINMENT PENETRATION NUMBER	VALVE TYPE	ASME SECTION III CLASS	PRIMARY METHOD OF ACTUATION	SECONDARY METHOD OF ACTUATION	NORMAL VALVE POSITION	SHUTDOWN VALVE POSITION	POST ACCIDENT POSITION	POWER FAILURE VALVE POSITION (6)	ISOLATION SIGNAL	VALVE CLOSURE TIME (7)	POWER SOURCE	REMARKS
M-72	MO Gate	2	RM	M	O	O	O	As is	RM (Note 36)	Standard	Note (22)	Note (20)
M-73 & M-74	MO Gate	2	Auto	RM	C	C	C	As is	G, RM	30 sec	Note (22)	Note (2, 20,56)
M-75	MO Gate	2	Auto	RM	C	C	C	As is	RM (Note 36)	Note (59)	ESS 1 (DC)	Note (20,57)
M-76	MO Gate	2	Auto	RM	O	O	O	As is	RM (Note 36)	Note (59)	ESS 1	Note (20,54))
	Check	2	Process	NA	C	C	C	As is	Reverse Flow	Instantan.		
M-77	MO Globe	2	RM	M	C	C	C	As is	Rm(Notes 31,36)	Note (47)	ESS 1	Note (20)
	MO Gate	2	RM	M	O	O	C	As is	Rm(Notes 31,36)	Standard	ESS 1	Note (20)
	Relief	2	Process	NA	C	C	C	NA		---	--	Note (20)
	Gate	2	Manual	NA	C	C	C	NA		---	--	--
	Gate	2	Manual	NA	C	C	C	NA		---	--	Note (20,54)
	MO Globe	2	Process	RM	C	C	C	As is	RM(Notes 31,36)	Note (59)	ESS 1	
	MO Globe	2	Process	RM	C	C	C	As is	RM(Notes 31,36)	Note (59)	ESS 1	--
M-78												
M-79 & M-84	MO Globe	2	Auto	RM	C	C	C	As is	G,RM	Standard	Note (22)	Note (2 20)
	MO Globe	2	Auto	RM	C	C	C	As is	G,RM	Standard	ESS 2	Note (20)
	Gate	2	M	NA	C	C	C	NA		--	--	Note (20)
	MO Gate	2	RM	M	O	C	C	As is	RM(Notes 31,36)	Standard Note	Note (22)	Note (20)
	MO Gate	2	RM	M	C	C	C	As is	GRM(Notes31,36)	(50) 22 sec	ESS 1	Note (20)
	Relief	2	Process	NA	C	C	C	NA		--	--	Note (20)
M-80	MO Globe	2	RM	M	C	C	C	As is	RM(Notes 31,36)	7 sec	ESS 1 (DC)	Note (20)
M-81	MO Globe	2	RM	M	O	O	O	As is	RM(Notes 31,36)	Note (59)	ESS 1	Note (20,54)
	No Slam Check	2	Process	NA	C	C	C	NA	Reverse Flow	Instantan.	NA	
M-82	MO Globe	2	Auto	M	C	C	C	As is	G,RM	Standard	ESS 3	Note (20)
	MO Gate	2	Auto	M	C	C	C	As is	G,RM	Standard	ESS 3	Note (20,56)
M-83 & M-93	Relief	2	Process	NA	C	C	C	NA	Process	NA	NA	Note (20)
	Relief	2	Process	NA	C	C	C	NA	Process	NA	NA	Note (20)
M-85	Relief	2	Process	NA	C	C	C	NA	Process	NA	NA	Note (20)
M-86	Relief	2	Process	NA	C	C	C	NA	Process	NA	NA	Note (20)
M-87	Relief	2	Process	NA	C	C	C	NA	Process	NA	NA	Note (20)
M-90	Relief	2	Process	NA	C	C	C	NA	Process	NA	NA	Note (20)
M-91	Relief	2	Process	NA	C	C	C	NA	Process	NA	NA	Note (20)
M-99	Relief	2	Process	NA	C	C	C	NA	Process	NA	NA	Note (20)

CONTAINMENT PENETRATION NUMBER	NRC GDC	LINE ISOLATED	FLUID CONTAINED	LINE SIZE (in)	ESF SYSTEM (NOTE 21)	THROUGH LINE LEAKAGE CLASSIFICATION (NOTE 14)	VALVE ARRANGEMENT FIGURE 6.2-31	VALVE NUMBER	LOCATION WITH RESPECT TO CONTAINMENT	TYPE C TEST	LENGTH OF PIPE FROM CONTAINMENT TO OUTERMOST VALVE (ft)
M-88 & M-89	56 (Note 28)	RHR Safety/Relief Valve Discharge and H_x Vent Line	Steam	3/4 3/4 6 2	Yes Yes Yes Yes	B B C C	Detail (p)	1&2E12-F073A,B 1&2E12-F074A,B 1&2E12-F055A,B 1&2E12-F311A,B	Outside Outside Outside Outside	No (Note 29) No (Note 29) No (Note 29) No (Note 29)	N/A 56 Max.
M-92	56 (Note 28)	RCIC Safety/Relief Valve Discharge	Condensate	4	No	C	Detail (AK)	1&2E12-F036B	Outside	No (Note 29)	5
M-94	56 (Note 28)	HPCS Safety/Relief Valve Discharge	Condensate	2	Yes	C	Detail (AK)	1&2E22-F014	Outside	No (Note 29)	27
M-95	Spare										
M-96	56	Drywell Equip. Drains	Water	4 4	No No	A (b) A (b)	Detail (g)	1&2RE025 1&2RE024	Outside Outside	Yes Yes	10 N/A
M-97	56	Drywell Equip. Drain Cooling	Water	2 2	No No	A (b) A (b)	Detail (g)	1&2RE029 1&2RE026	Outside Outside	Yes Yes	10 N/A
M-98	56	Drywell Floor Drains	Water	4 4	No No	A (b) A (b)	Detail (g)	1&2RF012 1&2RF013	Outside Outside	Yes Yes	N/A 10
M-100	56 (Note 28)	SUPR CHBR Oxygen Monitor	N_2/O_2	1/2	No	A (b)	Detail (g)	1&2CM019A 1&2CM020A	Outside Outside	Yes Yes	60 60

CONTAINMENT PENETRATION NUMBER	VALVE TYPE	ASME SECTION III CLASS	PRIMARY METHOD OF ACTUATION	SECONDARY METHOD OF ACTUATION	NORMAL VALVE POSITION	SHUTDOWN VALVE POSITION	POST ACCIDENT POSITION	POWER FAILURE VALVE POSITION (6)	ISOLATION SIGNAL	VALVE CLOSURE TIME (7)	POWER SOURCE	REMARKS
M-88 & M-89	MO Globe MO Globe Relief Relief	2 2 2 2	RM RM Process Process	M M NA NA	C C C C	C C C C	C C C C	As is As is NA NA	RM (Note 36) RM (Note 36) Process Process	Standard Standard NA NA	ESS 1 ESS 1 NA NA	Note (20) Note (20) Note (20) Note (20)
M-92	Relief	2	Process	NA	C	C	C	NA	Process	NA	NA	Note (20)
M-94	Relief	2	Process	NA	C	C	C	NA	Process	NA	NA	Note (20)
M-95												
M-96	AO Globe AO Globe	2 2	Auto Auto	RM RM	C C	C C	C C	C C	B,F,RM B,F,RM	Standard Standard	ESS 1 ESS 2	Note (20)
M-97	AO Globe AO Globe	2 2	Auto Auto	RM RM	C C	C C	C C	C C	B,F,RM B,F,RM	Standard Standard	ESS 1 ESS 2	Note (20,42,54)
M-98	AO Globe AO Globe	2 2	Auto Auto	RM RM	C C	C C	C C	C C	B,F,RM B,F,RM	Standard Standard	ESS 2 ESS 1	Note (20,42)
M-100	SOL Globe SOL Globe	2 2	Auto Auto	RM RM	O O	O O	C C	C C	B, F, RM B, F, RM	5 sec 5 sec	ESS 1 ESS 2	Note 20

CONTAINMENT PENETRATION NUMBER	NRC GDC	LINE ISOLATED	FLUID CONTAINED	LINE SIZE (in.)	ESF SYSTEM (NOTE 21)	THROUGH LINE LEAKAGE CLASSIFICATION (NOTE 14)	VALVE ARRANGEMENT FIGURE 6.2-31	VALVE NUMBER	LOCATION WITH RESPECT TO CONTAINMENT	TYPE C TEST	LENGTH OF PIPE FROM CONTAINMENT TO OUTERMOST VALVE (ft)
M-101	56 56 (Note 28)	RCIC Turbine Exhaust Breaker Line RCIC Safety/Relief Valve Discharge	Air Condensate	2 2 4	Yes Yes No	A (b) A (b) C	Detail (o) Detail (AK)	1&2E51-F080 1&2E51-F086 1&2E12-F036A	Outside Outside Outside	Yes Yes No (Note 29)	17 NA 5
M-102	Spare										
M-103	NA	Vacuum Breaker	Air	24	Yes	Exempt	Detail (y)	1&2PC003C	Outside	No	4
M-104	56 (Note 32)	Supp. Pool Water Level	Supp. Pool Water	3/4	No	C	Detail (w)	1&2CM012	Outside	No (Note 32)	10 Max.
	56	Combustible Gas Control Return	Air Vapor Mixture	6 6	Yes Yes	A (b) A (b)	Detail (g)	1&2HG005A 1&2HG006A	Outside Outside	Yes Yes	NA
	NA	Vacuum Breaker	Air	24	Yes	Exempt	Detail (y)	1&2PC003A	Outside	No	4
	56	HCVS Vent	Air Vapor Mixture	12	No	A (b)	Detail (an)	2PC009A	Outside	Yes	40
	56	HCVS Vent	Air Vapor Mixture	12	No	A (b)	Detail (an)	2PC010A	Outside	Yes	42
M-105	56 (Note 32) NA	Supp. Pool Water Level Vacuum Breaker	Supp. Pool Water Air	3/4 24	No Yes	C Exempt	Detail (w) Detail (y)	1&2CM004 1&2PC003D	Outside Outside	No (Note 32) No	10 Max. 4
M-106	NA 56	Vacuum Breaker Combustible Gas Control Return	Air Air Vapor Mixture	24 6 6	Yes Yes Yes	Exempt A (b) A (b)	Detail (y) Detail (g)	1&2PC003B 1&2HG005B 1&2HG006B	Outside Outside Outside	No Yes Yes	4 N/A
M-107	NA NA	Vacuum Breaker Vacuum Breaker	Air Air	24 24	Yes Yes	Exempt C	Detail (y) Detail (y)	1&2PC002C 1&2PC001C	Outside Outside	No No	2
M-108	NA NA	Vacuum Breaker Vacuum Breaker	Air Air	24 24	Yes Yes	Exempt C	Detail (y) Detail (y)	1&2PC002A 1&2PC001A	Outside Outside	No No	2
M-109	NA NA	Vacuum Breaker Vacuum Breaker	Air Air	24 24	Yes Yes	Exempt C	Detail (y) Detail (y)	1&2PC002D 1&2PC001D	Outside Outside	No No	2
M-110	NA NA	Vacuum Breaker Vacuum Breaker	Air Air	24 24	Yes Yes	Exempt C	Detail (y) Detail (y)	1&2PC002B 1&2PC001B	Outside Outside	No No	2

CONTAINMENT PENETRATION NUMBER	VALVE TYPE	ASME SECTION III CODE CLASS	PRIMARY METHOD OF ACTUATION	SECONDARY METHOD OF ACTUATION	NORMAL VALVE POSITION	SHUTDOWN VALVE POSITION	POST ACCIDENT POSITION	POWER FAILURE VALVE POSITION (6)	ISOLATION SIGNAL	VALVE CLOSURE TIME (7)	POWER SOURCE	REMARKS
M-101	MO Globe	2	RM	M	O	O	C	As is	F,RM (Note 36)	Note (59)	ESS 1	
	MO Globe	2	RM	M	O	O	C	As is	F,RM (Note 36)	Standard	ESS 2	Note (20)
	Relief	2	Process	NA	C	C	C	NA	Process	NA	NA	Note (20)
M-102												
M-103	Butterfly	2	M	NA	O	O	O	NA	NA	NA	NA	Note (4, 55)
M-104	EFCV	2	Process	NA	O	O	O	NA	Flow	Instantan.		NA
	MO Gate	2	RM	M	C	C	O	As is	RM (Note 37)	Standard	Note (23)	Note (20,54)
	MO Gate	2	RM	M	C	C	O	As is	RM (Note 37)	Standard	Note (23)	
	Butterfly	2	M	NA	O	O	O	NA	NA	NA	NA	Note (4,55)
	Butterfly	2	RM	M	C	C	C	C	NA	NA	HCVS	
	Butterfly	2	RM	M	C	C	C	C	NA	NA	HCVS	
M-105	EFCV	2	Process	NA	O	O	O	NA	Flow	Instantan.		NA
	Butterfly	2	M	NA	O	O	O	NA	NA	NA	NA	Note (4,55)
M-106	Butterfly	2	M	NA	O	O	O	NA	NA	NA	NA	Note (4,55)
	MO Gate	2	RM	M	C	C	O	As is	RM (Note 37)	Standard	Note (23)	Note (20,54)
	MO Gate	2	RM	M	C	C	O	As is	RM (Note 37)	Standard	Note (23)	
M-107	Butterfly	2	M	N/A	O	O	O	NA	NA	NA	NA	Note (4,55)
	Vacuum Breaker	2	Process	N/A	C	C	C/O	NA	Pressure Differential	NA		Note (4)
M-108	Butterfly	2	M	NA	O	O	O	NA	NA	NA	NA	Note (4,55)
	Vacuum Breaker	2	Process	NA	C	C	C/O	NA	Pressure Differential	NA		Note (4)
M-109	Butterfly	2	M	NA	O	O	O	NA	NA	NA	NA	Note (4,55)
	Vacuum Breaker	2	Process	NA	C	C	C/O	NA	Pressure Differential	NA		Note (4)
M-110	Butterfly	2	M	NA	O	O	O	NA	NA	NA	NA	Note (4,55)
	Vacuum Breaker	2	Process	NA	C	C	C/O	NA	Pressure Differential	NA		Note (4)

CONTAINMENT PENETRATION NUMBER	NRC GDC	LINE ISOLATED	FLUID CONTAINED	LINE SIZE (in)	ESF SYSTEM (NOTE 21)	THROUGH LINE LEAKAGE CLASSIFICATION (NOTE 14)	VALVE ARRANGEMENT FIGURE 6.2-31	VALVE NUMBER	LOCATION WITH RESPECT TO CONTAINMENT	TYPE C TEST	LENGTH OF PIPE FROM CONTAINMENT TO OUTERMOST VALVE (ft)
I-1A, B, C, D, E, F	---	---	---	---	---	---	---	---	---	---	---
I-2	55 (Note 26)	RPV Level and Pressure	Reactor Water	3/4	Yes	C	Detail (w)	1&2B21-F374	Outside	No (Note 33)	10 max.
I-3	---	---	---	---	---	---	---	---	---	---	10 max.
I-4A	55 (Note 26) 55 (Note 33)	RPV Level and Pressure Backfill	Reactor Water Reactor Water	3/4 1/2	Yes No	C C(b)	Detail (w) Detail (ac)	1&2B21-F376 1&2C11-F423G/ 1&2C11-F422G	Outside Outside	No (Note 33) Yes (Note 33)	10 max. 10 max
I-4B, C, D, E	---	---	---	---	---	---	---	---	---	---	10 max.
I-4F	56	SUPR CHBR/DW Oxygen Monitor	Air	3/4 3/4	No No	A (b) A (b)	Detail (g)	1&2CM017A 1&2CM018A	Outside Outside	Yes Yes	10 max. 10 max.
I-5A	55 (Note 26) 55 (Note 33)	RPV Level and Pressure Backfill	Reactor Water Reactor Water	3/4 1/2	Yes No	C C (b)	Detail (w) Detail (ac)	1&2B21-F359 1&2C11-F423B/ 1&2C11-F422B	Outside Outside	No (Note 33) Yes (Note 33)	10 max. 18 max.
I-5B, C, D, E	---	---	---	---	---	---	---	---	---	---	10 max.
I-5F	56	Drywell Tritium Grab Sample	Air	3/4 3/4	No No	A (b) A (b)	Detail (g)	1&2CM017B 1&2CM018B	Outside Outside	Yes Yes	10 max. 10 max
I-6	55 (Note 26)	RPV Level and Pressure	Reactor Water	3/4	Yes	C	Detail (w)	1&2B21-F355	Outside	No (Note 33)	10 max.
I-7	55 (Note 26) 55 (Note 33)	RPV Level and Pressure Backfill	Reactor Water Reactor Water	3/4 1/2	Yes No	C C (b)	Detail (w) Detail (ac)	1&2B21-F361 1&2C11-F423D/ 1&2C11-F422D	Outside Outside	No (Note 33) Yes (Note 33)	10 max. 13 max
I-8A	55 (Note 26) 55 (Note 33)	RPV Level and Pressure Backfill	Reactor Water Reactor Water	3/4 1/2	Yes No	C C (b)	Detail (w) Detail (ac)	1&2B21-F378 1&2C11-F423F/ 1&2C11-F422F	Outside Outside	No (Note 33) Yes (Note 33)	10 max. 54 max.
I-8B, C, F	---	---	---	---	---	---	---	---	---	---	---
I-8D	56	Drywell Pressure	Air	3/4	No	C	Detail (w)	1&2VQ061	Outside	No (Note 32)	10 max.

CONTAINMENT PENETRATION NUMBER	VALVE TYPE	ASME SECTION III CLASS	PRIMARY METHOD OF ACTUATION	SECONDARY METHOD OF ACTUATION	NORMAL VALVE POSITION	SHUTDOWN VALVE POSITION	POST ACCIDENT POSITION	POWER FAILURE VALVE POSITION (6)	ISOLATION SIGNAL	VALVE CLOSURE TIME (7)	POWER SOURCE	REMARKS
I-1A,B,C,D,E,F	--	--	--	--	--	--	--	--	--	--	--	Spare
I-2	Excess Flow Check	2	Process	NA	O	O	O	NA	Flow	Instantaneous	NA	
I-3	--	--	--	--	--	--	--	--	--	--	--	Spare
I-4A	Excess Flow Check Checks	2 2	Process Process	NA NA	O O	O C	O C	NA NA	Flow Flow	Instantaneous Instantaneous	NA NA	 Note (33)
I-4B,C,D,E	--	--	--	--	--	--	--	--	--	--	--	Spare
I-4F	SO Globe SO Globe	2 2	Auto Auto	RM RM	O O	O O	C C	C C	B,F,RM B,F,RM	5 sec. 5 sec.	ESS 2 ESS 1	Note (20)
I-5A	Excess Flow Check Checks	2 2	Process Process	NA NA	O O	O C	O C	NA NA	Flow Flow	Instantaneous Instantaneous	NA NA	 Note (33)
I-5B,C,D,E	--	--	--	--	--	--	--	--	--	--	--	Spare
I-5F	SO Globe SO Globe	2 2	Auto Auto	RM RM	O O	O O	C C	C C	B,F,RM B,F,RM	5 sec. 5 sec.	ESS 2 ESS 1	Note (20)
I-6	Excess Flow Check	2	Process	NA	O	O	O	NA	Flow	Instantaneous	NA	
I-7	Excess Flow Check Checks	2 2	Process Process	NA NA	O O	O C	O C	NA NA	Flow Flow	Instantaneous Instantaneous	NA NA	 Note (33)
I-8A	Excess Flow Chk Checks	2 2	Process Process	NA NA	O O	O C	O C	NA NA	Flow Flow	Instantaneous Instantaneous	NA NA	 Note (33)
I-8B,C,F	--	--	--	--	--	--	--	--	--	--	--	Spare
I-8D	Excess Flow Check	2	Process	NA	O	O	O	NA	Flow	Instantaneous	NA	

CONTAINMENT PENETRATION NUMBER	NRC GDC	LINE ISOLATED	FLUID CONTAINED	LINE SIZE (in)	ESF SYSTEM (NOTE 21)	THROUGH LINE LEAKAGE CLASSIFICATION (NOTE 14, 15)	VALVE ARRANGEMENT FIGURE 6.2-31	VALVE NUMBER	LOCATION WITH RESPECT TO CONTAINMENT	TYPE C TEST	LENGTH OF PIPE FROM CONTAINMENT TO OUTERMOST VALVE (ft)
I-8E	57 (Note 44)	RPV Head Seal Leak Detection	Air	3/4	No	---	Detail (j)	1&2E31-F303	Outside	No	10 max.
I-9a	55 (Note 26)	RPV Level and Pressure	Reactor Water	3/4	Yes	C	Detail (w)	1&2B21-F370	Outside	No (Note 33)	10 max.
I-9B, C	---	---	---	---	---	---	---	---	---	---	10 max.
I-9D, E, F	57 (Note 44)	ADS Accumulator Pressure	Air	3/4	Yes	B	Detail (j)	1&2B21-F342D, V, S	Outside	No	10 max.
I-10A & B	55 (Note 26)	RPV Level and Pressure	Reactor Water	3/4 3/4	Yes Yes	C C	Detail (w) Detail (w)	1&2B21-F363 1&2B21-F353	Outside Outside	No (Note 33) No (Note 33)	10 max. 10 max.
I-10C & D	55 (Note 26)	RCIC Steam Flow		3/4 3/4	Yes Yes	C C	Detail (w) Detail (w)	1&2B21-F415B 1&2B21-F415A	Outside Outside	No (Note 33) No (Note 33)	10 max. 10 max
I-10E & F	---	---	---	---	---	---	---	---	---	---	10 max.
I-11A	56	Primary Cont. Air Sample	Air Air	1/2 1/2	No No	A (b) A (b)	Detail (g)	1&2CM031 1&2CM032	Outside Outside	Yes Yes	10 max. 10 max.
I-11B	56 (Note 28)	Post LOCA Containment Monitoring	Air	1/2 1/2 1/2	Yes No No	B A (b) A (b)	Detail (k) Detail (g) Detail (g)	1&2CM022A 1&2CM029 1&2CM030	Outside Outside Outside	No (Note 40) Yes Yes	10 max. NA 10 max.
I-12A	55	RPV Level and Pressure	Reactor Water	3/4	Yes	---	Detail (w)	1&2B21-F357	Outside	No (Note 33)	10 max.
I-12B, C, E, F	57 (Note 44)	ADS Accumulator Pressure	Air	3/4	Yes	B	Detail (j)	1&2B21-E342E, R, U, C	Outside	No	10 max.
I-12D	---	---	---	---	---	---	---	---	---	---	---
I-13	56 (Note 32)	Drywell Pressure	Air	3/4	Yes	C	Detail (w)	1&2B21-F382	Outside	No (Note 32)	10 max.
I-14A, B, C, D, E, F	---	---	---	---	---	---	---	---	---	---	10 max.
I-15A, B, C, D	55 (Note 26)	Steam Flow	Steam	3/4 3/4 3/4 3/4	Yes Yes Yes Yes	C C C C	Detail (w) Detail (w) Detail (w) Detail (w)	1&2B21-F328B 1&2B21-F327B 1&2B21-F327A 1&2B21-F328A	Outside Outside Outside Outside	No (Note 33)	10 max. 10 max. 10 max. 10 max.
I-15 E & F	55 (Note 26)	RWCU Flow	Reactor Water	3/4 3/4	No No	C C	Detail (w) Detail (w)	1&2G33-F312A 1&2G33-F312B	Outside Outside	No (Note 33) No (Note 33)	10 max. 10 max.
I-16A	55 (Note 26)	RHR Line Integrity	Reactor Water	3/4	Yes	C	Detail (w)	1&2E12-F315	Outside	No (Note 33)	10 max.
I-16B & C	---	---	---	---	---	---	---	---	---	---	10 max.

CONTAINMENT PENETRATION NUMBER	VALVE TYPE	ASME SECTION III CLASS	PRIMARY METHOD OF ACTUATION	SECONDARY METHOD OF ACTUATION	NORMAL VALVE POSITION	SHUTDOWN VALVE POSITION	POST ACCIDENT POSITION	POWER FAILURE VALVE POSITION (6)	ISOLATION SIGNAL	VALVE CLOSURE TIME (7)	POWER SOURCE	REMARKS
I-8E	Globe	2	Manual	NA	O	O	O	NA	--	--	NA	
I-9A	Excess Flow Check	2	Process	NA	O	O	O	NA	Flow	Instantaneous	NA	
I-9B,C	--	--	--	--	--	--	--	--	--	--	--	Spare
I-9D,E,F	Manual	2	Manual	NA	O	O	O	NA	--	--	--	
I-10A & B	Excess Flow Chk Excess Flow Chk	2 2	Process Process	NA NA	O O	O O	O O	NA NA	Flow Flow	Instantaneous Instantaneous	NA NA	
I-10C & D	Excess Flow Chk Excess FlowChk	2 2	Process Process	NA NA	O O	O O	O O	NA NA	Flow Flow	Instantaneous Instantaneous	NA NA	
I-10E & F	--	--	--	--	--	--	--	--	--	--	--	Spare
I-11A	SO Globe SO Globe	2 2	Auto Auto	RM RM	O O	O O	C C	C C	B,F,RM B,F,RM	5 sec. 5 sec.	ESS 2 ESS 1	Note (20)
I-11B	SO Globe SO Globe SO Globe	2 2 2	Auto Auto Auto	RM RM RM	C/O O O	C O O	O C C	O C C	RM (Note 37) B,F,RM B,F,RM	5 sec. 5 sec. 5 sec.	ESS 1 ESS 2 ESS 1	Note (20) Note (20)
I-12A	Excess Flow Check	2	Process	NA	O	O	O	NA	Flow	Instantaneous	NA	
I-12B,C,E,F	Manual	2	Manual	NA	O	O	O	NA	--	--	--	
I-12D	--	--	--	--	--	--	--	--	--	--	--	Spare
I-13	Excess Flow Check	2	Process	NA	O	O	O	NA	Pressure	Instantaneous	NA	
I-14A,B,C,D,E	--	--	--	--	--	--	--	--	--	--	--	Spare
I-15A,B,C,D	Excess Flow Chk Excess Flow Chk Excess Flow Chk Excess Flow Chk	2 2 2 2	Process Process Process Process	NA NA NA NA	O O O O	O O O O	O O O O	NA NA NA NA	Flow Flow Flow Flow	Instantaneous Instantaneous Instantaneous Instantaneous	NA NA NA NA	
I-15E & F	Excess Flow Chk Excess Flow Chk	2 2	Process Process	NA NA	O O	O O	O O	NA NA	Flow Flow	Instantaneous Instantaneous	NA NA	
I-16A	Excess Flow Check	2	Process	NA	O	O	O	NA	Flow	Instantaneous	NA	
I-16B & C	--	--	--	--	--	--	--	--	--	--	--	Spare

CONTAINMENT PENETRATION NUMBER	NRC GDC	LINE ISOLATED	FLUID CONTAINED	LINE SIZE (in.)	ESF SYSTEM (NOTE 21)	THROUGH LINE LEAKAGE CLASSIFICATION (NOTE 14)	VALVE ARRANGEMENT FIGURE 6.2-32	VALVE NUMBER	LOCATION WITH RESPECT TO CONTAINMENT	TYPE C TEST	LENGTH OF PIPE FROM CONTAINMENT TO OUTERMOST VALVE (ft)
I-16D & E	55 (Note 26)	RCIC Steam Flow	Steam	3/4 3/4	Yes Yes	C C	Detail (w)	1&2B21-F413B 1&2B21-F413A	Outside Outside	No (Note 33) No (Note 33)	10 Max. 10 Max.
I-16F	55 (Note 26)	LPCS/LPCI ΔP	Reactor Water	3/4	Yes	C	Detail (w)	1&2E21-F304	Outside	No (Note 33)	10 Max.
I-17A	55 (Note 26)	Jet Pump Pressure	Reactor Water	3/4	No	C	Detail (w)	1&2B21-F344	Outside	No (Note 33)	10 Max.
I-17B,C,D,E,F	---	---	---	--	--	--	--	--	--	---	10 Max.
I-18	56 (Note 32)	Drywell Pressure	Air	3/4	Yes	--	Detail (w)	1&2B21-F365	Outside	No (Note 32)	10 Max.
I-19A I-19B I-19C I-19D I-19E I-19F	55 (Note 26)	Jet Pump Flow	Reactor Water	3/4 3/4 3/4 3/4 3/4 3/4	No No No No No No	C C C C C C	Detail (w) Detail (w) Detail (w) Detail (w) Detail (w) Detail (w)	1&2B21-F443 1&2B21-F439 1&2B21-F437 1&2B21-F441 1&2B21-F445A 1&2B21-F447	Outside Outside Outside Outside Outside Outside	No (Note 33) No (Note 33) No (Note 33) No (Note 33) No (Note 33) No (Note 33)	10 Max. 10 Max. 10 Max. 10 Max. 10 Max. 10 Max.
I-20A I-20B I-20C I-20D I-20E I-20F	55 (Note 26)	Jet Pump Flow	Reactor Water	3/4 3/4 3/4 3/4 3/4 3/4	No No No No No No	C C C C C C	Detail (w) Detail (w) Detail (w) Detail (w) Detail (w) Detail (w)	1&2B21-F455A 1&2B21-F451 1&2B21-F449 1&2B21-F453 1&2B21-F445B 1&2B21-F455B	Outside Outside Outside Outside Outside Outside	No (Note 33) No (Note 33) No (Note 33) No (Note 33) No (Note 33) No (Note 33)	10 Max. 10 Max. 10 Max. 10 Max. 10 Max. 10 Max.
I-21A,B,C,D,E,F	---	---	---	---	--	-	---	---	---	---	10 Max.
I-22A & D	55 (Note 26)	Recirc. Pump Seal Press.	Reactor Water	3/4 3/4	No No	C C	Detail (w) Detail (w)	1&2B33-F319A 1&2B33-F317A	Outside Outside	No (Note 33) No (Note 33)	10 Max. 10 Max.
I-22B & C	55 (Note 26)	Recirc. Pump Flow	Reactor Water	3/4 3/4 3/4 3/4	No No No No	C C C C	Detail (x) Detail (x) Detail (x) Detail (x)	1&2B33-F313C 1&2B33-F313D 1&2B33-F311C 1&2B33-F311D	Outside Outside Outside Outside	No (Note 33) No (Note 33) No (Note 33) No (Note 33)	10 Max. 10 Max. 10 Max. 10 Max.

CONTAINMENT PENETRATION NUMBER	VALVE TYPE	ASME SECTION III CODE CLASS	PRIMARY METHOD OF ACTUATION	SECONDARY METHOD OF ACTUATION	NORMAL VALVE POSITION	SHUTDOWN VALVE POSITION	POST ACCIDENT POSITION	POWER FAILURE VALVE POSITION (6)	ISOLATION SIGNAL	VALVE CLOSURE TIME (7)	POWER SOURCE	REMARKS
I-16D & E	Excess Flow Check	2	Process	NA	O	O	O	NA	Flow	Instantaneous	NA	
	Excess Flow Check	2	Process	NA	O	O	O	NA	Flow	Instantaneous	NA	
I-16F	Excess Flow Check	2	Process	NA	O	O	O	NA	Flow	Instantaneous	NA	
I-17A	Excess Flow Check	2	Process	NA	O	O	O	NA	Flow	Instantaneous	NA	
I-17B,C,D,E,F	---	--	---	---					---	---		Spare
I-18	Excess Flow Check	2	Process	NA	O	O	O	NA	Pressure	Instantaneous	NA	
I-19A	Excess Flow Check	2	Process	NA	O	O	O	NA	Pressure	Instantaneous	NA	
I-19B	Excess Flow Check	2	Process	NA	O	O	O	NA	Pressure	Instantaneous	NA	
I-19C	Excess Flow Check	2	Process	NA	O	O	O	NA	Pressure	Instantaneous	NA	
I-19D	Excess Flow Check	2	Process	NA	O	O	O	NA	Pressure	Instantaneous	NA	
I-19E	Excess Flow Check	2	Process	NA	O	O	O	NA	Pressure	Instantaneous	NA	
I-19F	Excess Flow Check	2	Process	NA	O	O	O	NA	Pressure	Instantaneous	NA	
I-20A	Excess Flow Check	2	Process	NA	O	O	O	NA	Pressure	Instantaneous	NA	
I-20B	Excess Flow Check	2	Process	NA	O	O	O	NA	Pressure	Instantaneous	NA	
I-20C	Excess Flow Check	2	Process	NA	O	O	O	NA	Pressure	Instantaneous	NA	
I-20D	Excess Flow Check	2	Process	NA	O	O	O	NA	Pressure	Instantaneous	NA	
I-20E	Excess Flow Check	2	Process	NA	O	O	O	NA	Pressure	Instantaneous	NA	
I-20F	Excess Flow Check	2	Process	NA	O	O	O	NA	Pressure	Instantaneous	NA	
I-21A,B,C,D,E,F	---		---	--	--	--	--	--	---	---	--	Spare
I-22A & D	Excess Flow Check	2	Process	NA	O	O	O	NA	Flow	Instantaneous	NA	
	Excess Flow Check	2	Process	NA	O	O	O	NA	Flow	Instantaneous	NA	
I-22B & C	Excess Flow Check	2	Process	NA	O	O	O	NA	Flow	Instantaneous	NA	
	Excess Flow Check	2	Process	NA	O	O	O	NA	Flow	Instantaneous	NA	
	Excess Flow Check	2	Process	NA	O	O	O	NA	Flow	Instantaneous	NA	
	Excess Flow Check	2	Process	NA	O	O	O	NA	Flow	Instantaneous	NA	

CONTAINMENT PENETRATION NUMBER	NRC GDC	LINE ISOLATED	FLUID CONTAINED	LINE SIZE (in.)	ESF SYSTEM (NOTE 21)	THROUGH LINE LEAKAGE CLASSIFICATION (NOTE 14)	VALVE ARRANGEMENT FIGURE 6.2-32	VALVE NUMBER	LOCATION WITH RESPECT TO CONTAINMENT	TYPE C TEST	LENGTH OF PIPE FROM CONTAINMENT TO OUTERMOST VALVE (ft)
I-22E & F	55 (Note 26)	Recirc. Pump ΔP	Reactor Water	3/4 3/4	No No	C C	Detail (w) Detail (w)	1&2B33-F315A 1&2B33-F315B	Outside Outside	No (Note 33) No (Note 33)	10 Max. 10 Max.
I-23A	---	---	---	---	--	-	---	---	---	---	10 Max.
I-23B	55 (Note 26)	Recirc. Pump Suction Press.	Reactor Water	3/4	No	C	Detail (w)	1&2B33-F301A	Outside	No (Note 33)	10 Max.
I-23C & D	55 (Note 26)	Recirc. Pump Flow	Reactor Water	3/4 3/4 3/4 3/4	No No No No	C C C C	Detail (x) Detail (x)	1&2B33-F307C 1&2B33-F307D 1&2B33-F305C 1&2B33-F305D	Outside Outside Outside Outside	No (Note 33) No (Note 33) No (Note 33) No (Note 33)	10 Max. 10 Max. 10 Max. 10 Max.
I-23E & F	55 (Note 26)	RHR Shutdown Flow	Reactor Water	3/4 3/4	Yes Yes	C C	Detail (w) Detail (w)	1&2E12-F359B 1&2E12-F359A	Outside Outside	No (Note 33) No (Note 33)	10 Max. 10 Max.
I-24A,B,C,D,E,F	--	---	---	---	--	--	--	---	---	---	10 Max.
I-25A & B	55 (Note 26)	RHR Line Integrity	Reactor Water	3/4 3/4	Yes Yes	C C	Detail (w) Detail (w)	1&2E12-F319 1&2E12-F317	Outside Outside	No (Note 33) No (Note 33)	10 Max. 10 Max.
I-25C, D, E, F	---	---	---	---	---	-	--	---	---	---	10 Max.
I-26	56 (Note 32)	Drywell Press.	Air	3/4	Yes	C	Detail (w)	1&2B21-F367	Outside	No (Note 33)	10 Max.
I-27A & D	55 (Note 26)	Recirc. Pump Flow	Reactor Water	3/4 3/4 3/4 3/4	No	C C C C	Detail (x) Detail (x)	1&2B33-F307A 1&2B33-F307B 1&2B33-F305A 1&2B33-F305B	Outside Outside Outside Outside	No (Note 33) No (Note 33) No (Note 33) No (Note 33)	10 Max. 10 Max. 10 Max. 10 Max.

CONTAINMENT PENETRATION NUMBER	VALVE TYPE	ASME SECTION III CODE CLASS	PRIMARY METHOD OF ACTUATION	SECONDARY METHOD OF ACTUATION	NORMAL VALVE POSITION	SHUTDOWN VALVE POSITION	POST ACCIDENT POSITION	POWER FAILURE VALVE POSITION (6)	ISOLATION SIGNAL	VALVE CLOSURE TIME (7)	POWER SOURCE	REMARKS
I-22E & F	Excess Flow Check	2	Process	NA	O	O	O	NA	Flow	Instantaneous	NA	
	Excess Flow Check	2	Process	NA	O	O	O	NA	Flow	Instantaneous	NA	
I-23A	---	-	---	--	-	-	-	--	----	---	--	Spare
I-23B	Excess Flow Check	2	Process	NA	O	O	O	NA	Flow	Instantaneous	NA	
I-23C & D	Excess Flow Check	2	Process	NA	O	O	O	NA	Flow	Instantaneous	NA	
	Excess Flow Check	2	Process	NA	O	O	O	NA	Flow	Instantaneous	NA	
	Excess Flow Check	2	Process	NA	O	O	O	NA	Flow	Instantaneous	NA	
	Excess Flow Check	2	Process	NA	O	O	O	NA	Flow	Instantaneous	NA	
I-23E & F	Excess Flow Check	2	Process	NA	O	O	O	NA	Flow	Instantaneous	NA	
	Excess Flow Check	2	Process	NA	O	O	O	NA	Flow	Instantaneous	NA	
I-24A,B,C,D,E,F	---	-	---	---	-	-	-	--	----	---	--	Spare
I-25A & B	Excess Flow Check	2	Process	NA	O	O	O	NA	Flow	Instantaneous	NA	
	Excess Flow Check	2	Process	NA	O	O	O	NA	Flow	Instantaneous	NA	
I-25C, D, E, F	---	-	---	--	-	-	-	--	--	---	--	Spare
I-26	Excess Flow Check	2	Process	NA	O	O	O	NA	Pressure	Instantaneous	NA	
I-27A & D	Excess Flow Check	2	Process	NA	O	O	O	NA	Pressure	Instantaneous		
	Excess Flow Check	2	Process	NA	O	O	O	NA	Pressure	Instantaneous	NA NA	
	Excess Flow Check	2	Process	NA	O	O	O	NA	Pressure	Instantaneous	NA NA	
	Excess Flow Check	2	Process	NA	O	O	O	NA	Pressure	Instantaneous		

CONTAINMENT PENETRATION NUMBER	NRC GDC	LINE ISOLATED	FLUID CONTAINED	LINE SIZE (in)	ESF SYSTEM (NOTE 21)	THROUGH LINE LEAKAGE CLASSIFICATION (NOTE 14,15)	VALVE ARRANGEMENT FIGURE 6.2-31	VALVE NUMBER	LOCATION WITH RESPECT TO CONTAINMENT	TYPE C TEST	LENGTH OF PIPE FROM CONTAINMENT TO OUTERMOST VALVE (ft)
I-27B & C	55 (Note 26)	RHR Shutdown Flow	Reactor Water	3/4 3/4	Yes	C C	Detail (w) Detail (w)	1&2E12-F360A 1&2E12-F360B	Outside Outside	No (Note 33) No (Note 33)	10 Max. 10 Max.
I-27E&F	55 (Note 26)	Recirc. Pump Seal Press.	Reactor Water	3/4 3/4	No No	C C	Detail (w)	1&2B33-F317B 1&2B33-F319B	Outside Outside	No (Note 33) No (Note 33)	10 Max. 10 Max.
I-28A	55 (Note 26)	Recirc. Pump Suction Press.	Reactor Water	3/4	No	C	Detail (w)	1&2B33-F301B	Outside	No (Note 33)	10 Max.
I-28B & C	55 (Note 26)	Recirc. Pump P	Reactor Water	3/4 3/4	No No	C C	Detail (w) Detail (w)	1&2B33-F315D 1&2B33-F315C	Outside Outside	No (Note 33) No (Note 33)	10 Max. 10 Max.
I-28D & E	55 (Note 25)	Recirc. Pump Flow	Reactor Water	3/4 3/4 3/4 3/4	No No No No	C C C C	Detail (x) Detail (x)	1&2B33-F313A 1&2B33-F313B 1&2B33-F311A 1&2B33-F311B	Outside Outside Outside Outside	No (Note 33) No (Note 33) No (Note 33) No (Note 33)	10 Max. 10 Max. 10 Max. 10 Max.
I-28F	55 (Note 26)	RPV Drain Flow	Reactor Water	3/4	No	C	Detail(w)	1&2G33-F309	Outside	No (Note 33)	10 Max.
I-29A, D, E, F	55 (Note 26)	Steam Flow	Steam	3/4 3/4 3/4 3/4	No No No No	C C C C	Detail(w) Detail(w) Detail(w) Detail(w)	1&2B21-F326D 1&2B21-F325D 1&2B21-F325C 1&2B21-F326C	Outside Outside Outside Outside	No (Note 33) No (Note 33) No (Note 33) No (Note 33)	10 Max. 10 Max. 10 Max. 10 Max.
I-29B	55 (Note 26)	Core ΔP	Reactor Water	3/4	Yes	C	Detail(w)	1&2B21-F350	Outside	No (Note 33)	10 Max.
I-29C	55 (Note 26)	RPV Bottom Head Drain Flow	Reactor Water	3/4	No	C	Detail(w)	1&2B21-F346	Outside	No (Note 33)	10 Max.
I-30A & B	55 (Note 26)	RPV/HPCS ΔP	Reactor Water	3/4 3/4	No No	C C	Detail(w) Detail(w)	1&2B21-F348 1&2E22-F304	Outside Outside	No (Note 33) No (Note 33)	10 Max. 10 Max.
I-30C, D, E, F	57 (Note 44)	MSIV Accumulator Pressure	Air	3/4	No	B	Detail(j)	1&2B21-F329A,B,C,D	Outside	No	10 Max.
I-31A I-31B I-31C I-31D I-31E I-31F	55 (Note 26)	Jet Pump Flow	Reactor Water	3/4 3/4 3/4 3/4 3/4 3/4	No No No No No No	C C C C C C	Detail(w) Detail(w) Detail(w) Detail(w) Detail(w) Detail(w)	1&2B21-F471 1&2B21-F469 1&2B21-F473 1&2B21-F465B 1&2B21-F475B 1&2B21-F475A	Outside Outside Outside Outside Outside Outside	No (Note 33) No (Note 33) No (Note 33) No (Note 33) No (Note 33) No (Note 33)	10 Max. 10 Max. 10 Max. 10 Max. 10 Max. 10 Max

CONTAINMENT PENETRATION NUMBER	VALVE TYPE	ASME SECTION III CLASS	PRIMARY METHOD OF ACTUATION	SECONDARY METHOD OF ACTUATION	NORMAL VALVE POSITION	SHUTDOWN VALVE POSITION	POST ACCIDENT POSITION	POWER FAILURE VALVE POSITION (6)	ISOLATION SIGNAL	VALVE CLOSURE TIME (7)	POWER SOURCE	REMARKS
I-27B & C	EFC	2	Process	NA	O	O	O	NA	Flow	Instantaneous	NA	
	EFC	2	Process	NA	O	O	O	NA	Flow	Instantaneous	NA	
I-27E & F	EFC	2	Process	NA	O	O	O	NA	Flow	Instantaneous	NA	
	EFC		Process	NA	O	O	O	NA	Flow	Instantaneous	NA	
I-28A	EFC	2	Process	NA	O	O	O	NA	Flow	Instantaneous	NA	
I-28B & C	EFC	2	Process	NA	O	O	O	NA	Flow	Instantaneous	NA	
	EFC	2	Process	NA	O	O	O	NA	Flow	Instantaneous	NA	
I-28D & E	EFC	2	Process	NA	O	O	O	NA	Flow	Instantaneous	NA	
	EFC	2	Process	NA	O	O	O	NA	Flow	Instantaneous	NA	
	EFC	2	Process	NA	O	O	O	NA	Flow	Instantaneous	NA	
	EFC	2	Process	NA	O	O	O	NA	Flow	Instantaneous	NA	
I-28F	EFC	2	Process	NA	O	O	O	NA	Flow	Instantaneous	NA	
I-29A,D,E,F	EFC	2	Process	NA	O	O	O	NA	Flow	Instantaneous	NA	
	EFC	2	Process	NA	O	O	O	NA	Flow	Instantaneous	NA	
	EFC	2	Process	NA	O	O	O	NA	Flow	Instantaneous	NA	
	EFC	2	Process	NA	O	O	O	NA	Flow	Instantaneous	NA	
I-29B	EFC	2	Process	NA	O	O	O	NA	Flow	Instantaneous	NA	
I-29C	EFC	2	Process	NA	O	O	O	NA	Flow	Instantaneous	NA	
I-30A & B	EFC	2	Process	NA	O	O	O	NA	Flow	Instantaneous	NA	
	EFC	2	Process	NA	O	O	O	NA	Flow	Instantaneous	NA	
I-30C,D,E,F	Manual	2	Manual	NA	O	O	O	NA	--	--	--	
I-31A	EFC	2	Process	NA	O	O	O	NA	Flow	Instantaneous	NA	
I-31B	EFC	2	Process	NA	O	O	O	NA	Flow	Instantaneous	NA	
I-31C	EFC	2	Process	NA	O	O	O	NA	Flow	Instantaneous	NA	
I-31D	EFC	2	Process	NA	O	O	O	NA	Flow	Instantaneous	NA	
I-31E	EFC	2	Process	NA	O	O	O	NA	Flow	Instantaneous	NA	
I-31F	EFC	2	Process	NA	O	O	O	NA	Flow	Instantaneous	NA	

EFC = Excess Flow Check

CONTAINMENT PENETRATION NUMBER	NRC GDC	LINE ISOLATED	FLUID CONTAINED	LINE SIZE (in)	ESF SYSTEM (NOTE 21)	THROUGH LINE LEAKAGE CLASSIFICATION (NOTE 14,15)	VALVE ARRANGEMENT FIGURE 6.2-31	VALVE NUMBER	LOCATION WITH RESPECT TO CONTAINMENT	TYPE C TEST	LENGTH OF PIPE FROM CONTAINMENT TO OUTERMOST VALVE (ft)
I-32A I-32B I-32C I-32D I-32E I-32F	55 (Note 26)	Jet Pump Flow	Reactor Water	3/4 3/4 3/4 3/4 3/4 3/4	No No No No No No	C C C C C C	Detail (w) Detail (w) Detail (w) Detail (w) Detail (w) Detail (w)	1&2B21-F465A 1&2B21-F467 1&2B21-F463 1&2B21-F459 1&2B21-F457 1&2B21-F461	Outside Outside Outside Outside Outside Outside	No (Note 33) No (Note 33) No (Note 33) No (Note 33) No (Note 33) No (Note 33)	10 Max. 10 Max. 10 Max. 10 Max. 10 Max. 10 Max.
I-33	56 (Note 32)	Drywell Pressure	Air	3/4	Yes	C	Detail (w)	1&2B21-F380	Outside	No (Note 33)	10 Max.
I-34A, D, E, F	55 (Note 26)	Steam Flow	Steam	3/4 3/4 3/4 3/4	Yes Yes Yes Yes	C C C C	Detail (w) Detail (w) Detail (w) Detail (w)	1&2B21-F328D 1&2B21-F328C 1&2B21-F327C 1&2B21-F327D	Outside Outside Outside Outside	No (Note 33) No (Note 33) No (Note 33) No (Note 33)	10 Max. 10 Max. 10 Max. 10 Max.
I-34B & C	---	---	---	---	---	-	---	---	---	---	---
I-35	56 (Note 28) 56	Post LOCA Containment Monitoring HRSS Sampling	Air Air	1/2 1/2	Yes No	B A(b) A(b)	Detail (k) Detail (g) Detail (g)	1&2CM023B 1&2CM085 1&2CM086	Outside Outside Outside	No (Note 40) Yes Yes	10 Max. 10 Max. 10 Max.
I-36	56 (Note 28)	Post LOCA Containment Monitoring	Air	1/2 1/2 1/2	Yes No No	B A (b) A (b)	Detail (k) Detail (g) Detail (g)	1&2CM024A 1&2CM027 1&2CM028	Outside Outside Outside	No (Note 40) Yes Yes	10 Max. Not Applicable 10 Max.
I-37A, B, C, D	55 (Note 26)	Steam Flow	Steam	3/4 3/4 3/4 3/4	Yes Yes Yes Yes	C C C C	Detail (w) Detail (w) Detail (w) Detail (w)	1&2B21-F325A 1&2B21-F326A 1&2B21-F325B 1&2B21-F326B	Outside Outside Outside Outside	No (Note 33) No (Note 33) No (Note 33) No (Note 33)	10 Max. 10 Max. 10 Max. 10 Max.
I-37E & F	---	---	---	---	---	---	---	---	---	---	10 Max.
I-38 & 39	NA	Supp. Chamber	Air	1 1/4	No	--- ---	--- ---	--- ---	--- ---	--- ---	10 Max. 10 Max.

CONTAINMENT PENETRATION NUMBER	VALVE TYPE	ASME SECTION III CLASS	PRIMARY METHOD OF ACTUATION	SECONDARY METHOD OF ACTUATION	NORMAL VALVE POSITION	SHUTDOWN VALVE POSITION	POST ACCIDENT POSITION	POWER FAILURE VALVE POSITION (6)	ISOLATION SIGNAL	VALVE CLOSURE TIME (7)	POWER SOURCE	REMARKS
I-32A	EFC	2	Process	NA	O	O	O	NA	Flow	Instantaneous	NA	
I-32B	EFC	2	Process	NA	O	O	O	NA	Flow	Instantaneous	NA	
I-32C	EFC	2	Process	NA	O	O	O	NA	Flow	Instantaneous	NA	
I-32D	EFC	2	Process	NA	O	O	O	NA	Flow	Instantaneous	NA	
I-32E	EFC	2	Process	NA	O	O	O	NA	Flow	Instantaneous	NA	
I-32F	EFC	2	Process	NA	O	O	O	NA	Flow	Instantaneous	NA	
I-33	EFC	2	Process	NA	O	O	O	NA	Pressure	Instantaneous	NA	
I-34A,D,E,F	EFC	2	Process	NA	O	O	O	NA	Pressure	Instantaneous	NA	
	EFC	2	Process	NA	O	O	O	NA	Pressure	Instantaneous	NA	
	EFC	2	Process	NA	O	O	O	NA	Pressure	Instantaneous	NA	
	EFC	2	Process	NA	O	O	O	NA	Pressure	Instantaneous	NA	
I-34B,C	--	--	--	--	--	--	--	--	--	--	--	Spare
I-35	SO Globe	2	RM	N/A	C/O	C	O	O	RM	5 sec.	ESS 2	
	SO Globe	2	Manual	N/A	C	C	C/O	C	---	---	N/A	
	SO Globe	2	Manual	N/A	C	C	C/O	C	---	---	N/A	
I-36	SO Globe	2	RM	N/A	C/O	C	O	O	RM	5 sec.	ESS 1	Note (20)
	SO Globe	2	Auto	RM	O	O	C	C	B,F,RM	5 sec.	ESS 2	
	SO Globe	2	Auto	RM	O	O	C	C	B,F,RM	5 sec.	ESS 1	
I-37A,B,C,D	EFC	2	Process	NA	O	O	O	NA	Flow	Instantaneous	NA	
	EFC	2	Process	NA	O	O	O	NA	Flow	Instantaneous	NA	
	EFC	2	Process	NA	O	O	O	NA	Flow	Instantaneous	NA	
	EFC	2	Process	NA	O	O	O	NA	Flow	Instantaneous	NA	
I-37E&F	--	--	--	--	--	--	--	--	--	--	--	Spare
I-38 & 39	--	--	--	--	--	--	--	--	--	--	--	RTDs are provided through these connections
	--	--	--	--	--	--	--	--	--	--	--	

EFC = Excess Flow Check

CONTAINMENT PENETRATION NUMBER	NRC GDC	LINE ISOLATED	FLUID CONTAINED	LINE SIZE (in.)	ESF SYSTEM (NOTE 21)	THROUGH LINE LEAKAGE CLASSIFICATION (NOTE 14)	VALVE ARRANGEMENT FIGURE 6.2-32	VALVE NUMBER	LOCATION WITH RESPECT TO CONTAINMENT	TYPE C TEST	LENGTH OF PIPE FROM CONTAINMENT TO OUTERMOST VALVE (ft)
I-40,41, 42,43	56 (Note 32)	Supp. Pool Water Level	Supp. Pool Water	3/4 3/4 3/4 3/4 3/4 3/4 3/4 3/4	No No No No No No No No	Exempt Exempt Exempt Exempt Exempt Exempt Exempt Exempt	Detail (v) Detail (v) Detail (v) Detail (v) Detail (v) Detail (v) Detail (v) Detail (v)	1&2CM039 1&2CM040 1&2CM041 1&2CM042 1&2CM043 1&2CM044 1&2CM045 1&2CM046	Outside Outside Outside Outside Outside Outside Outside Outside	No (Note 32) No (Note 32) No (Note 32) No (Note 32) No (Note 32) No (Note 32) No (Note 32) No (Note 32)	10 Max. 10 Max. 10 Max. 10 Max. 10 Max. 10 Max. 10 Max. 10 Max.
I-44 & 46	--	Supp. Pool Water Temp.	--	1 1/4 1 1/4	--	--	--				10 Max. 10 Max.
I-45	56 (Note 28)	Drywell Air Sampling Post LOCA Cont. Mont. Drywell Tritium Grab Sample	Air	1	No No Yes No No	A (b) A (b) B A(b) A(b)	Detail (g) Detail (k) Detail (g) 	1&2CM034 1&2CM033 1&2CM025A 1&2CM020B 1&2CM019B	Outside Outside Outside Outside Outside	Yes Yes No (Note 40) Yes Yes	10 Max. 10 Max. 10 Max. 10 Max. 10 Max.
I-47	56(Note 28)	Post LOCA Containment Monitoring	Air	1 1/4	Yes	B	Detail (w)	1&2CM026B	Outside	No(Note 40)	10 Max
	56	HRSS Sampling	Air	1/2	No	A(b) A(b)	Detail (g) Detail (g)	1&2CM089 1&2CM090	Outside Outside	Yes Yes	10 Max. 10 Max..
I-48 & 49	56 (Note 32)	Supp. Pool Water Level	Supp. Pool Water	1 1/4 1 1/4	No No	C C	Detail (w) Detail (w)	1&2E22-F341 1&2E22-F342	Outside Outside	No(Note 32) No(Note 32)	10 Max. 10 Max.
I-50	56 (Note 28)	Post LOCA Containment Monitoring	Air	1/2	Yes	B	Detail (k)	1&2CM021B	Outside	No (Note 40)	10 Max.
	56	HRSS Sampling	Air	1/2	No	A(b) A(b)	Detail (g) Detail (g)	1&2CM085 1&2CM086	Outside Outside	Yes Yes	10 Max. 10 Max.

CONTAINMENT PENETRATION NUMBER	VALVE TYPE	ASME SECTION III CODE CLASS	PRIMARY METHOD OF ACTUATION	SECONDARY METHOD OF ACTUATION	NORMAL VALVE POSITION	SHUTDOWN VALVE POSITION	POST ACCIDENT POSITION	POWER FAILURE VALVE POSITION (6)	ISOLATION SIGNAL	VALVE CLOSURE TIME (7)	POWER SOURCE	REMARKS
I-40,41, 42,43	Globe	2	Manual	N/A	C	C	C	NA	Flow	---	NA	
	Globe	2	Manual	N/A	C	C	C	NA	Flow	---	NA	
	Globe	2	Manual	N/A	C	C	C	NA	Flow	---	NA	
	Globe	2	Manual	N/A	C	C	C	NA	Flow	---	NA	
	Globe	2	Manual	N/A	C	C	C	NA	---	---	NA	
	Globe	2	Manual	N/A	C	C	C	NA	---	---	NA	
	Globe	2	Manual	N/A	C	C	C	NA	---	---	NA	
	Globe	2	Manual	N/A	C	C	C	NA	---	---	NA	
I-44 & 46												RTDs are provided through these connecti
I-45	SO Globe	2	Auto	RM	O	O	C	C	B,F,RM	5 sec.	ESS 2	(Note 20)
	SO Globe	2	Auto	RM	O	O	C	C	B,F,RM	5 sec.	ESS 1	
	SO Globe	2	Auto	RM	C/O	C	O	O	RM (Note 37)	5 sec.	ESS 1	
	SO Globe	2	Auto	RM	O	O	C	C	B,F,RM	5 sec.	ESS 2	(Note 20)
	SO Globe	2	Auto	RM	O	O	C	C	B,F,RM	5 sec.	ESS 1	
I-47	SO Globe	2	Auto	RM	C/O	C	O	O	RM(Note37)	5 sec.	ESS 2	
	SO GLOBE	2	Manual	N/A	C	C	C/O	C	---	---	N/A	
	SO GLOBE	2	Manual	N/A	C	C	C/O	C	---	---	N/A	
I-48 & 49	Excess Flow Check	2	Process	NA	O	O	O	NA	Flow	Instantaneous	NA	
	Excess Flow Check	2	Process	NA	O	O	O	NA	Flow	Instantaneous	NA	
I-50	SO Globe	2	Auto	RM	C/O	C	O	O	RM (Note37)	5 sec.	ESS 2	
	SO Globe	2	Manual	N/A	C	C	C/O	C	---	---	N/A	
	SO Globe	2	Manual	N/A	C	C	C/O	C	---	---	N/A	

SIGNAL	DESCRIPTION
A	Reactor vessel low water level level 3 - (A scram occurs at this level also. This is the higher of the two low water level signals.)
B	Reactor vessel low low water level level 2 - (The RCIC and HPCS systems are initiated at this level also. (This is the lower of the two low water level signals.)
C	High radiation - Main steam
D	Line break - High area temperature or very high system flow.
E	Main condenser low vacuum.
F	High drywell pressure.
G	Reactor vessel low low low water level (Level 1) or high drywell pressure (Emergency Core Cooling System are started).
H	Reactor vessel low low low water level (Level 1)
J	Line break in cleanup system - high space temperature.
M	Line break in RHR shutdown and head cooling (high space temberature).
P	Low main steamline pressure at inlet turbine (RUN mode only).
U	High reactor vessel pressure - close RHR shutdown cooling valves and head cooling valves.
Y	High radiation, fuel pool ventilation exhaust.
Z	High radiaion, reactor building ventilation exhaust.
RM	Remote manual switch from control room. (All regular Class A and Class B isolation valves are capable of remote manual operation from the control room.)
RME	Remote manual switch from Auxiliary Electric Equipment Room. Note - position indication also available in Control Room in group summary position indicator lights.

These notes are keyed by number to correspond to numbers in parenthesis in Table 6.2-21.

1. Main steam isolation valves require that both solenoid pilots be de-energized to close valves. Accumulator air pressure plus spring force together close valves when both pilots are de-energized. Voltage failure at only one pilot does not cause valve closure. The valves are designed to fully close in less than 5 seconds.

2. Suppression pool spray (1(2)E12-F027A/B) and suppression pool cooling valves (1(2)E12-F024A/B) have interlocks that allow them to be manually reopened after automatic closure. This setup permits suppression pool spray, for high drywell pressure conditions, and/or suppression pool water cooling. The drywell spray valves (1(2)E12-F016A/B, 1(2)E12-F017A/B), do not receive any automatic closure signals.

3. Testable check valves are provided with an air operator for remote opening with zero differential pressure across the valve seat. These valves will close on reverse flow even though the test switches may be positioned for open. The valves open when pump pressure exceeds reactor pressure even though the test switch may be closed. The remote testable feature and control room indication has been eliminated from the Division 1, 2, and 3 ECCS and RHR Shutdown Cooling Return testable check valves. The air operators are removed from check valves 1(2)E12-F041A/B/C, 1(2)E12-F050A/B, 1(2)E21-F006, and 1(2)E22-F005, and a mechanism is installed on each valve to pin it open for maintenance and testing.

4. In the normal configuration the lines are considered to be an extension of primary containment. If a vacuum breaker valve is inoperable, the butterfly valve will be closed to prevent bypass leakage. If a vacuum breaker valve is subsequently removed, a blind flange will be added, and the flange and butterfly valve will form the containment boundary. The vacuum breaker valves will be leakage tested as part of the periodic low pressure suppression bypass leakage test. The acceptance limits are based on the allowable suppression bypass capability of the containment.

5. A-c motor-operated valves required for isolation functions are powered from the a-c standby power buses. D-c operated isolation valves are powered from the station batteries.

6. All motor-operated isolation valves remain in the last position upon failure of valve power. All air-operated valves close on motive air failure except the VQ Butterfly valves which require their solenoid valves to be deenergized.

7. The standard operating times for power actuated valves based on actual stem travel shall be less than or equal to 110% of the nominal values below:

	Motor-operated	Air-Operated
Gate valves	12 in./min	Not applicable
Globe valves	4 in./min	4 in./min
Butterfly valves	30 - 90 seconds	0 - 10 seconds

8. Reactor building vent exhaust high radiation signal “z” and fuel pool ventilation exhaust high radiation signal "Y" are generated by two trip units; this requires one unit at high trip or both units at downscale (instrument failure trip), in order to initiate isolation.

9. Valves can be opened or closed by remote manual switch for operating convenience during any mode of reactor operation except when an automatic signal is present.

10. Normal status position of valve (open or closed) is the position during normal power operation of the reactor (see "Normal Status" column).

11. Deleted.

12. Deleted.

13. Deleted.

14. Categories indicated are in accordance with Subsection ISTC – 1300 of ASME OM Code, 2001 Edition through 2003 Addenda. The types of leakage tests are as follows: (a) water test and (b) air test. Exempt valves are those used for testing, draining, venting, maintenance or operational convenience.

15. The leakage criteria for these valves is specified in 10 CFR 50 Appendix J and the LaSalle Primary Containment Leak Rate Testing Program.

16. Deleted.

17. The outboard check valves on the feedwater return lines are provided with an air operator for testing the valves to ensure that the disks are not frozen in the open position. The actuator moves the disk partially into the flow stream, but is not capable of completely closing the valve against flow. The feedwater valve actuator is used to apply seating force to the valve for ensuring leaktightness at low differential pressures. The actuator will be exercised to assure operability prior to leak testing.

18. The TIP drive guide tubes provide a sealed path for the flexible drive cable of the TIP probes. The TIP tubing seals the TIP system from the reactor coolant and forms a leak tight boundary designed for reactor coolant pressure boundary conditions. The shear valve is provided to cut the cable in the event that the drive cable cannot be withdrawn, and the ball provides the guide tubes with shut-off capability.

The LaSalle TIP system design specifications require that the maximum leakage rate of the ball and shear valves shall be in accordance with the Manufacturers Standardization Society (Hydrostatic Testing of Valves). The ball valves are 100% leak tested to the following criteria by the manufacturer:

Pressure	0 - 62 psig
Temperature	340°F
Leak Rate	10^{-3} cm^3 /s

A statistically chosen sample of the shear valves is tested by the manufacturer to the following criteria:

Pressure	0 - 125 psig
Temperature	340°F
Leak Rate	10^{-3} cm^3 /sec STP.

The shear valves have explosive squibs and require testing to destruction. They cannot therefore be 100% tested nor can they be tested in accordance with 10 CFR 50 Appendix J requirements after installation.

Isolation is accomplished by a seismically qualified solenoid-operated ball valve, which is normally closed. Ball valve position is indicated in the control room. The ball valve is periodically leak tested in accordance with the LaSalle 10 CFR 50 Appendix J Program and the acceptable leakage limits for these valves are in accordance with the Appendix J program.

When the TIP system cable is inserted, the ball valve of the selected tube opens automatically so that the probe and cable may advance. A maximum of four valves may be opened at any one time to conduct calibration, and any one guide tube is used, at most, a few hours per year.

If closure of the line is required during calibration, a signal causes the cable to be retracted and the ball valve to close automatically after completion of cable withdrawal. If a TIP cable fails to withdraw or a ball valve fails to close, each line is equipped with an explosive shear valve.

If a failure occurs, the shear valve would be manually actuated from the Main Control Room to shear the TIP cable and isolate the penetration. Because the TIP shear valve requires testing to destruction, it is not tested in accordance with 10 CFR 50 Appendix J, but instead is tested as specified in Technical Specification. The Technical Specification verifies continuity of the explosive charge and batch sampling testing of the explosive squib charges, with replacement of the explosive squib before expiration of the shelf-life and operating life. A statistical sample of the shear valves are leak tested in the manufacturers shop to ensure that the leakage limits conform to the design specification limits of 10^{-3} cm^3/sec.

19. The hydraulic lines are sealed pipe designed for 2000 psig operating pressure.

20. Test pressure is not in the same direction as the pressure existing when the valve is required to perform the safety function as required by Appendix J to 10 CFR 50. Either manufacturers' test data, site test results or justification (e.g., reverse test pressure tending to lift disk from seat) will be available on site to verify that testing in the reverse direction will provide either equivalent or more conservative results.

21. Although the valves listed may be included in the containment isolation system which is an ESF system, a "yes" designation is given only for those valves in systems where the parent system containing the valve is an ESF system.

22. The valves associated with RHR "A" loop are powered from ESS1 sources. The valves associated with RHR "B" and "C" loops are supplied from ESS2 power sources.

23. The power source for the valves associated with penetrations M-23 (Unit 2), M-33 (Unit l) and M-106 is ESS1. The power source for the valves associated with penetrations M-53 and M-104 is ESS2. This arrangement was used to maintain redundancy of function for the combustible gas control system. The valves are closed during normal plant operation, and are open only for periodic testing and following a LOCA.

24. Criterion 55 concerns those lines of the reactor coolant pressure boundary penetrating the primary reactor containment. The control rod drive (CRD) insert and withdraw lines are not part of the reactor coolant pressure boundary. The basis to which the CRD lines are designed is commensurate with the safety importance of isolating these lines. Since these lines are vital to the scram function, their operability is of utmost concern.

    In the design of this system, it has been accepted practice to omit automatic valves for isolation purposes, as this introduces a possible failure mechanism. As a means of providing positive actuation, manual shutoff valves (1&2C11D001-101 and –102) are used. The charging water, drive water and cooling water headers are provided with a check valve (1&2C11D001-115, -137 and -138) within the hydraulic control unit (HCU), a Seismic Category I module, and the normally closed solenoid valves (1&2C11D001-120, -121, -122 and -123). These valves will prevent any direct flow away from containment. These valves are shown on Sheet 3 of Drawing M-100 (Unit 1) and M-146 (Unit 2).

    If an insert line fails, a ball check valve provided in each drive is designed to seal off the broken line by using reactor pressure to shift the ball check valve to the upper seat. This feature also prevents any direct flow away from the primary containment.

When the HCU's are pressurized, leaks resulting from degraded piping integrity would be observed by the Operators on their daily rounds. In addition, several indicators in the control room, such as temperature and pressure of CRD cooling water or drywell sump pump operation, indicates whether leakage is excessive. The maximum leakage expected at this penetration is 3 gpm when the RPV is still pressurized (about 1000 psi). This leakage also assumes a single active failure of a check valve inside the HCU. After the reactor vessel is depressurized, the CRD leakage will decrease to about 0.5 gpm. It may also be said that leakage monitoring of the CRD insert and withdraw lines is provided by the overall type A leakage rate test. Since the RPV and nonseismic portions of the CRD system are vented during the performance of the Type A test, any leakage from these lines would be included in the total Type A test leakage.

The flowout of the CRD is restricted through the HCU performance test requirements to ensure that HCU leakage does not exceed 0.2 gpm. The maximum leakage expected for these penetrations is 0.2 gpm per HCU. If a single failure is assumed, the maximum leakage would be 3 gpm. Seismic tests have demonstrated the seal integrity of the CRD system. Maximum leakage following these tests did not exceed 3 gpm.

The system design criteria are as follows:

	Seismic Category	Quality Group Classification	Quality Assurance Classification
Valves; insert and withdraw	I	B	I
Insert and withdraw line piping	I	B	I

The CRD insert and withdraw lines are compatible with the criteria intended by 10 CFR 50, Appendix J for Type C testing, since the acceptance criterion for Type C testing allows demonstration of fluid leakage rates by associated bases. The maximum leakage expected has been factored in with the total allowable containment penetration leakage and determined to be acceptabe.

25. The recirculation pump seal water line extends from the recirculation pump through the drywell and connects to the CRD supply line outside the primary containment. The seal water line forms a part of the reactor coolant pressure boundary; therefore, the consequences of failing this line have been evaluated. This evaluation shows that the consequences of breaking this line are less severe than failing an instrument line. Therefore, the two check valves in series provide sufficient isolation capability for postulated failure of this line.

    These lines are high-pressure lines coming from the discharge of the CRD pumps to the recirculation pump seals. They are provided with a check valve inside the containment and a check valve outside the containment.

    The inside and outside check will receive a Type C local leak test with water as the testing mechanism during refueling outages.

26. See Note 33.

27. The Hydraulic Control Unit (HCU) is a factory-assembled engineered module of valves, tubing, piping, and stored water which controls a single control rod drive by the application of precisely timed sequences of pressures and flows to accomplish slow insertion or withdrawal of the control rods for power control, and rapid insertion for reactor scram.

Although the hydraulic control unit, as a unit, is field installed and connected to process piping, many of its internal parts differ markedly from process piping components because of the more complex functions they must provide.

Thus, although the codes and standards invoked by Groups A, B, C and D pressure integrity quality levels clearly apply at all levels to the interfaces between the HCU and the connecting conventional piping components (e.g., pipe nipples, fittings, simple hand valves, etc.), it is considered that they do not apply to the specialty parts (e.g., solenoid valves, pneumatic components, and instruments). The HCU shutoff (isolation) valves are Quality Group B.

The design and construction specifications for the HCU do invoke such codes and standards as can be reasonably applied to individual parts in developing required quality levels, but these codes and standards are supplemented with additional requirements for these parts and for the remaining parts and details. For example, 1) all welds are penetrant tested (PT), 2) all socket welds are inspected for gaps between pipe and socket bottom, 3) all welding is performed by qualified welders, and 4) all work is done per written procedures. Quality Group D is generally applicable because the codes and standards involked by that group contain clauses which permit the use of manufacturer's standards and proven design techniques which are not explicitly defined within the codes of Quality Group A, B, or C. This is supplemented by the QC techniques.

28. These lines have been evaluated to an acceptable alternative design basis other than that specifically listed in GDC 55 and 56. This alternate basis is found in SRP 6.2.4, Rev. 2, Item II.e and the evaluation to the criteria specified therein is as follows:

a. All lines are in engineered safety feature or engineered safety featured-related systems.

b. System reliability can readily be seen to be greater when only a single valve is provided, since the addition of another valve in series provides an additional potential point of failure, and, in the case of relief valve discharge lines, the installation of an additional valve is actually prohibited by the ASME Code.

c. The systems are closed outside containment.

d. A single active failure of these ESF systems can be accommodated.

e. The systems outside containment are protected from missiles consistent with their classification as ESF systems.

f. The systems are designed to Seismic Category I standards.

g. The systems are classified as Safety Class 2.

h. The design ratings of these systems meet or exceed those specified for the primary containment.

i. The leaktightness of these systems is assured by normal surveillance, inservice testing and leak detection monitoring.

j. The single valve on these lines is located outside containment.

29. 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 water leg pumps; 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.

These valves are under continuous leakage test because they are always subjected to a differential pressure acting across the seat. Leakage through these valves is continuously monitored by the pressure switches in the pump discharge lines, which have a low alarm setpoint in the main control room.

Even though a special leakage test is not merited on these valves for the reasons discussed above, a system leakage test will be performed and compared to an acceptance limit based on site boundary dose considerations.

30. The leakages through the Main Steamline valves will not be included in establishing the acceptance limits for the combined leakage in accordance with the 10 CFR 50, Appendix J, Type B and C tests. The NRC granted exemption to 10 CFR 50, Appendix J, for not including MSIV leakage in the Type A, B, or C acceptance criteria. This exemption is based on the use of the MSIV Isolated Condenser Leakage Treatment Method discussed in Section 6.8, and associated analyses.

31. Although only one isolation valve signal is indicated for these valves, the valves also receive automatic signals from various system operational parameters. For example, the ECCS pump minimum flow valves close automatically when adequate flow is achieved in the system; the ECCS test lines close automatically on receipt of an accident signal. Although these signals are not considered isolation signals; and are therefore, excluded from this table, there are other system operation signals that control these valves to ensure their proper position for safe shutdown. Reference to the logic diagrams for these valves indicates which other signals close these valves.

32. To satisfy the requirements of General Design Criterion 56 and to perform their function, these instrument lines have been designed to meet the requirements of Regulatory Guide 1.11 (Safety Guide 11).

    These lines are Seismic Category I and terminate in instruments that are Seismic Category I. They are provided with manual isolation valves and excess flow check valves.

The integrity of these lines is to be tested during the Type “A” Test. These lines and their associated instruments are to be pressurized to P_a. Surveillance inspections are performed to ensure that the leaktight integrity of these lines and their associated instruments. Additional inservice inspection is included in the Technical Specifications. This inservice inspection verifies the function of the excess flow check valves.

Isolation is provided by the excess flow check valve. In the event of a line rupture downstream of the check valve and a containment pressure above 2 psig this valve would close to limit the amount of leakage.

33. To perform their function and to satisfy the requirements of General Design Criterion 55, these instrument lines have been designed to meet the requirements of Regulatory Guide 1.11 (Safety Guide 11).

These lines are Seismic Category I and terminate in instruments that are Seismic Category I. They-are provided with flow-restricting orifices, manual isolation valves, and excess flow check valves.

The flow-restricting orifice is sized to assure that in the event of a postulated failure of the piping or component, the potential offsite exposure would be substantially below the guidelines of 10 CFR 100.

Isolation is provided by the excess flow check valve. In the event of a line rupture downstream of the check valves, this valve would close to limit the amount of leakage.

The integrity of these lines are tested during the Type "A" Test. Surveillance inspections are performed to ensure the leaktight integrity of these lines and their associated instruments. Additional inservice testing is included in the Technical Specifications. This inservice inspection verifies the function of the excess flow check valves.

For Unit 1 Penetrations M-21 and M-59, and Unit 2 Penetrations M-52 and M-65 reference leg backfill lines have been installed to comply with NRC Bulletin 93-03. These lines tap into the reference legs outboard of the excess flow check valves. Two safety related, Seismic Category I, check valves provide the boundary between the non-safety related CRD system and the safety related reference leg. These two check valves also form part of the boundary that will be checked by surveillance inspections in accordance with Check Valve Monitoring and Preventative Maintenance Program.

For Penetrations I-4A, I-5A, I-7 and I-8A, reference leg backfill lines have been installed to comply with NRC Bulletin 93-03. These lines tap into reference lines 1(2)NB10A-3/4”, 1(2)NB12A-3/4", 1(2)NB23A-3/4" and 1(2)NB25A-3/4" between the containment penetration and the manual isolation valve/excess flow check valve combination. This makes these lines part of the reactor coolant pressure boundary. This location was chosen to prevent the mispositioning of the manual isolation valve (while the injection line is

functioning) from over pressurizing all the instruments on the instrument panel. Two safety related, Seismic Category I, check valves in series act as the outboard containment isolation valves. These two valves also provide the boundary between the non-safety related CRD system and the safety related reference leg as well as form part of the boundary that will be checked by surveillance inspections in accordance with Check Valve Monitoring and Preventative Maintenance Program.

34. These valves are provided for long-term leaktightness only. Feedwater check valves in each line provide immediate isolation. These MO valves are remote manually closed from the control room upon indication of loss of feedwater flow. Therefore, no additional isolation signals are required.

35. Penetrations M-49 and M-50 contain lines for the hydraulic control of the reactor recirculation flow control valves. The hydraulic fluid in these lines is used to position the flow control valves.

Three of four lines of each penetration in this system are under a constant pressure test during normal plant operations due to its high operating pressure of 1800 psig. The fourth line of each penetration in this system is a seal leakage return line back to the HPU Reservoir. Any leakage from this system would be limited to hydraulic fluid which fills these lines and is independent of the containment atmosphere.

In order to perform Type C leakage tests on the isolation valves associated with this system, the system would have to be disabled and the hydraulic fluid drained. This is detrimental to the proper operation of the system in that possible damage could occur in establishing the test condition or restoring the system to normal.

Therefore, these hydraulic isolation valves are exempted from Type C testing.

36. The feedback information available to the plant operator which enables him to determine when the valves with only a "Remote Manual (RM)" closure should be closed is summarized as follows:

a. Leak detection information, as described in Subsection 7.6.2.2 is available to enable the operator to determine the location of a leak or line failure, and close the isolation valve associated with that line.

b. RPV level information is available to the operator to ascertain whether the flow is actually reaching the RPV.

c. Suppression pool water level information would also identify the occurrence of a line failure or leakage.

37. These valves are required to open on signals B and F during the post-LOCA conditions. They remain closed during all other plant operating states, except cold shutdown. Therefore, there is no reason to provide them with any isolation signal other than remote manual.

38. The ADS supply lines are maintained at a minimum pressure of 160 psig at all times. Leakage in these lines is monitored by pressure instrumentation which alarms in the main control room on low pressure. Therefore, these lines are always under a continuous leak test, and a specific local leak rate test (Type C) will not be performed. The intent of the requirement is satisfied however, by the system design itself.

39. 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 ECCS. However, because these ECCS systems are constantly monitored for excessive leakage, this is not a credible event for design.

40. These valves are required to open and remain open following a LOCA to allow the containment air to be sampled. They are part of a system which constitutes a closed loop outside of the containment and will be open during Type A testing. Therefore there is no reason to perform a Type C test on these valves.

41. The inboard flange of these butterfly valves has been provided with a double O-ring type gasket with a leakoff test connection provided between the O-rings. This permits the performance of a Type B leak rate test on this non-welded containment boundary, in addition to the Type C leak test on the valve seats.

42. These valves are capable of being manually overrided by applying jumpers to the isolation logic when a containment isolation signal is present, in order to obtain reactor coolant sample at the High Radiation Sample System Panels under post-accident conditions.

43. These penetrations are provided with removable spools outboard of the outboard isolation valve. During operation these lines will be blind flanged using a double O-ring and Type B leak tested. In addition, the packing of these isolation valves will be soap-bubble tested to ensure insignificant or no leakage at containment test pressure.

44. These lines have been evaluated to an acceptable alternate design basis other than that specifically listed in GDC 57. This alternate basis is found in SRP 6.2.4, Rev. 2, Item II.a.

45. High Radiation Detectors (1&2 RE-CM011 and 1&2 RE-CM017) have been installed in Containment Penetrations M-31 & M-32. These detectors are mounted in steel sleeves which protrude into the Primary Containment at diverse locations, so as to view a larger segment of the containment atmosphere, maintain accessibility for maintenance and calibration, and to minimize exposure during maintenance and calibration. The Containment Penetration is Seal Welded on the inside of the containment and Blind Flanged on the outside of the Containment.

46. These valves are provided with plugged Tees between the solenoid valve and the air cylinder for applying air pressure to the air cylinder using an air bellows hand pump for opening the valve, if instrument air is not available.

47. These valves have different closure time.

1E21-F012	Closure time - less than or equal to 40 seconds
2E21-F012	Closure time - slower than standard (see below)

48. These valves have a slower than standard stem speed, but operate faster than the Tech Spec requirement. The valves' stroke time has been evaluated and is acceptable.

49. In Test Mode 1 the RCIC System is aligned to take suction from the Condensate Storage Tank (CST) and the full flow test return line is aligned to the CST. Valves E51-F362 and E51-F363 will become primary containment isolation valves. In Test Mode 2 the RCIC System is aligned to take suction from the Suppression Pool (SP). Valves E51-F362 and E51-F363 will no longer be containment isolation valves. Valves E51-F022, and E51-F059 will become containment isolation valves and spectacle flange E51-D316 (blind side) will be a containment isolation boundary.

50. General Electric Specification 22A2817AK Rev. 6 states that the maximum operating time for valves 1(2)E12-F064 A/B/C is eight seconds. The intent is to insure that RHR pump minimum flow requirements are met. The downstream orifice becomes the limiting device before the valve fully opens. An evaluation (NTS 373-201-98-CAQ05833.00) concluded as long as the minimum flow valves pass the required minimum flow in 8 seconds or less, the GE specification requirements are met.

51. These valves are subject to bonnet pressure locking. The reactor side valve discs have vent holes drilled in them to prevent pressure accumulation in the bonnet.

52. Exempt Change DCPs 9500254, 255, 256, and 257 change the Valve Closure time for the 1E12-F017B, 17A, 16B, and 16A valves from approximately 75 seconds to approximately 95 seconds. Exempt Change E01-2-94-934A, B, C and D change the Valve Closure time for the 2E12-F016A, B and 2E12-F017A, B valves from approximately 75 seconds to approximately 95 seconds. These are no longer in the standard operating time range for a motor operated gate valve.

53. Exempt Changes E01-1-94-433 and E01-2-94-939-E changed the valve closure times for the 1G33-F040 and 2G33-F040 valves, respectively, from approximately 21 seconds to 39 seconds. This is no longer in the standard operating time range for a motor operated gate valve.

54. The stem packing of these inboard primary containment isolation valves (located outside primary containment) is not tested for leakage during Type C Local Leak Rate Testing. The packing itself is either local leak rate tested via test port or subjected to pressure and subsequently soap bubble tested during primary containment pressurization on a periodic basis in accordance with 10 CFR 50 Appendix J and the LaSalle Station Leak Rate Test Program.

55. The Vacuum Breaker line manual isolation valves have a double-gasketed flange on the inboard or containment side provided with test connections for leak testing. The outboard flanges on the manual isolation valves are leak tested by pressurizing the entire vacuum breaker line and performing a soap bubble test on the outboard flange. The stem seal or packing of these valves will be tested either locally or by primary containment pressurization and subsequent soap bubble inspection.

56. This valve is subject to bonnet pressure locking. The non-containment side valve disc has a vent hole drilled in it to prevent pressure accumulation in the bonnet.

57. The bonnet of this valve, on Units 1 and 2, has a hole drilled in it discharging through piping to downstream of the primary containment isolation valve.

58. These lines have been evaluated to an acceptable alternative design basis other than that specifically listed in GDC 56 and SRP 6.2.4.II. NRC approval of this design is found in the LaSalle Safety Evaluation Report (SER), NUREG 0519 Section 22.2.II.E.4.2.

59. These valves are monitored by the IST/MOV program as implemented by Subsection ISTC of ASME OM Code 2001 Edition through 2003 Addenda, and Code Case OMN-1 "Alternative Rules for Pressure and Inservice Testing of Certain Electric Motor Operated Valve Assemblies in Light Water Reactor Power Plants".

60. Valves 1(2)E51-F064 have been replaced by spectacle flanges 1(2)E51-D324.

61. In response to Generic Letter 96-06, a hole exists in the inboard disc at the inboard containment isolation valve to prevent thermal over-pressurization of the penetration.

62. Penetration M-34 contains the Standby Liquid Control System Injection line.

    The Standby Liquid Control System (SBLC) Line enters the reactor vessel below the core plate. Under post LOCA conditions, the reflooding capability of the jet pumps will always assure the core to be two-thirds covered. This provides assurance that the SBLC line will always be water filled post-LOCA. Thus, the SBLC line is not a potential primary containment atmospheric pathway either during or following a Design Basis Accident (DBA).

    Type C testing is not required on boundaries that do not constitute potential primary containment atmospheric pathways during and following a DBA. Thus, it is not required to Type C test any of the containment isolation valves in that pathway.

    The SBLC line including valves 1&2C41-F007 and 1&2C41-F004A,B will be hydrostatically tested on a periodic basis to insure their leak tight integrity and evaluated against the leakage requirements of Technical Specifications SR 3.6.1.3.11.

63. Exempt from Appendix J testing per EC 404598. Type C testing is not required on boundaries that do not constitute potential primary containment atmospheric pathways during and following a DBA. Thus, it is not required to Type C test any of the containment isolation valves in that pathway.

TABLE 6.2-22
(SHEET 1 OF 2)

PARAMETERS USED TO DETERMINE HYDROGEN CONCENTRATION

1.	Reactor power	3,559 MWt
2.	Number of assemblies	764
3.	Total Zr mass in active clad/assembly	101 lb
4.	Zirconium clad mass	77,187 lb
5.	Fraction of Zr clad reacted	0.945%
6.	Drywell free volume	229,538 ft^3
7.	Suppression chamber volume	165,100 ft^3
8.	Drywell initial temperature	135° F
9.	Drywell initial pressure	0.75 psig
10.	Drywell initial relative humidity	20%
11.	Suppression chamber initial temperature	105° F**
12.	Suppression chamber initial pressure	0.75 psig
13.	Suppression chamber initial relative humidity	100%
14.	Thermal recombiner capacity	125 scfm

TABLE 6.2-22
(SHEET 2 OF 2)

15. The guidelines as set forth in Regulatory Guide 1.7 were followed:

a) 50% of the halogens and 1% of the solids present in the core are intimately mixed with the coolant water.

b) 25% of the halogens plate out on surfaces in the containment.

c) All noble gases and 25% of the halogens are released from the core to the containment atmosphere.

d) All other fission products remain in the fuel rods.

e) $G(H_2)^*$is 0.5 molecules/100eV

f) $G(O_2)$*is 0.25 molecules/100eV

g) The following percentage of fission product radiation energy is absorbed by the coolant:

Percentage	Radiation Type	Location of Source
0%	Beta	Fuel Rods
100%	Beta	Coolant
10%	Gamma	Fuel Rods
100%	Gamma	Coolant

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* For water, borated water, and borated alkaline solutions.

** As discussed in Section 6.2.1.8 supplementary evaluations have been satisfactorily completed with a 105°F initial suppression pool temperature. (Reference 14)

TABLE 6.2-23

CONTAINMENT LEAKAGE TESTING

TYPE OF TEST PER APPENDIX J OF 10 CFR 50	DESCRIPTION OF TEST	CALCULATED PEAK PRESSURE Pa (psig)	LEAK RATES at Pa (%/24 hours) MAXIMUM ALLOWABLE (La)	DESIGN (Ld)	TEST PRESSURE Pt (psig)
A	Integrated Leak Rate	42.6	1.00(3)	0.5	(6)
B	Local Penetration Leakage Rate	42.6	(1)	(1)	(6)
C	Local Containment Isolation Valve Leakage Rate	42.6	(1)(2)	0.1 SCFH per inch of nominal valve size at 50 psig	(6)
-	MSIV Leakage Rate	42.6	(5)	200 scfh	25(4)

(1) The combined leakage rate of all penetrations and valves exclusive of MSIV leakage subject to Type B and C tests shall be less than 0.60 La, as specified in Appendix J to 10 CFR 50.

(2) See Table 6.2-21, Note 15.

(3) Exclusive of the MSIV leakage rates.

(4) Exemption of 10 CFR 50, as stated in III C.3 of Appendix J.

(5) The sum of all four main steam lines shall be less than 400 SCFH. Any MSIV exceeding the proposed limit will be repaired and retested to meet a leakage rate of less than 200 SCFH.

(6) Test pressure shall be, as a minimum, equal to Pa. Variance in test pressure shall be in accordance with ANSI/ANS 56.8-1994.

TABLE 6.2-24
(SHEET 1 OF 2)

SUBCOMPARTMENT VENT PATH DESCRIPTION
RECIRCULATION OUTLET LINE BREAK WITH SHIELDING DOORS

VENT PATH NO.	FROM VOL. NODE NO.	TO VOL. NODE NO.	DESCRIPTION OF VENT PATH FLOW	AREA* (ft^2)	LENGTH (ft)	Σ (L/A) (ft^{-1})	HYDRAULIC DIAMETER (ft)	HEAD LOSS, K: FRICTION LOSS, K_f	HEAD LOSS, K: TURNING LOSS, K_{bl}	HEAD LOSS, K: EXPANSION AND CONTRACTION, K_g	HEAD LOSS, K: TOTAL
1	1	2	unchoked	14.86	5.98	0.40	4.05	-	0.10	0.14	0.24
2	2	3	unchoked	14.86	5.98	0.40	4.05	-	0.10	0.14	0.24
3	3	4	unchoked	14.86	7.48	0.50	4.05	-	0.12	0.28	0.40
4	4	5	unchoked	14.86	8.97	0.60	4.05	-	0.14	0.28	0.42
5	6	7	choked	20.19	6.04	0.30	4.40	-	0.06	0.16	0.22
6	7	8	choked	20.19	6.04	0.30	4.40	-	0.06	0.16	0.22
7	8	9	choked	20.19	7.55	0.38	4.40	-	0.07	0.32	0.39
8	9	10	unchoked	20.19	9.06	0.45	4.40	-	0.09	0.32	0.41
9	35	34	choked	7.04	2.50	0.30	2.42	-	0.85	0.00	0.85
10	34	11	choked	10.02	3.19	0.32	2.95	-	0.03	0.32	0.35
11	11	12	choked	7.47	4.78	0.64	2.70	-	0.56	0.00	0.56
12	12	13	choked	7.09	6.37	0.90	2.70	-	0.52	0.32	0.84
13	13	14	unchoked	7.09	7.96	1.13	2.70	-	0.53	0.32	0.85
14	14	15	unchoked	7.09	9.55	1.35	2.70	-	1.00	0.64	1.64
15	11	17	choked	2.11	4.78	2.26	2.70	-	0.05	0.00	0.05
16	16	17	choked	3.87	6.37	1.46	2.20	-	0.07	0.31	0.38
17	17	18	unchoked	6.79	6.37	0.94	2.70	-	0.52	0.31	0.83
18	18	19	unchoked	6.79	7.96	1.17	2.70	-	0.54	0.31	0.85
19	19	20	unchoked	6.79	9.55	1.41	2.70	-	1.01	0.62	1.63
20	21	22	unchoked	9.83	6.35	0.65	3.00	-	0.06	0.30	0.36
21	22	23	choked	9.83	6.35	0.65	3.00	-	0.06	0.30	0.36
22	23	24	unchoked	9.83	7.93	0.81	3.00	-	0.07	0.60	0.67
23	24	25	unchoked	9.83	9.52	0.97	3.00	-	0.08	0.60	0.68
24	26	27	unchoked	14.68	9.52	0.65	3.25	-	0.98	0.30	1.28
25	27	28	unchoked	14.68	9.52	0.65	3.25	-	0.08	0.60	0.68
26	28	29	unchoked	14.68	9.52	0.65	3.25	-	0.98	0.30	1.28
27	30	31	unchoked	13.49	9.52	0.71	3.20	-	0.97	0.30	1.27
28	31	32	unchoked	13.49	9.52	0.71	3.20	-	0.53	0.60	1.13
29	32	33	unchoked	13.49	9.52	0.71	3.20	-	0.97	0.30	1.27
30	6	1	unchoked	18.40	6.27	0.33	5.80	0.03	0.00	0.00	0.03, 0.03**
31	7	2	unchoked	18.40	6.27	0.33	5.80	0.03	0.00	0.00	0.03, 0.03**
32	8	3	unchoked	18.40	6.27	0.33	5.80	0.03	0.00	0.00	0.03, 0.03**
33	9	4	unchoked	23.36	6.27	0.22	5.80	0.03	0.00	0.00	0.03, 0.03**
34	10	5	unchoked	23.36	6.27	0.22	5.80	0.03	0.00	0.00	0.03, 0.03**
35	34	6	choked	3.61	7.20	1.40	3.70	0.01	0.00	1.12	1.13, 0.90**
36	11	6	choked	3.61	7.20	1.40	3.70	0.01	0.00	1.12	1.13, 0.90**
37	12	7	unchoked	7.22	6.19	0.62	3.70	0.01	0.00	1.12	1.13, 0.90**
38	13	8	unchoked	7.22	6.19	0.62	3.70	0.01	0.27	1.12	1.40, 1.17**
39	14	9	unchoked	10.84	6.19	0.41	3.70	0.01	0.00	1.12	1.13, 0.90**
40	15	10	unchoked	10.84	6.19	0.41	3.70	0.01	0.00	1.12	1.13, 0.90**
41	12	17	unchoked	8.56	4.80	0.56	3.70	0.01	0.45	0.00	0.46
42	13	18	unchoked	8.56	4.80	0.56	3.70	0.01	0.45	0.00	0.46

TABLE 6.2-24
(SHEET 2 OF 2)

VENT PATH NO.	FROM VOL. NODE NO.	TO VOL. NODE NO.	DESCRIPTION OF VENT PATH FLOW	AREA* (ft^2)	LENGTH (ft)	Σ (L/A) (ft^{-1})	HYDRAULIC DIAMETER (ft)	HEAD LOSS, K: FRICTION LOSS, K_f	HEAD LOSS, K: TURNING LOSS, K_{bl}	HEAD LOSS, K: EXPANSION AND CONTRACTION, K_g	HEAD LOSS, K: TOTAL
43	14	19	unchoked	12.84	4.80	0.37	3.70	0.01	0.45	0.00	0.46
44	15	20	unchoked	11.65	4.80	0.41	3.70	0.01	0.43	0.00	0.44
45	34	16	choked	5.94	4.80	0.94	3.70	0.03	0.00	0.00	0.03
46	11	16	unchoked	5.94	4.80	0.94	3.70	0.03	0.85	0.00	0.88
47	16	21	choked	7.72	4.54	0.44	3.70	0.01	0.00	0.66	0.67
48	17	22	choked	7.72	5.55	0.59	3.70	0.02	0.00	0.66	0.68
49	18	23	unchoked	7.72	5.55	0.59	3.70	0.02	0.00	0.66	0.68
50	19	24	unchoked	11.57	5.55	0.40	3.70	0.02	0.00	0.66	0.68
51	20	25	unchoked	11.57	5.50	0.40	3.70	0.02	0.00	0.66	0.68
52	21	26	choked	7.72	8.00	0.80	3.90	0.03	0.27	0.66	0.96
53	22	26	choked	3.86	8.00	1.60	3.90	0.03	0.35	0.66	1.04
54	22	27	choked	3.86	8.00	1.60	3.90	0.03	0.35	0.66	1.04
55	23	27	choked	7.72	8.00	0.80	3.90	0.03	0.00	0.66	0.69
56	24	28	unchoked	11.57	8.00	0.54	3.90	0.03	0.27	0.66	0.96
57	25	29	unchoked	11.57	8.00	0.54	3.90	0.03	0.28	0.66	0.97
58	26	30	choked	11.57	9.20	0.60	3.90	0.03	0.31	0.66	1.00
59	27	31	choked	11.57	9.20	0.60	3.90	0.03	0.35	0.66	1.04
60	28	32	choked	11.57	9.20	0.60	3.90	0.03	0.28	0.66	0.97
61	29	33	choked	11.57	9.20	0.60	3.90	0.03	0.31	0.66	1.00
62	30	36	choked	9.27	-	1.05	-	0.01	0.00	0.74	0.75
63	31	36	choked	13.90	-	0.70	-	0.02	0.00	1.67	1.69
64	32	36	choked	13.90	-	0.70	-	0.02	0.00	1.67	1.69
65	33	36	choked	9.27	-	1.05	-	0.01	0.00	0.74	0.75
66	33	36	choked	2.04	-	1.05	-	-	-	-	1.72
67	32	36	choked	0.68	-	3.39	-	-	-	-	1.71
68	31	36	choked	2.10	-	1.11	-	-	-	-	1.71
69	30	36	choked	1.77	-	1.25	-	-	-	-	1.72
70	36	37	unchoked	400.	-	0.06	-	-	-	-	0.05
71	29	37	choked	1.39	-	1.50	-	-	-	-	1.73
72	28	37	choked	0.71	-	3.30	-	-	-	-	1.71
73	27	37	choked	0.71	-	3.30	-	-	-	-	1.71
74	26	37	choked	1.39	-	1.50	-	-	-	-	1.71
75	37	38	unchoked	965.	-	0.03	-	-	-	-	0.05
76	20	38	choked	1.25	-	1.97	-	-	-	-	1.71
77	19	38	choked	1.07	-	2.20	-	-	-	-	1.71
78	18	38	choked	0.71	-	3.30	-	-	-	-	1.71
79	17	38	choked	0.71	-	3.30	-	-	-	-	1.71
80	15	38	choked	1.25	-	1.97	-	-	-	-	1.71
81	14	38	choked	1.07	-	2.20	-	-	-	-	1.71
82	13	38	choked	1.47	-	1.50	-	-	-	-	1.71
83	12	38	choked	0.71	-	3.30	-	-	-	-	1.71
84	11	38	choked	0.71	-	3.30	-	-	-	-	1.71
85	35	38	choked	1.08	-	2.43	-	-	-	-	1.71
86	0	35	choked	1.00	-	0.00	-	-	-	-	0.00

* Minimum cross-sectional area.

**Loss coefficient for reverse flow.

TABLE 6.2-25
(SHEET 1 OF 2)

SUBCOMPARTMENT VENT PATH DESCRIPTION FEEDWATER LINE BREAK WITH SHIELDING DOORS

								HEAD LOSS, K			
VENT PATH NO.	FROM VOL. NODE NO.	TO VOL. NODE NO.	DESCRIPTION OF VENT PATH FLOW	AREA* (ft^2)	LENGTH (ft)	Σ (L/A) (ft^{-1})	HYDRAULIC DIAMETER (ft)	FRICTION LOSS, K_f	TURNING LOSS, K_{bl}	EXPANSION AND CONTRACTION, K_E	TOTAL
1	1	2	unchoked	14.86	8.97	0.60	4.05	-	0.15	0.14	0.29
2	2	3	unchoked	14.86	8.97	0.60	4.05	-	0.15	0.28	0.43
3	3	4	unchoked	14.86	8.97	0.60	4.05	-	0.15	0.14	0.29
4	5	6	unchoked	20.19	9.06	0.45	4.40	-	0.09	0.16	0.25
5	6	7	unchoked	20.19	9.06	0.45	4.40	-	0.09	0.32	0.41
6	7	8	unchoked	20.19	9.06	0.45	4.40	-	0.09	0.16	0.25
7	9	10	unchoked	13.88	9.55	0.69	3.10	-	1.00	0.31	1.31
8	10	11	unchoked	13.88	9.55	0.69	3.10	-	0.65	0.62	1.27
9	11	12	unchoked	13.88	9.55	0.69	3.10	-	1.00	0.31	1.31
10	13	14	unchoked	9.83	6.35	0.65	3.00	-	0.06	0.45	0.51
11	14	15	unchoked	9.83	6.35	0.65	3.00	-	0.06	0.45	0.51
12	15	16	unchoked	9.83	7.80	0.81	3.00	-	0.08	0.30	0.38
13	16	17	unchoked	9.83	9.52	0.97	3.00	-	0.09	0.30	0.39
14	18	19	unchoked	14.68	6.35	0.44	3.25	-	0.49	0.30	0.79
15	19	20	unchoked	14.68	6.35	0.44	3.25	-	0.53	0.30	0.83
16	20	21	unchoked	14.68	6.35	0.54	3.25	-	0.51	0.00	0.51
17	21	22	unchoked	14.68	6.35	0.65	3.25	-	0.55	0.30	0.85
18	29	23	choked	5.42	2.50	0.40	2.52	-	0.85	0.00	0.85
19	23	24	choked	16.19	3.17	0.20	3.20	-	0.03	0.30	0.33
20	24	25	choked	16.19	4.76	0.30	3.20	-	0.05	0.00	0.05
21	25	26	unchoked	16.19	6.35	0.40	3.20	-	0.73	0.60	1.33
22	26	27	unchoked	16.19	7.93	0.50	3.20	-	0.74	0.60	1.34
23	27	28	unchoked	16.19	9.52	0.60	3.20	-	0.09	0.30	0.39
24	5	1	unchoked	23.80	6.27	0.26	5.80	-	0.00	0.00	0.03
25	6	2	unchoked	23.80	6.27	0.26	5.80	-	0.00	0.00	0.03
26	7	3	unchoked	23.80	6.27	0.26	5.80	0.03	0.00	0.00	0.03
27	8	4	unchoked	23.80	6.27	0.26	5.80	0.03	0.00	0.00	0.03
28	9	5	unchoked	10.84	8.53	0.54	3.70	0.02	0.26	0.85	1.13, 1.28**
29	10	6	unchoked	10.84	8.53	0.54	3.70	0.02	0.26	0.85	1.13, 1.28**
30	11	7	unchoked	10.84	8.53	0.54	3.70	0.02	0.26	0.85	1.13, 1.28**
31	12	8	unchoked	10.84	8.53	0.54	3.70	0.02	0.26	0.85	1.13, 1.28**

TABLE 6.2-25
(SHEET 2 OF 2)

								HEAD LOSS, K			
VENT PATH NO.	FROM VOL. NODE NO.	TO VOL. NODE NO.	DESCRIPTION OF VENT PATH FLOW	AREA* (ft^2)	LENGTH (ft)	Σ (L/A) (ft^{-1})	HYDRAULIC DIAMETER (ft)	FRICTION LOSS, K_f	TURNING LOSS, K_{bl}	EXPANSION AND CONTRACTION, K_g	TOTAL
32	13	9	unchoked	7.22	8.00	0.83	3.70	0.02	0.31	0.63	0.96
33	14	9	unchoked	3.61	8.00	1.66	3.70	0.02	0.31	0.63	0.96
34	14	10	unchoked	3.61	8.00	1.66	3.70	0.02	0.31	0.63	0.96
35	15	10	unchoked	7.22	8.00	0.83	3.70	0.02	0.31	0.63	0.96
36	16	11	unchoked	10.84	8.00	0.56	3.70	0.02	0.31	0.63	0.96
37	17	12	unchoked	10.84	8.00	0.56	3.70	0.02	0.36	0.63	1.01
38	18	13	choked	7.71	8.00	0.80	3.90	0.02	0.00	0.66	0.68
39	19	14	choked	7.71	8.00	0.80	3.90	0.02	0.35	0.66	1.03
40	20	15	unchoked	7.71	8.00	0.80	3.90	0.02	0.28	0.66	0.96
41	21	16	unchoked	11.57	8.00	0.54	3.90	0.02	0.29	0.66	0.97
42	22	17	unchoked	11.57	8.00	0.54	3.90	0.02	0.28	0.66	0.96
43	23	18	choked	3.86	10.08	1.94	3.90	0.04	0.00	0.66	0.70
44	24	18	choked	3.96	10.08	1.94	3.90	0.04	0.00	0.66	0.70
45	25	19	choked	7.71	10.08	0.97	3.90	0.04	0.28	0.66	0.98
46	26	20	choked	7.71	10.08	0.97	3.90	0.04	0.30	0.66	1.00
47	27	21	unchoked	11.57	10.08	0.65	3.90	0.04	0.29	0.66	0.99
48	28	22	unchoked	11.57	10.08	0.65	3.90	0.04	0.27	0.66	0.97
49	23	30	choked	1.54	-	3.60	-	0.01	0.00	1.60	1.61
50	24	30	choked	3.86	-	1.30	-	0.02	0.00	1.05	1.07
51	25	30	choked	7.71	-	1.06	-	0.02	0.00	1.97	1.99
52	26	30	choked	7.71	-	1.06	-	0.02	0.00	1.97	1.99
53	27	30	unchoked	9.27	-	0.79	-	0.01	0.00	2.39	2.40
54	28	30	unchoked	11.57	-	0.65	-	0.02	0.00	1.80	1.82
55	29	30	choked	0.68	-	3.96	-	-	-	-	1.71
56	28	30	choked	0.68	-	3.96	-	-	-	-	1.71
57	27	30	unchoked	1.36	-	1.98	-	-	-	-	1.71
58	26	30	unchoked	1.36	-	1.70	-	-	-	-	1.73
59	25	30	unchoked	0.68	-	3.96	-	-	-	-	1.71
60	30	31	unchoked	400.	-	0.06	-	-	-	-	0.05
61	22	31	choked	0.71	-	3.86	-	-	-	-	1.71
62	21	31	unchoked	1.39	-	1.70	-	-	-	-	1.73
63	20	31	unchoked	0.68	-	2.98	-	-	-	-	1.74
64	19	31	unchoked	1.42	-	1.93	-	-	-	-	1.71
65	31	32	unchoked	965.	-	0.03	-	-	-	-	0.05
66	12	32	choked	2.89	-	0.90	-	-	-	-	1.71
67	11	32	choked	2.50	-	1.17	-	-	-	-	1.71
68	10	32	unchoked	2.50	-	1.17	-	-	-	-	1.71
69	9	32	unchoked	2.14	-	1.29	-	-	-	-	1.71
70	0	32	choked	1.0	-	0.0	-	-	-	-	0.0

* Minimum cross-sectional area.
** Loss coefficient for reverse flow.

TABLE 6.2-26

MASS AND ENERGY RELEASE RATE DATA

RECIRCULATION OUTLET LINE BREAK

(For Biological Shield Pressurization Analysis)

BREAK AREA ≅ 2.753 ft^2

TIME (sec)	LIQUID MASS FLOW RATE (lb_m/sec)	STEAM MASS FLOW RATE (lb_m/sec)	LIQUID ENTHALPY (Btu/lb_m)	STEAM ENTHALPY (Btu/lb_m)	TOTAL MASS RELEASE RATE (lb_m/sec)	TOTAL ENERGY RELEASE RATE (Btu/sec)
0.0	0.	0.	527.4	1195.9	0.	0.
0.0020	742.	0.	527.4	1195.9	742.	3.92×10^5
0.0040	2388.	0.	527.4	1195.9	2388.	1.26×10^6
0.0060	4958.	0.	527.4	1195.9	4958.	2.62×10^6
0.0080	8926.	0.	527.4	1195.9	8926.	4.71×10^6
0.0100	14162.	0.	527.4	1195.9	14162.	7.47×10^6
0.0173	36184.	0.	527.4	1195.9	36184.	1.91×10^6
0.0194	36184.	0.	527.4	1195.9	36184.	1.91×10^7
0.0194	18324.	0.	527.4	1195.9	18324.	9.67×10^6
0.0220	21146.	0.	527.4	1195.9	21146.	1.12×10^7
0.0240	22890.	0.	527.4	1195.9	22890.	1.21×10^7
0.0260	24294.	0.	527.4	1195.9	24294.	1.28×10^7
0.0280	25222.	0.	527.4	1195.9	25222.	1.33×10^7
0.0300	25730.	0.	527.4	1195.9	25730.	1.36×10^7
0.0310	25770.	0.	527.4	1195.9	25770.	1.36×10^7
5.0	25770.	0.	527.4	1195.9	25770.	1.36×10^7

TABLE 6.2-27

MASS AND ENERGY RELEASE RATE DATA

FEEDWATER LINE BREAK

(For biological shield pressurization analysis)

BREAK AREA ≅ 1.538 ft						
TIME (sec)	LIQUID MASS FLOW RATE (lb_m/sec)	STEAM MASS FLOW RATE (lb_m/sec)	LIQUID ENTHALPY (Btu/lb_m)	STEAM ENTHALPY (Btu/lb_m)	TOTAL MASS RELEASE RATE (lb_m/sec)	TOTAL ENERGY RELEASE RATE (Btu/sec)
0.0	14,197.	0.	397.8	1190.	14,197.	5.65×10^6
0.00105	14,197.	0.	397.8	1190.	14,197.	5.65×10^6
0.00106	21,599.	0.	397.8	1190.	21,599.	8.60×10^6
1.0	21,599.	0.	397.8	1190.	21,599.	8.60×10^6

TABLE 6.2-28
(SHEET 1 OF 8)
PRIMARY CONTAINMENT ISOLATION VALVES

VALVE FUNCTION AND NUMBER		VALVE GROUP[a]	MAXIMUM ISOLATION TIME (Seconds)
A.	AUTOMATIC ISOLATION VALVES		
1.	Main Steam Isolation Valves 1(2)B21-F022A, B, C, D 1(2)B21-F028A, B, C, D	1	5*
2.	Main Steam Line Drain Valves 1(2)B21-F016 1(2)B21-F019 1(2)B21-F067A, B, C, D	1	 ≤ 15 ≤ 15 ≤ 23
3.	Reactor Coolant System Sample Line Valves [b] 1(2)B33-F019 1(2)B33-F020	3	≤ 5
4.	Drywell Equipment Drain Valves 1(2)RE024 1(2)RE025 1(2)RE026 1(2)RE029	2	 ≤ 20 ≤ 20 ≤ 15 ≤ 15
5.	Drywell Floor Drain Valves 1(2)RF012 1(2)RF013	2	≤ 20
6.	Reactor Water Cleanup Suction Valves 1(2)G33-F001[c] 1(2)G33-F004	5	≤ 10
7.	RCIC Steam Line Valves 1(2)E51-F008[d] 1(2)E51-F063 1(2)E51-F076	8	 ≤ 20 ≤ 15 ≤ 15
8.	Containment Vent and Purge Valves 1(2)VQ026 1(2)VQ027 1(2)VQ029 1(2)VQ030 1(2)VQ031 1(2)VQ032 1(2)VQ034 1(2)VQ035 1(2)VQ036 1(2)VQ040 1(2)VQ042 1(2)VQ043 1(2)VQ047 1(2)VQ048 1(2)VQ050 1(2)VQ051 1(2)VQ068	4	≤ 10 ≤ 10 ≤ 10 ≤ 10 ≤ 10 ≤ 5 ≤ 10 ≤ 5 ≤ 10 ≤ 10 ≤ 10 ≤ 10 ≤ 5 ≤ 5 ≤ 5 ≤ 5 ≤ 5
9.	RCIC Turbine Exhaust Vacuum Breaker Line Valves 1(2)E51-F080 1(2)E51-F086	9	N/A

TABLE 6.2-28
(SHEET 2 OF 8)
PRIMARY CONTAINMENT ISOLATION VALVES

VALVE FUNCTION AND NUMBER		VALVE GROUP[(a)]	MAXIMUM ISOLATION TIME (Seconds)
A. AUTOMATIC ISOLATION VALVES (CONTINUED)			
10.	Containment Monitoring Valves 1(2)CM017A,B 1(2)CM0l8A,B 1(2)CM019A,B 1(2)CM020A,B 1(2)CM021B [(f)] 1(2)CM022A[(f)] 1(2)CM025A[(f)] 1(2)CM026B[(f)] 1(2)CM027 1(2)CM028 1(2)CM029 1(2)CM030 1(2)CM031 1(2)CM032 1(2)CM033 1(2)CM034	2	≤5
11.	Drywell Pneumatic Valves 1(2)IN001A and B 1(2)IN017 1(2)IN074 1(2)IN075 1(2)IN031	 10 10 10 10 2	 ≤ 30 ≤ 22 ≤ 22 ≤ 22 ≤ 5
12.	RHR Shutdown Cooling Mode Valves 1(2)E12-F008 1(2)E12-F009 1(2)E12-F023 1(2)E12-F053A and B	6	 ≤40 ≤ 40 ≤ 90 ≤ 29
13.	Tip Guide Tube Ball Valves (Five Valves) 1(2)C51-J004	7	N/A
14.	Reactor Building Closed Cooling Water System Valves 1(2)WR029 1(2)WR040 1(2)WR179 1(2)WR180	2	≤ 30
15.	Primary Containment Chilled Water Inlet Valves 1(2)VP113A and B 1(2)VP063A and B	2	 ≤ 90 ≤ 40
16.	Primary Containment Chilled Water Outlet Valves 1(2)VP053A and B 1(2)VP114A and B	2	 ≤ 40 ≤ 90

TABLE 6.2-28
(SHEET 3 OF 8)
PRIMARY CONTAINMENT ISOLATION VALVES

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VALVE FUNCTION AND NUMBER		VALVE GROUP[a]	MAXIMUM ISOLATION TIME (Seconds)
A. AUTOMATIC ISOLATION VALVES (CONTINUED)			
17.	Recirc. Hydraulic Flow Control Line Valves 1(2)B33-F338 A and B 1(2)B33-F339 A and B 1(2)B33-F340 A and B 1(2)B33-F341 A and B 1(2)B33-F342 A and B 1(2)B33-F343 A and B 1(2)B33-F344 A and B 1(2)B33-F345 A and B	2	≤ 5
18.	Feedwater Testable Check Valves 1(2)B21-F032 A and B	2	N/A
B. MANUAL ISOLATION VALVES			
1.	1(2)FC086		N/A
2.	1(2)FC113		N/A
3.	1(2)FC114		N/A
4.	1(2)FC115		N/A
5.	1(2)MC027 [h]		N/A
6.	1(2)MC033 [h]		N/A
7.	1(2)SA042 [h]		N/A
8.	1(2)SA046 [h]		N/A
9.	1(2)CM039		N/A
10.	1(2)CM040		N/A
11.	1(2)CM041		N/A
12.	1(2)CM042		N/A
13.	1(2)CM043		N/A
14.	1(2)CM044		N/A
15.	1(2)CM045		N/A
16.	1(2)CM046		N/A
17.	1(2)CM085		N/A
18.	1(2)CM086		N/A
19.	1(2)CM089		N/A
20.	1(2)CM090		N/A
C. EXCESS FLOW CHECK VALVES			
1.	1(2)B21-F374		
2.	1(2)B21-F376		
3.	1(2)B21-F359		
4.	1(2)B21-F355		
5.	1(2)B21-F361		
6.	1(2)B21-F378		
7.	1(2)B21-F372		
8.	1(2)B21-F370		
9.	1(2)B21-F363		
10.	1(2)B21-F353		
11.	1(2)B21-F415A, B		
12.	1(2)B21-F357		

TABLE 6.2-28
(SHEET 4 OF 8)
PRIMARY CONTAINMENT ISOLATION VALVES

VALVE FUNCTION AND NUMBER	VALVE GROUP[a]	MAXIMUM ISOLATION TIME (Seconds)
C. EXCESS FLOW CHECK VALVES (CONTINUED)		
13. 1(2)B21-F382		
14. 1(2)B21-F328A, B, C, D		
15. 1(2)B21-F327A, B, C, D		
16. 1(2)B21-F413A, B		
17. 1(2)B21-F344		
18. 1(2)B21-F365		
19. 1(2)B21-F443		
20. 1(2)B21-F439		
21. 1(2)B21-F437		
22. 1(2)B21-F441		
23. 1(2)B21-F445A, B		
24. 1(2)B21-F453		
25. 1(2)B21-F447		
26. 1(2)B21-F455A, B		
27. 1(2)B21-F451		
28. 1(2)B21-F449		
29. 1(2)B21-F367		
30. 1(2)B21-F326A, B, C, D		
31. 1(2)B21-F325A, B, C, D		
32. 1(2)B21-F350		
33. 1(2)B21-F346		
34. 1(2)B21-F348		
35. 1(2)B21-F471		
36. 1(2)B21-F473		
37. 1(2)B21-F469		
38. 1(2)B21-F475A, B		
39. 1(2)B21-F465A, B		
40. 1(2)B21-F467		
41. 1(2)B21-F463		
42. 1(2)B21-F380		
43. 1(2)G33-F312A, B		
44. 1(2)G33-F309		
45. 1(2)E12-F315		
46. 1(2)E12-F359A, B		
47. 1(2)E12-F319		
48. 1(2)E12-F317		
49. 1(2)E12-F360A, B		
50. 1(2)E21-F304		
51. 1(2)E22-F304		
52. 1(2)E22-F341		

TABLE 6.2-28
(SHEET 5 OF 8)
PRIMARY CONTAINMENT ISOLATION VALVES

VALVE FUNCTION AND NUMBER		VALVE GROUP[a]	MAXIMUM ISOLATION TIME (Seconds)
C. EXCESS FLOW CHECK VALVES (CONTINUED)			
53.	1(2)E22-F342		
54.	1(2)B33-F319A, B		
55.	1(2)B33-F317A, B		
56.	1(2)B33-F313A, B, C, D		
57.	1(2)B33-F311A, B, C, D		
58.	1(2)B33-F315A, B, C, D		
59.	1(2)B33-F301A, B		
60.	1(2)B33-F307A, B, C, D		
61.	1(2)B33-F305A, B, C, D		
62.	1(2)CM004		
63.	1(2)CM002		
64.	1(2)CM012		
65.	1(2)CM010		
66.	1(2)VQ061		
67.	1(2)B21-F457		
68.	1(2)B21-F459		
69.	1(2)B21-F461		
70.	1(2)CM102		
71.	1(2)B21-F570		
72.	1(2)B21-F571		
D. OTHER ISOLATION VALVES			
1.	Deleted		
2.	Reactor Feedwater and RWCU System Return 1(2)B21-F010A, B 1(2)B21-F065A, B 1(2)G33-F040		
3.	<u>Residual Heat Removal/Low Pressure Coolant Injection System</u> 1(2)E12-F042A, B, C 1(2)E12-F016A, B 1(2)E12-F017A, B 2E12-F480A,B 1(2)E12-F004A, B, C 1(2)E12-F027A, B 1(2)E12-F024A, B 1(2)E12-F021 1(2)E12-F302 1(2)E12-F064A, B, C 1(2)E12-F011A, B 1(2)E12-F088A, B, C 1(2)E12-F025A, B, C 1(2)E12-F030 1(2)E12-F005		

TABLE 6.2-28
(SHEET 6 OF 8)
PRIMARY CONTAINMENT ISOLATION VALVES

VALVE FUNCTION AND NUMBER	VALVE GROUP[(a)]	MAXIMUM ISOLATION TIME (Seconds)
D. OTHER ISOLATION VALVES (CONTINUED)		
3. Residual Heat Removal/Low Pressure Coolant Injection System (Continued) 1(2)E12-F073A, B 1(2)E12-F074A, B 1(2)E12-F055A, B 1(2)E12-F036A, B 1(2)E12-F311A, B		
4. Low Pressure Core Spray System 1(2)E21-F005 1(2)E21-F001 1(2)E21-F012 1(2)E21-F011 1(2)E21-F018 1(2)E21-F031		
5. High Pressure Core Spray System 1(2)E22-F004 1(2)E22-F015 1(2)E22-F023 1(2)E22-F012 1(2)E22-F014		
6. Reactor Core Isolation Cooling System 1(2)E51-F013 1(2)E51-F069 1(2)E51-F028 1(2)E51-F068 1(2)E51-F040 1(2)E51-F031 1(2)E51-F019 1(2)E51-F059(i) 1(2)E51-F022(i) 1(2)E51-F362(j) 1(2)E51-F363(j)		
7. Post LOCA Hydrogen Control 1(2)HG001A, B 1(2)HG002A, B 1(2)HG005A, B 1(2)HG006A, B		

TABLE 6.2-28
(SHEET 7 OF 8)
PRIMARY CONTAINMENT ISOLATION VALVES

VALVE FUNCTION AND NUMBER	VALVE GROUP[a]	MAXIMUM ISOLATION TIME (Seconds)
D. OTHER ISOLATION VALVES (CONTINUED)		
8. Standby Liquid Control System 1(2)C41-F004A, B 1(2)C41-F006 1(2)C41-F007		
9. Reactor Recirculation Seal Injection 1(2)B33-F013A, B 1(2)B33-F017A, B		
10. Drywell Pneumatic System 1(2)IN018 1(2)IN100 1(2)IN101		
11. Reference Leg Backfill 1(2)C11-F422B 1(2)C11-F422D 1(2)C11-F422F 1(2)C11-F422G 1(2)C11-F423B 1(2)C11-F423D 1(2)C11-F423F 1(2)C11-F423G		
12. Control Rod Drive Insert Lines 1(2)C11-D001-120 1(2)C11-D001-123		
13. Control Rod Drive Withdrawal Lines 1(2)C11-D001-121 1(2)C11-D001-122		
14. RHR Shutdown Cooling 1(2)E12-F460		
15. Reactor Coolant System Sample Line Valve 1(2)B33-F395		
16. Reactor Building Closed Cooling Water 1(2)WR225/226		
17. Primary Containment Chilled Water Inlet Valve 1(2)VP198A/B		
18. Primary Containment Chilled Water Outlet Valve 1(2)VP197A/B		
19. Containment Monitoring System 1(2)CM023B 1(2)CM024A		
20. Hardened Containment Vent System 2PC009A 2PC010A		

TABLE 6.2-28
(SHEET 8 OF 8)
PRIMARY CONTAINMENT ISOLATION VALVES

* But $\geq$ 3 seconds.
a) See Technical Specification for isolation signal(s) that operates each valve group.
b) May be opened on an intermittent basis under administrative control.
c) Not closed by SLCS actuation.
d) Deleted.
e) Not closed by Trip Functions 4a, c, d, e or f of Technical Specification 3.3.2, Table 3.3.2-1.
f) Opens on an isolation signal.
g) Also closed by drywell pressure-high signal
h) These penetrations are provided with removable spools outboard of the outboard isolation valve. During operation, these lines will be blind flanged using a double O-ring.
i) If valves 1(2)E51-F362 and 1(2)E51-F363 are locked closed and acceptably leak rate tested, then valves 1(2)E51-F059 and 1(2)E51-F022 are not considered to be primary containment isolation valves and are not required to be leak rate tested.
j) Either the 1(2)E51-F362 or the 1(2)E51-F363 valve may be open when the RCIC system is in the standby mode of operation, and both valves may be open during operation of the RCIC system in the full flow test mode, providing that:

    (1) valve 1(2)E51-F022 is acceptably leak rate tested, and

    (2) valve 1(2)E51-F059 is deactivated, locked closed and acceptably leak rate tested, and

    (3) the spectacle flange, installed immediately downstream of the 1(2)E51-F059 valve, is closed and acceptably leak rate tested.

6.3 EMERGENCY CORE COOLING SYSTEMS

6.3.1 Design Bases

The objective of the emergency core cooling systems (ECCS), in conjunction with the containment, is to limit the release of radioactive materials following a loss-of-coolant accident so that resulting radiation exposures are within the guideline values given in published regulations.

Safety design bases for the emergency core cooling systems are given in the following subsections.

6.3.1.1 Summary Description of the Emergency Core Cooling System

The emergency core cooling system (ECCS) consists of a high-pressure core spray (HPCS) system, a low-pressure core spray (LPCS) system, a low-pressure coolant injection (LPCI) system, and an automatic depressurization system (ADS).

The HPCS consists of a single, motor-driven pump and associated piping, valves, controls and instrumentation. The system is designed to pump water over the entire range of operating pressures, and thus can spray water into the reactor vessel even if the reactor pressure remains near normal operating levels. For small breaks which do not result in rapid vessel depressurization, the HPCS maintains the proper reactor water level and depressurizes the vessel.

The HPCS sprays the top surface of the core until sufficient water accumulates in the vessel to reflood the core. Water is injected into the vessel through nozzles in a circular sparger above and around the periphery of the core.

The LPCS is a loop similar to, but independent of, the HPCS. The low pressure system is designed to provide protection in case of larger breaks which would rapidly depressurize the reactor vessel. Like the HPCS, water from the LPCS enters the vessel through nozzles in a circular sparger located above and around the core periphery. The LPCS limits the maximum cladding temperature and cools it to saturation upon flooding the core. This system acts to protect the core for intermediate and large breaks, and is assisted by the HPCS and ADS for small breaks.

The LPCI is capable of delivering a large flood of water into the core to refill the vessel once it depressurizes. It consists of three residual heat removal subsystem pumps, each of which injects water into the vessel through its own separate piping and penetrations. The function of this system is to cool the core by flooding, thereby cooling the cladding to saturation after a LOCA. The LPCI acts to protect the core for intermediate or large breaks, and is assisted by the HPCS and ADS for small breaks.

Because the spraying and flooding systems can draw water from the suppression pool, they have a continuous supply of water. Water and steam from the vessel which would be lost through a postulated pipe break are collected in the suppression pool. Likewise, water pumped by the ECCS and lost through a break would also accumulate in the suppression pool.

The ADS utilizes 7 of the 13 safety/relief valves. These are activated as a backup to the HPCS to reduce vessel pressure in case of breaks for which depressurization is required, so that flow from the LPCI and LPCS can enter the vessel in time to cool the core and limit cladding temperature.

6.3.1.1.1 Range of Coolant Ruptures and Leaks

The emergency core cooling systems provide adequate core cooling in the event of any size break or leak in the nuclear system process barrier up to and including the limiting design basis break, which is the double ended break of the recirculation suction line.

6.3.1.1.2 Fission Product Decay Heat

In the event of a loss-of-coolant accident, the emergency core cooling systems remove both residual stored heat and radioactive decay heat from the reactor core at a rate that limits the maximum fuel cladding temperature to a value less than the 10 CFR 50 limit of acceptability of 2200° F. The amount of heat to be removed is discussed in Section 6.2.

6.3.1.1.3 Reactivity Required for Cold Shutdown

The reactor is designed to be in the cold shutdown condition with the control rod of highest reactivity worth fully withdrawn and all other control rods fully inserted. Refer to Subsection 4.3.2 for a complete discussion.

6.3.1.1.4 Steam Flow Induced Process Measurement Error

An additional steam flow induced process measurement error in the Level 3 scram was accounted for impact on the ECCS-LOCA analysis. Reference 9 provides an evaluation of this process error. For breaks outside the containment, with the exception of a feedwater line break in the turbine building, there are other sensors independent of water level that will detect the break and generate a scram signal before the L3 scram is reached. Thus, for these breaks there would be no impact on the LOCA response due to a change in the L3 analytical limit. Feedwater line break in the turbine building is insensitive to the delay in L3 scram, so the impact is inconsequential. There is no impact due to a change in the L3 analytical limit for large breaks inside the containment as ECCS-LOCA analysis initializes the reactor at normal water level and scram is assumed to occur on a high drywell pressure

signal at the start of the LOCA event.

For small breaks inside the containment, the ECCS-LOCA analysis Appendix K assumptions initializes with the reactor at the scram water level and scram is assumed to occur on a low water level signal at the start of the LOCA event. Therefore the small break Appendix K cases are potentially affected by a change in the L3 analytic limit. A reduction in the L3 analytical limit has both a negative and a positive impact on the ECCS-LOCA analysis. A lower reactor water level at the time of scram means there is less vessel inventory, which can result in a longer period of core uncovery and a higher PCT. On the other hand, a lower reactor water level at the time of scram will result in earlier actuation of the automatic depressurization system (ADS) and earlier low pressure ECCS injection which can result in a shorter period of core uncovery and a lower PCT. These competing effects insure that a small change in the L3 analytical limit will have a minor impact on the calculated PCT.

The evaluation was done by repeating each of the Appendix K small break case calculations with a lower L3 analytical limit (by 7 inches). Results of the evaluation show that the change in PCT is +10ºF. Accordingly, the impact of this change in PCT is negligibly small on the oxidation calculation, whereas, the other two criteria mandated by 10CFR50.46; coolable geometry and long-term cooling provisions, are unaffected. For LaSalle, all 10CFR50.46 criteria for ECCS performance are satisfied.

It is important to note that the impact of a reduction in the L3 analytical limit on the Appendix K small break PCT calculation is primarily a calculation issue and does not affect the safety margin. From a safety point of view it can be readily shown that if the inventory in the reactor, between normal water level and the current scram water level analytical set-point, is discharged into the drywell due to a LOCA, a high drywell pressure signal will occur before the water level inside the reactor reaches the current L3 analytical limit. Therefore for the case where credit is taken for drywell pressure scram, the actual water inventory and PCT will not be affected by changes in the L3 analytical limit.

The GEH LOCA analysis for GNF2 fuel has incorporated the Steam Flow Induced Process Measurement Error correction in Reference 10.

6.3.1.2 Functional Requirement Design Bases

a. Emergency core cooling systems are provided with sufficient capacity, diversity, reliability, and redundancy to cool the reactor core under all design-basis accident conditions.

b. Emergency core cooling systems are initiated automatically by conditions that sense the potential inadequacy of the normal core cooling.

c. The emergency core cooling systems are capable of startup and operation regardless of the availability of offsite power supplies and the normal generating system of the plant.

d. Action taken to effect containment integrity does not negate the ability to achieve core cooling. All ECCS pumps are designed to operate without benefit of containment back pressure.

e. The components of the emergency core cooling systems within the reactor vessel are designed to withstand the transient mechanical loadings during a loss-of-coolant accident so that the required core cooling flow is not restricted.

f. The equipment of the emergency core cooling systems can withstand the physical effects of a loss-of-coolant accident so that the core can be effectively cooled. Such effects considered are missiles, fluid jets, pipe whip, high temperature, pressure, humidity, and seismic acceleration.

g. To provide a reliable supply of water for the emergency core cooling systems, the prime source of liquid for cooling the reactor core after a loss-of-coolant accident is a stored source located within the containment. The source is located so that a closed cooling water path is established during emergency core cooling systems operation.

6.3.1.3 Reliability Requirements Design Bases

The flow rate and sensing networks of each emergency core cooling system are testable during reactor shutdown. All active components are testable during normal operation of the nuclear system.

6.3.2 System Design

The ECCS, containing four separate subsystems, is designed to satisfy the following performance objectives:

a. to prevent fuel cladding fragmentation for any mechanical failure of the nuclear boiler system up to, and including, a break equivalent to the largest nuclear boiler system pipe;

b. to provide this protection by at least two independent, automatically actuated cooling systems;

c. to function with or without external (offsite) power sources; and

d. to permit testing of all ECCS by acceptable methods including, wherever practical, testing during power plant operations.

The aggregate of these emergency core cooling systems is designed to protect the reactor core against fuel cladding damage (fragmentation) across the entire spectrum of line break accidents.

The power for operation of the ECCS is from regular a-c power sources. Upon loss of the regular power, operation is from onsite standby a-c power sources. Standby sources have sufficient diversity and capacity so that all ECCS requirements are satisfied. The HPCS is powered from one a-c supply bus. The LPCS and one LPCI are powered from a second a-c supply bus and the two remaining LPCI are powered from a third and separate a-c supply bus. The HPCS has its own diesel generator as its alternate power supply. The LPCS and one LPCI loops switch to one site backup power supply and the other two LPCI loops switch to a second site backup power supply.

All systems start automatically. The starting signal comes from at least two independent and redundant sensors of drywell pressure and low reactor vessel water level. Refer to Subsection 7.3.1.2 for a complete discussion of the ECCS instrumentation and starting and control logic.

Further discussion of the integrated performance of the ECCS is presented in Subsection 6.3.3.7. The bounds within which system parameters must be maintained and the acceptable inoperable components are discussed in the Technical Specifications.

6.3.2.1 Schematic Piping and Instrumentation Diagrams

Piping and instrumentation diagrams for the subsystems and components which constitute the ECCS are provided and are referenced under the discussion of that subsystem or component.

6.3.2.2 Equipment and Component Descriptions

6.3.2.2.1 High-Pressure Core Spray (HPCS) System

The high-pressure core spray (HPCS) system consists of a single motor-driven pump located outside the primary containment 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 (P&ID) for the HPCS is shown in Drawing Nos. M-95 and M-141. The HPCS system process diagram is shown in Figure 6.3-1.

The principal HPCS equipment is located outside the primary containment. Suction piping is provided from the suppression pool. The suppression pool water source assures a closed cooling water supply for extended operation of the HPCS system. 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 injection piping is provided with an isolation valve on each side of the containment barrier. Remote controls for operating the valves and diesel generator are provided in the plant control room. The controls and instrumentation of the HPCS system are described, illustrated, and evaluated in detail in Chapter 7.0.

The HPCS system is designed to cool the reactor core sufficiently to prevent fuel cladding temperatures from exceeding the 10 CFR 50 limit of 2200° F following any break in the nuclear system piping. 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 loss-of-coolant accident should occur, a low water level signal or a high drywell pressure signal initiates a reactor scram, the HPCS and its support equipment. The HPCS flow automatically stops when a high water level in the reactor vessel is signaled. 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 system vessel pressure versus flow characteristic assumed in LOCA analyses is shown in Figure 6.3-2. Figure 6.3-10 shows the minimum required pump head for HPCS system in order to meet the LOCA analyses assumptions. 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 is below the water level of the suppression pool. This assures a flooded pump suction. Pump NPSH requirements are met even with the containment at atmospheric pressure by providing adequate suction head and suction line size. The HPCS pump characteristics, head, flow, horsepower, and required NPSH are shown in Figure 6.3-3.

If the HPCS line should break outside the containment, a check valve in the line inside the drywell will prevent loss of reactor water outside the containment. The HPCS pump 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.

To assure continuous core cooling, signals to isolate the containment do not operate any HPCS valves which could affect flow to the reactor pressure vessel.

The HPCS equipment and support structures are designed in accordance with Seismic Category I criteria (Chapter 3.0). The system is assumed to be filled with water for seismic analysis.

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 nuclear system pressure relief valves to relieve high-pressure steam to the suppression pool. The design, number, location, description, and evaluation of the pressure relief valves are discussed in detail in Subsection 5.2.2.4.1. The operation of the ADS is discussed in Subsection 7.3.1.2.2. The piping and instrument diagram (P&ID) for the ADS is shown in Drawings M-55 and M-116.

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 a-c 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. Drawing Nos. M-94 and M-140 show the P&ID for the low-pressure core spray system, and Figure 6.3-4 shows the process diagram for the low-pressure core spray system.

When low water level in the reactor vessel or high pressure in the drywell is sensed, with reactor vessel pressure low enough, the low-pressure core spray system automatically sprays water into the top of the fuel assemblies to cool the core. This

action is initiated in conjunction with other ECCS subsystems soon enough, and at a sufficient flow rate to maintain the fuel cladding temperature below 2200° F. (The low-pressure coolant injection system starts from the same signals and operates independently to achieve the same objective by flooding the reactor vessel.)

The low-pressure core spray system protects the core in the event of a large break in the nuclear system and when the HPCS is unable to maintain reactor vessel water level. Such protection extends to a small break in which the ADS or HPCS has operated to lower the reactor vessel pressure to the operating range of the LPCS. The system vessel pressure versus flow characteristic assumed for LOCA analyses is shown in Figure 6.3-5. Figure 6.3-11 shows the minimum required pump head for the LPCS system in order to meet the LOCA analyses assumption.

The LPCS pump receives power from an a-c power bus having standby power source backup supply. The pump motor and associated automatic motor-operated valves for the LPCS and one LPCI loop receive a-c power from the same bus, while another bus provides a-c power for equipment on the other two LPCI loops (Section 8.3).

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.

To assure continuity of core cooling, signals to isolate the containment do not operate any low-pressure core spray system valves which could affect flow to the reactor pressure vessel.

The LPCS injection check valve is the only low-pressure core spray equipment in the containment required during a loss-of-coolant accident that requires consideration for the high temperature and humidity environment in the drywell resulting from the accident. The valve actuates on flow through the pipeline, independent of any external signal. Thus, neither the normal nor accident environment in the drywell affects the operability of the low-pressure core spray equipment for the accident.

The LPCS system piping and support structures are designed in accordance with Seismic Category I criteria (Chapter 3.0). The system is assumed to be filled with water for seismic analysis.

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 that system. This valve is located in the core spray pump suction line as close to the suppression pool penetration as practical. Because the LPCS conveys water from the suppression pool, a closed loop is established for the spray water escaping from the break.

The LPCS pump is located in the reactor building below the water level in the suppression pool to assure positive pump suction. Pump NPSH requirements are met with the containment at atmospheric pressure. A pressure gauge is provided to indicate the suction head. The LPCS pump characteristics are shown in Figure 6.3-6.

6.3.2.2.4 Low-Pressure Coolant Injection (LPCI) Subsystem

The low-pressure coolant injection subsystem is one of the independent operating subsystems of the RHR system. The LPCI subsystem is actuated by low water level in the reactor or high pressure in the drywell. The subsystem, in conjunction with other ECC subsystems, is required to flood the core before fuel cladding temperature reaches 2200° F and then to maintain water level.

LPCI operation provides protection to the core for a large break in the nuclear system in addition to the LPCS and HPCS. Protection provided by LPCI also extends to a small break in which the ADS or HPCS have reduced the reactor vessel pressure to the LPCI operating range. The vessel pressure versus flow characteristic assumed in the LOCA analyses for the LPCI pumps is shown in Figure 6.3-7. Figure 6.3-12 shows the minimum required pump head for the LPCI system in order to meet the LOCA analyses assumptions.

Figure 6.3-8 shows the schematic process diagram (and process data) of the RHR system. The LPCI subsystem uses the three RHR motor-driven centrifugal pumps to convey water from the suppression pool to the reactor vessel through three separate nozzles. The RHR pumps receive power from a-c power buses having standby power source backup supply. Two RHR pump motors and the associated automatic motor-operated valves receive a-c 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, control 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.

The LPCI injection check valves on each LPCI line are the only LPCI components in the drywell required to actuate during a loss-of-coolant accident that require consideration for the high temperature and humidity environment in the drywell resulting from the accident. The valves actuate on flow through the pipeline, independent of any external signal. Thus, neither the normal nor accident environment in the

containment affects the operability of the low-pressure coolant injection equipment for the accident.

Using the suppression pool as the source of water for LPCI establishes a closed loop for recirculation of LPCI water escaping from the break. LPCI pumps and equipment are described in detail in Subsection 5.4.7, which also describes the other functions served by the same pumps if not needed for the LPCI function. The portions of the RHR required for accident protection are designed in accordance with Seismic Category I criteria (Chapter 3.0). The piping and instrument diagram (P&ID) for the LPCI is shown in Drawings M-96 and M-142.

6.3.2.2.5 ECCS Discharge Line Fill System

One design requirement of any core cooling system 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 a-c power source. The lag between the signal pump start and the initiation of flow into the RPV can be minimized by always keeping the core cooling pump discharge lines full. [redacted] The discharge piping is filled and vented using high point vents in conjunction with the ECCS discharge line fill system to remove gas (air) that may be introduced through maintenance and operational activities. During normal operations, the ECCS discharge line fill system maintains the discharge lines in a “water filled” condition.

On January 11, 2008, the NRC issued Generic Letter 2008-01, Managing Gas Accumulation in Emergency Core Cooling, Decay Heat Removal, and Containment Spray Systems (Reference 13). Generic Letter 2008-01 requested licensees to evaluate the licensing basis, design, testing, and corrective action programs for the Emergency Core Cooling, Decay Heat Removal, and Containment Spray Systems to ensure that gas accumulation is maintained less than the amount that challenges operability of these systems, and that appropriate action is taken when conditions adverse to quality occurred at the station:

- evaluations have been performed that identified locations in piping systems that are susceptible to gas accumulation,
- inspections have been performed, where possible, to substantiate the existence of accumulated gas and quantify volume,
- evaluations have been performed to determine if it is possible to eliminate gas from accumulating, and where possible, the associated actions have been implemented (i.e. procedure changes, physical changes),
- evaluations have been performed to demonstrate the acceptability of gas pockets that cannot be eliminated, and
- a program for periodic gas monitoring has been developed

The piping systems addressed in the response to Generic Letter 2008-01 have the potential to develop voids and pockets of entrained gases. Maintaining the pump suction and discharge piping sufficiently full of water is necessary to ensure that the system will perform properly and will inject the flow assumed in the safety analyses into the Reactor Coolant System or containment upon demand. This will also prevent damage from pump cavitation or water hammer, and pumping of unacceptable quantities of non-condensable gas (e.g., air, nitrogen, or hydrogen) into the reactor vessel following an ECCS start signal or during shutdown cooling. There are some piping locations (e.g., Div 1 RHR/LPCI) that cannot be fully vented due to the physical layout and inability to dynamically vent the piping. These locations have been evaluated on a case by case basis.

Some configurations exist (e.g., Div 1 RHR/LPCI) where it is not possible to totally remove all of the entrained air through the high point vents. These are evaluated on a case by case basis and analysis demonstrates the acceptability of the small amount of gas. These analyses address the concerns identified in NRC Generic Letter 2008-01 "Managing Gas Accumulation in Emergency Core Cooling, Decay Heat Removal, and Containment Spray Systems".

By application to the NRC dated July 14, 2014, Exelon requested a change to the Technical Specifications (TSs) for LaSalle Units 1 and 2 to adopt NRC-approved Technical Specifications Task Force (TSTF) Standard Technical Specifications Change Traveler TSTF-523, Revision 2, "Generic Letter 2008-01, Managing Gas Accumulation." (Reference 15), dated February 21, 2013. The proposed changes included:

- Revising the wording of the Surveillance Requirements (SRs) related to gas accumulation for the ECCS from "verify (system name) piping is full of water" to "verify the (system name) locations susceptible to gas accumulation are sufficiently filled with water."
- Adding a new SR with similar wording for the RCIC and RHR shutdown cooling, suppression pool cooling, and suppression pool spray systems.
- Revising the Limiting Condition for Operating (LCO) Bases for the specifications governing the ECCS, RCIC, and RHR systems to reflect the changes to the SRs and adding an acknowledgement that management of gas voids is important to system operability.

On June 19, 2015, the NRC issued Amendment Nos. 214 and 200 to Facility Operating License Nos. NPF-11 and NPF-18 for LaSalle Units 1 and 2 (Reference 16), respectively, approving the proposed changes.

Since the ECCS discharge lines are elevated above the suppression pool, 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 ECCS division.

The power supply to these pumps is classified as essential when the main ECCS pumps are deactivated. Indication is provided in the control room as to whether these pumps are operating, and ESF system status lights indicate low discharge lines pressure. The piping and instrument diagram (P&ID) for the ECCS is shown on the P&IDs for HPCS, LPCS, and LPCI.

6.3.2.2.6 ECCS Pumps NPSH

The ECCS pump specifications are such that the NPSH requirements for HPCS, LPCS and LPCI are met with the containment at atmospheric pressure and the suppression pool at saturation temperature. Calculations were performed to evaluate ECCS NPSH requirements post DBA-LOCA. The calculations used the following conservative inputs:

1. Maximum ECCS pump flow - unthrottled system, reactor pressure at 0 psid, maximizing suction friction losses and NPSH required.

   LPCI pump - 8100 gpm
   LPCS pump - 8100 gpm
   HPCS pump - 7000 gpm

2. Increased clean, commercial steel piping friction losses to account for potential aging effects, thus maximizing suction losses. An absolute roughness of 0.0005 ft was used (vs. 0.00015 ft. for clean pipe), resulting in an increase in calculated head loss of about 22 percent.

3. To account for strainer plugging, the head loss across the debris bed formed on the stacked disk replacement strainers installed at the suction of the ECCS pumps due to accumulation of insulation debris and miscellaneous fibrous and particulate matter debris produced as a result of a LOCA is determined. This head loss is added to the head loss associated with a clean strainer.

4. Containment conditions used in the analysis are containment at atmospheric pressure and the suppression pool at saturation temperature (212F).

5. A minimum suppression pool elevation of 695' 11-1/2" is used. This includes a worst-case post-LOCA drawdown of 43 inches.

6. NPSH Required values for the ECCS pumps are taken from the vendor pump curves. With respect to the pump suction inlet centerline, the NPSH Required is:

   LPCI pump - 14.0 ft. @8100 gpm
   LPCS pump - 2.0 ft. @8100 gpm
   HPCS pump - 5.0 ft. @7000 gpm

The calculations determined that adequate NPSH exists to meet ECCS pump requirements post LOCA for all ECCS pumps. Additionally, adequate margin exists to ensure that flashing does not occur in any of the ECCS pump suction lines post-LOCA.

ECCS PUMP NPSH AND FLASHING MARGINS FOR LIMITING SUPPRESSION POOL CONDITIONS

Pump	Pump Flow Rate (gpm)	Strainer Margin for NPSH (ft.)	Strainer Margin for Flashing (ft.)	Clean Strainer Head Loss[1] (ft.)	Head Loss due to post-LOCA debris (ft.)	NPSH Margin (ft.)
RHR/LPCI	8100	5.4	12.4	0.71	3.6[2]	1.1
LPCS	8100	17.6	12.6	0.71	3.6[2]	8.3
HPCS	7000	14.0	11.6	0.53	2.4	8.7

[1] 0.76 feet @8400 gpm
[2] Maximum value (@8100 gpm, Unit 2)

6.3.2.2.7 <u>Design Pressures and Temperatures</u>

The design pressures and temperatures at various points in the system, during each of the several modes of operation of the ECC subsystems, can be obtained from the miscellaneous information blocks on the following process diagrams: Figure 6.3-1 for the HPCS, Figure 6.3-4 for the LPCS, and Figure 6.3-8 for the LPCI.
The operational characteristics of the ADS valves are presented in Subsection 5.2.2.

6.3.2.2.8 <u>Coolant Quantity</u>

With reference to the Mark II containment at LaSalle County Station Units 1 and 2, the HPCS system normally takes suction from the suppression pool which contains a minimum of 128,800 cubic feet of water. The LPCS and LPCI systems also take suction from the suppression pool for their source of water.

The CSCS equipment cooling water system source (cooling lake) which provides the ultimate heat sink for cooling the suppression pool during the recovery from a DBA has sufficient capacity to accept heat from the suppression pool and prevent it from exceeding 200° F.

6.3.2.2.9 <u>Pump Characteristics</u>

Pump characteristic curves and the pump power requirements for all ECCS pump are shown in Figures 6.3-3, 6.3-6, and 6.3-9. Pump power requirements are given in Chapter 8.0.

6.3.2.2.10 Heat Exchanger Characteristics

There are no heat exchangers in the closed cooling water path associated with the emergency core cooling subsystems. The heat exchangers in the RHR system are discussed in Section 6.2.

6.3.2.2.11 ECCS Flow Diagrams

A schematic diagram and the flow rates and pressures of the various ECCS subsystems can be obtained from the following process diagrams: Figure 6.3-1, High-Pressure Core Spray System; Figure 6.3-4, Low-Pressure Core Spray System; and Figure 6.3-8, Residual Heat Removal System. (The RHR process diagrams show the low-pressure coolant injection system.) These parameters are presented for several modes of operation, including loss-of-coolant accident and test conditions.

6.3.2.2.12 Relief Valves and Vents

The ECC subsystems contain relief valves to protect the components and piping from inadvertent overpressure conditions.

The HPCS system has one relief valve on the discharge side of the pump downstream of the check valve to relieve thermally expanded fluid:

Nominal relief setting: 1500 psig.

HPCS suction side relief valve:

Nominal relief setting: 100 psig

Capacity: > 10 gpm, 10% Accumulation.

The LPCS system pump discharge relief valve:

Nominal relief setting: 550 psig

Capacity: 100 gpm, 10% Accumulation.

LPCS suction side relief valve:

Nominal relief setting: 100 psig

Capacity: > 10 gpm, 10% Accumulation.

The LPCI system pump discharge relief valve (one for each of three pumps):

Nominal relief setting: 500 psig.

6.3.2.2.13 Motor-Operated Valves and Controls (General)

Motor-operated valves are used in the RHR, HPCS, and LPCS emergency core cooling (ECC) systems; they are also used in the RCIC, feedwater, recirculation, reactor water cleanup (RWCU), standby gas treatment, standby liquid control, main steam, and hydrogen recombiner systems. In addition, motor-operated valves are installed on various primary and secondary containment isolation lines, certain sample lines for containment sampling in the post-LOCA condition, and other lines as indicated in Table 6.3-9.

Valve motor operators in these safety systems are provided with thermal overload protection devices. To ensure that the thermal overloads will not prevent the motor-operated valves from performing their safety-related functions under emergency conditions, the thermal overload protection devices are either bypassed under accident conditions or have sufficiently high trip setpoints to prevent inadvertent trips during valve operation per Regulatory Guide 1.106, Rev. 1. Thermal overload bypass circuits are normally installed on the safety-related motor-operated valves that are required to operate during or immediately following an accident such as the primary containment automatic isolation, emergency core cooling, and RCIC system valves. Thermal overload bypass circuits are not installed on the hydrogen recombiner valves since these valves are not required to be operated until several hours after the accident has occurred. In addition, these valves are normally closed and are provided with only a remote manual control system.

For the valves equipped with thermal overload bypass circuits, the thermal overload protection is either (1) normally in the circuit but automatically bypassed whenever any safety-related use of the valve is initiated, or (2) continuously bypassed and temporarily placed in the circuit via a test switch when the motors are undergoing periodic surveillance or maintenance testing.

To prevent the valve motors from being damaged during normal operation or surveillance testing when the thermal overloads are not bypassed, the thermal overloads are set to trip the valve motor operators during locked rotor conditions. A schematic or typical thermal overload bypass arrangement is shown in Figure 6.3-47 and a list of motor-operated valves which have their thermal overload protection bypassed during an accident condition is given in Table 6.3-9.

For the hydrogen recombiner motor-operated valves, the thermal overloads are always in the circuit. However, setting calculations based on IEEE-741-1990 demonstrate that the thermal overloads for these valves will not inadvertently trip

during required valve operation. The trip setpoints of these thermal overloads have been verified to account for the uncertainties due to the ambient temperature at the location of the overload device following an accident and the inaccuracies in the device trip characteristics.

Further information on motor-operated valves and controls is provided in Subsection 6.2.4.

6.3.2.2.14 Process Instrumentation

Multiple instrumentation is available to the operator in the control room to assist him in assessing the post-LOCA conditions.

Basically, these indications are of two varieties: those which indicate the pressures, temperatures and level in the reactor vessel and in the containment; and those that provide indication of operation of the ECCS, position of valves and circuit breakers and flows of ECCS systems.

The most significant instruments in the first category would be:

a. reactor vessel level,

b. reactor vessel pressure,

c. containment pressure,

d. containment temperature,

e. suppression pool level, and

f. suppression pool temperature,

and in the category of ECCS:

a. LPCI flow,

b. LPCS flow, and

c. HPCS flow,

Other available instrumentation is listed in the P&ID included with the description of the above system in Chapters 5.0 and 6.0. Discussion of instrumentation also appears in some detail in Chapter 7.0.

6.3.2.2.15 Scram Discharge System Pipe Break

In August 1981, the U. S. Nuclear Regulatory Commission published NUREG-0803, “Generic Safety Evaluation Report regarding integrity of BWR Scram System Piping”. This document addressed the possibility of Scram System pipe breaks outside the primary containment. Specifically, a generic BWR probabilistic risk assessment in that document indicated that the postulated Scram Discharge Volume (SDV) event is not a dominant contributor to the probability of core damage. However, NRC guidance in Chapter 5 of NUREG-0803 required that certain plant specific issues be addressed by BWR owners. These plant specific issues included (1) Piping Integrity, (2) Mitigation Capability, and (3) Environmental Qualification.

LaSalle Station has addressed the plant-specific recommendations of NUREG-0803 in the response to NRC per Reference 6. The plant-specific evaluation established that even with the postulated break in the Scram Discharge System piping, the LaSalle leak detection equipment and the Station Operating Procedures will guide the Reactor Operators to prompt and successful mitigation of the event with equipment that is qualified for safe shutdown, adequate core cooling, and capable of maintaining secondary containment integrity.

6.3.2.2.16 ECCS Spray Flows Needed for Long Term Core Cooling

The licensing acceptance criterion for the long-term cooling requirement is satisfied if the core is reflooded above the top of the active fuel, or if the core is reflooded to the top of the jet pump suction and one core spray system is operational. During construction of LaSalle Units 1 and 2, a full scale mock-up of the LaSalle core spray spargers and fuel assemblies was constructed. Testing performed at this facility verified that a core spray flow rate of 6250 gpm resulted in an adequate core spray distribution to support long-term cooling of the fuel. Though short-term ECCS-LOCA analyses have been performed with less core spray flow a flow rate of 6250 gpm at 0 psid to the core spray sparger is still required to ensure that long-term cooling of the fuel is maintained. (Reference 10)

6.3.2.3 Applicable Codes and Classification

All piping systems and components (pumps, valves, etc.) for the ECCS comply with the applicable codes, addenda, code cases, and errata in effect at the time the equipment is procured. See Tables 3.2-1, 3.2-2, 3.2-3 and 3.2-4 for code requirements pertaining to components and systems. Tables 3.2-1, 3.2-2, and 3.2-3 list code editions in effect at the time of original equipment procurement.

The piping and components of the ECCS subsystems within the containment and out to and including the pressure retaining injection valve are Class I. All other piping and components are Class 2, 3, or non-Code as indicated on the system P&ID. Subsection NA, NB, NC and ND of the Code apply to the ECCS.

The equipment and piping of the ECCS, in order to meet specified seismic capabilities, are designed to the requirements of Seismic Category I. This class includes all structures and equipments essential to the safe shutdown and isolation of the reactor, or the failure or damage of which could result in undue risk to the health and safety of the public.

6.3.2.4 Materials Specifications and Compatibility

Refer to Table 5.2-7, Reactor Coolant Pressure Boundary Materials (Section 5.2) for a presentation of the specifications which generally apply to the selection of materials used in the emergency core cooling system. 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 in or on the emergency core cooling system are reviewed and evaluated with regard to radiolytic and pyrolytic composition and attendant effects on safe operation of the ECCS. For example, guidance on the use of fluoro carbon plastic (Teflon) is provided to address IGSCC and FME concerns associated with use of Teflon. Only inorganic thermal insulation, which does not decompose due to radiation or temperature, is used in these environments. All paints used are suitable for the temperature conditions anticipated for their service. Additional information is presented in Section 6.1.

6.3.2.5 System Reliability

As applied to the ECCS, availability is defined as the probability that the system is operable when required. The ECCS availability is a function of the component system test intervals and the failure rates of the component parts used in the systems. The component parts used in the ECCS have low failure rates, as evidenced by historical field operating experience. The ECCS availability required

to assure adequate plant safety is established as a system design requirement. System availability is evaluated to assure adherence to the availability design requirement, the periodic surveillance test intervals, and allowable repair times for inoperable systems. When applicable, analyses are performed by the methods outlined in Reference 1. The levels of redundancy, diversity, and surveillance requirements combine to yield a high order of system availability.

ECCS analyses to determine peak core temperatures are based on the most limiting single failures, assuming no offsite power is available. The analyses demonstrate that the ECCS function is sufficient to meet the Appendix K criteria. The analyses do not consider various minimum combinations of the remaining systems, following a postulated single failure, which are sufficient to meet the Appendix K criteria.

6.3.2.6 Protection Provisions

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.

The component supports which protect against damage from movement and from seismic events are discussed in Subsection 5.4.14. The methods used to provide assurance that thermal stresses do not cause damage to the ECCS are described in Subsection 3.9.1.

The ECCS are protected against the effects of pipe whip, which might result from piping failures up to and including the LOCA. This protection is provided by separation, pipe whip restraints, or energy absorbing materials if required. One of these three methods will be 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 ECCS pump rooms. In addition, the watertight construction of the ECCS pump rooms, when required, protects against mass flooding.

6.3.2.7 Provisions for Performance Testing

High-Pressure Core Spray System

a. A full flow test line is provided to route water from and to the suppression pool without entering the reactor pressure vessel.

b. Instrumentation is provided to indicate system performance during normal test operations.

c. All motor-operated valves are capable of manual operation either local or remote for test purposes with the exception of valves E22-F010 and E22-F011. Valves E22-F001, E22-F010, and E22-F011 are no longer considered part of the design basis for the HPCS System.

d. System relief valves are removable for bench testing during plant shutdown.

e. Drains are provided to leak test the major system valves.

Low-Pressure Core Spray System

a. A full flow test line is provided to route water from and to the suppression pool without entering the reactor pressure vessel.

b. A provision exists to crosstie to the RHR Shutdown Cooling suction line to utilize reactor quality water when testing the pump discharge into the reactor pressure vessel during normal plant shutdown. Utilization of this crosstie is optional as testing can be performed with suction from the Suppression Pool.

c. Instrumentation is provided to indicate system performance during normal and test operations.

d. All motor-operated valves and check valves are capable of operation for test purposes.

e. Relief valves are removable for bench testing during plant shutdown.

Low-Pressure Coolant Injection System

a. A discharge test line is provided for each of the three pump loops to route suppression pool water back to the suppression pool without entering the reactor pressure vessel.

b. A suction test line supplying reactor grade water, is provided to test loop "C" discharge into the reactor pressure vessel during normal plant shutdown.

c. Instrumentation is provided to indicate system performance during normal and test operations.

d. All motor-operated valves, air-operated valves, and check valves are capable of manual operation for test purposes.

e. Shutdown lines taking suction from the reactor system water are provided for loops "A" and "B" to test pump discharge into the reactor pressure vessel during normal plant shutdown and to provide for shutdown cooling.

f. All relief valves are removable for bench testing during plant shutdown.

6.3.2.8 Manual Actions

The initiation of the ECCS is completely automatic. No operator action is assumed for at least 10 minutes after initiation. As shown by analysis results (Reference 10), something less than 4 minutes is required to reflood the core following the design-basis accident. The length of time required is a function of the size and location of the break and the location of the postulated single failure, if any. A time sequence of events for these operations is given in Table 6.3-3.

The design evaluations are all based on these rather long operator delays, and indicate considerable safety margin is still available.

6.3.3 ECCS Performance Evaluation

The performance of the ECCS is evaluated through application of evaluation models, approved by the NRC, which conform to the requirements of 10 CFR 50 Appendix K and then showing conformance to the acceptance criteria of 10 CFR 50.46 (References 1, 4, 8 and 12 provide a complete description of the methods used to perform the calculations). These are summarized herein. A summary description of the loss-of-coolant accident results are also provided herein. LOCA Analysis for Power Uprate to 3489 MWt was performed in Reference 7. The GEH LOCA analyses for GNF2 fuel at Thermal Power Optimization (TPO) power level of 3546 MWt was performed in Reference 10.

The information provided herein is applicable to the current licensing basis LOCA analyses from References 3 and 10.

The ECCS performance is evaluated for the entire spectrum of break sizes for postulated LOCA's. The accidents, as listed in Chapter 15.0, for which ECCS operation is required are:

a. 15.2.8 feedwater piping break;

b. 15.6.4 spectrum of BWR steam system piping failures outside of containment; and

c. 15.6.5 loss-of-coolant accidents.

Chapter 15.0 provides the radiological consequences of the above listed events.

6.3.3.1 <u>ECCS Bases for Technical Specifications</u>

The maximum average planar linear heat generation rates 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 Subsections 6.3.3.4 and 6.3.3.5, and testing requirements are discussed in Subsection 6.3.4. Limits on minimum suppression pool water level are discussed in Section 6.2.

6.3.3.2 <u>Acceptance Criteria for ECCS Performance</u>

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, and for each criterion applicable parts of Subsection 6.3.3, where conformance is demonstrated, are indicated. A detailed description of the methods used to show compliance is provided in Reference 4.

<u>Criterion 1; Peak Cladding Temperature</u>

"The calculated maximum fuel element cladding temperature shall not exceed 2200°F." Conformance to Criterion 1 is shown in Table 6.3-8. Compliance with Criterion 1 is demonstrated in Reference 10.

<u>Criterion 2: Maximum Cladding Oxidation</u>

"The calculated total local oxidation of the cladding shall nowhere exceed 0.17 times the total cladding thickness before oxidation." Conformance to Criterion 2 is shown in Table 6.3-8. Compliance with Criterion 2 is demonstrated in Reference 10.

<u>Criterion 3: Maximum Hydrogen Generation</u>

"The calculated total amount of hydrogen generated from the chemical reaction of the cladding with water or steam shall not exceed 0.01 times the hypothetical

amount that would be generated if all the metal in the cladding cylinder surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react." Conformance to Criterion 3 is shown in Table 6.3-8. Compliance with Criterion 3 is demonstrated in Reference 10.

Criterion 4: Coolable Geometry

"Calculated changes in core geometry shall be such that the core remains amenable to cooling." As described in Reference 1, Section III, conformance to Criterion 4 is demonstrated by conformance to Criteria 1 and 2. Compliance with Criterion 4 for GE fuels is demonstrated in Reference 10.

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." Conformance to Criterion 5 is demonstrated generically for General Electric BWR's in Reference 4. Briefly summarized, when the core refloods shortly following the postulated LOCA, the fuel rods will return quickly to saturation temperature over their entire length. For large pipe breaks the heat flux in the core will eventually be inadequate to maintain a two-phase water flow over the entire length of the core. The static water level inside the core shroud is approximately that of the jet pump suctions.

When at least one spray system is available long-term, the upper third of the core will remain wetted by the core spray water as in non-jet pump BWRs, and there will be no further perforation or metal-water reaction.

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 Subsection 6.3.2.5 and Tables 6.3-5, 6.3-6. Table 6.3-6 summarizes that all potential single failures can be identified as no more severe than one of the following failures:

a. Low-pressure coolant injection (LPCI), emergency diesel-generator, which powers two LPCI pumps. For example, failure of one LPCI pump or one LPCI injection valve is less severe than the diesel-generator failure which disables two LPCI pumps.

b. Low-pressure core spray (LPCS) emergency diesel-generator, which powers one LPCI pump and one LPCS pump.

c. High-pressure core spray (HPCS).

d. One automatic depressurization system (ADS) valve.

It is, therefore, only necessary to consider each of the above single failures in the emergency core cooling system performance analyses. For large breaks, failure of one of the diesel generators is, in general, the most severe failure. For small breaks, the HPCS is the most severe failure. The systems of the ECCS which remain operational after these failures are shown in Table 6.3-6.

For the LOCA evaluation model which covers the entire spectrum of break sizes (large breaks to small breaks), failure of the HPCS ECCS subsystem in Division 3 due to failure of its associated diesel generator is, in general, the most severe failure. The remaining operable ECCS subsystems, which include one spray subsystem, provide the capability to adequately cool the core, under near-term and long-term conditions, and prevent excessive fuel damage. For all LOCA analyses, only six ADS valves are assumed to function.

A single failure in the ADS (one ADS valve) has no effect in large breaks.

6.3.3.4 System Performance During the Accident

In general, the system response to an accident can be described as follows:

a. receiving an initiation signal;

b. a small lag time (to open all valves and have the pumps up to rated speed); and

c. finally, the ECCS flow entering the vessel.

Key ECCS actuation setpoints and time delays for all the emergency core cooling systems are provided in Table 6.3-2 for the GE LOCA analysis.

The flow delivery rates analyzed in Subsection 6.3.4 can be determined from the head-flow curves and the pressure versus time plots discussed in Subsection 6.3.3.7. Simplified piping and instrumentation and functional control diagrams for the

ECCS are provided in Subsection 6.3.2. A representative operational sequence of ECCS for the DBA is shown in Table 6.3-3 for the GE LOCA analysis.

Operator action is not required for ECCS operation, except as a monitoring function, during the short-term cooling period following the LOCA. During the short-term cooling period, the operator will take action as specified in Subsection 6.2.2.3 to place the containment cooling system into operation.

6.3.3.5 Use of Dual Function Components for 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 is configured from the RHR pumps and some of the RHR valves and piping. When the reactor water level is low, the LPCI subsystem (line up) has priority through the valve control logic over the other RHR subsystems for containment cooling. Immediately following a LOCA, the RHR system is directed to the LPCI mode. When the RHR shutdown cooling mode is utilized, the transfer to the LPCI mode must be remote manually initiated.

6.3.3.6 Limits on ECC System Parameters

The limits on the ECC system parameters are identified in Subsections 6.3.3.2, 6.3.3.7.3 and 6.3.3.7.4.

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 redundance and the specified test intervals.

6.3.3.7 ECCS Analysis for LOCA

6.3.3.7.1 LOCA Analysis Procedures and Input Variables

The procedures approved for LOCA analysis conformance calculations are described in detail in References 1, 4, 8 and 12. These procedures were used in the calculations enumerated in Subsection 6.3.3. For convenience, the major computer codes are briefly described below. The interfaces between the codes are shown schematically in Figure 1 of Reference 3. The major interfaces are briefly noted below.

Short-Term Thermal-Hydraulic Model (LAMB)

The LAMB code is a model which is used to analyze the short-term thermodynamic and thermal-hydraulic behavior of the coolant in the vessel during a postulated LOCA. In particular, LAMB predicts the core flow, core inlet enthalpy and core pressure during the early stages of the reactor vessel blowdown. For a detailed description of the model and a discussion regarding sources of input to the model, refer to the "LAMB Code Documentation," Section II.A.3 of Reference 1.

Transition Boiling Transition Model (TASC)

The TASC model is used to evaluate the short-term thermal-hydraulic response of the coolant in the core during a postulated loss-of-coolant accident. In particular, the convective heat transfer response in the thermally limiting fuel bundle is analyzed during the transient.

Fuel Rod Thermal-Mechanical Response (PRIME Model)

The PRIME code is used to calculate the thermal-mechanical response of nuclear fuel to time varying power histories. It provides the parameters to initialize the fuel stored energy and fuel rod fission gas inventory at the onset of postulated LOCA. The PRIME model is described in Reference 11 as implemented in Reference 12.

Long-term System Response (SAFER)

This code is used to calculate the long-term system response of the reactor for reactor transients over a complete spectrum of hypothetical break sizes and locations. SAFER is compatible with the PRIME fuel rod model for gap conductance and fission gas release. SAFER tracks, as a function of time, the core water level, system pressure response, ECCS performance, and other primary thermal-hydraulic phenomena occurring in the reactor. The SAFER code employs a heatup model with a simplified radiation heat transfer correlation to calculate PCT and local maximum oxidation. SAFER realistically models all regimes of heat transfer which occur inside the core during the event, and it provides the outputs as a function of time for heat transfer coefficients and PCT.

A listing of significant input variables used by the evaluation model codes is presented in Table 4-1 and Figure 3-1 in Reference 3 (the numerical values of which are subject to revision in later analysis updates).

SAFER/PRIME-LOCA Model Application Methodology

Using the SAFER/PRIME-LOCA models, the LOCA events are analyzed with nominal values of inputs and correlations. A calculation is performed in conformance to Appendix K and checked for consistency with generic statistical upper bound analyses that encompass modeling uncertainties in SAFER/PRIME-LOCA and uncertainties related to plant parameters.

6.3.3.7.2 Accident Description

A detailed description of the Initial LOCA calculation methodology is provided in References 1, 4, 8 and 12. The GEH ECCS-LOCA analysis is summarized in References 3, 7 and 10. For convenience, a short description of the major events during a 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 slow. This pump coastdown governs the core flow response for the next several seconds. When the jet pump suctions uncover, 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 low plenum flashing continues at a reduced rate for the next several seconds.

Heat transfer rates on the fuel cladding during the early stages of the blowdown are governed primarily by the core flow response. Nucleate boiling continues in the high power plane until shortly after jet pump uncovery. Boiling transition follows shortly after the core flow loss that results from jet pump uncovery. Film boiling heat transfer rates then apply, with increasing heat transfer resulting from the core flow increase during the lower plenum flashing period. Heat transfer then slowly decreases until the high power axial plane uncovers. At that time, convective heat transfer is assumed to cease.

Water level inside the shroud remains high during the early stages of the blowdown because of flashing of the water in the core. After a short time, the level inside the shroud has decreased to uncover the core. Several seconds later the ECCS is actuated. As a result the vessel water level begins to increase. Some time later, the lower plenum is filled, and the core is subsequently rapidly recovered.

The cladding temperature at the high power plane increases initially because nucleate boiling is not maintained even though, the heat input decreases and the sink temperature decreases. A rapid, short duration cladding heatup follows the time of boiling transition when film boiling occurs and the cladding temperature approaches that of the fuel. The subsequent heatup is slower, being governed by decay heat and core spray heat transfer. Finally, the heatup is terminated when the core is recovered by the accumulation of ECCS water.

6.3.3.7.3 Break Spectrum Calculations

A complete spectrum of postulated break sizes and location is considered in the evaluation of ECCS performance. The general analytical procedures for conducting break spectrum calculations are discussed in Reference 19 for GE fuel. For ease of reference, a summary of all figures and tables presented in subsection 6.3.3 is shown in Table 6.3-4. All figures and tables for the LaSalle specific GEH ECCS-LOCA analysis are presented in References 3, 7 and 10.

A complete break spectrum for GE fuel was initially evaluated in Reference 3. Subsequent analysis of the limiting cases from Reference 3 have been repeated to account for plant changes since that time and confirm continued compliance to Acceptance Criteria of 10 CFR 50.46. The ECCS-LOCA analysis basis currently includes assessment of effect from:

-relaxation of certain ECCS parameters (i.e. HPCS injection valve stroke time increased from 14 to 28 seconds; LPCI and LPCS injection valve stroke time increased from 20 to 40 seconds), Reference 3,
-uprate of power to 3489 MWt, Reference 7,
-implementation of Thermal Power Optimization (which was concluded not to impact the ECCS-LOCA analysis basis for compliance), Reference 14, subsequent insertion of the GNF2 fuel bundle in the core, at TPO power of 3546 MWt, Reference 51

A summary of the current SAFER/PRIME-LOCA results, limiting across the break spectrum, is shown in tabular form in Table 6.3-8. Conformance to the acceptance criteria (PCT < 2200°F, local clad oxidation < 17% and a core wide metal water reaction < 1%) is demonstrated. Details of calculations for specific breaks are included in subsequent paragraphs. The LOCA analysis for GNF2 fuel was performed in Reference 51.

6.3.3.7.4 Large Recirculation Line Break Calculations

Important results from the GE LOCA analyses of the DBA (double ended guillotine break of the recirculation suction line with a single failure of the HPCS diesel generator) are shown in the referenced reports. The following results characterize the DBA recirculation line large break transient using the GE ECCS-LOCA evaluation model:

a) Water level as a function of time from SAFER.

b) Reactor vessel pressure as a function of time from SAFER.

c) Fuel rod convective heat transfer coefficient as a function of time from SAFER.

d) Peak cladding temperature as a function of time from SAFER.

The maximum local oxidation and peak cladding temperature from the GE LOCA (SAFER/PRIME) analysis of the record for DBA as well as other break sizes, single failures and break locations are shown in Table 6.3-8. Representative figures showing the transient for these analyses for the DBA break are shown in Figures 6.3-81-a through d. Power uprate results are shown in Reference 5, the GNF2 results are shown in Reference 10.

The use or application of an approved evaluation model, per 10 CFR 50.46, bears with it the need to track and report errors or changes which affect any of the LOCA analyses and the current licensing basis PCT.

6.3.3.7.5 Deleted.

6.3.3.7.6 Small Recirculation Line Break Calculations

Important results from the GE LOCA analysis of the small break (0.08 ft^2 recirculation piping suction break with a single failure of the HPCS diesel generator) are shown in Figures 4-a, 4-b, 4-c, 4-d, and 4-e of Reference 10, for GNF2 fuel. These figures are not included in this section because GE considers this information proprietary and will not release them for use in a public domain document. The following results are shown in Reference 10 for the 0.08 ft^2 small break LOCA:

For the GE LOCA analyses:

a) Water level as a function of time from SAFER. (Figure 4-a)

b) Reactor vessel pressure as a function of time from SAFER. (Figure 4-b)

c) Fuel rod convective heat transfer coefficient as a function of time from SAFER. (Figure 4-d)

d) Peak cladding temperature as a function of time from SAFER. (Figure 4-c)

e) ECCS flow rate as a function of time from SAFER. (Figure 4-e)

The limiting large break GNF2 fuel is not the overall limiting break from the break spectrum analysis. The small break is the limiting case for the licensing basis for GNF2 fuel as shown in Reference 10.

6.3.3.7.7 Calculations For Other Break Locations

GE analyzed four non-recirculation break locations to determine the limiting non-recirculation line break and whether or not the results of this break were bound by the limiting recirculation line break. These breaks are the HPCS line break, the feedline break, the main steamline break inside containment, and the steamline break outside of containment. The main steamline break outside containment (see Section 6.3.3.7.8.1) was determined to be the limiting non-recirculation line break in Reference 2. Reference 2 also shows that the HPCS line break, the feedline break, and the main steamline break inside containment result in no cladding heatup beyond the initial cladding temperature. For these reasons no other non-recirculation line breaks needed to be examined in References 3 and 7.

6.3.3.7.8 Steamline Break Outside Containment

Any break outside the primary containment in a line which connects directly to the reactor pressure vessel will initiate ADS action if conditions as described in subsection 7.3.1.2.2.3 are met. Therefore, given the LOCA assumptions of no feedwater or RCIC, and assuming the failure of HPCS if the main steamline isolation valves (MSIV) close and the break becomes isolated or is too small to depressurize the vessel to below the shutoff head of the low-pressure ECC systems, then actuation of the ADS is necessary to reduce the vessel pressure so that the low-pressure ECC systems can terminate the transient. This will occur automatically after the time delay bypass of high drywell pressure.

The outside steamline break is a representative analysis of this class of breaks, since a large amount of vessel inventory is lost through the broken steamline before the MSIV's can isolate the break. All these types of breaks have the same characteristic sequence of events once the MSIV's close culminating in automatic ADS actuation and subsequent vessel reflooding by the low-pressure ECC systems.

A GE outside steamline break analysis was investigated assuming automatic ADS action 12 minutes after RPV level reaches level 1. A complete set of results using the small-break method is provided in Reference 3. The steamline break outside containment analysis for Power Uprate to 3489 MWt was performed in Reference 7. The peak cladding temperature predicted is far below the 2200° F limit. Table 6.3.7 lists a representative sequence of events associated with this break.

6.3.3.8 <u>LOCA Analysis Conclusions</u>

A new LOCA analysis (Reference 10) was performed for GE fuel to support the introduction of GNF2 fuel for LaSalle Units 1 and 2. The licensing basis PCT for the GNF2 fuel is 1540°F (Reference 10).

Beginning with LaSalle Unit 2 Cycle 16 and continuing in subsequent cycles of Unit 2, GNF3 Lead Use Assemblies (LUA) are inserted into non-limiting location for demonstration purposes. The licensing basis PCT for GNF3 LUAs is 1550°F.

There exist changes or errors that affect the anlysis-of-record PCT calculation. These changes or errors are documented via 10 CFR 50.46 reporting. After accounting for all changes or errors, the Licensing Basis PCT remains below the 2,200 F limit. Refer to the docketed annual or thirty-day 10 CFR 50.46(a)(3) report for details.

Having shown compliance by analysis with an Approved Evaluation Model, it is concluded that the ECCS equipment will perform its function in an acceptable manner and meet all of the 10 CFR 50.46 acceptance criteria, given operation at or below the maximum average planar linear heat generation rates for GNF2 fuel given in the COLR.

6.3.4 Tests and Inspections

Each active component of the emergency core cooling systems that is provided to operate in a design-basis accident is designed to be tested during normal operation of the nuclear system.

The HPCS, ADS, LPCI, and LPCS loops are tested periodically to assure that the emergency core cooling systems will operate.

Preoperational tests of the emergency core cooling systems were conducted during the final stages of plant construction prior to initial startup (Chapter 14.0 of the FSAR). These tests assure correct functioning of all controls, instrumentation, pumps, piping, and valves. System reference characteristics, such as pressure differentials and flow rates, are documented following the preoperational tests and are used to establish the limits of acceptability for measurements obtained in subsequent operational tests.

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[redacted]

[redacted]

6.3.5 Instrumentation Requirements

Design details, including redundancy and logic, of the instrumentation of the ECCS are discussed in Subsection 7.3.1.

6.3.5.1 HPCS Actuation Instrumentation

The HPCS is automatically actuated by the following sensed variables: reactor vessel low water level, or drywell high pressure.

In addition, the HPCS can be manually actuated from the control room.

6.3.5.2 ADS Actuation Instrumentation

The ADS is automatically actuated by the following sensed variables: reactor vessel low water level and drywell high pressures. The drywell high pressure signal is not required for auto initiation if the drywell pressure bypass timer (DPBT) times out. Another time delay allows the logic to reset or the operator to bypass automatic blowdown if conditions have corrected themselves or the signals are erroneous. A manual switch may be used to inhibit ADS action if necessary. For further discussion see subsection 7.3.1.2.2.3.

In addition, the ADS can be manually actuated from the control room.

6.3.5.3 LPCS Actuation Instrumentation

The LPCS is automatically actuated by the following sensed variables: reactor vessel low water level, or drywell high pressure.

In addition the LPCS can be manually actuated from the control room.

6.3.5.4 LPCI Actuation Instrumentation

The LPCI is automatically actuated by the following sensed variables: reactor vessel low water level, or drywell high pressure. Reactor vessel low water level or drywell high pressure also stops other modes of RHR system operation so that LPCI is not inhibited.

In addition, the LPCI can be manually actuated from the control room. Subsection 7.3.1.3.2.3 discusses conformance to IEEE-279 and other applicable regulatory requirements for the ECCS instrumentation and controls.

6.3.6 References

1. "Analytical Model for Loss-of-Coolant Analysis in Accordance with 10 CFR 50 Appendix K," NEDO-20566-A, September 1986.

2. "LaSalle County Nuclear Station Unit 1 ECCS Flow Uncertainty Evaluation," NEDC-32835P, June 1998.

3. GE Document, "LaSalle County Station Units 1 and 2 SAFER/GESTR-LOCA Loss-Of-Coolant Accident Analysis," NEDC-32258P, General Electric Company, October 1993.

4. "The GESTR-LOCA and SAFER Models for the Evaluation of the Loss-Of-Coolant Accident, Volume I, GESTR-LOCA - A Model for the Prediction of Fuel and Thermal Performance, Volume II, SAFER - Long Term Inventory Model for BWR Loss-Of-Coolant Analysis, Volume III, SAFER/GESTR Application Methodology, NEDE-23785-1-P-A, February 1985 and Volume III, Supplement 1, Revision 1, "Additional Information for Upper Bound PCT Calculation," March 2002.

5. GE document GE-NE-208-21-1093, "Engineering Evaluation Requirements for the LaSalle County Station Units 1 and 2 SAFER-GESTR Loss of Coolant Accident Analysis with ECCS Relaxations," dated November 1993.

6. Letter, C. E. Sargent (ComEd) to A. Schwencer (NRC) "LaSalle County Station Units 1 and 2 Response to NUREG-0803, NRC Docket Nos. 50-373 and 50-374", January 21, 1982 (SEAG Number 00-000505).

7. "LaSalle County Station Power Uprate Project, Task 407, ECCS Performance," GE-NE-A1300384-39-01, Revision 1, September 1999.

8. Compilation of Improvements to GENE's SAFER ECCS-LOCA Evaluation Models, NEDC-32950P January 2000.

9. "BWR Owners Group Evaluation of Steam Flow Induced Error (SFIE) Impact on the L3 Setpoint Analytic Limit" GEH-NE-0000-0077-4603-R1, October 2008.

10. GE Hitachi Nuclear Energy, "LaSalle County Station GNF2 ECCS-LOCA Evaluation," 0000-0121-8990-R0, January 2012.

11. Global Nuclear Fuel, "The PRIME Model for Analysis of Fuel Rod Thermal-Mechanical Performance," Technical Bases-NEDC-33256P-A, Qualification-NEDC-33257P-A, and Application Methodology-NEDC-33258P-A, September 2010.

12. GE Hitachi Nuclear Energy, "Implementation of PRIME Models and Data in Downstream Methods," NEDO-33173 Supplement 4-A, September 2011.

13. NRC Generic Letter 2008-001, "Managing Gas Accumulation In Emergency Core Cooling, Decay Heat Removal, and Containment Spray Systems," January 11, 2008.

14. NEDC-33485P-A, "Safety Analysis Report for LaSalle County Station Units 1 and 2 Thermal Power Optimization," January 2010.

15. TSTF-523, Revision 2, "Generic Letter 2008-01, Managing Gas Accumulation," February 21, 2013.

16. Letter, B. Purnell (NRC) to B. Hanson (Exelon Generation Company) "Braidwood Station, Units 1 and 2; Byron Station, Unit Nos. 1 And 2; Clinton Power Station, Unit 1 And 2; And Quad Cities Nuclear Power Station, Units 1 and 2 Issuance of Amendments Regarding Adoption Of Technical Specification Task Force (TSTF) Traveler TSTF-523, "Generic Letter 2008-01, Managing Gas Accumulation" (TAC Nos. MF4436, MF4437, MF4438, MF4439, MF4440, MF4441, MF4442, MF4443, MF4444, MF4445, MF4446). "Enclosures 8, 9, and 12, June 19, 2015.

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TABLE 6.3-2
(SHEET 1 of 5)

Significant Input Variables Used In the GE
Loss-Of-Coolant Accident Analyses

A. Plant Parameters

	Units	GNF2 Nominal(1)(3)	GNF2 Analysis Value(1)(3)
[redacted]	[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]	[redacted]

[redacted]

TABLE 6.3-2
(SHEET 2 of 5)

Significant Input Variables Used In the GE
Loss-Of-Coolant Accident Analyses

B. Emergency Core Cooling System Parameters

Low Pressure Coolant Injection System

Initiating Signals	Units	Analysis Value
[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]

[redacted]

TABLE 6.3-2
(SHEET 3 of 5)

Significant Input Variables Used In the GE
Loss-Of-Coolant Accident Analyses

Low Pressure Core Spray System

[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]

[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]

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TABLE 6.3-2
(SHEET 4 of 5)

Significant Input Variables Used In the GE
Loss-Of-Coolant Accident Analyses

High Pressure Core Spray System

[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]

[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]
[redacted]		[redacted]

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TABLE 6.3-2
(SHEET 5 of 5)

Significant Input Variables Used In the GE
Loss-Of-Coolant Accident Analyses

Automatic Depressurization System

[redacted]		[redacted]
[redacted]		[redacted]
[redacted]	[redacted]	[redacted]

[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]

[redacted]

C. Fuel Parameters

[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]

[redacted]

[redacted] [redacted]

TABLE 6.3-3
(SHEET 1 of 2)

OPERATIONAL SEQUENCE OF EMERGENCY CORE COOLING SYSTEMS FOR DESIGN-BASIS ACCIDENT ANALYSIS[1]

(The information in this table is historical; and presented as representative for current analyses.) Note 3

TIME(sec)	EVENTS
0	Design-basis loss-of-coolant accident assumed to start; normal auxiliary power assumed to be lost.
~ 0	Drywell high pressure[2] and reactor low water level reached. All diesel generators signaled to start; scram; HPCS, LPCS, LPCI signaled to start on high drywell pressure.
$t_1 \rightarrow 6$	Reactor low-low water level reached. HPCS receives second signal to start.
$t_2 \rightarrow 7$	Reactor low-low-low water level reached. Main steam isolation valve close. Second signal to start LPCI and LPCS; auto-depressurization sequence begins.
(t_1+13)	HPCS diesel generators ready to load; energize HPCS pump motor, open HPCS injection valve.
(t_2+13)	Division 1 and 2 diesel generators ready to load; start to close containment isolation valves.
(t_1+41)	HPCS injection valve open and pump at design flow, which completes HPCS startup; LPCS and LPCI (RHR "C") pumps at rated speed.
$t_3 \rightarrow 28$	Low pressure permissive for LPCS & LPCI injection valve
$(t_3+40) \rightarrow 68$	LPCI and LPCS pumps at rated flow, LPCS and LPCI injection valves open, which completes the LPCI and LPCS startups.
~150	Core effectively reflooded assuming worst single failure; heatup terminated.
>10 min.	Operator shifts to containment cooling.

TABLE 6.3-3
(SHEET 2 of 2)

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. (See Subsections 6.3.2.5 and 6.3.3.3.) The recirculation suction line break DBA with limiting HPCS EDG failure case, using Appendix K assumptions, is used.

2. Credit is taken in LOCA analyses for ECCS start on high drywell pressure signal.

3. A comparable sequence of events for small break scenarios would be similarly available.

Source of information: Reference 5 analysis results from GE.

TABLE 6.3-5
(SHEET 1 of 6)

ECCS SINGLE VALVE FAILURE ANALYSIS

SYSTEM	VALVE	POSITION FOR NORMAL PLANT OPERATION CLOSED	OPEN	CONSEQUENCES OF VALVE FAILURE ASSUMED TOGETHER WITH DESIGN-BASIS (DBA) LOCA	REMAINING ECCS COOLANT DELIVERY SYSTEMS
High-pressure core spray (HPCS)	Suppression pool suction E22-F015		X	If MO valve fails to remain open during system operation, HPCS will no longer function.	LPCS + 3 LPC1 loops
	Drains and pressure test connections on suction line			If these manual valves are placed in the incorrect open position, part of the flow could be diverted to locations other than the RPV. However, since all connections, except that for E22-F019, have two valves that must be left open before flow is diverted, and the leak detection system would alarm, three failures would be required for this improper position to result and go undetected. In the case of E22-F019, two failures would be required.	LPCS + 3 LPCI loops + partial HPCS
	E22-F019	X			
	E22-F017/E22-F308	X			
	E22-F339/E22-F340	X			
	Minimum flow			If MO valve fails to open, HPCS pump may overheat and fail.	LPCS + 3 LPCI loops
	E22-F012	X		If valve fails to reclose, approximately 10% of system flow returns to suppression pool	90% HPCS + LPCS +3 LPCI loops
	Condensate tank suction to HP Core Spray			Valves are isolated from HPCS System by means of blind flange. Failure will have no effect on HPCS operation.	HPCS + LPCS + 3 LPCI loops
	E22-F001 (MO)	X			
	E22-F302 (Manual)	X			
	E22-F030/E22-F309 (Pressure test connection)	X			
	Test return to suppression pool			If MO valve is open on start of LOCA, auto close signal recloses valve.	HPCS + LPCS + 3 LPCI loops.
	E22-F023	X		If valve fails to remain closed during system operation, approximately 90% of HPCS flow returns to suppression poo1. HPCS will no longer function.	LPCS + 3 LPCI loops
	Abandoned test return to condensate tank			If these valves are placed in the incorrect open position, part of flow could be diverted to other locations than RPV. However, valves are closed and handwheels are removed.	LPCS +3 LPCI loops + partial HPCS
	E22-F010	X			
	E22-F011	X			

TABLE 6.3-5
(SHEET 2 of 6)

SYSTEM	VALVE	POSITION FOR NORMAL PLANT OPERATION CLOSED	OPEN	CONSEQUENCES OF VALVE FAILURE ASSUMED TOGETHER WITH DESIGN-BASIS (DBA) LOCA	REMAINING ECCS COOLANT DELIVERY SYSTEMS
	Injection valve E22-F004	X		If MO valve fails to remain open, HPCS will no longer function.	LPCS + 3 LPCI loops
	Maintenance valve E22-F038		X	This manual valve is located in the discharge line inside the drywell, and if closed, would result in blocking system injection. Since the valve has position (open/closed) indication in the control room, two error/failures would be required for blockage of system flow to result (i.e., valve is placed in wrong position and operator fails to take corrective action, or position indicating lights do not properly function.	LPCS + 3 LPCI loops
	Water leg valves			These manual valves must be in the position shown to ensure that the discharge line remain filled, thus avoiding water hammer on pump start. Improper positioning would result in a pressure switch/alarm indicating the discharge line is not filled. Therefore, two failures (valve in improper position and switch/alarm failure) must occur before the error goes undetected.	LPCS + 3 LPCI loops
	E22-F026	X			
	E22-F034		X		
	E22-F006		X		
	E22-F033		X		
	Drains, vents and pressure test connections on discharge lines			These manual valves are normally closed, connected in series, and located on the pump discharge line. Both valves in each group must be open before water is diverted from the normal discharge path. Also, as in the case of valves F030 and F033 above, improper position would be detected by the Leak Detection System(i.e., 3 failures required for improper position to result and go undetected.	LPCS + 3 LPCI loops + partial HPCS
	E22-F003/E22-F031	X			
	E22-F021/E22-F022	X			
	E22-F348/E22-F347	X			
	E22-F349/E22-F350	X			
Low-pressure core spray (LPCS)	Suppression pool suction E21-F001		X	If valve fails to remain open during system operation, LPCS will no longer function.	HPCS + LPCS + 3LPCI loops. HPCS + 3 LPCI loops

TABLE 6.3-5
(SHEET 3 of 6)

SYSTEM	VALVE	POSITION FOR NORMAL PLANT OPERATION CLOSED	OPEN	CONSEQUENCES OF VALVE FAILURE ASSUMED TOGETHER WITH DESIGN-BASIS (DBA) LOCA	REMAINING ECCS COOLANT DELIVERY SYSTEMS
	Drains, vents and pressure test connections on suction line			If these manual valves are incorrectly placed in the open position, the leak detection system would alarm. In addition, all connections except E21-F008 require that two valves in series be left in an incorrect position before suction flow is affected. Thus, three failures would be required for the improper valve positions to result in flow loss, except in the case of E21-F008 which requires two failures.	HPCS + 3 LPCI loops + partial LPCS
	E21-F008	X			
	E21-F327/E21-F328	X			
	E21-F334/E21-F335	X			
	E21-F329/E21-F330	X			
	E21-F331/E21-F332	X			
	Test return line E21-F012	X		If MO valve is open on start of LOCA, auto close signal recloses valve.	HPCS + LPCS + LPCI loops.
				If valve fails to remain closed during system operation, approximately 90% of LPCS flow returns to suppression pool. LPCS will no longer function.	HPCS + 3 LPCI loops
	Injection valve E21-F005	X		If MO valve fails to remain open, LPCS will no longer function.	HPCS + 3 LPCI loops
	Maintenance Valve E21-F051		X	Since this manual valve has position indication in the control room, the valve would have to be in the wrong position (closed) and the position indication fail in order for injection blockage to occur; a malfunction requires 2 failures.	HPCS + 3 LPCI loops
	Minimum flow			If valves are not open, LPCS pump may overheat and fail. If valve E21-F011 fails to close approximately 10% of system flow returns to suppression pool.	HPCS + 3 LPCI loops For E21-F011 failure to close, HPCS + 90% LPCS + 3 LPCI loops.
	E21-F011 (MO)		X		
	E21-F052 (Manual)		X		
	Drain, vent and pressure test connections on discharge line			Incorrect position could degrade injection flow. Since both manual valves are in the same drain line, both valves would have to be in the wrong position in order for injection flow to degrade; a malfunction requires 2 failures.	HPCS + 3 LPCI loops + partial LPCS
	E21-F325/E21-F326	X			
	E21-F025/E21-F305	X			
	E21-F013/E21-F014	X			
	E21-F321/E21-F322	X			

TABLE 6.3-5
(SHEET 4 of 6)

SYSTEM	VALVE	POSITION FOR NORMAL PLANT OPERATION CLOSED	OPEN	CONSEQUENCES OF VALVE FAILURE ASSUMED TOGETHER WITH DESIGN-BASIS (DBA) LOCA	REMAINING ECCS COOLANT DELIVERY SYSTEMS
	Water leg Valves E21-F004 E21-F032 E21-F034 E21-F035	 X	 X X X	These manual valves must be in the indicated position to ensure discharge line remains filled. Since a low pressure alarm indicates a fill system failure, both sensor and valve position would have to be incorrect in order for the failure to go undetected. Two failures would be required	HPCS + 3 LPCI loops
Low-pressure coolant injection (LPCI)					
LPCI loop A	Suppression pool suction E12-F004A		X	If valve fails to remain open during system operation, LPCI loop will no longer function	HPCS + LPCS + 3 LPCI loops. HPCS + LPCS + 2 LPCI loops.
	Minimum flow E12-F064A (MO) E12-F018A (Manual)		 X X	If valves are not open, LPCI pump may overheat and fail. If valves E12-F064A fails to close approximately 10% of loop, flow returns to suppression pool	HPCS + LPCS + 2 LPCI loops. HPCS + LPCS + 2 For E12-F064A failure to close. LPCI loops + 90% LPCI loop.
	Test return line E12-F024A	 X		If MO valve is open on start of LOCA, auto close signal recloses valve. If valve fails to remain closed during system operation approximately 90% of loop flow returns to suppression pool. LPCI loop will no longer function.	HPCS + LPCS + 3 LPCI loops. HPCS + LPCS + 2 LPCI loops.
	Drain, vent and pressure test connections on the suction line E12-F370A/E12-F369A E12-F397/E12-F398 E12-F356A/E12-F379A E12-F071A/E12-F070	 X X X X		If these manual valves are in the incorrect position, part of the flow could be diverted. However, all connections are provided with two valves in series, and the leak detection system would alarm. Thus, three failures must be postulated for the incorrect condition to go undetected.	HPCS + LPCS + 2 LPCI loops + partial LPCIA

TABLE 6.3-5
(SHEET 5 of 6)

SYSTEM	VALVE	POSITION FOR NORMAL PLANT OPERATION CLOSED	OPEN	CONSEQUENCES OF VALVE FAILURE ASSUMED TOGETHER WITH DESIGN-BASIS (DBA) LOCA	REMAINING ECCS COOLANT DELIVERY SYSTEMS
Low-pressure coolant injection (LPCI) (cont'd) LPCI loop A					
	Heat exchanger bypass E12-F048A		X	No effect. LPCI flow will be through heat exchanger. Heat exchanger pressure drop will not degrade loop flow.	HPCS + LPCS + 3 LPCI loops.
	Injection valve(s) E12-F042A	X		If MO valve fails to remain open, LPCI loop will no longer function.	HPCS + LPCS + 2 LPCI loops.
	Maintenance valve E12-F092A (Manual) E12-F098A (Manual)		 X X	The valve E12-F092A with position indication in the main control room. Therefore, for this valve to be incorrectly positioned (closed), a failure of this indication as well as incorrect valve positioning (two failures) must be assumed. Valve E12-F098A could block LPCI flow if left in the incorrect (closed) position.	HPCS + LPCS + 2 LPCI loops.
	Water leg valves E12-F085A		X	This manual valve must be open to ensure a filled discharge line. Incorrect positioning would be detected and alarmed in the control room by a pressure switch signal on low pressure. Thus, two failures would be required in order for valve to be incorrectly positioned.	HPCS + LPCS + 2 LPCI
	Drains, vents and pressure test connections on discharge line E12-F361A/E12-F362A E12-F363A/E12-F364A E12-F385A/E12-F386A E12-F080A/E12-F081A E12-F060A/E12-F075A E12-F367/E12-F368 E12-F372A/E12-F371A E12-F056A/E12-F057A E12-F321A/E12-F322A E12-F086/E12-F389 E12-F063A/E12-F388A	 X X X X X X X X X X X		All connections are double valved; therefore, two valves in series would have to be in an incorrect position before any flow would be diverted. In addition, the low pressure alarm would be sounded in the control room since the water leg pump would not maintain the line filled, and leak detection alarms would also be triggered by leakage into the areas. Therefore, four failures must be postulated before any adverse effects on the system could go undetected.	HPCS + LPCS + 2 LPCI + Partial LPCI A

TABLE 6.3-5
(SHEET 6 of 6)

SYSTEM	VALVE	POSITION FOR NORMAL PLANT OPERATION CLOSED	OPEN	CONSEQUENCES OF VALVE FAILURE ASSUMED TOGETHER WITH DESIGN-BASIS (DBA) LOCA	REMAINING ECCS COOLANT DELIVERY SYSTEMS
LPCI (Cont'd)	Combustible gas control cooling water supply E12-F312A	X		This MO valve, if left in the incorrect position, could divert flow away from LPCI. However, position indication is provided in the main control room.	HPCS + LPCS + 2 LPCI + partial LPCI A
LPCI Loop A	Head spray E12-F023	X		This MO valve in an incorrect position (open) would be closed by an isolation signal if LPCI were activated. In addition, position indication is provided in the main control room, and the flow diverted would be sprayed into the RPV head.	HPCS + LPCS + 2 LPCI + partial LPCI A
Loops B and C are identical to Loop A except for the following instances: 1) No heat exchanger bypass valve (E12-F048) exists for Loop C; however, it is provided for Loop B. 2) No combustible gas control cooling water cross-tie exists for Loop C. 3) No head spray line exists for either Loop B or C. 4) The following additional connections and valves exist on Loop C and not on Loops A or B					
	Suppression pool cleanup suction lines			These manual valves located in branch lines off the LPCI suction are also provided with a normally blind flanged connection. A spool piece can be added during plant shutdown to clean-up the suppression pool. Therefore, both the valve and blind flange would have to be incorrect before flow could be diverted.	HPCS + LPCS + LPCI A&B
	E12-F303	X			
	E12-F402	X			
	Water leg valves			Pressure switches are provided to alarm at low pressure if the water leg pumps are not maintaining the proper fill in the lines.	HPCS + LPCS + LPCI A
	E12-F082		X		
	E12-F380		X		

TABLE 6.3-6

SINGLE FAILURES CONSIDERED

FOR ECCS ANALYSIS

Assumed Failure (1)	Remaining ECCS (2)
HPCS D/G	LPCS + 3 LPCI + ADS(3)
LPCS D/G	HPCS + 2 LPCI + ADS(3)
LPCI D/G	HPCS + LPCS + LPCI + ADS(3)
ADS	HPCS + LPCS + 3 LPCI + 5 ADS valves

(1) Other postulated failures are not specifically considered because they 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 remaining systems are those listed for the recirculation line break, less the ECCS in which the break is assumed.

(3) The analysis was performed assuming only 6 of the 7 ADS Valves were functional. This was done to support operation with one SRV out-of-service. In the case of a single failure of the ADS, only 5 ADS valves were assumed.

TABLE 6.3-7

SEQUENCE OF EVENTS FOR STEAMLINE BREAK OUTSIDE CONTAINMENT

(The information in this table is historical; and presented as representative for current analyses.)

TIME (sec)	EVENT
0	Guillotine break of one main steamline outside primary containment.
~0.5	High steamline flow signal initiates closure of main steamline isolation valve.
<1.0	Reactor begins scram.
≤5.5	Main steamline isolation valves fully closed.
~60	RCIC and HPCS would initiate on low water level (RCIC considered unavailable, HPCS assumed single failure, and therefore, may not be available).
~6	Safety relief valves open high vessel pressure. The valves open and close to maintain vessel pressure at approximately 1100 psi.
~300	Reactor water level above core begins to drop slowly due to loss of steam through the safety valves. Reactor pressure still at approximately 1100 psi.
~1150	ADS auto initiates after 10 minute drywell pressure bypass timer plus the existing 2 minute initiation delay. Vessel depressurizes rapidly.
~1350	Low-pressure ECC systems initiated. Reactor fuel uncovered partially.
~1400	Core effectively reflooded and cladding temperature heatup terminated. No fuel rod failure.

SUMMARY OF SAFER/PRIME-LOCA ANALYSIS RESULTS (10 CFR 50 Appendix K)

LASALLE 1 & 2 SPECIFIC BREAK SPECTRUM

Fuel Type: GNF2[1] (SAFER/PRIME)

Break Size	Break Location	Single Failure	1st PCT	2nd PCT
DBA	Suction	HPCS/DG	1294	1445
DBA	Suction	LPCS/DG	1294	1411
DBA	Suction	LPCI/DG	1294	1353
80% DBA	Suction	HPCS/DG	1341	1291
60% DBA	Suction	HPCS/DG	1345	1168
0.07	Suction	HPCS/DG	NA [2]	1487
0.08	Suction	HPCS/DG	NA [2]	1508
0.09	Suction	HPCS/DG	NA [2]	1458

Limiting Break	0.08 ft^2 Recirculation Suction Line Break
Limiting ECCS Failure	HPCS/Diesel Generator Failure
Peak Cladding Temperature (Licensing Basis)	< 1540°F[3]
Maximum Local Oxidation	< 1%
Core-Wide Metal-Water Reaction	<0.1%

(1) Source of Information: Reference 10.
(2) There is no early boiling transition for break areas less than 1.0 ft^2. Therefore, N/A is used for the first PCT and the value in the second PCT column is the peak PCT for the entire transient.

TABLE 6.3-9
(Sheet 1 of 4)

MOTOR-OPERATED VALVES THERMAL OVERLOAD PROTECTION

	VALVE NUMBER	BYPASS DEVICE (Continuous, Accident Conditions, or None)	SYSTEM(S) AFFECTED
a.	1VG001	Accident Conditions	SBGTS
	1VG003	Accident Conditions	
	2VG001	Accident Conditions	
	2VG003	Accident Conditions	
b.	1(2)VP113A	Accident Conditions	Primary containment chilled water coolers
	1(2)VP113B	Accident Conditions	
	1(2)VP114A	Accident Conditions	
	1(2)VP114B	Accident Conditions	
	1(2)VP053A	Accident Conditions	
	1(2)VP053B	Accident Conditions	
	1(2)VP063A	Accident Conditions	
	1(2)VP063B	Accident Conditions	
c.	1VQ038*	Accident Conditions	Primary containment vent and purge system
	1(2)VQ032	Accident Conditions	
	1(2)VQ035	Accident Conditions	
	1(2)VQ047	Accident Conditions	
	1(2)VQ048	Accident Conditions	
	1(2)VQ050	Accident Conditions	
	1(2)VQ051	Accident Conditions	
	1(2)VQ068	Accident Conditions	
	1VQ037*	Accident Conditions	
	2VQ037*	Accident Conditions	
	2VQ038*	Accident Conditions	
d.	1(2)WR179	Accident Conditions	RBCCW system
	1(2)WR180	Accident Conditions	
	1(2)WR040	Accident Conditions	
	1(2)WR029	Accident Conditions	
e.	1(2)B21 - F067A	Accident Conditions	Main steam system
	1(2)B21 - F067B	Accident Conditions	
	1(2)B21 - F067C	Accident Conditions	
	1(2)B21 - F067D	Accident Conditions	
	1(2)B21 - F019	Accident Conditions	
	1(2)B21 - F016	Accident Conditions	
	1(2)B21 - F020	Continuous	
	1(2)B21 - F068	Continuous	
	1(2)B21 - F070	Continuous	
	1(2)B21 - F069	Continuous	
	1(2)B21 - F071	Continuous	
	1(2)B21 - F072	Continuous	
	1(2)B21 - F073	Continuous	
	1(2)B21 - F418A	Continuous	
	1(2)B21 - F418B	Continuous	

* These valves have thermal overload bypass for accident conditions from both Unit 1 and Unit 2

TABLE 6.3-9
(Sheet 2 of 4)

	VALVE NUMBER	BYPASS DEVICE (Continuous, Accident Conditions, or None)	SYSTEM(S) AFFECTED
f.	1(2)B21 - F065A	Continuous	Main feedwater system
	1(2)B21 - F065B	Continuous	
g.	1(2)E21 - F001	Continuous	LPCS system
	1(2)E21 - F005	Accident Conditions	
	1(2)E21 - F011	Accident Conditions	
	1(2)E21 - F012	Accident Conditions	
h.	1(2)C41 - F001A	Accident Conditions	SBLCS
	1(2)C41 - F001B	Accident Conditions	
i.	1(2)G33 - F001	Accident Conditions	RWCU
	1(2)G33 - F004	Accident Conditions	
	1(2)G33 - F040	Continuous	
j.	1(2)E12 - F052A	Accident Conditions	RHR system
	1(2)E12 - F064A	Accident Conditions	
	1(2)E12 - F087A	Accident Conditions	
	1(2)E12 - F004A	Continuous	
	1(2)E12 - F047A	Continuous	
	1(2)E12 - F048A	Accident Conditions	
	1(2)E12 - F003A	Continuous	
	1(2)E12 - F026A	Accident Conditions	
	1(2)E12 - F068A	Continuous	
	1(2)E12 - F073A	Continuous	
	1(2)E12 - F074A	Continuous	
	1(2)E12 - F011A	Accident Conditions	
	1(2)E12 - F024A	Accident Conditions	
	1(2)E12 - F016A	Accident Conditions	
	1(2)E12 - F017A	Accident Conditions	
	1(2)E12 - F027A	Accident Conditions	
	1(2)E12 - F004B	Continuous	
	1(2)E12 - F047B	Continuous	
	1(2)E12 - F048B	Accident Conditions	
	1(2)E12 - F003B	Continuous	
	1(2)E12 - F068B	Continuous	
	1(2)E12 - F073B	Continuous	
	1(2)E12 - F074B	Continuous	
	1(2)E12 - F026B	Accident Conditions	
	1(2)E12 - F011B	Accident Conditions	
	1(2)E12 - F024B	Accident Conditions	
	1(2)E12 - F006B	Continuous	
	1(2)E12 - F016B	Accident Conditions	
	1(2)E12 - F017B	Accident Conditions	
	1(2)E12 - F042B	Accident Conditions	
	1(2)E12 - F064B	Accident Conditions	
	1(2)E12 - F093	Continuous	
	1(2)E12 - F021	Accident Conditions	
	1(2)E12 - F004C	Continuous	

TABLE 6.3-9
(Sheet 3 of 4)

	VALVE NUMBER	BYPASS DEVICE (Continuous, Accident Conditions, or None)	SYSTEM(S) AFFECTED
j. (cont'd)	1(2)E12 - F052B	Accident Conditions	RHR system
	1(2)E12 - F087B	Accident Conditions	
	1(2)E12 - F099B	Accident Conditions	
	1(2)E12 - F099A	Accident Conditions	
	1(2)E12 - F008	Accident Conditions	
	1(2)E12 - F009	Accident Conditions	
	1(2)E12 - F040A	Accident Conditions	
	1(2)E12 - F040B	Accident Conditions	
	1(2)E12 - F049A	Accident Conditions	
	1(2)E12 - F049B	Accident Conditions	
	1(2)E12 - F053A	Accident Conditions	
	1(2)E12 - F053B	Accident Conditions	
	1(2)E12 - F006A	Continuous	
	1(2)E12 - F023	Accident Conditions	
	1(2)E12 - F027B	Accident Conditions	
	1(2)E12 - F042A	Accident Conditions	
	1(2)E12 - F042C	Accident Conditions	
	1(2)E12 - F064C	Accident Conditions	
	1(2)E12 - F094	Continuous	
k.	1(2)E51 - F086	Accident Conditions	RCIC system
	1(2)E51 - F022	Accident Conditions	
	1(2)E51 - F068	Continuous	
	1(2)E51 - F069	Continuous	
	1(2)E51 - F080	Accident Conditions	
	1(2)E51 - F046	Accident Conditions	
	1(2)E51 - F059	Accident Conditions	
	1(2)E51 - F063	Accident Conditions	
	1(2)E51 - F019	Accident Conditions	
	1(2)E51 - F031	Continuous	
	1(2)E51 - F045	Accident Conditions	
	1(2)E51 - F008	Accident Conditions	
	1(2)E51 - F010	Accident Conditions	
	1(2)E51 - F013	Accident Conditions	
	1(2)E51 - F076	Accident Conditions	
	1(2)E51 - F360	Accident Conditions	
l.	1(2)E22 - F004	Accident Conditions	HPCS system
	1(2)E22 - F012	Accident Conditions	
	1(2)E22 - F015	Continuous	
	1(2)E22 - F023	Accident Conditions	

TABLE 6.3-9
(Sheet 4 of 4)

	VALVE NUMBER	BYPASS DEVICE (Continuous, Accident Conditions, or None)	SYSTEM(S) AFFECTED
m.	1(2)HG001A	None	Hydrogen recombiner system
	1(2)HG001B	None	
	1(2)HG002A	None	
	1(2)HG002B	None	
	1(2)HG005A	None	
	1(2)HG005B	None	
	1(2)HG006A	None	
	1(2)HG006B	None	
	1(2)HG003	None	
	2(1)HG009	None	
	2(1)HG018	None	
	1(2)HG025	None	
	1(2)HG026	None	
	1(2)HG027	None	
	1(2)E12-F312A	None	
	1(2)E12-F312B	None	

6.4 HABITABILITY SYSTEMS

Habitability systems are designed to ensure habitability inside the control and the auxiliary electric equipment (AEE) rooms for both Units 1 and 2 during all normal and abnormal station operating conditions including the post-LOCA requirements, in compliance with 10 CFR 50.67. The habitability systems cover all the equipment, supplies, and procedures related to the control and auxiliary electric equipment so that control room operators are safe against postulated releases of radioactive materials, noxious gases, smoke, and steam. Adequate sanitary facilities and medical supplies are provided to meet the requirements of operating personnel during and after the accident. Adequate food and water storage in the control room are also provided for operators during the accident. In addition, the environment of the control and auxiliary electric equipment rooms is maintained in order to ensure the integrity of the contained safety-related controls and equipment, during all the station operating conditions.

6.4.1 Design Bases

The design bases of the habitability systems upon which the functional design is established, are summarized as follows:

a. Independent HVAC systems are provided for the control room and the auxiliary electric equipment room which contains the remote shutdown panels and consists of auxiliary electric equipment room Unit 1 and Unit 2.

b. The control and auxiliary electric equipment rooms are occupied continuously on a year-round basis. The occupancy of the operating personnel is assured for a minimum period of 30 days, after a design-basis accident (DBA).

c. The habitability systems are designed to support a minimum of 5 people during normal and abnormal station operating conditions. The control room is supplied by three separate and independent breathing air subsystems which are each comprised of three 300 cubic foot air cylinders with appropriate pressure regulators, low pressure alarms and face masks. Two of these subsystems are for the Unit 1 and Unit 2 control room operators, while the third system supplies a manifold in the control room which can support the senior reactor operator, the control room technical adviser, and a third user as deemed necessary. All three subsystems are designed to provide a minimum of 6 hours of breathing air for each user.

d. Sanitary facilities and medical supplies for minor injuries are provided for the control room. In addition, food and bottled water for a day (at least three meals) are stored in the control room for a minimum of 10 people. This food is for use in accident conditions when access to the control room with food and water would be limited by dose rates.

e. The radiological effects on the control and auxiliary electric equipment rooms that could exist as a consequence of any accident described in Chapter 15.0 are considered in the design of the habitability system.

f. The design includes provisions to preclude the effect of noxious gas and smoke from inside or outside the plant.

g. In addition to the subsystems mentioned in (c) above, carts containing self-contained breathing air systems, e.g., Air-Paks, are located inside the control room. These portable carts are intended for emergency use.

Each Air-Pak has a nominal 1/2 hour air breathing bottle which is rechargeable. These carts contain adequate spares to provide necessary replacement bottles. A self-contained recharging system is provided for refilling expended air bottles on a timely basis to assure an adequate air supply to emergency personnel. Proceduralized methods are available to refill SCBAs if required for long term use.

At least ten total air paks are dedicated for fire brigade use and are located where brigade members can readily obtain them. These air packs are also rechargeable to assure adequate air supply to the fire brigade.

h. The habitability systems are designed to operate effectively during and after a DBA such as a LOCA with the simultaneous loss of offsite power, design-basis earthquake, or failure of any one of the HVAC system components.

i. Radiation monitors, ammonia, and ionization detectors continuously monitor the air supply from the control room and AEE room outside air inlets (see Figure 6.4-2). The detection of high radiation, ammonia, or smoke is alarmed in the control room. Related protection functions are simultaneously initiated for high radiation or smoke. Pressure differential indicators are provided in the control room and AEE room to monitor the pressure differential between control/AEE room and surrounding areas respectively.

Outdoor air and individual room temperature indicators are provided for the control room HVAC system and the AEE room HVAC system.

j. Each control room and AEER HVAC subsystem has a supply air filter unit that contains a charcoal filter unit, called the recirculation filter. Each filter unit consists of a pre-filter and a normally bypassed charcoal filter. Upon detection of smoke in the return ductwork, the

charcoal filter is automatically placed in service. Within 2 minutes of detection of high ammonia concentration in the air intake, the Operator will align the CRE HVAC systems in recirculation mode and will don a self-contained breathing apparatus. Upon detection of high radiation, the Operator must manually place the charcoal filter on-line within 4 hours of detection to maintain the control room dose to within 10 CFR 50.67.

k. The Control Room Envelope (CRE) boundary is maintained to ensure that filtered and unfiltered inleakage into the CRE will not exceed the inleakage assumed in the dose calculation (subsection 9.4.1.1.1.1g and 9.4.1.2.1.1f) in maintaining post accident dose in the CRE within the limits of 10 CFR 50.67.

6.4.2 System Design

6.4.2.1 Definition of Control Room Envelope

The control room envelope consists of control room and auxiliary electric equipment rooms for both Units 1 and 2, control room toilet and the main security control center. Air handling units, filter trains, ducts and dampers are also part of the CRE.

6.4.2.2 Ventilation System Design

The detailed ventilation system design is presented in Subsection 9.4.1.

All the components are designed to perform their function during and after the design basis earthquake except for the electric heating equipment, which is supported to stay in position, but may not function.

All components are protected from internally and externally generated missiles. A layout diagram of the control and AEE rooms, showing doors, corridors, stairways, shield walls and the equipment layout is given in Figure 6.4-1.

The description of controls, instruments, radiation, smoke, and ammonia monitors for the control/AEE room HVAC systems is included in Subsections 7.2. and 7.3.4.3. The locations of outside air intakes and potential sources of radioactive and toxic gas releases are indicated in Figure 6.4-2.

A detailed description of the emergency makeup air filter trains is presented in Subsection 6.5.1.

6.4.2.3 Leaktightness

The control room ductwork was leak tested during start-up and the leakage through the isolation dampers was determined from vendor data. All cable pans and duct

penetrations are sealed. Approximately 1500 cfm of outside air is introduced in the control room to maintain positive pressure with respect to adjacent areas and approximately 2500 cfm of outside air is introduced in the AEE room to maintain positive pressure. During isolation of the control room or AEE room, due to the presence of toxic gases in the intake stream, the outside air dampers are shut.

6.4.2.4 Interaction with Other Zones and Pressure-Containing Equipment

The control room is surrounded by the auxiliary building offices (elevation 768 feet). These offices are served by an independent HVAC system as described in Subsection 9.4.3. There is a ventilation barrier between the control room and auxiliary building office HVAC systems through concrete wall construction and leaktight doors. The control room is isolated from the turbine building through leaktight double doors.

The auxiliary electric equipment room is surrounded by the switchgear room and the cable spreading rooms. These are served by independent HVAC systems as described in subsection 9.4.5.

6.4.2.5 Shielding Design

The shielding for the control and AEE rooms is designed so that the doses experienced by control room personnel during normal operation and during design-basis accidents are as low as reasonably achievable (ALARA). However, the main function of the shielding is to protect occupants from the radiation associated with a LOCA.

During normal operation the control and AEE rooms are shielded from radiation sources in reactor water, steam processing equipment, station vent stack, and in the calibration facility. The sources, shielding, areas affected, and the dose rates are given in Table 6.4-1.

The design-basis accident which requires excessive radiation protection for the control and AEE rooms is the LOCA. The radiation sources due to a LOCA are distributed throughout the containment and the environment surrounding the control and AEE rooms as specified in Chapter 15.0. The shielding design and doses are based on airborne, cloud, and plate out sources given in Table 6.4-2. The location of the sources is shown in Figure 6.4-3.

The shielding reduces the radiation dose rates inside the control room (from outside sources) to levels where the accumulated dose is a small fraction of the limit specified in 10 CFR 50.67.

The shielding arrangement for the control and AEE rooms is presented in Figure 6.4-1, the sources and accident doses are given in Table 6.4-2, and the LOCA

shielding model is shown in Figure 6.4-3. Exposure of control room personnel due to airborne radiation inside the control room is discussed in Chapter 15.0.

The shielding which protects the control and AEE rooms during normal operation is directly associated with the radiation sources, i.e is not part of the control and AEE rooms shielding, which provides additional radiation protection. Table 6.4-1 lists the sources, total shielding thickness, and calculated dose rates during normal operation.

6.4.3 System Operational Procedures

During normal plant operation, the mixture of recirculated air and outside air for the control room HVAC system is filtered by high-efficiency, water and fire resistant glass fiber filters. The control room HVAC system is started through a remote control switch located in the control room. The sequence of operation is given in Chapter 7.0.

To remove any noxious gases, odors, and smoke from the control room environs, a bank of charcoal absorber beds is provided with each control room air handling equipment train. These charcoal beds, located downstream of high-efficiency filters, are normally bypassed. Within 2 minutes of detection of high ammonia concentration in the air intake, the Operator will align the CRE HVAC systems in recirculation mode and will don a self-contained breathing apparatus. The control room HVAC system is manually put in the recirculation mode, by which the outside air intake dampers are closed and the recirculation air from the control room system is routed through the charcoal absorber banks by depressing a pushbutton provided on the main control panel.

On the smoke detection signal in the return duct, the supply air to the control room HVAC system is automatically routed through the charcoal absorber and annunciated on the main control board. The operator may continue to route the system supply air through the charcoal absorber for smoke removal, or depending on the condition of the outside air, may manually bypass the charcoal absorber and purge the system with outside air. Prior to manually placing the HVAC systems in purge, e.g., maximum outside and exhaust air, the operator shall align the supply air through the charcoal absorber.

In the event of high radiation detection from the outside air intake of the control room HVAC system, the radiation monitoring system automatically shuts off normal and maximum outside air supply, and maximum exhaust air to the system. The minimum outside air requirement is routed through the emergency makeup air filter train and fan (for removal of radioactive particulates and iodine), before being supplied to the system.

Two emergency makeup air filter trains and fans are provided, each capable of handling minimum requirements of outside air for the system. In the event of high radiation levels, each train is sized to process 4000 cfm of outside air, providing 1500 cfm to the control room HVAC system and 2500 cfm to the auxiliary electric equipment room HVAC system. Each train contains a supply air filter, which must

be placed on-line within the first four hours of an accident to maintain CR doses within 10 CFR 50.67 values. The emergency makeup air filter units are described in detail in Subsection 6.5.1.

6.4.4 Design Evaluation

The control room HVAC system is designed to maintain a habitable environment and to ensure the operability of all the components in the control room under all the station operating conditions. The system is provided with redundant equipment to meet the single failure criteria. The redundant equipment is supplied with separate essential power sources and is operable during loss of offsite power. The power supply and control and instrumentation meet IEEE-279 and IEEE-308 criteria. All the HVAC equipment except heating are designed for Seismic Category I.

The likelihood of an equipment fire affecting control room habitability is minimized because early ionization detection is assured, fire fighting apparatus is available, and filtration and purging capability are provided.

The following provisions are made to minimize fire and smoke hazards inside the control room and damage to nuclear safety- related circuits:

a. Most electrical wiring and equipment are surrounded by, or mounted in metal enclosures.

b. The nuclear safety-related circuits for redundant divisions (including wiring) are physically segregated by space or fire partitions to allow only isolated damage to electrical equipment.

c. Cables used throughout the control room are flame retardant.

d. Structural and finish materials (including furniture) for the control room and interconnecting areas have been selected on the basis of fire resistant characteristics. Structural floors and interior walls are of reinforced concrete. Interior partitions incorporate metal, masonry, or gypsum dry walls on metal joists. The control room ceiling, door frames, and doors are metallic. Wood trim is not used.

The air distribution in the control room is designed to supply air into the occupied area and exhaust through the control panels. In the event of smoke or products of combustion in the control panels, the ionization detection system alerts the operator and automatically positions dampers to pass all the supply air delivered to the conditioned spaces through a normally bypassed absorber for smoke and odor absorption. A manual override is also provided for this function as well as the ability to introduce 100% outside air to purge the control room as may be necessary.

Two redundant ammonia detectors are provided at each outside air intake duct to the control room HVAC system. Upon detection of ammonia in the outside air, a control room annunciator alarms. Within 2 minutes, the Operator will align the control room HVAC system in recirculation mode and don a self-contained breathing apparatus. The control room HVAC system will operate in 100% recirculation mode, thus routing the recirculating air through its charcoal absorbers.

Four radiation monitor channels (A, B, C, and D) are provided to detect high radiation at each outside air intake to the control room HVAC system. These monitor channels alarm the control room upon detection of high radiation. The emergency makeup air filter trains, designed to remove radioactive particulates and absorb radioactive iodine from the minimum quantity of outside air, are automatically started upon high radiation signals from two-out-of-four radiation monitor channels. The four monitor channels are divided into two trip systems. High radiation signals from Monitor channels A and B or C and D will start the emergency makeup filter train for each intake.

The emergency makeup air filter trains, recirculation filters, and control room shielding are designed to limit the occupational dose below levels required by 10 CFR 50.67.

The introduction of the minimum quantity of outside air to maintain the control room and other areas served by the control room HVAC system at a positive pressure with respect to surrounding potentially contaminated areas, at all the station operating conditions except when the system is in recirculation mode, precludes infiltration of unfiltered air into the control room.

The physical location of two redundant outside air intakes provides the option of drawing makeup air to the control room HVAC system from either of them depending upon the lesser contamination level, during and after a LOCA. It is possible that due to outside wind direction after a LOCA, one of the air intakes may not have any contaminants, while the other intake may have contaminants. The former may be utilized for makeup air in the control room. This provides additional security towards maintaining the habitability of the control room. The radiological consequences due to radioactivity drawn into the control room or AEER are provided in section 15.6.5.5.

6.4.5 Testing and Inspection

The control room HVAC system and its components are thoroughly tested in a program consisting of the following:

a. factory and component qualification tests,

b. onsite preoperational testing, and

c. onsite subsequent periodic testing.

d. The CRE is tested in general compliance to regulatory Guide 1.197, sections C.1 and C.2.

Written test procedures establish minimum acceptable values for all tests. Test results are recorded as a matter of performance record, thus enabling early detection of faulty performance.

All equipment is factory inspected and tested in accordance with the applicable equipment specifications, codes, and quality assurance requirements. System ductwork and erection of equipment is inspected during various construction stages for quality assurance. Construction tests are performed on all mechanical components and the system is balanced for the design airflows and system operating pressures. Controls, interlocks, and safety devices on each system are cold checked, adjusted, and tested to ensure the proper sequence of operation.

The inplace HEPA and Charcoal filter testing acceptance criteria, and the decontamination efficiency for the emergency makeup unit comply with the values listed in Reg. Guide 1.52, Revision 2.

6.4.6 Instrumentation Requirements

All the instruments and controls for the control room HVAC system are electric or pneumatic.

a. Each redundant control room HVAC system has a local control panel and each is independently controlled. Important operating functions are controlled and monitored from the main control room.

b. Instrumentation is provided to monitor important variables associated with normal operation. Instruments to alarm abnormal conditions are provided in the control room.

c. A radiation detection system (instrument range 0.10 to 10,000 mr/hr.) is provided to monitor the radiation levels at the system outside air intakes and inside the control room. A high radiation signal is alarmed on the main control board.

d. The ammonia detection system is provided to detect the presence of ammonia at outside air intakes. Ammonia detection is annunciated locally and in the main control room.

e. The ionization detection is provided in the outside and return air path from associated areas. Ionization detection is annunciated locally and on the main control board via the fire detection control panel.

f. The control room HVAC system is designed for automatic environmental control with the manual starting of fans. The refrigeration equipment has a manual auto switch.

g. A fire protection water spray system is provided to each charcoal adsorber / absorber bed.

h. The various instruments of the control system are described in detail in Chapter 7.0.

i. The emergency makeup air filter train airflow rate and upstream HEPA filter differential pressure is transmitted to the main control board, recorded, and alarmed.

TABLE 6.4-1

DOSE RATES IN THE CONTROL AND AUXILIARY ELECTRIC

EQUIPMENT (AEE) ROOMS DURING NORMAL OPERATIONS

COMPONENT	SOURCE	RADIATION	AREAS AFFECTED	TOTAL SHIELD THICKNESS (INCHES)*	CALCULATED DOSE RATE (mr/hr)
RWCU pump	Reactor water	Direct gamma	Control room	56	<0.1
			AEE room	42	<0.2
Skyshine	Reactor steam	Scattered gamma	Control room	30	<0.1
			Computer room	12	<0.5
Main steam tunnel	Reactor steam	Direct gamma	Control room	56	<0.5
			AEE room	56	<0.5
Station vent stack	Off-gas	Direct gamma	Control room	40	<0.1
Feedwater pump	Reactor steam	Direct gamma	Computer room	48	<0.1
Calibration facility	Cs-137	Direct gamma	AEE room	24	<0.1

* Thickness is given in inches of ordinary concrete with density of 140 pounds per cubic foot

TABLE 6.4-2

SHINE DOSE EXPERIENCED BY CONTROL ROOM PERSONNEL FOLLOWING LOSS-OF-COOLANT ACCIDENT*

SOURCE	SOURCE DISTRIBUTION **	SHIELD MODEL***	ACTUAL SHIELD***	MAXIMUM DOSE RATE (R/hr)	ACCUMULATED** DOSE (rem)
1. Primary Containment	100% Nobles, 50% Halogens, 1% Particulates evenly distributed	72 in. R.B. + 56 in. wall	72 in. R.B. + 56 in. wall		
a. Airborne	100% on west side†			$.6 \times 10^{-7}$	4E-6
b. Plate out				4×10^{-1}	2E-8
2. Reactor Building	0.5% per day from 1 above	56 in. wall, 36 in. ceiling	56 in. wall, 48 in. ceiling		
a. Airborne	evenly distributed			2×10^{-5}	3.5E-3
b. Plate out A	87% on west side†			1×10^{-5}	1.6E-2
c. Refueling floor plate out B	13% on west side†			1.2×10^{-3}	6E-6
3. SGTS Filter Unit	100% Halogens, particulates from 2a	36 in. R.B. + 56 in. wall	124 in. R.B. + 56 in. wall	2×10^{-9}	1.3E-5
4. Exhaust Clouds	Exhaust from 3, 100% Nobles, 10% Halogens				
a. External to stations		40 in. wall	40 in. wall	2×10^{-4}	9.9E-4
b. Airborne adjacent to control room		24 in. wall	24 in. wall	$<1 \times 10^{-7}$	1.6E-5
5. Control Room Air Intake Filter Unit	Exhaust from 3 100% Nobles, 10% Halogens	24 in. ceiling	36 in. ceiling	2×10^{-2}	1.2E-2
			Total(rem):	$<9.4 \times 10^{-1}$ leak rate of a 0.005/day <1 2 leak rate of 0.00635/day	

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* Due to sources outside the control room an average χ/Q was used to calculate the sources on the control room intake filter; more than 2/3 of this value is due to fumigation.

** For calculation purposes, the duration of the LOCA was chosen to be 30 days. No credit was taken for containment spray or mixing in the secondary containment. The filter efficiency for the SGTS filter units is 99% for halogens and 99.95%, including filter bank bypass for particulates.

*** Thickness of ordinary concrete with density of 140 pounds per cubic foot.

† 50% of the available halogens particulates are plated out as indicated.

Note 1: The doses due to radioactivity drawn into the Control Room and Auxiliary Electric Equipment Room are given in section 15.6.5.5.

Note 2: This table was developed based upon the original source term used in the DBA LOCA analysis. The source term and distribution have been revised and the primary containment leak rate increased from 0.625% to 1%/day, but the resultant dose is negligible compared to the 10 CFR 50.67 limits.

6.5 FISSION PRODUCT REMOVAL AND CONTROL SYSTEMS

6.5.1 Engineered Safety Feature (ESF) Filter Systems

The following filtration systems which are required to perform safety-related functions are provided:

a. Standby gas treatment system: This system is utilized to reduce halogen and particulate concentrations in gases leaking from the primary containment and which are potentially present in the secondary containment (reactor building) following the accident.

b. Control room and Auxiliary Electric Equipment Room (AEE Room) HVAC emergency makeup air filter units and recirculation filters: These systems are utilized to clean the outside air of halogen and particulates, which are potentially present in outside air following an accident, before introducing air into the control room or AEER HVAC system.

6.5.1.1 Design Bases

6.5.1.1.1 Standby Gas Treatment System

a. The standby gas treatment system is designed to automatically start in response to any one of the following signals:

   1. high pressure in Unit 1 or Unit 2 drywell,

   2. low-water level in Unit 1 or Unit 2 reactor,

   3. high radiation in exhaust air from over the fuel handling pools in the reactor building for either Unit 1 or Unit 2,

   4. high radiation in the ventilation exhaust plenum for reactor building for either Unit 1 or Unit 2, and

   5. manual activation from the main control room.

b. The radioactive gases leaking from the primary containment and which are potentially present in the secondary containment after a LOCA are treated in order to remove particulate and radioactive and nonradioactive forms of iodine to limit the offsite dose to the guidelines of 10 CFR 50.67.

c. The capability of one SGTS train to draw down the pressure in the secondary containment to -0.25 in. H_2O, and to maintain that secondary containment pressure, is verified on a staggered basis in accordance with Technical Specifications.

d. Any primary containment leakage (except that which is treated by the MSIV-ICLTM) will be contained within the secondary containment free air volume and will only reach the outside after passing through the SGTS. The secondary containment inleakage is determined by utilizing published leakage data for applicable building construction and incorporating known leakage values for piping, electrical, and duct penetrations at pressure control boundaries. The SGTS flow rate is approximately equal to the total free air volume of the reactor buildings for both Units 1 and 2 evacuated at a rate of one per day. The design flow rate through the SGTS also accounts for volumetric expansion of both reactor building air volumes due to temperature rises as equipment residual heat is released after ventilation and process system shutdown.

e. The secondary containment leakage is calculated in the following manner:

1. Assume laminar flow through small cracks, thus

$$Q = K \Delta P$$

where:

ΔP is the pressure differential across the secondary containment boundary; Q is the airflow rate (leakage); and K is the loss coefficient.

2. Take a secondary containment leak rate of 4000 ft^3/min at still wind conditions with -0.25 inch (H_2O) differential pressure between the outdoor ambient condition and the in-containment pressure.

3. Assume the manufacturer's certified leak test results on the siding for the reactor building.

4. Accept the air leakage test results contained in "Conventional Building for Reactor Containment," NAA-SR-10100.

f. Two full-capacity standby gas treatment system equipment trains and associated dampers, ducts, instruments, and controls are provided.

g. Each train is sized and specified for the worst conditions, treating incoming air-steam mixtures saturated at 150° F containing fission products and incoming particulates released from primary containment at the Tech. Spec. leakage rate as determined in accordance with Regulatory Guide 1.183. The design nominal volume rate for each train was established at 4000 cfm.

h. Each equipment train contains the amount of charcoal required to absorb the inventory of fission products leaking from the primary containment, based on a one unit LOCA.

i. Each train is designed with the proper air heaters, demister, and prefilters needed to assure the optimum gas conditions entering the high-efficiency particulate air (HEPA) and charcoal filters. The air heater is sized to reduce air entering at 150° F, 100% relative humidity to a maximum 70% relative humidity. The demister is specified to remove any entrained moisture in the airstream.

j. A standby cooling air fan is provided for each equipment train to remove heat generated by fission product decay on the HEPA filters and charcoal adsorbers after shutdown of the train.

The standby cooling air fan is conservatively sized to remove approximately 7700 Btu/hr of heat (generated by instantaneous deposition of iodine, on a HEPA filter bank and charcoal adsorbers) with less than a 50° F rise in cooling air temperature. This will limit the air temperature in the SGTS to 200° F maximum to prevent possible desorption and fire. Charcoal desorption temperature is given in ORNL-NSIC-65. No credit is taken for equipment or environment heat sink. Reactor building cooling air is routed through the shutdown train and exhausted to the atmosphere via the plant vent stack.

k. The SGTS exhibits a removal efficiency of no less than 99% on radioactive and nonradioactive forms of iodine and no less than 99.95%, including filter bank bypass on all particulate matter 0.3 micron and larger in size. The particulate removal efficiency is predicated on the use of 99% particulate removal efficiency. The physical property of new charcoal purchased shall meet requirements specified in Table 5-1 of ANSI/ASME N509-1980. Performance requirement shall be as specified in Table 5-1 of ANSI/ASME N509-1980 with penetration less than 0.5% as tested per ASTM D3803-1989. The charcoal is contained in gasketless, all welded construction adsorbers to preclude bypass of the charcoal and to ensure the highest removal efficiencies on methyl iodine.

   The exhaust air from each SGTS is routed through a seismically supported duct and is an elevated release at an elevation of 1080 feet above mean sea level, approximately 186 feet 8 inches above the highest structure. The discharge air velocity from the SGTS vent exhaust pipe is approximately 1270 fpm. This high point release provides effluent dispersion ratios sufficient to meet the dose requirements of 10 CFR 50.67.

l. The SGTS is designed with redundancy to meet single failure criteria.

m. The power supplies meet IEEE 308 criteria and ensure uninterrupted operation in the event of loss of normal a-c power. The controls meet IEEE 279.

n. The SGTS is designed to Seismic Category I requirements.

o. The SGTS is designed to permit periodic testing and inspection of the principal system components described in the following subsections.

6.5.1.1.2 Emergency Makeup Air Filter Units:

a. The emergency makeup air filter unit is designed to start automatically and provide outside air to the control room and auxiliary electric equipment room HVAC systems in response to any one of the following signals:

   1. high radiation signal from the radiation monitors installed in outside air intake louvers for the control room and auxiliary electric equipment room HVAC systems; and

   2. manual activation from the main control room.

b. The Regulatory Guide 1.183 source model in conjunction with approved methods is used to calculate the quantity of activity released as a result of an

accident and to determine inlet concentrations to the emergency makeup air filter train. See section 15.6.5.5 for additional details.

c. The capacity of the emergency makeup air filter units is based on the air quantity required to maintain the rooms served by the control room HVAC and auxiliary electric equipment room HVAC systems at a positive pressure with respect to adjacent areas.

d. Two full capacity emergency makeup air filter units and associated dampers, ducts, and controls are provided.

e. Each unit is designed with the proper air heaters, demister, and prefilters needed to assure the optimum air conditions entering the high-efficiency particulate air (HEPA) and charcoal filters.

f. The emergency makeup filter unit removal efficiency utilized in the AST dose analysis is 90% on radioactive and non radioactive forms of iodine and 99%, including filter bypass on all particulate matter 0.3 micron and larger size. The emergency makeup filter unit is conservatively tested in accordance with the ventilation filter test program to have a removal efficiency of no less than 95% on radioactive and non radioactive forms of iodine and no less than 99.95%, including filter bypass on all particulate matter 0.3 micron and larger size.

g. The emergency makeup air filter unit is designed to meet single failure criteria.

h. The power supplies meet IEEE 308 criteria and ensure uninterrupted operation in the event of loss of normal a-c power. The controls meet IEEE 279.

i. The emergency makeup air filter units are designed to Seismic Category I requirements.

j. The emergency makeup air filter units are designed to permit periodic testing and inspection of principal system components described in the following subsections.

k. Each control room and AEER HVAC subsystem has a supply air filter unit that contains a charcoal filter unit, called the recirculation filter. Each filter unit consists of a pre-filter and a normally bypassed charcoal filter. Upon detection of smoke in the return ductwork, the charcoal filter is automatically placed in service. Within 2 minutes of detection of high ammonia concentration in the air intake, the Operator will align the CRE HVAC systems in recirculation mode and

will don a self –contained breathing apparatus. Upon detection of high radiation, the Operator must manually place the charcoal filter on-line within 4 hours of detection to maintain the control room dose to within 10 CFR 50.67.

6.5.1.2 System Design

6.5.1.2.1 Standby Gas Treatment System

a. The schematic design of the SGTS is shown in Drawing No. M-89. Nominal size of principal system components are listed in the Table 6.5-1.

b. The SGTS is automatically or manually started to treat air exhausted from either reactor building. Two completely redundant parallel process systems are provided, each having a nominal capacity of 4000 ft3/min (at 150° F).

   As indicated on the schematic in Drawing No. M-89, each process system may be considered as an installed spare. The process systems have separate equipment trains, isolation valves, power feeds, controls, and instrumentation. Two full capacity redundant standby gas treatment system equipment trains are provided. One equipment train is located in the Unit 1 reactor building and the other equipment train is located in the Unit 2 reactor building. The suction and discharge side of both trains are headered together so that either of the trains can treat the air from both reactor buildings. Each SGTS equipment train and damper on the suction and discharge side of corresponding trains are powered by electrical essential Division 2 of the related unit. Either secondary containment isolation power signal starts both equipment trains and activates both alarms in the main control room. The operator then shuts down one of the standby gas treatment system equipment trains after ensuring that at least one of the two redundant trains is operating. The intake connections used for the standby gas treatment system are located on reactor building Units 1 and 2 floor elevation 820 feet 0 in. No redundant duct system component is located within 20 feet of its counterpart in areas where credible internal missiles or pipe whips might compromise redundancy.

c. Each SGTS has the following components:

   1. A primary fan for inducing the air from the spaces listed previously and discharging it through the filter train and common discharge pipe for elevated release to atmosphere. The fan performance and motor selection are based on the worst environmental conditions inside the reactor building. The flow and pressures are listed in Table 6.5-1.

2. A standby cooling air fan is sized to dissipate heat generated by fission product decay on the filters. The 200 ft^3/min flow capacity limits the maximum temperature in the train to 200° F for 150° F entering air temperature. The fan is used only after train shutdown and when the electric heater and primary fan are not operating.

3. A demister which removes any entrained water droplets and moisture to minimize water loading on the prefilter. The

demister meets qualification requirements similar to those in MSAR 71-45 and is in UL Class I.

4. A single stage electric heater is sized to reduce the humidity of the airstream to at least 70% relative humidity for the worst inlet conditions. An analysis of heater capabilities for various entering saturated air conditions ranging from 65° F to 150° F yields a peak heating requirement of 47,000 Btu/hr at 95° F entering air temperature. A 23-kW heater is provided.

5. A prefilter, UL listed, all-glass media, exhibiting no less than 85% efficiency based on ASHRAE atmospheric dust spot test.

6. A high-efficiency particulate air (HEPA) filter, water resistant, capable of removing 99.95% minimum of particulate matter which is 0.3 micron or larger in size. The filter is designed to be fire resistant. Four, 1000-ft /min elements are provided. All elements are fabricated in accordance with Military Specification MIL-F-51068, MIL-F-51079 and UL-586. The elements are size 5 with IIB element frame material. Gasket material will be SCE 43 per ASTM D1056. Testing of the HEPA filter banks is described in Subsection 6.5.1.4.

7. A charcoal adsorber capable of removing not less than 99% of radioactive and nonradioactive forms of iodine. The charcoal adsorber is a gasketless, welded seam type, filled with impregnated coconut shell charcoal. The bank holds a total of approximately 5800 pounds of charcoal.

   The charcoal specification requires an ignition temperature test and a methyl iodide test on each batch of charcoal supplied. In addition, model tests or previous qualification test data were required to demonstrate the effectiveness of the bed design before construction of the actual beds. Test data proving uniform packing density of charcoal in beds was also required.

   Ten test canisters are provided for each adsorber. These canisters contain the same depth of the same charcoal as is in the adsorber. The canisters are mounted, so that a parallel flow path is created between each canister and the adsorber. Periodically one of the canisters is removed and laboratory

tested to reverify the adsorbent efficiency. Two deluge valves in parallel connected to the station fire protection system are mounted outside of the charcoal adsorber. The charcoal bed is provided with a high temperature detector. The detector sensing high adsorber temperature will actuate an alarm in the main control room. High temperature alarms are nominally set at 310 °F. Manual charcoal deluge valves are operated locally and then solenoid operated valves are operated from the control room. The normally manual closed isolation valves upstream of the deluge valve in all cases require local actions to initiate water flow.

8. A high efficiency particulate filter identical to the one described in item 6 previously is provided to trap charcoal fines which may be entrained by the airstream.

d. Flow control valves are utilized upstream to regulate flow through the train. The train upstream static pressure will fluctuate between +1 and -1 inches water gauge.

e. Full-size access doors to each filter compartment are provided in the equipment train housing. Access doors are provided with transparent portholes to allow inspection of components without violating the train integrity.

f. The housing is of all welded construction, heavily reinforced.

g. Interior lights with external light switches, are provided between all train components to facilitate inspection, testing, and replacement of components.

h. Filter frames are in accordance with recommendations of Section 4.3 of ORNL-NSIC-65.

i. The height of release of the standby gas treatment system vent to the atmosphere is at elevation 1080 feet (186 feet 8 inches above the highest structure on the station).

6.5.1.2.2 <u>Emergency Makeup Air Filter Units</u>

a. The emergency makeup air filter units work in conjunction with the control room and auxiliary electric equipment room HVAC system as described in Subsection 9.4.1. The nominal size of principal system components is listed in Table 9.4-1.

b. In the event of high radiation detection in the outside air intakes of the control room HVAC system, the radiation monitoring system automatically shuts off normal outside air supply to the system and routes the outside air through the emergency makeup air filter train and fan (for removal of radioactive particulates and iodine), before being supplied to the control room and auxiliary electric equipment room HVAC systems.

c. Two emergency makeup air filter trains and fans are provided, each capable of handling 4000 cfm nominal of outside air, providing approximately 1500 cfm to the control room HVAC system and approximately 2500 cfm to the auxiliary electric equipment room HVAC system.

d. Each emergency makeup air filter unit is comprised of the following components in sequence:

    1. A demister which removes any entrained water droplets and moisture to minimize water droplets and water loading of the prefilter. The demister will meet qualification requirements similar to those in Mine Safety Appliance Research (MSAR) report 71-45 and will be UL Class I.

    2. A single stage electric heater, sized to reduce the humidity of the airstream to at least 70% relative humidity for the worst inlet conditions. An analysis of heater capacities for various entering saturated air conditions ranging from - 10° F to 95° F yields a peak heating requirement of 60,000 Btu/hr at 95° F. A 20-kW heater is provided.

    3. A prefilter, UL listed, all glass media, exhibiting no less than 85% efficiency based on ASHRAE Standard 52.2 method of testing.

    4. A high-efficiency particulate (HEPA) filter, water resistant, capable of removing 99.97% minimum of particulate matter which is 0.3 micron or larger in size. The filter is designed to be fire resistant, as may be required after consideration of heat generation from postulated deposit of fission products. Four 1000 cfm elements are provided. All elements are fabricated in accordance with Military Specification MIL-F-51068, MIL-F-51079, and UL-586.

5. A charcoal adsorber capable of removing not less than 95% of radioactive forms of iodine is provided. The charcoal absorber is an all welded gasketless type filled with impregnated coconut shell charcoal. The charcoal adsorber beds hold approximately 650 pounds of charcoal.

   The bed dimensions are so designed that the air has at least 0.25 seconds of residence time through the charcoal. The physical property of new charcoal purchased shall meet requirements specified in Table 5-1 of ANSI/ASME N509-1980. Performance requirement shall be as specified in Table 5-1 of ANSI/ASME N509-1980 with penetration less than 0.5% as tested per ASTM D3803-1989.

   The charcoal specification requires an ignition temperature test and a methyl iodine test on each batch of charcoal supplied.

   Ten test canisters are provided for the charcoal adsorber. These canisters contain the same depth of the same charcoal as in the charcoal adsorber. The canisters are so mounted that a parallel flow path is created between each canister and the charcoal adsorber. Thus, the charcoal in the canisters is subjected to the same contaminants as the charcoal in the bed. Periodically, one of the canisters is removed and laboratory tested to reverify the absorbent efficiency.

   Two deluge valves connected to the station fire water system are mounted adjacent to each charcoal adsorber. Manual charcoal deluge valves are operated locally. The normally closed manual isolation valves upstream of the solenoid deluge valve, in all cases, require local actions to initiate water flow. The deluge system will spray the adsorber compartment and thereby precluding the chance of an adsorber fire.

6. A high-efficiency particulate filter identical to the one described in item 4 is provided to trap charcoal fines which are entrained by the airstream.

7. A fan induces the air from the intake louvers and the makeup air filter train and discharges it to the suction side of the control room air handling equipment train. The fan performance is based on the maximum density and worst pressure condition, when it is inducing -10° F air from the outdoors and the makeup air filter train, containing filters which operate at no less than

twice their clean pressure drop.

8. Full size access doors adjacent to each filter are provided in the equipment train housing. Access doors are provided with transparent portholes to allow inspection and maintenance of components without violating the train integrity. Spacing between filter sections is based on ease of maintenance considerations.

9. The housing is an all welded construction, heavily reinforced, and built to tight leakage requirements.

10. Interior lights with external light switches are provided between all train components to facilitate inspection, testing, and replacement of components.

6.5.1.2.3 Supply Air Filter Unit Recirculation Filter

Each control room and AEER HVAC subsystem has a supply air filter unit that contains a charcoal filter unit, called the recirculation filter. Each filter unit consists of a pre-filter and a normally bypassed charcoal filter. Upon detection of smoke in the return ductwork, the charcoal filter is automatically placed in service. [redacted]

6.5.1.3 Design Evaluation

6.5.1.3.1 Standby Gas Treatment System

The Standby Gas Treatment System (SGTS) is designed to preclude direct exfiltration of contaminated air from either reactor building following an accident or abnormal occurrence which could result in abnormally high airborne radiation in the secondary containment. Equipment is powered from essential buses and all power circuits will meet IEEE 279 and IEEE 308. Redundant components are provided where necessary to ensure that a single failure will not impair or preclude system operation. A standby gas treatment system failure analysis is presented in Table 6.5-2.

An analysis was performed to determine the SGTS equipment capacity, based on the total inleakages to the secondary containment for both Units 1 and 2, while all the areas in the secondary containment are maintained at 0.25-inch water gauge negative. The secondary containment air pressure will begin to increase and

approach 0 in. H_2O (i.e., rises from initial -0.25 in. H_2O to 0 in. H_2O) due to inleakage into the secondary containment during post-LOCA and at times when SGTS is started. The secondary containment air pressure begins to decrease exponentially at the time the SGTS reaches its full capacity. As required by the Technical Specifications, within 15 minutes the secondary containment pressure will be reduced to -0.25 in. H_2O with the SGTS at full capacity (see Figure 6.3-80). During this time period, the pressure difference is always negative (assuming 0 wind speed); therefore, only inleakage from the outside atmosphere can occur.

6.5.1.3.2 Emergency Makeup Air Filter Units

The emergency makeup air filter units work in conjunction with the control room and auxiliary electric equipment room HVAC systems to maintain habitability in the control room and auxiliary equipment rooms. The design evaluation is given in Subsection 6.4.4.

6.5.1.4 Tests and Inspections

6.5.1.4.1 Standby Gas Treatment System

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

   1. factory and component qualification tests,

   2. onsite preoperational testing, and

   3. onsite periodic testing.

   Written test procedures establish minimum acceptable values for all tests. Test results are recorded as a matter of performance record, thus enabling early detection of depleted performance.

b. The factory and component qualification tests consist of the following:

   1. equipment train housing - a leak test $\pm$2.0 psig internal pressure, and magnetic particle or liquid penetrant testing per Section III of ASME Boiler and Pressure Vessel Code of all welds which could cause bypass leakage around HEPA filters or adsorber beds;

   2. demister - qualification test or objective evidence to demonstrate compliance with specified design criteria;

   3. HEPA filters - elements tested individually by applicable inspection and testing methods;

   4. HEPA filter frames - liquid penetrant test per ASME B&PV Code Section III of all welds which could cause bypass leakage around HEPA filters.

   5. adsorbent beds - model test of bed or objective evidence to demonstrate flow pressure characteristics, channeling effects;

6. adsorbent - qualification tests for ignition temperature and methyl iodine removal efficiency test;

7. fans - tested in accordance with the latest revision of AMCA Standard 210 "Air Moving and Conditioning Association Test Code for Air Moving Devices," to establish characteristic curves, etc.;

8. heater - uniform temperature test, high temperature cutout test, and adjacent equipment temperature test;

9. prefilter - objective evidence or certification that ASHRAE efficiency specified is attained; and

10. valves - shop tests demonstrating leaktightness, closure times.

c. The onsite preoperational tests are discussed in Section 14.1 of the FSAR.

d. Onsite periodic testing - Operating personnel are trained and required to make surveillance checks. These checks shall include visual inspection and periodically running the equipment trains for performance testing as outlined in the Technical Specifications.

6.5.1.4.2 Emergency Makeup Air Filter Units

a. The emergency makeup air filter unit and its components were thoroughly tested in a program consisting of the following:

1. factory and component qualification tests,

2. onsite preoperational testing, and

3. onsite subsequent periodic testing.

Written test procedures establish minimum acceptable values for all tests. Test results are recorded as a matter of performance record, thus enabling early detection of faulty performance.

b. The factory and component qualification tests consisted of the following:

1. Filter Train Housing

a) leak test at design internal pressure, and

b) magnetic particle or liquid penetrant testing per Section III of ASME Boiler and Pressure Vessel Code of all welds which could cause bypass leakage around HEPA filters or absorber bed.

2. Demister

qualification test or objective evidence to demonstrate compliance with specified design criteria.

3. Prefilter

objective evidence or certification that ASHRAE efficiency specified were attained.

4. HEPA Filters

elements tested individually in accordance with applicable inspection and testing methods.

5. HEPA Filter Frames

liquid penetrant testing per ASME B&PV Code Section III of all welds which could cause bypass leakage around HEPA filters or adsorber bed.

6. Adsorbent Beds

model test of bed or objective evidence to demonstrate flow pressure characteristics, channeling effects.

7. Adsorbent

qualification tests for ignition temperature and methyl iodine removal efficiency test.

8. Fans were tested in accordance with the latest revision of AMCA Standard 210 "Air Moving and Conditioning Association Test Code for Air Moving Devices," to establish characteristic curves, etc.

9. Heater

   a) uniform temperature test,

   b) high-temperature cutout test, and

   c) adjacent equipment temperature test.

10. The onsite preoperational testing as described in Chapter 14.0 of the FSAR.

11. Onsite subsequent periodic testing as described in the Technical Specifications.

6.5.1.5 <u>Instrumentation Requirements</u>

a. Differential pressure indicators are provided to measure the pressure drop across each filter. Pressure differential across the upstream HEPA filter is transmitted to the main control board, recorded, and alarmed on high-pressure differential.

b. Each adsorber bed is provided with high-temperature detectors. The temperature detector actuates an alarm in the control room when the increase in adsorber temperature is beyond a preset value.

c. Manual charcoal deluge valves are operated locally. The normally closed manual isolation valves upstream of the solenoid deluge valve, in all cases, require local actions to initiate water flow. The deluge system will spray the adsorber compartment and thereby precluding the chance of an adsorber fire.

d. All power-operated isolation valves are supplied with position switches to provide positive indication on the main control board.

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

f. Flow signals are transmitted to the main control board for indication recording and are used as an input to a flow control valve provided upstream of each equipment train.

g. Remote manual operation is provided on the main control board for each fan, and each deluge valve.

6.5.1.6 Materials

a. All component material is capable of a service life of 40 years normal operation plus 6 months post-LOCA at the maximum cumulative radiation exposure without any adverse effects on service, performance, or operation. All materials of construction are compatible with the radiation exposure set forth. This includes but is not limited to all metal components, seals, gaskets, lubricants, and finishes, such as paints, etc. The integrated dose following the once-in-a-lifetime post-LOCA, uses the values given in UFSAR Section 3.11.

b. Care is taken to avoid the use of any compounds or other chemicals during fabrication or production that contain chlorides or other constituents capable of inducing stress corrosion in stainless steels which are used in the adsorber bed.

c. Pressure and temperature - All components, including the housings, shall be designed in accordance with the applicable pressure and temperature conditions.

d. All filter unit gaskets and seal pads are closed-cell, ozone resistant, oil-resistant neoprene or silicone-rubber sponge, Grade SCE-43 in accordance with ASTM D1056.

e. Only adhesives as listed and approved under AEC Health and Safety Bulletin 306, dated March 31, 1971, covering Military Specification MIL-F-51068C, dated June 8, 1970, and all the latest amendments and modifications are used.

f. The organic compounds included in the filter train are as follows:

    1. charcoal;

    2. the binder in the HEPA filter media (the total weight of media per filter element is approximately 4 pounds, or a total of 32 pounds per equipment train);

    3. adhesive used in HEPA filters - approximately 1 liquid quart of fire-retardant neoprene adhesive is used to manufacture each HEPA filter;

    4. neoprene gaskets used on HEPA filters and o-rings are used on the charcoal filter sample canisters; and

5. the binder in the glass pads used in the demister section (this is a phenolic compound).

6.5.2 Containment Spray Systems

The containment spray systems are described in Section 6.3. The containment spray systems are not required for fissions product removal.

6.5.3 Fission Product Control System

The standby gas treatment system (SGTS) is used to control the cleanup of fission products from the containment following an accident and is described in detail in Subsection 6.5.1.

6.5.4 Ice Condenser as a Fission Product Cleanup System

Not applicable.

TABLE 6.5-1
(SHEET 1 OF 4)

STANDBY GAS TREATMENT SYSTEM COMPONENTS

NAME OF EQUIPMENT	TYPE, QUANTITY AND NOMINAL CAPACITY (per component)
A. Filter Unit	
1. Equipment Numbers	1VG01S, 2VG01S
2. Type	Package
3. Quantity	2
4. Components of Each Unit	
a. Fan	
Type	Centrifugal
Quantity	1
Drive	Direct
Capacity (ft^3/min)	4000 (nominal)
Static Pressure (in. H_2O)	14.8
b. Demister	
Type	Impingement
Quantity	1 Bank with 4 elements
Static resistance	
clean (in. H_2O)	0.95
dirty (in. H_2O)	1.7
c. Heater	
Type	Electric, sheathed, single stage

TABLE 6.5-1
(SHEET 2 OF 4)

NAME OF EQUIPMENT	TYPE, QUANTITY AND NOMINAL CAPACITY (per component)
Quantity	1
Capacity (kW)	23
Accessories	Overload cutout
d. Prefilter	
Type	High Efficiency
Quantity	1 Bank With 4 Elements
Efficiency (per ASHRAE) Dust Spot Test)	90%
Static resistance	
clean (in. H_2O)	0.35
dirty (in. H_2O)	1
e. HEPA Filters	
Type	Absolute High Efficiency
Quantity	4 Elements per Bank. Two Banks per Train
Media	Glass Fiber, Waterproof, Fire Resistant
Bank Efficiency (% with 0.3 micron particles)	99.97 (Purchased) 99.95 (Operational Requirement)
Static Resistance	
clean (in. H_2O)	0.7
dirty (in. H_2O)	2

TABLE 6.5-1
(SHEET 3 OF 4)

NAME OF EQUIPMENT	TYPE, QUANTITY AND NOMINAL CAPACITY (per component)
f. Charcoal Adsorber Bed	
Type	Vertical gasketless
Quantity	8 - 8 in. thick
Media	Impregnated Charcoal
Iodine Removal Efficiency (%)	99 (Operational Requirement) 99 (Operational Requirement)
Quantity of Media (lb)	5800
Depth of Bed (in.)	8
Residence Time for 8 in. bed (sec)	2.0
Static Resistance (in. H_2O)	4.6
g. Standby Cooling Air Fan	
Type	Centrifugal
Quantity	1
Drive	Direct
Capacity (ft^3/min)	200
Static Pressure (in. H_2O)	5

TABLE 6.5-1
(SHEET 4 OF 4)

NAME OF EQUIPMENT	TYPE, QUANTITY AND NOMINAL CAPACITY (per component)
B. Secondary Containment Isolation Dampers	
1. Equipment Numbers	1VQ037, 1VQ038 2VQ037, 2VQ038 1VR04YA&B, 1VR05YA&B 2VR04YA&B, 2VR05YA&B
2. Type	Special
3. Quantity	8
4. Operator	Air Cylinder
5. Diameter (in.)	72

TABLE 6.5-2

STANDBY GAS TREATMENT SYSTEM EQU1PMENT FAILURE ANALYSIS

COMPONENT	FAILURE	FAILURE DETECTED BY	ACTION
[redacted]	[redacted]	[redacted]	[redacted]
[redacted]		[redacted]	[redacted]
[redacted]	[redacted]	[redacted]	[redacted]
[redacted]		[redacted]	[redacted]
[redacted]	[redacted]	[redacted]	[redacted]
[redacted]	[redacted]		[redacted]
	[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]	[redacted]

6.6 INSERVICE INSPECTION OF ASME CODE CLASS 2 AND 3 COMPONENTS

6.6.1 Components Subject to Examination

All ASME Class 2 components (pressure vessels, piping, pumps, and valves) are inservice inspected according to ASME, B&PVC, Section XI, Subsection IWC, with appropriate addendum(s). The main steamlines (four) are inspected from the first outside containment isolation valve to the turbine stop valves. Inspection requirements are the same as for ASME Class 2 components.

All ASME Class 3 components (pressure vessels, piping, and valves) are inservice inspected according to ASME, B&PVC, Section XI, Subsection IWD, with appropriate addendum(s).

6.6.2 Accessibility

The design and arrangement of the ASME Class 2 and ASME Class 3 piping, pump, and valve components have been made accessible for inspection and examination as follows:

Pipe and Equipment Welds

Location and clearance envelopes have been established for inspection and examination. Contours and surface finish are acceptable for inspection and examination.

Insulation Removal

Piping or components to be inspected according to the Section XI code which are insulated, have been designed with removable numbered insulation panels.

Shielding

Piping or components to be inspected according to the Section XI code and are radiologically shielded have been designed with removable or accessible shields.

6.6.3 Examination Techniques and Procedures

Inservice inspection will be in accordance with ASME, B&PV Section XI.

6.6.4 Inspection Intervals

The initial 10-year inspection program for LaSalle units 1 and 2 was submitted to the NRC on July 13, 1982 and December 21, 1982, respectively. The inservice

inspection program for both units 1 and 2 are based on the requirements of the ASME, Section XI 1980 edition including addenda through winter 1980. The inservice examinations conducted during the second 120 month Inspection Interval will comply with the 1989 Edition of ASME Section XI, except in cases where relief has been granted by the NRC. The inservice examinations conducted during the third 120 month Inspection Interval will comply with the 2001 Edition through the 2003 addenda, including the December of 2003 Erratum of ASME Section XI, except in cases where relief has been granted by the NRC.

6.6.5 Examination Categories and Requirements

The inservice inspection categories and requirements for Class 2, and Class 3 components are in agreement with ASME Section XI.

Specific written requests for relief from ASME code requirements determined to be impractical were contained in the initial inservice inspection program. Relief from those requirements was granted by the NRC, detailed evaluation is included in Appendix C of NUREG-0519, Supplement No. 5, Safety Evaluation Report related to the operation of LaSalle County Station, Units 1 and 2.

6.6.6 Evaluation of Examination Results

The evaluation of Class 2 components examination results will comply with the requirements of Section XI.

The repair procedures for Class 2 and 3 components comply with the requirements of Section XI.

6.6.7 System Pressure Tests

All Class 2 system pressure testing complies with the criteria of Code Section XI, Article IWC-5000. All Class 3 system pressure tests comply with the criteria of Article IWD-5000.

6.6.8 Augmented Inservice Inspection to Protect Against Postulated Piping Failures

This inspection has been adequately covered by the requirements of Section XI already adhered to previously.

6.7 MAIN STEAM ISOLATION VALVE LEAKAGE CONTROL SYSTEM (MSIV-LCS) Unit 2 deleted, Unit 1 abandoned in place

The Main Steam Isolation Valve Leakage Control System provided originally has been deleted. The valve leakages are processed by the Isolated Condenser Leakage Treatment Method as discussed in Section 6.8.

6.8 Main Steam Isolation Valve - Isolated Condenser Leakage Treatment Method

The Main Steam Isolation Valve - Isolated Condenser Leakage Treatment Method (MSIV - ICLTM) (Also called the MSIV Alternate Treatments Leakage Paths) controls and minimizes the release of fission products which could leak through the closed main steam isolation valves (MSIV's) after a LOCA. The system provides this control by processing valve leakage through the main steamlines, main steamline drains, and the main condenser.

6.8.1 Design Bases

6.8.1.1 Safety Criteria

The following general and specific design criteria represent system design, safety, and performance requirements imposed upon the MSIV-ICLTM:

a. The safety function of the main steamlines and main steamline drains are described in LSCS-UFSAR Section 10.3.

b. The safety function of the main condenser is described in LSCS-UFSAR Section 10.4.1.

6.8.1.2 Regulatory Acceptance Criteria

The classification of the components and piping of the main steam supply system is listed in Table 3.2-1. All components and piping for the main steam supply system are designed in accordance with the codes and standards listed in Table 3.2-2 for the applicable classification.

The classification of the main condenser is described in LSCS-UFSAR Section 10.4.1.3.

6.8.1.3 Leakage Rate Requirements

The MSIV-ICLTM has been incorporated as an integral part of the BWR plant design. The design features employed with this systems are established to reduce the leakage rate of radioactive materials to the environment during a postulated LOCA. Leakage control requirements are imposed upon the MSIV-ICLTM in order to:

a. eliminate the possibility of secondary containment bypass leakage of accident induced radioactive releases,

b. allow for higher MSIV leakage limits, and

c. assure reasonable leakage verification test frequencies (once per fuel cycle).

The design and operational requirements imposed upon the MSIV-ICLTM relative to the foregoing criteria are established to:

a. allow MSIV leakage rates up to a total of 400 scfh for all MSIV valves,

b. allow a MSIV leakage rate verification testing frequency compatible with the requirements of plant operating technical specifications, and

c. assure and restrict total plant dose impacts below 10 CFR 50.67 guidelines.

6.8.2 System Description

6.8.2.1 General Description

The system provides this control by processing valve leakage through the main steamlines, main steamline drains, and the main condenser.

6.8.2.2 System Operation (U2 MSIV LCS delete, U1 Abandon-in-place)

With the deletion of the MSIV-LCS, MSIV leakage will pass from the outboard MSIV, through the main steamlines, main steamline drains and into the condenser. The large wetted volume in the main condenser plates out inorganic iodine and holds up other fission products that escape through the MSIVs, limiting release to the environment. This alternate pathway is more reliable than the MSIV-LCS since less equipment is employed. The alternate pathway also has a much higher capacity for processing leakage than does the MSIV-LCS, with a capacity of only 100 scfh. In addition, the MSIV-LCS will only operate at less than 35 psig reactor vessel steam dome pressure, whereas the alternate pathway is independent of reactor pressure.

To properly align the pathway, in addition to closing the MSIVs and the containment isolation valves, operators will close valves to isolate the leakage pathway from the auxiliary steam supplies. The operating drains will remain open and either one of two startup drains will be opened. All of the remote manually operated valves that need to be moved are powered from Class 1E power supplies. Although these valves and their power supplies (with the exception of the MSIVs) are not maintained as safety-related, design control for all of these valves is maintained with respect to their importance to safety. Appropriate changes to station

procedures have been made to reflect deletion of the MSIV-LCS and use of the alternate leakage treatment method.

6.8.2.3 Equipment Required

The following equipment components are provided to facilitate system operation:

a. piping - process piping is carbon steel throughout;

b. valves - motor-operated, standard closing speeds;

c. main condenser

6.8.3 System Evaluation

An evaluation of the capability of the MSIV-ICLTM to prevent or control the release of radioactivity from the main steamlines during and following a LOCA has been conducted. The results of this evaluation are presented in LaSalle County Nuclear Power Stations Units 1 and 2 Application for Amendment of Facility Operating Licenses NPF-11 and NPF-18, Appendix A, Technical Specifications, and Exemption to Appendix J of 10CFR50 Regarding Elimination of MSIV Leakage Control System and Increased MSIV Leakage Limits, NRC Docket Nos. 50-373 and 50-374.

Additionally, Sargent & Lundy performed an evaluation on the piping, condenser and turbine building, to assure they would remain functional during a seismic event to mitigate the radiologically consequences of MSIV leakage (Reference Sargent & Lundy Calculation 068078 (EMD), Rev. 2, dated 8/9/95 for Unit 1 and 067927 (EMD), Rev. 2 dated 8/10/95 for Unit 2).

See Section 15.6.5.5 for more information in the dose analysis and dose consequences.

6.8.4 Instrumentation Requirements

The instrumentation necessary for control and status indication of the MSIV-ICLTM is designed to function under Seismic Category I and environmental loading conditions appropriate to its installation with the control circuits designed to satisfy separation criteria. MSIV closed indication is powered from Class 1E power and is maintained as safety-related.

6.8.5 Inspection and Testing

Preoperational tests for the main steamlines, main steamline drains, and the main condenser are discussed in LSCS-UFSAR Sections 10.3.4 and 10.4.1.4. No additional testing is required to support this operating mode.

TABLE 6.8-1

DOSE CONSEQUENCES OF MSIV LEAKAGE

LEAKAGE 30 DAYS FOLLOWING LOCA-UNIT 1

(100 SCFH per line)

	WHOLE BODY DOSE (rem)	THYROID DOSE (rem)
Exclusion Area (509 meters)	1.451E-3	3.14E-2
Low Population Zone (6400 meters)	3.3E-2	10.47

ATTACHMENT 6.A

ANNULUS PRESSURIZATION

ATTACHMENT 6.A

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6.A ANNULUS PRESSURIZATION

6.A.1 INTRODUCTION

Annulus pressurization refers to the loading on the shield wall and reactor vessel caused by a postulated pipe rupture at the reactor pressure vessel nozzle safe-end to pipe weld. The pipe break assumed is an instantaneous guillotine rupture which allows mass/energy release into the drywell and annular region between the biological shield wall and the reactor pressure vessel (RPV).

The mass and energy released during the postulated pipe rupture cause:

a. A rapid asymmetric decompression acoustic loading of the annular region between the vessel and shroud from the pipe break at or beyond the vessel nozzle safe-end weld.

b. A transient asymmetric differential pressure within the annular region between the biological shield wall and the reactor pressure vessel (annulus pressurization).

c. A jet-stream release of the reactor pressure vessel inventory and the impact of the ruptured pipe against the whip restraint attached to the biological shield wall.

The results of the mass and energy release evaluation are then used to produce a dynamic structural analysis (force-time history) of the RPV and shield wall. The force time history output from the dynamic analysis is subsequently used to compute loads on the reactor components. The following is a more detailed description of the annulus pressurization calculation performed for the LaSalle County Station.

6.A.2 SHORT-TERM MASS ENERGY RELEASE

The postulated pipe rupture at the weld between recirculation or feedwater piping and the reactor nozzle safe end leads to a high rate of water and steam mixture into the annulus between the RPV and the shield wall. Figure 6.A-1 illustrates the location of this break. Calculation of the mass/energy release is performed using the generic method for short-term mass releases. This method and a sample calculation are described below. Figure 6.A-2 illustrates a typical mass flux vs. time for a feedwater line break.

The purpose of this procedure is to document the method by which short-term mass release rates are calculated. The flow rates which could be produced by a primary system line break for the first 5 seconds include the effects of inventory and subcooling. Optionally, credit may be taken for a finite break opening time.

ASSUMPTIONS

The assumptions are as follows:

a. The initial velocity of the fluid in the pipe is zero. When considering both sides of the break, the effects of initial velocities would tend to cancel out.

b. Constant reservoir pressure.

c. Initial fluid conditions inside the pipe on both sides of the break are similar.

d. Wall thickness of the pipe is small compared to the diameter.

e. Subcompartment pressure $\simeq$ 0.

f. Mass flux is calculated using the Moody steady slip flow model with subcooling.

g. For steamline breaks, level swell occurs at 1 second after the break with a quality of 7%.

NOMENCLATURE (See Figure 6.A-3)

A_{BR} - Break area.

A_L - Minimum cross-sectional area between the vessel and the break. This can be the sum of the areas of parallel flow paths.

C - Sonic velocity (see Figure 6.A-4).

D - Pipe inside diameter at the break location.

F_I - Inventory flow multiplier.

F_I = 0.75 for saturated steam.

FI = 0.50 for liquid and saturated steam-liquid mixtures.

g_c - Proportionality constant (=32.17 ^{2} lbm-ft/lbf-sec^2).

G - Mass flux.

G_C - Maximum mass flux (see Figure 6.A-5).

h_O - Reservoir or vessel enthalpy.

h_P - Initial enthalpy of the fluid in the pipe.

h_7 - Enthalpy at P_O and a quality of 7%.

L_I - Inventory length. The distance between the break and the nearest area increase of A_L whichever is less.

$\dot{M}$ - Mass flow rate.

$\dot{M}_I$ - Mass flow rate during the inventory period.

P_O - Reservoir or vessel pressure.

P_{SAT} - Saturation pressure for liquid with an enthalpy of h_P.

t - Time.

t_I - Length of the inventory period.

v - Specific volume of the fluid initially in the pipe.

V_I - Volume of the pipe between the break and A_L .

X - Separation distance of the break.

6.A.2.1 Instantaneous Guillotine Break

The following method should be applied to each side of the break and the results summed to determine the total flow.

Inventory Period

Prior to a pipe break, the fluid in the pipe is moving at a relatively low velocity. After the break occurs, a finite time is required to accelerate the fluid to steady-state velocities. The length of this time period is conservatively estimated as follows:

a. If $A_L / A_{BR} > F_I$,

$$t_I = \frac{2 L_I}{c} \tag{6.A-1}$$

b. If $A_L / A_{BR} < F_I$,

$$t_I = \frac{V_I}{A_{BR} \, G \, F_I \, v} \tag{6.A-2}$$

where G is calculated as shown in Subsection 6.A.2.4 for a large separation distance and $t < t_I$.

During this time period, the mass flow rate is calculated as

$$\dot{M}_I = G \, A_{BR} \, F_I \tag{6.A-3}$$

Steady-State Period

Following the inventory period, the flow is assumed to be choked at the limiting cross-sectional flow area.

For $t_I < t < 5.0$ seconds, (6.A-4)

$$\dot{M} = A_L \, G$$

6.A.2.2 Break Opening Flow Rate

See Table 6.A-1 for the pipe displacement time history for postulated recirculation suction pipe rupture and Figure 6.A-7 for the nomenclature used.

Inventory Period

The inventory period is determined as described in Subsection 6.A.2.1. The flow rate as a function of pipe separation distance is given by

$$\dot{M} = G \, \pi \, D \, X \tag{6.A-5}$$

where G is obtained by using the methods of Subsection 6.A.2.4 (a or b).

Determining Flow Rate

Following the inventory period, equation 6.A-5 is used to deter mine the flow rate where the mass flux, G, is determined from Subsection 6.A.2.4 (a, c, or d).

6.A.2.3 Combined Break Flow

To determine the total flow rate released from the break, the results of Subsections 6.A.2.1 and 6.A.2.2 are compared and whichever produces the smallest flow rate at any time is used (see Figure 6.A-6). Both methods produce maximum flow rates based on different limiting areas. The transfer from one curve to the other represents a change in the point where the flow is choked.

6.A.2.4 Determination of the Mass Flux, G

Depending on the time period, fluid conditions, and break separation distance, the mass flux is determined as follows:

$$X_B = \sqrt{1-(P_{SAT}/P_o)} \quad (D/2) \tag{6.A-6}$$

a. If $X < X_B$,

$$G = \sqrt{2 g_c P_o / v}$$

b. If $X > X_B$ and $t < t_I$
   $G = G_c\ (P_o\ ,\ h_p)$ from Figure 6.A-5

c. If $X > X_B$ and $t > t_I$
   $G = G_c\ (P_o\ ,\ h_o)$ from Figure 6.A-5

d. If the break is a steamline and $T > 1.0$, level swell occurs.
   $G = G_c\ (P_o\ ,\ h_7)$ from Figure 6.A-5

Note that for complete break separation (Subsection 6.A.2.1), X is always greater than X_B, and for saturated water, X_B is equal to zero.

6.A.2.5 Biological Shield Wall

For the purpose of analyzing the biological shield wall pressurization, credit may be taken for flow which escapes through the wall penetration. If the initial break location is in the annulus region between the wall and the vessel, no flow is assumed to escape through the penetration. If, however, it is located within the penetration itself, some of the flow may be assumed to escape. It is recommended

that the fraction of the flow which escapes be calculated based on the ratio of the minimum annular flow area between the penetration and pipe surface and between the penetration and pipe surface and between the penetration and the safe-end nozzle.

6.A.2.6 Comparison of the GE model with the Henry/Fauske Correlation

The GE methodology for calculating the mass energy release from a recirculation line break which results in an annulus pressurization event was provided the NRC's Mr. Denwood F. Ross, Assistant Director for Reactor Safety, via GE letter dated May 2, 1978, from Mr. E. D. Fuller of BWR Licensing. This methodology was used in the adequacy assessment made for LSCS.

The definition of the annulus pressurization is given in the introduction (Subsection 6.A.1). A description of the time aspects of the calculated mass and energy flow rates followed by a description of the modeling for the feedwater line and separately for the recirculation line is provided below. A comparison is then made between GE's analytical method and the method used in RELAP4/MOD5. Finally, both graphical and numerical results of this comparison are provided to substantiate the conclusion that the resulting break flows using the GE methods are much more conservative than those predicted by the use of RELAP for the LaSalle plant.

Timing Aspects of Mass and Energy Flow Rates

The GE method for calculating the short-term mass/energy release assumes that the overall time for mass release may be divided into two periods, the inventory period and the quasi-steady period. The inventory period is defined as the time required to accelerate the pipe fluid to steady-state velocities, at which time the flow is assumed to choke at minimum flow cross sections. During this time, the mass flux is based on initial thermodynamic conditions existing within the pipe. In the quasi-steady period, during which the flow is choked, the mass flux is based on thermodynamic conditions upstream from the choke points. For both time periods the mass flux is determined from a graph of critical mass flux versus enthalpy, as calculated by the Moody Slip Flow Method. Each side of the break is analyzed separately and the results summed to give the total mass release rate.

Method for Feedwater Line Modeling

The feedwater system for LaSalle County Station consists of the pumps, heaters, valves, and piping necessary for the transmission of hotwell condensate to the reactor vessel as part of the closed cycle cooling loop. LSCS has three feedwater pumps, two steam- driven and one electric-driven. During normal operation, the electric pump is in standby. The flow passes through a complex series of pipes and components from the feedwater pumps to the reactor vessel.

The break location for the feedwater line break is the safe-end to the pipe weld housed within the vessel to shield wall subcompartment. For the feedwater line break, instantaneous break opening is assumed. Flow for the vessel side passes through the feedwater nozzles of the broken line and out the break. Flow from the system side passes through the feedwater piping network and out the break.

The nodalization of the feedwater system is shown in Figures 6.A-8 and 6.A-9. A series of 24 modes was selected after sensitivity studies were completed which demonstrated that a 24-node model was conservative relative to higher-noded systems.

The broken feedwater leg to be analyzed was chosen by multiple RELAP runs to determine the limiting break location. The critical assumptions in the analysis are as follows:

a. The feedwater pumps are simulated as (constant) mass flow sources.

b. The reactor pressure vessel (RPV) is an infinite reservoir at constant temperature and pressure.

c. The temperature of the pump-side hydraulic network remains constant.

d. Appropriate sections of the hydraulic network are combined by means of "Ohm's Law" expressions for series and parallel circuits, assuming constant fanning friction actions.

e. The RPV thermodynamics state is subcooled at the prevailing temperature in the lower plenum (532° F).

The break is modeled as an instantaneous guillotine pipe break with complete pipe offset. Before the break occurs, a fully open valve connects, Volumes 18 and 19. Closed valves connect those volumes to Volume 1, an infinite sink at constant pressure and temperature (atmospheric conditions). The break is initiated at time zero by closure of the valve between Volumes 18 and 19 simultaneous opening of the valves to Volume 1.

<u>Method of Recirculation Line Modeling</u>

The recirculation system for LaSalle County Station is similar to the recirculation system of other BWR's. Flow is taken from the lower jet pump diffuser region, passed through 21-inch lines to a constant-speed pump, and then through a flow control valve to a header which feeds flow to five risers which provide flow to two jet pump nozzles each.

The nodalization for the recirculation line leak is shown in Figure 6.A-10. The system has been modeled using 21 nodes. The break is located at the vessel nozzle safe-end to pipe weld on the recirculation pump suction side. The type of break considered here has a finite break opening time. For this case the break opening is complete after 30 milliseconds, at which time the pipe offset longitudinal distance is 5.8 inches. The break area is modeled as the surface area of an imaginary volume having a length of 5.8 inches and a diameter equal to that of the recirculation pipe ID. This volume (#18) is connected by a valve (Type 3) to an infinite reservoir (volume #19), and also by valves (Type 2) to the vessel side volume (#27) and pump side volume (#21). Both valves (Type 1) also connect Volumes 17 and 21. It is normally open before the break, and at the initiation of the break, closes at the same rate as the other valves open. The sum of the areas of the Type 2 valves equals the pipe area.

This network of valves best represents the break with finite opening time. Valves of Type 2 are opened at the same rate as Type 3 to ensure that choking occurs at Junctions 21 and/or 22. Junction 23 (having valve Type 3) is in reality a fluid surface, and choking cannot physically occur there. Choking must at least occur at Junctions 21 and/or 22, where the fluid is constrained by the pipe diameter.

Other assumptions in the analysis include:

a. Negligible effects of core reactor kinetics on rated heat transfer to the core volume (Volume 2).

b. Constant flow of steam from the steam dome (Volume 5) at rated conditions.

c. Constant flow of feedwater at rated conditions.

d. Recirculation pumps trip at the time zero and are modeled via pump characteristic curves for coastdown.

e. Jet pump hydraulics were modeled as one equivalent pump per recirculation loop.

Comparison of General Electric Analysis to RELAP4/MOD5

For the annulus pressurization event, the NRC has questioned General Electric's method for computing mass and energy flow rates following a postulated LOCA from long lines containing subcooled fluid. A program was developed to expedite the licensing of the LaSalle County Station to perform RELAP analyses using appropriate assumptions and to compare the results with those obtained using General Electric's method.

RELAP4/MOD5 is a general computer program which can be used to analyze the thermal hydraulic transient behavior of a water- cooled nuclear reactor subjected to postulated accidents such as loss-of-coolant accidents. The program simultaneously solves the fluid flow, heat transfer, the reactor kinetics equations describing the behavior of the reactor.

Numerical input data is utilized to describe the initial conditions and geometry of the system being analyzed. This data includes fluid volume, geometry, pump characteristics, power generation, heat exchanger properties, and nodalization of fluid flow paths. Once the system has been described with initial flow, pressure, temperature, and power level boundary conditions, transients such as loss-of-coolant accident can be simulated by control action inputs. RELAP then computes fluid conditions such as flow, pressure, mass inventory and quality as a function of time.

For the brief transients considered here ($t \leq 0.5$ seconds), appreciable simplification of the overall thermal-hydraulic system, including the reactor pressure vessel, is justified owing to the relatively longer time constants which apply for heat transfer. Brief summaries of the modeling approaches for feedwater and recirculation line breaks are given below.

The assumptions applied to these analyses are as follows:

a. Feedwater line:

    1. LaSalle RELAP deck as basis.

    2. Henry-Fauske-Moody flow model is used.

    3. Instant break opening.

    4. Mass flux terms between vessel and break (short side) are eliminated.

b. Recirculation line:

    1. LaSalle RELAP deck as basis.

    2. Finite break opening time is allowed for.

    3. Henry-Fauske-Moody flow model is used.

    4. Momentum flux terms in RELAP between vessel and break (short side) are eliminated.

<u>Results of the Analysis</u>

The resulting break flows using the GE methods are much more conservative than those obtained by the use of RELAP. This is indicated graphically in Figures 6.A-11 through 6.A-13.

Conclusions

The mass release result for the GE mass energy release method and the RELAP4/Mod 5 calculations are compared in Figures 6.A-11 through 6.A-13 for the postulated feedwater line break and recirculation line break respectively. The analyses show that the GE method is conservative relative to RELAP 4/Mod 5 for both cases. The ration (r) of the GE method flow rates to those from RELAP/MOD5 is as follows:

Break Location	r(t = 0.1 sec)	r(t = 0.5 sec)
Feedwater (Leg EA)	2.300	1.70
Feedwater (Leg EB)	2.200	1.60
Recirculation Line	1.065	1.21

6.A.3 LOAD DETERMINATION

6.A.3.1 Acoustic Loads

Because the boiling water reactor (BWR) is a two-phase system that operates at or close to saturation pressure (1000 psi), the differential pressure across the reactor shroud is of short duration, and the BWR system is not subjected to a significant shock-type load with respect to structural supports. This short- duration acoustic load is confined to a bending moment and shear force on the reactor pressure vessel and reactor shroud support. The results of the integrated force acting on the reactor pressure vessel shroud determined by Method of Characteristics, are given in Table 6.A-2.

6.A.3.2 Pressure Loads

The pressure responses of the RPV-shield wall annulus for a recirculation suction line and a feedwater line were investigated using the RELAP4 computer code. An asymmetric model using several nodes and flow paths was developed for the analysis of the recirculation and feedwater line breaks. Further description of these analytical models and detailed discussion of the analyses may be found in Section 6.2.

The pressure histories generated by the RELAP4 code were in turn used to calculate the loads on the sacrificial wall and the reactor pressure vessel. The annulus was divided into seven zones and an eighth-order Fourier fit to the output

pressure histories made for each zone to produce the Fourier coefficients required for the structural analysis of the shield wall. The specific loading data consisted of the time-pressure (psia) histories for each node within the annulus. Time-force histories representing the resultant loads on the RPV for each node through its geometric center were generated by taking the product of the node pressure and its "effective" surface area.

A sample pressure-to-force calculation is shown in Subsection 6.A.4. Subsection 6.A.5 shows the nodalization schemes and pressure areas used in this calculation. The time-force histories (forcing functions) calculated at each nodal point for both a recirculation and a feedwater line break are shown in Subsection 6.A.7. The nodal points are illustrated in Figure 6.A-14.

6.A.3.3 Jet Loads

To address structural loads on the vessel and internals completely, jet thrust, jet impingement, and pipe whip restraint loads must be considered in conjunction with the above mentioned pressure loads. Jet thrust refers to the vessel reaction force with results as the jet stream of liquid is released from the break. Jet impingement refers to the jet stream force which leaves the broken pipe and impacts the vessel. The pipe whip restraint load is the force which results when the energy-absorbing pipe whip restraint restricts the pipe separation to less than one full pipe diameter. This restricted separation is accounted for as a finite break opening time in the mass/energy release calculation. These jet loads are calculated as described in ANSI 176 (draft), "Design Basis For Protection Of Nuclear Power Plants Against Effects Of Postulated Pipe Ruptures", January 1977.

The jet load forces used in this analysis are shown in Subsection 6.A.6. Although these values have been calculated for a recirculation line break only, they are also conservatively used for the feedwater load evaluation. This is conservative because the calculation of these jet effects depends largely on the area of the break, and the recirculation line is about 2.5 times larger in area. Figure 6.A-15 illustrates the location of the pressure loads and jet loads with respect to the RPV and shield wall.

The pressure loads and jet loads described above are then combined to perform a structural dynamic analysis. Both of these loads are appropriately distributed along a horizontal beam model, which is shown in Figure 6.A-14. The vessel coordinates of these nodal points are described in Table 6.A-3.

The force time histories are then applied to a composite lumped- mass model of the pedestal, shield wall, and a detailed representation of the reactor pressure vessel and internals. The DYSEA01 computer program is used for this analysis. This computer program is described in Subsection 6.A.3.4. The analysis produces acceleration time histories at all nodes for use in evaluating the reactor pressure vessel and internal components. Response spectra at all nodes are also computed.

The peak loading on the major components used to establish the adequacy of the component design is shown in Tables 6.A-4 and 6.A-5.

6.A.3.4 Dynamic and Seismic Analysis (DYSEA) Computer Program

The DYSEA (Dynamic and Seismic Analysis) program is a GE proprietary program developed specifically for seismic and dynamic analysis of RPV and internals/building systems. It calculates the dynamic response of linear structural systems by either temporal modal superposition or response spectrum method. Fluid- structure interaction effect in the RPV is taken into account by way of hydrodynamic mass.

The DYSEA program was based on the program SAP-IV with added capability to handle the hydrodynamic mass effect. Structural stiffness and mass matrices are formulated similar to SAP-IV. Solution is obtained in the time domain by calculating the dynamic response mode by mode. Time integration is performed by using Newmark's β-method. Response spectrum solution is also available as an option.

Program Version and Computer

The DYSEA version now operating on the Honeywell 6000 computer of GE, Nuclear Energy Systems Division, was developed at GE by modifying the SAP-IV program. Capability was added to handle the hydrodynamic mass effect due to fluid-structure interaction in the reactor. The program can handle three-dimensional dynamic problems with beam, trusses, and springs. Both acceleration time histories and response spectra may be used as input.

History of Use

The DYSEA program was developed in the summer of 1976. It has been adopted as a standard production program since 1977 and it has been used extensively in all dynamic and seismic analysis of the RPV and internals/building systems.

Extent of Application

The current version of DYSEA has been used in all dynamic and seismic analysis since its development. Results from test problems were found to be in close agreement with those obtained from either verified programs or analytic solutions.

Test Problems

Problem 1:

The first test problem involves finding the eigenvalues and eigenvectors from the following characteristic equation:

$$(\omega^2 [M]-[K]) \{x\} = 0$$

where ω is the circular frequency, x is the eigenvector, and [K] and [M] are the stiffness and the mass matrices given by:

$$[M] = \begin{bmatrix} 1 - \frac{4}{\pi^2} & \frac{4}{\pi^2} & -\frac{4}{q\pi^2} \\ & 1 - \frac{4}{q\pi^2} & \frac{4}{\dot{\pi}^2} \\ \text{Symmetric} & & 1 - \frac{4}{25\pi^2} \end{bmatrix} \quad (6.A\text{-}7)$$

$$[K] = \begin{bmatrix} 1 + \frac{\pi^2}{4} & 3 & \frac{5}{q} \\ & 1 + \frac{g\pi^2}{4} & 15 \\ \text{Symmetric} & & 1 + \frac{25\pi^2}{4} \end{bmatrix} \quad (6.A\text{-}8)$$

The analytical solution and the solution from DYSEA are:

a) Eigenvalues ω_i:

i	DYSEA SOLUTION	ANALYTIC SOLUTION
1	5.7835	5.7837
2	30.4889	30.4878
3	75.0493	75.0751

b) Eigenvectors ϕ_i:

1. DYSEA SOLUTION — ANALYTIC SOLUTION

$$0.0319 \quad \begin{bmatrix} 1.000 & 1.000 & 1.000 \\ -0.0319 & -1.5536 & -1.2105 \\ -0.0072 & -0.0666 & 2.0271 \end{bmatrix} \qquad \begin{bmatrix} 1.000 & 1.000 & 1.000 \\ -0.0319 & -1.554 & -1.211 \\ -0.0072 & -0.0666 & 2.027 \end{bmatrix}$$

Problem 2:

The second test problem compares the dynamic responses of the reactor pressure vessel, internals and reactor building subjected to earthquake ground motion.

The mathematical model of the reactor pressure vessel, internals and reactor building is given in Figure B-1. The inputs in the form of ground spectra are applied at the basement level. Response spectrum analysis was used in the analysis.

Natural frequencies of the system and the maximum responses at key locations have been calculated by both DYSEA and SAMIS. Result comparison are given in B-1 and B-2. It can be seen that the results calculated by DYSEA agree closely with those obtained by SAMIS.

6.A.4 PRESSURE TO FORCE CONVERSION

The RELAP4 pressure distribution output is converted to equivalent forces which are input into the DYSEA01 computer program. Each pressure is represented by a force acting normal to the RPV or shield wall at the center of the given pressure surface area. These forces are then converted into resulting forces (x component) acting on the respective DYSEA01 RPV beam nodal points. Mathematically, this is described as:

$$F_R = PA \cos \theta$$

where:

F_R = resultant force (lb),

P = RELAP4 node pressure (psia),

A = RELAP4 node surface area (in^2), and

θ = Component angle.

The results of these calculations are summarized in Table 6.A-4.

As an example, the pressure to force conversion at DYSEA01 node points 31 and 32 is shown below:

Time = 0.0800 seconds

NODE	ELEV (inches)	PRESSURE (lb/in^2)	AREA* (in^2)	ANGLE (degrees)	FORCE (lb)
6	1089.14	43.61	5828.44	15	245516
7	1089.14	35.34	5828.44	45	145660
8	1089.14	39.24	5828.44	75	59188
9	1089.14	41.40	8617.79	112.5	-136539
10	1089.14	39.99	8617.79	157.5	-318367
					- 4543

*See Table 6.A-8

For 360°, the resultant force is 2 times 4543 lb or an inward (positive) force of 9086 lb.

Since DYSEA nodal points 31 and 32 are at Elevations 1065.2 inches and 1125.7 inches respectively, the RELAP4 pressure/force at Elevation 1089.14 inches is distributed accordingly.
Consequently:

$$F_{31} = \frac{1125.7 - 1089.14}{1125.7 - 1065.2} (9086) = 5491 \text{ lb, and}$$

$$F_{32} = \frac{1065.2 - 1089.14}{1065.2 - 1125.7} (9086) = 3595 \text{ lb.}$$

These values can be compared to the computer-calculated DYSEA01 results, which are 5832.6 lb and 3252.7 lb respectively (Reference 1).

In the matrix displacement method of structural analysis, externally applied nodal forces and moments are required to produce nodal displacements equivalent to those that would be produced by forces or pressures applied between nodes. GE

considers the external moment effects for LaSalle AP to be negligible because of the close nodal spacing of the LaSalle RPV mathematical model.

6.A.5 SACRIFICIAL SHIELD, ANNULUS PRESSURIZATION, AND RPV LOADING DATA

This subsection provides a brief description of the analyses performed and the nodalization schemes, force constants, and load centers for the recirculation and feedwater line breaks. These data are used as input to the pressure to force conversion calculation.

The pressure responses of the RPV-sacrificial shield wall annulus to postulated pipe breaks at the RPV nozzle safe-end to pipe weld in a recirculation outlet line and a feedwater line were investigated using the RELAP4 computer code. Throughout the analyses the following assumptions were made:

a. RPV thermal insulation displaces to the shield wall while retaining its original volume and leaving its support structure in place.

b. Insulation above the shield wall yields to elevated pressures and blows out into the drywell allowing venting of annulus at the summit of the shield wall.

c. sacrificial shield penetration doors remain closed, allowing for limited venting of the annulus through all nozzle penetrations.

The nodalization schemes for both studies remain consistent with the guidelines cited above, with the exception of the region directly above the break, where it was anticipated that a finer mesh would be necessary to properly account for the highly localized pressure gradients expected there (see Figures 6.A-16 and 6.A-17). The final nodalization was determined on the basis of available sensitivity studies for similar analyses.

The mass and energy release rates were derived with the methods outlined in Subsection 6.A.2. The blowdown rates for the recirculation outlet line break analysis account for actual pipe displacement, while those for the feedwater line reflect an assumption of instantaneous pipe displacement (see RELAP4 input listings, Tables 6.A-6 and 6.A-7).

The specific loading data compiled for the NSSS adequacy evaluation for postulated pipe breaks within the annulus consists of the time-pressure history (psia) and two time-force (lbf) histories for each node within the annulus. The latter two histories represent integrated forces acting through the center of each node on the RPV and the sacrificial shield wall respectively. The time-force histories were generated by

taking the product of the node pressure and a predetermined constant, or η_{ss}, which accounts for the curved surface of the RPV and the sacrificial shield respectively (see Tables 6.A-8 and 6.A-9). The two loading histories, one for the RPV and one for the shield wall, are defined below.

$$F_{v_i} = \int_{-\Delta\theta/2}^{+\Delta\theta/2} P_i \ \ell_i \, R_V \cos\ 0 d\theta \ - \xi \sum_j P_i \left\{ \frac{\pi D_{p_j}^2}{4} \right\} \qquad (6.A\text{-}9)$$

$$= P_i \, 2 \ell_i \, R_V \ \sin\ (\Delta\theta/2) \ - P_i \ \xi \sum_j \left\{ \frac{\pi D_{p_j}^2}{4} \right\}$$

$$= P_i \ \eta_v$$

Where:

F_{v_i}	≡	nodal resultant force on RPV (lbf),
P_i	≡	node absolute pressure (psia),
ℓ_i		node height (inches),
R_V	≡	RPV radius (inches),
$\Delta\theta$	≡	azimuthal width of node (degrees), and
D_{p_j}	≡	pipe OD (in.).

(6.A-10)

$$F_{ss_i} = \int_{-\Delta\theta/2}^{+\Delta\theta/2} P_i \ \ell_i \ R_{ss} \cos \theta \, d\theta \ - \xi \sum_j P_i \left\{ \frac{\pi D^2 ss_j}{4} \right\}$$

$$= P_i \ 2\ell_i \ R_{ss} \ \sin \left(\Delta\theta/2\right) - P_{i\,u} \ \xi \sum_j \left\{ \frac{\pi D^2 ss_j}{4} \right\}$$

$$= P_i \ \eta_{ss}$$

Where:

F_{ss_i} ≡ nodal resultant force on shield wall (lbf),

P_i ≡ node absolute pressure (psia),

ℓ_i ≡ node height (inches),

R_{ss} ≡ shield wall inner radius (inches),

$\Delta\theta$ ≡ azimuthal width of node (degrees),

D_{ss_j} ≡ penetration ID (inches), and

ξ ≡ proportionality factor $= \dfrac{\sin \dfrac{(\Delta\theta)}{2}}{\dfrac{\Delta\theta}{2}} \left\{ \dfrac{360}{2\pi} \right\}$

6.A.6 <u>JET LOAD FORCES</u>

This subsection provides the jet load forces which result from pipe separation during the postulated accident. The pipe whip schematic is shown in Figure 6.A-7, and the resulting loads are listed in Table 6.A-1.

These loads are applied to the appropriate nodal points for input to the DYSEA01 computer program. The DYSEA01 program input is provided in Table 6.A-10.

6.A.7 RECIRCULATION AND FEEDWATER LINE BREAK FORCING FUNCTION

The time force histories provided in Reference 1 are those values converted from the time-pressure histories which were calculated with the RELAP4 computer program. These time forces histories are used as input to the DYSEA01 computer program.

6.A.8 REFERENCES

1. Calculation NSLD 3C7-0477-002, Sacrificial Shield Annulus Pressurization and Reactor Pressure Vessel Loading Data for General Electric NSSS Adequacy Evaluation, Rev. 000A.

TABLE 6.A-1
(SHEET 1 OF 5)

TIME HISTORY FOR POSTULATED RECIRCULATION

SUCTION PIPE RUPTURE*, **

Time (sec)	Pipe Displ. At Restraint (in.)	Pipe Velocity At Restraint (ft/sec)	Pipe Acceler. At Restraint (ft/sec^2)	Rel. Displ. Of End (in.)	Total Displ. Of End (in.)	Restr. Load Comp. PD1 (lb)	Restr. Load Comp. PD2 (lb)	Blowdown Force (lb)
0.00153	4.147E-02	3.547E 00	1.679E 03	0.	4.648E-02	0.	0.	346919.
0.00233	8.294E-02	4.889E 00	1.655E 03	0.	9.295E-02	0.	0.	346919.
0.00297	1.244E-01	5.932E 00	1.645E 03	0.	1.394E-01	0.	0.	346919.
0.00351	1.659E-01	6.816E 00	1.640E 03	0.	1.859E-01	0.	0.	346919.
0.00398	2.074E-01	7.597E 00	1.635E 03	0.	2.324E-01	0.	0.	346919.
0.00441	2.488E-01	8.304E 00	1.632E 03	0.	2.789E-01	0.	0.	346919.
0.00481	2.903E-01	8.955E 00	1.630E 03	0.	3.253E-01	0.	0.	346919.
0.00519	3.318E-01	9.561E 00	1.628E 03	0.	3.718E-01	0.	0.	346919.
0.00554	3.732E-01	1.013E 01	1.626E 03	0.	4.183E-01	0.	0.	346919.
0.00587	4.147E-01	1.067E 01	1.624E 03	0.	4.648E-01	0.	0.	346919.
0.00687	5.427E-01	1.077E 01	3.194E 02	2.689E-02	6.351E-01	50588.	0.	346919.
0.00787	6.742E-01	1.117E 01	4.350E 02	9.147E-02	8.471E-01	108204.	0.	346919.
0.00887	8.108E-01	1.159E 01	3.863E 02	1.808E-01	1.089E 00	168037.	0.	346919.
0.00987	9.519E-01	1.190E 01	2.419E 02	2.875E-01	1.354E 00	229892.	0.	346919.
0.01087	1.096E 00	1.203E 01	3.532E 01	4.076E-01	1.636E 00	293042.	0.	346919.
0.01187	1.240E 00	1.194E 01	-2.099E 02	5.388E-01	1.928E 00	356421.	0.	346919.

* Output parameters are listed at the end of this table.

** Except for the restraint load components PD1 and PD2, all variables below are in a direction parallel to the blowdown force.

TABLE 6.A-1
(SHEET 2 OF 5)

Time (sec)	Pipe Displ. At Restraint (in.)	Pipe Velocity At Restraint (ft/sec)	Pipe Acceler. At Restraint (ft/sec^2)	Rel. Displ. Of End (in.)	Total Displ. Of End (in.)	Restr. Load Comp. PD1 (lb)	Restr. Load Comp. PD2 (lb)	Blowdown Force (lb)
0.01287	1.381E 00	1.158E 01	-4.744E 02	6.802E-01	2.228E 00	418752.	0.	346919.
0.01387	1.517E 00	1.096E 01	-7.414E 02	8.316E-01	2.531E 00	478650.	0.	346919.
0.01487	1.643E 00	1.007E 01	-1.027E 03	9.934E-01	2.835E 00	538908.	0.	346919.
0.01587	1.757E 00	8.948E 00	-1.197E 03	1.166E 00	3.136E 00	581800.	0.	346919.
0.01687	1.857E 00	7.672E 00	-1.335E 03	1.350E 00	3.431E 00	618871.	0.	346919.
0.01787	1.941E 00	6.278E 00	-1.438E 03	1.543E 00	3.719E 00	649762.	0.	346919.
0.01887	2.008E 00	4.801E 00	-1.504E 03	1.746E 00	3.996E 00	674226.	0.	346919.
0.01987	2.056E 00	3.279E 00	-1.531E 03	1.956E 00	4.261E 00	692131.	0.	346919.
0.02087	2.086E 00	1.751E 00	-1.519E 03	2.172E 00	4.510E 00	703465.	0.	346919.
0.02187	2.098E 00	2.567E-01	-1.469E 03	2.392E 00	4.744E 00	708338.	0.	346919.
0.02222	2.098E 00	0.	0.	2.470E 00	4.822E 00	708572.	0.	346919.
0.02242	2.098E 00	0.	0.	2.513E 00	4.865E 00	708572.	0.	346919.
0.02262	2.098E 00	0.	0.	2.555E 00	4.907E 00	708572.	0.	346919.
0.02283	2.098E 00	0.	0.	2.598E 00	4.950E 00	708572.	0.	346919.
0.02304	2.098E 00	0.	0.	2.640E 00	4.992E 00	708572.	0.	346919.
0.02325	2.098E 00	0.	0.	2.683E 00	5.035E 00	708572.	0.	346919.
0.02347	2.098E 00	0.	0.	2.725E 00	5.077E 00	708572.	0.	346919.
0.02370	2.098E 00	0.	0.	2.768E 00	5.120E 00	708572.	0.	346919.
0.02393	2.098E 00	0.	0.	2.810E 00	5.162E 00	708572.	0.	346919.

TABLE 6.A-1
(SHEET 3 OF 5)

Time (sec)	Pipe Displ. At Restraint (in.)	Pipe Velocity At Restraint (ft/sec)	Pipe Acceler. At Restraint (ft/sec^2)	Rel. Displ. Of End (in.)	Total Displ. Of End (in.)	Restr. Load Comp. PD1 (lb)	Restr. Load Comp. PD2 (lb)	Blowdown Force (lb)
0.02417	2.098E 00	0.	0.	2.853E 00	5.2O5E 00	708572.	0.	346919.
0.02442	2.098E 00	0.	0.	2.895E 00	5.247E 00	708572.	0.	346919.
0.02467	2.098E 00	0.	0.	2.938E 00	5.290E 00	708572.	0.	346919.
0.02494	2.098E 00	0.	0.	2.980E 00	5.332E 00	708572.	0.	346919.
0.02522	2.098E 00	0.	0.	3.023E 00	5.375E 00	708572.	0.	346919.
0.02551	2.098E 00	0.	0.	3.065E 00	5.417E 00	708572.	0.	346919.
0.02582	2.098E 00	0.	0.	3.108E 00	5.460E 00	708572.	0.	346919.
0.02614	2.098E 00	0.	0.	3.150E 00	5.502E 00	708572.	0.	3469l9.
0.02649	2.098E 00	0.	0.	3.193E 00	5.545E 00	708572.	0.	346919.
0.02687	2.098E 00	0.	0.	3.235E 00	5.587E 00	708572.	0.	346919.
0.02728	2.098E 00	0.	0.	3.278E 00	5.630E 00	708572.	0.	346919.
0.02774	2.098E 00	0.	0.	3.320E 00	5.672E 00	708572.	0.	346919.
0.02827	2.098E 00	0.	0.	3.363E 00	5.715E 00	708572.	0.	346919.
0.02893	2.098E 00	0.	0.	3.405E 00	5.757E 00	708572.	0.	346919.
0.02992	2.098E 00	0.	0.	3.448E 00	5.800E 00	708572.	0.	346919.

TABLE 6.A-1
(SHEET 4 OF 5)

Output Parameters Summary

Effective clearance (inches) 0.415	Length from restraint to break (ft) 3.542	Restraint loading direction 0 degrees
Pipe bending strain limit (in/in) 9.054E-02	Pipe rotation stability limit (degr.) 7.7815	Max. allowable bending moment (ft-lbs) 1417307

Impact Velocity = 10.67 ft/sec   Impact Time = 0.0059 seconds

Number of bars composing the restraint 2	Defl. of struc. in direction of thrust (in.) 0.7086	Defl. of restr. in direction of thrust (in.) 0.9754
Force on restr. in direction of thrust (lb) 708572	Force on struc. in direction of thrust (lb) 708572	Time at peak dynamic load (seconds) 0.0221
Total energy absorbed by the restraint (ft-lb) 30522	`Energy absorbed by the structure (ft-lb) 20920	Energy absorbed by the bottom hinge (ft-lb) 1956

Energy absorbed by the top top hinge (ft-lb)	Restraint load (peak) components (lb)		Restraint load (static) components (lb)	
	PD1	PD2	PS1	PS2
0.	708572	0.	138258	0.

TABLE 6.A-1
(SHEET 5 OF 5)

Relative defl. of pipe end in the direction of the thrust (in.)	Total defl. of the pipe end
3.4649	5.8168

Defl. time for pipe end (seconds after impact)	Total time of movement
0.0250	0 0309

Energy absorbed by the restraint hinge (ft-lb)	Total absorbed energy (ft-lb)
115445	168843

Pipe defl. at restraint components (in.)		Pipe defl. at the break components (in.)	
XR1	XR2	XP1	XP2
2.0986	0.	5.8168	0.

TABLE 6.A-2

ACOUSTIC LOADING ON REACTOR PRESSURE VESSEL SHROUD

TIME (msec)	ACOUSTIC LOAD (kips)
0.0	0
0.7	0
5.9	1076
10.2	2133
16	50

TABLE 6.A-3
(SHEET 1 OF 2)

RPV COORDINATES OF NODAL POINTS

NODE NUMBER	NODAL COORDINATES X-ORDINATE	Y- ORDINATE	Z-ORDINATE
1	-912.000	774.000	1563.000
2	-912.000	774.000	1556.000
3	-912.000	774.000	981.200
4	-912.000	774.000	740.000
5	-912.000	774.000	1356.000
6	-912.000	774.000	1316.000
7	-912.000	774.000	1279.200
8	-912.000	774.000	1240.400
9	-912.000	774.000	1201.600
10	-912.000	774.000	1163.600
11	-912.000	774.000	1141.700
12	-912.000	774.000	1125.700
13	-912.000	774.000	1065.200
14	-912.000	774.000	1035.200
15	-912.000	774.000	1021.300
16	-912.000	774.000	994.200
17	-912.000	774.000	1601.700
18	-912.000	774.000	1559.700
19	-912.000	774.000	1499.700
20	-912.000	774.000	1436.900
21	-912.000	774.000	1398.500
22	-912.000	774.000	1318.000
23	-912.000	774.000	1279.200
24	-912.000	774.000	1240.400
25	-912.000	774.000	1201.600
26	-912.000	774.000	1163.600
27	-912.000	774.000	1141.700
28	-912.000	774.000	1125.700
29	-912.000	774.000	1021.300
30	-912.000	774.000	1035.200
31	-912.000	774.000	1065.200
32	-912.000	774.000	1125.700
33	-912.000	774.000	1141.700
34	-912.000	774.000	1163.600
35	-912.000	774.000	1201.600
36	-912.000	774.000	1240.400
37	-912.000	774.000	1279.200
38	-912.000	774.000	1318.000
39	-912.000	774.000	1356.600
40	-912.000	774.000	1398.500
41	-912.000	774.000	1436.900
42	-912.000	774.000	1499.700
43	-912.000	774.000	1559.700
44	-912.000	774.000	1563.600

TABLE 6.A-3
(SHEET 2 OF 2)

NODAL COORDINATES

NODE NUMBER	X-ORDINATE	Y- ORDINATE	Z-ORDINATE
45	-912.000	774.000	1601.700
46	-912.000	774.000	1619.800
47	-912.000	774.000	1724.200
48	-912.000	774.000	1743.600
49	-912.000	774.000	1768.200
50	-912.000	774.000	1817.100
51	-912.000	774.000	1866.000
52	-912.000	774.000	1563.000
53	300.000	774.000	886.000
54	-912.000	774.000	446.000
55	-912.000	774.000	318.000
56	-912.000	774.000	0.
57	-912.000	774.000	740.000

TABLE 6.A-4
MAXIMUM MEMBER FORCES DUE TO ANNULUS PRESSURIZATION

COMPONENT DESCRIPTION	ELEMENT NUMBER	FEEDWATER	RECIRC.	JET REACTION
Top guide (L)*	4	22.20	38.00	29.0
Core plate (L)	7	20.80	42.00	30.0
Fuel support (L)	8	19.00	69.00	74.0
CRD housing (L)		9.10	22.00	70.0
CRD housing (M)		.24	.56	1.9
Shroud head (L)	19	59.80	78.00	133.0
Shroud head (M)	19	6.40	5.90	6.1
Shroud support (L)	26	184.00	296.00	246.0
Shroud support (M)	26	19.80	40.00	22.0
Vessel skirt (L)	50	1220.00	3204.00	1858.0
Vessel skirt (M)	50	216.00	221.00	130.0
Pedestal cont. (L)	3	486.00	2325.00	859.0
Pedestal cont. (M)	3	326.00	680.00	206.0
Stabilizer (L)	III	1722.00	1949.00	746.0
CRD support beam (L)		4.50	27.00	50.0

---

* *(L) Load - 10^3 x lb
(M) Moment - 10^6 x in. x lb
All loads incorporate appropriate factor to account for shell behavior

TABLE 6.A-5

MAXIMUM ACCELERATION* DUE TO ANNULUS PRESSURIZATION

(in./sec²)

COMPONENT DESCRIPTION	NODE NUMBER	FEEDWATER	RECIRC. LINE BREAK	JET LOAD
–P line	9	80	283	675
CRD guide tube	11	86	298	309
Separators	17	155	306	342
Head spray	51	178	416	898
Steam dryer	46	118	200	451
Feedwater sparger	43	109	157	538
Jet pump	38	133	362	406
RPV	30	62	253	514
RPV (bottom)	16	61	254	598
Shield wall	2	282	398	229
Top of shield wall	1	190	326	254
Fuel	5	74	198	394
Fuel	7	27	51	77
Fuel	9	80	283	675

- *All accelerations incorporate a factor to account for shell behavior.

TABLE 6.A-6
(SHEET 1 OF 3)

RELAP 4 INPUT DATA, RECIRCULATION LINE OUTLET BREAK

```
= LASALLE RPV-SHIELD ANNULUS PRESSURIZATION STUDY – NSLD CALC NO 3C7-0976-001
* PROJECT NO 4266-00  R.M. HOGAN – D.L. ROBINSON – NUCLEAR ANALYSTS
* RECIRCULATION OUTLET LINE BREAK
  *
* CASE "A" BASE LISTING 12/27/76
  *
*234567890123456789012345789012345789012345789012345789012345789012345789
0
* PROBLEM DIMENSIONS
* CARD  LDMP-NEDI-NTC—NTRP-NVOL-NBUB-NTDV-NJUN-NONE-NFLL-NONE
 010001 -2  0    3    6    38   0    0    86   0    4    0    1    0    0  0  0  0
  *
 *PROBLEM CONSTANTS
 010002    0.0     1.0
  *
* TIME STEPS
  030010      1    1    10   0    0.0001   1E-06   0.025
  030020      1    1    5    0    0.001    1E-06   0.2
  030030      1    1    1    0    0.01     1E-06   1.0
 *
 * TRIP CONTROLS
  040010      1    1    0    0    0.2   0.0   *END PROBLEM ON ELAPSED TIME
  040020      2    1    0    0    0.0   0.0   * ACTION #2 ON ELAPSED TIME (FILL)
  040030      3    4    30   36   3.0   0.0   * ACTION #3 ON DP (OPEN VALVE)
  040040      4    4    31   36   3.0   0.0   * ACTION #4 ON DP (OPEN VALVE)
  040050      5    4    32   36   3.0   0.0   * ACTION #5 ON DP (OPEN VALVE)
 040060       6    4    33   36   3.0   0.0   * ACTION #6 ON DP (OPEN VALVE)
 *
 * BEGIN VOLUME DATA
* 234567890123456789012345678901234567890123456789012345678901234567890
VOLUME  B  R  PRESS  TEMP  QUAL   VOLUME  MT    MIX   TP  FLOWA  DIAMV  ELEV
050011  0  0  15.45  -1.   0.946  100.6   5.07  5.07  0   18.40  0.0    755.29
050021  0  0  15.45  -1.   0.946  100.6   5.07  5.07  0   18.40  0.0    755.29
050031  0  0  15.45  -1.   0.946  100.6   5.07  5.07  0   18.40  0.0    755.29
050041  0  0  15.45  -1.   0.946  150.9   5.07  5.07  0   23.36  0.0    755.29
050051  0  0  15.45  -1.   0.946  150.9   5.07  5.07  0   23.36  0.0    755.29
050061  0  0  15.45  -1.   0.946  121.0   7.47  7.47  0   20.98  0.0    760.36
050071  0  0  15.45  -1.   0.946  121.0   7.47  7.47  0   20.98  0.0    760.36
050081  0  0  15.45  -1.   0.946  121.0   7.47  7.47  0   20.98  0.0    760.36
050091  0  0  15.45  -1.   0.946  181.5   7.47  7.47  0   25.64  0.0    760.36
050101  0  0  15.45  -1.   0.946  181.5   7.47  7.47  0   25.64  0.0    760.36
050111  0  0  15.45  -1.   0.946  39.87   6.92  6.92  0   10.02  0.0    767.83
050121  0  0  15.45  -1.   0.946  54.28   4.90  4.90  0   10.50  0.0    767.83
050131  0  0  15.45  -1.   0.946  61.94   4.90  4.90  0   10.50  0.0    767.83
050141  0  0  15.45  -1.   0.946  81.43   4.90  4.90  0   13.47  0.0    767.83
050151  0  0  15.45  -1.   0.946  80.54   4.90  4.90  0   13.47  0.0    767.83
050161  0  0  15.45  -1.   0.946  26.77   2.67  2.67  0   8.43   0.0    774.75
050171  0  0  15.45  -1.   0.946  52.18   4.69  4.69  0   10.30  0.0    772.73
050181  0  0  15.45  -1.   0.946  52.18   4.69  4.69  0   10.30  0.0    772.73
050191  0  0  15.45  -1.   0.946  78.28   4.69  4.69  0   13.27  0.0    772.73
050201  0  0  15.45  -1.   0.946  77.39   4.69  4.69  0   13.27  0.0    773.73
050211  0  0  15.45  -1.   0.946  67.48   6.41  6.41  0   12.44  0.0    777.42
050221  0  0  15.45  -1.   0.946  67.48   6.41  6.41  0   12.44  0.0    777.42
050231  0  0  15.45  -1.   0.946  67.48   6.41  6.41  0   12.44  0.0    777.42
050241  0  0  15.45  -1.   0.946  101.2   6.41  6.41  0   15.52  0.0    777.42
050251  0  0  15.45  -1.   0.946  101.2   6.41  6.41  0   15.52  0.0    777.42
050261  0  0  15.45  -1.   0.946  171.1   9.59  9.59  0   18.61  0.0    783.83
050271  0  0  15.45  -1.   0.946  155.8   9.59  9.59  0   18.61  0.0    783.83
050281  0  0  15.45  -1.   0.946  155.8   9.59  9.59  0   18.61  0.0    783.83
050291  0  0  15.45  -1.   0.946  171.1   9.59  9.59  0   18.61  0.0    783.83
050301  0  0  15.45  -1.   0.946  155.8   8.81  8.81  0   17.86  0.0    793.42
050311  0  0  15.45  -1.   0.946  153.4   8.81  8.81  0   17.86  0.0    793.42
050321  0  0  15.45  -1.   0.946  143.9   8.81  8.81  0   17.86  0.0    793.42
050331  0  0  15.45  -1.   0.946  164.1   8.81  8.81  0   17.86  0.0    793.42
050341  0  0  15.45  -1.   0.946  19.76   6.92  6.92  0   10.02  0.0    767.83
050351  0  0  15.45  -1.   0.946  19.52   4.92  4.92  0   7.04   0.0    769.56
050361  0  0  15.45  -1.   0.557  16315.  41.0  41.0  0   400.   0.0    793.42
050371  0  0  15.45  -1.   0.557  11665.  12.1  12.1  0   965.   0.0    781.32
050381  0  0  15.45  -1.   0.557  82775.  44.7  44.7  0   1850.  0.0    736.62
VOLUME  B  R  PRESS  TEMP  QUAL   VOLUME  MT    MIX   TP  FLOWA  DIAMV  ELEV
* 234567890123456789012345678901234567890123456789012345678901234567890
* END VOLUME DATA
*
* BEGIN HORIZONTAL FLOW PATHS WITHIN S.S. ANNULUS
* 234567890123456789012345678901234567890123456789012345678901234567890
```

TABLE 6.A-6
(SHEET 2 OF 3)

RELAP 4 INPUT DATA, RECIRCULATION LINE OUTLET BREAK

JUNCT	IN	OT	P	V	FLO	AJUN	ZJUN	IN	FJUF	FJUR	V	C	1	EQ	DM	CC	C	E
080011	1	2	0	0	0.0	14.86	757.82	0.40	0.24	0.00	0	0	0	0	0.0	0.6	1	0
080021	2	3	0	0	0.0	14.86	757.82	0.40	0.24	0.00	0	0	0	0	0.0	0.6	1	0
080031	3	4	0	0	0.0	14.86	757.82	0.50	0.40	0.00	0	0	0	0	0.0	0.6	1	0
080041	4	5	0	0	0.0	14.86	757.82	0.60	0.42	0.00	0	0	0	0	0.0	0.6	1	0
080051	6	7	0	0	0.0	20.19	764.10	0.30	0.22	0.00	0	0	0	0	0.0	0.6	1	0
080061	7	8	0	0	0.0	20.19	764.10	0.30	0.22	0.00	0	0	0	0	0.0	0.6	1	0
080071	8	9	0	0	0.0	20.19	764.10	0.38	0.39	0.00	0	0	0	0	0.0	0.6	1	0
080081	9	10	0	0	0.0	20.19	764.10	0.45	0.41	0.00	0	0	0	3	0.0	0.6	1	0
080091	35	34	0	0	0.0	7.04	772.02	0.30	0.85	0.00	0	0	0	0	0.0	0.6	1	0
080101	34	11	0	0	0.0	10.02	771.29	0.32	0.35	0.00	0	0	0	0	0.0	0.6	1	0
080111	11	12	0	0	0.0	7.47	770.28	0.64	0.56	0.00	0	0	0	3	0.0	0.6	1	0
080121	12	13	0	0	0.0	7.09	770.28	0.90	0.84	0.00	0	0	0	0	0.0	0.6	1	0
080131	13	14	0	0	0.0	7.09	770.28	1.13	0.85	0.00	0	0	0	3	0.0	0.6	1	0
080141	14	15	0	0	0.0	7.09	770.28	1.35	1.64	0.00	0	0	0	3	0.0	0.6	1	0
080151	11	17	0	0	0.0	2.11	773.74	2.26	0.05	0.00	0	0	0	0	0.0	0.6	1	0
080161	16	17	0	0	0.0	3.87	776.09	1.46	0.38	0.00	0	0	0	3	0.0	0.6	1	0
080171	17	18	0	0	0.0	6.79	775.07	0.94	0.83	0.00	0	0	0	3	0.0	0.6	1	0
080181	18	19	0	0	0.0	6.79	775.07	1.17	0.85	0.00	0	0	0	3	0.0	0.6	1	0
080191	19	20	0	0	0.0	6.79	775.07	1.41	1.63	0.00	0	0	0	3	0.0	0.6	1	0
080201	21	22	0	0	0.0	9.83	780.62	0.65	0.36	0.00	0	0	0	3	0.0	0.6	1	0
080211	22	23	0	0	0.0	9.83	780.62	0.65	0.36	0.00	0	0	0	3	0.0	0.6	1	0
080221	23	24	0	0	0.0	9.83	780.62	0.81	0.67	0.00	0	0	0	3	0.0	0.6	1	0
080231	24	25	0	0	0.0	9.83	780.62	0.97	0.68	0.00	0	0	0	3	0.0	0.6	1	0
080241	26	27	0	0	0.0	14.68	788.62	0.65	1.28	0.00	0	0	0	3	0.0	0.6	1	0
080251	27	28	0	0	0.0	14.68	788.62	0.65	0.68	0.00	0	0	0	3	0.0	0.6	1	0
080261	28	29	0	0	0.0	14.68	788.62	0.65	1.28	0.00	0	0	0	3	0.0	0.6	1	0
080271	30	31	0	0	0.0	13.49	797.83	0.71	1.27	0.00	0	0	0	3	0.0	0.6	1	0
080281	31	32	0	0	0.0	13.49	797.83	0.71	1.13	0.00	0	0	0	3	0.0	0.6	1	0
080291	32	33	0	0	0.0	13.49	797.83	0.71	1.27	0.00	0	0	0	3	0.0	0.6	1	0
JUNCT	IN	OT	P	V	FLO	AJUN	ZJUN	IN	FJUF	FJUR	V	C	1	EQ	DM	CC	C	E
* 234567890123456789012345678901234567890123456789012345678901234567890																		
* END HORIZONTAL FLOW PATHS WITHIN 5.5. ANNULUS																		
*																		
* BEGIN VERTICAL FLOW PATHS WITHIN S.S. ANNULUS																		
* 234567890123456789012345678901234567890123456789012345678901234567890																		
JUNCT	IN	OT	P	V	FLO	AJUN	ZJUN	IN	FJUF	FJUR	V	C	I	EQ	DM	CC	C	E
080301	6	1	0	0	0.0	18.40	760.36	0.33	0.03	0.03	1	0	0	3	0.0	0.6	1	0
080311	7	2	0	0	0.0	18.40	760.36	0.33	0.03	0.03	1	0	0	3	0.0	0.6	1	0
080321	8	3	0	0	0.0	18.40	760.36	0.33	0.03	0.03	1	0	0	3	0.0	0.6	1	0
080331	9	4	0	0	0.0	23.36	760.36	0.22	0.03	0.03	1	0	0	3	0.0	0.6	1	0
080341	10	5	0	0	0.0	23.36	760.36	0.22	0.03	0.03	1	0	0	0	0.0	0.6	1	0
080351	34	6	0	0	0.0	3.61	767.83	1.40	1.13	0.90	1	0	0	3	0.0	0.6	1	0
080361	11	6	0	0	0.0	3.61	767.83	1.40	1.13	0.90	1	0	0	3	0.0	0.6	1	0
080371	12	7	0	0	0.0	7.22	767.83	0.62	1.13	0.90	1	0	0	3	0.0	0.6	1	0
080381	13	8	0	0	0.0	7.22	767.83	0.62	1.40	1.17	1	0	0	3	0.0	0.6	1	0
080391	14	9	0	0	0.0	10.84	767.83	0.41	1.13	0.90	1	0	0	0	0.0	0.6	1	0
080401	15	10	0	0	0.0	10.84	767.83	0.41	1.13	0.90	1	0	0	0	0.0	0.6	1	0
080411	12	17	0	0	0.0	8.56	772.73	0.56	0.46	0.00	1	0	0	3	0.0	0.6	1	0
080421	13	18	0	0	0.0	8.56	772.73	0.56	0.46	0.00	1	0	0	0	0.0	0.6	1	0
080431	14	19	0	0	0.0	14.50	772.73	0.33	0.59	0.00	1	0	0	0	0.0	0.6	1	0
080441	15	20	0	0	0.0	14.50	772.73	0.33	0.68	0.00	1	0	0	0	0.0	0.6	1	0
080451	34	16	0	0	0.0	5.94	774.75	0.94	0.03	0.00	1	0	0	3	0.0	0.6	1	0
080461	11	16	0	0	0.0	5.94	774.75	0.94	0.88	0.00	1	0	0	3	0.0	0.6	1	0
080471	16	21	0	0	0.0	7.72	777.42	0.44	0.67	0.00	1	0	0	0	0.0	0.6	1	0
080481	17	22	0	0	0.0	7.72	777.42	0.59	0.68	0.00	1	0	0	0	0.0	0.6	1	0
080491	18	23	0	0	0.0	7.72	777.42	0.59	0.68	0.00	1	0	0	0	0.0	0.6	1	0
080501	19	24	0	0	0.0	11.57	777.42	0.40	0.68	0.00	1	0	0	0	0.0	0.6	1	0
080511	20	25	0	0	0.0	11.57	777.42	0.40	0.68	0.00	1	0	0	0	0.0	0.6	1	0
080521	21	26	0	0	0.0	7.72	783.83	0.80	0.96	0.00	1	0	0	0	0.0	0.6	1	0
080531	22	26	0	0	0.0	3.86	783.83	1.60	1.04	0.00	1	0	0	3	0.0	0.6	1	0
080541	22	27	0	0	0.0	3.86	783.83	1.60	1.04	0.00	1	0	0	0	0.0	0.6	1	0
080551	23	27	0	0	0.0	7.72	783.83	0.80	0.69	0.00	1	0	0	3	0.0	0.6	1	0
080561	24	28	0	0	0.0	11.57	783.83	0.54	0.96	0.00	1	0	0	0	0.0	0.6	1	0
080571	25	29	0	0	0.0	11.57	783.83	0.54	0.97	0.00	1	0	0	0	0.0	0.6	1	0
080581	26	30	0	0	0.0	11.57	793.42	0.60	1.00	0.00	1	0	0	0	0.0	0.6	1	0
080591	27	31	0	0	0.0	11.57	793.42	0.60	1.04	0.00	1	0	0	0	0.0	0.6	1	0
080601	28	32	0	0	0.0	11.57	793.42	0.60	0.97	0.00	1	0	0	0	0.0	0.6	1	0
080611	29	33	0	0	0.0	11.57	793.42	0.60	1.00	0.00	1	0	0	0	0.0	0.6	1	0
JUNCT	IN	OT	P	V	FLO	AJUN	ZJUN	IN	FJUF	FJUR	V	C	I	EQ	DM	CC	C	E
*234567890123456789012345678901234567890123456789012345678901234567890																		
*END VERTICAL FLOW PATHS WITHIN S.S. ANNULUS																		
*																		
* BEGIN FLOW PATHS TO CONTAINMENT – PENETRATIONS WITH SHIELDING DOORS																		
*234567890123456789012345678901234567890123456789012345678901234567890																		
JUNCT	IN	OT	P	V	FLO	AJUN	ZJUN	IN	FJUF	FJUR	V	C	I	EQ	DM	CC	C	E
080621	30	36	0	1	0.0	9.27	797.83	1.05	0.75	0.00	0	0	0	0	0.0	0.6	1	0
080631	31	36	0	2	0.0	13.90	797.83	0.70	1.69	0.00	0	0	0	0	0.0	0.6	1	0

TABLE 6.A-6
(SHEET 3 OF 3)

RELAP 4 INPUT DATA, RECIRCULATION LINE OUTLET BREAK

```
080641  32  36  0  3  0.0  13.90  797.83  0.70  1.69  0.00  0  0  0  0  0.0  0.6  1  0
080651  33  36  0  4  0.0  9.27   797.83  1.05  0.75  0.00  0  0  0  0  0.0  0.6  1  0
080661  33  36  0  0  0.0  2.04   797.83  1.05  1.72  0.00  0  0  0  3  0.0  0.6  1  0
080671  32  36  0  0  0.0  0.68   797.83  3.39  1.71  0.00  0  0  0  3  0.0  0.6  1  0
080681  31  36  0  0  0.0  2.10   797.83  1.11  1.71  0.00  0  0  0  3  0.0  0.6  1  0
080691  30  36  0  0  0.0  1.77   797.83  1.25  1.72  0.00  0  0  0  3  0.0  0.6  1  0
080701  36  37  0  0  0.0  400.   793.42  0.06  0.05  0.00  1  0  0  3  0.0  0.6  1  0
080711  29  37  0  0  0.0  1.39   788.62  1.50  1.73  0.00  0  0  0  3  0.0  0.6  1  0
080721  28  37  0  0  0.0  0.71   788.62  3.30  1.71  0.00  0  0  0  3  0.0  0.6  1  0
080731  27  37  0  0  0.0  0.71   788.62  3.30  1.71  0.00  0  0  0  3  0.0  0.6  1  0
080741  26  37  0  0  0.0  1.39   788.62  1.50  1.71  0.00  0  0  0  3  0.0  0.6  1  0
080751  37  38  0  0  0.0  965.   781.32  0.03  0.05  0.00  1  0  0  3  0.0  0.6  1  0
080761  20  38  0  0  0.0  1.25   775.07  1.97  1.71  0.00  0  0  0  3  0.0  0.6  1  0
080771  19  38  0  0  0.0  1.07   775.07  2.20  1.71  0.00  0  0  0  3  0.0  0.6  1  0
080781  18  38  0  0  0.0  0.71   775.07  3.30  1.71  0.00  0  0  0  3  0.0  0.6  1  0
080791  17  38  0  0  0.0  0.71   775.07  3.30  1.71  0.00  0  0  0  3  0.0  0.6  1  0
080801  15  38  0  0  0.0  1.25   770.28  1.97  1.71  0.00  0  0  0  3  0.0  0.6  1  0
080811  14  38  0  0  0.0  1.07   770.28  2.20  1.71  0.00  0  0  0  3  0.0  0.6  1  0
080821  13  38  0  0  0.0  1.47   770.28  1.50  1.71  0.00  0  0  0  3  0.0  0.6  1  0
080831  12  38  0  0  0.0  0.71   770.28  3.30  1.71  0.00  0  0  0  3  0.0  0.6  1  0
080841  11  38  0  0  0.0  0.71   772.02  3.30  1.71  0.00  0  0  0  3  0.0  0.6  1  0
080851  35  38  0  0  0.0  1.08   772.02  2.43  1.71  0.00  0  0  0  0  0.0  0.6  1  0
JUNCT   IN  OT  P  V  FLO  AJUN   ZJUN    IN    FJUF  FJUR  V  C  I  EQ DM   CC   C  E
*23456789012345678901234567890123456789012345678901234567890123456789012345678901234567890
*END FLOW PATHS TO CONTAINMENT – PENETRATIONS WITH SHIELDING DOORS
*
*BEGIN FILL PATH
*23456789012345678901234567890123456789012345678901234567890123456789012345678901234567890
JUNCT   IN  OT  P  V  FLO  AJUN   ZJUN    IN    FJUF  FJUR  V  C  I  EQ DM   CC   C  E
080861  0   35  1  0  0.0  1.00   772.02  0.00  0.00  0.00  0  0  0  3  0.0  1.0  1  0
JUNCT   IN  OT  P  V  FLO  AJUN   ZJUN    IN    FJUF  FJUR  V  C  I  EQ DM   CC   C  E
*23456789012345678901234567890123456789012345678901234567890123456789012345678901234567890
* END FILL PATH
*
* VALVE DATA CARDS
 110010  -3   0.0   0.0   0.0
 110020  -4   0.0   0.0   0.0
 110030  -5   0.0   0.0   0.0
 110040  -6   0.0   0.0   0.0
*
* FILL TABLE  DATA CARDS
* FILL CONTROL
 130100  16   2   0   0   1060.   533.
* CARD    TIME     FLOW     TIME      FLOW     TIME      FLOW
130101    0.0      0.0      0.002     371.     0.004     1194.
130102    0.006    2476.    0.008     4463.    0.010     7081.
130103    0.0173   18092.   0.019395  18092.   0.019405  9162.
130104    0.022    10573.   0.024     11445.   0.026     12147.
130105    0.028    12611.   0.030     12865.   0.031     12885.
130106    5.0      12885.
*
*23456789012345678901234567890123456789012345678901234567890123456789012345678901234567890
*****************************************************************************
* MODEL REVISIONS
*****************************************************************************
```

*

TABLE 6.A-7
(SHEET 1 OF 3)

RELAP 4 INPUT DATA, FEEDWATER LINE BREAK

```
= LASALLE RPV-SHIELD ANNULUS PRESSURIZATION STUDY – NSLD CALC NO 3C7-0976-001
* PROJECT NO 4266-00  R.M. HOGAN – D.L. ROBINSON – NUCLEAR ANALYSTS
* FEEDWATER LINE BREAK
  *
* CASE "C"  BASE LISTING 1/3/77
  *
*234567890123456789012345789012345789012345789012345789012345789012345789012345789012345789012345789
* PROBLEM DIMENSIONS
* CARD  LDMP----NEDI---------NTS--------NTRP---------NVOL------NBUB--------NTDV-------NJUN-------NONE--------NFLL---------NONE
 010001    -2      0       3       8       32      0       0       70      060      1       00000
  *
 *PROBLEM CONSTANTS
 010002            0.0     1.0
  *
* TIME STEPS
  030010           1       1       50      0       0.0001  1E-06   0.025
  030020           1       1       25      0       0.001   1E-06   0.2
  030030           1       1       1       0       0.01    1E-06   1.0
 *
 * TRIP CONTROLS
  040010           1       1       0       0       0.2     0.0     *END PROBLEM ON ELAPSED TIME
  040020           2       1       0       0       0.0     0.0     * ACTION #2 ON ELAPSED TIME (FILL)
  040030           3       4       23      30      3.0     0.0     * ACTION #3 ON DP (OPEN VALVE)
  040040           4       4       24      30      3.0     0.0     * ACTION #4 ON DP (OPEN VALVE)
  040050           5       4       25      30      3.0     0.0     * ACTION #5 ON DP (OPEN VALVE)
 040060            6       4       26      30      3.0     0.0     * ACTION #6 ON DP (OPEN VALVE)
040070        7       4       27      30      3.0     0.0     *ACTION #7 ON DP (OPEN VALVE)
040080        8       4       28      30      3.0     0.0     * ACTION #8 ON DP (OPEN VALVE)
*
* BEGIN VOLUME DATA
* 234567890123456789012345789012345789012345789012345789012345789012345789012345789012345789012345789
      VOLUME   B     R     PRESS   TEMP   QUAL    VOLUME   HT      MIX     TP    FLOWA   DIAMV   ELEV
      050011   0     0     15.45   -1.    0.946   150.9    5.07    5.07    0     23.36   0.0     755.29
      050021   0     0     15.45   -1.    0.946   150.9    5.07    5.07    0     23.36   0.0     755.29
      050031   0     0     15.45   -1.    0.946   150.9    5.07    5.07    0     23.36   0.0     755.29
      050041   0     0     15.45   -1.    0.946   150.9    5.07    5.07    0     23.36   0.0     755.29
      050051   0     0     15.45   -1.    0.946   181.5    7.47    7.47    0     23.80   0.0     760.36
      050061   0     0     15.45   -1.    0.946   181.5    7.47    7.47    0     23.80   0.0     760.36
      050071   0     0     15.45   -1.    0.946   181.5    7.47    7.47    0     23.80   0.0     760.36
      050081   0     0     15.45   -1.    0.946   181.5    7.47    7.47    0     23.80   0.0     760.36
      050091   0     0     15.45   -1.    0.946   159.7    9.59    9.59    0     17.83   0.0     767.83
      050101   0     0     15.45   -1.    0.946   157.9    9.59    9.59    0     17.83   0.0     767.83
      050111   0     0     15.45   -1.    0.946   157.9    9.59    9.59    0     17.83   0.0     767.83
      050121   0     0     15.45   -1.    0.946   167.4    9.59    9.59    0     17.83   0.0     767.83
      050131   0     0     15.45   -1.    0.946   67.48    6.41    6.41    0     12.44   0.0     777.42
      050141   0     0     15.45   -1.    0.946   67.48    6.41    6.41    0     12.44   0.0     777.42
      050151   0     0     15.45   -1.    0.946   67.48    6.41    6.41    0     12.44   0.0     777.42
      050161   0     0     15.45   -1.    0.946   101.2    6.41    6.41    0     15.79   0.0     777.42
      050171   0     0     15.45   -1.    0.946   101.2    6.41    6.41    0     15.79   0.0     777.42
      050181   0     0     15.45   -1.    0.946   100.8    9.59    9.59    0     15.52   0.0     783.83
      050191   0     0     15.45   -1.    0.946   110.0    9.59    9.59    0     15.52   0.0     783.83
      050201   0     0     15.45   -1.    0.946   116.1    9.59    9.59    0     15.52   0.0     783.83
      050211   0     0     15.45   -1.    0.946   171.1    9.59    9.59    0     18.61   0.0     783.83
      050221   0     0     15.45   -1.    0.946   155.8    9.59    9.59    0     18.61   0.0     783.83
      050231   0     0     15.45   -1.    0.946   45.22    10.58   10.58   0     13.39   0.0     793.42
      050241   0     0     15.45   -1.    0.946   55.63    10.58   10.58   0     13.39   0.0     793.42
      050251   0     0     15.45   -1.    0.946   116.2    10.58   10.58   0     16.48   0.0     793.42
      050261   0     0     15.45   -1.    0.946   131.5    10.58   10.58   0     16.48   0.0     793.42
      050271   0     0     15.45   -1.    0.946   176.7    10.58   10.58   0     19.57   0.0     793.42
      050281   0     0     15.45   -1.    0.946   171.8    10.58   10.58   0     19.57   0.0     793.42
      050291   0     0     15.45   -1.    0.946   16.12    4.00    4.00    0     5.42    0.0     796.75
      050301   0     0     15.45   -1.    0.557   16315.   41.00   41.00   0     400.    0.0     793.42
      050311   0     0     15.45   -1.    0.557   11665.   12.10   12.10   0     965.    00      781.32
      050321   0     0     15.45   -1.    0.557   82775.   44.70   44.70   0     1850.   00      736.62
      VOLUME   B     R     PRESS   TEMP   QUAL    VOLUME   HT      MIX     TP    FLOWA   DIAMV   ELEV
  *234567890123456789012345789012345789012345789012345789012345789012345789012345789012345789012345789
  * END VOLUME DATA
  *
  * BEGIN HORIZONTAL FLOW PATHS WITHIN S.S. ANNULUS
   *  234567890123456789012345789012345789012345789012345789012345789012345789012345789012345789012345789
  *      JUNCT----IN——————0T——————P-------------V--------------FLO---------AJUN--------ZJUN---------IN-----------FJUF————FJUR——-----V –C-I-EQ---DM----------CC------------C-E
         080011   1    2    0    0    0.0    14.86   757.82   0.60   0.29   0.00   0 0 0 0   0.0   0.6   1 0
         080021   2    3    0    0    0.0    14.86   757.82   0.60   0.43   0.00   0 0 0 0   0.0   0.6   1 0
         080031   3    4    0    0    0.0    14.86   757.82   0.60   0.29   0.00   0 0 0 0   0.0   0.6   1 0
         080041   5    6    0    0    0.0    20.19   764.10   0.45   0.25   0.00   0 0 0 0   0.0   0.6   1 0
```

TABLE 6.A-7
(SHEET 2 OF 3)

RELAP 4 INPUT DATA, FEEDWATER LINE BREAK

```
080051  6   7   0  0  0.0  20.19  764.10  0.45  0.41  0.00  0 0 0 0  0.0  0.6  1 0
080061  7   8   0  0  0.0  20.19  764.10  0.45  0.25  0.00  0 0 0 0  0.0  0.6  1 0
080071  9   10  0  0  0.0  13.88  772.63  0.69  1.31  0.00  0 0 0 0  0.0  0.6  1 0
080081  10  11  0  0  0.0  13.88  772.63  0.69  1.27  0.00  0 0 0 3  0.0  0.6  1 0
080091  11  12  0  0  0.0  13.88  772.63  0.69  1.31  0.00  0 0 0 3  0.0  0.6  1 0
080101  13  14  0  0  0.0  9.83   780.62  0.65  0.51  0.00  0 0 0 0  0.0  0.6  1 0
080111  14  15  0  0  0.0  9.83   780.62  0.65  0.51  0.00  0 0 0 3  0.0  0.6  1 0
080121  15  16  0  0  0.0  9.83   780.62  0.81  0.38  0.00  0 0 0 3  0.0  0.6  1 0
080131  16  17  0  0  0.0  9.83   780.62  0.97  0.39  0.00  0 0 0 3  0.0  0.6  1 0
080141  18  19  0  0  0.0  14.68  788.62  0.44  0.79  0.00  0 0 0 3  0.0  0.6  1 0
080151  19  20  0  0  0.0  14.68  788.62  0.44  0.83  0.00  0 0 0 3  0.0  0.6  1 0
080161  20  21  0  0  0.0  14.68  788.62  0.54  0.51  0.00  0 0 0 3  0.0  0.6  1 0
080171  21  22  0  0  0.0  14.68  788.62  0.65  0.85  0.00  0 0 0 3  0.0  0.6  1 0
080181  29  23  0  0  0.0  5.42   798.75  0.40  0.85  0.00  0 0 0 0  0.0  0.6  1 0
080191  23  24  0  0  0.0  16.19  798.75  0.20  0.33  0.00  0 0 0 3  0.0  0.6  1 0
080201  24  25  0  0  0.0  16.19  798.75  0.30  0.05  0.00  0 0 0 3  0.0  0.6  1 0
080211  25  26  0  0  0.0  16.19  798.75  0.40  1.33  0.00  0 0 0 3  0.0  0.6  1 0
080221  26  27  0  0  0.0  16.19  798.75  0.50  1.34  0.00  0 0 0 3  0.0  0.6  1 0
080231  27  28  0  0  0.0  16.19  798.75  0.60  0.39  0.00  0 0 0 3  0.0  0.6  1 0
* JUNCT----IN—---------0T—----------P-------------V--------------FLO---------AJUN--------ZJUN---------IN-----------FJUF—-------FJUR——---V C-I-EQ---DM----------CC-----——--C-E
*2345678901234567890123456789012345678900123456789012345678900123456789012345678901234567890
* END HORIZONTAL FLOW PATHS WITHIN 5*5* ANNULUS
*
* BEGIN VERTICAL FLOW PATHS WITHIN S*S* ANNULUS
*01234567890123456789012345678901234567890123456789012345678901234567890123456789012345678901234567890
* JUNCT----IN—---------0T—----------P-------------V--------------FLO---------AJUN--------ZJUN---------IN-----------FJUF—-------FJUR——---V –C-I-EQ---DM----------CC-----——C-E
080241  5   1   0  0  0.0  23.80  760.36  0.26  0.03  0.00  1 0 0 3  0.0  0.6  1 0
080251  6   2   0  0  0.0  23.80  760.36  0.26  0.03  0.00  1 0 0 3  0.0  0.6  1 0
080261  7   3   0  0  0.0  23.80  760.36  0.26  0.03  0.00  1 0 0 3  0.0  0.6  1 0
080271  8   4   0  0  0.0  23.80  760.36  0.26  0.03  0.00  1 0 0 0  0.0  0.6  1 0
080281  9   5   0  0  0.0  10.84  767.83  0.54  1.13  1.28  1 0 0 0  0.0  0.6  1 0
080291  10  6   0  0  0.0  10.84  767.83  0.54  1.13  1.28  1 0 0 3  0.0  0.6  1 0
080301  11  7   0  0  0.0  10.84  767.83  0.54  1.13  1.28  1 0 0 0  0.0  0.6  1 0
080311  12  8   0  0  0.0  10.84  767.83  .054  1.13  1.28  1 0 0 0  0.0  0.6  1 0
080321  13  9   0  0  0.0  7.22   777.42  0.83  0.96  0.00  1 0 0 3  0.0  0.6  1 0
080331  14  9   0  0  0.0  3.61   777.42  1.66  0.96  0.00  1 0 0 3  0.0  0.6  1 0
080341  14  10  0  0  0.0  3.61   777.42  1.66  0.96  0.00  1 0 0 3  0.0  0.6  1 0
080351  15  10  0  0  0.0  7.22   777.42  0.83  0.96  0.00  1 0 0 0  0.0  0.6  1 0
080361  16  11  0  0  0.0  10.84  777.42  0.56  0.96  0.00  1 0 0 0  0.0  0.6  1 0
080371  17  12  0  0  0.0  10.84  777.42  0.56  1.01  0.00  1 0 0 0  0.0  0.6  1 0
080381  18  13  0  0  0.0  7.71   783.83  0.80  0.68  0.00  1 0 0 0  0.0  0.6  1 0
080391  19  14  0  0  0.0  7.71   783.83  0.80  1.03  0.00  1 0 0 0  0.0  0.6  1 0
080401  20  15  0  0  0.0  7.71   783.83  0.80  0.96  0.00  1 0 0 0  0.0  0.6  1 0
080411  21  16  0  0  0.0  11.57  783.83  0.54  0.97  0.00  1 0 0 0  0.0  0.6  1 0
080421  22  17  0  0  0.0  11.57  783.83  0.54  0.96  0.00  1 0 0 0  0.0  0.6  1 0
080431  23  18  0  0  0.0  3.86   793.42  1.94  0.70  0.00  1 0 0 3  0.0  0.6  1 0
080441  24  18  0  0  0.0  3.86   793.42  1.94  0.70  0.00  1 0 0 0  0.0  0.6  1 0
080451  25  19  0  0  0.0  7.71   793.42  0.97  0.98  0.00  1 0 0 0  0.0  0.6  1 0
080461  26  20  0  0  0.0  7.71   793.42  0.97  1.00  0.00  1 0 0 0  0.0  0.6  1 0
080471  27  21  0  0  0.0  11.57  793.42  0.65  0.99  0.00  1 0 0 0  0.0  0.6  1 0
080481  28  22  0  0  0.0  11.57  793.42  0.65  0.97  0.00  1 0 0 0  0.0  0.6  1 0
* JUNCT----IN—---------0T—----------P-------------V--------------FLO---------AJUN--------ZJUN---------IN-----------FJUF—-------FJUR——---V –C-I-EQ---DM----------CC----------C-E
*23457789012345678901234567890123456789012345678901234567890123456789012345678901234567890
* END VERTICAL FLOW PATHS WITHIN S*S*ANNULUS
*
* BEGIN FLOW PATHS TO CONTAINMENT – PENETRATIONS WITH SHIELDING DOORS
* 23457789012345678901234567890123456789012345678901234567890123456789012345678901234567890
* JUNCT----IN—---------0T—----------P-------------V--------------FLO---------AJUN--------ZJUN---------IN-----------FJUF—-------FJUR——---V –C-I-EQ---DM----------CC-----——C-E
080491  23  30  0  1  0.0  1.54   798.75  3.60  1.61  0.00  0 0 0 0  0.0  0.6  1 0
080501  24  30  0  2  0.0  3.86   798.75  1.30  1.07  0.00  0 0 0 0  0.0  0.6  1 0
080511  25  30  0  3  0.0  7.71   798.75  1.06  1.99  0.00  0 0 0 0  0.0  0.6  1 0
080521  26  30  0  4  0.0  7.71   798.75  1.06  1.99  0.00  0 0 0 0  0.0  0.6  1. 0
080531  27  30  0  5  0.0  9.27   798.75  0.79  2.40  0.00  0 0 0 0  0.0  0.6  1 .0
080541  28  30  0  6  0.0  11.57  798.75  0.65  1.82  0.00  0 0 0 0  0.0  0.6  1 0
080551  29  30  0  0  0.0  0.68   798.75  3.96  1.71  0.00  0 0 0 0  0.0  0.6  1 0
080561  28  30  0  0  0.0  0.68   798.75  3.96  1.71  0.00  0 0 0 3  0.0  0.6  1 0
080571  27  30  0  0  0.0  1.36   798.75  1.98  1.71  0.00  0 0 0 3  0.0  0.6  1 0
080581  26  30  0  0  0.0  1.36   798.75  1.70  1.73  0.00  0 0 0 3  0.0  0.6  1 0
080591  25  30  0  0  0.0  0.68   798.75  3.96  1.71  0.00  0 0 0 3  0.0  0.6  1 0
080601  30  31  0  0  0.0  400.   793.42  0.06  0.05  0.00  1 0 0 3  0.0  0.6  1 0
080611  22  31  0  0  0.0  0.71   788.62  3.86  1.71  0.00  0 0 0 3  0.0  0.6  1 0
080621  21  31  0  0  0.0  1.39   788.62  1.70  1.73  0.00  0 0 0 3  0.0  0.6  1 0
080631  20  31  0  0  0.0  0.68   788.62  2.98  1.74  0.00  0 0 0 3  0.0  0.6  1 0
080641  19  31  0  0  0.0  1.42   788.62  1.93  1.71  0.00  0 0 0 3  0.0  0.6  1 0
080651  31  32  0  0  0.0  965.   781.32  0.03  0.05  0.00  1 0 0 3  0.0  0.6  1 0
080661  12  32  0  0  0.0  2.89   772.63  0.90  1.71  0.00  0 0 0 3  0.0  0.6  1 0
080671  11  32  0  0  0.0  2.50   772.63  1.17  1.71  0.00  0 0 0 3  0.0  0.6  1 0
```

TABLE 6.A-7
(SHEET 3 OF 3)

RELAP 4 INPUT DATA, FEEDWATER LINE BREAK

```
080681  10      32      0       0       0.0     2.50    772.63  1.17    1.71    0.00    0 0 0 3  0.0     0.6
1 0
080691  9       32      0       0       0.0     2.14    772.63  1.29    1.71    0.00    0 0 0 3  0.0     0.6
1 0
* JUNCT----IN—---------0T—----------P-------------V--------------FLO---------AJUN--------ZJUN---------IN-----------FJUF—------FJUR——----V –C-I-EQ---DM----------CC-----
-—C-E
* 2345778901234567890123456789012345678901234567890123456789012345678901234567890
* END FLOW PATHS TO CONTAINMENT – PENETRATIONS WITH SHIELDING DOORS
*
* BEGIN FILL PATH
* 2345778901234567890123456789012345678901234567890123456789012345678901234567890
* JUNCT----IN—---------0T—----------P-------------V--------------FLO---------AJUN--------ZJUN---------IN-----------FJUF—------FJUR——----V –C-I-EQ---DM----------CC----
----—C-E
080701  0       29      1       0       0.0     1.0     789.75  0.0     0.0     0.0     0 0 0 3  0.0     1.0
1 0
* JUNCT----IN—---------0T—----------P-------------V--------------FLO---------AJUN--------ZJUN---------IN-----------FJUF—------FJUR——----V –C-I-EQ---DM----------CC-----
-—C-E
* 2345778901234567890123456789012345678901234567890123456789012345678901234567890
* END FILL PATH
*
* VALVE  DATA CARDS
110010  -3      0.0     0.0     0.0     0.0
110020  -4      0.0     0.0     0.0     0.0
110030  -5      0.0     0.0     0.0     0.0
110040  -6      0.0     0.0     0.0     0.0
110050  -7      0.0     0.0     0.0     0.0
110060  -8      0.0     0.0     0.0     0.0
*
* FILL TABLE  DATA CARDS
* FILL CONTROL
130100  4       2       0       0       1045.   420.
* CARD  TIME    FLOW    TIME    FLOW
1030101 0.0     14200.  0.001050 14200.
1030102 0.001060 21600. 1.00    21600.
*
* 2345778901234567890123456789012345678901234567890123456789012345678901234567890
***********************************************************************************************************************************
**********M
* MODEL REVISIONS
130101  0.0     7100.   0.001050 7100.
130102  0.001060 10800. 1.00    10800.
CARD ABOVE  IS REPLACEMENT CARD.

***********************************************************************************************************************************
*********M
*
```

TABLE 6.A-8
(SHEET 1 OF 2)

FORCE CONSTANTS AND LOAD CENTERS FOR RECIRCULATION LINE OUTLET BREAK

NODE	O_v	O_{ss}	ELEVATION	Π
1	3696.03	4948.35	757.82	15.0°, 345.0°
2	3696.03	4948.35	757.825	45.0°, 315.0°
3	3696.03	4948.35	757.825	75.0°, 285.0°
4	5464.86	7316.51	757.825	112.5°, 247.5°
5	5464.86	7316.51	757.825	157.5°, 202.5°
6	5828.44	7290.77	764.095	15.0°, 345.0°
7	5828.44	7290.77	764.095	45.0°, 315.0°
8	5828.44	7290.77	764.095	75.0°, 285.0°
9	8617.79	10779.95	764.095	112.5°, 247.5°
10	8617.79	10779.95	764.095	157.5°, 202.5°
11	2857.42	2503.87	771.290	22.5°, 337.5°
12	4038.29	3887.97	770.280	45.0°, 315.0°
13	4022.57	2990.40	770.280	75.0°, 285.0°
14	5970.91	5748.63	770.280	112.5°, 247.5°
15	5891.80	5523.42	770.280	157.5°, 202.5°
16	2234.85	2605.94	776.085	15.0°, 345.0°
17	3862.52	3683.01	775.075	45.0°, 315.0°
18	3862.52	3683.01	775.075	75.0, 285.0
19	5711.02	5445.58	775.075	112.5°, 247.5°
20	5631.91	5220.37	775.075	157.5°, 202.5°
21	5325.49	6256.20	780.625	15.0°, 345.0°
22	5325.49	6256.20	780.625	45.0°, 315.0°

TABLE 6.A-8
(SHEET 2 OF 2)

NODE	O_v	O_{ss}	ELEVATION	Π
23	5325.49	6256.20	780.625	75.0°, 285.0°
24	7874.13	9250.27	780.625	112.5°, 247.5°
25	7874.13	9250.27	780.625	157.5°, 202.5°
26	11713.96	11338.09	788.625	22.5°, 337.5°
27	11713.28	12957.61	788.625	67.5°, 297.5°
28	11713.28	12957.61	788.625	112.5°, 247.5°
29	11713.96	11338.09	788.625	157.5°, 202.5°
30	12864.45	12694.81	798.710	22.5°, 337.5°
31	12809.98	12622.87	798.710	67.5°, 297.5°
32	12934.41	14386.28	798.710	112.5°, 247.5°
33	12867.88	11885.05	798.710	157.5°, 202.5°
34	1557.92	2042.96	771.290	7.5°, 352.5°
35	1140.80	0.00	772.020	0.0°, 360.0°

TABLE 6.A-9
(SHEET 1 OF 2)

FORCE CONSTANTS AND LOAD CENTERS
FOR FEEDWATER LINE BREAK

NODE	O_v	O_{ss}	ELEVATION	Π
1	5464.86	7316.51	757.825	22.5°, 337.5°
2	5464.86	7316.51	757.825	67.5°, 292.5°
3	5464.86	7316.51	757.825	112.5°, 147.50
4	5464.86	7316.51	757.825	157.5°, 202.5°
5	8617.79	10779.95	764.095	22.5°, 337.5°
6	8617.79	10779.95	764.095	67.5°, 292.5°
7	8617.79	10779.95	764.095	112.5°, 247.5°
8	8617.79	10779.95	764.095	157.5°, 202.5°
9	11681.94	11194.20	772.625	22.5°, 337.5°
10	11523.72	10743.78	772.625	67.5°, 292.5°
11	11523.72	10743.78	772.625	112.5°, 247.5°
12	11666.44	10309.43	772.625	157.5, 202.5
13	5325.49	6256.20	780.625	15.0°, 345.0°
14	5325.49	6256.20	780.625	45.0°, 315.0°
15	5325.49	6256.20	780.625	75.0°, 285.0°
16	7874.13	9250.27	780.625	112.5°, 247.5°
17	7874.13	9250.27	780.625	157.5°, 202.5°
18	7967.46	9359.90	788.625	15.0°, 345.0°
19	7841.24	7570.97	788.625	45.0°, 315.0°
20	7963.08	7716.94	788.625	75.0°, 285.0°
21	11713.96	11338.09	788.625	112.5°, 247.5°
22	11718.28	12957.61	788.625	157.5°, 202.5°
23	3530.66	4305.38	798.710	7.5°, 352.5°

TABLE 6.A-9
(SHEET 2 OF 2)

NODE	O_v	O_{ss}	ELEVATION	Π
24	4432.90	5207.63	798.710	22.5°, 337.5°
25	8726.85	9431.69	798.710	45.0°, 315.0°
26	8722.47	7788.73	798.710	75.0°, 285.0°
27	12872.20	13504.58	798.710	112.5°, 247.5°
28	12934.41	14386.28	798.710	157.5°, 202.5°
29	840.94	0.00	798.710	0.0°, 360.0°

TABLE 6.A-10
(SHEET 1 OF 2)
DYSEA01 PROGRAM INPUT FOR JET LOAD FORCES

TIME FUNCTION NUMBER = (1)

FUNCTION DESCRIPTION = (RESTRAINT LOAD AT NODE 2)

NUMBER OF ABSCISSAE = (51)
FUNCTION SCALE FACTOR = (3.8880E-01)

TIME VALUE	FUNCTION	TIME VALUE	FUNCTION	TIME VALUE	FUNCTION	TIME VALUE	FUNCTION	TIME VALUE	FUNCTION
0.00163	0.	0.00233	0.	0.00297	0.	0.00351	0.	0.00398	0.
0.00441	0.	0.00481	0.	0.00519	0.	0.00554	0.	0.00667	0.
0.00887	5.0588E 04	0.00787	1.0820E 05	0.00887	1.6604E 05	0.00987	2.2989E 05	0.01087	2.9304E 05
0.01187	3.5842E 05	0.01287	4.1875E 05	0.01387	4.7365E 05	0.01487	5.3891E 05	0.01587	5.8180E 05
0.01687	6.1887E 05	0.01787	6.4976E 05	0.01887	6.7423E 05	0.01987	6.9213E 05	0.02087	7.0347E 05
0.02187	7.0834E 05	0.02222	7.0857E 05	0.02242	7.0857E 05	0.02262	7.0857E 05	0.02283	7.0857E 05
0.02304	7.0857E 05	0.02325	7.0857E 05	0.02347	7.0857E 05	0.02370	7.0857E 05	0.02393	7.0857E 05
0.02417	7.0857E 05	0.02442	7.0857E 05	0.02467	7.0857E 05	0.02494	7.0857E 05	0.02522	7.0857E 05
0.02551	7.0857E 05	0.02582	7.0857E 05	0.02614	7.0857E 05	0.02649	7.0857E 05	0.02687	7.0858E 05
0.02728	7.0857E 05	0.02774	7.0857E 05	0.02827	7.0857E 05	0.02893	7.0857E 05	0.02992	7.0858E 05
0.19740	7.0857E 05								

TIME FUNCTION NUMBER = (2)

FUNCTION DESCRIPTION = (RESTRAINT LOAD AT NODE 3)

NUMBER OF ABSCISSAE = (51)
FUNCTION SCALE FACTOR = (6.1120E-01)

TIME VALUE	FUNCTION	TIME VALUE	FUNCTION	TIME VALUE	FUNCTION	TIME VALUE	FUNCTION	TIME VALUE	FUNCTION
0.00153	0.	0.00233	0.	0.00297	0.	0.00351	0.	0.00398	0.
0.00441	0.	0.00481	0.	0.00519	0.	0.00554	0.	0.00587	0.
0.00687	5.0588E 04	0.00787	1.0820E 05	0.00887	1.6604E 05	0.00987	2.2989E 05	0.01087	2.9304E 05
0.01187	3.5642E 05	0.01287	4.1875E 05	0.01387	4.7365E 05	0.01487	5.3891E 05	0.01587	5.8180E 05
0.01687	6.1887E 05	0.01787	8.4976E 05	0.01887	6.7423E 05	0.01987	6.9213E 05	0.02087	7.0347E 05
0.02187	7.0834E 05	0.02222	7.0857E 05	0.02242	7.0857E 05	0.02202	7.0857E 05	0.02283	7.0857E 05
0.02304	7.0857E 05	0.02325	7.0857E 05	0.02347	7.0857E 05	0.02370	7.0857E 05	0.02393	7.0857E 05
0.02417	7.0857E 05	0.02442	7.0857E 05	0.02467	7.0857E 05	0.02494	7.0857E 05	0.02522	7.0857E 05
0.02551	7.0857E 05	0.02582	7.0857E 05	0.02614	7.0857E 05	0.02649	7.0857E 05	0.02687	7.0858E 05
0.02728	7.0857E 05	0.02774	7.0857E 05	0.02827	7.0857E 05	0.02893	7.0857E 05	0.02992	7.0858E 05
0.19740	7.0857E 05								

TABLE 6.A-10
(SHEET 2 OF 2)
DYSEA01 PROGRAM INPUT FOR JET LOAD FORCES

TIME FUNCTION NUMBER = (3)

FUNCTION DESCRIPTION = (BLOWDOWN LOAD AT NODE 34 & JET LOAD)

NUMBER OF ABSCISSAE = (51)
FUNCTION SCALE FACTOR = (-2.4270E 00)

TIME VALUE	FUNCTION	TIME VALUE	FUNCTION	TIME VALUE	FUNCTION	TIME VALUE	FUNCTION	TIME VALUE	FUNCTION
0.00153	2.8666E 05	0.00233	2.8666E 05	0.00297	2.8666E 05	0.00351	2.8666E 05	0.00398	2.8666E 05
0.00441	2.8666E 05	0.00481	2.8666E 05	0.00519	2.8666E 05	0.00554	2.8666E 05	0.00587	2.8666E 05
0.00887	2.8666E 05	0.00787	2.8666E 05	0.00887	2.8666E 05	0.00987	2.8666E 05	0.01087	2.8666E 05
0.01187	2.8666E 05	0.01287	2.8666E 05	0.01387	2.8666E 05	0.01487	2.8666E 05	0.01587	2.8666E 05
0.01687	2.8666E 05	0.01787	2.8666E 05	0.01887	2.8666E 05	0.01987	2.8666E 05	0.02087	2.8666E 05
0.02187	2.8666E 05	0.02222	2.8666E 05	0.02242	2.8666E 05	0.02262	2.8666E 05	0.02283	2.8666E 05
0.02304	2.8666E 05	0.02325	2.8666E 05	0.02347	2.8666E 05	0.02370	2.8666E 05	0.02390	2.8666E 05
0.02417	2.8666E 05	0.02442	2.8666E 05	0.02467	2.8666E 05	0.02494	2.8666E 05	0.02522	2.8666E 05
0.02551	2.8666E 05	0.02582	2.8666E 05	0.02614	2.8666E 05	0.02649	2.8666E 05	0.02687	2.8666E 05
0.02728	2.8666E 05	0.02774	2.8666E 05	0.02827	2.8666E 05	0.02893	2.8666E 05	0.02992	2.8666E 05
0.19740	2.8666E 05								

TIME FUNCTION NUMBER = (4)

FUNCTION DESCRIPTION = (BLOWDOWN LOAD AT NODE 35 & JET LOAD)

NUMBER OF ABSCISSAE = (51)
FUNCTION SCALE FACTOR = (-2.4270E 00)

TIME VALUE	FUNCTION	TIME VALUE	FUNCTION	TIME VALUE	FUNCTION	TIME VALUE	FUNCTION	TIME VALUE	FUNCTION
0.00163	6.0254E 04	0.00233	6.0254E 04	0.00297	6.0254E 04	0.00351	6.0254E 04	0.00398	6.0254E 04
0.00441	6.0254E 04	0.00481	6.0254E 04	0.00519	6.0254E 04	0.00554	6.0254E 04	0.00587	6.0254E 04
0.00887	6.0254E 04	0.00787	6.0254E 04	0.00887	6.0254E 04	0.00987	6.0254E 04	0.01087	6.0254E 04
0.01187	6.0254E 04	0.01287	6.0254E 04	0.01387	6.0254E 04	0.01487	6.0254E 04	0.01587	6.0254E 04
0.01687	6.0254E 04	0.01787	6.0254E 04	0.01887	6.0254E 04	0.01987	6.0254E 04	0.02087	6.0254E 04
0.02187	6.0254E 04	0.02222	6.0254E 04	0.02242	6.0254E 04	0.02262	6.0254E 04	0.02283	6.0254E 04
0.02304	6.0254E 04	0.02325	6.0254E 04	0.02347	6.0254E 04	0.02370	6.0254E 04	0.02390	6.0254E 04
0.02417	6.0254E 04	0.02442	6.0254E 04	0.02467	6.0254E 04	0.02494	6.0254E 04	0.02522	6.0254E 04
0.02551	6.0254E 04	0.02582	6.0254E 04	0.02614	6.0254E 04	0.02649	6.0254E 04	0.02687	6.0254E 04
0.02726	6.0254E 04	0.02774	6.0254E 04	0.02827	6.0254E 04	0.02893	6.0254E 04	0.02992	6.0254E 04
0.19740	6.0254E 04								

TABLE 6.A-11
(Sheets 1 through 32)

TIME FORCE HISTORIES – RECIRCULATION LINE BREAK

Deleted

TABLE 6.A-12
(Sheets 1 through 28)

TIME FORCE HISTORIES – FEEDWATER LINE BREAK

Deleted

ATTACHMENT 6.B
RECIRCULATION SYSTEM SINGLE-LOOP OPERATION

ATTACHMENT 6.B

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ATTACHMENT 6.B

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ATTACHMENT 6.B

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6.B RECIRCULATION SYSTEM SINGLE-LOOP OPERATION

6.B.1 INTRODUCTION AND SUMMARY

Sections 6.B.2, 6.B.3, 6.B.4, and 6.B.5 describe the GE methodology for the MCPR safety limit calculation and single loop operation transient analyses. The transient analyses presented in this chapter are for a specific cycle, and are not re-performed for each reload.

6.B.1.1 GE Analysis

Single-loop operation at reduced power is highly desirable in the event recirculation pump or other component maintenance renders one loop inoperative. To justify single-loop operation, accidents and abnormal operational transients associated with power operations, as presented in Section 6.3 and Chapter 15.0, were reviewed for the single loop case with only one pump in operation.

Increased uncertainties in the core total flow and TIP readings resulted in an incremental increase in the MCPR fuel cladding integrity safety limit during single-loop operation. This increase is reflected in the MCPR operating limit. No other increase in this limit is required because all abnormal operational transients are bounded by the rated power/flow analyses performed. The least-stable power/flow condition, achieved by tripping both recirculation pumps, is not affected by one-pump operation.

6.B.2 MCPR FUEL CLADDING INTEGRITY SAFETY LIMIT

Except for core total flow and TIP reading, the uncertainties used in the statistical analysis to determine the MCPR fuel cladding integrity safety limit are not dependent on whether coolant flow is provided by one or two recirculation pumps. A 6% core flow measurement uncertainty has been established for single-loop operation (compared to 2.5% for two-loop operation). As shown below, this value conservatively reflects the one standard deviation (one sigma) accuracy of the core flow measurement system documented in Reference 1. The random noise component of the TIP reading uncertainty was revised for single recirculation loop operation to reflect the operating plant test results given in Subsection 6.B.2.4. This revision resulted in a single-loop operation process computer uncertainty of 6.8% for initial cores. Comparable two-loop process computer uncertainty values are 6.3% for initial cores. The net effect of these two revised uncertainties is an incremental increase in the required MCPR fuel cladding integrity safety limit.

6.B.2.1 Core Flow Uncertainty

6.B.2.2 Core Flow Measurement During Single-Loop Operation

The jet pump core flow measurement system is calibrated to measure core flow when both sets of jet pumps are in forward flow; total core flow is the sum of the indicated loop flows. For single-loop operation, however, some inactive jet pumps will be backflowing. Therefore, the measured flow in the backflowing jet pumps must be subtracted from the measured flow in the active loop. In addition, the jet pump coefficient is different for reverse flow than for forward flow, and the measurement of reverse flow must be modified to account for this difference.

For single-loop operation the total core flow is derived by the following formula:

$$\begin{pmatrix} \text{Total Core} \\ \text{Flow} \end{pmatrix} = \begin{pmatrix} \text{Active Loop} \\ \text{Indicated Flow} \end{pmatrix} - C \begin{pmatrix} \text{Inactive Loop} \\ \text{Flow} \end{pmatrix}$$

Where C (= 0.95) is defined as the ratio of "Inactive Loop True Flow" to "Inactive Loop Indicated Flow," and "Loop Indicated Flow" is the flow indicated by the jet pump "single-tap" loop flow summers and indicators, which are set to indicate forward flow correctly.

The 0.95 factor was the result of a conservative analysis to appropriately modify the single-tap flow coefficient for reverse flow. (NOTE: The LSCS value of the "C" coefficient is 0.78 (±0.078) at reactor operating conditions.) If a more exact, less conservative core flow is required, special in-reactor calibration tests would have to be made. Such calibration tests would involve calibrating core support plate ΔP versus core flow during two-pump operation along the 100% flow control line, operating on

one pump along the 100% flow control line, and calculating the correct value of C based on the core derived from the core support plate ΔP and the loop flow indicator readings.

6.B.2.3 Core Flow Uncertainty Analysis

The uncertainty analysis procedure used to establish the core flow uncertainty for one-pump operation is essentially the same as for two-pump operation, except for some extensions. The core flow uncertainty analysis is described in Reference 1. The analysis of one-pump core flow uncertainty is summarized below.

For single-loop operation, the total core flow can be expressed as follows (refer to Figure 6.B-1):

$$W_C = W_A - C\,W_I$$

Where

W_C = total core flow,

W_A = active loop flow, and

W_I = inactive loop (true) flow.

By applying the "propagation of errors" method to the above equation, the variance of the total flow uncertainty can be approximated by:

$$\sigma_{W_C}^{\;2} = \left(\frac{1}{1 - a}\right)^2 \sigma_{W_A}^{\;2} + \left(\frac{a}{1 - a}\right)^2 \sigma_{W_I}^{\;2}$$

where

σW_C = uncertainty in total core flow (%),

σW_A = uncertainty in active loop flow (%),

σW_I = uncertainty in inactive loop flow (%), and

a = W_I / W_A

The uncertainty of σW_A was analyzed to be 2.8%. A conservative, bounding value of 3.0% was used for σW_A in the total flow uncertainty variance calculation. The

uncertainty, σW_I is comprised of the uncertainty in the "C" coefficient and random uncertainties such as jet pump ΔP measurement uncertainty and instrumentation errors. The bounding value of 3.75% for σW_I was used in the determination of σW_I. Based on the above uncertainties and a bounding value of 0.36 for a, the variance of the total flow uncertainty is approximately:

$$\sigma W_C{}^2 = \left(\frac{1}{1 - 0.36}\right)^2 (3.0\%)^2 + \left(\frac{0.36}{1 - 0.36}\right)^2 (3.75\%)^2$$

$$= (5.0\%)^2$$

When the effect of 4.1% core bypass flow uncertainty at 12% (bounding case) bypass flow fraction is added to the above total core flow uncertainty, the active coolant flow uncertainty is:

$$\sigma^2_{\text{active}} = (5.0\%)^2 + \left(\frac{0.12}{1-0.12}\right)^2 (4.1\%)^2 = (5.7\%)^2$$

which is less than the 6% core flow uncertainty assumed in the statistical analysis.

In summary, core flow during one-pump operation is established in a conservative way and its uncertainty has been conservatively evaluated.

6.B.2.4 TIP Reading Uncertainty

TIP uncertainties used in the Safety Limit MCPR analysis can be found in Reference 8.

6.B.3 MCPR OPERATING LIMIT

6.B.3.1 Abnormal Operational Transients

The consequences of an Anticipated Operational Occurrence (AOO) initiated from Single Loop Operation (SLO) are no different than the consequences of the same event initiated from two-loop operation, given the same initial power/flow conditions. One transient analyzed only for single loop operation, the abnormal startup of an idle recirculation loop, results in more severe consequences at low power levels than similar cold water injection transients (i.e. feedwater controller failure) as analyzed for two loop operation. An analysis of this event is given in Section 15.4.4. The fuel thermal-mechanical integrity and safety limit MCPR (as increased for SLO) are protected during a postulated AOO in SLO mode by adhering to thermal limits derived from the more limiting of either the two-loop operation AOO results or the results from the idle recirculation loop startup event. Results of these analyses, and a discussion of the applicability of these analyses to SLO, may be found in the LaSalle Administrative Technical Requirements and its associated references.

Figure 6.B-2 shows the consequences of a typical pressurization transient (turbine trip) as a function of power level. As can be seen, the consequences of operation at lower power (such as would occur during SLO) result in lower reactor pressurization and neutron flux levels. Therefore, in absolute terms of maximum pressure and flux, SLO results in a milder transient than two-loop operation.

The power and flow dependent thermal limits developed for two loop operation are applicable for SLO, except for portions of the thermal limits which must be adjusted for the more severe consequences of the idle recirculation loop startup event discussed above. The flow dependent thermal limits are based on the event where both recirculation loop controllers fail (in the case of SLO, this event bounds failure of one controller, as the flow and power increase would be less). However, for operation in SLO, the flow dependent thermal limits are adjusted to also bound the results of the idle recirculation loop startup event. These thermal limits are found in the LaSalle Administrative Technical Requirements.

The power dependent thermal limits are based on pressurization transients, such as the load rejection without bypass event, and the feedwater controller failure event (which is also a cold water injection event). As described above, the two loop results bound the SLO results for these events. Therefore, these SLO thermal limits are only different from the dual loop thermal limits in that they have been adjusted to protect a MCPR safety limit that is higher than the dual loop value.

In the following sections, three of the most limiting transients of cold water increase, pressurization, and flow decrease events are analyzed for single-loop operation. These analyses were performed for the initial cycle core. For reload

cores, the bounding two loop operation analysis results for events a and b below are found in the LaSalle Administrative Technical Requirements. The transients are, respectively:

a. feedwater flow controller failure (maximum demand),

b. generator load rejection with bypass failure, and

c. one pump seizure accident.

The plant initial conditions are given in Table 6.B-1.

6.B.3.2 Feedwater Controller Failure - Maximum Demand

This section presents initial cycle GE results.

6.B.3.2.1 Identification of Causes and Frequency Classification

This event is postulated on the basis of a single failure of a control device, specifically one which can directly cause an increase in coolant inventory by increasing the feedwater flow. The most severe applicable event is a feedwater controller failure during maximum flow demand. The feedwater controller is forced to its upper limit at the beginning of the event.

This event is considered to be an incident of moderate frequency.

6.B.3.2.2 Sequence of Events and Systems Operation

With excess feedwater flow the water level rises to the high-level reference point at which time the feedwater pumps and the main turbine are tripped and a scram is initiated. Table 6.B-2 lists the sequence of events for Figure 6.B-3. The figure shows the changes in important variables during this transient.

Identification of Operator Actions

a. Observe that high feedwater pump trip has terminated the failure event.

b. Switch the feedwater controller from auto to manual control in order to try to regain a correct output signal.

c. Identify causes of the failure and report all key plant parameters during the event.

Systems Operation

In order to properly simulate the expected sequence of events, the analysis of this event assumes normal functioning of plant instrumentation and controls, plant protection and reactor protection systems. Important system operational actions for this event are high level tripping of the main turbine, feedwater turbine, turbine stop valve scram trip initiation, recirculation pump trip (RPT), and low-water level initiation of the reactor core isolation cooling system and the high-pressure core spray system to maintain long-term water level control following tripping of feedwater pumps (not simulated).

6.B.3.2.3 Effect of Single Failures and Operator Errors

In Table 6.B-2 the first sensed event to initiate corrective action to the transient is the vessel high-water level (L8) trip. Multiple level sensors are used to sense and detect when the water level reaches the L8 setpoint. At this point in the logic, a single failure will not initiate or prevent a turbine trip signal. Turbine trip signal transmission, however, is not built to single-failure criterion. The result of a failure at this point would have the effect of delaying the pressurization "signature." However, high moisture levels entering the turbine will be detected by high levels in the moisture separators which are designed to trip the unit. In addition, excessive moisture entering the turbine will cause vibration to the point where it too will trip the unit.

Scram trip signals from the turbine are designed such that a single failure will neither initiate nor impede a reactor scram trip initiation.

6.B.3.2.4 Core and System Performance

Mathematical Model

The computer model described in Subsection 15.1.2A.3 was used to simulate this event.

Input Parameters and Initial Conditions

The analysis has been performed with the plant condition tabulated in Table 6.B-1, except that the initial vessel water level is at level setpoint L4 for conservation. By lowering the initial water level, more feedwater will get in, hence higher neutron flux will be attained before Level 8 is reached.

The same void reactivity coefficient used for pressurization transient is applied since a more negative value conservatively increases the apparent severity of the power increase. End of cycle (all rods out) scram characteristics are assumed. The safety/relief valve action is conservatively assumed to occur with higher than

nominal setpoints. The transient is simulated by programming an upper limit failure in the feedwater system such that 135% feedwater flow occurs at design pressure of feedwater spargers (1075 psia). Since the reactor is initially operating at a lower power level, the feedwater sparger experiences a pressure which is much lower than the design pressure, hence the feedwater runout capacity reaches 160% of rated.

Results

The simulated feedwater controller transient is shown in Figure 6.B-3 for the case of 78% power 63% core flow. The high-water level turbine trip and feedwater pump trip are initiated at approximately 5.46 seconds. Scram occurs simultaneously from stop valve closure, and limits the neutron flux peak and fuel thermal transient so that no fuel damage occurs. MCPR remains above safety limit and peak fuel center temperature increases less than 170° F. The turbine bypass system opens to limit peak pressure in the steamline near the safety valves to 1103 psig and the pressure at the bottom of the vessel to about 1118 psig.

Consideration of Uncertainties

All systems utilized for protection in this event were assumed to have the poorest allowable response (e.g., relief setpoints, scram stroke time, and work characteristics). Expected plant behavior is, therefore, expected to lead to a less severe transient.

6.B.3.2.5 Barrier Performance

As noted above, the consequences of this event do not result in any temperature or pressure transient in excess of the criteria for which the fuel, pressure vessel, or containment are designed; therefore, these barriers maintain integrity and function as designed.

6.B.3.2.6 Radiological Consequences

The consequences of this event do not result in any fuel failures; however, radioactive steam is discharged to the suppression pool as a result of SRV activation.

6.B.3.3 Generator Load Rejection Without Bypass With RPT

This section presents initial cycle GE results.

6.B.3.3.1 Identification of Causes and Frequency Classification

Fast closure of the turbine control valves (TCV) is initiated whenever electrical grid disturbances occur which result in significant loss of electrical load on the generator. The turbine control valves are required to close as rapidly as possible to prevent overspeed of the turbine-generator rotor. Closure of the main turbine control valves will increase system pressure.

This event is categorized as an infrequent incident with the following characteristics:

Frequency:	0.0036/plant-year
MTBE:	278 years

Frequency basis: thorough searches of domestic plant operating records have revealed three instances of bypass failure during 628 bypass system operations. This gives a probability of bypass failure of 0.0048. Combining the actual frequency of a generator load rejection with the failure rate of the bypass yields a frequency of a generator load rejection with bypass failure of 0.0036 event/plant year.

6.B.3.3.2 Sequence of Events and System Operation

Sequence of Events

A loss of generator electrical load at 78% and 63% flow under single recirculation loop operation produces the sequence of events listed in Table 6.B-3. Notice that the vessel level reaches L8 at 5.3 seconds. The trip of feedwater pumps on L8 is not simulated.

Identification of Operator Options

a. Verify proper bypass valve performance.

b. Observe that the pressure regulator is controlling reactor pressure at the desired value.

c. Record peak power and pressure.

d. Verify relief valve operation.

System Operation

Turbine control valve (TCV) fast closure initiates a scram trip signal for power levels greater than or equal to 25% of rated core thermal power. In addition,

recirculation pump trip is initiated. Both of these trip signals satisfy single failure criterion and credit is taken for these protection features.

The pressure relief system which operates the relief valves independently when system pressure exceeds relief valve instrumentation setpoints is assumed to function normally during the time period analyzed.

All plant control systems maintain normal operation unless specifically designated to the contrary.

Mitigation of pressure increase, the basic nature of this transient, is accomplished by the reactor protection system functions. Turbine control valve trip scram and RPT are designed to satisfy the single failure criterion.

Mathematical Model

The computer model described in Subsection 15.2.2A.3 was used to simulate this event.

Input Parameters and Initial Conditions

These analyses have been performed, unless otherwise noted, with the plant conditions tabulated in Table 6.B-1.

The turbine electrohydraulic control system (EHC) power/load imbalance device detects load rejection before a measurable speed change takes place.

The closure characteristics of the turbine control valves are assumed such that the valves operate in the full arc (FA) mode and have a full stroke closure time, from fully open to fully closed, of 0.15 second.

Auxiliary power would normally be independent of any turbine-generator overspeed effect and continuously supplied at rated frequency as automatic fast transfer to auxiliary power supplies normally occurs. For the purposes of worst case analysis, the recirculation pumps are assumed to remain tied to the main generator and thus increase in speed with the T-G overspeed until tripped by the recirculation pump trip system (RPT).

The reactor is operating in the manual flow-control mode when load rejection occurs. Results do not significantly differ if the plant had been operating in the automatic flow-control mode.

6.B.3.3.3 Results

Figure 6.B-4 shows that, for the case of bypass failure, peak neutron flux reaches about 135.6% of rated, average surface heat flux reaches 8% of rated. The calculated MCPR is 1.29, which is well above the safety limit.

Consideration of Uncertainties

The full-stroke closure rate of the turbine control valve of 0.15 second is conservative. Typically, the actual closure rate is more like 0.2 second. Clearly the less time it takes to close, the more severe the pressurization effect.

All systems utilized for protection in this event were assumed to have the poorest allowable response (e.g., relief setpoints, scram stroke time, and worth characteristics). Expected plant behavior is, therefore, expected to reduce the actual severity of the transient.

Peak pressure at the valves reaches 1128 psig. The peak nuclear system pressure reaches 1153 psig at the bottom of the vessel, well below the nuclear barrier transient pressure limit of 1375 psig.

6.B.3.3.4 Barrier Performance

The consequences of these events do not result in any temperature or pressure transient in excess of the criteria for which the fuel, pressure vessel, or containment are designed and, therefore, these barriers maintain their integrity as designed.

6.B.3.3.5 Radiological Consequences

The consequences of the events identified previously do not result in any fuel failures; however, radioactivity is nevertheless discharged to the suppression pool as a result of SRV activation.

6.B.3.4 Recirculation Pump Seizure Accident

This analysis presents initial cycle GE results.

6.B.3.4.1 Identification of Causes and Frequency Classification

The case of recirculation pump seizure represents the extremely unlikely event of instantaneous stoppage of the pump motor shaft of one recirculation pump. This produces a very rapid decrease of core flow as a result of the large hydraulic resistance introduced by the stopped rotor.

This event is considered to be a limiting fault.

Actual occurrence data is not available at this time.

6.B.3.4.2 Sequence of Events and Systems Operations

Table 6.B-4 lists the sequence of events for this recirculation pump seizure accident.

Identification of Operator Actions

The operator should ascertain that the reactor scrams with the turbine trip resulting from reactor water level swell. The operator should regain control of reactor water level through RCIC operation or by restart of a feedwater pump, and must monitor reactor water level and pressure control after shutdown.

6.B.3.4.3 Systems Operation

In order to properly simulate the expected sequence of events, the analysis of this event assumes normal functioning of plant instrumentation and controls, plant protection, and reactor protection systems.

Operation of HPCS and RCIC systems, though not included in this simulation, are expected to occur in order to maintain adequate water level.

6.B.3.4.4 Core and System Performance

Mathematical Model

The nonlinear dynamic model described briefly in Subsection 15.3.3.3 is used to simulate this event.

Input Parameters and Initial Conditions

This analysis has been performed, unless otherwise noted, with plant conditions tabulated in Table 6.B-1. For the purpose of evaluating consequences to the fuel thermal limits this transient event is assumed to occur as a consequence of an unspecified, instantaneous stoppage of the active recirculation pump shaft while the reactor is operating at 78% NB rated power under SLO conditions. Also, the reactor is assumed to be operating at thermally limited conditions.

The void coefficient is adjusted to the most conservative value; that is, the least negative value in Table 6.B-1.

6.B.3.4.5 Results

Figure 6.B-5 presents the results of the accident. Core coolant flow drops rapidly, reaching a minimum value of 76.4 at about 1.09 second. The level swell produces a trip of both the main and feedwater turbines which, in turn, results in stop valve closure scram. The turbine trip, occurring after the time at which MCPR results, does not significantly retard the heat flux decrease and imposes no threat to fuel thermal limits. Considerations of uncertainties are included in the GETAB analysis.

6.B.3.4.6 Barrier Performance

The bypass valves and momentary opening of some of the safety/relief valves limit the pressure to well within the range allowed by the ASME vessel code. Therefore, the reactor coolant pressure boundary is not threatened by overpressure.

6.B.3.4.7 Radiological Consequences

The consequences of this event do not result in any fuel failures; however, radioactivity is nevertheless discharged to the suppression pool as a result of SRV activation.

6.B.3.5 Summary and Conclusions

The transient results for these initial cycles analyses are summarized in Table 6.B-5. This table indicates that for the transient events analyzed here, the MCPRs are well above the safety limit value of 1.06 (original analysis MCPR safety limit). It is concluded that the thermal margin safety limits established for two-pump operation are also applicable to single-loop-operation conditions.

For pressurization, Table 6.B-5 indicates that the peak pressures are below the ASME code value of 1375 psig. Hence, it is concluded that the pressure barrier integrity is maintained under single-loop-operation conditions.

6.B.4 OPERATING MCPR LIMIT

For single-loop operation, the rated condition steady-state MCPR limit is increased to account for the increase in the fuel cladding integrity safety limit (Section 6.B.2). At lower flows, the steady-state operating MCPR limit is conservatively established by a flow dependent MCPR. The operating limit is the more conservative of the two. This ensures that the 99.9% statistical limit requirement is always satisfied for any postulated abnormal operational occurrence.

6.B.5 STABILITY ANALYSIS

The least stable power/flow condition attainable under normal conditions occurs at natural circulation with the control rods set for rated power and flow. This condition may be reached following the trip of both recirculation pumps. As shown in Figure 6.B-5, operation along the minimum forced recirculation line with one pump running, at minimum speed, is more stable than operating with natural circulation flow only, but is less stable than operating with both pumps operating at minimum speed. Because of the increased flow fluctuation during one-recirculation-loop operation, the flow control should be left in manual operation to preclude unnecessary wear on the automatic controls.

6.B.6 Loss-of-Coolant Accident Analysis

An analysis of single recirculation loop operation utilizing the models and assumptions documented in Reference 4 was performed for the LSCS units. Using this method SAFER/GESTR-LOCA calculations were performed for the DBA. The SLO PCTs were calculated without a MAPLHGR reduction. GE determined the results were within the 10 CFR50.46 acceptance criteria. However, SLO without MAPLHGR reduction results in more limiting PCTs than the two loop LOCA.

Approval for single loop operation with regard to ECCS-LOCA response is based on an evaluation model methodology contingent on a single, overall bounding, licensing basis PCT being reported, which would demonstrate compliance for all allowed operating domains and regulatory requirements – break locations, break sizes, power distributions and power/flow conditions. Typical operation would presume all loops in service as a more representative initial condition when assuming occurrence of the loss of coolant accident. This is directed as the basis for the reported licensing basis PCT. To provide a complaint operating space with single loop operation, then, a power multiplier is defined that restricts SLO operation, based on the ECCS flow capacity for the single loop, consistent with assumptions of the limiting DBA event with two loops available. Operation within that power restriction is enforced by specification so that the PCT result for SLO based on nominal conditions will remain below the nominal PCT for the bounding case, the basis for the licensing basis PCT.

For LaSalle, the power multiplier was found and reported in Reference 5, upon insertion of GE14 fuel into the core, as 0.78. In Reference 6, this value for the power multiplier was confirmed to remain acceptably bound compared to two-loop operation with GNF2 fuel, based on the SAFER/PRIME-LOCA evaluation model.

6.B.6.1 Break Spectrum Analysis

For GE Fuel, SAFER/GESTR-LOCA calculations were performed for LaSalle Units 1 and 2 for SLO using very conservative and bounding assumptions given in Section 5.4 of Reference 5. The most limiting SLO break was consistent with the limiting

two-loop operation DBA recirculation suction side break, assuming single failure of the HPCS diesel generator. The licensing basis PCT for GE14 and GNF2 fuel bundles is determined from the more limiting 0.08 sq. ft. small recirculation line break.

6.B.6.2. Single-Loop MAPLHGR Determination

For GEH fuel, the SLO analysis assumes the same MAPLHGR limits as the two-loop operation DBA analysis. The affect is taken as a power multiplier limit applied under nominal assumptions. Demonstration of a bounded PCT for SLO initial conditions with the power multiplier limit assures compliance by virtue of the overall acceptable licensing basis PCT.

The accident response for the small break, for GEH fuel and evaluation models, is less affected by ECCS assets. Rather, it is characterized as a slower depletion of inventory until low level setpoints are reached. Then, depressurization is accomplished by the ADS. The result is a partial uncover of the core, with bounding PCT occurring under a top-peak power distribution assumption, when low pressure injection is able to be injected, initiating a recovery of the core. The core/vessel small break response is not appreciably different for two-loop vs. single-loop operation. Assuring the more affected DBA transient is bounding – under the assigned power multiplier, compared to two-loop operation – the small break result would consistently be bounding as well, as the PCT effect would be driven by the power differential. No SLO case is required for the small break under the GEH methodology.

6.B.6.3 Small Break Peak Cladding Temperature

Section 5.3.1 of Reference 4 discusses why the DBA break is more limiting than the smaller break sizes for SLO. Section 5.3.1 of Reference 4 also discusses the effect of the assumptions used in the one-pump operation analysis and the duration of nucleate boiling. GE did not calculate small break results for SLO because they are non-limiting.

6.B.7 REFERENCES

1. General Electric BWR Thermal Analysis Basis (GETAB): Data, Correlation, and Design Application, General Electric Company (NEDO-10958-A), January 1977.

2. Deleted

3. Deleted

4. “LaSalle County Station Units 1 and 2 SAFER/GESTR-LOCA Loss-of-Coolant Accident Analysis,” NEDC-32258P, General Electric Company, October 1993.

5. “LaSalle Units 1 and 2 SAFER/GESTR Loss-of-Coolant-Accident Analysis for GE14 Fuel”, GE-NE-0000-0022-8684-R2, November 2006.

6. “LaSalle County Station GNF2 ECCS-LOCA Evaluation”, 0000-0121-8990-R0, GE Hitachi, January 2012.

7. Deleted

INPUT PARAMETERS AND INITIAL CONDITIONS

FOR ANALYSIS OF INITIAL CORE TRANSIENTS AND ACCIDENTS

FOR SINGLE-LOOP OPERATION

(INITIAL CORE VALUES)**

1. Thermal Power Level, Analysis Value, % NBR	78
2. Steam Flow, lb/h	10.71×10^6
3. Core Flow, lb/h	68.26×10^6
4. Feedwater Flow Rate, lb/sec	2976
5. Feedwater Enthalpy, Btu/lb	367.3
6. Vessel Dome Pressure, psig	1001
7. Vessel Core Pressure, psig	1006
8. Turbine Bypass Capacity, % NBR	25
9. Core Coolant Inlet Enthalpy, Btu/lb	516.8
10. Turbine Inlet Pressure, psig	969.3
11. Fuel Lattice	8 x 8
12. Core Average Gap Conductance, Btu/sec-ft^2-°F	0.1662
13. Core Leakage Flow, %	12
14. Required MCPR Operating Limit	1.41 *
15. MCPR Safety Limit	1.06
16. Doppler Coefficient, -¢/°F	
Nominal EOC-1	0.221
Analysis Data	0.221
17. Void Coefficient, -¢/% Voids	
Nominal EOC-1	7.429
Analysis Data for Power Increase Events	12.63
Analysis Data for Power Increase Events	7.01
18. Core Average Rated Void Fraction, %	0.414
19. Scram Reactivity, Analysis Data	FSAR Figure 15.0-2
20. Control Rod Drive Speed, position versus time	FSAR Figure 15.0-2

* Dual-pump operation operating limit for 63% core flow, obtained by applying K_f-curve to operating limit CPR at rated condition (1.24).

** For cycle specific inputs, see the transient analysis input parameters.

TABLE 6.B-1
(SHEET 2 OF 2)

(INITIAL CORE VALUES)

21. Jet Pump M Ratio	3.20
22. Safety/Relief Valve Capacity, % NBR at 1165 psig	111.5
Manufacturer	Crosby
Quantity Installed	18
23. Relief Function Delay, sec	0.1
24. Relief Function Response, sec	0.1
25. Setpoints for Safety/Relief Valves	
Safety Function, psig	1150, 1175, 1185, 1195, 1205
Relief Function, psig	1076, 1086, 1096, 1106, 1116
26. Number of Valve Groupings Simulated	
Safety Function, No.	5
Relief Function, No.	5
27. Vessel Level Trips, Inches above Steam Dryer Skirt Bottom (Instrument Zero)	
Level 8 - (L8)	55.5
Level 3 - (L3)	12.5
Level 2 - (L2)	-50
28. RPT Delay, sec	0.14
29. RPT Inertia Time Constant for Analysis, sec	6.0

TABLE 6.B-2

SEQUENCE OF EVENTS FOR FIGURE 6.B-3
(INITIAL CORE RESULTS)

TIME (sec)	EVENT
0	Initiate simulated failure of 160% upper limit on feedwater flow.
5.46	L8 vessel level setpoint trips main turbine and feedwater pumps.
5.47	Reactor scram trip actuated from main turbine stop valve position switches.
5.47	Recirculation pump (RPT) actuated by turbine stop valve position switches.
5.57	Main turbine stop valves closed and main turbine bypass valves start to open.
8.01, 8.29	Relief valves actuated (groups 1, 2).
11.67, 12.23	Relief valves close (groups 2, 1).
29.32	Main turbine bypass valves closed.
48.35	Main turbine bypass valves start to open.

TABLE 6.B-3

SEQUENCE OF EVENTS FOR FIGURE 6.B-4
(INITIAL CORE RESULTS)

TIME (sec)	EVENT
-0.015 (approx)	Turbine-generator detection of loss of electrical load
0	Turbine-generator power load unbalance (PLU) devices trip to initiate turbine control valve fast closure
0	Turbine bypass valves fail to operate
0	Fast control valve closure (FCV) initiates scram trip
0	Fast control valve closure (FCV) initiates recirculation pump trip (RPT)
0.039	Turbine control valves closed
0.14	Recirculation pump motor circuit breakers open, causing decrease in core flow to natural circulation
1.98, 2.12, 2.27, 2.45, 2.74	Relief valves actuated (groups 1, 2, 3, 4, 5)
4.58, 4.91, 5.20 (est)	Relief valves close (groups 5, 4, 3)
5.30	Vessel level reaches L8 setpoint, feed water pumps tripped (not simulated)
5.50, 5.84 (est)	Relief valves close (groups 2, 1)
12.00	Relief valves actuated (group 1)
19.0 (est)	Relief valves close (group 1)
33 2	Relief valves actuated (group 1)
38.0 (est)	Relief valves close (group 1)

TABLE 6.B-4

SEQUENCE OF EVENTS FOR FIGURE 6.B-5
(INITIAL CORE RESULTS)

TIME (sec)	EVENT
0	Single pump seizure was initiated, core flow decreases to natural recirculation
1.23	Reverse flow ceases in the idle loop
4.93	High vessel water level (L8) trip initiates main turbine trip
4.93	High vessel water level (L8) trip initiates feedwater turbine trip
4.93	Main turbine trip initiates bypass operation
4.96	Main turbine valves reach 90% open position and initiate reactor scram trip
5.03	Turbine stop valves closed and turbine bypass valves start to open to regulate pressure
10.0 (est)	Turbine bypass valves start to close
25.1	Turbine bypass valves closed
38.6	Turbine bypass valves reopen on pressure increase at turbine inlet

TABLE 6.B-5

SUMMARY OF EVENT RESULTS

SINGLE RECIRCULATION LOOP OPERATION
(Typical)

PARAGRAPH	FIGURE	DESCRIPTION	MAXIMUM NEUTRON FLOW (% NBR)	MAXIMUM DOME PRESSURE (psig)	MAXIMUM VESSEL PRESSURE (psig)	MAXIMUM STEAMLINE PRESSURE (psig)	MAXIMUM CORE AVERAGE SURFACE HEAT FLUX (% of Initial)	MCPR	FREQUENCY* CATEGORY
6.B.3.2	6.B-3	Feedwater flow Controller Failure (Maximum Demand)	119.3	1112	1126	1103	108.8	1.26	a
6.B.3.3	6.B-4	Generator Load Rejection	135.6	1138	1153	1128	103.5	1.29	b
6.B.3.4	6.B-5	Seizure of Active Recirculation Pump	78.0	1021	1031	1018	100.0	1.17	c

* a = incident of moderate frequency; b = infrequent incident; c = limiting faults

ATTACHMENT 6.C

HISTORICAL BASE ANALYSIS

ATTACHMENT 6.C

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ATTACHMENT 6.C

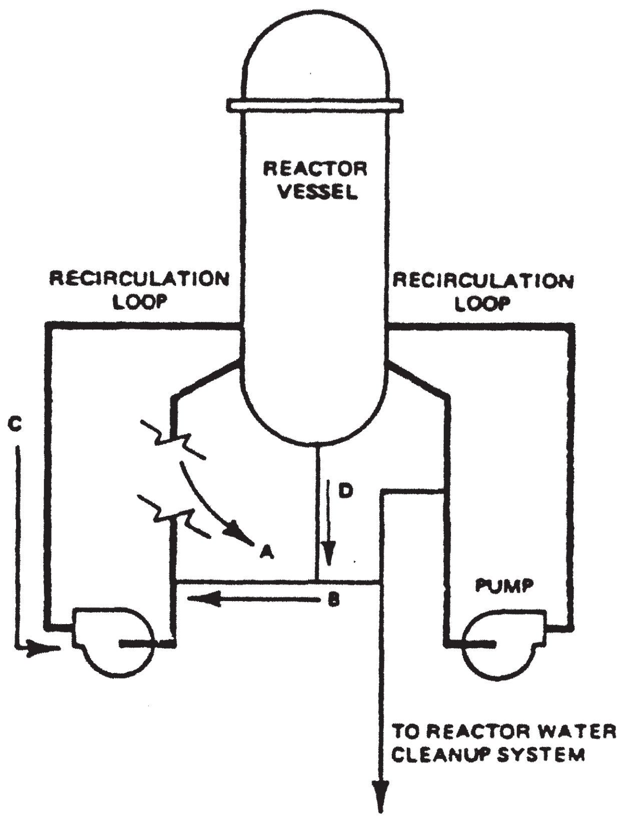
LIST OF FIGURES

6.C HISTORICAL BASE ANALYSIS

6.C.1 INTRODUCTION AND SUMMARY

This section documents the base analyses done prior to power uprate at 3434 MWt – see Figures 6.C-1 through 6.C-4. This section is retained for its historical information only. Section 6.2 describes the analyses and results for these base analyses performed at 3434 MWt.

POINT OF CRITICAL FLOW	
A.	RECIRCULATION LINE
B.	CLEANUP LINE
C.	COMBINED AREA OF ALL JET PUMP NOZZLES ASSOCIATED WITH THE BROKEN LOOP
D.	BOTTOM HEAD DRAIN



SCHEMATIC SHOWING COMPOSITION OF TOTAL RECIRCULATION LINE BREAK AREA

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FIGURE 6.2-1 DIAGRAM OF THE RECIRCULATION LINE BREAK LOCATION

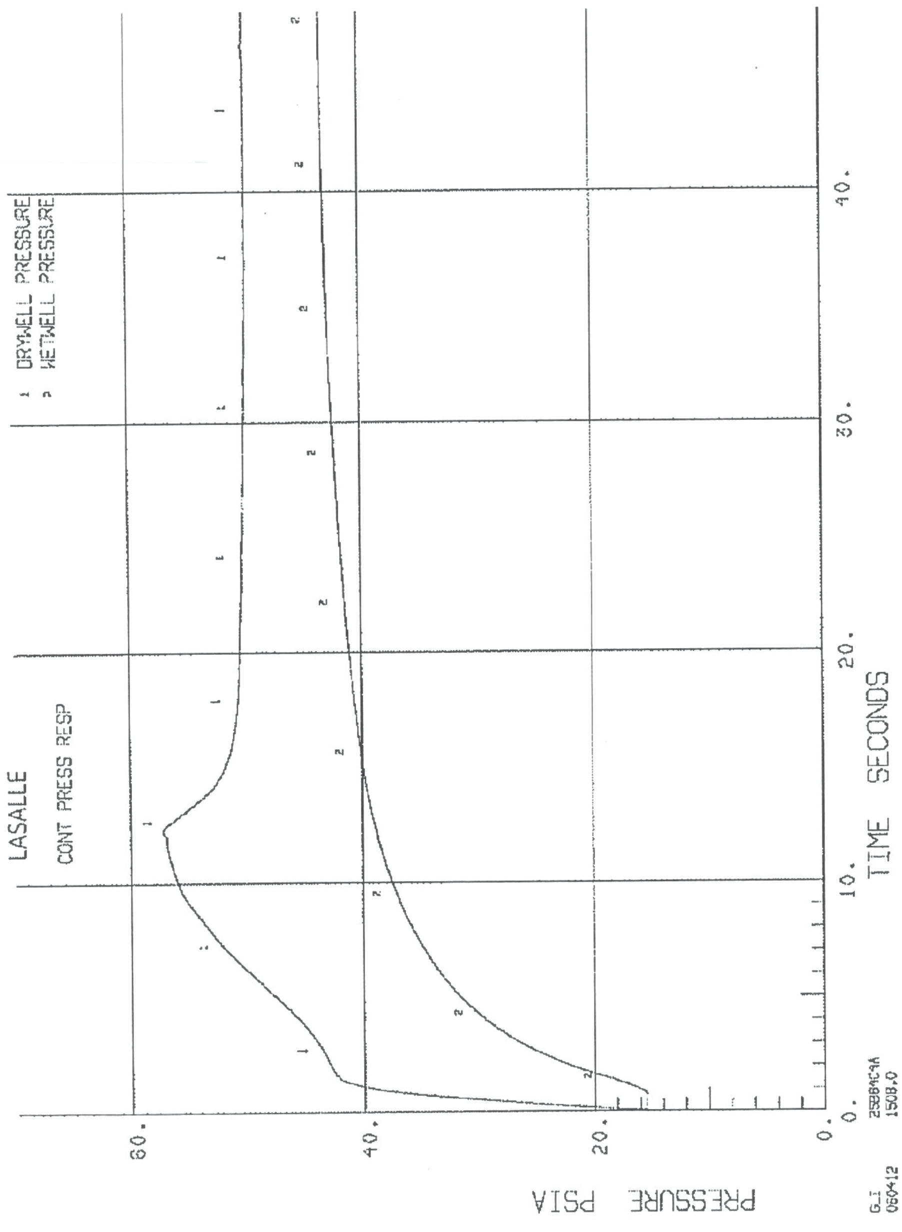

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FIGURE 6.2-2

SHORT-TERM PRESSURE RESPONSE FOLLOWING A RECIRCULATION LINE BREAK (At 3559 MWt)

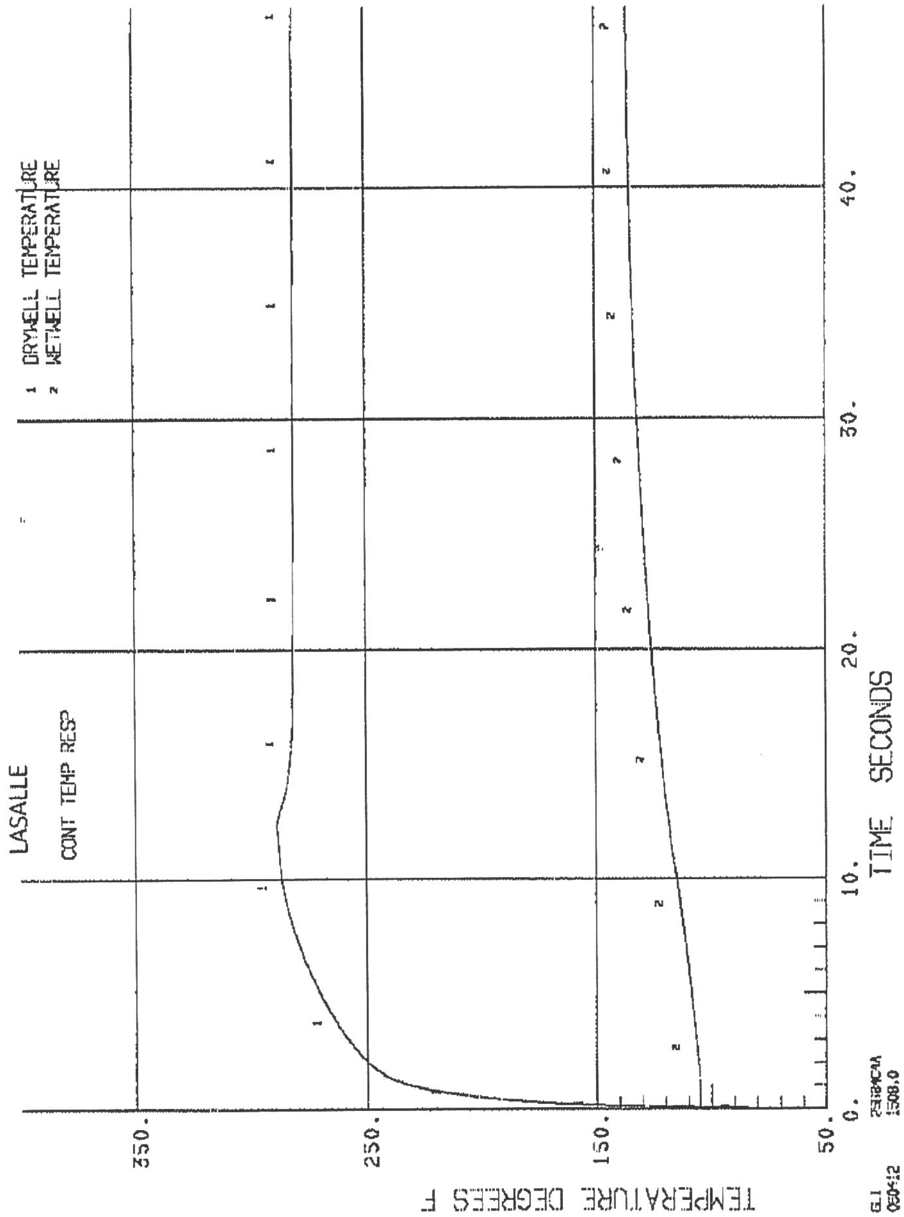

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FIGURE 6.2-3

SHORT-TERM TEMPERATURE RESPONSE FOLLOWING A RECIRCULATION LINE BREAK (At 3559 MWt)

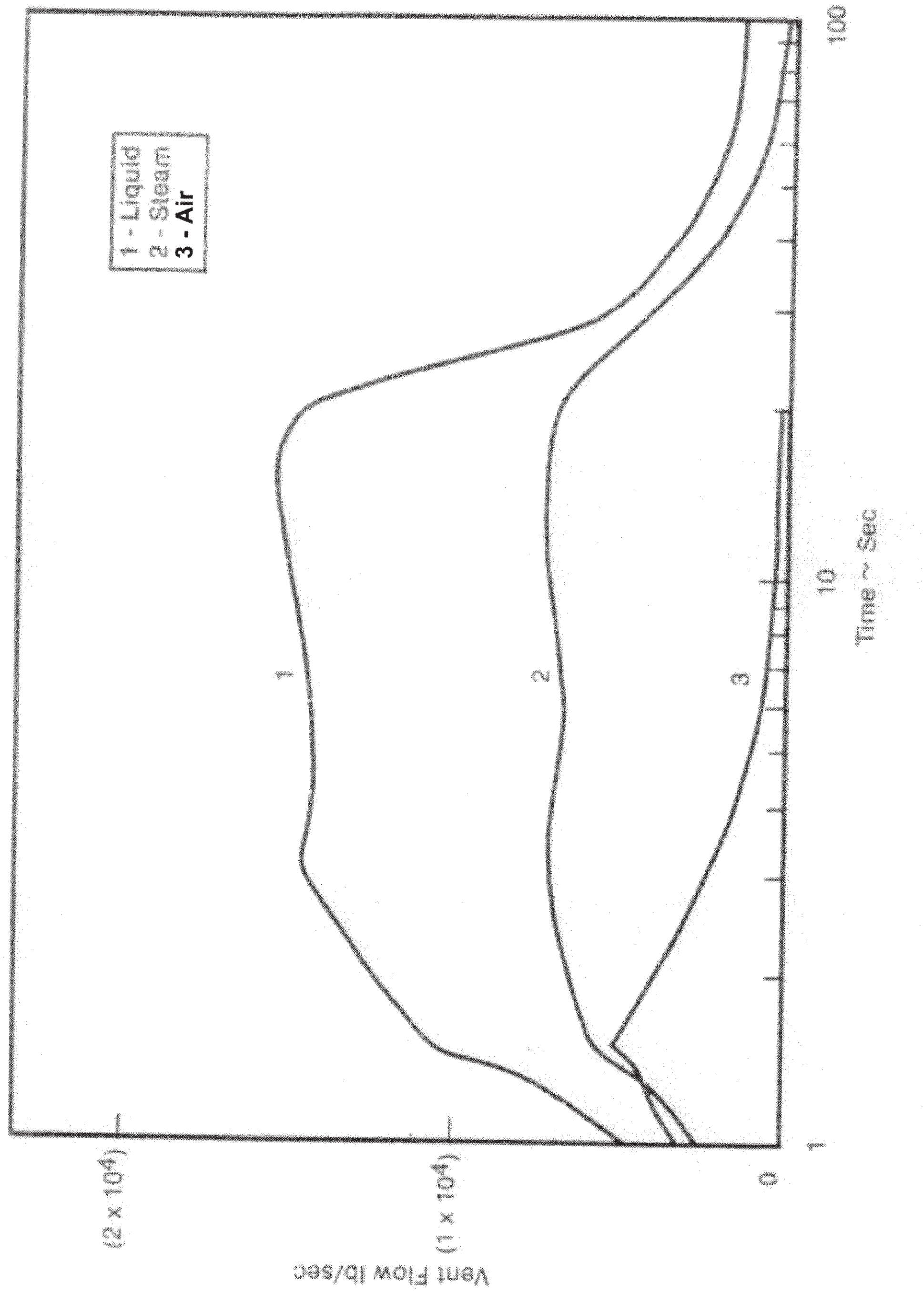

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FIGURE 6.2-4 CONTAINMENT VENT SYSTEM FLOW RATE VS. TIME FOR RECIRCULATION LINE BREAK (At 3434 MWt)

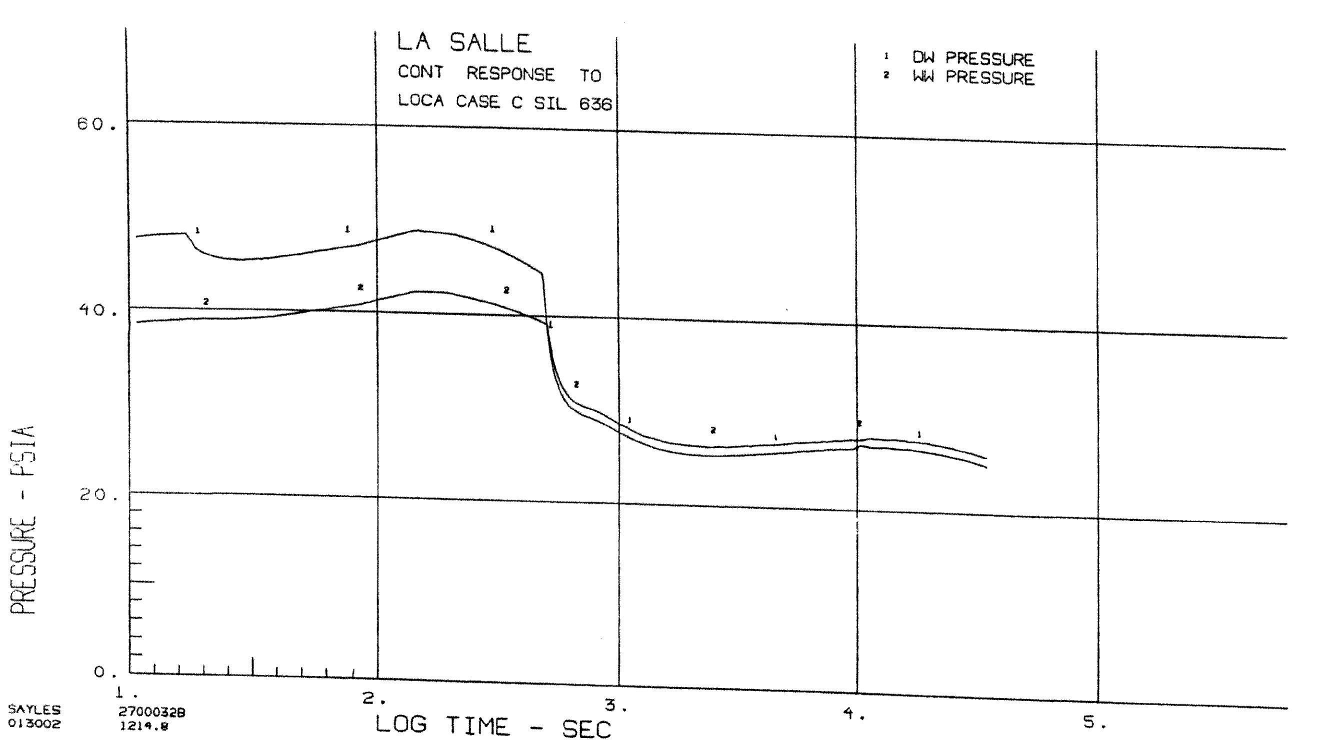

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FIGURE 6.2-5a

LONG TERM CONTAINMENT PRESSURE RESPONSE FOLLOWING A RECIRCULATION
LINE BREAK (At 3559 MWt)
CASE C (2 PUMPS, 1 HEAT EXCHANGER WITHOUT CONTINUOUS SPRAY)

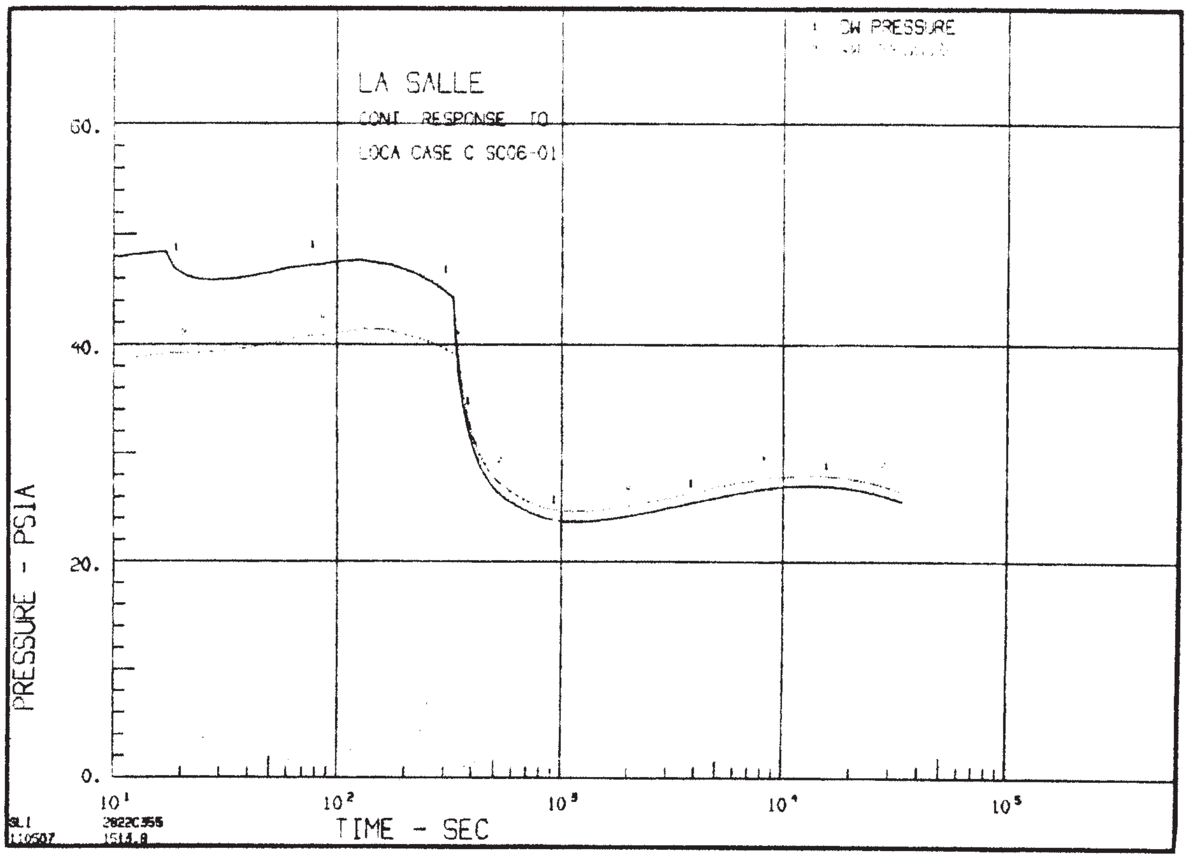

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FIGURE 6.2-5b

LONG TERM CONTAINMENT PRESSURE RESPONSE FOLLOWING A RECIRCULATION LINE BREAK (At 3559 MWt)

Case C with GE SC06-01 CONSIDERATION (5 PUMPS, 1 HEAT EXCHANGER WITHOUT CONTINUOUS SPRAY)

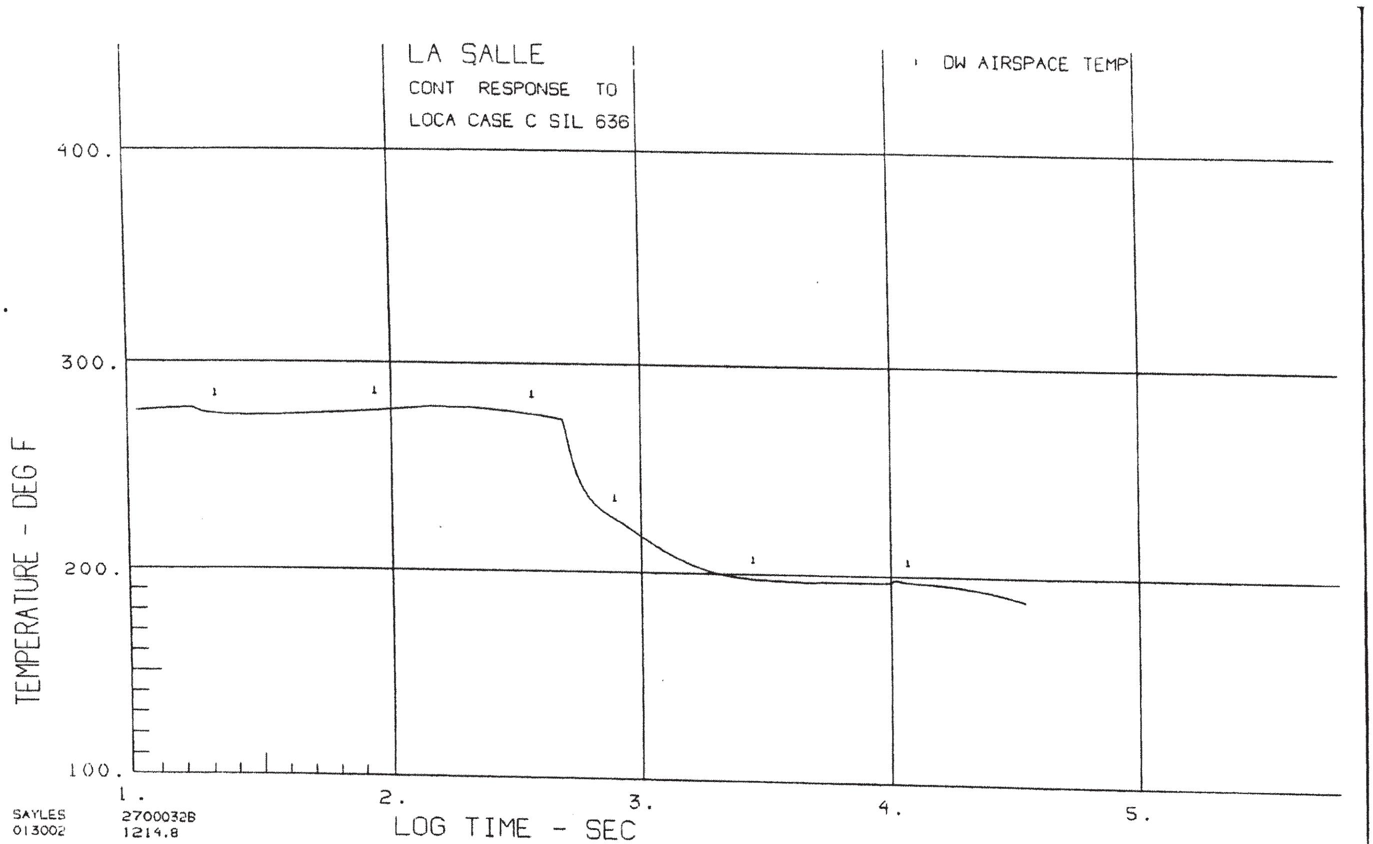

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FIGURE 6.2-6a LONG TERM DRYWELL TEMPERATURE RESPONSE FOLLOWING A RECIRCULATION LINE BREAK (At 3559 MWt) CASE C (2 PUMPS, 1 HEAT EXCHANGER WITHOUT CONTINUOUS SPRAY)

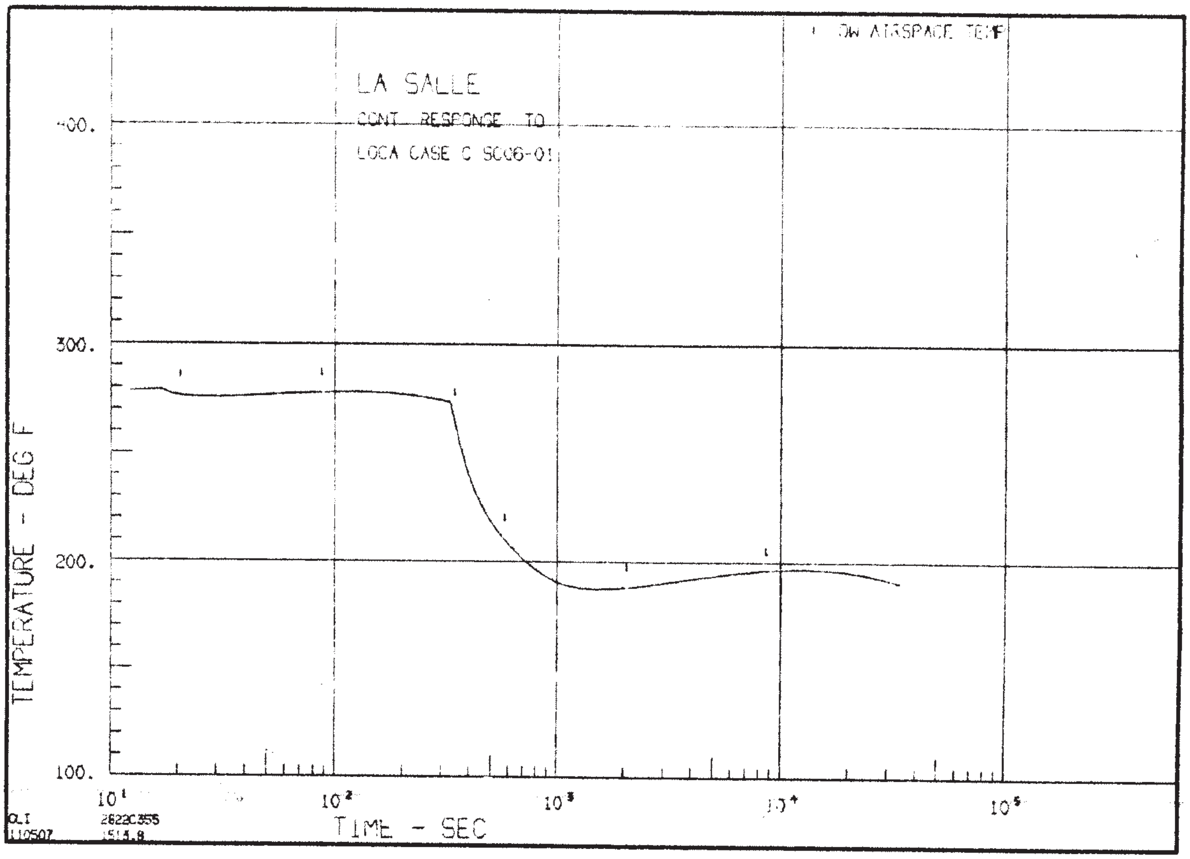

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FIGURE 6.2-6b

LONG TERM DRYWELL TEMPERATURE RESPONSE
FOLLOWING A RECIRCULATION
LINE BREAK (At 3559 MWt)

Case C with GE SC06-01 CONSIDERATION
(5 PUMPS, 1 HEAT EXCHANGER WITHOUT CONTINUOUS SPRAY)

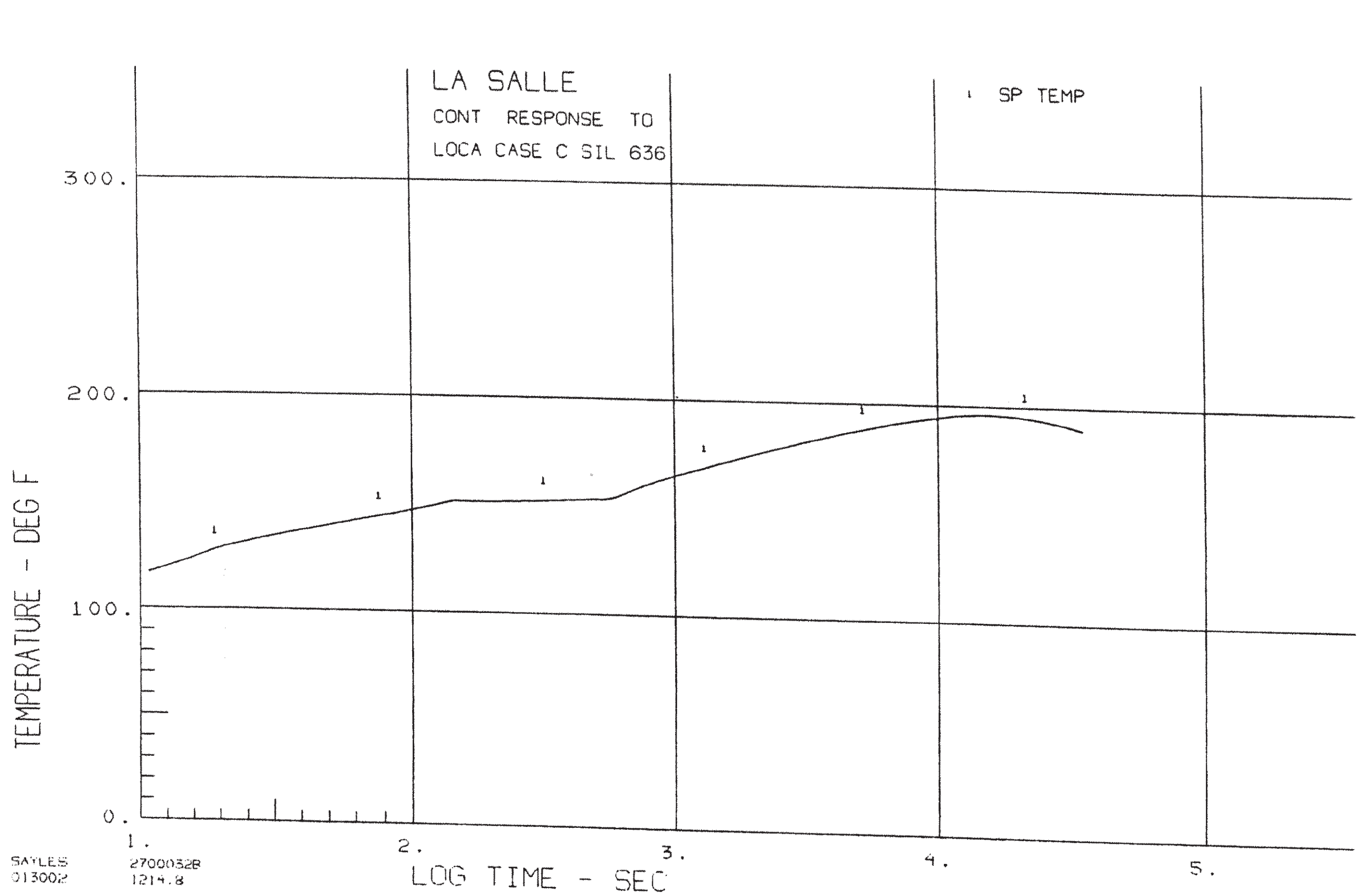

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FIGURE 6.2-7a LONG TERM SUPPRESSION POOL RESPONSE FOLLOWING A RECIRCULATION LINE BREAK (At 3559 MWt) CASE C (2 PUMPS, 1 HEAT EXCHANGER WITHOUT CONTINUOUS SPRAY)

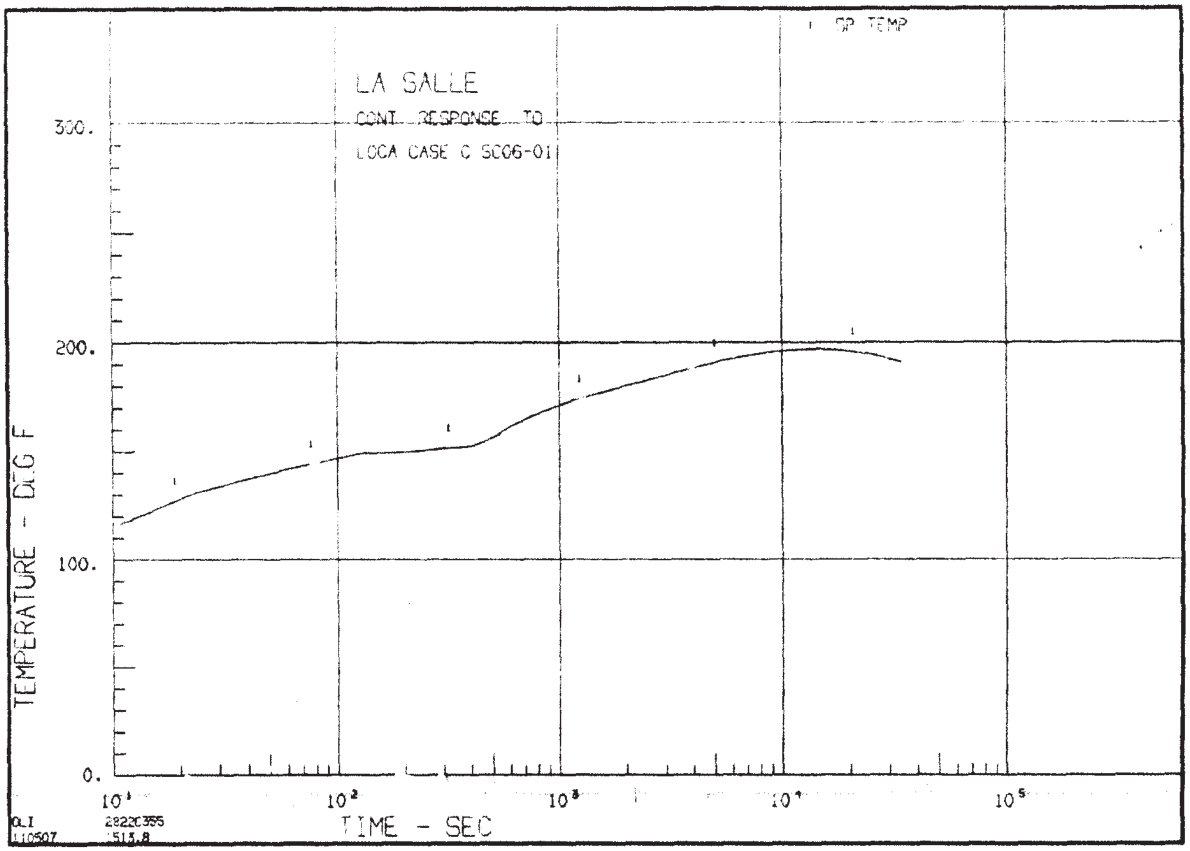

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FIGURE 6.2-7b

LONG TERM SUPPRESSION POOL RESPONSE
FOLLOWING A RECIRCULATION
LINE BREAK (At 3559 MWt)

Case C with GE SC06-01 CONSIDERATION
(5 PUMPS, 1 HEAT EXCHANGER WITHOUT CONTINUOUS SPRAY)

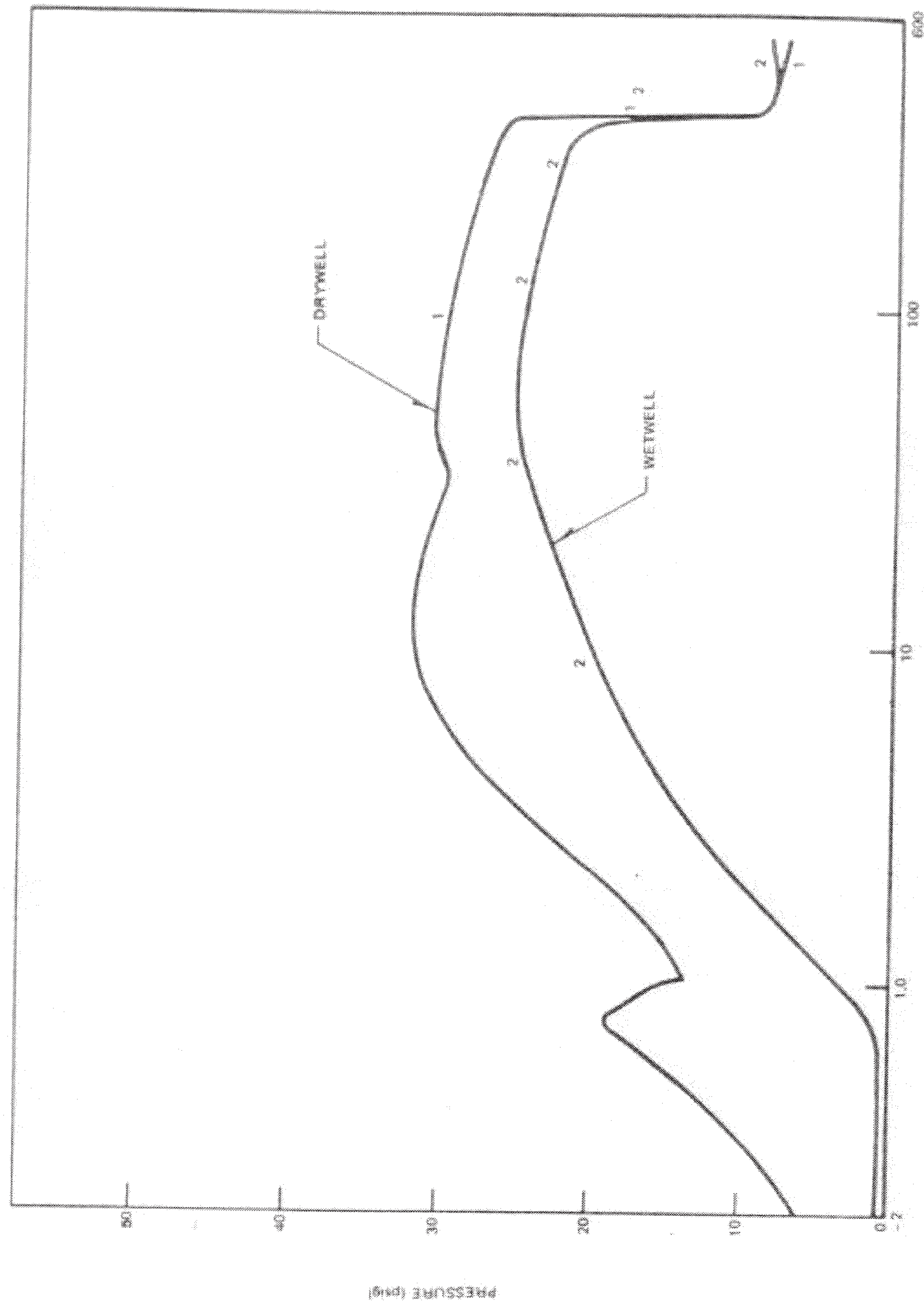

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FIGURE 6.2-8

PRESSURE RESPONSE FOR A MAIN STEAMLINE BREAK

(At 3434 MWt)

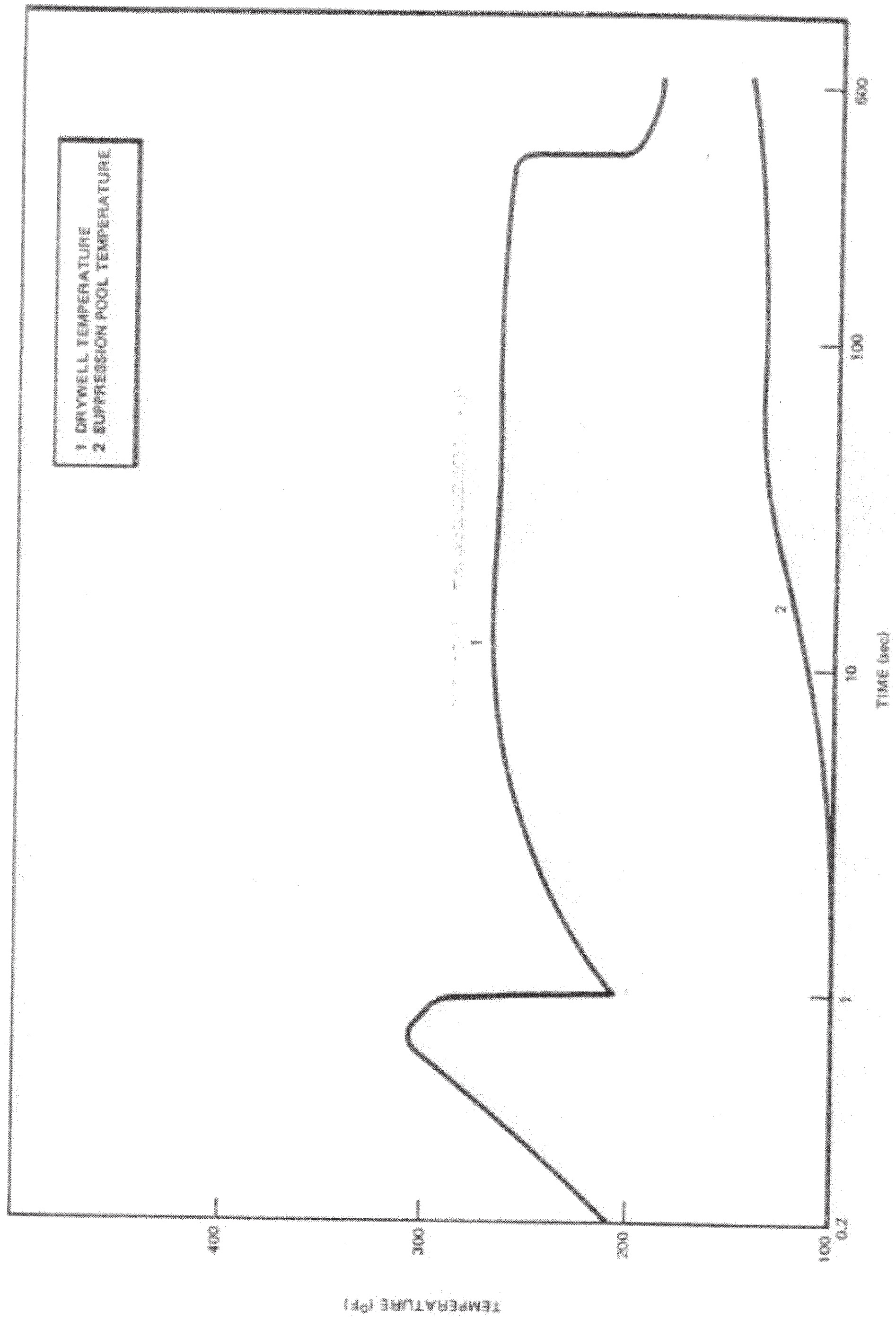

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FIGURE 6.2-9

TEMPERATURE RESPONSE FOLLOWING
A MAIN STEAMLINE BREAK
(At 3434 MWt)

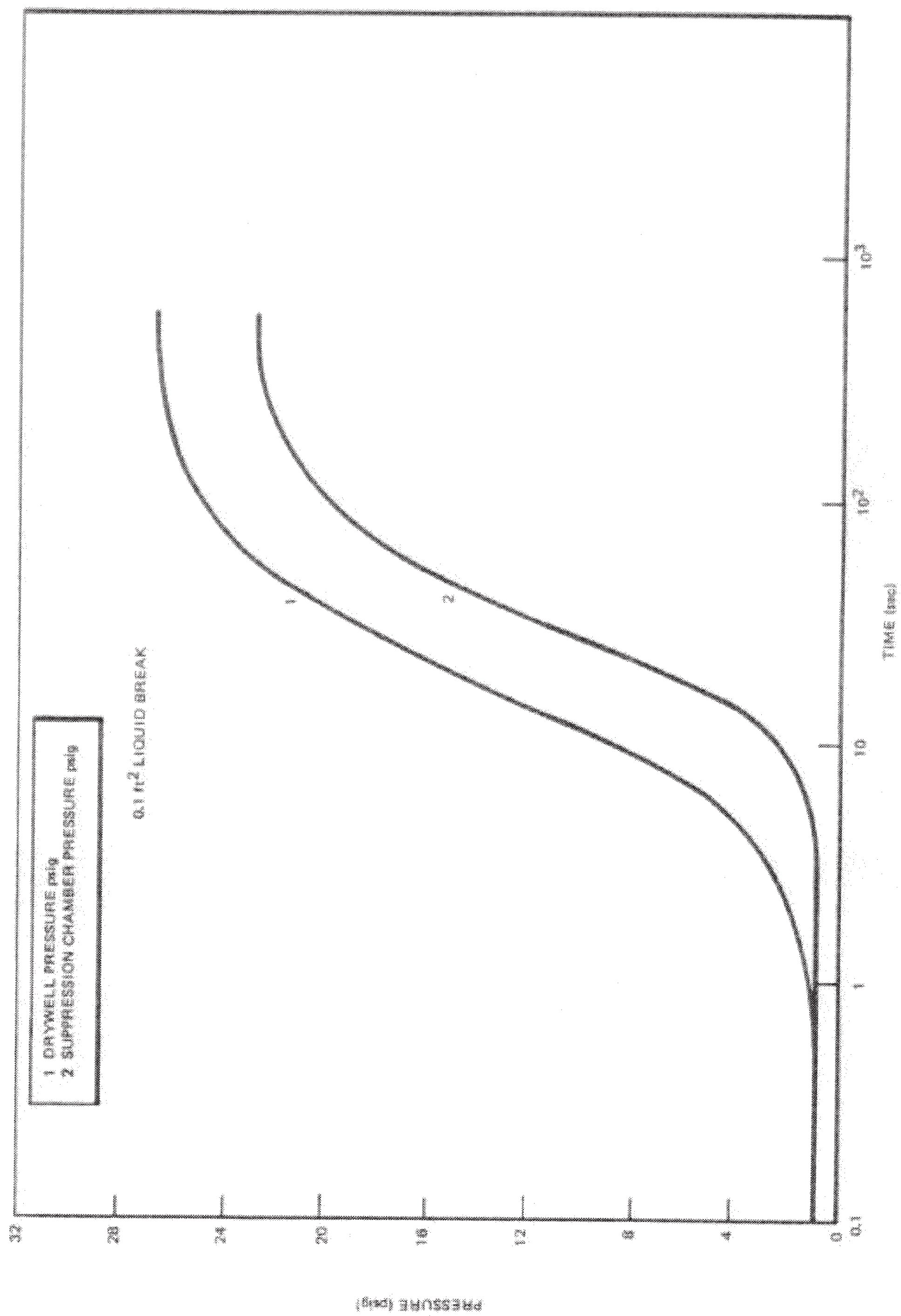

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FIGURE 6.2-10

PRESSURE RESPONSE FOR 0.1 FT^2

LIQUID LINE BREAK

(At 3434 MWt)

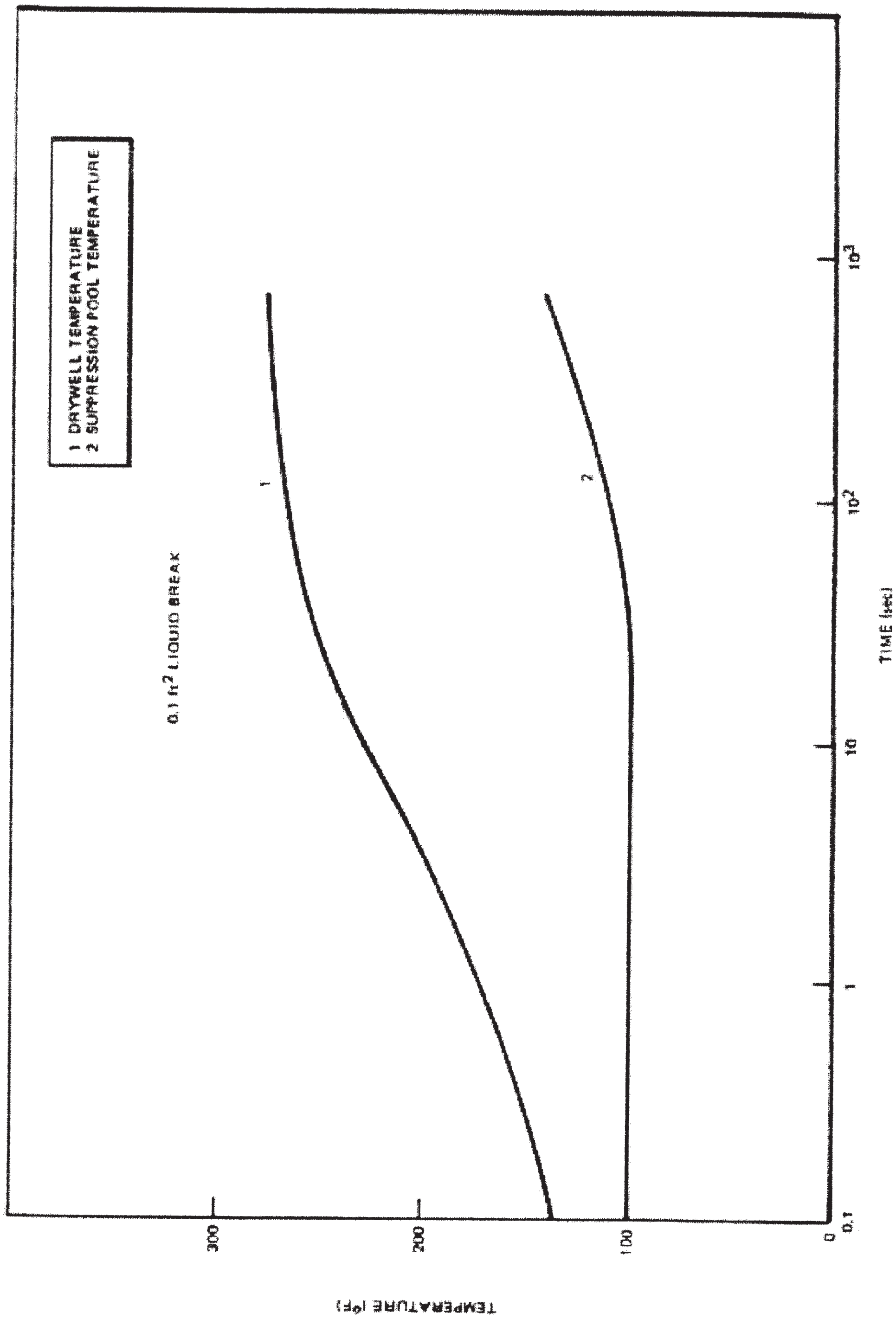

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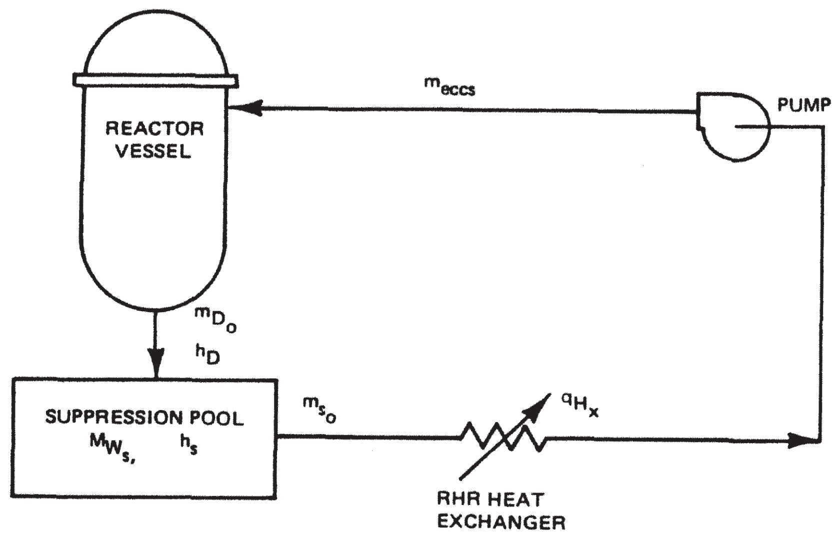
FIGURE 6.2-11 TEMPERATURE RESPONSE FOR 0.1 FT² LIQUID LINE BREAK (At 3434 MWt)



h_D	=	ENTHALPY OF WATER LEAVING REACTOR, Btu/lb
m_{D_o}	=	FLOW RATE OUT OF REACTOR, lb/sec
h_s	=	ENTHALPY OF WATER IN SUPPRESSION POOL, Btu/lb
m_{s_o}	=	FLOW OUT OF SUPPRESSION POOL, lb/sec
q_{H_x}	=	HEAT REMOVAL RATE OF HEAT EXCHANGER, Btu/sec
M_{W_s}	=	MASS OF WATER IN SUPPRESSION POOL
q_D	=	CORE DECAY HEAT RATE, Btu/sec
q	=	STORED ENERGY RELEASE RATE, Btu/sec

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FIGURE 6.2-12

SCHEMATIC OF ECCS LOOP

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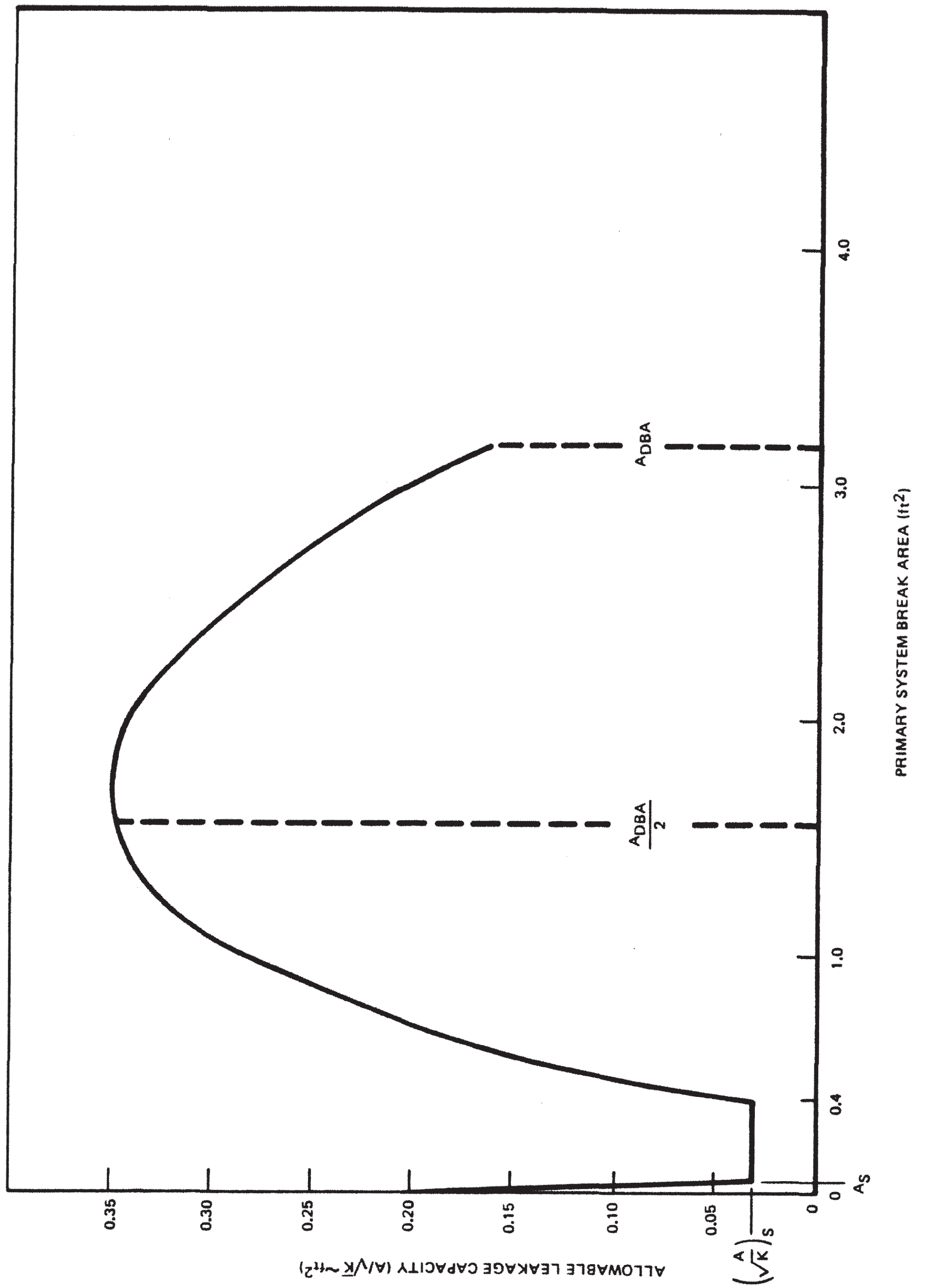

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FIGURE 6.2-13

ALLOWABLE STEAM BYPASS
LEAKAGE CAPACITY

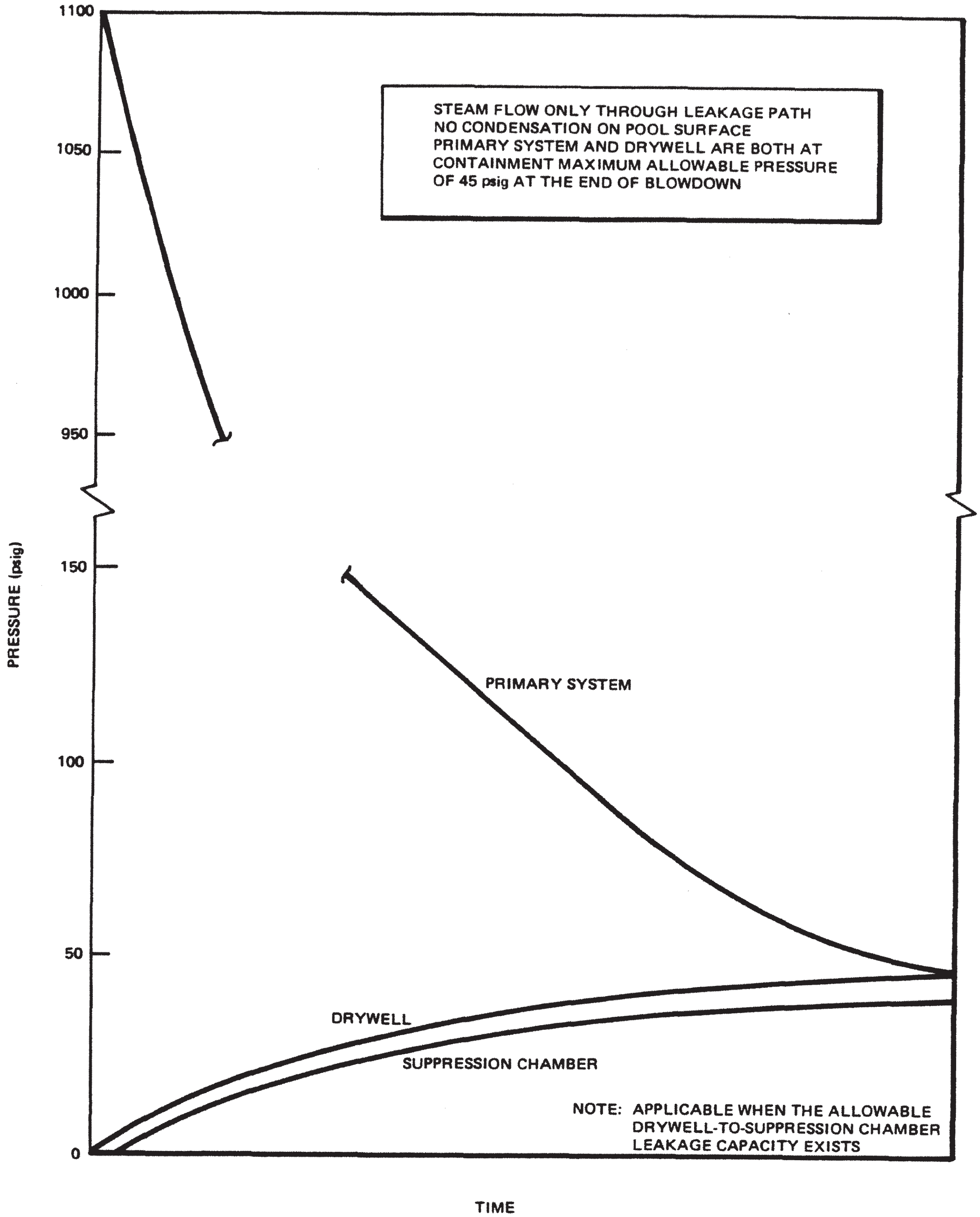

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FIGURE 6.2-14

CONTAINMENT RESPONSE TO LARGE
PRIMARY SYSTEM BREAKS

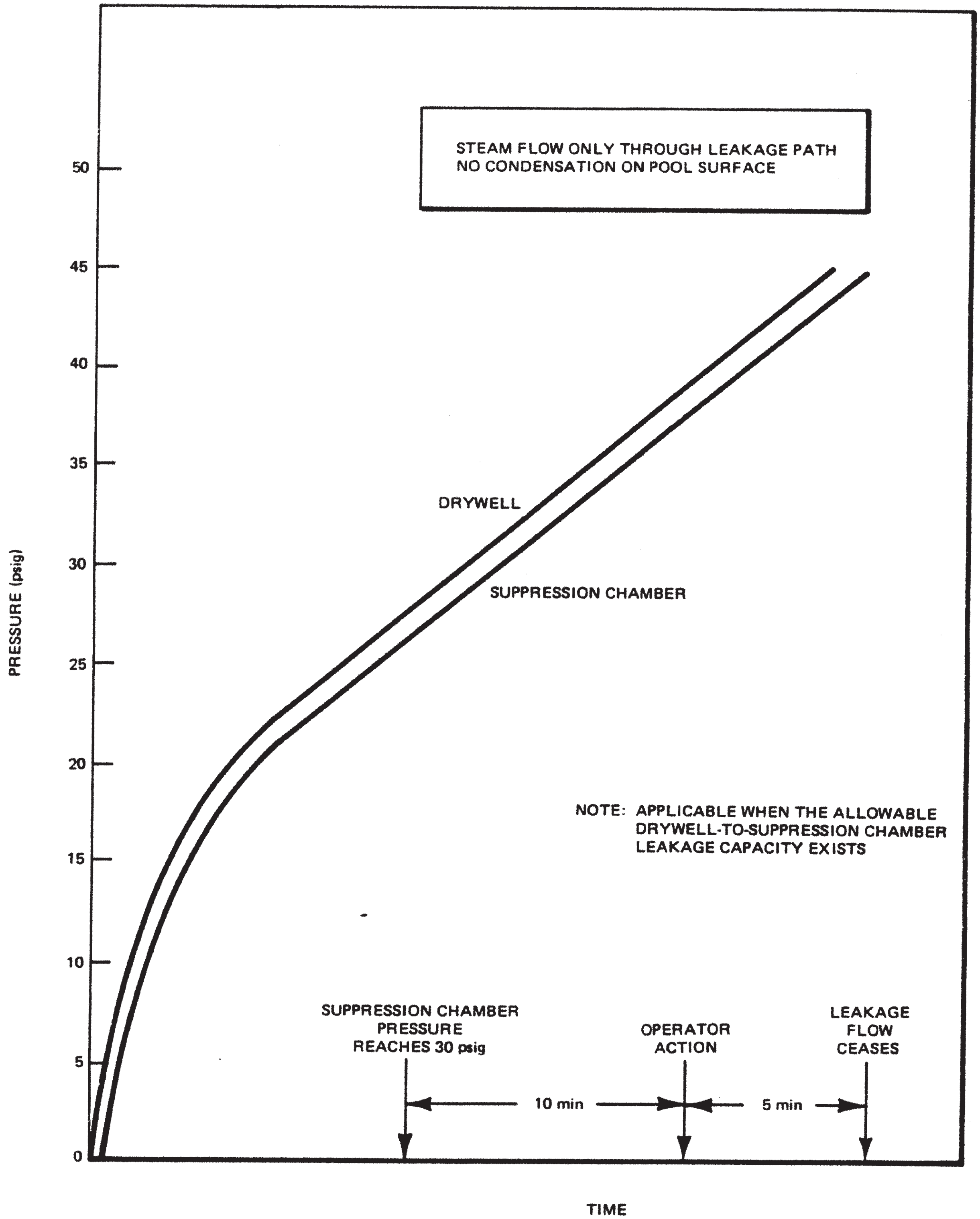

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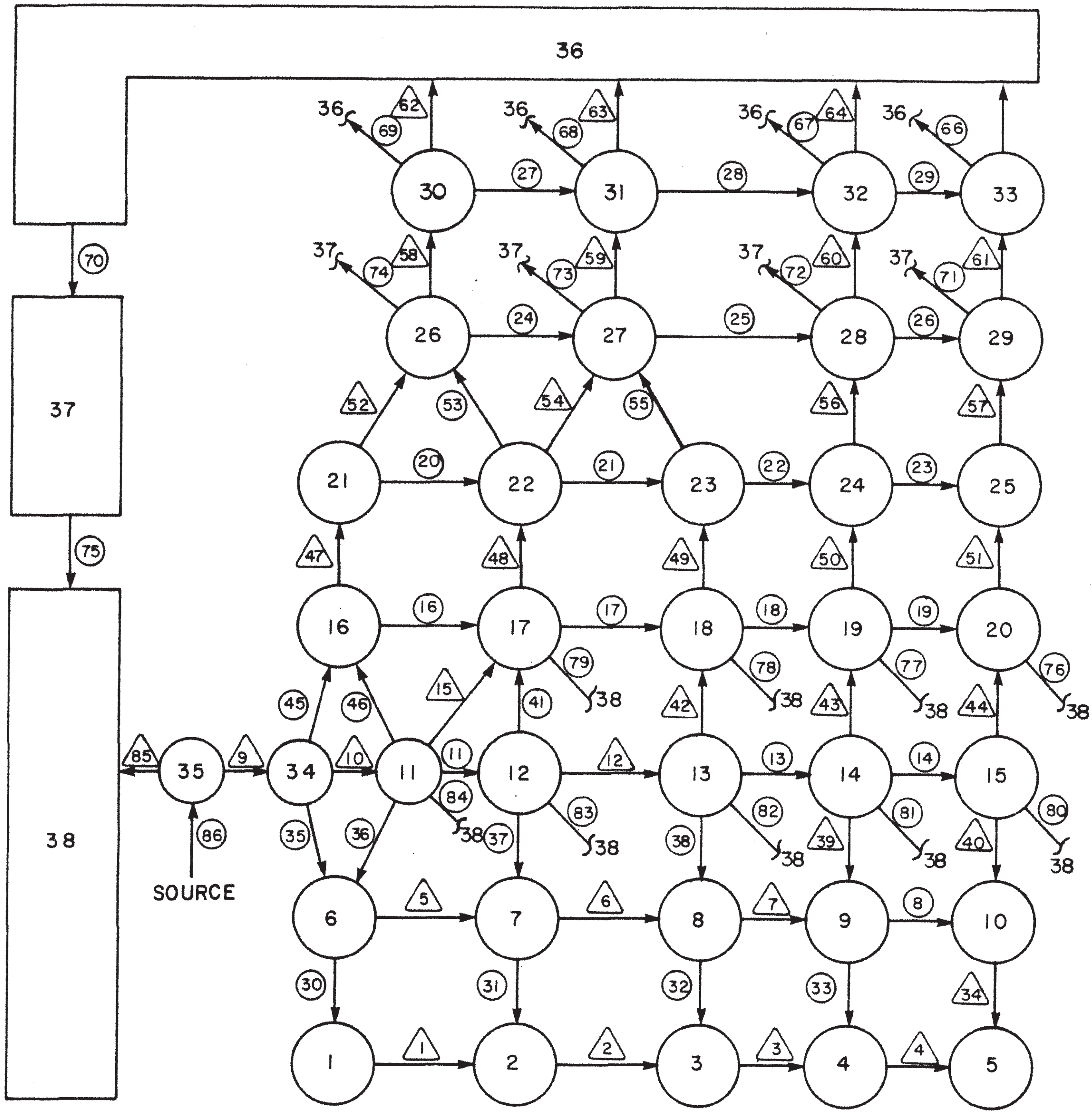
FIGURE 6.2-15

CONTAINMENT RESPONSE TO SMALL PRIMARY SYSTEM BREAKS

REV. 0 - APRIL 1984



INDICATES NODE

INDICATES INCOMPRESSIBLE VENT PATH

INDICATES COMPRESSIBLE VENT PATH

VENT PATH TO CONTAINMENT

LA SALLE COUNTY STATION
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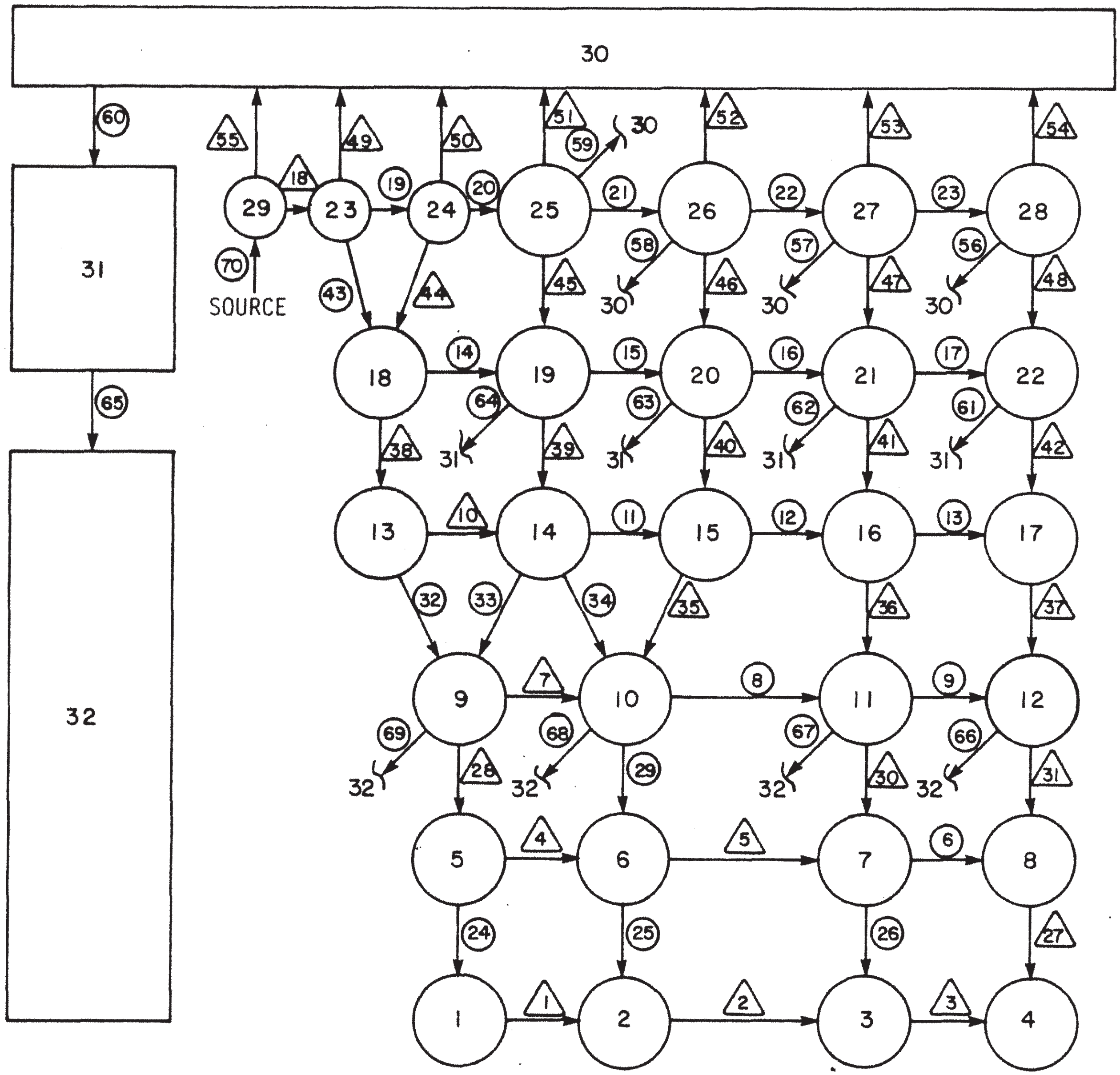
FIGURE 6.2-16

NODALIZATION SCHEMATIC FOR RECIRCULATION LINE BREAK

REV. 0 - APRIL 1984



INDICATES NODE

INDICATES INCOMPRESSIBLE VENT PATH

INDICATES COMPRESSIBLE VENT PATH

* VENT PATH TO CONTAINMENT

LA SALLE COUNTY STATION
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE 6.2-17

NODALIZATION SCHEMATIC FOR
FEEDWATER LINE BREAK

REV. 0 - APRIL 1984

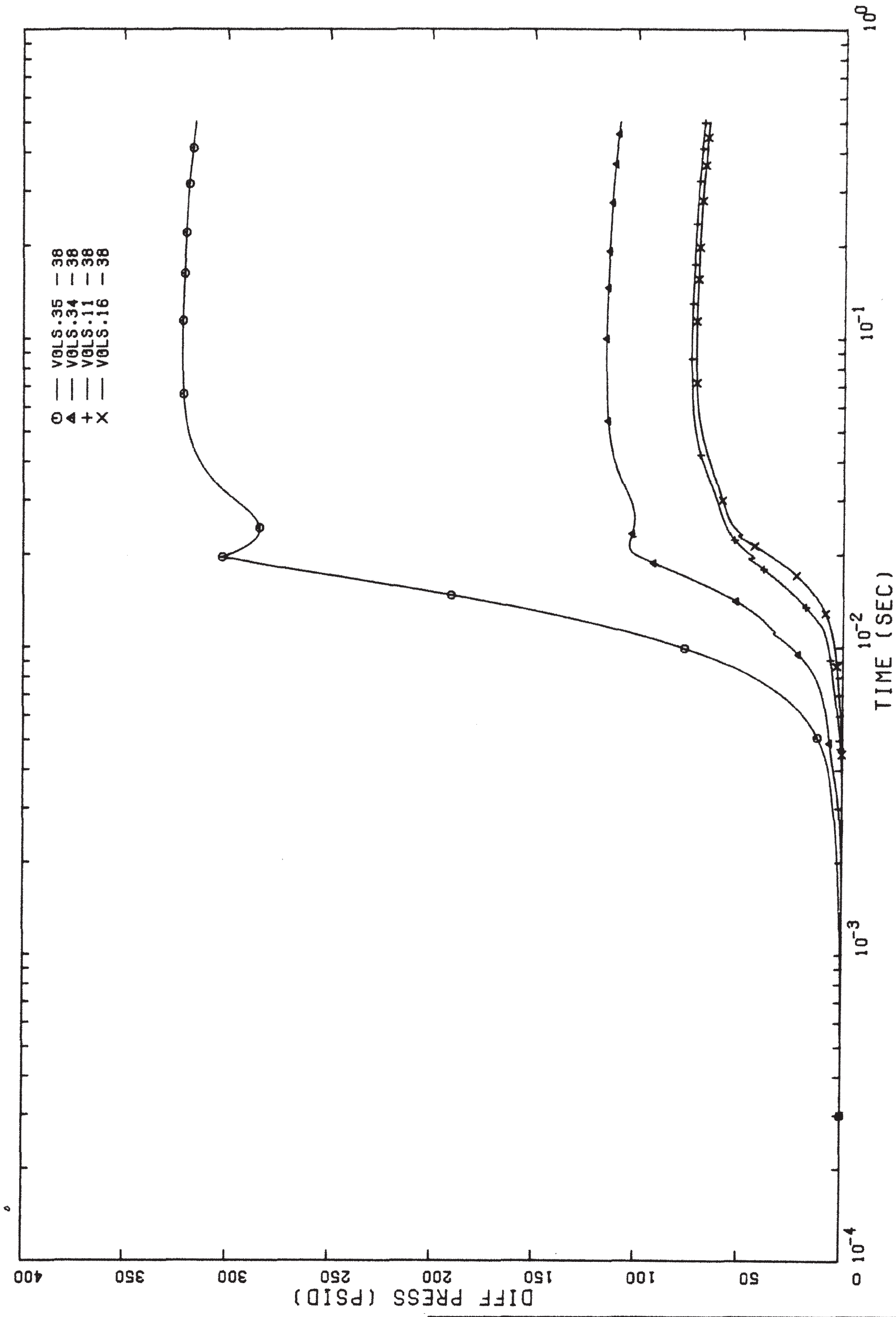


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FIGURE 6.2-18

ΔP VS. LOG t ABOUT BREAK -
RECIRCULATION LINE BREAK

REV. 0 - APRIL 1984

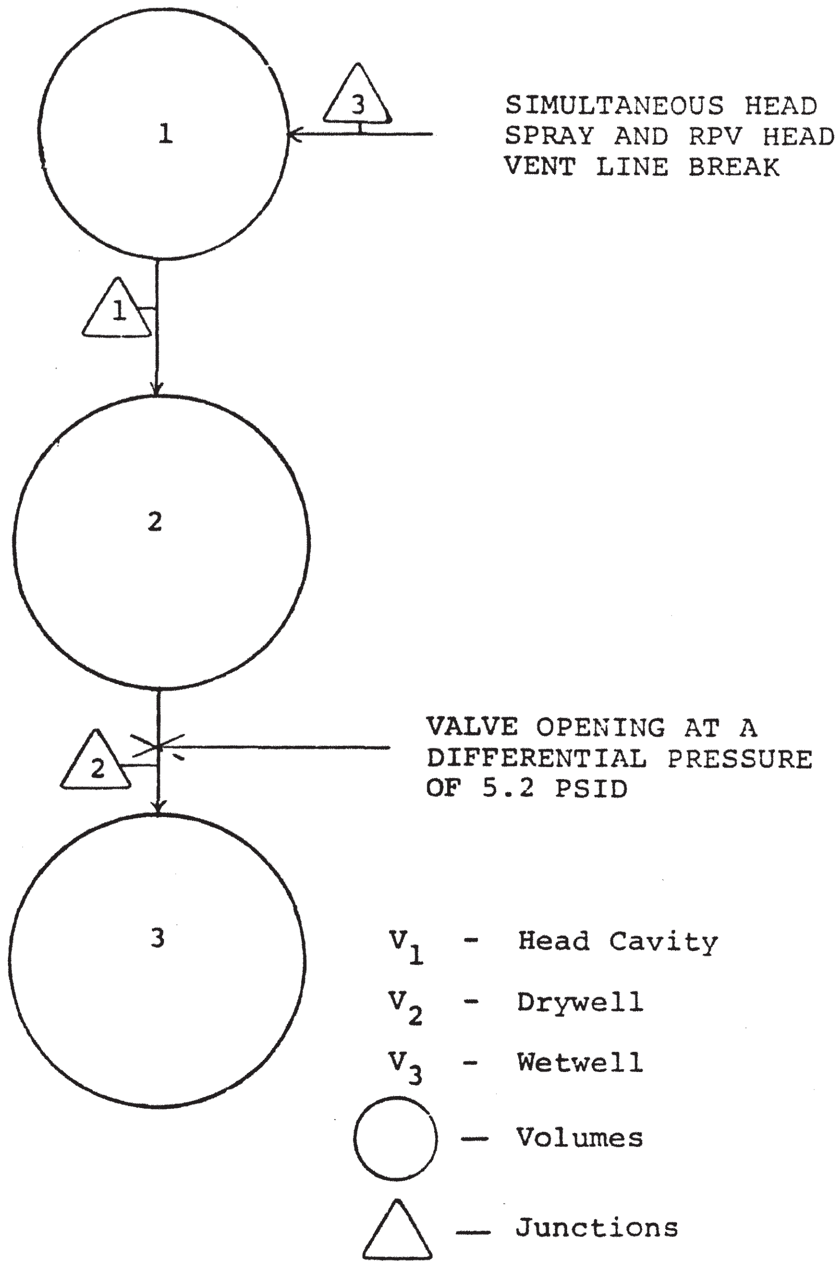

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FIGURE 6.2-19

HEAD SPRAY LINE BREAK NODALIZATION

REV. 0 - APRIL 1984

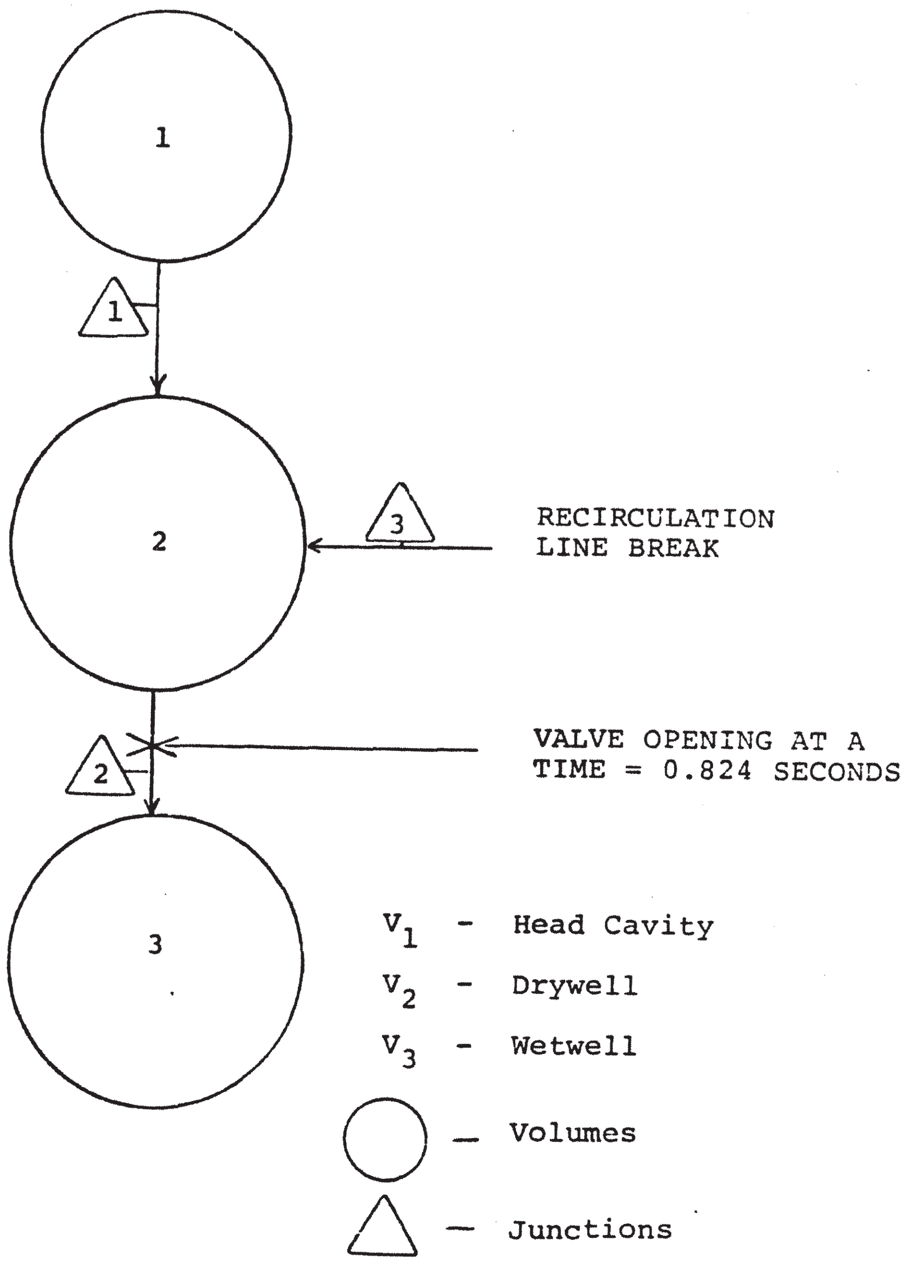

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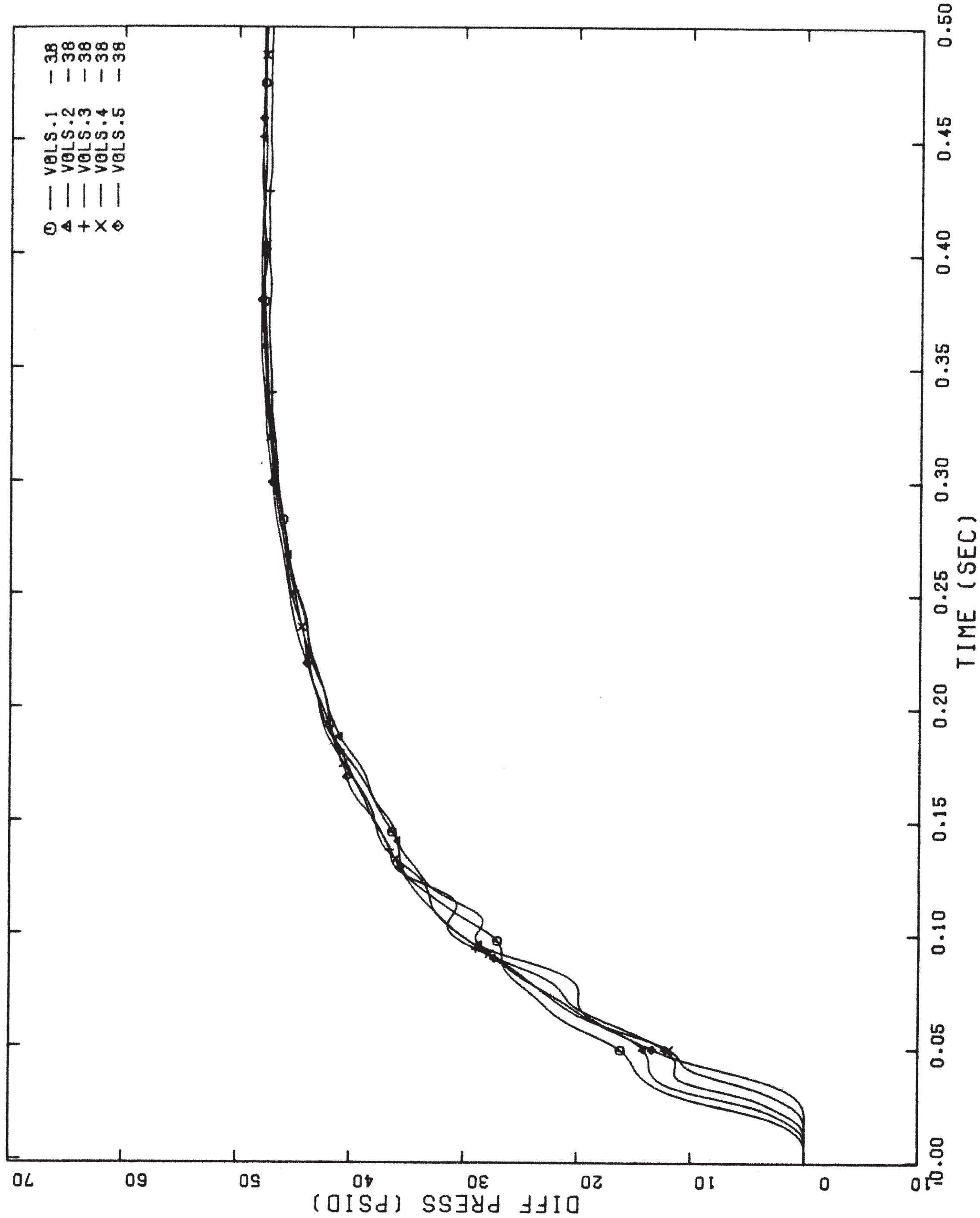
FIGURE 6.2-20

RECIRCULATION LINE BREAK NODALIZATION



ΔP VS.t FOR LOWER REACTOR SKIRT

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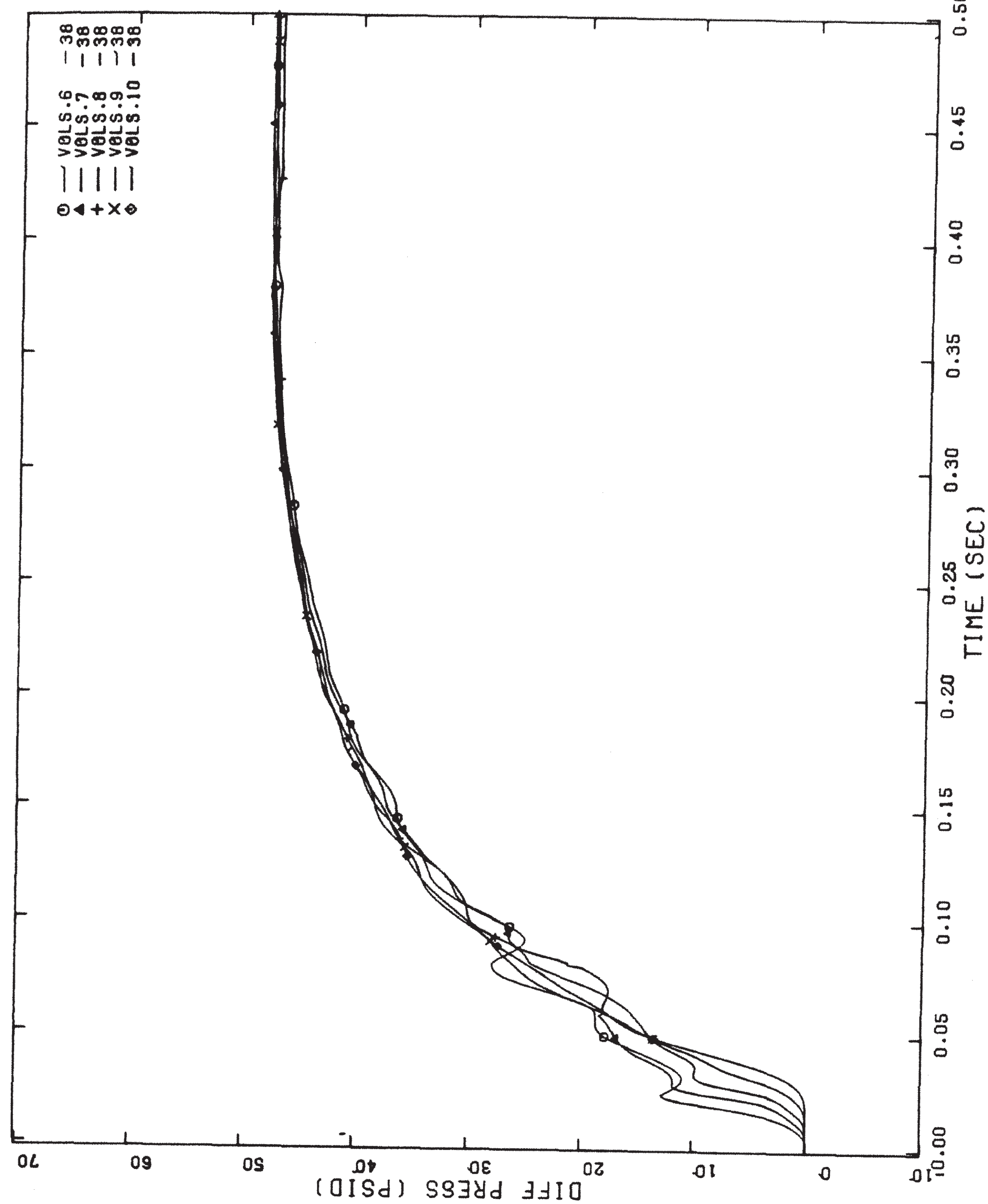
FIGURE 6.2-21

PRESSURE RESPONSE FOR RECIRCULATION LINE BREAK
(SHEET 1 of 9)

REV. 0 - APRIL 1984



ΔP VS. t FOR UPPER REACTOR SKIRT

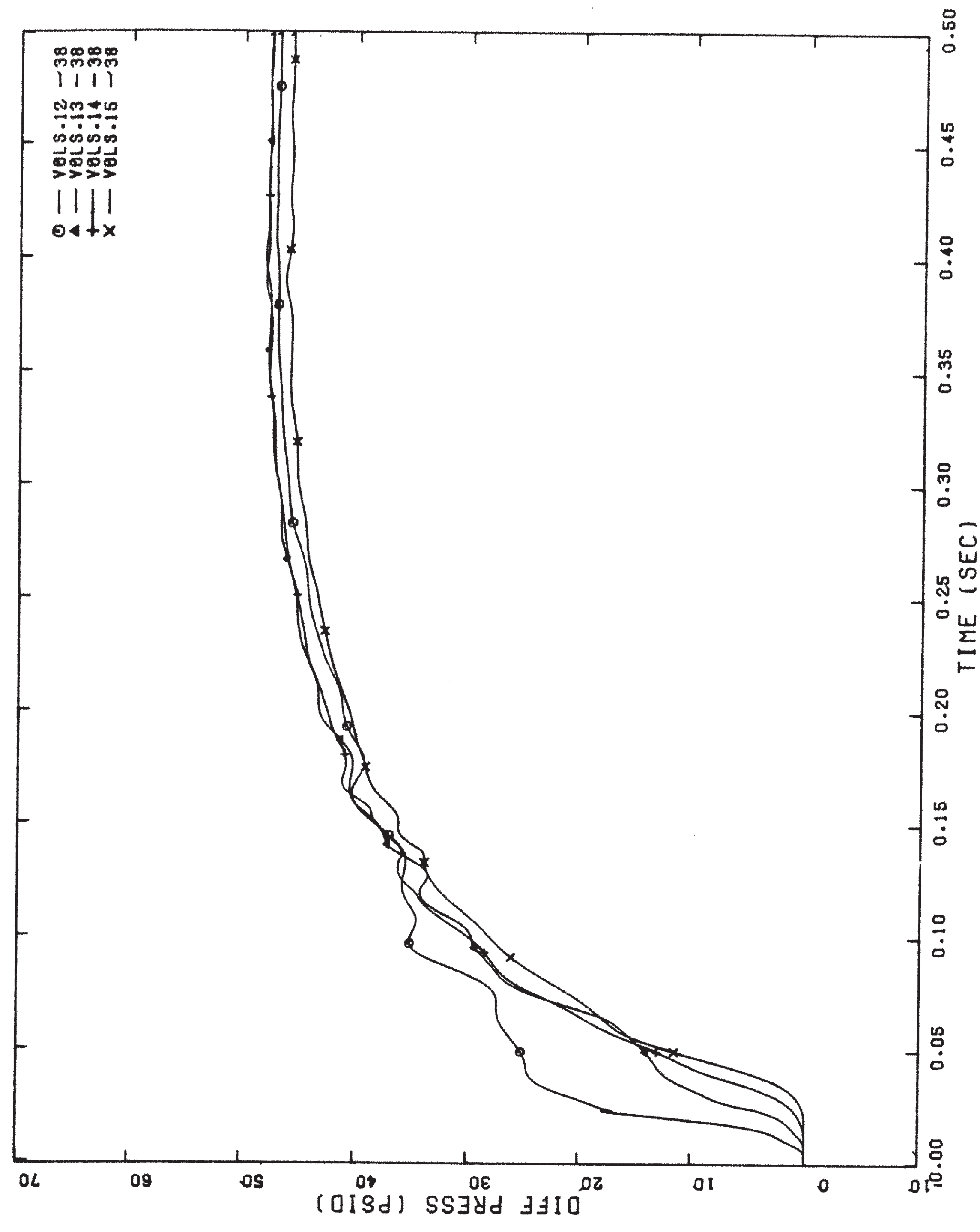
LA SALLE COUNTY STATION
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE 6.2-21

PRESSURE RESPONSE FOR RECIRCULATION LINE BREAK

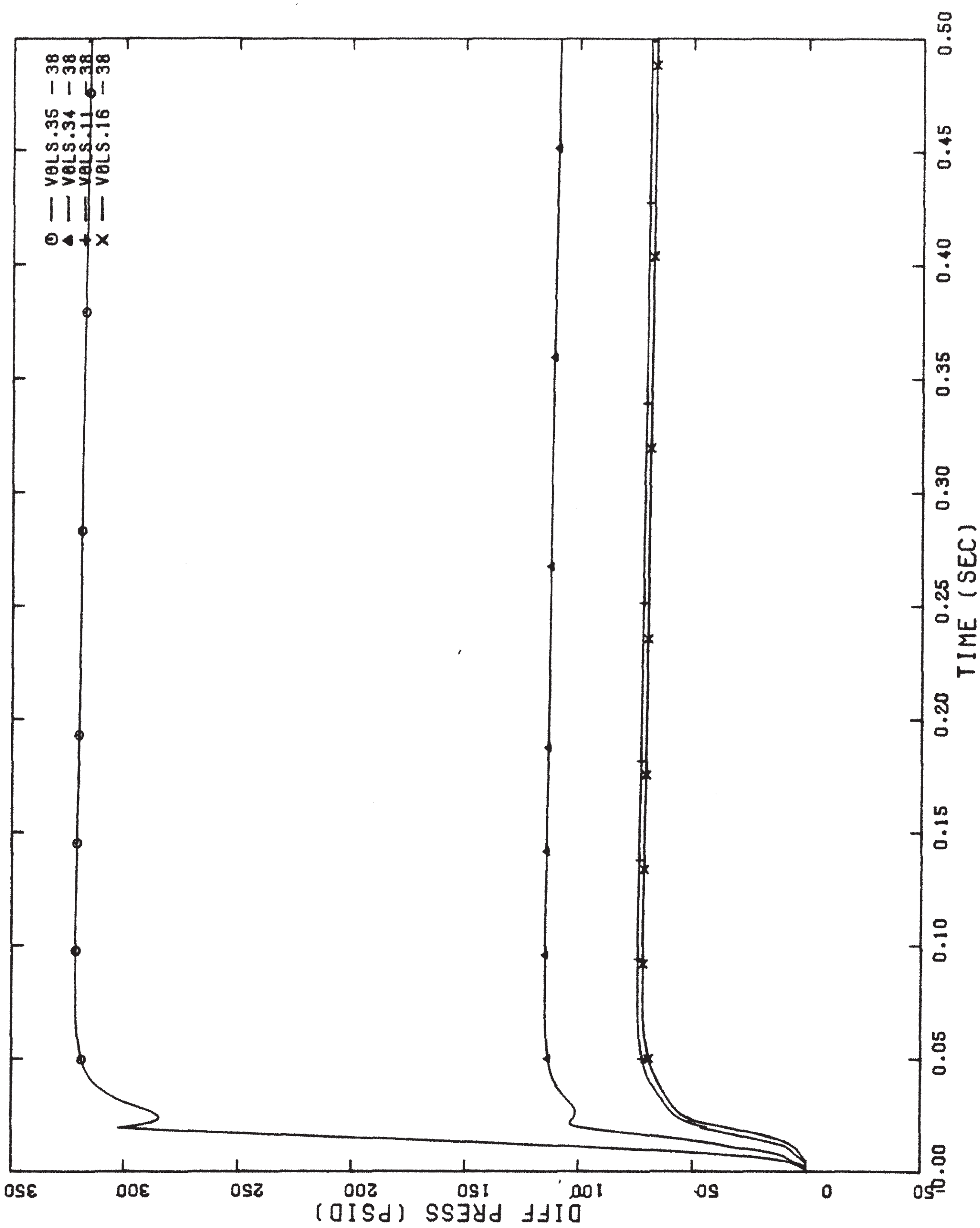
(SHEET 2 of 9)

REV. 0 - APRIL 1984



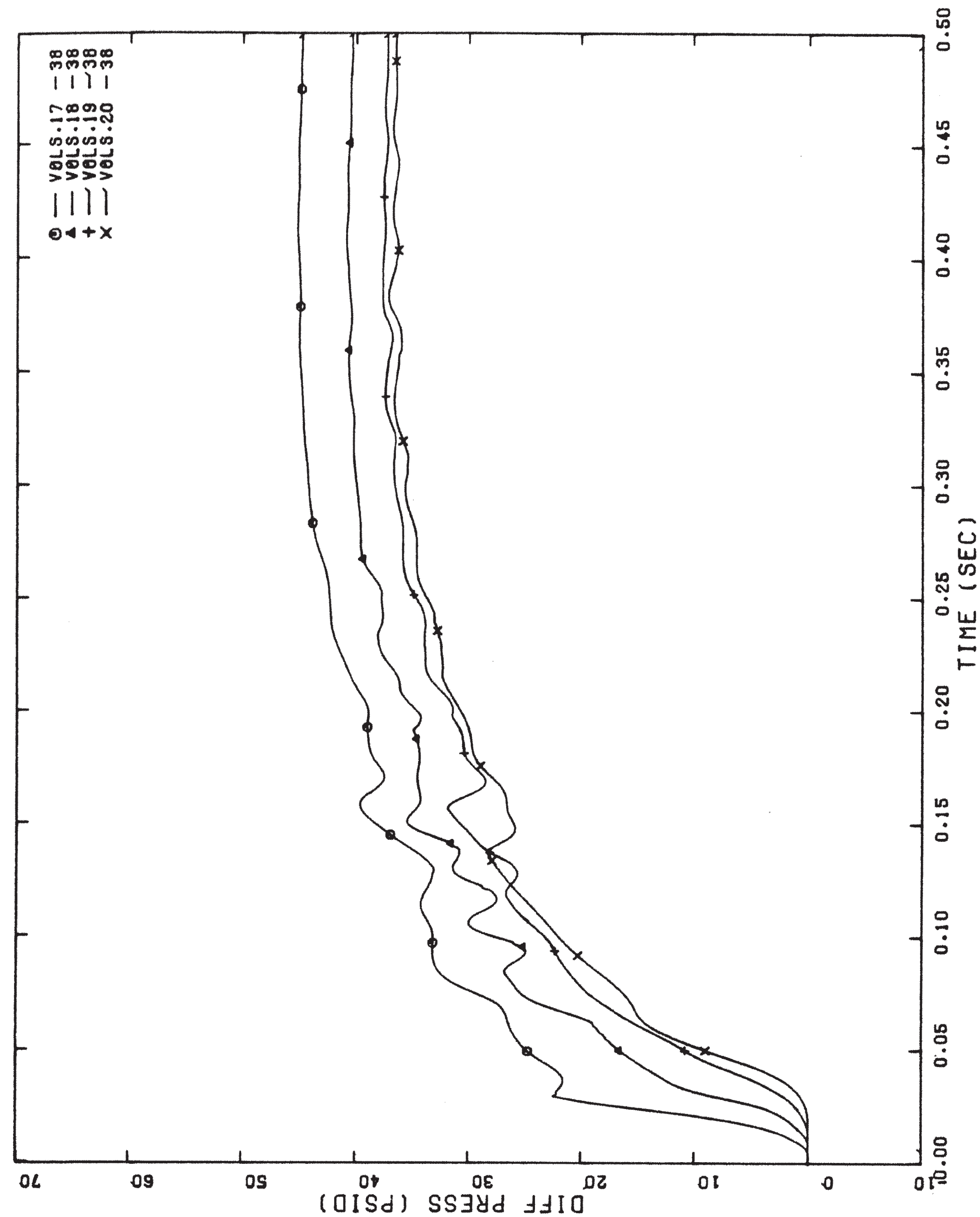
ΔP VS. t FOR LOWER RECIRCULATION
NOZZLE SECTION

LA SALLE COUNTY STATION UPDATED FINAL SAFETY ANALYSIS REPORT
FIGURE 6.2-21 PRESSURE RESPONSE FOR RECIRCULATION LINE BREAK (SHEET 3 of 9) REV. 0 - APRIL 1984



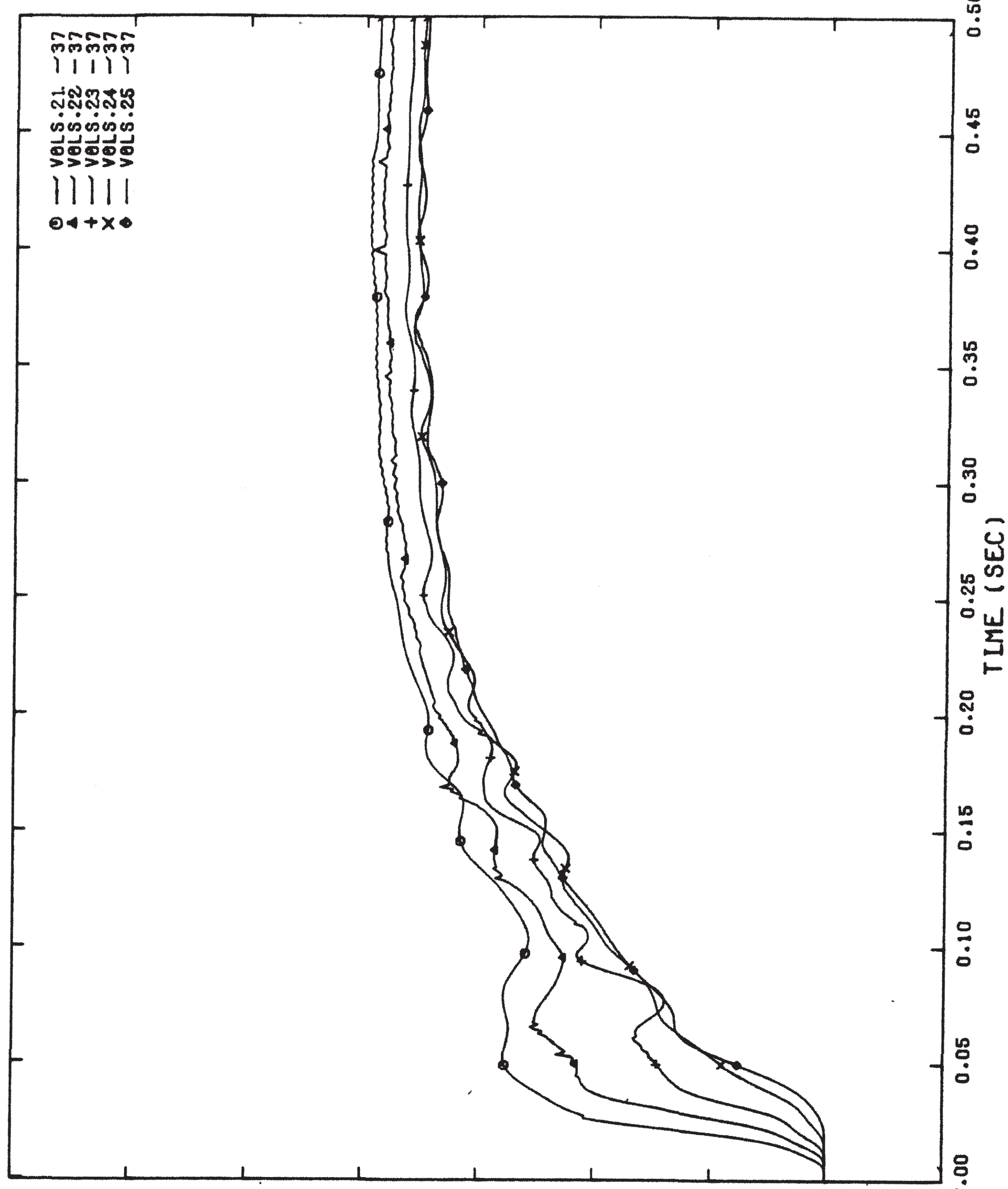
ΔP VS. t ABOUT BREAK

LA SALLE COUNTY STATION UPDATED FINAL SAFETY ANALYSIS REPORT
FIGURE 6.2-21 PRESSURE RESPONSE FOR RECIRCULATION LINE BREAK (SHEET 4 of 9) REV. 0 - APRIL 1984



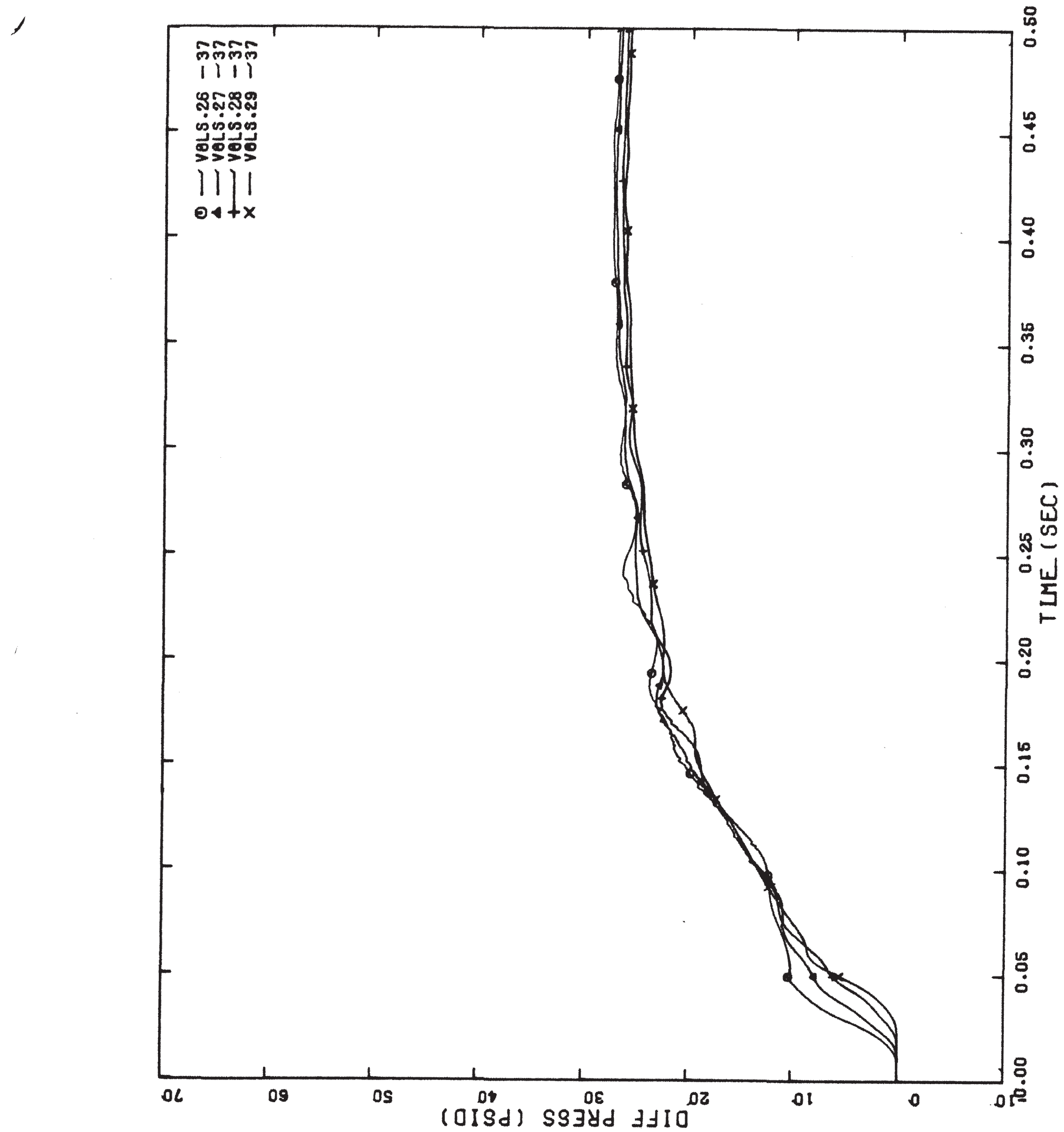
ΔP VS. t FOR UPPER RECIRCULATION NOZZLE SECTION

LA SALLE COUNTY STATION UPDATED FINAL SAFETY ANALYSIS REPORT
FIGURE 6.2-21 PRESSURE RESPONSE FOR RECIRCULATION LINE BREAK (SHEET 5 of 9) REV. 0 - APRIL 1984



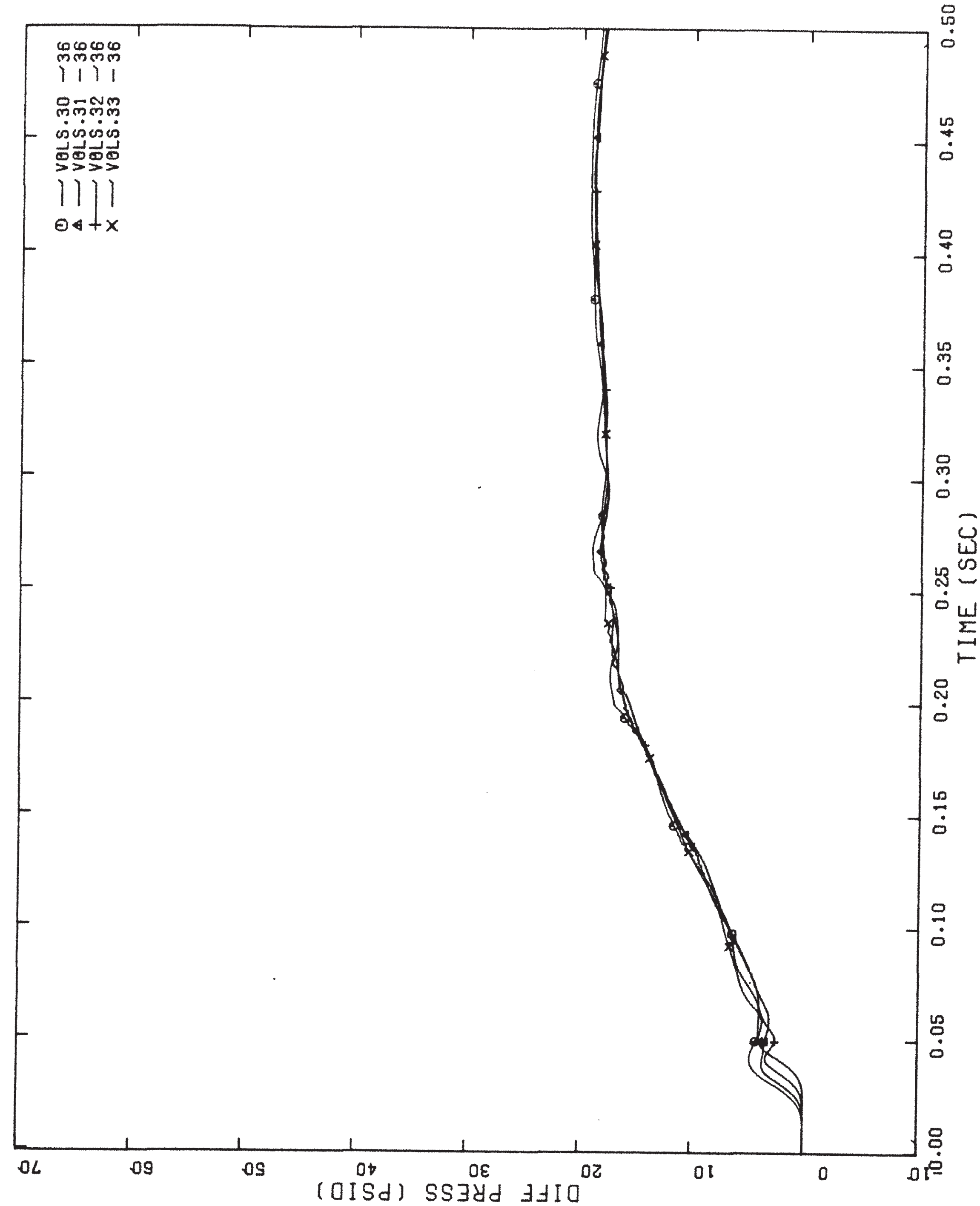
ΔP VS. t FOR MID SECTION

LA SALLE COUNTY STATION UPDATED FINAL SAFETY ANALYSIS REPORT
FIGURE 6.2-21 PRESSURE RESPONSE FOR RECIRCULATION LINE BREAK (SHEET 6 of 9) REV. 0 - APRIL 1984



ΔP VS. t FOR LPCI NOZZLE SECTION

LA SALLE COUNTY STATION UPDATED FINAL SAFETY ANALYSIS REPORT
FIGURE 6.2-21 PRESSURE RESPONSE FOR RECIRCULATION LINE BREAK (SHEET 7 of 9) REV. 0 - APRIL 1984



ΔP VS. t FOR FEEDWATER NOZZLE SECTION

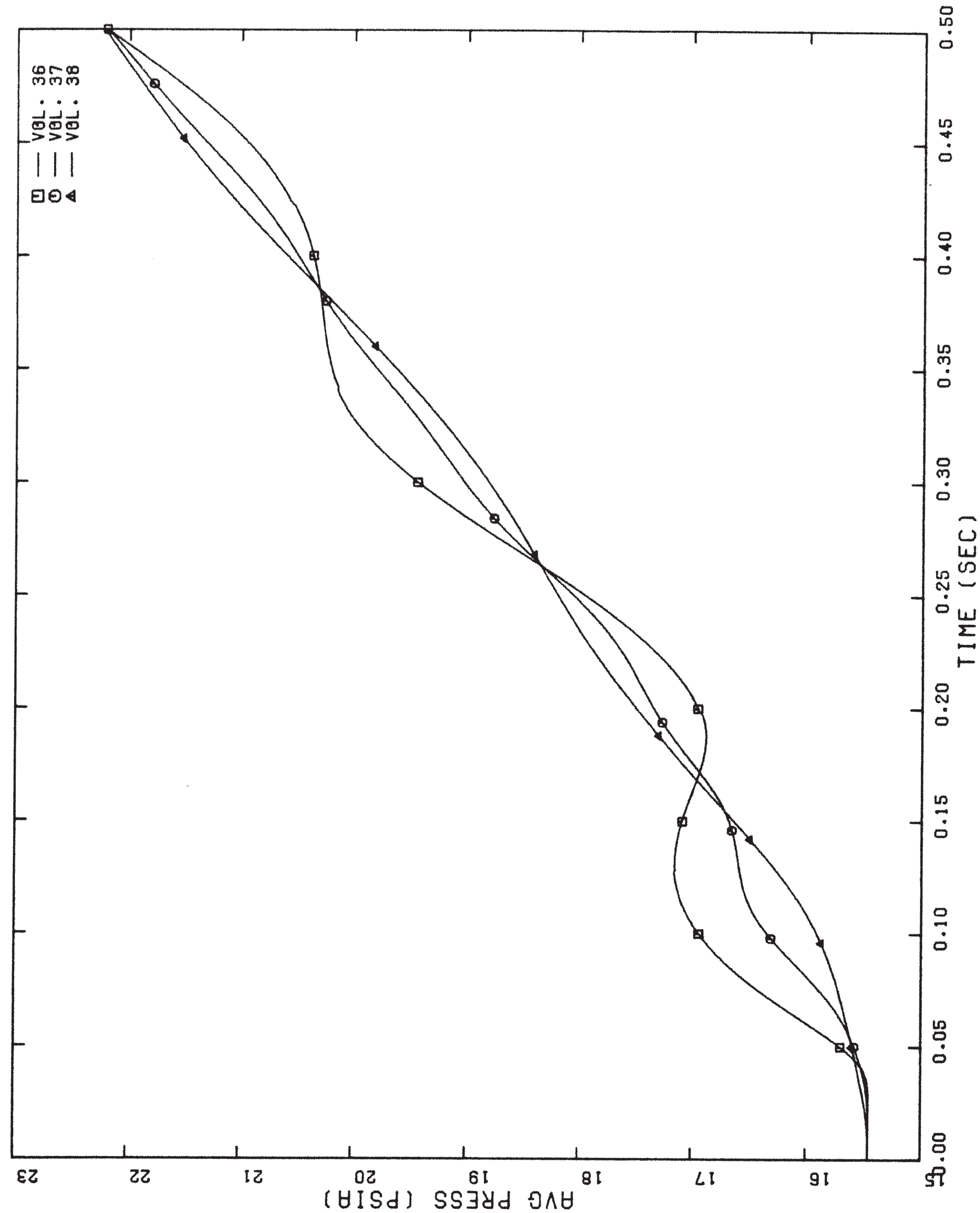
LA SALLE COUNTY STATION
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE 6.2-21

PRESSURE RESPONSE FOR RECIRCULATION LINE BREAK

(SHEET 8 of 9)

REV. 0 - APRIL 1984



CONTAINMENT PRESSURE RESPONSE

LA SALLE COUNTY STATION UPDATED FINAL SAFETY ANALYSIS REPORT
FIGURE 6.2-21 PRESSURE RESPONSE FOR RECIRCULATION LINE BREAK (SHEET 9 of 9) REV. 0 - APRIL 1984

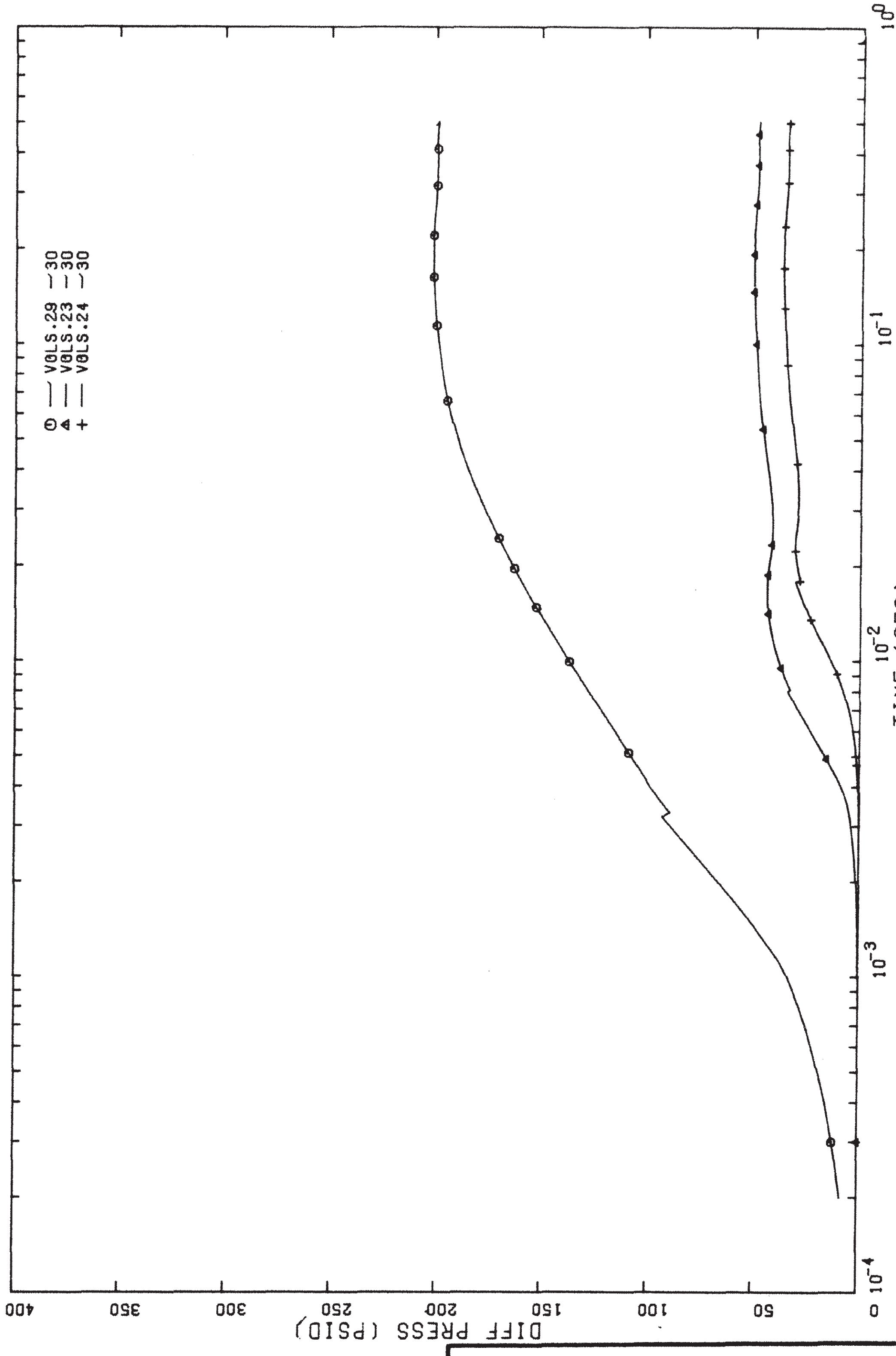


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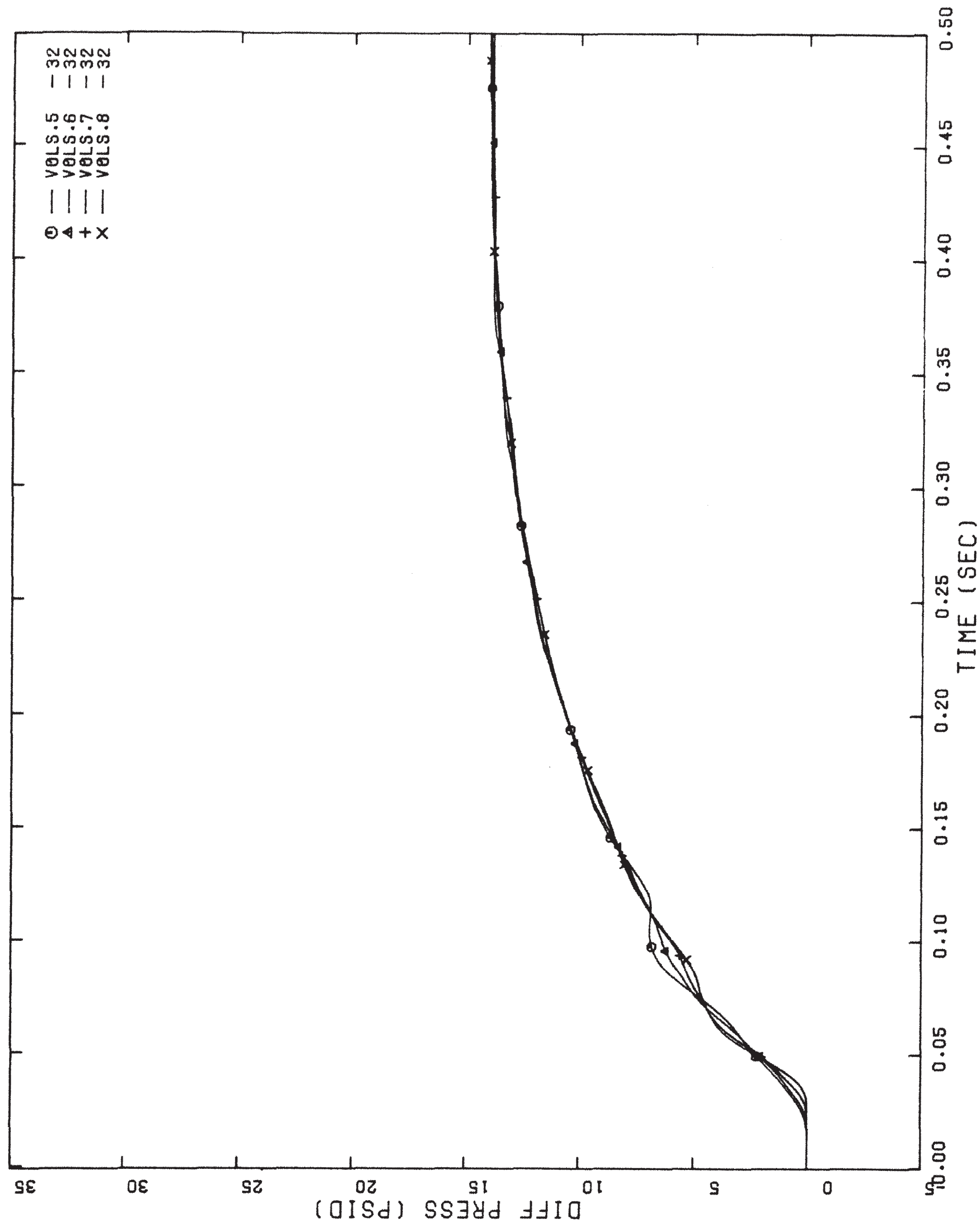
FIGURE 6.2-22

ΔP VS. LOG t ABOUT BREAK -
FEEDWATER LINE BREAK

REV. 0 - APRIL 1984



ΔP VS. t FOR LOWER REACTOR SKIRT

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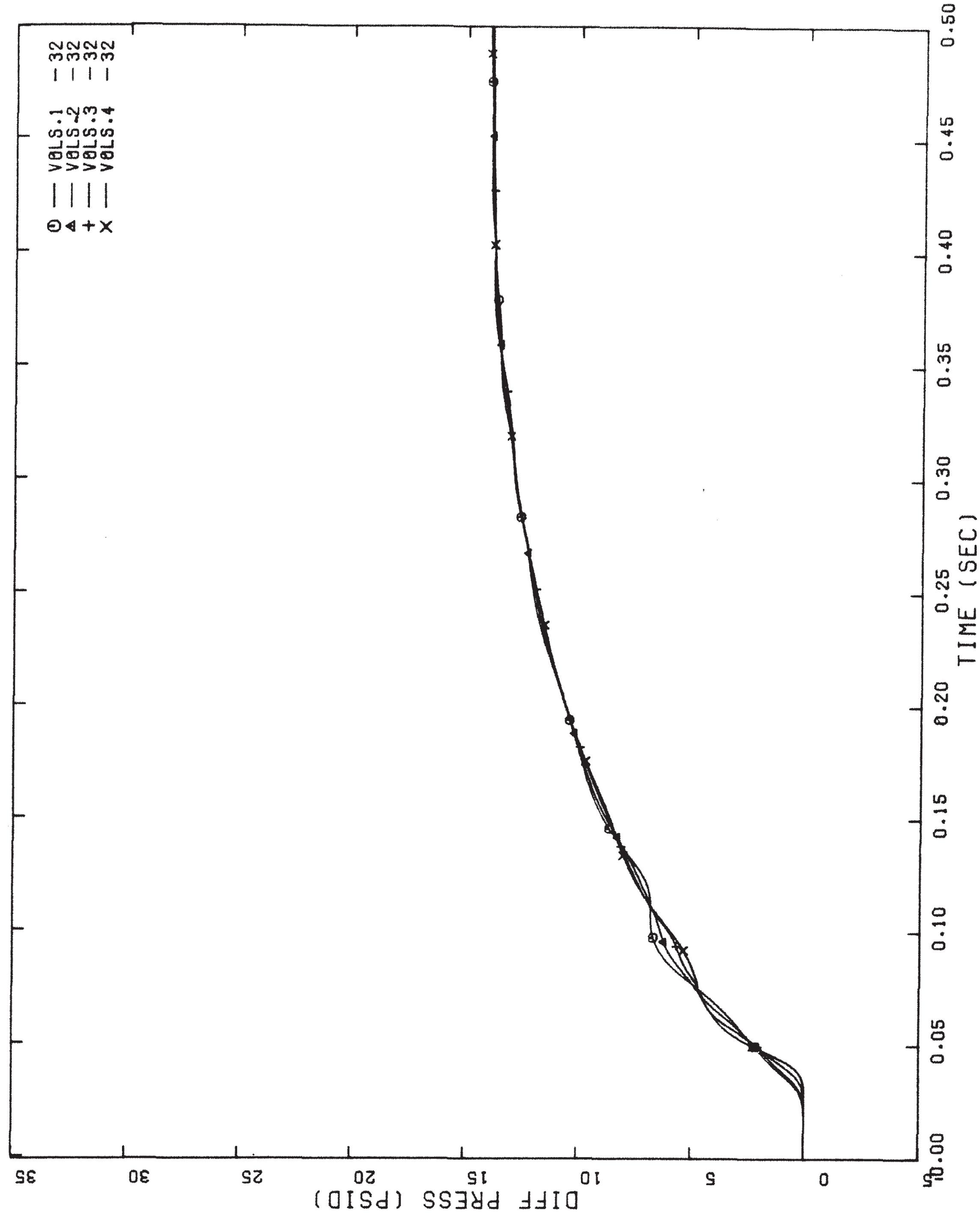
FIGURE 6.2-23 PRESSURE RESPONSE FOR FEEDWATER LINE BREAK (SHEET 1 of 8)

REV. 0 - APRIL 1984



ΔP VS. t FOR UPPER REACTOR SKIRT

LA SALLE COUNTY STATION
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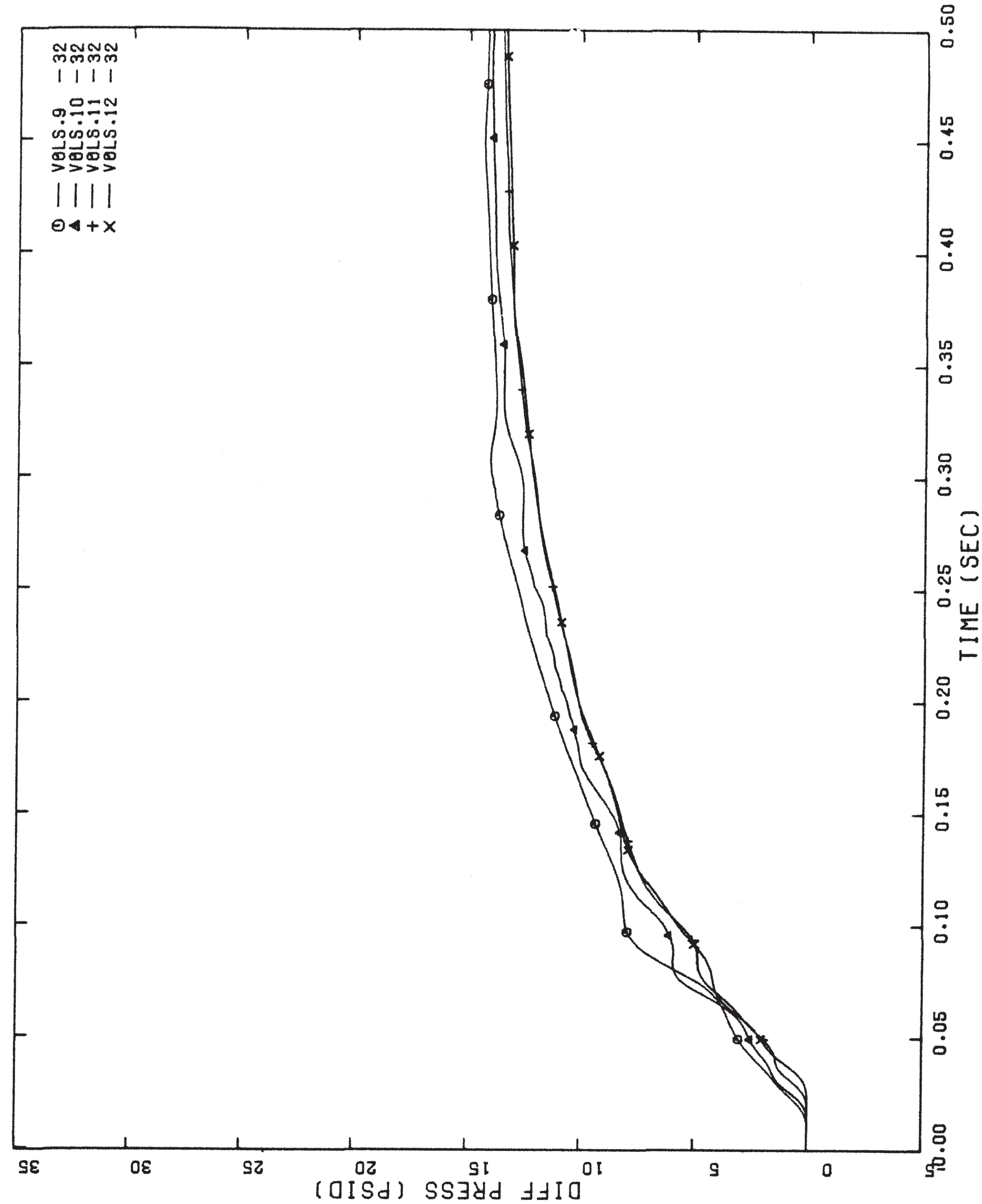
FIGURE 6.2-23

PRESSURE RESPONSE FOR FEEDWATER LINE BREAK
(SHEET 2 of 8)

REV. 0 - APRIL 1984



ΔP VS. t FOR RECIRCULATION NOZZLE SECTION

LA SALLE COUNTY STATION
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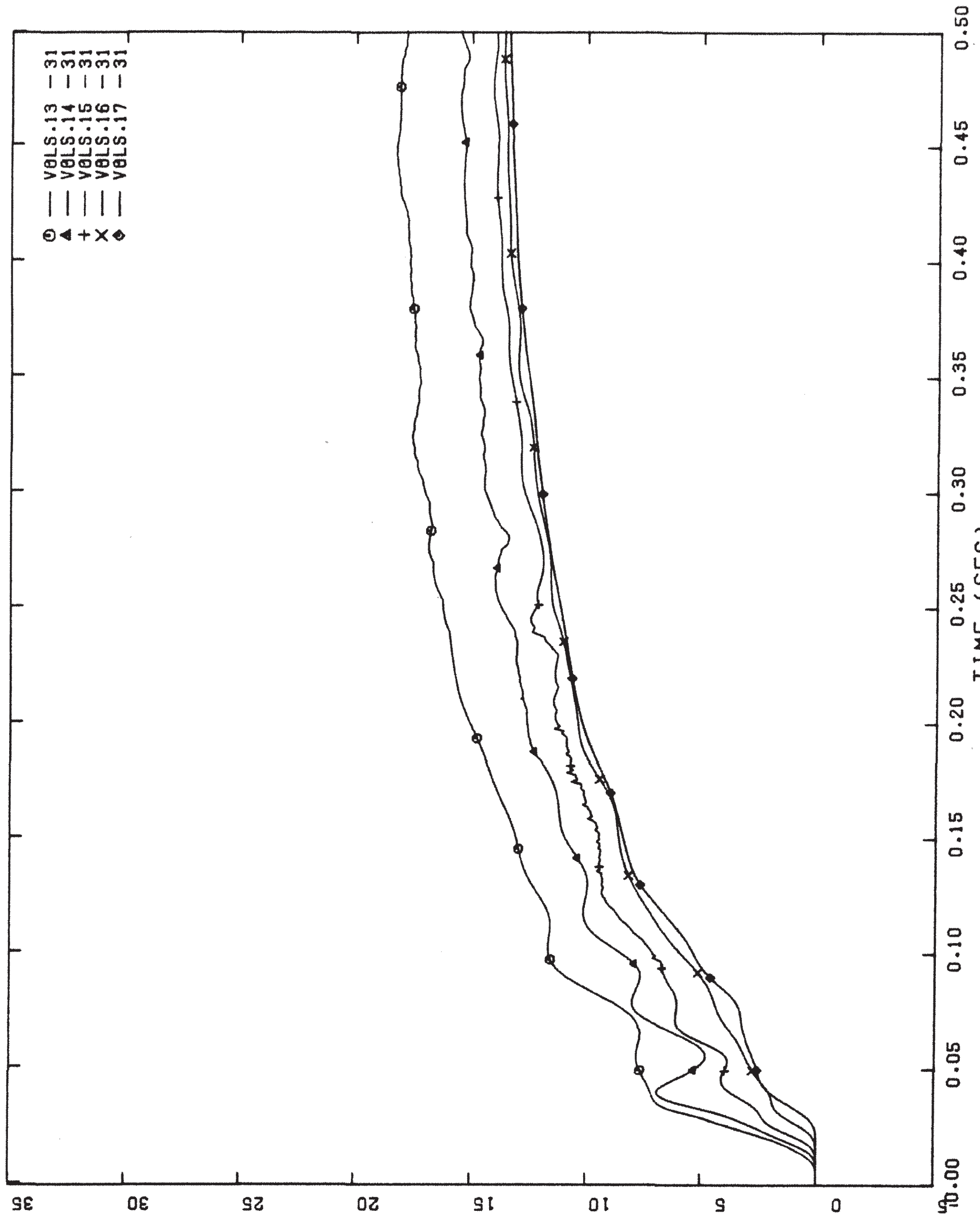
FIGURE 6.2-23

PRESSURE RESPONSE FOR FEEDWATER LINE BREAK

(SHEET 3 of 8)

REV. 0 - APRIL 1984



ΔP VS. t FOR MID SECTION

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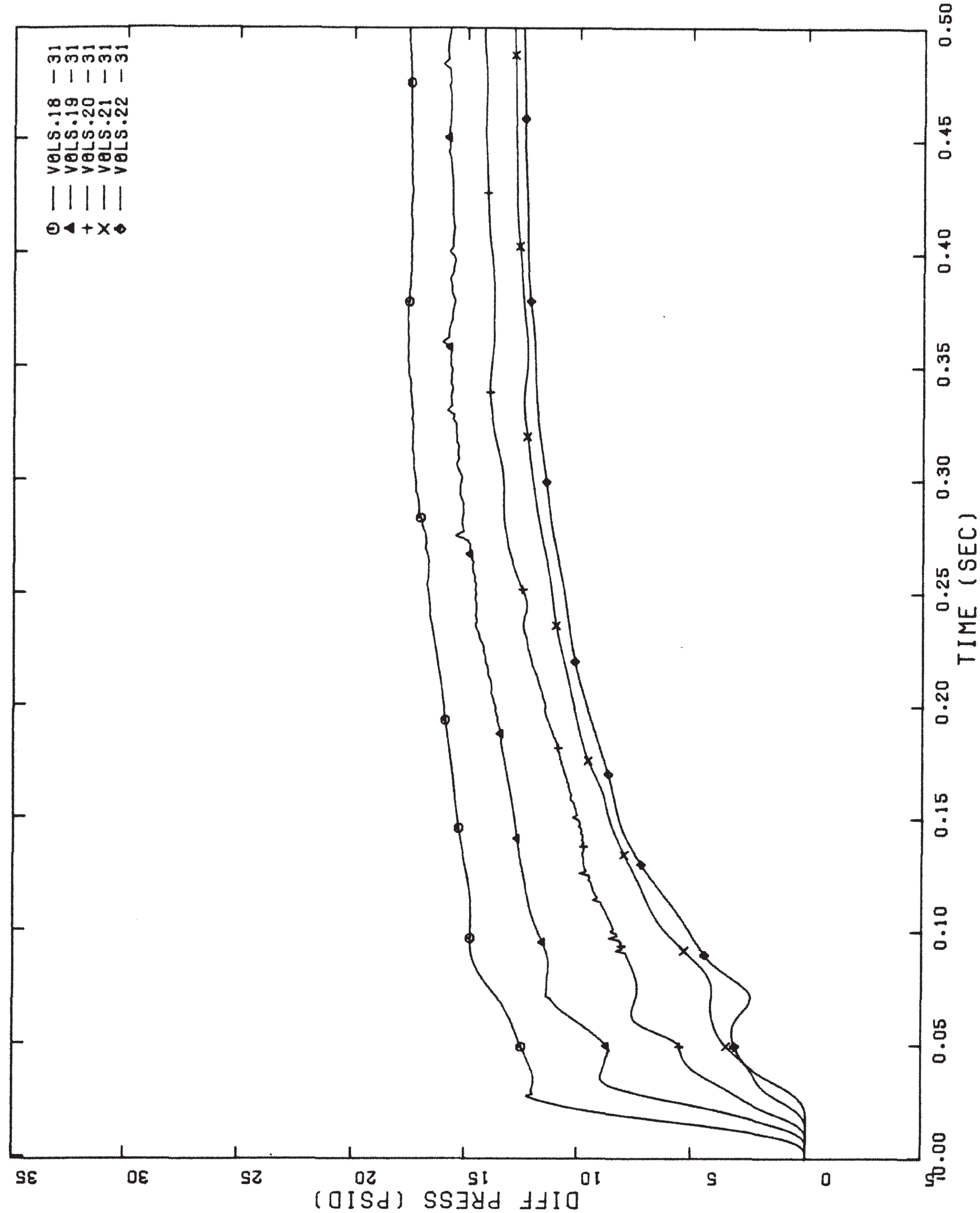
FIGURE 6.2-23 PRESSURE RESPONSE FOR FEEDWATER LINE BREAK (SHEET 4 of 8) REV. 0 - APRIL 1984



ΔP VS. t FOR LPCI NOZZLE SECTION

LA SALLE COUNTY STATION
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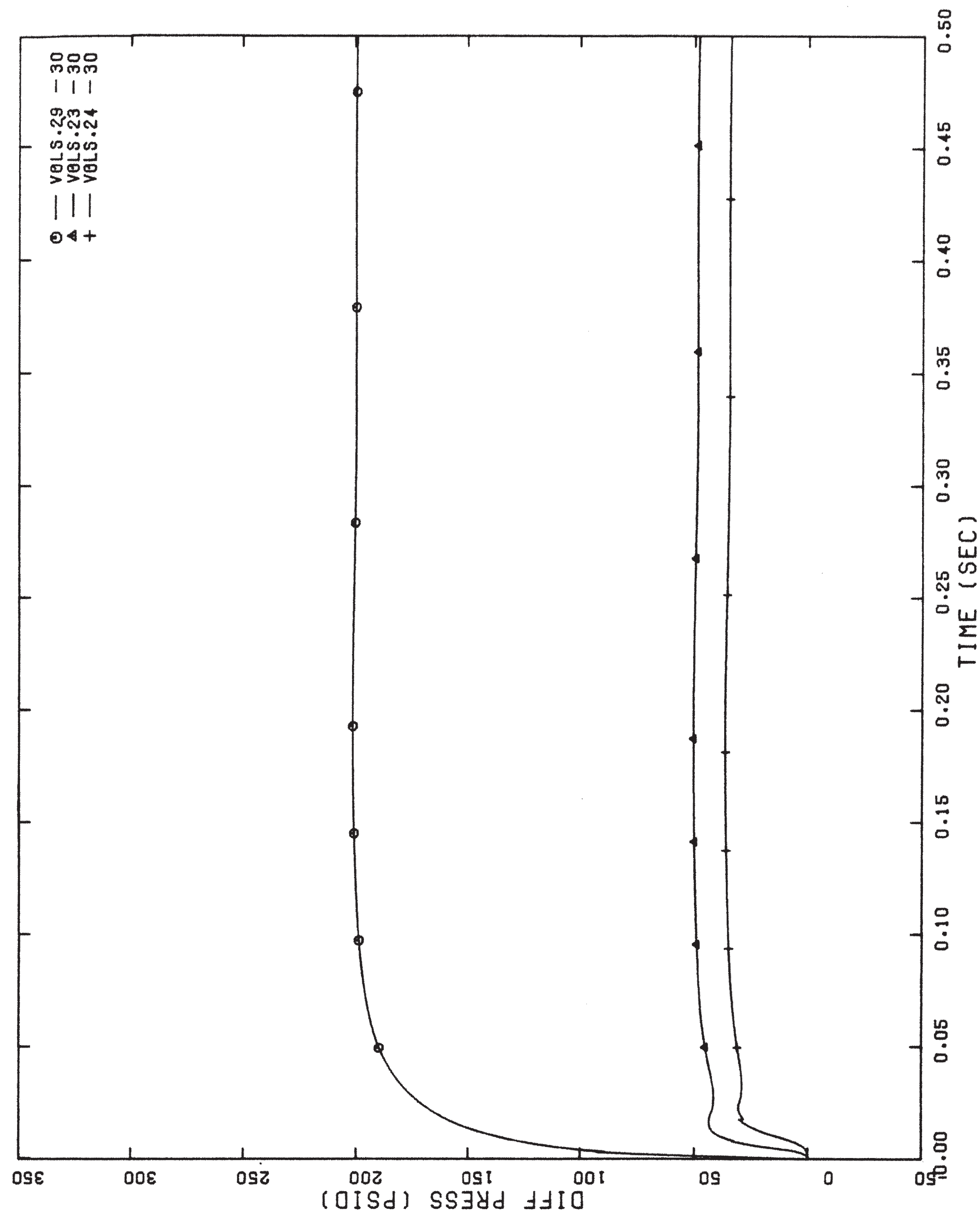
FIGURE 6.2-23

PRESSURE RESPONSE FOR FEEDWATER LINE BREAK

(SHEET 5 of 8)

REV. 0 - APRIL 1984



ΔP VS. t ABOUT BREAK

LA SALLE COUNTY STATION
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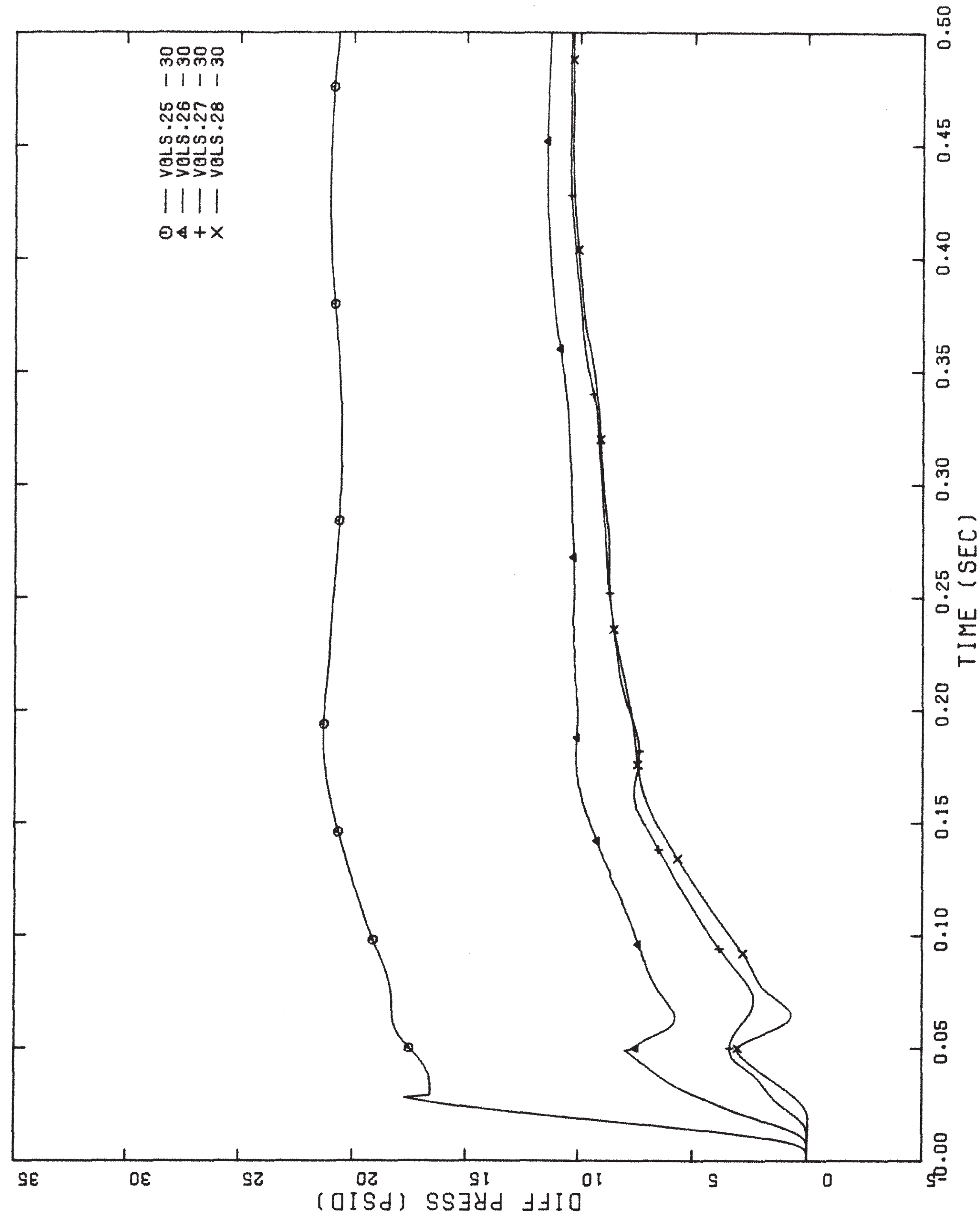
FIGURE 6.2-23

PRESSURE RESPONSE FOR FEEDWATER LINE BREAK

(SHEET 6 of 8)

REV. 0 - APRIL 1984



ΔP VS. t FOR FEEDWATER NOZZLE SECTION

LA SALLE COUNTY STATION
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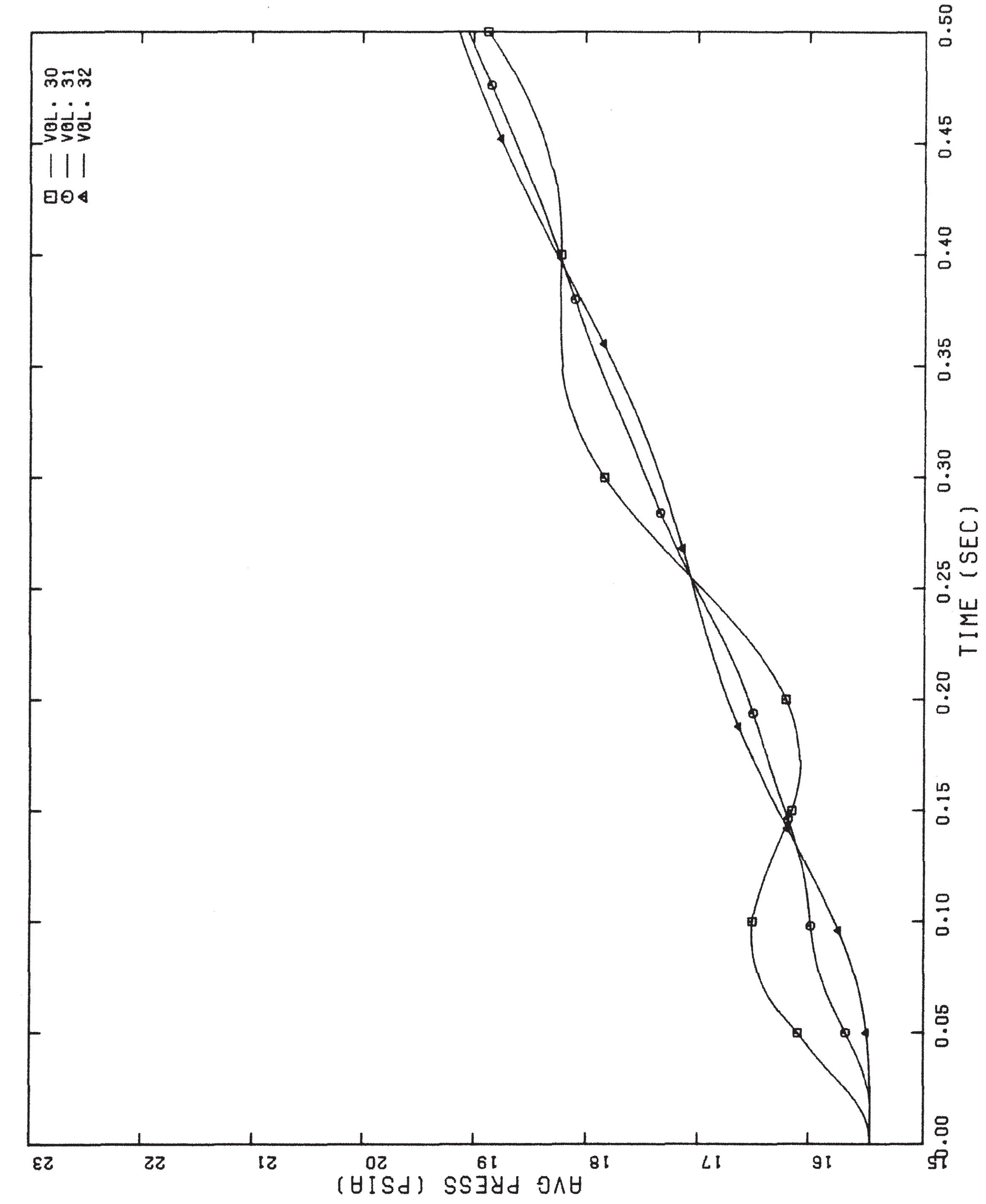
FIGURE 6.2-23

PRESSURE RESPONSE FOR FEEDWATER LINE BREAK

(SHEET 7 of 8)

REV. 0 - APRIL 1984



CONTAINMENT PRESSURE RESPONSE

LA SALLE COUNTY STATION UPDATED FINAL SAFETY ANALYSIS REPORT
FIGURE 6.2-23 PRESSURE RESPONSE FOR FEEDWATER LINE BREAK (SHEET 8 of 8) REV. 0 - APRIL 1984

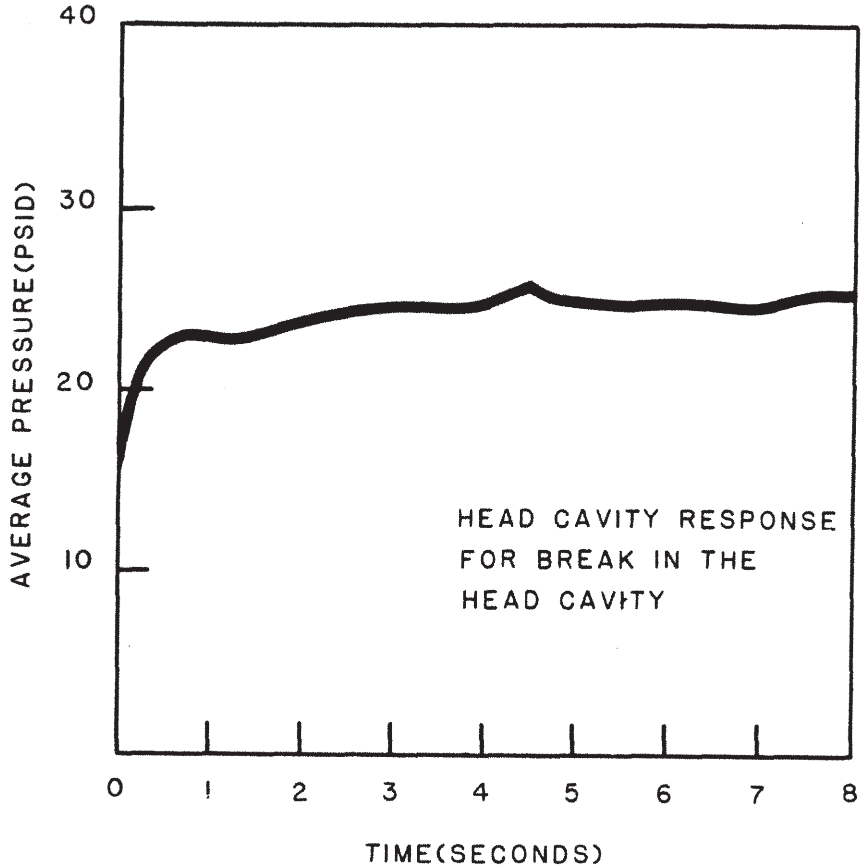


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FIGURE 6.2-24

PRESSURE HISTORIES OF NODES
FOR WORST BREAK CASES

(SHEET 1 of 4)

REV. 0 - APRIL 1984

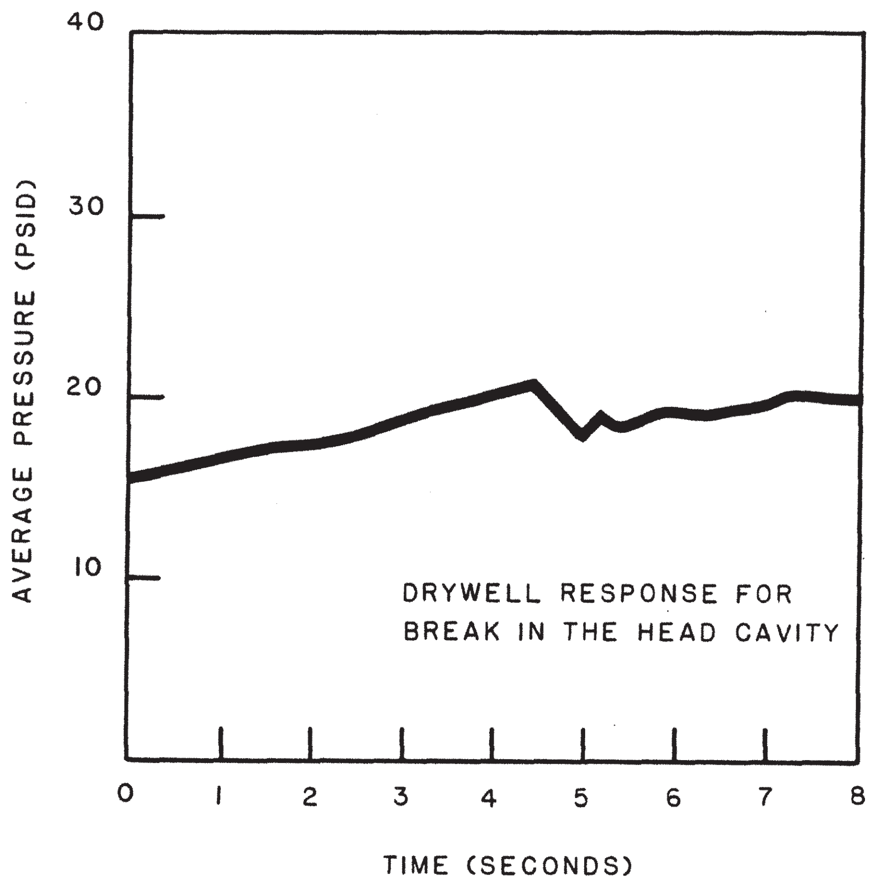

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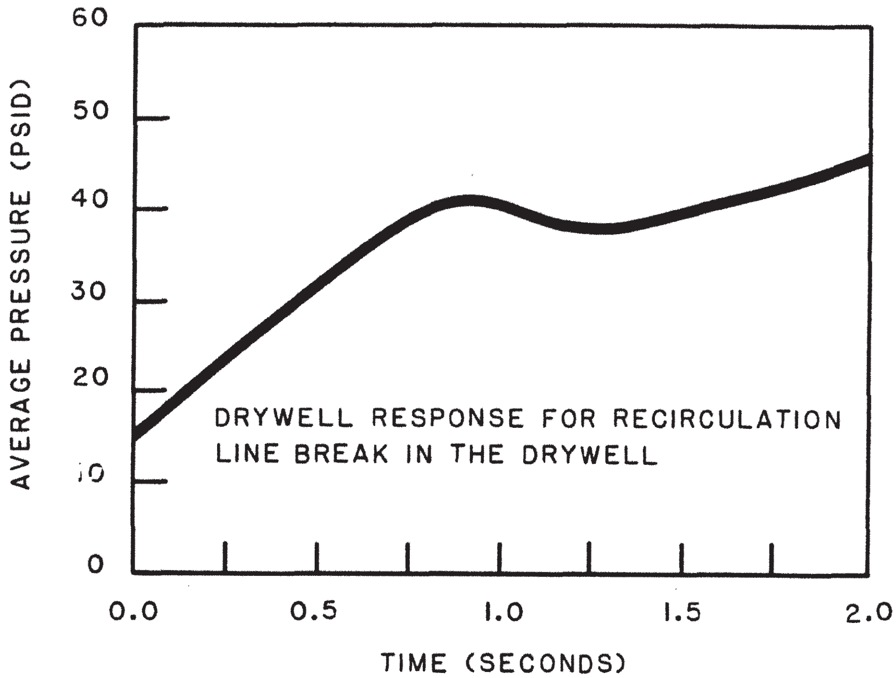
FIGURE 6.2-24

PRESSURE HISTORIES OF NODES
FOR WORST BREAK CASES

(SHEET 2 of 4)



LA SALLE COUNTY STATION
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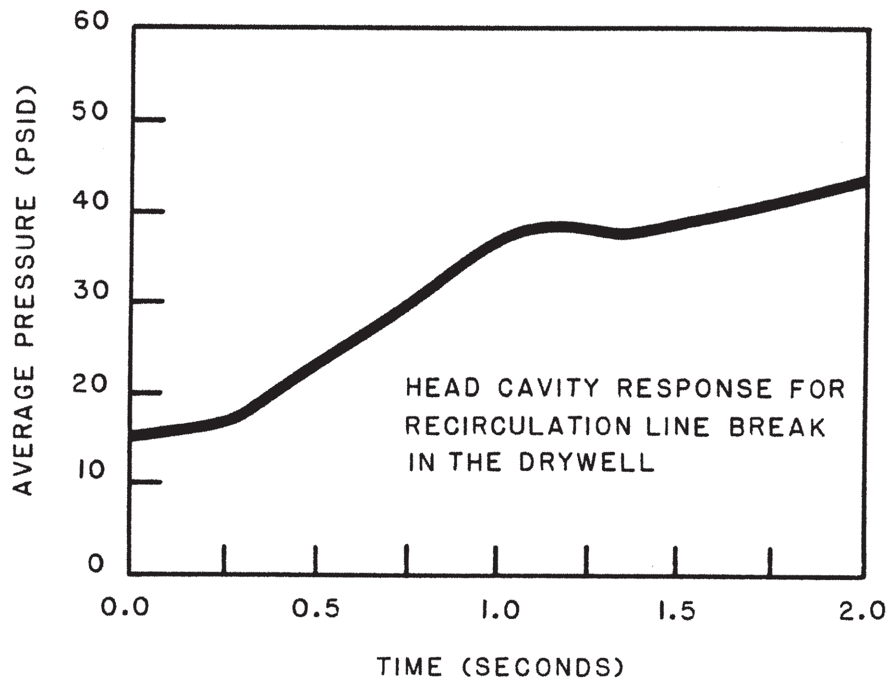
FIGURE 6.2-24

PRESSURE HISTORIES OF NODES
FOR WORST BREAK CASES

(SHEET 3 of 4)

REV. 0 - APRIL 1984



LA SALLE COUNTY STATION
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FIGURE 6.2-24

PRESSURE HISTORIES OF NODES
FOR WORST BREAK CASES

(SHEET 4 of 4)

REV. 0 - APRIL 1984

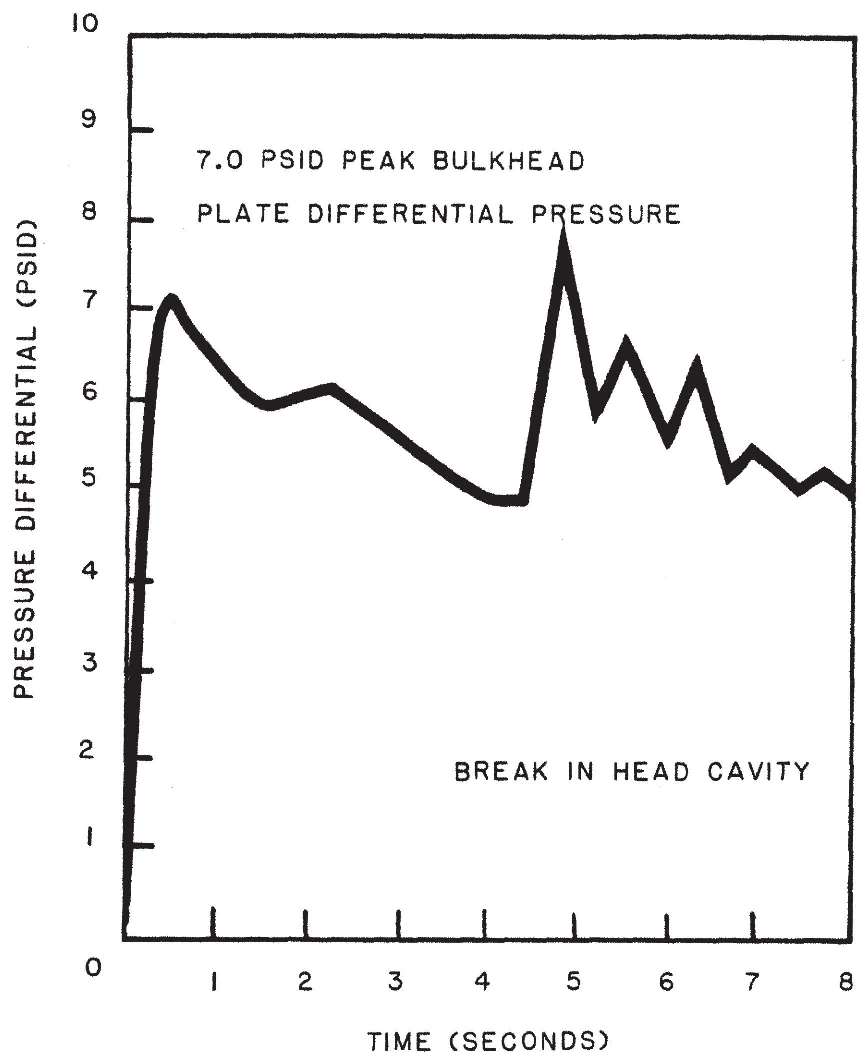

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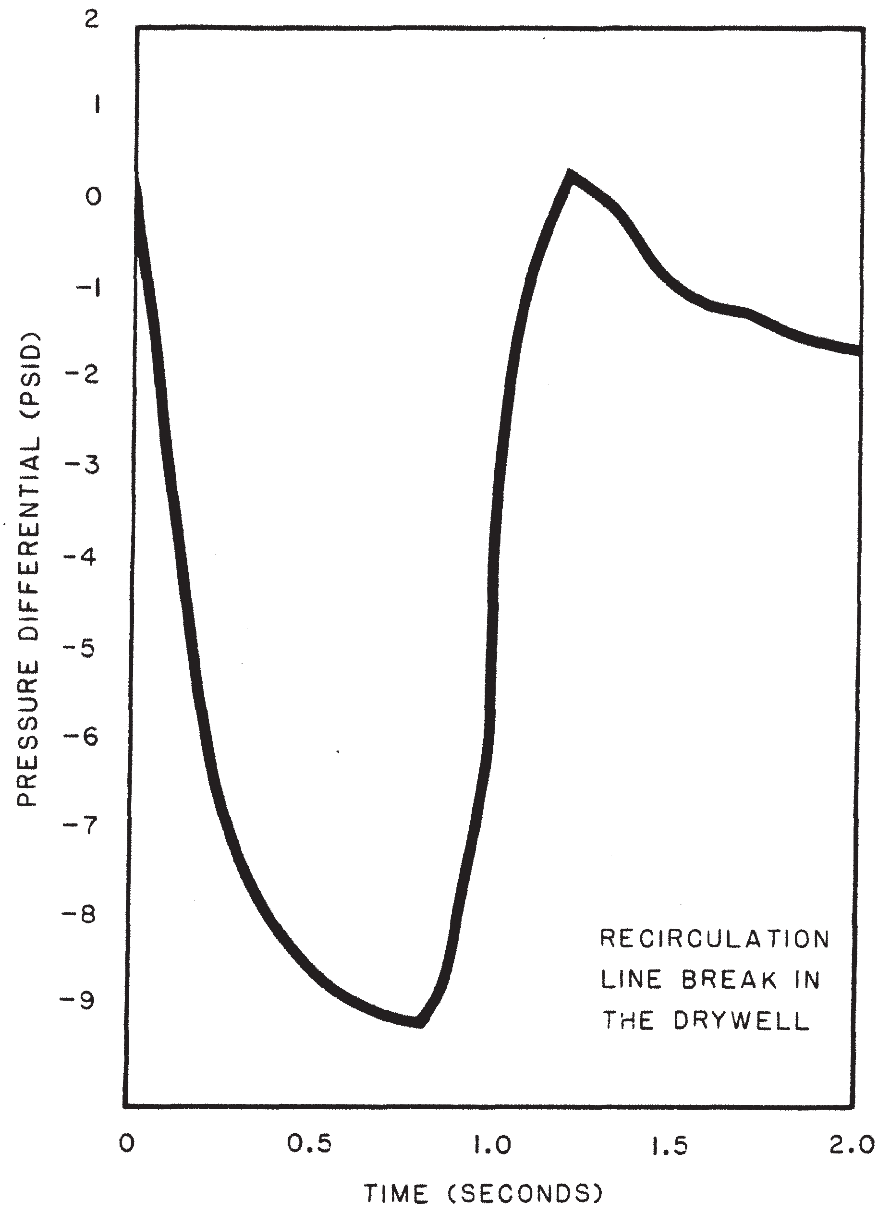
FIGURE 6.2-25

PRESSURE DIFFERENTIAL ACROSS THE BULKHEAD PLATE FOR THE WORST BREAK CASES

(SHEET 1 of 2)

REV. 0 - APRIL 1984



LA SALLE COUNTY STATION UPDATED FINAL SAFETY ANALYSIS REPORT
FIGURE 6.2-25 PRESSURE DIFFERENTIAL ACROSS THE BULKHEAD PLATE FOR THE WORST BREAK CASES (SHEET 2 of 2)

REV. 0 - APRIL 1984

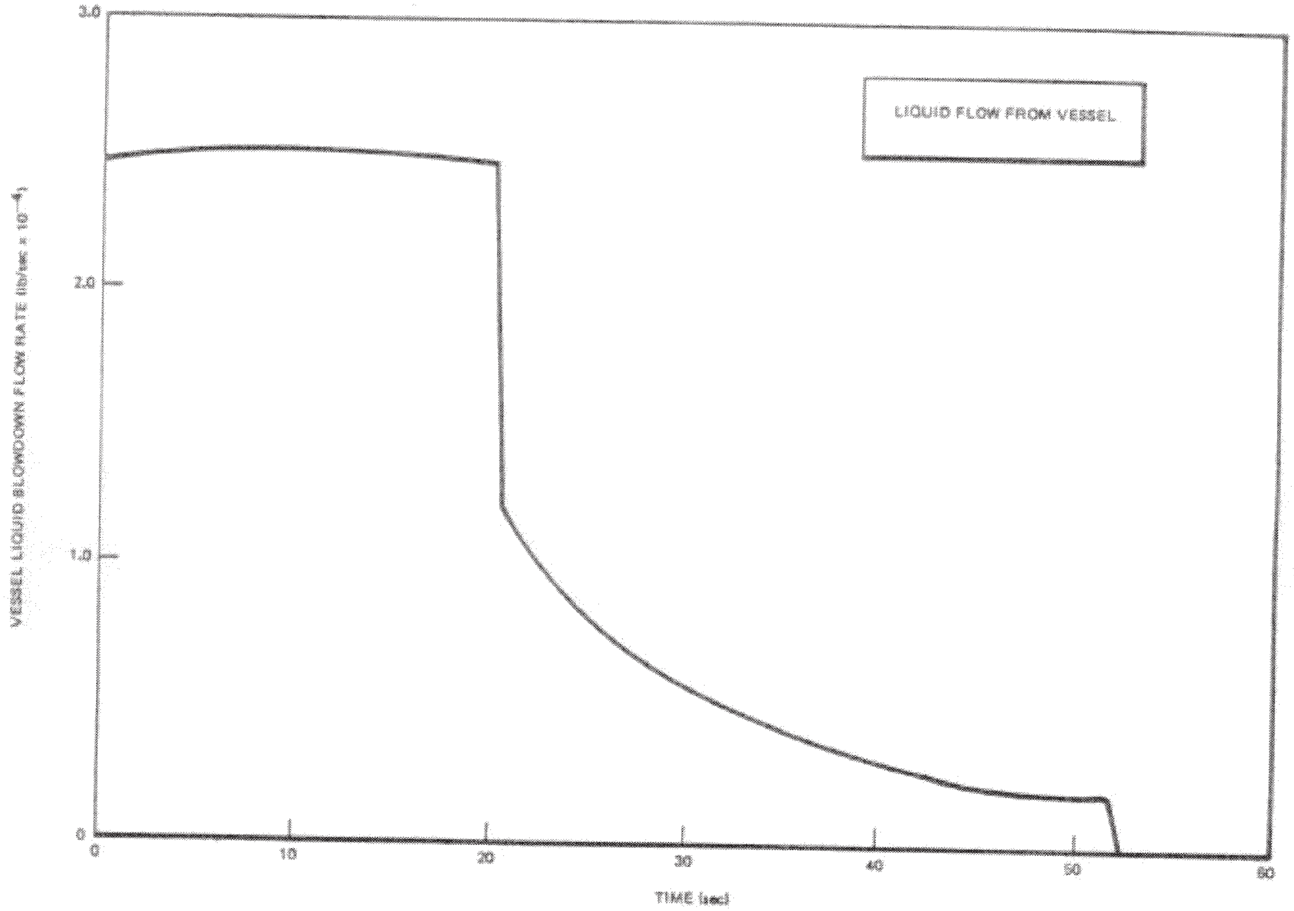


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FIGURE 6.2-26 VESSEL LIQUID BLOWDOWN RATE (At 3434 MWt)

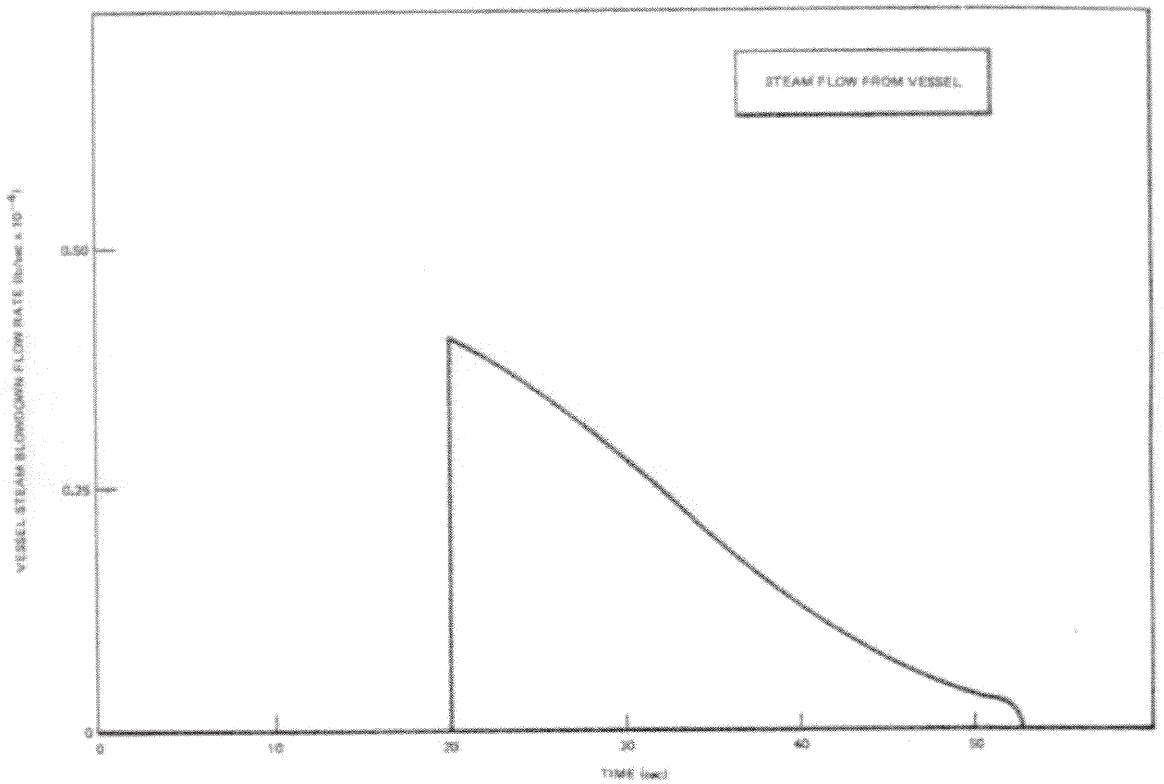

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FIGURE 6.2-27

VESSEL STEAM BLOWDOWN RATE
(At 3434 MWt)

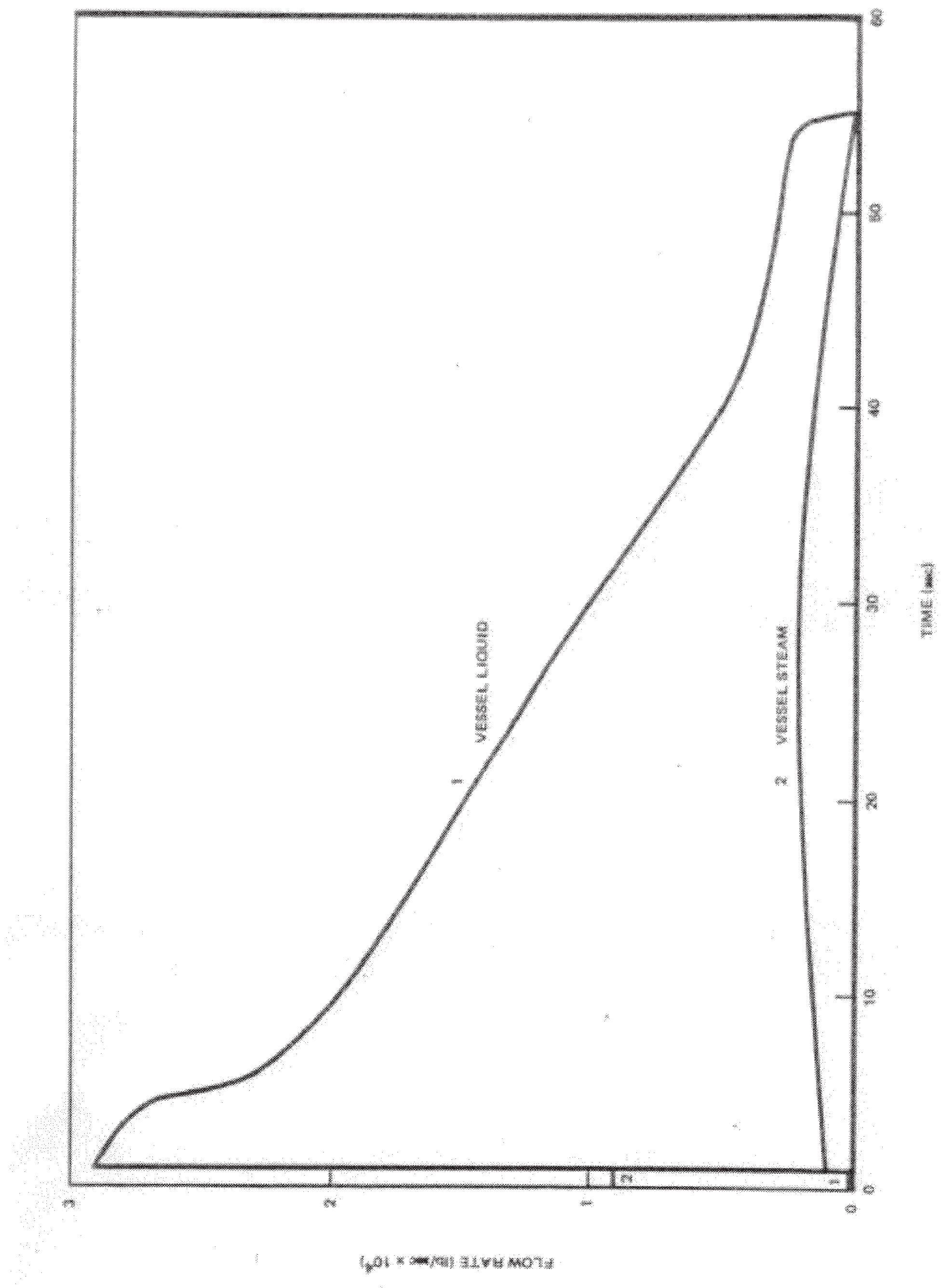

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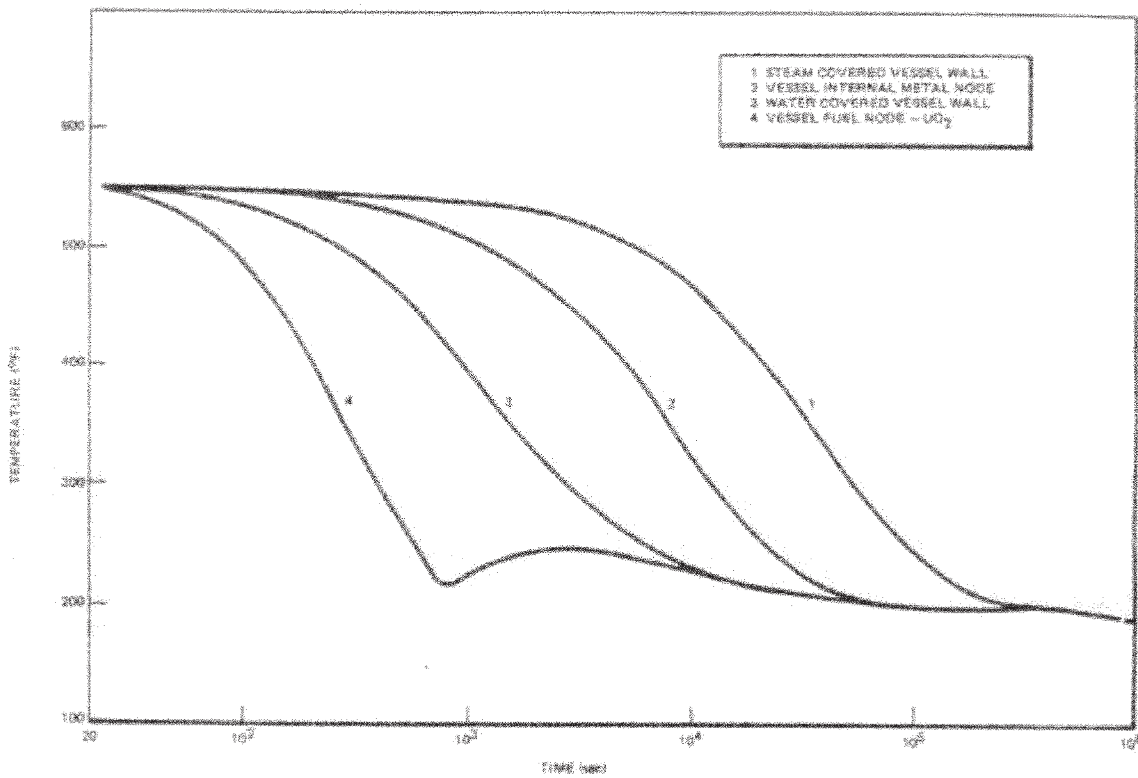
FIGURE 6.2-28

MAIN STEAMLINE BREAK RESPONSE
PARAMETERS BLOWDOWN FLOW
(At 3434 MWt)



Note: This figure is extracted from original analysis and is presented here as historical and representative of comparable response as would be expected for current analysis.

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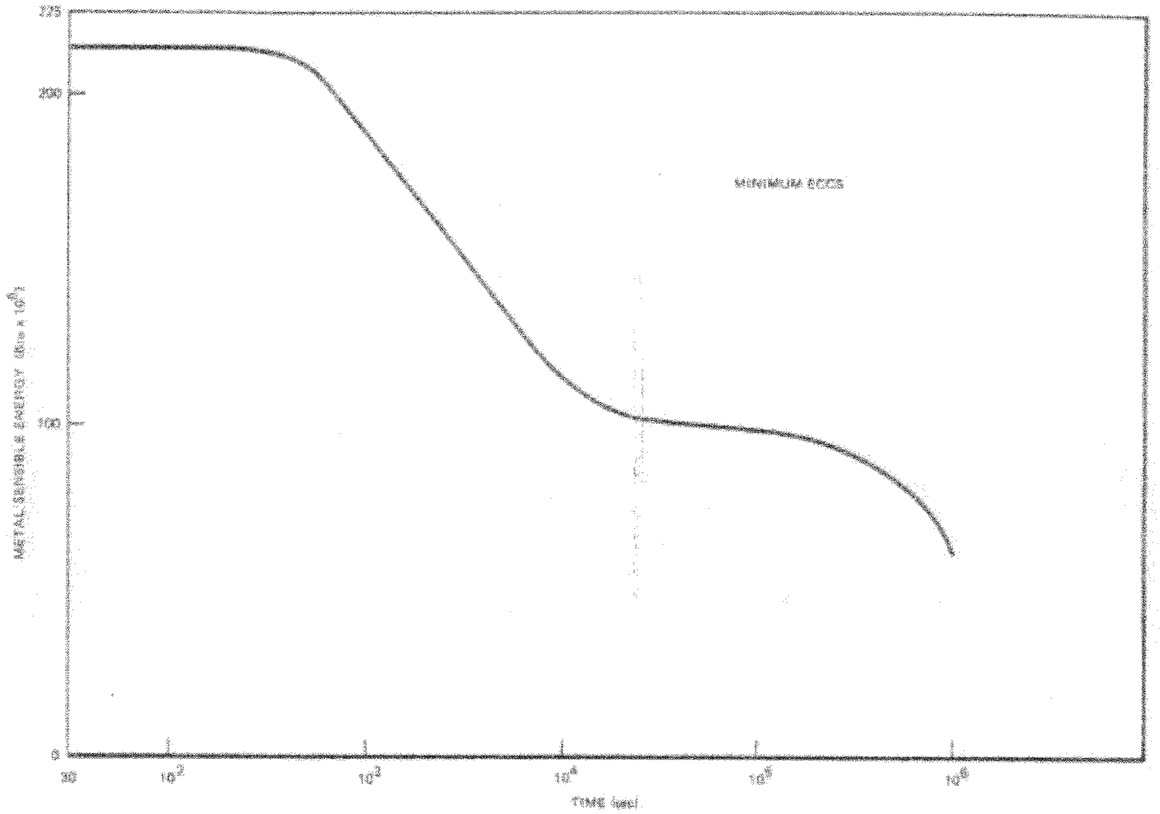
FIGURE 6.2-29 TEMPERATURE RESPONSE OR REACTOR VESSEL (At 3434 MWt)



Note: This figure is extracted from original analysis and is presented here as historical and representative of comparable response as would be expected for current analysis.

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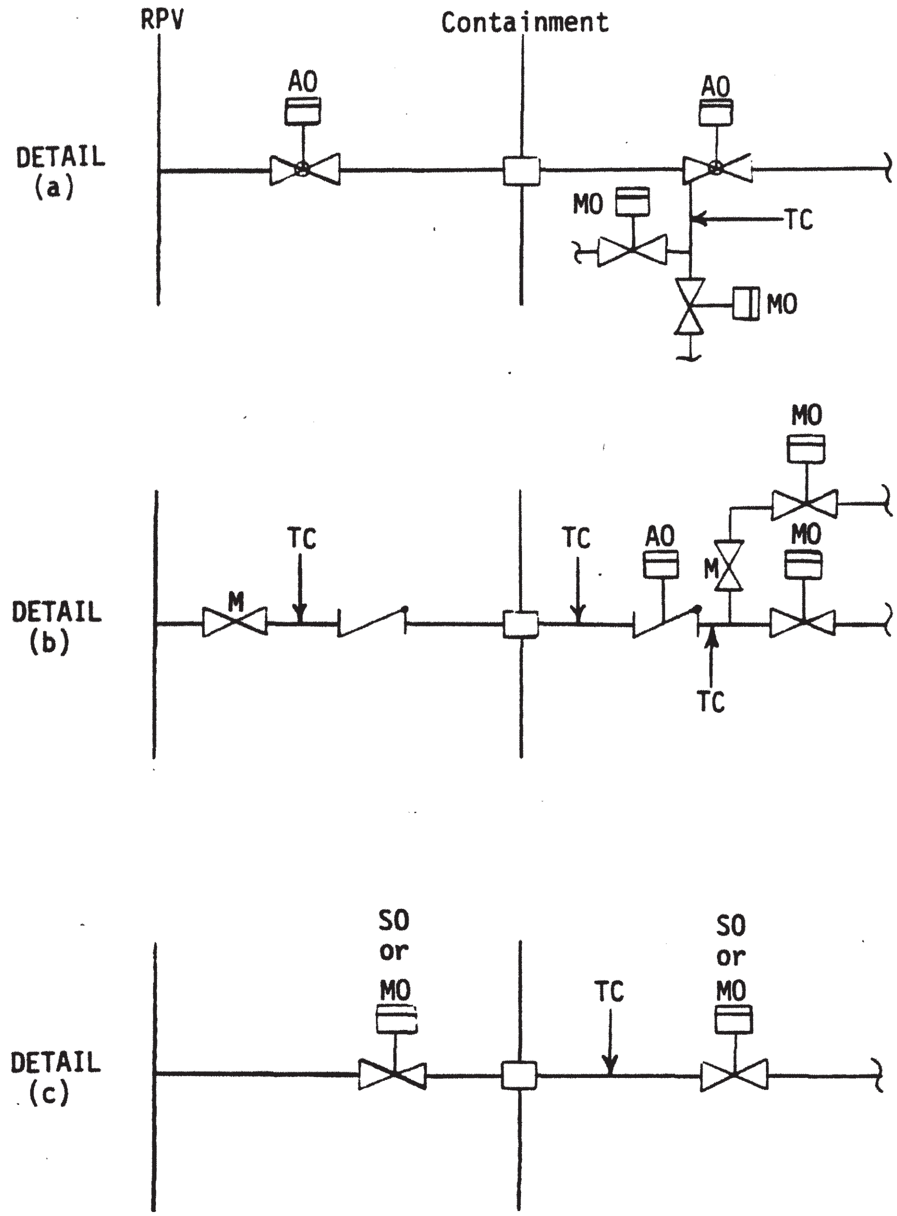
FIGURE 6.2-30 SENSIBLE ENERGY TRANSIENT IN THE REACTOR VESSEL AND INTERNAL METALS (At 3434 MWt)



NOTE: TC DESIGNATES TEST CONNECTION.

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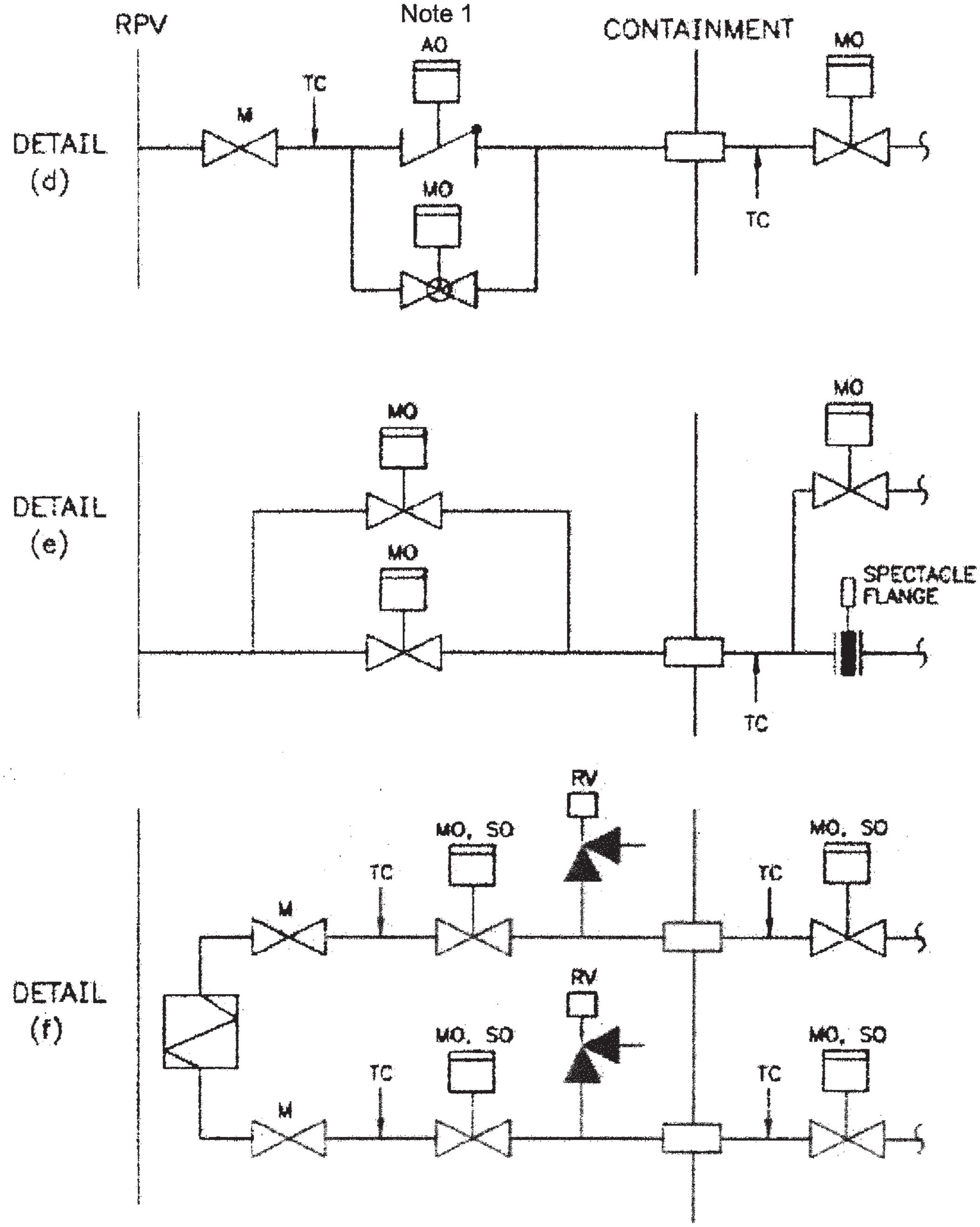
FIGURE 6.2-31

CONTAINMENT VALVE ARRANGEMENTS

(SHEET 1 of 10)

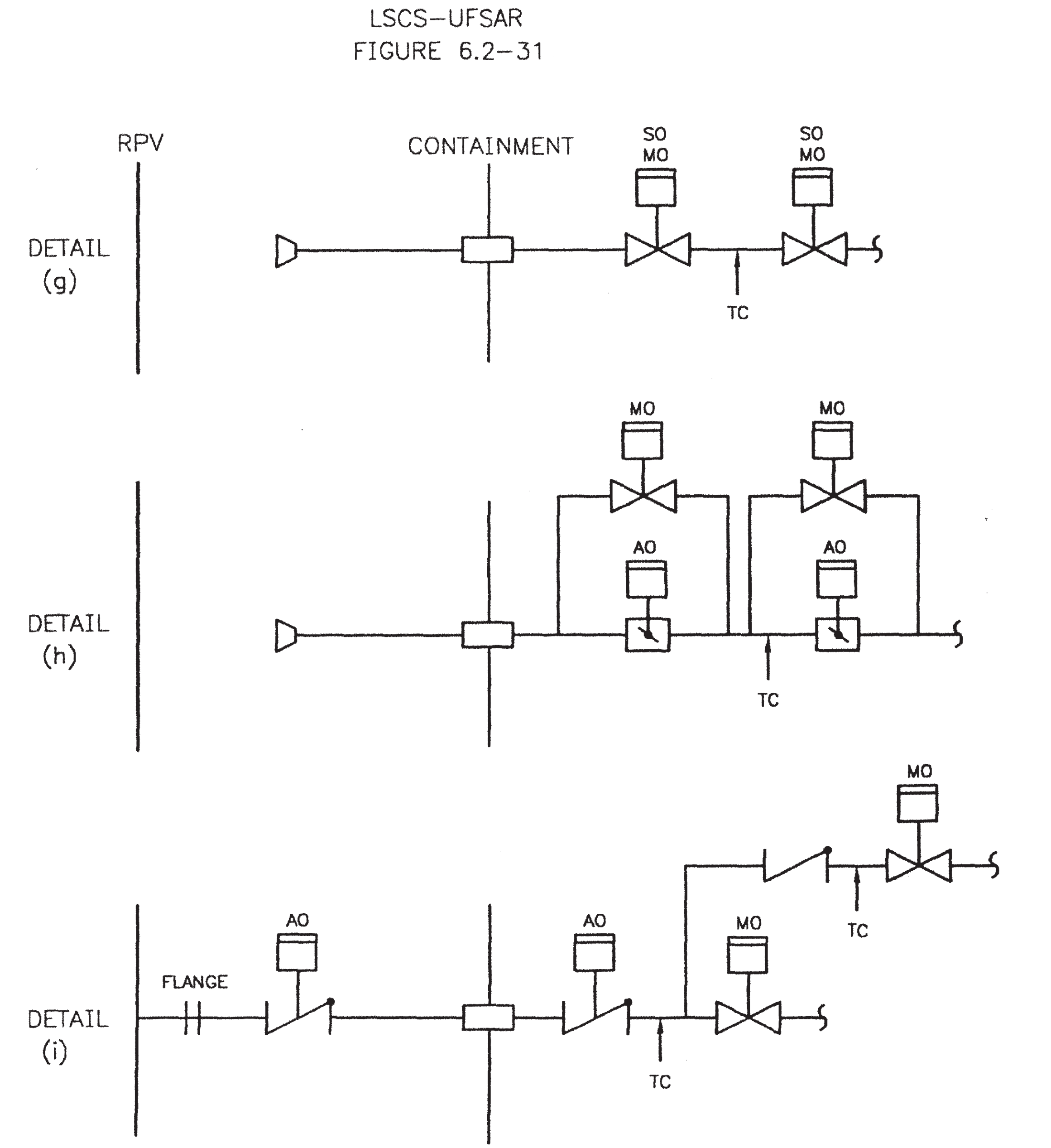
REV. 9 - APRIL 1993



Note 1: The Air Actuators are removed from Check Valves 1(2)E12-F050A/B.

FIGURE 6.2-31

LA SALLE COUNTY STATION UPDATED FINAL SAFETY ANALYSIS REPORT
FIGURE 6.2-31 CONTAINMENT VALVE ARRANGEMENTS (SHEET 2 OF 10)



LA SALLE COUNTY STATION
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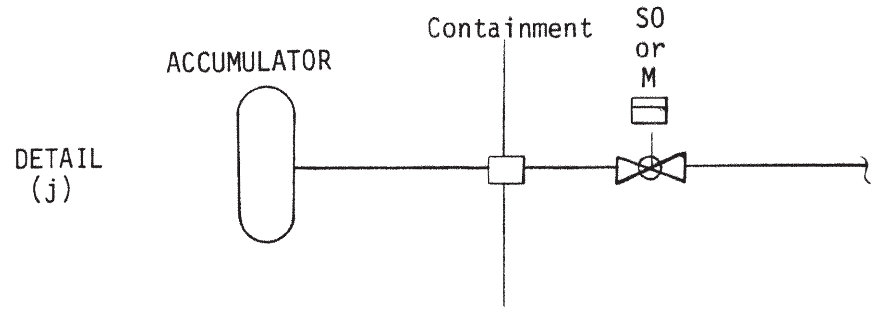
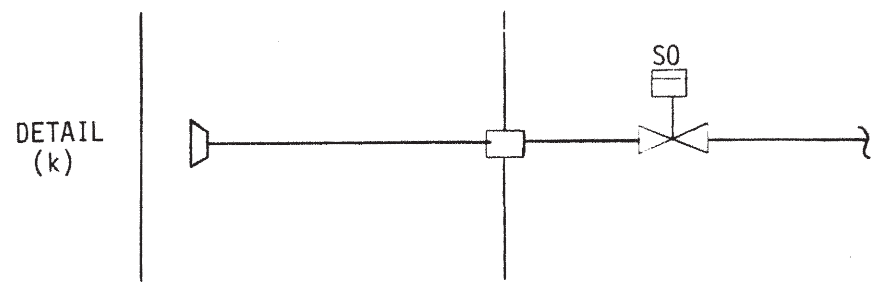
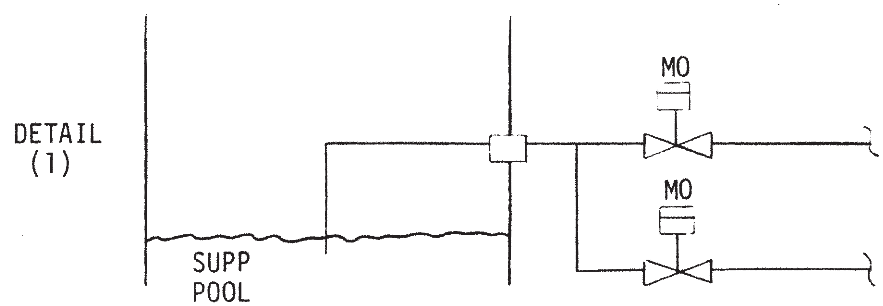
FIGURE 6.2-31

CONTAINMENT VALVE ARRANGEMENTS
(SHEET 3 OF 10)







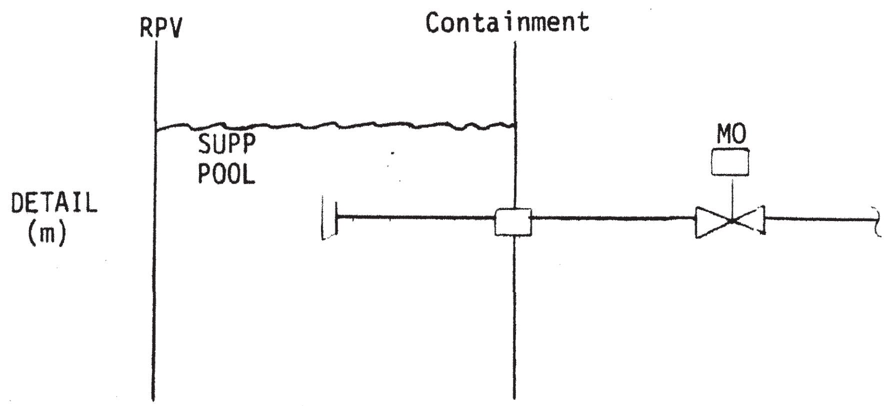
LA SALLE COUNTY STATION
UPDATED FINAL SAFETY ANALYSIS REPORT

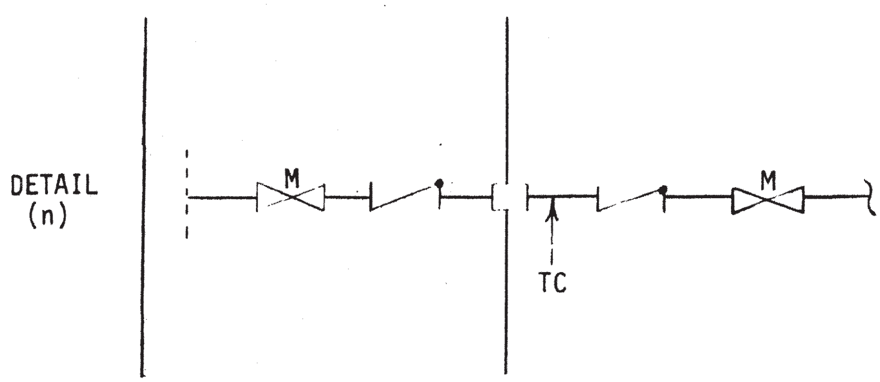
FIGURE 6.2-31

CONTAINMENT VALVE ARRANGEMENTS

(SHEET 4 of 10)

REV. 0 - APRIL 1984





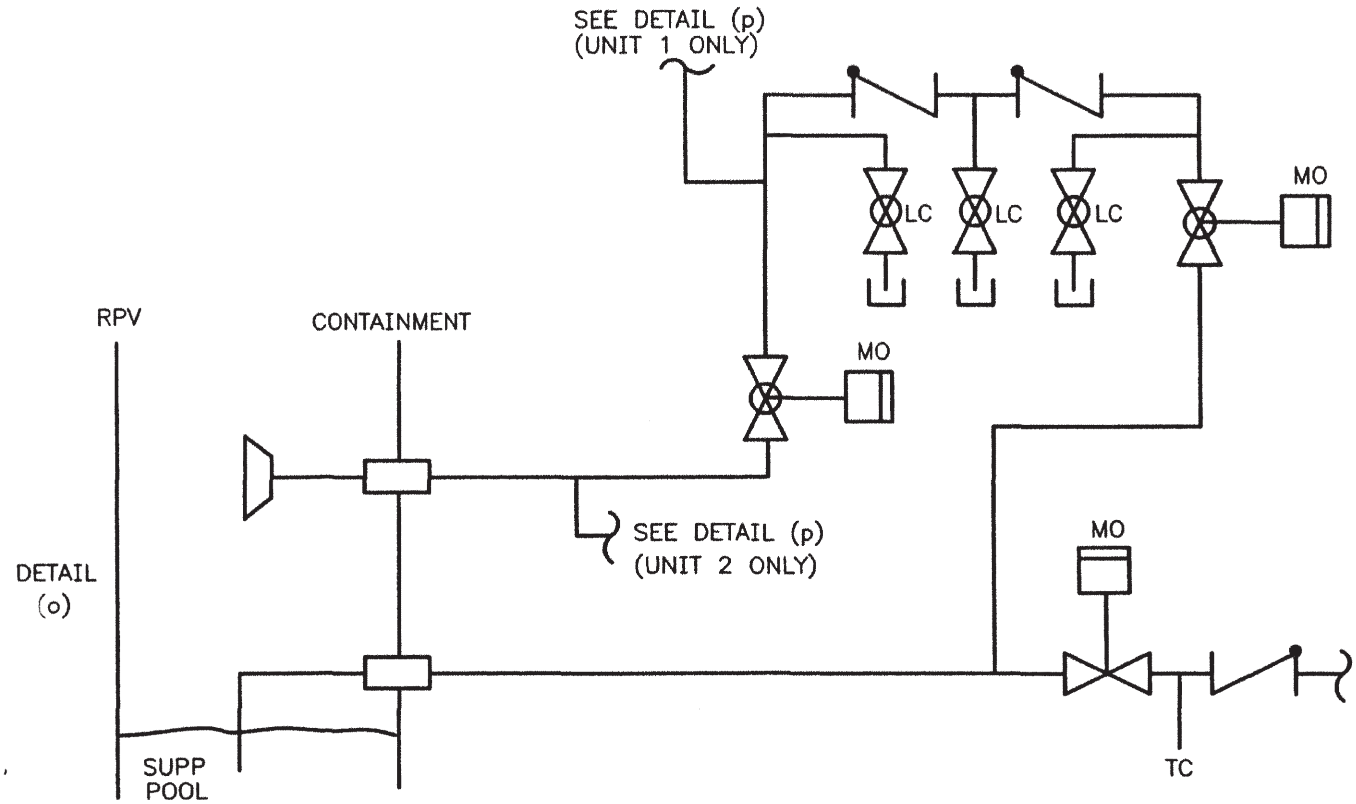
LA SALLE COUNTY STATION
UPDATED FINAL SAFETY ANALYSIS REPORT

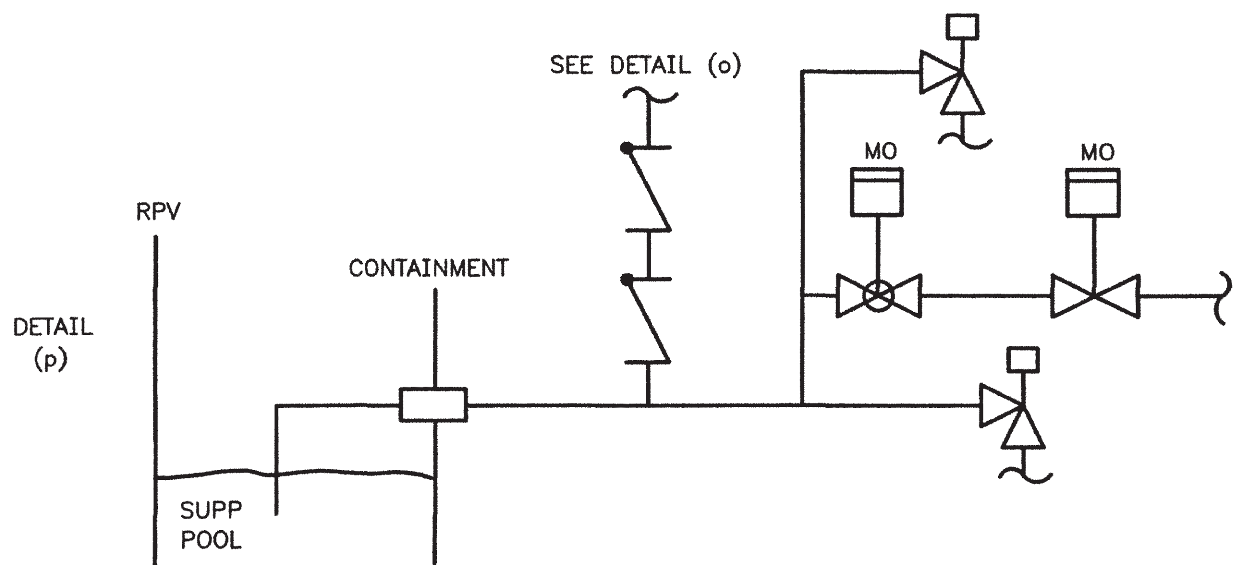
FIGURE 6.2-31

CONTAINMENT VALVE ARRANGEMENTS

(SHEET 5 of 10)

REV. 0 - APRIL 1984





LASALLE COUNTY STATION
UPDATED FINAL SAFETY ANALYSIS
REPORT

FIGURE 6.2–31

CONTAINMENT VALVE ARRANGEMENTS
(SHEET 6 OF 10)

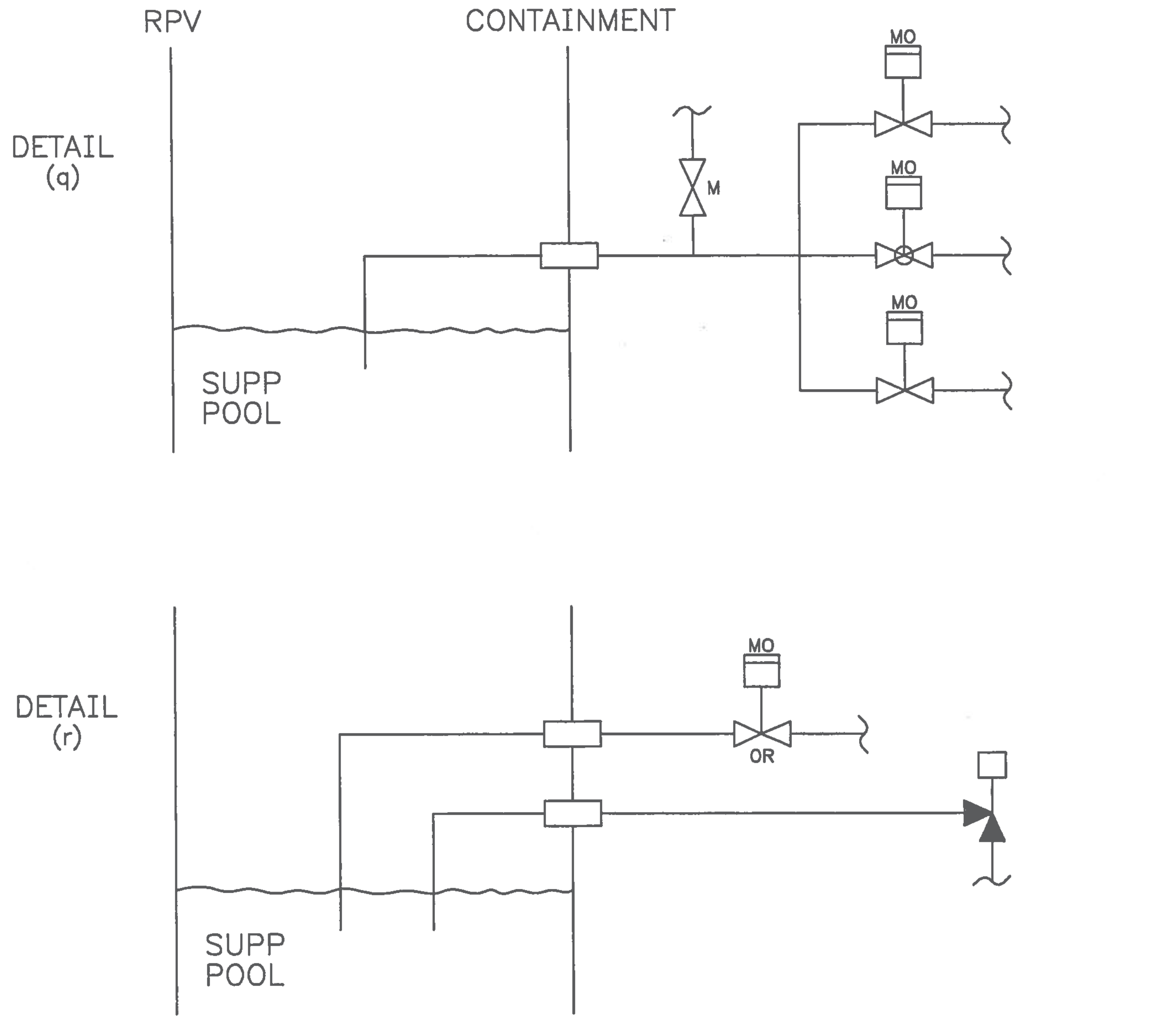


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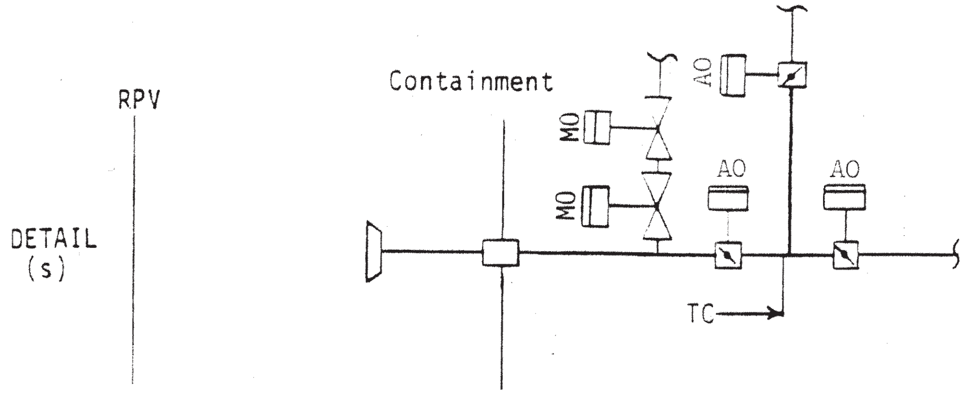
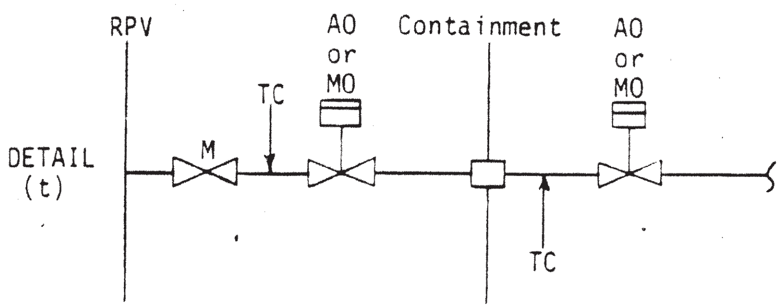
FIGURE 6.2-31

CONTAINMENT VALVE ARRANGEMENTS
(SHEET 7 OF 10)





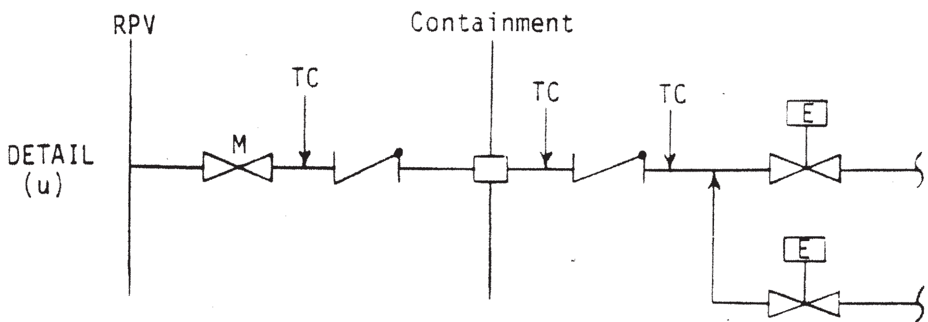


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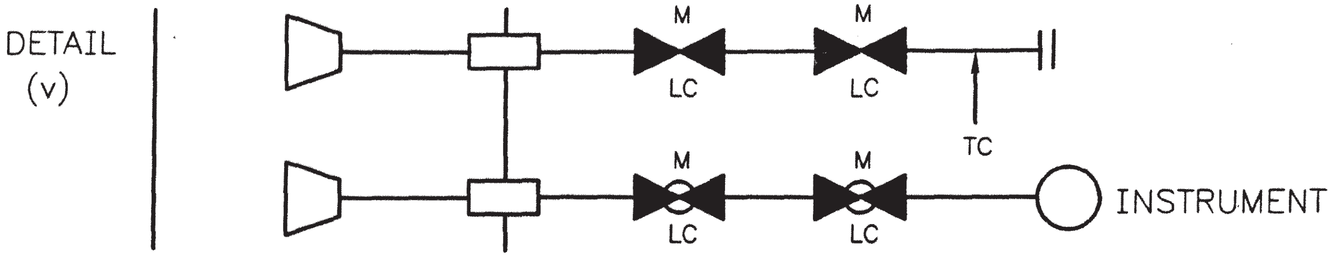
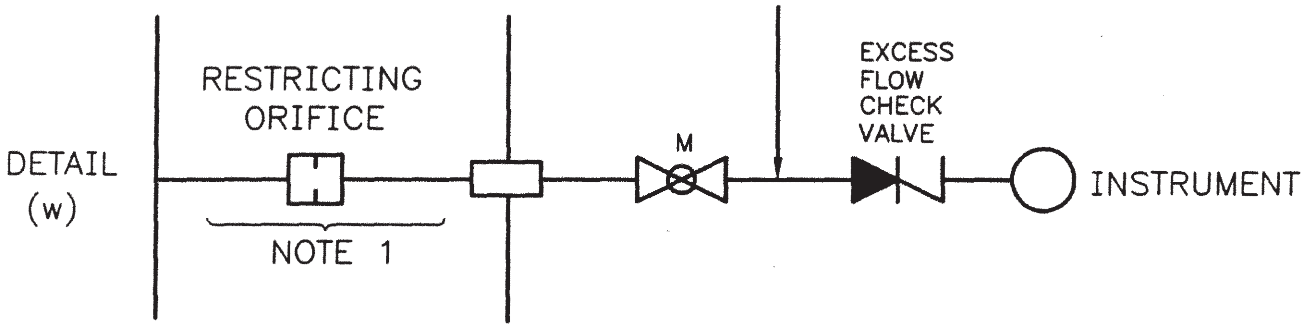
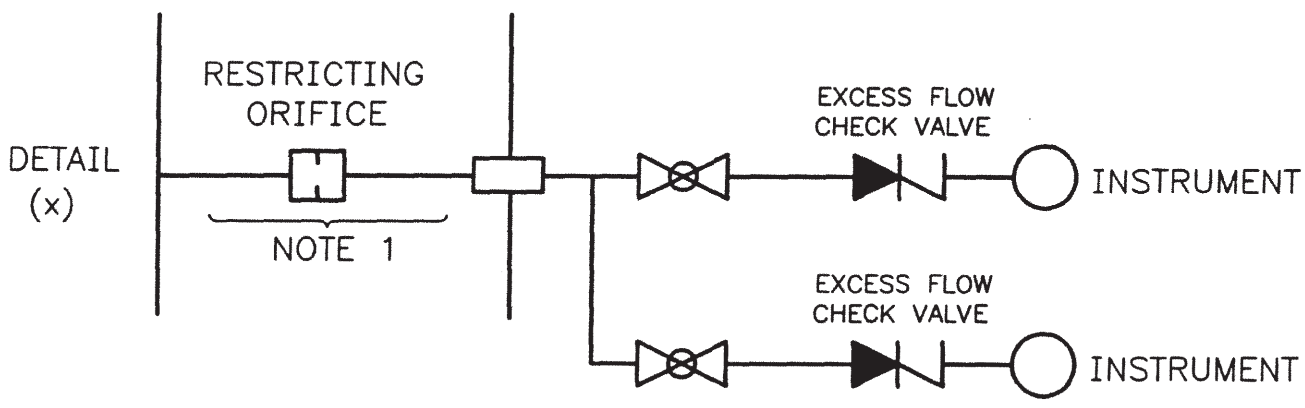
FIGURE 6.2-31 CONTAINMENT VALVE ARRANGEMENTS (SHEET 8 of 10)







NOTE 1: IN THOSE CASES WHERE INSTRUMENT LINES ARE DIRECTLY CONNECTED TO THE CONTAINMENT ATMOSPHERE, THE INBOARD PORTION IS BETTER REPRESENTED BY THE INBOARD PORTION IN DETAIL (v); HOWEVER, THE OUTBOARD PORTION REMAINS AS SHOWN HERE.

NOTE 2: WHERE PROVIDED, SEE CURRENT P & ID.

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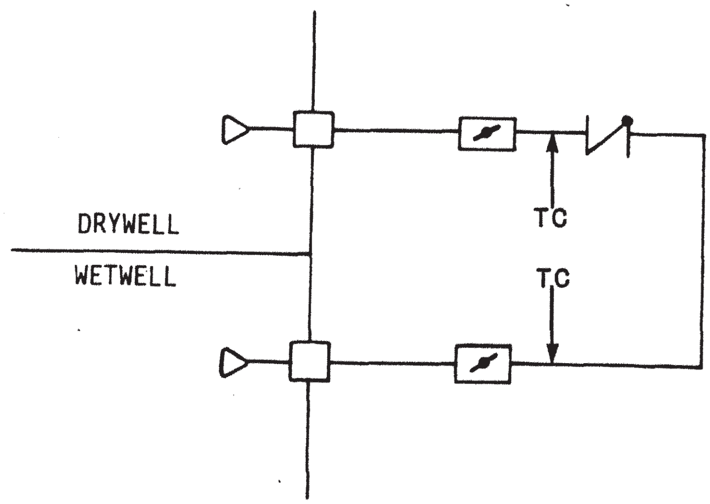
FIGURE 6.2-31 CONTAINMENT VALVE ARRANGEMENTS (SHEET 9 OF 10)

REVISION 13

DETAIL
(y)



DETAIL
(z)

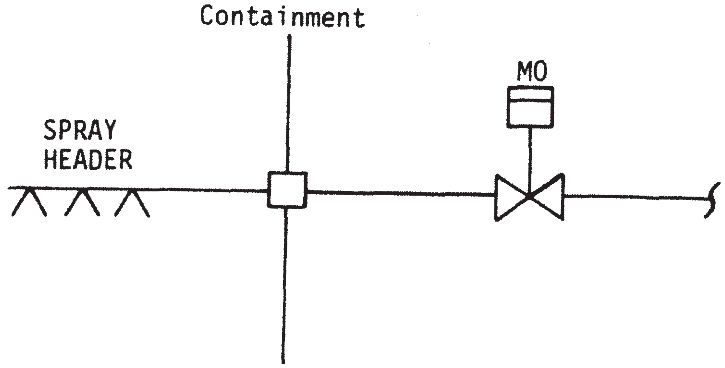


FIGURE 6.2-31

CONTAINMENT VALVE ARRANGEMENTS

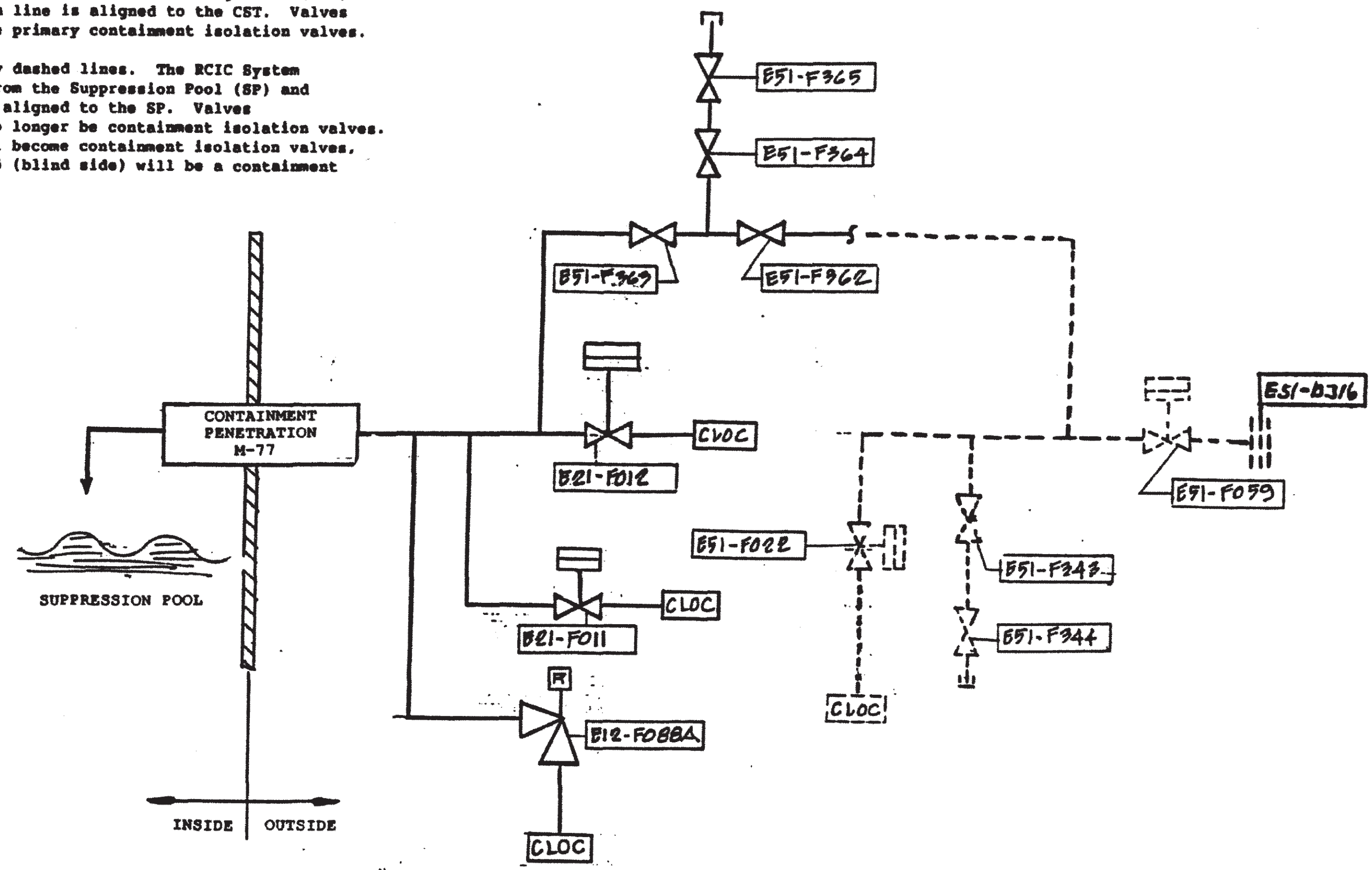
(SHEET 10 of 10)

Discussion:

The CLOC represents system boundaries (valves, flanges, pump seals, etc.) which are normally sealed closed, automatically closed, or are closed with a remote manual operator to accomplish containment isolation.

Test Mode 1 is represented by solid lines. The RCIC System is aligned to take suction from the condensate storage tank (CST) and the full flow test return line is aligned to the CST. Valves E51-F362 and F363 will become primary containment isolation valves.

Test Mode 2 is represented by dashed lines. The RCIC System is aligned to take suction from the Suppression Pool (SP) and the flow test return line is aligned to the SP. Valves E51-F362 and E51-F363 will no longer be containment isolation valves. Valves E51-F022 and F059 will become containment isolation valves, and spectacle flange E51-D316 (blind side) will be a containment isolation boundary.



DETAIL AA

LA SALLE COUNTY STATION
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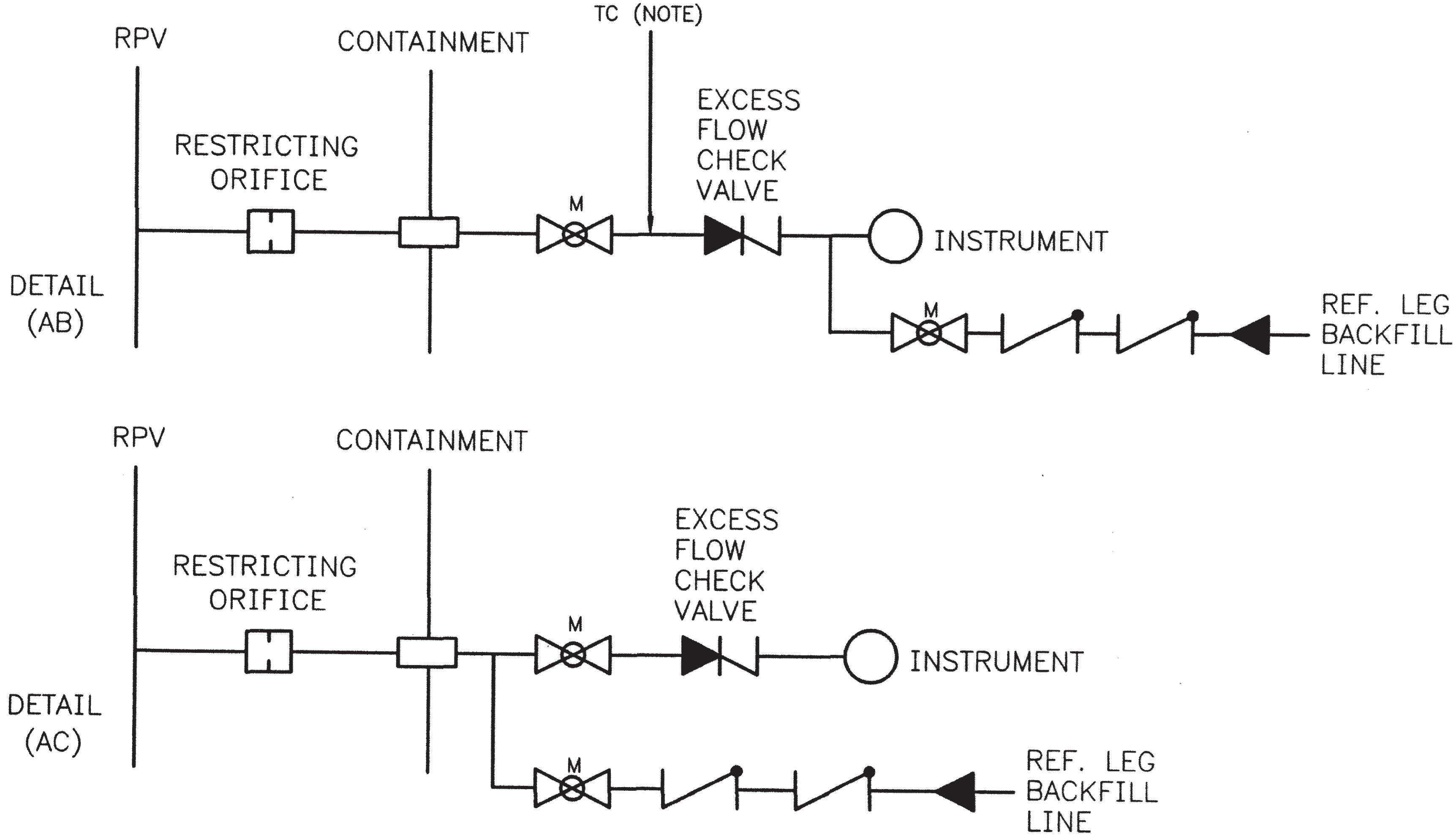
FIGURE 6.2-31

Containment Valve Arrangements
(Sheet 10a of 10)



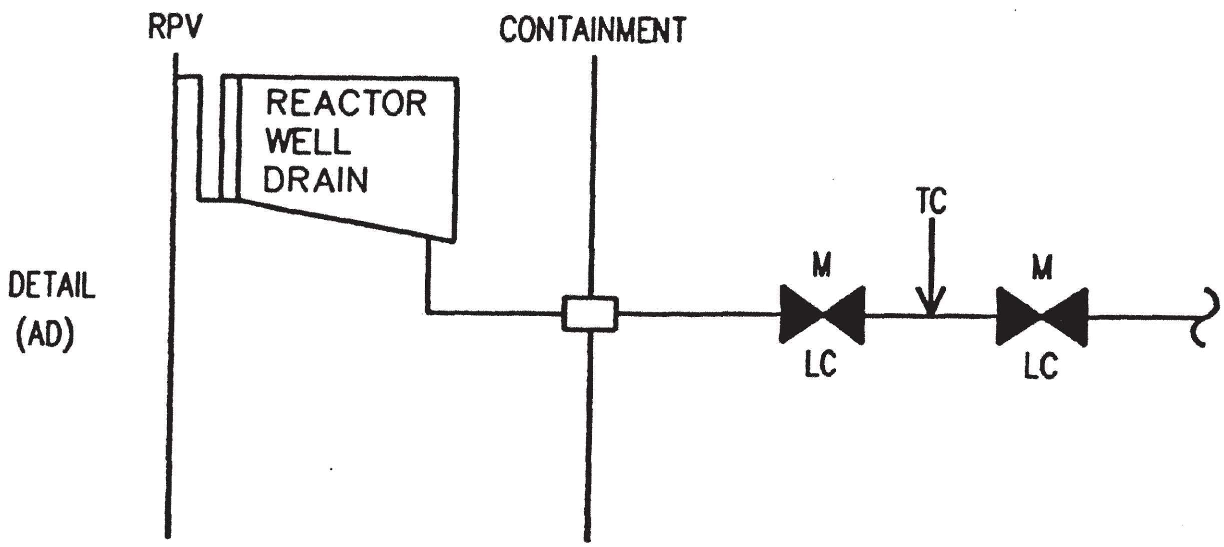
NOTE: WHERE PROVIDED, SEE CURRENT P & ID.

FIGURE 6.2-31

LA SALLE COUNTY STATION UPDATED FINAL SAFETY ANALYSIS REPORT
FIGURE 6.2-31 CONTAINMENT VALVE ARRANGEMENTS (SHEET 10B OF 10)

REVISION 13

FIGURE 6.2-31



NOTE: THIS FIGURE APPLIES TO UNIT 2 ONLY.

LASALLE COUNTY STATION UPDATED FINAL SAFETY ANALYSIS REPORT
FIGURE 6.2-31 CONTAINMENT VALVE ARRANGEMENTS (SHEET 10C OF 10)

FIGURE 6.2-31

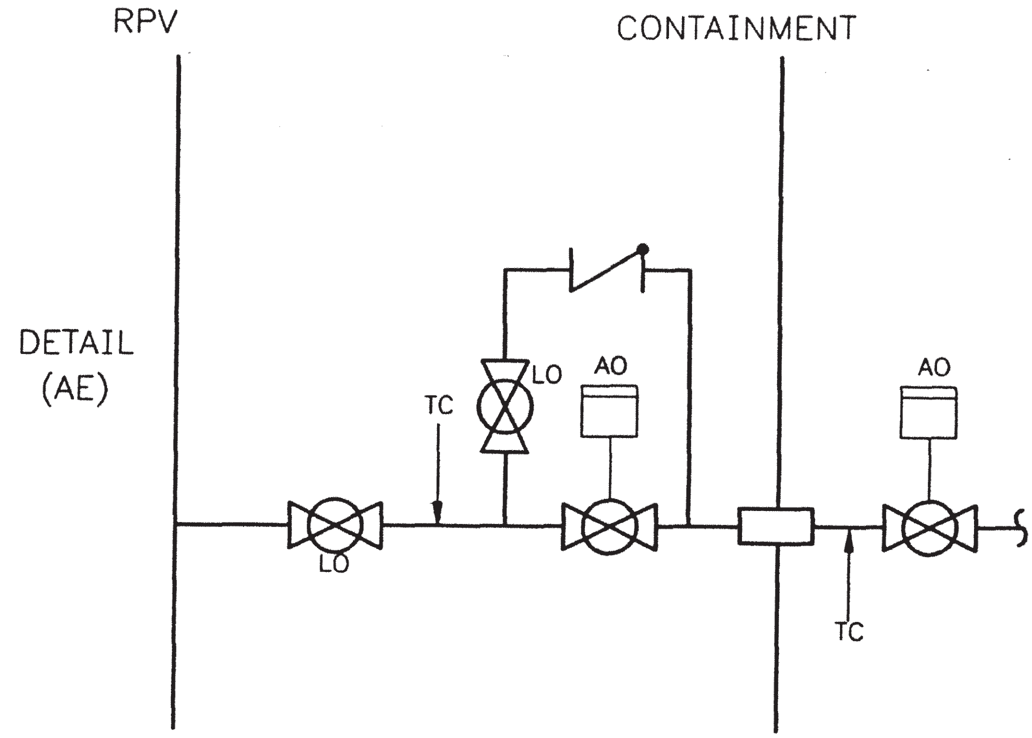


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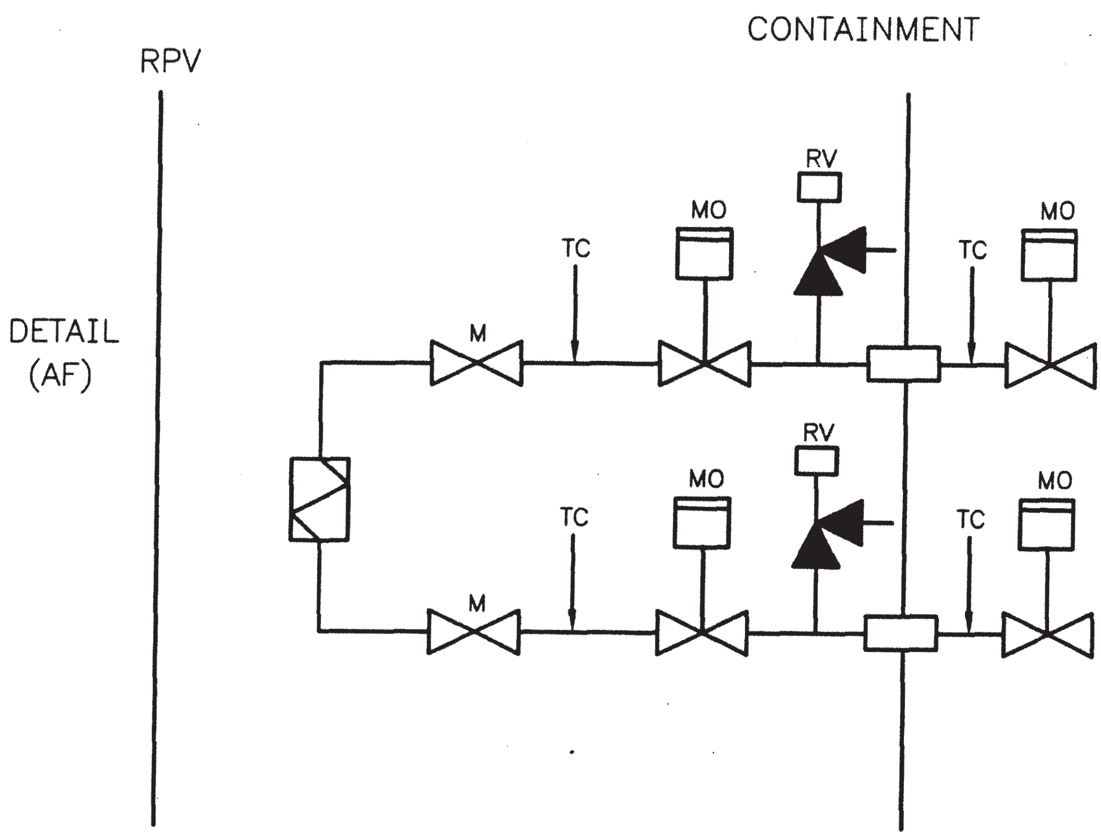
FIGURE 6.2-31

CONTAINMENT VALVE ARRANGEMENTS
(SHEET 10D OF 10)

REVISION 13



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REPORT

FIGURE 6.2-31

CONTAINMENT VALVE ARRANGEMENTS
(SHEET 10E OF 10)

REVISION 13

FIGURE 6.2-31

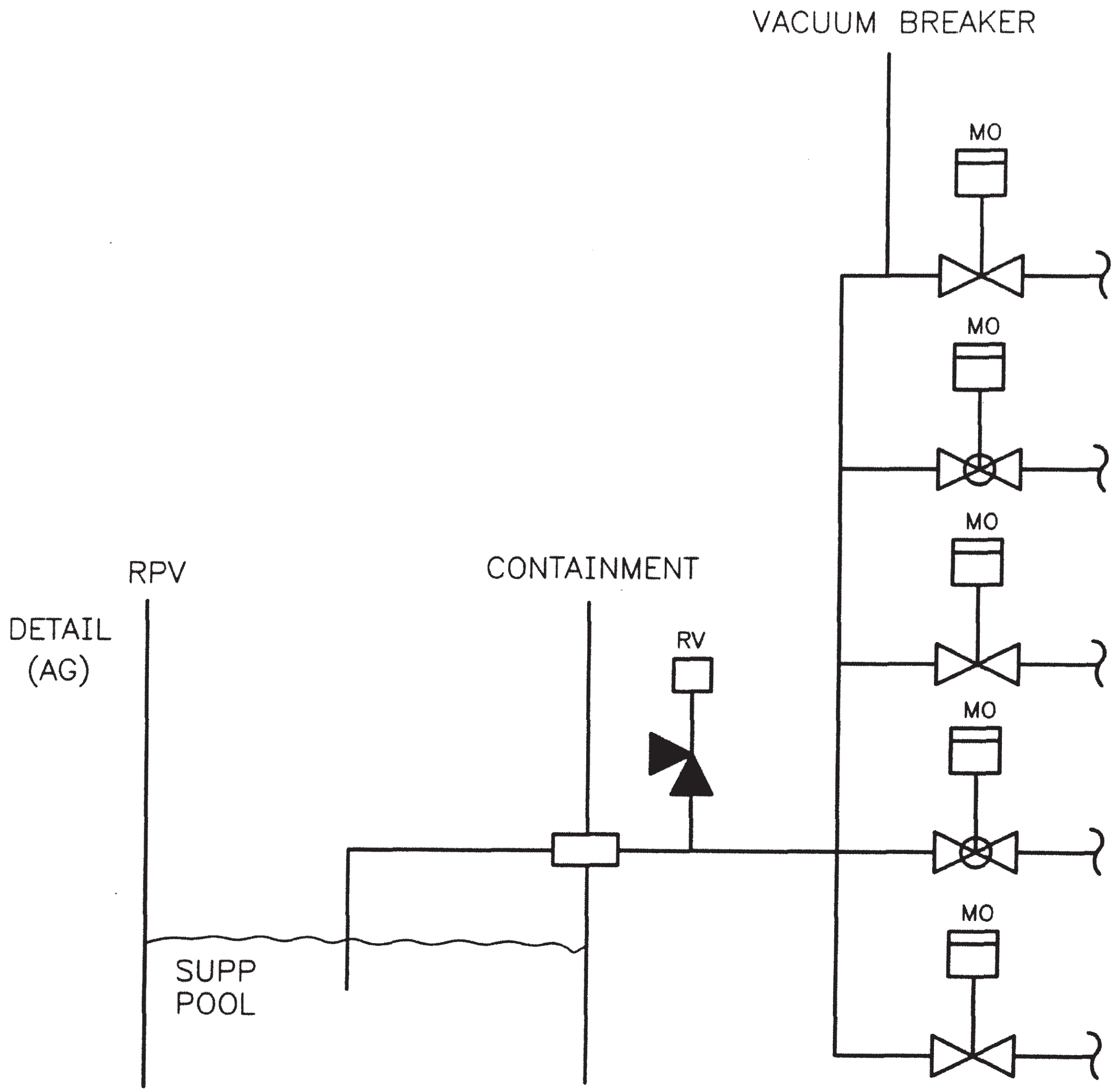

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FIGURE 6.2-31

CONTAINMENT VALVE ARRANGEMENTS
(SHEET 10F OF 10)

REVISION 13

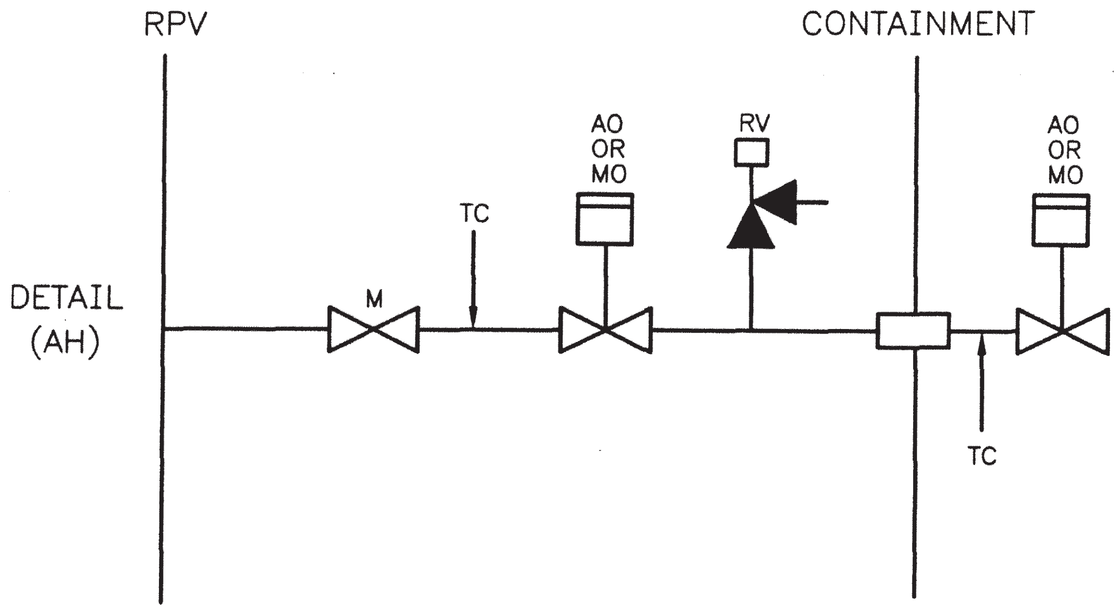

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FIGURE 6.2-31

CONTAINMENT VALVE ARRANGEMENTS
(SHEET 10G OF 10)

REVISION 13

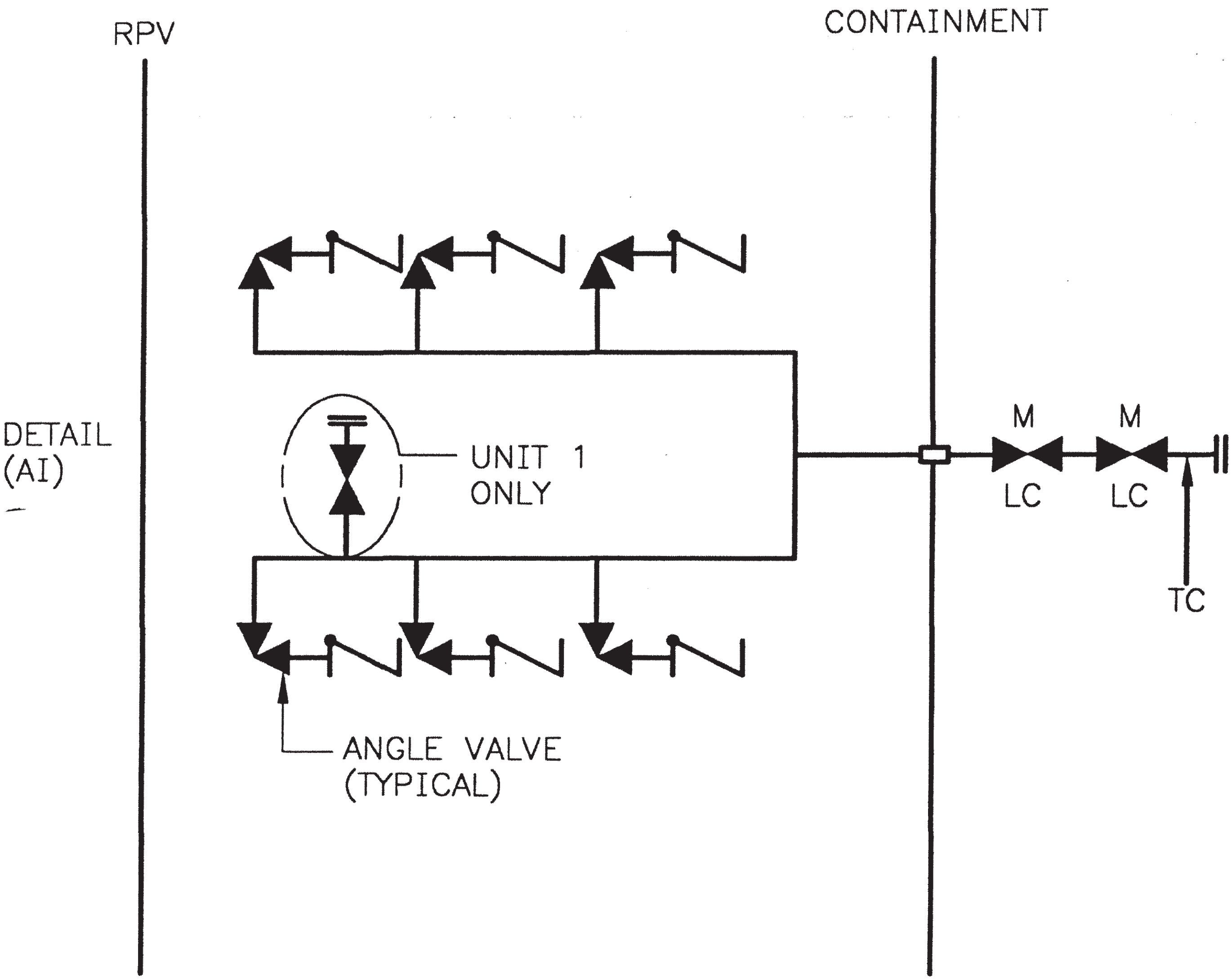

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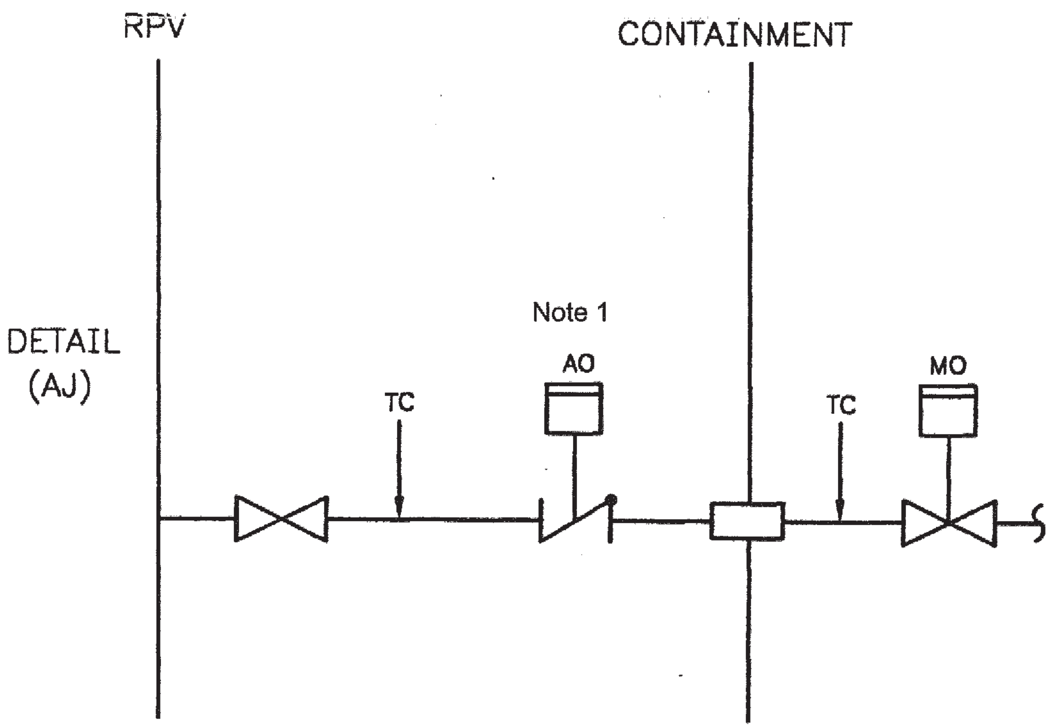
FIGURE 6.2–31
CONTAINMENT VALVE ARRANGEMENTS
(SHEET 10H OF 10)

REV. 13



Note: The Air Actuators are removed from Check Valves 1(2)E21-F006, 1(2)E22-F005, and 1(2)E12-F041A/B/C.

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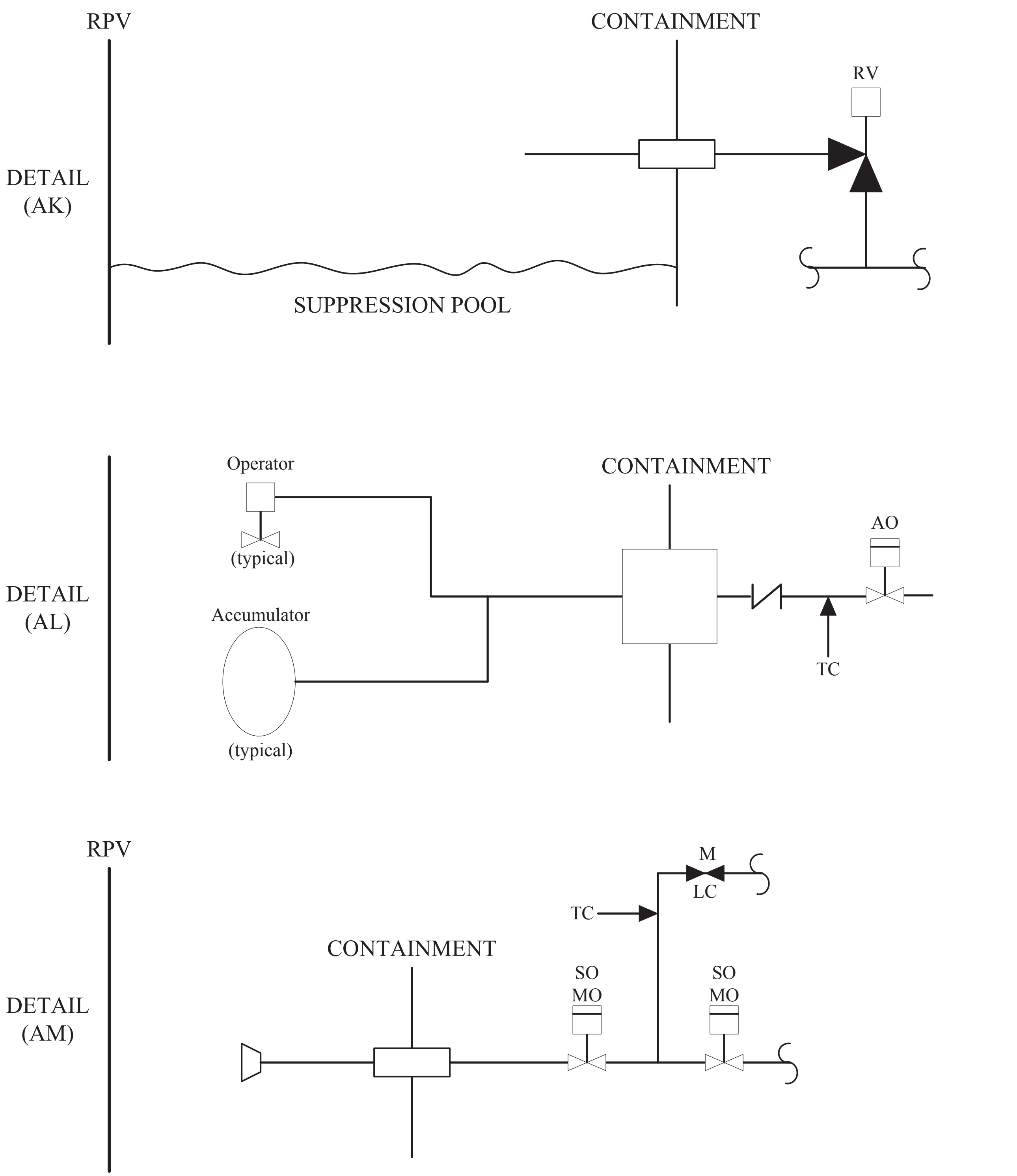
FIGURE 6.2–31 CONTAINMENT VALVE ARRANGEMENTS (SHEET 10I OF 10)



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UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE 6.2-31

CONTAINMENT VALVE ARRANGEMENTS
(SHEET 10J OF 10)

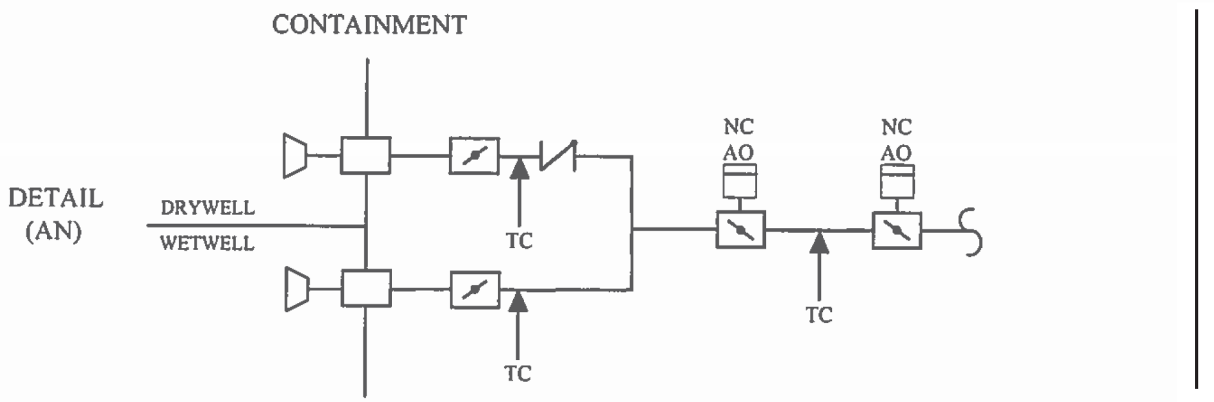

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FIGURE 6.2-31

CONTAINMENT VALVE ARRANGEMENTS
(SHEET 10K OF 10)

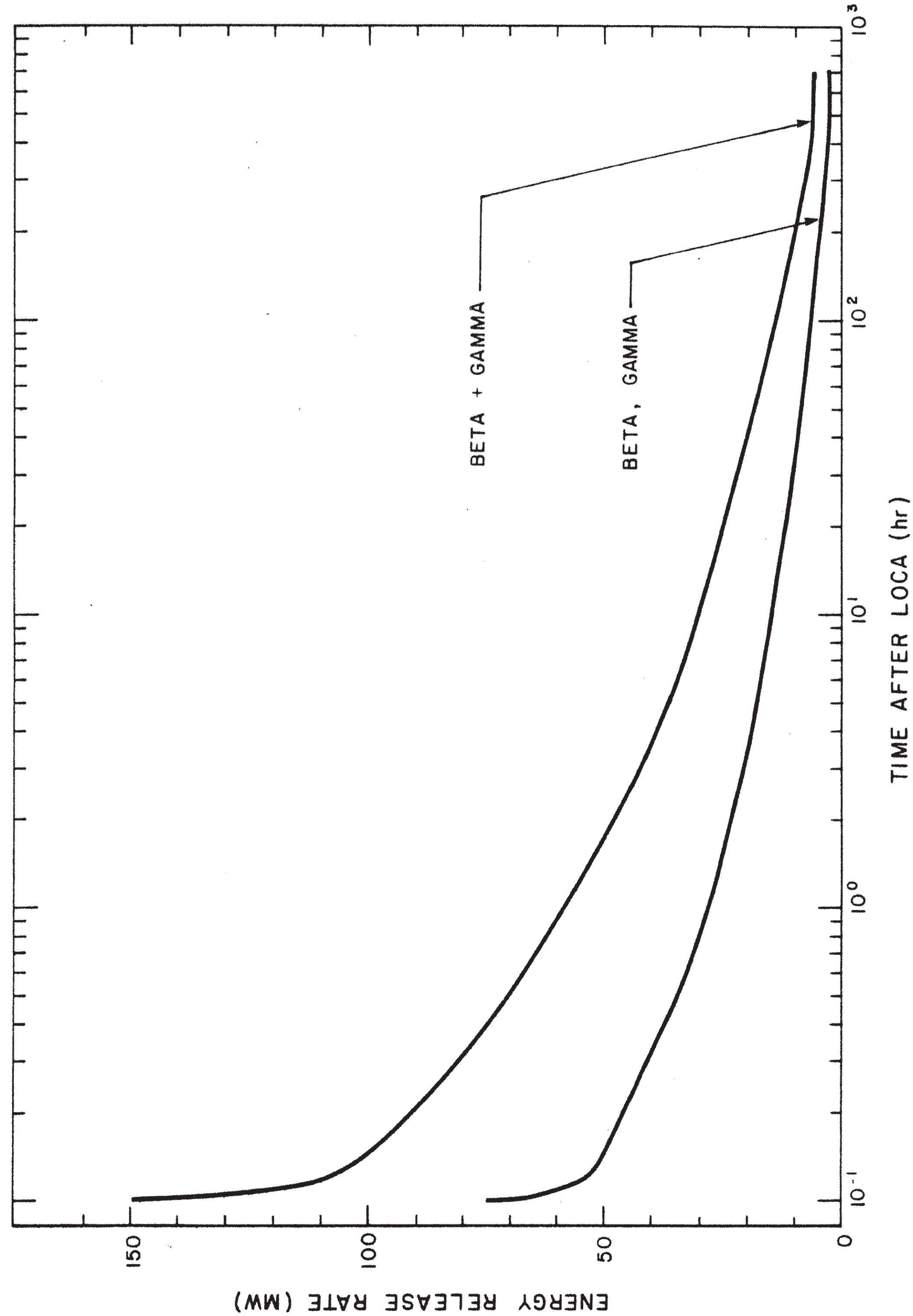

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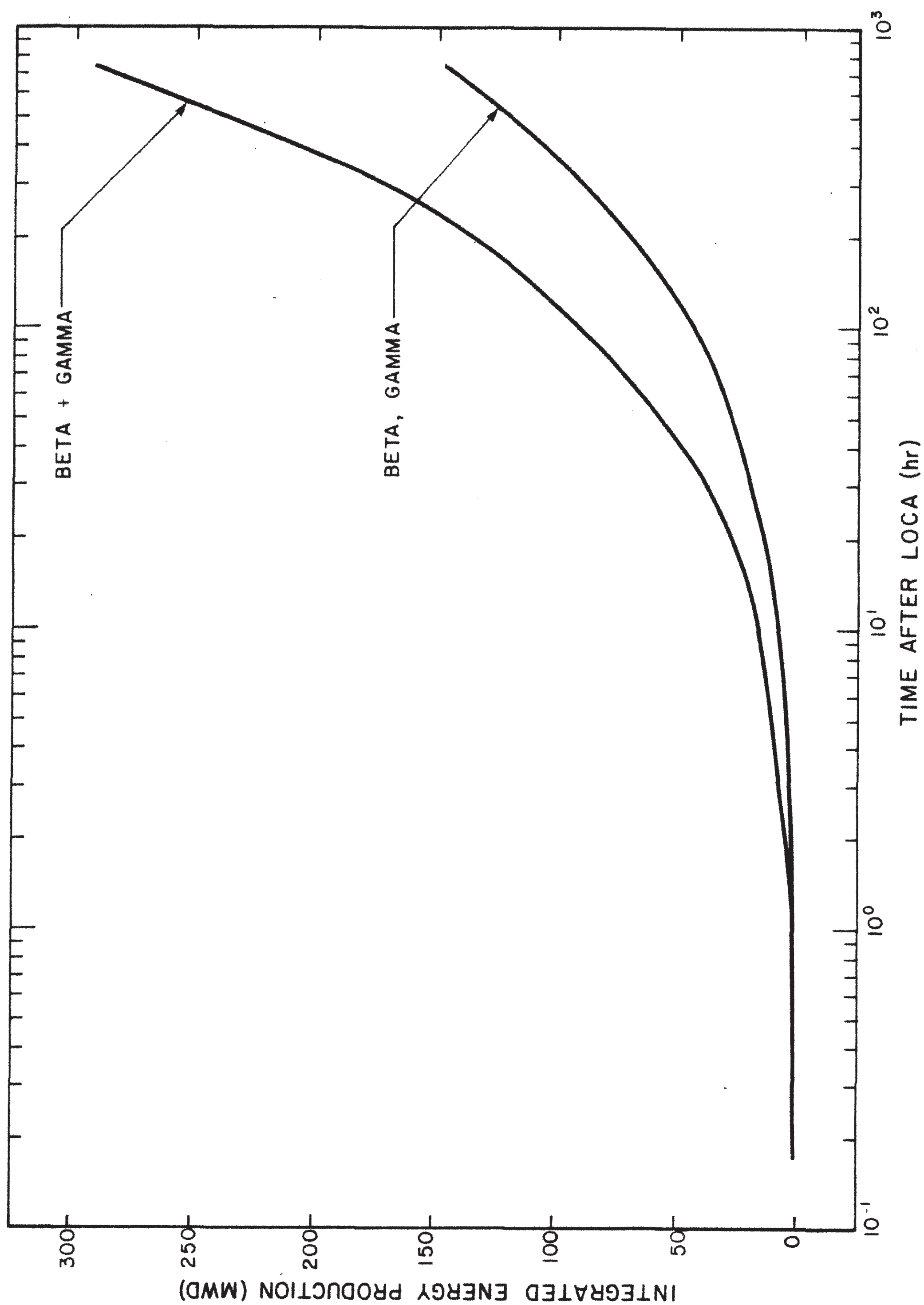
FIGURE 6.2-32

ENERGY RELEASE RATES AS
A FUNCTION OF TIME



LA SALLE COUNTY STATION
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FIGURE 6.2-33

INTEGRATED ENERGY RELEASE AS
A FUNCTION OF TIME

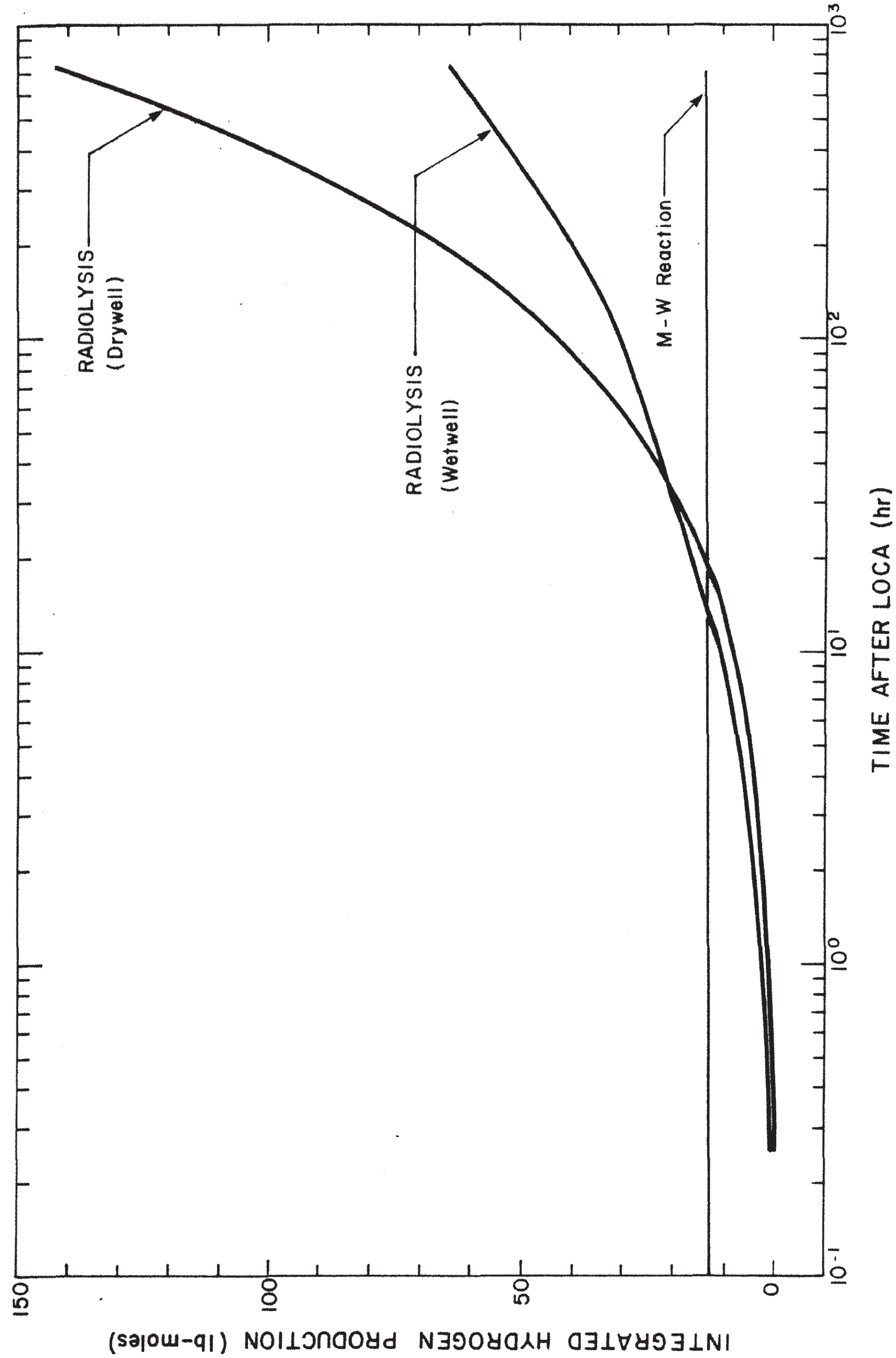

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FIGURE 6.2-34

INTEGRATED HYDROGEN PRODUCTION
AS A FUNCTION OF TIME

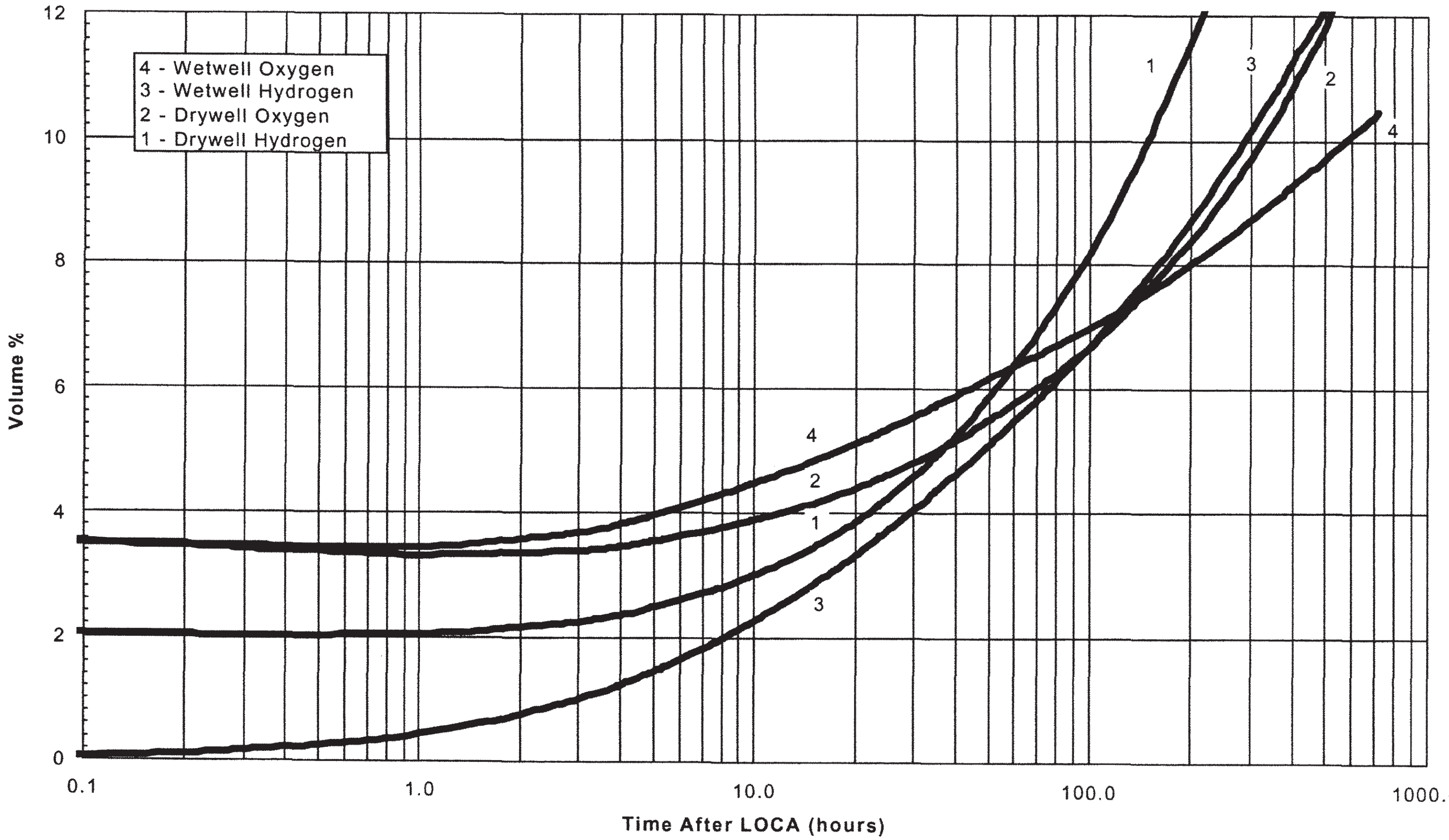

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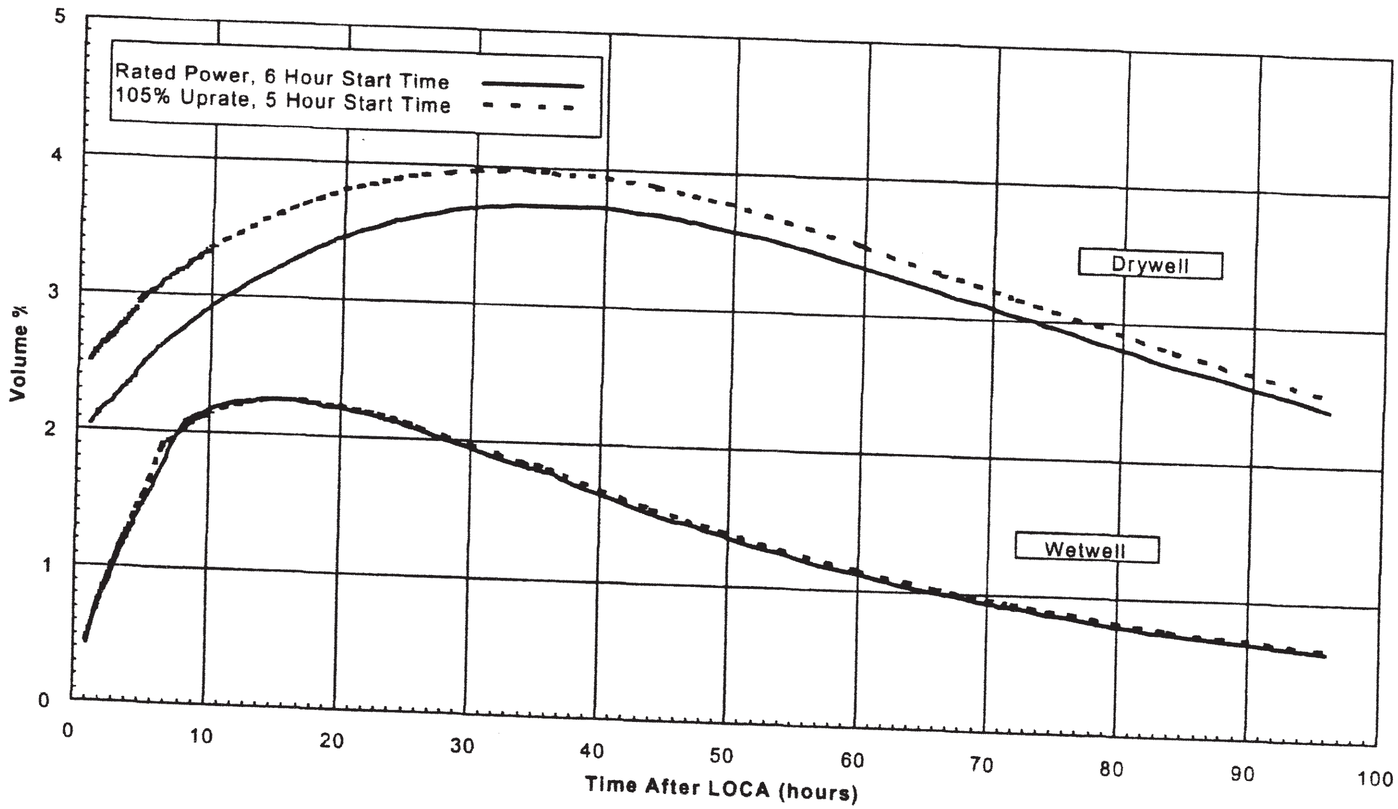
FIGURE 6.2-35

UNCONTROLLED HYDROGEN
AND OXYGEN GENERATION



Note: The information provide in this figure is historical. The hydrogen recombining function of the hydrogen recombiners is abandoned in place.

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FIGURE 6.2-36 HYDROGEN CONCENTRATION WITH 125 SCFM

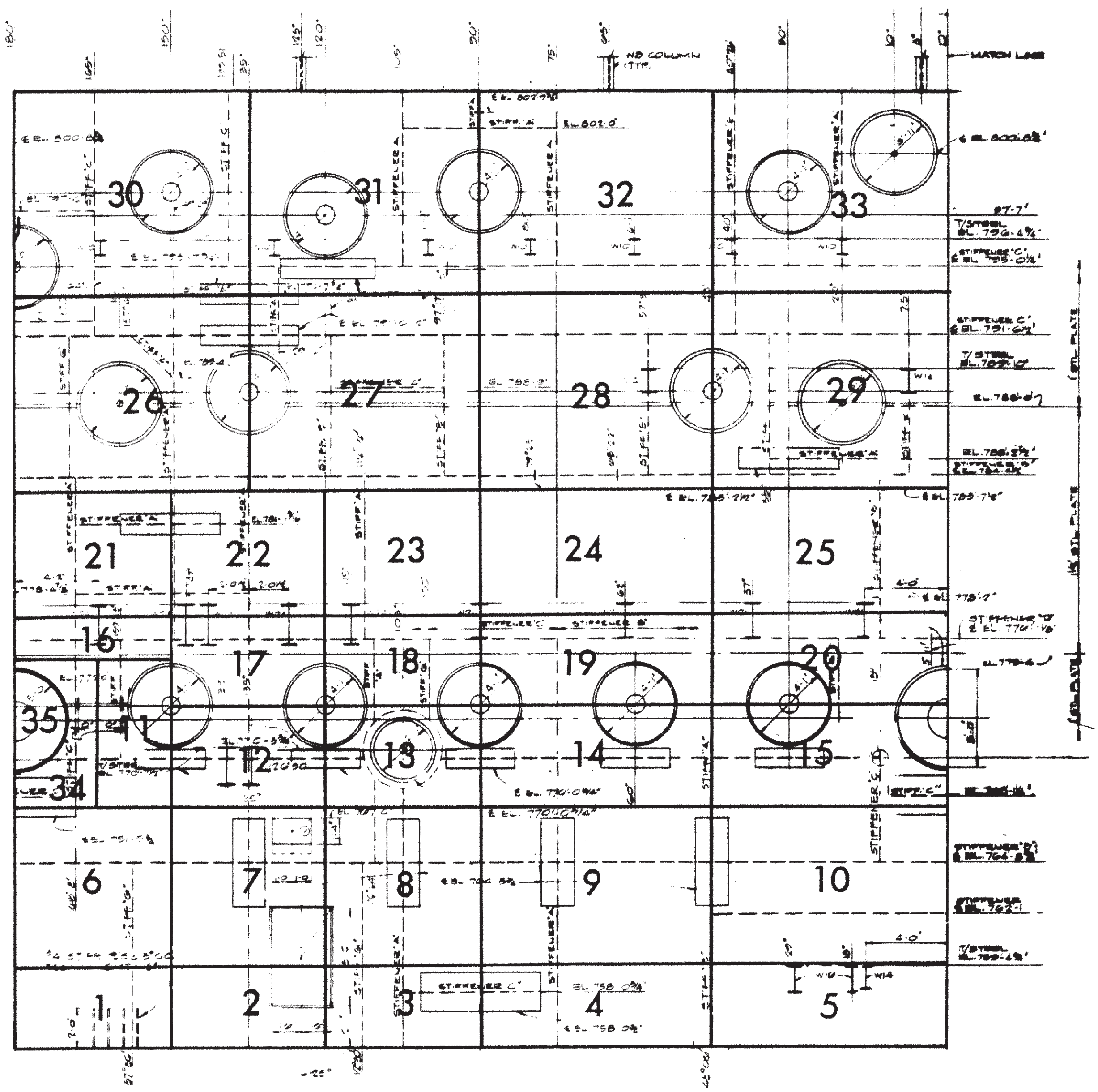

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FIGURE 6.2-37

NODALIZATION OVERLAY FOR
RECIRCULATION LINE BREAK

REV. 0 - APRIL 1984

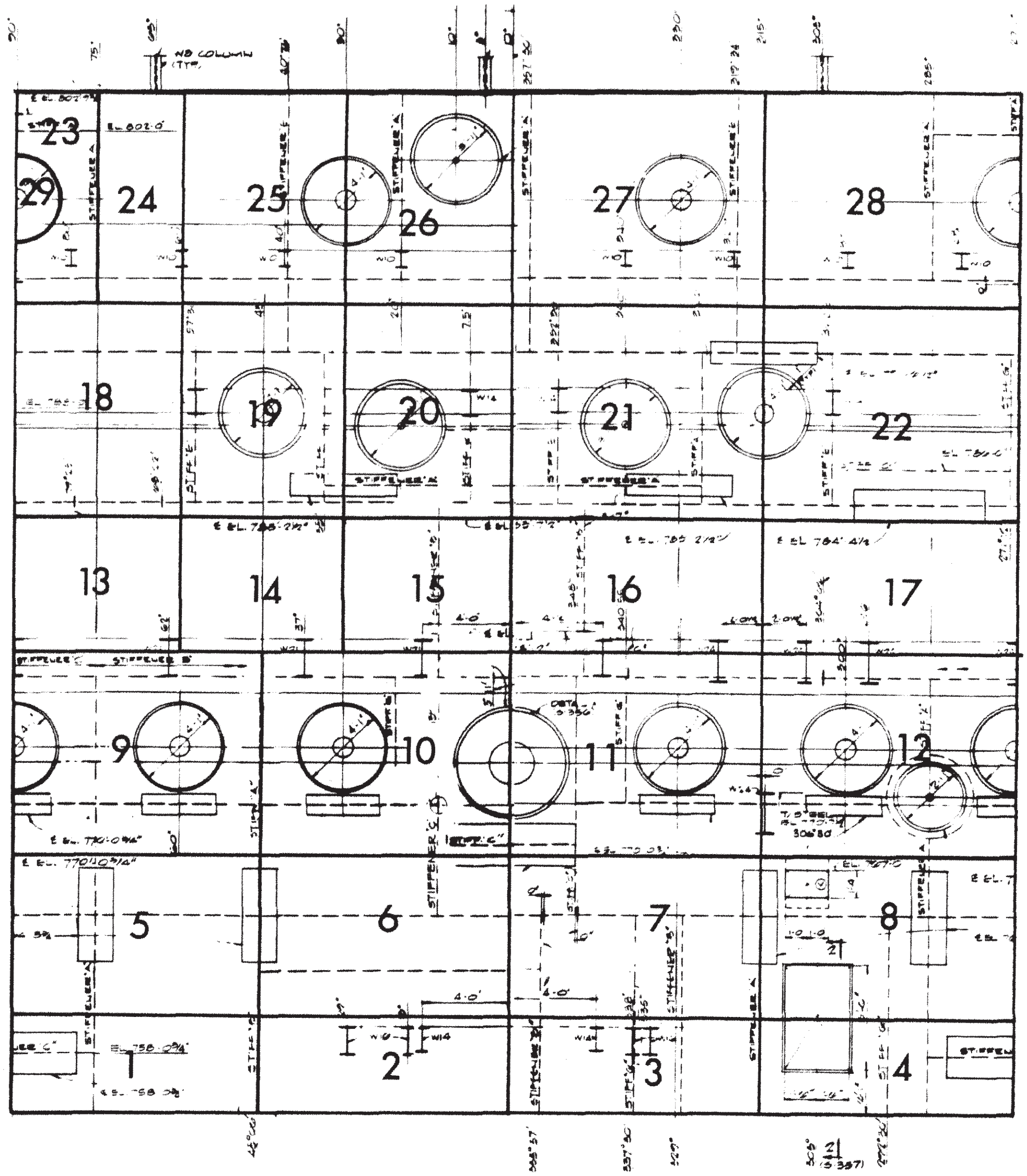

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FIGURE 6.2-38

NODALIZATION OVERLAY FOR
FEEDWATER LINE BREAK

REV. 0 - APRIL 1984

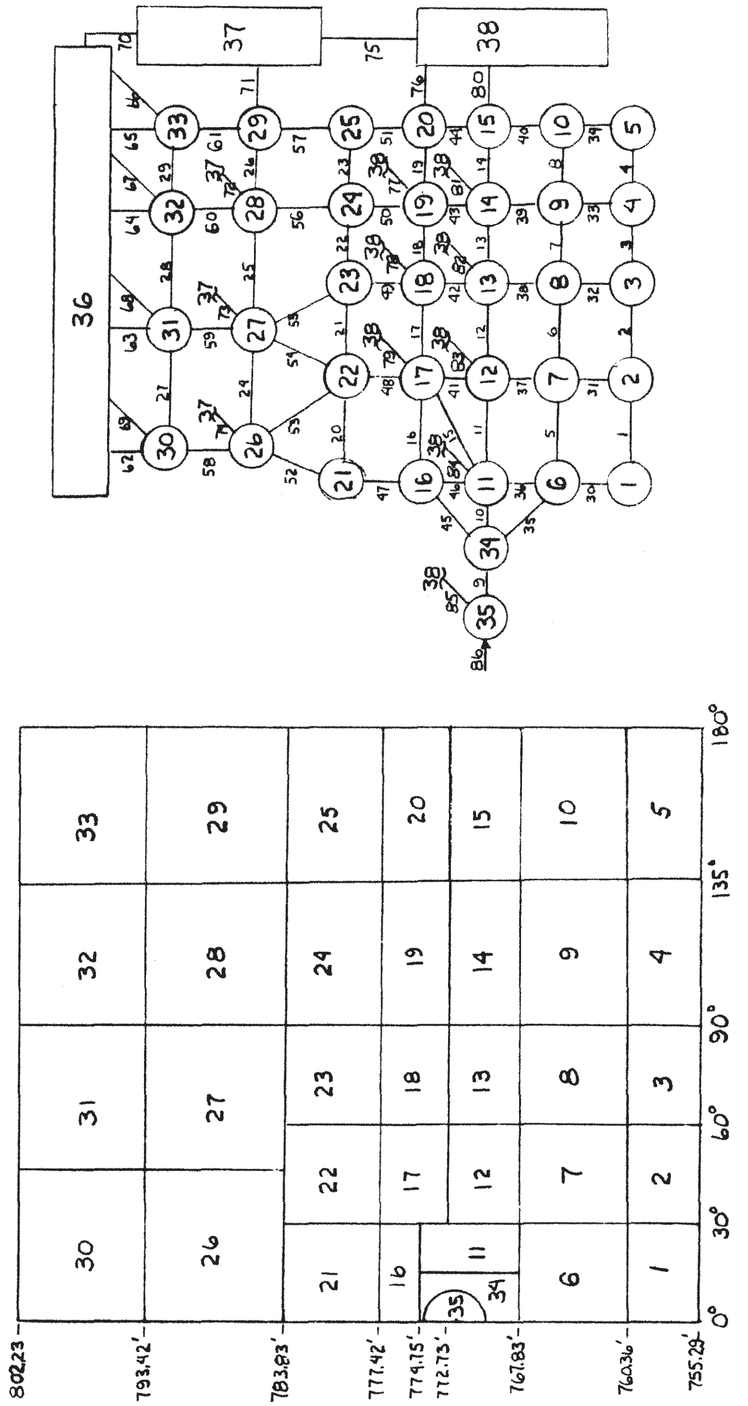

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FIGURE 6.2-39

NODALIZATION FOR ORIGINAL
RECIRCULATION LINE BREAK ANALYSIS

REV. 0 - APRIL 1984

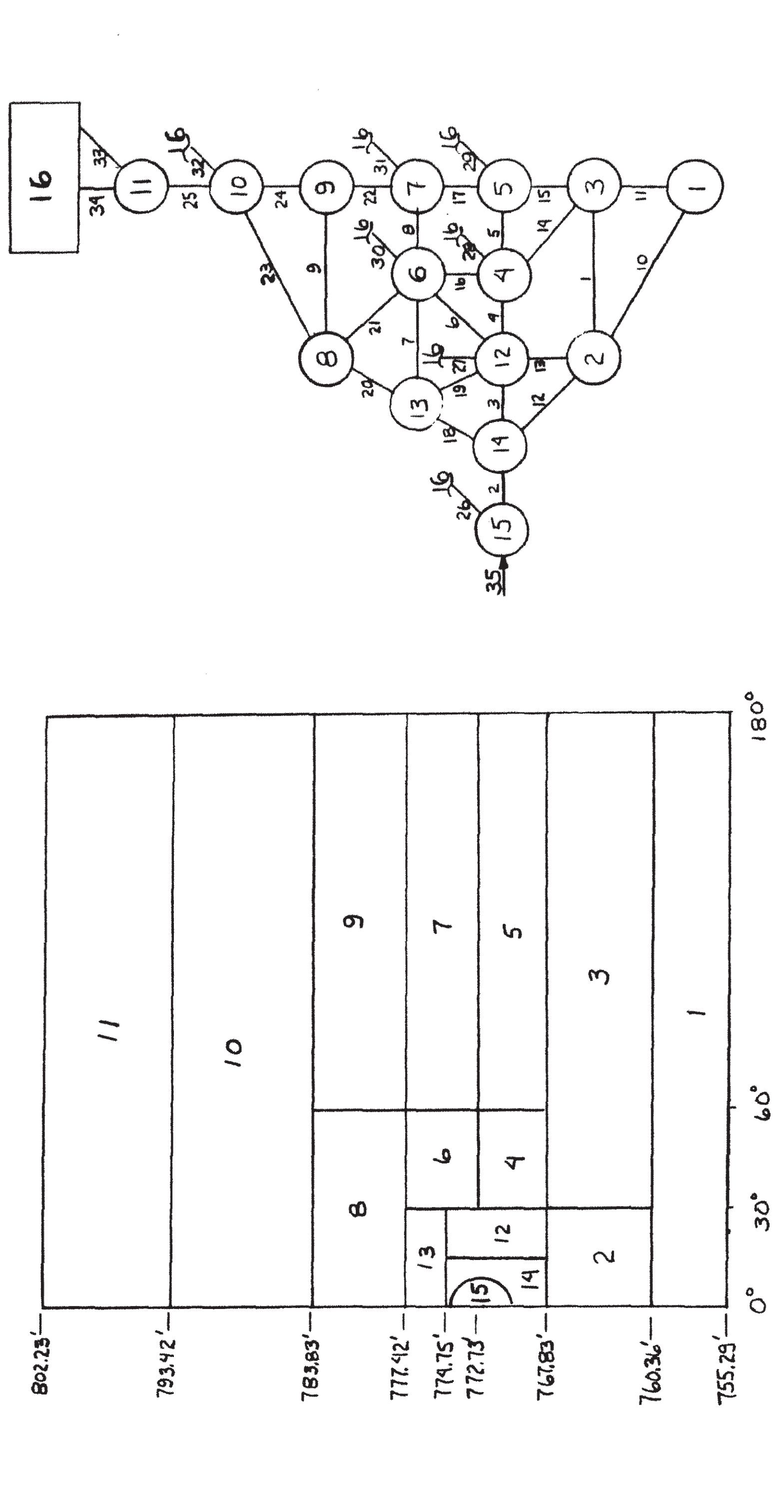

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FIGURE 6.2-40

"EQUIVALENT" NODALIZATION (CASE A)

REV. 0 - APRIL 1984

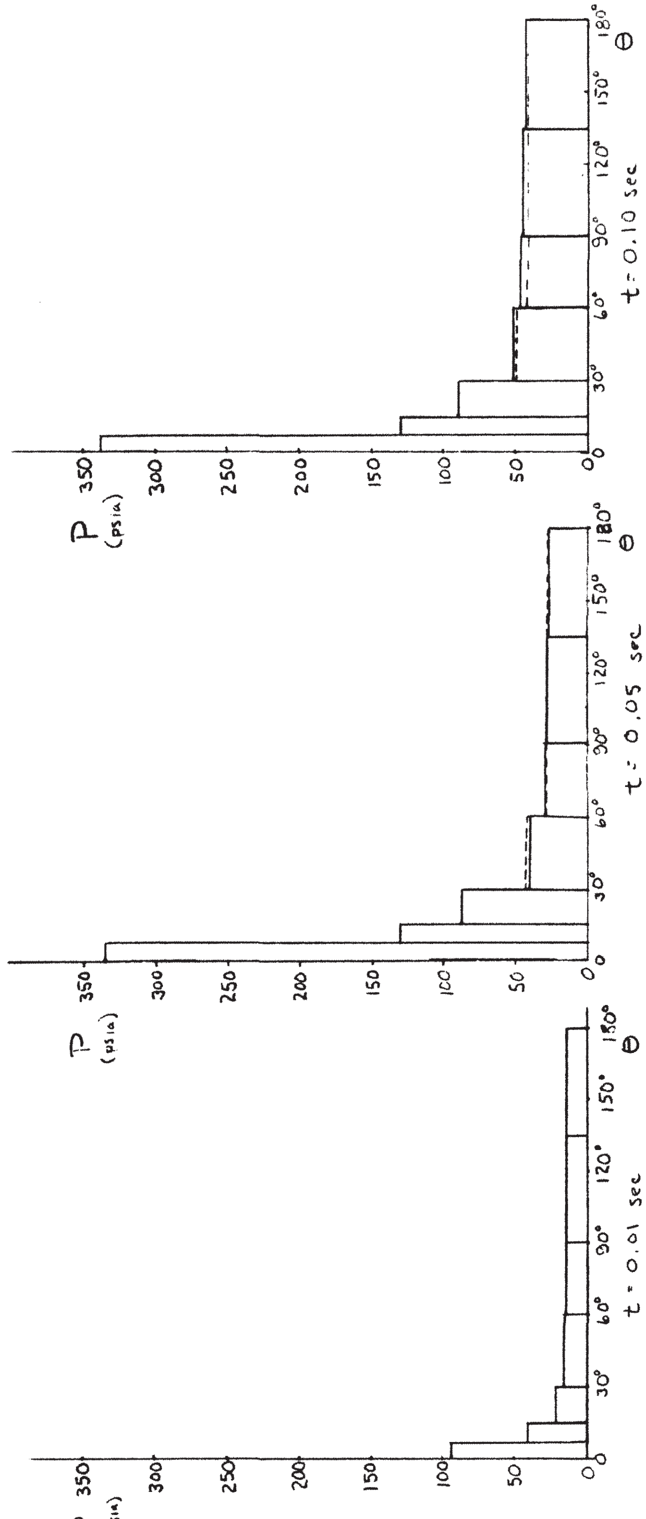

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FIGURE 6.2-41

AZIMUTHAL PRESSURE DISTRIBUTION
(AT ℄ RECIRCULATION OUTLET NOZZLE)
ORIGINAL DATA AND CASE A

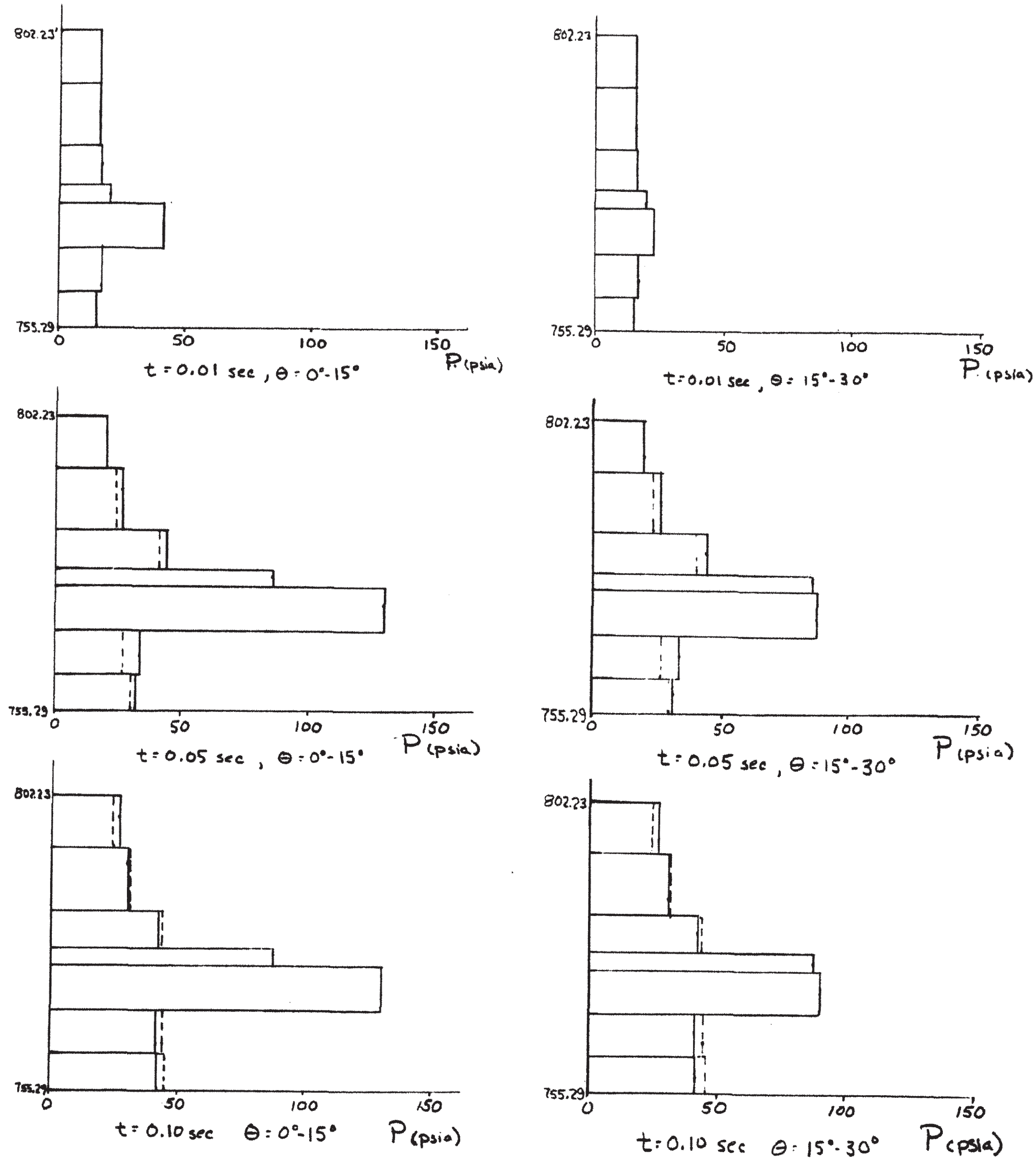

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FIGURE 6.2-42

AXIAL PRESSURE DISTRIBUTION
ORIGINAL DATA AND CASE A

REV. 0 - APRIL 1984

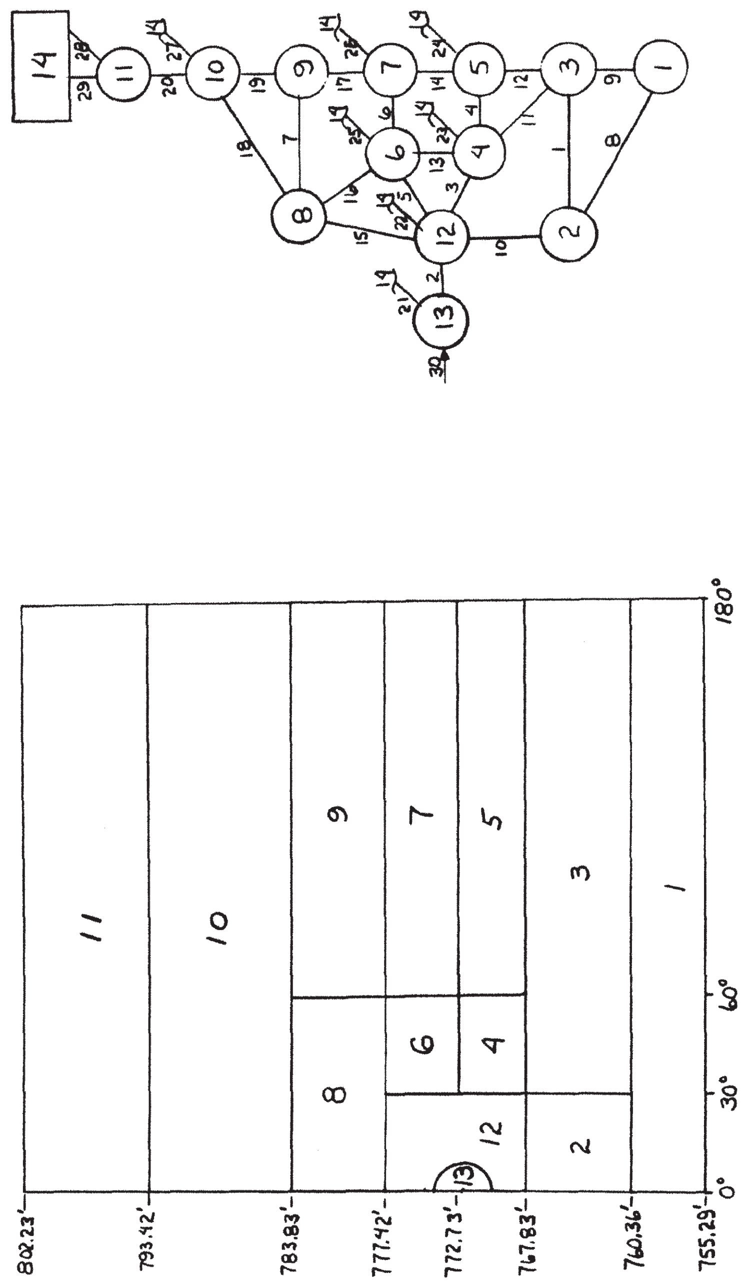

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FIGURE 6.2-43

SIMPLIFIED NODALIZATION (CASE B)

REV. 0 - APRIL 1984

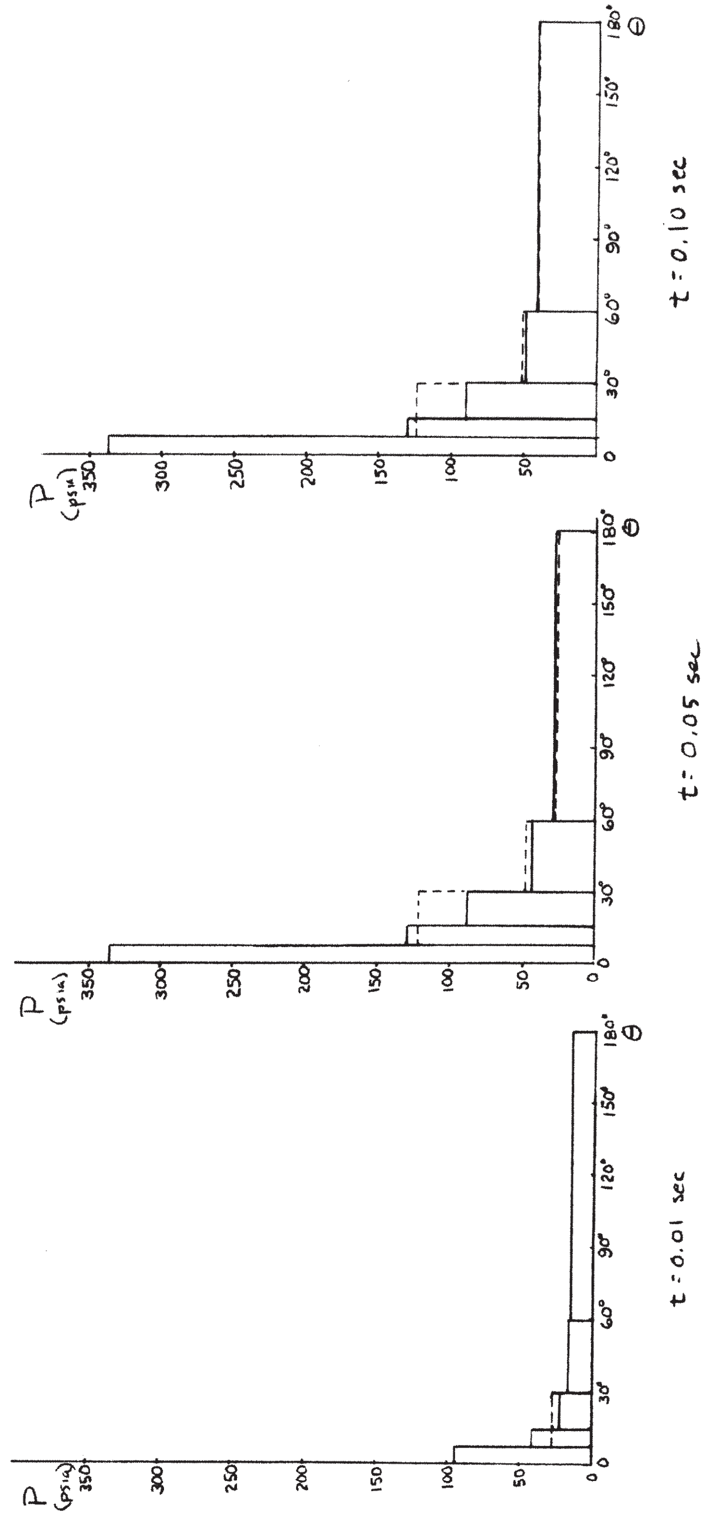

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FIGURE 6.2-44

AZIMUTHAL PRESSURE DISTRIBUTION
(AT ℄ RECIRCULATION OUTLET NOZZLE)
CASE A AND CASE B

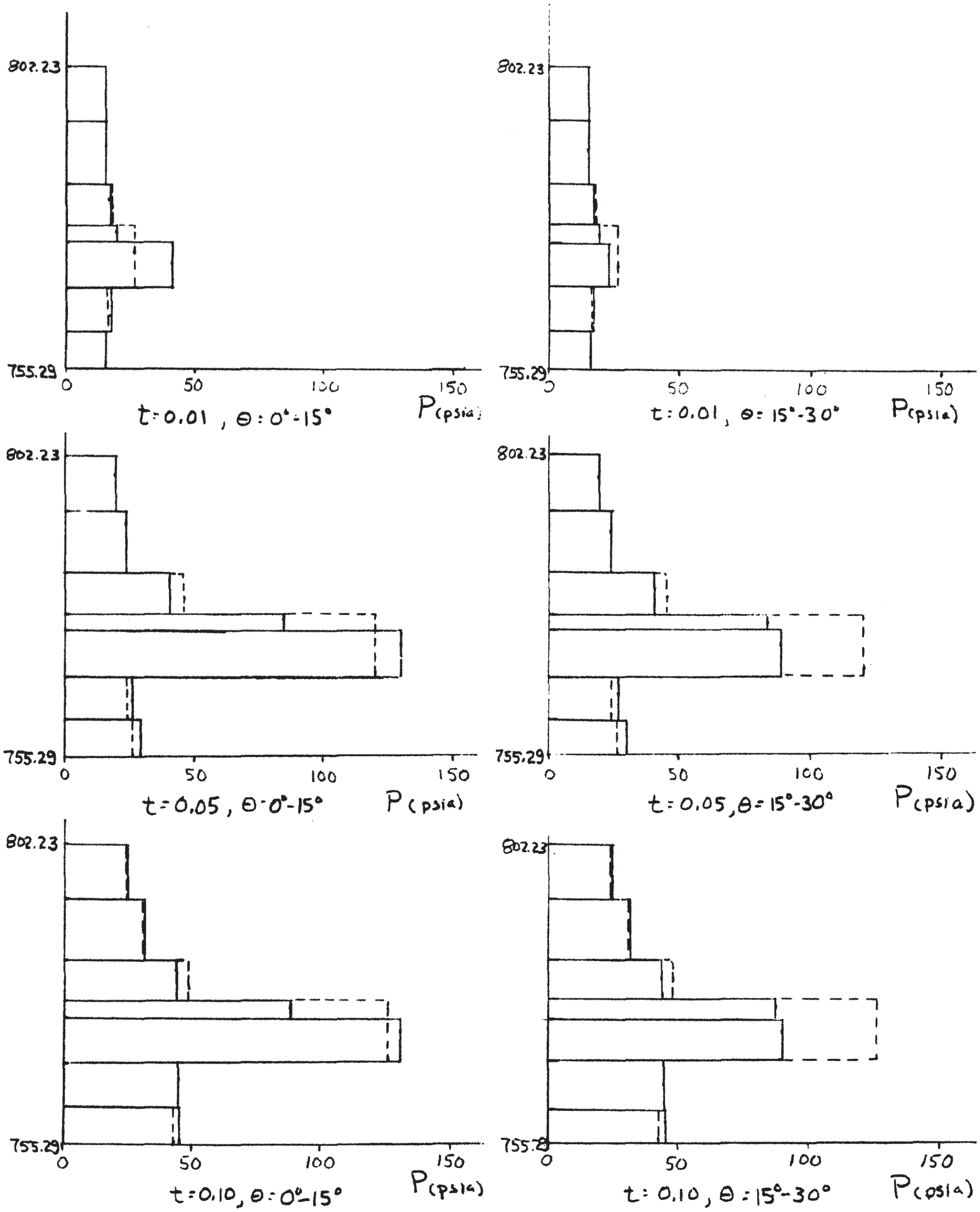

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FIGURE 6.2-45

AXIAL PRESSURE DISTRIBUTION CASE A AND CASE B

REV. 0 - APRIL 1984

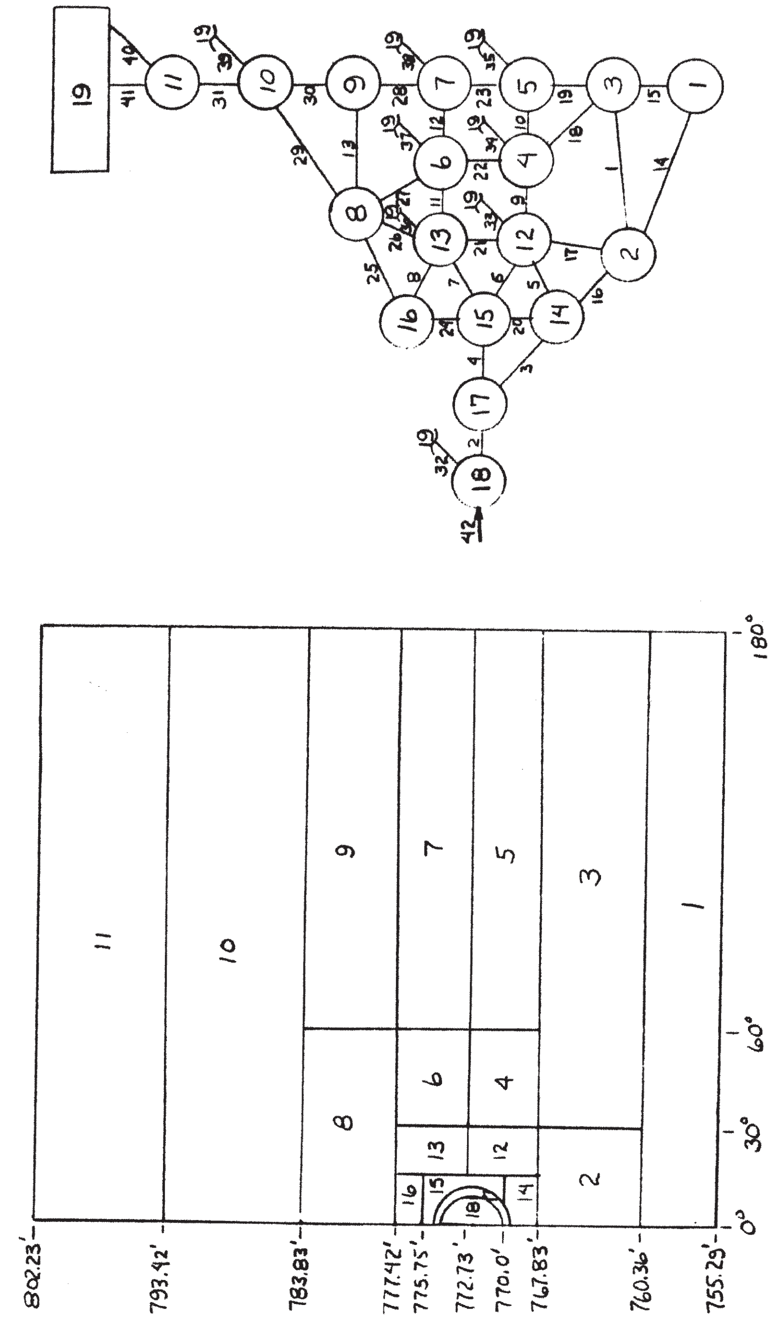

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FIGURE 6.2-46

COMPLEX NODALIZATION (CASE C)

REV. 0 - APRIL 1984

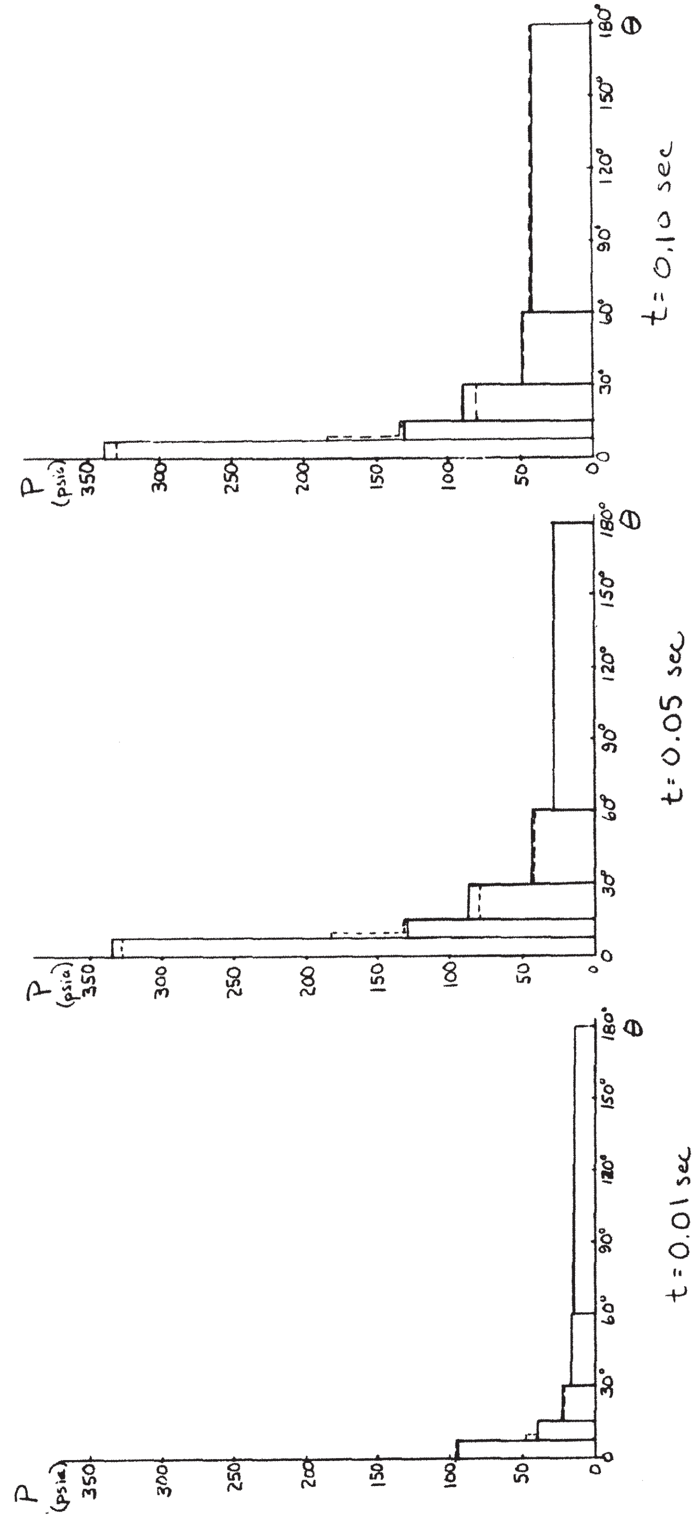

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FIGURE 6.2-47

AZIMUTHAL PRESSURE DISTRIBUTION
(AT ℄ RECIRCULATION OUTLET NOZZLE)
CASE A AND CASE C
REV. 0 - APRIL 1984

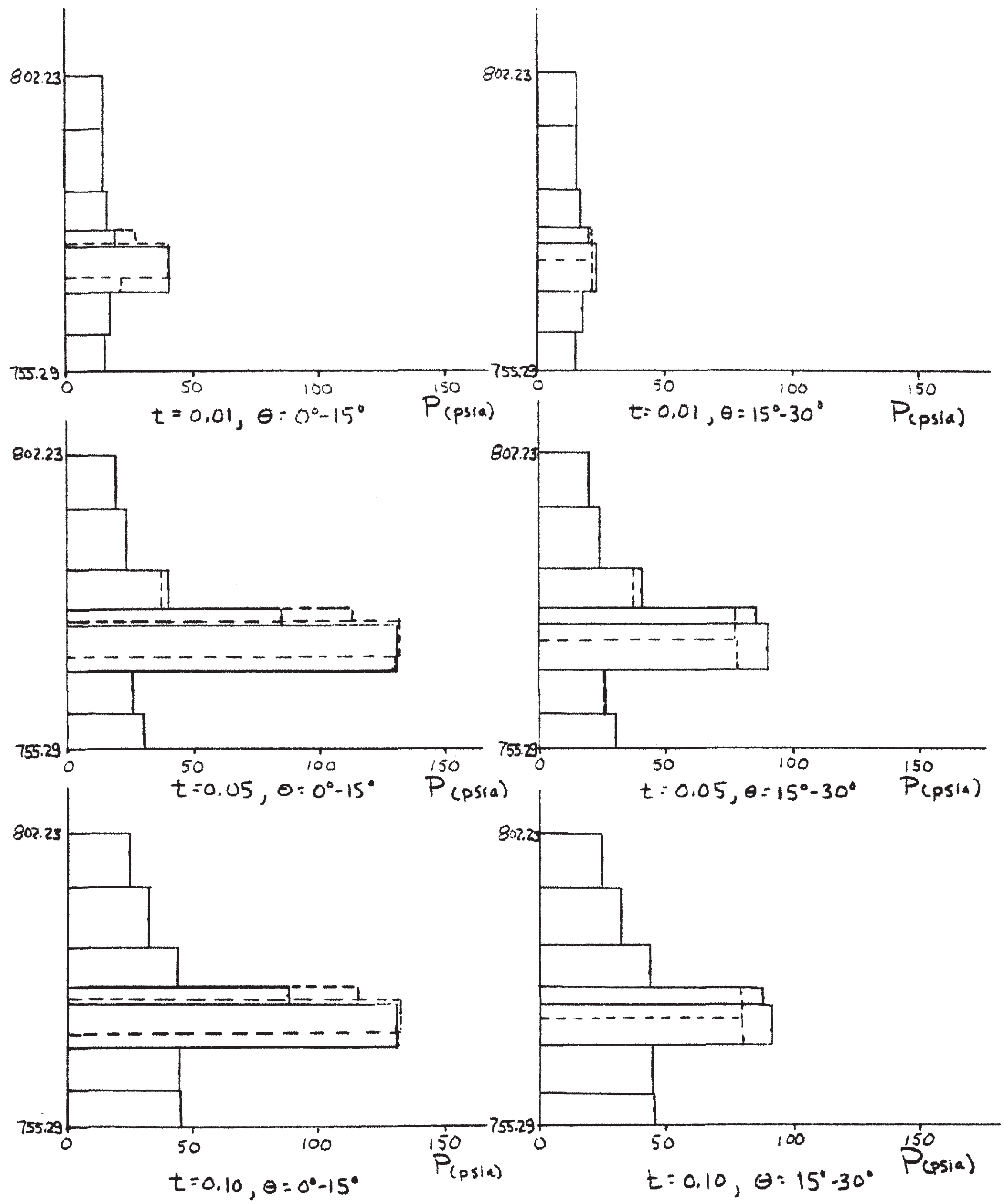

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FIGURE 6.2-48

AXIAL PRESSURE DISTRIBUTION
(CASE A AND CASE C)

REV. 0 - APRIL 1984

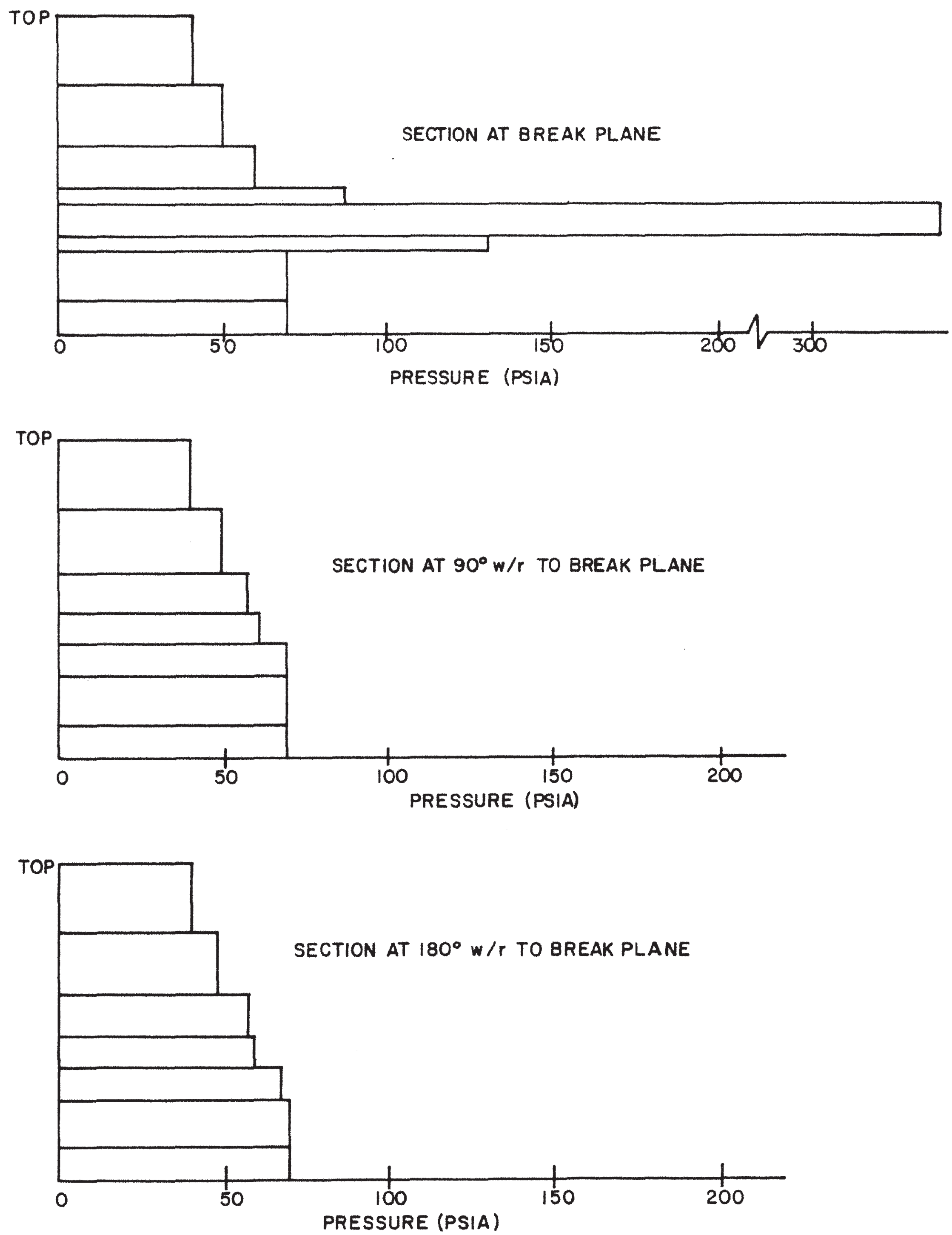

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FIGURE 6.2-49

AXIAL PRESSURE DISTRIBUTION AT
t = 0.500 SECONDS

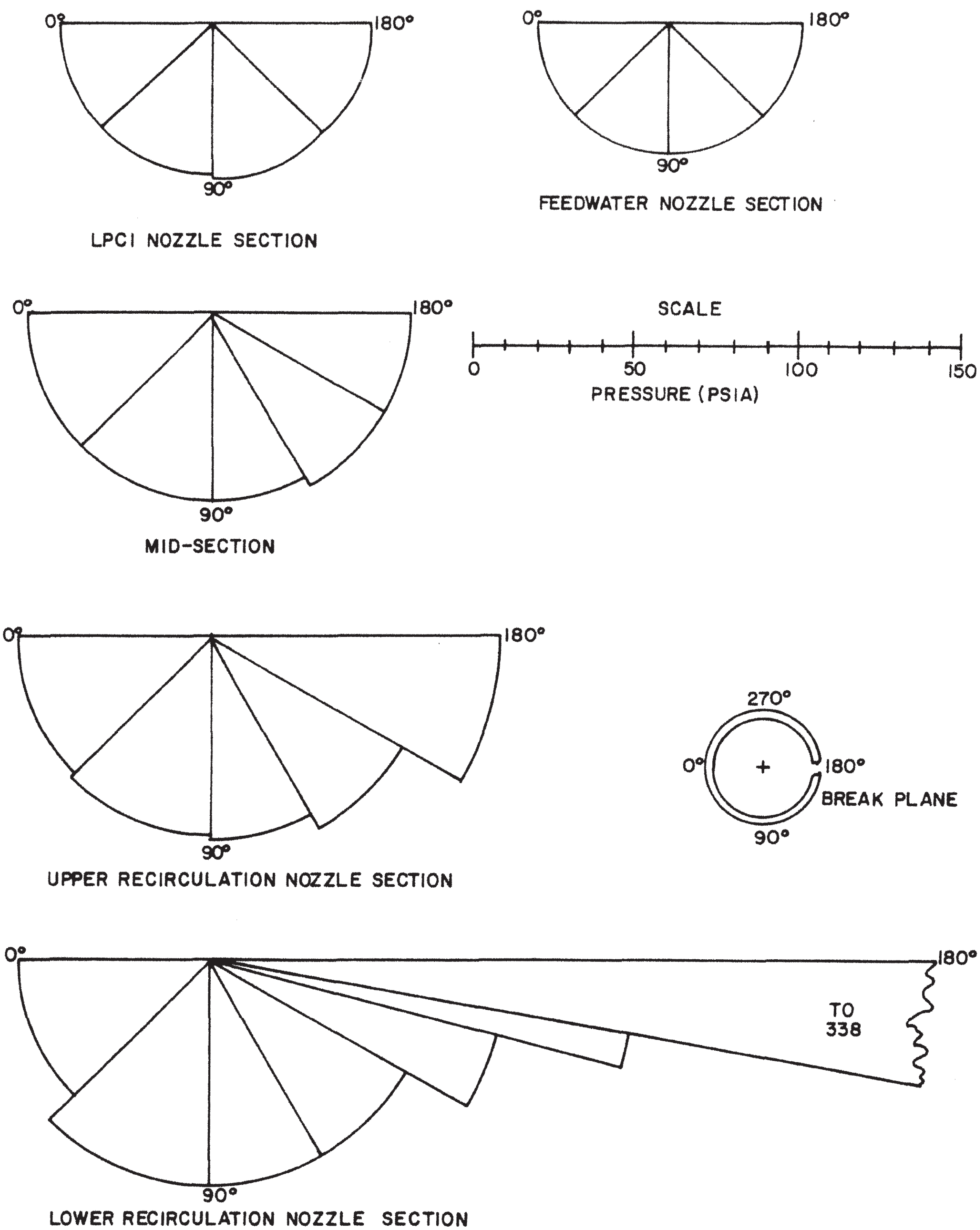

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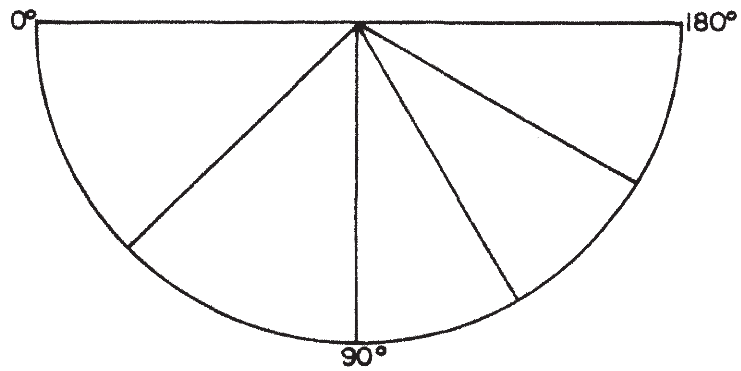
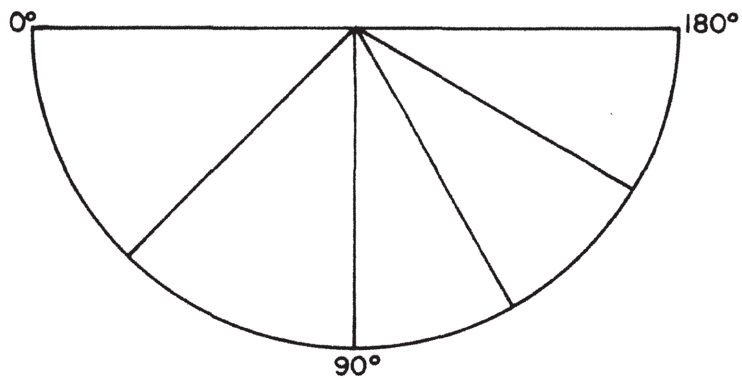
FIGURE 6.2-50

CIRCUMFERENTIAL PRESSURE DISTRIBUTION AT
t = 0.500 SECONDS
(SHEET 1 of 2)

REV. 0 - APRIL 1984



UPPER REACTOR SKIRT SECTION



LOWER REACTOR SKIRT SECTION

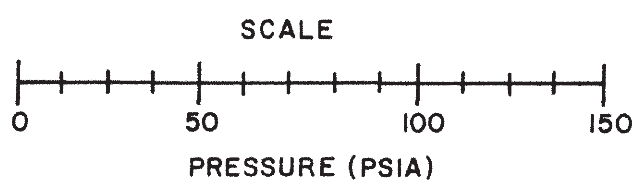


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FIGURE 6.2-50

CIRCUMFERENTIAL PRESSURE DISTRIBUTION AT
t = 0.500 SECONDS

(SHEET 2 of 2)

REV. 0 - APRIL 1984

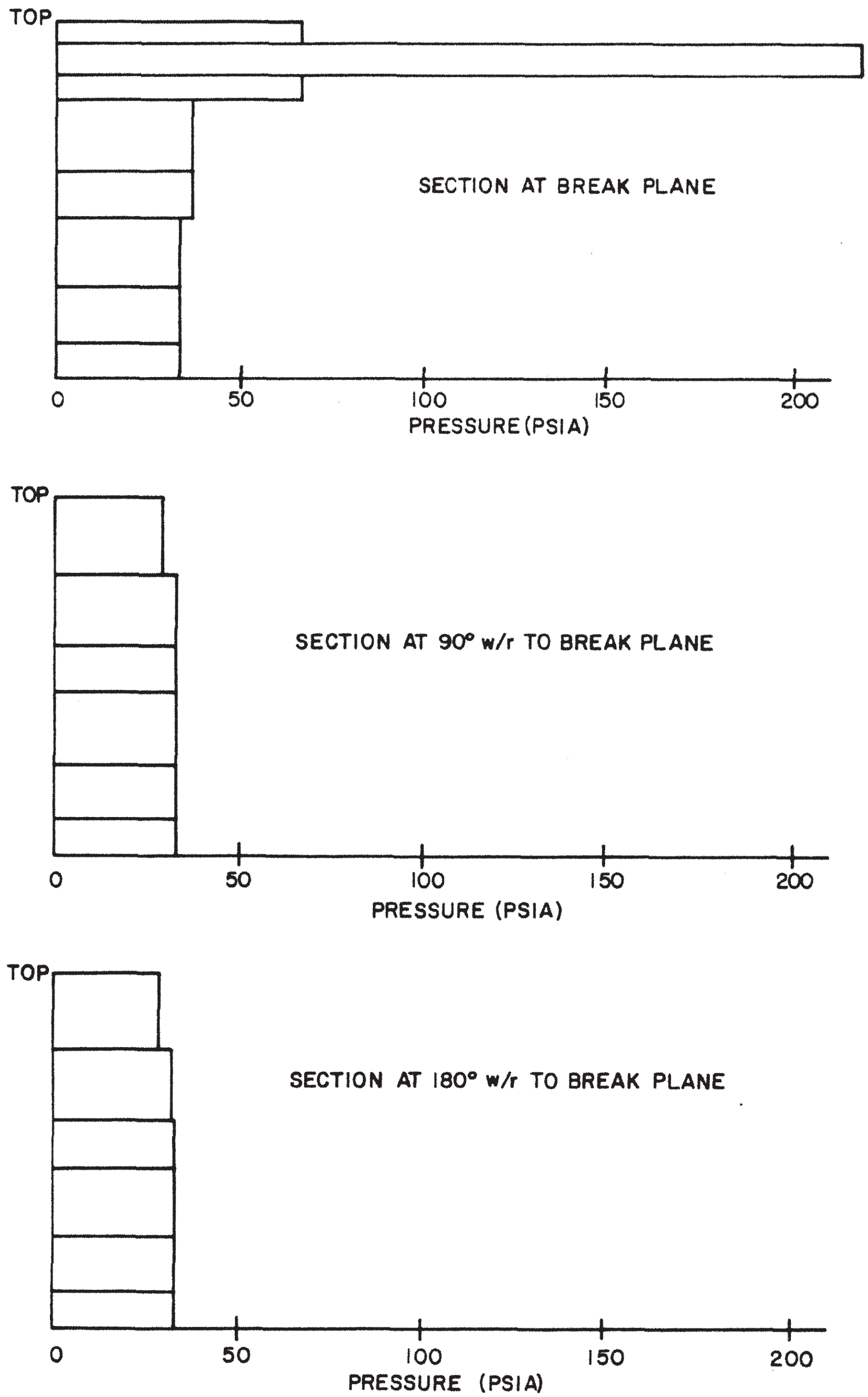

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FIGURE 6.2-51

AXIAL PRESSURE DISTRIBUTION AT
t = 0.500 SECONDS (CASE C)

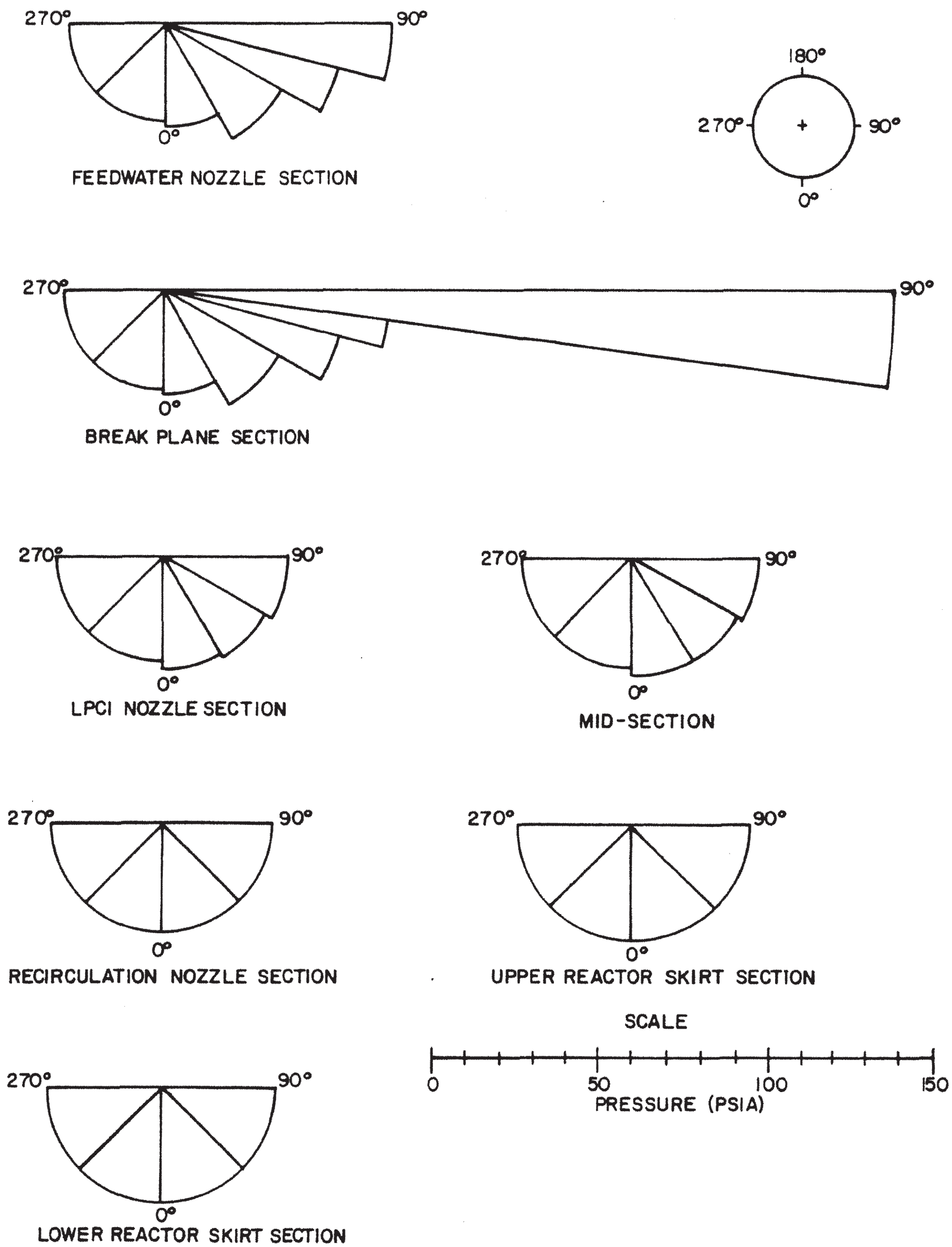

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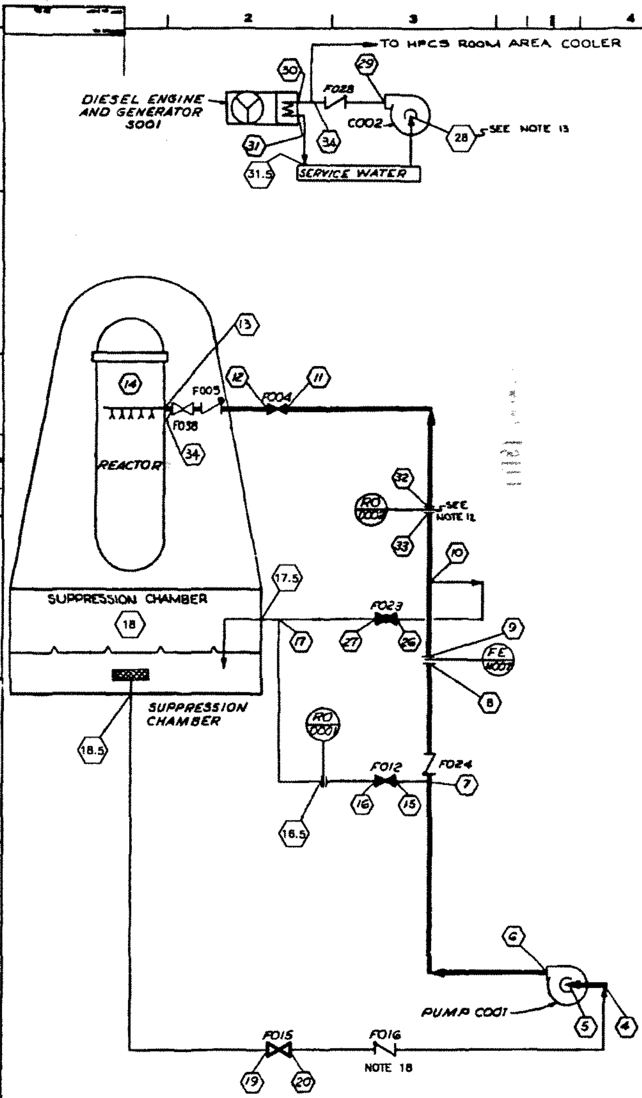
FIGURE 6.2-52

CIRCUMFERENTIAL PRESSURE DISTRIBUTION AT
t = 0.500 SECONDS (CASE C)



PRIMARY MODES

MODE B ACCIDENT, REACTOR AT HIGH PRESSURE SUCTION FROM SUPPRESSION POOL

POSITION ◯	4	5	6	7	8	9	10	11	12	13	14	19	20	18
FLOW GPM	1550	←							→	1550	N/A	1550	1550	N/A
PRESS PSIA											1143			14.7
TEMP °F	120/40	←							→	120/40	AMB	120/40	120/40	AMB
MAX PRESS DROP FEET		← 2787 →												

MODE C ACCIDENT SYSTEM INJECTION AT RATED CORE SPRAY SUCTION FROM SUPPRESSION POOL

POSITION ◯	4	5	6	7	8	9	10	11	12	13	14	19	20	18	28	29	30	31	34
FLOW GPM	6200	←							→	6200	N/A	6200	6200	N/A	1000	1000	800	800	800
PRESS PSIA											215			14.7	SEE NOTE 13				
TEMP °F	170/40	←							→	170/40	AMB	170/40	170/40	AMB	100	100	100	130	100
MAX PRESS DROP FEET		← 845 →			← 21 →			← 2.2 →		← 198 →		← 0.30 →			← 150 →		← 15 →		

MODE D ACCIDENT SYSTEM INJECTION AT CORE FLOOD SUCTION FROM SUPPRESSION POOL

POSITION ◯	4	5	6	7	8	9	10	11	12	13	14	19	20	18
FLOW GPM	6856	←							→	6856	N/A	6856	6856	N/A
PRESS PSIA											41			14.7
TEMP °F	170/40	←							→	170/40	AMB	170/40	170/40	AMB
MAX PRESS DROP FEET		← 596 →			← 24 →			← 2.5 →		← 251 →				

MODE F ACCIDENT SYSTEM OPERATING AT RUNOUT SUCTION FROM SUPPRESSION POOL

POSITION ◯	4	5	6	7	8	9	10	11	12	13	14	19	20	18
FLOW GPM SEE NOTE 9	7175	←							→	7175	N/A	7175	7175	N/A
PRESS PSIA		SEE NOTE 7									14.7			14.7
TEMP °F	212/40	←							→	212/40	AMB	212/40	212/40	AMB
MAX PRESS DROP FEET		← 450 →										← 0.48 →		

MODE G SYSTEM TEST SUCTION FROM SUPPRESSION POOL

POSITION ◯	4	5	6	7	8	9	10	21	26	27	17	19	20	18
FLOW GPM SEE NOTE 9	7175	←										→	7175	N/A
PRESS PSIA														14.7
TEMP °F	110/40	←										→	120/40	AMB
MAX PRESS DROP FEET		← 450 →							← 59 → SEE NOTE 10					

MODE S SYSTEM ON STANDBY DUTY

POSITION ◯		4	5	6	7	8	9	10	11	12	13	14	18	19	20
FLOW GPM		0	←							→	O	N/A	N/A	0	0
PRESS PSIA												SEE REF	14.7		
TEMP °F			←				→	110/40	AMB	←	→	AMB	AMB	170/40	170/40
MAX PRESS DROP FEET			← 0 →												

TABLE I
VALVE POSITION TABLE

VALVE	F001	F004	F010	F011	F012	F015	F023	F038
MODE B	C	O	C	C	C	O	C	O
MODE C	C	O	C	C	C	O	C	O
MODE D	C	O	C	C	C	O	C	O
MODE F	C	O	C	C	C	O	C	O
MODE G	C	O	C	C	C	O	O	O
MODE J	C	C	C	C	O	O	C	O
MODE S	C	C	C	C	C	O	C	O

O VALVE OPEN
C VALVE CLOSED

MODE J PUMP OPERATING ON BYPASS, SUCTION FROM SUPPRESSION POOL

POSITION ◯		4	5	6	7	15	16	17	18	19	20
FLOW GPM		1000	←					1000	N/A	1000	1000
PRESS PSIA									47		
TEMP °F		178/48	←					178/48	AMB	178/48	178/48
MAX PRESS DROP FEET			← 3000 →			← 2.0 →					

TABLE II
LIMITING LINE LOSS

MODE	FLOW PATH	COMMENT
F	18.5-19-20-4-5	SEE NOTE 7
C	6-7-8-9-10-11-12-13	SEE NOTE 5
J	7-15-16-16.5-17	
G	21-26-27-17-17.5	

NOTES:

1. ALL EMPTY PRESSURE DATA BLANKS CAN BE FILLED IN BY OTHERS (BASED ON ACTUAL ARRANGEMENT) OR EQUIV. HYDRAULIC DATA SUBMITTED TO APED FOR REVIEW NO DATES THE DATA IS NOT SIGNIFICANT.
2. MAX/MIN INDICATES MAXIMUM & MINIMUM VALUE OF PARAMETER FOR THE MODE SPECIFIED.
3. ELEVATIONS ARE NOT INCLUDED IN ΔP VALUES GIVEN. ELEVATIONS SHALL BE INCLUDED WHEN DETERMINING FINAL VALUES FOR THE EMPTY PRESSURE BLANKS.
4. THE PUMP MAXIMUM SHUTOFF HEAD WILL NOT EXCEED 3450 FT.
5. IN MODE E WITH A PUMP TDH OF 845 FT. AND A VESSEL PRESSURE OF 215 PSIA THE FLOW MUST BE EQUAL TO OR GREATER THAN 5200 GPM.
6. THE PUMP TDH GIVEN FOR MODE E BASED ON A MAXIMUM CONTAINMENT PRESSURE OF 45 PSIG. BWRSD BE ADVISED IF THE CONTAINMENT DESIGN IS BASED ON HIGHER PRESSURE AND THE IMPACT ON THE HIGH PRESSURE CORE SPRAY SYSTEM EVALUATED.
7. IN MODE F, THE NET POSITIVE SUCTION HEAD (NPSH) AVAILABLE AT THE CENTERLINE OF THE PUMP SUCTION NOZZLE MUST EQUAL OR EXCEED 16.0 FT.
8. DELETED.
9. THE FLOW SPECIFIED FOR MODES F AND G IS APPROXIMATE AND MUST BE DETERMINED BASED ON FINAL SYSTEM DESIGN. THE FLOW GIVEN FOR THE MAXIMUM ALLOWABLE.
10. THE ΔP GIVEN FOR THE VALVES IN MODE G IS THE MINIMUM POSSIBLE AND MAY BE INCREASED BY OTHERS (THROTTLING) TO ACCOMMODATE PIPING ARRANGEMENT.
11. DELETED.
12. THE ΔP BETWEEN LOCATION 32 AND 33 WILL BE DETERMINED IN PRE-OPERATIONAL TEST. THE ΔP WILL BE ADJUSTED TO MEET THE FLOW REQUIREMENTS OF MODE C, D AND/OR E, AND TO LIMIT MAXIMUM RUNOUT FLOW TO 7175 GPM.
13. THE MINIMUM AVAILABLE NPSH TO THE DIESEL SERVICE WATER PUMP OCCURS IN MODE C AND MUST BE 22 FEET OR GREATER.
14. ΔP VALUES FOR EQUIPMENT WITHIN GE-BWRSD SCOPE ARE AS NOTED.
15. TABLE 1 INDICATES VALVE POSITION DURING VARIOUS OPERATING MODES.
16. PIPING SYSTEM DESIGN PRESSURE AND TEMPERATURE AND THE ESTIMATED LINE SIZES ARE FOR INFORMATION ONLY. ACTUAL DESIGN TEMPERATURE AND PRESSURE AND LINE SIZES AS DETERMINED BY OTHERS SHALL MET THE PROCESS DIAGRAM HYDRAULIC REQUIREMENTS.
17. IN MODE D AND WITH A VESSEL PRESSURE OF 14.7 PSIA, THE FLOW SHALL NOT EXCEED 7175 GPM.
18. THE INTERNALS OF CHECK VALVE 1E22-F016 (2E22-F016) HAVE BEEN REMOVED PER DCP 9500324 (9500325).

SUPPLEMENTAL DOCUMENTS UNDER THE FOLLOWING IDENTITIES ARE TO BE USED IN CONJUNCTION WITH THIS DRAWING.

	REFERENCE DESIGNATOR
1. HIGH PRESSURE CORE SPRAY P&ID	E22-1010
2. NUCLEAR BOILER SYSTEM PROC. DIAGRAM	B21-1020
3. HIGH PRESSURE CORE SPRAY DESIGN SPEC	E22-4010

MISCELLANEOUS INFORMATION
(SEE NOTE 16)

LOCATION	1.5 3	4 16.5 6	5 6 8 11 34	15 7	16.5 17 10	27 17.5 21	25	29 30 31 31.5
DESIGN TEMP (°F)	148 [illegible]	[illegible]	212 * SEE REF 2	212	212	212 140		100 150
DESIGN PRESS (PSIG)	100	100	1325 * SEE REF 2	1325 100	1325 100	1325 100		100 100
EST MA ED LINE SIZE (")	16	18	16 12	4	12	10		5 8
	CONDENSATE STG SUCTION LINE	SUPPRESSION POOL SUCTION	MAIN HPCS LINE TO REACTOR	BY PASS LINE	SUPPRESSION POOL TEST LINE	CONDENSATE STORAGE TK RETURN LINE		DIESEL COOLING WATER LINE

* DUAL DESIGN CONDITION 250 PSIG @ 575 °F OR 1150 PSIG @ 140 °F.

731E932AA

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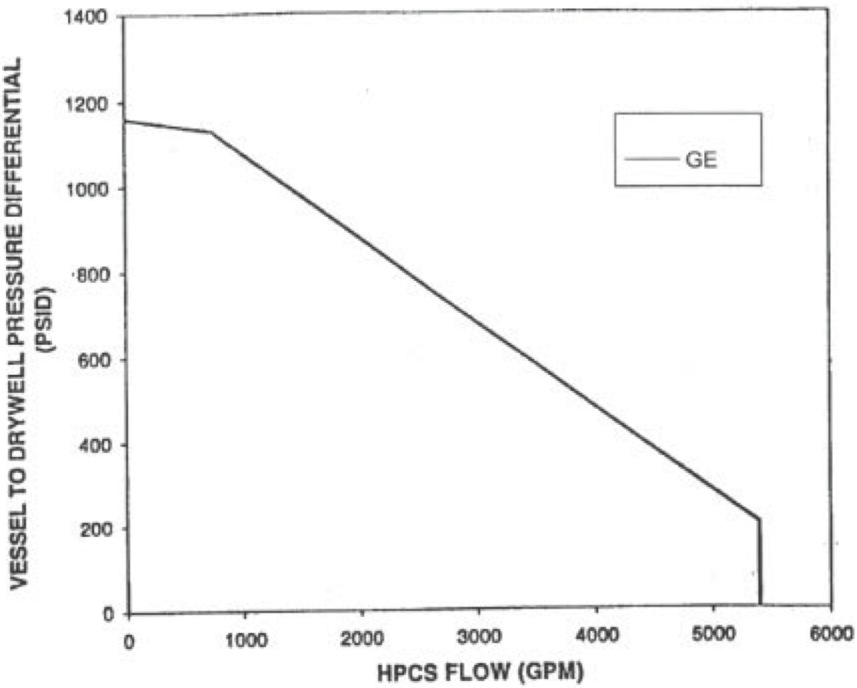
FIGURE 6.3-1

HPCS SYSTEM PROCESS DIAGRAM

REV. 13



Source: References 5

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FIGURE 6.3-2

VESSEL PRESSURE VS. HPCS FLOW
ASSUMED IN GE LOCA ANALYSES

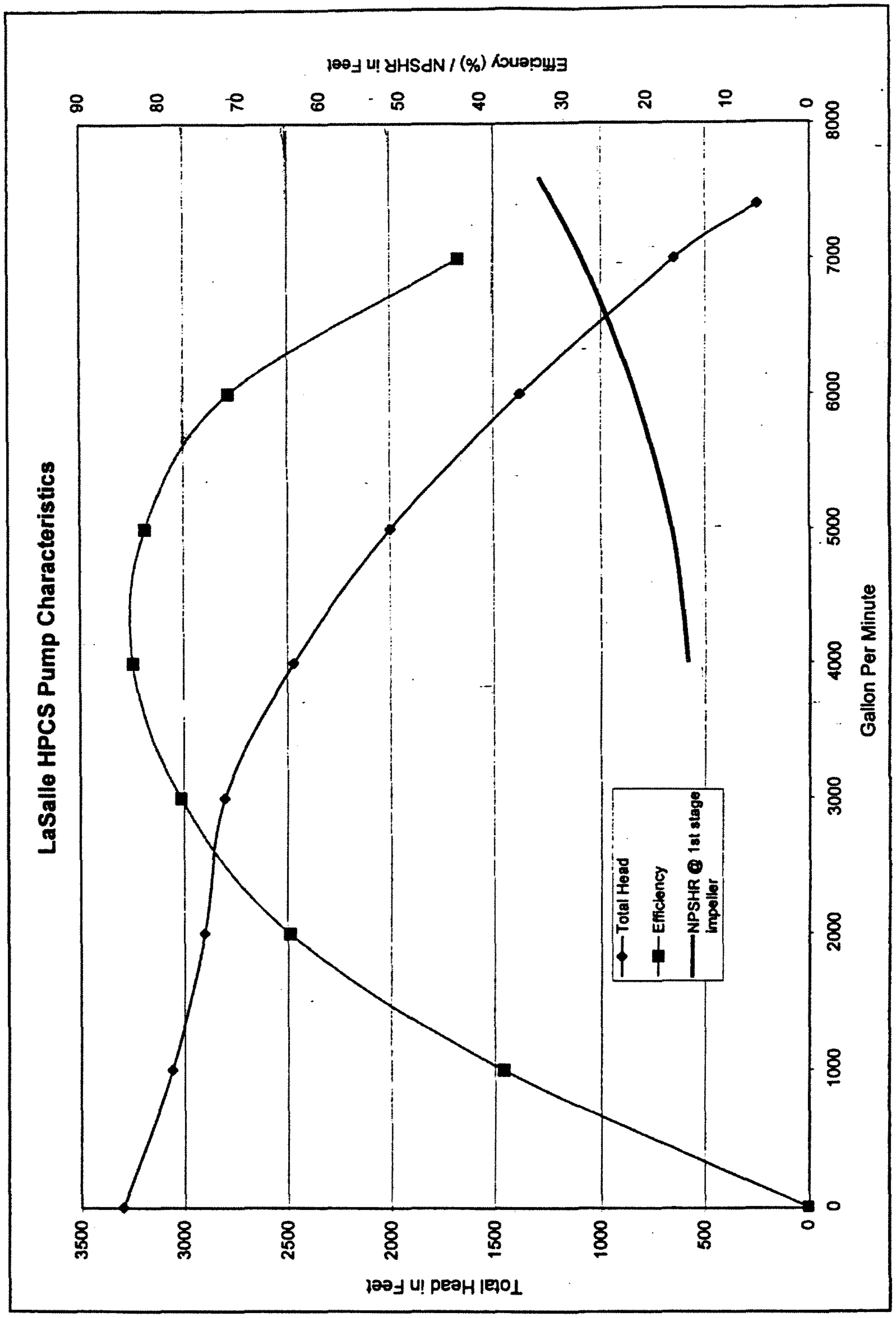

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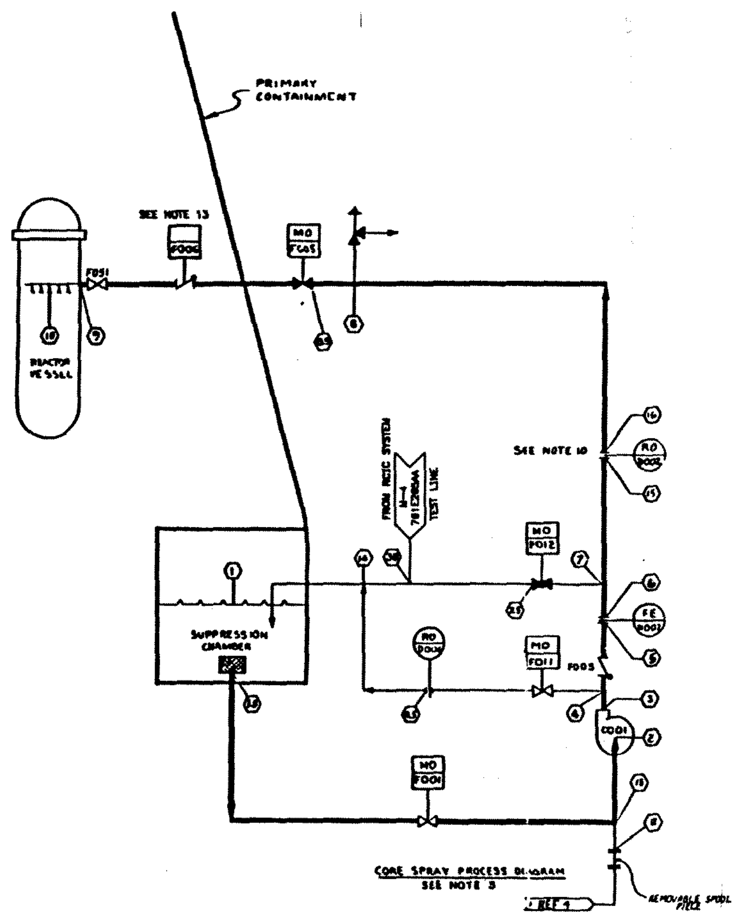
FIGURE 6.3-3 HPCS PUMP CHARACTERISTICS



FCF-259A2BTAA (E21 1020)

MODE A SYSTEM TEST, SUCTION FROM SUPPRESSION POOL

LOCATION ○	1	2	3	4	5	6	7	8	9	10	11	12	13	14
FLOW - GPM	N/A	[illegible] ←				→	[illegible]	0	0	0	0	0	0	[illegible]
PRESS - PSIA	14.7	X	X	X	X	X		X	X	X	X	X	X	
TEMP - °F	120/40	←				→	120/40	AMB	←			→	AMB	[illegible]/40
MAX PRESS DIFFERENCE - FT		+500+												

MODE B SYSTEM TEST, SUCTION FROM RESIDUAL HEAT REMOVAL SYSTEM

LOCATION ○	1	2	3	4	5	6	7	8	9	10	11	12	13	14
FLOW - GPM	N/A	8200 ←							→	8200	N/A	8200	8200	
PRESS - PSIA	14.7		X	X	X	X	X	X	X	14.7	14.7			X
TEMP - °F	N/A	125/40 ←							→	125/40	N/A	125/40	125/40	AMB
MAX PRESS DIFFERENCE - FT		+650+												

MODE C PUMP OPERATING ON BYPASS, SUCTION FROM SUPPRESSION POOL

LOCATION ○	1	2	3	4	5	6	7	8	9	10	11	12	13	14
FLOW - GPM	N/A	635	635	635	0	0	0	0	0	0	0	0	0	635
PRESS - PSIA	14.7	X	X		X	X	X	X	X	X	X	X	X	
TEMP - °F	212/40	←	→	212/40	AMB	←						→	AMB	212/40
MAX PRESS DIFFERENCE - FT		1185/1020												

MODE D ACCIDENT, SYSTEM INJECTION AT RATED CORE SPRAY (126 PSID)

LOCATION ○	1	2	3	4	5	6	7	8	9	10
FLOW - GPM	N/A	6350 ←							→	6350
PRESS - PSIA	14.7									142.7
TEMP - °F	170	←							→	170
MAX PRESS DIFFERENCE - FT		+674+			+17+				+178+	

MODE E ACCIDENT, SYSTEM INJECTION AT RATED CORE FLOOD

LOCATION ○	1	2	3	4	5	6	7	8	9	10
FLOW - GPM	N/A	7214 ←							→	7214
PRESS - PSIA	14.7	X	X	X	X	X	X	X	X	40.7
TEMP - °F	212/40	←							→	212/40
MAX PRESS DIFFERENCE - FT		+515+								

MODE F ACCIDENT, SYSTEM OPERATING AT RUNOUT

LOCATION ○	1	2	3	4	5	6	7	8	9	10
FLOW - GPM	N/A	7000 ←							→	[illegible]
PRESS - PSIA	14.7		X	X	X	X	X	X	X	14.2
TEMP - °F	212	←							→	212
MAX PRESS DIFFERENCE - FT		+500+								

MODE S SYSTEM ON STANDBY DUTY

LOCATION ○	1	2	3	4	5	6	7	8	9	10	11	12	13	14
FLOW - GPM	N/A	0	←										→	0
PRESS - PSIA	14.7		X	X	X	X	X	X	X	30	14.7	X	X	X
TEMP - °F	120/40	←				→	120/40	AMB	AMB	[illegible]	120/40	←	→	120/40
MAX PRESS DIFFERENCE - FT		+0+												

NOTES.

1. ALL EMPTY PRESSURE DATA BLANKS CAN BE FILLED IN BY OTHERS BASED ON ACTUAL ARRANGEMENTS OR EQUIVALENT HYDRAULIC DATA SUBMITTED TO [illegible] FOR REVIEW. [illegible] INDICATES THE DATA IS NOT SIGNIFICANT.
2. [illegible] INDICATES MAXIMUM & MINIMUM VALUE OF PARAMETER FOR THE MODE SPECIFIED.
3. ELEVATIONS ARE NOT INCLUDED IN THE ΔP VALUES GIVEN. ELEVATIONS SHALL BE INCLUDED WHEN DETERMINING FINAL VALUES FOR THE EMPTY DATA BLANKS.
4.
5. IN MODE F, THE NET POSITIVE SUCTION HEAD (NPSH) AVAILABLE AT A REFERENCE LOCATION 3 FEET ABOVE THE PUMP MOUNTING FLANGE MUST EQUAL OR EXCEED [illegible] FT. THE NPSH AVAILABLE AT THE PUMP SUCTION NOZZLE MUST EQUAL THIS VALUE PLUS THE DIFFERENCE IN ELEVATION BETWEEN THE REFERENCE LOCATION AND THE CENTERLINE OF THE PUMP SUCTION NOZZLE.
6. IN MODE B, THE NPSH AVAILABLE MUST EQUAL THE VALUE SPECIFIED IN NOTE 5 PLUS [illegible] FT.
7. [illegible] GPM IS INCLUDED IN THE FLOW GIVEN FOR MODE D TO COMPENSATE FOR LEAKAGE IN THE REACTOR INTERNALS. ([illegible] GPM IN MODE E).
8. IN MODE D, [illegible] PSID IS THE DIFFERENTIAL PRESSURE BETWEEN THE REACTOR VESSEL AND THE SUPPRESSION POOL.
9. THE FLOW SPECIFIED FOR MODE F IS THE MAXIMUM ALLOWABLE.
10. THE ΔP BETWEEN LOCATION [illegible] AND [illegible] WILL BE DETERMINED IN PRE-OPERATIONAL TEST. THE ΔP WILL BE ADJUSTED TO MEET THE FLOW REQUIREMENTS OF MODE D, E OR F.
11. THE PIPE ATTACHED TO THE PUMP SUCTION NOZZLE SHALL BE STRAIGHT (E.G. NO TEES OR ELBOWS ETC.) FOR AT LEAST [illegible] PIPE DIAMETER UPSTREAM OF THE NOZZLE TO INSURE PROPER VELOCITY PROFILE.
12. PIPING SYSTEM DESIGN PRESSURE AND TEMPERATURE AND THE ESTIMATED LINE SIZES ARE FOR INFORMATION ONLY. ACTUAL DESIGN TEMPERATURE AND PRESSURE AND LINE SIZES AS DETERMINED BY OTHERS SHALL MEET THE PROCESS DIAGRAM HYDRAULIC REQUIREMENTS.
13. FREE SWINGING DISC. THE ACTUATOR IS USED FOR LOCAL LEAK RATE TESTING ONLY.

REFERENCE DOCUMENTS:	REF. ITEM NO.
1. LPCS SYSTEM P&ID	E21-1010
2. LPCS SYS DESIGN SPEC	E21-4010
3. NUCLEAR BOILER SYS PROC DIAG	B21-1020
4. RESIDUAL HEAT REMOVAL SYS P&ID	E12-1010

SUPPORTING DOCUMENTS:	REF. ITEM NO.
1. PIPING & INSTRUMENT SYM	A42-1010

TABLE II
LIMITING LINE LOSS

MODE	FLOW PATH	COMMENTS
F	15-13-2	SEE NOTE 6
D OR E	3-4 5-6 15 16-6-6.5 9-10	
A	7-7.5-14	
C	4-4.5-14	
S	12-13	

VALVE POSITIONS

CONDITION	VALVE NO					
	F001		F005	F007	F012	F011
MODE A	O		C	O	P	C
MODE B	C		O	O	C	C
MODE C	O		C	O	C	O
MODE D	O		O	O	C	C
MODE E	O		O	O	C	C
MODE F	O	–	O	O	C	C
MODE S	O		C	O	C	O

P - PARTIALLY OPEN
C - FULLY CLOSED
O - FULLY OPEN

MISCELLANEOUS INFORMATION - SEE NOTE 12

LOCATION ○	1.5	2	3 4 5 6 7 8	8.5	9	4 4.5	16	7 7.5	14	12 13
DESIGN TEMP (°F)	212	LPCS PUMP	← 400 →	SEE NOTE 3		← 212 →		400 | 212		← 212 →
DESIGN PRESS (PSIG)	100		← 800 →	SEE NOTE 3		550 | 100		550 | 100		← 100 →
ESTIMATED LINE SIZE (IN)	18"		← 16" →	12"		← 4" →		← 12 →		← 16" →
	MAIN CORE SPRAY LINE TO REACTOR					BYPASS LINE		TEST LINE		CONDENSATE LINE

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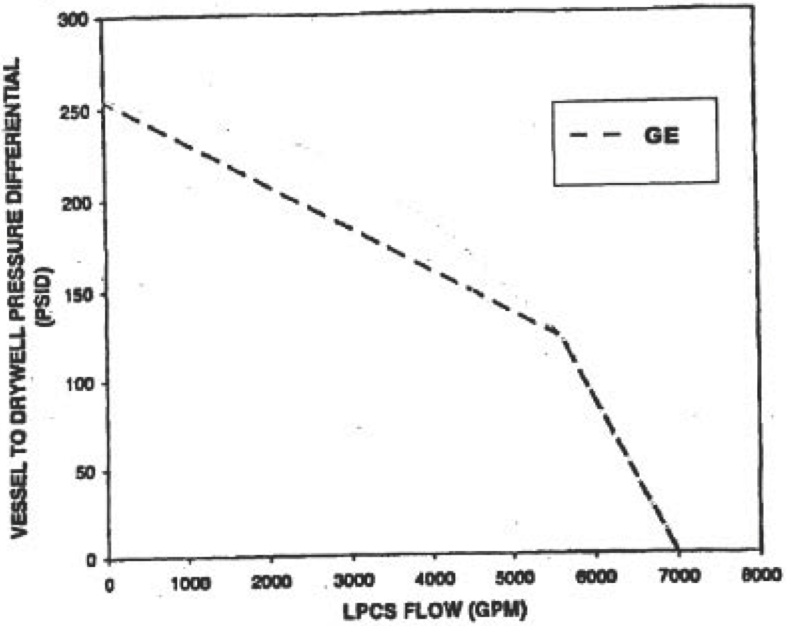
FIGURE 6.3-4

LPCS SYSTEM PROCESS DIAGRAM

REV. 13



Source: References 5

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FIGURE 6.3-5

VESSEL PRESSURE VS. LPCS FLOW
ASSUMED IN GE LOCA ANALYSES

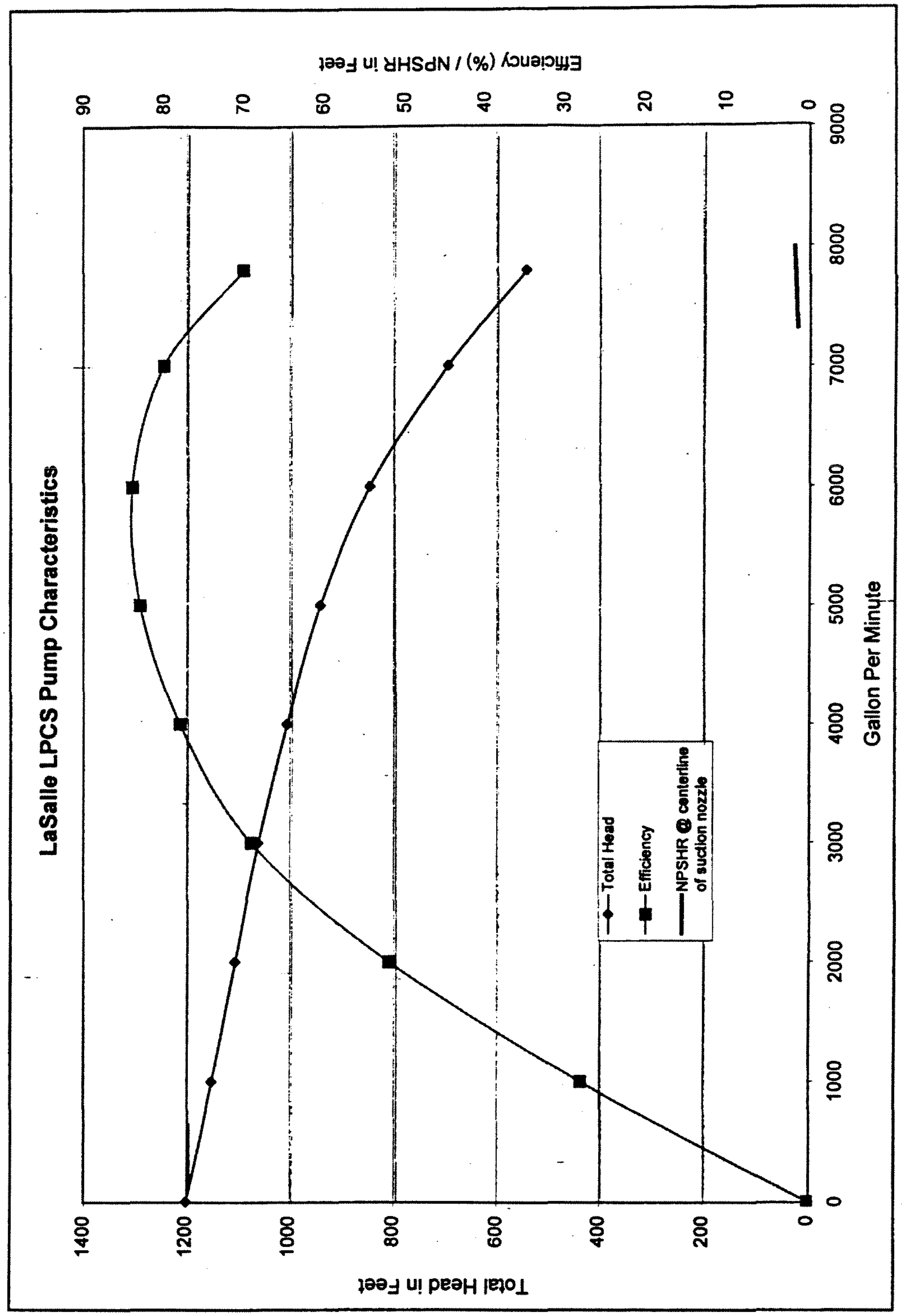

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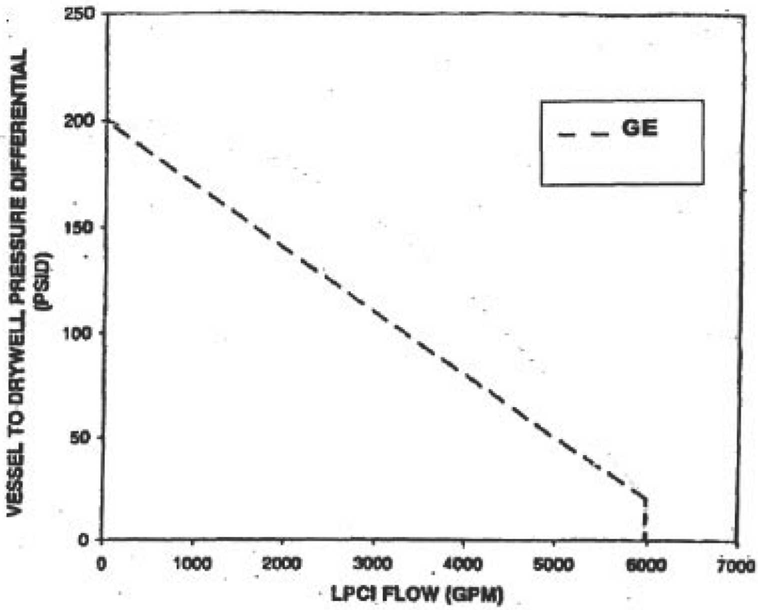
FIGURE 6.3-6

LPCS PUMP CHARACTERISTICS



Source: Reference 5

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FIGURE 6.3-7

VESSEL PRESSURE VS. LPCI FLOW
ASSUMED IN GE LOCA ANALYSES

FCF 238X184AA G1[illegible], E12-1020

MODE F

POSITION ○	1	2	3	4	5	6	49	7	8	9	10	13	24	1
FLOW-GPM	—	7450	←									→	7450	—
PRESS-PSIA	14.7	X	X	X	X	X	X	X	X	X	X			14.7
TEMP °F	—	120/40	←									→	120/40	—
MAX. PRESS DROP-FEET					TDH=267									

LOOP A & B TEST　LOOP C TEST

MODE G

Rx PRESS 110 PSIG

POSITION ○	29	25	55	26	4	5	6	43 A,B	24 A,B	1		1	2	3	4	5	6	42	45 A,B	24 A,B	1
FLOW-GPM	—	550	←						→ 550	—		—	550	←					→	550	—
PRESS-PSIA	125	X	X	X	X			X		14.7		14.7	X	X	X	X				X	14.7
TEMP °F	—	344	←						→ 344	—		—	125/40	←					→	125/40	—
MAX. PRESS DROP-FEET						TDH=[illegible] SEE NOTE 22										TDH=[illegible] SEE NOTE 22					

LOOP A & B　LOOP A　B&C LOOPS

MODE S

POSITION ○	1	2	3	4	5	6	18 A,B	19 A,B	20 A,B	9	10	46	11	48 A,B	21 A,B	50 A,B	27 A,B	51	33	52 A,B	53 A,B
FLOW-GPM	N/A	←																	→		N/A
PRESS-PSIA	14.7	X	X	X	X	X	X	X	X	X	X	X	1015	X	1015	X	1015	X	1015	X	X
TEMP °F	90/40	←									→	90/40	547	90/40	547	90/40	547	90/40	547	90/40	90/40
MAX PRESS DROP-FEET						0															

MODE S (CONT'D)

POSITION ○	54	56	55 A,B		34	35	36 A,B	37 A,B	12 A,B		30 A,B	38 A,B	40 A,B	41	42
FLOW-GPM	N/A	N/A	N/A		N/A	←		→	N/A		N/A	←		→	N/A
PRESS-PSIA	X	X	X		X	X	X	X	X		X	X	X	X	X
TEMP °F	AMB	AMB	AMB		AMB	AMB	AMB	90/40	90/40		90/40	90/40	AMB	AMB	AMB
MAX PRESS DROP-FEET															

MODE A-1

(SEE NOTE 14)

POSITION ○	1	2	3	4	5	6	7	8	9	10	16	11	29
FLOW-GPM	—	7450	←								→	7450	—
PRESS-PSIA	59.7/14.7	X	X	X	X								857/14.7
TEMP °F	—	170/40	←								→	170/40	—
MAX PRESS DROP FEET					TDH=267				ΔH=17			ΔH=40	

LOOP A & B　LOOP C

MODE A-2

(SEE NOTE 14)

POSITION ○	1	2	3	4	5	6	7	8	9	10	16	11	29
FLOW-GPM	—	8400	←								→	8400	—
PRESS-PSIA	14.7	X	X	X	X	X	X	X	X	X	X	X	14.7
TEMP °F	—	150/90	←								→	150/90	—
MAX PRESS DROP-FEET					TDH=235								

LOOP A & B　LOOP C

MODE B

POSITION ○	1	2_B	3_B	4_B	5_B	6_B	7_B	17_B	18_B	19_B	20_B	8_B	9_B	10_B	13_B	14	15_B	21_B	22	23_B	1		60_B	61_B
FLOW-GPM	—	8400	←													→	8400	7750	—	450	—		7400	7400
PRESS-PSIA	59.7/14.7					X	X	X	X	X	X	X	X	X	X	X			59.7/14.7		59.7/14.7		X	X
TEMP °F	—	212	←					→	212	170.4	←						→	170.4	—	170.4	—		140	142
MAX PRESS DROP-FEET					TDH=235																			

HEAT REMOVAL PER HX 155×10^6 SEE NOTE 16 (1 HX OPERATING) (SEE NOTE 21)

INDICATES FLOW PASSING THRU ABOVE POSITIONS (TYP)

SEE NOTE 7 (TYP)

MODE D

SEE NOTE 23　RX PRESS 110 PSIG

POSITION ○	29	25	26	5_A	6_A	7_A	18_A	19_A	8_A	9_A	10_A	14	30	31	66	33	29	47_A	16_A	27_A	28	29	$60_{A,B}$	$61_{A,B}$
FLOW-GPM	—	7450	7450	7450	7450	7450	7450	7450	7450	←		→	7450	870/0	870/0	870/0	—	[illegible]	[illegible]	[illegible]	[illegible]	—	7400	7400
PRESS-PSIA	125			X	X	X	X	X	X	X	X	X					125	X	X	X	X	125	X	X
TEMP °F	—	344	←				→	344	252.5	←					252.5	REF 3	—	252.5	252.5	252.5	252.5	—	90	182.1
MAX PRESS DROP-FEET					TDH=267								ΔH=17			ΔH=4								

HEAT REMOVAL PER HX 35.2×10^6 BTU/HR (1 HX OPERATING) (SEE NOTE 21)

HEAD SPRAY　SEE REF 3　VESSEL COOLANT　HEAD SPRAY (VESSEL COOLANT)

MODE E

RX PRESS 0 PSIG

POSITION ○	29	25	26	56	4	5	6	$18_{A,B}$	$19_{A,B}$	9	10	$16_{A,B}$	11	$21_{A,B}$	28	29	$60_{A,B}$	$61_{A,B}$
FLOW-GPM	—	[illegible]	5900	7450	←									→	7450		7400	7400
PRESS-PSIA	14.7	X	X	X	X	X	X	X	X	X	X	X	X		X	14.7	X	X
TEMP °F	—	120	[illegible]												[illegible]		90	131.3
MAX PRESS DROP-FEET																	ΔH=23	

DUTY PER HX 41.6×10^6 BTU/HR (2 HX'S OPERATING)

LOOP A & B TEST　LOOP C TEST

MODES

A-1. ACCIDENT WITH REFLOODING [illegible]

A-2. ACCIDENT WITH REFLOODING [illegible]

B. POST ACCIDENT CONTAINMENT SPRAY [illegible]

D. NORMAL SHUTDOWN AFTER FLOODING TO MAIN CONDENSER [illegible]

E. CONTINUATION OF NORMAL SHUTDOWN [illegible]

F. RHR SYSTEM TEST DURING PLANT OPERATION.

G. MINIMUM FLOW BYPASS MODE.

S. SYSTEM ON STANDBY DUTY

LEGEND

RX PRESS - - - - - REACTOR VESSEL PRESSURE

TDH - - - - - TOTAL DYNAMIC HEAD

SHO - - - - - SHUTOFF HEAD

ΔH - - - - - HEAD LOSS

ΔP - - - - - PRESSURE LOSS

REFERENCE DRAWINGS:

1. RESIDUAL HEAT REMOVAL SYSTEM P&ID - - - - - E12-1010
2. RESIDUAL HEAT REMOVAL SYSTEM DESIGN SPEC - - - - - E12-4010
3. RCIC P&ID - - - - - E51-1005
4. RECIRC SYSTEM PROCESS FLOW DIAGRAM - - - - - [illegible]
5. LOW PRESSURE CORE SPRAY PFD - - - - - E21-1020

NOTES:

1. [illegible]
2. PIPING BETWEEN POINTS WITH EMPTY DATA BLANKS [illegible]
3. SHOWN AS TYPICAL FOR ONE LOOP. [illegible]
4. [illegible]
5. ELEVATIONS ARE NOT INCLUDED [illegible]
6. [illegible] INDICATES MAXIMUM (X) AND MINIMUM (Y) VALUES FOR THE MODE SPECIFIED.
7. DASHED LINES INDICATE FLOW DOES NOT PASS THRU THESE POINTS.
8. THE NPSH AVAILABLE IN MODES [illegible]
9. FLOODING LINE SHALL BE SIZED TO [illegible]
10. THIS PORTION OF PIPING TO BE SIZED BASED ON FLOW [illegible] FUEL POOL COOLING AND FILTERING SYSTEM.
11. TABLE 1 INDICATES VALVE POSITIONS DURING VARIOUS MODES OF OPERATION.
12.
13.
14. TYPICAL VALUES FOR MAX. AND MIN. SUPPRESSION POOL TEMP SHOWN. [illegible]
15. WATER FLOWS ARE IN GPM, STEAM FLOWS ARE IN 1000 LBS/HR.
16. HEAT EXCHANGER HEAT REMOVAL AND SPRAY BASED UPON 7400 GPM SHELL SIDE FLOW.
17. FOR LINE SIZING INFORMATION SEE REF. 3.
18. THE WEIGHT OF WATER IN THE SHUTDOWN COOLING SUBSYSTEM PIPING, [illegible]
19.
20. PIPING SYSTEM DESIGN PRESSURE AND TEMPERATURE AND THE ESTIMATED LINE SIZES ARE FOR INFORMATION ONLY. [illegible]
21. [illegible]
22. TDH = 700 FT. MAX.
23. ONLY ONE LOOP IS REQUIRED AT THIS STAGE OF SHUTDOWN. [illegible]

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FIGURE 6.3-8

RESIDUAL HEAT REMOVAL SYSTEM (RHR)

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SEE NOTE 20

POSITION ○	3.1	3.2	4	5	6	7	8	9	10	47	16	46	11	7	7.1	18	19	20	8	31	4 C	5 C	6	9	10	16	46	11	25	54.1	26	4 A,B	5 A,B	16	50	27
DESIGN PRESS. IN PSIG	125	220		[illegible]	500							1250		500		RHR Hx	500			220		RHR Hx	500			500	1250		SEE REF 4 M05-1-2	220				500	SEE REF 4 M05-1-[illegible]	
DESIGN TEMP IN °F	212	358			358						480	575		358			358			212			212			480	575			358				480		
ESTIMATED LINE SIZE	24"				18"						12"			18"			18"			24"			18"				12"		20"		18"	24"		12"		
	LPCI LINE (VIA RHR Hx BYPASS)													Hx LINE						LPCI LINE (DIRECT TO REACTOR)									SHUTDOWN SUCTION					SHUTDOWN RETURN LINE		

POSITION ○	56	56.1	4	49	49.1	45	6 C	6.1 C	6.2	43	6 A&B	6.1 A&B	43
DESIGN PRESS. IN PSIG	125			500	125		500	125			500	125	
DESIGN TEMP IN °F	212			212			212					212	
ESTIMATED LINE SIZE	18"			18"			3"		3"		3"		
	SYSTEM TEST LINES (LOOP C)						MIN FLOW BYPASS LINES (LOOP C)				MIN FLOW BYPASS (L[illegible]		

TABLE 1 - VALVE POSITION CHART

SEE NOTE 11

F004 A&B / F004C			[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	F016	STRAINER AB	C
MODE A-1	O	O						O		O	O																						1-50% P	
MODE A-2	O	O						O		O	O																						1-50% P	
MODE B		O					O		O							O	O	O													O		[illegible]	---
MODE D			O				O	TC	O					T	O					T		O							O	O	O			
MODE E			O				O		O					O-T	O													O	O	O	O			
MODE F	O	O						O											T-O				T-O										CLEAR	CLEAR
MODE G	O	O	O																									O	O	O		O	CLEAR	CLEAR
MODE S	O	O					O	O	O																							O	CLEAR	CLEAR

O - VALVE OPEN
T - VALVE THROTTLED
P - STRAINER PLUGGED
T-C - VALVE THROTTLED OR CLOSED
O-T - VALVE OPEN OR THROTTLED
BLANK SPACE INDICATES VALVE IS CLOSED

TABLE 3 LIMITING LINE LOSSES (NUMBERS REFER TO POSITION ○)

MODE A-1	6-11 PUMP DISCHARGE LINE TO RPV FLOODING PENETRATION
MODE A-2	NONE
MODE B	1-2-3 (SUCTION LINE SUPPRESSION POOL TO PUMP), 47-21 & 15-23 (CONTAINMENT SPRAY LINES PUMP DISCHARGE TO TO SPRAY HEADERS)
MODE D	14-51-33-29 (VESSEL HEAD SPRAY LINE) SEE NOTE 10 29 25-5 (SHUTDOWN SUCUTION LINE RPV TO PUMP) 56-4 (SHUTDOWN SUCUTION LINE TEST BRANCH TO PUMP)
MODE E	16-27 (SHUTDOWN DISCHARGE LINE LPCI BRANCH TO RECIRC LOOP)
MODE F	13-43-24 14 49-45 (TEST LINE TO SUPPRESSION POOL)
MODE G	6-43 (PUMP MINIMUM BYPASS LINE)

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FIGURE 6.3-8

RESIDUAL HEAT REMOVAL SYSTEM (RHR)
(SHEET 2 of 3)

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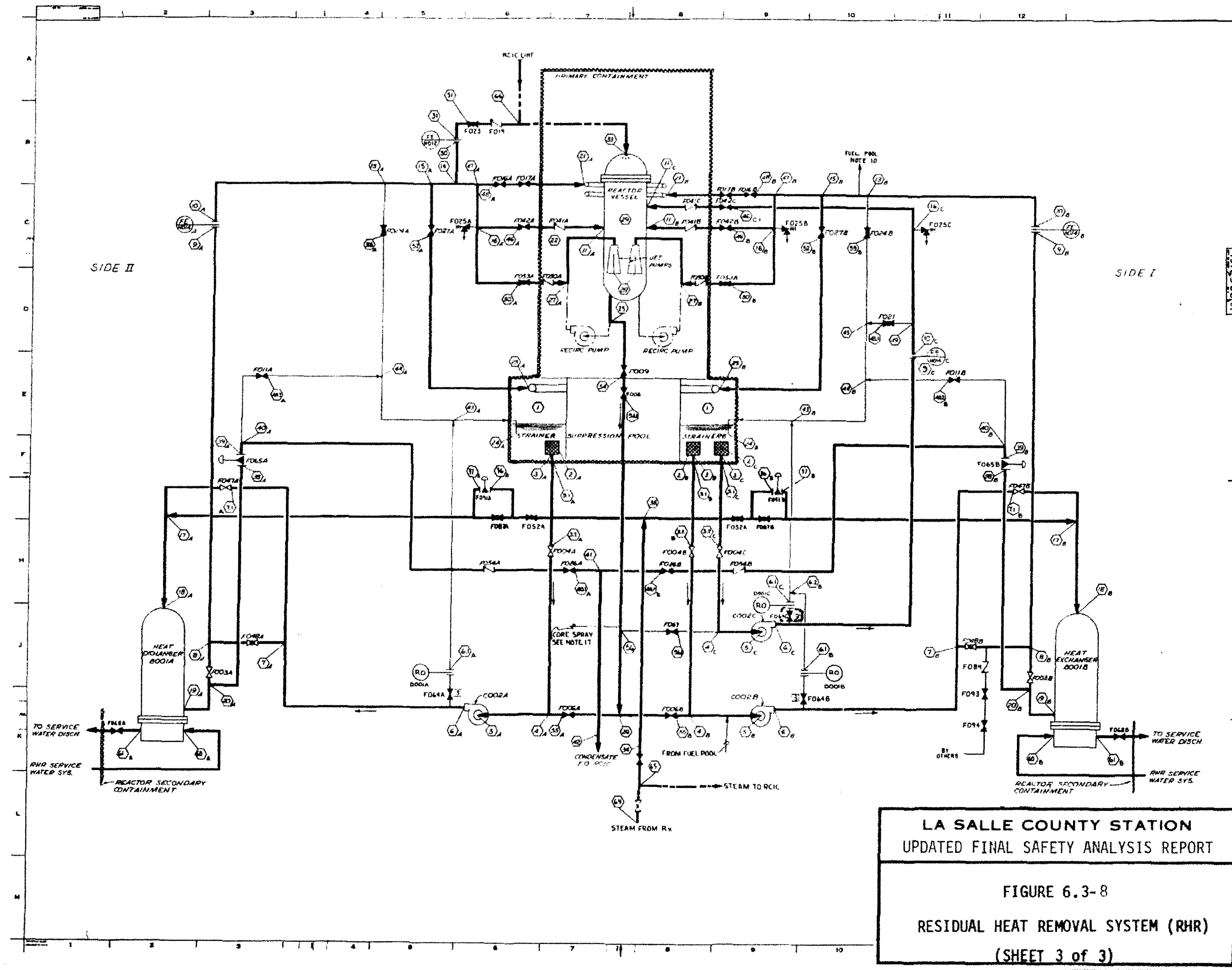

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FIGURE 6.3-8

RESIDUAL HEAT REMOVAL SYSTEM (RHR)

(SHEET 3 of 3)

REV. 14, APRIL 2002

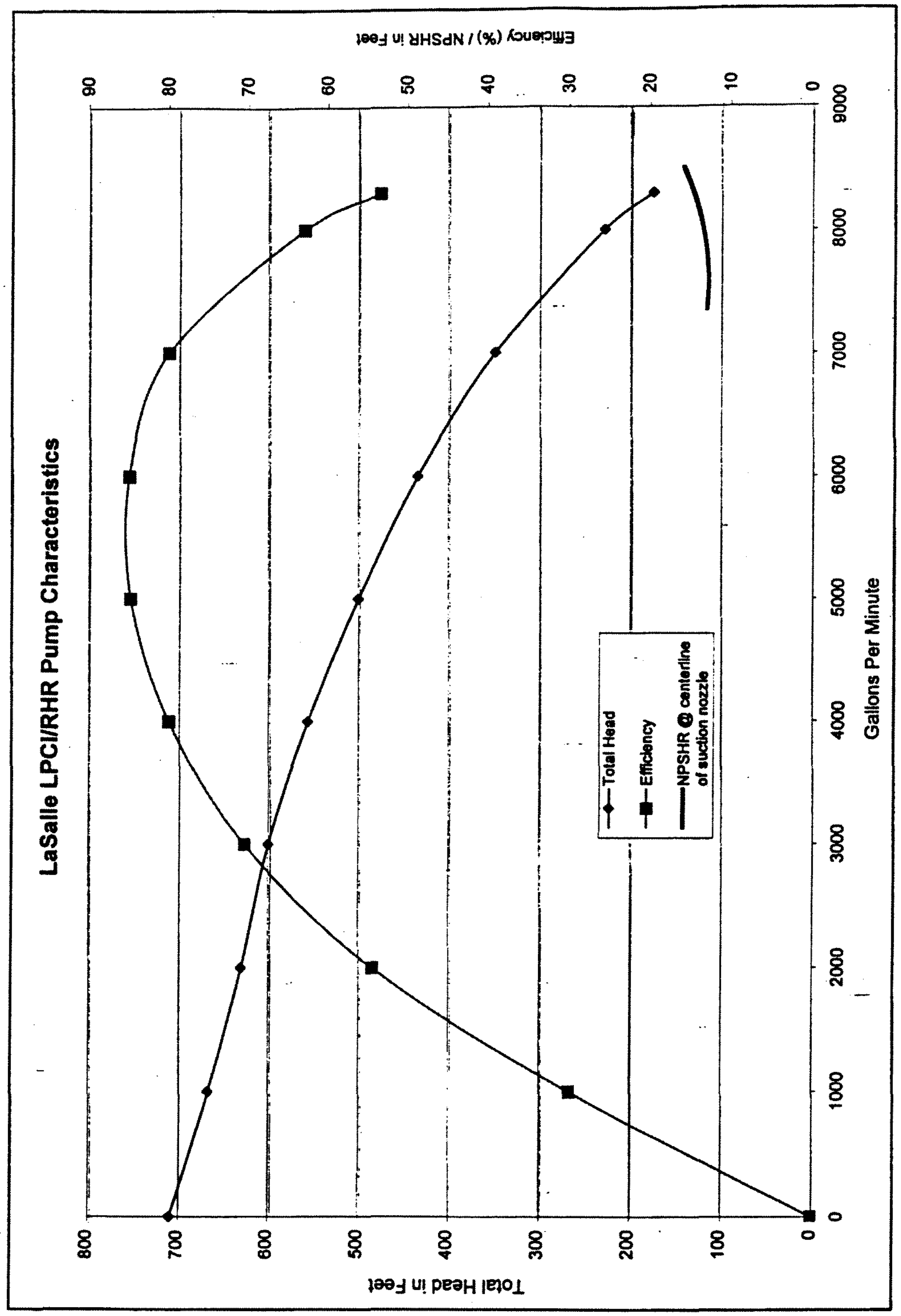

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FIGURE 6.3-9

LPCI PUMP CHARACTERISTICS

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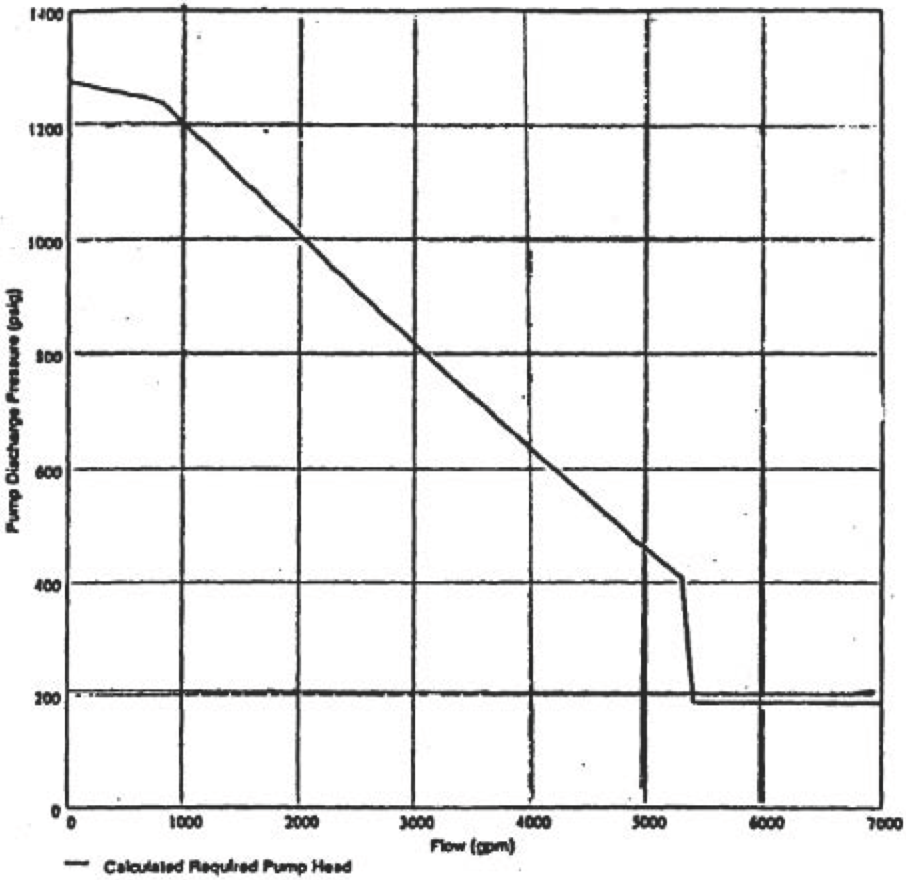

Source: Reference 2

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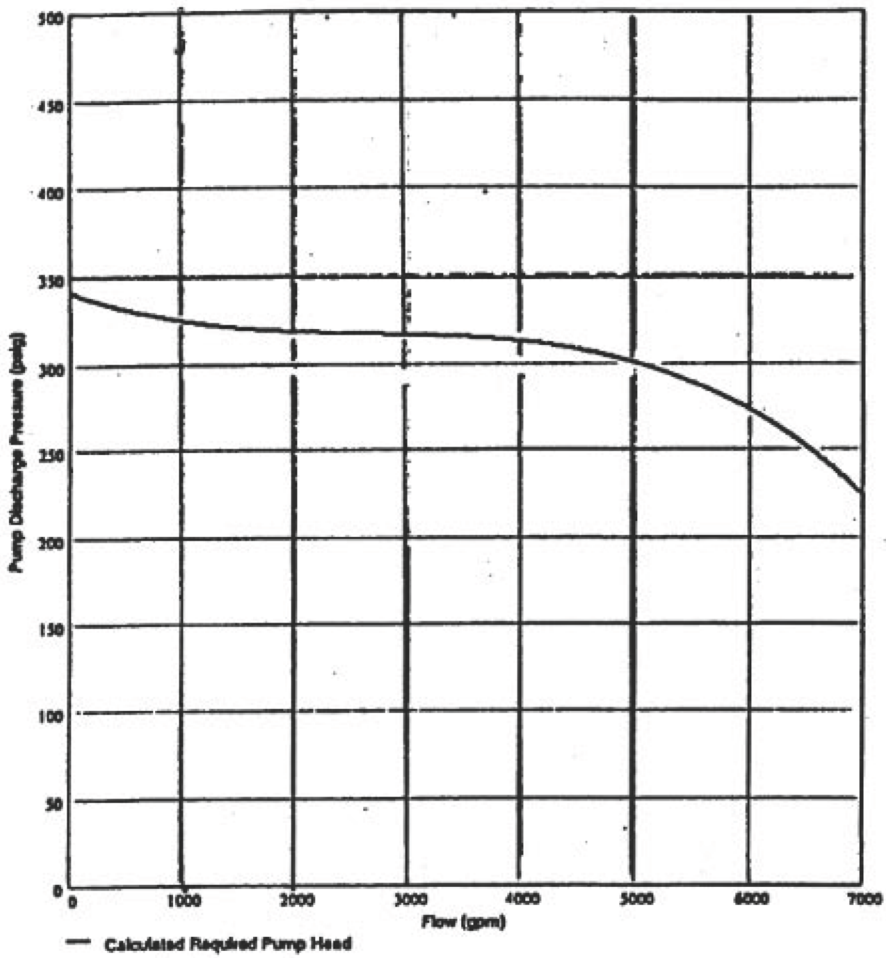
FIGURE 6.3-10

HPCS MINIMUM REQUIRED PUMP HEAD TO MEET LOCA ANALYSES ASSUMPTIONS



Source: Reference 2

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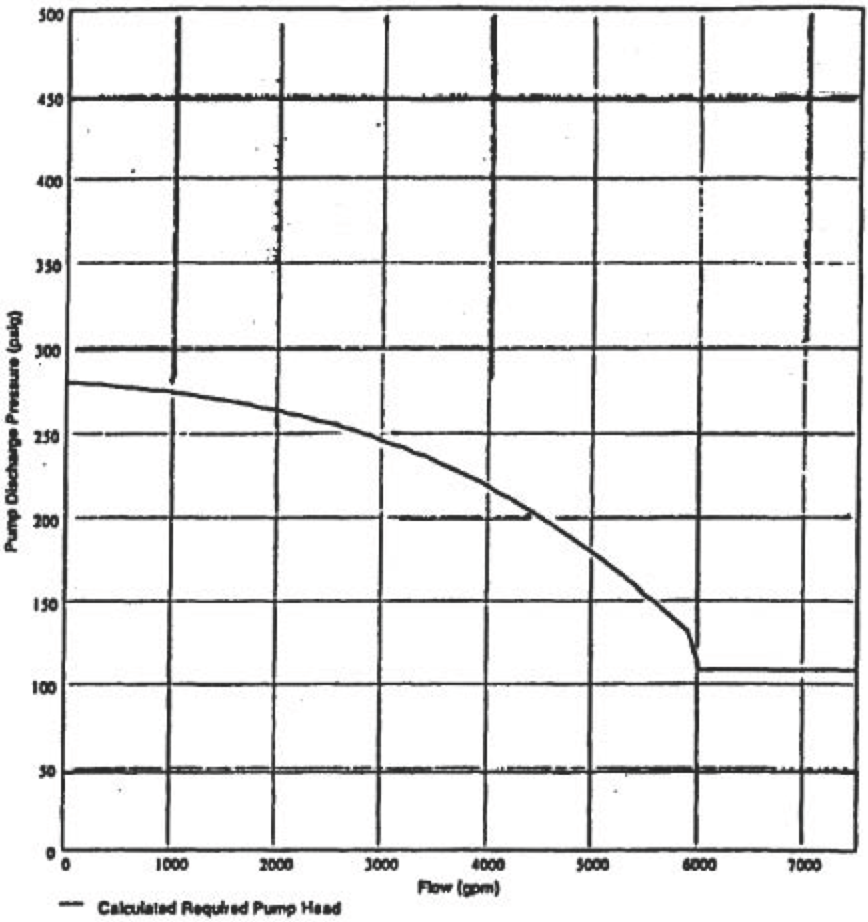
FIGURE 6.3-11 LPCS MINIMUM REQUIRED PUMP HEAD TO MEET LOCA ANALYSES ASSUMPTIONS



Source: Reference 2

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FIGURE 6.3-12 LPCI MINIMUM REQUIRED PUMP HEAD TO MEET LOCA ANALYSES ASSUMPTIONS

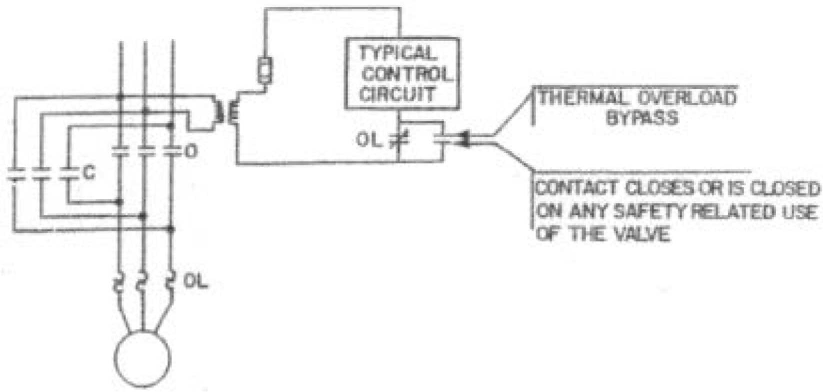

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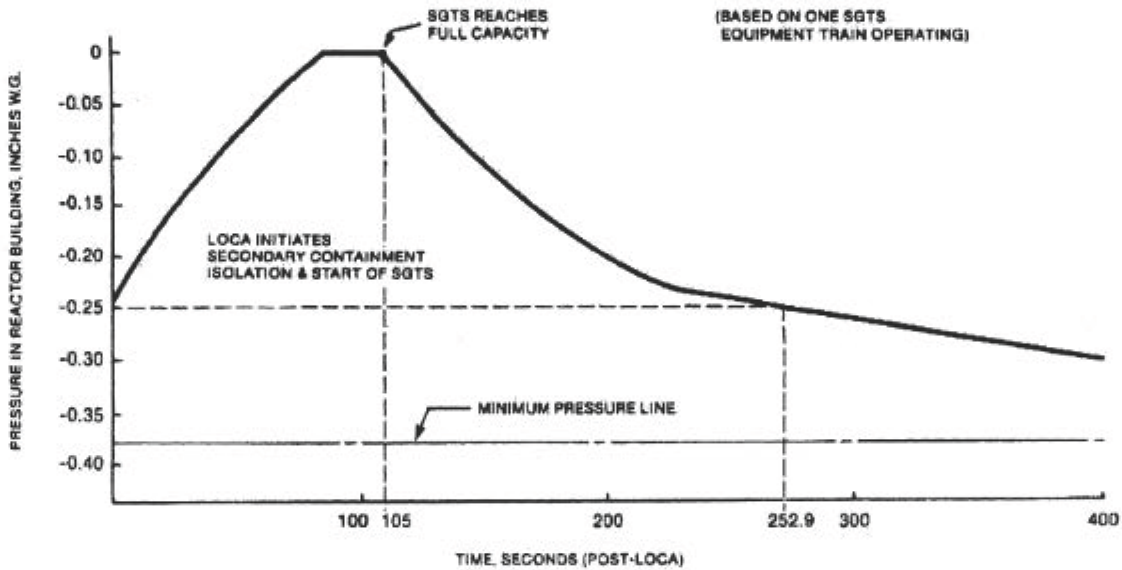
FIGURE 6.3-13 SCHEMATIC OF THERMAL OVERLOAD BYPASS CIRCUITRY



NOTE: This figure was used to support original licensing. For current licensing requirements for system pressure-time response, see the Technical Specifications.

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FIGURE 6.3-14 POST LOCA TIME-PRESSURE IN SECONDARY CONTAINMENT (BASED ON ONE SGTS EQUIPMENT TRAIN OPERATING)

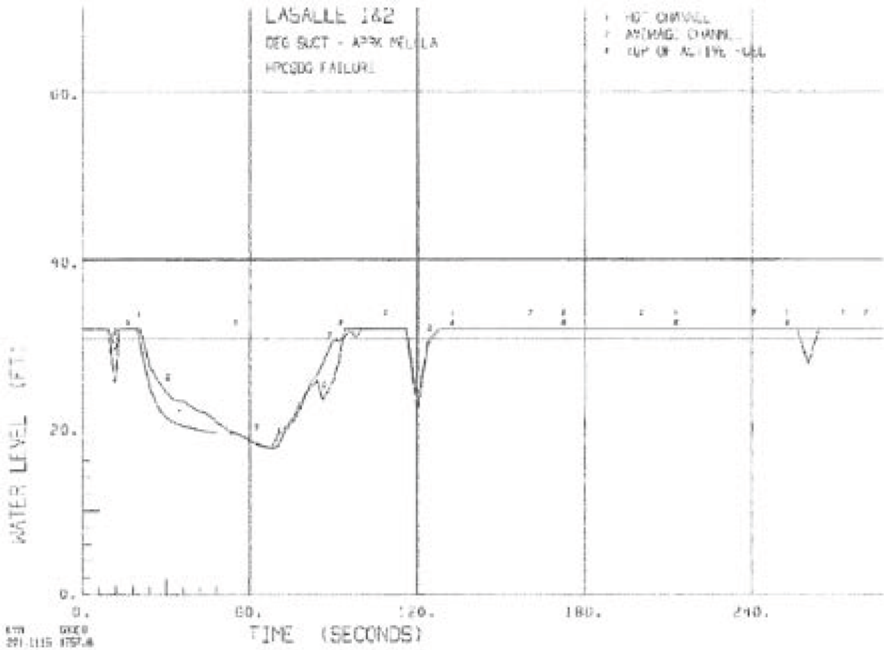

Figure 6.3-15-a Water Level in Hot and Average Channels,
Limiting Large Recirculation Suction Line Break (DEG), HPCS-DG Failure, GNF2 Fuel
LPCS + 3 LPCI + 6 ADS Available, Appendix K Assumptions

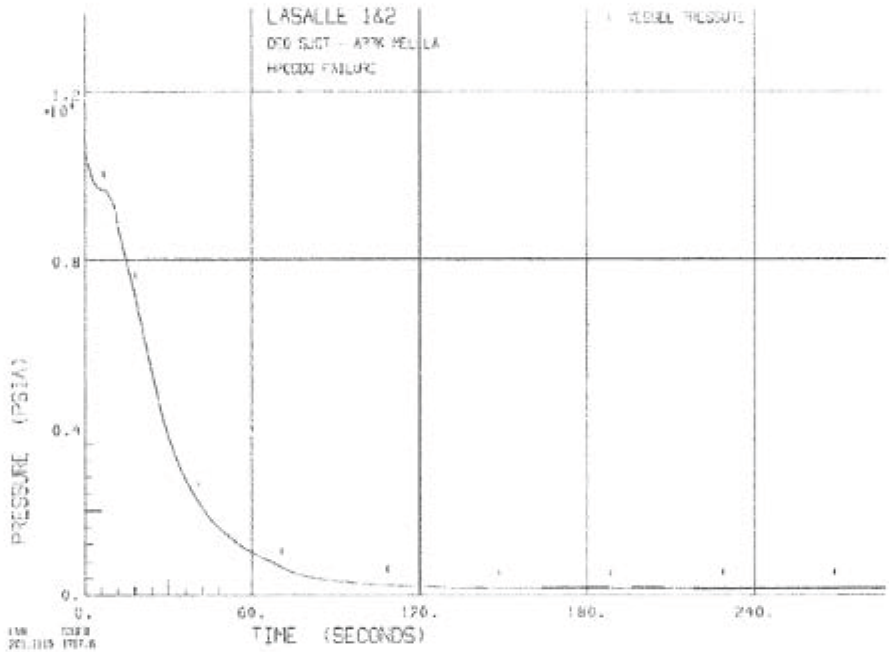

Figure 6.3-15-b Reactor Vessel Dome Pressure,
Limiting Large Recirculation Suction Line Break (DEG), HPCS-DG Failure, GNF2 Fuel
LPCS + 3 LPCI + 6 ADS Available, Appendix K Assumptions

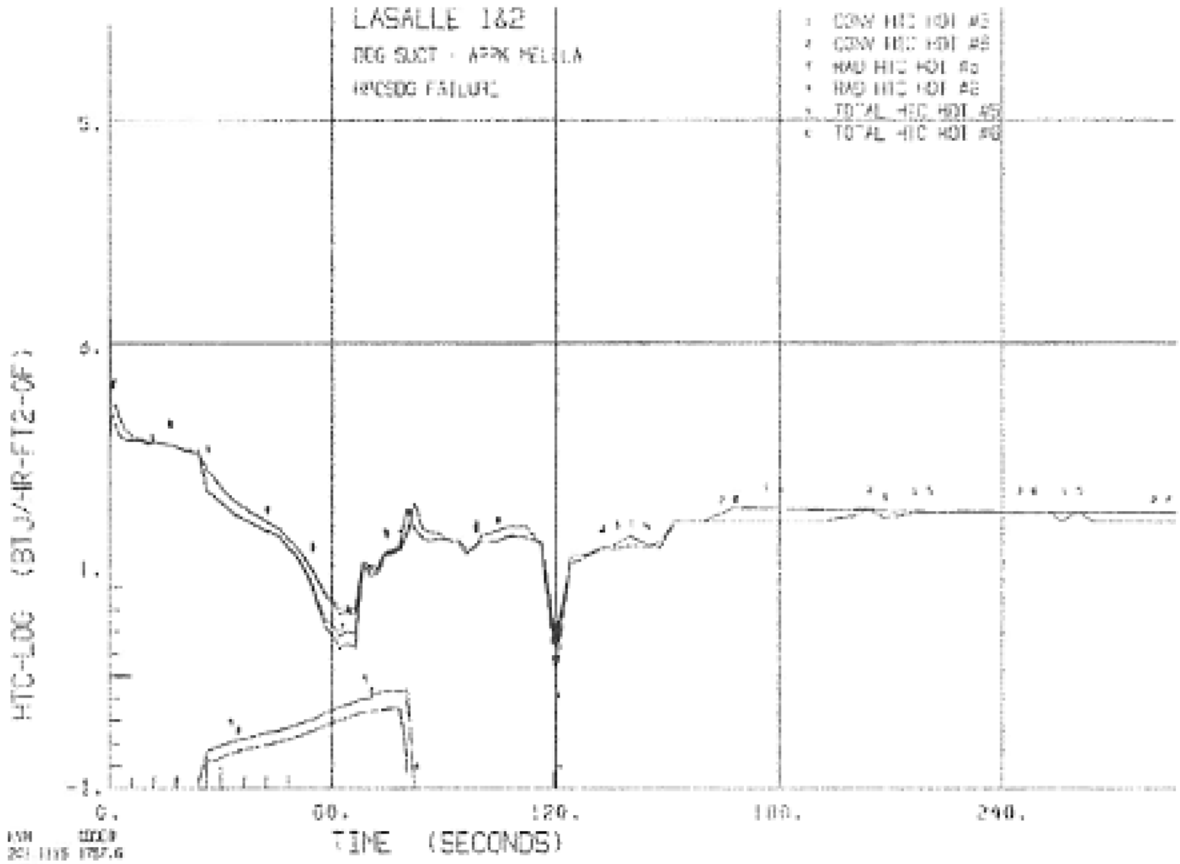

Figure 6.3-15-c Heat Transfer Coefficients,
Limiting Large Recirculation Suction Line Break (DEG), HPCS-DG Failure, GNF2 Fuel
LPCS + 3 LPCI + 6 ADS Available, Appendix K Assumptions

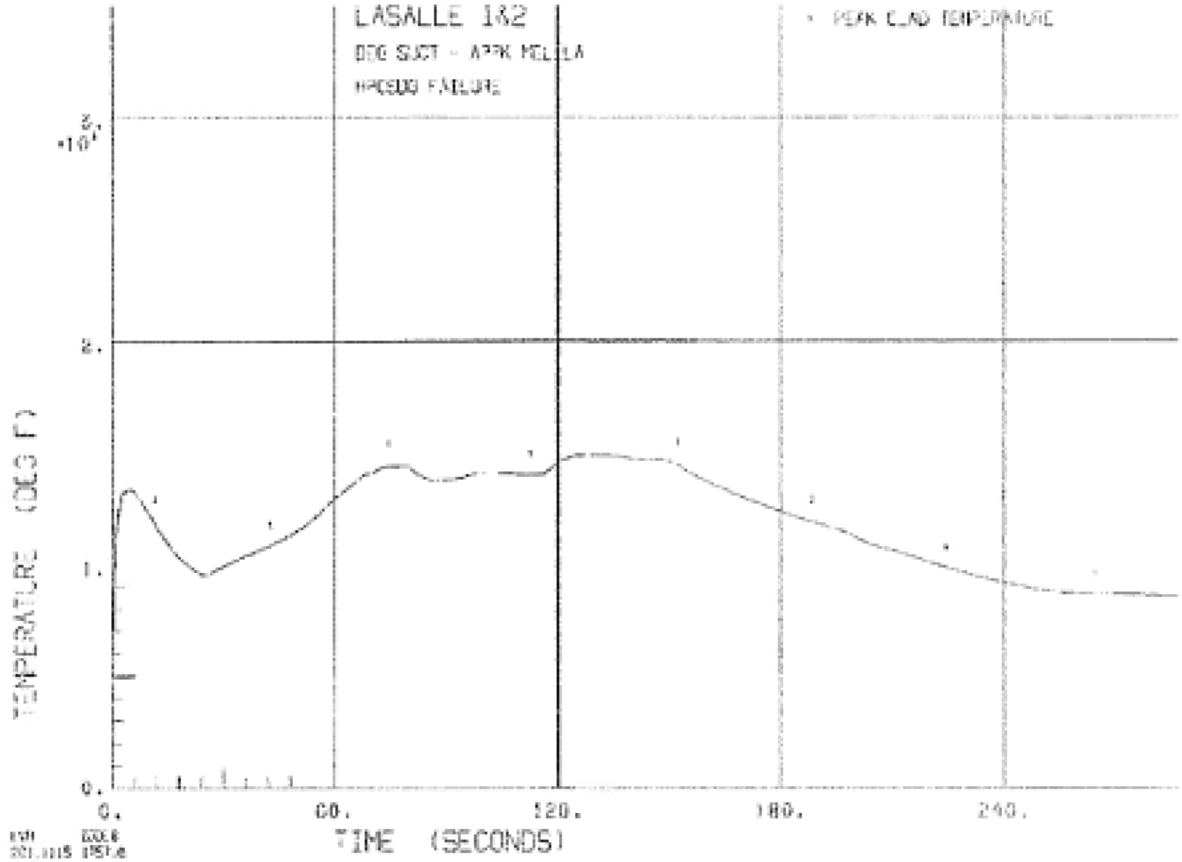

Figure 6.3-15-d Peak Cladding Temperature,
Limiting Large Recirculation Suction Line Break (DEG), HPCS-DG Failure, GNF2 Fuel
LPCS + 3 LPCI + 6 ADS Available, Appendix K Assumptions

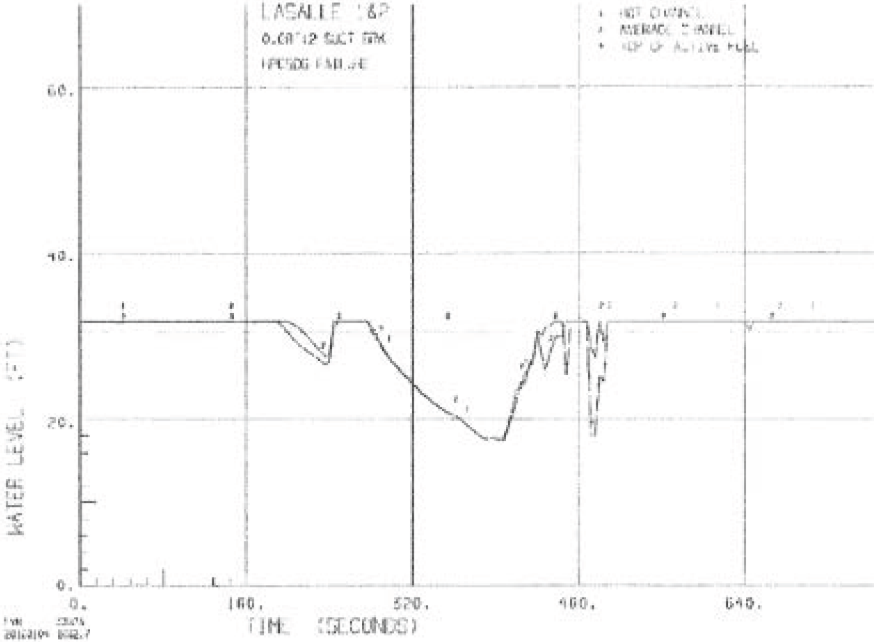

Figure 6.3-16-a Water Level in Hot and Average Channels,
Limiting Small Recirculation Suction Line Break (0.08 ft²), HPCS-DG Failure, GNF2 Fuel
LPCS + 3 LPCI + 6 ADS Available, Appendix K Assumptions

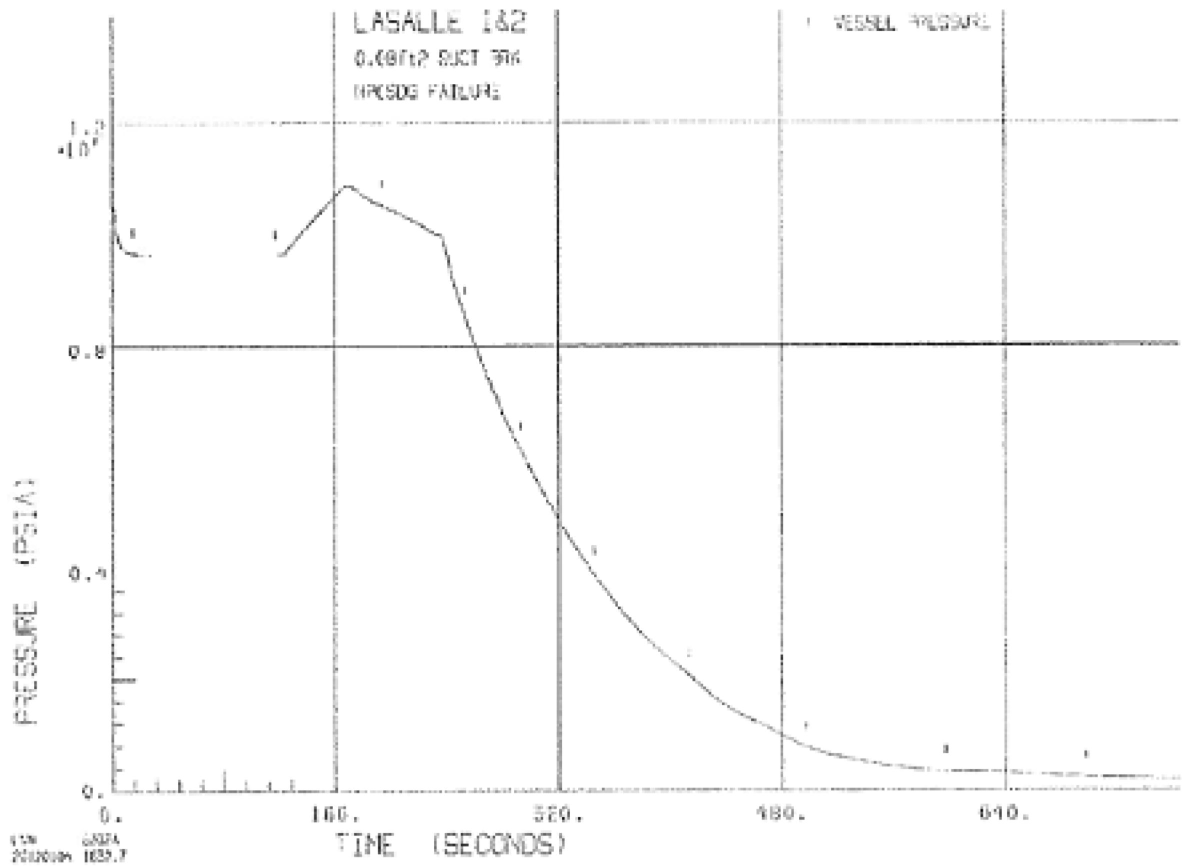

Figure 6.3-16-b Reactor Vessel Dome Pressure,
Limiting Small Recirculation Suction Line Break (0.08 ft^2), HPCS-DG Failure, GNF2 Fuel
LPCS + 3 LPCI + 6 ADS Available, Appendix K Assumptions

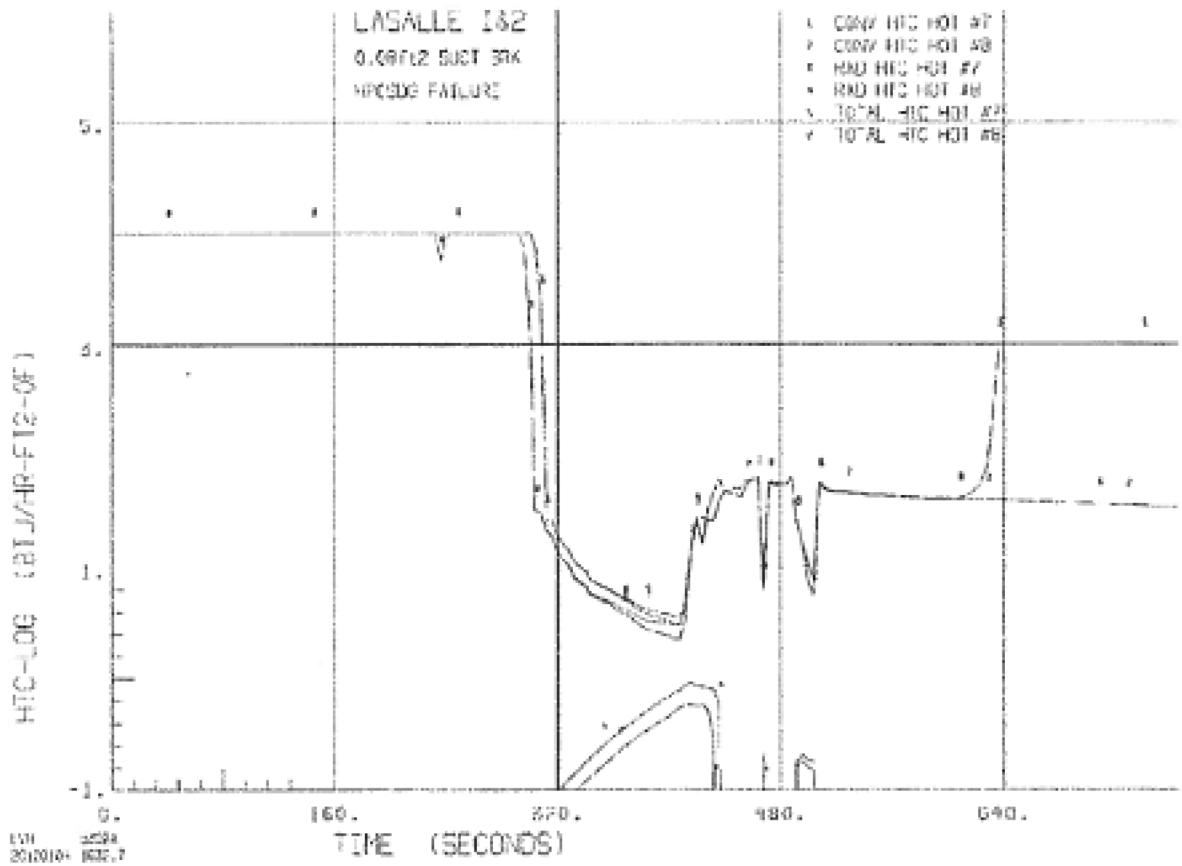

Figure 6.3-16-c Heat Transfer Coefficients,
Limiting Small Recirculation Suction Line Break (0.08 ft²), HPCS-DG Failure, GNF2 Fuel
LPCS + 3 LPCI + 6 ADS Available, Appendix K Assumptions

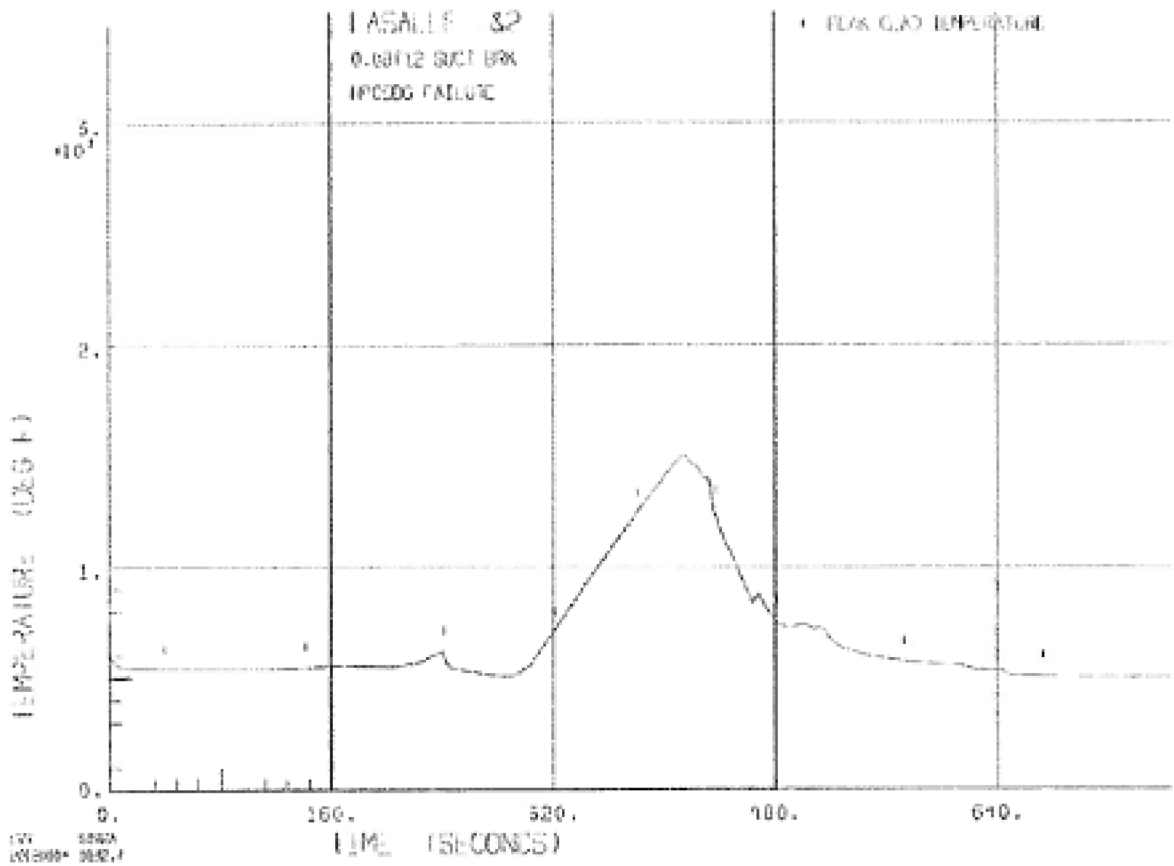

Figure 6.3-16-d Peak Cladding Temperature,
Limiting Small Recirculation Suction Line Break (0.08 ft²), HPCS-DG Failure, GNF2 Fuel
LPCS + 3 LPCI + 6 ADS Available, Appendix K Assumptions

[redacted]

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FIGURE 6.4-1

CONTROL AND AUXILIARY ELECTRIC ROOM LAYOUT
(SHEET 1 OF 2)

[redacted]

LA SALLE COUNTY STATION
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FIGURE 6.4-1

CONTROL AND AUXILIARY ELECTRIC
EQUIPMENT ROOM LAYOUT

(SHEET 2 OF 2)

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[redacted]

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FIGURE 6.4-2

LOCATION OF OUTSIDE AIR INTAKES

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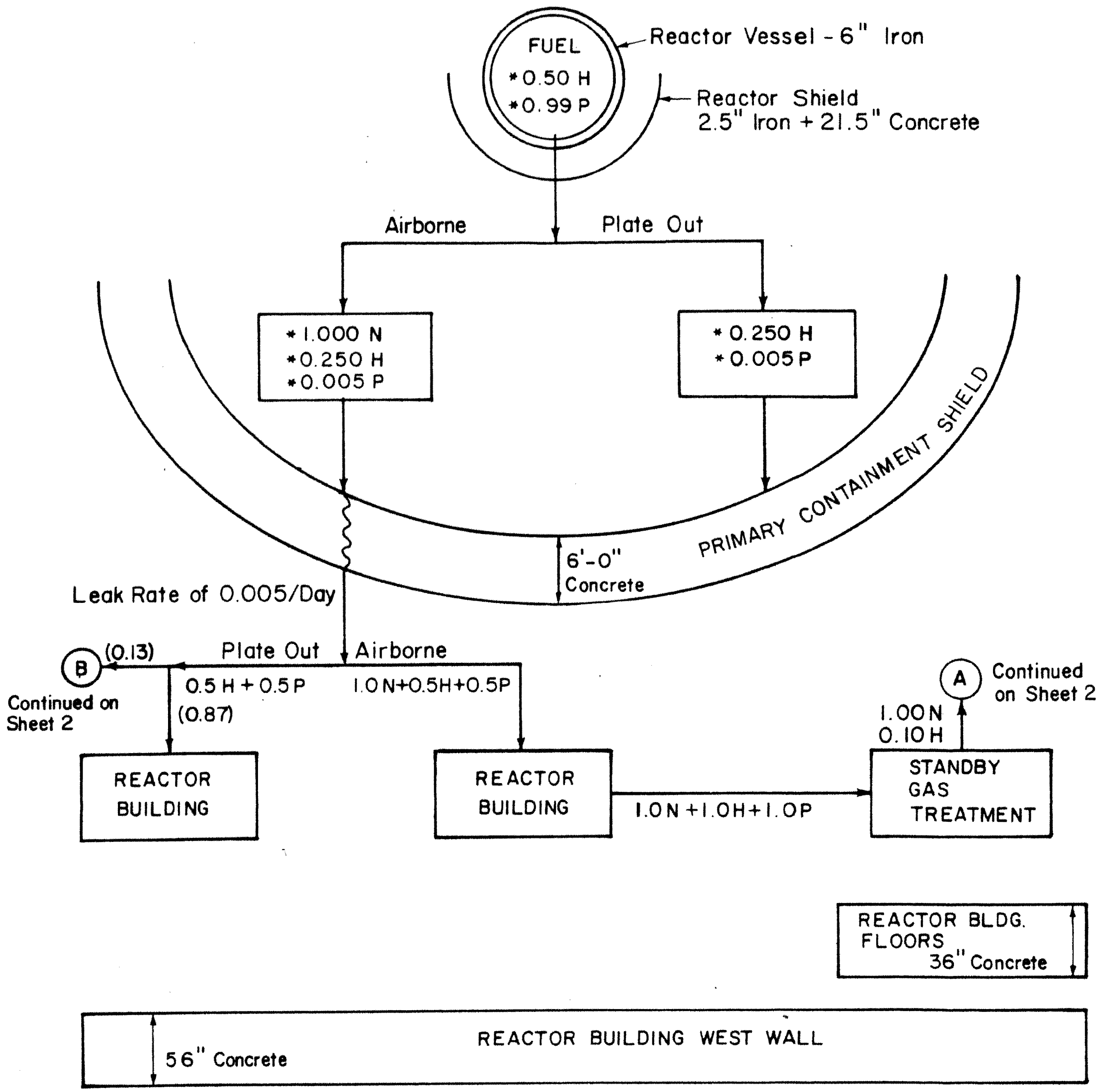

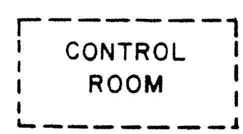

LEGEND

N - Noble Gases

H - Halogen

P - Particulates

* - Distribution of fission products immediately following a LOCA

NOTES

1. Flows beyond the primary containment are fractions of the upstream input.
2. The .635 % per day leak rate will increase the downstream sources by approximately 25%
$[(1-e^{-.00635t})/(1-e^{-.005t}) \simeq 1.25]$

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FIGURE 6.4-3

CONTROL ROOM SHIELDING MODEL

(SHEET 1 of 2)

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[redacted]

LASALLE COUNTY STATION
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FIGURE 6.4-3

CONTROL ROOM SHIELDING MODEL
(SHEET 2 OF 2)

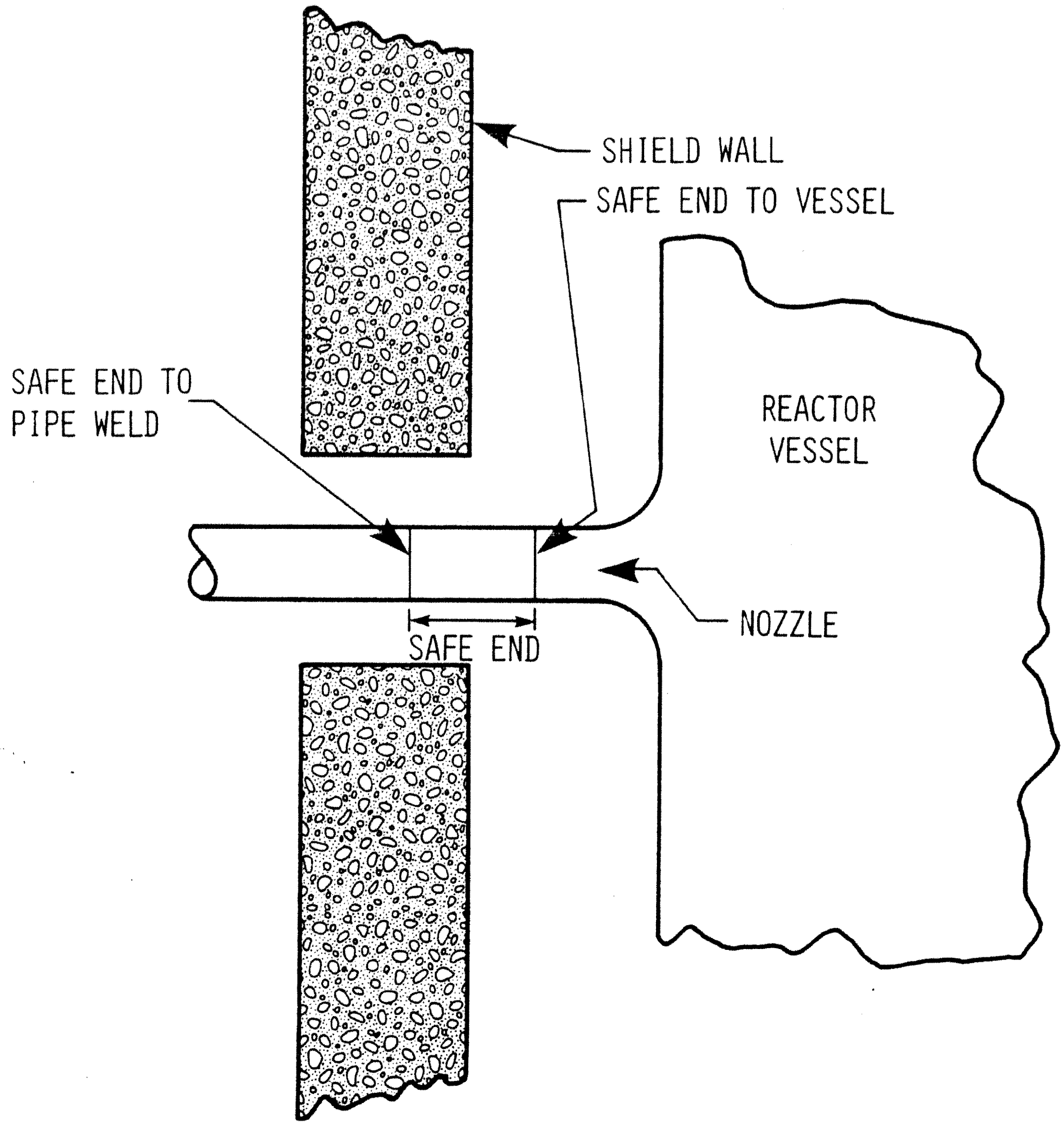

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FIGURE 6.A-1

SAFE END BREAK LOCATION

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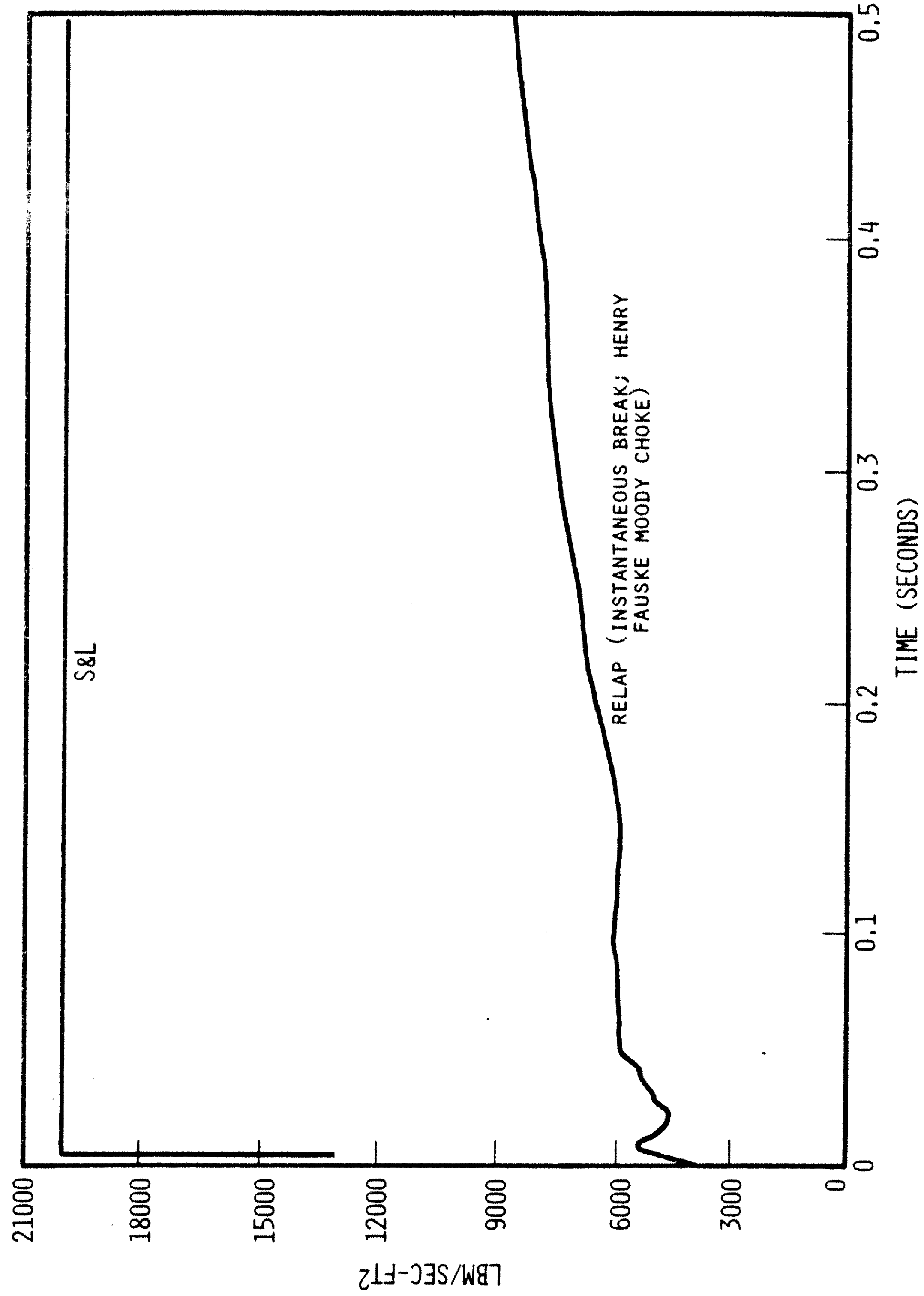

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FIGURE 6.A-2

BREAK FLOW VS. TIME - FEEDWATER LINE BREAK

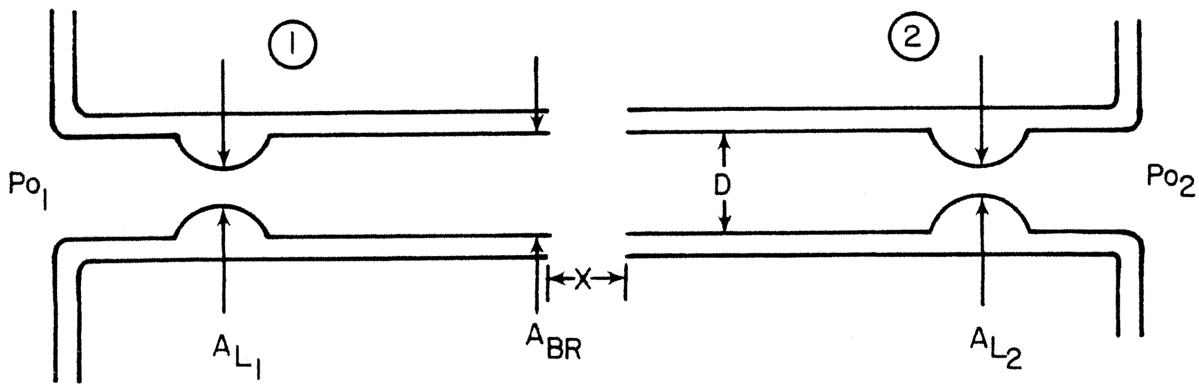

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FIGURE 6.A-3

GEOMETRY

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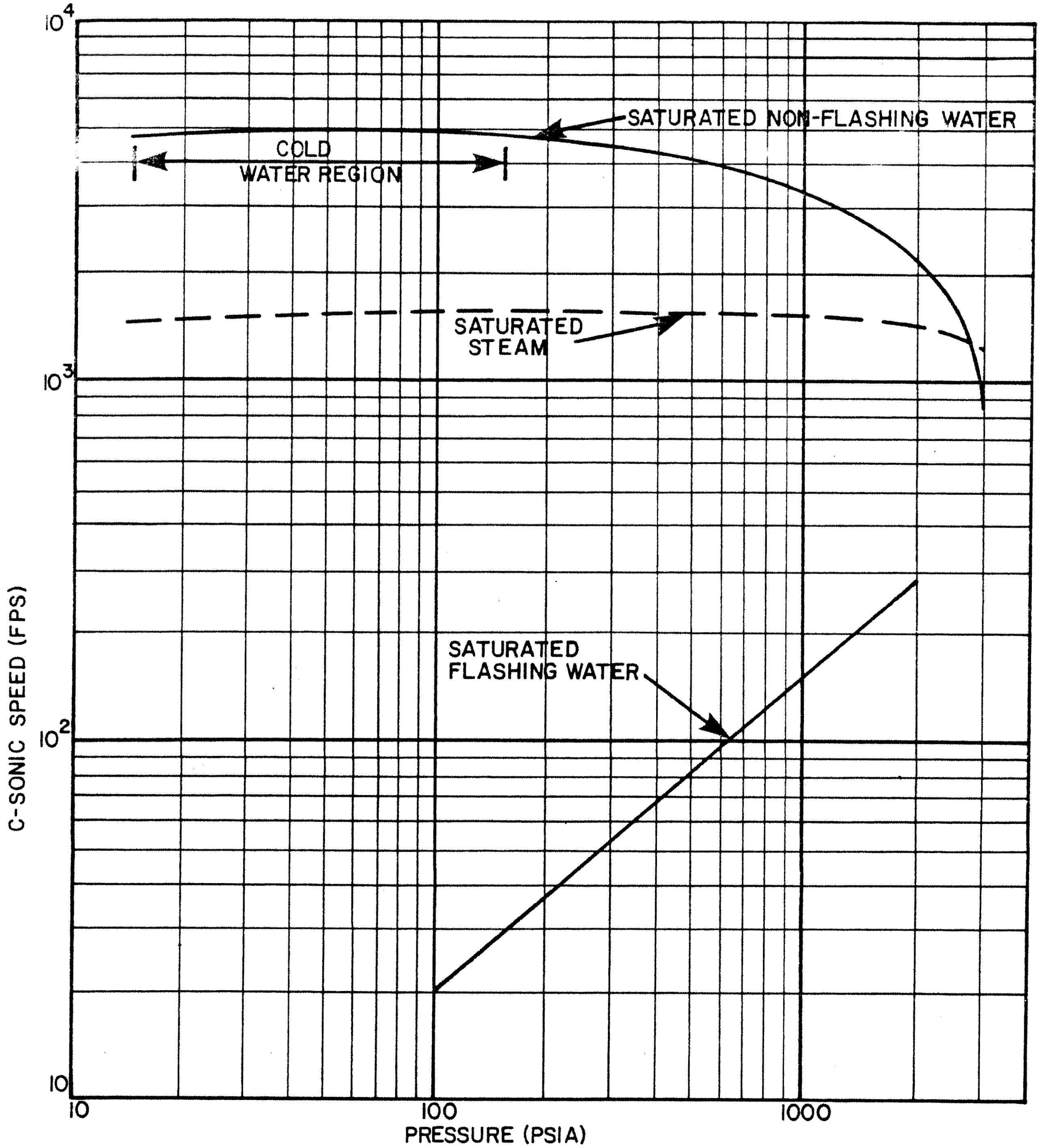

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FIGURE 6.A-4

WAVE SPEED

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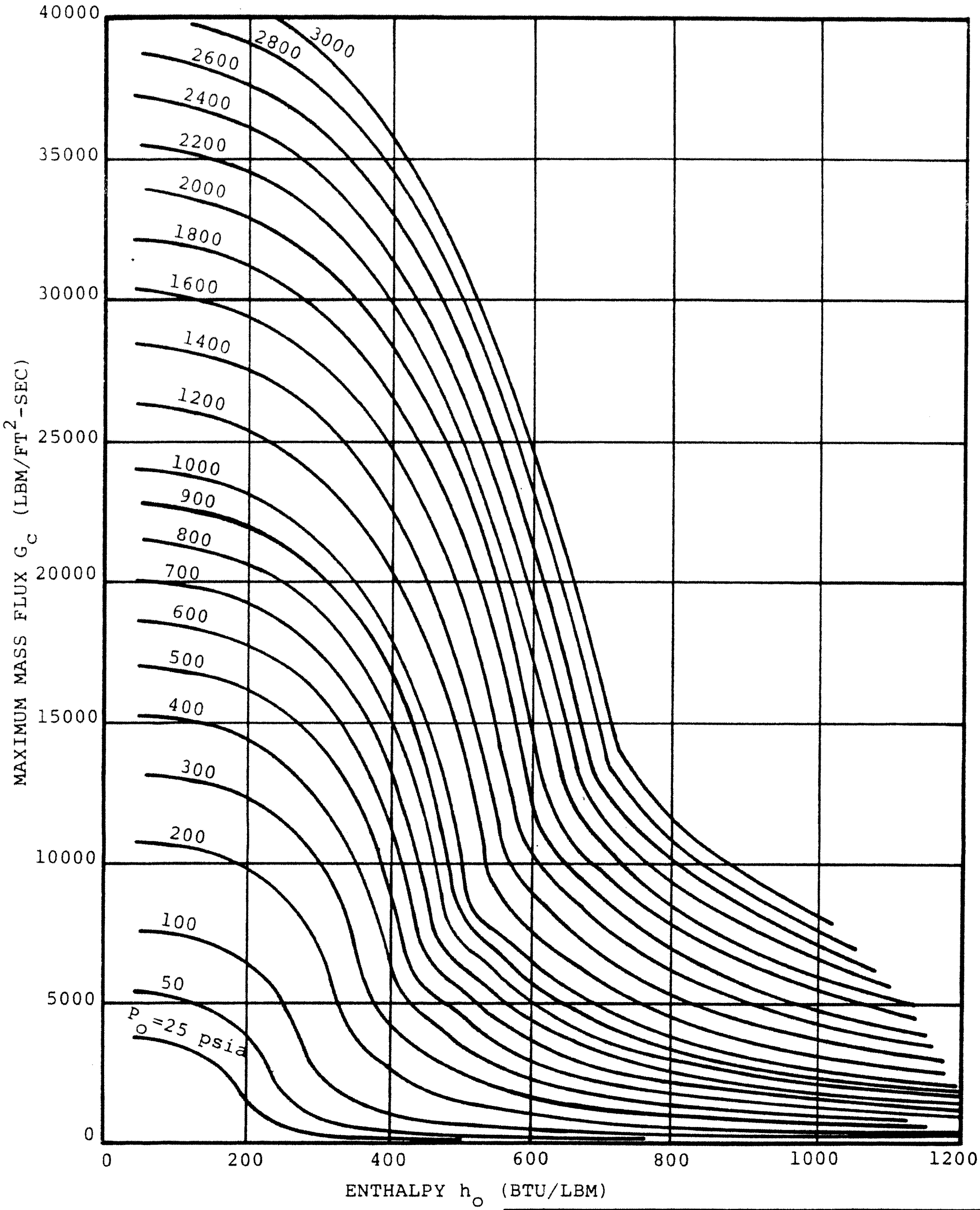

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FIGURE 6.A-5

MASS FLUX, MOODY STEADY SLIP FLOW

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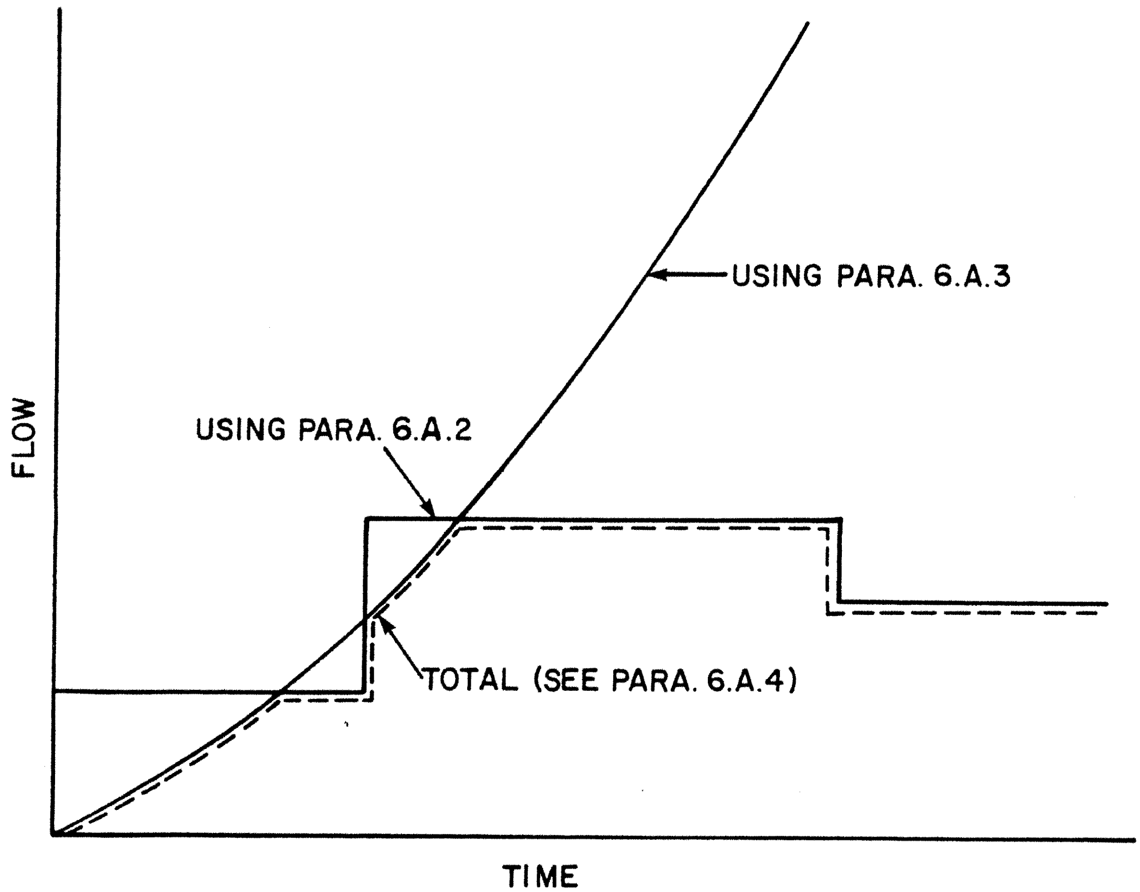

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FIGURE 6.A-6

BREAK FLOW VS. TIME

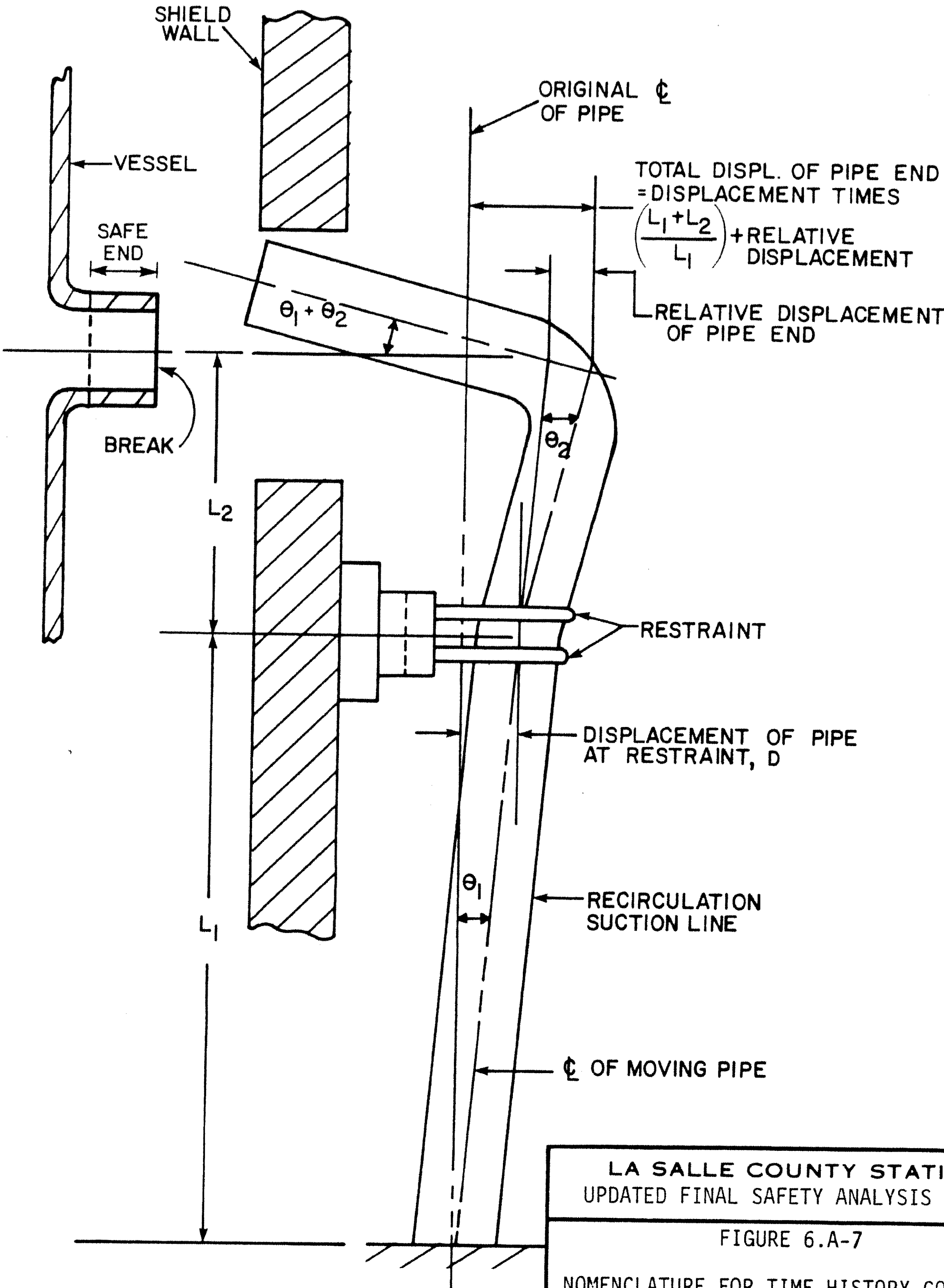

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FIGURE 6.A-7

NOMENCLATURE FOR TIME HISTORY COMPUTER PRINTOUT

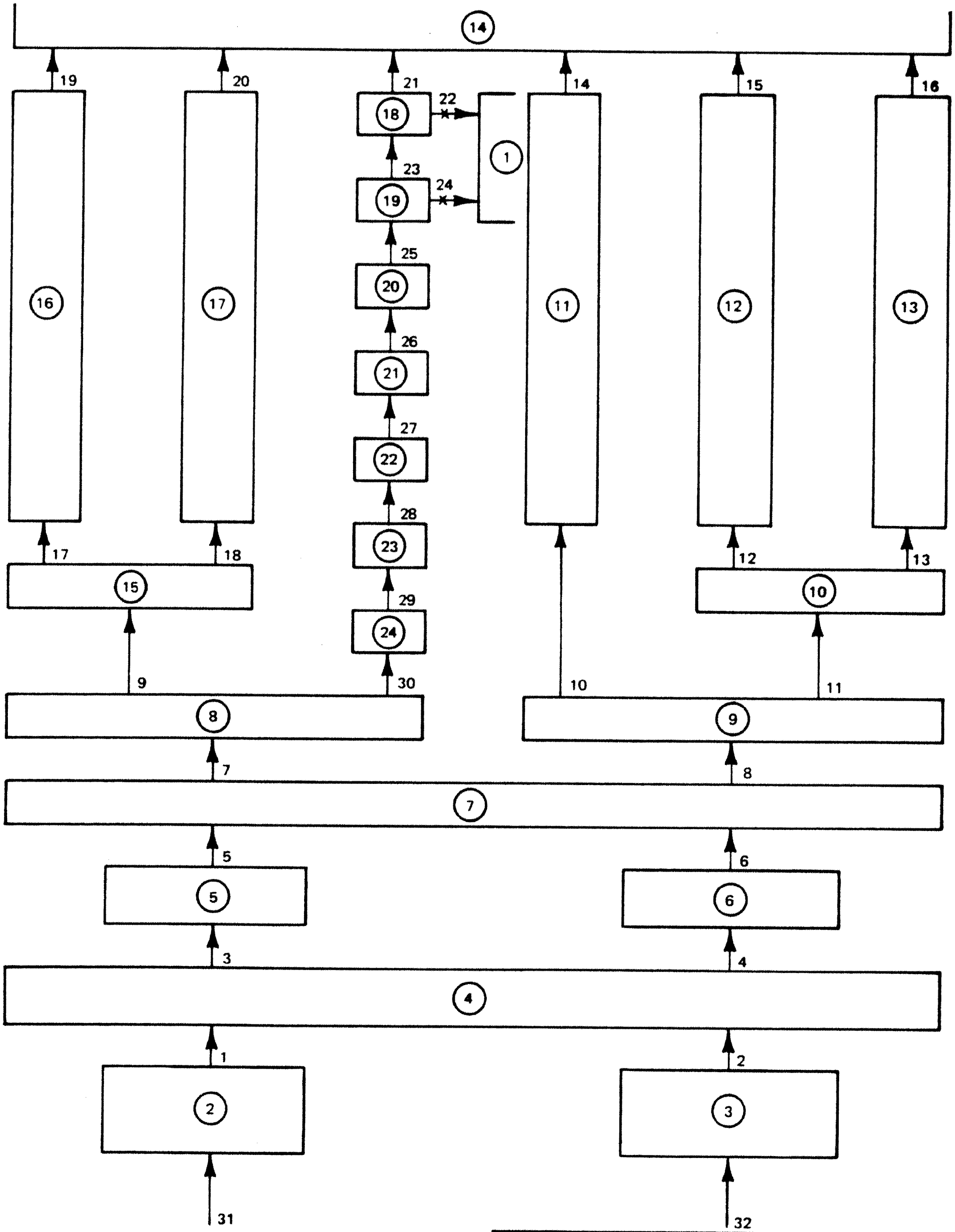

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FIGURE 6.A-8

FEEDWATER LINE SYSTEM NODALIZATION - LEG EA

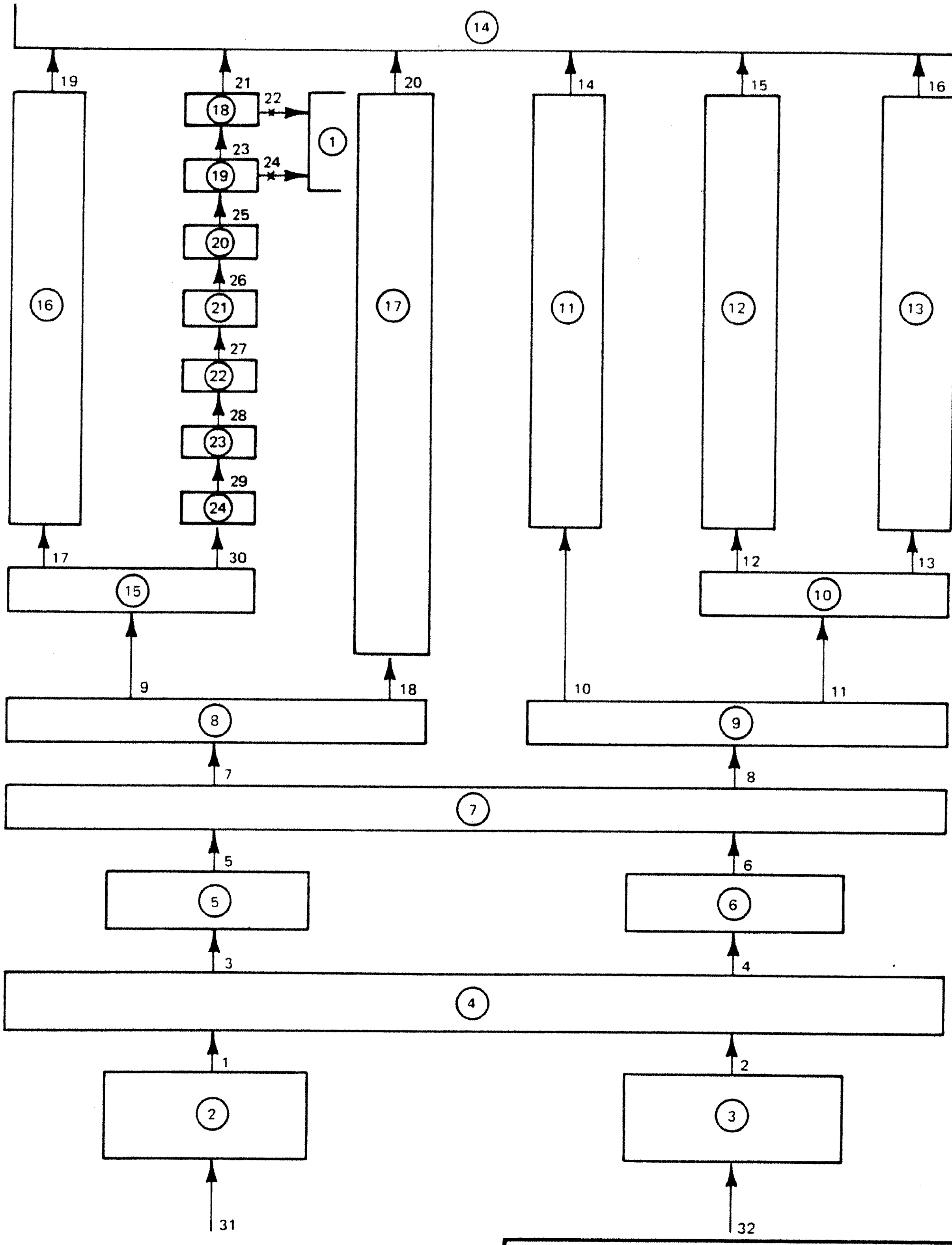

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FIGURE 6.A-9

FEEDWATER LINE SYSTEM NODALIZATION - LEG EB

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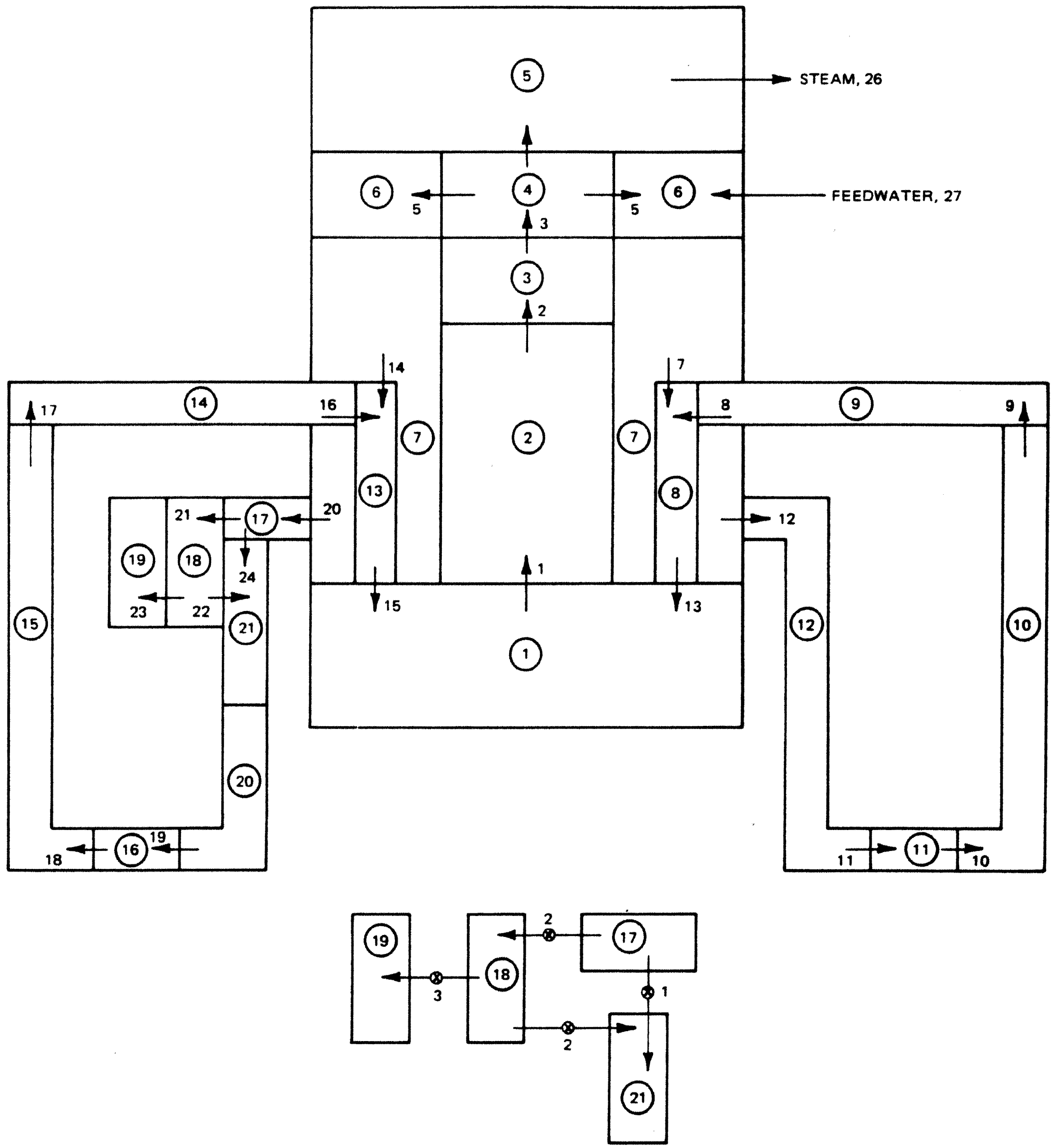

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FIGURE 6.A-10

RECIRCULATION LINE SYSTEM
NODALIZATION

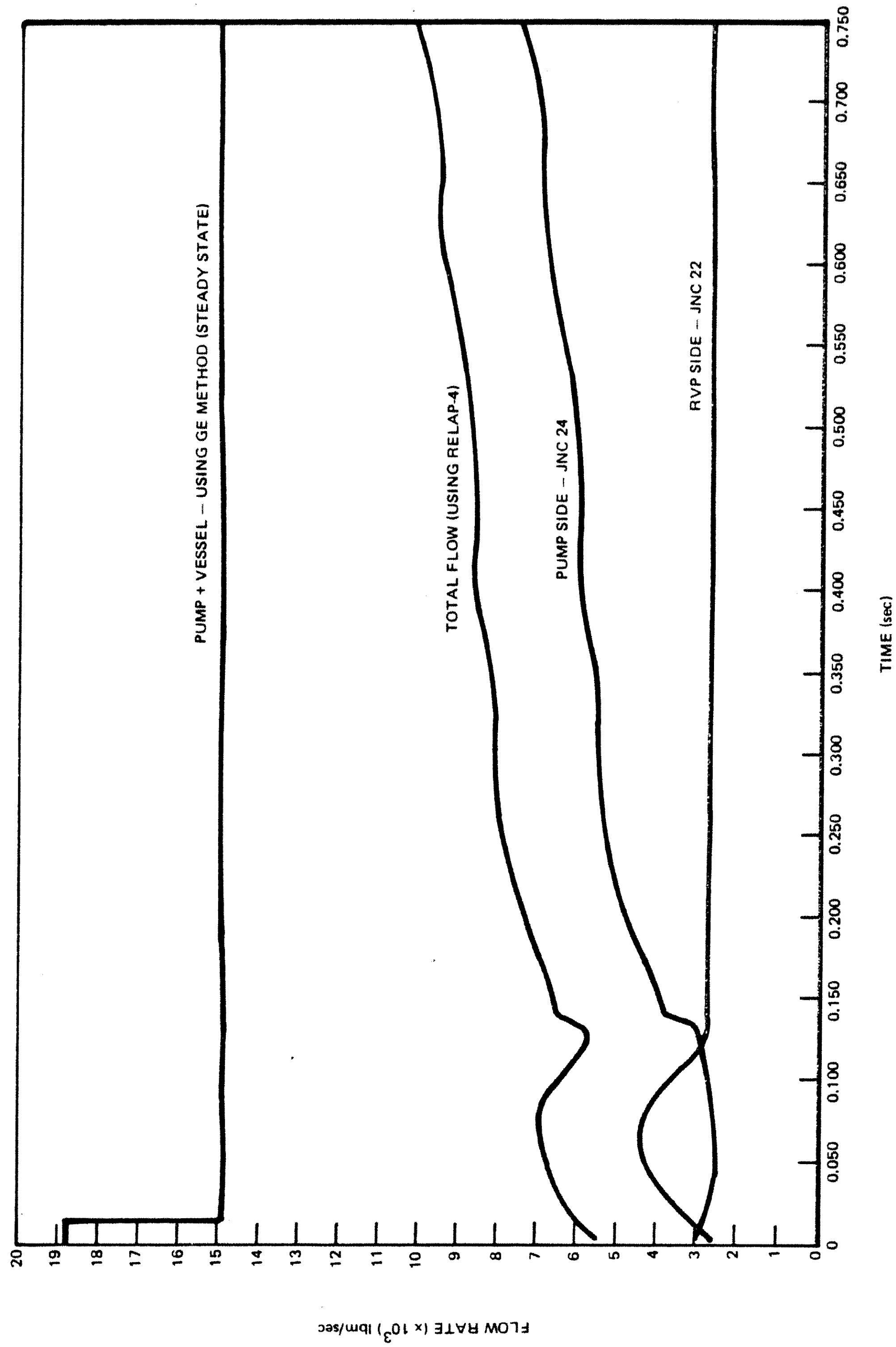

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FIGURE 6.A-11

COMPARISON OF THE GE AND RELAP4/MOD5 METHODS - FEEDWATER LINE BREAK, LEG EA

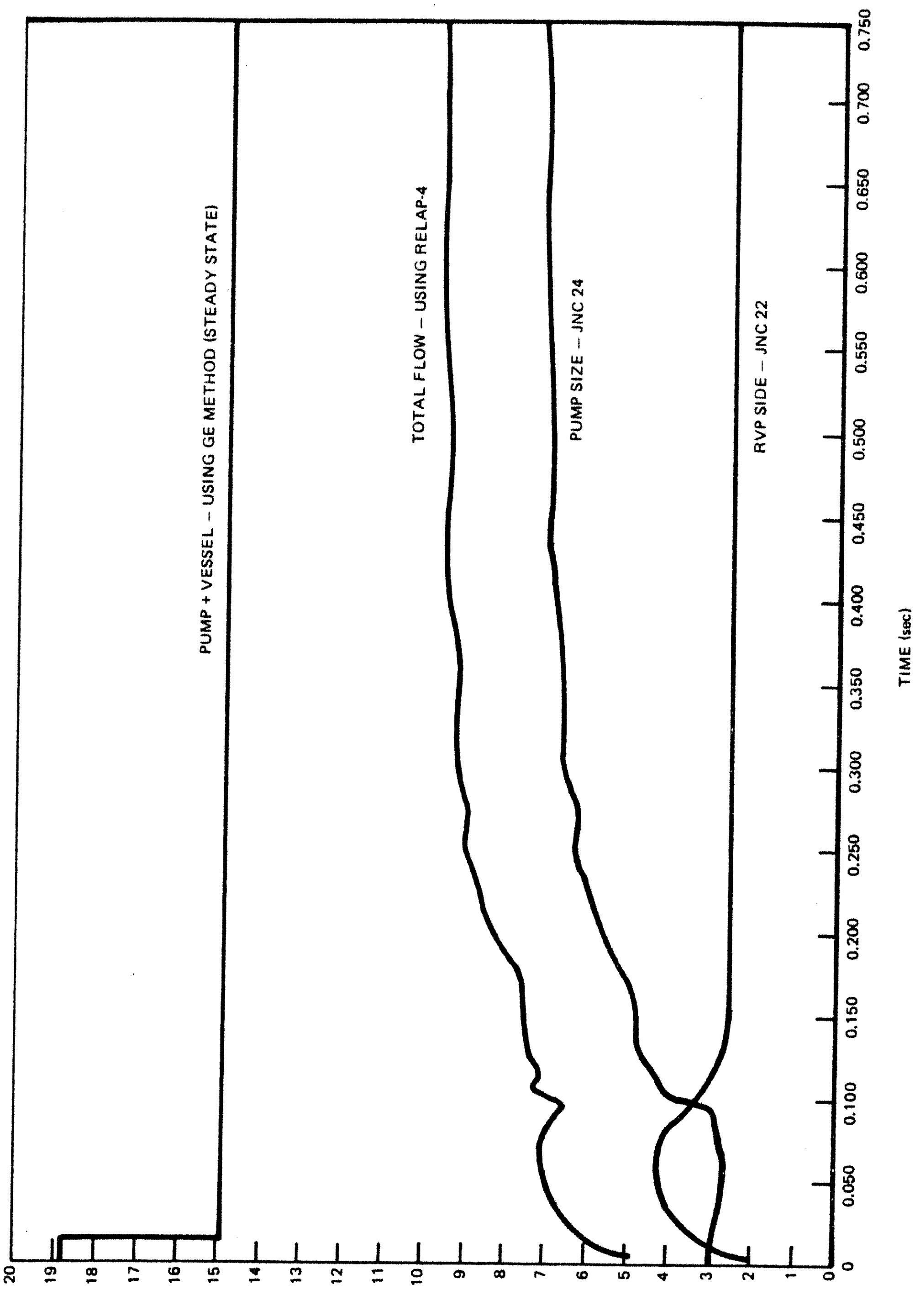

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FIGURE 6.A-12

COMPARISON OF THE GE AND
RELAP4/MOD5 METHODS - FEEDWATER
LINE BREAK, LEG EB

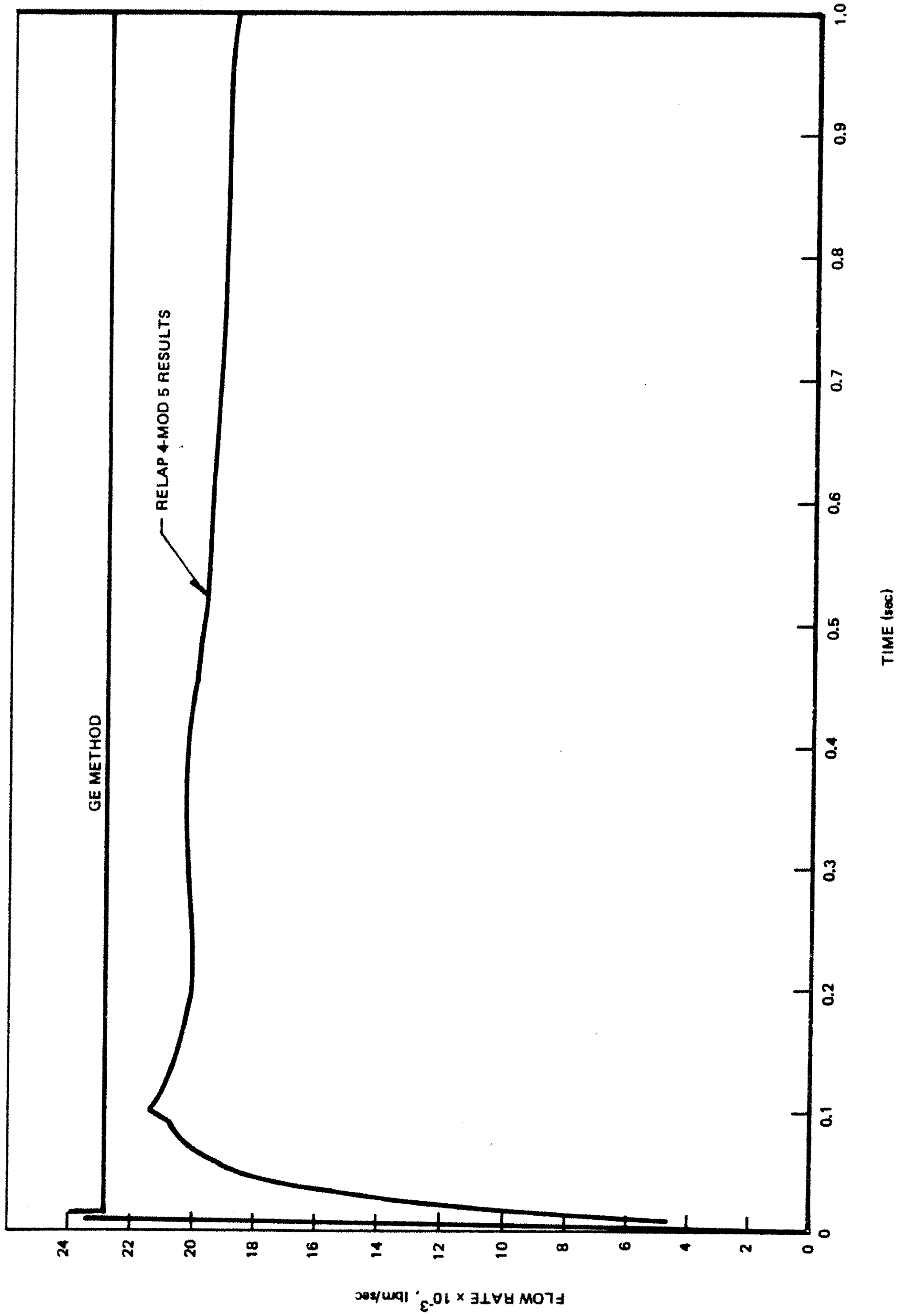

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FIGURE 6.A-13

COMPARISON OF THE GE AND RELAP4/MOD5 METHODS - RECIRCULATION LINE BREAK, FINITE OPENING TIME

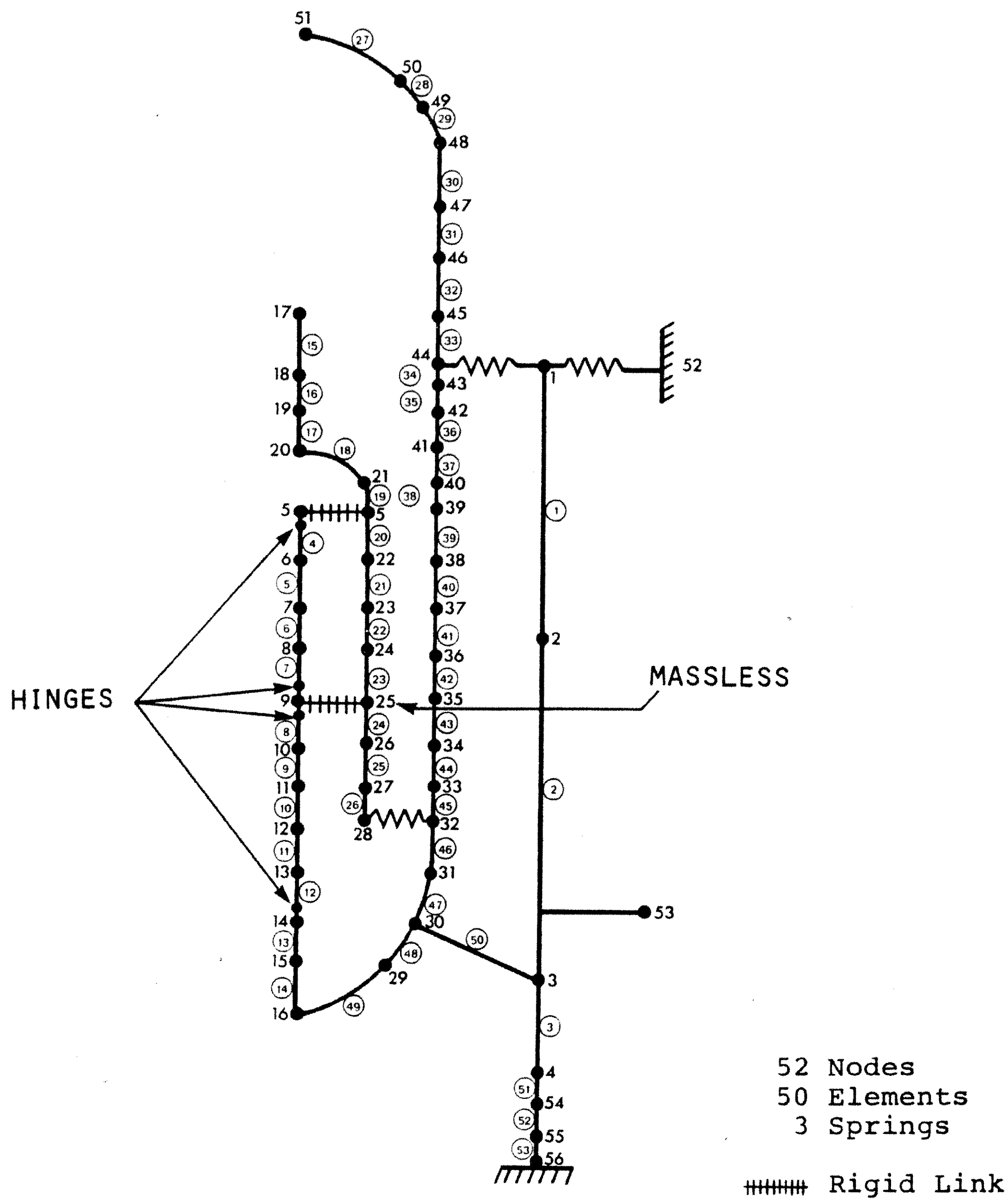

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FIGURE 6.A-14

HORIZONTAL MODEL FOR ANNULUS PRESSURIZATION

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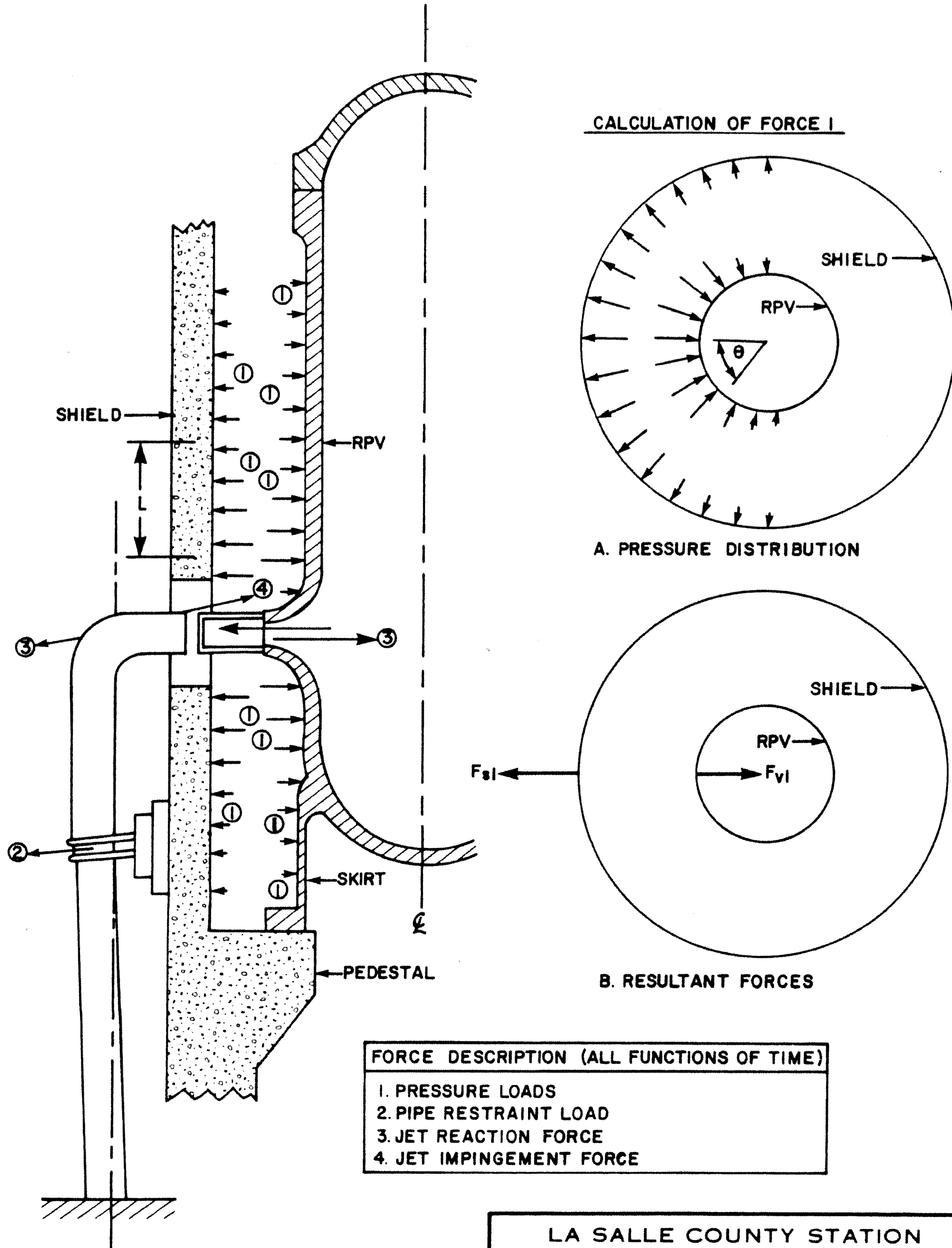

FORCE DESCRIPTION (ALL FUNCTIONS OF TIME)
1. PRESSURE LOADS 2. PIPE RESTRAINT LOAD 3. JET REACTION FORCE 4. JET IMPINGEMENT FORCE

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FIGURE 6.A-15

ANNULUS PRESSURIZATION
LOADING DESCRIPTION

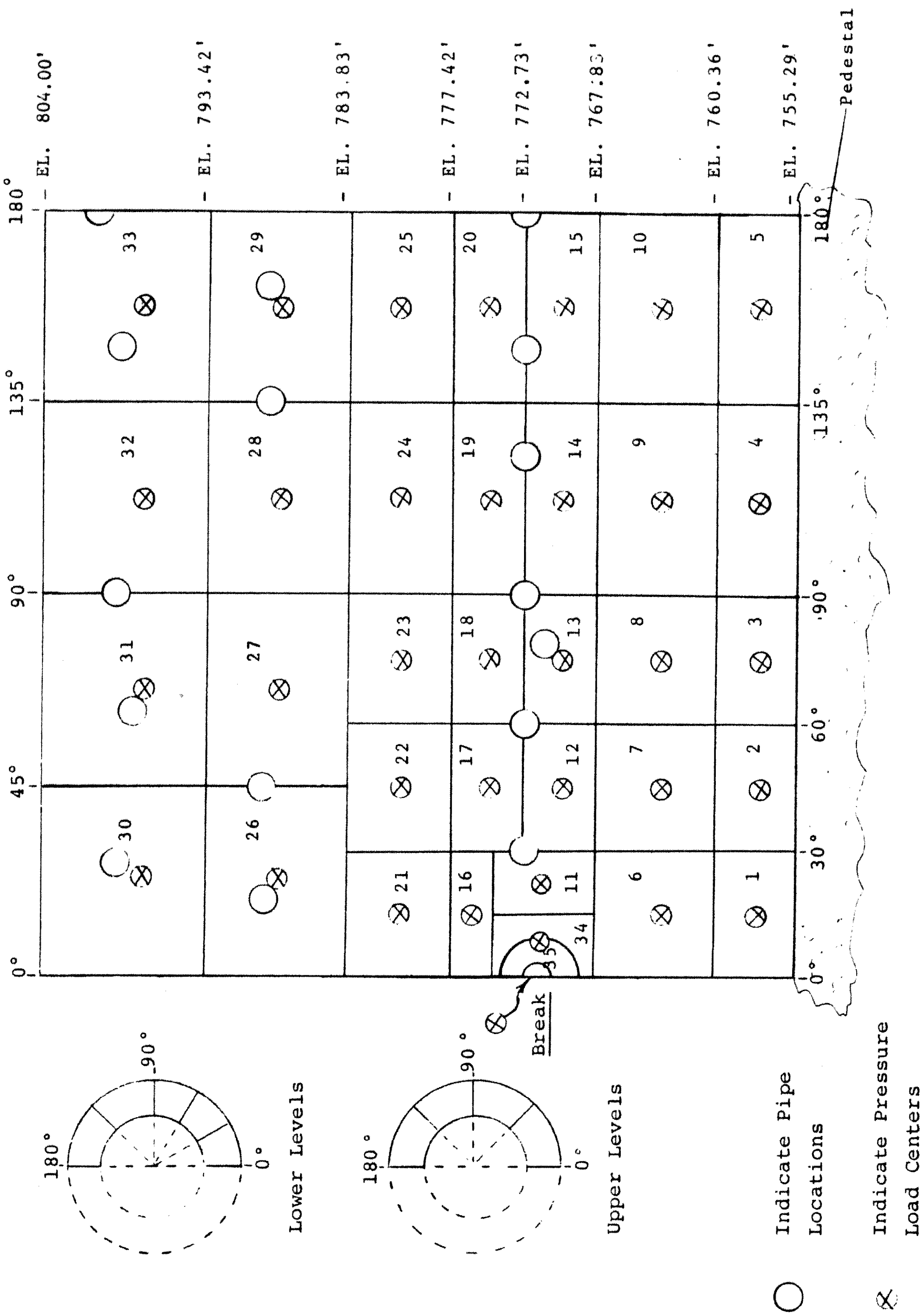

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FIGURE 6.A-16

ANNULAR SPACE NODALIZATION FOR RECIRCULATION LINE BREAK

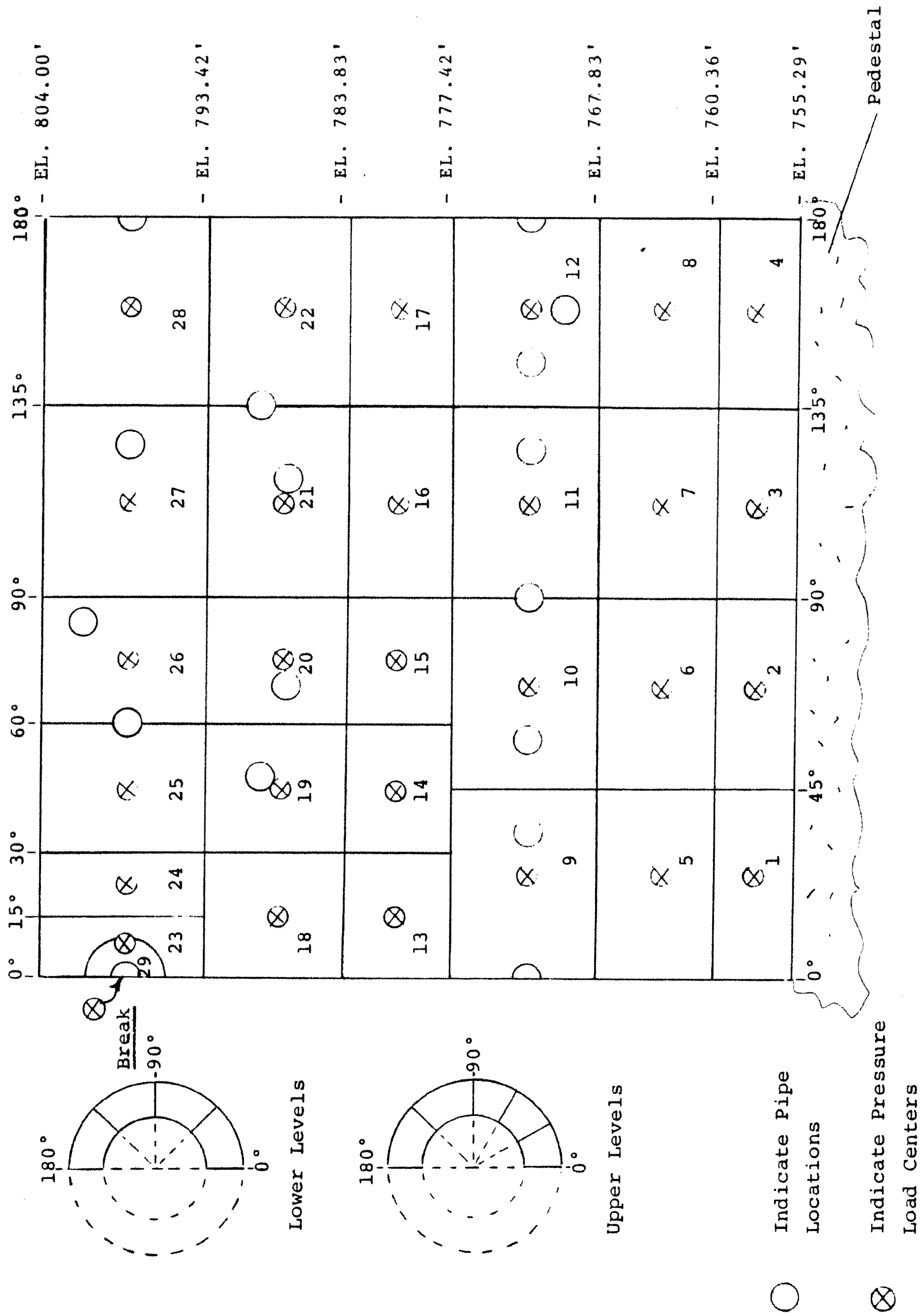

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FIGURE 6.A-17

ANNULAR SPACE NODALIZATION FOR FEEDWATER LINE BREAK

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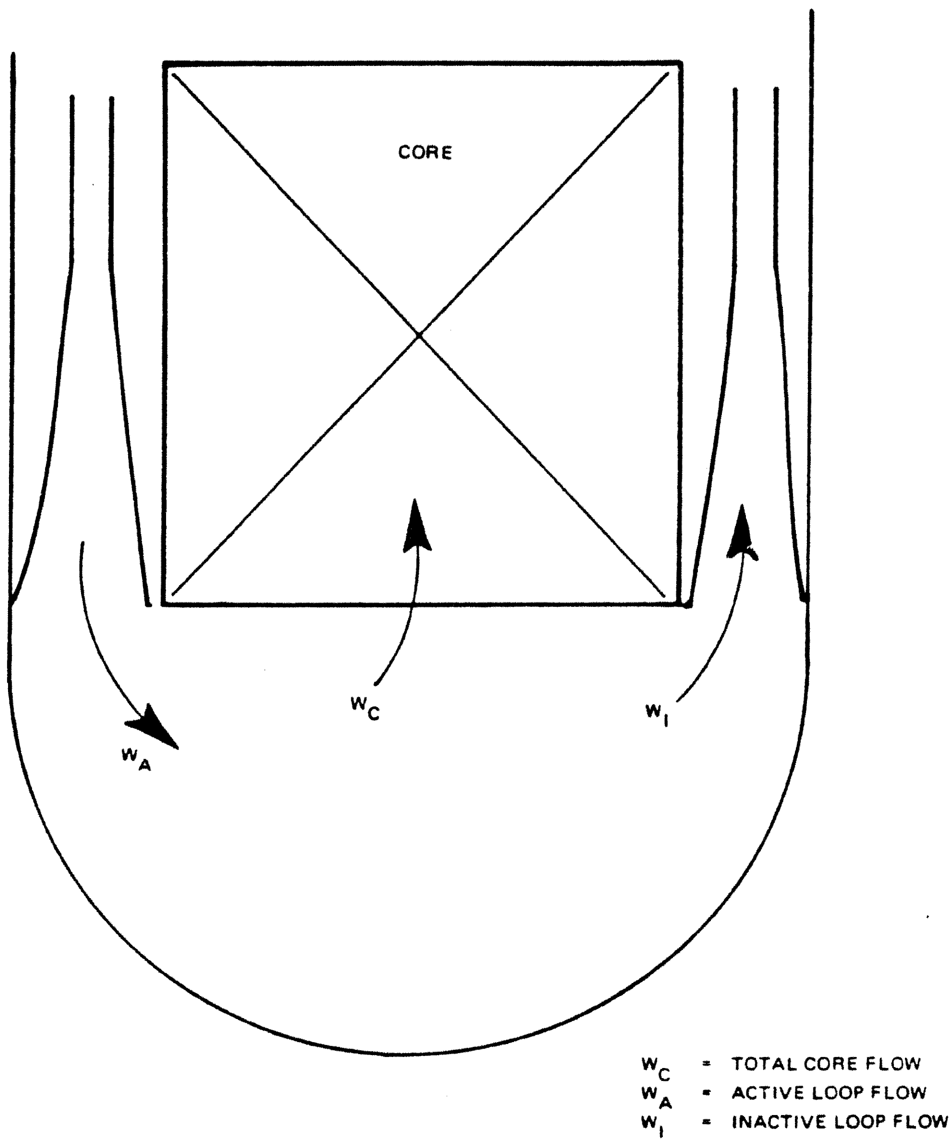

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FIGURE 6.B-1

ILLUSTRATION OF SINGLE RECIRCULATION LOOP OPERATION FLOWS

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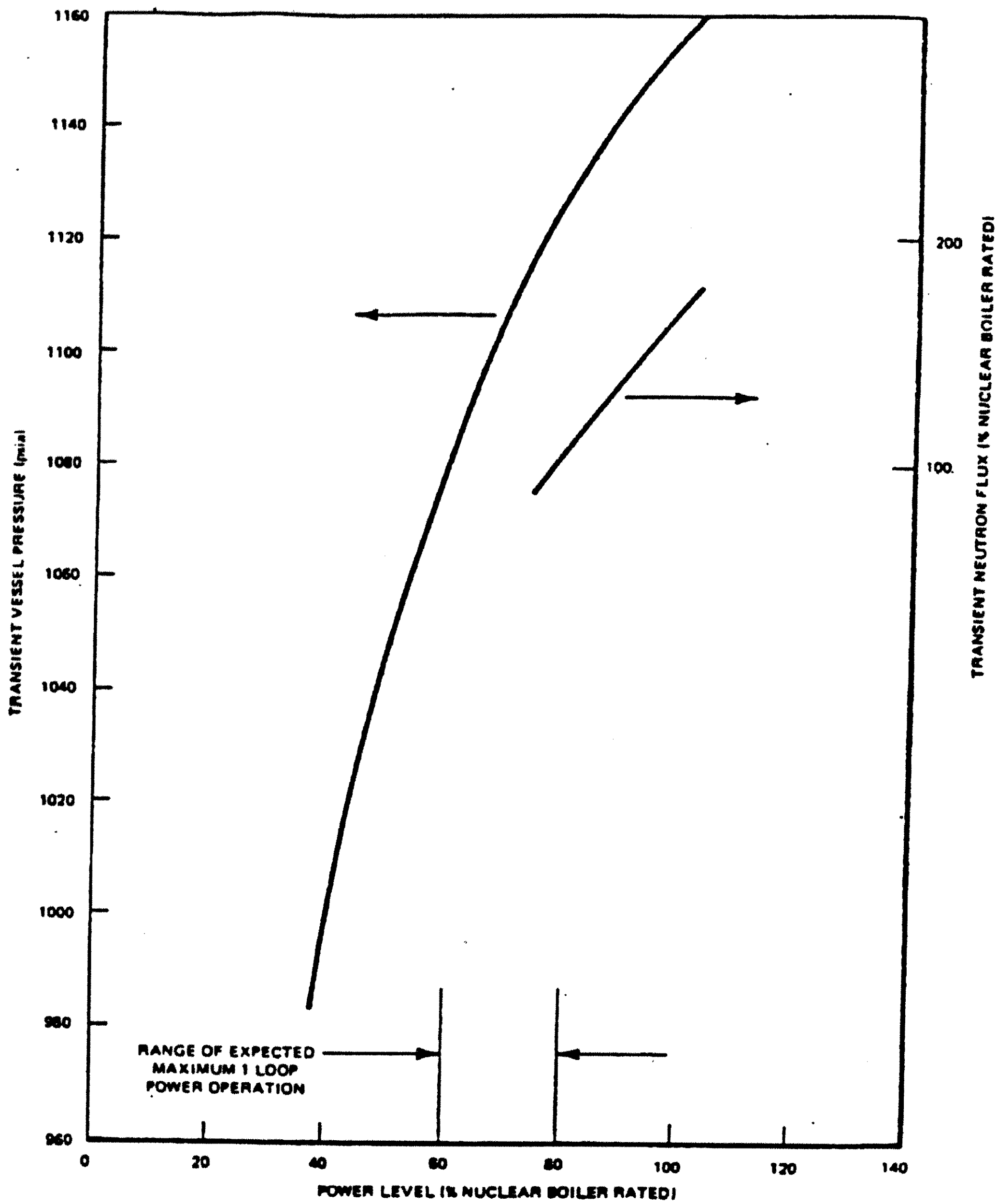

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FIGURE 6.B-2 FEEDWATER CF WITH ONE PUMP OPERATION TYPICAL (GE)

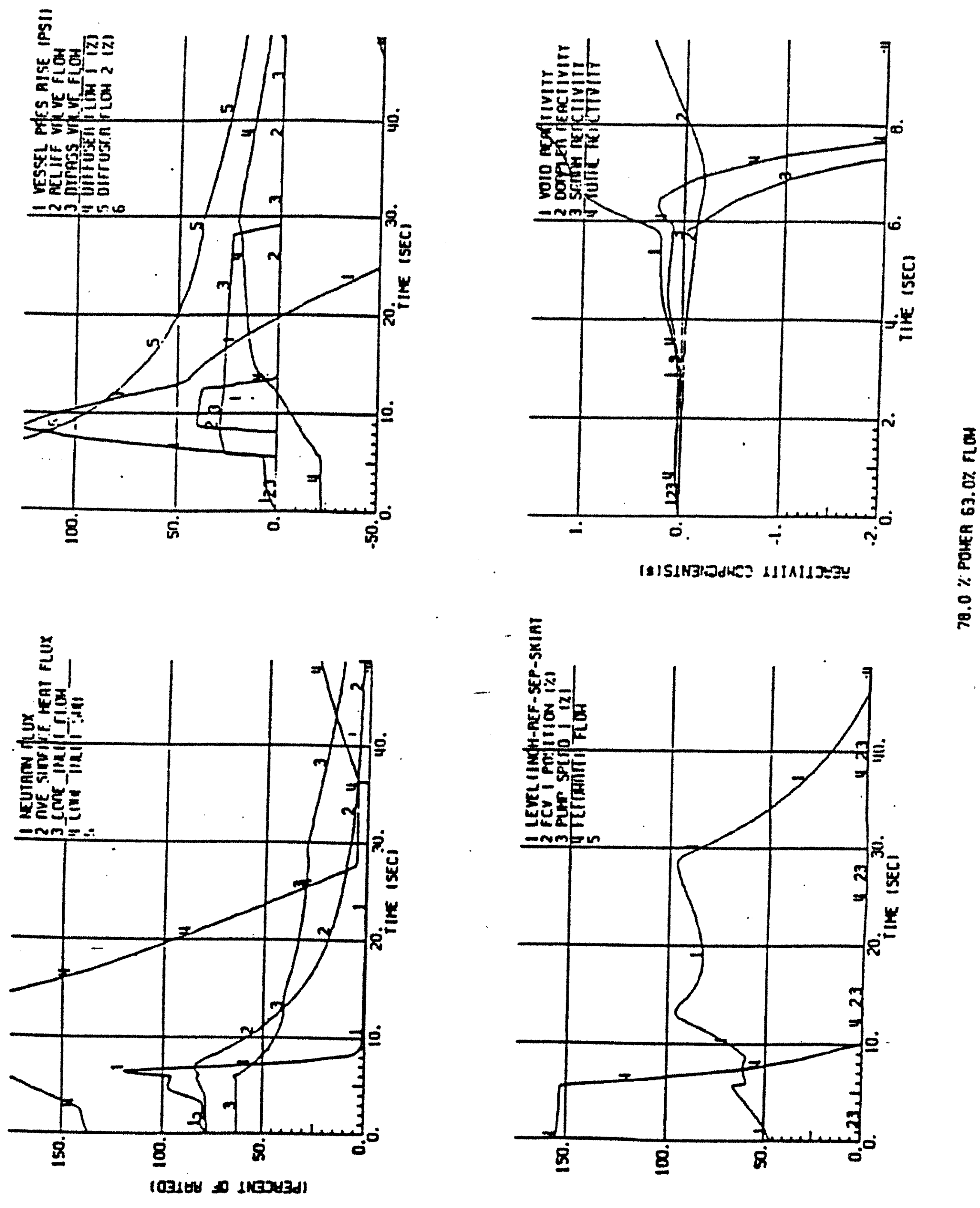

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FIGURE 6.B-3

FEEDWATER CF WITH ONE PUMP OPERATION
TYPICAL (GE)

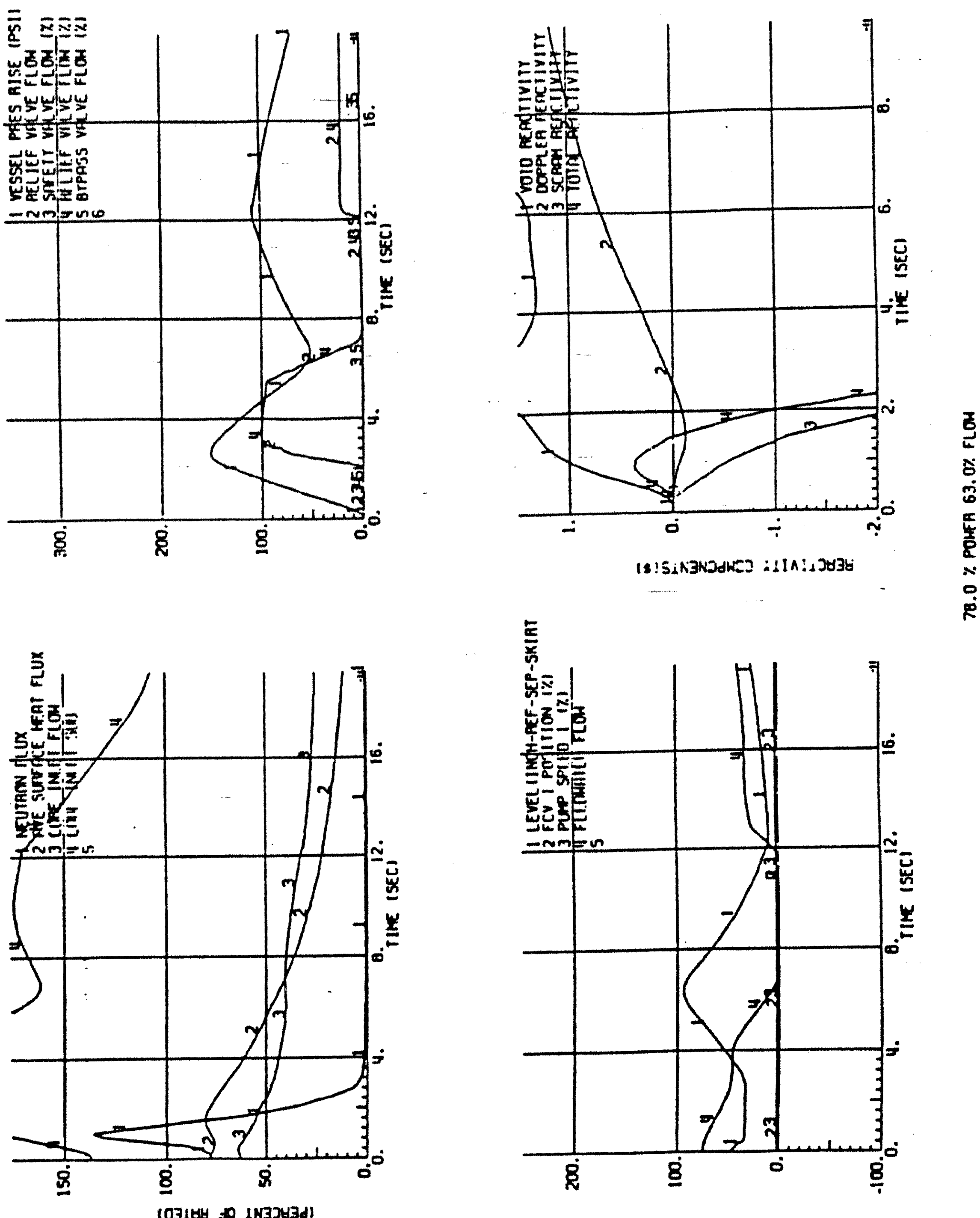

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FIGURE 6.B-4
TYPICAL LOAD REJECTION WITH ONE PUMP OPERATION

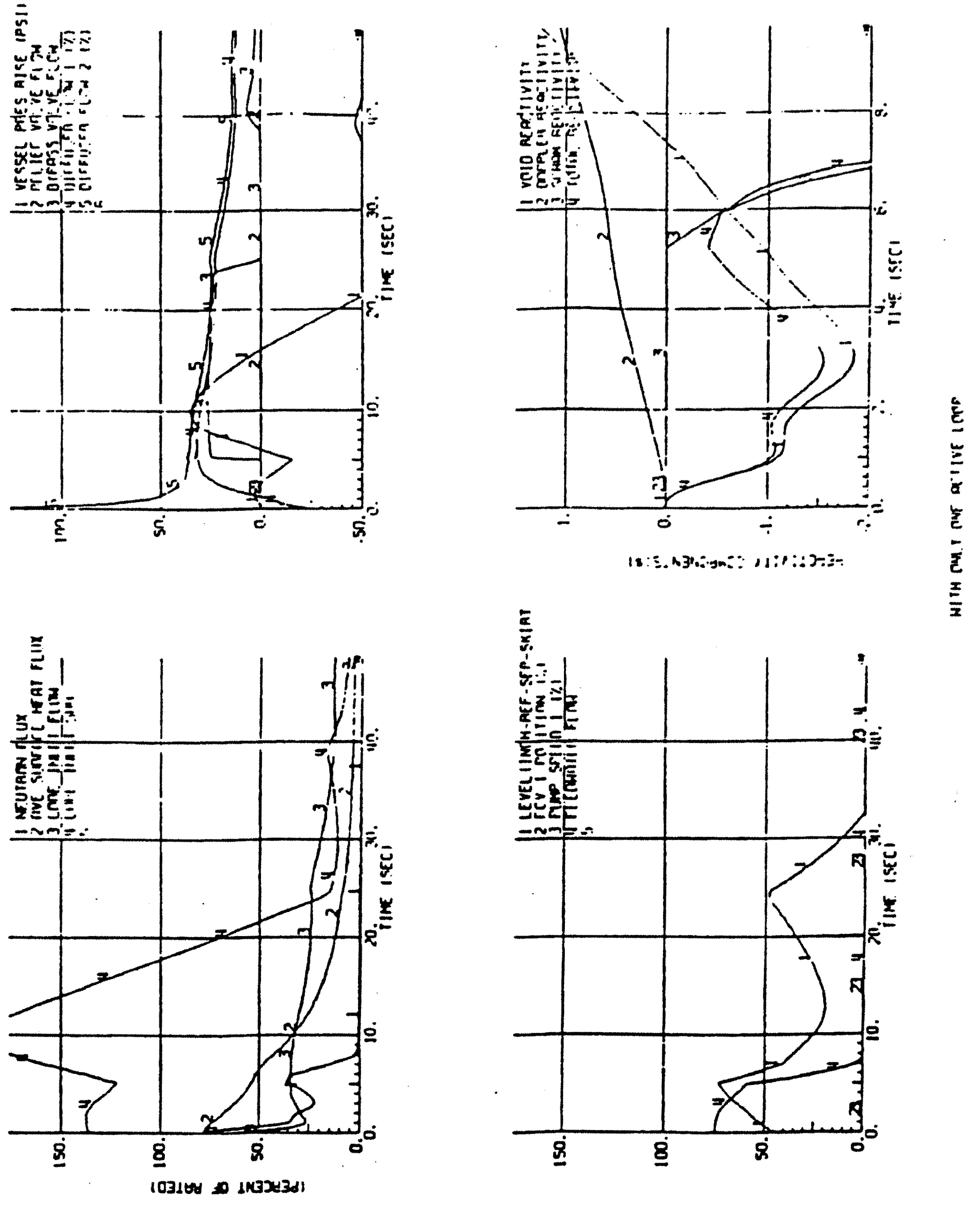

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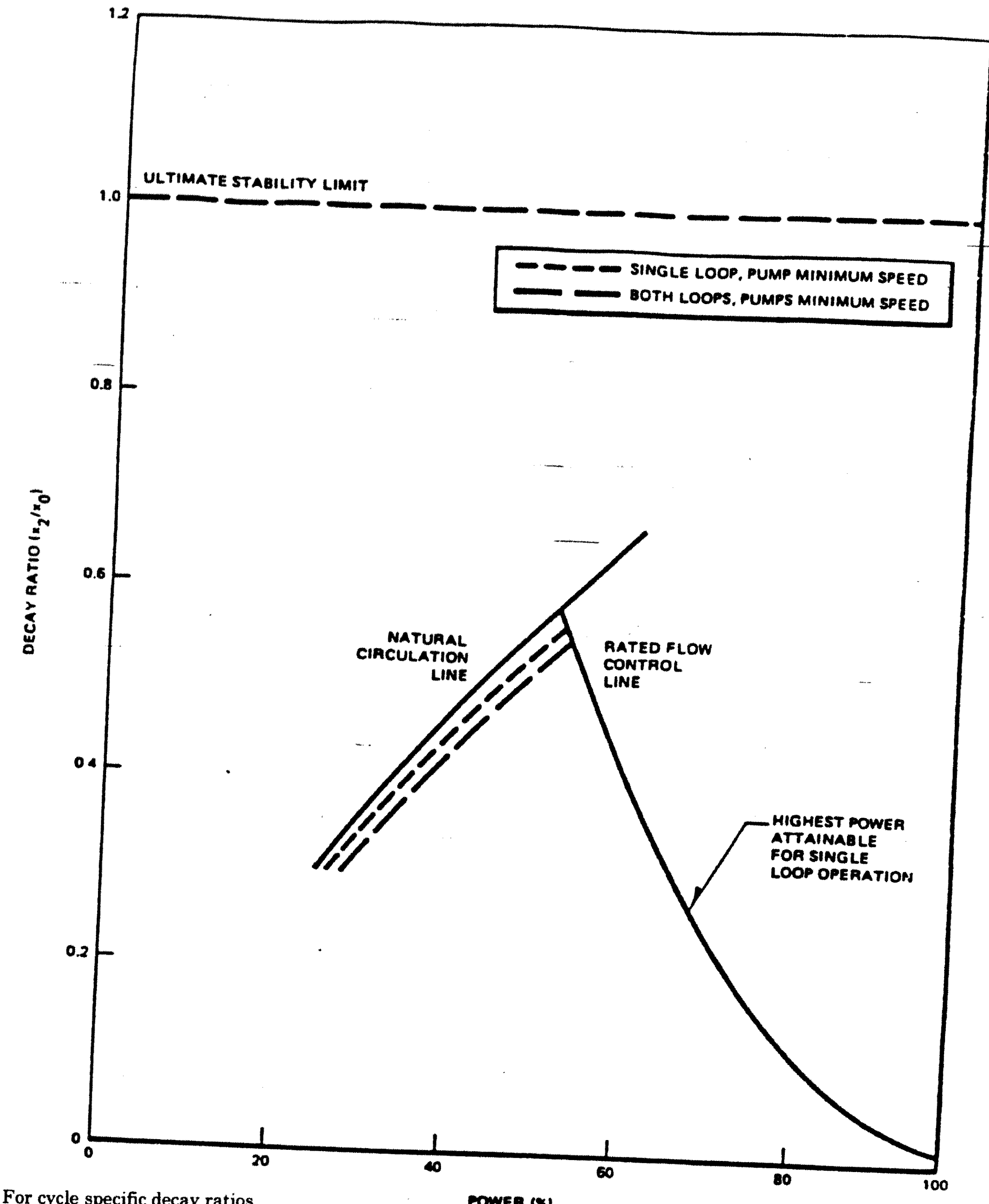
FIGURE 6.B-5

TYPICAL
SEIZURE OF ONE RECIRCULATION PUMP



* For cycle specific decay ratios, SEE the LaSalle Administrative Technical requirements

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FIGURE 6.B-6

Typical, GE

DECAY RATIO VERSUS POWER CURVE FOR TWO-LOOP AND SINGLE-LOOP OPERATION*

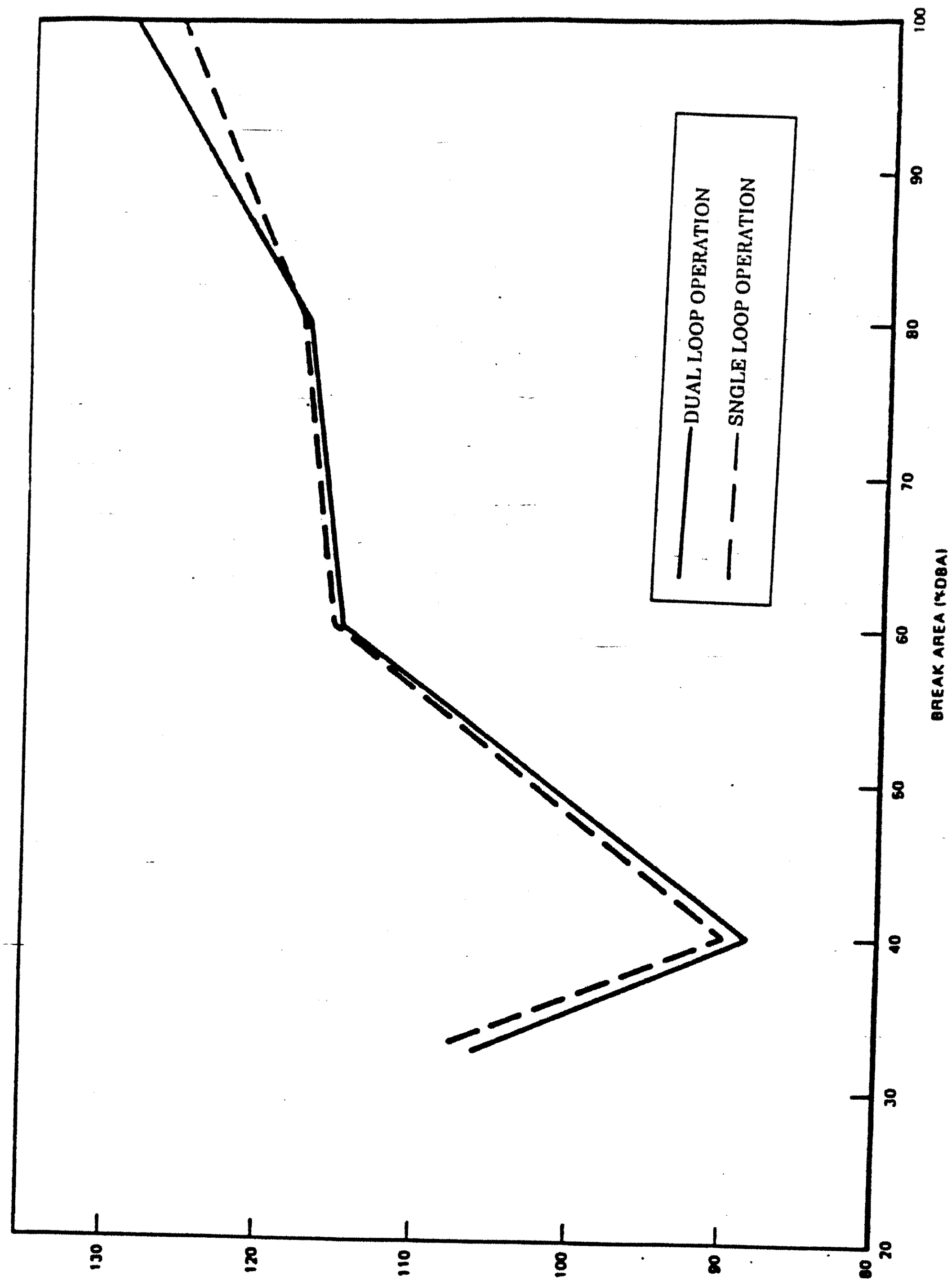

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FIGURE 6.B-7 UNCOVERED TIME VS. BREAK AREA – LASALLE 1 AND 2 SUCTION BREAK LPCS/DG FAILURE

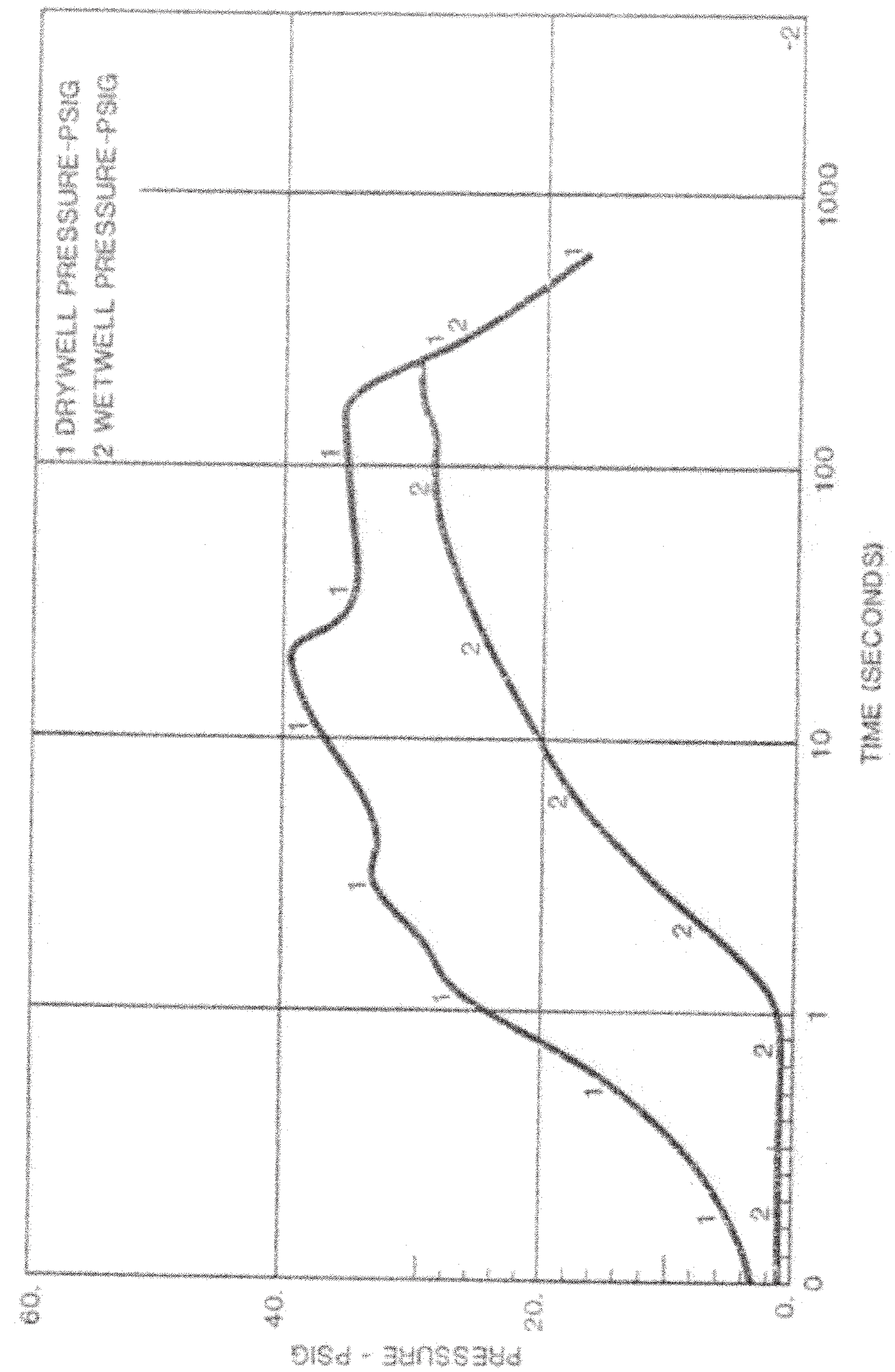

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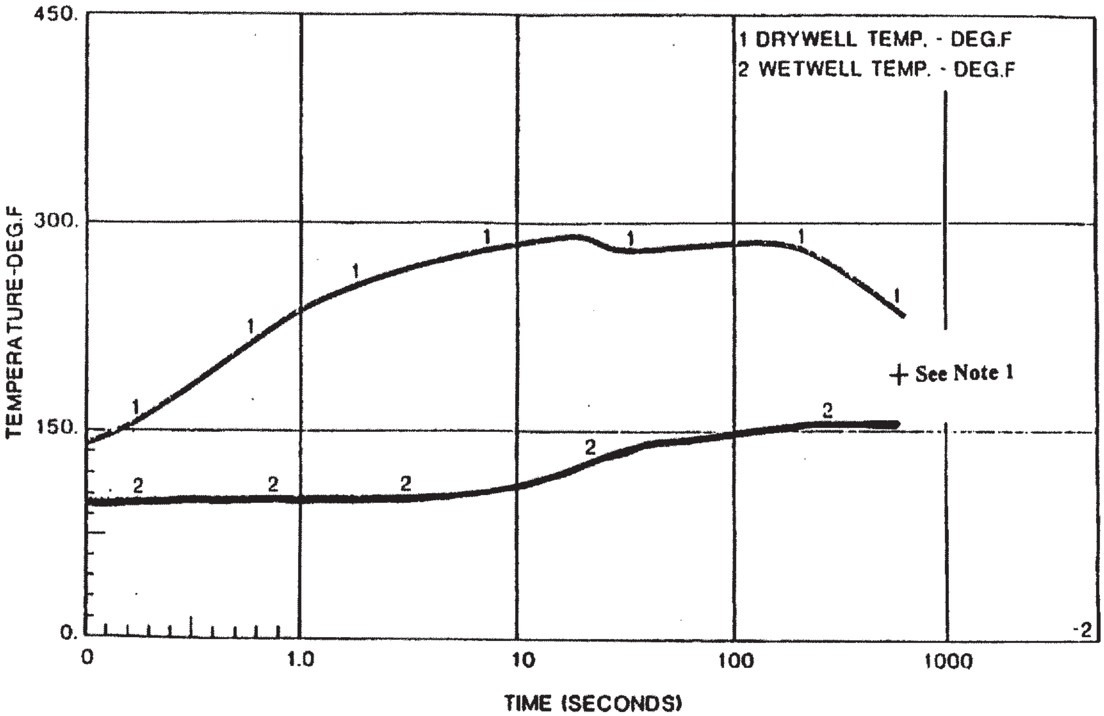
FIGURE 6.C-1
RECIRCULATION LINE BREAK
PRESSURE RESPONSE



Notes: 1. This point represents the projected suppression pool temperature due to the feedwater coastdown/injection. This point is a starting temperature for the assessment of peak long term suppression pool temperature.

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FIGURE 6.C-2 TEMPERATURE RESPONSE FOR RECIRCULATION LINE BREAK

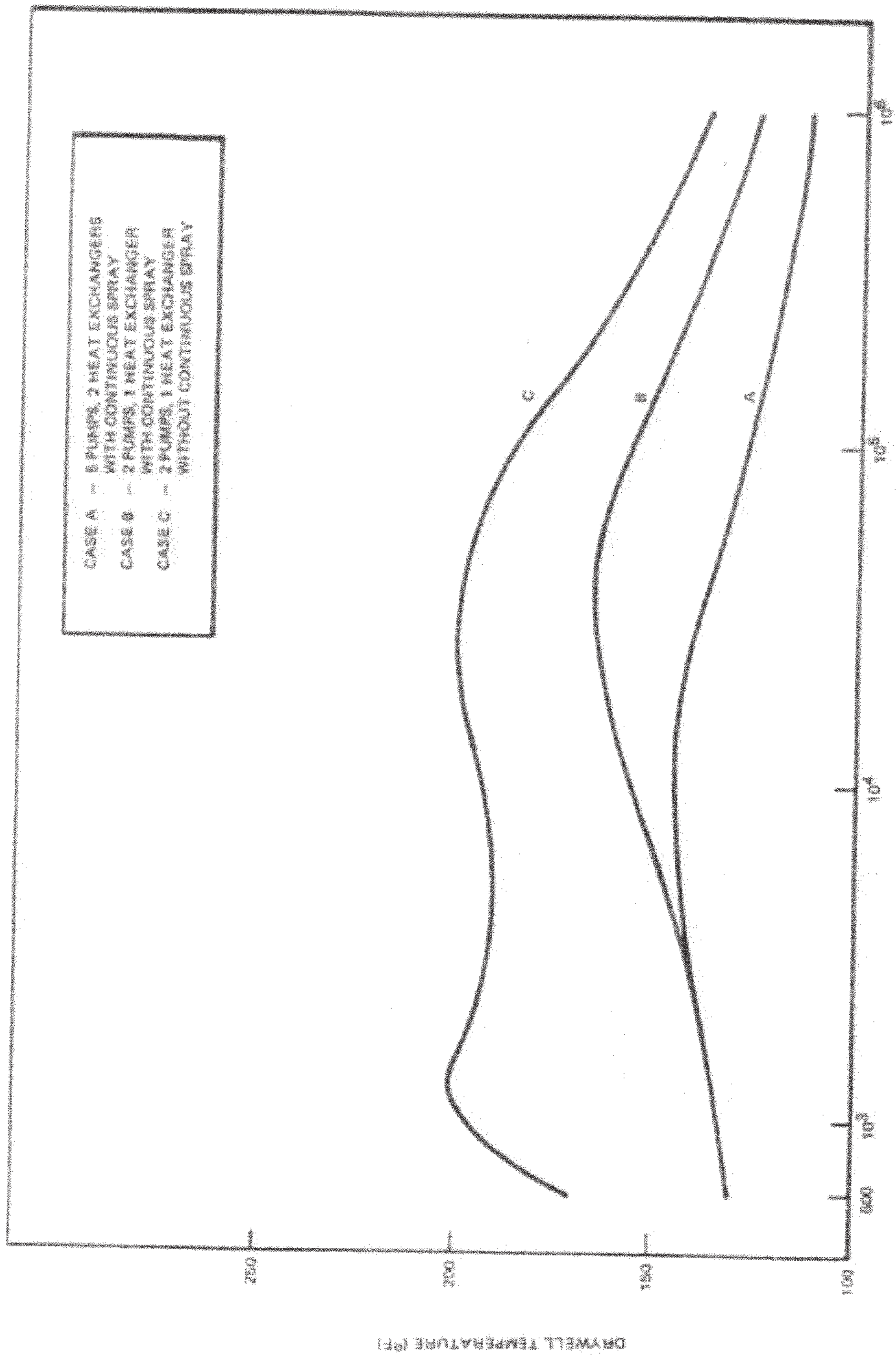

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FIGURE 6.C-3
DRYWELL TEMPERATURE RESPONSE

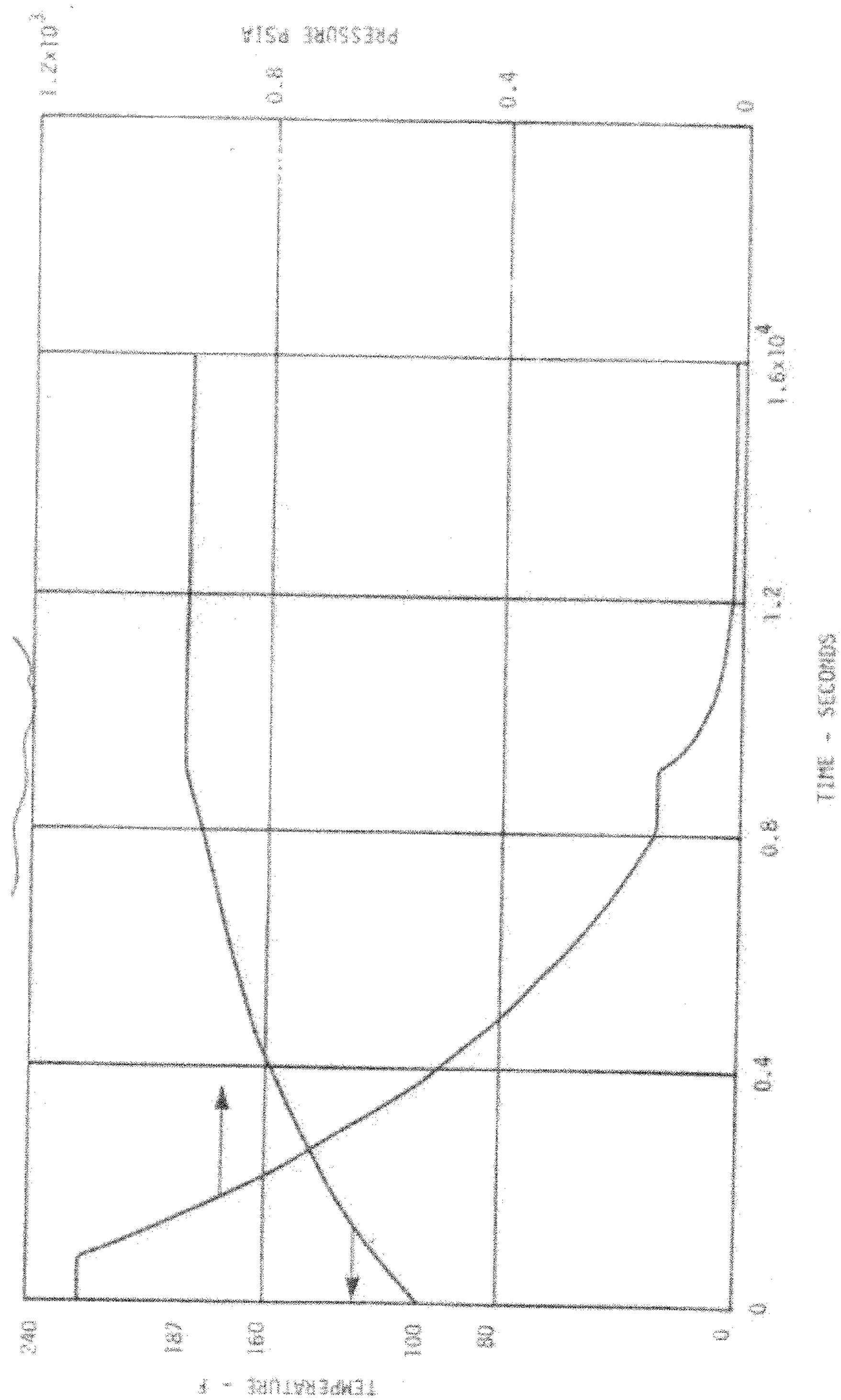

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FIGURE 6.C-4
POOL TEMPERATURE RESPONSE –
ISOLATION/SCRAM, 1RHR AVAILABLE