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DRAWINGS CITED IN THIS CHAPTER*

* 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 USAR. They are controlled by the Controlled Documents Program.

DRAWING*	SUBJECT
762E260AC	Process Diagram MSIV Leakage Control System
762E425AC	Process Diagram Residual Heat Removal
762E454	High Pressure Core Spray
762E467AC	Low Pressure Core Spray
M01-1105	General Arrangement - Basement Floor Plan
M01-1106	General Arrangement - Grade Floor Plan El. 737'-0"
M01-1107	General Arrangement - Mezzanine Floor Plan El. 762'-0"
M01-1108	General Arrangement - Main Floor Plan'
M01-1109	General Arrangement - Miscellaneous Floor Plans
M01-1111	General Arrangement - Sections "C-C", "D-D", and "E-E"
M01-1524	Radiation Shielding Design - Containment Building El 800'-0"
M01-1526	Radiation Shielding Design - Containment Building El 825'-0"
M05-1002	Main Steam
M05-1004	Reactor Feedwater System
M05-1010	Circulating Water System
M05-1012	Cycled Condensate Storage
M05-1032	Component Cooling Water
M05-1037	Fuel Pool Cooling and Cleanup System
M05-1039	Fire Protection
M05-1040	Instrument Air
M05-1045-12	Postaccident Sampling & Analysis System (PASS)
M05-1048	Service Air Piping and Instrumentation Diagram
M05-1052	Shutdown Service Water
M05-1060	Suppression Pool Cleanup System
M05-1065	Breathing Air System
M05-1069	Suppression Pool Cleanup System
M05-1070	Main Steam Isolation Valve-Leakage Control System
M05-1073	Low Pressure Core Spray System
M05-1074	High Pressure Core Spray System
M05-1075	Residual Heat Removal System
M05-1076	Reactor Water Clean-up System
M05-1077	Standby Liquid Control System
M05-1078	Control Rod Drive System
M05-1079	Reactor Core Isolation Cooling System
M05-1080	Fuel Transfer System
M05-1102	Control Room HVAC
M05-1105	Standby Gas Treatment System
M05-1109	Drywell Cooling Chilled Water System
M05-1110	Drywell Purge
M05-1111-1	Containment Building HVAC System
M05-1111-4	Combustible Gas Control System Equipment Room Cooling

CHAPTER 6 - ENGINEERED SAFETY FEATURES

DRAWINGS CITED IN THIS CHAPTER*

DRAWING*	SUBJECT
M05-1117	Station Chilled Water System
M06-1075-6	Containment Spray System, Spray Nozzle Orientation and Spray Header Plan View
M06-1075-11	Containment Spray System - Spray Header Plan View
M27-1305	Containment Building Piping - Elevation 816'-7"
M27-1306	Containment Building Piping - Elevation 803'-3" (Lower Elevation)
M27-1311	Containment Building Piping Plan - Elevations 770'0" to 828'-3"
M27-1312	Containment Building Piping - Section 1-1
M27-1314	Reactor Head Piping Plan Elevation 804'-2-1/4"

CHAPTER 6 - ENGINEERED SAFETY FEATURES

6.0 GENERAL

The engineered safety features of the Clinton Power Station (CPS) are those systems provided to mitigate the consequences of postulated design-basis accidents. The features can be divided into five general groups: containment systems, emergency core cooling systems, habitability systems, standby gas treatment system, and other engineered safety features. The systems in each general group are:

a. Containment Systems:

   Containment Building
   Secondary Containment
   Containment Heat Removal Systems (including the RHR Containment Spray Mode and the RHR Suppression Pool Cooling Mode)
   Containment and Reactor Vessel Isolation System
   Combustible Gas Control System (including the drywell to containment vacuum relief system)
   Suppression Pool Makeup System
   RHR System, Feedwater Leakage Control Mode (FWLC)

b. Emergency Core Cooling Systems:

   High-Pressure Core Spray System
   Low-Pressure Core Spray System
   Automatic Depressurization System
   Low-Pressure Coolant Injection (RHR System)

c. Habitability Systems:
   Control Room HVAC System

d. Standby Gas Treatment Systems

e. Other Systems:

   Overpressurization Protection System
   Main Steam Isolation Valve Leakage Control System
   Reactor Core Isolation Cooling System
   Control Rod Drive Support System
   Control Rod Velocity Limiter

6.1 ENGINEERED SAFETY FEATURE MATERIALS

Materials used in the engineered safety feature (ESF) components have been evaluated to ensure that material interactions will not occur that could potentially impair operation of the ESF. Materials have been selected to withstand the service conditions, environmental conditions, and radiation levels encountered during normal operation and any postulated accident.

Coatings used on exterior surfaces, within the primary containment, are suitable for the environmental conditions expected.

6.1.1 Metallic Materials

6.1.1.1 Materials Selection and Fabrication

6.1.1.1.1 Material Specifications

Table 5.2-4 lists the principal pressure retaining materials and the appropriate material specifications for the reactor coolant pressure boundary components. Table 6.1-1 lists the principal pressure retaining materials and the appropriate material specifications for the engineered safety features of the plant.

Pressure retaining components 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 ESF, in general, have been designed with corrosion allowances, in compliance with the general requirements of Section III of the ASME B&PV Code, Article NC-3120.

6.1.1.1.2 Compatibility of Construction Materials with Core Cooling Water and Containment Sprays

Subsection 5.2.3.2.3 discusses compatibility of the reactor coolant with materials of construction exposed to the reactor coolant. These same materials of construction are used for the engineered safety feature components.

Demineralized water, with no additives, is used in BWR core cooling water and containment sprays. No detrimental effects will occur on any of the ESF construction materials from the allowable contaminant levels in this high purity water.

6.1.1.1.3 Controls for Austenitic Stainless Steel

a. Controls of the Use of Sensitized Stainless Steel:

   Controls to avoid severe sensitization are discussed in Subsection 5.2.3.4.1.1. General compliance or alternate approach assessment for Regulatory Guide 1.44 may be found in Subsection 5.2.3.4.1.2.

b. Process Controls to Minimize Exposure to Contaminants:

   Process controls for austenitic stainless steel are discussed in Subsection 5.2.3.4.1.2. General compliance or alternate approach assessment for Regulatory Guide 1.44 may also be found in Subsection 5.2.3.4.1.2.

c. Use of Cold Worked Austenitic Stainless Steel:

   Austenitic stainless steel with a yield strength greater than 90,000 psi was not used in ESF systems.

d. Thermal Insulation Requirements:

Nonmetallic thermal insulation materials in ESF systems are required to have the proper ratio of leachable sodium plus silicate ions to leachable chloride plus fluoride ions. General compliance or alternate approach assessment for Regulatory Guide 1.36 may be found in Subsection 5.2.3.2.4.

e. Avoidance of Hot Cracking of Stainless Steel:

Process controls to avoid hot cracking of stainless steel are discussed in Subsection 5.2.3.4.2.1. General compliance or alternate approach assessment for Regulatory Guide 1.31 may also be found in Subsection 5.2.3.4.2.1.

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

Demineralized water, with no additives, is used in the core cooling water and containment sprays. No detrimental effects will occur on any of the ESF materials from this high purity water.

6.1.2 Organic Materials

6.1.2.1 Protective Coatings

The use of organic protective coating within the containment is kept to a minimum. Table 6.1-2 lists the organic compounds that exist within the containment. These materials in or on 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.

Exposed carbon steel surfaces inside the containment are to be coated with an inorganic zinc primer which has been qualified in accordance with ANSI Standards N101.2, N101.4 (Regulatory Guide 1.54), and ANSI-N512. Inorganic zinc paint, unlike organic paint, will not radiolytically or pyrolytically decompose or otherwise interact with other ESF materials in the containment.

Certain structural items and mechanical components are painted with organic coatings. These items are typically painted with coatings that are in accordance with Regulatory Guide 1.54 and applicable ANSI Standards. Some items are protected with unqualified coatings. The items are typically small size equipment such as electrical/electronic trim, covers, face plates, valves, valve handles, etc.

Hydrogen generation from corrosion of zinc primers is included in Subsection 6.2.5. However, hydrogen produced from thermal, chemical, and radiolytic decomposition of organic coatings as well as from other organic materials will be insignificant compared with other sources. The significant sources of hydrogen are addressed in Subsection 6.2.5. These sources are zircaloy-water reaction and radiolysis of the large volume of water. The small quantity of hydrogen from organics will not adversely affect the ESF.

In containment, the volume of solid debris that can be formed from all unqualified organic materials (including unqualified organic coatings) and reach the containment sump under DBA conditions is minimal. This is because typically the exceptions to using qualified organic coatings are limited to small equipment.

Solid debris from within the drywell could also reach the containment ECCS suction lines in the suppression pool. The only part of the suppression pool within the drywell is the annular region enclosed by the weir wall. Transport of debris to the suppression pool can be accomplished only through the vents.

Solid debris formed in containment outside the drywell would not present a problem. Even if all the debris reached the suppression pool it would not pose a problem to the ECCS suction lines because of the following factors.

1. The large volume of the pool (135,700 ft^3 at low water level, see Note ** for item A.7 in Table 6.2-1).

2. The low-design inlet velocity of the suction strainers (less than 0.02 ft/sec).

3. The ECCS system suction strainer is designed to prevent passage of particles over 3/32 inch in diameter which has been determined to be sufficient to prevent damage to or clogging of the ECCS or the components which they serve.

4. The suction lines are designed such that adequate NPSH and flow are available for the ECCS pumps even if the strainer is fully loaded (i.e., conservatively specified debris loading resulting from LOCA-generated and pre-LOCA debris materials). (Q&R 281.2)

See subsection 6.2.2.2 for a discussion of the methodology used for determining debris generation and the treatment of miscellaneous debris such as paint.

6.1.3 References

1. NUREG/CR-2726, "Light Water Reactor Hydrogen Manual," August 1983.

2. Letter from Roger J. Mattson, Director, NRC-DSS, to R.S. Boyd, "Post-LOCA Hydrogen Production from Material Inside Containment," October 17, 1978.

TABLE 6.1-1
PRINCIPAL PRESSURE-RETAINING MATERIAL
FOR ESF SYSTEM COMPONENTS

I. Containment Systems	
Containment Walls	4000 psi concrete
Containment Liner	SA-516, Grade 60
Suppression Pool Liner	SA-240, Type 304
Drywell Head	SA-240, Type 304
Drywell Head Bolting	SA-193, Grade B6
Containment Penetration Sleeves	SA-333, Grade 1 SA-333, Grade 6 SA-516, Grade 60 SA-516, Grade 70 SA-312, Type 304 SA-240, Type 304
Hatches (Equipment and Personnel)	SA-516, Grade 70
II. Secondary Containment	
Walls	3500 psi concrete
Doors	Cold-rolled carbon steel with galvanizing or rust inhibitor
Metal Sidings	A525 (or as per specification K-2950)
Dampers	A446, A526, or A527
Ducts	A526 or A527, and A36
III. Containment Heat Removal Systems	
RHR Pumps	
Bowl Assembly	A216, Grade WCB
Discharge Head Shell	SA-516, Grade 70
Discharge Head Cover	SA-105
Suction Barrel Shell and Dished Head	SA-516, Grade 70
Flanges	SA-105
Pipe	SA-106, Grade B
Shaft	A-276, Type 410 Condition H
Impeller	A-351, Grade CA6NM
Studs	SA-193, Grade B7
Nuts	SA-194, Grade 7
Cyclone Separator Body and Cover	SA-479, Type 304
RHR Heat Exchanger	
Shell, Head and Channel	SA-516, Grade 70
Tubesheet	SA-516, Grade 70
Nozzles	SA-105
Flanges	SA-105
Tubes	SA-249, Type 304L
Bolts	SA-193-B7
Nuts	SA-194, Grade 7
Piping	SA-106, Grade B SA-155, Grade KC65 C1.1 SA-312, Grade Type 304 SA-376, Grade Type 304 SA-358, Grade 304 C1.1 SA-333, Grade 6

TABLE 6.1-1 (Cont'd)

	SA-312, Grade Type 316L
	SA-376, Grade Type 316L
	SA-213, Grade Type 304
	SA-213, Grade Type 316L
	SA-358, Grade 316, Class 1 with 0.035% max carbon
Valves	SA-105
	SA-216, Grade WCB or WCC
	SA-182, Grade F304
	SA-479, Type 304 or 316
	SA-350, Grade LF-2
	SA-352, Grade LCB
	SA-351, Grade CF8
ECCS/RCIC Suction Strainer	SA-240, Type 304
	SA-182, Grade F304 or F316
	SA-312, Grade Type 304
	SA-193, Grade B8 C1 II
	SA-194, Grade 8
	SA-403, Grade WP304
	A693, Type 631
Spray Nozzles	SA-351, Grade CF8
IV. Containment Isolation System	
Piping	SA-106, Grade B
	SA-155, Grade KC65
	SA-312, Grade Type 304
	SA-376, Grade Type 304
	SA-358, Grade 304
	SA-333, Grade 6
Valves	SA-105, SA-479 Type 316 or WCC
	SA-216, Grade WCB
	SA-182, Grade F304
	SA-351, Grade CF8
V. Combustible Gas Control System	
Hydrogen Recombiner	304 SS*, 321 SS*,Carbon Steel*
Piping	SA-106, Grade B
	SA-312 or 376, Grade TP304
Valves	SA-105
Compressors	SME Approved Carbon Steel
VI. Suppression Pool Makeup System	
Piping	SA-106, Grade B
	SA-155, Grade KC65
	SA-312, Grade Type 304
	SA-376, Grade Type 304
	SA-358, Grade Type 304
Valves	SA-105, SA-182
	SA-216, Grade WCB, SA-351
	SA-216, Grade WCC
	SA-479, Type 316

* These materials are documented in Rockwell International Stress Report N139SR220006, Table 1.

TABLE 6.1-1 (Cont'd)

VII. Drywell to Containment Vacuum Relief System	
Vacuum Relief Valves	SA-106, Grade B SA-350, Grade LF-1
Piping	SA-106, Grade B
Emergency Core Cooling System	
VIII. High-Pressure Core Spray System	
HPCS Pump	
Bowl Assembly	A-216, Grade WCB
Discharge Head Shell	SA-516, Grade 70
Discharge Head Cover	SA-105
Suction Barrel Shell and Dished Head	SA-516, Grade 70
Flanges	SA-105
Pipe	SA-516, Grade 70
Shaft	A-276, Type 410 Condition H
Impeller	A-351, Grade CA6NM
Studs	SA-193, Grade B7
Nuts	SA-194, Grade 7
Cyclone Separator Body and Cover	SA-479, Type 304
Piping	SA-106, Grade B SA-312, Grade Type 304
Valves	SA-479, Type 316 SA-105 SA-216, Grade WCB, WCC SA-182, Grade F304 SA-351, Grade CF8
Suction Strainer	See Item III
IX. Low-Pressure Core Spray System	
LPCS Pump	
Bowl Assembly	A-216, Grade WCB
Discharge Head Shell	SA-516, Grade 70
Discharge Head Cover	SA-105
Suction Barrel Shell and Dished Head	SA-516, Grade 70
Flanges	SA-105
Pipe	SA-106, Grade B
Shaft	A-276, Type 410 Condition H
Impeller	A-351, Grade CA6NM
Studs	SA-193, Grade B7
Nuts	SA-194, Grade 7
Cyclone Separator Body and Cover	SA-479, Type 304
Piping	SA-312, Grade TP304 SA-312, Grade TP316L SA-213, Grade TP316L SA-358, Grade 304, Class 1 SA-213, Grade TP304 SA-106, Grade B

TABLE 6.1-1 (Cont'd)

Valves	SA-105 SA-216, Grade WCB, WCC SA-479, Type 304 or 316 SA-352, Grade LCB SA-350, Grade LF-2 SA-182, Grade F304 SA-351, Grade CF8
Suction Strainer	See Item III
X. Automatic Depressurization System	
Safety Relief Valves	SA-350, Grade LF-2 SA-352, Grade LCB SA-351, Grade CF3A
Accumulator Tanks Instrument Air Piping	SA-312, Grade Type 304 SA-376, Grade Type 304
SRV Discharge Piping	SA-358, Grade 304,Class 1 SA-106, Grade B
XI. Low-Pressure Coolant Injection	
RHR Pumps	See Item III
Valves	See Item III
Suction Strainer	See Item III
Standby Gas Treatment System	
Filter Housing	A-36
Fan Housing	Carbon Steel
Piping	A-106, Grade B
Valves	A-105 A-216, Grade WCB
Charcoal Adsorbers	A-167, Type 304 A-240, Type 304
Dampers	A-36, coated to A164-RS
Habitability System	
XII. Control Room HVAC Makeup Filter System	
Filter Housing	A-36
Fan Housing	Carbon Steel
Valves	A-105 A-216 Grade WCB
Charcoal Absorbers	A-167, Type 304 A-240, Type 304
Dampers	A-446, A-527 or A-526 A-525-G90 (coating)
Other Systems	
XIII. Overpressurization Protection System	
Safety Relief Valve	SA-350, Grade LF2 SA-352, Grade LCB SA-351, Grade CF3A
Piping	SA-358, Grade 304, Class 1 SA-106, Grade B
XIV. Main Steam Isolation Valve Leakage Control System	
Blowers Housing	N/A

TABLE 6.1-1 (Cont'd)

Piping	SA-106, Grade B SA-312, Grade TP304 SA-376, Grade TP304
Valves	SA-105 A-182 Grade F304 or Grade F316
XV. Reactor Core Isolation Cooling System	
Piping	SA-106, Grade B SA-333, Grade 6 SA-155, Grade KC65 SA-312, Grade Type 304 SA-376, Grade Type 304 SA-358, Grade Type 304
Valves	SA-105, SA479, Type 316 SA-216, Grade WCB, WCC SA-182, Grade F304 SA-351, Grade CF8
Suction Strainer	See Item III
XVI. Control Rod Velocity Limiter	A351, Grade CF8, CF3
<u>Standby AC Power System</u>	
XVII. Fuel Oil Piping	SA-106, Grade B
Engine Crank Shaft	Forged Carbon Steel, Heated Treated
Cylinders	Cast Iron
Piping	Carbon Steel
XVIII. Shutdown Service Water System	
Pumps	SA-216, Grade WCB
Piping	SA-106, Grade B SA-155, Grade KC65
Valves	SA-216, Grade WCB, WCC SA-105, SA479, Type 316
Strainers	304 SS

TABLE 6.1-2
ORGANIC MATERIALS WITHIN THE
PRIMARY CONTAINMENT

MATERIALS	USE	QUANTITY
Chlorosulfinated Polysethylene (Hypalon)	Low Voltage and Medium Voltage Electrical Power Cable Jacketing Material Control Cable and Instrumentation Cable Jacketing Material	Throughout drywell
Etylene Propylene Rubber (EPR)	Low Voltage and Medium Voltage Electrical Power Cable Jacketing Material Control Cable and Instrumentation Cable Jacketing Material	Throughout drywell
Cross-Linked Polyolefin/Polymer LD	Instrumentation Coaxial and Triaxial Insulation/Jacketing Material	Throughoutdrywell
Epoxy-Polyamide Finish	Coating for Exposed Carbon Steel Surfaces	Approx. 6,400 lb
Modified Epoxy-Polyamide Surfacer	Coating for Exposed Concreet Surfaces	Approx. 11,900 lb
Epoxy-polyamide Finish	Coating for Exposed Concrete Surfaces	Approx. 2,800 lb
Lube Oil	Reactor Recirculation Pump Motor (2 motors)	Approx. 110 gal.
Firquel 220/or Firquel EHC (Stauffer)	Recirculation Control Valve Hydraulic Fluid (2 valves)	80 gal. per valve

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 USAR.

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. The containment and drywell structure has the capability to withstand the peak transient pressures and temperatures that could occur due to any postulated loss-of-coolant accident (LOCA) as discussed in Subsection 6.2.1.1.3. The LOCA includes the worst single failure (which leads to maximum containment and drywell pressure and temperature) and is further postulated to occur simultaneously with loss of offsite power and a safe shutdown earthquake (SSE).

b. The containment and drywell have the capability to maintain their functional integrity indefinitely after any postulated LOCA.

c. The containment system and drywell will withstand coincident fluid jet forces associated with the flow from the postulated rupture of any pipe within the containment or drywell.

d. The sources and amounts of mass and energy release as well as the postaccident time dependence of the mass and energy release for the most severe of the postulated loss-of-coolant accidents are described in Subsection 6.2.1.3.

e. Energy released to the containment atmosphere as a result of the postulated accidents referred to in item d above is removed by the containment heat removal system (i.e., containment spray system, and suppression pool cooling system) discussed in Subsection 6.2.2.

   For the purpose of the containment peak pressure analysis, the containment heat removal system was assumed to be affected by the most restrictive single active failure resulting in the minimum heat removal capability.

f. Flow from postulated pipe ruptures is directed to the pressure suppression pool through the drywell to suppression pool vents, and distributed throughout the pool to condense the steam portion of the flow rapidly, and to limit the pressure differentials between the drywell and containment during various postaccident cooling modes.

g. The containment design permits removal of fuel assemblies from the reactor core after any postulated LOCA.

h. The containment system is protected against missiles, from internal or external sources, as discussed in Subsection 3.5.1, and excessive motion of the pipes that could directly or indirectly jeopardize containment integrity.

i. The containment limits leakage during and following any of the postulated LOCA's to a value less than the leakage rates that would result in an offsite dose greater than specified in 10 CFR 50.67.

j. Periodic leak tests, as discussed in Chapter 14 and Subsection 6.2.6, may be conducted to confirm the integrity of the containment.

For the purposes of the containment design there is no single design-basis accident (DBA). The various postulated LOCA's analyzed to determine peak temperature and pressure are discussed in Subsection 6.2.1.1.3.

6.2.1.1.2 Design Features

The design features of the containment structure and internal structures are described in the text and figures of Section 3.8.

6.2.1.1.2.1 Protection from the Dynamic Effects of Postulated Accidents

The containment structure, internal structures, and engineered safety feature systems are protected from loss of safety function due to dynamic effects of postulated accidents. The containment is designed to provide separation and inclusion of barriers, restraints, and impingement shields to protect essential structures and safe-shutdown systems and components from internally generated missiles, flooding, pipe whip and jet impingement forces. The detailed criteria, locations, and descriptions of devices used for protection are given in Sections 3.5 and 3.6.

6.2.1.1.2.2 Codes and Standards

Codes and standards applied to the design, fabrication, and erection of the containment and internal structures are discussed in Section 3.8. In each case the codes and standards used are consistent with the equipment safety function.

6.2.1.1.2.3 Qualification Tests

Testing of containment structure, systems, and components is described in Section 3.8, Subsection 6.2.6, and Chapter 14.

6.2.1.1.2.4 Protection Against External Pressure Loads

Inadvertent operation of containment heat removal systems and other possible modes of plant operation that could possibly result in significant external structural loading result in lower pressures than the design containment external pressure as discussed in Subsection 6.2.1.1.4. Therefore, no special provisions for loss of containment integrity under these conditions are required.

6.2.1.1.2.5 Potential Water Traps Inside Containment

Potential water entrapment volumes may cause suppression pool level drawdown. The suppression pool makeup system provides adequate makeup water from the upper containment pool to account for all conceivable entrapment volumes while maintaining long-term drywell vent water coverage and sufficient suction head for the ECCS pumps. See Subsection 6.2.7 for further discussion.

6.2.1.1.2.6 Containment Cooling and Ventilation Systems

The functional capability of the normal containment ventilation system to maintain the temperature pressure and humidity in the containment and subcompartments within the prescribed limits, and the action to be taken if these conditions are exceeded, are discussed in Section 9.4. The maximum allowable containment conditions for normal plant operation are listed in Section 3.11, Table 3.11-5.

Complete and total failure of these systems will necessitate a plant shutdown because the operating temperature limits will be exceeded.

6.2.1.1.3 Design Evaluation

6.2.1.1.3.1 Summary Evaluation

The key design parameters and the maximum calculated accident parameters for the pressure suppression containment are as follows:

Parameter	Design Parameter	Calc Accident Parameter
a. Containment design pressure	15 psig	6.97 psig(1)
b. Containment design temperature	185°F	182.2°F(2)
c. Drywell design pressure	30 psig	22.23 psig
d. Drywell design temperature	330°F	330°F(3)

(1) In the short-term analysis, a higher pressure peak (9.22 psig) occurs in the wetwell region below the HCU floor during pool swell.

(2) This is the peak suppression pool temperature. The peak containment temperature is 160.8°F (Reference 31).

(3) In the short-term, a higher drywell atmosphere temperature peak (335°F) occurs briefly (0.4 seconds) during the blowdown on a main steam line break. This temperature does not present a threat to the drywell structural materials during this short duration because it takes a much longer time for the drywell structural materials to increase to the limit.

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

The maximum drywell pressure occurs during the blowdown phase of a main steam line break. The peak containment pressure occurs during the long-term phase of the transient after the peak suppression pool temperature is reached.

The most severe drywell temperature condition (peak temperature and duration combined) occurs for a small primary system rupture above the reactor water level that results in the blowdown of reactor steam to the drywell (small steam break). In order to demonstrate that breaks smaller than the rupture of the largest primary system pipe will not exceed the containment design parameters, the containment system responses to an intermediate size liquid break and a small size steam break are evaluated. The results show that the containment design conditions are not exceeded for these smaller break sizes.

Each event is divided into short-term and long-term analyses. The short-term analysis (0 to 30 seconds) determine the peak drywell pressure and temperature and provides inputs for the containment loads evaluations. The long-term analysis (through 30 days) determines the peak containment and suppression pool temperature as well as the long term containment pressure.

The analyses assume that the primary system and containment are initially at the limiting normal operating conditions that result in the peak pressures and temperatures. References are provided that describe relevant experimental verification of the analytical models used to evaluate the containment system response.

6.2.1.1.3.2 Containment Design Parameters

Table 6.2-1 provides a listing of the key design parameters of the primary containment system including the design characteristics of the drywell, suppression pool 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 post blowdown long-term accident operation. Performance parameters given include those applicable to full operation and to those conservatively reduced capacities assumed for containment analyses.

6.2.1.1.3.3 Accident Response Analysis

The containment functional evaluation is based upon 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 steamline,

c. an intermediate size liquid line rupture, and

d. a small size steamline rupture.

Bounding energy release from these accidents is reported in Subsection 6.2.1.3.

6.2.1.1.3.3.1 Recirculation Line Break

Immediately following the rupture of the recirculation line, the flow out both sides of the break will be limited to the maximum allowed by critical flow considerations. The total effective flow area is given in Figure 6.2-1. In the side adjacent to the suction nozzle, the flow will correspond to critical flow in the pipe cross section. In the side adjacent to the injection nozzle, the flow will correspond to critical flow at the ten jet pump nozzles associated with the broken loop. In addition, the cleanup line crosstie will add to the critical flow area. Table 6.2-3 provides a summary of the break areas.

6.2.1.1.3.3.1.1 Assumptions for Reactor Blowdown

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

a. The initial conditions for the recirculation line break accident are such that the system energy is maximized and the system mass is minimized. That is:

   1. The reactor is operating at 102% of rated power. This maximizes the postaccident decay heat.

   2. The service water temperature is the maximum normal.

   3. The suppression pool mass is at the low water level for the long-term analysis and at the high water level for the short-term analysis.

   4. The suppression pool temperature is the maximum normal.

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

c. Reactor power generation ceases 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 the shutdown times is negligible.

d. The vessel depressurization flowrates are calculated using Moody's critical flow model (Reference 3) assuming "liquid only" outflow, since this assumption maximizes the energy release to the drywell. "Liquid only" outflow implies that all vapor formed in the reactor pressure vessel (RPV) by bulk flashing rises to the surface rather than being entrained in the existing flow. In reality, some of the vapor would be entrained in the break flow which would significantly reduce the RPV discharge flowrates. Further, Moody's critical flow model, which assumes annular, isentropic flow, thermodynamic phase equilibrium, and maximized slip ratio, accurately predicts vessel outflows through small diameter orifices. Actual rates through larger flow areas, however, are less than the model indicates because of the effects of a near homogeneous two-phase flow pattern and phase nonequilibrium. These effects are conservatively neglected in the analysis.

e. The core decay heat and the sensible heat released in cooling the fuel to initial average coolant temperature are included in the reactor pressure vessel depressurization calculation. The rate of energy release is calculated using a conservatively high heat transfer coefficient throughout the depressurization period. The resulting high energy release rate causes the RPV to maintain nearly rated pressure for approximately 20 seconds. The high RPV pressure increases the calculated blowdown flowrates which is again conservative for analysis purposes. The sensible energy of the fuel stored at temperatures below the initial average coolant temperature is released to the vessel fluid along with the stored energy in the vessel and internals as vessel fluid temperatures decrease during the remainder of the transient calculation.

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

g. A complete loss of offsite power occurs simultaneously with the pipe break. This condition results in the loss of power conversion system equipment and also requires that all vital systems for long-term cooling be supported by onsite power supplies.

h. Feedwater flow to the reactor is assumed to continue after the break. For the short-term analysis, the feedwater inventory out to the last feedwater heater is included as part of the initial vessel fluid. For the long-term analysis, the feedwater flow into the vessel continues until all the high-energy feedwater with temperatures above the peak suppression pool temperature is injected into the reactor vessel. Including the feedwater at temperatures greater than the peak suppression pool temperature maximizes the peak suppression pool temperature.

6.2.1.1.3.3.1.2 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 containment during the short-term. Since highly turbulent conditions are expected due to the blowdown flow, the analysis assumes complete mixing.

b. The fluid flowing through the drywell-to-suppression pool vents is formed from a homogeneous mixture of the fluid in the drywell. The use of this assumption results in complete carryover of the drywell air and a higher positive flow rate of liquid droplets which conservatively maximizes vent pressure losses.

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

d. In the short-term analysis, no heat loss occurs from the gases inside the containment. In reality, condensation of some steam on the drywell surfaces would occur.

6.2.1.1.3.3.1.3 Assumptions for Long-Term Cooling

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

a. The LPCI pumps are used to flood the core prior to 600 seconds after the accident. The HPCS is available for the entire accident.

b. The effects of decay energy, stored energy, sensible energy, energy added by ECCS pumps, and energy from the zirconium water reaction on the suppression pool temperature are considered.

c. The suppression pool is available as a heat sink in the containment system. After 1800 seconds, makeup from the upper containment pool is included.

d. After approximately 1800 seconds, the RHR heat exchangers are activated to remove energy from the containment via recirculation cooling of the suppression pool with the RHR service water systems. It is conservatively assumed that containment spray is not utilized.

e. Passive heat sinks in the drywell, and containment are modeled for the accident and transient events (Table 6.2-9). The Uchida convective heat transfer coefficients are used based on the local steam-to-air ratio.

f. The mixing between the containment airspace and the suppression pool and the resulting containment airspace heatup is modeled mechanistically by realistic heat and mass transfers between the suppression pool and the wetwell airspace.

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

Case A. Offsite power available - All ECCS equipment operating (at 2952 MWt).

Case B1. Loss of offsite power - Minimum diesel power available for ECCS (at 2952 MWt).

Case B2. Loss of offsite power - Minimum diesel power available for ECCS (at 3543 MWt).

6.2.1.1.3.3.1.4 Initial Conditions for Accident Analyses

Table 6.2-4 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, and the containment.

Table 6.2-3 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.

6.2.1.1.3.3.1.5 Short-Term Accident Response

The calculated containment pressure and temperature responses for the recirculation line break are shown in Figures 6.2-2 and 6.2-3, respectively. Following the break, the drywell pressure increases rapidly due to the injection of the break flow. The peak drywell pressure occurs during the vent clearing phase of the transient as suppression pool water is being cleared from the vents. Following vent clearing, the drywell pressures decreases as the break flow decreases.

The containment is pressurized early in the transient by the carryover of noncondensables from the drywell. As the transient continues, break flow enters the suppression pool, and the temperature of the suppression pool water increases, causing the containment pressure to increase.

Table 6.2-5 provides the peak pressure, temperature and time parameters for the recirculation line break as predicted for the conditions of Tables 6.2-3 and 6.2-4 and in correspondence with Figures 6.2-2 and 6.2-3. Figure 6.2-2 shows the time dependent response of the drywell differential pressure.

During the blowdown period of the LOCA, the pressure suppression vent system conducts the flow of the steam-water gas mixture in the drywell to the suppression pool for condensation of the steam. The pressure differential between the drywell and suppression pool controls this flow. Figure 6.2-5 provides the mass flow versus time relationship through the vent system for this accident.

6.2.1.1.3.3.1.6 Long-Term Accident Responses

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 in Subsection 6.2.1.1.3.3.1.3 for the two cases of interest.

CASE A: All ECCS equipment operating (at 2952 MWt).

This case assumes that offsite a-c power is available to operate all cooling systems. During the first 1800 seconds following the pipe break, the high-pressure core spray (HPCS), low-pressure core spray (LPCS) and all LPCI pumps are assumed operating. All flow is injected directly into the reactor vessel.

After 1800 seconds, both RHR heat exchangers are activated to remove energy from the containment. During this mode of operation the LPCI flow is routed through both RHR heat exchangers where it is cooled before being returned to the suppression pool.

The containment pressure response to this set of conditions is shown as curve A in Figure 6.2-6. The corresponding drywell and suppression pool temperature responses are shown in Figures 6.2-7 and 6.2-8. 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 removal rate of the RHRS equals the energy addition rate from the decay heat, the containment pressure and temperature reach a second peak value and decrease gradually. Table 6.2-6 summarizes the equipment operation, the peak long-term containment pressure, and the peak suppression pool temperature.

CASE B1: Loss of offsite power - minimum ECCS equipment operating (at 2952 MWt).

This case assumes no offsite power is available following the accident with only minimum diesel power. After 1800 seconds, the LPCI flow through only one RHR heat exchanger is returned to the suppression pool. The containment pressure response to this set of conditions is shown as curve B in Figure 6.2-6. The corresponding drywell and suppression pool temperature responses are shown in Figures 6.2-7 and 6.2-8. A summary of this case is given in Table 6.2-6.

Figure 6.2-9 shows the rate at which the RHR system heat exchanger will remove heat from the suppression pool following a LOCA (Subsection 6.2.2 describes the containment cooling mode of the RHR system). The heat removal rate is shown for the two cases at 2952 MWt. The first assumes that all the ECCS equipment is available, including both RHR heat exchangers and the associated RHR service water pumps. The second case is for the very degraded minimum cooling condition that would limit the heat removal capacity to one heat exchanger. For both cases, it was conservatively assumed that at the time of the accident the residual heat removal service water was at its maximum design temperature as defined in Table 6.2-2.

CASE B2: Loss of offsite power-minimum ECCS equipment operating (at 3543 MWt).

Case B1, above, is reanalyzed at 3543 MWt. The drywell and containment pressure responses to this set of conditions are shown in Figure 6.2-6a. The corresponding drywell, containment and suppression pool temperature responses are shown in Figure 6.2-7a. A summary of this case is given in Table 6.2-6a.

For Case B2, rather than taking the results of the short-term evaluation as a starting point, the beginning of the event, including blowdown, is modeled. At the end of the blowdown, the drywell pressure stabilizes at a slightly higher pressure than the containment, the difference being equal to the hydrostatic head of vent submergence. During the RPV depressurization phase, most of the noncondensable gases initially in the drywell are forced into the containment. However, following the depressurization the noncondensables will redistribute between the drywell and containment via the vacuum breaker system. This redistribution takes place as steam in the drywell is condensed by the relatively cool ECCS water which is beginning to cascade from the break causing the drywell pressure to decrease.

The ECCS supplies sufficient core cooling water to control core heatup and limit metal-water reaction to less than 1%. 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 pool via the drywell-to-suppression pool vent system.

Figure 6.2-9a shows the rate at which the RHR system heat exchanger will remove heat from the suppression pool for Case B2.

6.2.1.1.3.3.1.7 Energy Balance During Accident

In order to establish an energy distribution in the containment 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 sensible energy,

c sensible heat rate (vessel and internals),

d. pump heat rate,

e. heat removal rate from suppression pool (Figure 6.2-9 or 6.2-9a), and

f. metal-water reaction heat rate.

Items a, b, c, d and f are provided in Subsection 6.2.1.3

6.2.1.1.3.3.1.8 Chronology of Accident Events

A complete description of the containment response to the recirculation line break has been given in Subsections 6.2.1.1.3.3.1.5 through 6.2.1.1.3.3.1.7. Results for this accident are shown in Figures 6.2-2, 6.2-3, 6.2-5, 6.2-6, 6.2-6a, 6.2-7, 6.2-7a, 6.2-8, 6.2-9 and 6.2-9a.

6.2.1.1.3.3.2 Main Steamline Break

The assumed sudden rupture of a main steamline between the reactor vessel and the flow limiter would result in the maximum flow rate of primary system fluid and energy to the drywell. This would in turn result in the maximum drywell differential pressure. The sequence of events immediately following the rupture of a main steamline between the reactor vessel and the flow limiter have been determined. The flow in both sides of the break will accelerate to the maximum allowed by the critical flow considerations. In the side adjacent to the reactor vessel, the flow will correspond to critical flow in the steamline break area. Blowdown through the other side of the break will 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 steamline flow restrictor. The total effective flow area is given in Figure 6.2-10.

6.2.1.1.3.3.2.1 Assumptions for Reactor Blowdown

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

a. The vessel depressurization flowrates are calculated using Moody's critical flow model (Reference 3). During the first second of blowdown, the flow consists of saturated steam.

Immediately following the break, the total steam flow rate leaving the vessel exceeds the steam generation rate in the core, causing an initial depressurization of the reactor pressure vessel. 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.

b. The main steam isolation valves start closing at 0.5 second after the accident and are fully closed in the maximum time of 5 seconds following closure initiation. By assuming slow closure of these valves, a large effective break area is maintained for a longer period of time. The peak drywell pressure occurs before the reduction in effective break area and is therefore insensitive to any additional delay in closure of the isolation valves.

6.2.1.1.3.3.2.2 Assumptions for Containment Pressurization

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

6.2.1.1.3.3.2.3 Assumptions for Long-Term Cooling

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

6.2.1.1.3.3.2.4 Initial Conditions for Accident Analyses

Table 6.2-4 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, and the containment.

Table 6.2-3 provides the initial conditions and numerical values assumed for the main steamline break accident as well as the sources of energy considered prior to the postulated pipe rupture.

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

6.2.1.1.3.3.2.5 Short-Term Accident Response

Figures 6.2-11 and 6.2-12 show the pressure and temperature responses of the drywell and suppression chamber during the primary system blowdown phase of the steamline break accident. Figure 6.2-14 shows the vent mass flow versus time.

The drywell atmosphere temperature approaches a peak at approximately 1 second after 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. The peak differential pressure occurs shortly after the vent clearing transient. As the blowdown proceeds, the primary system pressure and fluid inventory will decrease resulting in reduced break flow rates. As a consequence, the flow rate in the vent system and the differential pressure between the drywell and suppression chamber begin to decrease.

Table 6.2-5 presents the peak pressures, peak temperatures and times of this accident as compared to the recirculation line break.

6.2.1.1.3.3.2.6 Long-Term Accident Responses

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 in Subsection 6.2.1.1.3.3.1.3. The accident analyzed corresponded to the minimum ECCS flow condition.

The drywell and containment pressure responses to this of conditions are shown in Figure 6.2-6b. The corresponding drywell, containment and suppression pool temperature responses are shown in Figure 6.2-7b. A summary of this case is given in Table 6.2-6a. The beginning of the event, including blowdown, is modeled. After the primary system pressure has dropped to the drywell pressure, the blowdown will be over. At this time the drywell will contain saturated steam, and the drywell and containment pressures will stabilize. The pressure difference corresponds to the hydrostatic pressure of vent submergence.

The drywell and suppression pool will remain in this equilibrium condition until the reactor vessel refloods. During this period, the emergency core cooling pumps will be injecting cooling water from the suppression pool into the reactor.

For this event, it is assumed that the operators control the water level in the vessel below the elevation of the break. This results in the drywell temperature and pressure remaining high to the end of the event (approximately 240°F and 25 psia). If ECCS flow were allowed to flood the reactor vessel to the steamline nozzle and to spill out into the drywell, the water spillage would condense the steam in the drywell and thus reduce the drywell temperature and pressure. As soon as the drywell temperature dropped below the containment pressure, the drywell vacuum breakers would open and noncondencensable gases from the containment would flow back into the drywell until the pressure in the two regions equalized. The resultant condition would be similar to that seen for the recirculation line break.

Figure 6.2-9b shows the rate at which the RHR system heat exchanger will remove heat from the suppression pool for the main steamline break. It was conservatively assumed that at the time of the accident the residual heat removal service water was at its maximum design temperature as defined in Table 6.2-2.

6.2.1.1.3.3.2.7 Energy Balance During Accident

In order to establish an energy distribution in the containment 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 and fuel relaxation sensible energy,

c. c. sensible heat rate (vessel and internals),

d. pump heat rate,

e. heat removal rate from suppression pool (Figure 6.2-9 or 6.2-9b), and

f. metal-water reaction heat rate.

Items a, b, c, d and f are provided in Subsection 6.2.1.3. A complete energy balance for the main steamline break accident at 2952 MWt is given in Table 6.2-7 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 peak pressure in the containment.

6.2.1.1.3.3.2.8 Chronology of Accident Events

A complete description of the containment response to the main steam line break has been given in Subsections 6.2.1.1.3.3.2.5 through 6.2.1.1.3.3.2.7. Results for this accident are shown in Figures 6.2-6, 6.2-6b, 6.2-7, 6.2-7b, 6.2-8, 6.2-9, 6.2-9b, 6.2-11, 6.2-12, and 6.2-14. A chronological sequence of events for this accident from time zero is provided in Tables 6.2-8 and 6.2-8a.

6.2.1.1.3.3.3 Hot Standby Accident Analysis

This accident was not reanalyzed at current power (3543 MWt) since it was not the limiting case. The analysis presented below is based on a power of 2952 MWt.

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

The peak drywell pressure following a main steamline break is dependent upon the rise time of the reactor water level as this determines the time at which the two phase blowdown begins. A 1-second level rise time is a conservative bounding condition for a main steamline break with a reduced reactor power level. However, since a 1-second level rise time was conservatively assumed for the LOCA at 102% of rated power, the peak drywell pressure following a blowdown at hot standby will be no higher than is shown in Figure 6.2-11.

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

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

a. 102% of rated power/maximum normal pool temperature, or

b. approximately 1/2 hour of hot standby operation.

In both cases, containment cooling (RHR system) is initiated 30 minutes after the LOCA.

A blowdown at 1/2 hour after an isolation results in the highest peak long-term temperature because it is assumed that no heat is rejected from the system for 1/2 hour after the start of an isolation event. Thus, Figure 6.2-15 indicates that the longterm consequences of a LOCA which occurs after a period of hot standby operation are no more severe than for a LOCA at 102% of rated power.

6.2.1.1.3.3.4 Intermediate Size Breaks

This accident was not reanalyzed at current power (3543 MWt) since it was not the limiting case. The analysis presented below is based on a power of 2952 MWt.

An intermediate size break is analyzed as part of the containment performance evaluation to demonstrate that the consequences are no more severe than from a rupture of the largest primary system pipe. This classification covers those breaks for which the blowdown will result in reactor depressurization and operation of the ECCS. This section describes the consequences to the containment 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 per second. 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- containment differential pressure is equal to the vent submergence hydrostatic pressure.

Figures 6.2-16 and 6.2-17 show the drywell and containment 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 containment free space. The continual purging of drywell air to the containment will result in a gradual pressurization of both the containment and drywell. The containment will continue to gradually increase in pressure due to the long-term pool heatup.

The ECCS will be initiated as a result of 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 1,100 seconds. This will terminate the blowdown phase of the transient.

In addition, the suppression pool end of blowdown temperature will be the same as that of the main steamline break because essentially the same amount of primary system energy is released during the blowdown. After reactor depressurization and reflood, water from the ECCS will begin to flow out the break. This flow will condense the drywell steam and eventually cause the drywell and containment pressures to equalize in the same manner as following a main steamline break.

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

From this description, it can be concluded that the consequences of an intermediate size break are less severe than from a main steamline break.

6.2.1.1.3.3.5 Small Size Breaks

6.2.1.1.3.3.5.1 Reactor System Blowdown Considerations

This section discusses the containment transient associated with small primary systems blowdowns. The sizes of primary system ruptures in this category are those that will not result in reactor depressurization due to either loss of reactor coolant or automatic operation of the ECCS equipment. Following 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 (for example) the steam and liquid associated with a liquid blowdown would be at 212°F.

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 constant enthalpy depressurization of high pressure, saturated steam will result in superheated conditions. For example, decompression of 1000 psia saturated steam to atmospheric pressure will result in 298°F superheated steam (86°F of superheat).

A small reactor steam leak (resulting in superheated steam) 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 for the larger break 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.

6.2.1.1.3.3.5.2 Containment Response

For drywell design considerations, the following sequence of events is assumed to occur. With the reactor and containment operating 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. The pressure increase will lower the water level in the annulus until the level begins to clear 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 containment free space. The air carryover 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 the containment, short-term pressurization of the containment will cease and the system will reach an equilibrium condition. The drywell will contain only superheated steam, and continued blowdown of reactor steam will condense in the suppression pool. The suppression pool temperature will continue to increase until the RHR heat exchanger heat removal rate is equal to the decay heat release rate.

6.2.1.1.3.3.5.3 Recovery Operations

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

6.2.1.1.3.3.5.4 Drywell Design Temperature Considerations

For drywell design purposes, it is assumed that there is a blowdown of reactor steam for the 6-hour cooldown period. The corresponding design temperature is determined by finding the combination of primary system pressure and drywell pressure that produces the maximum superheat temperature. This temperature is then assumed to exist for the entire 6-hour period. The maximum drywell steam temperature occurs when the primary system is at approximately 450 psia and the drywell pressure is maximum. Thus, for small size break analysis, it is assumed that the drywell is at 15 psig; this results in a temperature of 330°F.

The small break results in the maximum sustained temperature in the drywell. A large steam line break results in a slightly higher drywell atmosphere peak temperature, but the duration of this spike is too short (approximately 0.4 seconds) to effect the drywell structure. Therefore, the drywell design temperature of 330°F is adequate to ensure integrity in all design basis events.

6.2.1.1.3.4 Accident Analysis Models

6.2.1.1.3.4.1 Short-Term Pressurization Model

The analytical models, assumptions, and methods used by General Electric to evaluate the containment response during the reactor blowdown phase of a LOCA are described in References 1 and 2.

6.2.1.1.3.4.2 Long-Term Cooling Model

The analytical models, assumptions, and methods used by General Electric to evaluate the long-term pressure/temperature response are described in References 1 and 2. The methodology is incorporated into the long-term containment response computer code. The computer code performs mass and energy balances on models of the reactor vessel, drywell airspace, drywell pool, weir annulus, suppression pool, and containment airspace. The code has a static vent clearing model but incorporates a complex system model, which permits realistic simulation of the ECCS systems, containment spray, and upper pool dump. Schematically, the cooling loop used for the analysis is shown in Figure 6.2-18.

6.2.1.1.3.5 High Energy Line Rupture Within the Containment

In order to pass from the drywell to the auxiliary building, some primary system pipes pass through the containment (the main steamlines for example). If these pipes were unguarded, rupture within the containment would result in a direct release of primary system fluid to the containment atmosphere. The pressure suppression features of the containment would thus be bypassed and the potential would exist for a pipe rupture to produce significant containment pressures.

Because of this potential, all reactor coolant pressure boundary pipes of a size which would result in containment overpressurization which pass through the containment, with the exception of the LPCI, HPCS, and LPCS, are provided with guard pipes that vent to the drywell. Thus, in the event of a pipe rupture, the blowdown flow will pass through the suppression pool vent system, and the steam will be condensed. The traversing incore probe (TIP), control rod drive (CRD) insert and withdraw, reactor water cleanup system and instrument lines could also discharge primary system coolant to the containment in the event of a rupture. Unisolatable instrument line rupture results in the maximum discharge of primary system coolant to the containment. This accident is discussed in Chapter 15. Each instrument line contains a 1/4-inch diameter flow restricting orifice to limit the containment pressure increase to values well below the design pressure.

The LPCI, LPCS, and HPCS lines have check valves inboard of the drywell penetration that will prevent blowdown to the containment. The major components of the reactor water cleanup system are located within the containment. The system suction line penetrates the drywell and is provided with a guard pipe. The cleanup system components located inside the containment are provided with break detection and isolation systems that will limit the total blowdown fluid flow to the containment to acceptable values.

6.2.1.1.4 Negative Pressure Design Evaluation

6.2.1.1.4.1 Evaluation of Drywell Negative Differential Pressure

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

a. All air has been purged out of the drywell.

b. Drywell vacuum breakers do not open.

c. The suppression pool is at the post-blowdown (1800 seconds) temperature, as determined from Figure 6.2-7b.

d. The containment is at suppression pool temperature and 100% relative humidity.

e. Steam in the drywell is cooled to suppression pool temperature.

The negative pressure evaluation uses the initial conditions in Table 6.2-4 for the long term except for the drywell initial pressure. The negative pressure evaluation assumes that the drywell initial pressure is atmospheric (0 psig) to reduce the initial air mass and conservatively predict the most severe negative pressure.

The final drywell pressure is equal to the saturation pressure as the suppression pool post-blowdown temperature. The final containment pressure is equal to the partial pressure of the vapor at the suppression pool post-blowdown temperature plus the partial pressure of the air assuming the entire drywell air mass has been purged into containment. Using these assumptions and initial conditions, the bounding negative pressure load across the drywell wall is:

$$\Delta P_D = P_D - P_C$$

$$\Delta P_D \leq -16.9 \text{ psid}$$

6.2.1.1.4.2 Evaluation of Containment Negative Pressure

The transients which could result in significant negative pressure within the containment all involve the inadvertent actuation of the containment spray while the containment atmosphere is at high temperature and humidity. The greatest negative pressure condition would occur if there is a break in the reactor water cleanup (RWCU) system followed by actuation of containment spray.

The calculated maximum negative containment pressure for this case is less than 2.2 psid. The conservative initial conditions assumed for the evaluation are:

a.	Containment free volume	1.51×10^6 ft^3
	1. Temperature	104°F
	2. Pressure	14.7 psia
	3. Relative humidity	60%
b.	Drywell free air volume	2.47×10^5 ft^3
	1. Temperature	135°F
	2. Pressure	14.7 psia
	3. Relative humidity	30%
c.	Suppression pool temperature	60°F.

The peak containment pressure resulting from the break would be less than 3 psig. The steam released to the containment is assumed to result in a change of temperature and relative humidity at the time of spray initiation to the following:

a. Temperature 137°F

b. Relative humidity 100%.

Assumptions used for this calculation of negative pressure are as follows:

a. There is no heat transfer between the suppression pool and the containment atmosphere. This assumption is conservative because the suppression pool is the source for the containment spray fluid, and the lower the spray temperature the greater the value of the negative pressure.

b. The containment sprays are assumed to be 100% efficient to maximize the negative pressure.

c. The decay heat input and/or metal-to-water reaction heat input after blowdown is neglected. This is conservative because any additional energy would tend to reduce the magnitude of the negative pressure.

d. The drywell is treated as a compartment having vacuum relief valves permitting flow only from the containment to the drywell. When the containment pressure exceeds the drywell pressure by 0.2 psi, the containment vents to the drywell in order to relieve this negative differential pressure.

e. The containment air volume is maximized by assuming low water level.

The maximum negative containment pressure for the RWCU break is calculated to be less than 2.2 psid, which is less that the 3.0 psid negative limit determined by liner plate deformation without exceeding the allowable stress level, therefore a containment vacuum relief system is not required.

6.2.1.1.5 Steam Bypass of the Suppression Pool

6.2.1.1.5.1 Introduction

The concept of the pressure suppression reactor containment is that any steam released from the primary system will be condensed by the suppression pool and will not have an opportunity to produce a significant pressurization effect on the containment. This is accomplished by channeling the steam into the suppression pool through a vent system. This arrangement forces steam released from the primary system to be condensed in the pool. If a leakage path were to exist between the drywell and the containment, the leaking steam would produce pressurization of the containment. To mitigate the consequences of any steam which bypasses the suppression pool, a high containment pressure signal will automatically initiate the containment spray system any time after LOCA + 10 minutes. Realignment logic and interlock affecting operation of containment sprays are discussed in Subsection 7.3.1.

A possible scenario that has been hypothesized is that the suppression pool may be bypassed during normal operating conditions due to a stuck open check valve. Bypass of the suppression pool will not exist unless both valves in a given line are open. These valves are passive valves and are periodically checked to determine that they will open. The position of each of the valves is indicated to the operator. (Q&R 421.12)

Position indicating lights are provided on the Standby Information Panel in the main control room for each of the eight check valves in the four vacuum relief lines. These indicating lights are controlled by limit switches on the check valves and indicate closed, intermediate, and open valve-position. (see Subsection 7.5.1.4.2.8.1)

Test switches for the valves are provided near the vacuum breaker position indicators. There are no alarms associated with these valves. The valves will be tested in accordance with approved technical specifications.

As discussed in Subsection 6.2.6.5.1, a drywell bypass leakage rate tests was performed initially during the preoperational test program at the design pressure (30 psig). Drywell bypass leakage tests will continue to be performed periodically at reduced pressure (3 psig) in accordance with the Technical Specifications. The acceptance criterion developed as indicated in NUREG 75/087 Section 6.2.1.1.C, Item I1.5.C, is specified in Subsection 6.2.6.5.1. (Q&R 480.25)

The following presents the results of calculations performed to determine the allowable leakage capacity between the drywell and containment.

6.2.1.1.5.2 Criteria

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

6.2.1.1.5.3 Analysis

The allowable drywell leakage capacity has been evaluated for the complete spectrum of credible primary system rupture areas. It is expressed in terms of the parameter $A/\sqrt{K}$ where

A = Flow area of leakage path, ft^2.

K = Geometric and friction loss coefficient.

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

$$P_D - P_C = K \cdot \frac{V^2}{2g_c} \cdot \frac{1}{144v}$$

Where:

P_D = Drywell pressure, psia

P_c = Containment pressure, psia

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

V = Velocity of flow, ft/sec.

g_c = Proportionality constant, 32.2 lbm-ft/lbf-sec^2

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

If the leakage path flow rate is $\dot{M}$ (lbm/sec) and the flow area is A (ft^2), the above equation can be rewritten to give

$$\dot{M} = A/\sqrt{K}\ \sqrt{2g_c\left(P_D - P_C\right)\cdot\frac{144}{v}}$$

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

6.2.1.1.5.4 Bypass Capability Without Containment Spray and Heat Sinks

The following description applies to an evaluation performed at the original licensed thermal power of 2952 MWt. Since the results of this evaluation are not used for determining the limiting allowable leakage area, it was not performed at the current power level.

Although containment spray will be automatically initiated on high containment pressure if required any time after LOCA + 10 minutes, this analysis demonstrates the allowable bypass leakage capability without containment spray. Figure 6.2-19 shows the allowable leakage as a function of primary system break area. It is a composite of two curves. Large primary system ruptures generate high pressure differentials across the assumed leakage path which in turn give proportionally higher leakage flow rates. However, large primary system breaks also rapidly depressurize the reactor and terminate the blowdown. Once this has occurred, there will no longer be a pressure differential across the drywell leakage path so that leakage flow and containment pressurization will cease. Since leakage into the containment is of limited duration, the maximum allowable area of the leakage path is large. Assuming a primary system rupture of 3.2 ft^2, Figure 6.2-19 shows the allowable leakage flow path could have an $A/\sqrt{K}$ of 10.15 ft^2.

As the size of the assumed primary system rupture decreases, the magnitude of the differential pressure across any leakage path also decreases. However, smaller breaks result in an increasingly longer reactor blowdown period which, in turn, results in longer durations of the leakage flow. The limiting case is a very small reactor system break which will not automatically result in reactor depressurization. For this case, it is assumed that the response of the plant operators is to shut the reactor down in an orderly manner at 100°F/hr cooldown rate. This would result in the reactor being depressurized and the break flow being terminated within approximately 6 hours. During this 6-hour period, the blowdown flow from the reactor primary system would have swept all the drywell air over to the containment. The blowdown steam would be condensed in the suppression pool, but in order for this to occur, the water level in the vent annulus would have to be depressed to the top upper row of vents. This continuous pressure differential, combined with a 6-hour duration, results in the most severe drywell leakage requirement. The maximum allowable leakage path area under these circumstances is an $A/\sqrt{K}$ of 0.02 ft^2.

A study has been made of potential cracking of the reinforced concrete drywell due to shrinkage, thermal gradients, seismic events, small break and LOCA accidents, and combinations of these (Reference 4). The report indicates no significant cracking of the drywell walls.

6.2.1.1.5.5 Bypass Capability with Containment Spray and Heat Sinks

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

The flow rate of one containment spray loop is 3800 gpm and is assumed to be initiated no sooner than 10 minutes after the accident. The suppression pool water passes through the RHR heat exchanger and is injected into the upper Containment region. The spray will rapidly condense the steam and would therefore create a homogeneous air-steam mixture in the containment. The available containment heat sinks shown in Table 6.2-9 were considered with variable convective heat transfer coefficients based on the local instantaneous air-steam ratio. The shutdown rate was assumed to be 100°F/hr, and the maximum design service water temperature (Table 6.2-2) was used. The shutdown rate corresponds to the maximum rate which does not impose thermal cycle on the reactor vessel. This analysis results in an allowable drywell leakage capability of $A/\sqrt{K}$ of 1.00 ft^2. The corresponding pressure transient is shown in Figure 6.2-20.

The assumptions for allowable bypass calculations utilizing heat sinks are as follows:

a. Following the occurrence of a pipe line break within the drywell, air is purged through the vents into the containment.

b. The air in the containment is compressed by the incoming mixture of air and steam.

c. The containment sprays are activated 180 seconds after the containment pressure reaches 9 psig, or at LOCA + 13 minutes, whichever comes later.

d. The RHR heat exchanger in containment spray mode is initiated no later than 6 minutes after the spray is initiated.

e. The efficiency of the sprays are based upon the local steam to air ratio.

f. The air and steam in the containment is homogeneously mixed throughout the event.

g. Heat is transferred to exposed concrete and steel in the containment. The Uchida convective heat transfer coefficients used are based on the local steam to air ratio.

h. No energy is assumed to leave the containment except through the RHR heat exchanger.

The following analysis provides an illustration of the methods used to calculate steam condensing capability under typical post-LOCA conditions.

The spray water temperature is calculated from:

$$T_S = T_P - \frac{KHX}{\dot{M}_S} \frac{(T_P - T_{SW})}{C_p}$$

Where:

T_S = Spray temperature at the nozzle, °F

T_p = Suppression pool temperature, °F

KHX = Heat exchange effectiveness, Btu/sec °F (degraded)

T_{SW} = Service water temperature, °F

$\dot{M}_s$ = Spray flow rate, lbm/sec

The containment spray takes water from the suppression pool and injects it into the containment airspace. The spray droplets absorb some of the heat from the containment airspace and reach a new temperature, depending on the spray efficiency, which is a function of the air-to-steam mass ratio in the airspace. Some of the droplets may evaporate, and the remainder settle to the suppression pool. A mass and energy balance of the containment airspace determines the resultant conditions in the containment (pressure, temperature, air and steam masses) for each time step.

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

Based on the above numbers, the allowable drywell leakage rate as established by the small break accident is $A/\sqrt{K}$ = 1.0. The equivalent flow rate in standard cubic feet per minute for a test at 3 psid is provided in Table 6.2-1. The fact that the leak rate is not exceeded will be verified by periodic tests in accordance with the Technical Specifications.

6.2.1.1.6 Suppression Pool Dynamic Loads

The methodologies, from which the structural design basis suppression pool dynamic loads due to SRV discharge and LOCA events are determined, are discussed in Attachment A3.8. Attachment A3.8 presents a detailed discussion of the development of the design basis SRV and LOCA loads as well as their treatment in establishing the structural design basis.

Similarly the design basis submerged structure loads on piping and equipment are discussed in detail in Attachment A3.9.

The piping and equipment located in the containment and internal containment structures have been designed in accordance with the SRV- and LOCA-related loads defined in GE Topical Report NEDO-11314-08 (GESSAR Appendix 3B). The Mark III Confirmatory

Test Program has provided additional information regarding SRV and LOCA phenomena in the Mark III containment. A bibliography of the test reports is provided in Section 3B.13 on Pages 3B-61 and 3B-62 of General Electric document 22A7000 (Appendix 3B of GESSAR-FDA). In addition, Table 3B-2 of 22A7000 presents a detailed, load by load comparison of the test results and the Mark III design basis loads.

Illinois Power Company has performed a conformance evaluation assessing the status of CPS with respect to NUREG-0763, "Guidelines for Confirmatory Inplant Tests of Safety Relief Valve Discharges for BWR Plants." It has been concluded that CPS is exempt from confirmatory inplant SRV discharge testing due to its similarity to Kuosheng and Grand Gulf. (Q&R 480.26)

The design basis loads defined in Table 3B-2 of 22A7000 are considered in the design verification of the CPS piping and equipment located within the containment and internal containment structures. Since CPS was originally designed to accommodate ramshead SRV loads defined and provided by General Electric, it is unlikely that any design changes in the containment will be required.

The structural and BOP piping and equipment assessments for the SRV and LOCA loads are presented in Attachment B3.8 and Attachment B3.9, respectively. The assessment of NSSS piping and equipment capability with respect to these loads is discussed in Attachment C3.9. The structural implications relating to the NSSS piping and equipment load assessment are discussed in Attachment C3.9.

6.2.1.1.7 Asymmetric Loading Conditions

Containment and internal structure asymmetric loading conditions including localized pipe forces, pool swell, and safety/relief valve actuations are discussed in Section 3.8 and Attachment A3.8. Tornado and design wind loads are also asymmetric loads on the containment structure and are described in Section 3.3.

Analytical models used to evaluate the containment and drywell responses to postulated accidents and transients combined with the effects of operating basis and safe shutdown earthquakes are discussed in Section 3.8 and Attachment A3.8. The assumptions used in modeling these structures are also discussed.

6.2.1.1.8 Containment Environmental Control

The functional capability of the normal containment ventilation system to maintain the temperature pressure and humidity in the containment and subcompartments within the prescribed limits and the action to be taken if these conditions are exceeded are discussed in Section 9.4 The maximum allowable containment conditions for normal plant operation are listed in Section 3.11, Table 3.11-5.

6.2.1.1.9 Postaccident Monitoring

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

6.2.1.2 Containment Subcompartments

The containment is a large continuous volume that encloses both the drywell and the containment pipe tunnel. The containment, containment pipe tunnel, and drywell are interrupted at various locations by walls, piping, grating, ventilation ducts, etc., that form subcompartments which could experience pressure loadings if high-energy lines were to break inside them. The volumes within the containment which can be classified as subcompartments include:

a. the volume bounded by the drywell head, the reactor head and the connecting bulkhead (hereafter referred to as the head cavity);

b. the annular area between the reactor pressure vessel and the biological shield (hereafter referred to as the shield annulus);

c. the containment pipe tunnel;

d. the RWCU heat exchanger rooms;

e. the RWCU valve rooms;

f. the RWCU crossover pipe tunnel;

g. the RWCU filter-demineralizer holding pump room;

h. the RWCU filter-demineralizer rooms; and

i. the RWCU filter-demineralizer valve room.

6.2.1.2.1 Design Basis

6.2.1.2.1.1 Drywell Head Cavity

The drywell head cavity has been analyzed for specific line breaks. These were: (1) a break of the recirculation outlet line within the drywell, (2) a break of the main steamline in the drywell, and (3) a break of the head spray line within the head cavity. These analyses have been carried out to establish the pressure differentials that would exist across the refueling bulkhead as a result of these accident conditions. These analyses were performed at original licensed power conditions (2952 MWt) and the structures were designed using differential pressures that bounded those calculated. An assessment of the differential pressures was performed for the current licensed power. The differential pressures used for the structure design were determined to envelop the expected differential pressures for current licensed power. The differential pressures used for design are included in below and in Tables 6.2-12 and 13. A break of the recirculation outlet line was found to produce a higher pressure differential across the refueling bulkhead than a break of the main steamline, a value of 5.49 psid upward. The head spray line break resulted in a pressure differential of 8.21 psid downward. The main steamline break data are not presented due to the fact that the recirculation outlet line break produces the highest upward differential pressure.

The break size, mass flow rate, and energy content for the recirculation line are identified in Subsection 6.2.1.2.3 and Table 6.2-10. 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 break were determined using Moody's flow through the 3.72-inch head spray nozzle at the reactor conditions with a Moody multiplier of 1.0. The rate, break size, and energy content is presented in Table 6.2-11. Flow from the other side of the head spray line break was neglected.

The factor, 1.4, required by Paragraph II.5 of Standard Review Plan 6.2.1.2 was not applied to the results since the analysis was done to establish final design margin. Adequate pressure margin does exist as is indicted in Tables 6.2-12 and 6.2-13.

6.2.1.2.1.2 Shield Annulus

Pressure transients within the biological shield annulus are important from two considerations: (1) determination of the design conditions for the shield wall, and (2) determination of the tipping forces on the reactor pressure vessel and associated piping. The results of calculations that determine the design conditions for the shield wall are presented below. The results of the analyses that determine the tipping forces on the reactor pressure vessel and associated piping appear in Sections 5.3 and 5.4.

It is not clear that one line break will yield the most severe conditions for both considerations. Therefore, the consequences of three line breaks were studied: (1) a complete circumferential fracture of one of the two recirculation outlet lines at the safe-end-to-pipe weld, (2) a complete circumferential fracture of one of the four feedwater lines at the safe-end-to-pipe weld and (3) a complete circumferential of one of the ten recirculation inlet lines at the safe-end-to-pipe weld. The pressure transients resulting from these postulated breaks were used in the determination of the adequacy of the shield wall, vessel, and associated piping.

The pressurization analyses for the postulated breaks in the recirculation inlet, recirculation outlet, and feedwater lines were based on the nodalization schemes depicted in Figures 6.2-21, 6.2-22, and 6.2-148. The results of the analyses performed at 2952 MWt are described in Sections 6.2.1.2.1.2.1 through 6.2.1.2.1.2.3. An assessment of the analyses determined that the mass and energy releases at 3473 MWt are bounded by those determined at 2952 MWt. Therefore, the differential pressures provided in Table 6.2-14 are still bounding.

6.2.1.2.1.2.1 Recirculation Outlet Line Break

The injection of initially subcooled liquid into the annulus results in a significant fraction of the liquid flashing to steam and pressurization the annulus. The responses of the break volume and adjoining nodes are shown in Figures 6.2-23 through 6.2-54. Within 20 milliseconds after the postulated break, flow out of the flow diverter (see Subsection 6.2.1.2.2.2.1 for flow diverter description) is choked. Approximately 20 milliseconds later, the pressure in the flow diverter and the pressure in the surrounding annulus nodes peak, reflecting subcooling and inventory effects addressed in blowdown flow rates. Flow in the annulus initially proceeds in all directions, but soon swings preferentially upward in response to increasing pressure within the dead-ended reactor skirt region. By 20 milliseconds into the transient, the pressures in and about the penetration have stabilized and shortly thereafter the differential pressures across the shield node. The peak pressure in the flow diverter, however, reaches 635.0 psid. The peak differential pressure for all nodes are given in Table 6.2-14.

As stated in Standard Review Plan 6.2.1.2 (II.1), the initial conditions "assume air at the maximum allowable temperature, minimum absolute pressure, and zero percent relative humidity."

The RELAP4/MOD3 code will not accept zero percent relative humidity. Therefore, a reasonable approximation to zero percent relative humidity that produces a stable solution (0.1 percent) was used in the calculation.

The subcompartment under consideration is the biological shield annulus. One side of this subcompartment is the RPV. The other side of the subcompartment is the reflective thermal insulation on the inside of the biological shield wall. Therefore, the air temperature is conservatively assumed to be at the maximum surface temperature of the RPV, i.e., 528°F.

The operating design thermal limits for the biological shield are as follows:

$T_{Inside\ Top}$ - 144°F　　$T_{Outside\ Top}$ - 139°F

$T_{Inside\ Bottom}$ - 118°F　　$T_{Outside\ Bottom}$ - 128°F

The design thermal limits are obtainable because the annulus between the reflective thermal insulation and the shield wall has HVAC flow. Therefore, the temperature in this region is kept within the thermal limits of the shield wall, see Figures 6.2-57, 6.2-58 and Drawing M05-1109.

The governing thermal loading for accident condition is after a small line break accident and the design temperatures of this condition are:

$T_{Inside\ Top}$ - 97°F　　$T_{Outside\ Top}$ - 280°F

$T_{Inside\ Bottom}$ - 70°F　　$T_{Outside\ Bottom}$ - 278°F

(Q&R 480.02)

The break for the recirculation outlet line was assumed to be a guillotine rupture of the pipe with a time dependent displacement of the pipe. The flow area changes with time as detailed in Reference 5. This area includes the total cross-sectional area of the pipe plus the area of the jet pump nozzles and the cross-sectional area of the cleanup line.

The mass and energy release is calculated by use of the GE Short-Term Energy Method accounting for the time dependent break area as described in Reference 5.

6.2.1.2.1.2.2 Feedwater Line Break

Pressurization effects from the postulated feedwater line break are much less pronounced than for the recirculation outlet line break. Much of the injected fluid finds its way up and out of the annulus and over the top of the shield wall into the drywell. Nevertheless, the differential pressure across the shield wall at nodes surrounding the break node peaks at 20.6 psid, while the differential pressure across the break node reaches 127.1 psid. By 0.5 seconds into the transient, all the differential pressures across the shield wall have peaked and are decreasing.

Reactor shield wall and reactor pressure vessel details are discussed in Sections 3.8 and 5.3, respectively.

The feedwater line break was assumed to be an instantaneous double-ended break. The break model included the effects of subcooled liquid inventory in the determination with the short-term release method specified by GE.

6.2.1.2.1.2.3 Recirculation Inlet Line Break

The presence of the flow diverter on the recirculation outlet line significantly mitigates the pressure transient within the shield annulus resulting from a postulated break in the recirculation outlet line. Therefore, the recirculation inlet line break is also considered as part of the design basis because the pressure transient from these unguarded lines has the potential for producing the design controlling loads on the vessel skirt and the lower portion of the shield.

The pressurization effects of the recirculation inlet line break are quite similar to those of the recirculation outlet line break discussed in Subsection 6.2.1.2.1.2.1. The peak differential pressure at nodes surrounding the break node reaches 32.3 psid while the differential pressure at the break node is calculated to be 76.9 psid. By 0.5 seconds after the break occurs the pressure at all nodes has peaked and begun to decrease.

The break postulated for this analysis is an instantaneous double-ended guillotine break of one of the recirculation pump discharge lines at the RPV nozzle safe-end-to-pipe weld. The break flow is determined by the short-term mass-energy release method (Reference 5).

6.2.1.2.1.3 Containment Pipe Tunnel

The containment pipe tunnel has been analyzed for an RWCU line break. This analysis was specifically carried out to establish the differential pressure that would exist in the pipe tunnel or connected subcompartments as a result of this postulated accident condition. The results are below the design differential pressures.

The RWCU line break in the containment pipe tunnel was found to produce a maximum differential pressure of <3 psid. This maximum differential pressure is below the design value of 5.11 psid.

The postulated break of the RWCU line in the containment pipe tunnel was modeled as a double-ended guillotine break of the RWCU line. Mass flow into the subcompartment is terminated by the automatic isolation of the suction and discharge lines upon detection of the leak by either differential temperature in the containment pipe tunnel or unbalanced flow in the RWCU system. The maximum total mass released consists of the RWCU system inventory plus the mass added from the reactor pressure vessel from the upstream side of the break. The blowdown mass and energy release rates associated with the upstream and downstream sides of the break are given in Table 6.2-16. A choked flow Moody multiplier of 0.6 is used for all flow paths other than the break flow path.

6.2.1.2.1.4 RWCU Heat Exchanger Subcompartments

The RWCU heat exchanger subcompartments were analyzed for an RWCU line break. This analysis was performed to determine the maximum resulting differential pressure within these subcompartments. The line break analysis indicated a maximum pressure differential of <4 psid. The design pressure is 4.90 psid.

A brief discussion of the modeling of this study is discussed in Subsection 6.2.1.2.1.3 and the transient mass and energy flow rates utilized in the break analysis are presented in Table 6.2-17.

6.2.1.2.1.5 RWCU Valve Subcompartments

Analysis was performed for a postulated RWCU line break in the RWCU valve subcompartments to determine the design pressure margin of the subcompartments' design. The maximum differential pressure, <4 psid. The design value is 4.90 psid.

The modeling of the break analysis is presented in Subsection 6.2.1.2.1.3 and the transient mass and energy flow rates utilized in the analysis are given in Table 6.2-17.

6.2.1.2.1.6 RWCU Crossover Pipe Tunnel

A postulated RWCU line break in the RWCU crossover pipe tunnel was investigated to verify the adequacy of the tunnel's design. The tunnel has been evaluated for the maximum differential pressure of 8.6 psid.

The modeling of the break analysis is discussed in Subsection 6.2.1.2.1.3 and a further discussion of the calculations are presented in Subsection 6.2.1.2.3. The mass and energy flux terms are given in Table 6.2-18.

6.2.1.2.1.7 RWCU Filter-Demineralizer Holding Pump Subcompartment

A transient analysis was performed for a postulated RWCU line break in the RWCU filter-demineralizer holding pump room. This subcompartment has been designed for the maximum differential pressure of 4.1 psid. Subsection 6.2.1.2.1.3 discusses the modeling of the transient analysis. The mass and energy release rates are described in Table 6.2-18, and a detailed description of the accident analysis is presented in Subsection 6.2.1.2.3.

6.2.1.2.1.8 RWCU Filter-Demineralizer Subcompartments

The RWCU filter-demineralizer subcompartments were investigated for a postulated double-ended guillotine rupture of the RWCU line in these subcompartments. Peak differential pressure is 12.7 psid. The subcompartment design differential pressure is 21.0 psid.

Subsection 6.2.1.2.3 discusses the details of the analysis and Subsection 6.2.1.2.1.3 summarizes the assumptions relating to the modeling of the mass and energy release rates. Table 6.2-18 presents the specific mass and energy blowdown fluxes used in the analysis.

6.2.1.2.1.9 RWCU Filter-Demineralizer Valve Subcompartment

An RWCU line break in the RWCU filter-demineralizer valve room was investigated and a peak differential pressure of <5 psid was determined. The peak calculated differential pressure is below the design differential pressure of 6.0 psid.

The mass and energy flux terms used in this analysis are described in Table 6.2-18 with the supporting justification presented in Subsection 6.2.1.2.1.3. Subsection 6.2.1.2.3 discusses the particular analytical model used in the analysis.

6.2.1.2.2 Design Features

6.2.1.2.2.1 Drywell Head Cavity

The drywell head cavity is the annular volume between the reactor pressure vessel head, the drywell head, and the drywell bulkhead. The reactor pressure vessel head is considered to be hemispherical with a radius of 9.4 feet (see Drawing M01-1111-4). The drywell head is composed of a dome described as 2:1 ellipse of revolution and cylindrical base. The diameter of the cylindrical base and the major axis of the elliptical dome is 29.7 feet. The bulkhead encloses the volume by connecting the base of the drywell head to the reactor pressure vessel. The reactor pressure vessel head is enclosed in insulation that forms cylindrical volumes as shown in Drawing M01-1111-4.

Within the drywell head cavity there is a single high energy line; the head spray line. The routing of this line is shown in Drawing M27-1314. The bulkhead contains a number of penetrations as shown in Drawing M27-1314. The most significant vent area is associated with the HVAC inlet and outlet penetrations. The inlet penetrations are 12 inches in diameter and are connected to HVAC supply ducts that are assumed to remain in place during the postulated accident. The outlet penetrations exhaust vents or equivalent total-area openings shall be as a minimum equivalent to the path description specified in Table 6.2-19. Details of the geometry of the supply and exhaust vents are shown in Drawing M27-1314. The vent areas, compartment volumes, vent loss coefficients, etc., are tabulated in Tables 6.2-12 and 6.2-19, and Tables 6.2-13 and 6.2-20 for the postulated head spray line break in the head cavity and recirculation outlet line break in the drywell respectively.

6.2.1.2.2.2 Shield Annulus

The biological shield annulus is the volume between the reactor pressure vessel and the biological shield, as shown in Figures 6.2-57, 6.2-58 and Drawing M01-1111-4.

The biological shield annulus is approximately 36 inches in width and approximately 51 feet in height. The shield wall contains 24 major mechanical penetrations for piping to the reactor pressure vessel as well as a number of smaller instrumentation and HVAC lines. A total of nine personnel openings, each 2.5 by 4.0 feet, are provided above and below the reactor core region and, like nozzle penetrations, require no shielding doors. The 3.8 inch thick RPV side shell thermal insulation is spaced 3.0 inches from the shield wall and extends from the service walk below the recirculation lines to the head cavity seal. Steel gratings at several elevations within the annulus provide access to nozzle assemblies and primary systems welds during inservice inspection (ISI). These ISI platforms are rotated to a vertical position during plant operation in order to minimize obstructions in the annulus.

Within the shield annulus, all lines are high energy lines. A double-ended guillotine rupture of a feedwater line, a recirculation outlet line with flow diverter, and a recirculation inlet line have been investigated.

6.2.1.2.2.2.1 Recirculation Outlet Line Break

The recirculation outlet lines are each equipped with a flow diverter to ensure a maximum of 15% bypass flow into the shield annulus. Eighty-five percent of the flow is diverted into the drywell (see Figure 6.2-59). The vessel thermal insulation is assumed to remain in place throughout the transient thereby limiting the available free volume within the annulus. The vessel insulation above the shield wall is dislodged and blown into the drywell when an outward directed pressure differential of 1.5 psi is developed. The ISI platforms are rotated to their vertical positions except for those at elevations of 752 feet 3-1/2 inches and 794 feet 1/2 inch which are assumed fixed as shown in Figure 6.2-57.

A complete review of all volume and junction parameters is given in Tables 6.2-14 and 6.2-21.

6.2.1.2.2.2.2 Feedwater Line Break

In the feedwater break analysis, the three inspection openings at elevation 778 feet 3-1/2 inches are considered to open linearly with the pressure rise in the shield annulus. The openings are assumed to begin opening when the differential pressure between the annulus and the drywell at the elevation of the opening reaches 1.0 psi. The vent is assumed to be full-open when the differential pressure across the opening is 2.0 psi. The remaining inspection openings are not considered as vent areas in the feedwater break analysis. Further, no credit is taken for the vent areas associated with the mechanical and HVAC penetrations.

All ISI platforms are assumed to be rotated to the vertical position except the service walk at elevation 752 feet 3-1/2 inches which is assumed fixed as shown in Figure 6.2-57.

Each section of the horizontal closure ring insulation below the service walk at the bottom of the reactor vessel starts to bend downward at a differential pressure of 1.0 psid and is fully bent downward at 2.0 psid to allow restricted passage of flow into the reactor cavity. The remainder of the insulation within the shield annulus is conservatively assumed to remain in place.

A complete review of all input parameters may be found in Tables 6.2-22 and 6.2-23.

6.2.1.2.2.2.3 Recirculation Inlet Line Break

This analysis examines the shield annulus pressurization due to the double-ended guillotine rupture of a recirculation inlet line. The recirculation inlet line chosen for analysis is located at 90° azimuth as shown in Figure 6.2-57. The factors contributing to the selection of this line are the proximity of the inservice inspection platforms at elevation 762 feet 0 inches and the nozzle penetration (N9 in Figure 6.2-149) adjacent to this line.

Thermal insulation is installed adjacent to the shield wall as shown in Figures 6.2-57 through 6.2-59. This insulation is conservatively assumed to remain in place during the transient, thus limiting the free volume of the annulus and preventing the venting of the annulus through the shield wall penetrations and inspection openings.

The horizontal insulation above the reactor skirt is instantaneously displaced downward at the start of the transient and the free volumes of the nodes in the skirt area are taken as 50% of the free volume in the absence of this insulation.

The insulation above the shield wall is assumed to blow off (into the drywell) when the differential pressure between the upper nodes and the drywell reaches 2 psid. The resulting flow path is the only flow path for venting the annulus to the drywell during the transient.

The in-service inspection platforms are assumed rotated to the vertical position except the platforms at elevations 752 feet 3-1/2 inches and 794 feet 1/2 inch. These platforms are assumed fixed and are represented as nodal boundaries (see Figure 6.2-149).

Complete nodal volume, flow path and flow area information is given in Tables 6.2-67 and 6.2-68.

6.2.1.2.2.3 Containment Pipe Tunnel

The containment pipe tunnel is located between the drywell and containment walls and serves as an enclosure for protecting the major NSSS piping (main steam, feedwater, etc.) from the effects of LOCA related transients within the containment. The containment pipe tunnel is approximately 33.6 feet long, 22.0 feet wide and 34 feet high as shown in Drawings M27-1311 and M27-1312. The containment pipe tunnel is connected to the containment by two labyrinth passages adjacent to the drywell wall.

All high energy lines within the containment pipe tunnel are enclosed in guard pipes with the exception of the RWCU pump discharge line. A guillotine break in this line was analyzed. Tables 6.2-15 and 6.2-24 provide a tabulation of the volumes, vent areas, vent loss coefficients, etc.

6.2.1.2.2.4 RWCU Heat Exchanger Subcompartments

The RWCU heat exchanger rooms are located in the containment immediately above the containment pipe tunnel as shown in Drawings M27-1311 and M27-1312. The irregularly shaped cubicles are approximated by a volume that is approximately 15.2 feet wide, 16.6 feet long, and 24.0 feet high above elevation 800 feet and a volume that is approximately 6.7 feet wide, 20.7 feet long, and 11.2 feet high below elevation 800 feet. The heat exchanger rooms are vented through the adjacent RWCU valve rooms and a labyrinth to the containment. In addition, RWCU heat exchanger Room No. 1 is vented to the RWCU crossover pipe tunnel but this vent area is ignored. The major equipment in these cubicles are the RWCU heat exchangers and their associated piping. Tables 6.2-25 through 6.2-28 provide a summary of the volumes, vent areas, etc., for these rooms.

6.2.1.2.2.5 RWCU Valve Subcompartments

The RWCU valve rooms are located adjacent to the RWCU heat exchanger rooms as shown in Drawings M27-1311 and M27-1312. These rooms are approximately 21.0 feet long, 6.6 feet wide, and 8.9 feet high. Each room contains RWCU lines, valves, and associated instrumentation. Each room is vented to the drywell through a labyrinth on one end and an RWCU heat exchanger room on the other end. The volumes, vent areas, etc., for these rooms are tabulated in Tables 6.2-29 through 6.2-32.

6.2.1.2.2.6 RWCU Crossover Pipe Tunnel

The RWCU crossover pipe tunnel is a pipe chase for the RWCU system connecting the RWCU heat exchanger rooms with the RWCU filter-demineralizer valve cubicle, as shown in Drawing M27-1305. The RWCU crossover pipe tunnel is approximately 2.3 feet wide, 14.2 feet high, and 16.3 feet long. The pipe tunnel is vented on one end to the RWCU heat exchanger room and to the RWCU valve cubicle on the other end. The vent area to the heat exchanger room is ignored. Tables 6.2-33 and 6.2-34 provide the modeling information for the pipe tunnel.

6.2.1.2.2.7 RWCU Filter-Demineralizer Holding Pump Subcompartment

The RWCU filter-demineralizer holding pump room is situated as shown in Drawing M27-1306. The cubicle is approximately 11.0 feet wide, 30.2 feet long, and 11.3 feet high. The holding pump room contains the RWCU filter-demineralizer holding pumps and associated piping. The cubicle is vented to the filter demineralizer valve room and the containment. Data for vent areas, vent coefficients, volumes, etc., are presented in Tables 6.2-35 and 6.2-36.

6.2.1.2.2.8 RWCU Filter-Demineralizer Subcompartments

The RWCU filter-demineralizer rooms are located adjacent to the RWCU filter-demineralizer valve and holding pump rooms as depicted in Drawing M27-1306. Filter-demineralizer Room No. 1 is approximately 9.5 feet wide, 11.7 feet long, and 21.6 feet high. Filter-demineralizer Room No. 2 is approximately 9.5 feet wide, 15.4 feet long, and 21.6 feet high. These rooms contain the filter-demineralizer unit for the RWCU system and associated piping and instrumentation. These rooms are vented to the RWCU filter-demineralizer valve room and containment. The filter-demineralizer rooms have removable slabs in the roof of each room large enough to remove the major equipment within the rooms. The modeling data for these rooms are presented in Tables 6.2-37 and 6.2-38.

6.2.1.2.2.9 RWCU Filter-Demineralizer Valve Subcompartment

The RWCU valve room is situated as shown in Drawing M27-1305. The room contains the valves for the RWCU filter-demineralizer system, the RWCU piping and associated instrumentation and controls. The valve room is vented to the filter-demineralizer rooms, the holding pump room, and the crossover pipe tunnel. Data for the vent areas, vent loss coefficients, volumes, etc., are presented in Table 6.2-39 and 6.2-40.

6.2.1.2.3 Design Evaluation

To calculate the forces and moments upon the reactor pressure vessel (RPV) and biological shield wall (BSW) a beam model was devised as illustrated in Figures 6.2-180 and 6.2-181. The force and moment time histories were developed within the volumes shown in Figures 6.2-182 and 6.2-183, which also depict the coordinate system utilized. Azimuth location models are illustrated in Figures 6.2-184 and 6.2-185. The force and moment time histories for the RPV and the BSW are shown in Figures 6.2-186 through 6.2-197. These forces were resolved by the General Electric developed computer code GEAPL. This code considers a pressure time history within each volume as defined by the RELAP 4/MOD5 analysis and resolves it at each instant of time into equivalent centerline forces and moments at each affected node point in the beam model. The data for the projected area used to calculate these loads is contained in Tables 6.2-70 through 6.2-73. These tables and the information contained in Figures 6.2-182 and 6.2-183 can be used for the confirmatory calculation. The moment arms were calculated relative to a reference point, which, as measured above RPV datum (elevation 744.059 ft.), are 135.5 in. and 484.5 in. for the recirculation line and feedwater line nodalization, respectively. (Q&R 480.02)

6.2.1.2.3.1 Drywell Head Cavity

The drywell bulkhead is subject to pressure differentials when a head spray line or a recirculation suction line break is postulated. The directions in which these pressure differentials act are counter to one another. Thus, the consideration of each break is required to properly determine the design conditions for the bulkhead. The analyses that have been performed to evaluate these design pressure differentials are discussed in the following.

6.2.1.2.3.1.1 Head Spray Line Break

The head spray line break was assumed to be an instantaneous guillotine fracture of the head spray line at the reactor pressure vessel head. The head spray nozzle presents the minimum flow area in the flow path from the reactor pressure vessel to the head cavity after the occurrence of the break. The mass and energy release rates for this postulated break were calculated using the Moody table data from RELAP4/MOD3 with the assumed reactor pressure of 1060.0 psia and steam enthalpy of 1190.0 Btu/lbm as given in Table 6.2-11. The maximum mass velocity calculated through the 3.72 inch diameter head spray nozzle with a Moody multiplier of 1.0 is 2207.0 lbm/sec-ft^2. This mass velocity is conservatively assumed to remain constant throughout the transient.

The transient was modeled utilizing the RELAP4/MOD3 computer code. A three-node system was used to represent the head cavity, drywell, and wetwell along with two vent paths between the three nodes as shown in Figure 6.2-110. The vent path between the head cavity and drywell represents the bulkhead vents. As a result of the ductwork attached to three of the six bulkhead vents, the ductwork was assumed to completely block flow through three vents, and therefore, only three vents were considered open to flow during the transient. The flow area and properties of the three vents were combined into one large vent path. The second vent path was between the drywell and containment and represents the drywell vent system. Since the head spray line break is a small break and results in a relatively slow pressurization of the drywell, a valve was placed in the flow path and was dependent upon the drywell pressure exceeding the hydrostatic head at the drywell vent system exit.

The head cavity volume was calculated with the head insulation in place, yielding a minimum volume. In order to maximize the differential pressure across the bulkhead, the drywell volume was chosen as the maximum volume, i.e., low water level volume. The containment volume used, corresponded to the suppression pool level dictated by the drywell volume. See Table 6.2-12 for the nodal information.

The vent path between the head cavity and drywell was modeled as the combination of an entrance into a pipe removed from a wall, and an exit of a pipe removed from a wall. The corresponding loss coefficients were 0.72 and 4.95 (Reference 6), which yielded a total loss coefficient of 5.67. The vent area was determined to be 6.55 ft^2. Table 6.2-19 lists all of the vent parameters.

The RELAP4/MOD3 computer program does not allow for air flow between volumes. In order to closely represent the true conditions, the volumes were input as initially filled with a water-steam mixture of appropriate quality to match the density and pressure in the actual volumes. The initial conditions are listed in the nodal parameter description shown in Table 6.2-12.

The maximum downward differential pressure across the refueling bulkhead of 8.21 psid was reached 2.35 seconds after the break was initiated. See Figures 6.2-64 through 6.2-66 for the pressure transients of the three nodes. In addition, Figure 6.2-67 presents the differential pressure history across the bulkhead. Flow in all the vent paths was less than sonic up through the time the peak differential pressure was reached.

6.2.1.2.3.1.2 <u>Recirculation Outlet Line Break</u>

The recirculation line break was assumed to be an instantaneous guillotine rupture of the recirculation suction line at the outside the shield wall. The mass and energy rates for this postulated break are shown in Table 6.2-10 and account for liquid flow only until 19.0 seconds after start of the transient. Since the accident transient is very short in duration, the liquid enthalpy in Table 6.2-10 was used.

The transient was modeled utilizing the RELAP4/MOD3 computer code. The head cavity, drywell, and containment were modeled in the same way as described in Subsection 6.2.1.2.3.1.1, and as shown in Figure 6.2-68. The modeling of the drywell vent clearing and initiation of flow was done by placing a valve in the vent path. A previous study found that the clearing time was 0.7 seconds for the recirculation outlet line break. From this, the valve in the flow path was opened 0.7 seconds after the line break.

In this case, the head cavity volume was maximized by not considering the volume displaced by the reactor head insulation and the drywell volume was minimized by considering the suppression pool at high water level in order to increase the maximum upward differential pressure. The containment volume used in this analysis was smaller than actually exists. See Table 6.2-13 for the nodal information.

The vent path between the head cavity and drywell in the upward direction was modeled as an entrance to a sharp-edged orifice and exit from a pipe. The corresponding loss coefficients were determined to be 2.66 and 1.0 (Reference 6), and the total loss coefficient was then calculated to be 3.66. The flow area for the vent path was 6.55 ft^2. Table 6.2-20 describes all the vent path parameters.

The initial conditions were determined in the same manner as described in Subsection 6.2.1.2.3.1.1.

The maximum upward differential pressure across the refueling bulkhead of 5.49 psid was reached 0.77 seconds after the break. The flow through the vent paths remained unchoked throughout the pressure transient. Figure 6.2-72 shows the differential pressure transient across the bulkhead for this accident.

6.2.1.2.3.2 Shield Annulus

The biological shield wall in the drywell is subject to pressure differentials when a recirculation suction line, recirculation discharge line, or feedwater line breaks within the annular space between the reactor pressure vessel and the biological shield wall is postulated.

Consideration of each break is required to determine properly the design conditions for the shield wall. The analyses that have been performed to evaluate these design pressure differentials are discussed below.

A generic sensitivity study has been performed to demonstrate the appropriateness of the nodalization scheme and nodal sizes selected for evaluation of line breaks in the shield annulus. The RELAP4/MOD3 computer code was used to perform the analyses. The assumptions made in modeling the problem were in accordance with the applicable NRC guidelines. The mass and energy blowdown rates were determined according to the methods described in Reference 5.

In subsonic flow conditions, two flow mode!s were used, as defined for RELAP4/MOD3: (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 was 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 6.2-73 and 6.2-74). Therefore, the assumptions were made that since RELAP4/MOD3 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 biological shield nodalization. Using the original nodalization (Figure 6.2-75) 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 6.2-76). This model demonstrated identical pressure response close to the break and only minor differences away from the break (Figures 6.2-77 and 6.2-78). 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 6.2-79). The pressure response was not consistent with the original runs (Figures 6.2-80 and 6.2-81). 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 6.2-82). 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 6.2-83 and 6.2-84).

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 RELAP4 model would not result in a significant change in the results.

6.2.1.2.3.2.1 Recirculation Outlet Line Break

The recirculation line break was assumed to be an instantaneous guillotine rupture of the recirculation suction line in the annular space between the reactor pressure vessel and the biological shield wall with flow diverters incorporated on the recirculation suction lines.

The physical system, described previously, was modeled for analysis with the RELAP4/MOD5 computer code. The mass and energy release rates were determined by a generic method for short-term mass energy release supplied by General Electric and is tabulated in Table 6.2-41. Each of the recirculation suction lines were equipped with a flow diverter which provided for 15% bypass flow into the shield annulus. It was assumed that the vessel insulation remained in place throughout the transient; thereby limiting the available free volume within the annulus. Any venting of the annulus through openings in the shield wall was ignored. In addition, the vessel insulation above the shield wall was assumed to blow away into the drywell when an outwardly directed differential pressure of greater than 1.5 psid developed across it. The inservice inspection (ISI) platforms are rotated to their vertical positions, except those at elevations 752 feet 3-1/2 inches and 794 feet 1/2 inch which are assumed fixed as shown in Figures 6.2-57 and 6.2-58. The annulus pressure response will be circumferentially symmetric with respect to the break plane and therefore only one-half of the annulus was modeled. The sonic flow conditions away from the break were adequately accounted for with the Moody Slip Model and a Moody area multiplier of 0.6. However, a Moody multiplier of 1.0 was utilized at the break node junctions. The initial conditions within each subcompartment corresponded to the maximum operating temperature and minimum operating pressure cited in the plant design criteria along with a relative humidity approximately equal to 0.0%. The initial conditions are listed in Table 6.2-14.

Nodalization of the biological shield annulus was determined on the basis of natural geometric boundaries and the constraint that the pressure and/or temperature gradients within a node be minimized. Consistent with this philosophy an analytical model was developed consisting of 34 nodes and 61 flow paths (see Figures 6.2-21 and 6.2-85). The incorporation of the flow diverter into the model affected a reduction in the model's size but not its sensitivity.

As the vessel insulation within the annulus was assumed to remain fixed throughout the duration of the transient, no insulation-plugging was considered. The volume of each node was calculated as a net free volume, i.e., the respective volume of the annulus less any included nozzle piping and/or structural steel (see Table 6.2-14). The determination of individual vent path characteristics was made assuming homogeneous, fully developed, incompressible flow. All partial loss coefficients, k_i's were derived from Reference 6. The total loss coefficient k_t was then determined by adding the weighted partial loss coefficients in series:

$$k_t = \sum_i k_i \frac{A_t^2}{A_i}$$

where A_t is the junction area and A_i is the area within the junction

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:

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

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

where d is a "characteristic" diameter of the constriction of length L and with an area A (for an orifice the characteristic diameter is taken to be the diameter of the orifice). L_1, A_1, L_2, and A_2 are the length and flow area of the conduit partitioned by the constriction.

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: 22, 46, and 60.

Junction 22 represents the horizontal path connecting nodes 12 and 13. The junction area is the minimum cross-sectional area in the path between the geometric centers of the two nodes and equals the nominal cross-sectional area less the area included by one-half of a N2 nozzle, i.e., approximately 15.00 ft^2. The inertia coefficient, I, was calculated as the distance between the node centers divided by the junction area which gives a value of 0.40 ft^{-1}. The total loss coefficient, k_t, represents the combined losses due to skin friction and a single piping obstruction. The first part was evaluated using Diagram 6-2 of Reference 6 at 0.12 for an area of 16.89 ft^2. The loss due to the presence of the nozzle was estimated with Diagram 4-16 of Reference 6 at 0.02 for an area of 15.00 ft^2. The total loss coefficient, then, based on a junction area of 15.00 ft^2 is 0.11.

Junction 46 represents an axial flow path connecting nodes 24 and 29. The junction area is equal to one-eighth of the total cross-sectional area of the annulus less one-half of the area occupied by a N4 nozzle, yielding a flow area of 21.77 ft^2. An inertia coefficient of 0.48 ft^{-1} was calculated as before. The total loss coefficient consisted of a friction loss, K_1, and a local loss, K_2. K_1 was determined using Diagram 2-3 of Reference 2 to be 0.034 for an area of 23.95 ft^2. K_2 was found to, be 0.10 for 21.77 ft^2 using Diagram 4-16 of Reference 6. The sum of these loads gives 0.040 for a junction area of 21.77 ft^2.

Junction 60 connects the flow diverter, node 33, with the drywell, node 34. The junction area equals one-half of the annular area between the recirculation outlet line with insulation and the surrounding penetration liner, or approximately 1.88 ft^2. The inertia coefficient was determined using the three-part orifice equation described earlier and resulted in an I = 2.10 ft-1. The loss coefficient of 1.35 for 1.88 ft^2 was found using Diagram 11-28 of Reference 6 and accounts for the sudden contraction and expansion of the flow as it passes through the penetration and out into the drywell.

A complete review of all the junction parameters used in this analysis is given in Table 6.2-21. No sonic flow was expected or observed within the annulus except that issuing from the diverter. This 15% bypass flow recommended by GE and incorporated in the flow diverter design constitutes the effective break flow for this analysis and thus was treated using the Moody Slip Model with a Moody multiplier of 1.0 instead of the 0.6 value normally input. Note that the above treatment was also applied to the diverted flow into the drywell in order to preserve the recommended flow ratio.

The incorporation of flow diverters on the recirculation suction lines effectively throttle slowdown into the annulus and increase the blowdown to the drywell. The result is a significant reduction in the pressurization of the annulus. As the break flow into the flow diverter volume is increased, so does the node pressure, which reaches a quasi-steady-state value of approximately 650 psia (see Figure 6.2-23) when full break flow is established. Within 20 milliseconds following blowdown commencement, all flows out of the diverter have choked. The maximum pressure differentials across the biological shield wall were no greater than 6.10 psid.

See Figures 6.2-24 through 6.2-54 for the differential pressure transients across the shield wall, and Table 6.2-14 for all the peak differential pressures that were reached.

6.2.1.2.3.2.2 Feedwater Line Break

The feedwater line break was assumed to be an instantaneous guillotine rupture of the feedwater line and the break occurred in the annular space between the reactor pressure vessel and the biological shield wall.

The physical system just described was modeled for analysis with the WARLOC computer code. The mass and energy release rate was determined by a generic method for short-term mass/energy release supplied by General Electric and is tabulated in Table 6.2-42. All of the breakflow was assumed to enter the annulus. The three inspection openings above elevation

778 feet 3-1/2 inches were considered to be opened linearly at a pressure differential of 1.0 psid and to be fully open at 2.0 psid. No allowance was made for flow out of the other five personnel openings below the 778 feet 3-1/2 inches elevation, and no flow was assumed out of any mechanical or HVAC penetrations in the shield wall. The inservice inspection (ISI) platforms were rotated to their vertical positions with the exception of the service walk at elevation 752 feet 3-1/2 inches. In addition, it was assumed that each section of the horizontal closure ring insulation below the service walk started to bend downward at a differential pressure of 1.0 psid and is fully bent downward at 2.0 psid to allow restricted passage of flow into the reactor cavity. No credit was taken for increases in free volume and/or flow area due to loss of insulation materials. Moreover, the sonic flow conditions away from the break were adequately accounted for by the flow out of the break. The initial conditions within the annulus are listed in Table 6.2-22.

The nodalization of the feedwater break utilized an asymmetric model which modeled the complete annulus (360°) and was made up of 34 nodes and 75 flow paths (see Figures 6.2-22 and 6.2-86, respectively). Accordingly, a coarse mesh was used away from the break while an increasingly finer mesh was used nearer the break where substantial pressure gradients would be expected.

The calculation of volume and junction characteristics was performed in a manner similar to that outlined in the recirculation line analysis in Subsection 6.2.1.2.3.2.1. A complete review of all input parameters may be found in Tables 6.2-22 and 6.2-23.

The rupture of a feedwater line within the annular space between the RPV and the biological shield wall produced a short-lived highly localized pressure transient.

The small flow losses and inertial effects presented by the 3-foot wide annulus coupled with the sudden step changes in the blowdown flow rate resulted in a rather rapid transient and a quick pressure relief. The maximum differential pressure reached in the transient was 127.1 psid and was observed in the break node. The differential pressure transients for all of the nodes are presented in Figures 6.2-87 through 6.2-93 and the maximum peak differential pressures reached during the transient are tabulated in Table 6.2-22.

6.2.1.2.3.2.3 Recirculation Inlet Line Break

The recirculation inlet line break was assumed to be an instantaneous guillotine rupture of one of the recirculation inlet lines at the safe-end-to-pipe weld in the shield annulus.

The physical system described above was modeled for analysis with the RELAP4/MOD5 computer code. The mass energy release rate at original licensed power (2894 MWt) was determined by a generic method for short-term mass/energy release supplied by General Electric (Reference 5) which had been modified to include the effects of subcooled liquid by substituting the more conservative Henry-Fauske critical flow model of Reference 16 for the Moody critical flow model of Reference 5. The flow from the pump side of the break is assumed to be limited by the recirculation inlet line area of 0.51 ft^2. After the pipe inventory has discharged through the break, the flow is assumed to be critical flow through the break area for the remainder of the transient. The flow on the reactor side of the break is limited by the jet pump nozzles after the inventory period. Each jet pump contains ten 1.13-inch diameter nozzles for a total flow area of 0.070 ft^2. Thus, the flow following the inventory period is critical flow for the remainder of the transient which is limited by the jet pump nozzle area. The resulting mass energy release determined at 2894 MWt is tabulated in Table 6.2-69.

Reference 5 recommends using the Moody subcooled critical flow model without modification for the Henry-Fauske model. The mass and energy releases provided in Table 6.2-69 bound the mass and energy release at 3473 MWt determined using the Moody subcooled critical flow model.

The flow in the annulus during the transient is assumed homogeneous and incompressible, i.e., the phases are completely intermixed and momentum flux distributions are negligible. The internodal flow away from the break location is assumed choked and adequately represented by the Moody slip model and an area multiplier of 0.6.

Since loss coefficients for rapid expansions or contractions are substantially larger than those due to either skin friction or turning losses, the nodalization of the annulus was determined primarily by the location of structures within the shield annulus which act as flow obstructions. For example, see the discussion of insulation and in-service inspection platform effects on nodal boundaries in Subsection 6.2.1.2.2.2.3. Due to the absence of major flow obstructions to horizontal flow, vertical boundaries between nodes were chosen to allow a finer nodalization near the break node. The break node was chosen so that its height and width would be comparable with the width of the annulus (3.5 feet by 3.5 feet by 2.7 feet).

In all, the model incorporated 29 nodes and 51 flow paths as shown in Figure 6.2-148. The volume of each node was calculated as the net free volume assuming the thermal insulation along the shield wall remained in place. The determination of vent path loss coefficients was made assuming homogeneous, fully developed, high Reynolds number flow from Reference 6. Inertia coefficients were calculated conservatively as L/A, where L is the distance between node centers. A summary of all volume and junction parameters is presented in Tables 6.2-67 and 6.2-68. The initial conditions are given in Table 6.2-67.

The small flow losses and inertial effects presented by the annulus coupled with the sudden step changes in the blowdown flow rate resulted in a rather rapid transient and quick pressure relief. The maximum differential pressure reached during the transient was 76.9 psid and was observed in the break node. The differential pressure transients for all of the nodes are presented in Figures 6.2-150 through 6.2-178. The maximum differential pressures for the nodes are tabulated in Table 6.2-67.

6.2.1.2.3.3 <u>Containment Pipe Tunnel</u>

The containment pipe tunnel is subject to pressure differentials when a break in the reactor water cleanup (RWCU) system within the pipe tunnel is postulated. Consideration of this break is required to properly determine the design conditions for the pipe tunnel.

The RWCU line break was assumed to be an instantaneous guillotine rupture of the RWCU line within the containment pipe tunnel. The maximum total mass and energy release consists of the RWCU inventory plus the mass and energy from the reactor pressure vessel from the upstream side of the break during the time before the suction and discharge lines have been isolated by motor-operated valves.

In addition, only hot water inventory was considered in determining the blowdown flowrates. The transient spectrum of the blowdown mass and energy rates associated with the upstream and downstream side of the break are presented in Table 6.2-16. A Moody choked flow multiplier of 0.6 was used for each flow path other than the break path which utilized a Moody multiplier of 1.0. The mixture of the blowdown fluid and air was considered as a homogeneous mixture with complete liquid carryover. The subcompartment volumes and initial conditions for the temperature, pressure and relative humidity are shown in Table 6.2-15.

The nodalization of the containment pipe tunnel required the nodalization of pipe tunnel and primary containment and is shown in Figure 6.2-94A. The determination of vent path characteristics was performed in a similar manner as described in Subsection 6.2.1.2.3.2.1. The nodal and vent path descriptions are tabulated in Tables 6.2-15 and 6.2-24.

The rupture of a RWCU line within the containment pipe tunnel resulted in a maximum differential pressure of <3 psid.

6.2.1.2.3.4 RWCU Heat Exchanger Subcompartments

The RWCU heat exchanger subcompartments are subject to pressure differentials when a break in a RWCU system within these subcompartments is postulated. There are two such heat exchanger subcompartments that require analysis in order to properly determine the design conditions of each subcompartment.

The modeling of this event is similar to that described in Subsection 6.2.1.2.3.3 and the transient mass and energy release rates utilized in the break analysis are presented in Table 6.2-17. The nodalization scheme is presented in Figure 6.2-96 for heat exchanger rooms. All of the nodal and vent path descriptions are tabulated in Tables 6.2-25, 6.2-26, 6.2-27, and 6.2-28.

The rupture of a RWCU line in either heat exchanger room resulted in a maximum differential pressure of <4 psid. The differential pressure transient for heat exchanger rooms is provided in Reference 28.

6.2.1.2.3.5 RWCU Valve Subcompartments

The RWCU valve subcompartments are subject to pressure differentials when a break in the RWCU system within these subcompartments is postulated. There are two such valve rooms that require analysis in order to properly determine the design conditions of each subcompartment.

The modeling of this event is similar to that described in Subsection 6.2.1.2.3.3 and the transient mass and energy release rates utilized in the break analysis are presented in Table 6.2-17. The nodalization scheme is presented in Figure 6.2-99. Nodal and vent path descriptions are tabulated in Tables 6.2-29, 6.2-30, 6.2-31, and 6.2-32.

The rupture of a RWCU line within the valve rooms resulted in a maximum differential pressure of <4 psid. The differential pressure transient for this event is provided in Reference 28.

6.2.1.2.3.6 RWCU Crossover Pipe Tunnel

The RWCU crossover pipe tunnel is subject to pressure differentials when a break in the RWCU system within the crossover pipe tunnel is postulated.

The modeling of this event is similar to that described in Subsection 6.2.1.2.3.3 and the transient mass and energy release rates utilized in the break analysis are presented in Table 6.2-18.

The nodalization scheme is illustrated in Figure 6.2-102. All of the nodal and vent path characteristics are shown in Tables 6.2-33 and 6.2-34 respectively.

The rupture of a RWCU line in the crossover pipe tunnel resulted in a maximum differential pressure of 8.6 psid. The differential pressure history for the accident is presented in Reference 28.

6.2.1.2.3.7 RWCU Filter-Demineralizer Holding Pump Subcompartment

The RWCU filter-demineralizer holding pump room is subject to pressure differentials when a break in the RWCU system within the filter-demineralizer holding pump room is postulated.

The modeling of this event is similar to that described in Subsection 6.2.1.2.3.6 and the transient mass and energy release rates are presented in Table 6.2-18. The nodalization is shown in Figure 6.2-104. The nodal and vent path characteristics are shown in Tables 6.2-35 and 6.2-36, respectively.

The rupture of a RWCU line in the filter-demineralizer holding pump room resulted in a maximum differential pressure of 4.1 psid. Reference 28 shows the differential pressure history for this accident.

6.2.1.2.3.8 RWCU Filter-Demineralizer Subcompartments

The RWCU filter-demineralizer rooms are subject to pressure differentials when a break in the RWCU system within the filter-demineralizer rooms is postulated.

The modeling of the event is similar to that described in Subsection 6.2.1.2.3.6 and the transient mass and energy release rates for this break are presented in Table 6.2-18. The nodalization scheme is presented in Figure 6.2-94 for filter-demineralizer rooms. The nodal and vent path descriptions are correspondingly given in Tables 6.2-37 and 6.2-38.

The rupture of the RWCU line in the filter-demineralizer rooms resulted in a maximum differential pressure of 12.7 psid. Reference 28 shows the transient for both filter-demineralizer rooms.

6.2.1.2.3.9 RWCU Filter-Demineralizer Valve Subcompartment

The RWCU filter demineralizer valve room is subject to pressure differentials when a break in the RWCU system within the filter-demineralizer valve room is postulated.

The modeling of this event and nodalization is similar to that described in Subsection 6.2.1.2.3.6. In addition, the transient mass and energy release rates are tabulated in Table 6.2-18. The nodalization scheme is presented in Figure 6.2-108 and the nodal and vent path descriptions shown in Tables 6.2-39 and 6.2-40, respectively.

The rupture of a RWCU line in the filter-demineralizer valve room resulted in a maximum differential pressure of <5 psid. Reference 28 illustrates the differential pressure history in the filter-demineralizer valve room.

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

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

6.2.1.3.1 Mass and Energy Release Data

Table 6.2-43 provides the mass and enthalpy release data for the recirculation line break.

Figure 6.2-115 shows the blowdown flowrates for the recirculation line break. This data was employed in the containment pressure-temperature transient analyses reported in Subsection 6.2.1.1.3.3.1.

Table 6.2-44 provides the mass and enthalpy release data for the main steam line break. Figure 6.2-116 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 6.2.1.1.3.3.2.

6.2.1.3.2 Energy Sources

The reactor coolant system conditions prior to the 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.

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

a. Stored energy in the reactor system,

b. Energy generated by fission product decay,

c. Energy from fuel relaxation,

d. Sensible energy stored in the reactor structures,

e. Energy being added by the ECCS pumps, and

f. Metal-water reaction energy.

g. All but the pump heat energy addition is discussed or referenced in this section. The pump heat rate used in evaluating the containment response to the LOCA, is conservatively selected as a constant input equal to the horsepower rating of all operating ECCS pumps.

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 6.2-45 and 6.2-45a as a function of time after accident initiation.

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 6.2-117 shows the variation of the sensible heat content of the reactor vessel and internal structures during a main steam line break accident based upon the temperature transient responses.

6.2.1.3.3 Reactor Blowdown Model Description

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

6.2.1.3.4 Effects of Metal-Water Reaction

The containment systems are designed to accommodate the effects of metal-water reactions and other chemical reactions which may occur following a loss-of-coolant accident. The amount of metal-water reaction which can be accommodated is consistent with the performance objectives of the emergency core cooling systems (ECCS). Subsection 6.2.5.1.3.2 provides a discussion on the generation of hydrogen within the containment by metal-water reaction. In evaluating the containment response 14,706 Btu/sec of heat from metal-water reaction is included for the first 120 seconds. The containment response is insensitive to the reaction time, even for the extremely conservative case where all of the energy is included prior to the occurrence of peak drywell pressure.

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

This section is not applicable to the Clinton Power Station as it is a BWR type reactor.

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

This section is not applicable to the Clinton Power Station as it is a BWR type reactor.

6.2.1.6 Testing and Inspection

Containment testing and inspection programs are fully described in Subsection 6.2.6 and in Chapter 14. The requirements and bases for acceptability are outlined completely in the CPS Technical Specifications.

6.2.1.7 Instrumentation Requirements

The following containment parameters are monitored by redundant, safety-related instrumentation:

a. drywell pressure,

b. containment pressure,

c. suppression pool level,

d. suppression pool temperature,

e. containment and drywell area temperatures,

f. containment and drywell high range gamma radiation, and

g. containment and drywell ventilation exhaust radiation.

Containment and drywell hydrogen concentration is monitored by one channel of non-safety related instrumentation.

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.

6.2.2 Containment Heat Removal Systems

6.2.2.1 Design Bases

The containment heat removal system, consisting of the containment cooling system, is an integral part of the RHR system. The purpose of this system is to prevent excessive containment temperatures and pressures thus maintaining containment integrity following a LOCA. To fulfill this purpose, the containment cooling system meets the following safety design bases:

a. The system limits the long-term bulk temperature of the suppression pool to 185°F without spray operation when considering the energy additions to the containment following a LOCA. These energy additions, as a function of time, are provided in the previous section.

b. The single failure criteria applies to the system.

c. The system is designed to safety grade requirements including the capability to perform its function following a loss-of-coolant accident.

d. The system is operable during those environmental conditions imposed by the LOCA.

e. Each active component of the system, except for the Feedwater Leakage Control (FWLC) valves (MOVs and Check Valves), is testable during normal plant operations. The FWLC valves are tested during shutdown to assure component reliability.

f. The containment heat removal system is designed to Seismic Category I requirements. System components, as appropriate, are designed to meet ASME Code Section III, Class 2 requirements.

g. The RHR pump suction strainer is sized to prevent passage of particles over 3/32 inch in diameter which could cause malfunction of the containment heat removal system equipment or plug the containment spray nozzles.

To meet the requirements of General Design Criterion 34 and as indicated in USAR Appendix 15A, a single failure analysis has been done on the shutdown cooling system of the RHR system (Reference Article 15A.6.3.3, event 18, and Figure 15A.6-18 which addresses shutdown cooling specifically).

In the event that offsite power is lost, and assuming the single active failure as the loss of one division of emergency power, actuation of one of the shutdown cooling line valves to the open position would be prevented. In the postulated single failure described above, the single shutdown suction line will be unable to draw from the vessel for the cooldown operation.

Given the condition that suction line isolation loss occurred and could not be corrected by operator actions, alternate shutdown cooling is accomplished by depressurizing the reactor and supplying cooling water from the suppression pool by an available ECCS. The water is then returned to the suppression pool via any of the safety/relief valve discharge lines, and the decay heat will be removed by operation of the suppression pool cooling mode of the RHR system. (Q&R 480.28)

6.2.2.2 System Design

The containment cooling system is an integral part of the RHR system. Water is drawn from the suppression pool, pumped through one or both RHR heat exchangers and delivered to the suppression pool, or to the containment spray header. Water from the shutdown service water system is pumped through the heat exchanger tube side to exchange heat with the processed water. Two cooling loops are provided; each being mechanically and electrically separate from the other to achieve redundancy. A process and instrumentation diagram is provided in Section 5.4. The process diagram, including the process data, is provided in the Section 5.4 for all design operating modes and conditions.

Each train of the containment spray system consists of two headers: On the 'A' Train the top header consists of 63 equally-spaced spray nozzles and the bottom header consists of 186 equally-spaced spray nozzles. On the 'B' Train, the top header consists of 64 equally-spaced spray nozzles and the bottom header consists of 187 equally spaced spray nozzles. The nozzle orientation for each spray header is shown in Drawing M06-1075-6. A plan view of the spray headers is shown in Drawing M06-1075 Sheets 6 and 11. The spray nozzles used are Spraco 1713A nozzles, each of which is capable of a flow of 15.5 gpm with a pressure drop of 40 psid. These nozzles have an approximate 3/8-inch spray orifice. The nozzles produce spray droplets in varying sizes, the largest being less than 1600 microns at rated service conditions. Each nozzle header is independently oriented to ensure efficient coverage of the containment volume.

All portions of the containment cooling system are designed to withstand operating loads and loads resulting from natural phenomena. All operating components, except for the FWLC motor operated valves and check valves, can be tested during normal plant operation. The FWLC keep fill valves are tested during shutdown so that component reliability can be assured. Construction codes and standards are covered in Subsection 5.4.7.

The containment cooling system is started manually or automatically in the case of containment sprays. The LPCI mode is automatically initiated from ECCS signals, as discussed in Section 6.3, and the RHR system realigned for containment cooling by the plant operator after the reactor vessel water level has been recovered (see Subsection 6.2.1). The RHR pumps are already operating. Containment cooling is initiated in loop A or B by manually starting the shutdown service water pump, opening the service water valves at the heat exchanger, closing the LPCI injection valve, closing the heat exchanger bypass valve on the RHR (shell) side, and opening the pool return valve. In the event that a single failure has occurred, and the action which the plant operator is taking does not result in system operation, then the operator will place the other totally redundant system into operation by following the same startup procedure. If the operator chooses to use the containment spray, he must close the LPCI injection valves or the pool return valves and open the spray valves. The containment spray mode is also initiated automatically after 10 minutes of a LOCA signal if containment pressure exceeds 9 psig and drywell pressure exceeds 2 psig.

The 10-minute timer can be reset at the discretion of the operator to delay automatic spray initiation. Resetting of the timer does not preclude manual spray initiation. Automatic initiation is provided to protect the containment in the event of suppression pool bypass leakage as described in Subsection 6.2.1.1.5.5 The NPSH available to the RHR pumps is greater than the NPSH required by the pump for all credible suppression pool water levels, and is further discussed in Subsection 5.4.7.

Each ECCS pump takes suction directly from the suppression pool. To prevent foreign objects in the suppression pool from entering the ECCS flow path, a strainer is located on the ECCS pump suction lines in the suppression pool. The design of the strainer precludes the entrance of foreign materials greater than 3/32 inch diameter into the ECCS flow path.

Suppression pool water quality will be monitored and controlled, however, debris resulting from accident conditions can be postulated to enter the suppression pool. The effective area of the strainer would be reduced as the strainer removes the debris.

To ensure system function is maintained, the strainer is designed with sufficient strainer surface to provide very low fluid approach velocities (~0.02 fps). This will minimize head loss under postulated debris loading conditions in the event the strainers become fully loaded (i.e., conservatively specified debris loading resulting from LOCA-generated and pre-LOCA debris materials).

In response to NRC Bulletin 96-03, a large passive strainer was installed for the ECCS (and RCIC) pumps. The strainer is located on the floor of the suppression pool and completely circumscribes the suppression pool. The ECCS/RCIC suction strainer connects to each of the ECCS (and RCIC) suction piping penetrations. The ECCS/RCIC suction strainer is fabricated from stainless steel plate and perforated (3/32" holes) stainless steel plate. The ECCS/RCIC suction strainer is semi-circular in cross-section with two separate flow channels separated by an open central channel. The three ECCS divisions which to the ECCS/RCIC suction strainer are physically separated through the use of internal divider plates in the flow channels; two divider plates are installed at each divisional interface location. The ECCS/RCIC suction strainer rests on the floor of the suppression pool and the top of the ECCS/RCIC suction strainer is approximately 3' – 2" above the suppression pool floor.

The ECCS pump suction strainer located in the containment suppression pool meets the following safety design basis:

1. The strainer is designed to prevent the introduction of objects greater than 3/32-inch diameter into the reactor pressure vessel.
2. Adequate net positive suction head to the ECCS pumps shall be provided with the strainer full loaded (i.e., conservatively specified debris loading resulting from LOCA-generated and pre-LOCA debris materials).
3. The strainer is designed to withstand any loads anticipated during suppression pool transients including temperature, pressure and water level.

4. The strainer is designed to permit testing in conjunction with the periodic ECCS testing to demonstrate strainer operability.

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

Essentially 100% of the thermal piping insulation used in the CPS drywell is reflective metallic insulation (RMI) (e.g., main steam piping, reactor water clean-up piping and equipment, feedwater piping, reactor recirculation piping, and the reactor pressure vessel). The exceptions are a small number of locations where a minimal amount of other insulation was used due to space limitations (see Subsection 5.2.3.2.4). In addition, anti-sweat piping insulation is used on certain chilled water system piping in the drywell. This anti-sweat insulation is hydrophobic, is less dense than water and would float on the pool surface, and therefore, has been determined not to be a potential clogging threat to the ECCS pump suction strainers (Reference 24).

The metallic-reflective insulation is installed in sections with overlapping edges and seismic quick-release latches with integral keepers. Anti-sweat insulation is installed in varying lengths using manufacturer's recommended adhesive.

Instead of postulating the maximum potential insulation debris that may be generated by LOCA at various break locations, as recommended by Regulatory Guide 1.82, Rev. 2, the maximum pressure drop that RMI is capable of producing has been considered in the design of the ECCS suction strainer.

The CPS strainer was designed based on results of 1/4-scale model strainer testing performed for the Perry Nuclear Power Plant which did not use Reflective Metal Insulation (RMI) as part of the debris recipe. RMI will have a very small effect on the head loss for the CPS large passive strainer. This is due to several effects, principally settling and less affinity for transport, but most importantly because the RMI head loss has been shown to be minimal in BWR Owners Group testing of smaller strainers.

The quantity of "other" LOCA-generated debris (debris resulting from painted surfaces, fibrous, cloth, plastic, or particulate materials within the zone of influence that may produce debris) is based on the recommendations contained in Reference 24. The debris quantities used in the strainer design are considered very conservative.

Due to low approach velocities associated with the large, passive strainer, metal tags and other metallic materials were assumed to settle in the suppression pool. CPS takes no credit for settling of RMI, fiber insulation material, corrosion products/sludge, paint or coating debris, and plastic debris materials at the onset of a LOCA; however, significant fiber debris settling was observed in the 1/4-scale test program which is considered prototypical of the settling which would occur for fibrous debris in the suppression pool after LOCA-vent condensation oscillation and chugging have ceased. Since RMI is more dense than the fiber debris it is expected that similar settling of RMI would occur.

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

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

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

The large toroidal passive strainer has been designed in accordance with Regulatory Guide 1.82, Revision 2. The suction strainer has been designed to preclude the potential for loss of NPSH caused by debris blockage during the period that the ECCS is required to maintain long-term cooling. The large toroidal passive strainer design results in a very low approach velocity for water entering the strainer. Debris collected on the strainer surface is not expected to compact significantly (due to the very low approach velocity), resulting in minimal head loss. The testing of a 1/4-scale model of the strainer design confirmed the performance of the strainer and the behavior of the postulated debris bed as a function of time after the postulated LOCA. Because the debris bed will not be significantly compacted, flow will continue to pass through the debris (and the strainer) and thus the overall differential pressure will remain low. CPS uses essentially 100% RMI in the drywell. Head losses due to RMI in the suppression pool on the strainer have been evaluated and are considered to be negligible. Maintaining a low differential pressure will ensure adequate NPSH for the ECCS pumps.

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. Subsequent to the accident, fission product decay heat will result in a continuing energy input to the pool. The containment cooling system will remove this energy which is released into the primary containment system thus resulting in acceptable suppression pool temperatures and containment pressures.

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

a. With the reactor initially operating 102% of rated power, a LOCA occurs.

b. A loss of offsite power occurs and one emergency diesel fails to start and remains out of service during the entire transient. This is the worst single failure.

c. Only three ECCS pumps are activated and operated as a result of there being no offsite power and minimum onsite power. (Section 6.3 describes the ECCS equipment.)

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

The sprayed and unsprayed volumes and regions of the containment are shown in Figures 6.2-120 through 6.2-122. These sprayed regions represent 90% of the net containment volume and, therefore, full credit (thermal effectiveness, e_x = 1.0) for both pressure suppression and heat removal could be claimed for the spray system (Reference 7). This contention is also supported analytically by using the CONSPRAY code which is essentially the HEATDROP code of Parsley (Reference 8) with a few improvements, such as the ability to determine the drop size distribution groups from the mean drop size plotting capability, and direct evaluation of the overall spray thermal effectiveness.

6.2.2.3.1 Summary of Containment Cooling Analysis

When calculating the long-term post-LOCA pool temperature transient, it is assumed that the initial suppression pool temperature and the RHR service water temperature are at their maximum values. This assumption maximizes the heat sink temperature to which the containment heat is rejected and thus maximizes the containment temperature. In addition, the RHR heat exchanger is assumed to be in fully fouled condition at the time the accident occurs. This conservatively minimizes the heat exchanger heat removal capacity. The resultant suppression pool temperature transient is described in Subsection 6.2.1.1.3.3.1 and 6.2.1.1.3.3 and is shown in Figures 6.2-7a, 6.2-7b and 6.2-8. Even with the degraded conditions outlined above, the maximum temperature is maintained below the design limit specified in Subsection 6.2.2.1.

It should be noted that, when evaluating this long-term suppression pool transient, all heat sources in the containment are considered with no credit taken for any heat losses other than through the RHR heat exchanger and heat sinks in the containment and drywell air spaces. These heat sources are discussed in Subsection 6.2.1.3. Figures 6.2-9, 6.2-9a, and 6.2-9b show the actual heat removal rate of the RHR heat exchanger.

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

6.2.2.4 Tests and Inspections

Preoperational tests were performed to verify individual component operation, individual logic element operation, and system operation up to the containment spray nozzles. A sample of the sparger nozzles were bench tested for flow rate versus pressure drop to evaluate the original hydraulic calculations. Refer to Subsection 5.4.7.4 for further discussion of preoperational testing.

The containment spray nozzles may be operationally tested by connecting an air line to a test connection on the spray header, and blowing air out the nozzles. Unobstructed (free) air flow will be verified for each nozzle by either thermography (infrared camera) or physical inspection, which may include the addition of streamers to the nozzles or actual inspection of each nozzle for air flow.

6.2.2.5 Instrumentation Requirements

The details of the instrumentation are provided in Subsection 7.3.1.1.4. The suppression pool cooling mode of the RHR system is manually initiated from the control room.

6.2.3 Secondary Containment Functional Design

The secondary containment completely encloses the primary containment, except for the upper personnel hatch, and consists of the containment gas control boundary, the containment gas control boundary extension (siding within the auxiliary building), the fuel building, the ECCS RHR heat exchanger rooms, the pump rooms, the RWCU pump room, and the main steam pipe tunnel. During normal operation, the fuel building ventilation system maintains the secondary containment at a slightly negative pressure. Following a design basis accident, the standby gas treatment system (SGTS) achieves and maintains a negative pressure in the areas that comprise the secondary containment.

In addition, the SGTS provides the capability to remove potential contamination released to the secondary containment volume after an accident in primary containment. The design and operation of the fuel building ventilation system and the SGTS are discussed in Subsections 9.4.2 and 6.5.1 respectively. Chapter 15 discusses the operation of these systems under accident conditions.

6.2.3.1 Design Basis

The functional requirements for the secondary containment arise from the Code of Federal Regulation limits for the release of radioactive materials within the plant and at the plant boundary during normal operation and following postulated accidents within the primary containment. The specific design criteria implemented to meet these functional requirement are set forth below.

a. The secondary containment structures is of Seismic Category I design and is sufficiently leak tight that the SGTS can maintain the required negative pressure within the secondary containment volume for wind speeds up to approximately 30 mph. The secondary containment, in conjunction with the operation of the SGTS, is designed to achieve and maintain an 0.25-inch water gauge negative pressure in the boundary region within 19 minutes of the initiation of SGTS.

The pressure will prevent exfiltration of the secondary containment atmosphere for wind speeds less than 20 mph.

b. The secondary containment, in conjunction with the operation of the SGTS is designed to limit the total effective dose equivalent (TEDE) within the guidelines of 10 CFR 50.67 at the site boundary and low population zone. Also, the design limits the TEDE dose for the control room within the guidelines of 10 CFR 50, General Design Criteria 19.

c The internal and external design pressures and leak tightness of the secondary containment structures are discussed in Chapter 3. A description of the potential paths of primary containment leakage bypassing the secondary containment is given in Table 6.2-47.

d. The secondary containment and the SGTS are designed to permit periodic inspection and testing of principal systems and components such as fans, dampers, and filters.

6.2.3.2 System Design

The secondary containment consists of the fuel building; the portion of the auxiliary building enclosing the ECCS pump rooms; the RWCU pump and heat exchanger rooms; and the main steam tunnel to S-line; the gas control boundary that encloses the primary containment above the level of the auxiliary and fuel building roofs; the radwaste tunnel; the auxiliary building pipe tunnel; MSIV rooms (for MSIV blowers); auxiliary building floor drain pump room and the gas control boundary extension in the auxiliary building. The general arrangement of the various structures that comprise the secondary containment are shown in Figure 6.2-132 Sheets 1-6.

The free volume of the Secondary Containment is 1,710,000 ft^3.

The performance objective of the secondary containment is to provide a volume completely surrounding the primary containment which can capture fission products that might otherwise leak to the environment following a design basis accident. To achieve this, the fuel building and portions of the auxiliary building are of reinforced concrete construction, which has an inherently low leak rate. In addition, a low-leakage metal-siding enclosure is provided for the remainder of the secondary containment boundary in the auxiliary building and the containment gas control boundary. Following the postulated design basis accident, the SGTS functions to achieve and maintain the secondary containment volume at or below a negative pressure of 0.25-inch water gauge. The exhaust air discharge required to maintain this negative pressure is routed through the SGTS equipment trains which are designed to remove 99% of the elemental iodine and organic iodides. (Refer to Subsection 6.5.1 for a discussion of the SGTS filter system.)

The secondary containment is designed for a conservative inleakage at 0.25-inch water gauge differential pressure. The secondary containment is also designed to ensure that a uniform negative pressure is maintained throughout the volume during normal operating and post-accident conditions.

The design and construction codes, standards, and guides applied to the auxiliary and fuel buildings are discussed in Section 3.8.

In order to minimize the amount of radioactive material that leaks to the secondary containment following a design basis accident, all primary containment penetrations are provided with redundant, ASME code, Section III, Class 1 or Class 2, Seismic Category I isolation valves. These isolation valves are located in the primary and/or secondary containment and thus minimize the possibility of leakage bypassing the secondary containment. Table 6.2-47 presents a list of all piping penetrations for the primary containment along with attendant information relating to the types and location of isolation valves, the types of valve operators, power and isolation signal sources, etc. Subsection 6.2.4 provides a discussion of the design of the containment isolation system; the containment and reactor vessel isolation control systems are described in Subsection 7.3.1.1.2.

The primary containment leakage rate is specified in Subsection 6.2.1. To ensure that this leakage rate is not exceeded, the primary containment and containment components are subject to a leak rate testing program which is described in Subsection 6.2.6. Primary containment integrity is verified and assured in accordance with the CPS Technical Specifications.

Access openings into the secondary containment have been provided with air locks or are otherwise administratively controlled so that there will be one door closed at all times, in accordance with the CPS Technical Specifications. Access openings will have no adverse impact on operation of the SGTS and the integrity of the secondary containment.

Access openings are shown on plant general arrangement Drawings M01-1105-1, M01-1106-1, M01-1107-1, and M01-1109-1.

Instrumentation to monitor the status of the openings or position indicators and alarms with alarm capability in the main control room are not provided.

However, the integrity of the secondary containment is maintained because the access doors and hatches have been provided with one of the following:

a. Electrical interlocks between the two airlock doors to prevent both being simultaneously opened.

b. Administrative controls which will preclude access hatches being opened during reactor operation or during refueling operations. (Q&R 480.04)

6.2.3.3 Design Evaluation

6.2.3.3.1 Standby Gas Treatment System

The standby gas treatment system will maintain the secondary containment at a negative pressure with respect to the environment following the design-basis loss-of-coolant accident. The design flow rate of the SGTS is based on the following criteria:

a. The exhaust flow is based on the sum of potential inleakages when an 0.25-inch water gauge negative pressure is maintained in the secondary containment.

   The standby gas treatment system flow rate is nominally 4,000 cfm.

The leakage is determined by evaluation flow characteristics through small cracks based on manufacturer's certified leak test results on building siding; air leakage test results contained in "Conventional Building for Reactor Containment," NAA-SR-10100; and specified leak rates on valves, dampers, and penetrations. Wind effect are considered as described in Subsection 6.5.1.1.

b. Calculations indicate that the SGTS fan has been adequately sized to achieve an 0.25-inch water gauge negative pressure in less than 19 minutes after the LOCA event.

6.2.3.3.2 Secondary Containment System

The secondary containment system was analyzed to determine the effects of a DBA in primary containment on the pressure and temperature histories for subcompartments within the system. This analysis was carried out using a modified version of COMPARE/MOD1 computer code. Two separate analyses were performed. These were:

a. a short-term analysis for the pressure histories, and,

b. a long-term analysis for the temperature histories.

6.2.3.3.2.1 Short-Term Pressure History

The short-term analysis was performed using three interconnected nodes within the secondary containment and one node representing the outside environment. The transient was calculated by performing a balance of mass and energy addition, removal, and accumulation within each of the nodes. Mass flow between the nodes was prescribed in a conservative manner (i.e., minimum flow for a given pressure differential). Mass addition to nodes was due to infiltration of outside air, evaporation from the spent fuel pool, and junction flow. Mass removal from nodes was due to operation of the SGTS as well as junction flow. Energy addition to nodes was due to the presence of various heat sources while energy removal was due to heat transfer to heat sinks and the operation of fan coolers.

It should be noted that the SGTS has the ability to draw down the secondary containment to below -0.25 inch water gauge well within the first 19 minutes following a LOCA. Furthermore, the secondary containment pressure would remain below this value as long as the SGTS continues to operate, since the secondary containment heat loads will be balanced by the temperature dependent heat sinks and fan coolers as the transient proceeds. Specifically, design pressure in the secondary containment (-0.25" water gauge) is reached in less than 19 minutes.

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The time period until the secondary containment reaches a negative pressure of 0.25-inch water gauge should not be considered as a period of direct outleakage for the following reasons:

a. The pressure gradient forcing leakage from the primary containment is less than 4 psig during this time period. The containment design and construction, and testing requirements provide leakage integrity and such a small pressure difference provides little driving force for leakage across small leak paths.

b. The most predominant potential containment leak paths are piping penetrations and door seals which are located in the containment at elevations which are enclosed by the secondary containment which consists of the ECCS pump rooms, steam tunnel, and fuel building. Due to the large volume of these areas, the small amount of radioactive gases leaking through would require some interval of time to diffuse through the secondary containment to the outside.

c. Fuel cladding does not fail for at least several minutes.

d. The entire secondary containment, including the containment gas control boundary (CGCB), is maintained at a slight negative pressure during normal operations.

Thus, any primary containment leakage will be contained within the secondary containment and only will reach the outside after passing through the standby gas treatment system.

6.2.3.3.2.2 Long-Term Temperature Analysis

The secondary containment is the modeled using multiple nodes. The outside environment and primary containment are used as boundary nodes. The calculation consisted of a balance of energy addition, removal, and accumulation within each of the nodes. Energy addition was due to heat transfer through the primary containment wall as well as other walls, the operation of various safety-related and non-safety-related equipment, the spent fuel cask heat load, and sensible and latent heat of equipment and piping which was hot before the start of the transient. Energy removal was due to heat transfer to walls and operation of safety-related fan coolers.

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The fuel building, ECCS Equipment Room, and main steam tunnel temperature responses are within design limits.

6.2.3.4 Tests and Inspections

Test and inspections of the primary containment isolation system are discussed in Subsections 6.2.4, 6.2.6, and 7.3.1.1.2. Tests and inspections of the secondary containment system are discussed in Subsection 6.2.6.5.3. Test and inspections of the standby gas treatment system are discussed in Subsection 6.5.1.4. Primary containment leak-rate testing is discussed in the CPS Technical Specifications and Subsection 6.2.6.

6.2.3.5 Instrumentation Requirement

The instrumentation and controls for the fuel building ventilation system are described in Subsection 7.7.1.14 and for the standby gas treatment system (SGTS) in Subsection 7.3.1.1.7.

6.2.4 Containment Isolation System

The containment isolation system consists of the piping, valves, and actuators required to isolate the containment in the event of accidents or other conditions which can lead to excessive releases of radioactivity. The containment isolation systems, in general, close those fluid penetrations which support systems that are not required for emergency operation. Those fluid penetrations supporting engineered safety feature systems have remote manual isolation valves which may be closed from the control room, if required. Redundancy and physical separation is provided in the electrical and mechanical design to ensure that no single failure in the containment isolation systems prevents the system from performing its intended functions.

6.2.4.1 Design Bases

a. The containment isolation system is designed to function if operating conditions or radioactivity releases are in danger of causing doses in excess of those dosages specified in 10 CFR 50.67.

b. The design of isolation valving for lines penetrating the containment follows the requirements of General Design Criteria 54 through 57 as noted in Subsection 6.2.4.3.

c. Isolation valving for instrument lines which penetrate the containment conforms to the requirements of Regulatory Guide 1.11, with the exceptions stated in Section 1.8.

d. Capability for rapid closure or isolation of all pipes or ducts that penetrate the containment provides means of establishing a containment barrier in such pipes or ducts sufficient to maintain leakage within permissible limits.

e. Main steam isolation valve closure speeds shall limit radiological effects from exceeding guideline values established by 10 CFR 50.67.

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

g. The containment isolation valves and associated piping and penetrations are designed to Seismic Category I requirements.

h. Isolation valves, actuators, and controls are protected against damage by missiles and postulated effects of high- and moderate-energy line breaks.

6.2.4.2 System Design

The general criteria governing the design of the containment isolation systems are provided in Subsections 3.1.2 and 6.2.4.1. Table 6.2-47 summarizes the containment penetration isolation valves and provides information on the following:

a. Open or closed status under normal operating conditions and postulated accident conditions.

b. The primary and secondary modes of actuation provided for isolation valves.

c. The parameters sensed to initiate isolation valve closure.

d. The closure time and sequence of timing for principal isolation valves to secure containment isolation.

e. Applicable General Design Criteria.

f. General compliance or alternate approach assessment for Regulatory Guides 1.26 and 1.29 may be found in Section 3.2.

g. Section 3.11 presents a discussion of the environmental conditions, both normal and accidental, for which the containment isolation valve system is designed. The section also discusses the qualification tests that are required to assure the performance of the isolation valves under those environmental conditions.

For the particular systems that penetrate the containment listed in Table 6.2-47, a cross-reference to a USAR Figure is provided to depict the respective isolation valve arrangement. Closure time for containment isolation valves limits potential radioactive releases below permissible levels specified in 10 CFR 50.67.

Protection is provided for isolation valves, actuators, and controls against damage from missiles, pipe whip, and jet impingements. Potential sources of missiles, pipe whip, and jet impingement have been evaluated and are discussed in Subsections 3.6.1, 3.5.2 and 3.6.2.

The containment isolation system has been designed so that debris entering any piping system which penetrates containment will not prevent the containment isolation system from fulfilling its intended function. Three cases are considered below to demonstrate the integrity of the containment isolation system in spite of unspecified debris within the piping system.

The first case considered involves piping systems similar to the configuration shown on Drawing M05-1052, Sheet 3. This configuration consists of isolation valves inside and outside containment. Debris within the piping system should either pass through both containment isolation valves or will be stopped by one containment isolation valve. In either event, at least one isolation valve will still be available to fulfill the containment isolation function.

The second case considered involves piping systems similar to the configuration shown on Figure 6.2-146. This configuration consists of a single isolation valve and a closed loop piping system outside containment (CLOC). Debris in the piping system could block closure of the isolation valve, however this will not affect the integrity of the CLOC. Therefore the containment isolation function will not be impaired.

The third case considered involves piping systems similar to the configuration shown on Sheet 2 of Figure 6.2-123. This configuration contains an excess flow check valve in piping to an instrument. During normal plant operation, no flow passes through these lines. The lines are used for sensing suppression pool level, building ventilation pressure, containment pressure, drywell pressure and reactor coolant system pressure. If during a design basis accident, a failure occurs which initiates flow in the line, the check valve closes. Debris cannot prevent the valve from functioning because no mechanism exists for the debris to reach the valve.

Isolation valves are designed to be operable under the most adverse environmental conditions such as operation under maximum differential pressures, extreme seismic occurrences, steam laden atmosphere, high temperature, and high humidity. Electrical redundancy is provided for power-operated valves. Power for the actuation of two isolation valves in a line (inside and outside of containment) is supplied by two redundant, independent power sources without cross ties. In general, outboard isolation valves receive power from the Division 1 power source while isolation valves within containment receive power from the Division 2 power source. Further, the power supply is a-c for both Division 2 and Division 1 valves, depending upon the system under consideration.

The containment Atmosphere Monitoring System has been designed to perform its intended function in the event of failure of one of the emergency diesel generators. Therefore, inboard and outboard isolation valves in one monitoring division train receive power from the same source.

The main steamline isolation valves are spring loaded, pneumatic, double acting piston-operated globe valves designed to fail closed on loss of pneumatic supply pressure or loss of power to the solenoid-operated pilot valves. Each valve has two independent pilot valves supplied from independent power sources. Each main steamline isolation valve has an air accumulator to assist in its closure upon loss of air supply, loss of electrical power to the pilot valves, and/or failure of the loaded spring.

The separate and independent action of either air pressure or spring force is capable of closing an isolation valve.

It should be noted that all motor-operated isolation valves remain in their last position upon loss of valve power. On the other hand, all air-operated isolation valves, (not applicable to air-testable check valves), close on loss of air pressure or electric power.

Each ECCS system outside containment is located within its own watertight compartment. Each compartment has leak detection devices with appropriate alarms. Also, each compartment has a sump that has level switches mounted on the sump pumps. Each compartment contains area temperature monitors which alarm in the control room. Should any leakage (e.g., from a valve shaft) be detected by these devices, the plant operator can shut down the affected ECCS system and isolate it. Direct determination for leakage from the valve shaft and/or bonnet seals has not been provided. The remaining ECCS systems are capable of providing adequate core cooling.

The design of the isolation valve system includes consideration of single failures.

Each closed system outside containment for which credit is taken as an isolation barrier is shown in Figures 6.2-143 through 6.2-147.

Closed loop piping systems commence immediately beyond the outboard containment isolation valves. For penetrations 15 and 16, the relief valve lines which discharge into the suppression pool through penetrations 87 and 27 are located prior to the outboard containment isolation valves.

For penetration 28, the closed loop piping system originates after the outboard containment isolation valve, IE51-F031. The three-quarter inch pressure tap located inboard of this valve does not constitute a branch line as clarified in the response to Question 480.15.

The piping system shown on Figure 6.2-145 represents the boundary of the closed loop. Thus the piping from outboard containment isolation valve IE51-F019 to the dashed piping is part of the closed loop piping system.

For penetration 18, the closed loop piping system commences beyond the outboard containment isolation valves IE12-F024A, IE12-F011A, and IE12-F064A. A failure in the piping between these isolation valves and the containment will not result in leakage of the containment atmosphere because the piping inside the containment is sealed from the containment atmosphere by the suppression pool. (Q&R 480.11 & 480.12)

6.2.4.3 Design Evaluation

6.2.4.3.1 Introduction

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

Mechanical components are redundant, such that isolation valve arrangements provide back-up in the event of accident conditions. Isolation valve arrangements satisfy all requirements specified in General Design Criteria 54, 55, 56, and 57, and Regulatory Guide 1.11 as noted in Table 6.2-47.

The arrangements are described in Figure 6.2-123. The automatic isolation valves have redundance in the mode of actuation with the primary mode being automatic and the secondary mode being remote manual.

A program of testing, described in Subsection 6.2.4.4, is maintained to ensure valve operability and leaktightness.

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

Electrical redundancy is provided in isolation valve arrangements which eliminates dependency on one power source to attain isolation. Electrical cables for isolation valves in the same line powered from different divisions have been separated in accordance with IEEE-384. Cable selection was based on the specific environment to which they may be subjected, such as magnetic fields, high radiation, high temperature, and high humidity.

The Containment Atmosphere Monitoring system has been designed to perform its intended function in the event of failure of one of the emergency diesel generators. Therefore, inboard and outboard isolation valves in one monitoring division train receive power from the same source.

Provisions for operator control and/or locks ensure that the position of all nonpowered isolation valves is maintained and known. For all power-operated valves the position is indicated in the main control room. Nonpowered isolation valves are locked in either open or closed position, as required. Discussion of instrumentation and controls for the isolation valves is included in Section 7.3.

6.2.4.3.2 Evaluation Against General Design Criteria

6.2.4.3.2.1 Evaluation Against Criterion 55

The reactor coolant pressure boundary (RCPB), as defined in 10 CFR 50, Section 50.2 (v), consists of the reactor pressure vessel, pressure retaining appurtenances attached to the vessel, and valves and pipes which extend from the reactor pressure vessel up to and including the outermost isolation valve. The lines of the reactor coolant pressure boundary which penetrate the containment include provisions for isolation of the containment, thereby precluding any significant release of radioactivity. Similarly, for lines which do not penetrate the containment but which form a portion of the reactor coolant pressure boundary external to the drywell, the design ensures that isolation of the reactor coolant pressure boundary can be achieved.

6.2.4.3.2.1.1 Influent Lines

Influent lines which penetrate the primary containment and connect directly . to the RCPB are equipped with at least two isolation valves; one inside the drywell, and the other outside the containment as close to containment as practical. Protection against accidental releases to the environment is provided by these isolation valves.

6.2.4.3.2.1.1.1 Feedwater Lines

The feedwater lines are part of the reactor coolant pressure boundary as they penetrate the drywell to connect with the reactor pressure vessel. Each line has three isolation valves. The isolation valve inside the drywell is a simple check valve, located as close as practicable to the drywell wall. Outside the containment is an air-operated check valve located as close as practicable to the containment wall and farther away from the containment is a motor-operated gate valve. Should a break occur in the feedwater line, the check valves prevent significant loss of reactor coolant inventory and offer immediate isolation. The air-operated check valve is "power assisted" closed and is actuated by the protection system. During the postulated loss-of-coolant accident, it is desirable to maintain reactor coolant makeup from all sources of supply. For this reason, the outermost valve does not automatically isolate upon signal from the protection system. However, this valve is capable of being remotely closed from the control room to provide long-term leakage protection when continued makeup from the feedwater source is unnecessary.

In order to reduce potentially radioactive leakage and to limit potential dose from the feedwater lines, the feedwater lines are kept filled with water for 30 days post-accident. See Section 5.4.7.1.1.6 for description of the method of keeping the lines filled.

6.2.4.3.2.1.1.2 HPCS Line

The HPCS line penetrates the drywell to inject directly into the reactor pressure vessel. Reactor coolant pressure boundary isolation is provided by a testable check valve, located inside the drywell and a remote-manually controlled motor operated gate valve located as close as practicable to the exterior wall of the containment. Containment isolation is maintained by this gate valve. Redundant isolation is provided by the HPCS system piping and components which comprise a closed loop boundary. If a loss-of-coolant accident occurred, this gate valve would receive an automatic signal to open.

6.2.4.3.2.1.1.3 LPCI and LPCS Lines

Satisfaction of reactor coolant pressure boundary isolation criteria for the LPCI "C" and LPCS lines is accomplished by use of remote manually controlled, motor-operated, normally closed gate valves and testable check valves. Both types of valves are normally closed with the gate valves receiving an automatic signal to open at the appropriate time to assure that acceptable fuel design limits are not exceeded in the event of a loss-of-coolant accident. The normally closed check valves protect against containment overpressurization in the event of a break in the line between the check valve and containment wall by preventing high energy reactor water from entering the primary containment. Redundant containment isolation is provided by system piping and components which comprise a closed loop boundary.

For LPCI "A" and "B" lines in addition to the above described testable check valves, remote manually controlled, motor-operated, normally closed gate valves are located inside containment upstream of the testable check valves to provide the second barrier to reactor coolant pressure boundary, and the first barrier for containment isolation. They will receive automatic signal to open in the event of a loss-of-coolant accident. LPCI "A" and "B" lines also have remote manually controlled motor-operated gate valves, located outside containment, that are normally open. They can be closed to provide the second barrier for containment isolation in the event of a high energy line break inside containment occurring upstream of the normally closed motor-operated gate valves. Once the LPCI and LPCS systems are in operation, the low energy of the influent fluid excludes any possibility of containment overpressurization should a break occur.

6.2.4.3.2.1.1.4 Control Rod Drive Lines

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

a. Isolation in the supply line is provided by a simple check valve inside containment and a remote manually-operated globe valve located outside the containment, and as close to the containment as practical.

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

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 into the shutdown (scram) function.

As a means of providing isolation, manual shutoff valves are used. In the event of a break in these lines, the manual valves would provide isolation capability. 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. Containment overpressurization will not result from a line break in the containment since these lines contain small volumes, resulting in relatively small blowdown masses.

6.2.4.3.2.1.1.5 RHR and RCIC Lines

The RHR head spray and RCIC lines merge outside the containment to form a common line which penetrates the drywell and discharges directly into the reactor pressure vessel. The check valve inside the drywell is normally closed. This check valve is located as close as practicable to the reactor pressure vessel. Two types of valves, a simple check valve and a remote-manually controlled motor operated gate valve, are located outside the containment. The check valve assures immediate isolation of the reactor coolant pressure boundary in the event of a line break. The gate valve in the RHR line is closed automatically on isolation signals, as given in Table 6.2-47. The gate valve in the RCIC line receives its closure signal from closure of either of the RCIC turbine steam stop or supply valves, whether they are closed automatically, remote manually, or due to high reactor vessel water, level 8. Redundant containment isolation is provided by RHR B and RCIC system piping and components which comprise a closed loop boundary.

6.2.4.3.2.1.1.6 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 isolated, non-flowing line, the rupture of this line is extremely improbable. However, should a break occur subsequent to the explosive valve actuation, the check valves ensure continuing isolation.

6.2.4.3.2.1.2 Effluent Lines

Effluent lines which form part of the reactor coolant pressure boundary and penetrate containment are equipped with at least two isolation valves; one inside the drywell, and the other outside containment and located as close to the containment as practicable.

6.2.4.3.2.1.2.1 Main and RCIC Steamlines

The main steamlines extend from the reactor pressure vessel to the main turbine and condenser system, and penetrate the primary containment. The RCIC turbine steamline connects to the main steamline inside the drywell and penetrates the primary containment. For these lines, isolation is provided by automatically actuated gate valves, one of these valves is located inside the drywell and the other valve is located just outside the containment.

6.2.4.3.2.1.2.2 Reactor Water Cleanup System Lines

The majority of the reactor water cleanup (RWCU) system is located in the containment, with the RWCU pumps located in the auxiliary building. The suction line from the recirculation system penetrates the drywell and containment walls. Two automatically actuated isolation valves, one located inside the drywell and one outside the containment, are provided to prevent releases to the auxiliary building; both valves are located as close to their respective walls as possible.

The discharge line from the RWCU pumps penetrates the containment. Inside the containment the line feeds the RWCU regenerative heat exchangers. Isolation criteria are satisfied by automatically actuated gate valves inside and outside the containment.

A blowdown line off the regenerative heat exchanger bypass line penetrates the containment. Outside the containment the line branches to connect separate lines to the condenser and radwaste system. Isolation criteria are satisfied by means of automatically actuated block valves inside and outside the containment.

The reactor water cleanup pumps, heat exchangers, and filter demineralizers are located outside the drywell. The return line from the filter demineralizers connects to the feedwater line outside the containment between the outboard shutoff valve and the outside containment feedwater check valve. Isolation of this line is provided by the feedwater system check valve inside the containment and a check valve and motor-operated gate valve outside the containment. The motor-operated gate valve functions as a third isolation valve.

During the postulated loss-of-coolant accident, valves will automatically close. 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.

6.2.4.3.2.1.2.3 Recirculation System Sample Lines

A sample line from the recirculation system penetrates the drywell. The sample line is 3/4-inch diameter and designed to ASME Section III, Class 2. A sample probe with a 1/8-inch diameter hole is located inside one recirculation discharge line inside the drywell. In the event of a line break, the probe acts as a restricting orifice and limits the escaping fluid. Two air-operated valves which fail closed are provided, one inside and one outside the drywell.

6.2.4.3.2.1.2.4 RHR Shutdown Cooling Line

The RHR shutdown cooling line connects to the recirculation system and penetrates the primary containment. Isolation is provided by two automatically actuated gate valves, one inside the drywell and the other outside the containment.

A small leak-off line from the RHR shutdown cooling (SDC) header, downstream of the outboard containment isolation valve 1E12F008, connects the header to the closed loop outside containment (CLOC) boundary of the RHR Division 2 water leg pump 1E12C003 minimum flow line. This line provides the operators with a means to contend with small leak rates past containment isolation valves 1E12F008 and 1E12F009, which can pressurize the SDC header and cause a high pressure alarm at the CPS main control room (MCR) annunciator 5064-8F. As a result, the section of the SDC header bounded by normally closed valves 1E12F008, 1E12F006A, 1E12F006B, and 1E12F067 are part of the RHR Division 2 CLOC boundary. Reference Figure 6.2-147 Sheets 1, 2, and 3.

6.2.4.3.2.1.3 Summary

In order to assure protection against the consequences of accidents involving the release of radioactive material, pipes which form the reactor coolant pressure boundary have been shown to provide adequate isolation capabilities on a case-by-case basis. In all cases, two or more barriers protect against the release of radioactive materials.

In addition to meeting the isolation requirements stated in Criterion 55, the pressure-retaining components which comprise the reactor coolant pressure boundary are designed to meet other appropriate requirements which minimize the probability or consequences of an accidental pipe rupture. The quality requirements for these components ensure that they are designed, fabricated, and tested to the highest quality standards of all reactor plant components. The classification of components which comprise the reactor coolant pressure boundary are designed in accordance with the ASME Boiler and Pressure Vessel Code, Section III, Class 1.

It is, therefore, concluded that the design of the piping system which comprise the reactor coolant pressure boundary and penetrate the containment satisfies Criterion 55.

6.2.4.3.2.2 Evaluation Against Criterion 56

Criterion 56 requires that lines which penetrate the containment and communicate with the containment atmosphere must have two isolation valves, one inside the containment and the other outside. It should be noted that this criterion does not reflect consideration of the BWR suppression pool design. Penetrations for certain engineered safety features do not provide inside and outside containment isolation valves. These penetrations are provided with two redundant mechanisms to assure that no significant release of radioactivity will occur. The isolation provisions for these penetrations comply with Criteria 56 which states that:

> "Each line that connects directly to the containment atmosphere and penetrates primary reactor containment shall be provided with containment isolation valves as follows, unless it can be demonstrated the containment isolation provisions for a specific class of lines, such as instrument lines, are acceptable on some other defined basis."

In situations where isolation valves inside and outside containment are not provided, the basis for this design is defined in this subsection.

6.2.4.3.2.2.1 Influent Lines to Suppression Pool

These lines are provided with motor-operated isolation valves outside the containment as shown in Table 6.2-47. The second containment isolation barrier consists of the piping outside of the containment which is a closed loop (CLOC). The CLOC will not permit any significant escape of radioactivity.

The CLOC is designed on the following basis:

a. The CLOC is designed to Seismic Category I, Safety Class 2 requirements.

b. The CLOC design temperature and pressure rating is at least equal to that of the containment.

c. Each CLOC is located in a missile protected room and is protected against pipe whip.

d. The rooms provide a leaktight housing, thus affording leakage control and flood protection.

e. Leakage detection is provided in the control room for each CLOC as described in Subsection 7.6.1.4.

f. The CLOC will be able to withstand the single active failure of any component in the CLOC.

Two other design features, besides the closed loop outside of the containment, provide additional isolation. All of these lines discharge below the expected low water level of the suppression pool. Thus the suppression pool will, at all times, provide a seal between the containment atmosphere and the lines for at least 30 days post-DBA. The water leg pumps of these systems also provide a positive water seal above containment design pressure and demonstrate that system integrity is being maintained during normal plant operation.

6.2.4.3.2.2.1.1 LPCS, HPCS, and RHR Test and Pump Minimum Flow Bypass Lines

The LPCS, HPCS, and RHR test return and pump minimum flow bypass lines have test isolation capabilities commensurate with the importance to safety of isolating these lines. Each line has a motor-operated valve located outside the containment. Containment isolation requirements are met on "other design basis" as described in Subsection 6.2.4.3.2.2.1.

The test return lines are also used for suppression pool return flow during other modes of operation. In this manner the number of penetrations are reduced, minimizing the potential pathways for radioactive material release. Typically, pump minimum flow bypass lines join the respective test return lines downstream of the test return isolation valve. The bypass lines are isolated by motor-operated valves in series with a restricting orifice.

6.2.4.3.2.2.1.2 RCIC Turbine Exhaust and Pump Minimum Flow Bypass Lines

The lines which penetrate the containment and discharge to the suppression pool are provided with a normally open, motor-operated, remote manually actuated gate valve located as close to the outside of the containment as possible. There is a simple check valve upstream of the gate 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 turbine inlet steam valve while the turbine exhaust valve is not in a full open position. The RCIC pump minimum flow bypass line is isolated by a normally closed, remote manually actuated valve with a check valve and a restricting orifice installed upstream.

6.2.4.3.2.2.1.3 RHR Heat Exchanger Vent and Relief Valve Discharge Lines

The RHR heat exchanger vent lines are capped off immediately inside the containment. The vent lines are part of RHR steam condensing which is not used at Clinton. The relief discharge lines are isolated by the relief valves themselves in a fashion similar to a check valve. The addition of block valves for isolation would therefore defeat the purpose for which the relief valves are installed. Additionally, the relief valve setpoints are greater than 1.5 times the containment design pressure.

6.2.4.3.2.2.2 Effluent Lines from Suppression Pool

The RHR, RCIC, LPCS, and HPCS suction lines contain motor-operated, remote manually actuated, gate valves located outside the containment, which provide isolation of these lines in the event of a break. System reliability is greater with a single isolation valve because the possibility of valve failure is reduced. Information available to the operator, which enables him to determine when the valves would be shut, includes leak detection from the leak detection system, RPV level to ascertain whether flow is actually reaching the RPV, and suppression pool level which would also indicate a line failure. The isolation provisions for these lines meet the same requirements as described in items a through e in Subsection 6.2.4.3.2.2.1. Submergence in the suppression pool also assures a water seal between these lines and the containment atmosphere for at least 30 days post-DBA.

6.2.4.3.2.2.3 Containment Influent and Effluent Lines

6.2.4.3.2.2.3.1 Combustible Gas Control

The combustible gas control system (CGCS) lines which penetrate the containment are each equipped with an automatic motor-operated butterfly valve outside the containment, normally closed, which can be 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. The piping outside of the containment for the CGCS is a closed loop meeting the same criteria given in Subsection 6.2.4.3.2.2.1. The closed loop outside the containment, in conjunction with the motor- operated isolation valve, provide the required two isolation barriers. These lines must be available for long-term use following a design-basis loss-of-coolant accident. System reliability will be higher with the isolation barriers provided.

6.2.4.3.2.2.3.2 Standby Liquid Control System

This line passes through containment and connects to the standby liquid storage tank. It is closed outside containment by a welded blind coupling.

6.2.4.3.2.2.3.3 Other Lines

All other lines penetrating the containment, which must be designed to Criterion 56, are provided with two automatic isolation valves as required by Criterion 56. These penetrations include the following systems. The equipment and floor drains system, suppression pool cleanup system, drywell purge system, containment building HVAC system, solid radwaste reprocessing and disposal system, component cooling water system, cycle condensate system, fire protection system, instrument air system, makeup condensate system, service air system, drywell cooling system, and plant chilled water system. These lines are included in Table 6.2-47.

6.2.4.3.2.2.3.4 Containment Atmosphere Monitoring System

The Containment Atmosphere Monitoring System lines which penetrate the containment are each equipped with two automatic solenoid operated, spring loaded, normally open, fail closed valves. These valves provide assurance of isolating the lines in which they are located, in the event of a break. In addition, the piping outside of the containment for the system forms closed paths with the following features:

a. The piping and the analyzer panel [Analysis Sample Conditioning Module (ASCM)] are designed to seismic category I requirements.

b. The piping and the tubing on the ASCM are designed to withstand the worst design-basis loss-of-coolant accident (LOCA) temperature and pressure condition.

c. The tubing on the ASCM is tested for leakage.

d. Leakage detection is provided in the control room by the area radiation monitors 1RE-PR019A thru E and the fuel building exhaust duct hi-rad monitors 1RIX-PR006A thru D.

The tubing inside ASCM and sample lines to and from ASCM are not postulated to break following LOCA. Since these lines must be available for containment atmosphere sample following a LOCA and loss of off-site power both isolation valves of the Containment Atmosphere Monitoring System are powered from a single division. However, since both valves fail closed on loss of power and there is a seismically designed loop outside containment in addition to the two isolation valves, this system contains sufficient provisions for containment isolation.

6.2.4.3.2.2.4 Summary

To assure protection against the consequences of accidents involving release of significant amounts of radioactive materials, pipes that penetrate the containment are provided with isolation capabilities in accordance with Criterion 56.

In addition to meeting isolation requirements, the pressure retaining components of these systems are designed to the same quality standards as the containment.

6.2.4.3.2.3 Evaluation Against Criterion 57

Lines penetrating the primary containment and for which neither Criterion 55 nor Criterion 56 is applicable, compose the closed system inside containment isolation valve group.

Generally, both influent and effluent lines are isolated by automatic or remote manual isolation valves located inside containment, as well as the valves located outside containment as close as possible to the containment boundary which satisfy Criterion 57. Certain lines not used during power operation are isolated by locked closed valves located inside and outside the containment. Provisions for detecting leakage from remote, manually-controlled systems for determining when to isolate are described in Section 5.2.5.

Influent and effluent lines which are evaluated against Criterion 57 include the shutdown service water system, the reactor water cleanup system, the component cooling water system, the plant chilled water system, and the drywell chilled water system. These lines are included in Table 6.2-47.

6.2.4.3.2.4 Evaluation Against Regulatory Guide 1.11

Instrument lines which penetrate the containment conform to Regulatory Guide 1.11, with the exceptions stated in Section 1.8 They are equipped with automatic isolation valves (excess flow check valves), whose status (open or closed) is indicated in the control room. These valves are included in Table 6.2-47.

6.2.4.3.3 Failure Mode and Effects Analyses

The containment isolation system is designed so that no single active or passive failure will prevent the system from performing its intended function. A minimum of two reliable barriers exists in all cases in order to maintain isolation of the containment.

For penetrations provided with two remote manual or remote automatic isolation valves, the valves are powered from separate divisional busses such that the active failure of a diesel generator will not disable both devices capable of isolating the penetration. The only exceptions to this rule are certain systems which are required to function continuously after the postulated LOCA such as the shutdown service water system and the Containment Atmosphere monitoring system. The shutdown service water system is a Safety Class 3 system inside containment. Thus, in order to violate the integrity of the containment, both an active failure of the diesel generator and a passive failure in the piping system would have to occur.

For penetrations provided with a single remote manual or remote automatic isolation valve and a closed loop outside of containment (CLOC), the isolation valve is powered from a divisional bus. If the valve could not be used to isolate the penetration, the CLOC would maintain the containment integrity. The failure of the CLOC would be mitigated by the use of the isolation valve. Only certain ECCS penetrations are designed in this fashion. The single isolation valve is used to ensure greater system availability.

For ECCS relief valve discharge lines which are returned to the containment and terminated in the suppression pool, the lines penetrate containment from above water level of the suppression pool. The discharge lines extend below the minimum vent coverage elevation of 727 feet 1 inch of the suppression pool. The active failure of the relief valve will not compromise the containment isolation since the water seal from the suppression pool will be maintained. The passive failure of piping will not adversely affect containment integrity since the valve and the water seal will be intact.

6.2.4.3.4 Calculated Secondary Containment Bypass Leakage

The initial functional capability of the containment isolation system with respect to meeting allowable limits for secondary bypass leakage was evaluated in response to USAR Q & R 480.06 and 480.07. The containment penetrations listed in Table 6.2-47 were evaluated for potential secondary containment bypass leakage. The process pipelines passing through these penetrations are classified as Class 1 or 2. The Class 1 pipelines terminate or are completely enclosed within the secondary conatinment and are not considered potential sources of bypass leakage. The Class 2 pipelines penetrate the secondary containment and are considered sources of potential bypass leakage. The evaluation performed in response to Q & R 480.07 concluded that the estimated potential secondary containment leak rate was less than the allowable leak rate.

To ensure the functional capability of the containment isolation system is maintained in accordance with regulatory requirements limiting secondary containment bypass leakage, the functional performance of the containment isolation system is evaluated on an approved schedule with the CPS Primary Containment Leakage Rate Testing Program. The CPS Primary Containment Leakage Rate Testing Program evaluates the potential secondary containment bypass leakage pathways identified in Table 6.2-47 as described in Section 6.2.6.

6.2.4.4 Tests and Inspections

The containment isolation systems will undergo periodic testing. The functional capabilities of power-operated isolation valves are tested by remote-manual operation from the control room. By observing position indicator and changes in the affected system operation, the closing ability of a particular isolation valve is demonstrated.

Testable check valves are employed for certain system influent lines. These valves are not considered containment isolation valves, they serve as pressure isolation valves. These valves are inside drywell and tested during plant shutdown to ensure functional capability when required for operation.

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

In addition, components in the containment isolation system will be tested for correct functional performance during the preoperational test program.

An inservice inspection program for the valves in the reactor coolant pressure boundary is described in Subsection 5.2.4.

6.2.5 Hydrogen Mitigation In Containment

In order to mitigate the affects of hydrogen generated during an accident, there are systems designed and operated in accordance with 10 CFR 50.44, Standards for Combustible Gas Control System in Light Water Cooled Power Reactors. These systems are used to monitor and control the concentration of hydrogen in the drywell and containment following an accident with either non-degraded or degraded core conditions.

Hydrogen generation by a LOCA which has a non-degraded core is controlled to a concentration less than the flammability limit of 4 volume percent in accordance with Regulatory Guide 1.7, Control of Combustible Gas Concentration in Containment Following a Loss-of-Coolant Accident. For this situation hydrogen mitigation is accomplished by the Combustible Gas Control System (CGCS). The CGCS includes provisions for promoting atmospheric mixing and for reducing combustible gas concentrations by use of a hydrogen recombiner. This system is an Engineered Safety Feature (ESF) system.

Hydrogen generation due to a degraded core is possible only following extensive core uncovery. It requires the simultaneous occurrence of either a LOCA or a transient event, and the failure of emergency coolant supply to the core. For this situation hydrogen mitigation is accomplished by the Hydrogen Ignition System (HIS). This system is not an Engineered Safety Feature (ESF) system since the conditions requiring operation of this system are beyond the scope of a design basis accident.

The H_2 Monitoring Subsystem, a part of the Containment Atmosphere Monitoring System, is provided to monitor and alarm at pre-determined hydrogen levels during events involving both non-degraded and degraded core conditions. The H_2 Monitoring Subsystem provides the capability of monitoring and indicating normal operating and postaccident wet, i.e. at ambient humidity, and hydrogen concentration in the drywell and containment. The H_2 Monitoring Subsystem is described along with the entire Containment Atmosphere Monitoring System in Subsection 7.6.1.10. The sample point locations for the H_2 Monitoring Subsystem are described in Subsection 6.2.5.1.2.3.

The CGCS and HIS are described in more detail in Subsections 6.2.5.1 and 6.2.5.2, respectively.

6.2.5.1 Combustible Gas Control System (CGCS)

In order to ensure that the drywell-containment integrity is not endangered due to the generation of combustible gases following a postulated LOCA, systems for controlling the concentrations of combustible gases are provided within the plant according to General Design Criterion 41 and Regulatory Guide 1.7. Combustible gas control includes provisions to ensure thorough mixing in both the drywell and containment atmospheres, and for reducing combustible gas concentrations within the containment using a hydrogen recombiner. (As a backup means of control, containment atmosphere can be purged through the standby gas treatment system.)

Also, the purge system may aid in long-term, post-LOCA containment cleanup.

6.2.5.1.1 Design Bases

The design bases considered for the Combustible Gas Control design follow:

a. In the event of a LOCA, the hydrogen concentration in the drywell may begin to increase due to a metal-water reaction, radiolysis of water, and material corrosion or decomposition. The Containment Air Monitoring System, when manually placed in the accident mode, will be operational within 30 minutes (Subsection 7.6.1.10.3).

b. The drywell and containment are not inerted during normal operation. Therefore, for a LOCA, 4% by volume hydrogen concentration is an upper limit. Consistent with Reference 22, the CGCS will be manually activated at or below 3.5% by volume hydrogen to prevent the hydrogen concentration from exceeding 4% by volume in either the drywell or containment atmospheres. The CGCS is the primary means of reducing the hydrogen concentration in containment with backup provided by controlled purging.

c. The CGCS is designed to perform in the event of failure of one of the emergency diesel generators. This failure causes loss of one of two redundant mixing compressors and one of two recombiners. The components of the CGCS are protected from missiles and pipe whip to assure proper operation under accident conditions as required for safety class systems. Continuous operation of the drywell/containment mixing system is assumed until containment environmental parameters return to normal.

d. The CGCS is designed to meet Seismic Category I requirements. The mixing compressors are designed to remain operable in the post accident environment in the containment building. All components that can be subjected to containment atmosphere are capable of withstanding the humidity, temperature, pressure, and radiation conditions in the containment (or in the drywell, for those components subject to drywell conditions) following a LOCA.

e. There are two redundant, independent 100% capacity mixing compressors and two 100% capacity recombiners. The recombiners are located outside of the containment in an accessible area. Routine maintenance, periodic testing and inspection can be performed during normal plant operation or shutdown conditions.

f. The hydrogen recombiner system units are permanently installed; therefore, it is not necessary to have the ability to transport any portable hydrogen recombiner units to the plant after a LOCA.

g. The CGCS recombiners are remotely started from either the main control room or from the local control panels which are located in the control building. The compressors are started only from the main control room. The controls and instruments are designed such that no local operating adjustments are required while operating in a LOCA environment.

h. The CGCS does not introduce safety problems that affect the drywell-containment integrity.

6.2.5.1.2 System Design

6.2.5.1.2.1 Principles of Operation

The combustible gas control system is based on the following concepts:

a. In the unlikely event of a LOCA, the hydrogen concentration in the drywell will begin to increase due to the release of hydrogen from metal-water reactions, radiolysis of water, material corrosion and material decomposition. The hydrogen concentration is indicated in the control room on the hydrogen monitoring instrumentation. The drywell containment mixing system will be manually activated so that the hydrogen in the drywell will be dispersed throughout the total containment volume, thereby ensuring the concentration remains below 4%.

   The alarm level for hydrogen in the drywell and in containment is at 1.0% by volume. The alarm setpoint is based upon (1) a study done according to Regulatory Guide 1.7, and (2) the accuracy of the hydrogen monitors as explained in Subsection 7.6.1.10.

b. The hydrogen concentration in the drywell and the containment would continue to increase due to radiolysis and corrosion process. The recombiners are manually started at 0.5% containment hydrogen concentration. The recombiner is a thermal system, using heat to recombine the hydrogen and oxygen to form water.

   The operation of the recombiner system reduces overall hydrogen concentration to ensure that 4% by volume is not exceeded. Mixing and natural turbulence in both drywell and containment atmospheres, resulting from diffusion and convection caused by the elevated temperatures, ensure that any hydrogen formed is homogeneously dispersed throughout either the drywell or containment.

c. As a backup to the hydrogen recombiner system, a low volume purge system can be used to remove the hydrogen-air mixture out of the containment through the Standby Gas Treatment System filter to the atmosphere. When the containment pressure is less than the ambient pressure, the gaseous mixture removed is replaced with air, thus reducing the overall hydrogen concentration.

6.2.5.1.2.2 Atmosphere Mixing

Mixing of the atmospheres within the drywell and within the containment is required to ensure that local concentrations with greater than 4% by volume hydrogen cannot occur following a LOCA.

The atmospheres in both the drywell proper and the containment are each well mixed. This assumption is supported by the test data reported in Reference 6. The mixing in the drywell is, prior to the actuation of the combustible gas control system, 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 drywell wall. The natural convective action is enhanced by the momentum of steam and water emitted from the point of rupture to the drywell.

6.2.5.1.2.2.1 Location of Combustible Gas Control System Suction and Discharge Points

The potential concentration difference between the drywell and containment areas is minimized via the hydrogen mixing system which is part of the CGCS. The effluent from the mixing system exhaust lines is routed to the suppression pool and exhausts below the water surface. The hydrogen mixing system takes its suction from the drywell resulting in a slightly lower pressure in the drywell than in the containment. Any leakage will be from the containment to the drywell. Vacuum relief lines located in the drywell will allow the containment atmosphere to flow into the drywell, thus completing the mixing cycle. This allows mixing of the containment and drywell contents, thereby minimizing any concentration difference.

The suction and discharge points of the CGCS are located close to the containment liner near the following azimuths and elevations:

Suction -

58°, 752,-0"

291°, 789,-0"

Discharge -

66°, 752'-0"

294°, 763'-6"

6.2.5.1.2.3 Containment Atmosphere H_2/O_2 Monitoring Subsystem Sample Points

As mentioned previously, the concentration of hydrogen will be the limiting parameter. The hydrogen concentration will be continuously monitored following the LOCA, and displayed in the control room. The H_2 Monitoring Subsystem is a part of the Containment Atmosphere Monitoring System which is described in detail in Subsection 7.6.1.10.

Hydrogen sample points are located in the following major compartments inside primary containment:

Drywell:	- general area (ceiling) - control rod drive area
Containment:	- above suppression pool - containment dome area

The following provides a more detailed description of the sampling areas:

Drywell

1. There is one sample point in the drywell general area just below the drywell ceiling at approximately 180° azimuth. This sample point is midway between the shield wall and the drywell wall to monitor hydrogen in the general area.
2. There is one sample point in the control rod drive area at about El. 734.
3. Deleted

Containment:

4. There is one sample point above the suppression pool at approximately 0° azimuth, below floor elevation near 755'-0" beneath the steam tunnel and closer to the drywell wall than the containment wall since the SRV discharge points are inboard. This provides sampling in potential pocket areas above the suppression pool.
5. To monitor hydrogen released from the suppression pool area into the dome region, one sample point is located in the containment dome area.

Figures 6.2-198 and 6.2-199 depict the general locations of these four sampling areas on general arrangement elevation views of the containment. The locations are indicated by circles circumscribed about the appropriate number of the sampling area as presented above. (Q&R 480.22)

6.2.5.1.2.4 Combustible Gas Control System

The primary means for controlling non-degraded core combustible gas concentrations following a LOCA will be accomplished by the use of a drywell-containment mixing system and a hydrogen recombiner system. Table 6.2-48 lists the design and performance data for combustible gas control system components.

6.2.5.1.2.4.1 Drywell-Containment Mixing System

The primary function of the mixing system is to mix the drywell and the containment atmosphere to ensure the hydrogen concentration will not exceed the flammability limit. The mixing system will accomplish the initial control of hydrogen in the drywell following a LOCA. The system is Safety Class 2, Seismic Category I. It is 100% redundant, with duplicate piping, equipment, and instrumentation divisionally separated in different quadrants of the containment for maximum functional independence and reliability.

The hydrogen mixing system consists of two redundant 800 cfm centrifugal air compressors exhausting from the drywell through two 6 inch lines. These exhaust lines are routed to the suppression pool and exhaust below the water surface. The two 100% systems are designed to meet IEEE 279-1971 requirements. The P&ID for the system is shown in Figure 6.2-146. Since the hydrogen mixing system is exhausted below the water surface in the suppression pool, steam from the drywell will be condensed.

The hydrogen mixing system takes suction from the drywell resulting in a slightly lower pressure in the drywell than in the containment and any leakage will be from the containment to the drywell. There are vacuum relief lines between the drywell and containment. These vacuum relief lines are located above the pool swell zone.

The vacuum relief valves are designed to start opening when the drywell pressure is approximately 0.2 psid less than the containment and will be fully opened when this differential pressure is 0.5 psid. This allows the containment atmosphere to flow into the drywell to complete the mixing cycle. There are four vacuum relief valve assemblies in parallel, each consisting of two valves in series. Since these valves serve a safety-related function, they meet the design quality assurance and redundancy requirements for an engineered safety feature. The valves perform a vacuum relief function during the operation of the drywell-containment mixing system.

6.2.5.1.2.4.2 Hydrogen Recombiner System

A thermal recombiner system will be used to control combustible gas concentration. The system, located on elevation 702 feet in the control building, will pump gases from the containment through a reaction chamber in which the temperature is maintained by radiant heaters. The gases reach a temperature sufficient to cause the hydrogen and oxygen to recombine to form water without flame. After passing through the reaction chamber, the gas is cooled and returned to the containment below the surface of the suppression pool.

The recombiner system has two Rockwell Thermal Hydrogen Recombiners each with 70 scfm flow capacity. The gas passes through a 7.5-hp blower into a coiled pipe located in an insulated, electrically heated enclosure where the temperature is raised to 1,325° F. The heated gas then passes into a reaction chamber where the hydrogen and oxygen combine to form water vapor. The gas then passes through a heat exchanger where it is cooled to 150° F and it is then returned to the containment.

These recombiners were qualified for operation at 70 scfm with a specific acceptance test for CPS. The results of this test are contained in report number N139TR12013, "Performance Acceptance Test Results." (Q&R 480.21)

The hydrogen recombiner unit 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 Section III, Class 2. The equipment is designed to meet Seismic Category I requirements. The system is designed to accommodate conditions present following a LOCA event. Piping and instrumentation for the system are shown in Figure 6.2-146. The unit is started up manually at hydrogen volumetric concentration at 0.5% in the containment and requires a 90 minute warmup period. Once placed in operation, the system continues to operate until it is manually shut down. The system can be operated from the control room or local panels. Testing consists of energizing the blower and heaters and observing system operation to see if components are performing properly .

The recombiner units include independent control switches located on panels in the main control room. All functions and controls necessary to start the combustible gas control system are located in the control room.

6.2.5.1.2.5 Purge System

The standby gas treatment system can serve as a backup to the combustible gas recombiner to ensure control of potential postaccident hydrogen concentrations.

This backup purge method, with a 300 cfm capacity, can treat gases from the containment and/or drywell when hydrogen concentration is less than 6%. The flow passes through charcoal beds and high efficiency particulate air (HEPA) filters to minimize release of .radioactivity, and hydrogen-free makeup air is supplied to the containment and/or drywell from the atmosphere. Provision is made to use the standby gas treatment system for the post-LOCA situation.

6.2.5.1.3 Design Evaluation

6.2.5.1.3.1 General

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

a. hydrogen generated in the post-LOCA environment,

b. resultant drywell and containment concentrations as a function of time if uncontrolled, and

c. the functional requirements of the CGCS.

The results of the evaluation of the CGCS follow:

a. The total residual decay power as a fraction of operating power plotted as functions of time are in Figure 6.2-125.

b. The beta, gamma, and beta plus gamma energy release rates plotted as functions of time are in Figure 6.2-126.

c. The integrated beta, gamma and beta plus gamma energy release plotted as functions of time are in Figure 6.2-127.

d. The integrated production of combustible gas within the drywell and containment plotted as a function of time for each source are Figures 6.2-128a, 6.2-128b, 6.2-129a and 6.2-129b.

e. The volume percent concentration of combustibles as a function of time with and without the operation of CGCS are in Figures 6.2-130a and 6.2-130b. When credit for operation of the CGCS is taken, only one of two redundant subsystems was assumed to be in operation. This subsystem consisted of a drywell containment mixer operating at 800 cfm, and a hydrogen recombiner, operating at 70 cfm with an efficiency of 96%, both of which are assumed to start at hydrogen concentrations that are conservative relative to the emergency operating procedures.

6.2.5.1.3.2 Sources of Hydrogen

The potential hydrogen sources after a LOCA are classified as short- and long-term generation sources.

6.2.5.1.3.2.1 Short-Term Hydrogen Generation

In the period after the LOCA, hydrogen gas will accumulate within the drywell and containment as a result of hydrogen generation from the zircaloy metal-water reaction, radiolysis of water, and corrosion of metals. The first item is considered to be the short-term hydrogen generation source. (The hydrogen gas generated by radiolysis of the coolant during pre-accident operation is swept down the steamline before the MSIV's close, and a negligible amount is released out the break.) The only short-term metal-water reaction considered to be significant is reaction of water with the zircaloy fuel cladding which produces hydrogen by the following reaction:

$$Zr + 2H_20 \rightarrow Zr0_2 + 2\ H_2$$

The result of the analyses of the Emergency Core Cooling System (ECCS) in conformance with 10 CFR 50.46 and Appendix K are included in Appendix 15D. According to the criteria set forth in Regulatory Guide 1.7, the reaction fraction is conservatively assumed to be 5 times the analyzed value or 0.00023 inches of cladding thickness, whichever is greater. The analysis reflected in Table 6.2-52 and Figures 6.2-128a/b, 6.2-129a/b is based on 0.945% of 59,379 pounds of zirconium, which is conservative for all expected core configurations up to and including GE14 fuel. The fraction of zirconium that reacts is determined by using SAFER/GESTR methods described in Reference 20 and shown on Table 6.3-3. The duration of this reaction is assumed to be 2 minutes with a constant reaction rate. The resulting hydrogen is assumed to be uniformly distributed in the drywell.

6.2.5.1.3.2.2 Long-Term Hydrogen Generation

Hydrogen can also be formed by metallic corrosion. Three cathodic reactions were considered important to the post-LOCA evaluation of hydrogen evolution in drywell and containment. These cathodic reactions are:

$$2H^+ + 2e \rightarrow H_2$$

$$0_2 + 4H^+ + 4e \rightarrow H_2$$

$$0_2 + 2H_20 + 4e \rightarrow 40H-$$

The reaction between aluminum or zinc (in the form of galvanized steel and zinc-rich primers) and the solution used for emergency core cooling and the containment spray system provided the only significant hydrogen source due to metallic corrosion.

Table 6.2-49 lists the mass and exposed surface area of the corrodible materials: aluminum, aluminum-alloys, galvanized steel, and zinc-rich paints used in the drywell and containment. Most of this inventory is subject to rapid wet-dry cycling in a post LOCA environment. Only the inventory which can be completely and continuously submerged in relatively still water will corrode by hydrogen evolution, therefore, a majority of the inventory will corrode but fails to evolve significant hydrogen gas in the process.

Tables 6.2-50a and 6.2-50b and 6.2-51a and 6.2-51b contain the temperature envelopes for the drywell and containment, respectively. The transient temperature profiles in these tables envelop the complete range of accidents (main steamline break, recirculation line break and intermediate line break). Also listed in these tables are rates for hydrogen evolution for aluminum, aluminum alloys and for zinc at pH values of 5.6 and 8.6. The reaction rates were calculated for two groups of aluminum alloys since the reaction rate is different for each of these groups. The reaction rate for aluminum alloy group 1 was determined based on a composite of the reaction rates of alloys 413, 360, 1100, 1100H11, 1100H14, 5005H34, 5052H19, and ASTM B-108-72. While aluminum alloy group 1 may include other alloys such as 3003, 3002, 383, 416, 5005, and 5052, the reaction rate is governed by the composite reaction rate described above. Aluminum alloy group 2 consists of alloys 6061-T6, 6051-T5, and 6063-T5. The reaction rate for group 2 is determined as 1.25 times the reaction rate of group 1. These hydrogen evolution rates are based upon experimental data and are applicable when the metals are completely and continuously immersed in solution. The pH values 5.6 and 8.6 are the normal operational limits imposed upon the reactor pressure vessel water chemistry.

6.2.5.1.3.3 Accident Description

The postulated events associated with hydrogen generation and control following a large or intermediate break LOCA are:

a. The hydrogen release to the drywell lasting 2 minutes during which 5 times the reaction fraction of Appendix 15D or 0.00023 inches of thickness, whichever is greater, of the zircaloy cladding reacts with water in the reactor core region.

b. The radiolytic decomposition of the postaccident emergency cooling solution including: coolant in the core region, water in the drywell and drywell sumps and the suppression pool water. The fission product distribution model and water decomposition requirement were taken from Regulatory Guide 1.7 (Rev. 2). The initial radioactive source strength and decay constants were obtained from TID 14844. The I^{132} source term was conservatively assumed to continuously generate from the long lived Te^{132}.

c. The thermal, chemical, and radiolytic decomposition of organic materials in the drywell and containment is accounted for as discussed in Subsection 6.1.2.

d. The corrosion of aluminum and zinc is assumed to be by hydrogen evolution.

If uncontrolled, the above hydrogen sources result in a drywell hydrogen concentration exceeding 4 volume percent 3 hours after a Large Break LOCA if the coolant pH equals 5.6, and 5 hours after a Large Break LOCA if the coolant pH equals 8.6. The drywell hydrogen concentration is prevented from exceeding 4 volume percent by the drywell-containment mixing system which is manually started after the initiating LOCA event per the emergency procedure. This results in dilution of the hydrogen rich drywell atmosphere with the containment atmosphere.

In time, however, the hydrogen concentration in the drywell will begin to rise due to both the production of hydrogen in the drywell and to the hydrogen contained in the atmosphere being returned to the drywell from containment by the mixing system. The concentration of hydrogen in the containment rises due to both the production of hydrogen in the containment and to the continuous operation of the mixing compressor. Drywell and containment concentrations are prevented from exceeding 4 volume percent by a 70 scfm manually actuated hydrogen recombiner. As indicated in Figure 6.2-130a, a hydrogen recombiner must be ready to begin recombining hydrogen 2 days after the initiating LOCA event.

6.2.5.1.3.4 Analysis

Based upon the above postulated events associated with hydrogen generation and control, the hydrogen concentration in the drywell and containment can be calculated as a function of time. In formulating the model of the Mark III containment for these calculations a number of additional assumptions are made.

The assumptions include:

a. No hydrogen is removed from the drywell except through the operation of the hydrogen control system mixing compressor.

b. The hydrogen concentration is calculated with the initial 50% humidity in the atmosphere prior to the LOCA (The effect of steam dilution is neglected).

c. The transient temperature profiles, used to define the hydrogen evolution rates from metallic corrosion, envelop the complete range of accidents (i.e., main steamline break, recirculation line break and intermediate line break).

d. The operational limits imposed upon the reactor pressure vessel pH are assumed when calculating metal corrosion. This is conservative because the pH is normally neutral in a BWR.

e. A conservative value of the product of mass of zirconium and reaction fraction is used in order to preclude the necessity for reanalysis each cycle.

f. An allowance for future increases in quantities of corrodable metal is included in the input to the analysis.

g. To account for possible entrapped water volumes containing corrodible metals in the containment, the drywell volume below the top of the weir is considered to be an entrapped water volume during a large break LOCA.

h. Because the H_2 Monitoring Subsystem measures wet hydrogen concentration, the analysis and manual actuation levels are based on wet hydrogen concentration. As indicated in Regulatory guide 1.7, the flammability limit is 4% by volume for either wet or dry conditions.

Tables 6.2-49 through 6.2-52 list the parameters used to determine the amount of hydrogen generated. The results of two analyses are summarized in Figures 6.2-125 through 6.2-130b. The hydrogen concentration includes the moisture from the 50% relative humidity at the initial conditions, which for the postulated design basis accident is conservatively low.

The post-LOCA radiolytic source terms used to determine the radiolytic decomposition of water in the containment were developed to be consistent with the requirements of Regulatory Guide 1.7, Revision 2 and are depicted in Figures 6.2-125 through 6.2-127.

The hydrogen concentration following a large break LOCA, pH equal to 5.6, and with and without the operation of the CGCS, is presented in Figure 6.2-130a. The maximum hydrogen concentration in the drywell remains below 4.0 volume percent with only a single CGCS train operating.

The hydrogen concentration following a large break LOCA, pH equal to 8.6, and with and without the operation of the CGCS, is presented in Figure 6.2-130b. The maximum hydrogen concentration in the drywell remains below 4.0 volume percent.

6.2.5.1.4 Tests 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 states of plant construction prior to initial startup (Chapter 14). These tests assure correct function of all controls, instrumentation, recombiners, piping, and valves. System reference characteristics, such as pressure differentials and flow rates are documented during preoperational tests and are used as base points for measurements in subsequent operational tests.

6.2.5.1.5 Instrumentation Requirements

The instrumentation provisions for actuating the combustible gas control system and monitoring the system are described in Subsection 7.3.1.1.7.

6.2.5.2 Hydrogen Ignition System (HIS)

Effective February 25, 1985, the NRC amended the hydrogen control requirements of 10 CFR 50.44 for all boiling water reactor facilities with Mark III type containments for which construction permits were issued prior to March 28, 1979. The revised rule (Reference 18) requires the installation of a hydrogen control system capable of accommodating an amount of hydrogen equivalent to that generated from the reaction of 75% of the fuel cladding (surrounding the active fuel region) with water, without loss of containment integrity. The HIS is designed to accomplish this goal.

CPS participated in a hydrogen control owners group effort to develop and implement a hydrogen control program for Mark III containments. This group conducted analytical and testing activities to support the hydrogen control program development.

As a result of participation in this group, CPS now has an operational hydrogen control system based on the igniter systems developed and employed at the Grand Gulf, Sequoyah, McGuire, and D. C. Cook Nuclear Stations. (Q&R 480.24)

6.2.5.2.1 Design Bases

The HIS is designed to ignite hydrogen in the unlikely occurrence of a degraded core event which results in the generation of excessive quantities of hydrogen from a large metal-water reaction in the reactor pressure vessel. The HIS is designed to burn hydrogen at low concentrations, thereby maintaining the concentration of hydrogen below that, if ignited, could lead to containment overpressurization failure. The potential for significant pocketing of hydrogen will be precluded by:

a. Utilization of distributed ignition sources; and

b. Mixing caused by turbulence resulting from localized burns and the churning effect of the containment sprays.

The HIS is designed with suitable redundancy such that no single active component failure, including failure of power supplies, will prevent functioning of the system. The HIS is comprised of 115 igniter assemblies which are powered from two Class 1E power distribution panels. Each panel supplies a division of the igniter assemblies. The HIS is designed to operate for a minimum of 168 hours following initiation in an accident condition.

6.2.5.2.2 System Description

6.2.5.2.2.1 Location Criteria

The locations of the igniter assemblies are based on the following criteria:

a. Hydrogen can be released to the containment atmosphere via the safety-relief valves which exhaust to the suppression pool. Therefore, igniter assemblies are located in a ring above the suppression pool as well as at other locations throughout the containment.

b. Hydrogen can be released directly to the drywell atmosphere via a pipe break in the drywell. Therefore, igniter assemblies are located throughout the drywell.

c. For enclosed areas within the containment, two igniter assemblies are located in each room with each igniter fed from a separate Class IE power distribution panel. Note that the fuel transfer tube valve cubicle at elevation 770', azimuth 165° does not contain igniters since it is isolated from the containment and contains no safe shutdown equipment.

d. For open areas in the drywell and containment volumes, the following location criteria are used:

    1. A maximum distance of 60 feet exists between adjacent igniters of a given electrical power division (Division 1 or Division 2) unless specifically justified otherwise.

    2. A maximum distance of 30 feet exists between adjacent igniters of the two different electrical power divisions unless specifically justified otherwise.

e. The igniter assemblies are supported to withstand, without loss of function, the loads associated with seismic events, hydrodynamic events, and also thermal and pressure loads due to cyclic hydrogen burns.

f. The hydrogen igniter assemblies located in high-traffic areas are adequately protected against physical damage to the igniter assembly and shall provide protection for personnel against physical injury.

g. All the igniter assemblies located in the vicinity of the pool swell zone are capable of operation after exposure to froth load. The igniters also perform their function during operation of the containment spray system.

h. The igniter location and mounting permit surveillance and maintenance activity while keeping radiation exposure to the operators as low as reasonably achievable (ALARA).

i. The igniter system is designed to ensure that adequate coverage is maintained in the event of postulated high energy line breaks.

6.2.5.2.2.2 Igniter Locations

Based on the criteria discussed in Subsection 6.2.5.2.2.1, evaluations have concluded that 115 locations in the containment and drywell require the installation of igniter assemblies.

6.2.5.2.2.3 Igniter Assembly Description

The igniter assemblies used in the HIS are divided into two components:

a. The igniter enclosure which partially encloses the igniter and contains the terminal block, transformer and associated electrical wiring and;

b. The junction box which contains the cable termination.

The approximate weight of each assembly is 28 pounds. A hooded spray shield is provided for protection against the containment sprays. The igniter enclosure, junction box, and spray shield are constructed of stainless steel. The enclosure is 1/8" thick, the junction box is 14 gauge. Gasketing material and sealant is provided to ensure leaktightness of the igniter enclosure and junction box. Access to the enclosure interior is through a removable plate.

The igniter chosen for the HIS is identical to those used at the Sequoyah and Grand Gulf nuclear plants. The transformer is rated 200 VA for 120 VAC, 60 Hz ($\pm$10%), with primary and multiple secondary taps at 6, 8, 10, 12, 13, 16 and 18 VAC.

6.2.5.2.2.4 Igniter Supports

The igniter assemblies will be supported to withstand, without loss of function, the loads associated with seismic and hydrodynamic events appropriate for the given location. The design of the wetwell igniters is such that no igniters are located below elevation 751'-0" which is more than 19'-6" above the normal pool water level. These igniters have been purposely located above the bulk swell zone as defined by GESSAR II to avoid loading due to poolswell impact and drag.

Additionally, igniter assemblies located in the froth zone (13 igniters) will be either protected from or supported and designed to withstand the loads associated with froth swell.

The design of the drywell igniters is such that no igniters are located below elevation 765'-6" which is 29'-9" above the top of the weir wall. These igniters have been purposely located to avoid the impact and drag loads resulting from a drywell negative pressure response. Per CPS-unique load definitions, the maximum weir swell height is 5 feet above the top of the weir wall.

Hence, the drywell igniters are located with a margin of over 24 feet above the maximum weir swell zone.

6.2.5.2.2.5 Power Supplies

The igniter assemblies are fed by 120VAC, 60Hz power from two divisional Class 1E distribution panels. These distribution panels receive their power from Class 1E transformers rated 480V/120-208V, 60Hz, 3-phase with grounded neutral. Each transformer is fed from divisional Class 1E, 480V, MCC feeder breakers, which can be powered from one of the station standby diesel generators.

Of the 115 igniter assemblies installed, 56 are powered from the Division 1 power panel and 59 from the Division 2 power panel. Furthermore, for each division, the igniter assemblies are connected through six circuits with each circuit comprised of two series connected breakers (for backup protection of electrical penetration) tied to a contactor. Five to twelve igniter assemblies are connected to each of these circuits. A control switch, one for each divisional group of igniter assemblies, is located in the control room to provide remote operation of the HIS.

Power from each of the contactor circuits is brought into the containment through electrical penetrations to junction boxes where power is distributed to individual igniter assemblies. Figure 6.2-179 provides a single line diagram of the igniter power supply for one division.

6.2.5.2.3 HIS Component Qualification

All components of the HIS are qualified for environmental conditions resulting from a postulated loss of coolant accident or high energy line break in accordance with IEEE 323-1974 and NUREG-0588.

In addition, igniter assemblies are dynamically qualified to ensure the operability and structural integrity of the assembly under the following conditions:

a. seismic events; and

b. hydrodynamic loads resulting from suppression pool transients.

The igniter assemblies comply with the requirements of IEEE Standard 344-1975 as well as Regulatory Guide 1.100.

The qualification program included radiation exposure testing, followed by thermal aging, wear aging, seismic and hydrodynamic testing and accident environment testing. The igniter assemblies were subjected to a LOCA environment as well as a simulated hydrogen burn. The igniter demonstrated its capability to ignite a combustible concentration of hydrogen and air mixtures of 4 to 12 percent hydrogen by volume. In addition, the igniter demonstrated the ability to burn hydrogen in varying concentrations over an applied voltage range of 120VAC±10%. Furthermore, as part of the program demonstrating CPS compliance with the Hydrogen Rule, it was analytically shown that the igniters are capable of performing their functions during and after exposure to the environmental conditions created by the burning of hydrogen.

6.2.5.2.4 System Operation

The HIS is designed to prevent the accumulation of hydrogen in concentrations that, if ignited, could lead to containment overpressurization failure. The operation of the RHR containment sprays during HIS operation will be based solely on containment temperature. The HIS is not required for events which result in the generation of hydrogen less than or equal to the amounts and release rates considered in the design of the Combustible Gas Control System as described in Subsection 6.2.5.1. It is intended, though, that the HIS be manually actuated for all event sequences which possess the potential to generate excessive amounts of hydrogen which can be communicated into the containment or drywell.

6.2.5.2.4.1 Initiation Criteria

The HIS is initiated by manual actuation of hand switches when RPV communication with the containment or drywell (primary containment control emergency operating procedure conditions) and RPV level can not be maintained above the top of active fuel, or when the hydrogen concentration exceeds 0.5%. The basis for using the top of the active fuel for initiating the HIS is that this will provide for a conservative amount of time for the glow plug to heat up to its functioning temperature, prior to the release of hydrogen. The basis for HIS initiation at a 0.5% hydrogen concentration is to actuate the system as hydrogen is generated. This will avoid the potential for excessive hydrogen build-up and pocketing. Also, this will ensure the hydrogen deflagration limit is not exceeded. There are two hand switches located in the main control room, one for each divisional set of igniters. The details of the initiation procedures have been incorporated into the plant Emergency Operating Procedures.

6.2.5.2.4.2 Duration of Operation

The HIS glow plug assemblies will attain a nominal temperature of at least 1700°F. The glow plug heat up period is on the order of a minute following manual activation. The system is capable of continuous operation for a minimum of seven days following manual activation in an accident condition. All components of the system are designed for forty years of intermittent operation, provided environmental qualification maintenance requirements are followed.

6.2.5.2.5 Tests and Inspections

6.2.5.2.5.1 Preoperational Testing

The HIS was preoperationally tested to ensure correct functioning of all controls, wiring and igniter components providing baseline data for subsequent surveillance testing and maintenance.

The test included energizing one of the two divisional sets of igniters from the control room and verifying that all igniters powered from the associated panel are functional. An identical testing procedure was followed for the second divisional set of igniters.

Functional testing of the HIS included, as a minimum, measuring and recording the following:

a. Surface temperature of each igniter to verify that it is operating at/or above 1700°F with 120 Vac applied.

b. Voltage and current drawn by each of six circuits feeding the igniters in each of the two divisions.

6.2.5.2.5.2 Surveillance Testing

At specified intervals during normal plant operation, as defined in the CPS Technical Specifications, each of the two power divisional sets of igniters is energized and individual igniter current and voltage are recorded.

In addition, all accessible igniter assemblies will be tested to verify a surface temperature of at least 1700°F per the CPS Technical Specifications.

6.2.5.2.6 Instrumentation and Controls

The HIS is manually initiated from the control room. Instrumentation for the HIS consists of two control room hand switches, one for each of the two Class IE power divisions. Each hand switch energizes the igniters in its respective division.

6.2.5.2.6.1 Trips

The HIS is capable of manual trip. The HIS also trips automatically on loss of power. System restarts on restoration of power can be accomplished after manual reinitiation by the hand switches.

6.2.5.2.6.2 Indication

The following indication is provided in the main control room:

a. HIS "ON" and "OFF" lights;

b. HIS "AUTO TRIP" light; and

c. HIS status "TROUBLE" light.

6.2.5.2.6.3 Alarms

HIS Division 1 and Division 2 Trouble alarms are provided in the main control room. These trouble alarms result from HIS Failure to Start, Loss of Power, and/or Auto Trip.

6.2.6 Containment Leakage Testing

The containment leakage testing program is summarized in this section. The program is implemented in accordance with the plant Technical Specifications and is consistent with the requirements of Appendix J to 10CFR50 (with the exception of exemptions granted for the main steam isolation valves and airlocks). The following subsections describe the tests performed as preoperational or surveillance type tests. The tests described include Type A (integrated leak rate test), Type B (penetration leak rate test). Type C (isolation valves leak rate test), and the drywell leak rate test.

6.2.6.1 Containment Integrated Leak Rate Test

A preoperational containment integrated leak rate test (ILRT) was performed in accordance with the requirements for Type A test outlined in ANS N45.4 and Bechtel Corporation Topical Report No. BN-TOP-I, Rev. 1, dated November 1, 1972. Additional guidelines were extracted from ANSI/ANS-56.8. The Type A test was performed following the structural acceptance test which is outlined in Subsection 3.8.1.7 of this USAR. Test procedure preparation complied with the requirements provided in Chapter 14.

Prior to the preoperational ILRT, all construction, repair, inspection, and testing of welded joints, penetrations and mechanical closures was completed. In addition, a general visual inspection to uncover evidence of structural deterioration which may affect the leak-tightness was performed prior to the preoperational Type A test and will always be performed prior to each performance of the periodic Type A surveillance test as required by the plant Technical Specifications. Any such structural deterioration and corrective actions taken are included as part of the test record. For purposes of the test, containment isolation provisions are aligned as close to the postaccident conditions as practicable.

Isolation valves are positioned according to their normal mode of operation. Systems that are part of the containment boundary which may be exposed to the postaccident containment atmosphere (and not needed for core cooling) are opened, vented, and/or drained to the appropriate atmosphere, if practical. Containment isolation valves in these systems are exposed to the test medium pressure and to the expected postaccident differential pressure. Containment isolation valves in systems that are not vented or drained during the Type A test, but which may be exposed to the postaccident containment atmosphere are Type C tested, and the Type C test leak rate for the penetration path is added to the Type A test results. Containment isolation valves which would be sealed by a fluid (suppression pool and FWLC) during postaccident conditions are likewise sealed during the test.

System configuration and valve positions are specified in the test procedure. Table 6.2-47 provides a listing of piping systems which penetrate the containment and identifies those which are not vented.

The containment is pressurized using a medium that is reasonably clean, dry, and free of contaminants. Pressurizing facilities are isolated and vented or disconnected during the pressure decay portion of the test. The location of temperature detectors and appropriate weighting factors is determined to provide optimum test data.

The integrated leak rate tests are performed at or above calculated peak accident pressure but less than the design pressure using the Absolute Method outline in ANSI/ANS-56.8. The tests are to confirm that the actual containment leak rate does not exceed 75% of the maximum allowable leakage from the containment (La) in 24 hours (actual test duration may be less than 24 hours, but not less than 8 hours following a 4-hour stabilization period). Computation of leak rate is performed using the mass point analysis technique discussed in ANSI/ANS-56.8. Instrumentation selection and placement; determination of instrument calibration, accuracy, and acceptability; and recording and reporting of test data follows the guidelines in ANSI/ANS-56.8 and requirements of the Technical Specifications.

In the event that the Type A test results fail to meet the applicable test criteria, the guidelines as listed in Appendix J of 10 CFR 50 and the Technical Specifications are followed.

System containment penetrations which typically may be not vented and drained for the Type A containment leak test are identified in Table 6.2-47 with notes 18, 20, 27, or 40. These systems are those for which operability is desired during the Type A test and typically includes the feedwater, RHR shutdown cooling, RHR injection, HPCS injection, LPCS injection, RCIC injection, fire protection, control rod drive, plant chilled water, drywell chilled water, containment HVAC, and other systems. For any penetration not in a post LOCA alignment, the minimum pathway leakage obtained from the Type C local leak rate tests will be added to the Type A test results.

Not all of these systems meet the Type A test requirements because they are not all considered as "closed systems" as specified in NUREG-0800, Section 6.2.4. (Q&R 480.16)

6.2.6.2 Containment Penetration Leak Rate Test

Containment penetrations whose design incorporates resilient seals, gaskets, flanged joints, or sealant compounds receive periodic Type B tests in accordance with Appendix J of 10 CFR 50, following the guidelines of ANSI/ANS-56.8. The following penetrations will be Type B tested:

a. equipment access hatch,

b. two containment personnel access locks,

c. fuel transfer penetration,

d. electrical penetrations, and

e. the containment pressurizing penetration (1MC-67).

f. the RT decontamination penetration (1MC-74).

Testing used in determining the leakage through these penetrations is described in the following paragraphs for each type of penetration. The physical description of each penetration is given in Subsection 3.8.1.1.

a. Equipment Access Hatch

The equipment access hatch is furnished with a double-gasketed flange and bolted dish door. The hatch has provisions to pressurize the space between the double gaskets of the door flanges.

b. Personnel Access Locks

There are two personnel locks which will be Type B tested, one is used for access to the refueling floor and the other is used for normal access to the containment building. Both locks are doubledoor assemblies with double-gasket type seals as described in Subsection 3.8.1.1 and Figure 3.8-13.

Type B testing of these containment penetrations includes the following:

1. leak testing of the personnel door gasketed seals by pressurizing the space between the seals to the pressure Pa. The doors will be closed and latched, but the inner door tiedowns will not be installed.
2. leak testing of the airlock assembly by pressurizing the space between the airlock doors to the pressure Pa. Tiedowns are required on the interior door for air lock tests greater than 2 psig. The tiedowns will be installed such that the test pressure will not unseal the interior door and to ensure that leakage past the door seals will be minimized. The restraining force on the door is not critical for the performance of this test, since the door seals are tested separately. The tiedowns are installed inside the airlock and no mechanism for monitoring the force on the door has been provided.

c. Fuel Transfer Penetration

This penetration consists of a 40-inch pipe sleeve containing the transfer tube. The tube and sleeve are connected inside the containment through a double-bellows assembly and a double O-ring sealed flange. During normal operation, the transfer tube is fitted with a double gasketed blind flange. Each of the two bellows assemblies is of the double wall type. The bellows and gaskets will be leak tested by pressurizing with air.

d. Electrical Penetrations

Electrical penetrations are provided with a leakage surveillance system. Included are provisions for pressurization between the double O-rings which seal the closure plate and the weld-neck flange which is welded to the containment (penetration) nozzle.

e. Containment Pressurizing Penetration

The containment pressurizing penetration is spool connected to the pressurizing source during ILRT. When not in use, this penetration is blank flanged on both the inside and outside of the containment, and is provided with a test connection to pressurize the volume for leak testing.

f. RT Decontamination Penetration

Penetration 1MC-74 is a spare penetration designed primarily for use in chemical decontamination, however it may be used for other purposes when it is necessary to penetrate primary containment in modes 4 or 5. The penetration is designed with a gasket flange inboard and outboard.

Test pressure for Type B testing will be the calculated peak accident pressure (Pa).

The acceptance criteria for preoperational and periodic testing is given in Subsection 6.2.6.3.

Type B testing will be performed at the intervals specified in Subsection 6.2.6.4.

6.2.6.3 Containment Isolation Valve Leak Rate Test

Containment penetrations are provided with isolation valving to meet the requirements of General Design Criteria 54, 55, 56. or 57. Type C testing will follow the guidelines in ANSI/ANS-56.8 and Appendix J of 10 CFR 50. Test procedure preparation complied with the requirements provided in Chapter 14.

Containment isolation valves tested during Type C testing will include those valves that are closed or that close automatically upon receipt of an isolation signal in response to controls intended to effect containment isolation or that operate under postaccident conditions to effect containment isolation. A listing of containment isolation valves is provided in Table 6.2-47.

Normally, isolation valves are tested in the “forward” direction, test pressure applied in the same direction as that when the valve would be required to perform its safety function. However, “reverse” testing, test pressure applied in the opposite direction as that when the valve would be required to perform its safety function, is permitted for certain cases. Valves typically tested in the reverse direction are those which provide the first of two containment barriers, which are tested by applying the test pressure between the barriers. All inboard gate valves inside containment may be tested in the reverse direction. Inboard globe valves may be reverse tested if the test pressure will try and lift the disc from the seat. Test connections may be tested in the forward or reverse direction. (Refer to FPR 201925, letter Y-97243, and S&L Calculation 01ME112 for details.)

In general, containment isolation valves which may be exposed to the post-LOCA containment atmosphere will be tested with air or nitrogen; containment isolation valves which will be sealed by water, or other fluid for at least 30 days following a LOCA, may be tested with water or the sealing fluid. The test data obtained from tests performed on sealed valves will be segregated and a separate acceptance criteria applied, based on a leakage limit for the sealing fluid. These leakage limits are controlled by the CPS Technical Specifications.

For penetrations which normally use water for leak rate testing, it is an acceptable alternative to test these barriers with air for convenience. Then the results will be reported to the NRC with the ILRT report but need not be included in the type B & C totals.

Justification: The bases for using a direct volumetric conversion is as follows. The equation used to provide the justification for this is Bernoulli obstruction theory obtained from Fluid Mechanics Second ED.: White F. M.: McGraw Hill. This equation can be written as:

$$\frac{Q}{A_2} = \sqrt{\frac{2(P_1 - P_2)}{x\left[1 - \frac{D_2{}^4}{D_1{}^4}\right]}}$$

Where:

Q = flow rate

$(P_1 - P_2)$ = the differential pressure in Psid

X = the density of the medium in Pounds mass per cubic Foot (lbm/ft3)

D_1 = full diameter of the test or component body

D_2 = orifice diameter

A_2 = the cross-sectional area of the orifice

Note it was assumed that the value of $\left[1 - (D)_2^4 / (D)_1^4\right] = 1$ due to the relative size of the orifice being several orders of magnitude smaller than the size of the body.

It is easy to see from this equation and with the assumptions made that the density of the fluid is the determining factor when determining the leakage rate. It has further determined that a 0.10291" diameter orifice will result in about 1 GPM water leakage at 10 Psid and that 0.10148" diameter orifice will result in about 100,000 SCCM air leakage at 9 Psid. Therefore converting an air leakage to water is a very conservative conversion.

Test pressure for Type C testing will be the calculated peak accident pressure (Pa) for valves which are not sealed and 1.10 Pa for valves which are sealed.

Provisions for Type C testing are provided as required by the General Design Criteria.

The acceptance criteria for Type C testing is specified in Appendix J of 10 CFR 50. The combined leak rate for all penetrations and valves subject to post-LOCA containment atmosphere and Type B and C test shall be less than 0.60 La. La is the maximum allowable containment leak rate at the calculated peak accident pressure. Leak rates from valves tested with water shall be excluded from the combined B and C leak rate. The testing methods for Type C tests may be pressure decay, flow rate or vacuum retention as described in ANSI/ANS-56.8.

Periodical Type C testing will be performed at the intervals outlined in Subsection 6.2.6.4.

The only containment isolation valves which are exempted from Appendix J Type C local leak rate testing are pressure relief valves which discharge beneath the expected low water level of the suppression pool, isolation valves for instrument lines and test connections, vents and drains with two barriers (one inch or less). The suppression pool will provide an adequate seal to prevent any atmospheric leakage for at least 30 days post-DBA. The inservice inspection program will provide assurance of the operability and integrity of the instrument line containment isolation valve provisions. The isolation provisions for instrument lines penetrating containment are in accordance with Regulatory Guide 1.11, with the exceptions stated in Section 1.8. The isolation provisions for test connections, vents and drains are in accordance with ANSI/ANS 56.8 to facilitate testing.

These connections consist of a double barrier (e.g. two valves in series, or one valve a nipple and cap, or one valve with a nipple and blind flange. These connections are part of the containment system barriers, but due to their infrequent use and multiple barriers, they do not require leakage rate testing as long as the barrier configurations are maintained using an administrative control program. However some test connections, vents or drains will continue to be included in the test program due to the ease of testing. (A test will not be performed for the sole purpose of testing a test connection, vent or drain. A test should be performed when a test connections, vent or drain is part of the LLRT test boundary. Test connections, vents or drains which will be included within the test boundary will be tested along with the other valves in the penetration. Any barriers such as pipe caps that are not administratively controlled shall be removed during any air testing.) Test connections, vents and drains identified as being water sealed shall be exempt from all local testing, since these valves are designed to be sealed for 30 days post LOCA from the suppression pool and are not to be considered Appendix J atmospheric leakage paths. The Inservice inspection program verifies the integrity of these valves through visual inspections.

MSIV leakage testing is conducted with the MSIV LC system testing which is discussed in Subsection 6.7.5. (Q&R 480.14)

6.2.6.4 Scheduling and Reporting of Periodic Tests

The periodic leakage rate test schedules for Type A, B, and C tests are given in the CPS Technical Specifications.

The preoperational leakage rate tests for Type A, B, and C were performed as late in the construction phase as possible, but prior to initial operation.

Type B and C tests may be conducted at any time during normal plant operation or during shutdown periods as long as the interval between tests does not exceed the maximum interval specified in Chapter 16. Each time a Type B or C test is performed, the total of Type B and C leakage rates is updated to reflect the most recent test results. Type A, B, and C tests results will be submitted to the NRC in a summary report in accordance with 10 CFR 50 Appendix J approximately 3 months after each test.

6.2.6.5 Special Testing Requirements

6.2.6.5.1 Drywell Leakage Rate Test

Following the drywell structural integrity test described in Subsection 3.8.3.7, a drywell preoperational leakage rate test was conducted at about drywell design pressure (30 psig). In addition, periodic drywell leakage rate tests are performed at a reduced pressure in accordance with the Technical Specifications. These drywell leakage rate tests verify over the life of the plant that the steam leakage bypassing the suppression pool for the full range of postulated primary system breaks is less than the maximum allowable leakage.

Drywell leakage rate tests are performed with the drywell isolated from the containment. The containment space exterior to the drywell will be near atmospheric pressure. The horizontal suppression pool vents were capped during the preoperational tests to achieve approximately design pressure in the drywell. The reduced pressure test pressure is less than that required to bubble drywell air through the horizontal vents. The drywell will be maintained at test pressure for a minimum of 1 hour to allow the drywell atmosphere to stabilize.

Following stabilization the leakage rate test will commence. Drywell leakage rate may be determined by the air flow or pressure decay method. The air flow method directly measures the makeup air required to maintain the drywell at the test pressure. This makeup air flow rate is then equal to the drywell leakage rate. The pressure decay method is based on calculating the leakage rate knowing change in pressure as a function of time for a known free air volume of the drywell which is given in Table 6.2-52. The leakage rate is also corrected for temperature of the drywell atmosphere.

Preoperational drywell leakage rate tests at design pressure (30 psig) and reduced pressure (3 psig) were performed as late as practical in the construction phase, but prior to initial operation.

The acceptable leakage rate at design pressure (30 psig) and $A/\sqrt{K}$ of 1.18 ft^2 was 13,640 scfm and applied for the pre-operational test. The acceptable leakage rate at reduced pressure (3 psig) and $A/\sqrt{K}$ of 1.00 ft^2 is 3654 scfm. As required by SER 6.2.1.7, these acceptable leakage rates are 10% of the maximum allowable leakage rate using the specified value of $A/\sqrt{K}$. The periodic reduced pressure drywell leakage rate test intervals are specified in Technical Specifications.

6.2.6.5.2 Bypass Leakage Testing

The test schedule and maximum allowable leakage rate for potential bypass leakage paths of the secondary containment are given in the CPS Technical Specifications.

6.2.6.5.3 Secondary Containment Testing

The secondary containment leakage test is part of the secondary containment functional test. The periodic test and test intervals to ensure the functional capability of the secondary containment are specified in the CPS Technical Specifications. The secondary containment preoperational testing is discussed in Subsection 14.2.12.1.

6.2.6.5.4 Isolation Valve Leakage Control System

Note: As a result of the re-analysis of the Loss of Coolant Accident (LOCA) using Alternative Source Term (AST) Methodology, it is no longer necessary to credit the Main Steam Isolation Valve Leakage Control System (MSIVLCS) for post-LOCA activity leakage mitigation. The system has been left in place as a passive system and is not required to perform any safety function.

6.2.6.5.5 Containment Penetration Draining

The following penetrations are drained to prevent possible overpressure due to thermal expansion of trapped fluid:

a. Penetration 1MC-088, Shutdown Service Water Cooling Water Return from Reactor Recirculation Pumps

b. Penetration 1MC-078, Shutdown Service Water Cooling Water Supply to Reactor Recirculation Pumps

c. Pentration 1MC-116, Standby Liquid Control Boron Make-Up

d. Penetration 1MC-48, Shutdown Service Water Supply to Combustible Gas Control System Room Cooling Coil Cabinet

e. Penetration 1MC-208, Shutdown Service Water Return to Combustible Gas Control System Room Cooling Coil Cabinet

f. Penetration 1MC-205, Shutdown Service Water Supply to Combustible Gas Control System Room Cooling Coil Cabinet

g. Penetration 1MC-204, Shutdown Service Water Return to Combustible Gas Control System Room Cooling Coil Cabinet

6.2.7 Suppression Pool Makeup System

The suppression pool makeup system provides water from the upper containment pool to the suppression pool by gravity flow following a LOCA. The quantity of water provided is sufficient to account for all conceivable postaccident entrapment volumes (i.e., places where water can be stored) while maintaining long term drywell vent water coverage. Normal pool makeup resultant from evaporation, etc. is accomplished by remote manual operation of the makeup valve from the cycled condensate storage system.

6.2.7.1 Design Basis

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

a. The system is to be redundant with two 100% capacity lines. The redundant lines shall be physically separated and the electrical power and control shall be separated into two divisions in accordance with IEEE-279.

b. The system shall be Safety Class 2, Seismic Category I, and Quality Group B.

c. The minimum long-term postaccident suppression pool water coverage over the top of the top drywell vents shall be 2 feet.

d. The suppression pool volume, between normal operation low level and the minimum postaccident pool level, plus the makeup volume from the upper pool shall be adequate to supply all possible postaccident entrapment volumes for suppression pool water.

e. The postaccident entrapment volumes causing suppression pool level drawdown shall include:

    1. The free volume inside and below the top of the drywell weir wall.

    2. The added water volume needed to fill the vessel from a condition of normal power operation to a post accident complete fill of the vessel including top dome.

    3. Volume in the steam lines out to the first MSIV for three lines and out to the second MSIV on one line.

    4. An allowance for containment spray hold up on equipment and structural surfaces.

f. No credit for feedwater or HPCS injection from RCIC storage tank shall be taken in calculating minimum postaccident suppression pool level.

g. The minimum normal operation freeboard distance from suppression pool high level to the top of the weir wall shall be adequate to store the upper containment pool makeup volume without flooding into the drywell over the weir wall in case of an inadvertant dump of the upper pool.

h. The minimum normal operation suppression pool volume at low level shall be adequate to act as a short-term energy sink without taking credit for upper pool dump.

i. The long-term containment pressure and suppression pool temperature shall take credit for the volume added post accident from the upper containment pool.

j. The system shall dump the makeup volume through one of two redundant lines within a time period such that minimum vent coverage is maintained with all ECCS pumps operating at maximum runout flow rate.

6.2.7.2 <u>System Design</u>

The piping system consists of two lines which penetrate the separator end of the upper containment pool through the side walls. One line is on either side of the separator pool and they are then routed down to the suppression pool on opposite sides of the steam tunnel. The elevation of the separator pool penetrations is such as to limit the volume of water which can be dumped to the lower pool. This volume limitation along with adequate weir wall freeboard ensures that no drywell flooding over the weir wall will occur for inadvertant opening of the valves on the suppression pool makeup lines.

The volume of the upper containment pool which is available for suppression pool makeup consists of the difference between normal water level and the drawdown water level shown in Drawing M05-1069. The refueling gate leading to the dryer storage fuel transfer pool may be open during power operation.

Each suppression pool makeup line has two normally closed valves in series. The series valves on a line are powered from the same division, and each line is a separate division. All electrical power is available from on-site emergency power sources which have divisional separation and redundancy.

Initiation of the dump valve opening when the low-low suppression pool water level is reached ensures adequate water volume to keep the suppression pool vents covered for all break sizes.

The upper pool is dumped by gravity flow after opening the two normally closed valves in series in each line. The valves on the two separate lines receive divisionally separate signals to open. The open signal for each valve is derived from either of two separate suppression pool level sensors. There are a total of four level sensors, two per division.

There is also a permissive permitting valve opening only when the LOCA signal exists and the keylock switch is in "Enable". This LOCA signal is the same signal which initiates actuation of the ECCS pumps. An additional signal will open the valves 30 minutes after a LOCA, even if the suppression pool level is not low.

This combination provides high reliability for the upper containment pool dumping when required by low probability of inadvertant dump by spurious signals. See Drawing M05-1069 for the system P&ID.

The two valves in series in each of the two makeup system dump lines are located in the containment near the top of the drywell and outside the range of suppression pool dynamic effects. The pipes terminate just below the lowest operating floor in the containment to provide an unobstructed free fall to the suppression pool surface. The termination is above the suppression pool high level in order to avoid any air clearing loads. The pool dynamic loading on the makeup system pipe is expected to be relatively small due to the minimum drag cross section of the vertical open ended cylindrical geometry. The pipe schedule and support design include the effects of internal pressure, seismic loads, and pool dynamic loads near the suppression pool surface.

6.2.7.3 System Evaluation

6.2.7.3.1 Initiation

The opening of the makeup system valves is signaled by a series combination of low-low suppression pool level and a LOCA signal permissive (further discussion in Subsection 6.2.7.2). The low-low level signal has an allowable value of 18 inches below the normal low level. Due to instrument inaccuracies, the actual initiation level is between .8 to 18 inches below the normal low level. Since maximum ECCS pump flow lowers the suppression pool at a rate of approximately 0.54 ft/min, there is a minimum 7.4 seconds delay between start of ECCS flow and dumping of the upper pool.

The delay is actually 1-2 minute longer than this because vessel inventory mass is added to the suppression pool during blowdown steam condensation. This built-in delay assures that the drywell pressure increase due to vessel blowdown has been terminated by vent clearing prior to dumping of the upper pool and the corresponding increase in vent submergence.

6.2.7.3.2 Long-Term Vent Coverage

The suppression pool makeup system is capable of maintaining a minimum long-term vent coverage of 2 feet with only one of the makeup lines assumed to be operative in the event of a main steamline break.

6.2.7.3.3 Inadvertent Dump

The design of the opening signal for the suppression pool makeup valves assures high probability that no inadvertent dump will occur. The suppression pool low-low level signal to open the valves is in series with a permissive which only allows the open signal to pass through when a LOCA signal exists on that division. A simultaneous signal of suppression pool low-low level and LOCA will automatically open both pairs of valves in series in each of the dump lines to allow gravity drain of the upper pool to the suppression pool.

Each dump valve is provided with a remote manual control switch in the main control room which will open the valves. Thus, the upper pool can be dumped manually in accordance with IEEE-279, however, there is still single failure protection against inadvertent dump.

There are four level sensors measuring suppression pool water level with two sensors per electrical division. Level sensors on one electrical division cannot initiate flow from the makeup line whose valves are in a separate electrical division.

Pool swell induced structural loading will occur during the first ten seconds following a large break LOCA. Thus, the structural loading would occur prior to any significant flow of water from a makeup line which was erroneously signalled to open at the same instant as the break.

The peak structural loadings associated with smaller breaks are all less than the large break case and only slightly extended in time.

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

An inadvertent dump of the upper pool during any period of plant operation with a pressurized vessel does not represent, in and of itself, any hazard to the public, the plant operating personnel or any plant equipment. The drywell weir wall has sufficient freeboard height between the suppression pool high water level and the top of the weir wall to store the entire upper pool makeup volume without flooding over the weir wall into the drywell. The only concern is for the extremely low probability that a LOCA might occur during this period of high vent submergence following inadvertent dump. The dumped upper pool makeup volume can be transferred back to the upper pool through the RHR pumps, thus restoring the initial suppression pool water level.

No spent fuel is stored in the upper pool during plant operation so shielding is not an issue for this case. New fuel, while stored in the racks, does not require water for shielding purposes. Spent fuel can be temporarily stored in one end of the upper pool (steam dryer storage pool) during fuel transfer as part of the refueling operation. This storage area has sufficient water depth that adequate shielding is maintained over the fuel even following inadvertent dump of the upper pool makeup volume to the suppression pool. The separator storage pool wall limits the water height drop over the temporarily stored fuel in the dryer storage pool. The remaining water provides shielding over the top of active fuel temporarily stored even after inadvertent dump.

The only inadvertent dump event which represents a possible hazard to plant operating personnel is a dump event which occurs while fuel is in an elevated position, such as for transit between the reactor cavity and the fuel transfer pit. A 6-foot upper pool water level drop with a fuel bundle in the highest position leaves approximately 2 feet of water shielding over the top of the active fuel. This is adequate for bundle cooling but could represent a potential radiological hazard to operating personnel. Radiation alarms at the top of the upper pool would warn personnel of the high radiation levels. Sufficient time would be available for personnel to step to a safe shielded area out of line of sight of the suspended fuel bundle. The valve initiation logic includes an SPMS modeswitch which is administratively controlled and may be keylocked in the "DISABLE" position during refueling operations to prevent inadvertent opening of the dump valves. An alarm sounds in the control room if the SPMS mode switch is not in its ENABLE position.

6.2.7.3.4 Long-Term Heat Sink Capabilitv

The capacity of the RHR heat exchangers to safely limit the long term, post-LOCA suppression pool heatup transient is evaluated on the basis that the drawdown makeup system is activated early in the transient. Specifically, the evaluation assumes that the heat exchangers are activated 30 minutes after the LOCA, and that at this time the drawdown makeup system water has been added to the suppression pool inventory. The makeup 30 minute timer will ensure that this condition will exist. The dump period is not significant compared to the time it takes for the suppression pool peak temperature to be reached.

6.2.7.3.5 Mode 3 Suppression Pool Makeup

Draining of the reactor cavity portion of the upper containment pool is allowed in Mode 3 with reactor pressure < 235 psig. During this condition the allowable suppression pool level band must be raised to allow the suppression pool makeup function to be met by the combined volume of the suppression pool and upper containment pool.

A higher lower limit on the suppression pool level maintains the minimum vent coverage. The higher upper limit still retains acceptable drywell freeboard to prevent equipment from being submerged. The increased suppression pool level increases the hydrodynamic loads on the containment following a LOCA or SRV discharge. However, with the reactor pressure ≤ 235 psig, the loads are bounded by those from a full reactor pressure and power design basis accident.

In the small break LOCA with drywell bypass event, the higher suppression pool level increases the analyzed drywell pressure that does not clear the top row of horizontal vents. The higher pressure drives more steam through the bypass path. This is countered by the lower reactor pressure and lessened cool-down time.

The suppression pool provides the heat sink for the decay and sensible heat released by a LOCA. During this Mode 3 allowance, there is a reduction in the long term amount of water in the suppression pool following the accident. In order to compensate for the reduction in upper containment pool water inventory, there is a reduction in the amount of water entrapped by reducing the amount of water filling the reactor vessel. The reactor vessel is assumed filled to Level 8, which is consistent with Clinton Emergency Operating Procedures.

Analysis of the effect that the reduction in long term amount of water has on containment shows less than a 2 °F increase in the suppression pool temperature as a result of the inventory reduction. (Reference 29, 30)

6.2.7.4 Testing

The suppression pool makeup valves can be periodically manually tested, one at a time, during plant power operation. Each dump valve is provided with a separate remote manual control switch which along with administrative procedures will prevent both valves in series on the same line from opening simultaneously. The test will verify that the valve will open and close.

Instruments will be periodically tested and inspected.

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

6.2.7.5 Instrumentation

There are four suppression pool level sensors, and four suppression pool instrumentation channels, two per division that monitor suppression pool level, which provide the suppression pool makeup function. Two of these channels are from the Suppression Pool Makeup System (SM), one channel in each division; the other two channels are from the Containment Monitoring System (CM), also with one channel per division. Each level sensor provides a signal to a trip unit, which in turn, contributes to a one-out-of-two actuation signal, which will initiate the opening of both valves in either line, provided that a LOCA signal is present.

In addition, the level channels are used to continuously monitor suppression pool level. Each channel level indication is recorded in the control room.

Each level channel consists of (1) a differential pressure transmitter mounted locally in the auxiliary building with sensing lines which penetrate the primary containment (suppression pool wall) at a level below the low-low water level and at an upper elevation in the containment; (2) power supply and current alarm trip unit located in the main control room; and (3) information outputs located in the control room. The four level sensors are distributed around the suppression pool.

A level alarm for the upper pool is also provided to obtain the attention of plant operating personnel if the level drops below the required level. Level in the upper pool is normally maintained by a continuous overflow of level control weirs. The level is expected to stay nearly constant during plant power operation.

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

6.2.7.6 Materials

The piping which penetrates the separator storage pool and welds to the stainless steel pool liner is stainless steel to the first series valve. Both valve and subsequent downstream piping are carbon steel.

6.2.8 Humphrey Concerns

The Humphrey concerns, raised in 1982, are alleged safety issues related to the Mark III containment design. There are 66 individual concerns covering 22 major areas. The resolutions are in the CPS licensing basis contained in SSER 6, Appendix O. The issues include loads from pool swell on encroachments, loads on submerged structures from encroachments, loads on SRV discharge lines, loads created by RHR and other relief valves, drywell pool formation post LOCA, suppression pool thermal stratification, containment spray cooling capacity and operation, drywell bypass leakage, upper pool dump impacts, hydrogen mitigation, conflicts between accident analysis inputs and technical specification allowable conditions and EOP's, small break accident stresses, drywell flooding from the suppression pool, LPCI check valve failure, suppression pool temperature indication, insulation blockage of the weir gap and suction strainers and chugging loads. The information in SSER 6, Appendix O, is historical. Changes in analysis or configuration should be evaluated against the resolutions to ensure the issue remains closed. The resolutions do not need to be updated.

6.2.9 References

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5. D. K. Sharma, "Technical Description: Annulus Pressurization Load Adequacy Evaluation," General Electric Document No. NEDO-24548, Class I, January 1979.

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8. L. F. Parsly, "Design Considerations of Reactor Containment Spray Systems - Part VI. The Heating of Spray Drops in Air-Steam Atmospheres", ORNL-TM-2412, Part VI, Oak Ridge National Laboratory, 1970.

9. Regulatory Guide 1.7, "Control of Combustible Gas Concentrations in Containment Following a Loss-of-Coolant Accident," Revision 2, November 1978.

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13. Regulatory Guide 1.70, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants," Revision 3, November 1978.

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17. “Minimum Containment Pressure Analysis for Emergency Core Cooling System Capability Studies,” NUREG-75/087, Section 6.2.1.5, Revision 1.

18. Federal Register, Volume 50, Number 17, January 25, 1985.

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23. Final Analysis of Clinton Power Station Compliance With the Hydrogen Rule, Revision 0, March 1994.

24. NEDO-32686, "Utility Resolution Guidance for ECCS Suction Strainer Blockage," November 1996.

25 NEDE-24655, Technical Description Annulus Pressurization Load Adequacy Evaluation for Clinton Power Station Units 1 and 2, May 1979.

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28. DC-ME-09-CP, "Equipment Environmental Design Conditions Design Criteria."

29. NRC Safety Evaluation related to Tech. Spec. Amendment 156 for upper containment pool draindown in Mode 3 with RPV pressure < 235 psig.

30. NAI 8907-09, Revision 5, "GOTHIC Containment Analysis Package Qualification Report," Version 6.1, July 1999.

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TABLE 6.2-1
CONTAINMENT DESIGN PARAMETERS

		DRYWELL	CONTAINMENT
A.	DRYWELL AND CONTAINMENT		
1.	Internal Design Pressure, psig	30	15
2.	External Design Pressure, psig	17	3
3.	Design Temperature, °F	330	185
4.	Net Free Volume, ft^3	241,699	1,512,341
5.	Design Leak Rate	115,574 scfm at 30 psig	0.50%/day
6.	Maximum Acceptable Leak Rate	3654 scfm at 3 psig	La=0.65%/day*
7.	Suppression Pool Water Volume, ft^3	(1) LWL 10,707	135,220
		HWL 10,934	138,806
8.	Suppression Pool Surface Area, ft^2	454.88	7,174.6
9.	Suppression pool depth, ft		
	Low Water Level	18.92	18.92
	High Water Level	19.42	19.42
10.	Upper pool makeup volume, ft^3		14,748
11.	Secondary Containment Bypass Path Leakage, % of Maximum allowable leak rate		8%
B.	VENT SYSTEM		
1.	No. of Vents		102
2.	Nominal Vent Diameter, ft		2.29
3.	Total Vent Area, ft^2		420
4.	Vent Centerline Submergence (Low Level), ft		
	- Top Row		7
	- Middle Row		11.5
	- Bottom Row		16
5.	Vent Loss Coefficient (Varies with no. of vents open)		2.5 - 5

Note (1) Including horizontal vents

*Exclusive of MSIV leakage.

TABLE 6.2-2
ENGINEERED SAFETY SYSTEMS INFORMATION
FOR CONTAINMENT RESPONSE ANALYSES

	FULL CAPACITY	CONTAINMENT CASE A[(1)]	ANALYSIS VALUE CASES B1[(1)] and B2
A. CONTAINMENT SPRAY			
1. Number of RHR Pumps	2	0	0
2. Number of Lines	2	0	0
3. Number of Heaters	2	0	0
4. Flow rate, gpm/pump	3800	0	0
B. Containment cooling system:			
1. Number of RHR Pumps	2	2	1
2. Pump Capacity, gpm/pump	5050	5050[(2)]	5050[(2)]
3. RHR Heat Exchangers			
a. Type – Inverted U-tube, single pass Shell, multi-pass Tube, vertical mounting			
b. Number	2	2	1
c. Not Used			
d. Not Used			
e. Service Water Flow-rate, gpm/unit	5800	5800	5800
f. Service Water Temperature, °F			
Minimum Design	32		
Maximum Design	95	95	95
g. Containment Heat Removal Capability per unit, using 95°F Service Water and 185°F Pool Temperature, Btu/hr	116.7×10^6		

Note:

(1)See Subsection 6.2.1.1.3.3.1.6

(2)Case B2 was analyzed assuming that 500 gpm of the RHR pump flow is diverted from suppression pool cooling to the feedwater leakage control system. The heat transfer rate used in cases A and B1 has been demonstrated to be acceptable with the reduced flow.

TABLE 6.2-3
ACCIDENT ASSUMPTIONS AND INITIAL CONDITIONS FOR CONTAINMENT RESPONSE ANALYSES (CASE B2)

A.	Components of Effective Break Area (Recirculation Line Break) ft^2		
	1.	Recirculation line safe end	1.773
	2.	Cleanup line	0.080
	3.	Jet pumps	0.354
B.	Primary Steam Energy Distribution*, 10^6 Btu		
	1.	Steam energy	21.15
	2.	Liquid energy	266.5
	3.	Sensible energy	
		a. Reactor vessel and piping	74.9
		b. Reactor internals (less core)	62.1
		c. Fuel**	6.0
C.	Other Assumptions Used in Analysis		
	1.	MS closure time, sec	
		a. Recirculation line break	3.5
		b. Main steamline break	5.5
	2.	Scram time, sec	< 1

* All energy values except fuel are based on a 32°F datum.

** Fuel energy is based on a datum of 285°F.

TABLE 6.2-4
INITIAL CONDITIONS EMPLOYED IN CONTAINMENT RESPONSE ANALYSES (CASE B2)

A. REACTOR COOLANT SYSTEM: (at 102% of rated power and normal liquid levels)

1.	Reactor power level, MWt	3,543*
2.	Average coolant pressure, psia	1,040
3.	Average coolant temperature, °F	549
4.	Mass of reactor coolant system liquid, lb	485,707
5.	Mass of reactor coolant system steam, lb	17,753
6.	Volume of liquid in vessel, ft^3	8,977
7.	Volume of steam in vessel, ft^3	6,356
8.	Volume of liquid in recirculation loops, ft^3	580
9.	Volume of steam in steamlines, ft^3	1,221
10.	Volume of liquid in feedwater system, ft^3	13,246
11.	Volume of liquid in miscellaneous lines, ft^3	88

(Continued)

TABLE 6.2-4 (CONT'D)

B. CONTAINMENT

	DRYWELL	CONTAINMENT
1. Pressure, psig		
Short-term	-0.2	0
Long-term	1.0	0
2. Air temperature, °F		
Short-term	150	95
Long-term	150	122
3. Relative humidity, %		
Short-term	25	100
Long-term	25	40
4. Suppression pool water temperature, °F	95	95
5. Suppression pool water volume, ft^3		
High water level	10,934**	138,806
Low water level	10,707**	135,220
6. Top row vent centerline		
High water level	7.5**	7.5
Low water level	7**	7
7. Upper pool water temperature, °F	NA	120
8. Upper pool makeup water volume, ft^3	NA	14,748

Note

* Some results for non-limiting analyses performed at 102% of the original licensed thermal power (2952 MWt) are also included where corresponding analyses at 3543 MWt were not done.

**The drywell-side initial suppression pool volume and vent submergence correspond to equal pressure between the drywell and containment. The actual values used in the containment response analyses varied from these values due to the initial pressure difference between the drywell and containment.

TABLE 6.2-5
SUMMARY OF SHORT TERM CONTAINMENT RESPONSES TO RECIRCULATION LINE AND MAIN STEAMLINE BREAKS MINIMUM ECCS (CASE B2)

		RECIRCULATION LINE BREAK	MAIN STEAM LINE BREAK
1.	Peak drywell pressure, psig	21.30	22.23
2.	Time of peak drywell pressure, sec	1.745	2.052
3.	Peak drywell differential pressure, psid	20.09	21.38
4.	Time of peak drywell differential pressure, sec	1.745	2.052
5.	Peak drywell atmospheric temperature, °F	248.6	335.14*
6.	Peak wetwell pressure, psig	5.06	9.22
7.	Time of peak wetwell pressure, sec	2.950	3.005
8.	Calculated drywell margin	29.0	25.9

Note

* In the short-term, a higher drywell atmosphere temperature peak (335°F) occurs briefly (0.4 seconds) during the blowdown on a main steam line break. This temperature does not present a threat to the drywell structural materials during this short duration because it takes a much longer time for the drywell structural materials to increase to the limit.

TABLE 6.2-6
SUMMARY OF LONG-TERM CONTAINMENT RESPONSES TO
RECIRCULATION LINE OR MAIN STEAMLINE BREAKS (at 2952 MWt)

		CASE A	CASE B1
1.	Peak containment pressure, psig	4.60	8.74
2.	Time of peak containment pressure, sec	6515	36953
3.	Peak suppression pool temperature, °F	155.5	180.3
4.	Calculated containment margin, %	69.3	41.7
5.	HPCS flow rate, gpm	4900	4900
6.	LPCS flow rate, gpm	4900	0
7.	RHR flow rate, gpm	10100	5050

Note: Case A assumes all ECCS equipment operating.
Case B1 assumes LOOP and minimum ECCS equipment operating.

TABLE 6.2-6a

SUMMARY OF LONG-TERM CONTAINMENT RESPONSE TO RECIRCULATION LINE AND MAIN STEAMLINE BREAKS (at 3543 MWt - CASE B2)

	RECIRCULATION LINE BREAK	MAIN STEAM LINE BREAK
1. Peak containment pressure, psig	3.92	6.97
2. Time of peak containment pressure, sec	53,249	98,304
3. Peak suppression pool temperature, °F	182.2 [(1)]	181.2
4. Calculated containment margin, %	73.9	53.5
5. RHR flow rate, gpm	4,550	4,550

Note: Case B2 assumes LOOP and minimum ECCS equipment operating with the exception of the peak suppression pool temperature. A more limiting single failure is the RHR heat exchanger alone. If the full compliment of ECCS pumps run, a concervative amount of pump heat and sensible heat is transferred to the suppression pool.

TABLE 6.2-7
ENERGY BALANCE FOR MSL BREAK ACCIDENTS
(at 2952 MWt)

		ENERGY, Btu			
		INITIAL TIME	DW PEAK PRESSURE	END OF BLOWDOWN	WW PEAK PRESSURE
1.	Reactor Coolant (Vessel and Piping Inventory)	4.3E 08	4.1E 08	4.8E 07	1.6E 08
2.	Fuel and Cladding				
	Fuel	6.0E 06	6.0E 06	2.2E 06	0.
	Cladding	2.9E 06	2.9E 06	2.0E 06	1.4E 06
3.	Core Internals, also Reactor Coolant Piping, Pumps and Valves	7.9E 07	7.9E 07	7.8E 07	2.5E 07
4.	Reactor Vessel Metal	7.6E 07	7.6E 07	7.5E 07	2.5E 07
5.	Reactor Coolant System Piping Pumps and Valves	Included in (3)			
6.	Blowdown Enthalpy				
	Liquid	0.	3.8E 06	5.2E 08	5.2E 08
	Steam	0.	8.8E 06	9.2E 07	9.2E 07
7.	Decay Heat	0.	2.5E 06	6.4E 07	1.4E 09
8.	Metal-Water Reaction Heat	0.	1.3E 04	1.4E 06	1.4E 06
9.	Drywell Structures	0.	0.	0.	0.
10.	Drywell Air	1.6E 06	1.9E 06	6.7E 00	1.1E 06
11.	Drywell Steam	6.7E 05	9.3E 06	1.5E 07	7.2E 06
12.	Containment Air	1.0E 07	1.0E 07	1.2E 07	1.6E 07
13.	Containment Steam	4.0E 06	4.0E 06	1.0E 07	3.3E 07
14.	Suppression Pool Water*	6.5E 08	6.5E 08	9.9E 08	1.2E 09
15.	Energy Transferred by Heat Exchangers	0.	0.	0.	9.6E 08
16.	Passive Heat Sinks	0.	0.	0.	0.

* In response to NRC Bulletin 96-03, Hardware Change No. M-083 installed a new strainer, which rests of the floor of the suppression pool, to replace the individual strainers for the ECCS and RCIC system pumps. The new strainer displaces ~ 500 ft^3 of suppression pool water. Analysis has shown that the removal of the water does not invalidate the short-term or long-term containment LOCA response analyses.

TABLE 6.2-8
ACCIDENT CHRONOLOGY FOR
MAIN STEAMLINE BREAK ACCIDENT
(at 2952 MWt)

	EVENT	CASE A ALL ECCS IN OPERATION	CASE B1 MIN ECCS AVAILABLE
1.	1st row vent cleared	.947	.947
2.	2nd row vent cleared	1.169	1.169
3.	3rd row vent cleared	1.576	1.576
4.	Drywell reaches peak pressure	1.33	1.33
5.	Maximum Positive Differential Pressure occurs	1.33	1.33
6.	3rd row vent recovered	29	29
7.	Initiation of the ECCS	30	30
8	2nd row vent recovered	42	42
9.	1st row vent recovered	293	617
10.	End of blowdown	77	397
11.	Vessel reflooded	249	719
12.	Initiation of RHR heat exchanger	1980	1980
13.	Containment reaches peak pressure	6515	36953

TABLE 6.2-8a
ACCIDENT CHRONOLOGY FOR
MAIN STEAMLINE BREAK ACCIDENT
(at 3543 MWt)

	EVENT	CASE B2 MIN ECCS AVAILABLE
1.	1st row vent cleared	0.978
2.	2nd row vent cleared	1.267
3.	3rd row vent cleared	1.688
4.	Drywell reaches peak pressure	2.052
5.	Maximum positive differential pressure occurs	2.052

TABLE 6.2-9
AVAILABLE CONTAINMENT HEAT SINKS

	ITEM	VOLUME (ft^3)	SURFACE AREA (ft^3)	MATERIAL
A.	Drywell Structures	106068	22647	Concrete
B.	Containment Shell	1566	75215	Steel
C.	Misc. Steel Structures and Equipment	2589	93859	Steel
D.	Misc. Concrete Structures	159820	39130	Concrete

Note: The above Table represents the values used for containment heat sinks in the containment analysis. For drywell heat sinks, additional detail on the heat sinks and a comparison of the actual to analytical values, see Reference 26.

TABLE 6.2-10
MASS AND ENERGY RELEASE DATA*
FOR RECIRCULATION OUTLET LINE BREAK FOR HEAD
CAVITY DIFFERENTIAL PRESSURE TRANSIENT

Time (sec)	Mass Release Rate (lb_m/sec)		Enthalpy (Btu/lb_m)		Energy Release Rate (Btu/sec)	
	Liquid	Steam	Liquid	Steam	Liquid	Steam
0	21,070	0	551.2	-	11,613,784	-
1.8	21,040	0	550.7	-	11,586,728	-
1.9	17,790	0	550.7	-	9,796,953	-
5.0	18,060	0	557.0	-	10,059,420	-
7.5	18,230	0	560.5	-	10,217,915	-
10.0	18,300	0	562.0	-	10,284,600	-
15.0	18,270	0	561.5	-	10,258,605	-
18.9	18,190	0	559.8	-	10,182,762	3,732,204
19.0	7,720	3,140	559.4	1,188.6	4,318,568	3,049,892
25.0	4,757	2,542	507.1	1,199.8	2,412,275	2,384,530
30.0	3,122	1,981	463.0	1,203.7	1,445,486	978,955
40.0	1,593	816	365.0	1,199.7	581,445	121,056
50.0	1,438	103	254.3	1,175.3	365,683	172,612
54.0	0	148	-	1,166.3	-	-
57.0	0	0	-	-	-	-

* Mass and energy flowrates based on methodology presented in Reference 3.

TABLE 6.2-11
MASS AND ENERGY RELEASE DATA*
FOR HEAD SPRAY LINE BREAK IN HEAD CAVITY

Time (sec)	Mass Release Rate (lb_m/sec)		Enthalpy (Btu/lb_m)		Energy Release Rate (Btu/sec)	
	Liquid	Steam	Liquid	Steam	Liquid	Steam
Duration of Analysis	0	166.58	-	1190.0	-	198,226.9

* Mass and energy flowrates based on methodology presented in Reference 3.

TABLE 6.2-12
SUBCOMPARTMENT NODAL DESCRIPTION
OF HEAD CAVITY - HEAD SPRAY LINE BREAK

VOL. NO.	DESCRIPTION	HEIGHT ft.	SECTIONAL AREA ft^2	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.	DESIGN MARGIN %
1	Head Cavity	15.12	325.4	135	15.45	20	1	Spray Line	0.07548	Double-ended guillo-tine	8.21	30.0	265.4
2	Drywell	80.08	3163.6	135	15.45	20	-	-	-	-	-	-	
3	Containment	26.31	5591	90	20.02	100	-	-	-	-	-	-	

TABLE 6.2-13
SUBCOMPARTMENT NODAL DESCRIPTION
OF HEAD CAVITY - RECIRCULATION LINE BREAK

VOLUME NO.	DESCRIPTION	HEIGHT ft.	CROSS-SECTIONAL AREA ft^2	INITIAL CONDITIONS TEMP, °F	PRESS, psia	HUMID, %	DBA BREAK CONDITIONS BREAK LOC. VOL. NO.	BREAK LINE	BREAK AREA ft^2	BREAK TYPE	CALCU-LATED PEAK PRESS. DIFF. psid	DESIGN PEAK PRESS. DIFF, psid	MARGIN %
1	Head Cavity	15.12	443.5	135	15.45	20	-	-	-	-	-	-	
2	Drywell	80.09	3140.0	135	15.45	20	2	Recirc.	-	Double-ended guillot ine rupture	5.49	30.0	446.4
3	Containment	26.31	5512.0	90	20.19	100	-	-	-	-	-	-	

TABLE 6.2-14
SUBCOMPARTMENT NODAL DESCRIPTION
OF BIOLOGICAL SHIELD ANNULUS - RECIRCULATION OUTLET LINE BREAK WITH DIVERTER

					INITIAL CONDITIONS			DBA BREAK CONDITIONS				
VOLUME NO.	DESCRIPTION	VOLUME ft^3	HEIGHT ft.	CROSS-SECTIONAL AREA ft^2	TEMP, F	PRESS, psia	HUMID, %	BREAK LOC VOL. NO.	BREAK LINE	BREAK AREA ft^2	BREAK TYPE	CALC PEAK PRESS. DIFF. psig
1	Reactor Skirt Sect.	179.79	9.65	29.90	528.	14.20	0.1					3.5
2	Reactor Skirt Sect.	179.79	9.65	29.90	528.	14.20	0.1					3.9
3	Reactor Skirt Sect.	179.79	9.65	29.90	528.	14.20	0.1					3.1
4	Reactor Skirt Sect.	269.69	9.65	29.90	528.	14.20	0.1					3.1
5	Reactor Skirt Sect.	269.69	9.65	29.90	528.	14.20	0.1					2.7
6	Lower Recirc. Nozzle Sect.	80.53	6.00	15.95	528.	14.20	0.1					6.1
7	Lower Recirc. Nozzle Sect.	92.06	6.00	15.95	528.	14.20	0.1					6.0
8	Lower Recirc. Nozzle Sect.	90.72	6.00	15.95	528.	14.20	0.1					5.5
9	Lower Recirc. Nozzle Sect.	138.08	6.00	19.90	528.	14.20	0.1					3.2
10	Lower Recirc. Nozzle Sect.	126.25	6.00	19.90	528.	14.20	0.1					2.7
11	Upper Recirc. Nozzle Sect.	80.02	5.15	15.00	528.	14.20	0.1					5.9
12	Upper Recirc. Nozzle Sect.	81.00	5.15	15.00	528.	14.20	0.1					5.2
13	Upper Recirc. Nozzle Sect.	81.00	5.15	15.00	528.	14.20	0.1					3.5
14	Upper Recirc. Nozzle Sect.	121.47	5.15	20.75	528.	14.20	0.1					2.9
15	Upper Recirc. Nozzle Sect.	120.52	5.15	20.75	528.	14.20	0.1					2.0
16	Mid-Section	159.52	10.00	27.60	528.	14.20	0.1					4.0
17	Mid-Section	159.52	10.00	27.60	528.	14.20	0.1					3.4
18	Mid-Section	159.52	10.00	27.60	528.	14.20	0.1					2.8
19	Mid-Section	239.27	10.00	27.60	528.	14.20	0.1					3.0
20	Mid-Section	239.27	10.00	27.60	528.	14.20	0.1					2.0
21	LPCI Nozzle Sect.	171.60	10.85	29.95	528.	14.20	0.1					2.9
22	LPCI Nozzle Sect.	165.44	10.85	29.95	528.	14.20	0.1					2.0
23	LPCI Nozzle Sect.	169.75	10.85	29.95	528.	14.20	0.1					2.6
24	LPCI Nozzle Sect.	252.37	10.85	29.95	528.	14.20	0.1					1.8
25	LPCI Nozzle Sect.	254.95	10.85	29.95	528.	14.20	0.1					2.3
26	Feedwater Nozzle Sect.	153.07	9.75	26.30	528.	14.20	0.1					2.6
27	Feedwater Nozzle Sect.	150.76	9.75	26.30	528.	14.20	0.1					1.7
28	Feedwater Nozzle Sect.	153.41	9.75	26.30	528.	14.20	0.1					1.7
29	Feedwater Nozzle Sect.	228.35	9.75	26.30	528.	14.20	0.1					1.7
30	Feedwater Nozzle Sect.	228.72	9.75	26.30	528.	14.20	0.1					1.5
31	Main Steam Nozzle Sect.	292.70	8.05	46.50	528.	14.20	0.1					2.0
32	Main Steam Nozzle Sect.	292.70	8.05	46.50	528.	14.20	0.1					2.0
33	Flow Diverter	9.05	4.00	4.10	528.	14.20	0.1	33	RR		Double-ended guillotine rupture	635.
34	Drywell	121000.	80.25	1520.	150.	14.20	0.1					---

TABLE 6.2-15
SUBCOMPARTMENT NODAL DESCRIPTION
OF CONTAINMENT PIPE TUNNEL - RWCU LINE BREAK
NODE PARAMETERS

			INITIAL CONDITIONS		
NODE	DESCRIPTION	VOLUME (ft^3)	TEMPERATURE (°F)	PRESSURE (psia)	HUMIDITY (%)
1	MST Inside Containment	21855	142	14.695	90
2	Containment	1528945	104	14.695	90

Notes:

For initial conditions, maximum temperature, pressure, and relative humidity are assumed. These assumptions maximize long-term temperature and pressure response.

TABLE 6.2-16
MASS AND ENERGY RELEASE DATA
FOR RWCU LINE BREAK IN CONTAINMENT PIPE TUNNEL

Time (sec)	Forward Flow: Mass Flow Rate (lb_m/sec)	Forward Flow: Enthalpy (Btu/lb_m)	Forward Flow: Energy Flow Rate (Btu/sec)	Reverse Flow: Mass Flow Rate (lb_m/sec)	Reverse Flow: Enthalpy (Btu/lb_m)	Reverse Flow: Energy Flow Rate (Btu/sec)
0.00	374.	525.3	196,462	325.	303.5	98,638
0.17	374.	525.3	196,462	325.	303.5	98,638
0.1701	190.	525.3	99,807	325.	303.5	98,638
13.14	190.	525.3	99,807	325.	303.5	98,638
13.64	190.	525.3	99,807	0.	0.	0.
84.0	190.	525.3	99,807	0.	0.	0.
84.5	0.	0.	0.	0.	0.	0.
1.E7	0.	0.	0.	0.	0.	0.

Note:

The mass and energy releases presented above are based on LPU conditions at normal feedwater temperatures (NFWT). The subcompartment analyses were also completed for the case of reduced feedwater temperature (RFWT) at LPU conditions. The resulting RWCU mass releases are 5% higher than those provided above. The energy releases are 3% higher. The RWCU subcompartment analyses were completed at both NFWT and RFWT with the bounding results provided.

TABLE 6.2-17
MASS AND ENERGY RELEASE DATA FOR RWCU LINE BREAK
IN RWCU VALVE AND HEAT EXCHANGER ROOMS

	Forward Flow			Reverse Flow		
Time (sec)	Mass Flow Rate (lb_m/sec)	Enthalpy (Btu/lb_m)	Energy Flow Rate (Btu/sec)	Mass Flow Rate (lb_m/sec)	Enthalpy (Btu/lb_m)	Energy Flow Rate (Btu/sec)
0.00	374.	525.3	196,462	325.	328.3	106,698
1.04	374.	525.3	196,462	325.	328.3	106,698
1.0401	190.	525.3	99,807	325.	328.3	106,698
12.4	190.	525.3	99,807	325.	328.3	106,698
12.45	190.	525.3	99,807	0.	0.	0.
84.0	190.	525.3	99,807	0.	0.	0.
84.5	0.	0.	0.	0.	0.	0.
1.E7	0.	0.	0.	0.	0.	0.

Note:

The mass and energy releases presented above are based on LPU conditions at normal feedwater temperatures (NFWT). The subcompartment analyses were also completed for the case of reduced feedwater temperature (RFWT) at LPU conditions. The resulting RWCU mass releases are 5% higher than those provided above. The energy releases are 3% higher. The RWCU subcompartment analyses were completed at both NFWT and RFWT with the bounding results provided.

TABLE 6.2-18
MASS AND ENERGY RELEASE DATA FOR RWCU LINE BREAK IN FILTER-DEMINERALIZER FILTER-DEMINERALIZER HOLDING PUMP, FILTER-DEMINERALIZER VALVE ROOMS AND RWCU CROSSOVER PIPE TUNNEL

	Forward Flow			Reverse Flow		
Time (sec)	Mass Flow Rate (lb_m/sec)	Enthalpy (Btu/lb_m)	Energy Flow Rate (Btu/sec)	Mass Flow Rate (lb_m/sec)	Enthalpy (Btu/lb_m)	Energy Flow Rate (Btu/sec)
0.00	374.	525.3	196,462	325.	324.9	105,593
4.52	374.	525.3	196,462	325.	324.9	105,593
4.5201	190.	525.3	99,807	325.	324.9	105,593
8.2	190.	525.3	99,807	325.	324.9	105,593
8.25	190.	525.3	99,807	0.	0.	0.
84.0	190.	525.3	99,807	0.	0.	0.
84.5	0.	0.	0.	0.	0.	0.
1.E7	0.	0.	0.	0.	0.	0.

Note:

The mass and energy releases presented above are based on LPU conditions at normal feedwater temperatures (NFWT). The subcompartment analyses were also completed for the case of reduced feedwater temperature (RFWT) at LPU conditions. The resulting RWCU mass releases are 5% higher than those provided above. The energy releases are 3% higher. The RWCU subcompartment analyses were completed at both NFWT and RFWT with the bounding results provided.

TABLE 6.2-19

SUBCOMPARTMENT VENT PATH DESCRIPTION
OF HEAD CAVITY - HEAD SPRAY LINE BREAK

VENT PATH NO.	FROM VOL. NODE NO.	TO VOL. NODE NO.	DESCRIPTION OF VENT PATH FLOW		AREA ft^2	LENGTH ft	HYDRAULIC DIAMETER ft	HEAD LOSS, K			
			CHOKED	UNCHOKED				FRICTION K, ft/d	TURNING LOSS, K	EXPAN- SION, K	CONTRAC- TION, K
								FORWARD		REVERSE	
1	1	2	choked at 5.3 sec for ~ 0.1 sec		6.55	1.75	2.89	5.67		3.66	
2	2	3	-	√	251.3	41.54	17.88	2.5		2.5	

TABLE 6.2-20
SUBCOMPARTMENT VENT PATH DESCRIPTION
OF HEAD CAVITY - RECIRCULATION LINE BREAK

VENT PATH NO.	FROM VOL. NODE NO.	TO VOL. NODE NO.	DESCRIPTION OF VENT PATH FLOW: CHOKED	DESCRIPTION OF VENT PATH FLOW: UNCHOKED	AREA ft^2	LENGTH ft	HYDRAULIC DIAMETER ft	HEAD LOSS, K (FRICTION K, ft/d; TURNING LOSS, K; EXPAN-SION, K; CONTRAC-TION, K): FORWARD	HEAD LOSS, K (FRICTION K, ft/d; TURNING LOSS, K; EXPAN-SION, K; CONTRAC-TION, K): REVERSE
1	2	1	-	√	6.55	1.75	2.89	3.66	5.67
2	2	3	-	√	251.3	41.54	17.88	2.5	2.5

TABLE 6.2-21
SUBCOMPARTMENT VENT PATH DESCRIPTION
OF BIOLOGICAL SHIELD ANNULUS - RECIRCULATION OUTLET LINE BREAK WITH DIVERTER

VENT PATH NO.	FROM VOL. NODE NO.	TO VOL. NODE NO.	DESCRIPTION OF VENT PATHFLOW	AREA* (ft^2)	LENGTH (ft)	$\sum \frac{L}{A}$ (ft^{-1})	HYDRAULIC DIAMETER (ft)	HEAD LOSS, K: FRICTION LOSS, K_f	HEAD LOSS, K: TURNING LOSS, K_{bl}	HEAD LOSS, K: EXPANSION & CONTRACTION, K_E	HEAD LOSS, K: TOTAL
1	1	2	Unchoked	29.90	5.90	0.20	4.85	0.08	0.00	0.00	0.08
2	2	3	Unchoked	29.90	5.90	0.20	4.85	0.08	0.00	0.00	0.08
3	3	4	Unchoked	29.90	7.40	0.25	4.85	0.09	0.00	0.00	0.09
4	4	5	Unchoked	29.90	8.90	0.30	6.50	0.11	0.00	0.00	0.11
5	6	1	Unchoked	11.75	7.83	0.65	6.50	0.02	0.00	0.43	0.45
6	7	2	Unchoked	11.75	7.83	0.65	6.50	0.02	0.00	0.43	0.45
7	8	3	Unchoked	11.75	7.83	0.65	6.50	0.02	0.00	0.43	0.45
8	9	4	Unchoked	17.60	7.83	0.45	6.50	0.02	0.00	0.43	0.45
9	10	5	Unchoked	17.60	7.83	0.45	6.50	0.02	0.00	0.43	0.45
10	33	6	Choked	0.17	3.50	3.30	0.15	0.05	0.00	1.85	1.90
11	6	7	Unchoked	14.35	5.80	0.40	5.50	0.09	0.02	0.00	0.11
12	7	8	Unchoked	14.35	5.80	0.40	5.50	0.09	0.02	0.00	0.11
13	8	9	Unchoked	14.35	7.25	0.50	5.50	0.11	0.02	0.00	0.13
14	9	10	Unchoked	14.35	8.70	0.60	5.50	0.13	0.03	0.00	0.16
15	6	11	Unchoked	8.85	5.58	0.65	5.50	0.01	0.19	0.00	0.20
16	7	12	Unchoked	11.90	5.58	0.45	5.50	0.01	0.06	0.00	0.07
17	8	13	Unchoked	11.90	5.58	0.45	5.50	0.01	0.06	0.00	0.07
18	9	14	Unchoked	17.85	5.58	0.32	5.50	0.01	0.06	0.00	0.07
19	10	15	Unchoked	14.80	5.58	0.38	5.50	0.01	0.13	0.00	0.14
20	33	11	Choked	0.16	3.50	3.50	0.15	0.05	0.00	1.85	1.90
21	11	12	Unchoked	15.00	5.80	0.40	5.50	0.09	0.02	0.00	0.11
22	12	13	Unchoked	15.00	5.80	0.40	5.50	0.09	0.02	0.00	0.11
23	13	14	Unchoked	15.00	7.25	0.50	5.50	0.11	0.02	0.00	0.13
24	14	15	Unchoked	15.00	8.70	0.60	5.50	0.13	0.03	0.00	0.16
25	11	16	Unchoked	13.85	7.58	0.55	5.50	0.02	0.00	0.13	0.15

TABLE 6.2-21 (CONT'D)

VENT PATH NO.	FROM VOL. NODE NO.	TO VOL. NODE NO.	DESCRIPTION OF VENT PATH FLOW	AREA* (ft^2)	LENGTH (ft)	$\sum \frac{L}{A}$ (ft^{-1})	HYDRAULIC DIAMETER (ft)	HEAD LOSS, K: FRICTION LOSS, K_f	TURNING LOSS, K_{bl}	EXPANSION & CONTRACTION, K_E	TOTAL
27	13	18	Unchoked	13.85	7.58	0.55	5.50	0.02	0.00	0.13	0.15
28	14	19	Unchoked	20.75	7.58	0.37	5.50	0.02	0.00	0.13	0.15
29	15	20	Unchoked	20.75	7.58	0.37	5.50	0.02	0.00	0.13	0.15
30	16	17	Unchoked	27.60	5.80	0.21	5.50	0.12	0.00	0.00	0.12
31	17	18	Unchoked	27.60	5.80	0.21	5.50	0.12	0.00	0.00	0.12
32	18	19	Unchoked	27.60	7.25	0.27	5.50	0.14	0.00	0.00	0.14
33	19	20	Unchoked	27.60	8.70	0.32	5.50	0.17	0.00	0.00	0.17
34	16	21	Unchoked	14.10	10.43	0.74	5.50	0.03	0.00	0.13	0.16
35	17	22	Unchoked	12.10	10.43	0.86	5.50	0.03	0.06	0.09	0.18
36	18	23	Unchoked	14.10	10.43	0.74	5.50	0.03	0.00	0.13	0.16
37	19	24	Unchoked	21.15	10.43	0.50	5.50	0.03	0.01	0.12	0.16
38	20	25	Unchoked	21.15	10.43	0.50	5.50	0.03	0.01	0.12	0.16
39	21	22	Unchoked	23.60	5.80	0.25	5.50	0.08	0.04	0.00	0.12
40	22	23	Unchoked	29.60	5.80	0.20	5.50	0.12	0.00	0.00	0.12
41	23	24	Unchoked	25.30	7.25	0.30	5.50	0.11	0.02	0.00	0.13
42	24	25	Unchoked	23.60	8.70	0.37	5.50	0.11	0.04	0.00	0.15
43	21	26	Unchoked	14.97	10.30	0.70	5.50	0.03	0.01	0.00	0.04
44	22	27	Unchoked	11.64	10.30	0.90	5.50	0.03	0.06	0.00	0.09
45	23	28	Unchoked	13.80	10.30	0.75	5.50	0.03	0.02	0.00	0.05
46	24	29	Unchoked	21.77	10.30	0.48	5.50	0.03	0.02	0.00	0.05
47	25	30	Unchoked	21.77	10.30	0.48	5.50	0.03	0.01	0.00	0.04
48	26	27	Unchoked	24.15	5.80	0.24	5.50	0.10	0.01	0.00	0.11
49	27	28	Unchoked	26.30	5.80	0.22	5.50	0.12	0.00	0.00	0.12
50	28	29	Unchoked	26.30	7.25	0.28	5.50	0.14	0.00	0.00	0.14
51	29	30	Unchoked	24.15	8.70	0.36	5.50	0.14	0.01	0.00	0.15
52	26	31	Unchoked	11.55	8.05	0.70	5.50	0.02	0.83	2.00	2.85
53	27	31	Unchoked	11.55	8.05	0.70	5.50	0.02	0.83	2.00	2.85
54	28	31	Unchoked	11.55	8.05	0.70	5.50	0.02	0.83	2.00	2.85
55	29	32	Unchoked	17.33	8.05	0.45	5.50	0.02	0.83	2.00	2.85

TABLE 6.2-21 (CONT'D)

VENT PATH NO.	FROM VOL. NODE NO.	TO VOL. NODE NO.	DESCRIPTION OF VENT PATH FLOW	AREA* (ft^2)	LENGTH (ft)	$\sum \frac{L}{A}$ (ft^{-1})	HYDRAULIC DIAMETER (ft)	HEAD LOSS, K: FRICTION LOSS, K_f	HEAD LOSS, K: TURNING LOSS, K_{bl}	HEAD LOSS, K: EXPANSION & CONTRACTION, K_E	HEAD LOSS, K: TOTAL
56	30	32	Unchoked	17.33	8.05	0.45	5.50	0.02	0.83	2.00	2.85
57	31	32	Unchoked	9.75	17.40	1.75	5.50	0.07	0.17	0.00	0.24
58	31	34	Unchocked	46.50	0.00	0.01	5.50	0.00	0.00	0.00	0.00
59	32	34	Unchoked	46.50	0.00	0.01	5.50	0.00	0.00	0.00	0.00
60	33	34	Choked	1.88	4.75	2.10	0.90	0.00	0.00	1.35	1.35
61	0	33	Choked	1.00	0.00	0.00	----	----	----	----	0.00

* Minimum cross-sectional area

TABLE 6.2-22
SUBCOMPARTMENT NODAL DESCRIPTION
OF BIOLOGICAL SHIELD ANNULUS - FEEDWATER LINE BREAK

					INITIAL CONDITIONS			DBA BREAK CONDITIONS				
VOLUME NO.	DESCRIPTION	VOLUME ft^3	HEIGHT ft.	CROSS-SECTIONAL AREA ft^2	TEMP, °F	PRESS, psia	HUMID, %	BREAK LOC. VOL. NO.	BREAK LINE	BREAK AREA ft^2	BREAK TYPE	CALC. PEAK PRESS. DIFF. psig
1	Reactor Skirt Sect.	653.85	15.12	43.23	450.	15.0	0.24					5.4
2	Reactor Skirt Sect.	655.05	15.12	43.31	450.	15.0	0.24					4.5
3	Reactor Skirt Sect.	653.85	15.12	43.23	450.	15.0	0.24					5.0
4	Reactor Skirt Sect.	655.05	15.12	43.31	450.	15.0	0.24					11.4
5	Recirc. Nozzle Sect.	507.65	11.71	43.36	450.	15.0	0.24					8.0
6	Recirc. Nozzle Sect.	507.65	11.71	43.36	450.	15.0	0.24					7.0
7	Recirc. Nozzle Sect.	507.65	11.71	43.36	450.	15.0	0.24					7.2
8	Recirc. Nozzle Sect.	507.65	11.71	43.36	450.	15.0	0.24					9.5
9	Mid-Section	362.68	8.27	43.85	450.	15.0	0.24					5.4
10	Mid-Section	141.02	8.27	17.05	450.	15.0	0.24					8.1
11	Mid-Section	78.65	8.27	9.51	450.	15.0	0.24					5.5
12	Mid-Section	141.02	8.27	17.05	450.	15.0	0.24					8.2
13	Mid-Section	360.78	8.27	43.62	450.	15.0	0.24					4.5
14	Mid-Section	360.78	8.27	43.62	450.	15.0	0.24					7.8
15	Feedwater Nozzle Sect.	167.10	7.73	21.62	450.	15.0	0.24					4.4
16	Feedwater Nozzle Sect.	164.71	7.73	21.31	450.	15.0	0.24					3.7
17	Lower Break Boundary	75.20	4.48	16.79	450.	15.0	0.24					8.6
18	Lower Break Boundary	41.74	4.48	9.32	450.	15.0	0.24					20.6
19	Lower Break Boundary	76.18	4.48	17.01	450.	15.0	0.24					13.4
20	Feedwater Nozzle Sect.	169.80	7.73	21.97	450.	15.0	0.24					4.0
21	Feedwater Nozzle Sect.	163.70	7.73	21.18	450.	15.0	0.24					5.3
22	Feedwater Nozzle Sect.	329.80	7.73	42.67	450.	15.0	0.24					6.5

TABLE 6.2-22 (CONT'D)

VOLUME NO.	DESCRIPTION	VOLUME ft^3	HEIGHT ft.	CROSS-SECTIONAL AREA ft^2	INITIAL CONDITIONS TEMP, °F	PRESS, psia	HUMID, %	DBA BREAK CONDITIONS BREAK LOC. VOL. NO.	BREAK LINE	BREAK AREA ft^2	BREAK TYPE	CALC. PEAK PRESS. DIFF. psig
23	Lateral Break Boundary	54.21	3.25	16.68	450.	15.0	0.24					9.0
24	Break Node	26.88	3.25	8.27	450.	15.0	0.24	24	FW		Double-Ended Guillotine Rupture	127.1
25	Lateral Break Boundary	55.41	3.25	17.05	450.	15.0	0.24					13.3
26	Stabilizer Section	176.32	8.04	21.93	450.	15.0	0.24					7.0
27	Stabilizer Section	176.32	8.04	21.93	450.	15.0	0.24					8.0
28	Upper Break Boundary	571.24	8.04	71.05	450.	15.0	0.24					7.1
29	Upper Break Boundary	78.39	8.04	71.05	450.	15.0	0.24					8.3
30	Upper Break Boundary	571.24	8.04	71.05	450.	15.0	0.24					5.5
31	Stabilizer Section	176.32	8.04	21.93	450.	15.0	0.24					9.4
32	Stabilizer Section	176.32	8.04	21.93	450.	15.0	0.24					9.6
33	Stabilizer Section	352.55	8.04	43.85	450.	15.0	0.24					3.6
34	Drywell	213125.00	85.25	2500.00	120.	15.0	20.00					---

TABLE 6.2-23
SUBCOMPARTMENT VENT PATH DESCRIPTION
FEEDWATER LINE BREAK

VENT PATH NO.	FROM VOL. NODE NO.	TO VOL. NODE NO.	DESCRIPTION OF VENT PATH FLOW	AREA* (ft^2)	LENGTH (ft)	$\sum \frac{L}{A}$ (ft^{-1})	HYDRAULIC DIAMETER (ft)	HEAD LOSS, K: FRICTION LOSS, K_f	HEAD LOSS, K: TURNING LOSS, K_{bl}	HEAD LOSS, K: EXPANSION & CONTRACTION, K_E	HEAD LOSS, K: TOTAL
1	4	1	unchoked	34.063	17.541	0.515	5.00	0.216	0.062	0.033	0.311
2	5	1	unchoked	33.383	13.417	0.402	5.00	0.255	0.0	0.241	0.496
3	2	1	unchoked	35.833	17.541	0.490	5.00	0.263	0.058	0.129	0.452
4	6	2	unchoked	33.383	13.417	0.402	5.00	0.264	0.0	0.15	0.405
5	2	3	unchoked	35.063	17.541	0.490	5.00	0.216	0.062	0.033	0.311
6	7	3	unchoked	33.383	13.417	0.402	5.00	0.255	0.0	0.241	0.496
7	3	4	unchoked	35.833	17.541	0.490	5.00	0.265	0.058	0.129	0.452
8	8	4	unchoked	33.383	13.417	0.402	5.00	0.264	0.0	0.15	0.405
9	8	5	unchoked	27.525	17.541	0.637	5.00	0.184	0.056	0.027	0.267
10	9	5	unchoked	43.851	9.990	0.228	5.00	0.114	0.0	0.0	0.114
11	6	5	unchoked	27.525	17.541	0.637	5.00	0.185	0.057	0.030	0.271
12	10	6	unchoked	17.053	9.990	0.586	5.00	0.114	0.0	0.0	0.114
13	6	7	unchoked	27.525	17.541	0.637	5.00	0.184	0.056	0.027	0.267
14	11	6	unchoked	9.745	9.990	1.025	5.00	0.114	0.0	0.0	0.114
15	7	8	unchoked	27.525	17.541	0.637	5.00	0.185	0.057	0.030	0.271
16	12	6	unchoked	17.053	9.990	0.586	5.00	0.114	0.0	0.0	0.114
17	14	9	unchoked	19.805	17.541	0.886	5.00	0.242	0.051	0.008	0.301
18	13	7	unchoked	43.851	9.990	0.228	5.00	0.114	0.0	0.0	0.114
19	10	9	unchoked	20.678	12.181	0.589	5.00	0.119	0.045	0.0	0.164
20	14	8	unchoked	43.851	9.990	0.228	5.00	0.114	0.0	0.0	0.114
21	11	10	unchoked	19.805	5.847	0.295	5.00	0.027	0.023	0.008	0.058
22	15	9	unchoked	21.926	8.000	0.365	5.00	0.092	0.0	0.0	0.092
23	11	12	unchoked	19.805	5.847	0.295	5.00	0.027	0.023	0.008	0.058
24	16	9	choked	21.926	8.000	0.365	5.00	0.092	0.0	0.0	0.092
25	12	13	unchoked	20.678	12.181	0.589	5.00	0.127	0.044	0.0	0.171
26	17	10	choked	17.053	6.375	0.374	5.00	0.073	0.0	0.0	0.073
27	13	14	unchoked	18.932	17.541	0.927	5.00	0.236	0.045	0.016	0.297
28	18	11	choked	6.255	6.375	1.019	5.00	0.146	0.0	1.411	1.557
29	22	15	unchoked	16.500	13.155	13.155	5.00	0.136	0.030	0.232	0.398
30	19	12	choked	17.053	6.375	6.375	5.00	0.073	0.0	0.0	0.073
31	16	15	unchoked	15.416	8.770	8.770	5.00	0.050	0.020	0.446	0.516
32	20	13	choked	20.181	8.000	8.000	5.00	0.116	0.0	0.035	0.151
33	17	16	choked	10.416	7.796	7.796	5.00	0.07	0.031	0.015	0.132
34	21	13	unchoked	20.181	8.000	8.000	5.00	0.12	0.0	0.022	0.142
35	18	17	choked	10.850	5.360	5.360	5.00	0.023	0.029	0.007	0.059

TABLE 6.2-23 (CONT'D)

VENT PATH NO.	FROM VOL. NODE NO.	TO VOL. NODE NO.	DESCRIPTION OF VENT PATH FLOW	AREA* (ft^2)	LENGTH (ft)	$\sum \frac{L}{A}$ (ft^{-1})	HYDRAULIC DIAMETER (ft)	HEAD LOSS, K: FRICTION LOSS, K_f	HEAD LOSS, K: TURNING LOSS, K_{bl}	HEAD LOSS, K: EXPANSION & CONTRACTION, K_E	HEAD LOSS, K: TOTAL
36	22	14	unchoked	40.362	8.000	8.000	5.00	0.106	0.0	0.12	0.226
37	18	19	unchoked	10.850	5.360	5.360	5.00	0.023	0.029	0.007	0.059
38	23	17	unchoked	15.100	3.865	3.865	5.00	0.053	0.0	0.03	0.083
39	19	20	choked	9.430	7.795	7.795	5.00	0.068	0.024	0.302	0.394
40	24	18	choked	5.840	3.865	3.865	5.00	0.083	0.0	0.943	1.026
41	20	21	unchoked	13.671	8.770	8.770	5.00	0.044	0.016	0.464	0.524
42	25	19	choked	21.040	3.865	3.865	5.00	0.041	0.0	0.014	0.055
43	21	22	unchoked	16.500	13.155	13.155	5.00	0.188	0.028	0.678	0.894
44	15	26	unchoked	19.973	7.885	7.885	5.00	0.104	0.0	0.02	0.124
45	23	16	choked	6.172	5.847	5.847	5.00	0.021	0.027	0.101	0.149
46	16	27	unchoked	19.973	7.885	7.885	5.00	0.104	0.0	0.04	0.144
47	24	23	choked	6.172	5.360	5.360	5.00	0.018	0.025	1.667	1.710
48	23	28	choked	15.100	5.645	5.645	5.00	0.09	0.0	0.0	0.090
49	24	25	choked	8.125	5.360	5.360	5.00	0.03	0.044	0.0	0.074
50	24	29	choked	5.840	4.645	5.645	5.00	0.169	0.0	0.943	1.026
51	25	20	choked	8.125	7.796	7.796	5.00	0.049	0.065	0.0	0.074
52	25	30	choked	17.050	5.645	0.331	5.00	0.09	0.0	0.0	0.090
53	33	26	unchoked	20.100	13.155	0.654	5.00	0.139	0.045	0.0	0.184
54	20	31	choked	19.970	7.885	0.395	5.00	0.104	0.0	0.067	0.171
55	27	26	unchoked	12.600	8.770	0.696	5.00	0.024	0.013	0.155	0.192
56	21	32	unchoked	19.973	7.885	0.395	5.00	0.11	0.0	0.02	0.130
57	28	27	unchoked	12.600	5.847	0.545	5.00	0.01	0.01	0.395	0.415
58	22	33	unchoked	39.945	7.885	0.197	5.00	0.109	0.0	0.139	0.248
59	29	28	choked	12.600	5.360	0.425	5.00	0.010	0.008	0.215	0.233
60	26	34	unchoked	43.851	4.020	0.092	5.00	0.046	0.0	1.0	1.046
61	29	30	choked	20.100	5.360	0.267	5.00	0.023	0.22	0.0	0.045
62	27	34	unchoked	10.626	4.020	0.378	5.00	0.046	0.0	1.315	1.361
63	30	31	unchoked	20.100	7.796	0.388	5.00	0.049	0.022	0.0	0.071
64	28	34	unchoked	5.750	4.020	0.699	5.00	0.046	0.0	1.302	1.348
65	31	32	choked	12.600	7.796	0.619	5.00	0.020	0.008	0.204	0.232
66	29	34	choked	9.745	4.020	0.413	5.00	0.046	0.0	1.0	1.046
67	32	33	unchoked	12.600	13.155	1.044	5.00	0.054	0.018	0.606	0.678
68	30	34	unchoked	17.053	4.020	0.236	5.00	0.057	0.0	1.0	1.057
69	16	34	unchoked	10.000	2.000	0.200	3.05	0.0	0.0	2.85	2.850
70	31	34	choked	21.926	4.020	0.183	5.00	0.057	0.0	1.0	1.057
71	20	34	unchoked	10.000	2.000	0.200	3.05	0.0	0.0	2.85	2.850
72	32	34	unchoked	10.626	4.020	0.378	5.00	0.073	0.0	1.231	1.304
73	22	34	unchoked	10.000	2.000	0.200	3.05	0.0	0.0	2.85	2.850
74	33	34	unchoked	32.551	4.020	0.123	5.00	0.058	0.0	1.279	1.334
75	24	34	choked	2.990	2.000	0.669	0.95	0.0	0.0	2.85	2.850

* Minimum cross-sectional area

TABLE 6.2-24
SUBCOMPARTMENT VENT PATH DESCRIPTION
OF CONTAINMENT PIPE TUNNEL - RWCU LINE BREAK

Path	Description	Flow Area (ft^{-2})	$\sum l/A$ (ft^{-1})	K
1	From MST to Containment	46	3.08	2.90

TABLE 6.2-25
SUBCOMPARTMENT NODAL DESCRIPTION
FOR BREAK IN RWCU HEAT EXCHANGER ROOM A (CASE 2)

				INITIAL CONDITIONS			DBA BREAK CONDITIONS						
VOLUME NO.	DESCRIPTION	HEIGHT ft.	CROSS-SECTIONAL AREA ft^2	TEMP, °F	PRESS. psia	HUMID, %	BREAK LOC. VOL. NO.	BREAK LINE	BREAK AREA ft^2	BREAK TYPE	CALC. PEAK PRESS. DIFF. psig	DESIGN PEAK PRESS. DIFF, psig	DESIGN MARGIN %
1	RWCU Heat Exchanger Room A	35.2	198.5	122.0	14.7	~0	1	RWCU	-	Double-ended guillotine rupture	3.1	-	-
2	RWCU Heat Exchanger Valve Room A	9.2	139.3	104.0	14.7	~0	-	-	-	-	-	-	-
3	Containment	125.0	12000.0	104.0	14.7	~0	-	-	-	-	-	-	-
4	RWCU Crossover Pipe Tunnel	33.4	28.2	122.0	14.7	~0	-	-	-	-	-	-	-

TABLE 6.2-26
SUBCOMPARTMENT VENT PATH DESCRIPTION
FOR BREAK IN RWCU HEAT EXCHANGER ROOM A (CASE 2)

VENT PATH NO.	FROM VOL. NODE NO.	TO VOL. NODE NO.	DESCRIPTION OF VENT PATH FLOW: CHOKED	UNCHOKED	AREA ft^2	LENGTH ft / HYDRAULIC DIAMETER ft: INERTIA (ft^{-1})	HEAD LOSS, K: FRICTION K, ft/d	TURNING LOSS, K	EXPANSION K	CONTRACTION K	TOTAL
1	1	2	-	X	22.9	1.0					1.8
2	2	3	-	X	21.5	1.9					4.4
3	1	4	-	X	10.7	3.8					2.8
4	0	1	X	-	1.0	0.0					0.0

TABLE 6.2-27
SUBCOMPARTMENT NODAL DESCRIPTION FOR BREAK IN RWCU HEAT EXCHANGER ROOM B (CASE 5)

VOLUME NO.	DESCRIPTION	HEIGHT ft.	CROSS-SECTIONAL AREA ft^2	INITIAL CONDITIONS TEMP, °F	PRESS. psia	HUMID, %	DBA BREAK CONDITIONS BREAK LOC. VOL. NO.	BREAK LINE	BREAK AREA ft^2	BREAK TYPE	CALC. PEAK PRESS. DIFF. psig	DESIGN PEAK PRESS. DIFF, psig	DESIGN MARGIN %
1	RWCU Heat Exchanger Room B	35.2	195.0	122.0	14.7	~0	1	RWCU	-	Double-ended guillotine rupture	3.0	-	-
2	RWCU Heat Exchanger Valve Room B	9.2	129.4	104.0	14.7	~0	-	-	-	-	-	-	-
3	Containment	125.0	12000.0	104.0	14.7	~0	-	-	-	-	-	-	-

TABLE 6.2-28

SUBCOMPARTMENT VENT PATH DESCRIPTION FOR BREAK IN RWCU HEAT EXCHANGER ROOM B (CASE 5)

			DESCRIPTION OF VENT PATH FLOW			LENGTH ft / HYDRAULIC DIAMETER ft	HEAD LOSS, K				
VENT PATH NO.	FROM VOL. NODE NO.	TO VOL. NODE NO.	CHOKED	UNCHOKED	AREA ft^2	INERTIA (ft^{-1})	FRICTION K, ft/d	TURNING LOSS, K	EXPANSION K	CONTRACTION K	TOTAL
1	1	2	-	X	22.9	1.0					1.8
2	2	3	-	X	19.1	2.0					2.9
3	0	1	X	-	1.0	0.0					0.0

TABLE 6.2-29
SUBCOMPARTMENT NODAL DESCRIPTION FOR BREAK IN RWCU HEAT EXCHANGER VALVE ROOM A (CASE 6)

				INITIAL CONDITIONS			DBA BREAK CONDITIONS						
VOLUME NO.	DESCRIPTION	HEIGHT ft.	CROSS-SECTIONAL AREA ft^2	TEMP, °F	PRESS. psia	HUMID, %	BREAK LOC. VOL. NO.	BREAK LINE	BREAK AREA ft^2	BREAK TYPE	CALC. PEAK PRESS. DIFF. psig	DESIGN PEAK PRESS. DIFF, psig	DESIGN MARGIN %
1	RWCU Heat Exchanger Room A	35.2	198.5	122.0	14.7	~0	2	RWCU	-	Double-ended guillotine rupture	2.8	-	-
2	RWCU Heat Exchanger Valve Room A	9.2	139.3	104.0	14.7	~0	-	-	-	-	-	-	-
3	Containment	125.0	12000.0	104.0	14.7	~0	-	-	-	-	-	-	-
4	RWCU Crossover Pipe Tunnel	33.4	28.2	122.0	14.7	~0	-	-	-	-	-	-	-

TABLE 6.2-30
SUBCOMPARTMENT VENT PATH DESCRIPTION FOR BREAK IN RWCU HEAT EXCHANGER ROOM A (CASE 6)

VENT PATH NO.	FROM VOL. NODE NO.	TO VOL. NODE NO.	DESCRIPTION OF VENT PATH FLOW: CHOKED	DESCRIPTION OF VENT PATH FLOW: UNCHOKED	AREA ft^2	$\frac{\text{LENGTH ft}}{\text{HYDRAULIC DIAMETER ft}}$ INERTIA (ft^{-1})	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	2	1	-	X	22.9	1.0					1.7
2	2	3	-	X	21.5	1.9					4.4
3	1	4	-	X	10.7	3.8					2.8
4	0	2	X	-	1.0	0.0					0.0

TABLE 6.2-31
SUBCOMPARTMENT NODAL DESCRIPTION FOR BREAK IN RWCU HEAT EXCHANGER VALVE ROOM B (CASE 8)

				INITIAL CONDITIONS			DBA BREAK CONDITIONS						
VOLUME NO.	DESCRIPTION	HEIGHT ft.	CROSS-SECTIONAL AREA ft^2	TEMP, °F	PRESS. psia	HUMID, %	BREAK LOC. VOL. NO.	BREAK LINE	BREAK AREA ft^2	BREAK TYPE	CALC. PEAK PRESS. DIFF. psig	DESIGN PEAK PRESS. DIFF, psig	DESIGN MARGIN %
1	RWCU Heat Exchanger Room B	35.2	195.0	122.0	14.7	~0	2	RWCU	-	Double-ended guillotine rupture	3.1	-	-
2	RWCU Heat Exchanger Valve Room B	9.2	129.4	104.0	14.7	~0	-	-	-	-	-	-	-
3	Containment	125.0	12000.0	104.0	14.7	~0	-	-	-	-	-	-	-

TABLE 6.2-32
SUBCOMPARTMENT VENT PATH DESCRIPTION FOR BREAK IN RWCU HEAT EXCHANGER VALVE ROOM B (CASE 8)

VENT PATH NO.	FROM VOL. NODE NO.	TO VOL. NODE NO.	DESCRIPTION OF VENT PATH FLOW: CHOKED	DESCRIPTION OF VENT PATH FLOW: UNCHOKED	AREA ft^2	LENGTH ft / HYDRAULIC DIAMETER ft: INERTIA (ft^{-1})	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	2	1	-	X	22.9	1.0					1.7
2	2	3	-	X	19.1	2.0					2.9
3	0	2	X	-	1.0	0.0					0.0

TABLE 6.2-33
SUBCOMPARTMENT NODAL DESCRIPTION FOR BREAK IN RWCU CROSSOVER PIPE TUNNEL (CASE 7)

				INITIAL CONDITIONS			DBA BREAK CONDITIONS						
VOLUME NO.	DESCRIPTION	HEIGHT ft.	CROSS-SECTIONAL AREA ft^2	TEMP, °F	PRESS. psia	HUMID, %	BREAK LOC. VOL. NO.	BREAK LINE	BREAK AREA ft^2	BREAK TYPE	CALC. PEAK PRESS. DIFF. psig	DESIGN PEAK PRESS. DIFF, psig	DESIGN MARGIN %
1	RWCU Heat Exchanger Room A	35.2	198.5	122.0	14.7	~0	4	RWCU	-	Double-ended guillotine rupture	7.6	-	-
2	RWCU Heat Exchanger Valve Room A	9.2	139.3	104.0	14.7	~0	-	-	-	-	-	-	-
3	Containment	125.0	12000.0	104.0	14.7	~0	-	-	-	-	-	-	-
4	RWCU Crossover Pipe Tunnel	33.4	28.2	122.0	14.7	~0	-	-	-	-	-	-	-

TABLE 6.2-34

SUBCOMPARTMENT VENT PATH DESCRIPTION FOR BREAK IN RWCU CROSSOVER PIPE TUNNEL (CASE 7)

VENT PATH NO.	FROM VOL. NODE NO.	TO VOL. NODE NO.	DESCRIPTION OF VENT PATH FLOW: CHOKED	DESCRIPTION OF VENT PATH FLOW: UNCHOKED	AREA ft^2	LENGTH ft / HYDRAULIC DIAMETER ft: INERTIA (ft^{-1})	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	-	X	22.9	1.0					1.8
2	2	3	-	X	21.5	1.9					4.4
3	4	1	-	X	10.7	3.8					2.2
4	0	4	X	-	1.0	0.0					0.0

TABLE 6.2-35
SUBCOMPARTMENT NODAL DESCRIPTION FOR BREAK IN F/D HOLDING PUMP ROOM (CASE 3)

VOLUME NO.	DESCRIPTION	HEIGHT ft.	CROSS-SECTIONAL AREA ft^2	INITIAL CONDITIONS TEMP, °F	PRESS. psia	HUMID, %	DBA BREAK CONDITIONS BREAK LOC. VOL. NO.	BREAK LINE	BREAK AREA ft^2	BREAK TYPE	CALC. PEAK PRESS. DIFF. psig	DESIGN PEAK PRESS. DIFF, psig	DESIGN MARGIN %
1	F/D Valve Room	9.7	437.2	104.0	14.7	~0	2	RWCU	-	Double-ended guillotine rupture	4.1	-	-
2	F/D Holding Pump Room	10.0	290.1	104.0	14.7	~0	-	-	-	-	-	-	-
3	Containment	125.0	12000.0	104.0	14.7	~0	-	-	-	-	-	-	-

TABLE 6.2-36
SUBCOMPARTMENT VENT PATH DESCRIPTION FOR BREAK IN F/D HOLDING PUMP ROOM (CASE 3)

VENT PATH NO.	FROM VOL. NODE NO.	TO VOL. NODE NO.	DESCRIPTION OF VENT PATH FLOW: CHOKED	DESCRIPTION OF VENT PATH FLOW: UNCHOKED	AREA ft^2	LENGTH ft / HYDRAULIC DIAMETER ft: INERTIA (ft^{-1})	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	2	1	-	X	12.3	1.8					1.4
2	2	3	-	X	21.5	1.7					1.9
3	0	2	X	-	1.0	0.0					0.0

TABLE 6.2-37*
SUBCOMPARTMENT NODAL DESCRIPTION FOR BREAK IN F/D ROOM (CASE 4)

				INITIAL CONDITIONS			DBA BREAK CONDITIONS						
VOLUME NO.	DESCRIPTION	HEIGHT ft.	CROSS-SECTIONAL AREA ft^2	TEMP, °F	PRESS. psia	HUMID, %	BREAK LOC. VOL. NO.	BREAK LINE	BREAK AREA ft^2	BREAK TYPE	CALC. PEAK PRESS. DIFF. psig	DESIGN PEAK PRESS. DIFF, psig	DESIGN MARGIN %
1	F/D Valve Room	9.7	437.2	104.0	14.7	~0	4	RWCU	-	Double-ended guillotine rupture	10.5	-	-
2	F/D Holding Pump Room	10.0	290.1	104.0	14.7	~0	-	-	-	-	-	-	-
3	Containment	125.0	12000.0	104.0	14.7	~0	-	-	-	-	-	-	-
4	F/D Room	21.67	96.5	122.0	14.7	~0	-	-	-	-	-	-	-

* Applicable for F/D Rooms 1 and 2.

TABLE 6.2-38*
SUBCOMPARTMENT VENT PATH DESCRIPTION FOR BREAK IN F/D ROOM (CASE 4)

VENT PATH NO.	FROM VOL. NODE NO.	TO VOL. NODE NO.	DESCRIPTION OF VENT PATH FLOW		AREA ft^2	LENGTH ft / HYDRAULIC DIAMETER ft	HEAD LOSS, K				
			CHOKED	UNCHOKED		INERTIA (ft^{-1})	FRICTION K, ft/d	TURNING LOSS, K	EXPANSION K	CONTRACTION K	TOTAL
1	1	2	-	X	12.3	1.8					1.4
2	2	3	-	X	21.5	1.7					1.9
3	4	1	-	X	4.5	1.4					3.7
4	4	3	-	X	36.0	0.5					2.9
5	0	4	X	-	1.0	0.0					0.0

* Applicable for F/D Rooms 1 and 2.

TABLE 6.2-39
SUBCOMPARTMENT NODAL DESCRIPTION
FOR BREAK IN FILTER-DEMINERALIZER VALVE ROOM

				INITIAL CONDITIONS			DBA BREAK CONDITIONS						
VOLUME NO.	DESCRIPTION	HEIGHT ft.	CROSS-SECTIONAL AREA ft^2	TEMP, °F	PRESS. psia	HUMID, %	BREAK LOC. VOL. NO.	BREAK LINE	BREAK AREA ft^2	BREAK TYPE	CALC. PEAK PRESS. DIFF. psig	DESIGN PEAK PRESS. DIFF, psig	DESIGN MARGIN %
1	F/D Valve Room	9.7	437.2	104.0	14.7	~0	1	RWCU	-	Double-ended guillotine rupture	5.0	-	-
2	F/D Holding Pump Room	10.0	290.1	104.0	14.7	~0	-	-	-	-	-	-	-
3	Containment	125.0	12000.0	104.0	14.7	~0	-	-	-	-	-	-	-

TABLE 6.2-40
SUBCOMPARTMENT VENT PATH DESCRIPTION FOR BREAK IN F/D ROOM (CASE 1)

VENT PATH NO.	FROM VOL. NODE NO.	TO VOL. NODE NO.	DESCRIPTION OF VENT PATH FLOW: CHOKED	DESCRIPTION OF VENT PATH FLOW: UNCHOKED	AREA ft^2	LENGTH ft / HYDRAULIC DIAMETER ft: INERTIA (ft^{-1})	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	-	X	12.3	1.8					1.4
2	2	3	-	X	21.5	1.7					1.9
3	0	1	X	-	1.0	0.0					0.0

TABLE 6.2-41
MASS AND ENERGY RELEASE DATA*
FOR RECIRCULATION OUTLET LINE WITH DIVERTER

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** (BTU/sec)
0.	0.	0.	528.	1196.	0.	0.
0.00572	5585.3	0.	528.	1196.	5585.3	2.949×10^6
0.00875	11170.5	0.	528.	1196.	11170.5	5.899×10^6
0.01111	16751.4	0.	528.	1196.	16751.4	8.845×10^6
0.01311	22341.0	0.	528.	1196.	22341.0	1.180×10^7
0.01424	25920.0	0.	528.	1196.	25920.0	1.369×10^7
0.01575	25920.0	0.	528.	1196.	25920.0	1.369×10^7
0.01575	13638.1	0.	528.	1196.	13638.1	7.201×10^6
0.01647	14787.6	0.	528.	1196.	14787.6	7.808×10^6
0.01794	17250.3	0.	528.	1196.	17250.3	9.108×10^6
0.01932	19716.8	0.	528.	1196.	19716.8	1.041×10^7
0.02061	22179.5	0.	528.	1196.	22179.5	1.171×10^7
0.02183	24646.1	0.	528.	1196.	24646.1	1.301×10^7
0.02242	25920.0	0.	528.	1196.	25920.0	1.369×10^7
1.655	25920.0	0.	528.	1196.	25920.0	1.369×10^7
1.655	21395.0	0.	528.	1196.	21395.0	1.130×10^7
5.000	21395.0	0.	528.	1196.	21395.0	1.130×10^7

* Tabulated values were halved when used in symmetric model.

** Mass and energy flow rates based on methodology presented in Reference 3.

TABLE 6.2-42
MASS AND ENERGY RELEASE DATA*
FOR FEEDWATER LINE BREAK IN BIOLOGICAL SHIELD ANNULUS

Time (sec)	Mass Release Rate (lb_m/sec) Liquid	Mass Release Rate (lb_m/sec) Steam	Enthalpy (Btu/lb_m) Liquid	Enthalpy (Btu/lb_m) Steam	Energy Release Rate (Btu/sec) Liquid	Energy Release Rate (Btu/sec) Steam
0.000	13,959.	0.0	398.00	1190.4	5,555,682.	-
0.0046	10,749.	632.2	452.12	1190.4	4,859,838.	752,571
0.5660	3,769.	632.2	552.35	1190.4	2,081,807.	752,571
5.0000	3,769.	632.2	552.35	1190.4	2,081,807.	752,571

* Mass and energy flow rates based on methodology presented in Reference 3.

TABLE 6.2-43
REACTOR BLOWDOWN DATA FOR RECIRCULATION LINE BREAK

TIME (sec)	RPV PRESSURE (psia)	LIQUID FLOW (LBM/SEC)	STEAM FLOW (LBM/SEC)	BREAK ENTHALPY (Btu/LBM)
0.0	1040	16,310	0	525.1
0.00098	1040	16,310	0	525.1
0.00293	1040	24,520	0	525.1
0.110	1041	24,550	0	525.1
1.00	1042	24,620	0	525.1
1.997	1044	20,570	0	525.1
2.00	1044	17,720	0	549.2
3.01	1048	17,750	0	549.9
4.01	1064	17,880	0	552.4
5.01	1082	18,030	0	555.1
10.01	1157	18,580	0	565.9
17.122	1192	18,820	0	570.7
17.126	1192	9256	3015	721.7
20.00	1067	7688	2973	730.2
25.06	809.3	5160	2588	740.8
30.00	582.9	3259	2059	752.6
40.50	341.8	0	1547	1204.0
42.07	323.2	0	1463	1203.0
43.63	306.1	9828	0	396.0
45.19	292.2	9597	0	391.3
46.75	278.4	9353	0	386.6
48.32	265.3	9119	0	381.9
88.94	127.1	6277	0	317.2
128.00	100.2	5978	0	298.7
167.07	75.7	5881	0	278.2
323.32	17.1	1619	0	187.9

TABLE 6.2-44
REACTOR BLOWDOWN DATA FOR MAIN STEAMLINE BREAK

TIME (SEC)	RPV PRESSURE (PSIA)	LIQUID FLOW (LBM/SEC)	STEAM FLOW (LBM/SEC)	BREAK ENTHALPY (BTU/LBM)
0.0	1040	0	8249	1191
0.00293	1040	0	8248	1191
0.00488	1040	0	9626	1191
0.0811	1035	0	9574	1192
0.1045	1033	0	9562	1192
0.1064	1033	0	6934	1192
0.997	992.9	0	6650	1193
1.001	992.7	23,070	791.1	563.0
2.00	983.2	22,660	885.5	564.5
3.00	972.6	22,220	980.7	566.0
4.04	966.9	21,810	1088	568.6
5.00	967.0	19,190	1070	572.1
10.08	989.4	15,450	1448	596.8
15.08	939.5	13,060	1898	616.8
20.02	818.2	10,170	2129	631.4
25.02	650.9	7188	2066	642.6
30.08	478.6	4595	1772	655.7
40.11	289.7	0	1637	1203.0
81.05	144.9	0	828	1194.0
159.17	84.8	0	491	1184.0
198.23	71.1	0	413	1181.0
315.42	50.0	0	294	1174.0
432.61	40.0	0	236	1170.0
588.86	32.9	0	196	1166.0
706.5	29.5	0	176	1164.0
784.17	28.0	0	154	1163.0
1294.45	25.1	0	107	1161.0

TABLE 6.2-45
CORE DECAY HEAT FOLLOWING LOCA
FOR SHORT-TERM CONTAINMENT ANALYSES

TIME (SEC)	NORMALIZED CORE HEAT(1)
0	1.0879
2	0.5605
6	0.554
10	0.3898
20	0.1278
30	0.0811

Note:
(1) Normalized to 3543 MWt. Includes Metal Water Reaction and Fuel Relaxation Energy.

TABLE 6.2-45a
CORE DECAY HEAT FOLLOWING LOCA
FOR LONG-TERM CONTAINMENT ANALYSES

TIME (sec)	NORMALIZED CORE HEAT(1)
0	1.004380
1	0.588680
2	0.550580
4	0.575620
10	0.376350
20	0.116160
60	0.041223
80	0.038990
100	0.037540
120	0.036548
120(2)	0.032168
150	0.030680
400	0.025450
800	0.021930
1000	0.020750
2000	0.017030
4000	0.013660
6000	0.012050
8000	0.011100
10000	0.010430
20000	0.009149
40000	0.007719
60000	0.006956
80000	0.006434
100000	0.006047
200000	0.004884
400000	0.003809
800000	0.002872
1000000	0.002618
1500000	0.002207

Notes:

(1) Normalized to 3543 MWt. Includes Metal Water Reaction and Fuel Relaxation energy.

(2) Does not include Metal Water Reaction energy.

TABLE 6.2-45b
EPU 24-MONTH CYCLE DECAY HEAT DATA (1)

TIME (sec)	NORMALIZED CORE HEAT (2)
1	1.004E-00
2	5.886E-01
4	5.505E-01
10	5.756E-01
20	3.763E-01
60	1.162E-01
80	4.125E-02
100	3.903E-02
150	3.758E-02
200	3.073E-02
400	2.909E-02
600	2.551E-02
800	2.347E-02
1000	2.199E-02
1500	2.081E-02
2000	1.863E-02
4000	1.709E-02
6000	1.371E-02
8000	1.211E-02
10000	1.150E-02
15000	1.049E-02
20000	9.875E-03
40000	9.202E-03
60000	7.750E-03
80000	7.012E-03
86400	6.489E-03
100000	6.353E-03
150000	6.101E-03
173000	5.407E-03
200000	4.931E-03
259000	4.516E-03

TABLE 6.2-46

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TABLE 6.2-46 (CON'T)

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TABLE 6.2-47
ISOLATION VALVE SUMMARY FOR LINE PENETRATING CONTAINMENT

CONT. PENE. NO. NOTE 13	SYSTEM NAME	VALVE NUMBER	LINE SIZE (IN)	VALVE LOCA-TION	FLUID CON-TAINED	PIPE LENGTH-CONT. TO OUTER-MOST ISO VALV	NRC DESIGN CRITERIA	VALVE TYPE AND OPERATOR	PRIMARY MODE	SECON-DARY MODE	NORMAL NOTE 12	SHUT-DOWN	POST LOCA	POWER FAILURE	POWER SOURCE	CONT. ISOL. SIGNAL NOTE 8	CLOSURE TIME (SEC) NOTE 10	ENGNRD SAFETY FEATURE NOTE 15	THRU LINE LEAK CLASS NOTE 17	REFERENCE FIGURE/ DRAWING	TYPE C TEST	TYPE MEDIA	REMARKS
1.	Equip Hatch	1CM099	0.25	IC	A	2.5"	56	GL/MAN	Man	N/A	Shut Shut	Open/Shut Shut	Shut Shut	N/A	N/A	None	N/A N/A	Yes	1 1	3.6-1 Sh 41	N/A Yes	CA/G CA/G	Note 14 Test Conn. for Seals
2.	Personnel Lock 737'										Shut	Open/Shut	Shut				N/A		1		N/A	CA/G	Note 14
3.	Personnel Lock 828'										Shut	Open/Shut	Shut				N/A		2		N/A	CA/G	Note 14
4.	Fuel Trans Tube	 1F42F304A 1F42F304B	 0.75 0.75	 IC IC	 A A		 56 56	 GL/MAN GL/MAN	 Man Man	 N/A N/A	Shut Shut Shut	Open/Shut Shut Shut	Shut Shut Shut	 N/A N/A	 N/A N/A	 None None	N/A N/A N/A	 No No	1 1 1	M05-1080	N/A Yes Yes	CA/G CA/G G	Note 14
5.	Main Steam	1B21F022C	24	ID	S		55	GL/AO	Auto	RM	Open	Shut	Shut	Shut	Div. 2	C,D,E,F,G H,J,U,X,R	3-5	Yes	1	M05-1002 Sh 1 & 2	Yes	G	Note 1
		1B21F028C	24	OC	S	1'-7"	55	GL/AO	Auto	RM	Open	Shut	Shut	Shut	Div. 1	C,D,E,F,G H,J,U,X,R	3-5	Yes	1	M05-1070	Yes	G	
		1B21F067C	1.5	OC	S/W	15"	55	GL/MO	Auto	RM	Open	Shut	Shut	As-Is	Div. 1	C,D,E,F,G H,J,U,X,R	*14	Yes	1	6.2-123 Sh 1	Yes	G	*Note 28
		1B21F025C	0.75	OC	S		55	GL/MAN	Man	N/A	Shut	Shut	Shut	N/A	N/A	None	N/A	Yes	1		Yes	G	
		1E32F001J	1.5	OC	S/W	26'-1"	55	GL/MO	RM	Man	Shut	Shut	Shut	As Is	Div. 1	None	N/A	Yes	1		Yes	G	Note 6
	MSIV/ LCS	1E32F327C	0.5	OC	S/W		55	GL/MAN	Man	N/A	Shut	Shut	Shut	N/A	N/A	None	N/A	Yes	1		Yes	G	
		1E32F330A	0.5	OC	S/W		55	GL/MAN	Man	N/A	Shut	Shut	Shut	N/A	N/A	None	N/A	Yes	1		Yes	G	
6.	Main Steam	1B21F022A	24	ID	S		55	GL/AO	Auto	RM	Open	Shut	Shut	Shut	Div. 2	C,D,E,F,G H,J,U,X,R	3-5	Yes	1	M05-1002 Sh 1 & 2	Yes	G	Note 1
		1B21F028A	24	OC	S	1'-7"	55	GL/AO	Auto	RM	Open	Shut	Shut	Shut	Div. 1	C,D,E,F,G H,J,U,X,R	3-5	Yes	1	M05-1070	Yes	G	
		1B21F067A	1.5	OC	S/W	15'	55	GL/MO	Auto	RM	Open	Shut	Shut	As Is	Div. 1	C,D,E,F,G H,J,U,X,R	*14	Yes	1	6.2-123 Sh 1	Yes	G	*Note 28
		1B21F025A	0.75	OC	S		55	GL/MAN	Man	N/A	Shut	Shut	Shut	N/A	N/A	None	N/A	Yes	1		Yes	G	
	MSIV/ LCS	1E32F001A	1.5	OC	S/W	26'-1"	55	GL/MO	RM	Man	Shut	Shut	Shut	As Is	Div. 1	N/A	N/A	Yes	1		Yes	G	Note 6
		1E32F327A	0.5	OC	S/W		55	GL/MAN	Man	N/A	Shut	Shut	Shut	N/A	N/A	None	N/A	Yes	1		Yes	G	
		1E32F329A	0.5	OC	S/W		55	GL/MAN	Man	N/A	Shut	Shut	Shut	N/A	N/A	None	N/A	Yes	1		Yes	G	
7.	Main Steam	1B21F022D	24	ID	S		55	GL/AO	Auto	RM	Open	Shut	Shut	Shut	Div. 2	C,D,E,F,G, H,J,U,X,R	3-5	Yes	1	M05-1002 Sh 1 & 2	Yes	G	Note 1
		1B21F028D	24	OC	S	1'-7"	55	GL/AO	Auto	RM	Open	Shut	Shut	Shut	Div. 1	C,D,E,F,G, H,J,U,X,R	3-5	Yes	1	M05-1070	Yes	G	
		1B21F067D	1.5	OC	S/W	15'	55	GL/MO	Auto	RM	Open	Shut	Shut	As Is	Div. 1	C,D,E,F,G, H,J,U,X,R	*14	Yes	1	6.2-123 Sh 1	Yes	G	*Note 28
		1B21F025D	0.75	OC	S		55	GL/MAN	Man	N/A	Shut	Shut	Shut	N/A	N/A	None	N/A	Yes	1		Yes	G	
		1E32F001N	1.5	OC	S/W	26'-1"	55	GL/MO	RM	Man	Shut	Shut	Shut	As Is	Div. 1	None	N/A	Yes	1		Yes	G	Note 6
	MSIV/ LCS	1E32F327D	0.5	OC	S/W		55	GL/MAN	Man	N/A	Shut	Shut	Shut	N/A	N/A	None	N/A	Yes	1		Yes	G	
		1E32F330C	0.5	OC	S/W		55	GL/MAN	Man	N/A	Shut	Shut	Shut	N/A	N/A	None	N/A	Yes	1		Yes	G	

TABLE 6.2-47
ISOLATION VALVE SUMMARY FOR LINE PENETRATING CONTAINMENT (Continued)

CONT. PENE. NO. NOTE 13	SYSTEM NAME	VALVE NUMBER	LINE SIZE (IN)	VALVE LOCA-TION	FLUID CON-TAINED	PIPE LENGTH-CONT. TO OUTER-MOST ISO VALV	NRC DESIGN CRITERIA	VALVE TYPE AND OPERATOR	PRIMARY MODE	SECON-DARY MODE	NORMAL NOTE 12	SHUT-DOWN	POST LOCA	POWER FAILURE	POWER SOURCE	CONT. ISOL. SIGNAL NOTE 8	CLOSURE TIME (SEC) NOTE 10	ENGNRD SAFETY FEATURE NOTE 15	THRU LINE LEAK CLASS NOTE 17	REFERENCE FIGURE/ DRAWING	TYPE C TEST	TYPE MEDIA	REMARKS
8.	Main Steam	1B21F022B	24	ID	S		55	GL/AO	Auto	RM	Open	Shut	Shut	Shut	Div. 2	C,D,E,F,G, H,J,U,X,R	3-5	Yes	1	M05-1002 Sh 1 & 2	Yes	G	Note 1
		1B21F028B	24	OC	S	1'-7"	55	GL/AO	Auto	RM	Open	Shut	Shut	Shut	Div. 1	C,D,E,F,G, H,J,U,X,R	3-5	Yes	1	M05-1070	Yes	G	
		1B21F067B	1.5	OC	S/W	15'	55	GL/MO	Auto	RM	Open	Shut	Shut	As Is	Div. 1	C,D,E,F,G, H,J,U,X,R	*14	Yes	1	6.2-123 Sh 1	Yes	G	*Note 28
		1B21F025B	0.75	OC	S		55	GL/MAN	Man	N/A	Shut	Shut	Shut	N/A	N/A	None	N/A	Yes	1		Yes	G	
	MSIV/ LCS	1E32F001E	1.5	OC	S/W	26'-1"	55	GL/MO	RM	Man	Shut	Shut	Shut	As Is	Div. 1	None	N/A	Yes	1		Yes	G	Note 6
		1E32F327B	0.5	OC	S/W		55	GL/MAN	Man	N/A	Shut	Shut	Shut	N/A	N/A	None	N/A	Yes	1		Yes	G	
		1E32F329C	0.5	OC	S/W		55	GL/MAN	Man	N/A	Shut	Shut	Shut	N/A	N/A	None	N/A	Yes	1		Yes	G	
9.	Feed-Water	1B21F010A	18	ID	W		55	CH/-	RevFlow	N/A	Open	Shut/Open	Open/Shut	N/A	N/A	None	N/A	No	2	M05-1004	No	*CA/G	*Notes 20, 24
		1B21F032A	20	OC	W	2'-6"	55	CH/AO	RevFlow	Auto	Open	Shut/Open	Open/Shut	Shut	Div. 2	B,L,R	N/A	No	2	M05-1075 Sh 1	Yes	G	Notes 16, 24
		1B21F065A	20	OC	W	14'	55	GA/MO	RM	Man	Open	Shut	Open/Shut	As Is	Div. 1	None	N/A	No	2		Yes	G	Note 24
	RHR	1E12F053A	10	OC	W	90'	55	GL/MO	Auto	RM	Shut	Shut/Open	Shut	As Is	Div. 1	A,S,T,X,R	*65	No	1		Yes	G	*Notes 24, 28
		1E12F497	2.50	OC	W	88'	55	GL/MO	RM	Man	Shut	Shut	Open	As Is	Div. 1	None	N/A	No	1		Yes	G	
		1E12F501A	0.75	OC	W		55	GL/MAN	Man	N/A	Shut	Shut	Shut	N/A	N/A	None	N/A	No	1		Yes	G	
		1E12F503A	0.75	OC	W		55	GL/MAN	Man	N/A	Shut	Shut	Shut	N/A	N/A	None	N/A	No	1		Yes	G	
		1E12F507	0.75	OC	W		55	GL/MAN	Man	N/A	Shut	Shut	Shut	N/A	N/A	None	N/A	No	1		Yes	G	
		1E12F511A	0.50	OC	W		55	GL/MAN	Man	N/A	Shut	Shut	Shut	N/A	N/A	None	N/A	No	1		Yes	G	
		1E12F513	0.50	OC	W		55	GL/MAN	Man	N/A	Shut	Shut	Shut	N/A	N/A	None	N/A	No	1		Yes	G	
		1E12F523A	0.75	OC	W		55	GL/MAN	Man	N/A	Shut	Shut	Shut	N/A	N/A	None	N/A	No	1		Yes	G	
		1E12F525A	0.75	OC	W		55	GL/MAN	Man	N/A	Shut	Shut	Shut	N/A	N/A	None	N/A	No	1		Yes	G	
		1E12F058A	0.75	OC	W		55	GL/MAN	Man	N/A	Shut	Shut	Shut	N/A	N/A	None	N/A	No	1		Yes	G	
		1E12F349A	0.75	OC	W		55	GL/MAN	Man	N/A	Shut	Shut	Shut	N/A	N/A	None	N/A	No	1		Yes	G	
	Feed-Water	1B21F030A	0.75	OC	W		55	GL/MAN	Man	N/A	Shut	Shut	Shut	N/A	N/A	None	N/A	No	1		Yes	G	
		1B21F063A	0.75	ID	W		56	GL/MAN	Man	N/A	Shut	Shut	Shut	N/A	N/A	None	N/A	No	2		Yes	G	
		1B21F518A	0.75	OC	W		55	GL/MAN	Man	N/A	Shut	Shut	Shut	N/A	N/A	None	N/A	No	1		Yes	G	

TABLE 6.2-47
ISOLATION VALVE SUMMARY FOR LINE PENETRATING CONTAINMENT (Continued)

CONT. PENE. NO. NOTE 13	SYSTEM NAME	VALVE NUMBER	LINE SIZE (IN)	VALVE LOCA-TION	FLUID CON-TAINED	PIPE LENGTH-CONT. TO OUTER-MOST ISO VALV	NRC DESIGN CRITERIA	VALVE TYPE AND OPERATOR	PRIMARY MODE	SECON-DARY MODE	NORMAL NOTE 12	SHUT-DOWN	POST LOCA	POWER FAILURE	POWER SOURCE	CONT. ISOL. SIGNAL NOTE 8	CLOSURE TIME (SEC) NOTE 10	ENGNRD SAFETY FEATURE NOTE 15	THRU LINE LEAK CLASS NOTE 17	REFERENCE FIGURE/ DRAWING	TYPE C TEST	TYPE MEDIA	REMARKS
10.	Feed-Water	1B21F010B	18	ID	W		55	CH/-	RevFlow	N/A	Open	Shut/Open	Open/Shut	N/A	N/A	None	N/A	No	2	M05-1004	No	*CA/G	*Notes 20, 24
		1B21F032B	20	OC	W	2'-6"	55	CH/AO	RevFlow	Auto	Open	Shut/Open	Open/Shut	Shut	Div .2	B,L,R	N/A	No	2	M05-1075 Sh 2	Yes	G	Notes 16, 24
		1B21F065B	20	OC	W	14'	55	GA/MO	RM	Man	Open	Shut	Open/Shut	As Is	Div. 1	None	N/A	No	2		Yes	G	
	RHR	1E12F053B	10	OC	W	90'	55	GL/MO	Auto	RM	Shut	Shut/Open	Shut	As Is	Div. 1	A,S,T,X,R	*65	No	1	M05-1076 Sh 4	Yes	G	Note 24
		1E12F496	2.00	OC	W	88'	55	GL/MO	RM	Man	Shut	Shut	Open	As Is	Div. 2	None	N/A	No	1		Yes	G	*Notes 24, 28
	Feed-water	1B21F063B	0.75	ID	W		56	GL/MAN	Man	N/A	Shut	Shut	Shut	N/A	N/A	None	N/A	No	2		Yes	G	
		1B21F518B	0.75	OC	W		55	GL/MAN	Man	N/A	Shut	Shut	Shut	N/A	N/A	None	N/A	No	1		Yes	G	
		1B21F030B	0.75	OC	W		55	GL/MAN	Man	N/A	Shut	Shut	Shut	N/A	N/A	None	N/A	No	1		Yes	G	
	RHR	1E12F349B	0.75	OC	W		55	GL/MAN	Man	N/A	Shut	Shut	Shut	N/A	N/A	None	N/A	No	1		Yes	G	
		1E12F058B	0.75	OC	W		55	GL/MAN	Man	N/A	Shut	Shut	Shut	N/A	N/A	None	N/A	No	1		Yes	G	
		1E12F501B	0.75	OC	W		55	GL/MAN	Man	N/A	Shut	Shut	Shut	N/A	N/A	None	N/A	No	1		Yes	G	
		1E12F503B	0.75	OC	W		55	GL/MAN	Man	N/A	Shut	Shut	Shut	N/A	N/A	None	N/A	No	1		Yes	G	
		1E12F505	0.75	OC	W		55	GL/MAN	Man	N/A	Shut	Shut	Shut	N/A	N/A	None	N/A	No	1		Yes	G	
		1E12F516A	0.50	OC	W		55	GL/MAN	Man	N/A	Shut	Shut	Shut	N/A	N/A	None	N/A	No	1		Yes	G	
		1E12F518	0.50	OC	W		55	GL/MAN	Man	N/A	Shut	Shut	Shut	N/A	N/A	None	N/A	No	1		Yes	G	
		1E12F523B	0.75	OC	W		55	GL/MAN	Man	N/A	Shut	Shut	Shut	N/A	N/A	None	N/A	No	1		Yes	G	
		1E12F525B	0.75	OC	W		55	GL/MAN	Man	N/A	Shut	Shut	Shut	N/A	N/A	None	N/A	No	1		Yes	G	
		1G33F057	0.75	OC	W		55	GL/MAN	Man	N/A	Shut	Shut	Shut	N/A	N/A	None	N/A	No	1		Yes	G	
																							Note 6
11.#	RHR	1E12F004A	20	OC	W	8'	56	GA/MO	RM	Man	Open	Open/Shut	Open	As Is	Div. 1	None	N/A	Yes	1	M05-1075 Sh 1	Yes	SP/W	Note 21
		1E12F334A	0.75	OC	W		56	GL/MAN	Man	N/A	Shut	Shut	Shut	N/A	N/A	None	N/A	Yes	1		No	SP/W	
		1E12F335A	0.75	OC	W		56	GL/MAN	Man	N/A	Shut	Shut	Shut	N/A	N/A	None	N/A	Yes	1		No	W	#Note 49
12.#	RHR	1E12F004B	20	OC	W	8'	56	GA/MO	RM	Man	Open	Open/Shut	Open	As Is	Div. 2	None	N/A	Yes	1	M05-1075 Sh 2	Yes	SP/W	Note 21
		1E12F334B	0.75	OC	W		56	GL/MAN	Man	N/A	Shut	Shut	Shut	N/A	N/A	None	N/A	Yes	1		No	SP/W	
		1E12F335B	0.75	OC	W		56	GL/MAN	Man	N/A	Shut	Shut	Shut	N/A	N/A	None	N/A	Yes	1		No	W	#Note 49
13.#	RHR	1E12F105	20	OC	W	4'	56	GA/MO	RM	Man	Open	Open/Shut	Open	As Is	Div. 2	None	N/A	Yes	1	M05-1075 Sh 3	Yes	SP/W	Note 18
		1E12F334C	0.75	OC	W		56	GL/MAN	Man	N/A	Shut	Shut	Shut	N/A	N/A	None	N/A	Yes	1		No	SP/W	
		1E12F335C	0.75	OC	W		56	GL/MAN	Man	N/A	Shut	Shut	Shut	N/A	N/A	None	N/A	Yes	1		No	W	#Note 49
14.	RHR	1E12F009	18	ID	W		55	GA/MO	Auto	RM	Shut	Open/Shut	Shut	As Is	Div. 2	A,S,T,X,R	**53	Yes	1	M05-1075 Sh 1	Yes	*CA/G	*Note 20 **Note 28
		1E12F008	18	OC	W	1'-6"	55	GA/MO	Auto	RM	Shut	Open/Shut	Shut	As Is	Div. 1	A,S,T,X,R	*53	Yes	1		Yes	G	*Note 28
		1E12F001	0.75	OC	W		55	GL/MAN	Man	N/A	Shut	Shut	Shut	N/A	N/A	None	N/A	Yes	1		Yes	G	

TABLE 6.2-47
<u>ISOLATION VALVE SUMMARY FOR LINE PENETRATING CONTAINMENT</u> (Continued)

CONT. PENE. NO. NOTE 13	SYSTEM NAME	VALVE NUMBER	LINE SIZE (IN)	VALVE LOCA-TION	FLUID CON-TAINED	PIPE LENGTH-CONT. TO OUTER-MOST ISO VALV	NRC DESIGN CRITERIA	VALVE TYPE AND OPERATOR	PRIMARY MODE	SECON-DARY MODE	NORMAL NOTE 12	SHUT-DOWN	POST LOCA	POWER FAILURE	POWER SOURCE	CONT. ISOL. SIGNAL NOTE 8	CLOSURE TIME (SEC) NOTE 10	ENGNRD SAFETY FEATURE NOTE 15	THRU LINE LEAK CLASS NOTE 17	REFERENCE FIGURE/ DRAWING	TYPE C TEST	TYPE MEDIA	REMARKS
15.	RHR	1E12F027A	12	OC	W	2'-3"	55	GA/MO	RM	Man	Open	Open	Open	As Is	Div. 1	None	N/A	Yes	1	M05-1075 Sh 1	Yes	G	
		1E12F028A	10	IC	W		56	GA/MO	RM	Man	Shut	Shut	Open/Shut	As Is	Div. 1	None	N/A	Yes	1		Yes	*CA/G	*Note 18
		1E12F025A*	1.0	OC	W	36'	55	PR/-	OVPress	N/A	Shut	Shut	Shut	N/A	N/A	None	N/A	Yes	1		Yes	G	*Note 35
		1E12F042A	12	IC	W		55	GA/MO	RM	Man	Shut	Shut	Open	As Is	Div. 1	None	N/A	Yes	1		Yes	*CA/G	*Note 18
		1E12F037A	10	IC	W		56	GL/MO	Auto	RM	Shut	Open/Shut	Shut	As Is	Div. 1	A,S,T,L,R	**120	Yes	1		Yes	*CA/G	*Note 18, **Note 28 Note 43
		1E12F107A	0.75	IC	W		56	GL/MAN	Man	N/A	Shut	Shut	Shut	N/A	N/A	None	N/A	Yes	1		Yes	*CA/G	*Note 18
		1E12F331A	0.75	IC	W		56	GL/MAN	Man	N/A	Shut	Shut	Shut	N/A	N/A	None	N/A	Yes	1		Yes	*CA/G	*Note 18
		1E12F329A	0.75	IC	W		56	GL/MAN	Man	N/A	Shut	Shut	Shut	N/A	N/A	None	N/A	Yes	1		Yes	*CA/G	*Note 18
		1E12F044A	3.0	IC	W		57	GA/MAN	Man	N/A	Shut	Shut	Shut	N/A	N/A	None	N/A	Yes	1		Yes	*CA/G	*Note 18
16.	RHR	1E12F027B	12	OC	W	1'-6"	55	GA/MO	RM	Man	Open	Open	Open	As Is	Div. 2	None	N/A	Yes	1	M05-1075 Sh 2	Yes	G	
		1E12F028B	10	IC	W		56	GA/MO	RM	Man	Shut	Shut	Open/Shut	As Is	Div. 2	None	N/A	Yes	1		Yes	*CA/G	*Note 18
		1E12F025B*	1.0	OC	W	17'	55	PR/-	OVPress	N/A	Shut	Shut	Shut	N/A	N/A	None	N/A	Yes	1		Yes	G	*Note 35
		1E12F042B	12	IC	W		55	GA/MO	RM	Man	Shut	Shut	Open	As Is	Div .2	None	N/A	Yes	1		Yes	*CA/G	*Note 18
		1E12F037B	10	IC	W		56	GL/MO	Auto	RM	Shut	Open/Shut	Shut	As Is	Div. 2	A,S,T,L,R	**120	Yes	1		Yes	*CA/G	*Note 18, **Note 28 Note 43
		1E12F107B	0.75	IC	W		56	GL/MAN	Man	N/A	Shut	Shut	Shut	N/A	N/A	None	N/A	Yes	1		Yes	*CA/G	*Note 18
		1E12F331B	0.75	IC	W		56	GL/MAN	Man	N/A	Shut	Shut	Shut	N/A	N/A	None	N/A	Yes	1		Yes	*CA/G	*Note 18
		1E12F329B	0.75	IC	W		56	GL/MAN	Man	N/A	Shut	Shut	Shut	N/A	N/A	None	N/A	Yes	1		Yes	*CA/G	*Note 18
		1E12F044B	3.0	IC	W		57	GA/MAN	Man	N/A	Shut	Shut	Shut	N/A	N/A	None	N/A	Yes	1		Yes	*CA/G	*Note 18
17.	RHR	1E12F042C	12	OC	W	1'-6"	55	GA/MO	RM	Man	Shut	Shut	Open/Shut	As Is	Div. 2	None	N/A	Yes	1	M05-1075 Sh 3	Yes	G	Note 6 Note 18
		1E12F056C	0.75	OC	W		55	GL/MAN	Man	N/A	Shut	Shut	Shut	N/A	N/A	None	N/A	Yes	1		Yes	G	Note 39

TABLE 6.2-47
ISOLATION VALVE SUMMARY FOR LINE PENETRATING CONTAINMENT (Continued)

CONT. PENE. NO. NOTE 13	SYSTEM NAME	VALVE NUMBER	LINE SIZE (IN)	VALVE LOCA-TION	FLUID CON-TAINED	PIPE LENGTH-CONT. TO OUTER-MOST ISO VALV	NRC DESIGN CRITERIA	VALVE TYPE AND OPERATOR	PRIMARY MODE	SECON-DARY MODE	NORMAL NOTE 12	SHUT-DOWN	POST LOCA	POWER FAILURE	POWER SOURCE	CONT. ISOL. SIGNAL NOTE 8	CLOSURE TIME (SEC) NOTE 10	ENGNRD SAFETY FEATURE NOTE 15	THRU LINE LEAK CLASS NOTE 17	REFERENCE FIGURE/ DRAWING	TYPE C TEST	TYPE MEDIA	REMARKS
18.#	RHR	1E12F024A	14	OC	W	94'	56	GA/MO	Auto	RM	Shut	Shut	Shut	As Is	Div. 1	L,U	*45	Yes	1	M05-1075	Yes	SP/W	*Note 28
		1E12F011A	4.0	OC	W	204'	56	GL/MAN	Man	N/A	Shut	Shut	Shut	N/A	N/A	None	N/A	Yes	1	Sh 1 & 4	Yes	SP/W	
		1E12F064A	4.0	OC	W	210'	56	GA/MO	RM	Man	Open	Shut/Open	Open/Shut	As Is	Div. 1	None	N/A	Yes	1		Yes	SP/W	
	LPCS	1E21F011	4.0	OC	W	180'	56	GA/MO	RM	Man	Open	Open	Open/Shut	As Is	Div. 1	None	N/A	Yes	1	6.2-143	Yes	SP/W	
		1E21F012	10	OC	W	101'	56	GL/MO	Auto	RM	Shut	Shut	Shut	As Is	Div. 1	L,U	*90	Yes	1		Yes	SP/W	*Note 28
	RHR	1E12F366A	0.75	OC	W		56	GL/MAN	Man	N/A	Shut	Shut	Shut	N/A	N/A	None	N/A	Yes	1		No	SP/W	
		1E12F365A	0.75	OC	W		56	GL/MAN	Man	N/A	Shut	Shut	Shut	N/A	N/A	None	N/A	Yes	1		No	W	
		1E12F418	0.75	OC	W		56	GL/MAN	Man	N/A	Shut	Shut	Shut	N/A	N/A	None	N/A	Yes	1		No	SP/W	
		1E12F419	0.75	OC	W		56	GL/MAN	Man	N/A	Shut	Shut	Shut	N/A	N/A	None	N/A	Yes	1		No	W	
		1E12F420	0.75	OC	W		56	GL/MAN	Man	N/A	Shut	Shut	Shut	N/A	N/A	None	N/A	Yes	1		No	SP/W	
		1E12F421	0.75	OC	W		56	GL/MAN	Man	N/A	Shut	Shut	Shut	N/A	N/A	None	N/A	Yes	1		No	W	
		1E12F414	0.75	OC	W		56	GL/MAN	Man	N/A	Shut	Shut	Shut	N/A	N/A	None	N/A	Yes	1		No	SP/W	
		1E12F415	0.75	OC	W		56	GL/MAN	Man	N/A	Shut	Shut	Shut	N/A	N/A	None	N/A	Yes	1		No	W	
	LPCS	1E21F347	0.75	OC	W		56	GL/MAN	Man	N/A	Shut	Shut	Shut	N/A	N/A	None	N/A	Yes	1		No	W	
		1E21F346	0.75	OC	W		56	GL/MAN	Man	N/A	Shut	Shut	Shut	N/A	N/A	None	N/A	Yes	1		No	SP/W	#Note 49
19.#	RHR	1E12F021	14	OC	W	120'	56	GL/MO	Auto	RM	Shut	Shut	Shut	As Is	Div. 2	L,U	*123	No	1	M05-1075 Sh 3	Yes	SP/W	*Note 28
		1E12F064C	4.0	OC	W	212'	56	GA/MO	RM	Man	Open	Shut/Open	Open/Shut	As Is	Div. 2	None	N/A	Yes	1		Yes	SP/W	
		1E12F353	0.75	OC	W		56	GL/MAN	Man	N/A	Shut	Shut	Shut	N/A	N/A	None	N/A	No	1		No	SP/W	
		1E12F354	0.75	OC	W		56	GL/MAN	Man	N/A	Shut	Shut	Shut	N/A	N/A	None	N/A	No	1		No	W	
		1E12F428	0.75	OC	W		56	GL/MAN	Man	N/A	Shut	Shut	Shut	N/A	N/A	None	N/A	No	1		No	SP/W	
		1E12F429	0.75	OC	W		56	GL/MAN	Man	N/A	Shut	Shut	Shut	N/A	N/A	None	N/A	No	1		No	W	#Note 49
20.#	RHR	1E12F024B	14	OC	W	130'	56	GA/MO	Auto	RM	Shut	Shut	Shut	As Is	Div. 2	L,U	*45	Yes	1	M05-1075 Sh 2	Yes	SP/W	*Note 28
		1E12F011B	4.0	OC	W	130"	56	GL/MAN	Man	N/A	Shut	Shut	Shut	N/A	N/A	None	N/A	Yes	1		Yes	SP/W	
		1E12F064B	4.0	OC	W	154'	56	GA/MO	RM	Man	Open	Shut/Open	Open/Shut	As Is	Div. 2	None	N/A	Yes	1		Yes	SP/W	
		1E12F366B	0.75	OC	W		56	GL/MAN	Man	N/A	Shut	Shut	Shut	N/A	N/A	None	N/A	Yes	1		No	SP/W	
		1E12F065B	0.75	OC	W		56	GL/MAN	Man	N/A	Shut	Shut	Shut	N/A	N/A	None	N/A	Yes	1		No	W	
		1E12F426	0.75	OC	W		56	GL/MAN	Man	N/A	Shut	Shut	Shut	N/A	N/A	None	N/A	Yes	1		No	SP/W	
		1E12F427	0.75	OC	W		56	GL/MAN	Man	N/A	Shut	Shut	Shut	N/A	N/A	None	N/A	Yes	1		No	W	#Note 49
21.#	RHR	1E12F017A	2.0	OC	W	68'	56	PR/-	OVPres s	N/A	Shut	Shut	Shut	N/A	N/A	None	N/A	Yes	1	M05-1075 Sh 1	Yes	SP/W	#Note 46, 49
22.	Spare																					CA	
23.	RHR	1E12F005	2.0	OC	W	53'	56	PR/-	OVPres s	N/A	Shut	Shut	Shut	N/A	N/A	None	N/A	Yes	1	M05-1075 Sh 1	No	SP	Notes 46, 48

TABLE 6.2-47
ISOLATION VALVE SUMMARY FOR LINE PENETRATING CONTAINMENT (Continued)

CONT. PENE. NO. NOTE 13	SYSTEM NAME	VALVE NUMBER	LINE SIZE (IN)	VALVE LOCA-TION	FLUID CON-TAINED	PIPE LENGTH-CONT. TO OUTER-MOST ISO VALV	NRC DESIGN CRITERIA	VALVE TYPE AND OPERATOR	PRIMARY MODE	SECON-DARY MODE	NORMAL NOTE 12	SHUT-DOWN	POST LOCA	POWER FAILURE	POWER SOURCE	CONT. ISOL. SIGNAL NOTE 8	CLOSURE TIME (SEC) NOTE 10	ENGNRD SAFETY FEATURE NOTE 15	THRU LINE LEAK CLASS NOTE 17	REFERENCE FIGURE/ DRAWING	TYPE C TEST	TYPE MEDIA	REMARKS
24.#	RHR	*1E12F055A	12	OC	W	83'	56	PR/-	OVPress	N/A	Shut	Shut	Shut	N/A	N/A	None	N/A	Yes	1	M05-1075	Yes	SP/W	Notes 45, 46 *Note 50
		1E12F433A	0.75	OC	W		56	GL/MAN	Man	N/A	Shut	Shut	Shut	N/A	N/A	None	N/A	Yes	1	Sh 1 & 4	No	W	
		1E12F432A	0.75	OC	W		56	GL/MAN	Man	N/A	Shut	Shut	Shut	N/A	N/A	None	N/A	Yes	1		No	SP/W	
		1E12F112A	2.0	OC	W		56	PR/-	OVPress	N/A	Shut	Shut	Shut	N/A	N/A	None	N/A	Yes	1		Yes		#Note 49
25.#	RHR	1E12F017B	2.0	OC	W	58'	56	PR/-	OVPress	N/A	Shut	Shut	Shut	N/A	N/A	None	N/A	Yes	1	M05-1075 Sh 2	Yes	SP/W	#Note 46, 49
26.#	RHR	*1E12F055B	12	OC	W	92'	56	PR/-	OVPress	N/A	Shut	Shut	Shut	N/A	N/A	None	N/A	Yes	1	M05-1075	Yes	SP/W	Note 45, 46 *Note 50
		1E12F433B	0.75	OC	W		56	GL/MAN	Man	N/A	Shut	Shut	Shut	N/A	N/A	None	N/A	Yes	1	Sh 2 & 4	No	W	
		1E12F432B	0.75	OC	W		56	GL/MAN	Man	N/A	Shut	Shut	Shut	N/A	N/A	None	N/A	Yes	1		No	SP/W	
		1E12F112B	2.0	OC	W		56	PR/-	OVPress	N/A	Shut	Shut	Shut	N/A	N/A	None	N/A	Yes	1		Yes		#Note 49
27.	RHR	1E12F025B	1.5	OC	W	17'	56	PR/-	OVPress	N/A	Shut	Shut	Shut	N/A	N/A	None	N/A	Yes	1	M05-1075 Sh 2	No	SP	Note 35, 48, 46
28.#	RCIC	1E51F031	6.0	OC	W	4'	56	GA/MO	Auto	RM	Shut	Shut	Open/Shut	As Is	Div. 1	V,X,B, F,E,R	*48	Yes	1	M05-1079 Sh 2	Yes	SP/W	*Notes 28, 30
		1E51F336	0.75	OC	W		56	GL/MAN	Man	N/A	Shut	Shut	Shut	N/A	N/A	None	N/A	Yes	1		No	SP/W	
		1E51F337	0.75	OC	W		56	GL/MAN	Man	N/A	Shut	Shut	Shut	N/A	N/A	None	N/A	Yes	1		No	W	#Note 49
29.#	RHR	1E12F101	1.5	OC	W	61'-1"	56	PR/-	OVPress	N/A	Shut	Shut	Shut	N/A	N/A	None	N/A	Yes	1	M05-1075 Sh 3	Yes	SP/W	#Notes 46, 49
30.#	RHR	1E12F025C	1.5	OC	W	24'	56	PR/-	OVPress	N/A	Shut	Shut	Shut	N/A	N/A	None	N/A	Yes	1	M05-1075 Sh 3	Yes	SP/W	#Notes 46, 49
31.#	RHR	1E12F036	6	OC	W	35'-9"	56	PR/-	OVPress	N/A	Shut	Shut	Shut	N/A	N/A	None	N/A	Yes	1	M05-1075 Sh 2 & 4	Yes	SP/W	#Notes 46, 49
		1E12F437	0.75	OC	W		56	GL/MAN	Man	N/A	Shut	Shut	Shut	N/A	N/A	None	N/A	Yes	1		No	W	
		1E12F436	0.75	OC	W		56	GL/MAN	Man	N/A	Shut	Shut	Shut	N/A	N/A	None	N/A	Yes	1		No	SP/W	
32.#	LPCS	1E21F001	20	OC	W	6'-6"	56	GA/MO	RM	Man	Open	Open	Open	As Is	Div. 1	None	N/A	Yes	1	6.2-143	Yes	SP*/W	#Note 49
		1E21F331	0.75	OC	W		56	GL/MAN	Man	N/A	Shut	Shut	Shut	N/A	N/A	None	N/A	Yes	1		No	SP/W	
		1E21F344	0.75	OC	W		56	GL/MAN	Man	N/A	Shut	Shut	Shut	N/A	N/A	None	N/A	Yes	1		No	W	
33.#	HPCS	1E22F023	10	OC	W	62'	56	GL/MO	Auto	RM	Shut	Shut	Shut	As Is	Div. 3	B,L	*68	Yes	1	6.2-144	Yes	SP/W	*Note 28
		1E22F014	1.0	OC	W	90'	56	PR/-	OVPress	N/A	Shut	Shut	Shut	N/A	N/A	None	N/A	Yes	1		Yes	SP/W	
		1E22F012	4.0	OC	W	76'	56	GA/MO	RM	Man	Shut	Shut	Open/Shut	As Is	Div. 3	None	N/A	Yes	1		Yes	SP/W	
		1E22F035	1.0	OC	W	76'	56	PR/-	OVPress	N/A	Shut	Shut	Shut	N/A	N/A	None	N/A	Yes	1		Yes	SP/W	#Notes 46, 49
		1E22F039	1.0	OC	W	94'-6"	56	PR/-	OVPress	N/A	Shut	Shut	Shut	N/A	N/A	None	N/A	Yes	1		Yes	SP/W	
		1E22F376	0.75	OC	W		56	GL/MAN	Man	N/A	Shut	Shut	Shut	N/A	N/A	None	N/A	Yes	1		No	SP/W	
34.#	Supp Pool	1SF004	12	OC	W	6'	*56	GA/MO	Auto	RM	Shut	Shut	Shut	As Is	Div. 1	B,L,R	*84	Yes	2	M05-1060	Yes	SP/W	*Note 28
	Clean-Up	1SF034	0.75	OC	W		56	GL/MAN	Man	N/A	Shut	Shut	Shut	N/A	N/A	None	N/A	Yes	1		Yes	SP/W	#Notes 49, 51
35.	HPCS	1E22F004	10	OC	W	1'-6"	55	GA/MO	RM	Man	Shut	Shut	Open/Shut	As Is	Div. 3	None	N/A	Yes	1	6.2-144	Yes	G	Note 39
		1E22F021	0.75	OC	W		55	GL/MAN	Man	N/A	Shut	Shut	Shut	N/A	N/A	None	N/A	Yes	1		Yes	G	Note 18
36.	LPCS	1E21F005	10	OC	W	2'	55	GA/MO	RM	Man	Shut	Shut	Open	As Is	Div. 1	None	N/A	Yes	1	6.2-143	Yes	G	Note 6
		1E21F013	0.75	OC	W		55	GL/MAN	Man	N/A	Shut	Shut	Shut	N/A	N/A	None	N/A	Yes	1		Yes	G	Note 39, 18
37.#	HPCS	1E22F015	20	OC	W	6'	56	GA/MO	RM	Man	Shut	Shut	Open/Shut	As Is	Div. 3	None	N/A	Yes	1	6.2-144	Yes	SP*/W	*Note 18 #Note 49
38.#	LPCS	1E21F018	2.0	OC	W	30'	56	PR/-	OVPress	N/A	Shut	Shut	Shut	N/A	N/A	None	N/A	Yes	1	6.2-143	Yes	SP/W	#Notes 46, 49
		1E21F031	1.5	OC	W	152'	56	PR/-	OVPress	N/A	Shut	Shut	Shut	N/A	N/A	None	N/A	Yes	1		Yes	SP/W	
39.	Spare																					CA	

TABLE 6.2-47
ISOLATION VALVE SUMMARY FOR LINE PENETRATING CONTAINMENT (Continued)

CONT. PENE. NO. NOTE 13	SYSTEM NAME	VALVE NUMBER	LINE SIZE (IN)	VALVE LOCA-TION	FLUID CON-TAINED	PIPE LENGTH-CONT. TO OUTER-MOST ISO VALV	NRC DESIGN CRITERIA	VALVE TYPE AND OPERATOR	PRIMARY MODE	SECON-DARY MODE	NORMAL NOTE 12	SHUT-DOWN	POST LOCA	POWER FAILURE	POWER SOURCE	CONT. ISOL. SIGNAL NOTE 8	CLOSURE TIME (SEC) NOTE 10	ENGNRD SAFETY FEATURE NOTE 15	THRU LINE LEAK CLASS NOTE 17	REFERENCE FIGURE/ DRAWING	TYPE C TEST	TYPE MEDIA	REMARKS
40.#	RCIC	1E51F019	2.0	OC	W	5'-6"	56	GL/MO	RM	Man	Shut	Shut	Shut	As Is	Div. 1	None	N/A	Yes	1	M05-1079 Sh 2	Yes	SP/W	#Notes 46, 49
		1E51F090	1.0	OC	W	42'-4"	56	PR/-	OVPres s	N/A	Shut	Shut	Shut	N/A	N/A	None	N/A	Yes	1		Yes	SP	
41.#	RCIC	1E51F040	12	OC	S		56	CH/-	RVFlow	N/A	Open/Shut	Open/Shut	Open/Shut	N/A	N/A	None	N/A	Yes	1	M05-1079 Sh 1	Yes	*W	*Note 31
		1E51F041	0.75	OC	S		56	GL/MAN	Man	N/A	Shut	Shut	Shut	N/A	N/A	None	N/A	Yes	1		No	*W	*Note 31 #Note 49
		1E51F068	12	OC	S	1'-6"	56	GA/MO	RM	Man	Open	Open	Open	As Is	Div. 1	None	N/A	Yes	1		Yes	*W	*Note 31
		+1E51F077	1.5	OC	A	12'	56	GL/MO	Auto	RM	Open	Open	Shut	As Is	Div. 1	**L,V	*21	Yes	1		No	SP	*Note 28, **Note 32, +Note 37
42.	RCIC	1E51F013	6.0	OC	W	7'	55	GA/MO	RM	Man	Shut	Shut	Open	As Is	Div. 1	None	N/A	Yes	1	M05-1079 Sh 2 M05-1075 Sh 2	Yes	G	Note 39
	RHR	1E12F023	4.0	OC	W	70'	55	GL/MO	Auto	RM	Shut	Shut	Shut	As Is	Div. 1	A,S,T,X,R	*39	Yes	1		Yes	G	*Note 28
	RCIC	1E51F034	0.75	OC	W		55	GL/MAN	Man	N/A	Shut	Shut	Shut	N/A	N/A	None	N/A	Yes	1		Yes	G	
		1E51F391	0.75	OC	W		55	GL/MAN	Man	N/A	Shut	Shut	Shut	N/A	N/A	None	N/A	Yes	1		Yes	G	Note 18
	RHR	1E12F061	0.75	OC	W		55	GL/MAN	Man	N/A	Shut	Shut	Shut	N/A	N/A	None	N/A	Yes	1		Yes	G	
43.	RCIC	1E51F063	8.0	ID	S		55	GA/MO	Auto	RM	Open	Open	Open/Shut	As Is	Div. 2	V,X,E,F	*41	Yes	2	M05-1075 Sh 1	Yes	CA/G	*Note 28
		1E51F064	8.0	OC	S	2'	55	GA/MO	Auto	RM	Open	Open	Open/Shut	As Is	Div. 1	V,X,E,F **B,**R	*41	Yes	2		Yes	G	*Note 28, **Note 30
		1E51F076	1.0	ID	S		55	GL/MO	Auto	RM	Shut	Shut	Shut	As Is	Div. 2	V,X,E,F	*14	Yes	2		Yes	CA/G	*Note 28
		1E51F399	0.75	ID	S		56	GL/MAN	Man	N/A	Shut	Shut	Shut	N/A	N/A	None	N/A	Yes	1		Yes	CA/G	
		1E51F072	0.75	OC	S		55	GL/MAN	Man	N/A	Shut	Shut	Shut	N/A	N/A	None	N/A	Yes	1		Yes	G	
		1E51F401	0.75	OC	S		55	GL/MAN	Man	N/A	Shut	Shut	Shut	N/A	N/A	None	N/A	Yes	1		Yes	G	
44.	RCIC	1E51F078	3.0	OC	A	3'-6"	56	GA/MO	Auto	RM	Open	Open	Shut	As Is	Div. 2	L,V*	*27	Yes	2	M05-1075 Sh 1	Yes	CA/G	*Notes 28, 32
		1E51F375	0.75	OC	A		56	GL/MAN	Man	N/A	Shut	Shut	Shut	N/A	N/A	None	N/A	Yes	1		Yes	CA/G	
		1E51F376	0.75	OC	A		56	GL/MAN	Man	N/A	Shut	Shut	Shut	N/A	N/A	None	N/A	Yes	1		Yes	G	
		1E51F082	0.75	OC	A		56	GL/MAN	Man	N/A	Shut	Shut	Shut	N/A	N/A	None	N/A	Yes	1		Yes	G	Note 45
		1E51F080	0.75	OC	A		56	GL/MAN	Man	N/A	Shut	Shut	Shut	N/A	N/A	None	N/A	Yes	1		Yes	G	
		1E51F083	0.75	OC	A		56	GL/MAN	Man	N/A	Shut	Shut	Shut	N/A	N/A	None	N/A	Yes	1		Yes	G	
		+1E51F077	1.5	OC	A	12'	56	GL/MO	Auto	RM	Open	Open	Shut	As Is	Div. 1	**L,V	*21	Yes	2		Yes	G	*Note 28, **Note 32 +Note 37
45.	Main Steam	1B21F016z	3.0	ID	S		55	GA/MO	Auto	RM	Open	Open	Shut	As	Div. 2	C,D,E,G,H, J,U,X,F,R	*50	Yes	2	M05-1002 Sh 1 & 2	Yes	CA/G	*Note 28
		1B21F019	3.0	OC	S	1'-6"	55	GA/MO	Auto	RM	Open	Open	Shut	As Is	Div. 1	C,D,E,G,H, J,U,X,F,R	*50	Yes	2		Yes	G	*Note 28
		1B21F017	0.75	OC	S		55	GL/MAN	Man	N/A	Shut	Shut	Shut	N/A	N/A	None	N/A	Yes	1		Yes	G	

TABLE 6.2-47
ISOLATION VALVE SUMMARY FOR LINE PENETRATING CONTAINMENT (Continued)

CONT. PENE. NO. NOTE 13	SYSTEM NAME	VALVE NUMBER	LINE SIZE (IN)	VALVE LOCA-TION	FLUID CON-TAINED	PIPE LENGTH-CONT. TO OUTER-MOST ISO VALV	NRC DESIGN CRITERIA	VALVE TYPE AND OPERATOR	PRIMARY MODE	SECON-DARY MODE	NORMAL NOTE 12	SHUT-DOWN	POST LOCA	POWER FAILURE	POWER SOURCE	CONT. ISOL. SIGNAL NOTE 8	CLOSURE TIME (SEC) NOTE 10	ENGNRD SAFETY FEATURE NOTE 15	THRU LINE LEAK CLASS NOTE 17	REFERENCE FIGURE/ DRAWING	TYPE C TEST	TYPE MEDIA	REMARKS
46.	Compont. Cooling	1CC049	10	OC	W	2'	56	GA/MO	Auto	RM	Open	Open/Shut	Shut	As Is	Div. 1	B,L,R	*84	No	2	M05-1032 Sh 3	Yes	G	*Note 28
		1CC050	6.0	ID	W		57	GA/MO	Auto	RM	Open	Open/Shut	Shut	As Is	Div. 2	B,L,R	*45	No	2		Yes	CA/G	*Note 28
		1CC127	8.0	IC	W	2'	57	GA/MO	Auto	RM	Open	Open/Shut	Shut	As Is	Div. 2	B,L,R	*64	No	2		Yes	CA/G	*Note 28
		1CC164	0.75	OC	W		56	GA/MAN	Man	N/A	Shut	Shut	Shut	N/A	N/A	None	N/A	No	1		Yes	G	
		1CC266	0.75	IC	W		56	GL/MAN	Man	N/A	Shut	Shut	Shut	N/A	N/A	None	N/A	No	2		Yes	CA/G	
47.	Compont. Cooling	1CC053	6.0	ID	W		57	GA/MO	Auto	RM	Open	Open/Shut	Shut	As Is	Div. 2	B,L,R	*45	No	2	M05-1032 Sh 3	Yes	CA/G	*Note 28
		1CC054	10	OC	W	1'-6"	57	GA/MO	Auto	RM	Open	Open/Shut	Shut	As Is	Div. 1	B,L,R	*84	No	2		Yes	G	*Note 28
		1CC060	8.0	IC	W		57	GA/MO	Auto	RM	Open	Open/Shut	Shut	As Is	Div. 2	B,L,R	*64	No	2		Yes	CA/G	*Note 28
		1CC165	0.75	OC	W		57	GL/MAN	Man	N/A	Shut	Shut	Shut	N/A	N/A	None	N/A	No	1		Yes	G	
48.	S/D Serv Wtr	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	M05-1052 Sh 5	N/A	N/A	
49.	Breathing Air	ORA026	1.0	OC	A	1'	56	GA/AO	**Auto	**RM	Shut	Shut	Shut	Shut	Div. 1	B,L,R	N/A*	No	2	6.2-123 Sh 3	Yes	G	*Note 29 **Note 43
		ORA027	1.0	IC	A		56	GA/AO	**Auto	**RM	Shut	Shut	Shut	Shut	Div. 2	B,L,R	N/A*	No	2		Yes	CA/G	*Note 29 **Note 43
50.	Conden. Storage	OMC009	4.0	OC	W	1'-6"	56	GA/MO	Auto	RM	Open	Shut	Shut	As Is	Div. 1	B,L,R	*58	No	2	6.2-123 Sh 3	Yes	G	*Note 28
		OMC010	4.0	IC	W		56	GA/MO	Auto	RM	Open	Shut	Shut	As Is	Div. 2	B,L,R	*58	No	2		Yes	G	*Note 28
		1MC090	0.75	IC	W		56	PR/-	OVPress	N/A	Shut	Shut	Shut	N/A	N/A	None	N/A	No	2		Yes	G	
		1MC011	0.75	OC	W		56	GL/MAN	Man	N/A	Shut	Shut	Shut	N/A	N/A	None	N/A	No	1		Yes	G	
51.	Spare																					CA	
52.	Fuel Pool Cooling and Cleanup	1FC036	8.0	OC	W	1'-6"	56	GA/MO	Auto	RM	Open	Open/Shut	Shut	As Is	Div. 1	B,L,R	*75	No	1	M05-1037 Sh 1	Yes	G	*Note 28
		1FC037	8.0	IC	W		56	GA/MO	Auto	RM	Open	Open/Shut	Shut	As Is	Div. 2	B,L,R	*75	No	1		Yes	CA/G	*Note 28
		1FC180	0.75	OC	W		56	GL/MAN	Man	N/A	Shut	Shut	Shut	N/A	N/A	None	N/A	No	1		Yes	G	
53.	Fuel Pool Cooling and Cleanup	1FC007	10	IC	W		56	GA/MO	Auto	RM	Open	Open/Shut	Shut	As Is	Div. 2	B.L,R	*66	No	1	M05-1037 Sh 1	Yes	CA/G	*Note 28
		1FC008	10	OC	W	1'-6"	56	GA/MO	Auto	RM	Open	Open/Shut	Shut	As Is	Div. 1	B,L,R	*66	No	1		Yes	G	*Note 28
		1FC181	0.75	OC	W		56	GL/MAN	Man	N/A	Shut	Shut	Shut	N/A	N/A	None	N/A	No	1		Yes	G	
54.	Spare																					CA	
55.	Spare																					CA	
56.	Fire Prot	1FP052	10	IC	W		56	GA/MO	Auto	RM	Open	Open	Shut	As Is	Div. 2	B,L,R	**87	No	2	M05-1039 Sh 9	Yes	*CA/G	*Note 18 **Note 28
		1FP051	10	OC	W	1'-6"	56	GA/MO*	Man	N/A	Shut	Shut	Shut	As Is	Div. 1	None	N/A	No	2		Yes	G	*Note 44
		1FP199	0.75	OC	W		56	GL/MAN	Man	N/A	Shut	Shut	Shut	N/A	N/A	None	N/A	No	1		Yes	G	
57.	Inst. Air	1IA005	3.0	OC	A	1'-6"	56	CV/AO	Auto	RM	Open	Open/Shut	Shut	Shut	Div. 1	U	*36	No	2	M05-1040 Sh 3	Yes	G	*Note 28
		1IA006	3.0	IC	A		56	CV/AO	Auto	RM	Open	Open/Shut	Shut	Shut	Div. 2	U	*36	No	2		Yes	CA/G	*Note 28
		1IA039	0.75	OC	A		56	GL/MAN	Man	N/A	Shut	Shut	Shut	N/A	N/A	None	N/A	No	1		Yes	G	
		1IA175	0.5	IC	A	1'-6"	56	CH/-	RevFlow	N/A	Open	Open/Shut	Shut	N/A	N/A	None	N/A	No	1		Yes	CA/G	
58.	Inst. Air	1IA042B	1.0	ID	A		56	CH/-	RevFlow	N/A	Open/Shut	Shut	Shut/Open	N/A	N/A	None	N/A	Yes	2	M05-1040 Sh 5	Yes	CA/G	
		1IA012B	1.0	IC	A		56	GL/MO	Auto	RM	Open	Open	Shut	As Is	Div. 1	L,B,R	*25	Yes	2		Yes	CA/G	* Note 28
		1IA012A	1.0	OC	A		56	GL/MO	RM	Man	Shut	Shut	Open	As Is	Div. 1	None	N/A	Yes	2		Yes	G	*Note 6

TABLE 6.2-47
ISOLATION VALVE SUMMARY FOR LINE PENETRATING CONTAINMENT (Continued)

CONT. PENE. NO. NOTE 13	SYSTEM NAME	VALVE NUMBER	LINE SIZE (IN)	VALVE LOCA-TION	FLUID CON-TAINED	PIPE LENGTH-CONT. TO OUTER-MOST ISO VALV	NRC DESIGN CRITERIA	VALVE TYPE AND OPERATOR	PRIMARY MODE	SECON-DARY MODE	NORMAL NOTE 12	SHUT-DOWN	POST LOCA	POWER FAILURE	POWER SOURCE	CONT. ISOL. SIGNAL NOTE 8	CLOSURE TIME (SEC) NOTE 10	ENGNRD SAFETY FEATURE NOTE 15	THRU LINE LEAK CLASS NOTE 17	REFERENCE FIGURE/ DRAWING	TYPE C TEST	TYPE MEDIA	REMARKS
59.	Service Air	1SA030	3.0	IC	A		56	CV/AO	Auto	RM	Open	Open/Shut	Shut	Shut	Div. 2	B,L,R	*16	No	2	M05-1048 Sh 6	Yes	CA/G	*Note 28
		1SA029	3.0	OC	A	1'-6"	56	CV/AO	Auto	RM	Open	Open/Shut	Shut	Shut	Div. 1	B,L,R	*16	No	2		Yes	G	*Note 28
		1SA046	0.75	OC	A		56	GL/MAN	Man	N/A	Shut	Shut	Shut	N/A	N/A	None	N/A	No	1		Yes	G	
60.	RWCU	1G33F001	6.0	ID	W		55	GA/MO	Auto	RM	Open	Open	Shut	As Is	Div. 2	B,F,N,1,2, E,X,R,6,7	*20	No	1	M05-1076 Sh 4	Yes	**G	*Note 28 **Note 25
		1G33F004	6.0	OC	W	1'-6"	55	GA/MO	Auto	RM	Open	Open	Shut	As Is	Div. 1	B,F,N,1,2, E,X,R,6,7	*20	No	1		Yes	**G	*Note 28 **Note 25
		1G33F002	0.75	OC	W		55	GL/MAN	Man	N/A	Shut	Shut	Shut	N/A	N/A	None	N/A	No	1		Yes	G	
61.	RWCU	1G33F053	4.0	IC	W		57	GA/MO	Auto	RM	Open	Open	Shut	As Is	Div. 2	B,F,N,1, 2,E,X,R,7	*21	No	1	M05-1076 Sh 4	Yes	CA/G	*Note 28
		1G33F054	4.0	OC	W	1'-6"	57	GA/MO	Auto	RM	Open	Open	Shut	As Is	Div. 1	B,F,N,1, 2,E,X,R,7	*21	No	1		Yes	G	*Note 28
		1G33F061	0.75	OC	W		57	GL/MAN	Man	N/A	Shut	Shut	Shut	N/A	N/A	None	N/A	No	1		Yes	G	
62.	Combust. Gas Contr	1HG008	2.0	OC	A	3'-6"	56	BF/MO	Auto	RM	Shut	Shut	Open/Shut Shut	As Is	Div. 2	B,L,R	*117	Yes	2	6.2-146 Sh 1 & 2	Yes	G	Notes 19, 38 *Note 28
	CLOC	1HG019	0.75	OC	A		56	GL/MAN	Man	Man	N/A	Shut		N/A	N/A	None	N/A	Yes	1		Yes	CA/G	Note 38
63.	CRD	1C11F122	2.0	IC	W		55	CH/-	RevFlow	N/A	Open	Open	Open/Shut	N/A	N/A	None	N/A	No	2	M05-1078 Sh 1	Yes	CA/G	
		1C11F083	2.0	OC	W	2'-6"	55	GL/MO	RM	Man	Open	Open	Open/Shut	As Is	Div. 1	None	N/A	No	2		Yes	G	Note 18
		1C11F128	0.75	OC	W		55	GL/MAN	Man	N/A	Shut	Shut	Shut	N/A	N/A	None	N/A	No	1		Yes	G	
64.	RWCU	1G33F040	4.0	IC	W		57	GL/MO	Auto	RM	Open	Open	Shut	As Is	Div. 2	B,F,N,1, 2,E,X,R,7	*21	No	1	M05-1076 Sh 4	Yes	CA/G	*Note 28
		1G33F039	4.0	OC	W	1'-6"	57	GL/MO	Auto	RM	Open	Open	Shut	As Is	Div. 1	B,F,N,1, 2,E,X,R,7	*21	No	1		Yes	G	*Note 28
		1G33F055	0.75	OC	W		57	GL/MAN	Man	N/A	Shut	Shut	Shut	N/A	N/A	None	N/A	No	1		Yes	G	
65.	Radwaste	1WX019	2.0	IC	W		56	PL/AO	Auto	RM	Open/Shut	Shut	Shut	Shut	Div. 2	B,L,R	*2	No	2	M05-1010 Sh 2	Yes	CA/G	*Note 28
		1WX020	2.0	OC	W	2'-6"	56	PL/AO	Auto	RM	Open/Shut	Shut	Shut	Shut	Div. 1	B,L,R	*2	No	2		Yes	G	*Note 28
		1WX080	0.75	IC	W		56	PR/-	OVPress	N/A	Shut	Shut	Shut	N/A	N/A	N/A	N/A	No	2		Yes	G	
66.	Spare																					CA	
67.	Cont. Pressur-	1SA129	0.75	OC	A		56	GL/MAN	Man	N/A	Shut	Shut	Shut	N/A	N/A	None	N/A	No	1	M05-1048 Sh 9	Yes	G	
	ization	1SA127	6.0	OC	A		*	BF/MAN	Man	N/A	*	*	*	N/A	N/A	*	N/A	No	*		No	CA	*Note 27
68.	Process Sampling	1PS017	0.5	OC	W	1'-6"	56	GA/SO	Auto	RM	Shut	Shut	Open/Shut	Shut	Div. 1	B,L,R	*N/A	Yes	2	M05-1045-12	Yes	G	*Note 29
		1PS016	0.5	IC	W	1'-5"	56	GA/SO	Auto	RM	Shut	Shut	Open/Shut	Shut	Div. 2	B,L,R	*N/A	Yes	2		Yes	CA/G	*Note 29
	Post Accident	1PS023	0.5	OC	W	2'-3"	56	GA/SO	Auto	RM	Shut	Shut	Open/Shut	Shut	Div. 1	B,L,R	*N/A	Yes	2		Yes	G	*Note 29
	Sampling	1PS022	0.5	IC	W	1'-5"	56	GA/SO	Auto	RM	Shut	Shut	Open/Shut	Shut	Div. 2	B,L,R	*N/A	Yes	2		Yes	CA/G	*Note 29
		1PS035	0.75	OC	A	2'-0"	56	GA/SO	Auto	RM	Shut	Shut	Open/Shut	Shut	Div. 1	B,L,R	*N/A	Yes	2		Yes	G	*Note 29
		1PS034	0.75	IC	A	1'-5"	56	GA/SO	Auto	RM	Shut	Shut	Open/Shut	Shut	Div. 2	B,L,R	*N/A	Yes	2		Yes	CA/G	*Note 29
		1PS055	0.5	OC	A	3'-3"	56	GA/SO	Auto	RM	Shut	Shut	Open/Shut	Shut	Div. 1	B,L,R	*N/A	Yes	2		Yes	G	*Note 29
		1PS056	0.5	IC	A	2'-3"	56	GA/SO	Auto	RM	Shut	Shut	Open/Shut	Shut	Div. 2	B,L,R	*N/A	Yes	2		Yes	CA/G	*Note 29
		1PS070	0.75	IC	W	2'-3"	56	GA/SO	Auto	RM	Shut	Shut	Open/Shut	Shut	Div. 2	B,L,R	*N/A	Yes	2		Yes	CA/G	*Note 29
		1PS069	0.75	OC	W	2'-6"	56	GA/SO	Auto	RM	Shut	Shut	Open/Shut	Shut	Div. 1	B,L,R	*N/A	Yes	2		Yes	G	*Note 29
69.	Equip. Drain	1RE021	3.0	IC	W		56	CV/AO	Auto	RM	Open/Shut	Open/Shut	Shut	Shut	Div. 2	B,L,R	*16	No	1	6.2-123 Sh 3	Yes	CA/G	*Note 28
		1RE022	3.0	OC	W	1'-6"	56	CV/AO	Auto	RM	Open/Shut	Open/Shut	Shut	Shut	Div. 1	B,L,R	*16	No	1		Yes	G	*Note 28

TABLE 6.2-47
ISOLATION VALVE SUMMARY FOR LINE PENETRATING CONTAINMENT (Continued)

CONT. PENE. NO. NOTE 13	SYSTEM NAME	VALVE NUMBER	LINE SIZE (IN)	VALVE LOCA-TION	FLUID CON-TAINED	PIPE LENGTH-CONT. TO OUTER-MOST ISO VALV	NRC DESIGN CRITERIA	VALVE TYPE AND OPERATOR	PRIMARY MODE	SECON-DARY MODE	NORMAL NOTE 12	SHUT-DOWN	POST LOCA	POWER FAILURE	POWER SOURCE	CONT. ISOL. SIGNAL NOTE 8	CLOSURE TIME (SEC) NOTE 10	ENGNRD SAFETY FEATURE NOTE 15	THRU LINE LEAK CLASS NOTE 17	REFERENCE FIGURE/ DRAWING	TYPE C TEST	TYPE MEDIA	REMARKS
70.	Floor Drain	1RF021	3.0	IC	W		56	CV/AO	Auto	RM	Open/Shut	Open/Shut	Shut	Shut	Div. 2	B,L,R	*16	No	1	6.2-123 Sh 3	Yes	CA/G	*Note 28
		1RF022	3.0	OC	W	1'-6"	56	CV/AO	Auto	RM	Open/Shut	Open/Shut	Shut	Shut	Div. 1	B,L,R	*16	No	1		Yes	G	*Note 28
71.	H2 Recom-biner	1HG001	2.0	OC	A	3'	56	BF/MO	Auto	RM	Shut	Shut	Open/Shut	As Is	Div. 1	B,L,R	*117	Yes	2	6.2-146 Sh 1	Yes	G	Notes 19, 38 *Note 28
		1HG016	0.75	OC	A		56	GL/MAN	Man	N/A	Shut	Shut	Shut	N/A	N/A	None	N/A	Yes	1		Yes	CA/G	Note 38
	CLOC	1HG020	0.75	OC	A		56	GL/MAN	Man	N/A	Shut	Shut	Shut	N/A	N/A	None	N/A	Yes	1		No	CA	Note 38
72.	H2 Recom-biner	1HG004	2.0	OC	A	3'	56	BF/MO	Auto	RM	Shut	Shut	Open/Shut	As Is	Div. 1	B,L,R	*117	Yes	2	6.2-146 Sh 1	Yes	G	Notes 19, 38 *Note 28
		1HG017	0.75	OC	A		56	GL/MAN	Man	N/A	Shut	Shut	Shut	N/A	N/A	None	N/A	Yes	1		Yes	CA/G	Note 38
	CLOC	1HG021	0.75	OC	A		56	GL/MAN	Man	N/A	Shut	Shut	Shut	N/A	N/A	None	N/A	Yes	1		No	CA	Note 38
73.	Spare																					CA	
74.	RT Decon	1G33F428	0.75	OC	A		56	GL/MAN	Man	N/A	Shut	Shut	Shut	N/A	N/A	None	N/A	No	1	M05-1076 Sh 4	No	G	Note 48
75.	Spare																					CA	
76.#	RHR	1E12F030	1.5	OC	W	52'-6"	56	PR/-	OVPres s	N/A	Shut	Shut	Shut	N/A	N/A	None	N/A	No	1	M05-1075 Sh 2	Yes	SP/W	#Notes 46, 49
77.	Spare																					CA	
78.	Comp. Cooling Water	1CC074	4	ID	W		57	GA/MO	Man	N/A	Shut	Shut	Shut	As Is	Div. 2	L,U	*35	No	1	M05-1032 Sh 3	Yes	CA/G	*Notes 28, 44
		1CC073	4	OC	W	2'	57	GA/MO	Man	N/A	Shut	Shut	Shut	As Is	Div. 1	L,U	*35	No	1		Yes	G	*Notes 28, 44
		1CC170	0.75	OC	W		57	GL/MAN	Man	N/A	Shut	Shut	Shut	N/A	N/A	None	N/A	No	1		Yes	G	
79. #	Supp. Pool Cleanup	1SF001	10	OC	W	11'	56	GA/MO	Auto	RM	Shut	Shut	Shut	As Is	Div. 1	B,L,R	*114	No	1	M05-1060	Yes	W	*Note 28, Note 53
		1SF002	10	OC	W	2'-6"	56	GA/MO	Auto	RM	Shut	Shut	Shut	As Is	Div. 2	B,L,R	*126	No	1		Yes	SP/W	*Note 28, Note 53
		1SF023	0.75	OC	W		56	GL/MAN	Man	N/A	Shut	Shut	Shut	N/A	N/A	None	N/A	No	1		Yes	G	#Note 52
80.	Spare																					CA	
81.	Fire Prot.	1FP050	6	IC	W		56	GA/MO	Auto	RM	Open	Open	Shut	As Is	Div. 2	B,L,R	*58	No	2	M05-1039 Sh 9	Yes	*CA/G	*Notes 18, 28
		1FP092	6	OC	W	1'-6"	56	GA/MO	Auto	RM	Open	Open	Shut	As Is	Div. 1	B,L,R	*58	No	2		Yes	G	*Note 28
		1FP201	0.75	OC	W		56	GL/MAN	Man	N/A	Shut	Shut	Shut	N/A	N/A	None	N/A	No	1		Yes	G	
82.	Fire Prot.	1FP053	10	IC	W		56	GA/MO	Auto	RM	Open	Open	Shut	As Is	Div. 2	B,L,R	*68	No	2	MO5-1039 Sh 9	Yes	*CA/G	*Notes 18, 28
		1FP054	10	OC	W	1'-6"	56	GA/MO**	Man	N/A	Shut	Shut	Shut	As Is	Div. 1	None	N/A	No	2		Yes	G	**Note 44
		1FP200	0.75	OC	W		56	GL/MAN	Man	N/A	Shut	Shut	Shut	N/A	N/A	None	N/A	No	1		Yes	G	
83.	Spare																					CA	
84.	Spare																					CA	
85.	Cycled Cond.	1CY017	6	IC	W		56	GA/MO	Auto	RM	Open	Open	Shut	As Is	Div. 2	B,L,R	*75	No	2	M05-1012 Sh 6	Yes	CA/G	*Note 28
		1CY016	6	OC	W	1'-6"	56	GA/MO	Auto	RM	Open	Open	Shut	As Is	Div. 1	B,L,R	*75	No	2		Yes	G	*Note 28
		1CY019	0.75	OC	W		56	GL/MAN	Man	N/A	Shut	Shut	Shut	N/A	N/A	None	N/A	No	1		Yes	G	
86.	RWCU	1G33F028	4	IC	W		57	GA/MO	Auto	RM	Shut	Shut	Shut	As Is	Div. 2	B,F,N,1 2,E,X,R,7	*24	No	2	M05-1076 Sh 4	Yes	CA/G	*Note 28
		1G33F034	4	OC	W	1'-6"	57	GA/MO	Auto	RM	Shut	Shut	Shut	As Is	Div. 1	B,F,N,1 2,E,X,R,7	*24	No	2		Yes	G	*Note 28
		1G33F070	0.75	OC	W		57	GL/MAN	Man	N/A	Shut	Shut	Shut	N/A	N/A	None	N/A	No	1		Yes	G	
87.	RHR	1E12F025A	1.5	OC	W	36'	56	PR/-	OVPres s	N/A	Shut	Shut	Shut	N/A	N/A	None	N/A	Yes	1	M05-1075 Sh 1	No	SP	Notes 35,46,48
88.	Comp. Cooling Water	1CC071	4	ID	W		57	GA/MO	Man	N/A	Shut	Shut	Shut	As Is	Div. 2	L,U	*35	Yes	1	M05-1032 Sh 3	Yes	CA/G	*Notes 28, 44
		1CC072	4	OC	W	2'	57	GA/MO	Man	N/A	Shut	Shut	Shut	As Is	Div. 1	L,U	*35	Yes	1		Yes	G	*Notes 28, 44
		1CC171	0.75	OC	W		57	GL/MAN	Man	N/A	Shut	Shut	Shut	N/A	N/A	None	N/A	Yes	1		Yes	G	
89.	RHR HX Shell Vent																					CA	Note 41

TABLE 6.2-47
ISOLATION VALVE SUMMARY FOR LINE PENETRATING CONTAINMENT (Continued)

CONT. PENE. NO. NOTE 13	SYSTEM NAME	VALVE NUMBER	LINE SIZE (IN)	VALVE LOCA-TION	FLUID CON-TAINED	PIPE LENGTH-CONT. TO OUTER-MOST ISO VALV	NRC DESIGN CRITERIA	VALVE TYPE AND OPERATOR	PRIMARY MODE	SECON-DARY MODE	NORMAL NOTE 12	SHUT-DOWN	POST LOCA	POWER FAILURE	POWER SOURCE	CONT. ISOL. SIGNAL NOTE 8	CLOSURE TIME (SEC) NOTE 10	ENGNRD SAFETY FEATURE NOTE 15	THRU LINE LEAK CLASS NOTE 17	REFERENCE FIGURE/ DRAWING	TYPE C TEST	TYPE MEDIA	REMARKS
101.	Cont. HVAC	1VR001A	36	OC	A	2'	56	BF/AO	**Auto	**RM	Shut	Open	Shut	Shut	Div. 1	B,L,M,Z,5,R	*4	Yes	2	M05-1111 Sh 1	Yes	G	*Note 28 **Note 43
		1VR002A	4	OC	A	10'	56	GL/MO	*Auto	RM	Shut	Shut	Open/Shut	As Is	Div. 1	P	**≤10	Yes	1		Yes	G	*Note 44 **Note 42
		1VR001B	36	IC	A		56	BF/AO	**Auto	**RM	Shut	Open	Shut	Shut	Div. 2	B,L,M,Z,5,R	*4	Yes	2		Yes	CA/G	*Note 28 **Note 43
		1VR002B	4	IC	A		56	GL/MO	*Auto	RM	Shut	Shut	Open/Shut	As Is	Div. 2	P	**≤10	Yes	2		Yes	CA/G	*Note 44 **Note 42
		1VR003	0.75	OC	A		56	GL/MAN	Man	N/A	Shut	Shut	Shut	N/A	N/A	None	N/A	Yes	1		Yes	G	
102.	Drywell Purge	1VQ004A	36	OC	A	2'-6"	56	BF/AO	**Auto	**RM	Shut	Open/Shut	Shut	Shut	Div. 1	B,L,M,Z,5,R	*≤6	Yes	2	M05-1110 Sh 2	Yes	G	*Note 28 **Note 43
		1VQ006A	4	OC	A	4'	56	GL/MO	*Auto	RM	Shut	Open/Shut	Open/Shut	As Is	Div. 1	P	**≤10	Yes	2		Yes	G	*Note 44 **Note 42
		1VQ004B	36	IC	A		56	BF/AO	**Auto	**RM	Shut	Open/Shut	Shut	Shut	Div. 2	B,L,M,Z,5,R	*≤6	Yes	2		Yes	CA/G	*Note 28 **Note 43
		1VQ006B	4	IC	A		56	GL/MO	*Auto	RM	Shut	Open/Shut	Open/Shut	As Is	Div. 2	P	**≤10	Yes	2		Yes	CA/G	*Note 44 **Note 42
		1VQ007	0.75	OC	A		56	GL/MAN	Man	N/A	Shut	Shut	Shut	N/A	N/A	None	N/A	Yes	1		Yes	G	
103.	Plant Chill	1WO001A	6	OC	W	1'-2"	57	GA/MO	Auto	RM	Open	Open	Shut	As Is	Div. 1	+L,U	*44	No	2	M05-1117 Sh 19	Yes	G	*Note 28 +Note 47
	Water	1WO001B	6	IC	W		57	GA/MO	Auto	RM	Open	Open	Shut	As Is	Div. 2	+L,U	**44	No	2		Yes	CA*/G	*Note 18 **Note 28 +Note 47
104.	Plant Chill	1WO002A	6	OC	W	1'-2"	57	GA/MO	Auto	RM	Open	Open	Shut	As Is	Div. 1	+L,U	*44	No	2	M05-1117 Sh 19	Yes	G	*Note 28 +Note 47
	Water	1WO002B	6	IC	W		57	GA/MO	Auto	RM	Open	Open	Shut	As Is	Div. 2	+L,U	**44	No	2		Yes	CA*/G	*Note 18 **Note 28 +Note 47
105.	Spare																					CA	
106.	Contin- uous	1VR007B	12	IC	A	3'-6"	56	BF/AO	Auto	RM	Open	Open	Shut	Shut	Div. 2	B,L,M,Z,5,R B,L,M,Z,5,R	*6	No	2	M05-1111 Sh 4	Yes	CA/G	*Note 28
	Cnmt. Purge	1VR007A	12	OC	A	4'-0"	56	BF/AO	Auto	RM	Open	Open	Shut	Shut	Div. 1	None	*6	No	2		Yes	G	*Note 28
		1VR011	0.75	OC	A		56	GL/MAN	Man	N/A	Shut	Shut	Shut	N/A	N/A		N/A	No	1		Yes	G	
107.	DW Bldg Chill Water	1VP004B	10	OC	W	3'-6"	56	GA/MO	Auto	RM	Open	Open/Shut	Shut	As Is	Div. 1	+L,U	*84	No	2	M05-1109 Sh 2	Yes	G	*Note 28 +Note 47
		1VP005B	10	ID	W		57	GA/MO	Auto	RM	Open	Open/Shut	Shut	As Is	Div. 2	+L,U	**84	No	2		Yes	CA*/G	*Note 18 **Note 28 +Note 47
		1VP044B	0.75	OC	W		56	GL/MAN	Man	N/A	Shut	Shut	Shut	N/A	N/A	None	N/A	No	1		Yes	G	
		1VP077D	0.75	IC	W		56	GL/MAN	Man	N/A	Shut	Shut	Shut	N/A	N/A	None	N/A	No	1		Yes*	CA*/G	*Note 18
		1VP023B	0.75	IC	W		56	PR/-	OVPres s	N/A	Shut	Shut	Shut	N/A	N/A	None	N/A	No	1		Yes	CA*/G	*Note 18
108.	DW Bldg Chill Water	1VP014B	10	ID	W		57	GA/MO	Auto	RM	Open	Open/Shut	Shut	As Is	Div. 2	+L,U	**84	No	2	M05-1109 Sh 2	Yes	CA*/G	*Note 18 **Note 28 +Note 47
		1VP015B	10	OC	W	3'-6"	56	GA/MO	Auto	RM	Open	Open/Shut	Shut	As Is	Div. 1	+L,U	*84	No	2		Yes	G	*Note 28 +Note 47
		1VP047B	0.75	OC	W		56	GL/MAN	Man	N/A	Shut	Shut	Shut	N/A	N/A	None	N/A	No	1		Yes	G	
		1VP077B	0.75	IC	W		56	GL/MAN	Man	N/A	Shut	Shut	Shut	N/A	N/A	None	N/A	No	1		Yes	CA*/G	*Note 18
		1VP027B	0.75	IC	W		56	PR/-	OVPres s	N/A	Shut	Shut	Shut	N/A	N/A	None	N/A	No	1		Yes	CA*/G	*Note 18

TABLE 6.2-47
ISOLATION VALVE SUMMARY FOR LINE PENETRATING CONTAINMENT (Continued)

CONT. PENE. NO. NOTE 13	SYSTEM NAME	VALVE NUMBER	LINE SIZE (IN)	VALVE LOCA-TION	FLUID CON-TAINED	PIPE LENGTH-CONT. TO OUTER-MOST ISO VALV	NRC DESIGN CRITERIA	VALVE TYPE AND OPERATOR	PRIMARY MODE	SECON-DARY MODE	NORMAL NOTE 12	SHUT-DOWN	POST LOCA	POWER FAILURE	POWER SOURCE	CONT. ISOL. SIGNAL NOTE 8	CLOSURE TIME (SEC) NOTE 10	ENGNRD SAFETY FEATURE NOTE 15	THRU LINE LEAK CLASS NOTE 17	REFERENCE FIGURE/ DRAWING	TYPE C TEST	TYPE MEDIA	REMARKS
109.	DW Bldg. Chill Water	1VP004A	10	OC	W	3'-6"	56	GA/MO	Auto	RM	Open	Open/Shut	Shut	As Is	Div. 1	+L,U	*84	No	2	M05-1109 Sh 1	Yes	G	*Note 28 +Note 47
		1VP005A	10	ID	W		57	GA/MO	Auto	RM	Open	Open/Shut	Shut	As Is	Div. 2	+L,U	**84	No	2		Yes	CA*/G	*Note 18 **Note 28 +Note 47
		1VP044A	0.75	OC	W		56	GL/MAN	Man	N/A	Shut	Shut	Shut	N/A	N/A	None	N/A	No	1		Yes	G	
		1VP077C	0.75	IC	W		56	GL/MAN	Man	N/A	Shut	Shut	Shut	N/A	N/A	None	N/A	No	1		Yes	CA*/G	*Note 18
		1VP023A	0.75	IC	W		56	PR/-	OVPress	N/A	Shut	Shut	Shut	N/A	N/A	None	N/A	No	1		Yes	CA*/G	*Note 18
110.	DW Bldg. Chill Water	1VP014A	10	ID	W		57	GA/MO	Auto	RM	Open	Open/Shut	Shut	As Is	Div. 2	+L,U	**84	No	2	M05-1109 Sh 1	Yes	CA*/G	*Note 18 **Note 28 +Note 47
		1VP015A	10	OC	W	3'-6"	56	GA/MO	Auto	RM	Open	Open/Shut	Shut	As Is	Div. 1	+L,U	*84	No	2		Yes	G	*Note 28 +Note 47
		1VP047A	0.75	OC	W		56	GL/MAN	Man	N/A	Shut	Shut	Shut	N/A	N/A	None	N/A	No	1		Yes	G	
		1VP077A	0.75	IC	W		56	GL/MAN	Man	N/A	Shut	Shut	Shut	N/A	N/A	None	N/A	No	1		Yes	CA*/G	*Note 18
		1VP027A	0.75	IC	W		56	PR/-	OVPress	N/A	Shut	Shut	Shut	N/A	N/A	None	N/A	No	1		Yes	CA*/G	*Note 18
111.	Spare																					CA	
112.	Spare																					CA	
113.	Contin-uous Cnmt. Purge	1VR006A	12	OC	A	1'-6"	56	BF/AO	Auto	RM	Open	Open	Shut	Shut	Div. 1	B,L,M,Z,5,R	*6	No	2	M05-1111 Sh 4	Yes	G	*Note 28
		1VR006B	12	IC	A	7'	56	BF/AO	Auto	RM	Open	Open	Shut	Shut	Div. 2	B,L,M,Z,5,R	*6	No	2		Yes	CA/G	*Note 28
		1VR012	0.75	OC	A		56	GL/MAN	Man	N/A	Shut	Shut	Shut	N/A	N/A	None	N/A	No	1		Yes	G	
114.	Spare																					CA	
115.	Spare																					CA	
116.	Stby. Liquid Control																						*Note 26
150.	Cont. Monitoring	1CM003A	0.75	OC	A	7'	*56	EFCV/-	ExFlow	N/A	Open	Open	Open	N/A	N/A	None	N/A	Yes	1	3.6-1 Sh 39	No	CA	*Note 6
151.	Cont. Monitoring	1CM051	0.75	OC	A		*56	EFCV/-	ExFlow	N/A	Open	Open	Open	N/A	N/A	None	N/A	Yes	1	3.6-1 Sh 41, 55	No	CA	*Note 6
		1CM066	0.75	OC	W		*55	EFCV/-	ExFlow	N/A	Open	Open	Open	N/A	N/A	None	N/A	Yes	1		No	CA	*Note 6
		1CM072	0.5	OC	W		55	GL/MAN	Man	N/A	Shut	Shut	Shut	N/A	N/A	None	N/A	Yes	1		No	CA/G	Note 48
		1CM073	0.5	OC	W		55	GL/MAN	Man	N/A	Shut	Shut	Shut	N/A	N/A	None	N/A	Yes	1		No	G	Note 48
		1CM076	0.5	OC	W		56	GL/MAN	Man	N/A	Shut	Shut	Shut	N/A	N/A	None	N/A	Yes	1		No	CA/G	Note 48
152.	Cont. Monitoring Type A Test Instru-ments	1CM080A	0.75	OC	A	1.0"	56	GL/MAN	Man	N/A	Shut	Shut	Shut	N/A	N/A	None	N/A	No	1	3.6-1 Sh 41	Yes	G	Note 40
		1CM080B	0.75	OC	A	1.0"	56	GL/MAN	Man	N/A	Shut	Shut	Shut	N/A	N/A	None	N/A	No	1		Yes	G	Note 40
		1CM080C	0.75	OC	A	1.0"	56	GL/MAN	Man	N/A	Shut	Shut	Shut	N/A	N/A	None	N/A	No	1		Yes	G	Note 40
		1CM081A	0.75	IC	A		56	GL/MAN	Man	N/A	Shut	Shut	Shut	N/A	N/A	None	N/A	No	1		Yes	G	Note 40
		1CM081B	0.75	IC	A		56	GL/MAN	Man	N/A	Shut	Shut	Shut	N/A	N/A	None	N/A	No	1		Yes	G	Note 40
		1CM081C	0.75	IC	A		56	GL/MAN	Man	N/A	Shut	Shut	Shut	N/A	N/A	None	N/A	No	1		Yes	G	Note 40
153.	Cont. Monitoring	1CM022	0.75	OC	A		56	GA/SO	Auto	RM	Open	Open	Shut/Open	Shut	Div. 2	B,L,R	*N/A	Yes	1	3.6-1 Sh 40	Yes	G	*Note 29
		1CM023	0.75	IC	A		56	GA/SO	Auto	RM	Open	Open	Shut/Open	Shut	Div. 2	B,L,R	*N/A	Yes	1		Yes	CA/G	*Note 29
		1CM025	0.75	OC	A		56	GA/SO	Auto	RM	Open	Open	Shut/Open	Shut	Div. 2	B,L,R	*N/A	Yes	1		Yes	G	*Note 29
		1CM026	0.75	IC	A		56	GA/SO	Auto	RM	Open	Open	Shut/Open	Shut	Div. 2	B,L,R	*N/A	Yes	1		Yes	CA/G	*Note 29

TABLE 6.2-47
ISOLATION VALVE SUMMARY FOR LINE PENETRATING CONTAINMENT (Continued)

CONT. PENE. NO. NOTE 13	SYSTEM NAME	VALVE NUMBER	LINE SIZE (IN)	VALVE LOCA-TION	FLUID CON-TAINED	PIPE LENGTH-CONT. TO OUTER-MOST ISO VALV	NRC DESIGN CRITERIA	VALVE TYPE AND OPERATOR	PRIMARY MODE	SECON-DARY MODE	NORMAL NOTE 12	SHUT-DOWN	POST LOCA	POWER FAILURE	POWER SOURCE	CONT. ISOL. SIGNAL NOTE 8	CLOSURE TIME (SEC) NOTE 10	ENGNRD SAFETY FEATURE NOTE 15	THRU LINE LEAK CLASS NOTE 17	REFERENCE FIGURE/ DRAWING	TYPE C TEST	TYPE MEDIA	REMARKS
154.	Spare																					CA	
155.	Spare																					CA	
156.	SGTS	1VG056B	0.75	OC	A		*56	EFCV/-	ExFlow	N/A	Open	Open	Open	N/A	N/A	None	N/A	Yes	1	6.2-123 Sh 2	No	CA	*Note 6
157.	Cont. Monitoring	1CM002A	0.75	OC	A	4'	*56	EFCV/-	ExFlow	N/A	Open	Open	Open	N/A	N/A	None	N/A	Yes	1	3.6-1 Sh 39	No	CA	*Note 6
		1CM003B	0.75	OC	A		*56	EFCV/-	ExFlow	N/A	Open	Open	Open	N/A	N/A	None	N/A	Yes	1		No	CA	*Note 6
158.	Spare																					CA	
159.	Spare																					CA	
160.	Cont. Monitoring	1CM067	0.75	OC	W		*55	EFCV/-	ExFlow	N/A	Open	Open	Open	N/A	N/A	None	N/A	Yes	1	3.6-1 Sh 56	No	CA	*Note 6
		1CM074	0.5	OC	W		55	GL/MAN	Man	N/A	Shut	Shut	Shut	N/A	N/A	None	N/A	Yes	1		No	CA/G	Note 48
		1CM075	0.5	OC	W		55	GL/MAN	Man	N/A	Shut	Shut	Shut	N/A	N/A	None	N/A	Yes	1		No	G	Note 48
161.	Spare																					CA	
162.	Spare																					CA	
163.	Spare																					CA	
164.	Supp Pool Make-up	1SM010	0.75	OC	A		*56	EFCV/-	ExFlow	N/A	Open	Open	Open	N/A	N/A	None	N/A	Yes	1	M05-1069	No	CA	*Note 6
165.	Cont. Bldg. HVAC	1VR016A	0.75	OC	A		*56	EFCV/-	ExFlow	N/A	Open	Open	Open	N/A	N/A	None	N/A	Yes	1	6.2-123 Sh 2	No	CA	*Note 6
		1VR016B	0.75	OC	A		*56	EFCV/-	ExFlow	N/A	Open	Open	Open	N/A	N/A	None	N/A	Yes	1		No	CA	*Note 6
		1VR018A	0.75	OC	A		*56	EFCV/-	ExFlow	N/A	Open	Open	Open	N/A	N/A	None	N/A	Yes	1		No	CA	*Note 6
166.	H2 Re-combiner CLOC	1HG005	2.0	OC	A	4'	56	BF/MO	Auto	RM	Shut	Shut	Open/Shut	As Is	Div. 2	B,L,R	*117	Yes	2	6.2-146 Sh 1, 2	Yes	G	*Note 28 Note 38
		1HG018	0.75	OC	A		56	GL/MAN	Man	N/A	Shut	Shut	Shut	N/A	N/A	None	N/A	No	1		Yes	CA/G	Note 38
167.	SGTS	1VG057B	0.75	OC	A		*56	EFCV/-	ExFlow	N/A	Open	Open	Open	N/A	N/A	None	N/A	Yes	1	6.2-123 Sh 2	No	CA	*Note 6
168.	Cont Bldg HVAC	1VR018B	0.75	OC	A		*56	EFCV/-	ExFlow	N/A	Open	Open	Open	N/A	N/A	None	N/A	Yes	1	6.2-123 Sh 2	No	CA	*Note 6
169.	Cont. HVAC	1VR035	0.75	IC	A		56	GA/SO	Auto	RM	Open	Open/Shut	Shut	Shut	Div. 2	B,L,M,Z,5,R	N/A	No	2	6.2-123 Sh 3	Yes	CA*/G	*Note 18
		1VR036	0.75	OC	A		56	GA/SO	Auto	RM	Open	Open/Shut	Shut	Shut	Div. 1	B,L,M,Z,5,R	N/A	No	2		Yes	G	
		1VR040	0.75	IC	A		56	GA/SO	Auto	RM	Open	Open/Shut	Shut	Shut	Div. 2	B,L,M,Z,5,R	N/A	No	2		Yes	CA*/G	*Note 18
		1VR041	0.75	OC	A		56	GA/SO	Auto	RM	Open	Open/Shut	Shut	Shut	Div. 1	B,L,M,Z,5,R	N/A	No	2		Yes	G	
170.	Spare																					CA	
171.	Supp Pool Make-up	1SM009	0.75	OC	A	2'-6"	*56	EFCV/-	ExFlow	N/A	Open	Open	Open	N/A	N/A	None	N/A	Yes	1	M05-10693	No	CA	*Note 6
172.	RHR HX Shell Vent																					CA	Note 41
173.	Cont. Monitoring	1CM048	0.75	OC	A		56	GA/SO	Auto	RM	Open	Open	Shut/Open	Shut	Div. 1	B,L,R	*N/A	Yes	1	3.6-1 Sh 40	Yes	G	*Note 29
		1CM047	0.75	IC	A		56	GA/SO	Auto	RM	Open	Open	Shut/Open	Shut	Div. 1	B,L,R	*N/A	Yes	1		Yes	CA/G	*Note 29
		1CM011	0.75	OC	A		56	GA/SO	Auto	RM	Open	Open	Shut/Open	Shut	Div. 1	B,L,R	*N/A	Yes	1		Yes	G	*Note 29
		1CM012	0.75	IC	A		56	GA/SO	Auto	RM	Open	Open	Shut/Open	Shut	Div. 1	B,L,R	*N/A	Yes	1		Yes	CA/G	*Note 29
174.	Spare																					CA	
175.	Spare																					CA	
176.	Spare																					CA	
177.	RCIC	1E51F377B	0.75	OC	W	4'-6"	*56	EFCV/-	ExFlow	N/A	Open	Open	Open	N/A	N/A	None	N/A	Yes	1	6.2-123 Sh 2	No	SP	*Note 6
		1E51F437A	0.50	OC	W		56	GL/MAN	Man	N/A	Shut	Shut	Shut	N/A	NA	None	N/A	Yes	1		No	SP/W	Note 48
		1E51F437B	0.50	OC	W		56	GL/MAN	Man	N/A	Shut	Shut	Shut	N/A	NA	None	N/A	Yes	1		No	W	Note 48
178.	Spare																					CA	

TABLE 6.2-47
ISOLATION VALVE SUMMARY FOR LINE PENETRATING CONTAINMENT (Continued)

CONT. PENE. NO. NOTE 13	SYSTEM NAME	VALVE NUMBER	LINE SIZE (IN)	VALVE LOCA-TION	FLUID CON-TAINED	PIPE LENGTH-CONT. TO OUTER-MOST ISO VALV	NRC DESIGN CRITERIA	VALVE TYPE AND OPERATOR	PRIMARY MODE	SECON-DARY MODE	NORMAL NOTE 12	SHUT-DOWN	POST LOCA	POWER FAILURE	POWER SOURCE	CONT. ISOL. SIGNAL NOTE 8	CLOSURE TIME (SEC) NOTE 10	ENGNRD SAFETY FEATURE NOTE 15	THRU LINE LEAK CLASS NOTE 17	REFERENCE FIGURE/ DRAWING	TYPE C TEST	TYPE MEDIA	REMARKS
179.	HPCS	1E22F332	0.75	OC	W	3'	*56	EFCV/-	ExFlow	N/A	Open	Open	Open	N/A	N/A	None	N/A	Yes	1	M05-1069	No	SP	*Note 6
	Supp Pool Make-up	1SM011	0.75	OC	W		*56	EFCV/-	ExFlow	N/A	Open	Open	Open	N/A	N/A	None	N/A	Yes	1	6.2-123 Sh 2	No	SP	*Note 6
		1SM027A	0.50	OC	W		56	GL/MAN	Man	N/A	Shut	Shut	Shut	N/A	NA	None	N/A	Yes	1		No	SP/W	Note 48
		1SM027B	0.50	OC	W		56	GL/MAN	Man	N/A	Shut	Shut	Shut	N/A	NA	None	N/A	Yes	1		No	W	Note 48
	HPCS	1E22F381A	0.50	OC	W		56	GL/MAN	Man	N/A	Shut	Shut	Shut	N/A	NA	None	N/A	Yes	1		No	SP/W	Note 48
		1E22F381B	0.50	OC	W		56	GL/MAN	Man	N/A	Shut	Shut	Shut	N/A	NA	None	N/A	Yes	1		No	W	Note 48
180.	HPCS	1E22F330	0.75	OC	A		*56	EFCV/-	ExFlow	N/A	Open	Open	Open	N/A	N/A	None	N/A	Yes	1	6.2-123 Sh 2	No	CA	*Note 6
181.	Supp Pool Make-up	1SM008	0.75	OC	W		*56	EFCV/-	ExFlow	N/A	Open	Open	Open	N/A	N/A	None	N/A	Yes	1	M05-1069	No	SP	*Note 6
		1SM026A	0.50	OC	W		56	GL/MAN	Man	N/A	Shut	Shut	Shut	N/A	NA	None	N/A	Yes	1		No	SP/W	Note 48
		1SM026B	0.50	OC	W		56	GL/MAN	Man	N/A	Shut	Shut	Shut	N/A	NA	None	N/A	Yes	1		No	W	Note 48
182.	Spare																					CA	
183.	Cont. Montoring	1CM002B	0.75	OC	W	2'-6"	*56	EFCV/-	ExFlow	N/A	Open	Open	Open	N/A	N/A	None	N/A	Yes	1	3.6-1 Sh 39	No	SP	*Note 6
		1CM100A	0.50	OC	W		56	GL/MAN	Man	N/A	Shut	Shut	Shut	N/A	NA	None	N/A	Yes	1		No	SP/W	Note 48
		1CM100B	0.50	OC	W		56	GL/MAN	Man	N/A	Shut	Shut	Shut	N/A	NA	None	N/A	Yes	1		No	W	Note 48
184.	Spare																					CA	
200.	RCIC	1E51F377A	0.75	OC	A	7'	*56	EFCV/-	ExFlow	N/A	Open	Open	Open	N/A	N/A	None	N/A	Yes	1	6.2-123 Sh 2	No	CA	*Note 6
201.	Spare																					CA	
202.	Spare																					CA	
203.	Cont. Montoring	1CM053	0.75	OC	A		*56	EFCV/-	ExFlow	N/A	Open	Open	Open	N/A	N/A	None	N/A	Yes	1	3.6-1 Sh 41	No	CA	*Note 6
		1CM077	0.5	OC	A			GL/MAN	Man	N/A	Shut	Shut	Shut	N/A	NA	None	N/A	Yes	1		No	CA/G	Note 48
204.	S/D Serv. Water	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	M05-1052 Sh 5	N/A	N/A	
205.	S/D Serv. Water	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	M05-1052 Sh 5	N/A	N/A	
206.	Instr. Air	1IA013A	1.0	OC	A	1'	*56	GL/MO	RM	Man	Shut	Shut	Open	As Is	Div. 2	None	N/A	Yes	2	M05-1040 Sh 5	Yes	G	*Note 6
		1IA013B	1.0	IC	A		56	GL/MO	Auto	RM	Open	Open	Shut	As Is	Div. 2	L,B,R	*25	Yes	2		Yes	CA/G	*Note 28
		1IA042A	1.0	ID	A		56	CH/-	RevFlow	N/A	Open	Shut	Shut/Open	N/A	N/A	None	N/A	Yes	2		Yes	CA/G	
207.	Spare																					CA	
208.	S/D Serv. Water	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	M05-1052 Sh 5	N/A	G	
209	Spare																					CA	
210	Process Sampling	1PS038	0.75	OC	W	1'-9"	55	GA/SO	Auto	RM	Shut/Open	Shut/Open	Shut/Open	Shut	Div. 1	B,L,R	*N/A	Yes	2	M05-1045-12	Yes	G	*Notes 18, 29
		1PS037	0.75	IC	W	3'-11"	55	GA/SO	Auto	RM	Shut/Open	Shut/Open	Shut/Open	Shut	Div. 2	B,L,R	*N/A	Yes	2		Yes	CA/G	*Notes 18, 29
	Post Accident Sampling	1PS004	0.75	IC	W	3'-0"	56	GA/SO	Auto	RM	Shut	Shut	Shut/Open	Shut	Div. 2	B,L,R	*N/A	Yes	2		Yes	CA/G	*Note 29
		1PS005	0.75	OC	W	1'-4"	56	GA/SO	Auto	RM	Shut	Shut	Shut/Open	Shut	Div. 1	B,L,R	*N/A	Yes	2		Yes	G	*Note 29
		1PS010	0.75	OC	W	2'-10"	56	GA/SO	Auto	RM	Shut	Shut	Shut/Open	Shut	Div. 1	B,L,R	*N/A	Yes	2		Yes	G	*Note 29
		1PS009	0.75	IC	W	2'-6"	56	GA/SO	Auto	RM	Shut	Shut	Shut/Open	Shut	Div. 2	B,L,R	*N/A	Yes	2		Yes	CA/G	*Note 29
		1PS031	0.75	IC	A	2'-10"	56	GA/SO	Auto	RM	Shut	Shut	Shut/Open	Shut	Div. 2	B,L,R	*N/A	Yes	2		Yes	CA/G	*Note 29
		1PS032	0.75	OC	A	1'-4"	56	GA/SO	Auto	RM	Shut	Shut	Shut/Open	Shut	Div. 1	B,L,R	*N/A	Yes	2		Yes	G	*Note 29
211.	Spare																					CA	

TABLE 6.2-47

Isolation Valve Summary for Line Penetrating Containment
(Continued)

TABLE KEY:

Location

ID - Inside Drywell
IC - Inside Containment (outside drywell)
OC - Outside Containment

Fluid Contained

A - Air or Other Gas
W - Water
S - Steam

Valve Type and Operator

BF - Butterfly		PR	- Pressure Relief or Safety
GA - Gate		MO	- Motor Operated
GL - Globe		AO	- Air Operated
PL - Plug		Ex Flow	- Excess Flow
CH - Check		SO	- Solenoid Operated
CV - Control Valve			

Actuation Mode

OVPress - Over Pressure
Rev Flow - Reverse Flow
RM - Remote Manual
Ex Flow - Excess Flow

Auto - Valve actuation is initiated by an isolation signal intended to provide containment isolation for various conditions (see Note 8)

MAN - Manual (valve handwheel or other)

Test Type

A, B, & C - Types A, B, & C, respectively, as defined in 10CFR50 Appendix J

Test Media

CA = Containment Atmosphere
G = Gas
W = Water
SP = Suppression Pool

Power Source

All valves are fed from Nuclear Safety Related power sources with divisional identification as noted.

Other

N/A - Not applicable.

TABLE 6.2-47

Isolation Valve Summary for Line Penetrating Containment
(Continued)

NOTES

1. Operation and testing of the main steamline isolation valves is discussed in Subsection 5.4.5.
2. Deleted.
3. Deleted.
4. Deleted.
5. Deleted.
6. Isolation valving for instrument lines which penetrate the containment conform to the requirements of NRC Regulatory Guide 1.11, with the exceptions stated in Section 1.8. The in-service inspection program will provide assurance of the operability and integrity of these isolations provisions. Type "C" testing will not be performed on the instrument line isolation valves. The instrument lines will be within the boundaries of the Type "A" test, open to the media (containment atmosphere or suppression pool water) to which they will be exposed under postulated accident conditions. Instrument taps from the process line located between the process isolation valves and the penetration, and not themselves penetrating containment, will be Type "A" and/or "C" tested along with the process line isolation valves. Instrument rack-mounted isolation valves outside of containment penetrations 5, 6, 7, 8, 10, 17, 36, 58 and 206 are not Containment Isolation Valves per Regulatory Guide 1.11 (originally NRC Safety Guide 11).
7. Deleted.
8. Isolation trip signals are tabulated below:

Abbrev.	Description
A	Reactor Vessel Water Level Low (Level 3)
B	Reactor Vessel Water Level Low (Level 2)
C	Deleted
D	Main Steam Line High Flow
E	Main Steam Tunnel Temp. High
F	Main Steam Tunnel Differential Temp. High

TABLE 6.2-47

Isolation Valve Summary for Line Penetrating Containment
(Continued)

Note 8 (Cont'd)

G	Main Steam in Turbine Building Temp. High
H	Turbine Inlet Pressure Low
J	Condenser Vacuum Low
L	Drywell Pressure High
M	Containment Exhaust Duct High Rad.
N	RWCU High Temp.
P	Containment High Pressure
R	CRVICS Manual Initiation
S	RHR Equipment Area High Differential Temp.
T	RHR Equipment Area High Temp.
U	Reactor Water level Low (Level 1)
V	RCIC High Steam Line Space Temp. RCIC Low Steam Line Pressure RCIC High Steam Flow High Turbine Exhaust Pressure RCIC Area High Temp. RCIC Area High Differential Temp.
X	Permissively Interlocked with Other Equipment
Z	High Rad. in Containment Refueling Pool Exhaust Duct
1	RWCU Equipment High Differential Flow
2	RWCU Vent High Differential Temp.
5	Containment Purge Duct High Radiation.
6	Standby Liquid Control System Initiation
7	RWCU Isolation Manual Initiation

9. Deleted.

10. Initiation of valve closure to be simultaneous with receipt of isolation signal.

TABLE 6.2-47

Isolation Valve Summary for Line Penetrating Containment
(Continued)

11. Deleted.

12. Normal status position of valve (open or shut) is position during power operation of the reactor.

13. Electrical Penetrations are not included in this table; they are described in USAR Section 3.8.

14. See USAR Section 3.8 for a description of the isolation provisions for these penetrations.

15. For purposes of this table designation as an Engineering Safety Feature relates to the function of the fluid system boundary containing the valve. It is recognized, in USAR Chapter 6, that the Containment Isolation System including isolation valving is an Engineered Safety Feature regardless of the classification of the fluid system boundary containing the valve.

16. Feedwater valves 1B21F032A and 1B21F032B are testable check valves which are furnished with a solenoid/air operator. This operator provides a “closing” force by releasing air pressure on the valve actuator whenever the isolation signals identified in the table are initiated or on power failure to the solenoid valves. This closing force is not adequate to close the valve against full flow. During normal operation, the solenoid/air operators allow the valve disc to free swing. These valves are not provided with position indication since Reg. Guide 1.97 specifically excludes position indication requirements for check valves.

17. Thru line leakage classifications are as follows:

    “1” Potential leakage path from containment and/or drywell into the secondary containment.

    “2” Potential leakage path from containment and/or drywell bypassing the secondary containment.

18. Systems needed to maintain the reactor safely shutdown during the Type A test need not be vented. Type C test for each affected penetration will be added to the Type A test result as applicable in Appendix J to 10 CFR 50.

19. The hydrogen recombiner system will be open to the containment during the Type A test. If these valves are shut during the Type A test, the system leak rate results will be added to the Type A test results.

TABLE 6.2-47

Isolation Valve Summary for Line Penetrating Containment
(Continued)

20. The RHR system may be operating in the shutdown cooling mode during the Type A test. The RHR penetrations affected by this operation will be tested as indicated in Note 18. For the feedwater penetrations affected by this operation (1MC009 and 1MC010), Type C water tests will be performed and reported to the NRC in the ILRT report. The test results for the feedwater penetrations will not be added to the type A test results since the FWLC will keep the penetration piping filled with water during post-accident conditions. For this same reason, test results will also not be included in the type B & type C test totals.

21. The RHR system may be operating in the shutdown cooling mode during the Type A test. These valves are tested using water but the results are not required to be added to the Type A test results. The LPCS, HPCS, and RHR may be aligned in the normal standby or injection mode during the Type A test. This will expose the closed loop outside containment to containment pressure through the suppression pool. This is the closest valve alignment to the post-LOCA alignment possible. Type C water test results on these suction valves will not be added to the Type A test results.

22. Deleted

23. Deleted

24. Each feedwater penetration contains a check valve in the drywell, an air operated check valve and a motor operated gate valve outside the drywell. Post-accident containment integrity is provided by these valves in combination with a manually initiated keep fill system (FWLC mode of RHR). Type C water tests are performed for each penetration. The total combined allowable water leakage for penetrations 1MC009 and 1MC010 is limited to 2 gpm to maintain the assumptions contained in the offsite dose analysis. The total combined leakage is calculated by summing the highest single valve leakage rate (highest leak rate for F032/F065 valves) for each penetration.

    Per Licensing Amendment 127 leakage through 1B21-F010A/B is not included in the allowable Primary Containment leakage limit. The valves are tested in accordance with Inservice Testing Program requirements.

25. Valves 1G33F001 and 1G33F004 will not be Type A tested. A water seal is assured at all times since this line drains from the bottom of the reactor vessel.

26. The pipeline passing through this penetration has a welded blind coupling installed on the outboard side of the containment.

TABLE 6.2-47

Isolation Valve Summary for Line Penetrating Containment
(Continued)

27. This penetration is used for performing periodic Type A of the containment. During testing, Valve ISA127 will provide a Type A boundary. At all other times, the line will be sealed with blank flanges.

28. The valve closure time is monitored in accordance with the In-Service Inspection (ISI) program.

29. The valve closure time has been revised to N/A to make the data consistent with the Operational Requirements Manual. ISI program, however, specifies the stroke time for the valve operability.

30. Manual isolation permissive is provided by the "B" signal for valves 1E51-F031 and 1E51-F064. The manual isolation push-button must be pushed for these valves to close.

31. The RCIC Turbine Exhaust line terminates in the suppression pool. Type C water testing is performed on this penetration to verify that no unacceptable amount of water from inside the containment can leak past this penetration.

32. Isolates on RCIC low steam line pressure only.

33. Deleted.

34. Deleted.

35. Valves 1E12-F025A and 1E12-F025B are listed in the table once for each containment penetration in which they will be tested in order to accurately represent the testing performed since these valves are associated with two penetrations. These valves are relief valves from the first isolation barrier for one penetration and the second barrier for another penetration. These valves relieve into the suppression pool and will be subject to ILRT (type A) test pressure from one penetration while the LLRT (type B & C) may be performed in association with another penetration since these valves perform isolation functions for more than one penetration.

36. Deleted.

TABLE 6.2-47

Isolation Valve Summary for Line Penetrating Containment
(Continued)

37. 1E51F077 is listed twice in the table, first with MC041 as the first valve. Since this penetration is sealed by the suppression pool, the only place for this valve to leak is back into the containment atmosphere, and no local water test is required. 1E51F077 is also listed with MC044 as the second isolation valve. This penetration is exposed to post LOCA containment atmosphere and thus will be type C tested with air.

38. The closed loop piping provides the second isolation for two penetrations. The containment leakage rate and the bypass leakage for penetration 71 & 72 or 62 & 166 is determined using the following method. The inboard containment isolations shall be considered as parallel paths (i.e. one penetration) and the leakages added to determine the inboard leakage. The outboard leakage for the combined penetrations shall be the leakage associated with the CLOC (closed loop outside containment.

39. RCIC, LPCI C, LPCS and HPCS injection lines (containment penetrations MC017, 035, 036 & 042) have been evaluated to an acceptable alternate design basis other than specifically listed in GDC 55. This alternate basis is found in SRP 6.2.4.II.6.e and the evaluation to the criteria is as follows:

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

    b. System reliability has been proven to be more reliable when only a single valve is provided with a Closed Loop Outside Containment (CLOC). A probabilistic risk assessment (PRA) of these lines has shown that a CLOC is a more reliable containment barrier than a testable check valve.

    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.

TABLE 6.2-47

Isolation Valve Summary for Line Penetrating Containment
(Continued)

39. (Continued)

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

    i. The leak tightness 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.

40. 1CM080A, B & C and 1CM081A, B & C are not included in the Type A test since they are open to allow the measurement of containment atmosphere during the type A test. The minimum path leakage for these valves will be added to the measured Type A leakage.

41. These penetrations are now considered spares (reference modification RH-F015 which deleted the type C testing requirements for these penetrations). They were modified by cutting the process piping and inserting a welded blind coupling in the piping between the drywell and the containment. The only leakage point of concern is the welded connections. No local leak rate testing is required.

42. Valve cannot be opened if containment pressure is greater than or equal to 3 psi.

43. Valves 1VR001A, 1VR001B, 1VQ004A, 1VQ004B, 0RA026, 0RA027, 1E12F037A and 1E12F037B may each be considered manual valves as long as the valve is maintained closed by use of administrative control.

44. This valve shall be “sealed closed” under administrative control to assure that it cannot be inadvertently opened. Administrative control includes mechanical devices to seal and lock the valve closed and prevent power from being supplied to the valve operator. For valves supplied with keylocked control switches, removal of the key in conjunction with tagging the control switches in the main control room in the closed position satisfies this requirement. This valve is considered a manual valve as long as it is maintained closed by use of administrative control.

45. A blind coupling separates the RCIC Turbine Exhaust Vacuum Breaker piping from the RHR piping.

TABLE 6.2-47

Isolation Valve Summary for Line Penetrating Containment
(Continued)

46. The isolation for these penetrations does not satisfy the explicit requirements of GDC 56, but is acceptable on some other defined basis, which includes the requirement for overpressure protection, the fact that the relief valves will withstand temperatures and pressures at least equal to the containment design temperature and pressure, and the additional isolation barrier of the closed system outside containment.

47. This valve receives a closure signal upon manual initiation of the associated RHR division.

48. Pressure relief valves which discharge beneath the expected low water level of the suppression pool, isolation valves for instrument lines and test connections, and vents and drains with two barriers (one inch or less) are exempted from the Type C testing. See Subsection 6.2.6.3.

49. Containment isolation valves which are sealed by water for 30 days post-LOCA may be water tested. Test connections, vents, and drains which are water-sealed for 30 days post-LOCA are exempt from all testing. See Subsection 6.2.6.3.

50. Relief Valves 1E12-F055A & B are shut by raising the relief setpoint.

51. Penetration 1MC-34 follows the requirements of GDC criteria 56 and is acceptable on some other defined basis per ANSI/ ANS-56.2 Note 56-1 of ANS 56.2 states that “Lines which connect directly to the suppression pool are each provided with a single remote manual or automatic isolation valve. These valves are attached to lines which are an extension of the containment and are enclosed in a pump room adjacent to the containment which has provisions for environmental control of any fluid leakage. The lines to the suppression pool are always submerged so no containment atmosphere can impinge upon the valves.” The SF containment automatic isolation valve (1SF004) and SF pumps are located within a sealable room that assures the penetration isolation provision.

52. Penetration number 79 follows Criterion 56. ANSI N271 note 56-1 states, “An acceptable alternative to the above arrangement would be to provide a second isolation valve outside containment and eliminate the requirement for a closed system outside containment.” This explicitly covers 1MC-79, which has two automatic containment isolation valves outside containment.

53. USAR Table 6.2-47 indicates valves 1SF001 and 1SF002 are Type “C” tested with water. However, as a matter of convenience, these valves are tested with air/gas, not water. See USAR Section 6.2.6.3 for further discussion on penetrations which use air as an acceptable alternative to water testing.

TABLE 6.2-48
COMBUSTIBLE GAS CONTROL SYSTEM COMPONENTS
DESIGN AND PERFORMANCE DATA

CGCS Compressors	
Quantity	Two 100% capacity
Capacity (cfm)	800
Static Pressure (psig)	4.5
Drive	direct
Motor, hp	60
Manufacturer	Spencer Turbine
CGCS Recombiners	
Quantity	Two 100% capacity
Capacity, (scfm)	70
Static Pressure (inches H_20)	3.9
Design Pressure (at 1400°F/250°F, psig)	10/50
Pneumatic Test Pressure (psig)	72 ± 3
Heater Rating (kW)	48
Motors: 1. Fan hp	3
2. Blower hp	7.5
Recombination Rate (% H_2)	up to 5%
Hydrogen Return to Containment (at 3% H_2 conc. by vol. fuel)	1/30
Manufacturer	Rockwell International

TABLE 6.2-49
INVENTORY OF CORRODIBLE MATERIALS
INSIDE THE DRYWELL AND CONTAINMENT

VOLUME	MATERIAL	WEIGHT (lbm)	SURFACE AREA (ft^2)
Drywell	Aluminum Alloy Groups 1 and 2	4,126	54,171
	Aluminum or Aluminum Alloys (except as specified above)	940	528
	Aluminum Allowance for Future Modifications	200	100
	Zinc (Galvanized Steel and Surfaces Coated with Seine Based Paint) not subject to complete and continuous immersion	15,021	144,937
	Zinc (Galvanized Steel and Surfaces Coated with Zinc Based Paint) which may be subject to complete and continuous immersion	3,858	25,756
	Zinc Allowance for Future Modifications	100	1000
Containment	Aluminum Alloy Groups 1 and 2	6,519	45,512
	Aluminum or Aluminum Alloys (except as specified above)	4,492	2,494
	Aluminum Allowance for Future Modifications	200	500
	Zinc (Galvanized Steel and Surfaces Coated with Zinc Based Paint) not subject to complete and continuous immersion	---	---
	Zinc (Galvanized Steel and Surfaces Coated with Zinc Based Paint) which may be subject to complete and continuous immersion	---	---

TABLE 6.2-50a
TRANSIENT TEMPERATURE ENVELOPE IN THE DRYWELL AND THE CORRESPONDING HYDROGEN EVOLUTION RATE AS A FUNCTION OF TIME FOR MATERIAL CORROSION @pH = 8.6

$$\text{CORROSION RATE}\left(\frac{\text{scf of }H_2}{\text{hr}-\text{ft}^2}\right)$$

TIME	°F	°R	°K	Aluminum Set 1	Aluminum Set 2	Aluminum	Zinc
3	250	709.7	394.3	8.5639E-06	1.0705E-05	5.7545E-02	1.6509E-03
6	250	709.7	394.3	8.5639E-06	1.0705E-05	5.7545E-02	1.6509E-03
24	250	709.7	394.3	8.5639E-06	1.0705E-05	5.7545E-02	1.6509E-03
36	249	708.9	393.8	8.4895E-06	1.0612E-05	5.6873E-02	1.6186E-03
48	248	708.2	393.4	8.4156E-06	1.0520E-05	5.6208E-02	1.5868E-03
60	248	707.4	393.0	8.3422E-06	1.0428E-05	5.5550E-02	1.5556E-03
72	247	706.6	392.6	8.2692E-06	1.0337E-05	5.4897E-02	1.5249E-03
84	246	705.9	392.2	8.1968E-06	1.0246E-05	5.4251E-02	1.4948E-03
96	245	705.1	391.7	8.1248E-06	1.0156E-05	5.3612E-02	1.4652E-03
120	244	703.6	390.9	7.9823E-06	9.9779E-06	5.2351E-02	1.4075E-03
144	242	702.1	390.1	7.8417E-06	9.8021E-06	5.1114E-02	1.3519E-03
168	241	700.6	389.2	7.7030E-06	9.6287E-06	4.9901E-02	1.2983E-03
192	239	699.1	388.4	7.5661E-06	9.4576E-06	4.8712E-02	1.2465E-03
240	236	696.0	386.7	7.2979E-06	9.1224E-06	4.6404E-02	1.1486E-03
288	233	693.0	385.0	7.0370E-06	8.7963E-06	4.4187E-02	1.0576E-03
360	229	688.5	382.5	6.6591E-06	8.3238E-06	4.1025E-02	9.3311E-04
480	221	680.9	378.3	6.0638E-06	7.5797E-06	3.6169E-02	7.5456E-04
624	212	671.8	373.2	5.4041E-06	6.7552E-06	3.0979E-02	5.8111E-04
720	206	665.7	369.9	4.9960E-06	6.2450E-06	2.7874E-02	4.8631E-04

TABLE 6.2-50b

TRANSIENT TEMPERATURE ENVELOPE IN THE DRYWELL AND THE CORRESPONDING HYDROGEN EVOLUTION RATE AS A FUNCTION OF TIME FOR MATERIAL CORROSION @pH = 5.6

$$\text{CORROSION RATE}\left(\frac{\text{scf of H}_2}{\text{hr}-\text{ft}^2}\right)$$

TIME	°F	°R	°K	Aluminum Set 1	Aluminum Set 2	Aluminum	Zinc
3	250	709.7	394.3	9.8296E-07	1.2287E-06	2.6768E-03	4.5264E-03
6	250	709.7	394.3	9.8296E-07	1.2287E-06	2.6768E-03	4.5264E-03
24	250	709.7	394.3	9.8296E-07	1.2287E-06	2.6768E-03	4.5264E-03
36	249	708.9	393.8	9.8119E-07	1.2265E-06	2.6455E-03	4.4377E-03
48	248	708.2	393.4	9.7943E-07	1.2243E-06	2.6146E-03	4.3506E-03
60	248	707.4	393.0	9.7766E-07	1.2221E-06	2.5840E-03	4.2650E-03
72	247	706.6	392.6	9.7590E-07	1.2199E-06	2.5536E-03	4.1809E-03
84	246	705.9	392.2	9.7413E-07	1.2177E-06	2.5236E-03	4.0983E-03
96	245	705.1	391.7	9.7237E-07	1.2155E-06	2.4938E-03	4.0171E-03
120	244	703.6	390.9	9.6883E-07	1.2110E-06	2.4352E-03	3.8591E-03
144	242	702.1	390.1	9.6530E-07	1.2066E-06	2.3776E-03	3.7066E-03
168	241	700.6	389.2	9.6176E-07	1.2022E-06	2.3212E-03	3.5596E-03
192	239	699.1	388.4	9.5822E-07	1.1978E-06	2.2659E-03	3.4177E-03
240	236	696.0	386.7	9.5113E-07	1.1889E-06	2.1586E-03	3.1492E-03
288	233	693.0	385.0	9.4404E-07	1.1800E-06	2.0554E-03	2.8996E-03
360	229	688.5	382.5	9.3337E-07	1.1667E-06	1.9083E-03	2.5584E-03
480	221	680.9	378.3	9.1556E-07	1.1445E-06	1.6825E-03	2.0688E-03
624	212	671.8	373.2	8.9413E-07	1.1177E-06	1.4410E-03	1.5933E-03
720	206	665.7	369.9	8.7980E-07	1.0997E-06	1.2966E-03	1.3334E-03

TABLE 6.2-51a

TRANSIENT TEMPERATURE ENVELOPE IN THE CONTAINMENT AND THE CORRESPONDING HYDROGEN EVOLUTION RATE AS A FUNCTION OF TIME FOR MATERIAL CORROSION @pH = 8.6

CORROSION RATE $\left(\frac{\text{scf of } H_2}{\text{hr} - \text{ft}^2}\right)$

TIME	°F	°R	°K	Aluminum Set 1	Aluminum Set 2	Aluminum	Zinc Rate
3	185	644.7	358.2	3.7595E-06	4.6994E-06	1.9015E-02	2.5518E-04
6	185	644.7	358.2	3.7595E-06	4.6994E-06	1.9015E-02	2.5518E-04
24	185	644.7	358.2	3.7595E-06	4.6994E-06	1.9015E-02	2.5518E-04
36	185	644.2	357.9	3.7370E-06	4.6713E-06	1.8862E-02	2.5173E-04
48	184	643.8	357.7	3.7147E-06	4.6434E-06	1.8711E-02	2.4833E-04
60	184	643.4	357.4	3.6924E-06	4.6156E-06	1.8560E-02	2.4497E-04
72	183	643.0	357.2	3.6703E-06	4.5879E-06	1.8411E-02	2.4166E-04
84	183	642.5	357.0	3.6483E-06	4.5603E-06	1.8262E-02	2.3838E-04
96	182	642.1	356.7	3.6263E-06	4.5329E-06	1.8115E-02	2.3514E-04
120	182	641.2	356.2	3.5828E-06	4.4785E-06	1.7823E-02	2.2878E-04
144	181	640.4	355.8	3.5396E-06	4.4246E-06	1.7535E-02	2.2258E-04
168	180	639.5	355.3	3.4969E-06	4.3711E-06	1.7250E-02	2.1653E-04
192	179	638.7	354.8	3.4546E-06	4.3182E-06	1.6970E-02	2.1063E-04
240	177	636.9	353.9	3.3711E-06	4.2139E-06	1.6421E-02	1.9927E-04
288	176	635.2	352.9	3.2892E-06	4.1115E-06	1.5887E-02	1.8846E-04
360	173	632.6	351.5	3.1693E-06	3.9616E-06	1.5112E-02	1.7323E-04
480	169	628.4	349.1	2.9770E-06	3.7213E-06	1.3893E-02	1.5031E-04
624	164	623.2	346.2	2.7585E-06	3.4482E-06	1.2539E-02	1.2645E-04
720	160	619.8	344.3	2.6200E-06	3.2750E-06	1.1699E-02	1.1250E-04

TABLE 6.2-51b

TRANSIENT TEMPERATURE ENVELOPE IN THE CONTAINMENT AND THE CORRESPONDING HYDROGEN EVOLUTION RATE AS A FUNCTION OF TIME FOR MATERIAL CORROSION @pH = 5.6

$$\text{CORROSION RATE}\left(\frac{\text{scf of }H_2}{\text{hr}-\text{ft}^2}\right)$$

TIME	°F	°R	°K	Aluminum Set 1	Aluminum Set 2	Aluminum	Zinc Rate
3	185	644.7	358.2	8.2981E-07	1.0373E-06	8.8451E-04	6.9963E-04
6	185	644.7	358.2	8.2981E-07	1.0373E-06	8.8451E-04	6.9963E-04
24	185	644.7	358.2	8.2981E-07	1.0373E-06	8.8451E-04	6.9963E-04
36	185	644.2	357.9	8.2878E-07	1.0360E-06	8.7741E-04	6.9020E-04
48	184	643.8	357.7	8.2776E-07	1.0347E-06	8.7036E-04	6.8087E-04
60	184	643.4	357.4	8.2674E-07	1.0334E-06	8.6336E-04	6.7166E-04
72	183	643.0	357.2	8.2572E-07	1.0321E-06	8.5640E-04	6.6256E-04
84	183	642.5	357.0	8.2470E-07	1.0309E-06	8.4949E-04	6.5358E-04
96	182	642.1	356.7	8.2367E-07	1.0296E-06	8.4263E-04	6.4470E-04
120	182	641.2	356.2	8.2163E-07	1.0270E-06	8.2905E-04	6.2727E-04
144	181	640.4	355.8	8.1958E-07	1.0245E-06	8.1565E-04	6.1027E-04
168	180	639.5	355.3	8.1754E-07	1.0219E-06	8.0243E-04	5.9368E-04
192	179	638.7	354.8	8.1549E-07	1.0194E-06	7.8939E-04	5.7751E-04
240	177	636.9	353.9	8.1140E-07	1.0142E-06	7.6384E-04	5.4634E-04
288	176	635.2	352.9	8.0730E-07	1.0091E-06	7.3899E-04	5.1671E-04
360	173	632.6	351.5	8.0116E-07	1.0014E-06	7.0298E-04	4.7497E-04
480	169	628.4	349.1	7.9091E-07	9.8864E-07	6.4623E-04	4.1212E-04
624	164	623.2	346.2	7.7860E-07	9.7325E-07	5.8326E-04	3.4669E-04
720	160	619.8	344.3	7.7039E-07	9.6299E-07	5.4420E-04	3.0846E-04

TABLE 6.2-52
PARAMETERS USED FOR THE EVALUATION OF COMBUSTIBLE GASES IN THE DRYWELL AND THE CONTAINMENT AFTER A LOCA

Drywell Volume Post LOCA	208,204 ft^3
Suppression Chamber Free Volume	1,512,341 ft^3
Reactor Power	3543 MWt
Amount of Active Zirconium	59380 lbm
Fraction of Zirconium Reacted	0.945%
Initial Oxygen Concentration in Dry Air	0.21
Drywell Initial Temperature	250° F
Drywell Initial Pressure	14.7 psia
Drywell Initial Relative Humidity	50%
Wetwell Initial Temperature	185° F
Wetwell Initial Pressure	14.7 psia
Wetwell Initial Relative Humidity	50%
Capacity of Thermal Recombiner	70 scfm
Recombiner Efficiency	96%
Fraction Suppression Pool Water in Drywell	0.283
Capacity of Mixing System	800 cfm

TABLE 6.2-53

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TABLE 6.2-54

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TABLE 6.2-67
SUBCOMPARTMENT NODAL DESCRIPTION
OF BIOLOGICAL SHIELD ANNULUS - RECIRCULATION DISCHARGE LINE BREAK

NODE NO.	DESCRIPTION	VOLUME (ft^3)	HEIGHT (ft)	FLOW CROSS-SECTIONAL AREA (ft^2)	BOTTOM ELEVATION (ft)	INITIAL CONDITIONS Temp, (°F)	Press, (psia)	Humid,* (%)	CALC. PEAK PRESS DIFF, (psid)
1	Reactor Skirt Sect.	64.0	9.63	13.5	742.67	528.0	14.20	0.1	18.8
2	Reactor Skirt Sect.	85.5	9.63	13.5	742.67	528.0	14.20	0.1	20.7
3	Reactor Skirt Sect.	06.7	9.63	13.5	742.67	528.0	14.20	0.1	22.9
4	Reactor Skirt Sect.	128.2	9.63	13.5	742.67	528.0	14.20	0.1	16.8
5	Reactor Skirt Sect.	128.2	9.63	13.5	742.67	528.0	14.20	0.1	17.9
6	Lower Recirc. Nozzle Sect.	59.7	6.00	16.2	752.29	528.0	14.20	0.1	28.9
7	Lower Recirc. Nozzle Sect.	92.0	6.00	16.2	752.29	528.0	14.20	0.1	20.8
8	Lower Recirc. Nozzle Sect.	112.6	6.00	16.2	752.29	528.0	14.20	0.1	14.9
9	Lower Recirc. Nozzle Sect.	131.5	6.00	16.2	752.29	528.0	14.20	0.1	14.9
10	Lower Recirc. Nozzle Sect.	135.0	6.00	16.2	752.29	528.0	14.20	0.1	17.3
11	Upper Recirc. Nozzle Sect.	34.0	3.71	10.0	758.29	528.0	14.20	0.1	32.3
12	Upper Recirc. Nozzle Sect.	56.1	3.71	10.0	758.29	528.0	14.20	0.1	16.3
13	Upper Recirc. Nozzle Sect.	66.6	3.71	10.0	758.29	528.0	14.20	0.1	15.0
14	Upper Recirc. Nozzle Sect.	81.6	3.71	10.0	758.29	528.0	14.20	0.1	17.2
15	Upper Recirc. Nozzle Sect.	84.1	3.71	10.0	758.29	528.0	14.20	0.1	15.3
16	Mid-Section	187.4	16.29	12.0	762.00	528.0	14.20	0.1	15.7
17	Mid-Section	248.2	16.29	16.0	762.00	528.0	14.20	0.1	12.4
18	Mid-Section	314.0	16.29	19.9	762.00	528.0	14.20	0.1	12.7
19	Mid-Section	383.4	16.29	23.9	762.00	528.0	14.20	0.1	14.2
20	Mid-Section	388.2	16.29	23.9	762.00	528.0	14.20	0.1	13.8
21	Feedwater Nozzle Sect.	167.0	15.75	12.0	778.29	528.0	14.20	0.1	11.5
22	Feedwater Nozzle Sect.	231.5	15.75	16.0	778.29	528.0	14.20	0.1	10.4
23	Feedwater Nozzle Sect.	277.9	15.75	19.9	778.29	528.0	14.20	0.1	10.0

TABLE 6.2-67 (Cont'd)

NODE NO.	DESCRIPTION	VOLUME (ft^3)	HEIGHT (ft)	FLOW CROSS-SECTIONAL AREA (ft^2)	BOTTOM ELEVA-TION (ft)	INITIAL CONDITIONS Temp, (°F)	Press, (psia)	Humid,* (%)	CALC. PEAK PRESS DIFF, (psid)
24	Feedwater Nozzle Sect.	334.5	15.75	23.9	778.29	528.0	14.20	0.1	9.9
25	Feedwater Nozzle Sect.	338.8	15.75	23.9	778.29	528.0	14.20	0.1	9.8
26	Main Steam Nozzle Sect.	493.3	8.13	70.9	794.04	528.0	14.20	0.1	4.5
27	Main Steam Nozzle Sect.	493.3	8.13	70.9	794.04	528.0	14.20	0.1	4.4
28	Containment	73,250.0	80.25	1,520.0	723.15	150.0	14.20	0.1	-
29	Break Node	17.0	3.00	4.25	756.79	528.0	14.20	0.1	76.9

* Relative Humidity

TABLE 6.2-68
SUBCOMPARTEMENT VENT PATH DESCRIPTION
OF BIOLOGICAL SHIELD ANNULUS - RECIRCULATION DISCHARGE LINE BREAK

VENT PATH NO.	FROM VOL. NODE NO.	TO VOL. NODE NO.	DESCRIPTION OF VENT PATH FLOW	AREA* (ft^2)	LENGTH (ft)	$\sum \frac{L}{A}$ (ft^{-1})	HYDRAULIC DIAMETER (ft)	HEAD LOSS, K: Friction Loss, K_f	Turning Loss, K_{bl}	Expansion & Contraction K_E	Total
1	1	2	Unchoked	13.53	5.07	0.375	3.62	0.02	0.04	0.00	0.06
2	2	3	Unchoked	13.53	6.52	0.483	3.62	0.03	0.05	0.00	0.08
3	3	4	Unchoked	13.53	7.97	0.590	3.62	0.04	0.05	0.00	0.09
4	4	5	Unchoked	13.53	8.69	0.644	3.62	0.04	0.06	0.00	0.10
5	6	1	Unchoked	7.80	5.72	0.733	3.15	0.03	0.00	1.05	1.08
6	7	2	Unchoked	10.37	5.72	0.552	3.63	0.03	0.00	1.05	1.08
7	8	3	Unchoked	13.04	5.72	0.439	4.07	0.03	0.00	1.05	1.08
8	4	9	Unchoked	15.60	5.72	0.367	4.46	0.03	0.00	1.05	1.08
9	5	10	Unchoked	15.60	5.72	0.367	4.46	0.03	0.00	1.05	1.08
10	29	6	Unchoked	6.38	5.00	0.784	2.85	0.00	0.00	0.63	0.63
11	6	7	Unchoked	14.36	5.07	0.353	3.73	0.03	0.04	0.09	0.16
12	7	8	Unchoked	14.36	6.52	0.454	3.73	0.03	0.05	0.02	0.10
13	8	9	Unchoked	10.95	7.97	0.728	3.73	0.03	0.05	0.30	0.38
14	9	10	Unchoked	14.36	8.69	0.605	3.73	0.03	0.05	0.07	0.15
15	6	11	Unchoked	9.63	4.85	0.503	3.50	0.05	0.00	0.14	0.19
16	7	12	Unchoked	11.90	4.85	0.408	3.89	0.04	0.00	0.12	0.16
17	8	13	Unchoked	11.64	4.85	0.417	3.85	0.03	0.00	0.30	0.33
18	9	14	Unchoked	15.56	4.85	0.312	4.45	0.03	0.00	0.23	0.26
19	10	15	Unchoked	17.84	4.85	0.272	4.77	0.03	0.00	0.14	0.17
20	29	11	Unchoked	6.38	4.00	0.627	2.85	0.00	0.00	0.62	0.62
21	11	12	Unchoked	8.15	5.07	0.622	3.13	0.02	0.04	0.07	0.13
22	12	13	Unchoked	8.15	6.52	0.800	3.13	0.03	0.05	0.06	0.14
23	13	14	Unchoked	6.99	7.97	1.140	3.13	0.04	0.07	0.29	0.40
24	14	15	Unchoked	8.15	8.69	1.066	3.13	0.04	0.07	0.12	0.23
25	11	16	Unchoked	10.10	10.00	0.990	3.59	0.04	0.00	1.33	1.37

* Minimum cross-sectional area

TABLE 6.2-68 (Cont'd)

VENT PATH NO.	FROM VOL. NODE NO.	TO VOL. NODE NO.	DESCRIPTION OF VENT PATH FLOW	AREA* (ft^2)	LENGTH (ft)	$\sum \frac{L}{A}$ (ft^{-1})	HYDRAULIC DIAMETER (ft)	HEAD LOSS, K Friction Loss, K_f	Turning Loss, K_{bl}	Expansion & Contraction K_E	Total
26	12	17	Unchoked	15.52	10.00	0.664	4.45	0.07	0.00	0.00	0.07
27	13	18	Unchoked	19.61	10.00	0.510	5.00	0.07	0.00	0.00	0.07
28	14	19	Unchoked	22.36	10.00	0.447	5.34	0.07	0.00	0.37	0.44
29	15	20	Unchoked	23.53	10.00	0.425	5.47	0.07	0.00	0.00	0.07
30	16	17	Unchoked	42.08	5.07	0.121	4.72	0.02	0.05	0.27	0.34
31	17	18	Unchoked	42.58	6.52	0.153	4.72	0.02	0.06	0.06	0.14
32	18	19	Unchoked	44.80	7.97	0.178	4.72	0.02	0.06	0.05	0.13
33	19	20	Unchoked	42.91	8.69	0.203	4.72	0.02	0.06	0.05	0.13
34	16	21	Unchoked	8.99	16.04	1.784	3.38	0.05	0.00	0.43	0.48
35	17	22	Unchoked	11.52	16.04	1.392	3.83	0.04	0.00	0.47	0.51
36	18	23	Unchoked	16.94	16.04	0.947	4.64	0.05	0.00	1.66	1.71
37	19	24	Unchoked	19.97	16.04	0.803	5.04	0.06	0.00	0.85	0.91
38	20	25	Unchoked	18.97	16.04	0.846	4.91	0.05	0.00	1.07	1.12
39	21	22	Unchoked	40.31	5.07	0.126	4.70	0.02	0.05	0.27	0.34
40	22	23	Unchoked	36.09	6.52	0.181	4.70	0.02	0.06	0.05	0.13
41	23	24	Unchoked	41.31	7.97	0.193	4.70	0.02	0.07	0.06	0.15
42	24	25	Unchoked	36.09	8.69	0.241	4.70	0.02	0.08	0.27	0.37
43	21	26	Unchoked	7.80	11.88	1.523	3.15	0.04	0.00	1.27	1.31
44	22	26	Unchoked	10.47	9.88	0.944	3.65	0.03	0.00	1.05	1.08
45	23	26	Unchoked	13.09	11.38	0.869	4.08	0.04	0.00	1.21	1.25
46	24	27	Unchoked	15.60	10.88	0.697	4.46	0.04	0.00	1.22	1.26
47	25	27	Unchoked	15.71	10.88	0.693	4.47	0.03	0.00	1.05	1.08
48	26	27	Unchoked	27.64	18.52	0.670	5.12	0.08	0.18	0.00	0.26
49	26	28	Unchoked	66.00	2.50	0.056	9.17	0.07	0.00	4.28	4.35
50	27	28	Unchoked	66.00	2.50	0.056	9.17	0.07	0.00	4.28	4.35
51	0	29	Choked	1.00	0.00	0.000	----	----	----	----	----

* Minimum cross-sectional area

TABLE 6.2-69
MASS AND ENERGY RELEASE RATE DATA
FOR RECIRCULATION DISCHARGE LINE BREAK† IN BIOLOGICAL SHIELD ANNULUS

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	7522.0	0.0	528.8	1196.0	7522.0	3.98×10^6
0.0074	7522.0	0.0	528.8	1196.0	7522.0	3.98×10^6
0.0075	11283.0	0.0	528.8	1196.0	11283.0	5.97×10^6
0.1457	11283.0	0.0	528.8	1196.0	11283.0	5.97×10^6
0.1458	8290.0	0.0	528.7	1196.0	8290.0	4.38×10^6
5.0000	8290.0	0.0	528.7	1196.0	8290.0	4.38×10^6

† Tabulated values were halved when input to symmetric analysis

TABLE 6.2-70
RECIRCULATION OUTLET LINE BREAK[+] WITH FLOW DIVERTER
FORCE CONSTANTS FOR FORCE CALCULATIONS ON THE RPV

Vol. Node No.	Coefficient (in^2) Fx	Fy	Moment Arm** (in)
1	13361.9	0.0	94.462
2	9731.59	0.0	94.462
3	3580.31	0.0	94.462
4	-7827.22	0.0	94.462
5	-18896.59	0.0	94.462
6	7887.92	0.0	.706
7	6076.27	0.0	.706
8	2215.84	0.0	.706
9	-4861.48	0.0	.706
10	-11342.14	0.0	.706
11	6977.71	0.0	-66.29
12	5209.02	0.0	-66.29
13	1906.83	0.0	-66.29
14	-4167.5	0.0	-66.29
15	-9929.32	0.0	-66.29
16	13905.00	0.0	-157.286
17	10197.17	0.0	-157.286
18	3725.83	0.0	-157.286
19	-8145.36	0.0	-157.286
20	-19664.65	0.0	-157.286
21	13721.90	0.0	-282.29
22	9833.3	0.0	-282.29
23	3646.22	0,0	-282.29
24	-7945.62	0,0	-282.29
25	-19276.01	0.0	-282.29
26	13605.72	0.0	-405.794
27	9869.80	0.0	-405.794
28	3645.64	0.0	-405.794
29	-7945.62	0.0	-405.794
30	-19182.42	0.0	-405.794
31	22163.05	0.0	-513.05
32	-22163.05	0.0	-513.05
33*	991.80	0.0	-30.29

* Break node

** Distance measured above RPV datum to a height of 135.5” (Ref. 2)

+ Symmetric analysis

TABLE 6.2-71

RECIRCULATION OUTLET LINE BREAK[+] WITH FLOW DIVERTER
FORCE CONSTANTS FOR FORCE CALCULATIONS ON SHIELD WALL

Vol. Node No.	Coefficient (in^2) Fx	Fy	Moment Arm** (in)
1	17902.51	0.0	94.462
2	13105.55	0.0	94.462
3	4796.96	0.0	94.462
4	-10487.04	0.0	94.462
5	-25317.97	0.0	94.462
6	9768.55	0.0	.706
7	7734.29	0.0	.706
8	2634.18	0.0	.706
9	-6183.92	0.0	.706
10	-14167.76	0.0	.706
11	8676.66	0.0	-66.29
12	6599.61	0.0	-66.29
13	2415.63	0.0	-66.29
14	-5275.95	0.0	-66.29
15	-12413.45	0.0	-66.29
16	18600.00	0.0	-157.286
17	13616.15	0.0	-157.286
18	4983.86	0.0	-157.286
19	-10895.63	0.0	-157.286
20	-26304.38	0.0	-157.286
21	18032.95	0.0	-282.29
22	12096.66	0.0	-282.29
23	4738.61	0.0	-282.29
24	-10118.04	0.0	-282,29
25	-25080.17	0.0	-282.29
26	18135.01	0.0	-405.794
27	12775.92	0.0	-405.794
28	4859.26	0.0	-405.794
29	-10487.98	0.0	-405.794
30	-25320.23	0.0	-405.794
31	0	0.0	-513.05
32	0	0.0	-513.05
33*	1431.10	0.0	-30.29

* Break node

** Distance measured above RPV datum to a height of 135.5" (Ref. 2)

\+ Symmetric analysis

TABLE 6.2-72
FEEDWATER LINE BREAK FORCE CONSTANTS
FOR FORCE CALCULATIONS ON RPV

Vol. Node No.	Coefficient (in^2) Fx	Fy	Moment Arm** (in)
1	21253.18	21253.18	404.45
2	-21298.31	21298.31	404.45
3	-21253.18	-21253.18	404.45
4	21298.31	-21298.31	404.45
5	16499.50	16499.50	243.47
6	-16499.50	-16499.50	243.47
7	-16499.50	-16499.50	243.47
8	16499.50	16499.50	243.47
9	11810.99	11810.99	123.59
10	-2135.99	6774.51	123.59
11	-2822.98	2822.98	123.59
12	-6774.51	2153.99	123.59
13	-11740.85	-11740.85	123.59
14	11740.85	-11740.85	123.59
15	7680.05	3181.18	27.59
16	3129.70	7555.75	27.59
17	-1115.83	3538.97	47.09
18	1493.19	1493.19	47.09
19	-3641.96	1148.31	7.09
20	-7604.65	-3149.95	27.59
21	-3109.15	-7506.16	27.59
22	10703.20	-10703.20	27.59
23	-818.88	2597.15	.71
24*	-945.47	945.47	.71
25	-2661.97	839.31	.71
26	8118.38	3362.74	-67.03
27	3362.74	8118.38	-67.03
28	-2076.34	6585.30	-67.03
29	-2819.48	2819.48	-67.03
30	-6585.30	2076.34	-67.03
31	-8118.38	-3362.74	-67.03
32	-3362.74	-8118.38	-67.03
33	11481.12	-1148.12	-67.03
34***	0	0	

* Break node

** Distance measured above RPV datum to a height of 484.5" (Ref. 2)

*** Containment Volume

TABLE 6.2-73
FEEDWATER LINE BREAK
FORCE CONSTANTS FOR FORCE CALCULATIONS ON THE SHIELD WALL

Vol. Node No.	Coefficient (in^2) Fx	Fy	Moment Arm** (in)
1	25995.10	25995.10	404.45
2	-26237.11	26237.11	404.45
3	-25995.10	-25995.10	404.45
4	26237.11	-26237.11	404.45
5	20316.20	20316.20	243.47
6	-20316.20	20316.20	243.47
7	-20316.20	-20316.20	243.47
8	20316.20	-20316.20	243.47
9	14788.55	14788.55	123.59
10	-2674.48	8482.36	123.59
11	-3415.11	3415.11	123.59
12	-8482.36	2674.48	123.59
13	-14592.55	-14592.55	123.59
14	14592.55	-14592.55	123.59
15	9853.63	3915.82	27.59
16	3784.02	9135.43	27.59
17	-1343.68	4261.61	47.09
18	-1750.08	1750.08	47.09
19	-4473.99	1410.64	47.09
20	-9200.48	-3810.96	27.59
21	-3726.61	-8996.84	27.59
22	12856.92	-12856.92	27.59
23	-998.59	3167.12	.71
24*	-929.75	929.75	.71
25	-3333.05	1050.91	.71
26	10165.03	4210.49	-67.03
27	4210.49	10165.03	-67.03
28	-2599.78	8245.46	-67.03
29	-3530.28	3530.28	-67.03
30	-8245.46	2599.78	-67.03
31	-10165.03	-4210.49	-67.03
32	-4210.49	-10165.03	-67.03
33	14375.52	14375.52	-67.03
34***	0	0	

* Break node

** Distance measured above RPV datum to a height of 484.5" (Ref. 2)

*** Containment Volume

6.3 EMERGENCY CORE COOLING SYSTEM

6.3.1 Design Bases

Subsection 6.3.1 provides the design bases for the Emergency Core Cooling System (ECCS) and a summary description of the several systems as an introduction to the more detailed design descriptions provided in Subsection 6.3.2, and the performance analysis provided in Subsection 6.3.3.

6.3.1.1 Design Bases and Summary Description

6.3.1.1.1 Performance and Functional Requirements

The ECCS is designed to provide protection against postulated Loss-Of-Coolant Accidents (LOCA) caused by ruptures in primary system piping. The functional requirements (for example, coolant delivery rates) specified in detail in Table 6.3-8 are such that the system performance under all LOCA conditions postulated in the design satisfies the requirements of Paragraph 50.46, "Acceptance Criteria for Emergency Core Cooling System for Light Water Cooled Nuclear Power Reactors," of 10 CFR 50. These requirements, the most important of which is that the post-LOCA peak cladding temperature be limited to 2200°F, are summarized in Subsection 6.3.3.2. In addition, the ECCS is designed to meet the following requirements:

a. Protection is provided for any primary system line break up to, and including, the double-ended break of the largest line.

b. Two independent phenomenological cooling methods (flooding and spraying) are provided to cool the core.

c. One high-pressure cooling system is provided which is capable of maintaining water level above the top of the core and preventing automatic depressurization system (ADS) actuation for breaks of lines less than 1 inch nominal diameter.

d. No operator action is required until 10 minutes after an accident to allow for operator assessment and decision.

e. The ECCS is designed to satisfy all criteria specified in Section 6.3.

f. A sufficient water source and the necessary piping, pumps, and other hardware are provided so that the containment and reactor core can be flooded for possible core heat removal following a loss-of-coolant accident.

6.3.1.1.2 Reliability Requirements

The following reliability requirements apply:

a. The ECCS must conform to all licensing requirements and good design practices of isolation, separation, and common mode failure considerations.

b. In order to meet the above requirements, the ECCS network shall have built-in redundancy so that adequate cooling can be provided, even in the event of specified failures. As a minimum, the following equipment make up the ECCS:

   1. high-pressure core spray (HPCS),

   2. low-pressure core spray (LPCS),

   3. low-pressure coolant injection (LPCI) loops, and

   4. automatic depressurization system (ADS).

c. The system shall be designed so that a single active or passive component failure, including power buses, electrical and mechanical parts, cabinets, and wiring will not disable the ADS.

d. In the event of a break in a pipe that is not a part of the ECCS, no single active component failure in the ECCS shall prevent automatic initiation and successful operation of less than the following combination of ECCS equipment:

   1. three LPCI loops, the LPCS and the ADS (i.e., HPCS failure); or

   2. two LPCI loops, the HPCS and the ADS (i.e., "LPCS diesel generator" failure); or

   3. one LPCI loop, the LPCS, the HPCS and ADS (i.e., "LPC1 diesel generator" failure).

e. In the event of a break in a pipe that is a part of the ECCS, no single active component failure in the ECCS shall prevent automatic initiation and successful operation of less than the following combination of ECCS equipment:

   1. two LPCI loops and the ADS; or.

   2. one LPCI loop, the LPCS and the ADS; or

   3. one LPCI loop, the HPCS and the ADS; or

   4. the LPCS, the HPCS and ADS.

   These are the minimum ECCS combinations which result after assuming any failure (from 4 above) and assuming that the ECCS line break disables the affected system.

f. Long-term (10 minutes after initiation signal) cooling requirements call for the removal of decay heat via the service water system. In addition to the break which initiated the loss-of-coolant event, the system must be able to sustain one failure, either active or passive and still have at least one low-pressure ECCS pump (LPCI, HPCS or LPCS) operating with a heat exchanger and 100% service water flow.

g. Offsite power is the preferred source of power for the ECCS network and every reasonable precaution must be made to assure its high availability. However, onsite emergency power shall be provided with sufficient diversity and capacity so that all the above requirements can be met even if offsite power is not available.

h. The onsite diesel fuel reserve is in accordance with Regulatory Guide 1.137 with exceptions as discussed in Section 1.8.

i. Diesel-load configuration shall be as follows:

    1. one LPCI loop (with heat exchanger) and the LPCS connected to a single diesel generator;

    2. two additional LPCI loops (1 loop with heat exchanger) connected to a single diesel generator; and

    3. the HPCS connected to a single diesel-generator.

j. Systems which interface with, but are not part of the ECCS shall be designed and operated such that failure(s) in the interfacing systems shall not propagate to and/or affect the performance of the ECCS.

k. Non-ECCS systems interfacing with the ECCS buses shall automatically be shed from and/or be inhibited from the ECCS buses when a LOCA signal exists and offsite a-c power is not available.

l. No more than one storage battery shall be connectable to a d-c power bus.

m. Each system of the ECCS including flow rate and sensing networks must be capable of being tested during shutdown. All active components shall be capable of being tested during plant operation, including logic required to automatically initiate component action.

n. Provisions for testing the ECCS network components (electronic, mechanical, hydraulic and pneumatic, as applicable) shall be installed in such a manner that they are an integral and nonseparable part of the design.

6.3.1.1.3 ECCS Requirements for Protection from Physical Damage

The emergency core cooling system piping and components are protected against damage from movement, thermal stresses, the effects of the LOCA, and the safe shutdown earthquake.

The ECCS is protected against the effects of pipe whip which might result from piping failures up to, and including, the design-basis event LOCA. This protection is provided by separation, pipe whip restraints, or energy absorbing materials, if required. One of these three methods 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 ECCS pump room reinforced concrete walls.

In addition, the watertight construction of the ECCS pump rooms, when required, protects against mass flooding of redundant ECCS pumps.

The ECCS is capable of withstanding the passive failure of valve stem packings and pump seals following a LOCA. The maximum leakage due to a failure of this nature could be 23 gpm or less from an HPCS, LPCS, or RHR pump seal failure. Valve stem leakage would be significantly less than this.

GL 95-07 required that licensees evaluate the operational configuration of all their safety-related power operated gate valves (POVs) for susceptibility to pressure locking and/or thermal binding and to take appropriate actions to ensure that these valves are capable of performing their safety functions within the current licensing basis of the facility.

Consistent with industry/NRC discussions and feedback, the CPS scope of 95-07 valves was limited to those having an active safety function to open.

The results of the evaluation determined that several valves were susceptible to pressure locking. All of the valves identified have been modified to address pressure locking concerns.

Mechanical separation outside the drywell is achieved as follows:

a. The ECCS shall be separated into three functional groups:

   1. HPCS,

   2. LPCS + 1 LPCI + 100% service water and heat exchanger, and

   3. two LPCI pumps + 100% service water and heat exchanger.

b. The equipment in each group shall be separated from that in the other two groups. In addition, the HPCS and RCIC (which is not part of the ECCS) shall be separated within the Fuel Building, Auxiliary Building, and Containment.

c. Separation barriers are constructed between the functional groups as required to assure that environmental disturbances such as fire, pipe rupture, falling objects, etc., affecting one functional group will not affect the remaining groups. In addition, separation barriers have been provided to assure that such disturbances do not affect both the RCIC and the HPCS within the Fuel Building, Auxiliary Building, and Containment.

Flashing occurs when the vapor pressure of the liquid is equal to or greater than the total pressure of the liquid at that point. Bounding analyses have been performed for each of the ECCS suction lines that show that flashing will not occur in the ECCS suction lines. These analyses conservatively assume the suppression pool temperature is at 212°F and that the suppression pool elevation is at the post-LOCA draw down level of 727.08 ft. In addition, they assume the ECCS suction strainer is fully laden with post-LOCA debris.

6.3.1.1.4 <u>ECCS Environmental Design Basis</u>

Each emergency core cooling injection system, and the RCIC system, has a safety-related injection/isolation testable check valve located in piping within the drywell. In addition, the RCIC system has an isolation valve in the drywell portion of its steam supply piping. All valves are located above the highest water level expected in the drywell during any accident. The valves are quailified for the following environmental conditions:

a. Normal and upset plant operating ambient temperatures, relative humidities, and pressures as shown for each area of the drywell in Table 3.11-5.

b. Envelope-of-accident conditions for temperature, relative humidity, and pressure within the drywell for various time periods following the accident as shown in Table 3.11-5.

c. Normal and envelope-of-accident radiation environment as shown in Table 3.11-5.

The portions of ECCS and RCIC piping and equipment located outside the drywell and within the secondary containment are qualified for the following environmental conditions:

a. Normal and upset plant operating ambient temperatures, relative humidities, and pressures as shown in Table 3.11-5.

b. Envelope-of-accident conditions for temperature, relative humidity, and pressure for various time periods following the accident as shown in Table 3.11-5.

c. Normal and envelope-of-accident radiation environment as shown in Table 3.11-5.

6.3.1.2 Summary Descriptions of ECCS

The ECCS injection network comprises a High-Pressure Core Spray (HPCS) system, a Low-Pressure Core Spray (LPCS) system, and the Low-Pressure Coolant Injection (LPCI) mode of the residual heat removal system. Systems are briefly described here as an introduction to the more detailed system design descriptions provided in Subsection 6.3.2. The automatic depressurization system (ADS), which assists the injection network under certain conditions, is also briefly described. Boiling water reactors which employ the same ECCS design are listed in Table 1.3-1.

The LPCS system and the C loop of the LPCI system provide a connection point for makeup water to the reactor and suppression pool in a Beyond Design Basis External Event per NRC Order EA-12-049. This is not a safety function, nor is it an ECCS function. ECCS pumps are not used for this function.

6.3.1.2.1 High-Pressure Core Spray

The high-pressure core spray (HPCS) pumps water through a peripheral ring spray sparger mounted above the reactor core. Coolant is supplied over the entire range of system operation pressures. The primary purpose of the HPCS is to maintain reactor vessel inventory after small breaks which do not depressurize the reactor vessel. The HPCS also provides spray cooling heat transfer during breaks in which uncovering of the core is calculated.

6.3.1.2.2 Low-Pressure Core Spray

The LPCS is an independent loop similar to the HPCS, the primary difference being that the LPCS delivers water over the core at relatively low reactor pressures. The primary purpose of the LPCS is to provide inventory makeup and spray cooling during large breaks in which the core is calculated to uncover. Following ADS initiation, LPCS provides inventory makeup following a small break.

6.3.1.2.3 Low-Pressure Coolant Injection

LPCI is an operating mode of the residual heat removal system. Three pumps deliver water from the suppression pool to the bypass region inside the shroud through three separate reactor vessel penetrations to provide inventory makeup following large pipe breaks. Following ADS initiation, LPCI provides inventory makeup following a small break.

6.3.1.2.4 Automatic Depressurization System

The automatic depressurization system (ADS) utilizes a number of the reactor safety/relief valves to reduce reactor pressure during such small breaks in the event of HPCS failure. When the vessel pressure is reduced to within the capacity of the low-pressure system (LPCS and LPCI), these systems provide inventory makeup so that acceptable postaccident temperatures are maintained.

6.3.2 System Design

A more detailed description of the individual systems including individual design characteristics of the systems are covered in detail in Subsections 6.3.2.1 through 6.3.2.4. The following discussion will provide details of the combined systems; in particular, those design features and characteristics which are common to all systems.

Table 6.3-8 provides a listing of significant ECC system design parameters along with their design bases.

6.3.2.1 Schematic Piping and Instrumentation Diagrams

The P&IDs for the ECCS are identified in Subsection 6.3.2.2. The process diagrams which identify the various operating modes of each system are also identified in Subsection 6.3.2.2.

6.3.2.2 Equipment and Component Descriptions

The starting signal for the ECCS comes from at least two independent and redundant sensors of drywell pressure and low reactor water level. The ECCS is actuated automatically and requires no operator action during the first 10 minutes following an accident. A time sequence for starting of the systems is provided in Table 6.3-1.

Electric 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. One LPCI loop and the LPCS loops switch to one site backup power supply, and the other two LPCI loops switch to a second site backup power supply. Section 8.3 contains a more detailed description of the power supplies for the ECCS.

a. Regulatory Guide 1.82, Water Sources for Long-Term Recirculation Cooling Following a Loss-of-Coolant Accident

   General Compliance or Alternate Approach Assessment

The design of the large toroidal passive ECCS suction strainer was evaluated against the regulatory positions contained in Reg. Guide 1.82, Rev. 2.

The suction strainer is designed to preclude the potential for loss of NPSH caused by debris blockage during the period that the ECCS is required to maintain long-term cooling. The large toroidal passive strainer design results in a very low approach velocity for water entering the strainer. Debris collected on the strainer surface is not expected to compact significantly (due to the very low approach velocity), resulting in minimal head loss. A 1/4-scale model of the strainer design was tested with significant fiber debris loading to confirm the performance of the strainer and the behavior of the postulated fiber debris bed as a function time after the postulated LOCA. Because the fiber debris bed will not be significantly compacted, flow continues to pass through the debris (and the strainer) and thus the overall differential pressure will remain low. CPS uses essentially 100% RMI in the drywell. Given the large surface area available in the strainer, reaching a saturation bed thickness of RMI is not considered credible. Head losses due to RMI in the suppression pool on the strainer have been evaluated and are considered to be negligible. Maintaining a low differential pressure will ensure adequate NPSH for the ECCS pumps.

The size of the openings in the suppression pool suction strainer material has been chosen based on the minimum restrictions found in systems served by the suppression pool.

The ECCS pump suctions are designed to prevent degradation of pump performance through air ingestion and other adverse hydraulic effects. All of the suction piping remains below the surface of the suppression pool, and due to the very low approach velocity design of the strainer and the depth of the strainer in the pool, vortexing will not be present. The strainer is located at the bottom of the suppression pool, below the elevation of the S/RV quencher arms. Because of the location of the strainer, no encroachment into the recommended exclusion zone around the quenchers occurs. Additionally, the new strainer design is such that any air may enter the strainer would be released through the strainer mesh before traveling to the pump suction plenums; that is, air entrainment in the strainer will be minimized. Therefore, given the design and arrangement of the proposed strainer, air ingestion into the strainer and piping system will not occur.

The strainer does not involve any modification in the arrangement of drains from upper floors in the containment. There are no floor or equipment drains from the containment or reactor building that drain directly into the suppression pool. In addition, the strainer is located at the bottom of the suppression pool such that it is highly unlikely that any debris from drains from the upper regions of the containment could impinge directly on the suction strainer. The Suppression Pool Makeup dump lines discharge upper containment pool water into the suppression pool following a LOCA. The upper containment pools are maintained in a clean condition by operation of the Fuel Pool Cleanup system and through CPS's foreign material exclusion program. Therefore, debris from these lines will be limited to any corrosion products present in the dump lines carried along by the dump flow.

The suction strainer is designed such that its support structure will protect it from the effects of large debris. The strainer is designed so that it is capable of withstanding LOCA-induced hydrodynamic loads. CPS utilizes GESSAR II methods combined with acoustic methodology which demonstrates that the strainer can withstand the LOCA-induced hydrodynamic loads. Missile protection was evaluated and determined to be of no concern based on the location of the strainer and the postulated missile sources for CPS.

The suction strainer was evaluated and shown to be able to withstand loads associated with design basis seismic events without loss of structural integrity. In addition, the design incorporates provisions so that bolts do not lose torque during any vibratory motion, and incorporates restraints to preclude radial and axial movement of the strainer.

ASTM Type 304 stainless steel is used as the primary material for ECCS suction strainer to prevent corrosive degradation during periods of inactivity and normal operation.

CPS has established a containment cleanliness program for the control of foreign materials, and other programs to minimize the potential for strainer fouling from operations generated debris.

CPS takes no credit for LOCA generated debris hold up in the drywell, and the design does not include debris interceptors of any kind.

The strainer design requires no operator actions in response to debris accumulations or to otherwise assure availability of adequate NPSH for the ECCS pumps. Accordingly, no additional safety-related instrumentation is required.

The design of the ECCS suction strainer is passive.

CPS conducts comprehensive inspections during refueling outages to evaluate the cleanliness of the suppression pool. The strainer is periodically monitored by visual inspection for evidence of structural degradation or debris fouling. The frequency of suppression pool inspection and cleaning activities is determined based on plant specific debris collection data.

The large toroidal passive strainer does not require operator actions to prevent the accumulation of debris on the strainer or to mitigate the consequences of debris accumulation. The design of the strainer provides sufficient area to accomodate the maximum quantity of debris that is expected to be produced following a design basis LOCA combined with postulated in situ debris quantities. The Emergency Procedures contain guidance to the operator on the use of alternate water sources to provide a diverse means of providing long-term cooling to the core.

See USAR Section 6.2.2.2 for a discussion of the ECCS suction strainer compliance with Reg. Guide 1.82 as it pertains to debris generation and transport.

NPSH available to the ECCS pumps has been determined in accordance with Reg. Guide 1.1. Pressure drop across the suction strainer is based on results from testing and conservative analysis. The vapor pressure for suppression pool water used in NPSH calculations for events where significant debris generation is expected is based on a suppression pool bulk water temperature of 185° F, which is the maximum design temperature of the containment. Analyses show maximum suppression pool temperatures to be less that the containment design tempertature of 185° F. For events in which no significant debris generation is expected, NPSHA will continue to be evaluated for 212° F suppression pool water temperature. Containment pressure is assumed to be atmospheric in accordance with Reg. Guide 1.1 requirements.

Tests have quantified head loss caused by debris blockage on the strainer. Head loss measured during the testing accounts for the possible filtration of particulates by the debris bed. Tests were conducted to determine the performance characteristics of the passive strainer for the quantities and types of debris predicted following postulated accidents.

6.3.2.2.1 High-Pressure Core Spray (HPCS) System

The high-pressure core spray (HPCS) system consists of a single motor-driven centrifugal pump located outside the primary containment, a spray sparger in the reactor vessel located above the core (separate from the LPCS sparger), and associated system piping, valves, controls and instrumentation. The system is designed to operate from normal offsite auxiliary power or from a standby diesel-generator supply if offsite power is not available. The piping and instrumentation diagram, M05-1074, for the HPCS shows the system components and their arrangement. The HPCS system process diagram, 762E454, shows the design operating modes of the system. A simplified system flow diagram showing system injection into the reactor vessel is included in 762E454. Significant HPCS design parameters are provided on Table 6.3-8.

The principal active HPCS equipment is located outside the primary containment. Suction piping is provided from the RCIC storage tank and the suppression pool. Such an arrangement provides the capability to use reactor grade water from the RCIC storage tank when the HPCS system functions to back up the RCIC system. In the event that the RCIC storage water supply becomes exhausted or is not available, automatic switchover to the suppression pool water source will assure a closed cooling water supply for continuous operation of the HPCS system. HPCS pump suction is also automatically transferred to the suppression pool if the suppression pool water level exceeds a prescribed value. The RCIC storage tank reserves water for use only by the HPCS and RCIC.

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 radically over the core and into the fuel assemblies.

The HPCS discharge line to the reactor is provided with two isolation valves. One of these valves is a testable check valve located inside the drywell as close as practical to the reactor vessel. HPCS injection flow causes this valve to open during LOCA conditions (i.e., no power is required for valve actuation during LOCA). If the HPCS line should break outside the containment, the check valve in the line inside the drywell will prevent loss of reactor water outside the containment. The other isolation valve (which is also referred to as the HPCS injection valve) is a motor-operated gate valve located outside the primary containment as close as practical to HPCS discharge line penetration into the containment. This valve is capable of opening with the maximum differential pressure across the valve expected for any system operating mode including HPCS pump shutoff head. This valve is normally closed to back up the inside testable check valve for containment integrity purposes. A drain line is provided between the two valves. The test connection line is normally closed with two valves to assure containment integrity.

Remote controls for operating the motor-operated components and diesel generator are provided in the main control room. The controls and instrumentation of the HPCS system are described, illustrated, and evaluated in detail in Chapter 7.

The location and type of the manual valves in the HPCS system are detailed in Table 6.3-10 (see also Drawing M05-1074). Design considerations have been given to protect the system's safety functions from an undetected, incorrect positioning of any of these manual valves. Administrative controls likewise serve to minimize the possibility of such errors. These design/operations features are outlined in the table (see also Subsection 6.3.2.8).

The system is designed to pump water into the reactor vessel over a wide range of pressures. For small breaks that do not result in rapid reactor depressurization, the system maintains reactor water level and depressurizes the vessel. For large breaks, the HPCS system cools the core by a spray.

If a loss-of-coolant accident should occur, a low-water level signal, or a high drywell pressure signal, initiates the HPCS and its support equipment. The system can also be placed in operation manually.

When a high-water level in the reactor vessel is signaled, the HPCS is automatically stopped by a signal to the injection valve to close. The HPCS system also serves as a backup to the RCIC system in the event the reactor becomes isolated from the main condenser during operation and feedwater flow is lost. If normal auxiliary power is not available, the HPCS pump motor is driven by its own onsite power source. The HPCS standby power source is discussed in Section 8.3.

The HPCS pump head flow characteristic used in LOCA analyses is shown in Figure 6.3-3. 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* 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 both the RCIC storage tank and the suppression pool. This assures a flooded pump suction. At a predetermined low water level in the RCIC storage tank, the HPCS pump suction automatically transfers from this tank to the suppression pool. The available NPSH is greater than 41 feet. The required NPSH for the HPCS pump, as shown on Drawing 762E454, is 5 feet. Vortex formation in the RCIC storage tank at the suction piping entrance has been taken into consideration in determining the setpoint for automatic transfer.

Pump NPSH requirements are met by providing adequate suction head and suction line size. The available NPSH, calculated in accordance with Regulatory Guide 1.1, is based on the following conditions:

a. pump runout flow of 6,400 gpm,

b. atmospheric containment pressure,

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* psid = differential pressure between the reactor vessel and the suction source.

c. maximum suppression pool water temperature of 185° F,

d. suppression pool post-LOCA drawdown water level of elevation 727'-1",

e. suction piping losses, and

f. suction strainer fully loaded (i.e., conservatively specified debris loading resulting from LOCA-generated and pre-LOCA debris materials).

The calculated minimum available NPSH for the HPCS pump taking suction from the suppression pool is 26.1 feet which exceeds the minimum required NPSH of 5 feet.

A motor-operated valve is provided in the suction line from the suppression pool. The valve is located as close to the suppression pool penetration as practical. This valve is used to isolate the suppression pool water source when HPCS system suction is from the RCIC storage tank, and to isolate the system from the suppression pool in the event a leak develops in the HPCS System.

The HPCS pump characteristics are shown in Figure 6.3-79.

The design pressure and temperature of the system components are established based on the ASME Section III Boiler and Pressure Vessel Code. The design pressures and temperatures at various points in the system can be obtained from the miscellaneous information blocks on the HPCS process diagram (see Drawing 762E454).

A check valve, flow element, and restricting orifice are provided in the HPCS discharge line from the pump to the injection valve. The check valve is located below the minimum suppression pool water level and is provided so the piping downstream of the valve can be maintained full of water by the discharge line fill system (see Subsection 6.3.2.2.5). The flow element is provided to measure system flow rate during LOCA and test conditions, and for automatic control of the minimum low flow bypass gate valve. The measured flow is indicated in the main control room. The restricting orifice is sized during the preoperational test of the system to limit system flow to acceptable values as described in the HPCS system process diagram.

A low flow bypass line with a motor-operated gate valve connects to the HPCS discharge line upstream of the check valve on the pump discharge line. The line bypasses water to the suppression pool to prevent pump damage to overheating when other discharge line valves are closed. The valve automatically closes when flow in the main discharge line is sufficient to provide required pump cooling.

To assure continuous core cooling, signals to isolate the containment do not operate any HPCS valves.

The HPCS system incorporates relief valves to protect the components and piping from inadvertent overpressure conditions. One relief valve, set to relieve at 1560 psig, is located on the discharge side of the pump downstream of the check valve to relieve thermally expanded fluid. A second relief valve is located on the suction side of the pump and is set at 100 psig with a capacity of ≥ 10 gpm at 10% accumulation.

The HPCS components and piping are positioned to avoid damage from the physical effects of design-basis accidents, such as pipe whip, missiles, high temperature, pressure, and humidity.

The HPCS equipment and support structures are designed in accordance with Seismic Category I criteria (see Chapter 3). The system is assumed to be filled with water for seismic analysis.

Provisions are included in the HPCS system which will permit the HPCS system to be tested. These provisions are:

a. All active HPCS components are testable during normal plant operation.

b. A full flow test line is provided to route water from and to the RCIC storage tank without entering the reactor pressure vessel. The suction line from the RCIC storage tank also provides reactor grade water to fully test the HPCS including injection into the reactor pressure vessel during shutdown.

c. A full flow test line is provided to route water from and to the suppression pool without entering the reactor pressure vessel.

d. Instrumentation is provided to indicate system performance during normal test operations.

e. All motor-operated valves are capable of either local or remote manual operation for test purposes.

f. System relief valves are removable for bench-testing during plant shutdown.

6.3.2.2.2 Automatic Depressurization System (ADS)

If the RCIC and HPCS cannot maintain the reactor water level, the automatic depressurization system (ADS), which is independent of any other ECCS, reduces the reactor pressure so that flow from LPCS and LPCI 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, operational characteristics, and evaluation of the pressure relief valves are discussed in detail in Subsection 5.2.2. The operation of the ADS is discussed in Subsection 7.3.1.1.1.4.

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 M05-1073, the LPCS system P&ID presents the system components and their arrangement. The LPCS system process diagram, Drawing 762E467AC, shows the design operating modes of the system. Drawing 762E467AC includes a simplified system flow diagram showing injection into the reactor vessel by the LPCS system. Significant LPCS design parameters are provided on Table 6.3-8.

When low water level in the reactor vessel or high-pressure in the drywell is sensed, and with reactor vessel pressure low enough, the low-pressure core spray system automatically starts and sprays water into the top of the fuel assemblies to cool the core. The LPCS injection piping enters the vessel, divides and enters the core shroud at two points near the top of the shroud. A semicircular sparger is attached to each outlet. Nozzles are spaced around the sparger to spray the water radially over the core and into the fuel assemblies.

The LPCS is designed to provide cooling to the reactor core only when the reactor vessel pressure is low, as is the case for large LOCA break sizes. However, when the LPCS operates in conjunction with the ADS then the effective core cooling capability of the LPCS is extended to all break sizes because the ADS will rapidly reduce the reactor vessel pressure to the LPCS operating range.

The system head vs. flow characteristic assumed in LOCA analyses for the LPCS pump is shown in Figure 6.3-6.

The low-pressure core spray pump and all motor-operated valves can be operated individually by manual switches located in the control room. Operating indication is provided in the control room by a flowmeter and valve indicator lights.

The location and type of the manual valves in the LPCS system are detailed in Table 6.3-11 (see also Drawing M05-1073). Design considerations have been given to protect the system's safety functions from an undetected, incorrect positioning of any of these manual valves. Administrative controls likewise serve to minimize the possibility of such errors. These design/operations features are outlined in the table (see also Subsection 6.3.2.8).

To assure continuity of core cooling, signals to isolate the containment do not operate any low-pressure core spray system valves.

The LPCS discharge line to the reactor is provided with two isolation valves. One of these valves is a testable check valve located inside the drywell as close as practical to the reactor vessel. LPCS injection flow causes this valve to open during LOCA conditions (i.e., no power is required for valve actuation during LOCA). If the LPCS line should break outside the containment the check valve in the line inside the drywell will prevent loss of reactor water outside the containment.

The other isolation valve (which is also referred to as the LPCS injection valve) is a motor-operated gate valve located outside the primary containment as close as practical to LPCS discharge line penetration into the containment. This valve is capable of opening with the maximum differential across the valves expected for any system operating mode. This valve is normally closed to back up the inside testable check valve for containment integrity purposes. A drain line is provided between the two valves. The test connection line is normally closed with two valves to assure containment integrity.

The LPCS system components and piping are arranged to avoid unacceptable damage from the physical effect of design-basis accidents, such as pipe whip, missiles, high temperature, pressure and humidity.

All principal active LPCS equipment is located outside the primary containment.

A check valve, flow element and restricting orifice are provided in the LPCS discharge line from the pump to the injection valve. The check valve is located below the minimum suppression pool water level and is provided so the piping downstream of the valve can be maintained full of water by the discharge line fill system (see Subsection 6.3.2.2.5). The flow element is provided to measure system flow rate during LOCA and test conditions and for automatic control of the minimum low flow bypass gate valve. The measured flow is indicated in the main control room. The restricting orifice is sized during the preoperational test of the system to limit system flow to acceptable values as described on the LPCS system process diagram.

The LPCS pump (pump performance test results) characteristics, head, flow, horsepower, and required NPSH are shown in Figure 6.3-80.

A low flow bypass line with a motor-operated gate valve connects to the LPCS discharge line upstream of the check valve on the pump discharge line. The line bypasses water to the suppression pool to prevent pump damage due to overheating when other discharge line valves are closed or reactor pressure is greater than the LPCS system discharge pressure following system initiation. The valve automatically closes when flow in the main discharge line is sufficient to provide required pump cooling.

LPCS flow passes through a motor-operated pump suction valve that is normally open. This valve can be closed by a remote manual switch (located in the control room) to isolate the LPCS system from the suppression pool should a leak develop in the system. This valve is located in the core spray pump suction line as close to the suppression pool penetration as practical. Because the LPCS conveys water from the suppression pool, a closed loop is established for the spray water escaping from the LOCA-causing break.

The design pressure and temperature of the system components are established based on the ASME Section III Boiler and Pressure Vessel Code. The design pressures and temperatures at various points in the system can be obtained from the miscellaneous information blocks on the LPCS process diagram, 762E467AC.

The LPCS pump is located in the auxiliary building sufficiently below the water level in the suppression pool to assure a flooded pump suction and to assure that pump NPSH requirements are met with the containment at atmospheric pressure and the suction strainers fully loaded (i.e., conservatively specified debris loading resulting from LOCA-generated and pre-LOCA debris materials). A pressure gauge is provided to indicate the suction head. The LPCS pump characteristics are shown in Figure 6.3-6.

The available NPSH, calculated in accordance with Regulatory Guide 1.1, is based on the following conditions:

a. Pump runout flow of 6450 gpm,

b. Atmospheric containment pressure,

c. Maximum suppression pool water temperature of 185° F,

d. Suppression pool post-LOCA drawdown level of El. 727'-1",

e. LPCS pump suction piping losses, and

f. Suction strainer fully loaded (i.e., conservatively specified debris loading resulting from LOCA-generated and pre-LOCA debris materials).

The calculated minimum available NPSH for the LPCS pump is 26.4 feet which exceeds the minimum required NPSH of 5 feet. The LPCS pump characteristics are shown on Figure 6.3-6.

The LPCS system incorporates relief valves to prevent the components and piping from inadvertent overpressure conditions. One relief valve, located on the pump discharge, is set at 600 psig minimum elevation difference (in psi) between this valve and the discharge line check valve located upstream of the injection valve with the capacity of 100 gpm at 10% accumulation. The second relief valve is located on the suction side of the pump and is set for 100 psig at a capacity of >10 gpm at 10% accumulation.

The LPCS system piping and support structures are designed in accordance with Seismic Category I criteria (see Chapter 3). The system is assumed to be filled with water for seismic analysis.

Provisions are included in the LPCS system which will permit the LPCS system to be tested. These provisions are:

a. All active LPCS components are testable during normal plant operation, except the injection valve and check valve in the drywell which are testable when shutdown.

b. A full flow test line is provided to route water from and to the suppression pool without entering the reactor pressure vessel.

c. A suction test line supplying reactor grade water, is provided to test pump discharge into the reactor pressure vessel during normal plant shutdown.

d. Instrumentation is provided to indicate system performance during normal and test operations.

e. All motor-operated valves and check valves are capable of operation for test purposes.

f. Relief valves are removable for bench-testing during plant shutdown.

6.3.2.2.4 Low-Pressure Coolant Injection (LPCI) System

The low-pressure coolant injection subsystem is an operating mode of the RHR system. The LPCI system is automatically actuated by low-water level in the reactor or high-pressure in the drywell and uses the three RHR motor-driven pumps to draw suction from the suppression pool and inject cooling water flow into the reactor core and accomplish cooling of the core by flooding. Each loop has its own suction and discharge piping and separate vessel nozzle which connects with the core shroud through the LPCI couplings to deliver flooding water near the top of the core. The system is a high volume core flooding system.

The LPCI system, like the LPCS system, is designed to provide cooling to the reactor core only when the reactor vessel pressure is low, as is the case for large LOCA break sizes. However, when the LPCI operates in conjunction with the ADS then the effective core cooling capability of the LPCI is extended to all break sizes because the ADS will rapidly reduce the reactor vessel pressure to the LPCI operating range. The head flow characteristics assumed in the LOCA analyses for the LPCI pumps are shown in Figure 6.3-7.

Drawing 762E425AC shows a process diagram and process data for the RHR system, including LPCI. Significant LPCI design parameters are provided on Table 6.3-8. 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 (see Section 8.3).

The pump, piping, controls and instrumentation of the LPCI loops are separated and protected so that any single physical event, or missiles generated by rupture of any pipe in any system within the drywell, cannot make all loops inoperable.

To assure continuity of core cooling, signals to isolate the primary containment do not operate any RHR system valves which interfere with the LPCI mode of operation.

Each LPCI discharge line to the reactor is provided with two isolation valves. The valve inside the drywell is a testable check valve and the valve outside the drywell is a motor-operated gate valve. No power is required to operate the check valve inside of the drywell; rather, it opens as a result of LPCI injection flow. If a break were to occur outboard of the check valve it would shuttle closed to isolate the reactor from the line break.

The motor-operated valve outside of the drywell is called the LPCI injection valve and is located as close as practical to the drywell wall. It is capable of opening against the maximum differential expected for the LPCI modes; i.e., normal reactor pressure minus the upstream pressure with the RHR pump running at minimum flow.

The process diagram, 762E425AC, and the P&ID, M05-1075, indicate a great many flow paths are available other that the LPCI injection line. However, the low-water level or high drywell pressure signals which automatically initiate the LPCI mode are also used to isolate all other modes of operation and revert other system valves to the LPCI lineup (except shutdown cooling). Inlet and outlet valves from the heat exchangers receive no automatic signals as the system is designed to provide rated flow to the vessel whether they are open or not.

A check valve in the pump discharge line is used together with a discharge line fill system (see Subsection 6.3.2.2.5) to prevent water hammer resulting from pump start against a potential shutoff condition. A flow element in the pump discharge line is used to provide a measure of system flow and to originate automatic signals for control of the pump minimum flow valve. The minimum flow valve permits a small flow to the suppression pool in the event no discharge valve is open: or in the case of a LOCA, vessel pressure is higher than pump shutoff head.

Using the suppression pool as the source of water for the LPCI establishes a closed loop for recirculation of LPCI water escaping from the LOCA-causing break.

The design pressure and temperatures, at various points in the system, during each of the several modes of operation of the RHR subsystems, can be obtained from the miscellaneous information blocks on the LPCI process diagram, Drawing 762E425AC

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 heat exchangers are discussed in Subsection 6.2.2. The portions of the RHR required for accident protection including support structures are designed in accordance with Seismic Category I criteria (see Chapter 3). The LPCI pump characteristics are shown in Figure 6.3-7.

The LPCI system incorporates a relief valve on each of the pump discharge lines which protects the components and piping from inadvertent overpressure conditions. These valves are set to relieve pressure at 500 psig (minus elevation difference between this valve and pump discharge).

Provisions are included in the LPCI system to permit testing of the system. These provisions are:

a. All active LPCI components are designed to be testable during normal plant operation, except the injection valves, which are testable when shut down.
b. A discharge test line is provided for the three pump loops to route suppression pool water back to the suppression pool without entering the reactor pressure vessel.
c. A suction test line, supplying reactor grade water, is provided to test loop "C" discharge into the reactor pressure vessel during normal plant shutdown.
d. Instrumentation is provided to indicate system performance during normal and test operations.
e. All motor-operated valves, air-operated valves, and check valves are capable of manual operation for test purposes.
f. Shutdown lines taking suction from the recirculation system are provided for loops "A" and "B" to test pump discharge into the reactor pressure vessel after normal plant shutdown and to provide for shutdown cooling.
g. All relief valves are removable for bench-testing during plant shutdown.

6.3.2.2.5 ECCS Discharge Line Fill System

A requirement of the core cooling systems is that cooling water flow to the reactor vessel be initiated rapidly when the system is called on to perform its function. This quick-start system characteristic is provided by quick-opening valves, quick-start pumps, and a standby a-c power source. The lag between the signal to start the pump and the initiation of flow into the RPV can be minimized by keeping the core cooling pump discharge lines full. Additionally, if these lines were empty when the systems were called for, the large momentum forces associated with accelerating fluid into a dry pipe could cause physical damage to the piping. Therefore, the ECCS discharge line fill system is designed to maintain the pump discharge lines in a condition that is sufficiently full to accomplish this safety function.

Since the ECCS discharge lines are elevated above the suppression pool, check or stop-check valves are provided near the pumps to prevent back flow from emptying the lines into the suppression pool. Past experience has shown that these valves will leak slightly, producing a small back flow that will eventually empty the discharge piping. To ensure that this leakage from the discharge lines is replaced and the lines are always kept sufficiently filled, a water leg pump is provided for each of the three ECCS divisions. The power supply to these pumps is classified as essential when the main ECCS pumps are deactivated. The fill system, typical in principle and operation for each ECCS loop, consists of a water leg pump that takes suction from its corresponding ECCS pump suction line from the suppression pool and discharges downstream of the check valve on the ECCS pump discharge line. The piping and instrumentation diagrams for the fill systems are shown on Drawings M05-1073, M05-1074, and M05-1075 (Sheet 3).

Each water leg pump head and capacity are selected so that each ECCS pump discharge line can be maintained full up to the highest location on the discharge line, assuming all the valves on the pump discharge are leaking at the maximum expected rate. To prevent overheating of the water leg pump if the discharge line valves do not leak, a low-flow bypass line is provided to continuously circulate water back to the ECCS pump suction lines or suppression pool. Indication is provided in the main control room as to whether each pump is operating. Pressure instrumentation is provided on each water leg pump discharge line to initiate an alarm in the main control room when pressure in the discharge line is less than the hydrostatic head required to maintain the line full of water up to the injection valves.

6.3.2.2.6 Gas Management Program

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 21). 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 are identified.

The Clinton Power Station response to Generic Letter 2008-01 is contained in a letter from K. R. Jury to U. S. NRC, “Nine-Month Response to Generic Letter 2008-01,” dated October 14, 2008, RS-08-131 (Reference 22) and in a letter from Patrick R. Simpson to U. S. NRC, “Response to Request for Additional Information Regarding Generic Letter 2008-01,” dated December 15, 2009, RS-09-173 (Reference 23).

Clinton Technical Specification Surveillance Requirements for Shutdown Cooling, ECCS, RCIC, Containment Spray, and Suppression Pool Cooling have been revised to require gas management, which will maintain air less than the amount that challenges Operability. This Technical Specification change was submitted in accordance with Technical Specification Task Force Traveler (TSTF-523).

6.3.2.3 Applicable Codes and Classifications

The applicable codes and classification of the ECCS are specified in Section 3.2. All piping systems and components (pumps, valves, etc.) for the ECCS comply with applicable codes, addenda, code cases, and errata in effect at the time the equipment is procured. The piping and components of each ECCS within the containment and out to and including the pressure retaining injection valve are Safety Class 1. The RHR LPCI loop A&B injection valves are inside the containment. The remaining piping and components are Safety Class 2, 3, or non-code as system P&ID. The equipment and piping of the ECCS are designed to the requirements of Seismic Category I. This seismic designation applies to all structures and equipment's essential to the core cooling function. IEEE codes applicable to the controls and power supplies are specified in Section 7.1.

6.3.2.4 Materials Specifications and Compatibility

Materials specifications and compatibility for the ECCS are presented in Sections 6.1 and 3.2. Nonmetallic materials such as lubricants, seals, packing, paints and primers, insulation, as well as metallic materials, etc., are selected as a result of a review and evaluation for compatibility with other materials in the system and the surroundings, with concern for chemical, radiolytic, mechanical and nuclear effects. Materials used are reviewed and evaluated with regard to radiolytic and pyrolytic decomposition and attendant effects on safe operation of the ECCS.

6.3.2.5 System Reliability

A single failure analysis shows that no single failure prevents the starting of the ECCS when required, or the delivery of coolant to the reactor vessel. No individual system of the ECCS is single failure proof with the exception of the ADS, hence it is expected that single failures will disable individual systems of the ECCS. The most severe effects of single failures with respect to loss of equipment occur if a loss-of-coolant accident occurs in combination with an ECCS pipe break coincident with a loss of offsite power. The consequences of the most severe single failures are shown in Table 6.3-7.

A system level limiting mode failure analysis is presented in Appendix 15A, Nuclear Safety Operational Analysis, for all systems.

6.3.2.6 Protection Provisions

Protection provisions are included in the design of the ECCS. Protection is afforded against missiles, pipe whip, and flooding. Also accounted for in the design are thermal stresses, loading from a LOCA, and seismic effects.

The ECCS piping and components located outside the drywell are protected from internally and externally generated missiles by the ECCS pump room reinforced concrete walls. The HPCS piping and components between the RCIC Storage Tank and the Fuel Building are not provided with protection in accordance with section 3.5.2.4. The watertight construction of these ECCS pump rooms also protects the equipment against flooding. The layout of the ECCS pump rooms is described in Subsection 6.2.3.

The ECCS is protected against the effects of pipe whip, which might result from piping failures up to and including the design basis event LOCA. This protection is provided by separation, pipe whip restraints, and energy absorbing materials. These three methods are applied to provide protection against damage to piping and components of the ECCS which otherwise could result in a reduction of ECCS effectiveness to an unacceptable level. See Section 3.6 for criteria on pipe whip.

The component supports which protect against damage from movement and from seismic events are discussed in Subsection 5.4.14. The methods used to provide assurance that thermal stresses do not cause damage to the ECCS are described in Subsection 3.9.3.

The leak detection capability for the ECCS is discussed in Subsections 5.2.5 and 7.6.1.4. Loss of any one train of an ECCS will not negate the function of the ECCS. Flooding of one ECCS pump room cannot cause flooding of a redundant ECCS pump room.

A leak in any ECCS train outside these rooms would be detected by the leak detection methods described in Subsections 5.2.5 and 7.6.1.4. Any ECCS train which is found to have excessive leakage can be isolated and a redundant train initiated.

All leakage in the ECCS rooms is pumped to the auxiliary building floor tank drain from which the leakage is processed through the liquid radwaste system (refer to Drawings M05-1081, M05-1085 and M05-1086). For the operation of the floor and equipment drainage system and the liquid radwaste system, refer to Subsection 9.3.3 and Section 11.2.

The standards met by the leak detection system are described in Subsection 7.6.2.4.2.

The capability of the leak detection system to detect passive failures such as pump seals, valve seals, and measurement devices is described in Subsections 5.2.5 and 7.6.2.4.1.

The RHR, LPCS, and HPCS pumps are designed for the life of the plant (40 years) and tested for operability assurance and performance as follows:

a. In-shop tests including (1) hydrostatic tests of pressure retaining parts of 150% times the design pressure, (2) performance tests while the pump is operated with flow to determine the total developed head at zero and design flow, and (3) net positive suction head (NPSH) requirements.

b. After the pump is installed in the plant, it undergoes (1) the system hydro tests, (2) functional tests, (3) the required periodic inservice inspection of once every three months for a minimum of five minutes during normal plant operation, and (4) postulated 1 month of operation each year for shutdown cooling (RHR pumps only).

c. Plus designed for a postulated single operation of 100 day for one accident during the unit's 40-year life.

The following shows the maximum expected accumulated operating time for the life of the plant (40 years):

Mode of Operation	RHR	LPCS/HPCS
1. In-shop test	4 (hours)	4 (hours)
2. Preoperation	168	168
3. Monthly Testing	480	480
4. Yearly Testing	40	40
5. Post-LOCA	2400	2400
6. Shutdown	28800	N/A
Total	31892	3092

In order to verify these pumps will satisfy long-term operational requirements, GE was contracted to perform a 150 hour test on one of the ECCS pumps.

6.3.2.7 Provisions for Performance Testing

Periodic system and component testing provisions for the ECCS are described in Subsection 6.3.2.2 as part of the individual system descriptions.

6.3.2.8 Manual Actions

The ECCS is actuated automatically and requires no operator action during the first 10 minutes following the accident. During the long-term cooling period (after 10 minutes), the operator will take action as specified in Subsection 6.2.2.2 to place the containment cooling system into operation. Placing the containment cooling system into operation is the only manual action that the operator needs to accomplish during the course of the LOCA.

The operator has multiple instrumentation available in the control room for assistance in assessing the post-LOCA conditions. This instrumentation provides reactor vessel pressures, water levels, containment pressure, temperature and radiation levels as well as indicating the operation of the ECCS. ECC system flow indication is the primary parameter available to assess proper operation of the system. Other indications such as the position of valves, status of circuit breakers, and essential power bus voltage are also available to assist the operator in determining systems operating status. The electrical and instrumentation complement to the ECCS is discussed in detail in Section 7.3. Other available instrumentation is listed in the P&ID for the individual system. Much of the monitoring instrumentation available to the operator is discussed in more detail in Chapter 5 and Section 6.2.

The manual valves in the ECCS that could block system flow if placed in the incorrect (closed) position and their locations are as follows:

System	Valve No.	Location
RHR	E12-F039A-C	Drywell
RHR	E12-F029A-C	Auxiliary Building
RHR	E12-F304A-C	Auxiliary Building
LPCS	E21-F007	Drywell
LPCS	E21-F301	Auxiliary Building
LPCS	E21-F309	Auxiliary Building
HPCS	E22-F036	Drywell
HPCS	E22-F301	Fuel Building
HPCS	E22-F314	Fuel Building
HPCS	E22-F318	RCIC Storage Tank Room

The valves located in the drywell are provided with position-indicating lights located in the control room. The valves in the auxiliary building, fuel building, and RCIC Storage Tank Room are administratively controlled to ensure proper alignment. The incorrect positioning of ECCS valves (except the Manual Shutoff Valves in the drywell) would also be detected during normal plant operation when ECCS flow capacity verification tests are periodically conducted.

The position of each manually operated valve will be identified in a valve lineup sheet. System operating instructions for returning systems to operable status evaluates the need for a valve lineup to be completed before the system can be considered operable. For ECCS safety-related systems/components, this lineup will have independent verification, or concurrent verification for throttled valves with a pre-determined valve position. Other ECCS lines will receive alternate means of confirmation. Where appropriate, valves will be locked in the designated position to prevent inadvertent repositioning.

If valve positions are to be changed for surveillance purposes, the surveillance procedure will have steps requiring return to normal valve lineup prior to completion. Start and completion of surveillance procedures will be logged in the Unified Control Room Log (UCRL) maintained by shift personnel.

If maintenance is performed on a safety-related system that requires any valve position to be changed from that specified in the valve lineup, the following sequence will occur:

1. The request for the valve position change will be approved by the Tagging Authority before it is implemented. The Tagging Authority will assure that Technical Specifications, Operational Requirements Manual (ORM), and Off-site Dose Calculation Manual (ODCM) requirements are met as a result of the change.

2. A list of valves and/or boundaries of valves so changed will be kept in a file or the UCRL accessible to Shift Management. The change in status of any safety-related system from operable to inoperable or vice versa will be logged in the UCRL that will be reviewed by each oncoming operator at the controls and by Shift Management.

3. When work has been completed, the order to return all valves to their previous position will be approved by Shift Management.

4. Shift Management will not consider the system operable until all valves identified within the boundaries of the maintenance activities have been returned to the position specified in the valve lineup, and written evidence to this effect has been presented to him.

If valve positions in ECCS are changed for operational purposes (venting, filling, rotation of equipment, etc.), these changes will be made in accordance with procedures having similar adequate administrative controls to assure that:

1. Valves will be returned to the valve lineup position before the operational activity has been completed, or

2. Valve positions not in accordance with the valve lineup are known by Shift Management.

The maximum allowable time for operator action is the time required to drain the suppression pool to a level below that required to maintain the design minimum suppression pool water coverage of 2 feet over the top of the top drywell vent. The suppression pool water level margin available is 1 inch or 4,500 gallons. Based on a postulated seal failure for one of the ECCS pumps at a rate of 23 gpm, the leakage would be detected within 35 minutes upon receipt of an alarm in the control room from the leakage detection system (see Subsection 9.3.3). Following detection of the leakage, operator action to isolate the faulted ECCS loop would be required within approximately 2-1/2 hours to limit the total leakage volume to less than the 4,500 gallons.

The SRP 6.3 does not allow credit for operator action for 20 minutes following a loss-of-coolant accident (LOCA).

Issuance of the Standard Review Plans (SRP) post-dates the Clinton Power Station licensing docket by more than 2 years. Therefore, no attempt was made to design the plant to the requirements of the SRP. The CPS FSAR was prepared using Revision 3 of Regulatory Guide 1.70 as much as practical for a plant of its vintage, with assurance from NRC management that compliance with this regulatory guide assured submittal of all necessary licensing information.

As documented in a letter of August 5, 1977 from G. G. Sherwood to E. G. Case of the NRC, the SRP constitute a substantial increase in the information required just to describe the degree of compliance of various systems. This increase in turn represents a substantial resource expenditure which is unjustified and which could cause project delays if required of these projects. As stated in the reference letter, General Electric believes that SRP should be applied to the FSAR only to the extent they were required in the PSAR.

CPS believes the above position, which is the essence of a directive from Ben C. Rusche, Director of Nuclear Reactor Regulation, to the NRC Staff, dated January 31, 1977, is the appropriate procedure for review of the CPS FSAR.

6.3.3 ECCS Performance Evaluation

This section provides the results of the loss-of-coolant accident (LOCA) analysis for CPS. The analysis was performed using the SAFER/GESTR-LOCA Application Methodology approved by the Nuclear Regulatory Commission (NRC) (Reference 2).

This LOCA analysis was performed in accordance with NRC requirements to demonstrate conformance with the ECCS acceptance criteria of 10CFR50.46. A key objective of the LOCA analysis is to provide assurance that the most limiting combination of break size, break location and single failure has been considered for CPS. Reference 3 documents the requirements and the approved methodology to satisfy these requirements.

The SAFER/GESTR-LOCA application methodology is based on the generic studies presented in Reference 3. The approved application methodology consists of three main parts. First, potentially limiting LOCA cases are determined by applying realistic (nominal) analytical models across the entire break spectrum. Second, limiting LOCA cases are analyzed with an Appendix K model (inputs and assumptions) which incorporates the required features of 10CFR50 Appendix K. For the most limiting cases, a Licensing Basis Peak Cladding Temperature (PCT) is calculated based on the nominal PCT with an adder to account statistically for the differences between the nominal and Appendix K assumptions. Finally, a statistically derived Upper Bound PCT is calculated to demonstrate the conservatism of the Licensing Basis PCT. The Licensing Basis PCT conforms to all the requirements of 10CFR50.46 and Appendix K.

The results of the ECCS performance evaluation are discussed in References 13, 18, and 19. Reference 13 describes the base SAFER/GESTR-LOCA analysis performed using the application methodology described by Reference 3.

Reference 18 describes the ECCS-LOCA update for extended power uprate (EPU). The ECCS performance at EPU conditions was evaluated for CPS in a manner consistent with the constant pressure power uprate guidelines established in Reference 16. The most limiting cases from the Reference 13 analyses were evaluated at EPU conditions, using both nominal and Appendix K assumptions. The effect of constant pressure power uprate on the Licensing Basis PCT and Upper Bound PCT was based on the most limiting PCT change in the limiting nominal and Appendix K cases.

Reference 19 describes the SAFER/GESTR-LOCA EPU analysis for the GNF2 fuel type. In the Reference 19 GNF2 NFI, explicit Upper Bound PCT and Licensing Basis PCT calculations are performed.

Analyses supporting CPS operation in the Maximum Extended Operating Domain (MEOD) and Feedwater Heater Out of Service (FWHOS) are detailed in References 13, 18 and 19.

The ECCS performance is evaluated for the entire spectrum of break sizes for postulated LOCA's. The accidents, as listed in Chapter 15, for which ECCS operation is required are:

a. Feedwater Piping Break - Subsection 15.2.8;

b. Spectrum of BWR Steam System Piping Failures Outside of Containment - Subsection 15.6.4; and

c Loss-of-Coolant Accidents - Subsection 15.6.5.

Chapter 15 provides the radiological consequences of the above listed events.

6.3.3.1 ECCS Bases for Core Operating Limits

The maximum average planar linear heat generation rates calculated in this performance analysis provide the basis for the Core Operating Limits Report parameters designed to ensure conformance with the acceptance criteria of 10CFR50.46. 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 Acceptance Criteria for ECCS Performance

The applicable acceptance criteria, extracted from 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light-Water-Cooled Nuclear Power Reactors," are listed, and for each criterion, applicable parts of Subsection 6.3.3 where conformance is demonstrated are indicated.

Criterion 1, Peak Cladding Temperature

"The calculated maximum fuel element cladding temperature shall not exceed 2200° F." Conformance to Criterion 1 is shown in Subsections 6.3.3.7.3 Break Spectrum Calculations, 6.3.3.7.4 Compliance Evaluations, 6.3.3.7.5 Alternate Operating Mode Considerations, 6.3.3.7.6 MAPLHGR Limits, and specifically in Table 6.3-3 SAFER/GESTR-LOCA Licensing Results.

Criterion 2, Maximum Cladding Oxidation

"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 6.3-3 SAFER/GESTR-LOCA Licensing Results.

Criterion 3, Maximum Hydrogen Generation

"The calculated total amount of hydrogen generated from the chemical reaction of the cladding with water or steam shall not exceed 0.01 times the hypothetical amount that would be generated if all of the metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react." Conformance to Criterion 3 is shown in Table 6.3-3.

Criterion 4, Coolable Geometry

"Calculated changes in core geometry shall be such that the core remains amenable to cooling." As discussed in Reference 4, conformance to Criterion 4 is demonstrated by conformance to Criteria 1 and 2.

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 BWRs in Reference 4. Briefly summarized, the core remains covered to at least the jet pump suction elevation and the uncovered region is cooled by spray cooling.

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

It is therefore only necessary to consider each of these 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.

6.3.3.4 System Performance During the Accident

In general, the system response to an accident can be described as:

a. receiving an initiation signal,

b. a small lag time (to open all valves and have the pumps up to rated speed), and

c. finally the ECCS flow entering the vessel.

Key ECCS actuation setpoints and time delays for all the ECC systems are provided in Table 6.3-2. The minimization of the delay from the receipt of signal until the ECCS pumps have reached rated speed is limited by the physical constraints on accelerating the diesel-generators and pumps. Valve timing is shown in Table 6.3-2.

The flow delivery rates analyzed in Subsection 6.3.3 can be determined from the head-flow curves in Figures 6.3-3, 6.3-6, and 6.3-7 discussed in Subsection 6.3.2 and the pressure versus time plots in Reference 14. Simplified piping and instrumentation and functional control diagrams for the ECCS are discussed in Subsection 6.3.2. The operational sequence of ECCS for the DBA is shown in Table 6.3-1.

The original design basis values are included in Tables 6.3-1 and Table 6.3-2 for information and comparison. The design parameters were not changed when the ECCS analysis values were relaxed. The analysis now includes considerable margins from the design values in many cases. The systems and components continue to function per their design basis.

Operator action is not required, except as a monitoring function, during the short-term cooling period following the LOCA. During the long-term cooling period, the operator will take action as specified in Subsection 6.2.2.2 to place the containment cooling system into operation.

6.3.3.5 Use of Dual Function Components for ECCS

With the exception of the LPCI system, the systems of the ECCS are designed to accomplish only one function: to cool the reactor core following a loss of reactor coolant. To this extent, components or portions of these systems (except for pressure relief) are not required for operation of other systems which have emergency core cooling functions, or vice versa. Because either the ADS initiating signal or the overpressure signal opens the safety/relief valve, no conflict exists.

The LPCI subsystem, however, uses the RHR pumps and some of the RHR valves and piping. When the reactor water level is low, the LPCI subsystem has priority through the valve control logic over the other RHR subsystems including containment cooling. Immediately following a LOCA, the

RHR system is directed to the LPCI mode, however, realignment from the shutdown cooling mode requires manual actions.

6.3.3.6 Limits on ECCS Parameters

The limits on the ECC system parameters are discussed in Subsections 6.3.3.2 and 6.3.3.7.1.

Any number of components in any given system may be out of service, up to and including the entire system. The maximum allowable out of service time is a function of the level of redundance and the specified test intervals as discussed in Section 15.A.5.

6.3.3.7 ECCS Analyses for LOCA

6.3.3.7.1 LOCA Analysis Procedures and Input Variables

6.3.3.7.1.1 Description of Models

Four GE-NE computer models were used to determine the LOCA response for the CPS ECCS LOCA analysis. These models are LAMB, TASC, SAFER, and GESTR-LOCA. Together, these models evaluate the short-term and long-term reactor blowdown response to a pipe rupture, the subsequent core flooding by ECCS, and the final fuel rod heatup. Figure 6.3-10 is a flow diagram of these computer models, indicating the major code functions and the transfer of major parameters. The purpose of each model is described in the following subsections.

LAMB

The LAMB model (Reference 4) analyzes the short-term blowdown phenomena for postulated large pipe breaks in which nucleate boiling is lost before the water level drops sufficiently to uncover the active fuel. The LAMB output (most importantly, core flow as a function of time) is used in the TASC model for calculating blowdown heat transfer and fuel dryout time.

TASC

The TASC model (Reference 4) completes the transient short-term thermal-hydraulic calculation for large recirculation line breaks. The time and location of boiling transition are predicted during the period of recirculation pump coastdown. When the core inlet flow is low, TASC also predicts the resulting bundle dryout time and location. The calculated fuel dryout time is an input to the long-term thermal-hydraulic transient model, SAFER. A TASC model is used to predict the time and location of boiling transition and dryout time. TASC explicitly models the axially varying flow areas and heat transfer surface resulting from the part length fuel rods, and incorporates the critical power correlation for GE14 and GNF2.

GESTR-LOCA

This GESTR-LOCA (Reference 7) model provides the parameters to initialize the fuel stored energy and fuel rod fission gas inventory at the onset of a postulated LOCA for input to SAFER. GESTR-LOCA also establishes the initial transient pellet-cladding gap conductance for input to both SAFER and TASC.

SAFER

This SAFER model (References 8 through 12) calculates the long-term system response of the reactor over a complete spectrum of hypothetical break sizes and locations. SAFER is compatible with the GESTR-LOCA fuel rod model for gap conductance and fission gas release. SAFER calculates the core and vessel water levels, system pressure response, ECCS performance, and other primary thermal-hydraulic phenomena occurring in the reactor as a function of time. SAFER realistically models all regimes of heat transfer that occur inside the core, and provides the PCT and the heat transfer coefficients (which determine the severity of the temperature change) as a function of time. For current fuel analysis with the SAFER code, the part length fuel rods are treated as full-length rods, which conservatively overestimates the hot bundle power.

6.3.3.7.1.2 Analysis Procedure

The following procedures were used in the calculations documented in Subsection 6.3.3.

SAFER/GESTR-LOCA LICENSING METHODOLOGY

The SAFER/GESTR-LOCA licensing methodology approved by the NRC in Reference 3 allows the plant-specific break spectrum to be defined using nominal input assumptions. However, the calculation of the limiting PCT to demonstrate conformance with the requirements must include specific inputs and models documented in Appendix K.

The licensing basis PCT is based on the most limiting LOCA (highest PCT) and is defined as:

$$PCT_{Licensing} = PCT_{Nominal} + ADDER$$

The adder is calculated as follows:

$$ADDER^2 = [PCT_{App\ K} - PCT_{Nominal}]^2 + \sum(\delta PCT\iota)^2 .$$

Where:

$PCT_{App.\ K}$	=	Peak cladding temperature from calculation using Appendix K specified models and inputs.
$PCT_{Nominal}$	=	Peak cladding temperature from nominal case.
$\sum(\delta PCT\iota)$	=	Plant variable uncertainty term (defined in Reference 3).

The plant variable uncertainty term accounts statistically for the uncertainty in parameters which are not specifically addressed in 10CFR50 Appendix K.

To conform with 10CFR50.46 and the NRC SER requirements for use of the SAFER/GESTR-LOCA licensing methodology, the Licensing Basis PCT must be less than 2200°F.

Conformance evaluation of the nominal PCT is also required through the use of a statistical Upper Bound PCT as defined in the NRC SER documented in Reference 3. The Upper Bound PCT is a function of the limiting break Nominal PCT, modeling bias, and plant variable uncertainty. The Upper Bound PCT is defined as:

$$PCT_{Upper\ Bound} = PCT_{Nominal} + \Delta 4\text{-}max_{generic} + (\overline{\Delta 3} + 2s_{\Delta 3})$$

Where:

$D4\text{-}max_{generic}$ = Modeling Bias (defined in Reference 3). This term accounts for errors in modeling processes for which experimental data is available for comparison. These are primarily the LOCA thermal-hydraulic processes.

$(\overline{\Delta 3} + 2s_{\Delta 3})$ = Plant Variable Uncertainties (defined in Reference 3). This term accounts for uncertainties due to inputs to the model. These are typical plant parameters with associated uncertainties in their measured values.

The Upper Bound PCT is required to be less than the Licensing Basis PCT. This ensures that the Licensing Basis PCT is in all cases greater than the 95th percentile of the PCT distribution for the limiting case LOCA, and for all LOCAs within the design basis. As part of the development of the SAFER/GESTR-LOCA licensing methodology, GE-NE demonstrated that this criterion was satisfied for the BWR 5/6 class of plants. For CPS, fuel and plant-specific evaluations were performed to demonstrate conformance to these licensing criteria.

BWR 5/6 GENERIC ANALYSIS

For the GE BWR 5/6 product lines, a generic Appendix K conformance calculation was performed for the limiting hypothetical LOCA (Reference 3). The limiting LOCA was determined from the nominal break spectrum at that break size and ECCS component failure combination that yielded the highest nominal PCT. The Appendix K calculation was established as the basis for the licensing evaluation and determining reactor operating limits.

The PCT calculated as described above maintains margins for licensing evaluations (i.e., the licensing basis PCT is at least the upper 95th percentile PCT). This was verified by separate calculations to determine the upper 95th percentile values of PCT at the most limiting conditions determined from the nominal break spectrum calculations. These calculations were performed to qualify the “Appendix K Procedure” as being sufficiently conservative.

The Design Basis Accident (DBA) recirculation suction line break coincident with failure of the High Pressure Core Spray (HPCS) Diesel Generator (DG) was found to be the limiting break in the nominal break spectrum for the BWR 5/6 product lines. As a result, this case was used to perform the Appendix K calculation. The results of the Appendix K calculation demonstrated that a discharge coefficient of 1.0 in the Moody Slip Flow Model yields the highest calculated PCT. The Licensing Basis PCT for BWR 5/6 plants was then calculated by combining the nominal PCT with the adder described earlier. The Upper Bound PCT (95% probability PCT) was also established generically to demonstrate that the Licensing Basis PCT was above the Upper Bound PCT. This generic evaluation demonstrated that a PCT margin in excess of 80°F existed between the Upper Bound PCT and the Licensing Basis PCT (Reference 3).

CPS PLANT-SPECIFIC ANALYSIS

As discussed in the SER (Reference 3) the determination of the limiting case LOCA is based on:

1) The generic Appendix K PCT versus break size curve exhibits the same trends as the generic Nominal PCT versus break size curve for a given class of plants;

2) The limiting LOCA determined from Nominal calculations is the same as that determined from Appendix K calculations for a given class of plants; and

3) Both the generic Nominal PCT versus break size curve and Appendix K PCT versus break size curve for a given class of plants are shown to be applicable on a plant specific basis. Necessary conditions for demonstrating applicability include:

   a) Calculation of a sufficient number of plant specific PCT points to verify the shape of the curve;

   b) Confirmation that plant-specific Appendix K PCT calculations match the trend of the generic curve for that plant class;

   c) Confirmation that plant-specific operating parameters have been conservatively bounded by the models and inputs used in the generic calculations; and

   d) Confirmation that the plant-specific ECCS is consistent with the referenced plant class ECCS configuration.

The specific analysis performed for CPS consisted of break sizes ranging from 0.03ft^2 to the DBA recirculation suction line break (2.207 ft^2). An additional flow path area of 0.016 ft^2 was included in the analysis for the bottom head drain line. Therefore, the total effective break area for the DBA recirculation suction line break is 2.223 ft^2. Plant-specific operating parameters and ECCS configurations are consistent with those used in the generic calculations. Different single failures were investigated to identify the worst cases. The break spectrum was first evaluated using the analysis assumptions for nominal calculations (Table 6.3-12). The potentially limiting cases were then analyzed again with the analysis assumptions specified for the Appendix K calculation (Table 6.3-13). The CPS nominal and Appendix K PCT results were compared to assure that the PCT trends as a function of break size were consistent with one another and with those of the generic BWR 5/6 break spectrum curves documented in Reference 3. The nominal and Appendix K PCT results for CPS were used to calculate the Upper Bound PCT and Licensing Basis PCT.

6.3.3.7.1.3 Input To Analysis

PLANT INPUTS

The significant plant input parameters to the CPS LOCA analysis include plant operating conditions (Table 6.3-14), fuel parameters (References 18 and 19), and ECCS parameters (Table 6.3-2). Table 6.3-7 identifies the combinations of break locations, single-failures and available systems specifically analyzed for the CPS ECCS configuration (Figure 6.3-11).

FUEL PARAMETERS

The SAFER/GESTR-LOCA analyses were performed with conservative Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) at the most limiting combination of power and exposure. The most limiting power/exposure combination was determined by performing generic sensitivity studies for each fuel type along the peak power/exposure envelope used for fuel thermal/mechanical (T/M) design. The fuels evaluated for these analyses include GE14 and GNF2. For fuel-specific details, see References 18 and 19.

ECCS PARAMETERS

The CPS SAFER/GESTR-LOCA analysis incorporates values for some ECCS performance parameters that are more conservative than the original design basis values. Also, updates required by the SAFER/GESTR methodology were also included in the analysis. Table 6.3-2 contains the specific ECCS performance input parameters used in the evaluation.

The parameters discussed in this section together with Table 6.3-2 and Figure 6.3-9, make up the significant input variables used by the LOCA analysis codes identified in 6.3.3.7.1.1.

6.3.3.7.2 Accident Description

For convenience, a short description of the major events during the design-basis accident (DBA) is included here.

Immediately after the postulated double-ended recirculation line break, vessel pressure and core flow begin to decrease. The initial pressure response is governed by the closure of the main steam isolation valves and the relative values of energy added to the system by decay heat and energy removed from the system by the initial blowdown of fluid from the downcomer. The initial core flow decrease is rapid because the recirculation pump in the broken loop ceases to pump almost immediately due to loss of suction. The pump in the intact loop coasts down relatively slowly. This pump coastdown governs the core flow response for the next several seconds. When the jet pump suctions become uncovered, core flow decreases to near zero. When the recirculation pump suction nozzle becomes uncovered, the energy release rate from the break increases significantly and the pressure begins to decay more rapidly. As a result of the increased rate of vessel pressure loss, the initially subcooled water in the lower plenum saturates and flashes up through the core, increasing the core flow. This lower plenum flashing continues at a reduced rate for the next several seconds.

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 the jet pump is uncovered. Boiling transition follows shortly after the core flow loss that results from jet pump being uncovered. 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 becomes uncovered. 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 and the core becomes uncovered. 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 decreases initially because nucleate boiling is maintained, 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.

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 locations is considered in the Reference 13 evaluation of ECCS performance to determine the limiting cases. The subsequent GE14 (Reference 18) and GNF2 (Reference 19) analyses analyzed the limiting break sizes and locations identified by Reference 13.

For ease of reference, a summary of all figures presented in Reference 13 is shown in Table 6.3-4. Table 6.3-4A summarizes the figures presented in the Reference 18 GE14 analysis, and Table 6.3-4B summarizes the figures presented in the Reference 19 GNF2 analysis.

6.3.3.7.3.1 Recirculation Line Breaks

The maximum recirculation suction line break with potentially limiting single failures (HPCS DG, LPCI DG and LPCS DG) was analyzed for fuel types GE14 (Reference 18) and GNF2 (Reference 19) using nominal and Appendix K assumptions and inputs discussed in Subsection 6.3.3.7.1.2. The drain line flow path between the vessel bottom head and the broken recirculation line was considered in the analysis. The limiting fuel type was found to be GNF2 fuel and the limiting single failure was found to be the HPCS DG (Reference 19) for both nominal and Appendix K assumptions. A sufficient number of break sizes were analyzed to confirm that the shape of the PCT versus break area curve (break spectrum) was similar to that established in Reference 13. The mid-peaked and top-peaked axial power shapes are considered for GNF2 large break analysis. The evaluation confirms that the mid-peaked axial power shape yields the limiting PCT for large breaks. Reference 13 contains the nominal and Appendix K break spectrum results for the limiting fuel type and failure combination. This ensures that the limiting combination of break size, location, and single failure has been identified. The shape of the break spectrum curve from the base SAFER/GESTR ECCS-LOCA analysis (Reference 13) is also valid for GE14 and GNF2 fuel at EPU conditions.

The nominal assumptions used in the analysis are listed in Table 6.3-12. The results of the nominal calculations show that the nominal PCT decreases with decreasing break size in the DBA to the 1 ft^2 range which is consistent with the trend observed in the generic break spectrum (Reference 3). In this range, LOCA analysis results show two peak temperatures on the heatup curves. The first peak temperature is due to early boiling transition (dryout) which is not sensitive to differences in break sizes. The second peak temperature is caused by core uncovery. The relatively small heatup for CPS is typical of BWR 6 plants and is due to the short period of core uncovery. This short period of core uncovery is due to the relatively high capacity of the ECCS makeup systems and the relatively small total area of the DBA suction line break compared to other BWR designs. For CPS, the ECCS-LOCA analysis calculations in References 18 and 19 determine the single failure that results in the highest PCT in the large break range.

In the small break range (<1.0 ft^2), ECCS injection depends on reactor depressurization due to the initiation of the Automatic Depressurization System (ADS). The calculated PCT increases first with decreasing break size and then decreases again. For small breaks (which do not experience early dryout), the calculated PCT occurs due to core uncovery. For CPS, the calculations in References 18 and 19 determine the small break size that yields the highest nominal PCT. References 13 also contains a listing of all nominal cases analyzed for CPS. These results confirm that the limiting nominal LOCA case is the maximum recirculation line suction break with a HPCS DG failure. The system response time histories for the nominal cases are shown in References 13, 18, and 19.

The Appendix K assumptions used in the analyses are listed in Table 6.3-13. Using the Appendix K input assumptions, DBA analyses were performed for GE14 and GNF2 fuels with the HPCS DG failure. Then DBA analyses were performed with GE14 and GNF2 fuel types for HPCS DG, LPCI DG and LPCS DG failures. These results show that for the DBA case the GNF2 fuel with a HPCS DG failure is also the limiting fuel type and single failure for Appendix K analysis. Then analyses of three break sizes (60%, 80%, and 100% DBA recirculation suction line break) are performed for GE14 and GNF2 fuel with a HPCS DG failure. This is to examine the sensitivity of Appendix K PCT to break size and to assure that the limiting break is consistent with the generic Appendix K results. There is a rapid vessel depressurization due to vessel inventory loss through the break. This vessel depressurization is faster than the nominal case due to higher break flow from the Appendix K Moody Slip Flow Model (Table 6.3-13). This higher mass loss also causes an earlier core uncover. The fuel rods remain in film boiling for a longer time than the nominal case because of the more restrictive Appendix K assumptions, which do not allow the bundle to change from film boiling to transition boiling until the cladding superheat falls below 300°F. The LPCS and LPCI begin injection (following the same initiation sequence as the nominal case). Then there is a second bundle heatup, which lasts until the water level reaches the top of active fuel. Overall the bundle heatup for the Appendix K case is higher than the nominal case due to higher bundle power, decay heat and break flow. From these analyses it is concluded that the DBA recirculation suction line break with HPCS DG failure results in the highest Appendix K PCT for GE14 fuel and GNF2 (limiting fuel type). (see Table 6.3-3)

In the small break range (<1.0 ft^2), Appendix K evaluations were performed for break sizes around the limiting small break area from the nominal case. For CPS the calculations in References 13, 18 and 19 determine the small break size that yields the highest Appendix K PCT. References 13 and 19 also contain a listing of all Appendix K cases analyzed for CPS. These analyses show that the limiting Appendix K LOCA case is the maximum recirculation suction line break with a HPCS DG failure. The system response time histories for the limiting Appendix K cases in References 13, 18 and 19 are listed in Tables 6.3-4, 6.3-4a and 6.3-4b.

6.3.3.7.3.2 Non-Recirculation Line Breaks

Non-recirculation line breaks were analyzed for GE14 fuel, using nominal assumptions. The results of these analyses are presented in Reference 13. These results show that non-recirculation line breaks are significantly less limiting than the postulated recirculation line breaks. References 18 and 19 extend this conclusion to GE14 and GNF2 at EPU conditions.

6.3.3.7.4 Compliance Evaluations

6.3.3.7.4.1 Licensing Basis PCT Evaluation

The Appendix K results confirm that the limiting break is the recirculation suction line DBA, which is consistent with the BWR 5/6 generic conclusions. References 13, 18 and 19 provide Appendix K ECCS-LOCA DBA results for GE14 and GNF2 fuels. The Licensing Basis PCTs for CPS were calculated for GE14 and GNF2 fuel types with the limiting failure (HPCS DG) using the results presented above and the methodology presented in Subsection 6.3.3.7.1.2 with an adder applied to the nominal PCT. Reference 13 provides a description of the plant variable uncertainties used to determine a plant-specific adder for CPS. The Licensing Basis PCT adder determined in Reference 18 is applicable at EPU conditions.

The Analysis of Record PCT values for GE14 and GNF2 fuel listed in Table 6.3-3 are less than the 10 CFR 50.46 limit of 2200°F. (Note: For the purpose of 10 CFR 50.46 reporting requirements, the Licensing Basis PCT is the Analysis of Record PCT plus the PCT changes due to subsequent evaluations.) Subsequent to the Analysis of Record, several evaluations have been performed to address various issues and the Licensing Basis PCT is listed in Table 6.3-3. Refer to the latest annual or thirty day 10 CFR 50.46 report for details on the Licensing Basis as reported to the NRC.

6.3.3.7.4.2 Upper Bound PCT Evaluation

For the BWR 5/6 plants, the generic Appendix K PCT versus break size curve exhibits the same trends as the generic nominal PCT versus break size curve, and the limiting LOCA determined from nominal PCT calculations is the same as that determined from the Appendix K PCT calculations (Reference 3, Figure 3.4). The CPS specific results presented in Section 6.3.3.7.3 demonstrate the applicability of the BWR 5/6 generic nominal PCT and Appendix K PCT versus break size curves to the CPS plant.

Reference 13 provides a discussion of the calculation of the CPS specific Upper Bound PCTs for GE10 and GE14 fuels. The Upper Bound PCT adder determined in Reference 18 is applicable at EPU conditions for GE14 fuel.

Reference 19 provides a discussion of the calculation of the CPS specific Upper Bound PCTs for GNF2 fuel. Reference 20 eliminated the 1600°F restriction on the Upper Bound PCT and provided generic justification that the Licensing Basis PCT is conservative with respect to the Upper Bound PCT. The Upper Bound PCT determined in Reference 19 demonstrates that the Licensing Basis PCT is sufficiently conservative.

By verifying that the Licensing Basis PCT for CPS is greater that the Upper Bound (95th percentile) PCT, the level of safety and conservatism of this analysis meets the NRC acceptance criterion (Subsection 6.3.3.7.1.2) for the SAFER/GESTR methodology.

6.3.3.7.5 Alternate Operating Mode Considerations

6.3.3.7.5.1 Maximum Extended Load Line Limit (MELLL)

CPS can operate in an extended power/flow envelope called the Maximum Extended Operating Domain (MEOD). The MEOD region permits reactor operation at rated power over a wide range of core flows. Low core flow effects were generically addressed in Reference 5, which was approved by the NRC in Reference 6. These studies demonstrated that no ECCS MAPLHGR multiplier was required for low core flow operation for the BWR 5/6 plant class similar to CPS. References 13 and 19 provide analyses of the MEOD low flow condition.

The results of this evaluation show that the potential increase in PCT for a design basis LOCA at the MEOD low flow condition is small relative to the PCT margin (greater than 600°F) currently available with respect to the 2200°F criteria. Also, no low flow MAPLHGR or MCPR multipliers are required for ECCS considerations.

6.3.3.7.5.2 Increased Core Flow

References 18 and 19 provide analyses of the MEOD high flow condition. The impact on the PCT for a design basis LOCA at the increased core flow (ICF) condition is small relative to the PCT margin (greater than 600°F) currently available with respect to the 2200°F criteria.

6.3.3.7.5.3 Feedwater Temperature Reduction (FWTR)

References 18 and 19 provide analyses of the impact on LOCA results due to a feedwater temperature reduction. The impact on the PCT for a design basis LOCA at the FWTR condition is small relative to the PCT margin (greater than 600°F) currently available with respect to the 2200°F criteria.

6.3.3.7.5.4 Single-Loop Operation (SLO)

References 18 and 19 provide analyses to support single-loop operation for CPS. These analyses include the determination of MAPLHGR multipliers for SLO for each fuel type to assure that the SLO ECCS-LOCA results meet the acceptance criteria of 10CFR50.46 and the NRC SER requirements for the SAFER application methodology.

6.3.3.7.6 MAPLHGR Limits

Current GE BWR MAPLHGR limits (as a function of exposure) are based on the most limiting value of either the MAPLHGR determined from ECCS limits (PCT) or the MAPLHGR determined from fuel thermal-mechanical (T/M) design analysis limits.

The bounding MAPLHGR used in the CPS SAFER/GESTR-LOCA analysis for each fuel type are documented in References 18 and 19.

6.3.3.7.7 Summary of SAFER/GESTR Analysis Results

LOCA analyses have been performed for CPS utilizing the GE SAFER/GESTR-LOCA Application Methodology approved by the NRC. These analyses were performed in accordance with the NRC Safety Evaluation Report (SER) requirements for the use of the SAFER/GESTR-LOCA analysis methodology and demonstrate conformance with 10CFR50.46 and 10CFR50 Appendix K and thus establishes a revised licensing basis for CPS with the GE SAFER/GESTR-LOCA methodology.

The CPS SAFER/GESTR-LOCA results presented in Subsections 6.3.3.7.3 to 6.3.3.7.5 demonstrate that a sufficient number of plant-specific PCT points have been evaluated to establish the shape of both the nominal and Appendix K PCT versus break size curves. The analyses demonstrate that the limiting Licensing Basis PCT occurs for the recirculation suction line DBA.

Table 6.3-3 summarizes the key SAFER/GESTR licensing results for CPS. The analyses presented are performed in accordance with NRC requirements and demonstrate conformance with the ECCS acceptance criteria of 10 CFR 50.46. Therefore, these results provide a new LOCA Licensing Basis for CPS. The results are valid for fuel designs with comparable geometry to the GE14 and GNF2 fuel types analyzed.

With the explicit verification that the Licensing Basis PCT for CPS is greater than the Upper Bound (95th percentile) PCT, the level of safety and conservatism of this analysis meets the NRC approved criteria. Therefore, the requirements of Appendix K are satisfied.

6.3.3.7.8 An Analysis to Show that Diversion of ECCS to Containment Cooling at or less than 10 Minutes After a LOCA will Not Result in Exceeding Any Safety Criteria for the Entire Break Spectrum With Consideration of a Single Failure

6.3.3.7.8.1 Purpose

The purpose of this evaluation is to determine the impact of LOCI diversion to containment cooling at 10 minutes after the start of a LOCA event on the LOCA analysis results documented in Subsections 6.3.3.7.3 through 6.3.3.7.5.

6.3.3.7.8.2 Analysis Models and Methodology

The same analysis models and methodology identified in detail in Subsection 6.3.3.7.1 are used to perform the LOCA evaluation with LPCI diversion

6.3.3.7.8.3 Analysis Input

The same plant, fuel and ECCS parameters identified in detail in Subsection 6.3.3.7.1 are used to perform the LOCA evaluation with LPCI diversion except for the assumed number of LPCI pumps available for reflooding the core. A maximum of two LPCI pumps (specifically LPCI “A” and LPCI “B”) will be fully diverted at 10 minutes to the containment spray mode. (Note: LPCI “A” shares an emergency diesel-generator with the LPCS and LPCI “B” and “C” share an emergency diesel-generator. The pump associated with LPCI “C” cannot be diverted to containment spray.)

6.3.3.7.8.4 Analysis

Only those LOCA cases where the core is not reflooded before 10 minutes can be affected by the assumed LPCI diversion. Also for each break location, the break size that results in LPCI injection at about 10 minutes will show the biggest impact on the LOCA results due to LPCI diversion. Larger breaks will have LPCI injection sooner than 10 minutes and the reflooding flow will get some benefit from the diverted LPCI pumps before diversion. Smaller breaks will have less mass loss through the break and require less reflooding flow to restore the core water level.

Reference 13 summarizes the analysis results at each break location for the break size with LPCI injection at about 10 minutes after the start of the LOCA event. Appendix K assumptions were used because the higher decay heat and break flow compared to nominal assumptions will tend to maximize the possibility of core uncovery. Only breaks inside the containment were considered because the postulated pressurization of the suppression pool air space is due to high steam flow bypass from the drywell.

Recirculation Line Breaks

Since the LPCI diversion evaluation involves small break, the failure of the HPCS for non-core spray line breaks is the limiting single failure. For recirculation line breaks the ECCS remaining before diversion are three LPCI plus LPCS plus ADS, and after diversion are one LPCI plus

LPCS plus ADS. Reference 13 provides the ECCS-LOCA analysis that demonstrates the impact of LPCI diversion on recirculation line breaks.

Feedwater Line Breaks

Since the LPCI diversion evaluation involves small breaks, the failure of the HPCS for non-core spray line breaks is the limiting single failure. For feedwater line breaks the ECCS remaining before diversion are three LPCI plus LPCS plus ADS, and after diversion are one LPCI plus LPCS plus ADS. Reference 13 provides the ECCS-LOCA analysis that demonstrates the impact of LPCI diversion on feedwater line breaks.

Steam Line Breaks Inside Containment

Since the LPCI diversion evaluation involves small breaks, the failure of the HPCS for non-core spray line breaks is the limiting single failure. For the inside containment steam line break the ECCS remaining before diversion are three LPCI plus LPCS plus ADS, and after diversion are one LPCI plus LPCS plus ADS. Reference 13 provides the ECCS-LOCA analysis that demonstrates the impact of LPCI diversion on steam line breaks inside containment.

LPCI Line Breaks

Since the LPCI diversion evaluation involves small breaks, the failure of the HPCS for non-core spray line breaks is the limiting single failure. For the LPCI line break the ECCS remaining before diversion are two LPCI plus LPCS plus ADS, and after diversion are one LPCI plus LPCS plus ADS (for a break in LPCI line “C”) or LPCS plus ADS (for a break in LPCI line “A” or “B”). Reference 13 provides the ECCS-LOCA analysis that demonstrates the impact of LPCI diversion on LPCI line breaks.

Core Spray Line Breaks

Since the LPCI diversion evaluation involves small breaks, the HPCS line break is more limiting than the LPCS line break because a single failure will leave fewer systems available. For the HPCS line break, the failure of the LPCS diesel-generator is the more limiting case since this eliminates all core spray cooling. For the HPCS line break the ECCS remaining before diversion are two LPCI plus ADS, and after diversion are one LPCI plus ADS. Reference 13 provides the ECCS-LOCA analysis that demonstrates the impact of LPCI diversion on core spray line breaks.

Long Term Cooling

Automatic LPCI diversion to containment cooling at 10 minutes after the start of the LOCA event does not impact long term cooling capability. The core spray systems are unaffected and redundant. Therefore, following a single failure for all non-core spray line breaks there is at least one core spray system available for long term cooling to meet the long term cooling criteria given in Table 6-1 of the main report. For a core spray line break the elevation of the break is above the core. Therefore for these breaks the core can be completely flooded and the level maintained above the core by the undiverted LPCI system or the remaining core spray system depending on the assumed single failure.

6.3.3.7.8.5 Conclusions from LPCI Diversion Study

Based on the analyses in Reference 13, it is concluded that the ECCS-LOCA analysis results for CPS, considering LPCI diversion at 10 minutes after the start of the LOCA event, still meet all ECCS-LOCA licensing criteria.

6.3.3.7.9 Inadvertent Closure of the Reactor Recirculation System Line Suction Valve as a Single Failure in the LOCA Analysis, for the Break Size Most Affected by this Failure

This assumed single failure is not part of the standard ECCS analysis because it is not limiting. However, the consequences of this improbable single failure have been investigated throughout the break spectrum and for various times when failure is postulated to occur. Under all conditions, the resulting PCT is well below the maximum calculated Appendix K PCT. Furthermore, the resulting worst-case PCT for each break size falls below the current PCT vs. break area plot.

6.3.3.8 LOCA Analysis Conclusions

Having shown compliance with the applicable acceptance criteria of Subsection 6.3.3.2, it is concluded that the ECCS will perform its function in an acceptable manner and meet all of the 10 CFR 50.46 acceptance criteria, given operation at or below the maximum average planar linear heat generation rates in References 18 and 19. All ECCS-LOCA results are independent of cycle length.

6.3.4 Tests and Inspections

6.3.4.1 ECCS Performance Tests

All systems of the ECCS were tested for their operational ECCS function during the preoperational and/or startup test program. Each component was tested for power source, range, direction of rotation, setpoint, limit switch setting, torque switch setting, etc. Each pump was tested for flow capacity for comparison with vendor data. (This test is also used to verify flow measuring capability.) The flow tests involve the same suction and discharge source; i.e., suppression pool or RCIC storage tank.

All logic elements were tested individually and then as a system to verify complete system response to emergency signals including the ability of valves to revert to the ECCS alignment from other positions.

Finally the entire system was tested for response time and flow capacity taking suction from its normal source and delivering flow into the reactor vessel. This last series of tests was performed with power supplied from both offsite power and onsite emergency power.

See Chapter 14 for a thorough discussion of preoperational testing for these systems.

6.3.4.2 Reliability Tests and Inspections

The average reliability of a standby (non-operating) safety system is a function of the duration of the interval between periodic functional tests. The factors considered in determining the periodic test interval of the ECCS are: the desired system availability (average reliability), the number of redundant functional system success paths, the failure rates of the individual components in the system, and the schedule of periodic tests (simultaneous versus uniformly staggered versus randomly staggered). For the ECCS the above factors were used to determine safe test intervals utilizing the methods described in Reference 1. For current test intervals, refer to the CPS Technical Specifications.

All of the active components of the HPCS system, LPCS, and LPCI systems are designed so that they may be tested during normal plant operation. Full flow test capability is provided by a test line back to the suction source. The full flow test is used to verify the capacity of each ECCS pump loop while the plant remains undisturbed in the power generation mode. In addition, each individual valve may be tested during normal plant operation. ECCS system response time can be confirmed by routinely measuring the response time of major system components such as valve stroke times.

All of the active components of the ADS system except the safety/relief valves and their associated solenoid valves are designed so that they may be tested during normal plant operation. The safety/relief valves and associated solenoid valves are tested in accordance with the Technical Specifications. Safety/relief valves and their associated solenoid valves which have been overhauled during a plant outage are tested to verify operability.

Testing of the initiating instrumentation and controls portion of the ECCS is discussed in Subsection 7.3.1. The emergency power system, which supplies electrical power to the ECCS in the event that offsite power is unavailable, is tested as described in Subsection 8.3.1. The frequency of testing is specified in the Technical Specifications. Visual inspections of all the ECCS components located outside the drywell can be made at any time during power operation. Components inside the drywell can be visually inspected only during periods of access to the drywell. When the reactor vessel is open, the spargers and other internals can be inspected.

6.3.4.2.1 <u>HPCS Testing</u>

The HPCS can be tested at full flow with RCIC storage tank water at any time during plant operation except when the reactor vessel water level is low, or when the water level in the RCIC storage tank is below the reserve level, or when the valves from the suppression pool to the pump are open. If an initiation signal occurs while the HPCS is being tested, the system returns automatically to the operating mode. The two motor-operated valves in the test line to the RCIC storage system are interlocked closed when the suction valve from the suppression pool is open.

A design flow functional test of the HPCS system is performed by pumping water from the RCIC storage tank and back through the full flow test return line to the RCIC storage tank.

The suction valve from the suppression pool and the discharge valve to the reactor remain closed. These two valves are tested separately to ensure their operability.

The HPCS test conditions are tabulated on the HPCS process flow diagram, Drawing 762E454.

6.3.4.2.2 ADS Testing

Each ADS valve is manually actuated during initial reactor startup testing and following refurbishing of an ADS valve. The ADS circuitry can be checked any time.

During plant operation the ADS system can be checked as discussed in Subsection 7.3.1.1.1.4.

6.3.4.2.3 LPCS Testing

The LPCS pump and valves are tested periodically during reactor operation. With the injection valve closed and the return line open to the suppression pool, full flowing pump capability is demonstrated. The injection valve and the check valve are tested in a manner similar to that used for the LPCI valves. The system test conditions during reactor shutdown are shown on the LPCS system process diagram, Drawing 762E467AC.

6.3.4.2.4 LPCI Testing

Each LPCI loop can be tested during reactor operation. The test conditions are tabulated in Drawing 762E425AC. During plant operation, this test does not inject cold water into the reactor because the injection line check valve is held closed by the vessel pressure, and the shutoff valve remains in the closed position. Valves in the injection flow path are tested in accordance with the Inservice Inspection Program as applicable. To test an LPCI pump at rated flow, the test line valve to the suppression pool is opened, the pump suction valve from the suppression pool is opened (this valve is normally open), and the pumps are started using the remote/manual switches in the control room. Correct operation is determined by observing the instruments in the control room.

If an initiation signal occurs during the test, the LPCI system returns to the operating mode. The valves in the test bypass lines are closed automatically to assure that the LPCI pump discharge is correctly routed to the vessel.

6.3.5 Instrumentation Requirements

Design details including redundancy and logic of the ECCS instrumentation are discussed in Section 7.3.

All instrumentation required for automatic and manual initiation of the HPCS, LPCS, LPCI, and ADS is discussed in Subsection 7.3.2 and is designed to meet the requirements of IEEE-279 and other applicable regulator requirements. The HPCS, LPCS, LPCI and ADS can be manually initiated from the control room.

The HPCS, LPCS, and LPCI are automatically initiated on low reactor water level or high drywell pressure. The ADS is automatically actuated by sensed variables for reactor vessel low water level and drywell high pressure plus indication that at least one LPCI or LPCS pump is operating. The HPCS, LPCS, and LPCI automatically return from system flow test modes to the emergency core cooling mode of operation following receipt of an automatic initiation signal. The LPCS and LPCI system injection into the RPV begin when reactor pressure decreases to system discharge shutoff pressure.

HPCS injection begins as soon as the HPCS pump is up to speed and the injection valve is open since the HPCS is analyzed with the injection of water into the RPV over a pressure range from 1103 psid* to 0 psid*.

* psid-differential pressure between RPV and pump suction source.

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19. Clinton Power Station GNF2 ECCS-LOCA Evaluation, 0000-0121-9100-R0, October 2011.

20. NEDE-23785P-A, Vol. III, Supplement 1, “GESTR-LOCA and SAFER Models for the Evaluation of the Loss-of-Coolant Accident Volume III, Supplement 1, Additional Information for Upper Bound PCT Calculation, “Revision 1, General Electric Company, March 2002.

21. NRC Generic Letter 2008-01, “Managing Gas Accumulation in Emergency Core Cooling, Decay Heat Removal, and Containment Spray Systems,” dated January 11, 2008.

22. Letter from K. R. Jury to U. S. NRC, “Nine-Month Response to Generic Letter 2008-01,” dated October 14, 2008, RS-08-131.

23. Letter from Patrick R. Simpson to U. S. NRC, “Response to Request for Additonal Information Regarding Generic Letter 2008-01,” dated December 15, 2009, RS-09-173.

TABLE 6.3-1
OPERATIONAL SEQUENCE OF EMERGENCY CORE COOLING SYSTEMS FOR DESIGN-BASIS ACCIDENT (LARGE BREAK)

Orig. + D. B. TIME (sec)	S/G Analysis* TIME (sec)	EVENTS
0	0	Design basis loss-of-coolant accident assumed to start; normal auxiliary power assumed to be lost
~0	~0	Drywell high-pressure reached. All diesel generators signaled to start; scram: HPCS, LPCS, LPCI signaled to start on high drywell pressure.
~3	~5	Reactor low-low water level reached. HPCS receives second signal to start.
~7	~7	Reactor low-low-low water level reached. Second signal to start LPCI and LPCS; auto-depressurization sequence begins; main steam isolation valve signaled to close.
<12	<23	All diesel generators ready to load; energize HPCS pump motor; open HPCS injection valve: begin energizing LPCI and LPCS pump motors.
≤27	<47	HPCS injection valve open and pump at design flow, which completes HPCS startup.
~29	*	Pressure permissive for LPCI and LPCS injection valves reached.
~38	*	HPCS pump at rated flow.
~58	*	LPCS pump at rated flow, LPCS injection valve fully open, which completes the LPCS startup.
~59	*	LPCI pumps at rated flow, LPCI injection valve fully open, which completes the LPCI startup.
~210	*	Core effectively reflooded assuming worst single failure; heatup terminated.
≥10 min.	≥10 min.	Operator shifts to containment cooling.

NOTE: 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).

* Relaxed input values were used for the SAFER/GESTR Analysis. See References 13, 18, and 19 for the core response.

\+ The original design basis (D. B.) is included for information and comparison. The component design response times were not changed when the analysis times were relaxed.

TABLE 6.3-2
SIGNIFICANT INPUT VARIABLES USED IN THE LOSS-OF-COOLANT ACCIDENT ANALYSIS (LARGE BREAK)

A. Plant Parameters[(1)]

Variable	Units	S/G Analysis Value	Original D.B. Value[(4)]
a. Core thermal power (104.2% rated thermal power for original D. B. and 102% for S/G)	MWt	3542.5	3015
b. Vessel steam output	lbm/hr	15.52E6	13.08E6
c. Corresponding percent of rated steam flow	percent(%)	102.4	105
d. Vessel steam dome pressure	psia	1060	1060
e. Maximum recirculation line break area	ft^2	2.207	2.2
f. Bottom head drain line break area	ft^2	0.016	-

S/G = SAFER/GESTR

TABLE 6.3-2
SIGNIFICANT INPUT VARIABLES USED IN THE LOSS-OF-COOLANT ACCIDENT ANALYSIS (LARGE BREAK) (Continued)

B. Low Pressure Coolant Injection (LPCI) System[1]

Variable	Units	S/G Analysis Value	Original D.B. Value[4]
a. Maximum vessel pressure at which flow may commence	psid (vessel to drywell)	225	NC
b. Minimum rated flow inside shroud			
• Vessel pressure at which below listed flow rates are quoted	psid (vessel to drywell)	20	NC
• One LPCI pump into shroud	gpm	4400	4967
• Two LPCI pumps into shroud	gpm	8800	9933
• Three LPCI pumps into shroud	gpm	13200	14900
c. Minimum flow at 0 psid (vessel-to-drywell)			
• One LPCI pump into shroud	gpm	4800	5300
• Two LPCI pumps into shroud	gpm	9600	10600
• Three LPCI pumps into shroud	gpm	14400	15900
d. Initiating signals			
• Low-low-low water level(Level 1)	inches above vessel zero	360.0	368.0
or			
• High drywell pressure	psig	2.0	NC
e. Pressure at which injection valve may open	psig	315	415
f. Maximum time from initiating signal until power can be supplied to the valve with emergency diesel power	second	23*	12
g. Maximum time from initiating signal to pump at rated speed and capable of rated flow with emergency diesel power	second	48*	27
h. Injection valve stroke time-opening	second	55	30

NC = No change from SAFER/GESTR analysis value. Also, SAFER/GESTR analysis value is the same as the original design basis value.

* Includes an additional 1 second for instrument response time.

TABLE 6.3-2
SIGNIFICANT INPUT VARIABLES USED IN THE LOSS-OF-COOLANT ACCIDENT ANALYSIS (LARGE BREAK) (Continued)

C. Low Pressure Core Spray (LPCS) System(1)

Variable	Units	S/G Analysis Value	Original D.B. Value(4)
a. Maximum vessel pressure at which flow may commence	psid (vessel to drywell)	265	NC
b. Minimum flow inside the shroud at vessel pressure	gpm	4400	4900
	psid (vessel to drywell)	113	NC
c. Minimum flow at 0 psid (vessel to drywell)	gpm	4900	5500
d. Initiating signals			
• Low-low-low water level (Level 1)	inches above vessel zero	360.0	368.0
or			
• High drywell pressure	psig	2.0	NC
e. Pressure at which injection valve may open	psig	315	415
f. Maximum time from initiating signal until power can be supplied to the valve with emergency diesel power	seconds	23*	12
g. Maximum time from initiating signal to pump up to speed and capable of rated flow with emergency diesel power	seconds	48*	27
h. Injection valve stroke time-opening seconds	seconds	55	29

NC = No change from SAFER/GESTR analysis value. Also, SAFER/GESTR analysis value is the same as the original design basis value.

* Includes an additional 1 second for instrument response time.

TABLE 6.3-2
SIGNIFICANT INPUT VARIABLES USED IN THE LOSS-OF-COOLANT ACCIDENT ANALYSIS (LARGE BREAK) (Continued)

D. High Pressure Core Spray (HPCS) System[1]

Variable	Units	S/G Analysis Value	Original D.B. Value[4]
a. Maximum vessel pressure at which pump can inject flow	psid	1103	1200
b. Minimum flow inside the shroud at vessel pressure	gpm/psid	800/1103	467/1200
		4000/200	1400/1147
			4900/200
c. Minimum flow at 0 psid (vessel to drywell)	gpm	4900	4900
d. Initiating signals			
• Low-low water level (Level 2)	inches above vessel zero	456	468
or			
• High drywell pressure	psig	2.0	
e. Maximum allowable time delay from initiating signal to rated flow available (including injection valve stroke time opening) with emergency diesel power	seconds	47	27

NC = No change from SAFER/GESTR analysis value. Also, SAFER/GESTR analysis value is the same as the original design basis value.

TABLE 6.3-2
SIGNIFICANT INPUT VARIABLES USED IN THE LOSS-OF-COOLANT ACCIDENT ANALYSIS (LARGE BREAK) (Continued)

E. Automatic Depressurization System (ADS)[1]

Variable	Units	S/G Analysis Value	Original D.B. Value[4]
a. Total number of relief valves installed with ADS function	-	7	NC
b. Pressure at which flow capacity in 4.c. and 4.e. are quoted (vessel to suppression pool)	psid	1190	NC
c. Minimum flow capacity at pressure given in 4.b. with all available ADS valves open	lbm/hr	6.4745E6	NC
d. Total number of relief valves assumed in analysis	-	6	NC
e. Minimum flow rate at pressure given in 4.b. with number of relief valves given in 4.d	lbm/hr	5.5496e6	NC
f. Initiating signals			
• Low-low-low water level (Level 1)	inches above vessel zero	360.0	368.0
and			
• High drywell pressure			
or			
• Low-low-low water level (Level 1)	psig	2	NC
and			
• High drywell pressure bypass timer timed out	inches above vessel zero	360.0	368.0
g. High drywell pressure bypass timer	seconds	420	NC
h. ADS timer delay from initiating signal completed to the time valves are open with confirming signal that one LPCI or LPCS pump is running	seconds	125	120

NC = No change from SAFER/GESTR analysis value. Also, SAFER/GESTR analysis value is the same as the original design basis value.

TABLE 6.3-2
SIGNIFICANT INPUT VARIABLES USED IN THE LOSS-OF-COOLANT ACCIDENT ANALYSIS (LARGE BREAK) (Continued)

F. Fuel Parameters[1]

Variable	Units	S/G Analysis Value	Original D.B. Value[3]
a. Fuel type		See Note 1	Initial Core
b. Fuel bundle geometry		See Note 1	8 x 8
c. lattice		See Note 1	S
d. Number of fueled rods per assembly		See Note 1	62
e. Peak technical specification linear heat generation rate	KW/ft	See Note 1	13.4[3]
f. Initial minimum critical power ratio		See Note 1	1.17
g. Design axial peaking factor		See Note 1	1.4

Notes:

(1) The values presented for the SAFER/GESTR analysis are for the Appendix K calculations. Refer to References 13, 18, and 19 for the nominal values.

(2) This ECCS LOCA analysis was originally performed with the parameters specified in this table. As new fuel bundle designs are introduced the fuel type, fuel bundle geometry, lattice, number of fueled rods per bundle, peak linear heat generation rate, initial minimum critical power ratio and design axial peaking factor all may change. See References 13, 18, and 19 for the values used in the SAFER/GESTR analysis.

(3) The original DB values are for original licensed power using SAFE/REFLOOD methods. The original design basis is included for information because the component parameter design values were not changed when the analysis values were relaxed. This shows the margin between the design and analysis values.

TABLE 6.3-3
SAFER/GESTR-LOCA LICENSING RESULTS FOR CPS

1. Fuel Type	GE14 Fuel	GNF2 Fuel	Acceptance Criteria
2. Limiting Break	DBA Suction	DBA Suction	
3. Limiting Failure	HPCS DG	HPCS DG	
4. Peak Cladding Temperature (Analysis of Record)	<1570°F*	<1880°F*	≤ 2200°F
5. Estimated Upper Bound PCT (95% Probability PCT)	<1470°F	<1600°F	≤1600°F
6. Maximum Local Oxidation	<1.0%	<3.0%	≤17%
7. Core-Wide Metal-Water Reaction	<0.1%	<0.1%	≤1.0%
8. Coolable Geometry	Items 4 and 6	Items 4 and 6	PCT <2200°F and Maximum Local Oxidation <17%
9. Long Term Cooling	Core reflooded above TAF or Core reflooded to top of the jet pumps and 1 core spray available or	Core reflooded above TAF Core reflooded to top of the jet pumps and 1 core spray available	Core temperature acceptable low and long-term decay heat removed

* Subsequent to the Analysis of Record, several evaluations were performed to address various issues. Refer to the latest annual or thirty day 10 CFR 50.46 report for details on the Licensing Basis PCT as reported to the NRC. The 10 CFR 50.46 letter is on file at the site.

TABLE 6.3-4
Key to Figures in Reference 13

(Recirculation Line Breaks)

	Nominal Assumptions								Appendix K Assumptions	
	Large Break Methods						Small Break Methods (SBM)		LBM	SBM
	DBA	DBA	DBA	80% DBA	60% DBA	1.0 Ft^2	1.0 Ft^2	0.08 Ft^2	DBA	0.08 Ft^2
	HPCS Failure	LPCI DG Failure	LPCS DG Failure	HPCS DG Failure	HPCS DG Failure	HPCS DG Failure	HPCS DG Failure	HPCS DG Failure	HPCS DG Failure	HPCS DG Failure
Water Level in Hot and Avg Channels	A-1a	A-2a	A-3a	A-4a	A-5a	A-6a	A-7a	A-8a	B-1a	B-2a
Reactor Vessel Pressure	A-1b	A-2b	A-3b	A-4b	A-5b	A-6b	A-7b	A-8b	B-1b	B-2b
Peak Cladding Temp. (GE14)	A-1c	A-2c	A-3c	A-4c	A-5c	A-6c	A-7c	A-8c	B-1c	B-2c
Heat Transfer Coeff. (GE14)	A-1d	A-2d	A-3d	A-4d	A-5d	A-6d	A-7d	A-8d	B-1d	B-2d
ECCS Flow	A-1e	A-2e	A-3e	A-4e	A-5e	A-6e	A-7e	A-8e	B-1e	B-2e
Peak Cladding Temp. (GE10)	A-1f	NA	NA	NA	NA	NA	NA	NA	B-1f	NA
Heat Transfer Coeff. (GE10)	A-1g	NA	NA	NA	NA	NA	NA	NA	B-1g	NA
Peak Cladding Temp. (GE8)	A-1h	NA	NA	NA	NA	NA	NA	NA	B-1h	NA
Heat Transfer Coeff. (GE8)	A-1i	NA	NA	NA	NA	NA	NA	NA	B-1i	NA
Core Average Inlet Flow	A-1j	NA	NA	NA	NA	NA	NA	NA	B-1j	NA
Minimum Critical Power Radio	A-1k	NA	NA	NA	NA	NA	NA	NA	B-1k	NA

(Core Spray Line Break with LPCI Diversion)

	Appendix K Assumptions
	Small Break Method
	0.006 Ft^2
	LPCS DG Failure
Water Level in Hot and Avg Channels	C-1a
Reactor Vessel Pressure	C-1b
Peak Cladding Temp. (GE14)	C-1c
Heat Transfer Coeff. (GE14)	C-1d
ECCS Flow	C-1e

TABLE 6.3-4A
Key to Figures in Reference 18

(Recirculation Line Breaks)

	Nominal Assumptions		Appendix K Assumptions	
	Large Break (DBA)	Small Break ($0.07\ ft^2$)	Large Break (DBA)	Small Break ($0.06\ ft^2$)
Water Level in Hot and Average Channels	3-2	3-9	3-14	3-21
Reactor Vessel Pressure	3-3	3-10	3-15	3-22
Peak Cladding Temperature (GE14)	3-4	3-11	3-16	3-23
Heat Transfer Coefficient (GE14)	3-5	3-12	3-17	3-24
ECCS Flow	3-6	3-13	3-18	3-25
Core Average Inlet Flow	3-7	N/A	3-19	N/A
Minimum Critical Power Ratio	3-8	N/A	3-20	N/A

TABLE 6.3-4B
Key to Figures in Reference 19

(Recirculation Line Breaks)

	Nominal Assumption		Appendix K Assumptions	
	Large Break (DBA)	Small Break (0.07 ft^2)	Large Break (DBA)	Small Break (0.06 ft^2)
Water Level in Hot and Average Channels	1-a	3-a	2-a	4-a
Reactor Vessel Pressure	1-b	3-b	2-b	4-b
Peak Cladding Temperature (GNF2)	1-c	3-c	2-c	4-c
Heat Transfer Coefficient (GNF2)	1-d	3-d	2-d	4-d
ECCS Flow	1-e	3-e	2-e	4-e

TABLE 6.3-5

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NOT APPLICABLE TO BWR/6

TABLE 6.3-6

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TABLE 6.3-7
SINGLE FAILURE EVALUATION

The table below shows combinations of Automatic Depressurization System (ADS), High Pressure Core Spray (HPCS) System, Low Pressure Coolant Injection (LPCI) System and Low Pressure Core Spray (LPCS) System remaining operable following assumed single active failures. In performing the SAFER/GESTR-LOCA analysis, it was assumed that no postulated single active component failure will result in less than the minimum combinations of systems remaining operable as identified below.

The following single, active failures were considered in the ECCS performance evaluation for recirculation suction line break:

Assumed Failure(1)	**Systems Remaining**(2)(3)
LPCI Emergency Diesel Generator (D/G)	All ADS minus one, HPCS, LPCS,1 LPCI
LPCS Emergency D/G	All ADS minus one, HPCS, 2 LPCI
HPCS Emergency D/G	All ADS minus one, LPCS, 3 LPCI
One ADS Valve	All ADS minus two, HPCS, LPCS,3 LPCI

(1) Other postulated failures are not specifically considered because they all result in at least as much ECCS capacity as one of the assumed single failures listed above.

(2) The SAFER/GESTR-LOCA analysis was performed assuming one ADS valve inoperable in addition to the single failure being considered.

(3) Systems remaining, as identified in this table, are applicable to all non-ECCS line breaks. For a LOCA from an ECCS line break, the systems remaining are those listed, less the ECCS system in which the break is assumed.

TABLE 6.3-8
ECCS DESIGN PARAMETERS FOR CLINTON POWER STATION

SYSTEM	PARAMETER	DESIGN VALUE	BASIS
LPCS	Pool suction line design pressure	100 psig	Nominal value, suction from RPV (shutdown test)
RHR	" "	" "	" "
HPCS	Design pressure for suction from RCIC storage tank	" "	Nominal value, suction from RCIC storage tank
RHR	Shutdown suction line pressure	200	Max. vessel cut in pressure + max vessel water level above pump
LPCS	Pump discharge line pressure	600 psig	Shutoff head + max suction pressure
RHR(LPCI)	Pump discharge line pressure	500 psig	" "
HPCS	Pump discharge line pressure	1575 psig	" "
LPCS	Pump suction & discharge temp.	212°F***	Saturation at 1 atmosphere (Regulatory Guide 1.1)
HPCS	" "	" "	" "
RHR(LPCI)	Pool Suction Temp.	" "	" "
RHR	Shutdown Line Temperature	340°F	Max shutdown suction temperature (saturation @ 104 psig)
LPCS	Rated Flow	5010 gpm @ 119 psid*	Table 6.3-2
RHR(LPCI)	" "	@ 24 psid	5050 gpm/loop-three loops, Table 6.3-2
HPCS	" "	5010 gpm @ 200 psid*	Table 6.3-2 (Values selected to provide adequate core cooling for all design-basis events)
LPCS	RPV pressure at beginning flow	271 psid*	
RHR (LPCI)	RPV pressure at beginning flow	229 psid*	Table 6.3-2 (values selected to provide adequate core cooling for all design-basis events)
HPCS	" "	1200 psid*	" "
LPCSPump	Time to rated** speed	27 sec**	" "
RHR (LPCI)Pump	" "	"**	" "
HPCS Pump	" "	"**	" "
LPCS	Injection valve fully open	41 sec**	" "

TABLE 6.3-8
ECCS DESIGN PARAMETERS FOR CLINTON POWER STATION (CONTINUED)

SYSTEM	PARAMETER		DESIGN VALUE	BASIS	
RHR (LPCI)	"	"	42 sec**	"	"
HPCS	"	"	27 sec**	"	"
LPCS	Rated flow pump head		695 feet	Rated vessel pressure elevation difference from pool to vessel nozzle + frictional losses	
RHR (LPCI)	"	"	275 feet	"	"
HPCS	"	"	890 feet	"	“
				(suction from suppression pool)	

* psid, between suction source and RPV.
** Including power source availability.
*** 212°F is the temperature used for design of system components. The maximum expected temperature of pumped fluid, 185°F (i.e., maximum containment design temperature) is used for NPSH calculations.

TABLE 6.3-9

THIS TABLE HAS BEEN DELETED.

TABLE 6.3-10
MANUAL VALVES IN HPCS SYSTEM

VALVE NO.E22-	TYPE	LOCATION	SERVICE	FUNCTION	METHODS FOR MINIMIZING POSITIONING ERROR (See Note 2)
F036	10" Gate	Drywell	Main process line	Main process line block valve	Lo, position indicating light (control room mounted)
F026	3" Gate	Fuel BldgRR	Flushing	Backflush line by-passing check valve (F024)	Lc, closed
F003	3" Gate	Fuel Bldg	Flushing	Flushing water supply line to HPCS pump discharge	Lc, backed by check valve F024 and gate valve F031 (Lc)
F031	3" Gate	Fuel Bldg	Flushing	Flushing water supply line to HPCS pump discharge	Lc, backed by check valve F024 and gate valve F003 (Lc)
F034	2" Globe	Fuel Bldg	Water leg pump lines	Water leg pump suction isolation valve	Lo
F033	1-1/2" Globe	Fuel Bldg	Water leg pump lines	Water leg pump minimum flow line to HPCS pump suction	Open; during HPCS operation, position is not critical
F019	3" Gate	Fuel Bldg	Drain lines	HPCS drain to RHR system drains	Lc; backed by RHR valves F072A & B, F071 A&B, F070A and F069 all locked closed
F351	2" Globe	Fuel Bldg	Water leg pump test line	Water leg pump test line isolation	Lc, during HPCS operation, positions not critical

TABLE 6.3-10 (CONT'D)
MANUAL VALVES IN HPCS SYSTEM (CONTINUED)

VALVE NO.E22-	TYPE	LOCATION	SERVICE	FUNCTION	METHODS FOR MINIMIZING POSITIONING ERROR (See Note 2)
F314	20" Gate	Fuel Bldg	Main suction	Main suction maintenance	Lo
F301	14" Gate	Fuel Bldg	Main process line	Main process line maintenance	Lo

NOTES:

1. Piping low point drains, high point vents, and test connections are all double valved.

2. Lo = Locked open under administrative controls.
Lc = Locked closed under administrative controls
Backed by . . . = Double valve arrangement precluding impact on system operation without two positioning errors, and/or a non-manual valve failure.
Closed = Indicates valve is in line that forms closed loop with piping that,without a double positioning error, would have no effect on system functioning.

TABLE 6.3-11
MANUAL VALVES IN LPCS SYSTEM

VALVE NO.E21-	TYPE	LOCATION	SERVICE	FUNCTION	METHODS FOR MINIMIZING POSITIONING ERROR (See Note)
F007	10" Gate	Drywell	Main process line	Main process line block valve	Lo; position indicating light (control room mounted)
F025	3" Gate	Aux Bldg	Flushing	Flushing water supply to LPCS pump discharge piping to vessel	Lc; backed by check valve F306
F004	3" Gate	Aux Bldg	Flushing Draining Venting	Bypassing check valve F003	Lc*

*Bypass valve 1E21-F004 may be opened to vent the LPCS piping pressure prior to lifting the relief valve 1E21-F018.

F008	3" Gate	Aux Bldg	Flushing Servicing	LPCS pump suction line drain to radwaste system	Lc; backed by 1E12F072A, F071A, F071B, F072B, F072C, F069, F070 and 1E22F019, all locked closed
F302	4" Gate	Aux Bldg	Min, flow & test	Minimum flow & test return line to suppression pool	Lo; backed by motor operated F011
F032	2" Globe	Aux Bldg	Water leg pump lines	LPCS water leg pump suction isolation valve	Lo
F343	1-1/2" Globe	Aux Bldg	Water leg pump lines	Water leg pump minimum flow recirculation to LPCS suction lines	Lo; LPCS operation would not be affected by this valve's position (closed)
F301	20" Gate	Aux Bldg	Main suction	Main suction maintenance	Lo
F309	12" Gate	Aux Bldg	Main process line	Main process line	Lo
F310	2" Globe	Aux Bldg	Drain	Drain	Lc

TABLE 6.3-11 (CONT'D)

VALVE NO.E21-	TYPE	LOCATION	SERVICE	FUNCTION	METHODS FOR MINIMIZING POSITIONING ERROR (See Note)
F311	2" Globe	Aux Bldg	RHR test isolation	RHR test isolation	Lc
F312	2" Globe	Aux Bldg	RHR test isolation	RHR test isolation	Lc
F348	2" Globe	Aux Bldg	Water leg pump test line	Water leg pump test line isolation	Lc; during LPCS operation, position is not critical
F371	2" Globe	Aux Bldg	Water let pump injection line	Water leg pump injection line isolation	Lo
F372	2" Globe	Aux Bldg	Water let pump injection line	Water leg pump injection line isolation	Lo

NOTE: See Table 6.3-10

TABLE 6.3-12
ANALYSIS ASSUMPTIONS FOR NOMINAL CALCULATIONS

1.	Decay Heat	1979 American Nuclear Society (ANS) (Figure 6.3-9)
2.	Transition Boiling Temperature	Iloeje Correlation
3.	Break Flow	1.25 HEM(1) (Subcooled)
		1.0 HEM(1) (Saturated)
4.	Metal-Water Reaction	EPRI Coefficients
5.	Core Power	See References 13, 18, and 19
6.	Peak Linear Heat Generation Rate	See References 13, 18, and 19
7.	Bypass Leakage Coefficients	Nominal Values (Reference 7)
8.	Initial Operating Minimum Critical Power Ratio (MCPR)	See References 13, 18, and 19
9.	ECCS Water Enthalpy (Temperature)	88 Btu/lbm (120°F)
10.	ECCS Initiation Signals	See Table 6.3-2
11.	Automatic Depressurization System	125-Second Delay Time (Table 6.3-2)
12.	ECCS Available	Systems remaining after worst single failure
13.	Stored Energy	Best Estimate GESTR-LOCA
14.	Fuel Rod Internal Pressure	Best Estimate GESTR-LOCA
15.	Fuel Exposure	Limiting fuel exposure which maximizes PCT

(1) HEM: Homogeneous Equilibrium Model

TABLE 6.3-13
ANALYSIS ASSUMPTIONS FOR APPENDIX K CALCULATIONS

1.	Decay Heat	1971 ANS + 20% (Figure 6.3-9)
2.	Transition Boiling Temperature	Transition boiling allowed during blowdown only until cladding superheat exceeds 300°F
3.	Break Flow	Moody Slip Flow Model with discharge coefficients of 1.0, 0.8, and 0.6
4.	Metal-Water Reaction	Baker-Just
5.	Core Power	See References 13, 18, and 19
6.	Peak Linear Heat Generation Rate	See References 13, 18, and 19
7.	Bypass Leakage Coefficients	Same as Table 6.3-12
8.	Initial Operating Minimum Critical Power Ratio (MCPR)	See References 13, 18, and 19
9.	ECCS Water Enthalpy (Temperature)	Same as Table 6.3-12
10.	ECCS Initiation Signals	Same as Table 6.3-12
11.	Automatic Depressurization System	Same as Table 6.3-12
12.	ECCS Available	Same as Table 6.3-12
13.	Stored Energy	Same as Table 6.3-12
14.	Fuel Rod Internal Pressure	Same as Table 6.3-12
15.	Fuel Exposure	Same as Table 6.3-12

TABLE 6.3-14
PLANT OPERATIONAL PARAMETERS USED IN THE
CPS SAFER/GESTR-LOCA ANALYSIS[(1)]

Plant Parameters	Nominal	Appendix K
Core Thermal Power (MWt)	3473	3542.5
Corresponding Power (% of 3473 MWt)	100	102
Vessel Steam Output (Mlb/hr)	15.15	15.52
Vessel Steam Output (% rated)	100	102.4
Core Flow (Mlb/hr)	84.5	84.5
Core Flow (% rated)	100	100
Vessel Steam Dome Pressure (psia)	1040	1060
Maximum Recirculation Suction Line Break Area (ft^2)	2.207	2.207
Bottom Head Drain Line Break Area (ft^2)	0.016	0.016

(1) Per References 13, 18, and 19

6.4 HABITABILITY SYSTEMS

Habitability systems are designed to ensure that main control room personnel can remain inside all spaces served by the control room HVAC system during accident conditions by providing adequate protection against radiaton and toxic gases, in compliance with Criterion 19 of Appendix A to 10 CFR 50. The habitability systems cover all equipment, supplies, and procedures necessary to ensure that control room personnel are protected from postulated releases of radioactive materials, toxic gases, and steam. Adequate food, water storage, kitchen and sanitary facilities, and medical supplies are provided to meet the needs of operating personnel during and after an incident. In addition, the environment in all spaces served by the control room HVAC system (control room envelope) is controlled within specified limits which are conducive to prolonged service life of Safety Class 2 components during all station conditions. The control room HVAC is designed to ensure that main control room personnel can remain inside all spaces served by the control room HVAC system during normal station conditions, in compliance with Criterion 19 of Appendix A to 10CFR50, as detailed in section 9.4.1.

The control room HVAC smoke mode is not a part of the Habitability Design Basis. The smoke mode is addressed in section 9.4.1.

6.4.1 Design Basis

Design bases for the habitability systems' functional design are as follows:

a. Redundant trains of HVAC equipment maintain habitable environmental conditions in the control room envelope.

b. The habitability systems will support a minimum of seven people during normal operation and for 30 days of abnormal operation. A minimum of 5 days of food, water, and medical supplies is provided for an emergency control room staff, with additional food resupplied as needed.

c. Kitchen and sanitary facilities are provided in the control room envelope.

d. The potential radiological effects on the control room envelope of any incident described in Chapter 15 were considered.

e. Provisions to preclude significant entry of toxic gases (carbon dioxide, etc.) from inside or outside the plant.

f. Adequate self-contained breathing apparatus and a minimum of 6 hours of bottled air are provided inside the control room envelope for each member of the emergency staff. Unlimited offsite supplies are available from nearby locations.

g. The habitability systems will operate effectively during and after a design-basis accident (i.e., LOCA) with the simultaneous loss of offsite power, safe shutdown earthquake, or failure of any one of the control room HVAC components.

h. Radiation monitors continuously monitor the two control room HVAC system minimum outside air intakes. Detection of high radiation is alarmed in the control room and related protection functions are initiated. The pressure differential between the control room envelope and surrounding areas is measured and indicated in the control room. Individual room temperature indicators are provided for the control room envelope areas. Increased temperature due to failure of humidifier steam line is alarmed.

During Hi Rad mode, 3000 cfm of outside air is introduced into the control room envelope to maintain greater than or equal to 0.125 in. H_20 positive pressure with respect to the surrounding areas.

i. During station blackout a minimum of 5000 CFM of outside air may be introduced into the control room to maintain temperature less than 120° F. Portable gas operated fans are used to exhaust air from the control room.

j. The control room envelope (CRE) boundary is maintained to ensure that unfiltered inleakage into the CRE will not exceed the inleakage assumed in the licensing basis analysis of Design Basis Accident consequences to CRE occupants.

6.4.2 System Design

6.4.2.1 Definition of Control Room Envelope

The current control room envelope, which has a volume of 405,134 ft^3, consists of the control room and surrounding equipment and personnel support areas shown on Drawing M01-1108-6, and ventilated by the control room HVAC system.

6.4.2.2 Ventilation System Design

The detailed control room HVAC system description is in Subsection 9.4.1.

All of the system components are designed to perform their functions during and after the safe shutdown earthquake, except for the exhaust fan and the electric space heating and humidification equipment, which are physically supported to remain intact, but might not function other than to maintain airflow integrity. All active system components are protected from internally and externally generated missiles. A layout of the control room envelope, showing doors, corridors, stairways, shield walls, and the equipment layout is given in Drawing M01-1108-6.

The description of the controls and instrumentation, including the ionization and radiation monitors for the control room HVAC system is included in Subsections 7.1.2.1 and 7.3.1.1. The locations of the two minimum outside air intakes, the control room, and potential sources of radioactive and toxic gas releases are indicated in Figure 6.4-3.

A detailed description of the control room HVAC system makeup air filter trains is presented in Subsection 6.5.1.

6.4.2.3 Leak Tightness

The entire control room boundary is designed for low leakage. All boundary penetrations are sealed. The access doors are of airtight design with self-closing devices which shut the doors automatically following the passage of personnel.

The control room leakage criteria are based on the methods and assumptions given in "Conventional Building for Reactor Containment," Atomics International (NAA-SR-10100), and Regulatory Guide 1.78. Table 6.4-1 provides a listing of leakage data and total leakage for all leak paths.

Exfiltration sources include structural cracks, six doors, piping and duct penetrations, HVAC equipment, conduits and cables. In accordance with Regulatory Guide 1.78, position C.9, the control room makeup air flow is sized for a leakage from the control room boundary when the control room is pressurized to a positive pressure differential of 1/4 inch water gauge. The maximum mechanical exhaust and exfiltration from the control room envelope under all conditions except during postulated high radiation accidents and chlorine mode of operations is estimated to less than 4000 cfm. During a postulated high radiation accident and chlorine mode of operation, the makeup air dampers are closed and the exhaust fan for the toilets and locker room is shut off automatically, and the exfiltration is assumed to be 3000 cfm.

Determining the amount of unfiltered air inleakage is in accordance with the testing methods and frequencies specified in Sections C.1 and C.2 of Regulatory Guide 1.197, “Demonstrating Control Room Envelope Integrity at Nuclear Power Reactors” Revision 0, May 2003.

6.4.2.4 Interaction with Other Zones and Pressure-Containing Equipment

The control room HVAC system serves only rooms in the control room envelope. Exhaust air for the toilets and locker room is discharged to the atmosphere via a penthouse on the control building roof. Areas surrounding the control room envelope are served by the Auxiliary Building HVAC system.

During normal operation, the surrounding areas adjacent to the control room envelope will be at a lesser pressure with respect to the control room envelope. Normal access paths between other plant areas and the control room envelope are vestibules with two doors in series to minimize leakage.

There are no high energy lines in the proximity of or within the control room envelope that can affect control room habitability. The control room humidifier is located in the control room HVAC equipment room. Failure of humidifier steam line will actuate an alarm and operator action is initiated to isolate the humidifier. A Halon fire protection system is provided within the control room envelope (see Subsection 9.5.1.2.2.4).

6.4.2.5 Shielding Design

Radiation shielding for the control room is based upon radiation sources released from the core following a postulated loss-of-coolant accident (LOCA), and is to protect the inhabitants in accordance with Criterion 19 of 10 CFR 50, Appendix A.

Radiation source assumptions are in accordance with Regulatory Guide 1.3. The distribution of sources following the accident is calculated using the primary containment design-basis leak rate (see Subsection 6.2.1) and assuming that the standby gas treatment system is evacuating the secondary containment according to its design (see Subsection 6.5.1).

Specific radiation sources and the attenuation parameters associated with each of them used to determine the control room doses are tabulated in Table 6.4-2. The time dependent strength, radiation type and energy of each source, the shielding thickness, and the resulting doses to the control room personnel were calculated using techniques discussed in Subsection 12.3.2.6.

An isometric view of the control room is shown in Figure 6.4-3. Shielding details are provided in Table 6.4-2 and Drawings M01-1524 and M01-1526. Labyrinths at the entrances and shielding slabs over the ceiling penetrations are provided to preclude potential radiation streaming.

6.4.3 System Operating Procedures

During normal operation, one of the two 100% capacity trains of the main control room HVAC system continuously processes the recirculated and makeup air to maintain desired air temperature and quality. The mixture of recirculated and makeup air is filtered by high efficiency (85% by NBS dust spot method), waterproof and fire retardant fiberglass filters. Each control room HVAC system train, with the exception of the chiller, may be manually started through its respective control switch located in the main control room. The system chiller must be manually started from the chiller control panel. Local control of system train components is provided by allowing transfer of system control to the local control panel using the remote/local selector switch. The sequence of operation of the controls is described in Subsection 7.3.1.1.6.

In the event of high radiation detection at the minimum outside air intake(s), the radiation monitoring system will activate an alarm in the main control room, automatically start the makeup filter train, route the supply air stream through the charcoal beds in the recirculation air filter associated with the operating HVAC air handling system and trip the locker room exhaust fan. The 3000 cfm makeup air is automatically routed through the makeup air filter train, for removal of radioactive and nonradioactive particulates and iodine, before going into the control room. The control room operator can operate handswitches to close the minimum outside air intake damper, which was being used, and to open the other intake damper, to take advantage of the separation of the two air intakes and minimize radioactivity intake.

Areas above and below those spaces served by the main control room HVAC system will be maintained at a negative pressure, with respect to those spaces served by the control room HVAC system, by the auxiliary building HVAC system.

6.4.4 Design Evaluations

The control room HVAC system is designed to maintain a habitable environment compatible with prolonged service life of safety-related components in the control room under all station operating conditions. The system is provided with redundant equipment trains to meet the single failure criteria. The equipment trains are powered from redundant ESF buses and are operable during loss of offsite power. All of the control room HVAC system equipment except recirculation, heating and humidification equipment, and exhaust fan is designed for Seismic Category I loads.

6.4.4.1 Radiological Protection

The two outside minimum air intakes are separated from each other by being located on the east and west sides of the plant respectively. In the post-LOCA environment, air can be supplied to the control room from the intake where the airborne contamination is lowest.

Radiation monitors near the minimum outside air intakes are designed to initiate an alarm in the control room and a control signal to isolate the normal ventilation path, start up one of the makeup air filter trains and divert the outside air through it, route the supply air stream through the recirculation air charcoal adsorber and trip the locker room exhaust fan. Subsequently, readings from these monitors are used by the operator to select the minimum outside air intake with the lower airborne contamination. The makeup air filter trains are provided with HEPA and charcoal filters to remove particulates and iodine from the makeup air. The recirculation filter units are provided with high efficiency air and charcoal filters to further remove particulates and iodine from the supply air.

The control room is maintained at a positive pressure with respect to adjacent areas when operated in the radiation mode to minimize the ingress of unfiltered outside air. Air locks are also provided at the entrances for the same purpose.

The calculated dose to personnel inside the control room following accidents is reported in Chapter 15.

6.4.4.2 Toxic Gas Protection

Transportation and traffic surveys have shown that the frequency of transportation or delivery of chlorine does not dictate design for a potential chlorine hazard. Additionally, gaseous chlorine is no longer allowed on site by plant procedure and there are no other significant depots of chlorine within a five mile radius of the site. Therefore, no automatic initiation of the control room ventilation chlorine mode and no chlorine detectors are required.

A breathing air system is provided for control room operators. Protection of control room operators against other hazardous chemicals as required by Regulatory Guide 1.78 is provided.

There are three control room ventilation intakes. The first is located on the northwest corner of the Auxiliary Building. The second intake is located on the north wall of the Control Building. The third intake is located on the east wall of the Control Building.

In the highly unlikely event of a chlorine gas problem, the control room operator may manually initiate the chlorine mode of operation for the Control Room HVAC System. (Q&R 450.1)

For refrigerant vapors (freon) or their decomposition products, protection is provided by:

a. Locating refrigerant-using equipment (chillers) with the most refrigerant far from the control room, on the lowest elevation of the control and fuel buildings. These areas are ventilated independently from the control room system.

b. Providing pressure relief valves on each chiller piped directly to the outdoors, except for the drywell water chillers relief line which is connected to the fuel building exhaust system.

c. Pressurizing the control room to preclude infiltration.

d. Ventilating the control room HVAC equipment room independently from the control room HVAC system.

e. Utilizing chilled water cooling coils instead of direct expansion coils.

f. Requiring that the control room chillers are high quality machines that meet Safety Class 3, Quality Group C, Quality Assurance Requirement B, and Seismic Category I. The refrigeration loop is seismically qualified and is designed and tested to meet environmental conditions and radiation requirements. The units meet the applicable requirements of Section VIII, Division I of the ASME Code, ANSI B31.5 and ANSI B9.1.

Of the other potentially hazardous chemicals stored on site, listed in Table 2.2-6 only sulfuric acid, carbon dioxide, and nitrogen are included in Regulatory Guide 1.78. The following are features protecting against potential problems upon a release of sulfuric acid:

a. Sulfuric acid has a low vapor pressure (< 1 Torr),

b. The relative location of the sulfuric acid storage facility with respect to the control room minimum outside air intakes, and

c. The acid storage tank is vented to the outside. Fumes from spillage within the acid storage area are diluted by the exhaust air from the sulfuric acid storage area with the radwaste building and balance of the plant exhaust air streams.

   Analysis has shown that a postulated rupture in the carbon dioxide storage system does not result in an unacceptable concentration of CO_2 within the control room. Since the amount of nitrogen stored onsite is not a significant fraction of the control room volume, per Regulatory Guide 1.78, it does not need to be considered.

Design for toxic gases from offsite sources was not implemented since the frequency of transportation or delivery of those gases does not dictate it. As discussed in Section 2.2, the only identified hazardous product or material regularly stored, manufactured, or used within 5 miles of the station has been determined not to pose a significant risk to control room habitability and therefore the design basis for the control room ventilation system is unaffected.

The control room HVAC system does not provide automatic hazardous chemical protection to the control room envelope in case of either an onsite or offsite hazardous chemical accident. Operators are protected against the chemical threat by placing the control room HVAC system in the recirculation mode within two minutes of detection of a toxic chemical. This mode provides for 100% recirculated air with no outside air makeup.

The control room emergency breathing air system consists of two 14 bottle high pressure cascade breathing air systems (see Drawing M05-1065). Each bottle contains 300 scf of breathing air. Each system is capable of supplying sufficient breathing air for seven people for six hours. Each system of bottles has a remote fill station located outside the external south wall of the diesel generator building that serves to recharge the bottles if their use is required for longer than six hours. Air stations are located so that all areas of the control room are accessible. Each of the eight pressure demand masks available to control room personnel is supplied with a 5-minute or larger egress bottle. Since they will be stored in the control room and adjacent offices, and will be readily accessible, it is anticipated that donning time for the masks will be less than 2 minutes. The system meets the single failure criteria and is seismically designed and supported, however it is not nuclear safety-related. The system meets all applicable requirements of Regulatory Guide 1.95. Training in the use of this system is included in the CPS Respiratory Protection Program. This consists of detailed initial training and biennial retraining.

Testing and maintenance of the system is included in the CPS Respiratory Protection Program. The masks will be inspected at least monthly and after use, in accordance with manufacturers instructions.

6.4.4.3 Fire Protection

The likelihood of an equipment fire affecting control room habitability is minimized because early ionization detection is assured, fire-fighting apparatus is available, plus filtration and purging capability are provided.

The following provisions 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 barriers to prevent damage to both circuits by a single fire.

c. Most of the cable insulation's 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 studs. The control room ceiling, door frames, and doors are metallic. Wood trim is not used.

e. A fire hazards analysis of the Power Generation Control Complex by General Electric is contained in NED0-10466A.

Charcoal adsorbers are provided with deluge fire protection systems which are remote manually activated from the control room after the high or high-high temperature alarm points are reached. Halon is the only fire fighting chemical normally used in the main control room, therfore personnel protection from fire fighting chemicals is not required. Dry chemical fire extinguishers and fire hose stations are also provided within the habitability area outside the main control room.

6.4.5 Testing and Inspection

The main control room HVAC system and its components are thoroughly tested in a program consisting of the following:

a. factory component qualification tests,

b. onsite preoperational testing.

c. onsite subsequent periodic testing.

The CRE is tested in general conformance to Regulatory Guide 1.197 Revision 0, Sections C.1 and C.2.

Written test procedures establish minimum acceptable test values. Test results are recorded as a matter of performance record, thus enabling early detection of faulty performance.

NOTE: Discussions of factory inspection/testing and preoperational testing are considered historical.

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 compliance with installation specifications. Operating pressures and flows are balanced to design values. Controls, interlocks, and safety devices are cold checked (prior to operation), adjusted, and tested to ensure the proper sequence of operation.

Preoperational testing of the makeup air filter trains, recirculation air filter trains, and their components is described in Subsection 6.5.1.4. The preoperational test of the control room HVAC system, as described in Chapter 14, confirms the operability of the system and controls, in all modes and ensures that design air flow is attained.

The system is periodically tested, as described in the Technical Specifications. The smoke mode of operation and the associated components are subjected to the factory, preoperational, and subsequent periodic tests described in Subsection 9.4.1.4. The balance of the control room HVAC system is proven by its use during normal system operation.

6.4.6 Instrumentation Requirements

With the exception of fail-open, air-operated valves used to modulate chilled water flow through the equipment room chilled water coils, all instruments and controls for the control room HVAC system are electric or electronic. Control of various operating functions and the monitoring capabilities are described in detail in Subsection 7.3.1.1.6.

Important operating functions (e.g., start/stop system trains, etc.) are controlled and monitored from the main control room. Each redundant control room HVAC system train has a local control panel and is controlled independently from the other.

Instrumentation is provided to monitor important variables (e.g., computer room temperature and humidity, etc.) associated with normal operation, and to alarm abnormal conditions, (high radiation in minimum outside air intakes, etc.) on the main control board.

The control room HVAC system is designed for automatic environmental control after manual startup.

A redundant radiation detection system is provided to monitor the radiation levels in the minimum outside air intakes. A high radiation signal provides annunciation on the main control board, initiates startup of a control room makeup air train and actuates isolation dampers.

An ionization detection system is provided to detect smoke and products of combustion in areas served by the control room HVAC system. An abnormal signal provides annunciation on the fire protection status panel on the main control board. This signal results in opening of the charcoal adsorber isolation dampers and closing of the filter bypass damper.

A fire protection system supplies station fire protection water to each charcoal adsorber through motor-operated deluge valves. High temperature in the charcoal filters is annunciated on the main control board. The operator may elect to manually open the deluge valves (see 1 Subsection 7.3.1.1.6).

A pressure sensing system is provided to indicate, on the main control board, the control room pressure with respect to adjacent areas.

TABLE 6.4-1
CONTROL ROOM LEAKAGE ANALYSIS

LEAK PATH	DIFFERENTIAL PRESSURE (in. water gauge)	LEAK PATH (cfm)
Doors (Total of 6)	1/4	60
Structural (Construction Cracks)	1/4	820
Pipe and Instrumentation	1/4	20
Duct Penetration	1/4	10
HVAC Equipment and Ducts	varies	500
HVAC Mechanical Exhaust	N/A	1,000
Conduits and Cable Pans	1/4	500
Subtotal (Approx.)		2,910
Design Margin		1090
Total cfm		4,000

NOTE: The control room leakage is estimated at 1/4 inch water gauge, in accordance with Regulatory Guide 1.78, position C.9. This does not require the control room actually be pressurized to 1/4 inch during normal or accident conditions.

TABLE 6.4-2
RADIATION SOURCES AND THE ASSOCIATED ATTENUATION PARAMETERS USED IN THE DETERMINATION CONTROL ROOM DOSES

SOURCE	SOURCE GEOMETRY	DISTANCE FROM THE NEAREST CONTROL ROOM BOUNDARY	MINIMUM TOTAL SHIELDING THICKNESS*
Primary Containment Airborne Activity	Cylindrical	20.5 ft	5.5 ft
Primary Containment Plateout Activity	Annular	20.5 ft	5.5 ft
Secondary Containment Airborne			
Fuel Building	Rectangular Solid	35.5 ft	5.0 ft
Auxiliary Building	Rectangular Solid	2.8 ft	2.8 ft
Gas Control Building	Annular	17.5 ft	2.5 ft
Standby Gas Treatment Filter	Point	68.6 ft	6.08 ft
Cloud	Rectangular Solid	2.0 ft	2.0 ft
Control Room Air Intake Filter	Point	2.03 ft	2.0 ft

* Shielding material is structural concrete.

6.5 FISSION PRODUCT REMOVAL AND CONTROL SYSTEMS

Fission product removal and control systems are considered to be those systems for which credit is taken in reducing accidental release of fission products. The filter systems and containment spray systems for fission product removal are discussed in Subsections 6.5.1 and 6.5.2, and the fission product control systems in Subsection 6.5.3.

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 iodine and particulate concentrations in gases leaking from the primary containment and which are potentially present in the secondary containment following an accident.

b. Control room HVAC makeup air filter packages: This system is utilized to clean the outside air of radioactive and nonradioactive iodine and particulates, which are potentially present in outside air following an accident, before introducing air into the control room HVAC system.

c. Control room HVAC recirculation air filter packages: This system is utilized to clean internally recirculated air of residual radioactive and non-radioactive iodine.

6.5.1.1 Design Bases

6.5.1.1.1 Standby Gas Treatment System

a. The standby gas treatment system can be started manually and is designed to automatically start in response to any one of the following signals:

    1. high drywell pressure,

    2. low reactor water, level 2,

    3. high radiation level in exhaust air from the fuel transfer floor of the containment,

    4. high radiation level in the containment building ventilation exhaust,

    5. high radiation level in the exhaust air from the fuel building fuel handling floor, and

    6. high radiation level in the continuous containment purge exhaust duct.

b. The radioactive gases leaking from the primary containment to the secondary containment (which consists of containment gas control boundary extension, fuel building, ECCS pump rooms, RWCU pump rooms, and main steam tunnel) after a LOCA are treated in order to remove particulate and gaseous forms of iodine. This is to limit the offsite and control room dose to the guidelines of 10 CFR 50.67.

c. The standby gas treatment system (SGTS) equipment train air handling capability is based on the total inleakages to the secondary containment while all of the areas in the secondary containment are maintained at 0.25 inch water gauge negative pressure with respect to outside ambient pressure to preclude ground level leakage of untreated air to the environment. The secondary containment air pressure begins to decrease exponentially after the standby gas treatment system is started. For low wind speeds, a design pressure of 0.25 inch water gauge is reached within 19 minutes after this design basis LOCA. The time period until the secondary containment reaches a negative pressure of 0.25 inch water gauge should not be considered as a period of direct outleakage for the following reasons:

    1. The pressure gradient forcing leakage from the primary containment is less than 4 psig during this time period. The containment design and construction, and testing requirements provide leakage integrity and such a small pressure difference provides little driving force for leakage across small leak paths.

    2. The most predominant potential containment leak paths are piping penetrations and door seals which penetrate the containment at elevations enclosed by the secondary containment which consists of the ECCS pump rooms, steam tunnel, RWCU pump rooms, and fuel building. Due to the large volume of these areas, the small amount of radioactive gases leaking through would require some interval of time to diffuse through the secondary containment to the outside.

    3. Fuel cladding is not postulated to fail prior to containment isolation.

    4. The entire secondary containment, including the containment gas control boundary (CGCB), is maintained at approximately 0.25 inch water gauge negative pressure during normal operations.

d. Primary containment leakage, except for bypass leakage through the upper personnel air lock, will be contained within the secondary containment and will be processed 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 expected SGTS flow rate is approximately equal to the total free air volume of the fuel building, ECCS pump rooms, RWCU pump rooms, steam pipe tunnel, and the containment gas control boundary evacuated at a rate of one per day. The design flow rate through the SGTS also accounts for volumetric expansion of building air volumes due to temperature rises as equipment residual heat is released after the non-safety-related ventilation and process system shutdown.

e. The secondary containment leakage is calculated based laminar flow characteristic through small cracks.

The portion of secondary containment most affected by wind effects is the containment gas control boundary (CGCB) based on its construction, air change rate, and percentage of total secondary containment surface area exposed to direct winds. The SGTS fans are adequately sized to pull secondary containment negative for low to moderate wind speed.

The analysis shows that no secondary containment outleakage (bypass of SGTS) results for low to moderate wind speeds up to approximately 30 mph. Above this wind speed, the secondary containment outleakage increases gradually without increasing offsite dose rates due to more favorable atmospheric dispersion conditions.

Additional analyses indicate that for low wind speeds the secondary containment pressures of -0.25 inch water gauge is reached in less than 19 minutes after LOCA.

f. Two single unit capacity standby gas treatment system equipment trains (SGTSET) and associated dampers, piping, instruments, and controls are provided. The system is in compliance with the intent of Regulatory Guide 1.52 as described in Table 6.5-3.

g. Each SGTSET is sized and specified for the worst conditions, treating incoming air-steam mixtures saturated at 150°F containing fission products and particulates at a rate equivalent to the containment and main steam isolation valve design leakage. Fission products available in the containment atmosphere for release were determined in accordance with Regulatory Guide 1.3 and for Alternative Source Term analyses, Regulatory Guide 1.183.

h. Each equipment train contains the amount of charcoal required to adsorb the total amount of halogen fission products which leak from primary containment into secondary containment.

i. Each train is provided with a demister, air heater, and prefilter 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 SGTSET to remove heat generated by fission product decay on the HEPA filters and charcoal adsorbers after shutdown of the train.

Charcoal desorption temperature is given in ERDA 76-21. No credit is taken for equipment or environment heat sink. Control building cooling air is routed through the shutdown train and exhausted to the atmosphere.

k. The SGTS exhibits a removal efficiency of no less than 99% on radioactive and nonradioactive forms of iodine and no less than 99% on particulate matter. The charcoal is contained in gasketless, all welded construction absorbers to preclude bypass of the charcoal and to ensure the highest removal efficiencies for methyl iodine.

   The exhaust air from each SGTS is routed through seismically supported piping and released at an elevation of 935 feet 6 inches. The discharge air velocity from the SGTS vent exhaust pipe is 2460 fpm. This high point release (approximately 200 feet above grade) provides effluent dispersion ratios sufficient to meet the requirement of 10 CFR 50.67.

l. The SGTS is designed with redundancy to meet single failure criteria and conform to IEEE-323 and IEEE-344.

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 Control Room HVAC Makeup Air Filter Packages

a. The control room HVAC makeup air filter packages are designed to start automatically and provide outside air to the control room HVAC system in response to any one of the following signals:

   1. high radiation signal from the radiation monitors installed in minimum outside air intakes and control room HVAC system; and

   2. manual activation from the main control room.

b. Regulatory Guide 1.3, and for Alternative Source Term analyses, Regulatory Guide 1.183, assumptions are used to calculate the quantity of activity released as a result of an accident and to determine inlet concentrations to the control room HVAC makeup air filter package.

c. The capacity of the control room HVAC makeup air filter packages is based on the air quantity required to maintain the rooms served by the control room HVAC system at 1/8 inch H_2O positive pressure with respect to adjacent areas while providing makeup air for the control room toilets and locker room exhaust. The exhaust fan is isolated on a high radiation signal.

d. Two full capacity control room HVAC makeup air filter packages and associated dampers, ducts, and controls are provided.

e. Each package is provided with a demister, electric air heater, and prefilter needed to assure the optimum air conditions entering the high efficiency particulate air (HEPA) and charcoal filters.

f. The control room HVAC makeup air filter packages exhibit a removal efficiency of not less than 99% on radioactive and nonradioactive forms of iodine and not less than 99% on particulate matter.

g. Two trains are provided 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 the requirements of IEEE-279.

i. The control room HVAC makeup air filter packages are designed to Seismic Category I requirements.

j. The control room HVAC makeup air filter packages are designed to permit periodic testing and inspection of principal system components described in the following subsections.

k. The electrical components are qualified in accordance with IEEE-323 and IEEE-344.

6.5.1.1.3 Control Room HVAC Recirculation Air Filter Packages

a. The control room HVAC recirculation air filter packages are designed to divert the air flow from around to through the charcoal filter automatically in response to any one of the following signals:

1. High radiation signal from the radiation monitors installed in the minimum outside air intakes;

2. Detection of smoke or products of combustion in the areas served by the main control room HVAC system; and

3. Manual actuation from the main control room.

b. Two full capacity control room HVAC recirculation air filter packages and associated dampers, ducts, and controls are provided.

c. Each package consists of a prefilter, needed to remove excessive particulate matter to prevent excessive buildup in the charcoal adsorbent bed, and a charcoal filter.

d. The control room HVAC recirculation air charcoal filter has a removal efficiency of greater than 70% on all forms of iodine.

e. Two trains are provided to meet single failure criteria. These trains are physically separated so that damage to one unit does not cause damage to the second unit.

f. The power supplies meet IEEE-308 criteria and insure uninterrupted operation in the event of a loss of normal a-c power. The controls meet the requirements of IEEE-279. The electric components are qualified in accordance with IEEE-323 and IEEE-344.

g. The filter trains are designed to Seismic Category I requirements.

h. The control room HVAC recirculation air filter packages are designed to permit periodic testing and inspection of principal system components.

i. The operation of the recirculation filter units is compatible with the operation of the make-up filter units.

j. During normal operation of the control room HVAC system the recirculation filter charcoal adsorber bed is isolated (i.e., no flow through the bed). Thus the charcoal adsorbent retains its full capacity to remove iodine when placed in service.

k. The quality of charcoal for the recirculation filter units is identical to the quality of the charcoal adsorber used in the make-up filter package units.

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 M05-1105. Nominal sizes of principal system components are listed in Table 6.5-1. The equipment environmental design criteria are listed in Section 3.11 and redundant trains are physically separated as illustrated in Drawing M05-1073.

b. The SGTS is automatically or manually started to treat air exhausted from the CGCB, fuel building, steam tunnel, and ECCS pump rooms. Two completely redundant parallel process systems are provided, each having a minimum capacity of 4000 cfm (at 175° F) of which approximately 2000 cfm comes from the fuel building, 1000 cfm from the containment gas control annulus, 1000 cfm from all ECCS pump rooms and steam pipe tunnel.

   As indicated on the schematic in Drawing M05-1105 each process system may be considered as an installed spare, with either standby gas treatment system equipment train (SGTSET) capable of treating the required amount of air. The process systems have separate equipment trains, isolation valves, power feeds, controls and instrumentation. Each SGTSET is provided with segregated and independent suction and discharge pipes. The SGTSET are located in completely separate concrete equipment cubicles. Separation of filter trains is maintained in areas where credible internal missiles or pipe whips might compromise redundancy.

c. Each SGTSET has the following components:

   1. A primary fan for inducing the air from the spaces listed previously and through the filter train to the discharge pipe for the elevated release to atmosphere. The fan performance and motor selection are based on inducing maximum density (worst pressure condition) 40° F air from previously mentioned areas and forcing it through a filter train containing filters operating at a pressure drop of no less than twice their clean value. The flow and pressures are listed in Table 6.5-1. The fans are statically and dynamically balanced in conformance with ANSI N509-1976, Article 5.7.3.

   2. A standby cooling air fan is sized to dissipate heat generated by fission product decay on the filters. 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 prevent blinding of filters which reduces iodine removal efficiency of charcoal adsorbers. The demister meets qualification requirements of those specified in ANSI N509-1976, Section 5.4.

   4. A single stage electric heater is sized to maintain the humidity of the airstream to no more than 70% relative humidity for the worst inlet conditions. A 20-kW heater is provided. The electric air heaters meet the requirements of ANSI N509-1976, Article 5.5.

5. A prefilter, UL listed, all-glass media, exhibiting no less than 85% efficiency based on ASHRAE atmospheric dust spot test. The prefilters meet the requirements of ANSI N509-1976, Article 5.3.

6. A high-efficiency particulate air (HEPA) filter, water resistant, factory tested, to be capable of removing 99.97% minimum of particulate matter 0.3 micron or larger in size. The filter is designed to be fire resistant. Four 1000 cfm elements are provided. All elements are fabricated in accordance with Military Specifications MIL-F-51068D, MIL-F-51079B, and UL-586. Testing of the HEPA filter banks is described in Subsection 6.5.1.4. The HEPA filters meet the requirements of ANSI N509, Section 5.1 (See Table 6.5-3, Section C.3.d for year of compliance).

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 charcoal. The bank holds a total of 1800 pounds of charcoal of approximately 30 lb/ft^3 density having a minimum ignition temperature of 626° F.

   The bed is so designed that air has at least 0.5 seconds of residence time through the charcoal (i.e., a velocity of 40 fpm). Qualification of charcoal shall meet the requirements of ANSI N509-1980.

   Ten test canisters are provided for each adsorber. These canisters contain the same depth (i.e., 4 inches) 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 or a representative sample is removed and laboratory tested to reverify the adsorbent efficiency. Two deluge valves in parallel connected to the station fire protection system and shutdown service water systems are mounted outside the charcoal adsorber.

   Each charcoal bed is provided with a temperature detector with dual settings. High adsorber temperature will actuate an alarm in the main control room. High temperature alarms are set at 200° and 250° F. After receiving one of the alarms, the operator may manually activate any deluge valve via control switches on the main control board, spraying the adsorber compartment and thereby precluding the chance of an adsorber fire.

8. A high efficiency particulate air filter identical to the one described in Item 6 previously, is provided to trap the charcoal fines which may be entrained by the airstream.

d. A flow control single blade damper is utilized upstream of each train to regulate flow through it.

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 filter housing is of all welded construction, and is heavily reinforced. The materials and construction of the filter housings meet the requirements of ANSI N509-1976, Article 5.6.4.1.

g. Interior lights with external light switches are provided between all train components to facilitate inspection, testing, and replacement of components.

h. HEPA filter frames are in accordance with recommendations of Section 4.3.5 of ERDA 76-21.

i. The height of release of the standby gas treatment system vent to the atmosphere is at elevation 935 feet 6 inches above mean sea level, which is approximately 200 feet above grade.

j. The SGTS filter housings are drained in accordance with ERDA 76-21, Paragraph 4.5.8.

6.5.1.2.2 Control Room HVAC Makeup Air Filter Packages

a. The control room HVAC makeup air filter packages work in conjunction with the control room HVAC system as described in Subsections 9.4.1 and 6.4.2.2. The schematic design is shown in Drawing M05-1102. The nominal size of principal system components is listed in Table 9.4-1. The equipment environmental design criteria are listed in Section 3.11 and redundant trains are physically separated as illustrated in Drawing M01-1108-8.

b. In the event of high radiation detection in the minimum outside air intakes of the control room HVAC system, the radiation monitoring system automatically starts one of the makeup filter trains, shuts off the normal flow path of the makeup air, and draws outside air through the filter train.

c. Two control room HVAC makeup air filter trains and fans are provided, each capable of handling 3000 cfm of outside air.

d. Each control room HVAC 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 meets qualification requirements of those specified in ANSI N509-1976, Section 5.4.

    2. A single stage electric heater, sized to maintain the humidity of the airstream to no more than 70% relative humidity for the worst inlet conditions. An analysis of heater capacities for various entering saturated air conditions ranging from -2° F to 96° F yields a peak heating requirement of approximately 49,800 Btu/hr at 96°F. A 16-kW heater is provided. This heater meets the requirements of ANSI N-509-1976, Article 5.5.

3. A prefilter, UL listed, all glass media, exhibiting no less than 85% efficiency based on ASHRAE atmospheric dust spot test. This prefilter meets the requirements of ANSI N-509-1976, Article 5.3.

4. A high-efficiency particulate air (HEPA) filter, water resistant, factory tested, to be capable of removing 99.97% minimum of particulate matter 0.3 micron or larger in size. The filter is designed to be fire resistant. Four 1000 cfm elements are provided. All elements are fabricated in accordance with Military Specifications MIL-F-51068D, MIL-F-51079B, and UL-586. Testing of the HEPA filter banks is described in Subsection 6.5.1.4. The HEPA filters meet the requirements of ANSI N509, Section 5.1 (See FSAR Table 6.5-3, Section C.3.d for year of compliance).

5. A charcoal adsorber capable of removing not less than 99% of radioactive forms of iodine is provided. The charcoal adsorber is an all welded gasketless type filled with impregnated charcoal. The charcoal adsorber beds hold 1260 pounds of charcoal at approximately 30 lb/ft^3 density, having a minimum ignition temperature of 626° F.

   The bed is so designed that the air has at least 0.5 seconds of residence time through the charcoal (i.e., a velocity of 40 fpm). The charcoal adsorber shall meet the requirements of ANSI N-509-1976.

   Qualification of charcoal shall meet the requirements of ANSI N509-1980, Table 5-1.

   Ten test canisters are provided for the charcoal adsorber. These canisters contain the same depth (i.e., 4 inches) of the same charcoal as in the charcoal adsorber. The canisters are mounted so 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 or a representative sample is removed and laboratory tested to reverify the adsorbent efficiency.

   Two deluge valves connected to the station fire water and shutdown service water systems are mounted adjacent to each charcoal adsorber. In the case of high-temperature detection in the bed, an alarm is annunciated in the main control room. The station operator may elect to flood the adsorber by opening a deluge valve with a control switch on the main control board. High temperature alarms are set at 200° F and 250° F.

6. A high efficiency particulate air 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 outside air intake and discharges it first to the makeup air filter package, then to the inlet side of the control room supply air filter package, and finally to the control room air handling

equipment train. The fan performance is based on the maximum density and worst pressure condition when it is inducing -2° F air from the outdoors and the makeup air filter train contains filters which operate at no less than twice their clean pressure drop. The flow and pressures are listed in Table 9.4-1.

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 of components without violating the train integrity. Spacing between filter sections is based on ease of maintenance considerations.

9. The filter housing is an all welded construction, heavily reinforced, and built to low leakage requirements.

10. Interior lights with external light switches are provided between all train components to facilitate inspection, testing, and replacement of components.

11. The equipment parameters of various control room HVAC system components are given in Table 9.4-1.

12. HEPA filter frames are in accordance with recommendations of Section 4.3.5 of ERDA 76-21.

e. The system is in compliance with the intent of Regulatory Guide 1.52 as described in Table 6.5-3.

f. The control room packaged filter units are drained in accordance with ERDA 76-21, Paragraph 4.5.8.

6.5.1.2.3 Control Room HVAC Recirculation Air Filter Packages

a. The control room HVAC recirculation air filter packages work in conjunction with the control room HVAC system as described in Subsections 9.4.1 and 6.4.2.2. The schematic design is shown in Drawing M05-1102. The nominal size of principal system components is listed in Table 9.4-1.

b. In the event of high radiation detection in the minimum outside air intakes of the control room HVAC system, the charcoal filter of one of the recirculation filter trains is automatically placed in service.

c. Two control room HVAC recirculation air filter trains and fans are provided. Refer to Table 9.4-1 for system data.

d. Each control room HVAC recirculation air filter unit is comprised of the following components in sequence:

1. A prefilter, UL listed, all glass media, exhibiting no less than 85% efficiency based on ASHRAE atmospheric dust spot test. This prefilter meets the requirements of ANSI N-509-1976, Article 5.3.

2. An upstream isolation damper.

3. A charcoal adsorber capable of removing not less than 70% of all forms of iodine is provided. The charcoal adsorber is an all welded gasketless type filled with impregnated charcoal, having a minimum ignition temperature of 626° F.

   The bed is so designed that the air has at least 0.125 seconds of residence time through the charcoal (i.e., velocity of 80 fpm). Qualification of charcoal shall meet the requirements of ANSI N509-1980.

   Test canisters are provided for the charcoal adsorber. These canisters contain the same depth (i.e., 2 inches) of the same charcoal as in the charcoal adsorber. The canisters are mounted so 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 or a representative sample is removed and laboratory tested to reverify the adsorbent efficiency.

   Two deluge valves connected to the station fire water and shutdown service water systems are mounted adjacent to each charcoal adsorber. In the case of high-temperature detection in the bed, an alarm is annunciated in the main control room. The station operator may elect to flood the adsorber by opening a deluge valve with a control switch on the main control board. High temperature alarms are set at 200° F and 250° F.

4. A downstream isolation damper.

5. The equipment parameters of various control room HVAC system components are given in Table 9.4-1.

6.5.1.3 <u>Design Evaluation</u>

6.5.1.3.1 <u>Standby Gas Treatment System</u>

The standby gas treatment system (SGTS) is designed to minimize exfiltration of contaminated air from the secondary containment 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, IEEE-308, IEEE-323, and IEEE-344. 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.

The thyroid dose reduction factor at the site boundary and low population zone was calculated based on the following:

a. 100% of secondary containment inleakage treated and released through the SGTS; and

b. SGTS removal efficiency of 99% for all iodines.

6.5.1.3.2 Control Room HVAC Makeup Air and Recirculation Air Filter Packages

The control room HVAC makeup air and recirculation air filter packages work in conjunction with the control room HVAC system to maintain habitability in the control room. 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,

   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 at 1.0 psig internal pressure, and magnetic particle or liquid penetrant testing per AWS D1.1 of all welds which could cause bypass leakage around HEPA filters or adsorber bed;

   2. demister - qualification test or objective evidence to demonstrate compliance with specified design criteria;

   3. HEPA filters - elements procured prior to January 1, 1986 are tested individually by the 1 appropriate U.S. Department of Energy station in accordance with applicable inspection and testing bulletin; elements procured after January 1, 1986 are tested individually in accordance with applicable inspection and testing bulletin.

   4. HEPA filter frames - leak test at 1 psi pressure differential across filterless, covered bank;

   5. adsorbent beds - model test of bed or objective evidence to demonstrate flow pressure characteristics, channeling effects:

   6. adsorbent - qualification per the requirements of ANSI N509-1980, Table 5-1.

   7. fans - tested in accordance with ANSI N509-1976, Section 5.7, to establish characteristic curves;

8. heater - uniform temperature test, high-temperature cutout test, adjacent equipment temperature test, and high humidity qualification test;

9. prefilter - objective evidence or certification that ASHRAE efficiency specified is attained; and

10. valves - shop tests demonstrating leaktightness and closure times.

c. The onsite preoperational tests include the following:

1. visual inspection - performed prior to each subsequent test in accordance with ANSI N510-1980, Section 5;

2. housing - leak test after erection and field welding in accordance with ANSI N510-1980, Section 6 with the following exceptions:

    a. Field welds were painted prior to the duct and housing leak test.

    b. A calibrated orifice plate was used in lieu of totalizing gas volume meter.

3. heater - performance test in accordance with ANSI N510-1980, Section 14;

4. instruments - calibration and operability;

5. HEPA filter - leak test in accordance with ANSI N510-1980, Section 10 and mounting frame pressure leak test in accordance with ANSI N510-1980, Section 7;

6. adsorber - air flow capacity and distribution tests in accordance with ANSI N510-1980, Section 8, inplace leak test in accordance with ANSI N510-1980, Section 12, and mounting frame pressure leak test in accordance with ANSI N510-1980, Section 7;

7. adsorbent - laboratory testing in accordance with CPS Technical Specification requirements.

8. system - flow test, pressure test, mode test, and failure test; and

9. valves - closure time tests and leakage tests.

d. Onsite periodic testing - Operating personnel are trained and required to make surveillance checks. The checks performed and their required frequency are outlined in the CPS Technical Specifications.

6.5.1.4.2 <u>Control Room HVAC Makeup Air Filter Packages</u>

a. The control room HVAC makeup air filter packaqes and components are thoroughly tested in a program consisting of the following:

1. factory and component qualification tests,

2. onsite preoperational testing,

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 consist of the following:

1. Filter Train Housing

   a) leak test at design internal pressure, and

   b) magnetic particle or liquid penetrant testing per AWS DI.I of all welds which could cause bypass leakage around HEPA filters or adsorber 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 will be attained.

4. HEPA Filters

   Elements procured prior to January 1, 1986 are tested individually by the appropriate U.S. Department of Energy station in accordance with applicable inspection and testing bulletin. Elements procured after January 1, 1986 are tested individually in accordance with the applicable inspection and testing bulletin.

5. HEPA Filter Frames

   Leak test at 10-inch water gauge pressure differential across filterless, covered bank.

6. Adsorbent Beds

   Model test of bed or objective evidence to demonstrate flow pressure characteristics and channeling effects.

7. Adsorbent

   Qualification per the requirements of ANSI N509-1980, Table 5-1.

8. Fans will be tested in accordance with ANSI N509-1976, Section 5.7.2 to establish characteristic curves.

9. Heater

   a) uniform temperature test,

   b) high-temperature cutout test,

   c) adjacent equipment temperature test, and

   d) high humidity qualification test.

c. The onsite preoperational testing is the same as described in Subsection 6.5.1.4.1(c) for the SGTS.

d. Onsite subsequent periodic testing is the same as described in Subsection 6.5.1.4.1(d) for the SGTS.

6.5.1.4.3 <u>Control Room HVAC Recirculation Air Filter Packages</u>

a. The control room HVAC recirculation air filter packages and components are thoroughly tested in a program consisting of the following:

   1. factory and component qualification tests,

   2. onsite preoperational testing,

   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 consist of the following:

   1. Filter Train Housing

      Magnetic particle or liquid penetrant testing in accordance with AWS DI.I of all adsorber bed welds which could result in leakage bypassing adsorber beds.

   2. Prefilter

      Objective evidence or certification that ASHRAE efficiency specified will be attained.

   3. Adsorbent Beds

      Model test of bed or objective evidence to demonstrate flow pressure characteristics and channeling effects.

4. Adsorbent - Methyl iodine removal efficiency tests as indicated under Subsection 6.5.1.4.2.b.7.

c. The onsite preoperational tests include the following:

1. visual inspection - performed prior to each subsequent test in accordance with ANSI N510-1980, Section 5;

2. housing - leak test after erection and field welding in accordance with ANSI N510-1980, Section 6 with the following exceptions:

    a. Field welds were painted prior to the duct and housing leak test,

    b. A calibrated orifice plate was in lieu of totalizing gas volume meter;

3. instruments - calibration and operability;

4. adsorber - air flow capacity and distribution tests in accordance with ANSI N510-1980, Section 8, leak test in accordance with ANSI N510-1980 Section 12, and mounting frame pressure leak test in accordance with ANSI N510-1980, Section 7;

5. system - flow test, pressure test, mode test, and failure test: and

d. Onsite periodic testing - Operating personnel are trained and required to make surveillance checks. The checks performed and their required frequency are outlined in the CPS Technical Specifications.

6.5.1.5 Instrumentation Requirements

The instrument controls and devices for the control room HVAC and standby gas treatment control systems meet the requirements of the following Regulatory Guides: 1.29, 1.30, 1.47, 1.52, 1.53, 1.62, 1.75, 1.89, 1.100, and 1.105.

6.5.1.5.1 Control Room HVAC Makeup Air Filter Packages

The monitoring of primary makeup air filter package parameters is accomplished as follows:

a. Inlet and outlet temperature signals are transmitted to the main control board for indication.

b. Differential pressure across the demister and the prefilter is individually indicated locally on the filter train. Differential pressure across the demister, heater, and the prefilter combination is indicated, recorded and annunciated on the main control panel. Differential pressure across the upstream HEPA filter is indicated locally on the train and indicated, recorded and annunciated at the main control panel.

   Differential pressure across the downstream HEPA filter is indicated locally on the train and indicated and annunciated at the main control panel.

c. High-temperature and high-high temperature signals from each charcoal adsorber are transmitted to the main control board for high temperature annunciation. The station operator may elect to flood the adsorber by opening a deluge valve with a control switch on the main control board.

d. Makeup air flow rate is transmitted to the main control board for indication, recording, and annunciation.

e. Design details and logic of the instrumentation are described in Subsection 7.3.1.1.6. The system instrumentation is in compliance with the intent of Regulatory Guide 1.52, Rev. 2.

6.5.1.5.2 Control Room HVAC Recirculation Air Filter Packages

The monitoring of primary recirculation air filter packages parameters is accomplished as follows:

a. Differential pressure across the medium efficiency filter is indicated locally, and a signal is transmitted to the main control board for high differential pressure annunciation.

b. High-temperature and high-high temperature signals from each charcoal adsorber are transmitted to the main control board for high temperature annunciation. The station operator may elect to flood the adsorber by opening a deluge valve with a control switch on the main control board.

c. Design details and logic of the instrumentation are described in Subsection 7.3.1.1.6. The system instrumentation is in compliance with the intent of Regulatory Guide 1.52.

6.5.1.5.3 Standby Gas Treatment System Filter Trains

The monitoring of primary standby gas treatment system (SGTS) filter train parameters is accomplished as follows:

a. Inlet and outlet temperature signals are transmitted to the main control board for indication.

b. Differential pressure signal across the upstream HEPA filters is transmitted to the main control board for indication, recording and high differential pressure annunciation. It is also indicated locally on the train. In the downstream HEPA filter, the signal is transmitted into the main control board for indication and high pressure annunciation. It is also indicated locally on the train. High differential pressure across the prefilter and demister combination is annunciated on the main control board.

c. High-temperature and high-high temperature signals from each charcoal adsorber are transmitted to the main control board for high temperature annunciation. The station operator may elect to flood the adsorber by opening a deluge valve with a control switch on the main control board.

d. Flow signals from each standby gas treatment system filter train are transmitted to the main control board for indication, recording, and annunciation and are used as an input to a flow controller to modulate the control damper.

e. Design details and logic of the instrumentation are described in Subsection 7.3.1.1.9. The system instrumentation is in compliance with the intent of Regulatory Guide 1.52.

6.5.1.6 Materials

a. All component material is capable of a service life of 40 years normal operation at the maximum cumulative radiation exposure or post-LOCA operation without any adverse effects on service, performance, or operation. All materials of construction are compatible with the required radiation exposure. This includes, but is not limited to, all metal components, seals, gaskets, lubricants, and finishes, such as paints, etc. The integrated dose during the once-in-a-lifetime post-LOCA use is 2×10^8 rads for SGTS (2×10^7 rads for SGTS demister pads) and 1×10^5rads for control room HVAC makeup and supply (recirculation) air filter packages.

b. Care is taken to avoid the use during fabrication or production of any compounds or other chemicals that contain chlorides or other constituents capable of inducing stress corrosion in stainless steels 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 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 Military Specification MIL-F-51068D and all the latest amendments and modifications are used as of January 1, 1978.

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;

   5. the binder in the glass pads used in the demister section (this is a phenolic compound);

   6. phenolic compounds and elastomers associated with electrical components.

6.5.2 Containment Spray Systems

No credit is taken for fission product removal by the containment spray system. Further discussion of this system can be found in Subsection 6.2.2.

6.5.3 Fission Product Control System

The standby gas treatment system (SGTS) and the containment systems are used to control and cleanup the fission products released from the containment following an accident and are described in detail in Subsections 6.5.1, 3.8, and 6.2, respectively.

TABLE 6.5-3
COMPLIANCE WITH REGULATORY GUIDE 1.52, REV. 2 (MARCH 1978) (Continued)

TABLE 6.5-1
STANDBY GAS TREATMENT SYSTEM

NAME OF EQUIPMENT	TYPE, QUANTITY AND NOMINAL CAPACITY (PER COMPONENT)
A. EQUIPMENT TRAIN	
1. Type	Package
2. Quantity	2
3. Components of Train	
a. Fan	
Type	Centrifugal
Quantity	1
Drive	Direct
Capacity (cfm at 70°F)	4000
Static Pressure (in. H_2O)	17.0
Motor (hp)	30
b. Demister	
Type	Impingement
Quantity	1 Bank
Static resistance	
clean (in. H_2O)	1.0
dirty (in. of H_2O)	1.3
c. Heater	
Type	Fin Tubular, single stage
Quantity	1
Capacity (Kw)	20
Accessories	Overload cutout
d. Prefilter	
Type	Disposable
Quantity	1 Bank
Capacity (cfm at 175° F)	4000
Efficiency (per NBS Dust Spot Test)	85%
Static resistance	
clean (in. H_2O)	0.2
dirty (in. H_2O)	1.0

TABLE 6.5-1
STANDBY GAS TREATMENT SYSTEM (Continued)

NAME OF EQUIPMENT	TYPE, QUANTITY AND NOMINAL CAPACITY (PER COMPONENT)
e. HEPA Filters	
Type	High Efficiency Particulate Absolute Dry
Quantity	4 Elements per Bank/Two Banks per Train
Media	Glass Fiber, Waterproof, Fire Resistant
Individual Filter Efficiency (% with 0.3 micron cold generated dioctylphthalate smoke)	99.97
Static resistance	
clean (in.H_2O)	1.0
dirty (in.H_2O)	2.0
f. Charcoal Adsorber Bed	
Type	Gasketless
Quantity	1 per train
Flow Capacity (cfm at 175° F)	4000
Media	Impregnated Charcoal
Iodine Removal Efficiency (%)	99.9 on Methyl Iodine 99.9 on Elemental Iodine
Quantity of Media (lb)	1800 per Train
Depth of Bed (in)	4
Residence time (sec)	0.5
Charcoal Ignition Temperature (°F)	626° minimum
Maximum Heat Load (Btu/hr)	Less than 5,500 (No Containment Purge)
Static Resistance (in. H_2O)	1.8
g. Standby Cooling Air Fan	
Type	Centrifugal
Quantity	1
Drive	Direct
Capacity (cfm at 70°F)	350
Static Pressure (in. H_2O)	4.2
Motor (hp)	1.5

TABLE 6.5-3
COMPLIANCE WITH REGULATORY GUIDE 1.52, REV. 2 (MARCH 1978) (Continued)

TABLE 6.5-2
STANDBY GAS TREATMENT SYSTEM EQUIPMENT FAILURE ANALYSIS

COMPONENT	FAILURE	FAILURE DETECTED BY	ACTION REQUIRED
1) Primary Fan	Motor Burnout, Drive Shaft Break, etc.	Flow Monitor - Low-Pressure Switch and MCC Breakers	Main Control Board Alarm Operating Equipment Train Shutdown Manually Redundant Train Startup Manually Isolation Valves Positioned
2) Electric Heating Coil	Element Overheat	Overload Protection Circuit on Coil	Main Control Board Alarm Operating Equipment Train Shutdown Manually Redundant Train Startup Manually Isolation Valves Positioned
3) Standby Cooling Fan	No Startup Results In High Charcoal Adsorber Temperature	One Temperature Instrument with separate “high” and “high-high” switches for each Electrical Design.	High and high-high temperature alarms are annunciated in the main control room. The station operator may elect to flood the adsorber by opening a deluge valve with a control switch on the main control board. If the alarm is on the operating train, the operating train is shutdown and the redundant train started. The shutdown train isolation dampers automatically close.
4) Flow Control Valve	Fails Open	Flow Monitor - High-Pressure Switch	Main Control Board Alarm Operating Equipment Train Shutdown Manually Redundant Train Startup Manually Isolation Valves Positioned
	Fails Shut	Flow Monitor - Low-Pressure Switch	Main Control Board Alarm Operating Equipment Train Shutdown Manually Redundant Train Startup Manually Isolation Valves Positioned

TABLE 6.5-2
STANDBY GAS TREATMENT SYSTEM EQUIPMENT FAILURE ANALYSIS (Continued)

COMPONENT	FAILURE	FAILURE DETECTED BY	ACTION REQUIRED
5) Isolation Valve	Fails Open	-	None - Redundant Valves or Backflow Dampers Provided As Required
	Fails Shut	Flow Monitor - Low-Flow Switch	Main Control Board Alarm Operating Equipment Train Shutdown Manually Redundant Train Startup Manually Isolation Valves Positioned
6) HEPA Filter	High Particulate Loading	High Differential Pressure Switch	Main Control Board Alarm Operating Equipment Train Shutdown Manually Redundant Train Startup Manually Isolation Valves Positioned
7) Deluge Valve	Fails Closed	-	None Required - Two Valves Provided to Flood Bed
8) Controls and Interlocks	Division 1, Unit 1 Power Failure	ESF Status indication given at M.C.B.	Start Division II Train
9) Controls and Interlocks	Division 2, Unit 1 Power Failure	ESF Status indication given at M.C.B.	Start division I Train

TABLE 6.5-3
COMPLIANCE WITH REGULATORY GUIDE 1.52, REV. 2 (MARCH 1978)

REGULATORY POSITION	DISCUSSION
C.1. Environmental Design Criteria	C.1.
a. The design of an engineered-safety-feature atmosphere cleanup system should be based on the maximum pressure differential, radiation dose rate, relative humidity, maximum and minimum temperature, and other conditions resulting from the postulated DBA and on the duration of such conditions.	a. The CPS ESF filter systems are designed for the maximum environments, resulting from the postulated DBA, to which the systems will be exposed. Environmental conditions are specified in the applicable equipment specifications. (Refer to Section 3.11 for environmental conditions.)
b. The design of each ESF system should be based on the radiation dose to essential services in the vicinity of the adsorber section integrated over the 30-day period following the postulated DBA. The radiation source term should be consistent with the assumptions found in Regulatory Guides 1.3 (Reference 5), 1.4 (Reference 6), and 1.25 (Reference 7). Other engineered safety features, including pertinent components of essential services such as power, air, and control cables, should be adequately shielded from the ESF atmosphere cleanup systems.	b. Refer to Section 3.11 for applicable radiation doses. Source assumptions are consistent with Regulatory Guides 1.3, 1.4, and 1.25. Other ESF equipment and services are adequately shielded from the ESF filter systems.
c. The design of each adsorber should be based on the concentration and relative abundance of the iodine species (elemental, particulate, and organic), which should be consistent with the assumptions found in Regulatory Guides 1.3 (Reference 5), 1.4 (Reference 6), and 1.25 (Reference 7).	c. Refer to Section 3.11 for applicable radiation doses. Assumptions are consistent with Regulatory Guides 1.3, 1.4, and 1.25.
d. The operation of any ESF atmosphere cleanup system should not deleteriously affect the operation of other engineered safety features such as a containment spray system, nor should the operation of other engineered safety features such as containment spray system deleteriously affect the operation of any ESF atmosphere cleanup system.	d. The operation of the ESF filter systems is compatible with the operation of other ESF systems.

TABLE 6.5-3
COMPLIANCE WITH REGULATORY GUIDE 1.52, REV. 2 (MARCH 1978) (Continued)

REGULATORY POSITION	DISCUSSION
e. Components of systems connected to compartments that are unheated during a postulated accident should be designed for postaccident effects of both the lowest and highest predicted temperatures.	e. Components upstream of, and the ESF Air Filter Trains themselves, have been designed for temperatures in excess of the highest outdoor temperature (96° F) and the lowest predicted indoor temperature (40° F).
C.2 System Design Criteria	
a. ESF cleanup systems designed and installed for the purpose of mitigating accident doses should be redundant. The systems should consist of the following sequential components: (1) demisters, (2) prefilters (demisters may serve this function), (3) HEPA filters before the adsorbers, (4) iodine adsorbers (impregnated activated carbon or equivalent adsorbent such as, metal zeolites), (5) HEPA filters. after the adsorbers, (6) ducts and valves, (7) fans, and (8) related instrumentation. Heaters or cooling coils used in conjunction with heaters should be used when the humidity is to be controlled before filtration.	a. The ESF filter systems are redundant. Filter systems consist of the following sequential components: (1) demisters, (2) electric heaters, (3) prefilters, (4) HEPA filters, (5) impregnated activated carbon iodine adsorbers, (6) HEPA filters after the adsorbers, and (7) fans. During normal operation, control room supply air is continuously cleaned through the control room recirculation air train medium efficiency filter. No particulate infiltration exists as the control room is maintained at a positive pressure. HEPA filters are not required. During abnormal operation any additional air is filtered through the make-up air train. As humidity and entrained moisture conditions will not exceed 70%, an air heater and moisture separator for the control room recirculation air train is not required. Local RH indication has not been supplied because heaters are adequately sized to maintain the RH below the 70% limit, and the fan and heater are interlocked.
b. The redundant ESF atmosphere cleanup systems should be physically separated so that damage to one system does not also cause damage to the second system. The generation of missiles from high-pressure equipment rupture, rotating machinery failure, or natural phenomena should be considered in the design for separation and protection.	b. Redundant filter trains are physically separated from each other and are protected from internally generated missiles.

TABLE 6.5-3
COMPLIANCE WITH REGULATORY GUIDE 1.52, REV. 2 (MARCH 1978) (Continued)

REGULATORY POSITION	DISCUSSION
c. All components of an engineered-safety-feature atmosphere cleanup system should be designated as Seismic Category I (see Regulatory Guide 1.29) (Reference 8) if failure of a component would lead to the release of significant quantities of fission products to the working or outdoor environments.	c. All components of ESF filter systems are Seismic Category I (refer to Section 3.2).
d. If the ESF atmosphere cleanup system is subject to pressure surges resulting from the postulated accident, the system should be protected from such surges. Each component should be protected with such devices as pressure relief valves so that the overall system will perform its intended function during and after the passage of the pressure surge.	d. The standby gas treatment system is protected from such pressure surges. The control room system is not subject to such pressure surges.
e. In the mechanical design of the ESF system, the high radiation levels that may be associated with buildup of radioactive materials on the ESF system components should be given particular consideration. ESF system construction materials should effectively perform their intended function under the postulated radiation levels. The effects of radiation should be considered not only for the demisters, heaters, HEPA filters, adsorbers, and fans, but also for any electrical insulation, controls, joining compounds, dampers, gaskets, and other organic-containing materials that are necessary for operation during a postulated DBA.	e. The effects of radiation have been taken into account in the design of the ESF filter systems. All materials are suitable for the radiation levels encountered in this type of service.

TABLE 6.5-3
COMPLIANCE WITH REGULATORY GUIDE 1.52, REV. 2 (MARCH 1978) (Continued)

REGULATORY POSITION	DISCUSSION
f. The volumetric air flow rate of a single cleanup train should be limited to approximately 30,000 ft^3/min. If a total system air flow in excess of this rate is required, multiple trains should be used. For ease of maintenance, a filter layout three HEPA filters high and ten wide is preferred.	f. No ESF filter train capacity exceeds 30,000 cfm, except the control room recirculation air filter packages. The HEPA filter layout for the control room make-up air and SGTS trains is two wide by two high. The Control Room HVAC Recirculation Air Filter System operated in conjunction with the primary ventilation system during radiological emergencies to remove radioactive iodine from the Control Room atmosphere. This system is not in strict conformance with Regulatory Guide 1.52; however, an iodine decontamination efficiency of 70% (independent of chemical form) has been determined to be appropriate. Technical Specifications will include the following R.G. 1.52, Rev. 2, Section 5 and 6 requirements in support of the credit taken for the charcoal filter of the Control Room HVAC recirculation air filters: (1) the filter trains will be leak tested; and (2 the iodine removal efficiency of the activated charcoal will be determined by laboratory or field tests. The access area adjacent to the eight wide by six high filter bank of the control room HVAC circulation air filter is adequate to support their servicing and maintenance.
g The ESF atmosphere cleanup system should be instrumented to signal, alarm, and record pertinent pressure drops and flow rates at the control room.	g. Differential pressure is indicated locally for the demister and prefilter bank for the control room make-up air and SGTS filter trains. High differential pressure is alarmed in the control room for the demister, heater, and prefilter combination, and the downstream HEPA bank of each ESF filter train. In addition, differential pressure is recorded and alarmed in the control.

TABLE 6.5-3
COMPLIANCE WITH REGULATORY GUIDE 1.52, REV. 2 (MARCH 1978) (Continued)

REGULATORY POSITION	DISCUSSION
g. (Continued)	room for the upstream HEPA bank for the control room make-up air and SGTS filter trains. Appropriate low flow rates are annunciated in the control room. As high air flow though units can be monitored using the flow indicator or flow recorder provided in the main control room, a high flow alarm is not necessary As identified in the SER and Supplement 1, filter unit 0VC07SA/B is not in strict conformance with Reg Guide 1.52. High air flow through these units would result in changes to other system variables, such as control room differential pressure and temperature, which are indicated in the main control room. Therefore, high flow alarms are not necessary. Differential pressure is indicated locally for the control room recirculation air filter train and alarmed on high differential pressure in the control room.
h. The power supply and electrical distribution system for the ESF atmosphere cleanup system described in Section C.2.a above should be designed in accordance with Regulatory Guide 1.32 (Reference 9). All instrumentation and equipment controls should be designed to IEEE Standard 279 (Reference 10). The ESF system should be qualified and tested under Regulatory Guide 1.89 (Reference 11). To the extent applicable Regulatory Guides 1.30 (Reference 12), 1.100 (Reference 13) and 1.118 (Reference 14) and IEEE Standard 334 (Reference 15) should be considered in the design.	h. Electric power systems are per Regulatory Guide 1.32 and controls are per IEEE-279. Electrical components important to safety are qualified as Class 1E. The ESF filter trains and electrical items meet QA requirements of 10 CFR 50, Appendix B. Motors are in accordance with IEEE-334.

TABLE 6.5-3
COMPLIANCE WITH REGULATORY GUIDE 1.52, REV. 2 (MARCH 1978) (Continued)

REGULATORY POSITION	DISCUSSION
i. Unless the applicable engineered-safety-feature atmosphere cleanup system operates continuouslyduring all times that a DBA can be postulated to occur, the system should be automatically activated upon the occurrence of a DBA by (1) a redundant ESF signal (i.e., temperature, pressure), or (2) a signal from redundant Seismic Category I radiation monitors.	i. The standby gas system filter units are automatically activated by a signal from redundant radiation monitors for each division, while the control room filter units are automatically activated by a common signal from two divisional monitors. In both cases, single failure criteria is met.
j. To maintain radiation exposures to operating personnel as low as is reasonably achievable during plant maintenance ESF atmosphere cleanup systems should be designed to control leakage and facilitate maintenance in accordance with the guidelines of Regulatory Guide 8.8 (Reference 16). The ESF atmosphere cleanup train should be totally enclosed. Each train should be designed and installed in a manner that permits replacement of the train as an intact unit or as a minimum number of segmented sections without removal of individual components.	j. The filter trains are not designed to be removable from the plant as an intact unit, since the size of the entire train precludes shipment offsite and there are normally no offsite facilities for disposal of an intact unit. Filter elements will be removable for disposal through the radwaste system.
k. Outdoor air intake openings should be equipped with louvers, grills screens, or similar protective minimize the effects of high winds, rain, snow, ice, trash, or other contaminants on operation of the system. If atmosphere surrounding the plant could contain significant environmental contaminants, such as dusts and residues from smoke cleanup systems from adjacent coal burning power plants or industry, design of the system should consider these contaminants and prevent them from affecting the operation of any ESF atmosphere system.	k. Outdoor air openings are provided with suitable louvers designed to devices to minimize the entry of rain, snow, ice, or trash. The atmosphere surrounding the CPS plant does not the contain significant amounts of other contaminants referred to. Outdoor air is mixed with return air and filtered through the recirculation air filter unit prefilters. Under accident conditions, outside air is brought into the main control room cleanup through the makeup air filter unit.

TABLE 6.5-3
COMPLIANCE WITH REGULATORY GUIDE 1.52, REV. 2 (MARCH 1978) (Continued)

REGULATORY POSITION	DISCUSSION
l. ESF atmosphere cleanup system housings and ductwork should be designed to exhibit on test a maximum total leakage rate as defined in Section 4.12 of ANSI N509-1976 (Reference 1). Duct and housing leak tests should be performed in accordance with the provisions of Section 6 of ANSI N510-1975 (Reference 2).	l. The standby gas treatment system and the control room make-up air system filter housings have been shop leak tested. System ductwork will be constructed to exhibit a leakage no greater than the value as defined in in Section 4.12 of ANSI N509-1980, except that positive pressure duct leakage downstream of the control room make-up air filter is based on air cleaning effectiveness requirements, and will be leak tested as required to demonstrate compliance. Inleakage to the negative pressure duct upstream of the control room make-up air filter train is filtered and therefore leakage testing is not required. The standby gas treatment system negative pressure duct/pipe upstream of the filter train passes through either clean interspace or contaminated space. Inleakage is filtered prior to discharge and therefore leakage testing is not necessary.
C.3. Component Design Criteria and Qualification Testing	
a. Demisters should be designed, constructed, and tested in accordance with the requirements of Section 5.4 of ANSI N509-1976 (Reference 1). Demisters should meet Underwriters Laboratories (UL) Class 1 (Reference 17) requirements.	a. The demisters shall be designed, constructed, and tested in accordance with the requirements of Section 5.4 of ANSI N509-1976. Demisters meets UL requirements but UL no longer has classifications.
b. Air heaters should be designed, constructed, and tested in accordance with the requirements of Section 5.5 of ANSI N509-1976 (Reference 1).	b. Air heaters will be designed, constructed, and tested in accordance with the requirements of Section 5.5 of ANSI N509-1976.
c. Materials used in prefilters should withstand the radiation levels and environmental conditions prevalent during the postulated DBA. Prefilters should be designed, constructed, and tested in accordance with the provisions of Section 5.3 of ANSI N509-1976 (Reference 1).	c. Materials used in the prefilters are of the type commonly used in this application, and are specified for the environmental conditions at CPS. Prefilters shall be designed, constructed and tested in accordance with the provisions of Section 5.3 of ANSI N509-1976. Prefilters meets UL requirements but UL no longer has classifications.

TABLE 6.5-3
COMPLIANCE WITH REGULATORY GUIDE 1.52, REV. 2 (MARCH 1978) (Continued)

REGULATORY POSITION	DISCUSSION
d. The HEPA filters should be designed, constructed, and tested in accordance with Section 5.1 of ANSI N509-1976 (Reference 1). Each HEPA filter should be tested for penetration of dioctyl phthalate (DOP) in accordance with the provisions of MIL-F-51068 (Reference 19) and MIL-STD-282 (Reference 20).	d. HEPA filters procured prior to January 1, 1986 will be designed, constructed, and tested in accordance with Section 5.1 of ANSI N509-1976. HEPA filters procured after January 1, 1986 are designed, constructed and tested in accordance with ANSI-509-1980, Section 5.1. Also, the HEPA filters will be tested for penetration of DOP in accordance with the provisions of MILF-51068 and MIL-STD-282.
e. Filter and adsorber mounting frames should be constructed and designed in accordance with the provisions of Section 5.6.3 of ANSI N509-1976 (Reference 1)	e. Filter and adsorber mounting frames will be constructed and designed in accordance with Section of 5.6.3 of ANSI N509-1976. With respect to the requirements that penetrations through mounting frame should not exist, it should be noted that viewports are installed on the downstream side of filter unit adsorber frames. The charcoal adsorber frames are field leak-tested in accordance with ANSI N510-1980. These viewports are provided to observe adequate fill of the charcoal adsorber.
f. Filter and adsorber banks should be arranged in accordance with the recommendations of Section 4.4 of ERDA 76-21 (Reference 3).	f. Filter and adsorber banks are arranged in accordance with the recommendations of Section 4.4 of ERDA 76-21.

TABLE 6.5-3
COMPLIANCE WITH REGULATORY GUIDE 1.52, REV. 2 (MARCH 1978) (Continued)

REGULATORY POSITION	DISCUSSION
g. System filter housings, including floors and doors, should be constructed and designed in accordance with the provisions of Section 5.6 of ANSI N509-1976 (Reference 1).	g. System filter housings, including floors and doors, are constructed and designed in accordance with the provisions of Section 5.6 of ANSI N509-1976. Even though prefilter bank in unit 0VC07SA/B is higher than three HEPA filters, and no permanent service gallery is provided, the access areas are adequate to support service and maintenance of the prefilter and lighting. The SGTS and control room emergency outdoor air duct (pipe) are seamless ASTM A106 Grade B Schedule 40 or 3/8-inch wall pipe (depending on size). Corrosion will have a negligible effect on this duct (pipe) over plant life. Contribution of particulate matter will have no effect on operation, as systems are extensively operated and are cleaned during initial operation. The top and bottom of CPS filter unit doors are equipped with one latching lug each. Leak testing during initial system acceptance and after any major modification/repair verifies that the leak integrity of the filter housing is maintained.
h. Water drains should be designed in accordance with the recommendations of Section 4.5.8 of ERDA 76.21 (Reference 3).	h. Water drains are designed in accordance with the recommendations of Section 4.5.8 of ERDA 76.21.
i. The adsorber section of the ESF atmosphere cleanup system may contain any adsorbent material demonstrated to remove gaseous iodine (elemental iodine and organic iodines) from air at the required efficiency. Since impregnated activated carbon is commonly used, only this adsorbent is discussed in this guide. Each original or replacement batch of impregnated activated carbon used in the adsorber section should meet the	Each original and replacement batch of activated carbon used in the adsorber section will meet the qualification and batch test results summarized in Table 5.1 of ANSI N509-1980. The adsorbent beds are designed for an average atmospheric residence time of 0.25 seconds per 2 inch bed depth except for the Control Room Recirculation Air filter adsorber which is designed for a 0.125 second residence time per 2 inch bed depth.

TABLE 6.5-3
COMPLIANCE WITH REGULATORY GUIDE 1.52, REV. 2 (MARCH 1978) (Continued)

REGULATORY POSITION	DISCUSSION
i. (Continued) qualification and batch test results summarized in Table 5.1 of ANSI N509-1976 (Reference 1) In this table, a "qualification test" should be interpreted to mean a test that establishes the suitability of a product for a general application, normally a one-time test reflecting historical typical performance of material. In this table, a "batch test" should be interpreted to mean a test made on a production batch of product to establish suitability for a specific application. A "batch of activated carbon" should be interpreted to mean a quantity of material of the same grade, type, and series that has been homogenized to exhibit, within reasonable tolerance, the same performance and physical characteristics and for which the manufacturer can demonstrate by acceptable tests and quality control practices such uniformity. All material in the same batch should be activated, impregnated, and otherwise treated under the same process conditions and procedures in the same process equipment and should be produced under the same manufacturing release and instructions. Material produced in the charge of batch equipment constitutes a batch; material produced in different charges of the same batch equipment should be included in the same batch only if it can be homogenized as above. The maximum batch size should be 350 ft^3 of activated carbon. In an adsorbent other than impregnated activated carbon is proposed or if the mesh size distribution is different from the specifications in Table 5.1 of ANSI N509-1976 (Reference 1), the proposed adsorbent should have	The safety-related charcoal adsorbers are designed to a maximum loading of 2.5 milligram of total iodine per gram of charcoal based on the on the fact that post-LOCA containment purge will be accomplished through the use of drywell purge units, which are non-safety-related, non-seismic, and not qualified to Regulatory Guide 1.52.

TABLE 6.5-3
COMPLIANCE WITH REGULATORY GUIDE 1.52, REV. 2 (MARCH 1978) (Continued)

REGULATORY POSITION	DISCUSSION
i. (Continued) demonstrated the capability to perform as well as or better than activated carbon in satisfying the specifications in Table 5.1 of ANSI N509-1976 (Reference 1). If impregnated activated carbon is used as the adsorbent, the adsorber system should be designed for an average atmosphere residence time of 0.25 sec per two inches of adsorbent bed. The adsorption unit should be designed for a maximum loading of 2.5 mg of total iodine (radioactive plus stable) per gram of activated carbon. No more than 5% of impregnant (50 mg of impregnant per gram of carbon) should be used. The radiation stability of the type of carbon specified should be demonstrated and certified (see Section C.I.b of this guide for the design source term).	
j. Adsorber cells should be designed, constructed, and tested in accordance with the requirements of Section 5.2 of ANSI N509-1976 (Reference 1).	j. Adsorber cells are designed, constructed, and shop tested in accordance with Section 5.2 of ANSI N509-1976.
k. The design of the adsorber section should consider possible iodine desorption and adsorbent auto-ignition that may result from radioactivity-induced heat in the adsorbent and concomitant temperature rise. Acceptable designs include a low-flow air bleed system, cooling coils,water sprays for the adsorber section, or other cooling mechanisms. Any cooling mechanism should satisfy the single failure criterion. A low-flow air bleed system should satisfy the single failure criterion for providing low-humidity (less than 70% relative humidity) cooling air flow.	k. A low-flow fan has been provided for the SGTS to remove radioactivity-induced heat in the adsorbent.

TABLE 6.5-3
COMPLIANCE WITH REGULATORY GUIDE 1.52, REV. 2 (MARCH 1978) (Continued)

REGULATORY POSITION	DISCUSSION
l. The system fan, its mounting, and the ductwork connections should be designed, constructed, and tested in accordance with the requirements of Sections 5.7 and 5.8 of ANSI N509-1976 (Reference 1).	l. The system fan, its mounting, and the ductwork connections are designed, constructed, and tested in accordance with the requirements of Sections 5.7 and 5.8 of ANSI N509-1976. The fan motors are not specified to include provisions to indicate bearing and winding temperature or vibration limits. The motors have been qualified for the environmental conditions in which they are required to operate. The fans have been dynamically balanced in the factory and vibration is field verified to be within acceptance limits. As fans are located outside of containment and are accessible, provisions for remote monitoring of bearing temperatures and vibration limit switches are not required. The fan motors are in accordance with the applicable requirements of NEMA MG-1, IEEE 112A and IEEE 344, instead of ANSI C50-20 and ANSI N41-7. Fan motor rating, testing and monitoring instrumentation meet the intent of ANSI N509-1980.
m. The fan or blower used on the ESF atmosphere cleanup system should be capable of operating under the environmental conditions postulated, including radiation.	m. The fans and motors have been qualified to the environmental conditions of the postulated environment.
n. Ductwork should be designed, constructed, and tested in accordance with the provisions of Section 5.10 of ANSI N509-1976 (Reference 1).	n. The ductwork is designed, constructed, and tested in accordance with the intent of ANSI N509-1980. A fan pressure test for the field installed control room recirculation air cabinet is not necessary. All ESF air cleaning duct and housings are designed to withstand maximum loading conditions. Any ESF damper failure would result in flow alarming, area pressure changes, and/or status indication changes requiring operator action.

TABLE 6.5-3
COMPLIANCE WITH REGULATORY GUIDE 1.52, REV. 2 (MARCH 1978) (Continued)

REGULATORY POSITION	DISCUSSION
o. Ducts and housings should be laid out with a minimum of ledges, protrusions, and crevices that could collect dust and moisture and that could impede personnel or create a hazard to them in the performance of their work. Straightening vanes should be installed where required to ensure representative air flow measurement and uniform flow distribution through cleanup components.	o. Ducts are laid out with a minimum of ledges, protrusions, and crevices, except for external stiffeners, which are not considered a hazard. Straightening vanes are installed where required. Housings are laid out with stiffeners and internal framing, which are required for structural loading and are not considered a hazard.
p. Dampers should be designed, constructed, and tested in accordance with the provisions of Section 5.9 of ANSI N509-1976 (Reference 1).	p. Dampers are designed, constructed, and tested in accordance with the intent of ANSI N509-1976 Dampers 0VC10YA&B are opposed blade dampers instead of bubble-tight type. Design air leakage resulting from these dampers has been accounted for in control room dose analysis and is within acceptable limits. Actual leakage is field verified to not exceed design limit. To maintin housing leak integrity, operators for the control room recirculation air filter adsorber bed bypass and isolation dampers are exposed to the airstream. All environmental specifications are satisfied. In lieu of shop cycle test to ensure dampers are functioning when delivered; control room HVAC system isolation dampers were functional tested in the field during preoperational testing.
C.4	C.4
a. Accessibility of components and maintenance should be considered in the design of ESF atmosphere cleanup systems in accordance with the provisions of Section 2.3.8 of ERDA 76-21 (Reference 3) and Section 4.7 of ANSI N509-1976 (Reference 1).	a. Accessibility of components and maintenance is considered in the design of ESF atmosphere cleanup in accordance with Section 2.3.8 of ERDA 76-21 and Section 4.7 of ANSI N509-1976.

TABLE 6.5-3
COMPLIANCE WITH REGULATORY GUIDE 1.52, REV. 2 (MARCH 1978) (Continued)

b. For ease of maintenance, the system design should provide for a minimum of three feet from mounting frame to mounting frame between banks of components. If components are to be replaced, the dimensions to be provided should be the maximum length of the component plus a minimum of three feet.	b. All ESF filter units have been specified to provide a minimum dimension from mounting frame to mounting frame between banks of components equal to the maximum length of the component plus approximately 2 feet. The prefilters and HEPA filters for 0VC09SA/B and 0VG01SA/B are serviced from the downstream side away from adjacent components. Adequate clearance has been provided for the maintenance of these filters on the downstream side.
c. The system design should provide for permanent test probes with external connections in accordance with the provisions of Section 4.11 of ANSI N509-1976 (Reference 1).	c. The system design provides for permanent test probes with the external connections that are needed to conduct acceptance and subsequent periodic HEPA filter and adsorber testing per Section 4.11 of ANSI N509 1980, and ANSI N510-1980.
d. Each ESF atmosphere cleanup train should be operated at least 10 hours per month, with the heaters on (if so equipped), in order to reduce the buildup of moisture on the adsorbers and HEPA filters.	d. This requirement has been incorporated into the CPS Technical Specifications.
e. The cleanup components (i.e., HEPA filters, prefilters, and adsorbers) should not be installed while active construction is still in progress.	e. The cleanup components will not be installed while active construction is still in progress.
C.5 In-Place Testing Critria	C.5
a. A visual inspection of the ESF atmosphere cleanup system and all associated components in-place airflow distribution test, DOP test, or activated carbon adsorber section leak test in accordance with the provisions of Section 5 of ANSI N510-1975 (Reference 2).	a. A visual inspection of the ESF atmosphere cleanup system and all associated components will be made before each test per Section 5 of ANSI N510-1980. A Grey Card Illumination Test will be used to verify adequate lighting is available for visual inspection.

TABLE 6.5-3
COMPLIANCE WITH REGULATORY GUIDE 1.52, REV. 2 (MARCH 1978) (Continued)

b. The airflow distribution to the HEPA filters and iodine adsorbers should be tested in place for uniformity initially and after maintenance affecting the flow distribution. The distribution should be within ± 20% of the average flow per unit. The testing should be conducted in accordance with the provisions of Section 9 of "Industrial Ventilation" (Reference 21) and Section 8 of ANSI N510-1975 (Reference 2).	b. The airflow distribution testing will be conducted in accordance with the provisions of Section 9 of "Industrial Ventilation" and Section 8 of ANSI 510-1980. Temporary filters and/or blanks will be installed to artificially increase system resistance to 1.25 times the design dirty filter pressure drop during air flow capacity tests.
c. The in-place DOP test for HEPA filters should conform to Section 10 of ANSI N510-1975 (Reference 2). HEPA filter sections should be tested in place (1) initially, (2) at least once per 18 months thereafter, and (3) following painting, fire, or chemical release in any ventilation zone communicating with the system to confirm a penetration of less than 0.05% at rated flow. An engineered-safety-feature air filtration system satisfying this condition can be considered to warrant a 99% removal efficiency for particulates in accident dose evaluations. HEPA filters that fail to satisfy this condition should be replaced with filters qualified pursuant to regulatory position C.3.d of this guide. If the HEPA filter bank is entirely or only partially replaced, an in-place DOP test should be conducted. If any welding repairs are necessary on, within, or adjacent to the ducts, housing, or mounting frames, the filters and adsorbers should be removed from the housing during such repairs. The repairs should be completed prior to periodic testing, filter inspection, and in-place testing. The use of silicone sealants or any other temporary patching material on filters, housing, mounting frames, or ducts should not be allowed.	c. An in-place DOP test for HEPA filters will conform to Section 10 of ANSI N510-1980 and will be performed initially. The additional testing requirements will be incorporated into the CPS technical specifications. A 99% removal efficiency has been considered in the accident dose evaluation (see Chapter 15). Welding repairs will not be conducted on housings with filters and adsorbers installed. Silicone sealants or other temporary patching material shall not be used in the ESF filter housing. However, it is used as a permanent sealant for HVAC ductwork.

TABLE 6.5-3
COMPLIANCE WITH REGULATORY GUIDE 1.52, REV. 2 (MARCH 1978) (Continued)

d. The activated carbon adsorber section should be leak tested with a gaseous halogenated hydrocarbon refrigerant in accordance with Section 12 of ANSI N510-1975 (Reference 2) to ensure that bypass leakage through the adsorber section is less than 0.05%. After the test is completed, air flow through the unit should be maintained until the residual refrigerant gas in the effluent is less than 0.01 ppm. Adsorber leak testing should be conducted (1) initially, (2) at least once per 18 months thereafter, (3) following removal of an adsorber sample for laboratory testing if the integrity of the adsorber section is affected, and (4) following painting, fire, or chemical release in any ventilation zone communicating with the system.	d. An in-place refrigerant test for the carbon adsorber section will conform to Section 12 of ANSI N510-1980 and will be performed initially. The control room recirculation air adsorber has an acceptance of 2% penetration instead of 0.05%. Credit is taken for a removal efficiency of 70% as opposed to 99%. The additional testing requirements are incorporated into the CPS Technical Specifications.
C.6 Laboratory Testing Criteria for Activated Carbon	
a. The activated carbon adsorber section of the ESF atmosphere cleanup system should be assigned the decontamination efficiencies given in Table 2 for elemental iodine and organic iodides if the following conditions are met:	a. A 99% removal efficiency has been considered in the accident dose evaluation (see Chapter 15) for the control room make-up air and SGTS systems, while a 70% removal efficiency has been considered in the accident dose evaluation for the control room supply air system.
1) The adsorber section meets the conditions given in regulatory position C.5.d of the guide	1) See discussion for C.5.d.
2) New activated carbon meets the physical property specifications given in Table 5.1 of ANSI N509-1976 (Reference 1) and	2) New activated carbon meets the requirements of Table 5.1 of ANSI N509-1980.
3) Representative samples of used activated carbon pass the laboratory tests given in Table 2. If the activated carbon fails to meet any of the above conditions, it should not be used in engineered-safety-feature adsorbers.	3) Representative samples of used activated carbon will be laboratory tested in accordance with the CPS Technical Specifications. If the activated carbon fails to comply with Technical Specification requirements, it will be removed from ESF filter service.
b. The efficiency of the activated carbon adsorber section should be determined	b. Adsorbent samples have been designed to be exposed to the same service

TABLE 6.5-3
COMPLIANCE WITH REGULATORY GUIDE 1.52, REV. 2 (MARCH 1978) (Continued)

by laboratory testing of representative samples of the activated carbon exposed simultaneously to the same service conditions as the adsorber section. Each representative sample should be not less than two inches in both length and diameter, and each sample should have the same qualification and batch test characteristic as the system adsorbents There should be a sufficient number of representative samples located in parallel with the adsorber section to estimate the amount of penetration of the system adsorbent throughout its service life. The design of the samplers should be in accordance with the provisions of Appendix A of ANSI N509-1976 (Reference 1). Where the system activated carbon is greater than two inches deep, each representative sampling station should consist of enough two-inch samples in series to equal the thickness of the system adsorbent. Once representative samples are removed for laboratory test, their positions in the sampling array should be blocked off. Laboratory tests or representative samples should be conducted, as indicated in Table 2 of this guide, with the test gas flow in the same direction as the flow during service conditions. Similar laboratory tests should be performed on an adsorbent sample before loading into the adsorbers to establish an initial point for comparison of future test results. The activated carbon adsorber section should be replaced with new unused activated carbon meeting thephysical property specifications of Table 5.1 of ANSI N509-1976 (Reference 1) if (1) testing in accordance with the frequency specified in Footnote c of Table 2 results in a representative

b. (Continued)

sample failing to pass the applicable test in Table 2 or (2) no representative

conditions as the main adsorber and have been designed in accordance with the provisions of Appendix A of ANSI N509-1976. Charcoal used in the sample canisters is from the same lot as used in the main adsorber, and is representative of the thickness of the main bed.

Once removed, sample positions are sealed to prevent adsorber bypass.

A test canister or representative sample will be periodically removed for laboratory testing.

If the activated carbon fails to meet the CPS Technical Specification requirements, it will be removed from ESF filter service.

TABLE 6.5-3
COMPLIANCE WITH REGULATORY GUIDE 1.52, REV. 2 (MARCH 1978) (Continued)

sample is available for testing.

6.6 INSERVICE INSPECTION OF CLASS 2 AND 3 COMPONENTS

6.6.1 Components Subject to Examination

ASME Code Class 2 and 3 components including pressure vessels, piping, pumps, valves, bolting, and supports will be examined in accordance with Section XI of the Code, except for those components exempted per Section XI, or when specific relief is granted by the NRC in accordance with the provisions of 10 CFR 50.50a. The preservice examination coverage uses the indicated edition/addenda, ASME Code, Section XI, of Table 6.6-1.

Inservice testing of pumps and valves will be conducted in accordance with the requirements of ASME OM Code and is discussed in Subsection 3.9.6.

6.6.2 Accessibility

Physical arrangement of vessels, piping, pumps, valves, and supports provide personnel access for the examination and, if necessary, the repair of welds. Removable insulation is provided on those piping systems which require volumetric, surface examination, and other examinations. Temporary platforms, scaffolding, and ladders will be provided for the removal of insulation and pump and valve parts whose removal is necessary to permit access for examination. Consideration was given during design fabrication to weld joint configuration and surface finish to permit thorough ultrasonic examination.

6.6.3 Examination Techniques and Procedures

Examination techniques and procedures, including any special techniques and procedures will be written and performed on Class 2 and 3 components in accordance with the requirements of ASME Code Section XI.

6.6.4 Inspection Intervals

The inspection intervals will be in accordance with the requirements of ASME Code Section XI, each interval having a 10-year duration, with exceptions as authorized in Section XI.

The inspection schedule will be in accordance with Section XI, Subsections IWC and IWD for Class 2 and 3 components, respectively. Inspections will be performed during normal operations where possible, with the remainder performed concurrent with refueling and/or maintenance shutdowns, occurring during the inspection interval.

6.6.5 Examination Categories and Requirements

The inservice inspection examination categories and requirements for Class 2 components are in agreement with the requirements of ASME Code Section XI, Subsection IWC. Class 3 components will be examined in accordance with the requirements of ASME Code Section XI, Subsection IWD.

6.6.6 Evaluation of Examination Results

Evaluation of the preservice and subsequent inservice examination results will be conducted in accordance with the requirements of ASME Code Section XI.

Necessary repairs or replacement of components will be accomplished in accordance with the requirements of ASME Code Section XI.

The data obtained from the preservice inspection will establish the initial base line for subsequent inservice inspections.

6.6.7 System Pressure Tests

System pressure tests will be performed in accordance with the requirements of ASME Code Section XI, for Class 2 and 3 systems. System pressure tests will be conducted at the frequency required by ASME Code Section XI in order to perform visual examinations for evidence of component leakage.

6.6.8 Augmented Inservice Inspection to Protect Against Postulated Piping Failures

During the first ten-year inservice inspection (ISI) interval, high energy Class 2 piping located between the containment isolation valves (in the break exclusion area) was examined as follows:

One hundred percent of all circumferential and longitudinal welds of piping larger than 4 inch nominal pipe size and a surface examination of all socket welds.

Starting in the second ten-year ISI interval, in lieu of the above requirements, EPRI Topical Reports Risk-Informed ISI (TR-112657 Revision B-A), Break Exclusion Region (TR-1006937 Revision 0-A), and ASME Code Case N-578-x are used to establish the risk evaluation, selection criteria, and examination methods. The NRC approved the use of this alternate method in an SER dated June 27, 2002. The weld population subject to examination under the Risk-Informed BER Program are non-exempted piping welds as determined in accordance with the rules of ASME Section XI, Edition and Addenda as applicable to the existing ISI program.

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TABLE 6.6-1
PRESERVICE EXAMINATION COVERAGE

Code Class	Coverage	Code Edition/Addenda 1974/ Summer 1975	1977/ Summer 1978	1980/ Winter 1980
	Selection/Exemption Criteria			
2	Components, supports	X		
2	Bolting		X	
3	Supports			X
	NDE Methods and Acceptance Criteria			
2	Components, supports, bolting		X	
	Visual Methods and Acceptance Criteria			
2	Components, supports, bolting		X	
3	Supports		X	

6.7 MAIN STEAM ISOLATION VALVE LEAKAGE CONTROL SYSTEM (MSIVLCS)

Note: As a result of the re-analysis of the Loss of Coolant Accident (LOCA) using Alternative Source Term (AST) Methodology, it is no longer necessary to credit the Main Steam Isolation Valve Leakage Control System (MSIVLCS) for post-LOCA activity leakage mitigation. The system has been left in place as a passive system and is not required to perform any safety function.

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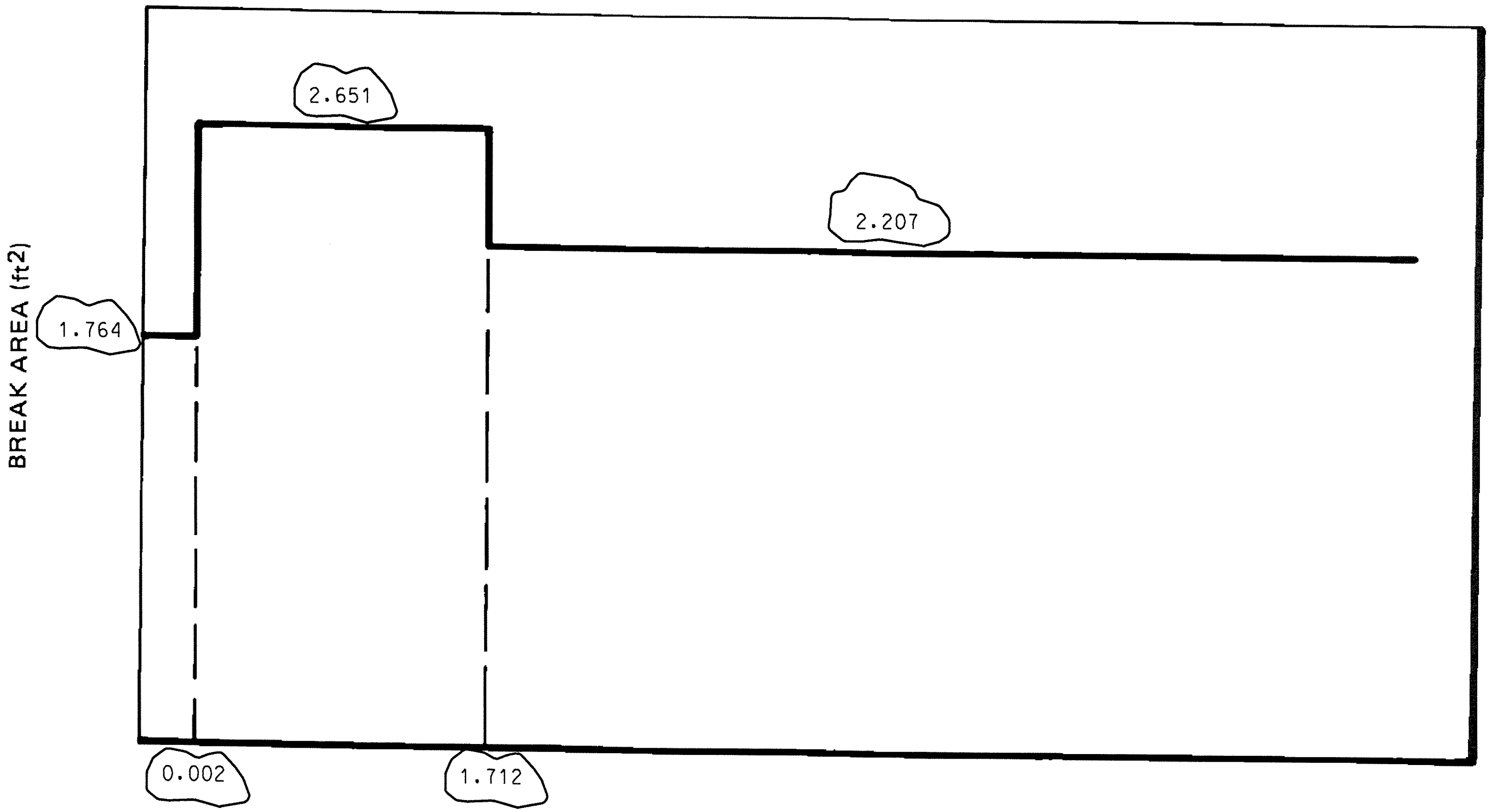


Figure 6.2-1. Effective Blowdown Area for Recirculation Line Break

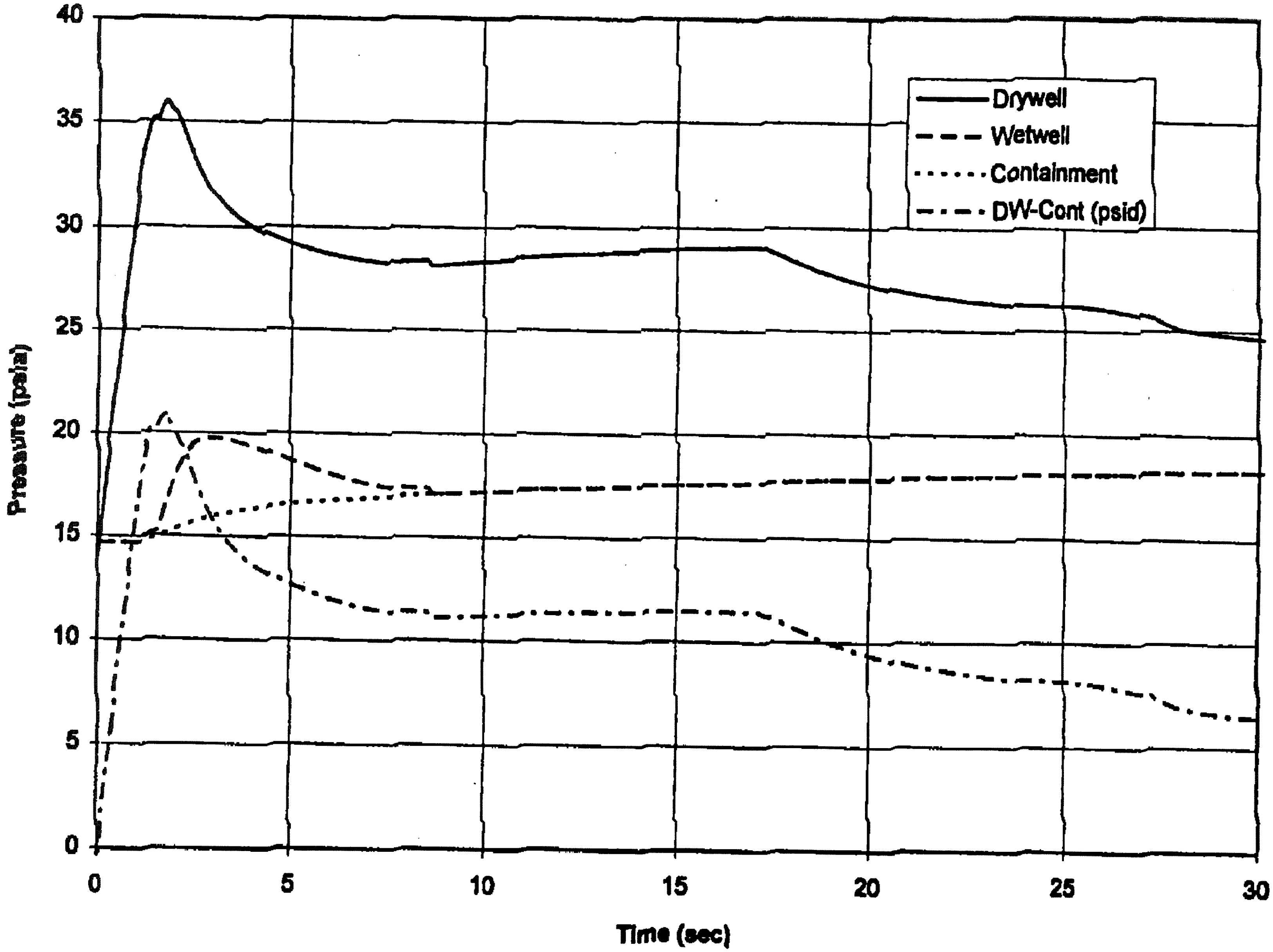
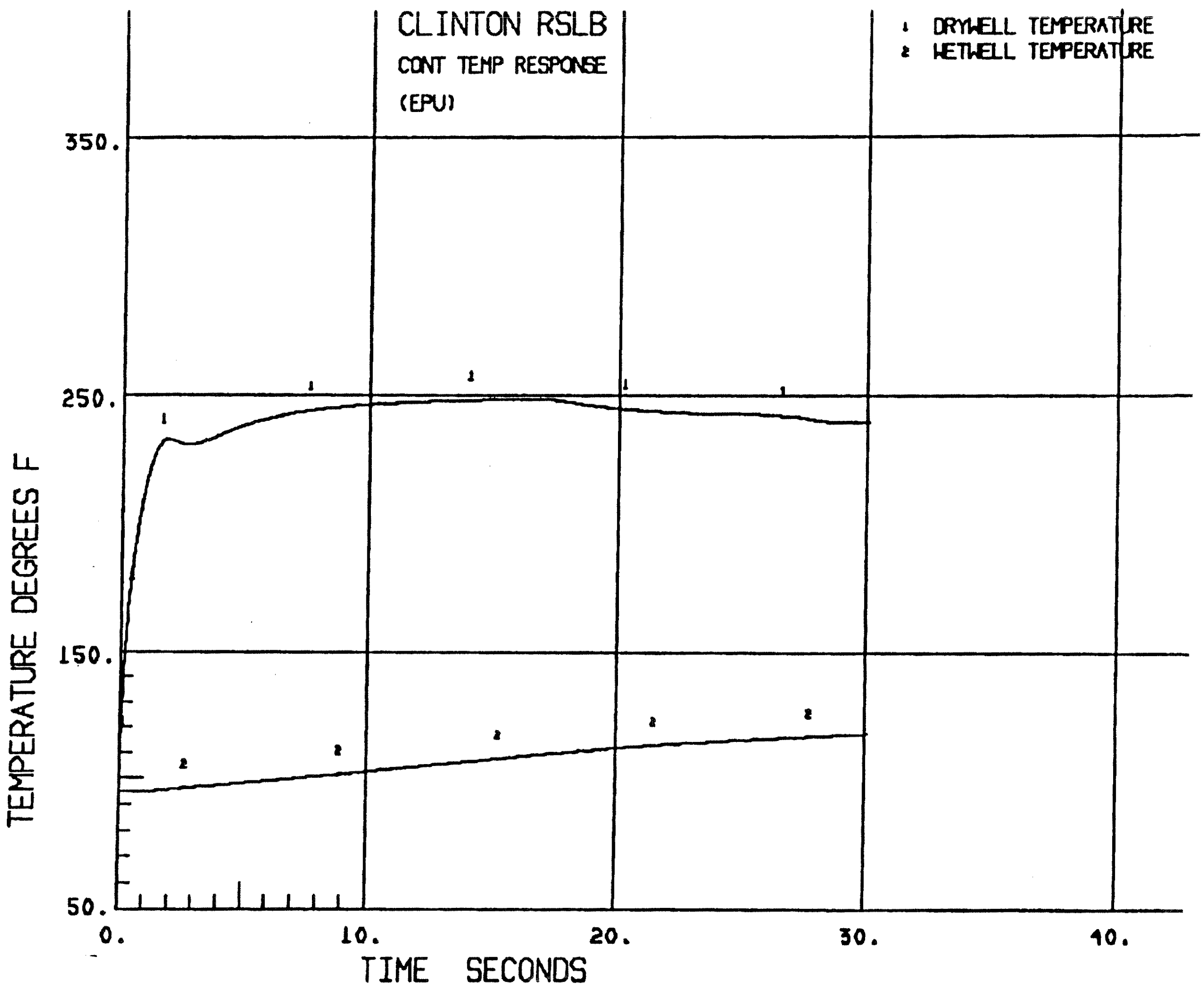


FIGURE 6.2-2: SHORT-TERM PRESSURE RESPONSE FOLLOWING A RECIRCULATION LINE BREAK



CLINTON POWER STATION

FIGURE 6.2-3
SHORT-TERM TEMPERATURE
RESPONSE FOLLOWING A
RECIRCULATION LINE BREAK

Figure 6.2-4
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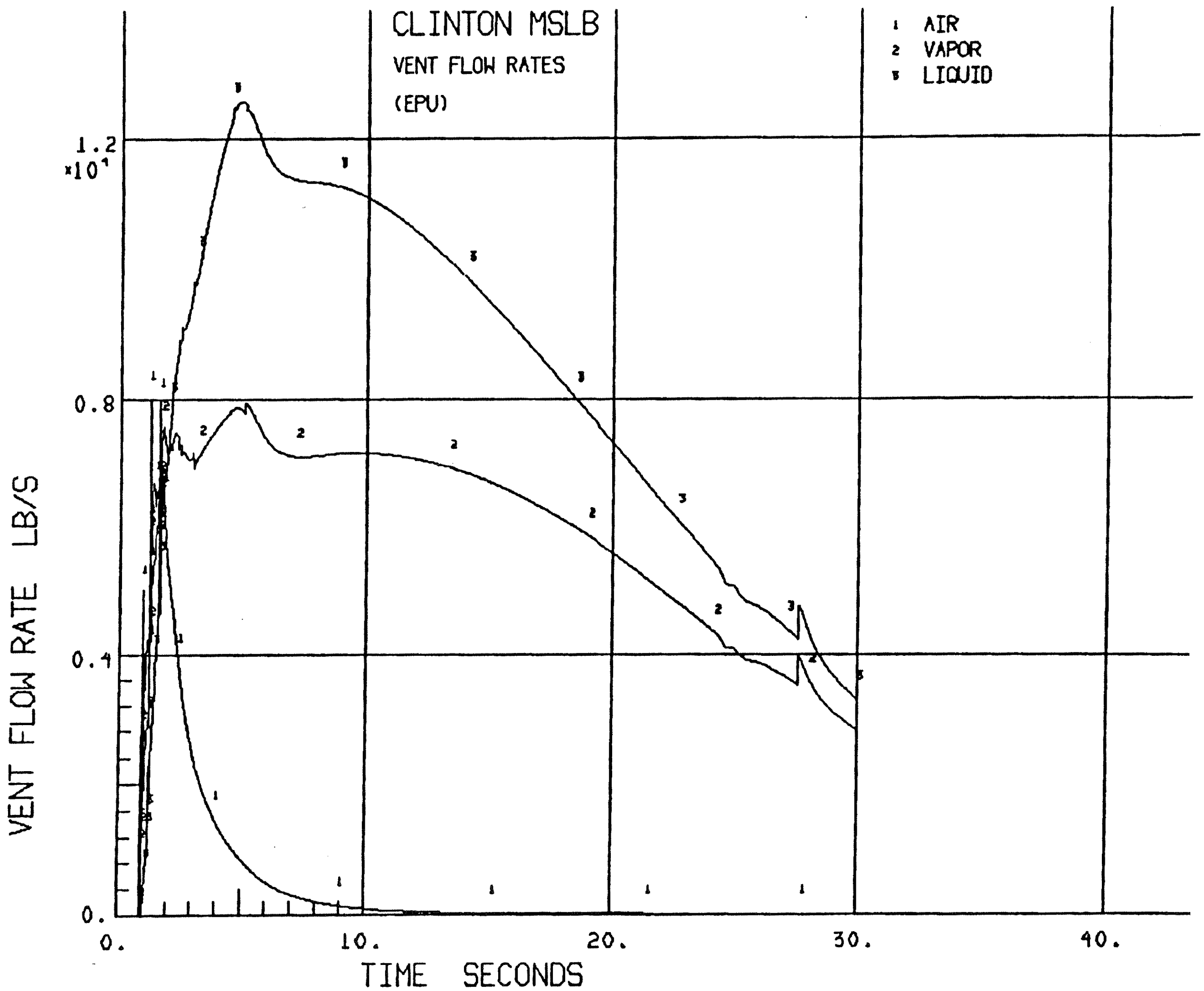

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FIGURE 6.2-5
SHORT-TERM VENT FLOW FOLLOWING
A RECIRCULATION LINE BREAK

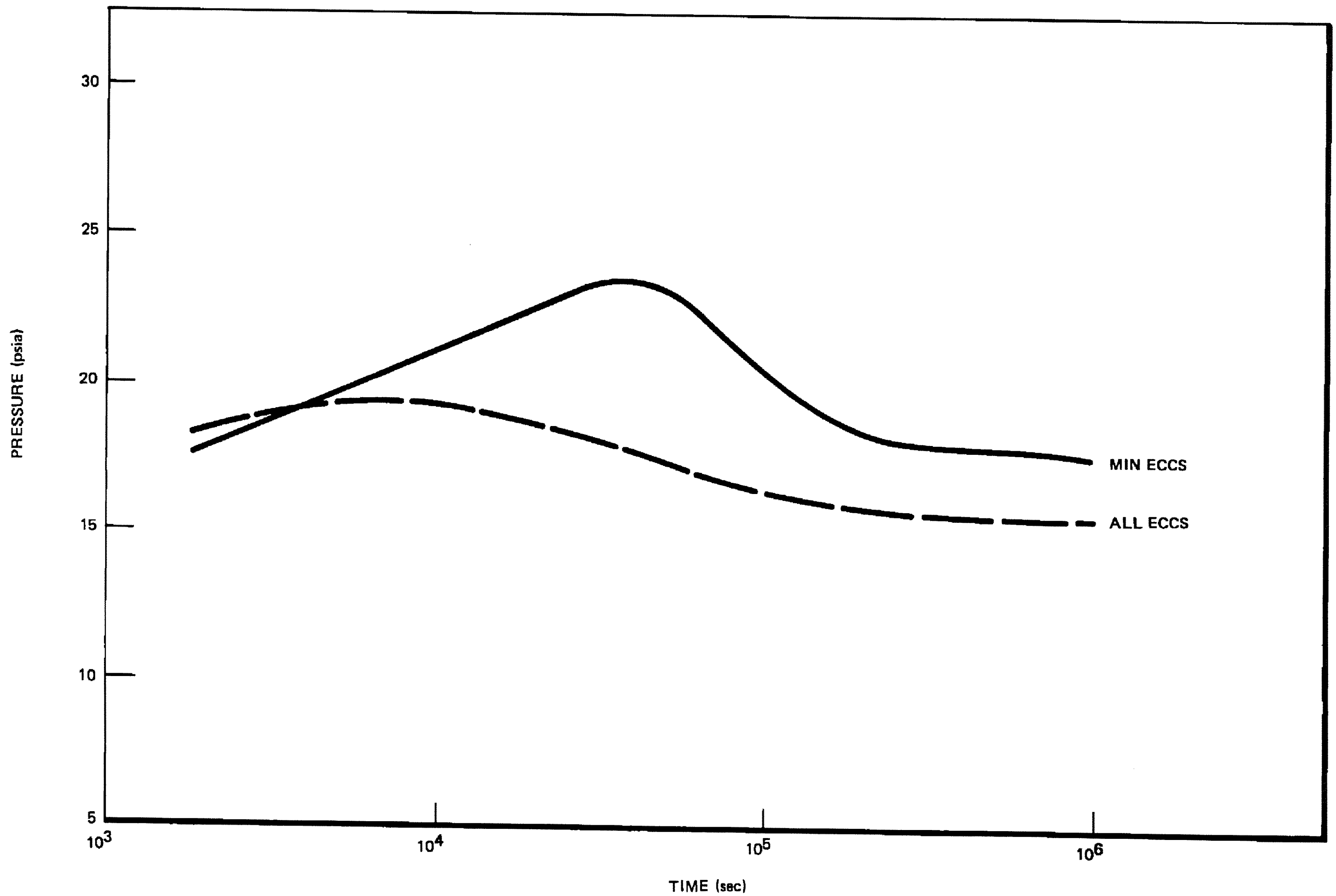

FIGURE 6.2-6. LONG-TERM PRESSURE RESPONSE FOLLOWING A RECIRCULATION LINE OR MAIN STEAMLINE BREAK (AT 2952 MWT)

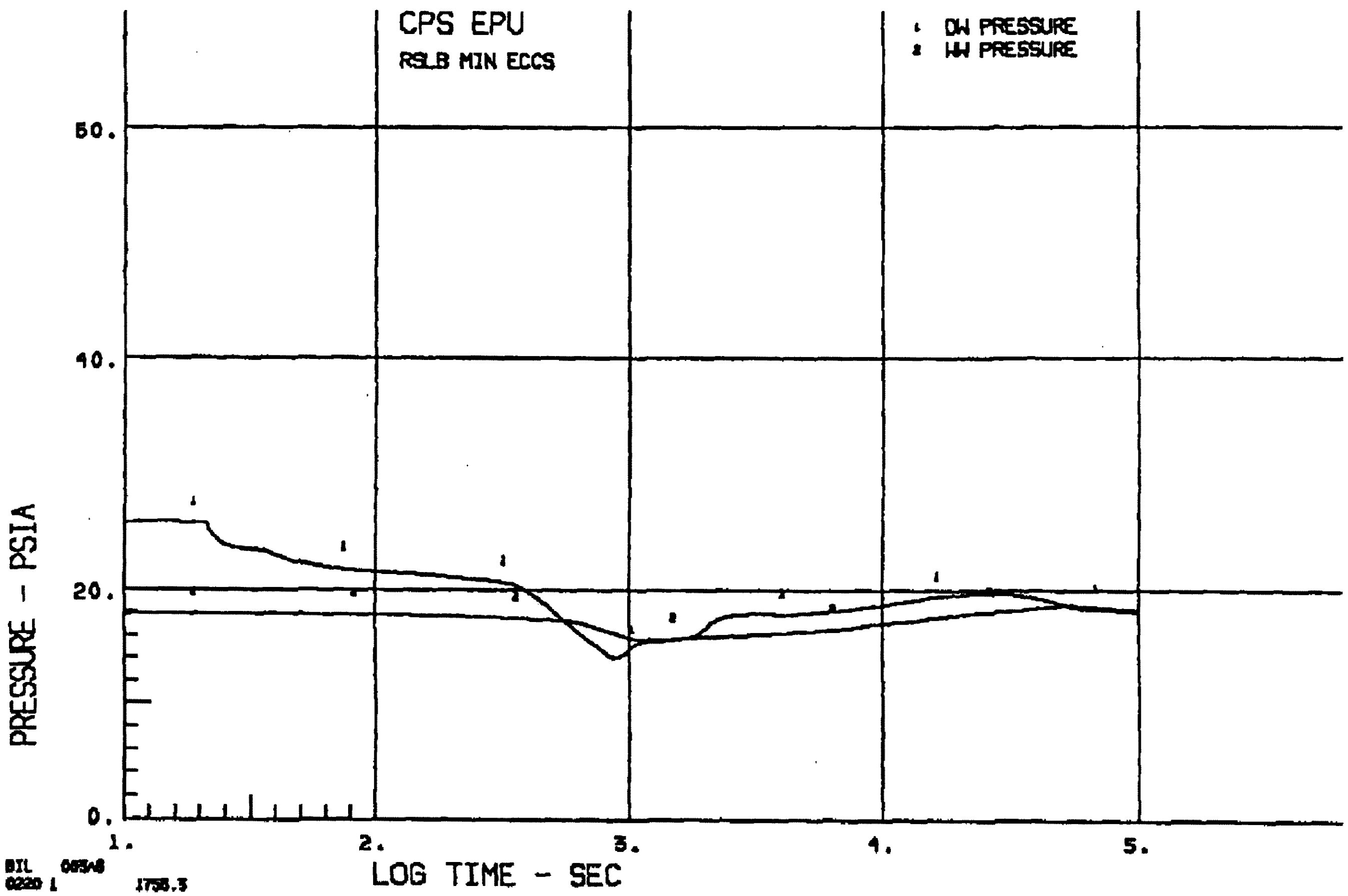

Figure 6.2-6a. Long-Term Pressure Response Following a Recirculation Line Break (at 3543 MWt)

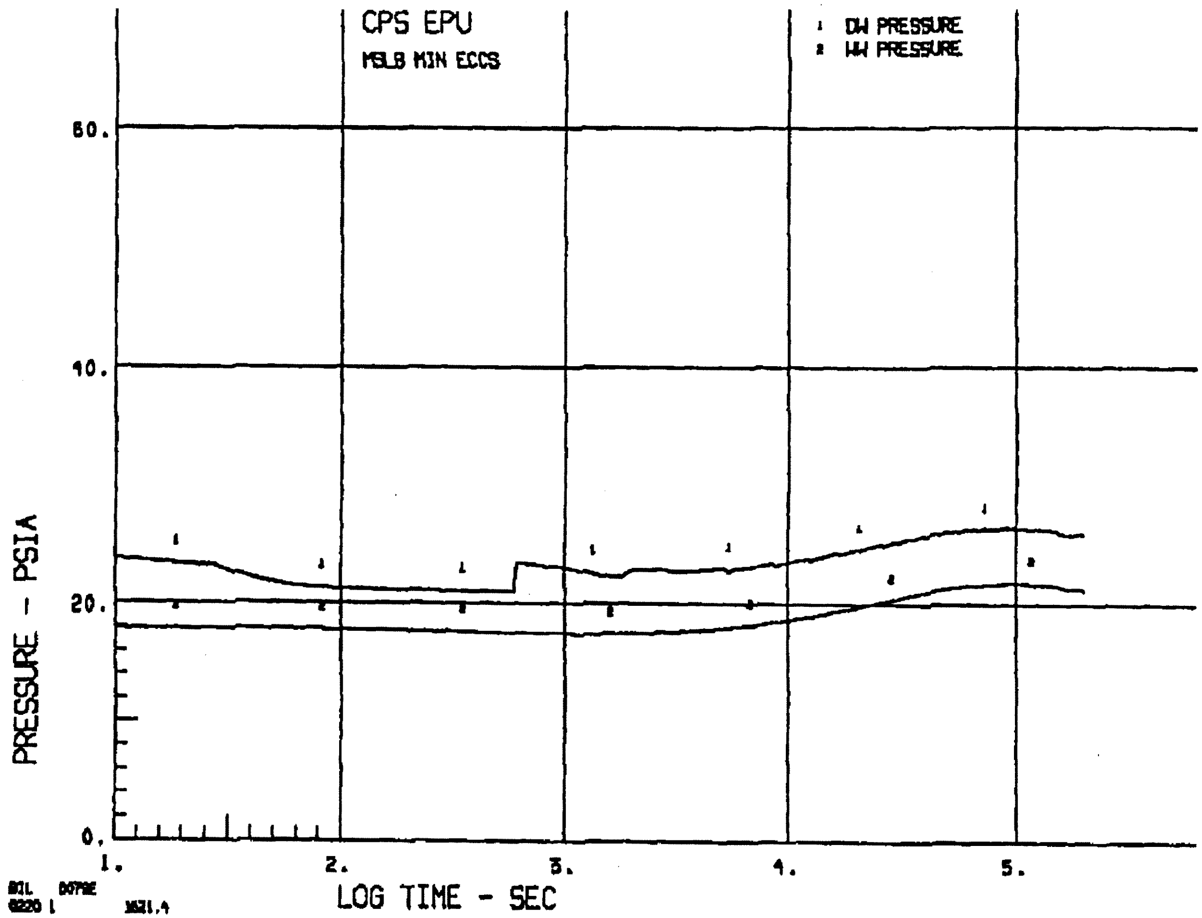

Figure 6.2-6b. Long-Term Pressure Response Following a Main Steamline Break (at 3543 MWt)

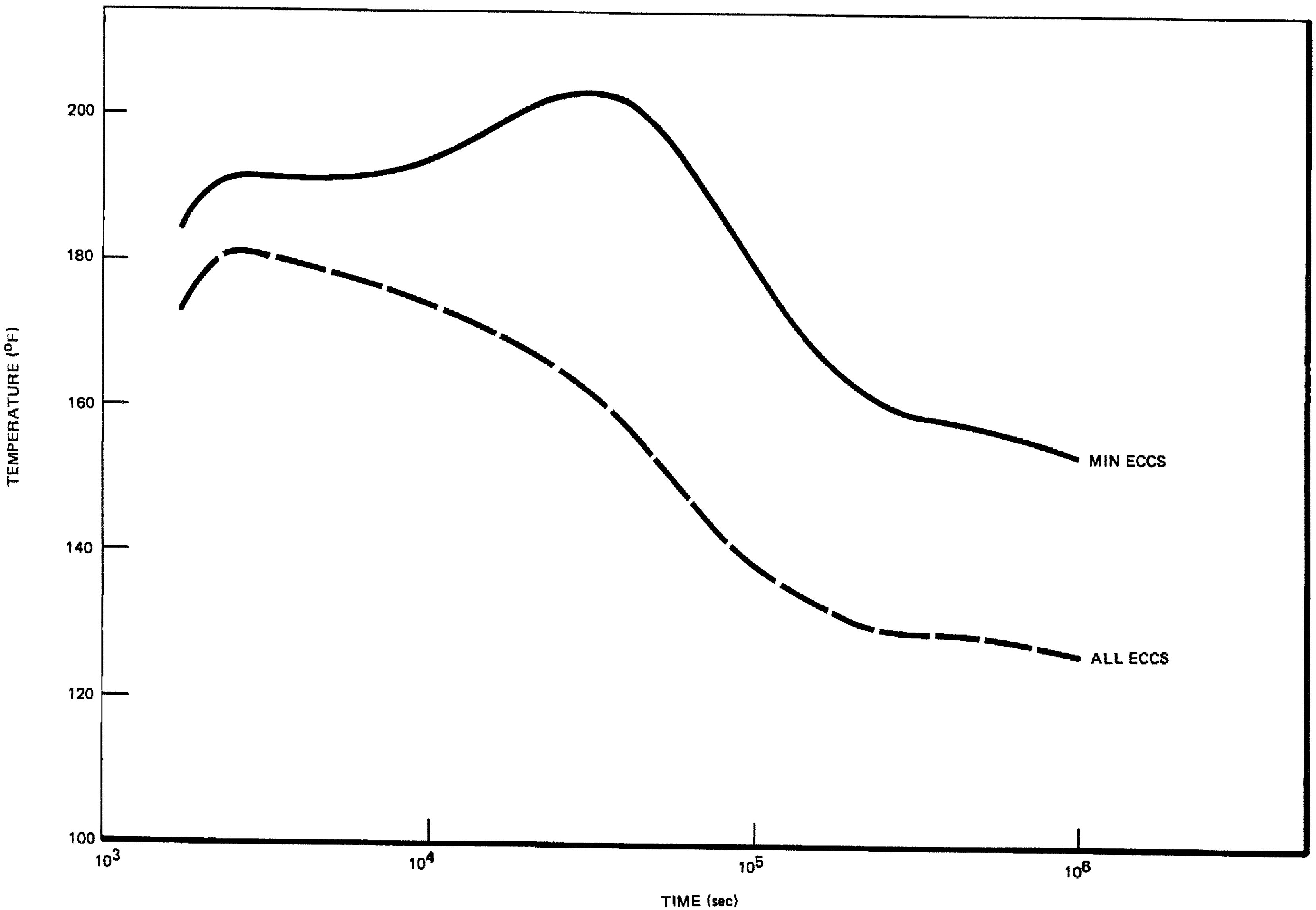

FIGURE 6.2-7. LONG-TERM DRYWELL TEMPERATURE RESPONSE FOLLOWING A RECIRCULATION LINE OR MAIN STEAMLINE BREAK (AT 2952 MWT)

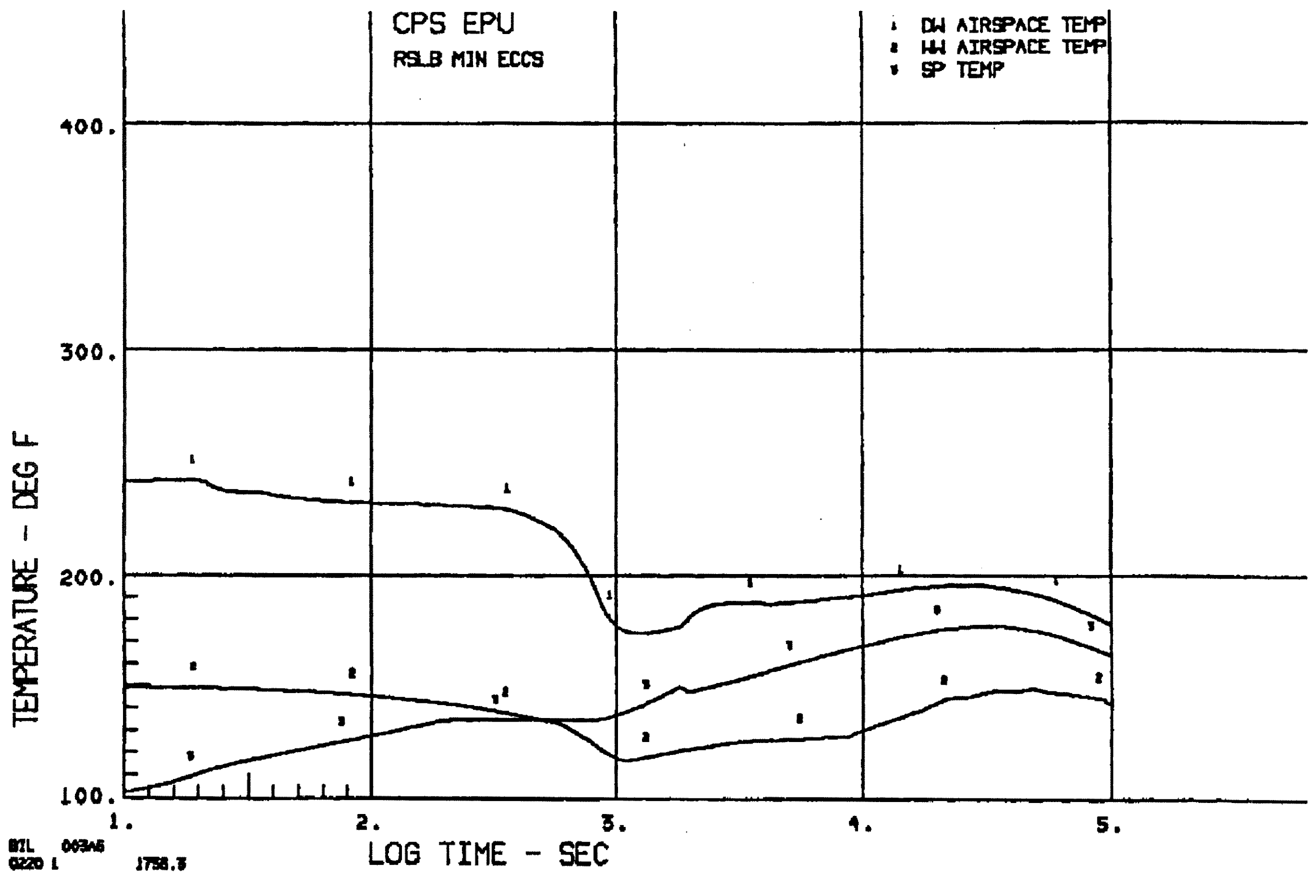

Figure 6.2-7a. Long-Term Temperature Response Following a Reciculation Line Break (at 3543 MWt)

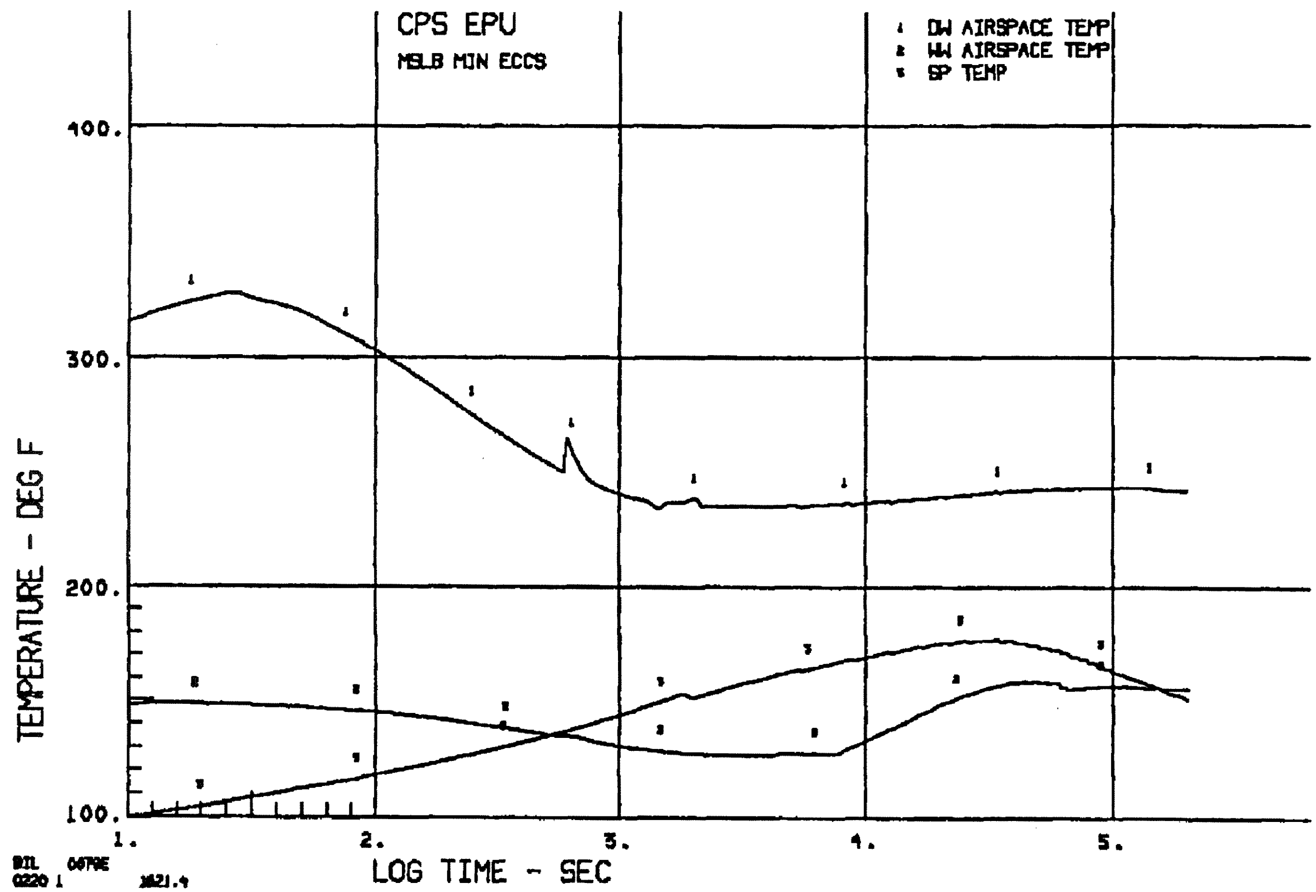

Figure 6.2-7b. Long-Term Temperature Response Following a Main Steamline Break (at 3543 MWt)

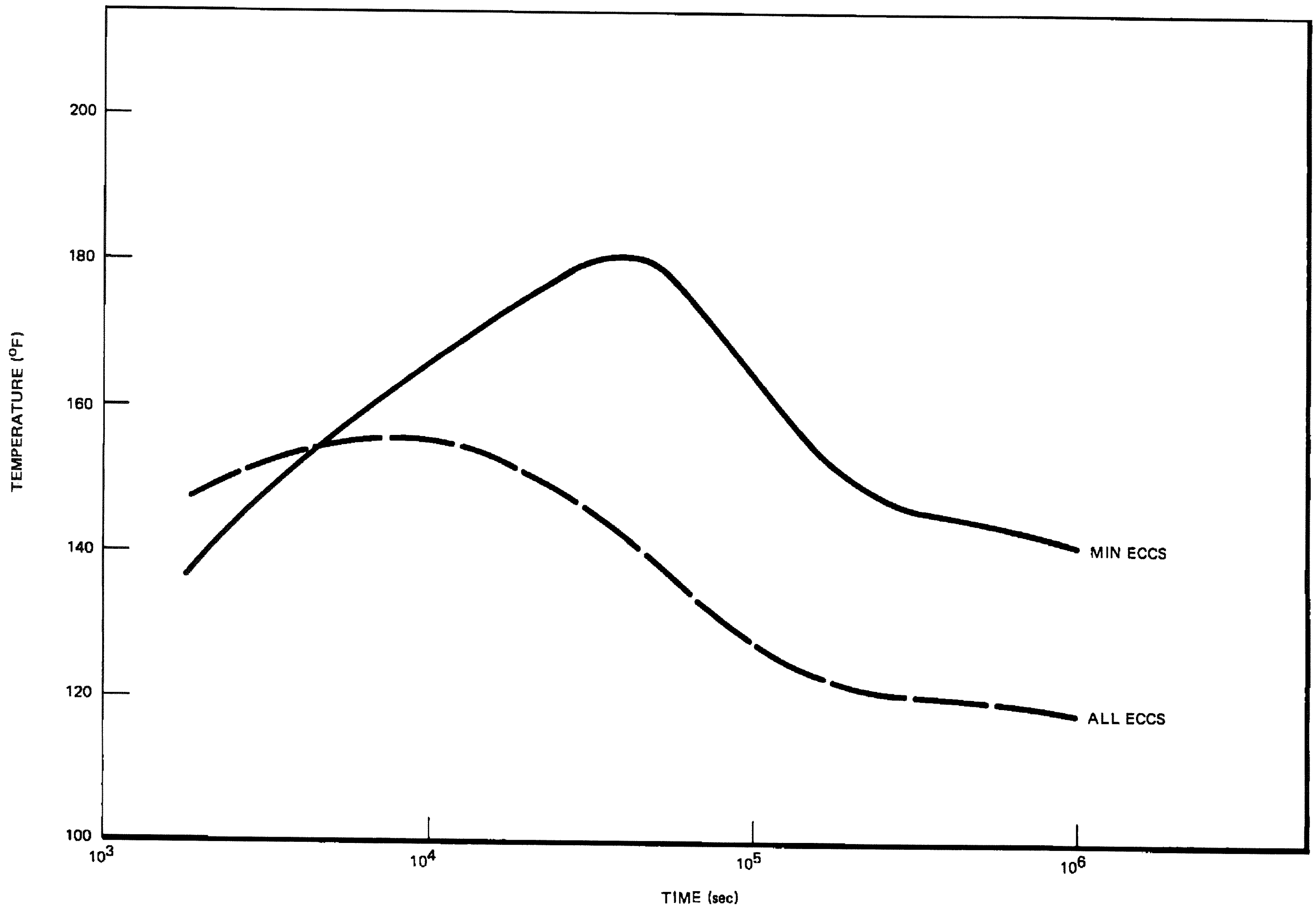

FIGURE 6.2-8. LONG-TERM SUPPRESSION POOL TEMPERATURE RESPONSE FOLLOWING A RECIRCULATION LINE OR MAIN STEAMLINE BREAK (AT 2952 MWT)

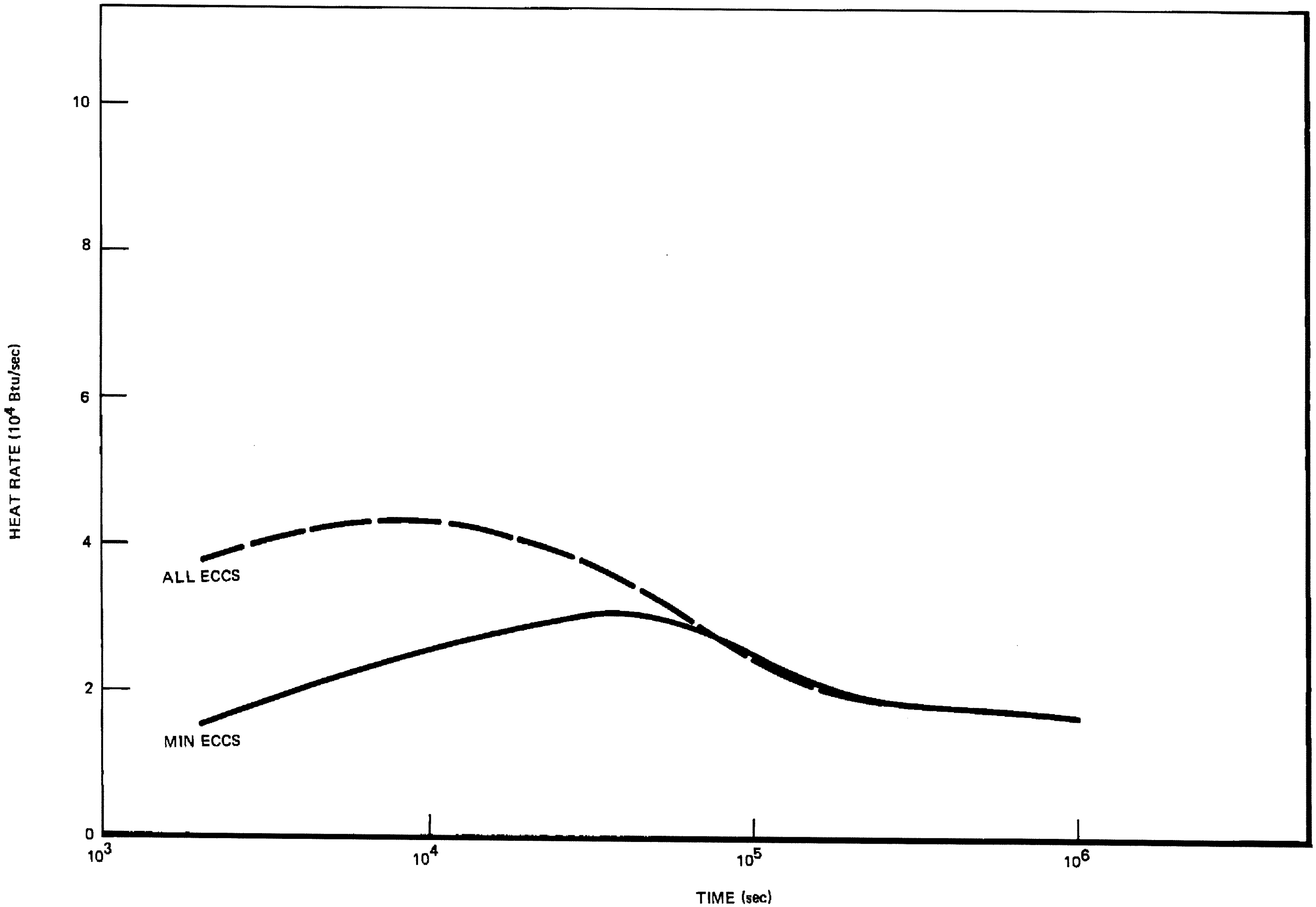

FIGURE 6.2-9. RHR HEAT REMOVAL RATE FOLLOWING A RECIRCULATION LINE OR MAIN STEAMLINE BREAK (AT 2952 MWT)

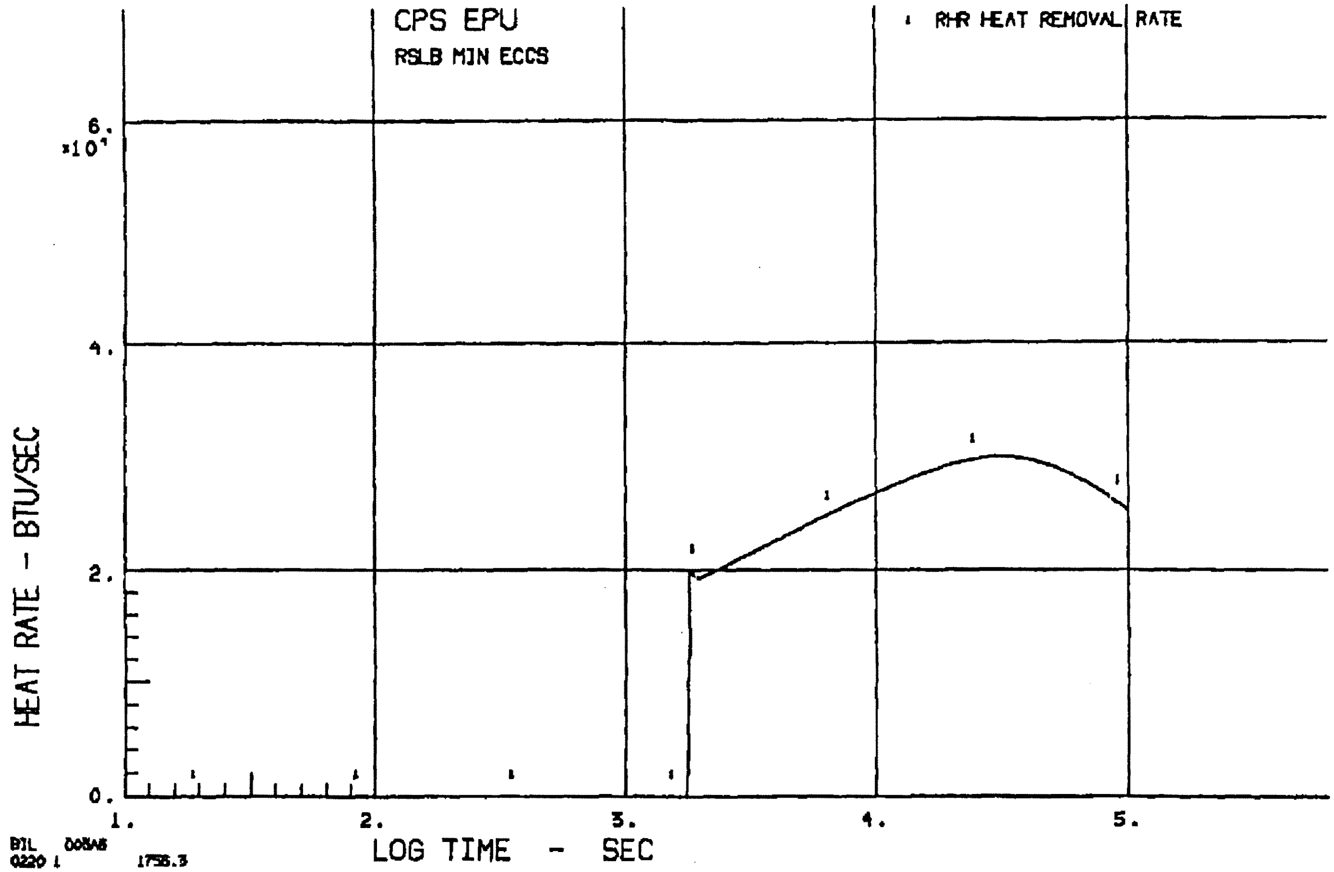

Figure 6.2-9a. RHR Heat Removal Rate Following a Recirculation Line Break (at 3543 MWt)

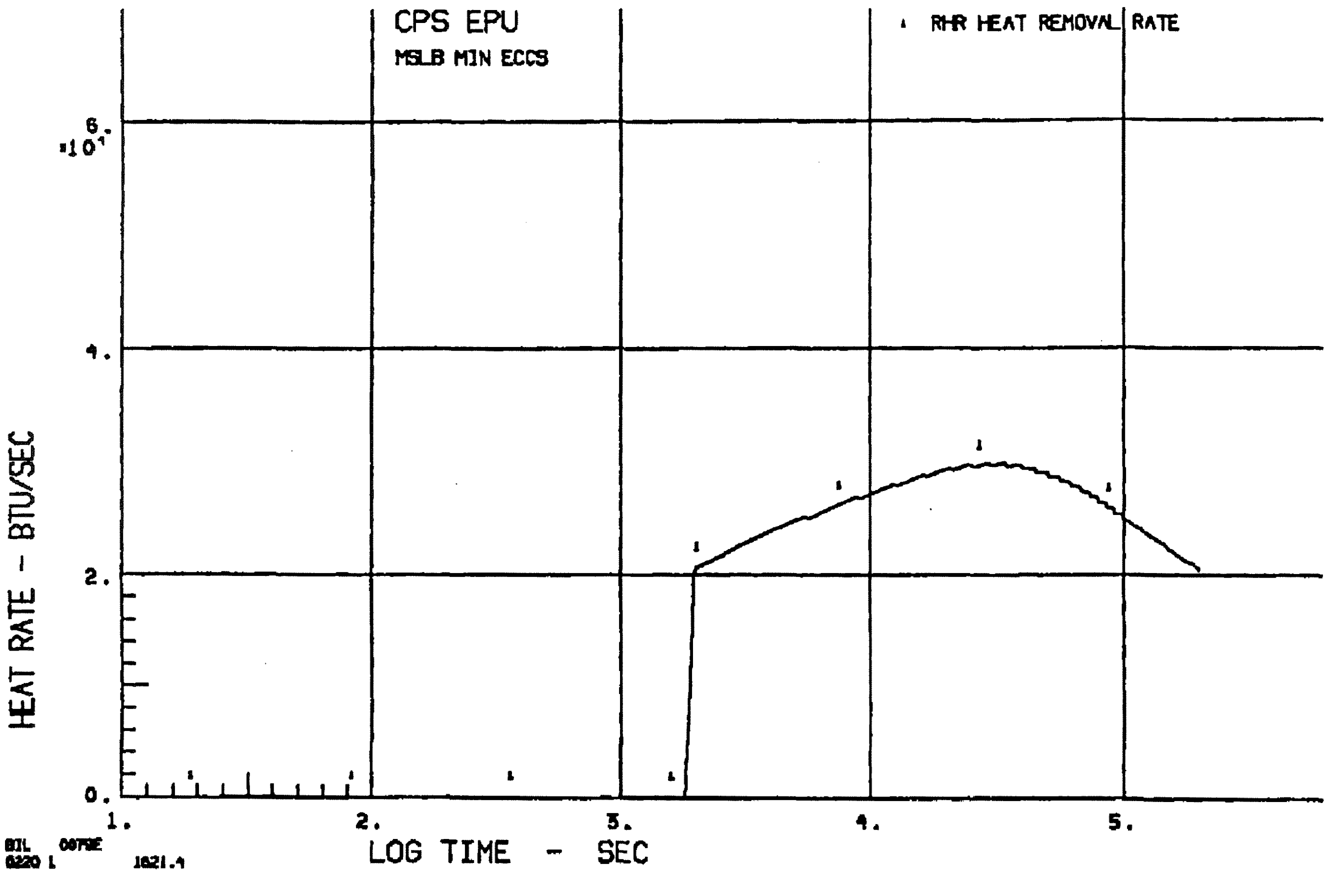

Figure 6.2-9b. RHR Heat Removal Rate Following a Main Steamline Break (at 3543 MWt)

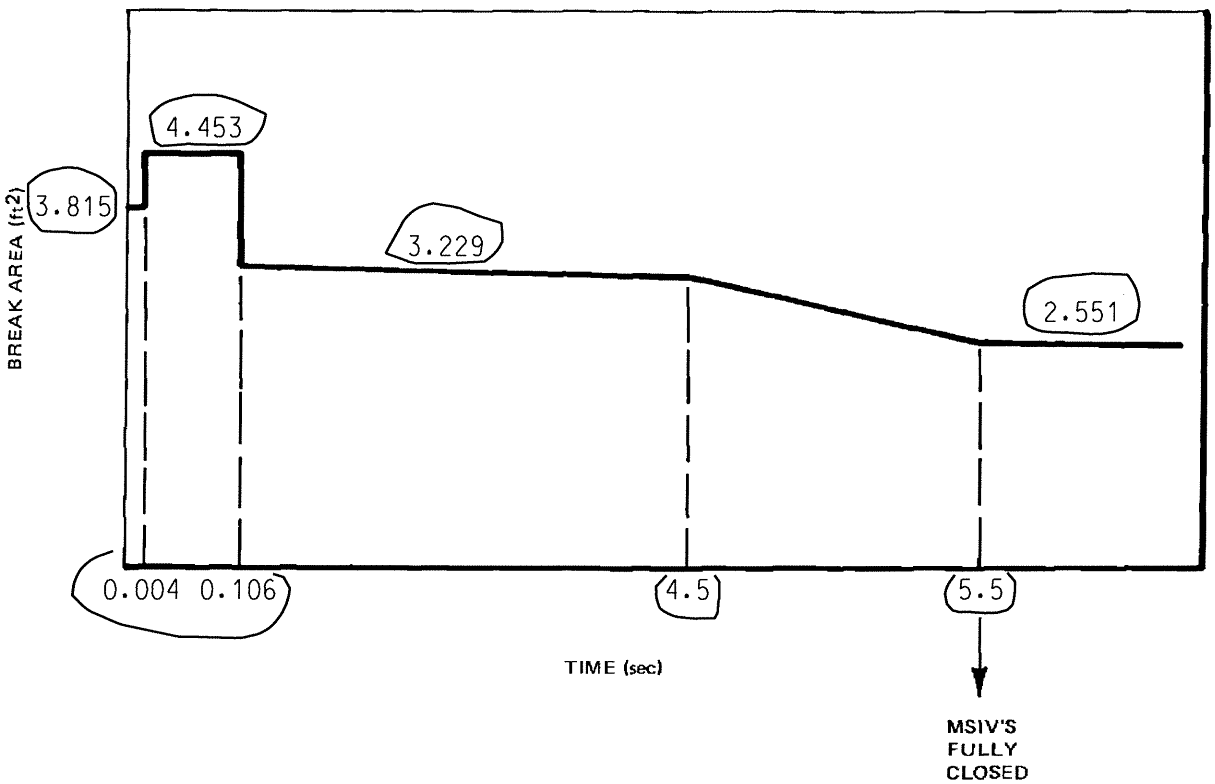

Figure 6.2-10. Effective Blowdown Area for Main Steamline Break

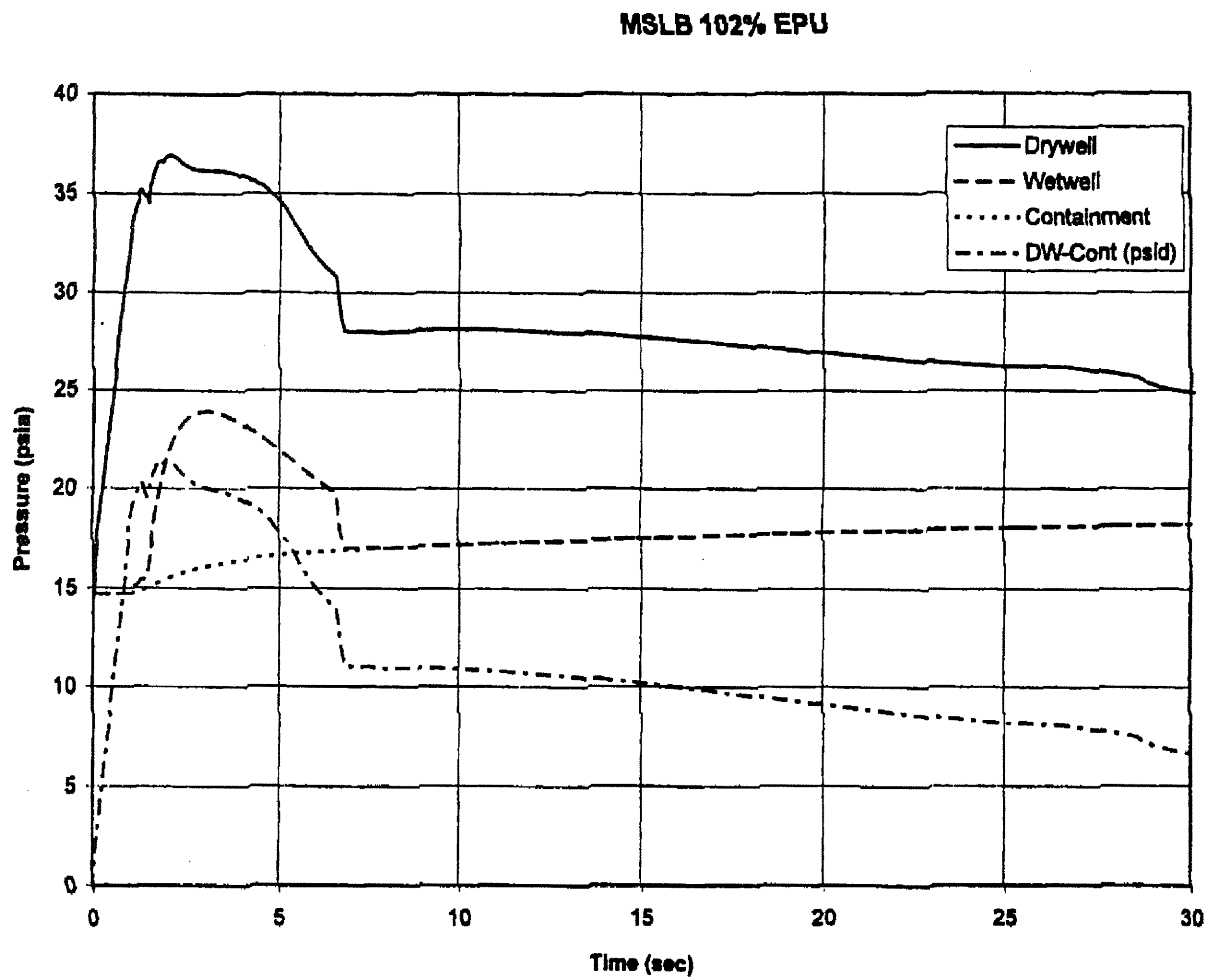

FIGURE 6.2-11. SHORT-TERM PRESSURE RESPONSE FOLLOWING A MAINSTEAM LINE BREAK

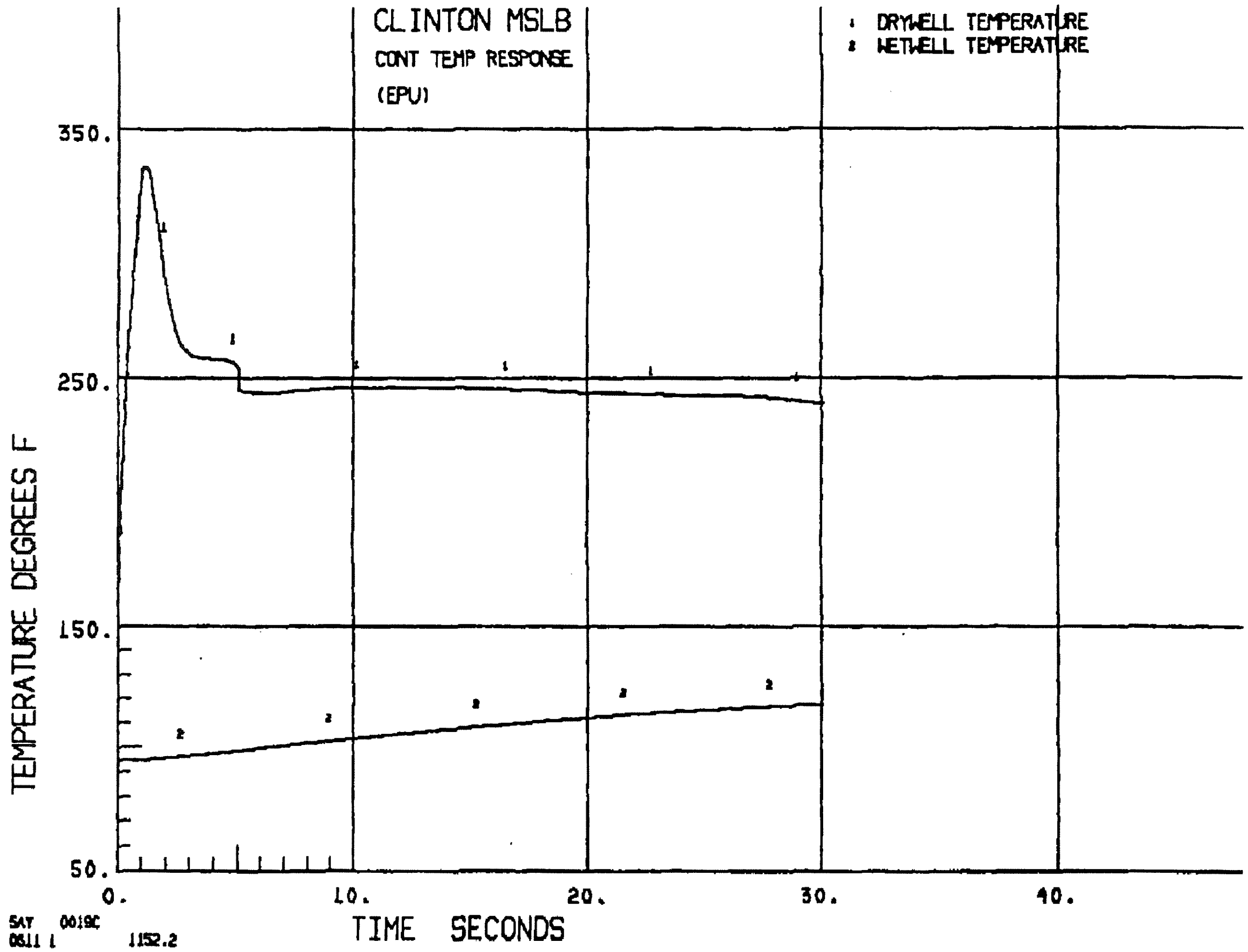

FIGURE 6.2-12. SHORT-TERM TEMPERATURE RESPONSE FOLLOWING A MAINSTEAM LINE BREAK

Figure 6.2-13
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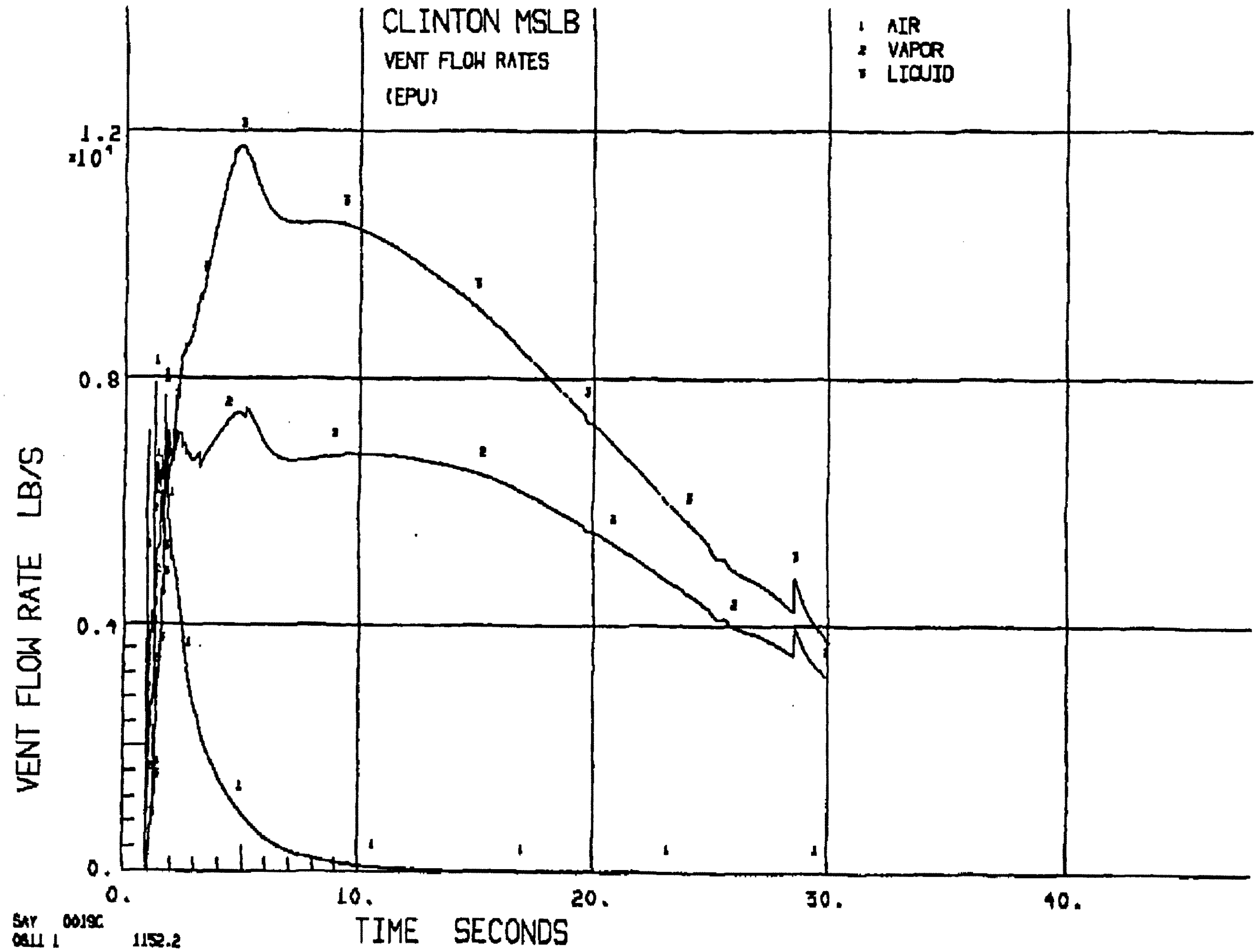

FIGURE 6.2-14. SHORT-TERM VENT FLOW FOLLOWING A MAINSTEAM LINE BREAK

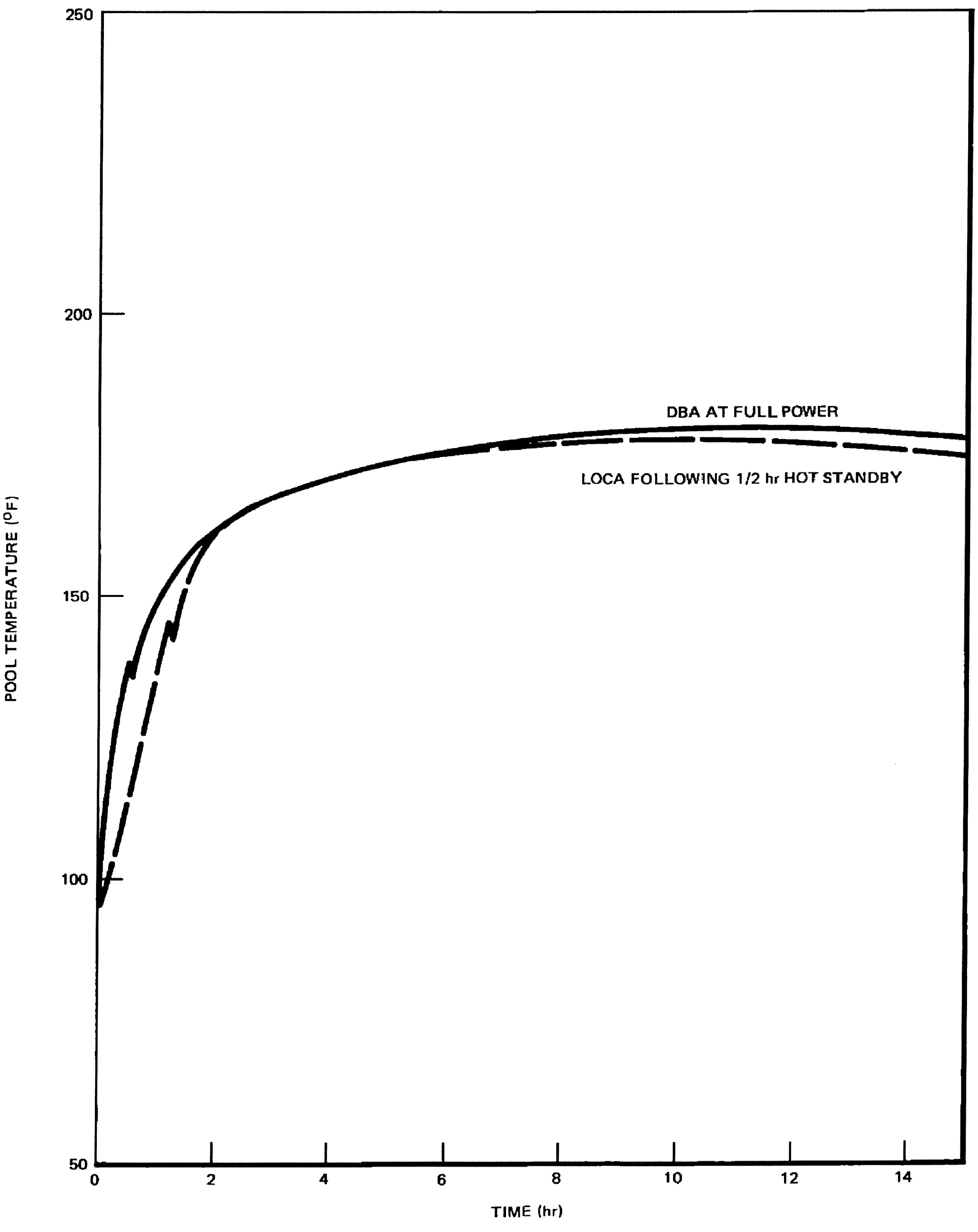

FIGURE 6.2-15. SUPRESSION POOL TEMPERATURE RESPONSE FOR DBA AND FOR BLOWDOWN FOLLOWING HOT STANDBY OPERATION (AT 2952 MWT).

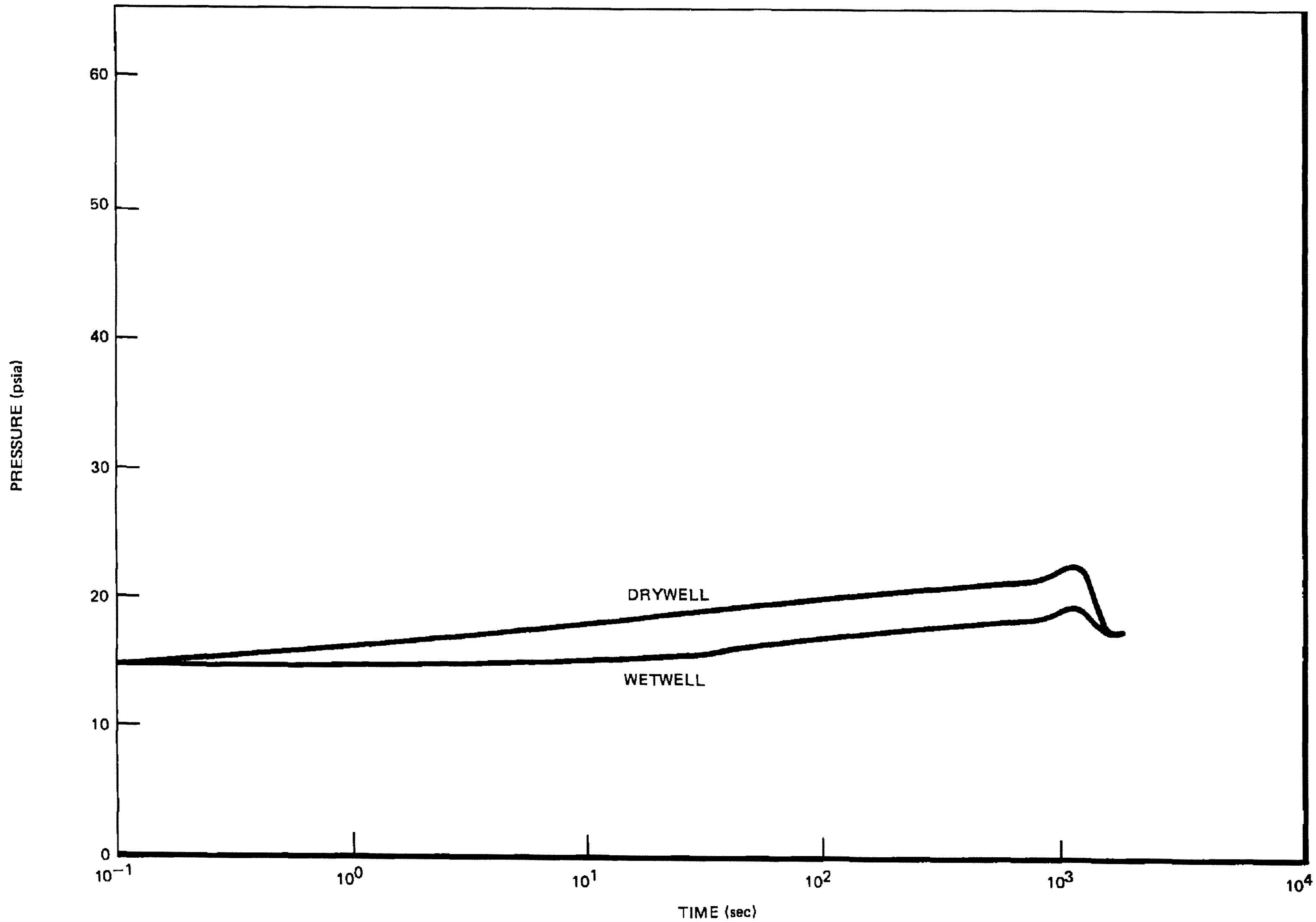

FIGURE 6.2-16. SHORT-TERM PRESSURE RESPONSE FOLLOWING AN INTERMEDIATE SIZE BREAK (AT 2952 MWT)

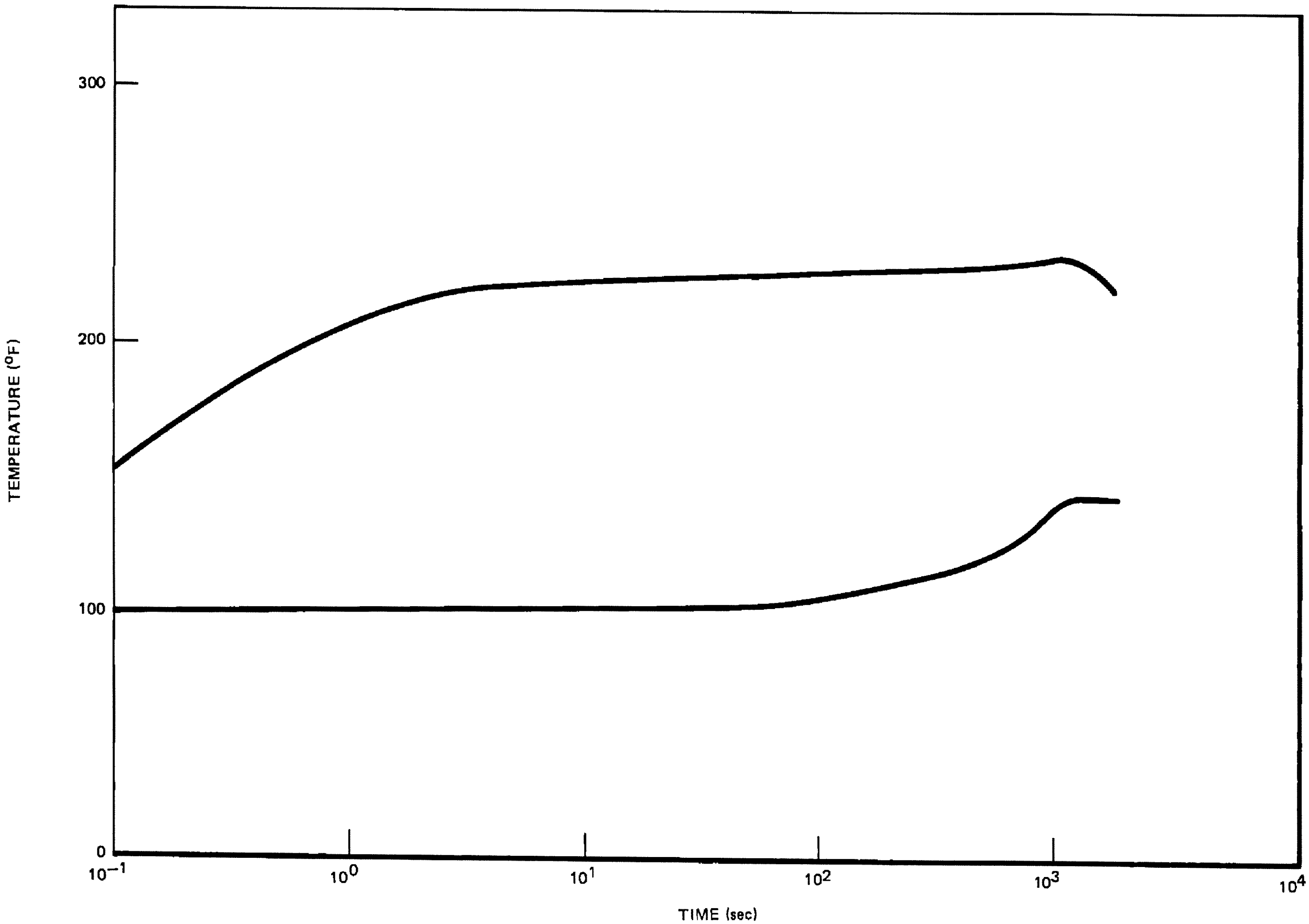

FIGURE 6.2-17. SHORT-TERM TEMPERATURE RESPONSE FOLLOWING AN INTERMEDIATE SIZE BREAK (AT 2952 MWT)

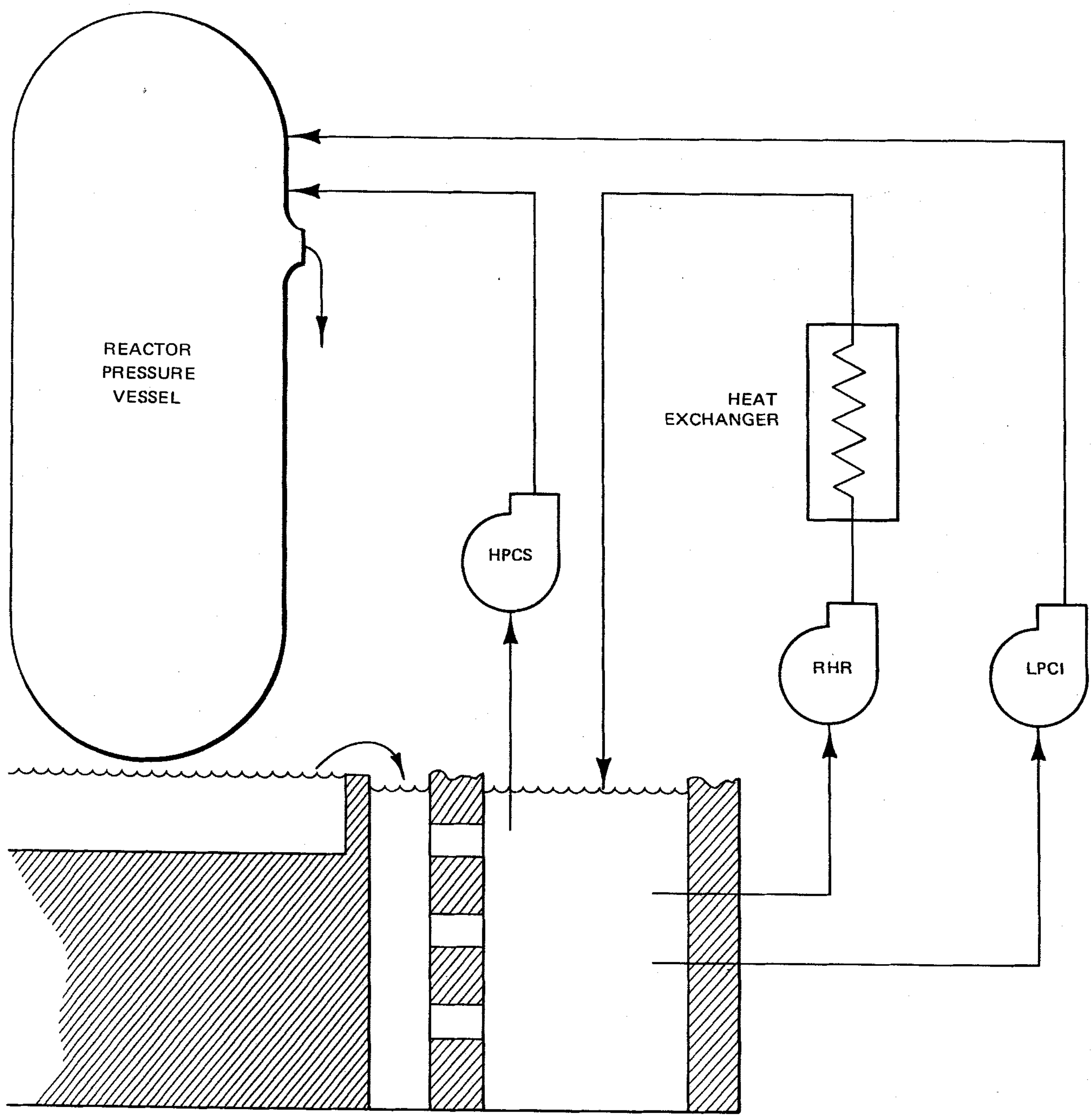

Figure 6.2-18. Schematic of the RHR Containment Cooling System Analytical Model (Minimum ECCS)

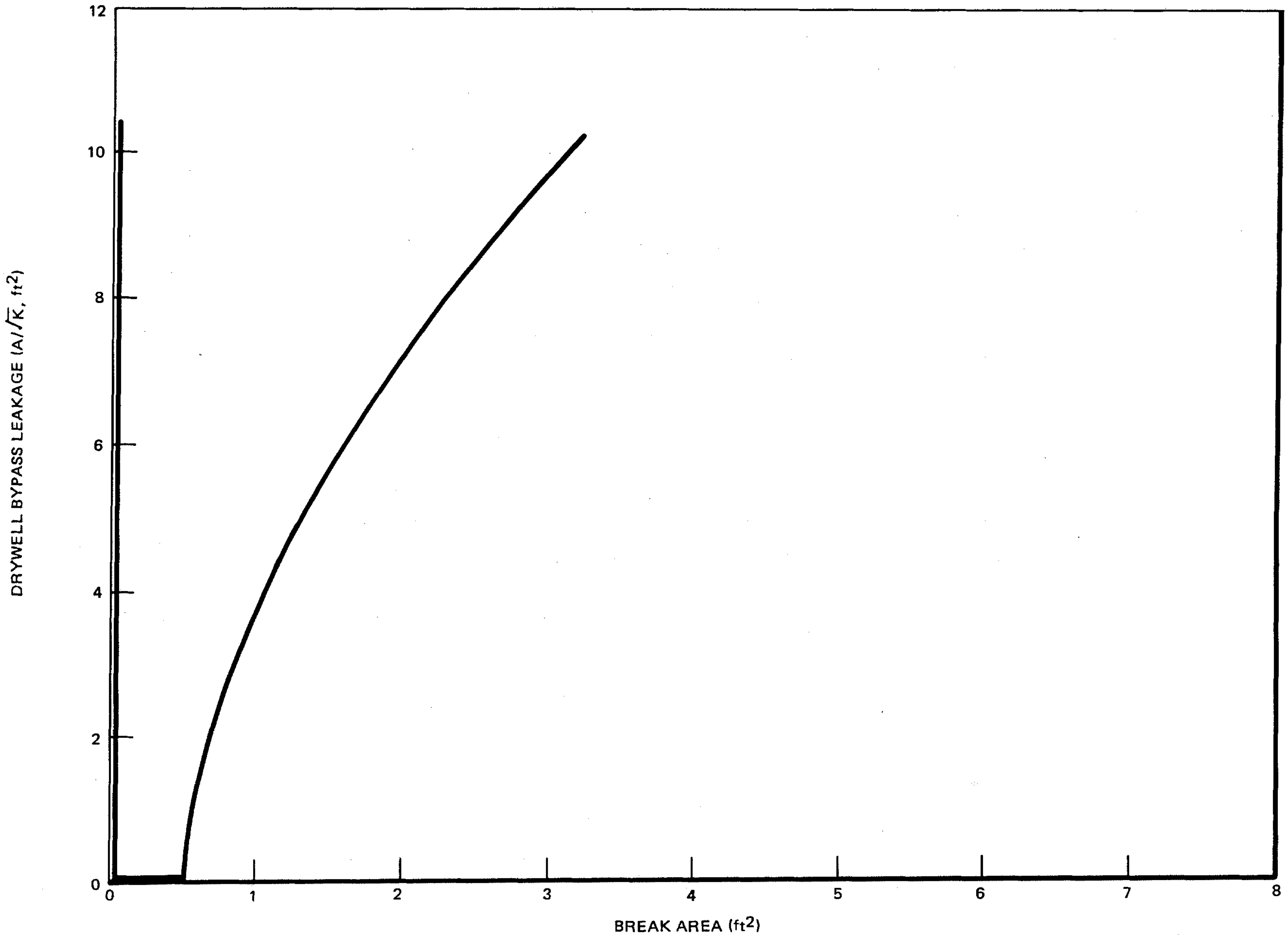

Figure 6.2-19. Maximum Allowable Steam Bypass Leakage Area
Without Containment Spray or Heat Sink

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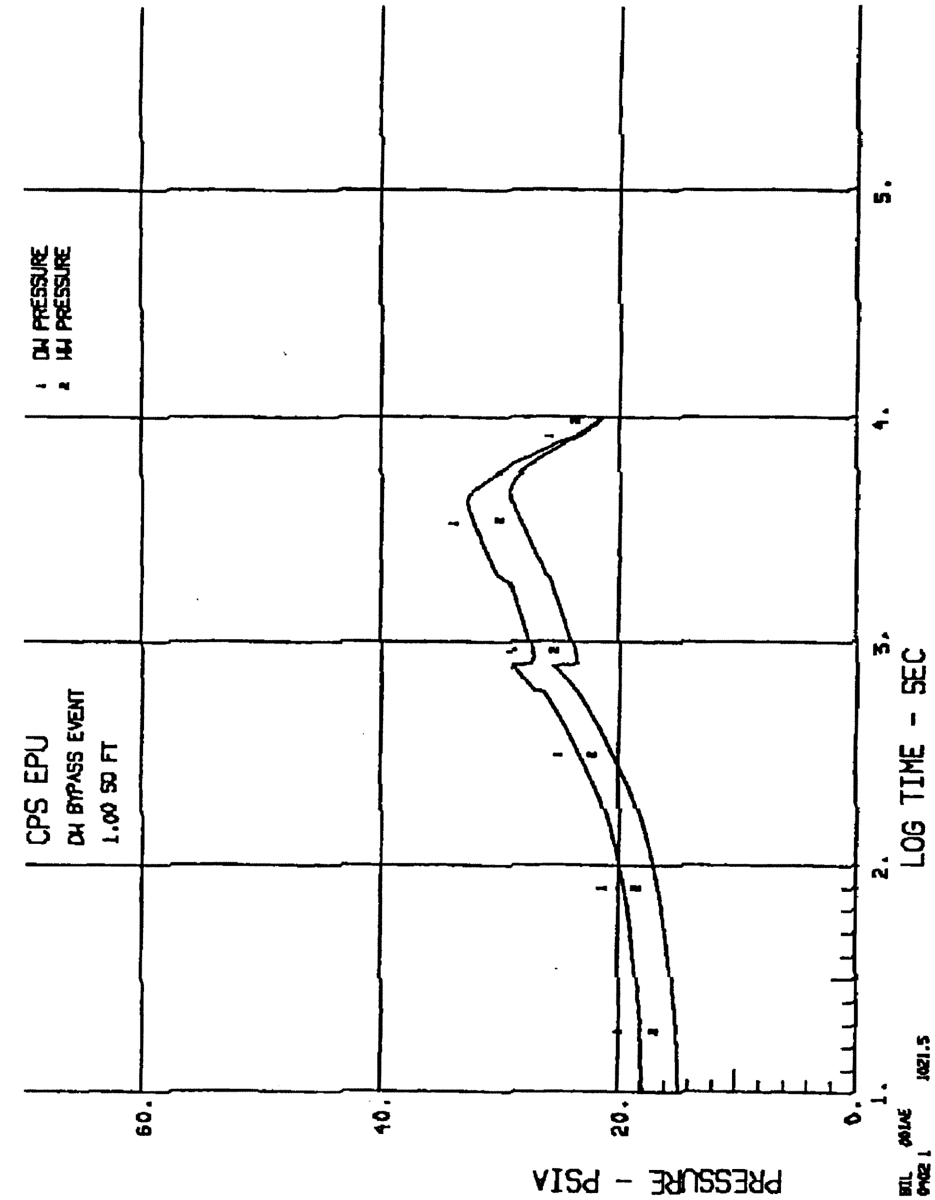
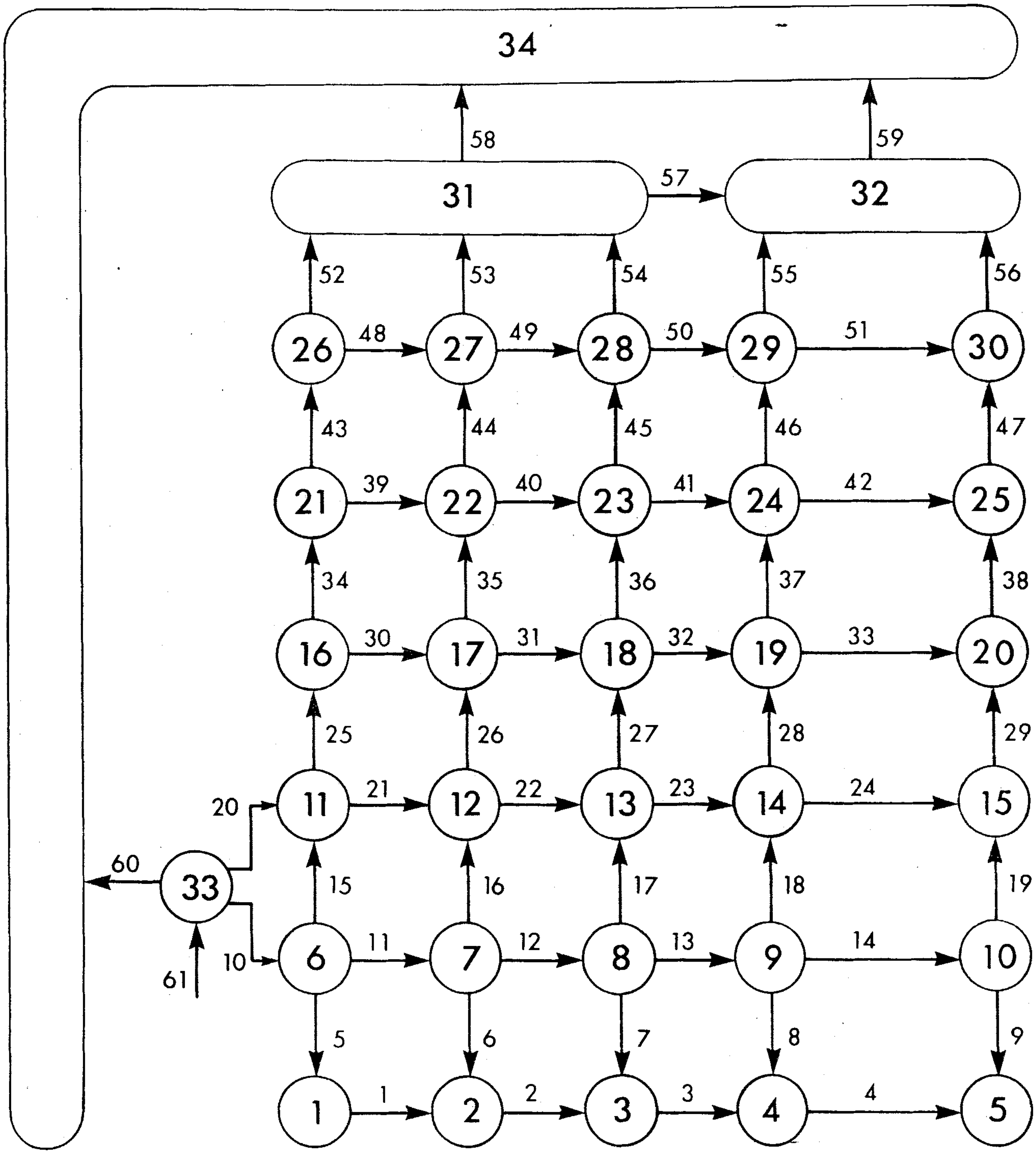


FIGURE 6.2-20



NOTE:

See Table 6.2-14 for a description of the nodes and Table 6.2-21 for a description of the vent paths.

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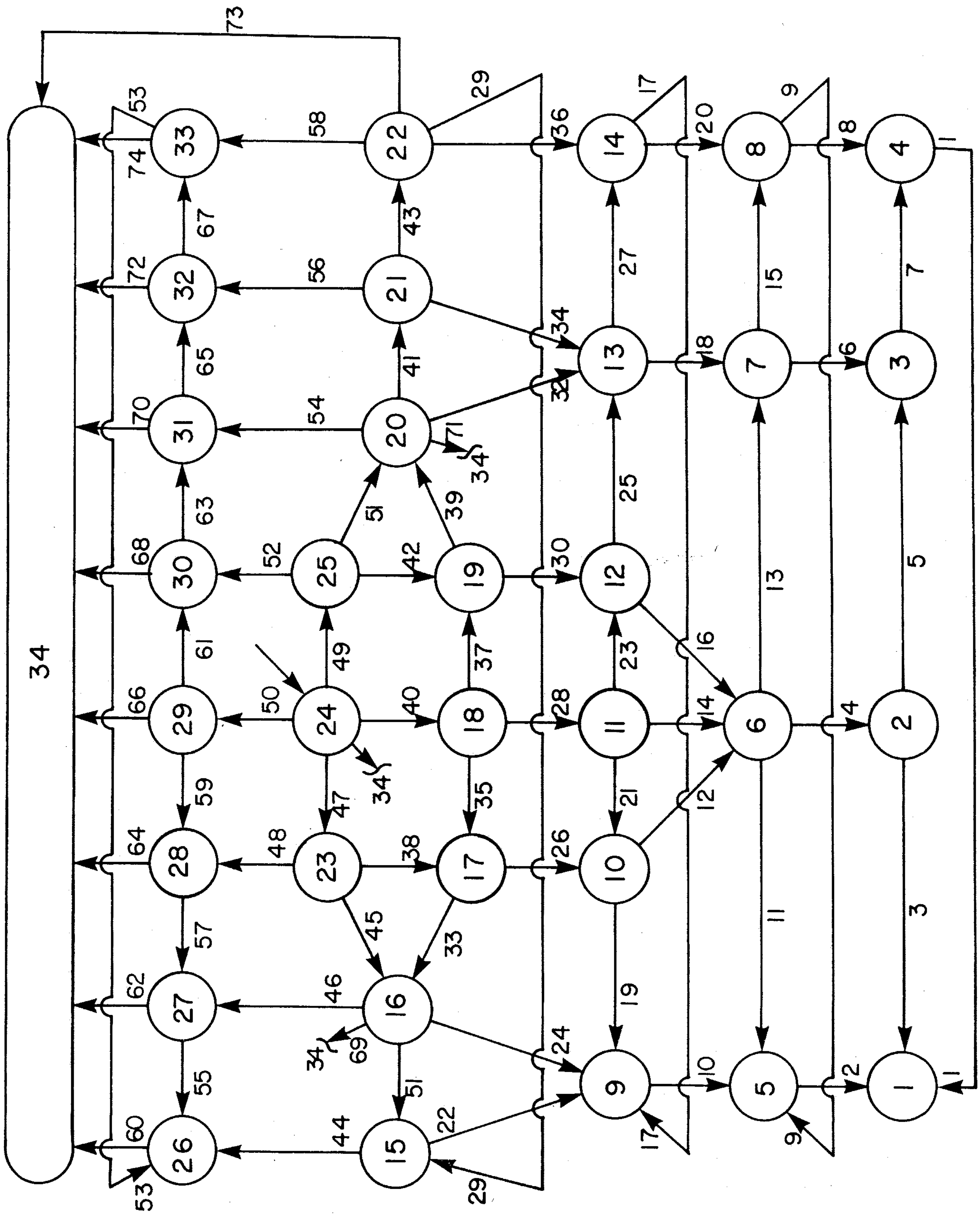
FIGURE 6.2-21 NODALIZATION SCHEMATIC FOR SACRIFICIAL SHIELD ANNULUS PRESSURIZATION ANALYSIS - RECIRCULATION OUTLET LINE BREAK



NOTE:

See Table 6.2-22 for a description of the nodes and Table 6.2-23 for a description of the vent paths.

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FIGURE 6.2-22

NODALIZATION SCHEMATIC FOR SACRIFICIAL SHIELD ANNULUS PRESSURIZATION ANALYSIS - FEEDWATER LINE BREAK

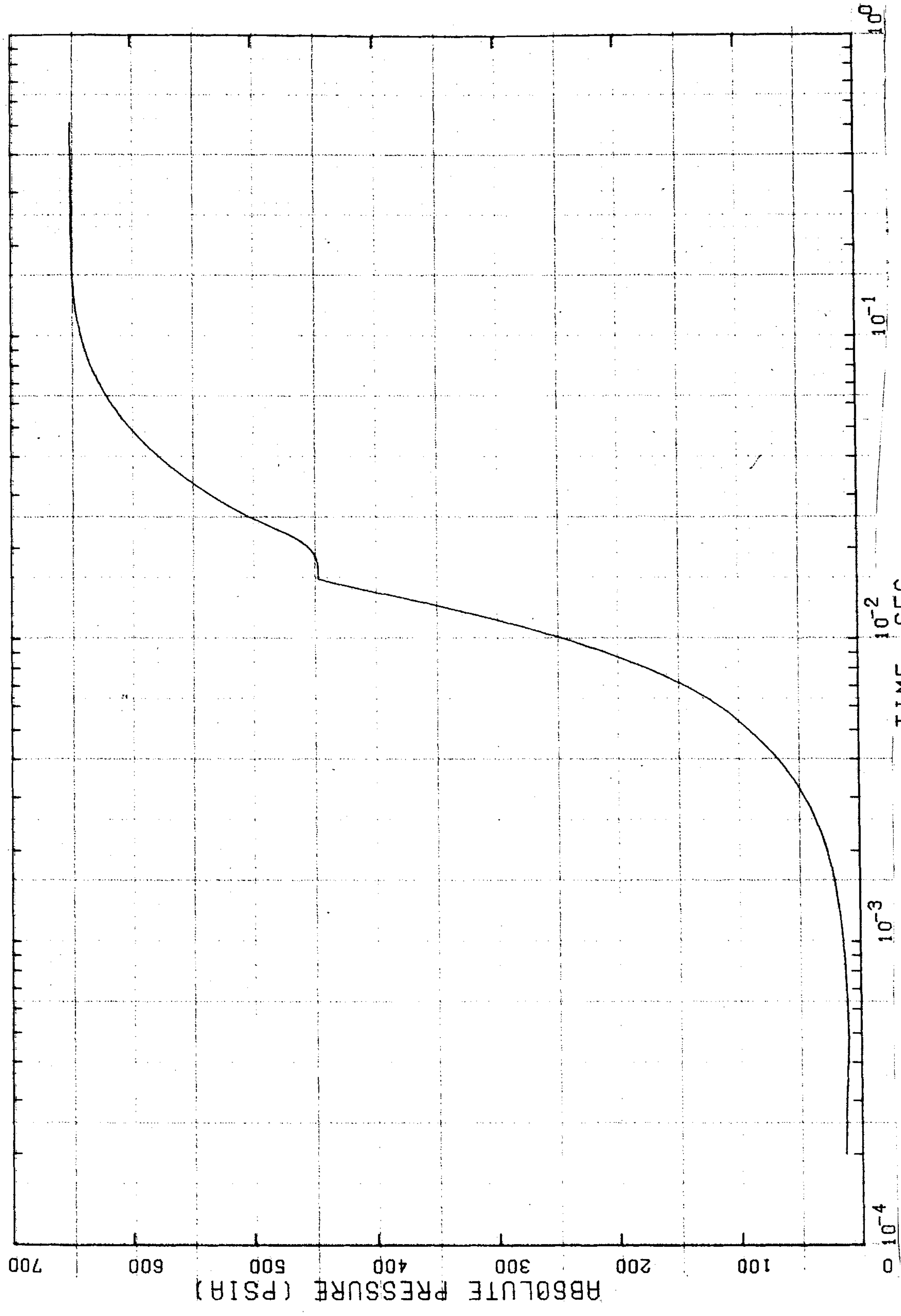

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FIGURE 6.2-23

PRESSURE RESPONSE
WITHIN FLOW DIVERTER
(RECIRCULATION OUTLET LINE BREAK)

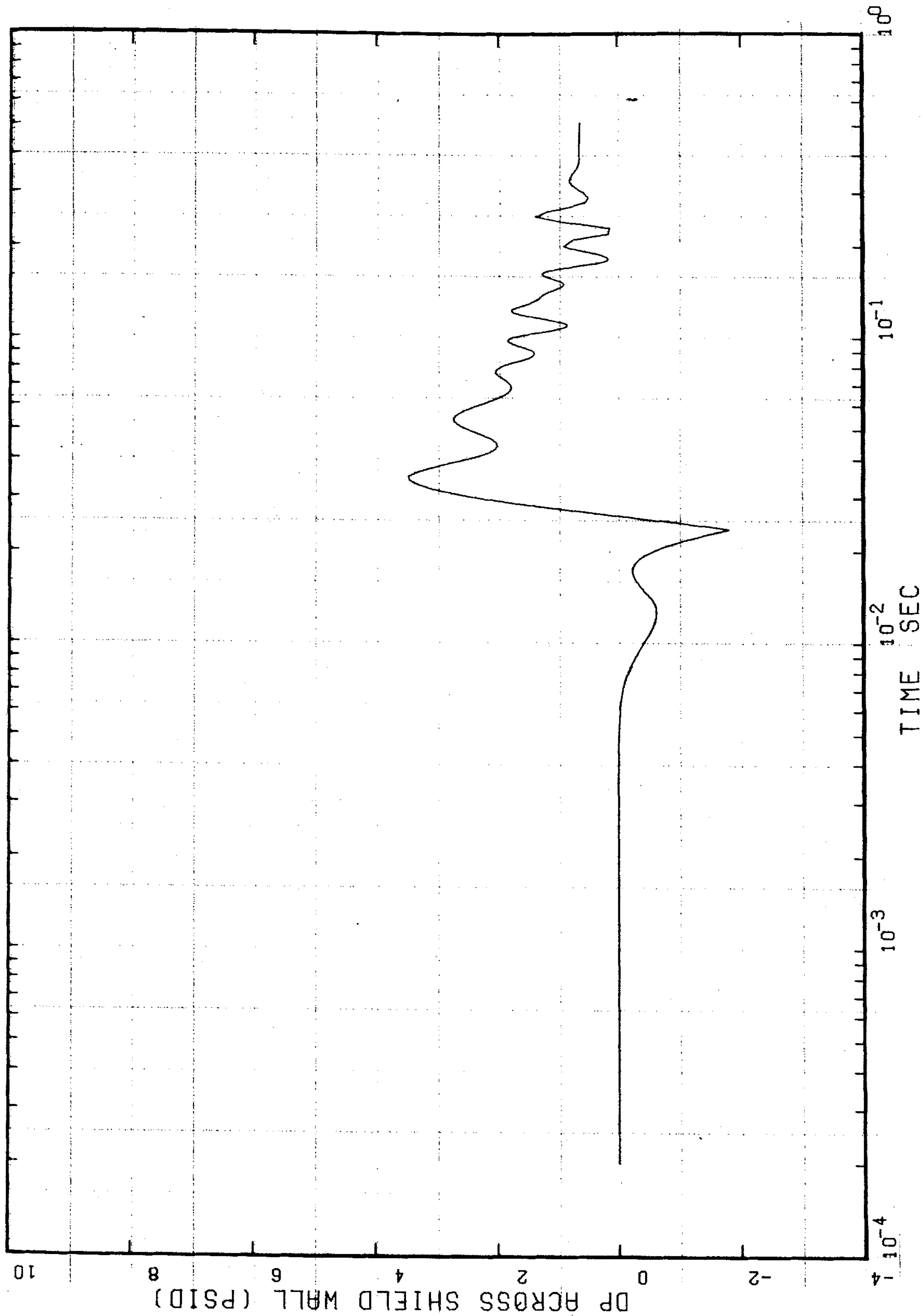

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FIGURE 6.2-24

ΔP VS. LOG T FOR NODE 1
(RECIRCULATION OUTLET LINE BREAK)

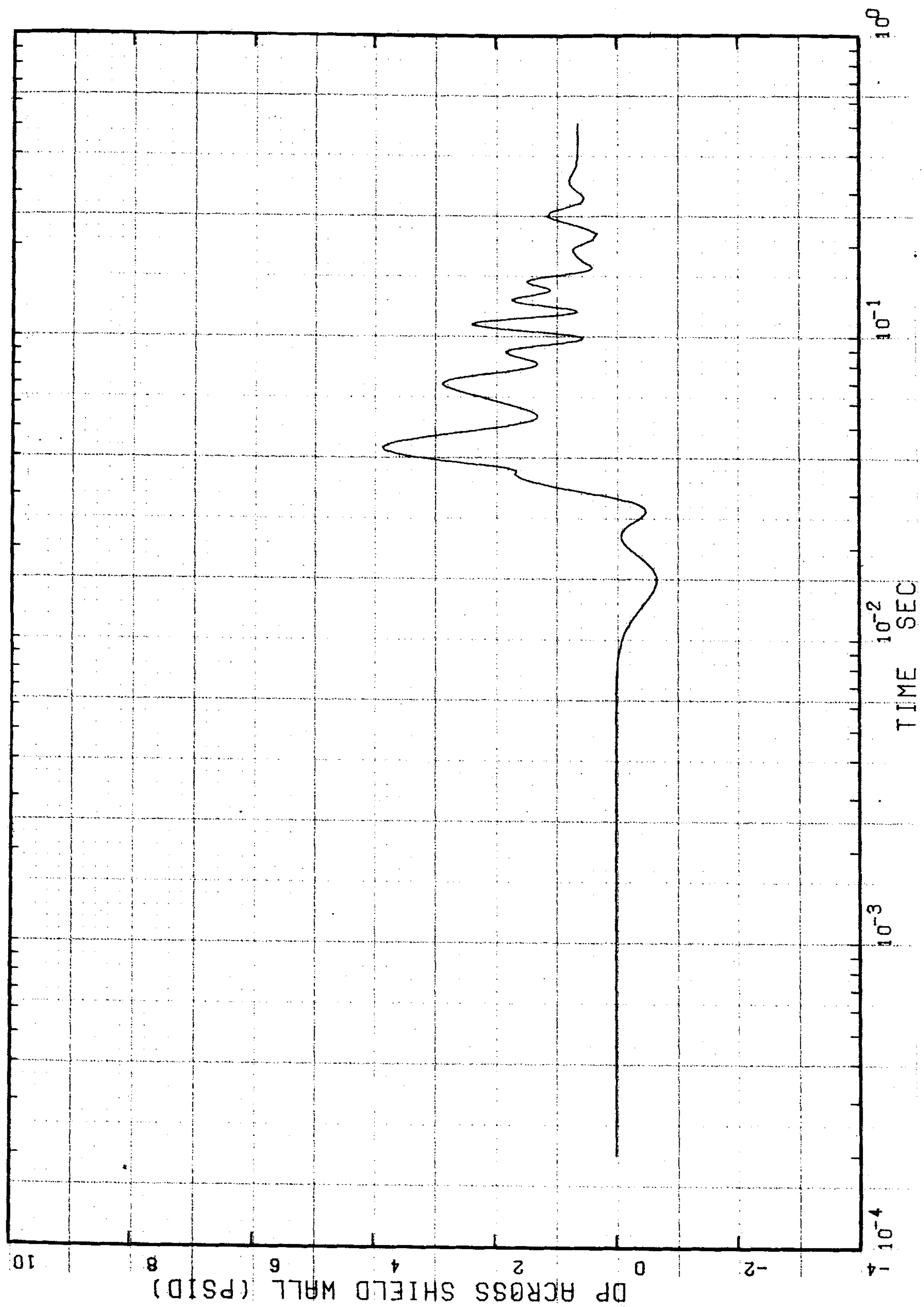

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FIGURE 6.2-25

ΔP VS. LOG T FOR NODE 2
(RECIRCULATION OUTLET LINE BREAK)

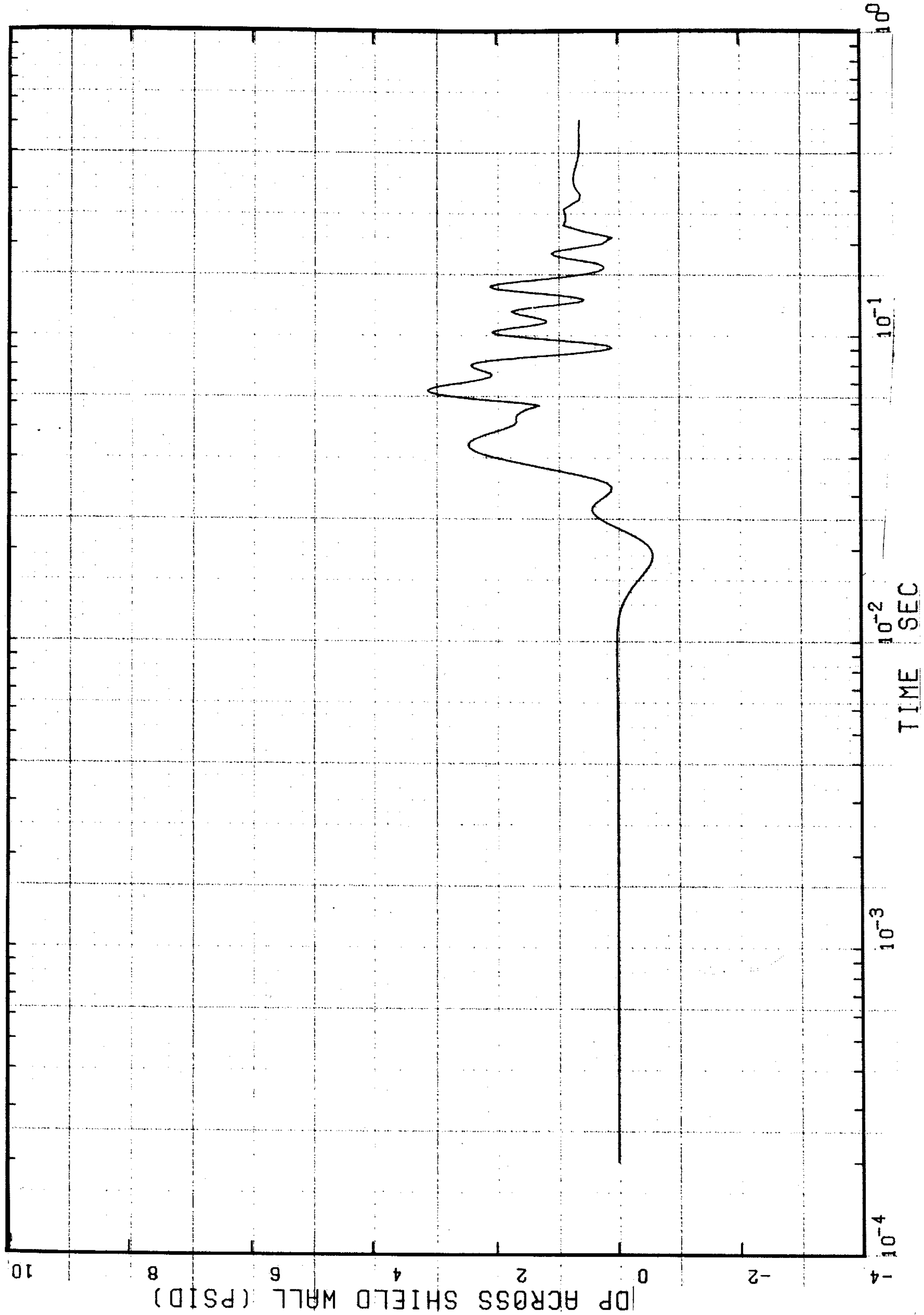

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FIGURE 6.2-26

ΔP VS. LOG T FOR NODE 3
(RECIRCULATION OUTLET LINE BREAK)

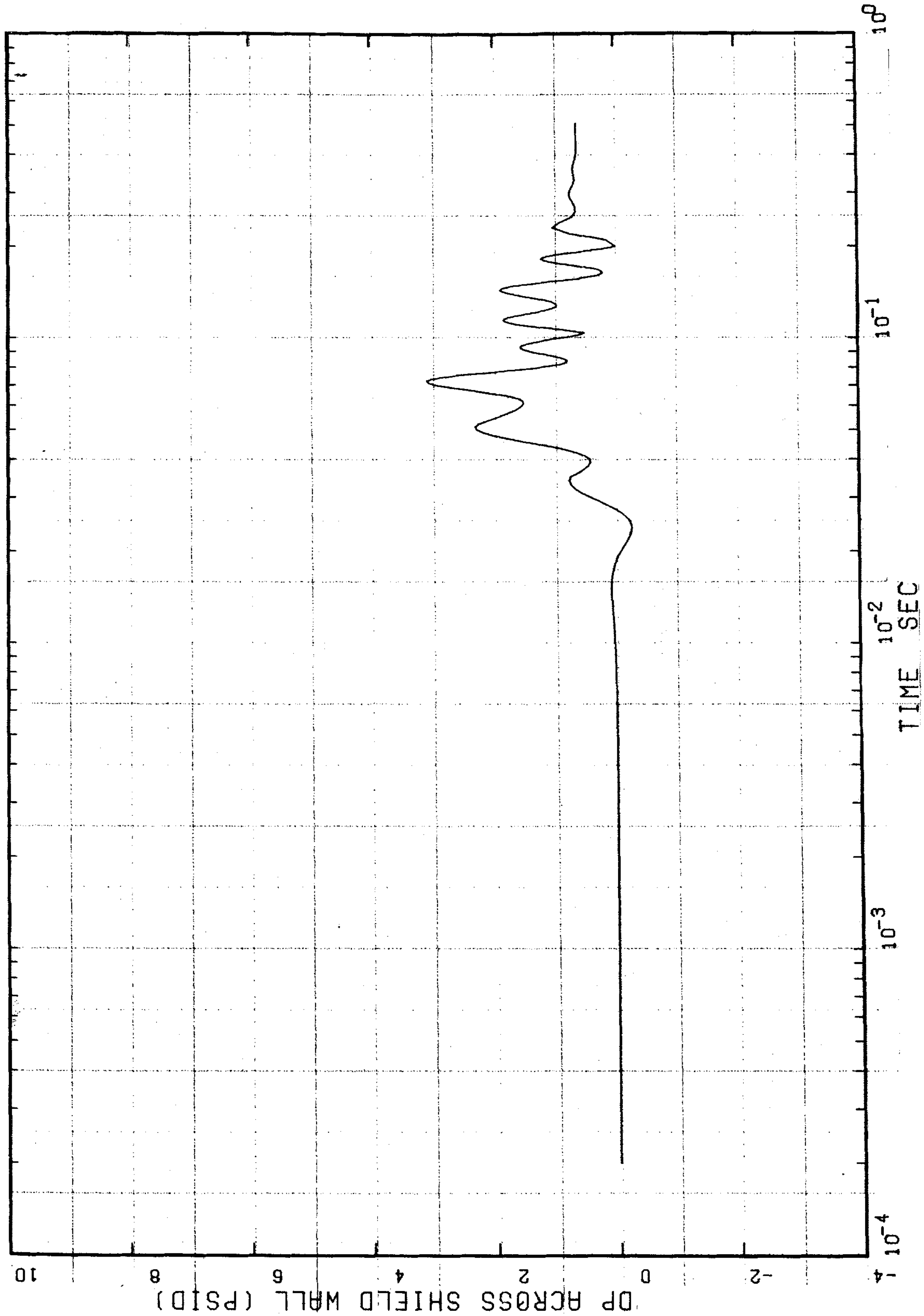

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FIGURE 6.2-27

ΔP VS. LOG T FOR NODE 4
(RECIRCULATION OUTLET LINE BREAK)

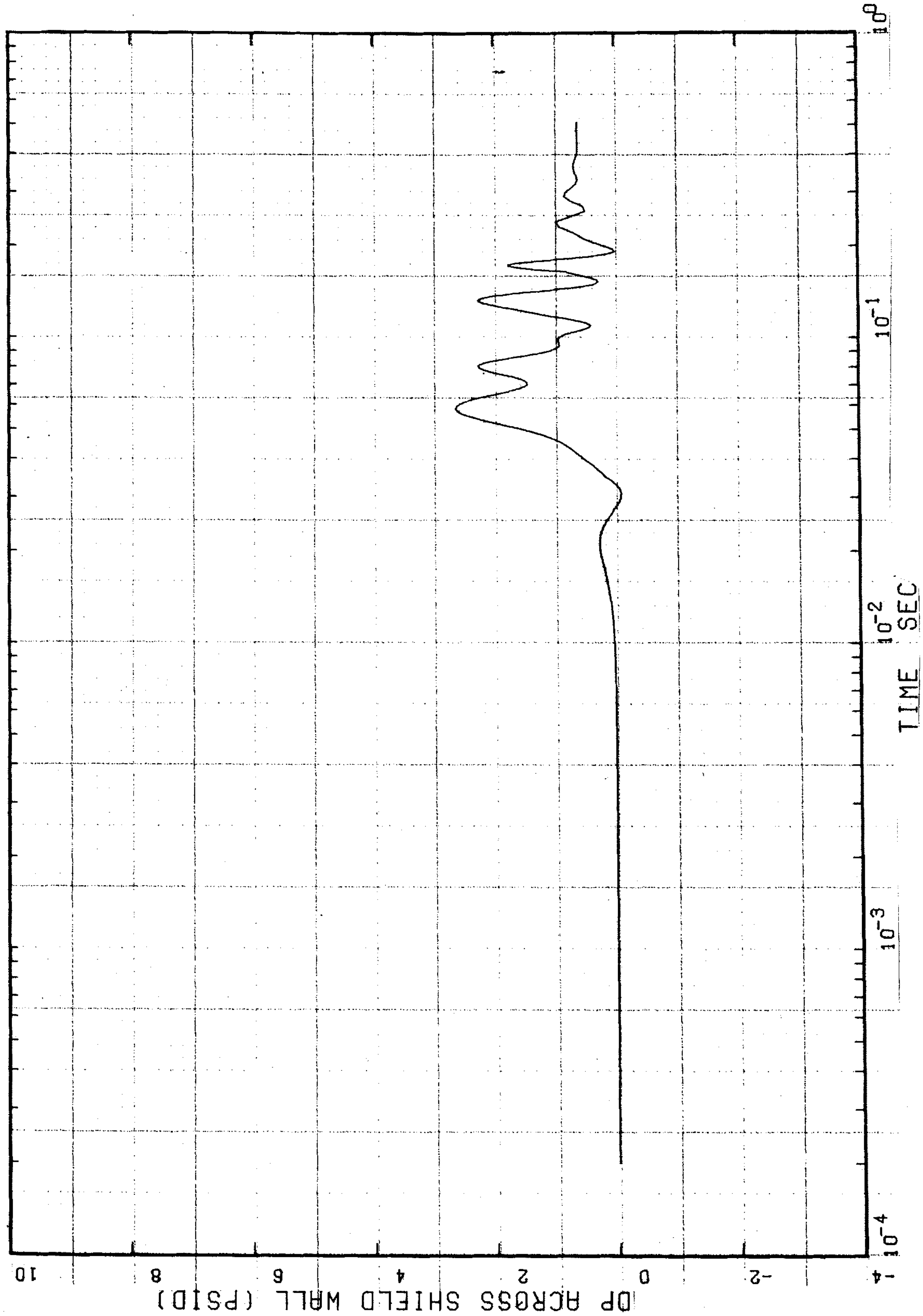

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FIGURE 6.2-28

ΔP VS. LOG T FOR NODE 5
(RECIRCULATION OUTLET LINE BREAK)

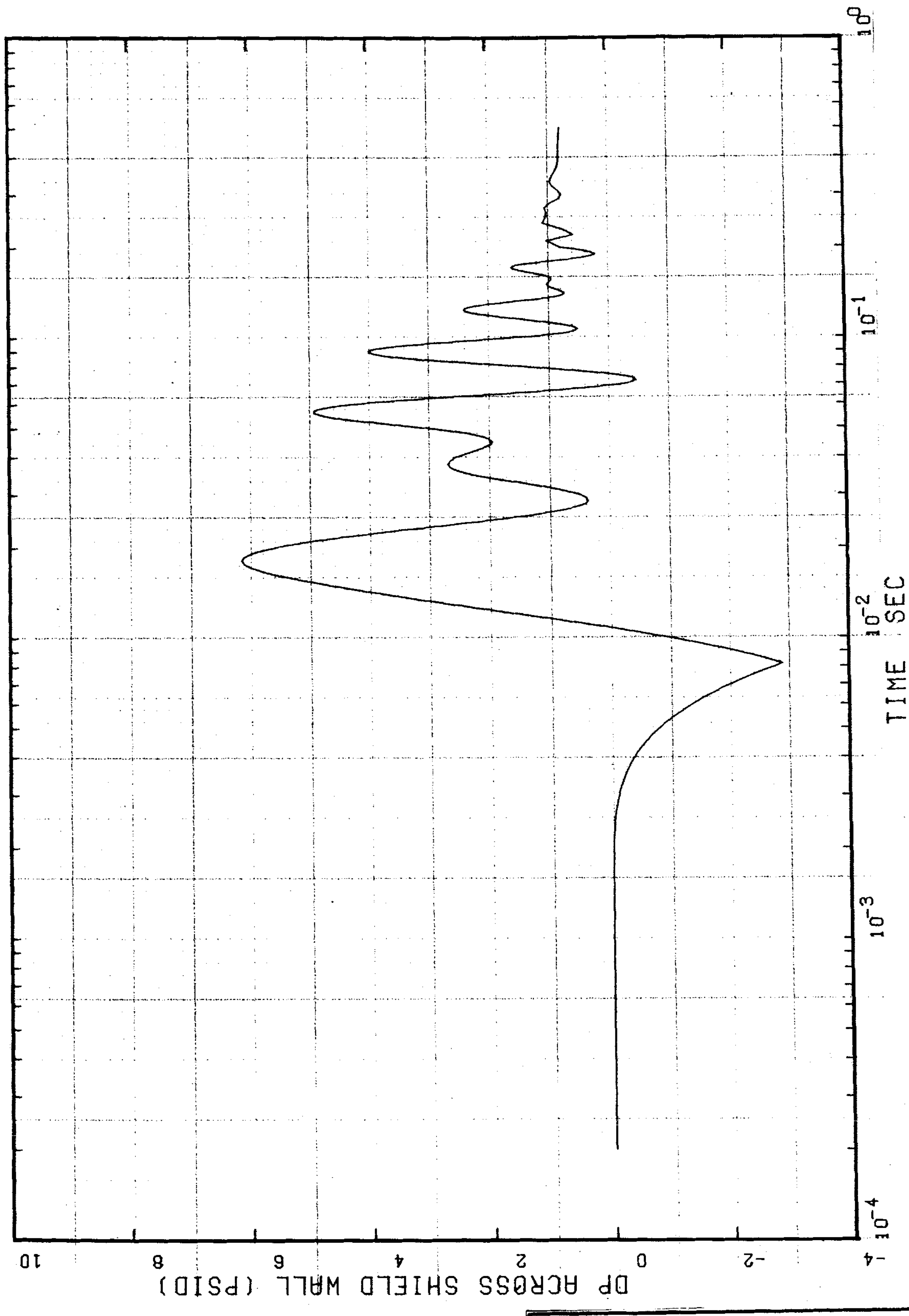

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FIGURE 6.2-29

ΔP VS. LOG T FOR NODE 6
(RECIRCULATION OUTLET LINE BREAK)

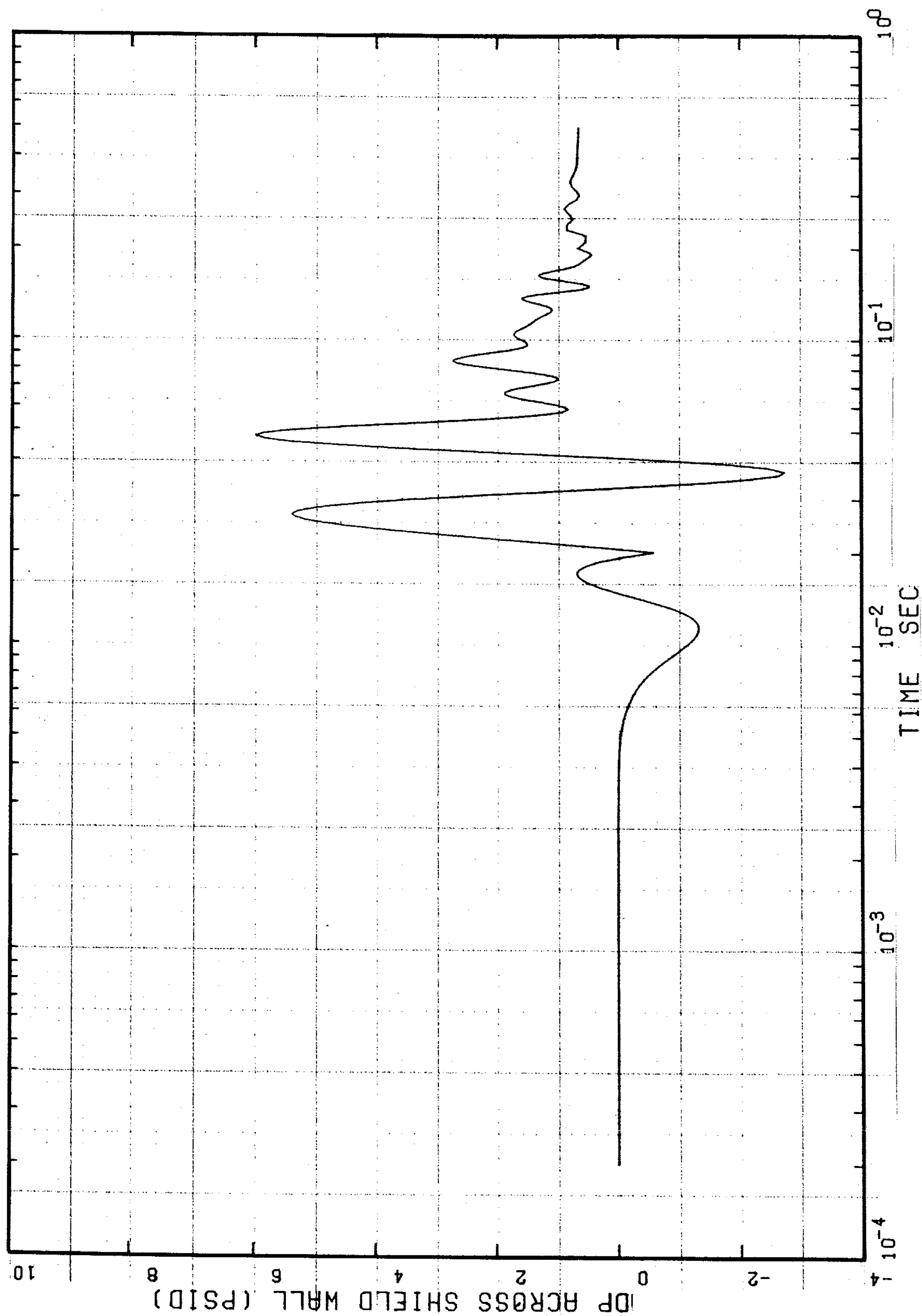

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FIGURE 6.2-30

ΔP VS. LOG T FOR NODE 7
(RECIRCULATION OUTLET LINE BREAK)

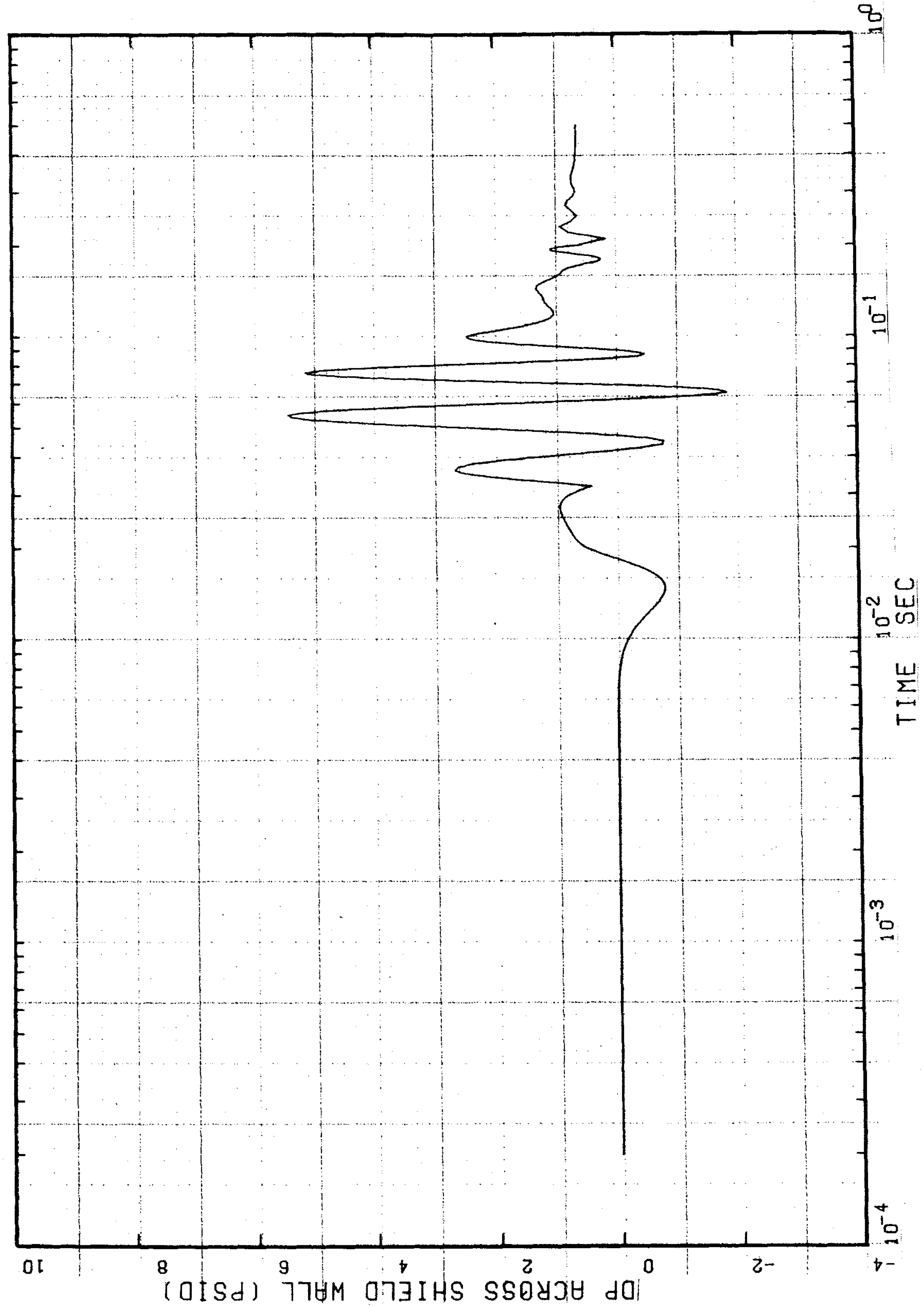

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FIGURE 6.2-31

ΔP VS. LOG T FOR NODE 8
(RECIRCULATION OUTLET LINE BREAK)

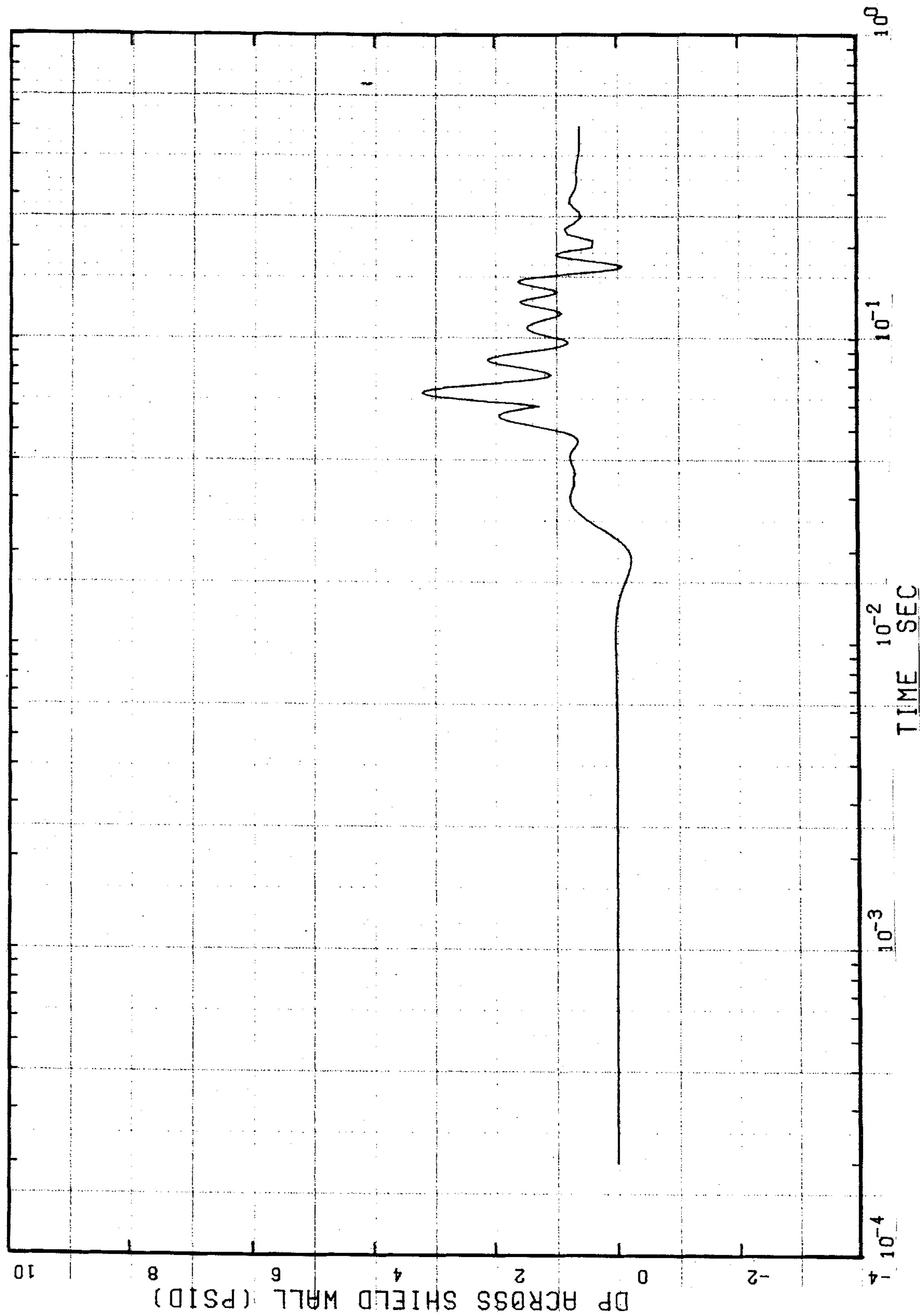

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FIGURE 6.2-32

ΔP VS. LOG T FOR NODE 9
(RECIRCULATION OUTLET LINE BREAK)

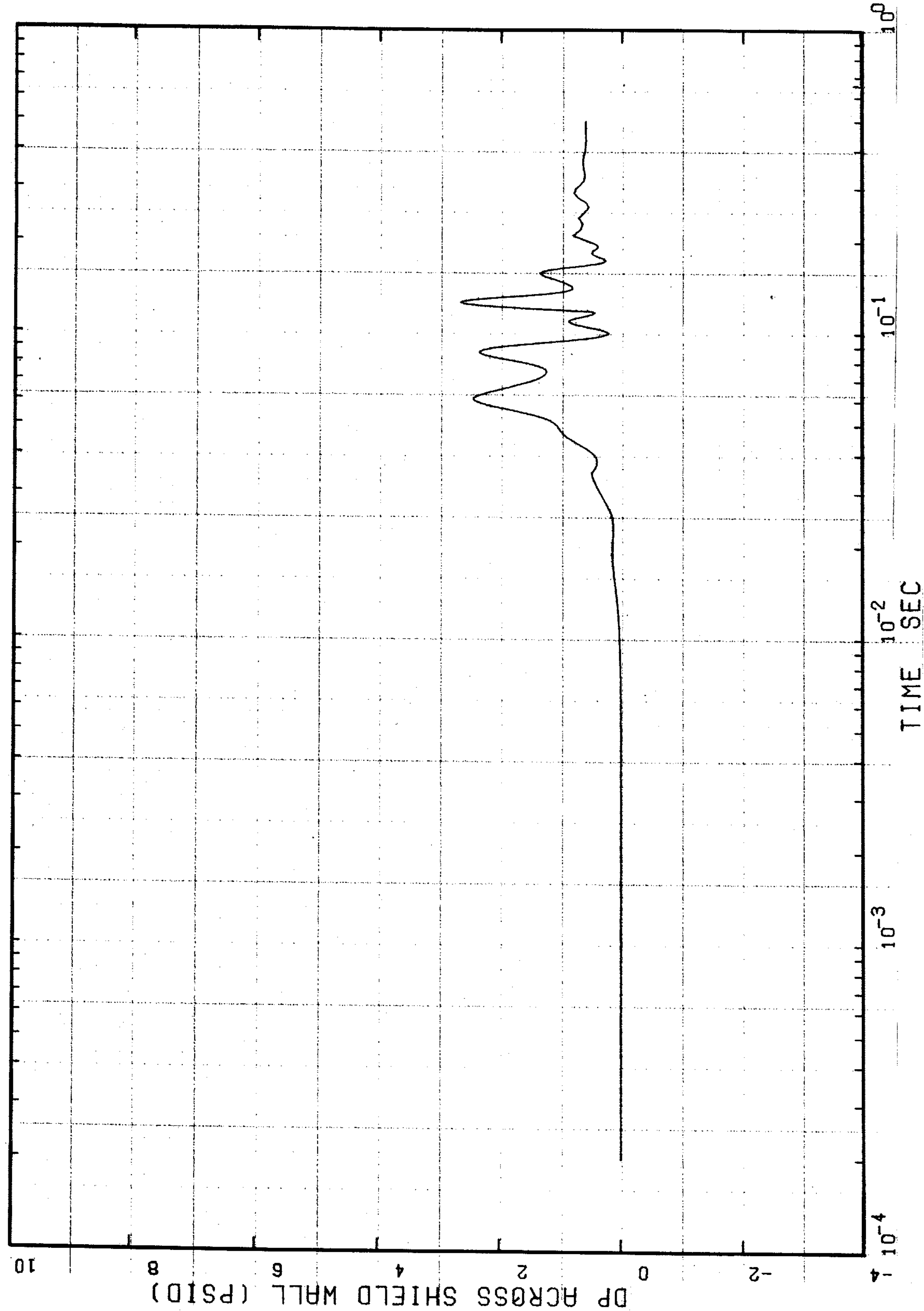

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FIGURE 6.2-33

ΔP VS. LOG T FOR NODE 10
(RECIRCULATION OUTLET LINE BREAK)

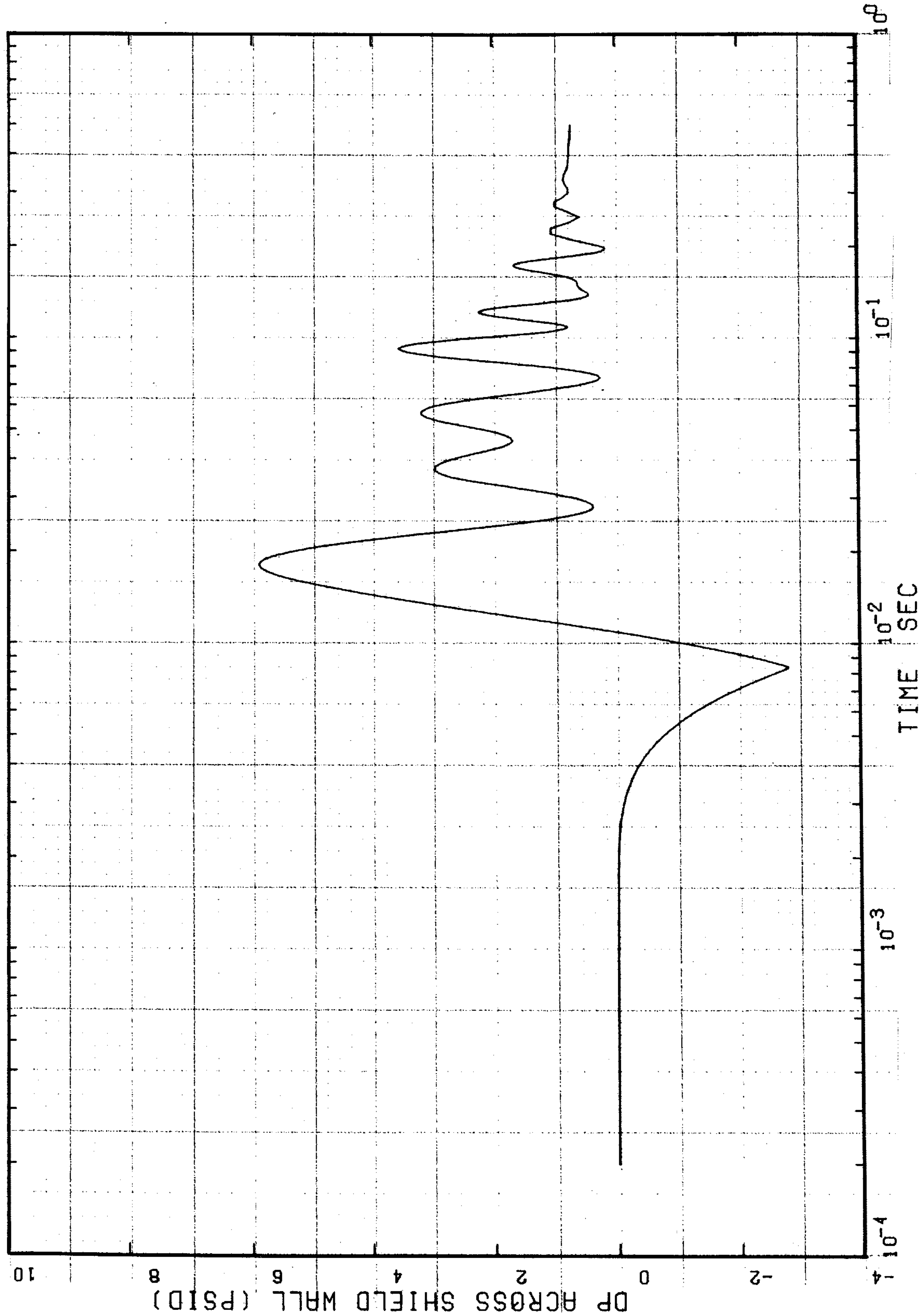

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FIGURE 6.2-34

ΔP VS. LOG T FOR NODE 11
(RECIRCULATION OUTLET LINE BREAK)

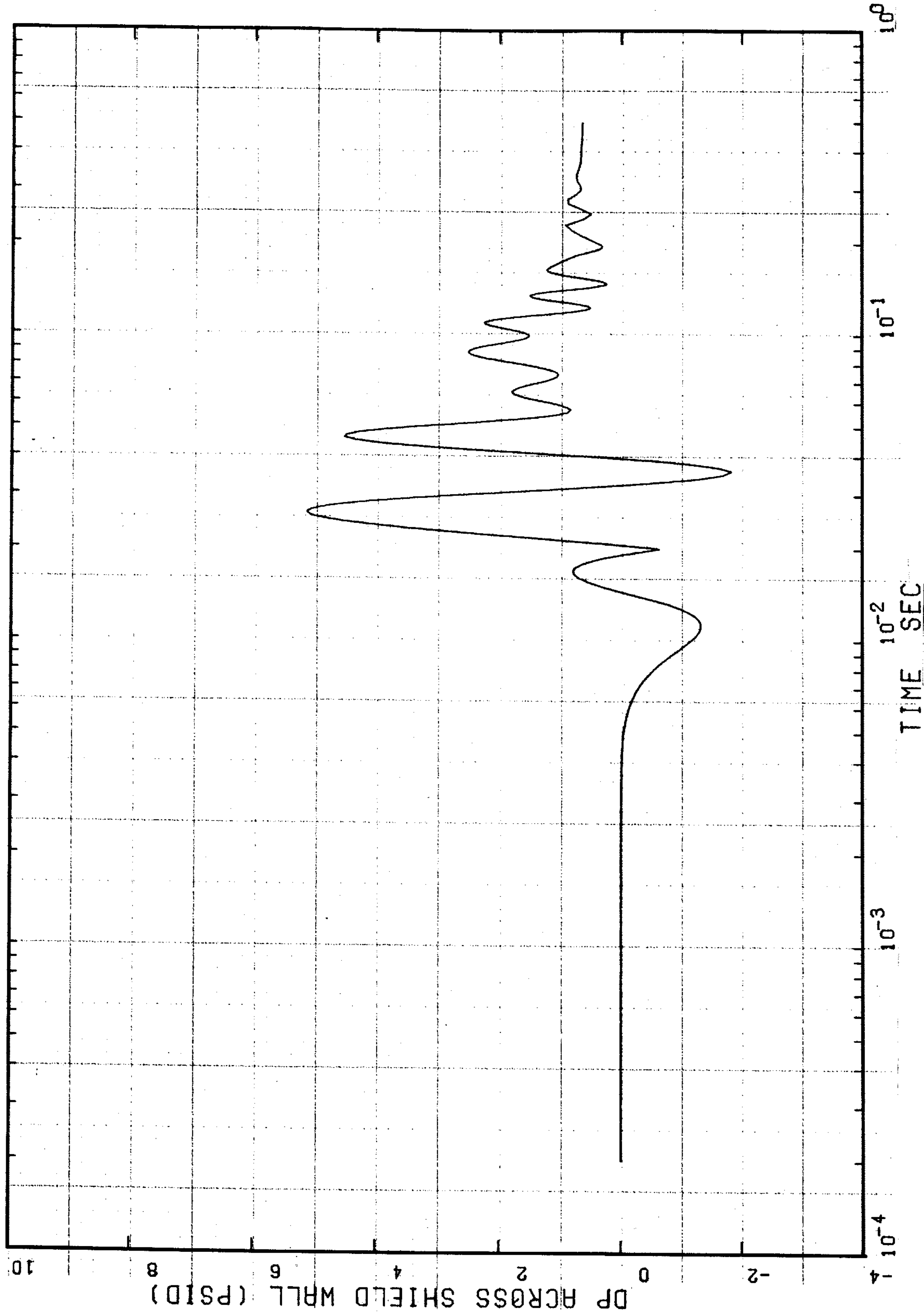

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FIGURE 6.2-35

ΔP VS. LOG T FOR NODE 12
(RECIRCULATION OUTLET LINE BREAK)

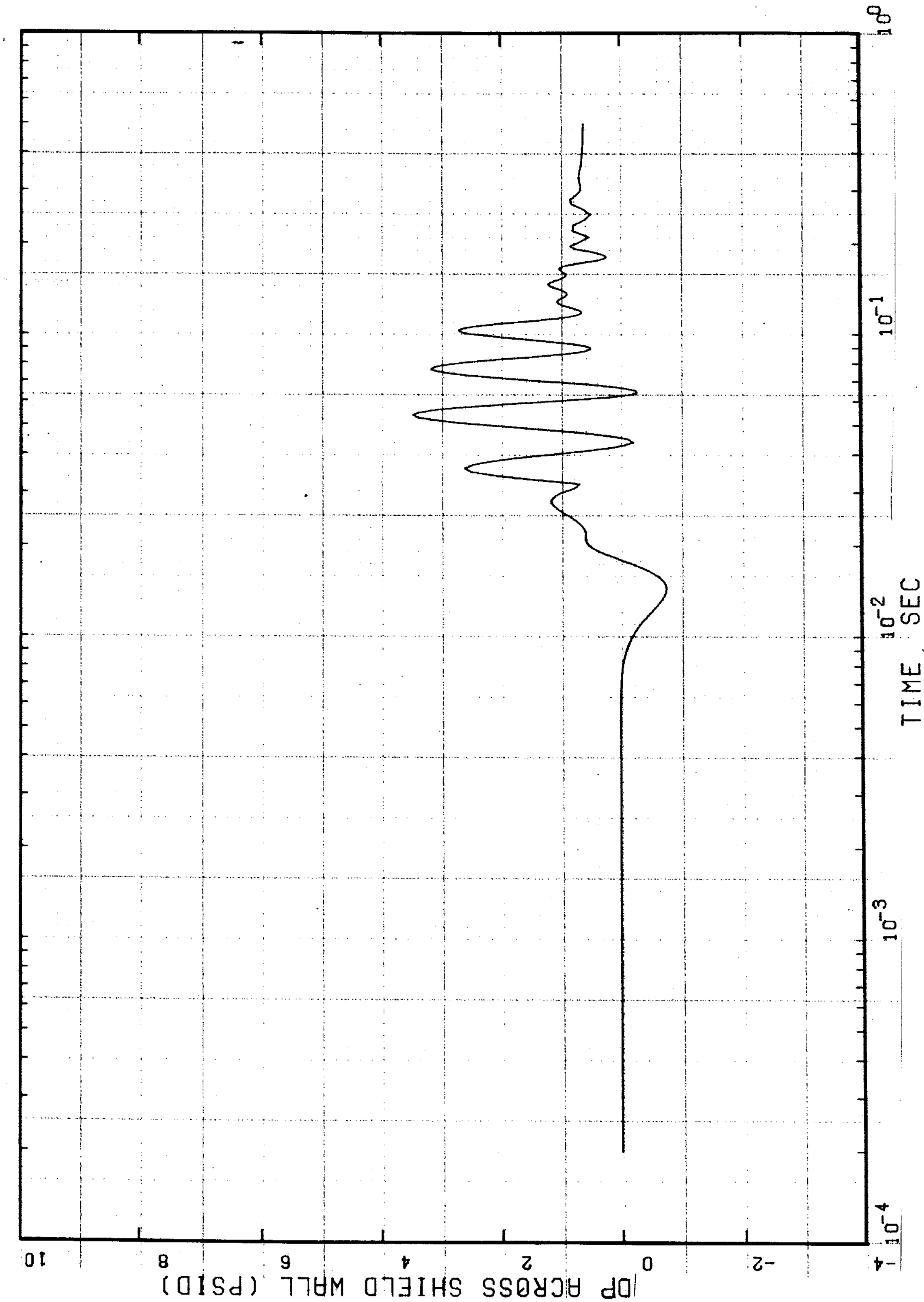

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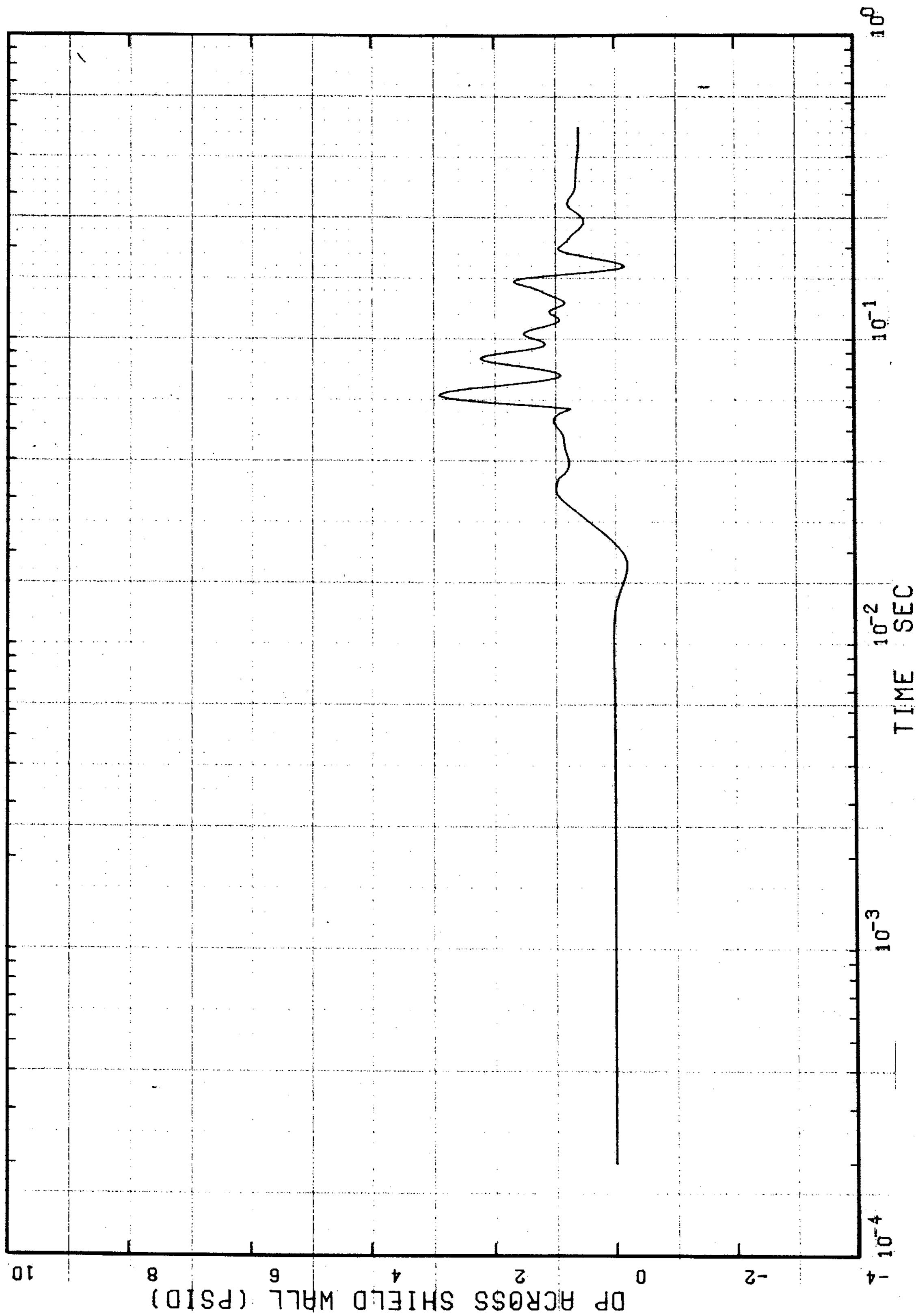
FIGURE 6.2-36

ΔP VS. LOG T FOR NODE 13
(RECIRCULATION OUTLET LINE BREAK)



CLINTON POWER STATION
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FIGURE 6.2-37

ΔP VS. LOG T FOR NODE 14
(RECIRCULATION OUTLET LINE BREAK)

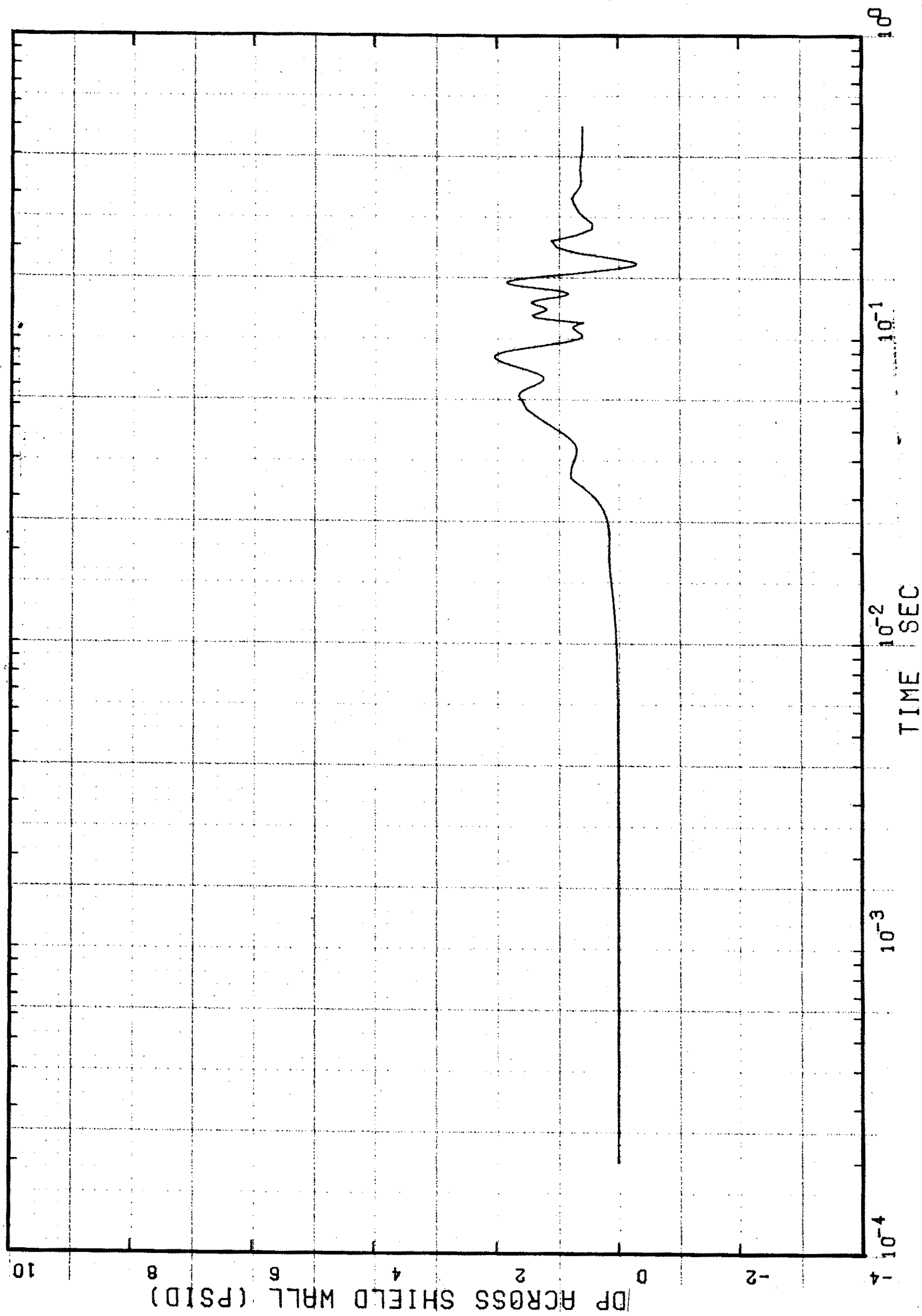

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FIGURE 6.2-38

ΔP VS. LOG T FOR NODE 15
(RECIRCULATION OUTLET LINE BREAK)

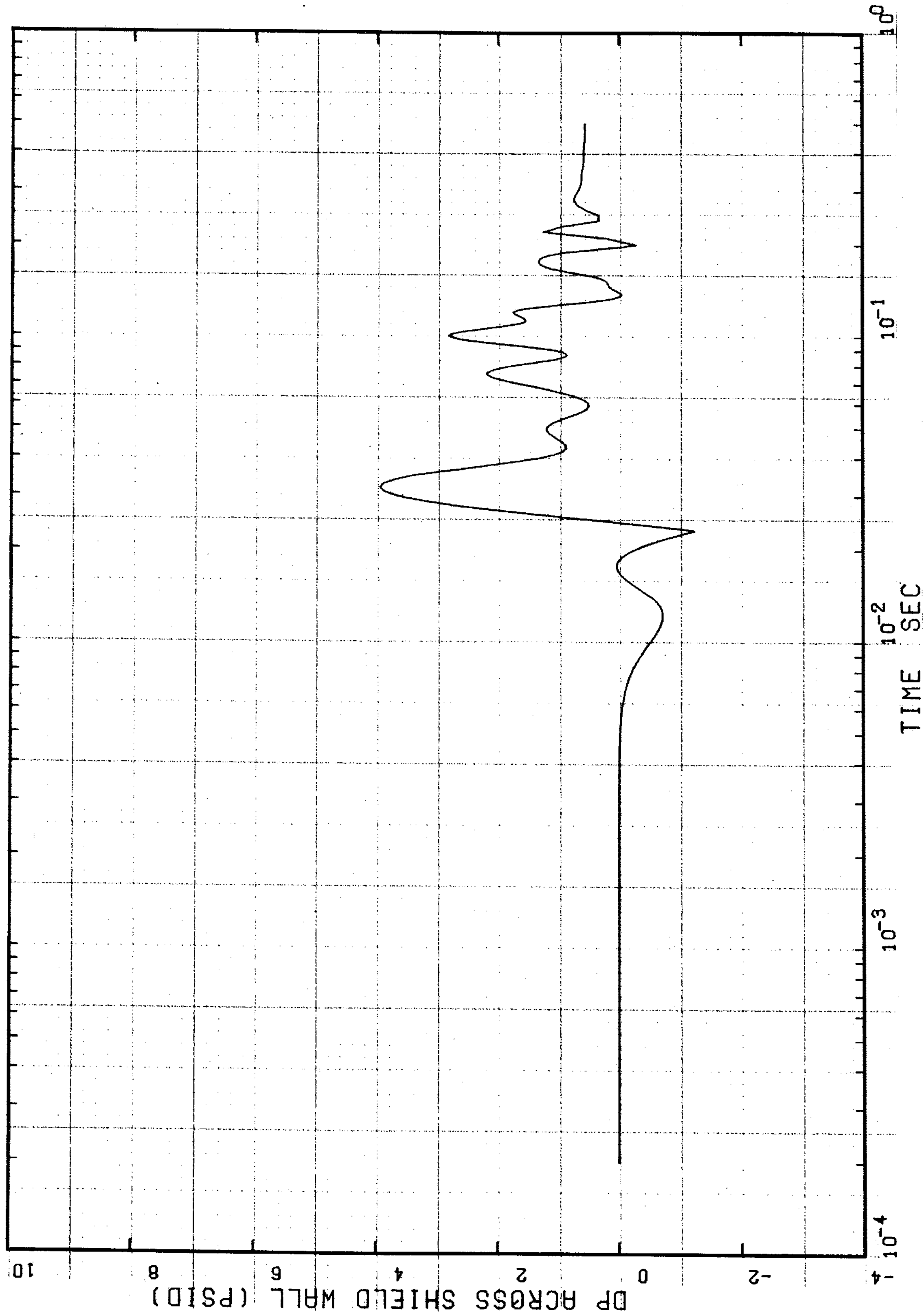

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FIGURE 6.2-39

ΔP VS. LOG T FOR NODE 16
(RECIRCULATION OUTLET LINE BREAK)

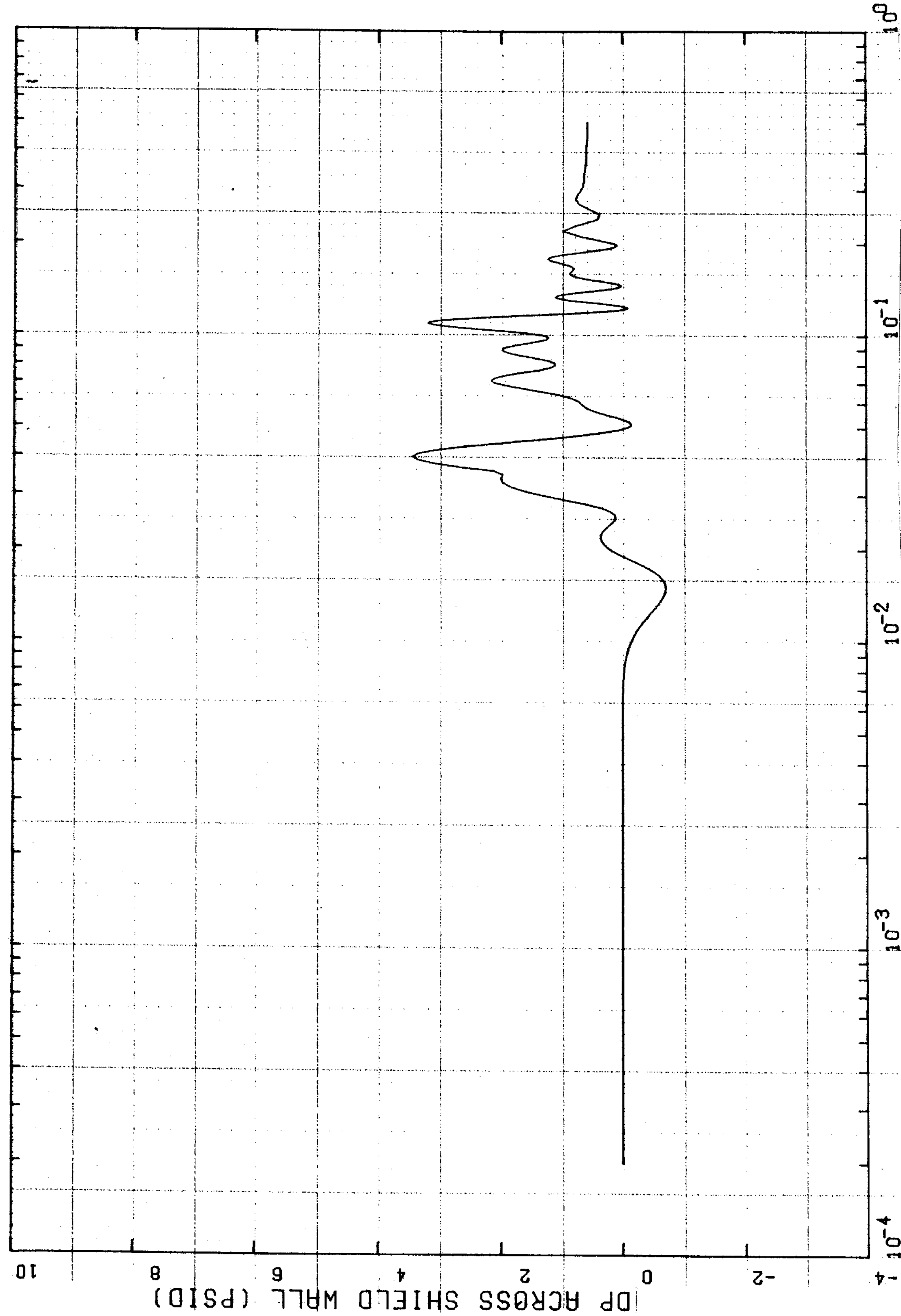

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FIGURE 6.2-40

ΔP VS. LOG T FOR NODE 17
(RECIRCULATION OUTLET LINE BREAK)

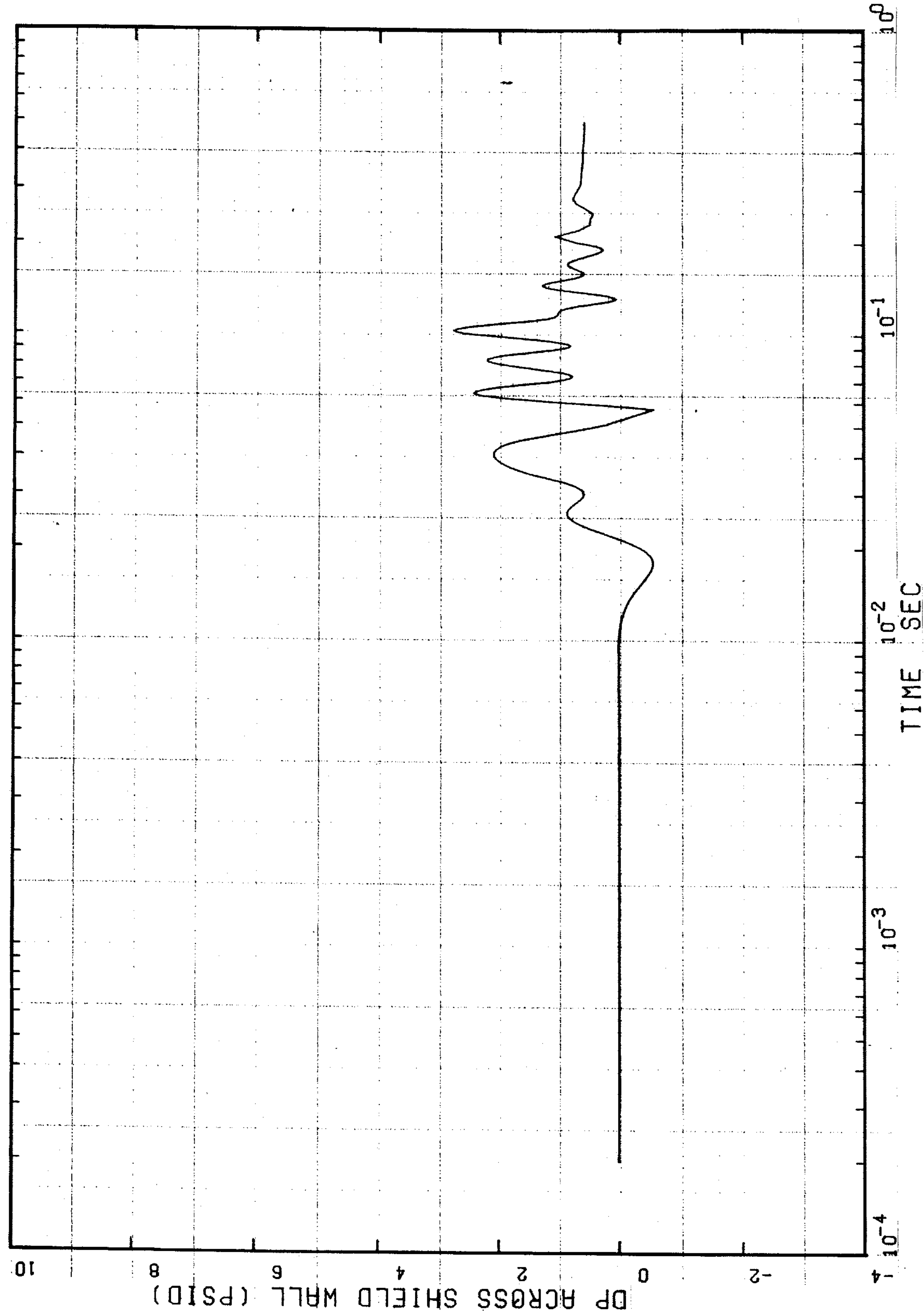

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FIGURE 6.2-41

ΔP VS. LOG T FOR NODE 18
(RECIRCULATION OUTLET LINE BREAK)

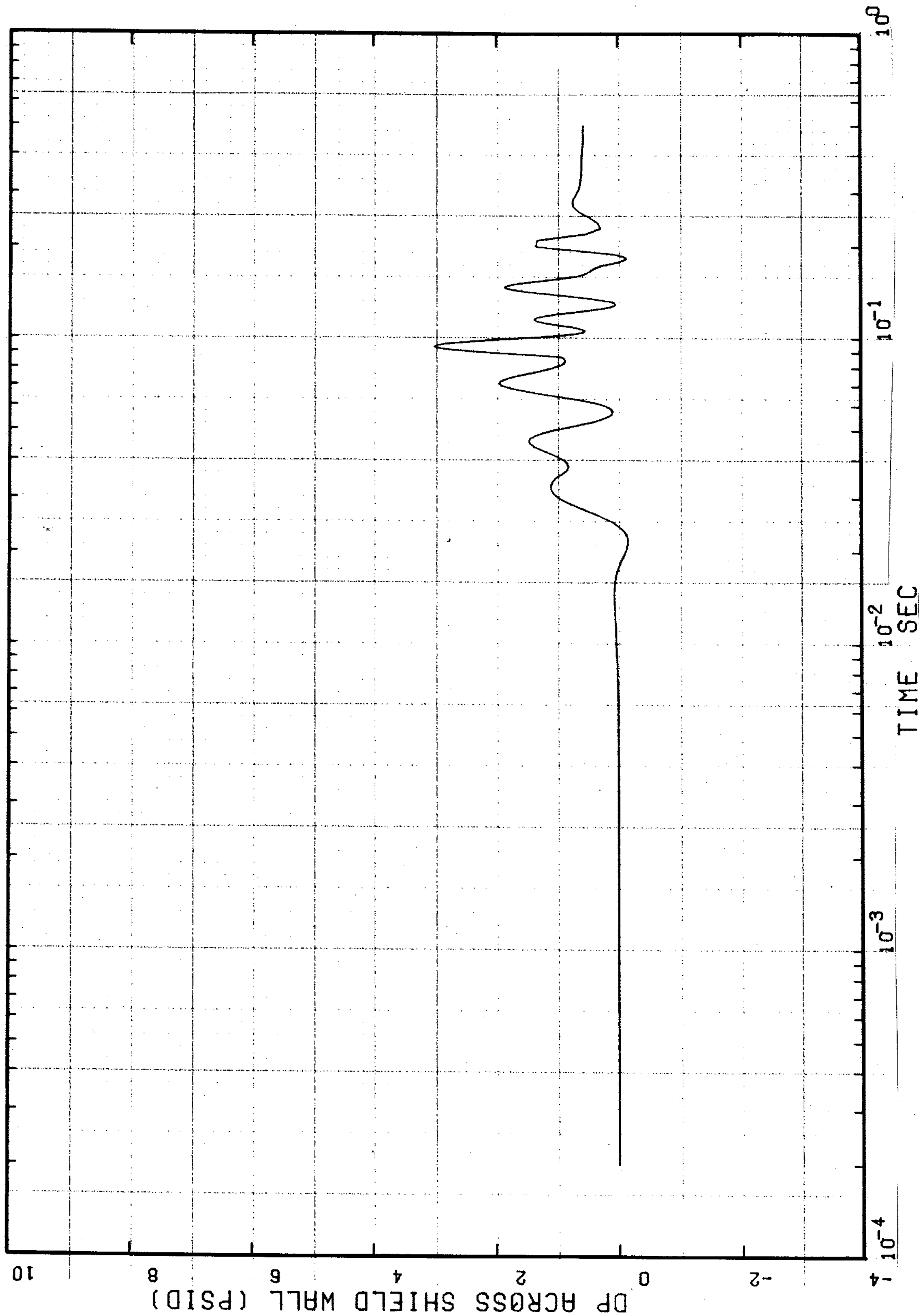

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FIGURE 6.2-42

ΔP VS. LOG T FOR NODE 19
(RECIRCULATION OUTLET LINE BREAK)

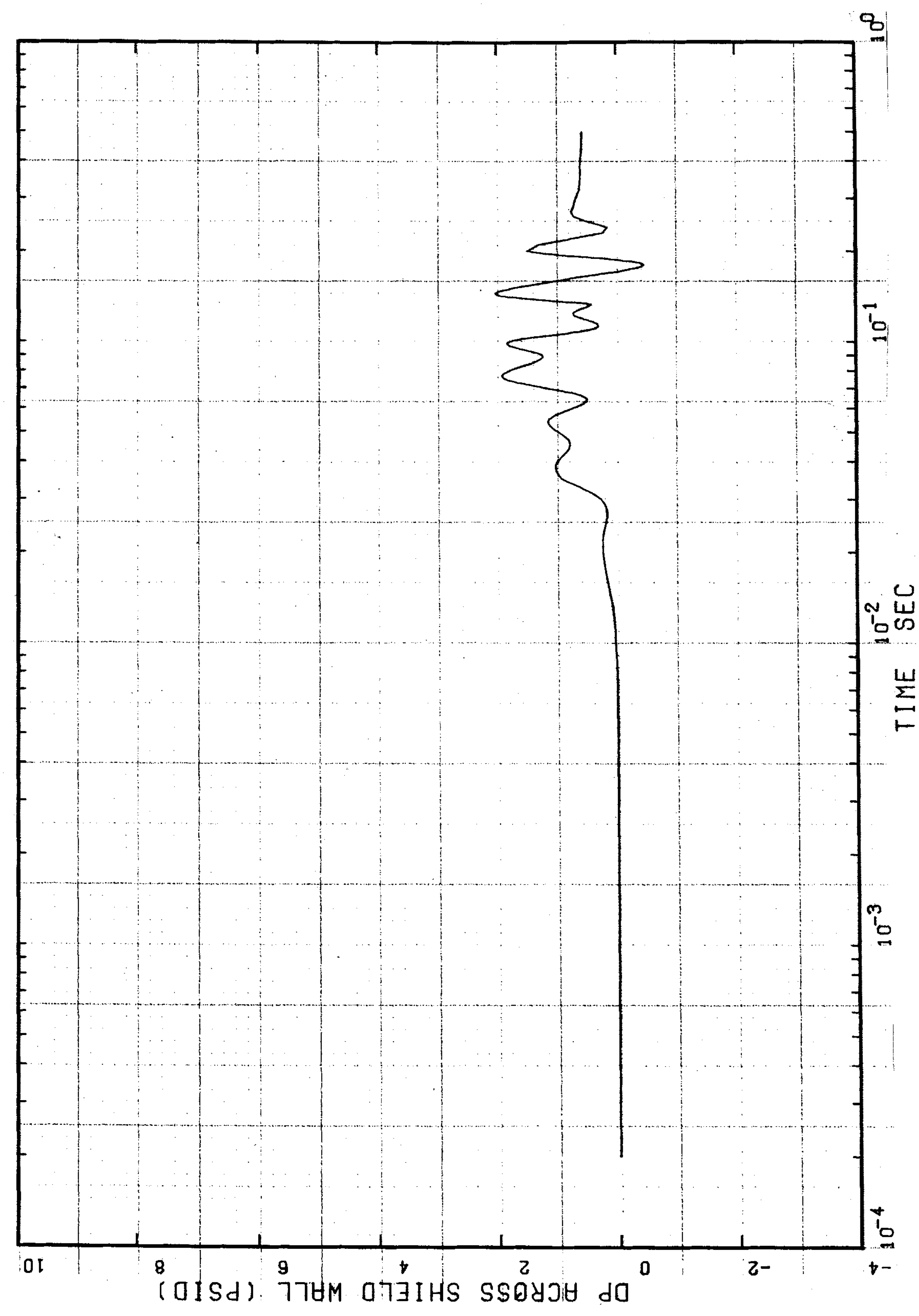

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FIGURE 6.2-43

ΔP VS. LOG T FOR NODE 20
(RECIRCULATION OUTLET LINE BREAK)

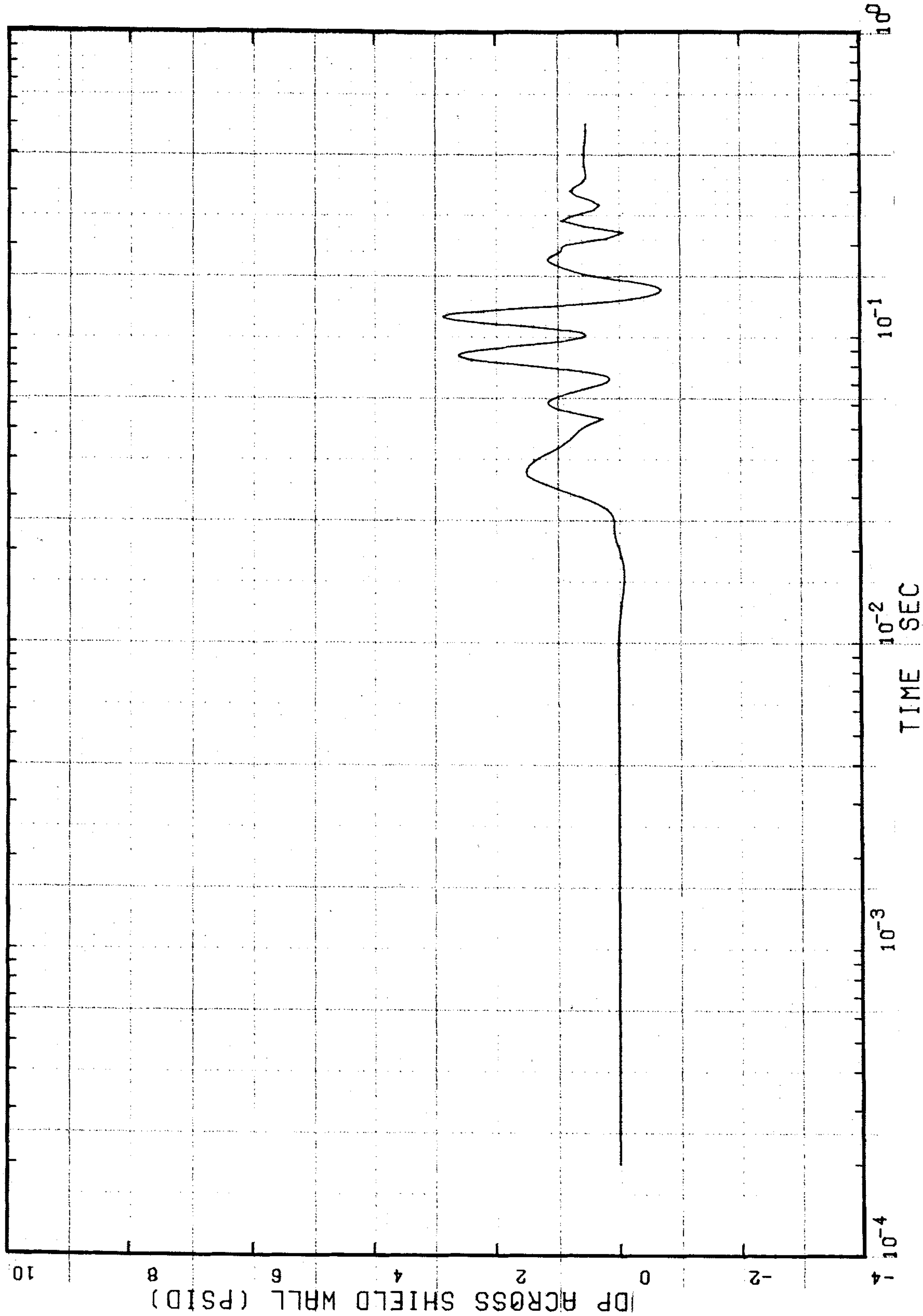

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FIGURE 6.2-44

ΔP VS. LOG T FOR NODE 21
(RECIRCULATION OUTLET LINE BREAK)

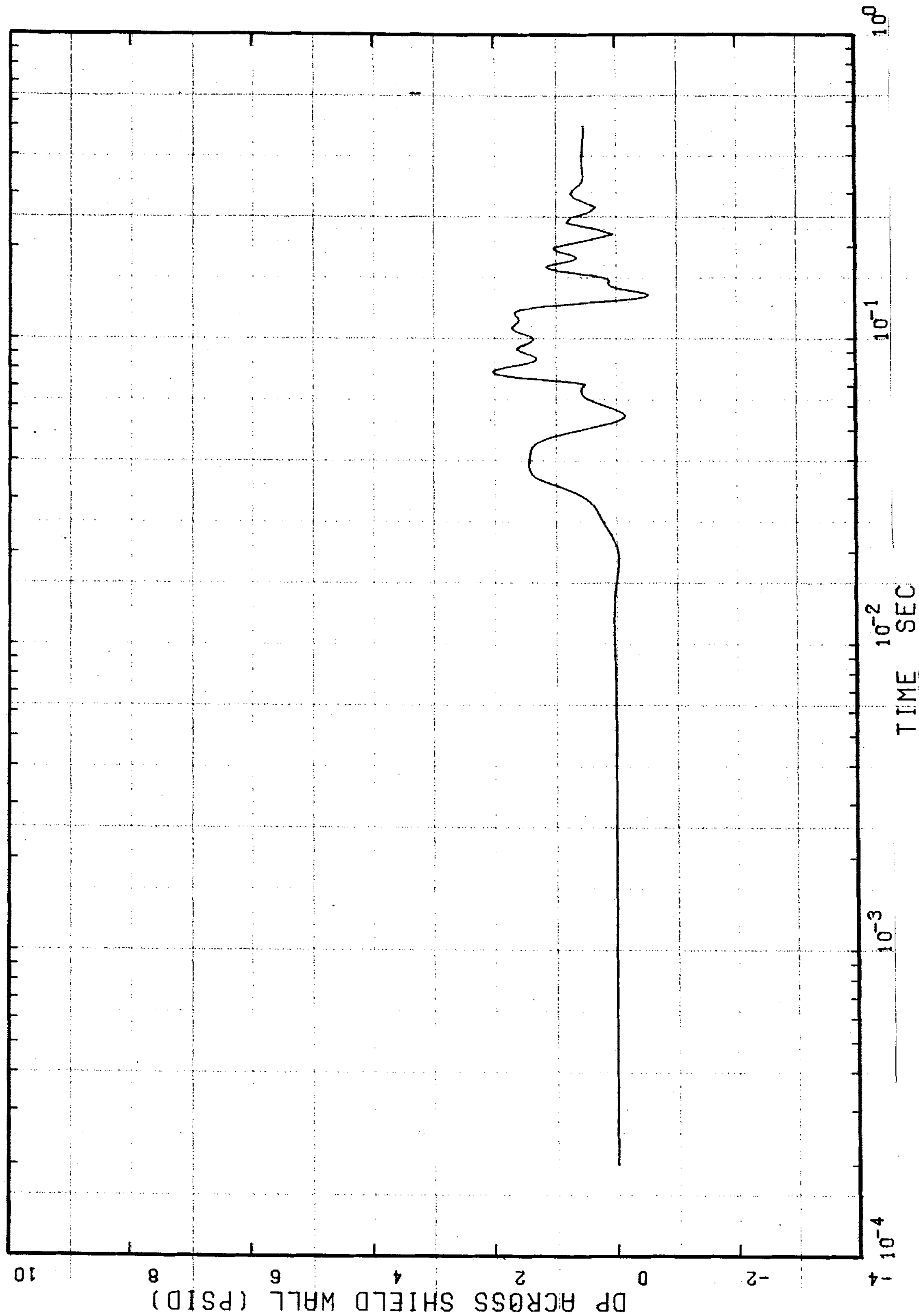

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FIGURE 6.2-45

ΔP VS. LOG T FOR NODE 22
(RECIRCULATION OUTLET LINE BREAK)

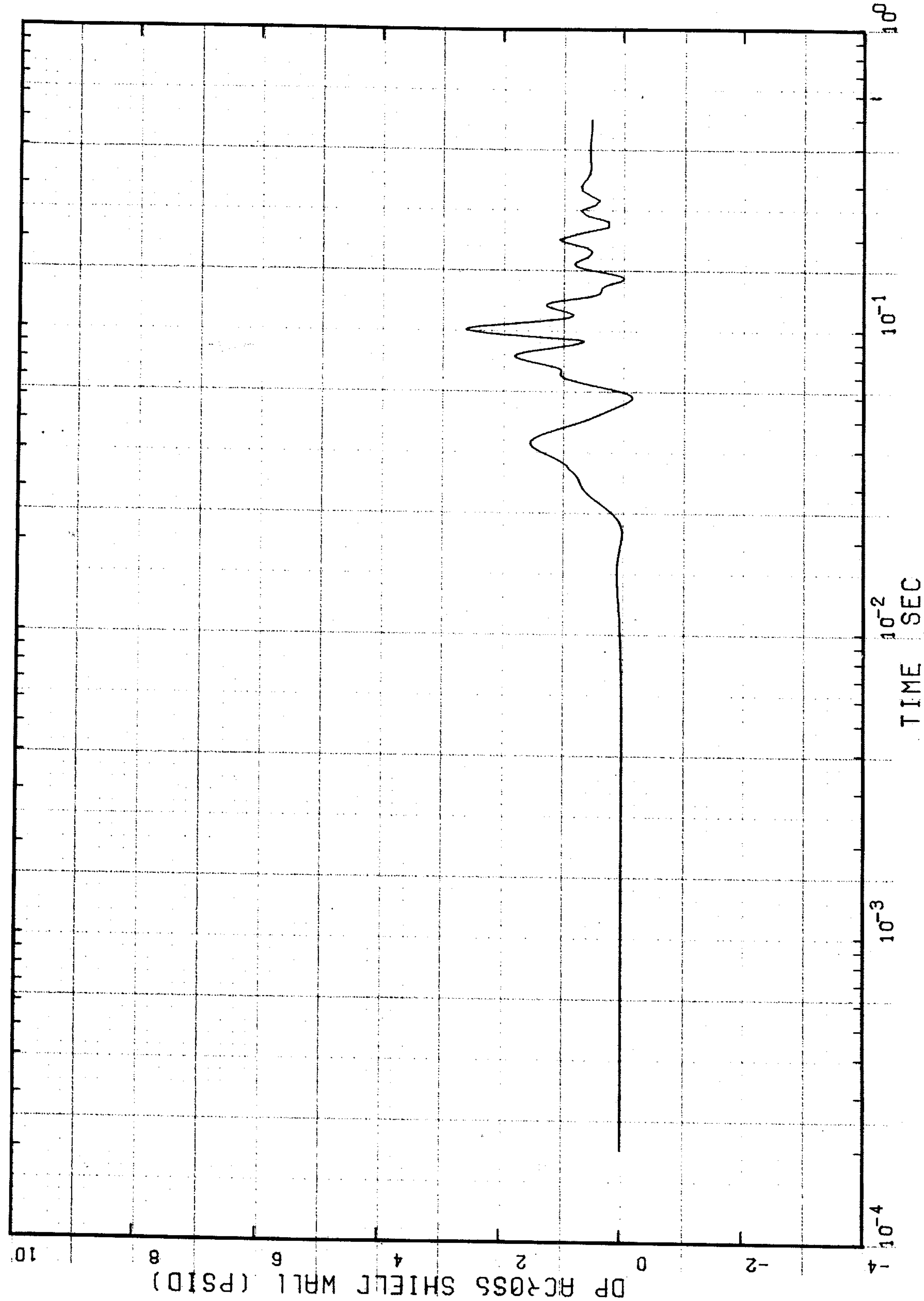

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FIGURE 6.2-46

ΔP VS. LOG T FOR NODE 23
(RECIRCULATION OUTLET LINE BREAK)

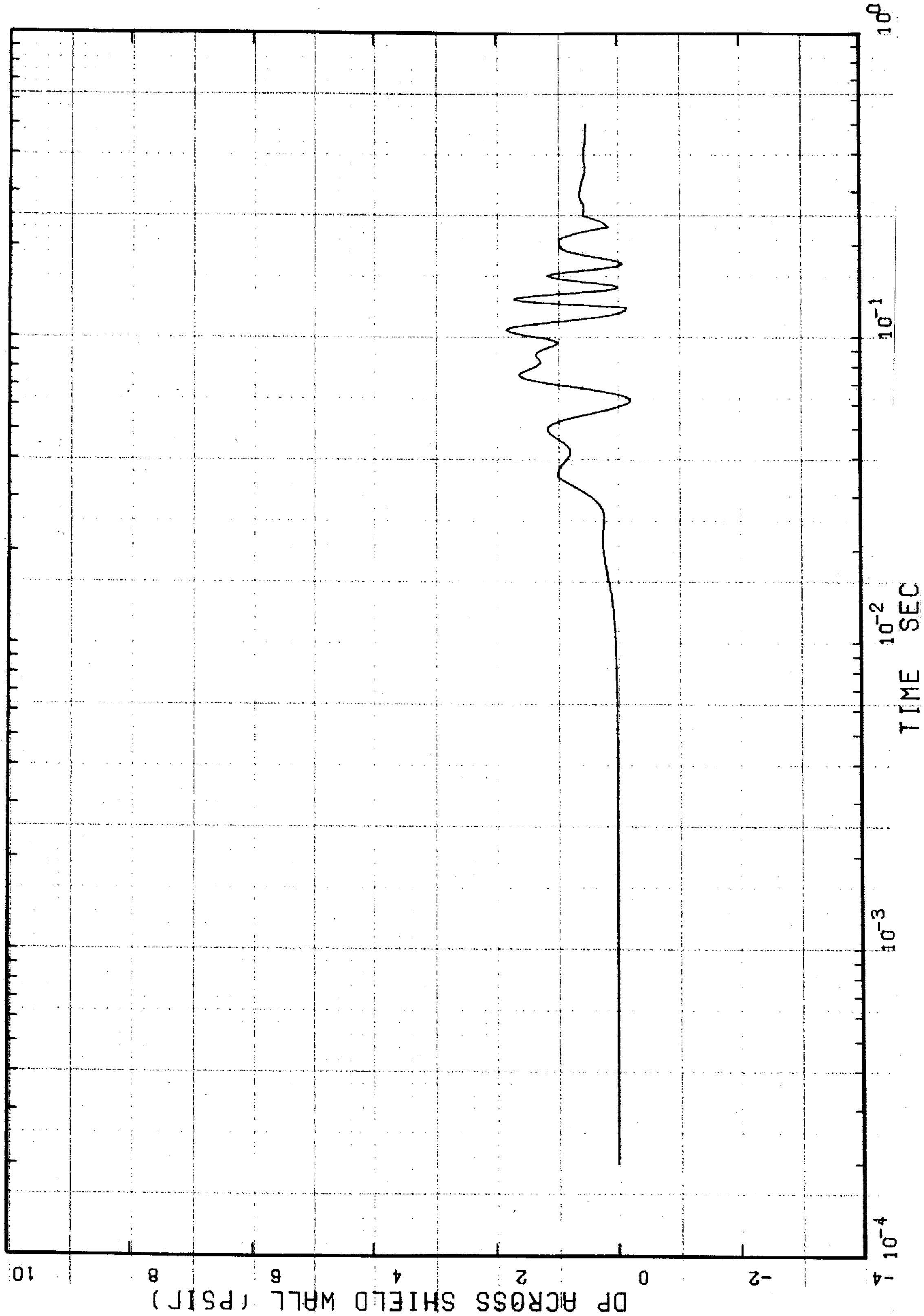

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FIGURE 6.2-47

ΔP VS. LOG T FOR NODE 24
(RECIRCULATION OUTLET LINE BREAK)

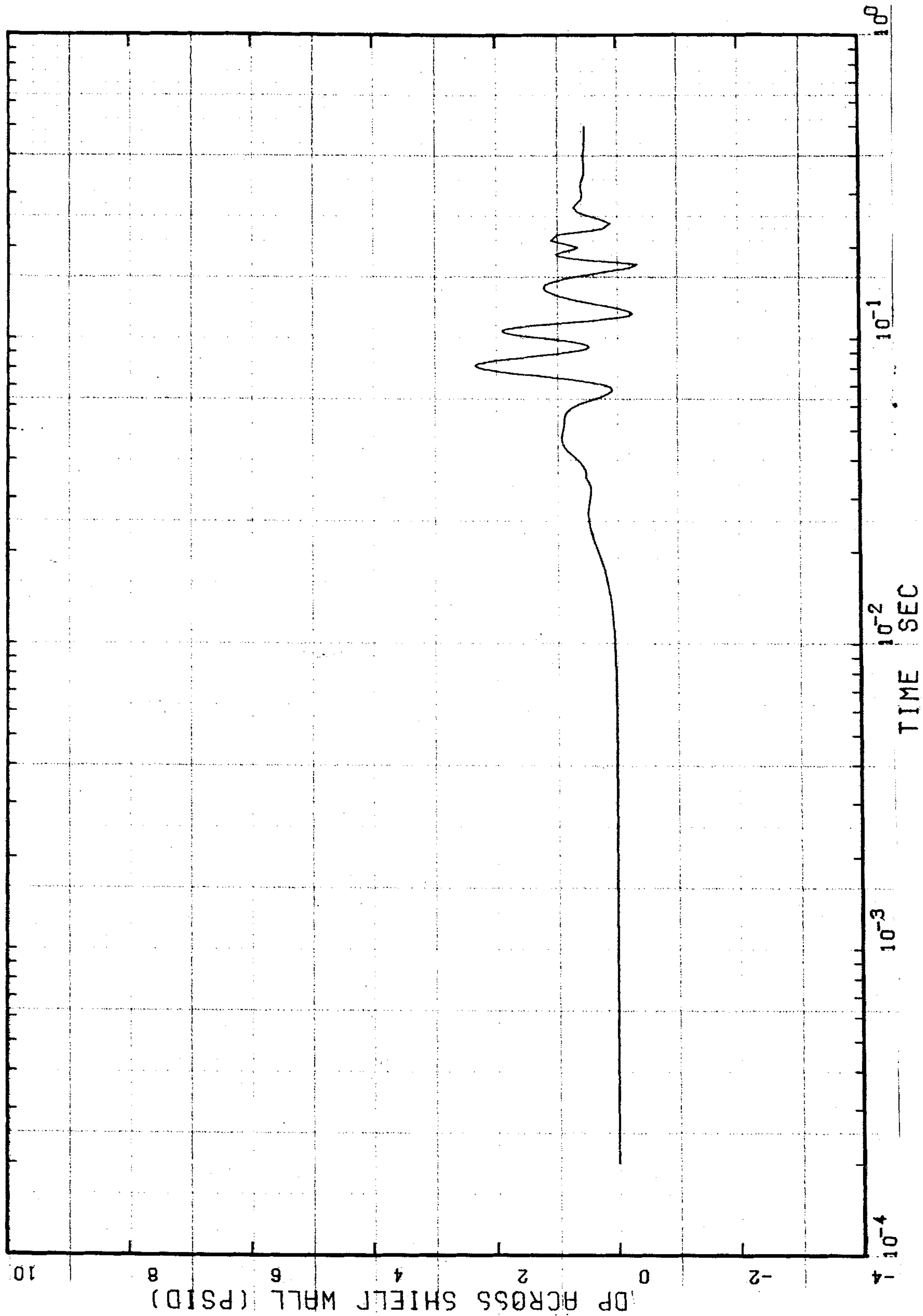

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FIGURE 6.2-48

ΔP VS. LOG T FOR NODE 25
(RECIRCULATION OUTLET LINE BREAK)

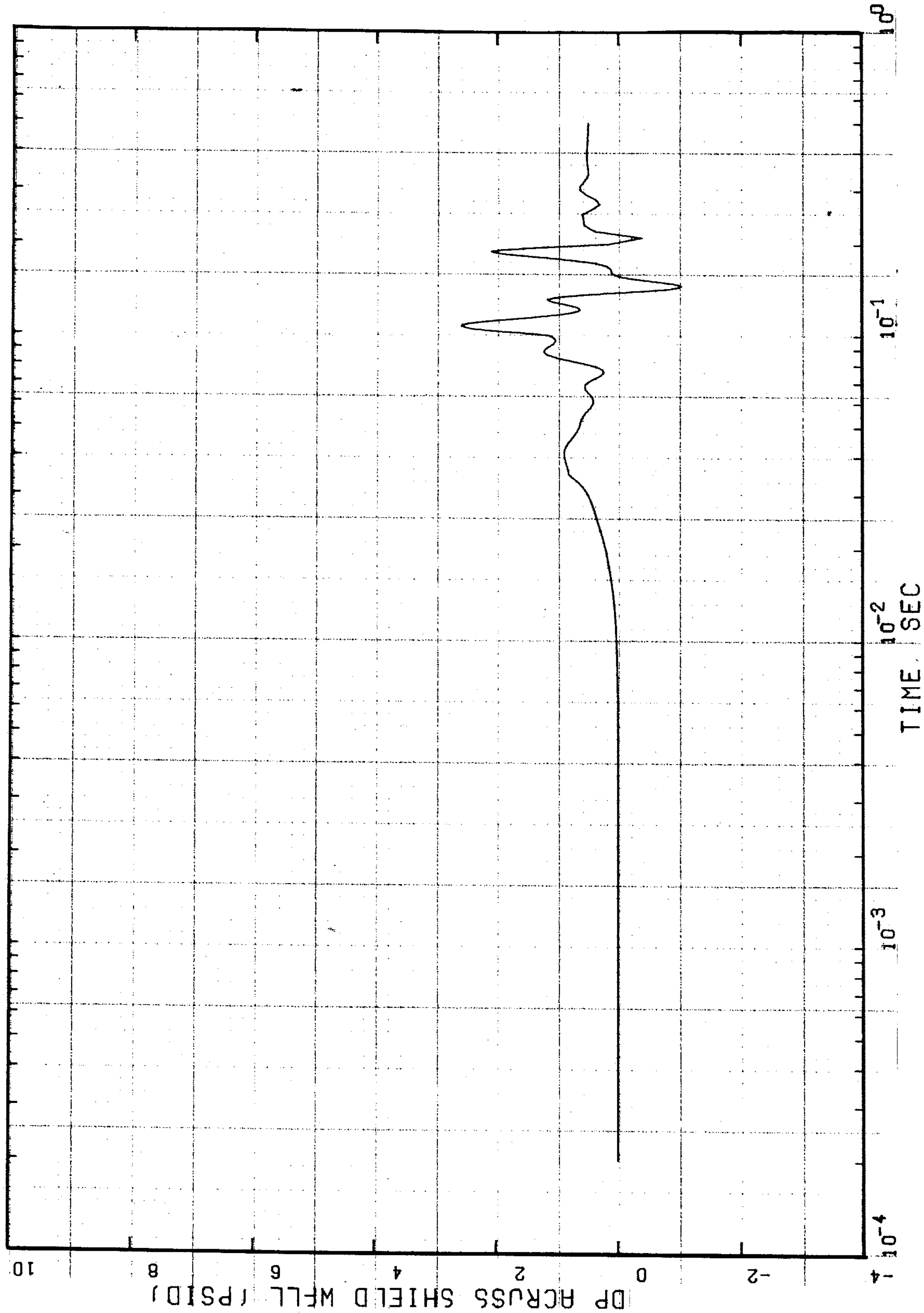

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FIGURE 6.2-49

ΔP VS. LOG T FOR NODE 26
(RECIRCULATION OUTLET LINE BREAK)

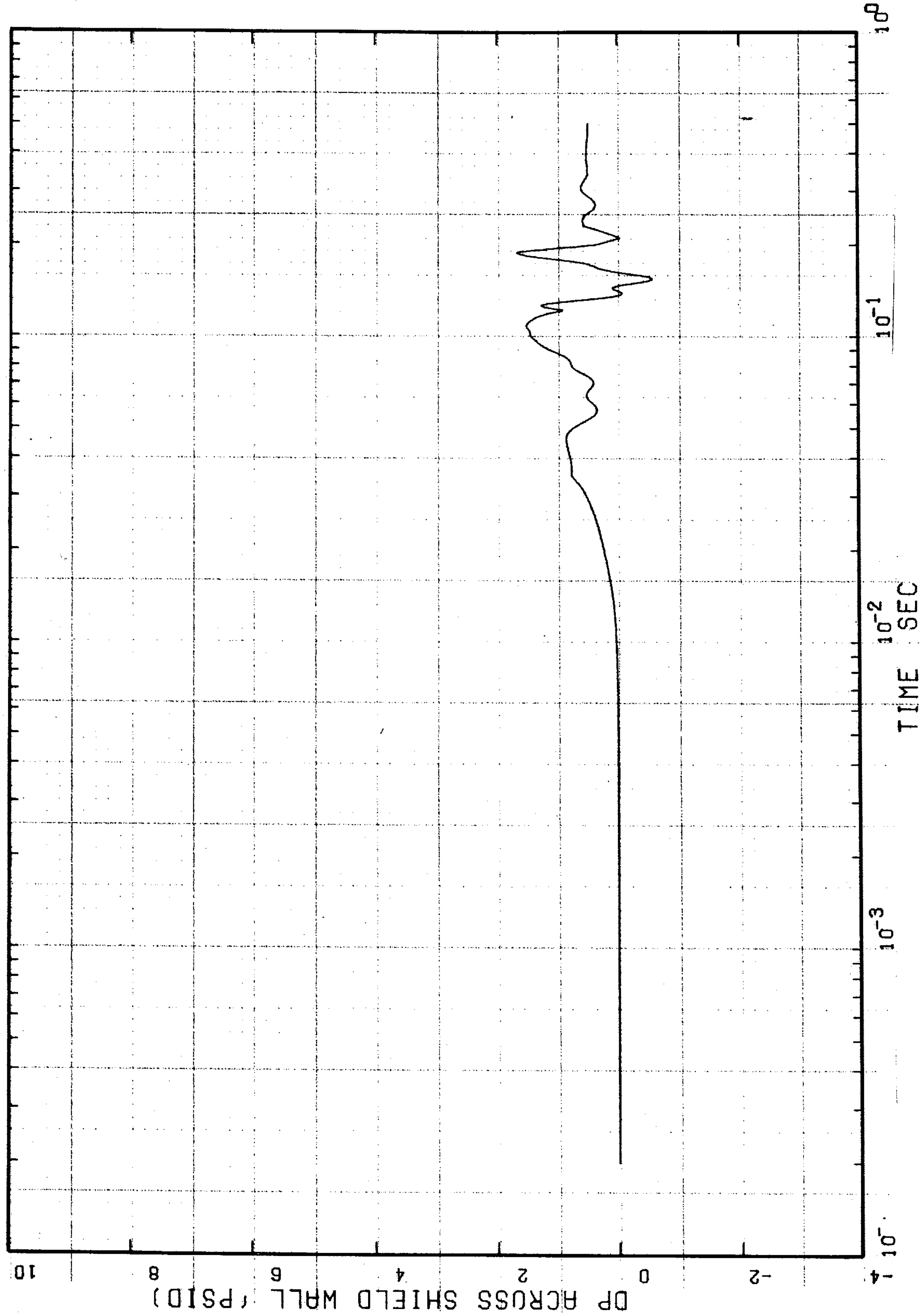

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FIGURE 6.2-50

ΔP VS. LOG T FOR NODE 27
(RECIRCULATION OUTLET LINE BREAK)

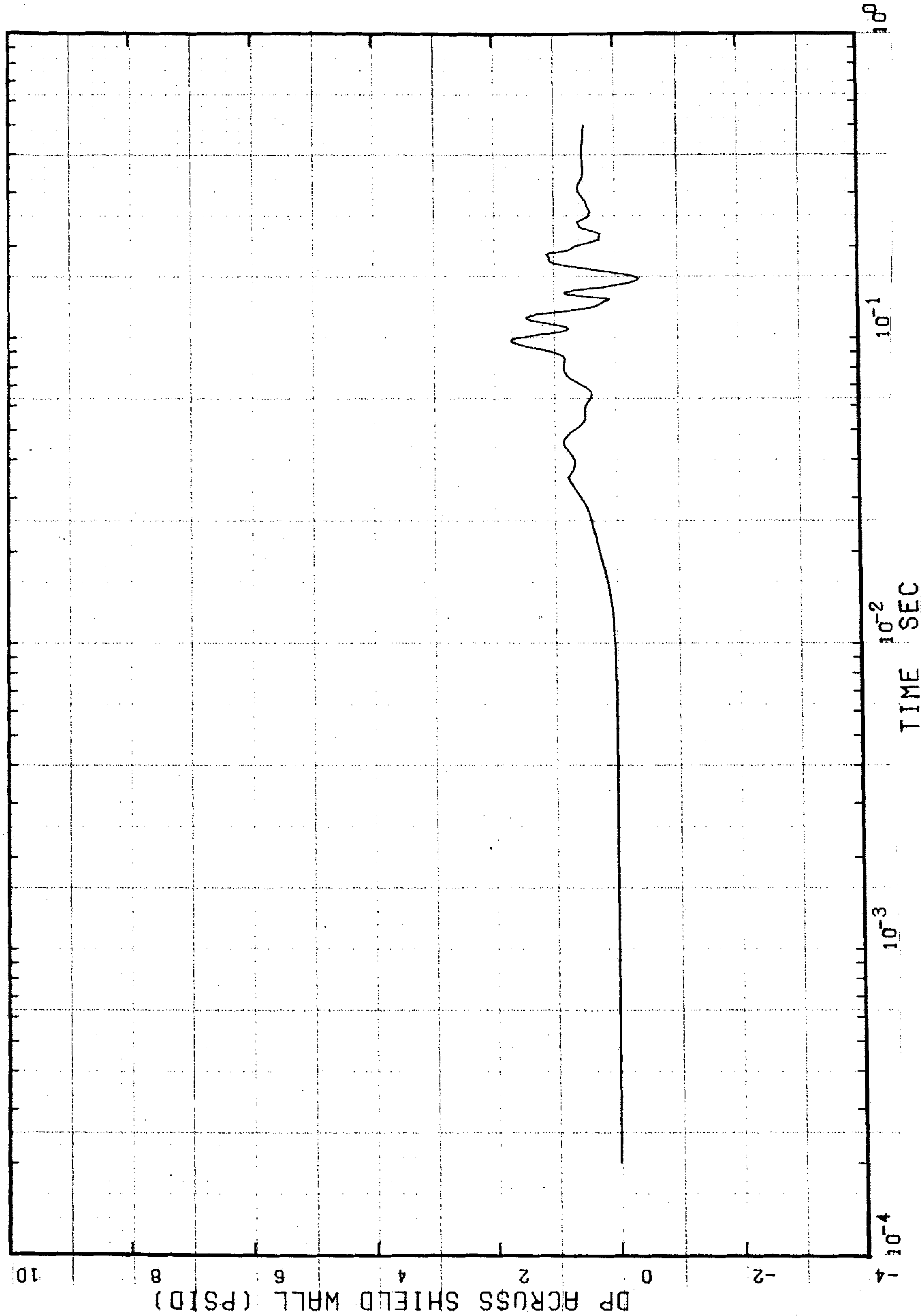

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FIGURE 6.2-51

ΔP VS. LOG T FOR NODE 28
(RECIRCULATION OUTLET LINE BREAK)

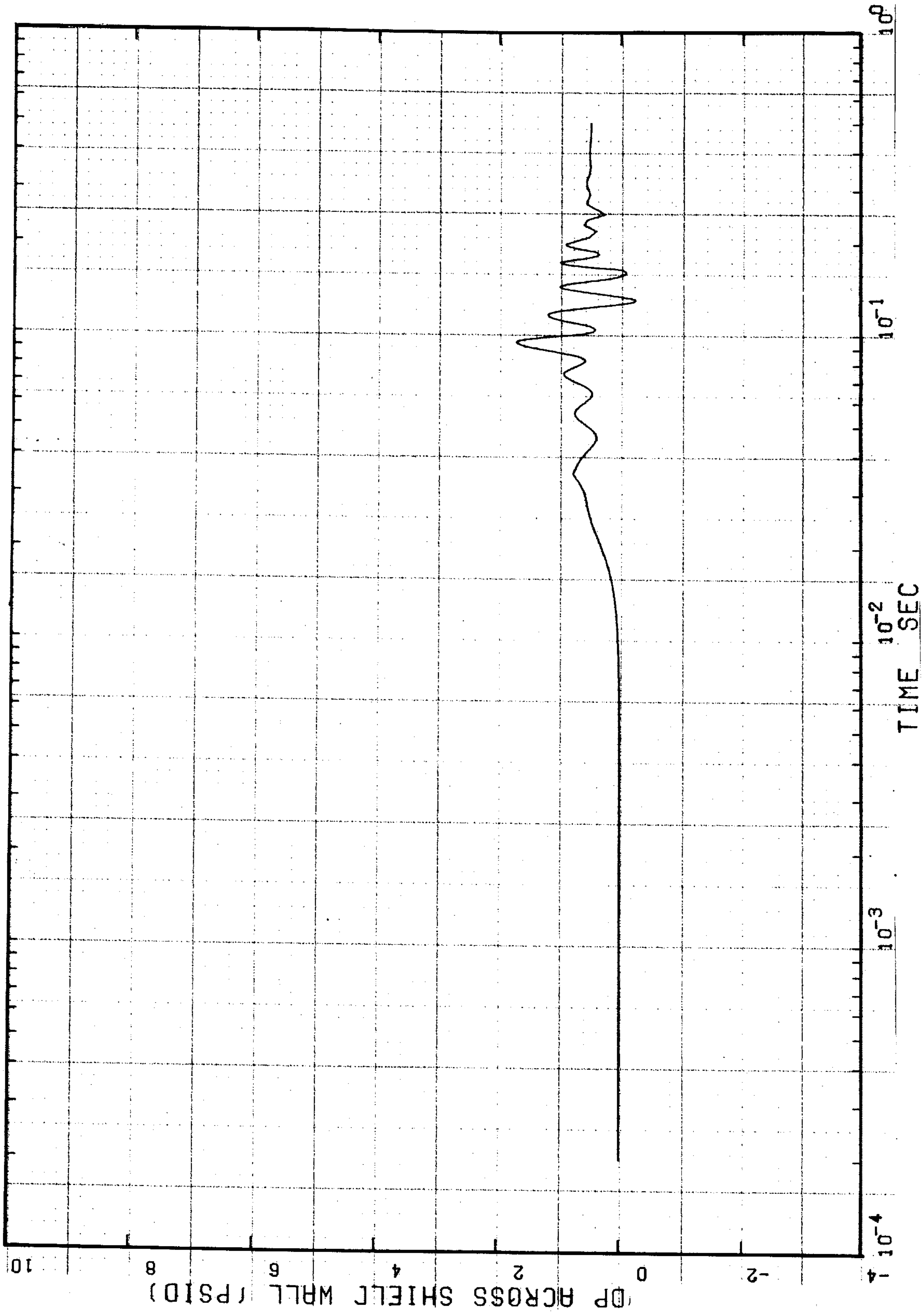

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FIGURE 6.2-52

ΔP VS. LOG T FOR NODE 29
(RECIRCULATION OUTLET LINE BREAK)

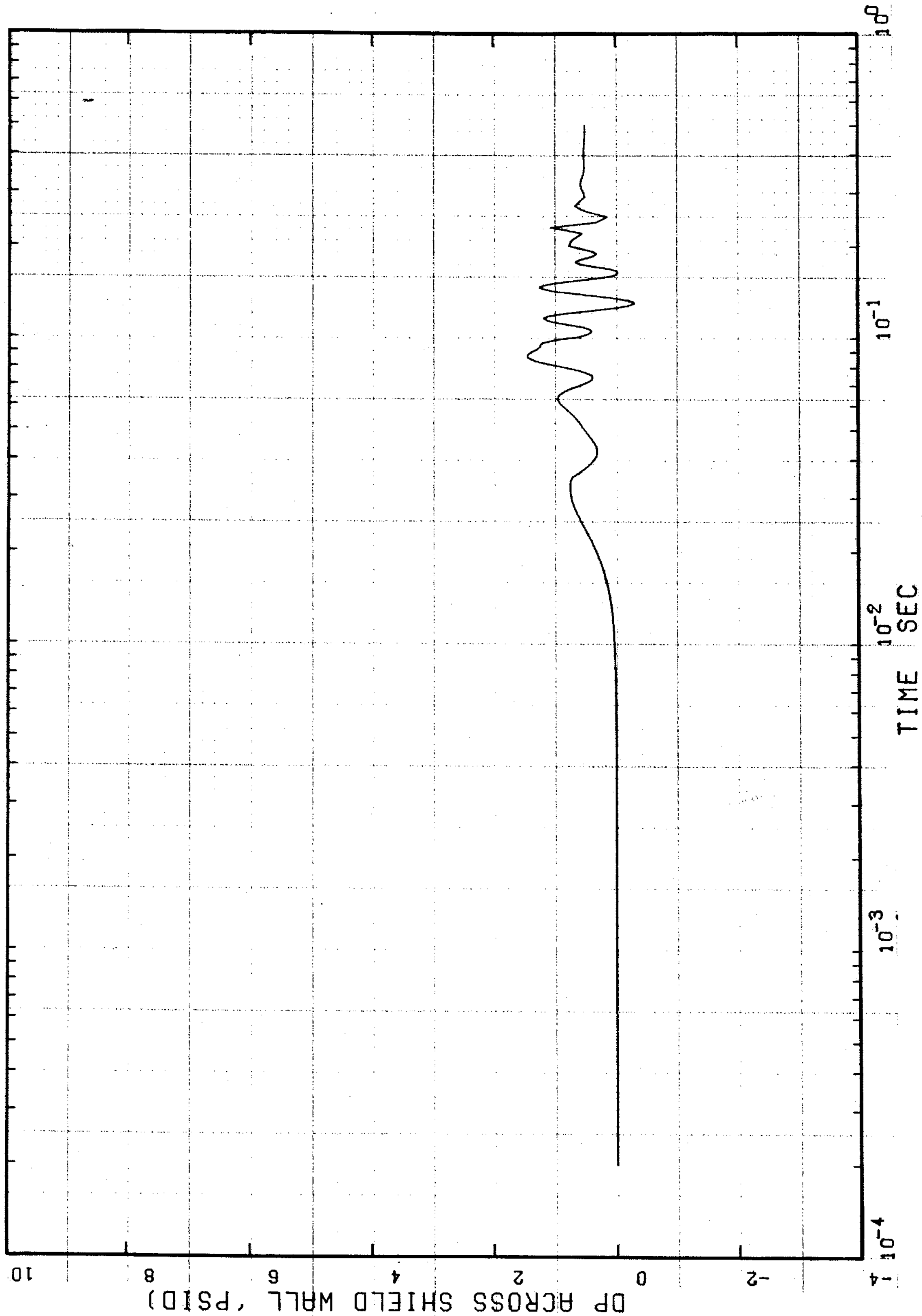

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FIGURE 6.2-53

ΔP VS. LOG T FOR NODE 30
(RECIRCULATION OUTLET LINE BREAK)

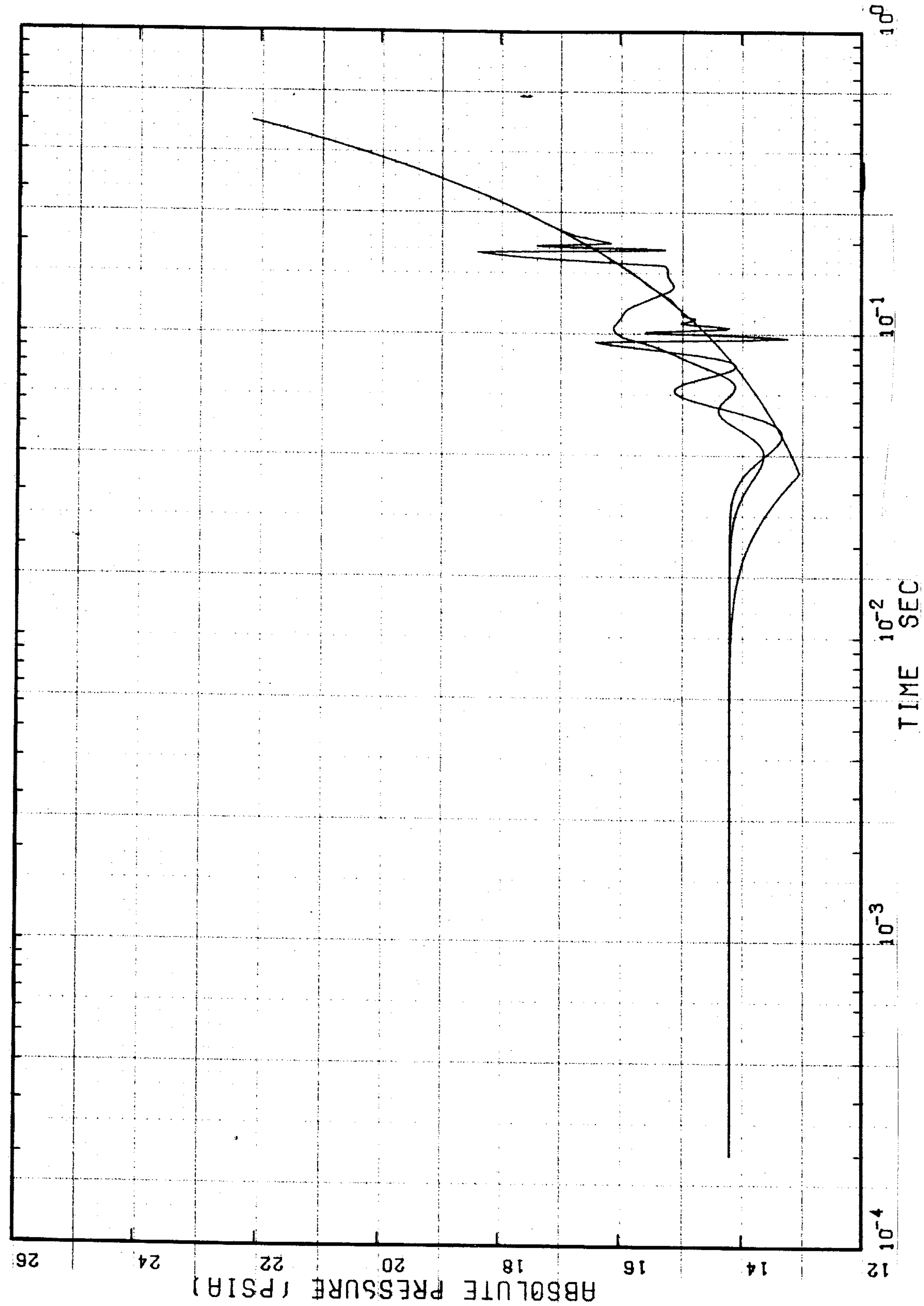

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FIGURE 6.2-54

PRESSURE RESPONSE OF
NODES 31, 32 AND 34
(RECIRCULATION OUTLET LINE BREAK)

Figures 6.2-55 and 6.2-56 have been deleted.

Security - Related Information Figure Withheld Under 10 CFR 2.390

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FIGURE 6.2-57

CROSS SECTION OF SHIELD ANNULUS AT LEVEL OF REACTOR RECIRCULATION PIPING

Security - Related Information Figure Withheld Under 10 CFR 2.390

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FIGURE 6.2-58

CROSS SECTION OF SHIELD ANNULUS AT
LEVEL OF MAIN STEAM AND FEEDWATER PIPING

Security - Related Information Figure Withheld Under 10 CFR 2.390

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FIGURE 6.2-59

FLOW DIVERTER SLEEVE - OPERATING DETAILS

Figures 6.2-60 and 6.2-63 have been deleted.

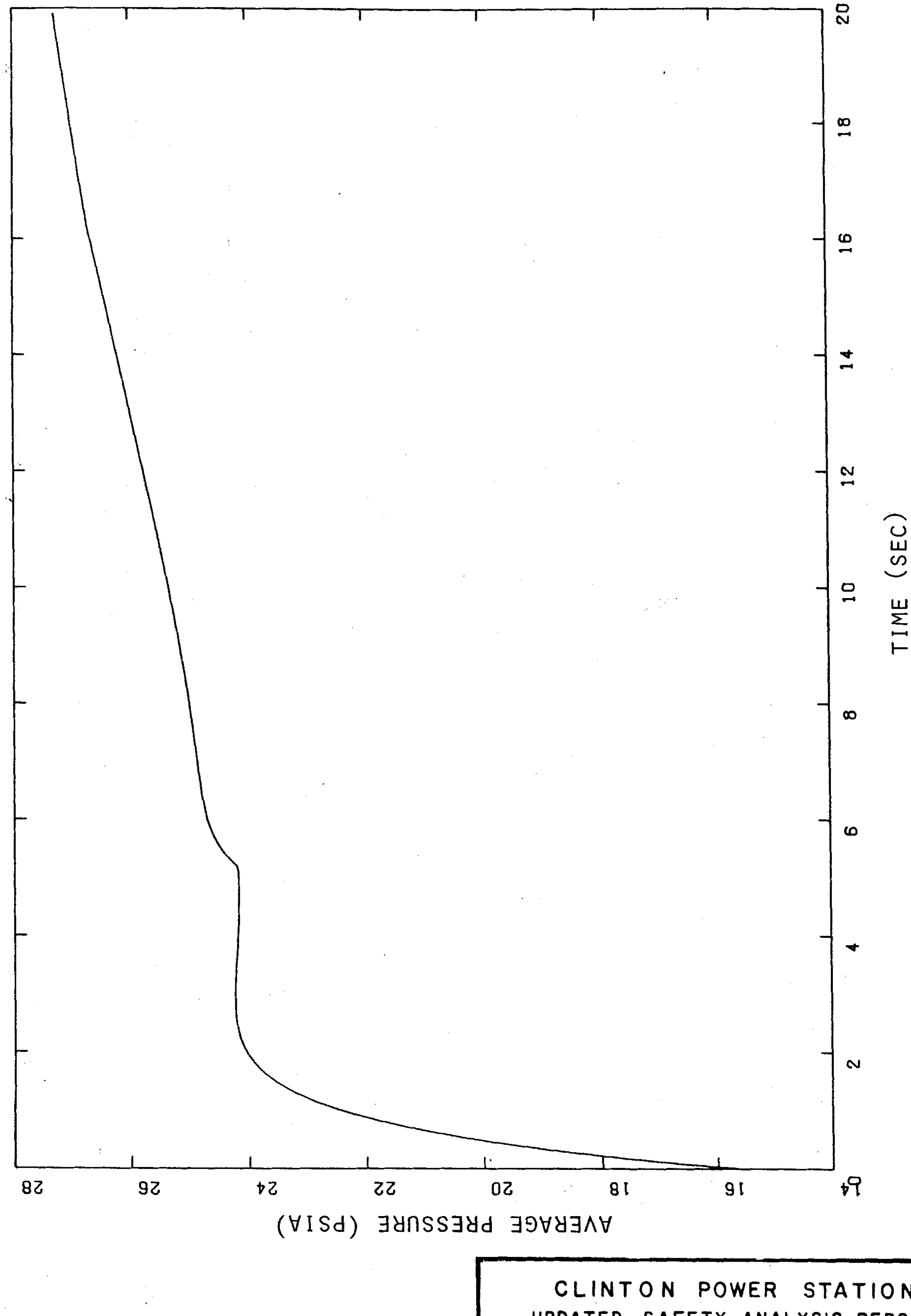

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FIGURE 6.2-64

AVERAGE HEAD CAVITY PRESSURE - SPRAY LINE BREAK

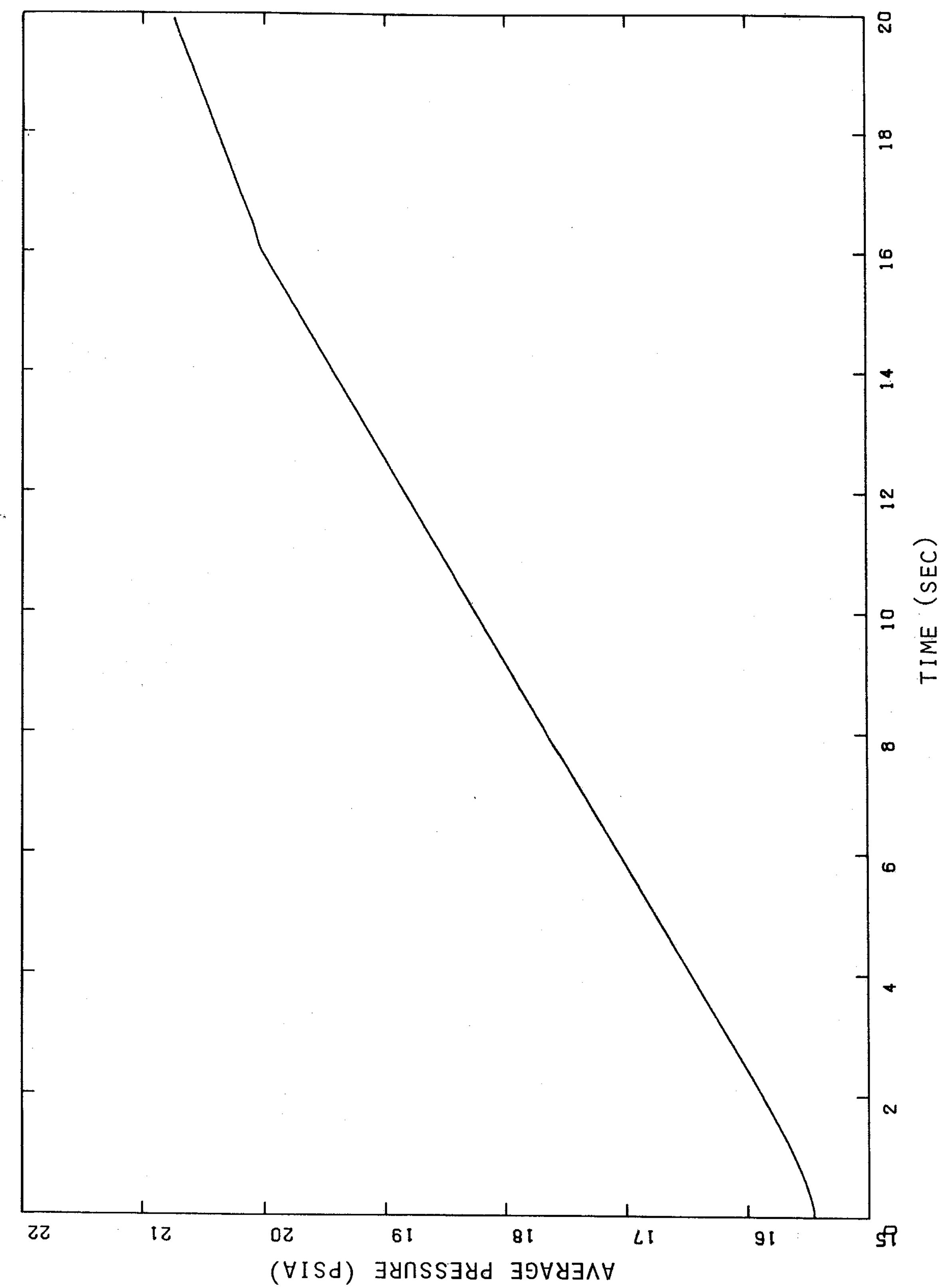

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FIGURE 6.2-65

AVERAGE DRYWELL PRESSURE - SPRAY LINE BREAK

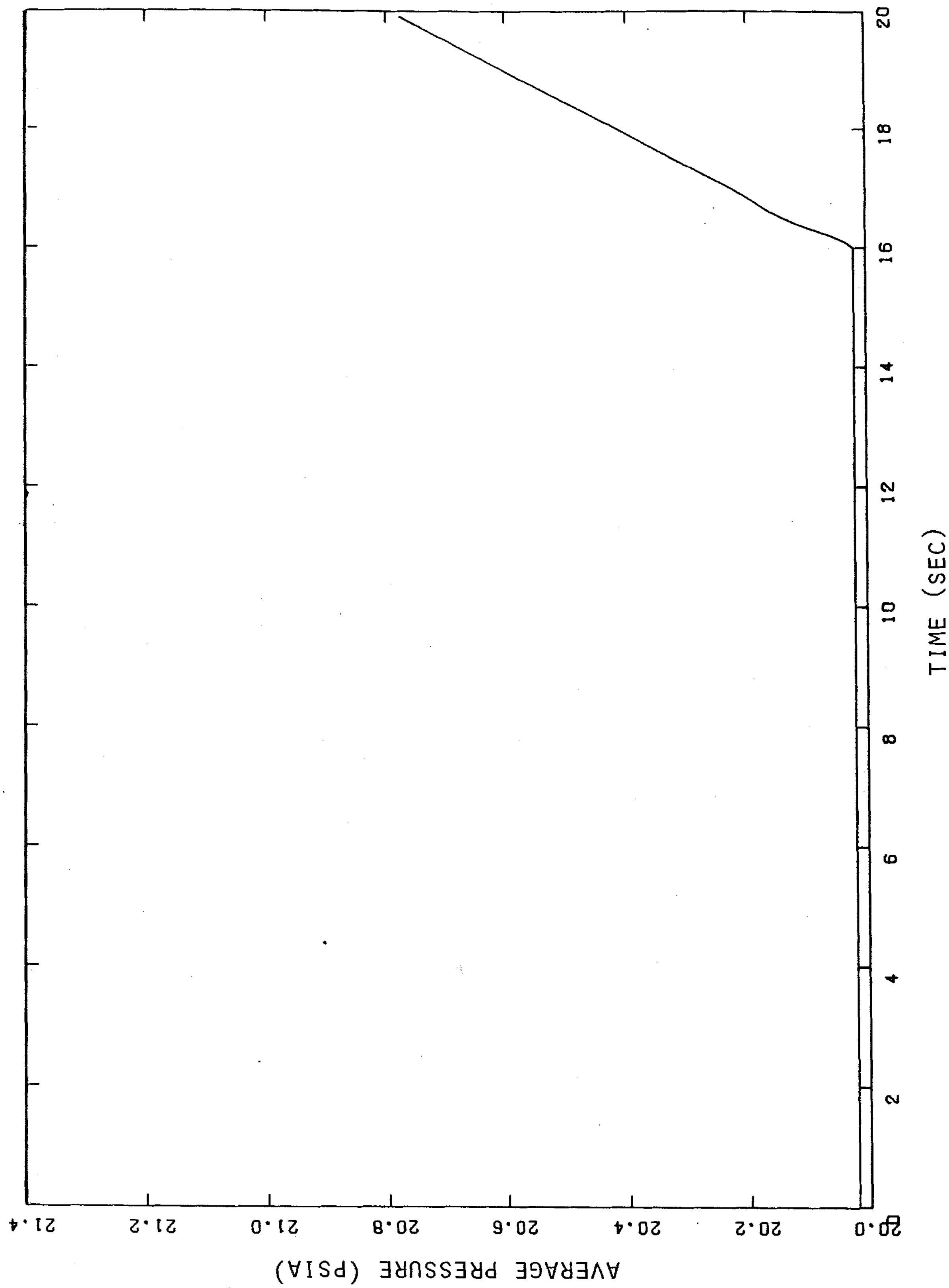

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FIGURE 6.2-66

AVERAGE WETWELL PRESSURE -
SPRAY LINE BREAK

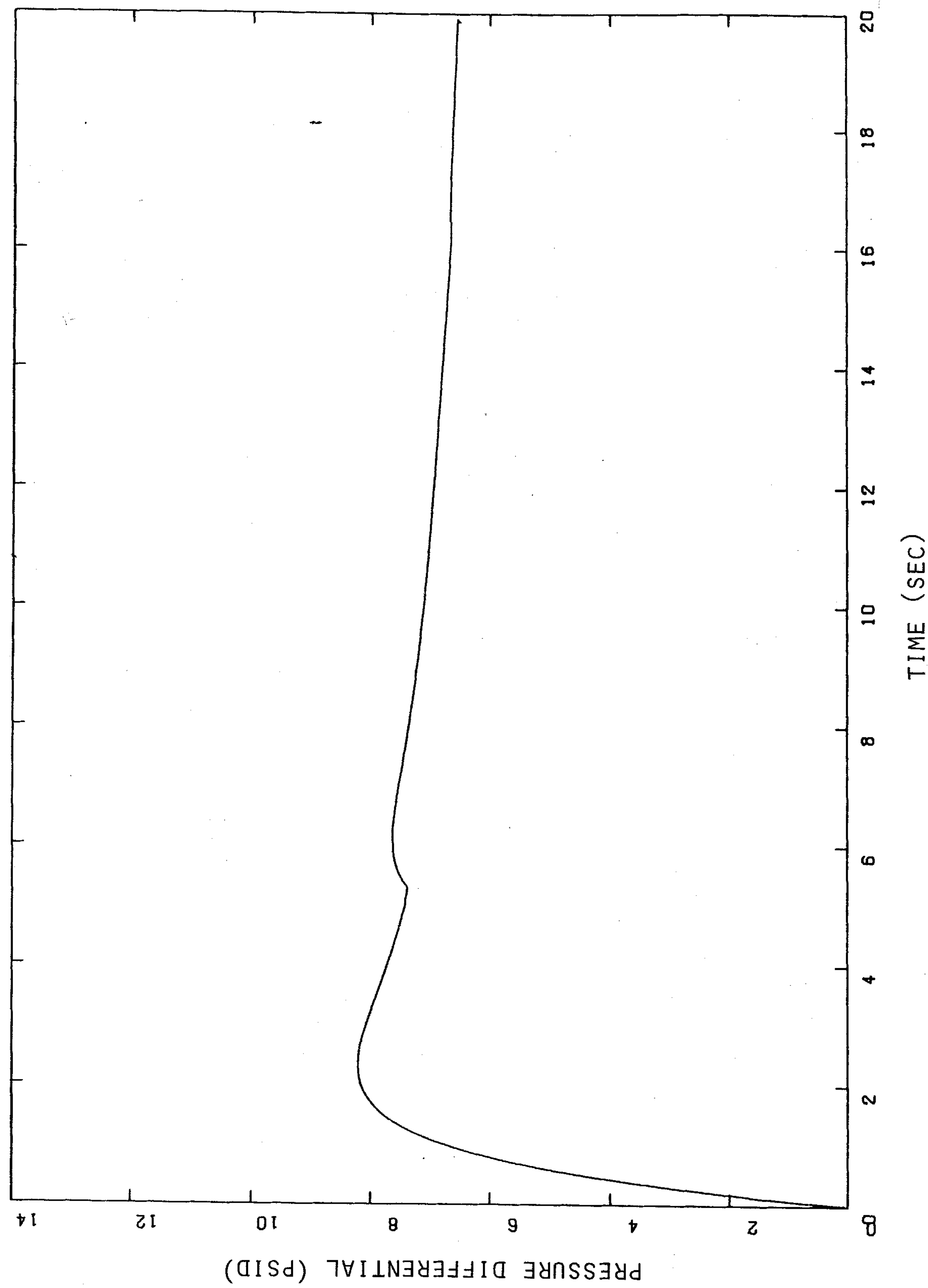

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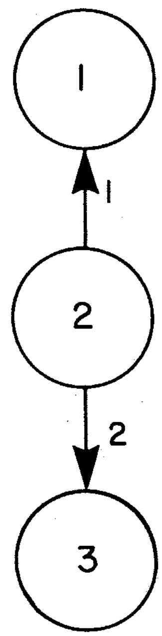
FIGURE 6.2-67

PRESSURE DIFFERENTIAL ACROSS BULKHEAD PLATE - SPRAY LINE BREAK



NOTE:

See Table 6.2-13 for a description of the nodes and
Table 6.2-20 for a description of the vent paths.

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FIGURE 6.2-68

NODALIZATION SCHEMATIC FOR RECIRCULATION
LINE BREAK IN DRYWELL FOR HEAD CAVITY
PRESSURIZATION STUDY

FIGURES 6.2-69 THROUGH 6.2-71

HAVE BEEN DELETED

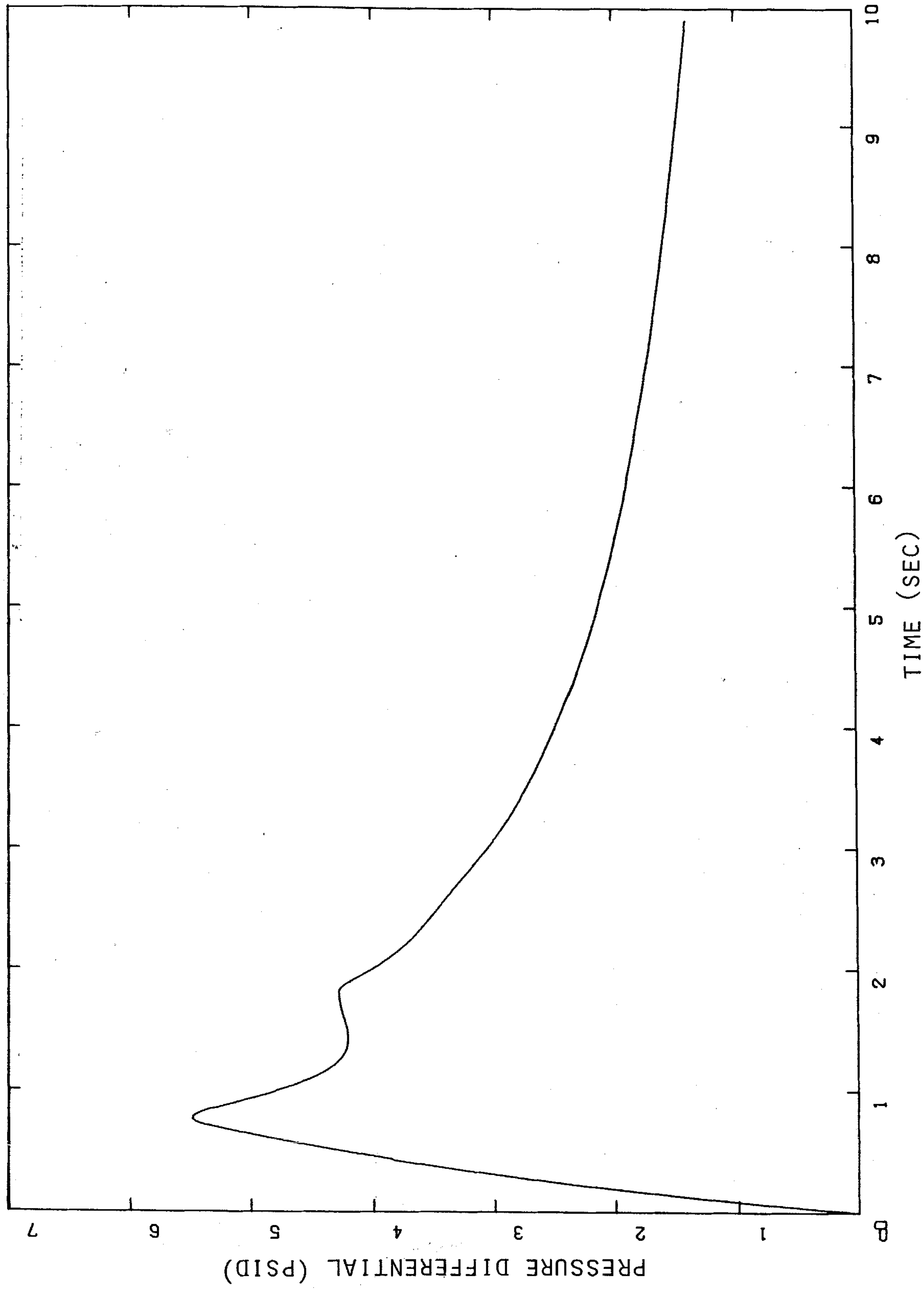

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FIGURE 6.2-72

PRESSURE DIFFERENTIAL ACROSS BULKHEAD PLATE - RECIRCULATION LINE BREAK

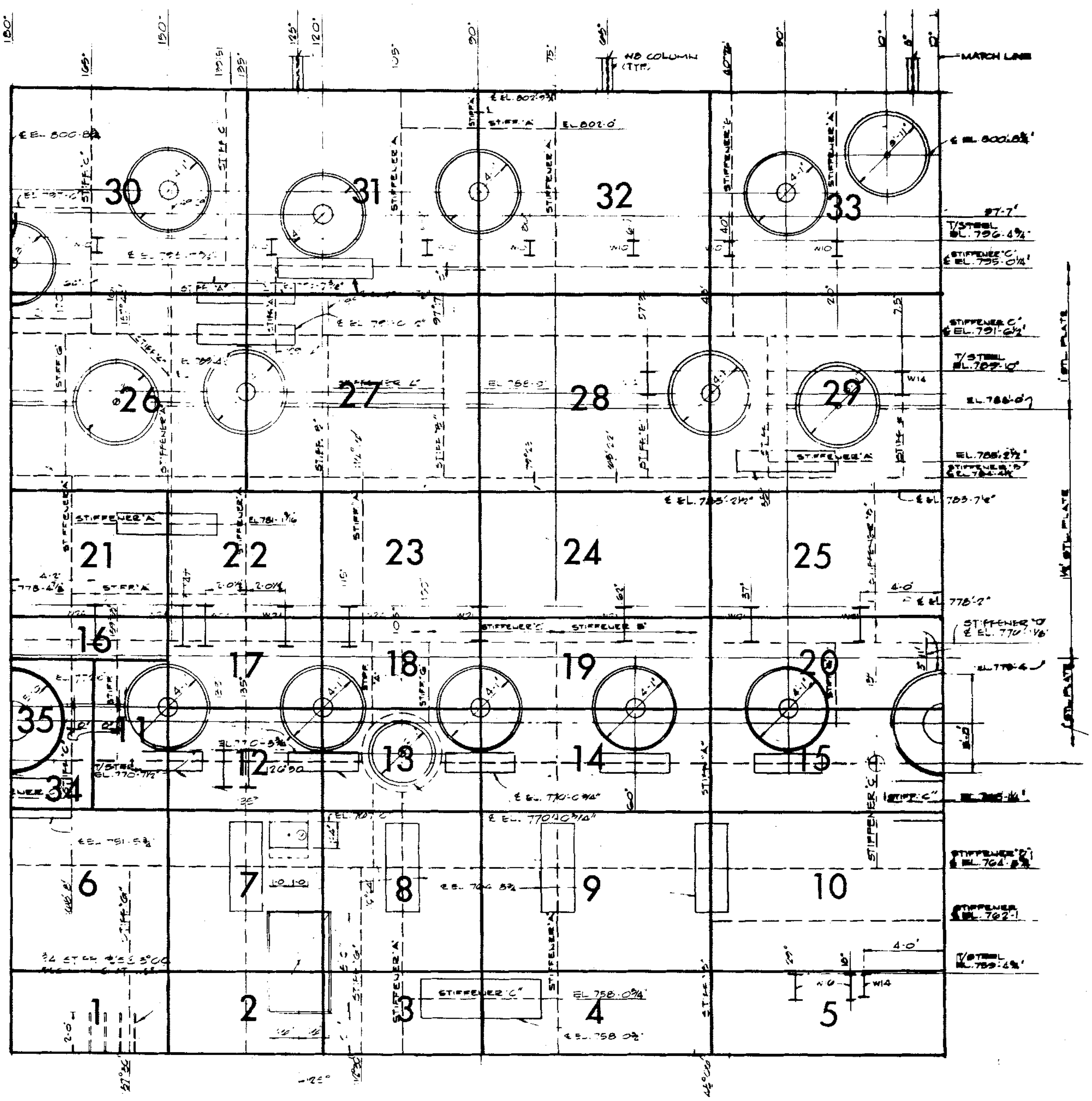

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FIGURE 6.2-73

NODALIZATION OVERLAY FOR RECIRCULATION LINE BREAK

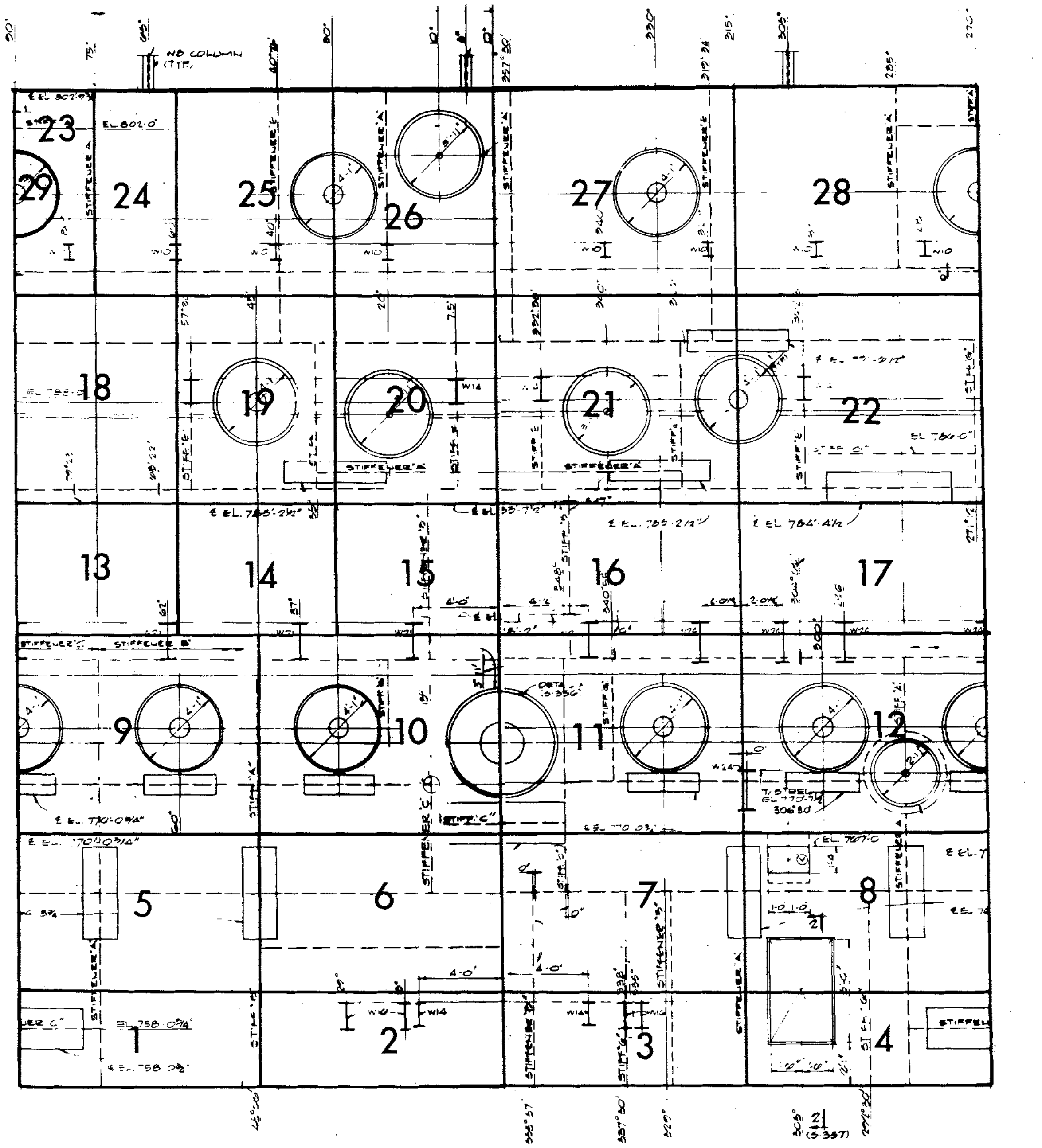

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FIGURE 6.2-74

NODALIZATION OVERLAY FOR
FEEDWATER LINE BREAK

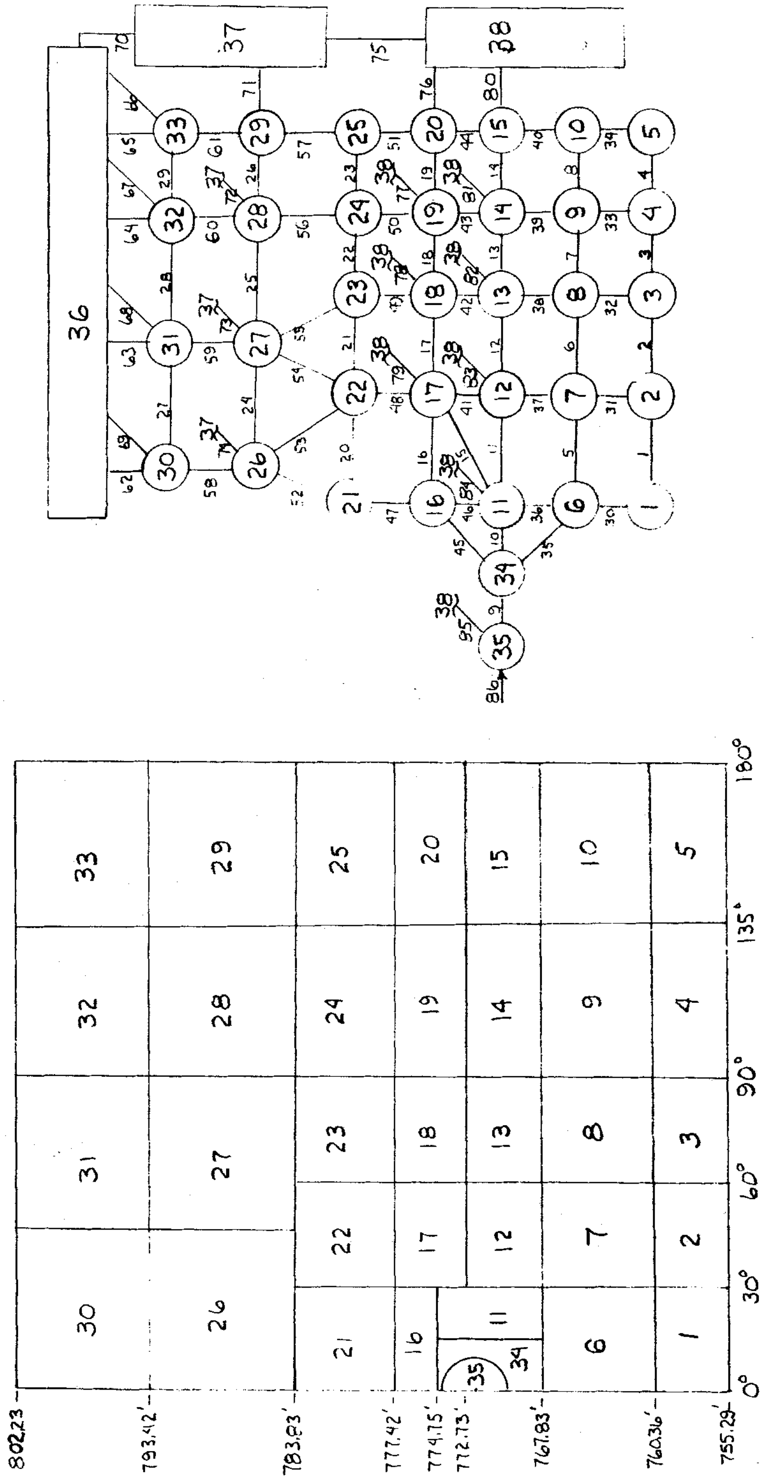

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FIGURE 6.2-75

NODALIZATION FOR ORIGINAL RECIRCULATION LINE BREAK ANALYSIS

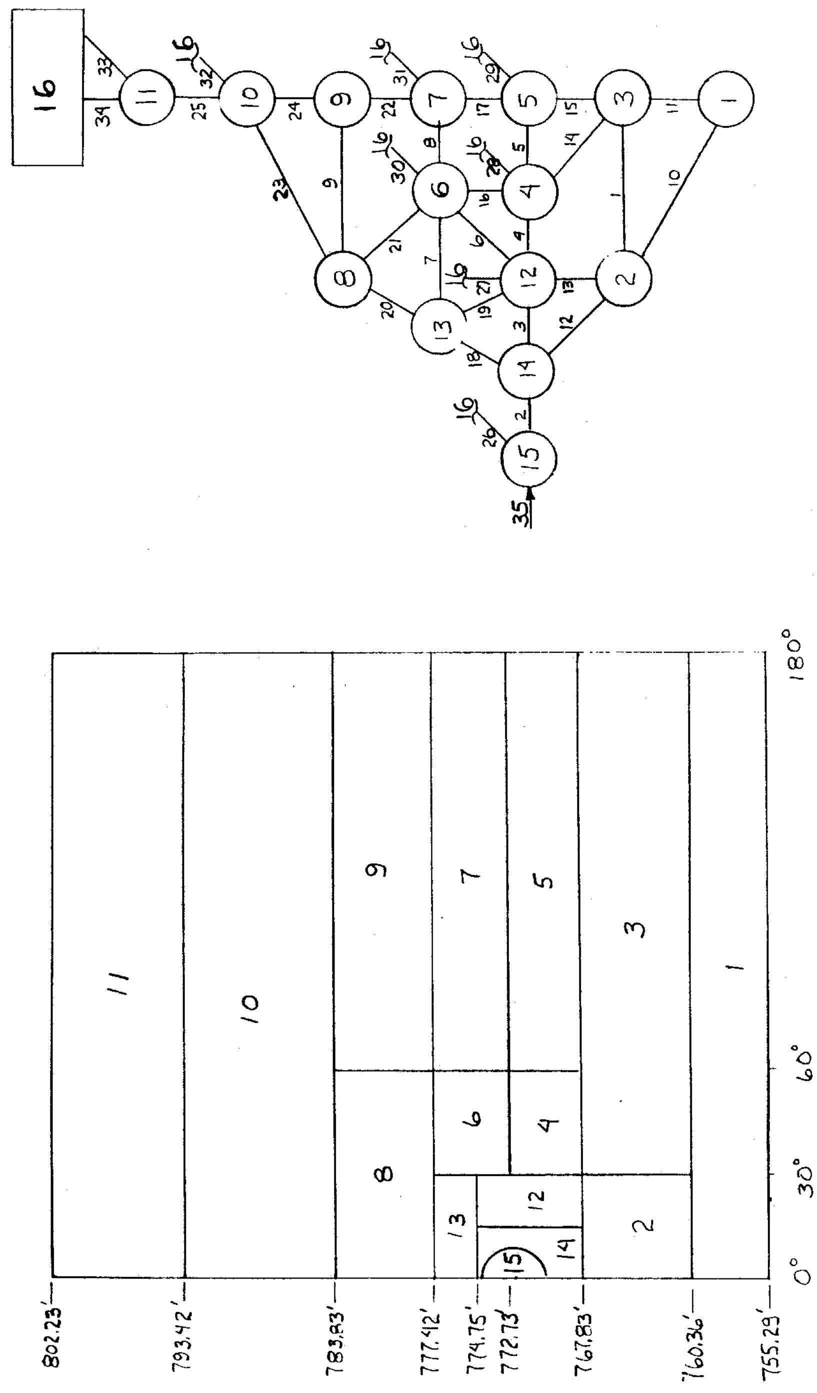

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FIGURE 6.2-76

"EQUIVALENT" NODALIZATION (CASE A)

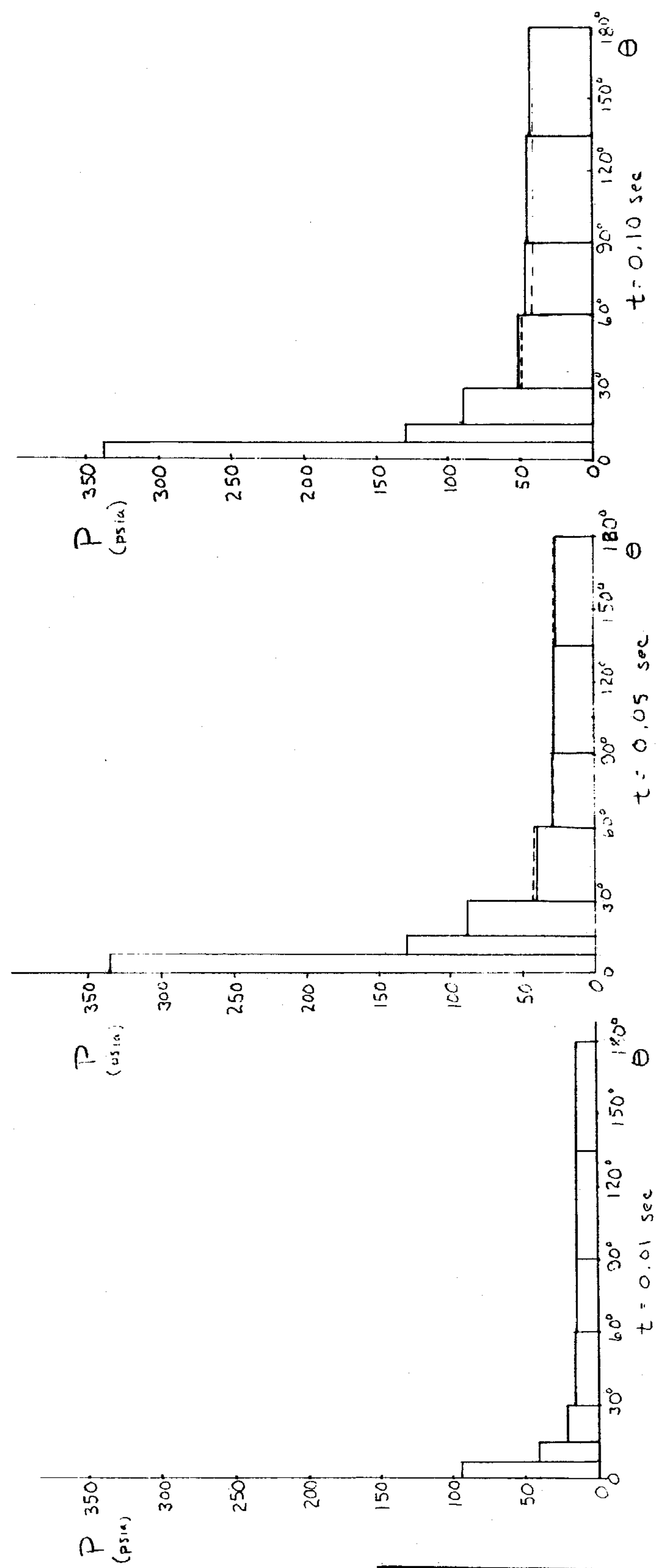

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FIGURE 6.2-77

AZIMUTHAL PRESSURE DISTRIBUTION
(AT ℄ RECIRCULATION OUTLET NOZZLE)
ORIGINAL DATA AND CASE A

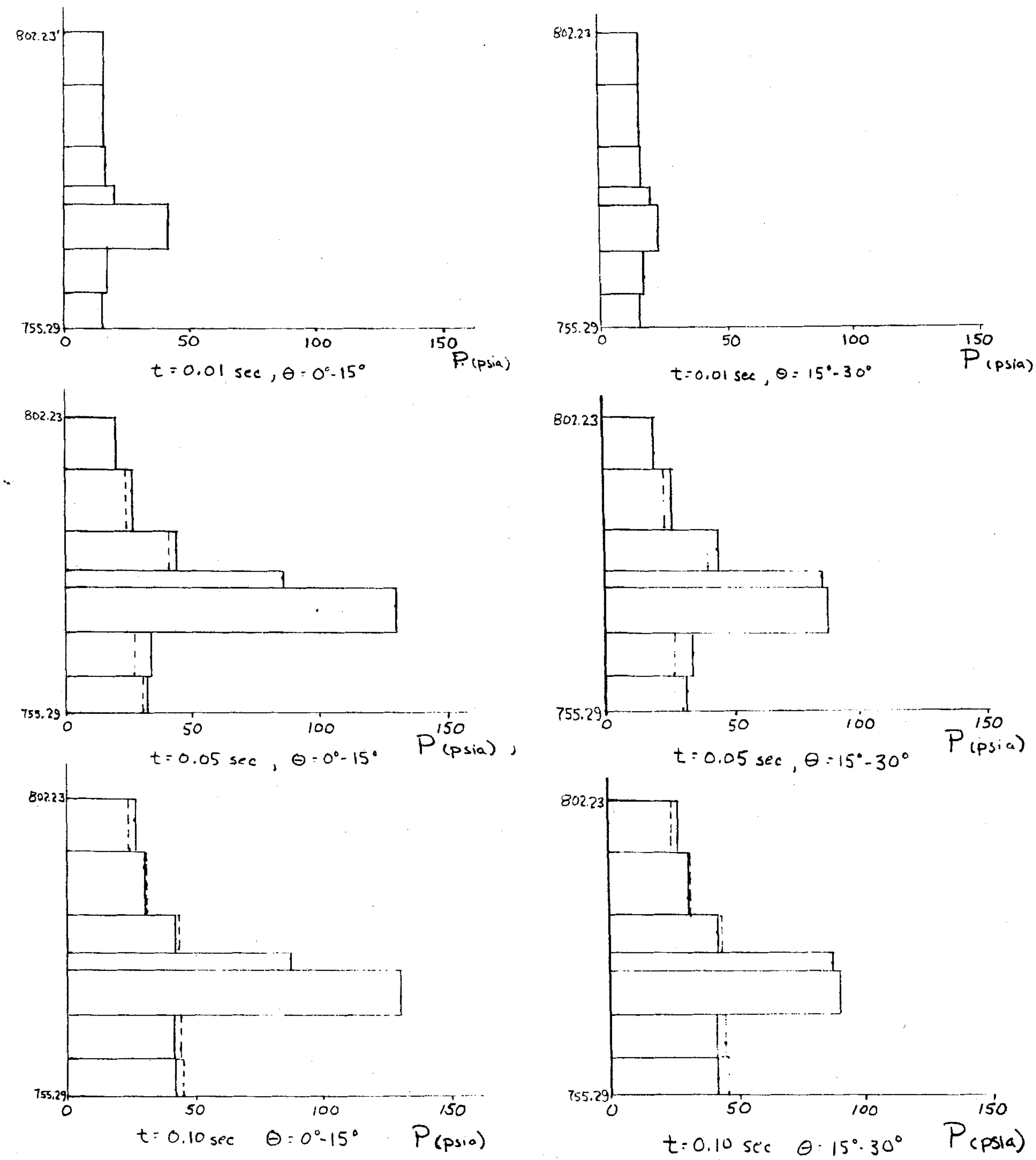

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FIGURE 6.2-78

AXIAL PRESSURE DISTRIBUTION
ORIGINAL DATA AND CASE A

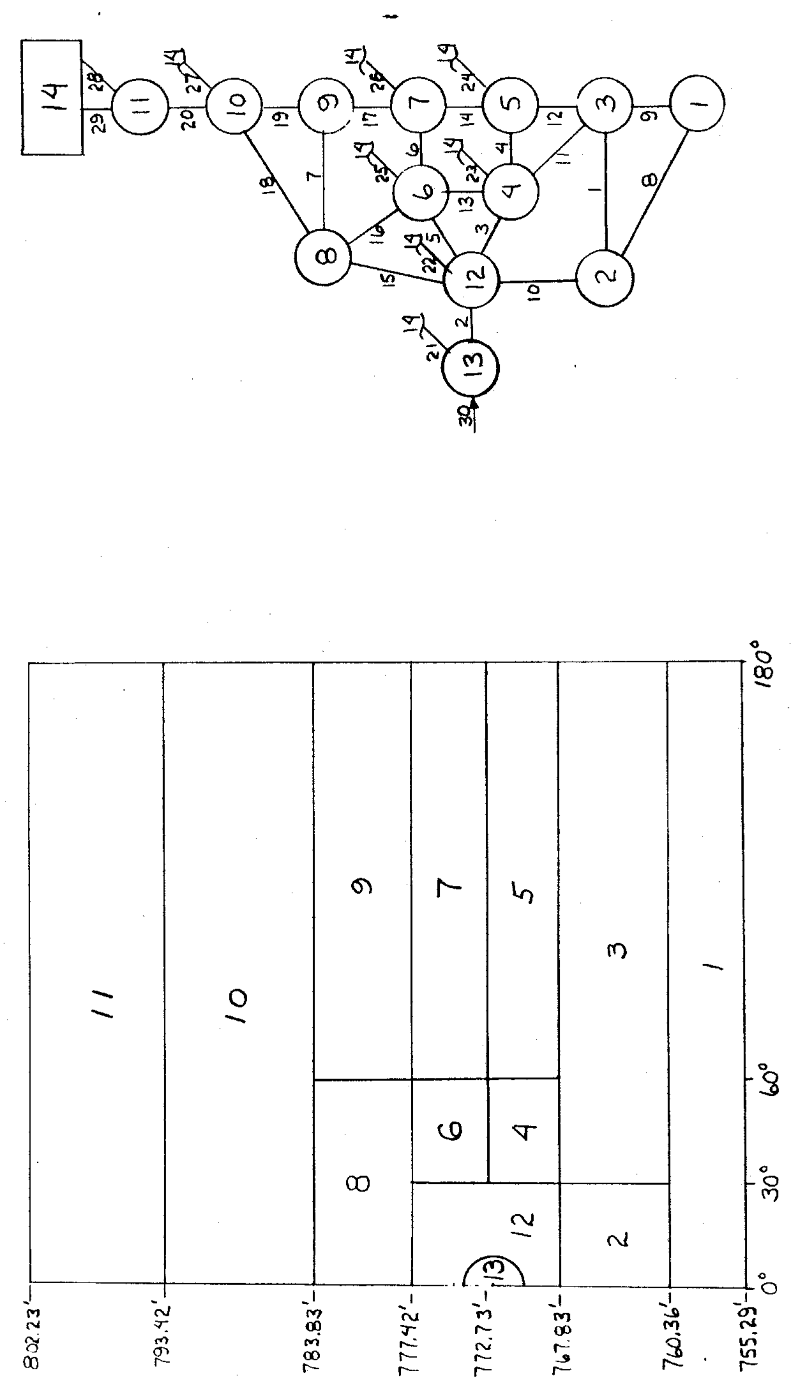

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FIGURE 6.2-79

SIMPLIFIED NODALIZATION (CASE B)

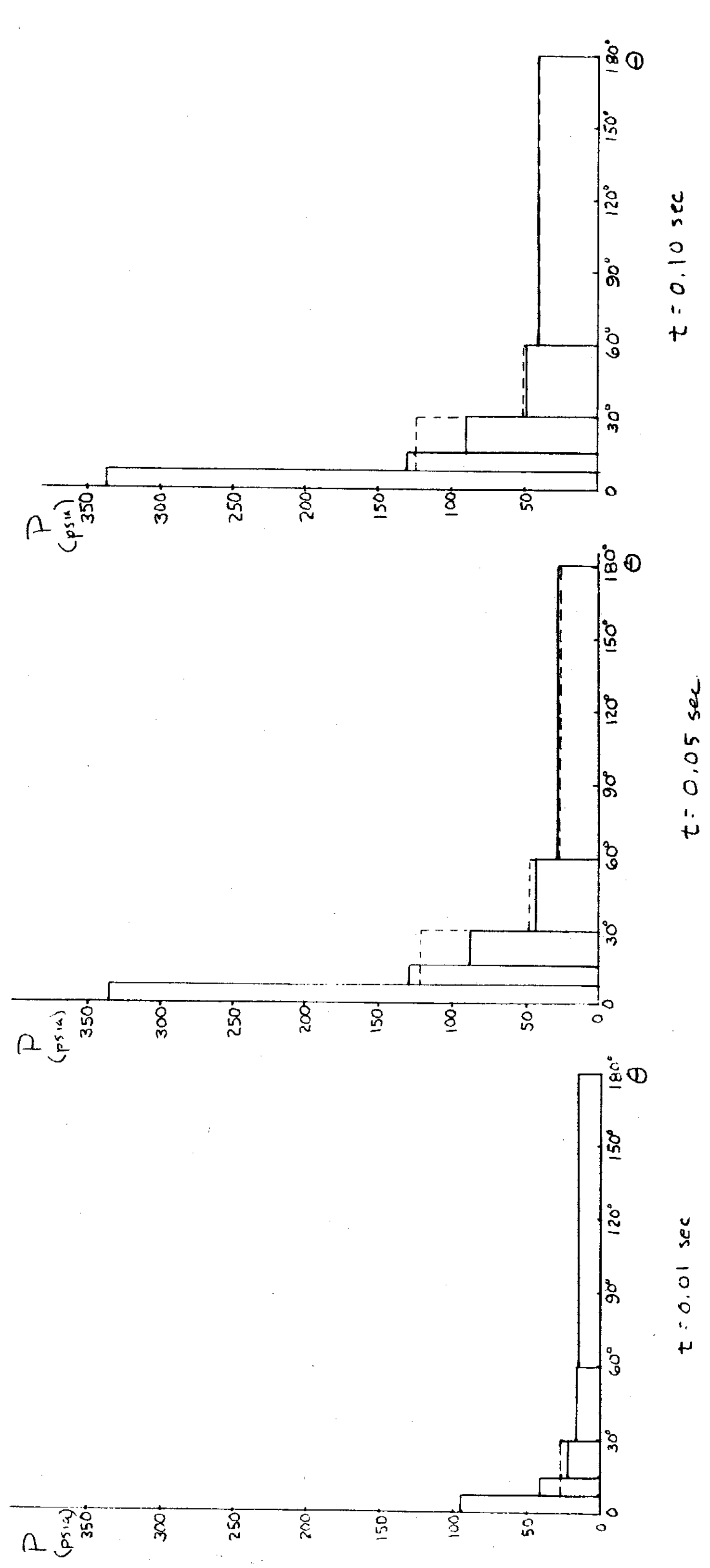

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FIGURE 6.2-80

AZIMUTHAL PRESSURE DISTRIBUTION
(AT ℄ RECIRCULATION OUTLET NOZZLE)
CASE A AND CASE B

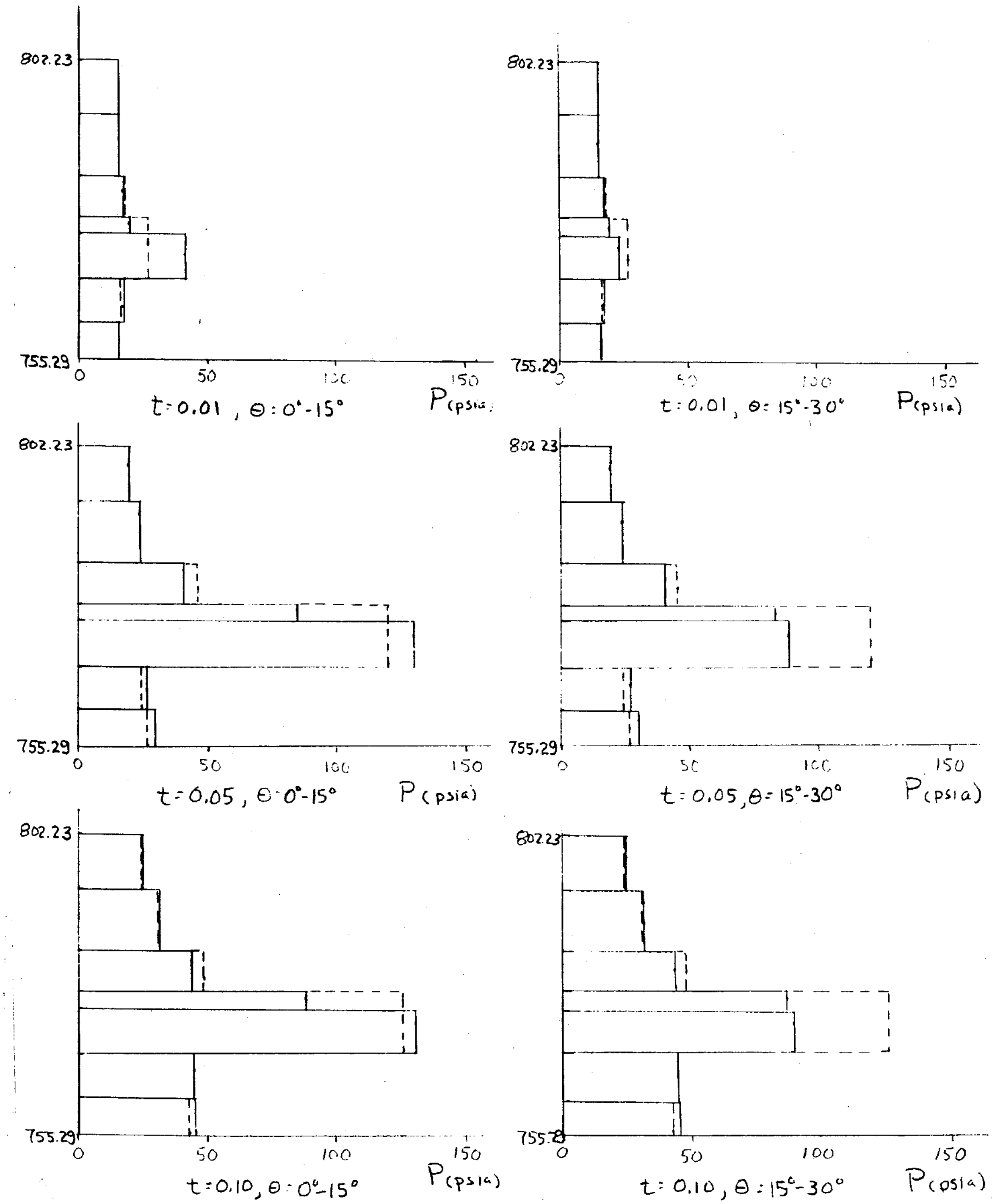

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FIGURE 6.2-81

AXIAL PRESSURE DISTRIBUTION CASE A AND CASE B

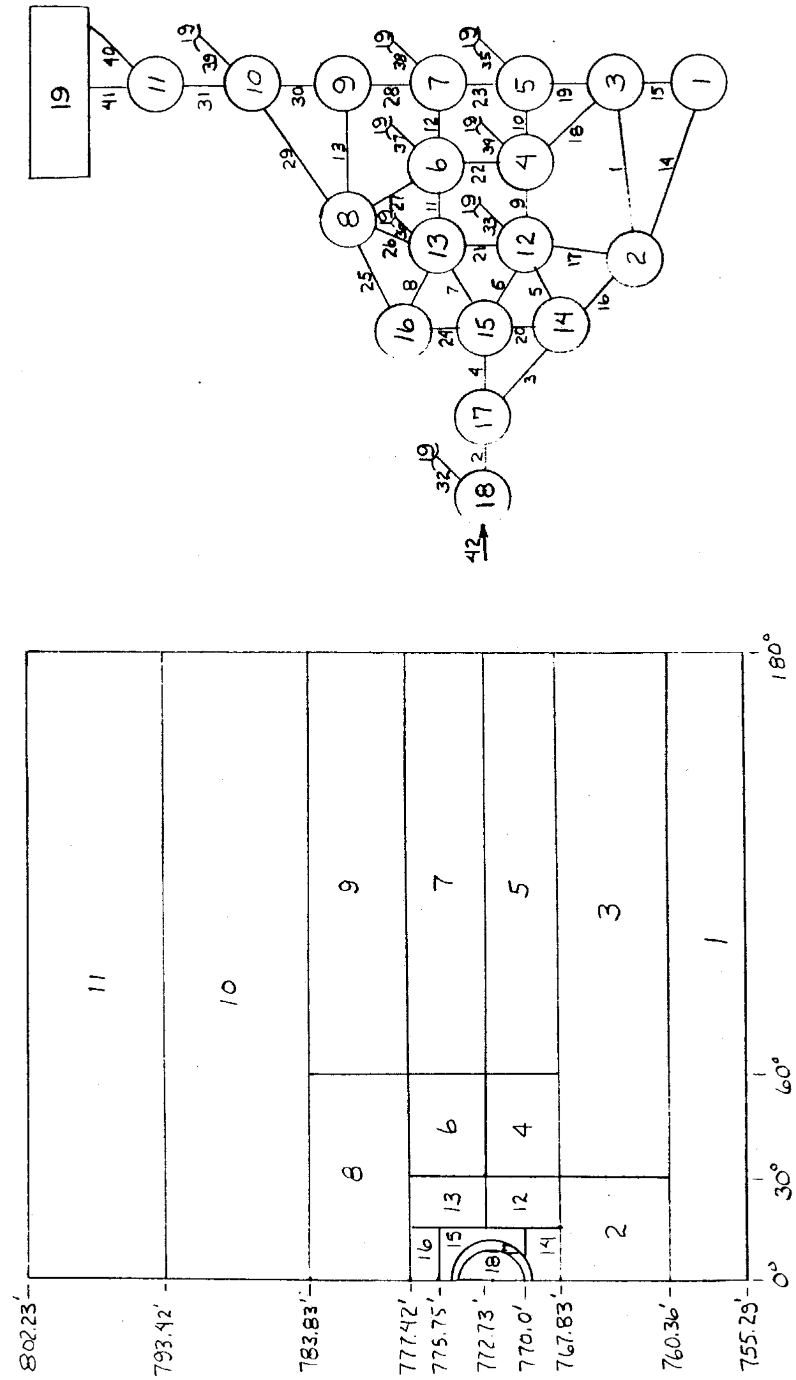

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FIGURE 6.2-82

COMPLEX NODALIZATION (CASE C)

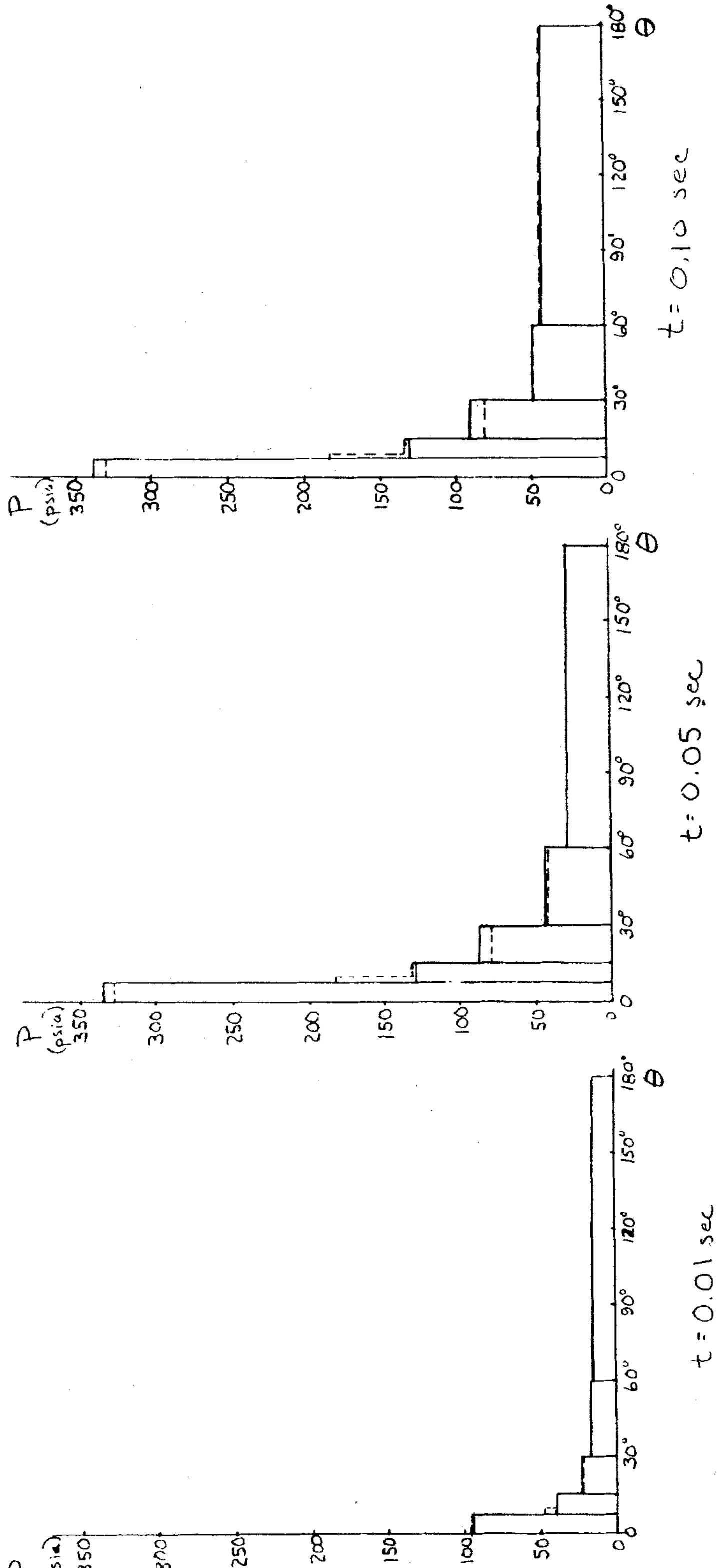

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FIGURE 6.2-83

AZIMUTHAL PRESSURE DISTRIBUTION
(AT ℄ RECIRCULATION OUTLET NOZZLE)
CASE A AND CASE C

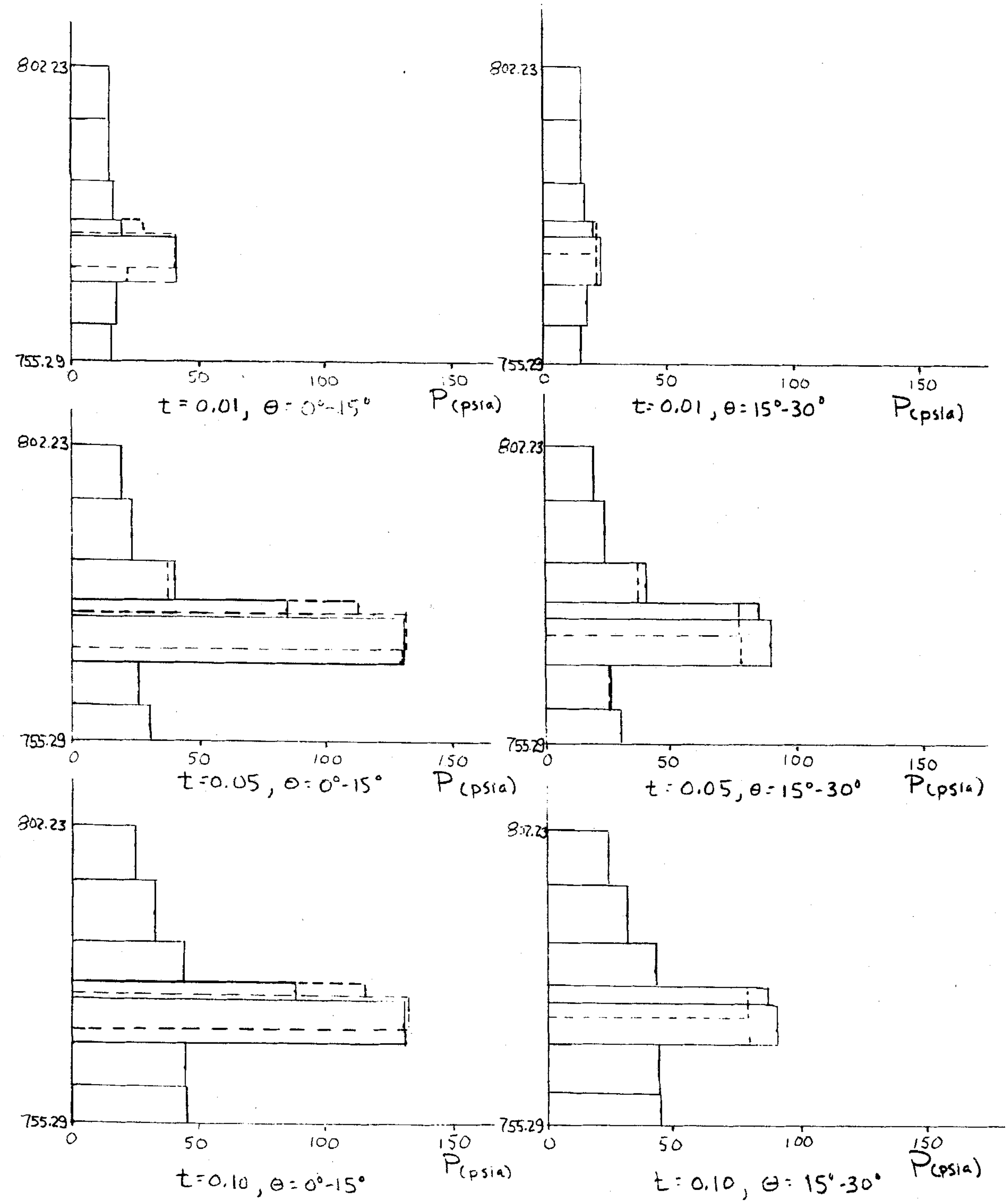

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FIGURE 6.2-84

AXIAL PRESSURE DISTRIBUTION
(CASE A AND CASE C)

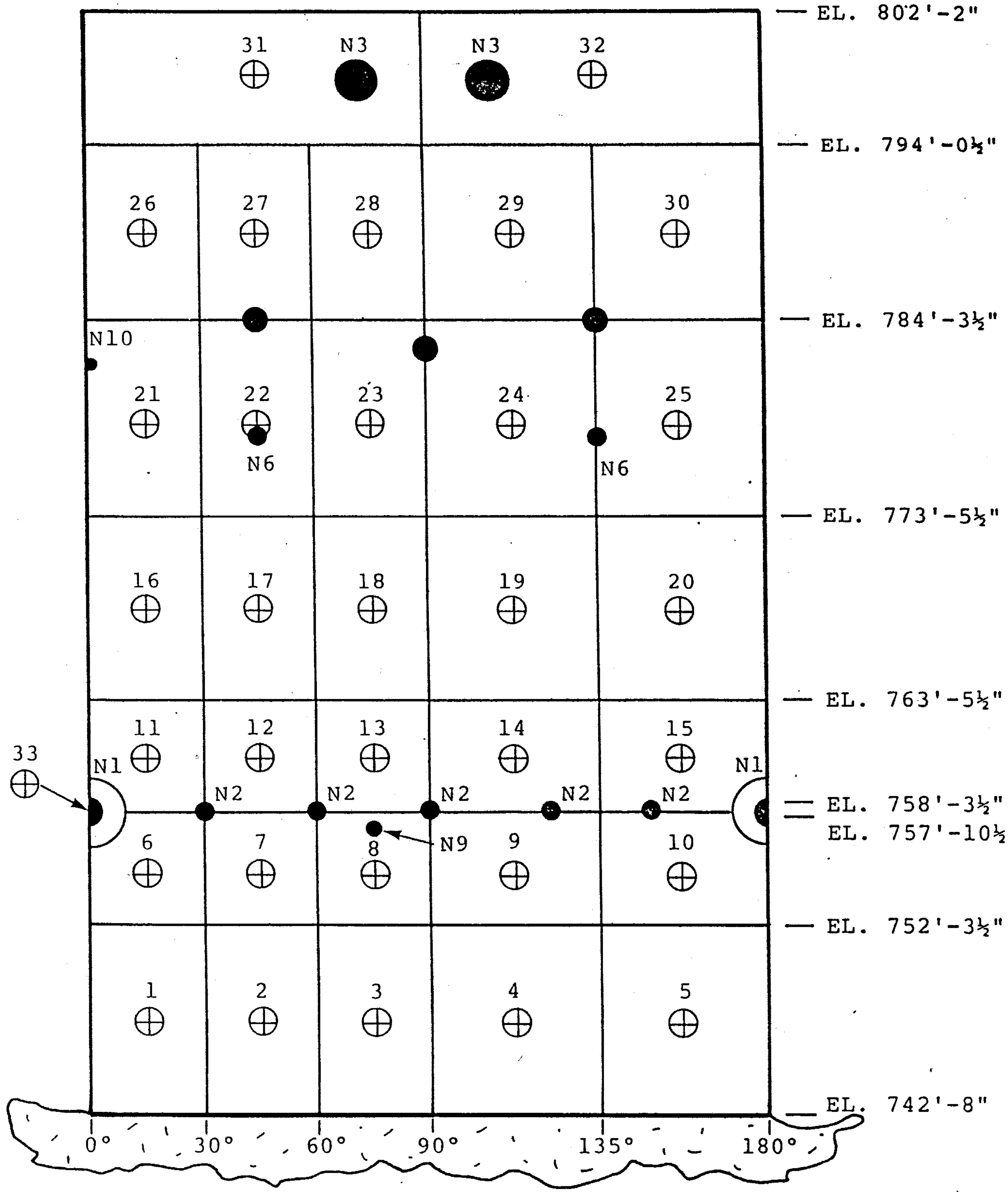

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FIGURE 6.2-85

ANNULUS NODALIZATION FOR RECIRCULATION OUTLET LINE BREAK

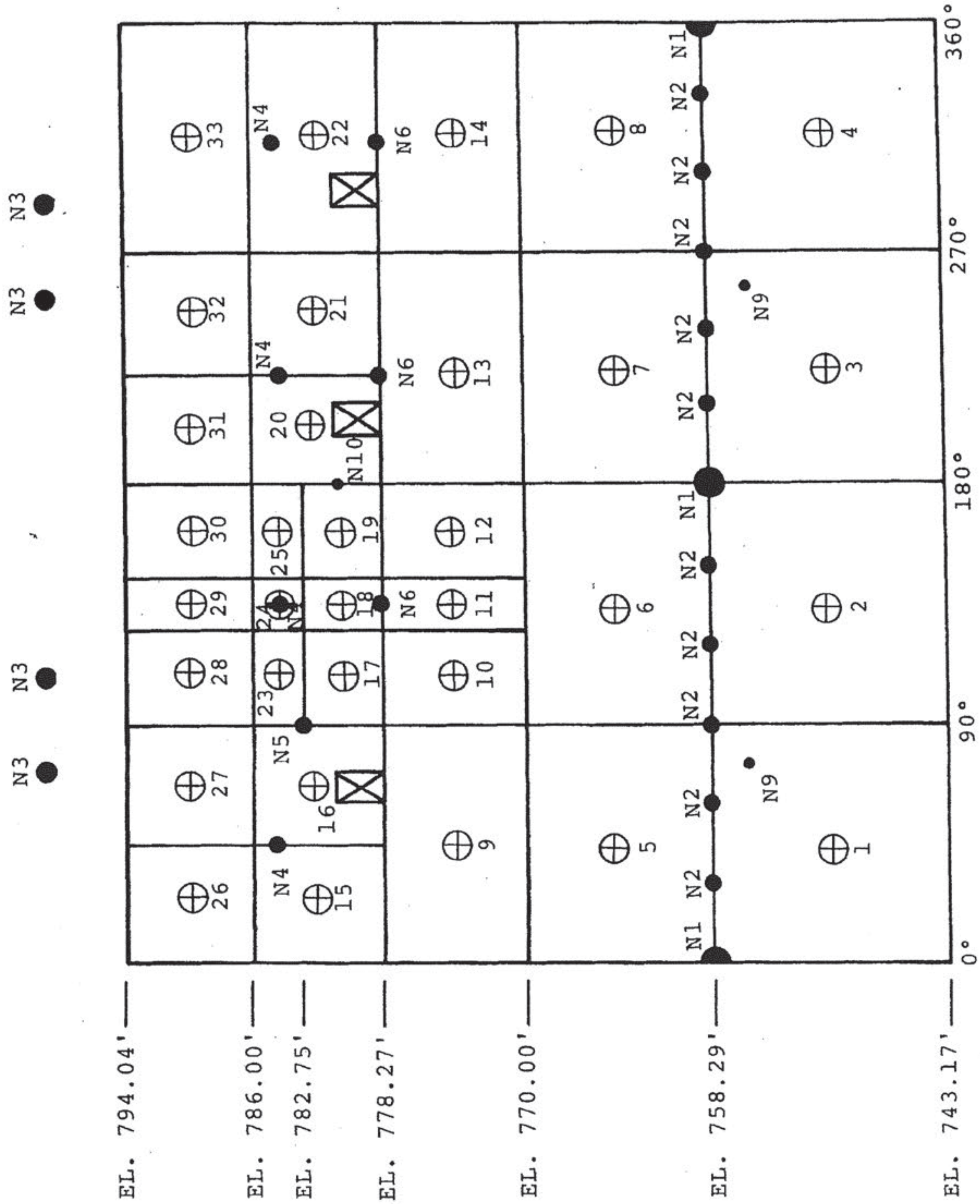

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FIGURE 6.2-86

ANNULUS NODALIZATION FOR
FEEDWATER LINE BREAK

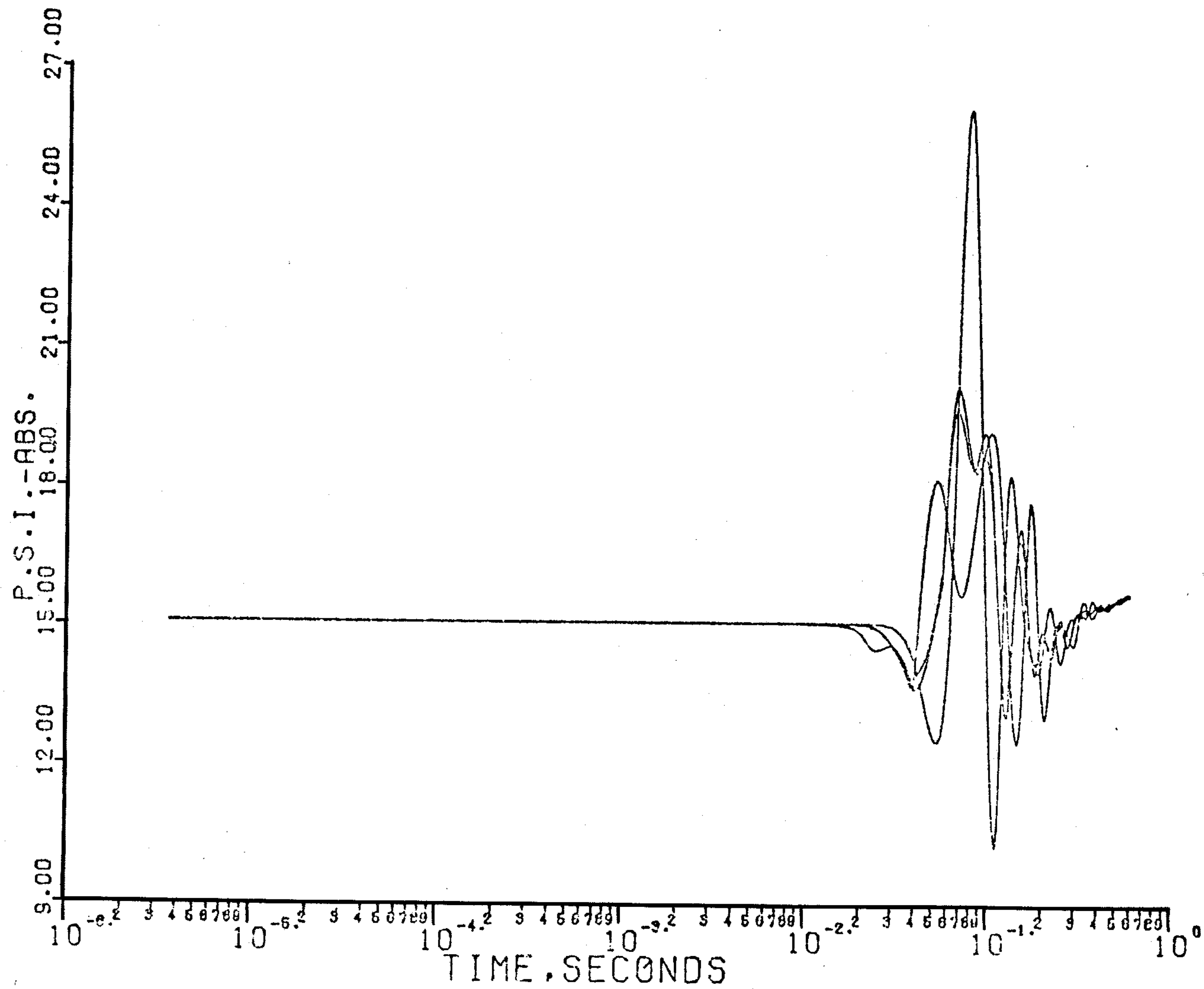

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FIGURE 6.2-87

RPV LOADING FEEDWATER LINE BREAK - TRANSIENT PRESSURES FOR VOLS. 1, 2, 3 AND 4

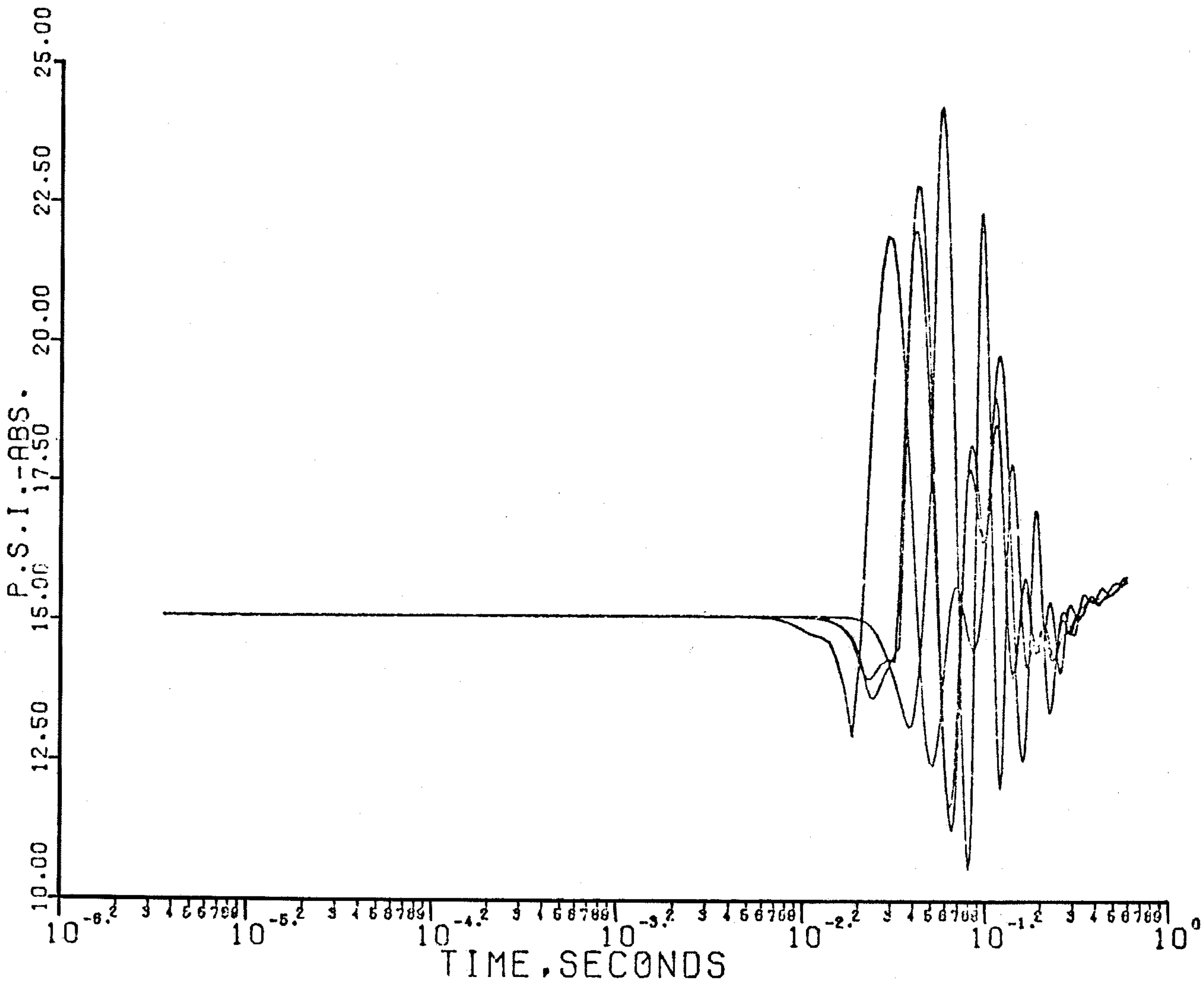

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FIGURE 6.2-88

RPV LOADING FEEDWATER LINE BREAK - TRANSIENT PRESSURES FOR VOLS. 5, 6, 7 AND 8

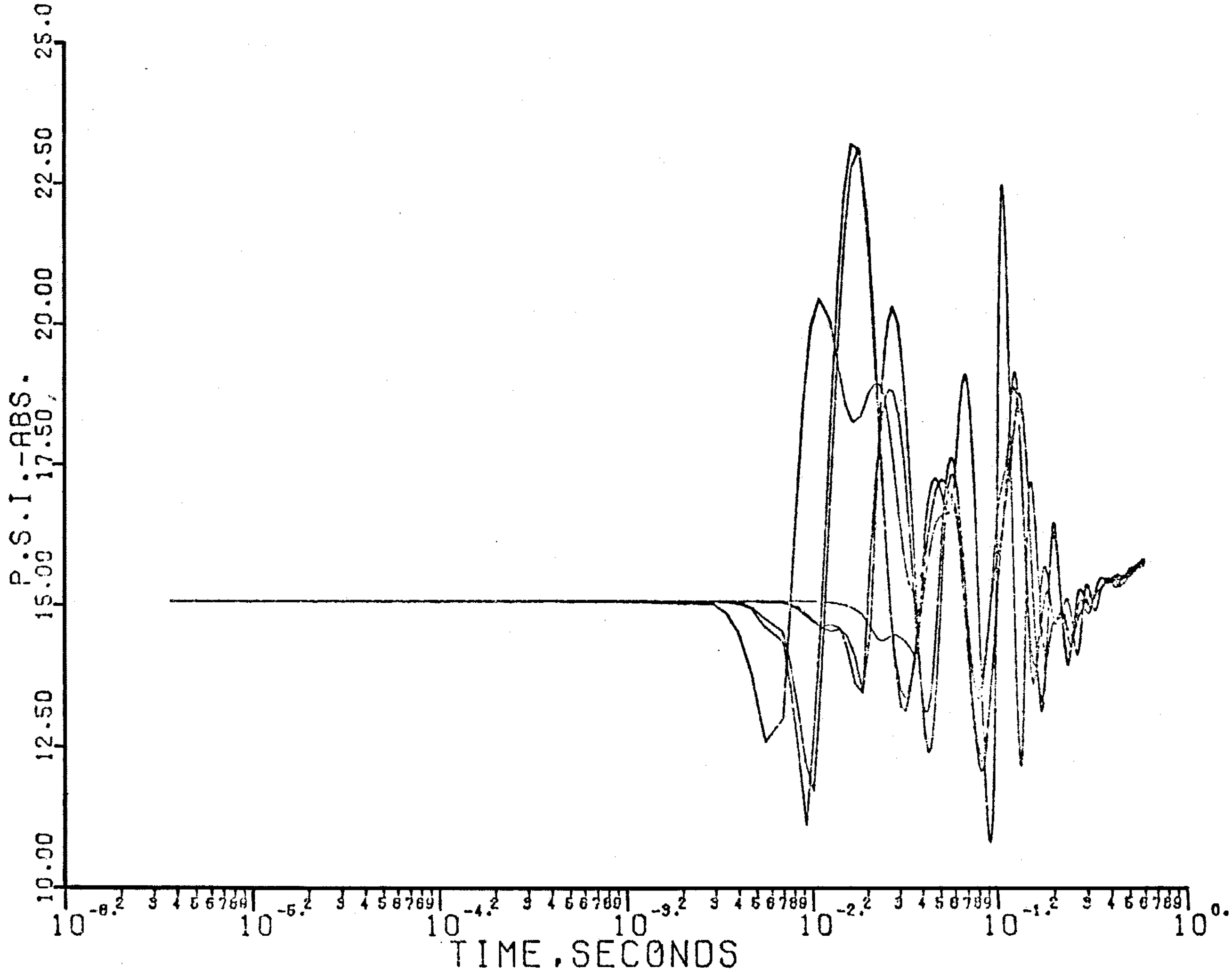

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FIGURE 6.2-89

RPV LOADING FEEDWATER LINE BREAK - TRANSIENT PRESSURES FOR VOLS. 9, 10, 11, 12, 13 AND 14

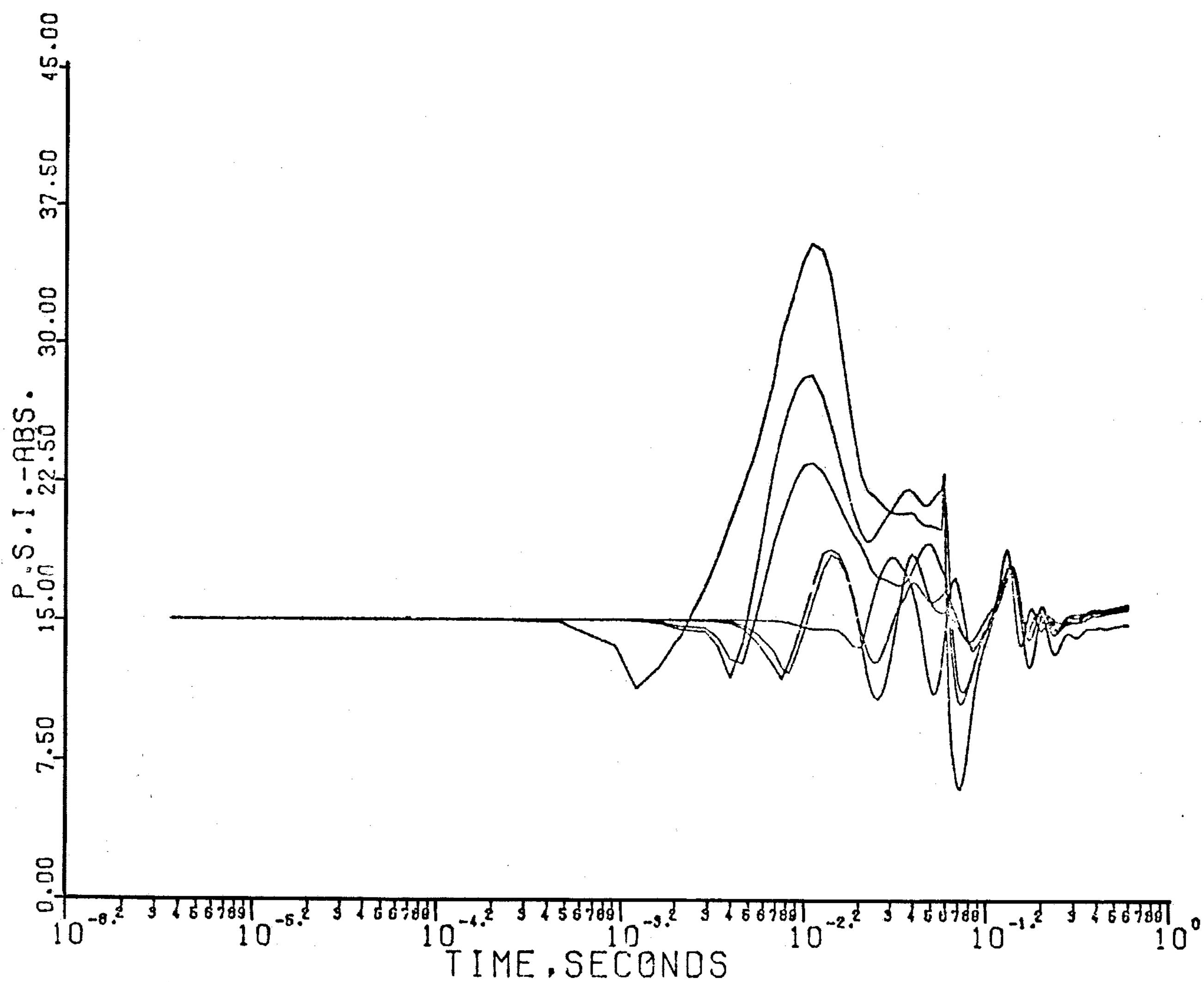

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FIGURE 6.2-90

RPV LOADING FEEDWATER LINE BREAK - TRANSIENT PRESSURES FOR VOLS. 15, 16, 17, 18, 19 AND 20

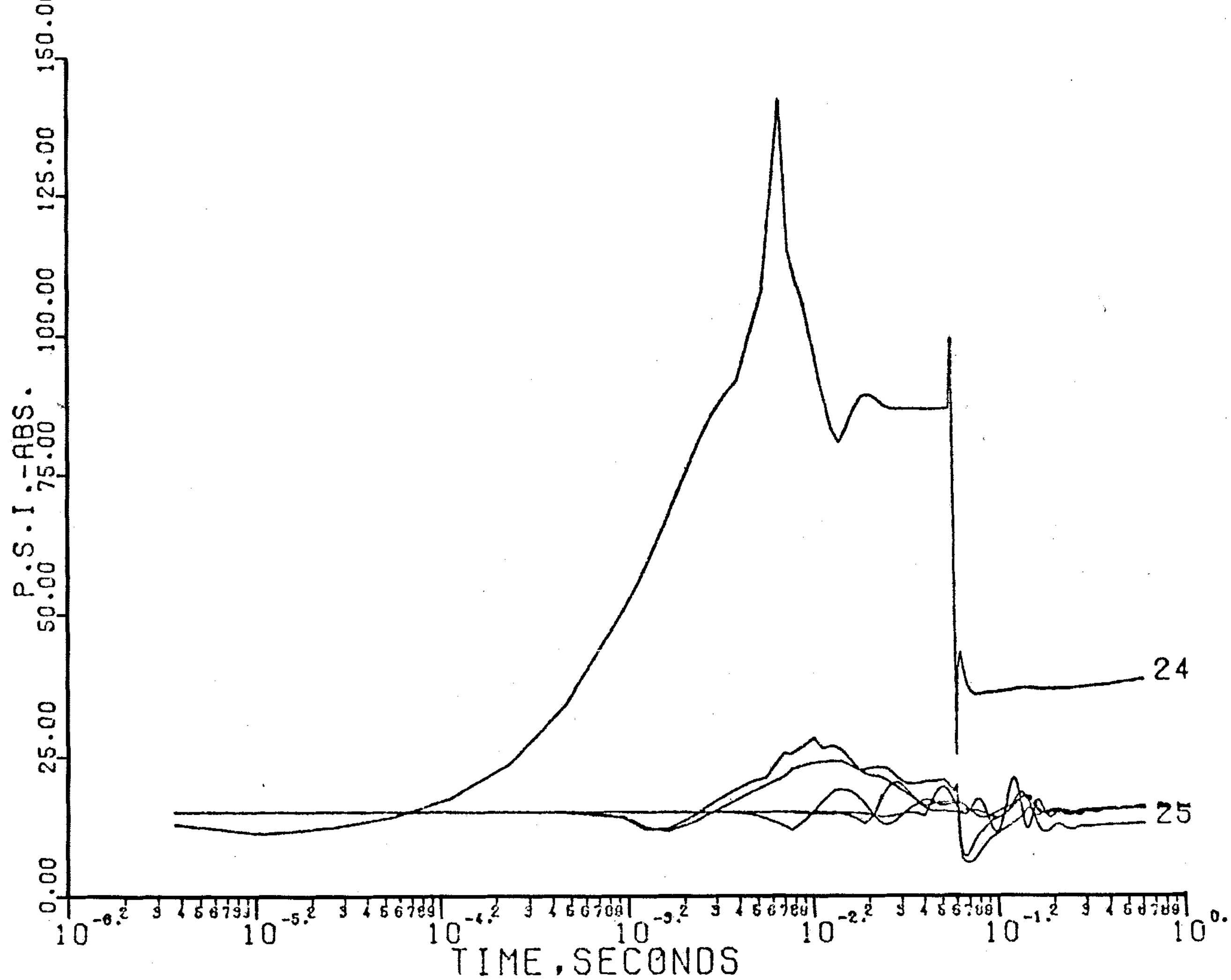

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FIGURE 6.2-91

RPV LOADING FEEDWATER LINE BREAK - TRANSIENT PRESSURES FOR VOLS. 20, 21, 22, 23, 24 AND 25

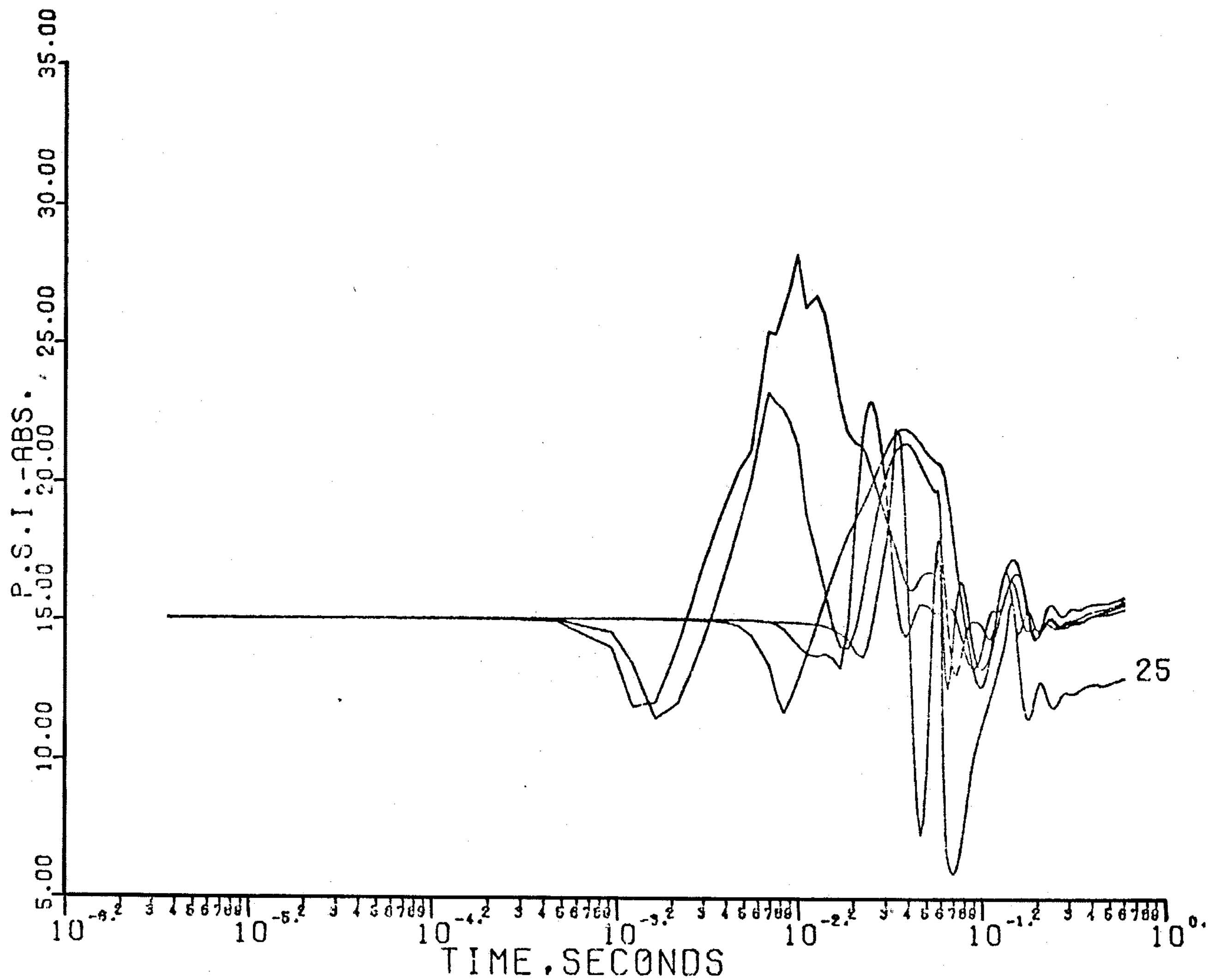

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FIGURE 6.2-92

RPV LOADING FEEDWATER LINE BREAK - TRANSIENT PRESSURES FOR VOLS. 25, 26, 27, 28 AND 29

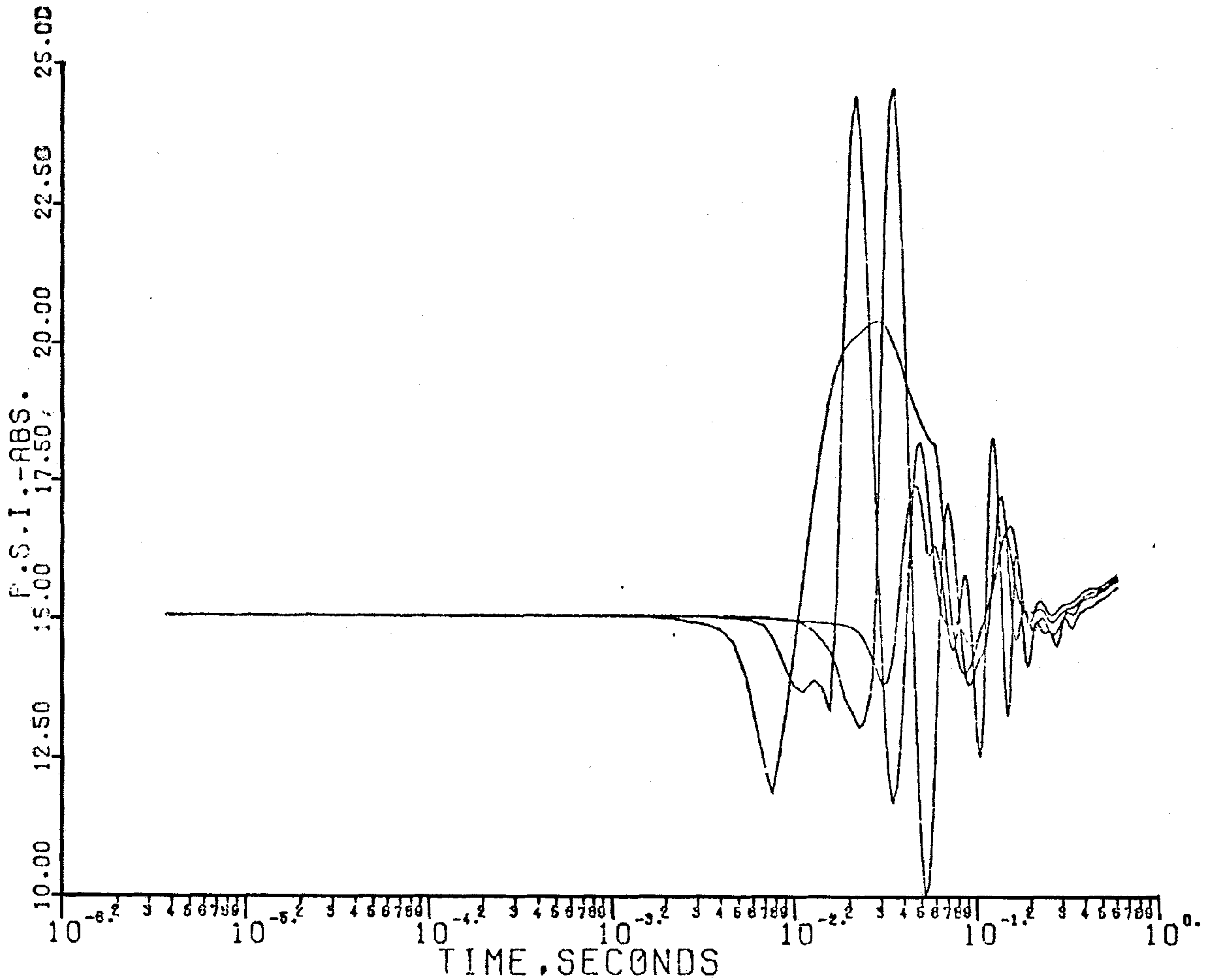

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FIGURE 6.2-93

RPV LOADING FEEDWATER LINE BREAK - TRANSIENT PRESSURES FOR VOLS. 30, 31, 32 AND 33

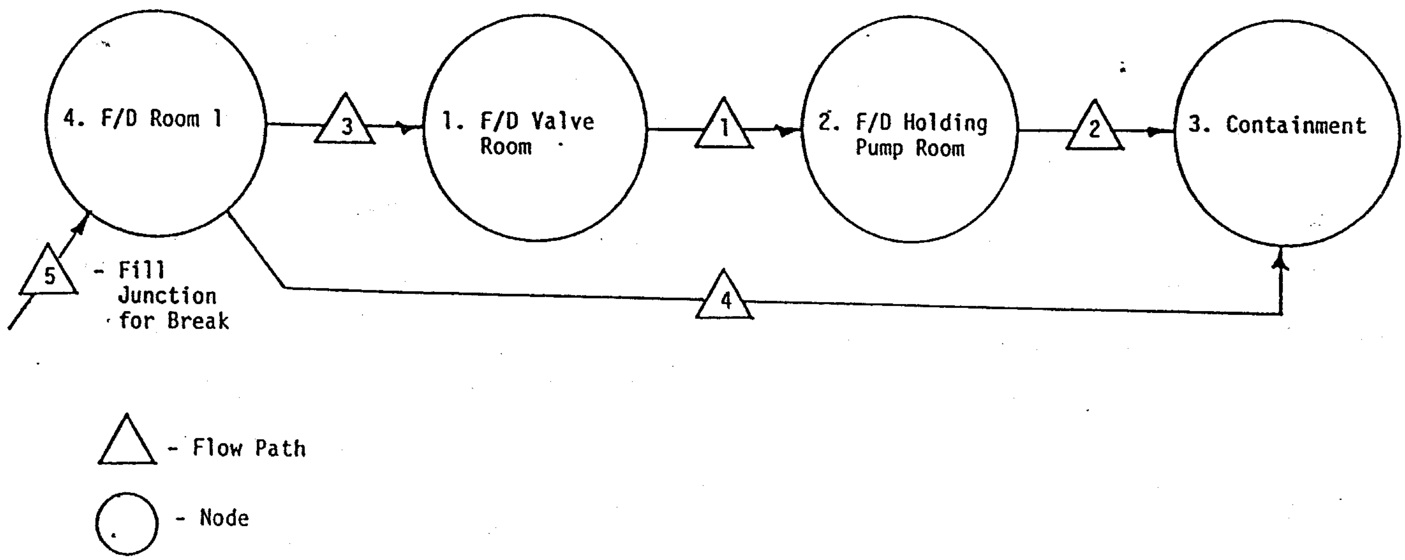

Figure 4: Flow Scheme for RWCU Line Break in F/D Rooms (Case 4)

NOTE:

See Table 6.2-37 for a description of the nodes and Table 6.2-38 for a description of the vent paths.

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FIGURE 6.2-94

NODALIZATION SCHEMATIC FOR RWCU LINE BREAK IN FILTER-DEMINERALIZER ROOMS

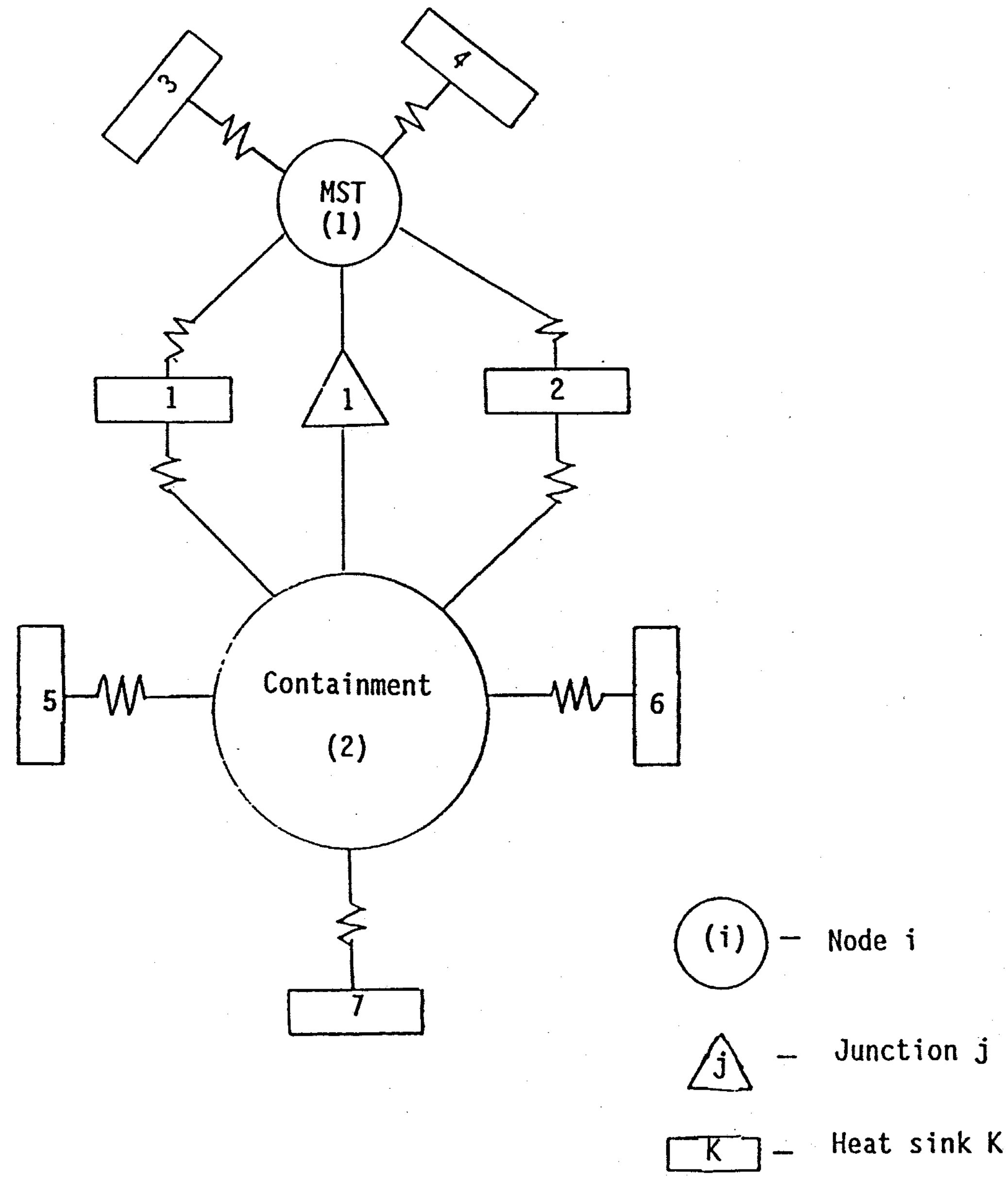
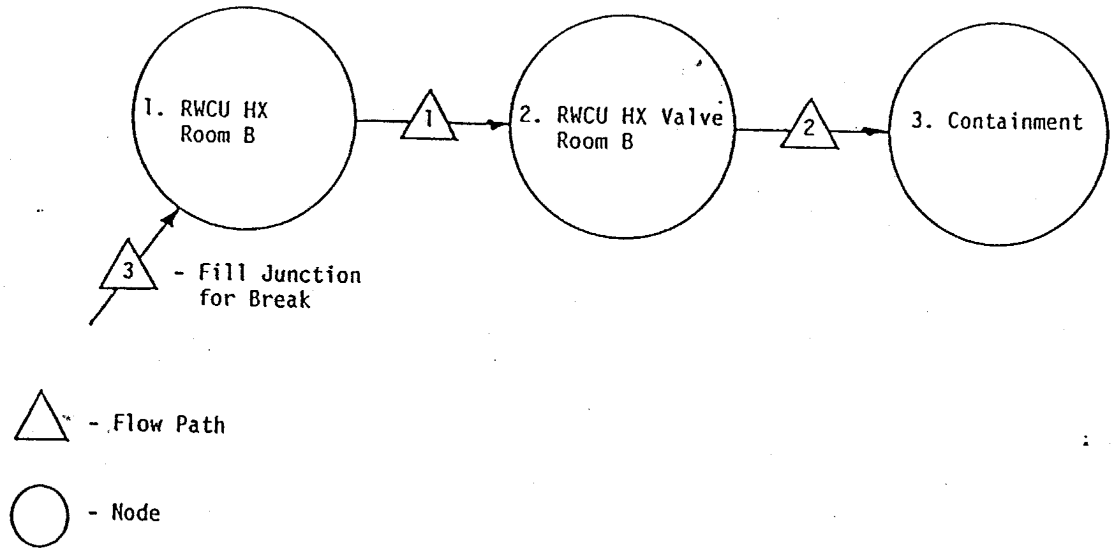


Figure 1
Nodalization Schematic

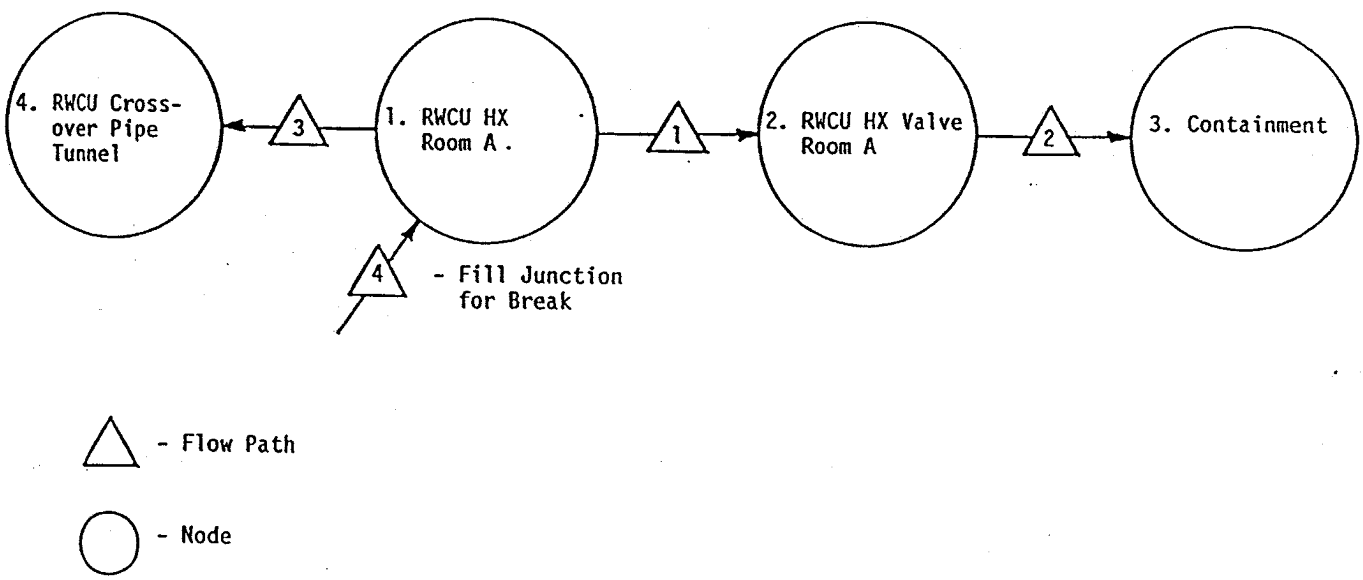
FIGURE 6.2-94A

NODALIZATION SCHEMATIC FOR RWCU LINE BREAK IN THE CONTAINMENT PIPE TUNNEL

Figure 6.2-95
Deleted



Flow Scheme for RWCU Line Break in RWCU Heat Exchanger Room B (Case 5)



Flow Scheme for RWCU Line Break in RWCU Heat Exchanger Room A (Case 2)

NOTE:

See Tables 6.2-25 and 6.2-27 for a description of the nodes and Tables 6.2-26 and 6.2-28 for a description of the vent paths.

CLINTON POWER STATION UPDATED SAFETY ANALYSIS REPORT
FIGURE 6.2-96 NODALIZATION SCHEMATIC FOR RWCU LINE BREAK IN HEAT EXCHANGER ROOMS NOS. 1 AND 2

Figures 6.2-97 and 6.2-98
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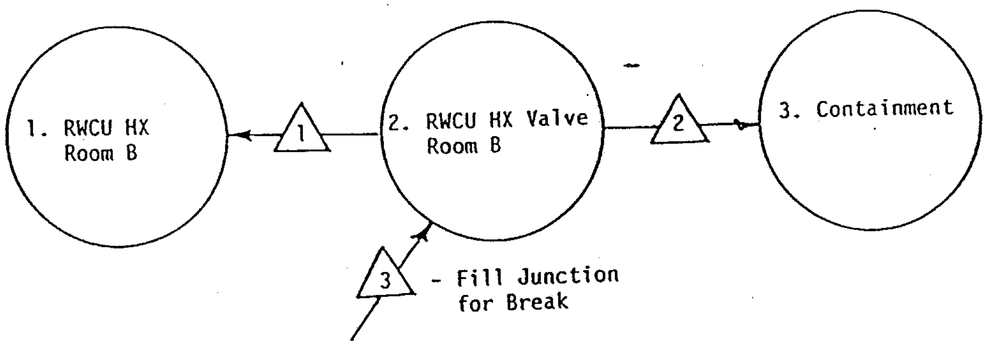
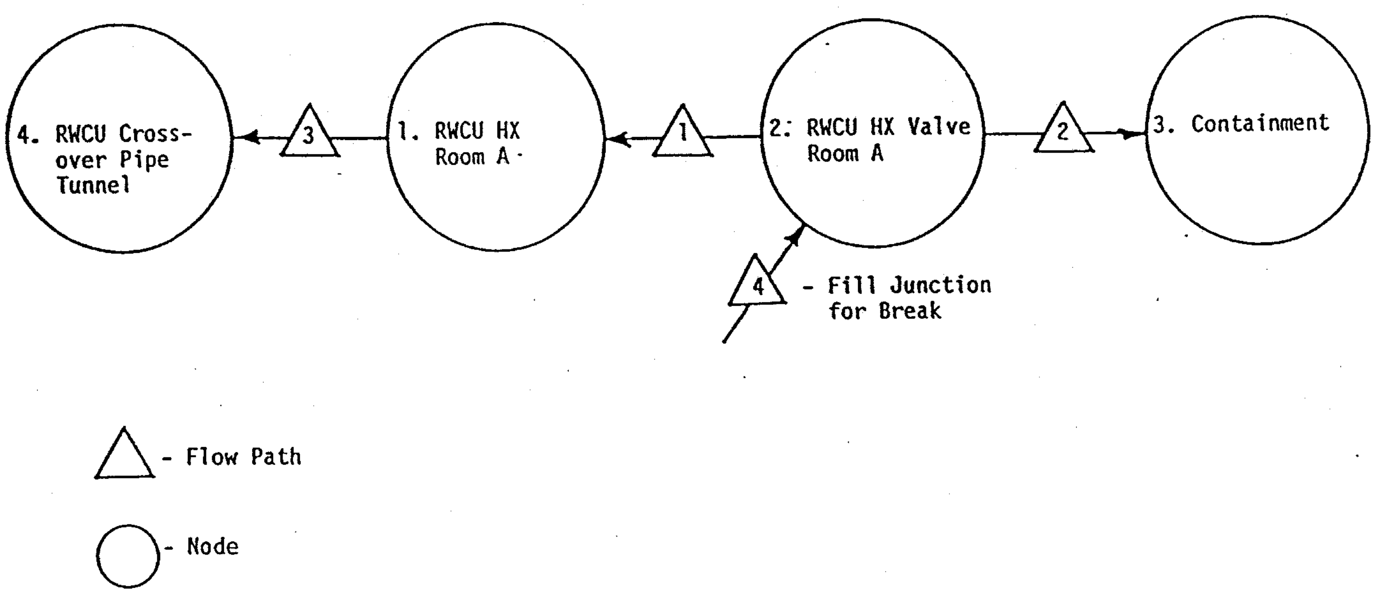


Flow Scheme for RWCU Line Break in RWCU
Heat Exchanger Valve Room B (Case 8)



Flow Scheme for RWCU Line Break in RWCU
Heat Exchanger Valve Room A (Case 6)

NOTE:

See Tables 6.2-29 and 6.2-31 for a description of the nodes and Tables 6.2-30 and 6.2-32 for a description of the vent paths.

CLINTON POWER STATION UPDATED SAFETY ANALYSIS REPORT
FIGURE 6.2-99 NODALIZATION SCHEMATIC FOR RWCU LINE BREAK IN RWCU VALVE ROOMS NOS. 1 AND 2

Figures 6.2-100 and 6.2-101
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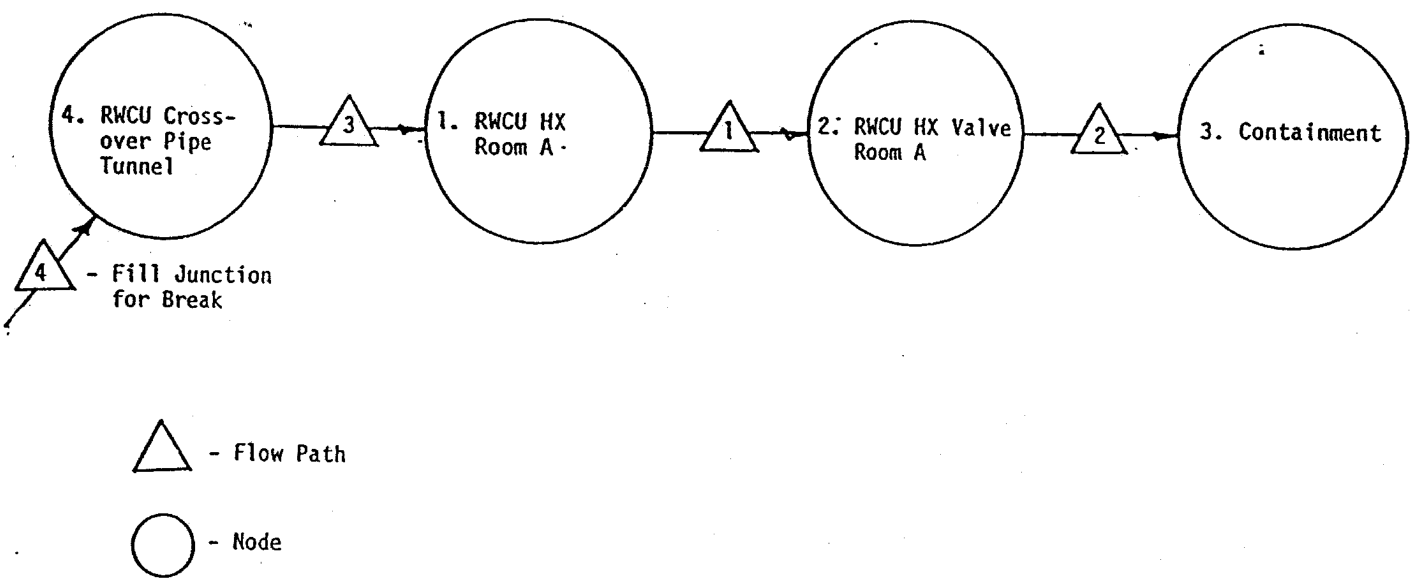

Figure 7: Flow Scheme for RWCU Line Break in RWCU Crossover Pipe Tunnel (Case 7)

NOTE:

See Table 6.2-33 for a description of the nodes and Table 6.2-34 for a description of the vent paths.

CLINTON POWER STATION
UPDATED SAFETY ANALYSIS REPORT

FIGURE 6.2-102

NODALIZATION SCHEMATIC FOR RWCU LINE BREAK IN RWCU CROSSOVER PIPE TUNNEL

Figure 6.2-103
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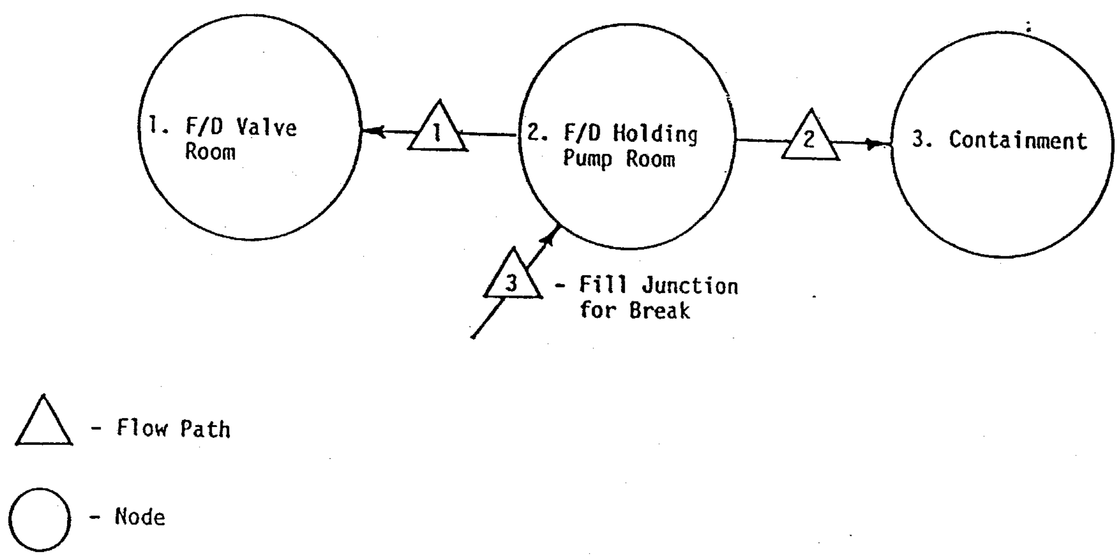

Figure 3: Flow Scheme for RWCU Line Break in Holding Pump Room (Case 3)

NOTE:

See Table 6.2-35 for a description of the nodes and Table 6.2-36 for a description of the vent paths.

CLINTON POWER STATION UPDATED SAFETY ANALYSIS REPORT
FIGURE 6.2-104 NODALIZATION SCHEMATIC FOR RWCU LINE BREAK IN FILTER-DEMINERALIZER HOLDING PUMP ROOM

Figures 6.2-105 and 6.2-107
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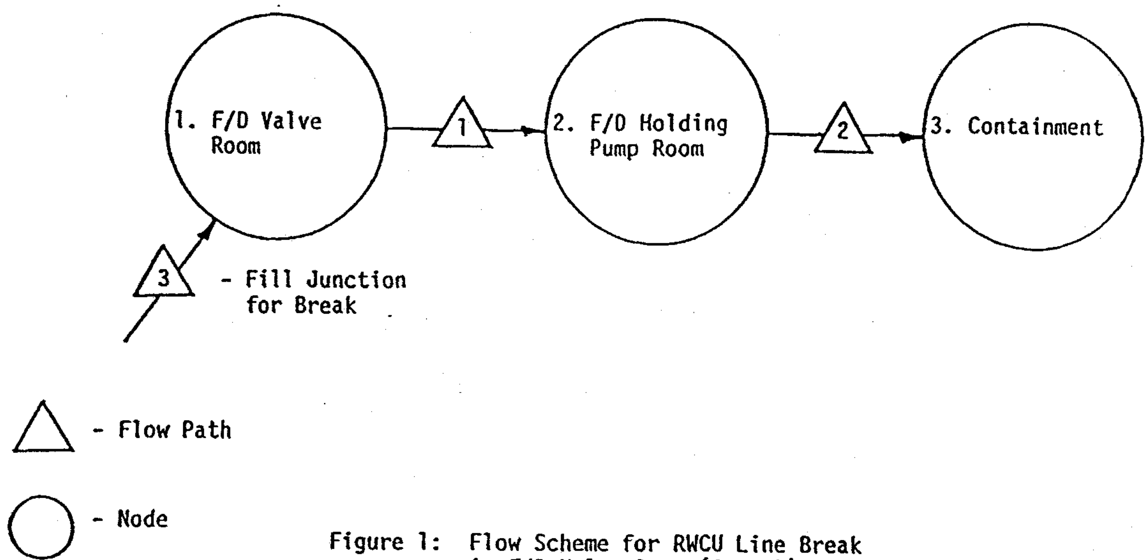

Figure 1: Flow Scheme for RWCU Line Break in F/D Valve Room (Case 1)

NOTE:

See Table 6.2-39 for a description of the nodes and Table 6.2-40 for a description of the vent paths.

CLINTON POWER STATION
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FIGURE 6.2-108

NODALIZATION SCHEMATIC FOR RWCU LINE BREAK IN FILTER-DEMINERALIZER VALVE ROOM

Figure 6.2-109
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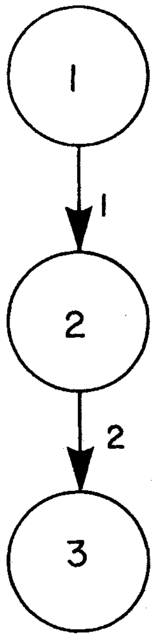

NOTE:

See Table 6.2-12 for a description of the nodes and Table 6.2-19 for a description of the vent paths.

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UPDATED SAFETY ANALYSIS REPORT

FIGURE 6.2-110

NODALIZATION SCHEMATIC FOR HEAD SPRAY LINE BREAK IN DRYWELL HEAD CAVITY

Figures 6.2-111 through 6.2-114
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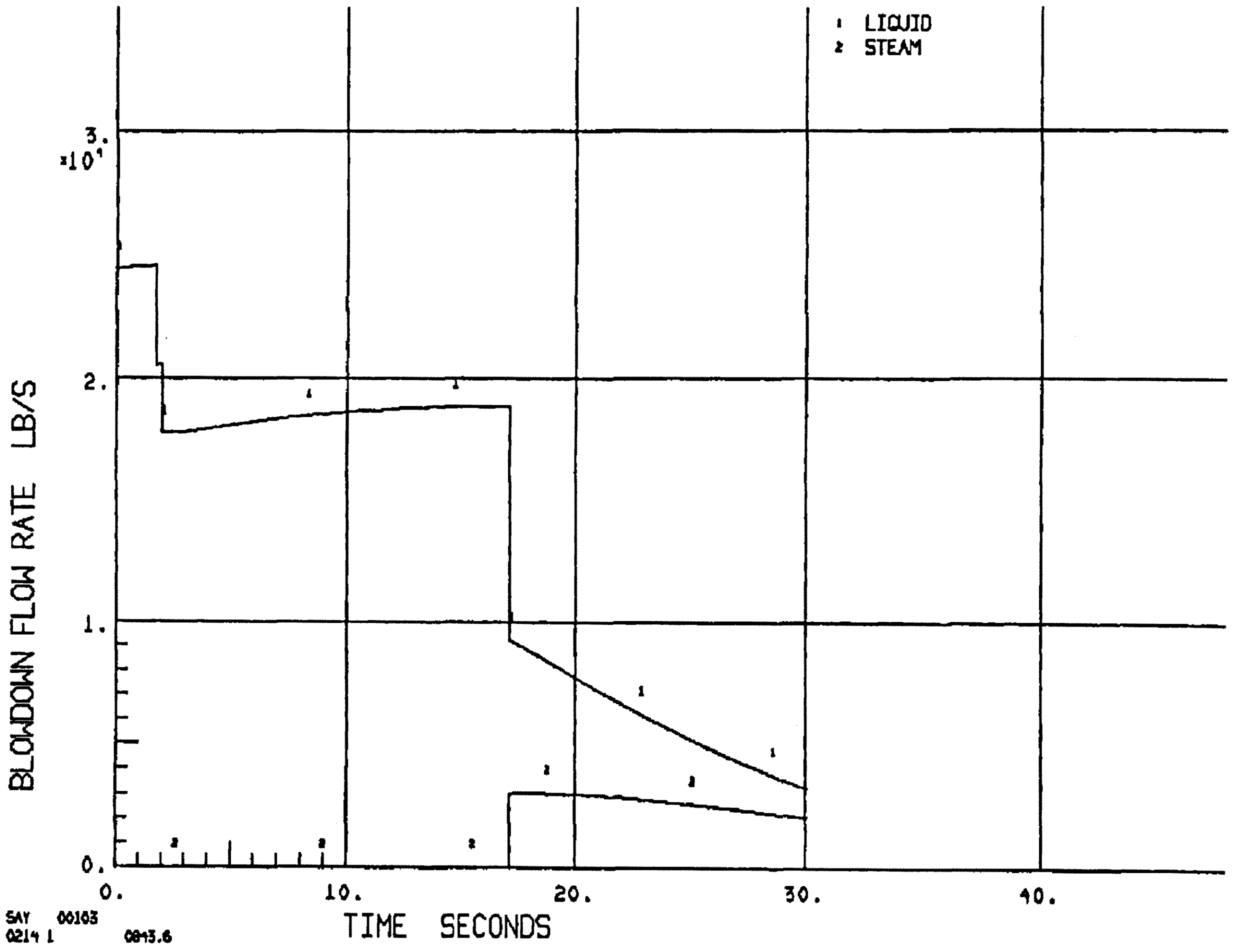

FIGURE 6.2-115. VESSEL BLOWDOWN FLOW RATE FOLLOWING A RECIRCULATION LINE BREAK

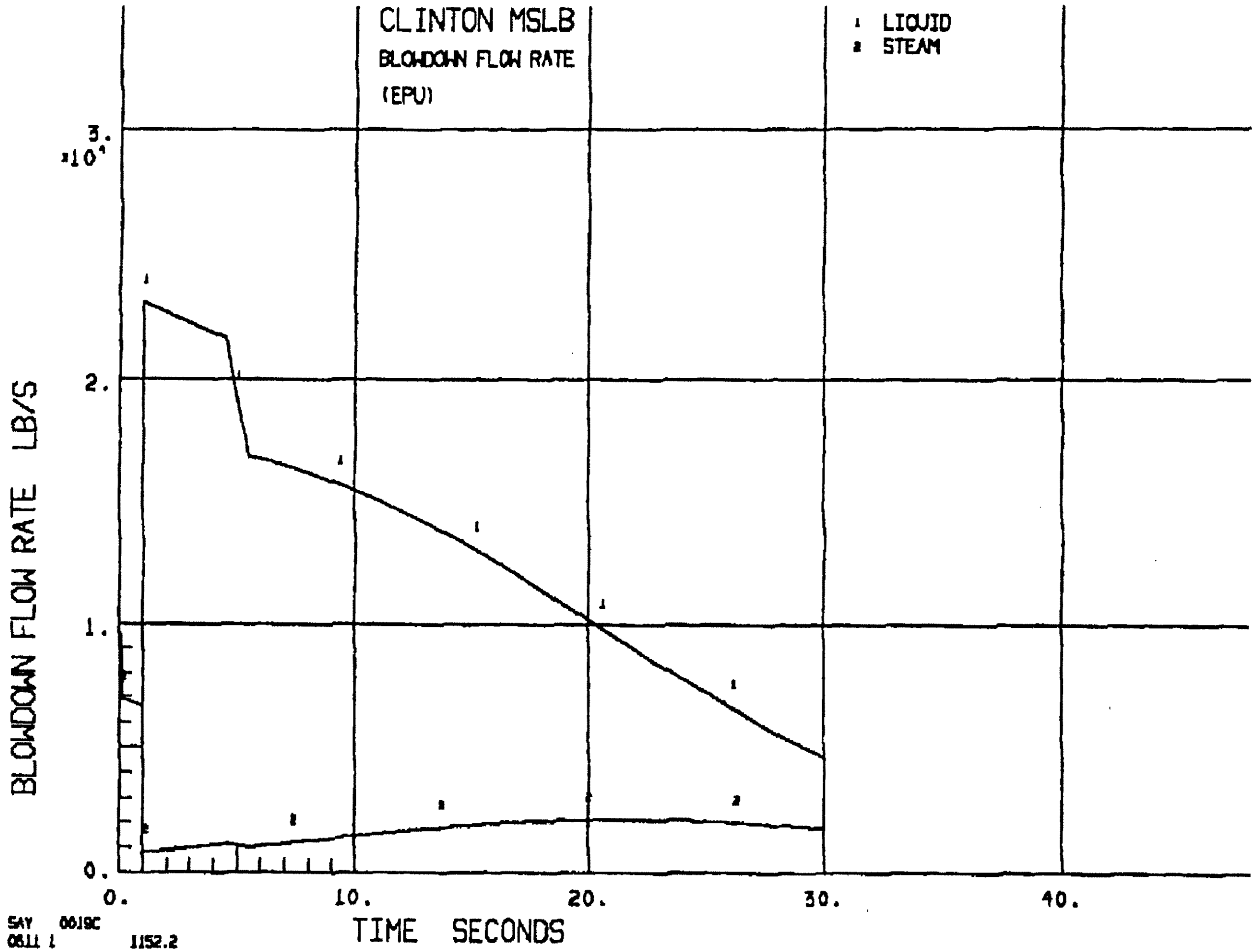

FIGURE 6.2-116. VESSEL BLOWDOWN FLOW RATES FOLLOWING A MAIN STEAM LINE BREAK

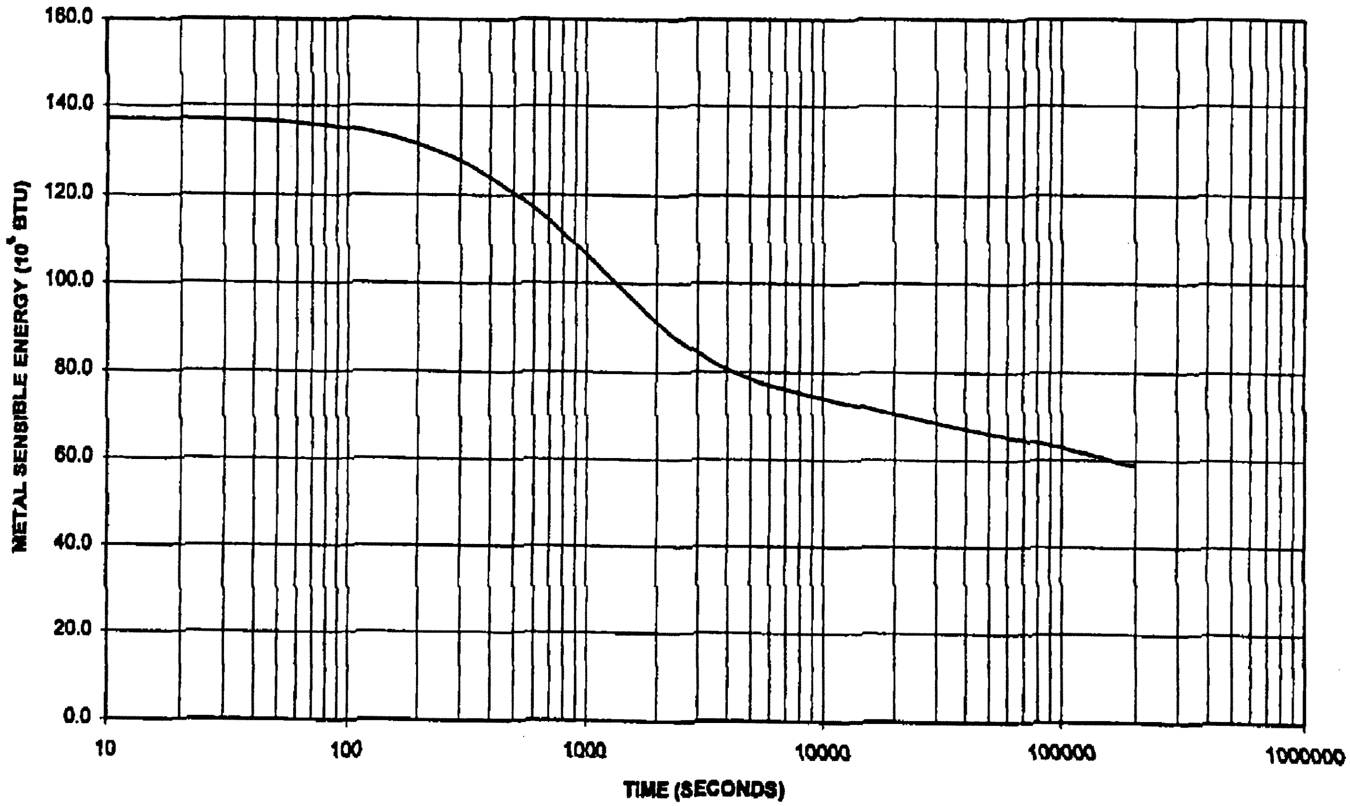

FIGURE 6.2-117. SENSIBLE ENERGY IN THE REACTOR PRESSURE VESSEL AND INTERNAL METALS FOLLOWING A MAIN STEAM LINE BREAK

Figures 6.2-118 and 6.2-119 have been deleted.

Security - Related Information Figure Withheld Under 10 CFR 2.390

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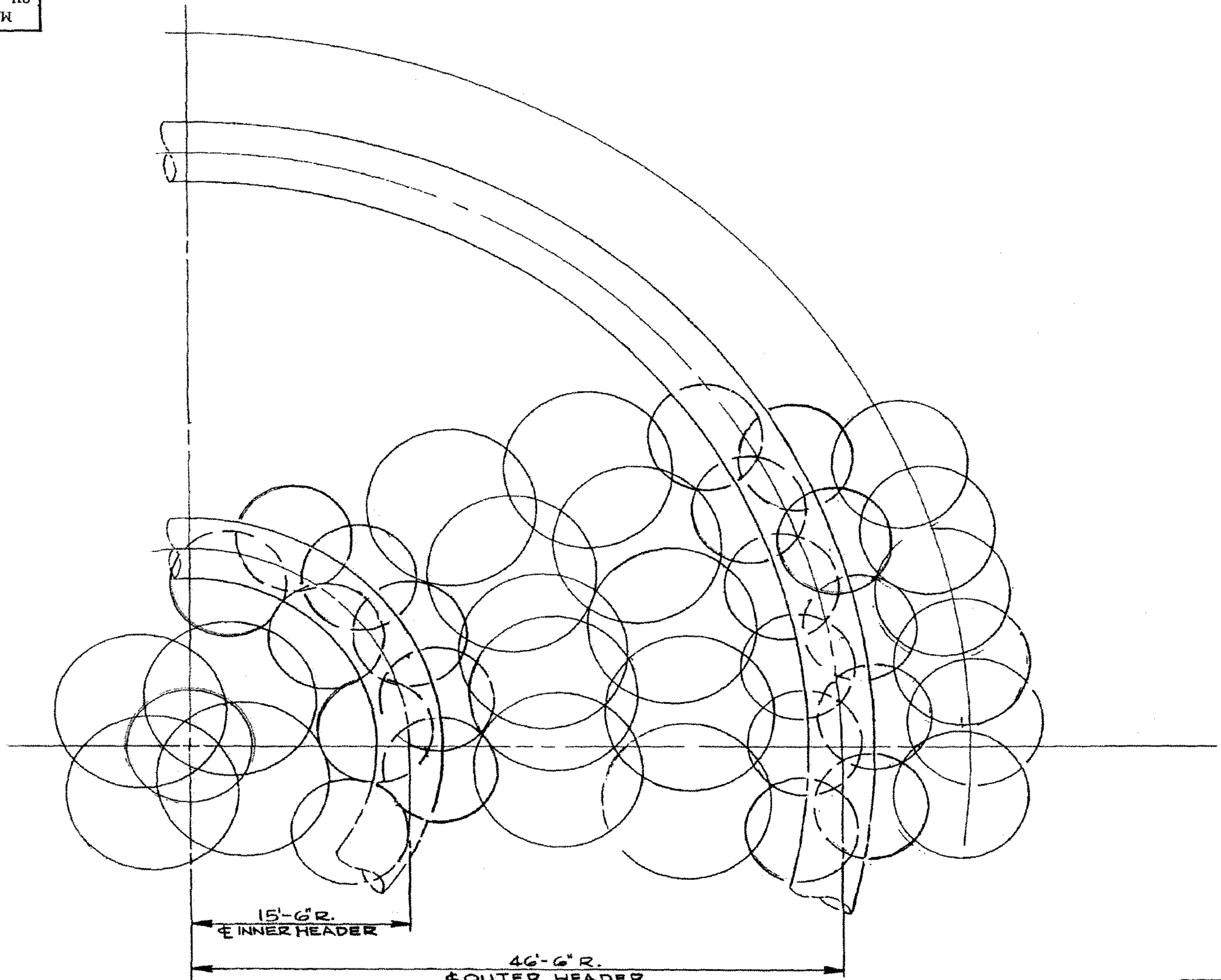
FIGURE 6.2-120

SPRAY NOZZLE COVERAGE CHARTS IN R,Z PLANE DRAWN WITH ALL NOZZLE ORIENTATIONS AND BROUGHT INTO ONE PLANE

Revision 12
January 2007



NOTE
THIS FIGURE REPRESENTS
SECTION "A-A"
OF FIGURE 6.2-120.

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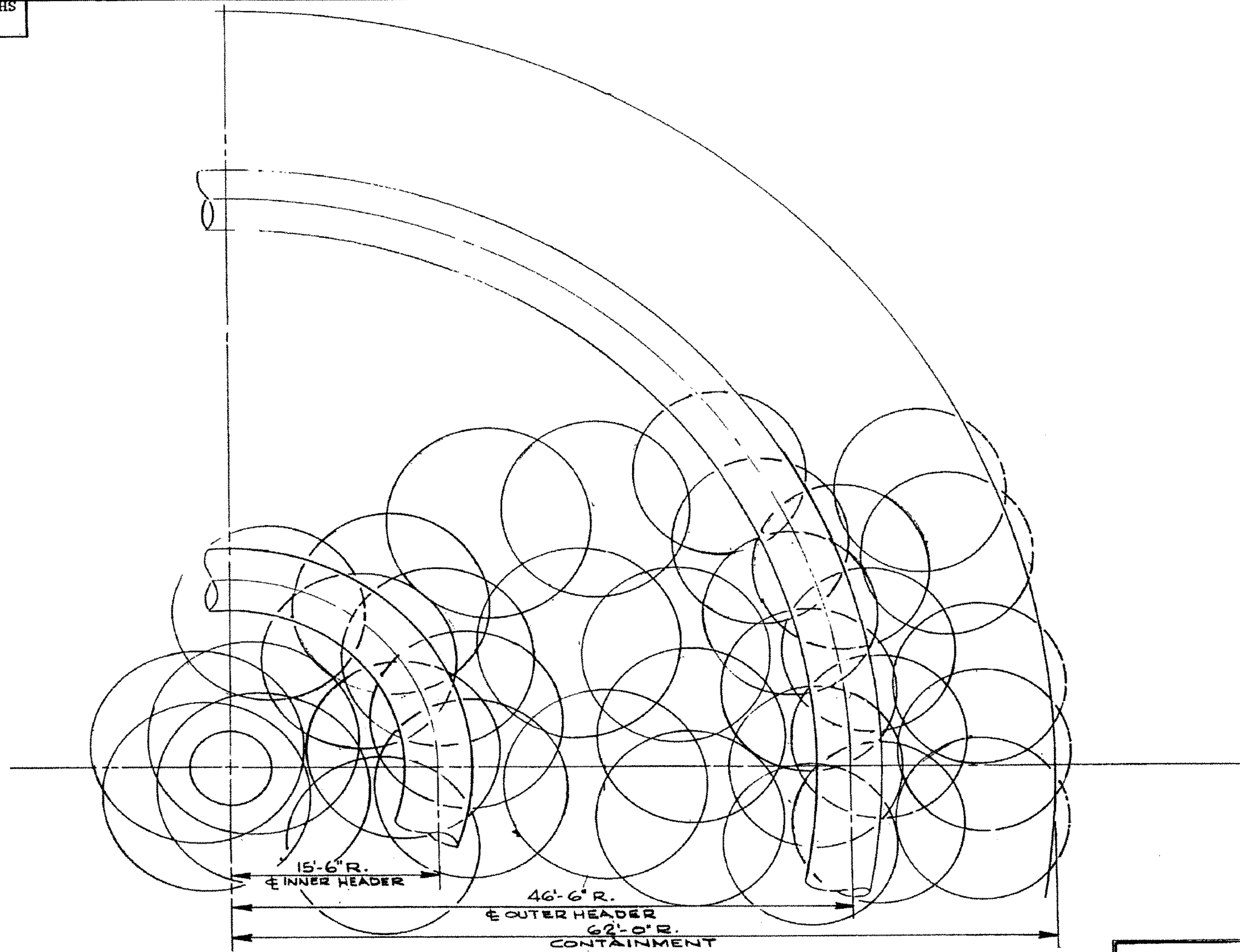
UPDATED SAFETY ANALYSIS REPORT

FIGURE 6.2-121

SPRAY SYSTEM AREAL COVERAGE IN A PLANE
10 FEET BELOW THE RESPECTIVE
HEADERS

Revision 12
January 2007



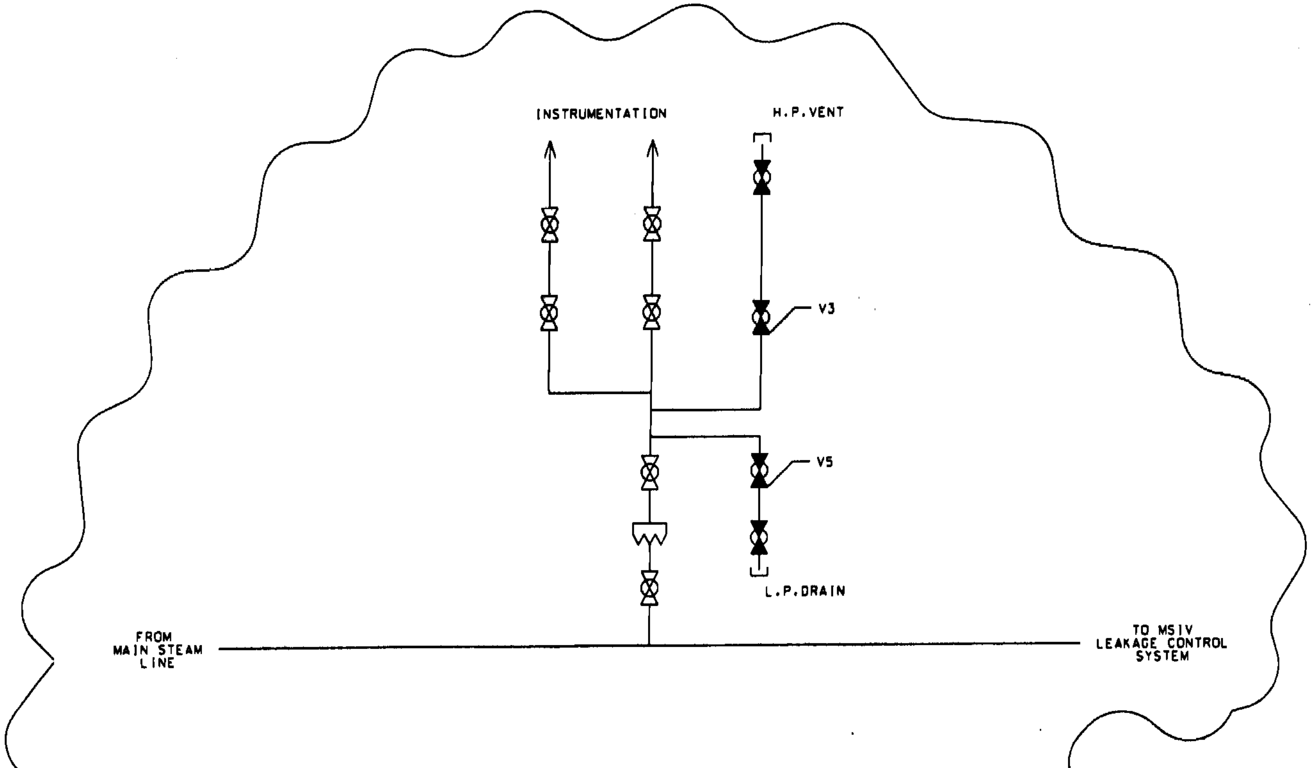
NOTE
THIS FIGURE REPRESENTS SECTION "B-B" OF FIGURE 6.2-120.

CLINTON POWER STATION
UPDATED SAFETY ANALYSIS REPORT

FIGURE 6.2-122

SPRAY SYSTEM AREAL COVERAGE AT ELEVATION 862'-6"

REVISION 9
NOVEMBER 2000



PENETRATION	MAIN STEAM LINE	V3	V5
1MC-5	1C	1E32-F327C	1E32-F330A
1MC-6	1A	1E32-F327A	1E32-F329A
1MC-7	1D	1E32-F327D	1E32-F330C
1MC-8	1B	1E32-F327B	1E32-F329C

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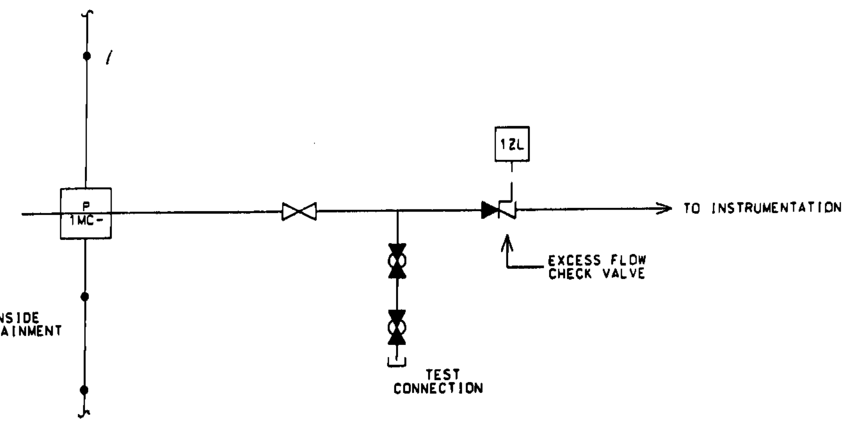
UPDATED SAFETY ANALYSIS REPORT

FIGURE 6.2-123

TYPICAL CONTAINMENT PENETRATIONS

(SHEET 1 OF 3)

REVISION 9
NOVEMBER 2000



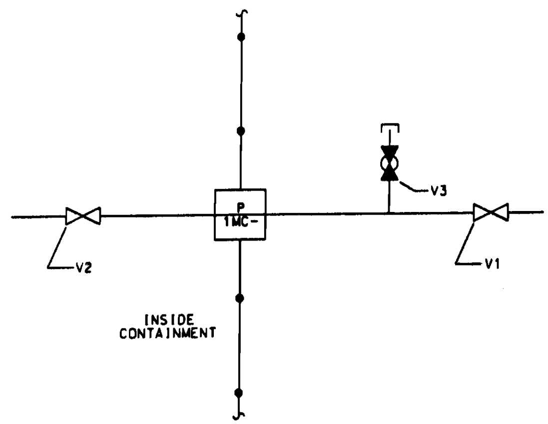
PENETRATION	EXCESS FLOW CHECK VALVE	TEST CONNECTION VALVES
1MC-156	1VG056B	NONE
1MC-165	1VR016A 1VR016B 1VR018A	NONE
1MC-167	1VG057B	NONE
1MC-168	1VR018B	NONE
1MC-177	1E51-F377B	1E51-F437A 1E51-F437B
1MC-179	1E22-F332	1E22-F381A 1E22-F381B
1MC-180	1E22-F330	NONE
1MC-200	1E51-F377A	NONE

CLINTON POWER STATION
UPDATED SAFETY ANALYSIS REPORT

FIGURE 6.2-123

TYPICAL CONTAINMENT PENETRATIONS

(SHEET 2 OF 3)



PENETRATION	V1	V2	V2
1MC-49	0RA026	0RA027	NONE
1MC-50	0MC009	0MC010	1MC011
1MC-69	1RE022	1RE021	NONE
1MC-70	1RF022	1RF021	NONE
1MC-169	1VR036 1VR041	1VR035 1VR040	NONE

CLINTON POWER STATION
UPDATED SAFETY ANALYSIS REPORT

FIGURE 6.2-123

TYPICAL CONTAINMENT PENETRATIONS

(SHEET 3 OF 3)

Figure 6.2-124
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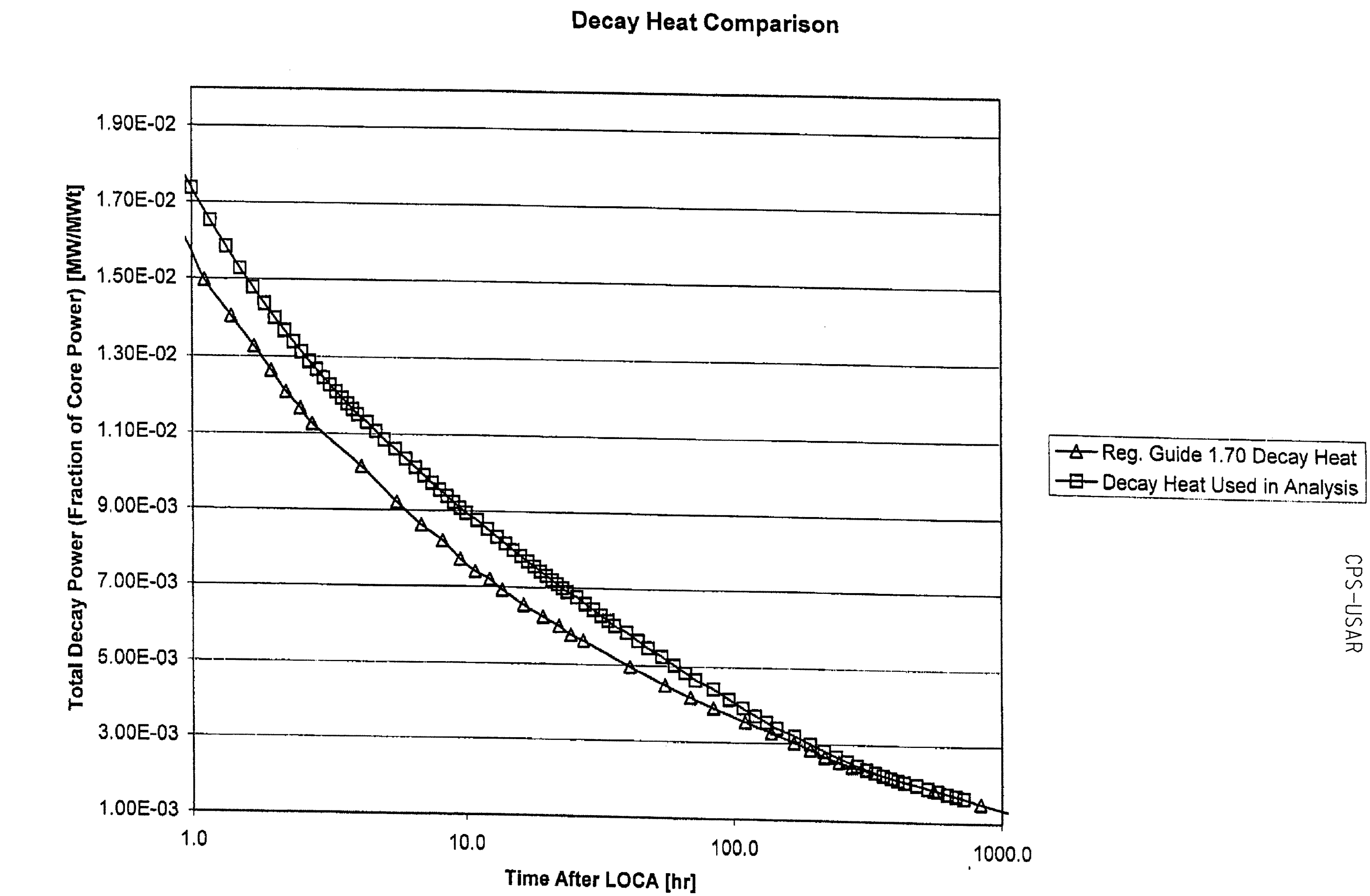

FIGURE 6.2-125. THE TOTAL RESIDUAL DECAY POWER AS A FRACTION OF OPERATING POWER VERSUS

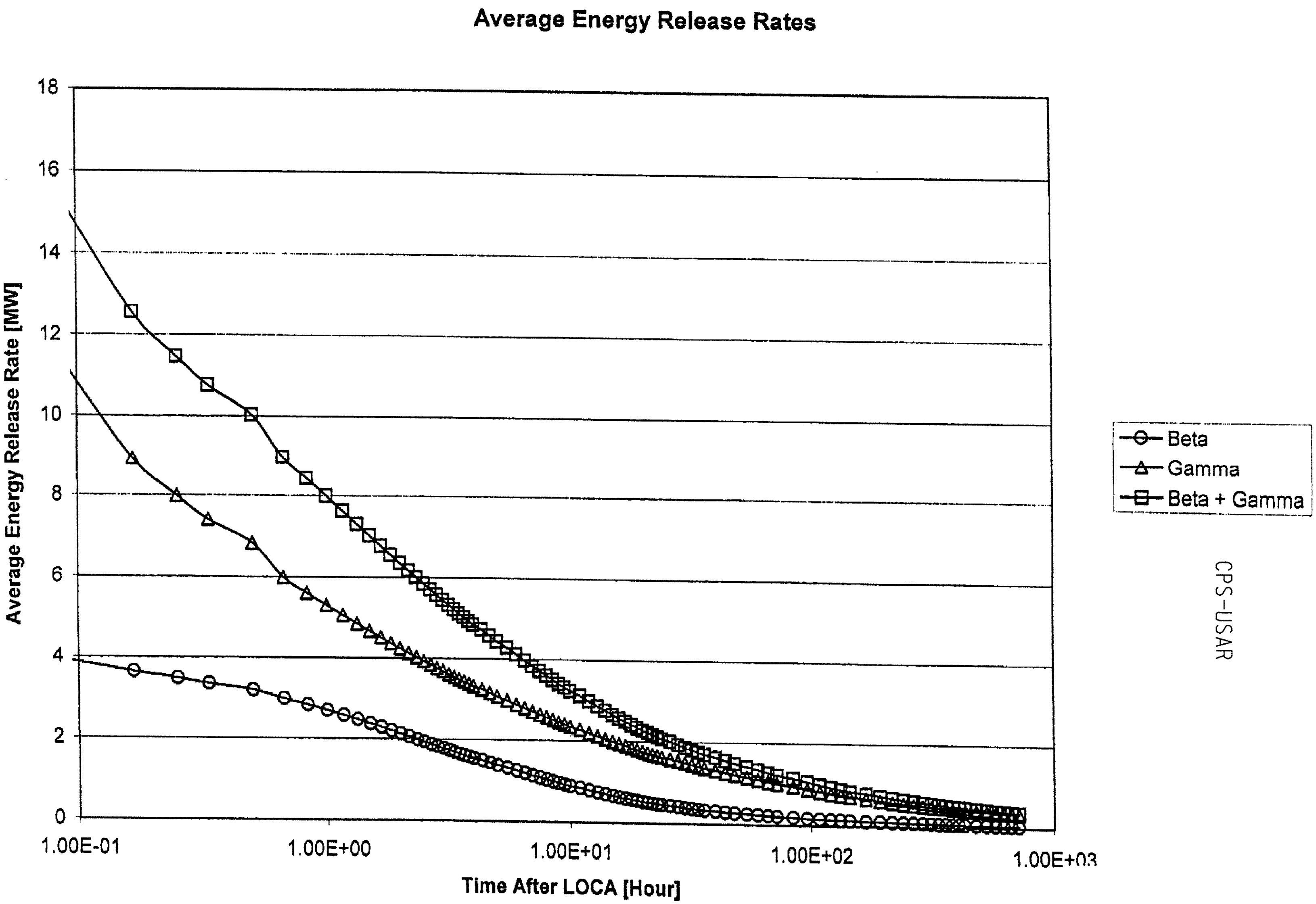

FIGURE 6.2-126. THE BETA, GAMMA, AND BETA PLUS GAMMA ENERGY AS A FUNCTION OF TIME

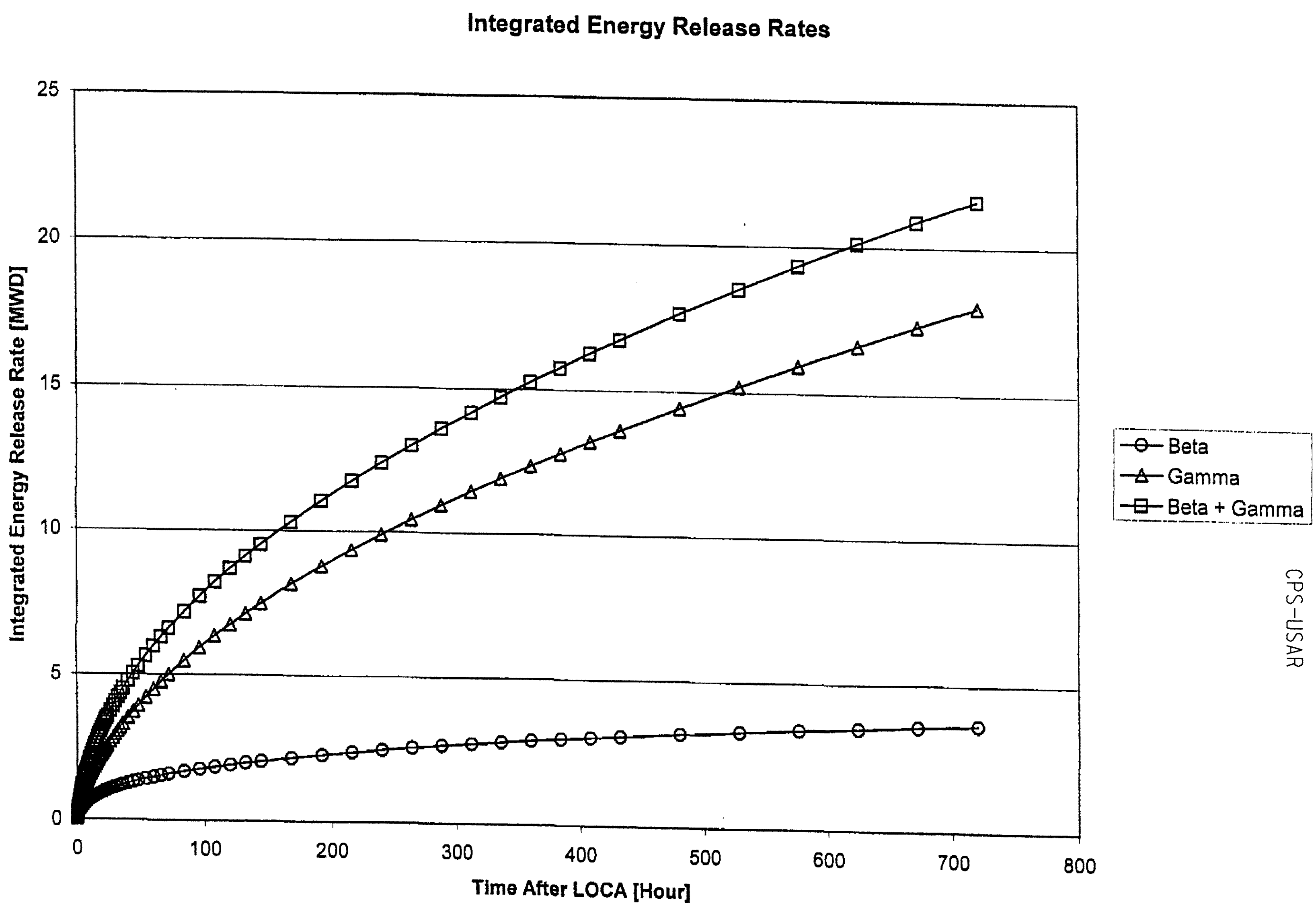

FIGURE 6.2-127. THE INTEGRATED ENERGY RELEASE OF BETA, GAMMA, AND BETA PLUS GAMMA AS A FUNCTION OF TIME

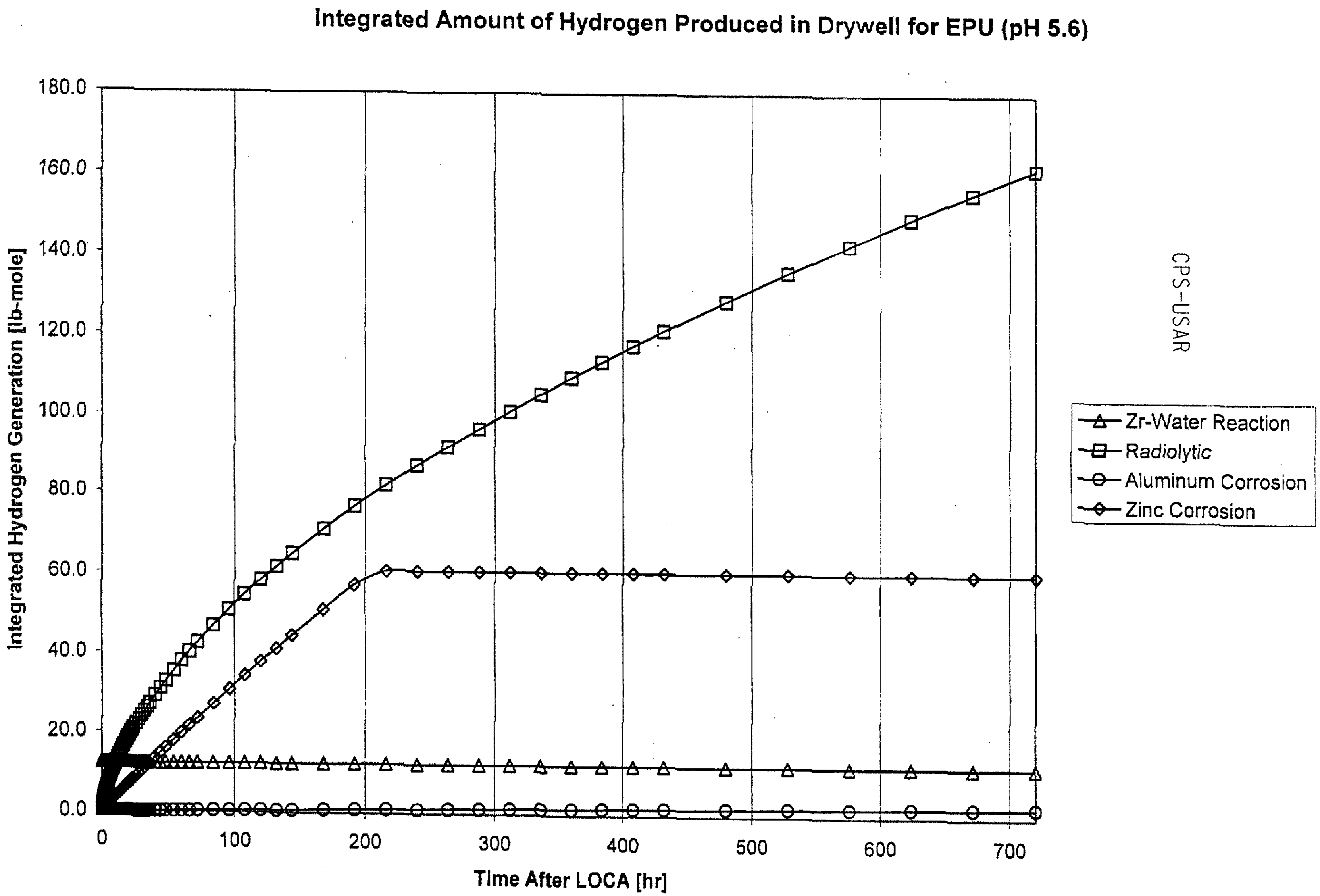

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FIGURE 6.2-128A

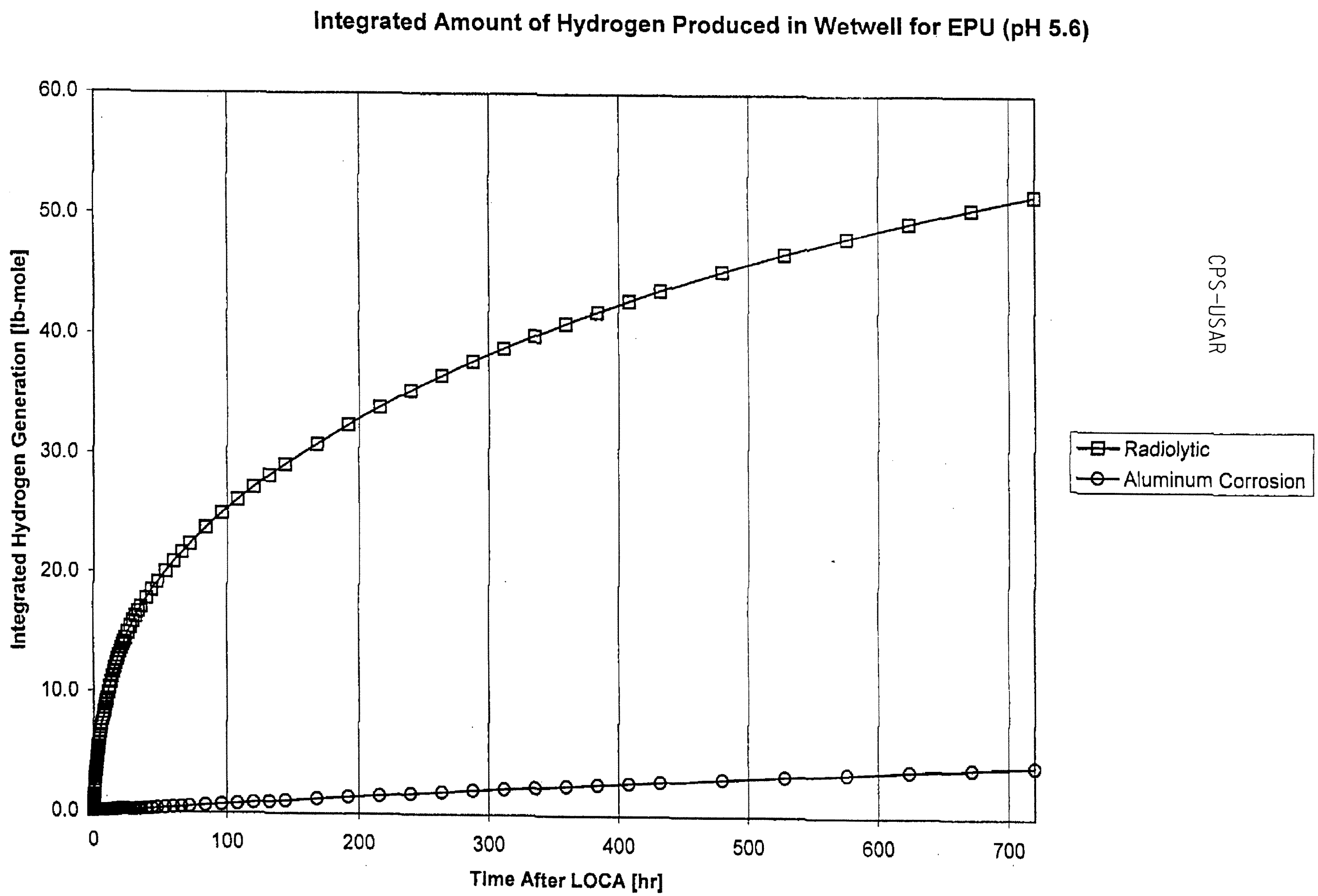

FIGURE 6.2-128B

Integrated Amount of Hydrogen Produced in Drywell for EPU (pH 8.6)

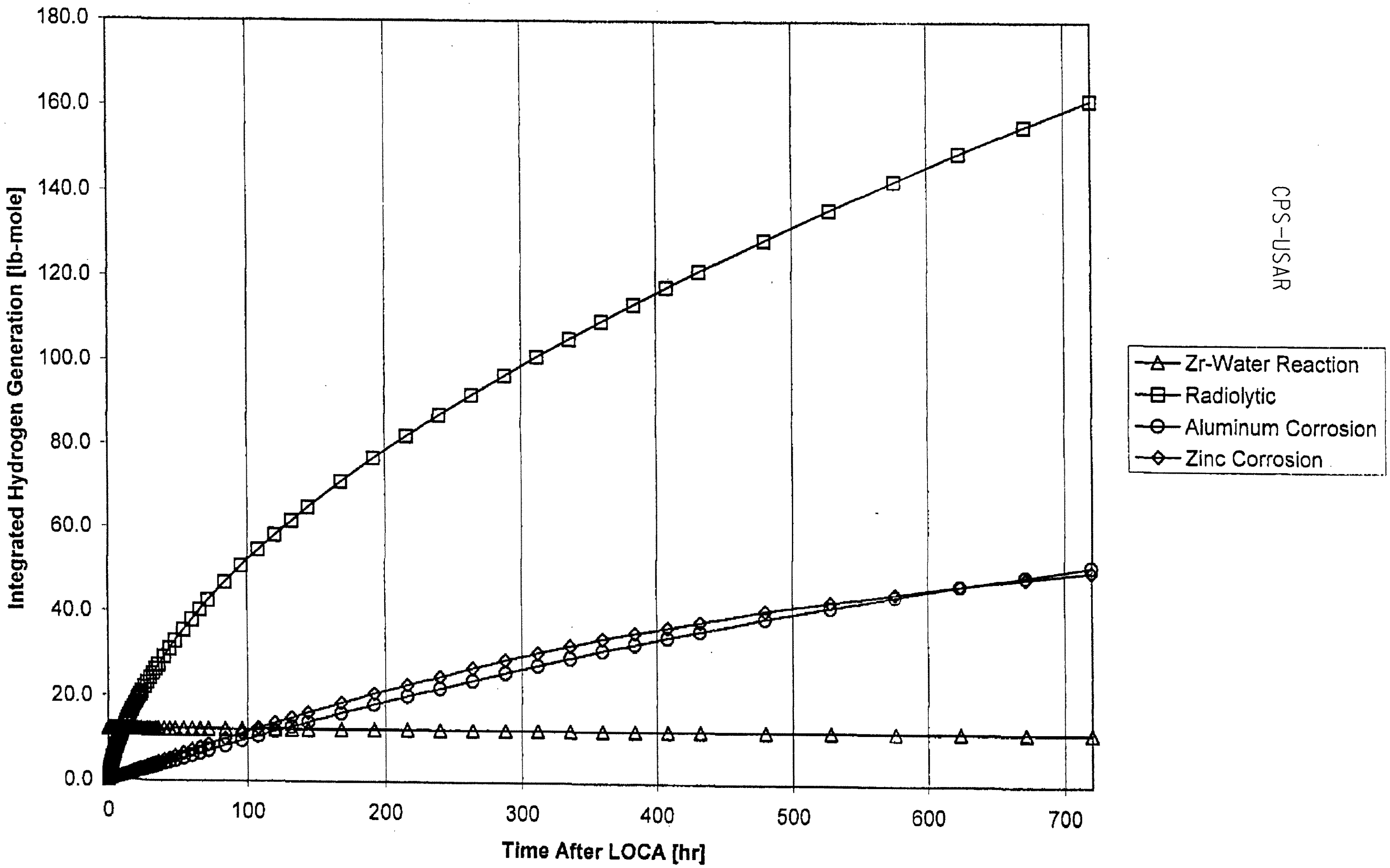

FIGURE 6.2-129A

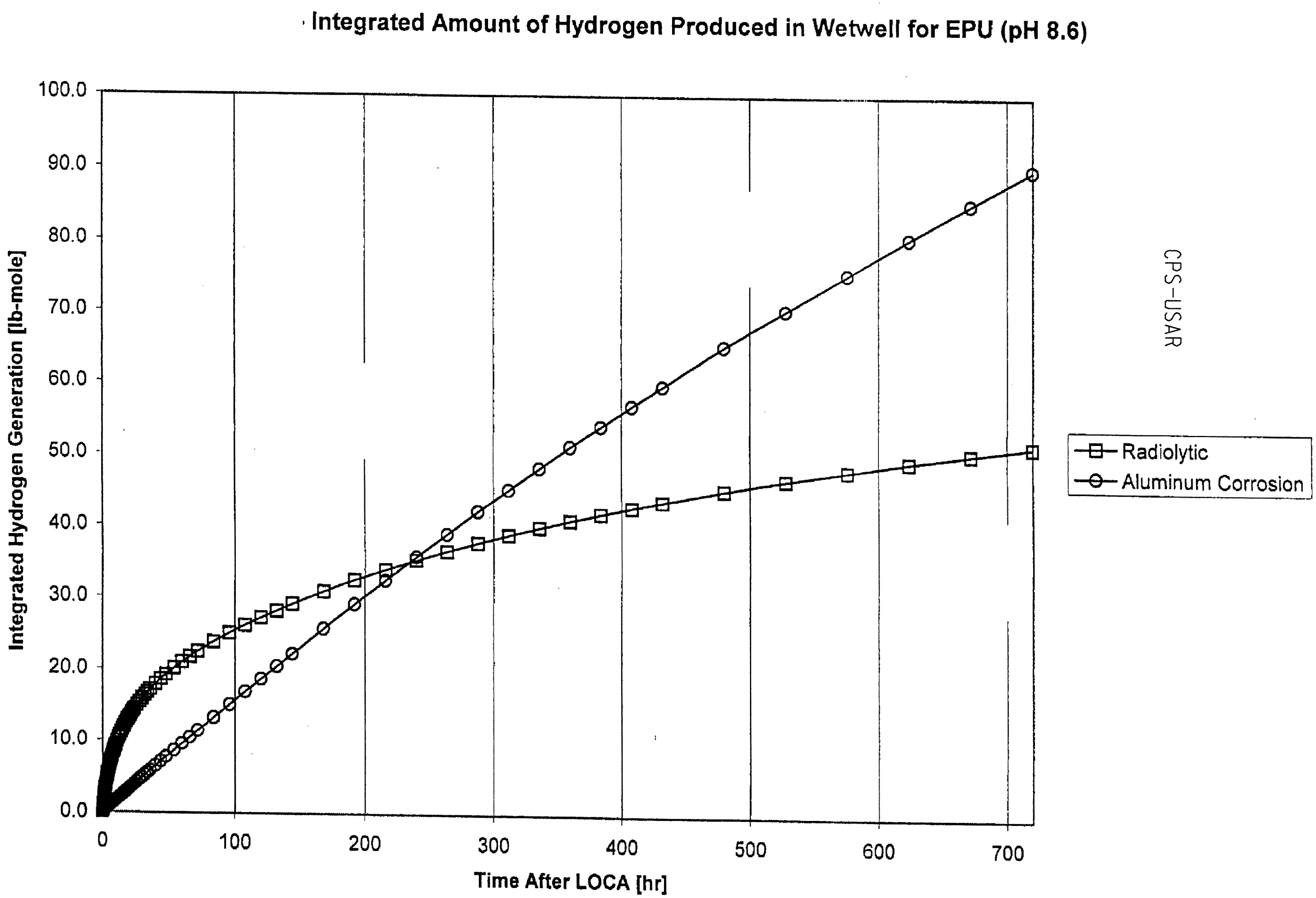

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FIGURE 6.2-129B

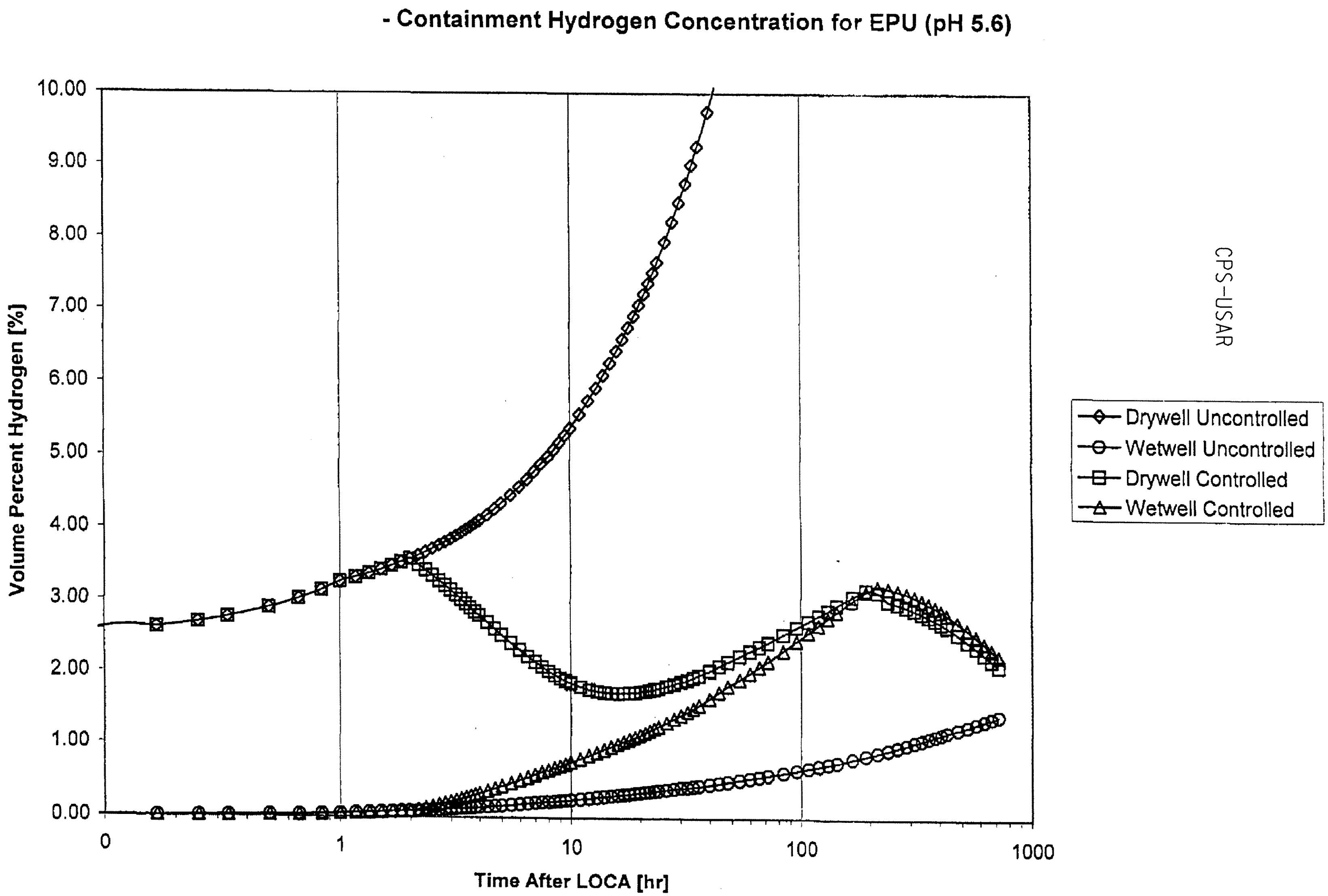

FIGURE 6.2-130A

CPS-USAR

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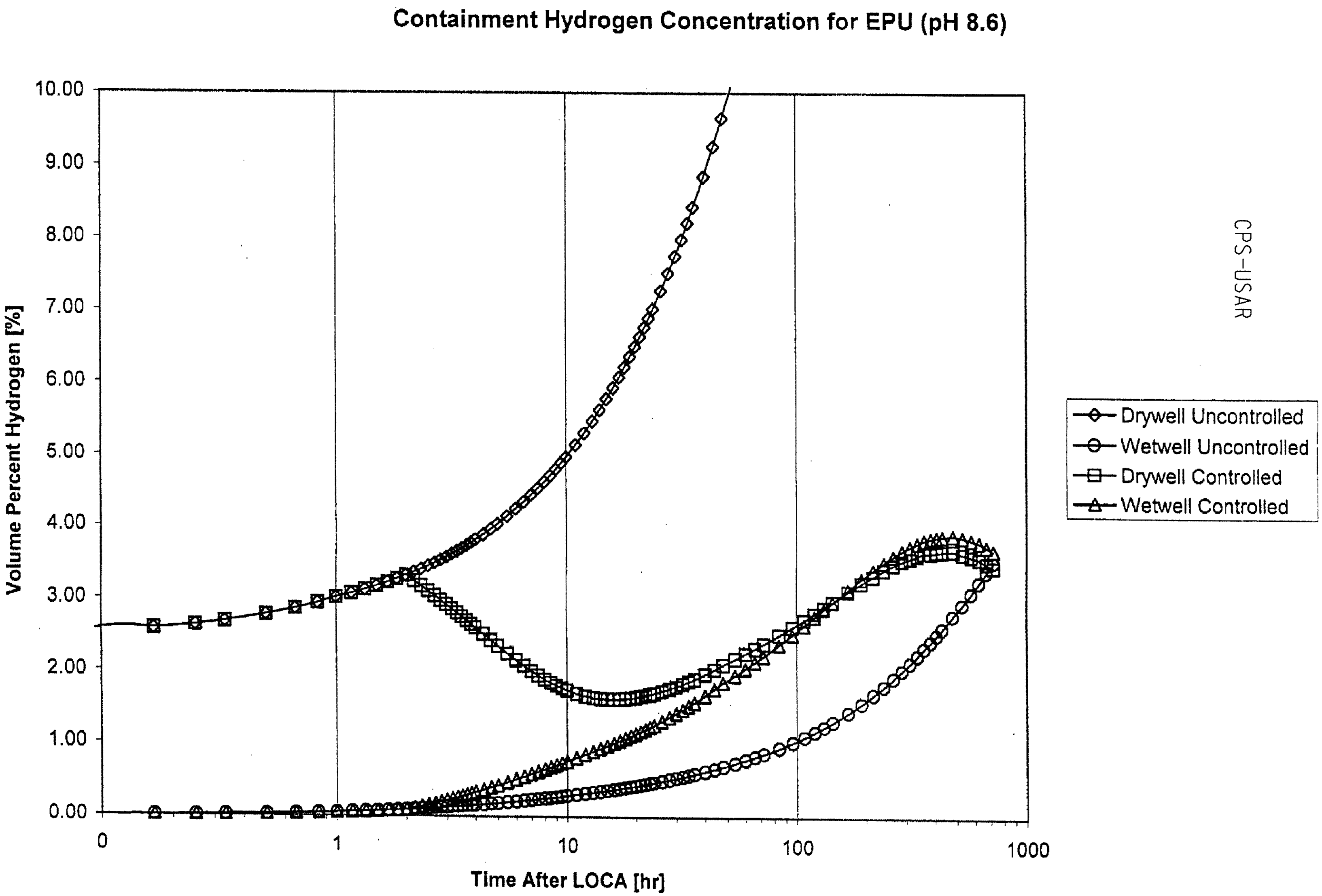

FIGURE 6.2-130B

Figure 6.2-131 has been deleted.

Security - Related Information Figure Withheld Under 10 CFR 2.390

CLINTON POWER STATION
UPDATED SAFETY ANALYSIS REPORT

FIGURE 6.2-132

SECONDARY CONTAINMENT BOUNDARY
(SHEET 1 of 6)

Security - Related Information Figure Withheld Under 10 CFR 2.390

CLINTON POWER STATION
UPDATED SAFETY ANALYSIS REPORT

FIGURE 6.2-132

SECONDARY CONTAINMENT BOUNDARY
(SHEET 2 of 6)

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CLINTON POWER STATION
FINAL SAFETY ANALYSIS REPORT

FIGURE 6.2-132

SECONDARY CONTAINMENT BOUNDARY

(SHEET 3 of 6)

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CLINTON POWER STATION
UPDATED SAFETY ANALYSIS REPORT

FIGURE 6.2-132

SECONDARY CONTAINMENT BOUNDARY
(SHEET 4 of 6)

Security - Related Information Figure Withheld Under 10 CFR 2.390

CLINTON POWER STATION
UPDATED SAFETY ANALYSIS REPORT

FIGURE 6.2-132

SECONDARY CONTAINMENT BOUNDARY

(SHEET 5 of 6)

Security - Related Information Figure Withheld Under 10 CFR 2.390

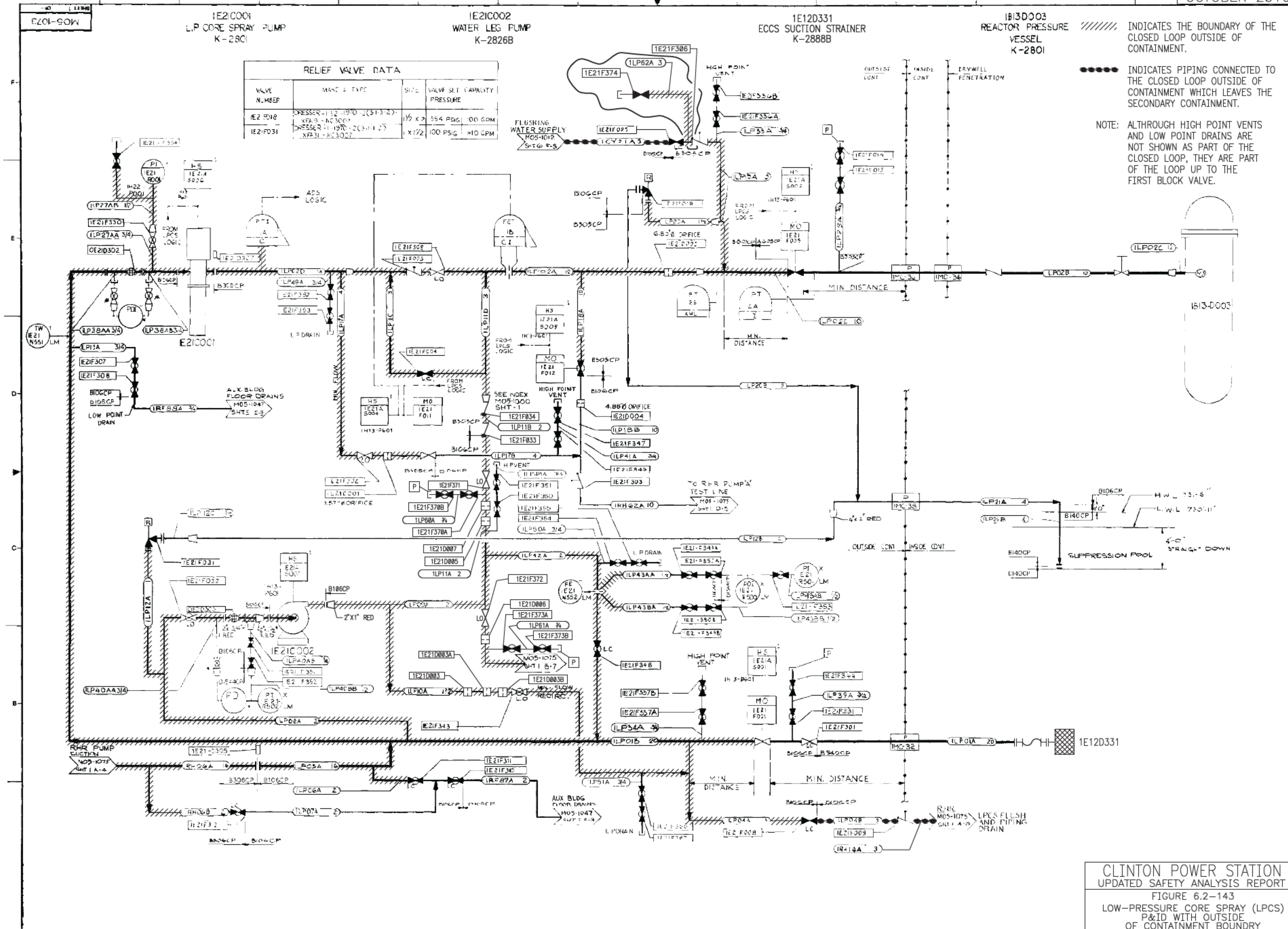
CLINTON POWER STATION
UPDATED SAFETY-ANALYSIS REPORT

FIGURE 6.2-132

SECONDARY CONTAINMENT BOUNDARY

(SHEET 6 of 6)

Figures 6.2-133 thru 6.2-142 have been deleted.



RELIEF VALVE DATA				
VALVE NUMBER	MAKE & TYPE	SIZE	VALVE SET PRESSURE	CAPACITY
1E21F018	DRESSER-1-1/2-1970-2(31-3-2)-XFA18-NC3007	1½ x 2	554 PSIG	100 GPM
1E21F031	DRESSER-1-1970-2(3-1-1-2)-XFA31-NC3007	1 x 1½	100 PSIG	>10 GPM

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FIGURE 6.2-143
LOW-PRESSURE CORE SPRAY (LPCS)
P&ID WITH OUTSIDE
OF CONTAINMENT BOUNDRY

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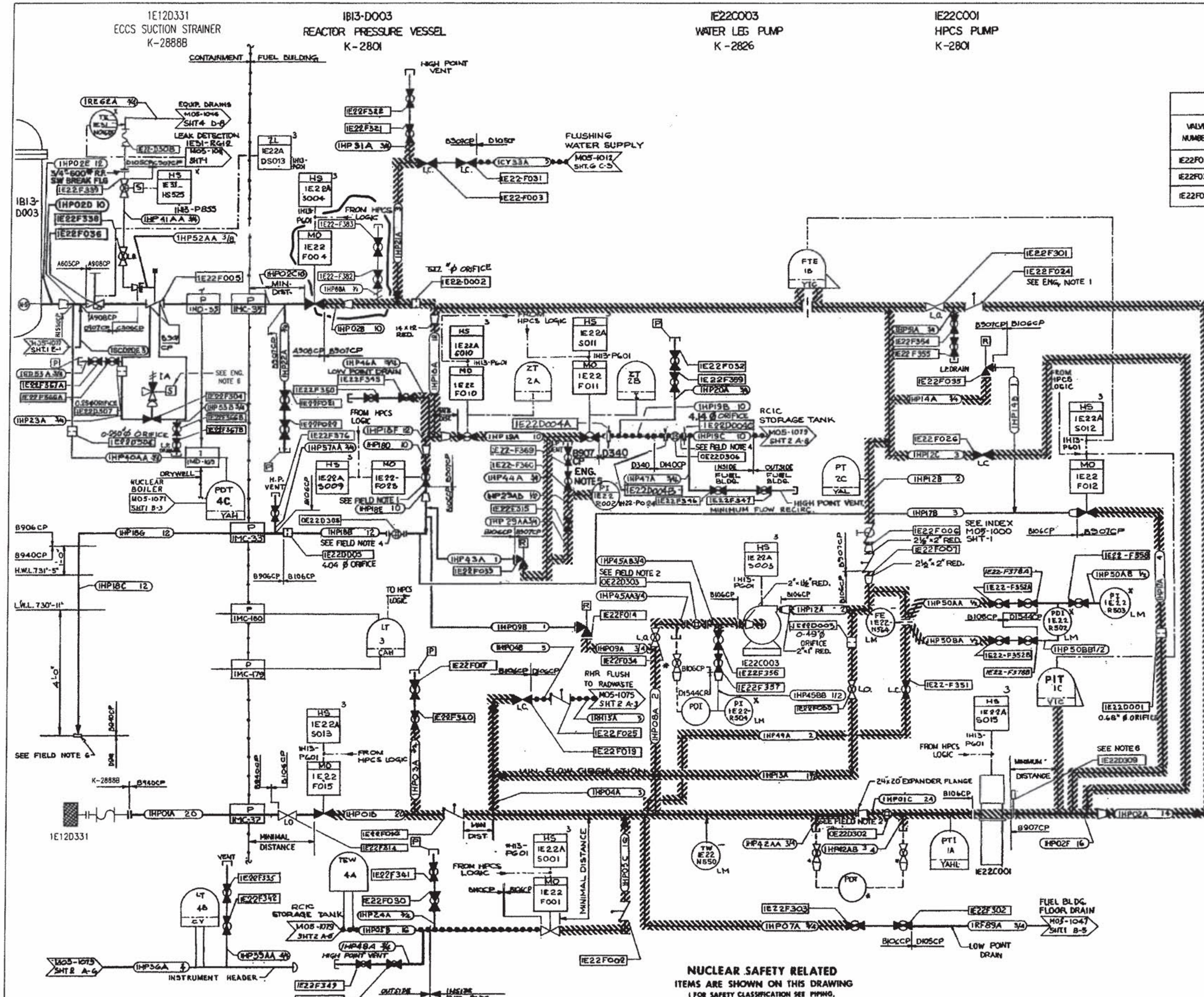
JANUARY 2011



RELIEF VALVE DATA					
VALVE NUMBER	MAKE & TYPE	SIZE	VALVE SET PRESSURE	BLOW DOWN (PSI)	CAPACITY
1E22F035	DRESSER 3/4-1970-2(3-4-1-2) -XFA55-NC 3007	3/4x1"	1560 PSIG.	FIXED	68GPM.
1E22F039	DRESSER 3/4-1975-3(3-4-4-2) -XFA50-NC3007	3/4"x1"	1560 PSIG.	FIXED	10 GPM
1E22F014	DRESSER 3/4-1975-3(3-1-1-2) -XFA49-NC3007	3/4"x1"	100 PSIG	FIXED	10 GPM.

Indicates the boundary of the closed loop outside of containment.

Indicates piping connected to the closed loop outside of containment which leaves the secondary containment.

Note: Although high point vents and low point drains are not shown as part of the closed loop, they are part of the loop up to the first block valve.

ENGINEERING NOTES:

1. CHECK VALVE 1E22-F024 SHALL BE LOCATED AT AN ELEVATION BELOW THE SUPPRESSION POOL LOW WATER LEVEL (730'-11").
2. ALL PRESSURE RELIEF VALVES SHALL BE REMOVABLE FOR TESTING.
3. ALL CHECK VALVES EXTERIOR TO THE CONTAINMENT SHALL BE TESTABLE TO VERIFY FREE MOVEMENT OF THE VALVE DISC. VALVE LOCATION SHOULD EASILY ACCOMMODATE THIS MANUAL TESTING.
4. "MINIMAL DISTANCE" MEANS THAT THE PIPING DESIGNER SHOULD HOLD THE INDICATED DISTANCE TO BE AS MINIMAL AS PRACTICALLY POSSIBLE, WITH CONSIDERATION FOR ISI ACCESS.
5. 1E22-R002 IS ABANDONED IN PLACE. SEE FDOR LHI-4162-1
6. EQUALIZING VALVE IS OPENED BY ENERGIZING THE AIR SOLENOID LOCALLY

FIELD NOTES:

1. VALVE 1E22F023 SHALL BE INSTALLED WITH THE PACKING GLAND ON THE UPSTREAM SIDE OF THE VALVE DISC.
2. TEMPORARY IN LINE STRAINER (CONICAL TYPE) & DIFFERENTIAL PRESSURE GAUGE AS SHOWN INSTALLED DURING PRE-OP. & INITIAL START UP TESTING ONLY. (SEE S & L STD. NF-270.9.1) REMOVE FOR NORMAL PLANT OPERATION & CAP & WELD GAUGE PRESSURE SENSING LINES AS SHOWN.
3. FOR HPCS LOGIC SEE GE DWG. 828E314.
4. SPOOL PIECE FOR TEMPORARY IN LINE STRAINER FOR PRE-OP & INITIAL START UP TESTING ONLY. REMOVE STRAINER FOR NORMAL PLANT OPERATION.
5. RESTRICTING ORIFICE SIZES INDICATED ARE FOR INITIAL SYSTEM START UP ONLY. ADJUST AS NECESSARY TO MEET HYDRAULIC REQUIREMENTS.
6. BLIND FLANGE FOR HYDROSTATIC PRESSURE TEST ONLY. REMOVE FOR NORMAL PLANT OPERATION.

NUCLEAR SAFETY RELATED
ITEMS ARE SHOWN ON THIS DRAWING
(FOR SAFETY CLASSIFICATION SEE PIPING, EQUIPMENT, VALVE, OR INSTRUMENT LISTS.)

NOTE
THE P & I D/C & I DIAGRAMS FOR THE C & I FUNCTION SYMBOLS ARE SHOWN ON DRAWING M10-9074 UNLESS OTHERWISE NOTED. SHEET NUMBER & DIAGRAM REFERENCE ARE SHOWN IN THE FUNCTION SYMBOL.

CLINTON POWER STATION
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FIGURE 6.2-144

HIGH-PRESSURE CORE SPRAY
P&ID SHOWING OUTSIDE
OF CONTAINMENT BOUNDARY

P146

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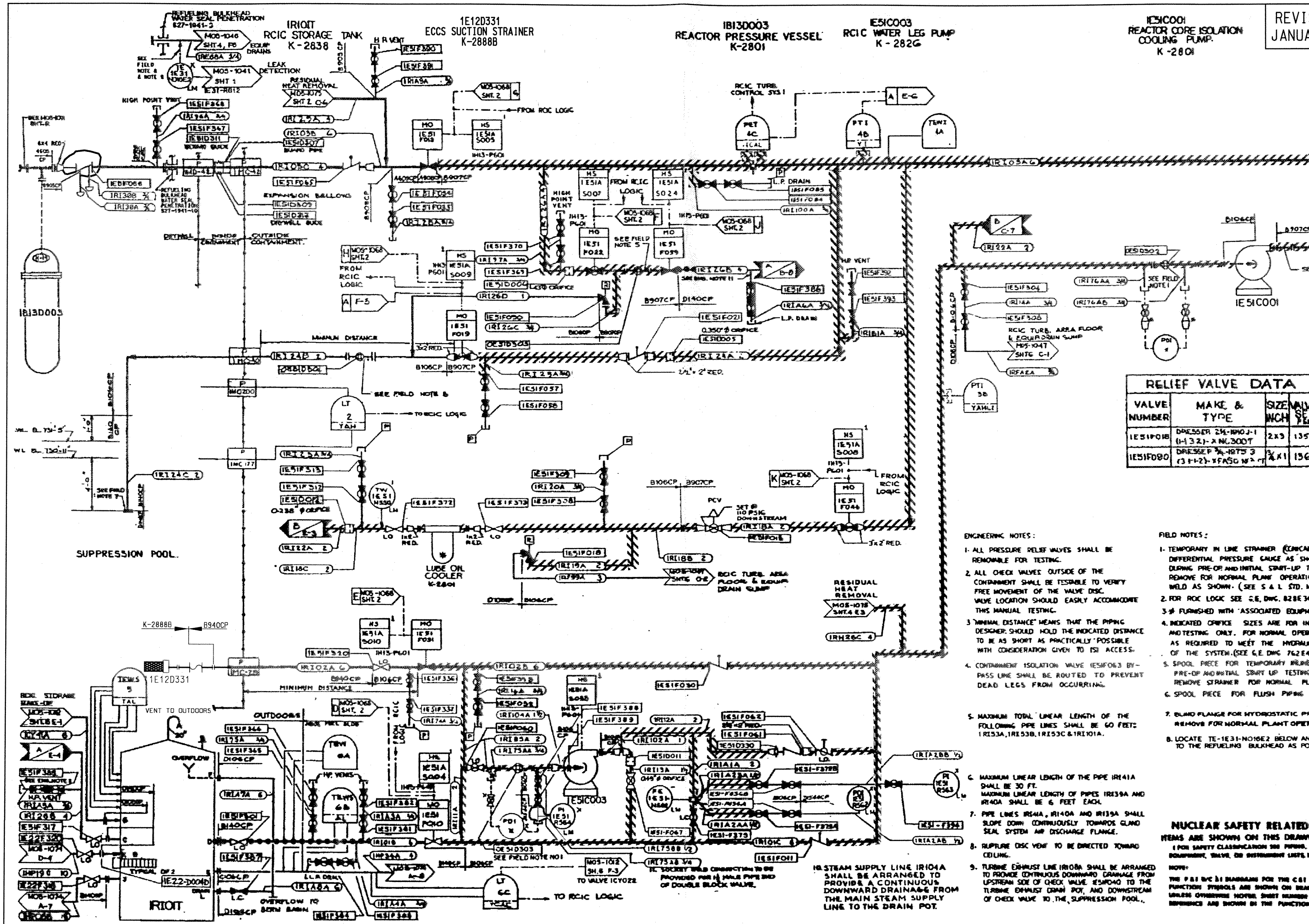
JANUARY 2007



RELIEF VALVE DATA

VALVE NUMBER	MAKE & TYPE	SIZE INCH	VALVE SET PSIG	CAPACITY GPM
1E51F018	DRESSER 2½-1910J-1 (1-32)-XNC300T	2x3	135	191
1E51F080	DRESSER ¾-1975 3 (3-1-2)-XFA50 N^x T	¾x1	1360	6E

ENGINEERING NOTES:

1. ALL PRESSURE RELIEF VALVES SHALL BE REMOVABLE FOR TESTING.
2. ALL CHECK VALVES OUTSIDE OF THE CONTAINMENT SHALL BE TESTABLE TO VERIFY FREE MOVEMENT OF THE VALVE DISC. VALVE LOCATION SHOULD EASILY ACCOMMODATE THIS MANUAL TESTING.
3. "MINIMAL DISTANCE" MEANS THAT THE PIPING DESIGNER SHOULD HOLD THE INDICATED DISTANCE TO BE AS SHORT AS PRACTICALLY POSSIBLE WITH CONSIDERATION GIVEN TO ISI ACCESS.
4. CONTAINMENT ISOLATION VALVE 1E51F063 BY-PASS LINE SHALL BE ROUTED TO PREVENT DEAD LEGS FROM OCCURRING.
5. MAXIMUM TOTAL LINEAR LENGTH OF THE FOLLOWING PIPE LINES SHALL BE 60 FEET: 1RI53A, 1RI53B, 1RI53C & 1RI101A.
6. MAXIMUM LINEAR LENGTH OF THE PIPE 1RI41A SHALL BE 30 FT. MAXIMUM LINEAR LENGTH OF PIPES 1RI39A AND 1RI40A SHALL BE 6 FEET EACH.
7. PIPE LINES 1RI41A, 1RI40A AND 1RI39A SHALL SLOPE DOWN CONTINUOUSLY TOWARDS GLAND SEAL SYSTEM AIR DISCHARGE FLANGE.
8. RUPTURE DISC VENT TO BE DIRECTED TOWARD CEILING.
9. TURBINE EXHAUST LINE 1RI08A SHALL BE ARRANGED TO PROVIDE CONTINUOUS DOWNWARD DRAINAGE FROM UPSTREAM SIDE OF CHECK VALVE 1E51F040 TO THE TURBINE EXHAUST DRAIN POT, AND DOWNSTREAM OF CHECK VALVE TO THE SUPPRESSION POOL.

10. STEAM SUPPLY LINE 1RI04A SHALL BE ARRANGED TO PROVIDE A CONTINUOUS DOWNWARD DRAINAGE FROM THE MAIN STEAM SUPPLY LINE TO THE DRAIN POT.

FIELD NOTES:

1. TEMPORARY IN LINE STRAINER (CONICAL TYPE) AND DIFFERENTIAL PRESSURE GAUGE AS SHOWN INSTALLED DURING PRE-OP AND INITIAL START-UP TESTING ONLY. REMOVE FOR NORMAL PLANT OPERATION, AND CAP AND WELD AS SHOWN. (SEE S & L STD. MF. 270.9.1)
2. FOR RCIC LOGIC SEE G.E. DWG. 828E306.
3. * FURNISHED WITH ASSOCIATED EQUIPMENT.
4. INDICATED ORIFICE SIZES ARE FOR INITIAL START-UP AND TESTING ONLY. FOR NORMAL OPERATION ADJUST AS REQUIRED TO MEET THE HYDRAULIC REQUIREMENTS OF THE SYSTEM. (SEE G.E. DWG 762E421A)
5. SPOOL PIECE FOR TEMPORARY INLINE STRAINER FOR PRE-OP AND INITIAL START UP TESTING ONLY. REMOVE STRAINER FOR NORMAL PLANT OPERATION.
6. SPOOL PIECE FOR FLUSH PIPING BY-PASS.
7. BLIND FLANGE FOR HYDROSTATIC PRESSURE TEST ONLY. REMOVE FOR NORMAL PLANT OPERATION.
8. LOCATE TE-1E31-N016E2 BELOW AND AS CLOSE TO THE REFUELING BULKHEAD AS POSSIBLE.

NUCLEAR SAFETY RELATED
ITEMS ARE SHOWN ON THIS DRAWING
(FOR SAFETY CLASSIFICATION SEE PIPING, EQUIPMENT, VALVE, OR INSTRUMENT LISTS.)

NOTE:
THE P&I D/C & I DIAGRAMS FOR THE C&I FUNCTION SYMBOLS ARE SHOWN ON DRAWING M10-9079 UNLESS OTHERWISE NOTED. SHEET NUMBER & DIAGRAM REFERENCE ARE SHOWN IN THE FUNCTION SYMBOL.

Note: Although high point vents and low point drains are not shown as part of the closed loop, they are part of the loop up to the first block valve.

////// Indicates the boundary of the closed loop outside of containment.

•••••• Indicates piping connected to the closed loop outside of containment which leaves the secondary containment.

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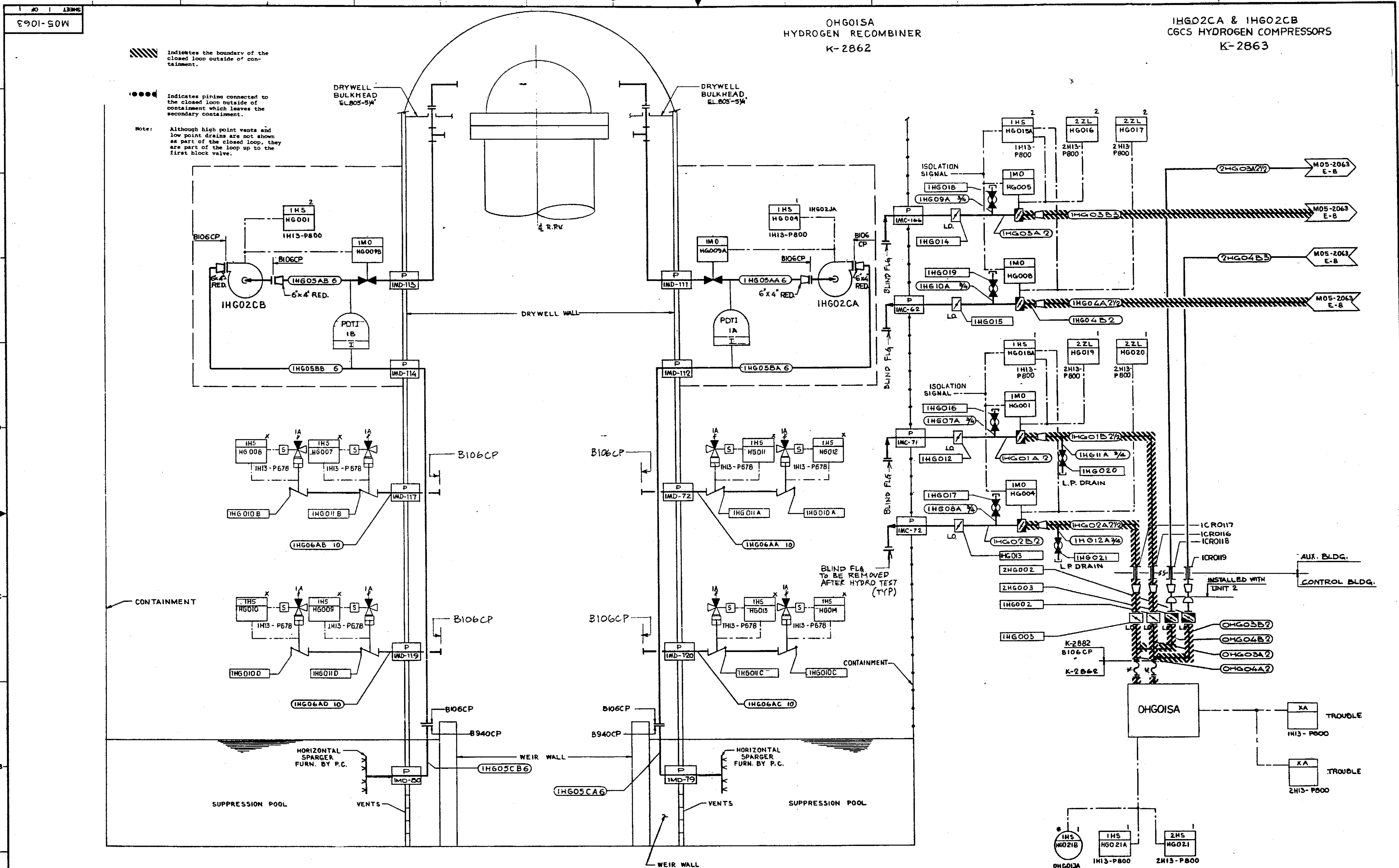
FIGURE 6.2-145

REACTOR CORE ISOLATION COOLING SYSTEM P&ID SHOWING OUTSIDE OF CONTAINMENT BOUNDARY

REV. 2 SEPT. 1990



NOTE: 1. WHEN UNIT 2 IS INSTALLED, VALVES 1HG006 AND 1HG007 SHALL BE LOCKED CLOSED, AND VALVES 2HG006 AND 2HG007 SHALL BE LOCKED OPEN.

NOTE:
THE P&I D/C&I DIAGRAMS FOR THE C&I FUNCTION SYMBOLS ARE SHOWN ON DRAWING M10-9063 UNLESS OTHERWISE NOTED. SHEET NUMBER & DIAGRAM REFERENCE ARE SHOWN IN THE FUNCTION SYMBOL.

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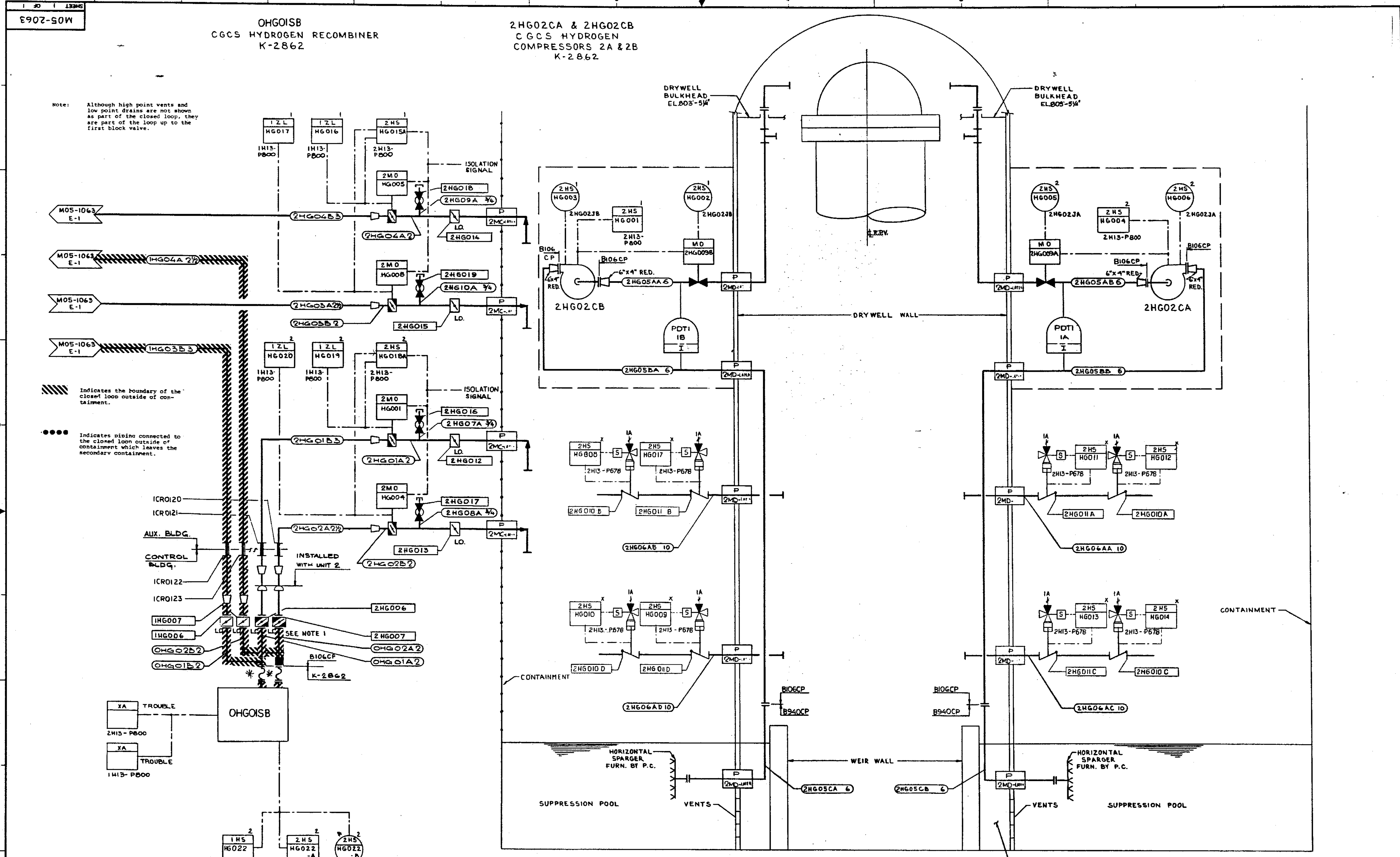
UPDATED SAFETY ANALYSIS REPORT

FIGURE 6.2-146

COMBUSTIBLE GAS CONTROL SYSTEM P&ID SHOWING OUTSIDE OF CONTAINMENT BOUNDARY (SHEET 1 of 2)



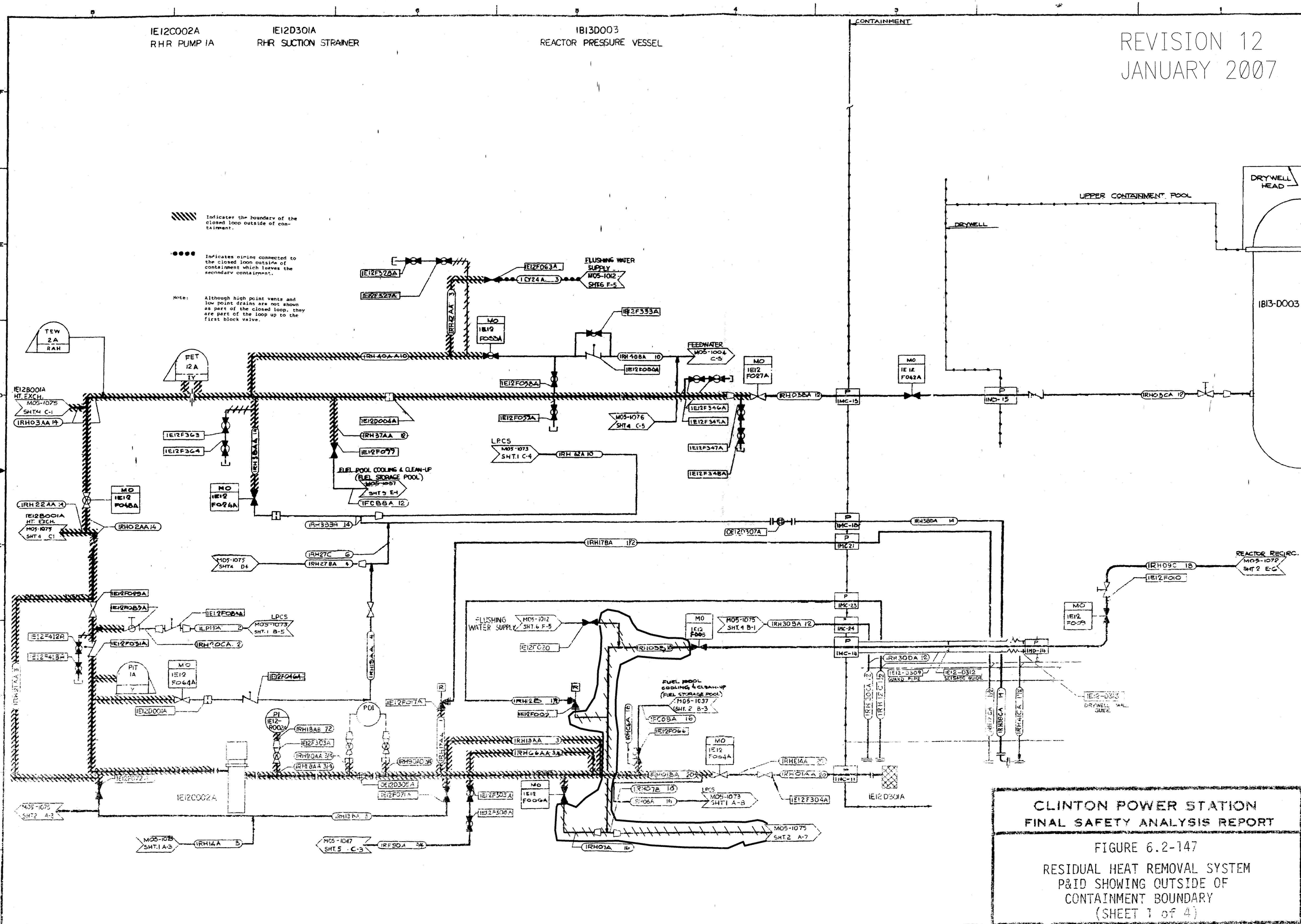
NOTES: 1. WHEN UNIT 2 IS INSTALLED, VALVES 1HG006 AND 1HG007 SHALL BE LOCKED CLOSED, AND VALVES 2HG006 AND 2HG007 SHALL BE LOCKED OPEN.

NOTE:
THE P&I D/C&I DIAGRAMS FOR THE C&I FUNCTION SYMBOLS ARE SHOWN ON DRAWING M10-2063 UNLESS OTHERWISE NOTED. SHEET NUMBER & DIAGRAM REFERENCE ARE SHOWN IN THE FUNCTION SYMBOL.

CLINTON POWER STATION
UPDATED SAFETY ANALYSIS REPORT

FIGURE 6.2-146

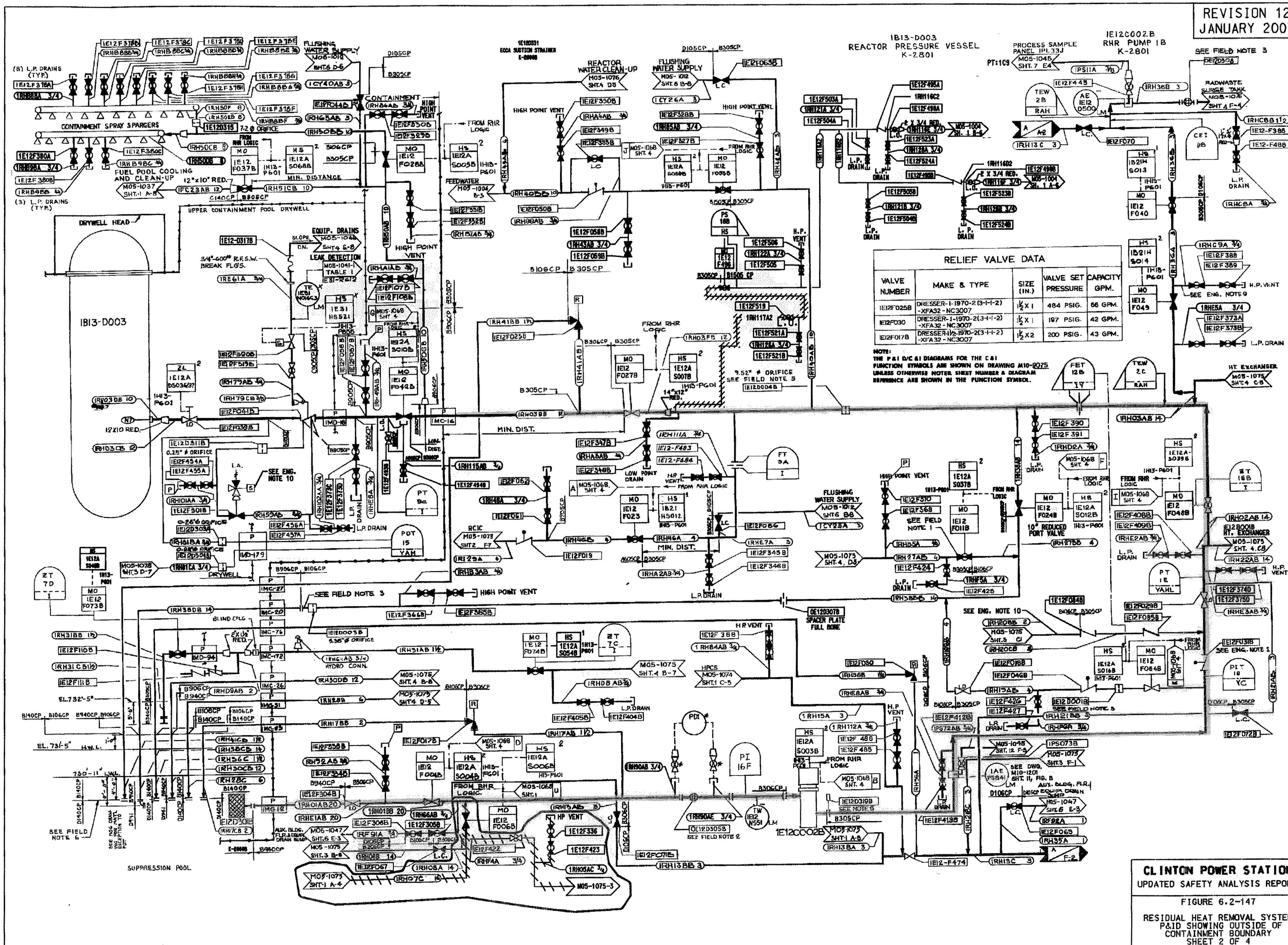
COMBUSTIBLE GAS CONTROL SYSTEM P&ID SHOWING OUTSIDE OF CONTAINMENT BOUNDARY (SHEET 2 of 2)



CLINTON POWER STATION
FINAL SAFETY ANALYSIS REPORT

FIGURE 6.2-147

RESIDUAL HEAT REMOVAL SYSTEM
P&ID SHOWING OUTSIDE OF
CONTAINMENT BOUNDARY
(SHEET 1 of 4)



RELIEF VALVE DATA

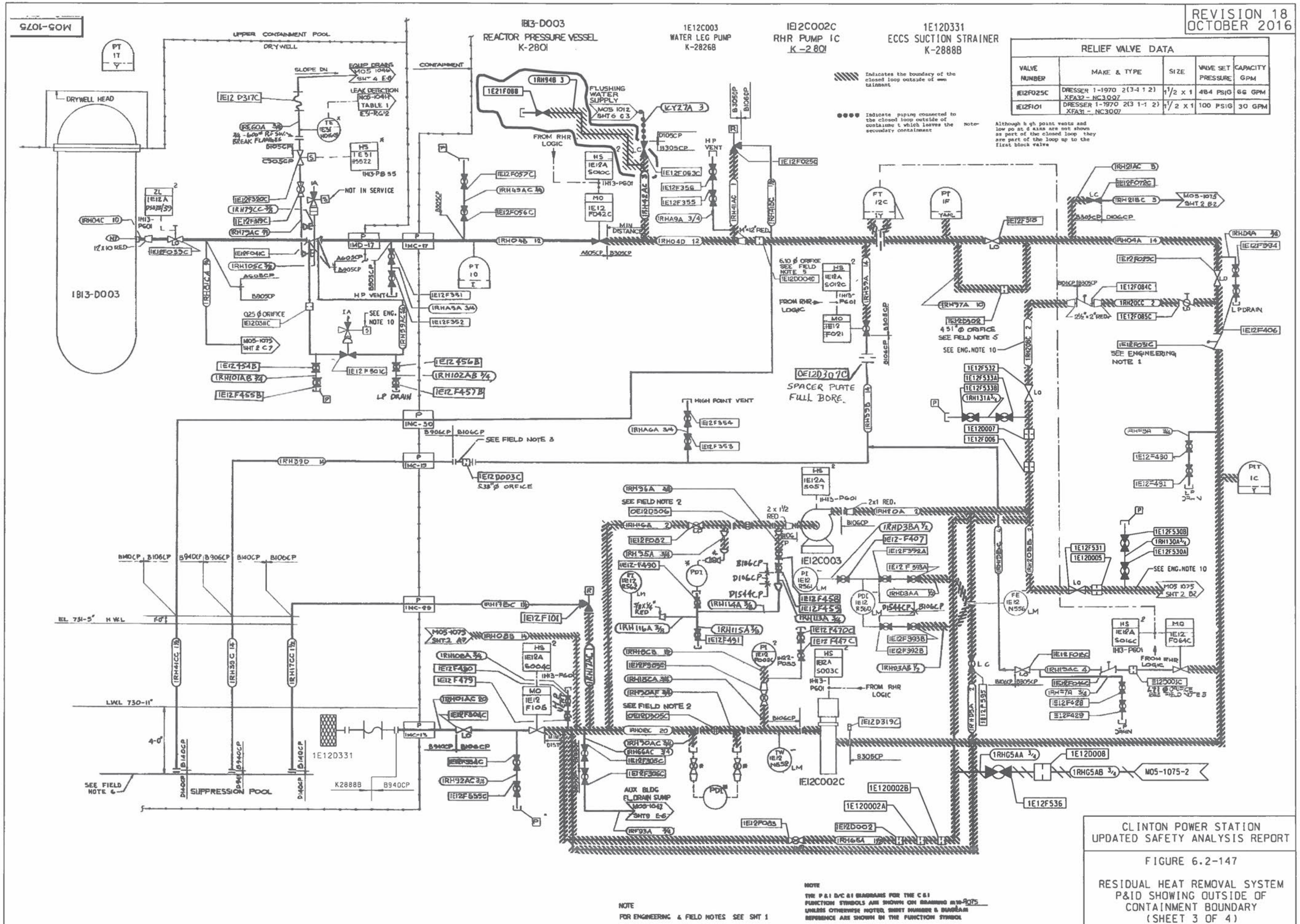
VALVE NUMBER	MAKE & TYPE	SIZE (IN.)	VALVE SET PRESSURE	CAPACITY GPM.
1E12F025B	DRESSER-1-1970-2 (3-1-1-2) -XFA32-NC3007	1½X1	484 PSIG.	66 GPM.
1E12F030	DRESSER-1-1970-2(3-1-1-2) -XFA32-NC3007	1½X1	197 PSIG.	42 GPM.
1E12F017B	DRESSER-1½-1970-2(3-1-1-2) -XFA32-NC3007	1½X2	200 PSIG.	43 GPM.

CLINTON POWER STATION
UPDATED SAFETY ANALYSIS REPORT

FIGURE 6.2-147

RESIDUAL HEAT REMOVAL SYSTEM P&ID SHOWING OUTSIDE OF CONTAINMENT BOUNDARY SHEET 2 OF 4

...\USAR\rev 12\fig6-2-147-2.dgn 09/13/2006 02:23:31 PM

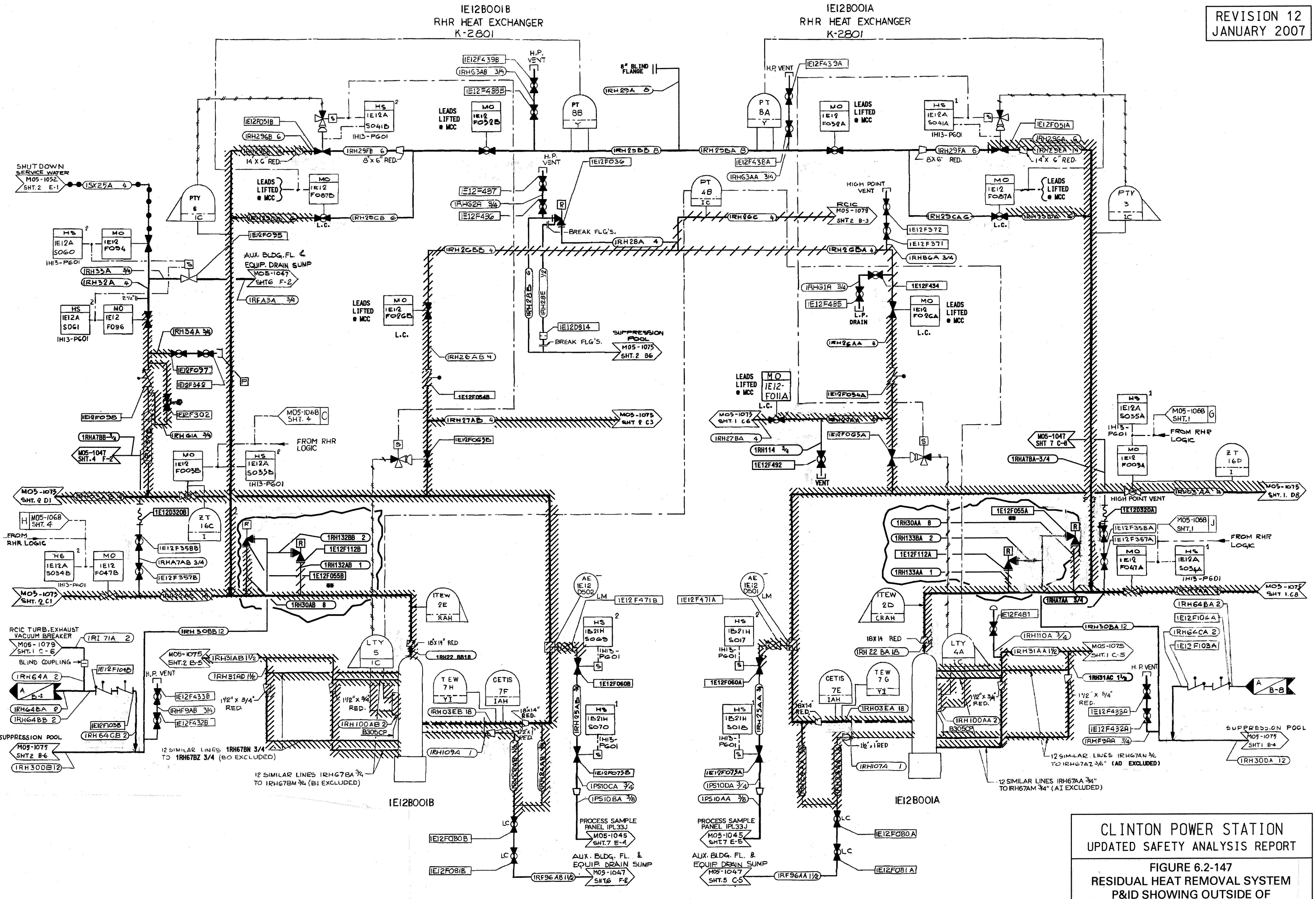


CLINTON POWER STATION
UPDATED SAFETY ANALYSIS REPORT

FIGURE 6.2-147

RESIDUAL HEAT REMOVAL SYSTEM
P&ID SHOWING OUTSIDE OF
CONTAINMENT BOUNDARY
(SHEET 3 OF 4)

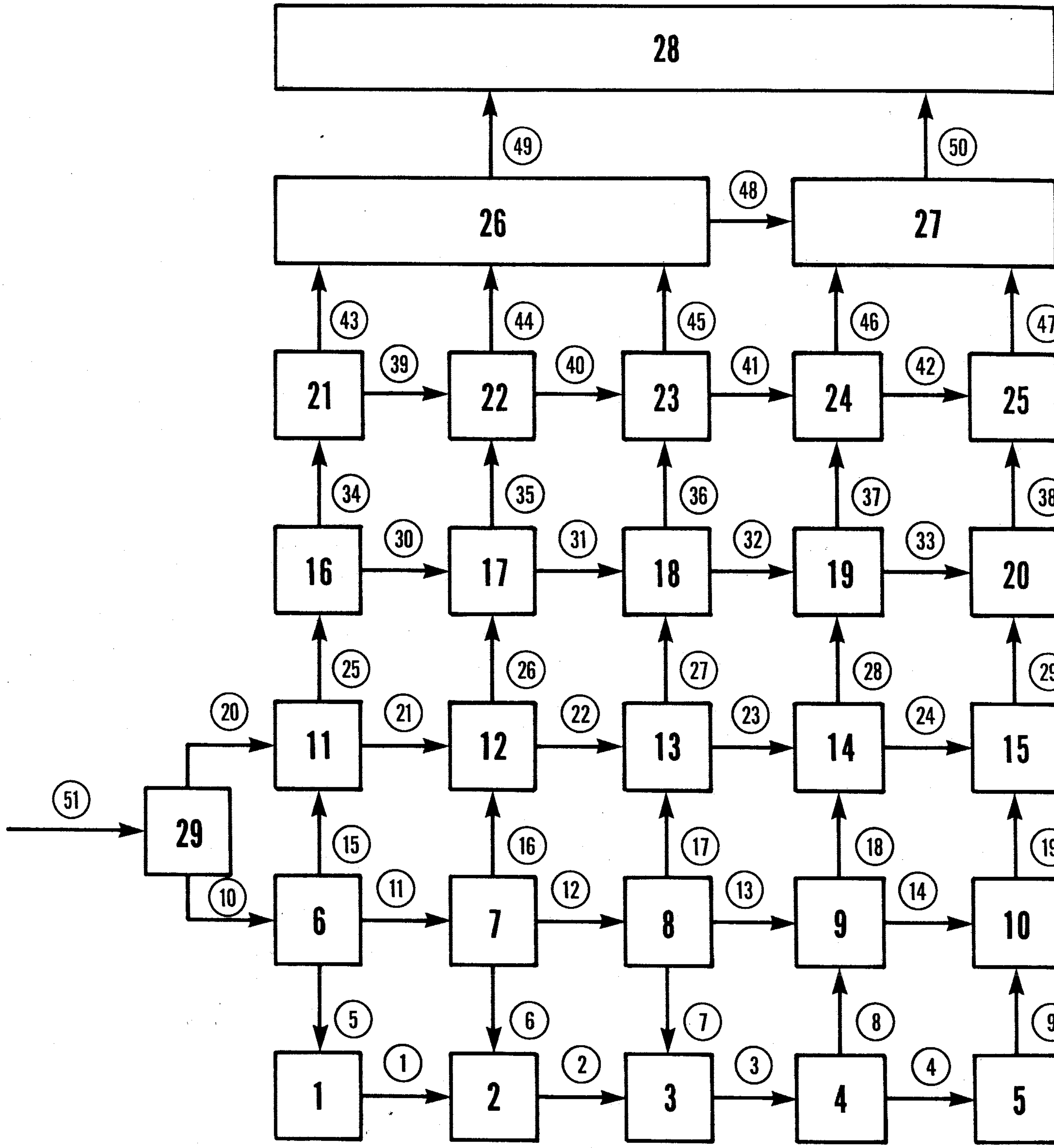
REVISION 12
JANUARY 2007



CLINTON POWER STATION
UPDATED SAFETY ANALYSIS REPORT

FIGURE 6.2-147
RESIDUAL HEAT REMOVAL SYSTEM
P&ID SHOWING OUTSIDE OF
CONTAINMENT BOUNDARY
(SHEET 4 OF 4)

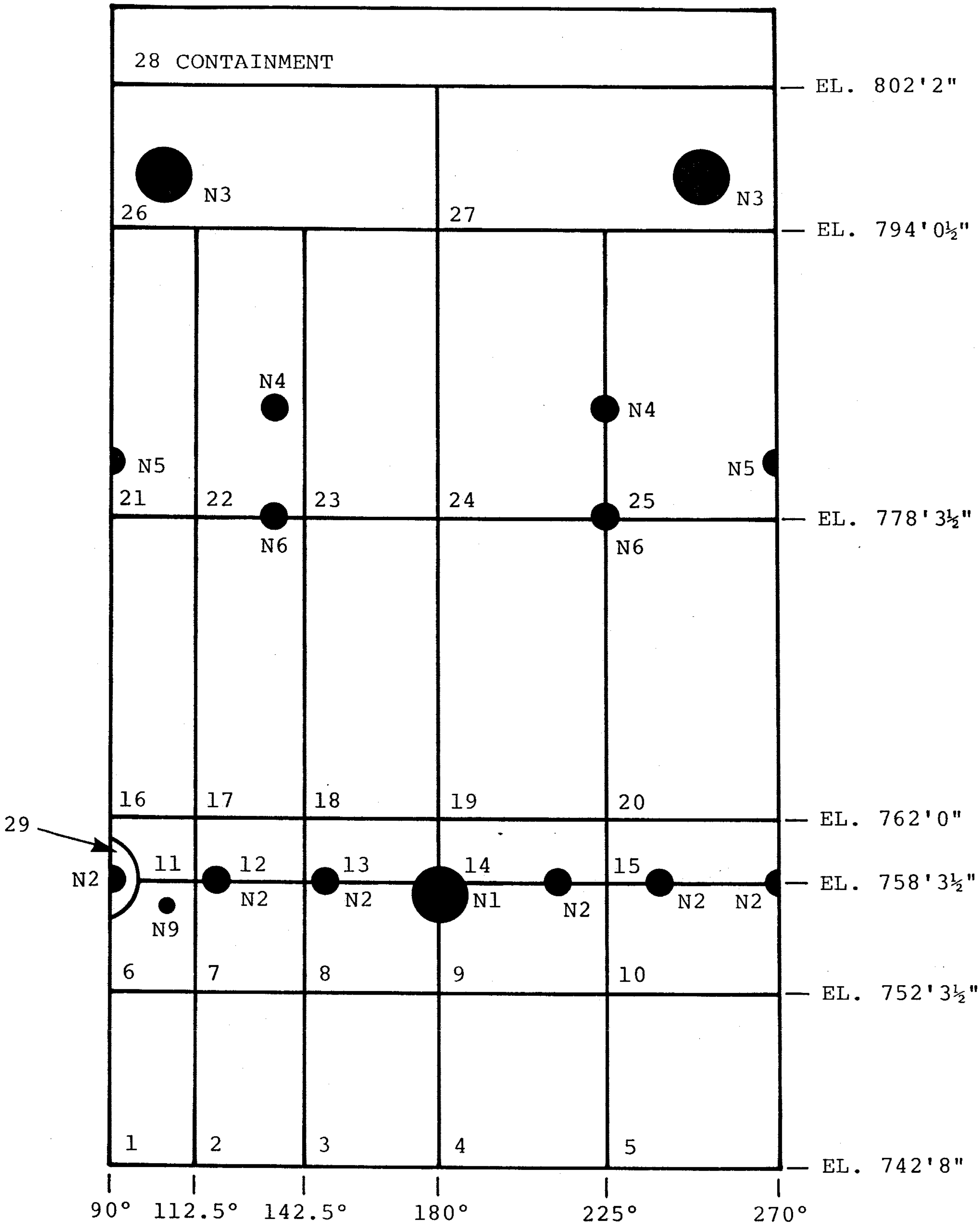
..\USAR\rev 12\FIG6-2-147-4.dgn 11/16/2005 05:35:57 PM



CLINTON POWER STATION
UPDATED SAFETY ANALYSIS REPORT

FIGURE 6.2-148

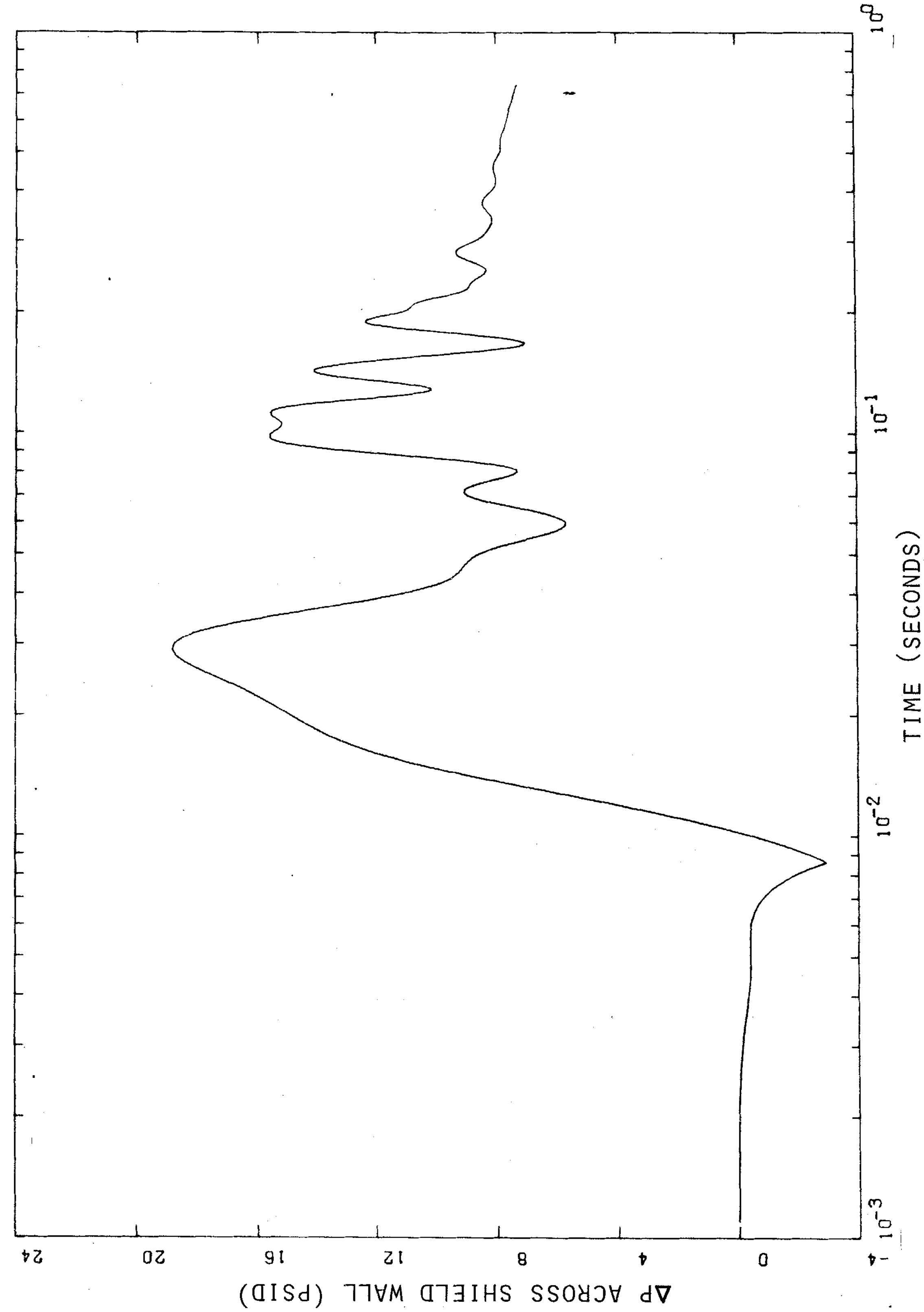
NODALIZATION SCHEMATIC FOR SACRIFICIAL SHIELD ANNULUS PRESSURIZATION ANALYSIS - RECIRCULATION INLET LINE BREAK



CLINTON POWER STATION
UPDATED SAFETY ANALYSIS REPORT

FIGURE 6.2-149

ANNULUS NODALIZATION FOR
RECIRCULATION INLET LINE BREAK



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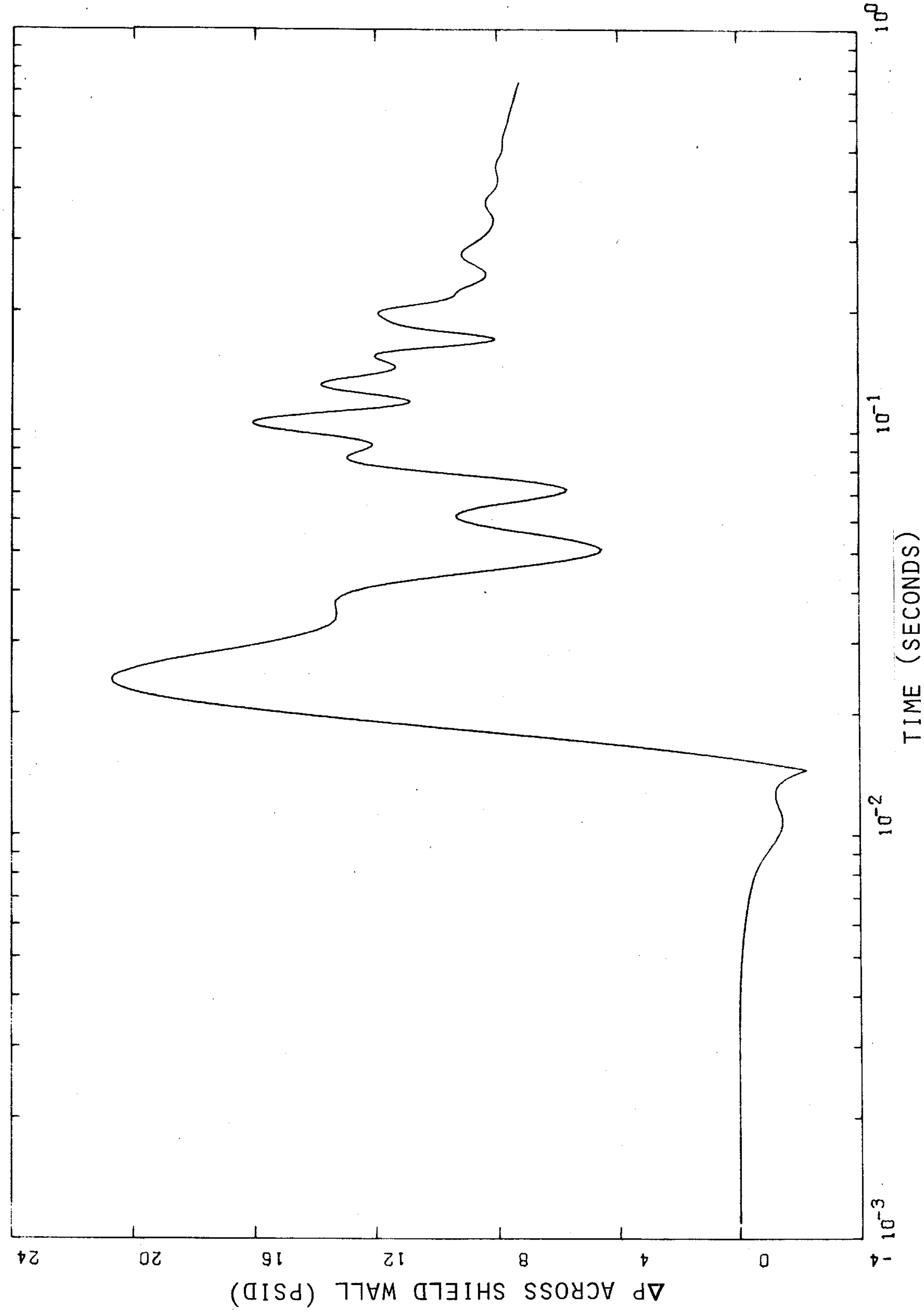
UPDATED SAFETY ANALYSIS REPORT

FIGURE 6.2-150

ΔP VS. LOG T FOR NODE 1
(RECIRCULATION INLET LINE BREAK)



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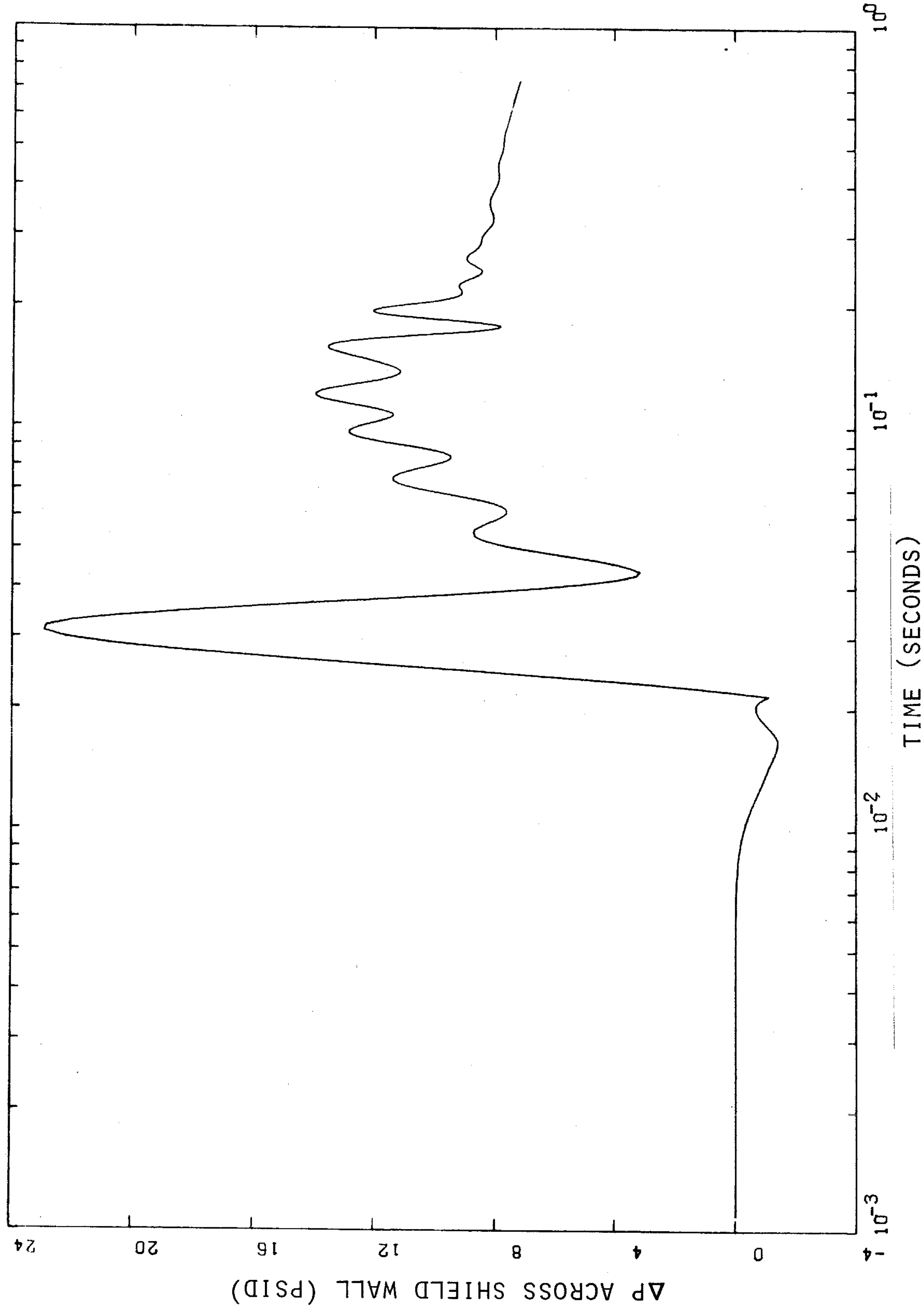
UPDATED SAFETY ANALYSIS REPORT

FIGURE 6.2-151

ΔP VS. LOG T FOR NODE 2
(RECIRCULATION INLET LINE BREAK)



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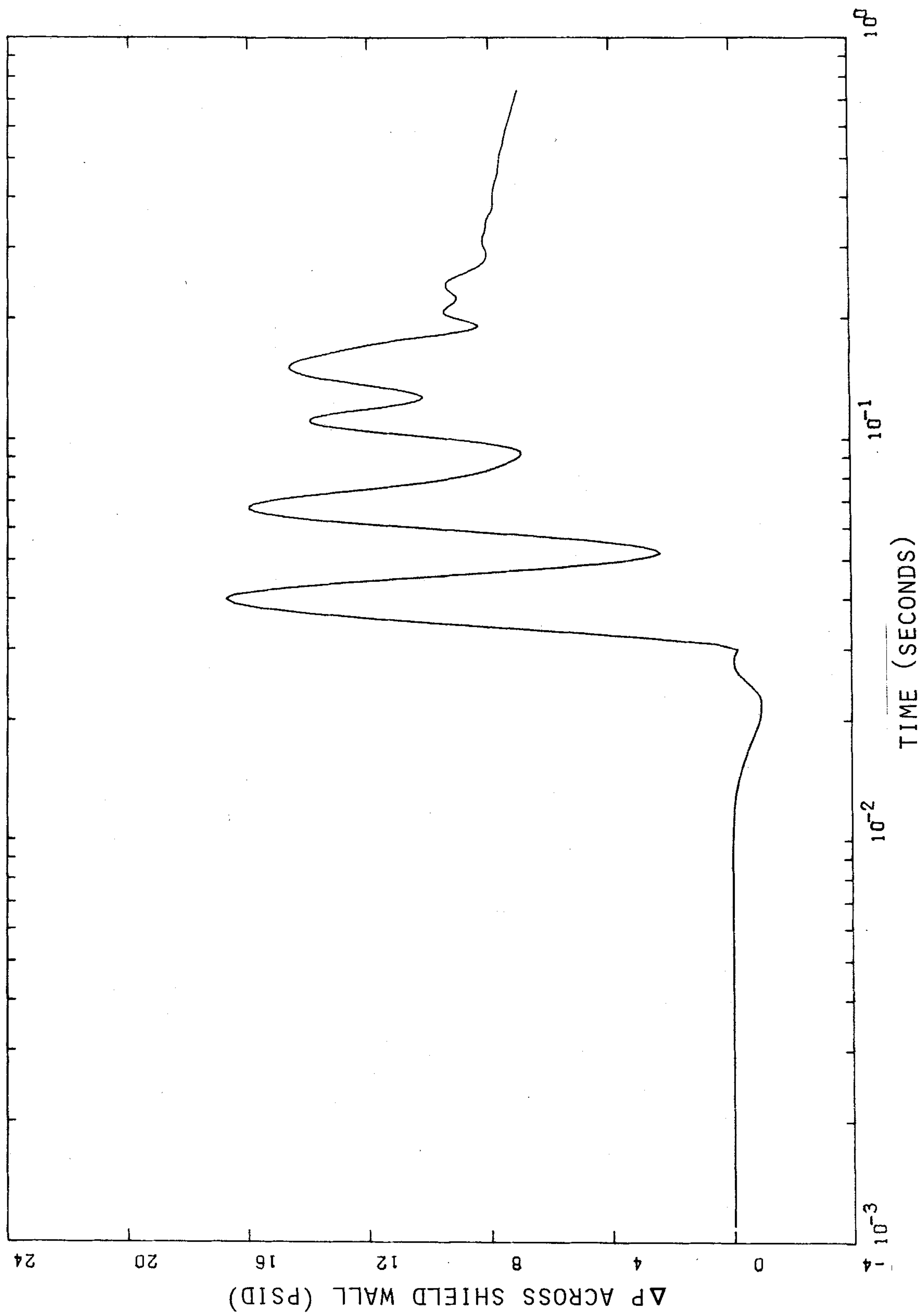
UPDATED SAFETY ANALYSIS REPORT

FIGURE 6.2-152

ΔP VS. LOG T FOR NODE 3
(RECIRCULATION INLET LINE BREAK)



CLINTON POWER STATION
UPDATED SAFETY ANALYSIS REPORT

FIGURE 6.2-153

ΔP VS. LOG T FOR NODE 4
(RECIRCULATION INLET LINE BREAK)

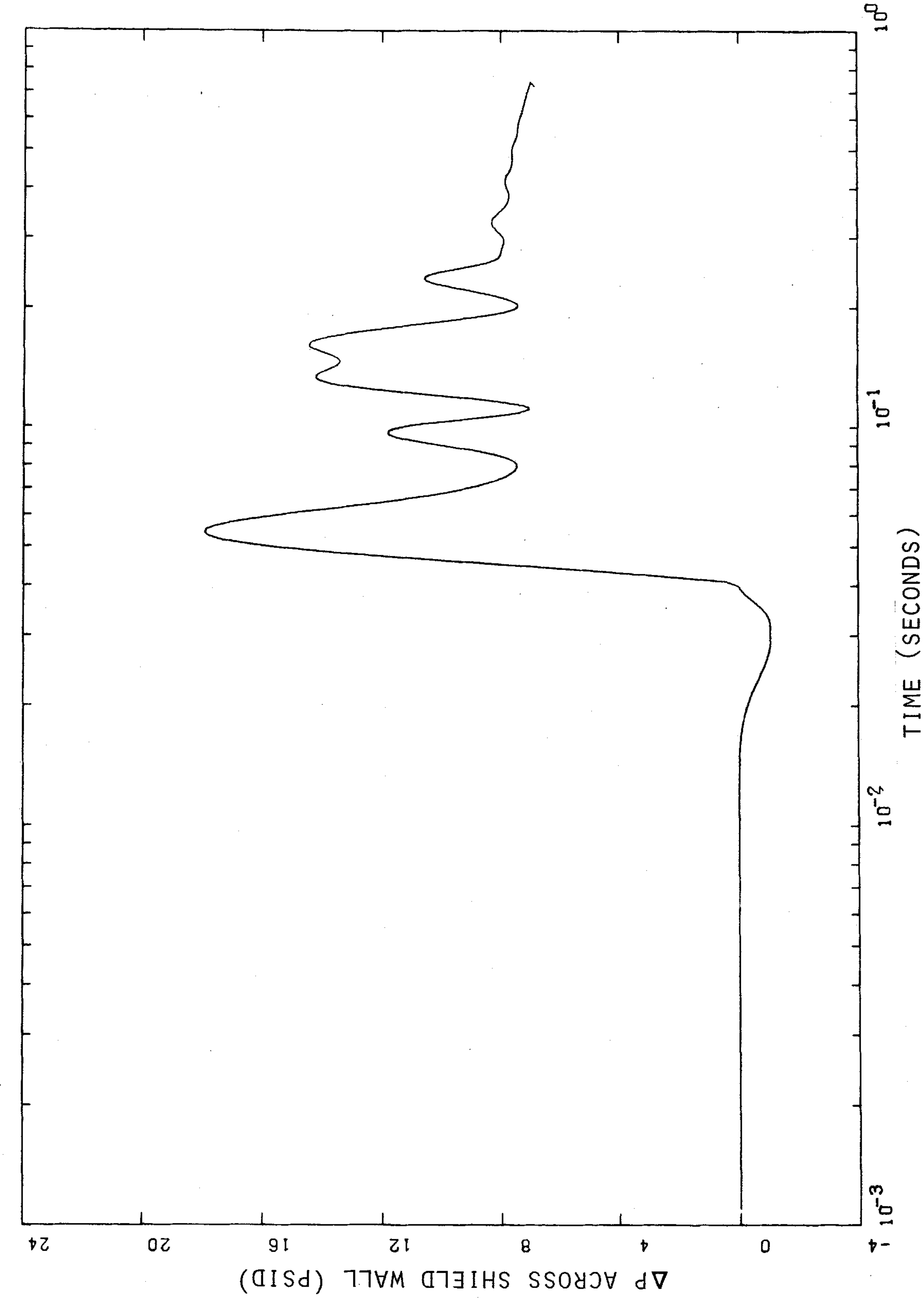


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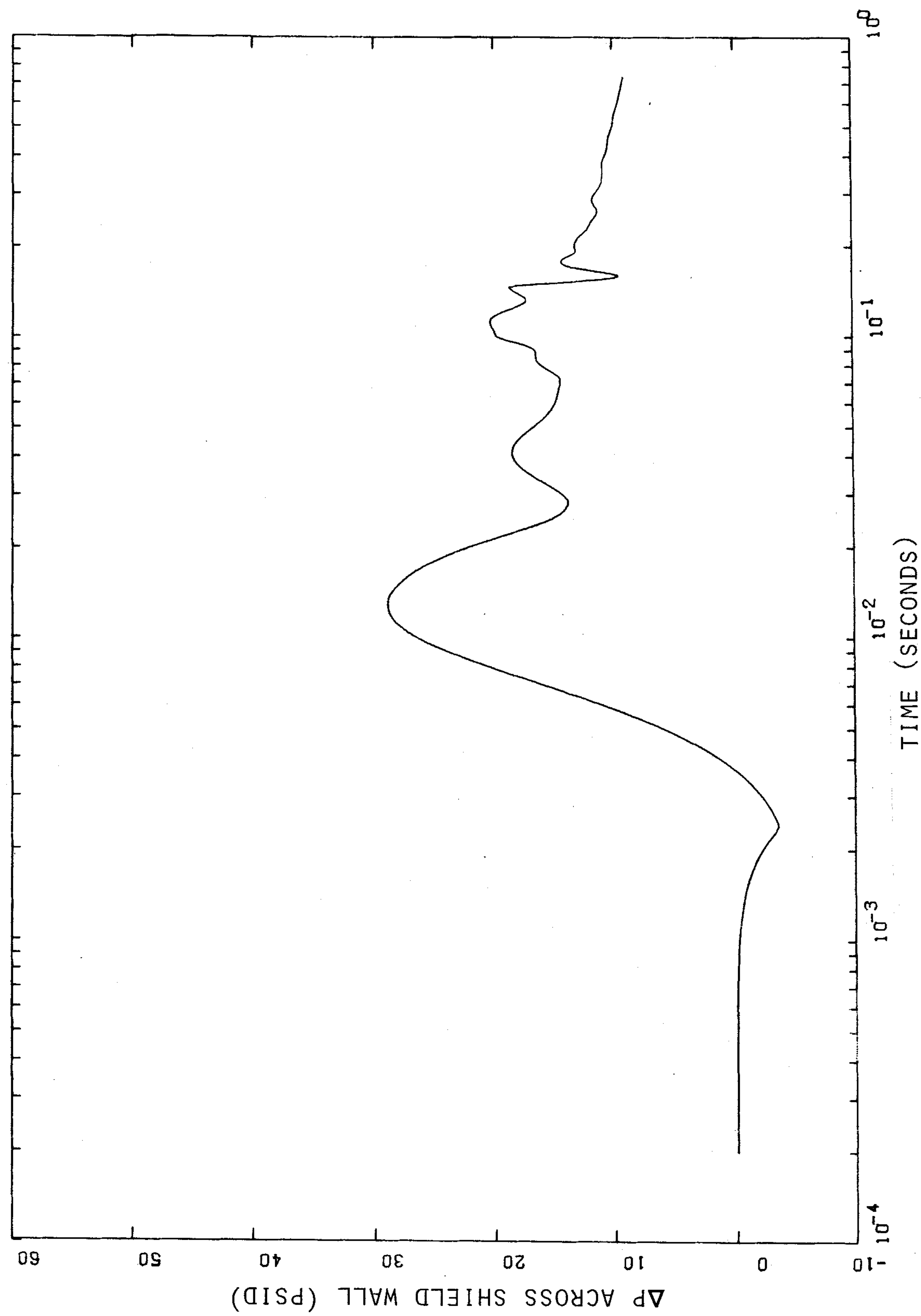
FIGURE 6.2-154 ΔP VS. LOG T FOR NODE 5 (RECIRCULATION INLET LINE BREAK)



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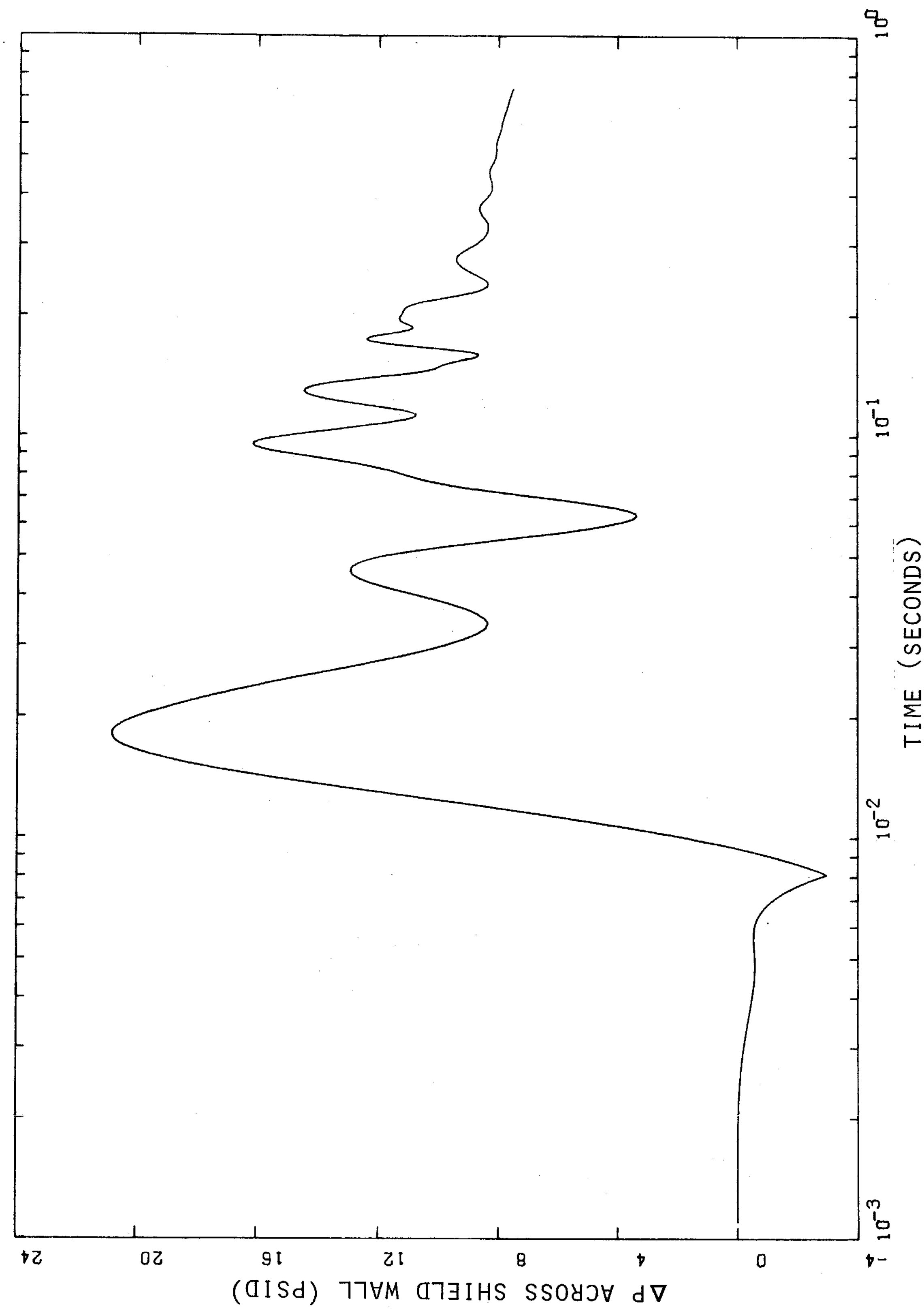
UPDATED SAFETY ANALYSIS REPORT

FIGURE 6.2-155

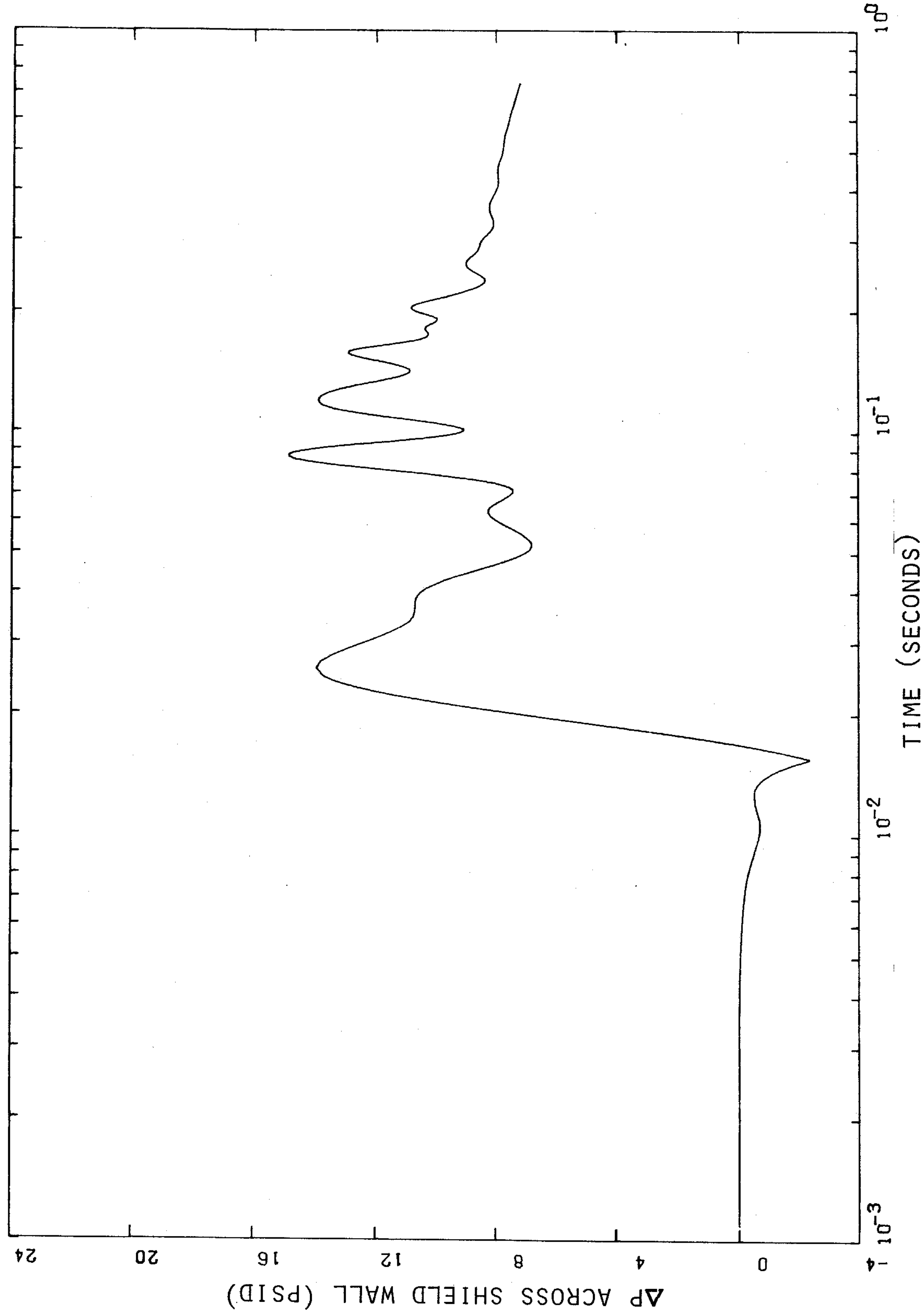
ΔP VS. LOG T FOR NODE 6
(RECIRCULATION INLET LINE BREAK)



CLINTON POWER STATION
UPDATED SAFETY ANALYSIS REPORT

FIGURE 6.2-156

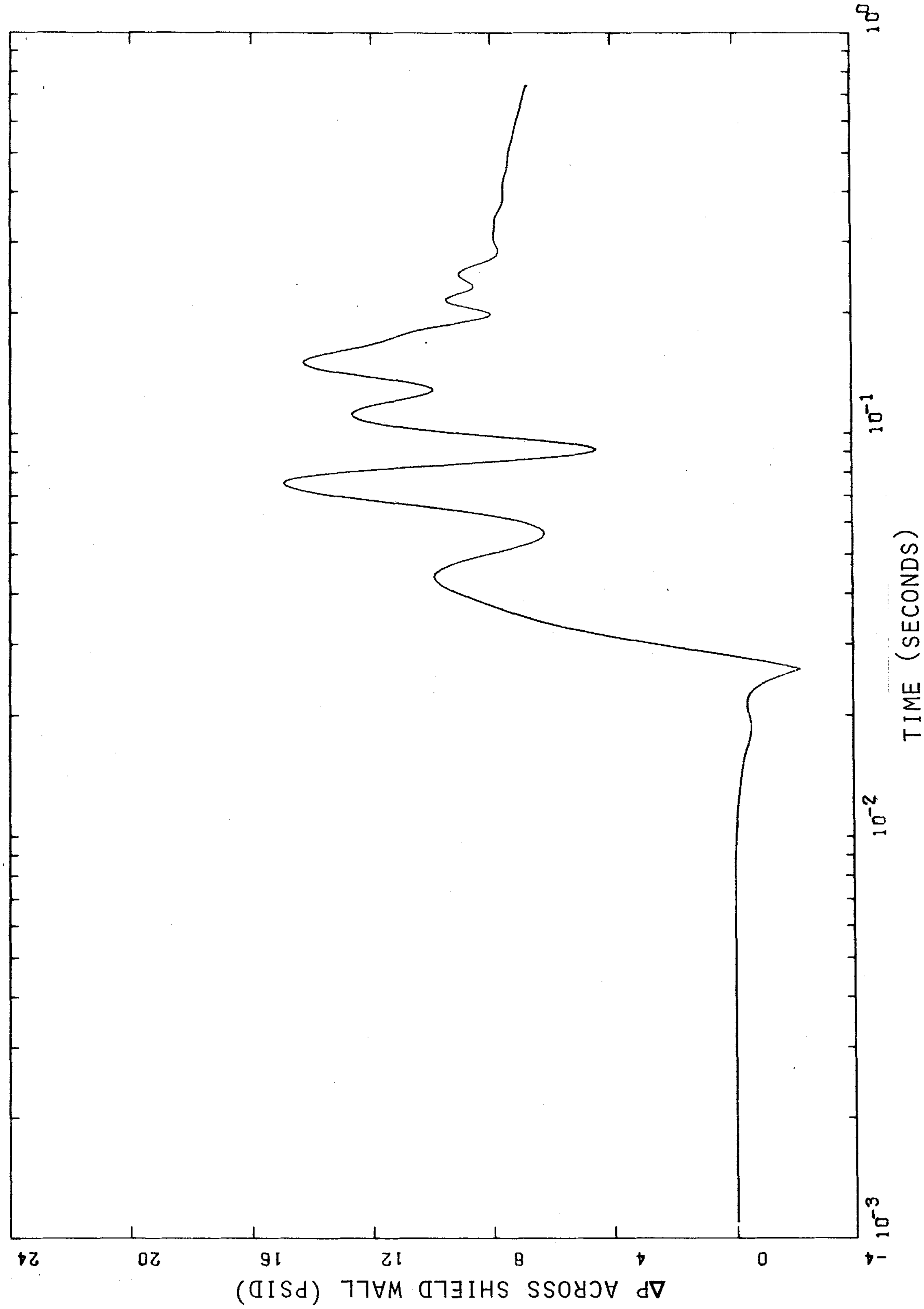
ΔP VS. LOG T FOR NODE 7
(RECIRCULATION INLET LINE BREAK)



CLINTON POWER STATION
UPDATED SAFETY ANALYSIS REPORT

FIGURE 6.2-157

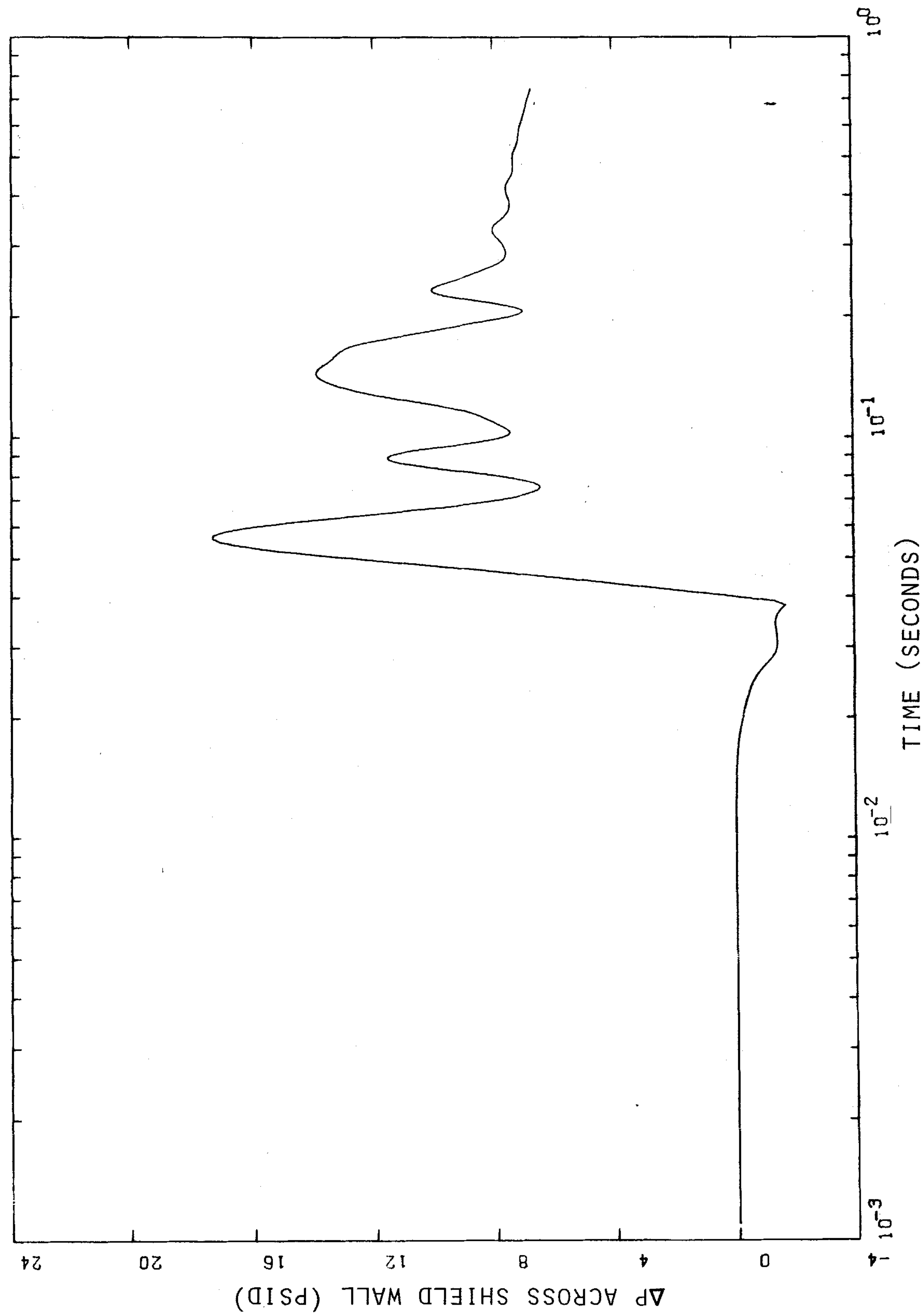
ΔP VS. LOG T FOR NODE 8
(RECIRCULATION INLET LINE BREAK)



CLINTON POWER STATION
UPDATED SAFETY ANALYSIS REPORT

FIGURE 6.2-158

ΔP VS. LOG T FOR NODE 9
(RECIRCULATION INLET LINE BREAK)



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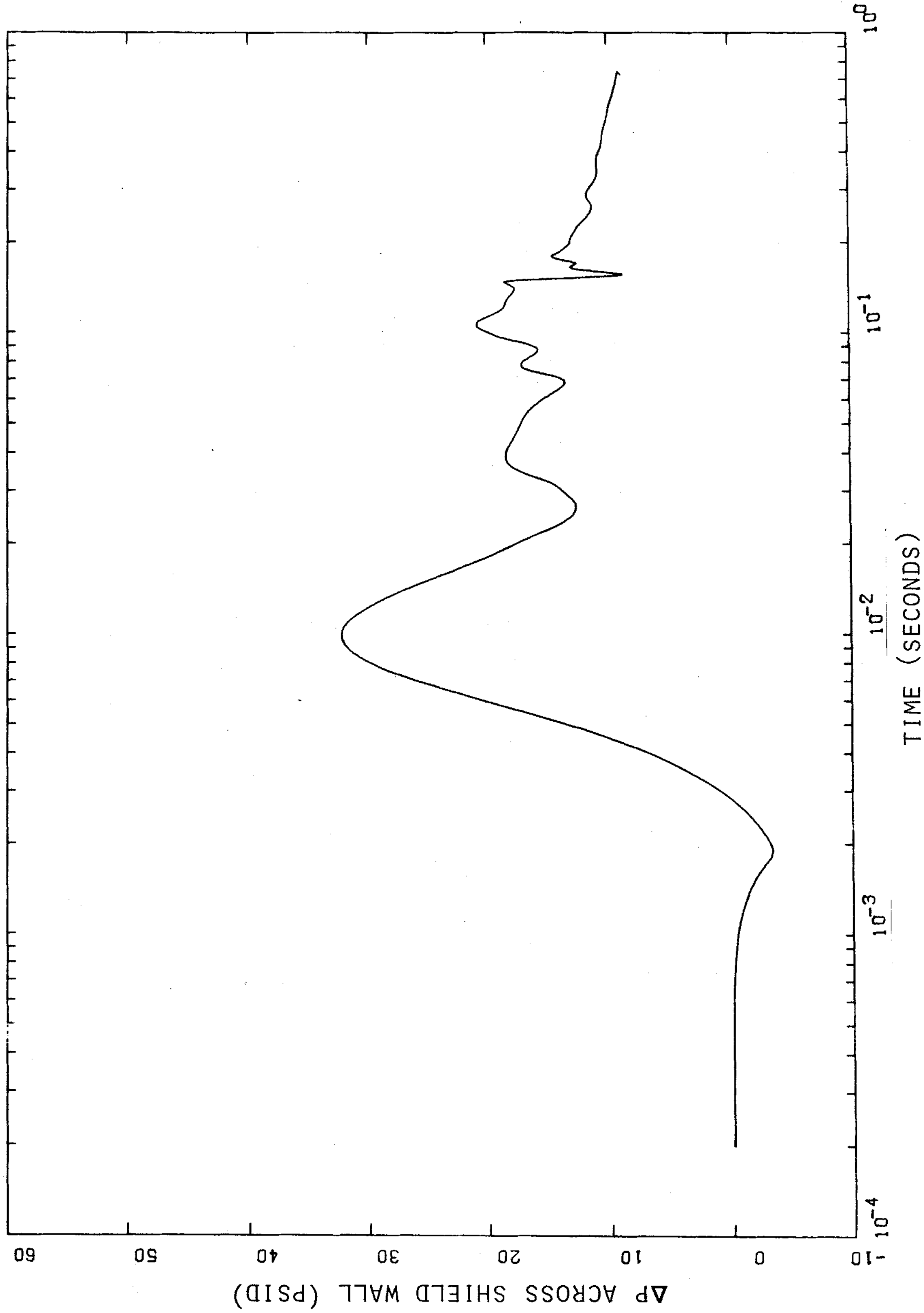
UPDATED SAFETY ANALYSIS REPORT

FIGURE 6.2-159

ΔP VS. LOG T FOR NODE 10
(RECIRCULATION INLET LINE BREAK)



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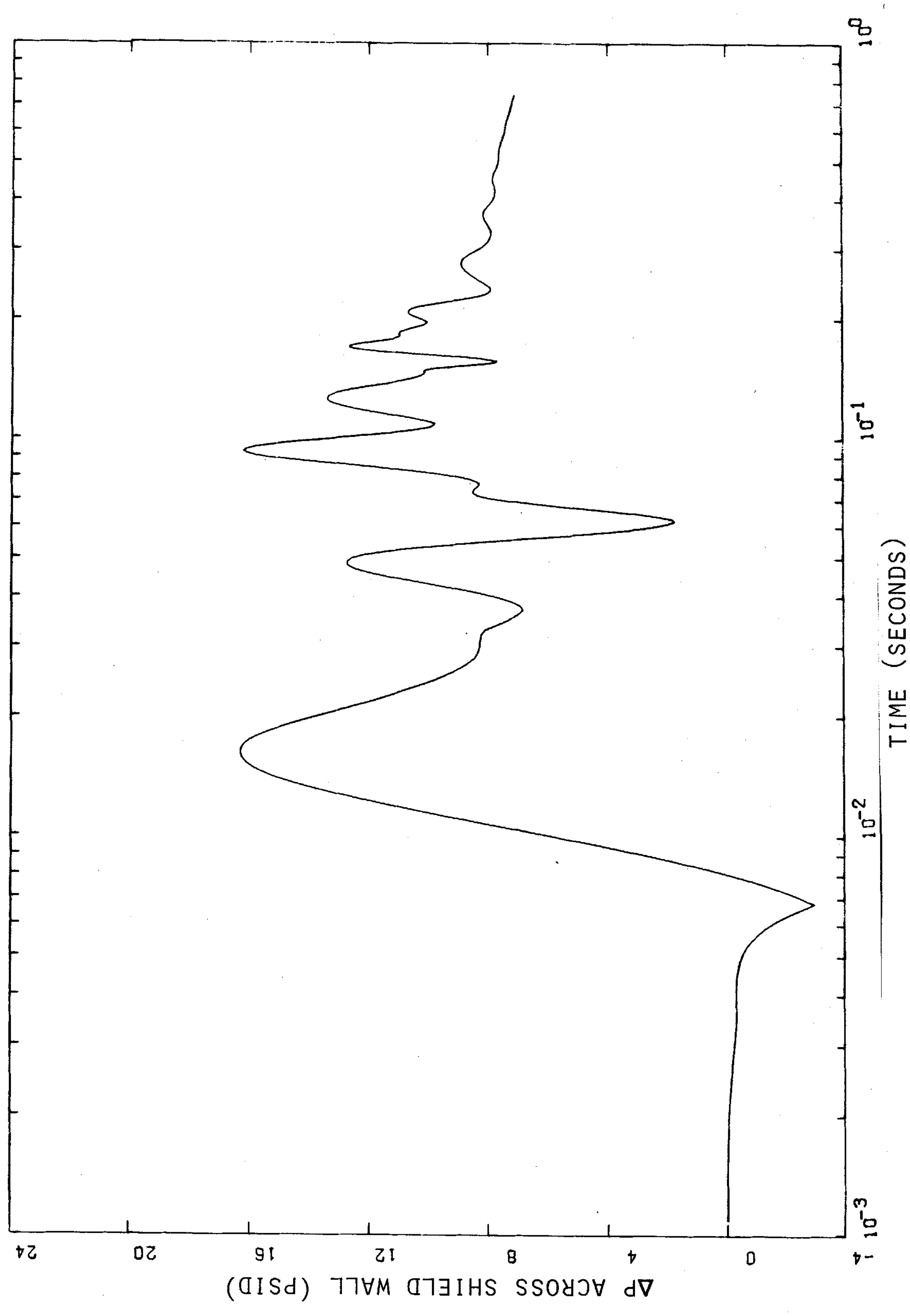
UPDATED SAFETY ANALYSIS REPORT

FIGURE 6.2-160

ΔP VS. LOG T FOR NODE 11
(RECIRCULATION INLET LINE BREAK)



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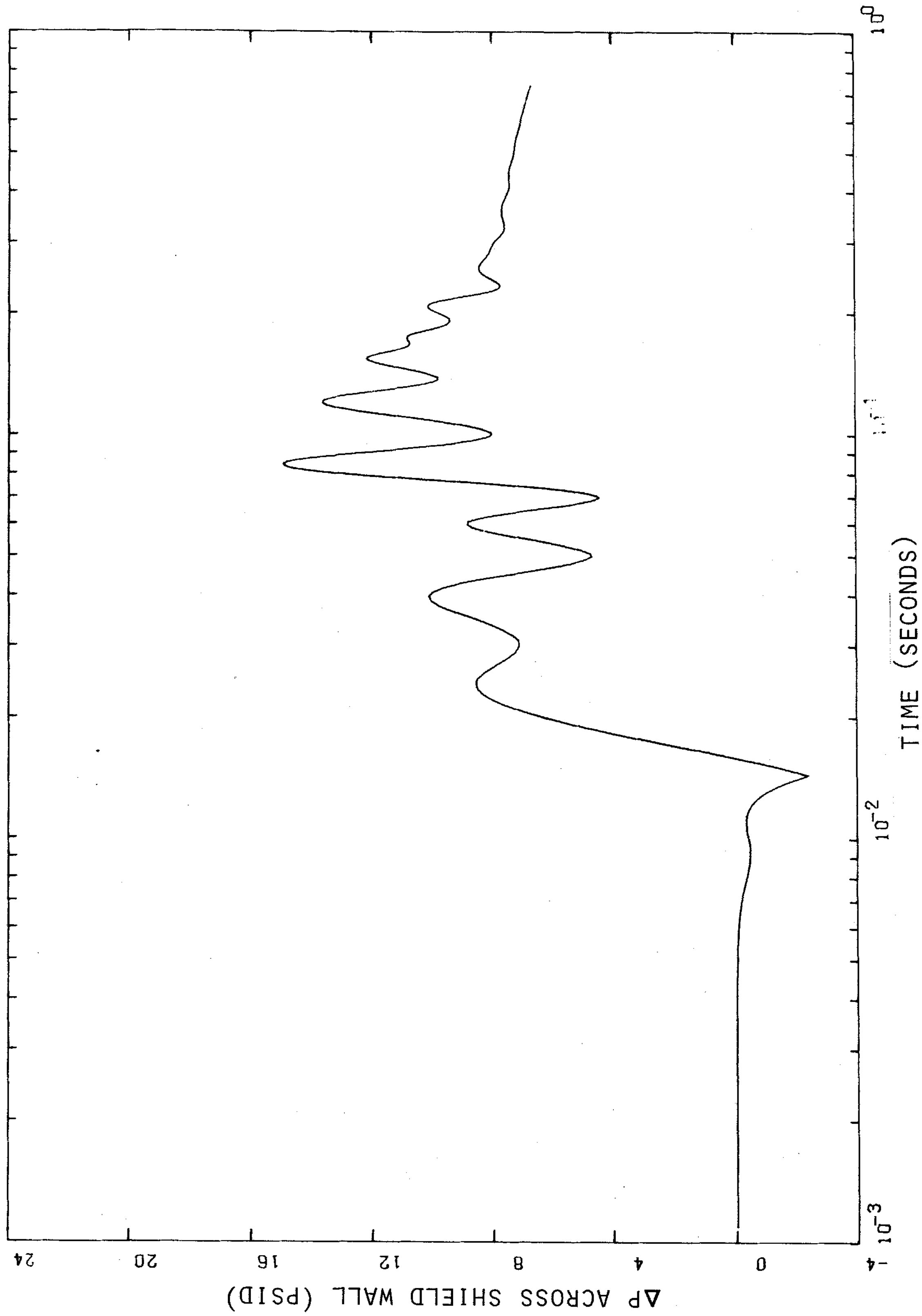
UPDATED SAFETY ANALYSIS REPORT

FIGURE 6.2-161

ΔP VS. LOG T FOR NODE 12
(RECIRCULATION INLET LINE BREAK)



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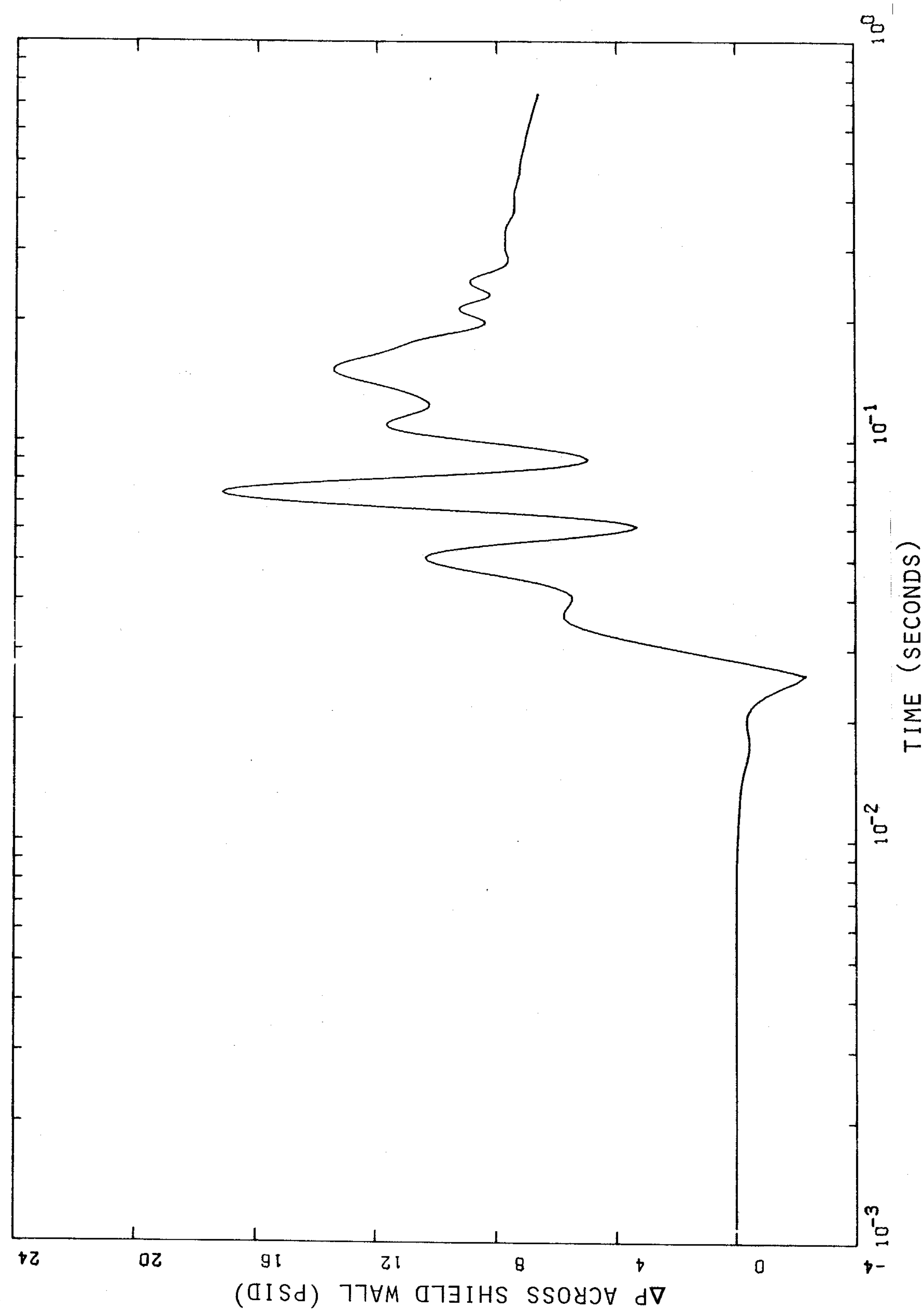
UPDATED SAFETY ANALYSIS REPORT

FIGURE 6.2-162

ΔP VS. LOG T FOR NODE 13
(RECIRCULATION INLET LINE BREAK)



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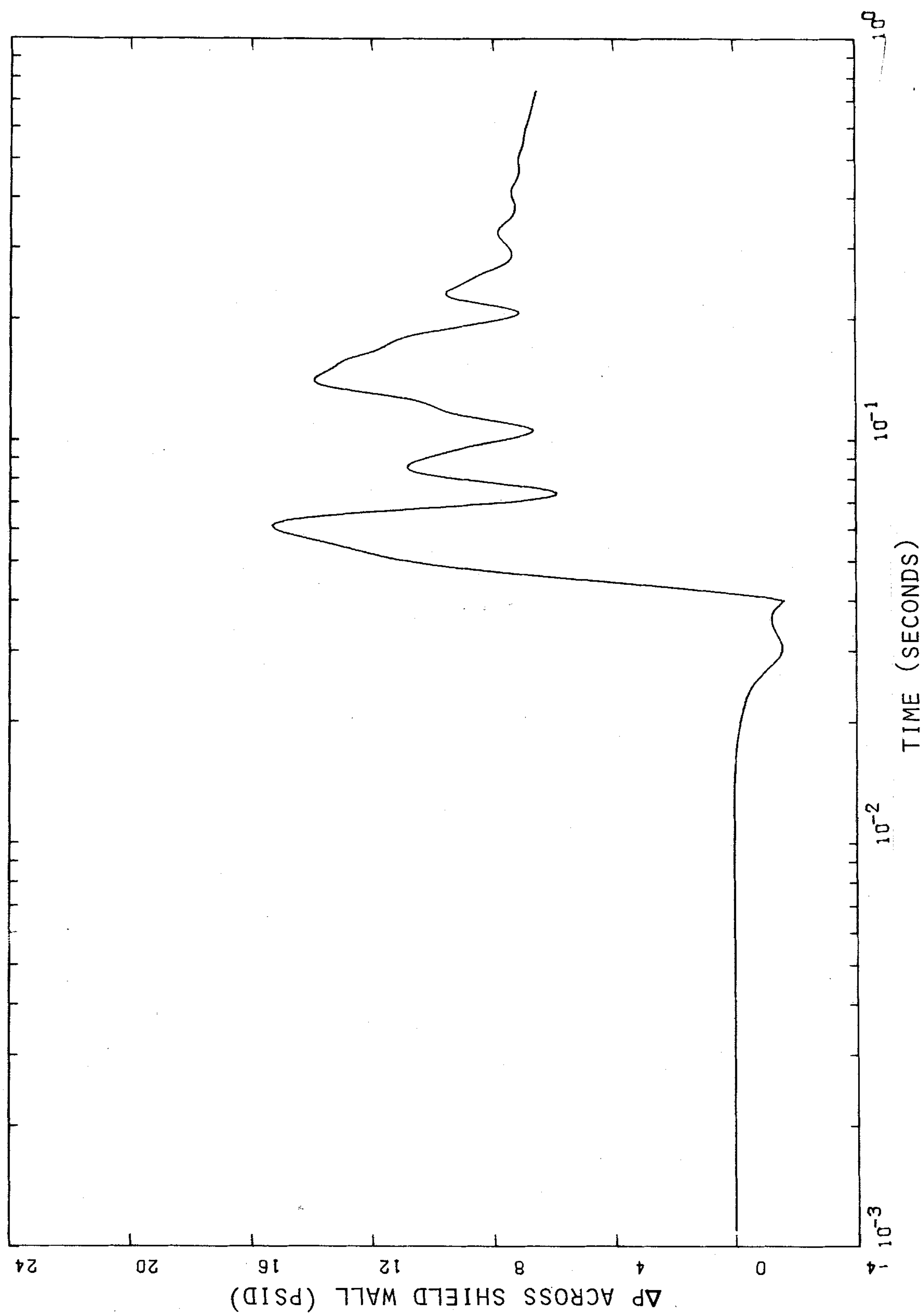
UPDATED SAFETY ANALYSIS REPORT

FIGURE 6.2-163

ΔP VS. LOG T FOR NODE 14
(RECIRCULATION INLET LINE BREAK)



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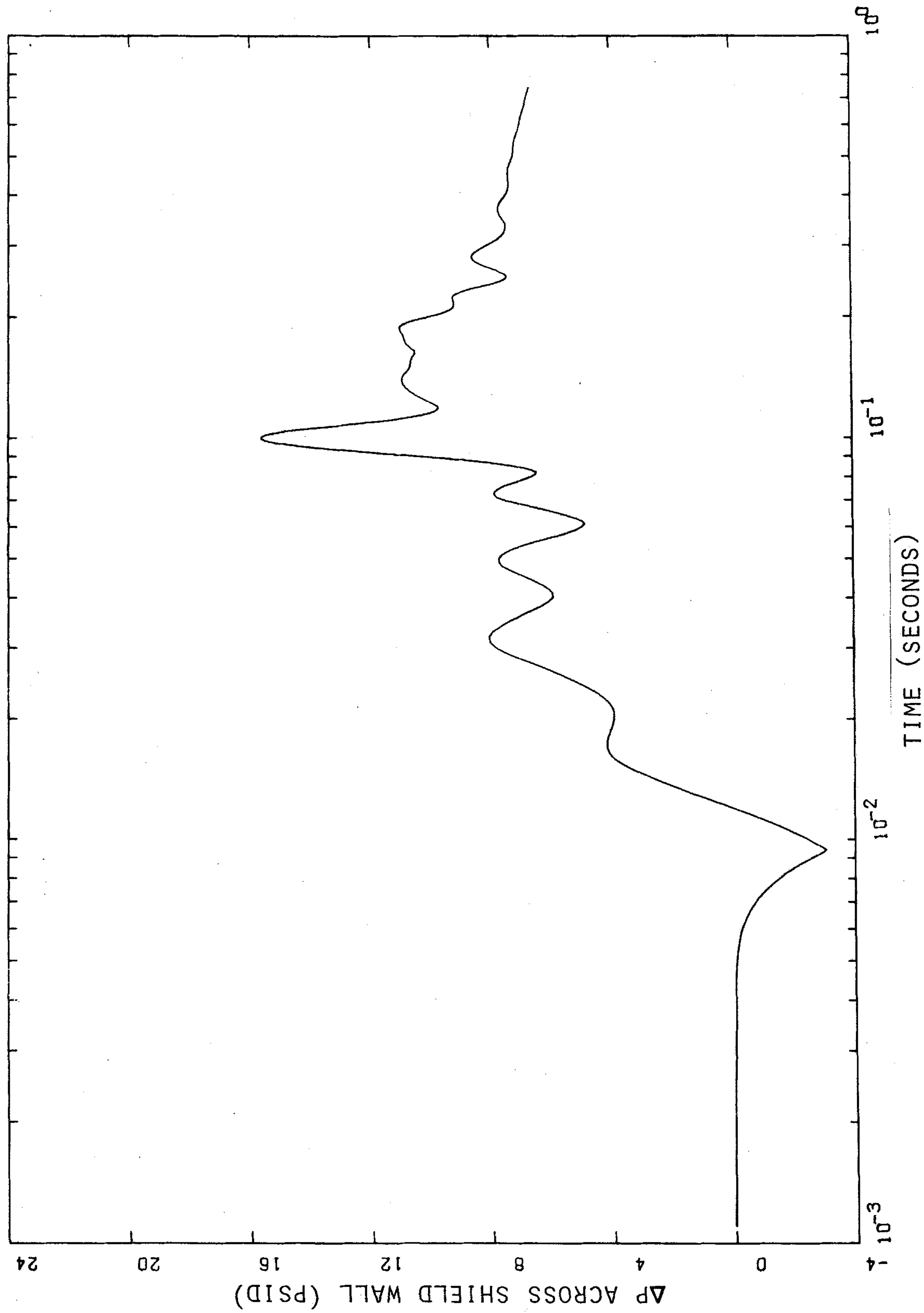
UPDATED SAFETY ANALYSIS REPORT

FIGURE 6.2-164

ΔP VS. LOG T FOR NODE 15
(RECIRCULATION INLET LINE BREAK)



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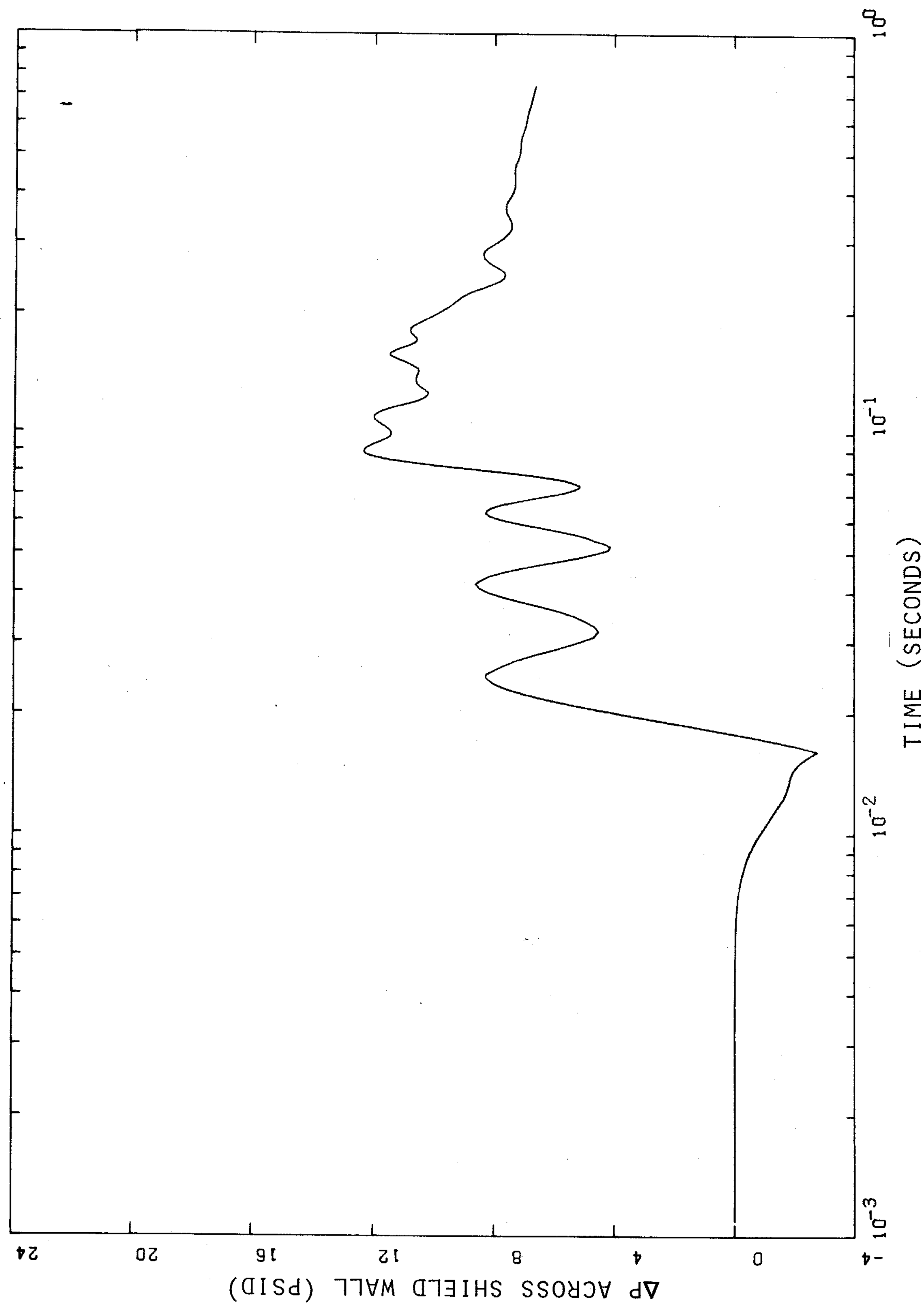
UPDATED SAFETY ANALYSIS REPORT

FIGURE 6.2-165

ΔP VS. LOG T FOR NODE 16
(RECIRCULATION INLET LINE BREAK)



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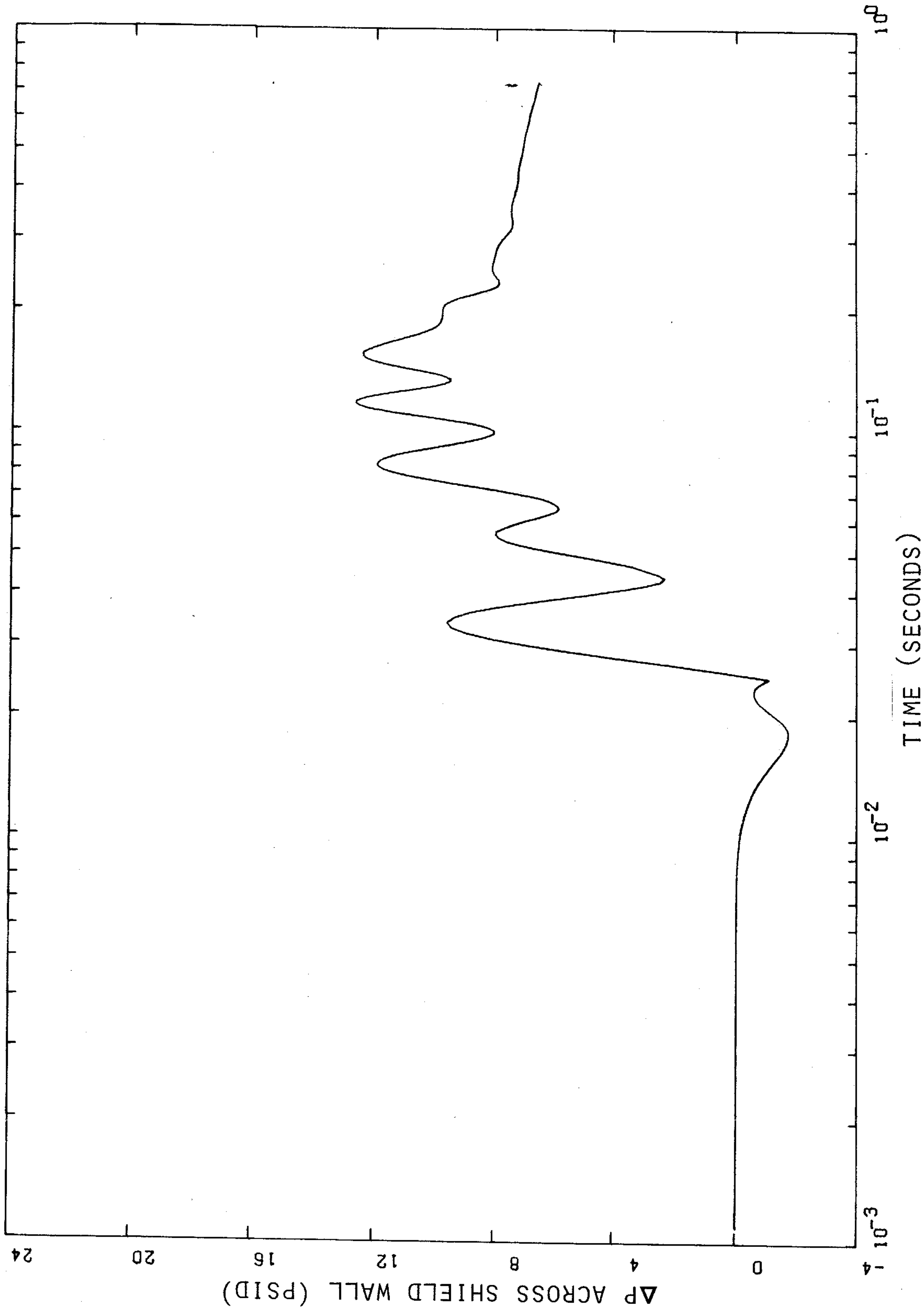
UPDATED SAFETY ANALYSIS REPORT

FIGURE 6.2-166

ΔP VS. LOG T FOR NODE 17
(RECIRCULATION INLET LINE BREAK)



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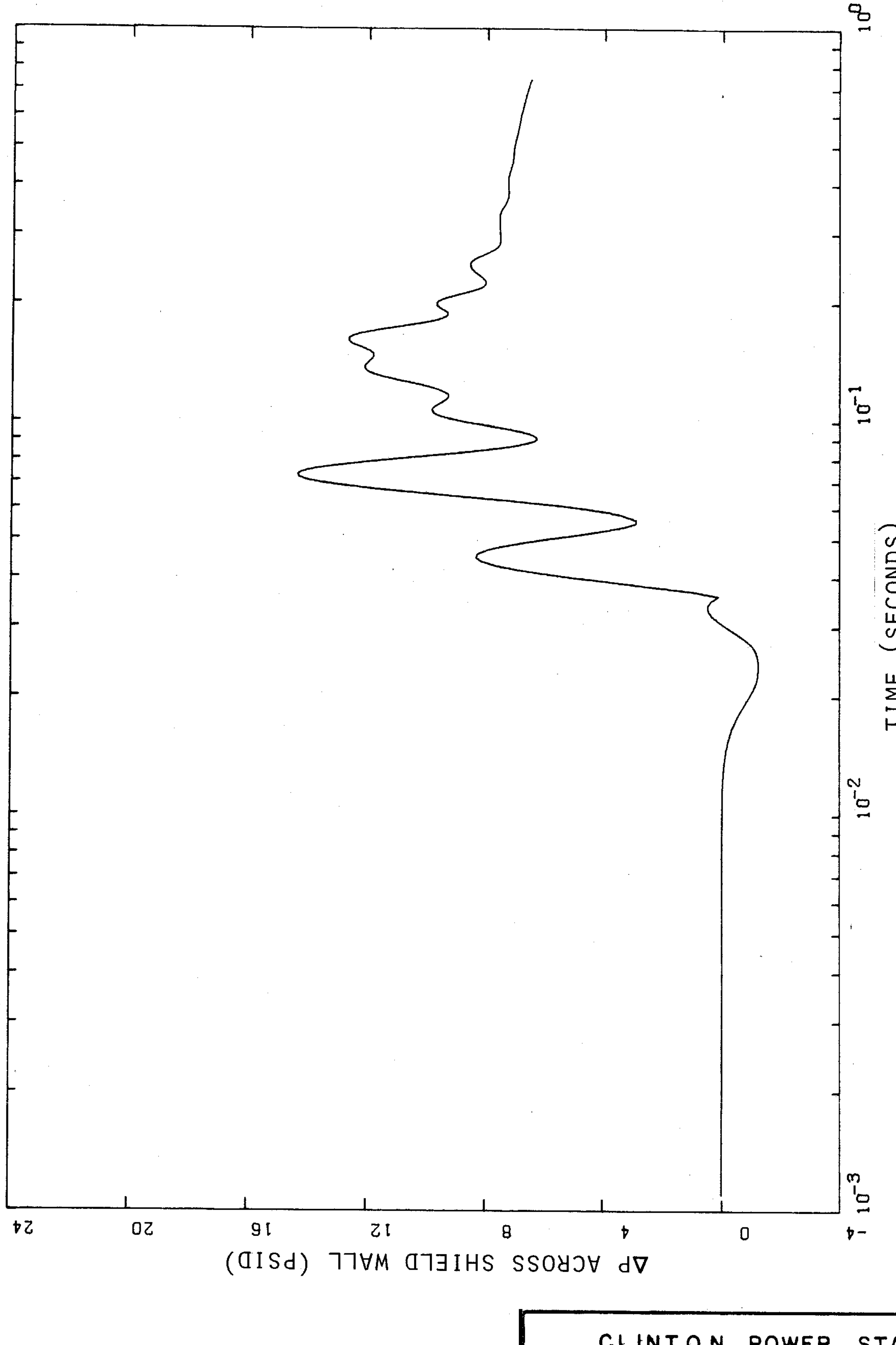
UPDATED SAFETY ANALYSIS REPORT

FIGURE 6.2-167

ΔP VS. LOG T FOR NODE 18
(RECIRCULATION INLET LINE BREAK)



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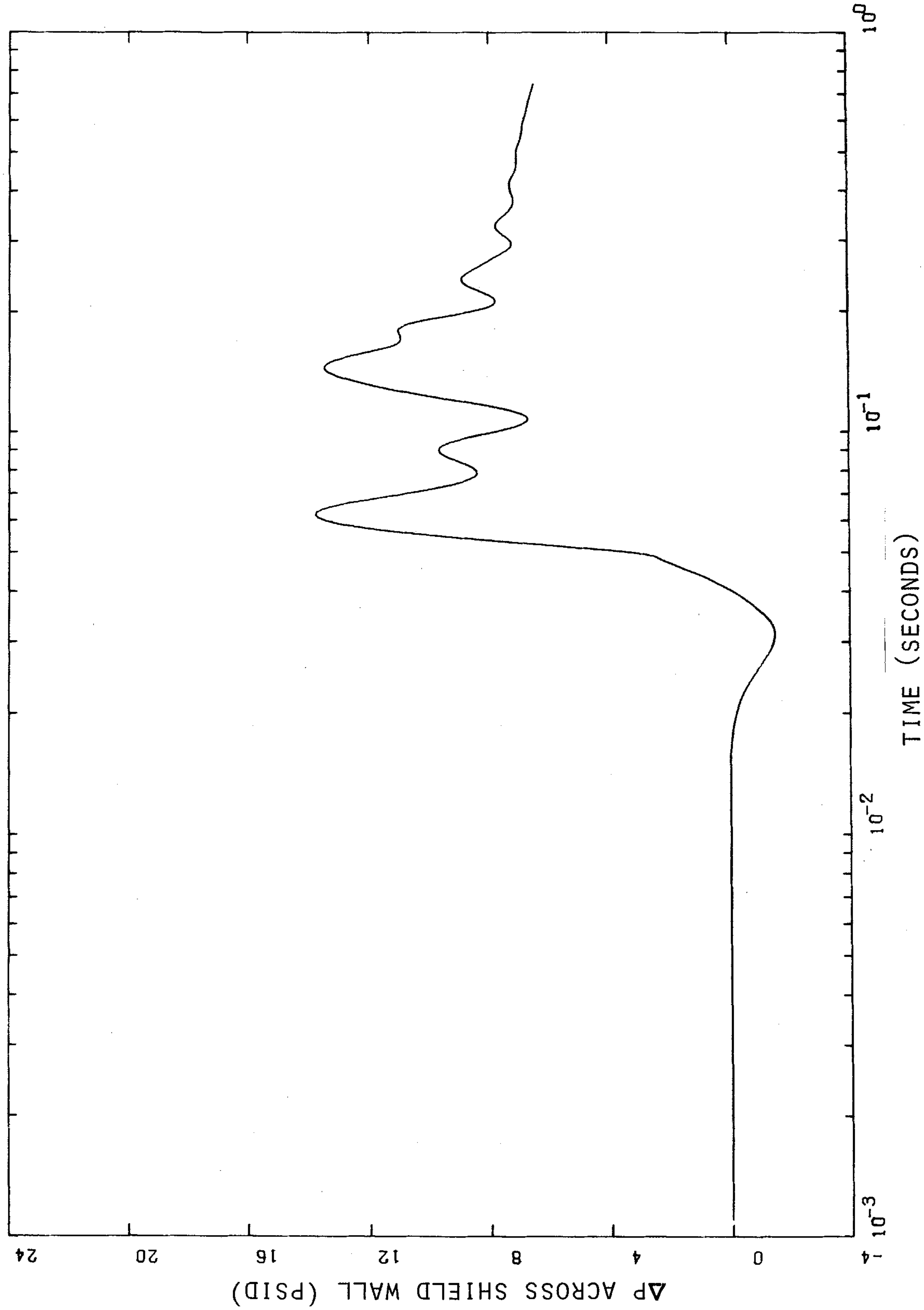
UPDATED SAFETY ANALYSIS REPORT

FIGURE 6.2-168

ΔP VS. LOG T FOR NODE 19
(RECIRCULATION INLET LINE BREAK)



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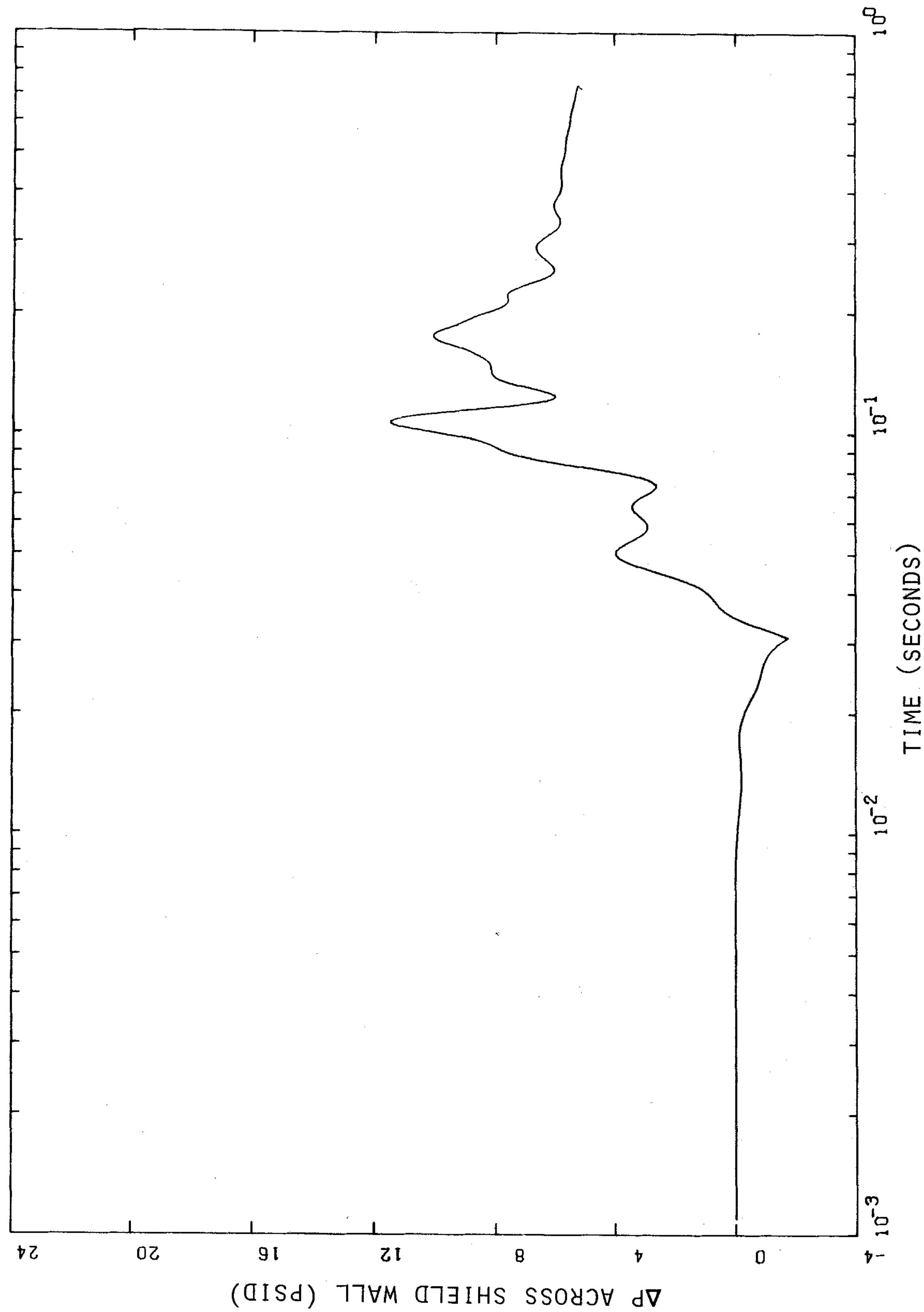
UPDATED SAFETY ANALYSIS REPORT

FIGURE 6.2-169

ΔP VS. LOG T FOR NODE 20
(RECIRCULATION INLET LINE BREAK)



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UPDATED SAFETY ANALYSIS REPORT

FIGURE 6.2-170

ΔP VS. LOG T FOR NODE 21
(RECIRCULATION INLET LINE BREAK)

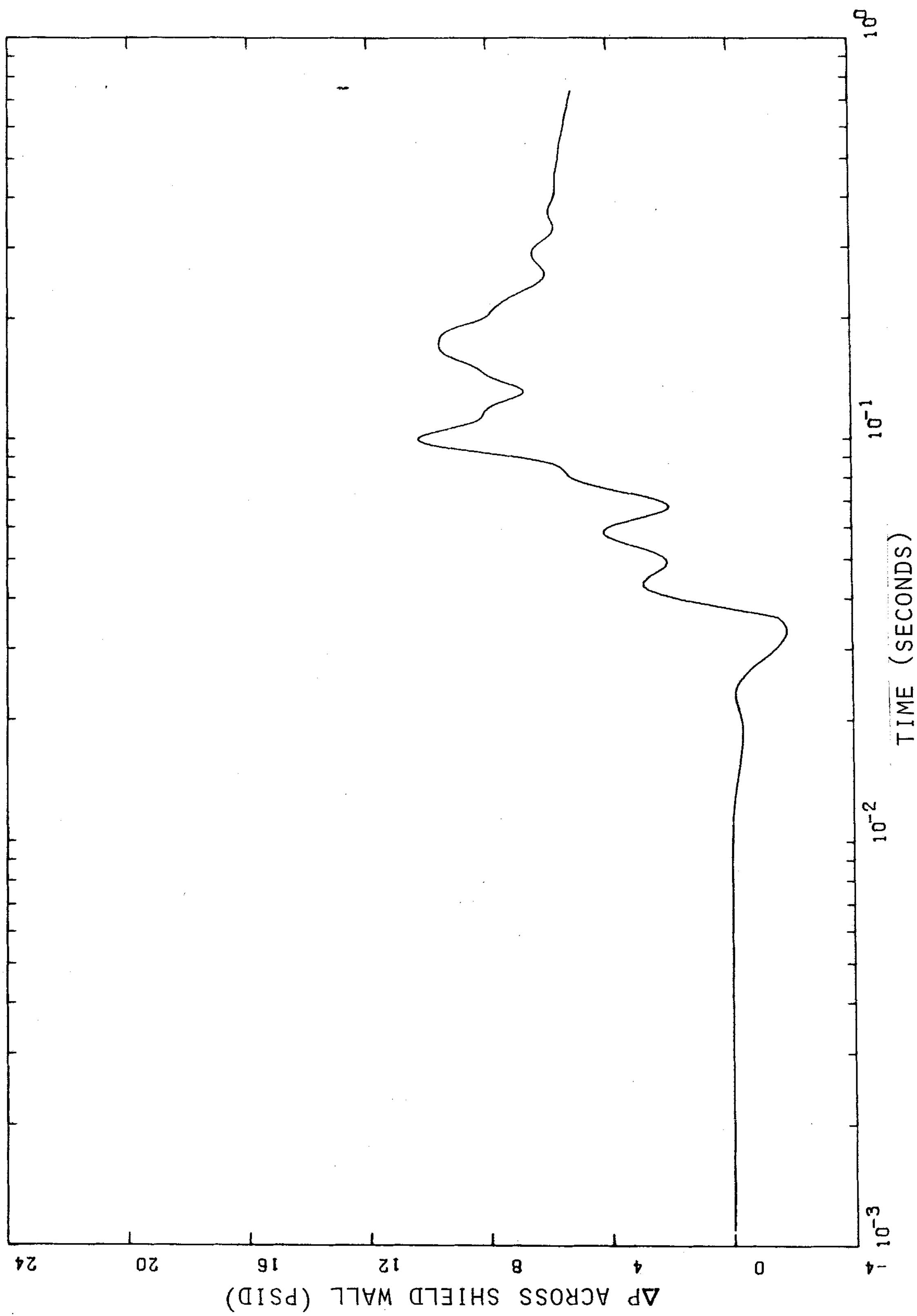

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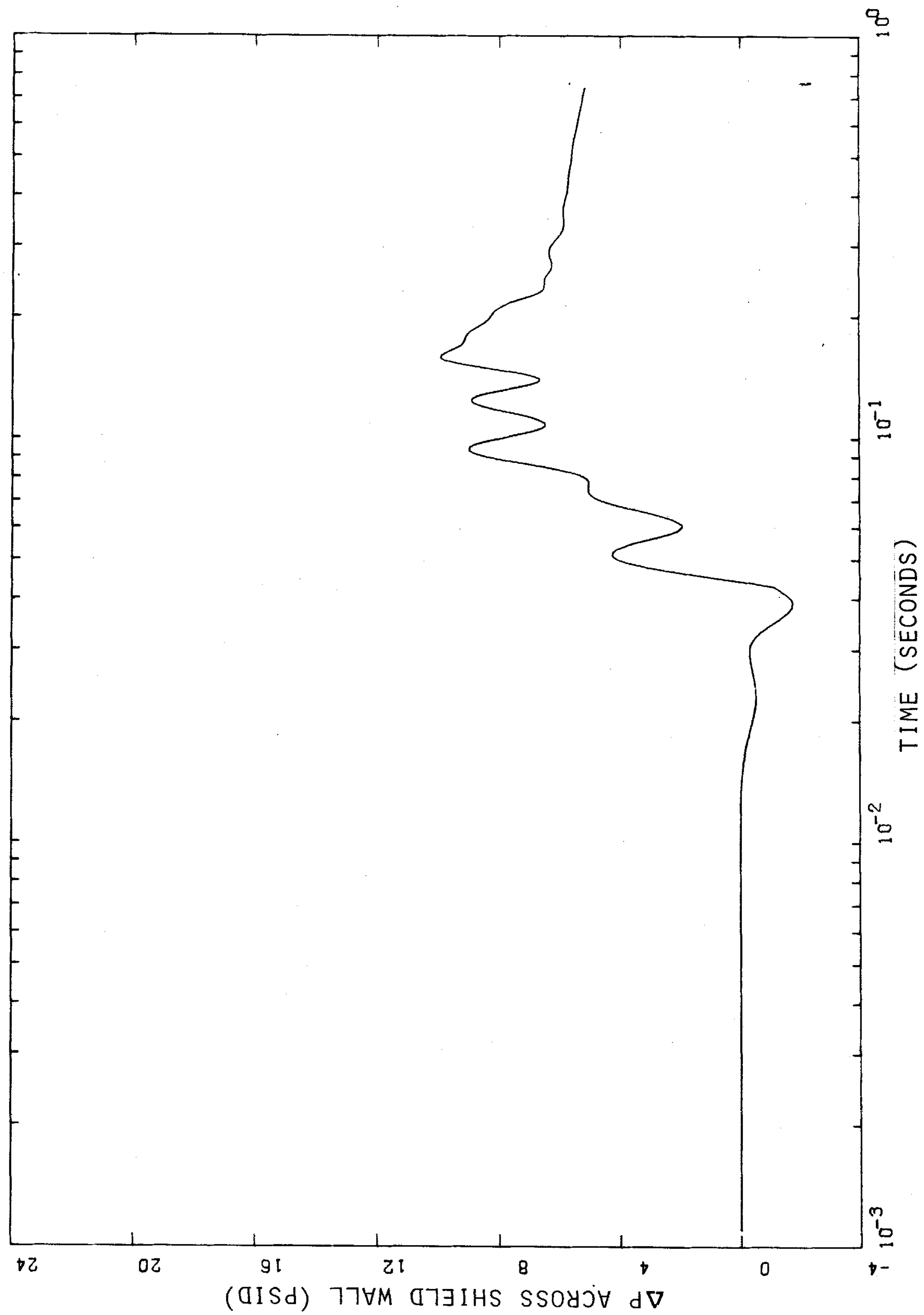
FIGURE 6.2-171

ΔP VS. LOG T FOR NODE 22
(RECIRCULATION INLET LINE BREAK)



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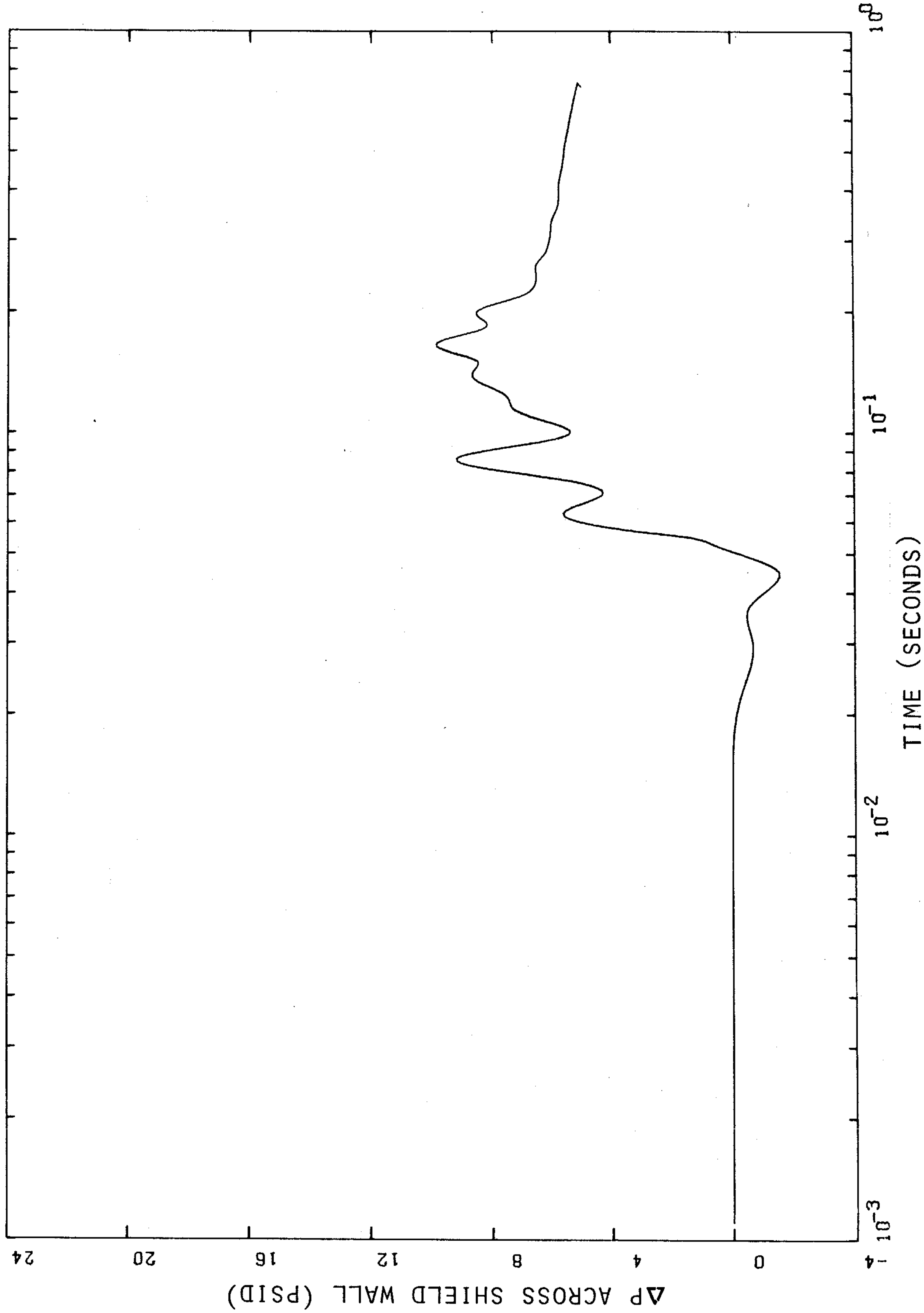
UPDATED SAFETY ANALYSIS REPORT

FIGURE 6.2-172

ΔP VS. LOG T FOR NODE 23
(RECIRCULATION INLET LINE BREAK)



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UPDATED SAFETY ANALYSIS REPORT

FIGURE 6.2-173

ΔP VS. LOG T FOR NODE 24
(RECIRCULATION INLET LINE BREAK)

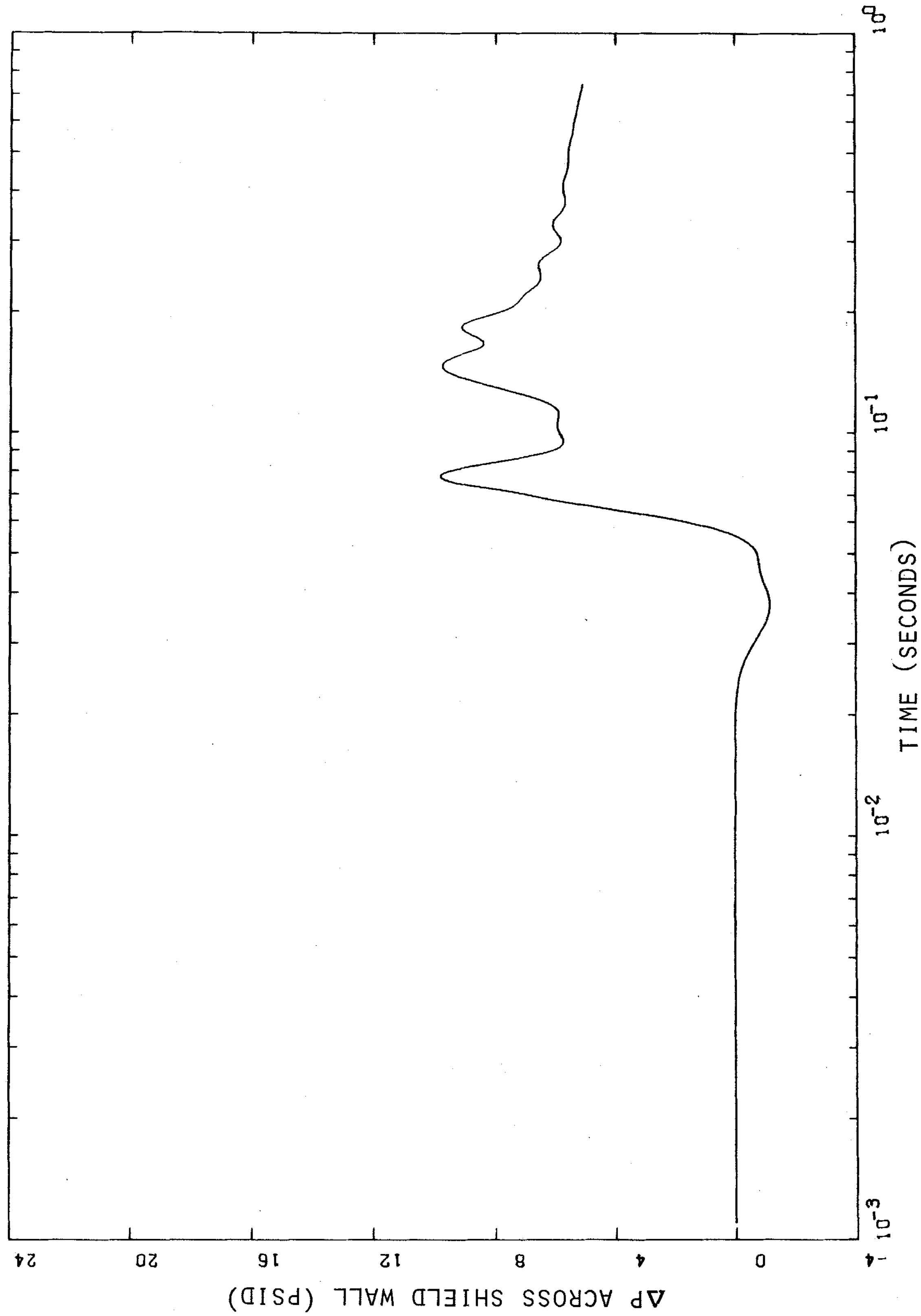

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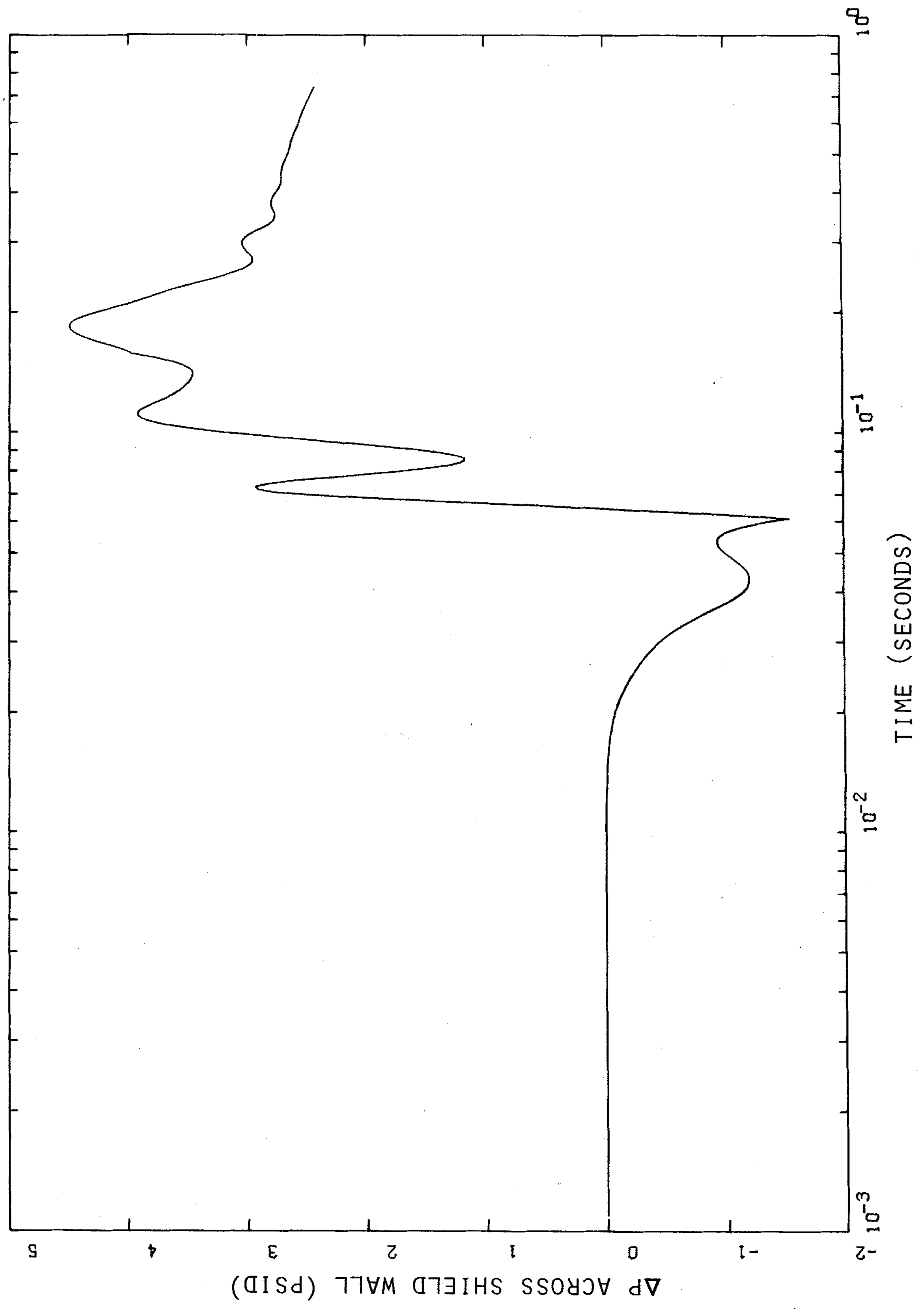
FIGURE 6.2-174

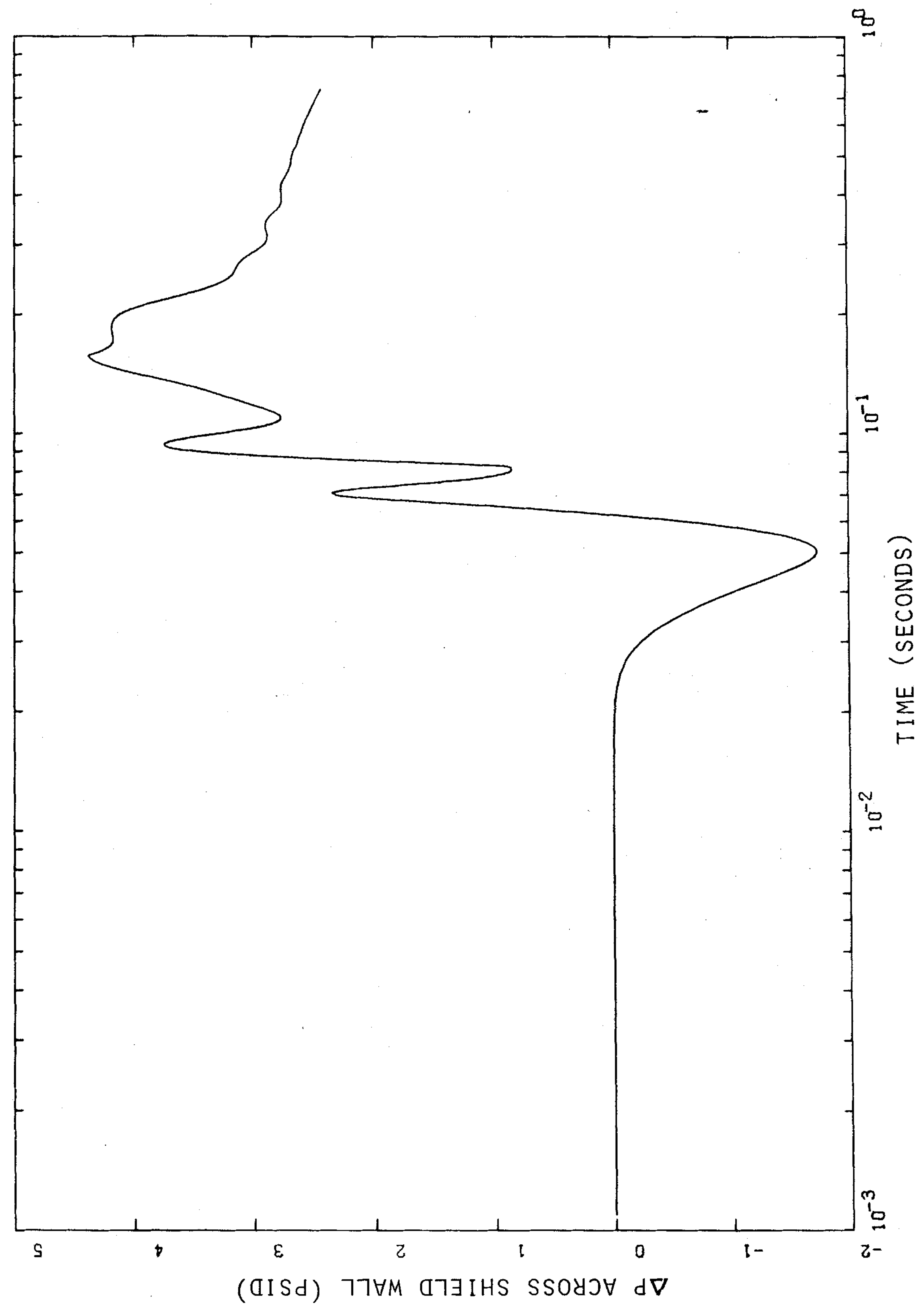
ΔP VS. LOG T FOR NODE 25
(RECIRCULATION INLET LINE BREAK)



CLINTON POWER STATION
UPDATED SAFETY ANALYSIS REPORT

FIGURE 6.2-175

ΔP VS. LOG T FOR NODE 26
(RECIRCULATION INLET LINE BREAK)

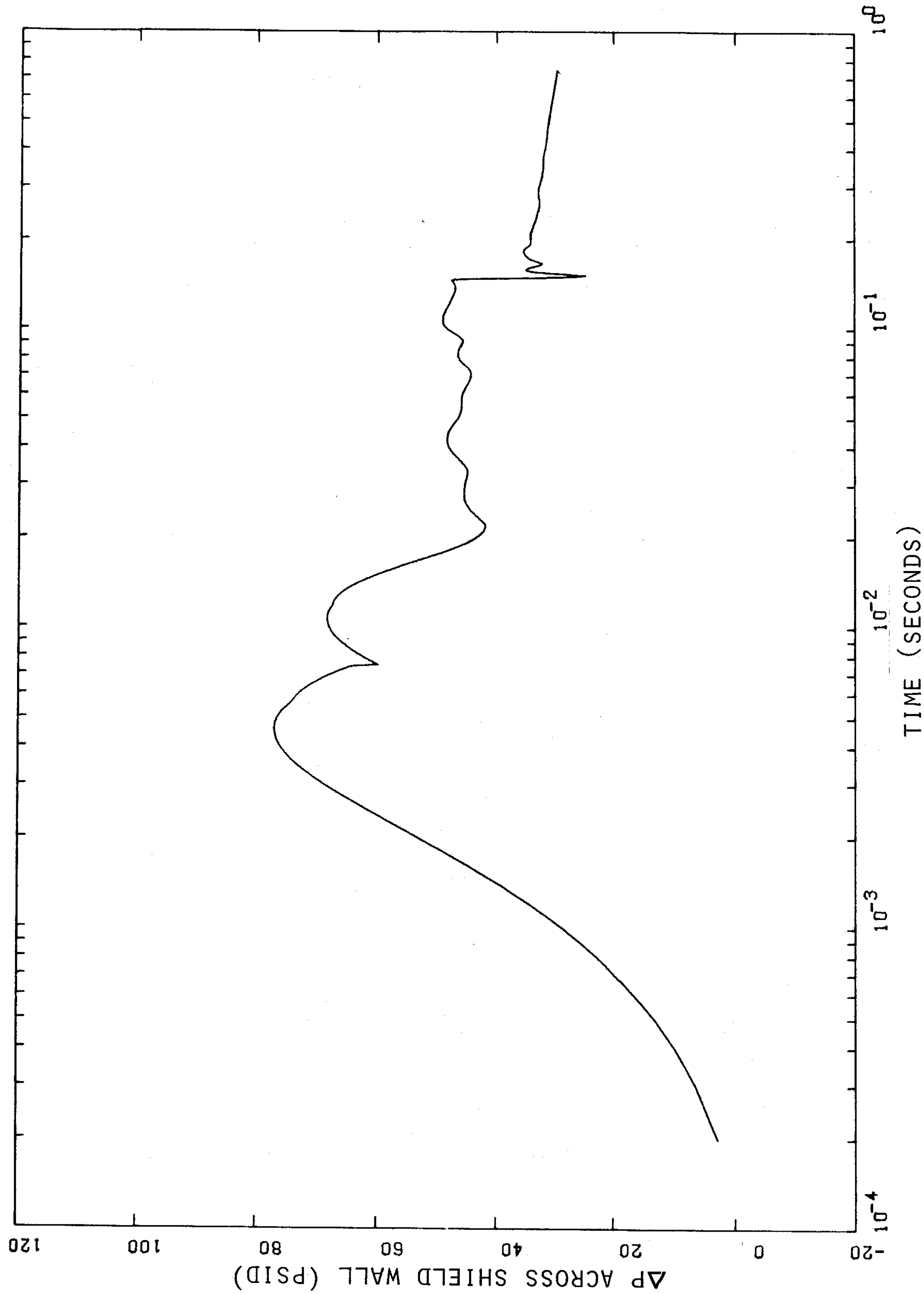


CLINTON POWER STATION

UPDATED SAFETY ANALYSIS REPORT

FIGURE 6.2-176

ΔP VS. LOG T FOR NODE 27
(RECIRCULATION INLET LINE BREAK)



CLINTON POWER STATION
UPDATED SAFETY ANALYSIS REPORT

FIGURE 6.2-177

ΔP VS. LOG T FOR BREAK NODE
(RECIRCULATION INLET LINE BREAK)

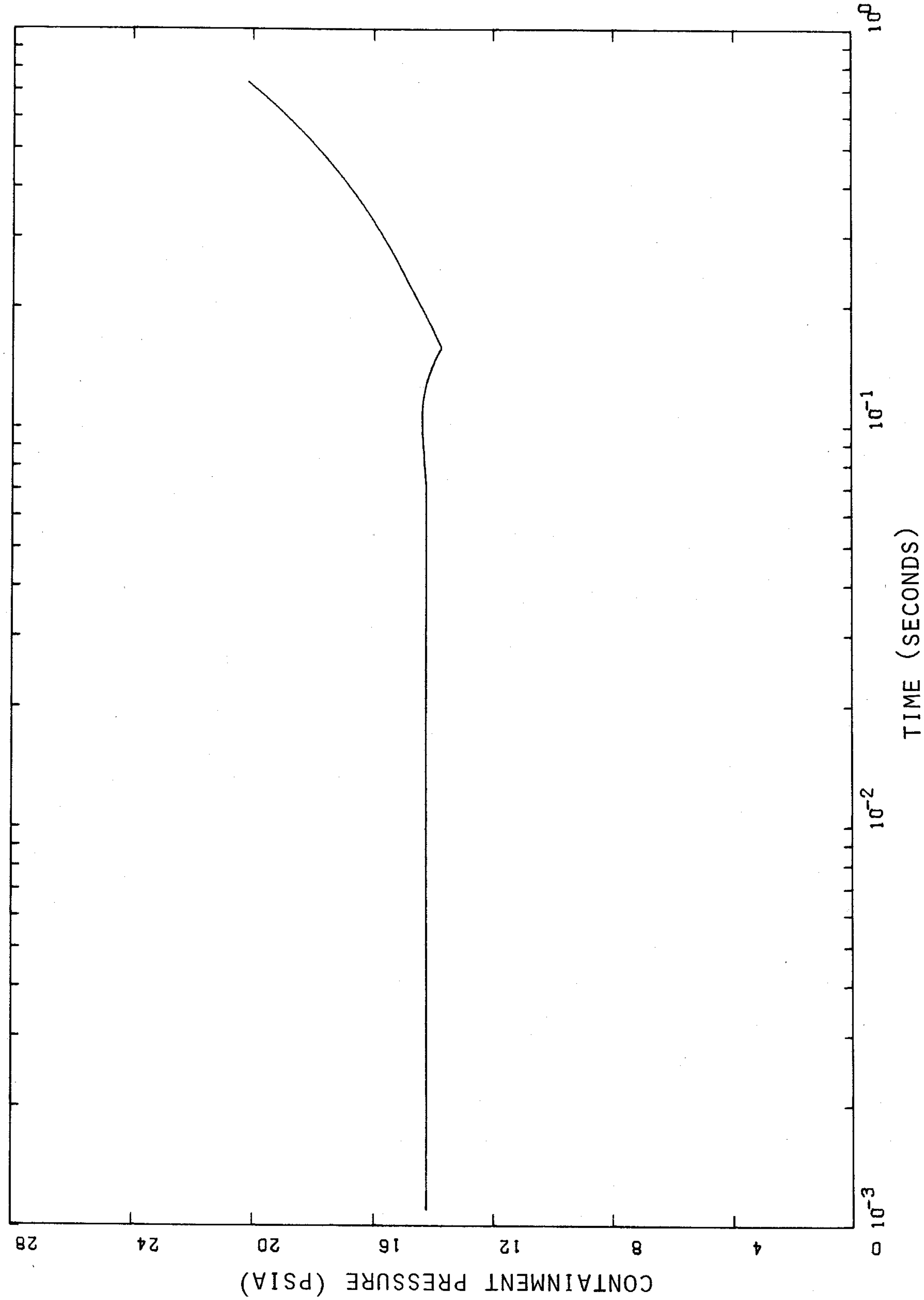


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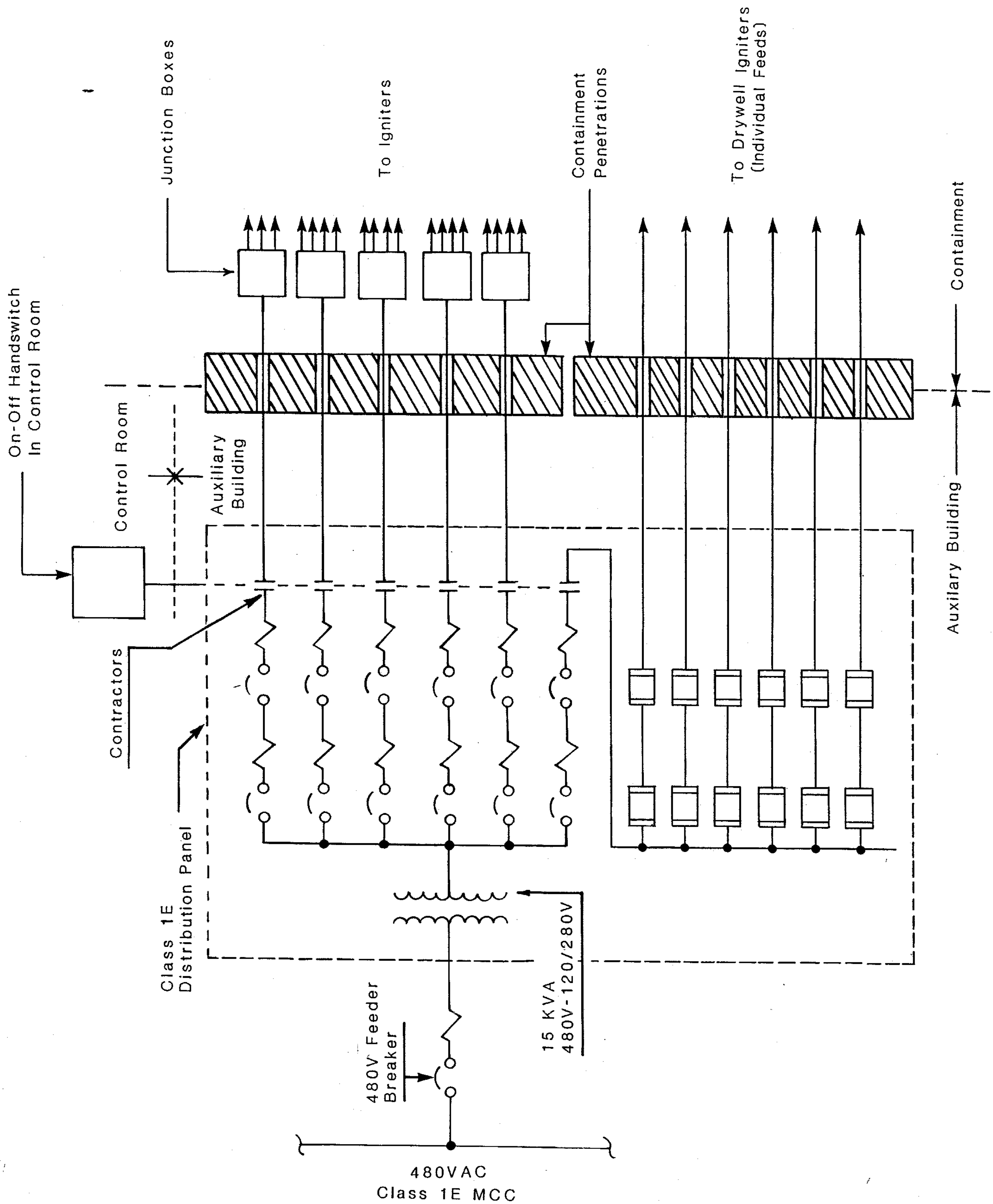
FIGURE 6.2-178

CONTAINMENT PRESSURE VS. LOG T
(RECIRCULATION INLET LINE BREAK)



CLINTON POWER STATION
UPDATED SAFETY ANALYSIS REPORT

FIGURE 6.2-179

SINGLE LINE DIAGRAM
IGNITER POWER SUPPLY FOR ONE DIVISION

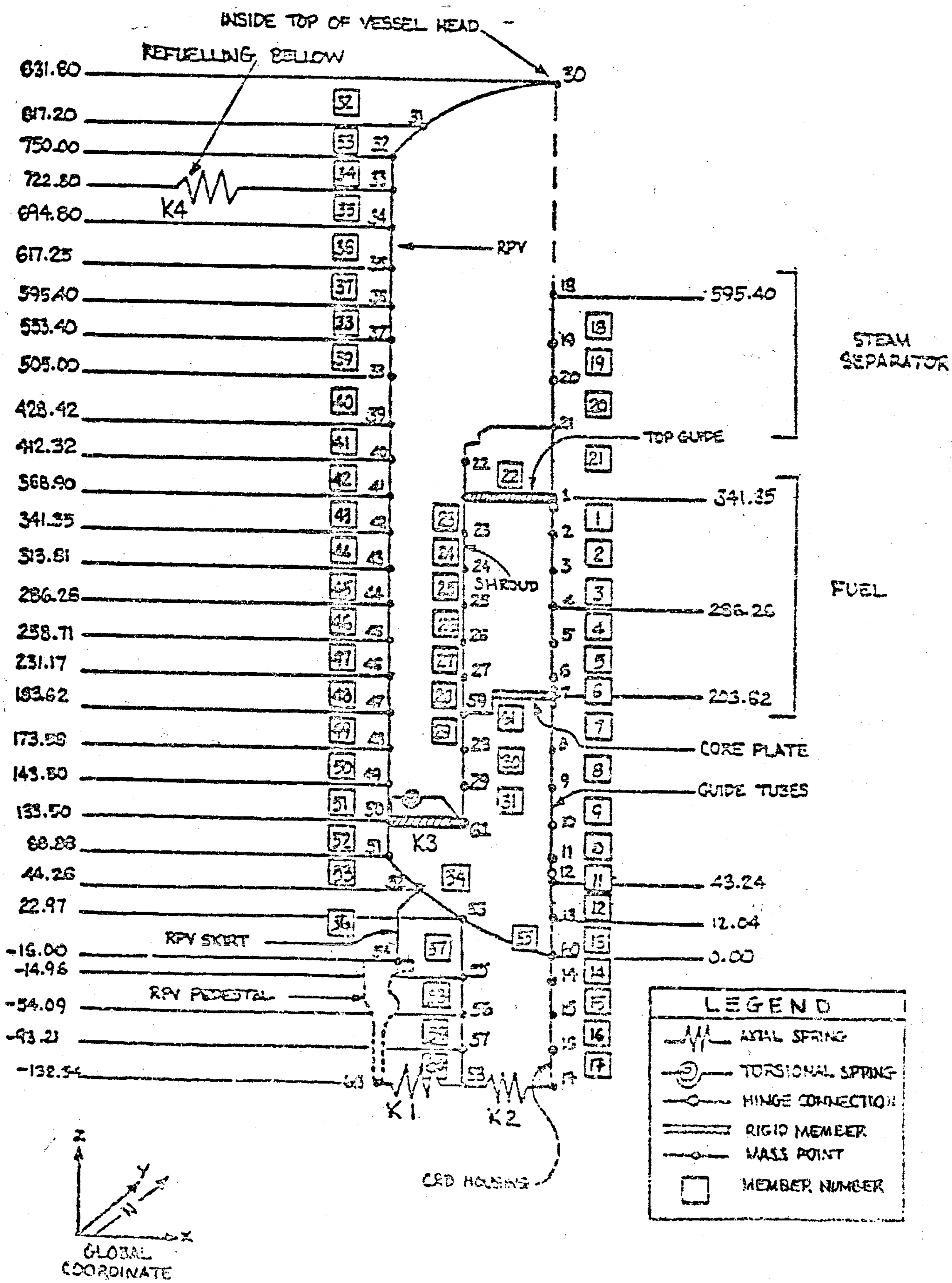


Figure 6.2-180 Reactor Pressure Vessel and Internals Horizontal Beam Math Model AP Event

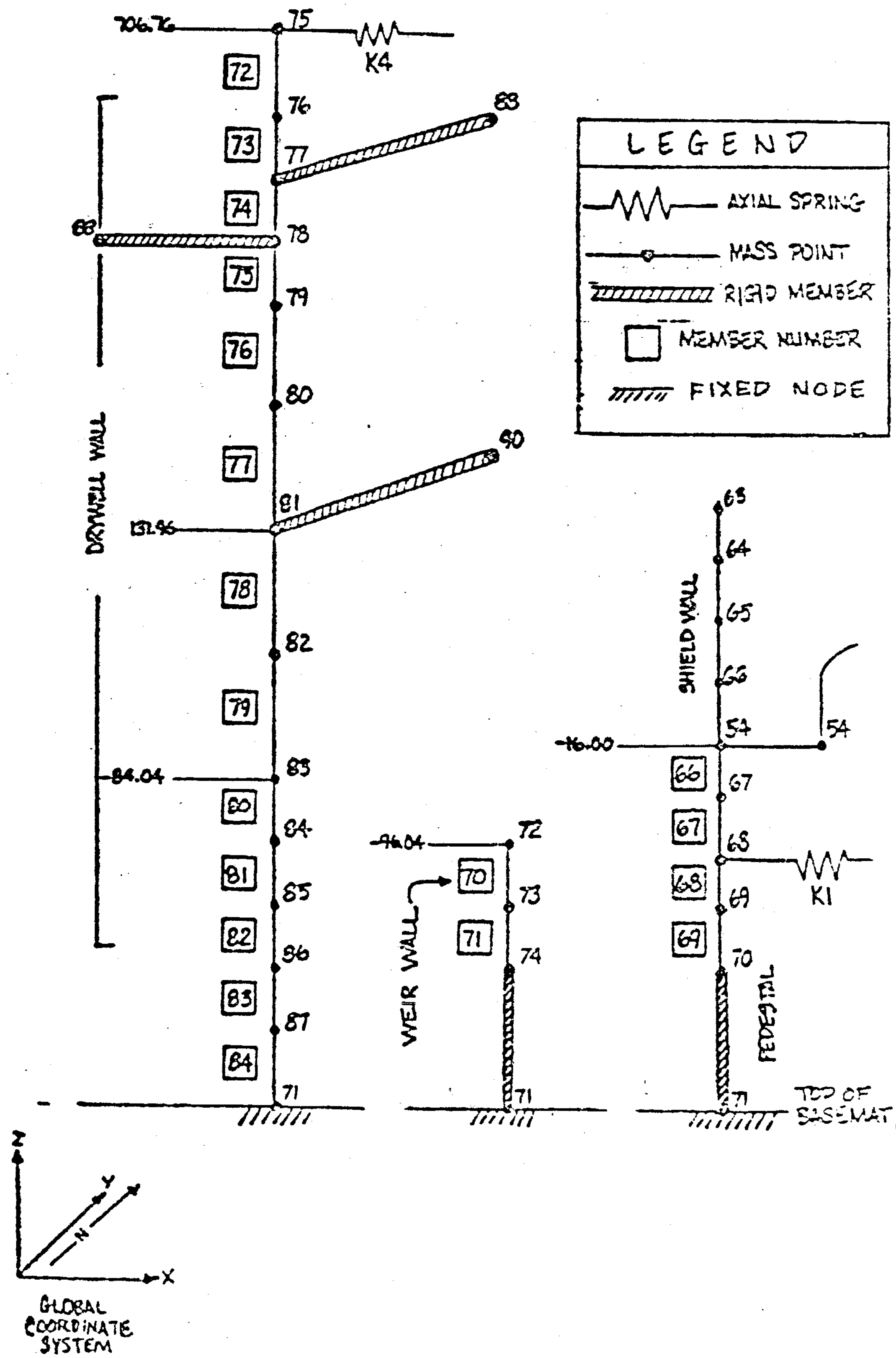


Figure 6.2-181 Drywell, Shield Wall and Pedestal Horizontal Beam Math Model AP Event

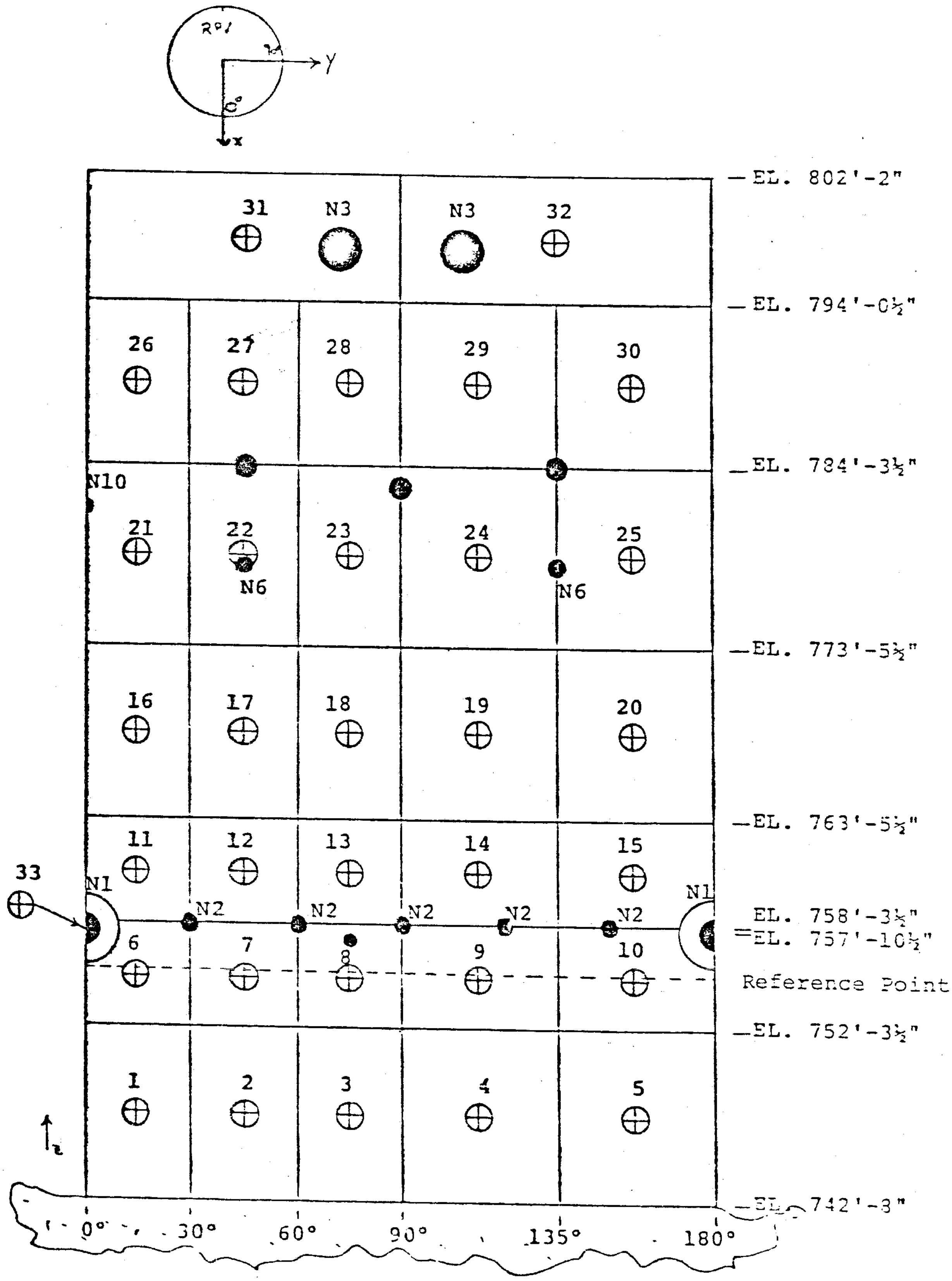


Figure 6.2-182 Annulus Nodalization For Recirculation Outlet Line Break

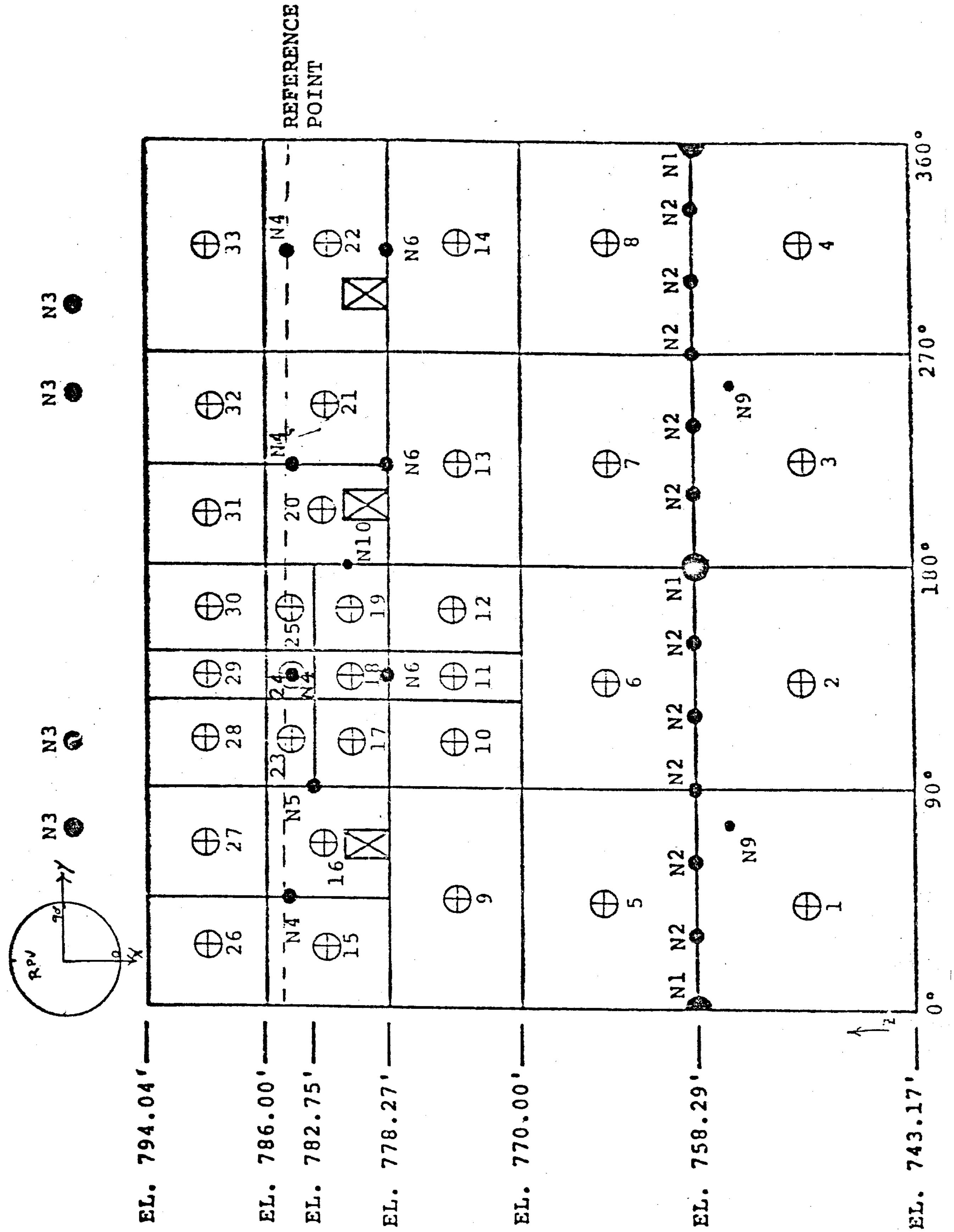


Figure 6.2-183 Annulus Nodalization for Feedwater Line Break

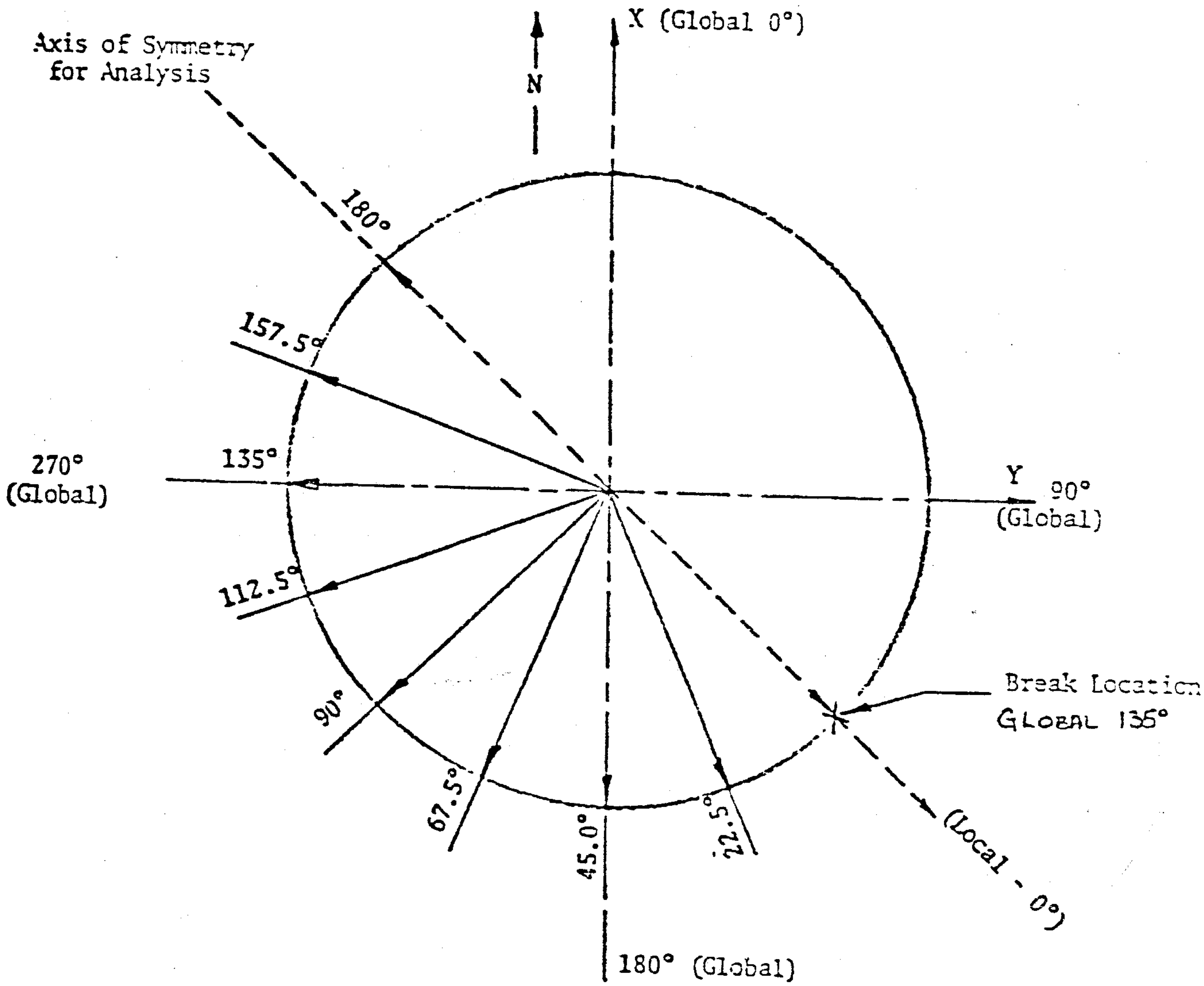


Figure 6.2-184 Break Location Feedwater Line Break

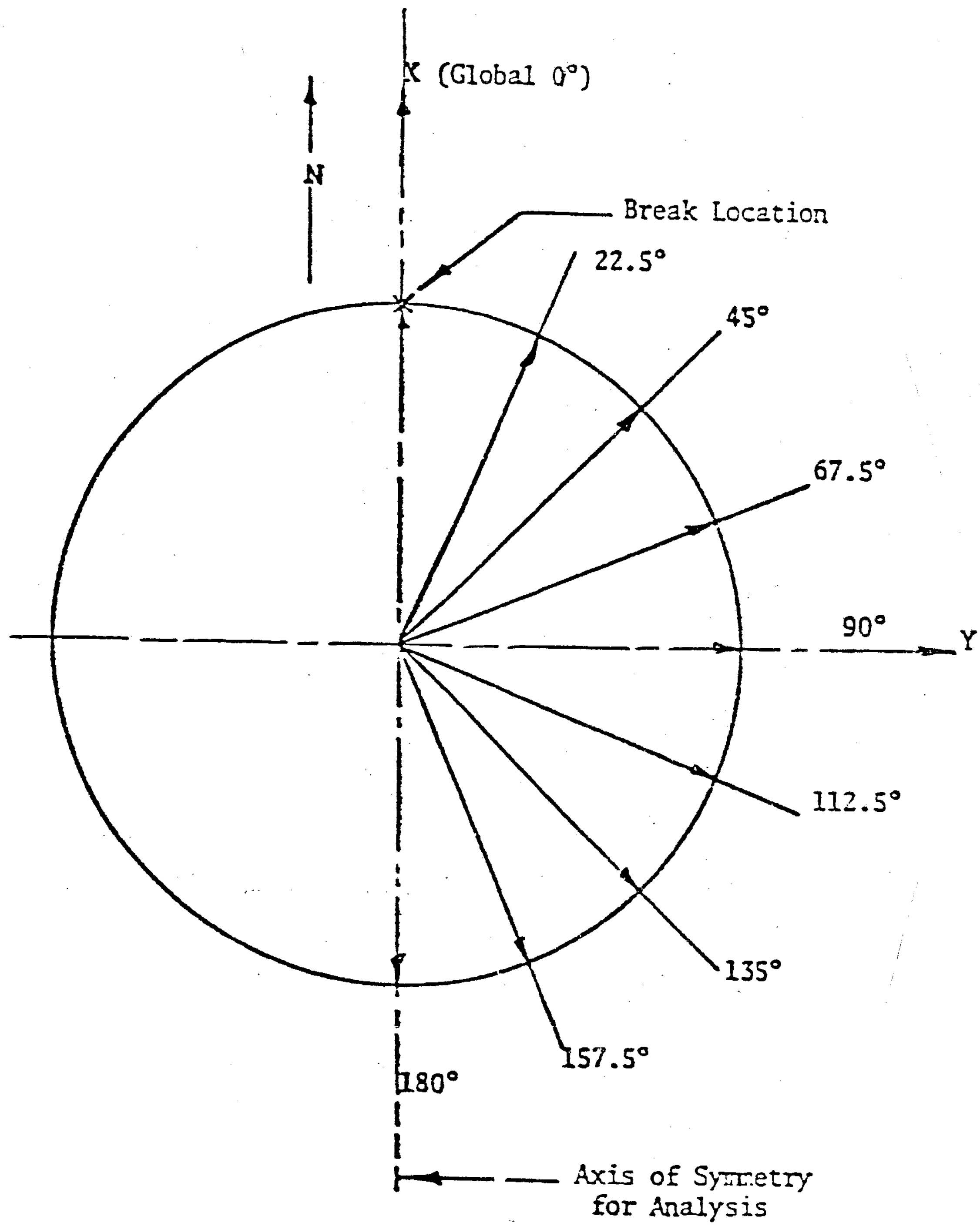


Figure 6.2-185 Break Location Recirculation Line Break

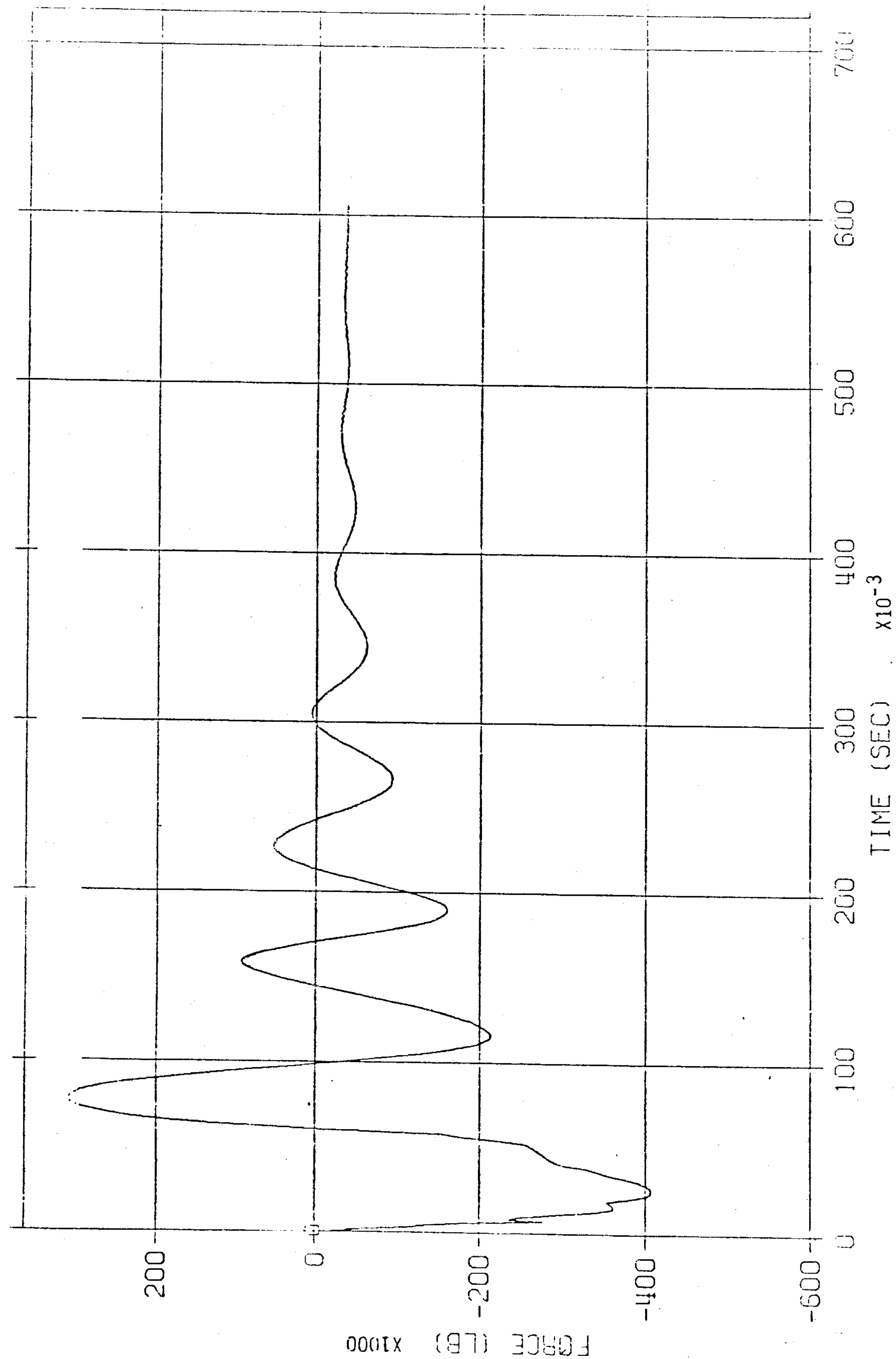


Figure 6.2-186 Clinton AP-FW Line Break Overall Load Resolution Time History on RPV at Elevation 484.5 in. X-Dir.

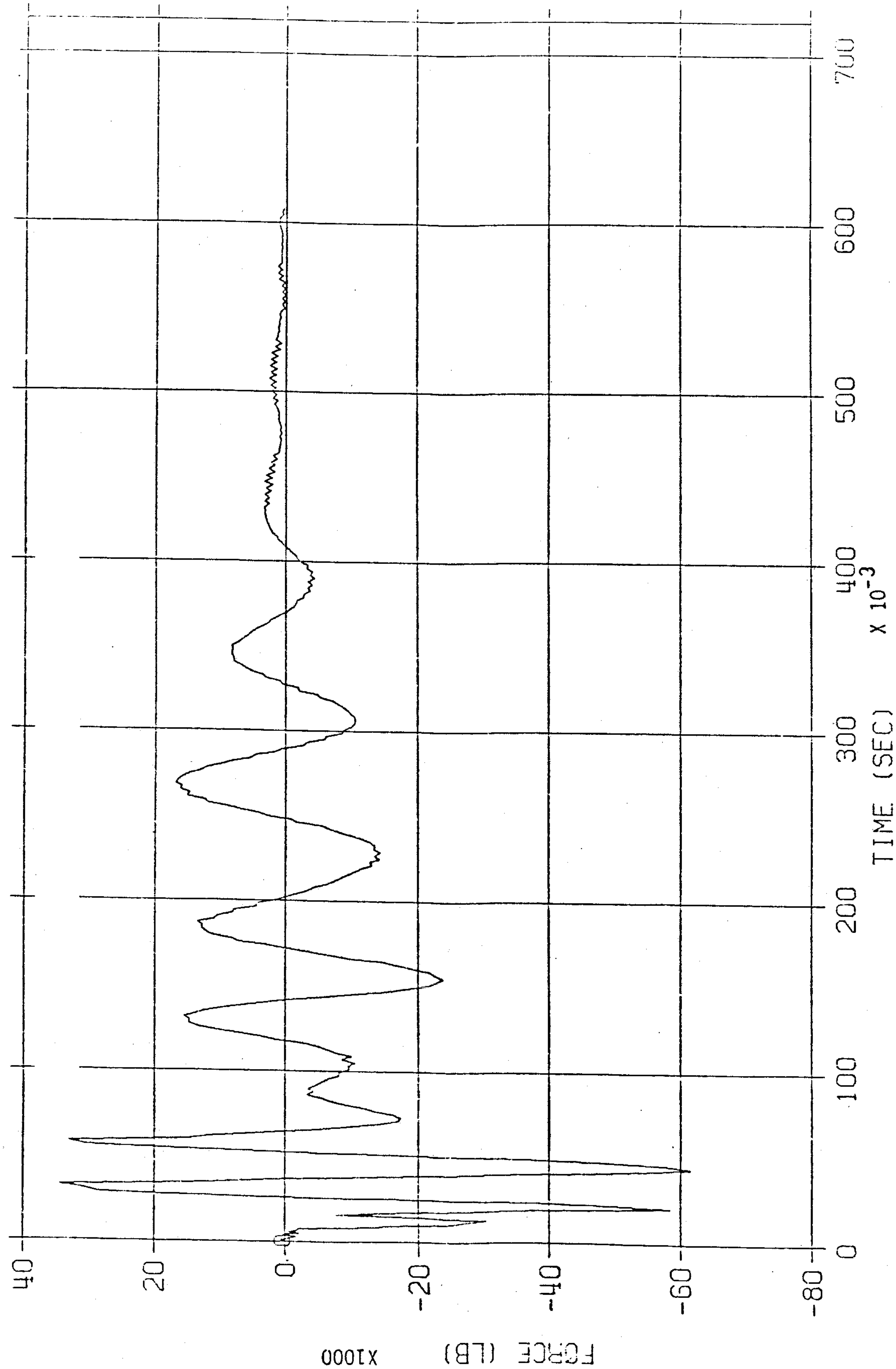


Figure 6.2-187 Clinton AP-FW Line Break Overall Load Resolution Time History on RPV at Elevation 484.5 in. Y-Dir.

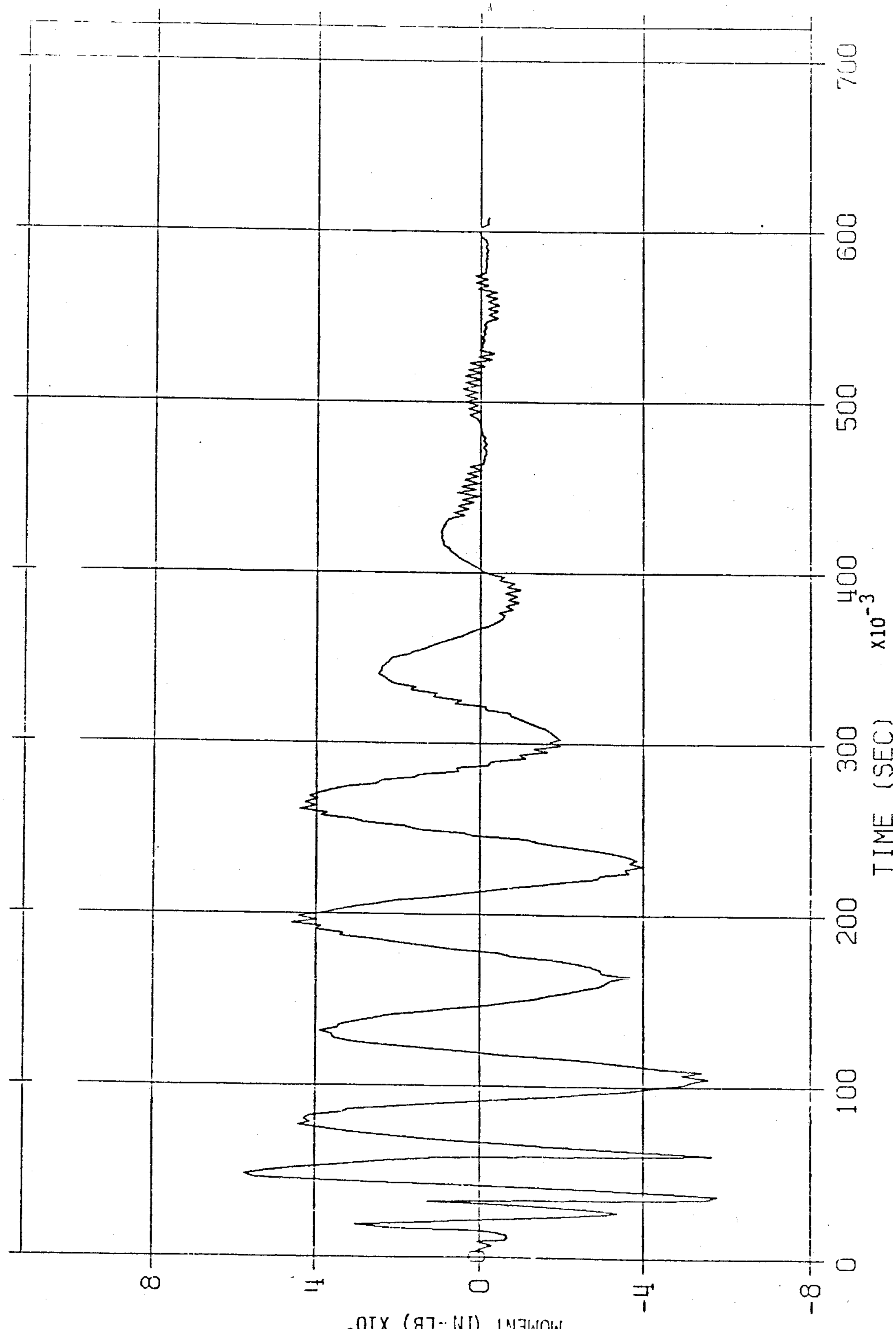


Figure 6.2-188 Clinton AP-FW Line Break Overall Load Resolution Time History on RPV at Elevation 484.5 in. About X-Axis

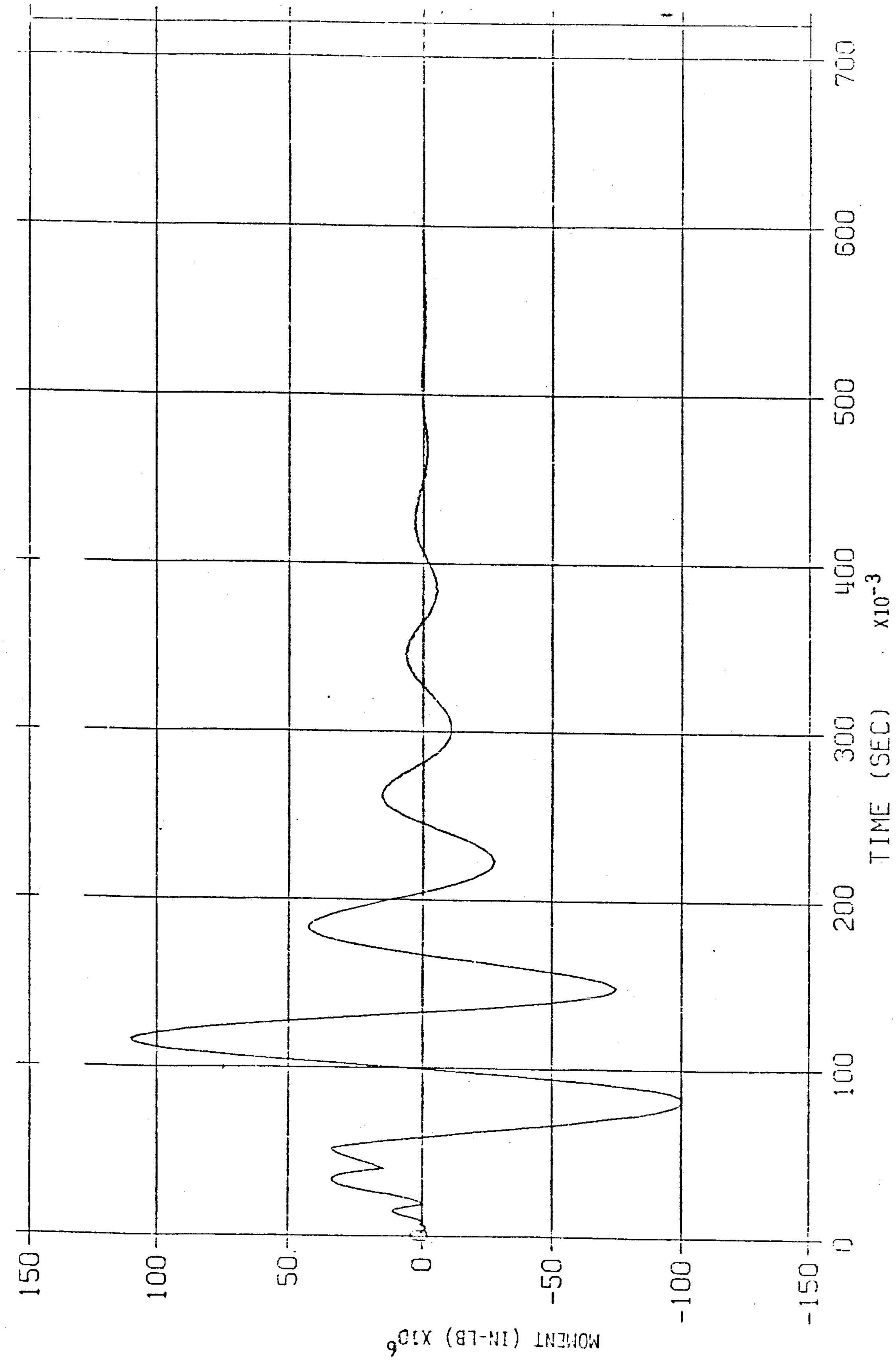


Figure 6.2-189 Clinton AP-FW Line Break Overall Load Resolution Time History on RPV at Elevation 484.5 in. In X-Dir.

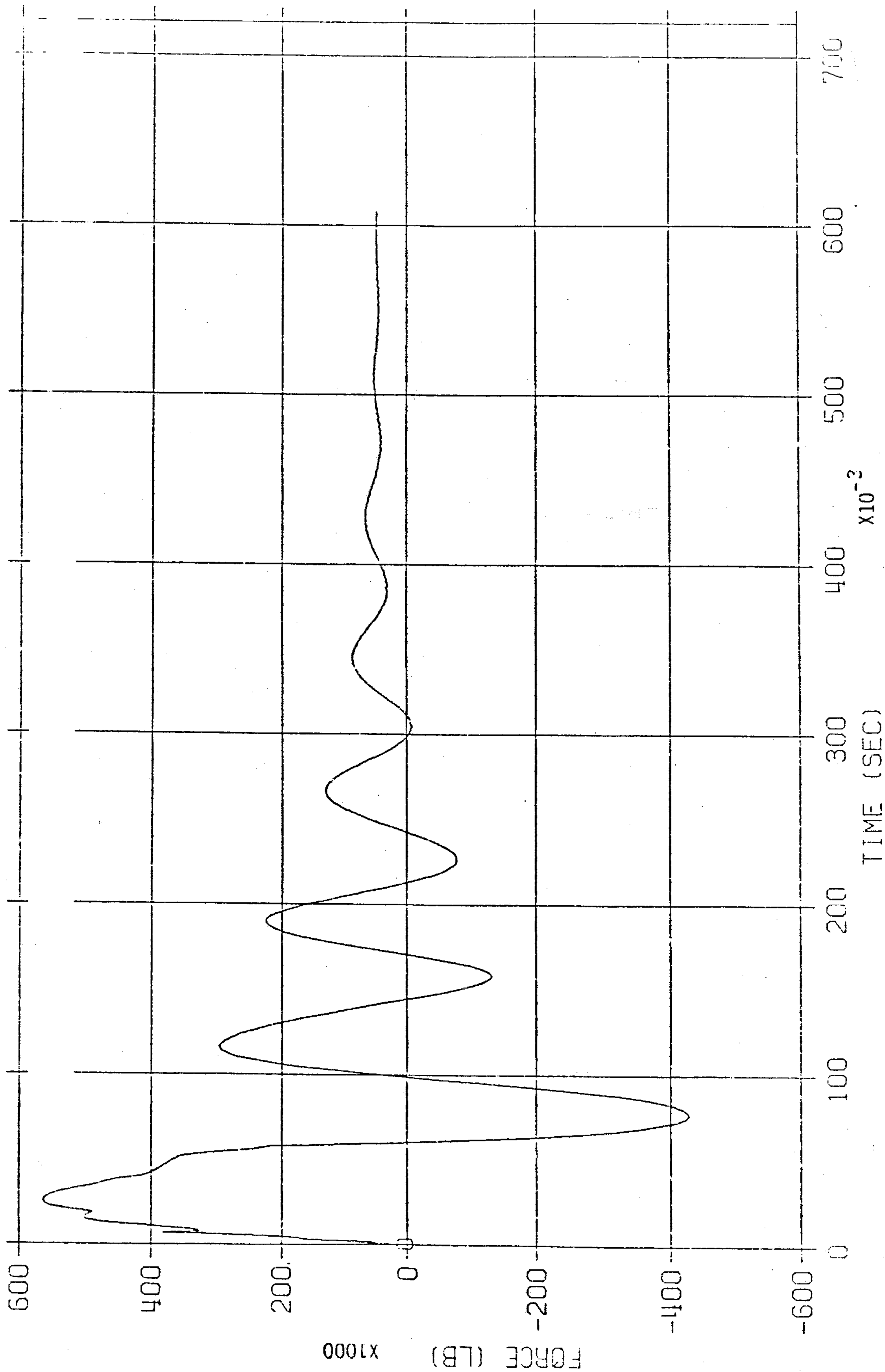


Figure 6.2-190 Clinton AP-FW Line Break Overall Load Resolution Time History on BSW at Elevation 484.5 in. In X-Dir.

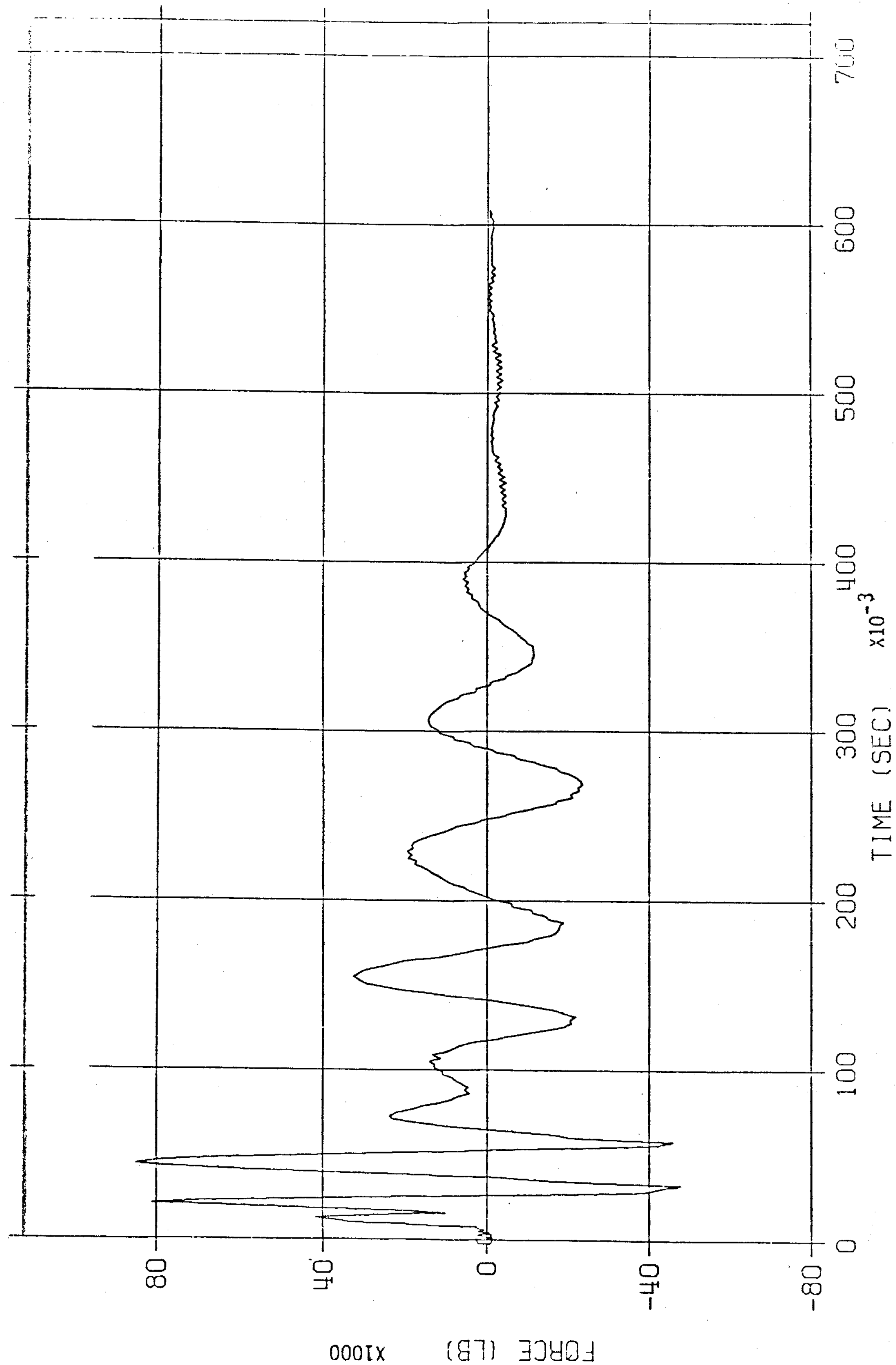


Figure 6.2-191 Clinton AP-FW Line Break Overall Load Resolution Time History on BSW at Elevation 484.5 in. In Y-Dir.

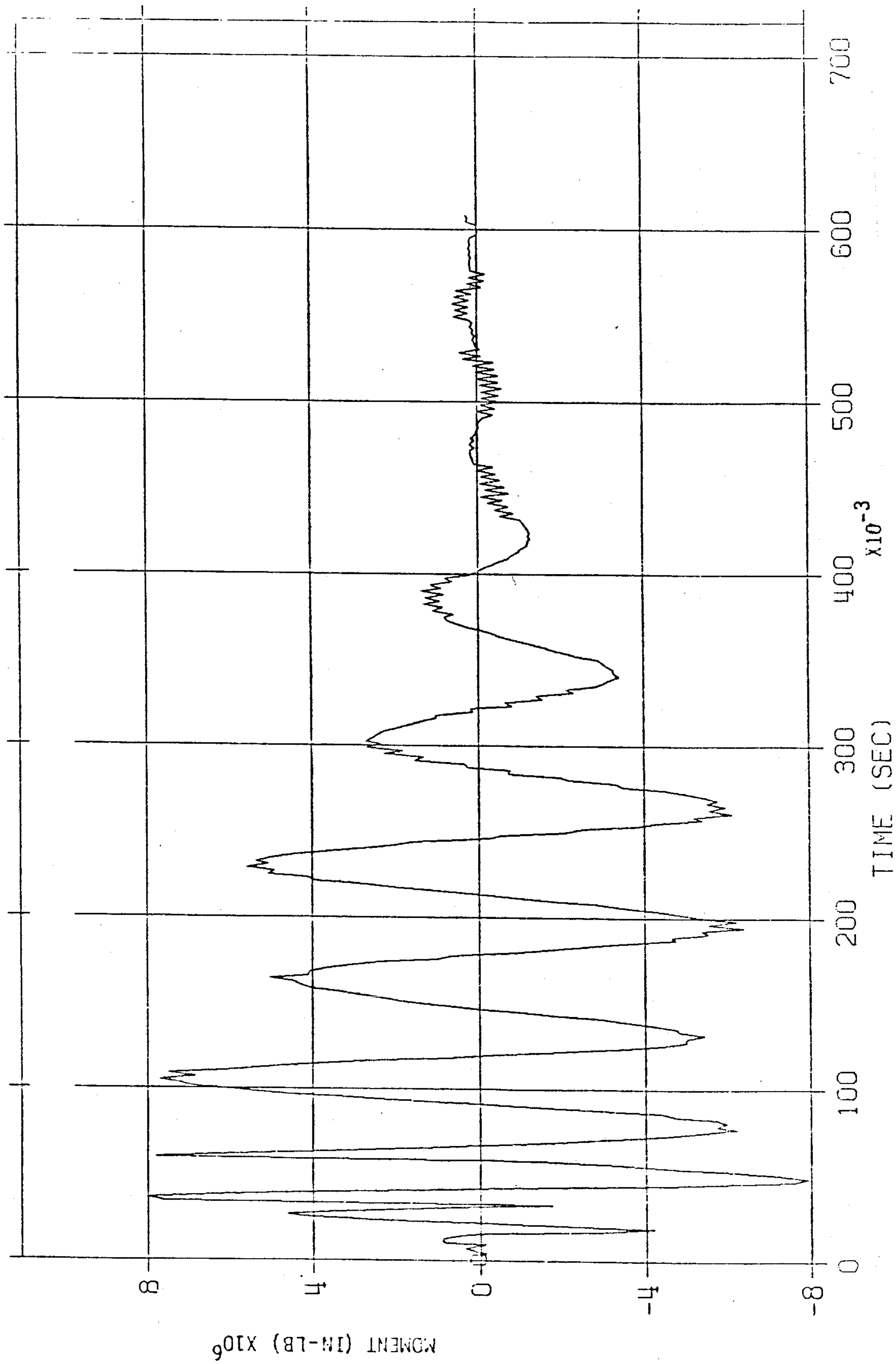


Figure 6.2-192 Clinton AP-FW Line Break Overall Load Resolution Time History on BSW at Elevation 484.5 in. About X-Axis

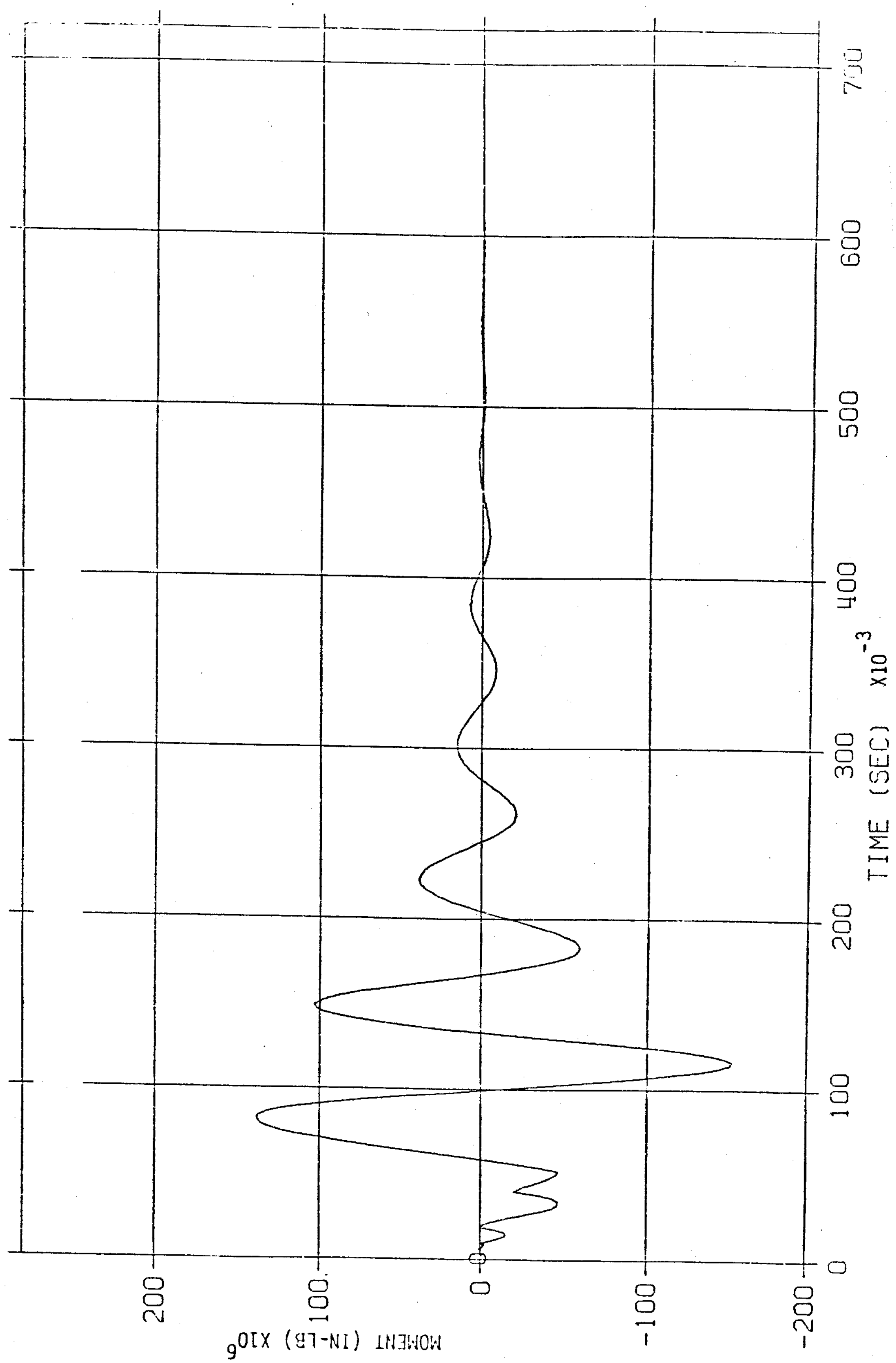


Figure 6.2-193 Clinton AP-FW Line Break Overall Load Resolution Time History on BSW at Elevation 484.5 in. About Y-Axis

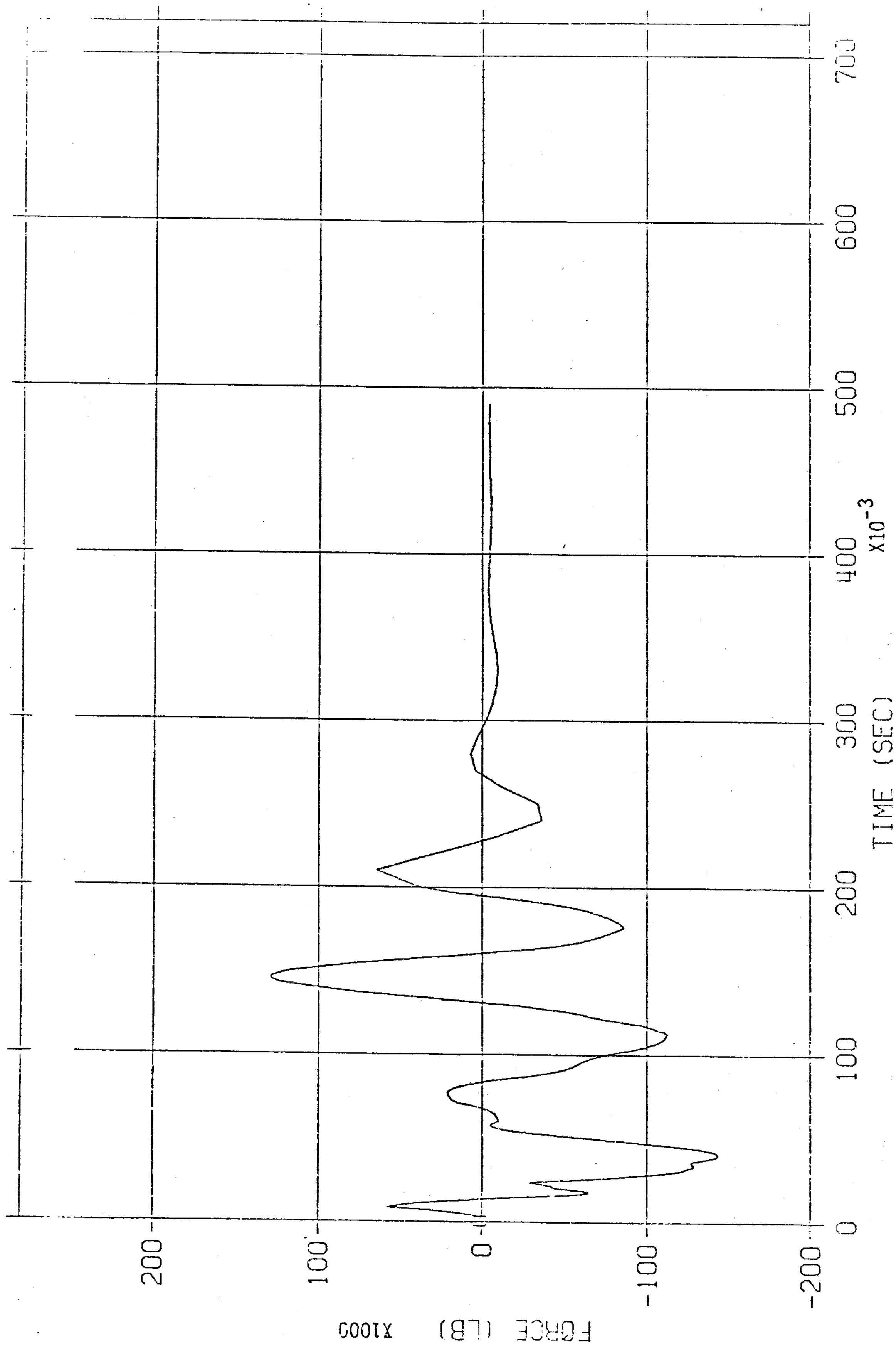


Figure 6.2-194 Clinton AP-RC Line Break Overall Load Resolution Time History on RPV at 135.5 in. X-Dir.

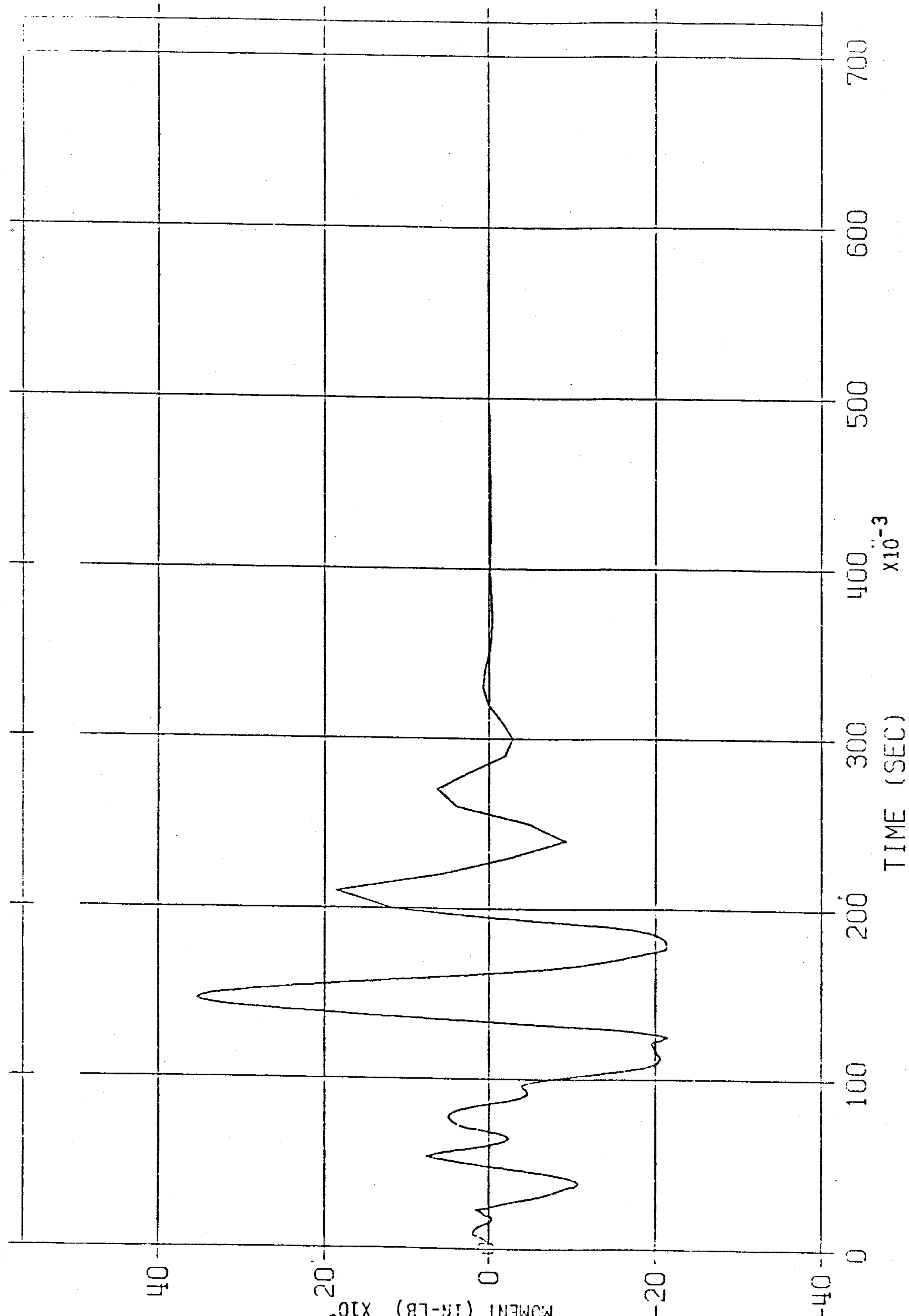


Figure 6.2-195 Clinton AP-RC Line Break Overall Load Resolution Time History on RPV at 135.5 in. About X-Aixis

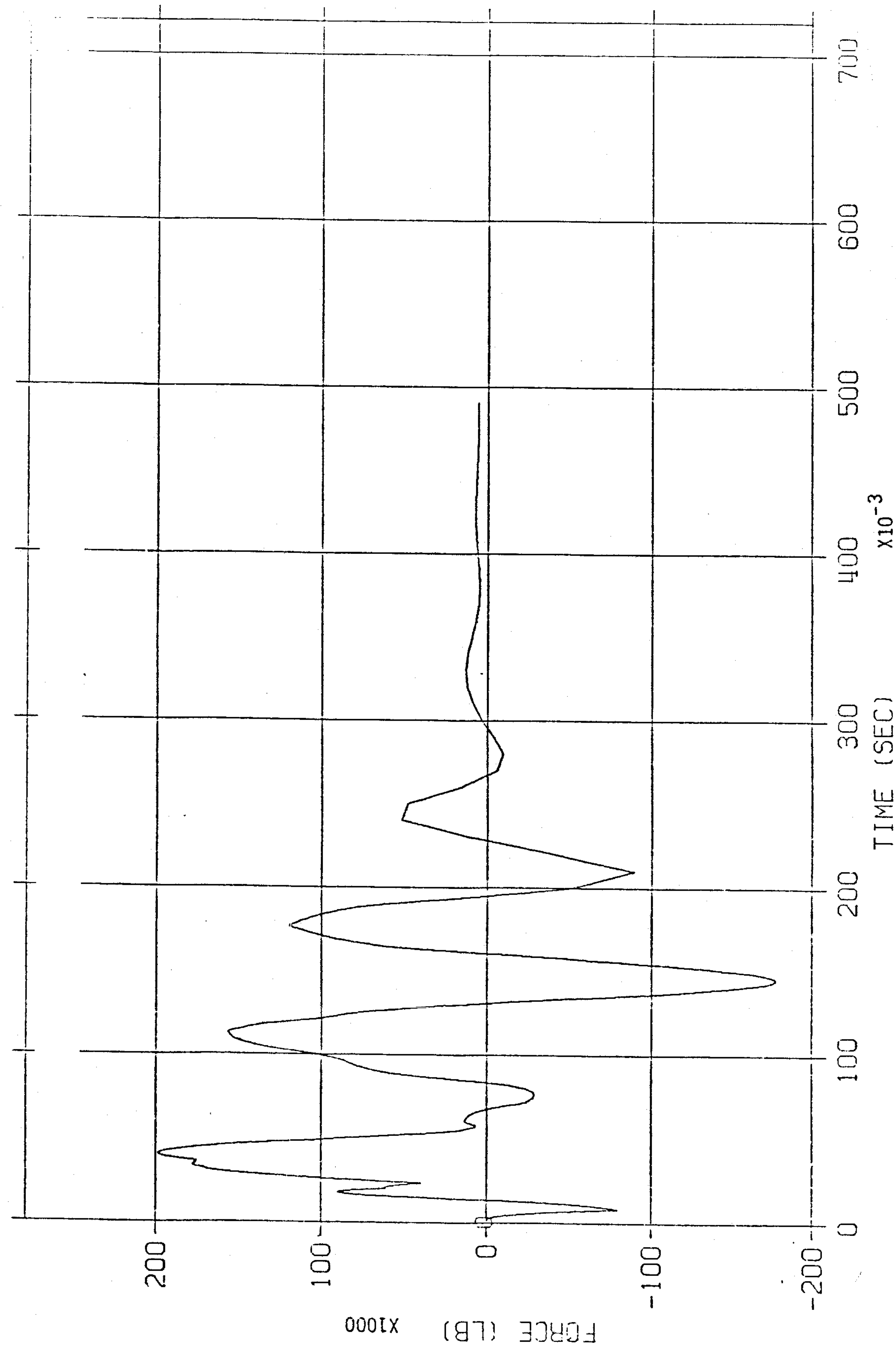


Figure 6.2-196 Clinton AP-RC Line Break Overall Load Resolution Time Hisory on BWS at 135.5 in. X-Dir.

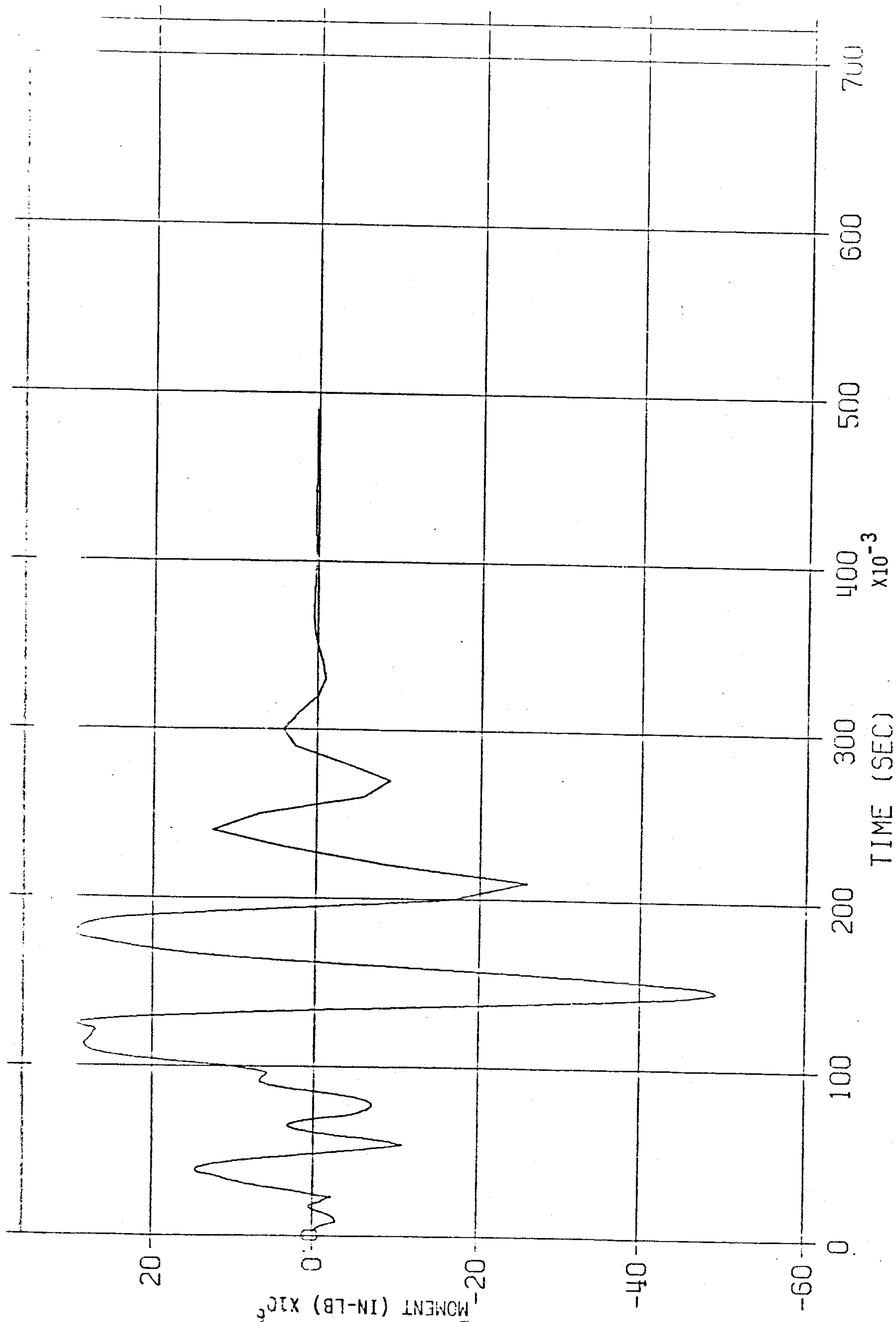


Figure 6.2-197 Clinton AP-RC Line Break Overall Load Resolution Time History on BSW at 135.5 in. About X-Axis

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FIGURE 6.2-198

SAMPLING POINT LOCATIONS
(SECTION G-G)

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FIGURE 6.2-199

SAMPLING POINT LOCATIONS
(SECTION D-D)

Figures 6.3-1 and 6.3-2
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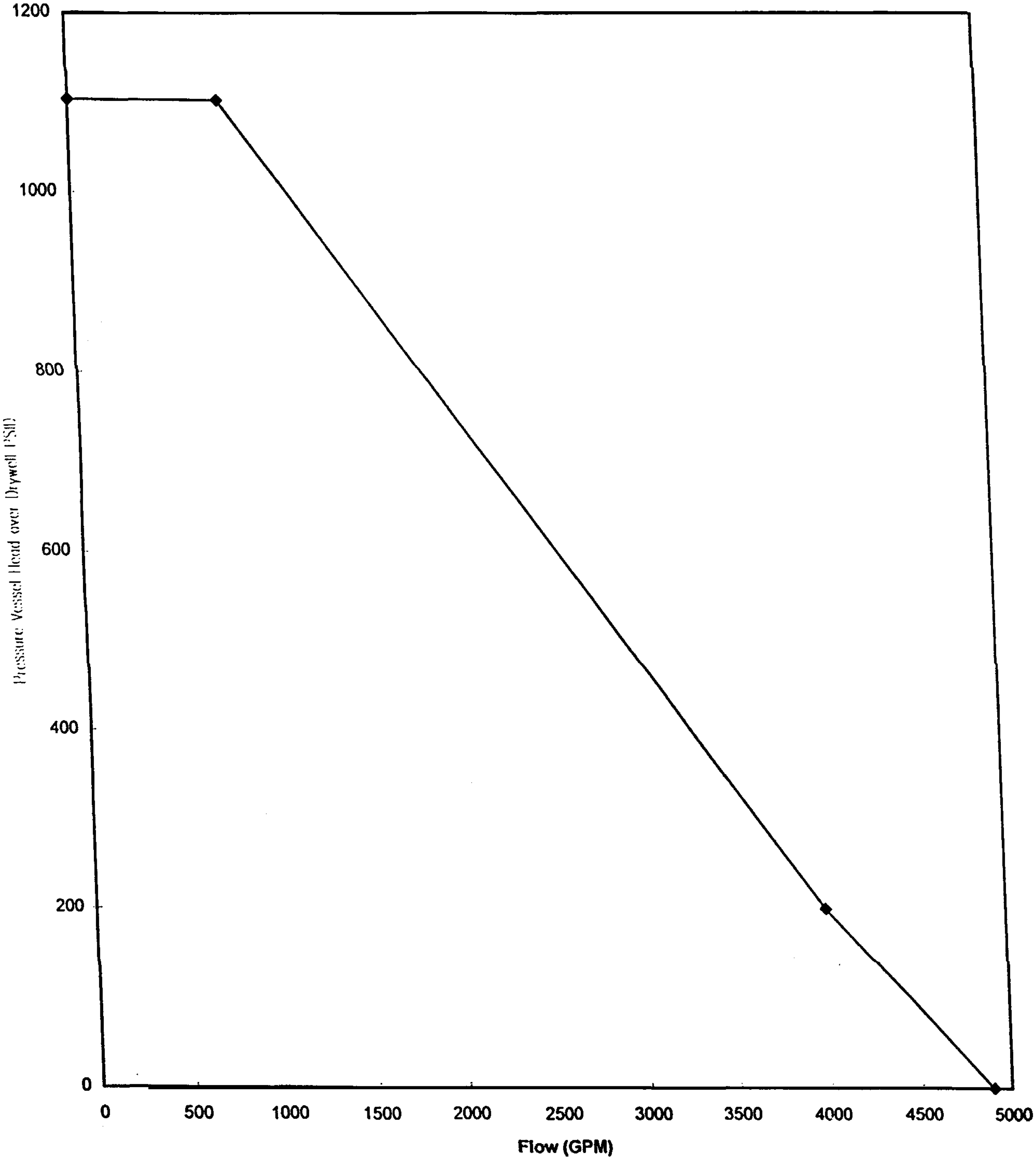

Figure 6.3-3 Head verses High Pressure Core Spray Flow Used in LOCA Analysis

Figures 6.3-4 and 6.3-5
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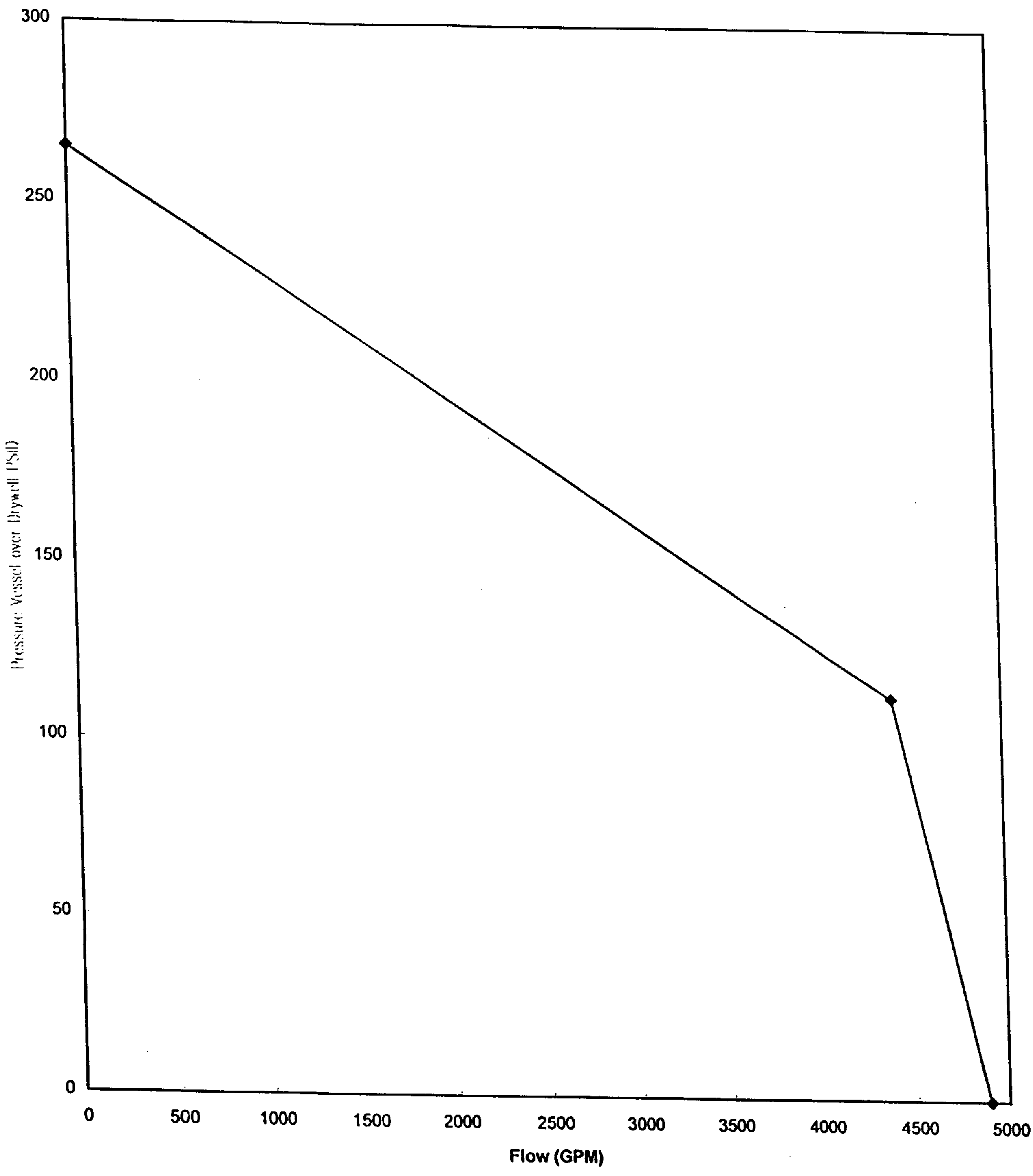

Figure 6.3-6 Head verses Low Pressure Core Spray flow Used in LOCA Analysis

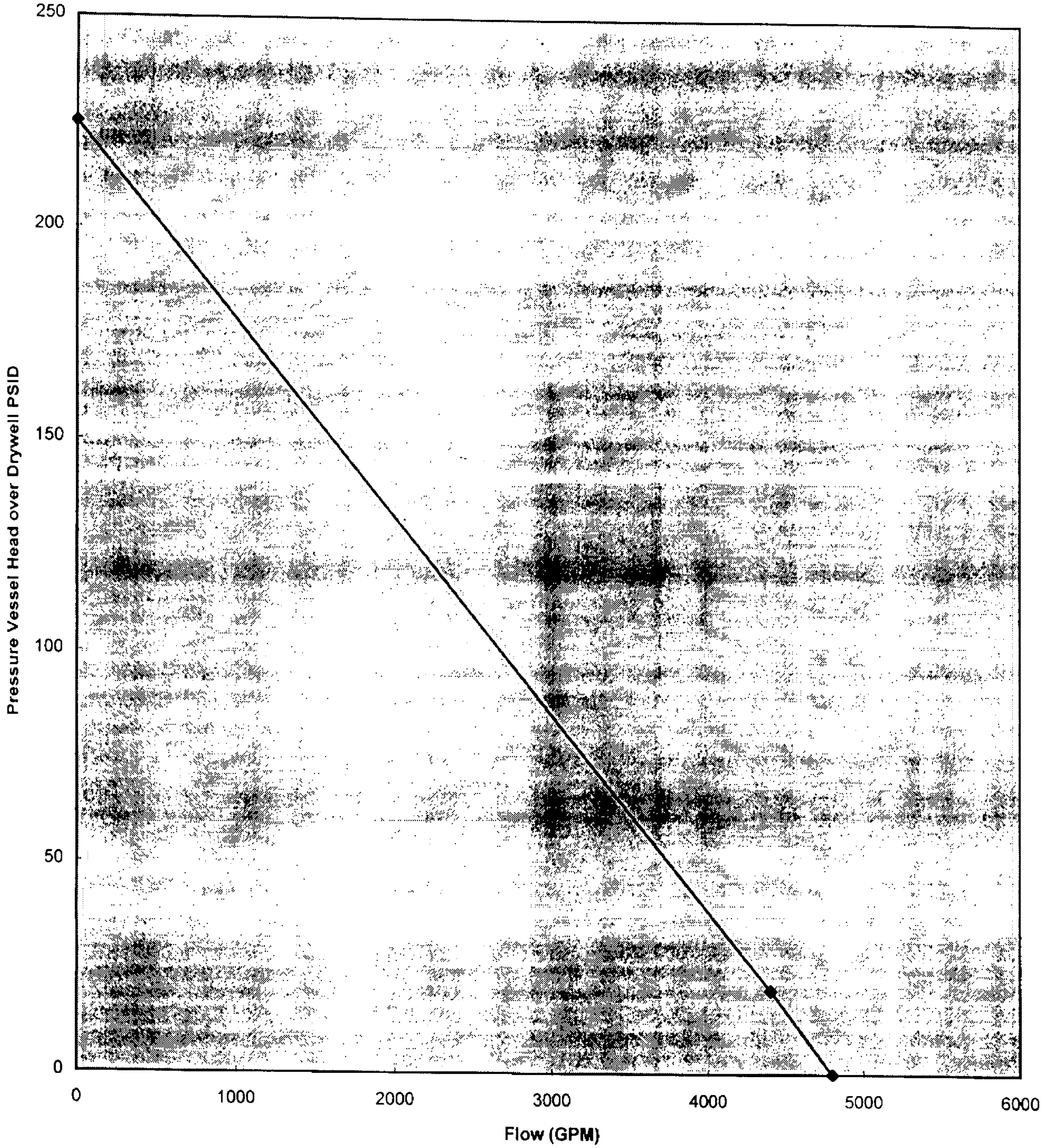

Figure 6.3-7 Head versus Low Pressure Coolant Injection Flow Used in LOCA Analysis for 1 Pump Only

Figure 6.3-8
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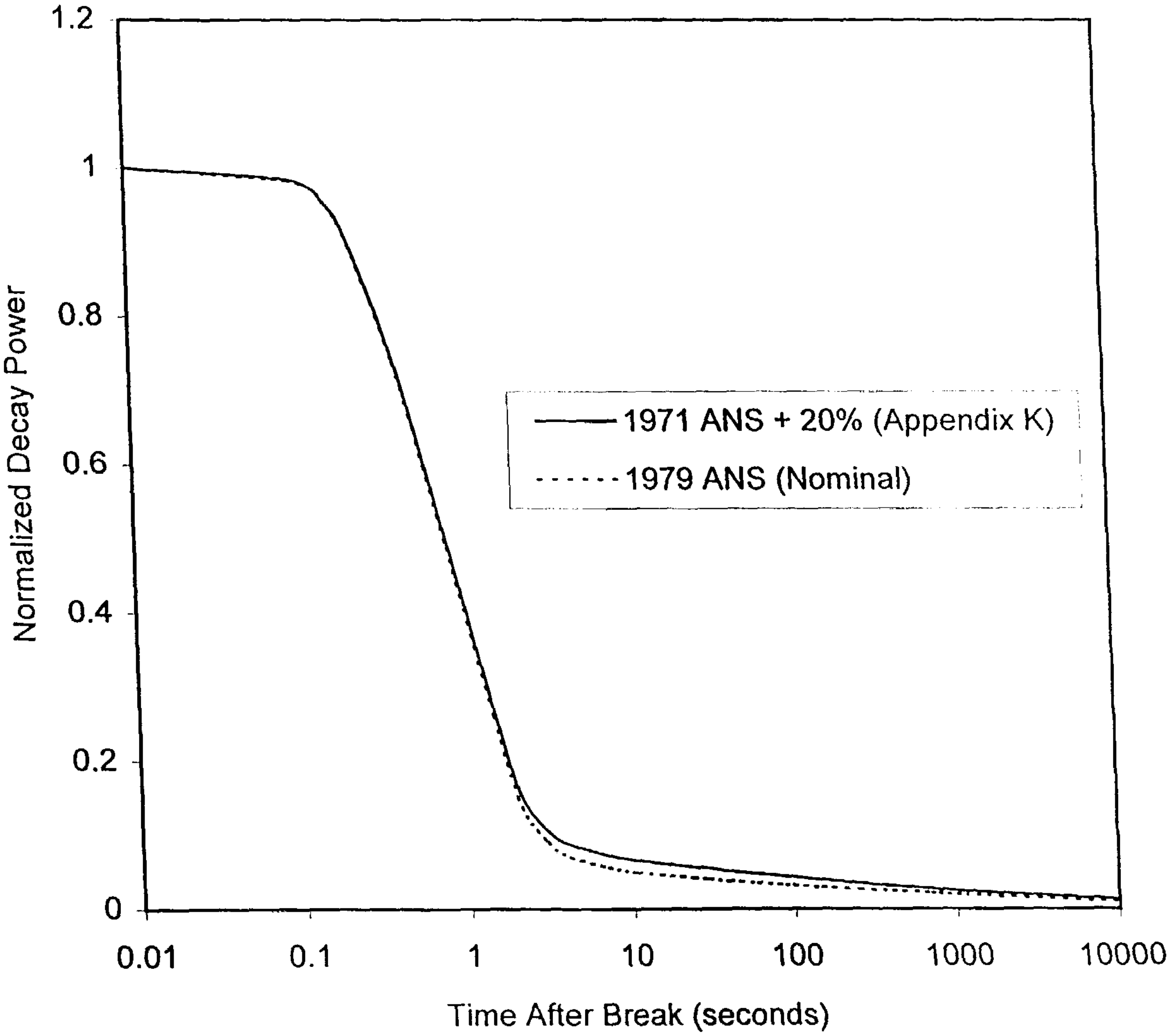

Figure 6.3-9 Normalized Decay Power

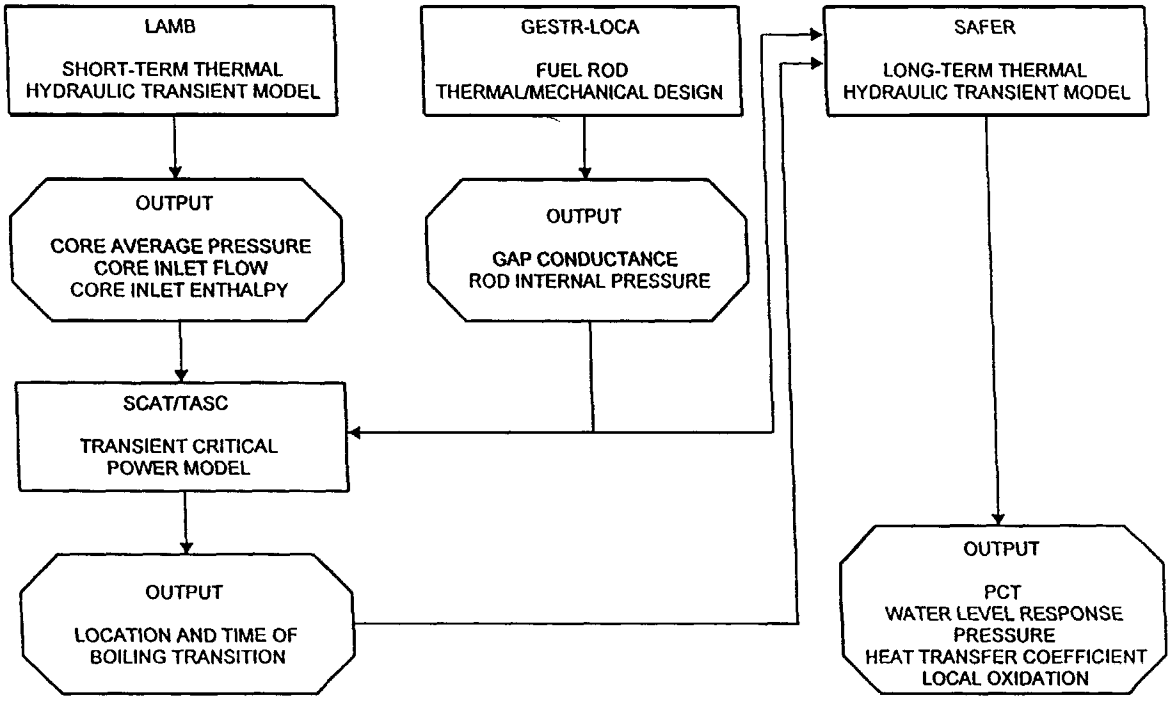

Figure 6.3-10 Flow Diagram of LOCA Analysis Using SAFER/GESTR

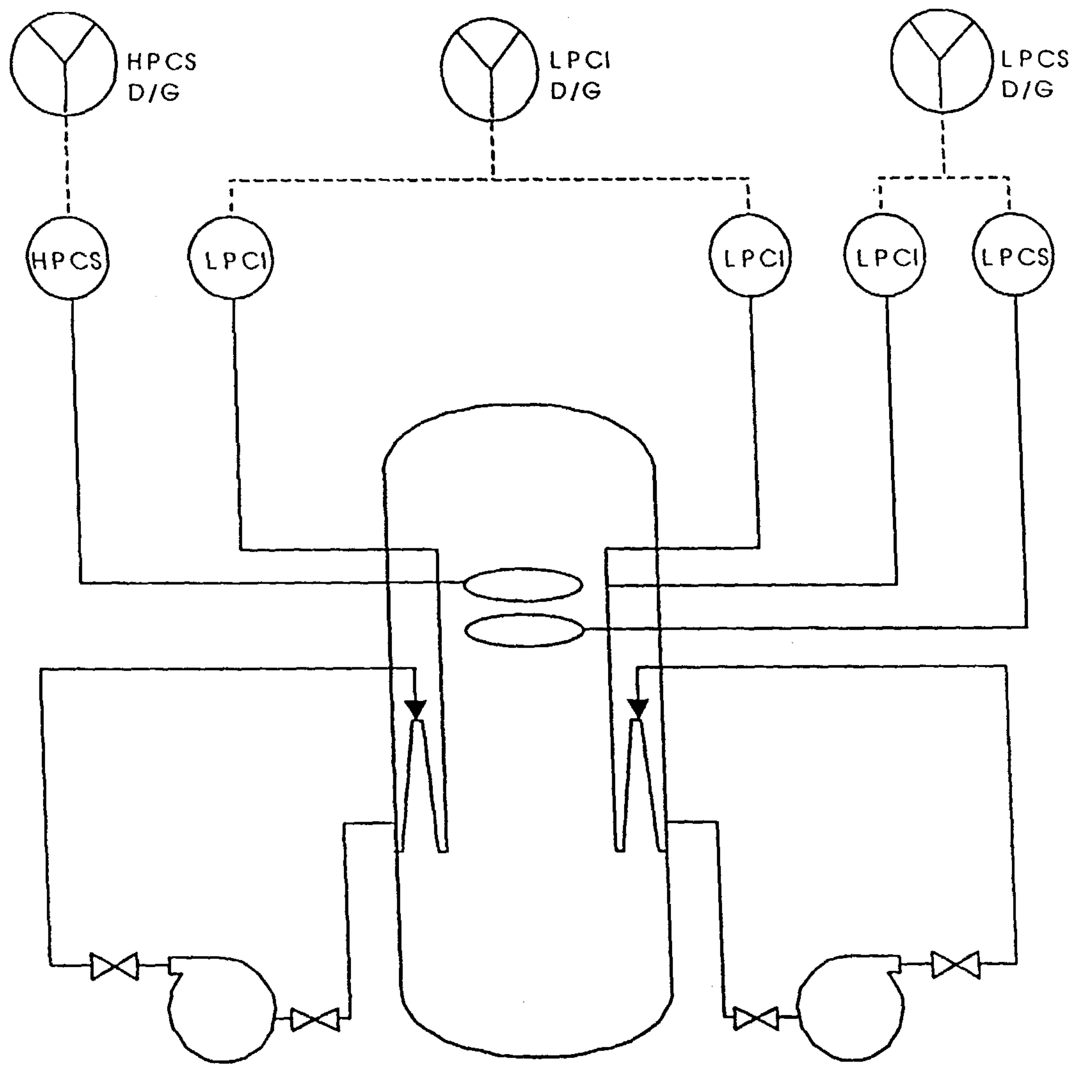

Figure 6.3-11 CPS ECCS Configuration

FIGURES 6.3-12 THROUGH 6.3-78

HAVE BEEN DELETED

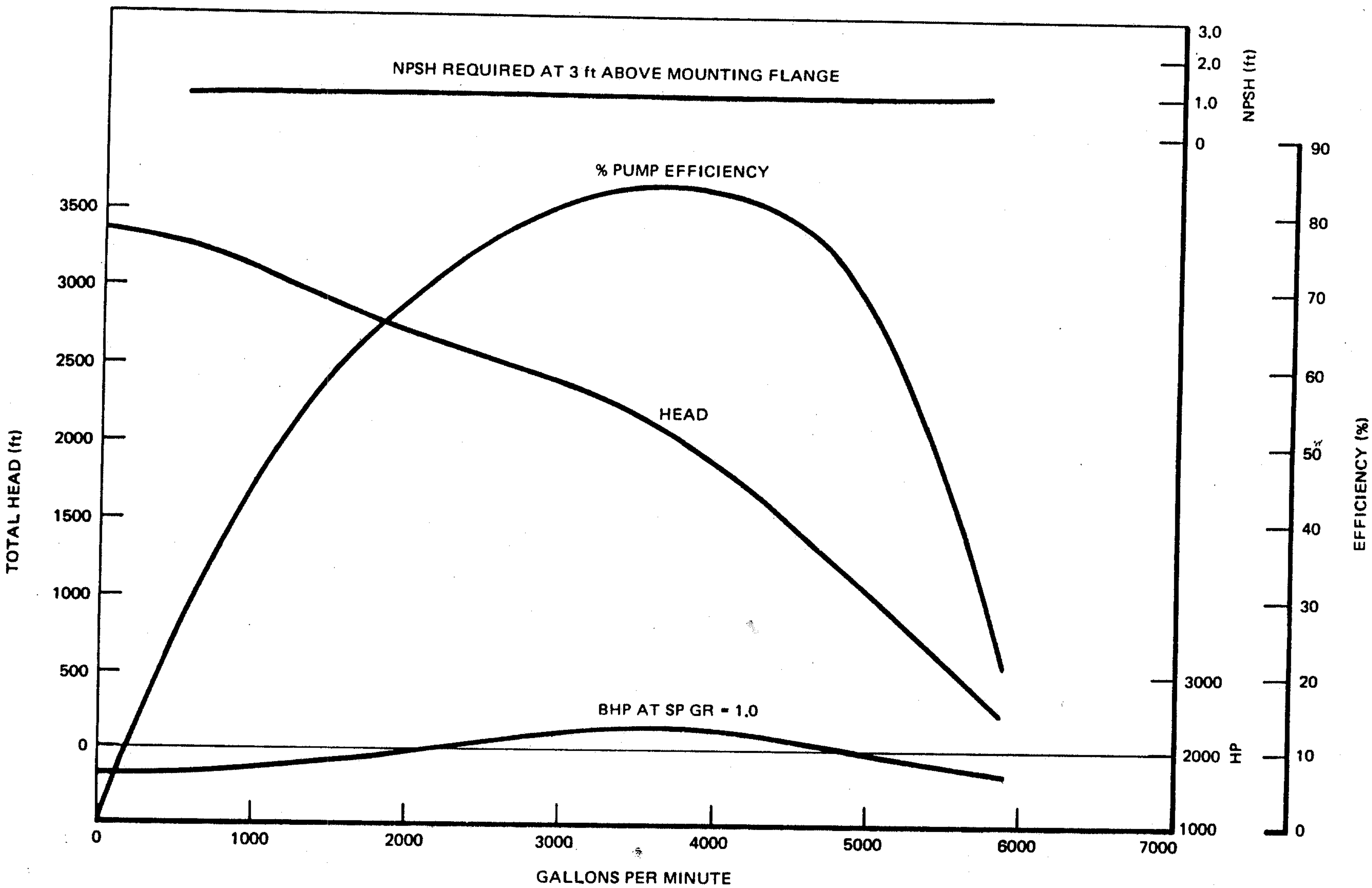

Figure 6.3-79. HPCS Pump Characteristic Curve

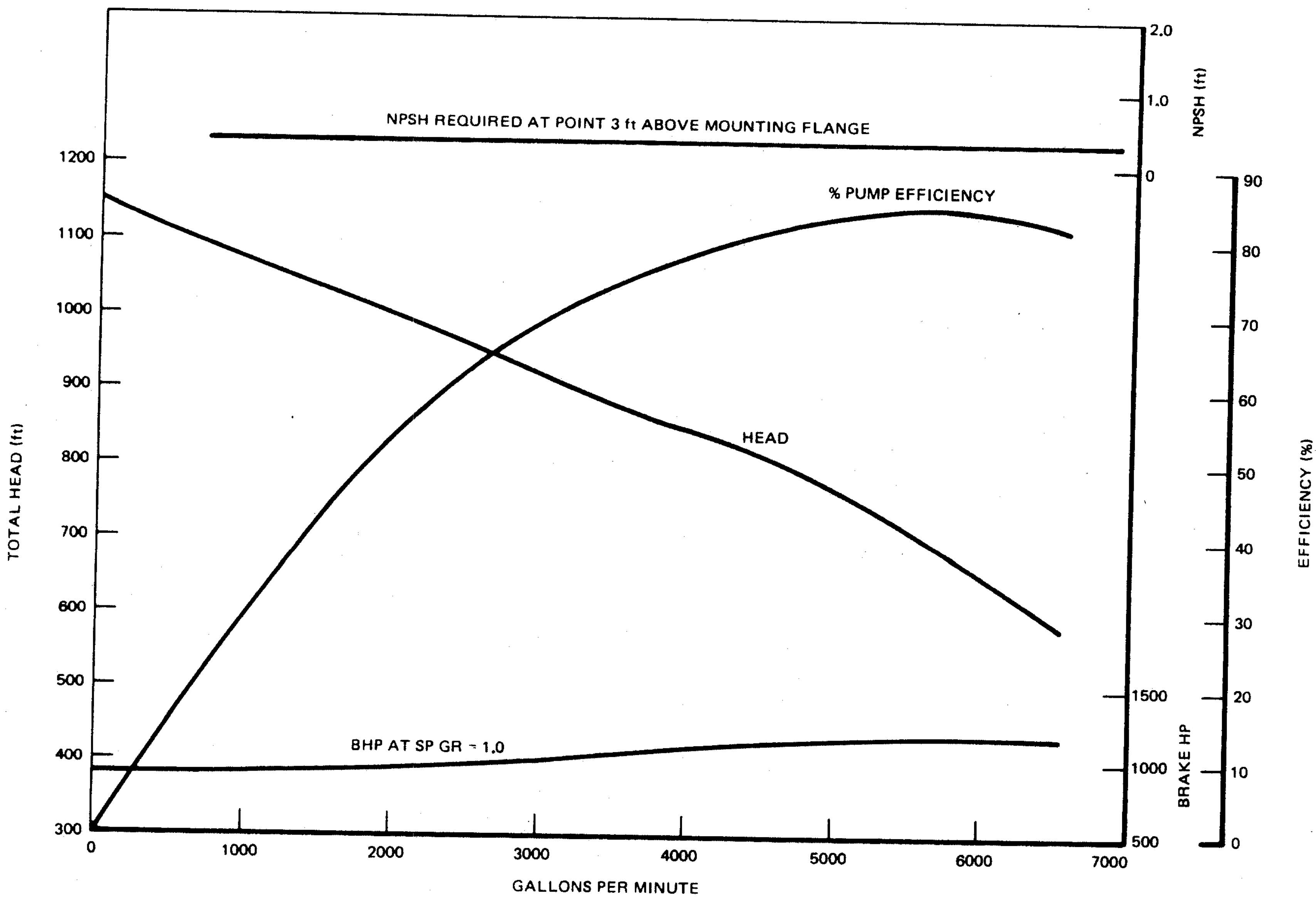

Figure 6.3-80 LPCS Pump Characteristic Curve

Figures 6.4-1 and 6.4-2
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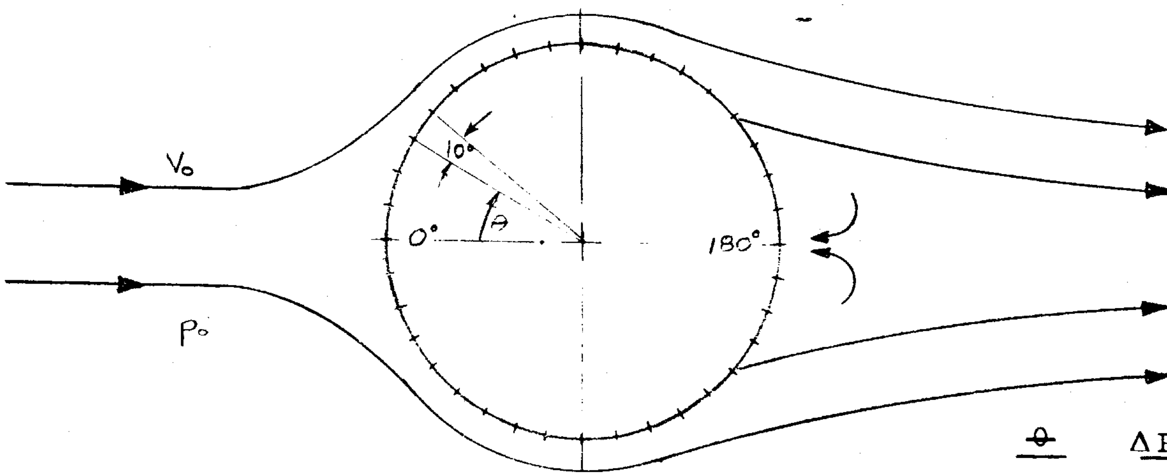
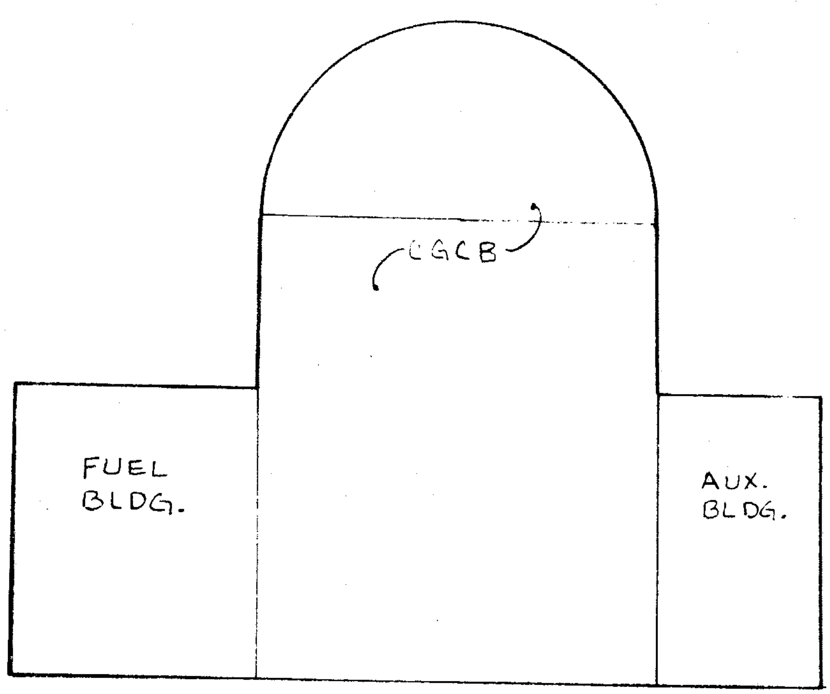
FIGURE 6.4-3

ISOMETRIC DRAWING OF THE CONTROL ROOM AND OTHER STRUCTURES



PLAN

θ	$\Delta P/(1/2\,\rho\,V_0^2)$
0	1.
10	.97
20	.77
30	.41
40	.1
50	-.3
60	-.62
70	-.9
80	-1.15
90	-1.2
100	-1.1
110	-1.0
120	-.6
130	-.15
140	.15
150	.15
160	.1
170	.1
180	.1



ELEVATION

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FIGURE 6.5-1

CONTAINMENT GAS CONTROL BOUNDARY
WIND PATTERNS AND COEFFICIENTS
AS A FUNCTION OF θ

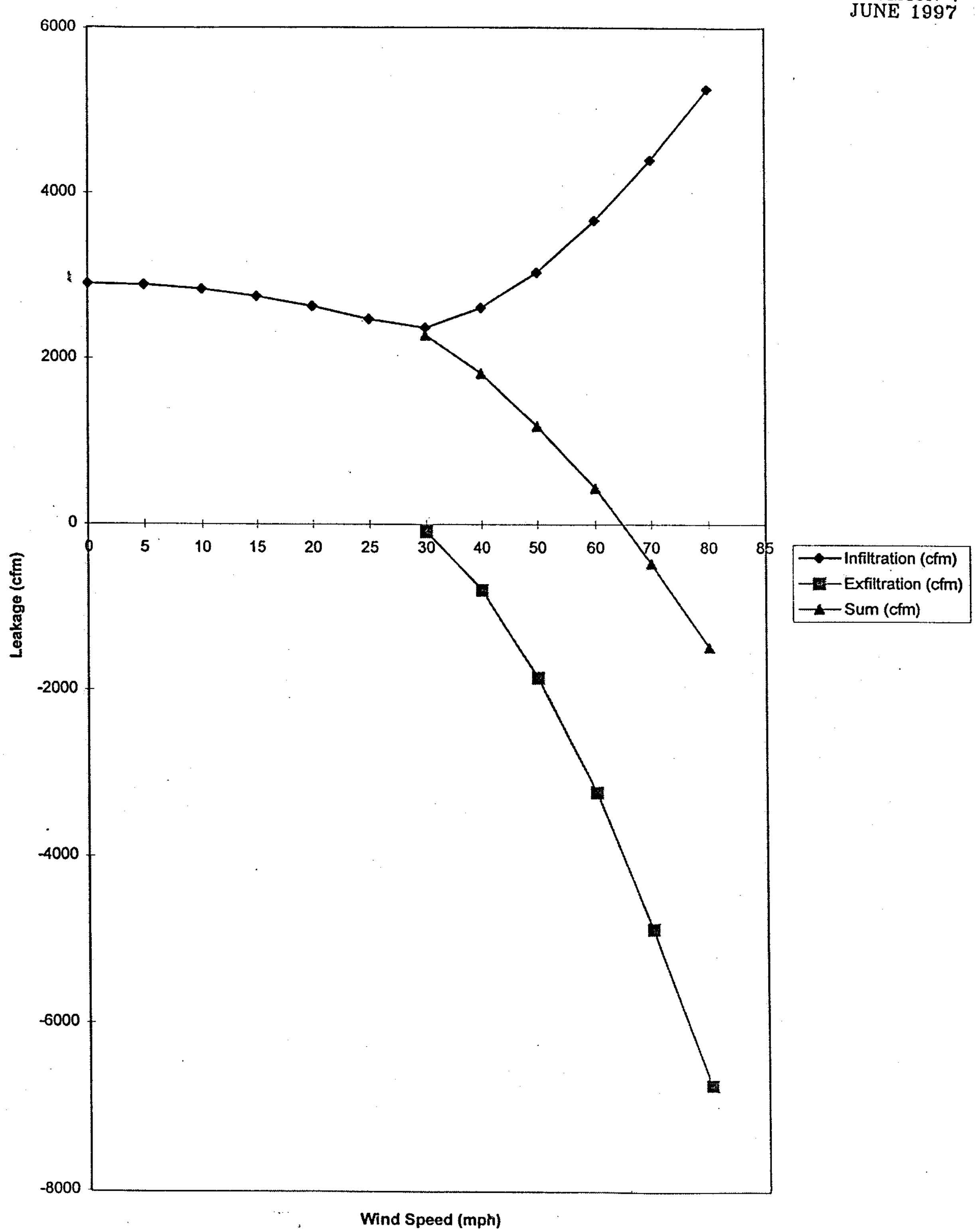

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FIGURE 6.5-2

SECONDARY CONTAINMENT LEAKAGE AS A FUNCTION OF WIND SPEED

Figures 6.7-1 through 6.7-3
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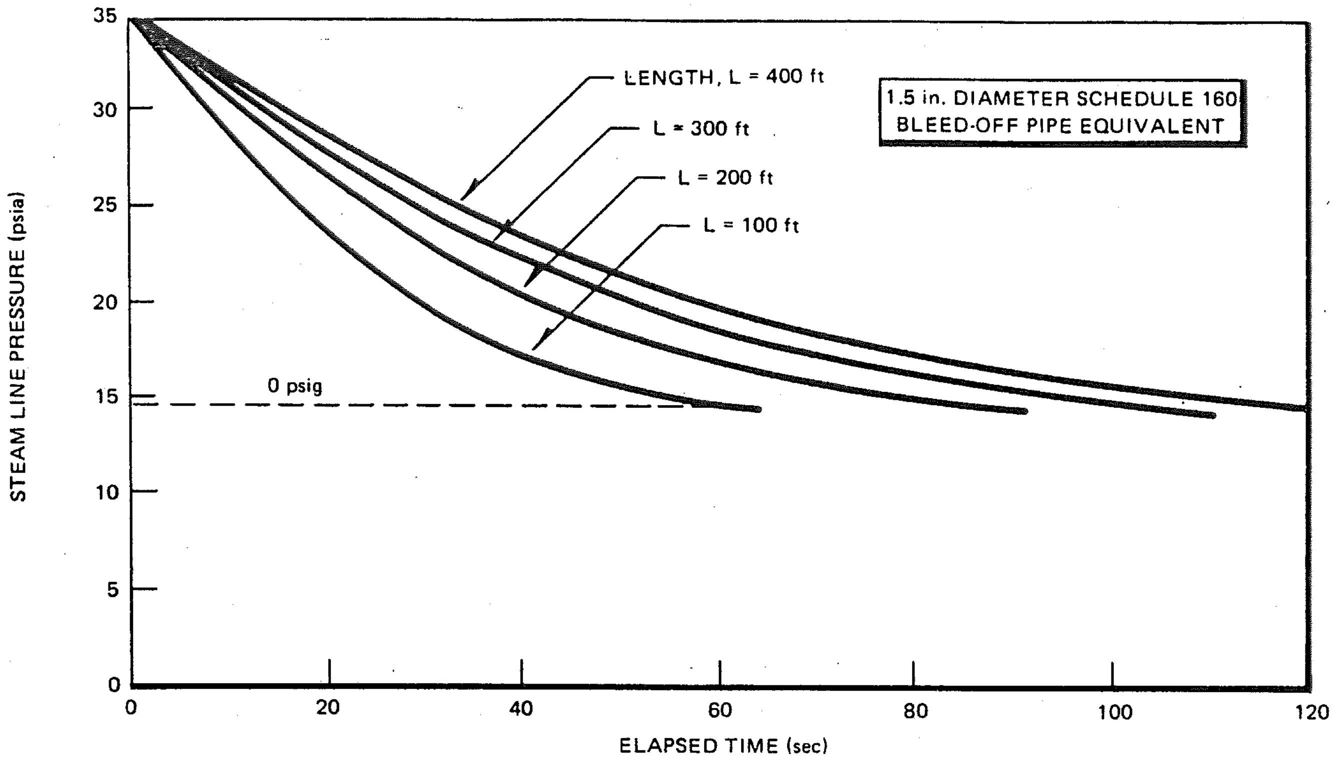

Figure 6.7-4. Effect of Bleed-off Line Length on Decompression of Main Steam Line Between Isolation Valves.