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	6.2-51h	0	August 1984		6.2-65	02-01	May 2002
	6.2-51i	0	August 1984		6.2-66	02-01	May 2002
	6.2-51j	0	August 1984		6.2-67	02-01	May 2002
	6.2-51k	0	August 1984		6.2-68	96-03	Sept. 1996
	6.2-51l	0	August 1984		6.2-69	02-01	May 2002
	6.2-51m	6	August 1990		6.2-70	02-01	May 2002
	6.2-51n	6	August 1990		6.2-71	0	August 1984
	6.2-51o	6	August 1990		6.2-72	0	August 1984
	6.2-51p	6	August 1990		6.2-73	96-03	Sept. 1996
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	6.2-79	96-03	Sept. 1996		6.3-30	RN12-034	July 2014
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	6.2-80	0	August 1984		6.3-32	00-01	December 2000
	6.2-81	0	August 1984		6.3-33	RN13-022	June 2015
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	6.2-83	RN07-028	November 2008		6.3-35	00-01	December 2000
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	6.3-3	00-01	December 2000		6.3-39	RN09-031	January 2010
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	6.3-6	RN99-105	January 2000		6.3-42	00-01	December 2000
	6.3-7	RN06-019	February 2007		6.3-43	00-01	December 2000
	6.3-8	RN99-013	February 2000		6.3-44	00-01	December 2000
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	6.3-11	00-01	December 2000		6.3-47	00-01	December 2000
	6.3-12	02-01	May 2002		6.3-48	00-01	December 2000
	6.3-13	02-01	May 2002		6.3-49	RN13-022	June 2015
	6.3-14	02-01	May 2002		6.3-50	00-01	December 2000
	6.3-15	RN02-039	June 2003		6.3-51	00-01	December 2000
	6.3-16	RN10-010	November 2011		6.3-52	RN10-010	November 2011
	6.3-17	RN06-029	September 2006		6.3-53	RN99-013	February 2000
	6.3-18	RN06-029	September 2006		6.3-54	RN06-029	September 2006
	6.3-19	RN06-029	September 2006		6.3-55	RN02-039	June 2003
	6.3-20	RN06-029	September 2006		6.3-56	RN10-010	November 2011
	6.3-21	RN13-022	June 2015		6.3-57	RN99-013	February 2000
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	6.3-24	RN13-022	June 2015		6.3-60	00-01	December 2000
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	6.3-3	0	August 1984
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	6.4-6	RN98-064	September 1998
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	6.5-15	RN02-034	May 2003

6.0 ENGINEERED SAFETY FEATURES

The central safety objective in reactor plant design and operation is control of reactor fission products. The methods used to assure this objective are:

1. Design of the reactor core in conjunction with the reactor control and protection systems to preclude release of fission products from the fuel (Chapters 4 and 7).

2. Retention of fission products in the Reactor Coolant System (Chapters 5 and 6).

3. Retention of fission products by the containment for operational and accidental releases beyond the reactor coolant boundary (Sections 3.8 and 6.2).

4. Limiting or optimizing fission product dispersal to minimize population exposure for an accidental release beyond the containment (Chapters 2 and 11).

The engineered safety features are the provisions in the plant which embody methods 2 and 3 above to prevent the occurrence, or to ameliorate the effects, of serious accidents.

6.1 GENERAL

The engineered safety features are:

1. Containment

   The reactor building provides a virtually leaktight barrier to the escape of fission products. Detailed information on the containment is provided in Section 6.2.1.

2. Reactor Building Heat Removal Systems

   These systems serve to reduce reactor building pressure and temperature. These systems are discussed in detail in Section 6.2.2.

3. Reactor Building Air Purification and Cleanup Systems

   The function of these systems is to provide air purification and cleanup services to the Reactor Building. These systems are discussed in Section 6.2.3. 02-01

4. Containment Isolation System

   This system provides containment isolation capability for the various system lines penetrating the containment. Detailed discussions of this system are presented in Section 6.2.4.

5. Combustible Gas Control System

   This system acts to control the concentration of hydrogen, oxygen, and other combustible gas which may be released to the reactor building to assure containment integrity. This system is discussed in Section 6.2.5.

6. Containment Leakage Testing.

   Containment leakage testing provisions are discussed in Section 6.2.6.

7. Emergency Core Cooling System

   This system delivers borated water to the reactor core to provide core cooling following postulated accidents. The boron, together with the control rods provides sufficient negative reactivity for safe shutdown following design basis accidents. The Emergency Core Cooling System is discussed in detail in Section 6.3.

8. Habitability Systems

   These systems provide the control room with adequate shielding, air purification, and climatic control. These systems are discussed in Section 6.4.

9. Fission Product Removal and Control Systems

   The fission product removal systems include the high efficiency particulate air filters in the Reactor Building cooling system and the control room emergency filter plenums. This equipment is discussed in Section 6.5.

6.2 CONTAINMENT SYSTEMS

6.2.1 CONTAINMENT FUNCTIONAL DESIGN

6.2.1.1 Design Bases

6.2.1.1.1 Reactor Building

The safety design basis for the Reactor Building is the requirement that the releases of radioactive material subsequent to a reactor accident do not result in doses which exceed the guideline values specified by 10 CFR 50.67. To satisfy this requirement, a design containment leak rate is specified in conjunction with performance requirements for the other engineered safety features. The containment must withstand the temperatures and pressures resulting from a spectrum of postulated loss of coolant (LOCA) and secondary system pipe break accidents without leakage in excess of the design leak rate of 0.2% per day. The radiological consequences of the most severe hypothetical LOCA are presented in Section 15.4. RN 12-034 02-01

The Reactor Building design pressure of 57 psig is above the peak calculated pressure of 53.0 psig resulting from the postulated double ended guillotine rupture (DER) of a main steam line. The bounding case, the 1.40 ft^2 DER at 25% power, is conservatively analyzed assuming an electrical channel A (CH-A) failure resulting in loss of one diesel generator (DG) and failure to isolate Emergency Feedwater (EFW) to the faulted steam generator.

The peak calculated Reactor Building temperature is 372.7°F. This peak calculated temperature results from the postulated 1.4 ft^2 DER at 102% power, assuming failure of a main steam isolation valve (MSIV) to close and a safety injection pump to start.

General containment design information is presented in Table 6.2.1. Figures 6.2-68 through 6.2-81 provide additional information on the general arrangement of Reactor Building structures and components. RN 02-021

6.2.1.1.1.1 Spectrum of Postulated Accidents

Five (5) postulated Reactor Coolant System pipe ruptures and 25 main steam line breaks have been analyzed to determine the peak Reactor Building pressure and temperature following an accident. For analytical purposes, it was assumed that each accident was coincident with loss of offsite power and a failure of at least 1 engineered safety feature. RN 03-003

The 5 postulated Reactor Coolant System pipe ruptures analyzed are as follows:

1. A double-ended pump rupture of the reactor coolant pump suction (DEPS) with maximum safety injection and 2 operable trains of the Reactor Building Spray (SP) System. RN 03-003

2. A DEPS with maximum safety injection and failure of 1 train of SP.

3. A DEPS with maximum safety injection, 2 operable trains of SP, and failure of 1 train of Reactor Building Cooling Units (RBCU). RN 03-003

4. A DEPS with minimum safety injection and failure of 1 train of emergency diesel generator coincident with a loss of offsite power.

5. A double-ended hot leg guillotine break (DEHL).

The mass and energy release information used for these enveloping Reactor Coolant System accidents is discussed in Section 6.2.1.3.10.1. The CONTEMP LT containment analysis code is used for this analysis and its use is discussed in Section 6.2.1.3.3.1. RN 03-003

The main steam line breaks (MSLB) analyzed to determine both peak reactor building pressure and peak reactor building temperature are listed in Table 6.2-1a. The mass and energy release data developed for these breaks is discussed in Section 6.2.1.3.10.3. The main steam line breaks are analyzed using the CONTEMPT LT/26 [27] containment analysis code discussed in Section 6.2.1.3.3.1.

Post accident pressure reduction is accomplished through the operation of the Reactor Building Spray System and Reactor Building cooling units and by the Reactor Building structures. The performance of the Reactor Building Spray System and Reactor Building cooling units is discussed in Section 6.2.1.3.4. Structural heat removal is also discussed in Section 6.2.1.3.4.

6.2.1.1.2 Subcompartments

The analysis of the pressures existing at any time in the reactor compartment and steam generator compartment is discussed in Section 6.2.1.3. This analysis is based upon mass and energy inputs described in Section 6.2.1.3.10. The compartment volumes and vent openings were obtained from design drawings and are consistent with the structural capacity of the walls discussed in Sections 3.8.1 and 3.8.3. All vent areas were calculated based upon "net" cross-sectional flow areas through pertinent openings in compartment walls and slabs. Net flow area is defined as the gross opening area minus any obstructions (piping, insulation, etc.), which are located in the particular opening. No credit was taken for the movement of obstructions or insulation in the analyses.

6.2.1.2 System Design

The design features of the Reactor Building and internal structures, as well as general arrangement drawings, are presented in Sections 3.8.1 and 3.8.3.

The codes, standards, and guides applied to the design of the Reactor Building and internal structures are included in Sections 3.8.1.2. and 3.8.3.2.

6.2.1.2.1 Design Provisions For Missiles and Pipe Whip

Design provisions are incorporated to protect the Reactor Building structure and Engineered Safety Features Systems against loss of function from dynamic effects that could occur following postulated accidents. The integrity of containment is maintained and functional capability of Engineered Safety Feature Systems is not compromised following postulated accidents that involve pipe rupture with subsequent pipe whip and jet impingement, and postulated accidents that involve missile impact. Design provisions include the following:

1. Physical separation of the redundant portions of Engineered Safety Features System.

2. Pipe rupture restraints that limit pipe movement following postulated pipe rupture.

3. Physical barriers, such as walls, floors, backfill, etc., to separate and protect equipment, as required.

4. Jet impingement shields, located where required, to protect specific piping or components from jet impingement from a postulated pipe break.

5. Detailed stress analyses to determine specific cases where provision of a shield or barrier is not required. This occasionally occurs in the path of low energy jets where sufficient stress margin is available in the pipe or component design to tolerate the impact.

Detailed discussion of design provisions for protection against postulated pipe rupture is presented in Section 3.6.

Detailed discussion of design provisions for protection against the effects of postulated missiles, both internal and external, is presented in Section 3.5.

6.2.1.2.2 External Loading

Inadvertent operation of the Reactor Building Spray System can result in a negative differential pressure in the Reactor Building. The external design pressure of the Reactor Building is 3.5 psig. Analysis of inadvertent Reactor Building Spray System operation is discussed in Section 6.2.1.3.6.

The external design pressure for tornado pressure loading is discussed in Section 3.3.

1. Reactor Building Pressure Equalization

   A narrow range differential pressure transmitter provides signals for monitoring Reactor Building internal pressure. Control room alarms are actuated by this pressure channel when the differential pressure approaches the baseline positive or negative pressure. Administrative procedures are implemented following the alarm to adjust Reactor Building pressure to within the required baseline pressure limits.

2. Inadvertent Reactor Building Spray

   During normal operation, inadvertent Reactor Building spray can occur only when performing the periodic inservice testing of one spray loop. An inadvertent safety injection signal to the spray loop discharge valve during spray loop test is the worst single failure.

6.2.1.2.3 Reactor Building Heating, Ventilating, and Air Conditioning System Functional Capability

The following Reactor Building ventilation systems operate continuously during normal plant operation to maintain suitable Reactor Building ambient temperature, pressure and humidity:

1. Reactor Building cooling units.

2. Reactor Building reactor compartment cooling system.

3. Reactor Building secondary compartment cooling system.

4. CRDM shroud cooling system.

The Reactor Building charcoal cleanup system may operate intermittently during normal plant operation to maintain suitable radiation levels inside the Reactor Building.

The 36 inch Reactor Building Purge System may be operated during cold shutdown and refueling to maintain suitable radiation levels inside the Reactor Building. During the other modes of plant operation (Modes 1-4), the valves are locked closed and operation of the system is not permitted. 98-01

The 6 inch alternate Reactor Building Purge System may be operated up to 1000 hours per year during modes 1-4, to maintain suitable moisture and radiation levels inside the Reactor Building. 98-01

Details of the operation of these systems are presented in Sections 6.2.2, 6.2.5, 6.5.1 and 9.4.8.

6.2.1.3 Design Evaluation

6.2.1.3.1 Reactor Building Peak Pressure and Temperature Analysis

In the event of a loss of coolant accident, much of the released reactor coolant will flash to steam. This release of mass and energy raises the temperature and pressure of the atmosphere within the Reactor Building. A rupture of a main steam pipe produces similar effects. The severity of the temperature and pressure peaks depends upon the nature, size, and location of the rupture.

A steam line break yields higher Reactor Building pressure than does a feedwater line break inside the Reactor Building. The same amount of water is added to the Reactor Building in each case. However, a steam line break causes the water to be added as superheated vapor, while a feedwater line break adds liquid with much lower enthalpy to the Reactor Building atmosphere. The feedwater line break is not considered to be as limiting as the main steam line break or primary system LOCA and, therefore, was not analyzed. RN 09-022

6.2.1.3.1.1 Peak Pressure Analysis

In order to identify the worst case, the spectrum of hypothetical LOCA and main steam line break accidents described in Section 6.2.1.1.1.1, based upon the initial conditions specified in Table 6.2-2, has been analyzed using the CONTEMPT LT computer program discussed in Section 6.2.1.3.3.1. RN 03-003

The break resulting in the maximum calculated pressure in the Reactor Building is the 1.40 ft^2 DER at 25% power main steam line break assuming a CH-A failure. The peak pressure in the Reactor Building is conservatively calculated to be 53.0 psig, at 1800 seconds following the break.

The following peak pressure analyses were performed:

1. Loss of Coolant Accident Analyses

   The loss of coolant accidents analyzed; information concerning break type, peak pressure, and temperature; time of peak pressure; single active failure assumed; and energy absorbed by passive heat sinks at the time of peak pressure are presented in Table 6.2-3.

   Figure 6.2-1 presents the calculated pressure as a function of time for the DEPS break with minimum safety injection.

   Figure 6.2-2 presents the calculated pressure as a function of time for the spectrum of pump suction DEPS break with maximum safety injection.

   Figure 6.2-3 presents the calculated pressure as a function of time for the double ended guillotine hot leg break (DEHL). This case was only run until the time at which the peak containment pressure is achieved. The break results in the peak RB pressure (45.1 psig) for the spectrum of LOCAs, and is unaffected with the reactor vessel upflow modification. RN 09-022

   Methods used, initial conditions, and single active failures assumed are discussed in Section 6.2.1.3.3.

2. Main Steam Line Break Analyses

   Figure 6.2-4 presents the calculated pressure versus time for the 1.40 ft^2 DER at 25% power main steam line break assuming a CH-A failure (the design basis accident).

   Methods used, initial conditions, and failures assumed for main steam line break analyses are discussed in Section 6.2.1.3.3.

   Table 6.2-3 presents a summary for each main steam line break analyzed.

6.2.1.3.1.2 Peak Temperature Analysis

The spectrum of steam line breaks listed in Table 6.2-1a, based upon the initial conditions specified in Table 6.2-2, has been analyzed using the CONTEMPT LT/26 [27] computer program discussed in Section 6.2.1.3.3.

The steam line break resulting in the maximum calculated temperature in the Reactor Building is the 1.4 ft^2 DER at 102% power with MSIV and SI failures. The peak temperature in the Reactor Building is calculated to be 372.7°F, 19 seconds after the break.

The following peak temperature analyses were performed.

1. Main Steam Line Break Analyses

   The main steam line break accidents analyzed; information concerning break type, peak temperature, and pressure; times of peak temperature and pressure; and active failures assumed are presented in Table 6.2-3.

   Figure 6.2-5a illustrates calculated temperature versus time for the worst case (temperature) main steam line break (1.4 ft^2 DER at 102% power with MSIV and SI failures).

   Figure 6.2-5b illustrates the calculated temperature as a function of time for a spectrum of main steam line split ruptures.

   Figures 6.2-5c and 6.2-5d illustrate the condensing steam heat transfer coefficient and the Reactor Building liner and interior concrete wall temperatures versus time for the worst case (temperature) main steam line break.

6.2.1.3.2 Long Term Reactor Building Performance

The long term system behavior during the design basis accident main steam line break and the worst case DEPS guillotine (minimum safety injection) LOCA have been evaluated to verify the ability of the ECCS and Reactor Building heat removal systems to maintain the Reactor Building pressure and temperature below design limits following a postulated high energy pipe rupture. This evaluation is based upon conservative predictions of the performance of these engineered safety features.

The primary system loss of coolant accident releases more energy into the Reactor Building over the course of the accident than the design basis accident main steam line break. The design basis accident mass and energy release is terminated after 1800 seconds. Since the plant operators are required to take action to terminate emergency feedwater flow to the faulted steam generator at 10 minutes, as discussed in Section 10.4.9.3, the analysis performed is conservative. Therefore, for long term Reactor Building performance, the LOCA is analyzed to 1440 hours and the design basis accident main steam line break is analyzed to 1800 seconds. The long term values of the Reactor Building parameters at 1440 hours for LOCA and 1800 seconds for the design basis accident main steam line break are presented in Table 6.2-4 and are shown graphically by Figures 6.2-1, 6.2-7 and Figures 6.2-4, 6.2-6, and 6.2-6a, respectively. The reactor vessel upflow modification has no impact on long-term Reactor Building parameters. RN 03-003

RN 09-022

Figures 6.2-8 through 6.2-14 are presented to demonstrate the long term cooling capability of the Reactor Building. The following conservative assumptions were used in the analysis:

1. No leakage.

2. Reactor Building Spray System is operated for a period of up to 40 days.

3. Reactor Building Cooling Units are operated for a period of up to 40 days.

4. Boiling in the reactor vessel is suppressed after 40 days by aligning the ECCS to the RCS hot legs and cold legs simultaneously after 8 hours during the hot leg recirculation phase. RN 03-003

The following parameters are illustrated by the figures indicated:

1. Total energy absorbed by passive heat sinks:

   a. Design basis accident main steam line break, Figure 6.2-8.

   b. LOCA, DEPS, minimum safety injection, Figure 6.2-9.

2. Condensing film heat transfer coefficients:

   a. Design basis accident main steam line break, Figure 6.2-10.

   b. LOCA, DEPS, minimum safety injection, Figure 6.2-11.

3. Reactor Building cooling units heat removal rate:

   a. Design basis accident main steam line break, Figure 6.2-12.

   b. LOCA, DEPS, minimum safety injection, Figure 6.2-13.

4. Residual Heat Removal System heat rate, Figure 6.2-14 (LOCA).

The accident chronology for the design basis accident main steam line break is given in Table 6.2-5.

The accident chronology for the LOCA (DEPS, minimum safety injection) is given in Table 6.2-6.

6.2.1.3.3 Methods of Analyses

6.2.1.3.3.1 Description of Computer Code

Calculation of Reactor Building pressure and temperature transients were performed using the digital computer code, CONTEMPT-LT. CONTEMPT-LT will model up to 4 containment volumes, consisting of primary system, wetwell, drywell, or dual containment. Only the primary system and drywell options are used. Transient phenomena within the Reactor Coolant System affect containment conditions by means of convective mass and energy transport through the pipe break. The containment drywell is separated into 2 regions. The first region consists of the air-steam phase; the second, the water phase. Sufficient relationships to describe the transient are provided by the equations of conservation of mass and energy as applied to each region, together with appropriate boundary conditions. RN 03-003

Two (2) versions of the CONTEMPT-LT computer program are used for analysis of Reactor Building pressure and temperature transients. CONTEMPT-LT/26 was used for analysis of MSLB and LOCA transients for steam generator replacement. Subsequent revision of LOCA Reactor Building pressure and temperature analysis used CONTEMPT-LT/28, which is described in NUREG/CR-0255 [47]. CONTEMPT-LT/28 is an update to version LT/26 described in this section. The analytical assumptions described in 6.2.1.3.3.2 apply to both versions of the program. No modifications to any models have been made CONTEMPT-LT/28 by SCE&G. The only significant modeling change resulting from the change to LT/28 from LT/26 is the use of the Uchida condensing heat transfer coefficient, rather than Tagami, for the condensing heat transfer to concrete and steel heat sinks during the LOCA transient, as described in 6.2.1.3.4.1. Containment pressure-temperature analysis for MSLB has not been revised since steam generator replacement and reflects the use of CONTEMPT-LT/26. Analysis of the double-ended hot leg rupture pressure and temperature transients has also not been revised since use of LT/28 was adopted. RN 03-003

The applicability of the CONTEMPT computer code to the containment pressure/temperature analyses can be verified by the Carolinas Virginia Tube Reactor (CVTR) experiments [24]. An excerpt from Reference [24] states the following:

> "The CVTR DBA tests and analyses have demonstrated that the CONTEMPT containment response analytical model will predict reasonably accurately the containment atmosphere pressure-temperature response provided correct assignment of condensing steam heat transfer coefficients and other key parameters are made. But, when common assumptions currently used for the heat transfer coefficient (Tagami) or similar were employed, the CVTR DBA test results indicated that the peak pressure prediction obtained from CONTEMPT was conservative by about 45%. However, the magnitude of overprediction of pressure depends on the blowdown conditions and other factors such that for a large pipe rupture (or fast blowdown), underestimating the heat transfer coefficient probably has a lesser effect on the peak pressure predictions (15% or less)."

CONTEMPT is basically a 1 node containment code. The containment temperature stratification, as measured in the CVTR tests, cannot be predicted by a 1 volume model which necessarily uses average overall heat transfer coefficients, such as Uchida[5] or Tagami[4]. However, the Uchida and Tagami correlation have been shown by the CVTR data and the work of Slaughterbeck [25] to be conservative for containment analysis.

6.2.1.3.3.2 Assumptions

The following are the major assumptions made in analysis:

1. Discharge mass and energy flow rates through the Reactor Coolant System break or main steam line break are established from the coolant blowdown and core thermal transient analysis described in Section 6.2.1.3.10

2. At the break point, the discharge water and corresponding energy are introduced into the vapor region. An energy balance is then made in 3 stages:

    a. Transfer of water and energy from the sump if the liquid region can boil.

    b. Determination of the masses of liquid and vapor in the atmosphere and the pressure and temperature of the atmosphere.

    c. Condensation of liquid from the vapor region to the liquid region and calculation of the liquid temperature.

    CONTEMPT LT/26 models condensate removal based upon the heat transferred to the containment heat sinks. For the main steam line break analysis, 92% of all condensate is assumed to be removed from the vapor region and added to the sump. For the LOCA analysis the isolated condensate model is not used. The use of the isolated condensate model results in a short term superheated temperature transient for dry steam blowdowns until containment sprays are actuated.

$$\dot{m} = F^{*} q\left(h_v - h_1\right)$$ 02-01

    Where:

$\dot{m}$	=	Rate of mass transfer of condensate from vapor region to pool region.
q	=	Heat transfer rate used to calculate condensate dropout rate.
h_v	=	Specific enthalpy of vapor in vapor region.
h_1	=	Specific enthalpy of liquid in vapor region.
F	=	Fraction of condensate formed on walls or cooling coils which is dropped into the pool region (F = 0.92).

3. Homogeneous mixing. The steam-air mixture has uniform properties, more specifically, thermal equilibrium between the air and steam.

4. Air is considered to be an ideal gas; water and steam tables are employed for thermodynamic properties.

During the transient, there is energy transfer from the steam-air and water regions to the internal structure and equipment within the Reactor Building shell. The temperature distribution through each of the heat conducting structures is determined by solving the 1 dimensional, multi-region heat conduction equation. Continuity of temperature and heat flow is assumed at composition interfaces. Boundary conditions are applied to the external surfaces of heat conducting structures as appropriate.

The following modifications have been made to CONTEMPT LT/26:

1. As originally programmed, CONTEMPT-LT/26 used the Uchida heat transfer coefficient even if the heat structure were superheated; a heat transfer coefficient of 2.0 Btu/hr-ft^2-°F (11.4w/m^2-°K) should be used. The following equations are now employed:

$$q_1 = h_u A \ (T_s - T_w)$$
$$q_2 = h_s A \ (T_v - T_w)$$

Where:

q_1, q_2 = Heat transfer rate.

h_u = Uchida heat transfer coefficient.

h_s = Superheated heat transfer coefficient.

= 2.0 Btu/hr-ft^2-°F(11.4 w/m^2-°K).

T_s = Containment vapor saturation temperature.

T_w = heat structure wall temperature.

T_v = Containment vapor temperature.

A = Heat structure surface area (ft^2).

2. The greater heat transfer rate is used: the Uchida transfer coefficient is used when the structure wall temperature is below saturation, while the superheated heat transfer coefficient is used when the structure surface is at, or above, saturation temperature.

3. Modifications have also been made to allow use of a tabular input for the condensing heat transfer coefficient and a constant forced convection heat transfer coefficient for each heat structure rather than using Uchida and natural convection coefficient of 2.0 Btu/hr-ft^2-°F.

4. These changes allow the isolated condensate model to be used in determining component temperature transients without restricting the heat transfer to the conservatively low values used in containment analysis.

5. A line change was also made to correct an error involving the driving potential for the structural heat sinks. This change corrected the energy imbalance that existed between heat removal from the vapor region and that absorbed by the passive heat sinks.

Provision is made in the computer analysis for the effects of several engineered safety features; including Reactor Building spray, Reactor Building cooling units and recirculation of sump water. Heat removal from the containment steam-air phase by Reactor Building spray is determined by allowing the spray water temperature to rise to the steam-air temperature.

6.2.1.3.3.3 Initial Conditions

The initial conditions used in the peak pressure LOCA and main steam line break analyses are presented in Table 6.2-2. The net free volume used is conservatively calculated from design drawings when establishing the heat sinks used for the ECCS analysis.

Under all operating conditions, the Reactor Building Cooling Systems maintain the following range of temperature and relative humidity conditions:

1. Temperature, 50°F to 120°F.

2. Relative humidity, 30% to 100%.

Under all operating conditions, Reactor Building pressure is maintained below 1.5 psig, the design limit achieved by manually relieving Reactor Building pressure, when required, as discussed in Section 6.2.1.2.2.

The initial conditions used in the peak temperature main steam line break analyses are the same as those used for the peak pressure analyses with the exception that initial Reactor Building pressure is 0.0 psig. (Used for breaks with no entrainment.) This maximizes the time it takes to reach the containment pressure setpoints for main steam isolation and Reactor Building spray actuation. It also reduces the mass of air available to absorb energy, resulting in higher containment vapor temperatures.

The use of the maximum Reactor Building operating temperature and a nominal relative humidity of 30% results in the highest calculated Reactor Building pressures.

Reactor Building heat removal through passive heat sinks and by the Reactor Building Spray System and cooling units is discussed in Section 6.2.1.3.4.

6.2.1.3.3.4 Failure of Engineered Safety Features

For the LOCA analyses, the peak containment temperature and pressure occur in the short term, near the end of blowdown and thus are not extremely sensitive to single failures. Long term pressures and temperatures are; however, maximized with the assumed failure of an emergency diesel resulting in the loss of 1 train of Reactor Building Spray, 1 Reactor Building cooling unit, and 1 train of ECCS.

For the DEPS guillotine breaks, the 4 cases analyzed are as follows:

1. A double-ended pump rupture of the reactor coolant pump suction (DEPS) with a maximum safety injection and 2 operable trains of Reactor Building Spray (SP) System. RN 03-003

2. A DEPS with maximum safety injection and failure of 1 train of SP.

3. A DEPS with maximum safety injection, 2 operable trains of SP, and failure of 1 train of Reactor Building Cooling Units (RBCU). RN 03-003

4. A DEPS with minimum safety injection and failure of 1 train of emergency diesel generator coincident with a loss of offsite power.

The main steam line break analyses conservatively assume one or more active failures. The active failures considered are as follows:

1. Electrical channel A (CH-A) failure, resulting in minimum Reactor Building heat removal from sprays and fan coolers and failure to isolate the flow to the broken loop steam generator, resulting in a 1000 gpm flow rate to that steam generator.

2. Main steam isolation valve (MSIV) on the broken loop steam generator fails to close, increasing the steam inventory released to the Reactor Building.

3. Main feedwater isolation valve fails (FWIV) to close, resulting in continued flow of feedwater.

4. Failure of a safety injection pump (SI) to operate.

5. Failure of 1 emergency diesel to start, resulting in minimum Reactor Building heat removal from sprays and fan coolers.

6. Failure of an emergency feedwater flow control valve to isolate flow to the faulted SG.

A summary of each main steam line break analyzed and its associated failures is shown in Table 6.2-1a.

The single active failure analysis of the systems relied upon to limit the mass and energy release and containment pressure/temperature response for the main steam line breaks are presented in the sections listed below:

1. Main steam, Section 10.3.3

2. Feedwater, Section 10.4.7.2.3

3. Emergency feedwater, Table 10.4-8

4. Reactor Building cooling, Tables 6.2-50 and 6.2-52

A summary of each main steam line break and LOCA analyzed is presented in Table 6.2-3.

6.2.1.3.4 Energy Sinks

6.2.1.3.4.1 Reactor Building Structures

Provision is made in the Reactor Building pressure transient analysis for heat storage in both interior and exterior walls. Every wall is divided into a large number of nodes. For each node, a conservation of energy equation expressed in finite difference form accounts for transient conduction into and out of the node and temperature rise of the node. Table 6.2-7 presents a detailed summary of the structural heat sinks. This summary includes the detailed information used in the ECCS analysis. For the Reactor Building peak pressure and long term analyses, the total amount of steel used was conservatively reduced from the values reported for the ECCS analysis. 02-01

The heat sinks were modeled for use in the CONTEMPT-LT computer code as shown by Table 6.2-8. The criteria used for mesh point spacing is based upon the work of Clausing [3]. RN 03-003

1. Concrete

    a. Mesh point of 0.05 inches for the first 3 inches.

    b. Mesh point spacing greater than, or equal to, 0.5 inches through the remaining thickness.

2. Steel

    Mesh point spacing of 0.05 inches to 0.08 inches.

The modeling of the Reactor Building liner/air/concrete was done using a volume weighted average for the steel/air mixture in the gap between the liner and the concrete. To assure conservatism, the thermal conductivity and volumetric heat capacity were halved.

The containment liner surface temperature and interior concrete surface temperature, as calculated by CONTEMPT LT/26 for worst case temperature main steam line break, are given by Figure 6.2-5d.

The thermophysical properties used in the peak pressure analyses are shown in Table 6.2-9. These represent conservatively low thermal conductivity and volumetric heat capacity values found in the literature.

The protective coatings on surfaces such as the Reactor Building liner and internal concrete were not considered in the analysis. This exclusion is based upon the following assumption.

If it is assumed that these coatings do not affect the steam condensation film coefficient, which most probably is the case, then the inclusion of an extra heat transfer "resistor", such as the coating, affects the overall heat transfer coefficient, U, by only 0.01%. This does not affect the Reactor Building pressure significantly.

Condensing film heat transfer coefficients are discussed in items 1 and 2, below:

1. Tagami

   During the blowdown period of a LOCA, heat transfer to steel heat sinks from the Reactor Building vapor region is based upon the Tagami [4] condensing heat transfer coefficient. Heat transfer to concrete heat sinks during blowdown is based on the Uchida correlation discussed later. RN 03-003

   The Tagami heat transfer coefficient increased parabolically to a peak value at the end of blowdown. Following the end of blowdown, the Uchida condensing heat transfer coefficient is used for both concrete and steel heat sinks. RN 03-003

   Tagami presents a plot of the maximum value of h as a function of "coolant energy transfer speed", defined as:

$$\frac{\text{total coolant energy transferred into containment}}{\text{(containment vessel volume) (time interval to peak pressure)}}$$

From this equation, the maximum heat transfer coefficient, h, for steel is calculated as follows:

$$h_{max} = 75\left(\frac{E}{t_p V}\right)^{0.60}$$

Where:

h_{max} = Maximum value of h (Btu/hr-ft^2-°F).

t_p = Time from start of accident to end of blowdown (sec.).

V = Reactor Building volume (ft^3).

E = Initial coolant blowdown energy (Btu).

The parabolic increase from the steady-state value of 2.0 Btu/hr-ft^2 -°F is given by the following equation:

$$h_s = h_{max}\left(\frac{t}{t_p}\right)^{1/2} ;\ 0 < t \leq t_p$$

Where:

h_s = Heat transfer coefficient for steel (Btu/hr-ft^2-°F).

t = Time from the start of accident (sec.).

For concrete, the heat transfer coefficient is taken as 40% of the value calculated for steel. RN 03-003

The Tagami condensing film heat transfer coefficient used for the DEPS with minimum safety injection is given by Figure 6.2.11.

2. Uchida

The main steam line break analysis uses the Uchida [5] correlation, rather than Tagami, for the condensing steam heat transfer coefficient. LOCA analysis uses the Uchida correlation for condensing heat transfer to concrete heat sinks during blowdown and for both concrete and steel heat sinks after the end of blowdown. RN 03-003

The Uchida correlation results in a lower heat transfer coefficient and, is therefore, the more conservative of the 2 for the main steam line break analyses. Slaughterbeck [25]

presents a discussion of the basis of heat transfer coefficients used for containment analysis.

Using the Tagami correlation with the peak heat transfer coefficient occurring when the main steam isolation valves close would result in higher heat transfer to the containment heat sinks. The Tagami correlation was not used due to the difficulty in defining the end of the turbulent portion of the blowdown.

The Uchida condensing film heat transfer coefficient is based primarily upon the mass ratio of air to steam in a nonturbulent environment. The Uchida correlation gives much lower heat transfer coefficients during the initial phase of the transient. The Uchida correlation, therefore, results in a conservative estimate of the heat removed from the vapor region during the main steam line blowdown.

The Uchida condensing steam heat transfer coefficients used during the worst case (pressure and temperature) main steam line breaks are presented by Figures 6.2-10 and 6.2-5c, respectively.

6.2.1.3.4.2 Reactor Building Spray

When a spray drop enters the hot saturated steam-air environment, the vapor pressure of the water at its surface is much less than the partial pressure of the steam in the atmosphere. Hence, there will be diffusion of steam to the drop surface and condensation on the drop. This mass flow will carry energy to the drop. Simultaneously, the temperature difference between the atmosphere and the drop will cause a heat flow to the drop. Both of these mechanisms will cause the spray drop temperature and vapor pressure to rise. The vapor pressure of the drop will eventually become equal to the partial pressure of the steam and the condensation will cease. The temperature of the drop will be essentially equal to the temperature of the steam-air mixture.

The terminal velocity of the drop can be calculated using the formula given by Weinberg[6] where the drag coefficient, C_D, is a function of the Reynolds (Re) number:

$$V^2 = \frac{4Dg(\rho - \rho_m)}{3C_D\rho_m}$$

Assuming a 700 micron drop size from the Reactor Building spray nozzles, the terminal velocity is less than 7 ft/sec. For a 1000 micron drop, the velocity would be less than 10 ft/sec. The Nusselt number for heat transfer, Nu, and the Nusselt number for mass transfer, Nu′ (Sherwood Number), can be calculated from the empirical relations given by Ranz and Marshall [7]:

$$Nu = 2 + 0.6\,(Re)^{1/2}\,(Pr)^{1/3}$$

$$Nu' = 2 + 0.6\,(Re)^{1/2}\,(Sc)^{1/3}$$

The Prandtl (Pr) number and the Schmidt (Sc) number for the conditions assumed are approximately 0.7 and 0.6, respectively. Both of these are sufficiently independent of pressure, temperature, and composition that they are assumed constant under containment conditions [8,9]. The coefficients of heat transfer (h_c) and mass transfer (k_G) are calculated from Nu and Nu′ respectively. The equations describing the temperature 02-01
rise of a falling drop are as follows:

$$\frac{d}{dt}(Mu) = mh_g + q$$

$$\frac{d}{dt}(M) = m$$

Where:

$$q = h_c A (T_s - T)$$
$$m = k_G A (P_s - P_v)$$

These equations can be integrated numerically to find the internal energy and mass of the drop as a function of time as it falls through the atmosphere. Analysis shows that the liquid drop temperature rises to the steam-air temperature in less than 0.5 seconds. This temperature increase occurs before the drop has fallen 5 feet. These results demonstrate that the Reactor Building spray will be 100% effective in removing heat from the Reactor Building atmosphere.

The Reactor Building Spray System is actuated based upon Hi-3 containment pressure. For the full double ended rupture main steam line break and Reactor Coolant System LOCA analyses, the pressure setpoint is reached well before the 10 seconds required for diesel generator startup. Reactor Building spray is assumed to actuate at 52 seconds for the LOCA and 53.1 seconds for the large main steam line breaks following accident initiation. This is conservative based upon the following:

1.	Diesel Start Time	10 seconds
2.	Time for RB Spray Pump to reach full capacity	5 seconds
3.	Time required to fill RB Spray piping and ring headers	32 seconds
4.	Total Elapsed Time	47 seconds

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Note: As a conservatism, filling of RB Spray piping is not credited during the 5 second pump acceleration phase.

Actuation timing calculations for the smaller main steam line breaks are presented in Table 6.2-47b . 98-01

6.2.1.3.4.3 Reactor Building Cooling Units

Heat removal by the Reactor Building cooling units is dependent upon the number of units operating, unit start and stop times, and the heat removal rate. The heat removal rate is dependent upon the Reactor Building atmosphere temperature and a number of thermal-physical cooler design parameters. The cooling coils of a Reactor Building cooling unit remove energy directly from the vapor region of the containment which results in reduced energy and pressure.

Heat removal due to Reactor Building cooling unit operation is simulated in the CONTEMPT code by specifying input values from a heat removal rate versus Reactor Building atmosphere temperature curve. The design values given on Figure 6.2-15, conservatively reduced to 60% to account for potential performance degradation, are used for MSLB analysis. LOCA analysis utilizes the RBCU performance curve corresponding to a conservative tubeside fouling factor of 0.0014 [46] shown on Figure 6.2-15. That curve is based on analytical results calculated by American Air Filter according to the methods described in Topical Report TR-7101A [16]. That analysis assumes the following boundary conditions: RN 03-003

Cooling water inlet temperature:	95° F
Cooling water flow:	2000 gpm
Tubeside fouling factor:	0.0014

The resulting performance curve gives a higher heat removal rate than the 60% design performance curve used in MSLB analysis, but still conservative compared to the design curve. All three curves are shown together for comparison on Figure 6.2-15.

The Reactor Building cooling units are calculated to be fully operational within 86.5 seconds following loss of offsite power. A description of the timing involved is presented in Section 6.2.2.2.2.2.2. The time used in the analyses is 86.5 seconds for the large main steam line breaks and the LOCA analyses. Actuation timing calculations for the main steam line breaks are presented in Table 6.2-47b.

6.2.1.3.4.4 Sump Water Recirculation Heat Exchangers

During the post accident long term cooling period, safety injection pumps (charging pumps and RHR pumps) supply recirculated water from the Reactor Building recirculation sumps to the reactor vessel. The RHR heat exchangers are placed in operation during the recirculation phase to remove energy directly from the reactor vessel to the outside environment.

The RHR heat exchangers are of the shell and U-tube type. Physical and thermal characteristics of each heat exchanger are specified in Table 6.2-1, as are the coolant inlet temperatures and flow rates.

6.2.1.3.5 Systems for Protection Against Negative Pressure Inside the Reactor Building

A narrow range differential pressure transmitter provides signals for monitoring the Reactor Building internal pressure. Control room alarms are actuated by this pressure channel when the differential pressure approaches the baseline pressure. Administrative procedures are used following actuation of the alarm to adjust Reactor Building pressure to within the required baseline pressure limits.

During normal operation, inadvertent Reactor Building spray can occur when doing the periodic inservice testing on 1 spray loop. An inadvertent safety injection signal to the spray loop discharge valve during spray loop testing is the worst single failure.

6.2.1.3.6 Reactor Building Analysis, Negative Pressure

Inadvertent operation of the Reactor Building Spray System causes depressurization of the Reactor Building due to the cooling effect of the spray water. A theoretical maximum depressurization can be calculated by assuming that the temperature of the Reactor Building atmosphere ultimately reaches the spray water temperature. (The evaporative cooling phase is not controlling for this case. See Reference [10]. Thus, the partial pressure of the air will simply change by a ratio of the absolute temperatures, while the vapor pressure will be equal to the saturation value at the spray water temperature. The theoretical maximum depressurization is:

$$P_d = P_o - \left[P_a \frac{T_{sp}}{T_o} + P_g\left(T_{sp}\right) \right]$$

Where:

P_d	=	Depressurization.
P_o	=	Initial pressure.
P_a	=	Initial air partial pressure.
$P_g\ (T_{sp})$	=	Water vapor partial pressure evaluated at T_{sp}.
T_{sp}	=	Spray water temperature.
T_o	=	Initial atmospheric temperature.

Using an initial temperature, pressure and relative humidity of 120°F, -0.1 psig, and 100%, respectively, for the Reactor Building atmosphere and a spray water temperature of 40°F, depressurization results in an external pressure of 3.5 psig. Therefore, the maximum external pressure does not exceed the design pressure given in Section 6.2.1.2.2. 98-01

6.2.1.3.7 Function Capability of the Normal Reactor Building Ventilation System

The normal ambient operating temperatures of the Reactor Building under various operating modes are indicted in Section 6.2.2.1.2. A single failure analysis is presented in Table 6.2-52. The maximum acceptable local compartment or area temperatures are described in Section 3.11.1.1. Instrumentation and alarms for the reactor building ventilation system are discussed in Section 9.4.8 . 98-01

6.2.1.3.8 Accident Pressure and Temperature Monitoring

The Reactor Building pressure and temperature and Reactor Building recirculation sump temperature instrumentation (including recording capability) is discussed in Section 7.5. The qualification of this instrumentation (including recording capability) is discussed in Section 7.5. The qualification of this instrumentation for use in the post accident containment environment is discussed in Sections 3.10, 3.11, and 7.4. The recorders are located in the control room, thus assuring accessibility to control room personnel during a LOCA. 02-01

6.2.1.3.9 Reactor Building Subcompartment Pressurization

Reactor Building subcompartments (steam generator compartments, pressurizer compartment, reactor cavity area) have been analyzed with respect to the appropriate design basis break to determine the peak pressure differential for pipe rupture. Each compartment has been broken down into volumes and flow paths. The steam generator and reactor cavity compartments were analyzed using the RELAP4/MOD5[30] code without air. The incompressible flow option without momentum flux was used for the momentum equation. Critical flow was predicted by the homogeneous equilibrium model, using the stagnation conditions of each node, consistent with the homogeneous equilibrium model critical flow table on RELAP4/MOD5. The transient pressure behavior of the pressurizer compartment has been calculated by the COMPARE-MOD 1 computer program [37]. Results of these analyses have been used in the compartment structural design.

The Subcompartments have been analyzed for the largest break(s) possible in each compartment. A summary is given below:

1. Pressurizer compartment - spray line and surge line breaks (see Section 6.2.1.3.9.1.)

2. Steam generator compartments - double ended hot leg and cold leg breaks (see Section 6.2.1.3.9.2)

3. Reactor cavity - cold leg break, 150 in^2 area (see Section 6.2.1.3.9.3).

The mass/energy release for each break used in subcompartment pressurization analyses is provided in Section 6.2.1.3.10.2.

The differential pressure on compartment walls is spatially varied for structural evaluation, where applicable. The vent parameters are realistically evaluated to provide a realistic pressure distribution following the accident.

The steam-water atmosphere was used to describe the initial conditions of Reactor Building subcompartments. The initial atmosphere within a subcompartment is modeled by a homogeneous water-steam mixture with an average density equivalent to the dry air model.

In addition, 100% entrainment has been assumed. The resulting calculated and design pressures are presented in Table 6.2.11. The analysis of each subcompartment is discussed separately in Sections 6.2.1.3.9.1 through 6.2.1.3.9.3.

The impact of the replacement steam generators and changes in plant operating parameters have been evaluated on the Reactor Building subcompartment analyses. Since the completion of the FSAR design analyses the Leak-Before-Break (LBB) Methodology has been applied in the plant design to the large primary piping. This eliminates the dynamic effects of postulated primary pipe ruptures from the design basis. Therefore, application of LBB means that the 150 in^2 cold leg break, DECL break and DEHL breaks need not be considered in the structural design basis. Continued use of these original bases results in a conservative design for the steam generator compartments and the reactor cavity which bounds any potential effects of the replacement steam generators and changes in the plant operating conditions.

For the pressurizer compartments, no reduction in break size is possible since the LBB methodology has only been applied to the large primary piping. Since the revised design power operating power parameters will allow for reduced Reactor Coolant System (RCS) temperatures, an increase in the pressurizer surge and spray line mass releases must be considered. The increase in the releases from previous analysis are conservatively bounded by increasing the surge and spray line mass releases by factors of 1.15 and 1.10, respectively. The corresponding enthalpies are conservatively 02-01

bounding for the lower RCS temperature. The results of this analysis are shown in Section 6.2.1.3.9.1.

6.2.1.3.9.1 Pressurizer Compartment Analysis

The pressurizer and surge tank compartments have been analyzed for pressure response from spray line and surge line breaks, respectively, using the COMPARE-MOD 1 computer program[37]. In each case, a 2 node model has been used. Figures 6.2-18 and 6.2-19 show the nodalization and appropriate parameters for the pressurizer compartment and surge tank compartment, respectively. Major relief areas for the pressurizer compartment include an opening at elevation 437'-6" and grating at elevation 488'-6". For the surge tank compartment, the major relief area is directly to the steam generator compartment. 02-01

The mass/energy releases are discussed in Section 6.2.1.3.10.2. Mass/energy 02-01 releases for the spray line rupture have been conservatively extended past the information provided by the NSSS vendor to permit calculation of the necessary transient differential pressure.

Figures 6.2-20 and 6.2-21 depict the pressure-time history for the pressurizer and surge tank compartments following the appropriate break. Differential pressures are also shown by these figures. Peak calculated and differential pressure are 41.9 psia (26.0 psid) and 45.0 psia (30.6 psid) for the pressurizer and surge tank compartments, respectively.

The peak calculated and design differential pressures for the pressurizer compartment are presented in Table 6.2-11.

The above mentioned calculated pressures are without margin. As shown by the valves in Table 6.2-11, the available margin between the calculated differential pressure and the structural design pressure is 59% for the pressurizer compartment and 68% for the surge tank compartment.

6.2.1.3.9.2 Steam Generator Compartment Analysis

The steam generator compartments were initially analyzed to determine the peak pressures on the compartment walls and the forces and moments acting upon the steam generator and reactor coolant pump supports. The general layout of the steam generator compartments is illustrated by Figure 6.2-22.

The analysis was performed using the RELAP4/MOD5 code[30]. Flow paths between nodes were modeled using inertial flow bounded by the homogeneous equilibrium critical flow model where significant changes in area occurred. In addition, all flow paths out of the break nodes were limited by the homogeneous equilibrium critical flow model regardless of geometry.

Initially, each steam generator compartment was analyzed for both the double ended hot and cold leg breaks. Models were set up to include significant restrictions and changes in area where practical.

From results of the initial studies it was determined that the loop C steam generator compartment was the worst case. Both the overall lower compartment pressure and specific nodal pressures due to geometric considerations were higher in loop C.

The loop C steam generator compartment was then modeled to include more definition and to establish loadings on the reactor coolant pump and steam generator.

The impact of the replacement steam generators and changes in plant operating parameters have been evaluated on the Reactor Building subcomponent analyses. Since the completion of the FSAR design analyses, the Leak-Before-Break (LBB) Methodology has been applied in the plant design to the large primary piping. This eliminates the dynamic effects of postulated primary pipe rupture from the design basis. Therefore, application of LBB means that the 150 in^2 cold leg break, DECL break and DEHL breaks need not be considered in the structural design basis. Continued use of these original bases results in a conservative design for the steam generator compartments and the reactor cavity which bounds any potential effects of the replacement steam generators and changes in the plant operating conditions.

6.2.1.3.9.2.1 Initial Steam Generator Compartment Models

The initial models for each loop were similar. In general, these models were comprised of 2 nodes below the steam generator and reactor coolant pump, 5 or 6 nodes between elevations 426.5’ and 436’, one node between elevations 436’ and 459.5’ and one node between elevations 459.5’ and 475.4’. Elevation 436’ was chosen as a dividing line because the steam generator supports and inservice inspection platforms form a natural boundary. The boundary at elevation 426.5’ is not real but does provide vertical symmetry for the break nodes.

The nodal arrangements, analytical results and model descriptions for the initial models are summarized by Table 6.2-11a. Tables 6.2-12 through 6.2-14a present model descriptions and results. Figures 6.2-22a through 6.2-22k illustrate nodal arrangements and results.

6.2.1.3.9.2.2 Final Loop C Steam Generator Compartment Models

A 20 node loop C steam generator compartment model was analyzed to determine nodal sensitivity and loadings on the reactor coolant pump due to a cold leg break. The volume between elevations 436’ and 451’ (top of reactor coolant pump) was divided into 5 nodes. The increase nodalization did not result in increased pressure in the break nodes since flow out of these nodes chokes early in the transient.

A 25 node compartment model was analyzed to determine nodal sensitivity and loadings on the steam generator due to a hot leg break. This model was similar to the 20 node model previously described but included more nodes around the steam generator above elevation 459.5’. The resultant peak pressure in break node 7 increased by less than 6%. This result was expected since the peak in node 7 is an inertial peak.

Forces and moments on the reactor coolant pump and steam generator were determined using results from the 20 and 25 node model, respectively.

The nodal arrangements, analytical results, model descriptions, and forces and moments data are summarized by Table 6.2-15. Tables 6.2-15a through 6.2-15f present model descriptions, and forces, and moments data. Figures 6.2-23 through 6.2-26 illustrate nodal arrangements, results and forces, and moments data.

6.2.1.3.9.2.3 Discussion of Results

From the tables and figures noted in Sections 6.2.1.3.9.2.1 and 6.2.1.3.9.2.2 it can be seen that the overall pressure response of the lower steam generator compartments (23 psid) is well below the design pressure of 41.2 psid.

A peak pressure of 47.9 psid was calculated (Node 4, cold leg) around the reactor coolant pump near the refueling canal wall.

A short term inertial peak of 45.9 psid (Node 7, hot leg) between the steam generator and secondary shield wall was calculated.

To study the impact of the above calculated pressures on the structural design, a simplified model of the lower steam generator compartment was developed. An analysis was performed using this model to evaluate the effects of the calculated pressures. Analytical results were compared to the results of a similar analysis for design pressure. It was determined that the design forces and moments for the above calculated pressures are less than for the design pressure. This reduction is due to the decrease in overall pressure even though pressure is increased locally.

6.2.1.3.9.3 Reactor Cavity Analysis

The impact of the replacement steam generators and changes in plant operating parameters have been evaluated on the Reactor Building subcomponent analyses. Since the completion of The FSAR design analyses the Leak-Before-Break (LBB) Methodology has been applied in the plant design to the large primary piping. This eliminates the dynamic effects of postulated primary pipe ruptures from the design basis. Therefore, application of LBB means that the 150 in^2 cold leg break, DECL break and DEHL breaks need not be considered in the structural design basis. Continued use of these original bases results in a conservative design for the steam generator

compartments and the reactor cavity which bounds any potential effects of the replacement steam generators and changes in the plant operating conditions.

1. Analyses

A pressurization analysis of the reactor cavity and penetration areas of the primary shield was performed using the RELAP4/MOD5 code[30] to predict pressures and forces and moments resulting from a postulated rupture. The analysis was performed by considering the maximum break size which could occur in the primary shield pipe penetrations. Cold leg and hot leg guillotine breaks are limited by restrictions imposed upon pipe movement by the penetration, supports, and loop geometry. Cold leg break geometry is illustrated by Figure 6.2-27. Due to the break geometry, the cold leg break area is larger than the hot leg break area. The maximum cold leg break area is limited to a value less than 150 in^2, the value used for mass/energy releases (see Section 6.2.1.3.10.2).

The general layout of the reactor cavity area is illustrated by Figure 6.2-28. A break in the loop B cold leg was selected for analysis because of the small inspection port opening. Dimensions on Figure 6.2-28 represent cold conditions. Thermal expansion under hot conditions was included in the calculation of the various model parameters in the RELAP4/MOD5 code. Insulation was assumed to move against the reactor vessel and/or piping, as appropriate, instantaneously with not change in thickness. Further details are provided by Figures 6.2-65 through 6.2-67.

A nodalization sensitivity study was performed for the forces and moments. Axes used in the orientation of the reactor vessel are illustrated by Figure 6.2-29. Forces in the x and z directions and the moment about y axis are the major components. These major components were considered in the acceptance of the nodalization sensitivity study.

Three (3) reactor cavity modes, incorporating 22, 31, and 33 nodes, were analyzed. The 22 node model included 3 nodes in the cold leg penetration and 15 nodes in the reactor vessel annulus. The 31 node model also used 3 nodes in the penetration but used 24 nodes in the annulus. The 33 node model incorporated 5 nodes in the penetration and 24 nodes in the annulus.

Information presented for the reactor cavity models include penetration, reactor vessel annulus, and overall models, as well as forces and moments. A cross reference between various models and appropriate tables and figures is presented by Table 6.2-16. Tables 6.2-16a through 6.2-16c address control volumes; Tables 6.2-17 through 6.2-17b, flow paths; and Tables 6.2-18 through 6.2-18b, force/moment areas. Figures 6.2-30 through 6.2-32a illustrate penetration nodes, reactor vessel annulus nodes, and overall model schematics.

2. Results

The results from the 3 models are the pressures in each node and the resulting forces and moments. The peak differential pressure with respect to containment pressure is tabulated for each node and differential pressure transient for the break node and reactor vessel annulus nodes adjacent to the break are also shown. In addition, forces and moments are tabulated for each model. A cross reference is provided by Table 6.2-18c. Tables 6.2-16a through 6.2-16c present peak differential pressures. Tables 6.2-18d through 6.2-18f present forces and moments. Figures 6.2-33 through 6.2-33e illustrate transient differential pressures for selected nodes.

The peak pressures listed in Table 6.2-16a through 6.2-16c are well below the design pressures in the penetration and reactor vessel annulus.

Peak forces and moments of interest are presented by Table 6.2-18g for each of the 3 models. The results of the nodalization sensitivity study indicate reasonable agreement among the 3 models.

Since the 33 node model is the most detailed, the numbers for this model are used for the design evaluation. A transient representation of forces and moments for the 33 node model is provided by Figures 6.2-34 through 6.2-35a.

6.2.1.3.10 Mass and Energy Releases

Rupture of any of the piping carrying pressurized high temperature reactor cooling water, termed a loss of cooling accident (LOCA), or rupture of a secondary system steam line within the containment will result in release of steam and water into the containment. This in turn will result in an increase in the containment pressure and temperature. In order to evaluate the response of the containment to such accidents, the rate of release as a function of time must be evaluated. Section 6.2.1.3.10.1 describes the long term releases following a LOCA in the primary cooling system. These results are used to evaluate the overall effect of such releases on the containment. Section 6.2.1.3.10.3 describes the releases resulting from rupture of a secondary system steam line located within the containment.

6.2.1.3.10.1 LOCA Mass and Energy Release

6.2.1.3.10.1.1 LOCA Release Rate Transient

The LOCA transient is typically divided into 4 phases:

1. Blowdown - which includes the period from accident initiation (when the reactor is at steady state operation) to the time that the RCS pressure reaches initial equilibrium with containment.

2. Refill - the period of time when the lower plenum is being filled by accumulator and safety injection water. At the end of blowdown, a large amount of water remains in the cold legs, downcomer, and lower plenum; therefore to conservatively consider the refill period for the purpose of containment mass and energy releases, this water is instantaneously transferred to the lower plenum. This allows an uninterrupted release of mass and energy to containment. Thus the refill period is conservatively neglected in the mass and energy release calculation.

3. Reflood - begins when the water from the lower plenum enters the core and ends when the core is completely quenched.

4. Post-Reflood (Froth) - describes the period following the reflood transient. For the pump suction break, a two-phase mixture exits the core, passes through the hot legs, and is superheated in the steam generators. After the broken loop steam generator cools, the break flow becomes 2 phase.

Tables 6.2-19 through 6.2-22, 6.2-25, and 6.2-26 provide the tabulations of the current long term mass and energy release rates versus time for all 4 phases over the spectrum of breaks analyzed.

6.2.1.3.10.1.2 Mass and Energy Release Analyses For Postulated Loss-of Coolant Accidents

The evaluation model used for the long term LOCA mass and energy release calculations used was the March 1979 model described in Reference [38]. This evaluation model has been reviewed and approved by the NRC, and has been used in the analyses of other dry containment plants.

The analyses presented in this section have also been performed to evaluate any potential effects the reactor vessel upflow conversion had on the mass and energy releases. This design change was shown to have a negligible effect on the calculated LOCA mass and energy releases. RN 09-022

For the long term mass and energy release calculations, operating temperatures for the highest average coolant temperature at full power (see Table 5.1-1) were selected as the bounding analysis conditions. The RCS temperature conditions for the rerating analysis for the long term LOCA transients are also bounding for asymmetric loop flow considerations. The modeled core power level of 2958 MWt, adjusted for calorimetric error (+2% of power), was the basis in the analysis. The use of higher temperatures is conservative because the initial fluid energy is based on coolant temperatures which are at the maximum levels attained in steady state operation. Additionally, an allowance of +5.3°F is reflected in the temperatures in order to account for instrument error and deadband. The initial RCS pressure in this analysis is based in a nominal value of 2250 psia. Also included is an allowance of +50 psia, which accounts for the uncertainty on pressurizer pressure. The resulting limiting pressure of 2300 psia affects the blowdown phase results, since this represents the initial pressure of the RCS. The RCS rapidly depressurizes from this value until the point at which it equilibrates with containment pressure.

RN 01-125

The rate at which the RCS blows down is initially more severe at the higher RCS pressure (2300 psia). Additionally, the RCS has a higher fluid density at 2300 psia (assuming a constant temperature) and subsequently has a higher RCS mass available for release. Thus, 2300 psia initial pressure was selected as the limiting case for the long term mass and energy release calculations. These assumptions conservatively maximize the mass and energy in the RCS.

The selection of fuel allowance for the long term mass and energy calculations and subsequent LOCA containment integrity calculation is based on the need to conservatively maximize the core stored energy. The margin is core stored energy was chosen to be +15%. Thus, the analysis very conservatively accounts for the stored energy in the core. The fuel conditions were adjusted up to provide a bounding analysis.

Margin in RCS volume of 3% (which is composed of 1.6% allowance for thermal expansion, and 1.4% for uncertainty) is modeled.

Regarding safety injection flow, the mass and energy calculation considers both minimum and maximum safety injection flow rates. Further details concerning these assumptions are contained in Section 6.2.1.3.10.1.9.

6.2.1.3.10.1.3 LOCA Break Size and Location

Generic studies have been performed with respect to the effect on LOCA mass and energy releases to postulated break size. The double-ended guillotine break has been found to be limiting due to larger mass flow rates during the blowdown phase of the transient. During the reflood and froth phases, the break size has little effect on the releases.

Three (3) distinct locations in the reactor coolant loop can be postulated for pipe rupture:

1. Hot leg (between vessel and steam generator)
2. Cold leg (between pump and vessel)
3. Pump suction (between steam generator and pump)

The break location analyzed and described herein is the double-ended pump suction guillotine break (10.48 ft^2). Pump suction break mass and energy releases have been calculated for the blowdown, reflood, and post-reflood phases of the LOCA.

The following information provides a discussion on each break location. The double-ended hot leg guillotine has been shown in previous studies to result in the highest blowdown mass and energy release rates. Although the core flooding rate would be highest for this break location, the amount of energy released from the steam generator secondary side is minimal because the majority of the fluid, which exists the core, bypasses the steam generators in venting to containment. As a result, the reflood mass and energy releases are reduced significantly as compared to either the pump suction or cold leg break locations where the core exit mixture must pass through the steam generators before venting through the break.

For the hot leg break, there is no reflood peak as determined by generic studies (i.e., from the end of the blowdown period the releases would continually decrease). Therefore, the reflood (and subsequent post reflood) releases are not calculated for a hot leg break. The mass and energy releases for the hot leg break blowdown phase have been included in the scope of this containment integrity analysis.

The cold leg break location has also been found in previous studies to be much less limiting in terms of the overall containment peak pressure. The cold leg blowdown is faster than that of the pump suction break and more mass is released into the containment. However, the core heat transfer is greatly reduced and this results in a considerably lower energy release into containment. Studies have determined that the blowdown transient for the cold leg is, in general, less limiting than that for the pump suction break. During reflood, the flooding rate is greatly reduced and the energy release rate into the containment is also reduced. Therefore, the cold leg break analysis is not usually performed.

The pump suction break combines the effects of the relatively high core flooding rate as in the hot leg break and the addition of the stored energy in the steam generators.

As a result, the pump suction break yields the highest energy flow rates during the post-blowdown period by including all of the available energy of the Reactor Coolant System in calculating the releases to containment. This break location has been determined to be the limiting break for typical dry containment plants. The choice of this break location, as the limiting break analyzed for the V. C. Summer plant, is consistent with other dry containment plants for the post-blowdown phase of the event.

In summary, the analysis of the limiting break location for a dry containment has been performed. The double-ended pump suction guillotine break has historically been considered to be the limiting break location for the post-blowdown phase of the event by virtue of its consideration of all energy sources present in the RCS. The analyses presented support the conclusions of the double-ended pump suction (DEPS) as the limiting break case for the post-blowdown period, considering both the minimum and maximum safety injection cases. This break location provides a mechanism for the release of the available energy in the Reactor Coolant System, including both the broken and intact loop steam generators.

6.2.1.3.10.1.4 LOCA Mass and Energy Release Data

The following items ensure that the mass and energy releases are conservatively calculated for maximum containment pressure:

1. Maximum expected operating temperature of the Reactor Coolant System.

2. Allowance in temperature for instrument error and deadband (+5.3°F).

3. Margin in volume of 3% (which is composed of 1.6% allowance for thermal expansion and 1.4% for uncertainty).

4. Nominal core power level of 2900 MWt.

5. Allowance for calorimetric error (+ 2% of power).

6. Conservative coefficients of heat transfer (i.e. steam generator primary/secondary heat transfer and Reactor Coolant System metal heat transfer).

7. Allowance in core stored energy effect of fuel densification.

8. Margin in core stored energy (+ 15%).

9. Allowance for RCS pressure uncertainty (+ 50 PSI).

6.2.1.3.10.1.5 Blowdown Mass and Energy Release Data

The SATAN-VI code is used for computing the blowdown transient and is the same as that used for the ECCS calculation in Reference [39]. The methodology for the use of this model is described in Reference [38].

Tables 6.2-19 and 6.2-20 present calculated mass and energy releases or the blowdown phase of the break analyzed for the DEPS and DEHL breaks, respectively. Break flow time histories from each side of the guillotine break are tabulated, where Break Flow Path No. 1 represents the flow from the reactor vessel outlet side of the break and Break Flow Path No. 2 represents the flow from the reactor vessel inlet side of the break. The mass and energy release for the double-ended pump suction break and the double-ended hot leg break given in Table 6.2-19 and 6.2-20, terminate 19.6 and 18.2 seconds, respectively, after the initiation of the postulated accident.

6.2.1.3.10.1.6 Reflood Mass and Energy Release Data

The WREFLOOD code is used for computing the reflood transient, is a modified version of that used in the ECCS calculation in Reference [39]. The methodology for the use of this model is described in Reference [38].

To clarify the mass and energy evaluation model described in Reference [38] steam/water mixing in the broken loop has been included in this analysis. This assumption is justified and is supported by test data, summarized as follows: 02-01

The model assumes a complete mixing condition (i.e., thermal equilibrium) for the steam/water interaction. The complete mixing process, however, is made up of 2 distinct physical processes. The first is a 2 phase interaction with condensation of steam by cold injection water. The second is a single phase mixing of condensate and injection water. Since the mass and energy of the steam released is the most important influence to the containment pressure transient, the steam condensation part of the mixing process is the only part that need be considered. (Any spillage directly heats only the sump.)

The most applicable steam/mixing test data has been reviewed for validation of the containment integrity reflood steam/water mixing model. This data is that generated in the 1/3 scale test (Reference [40]), which are the largest scale data available and thus most closely simulates the flow regimes and gravitational effects that would occur in a PWR. These tests were designed specifically to study the steam/water interaction for PWR reflood conditions.

From the entire series of 1/3 scale tests, a group corresponds almost directly to containment integrity reflood conditions. The injection flow rates for this group cover all phases and mixing conditions calculated during the reflood transient. The data from these tests were reviewed and discussed in detail in Reference [38]. For all these tests the data clearly indicates the occurrence of very effective mixing with rapid steam

condensation. The mixing model used in containment integrity reflood calculation is therefore, wholly supported by the 1/3 scale steam/water mixing data.

Additionally, the following justification is also noted. The limiting break for the containment integrity peak pressure analysis during post-blowdown phase is the double-ended pump suction break. For this break, there are 2 flow paths available in the RCS by which mass and energy may be released to containment. One (1) is through the outlet of the steam generator, the other via reverse flow through the reactor coolant pump. Steam, which is not condensed by ECCS injection in the intact RCS loops passes around the downcomer and through the broken loop cold leg and pump in venting to containment. This steam also encounters ECCS injection water as it passes through the broken loop cold leg; complete mixing occurs and a portion of it is condensed. It is this portion of steam, which is condensed, that is taken credit for in this analysis. This assumption is justified based upon the postulated break location and the actual physical presence of the ECCS injection nozzle. A description of the test and test results is contained in References [38] and [40].

The methodology previously discussed and described in Reference [38] has been utilized and approved on Dockets for Catawba Units 1 and 2, Indian Point 2 and 3, McGuire Units 1 and 2, Sequoyah Units 1 and 2, Millstone 3, Shearon Harris Unit 1, and Beaver Valley Unit 2. 02-01

Tables 6.2-21 and 6.2-22 present the calculated mass and energy release for the reflood phase of the double-ended pump suction break with minimum and maximum safety injection respectively. Flow time histories from each side of the double-ended pump suction break are tabulated where Break Flow Path No. 1 represents the flow through the outlet of the steam generator and Break Flow Path No. 2 represents reverse flow through the reactor coolant pump. A significantly higher mass and energy release occurs during the period the accumulators are injecting (from 22.3 to 43.8 seconds for minimum and maximum safety injection as illustrated in Table 6.2-21 and 6.2-22). The transient of the principal parameters during reflood are given in Tables 6.2-23 and 6.2-24 for the minimum and maximum safety injection double-ended pump suction break case. 02-01

6.2.1.3.10.1.7 Post-Reflood Mass and Energy Release Data

The FROTH code is used for computing the post-reflood transient. The methodology for the use of this model is described in Reference [38]. The mass and energy rates calculated by FROTH are used in the containment analysis until the time of containment depressurization.

After depressurization, the mass and energy release from decay heat is based on the 1979 ANSI/ANS Standard, Shown in Reference [41], and the following input:

1. Decay heat sources considered are fission product decay and heavy element decay of U-239 and Np-239.

2. Decay heat power from fissioning isotopes other than U-235 is assumed to be identical to that of U-235.

3. Fission rate is constant over the operating history of maximum power level.

4. The factor accounting for neutron capture in fission products was taken from Table 10 of Reference [41].

5. Operating time before shutdown is 3 years. 02-01

6. The total recoverable energy associated with 1 fission has been assumed to be 200 Mev/fission.

7. Two (2) sigma uncertainty (2 times the standard deviation) has been applied to the fission product decay.

Table 6.2-34 gives the core generated decay heat used in the long term mass and energy release calculation.

Tables 6.2-25 and 6.2-26 present the 2 phase (froth) mass and energy release data for the double-ended pump suction break minimum and maximum safety injection cases. Flow time histories from each side of the double-ended pump suction break are tabulated, where Break Flow Path No. 1 represents the flow through the outlet of the steam generator and Break Flow Path No. 2 represents reverse flow through the reactor coolant pump.

6.2.1.3.10.1.8 Mass and Energy Sources

The sources of mass considered in the LOCA mass and energy release analysis are given in Tables 6.2-27, 6.2-28, and 6.2-29. These sources are the Reactor Coolant System, accumulators, and pumped safety injection.

The energy inventories considered in the LOCA mass and energy release analysis are given in Tables 6.2-30, 6.2-31, and 6.2-32.

The components included in the mass and energy calculations are:

1. Reactor Coolant System water

2. Accumulator water

3. Pumped Injection water

4. Decay heat

5. Core stored energy

6. Reactor Coolant System metal

7. Steam generator metal

8. Steam generator secondary energy

9. Secondary transfer of energy (feedwater into and steam out of the steam generator secondary)

In the mass and energy release data presented, no Zirc-water reaction heat was considered because the clad temperature did not rise high enough for the energy released by this reaction to be of any significance.

System parameters needed to perform confirmatory analyses are provided in Table 6.2-33.

The consideration of the various energy sources in the mass and energy release analysis provides assurance that all available sources of energy has been included in this analysis. Thus, the review guidelines presented in the Standard Review Plan have been satisfied.

The mass and energy inventories are presented at the following times, as appropriate:

1. Time zero (initial conditions)

2. End of blowdown time

3. End of refill time

4. End of reflood time

5. Time of full depressurizations

The methods and assumptions used to release the various energy sources are given in Reference [38] except as noted in Section 6.2.1.3.10.1.6, which has been approved as a valid evaluation model by the Nuclear Regulatory Commission.

6.2.1.3.10.1.9 Application of Single Failure Criteria

An analysis of the effects of the single failure criteria has been performed on the mass and energy release rates for the (DEPS) break. For the DEPS results presented in this report an inherent assumption in the generation of the mass and energy release is that offsite power is lost. This results in the actuation of the emergency diesel generators, required to power the safety injection system. This is not an issue for the blowdown period, which is limited by the DEHL break.

Three (3) cases have been analyzed for the effects of a single failure. The double-ended pump suction case with both minimum and maximum safety injection for the 2900 MWt rerated conditions was analyzed. (RN 03-003) In the case of minimum safety injection, the single failure, postulated to occur, is the loss of an emergency diesel generator. This results in the loss of 1 pumped safety injection train, thereby, minimizing the safety injection flow and the loss of 1 train of RB spray and RBCU. For the case of maximum safety injection, two separate single failures are postulated to occur. One failure is the loss of 1 train of RB spray. The other is the loss of 1 train of RBCU. (RN 03-003) The analysis of both minimum and maximum safety injection cases ensures that the effect of all credible single failures is bounded.

6.2.1.3.10.2 Short Term Mass and Energy Releases

The model used to perform the short term blowdown analysis is fully described in Reference [12]. The release rates used for the reactor cavity, steam generator compartment, and pressurizer compartment analyses are listed in the following tables:

1. Reactor cavity blowdown for 150 in^2 cold leg break.
2. Cold leg DEG. (02-01)
3. Hot leg DEG.
4. Double ended pressurizer surge line break, Table 6.2-44.
5. Pressurizer spray line break, Table 6.2-45.

As indicated in Section 3.6, postulated breaks in the reactor coolant loop piping, except for branch line connections, have been eliminated through the use of Leak-Before-Break (LBB) methodology. The mass and energy releases previously used are retained; however, as bounding values to serve as the basis for the Reactor Building subcompartment pressurization analyses described in Section 6.2.1.3.9. (02-01) The RCS temperature conditions for the rerating analysis for the short term LOCA transients are also bounding for asymmetric loop flow considerations. (RN 01-125)

6.2.1.3.10.3 Mass And Energy Release Analyses (Containment Response) For Postulated Secondary System Pipe Ruptures Inside Containment

6.2.1.3.10.3.1 Method Of Analysis

The analysis of the main steam line break (MSLB) inside containment includes a calculation of the mass and energy release from the blowdown and the corresponding containment response. The mass and energy releases following a steam line rupture are dependent upon the configuration of the plant steam system and containment design, as well as the plant operating conditions and size of the rupture. The cases examined in this study were chosen to determine a limiting case, in terms of containment temperature and pressure response.

The following items must be considered for a steam line rupture inside containment analysis:

- nuclear kinetics characteristics and power generation in the reactor core,
- stored energy in both the primary reactor system and the secondary steam plant,
- main and auxiliary feedwater systems operation,
- safety systems operation (e.g. reactor trip, steam line isolation, ECCS),
- break size and location, and
- blowdown characteristics.

The analyses utilize a steam line break mass and energy release methodology which considers all of these factors, such that conservative steam line break blowdown transients may be determined.

The methods described in Reference [28] were used to generate the mass and energy release information. These methods form the basis for the assumptions and models used in the calculation of the mass and energy releases resulting from a steam line rupture. Reference [28] presents an extensive analysis of the nature of the effluent releases following a postulated main steam line rupture, as well as a discussion of the methods and models and models used.

Transient mass and energy releases following a postulated secondary side pipe break are calculated using the LOFTRAN code (Reference [32]). LOFTRAN is used for studies of the transient response of a PWR system to specified perturbations in process parameters. The code stimulates a multi-loop system including the reactor vessel, hot and cold leg piping, steam generator (shell and tube sides), and the pressurizer. A neutron point kinetics model is used and the reactivity effects of the moderator, fuel, boron, and rods are included. The secondary side of the steam generator is modeled as a homogeneous saturated mixture. Protection and control systems are simulated, as

well as the Safety Injection System. The calculation of the secondary side break flow is based on the Moody critical flow correlation (Reference [42]) with fl/D=0.

6.2.1.3.10.3.2 Definition Of Cases

The postulated break area can have competing effects on blowdown results. Larger break areas result in larger break mass flow rates. However, larger breaks also result in earlier generation of protective trip signals following the break and a reduction of both the power reduction by the plant and amount of high energy fluid available to be released to the containment.

To determine the effects of plant power level and break area on the mass and energy releases from a ruptured steam line, a spectrum of cases have been evaluated. For plant power levels of 102%, 75%, 50%, 25%, and 0% of the Engineered Safeguards Design (ESD) rating, 4 break sizes have been defined.

1. A full double-end rupture (DER) of main steam line. Note that integral steam generator flow restrictors limit the maximum break size to 1.4 ft^2 in any 1 steam line.

2. A large DER having an area just larger that that at which water entrainment occurs.

3. A small DER having an area just smaller than that at which water entrainment occurs.

4. A small split rupture defined as the largest break size in which the isolation signals must be generated by high containment pressure signals and result in no water entrainment on the break effluent.

Table 6.2-47a shows the steam line break mass and energy release to containment cases which were analyzed. Note that, at full power, no entrainment was predicted in any of the DER cases. However, full power large and small DER break sizes were analyzed, both with no entrainment, to provide a sensitivity. For the steam line break mass and energy releases to containment cases analyzed, a number of failures have been considered either individually, or in combination with other failures. The failures considered are: failure of the emergency feedwater runout control system to function, failure of a single train of the Safety Injection System, and failure of a main feedwater isolation valve to close.

Failure of a main steam isolation valve to close has not been explicitly considered as part of the mass and energy release calculation, but is included separately with the containment response portion of this analysis.

6.2.1.3.10.3.3 Mass And Energy Release Calculation Assumptions

The key assumptions made in the calculation of mass and energy releases for determination of the containment response analysis are discussed below.

6.2.1.3.10.3.3.1 Reactor Coolant System

Table 6.2-46b identifies the nominal plant design parameters considered in this analysis.

Initial Power Level

Steam line breaks can be postulated to occur with the plant in any operating condition ranging from zero to full power. Since steam generator mass decreases with increasing power level, breaks occurring at lower power levels will generally result in a greater total mass release to containment. However, because of increased stored energy in the primary plant, increased heat transfer in the steam generator, and additional heat generation in the fuel, the energy release to the containment from breaks postulated to occur while at power, may be greater than for breaks occurring with the plant at a zero power condition. Additionally, steam pressure and the dynamic conditions in the steam generators change with pressure and the dynamic conditions in the steam generators change with increasing power and have a significant influence on both the rate of blowdown and the amount of moisture entrained in the fluid leaving the break.

Because of the opposing effects (mass versus energy release) of changing power level on steam line break releases, no single power level can be singled out as a worst case initial condition for containment response following a steam line break. Therefore, several different power levels from zero to full power conditions were investigated.

In general, the plant initial conditions for each case are assumed to be at the nominal value, with appropriate uncertainties included. Table 6.2-46c identifies the values assumed for RCS flowrate, pressurizer pressure, RCS vessel average temperature, pressurizer water volume, steam generator water level, and feedwater enthalpy corresponding to each power level analyzed.

Core Decay Heat

Core residual heat generation assumed is based on the 1979 ANS Decay Heat + 2σ model.

Reactor Coolant Metal Heat Capacity

As the primary plant cools, the temperature of the primary coolant drops below the temperature of the reactor coolant piping, the reactor vessel, and the reactor coolant pumps. As this occurs, the heat stored in the metal is available to be transferred to the faulted steam generator. The effects of stored heat in the RCS metal mass are included in the results using conservative thick metal masses and heat transfer coefficients. However, stored heat does not have a major impact on the calculated mass and energy releases.

Core Reactivity Coefficients

Conservative core reactivity coefficients corresponding to end-of-cycle conditions were chosen to maximize the reactivity feedback effects resulting from the steam line break. Use of maximum reactivity feedback maximizes the potential for a post trip return to power, this conservatively maximizes heat transfer to the steam generator. The most reactive control rod assembly assumed to be stuck out of the core.

Rod Control

The rod control system was assumed to be in manual operation for all analyses.

6.2.1.3.10.3.3.2 Secondary System

Main Feedwater System

The rapid depressurization which occurs following a steam line rupture typically results in large amounts of water being added to the steam generators through the Main Feedwater System. Rapid closing feedwater control valves in the main feedwater lines limit this effect. The feedwater addition which occurs prior to closing of the feedwater line control valves influences the steam generator blowdown in several ways. First, the rapid addition increase the amount of entrained water in large break cases by lowering the bulk quality of the steam exiting the rupture. Secondly, because the water entering the steam generator is subcooled, it lowers the steam pressure thereby reducing the flow rate out of the break. As the steam generator pressure decreases, some of the fluid in the feedwater lines downstream of the control valves will flash into the steam generators providing additional secondary fluid which may exit out the rupture. Finally, the increased flow causes an increase in the heat transfer rate from the primary to secondary system resulting in greater energy being released out the break. Since these are competing effects on the total mass and energy release, no worst case feedwater transient can be defined for all plant conditions.

In the analysis, the worst effects of each variable have been assumed. For example, moisture entrainment for each break is calculated assuming conservatively small feedwater additions such that the entrained water is minimized. Determination of total steam generator inventory, however, is based on conservatively large feedwater additions as described below.

For double-ended-ruptures, main feedwater flow to the intact steam generators was modeled by assuming a constant feedwater flow until feedwater isolation. This minimizes the fluid inventory in the intact generators at the time of isolation, which reduces their cooling effect. This conservatively maximizes the energy which must be released through the break. The mass of feedwater added to the affected steam generator prior to feedline isolation is based upon the conservative assumption that the feedwater flow is at pump runout for the full-time period from initiation of the break until feedline isolation is complete. Feedline isolation may be provided by either the

feedwater isolation valve or closure of both the feedwater regulating valve and the feedwater bypass control valve. The total response time of 10 seconds includes a 1.5 second instrument response time and 8.5 second valve stroke time.

For the split breaks, feedwater flow to all generators was increased proportionally with the steam releases from the steam generators.

Following feedwater isolation, as the steam generator pressure decreases, some of the fluid in the feedwater lines downstream of the isolating valve may flash to steam if the feedwater temperature exceeds the saturation pressure. This unisolatable feedwater line volume is an additional source of high energy fluid that was assumed to be discharged out of the break and is conservatively maximized in this analysis.

Emergency Feedwater System

Generally, within the first 30 seconds following a steam line break, the Emergency Feedwater System will be initiated on any 1 of several protection system signals. Addition of emergency feedwater to the steam generators will increase the secondary mass available for release to containment as well as increase the heat transferred to the secondary fluid. The emergency feedwater flow rates were developed with the following conservatisms incorporated. The flow rates to the faulted loop were assumed to be conservatively high, maximizing the available mass for release to containment, and flow rates to the intact loops were assumed to be conservatively low, minimizing the available mass for secondary side heat removal. 99-01

The Emergency Feedwater System is designed to automatically isolate flow to the faulted steam generator and to limit flow to the faulted steam generator to less than 1000 gpm. With proper operation, emergency feedwater flow to the faulted generator is not more than 1000 gpm and generally automatically terminated within 48 seconds of detecting high flow to a depressurized SG. For conservatism, however, the emergency feedwater flow to the faulted generator was assumed to be 200 gpm until operator action is taken to terminate flow. 02-01

Steam Generator Fluid Mass

Conservative initial steam generator masses were used in all of the cases analyzed. The initial masses for the faulted steam generator were calculated as the mass corresponding to the programmed level + 5% of narrow range span. The initial masses for the intact steam generators were calculated as the mass corresponding to the programmed level -5% of narrow range span. Although the effect is not significant, this conservatively maximizes the secondary inventory available for release to containment, while minimizing the heat transfer capability of the intact steam generators.

Steam Generator Reverse Heat Transfer

Once steam line isolation is complete, those generators in the intact loops become sources of energy which can be transferred to the faulted steam generator. This energy transfer occurs via the primary coolant. As the primary plant cools, the temperature of the RCS coolant flowing in the steam generator tubes drops below the temperature of the secondary fluid in he intact steam generators resulting in energy being returned to the primary coolant. This energy is then available to be transferred to the steam generator with the broken steam line. The effects of reverse steam generator heat transfer are included in the results.

Break Flow Model

Piping discharge resistances were not included in the calculation of the releases from the steam line ruptures (Moody, Curve, Reference [42], for an fl/D = 0 was used). 02-01

6.2.1.3.10.3.3.3 Protection System

The protection systems available to mitigate the effects of a main steam line break accident inside containment are listed below along with the corresponding actuation signals modeled. The setpoints and delay times modeled in this analysis are provided in Tables 6.2-46d, 6.2-47a, and 6.2.-47b.

Reactor Trip

The following reactor trip signals were modeled in this analysis:

- Low pressurizer pressure
- Safety Injection System actuation
- Overpower ΔT
- High-1 containment pressure

Safety Injection

The following safety injection signals (SIS) were modeled in the analysis:

- Low pressurize pressure
- Low steam line pressure
- High-1 containment pressure

Steam Line Isolation

Main steam isolation valve actuation was modeled following receipt of any of the following:

- Low steam line pressure
- High-2 containment pressure

Feedwater Isolation

The following feedwater system isolation signals were modeled in this analysis:

- Low pressurizer pressure
- Low steam line pressure
- High-1 containment pressure

6.2.1.3.10.3.3.4 Safety System Failure Considerations

To minimize the number of cases to be analyzed, multiple failures were assumed in many, but not all, of the steam line break mass and energy release to containment cases analyzed. The following failures in the mass and energy release calculations are postulated which may significantly affect the containment results. The failure of a main steam isolation valve has been considered separately in the containment response analysis.

Failure of One Train of the Safety Injection System

A minimum Safety Injection System (SI) flow rate corresponding to the failure of one safety injection train was assumed in this analysis. A minimum SI flow is conservative since the reduced boron addition maximizes a return to power resulting from the RCS cooldown. The higher power generation increases heat transfer to the secondary side, maximizing both mass and energy flow out of the break. The delay time to achieve full SI flow was assumed to be 27 seconds.

Failure of Feedwater Isolation Valve (FWIV) in Faulted Loop

If the FWIV in the feedwater line to the faulted steam generator is assumed to fail in the open position, main feedwater would continue from the condensate pumps until isolation is provided via closure of the feedwater regulating valve and feedwater bypass control valve. Also, the unisolatable feedwater line volume will increase by the volume of the feedwater line between the isolation valve and the feedwater regulating valve and bypass control valve. This additional inventory would then be available to be released to containment.

Failure to Automatically Isolate Emergency Feedwater

The Emergency Feedwater System is designed to isolate emergency feedwater flow to the faulted steam generator via closure of control valves. Failure to automatically close a control valve to the faulted generator results in the addition of up to 1000 gpm of emergency feedwater flow to the faulted generator until operator action is taken to terminate the flow. The increased emergency feedwater flow conservatively provides additional steam generator inventory, which is available to be released to the Reactor Building. Furthermore, the increased emergency feedwater flow serves to maintain a higher pressure in the faulted generator, thus providing additional driving force for the mass and energy release.

Failure of Emergency Diesel

Failure of an emergency diesel to start results in the loss of one motor driven emergency feedwater pump and one train of the SI within the mass and energy calculations. Within the containment response analysis, one train of RB cooling would also be lost.

Failure of Electrical Channel

A loss of an electrical channel (CH-A) is postulated at event initiation. Within the mass and energy calculations, this results in loss of power to the feedwater control valves, feedwater isolation valves, and the main steam isolation valves; prevents the startup of one emergency diesel thus resulting in the loss of one motor driven emergency feedwater pump and one train of the SI; and prevents automatic isolation of emergency feedwater to the faulted SG. Within the containment response analysis, one train of RB cooling would also be lost.

Failure of Main Steam Line Isolation Valve (MSIV)

Failure of the faulted loop MSIV delays isolation of the intact loops and results in a larger unisolatable steam line volume. Prior to steam line isolation, all loops are feeding the break. Steam line isolation terminates fluid discharge from the intact steam generators while the faulted steam generator continues to blowdown. After isolation, reverse flow continues until the unisolatable volume is depleted. This failure is not explicitly considered within the mass and energy release, but is considered separately in the containment response analysis.

6.2.1.3.10.3.4 Results

Table 6.2-47a provides a summary of the mass and energy cases analyzed for use in the MSLB containment analysis. For each case, the initial NSSS power level, break size, single failure(s) assumed (if any), and entrainment assumptions are identified. The time of reactor trip (along with the actuation signal), feedwater isolation, and steam line isolation are provided for each case. Note that for cases that credit High-1 containment pressure or High-2 containment pressure the actuation times were determined iteratively from preliminary LOFTRAN and containment response analyses and confirmed in the final analysis.

Furthermore, in order to reduce the number of iterations during this process, the time to reach the High-1 setpoint was delayed until the High-2 setpoint was reached for the purpose of safeguards actuation. This delays actuation of the Safety Injection System and feedwater isolation, as well as reactor trip.

Tables of mass and energy release rates for the limiting containment pressure case and limiting containment temperature case are provided on Tables 6.2-48a and 6.2-48b, respectively.

6.2.1.3.11 Minimum Containment Pressure Analysis for Performance Capability Studies of Emergency Core Cooling System

The lower bound containment back pressure used for the best estimate large break LOCA analysis, presented in Section 15.4.1, is presented in Figure 15.4-1o. The containment back pressure is calculated using the methods and assumptions described in Section 15.4.1. Input parameters including the containment initial conditions, net free containment volume, passive heat sink materials thickness, and surface areas and starting time and number of containment cooling systems used in the analysis are presented in Tables 15.4-1a and 15.4-1b. Passive heat sink material thickness and surface area are shown in Table 6.2-60.

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6.2.1.4 <u>Testing and Inspection</u>

Containment leak rate testing is discussed in Section 6.2.6.

Testing and inservice inspection requirements for the Reactor Building are discussed in Section 3.8.1.7.

6.2.1.5 <u>Instrumentation Requirements</u>

The instrumentation to monitor containment conditions and perform the post accident monitoring function is discussed in Section 7.5. In addition, Reactor Building pressure instrumentation channels serve as inputs to both the Reactor Protection System and Engineered Safety Features Actuation System, discussed in Sections 7.2 and 7.3, respectively.

Qualification of safety related balance of plant instrumentation inside the Reactor Building is based upon the peak environmental conditions following the postulated design basis accident main steam line break and is discussed in Section 3.11.2.2.2. Qualification parameters are presents in Table 3.11-3.

Qualification of NSSS safety related instrumentation inside the Reactor Building is discussed in Section 3.11.2.2.1.

6.2.1.6 Materials

Safety-related coating systems are used for concrete and carbon steel surfaces. Both of these coating systems satisfy the quality assurance requirements of Reference [13]. In addition, the selected coating manufactures have documented test results certifying compliance with References [14] and [15] requirements for coating materials used inside containment.

The estimated quantity of unqualified paint inside containment is covered in calculation DC03190-025. RN 13-022

The amount of exposed organic material exposed directly to the containment atmosphere is considered negligible.

Metallic material for engineered safety features equipment are discussed in Sections 6.2.3.6, 6.2.4.5 and 6.2.5.6.

6.2.2 REACTOR BUILDING HEAT REMOVAL SYSTEMS

Post accident Reactor Building heat removal is provided by 2 systems acting in concert. One (1) system is the Reactor Building Spray System and the other is the Reactor Building Cooling System. Each of these systems also performs other functions. The Reactor Building Spray System removes iodine, which may be released from the core following a loss of coolant accident (LOCA) (see Section 6.2.3). The Reactor Building Cooling System removes heat from the Reactor Building during normal operation. This system also functions to remove particulate fission products from the post accident Reactor Building atmosphere as discussed in Section 6.5.1.

6.2.2.1 Design Bases

6.2.2.1.1 Post Accident Reactor Building Heat Removal Systems Design Bases

The design bases for the post accident Reactor Building heat removal systems are as follows:

1. The 2 systems, acting in concert, with the most disabling single failure as discussed in item 2 below, will limit the peak Reactor Building pressure to less than the Reactor Building design pressure of 57 psig. The heat sources considered in this analysis are discussed in Section 6.2.1.3.

2. The most disabling single failure is considered to be a loss of offsite power and failure of one onsite diesel generator to operate. This situation will result in the operation of one Reactor Building Spray Subsystem and one Reactor Building cooling unit. Other single failures will allow at least this amount of equipment to function.

3. Equipment inside the Reactor Building which is required to remove post accident heat is qualified to operate in the post accident Reactor Building environment.

4. Portions of these systems which are required to remove post accident heat are designed to withstand the most severe post accident pressure transient.

5. Portions of these systems which are required to remove post accident heat are qualified to experience the safe shutdown earthquake (SSE) without loss of function.

6. Portions of these systems which are required to remove post accident heat are protected from the dynamic effects of pipe rupture in accordance with the criteria presented in Section 3.6.

7. Reactor Building heat removal systems are designed for periodic testing and inspection.

8. Reactor Building heat removal systems are Seismic Category 1. The Reactor Building Spray System components are Safety Class 2a except the sodium hydroxide storage tank and related discharge lines which are Safety Class 2b. The Reactor Building Cooling System components are Safety Class 2b.

9. The Reactor Building heat removal systems are designed to satisfy containment depressurization requirements discussed in Section 6.2.1.

10. At strategically identified local high points, the Reactor Building Spray System is provided with indicating air-traps and vented orifices to ensure the Reactor Building Spray System remains void free (reference NRC GL 2008-01). RN 09-003

6.2.2.1.2 Reactor Building Cooling System Design Bases for Normal Operation

Design bases for the Reactor Building Cooling System during normal operation are as follows:

1. Operation of any 3 Reactor Building cooling units during normal power operation maintains average Reactor Building ambient air temperature at, or below, 120°F.

2. Operation of any 2 Reactor Building cooling units under loss of offsite power conditions is sufficient to remove the residual heat to bring the plant into a hot or 00-01
   cold shutdown condition.

3. Operation of any 2 Reactor Building cooling units during containment leak rate tests maintains an average Reactor Building ambient temperature with minimum stratification.

4. The system duct arrangement results in general Reactor Building air circulation and provides for discharge of cooled air in close proximity to suction inlets of 00-01
   various subsystems. These subsystems are described in Section 9.4.

5. System instrumentation provides for continuous monitoring of equipment and sounding of alarms to alert the operator to high vibration, smoke, high temperature, abnormally high coil condensate rate, or motor trip.

6.2.2.2 System Design

6.2.2.2.1 Reactor Building Spray System

This section discusses the design of the Reactor Building Spray System with respect to Reactor Building heat removal and its functions in common with fission product removal. Features that relate solely to fission product removal are described in Section 6.2.3.

The system diagram for the Reactor Building Spray System is provided as Figure 6.2-46. The system consists of 2 independent and redundant subsystems, each consisting of a pump, 2 Reactor Building sump isolation valves, 1 refueling water storage tank (RWST) isolation valve, 1 sodium hydroxide storage tank isolation valve, 1 discharge isolation valve, 3 ring headers, 165 spray nozzles, and connecting piping, manual maintenance valves, and check valves. The items shared by the 2 subsystems are the injection water source, the RWST; sodium hydroxide storage tank; and the full flow test line to the RWST. There are no shared active components. This design ensures single failure protection. Reactor Building Spray System component data are presented in Table 6.2-49. A single failure analysis is provided by Table 6.2-50.

6.2.2.2.1.1 System Design Features

Three (3) spray header and nozzle assemblies are provided for each spray subsystem. The headers are stainless steel ring manifolds that uniformly distribute the water mixture to each spray nozzle. Each spray subassembly has 165 spray nozzles and provides full coverage. Five (5) nozzles are located on the inner header and 80 on each of the middle and outer headers. The nozzles are attached to the headers with screwed nipple connectors. Figure 6.2-47 shows the spray header arrangement and nozzle orientation. The sizes and locations of headers inside the Reactor Building are listed Table 6.2-51.

The location of the Reactor Building Spray System ring headers within the Reactor Building dome and the position of the spray nozzles in the ring headers provide for 100% spray coverage of the operating floor with both spray loops operating. For the case of minimum engineered safety features, i.e., only one spray loop available, 93% of the Reactor Building at the operating floor level is covered by the spray.

During both normal and post accident plant operation, the Reactor Building cooling units and associated discharged ductwork supply cooling air flow to the unsprayed regions below the operating floor. This cooling air flow establishes a connective flow path whereby air or air/steam mixtures from the unsprayed regions and from inside the secondary shield walls are directed upward into the suction areas of the Reactor Building cooling units. Thus, the Reactor Building cooling units provide, for the normal or accident case, the required mixing and recirculation of Reactor Building atmosphere for the volumes between and lowest elevation inside the Reactor Building and the elevation of the Reactor Building cooling units (approximately 67 feet above the operating floor).

The spray nozzles are one piece, stainless steel units that deliver water at a nominal rate 15.2 gpm with a pressure differential of 40 psi across the nozzle. Flow is delivered in a hollow cone pattern with an included angle of approximately 60 degrees.

Orifices are located in each individual spray header and the common header supply line to compensate for the differences in elevation of the different headers and to balance Reactor Building Spray System characteristics with pump characteristics.

In the event of LOCA, water from the spray nozzles and water spilled through the break in the Reactor Coolant System is collected in either the A or the B Reactor Building recirculation sumps. Each of these two sumps has a suction line to one residual heat removal (RHR) pump and a suction line to one Reactor Building spray pump. Each recirculation sump is irregular in shape as shown in Figures 1.2-3 and 1.2-4. The overall plan dimensions of each sump are approximately 17 feet by 28 feet. Each of the recirculation sumps is surrounded by a 6 inch high curb. One side of Sump B abuts directly against the adjacent secondary compartment wall and does not have the 6 inch curb. The basement floor of the Reactor Building is at elevation 412'. The floor level of the recirculation sump area is at elevation 408'. There are four individual deep sump pits for the RHR and spray pump suctions – one RHR and one spray deep pit located in

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each of Sump A and Sump B. The deep pits are 4 foot by 4 foot in plan extending down to elevation 400'. The centerline of the suction pipe within each deep pit is at elevation 402'.

A schematic vertical section through a typical deep sump pit is shown in Figure 6.2-48. A removable welded stainless steel bar grating walkway is provided over the sump for a personnel access walkway and to support maintenance in the area. The grating is designed to remain in place during plant operation. The top of the grating walkway cover over the sumps is at elevation 412'-9". A large opening is provided through the grating to ensure adequate flow to the pumps through the opening for the postulated worst case assuming full blockage of the grating by debris transported to the sump following the postulated pipe break.

Each of the four deep sump pits is protected by a strainer assembly against the entry of potential types and quantities of debris generated as the result of hypothetical, postulated LOCA pipe break events. The RHR and Spray strainer modules are interconnected by a cross-duct to allow water to flow from one module to the other conservatively assuming one of the strainer modules becomes heavily blocked by postulated debris.

Each strainer assembly is composed of a single square module, the header box, equipped with 44 hollow fins, 11 on each of the 4 sides of the strainer header box. The fins are connected laterally to the approximately 4.75 foot high sides of the header box located directly over each sump pit. The fins are of varied length designed to fit within the available space in the sump. Each vertically oriented strainer fin consists of 18 gauge stainless steel sheet, perforated with 1/16 inch diameter holes. The performance of the strainer is enhanced by the extremely low approach velocity to the perforated fins of less than 0.1 inch/second. The area ratio of holes is about 41 percent and the surfaces of the fins are corrugated to increase their surface area. As the water level rises in the strainer during filling, air can escape through the fins and through the vent holes provided at the top of the strainer header box. This design ensures that there is no risk of air ingestion due to trapped air pockets during filling. RN 13-022

The total strainer surface areas provided are:

Sump A:	A RHR	-	1404 sq ft	Total Sump A:	2938 sq ft
	A Spray	-	1534 sq ft		
Sump B:	B RHR	-	1251 sq ft	Total Sump B:	2380 sq ft
	B Spray	-	1129 sq ft		

The design of both the Sump A and Sump B strainers includes a closed cross duct connecting the RHR and Spray header boxes within the sump. The cross duct design consists of 1/4 inch thick stiffened stainless steel plates. The cross duct provides a flow area approximately 5 inches high by 30 inches wide for flow between the interior of the two header boxes. The cross duct connection to the header boxes is located on the side and near the top of the header boxes. The cross duct is designed and fabricated to

the same criteria as the strainers. The cross duct serves to provide additional redundancy to the strainer design for both the Sump A and Sump B. For a postulated event where the fin strainers on either the RHR or Spray sides of the sump are assumed to be blocked by debris generated by the postulated LOCA pipe break event, the flow into the unblocked strainer header box provides sufficient recirculation flow through the cross duct to satisfy the NPSH requirements for the pumps on both the RHR and Spray sides of that sump.

The header boxes, strainer fins, and cross ducts are designed, fabricated, and installed in accordance with ASME Code and Seismic Category 1 requirements. Each strainer fin bank consisting of 11 fins is supported as an integral unit by bolting each fin to a horizontal truss located along the top and also along the bottom of each fin bank. The trusses are bolted to the header box assemblies. Each fin is also securely pinned at the bottom and bolted at the top to the header box assembly. Adjustable vertical supports for each fin bank are provided beneath the horizontal truss at the bottom of the fin bank to the sump floor at elevation 408'. The top of each header box consists of 3/8 inch stainless steel plate with stiffeners. A solid 3/8 inch thick hinged hatch cover plate for personnel access down into the deep pit is secured to the top of each header box by bolting all four sides of the personnel access hatch plate to the welded flange assembly on the header box top plate.

The bottom surface of each hatch cover plate is designed and fabricated to fit tightly with no gap when secured with the bolts against the continuous bearing bar located around the perimeter of the opening beneath the hatch cover plate. The continuous tight metal to metal closure with no gap precludes any potential debris entry without the need for any gasket material under the hinged cover plate. RN 13-022

Vortex formation was investigated at the controlling locations including at the engineered opening through the grating walkways over the sumps and also above the strainers. The controlling location is at the engineered opening through the grating where the cross section area is smaller and flow rate larger. The grating is assumed to be fully blocked by debris and the combined RHR and Spray streams flow through the engineered opening.

At the start of recirculation, the water level is assumed to be at the controlling minimum elevation of 414.6 feet corresponding to a small break LOCA. This is conservative in that the debris assumed to be blocking flow through the walkway grating could not be generated by a postulated small break LOCA rather only by postulated large break LOCA. The calculated Froude number corresponding to the flow rate and associated velocity through the engineered opening for this conservative case is 0.241. The critical submergence 15.3 inches maximum at the engineered opening is less than the minimum submergence of 22.2 inches to the engineered opening for the small break LOCA. Therefore, no air ingestion down into the sump through vortexing will occur.

Sump design features that prevent the generation of flow irregularities include:

1. Perforated fin and header design. As the water level rises in the strainer during filling, air can escape through the fins and also through 3/8 inch diameter vent holes located close to the top of the strainer header box. As the sump pit fills, a small air pocket may form in the top of the strainer header box. To allow the air pocket to escape, each strainer assembly is designed with 24 vent holes of 3/8 inch diameter close to the top of the strainer header box. Vent holes are covered by perforated sheet having 1/16 inch diameter holes. The pumps are activated well after the strainers are fully submerged and any air pocket within the strainer assembly has been displaced by water. As a result, ingestion of an air pocket into the pump intake is not considered to be a credible scenario. Air ingestion through the vent holes is similarly considered an incredible scenario due to the submergence of the strainers at the holes being much greater than the critical submergence at the holes. RN 13-022

2. The generation of vortices within the 4 foot by 4 foot plan deep sump pit standpipe at the inlet to the Reactor Building Spray System suction piping is prevented by the following:

   a. An established fluid velocity within the standpipe of less than 1 foot per second.

   b. A minimum submergence from the suction piping inlet to the minimum water level at the start of recirculation of at least 12.6 feet.

3. Generator of flow disturbances at the suction piping inlet is minimized by the inclusion of a 24 inch by 12 inch reducer which provides for a gradual change in fluid velocity between the standpipe and suction piping.

4. The location of the Reactor Building spray pumps relative to the top of the recirculation sumps (35 feet, minimum elevation difference) provides the fluid system conditions that ensure dissipation of any flow irregularities that might be generated within the sumps. These positive fluid system conditions are as follows:

   a. A minimum elevation head of 35 feet of water at the pump suction nozzles.

   b. The significant lengths of piping (75 and 100 feet) from the sump outlets to the pump suction nozzles.

Design of the RWST includes an internal vortex baffle at the entrance to the 20 inch suction connection through which cooling water is supplied to the RHR, safety injection Reactor Building spray pumps during operation in the safety injection mode. This internal baffle consists of a 4 foot by 4 foot by 4 foot, stainless steel box with open sides, which directs cooling water flow into the 20 inch suction nozzle from 2 sides

along the horizontal centerline of the baffle. This designed control of the flow path into the 20 inch suction nozzle prevents formation of vortices even under conditions of low fluid level in the RWST.

The recirculation water enters the strainer fins through the perforated steel sheet (1/16 inch diameter holes) and flows horizontally within the hollow fin corrugations into the header box. Due to the extremely low approach velocity to the perforated fins of less than 0.1 inches/second it is expected that most of the entrained particulate debris falls out of the flow stream to the sump floor. The top of the sump header box consists of stiffened 3/8 inch plate with a 3/8 inch plate single leaf bolted personnel hatch cover plate to permit entry into the deep sumps for inspection. All material in the sump area, including liner plate, strainer assembly, deep pit access ladder, etc., is stainless steel to prevent corrosion degradation. RN 13-022

Debris which washes down to the sump and passes through the strainer is evaluated for both blockage and erosion of downstream systems and components. The size of the debris is limited by the 1/16 inch round holes in the perforated plate strainer. Systems and component downstream effects are analyzed based on the methodology in WCAP-16406, Rev. 1 (Reference [48]). The analysis was completed as part of the response for Generic Letter 2004-02.

The potential for clogging the sump screens with debris is reduced by minimizing the use of non-metallic, high temperature insulation and by not using displaceable components, such as sand plugs, to provide vent area.

For a discussion of the potential for sump screen clogging, refer to Section 6.3.2.6.1. 98-01

Compliance with Regulatory Guide 1.82 is discussed in Appendix 3A.

High temperature insulation used inside the Reactor Building is generally stainless steel, reflective type insulation. Insulation encased in stainless steel is used on the sloping portion of the 6 reactor vessel nozzles and between the pipe restraint collars and Reactor Coolant System piping in the vicinity of the primary shield.

There are 2 independent pipe lines connecting the sump to the suction of the spray pumps. These lines are independent of the 2 lines for the Residual Heat Removal System . The lines from the sump can be isolated by remote, manually controlled, motor operated valves. Each line is enclosed in a concentric guard pipe which is 02-01 connected to a protective chamber completely surrounding the isolation valve closest to the Reactor Building.

The guard pipe and protective chamber collect any leakage due to a passive failure of the sump line up to and including the first isolation valve and thus prevent draining the sump to the Auxiliary Building. Passive failures beyond the first isolation valve can be isolated by the first isolation valve. The guard pipes and protective chambers are designed to a pressure of 57 psig at a temperature of 300°F. 98-01

All outdoor system piping is insulated and has heat tracing. The outdoor sodium hydroxide storage tank is neither insulated nor heat traced. This tank contains a 20 to 22% sodium hydroxide solution which is liquid at temperatures above -13°F.

6.2.2.2.1.2. System Operation

During normal plant operation, the Spray System is in standby condition with the suction aligned to receive water from the RWST. Operation of the system is initiated automatically following a LOCA or MSLB by signals from the Engineered Safety Features (ESF) Actuation System when the Reactor Building pressure increases to the actuation set point. The Spray System can also be started manually from the control room by the operator. A description of the ESF actuation signals is presented in Section 7.3.

If there is a spray actuation signal, the Reactor Building spray pumps are loaded onto the onsite diesel generators following a loss of offsite power (see Chapter 8). The total time for RB Spray water to start being delivered through the spray nozzles to the RB atmosphere is 47 seconds following LOCA initiation. This includes a 10 second start time for EDGs, 5 seconds for the spray pump to reach full capacity (during this 5 second 98-01
pump acceleration time, no filling of lines is assumed), and 32 seconds to fill the RB spray line up to the nozzles. For conservatism, it is assumed in the Reactor Building pressure analysis that spray starts 52 seconds after the LOCA occurs.

Each spray subsystem is started by separate ESF containment isolation Phase A and spray actuation signals. Normally, both subsystems operate; however, they are independent and can operate individually. The design basis for Reactor Building heat removal is one spray subsystem operating in conjunction with one Reactor Building cooling unit (with one RHR pump and one charging pump providing emergency core cooling water).

During the injection phase following a LOCA, the spray pumps receive fluid from the RWST and sodium hydroxide storage tank. The solutions are carried through the pump discharge lines to the spray headers and are then sprayed into the Reactor Building atmosphere through the spray nozzles. The spray is collected in the bottom of the Reactor Building with water from ECCS and the Reactor Coolant System in the Reactor Building recirculation sumps. During the recirculation phase, the spray pumps may take suction from the sumps, in the Reactor Building.

The spray droplets absorb heat as they fall through the steam-air atmosphere in the Reactor Building, thereby reducing Reactor Building atmosphere temperature.

A nominal spray flow of 2500 gpm is provided by each spray subsystem. When the system is receiving water from the RWST (injection phase) the flow nominally includes 50 gpm of a 20 to 22 weight percent solution of sodium hydroxide. This flow (nominal 2500 gpm) is delivered into the Reactor Building through 165 nozzles at a rate of 15.2

gpm per nozzle. The nozzles are arranged on ring headers in the top of the Reactor Building such that the spray from each subsystem covers essentially the same volume.

There are 4 significant operating combinations of Reactor Building spray pumps and Emergency Core Cooling System (ECCS) pumps:

1. Reactor Building Spray System design case: 1 Reactor Building spray pump; 1 residual heat removal (RHR) pump; and 1 charging pump operating (loss of offsite power and malfunction of one emergency diesel generator). 02-01

2. Normal case: 2 Reactor Building spray pumps; 2 RHR pumps; and 2 charging pumps (all ECCS pumps) operating. 02-01

3. One (1) Reactor Building spray pump inoperable, all ECCS pumps operating. 02-01

4. One (1) RHR pump operating, 2 charging pumps and 2 Reactor Building spray pumps operating. 02-01

The sodium hydroxide solution is gravity fed into the Reactor Building spray pump suction lines. The system is so designed that the sodium hydroxide storage tank empties at approximately the same time as does the refueling water storage tank (RWST) (e.g., the rate of change, proportional to the initial level, of the 2 liquid levels is the same) for Cases 1 and 2, above. This provides nominally constant sodium hydroxide flow rates for these 2 cases. The design flow rates are approximately 50 gpm to the single operating spray pump for Case 1 and approximately 45 gpm/pump for Case 2.

In Case 3, the initial flow of sodium hydroxide to the single operating Reactor Building spray pump is approximately 55 gpm. However, at this rate of flow from the sodium hydroxide storage tank, the liquid level lags the drop in liquid level in the RWST, since the equilibrium sodium hydroxide flow rate for this case is 73 gpm. (The equilibrium flow rate is the flow rate at which the sodium hydroxide storage tank level drops at the same proportional rate as does the RWST liquid level. This is the stable operating condition for the Reactor Building Spray System. The sodium hydroxide flow rate is unstable under any other condition and the sodium hydroxide flow rate gradually changes toward the equilibrium rate.) As a result, pressure head differential begins building up on the sodium hydroxide side of the Reactor Building Spray System at the junction of the sodium hydroxide and RWST lines. However, as the total pressure head on the sodium hydroxide side must balance the total pressure head on the RWST side of the junction, the sodium hydroxide flow rate increases as necessary to create an additional flow loss to offset the differential pressure head. The flow rate continues to increase as the system adjusts toward a stable operating condition at the equilibrium flow rate.

The rate at which the sodium hydroxide flow increases depends upon the flow impedance of the sodium hydroxide circuit from the sodium hydroxide storage tank to

the junction with the line from the RWST. If the impedance is very low, the flow rate increases very rapidly to the equilibrium rate. An increase in the impedance slows the rate of change of flow rate. Thus, the rate at which the sodium hydroxide flow rate increases from the initial rate can be controlled through discrete design of the impedance characteristics of the sodium hydroxide circuit. Flow characteristics of the circuit from the RWST must also be properly designed. Impedance elements must be limited to characteristics and locations that do not reduce the minimum available net positive suction head at the pump to less than the minimum required, or to zero, anywhere in the circuit from the RWST to the Reactor Building spray pumps.

Properly sized pressure loss inducing orifices are located in the common and individual sodium hydroxide lines (in conjunction with orifices in the individual spray pump lines from the RWST) to create sufficient and properly distributed flow impedance such that, in Case 3, flow increases to a maximum rate of 65 gpm at the end of the injection phase ("lo-lo" level in the RWST).

In the fourth case, the initial sodium hydroxide flow rate is approximately 42 gpm per Reactor Building spray pump. The equilibrium flow rate for this case is 34.5 gpm per pump. In this case, the liquid level in the sodium hydroxide storage tank drops at an increased rate compared to the liquid level in the RWST. The result is the reverse of the situation in Case 3. The sodium hydroxide flow rate decreases from the initial rate toward the equilibrium rate as the injection phase continues.

Upon receipt of the RWST "lo-lo" level signal in conjunction with the safety injection signal, the Reactor Building recirculation sump isolation valves (3004A & B and 3005A & B) automatically open. (See Section 7.6.11 for a discussion of the switch over logic.)

A design evaluation of the Reactor Building Spray System has been performed to track the fluid levels in the refueling water storage tank and sodium hydroxide tank during emergency safeguards operation. Refer to Section 6.2.2.3.1.4.

Sump water temperature data shown by Figures 6.2-6 and 6.2-7 were obtained from results of the transient analyses performed for both the main steam line break and the LOCA. The conservative assumptions applicable to the generation of this data are listed in Section 6.2.1.3.2. One of these assumptions, used in determining the temperature data illustrated by Figure 6.2-7, is that the Reactor Building Spray System is operated for up to 40 days to provide Reactor Building cooling following the design basis LOCA. RN 03-003

During post recirculation operations, the Reactor Building recirculation sump water can be sampled with the normal and Post-Accident Sampling System to monitor the pH. If additional sodium hydroxide is required, the sodium hydroxide feed line valve(s) from the sodium hydroxide storage tank is (are) opened and the sodium hydroxide solution is gravity fed into the Reactor Building spray pump suction line(s). The sodium hydroxide solution will flow into the pump suction line only when the Reactor Building pressure is less than or equal to atmospheric pressure.

A flow test line connecting spray pump discharge lines to the RWST provides a return loop for use in conducting periodic operating tests.

6.2.2.2.2 Reactor Building Cooling System

The system diagram for the Reactor Building Cooling System is provided as Figure 6.2-49. Design and performance data are included on this figure. Plan and elevation drawings of the system ductwork are provided as Figures 6.2-50 and 6.2-51. A single failure analysis is provided by Table 6.2-52.

6.2.2.2.2.1 System Design Features

The system consists of 4 plenums arranged to discharge filtered and cooled air into a common air supply main. The cooling air is further distributed from the supply main to the upper portions of the Reactor Building through supply air outlets in the supply main and to the lower elevations of the Reactor Building through vertical supply ducts connected to the supply main. Non-engineered safety features systems are provided for normal ventilation, cooling and filtering of the Reactor Building atmosphere and for transfer of heat to the Industrial Cooling Water System. These systems are described in Section 9.4.

Each plenum includes moisture separators, high efficiency particulate air (HEPA) filters, filter bypass opening and dampers, cooling coils and an axial flow fan driven by separate high speed and slow speed motors. The moisture separators and HEPA filters are arranged in vertical banks in the plenum. Air enters the plenum through vertical openings in two of the plenum walls or through a bypass opening in the plenum roof if the bypass damper is open. Air entering through the bypass opening is not filtered and passes directly to the cooling coils.

The Reactor Building cooling unit fans operate at high speed during normal periods and at slow speeds during post LOCA periods and during Reactor Building leak rate testing. The units are serviced by cooling water from the Industrial Cooling System during normal periods and by service water during post LOCA or loss of offsite power conditions. For normal operation, 3 out of 4 fans operate. For LOCA, one fan in each train operates.

The plenums are arranged along the perimeter of the Reactor Building with the base of each at elevation 514’. Two (2) of the Reactor Building cooling units are supplied with electrical power from the A channel and the other two from the B channel of the Class 1E electric system.

A single supply and return cooling water main serves the two Reactor Building cooling units powered from the A channel of the Class 1E system. A separate, single supply and return main serves the two units powered from the B channel. These two sets of cooling water mains are separated from each other.

Each Reactor Building cooling unit is provided with lighted service platforms inside the plenums with access doors for inspection, maintenance and replacement of filters, motor and fans. Information on monitoring and control of these units is presented in section 6.2.2.5.2.

The Reactor Building cooling units are designed and constructed in accordance with the following codes, standards, and guides:

1. Title 10, Code of Federal Regulation, Part 50, Appendices A and B.

2. "Health and Safety Information," Atomic Energy Commission, Issue No. 306, March 31, 1971. 99-01

3. "Standard Test Code For Air Moving Devices," Air Moving and Conditioning Association, Bulletin No. 210.

4. Boiler and Pressure Vessel Code, Sections III and IX, American Society of Mechanical Engineers, 1973.

5. "Standard Test Code for Sound Ratings, Air Moving Devices," Air Moving and Conditioning Association, Bulletin No. 300.

6. "Structural Welding Code," American Welding Society, AWS D1.1-72.

7. Military Specifications MIL-F-51079 and MIL-STD-282.

8. "Type Test of Continuous Duty Class 1 Motors Installed Inside the Containment of Nuclear Power Generating Stations," (Proposed Standard), Institute of Electrical and Electronic Engineers, IEEE 334-1971.

9. "High Efficiency Air Filter Units," Underwriters Laboratories, Bulletin UL-586.

10. "Air Filter Units," Underwriters Laboratories, Bulletin UL-900.

11. Regulatory guide 1.52 "Design, Testing and Maintenance Criteria for Atmospheric Cleanup System Air Filtration and Absorption Units of Light Water Cooled Nuclear Power Plants" (see Appendix 3A).

6.2.2.2.2.2 System Operation

The automatic change in operating mode from normal to emergency involves changing from motors operating at high speed to motors operating at slow speed, changing the unit cooling water from that supplied by the Industrial Cooling Water System to that from the Service Water System, and automatically closing the Reactor Building HEPA filter bypass damper. The active RBCU discharge valves will automatically align so that 2 of the 4 valves, one per train, will open when their associated RBCU fans are selected to run; the other two valves will close when their RBCU fans are not selected to run. The

units, when operating in the normal mode, are tripped upon the receipt of a safety injection or loss of offsite power signal and are then automatically started at slow speed in accordance with the ESF actuation system and the ESF loading sequence of the related emergency diesel generator. A description of the ESF loading sequence is provided in Section 7.3.

In the following discussion, the times given for the occurrence of events are referenced to the time at which the ESF loading sequence gives a start signal to the first electrical load. See Section 7.3 for information on the relationship between this time and the time of occurrence of the input signal which initiates the ESF loading sequence.

The slow speed motors are energized at approximately 25 seconds after the referenced time. An additional 8 seconds are required after energizing the slow speed motor for air flow to reach operating values. Thus, the total "air-side" switchover from normal to emergency operation is completed approximately 33 seconds after the reference time. The safety injection or loss of offsite power signals also initiate closing of the industrial cooling water supply and return valves connected to piping serving the plenum cooling coils. Following receipt of either of these signals, the ESF loading sequence energizes the service water pumps 10 seconds after the reference time and energizes the service water booster pumps 35 seconds after the reference time. The service water booster pumps require approximately 5 seconds after the motor is energized to reach the operating value for water flow. The total "water-side" valve switchover from normal to emergency operation is completed approximately 86.5 seconds after the reference time.

The Reactor Building cooling units can be manually operated from the control room at either high or slow speed. Control room selector switches also determine which one of the 2 A and which 1 of the 2 B electrical power channel units starts in response to an ESF loading sequence signal. The plenum unit HEPA filter bypass damper is in the open position during normal operation and is automatically closed upon receipt of a safety injection signal. Additionally, this damper can be manually opened or closed from the control room during the normal mode of operation. The damper can also be manually closed from the control room during operation in the emergency mode. The damper fails closed upon loss of electric power. The effects of the HEPA filter upon removal of fission products in the Reactor Building are discussed in Chapter 15.

6.2.2.3 Design Evaluation

An evaluation of the Reactor Building heat removal systems relative to their effect upon the Reactor Building pressure transient analysis is presented in Section 6.2.1.

6.2.2.3.1 Reactor Building Spray System

6.2.2.3.1.1 Net Positive Suction Head to Reactor Building Spray Pumps

It is important that the net positive suction head (NPSH) at the suction of the Reactor Building spray pumps be maintained at or above the minimum required value to assure the long term availability of the pumps. The NPSH available is a function of Reactor Building pressure, water temperature, static head of water, friction losses in the pump suction piping and pressure drop across the sump strainer as follows:

$$NPSH_a = P_c - P_s + \Delta H_z - \Delta H_f - \Delta H_s$$

Where:

$NPSH_a$ = NPSH available at the pump suction.

P_c = Reactor Building pressure.

P_s = Saturation pressure of the pumped water.

ΔH_z = Elevation pressure due to the height of water above the pump nozzle.

ΔH_f = Pressure loss due to friction in the suction line. RN 13-022

ΔH_s = Pressure loss across the sump strainer.

The pressure in the Reactor Building (P_c) will always equal or exceed the vapor pressure of the water in the sump because water will evaporate from the sump to increase the vapor pressure of the atmosphere if P_c starts to decrease below P_s. Generally, the partial pressure of the air in the atmosphere will maintain P_c greater than P_s. For conservatism, no containment overpressure is assumed, i.e., $P_c = P_s$. This is consistent with the guidance provided in Regulatory Guide 1.1. The elevation of the pump shaft centerline is 377 feet. The minimum sump water elevation in the Reactor Building during recirculation is 414.6 feet for small break LOCA with water holdup in the reactor vessel cavity. Therefore, the minimum elevation head is:

$$(414.6 \text{ feet} - 377 \text{ feet}) = 37.6 \text{ feet}$$

Friction losses are calculated using the runout flow (3300 gpm) rather than the design flow (2500 gpm). The minimum required NPSH at 3300 gpm is 17 feet based on vendor data. Head friction losses through the piping, fittings and valves have been determined

and yield a calculated head loss of 5.96 feet for Pump A and 6.16 feet for Pump B (Reference [44]). The pressure drop through the sump strainers was determined experimentally by small scale testing with chemical effects from aluminum corrosion precipitants. The maximum pressure drop through the strainer is 4.09 psid at 70°F for the maximum debris loading case.

A summary of the results is as follows:

	NPSH Available (ft)	NPSH Required (ft)	Margin (ft)
Pump A	22.1	17	5.1
Pump B	21.9	17	4.9

Design of the recirculation sumps prevents formation of vortices without the use of vortex suppressors. Plan and elevation drawings of the sump and piping to the Reactor Building spray pumps are provided by Figures 6.2-82 through 6.2-84. Preoperational testing established suction paths through the sump lines to the Reactor Building spray pumps for the purpose of determining the actual piping friction losses at full flow. The as-tested head loss for each train is 4.6 feet (Reference [45]).

The potential effects of Reactor Building spray water and reactor blowdown liquid becoming trapped (also called holdup volume) in areas of the Reactor Building and unavailable for post accident recirculation are considered in sump level analysis. Spray flow entering the refueling cavity drains through an 8 inch line to elevation 412' for recirculation. (This drain line is temporarily flanged off during refueling and contains no valves.) Spray flow entering the southern portion of the refueling cavity may drain into the reactor cavity through the reactor cavity seal ring. This is a small gap, so water accumulation in the reactor cavity would be very slow. The spray water may also drain through the access openings for inservice inspection of the reactor vessel "safe-end-welds". Most of the water flowing into these openings will drain out through pipe sleeves due to a unique baffle arrangement around each reactor vessel nozzle which prevents drainage to the reactor cavity. Spray flow landing on the 463 ft operating deck drains to the 412' elevation through the equipment hatch and two stair wells. There is no curbing around the equipment hatch.

RN 13-022

Leakage from breaks in the large Reactor Coolant System piping, including a "safe-end-weld" break, will drain to the recirculation sumps since leakage into the reactor cavity is prevented by the baffle arrangement previously mentioned. Flow exits the loop compartments by one of the three openings in the secondary shield wall and travels around the containment annulus to the recirculation sumps. After the water from the reactor coolant blowdown and Reactor Building spray collects above the Reactor Building floor to an elevation of 415 feet, it can enter the incore instrument pit through a ventilation opening in the shield wall that is part of the incore instrument chase. The volume of water collected in the incore instrument pit will vary, depending upon the height of the recirculation water above the Reactor Building floor.

The sump water elevation for the NPSH calculations assumed water fills the reactor cavity to the 430.75 ft elevation where it will spill out the loop openings in the primary shield wall. This is a conservative assumption which bounds any possible hold up concerns in the reactor vessel cavity. Holdup water volumes are also considered for droplet accumulation on vertical surfaces, steam in the containment free volume, water accumulation on the 463 ft and 436 ft floors, as well as transit water volumes in the spray and spill flows. RN 13-022

6.2.2.3.1.2 Spray Nozzles

Reactor Building spray nozzles are discussed in Section 6.2.3.2.

6.2.2.3.1.3 Leakage Considerations

Following a postulated LOCA the Residual Heat Removal System and the Reactor Building Spray System are operational. During the recirculation phase, leakage could occur through component flanges, pump shaft seals, valve flanges, and valve stems. Although leakage rates have been assumed for these sources, maintenance and periodic system testing will preclude all but a small percentage of the assumed leakage. Values for the expected maximum leakage through such components are listed in Tables 6.2-52b for Reactor Building Spray System components and 6.3-4 for Residual Heat Removal System components. The radiological assessment of this operational leakage is discussed in Section 15.4.1.3. Radioactivity release rates are listed in Table 15.4-10.

6.2.2.3.1.4 Drawdown Analysis

A design evaluation analysis has been performed to track fluid levels in the refueling water storage tank (RWST) and sodium hydroxide storage tank (SHST) during each of the 4 operating combinations outlined in Section 6.2.2.2.1.2. The analysis involves modeling all pumps and associated piping that take suction from the RWST and SHST as input to the PIPF computer program. This computer program then performs a steady state pressure balance on the piping network. The PIPF Computer Code [29] was approved by the NRC on March 29, 1978, for use in hydraulic analyses, providing 3 criteria were satisfied:

1. Fluid is low temperature and incompressible.

2. Energy conservation can be omitted.

3. For transient calculations, flow rate changes are slow and smooth.

In this analysis, the fluid is water at 70°F in solution with boric acid or sodium hydroxide. Energy is neither added to nor removed from the fluid, except through the pumps. The pumps are modeled to maintain a uniform flow rate throughout the transient. Models of

the Reactor Building Spray System and Emergency Core Cooling System for each of the 4 operating combinations are shown by Figures 6.2-51a through 6.2-51d.

The drawdown transient is modeled by determining the steady state flow from the tanks at the initial levels. The volume of fluid drawn from the tanks during a 5 minute interval (or smaller, if desired) is calculated from the above flow. Tank levels are then adjusted and another steady state balance is performed. The procedure is repeated until the RWST reaches a predetermined level, at which time drawdown is terminated.

Since sodium hydroxide concentrations range between 20 and 22% by weight and initial tank levels for both the RWST and SHST range between min-max values, 2 separate modes of analysis were considered. First, a minimum pH mode was analyzed in which the sodium hydroxide fluid concentration and initial tank level were at minimum values, while the RWST was a maximum value. Secondly, a maximum pH mode, in which the initial conditions were reversed, was considered. All 4 operating combinations were analyzed for each pH mode.

Input parameters used in the analysis are given in Tables 6.2-52c through 6.2-52f. Table 6.2-52c is a compilation of system branch data. Table 6.2-52d lists pump head data. Table 6.2-52e gives tank parameters. Table 6.2-52f lists specific volume and viscosity for the minimum and maximum concentrations of sodium hydroxide solution.

Drawdown transients for each of the 4 cases at both minimum and maximum sodium hydroxide initial conditions are plotted on Figures 6.2-51e through 6.2-51l. Summary tables for each initial sodium hydroxide condition are provided as Tables 6.2-52g and 6.2-52h. The analysis shows that the RWST and SHST drawdown uniformly.

The pH values of the spray solution, as a function of time during the injection and recirculation phases, for the 4 operating combinations of Reactor Building spray pumps and ECCS pumps are shown by Figures 6.2-51m through 6.2-51bb. Supplemental analyses have been performed to support a larger RCS volume, due to steam generator replacement, with RCS boron levels ranging from 0-2000 ppm. These analyses show a small increase (0.2 to 0.3) in the sump and spray pH values during the recirculation phase of the accident. 98-01

6.2.2.3.1.5 Preoperational Water Test

The model used to predict the results of chemical drawdown outlined in Section 6.2.2.3.1.4 has been verified by comparison to a site preoperational water test. The test involved filling the refueling water storage tank and sodium hydroxide storage tank with grade A water and conducting one complete drawdown using each of the 4 operating combinations of the Reactor Building spray pumps and ECCS pumps for a portion of the transient. A comparison of the site test results to the analytical predictions is shown for each pump configuration in Figures 6.2-51cc, dd, ee, and ff.

The good correlation between site test results and analytical prediction confirms the validity of the model used in the chemical drawdown analysis of Section 6.2.2.3.1.4.

6.2.2.3.2 Reactor Building Cooling System

The Reactor Building cooling units are sufficiently separated to confine the effects of pipe whip, jet impingement of external missiles to one unit. The 2 units powered from the A channel of the Class 1E electric system are located on the opposite side of the Reactor Building from the 2 units supplied from the B channel. Also the cooling water supply and return mains to these units are physically separated as is the A and B channel wiring.

Each unit can operate independently of the others and the discharge from each unit is isolated from the common air supply main by gravity operated dampers.

Reactor Building Cooling System components that must remain intact following a LOCA include: the 4 plenums and all internal components, plenum discharge ducts, common air supply main, and the 6 vertical supply ducts from the common air supply main to the lower elevation of the Reactor Building.

Design considerations employed to assure that components noted above will remain intact following a LOCA include:

1. Qualification of the plenum and components as described in American Air Filter (AAF) Topical Report, AAF-TR-7101 [16].

2. Testing and qualification of the fan motor assembly in accordance with IEEE-334 [17] as indicated in Reliance Electric Test Report NUC-9 [18]. 98-01 02-01

3. Testing and qualification of the HEPA filter bypass damper.

4. A seismic analysis for the duct systems and HEPA filter bypass dampers.

5. Design of the Reactor Building cooling unit plenum housing suitable for a 2 pound pressure differential and provision of plenum housing relief valves so that this differential is not exceeded during the LOCA pressure transient.

6. Design of duct systems suitable for a 2 pound pressure differential and provision of duct relief valves so that this differential is not exceeded during the LOCA pressure transient.

6.2.2.3.3 Heat Removal Rate

The Reactor Building Spray System utilizes 165 spray nozzles on each loop. The spray nozzles are Spray Engineering Company (SPRACO) Model 1713A and have a hollow cone spray pattern with an average droplet size of 230 microns at a differential pressure of 40 psi across the nozzles.

At a discharge flow rate of 2500 gpm, each subsystem absorbs 2 x 10^8 BTU/hr at a Reactor Building atmosphere temperature of 260°F. The spray droplets in this analysis reach thermal equilibrium with the Reactor Building atmospheres in less than 0.5 seconds.

The Reactor Building cooling units reject heat to the Service Water System following a LOCA. Four (4) Reactor Building cooling units are furnished, 2 piped and wired to the A loop and A channel, respectively, and the other 2 to the B loop and B channel, respectively. The system design anticipates simultaneous use of one Reactor Building cooling unit from each loop. The design heat removal rate for each unit following a LOCA is as follows: 125 x 10^6 BTU/hr with saturated air entering the cooling unit at 283°F, 69.39 psia, density of 0.182 lb/ft^3, with 2000 gpm of cooling service water entering at 95°F. In order to allow for degraded performance, containment pressure and temperature analysis for LOCA transients use the more conservative performance curve corresponding to a tubeside fouling factor of 0.0014, as shown on Figure 6.2-15. Analysis of MSLB transients use the design performance curve on Figure 6.2-15 times 60%. RN 03-003

6.2.2.4 <u>Tests and Inspections</u>

6.2.2.4.1 Reactor Building Spray System

In addition to the checkout tests performed prior to initial plant startup and following maintenance, system and component tests are performed periodically in accordance with the ASME Code, prescribed under 10CFR50.55a, to verify continued operability of the Reactor Building Spray System. Also, those elements of the Reactor Building Spray System that perform a containment isolation function must be leak tested periodically in accordance with the requirements of Appendix J to 10CFR50. RN 04-012

Prior to plant startup, flow tests were performed to verify that the various orifices are properly sized to provide the correct mixture of sodium hydroxide solution and RWST water during operation of the Reactor Building Spray System.

6.2.2.4.1.1 System Actuation Tests

Reactor Building Spray System actuation tests verify that the system will operate properly and as designed in the event of an accident. These tests are performed as a part of an overall test of the ESF actuation system. This procedure involves testing of analog and logic circuitry and the actuation of the ESF components (except those whose actuation is not acceptable while the plant is in operation).

The procedures followed in testing actuation of the Reactor Building Spray System components prevent introduction of spray into the Reactor Building or introduction of sodium hydroxide solution into the pump suction lines. Actuation of the header isolation valve and the spray pump in a spray subsystem are tested separately. By following this

procedure during testing, pump operation with an open header isolation valve will not occur.

Actuation of the valves in the line from the sodium hydroxide storage tank is blocked during testing when the plant is operating. These valves are tested only during shutdown and then only with the manual shutoff valve in the sodium hydroxide storage tank outlet line closed.

The test line is opened (only to the subsystem being tested) for testing of the pump. This allows the pump to be operated without spray flow into the Reactor Building. In conjunction with testing the actuation of the pump, an evaluation of the pump performance is made to ensure that performance is satisfactory. The opening signal to the valve (normally open) in the suction line from the RWST is tested simultaneously with the actuation of the Reactor Building spray pumps by a simulated High-3 Reactor Building pressure signal. The RWST discharge valves are not closed except during testing of the valves themselves or testing of the sump isolation valves. Testing pump actuation with the RWST discharge valves closed is not considered necessary.

Testing is performed on one spray subsystem at a time. The other subsystem remains in its normal condition for plant operation.

6.2.2.4.1.2 Component Tests

Sump isolation valves are tested by manual operator action with the shutoff valves in the suction lines from the RWST closed. The sump isolation valves are tested in one line at a time. One valve in the line is actuated open and closed, followed by actuation of the other valve in that line. One valve in the line is fully closed at all times. The response of the valves is monitored and evaluated to verify proper operation. The header isolation valves and RWST discharge valves are tested with the Reactor Building spray pumps off.

Spray nozzles are checked by injecting low pressure air or smoke into the lines to the spray headers through air test connections. The spray isolation valves are closed during this test. By visual observation of the smoke or of telltales (such as balloons or tufts), it is verified that the nozzles are not restricted.

6.2.2.4.1.3 Leak Tests of Containment Isolation Elements

Containment isolation is provided on each pump discharge line by a check valve located inside the Reactor Building and a motor operated gate valve located outside. Isolation is provided on each suction line from the Reactor Building sump by a motor operated gate valve located outside the Reactor Building combined with a concentric guard pipe and protective chamber enclosing the line from the sump to and including the gate valve. Test connections are provided for leak testing these isolation elements.

To leak test the check valves, it is necessary to replace the system balance orifice in the pump discharge line, located on the spray header side of the check valve, with a blank orifice plate. The orifice blank is provided with the orifice as a one piece unit. This provides a direct visual means of verifying that the orifice blank has been removed and the orifice replaced.

6.2.2.4.2 Reactor Building Cooling System

Various components of the Reactor Building cooling unit plenums have been tested and qualified for emergency conditions as reported in AAF-TR-7101[16], approved by the USAEC, DRL in November, 1972. The components tested included moisture separators, cooling coils, HEPA filter banks, and plenum housing relief valves. The fan and motor assembly has been tested and qualified by the Joy Manufacturing Company in a simulated environmental test in accordance with the procedure outlined in IEEE-334[17].

6.2.2.4.2.1 Preoperational and Periodic Tests

Prior to plant startup, check and balance tests are performed on the "air-side" and "water-side" of the Reactor Building Cooling System to adjust and verify that system components can develop the required design air and water flow rates. In addition to the tests performed prior to initial plant startup, system and component tests are performed periodically to verify continued operability of the Reactor Building cooling units in both normal and emergency modes.

6.2.2.4.2.2 System Actuation Tests

Reactor Building Cooling System actuation tests verify that the operating components of the system shift to the emergency mode. Actuation tests are performed to verify the following "air-side" and "water-side" functions:

1. "Air-Side" Actuation Tests

   a. Tripping of the Reactor Building cooling unit while operating at normal speed upon receipt of safety injection or loss of offsite power signals.

   b. Automatic start at low speed in accordance with the ESF actuation system.

   c. Closing of the bypass damper following receipt of safety injection signal.

2. "Water-Side" Actuation Tests

   a. Closing of the industrial cooling supply and return valves upon receipt of safety injection or loss of offsite power signals.

   b. Automatic starting of the service water pumps and service water booster pumps in accordance with the ESF actuation system.

   c. Automatic opening of the service water supply XVG-3106, 3108 and return XVB-3107 valves as the service water booster pumps start. RN 08-008

   d. Automatic opening of one of the service water Reactor Building cooling unit discharge XVG-3109 valves and automatic closure of the other XVG-3109 Reactor Building cooling unit discharge valve.

"Air-side" and "water-side" testing is performed on one of the 2 redundant subsystems at a time. The other subsystem continues normal operation during the tests. The manual shutoff valve downstream of the service water booster pumps is closed prior to the tests to prevent unnecessary contamination of the cooling coils with service water.

6.2.2.4.2.3 Component Inspections and Tests

The cooling plenum structure, cooling coils, fans, motors, and filters are inspected during refueling periods for signs of corrosion, metal fatigue, leakage, deterioration of gaskets or painted surfaces, loosening of instrumentation, and for adequate lubrication. The filter plenums are tested during each refueling period with dioctyl phthalate (DOP) in accordance with the requirements of ANSI N510-1975 [20]. 99-01

6.2.2.5 <u>Instrumentation Requirements</u>

6.2.2.5.1 Reactor Building Spray System

Reactor Building Spray System instrumentation provides for system performance and status monitoring during normal plant operation, during system tests, and during system operation following a LOCA. The Spray System instrumentation does not have an active function in the actuation or operation of the system. The data displayed and controls located in the control room are adequate for assessing malfunctions and accidents and performing necessary actions for recovery.

6.2.2.5.1.1 Flow Instrumentation

Flow instrumentation loops utilize orifices and electronic transmitters for measuring the flow rate in the discharge of each pump. Each transmitter supplies a signal to a flow indicator on the main control board (in the control room) and to a bistable mounted in the instrumentation panel in the Control Building relay room. The bistable has adjustable setpoints for the high and low flow alarm inputs to the main control board annunciator system. Flow alarms are defeated until the associated pump breaker has been closed for a time interval greater than the setpoint of a time delay relay. This alarm alerts the operator to an abnormality in the Reactor Building Spray System. Flow rate is also displayed locally at the transmitter and is monitored by the computer.

6.2.2.5.1.2 Level Instrumentation

Redundant liquid level measuring loops are provided on the sodium hydroxide storage tank. In each of the 2 measuring loops an electronic pressure differential transmitter is used to detect the fluid level in the tank. Fluid level is displayed on the main control board and is monitored by the computer. The level signal is also supplied to a bistable mounted on the instrumentation panel in the Control Building relay room. The bistable has adjustable setpoints for Hi and Lo alarm inputs to the main control board 98-01
annunciator system. A site glass also provides visible indication of level in the sodium hydroxide storage tank. The site glass is isolated and drained after each reading and remains valved out until another reading is required.

An empty alarm is not used on the sodium hydroxide storage tank. Should the tank be 98-01
emptied before the RWST is emptied, the liquid interface recedes into the tank discharge lines but the lines to not completely empty because of the pressure maintained at the junction with the RWST lines by the head of liquid in the RWST. Introduction of air into the pumps from vortex effects in the sodium hydroxide storage tank has no significant effect on pump operation because of the low flow velocity in the tank outlet line and the fact that the flow from the sodium hydroxide storage tank comprises only about 2% of the total pump flow.

Level instrumentation in the Reactor Building recirculation sumps permits verification that adequate water has been collected in the bottom of the Reactor Building to maintain sufficient flow and NPSH for the spray pumps in the recirculation mode.

The RWST is not a part of the Reactor Building Spray System but has level instrumentation relevant to the Reactor Building Spray System. An alarm is sounded if the water level drops to a "lo" level during startup or normal plant operations. Level instrumentation also initiates alarms in the Control Room on "Hi", "lo-lo" and "empty" 98-01
RWST level. Upon receipt of the RWST "lo-lo" level signal in conjunction with the safety injection signal, the Reactor Building sump isolation valves open automatically and the operator closes the valves in the line from the RWST. The operator will not terminate the Reactor Building Spray System operation until after the required operating period of RN 03-003
2 hours for a design basis LOCA.

6.2.2.5.1.3 Pressure Instrumentation

Local pressure gages are located in the suction lines and the discharge lines of each spray pump. A pressure gage is also located on the nitrogen gas supply line to the sodium hydroxide storage tank. The gages are used for system and component evaluation during testing. The latter gages are also used for monitoring and controlling the nitrogen cover gas pressure in the sodium hydroxide storage tank.

An electronic pressure transmitter is located on the discharge of each spray pump. These transmitters supply a signal to a pressure indicator on the main control board. Pressure data are also monitored by the computer.

Orifices are located in the individual lines from the sodium hydroxide storage tank. The purpose of these orifices is to create a specific flow impedance in the sodium hydroxide lines from the tank to the junctions with the lines from the RWST. However, test connections are provided to which test gages can be attached for measurement of flow through these lines.

Flow tests, performed prior to plant startup define the flow characteristics of the Reactor Building Spray System from the RWST and sodium hydroxide storage tank through the line junctions. These test data are utilized to verify that the system delivers the proper flow of sodium hydroxide solution or to resize the orifices to obtain correct system characteristics.

6.2.2.5.1.4 Temperature Instrumentation

A temperature loop provides local indication of the temperature of the liquid in the sodium hydroxide storage tank. Temperature loops (2) provide local indication of the temperature of the liquid at the spray pump suctions.

6.2.2.5.1.5 Valve Position Indication

Valve position indication for all 10 motor operated valves in this system is provided on the main control board by status lights.

6.2.2.5.1.6 Control

Actuation of the Reactor Building Spray System results from a safety injection and spray actuation signals or from manual action at the main control board. Four (4) manual spray actuation switches are provided. Manual actuation of the Spray System requires simultaneous operation of one of the pairs of the 4 switches. The safety injection signal and spray actuation signals are generated by the ESF Actuation System. Upon receipt of a phase "A" containment isolation signal, the valves in the discharge headers 98-01
(XVG-3003A and B) and in the sodium hydroxide storage tank discharge lines (XVG-3002A and B) open. The spray actuation signals result from the Reactor Building "Hi-3" pressure instrumentation (2 out of 4 instrument channels tripped). A separate actuation signal is supplied to each spray subsystem. Upon receipt of this signal, the

spray pump starts and the valve (XVG-3001A or B) in the suction line from the RWST is opened (if closed). The spray pump and each valve have individual switches on the main control board for manual operation of each component. Switches are also provided on the main control board for normal operation of the valves (XVG-3004A and B, and XVG-3005A and B) in the suction lines from the Reactor Building sumps. The spray pumps are automatically aligned to the Reactor Building sumps in response to signals from the RWST level instrumentation at the "lo-lo" level. The pumps can be restarted individually by switches located on the main control board. The operator will not terminate Reactor Building Spray System operation prior to the required 2 hour operating period. 98-01

Operator action from the control room is required to accomplish the following:

1. Shutdown of the Reactor Building Spray System.

2. Manual activation of a Reactor Building spray subsystem.

3. Reactor Building Spray System operation in the recirculation mode.

6.2.2.5.2 Reactor Building Cooling System

Reactor Building Cooling System instrumentation provides for performance and status monitoring of the system during normal operation, during system tests and, for certain components, during system operation following a LOCA. Some instrumentation components provide for control and interlock functions in the operation of the system. Instrumentation relevant to the "waterside" of the Reactor Building cooling system is further described in Sections 9.2.1.5 and 9.4.7.2.5.

6.2.2.5.2.1 Flow Instrumentation

Flow rate in the supply and return service water piping serving the plenum coils is displayed and alarms are provided in the control room. Main control board indication of these flow signals permits verification that adequate cooling water flow is established in the Reactor Building cooling unit coils for post accident Reactor Building heat removal. Refer to Section 9.2.1.5 for further information on this service water flow instrumentation.

6.2.2.5.2.2 Temperature Instrumentation

Redundant resistance temperature detectors are located inside the Reactor Building to measure Reactor Building ambient temperature. Each detector supplies a signal to a temperature indicator on the main control board and one detector, through an isolation device, supplies a signal to an indicator on the control room evacuation panel.

A temperature switch is located in the discharge duct of each Reactor Building cooling unit and provides a control room alarm upon occurrence of high discharge temperature during normal operation.

6.2.2.5.2.3 Pressure Instrumentation

A differential pressure switch and a direct reading differential pressure gage are installed across the moisture separator and across the HEPA filter bank of each Reactor Building cooling unit.

The pressure switches provide control room annunciation of high filter pressure drop.

The gages are used for component evaluation and inspection.

6.2.2.5.2.4 Vibration Instrumentation

A velomitor is mounted on each Reactor Building cooling unit fan. These switches provide signals to trip the fan motors from high speed and to annunciate the condition in the control room if high fan vibration exists for a period greater than approximately 10 seconds. The velomitor signal is not a part of the fan control circuit for emergency (slow) speed operation and, therefore, has no effect upon this mode of fan operation. RN 09-023

6.2.2.5.2.5 Smoke Detection Instrumentation

An ionization type smoke detector is located in the discharge duct of each Reactor Building cooling unit. Each detector provides a control room alarm upon detection of smoke in the duct. Note that these duct detectors are maintained by the plant, but are not credited because other means exist to detect a fire in the Reactor Building. Their purpose is to aid plant operators in their decision making process. RN 13-016

6.2.2.5.2.6 Motor Current Instrumentation

Current transformers, located at the motor controllers, supply signals to ammeters on the main control board for indication of fan motor current. One ammeter is provided for each motor (high speed and slow speed) of each of the 4 cooling units (a total of 8 ammeters). Thus, fan motor current is continuously indicated for all operating modes and provides information on cooling unit status.

6.2.2.5.2.7 Damper Position Indication

Position indication for each HEPA filter bypass damper is provided on the main control board by status lights on the damper control switch assembly.

Additionally, the status of each damper is independently indicated by means of ESF monitor lights on the main control board.

6.2.2.5.2.8 Special Instrumentation

The main control board ESF monitor lights (see Section 7.5 for more detailed explanation) provide an easily recognizable indication of the status of essential components and equipment. Included among the monitor lights for this system are status indication (actuated or not actuated) of the Reactor Building cooling unit slow speed fan motors and of the HEPA filter bypass dampers. See Section 9.2.1.5 for a description of similar status monitoring of the Service Water System components required for Reactor Building heat removal.

An audible and visual alarm is annunciated on the main control board if manual action by the operator deliberately bypasses the operation of the Reactor Building Cooling System.

Also, an alarm annunciation of motor trip is provided on the main control board for each Reactor Building cooling unit fan.

6.2.2.5.2.9 Control

Actuation of the Reactor Building Cooling System results from a safety injection signal, a loss of offsite power signal, or manual action in the control room. Automatic actuation of the system follows the sequence outlined in Section 6.2.2.2.2.2. A separate actuation signal is supplied to each subsystem. Control switches, located at the main control board, allow the operators to manually operate each component required for heat removal.

For each Reactor Building cooling unit, main control board switches are provided for the following components:

1. High speed fan motor.

2. Slow speed fan motor.

3. HEPA filter bypass damper.

Manual actuation of the Reactor Building Cooling System (air-side) is accomplished with these switches.

For automatic initiation of the system (air-side) operation, the position of main control board selector switches determines which 1 of the 2 A and which 1 of the 2 B power train units starts in response to the ESF loading sequence signal for that train. One (1) two-position selector switch per electrical power train is utilized for this purpose.

Actuation of the water-side of the Reactor Building cooling units is described in Sections 9.2.1.5 and 9.4.7.2.5.

Operator action from the control room is required to accomplish the following:

1. Shutdown of the Reactor Building Cooling System.

2. Manual actuation of a Reactor Building cooling subsystem.

6.2.3 REACTOR BUILDING AIR PURIFICATION AND CLEANUP SYSTEM

The Reactor Building Spray System has 2 functions. One is to remove heat from the Reactor Building atmosphere after a LOCA and the other is to remove fission products from the post LOCA atmosphere to reduce the dose to personnel offsite. The heat removal function is described and evaluated in Section 6.2.2. Additionally, the Reactor Building Cooling System HEPA filters aid in Reactor Building air purification. These filters are discussed in Section 6.5.

6.2.3.1 Design Bases

The Reactor Building Spray System is designed to reduce the concentration of radioactive iodine in the post LOCA Reactor Building atmosphere. In accordance with Regulatory Guide 1.183, it is assumed that of the radioiodine activity released from the core, 95% is in the form of cesium iodide (CsI), 4.85% in the form of elemental iodine and 0.15% in the form of organic iodine. RN 12-034

6.2.3.2 System Design (For Fission Product Removal)

The design of the Reactor Building Spray System is described in Section 6.2.2.2.1 for the overall system. The system diagram is included as Figure 6.2-46. This section discusses the design features related solely to the fission product removal function of the system.

6.2.3.2.1 System Design Features

The sodium hydroxide (NaOH) storage tank is designed and located to permit gravity draining into the spray pump suction line from the RWST. For the design case (one spray subsystem, one RHR pump, and one charging pump operating) it empties at approximately the same time as does the RWST. Orifices in RWST suction lines and the NaOH lines provide sufficient impedance to the system to maintain NaOH flow within the limits necessary for satisfactory spray pH level.

The 3300 gallon nominal capacity tank (3050 gallon minimum required usable volume) is 45-1/2 inches in diameter (OD), 40 feet-10 inches high, and is filled with NaOH solution at a minimum concentration of 20% by weight and a maximum concentration of 22% by weight. During injection, NaOH drains into each pump suction line at a junction at the 399’-6" elevation level at a nominal rate of 50 gpm. 99-01

A cover gas of nitrogen is maintained in the vapor space over the liquid to preclude precipitation or tank corrosion that could occur with air present. The tank is equipped with 2 pressure relief valves set to limit nitrogen overpressure to 3 psig. Redundant vacuum relief valves are sized to relieve vacuum in the tank as it is drained during spray operation. An orifice in the nitrogen supply line limits the flow of nitrogen to the relief capacity of one relief valve. This prevents any inadvertent overpressurization of the tank during pressurization operations.

A common discharge line from the NaOH storage tank is subsequently branched to provide flow to each pump suction line. In addition, nitrogen supply, vent, demineralized water supply, and liquid sample connections are provided on the tank. Redundant instrumentation is provided for measuring the liquid level and alarms in the control room. A temperature instrument is also provided. RN 13-010

The NaOH storage tank is provided with a recirculation loop attached at the fill and drain tank nozzles. The recirculation loop is provided with a recirculation pump and associated valves and instrumentation. The recirculation loop was added after initial plant operation under MRF 20168 to address concerns with NaOH tank stratification particularly when adding water to dilute the more dense NaOH solution and to provide a consistent NaOH concentration throughout the tank. To address industrial safety concerns related to handling NaOH, a feed pump and feed tank with a mixer are also provided, tying in to the recirculation loop. The feed is isolated from the recirculation loop by a check valve. The recirculation loop is normally isolated by locked closed valves as shown in Figure 6.2-46. The recirculation loop is non-safety related, but is seismically supported. RN 13-010

The NaOH storage tank is located outside in the Auxiliary Building tank pit at elevation 412'. For a concentration range between 20% and 22%, the maximum temperature at which freezing or separation of solids occurs is - 13°F. Therefore, a heat tracing system is not required.

The Reactor Building Spray System includes a total of 330 spray nozzles. These nozzles are ramp type, hollow cone nozzles. With a 40 psi pressure drop across the nozzle, the drop size distribution is a log normal distribution with a number mean diameter of 230 microns. The mass median diameter is approximately 680 microns. For conservatism, a mass mean diameter of 1300 microns is used in calculating iodine removal effectiveness (see Section 6.2.3.3.1). The spray nozzles are located on 6 rings as described in Section 6.2.2.2.1.1. A plan view of the spray nozzle locations is shown in Figure 6.2-47. The arrangement of the spray rings within the Reactor Building is such that essentially all of the free volume above the operating floor, elevation 463', is exposed to the spray. Below the operating floor elevation, areas outside the secondary shield walls are shielded from the spray. However, the atmosphere in this area is mixed with the rest of the volume inside the secondary shield walls. The volume inside the secondary shield walls with the exception of the pressurizer compartment, is open to the spray.

6.2.3.2.2 System Operation

The operation of the Reactor Building Spray System is presented in Section 6.2.2.2.1.2.

6.2.3.3 Design Evaluation

6.2.3.3.1 Fission Product Removal Effectiveness Calculations

The fission product removal capabilities of the Reactor Building Spray System under post accident conditions have been the subject of much research of government laboratories as well as NSSS vendors.

The effectiveness of this system is estimated by considering the chemical and physical processes that could occur under postulated accident conditions. Iodine is the primary fission product to be considered for removal by the Reactor Building Spray System, post accident. The principal forms of iodine that are of concern in the post accident environment are: elemental, aerosol particulates, and organic iodides.

The removal of elemental iodine from the Reactor Building atmosphere is considered a first order removal process which is mathematically represented by time dependent removal coefficients (λ) due to the action of the spray system (λ_s) and by wall deposition (λ_w). These removal coefficients are summed to obtain the total removal coefficient for elemental iodine.

Experimental results, References [33], [34] and [35], have shown that an important factor determining effectiveness of sprays in removing elemental iodine vapor is the concentration of iodine in the spray solution. Fresh sprays having no dissolved iodine were observed to be quite effective in the scrubbing of elemental iodine even at a pH as low as 5, References [34] and [35]. Sump solutions with dissolved iodine that recirculate post accident may revolatilize iodine if the solution is acidic, Reference [36]. Section 6.2.2.3.1.4 details the spray solution pH during the conditions of injection and sump recirculation under the extremes of system operating conditions. In all configurations analyzed, the spray/sump solution pH remains greater than 7.

Chemical additives in the spray solution have no significant effect upon aerosol particle removal since this process is principally mechanical.

1. Elemental iodine

During injection, the removal of elemental iodine by wall deposition is calculated by using the following equation:

$$\lambda_w = K_w A / V$$ 02-01

where:

k_w = mass transfer coefficient = 4.9 m/hr.

A = wetted surface area = 3457 m^2.

V = Reactor Building net free volume = 1.84 x 10^6 ft^3.

λ_w = removal coefficient by wall deposition = 0.325 hr^{-1}.

Based upon the well-mixed droplet model, the effectiveness of the spray in removing elemental iodine vapor is chiefly determined by the rate at which fresh solution area is introduced into the atmosphere. This can be estimated by the relationship (6F/VD) where F is the volume flow rate of the spray, V is the building net free volume, and D is the mass mean diameter of the spray drops. The first order spray removal coefficient is calculated using the following relationship:

$$\lambda_s = 6K_gTF/VD$$

where:

K_g = gas-phase mass transfer coefficient = 180 m/hr.

T = fall time of the drops = ratio of the average fall height (h) to the terminal velocity (U) where h = 114 ft and U = 410 cm/sec.

F = Flow rate = 2500 gpm.

V = Reactor Building volume - 1.84 x 10^6 ft^3.

D = drop diameter = 1300μ.

λ_s = 20.0hr^{-1}

λ_s = $\lambda_s + \lambda_w$ = 20.0 + 0.325 = 20.325hr^{-1}

2. Organic Iodides

It is conservatively assumed that organic iodides are not removed by the action of the sprays or wall deposition.

3. Particulate Iodine

The removal coefficient for particulate iodine by sprays is given by the following equation:

$$\lambda_p = 3hFE/2VD$$

where:

h = fall height of the droplets = 114 ft.

F = spray flow rate = 2500 gpm.

V = Reactor Building volume = 1.84×10^6 ft^3.

E/D = ratio of the dimensionless collection efficiency, (E), to the drop diameter, (D). E/D is assumed equal to 10 m^{-1} for a period of time until the aerosol mass has been depleted by a factor of 50. Thereafter, E/D is reduced by a factor of 10 to 1 m^{-1} (i.e. 98% of the suspended mass is 10 times more readily removed than the remaining 2%).

λ_p = 5.68 hr^{-1} (0 - 98% removal).

λ_p = 0.568 hr^{-1} (98 - 100% removal).

6.2.3.4 Tests and Inspections

Tests and inspections of the Reactor Building Spray System are described in Section 6.2.2.4.1.

6.2.3.5 Instrumentation Requirements

Instrumentation requirements of the Reactor Building Spray Systems are described in Section 6.2.2.5.1.

6.2.3.6 Materials

Spray system materials are compatible with the reactor coolant and the alkaline (NaOH) solution. The major components of the system are constructed of stainless steel, except the sodium hydroxide storage tank which is of carbon steel. Minor parts, such as pump seals, utilize other corrosion resistant materials.

6.2.4 CONTAINMENT ISOLATION SYSTEM

The objective of the containment isolation system is to allow the passage of fluids through the containment boundary under normal and accident conditions, while preserving the integrity of the containment boundary, when required to prevent or limit the escape of fission products as a result of a postulated LOCA.

The containment isolation system is not an independent system. Rather, the system is comprised of specific provisions included in each piping system that penetrates the Reactor Building. Actuation of containment isolation is accomplished through the Engineered Safety Features Actuation System (ESFAS). A detailed discussion of the ESFAS system is provided in Section 7.3.

6.2.4.1 Design Basis

In the unlikely event of a LOCA, the containment isolation system is designed to limit leakage of radioactive materials through lines penetrating the Reactor Building so that the site boundary dose guidelines specified in 10 CFR 50.67 are not exceeded. RN 12-034

Lines for which isolation is required are provided with 2 barriers, such that no single failure can prevent isolation. Each of these barriers is adequate to limit leakage of radioactivity within acceptable values over the entire range of normal and accident conditions. The selection of fast action automatically operated valves that receive isolation signals is based upon the safety analysis requirements described in Chapter 15.

An "isolation barrier" is either an isolation valve, blind flange, or a closed system. A closed system is defined as a fluid system which is not part of the Reactor Coolant System and which does not communicate with the Reactor Building atmosphere.

Fluid systems, including instrument lines, penetrating the Reactor Building satisfy the following specific design bases:

1. The design of isolation barriers for lines penetrating the Reactor Building follows the requirements of General Design Criteria 54 through 57 of 10CFR50, Appendix A.

2. The containment isolation system is protected against the effects of missiles and postulated pipe ruptures. See Sections 3.5 and 3.6.

3. The containment isolation system is designed to Safety Class 2a, Seismic Category 1 requirements. Regulatory Guides 1.26 and 1.29 are discussed in Appendix 3A.

4. The containment isolation system is designed with redundancy in accordance with General Design Criterion 54 so that no single active failure can result in loss of containment integrity.

5. The containment isolation system is designed to withstand accident environmental conditions (see Section 3.11).

6. Provisions are made for periodically testing the operability and leak tightness of the containment isolation valves to the extent necessary to ensure that the system satisfies performance requirements in the event of an accident.

6.2.4.2 System Design

A list of Reactor Building penetrations showing the means of containment isolation is included as Table 6.2-53. Containment isolation valves are listed by penetration number in Table 6.2-54. Each listing indicates valve number, size, methods of actuation and positions under various plant conditions. Figure 6.2-52 schematically shows the various valve arrangements used to isolate the fluid system penetrations.

6.2.4.2.1 Lines Penetrating the Reactor Building that Connect to the Reactor Coolant Pressure Boundary (General Design Criterion 55)

This category includes the process lines passing through penetrations numbered 221, 222, 223, 226, 227, 229, 314, 316, 318, 322, 325, 405, 408, 409, 410, 412, 415 and 426. Of these, 221, 222, 227, 229, 322, 325, 408, 409, 412, 415 and 426 have one or more check valves in parallel as the inside isolation valves and one or more power-operated valves in parallel as the outside isolation valves. Penetrations 223, 314, 318 and 405 have one or more power-operated valves in parallel as the inside isolation valves and one power-operated valve as the outside isolation valve.

Penetration 410 has a power-operated valve and check valve in parallel as the inside isolation device and power-operated valve as the outside isolation device. These arrangements all comply with the literal requirements of General Design Criterion 55.

Penetrations 226 and 316 do not have outside isolation valves. The closed Residual Heat Removal (RHR) System boundary is used as the outside containment device. These lines connect to the safety injection recirculation loops which are filled with sump water and at least one of which is in operation following an accident. Should a leak occur in the short length of pipe inside the Reactor Building between the valve and Reactor Building wall, the Reactor Building atmosphere would pass only as far as the fluid filled system containing sump water. This would prevent gas release to the outside. The design basis for these penetrations is that isolation valves outside the Reactor Building on the RHR suction lines are unnecessary and that inclusion of any such additional valves does not enhance plant safety.

The power-operated valves for penetrations 223, 314, 318, 405, 409, and 410 are automatically closed upon receipt of either a safety injection signal or containment isolation, Phase A signal, as noted in Table 6.2-54. The power-operated valves for penetration 426 are automatically opened by the safety injection signal to permit boron injection.

Valves in lines required to perform a safety function following an accident must be opened or must remain open for the Engineered Safety Features (ESF) Systems to operate. Automatic closing of these valves would defeat their intended purpose. Therefore, these valves are designed for remote manual operation from the control room. Lines in this category include low pressure safety injection lines and high pressure safety injection lines. Penetrations in this category are 222, 227, 322, 325, 412, and 415.

The seal injection lines to the reactor coolant pumps pass through penetrations 221, 229, and 408. Due to the sensitive nature of the pump seals, it is highly desirable to provide seal cooling flow at all times. Since the charging pumps are used for safety injection, the high pressure in-flow through the seal injection lines eliminates any need for automatic closure of the valves in these lines. Remote manual closure from the control room is available.

6.2.4.2.2 Lines Penetrating the Reactor Building That are Open to the Reactor Building Atmosphere (General Design Criterion 56)

This category includes the process lines passing through penetrations numbered 101, 103, 105A, 105B, 107, 201, 208, 209, 210, 211, 212, 216, 217, 231, 301A, 301B, 302, 303, 310, 311, 313, 317, 319, 320, 321, 324, 327, 328, 329, 401, 402, 404, 406B, 407A, 407B, 417D, 418, 419, 420, 421, 422, 423, 424, 425, 427, 500, 505, 600, 602, and 703. 98-01
Of these, 101, 103, 105A, 105B, 302, 303, 311, 317, 319, 320, 321, 401, 402, 407A, 407B, 417D, 418, 420, 422, 423, 424, and 427 have one power-operated valve outside and one power-operated or check valve inside. Penetrations 208 and 209 have one power operated valve and one check valve in parallel as the inside isolation and one power operated valve as the outside isolation. Penetration 301A has 2 power operated valves in parallel as the inside isolation and one power operated valve as the outside isolation. Penetration 301B has one power operated valve as the inside isolation and 2 power operated valves as the outside isolation. Penetrations 231, 310, 313, 324, 404, 419, and 421 have a locked closed manual valve outside and check valve or a locked closed manual valve inside. The process pipes through penetrations 201, 210, 211, 212, 216, 505, 600, and 602 are sealed with blind flanges inside and outside during operation. These modes of isolation are in literal compliance with General Design Criterion 56, considering that a blind flange is the equivalent of a locked closed valve.

Penetration 107, the fuel transfer tube, is sealed with a blind flange on the inside end. This flange has a double O-ring seal to provide a double barrier. Although there is a manual gate valve on the outside end of the tube, it is not considered a containment isolation barrier since the blind flange, when in place, is the containment boundary.

Penetrations 217, 406B, 500, and 703 are for the Reactor Building pressure sensing lines. Reactor Building pressure is sensed by 4 physically separated differential pressure transmitters mounted on Seismic Category 1 supports outside the Reactor Building. These devices are connected to the 1/2 inch stainless steel Reactor Building penetration by a length of 1/4 inch outside diameter, high pressure, stainless steel tubing. Bellows are connected to the ends of the tubes inside the Reactor Building. The capillary tubing between the bellows and the transmitter is fluid filled. The distance from penetration to transmitter is minimized and separation is maintained. This arrangement, together with the pressure sensors outside the Reactor Building, forms a double barrier. For a discussion of Regulatory Guide 1.11, see Appendix 3A.

98-01

RN 07-022

The lines from the Reactor Building recirculation sumps to the suctions of the redundant, low head safety injection recirculation (residual heat removal) pumps and Reactor Building spray pumps are each provided with a single motor-operated gate valve outside the Reactor Building. These valves are enclosed in containers which are leaktight at the Reactor Building design pressure. Piping from the sump to the valve containers is enclosed in a concentric guard pipe, also leaktight at the Reactor Building design conditions. The structural configuration is such that neither a valve contained nor the guard pipe is connected directly to the sump or to the Reactor Building atmosphere.

At the beginning of the safety injection recirculation phase the sump is full of water, the sump line valves are opened, and the line from the refueling water storage tank is closed. Since the sump is always submerged during recirculation following an accident and no Reactor Building atmosphere can impinge upon the valve, this valve is not a containment isolation valve in the usual sense. The valve does provide a barrier outside the Reactor Building to prevent loss of sump water should a leak develop in a recirculation loop. (This valve is capable of remote manual closure from the control room to accomplish this.) Should a leak develop in the valve body or in the pipe between the sump and the valve, the sump fluid is contained by the leaktight container and guard pipe arrangement. Since the sump line connection to the sump is always submerged during the recirculation phase, there is no need for a valve inside the Reactor Building to prevent escape of the Reactor Building atmosphere.

No single failure of either an active or passive component of this system will prevent recirculation of core cooling water nor will such a failure adversely affect the integrity of the Reactor Building. This arrangement satisfies safety requirements and no additional valves are necessary for containment isolation with respect to penetrations 327, 328, 329, and 425.

The power operated valves associated with penetrations 208, 209, 311, 317, 319, 320, 321, 407A, 407B, 417D, 418, 420, 422, 423, 424, and 427 are automatically closed by the containment isolation, Phase A signal. The hydrogen analyzer isolation valves associated with penetrations 105A, 105B, 301A, and 301B do not receive a closure signal because these valves are normally closed and will be opened post LOCA. The portion of penetration 301B shared with a Reactor Building pressure transmitter does isolate on a Phase A signal. The power operated valves associated with penetrations 101, 103, 302, and 402 are automatically closed by a containment ventilation isolation signal or high radiation in the Reactor Building purge exhaust vent (A train valves) or containment atmosphere high radiation (B train valves). The valves on penetrations 303 and 401, the Reactor Building spray supply lines, are automatically opened by a containment Phase A isolation signal.

6.2.4.2.3 Lines Penetrating Reactor Building That Do Not Connect With Either the Reactor Coolant Pressure Boundary or the Reactor Building Atmosphere (General Design Criterion 57)

The category includes the process lines passing through penetrations numbered 102, 202, 203, 204, 205, 206, 207, 213, 219, 220, 224, 225, 230A, 230B, 230C, 304, 305, 306, 308, 312, 323, 326, 330, 403, 411, 417A, 417B, 417C, and 428. Of these 102, 204, 304, 305, 312, 323, 330, and 403 have automatically actuated power operated valves outside the Reactor Building. These lines are in literal compliance with General Design Criterion 57. 98-01

6.2.4.3 <u>Design Evaluation</u>

The process piping through the penetrations and the isolation valves are designated as Safety Class 2a. They are designed in accordance with the ASME Code, Section III, Class 2.

Where power-operated valves are supplied inside and outside the Reactor Building, the inside valve and the outside valve are actuated by different trains of the actuation system. The train that actuates any isolation valve is indicated in the "Signal" column of Table 6.2-54.

Valves which may be required to operate in the post LOCA Reactor Building atmosphere to serve their safety function are qualified as described in Section 3.11.

The containment isolation system design complies with the recommendations of SRP 6.2.4; i.e., that there be diversity in the parameters sensed for the initiation of containment isolation. The containment isolation signals are given in Table 6.2-54.

Careful reconsideration as required by NUREG-0578, item 2.1.4, to the definition of essential and non-essential systems has been given. The basis of selection is that the essential systems are those which are required immediately to mitigate the consequences of an accident. All other systems are considered non-essential. The systems which penetrate containment and their classification are listed in Tables 6.2-53 and 6.2-54a. During the re-evaluation none of the systems were changed from nonessential to essential. 00-01

All non-essential systems, except those which have a blind flange or a locked closed manual valve, are automatically isolated by the containment isolation signals.

The design of control systems for automatic containment isolation valves is such that resetting the isolation signal will not result in the automatic reopening of containment isolation valves, with the exception of the letdown orifice isolation valves if the transfer switch on the Control Room Evacuation Panel is set for local operation and the control switch is in the open position. Reopening of containment isolation valves requires deliberate operator action. System operating procedures detail specific valves to operate during post accident recovery operations and valve operations for complete Safety Injection - Containment Isolation Recovery operations.

6.2.4.4 Test and Inspection

Leakage tests and containment isolation valve operability tests are discussed in Section 6.2.6.3. An isolation valve seal system is not employed.

6.2.4.5 Materials

The materials used in the components of the containment isolation system are selected so that the system can perform its engineered safety function throughout the life of the plant under all anticipated environmental situations. Criteria considered are as follows: RN 04-019

1. Materials selected for components of the containment isolation system inside the Reactor Building will withstand an integrated radiation dose of 10^8 rads without significant degradation of material properties.

2. All materials used in the containment isolation system that come in contact with the Reactor Building spray solution (alkaline solution of sodium hydroxide and boric acid) are resistant to corrosion by this solution. Nuclear steam supply systems that are normally wetted by borated water are made of austenitic stainless steel. Other containment isolation valves inside the Reactor Building are of carbon steel.

3. Materials used for containment isolation system components which contain zinc or aluminum are kept to a minimum. 98-01

6.2.4.6 Special Instrumentation

The main control board ESF monitor lights (see Section 7.5 for more detailed explanation) provide an easily recognizable indication of the status of essential components and equipment. Included among the monitor lights are status (position) indication of the containment isolation valves.

6.2.5 COMBUSTIBLE GAS CONTROL IN REACTOR BUILDING

Following a design basis accident, hydrogen gas may be generated inside the Reactor Building by reactions such as zirconium metal with water, corrosion of materials of construction, and core and sump water radiolysis. The following section is presented to describe the basis for the sizing of the combustible gas control system. To ensure that the hydrogen recombiners and an alternate Reactor Building purge system are provided.

To minimize hydrogen generation as a result of corrosion of containment metals, the use of exposed zinc and aluminum inside the Reactor Building is tracked and logged. In addition, the amount of hydrogen generated in the post accident environment is limited by maintaining a Reactor Building spray pH of 10.5 or less and a recirculation solution pH of less than 9.0. 98-01

6.2.5.1 Design Bases

6.2.5.1.1 Electric Hydrogen Recombiners

1. The recombiners are designed to sustain all normal loads as well as accident loads including safe shutdown earthquake and pressure transients from a design basis loss of coolant accident.

2. The recombiners are designed for a lifetime of 40 years, consistent with that of the plant. (Aging management will extend recombiner life for the duration of the plant life.) RN 04-019

3. All materials used in the recombiners are selected to be compatible with the environmental conditions inside the Reactor Building during normal operation or during accident conditions.

4. Process capacity is such that the Reactor Building hydrogen concentration will not exceed 4 volume percent based on the NRC TID-14844 release model as indicated in Regulatory Guide 1.7.

5. Each recombiner is provided with a separate power panel and control panel. Each recombiner is powered from a separate ESF bus.

6.2.5.1.2 Alternate Reactor Building Purge System

Supply and exhaust flow lines connected to pressure blowers are provided as a backup system to purge the Reactor Building for combustible gas control and to aid in cleanup of the Reactor Building atmosphere.

6.2.5.1.3 Hydrogen Concentration Monitor

The following design bases apply to the equipment provided to monitor hydrogen in the Reactor Building.

1. Redundant hydrogen concentration analyzers are provided to confirm that the electric hydrogen recombiners are functioning property following an accident. The analyzers are not required to initiate operation of the electric hydrogen recombiners since the recombiners will be started 24 hours after a LOCA and the ability of these recombiners to remove hydrogen from the Reactor Building atmosphere has been accepted by the NRC [22]. During normal plant operating conditions, one hydrogen analyzer is in standby. The redundant hydrogen analyzers will be started after a LOCA.

2. Dedicated sampling points for each monitor, capable of withstanding the SSE without loss of function, are provided.

3. The system is capable of remaining operable under both positive and negative pressure assuming a single failure.

4. The hydrogen monitoring system is class 1E and seismic Category 1.

5. Semi-automatic sampling capability to permit laboratory analysis as backup to the hydrogen analyzer is provided. This system is also capable of determining noble gases and iodine concentrations.

6. During a loss of coolant accident, Reactor Building hydrogen concentration is monitored and hydrogen recombiners are placed in service to minimize hydrogen buildup in the Reactor Building. Adequate shielding is provided to prevent personnel overexposure during hydrogen recombiner operation at local panels.

6.2.5.2 System Design

The system is designed for control of combustible H_2 concentrations in the Reactor Building following a loss of coolant accident. This system consists of a sampling system that provides Reactor Building atmosphere samples with electric hydrogen recombiners as the primary means of reducing H_2 Reactor Building concentrations and a purge system which is used as a backup system to the recombiners.

6.2.5.2.1 Electric Hydrogen Recombiners

Redundant electrical recombiners, as shown on Figure 6.2-53 are located inside the Reactor Building. The recombiner units are located in the Reactor Building such that they process a flow of Reactor Building air containing hydrogen at a concentration which is generally typical of the average concentration throughout the Reactor Building. Since the recombiner units are located in the Reactor Building, dedicated containment penetrations for hydrogen control are not required. In addition, the capability to install a recombiner unit outside the Reactor Building is not required.

To meet the requirements for redundancy and independence, 2 recombiners are provided, and each recombiner is provided with a separate power panel and control panel, and each is powered from a separate class 1E bus.

Reactor Building atmosphere is circulated through the recombiner by natural circulation where hydrogen is removed by heating to a temperature sufficient to cause recombination with the Reactor Building oxygen.

The recombiner consists of a thermally insulated vertical metal duct with electric resistance metal sheathed heaters provided to heat a continuous flow of Reactor Building air (containing hydrogen) up to a temperature which is sufficient to cause a reaction between hydrogen and oxygen. The recombiner is provided with an outer enclosure to keep out Reactor Building spray water. The recombiner consists of an inlet preheater section, a heater-recombination section, and a discharge mixing chamber that lowers the exist temperature of the air.

The unit is manufactured with corrosion resistant, high temperature materials except for the base which is carbon steel. The electric hydrogen recombiner uses commercial type electric resistance heaters sheathed with Incoloy-800 which is an excellent corrosion resistant material for this service.

Air drawn into the recombiner by natural convection and passes first through the preheater section. This section consists of a shroud placed around the center heater section to take advantage of heat conduction through the walls to preheat the incoming air. This accomplishes the dual functions of reducing heat losses from the recombiner and of preheating the air.

The warmed air passes through a flow orifice and then enters the electric heater section where it is heated to approximately 1150°F to 1400°F causing recombination to occur. Tests have verified that the recombination is not a catalytic surface effect associated with the heaters but occurs due to the increased temperature of the process gases. Since the phenomenon is not a catalytic effect, saturating of the unit by fission products will not occur. The heater section consists of 5 assemblies of electric heaters stacked vertically. Each assembly contains individual heating elements. Table 6.2-55 gives the recombiner design parameters.

The recombiner, power supply panel, and control panel are shown schematically in Figure 6.2-54. The power panel for the recombiner contains an isolation transformer plus a semiconductor controlled rectifier controller to regulate power into the recombiner. This equipment is not exposed to the post LOCA environment. To control the recombination process, the correct power input which will bring the recombiner above the threshold temperature for recombination will be set on the controller. The correct power required for recombination depends upon Reactor Building atmosphere conditions and will be determined when a recombiner operation is required. For equipment test and periodic checkout, a thermocouple readout instrument is also provided in the control panel for monitoring temperatures in the recombiner.

Reference [22] provides a description of the testing of a full scale prototype electric hydrogen recombiner.

6.2.5.2.2 Alternate Reactor Building Purge System

The alternate purge lines, which serve as backup to the hydrogen recombiners may be used in the unlikely event of failure of both hydrogen recombiners to operate following a LOCA. For this case, the supply pressure blower is operated to pressurize the Reactor Building to 3 psig with a clean supply of outside air (at approximately 600 scfm). This procedure is initiated when the H_2 concentration approaches 3 volume percent after a loss of coolant accident (approximately 2 1/2 days) and reduces the H_2 concentration to approximately 2.5 volume percent. As the H_2 concentration again approaches 3 volume percent, the backup purge isolation valves are opened and the Reactor Building atmosphere is released and throttled at a rate of 100 scfm through the charcoal and HEPA filters of the purge exhaust plenum while pressure returns to 0 psig. This operation is repeated each time H_2 concentration approaches 3 volume percent subsequent to initial accumulation. The alternate purge lines contain the required safety-related isolation valves as specified by Regulatory Guide 1.7. These components are designed for remote-manual operation from the control room, as required following a LOCA.

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6.2.5.2.3 Hydrogen Concentration Monitor

Figure 6.2-58 schematically shows the provisions for monitoring the Reactor Building atmosphere.

The capability for obtaining samples near each electric hydrogen recombiner and near the Reactor Building cooling units at approximately 530 foot elevation is provided. The redundant analyzers are operated post accident to continuously monitor the Reactor Building atmosphere. The plant operator can operate each analyzer from the control room. Two (2) sampling panels are provided for taking samples from connections in each hydrogen analyzer system. Each redundant sampling system and its associated control panel are accessible following an accident.

Seismic and environmental qualifications of the containment isolation valves for the sample lines and the hydrogen analyzers are discussed in Sections 3.10 and 3.11, respectively. The hydrogen analyzers are located outside the Reactor Building and are not subjected to extreme environmental conditions.

6.2.5.3 Design Evaluation

6.2.5.3.1 Electric Hydrogen Recombiners

Prediction of hydrogen generation following a LOCA (refer to Figures 15.4-71 through 15.4-73), shows that although hydrogen production rate decreases with time after the accident, total hydrogen accumulation could exceed 4 volume percent. The hydrogen recombiners limit hydrogen accumulation. Figure 15.4-71 shows the rate of hydrogen production from various sources with no recombiner operating after a LOCA. Figure 15.4-72 depicts the corresponding hydrogen accumulation in the containment. 98-01

For the purpose of showing that the electric recombiner is capable of maintaining safe hydrogen concentrations, an analysis was performed using the Regulatory Guide 1.7 model. The results for the Reactor Building volume are shown in Figure 15.4-73. The 98-01 model is based upon assuming a fission product activity release specified in TID-14844 and the values for post accident hydrogen generation specified in Regulatory Guide 1.7. The results, using no recombiner, are also shown on this figure.

Each electric recombiner is capable of continually processing a minimum of 100 scfm of Reactor Building atmosphere. The hydrogen contained in the processed atmosphere is converted to steam, which then exists to the Reactor Building atmosphere, thus reducing the overall Reactor Building hydrogen concentration. The hydrogen concentration in the Reactor Building was calculated for the models described above based on a recombiner capability of processing 100 scfm of Reactor Building atmosphere. This calculation shows that the maximum hydrogen concentration will be much less than 4 volume percent if the recombiner is started one day following the 98-01 accident. Therefore one of these recombiners meets the design criterion of maintaining a safe hydrogen concentration with considerable margin, and the second unit provides the redundance of a system of equal capability on a redundant power supply.

The peak hydrogen concentration occurs at the time when the amount of hydrogen being generated is equal to the amount of hydrogen being processed. Once this peak has been reached, the recombiner will be processing hydrogen at a faster rate than it is being produced. This will result in an overall reduction of the hydrogen concentration inside the Reactor Building. Thus, once the peak has been reached, the electric recombiner provides a continually increasing margin between the Reactor Building hydrogen concentration and the 4 volume percent limit.

The recombiner is designed to sustain all normal loads as well as accident loads such as seismic loads and temperature and pressure transients from a LOCA.

The Westinghouse electrical hydrogen recombiner, as described in topical report WCAP 7709-L (Proprietary) with Supplements 1 to 4 and in WCAP 7820 (non-proprietary) is considered by the Virgil C. Summer Nuclear Station to be the principal component used for combustible gas control. Whereas, bypass and inoperable status monitoring may be needed to provide information that certain protection equipment and systems that are normally operating are actually inoperative, such monitoring is not necessary for the electrical hydrogen recombiner because it is normally not operating and is not armed for automatic actuation. As described in WCAP 7709, the system is operated only during periodic testing and after the postulated Loss-of-Coolant Accident. Operation is initiated by operator action from the control station. Following an accident, the elapsed time prior to the needed start of the equipment is in terms of hours or days, and this elapsed time is evaluated in WCAP 7709. The design and environmental and seismic qualification of the Westinghouse electrical hydrogen recombiner was found acceptable for the prototype and production models by NRC, as reported in NRC's letter of May 1, 1975, from D. B. Vassallo to C. Eicheldinger, Manager of Westinghouse Nuclear Safety Department.

For further information on hydrogen production and accumulation see Section 15.4.1.2.

6.2.5.3.2 Alternate Reactor Building Purge System

If required, purging of the Reactor Building atmosphere following a LOCA is limited to a flow rate of 100 scfm by the alternate purge system. This flow rate is consistent with the minimum rated capacity of 100 scfm for one hydrogen recombiner.

6.2.5.3.3 Reactor Building Atmospheric Sampling System

Redundant sample systems have been designed to collect samples of the containment atmosphere. These samples are then analyzed in the radiochemistry laboratory for hydrogen, noble gases, and iodine. Radiation exposure and shielding considerations are discussed in FSAR Appendix 12A. Heat tracing is provided to prevent condensation of water vapor within the system.

6.2.5.4 Tests and Inspections

6.2.5.4.1 Electric Hydrogen Recombiners

The electric hydrogen recombiners have undergone extensive testing in the Westinghouse development program. These tests encompassed the initial analytical studies, laboratory proof-of-principal tests, and full scale prototype testing. The full scale prototype tests included the effects of:

1. Varying hydrogen concentrations.

2. Alkaline spray atmosphere.

3. Steam effects.

4. Convection currents.

5. Seismic.

A detailed discussion of these tests is given in Reference [22]. Post operational tests and inspections will be performed in accordance with Technical Specification requirements. Inspections will be performed to assure the capability of the recombiner to perform its function. Testing will be performed to verify operation of the control system to verify functional performance of the heaters to the required temperature level.

Seismic and environmental qualifications are discussed in Sections 3.10 and 3.11.

6.2.5.4.2 Alternate Reactor Building Purge System

The active components of the alternate purge lines, which are the containment isolation valves, are periodically tested in accordance with the requirements of Section 6.2.6, Containment Leakage Testing. 98-01

6.2.5.4.3 Sample Panels

The sample panels will be used on a periodic basis to maintain operator proficiency with system operation.

6.2.5.5 <u>Instrumentation Requirements</u>

6.2.5.5.1 Electric Hydrogen Recombiners

The recombiners do not require any instrumentation inside the Reactor Building for proper operation after a LOCA. The recombiners will be started manually after a LOCA. The sampling system will be used in obtaining Reactor Building atmosphere samples to verify proper operation of the recombiners or to indicate when the purge system should be actuated. This sample can be taken from any of 3 well ventilated locations within the Reactor Building.

6.2.5.5.2 Alternate Reactor Building Purge System

The powered components of the alternate Reactor Building purge line are designed for remote manual operation from the control room, as required following a LOCA.

6.2.5.5.3 Hydrogen Concentration Monitors

Manual action is required to start the redundant hydrogen analyzers. Since these analyzers provide only confirmation of proper operation of the redundant electric hydrogen recombiners, no automatic action is required.

The hydrogen analyzers have 0-10% and 0-20% range of H_2 in air (by volume) and an accuracy of $\pm$ 2% of range. This provides adequate confirmation of electric hydrogen recombiner operation. Hydrogen concentration in the Reactor Building is indicated in the control room.

6.2.5.5.4 Reactor Building Atmospheric Sampling System

Manual action is required to align the sampling system valves with the hydrogen analyzer supply and return lines. Once in service, the operational sequence is controlled remotely. The system is purged with nitrogen to containment following sampling completion. Following purge, the system is returned to the non-sampling mode manually, if required.

6.2.5.6 Materials

6.2.5.6.1 Electric Hydrogen Recombiners

The materials of construction for the electric recombiner are selected for their compatibility with the post LOCA environment.

The major structural components are manufactured from 300-Series stainless steel except for the base which is carbon steel. Incoloy-800 is used for the heater sheaths and Inconel-600 for other parts, such as the heat duct, which operate at high temperature.

There are no radiolytic or pyrolytic decomposition products from these materials. The carbon steel base of the recombiner unit is coated with a paint that satisfies the requirements of ANSI 101.2 (1972), "Protective Coating (Paints) for Light Water Nuclear Reactor Containment Facilities."

6.2.6 CONTAINMENT LEAKAGE TESTING

The Virgil C. Summer Nuclear Station containment and associated systems are designed to permit leakage testing in accordance with 10CFR50, Appendix J. 98-01

6.2.6.1 Containment Integrated Leakage Rate Test

The containment integrated leakage rate test is a Type A test as defined by 10CFR50, Appendix J. Integrated leakage rate testing is performed in accordance with a detailed, written, and approved procedure prepared in accordance with 10CFR50, Appendix J, 02-01 which satisfies the requirements of General Design Criteria 52, 53, and 54 (see Chapter 14).

6.2.6.1.1 Preoperational Type B and C Testing

Preoperational Type B and C leak testing is employed to detect leaks which may affect the integrity of the containment and the results of the initial integrated leakage rate test. These tests are performed at a pressure of not less than 45.8 psig.

The following areas are subjected to preoperational Type B or C tests, as applicable:

1. Containment penetrations (mechanical and electrical).

2. Equipment and personnel access hatches.

3. Fuel transfer tube.

4. Isolation valves in lines penetrating the containment boundary.

Any leaks detected are repaired as required.

6.2.6.1.2 Preoperational Testing - Integrated

An initial integrated leakage rate test of the Reactor Building is performed in accordance with 10CFR50, Appendix J, following completion of building construction, installation of all systems penetrating the containment boundary and after completion of Reactor Building inspection and structural integrity testing. This initial test is performed at the calculated peak containment internal pressure, P_a (47.1 psig). The absolute pressure-temperature method of measuring leakage from the Reactor Building, as described in ANSI N45.4-1972, Appendix B is employed during both of these leakage 02-01 rate tests. The objectives of the tests are as follows:

1. To establish the measured integrated leakage rate, L_{am}.

2. To obtain a leakage rate history and leakage characteristic of the containment system.

The leakage rate is determined by measuring the leakage from the Reactor Building over a period of at least 24 hours after Reactor Building temperature and pressure have been stable for a minimum of 4 hours. To ensure equal temperature distribution, throughout the building interior, fans are provided to circulate the air in the Reactor Building during the test. Verification of the leakage rate is obtained by supplemental testing as described in ANSI N45.4-1972, Appendix B. By measuring leakage from the 02-01
Reactor Building while a known leakage rate is superimposed on the normal building leakage rate, the difference between the total leakage and the superimposed known leakage results in the actual leakage rate. This leakage rate (difference) is compared with the original leakage rate as a check of its accuracy.

High accuracy instrumentation is provided to assure that building atmospheric temperature, pressure, and dewpoints are measured accurately.

Acceptance Criteria for this test are as follows:

1. The total measured containment leakage rate, L_{am}, at pressure, P_a, may not exceed O.75 L_a, the containment design basis leakage rate. L_a is 0.2% by weight of the contained atmosphere in 24 hours. If local leakage measurements are taken following repairs made to satisfy acceptance criteria, these measurements shall be taken at a test pressure, P_a.

In the event of failure of preoperational integrated testing to satisfy the acceptance outlined above, repairs and/or adjustments to equipment shall be made and a Type A test plus a supplemental test shall be performed. All repairs, adjustments to equipment, and test results shall be reported to the NRC as required by Section III.A.1.a of Appendix J to 10CFR50.

6.2.6.1.3 Periodic Testing - Integrated

Prior to December 1, 1996, Integrated Leakage Rate Testing was conducted in accordance with 10CFR50 Appendix J, Option A and ANSI N45.4 - 1972. Commencing with the implementation of Technical Specification Amendment 135, Integrated Leakage 98-01
Rate Testing will be conducted in accordance with 10CFR50 Appendix J, Option B as delineated in the Containment Leakage Rate Testing Program (CLRTP).

The performance of Type A tests shall be limited to periods when the plant is nonoperational and secured in the shutdown condition under the administrative control and in accordance with the safety procedure defined in the license.

6.2.6.1.4 Special Conditions During Testing

All containment penetrations except those in systems which are required to remain filled during and following a LOCA (i.e., Residual Heat Removal, Reactor Building Spray and Safety Injection Systems) will be tested in accordance with the Type A test requirements as described in the CLRTP. RN 03-035

6.2.6.1.5 Leakage Rate Test System Description

ECR 50504 removes existing leak rate components and installs a spool piece to provide a tie-in for vendor supplied equipment. The as-built system is a combination of vendor supplied and permanent plant components. The schematic system diagram is shown by Figure 6.2-59 and locations of temperature and humidity elements are shown by Figure 6.2-60. RN 03-043

The leakage rate test system consists of the applicable components, valves, instrumentation, controls, and associated piping required to accomplish the following:

1. Pressurize the Reactor Building with dry, compressed air.

2. Control the pressurization and blowdown of the Reactor Building.

3. Remove moisture and other impurities from the compressed air.

4. Protect against overpressurization.

5. Measure pressure, temperature and humidity of the Reactor Building atmosphere.

6. Superimpose a known leakage rate on the measured containment leakage rate.

RN 03-043

Instruments to monitor the pressure, temperature and dewpoint temperature of the contained air are mounted on a test panel located outside the Reactor Building. Sensors for these instruments are mounted in selected locations within the Reactor Building. Two (2) flow meters are also mounted on the test panel. These flow meters are used to superimpose a known leakage rate on the previously measured Reactor Building leakage rate. Additional temporary instrumentation may be installed to enhance the measurement of test parameters. 98-01

Two (2) 8 inch Reactor Building blowdown lines are provided for depressurization upon completion of the leakage rate tests. Associated piping, valves, instrumentation and controls are also provided.

Permanent components of the system are rated at 150 psig and 250°F. Piping penetrations and blind flanges are fabricated in accordance with the ASME Code, Section III [23], Class 2.

6.2.6.2 Containment Penetration Leakage Rate Test

Components which penetrate and seal the containment boundary with seals, gaskets or sealant compounds which are resilient; or piping penetrations fitted with an expansion bellows as the only barrier to leakage from containment are leak tested (Type B test) at periodic intervals in accordance the CLRTP during the lifetime of the plant to ensure their continuing integrity. 98-01

6.2.6.2.1 Applicability

This section discusses the various types of penetrations used in the Virgil C. Summer Nuclear Station and identifies which types are leakage tested and which do not require leakage testing.

6.2.6.2.1.1 Piping Penetrations and Spares

Table 6.2-53 is a listing of piping penetrations and spares. There are basically 4 types of penetration: spares, cold penetrations, hot penetrations, and specialty penetrations.

Spare penetrations are sealed by welding pipe caps to each end of the penetration sleeve. Since no resilient or flexible seals are used, these penetrations do not require Type B leakage tests.

Cold penetrations are sealed by a flat plate, of 1/2 inch or greater thickness, welded to both the sleeve and the process pipe at each end of the penetration sleeve. Since no resilient or flexible seals are used, these penetrations do not require Type B leakage tests.

Hot penetrations are sealed on the inside of containment by a flat plate in a manner similar to cold penetrations. On the outside, they are sealed by a single bellows, one end of which is attached to the penetration sleeve and the other to the process pipe. Since the containment barrier does not utilize a resilient or flexible seal, these penetrations do not require Type B leakage tests.

The specialty penetrations are No. 107, the fuel transfer tube; No. 309, is used to house an abandoned radiation detector and check source assembly originally assigned to RM-G7. Figure 6.2-60a illustrates details of this penetration. Nos. 327, 328, 329, and 425 are used for the recirculation sump line penetrations. The fuel transfer tube is sealed inside containment by a flat plate welded to both the penetration sleeve and to the fuel transfer tube. The fuel transfer tube closure design provides a double barrier and, therefore, constitutes the containment boundary and is tested. This penetration has a bellows seal in the refueling cavity, this bellows seal retains water within the cavity and is not a containment isolation seal. Similarly, a bellows seal is used in the transfer canal to retain water. Since no resilient or flexible seal is used for containment, this bellows seal does not require a Type B leakage test. The lines from the recirculation sumps are sealed by a 1/2 inch thick flat plate welded to both the penetration sleeve and the process pipe. Since no resilient or flexible seal is used for

containment isolation, no Type B leakage test is required. The outer end of the sleeve is attached to sump isolation valve containers which are leakage tested. The pipe to containment liner seal is leakage tested with the container.

For the reasons listed above, no piping penetrations are required to be subjected to Type B leakage tests. However, all piping penetrations, including those with a single bellows, have a design capability for Type B leakage testing. Other spare penetrations are as identified in Section 3.8.1.1.2.2. 02-01

6.2.6.2.1.2 Electrical Penetrations

The electrical penetrations are of the header plate type. The seal between the individual wires and the header plate is glass. This seal is neither resilient nor flexible. Therefore, no Type B leakage test is required. The header plate is bolted to the nozzle. The seal is formed by double O-ring gaskets. This type of seal is subjected to Type B leakage tests.

6.2.6.2.1.3 Personnel Airlocks and Equipment Hatch

There are 2 personnel airlocks and one equipment hatch that penetrate the containment boundary. There is a 16 foot diameter equipment hatch, a 9 foot diameter personnel airlock, and a 5 foot diameter personnel emergency airlock. In the case of these penetrations, the barrel is solidly welded to the Reactor Building liner. The cover door on the equipment hatch is sealed to the barrel assembly with a double O-ring seal. Both the inside and outside doors of both personnel airlocks are sealed with double O-ring seals. The personnel airlocks also have operating shafts which penetrate the inner and outer barriers of these penetrations. These barrier penetrations are also sealed with double O-ring seals. The double O-ring seals are subjected to Type B leakage tests.

The personnel airlock and emergency airlock are shown by Figures 3.8-19 through 3.8-20b. All mechanical and electrical penetrations through the atmosphere and containment bulkheads of each airlock are shown by these figures.

Each airlock design provides test connections on the atmosphere side of the atmosphere bulkhead for use in leak testing the entire airlock interspace at design pressure, P_a.

Clamps have been provided for the containment bulkhead door of each airlock. These clamps are installed from within the airlock and must be in place when testing the airlock interspace to P_a as defined in the CLRTP. The clamps are installed by sliding them from the stored position into the clamping position. For testing the personnel and emergency airlocks at P_a, test clamp bolts will be torqued in accordance with the manufacturer's recommendations as specified in surveillance procedures. 98-01

Sufficient clamps have been provided around the door perimeter to assure uniform door sealing. The design has accounted for the bolt forces generated from torquing as well as the design pressure, P_a, multiplied by 1.15 . 98-01

For local leak testing, a test panel is located on the atmosphere side of the atmosphere bulkhead of each airlock. Separate test connections are provided on each test panel for individual pressurization between the double seals for all handwheel shafts and between the double seals for each door.

Handwheel shaft seals are pressurized and leak tested at a pressure greater than or equal to 8 psig, for at least 3 minutes.

Double door seals are pressurized and leak tested at a pressure greater than or equal to 8 psig, for at least 3 minutes.

The door seals for each airlock have been designed for leak testing between the double seals at a maximum pressure of 10 psig when using only the normal door latching mechanism. 98-01

Testing the airlocks at a minimum of 8 psig will verify proper latching of the doors and 98-01 seating of the double seals, as well as indicating any seal degradation detrimental to leak tightness.

6.2.6.2.2 Procedures

This section discusses the procedures for conducting Type B leakage tests, including test pressure and acceptance criteria.

6.2.6.2.2.1 Piping Penetrations and Spares

As discussed in Section 6.2.6.2.1.1, the only pressure testing of the piping penetrations is testing of the sump isolation valve containers and the process piping that is isolated by gasketed, blind flanges.

The sump isolation valve containers are tested by pressurizing the containers and guard pipes with compressed air through a test connection outside the Reactor Building and monitoring pressure decay.

The blind flange that seals the fuel transfer tube has a double gasketed seal. The collar on the transfer tube that mates with the flange is drilled with passageways that allow pressurization between the gaskets. Because of the small volume between the gaskets, the pressure decay method is impracticable. The test is performed by measuring makeup flow into the test connection.

The leakage rate test system piping is sealed inside and outside the Reactor Building with blind flanges. The gaskets on these flanges are tested by pressurizing the piping between the flanges through test connections on the piping and measuring makeup flow into the test connection.

6.2.6.2.2.2 Electrical Penetrations

As discussed in Section 6.2.6.2.1.2, only the O-ring seals in the mechanical joints on some of the electrical penetrations are subjected to Type B leakage testing. These seals are tested by pressurizing the space between the O-rings and either performing a 98-01
pressure drop test or by measuring the makeup flow into the test connection.

6.2.6.2.2.3 Personnel Airlock and Equipment Hatch

Type B leakage tests are performed on these 3 penetrations by pressurizing the space between the double resilient seals and measuring the makeup flow into the test connection.

6.2.6.2.3 Test Pressure

Type B leakage tests are conducted using a pressure not less than P_a as defined in the 98-01
CLRTP.

6.2.6.2.4 Acceptance Criteria

The combined leakage rate for all penetrations and valves subjected to Type B and C tests shall be in accordance with the CLRTP. 98-01

6.2.6.3 Containment Isolation Valve Leakage Rate Test

Periodic tests (Type C tests) are conducted in accordance with the CLRTP, to 98-01
determine the operability and leakage rate characteristics of valves serving an isolation function in the testable fluid systems lines penetrating the Reactor Building.

The Engineered Safety Features Actuation System test circuitry provides the means for testing isolation valve operability.

6.2.6.3.1 Applicability

A list of all piping penetrations is included as Table 6.2-53. A column of this table indicates which penetrations are subjected to Type B leakage tests. Table 6.2-53a indicates containment isolation valves subject to Type C leakage tests. The containment isolation valves associated with the secondary side are not tested.

The boundary or barrier against leakage of containment atmosphere to the environment is the inside of the steam generator tubes, the outside of the steam generator shell, and the outside of all lines connected to the steam generator shell side. After a loss of coolant accident, these components form the actual barrier against release of the containment atmosphere to the environment.

Lines which emanate from the steam generator shell side will not be damaged during a loss of coolant accident. The design of these components is at least equivalent to, or better than, the Reactor Building liner with respect to quality assurance, pressure, temperature, testing, missile protection, etc. Thus, after a loss of coolant accident, components will be available and can be relied upon to limit release of radioactivity in the same manner as the Reactor Building liner.

In the event that the plant has been operating with steam generator tube leaks at the time of a loss of coolant accident, the barrier against release is as previously described. For large breaks, when an orderly shutdown cannot be executed, a plant trip will occur and the Emergency Core Cooling System will be started. Reactor Coolant System pressure will decrease at a rate dependent upon break size. Some time after the incident, containment atmosphere will enter the steam generator tube bundle through the pipe break. During this initial period (approximately 1/2 to 6 hours after the incident) the steam generator shell side is pressurized. Also, after the short term, the barrier is maintained by a head of water above the tube bundle.

Before, during, and after the initial relief of steam through the safety valves, radioactivity leakage is restricted by the barrier at the steam generator tubes, shell, steam lines, blowdown lines, and sample lines inside the Reactor Building.

In general, test connections are provided to permit pressurization of piping inside the inside isolation valve and a test vent is provided outside of the outside isolation valve to measure leakage through the valves. The test is conducted by connecting the pressure source to a test connection, aligning the valves so that there is only one closed valve between the test connection and test vent and measuring the leakage flow. In those cases where the inside isolation valve is a check valve, a second test connection is provided between the isolation valves to permit pressurizing the outside isolation valve. Where required, the test connection is attached to a blind flange which is attached to the piping inside the Reactor Building (e.g., Reactor Building purge valves).

The selection of the number and location of test connections and vents for each isolation valve to be subjected to Type C leakage testing is based upon satisfying the requirement that the test pressure be applied in the same direction as the pressure existing when the valve is performing its safety function, unless it can be determined that the results for a pressure applied in a different direction will provide equivalent or more conservative results. These valves are listed in Table 6.2-53a. The test media for XVG-08889-SI is directed between the disc. This method is identified per ANS 56.8 as an equivalent and conservative method of leak rate testing. RN 99-105

As previously discussed, test connections and vents have been incorporated into the design of piping penetrations that contain isolation valves. These connections, which permit testing of those isolation valves subject to Type C leakage testing, are shown schematically on the appropriate system flow diagrams. Leakage testing will be performed in such a manner that measurement of leakage past the valve seat will be from inside to outside containment except as listed in Table 6.2-53a. 98-01

6.2.6.3.2 Test Pressure

Type C leakage tests are conducted at a pressure not less than P_a as defined in the CLRTP. 98-01

6.2.6.3.3 Acceptance Criteria

See Section 6.2.6.2.4.

6.2.6.4 Scheduling and Reporting of Periodic Tests

6.2.6.4.1 Test Frequencies

6.2.6.4.1.1 Type A Tests

Test frequency shall be as established by the CLRTP. 98-01

6.2.6.4.1.2 Type B Tests

Test frequency shall be as established by the CLRTP. 98-01

6.2.6.4.1.3 Type C Tests

Test frequency shall be as established by the CLRTP. 98-01

6.2.6.4.2 Reports

The preoperational test summary technical report was submitted to the NRC approximately 3 months after the performance of the Type A test. 98-01

The report of the preoperational test included a schematic arrangement of the leakage rate measurement system, the instrumentation used and the test program selected as applicable to the preoperational test and subsequent periodic tests. The report contained an analysis and interpretation of the leakage rate test data for the Type A test results to the extent necessary for demonstration of the acceptability of the containment leakage rate in satisfying the acceptance criteria. 98-01

Periodic test results are documented and available for review in accordance with the CLRTP. 98-01

6.2.6.5 Special Testing Requirements

Leakage rate testing subsequent to containment modification is performed in accordance with the CLRTP. 98-01

Virgil C. Summer Nuclear Station does not utilize a subatmospheric containment, secondary containment, or pressure seal system.

6.2.7 REFERENCES

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TABLE 6.2-1

GENERAL CONTAINMENT DESIGN AND EVALUATION PARAMETERS

General Design Information		
Maximum Internal Design Pressure, psig		57
Maximum External Design Pressure, psig		3.5 psig
Design temperature, °F		283(1)
Free Volume, ft^3		$1.84x10^6$
Design Leak Rate, max. allowable, %/day		0.2

Engineered Safety Features	Full Capacity	Value Used for Analysis	
Passive Safety Injection			
Number of Accumulators	3	2 or 3	
Pressure Setpoint, Psig	600	600	
Active Safety Injection			
Residual Heat Removal Flow Rate, lb/sec	990.4	460.2 or 990.4	02-01
Reactor Building Spray System			
Number of Lines	2	1 or 2	
Number of Pumps	2	1 or 2	
Number of Headers	6	3 or 6	
Design Flow, gpm	5,000	2,500 or 5,000	02-01
Reactor Building Cooling Units			
Number	4	1 or 2	
Air Side Flow Rate, cfm/unit	60,270(3)	54,200(3)	RN 03-003
Heat Removal Rate at 283°F, Btu/hr/unit	$125x10^6$	$90.7x10^6$ (2)	
Residual Heat Removal System Heat Exchangers			
Number	2	1	
UA, BTU/hr-°F/unit	$1.519x10^6$ (4)	$1.519x10^6$ (4)	RN 03-003
Flow Rate			
Shell Side, gpm	5,600	5,600	02-01
Tube Side, gpm	3,750	3,750	

See Notes (1) through (4) on next page

Notes:

1. Peak reactor building temperature is 373.7°F as described in Section 6.2.1.1.1
2. Maximum analysis value assumed was an energy removal rate of 90.7 x10^6 Btu/Hr, at 283°F. RN 03-003
3. This parameter is used only in the Chapter 15 Radiological Consequence Analysis for particulate iodine removal post-LOCA. This parameter is not used in the Chapter 6 pressure/temperature analyses.
4. Based on the maximum component cooling water shell-side inlet temperature of 120° F [47]. RN 03-003

TABLE 6.2-1a

MAIN STEAM LINE BREAKS ANALYZED

Double Ended Ruptures(1) (No Entrainment) to Assess Impact of Single Failures

Area/Power	Failure(s) Assumed(2)
1.4 ft^2/ 102%	MSIV(3)
1.4 ft^2/ 102%	FWIV(4)
1.4 ft^2/ 102%	EFW FCV(5)
1.4 ft^2/ 102%	Diesel(6)
1.4 ft^2/ 102%	CH-A(7)

Double Ended Ruptures (with Entrainment) to Assess Impact of Power Level

Area/Power	Failures Assumed(2)
1.4 ft^2/ 75%	CH-A
1.4 ft^2/ 50%	CH-A
1.4 ft^2/ 25%	CH-A
1.4 ft^2/ 25%	FWIV(4)
1.4 ft^2/ 0%	CH-A

Double Ended Ruptures (No Entrainment) to Assess Impact of Break Size

Area/Power	Failures Assumed(2)
1.2 ft^2/ 102%	FWIV, MSIV, EFW FCV, Diesel
1.1 ft^2/ 102%	FWIV, MSIV, EFW FCV, Diesel

Small Double Ended Ruptures (with Entrainment)

Area/Power	Failures Assumed(2)
1.1 ft^2/ 75%	FWIV, MSIV, EFW FCV, Diesel
0.8 ft^2/ 50%	FWIV, MSIV, EFW FCV, Diesel
0.6 ft^2/ 25%	FWIV, MSIV, EFW FCV, Diesel
0.2 ft^2/ 0%	FWIV, MSIV, EFW FCV, Diesel

TABLE 6.2-1a (Continued)

MAIN STEAM LINE BREAKS ANALYZED

Small Double Ended Ruptures (No Entrainment)

Area/Power	Failures Assumed[(2)]
1.0 ft^2/ 75%	FWIV, MSIV, EFW FCV, Diesel
0.7 ft^2/ 50%	FWIV, MSIV, EFW FCV, Diesel
0.5 ft^2/ 25%	FWIV, MSIV, EFW FCV, Diesel
0.1 ft^2/ 0%	FWIV, MSIV, EFW FCV, Diesel

Split Ruptures (No Entrainment)

Area/power	Failures Assumed[(2)]
0.878 ft^2/ 102%	
0.871 ft^2/ 75%	FWIV, EFW FCV, Diesel
0.863 ft^2/ 50%	FWIV, EFW FCV, Diesel
0.849 ft^2/ 25%	FWIV, EFW FCV, Diesel
0.772 ft^2/0%	FWIV, EFW FCV, Diesel

Notes:

1. Effective break area for broken loop is 1.4 ft^2.
2. Failure of one Chg/SI pump is assumed in all analyses.
3. MSIV = Main steam isolation valve fails to close.
4. FWIV = Main feedwater isolation valve fails to close.
5. EFW FCV = EFW control valve which terminates flow to the faulted SG fails to close.
6. Diesel = One emergency diesel fails to start.
7. CH-A = Failure of Electrical Channel A.

TABLE 6.2-2

INITIAL CONDITIONS USED IN REACTOR BUILDING PEAK PRESSURE ANALYSIS

Reactor Coolant System		
Power Level, MWt (Includes +2% Allowance for Instrument Error and Deadband)	2958	02-01
Vessel Average Temperature, °F (Includes +5.3° F Allowance for Instrument Error and Deadband)	592.7	02-01
Mass of Reactor Coolant, lbm	421.3×10^3	
Reactor Coolant Energy,(1) BTU	251.3×10^6	
Reactor Coolant Pressure, psia	2300	
Reactor Building		
Pressure, psig(2)	1.5	
Temperature, °F	120	
Relative humidity, %	30	
Service Water Temperature °F	95	98-01
Refueling Water Temperature, °F	95	
Stored Water Quantity Assumed in Analysis		
Refueling Water Storage Tank, gal	404,000	
Accumulators, lbm	188,130	RN 02-009

(1) Energy relative to 32°F.

(2) Peak pressure analysis; peak temperature analysis uses pressure of 0.0 psig.

TABLE 6.2-3

COMPARATIVE RESULTS, PEAK PRESSURE AND TEMPERATURE

FOR A SPECTRUM OF ACCIDENTS

LARGE MSLBs

BREAK SIZE FT^2 / % POWER	1.4 / 102%	1.4 / 102%	1.4 / 102%	1.4 / 102%	1.4 / 102%	1.4 / 75%	1.4 / 50%	1.4 / 25%	1.4 / 25%	1.4 / 0%
	(CASE IG)	(CASE IH)	(CASE II)	(CASE IJ)	(CASE IK)	(CASE 2E)	(CASE 3E)	(CASE 4E)	(CASE 4G)	(CASE 5F)
RB SPRAY	2	2	2	1	1	1	1	1	2	1
RBCU'S	2	2	2	1	1	1	1	1	2	1
OTHER FAILURES	FWIV SI PUMP	MISV SI PUMP	EFW FCV SI PUMP	DIESEL	CH-A	CH-A	CH-A	CH-A	FWIV SI PUMP	CH-A
PEAK PRESSURE (PSIG)	50.4	48.1	47.9	48.2	49.3	49.6	50.9	53	52.6	51.7
TIME TO PEAK PRESSURE (SEC)	204	177	185	173	1,800	1,800	1,800	1,800	193	1,900
PEAK TEMPERATURE (°F)	360.8	372.7	360.9	361	352.1	283.7	275.4	278.1	277.8	276.4
TIME TO PEAK TEMPERATURE (SEC)	31	19	31	30	53	4	1,800	1,800	193	1,900

02-01

TABLE 6.2-3 (Continued)

SMALL MSLB'S WITHOUT ENTRAINMENT

BREAK SIZE FT^2 / % POWER	1.2/102% (CASE IB)	1.1/102% (CASE IC)	1.0/75% (CASE 2C)	0.7/50% (CASE 3C)	0.5/25% (CASE 4C)	0.1/0% (CASE 5C)
RB SPRAY	1	1	1	1	1	1
RBCU'S	1	1	1	1	1	1
OTHER FAILURES	(Note 1)	(Note 1)	(Note 1)	(Note 1)	(Note 1)	(Note 1)
PEAK PRESSURE (PSIG)	51.6	50.3	50.2	51.8	49.7	13.7
TIME TO PEAK PRESSURE (SEC)	1,200	1,200	1,200	1,200	1,200	1,200
PEAK TEMPERATURE (°F)	350.5	347.2	341.1	322	303.8	251.6
TIME TO PEAK TEMPERATURE (SEC)	49	53	53	58	67	480

02-01 02-01 02-01 02-01

Note: 1. Multiple failures assumed: MSIV, FWIV, EFW FCV, and Diesel. 02-01

TABLE 6.2-3 (Continued)

SMALL MSLB'S WITH ENTRAINMENT 02-01

BREAK SIZE FT^2/% POWER	1.1/75% (CASE 2B)	0.8/50% (CASE 3B)	0.6/25% (CASE 4B)	0.2/0% (CASE 5B)
RB SPRAY	1	1	1	1
RBCU'S	1	1	1	1
OTHER FAILURES	(Note 1)	(Note 1)	(Note 1)	(Note 1)
PEAK PRESSURE (PSIG)	51.3	48.7	49.9	25.9
TIME TO PEAK PRESSURE (SEC)	1,200	1,200	1,200	1,200
PEAK TEMPERATURE (°F)	276	272.4	274.1	232.8
TIME TO PEAK TEMPERATURE (SEC)	1,200	1,200	1,200	1,200

02-01

02-01

Note: 1. Multiple failures assumed: MSIV, FWIV, EFW FCV, and Diesel.

TABLE 6.2-3 (Continued)

SPLIT MSLB'S WITHOUT ENTRAINMENT

BREAK SIZE FT^2/% POWER	0.878/102%	0.871/75%	0.863/50%	0.849/25%	0.772/0%	02-01
	(CASE IB)	(CASE 2B)	(CASE 3B)	(CASE 4B)	(CASE 5B)	02-01
RB SPRAY	1	1	1	1	1	
RBCU'S	1	1	1	1	1	
OTHER FAILURES	(Note 1)	(Note 1)	(Note 1)	(Note 1)	(Note 1)	
PEAK PRESSURE (PSIG)	35	35	38.1	37.8	35.7	
TIME TO PEAK PRESSURE (SEC)	230	280	320	245	400.0	
PEAK TEMPERATURE (°F)	331.9	329.5	327.9	325.9	318.1	02-01
TIME TO PEAK TEMPERATURE (SEC)	67	67	67	67	67	

Note: 1. Multiple failures assumed: MSIV, FWIV, EFW FCV, and Diesel. 02-01

TABLE 6.2-3 (Continued)

COMPARATIVE RESULTS, PEAK PRESSURE AND TEMPERATURE
FOR A SPECTRUM OF ACCIDENTS

Large LOCA

BREAK SIZE FT^2/% POWER	DEPS[1]/102	DEPS/102	DEHL[2]/102	02-01
R B SPRAY	1	1	N/A	
RBCU'S	1	1	N/A	
SAFETY INJECTION	MIN	MAX	N/A	
PEAK PRESSURE (PSIG)	43.7	43.7	45.1	
TIME TO PEAK PRESSURE (SEC)	18	18	15	
PEAK TEMPERATURE (°F)	265.4	265.4	267.4	02-01
TIME TO PEAK TEMPERATURE (SEC)	18	18	15	
ENERGY ABSORBED BY PASSIVE HEAT SINKS AT TIME CF PEAK PRESSURE Btu x 10^6	23.6	23.6	22.1	98-01

Notes: 1. DEPS = Double-Ended Pump Suction
2. DEHL = Double-Ended Hot Leg

TABLE 6.2-4

LONG TERM VALUES OF REACTOR BUILDING PARAMETERS
FOR LOSS OF COOLANT ACCIDENT AND
MAIN STEAM LINE BREAK

	Design Basis [1] Accident Main Steam Line Break	LOCA [1] DEPS with Minimum Safety Injection	
Reactor Building Pressure, psig	38.3	1.8	RN 03-003
Reactor Building Vapor Temperature, °F	257	107	
Reactor Building Sump Temperature, °F	270	132	

Notes:

1. Long-term MSLB = 1800 Seconds
   Long-term LOCA = 1440 Hours RN 03-003

TABLE 6.2-5

CHRONOLOGY OF EVENTS FOR THE DESIGN BASIS ACCIDENT MAIN STREAM LINE BREAK [1]

		Time (Sec.)
1.	Break occurs, which blows down all three steam generators, coincident with Electrical Channel A Failure [2]	0.0
2.	Reactor building pressure reaches the safety injection setpoint.	1.0
3.	Feedwater pumps tripped (not assumed in analysis)	N/A
4.	Main steam isolation valves close (blowdown limited to one steam generator)	7.0
5.	Reactor building spray initiated	53.1
6.	Reactor building cooling units fully operational	86.5
7.	Peak pressure of 53.0 psig is reached	1800
8.	Emergency feedwater flow manually terminated	1800
9.	End of Analysis for DBA	1800.0

02-01

Notes: (1) 1.4 ft^2 DE Rupture from 25% power.

(2) Electrical Channel A Failure results in loss of power to the feedwater control and isolation valves, main steam isolation valves, prevents startup of an emergency diesel generator, and prevents automatic isolation of emergency feedwater to the faulted SG.

TABLE 6.2-6

CHRONOLOGY OF EVENTS FOR LOCA
(DEPS, Minimum Safety Injection)

		Time (Sec.)	
1.	Break occurs	0.0	
2.	Peak pressure of 43.7 psig is reached	18.0	
3.	Primary system blowdown complete	19.6	
4.	Reactor Building spray begins	52.0	
5.	Reactor building cooling units actuated	86.5	
6.	Recirculation begins	2499.0	
7.	Reactor building spray system and cooling unit operation is terminated	3.456×10^6	RN 03-003
8.	End of Analysis	5.184×10^6	

TABLE 6.2-7

PASSIVE HEAT SINKS

Description	Material of Each Layer	Thickness of Layer [1] (ft)	Steel Weight[2] (lbm)	Steel Temp (°F)	Area[3] (ft^2)	Exposure[4] (ft^2)	
							02-01
Reactor Building Concrete and steel							
Reactor Building Shell	Concrete	4			57,397(5)	0	
Reactor Building Dome	Concrete	3			20,241	0	
Floor Slab (el.412')	Concrete	4			8,868	8,868	
Floor Slab (el.436')	Concrete	2			5,019	5,019	
Floor Slab (el.463')	Concrete	2			6,675	6,675	
Decking under Floor Slabs at el. 436' and el.463'	Carbon Steel	1/8 in	44,315	Ambient	8,696	8,696	
Secondary Shield Walls between el. 412' and 436'	Concrete	3.66			6,168	11,919	
Secondary Shield Walls between el. 436' and 463'	Concrete	3.66			7,523	14,492	02-01
Secondary Shield Walls above el. 463'	Concrete	3.66			8,613	17,126	
Misc. Walls and Shield Slabs between el. 412' and 436'	Concrete	2			2,581	5,162	
Misc. Walls between el. 436' and el. 463'	Concrete	2			737	1,473	
Misc. Walls above el. 463'	Concrete	2			808	1,616	
Primary Shield Wall and Fuel Transfer Canal Walls between el. 412' and el. 436'	Concrete	8			2,729	5,457	
Primary Shield Wall and Fuel Transfer Canal Walls at el. 436'	Concrete	4.5			2,348	4,695	

TABLE 6.2-7 (Continued)

PASSIVE HEAT SINKS

Description	Material of Each Layer	Thickness of Layer [1] (ft)	Steel Weight[2] (lbm)	Steel Temp (°F)	Area[3] (ft^2)	Exposure[4] (ft^2)	
							02-01
Reactor Building Concrete and steel (Continued)							
Secondary Shield Slab	Concrete	2			1,280	2,560	
Reactor Building Liner	Carbon Steel	0.26 in	954,450[6]	Ambient	77,638	77,638	02-01
Liner Anchors	Carbon Steel	0.25 in	390,600[6]	Ambient	36,943	0	02-01
Fuel Transfer Canal Liners	Stainless Steel	0.266 in	131,250[6]	Ambient	11,497	11,497	02-01
Column Steel	Carbon Steel	1.55 in	212,000	Ambient	3,350	6,616	02-01
Framing Steel, Platforms, Stairs, Restraints	Carbon Steel	0.5 in	969,000	Ambient	47,281	88,887	
Special Steel Around Reactor in Primary Wall	Carbon Steel	1.9	305,000	Ambient	3,930	4,249	
Brackets	Carbon Steel	0.936 in	128,000	Ambient	3,349	3,339	
Airlocks and Hatch	Carbon Steel	0.87	45,000	Ambient	1,266	1,266	
Piping, Piping Supports and Equipment							
Spent Fuel Cooling System							
Sch 40 Pipe, 2 in	Stainless Steel	0.013	110	Ambient	19	19	
Sch 40 Pipe, 3 in	Stainless Steel	0.018	834	Ambient	101	101	
Leak Test Canal System							
Sch 40 Pipe, 1/2 in	Stainless Steel	0.009	26	Ambient	7	7	
OD Tubing, 1/2 in	Stainless Steel	0.005	24	Ambient	10	10	

TABLE 6.2-7 (Continued)

PASSIVE HEAT SINKS

Description	Material of Each Layer	Thickness of Layer (1) (ft)	Steel Weight(2) (lbm)	Steel Temp (°F)	Area(3) (ft^2)	Exposure(4) (ft^2)	
							02-01
Piping, Piping Supports and Equipment (Continued)							
R.B. Spray System							
Sch 40 Pipe, 3/4 in	Stainless Steel	0.009	11	Ambient	3	3	
Sch 40 Pipe, 1 in	Stainless Steel	0.011	544	Ambient	113	113	
Sch 40 Pipe, 2 in	Stainless Steel	0.013	37	Ambient	6	6	
Sch 40 Pipe, 3 in	Stainless Steel	0.018	2,274	Ambient	275	275	02-01
Sch 40 Pipe, 6 in	Stainless Steel	0.024	23,333	Ambient	2,133	2,133	
Sch 40 Pipe, 10 in	Stainless Steel	0.030	8,096	Ambient	563	563	
Safety Injection System							
Sch 140 Pipe, 12 in	Stainless Steel	0.094	23,043	Ambient	551	551	
Sch 40 Pipe, 12 in	Stainless Steel	0.034	1,338	Ambient	83	83	
Sch 150 Pipe, 3 in	Stainless Steel	0.036	501	Ambient	32	32	
Sch 160 Pipe, 2 in	Stainless Steel	0.029	4,690	Ambient	392	392	
Sch 40 Pipe, 2 in	Stainless Steel	0.013	110	Ambient	19	19	
Sch 160 Pipe, 1 in	Stainless Steel	0.021	924	Ambient	112	112	
Sch 40 Pipe, 1 in	Stainless Steel	0.011	285	Ambient	58	58	
Sch 160 Pipe, 3/4 in	Stainless Steel	0.018	620	Ambient	89	89	
Sch 40 Pipe, 1 in	Carbon Steel	0.011	445	Ambient	91	91	
Nuclear Utility System							
Sch Std Pipe, 1/2 in	Carbon Steel	0.009	204	Ambient	53	53	
Sch 40 Pipe, 1/2 in	Stainless Steel	0.009	247	Ambient	64	64	
Sch Std Pipe, 3/4 in	Carbon Steel	0.009	90	Ambient	22	22	
Sch 40 Pipe, 3/4 in	Stainless Steel	0.009	79	Ambient	19	19	
Sch Std Pipe, 1 in	Carbon Steel	0.011	1,595	Ambient	327	327	
Sch 40 Pipe, 1 in	Stainless Steel	0.011	487	Ambient	100	100	
Sch Std Pipe, 1-1/2 in	Carbon Steel	0.012	49	Ambient	9	9	
Sch Std Pipe, 2 in	Carbon Steel	0.013	877	Ambient	149	149	02-01
Sch 40 Pipe, 2-1/2 in	Carbon Steel	0.017	1,448	Ambient	188	188	
Sch 40 Pipe, 4 in	Carbon Steel	0.020	2,482	Ambient	271	271	
Sch 40 Pipe, 6 in	Carbon Steel	0.024	95	Ambient	9	9	

TABLE 6.2-7 (Continued)

PASSIVE HEAT SINKS

Description	Material of Each Layer	Thickness of Layer [1] (ft)	Steel Weight[2] (lbm)	Steel Temp (°F)	Area[3] (ft^2)	Exposure[4] (ft^2)	
							02-01
Piping, Piping Supports and Equipment (Continued)							
Nuclear Sampling System							
OD Tubing, 3/8 in	Stainless Steel	0.005	105	Ambient	48	48	02-01
Sch 40 Pipe, 3/4 in	Stainless Steel	0.009	23	Ambient	6	6	
Reactor Building Leak Rate Test System							
Sch 40 Pipe, 8 in	Carbon Steel	0.027	143	Ambient	11	11	02-01
ECCS Check Valve Testing System							
Sch 160 Pipe, 3/4 in	Stainless Steel	0.018	1,027	Ambient	147	147	02-01
Nuclear Drains System							
Sch 40 Pipe, 3 in	Stainless Steel	0.018	303	Ambient	37	37	
Sch 40 Pipe, 2 in	Stainless Steel	0.013	913	Ambient	155	155	
Sch 40 Pipe, 1/2 in	Stainless Steel	0.009	17	Ambient	4	4	
Post Accident Hydrogen Removal System							
Sch 40 Pipe, 3 in	Stainless Steel	0.018	2,426	Ambient	293	293	02-01
Floor & Equipment Drains							
Sch 10 Pipe, 8 in	Stainless Steel	0.013	630	Ambient	106	106	
Sch 10 Pipe, 6 in	Stainless Steel	0.011	2,685	Ambient	501	501	
Sch 10 Pipe, 4 in	Stainless Steel	0.010	7,349	Ambient	1,544	1,544	
Sch 10 Pipe, 2-1/2 in	Stainless Steel	0.010	1,077	Ambient	230	230	
Sch 10 Pipe, 2 in	Stainless Steel	0.009	92	Ambient	22	22	
Sch 10 Pipe, 1 in	Stainless Steel	0.009	1	Ambient	1	1	

TABLE 6.2-7 (Continued)

PASSIVE HEAT SINKS

Description	Material of Each Layer	Thickness of Layer [1] (ft)	Steel Weight[2] (lbm)	Steel Temp (°F)	Area[3] (ft^2)	Exposure[4] (ft^2)	
							02-01
Piping, Piping Supports and Equipment (Continued)							
Hanger Components and Supplementary Steel for Hanger Components	Carbon Steel	0.019	135,507	Ambient	14,555	14,555	
Polar Crane	Carbon Steel	0.198	205,000	Ambient	2,113	22,500	
Reactor Cavity Manipulator Crane	Carbon Steel	0.556	74,104	Ambient	3,000	3,000	
Fuel Transfer Control Panel	Carbon Steel	0.292	4,865	Ambient	34	34	
CRDM Shroud	Stainless Steel	0.015	2,896	Ambient	394	394	
CRDM Missile Shield	Concrete	1.500		Ambient	315	315	02-01
Service Water System [7]							
Sch Std Pipe, 16 in	Carbon Steel	0.034	19,400	95	1,298	1,298	
Sch 40 Pipe, 10 in	Carbon Steel	0.030	10,930	95	760	760	
Sch Std Pipe, 1 in	Carbon Steel	0.011	40	95	24	24	
Sch 40 Pipe, 3/4 in	Carbon Steel	0.009	9	95	8	8	
Reactor Makeup Water							
Sch 40 Pipe, 3 in	Stainless Steel	0.018	834	120	101	101	
Sch 40 Pipe, 3/4 in	Stainless Steel	0.009	418	120	103	103	

TABLE 6.2-7 (Continued)

PASSIVE HEAT SINKS

Description	Material of Each Layer	Thickness of Layer [1] (ft)	Steel Weight[2] (lbm)	Steel Temp (°F)	Area[3] (ft^2)	Exposure[4] (ft^2)	02-01
Piping, Piping Supports and Equipment (Continued)							
Reactor Coolant System							
OD Tubing, 3/8 in	Stainless Steel	0.005	2	120	1	1	
Sch 40 Pipe, 3/4 in	Stainless Steel	0.009	6	120	1	1	
Sch 40 Pipe, 1 in	Stainless Steel	0.011	84	120	17	17	
Sch 40 Pipe, 3 in	Stainless Steel	0.018	834	120	101	101	
Sch 40 Pipe ,4 in	Stainless Steel	0.020	2,644	120	289	289	
Sch 40 Pipe, 6 in	Stainless Steel	0.024	2,466	120	225	225	
Sch 160 Pipe, 6 in	Stainless Steel	0.059	277	120	9	9	
Sch 40 Pipe, 12 in	Stainless Steel	0.034	4,550	120	284	284	
Chemical and Volume Control System[7]							
Sch 160 Pipe, 3/4 in	Stainless Steel	0.018	39	130	6	6	
Sch 160 Pipe, 1-1/2 in	Stainless Steel	0.023	1,118	130	115	115	
Sch Pipe, 3 in	Stainless Steel	0.037	1,862	130	119	119	
Component Cooling System[7]							
Sch 40 Pipe, 1 in	Carbon Steel	0.011	319	135	65	65	
Sch 40 Pipe, 3/4 in	Carbon Steel	0.009	17	135	4	4	
Sch 40 Pipe, 1-1/2 in	Carbon Steel	0.012	734	135	135	135	
Sch 40 Pipe, 2 in	Carbon Steel	0.013	37	135	6	6	
Sch 40 Pipe, 3 in	Carbon Steel	0.018	2,880	135	348	348	
Sch 40 Pipe, 4 in	Carbon Steel	0.020	1,187	135	130	130	
Sch 40 Pipe, 6 in	Carbon Steel	0.024	13,279	135	1,214	1,214	02-01
Sch 40 Pipe, 8 In	Carbon Steel	0.027	2,998	135	237	237	
Sch 160 Pipe, 1-1/2 in	Carbon Steel	0.023	1,166	135	120	120	
Sch Std Pipe, 1/2 in	Carbon Steel	0.009	298	135	77	77	
Sch Std Pipe, 1 in	Carbon Steel	0.011	151	135	31	31	

TABLE 6.2-7 (Continued)

PASSIVE HEAT SINKS

Description	Material of Each Layer	Thickness of Layer [1] (ft)	Steel Weight[2] (lbm)	Steel Temp (°F)	Area[3] (ft^2)	Exposure[4] (ft^2)	
							02-01
Piping, Piping Supports and Equipment (Continued)							
Waste Disposal System[7]							
OD Tubing, 3/8 in	Stainless Steel	0.005	2	160	1	1	
Sch 40 Pipe, 3/4 in	Stainless Steel	0.009	102	160	25	25	
Sch 40 Pipe, 1/2 in	Stainless Steel	0.009	204	160	53	53	
Sch 40 Pipe, 1 in	Stainless Steel	0.011	84	160	17	17	
Sch 40 Pipe, 1-1/2 in	Stainless Steel	0.012	571	160	105	105	02-01
Sch 40 Pipe, 2 in	Stainless Steel	0.013	475	160	81	81	
Sch 160 Pipe, 2 in	Stainless Steel	0.028	893	160	75	75	
Sch 40 Pipe, 3 in	Stainless Steel	0.018	1,364	160	165	165	
Sch 160 Pipe, 3 in	Stainless Steel	0.037	1,002	160	65	65	
Sch 40 Pipe, 4 in	Stainless Steel	0.044	216	160	24	24	
Feedwater System[7]							
Sch 80 Pipe, 1 in	Carbon Steel	0.015	11	200	2	2	
Reactor Coolant System							
Sch 160 Pipe, 3/4 in.[8]	Stainless Steel	0.018	203	653	29	29	02-01
Reactor Coolant Drain Tank	Stainless Steel	0.043	2,000	120	93	93	
Reactor Coolant Drain Tank Heat Exchanger[8]	Stainless Steel	0.030	1,325	250	87	87	02-01
Reactor Coolant Drain Tank Pump[7]	Stainless Steel	0.05	720	200	30	30	02-01
Accumulator[8]	Stainless Steel	0.117	168,000	300	2,878	2,878	
Reactor Coolant Pumps Relief Tank[8]	Stainless Steel	0.071	23,000	340	650	650	02-01
CRDM[8]	Stainless Steel	0.152	114,000	400	1,527	1,527	
Stud Tensioners	Carbon Steel	0.314	12,000	Ambient	78	78	

TABLE 6.2-7 (Continued)

PASSIVE HEAT SINKS

Description	Material of Each Layer	Thickness of Layer [1] (ft)	Steel Weight[2] (lbm)	Steel Temp (°F)	Area[3] (ft^2)	Exposure[4] (ft^2)	
							02-01
Piping, Piping Supports and Equipment (Continued)							
Reactor Coolant Pump Handling Fixture	Carbon Steel	0.029	8,000	Ambient	547	547	
Reactor Vessel Head Guide Studs	Stainless Steel	0.127	4,500	Ambient	72	72	
Reactor Vessel Internals Lifting Device	Stainless Steel	0.077	15,000	Ambient	398	398	
Reactor Vessel Head Lifting Device	Stainless Steel	0.126	21,000	Ambient	340	340	
Upper Internals Storage Stand	Stainless Steel	0.053	2,435	Ambient	93	93	02-01
Lower Internals Storage Stand	Stainless Steel	0.063	2,176	Ambient	70	70	
Reactor Coolant Pump Tierods and Steam Generator Upper/Lower Supports	Carbon Steel	0.029	50,976	Ambient	3,510	3,510	02-01
Reactor Coolant Piping Crossover Supports	Carbon Steel	0.049	7,800	Ambient	330	330	
Incore Instrument Tubing I	Stainless Steel	0.025	7,900	Ambient	921	921	
Incore Instrument Tubing II	Stainless Steel	0.017	1,152	Ambient	176	176	
Misc. Supports, Fittings, Special Tools		0.083	104,565	Ambient	2,568	2,568	

TABLE 6.2-7 (Continued)

PASSIVE HEAT SINKS

Description	Material of Each Layer	Thickness of Layer (1) (ft)	Steel Weight(2) (lbm)	Steel Temp (°F)	Area(3) (ft^2)	Exposure(4) (ft^2)	02-01
HVAC Ductwork and Units							
Ring Duct	Carbon Steel	0.011	39,105	Ambient	6,952	6,952	
Vertical Risers	Carbon Steel	0.014	62,906	Ambient	9,150	9,150	
Duct Reinforcing Angles	Carbon Steel	0.015	9,772	Ambient	1,357	1,357	
Risers, Duct Reinforcing Angles	Carbon Steel	0.024	75,360	Ambient	6,384	6,384	
Duct Supports	Carbon Steel	0.015	17,640	Ambient	2,394	2,394	
Riser Supports	Carbon Steel	0.024	14,130	Ambient	1,197	1,197	
Reactor Building Cooling Unit to Ring Duct Connectors	Carbon Steel	0.014	15,400	Ambient	2,240	2,240	
Reactor Building Cleanup Units	Carbon Steel	0.006	26,000	Ambient	830	830	
Reactor Building Cooling Units	Carbon Steel	0.069	392,000	Ambient	11,640	11,640	
Reactor Compartment Cooling System	Carbon Steel	0.062	2,400	Ambient	79	79	
CRDM Shroud Vent System	Stainless Steel	0.004	4,186	125°F	2,042	2,042	98-01
Refueling Water Surface System Fans	Carbon Steel	0.047	1,110	Ambient	48	48	
Reactor Building Charcoal Cleanup Fans	Carbon Steel	0.016	1,200	Ambient	149	149	
Secondary Compartment Cooling System	Carbon Steel	0.017	4,500	Ambient	545	545	
Duct Support and Support Angles:							
Floor El. 412'	Carbon Steel	0.005	1,678	Ambient	635	635	
Floor El. 463'	Carbon Steel	0.005	49,500	Ambient	18,710	18,710	

TABLE 6.2-7 (Continued)

PASSIVE HEAT SINKS

Description	Material of Each Layer	Thickness of Layer [1] (ft)	Steel Weight[2] (lbm)	Steel Temp (°F)	Area[3] (ft^2)	Exposure[4] (ft^2)	
							02-01
Electrical							
Cable Trays	Carbon Steel	0.009	46,186	Ambient	10,473	10,473	
Cable Tray Supports	Carbon Steel	0.042	100,842	Ambient	4,900	4,900	
Conduits	Carbon Steel	0.018	37,220	Ambient	4,220	4,220	
Conduit Supports	Carbon Steel	0.007	7,080	Ambient	2,064	2,064	
Penetrations Type 1	Carbon Steel	0.009	7,285	Ambient	1,652	1,652	
Penetrations Type 2	Carbon Steel	0.056	50,407	Ambient	1,837	1,837	02-01
Misc. Panels	Carbon Steel	0.007	909	Ambient	265	265	

(1) Material thickness, where not indicated specifically, is in feet.

(2) A 5 percent (plus) ,margin (for ECCS analysis) is included in the calculated weights of reactor building liner, liner anchors and fuel transfer canal liner.

(3) The Area column shows a conservatively high surface area (for ECCS analysis) for one side (plus ends if significant) of the heat sink. 98-01

(4) The Exposure column indicates if both sides are exposed to the reactor building atmosphere (or one side to the outside).

(5) Carbon Steel plate (1/4 in.) lines interior surface. Theoretically, zero surface are is exposed directly to reactor building atmosphere.

(6) Uncertainty margin of 5 percent included.

(7) Heat sink modeled at ambient temperature of 120°F, weighted average temperature

(8) Not used for peak pressure analysis, only for ECCS analysis.

Table 6.2-8

PASSIVE HEAT SINK MODELS USED IN PEAK PRESSURE ANALYSIS

Reactor Building Cylinder

a. Surface = 57,397.0 ft^2
b. Materials
   First region - carbon steel, 0.0217 ft
   Second region - air/steel, 0.00521 ft
   Third region - concrete, 0.25 ft
   Fourth region - concrete, 0.75 ft

Reactor Building Dome

a. Surface = 20,241 ft^2
b. Materials
   First region - carbon steel, 0.0217 ft
   Second region - air/steel, 0.00521 ft
   Third region - concrete, 0.25 ft
   Fourth region - concrete, 0.75 ft

Reactor Building Material

a. Surface = 12,463 ft^2
b. Material
   First region - concrete, 0.33 ft
   Second region - concrete, 1.67 ft

Internal Concrete

a. Surface = 76,839 ft^2
b. Material
   First region - concrete, 0.25 ft
   Second region - concrete, 0.75 ft

Carbon Steel, 0.02 ft < thickness < 0.03 ft

a. Surface = 29,885.0 ft^2
b. Material, carbon steel, average thickness, 0.0266 ft

Carbon Steel, thickness < 0.025 ft

a. Surface = 68,946.0 ft^2 02-01
b. Material, carbon steel, average thickness, 0.0132 ft

Table 6.2-8 (Continued)

PASSIVE HEAT SINK MODELS USED IN PEAK PRESSURE ANALYSIS

Stainless Steel, thickness > 0.025 ft

a. Surface - 17, 051.0 ft^2
b. Material, stainless steel, average thickness, 0.033 ft

Stainless Steel, thickness < 0.025 ft

a. Surface = 4919.4 ft^2
b. Material, stainless steel, average thickness, 0.0138 ft

Carbon Steel, thickness > 0.03 ft

a. Surface = 121,616.0 ft^2
b. Material, carbon steel, average thickness, 0.0415 ft

TABLE 6.2-9

REACTOR BUILDING DESIGN EVALUATION PARAMETERS
PASSIVE HEAT SINK MATERIAL PROPERTIES

Material	Thermal Conductivity (BTU/hr/ft/°F	Volumetric Heat Capacity (BTU/ft^3/°F
Concrete	0.8	28.0
Stainless Steel	9.4	53.7
Carbon Steel	28.0	53.9
Air/steel	0.72	1.31

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TABLE 6.2-11

REACTOR BUILDING SUBCOMPARTMENT
CALCULATED AND DESIGN PRESSURES

Compartment	Peak Calculated Differential Pressure (psid)	Design Differential Pressure (psid)
Pressurizer		
Spray Spray Line Break	26.0	41.4
Surge Line Break	30.6	51.3
Steam Generator		
Lower Compartment	See Section 6.2.1.3.9.2	
Reactor Cavity	See Section 6.2.1.3.9.3	

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Amendment 02-01

TABLE 6.2-11a

SUMMARY OF INITIAL STEAM GENERATOR COMPARTMENT MODELS

Model	Nodal Arrangement Figure No.	Model Description/Results Listed in Table No.	Results Illustrated by Figure No.
16 Node Model, Loop A Steam Generator Compartment Cold Leg Break	6.2-22a	6.2-12, 6.2-12a	6.2-22b
17 Node Model, Loop A Steam Generator Compartment Hot Leg Break	6.2-22c	6.2-12b, 6.2-12c	6.2-22d
12 Node Model, Loop B Steam Generator Compartment Cold Leg Break	6.2-22e	6.2-13, 6.2-13a	6.2-22f
13 Node Model, Loop B Steam Generator Compartment Hot Leg Break	6.2-22g	6.2-13b, 6.2-13c	6.2-22h
14 Node Model, Loop C Steam Generator Compartment Hot and Cold Leg Breaks	6.2-22i	6.2-14, 6.2-14a	6.2-22j, 6.2-22k

TABLE 6.2-12

16 NODE LOOP A STEAM GENERATOR COMPARTMENT MODEL
COLD LEG BREAK, VOLUME DESCRIPTION

Control Volume No	Description	Height (ft)	Volume (ft^3)	Volume Flow Area (ft^2)	Bottom Elev. (ft)	Initial Conditions Temp (°F)	Initial Conditions Pressure (psia)	Initial Conditions Quality	Calc. Peak Diff. press[(1)] (psi)	Design Diff. press.[(2)] (psi)	Design Margin (%)	
												02-01
1	Lower 1/2 Compartment below Steam Generator	14.0	6,768.	615.	412.5	212	14.696	0.573	21.5	41.2	92	
2	Lower 1/2 Compartment below Reactor Coolant Pump	14.0	4,641.	413.	412.5	212	14.696	0.573	21.4	41.2	93	
3	Between Steam Generator and Secondary Shield Wall	9.5	2,084.	317.	426.5	212	14.696	0.573	21.6	41.2	91	
4	Between Steam Generator, Reactor Coolant Pump and Primary Shield Wall	9.5	2,195.	326.	426.5	212	14.696	0.573	21.8	41.2	89	02-01
5	Between Steam Generator, Reactor Coolant Pump and Secondary Shield Wall	9.5	1,380.	332.	426.5	212	14.696	0.573	21.6	41.2	91	
6	Between Reactor Coolant Pump, Cold Leg and Secondary Shield Wall	9.5	1,134.	314.	426.5	212	14.696	0.573	24.7	41.2	67	
7	Between Reactor Coolant Pump and Secondary Shield Wall	9.5	314.	90.	426.5	212	14.696	0.573	21.6	41.2	91	
8	Compartment above Elevation 436'	23.5	16,686.	758.	436.0	212	14.696	0.573	21.5	41.2	92	
9	Around Steam Generator above Elevation 459.5'	15.9	4,826.	392.	459.5	212	14.696	0.573	5.8	16.9	191	
10	Steam Generator B Compartment below Elevation 459.5'	47.0	34,094.	863.	412.5	212	14.696	0.573	NA[(3)]			
11	Steam Generator B Compartment above Elevation 459.5'	15.9	4,347.	413.	459.5	212	14.696	0.573	NA			

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Amendment 02-01

TABLE 6.2-12

16 NODE LOOP A STEAM GENERATOR COMPARTMENT MODEL
COLD LEG BREAK, VOLUME DESCRIPTION

Control Volume No	Description	Height (ft)	Volume (ft^3)	Volume Flow Area (ft^2)	Bottom Elev. (ft)	Initial Conditions Temp (°F)	Initial Conditions Pressure (psia)	Initial Conditions Quality	Calc. Peak Diff. press[1] (psi)	Design Diff. press.[2] (psi)	Design Margin (%)
12	Pressurizer Surge Tank Compartment	22.5	6,200.	100.	412.5	212	14.696	0.573	NA		
13	Gallery between Steam Generators A and C	11.5	6,900.	200.	412.0	212	14.696	0.573	NA		
14	Steam Generator C Compartment below Elevation 459.5'	47.0	34,922.	859.	412.5	212	14.696	0.573	NA		
15	Steam Generator C Compartment above Elevation 459.5'	15.9	5,461.	315.	459.5	212	14.696	0.573	NA		
16	Containment	200.	1.74+6	1.3+4	412.0	212	14.696	0.545	NA		

02-01

Notes:
1. With respect to containment.
2. Values to be multiplied by dynamic load factor of 1.1 for purposes of structural design.
3. NA, not applicable.

02-01

TABLE 6.2-12a

16 NODE LOOP A STEAM GENERATOR COMPARTMENT MODEL
COLD LEG BREAK, FLOW PATH DATA

02-01

Junction No.	Control Volume 1	Control Volume 2	Choking Model	Elevation (ft)	Flow Area [(2)]	Inertia (ft^{-1})	Head Loss, K [(1)] Friction fL/D	Bends	Forward Flow Expansion	Forward Flow Contraction	Forward Flow Total	Reverse Flow Expansion	Reverse Flow Contraction	Reverse Flow Total
1	1	2	Inertial[(2)]	419.5	276.	0.05	0.01	-	-	0.04	0.05	-	0.04	0.05
2	1	3	Inertial	426.5	245.	0.05	0.01	-	-	-	0.01	-	-	0.01
3	2	4	HEM[(3)]	426.5	108.	0.11	0.01	-	-	0.30	0.31	-	0.47	0.48
4	2	5	Inertial	426.5	28.	0.42	0.01	-	-	-	0.01	-	-	0.01
5	1	4	HEM	426.5	150.	0.08	0.01	-	-	0.31	0.32	-	0.49	0.50
6	1	5	Inertial	426.5	109.	0.11	0.01	-	-	-	0.01	-	-	0.01
7	2	6	HEM	426.5	127.	0.12	0.02	-	-	0.28	0.30	-	0.39	0.41
8	2	7	HEM	426.5	31.	0.31	0.01	-	0.20	-	0.21	0.20	-	0.21
9	1	16	HEM	415.9	26.8	0.28	-	0.68	0.51	0.38	1.57	0.51	0.38	1.57
10	1	12	HEM	419.5	91.	0.04	-	1.20	1.00	0.33	2.53	0.44	0.50	2.14
11	1	13	HEM	419.5	79.	0.12	-	-	0.25	0.50	0.75	1.00	0.29	1.29
12	2	10	HEM	419.5	105.	0.18	0.03	0.30	1.00	0.50	1.83	1.00	0.50	1.83
13	10	16	HEM	415.9	107.	0.28	-	0.68	0.51	0.38	1.57	0.51	0.38	1.57
14	10	16	HEM	418.0	44.	0.36	0.06	3.60	1.00	0.50	5.16	1.00	0.50	5.16
15	10	16	HEM	459.5	157.	0.07	0.01	-	1.00	0.50	1.51	1.00	0.50	1.51
16	10	11	HEM	459.5	193.	0.07	0.01	-	0.13	0.52	0.66	0.13	0.52	0.66
17	11	16	HEM	475.4	246.	0.03	0.01	-	1.00	-	1.01	1.00	-	1.01
18	5	7	HEM	431.25	50.	0.16	0.01	-	0.24	0.31	0.56	0.24	0.31	0.56
19	6	7	HEM	431.25	31.	0.30	0.01	-	1.70	0.25	1.96	1.70	0.31	2.02

02-01

02-01

TABLE 6.2-12a (Continued)

16 NODE LOOP A STEAM GENERATOR COMPARTMENT MODEL
COLD LEG BREAK, FLOW PATH DATA

02-01

Junction No.	Control Volume 1	Control Volume 2	Choking Model	Elevation (ft)	Flow Area [(2)]	Inertia (ft^{-1})	Head Loss, K [(1)]: Friction fL/D	Bends	Forward Flow: Expansion	Forward Flow: Contraction	Forward Flow: Total	Reverse Flow: Expansion	Reverse Flow: Contraction	Reverse Flow: Total
20	6	4	HEM	431.25	75.	0.10	0.01	-	0.36	0.24	0.61	0.26	0.31	0.58
21	4	5	HEM	431.25	51.	0.28	0.01	-	0.42	0.31	0.74	0.29	0.46	0.76
22	3	5	HEM	431.25	49.	0.38	0.01	-	0.24	0.31	0.56	0.22	0.43	0.66
23	3	4	HEM	431.25	51.	0.14	0.01	-	0.28	0.29	0.58	0.24	0.41	0.66
24	6	10	HEM	431.25	70.	0.18	0.03	0.30	1.00	0.50	1.83	1.00	0.50	1.83
25	3	8	HEM	436.0	102.	0.09	0.01	-	0.22	0.20	0.43	0.20	0.22	0.43
26	4	8	HEM	436.0	160.	0.08	0.01	-	0.07	0.07	0.15	0.07	0.07	0.15
27	5	8	HEM	436.0	77.	0.12	0.01	-	0.23	0.21	0.45	0.21	0.23	0.45
28	6	8	HEM	436.0	118.	0.14	-	-	0.01	-	0.01	0.01	-	0.01
29	7	8	Inertial	436.0	28.	0.49	0.02	-	0.035	0.035	0.09	0.035	0.035	0.09
30	8	16	HEM	459.5	185.	0.06	0.01	-	1.00	0.50	1.51	0.50	1.00	1.51
31	8	9	HEM	459.5	160.	0.05	0.01	-	0.41	0.28	0.70	0.28	0.41	0.70
32	9	16	HEM	475.4	269.	0.03	0.01	-	1.00	-	1.01	1.00	-	1.01
33	12	13	HEM	417.5	108.	0.12	-	1.20	0.20	0.38	1.78	0.57	0.23	2.00
34	13	14	HEM	417.8	162.	0.90	0.02	-	0.21	0.22	0.37	0.27	-	0.51
35	13	16	HEM	415.9	27.	0.75	-	0.80	0.71	0.19	1.70	0.71	0.19	1.70
36	13	16	HEM	420.0	33.	0.25	-	1.20	1.00	0.50	2.70	1.00	0.50	2.70
37	14	16	HEM	415.9	81.	0.28	-	0.68	0.51	0.38	1.57	0.51	0.38	1.57
38	14	16	HEM	418.0	54.	0.17	-	1.80	0.59	0.50	2.89	0.59	0.50	2.89
39	14	16	HEM	459.4	128.	0.05	-	-	1.00	0.50	1.50	1.00	0.50	1.50

02-01

TABLE 6.2-12a (Continued)

16 NODE LOOP A STEAM GENERATOR COMPARTMENT MODEL
COLD LEG BREAK, FLOW PATH DATA

							Head Loss, K [1]							
									Forward Flow			Reverse Flow		
Junction No.	Control Volume 1	Control Volume 2	Choking Model	Elevation (ft)	Flow Area [2]	Inertia (ft^{-1})	Friction fL/D	Bends	Expansion	Contraction	Total	Expansion	Contraction	Total
40	14	15	HEM	459.5	225.	0.05	0.02	-	0.08	0.64	0.74	0.08	0.64	0.74
41	15	16	HEM	475.4	318.	0.03	0.01	-	1.00	-	1.01	1.00	-	1.01
42	0	4	Inertial	430.75	Fill syst.									
43	0	6	Inertial	430.75	Fill syst.									

02-01

NOTES:

1. With respect to minimum flow area.

2. No choking allowed.

3. Homogeneous equilibrium model.

TABLE 6.2-12b

17 NODE LOOP A STEAM GENERATOR COMPARTMENT MODEL
HOT LEG BREAK, VOLUME DESCRIPTION

Control Volume No	Description	Height (ft)	Volume (ft^3)	Volume Flow Area (ft^2)	Bottom Elev. (ft)	Initial Conditions Temp (°F)	Initial Conditions Pressure (psia)	Initial Conditions Quality	Clac. Peak Diff. press[1] (psi)	Design Diff. press [2] (psi)	Design Margin (%)	02-01
1	Lower 1/2 Compartment below Steam Generator	14.0	6,768.	615.	412.5	212	14.696	0.573	19.4	41.2	112	
2	Lower 1/2 Compartment below Reactor Coolant Pump	14.0	4,641.	413.	412.5	212	14.696	0.573	19.4	41.2	112	02-01
3	Between Steam Generator and Secondary Shield Wall	9.5	1,044.	208.	426.5	212	14.696	0.573	19.5	41.2	111	02-01
4	Between Steam Generator, Reactor Coolant Pump and Primary Shield Wall	9.5	1,853.	374.	426.5	212	14.696	0.573	20.5	41.2	101	02-01
5	Between Steam Generator, Reactor Coolant Pump and Secondary Shield Wall	9.5	1,380.	332.	426.5	212	14.696	0.573	19.9	41.2	107	02-01
6	Between Reactor Coolant Pump, Cold Leg and Secondary Shield Wall	9.5	1,134.	314.	426.5	212	14.696	0.573	19.0	41.2	116	02-01
7	Between Reactor Coolant Pump and Secondary Shield Wall	9.5	314.	90.	426.5	212	14.696	0.573	19.5	41.2	111	02-01
8	Compartment above Elevation 436'	23.5	16,686.	758.	436.0	212	14.696	0.573	19.7	41.2	109	
9	Around Steam Generator above Elevation 459.5'	15.9	4,826.	392.	459.5	212	14.696	0.573	4.7	16.9	260	

TABLE 6.2-12b (Continued)

17 NODE LOOP A STEAM GENERATOR COMPARTMENT MODEL
HOT LEG BREAK, VOLUME DESCRIPTION

Control Volume No	Description	Height (ft)	Volume (ft^3)	Volume Flow Area (ft^2)	Bottom Elev. (ft)	Initial Conditions Temp (°F)	Pressure (psia)	Quality	Clac. Peak Diff. press[1] (psi)	Design Diff. press [2] (psi)	Design Margin (%)	
												02-01
10	Steam Generator B Compartment below Elevation 459.5'	47.0	34,094.	863.	412.5	212	14.696	0.573	NA[3]			02-01
11	Steam Generator B Compartment above Elevation 459.5'	15.9	4,347.	413.	459.5	212	14.696	0.573	NA			
12	Pressurizer Surge Tank Compartment	22.5	6,200.	100.	412.5	212	14.696	0.573	NA			
13	Gallery between Steam Generators A and C	11.5	6,900.	200.	412.0	212	14.696	0.573	NA			
14	Steam Generator C Compartment below Elevation 459.5'	47.0	34,922.	859.	412.5	212	14.696	0.573	NA			
15	Steam Generator C Compartment above Elevation 459.5'	15.9	5,461.	315.	459.5	212	14.696	0.573	NA			02-01
16	Containment	200.	1.74+7	1.3+4	412.0	212	14.696	0.573	NA			
17	Between Steam Generator Hot Leg and Primary Shield Wall	9.5	1,205.	218.	426.5	212	14.696	0.545	37.5	41.2	10	

Notes:

1. With respect to containment.
2. Values to be multiplied by dynamic load factor of 1.1 for purposes of structural design. 02-01
3. NA, not applicable.

TABLE 6.2-12c

16 NODE LOOP A STEAM GENERATOR COMPARTMENT MODEL
COLD LEG BREAK, FLOW PATH DATA

Junction No.	Control Volume 1	Control Volume 2	Choking Model	Elevation (ft)	Flow Area (ft^2)	Inertia (ft^{-1})	Head Loss, K (1): Friction fL/D	Bends	Forward Flow: Expansion	Forward Flow: Contraction	Forward Flow: Total	Reverse Flow: Expansion	Reverse Flow: Contraction	Reverse Flow: Total	
															02-01
1	1	2	Inertial (2)	419.5	276.	0.05	0.01	-	-	0.04	0.05	-	0.04	0.05	
2	1	3	Inertial	426.5	114.	0.01	0.01	-	-	-	0.01	-	-	0.01	02-01
3	2	4	HEM (3)	426.5	108.	0.11	0.01	-	-	0.30	0.31	-	0.47	0.48	
4	2	5	Inertial	426.5	28.	0.42	0.01	-	-	-	0.01	-	-	0.01	
5	1	4	HEM	426.5	117.	0.10	0.01	-	-	0.34	0.35	-	0.59	0.60	02-01
6	1	5	Inertial	426.5	109.	0.11	0.01	-	-	-	0.01	-	-	0.01	
7	2	6	HEM	426.5	127.	0.12	0.02	-	-	0.28	0.30	-	0.39	0.41	02-01
8	2	7	HEM	426.5	31.	0.31	0.01	-	0.20	-	0.21	0.20	-	0.21	
9	1	16	HEM	415.9	27.	0.28	-	0.68	0.51	0.38	1.57	0.51	0.38	1.57	02-01
10	1	12	HEM	419.5	91.	0.04	-	1.20	1.00	0.33	2.53	0.44	0.50	2.14	
11	1	13	HEM	419.5	79.	0.12	-	-	0.25	0.50	0.75	1.00	0.29	1.29	
12	2	10	HEM	419.5	105.	0.18	0.03	0.30	1.00	0.50	1.83	1.00	0.50	1.83	
13	10	16	HEM	415.9	107.	0.28	-	0.68	0.51	0.38	1.57	0.51	0.38	1.57	
14	10	16	HEM	418.0	44.	0.36	0.06	3.60	1.00	0.50	5.16	1.00	0.50	5.16	
15	10	16	HEM	459.5	157.	0.07	0.01	-	1.00	0.50	1.51	1.00	0.50	1.51	
16	10	11	HEM	459.5	193.	0.07	0.01	-	0.13	0.52	0.66	0.13	0.52	0.66	
17	11	16	HEM	457.4	246.	0.03	0.01	-	1.00	-	1.01	1.00	-	1.01	
18	5	7	HEM	431.25	50.	0.16	0.01	-	0.24	0.31	0.56	0.24	0.31	0.56	
19	6	7	HEM	431.25	31.	0.30	0.01	-	1.70	0.25	1.96	1.70	0.31	2.02	
20	6	4	HEM	431.25	75.	0.10	0.01	-	0.36	0.24	0.61	0.26	0.31	0.58	

TABLE 6.2-12c (Continued)

16 NODE LOOP A STEAM GENERATOR COMPARTMENT MODEL
COLD LEG BREAK, FLOW PATH DATA

Junction No.	Control Volume 1	Control Volume 2	Choking Model	Elevation (ft)	Flow Area (ft^2)	Inertia (ft^{-1})	Head Loss, K [1] Friction fL/D	Bends	Forward Flow Expansion	Forward Flow Contraction	Forward Flow Total	Reverse Flow Expansion	Reverse Flow Contraction	Reverse Flow Total	02-01
21	4	5	HEM	431.25	51.	0.28	0.01	-	0.43	0.27	0.71	0.29	0.38	0.68	
22	3	5	HEM	431.25	54.	0.28	0.01	-	0.28	0.14	0.43	0.22	0.14	0.37	
23	4	17	HEM	431.25	51.	0.19	0.01	-	0.30	0..28	0.59	0.24	0.40	0.65	02-01
24	6	10	HEM	432.25	70.	0.18	0.03	0.30	1.00	0.50	1.83	1.00	0.50	1.83	
25	3	8	HEM	436.0	66.	0.17	0.01	-	0.13	0.12	0.26	0.12	0.13	0.26	02-01
26	4	8	HEM	436.0	159.	0.09	0.01	-	0.04	0.04	0.09	0.04	0.04	0.09	
27	5	8	HEM	436.0	77.	0.12	0.01	-	0.23	0.21	0.45	0.21	0.23	0.45	
28	6	8	HEM	436.0	118.	0.14	0.01	-	0.01	-	0.02	0.01	-	0.02	
29	7	8	Inertial	436.0	28.	0.49	0.02	-	0.035	0.035	0.09	0.035	0.035	0.09	
30	8	16	HEM	459.5	185.	0.06	0.01	-	1.00	0.50	1.51	0.50	1.00	1.51	
31	8	9	HEM	459.5	160.	0.05	0.01	-	0.41	0.28	0.70	0.28	0.41	0.70	
32	9	16	HEM	475.4	269.	0.03	0.01	-	1.00	-	1.01	1.00	-	1.01	02-01
33	12	13	HEM	417.5	108.	0.12	-	1.20	0.20	0.38	1.78	0.57	0.23	2.00	
34	13	14	HEM	417.8	162.	0.10	0.02	-	0.21	0.22	0.45	0.27	-	0.29	
35	13	16	HEM	415.9	27.	0.75	-	0.80	0.71	0.19	1.70	0.71	0.19	1.70	
36	13	16	HEM	420.0	33.	0.25	-	1.20	1.00	0.50	2.70	1.00	0.50	2.70	
37	14	16	HEM	415.9	81.	0.28	-	0.68	0.51	0.38	1.57	0.51	0.38	1.57	
38	14	16	HEM	418.0	54.	0.17	-	1.80	0.59	0.50	2.89	0.59	0.50	2.89	
39	14	16	HEM	459.4	128.	0.05	-	-	1.00	0.50	1.50	1.00	0.50	1.50	
40	14	15	HEM	459.5	225.5	0.05	0.02	-	0.08	0.64	0.74	0.08	0.64	0.74	

TABLE 6.2-12c (Continued)

16 NODE LOOP A STEAM GENERATOR COMPARTMENT MODEL
COLD LEG BREAK, FLOW PATH DATA

							Head Loss, K (1)								02-01
									Forward Flow			Reverse Flow			
Junction No.	Control Volume 1	Control Volume 2	Choking Model	Elevation (ft)	Flow Area (ft^2)	Inertia (ft^{-1})	Friction fL/D	Bends	Expansion	Contraction	Total	Expansion	Contraction	Total	
41	15	16	HEM	475.4	318.	0.03	0.01	-	1.00	-	1.01	1.00	-	1.01	
42	3	17	HEM	431.25	54.	0.21	0.01	-	0.19	0.19	0.39	0.19	0.19	0.39	
43	17	8	HEM	436.0	51.	0.14	0.01	-	0.19	0.13	0.33	0.23	0.33	0.57	
44	1	17	HEM	426.5	159.	0.08	0.01	-	-	-	0.01	-	-	0.01	
45	0	4	Inertial	430.75	Fill syst.										
46	0	17	Inertial	430.75	Fill syst..										

NOTES:

1. With respect to minimum flow area.

2. No choking allowed.

3. Homogeneous equilibrium model. 02-01

TABLE 6.2-13

12 NODE LOOP B STEAM GENERATOR COMPARTMENT MODEL
COLD LEG BREAK, VOLUME DESCRIPTION

02-01

Control Volume No.	Description	Height (ft)	Volume (ft^3)	Volume Flow Area (ft^2)	Bottom Elev. (ft)	Initial Conditions Temp. (°F)	Pressure (psia)	Quality	Calc. Peak Diff. Press (1) (psi)	Design Diff. Press (2) (psi)	Design Margin (%)	
												02-01
1	Lower 1/2 Compartment below Reactor Coolant Pump	14.0	3,869.	364.	412.5	212	14.696	0.573	22.8	41.2	81	
2	Lower 1/2 Compartment below Steam Generator	14.0	6,708.	510.	412.5	212	14.696	0.573	22.8	41.2	81	
3	Between Reactor Coolant Pump and Secondary Shield Wall	9.5	369.	132.	426.5	212	14.696	0.573	22.9	41.2	80	
4	Between Reactor Coolant Pump, Cold Leg and Secondary Shield Wall	9.5	1,003.	332.	426.5	212	14.696	0.573	40.0	41.2	3	
5	Between Reactor Coolant Pump, Steam Generator and Primary Shield Wall	9.5	1,722.	744.	426.5	212	14.696	0.573	23.4	41.2	76	
6	Between Reactor Coolant Pump, Steam Generator and Secondary Shield Wall	9.5	1,430.	371.	426.5	212	14.696	0.573	22.8	41.2	81	
7	Between Steam Generator and Secondary Shield Wall	9.5	2,326.	460.	426.5	212	14.696	0.573	22.7	41.2	81	
8	Compartment above Elevation 436’	23.5	16,663.	863.	436.0	212	14.696	0.573	22.5	41.2	83	
9	Around Steam Generator above Elevation 459.5’	15.9	4,347.	413.	459.5	212	14.696	0.573	11.5	16.9	47	
10	Steam Generator A Compartment below Elevation 459.5’	47.0	35,254.	858.	412.5	212	14.696	0.573	NA (3)			02-01
11	Steam Generator A Compartment above Elevation 459.5’	15.9	4,826.	458.	459.5	212	14.696	0.573	NA			
12	Containment	200.	1.84+6	1.3+4	412.0	212	14.696	0.545	NA			02-01

Notes:

1. With respect to containment.
2. Values to be multiplied by dynamic load factor of 1.1 for purposes of structural design.
3. NA - not applicable.

TABLE 6.2-13a

12 NODE LOOP B STEAM GENERATOR COMPARTMENT MODEL
COLD LEG BREAK, FLOW PATH DATA

Junction No.	Control Volume 1	Control Volume 2	Choking Model	Elevation (ft)	Flow Area (ft^2)	Inertia (ft^{-1})	Head Loss, K [1] Friction fL/D	Bends	Forward Flow Expansion	Forward Flow Contraction	Forward Flow Total	Reverse Flow Expansion	Reverse Flow Contraction	Reverse Flow Total	
1	1	2	Inertial [2]	419.5	235.8	0.05	0.01	-	0.02	0.02	0.05	0.02	0.02	0.05	02-01
2	1	3	HEM [3]	426.5	28.4	0.31	0.01	-	-	0.41	0.42	0.81	-	0.82	02-01
3	1	4	HEM	426.5	47.7	0.17	0.01	-	-	0.36	0.37	0.69	-	0.70	
4	1	5	HEM	426.5	99.6	0.10	0.01	-	0.02	0.28	0.31	0.40	0.02	0.43	
5	1	6	HEM	426.5	22.7	0.27	0.01	-	0.19	0.23	0.43	0.21	0.21	0.43	
6	1	12	HEM	415.9	40.2	0.28	-	0.68	0.51	0.38	1.57	0.51	0.38	1.57	
7	1	12	HEM	418.0	43.9	0.36	0.06	3.6	1.0	0.5	5.16	1.0	0.5	5.16	
8	2	5	Inertial	426.5	83.6	0.14	0.01	-	-	-	0.01	-	-	0.01	
9	2	6	Inertial	426.5	90.8	0.11	0.01	-	-	0.06	0.07	-	0.06	0.07	
10	2	7	Inertial	426.5	277.8	0.04	0.01	-	-	-	0.01	-	-	0.01	02-01
11	2	10	HEM	419.5	104.5	0.18	0.03	0.30	1.0	0.5	1.83	1.0	0.5	1.83	02-01
12	2	12	HEM	415.9	67.0	0.28	-	0.68	0.51	0.38	1.57	0.51	0.38	1.57	
13	3	4	HEM	431.25	30.9	0.33	0.01	-	0.30	0.24	0.55	0.30	0.24	0.55	
14	3	6	HEM	431.25	58.3	0.17	0.01	-	0.15	-	0.16	-	0.15	0.16	
15	3	8	HEM	436.0	32.0	0.39	0.01	-	0.05	0.05	0.11	0.05	0.05	0.11	
16	4	5	HEM	431.25	61.1	0.17	0.01	-	0.33	0.25	0.59	0.33	0.25	0.59	
17	4	8	HEM	436.0	101.7	0.14	0.01	-	0.02	0.02	0.05	0.02	0.02	0.05	
18	5	6	HEM	431.25	50.4	0.19	0.01	-	0.28	0.23	0.52	0.28	0.23	0.52	
19	5	7	HEM	431.25	119.1	0.09	0.01	-	0.09	0.09	0.19	0.09	0.09	0.19	
20	5	8	HEM	436.0	134..2	0.09	0.01	-	0.06	0.12	0.19	0.12	0.06	0.19	

02-01

TABLE 6.2-13a (Continued)

12 NODE LOOP B STEAM GENERATOR COMPARTMENT MODEL
COLD LEG BREAK, FLOW PATH DATA

							Head Loss, K (1)								
									Forward Flow			Reverse Flow			02-01
Junction No.	Control Volume 1	Control Volume 2	Choking Model	Elevation (ft)	Flow Area (ft^2)	Inertia (ft^{-1})	Friction fL/D	Bends	Expansion	Contraction	Total	Expansion	Contraction	Total	
21	6	7	HEM	431.25	79.8	0.16	0.01	-	0.08	0.08	0.17	0.08	0.08	0.17	02-01
22	6	8	HEM	436.0	80.0	0.14	0.01	-	0.12	0.19	0.32	0.91	0.12	0.32	
23	7	8	HEM	436.0	181.3	0.08	0.01	-	0.03	0.12	0.16	0.3	0.12	0.16	
24	7	10	HEM	431.25	70.2	0.18	0.03	0.3	1.00	0.5	1.83	1.00	0.5	1.83	
25	8	9	HEM	459.5	193.0	0.07	0.01	-	0.13	0.52	0.66	0.13	0.52	0.66	
26	8	12	HEM	459.5	157.0	0.07	0.01	-	1.00	0.5	1.51	1.00	0.5	1.51	
27	9	12	HEM	475.4	245.9	0.03	0.01	-	1.00	-	1.01	1.00	-	1.01	
28	10	11	HEM	459.5	160.2	0.07	0.01	-	0.13	0.52	0.66	0.13	0.52	0.66	
29	10	12	HEM	415.9	26.8	0.28	-	0.68	0.51	0.38	1.57	0.51	0.38	1.57	
30	10	12	HEM	459.5	184.6	0.07	0.01	-	1.00	0.5	1.51	1.00	0.5	1.51	
31	11	12	HEM	475.4	269.1	0.03	0.01	-	1.00	-	1.01	1.00	-	1.01	
32	0	4	NA (4)	430.75	Fill syst.										02-01
33	0	5	NA	430.75	Fill syst.										02-01

NOTES:

1. With respect to minimum flow area.
2. No choking allowed.
3. Homogeneous equilibrium model. 02-01
4. NA, not applicable.

TABLE 6.2-13b

13 NODE LOOP B STEAM GENERATOR COMPARTMENT MODEL
HOT LEG BREAK, VOLUME DESCRIPTION

Control Volume No	Description	Height (ft)	Volume (ft^3)	Volume Flow Area (ft^2)	Bottom Elev. (ft)	Initial Conditions			Clac. Peak Diff. press[1] (psi)	Design Diff. press [2] (psi)	Design Margin (%)	02-01
						Temp (°F)	Pressure (psia)	Quality				
1	Lower 1/2 Compartment below Reactor Coolant Pump	14.0	3,869.	364.	412.5	212	14.696	0.573	19.9	41.2	107	
2	Lower 1/2 Compartment below Steam Generator	14.0	6,708.	510.	412.5	212	14.696	0.573	19.9	41.2	107	02-01
3	Between Reactor Coolant Pump and Secondary Shield Wall	9.5	369.	132.	426.5	212	14.696	0.573	20.3	41.2	103	
4	Between Reactor Coolant Pump, Cold Leg and Secondary Shield Wall	9.5	1,003.	332.	426.5	212	14.696	0.573	20.1	41.2	105	
5	Between Reactor Coolant Pump, Steam Generator and Primary Shield Wall	9.5	1,722.	744.	426.5	212	14.696	0.573	26.6	41.2	55	
6	Between Reactor Coolant Pump, Steam Generator and Secondary Shield Wall	9.5	1,430.	371.	426.5	212	14.696	0.573	20.3	41.2	103	
7	Between Steam Generator, Hot Leg and Secondary Shield Wall	9.5	1,396.	276.	426.5	212	14.696	0.573	26.0	41.2	58	
8	Compartment above Elevation 436’	23.5	16,663.	863.	436.0	212	14.696	0.573	20.2	41.2	104	
9	Around Steam Generator above Elevation 459.5’	15.9	4,347.	413.	459.5	212	14.696	0.573	9.6	16.9	76	
10	Steam Generator A Compartment below Elevation 459.5’	47.0	32,254.	858.	412.5	212	14.696	0.573	NA [3]			02-01
11	Steam Generator A Compartment above Elevation 459.5’	15.9	4,826.	458.	459.5	212	14.696	0.573	NA			
12	Containment	200.	1.74+6	1.3+4	412.0	212	14.696	0.545	NA			
13	Between Steam Generator and Secondary Shield Wall	9.5	930.	184.	426.5	212	14.696	0.573	20.1	41.2	105	02-01

Notes:

1. With respect to containment. 02-01
2. Values to be multiplied by dynamic load factor of 1.1 for purposes of structural design.
3. NA, not applicable. 02-01

TABLE 6.2-13c

13 NODE LOOP B STEAM GENERATOR COMPARTMENT MODEL
HOT LEG BREAK, FLOW PATH DATA

Junction No.	Control Volume 1	Control Volume 2	Choking Model	Elevation (ft)	Flow Area (ft^2)	Inertia (ft^{-1})	Head Loss, K[1] Friction fL/D	Bends	Forward Flow Expansion	Forward Flow Contraction	Forward Flow Total	Reverse Flow Expansion	Reverse Flow Contraction	Reverse Flow Total	02-01
1	1	2	Inertial[2]	419.5	235.8	0.05	0.01	-	0.02	0.02	0.05	0.02	0.02	0.05	02-01
2	1	3	HEM[3]	426.5	28.4	0.31	0.01	-	-	0.41	0.42	0.81	-	0.82	02-01
3	1	4	HEM	426.5	47.7	0.17	0.01	-	-	0.36	0.37	0.69	-	0.70	
4	1	5	HEM	426.5	99.6	0.10	0.01	-	0.02	0.28	0.31	0.42	0.02	0.43	
5	1	6	HEM	426.5	22.7	0.27	0.01	-	0.19	0.23	0.43	0.21	0.21	0.43	
6	1	12	HEM	415.9	40.2	0.28	-	0.68	0.51	0.38	1.57	0.51	0.38	1.57	
7	1	12	HEM	418.0	43.9	0.36	0.06	3.6	1.0	0.5	5.16	1.0	0.5	5.16	
8	2	5	HEM	426.5	83.6	0.14	0.01	-	-	0.37	0.38	0.69	-	0.70	
9	2	6	HEM	426.5	90.8	0.11	0.01	-	0.06	0.36	0.43	0.67	0.06	0.74	
10	2	7	HEM	426.5	166.7	0.04	0.01	-	-	0.18	0.19	0.18	-	0.19	
11	2	10	HEM	419.5	104.5	0.18	0.03	0.3	1.0	0.5	1.83	1.0	0.5	1.83	
12	2	12	HEM	415.9	67.0	0.28	-	0.68	0.51	0.38	1.57	0.51	0.38	1.57	
13	3	4	HEM	431.25	30.9	0.33	0.01	-	0.30	0.24	0.55	0.30	0.24	0.55	
14	3	6	HEM	431.25	58.3	0.17	0.01	-	0.15	-	0.16	-	0.15	0.16	
15	3	8	HEM	436.0	32.0	0.39	0.01	-	0.05	0.05	0.11	0.05	0.05	0.11	
16	4	5	HEM	431.25	61.1	0.17	0.01	-	0.33	0.25	0.59	0.33	0.25	0.59	
17	4	8	HEM	436.0	101.7	0.14	0.01	-	0.02	0.02	0.05	0.02	0.02	0.05	
18	5	6	HEM	431.25	50.4	0.19	0.01	-	0.28	0.23	0.52	0.28	0.23	0.52	
19	5	7	HEM	431.25	119.1	0.09	0.01	-	0.09	0.09	0.19	0.09	0.09	0.19	
20	5	8	HEM	436.0	134.2	0.09	0.01	-	0.06	0.12	0.19	0.12	0.06	0.19	02-01

TABLE 6.2-13c (Continued)

13 NODE LOOP B STEAM GENERATOR COMPARTMENT MODEL
HOT LEG BREAK, FLOW PATH DATA

Junction No.	Control Volume 1	Control Volume 2	Choking Model	Elevation (ft)	Flow Area (ft^2)	Inertia (ft^{-1})	Head Loss, K [1] Friction fL/D	Bends	Forward Flow Expansion	Forward Flow Contraction	Forward Flow Total	Reverse Flow Expansion	Reverse Flow Contraction	Reverse Flow Total
21	6	13	HEM	431.25	79.8	0.16	0.01	-	0.08	0.08	0.17	0.08	0.08	0.17
22	6	8	HEM	436.0	80.0	0.14	0.01	-	0.12	0.19	0.32	0.19	0.12	0.32
23	7	8	HEM	436.0	108.8	0.15	0.01	-	0.03	0.12	0.16	0.03	0.12	0.16
24	7	10	HEM	431.25	70.2	0.18	0.03	0.30	1.00	0.5	1.83	1.00	0.5	1.83
25	8	9	HEM	459.5	193.0	0.07	0.01	-	0.13	0.52	0.66	0.13	0.52	0.66
26	8	12	HEM	459.5	157.0	0.07	0.01	-	1.00	0.5	1.51	1.00	0.5	1.51
27	9	12	HEM	475.4	245.9	0.03	0.01	-	1.00	-	1.01	1.00	-	1.01
28	10	11	HEM	459.5	160.2	0.07	0.01	-	0.13	0.52	0.66	0.13	0.52	0.66
29	10	12	HEM	415.9	26.8	0.28	-	0.68	0.51	0.38	1.57	0.51	0.38	1.57
30	10	12	HEM	459.5	184.6	0.07	0.01	-	1.00	0.50	1.51	1.00	0.5	1.51
31	11	12	HEM	475.4	269.1	0.03	0.01	-	1.00	-	1.01	1.00	-	1.01
32	2	13	HEM	426.5	111.1	0.04	0.01	-	-	0.18	0.19	0.18	-	0.19
33	7	13	HEM	431.25	77.6	0.19	0.01	-	-	0.09	0.10	0.09	-	0.10`
34	8	13	HEM	436.0	72.5	0.23	0.01	-	0.03	0.12	0.16	0.03	0.12	0.16
35	0	5	NA [4]	Fill syst.										
36	0	7	NA	Fill syst.										

02-01

02-01

NOTES:

1. With respect to minimum flow area.
2. No choking allowed.
3. Homogeneous equilibrium model. 02-01
4. NA, not applicable.

TABLE 6.2-14

14 NODE LOOP C STEAM GENERATOR COMPARTMENT MODEL
HOT AND COLD LEG BREAKS, VOLUME DESCRIPTION

Control Volume No.	Description	Height (ft)	Volume (ft^3)	Flow Area (ft^2)	Volume Elevation (ft)	Bottom Temp. (°F)	Pressure (psia)	Quality	Calc. Peak Diff. Press.[1] Cold Leg (psi)	Calc. Peak Diff. Press.[1] Hot Leg (psi)	Design Press [2] (psi)	Design Margin (%)	02-01
1	Lower 1/2 Compartment below Reactor Coolant Pump	14.0	5,519.	442.	412.5	212	14.696	0.573	23.2	20.2	41.2	78	
2	Lower 1/2 Compartment below Steam Generator	14.0	6,232.	462.	412.5	212	14.696	0.573	23.3	20.2	41.2	77	
3	Between Reactor Coolant Pump, primary Shield and Steam Generator	9.5	1,528.	586.	426.5	212	14.696	0.573	26.3	26.3	41.2	57	
4	Between Cold Leg, Reactor Coolant Pump and Secondary Shield	9.5	609.	203.	426.5	212	14.696	0.573	47.9	20.1	41.2	-14 [4]	02-01
5	Between Reactor Coolant Pump, Steam Generator and Secondary Shield	9.5	2,640.	300.	426.5	212	14.696	0.573	23.2	20.1	41.2	78	
6	Between Steam Generator and Secondary Shield	9.5	1,022.	163.	426.5	212	14.696	0.573	23.3	20.2	41.2	77	
7	Between Steam Generator, Hot Leg and Secondary Shield Wall	9.5	864.	158.	426.5	212	14.696	0.573	23.4	43.4	41.2	-5 [4]	02-01
8	Compartment above Elevation 436'	23.5	16,470.	730.	436.0	212	14.696	0.573	22.9	20.4	41.2	80	
9	Around Steam Generator above Elevation 459.5'	15.9	5,461.	315.	459.5	212	14.696	0.573	6.4	6.0	16.9	164	
10	Gallery between Loops A and C	11.5	6,900.	200.	412.0	212	14.696	0.573	8.9	7.5	41.2	363	
11	Pressurizer Relief Tank Compartment	22.5	6,200.	100.	412.5	212	14.696	0.573	NA [3]				02-01
12	Steam Generator A Compartment below Elevation 459.5'	47.0	35,254.	858.	412.5	212	14.696	0.573	NA				
13	Steam Generator A Compartment above Elevation 459.5'	15.9	4,826.	260.	459.5	212	14.696	0.573	NA				02-01
14	Containment	200.	1.74+6	12,500.	412.0	212	14.696	0.573	NA				

Notes:

1. With respect to containment.
2. Values to be multiplied by dynamic load factor of 1.1 for purposes of structural design.
3. NA, not applicable.
4. For discussion of results, see Section 6.2.1.3.9.2.3.

TABLE 6.2-14a

14 NODE LOOP C STEAM GENERATOR COMPARTMENT MODEL
HOT AND COLD LEG BREAKS, FLOW PATH DATA

02-01

Junction No.	Control Volume 1	Control Volume 2	Choking Model	Elevation (ft)	Flow Area (ft^2)	Inertia (ft^{-1})	Head Loss, K (1): Friction fL/D	Bends	Forward Flow: Expansion	Forward Flow: Contraction	Forward Flow: Total	Reverse Flow: Expansion	Reverse Flow: Contraction	Reverse Flow: Total	
1	1	2	Inertial (2)	419.5	304.0	0.05	0.01	-	-	-	0.01	-	-	0.01	02-01
2	1	10	HEM (3)	417.8	161.8	0.09	0.02	-	0.21	0.22	0.45	0.27	-	0.29	02-01
3	1	14	HEM	415.9	26.8	0.28	-	0.68	0.51	0.38	1.57	0.51	0.38	1.57	
4	2	14	HEM	415.9	53.6	0.28	-	0.68	0.51	0.38	1.57	0.51	0.38	1.57	
5	2	14	HEM	418.0	53.8	0.17	-	1.8	0.59	0.5	2.89	0.59	0.5	2.89	02-01
6	3	1	HEM	426.5	63.0	0.08	0.01	-	0.72	-	0.73	-	0.06	0.07	
7	3	2	HEM	426.5	116.4	0.05	0.01	-	0.56	-	0.57	-	0.37	0.38	
8	3	4	HEM	431.25	44.3	0.28	0.02	-	1.86	1.13	3.01	0.41	-	0.43	
9	3	5	HEM	431.25	54.8	0.25	0.02	-	0.50	0.28	0.8	0.33	0.33	0.68	
10	3	7	HEM	431.25	74.8	0.14	0.03	-	0.04	0.20	0.27	0.15	0.11	0.29	
11	3	8	HEM	436.0	108.8	0.06	0.01	-	0.09	0.19	0.29	0.01	0.15	0.17	02-01
12	4	1	HEM	426.5	64.4	0.08	0.02	-	0.71	-	0.73	0.10	-	0.11	
13	4	5	HEM	431.25	40.0	0.20	0.02	-	0.22	0.24	0.48	0.22	0.24	0.48	
14	4	8	HEM	436.0	60.5	0.12	0.02	-	-	0.03	0.05	-	0.03	0.05	
15	5	1	HEM	426.5	144.4	0.03	0.01	-	0.12	0.15	0.28	0.09	0.17	0.27	
16	5	2	HEM	426.5	80.7	0.03	0.02	-	0.06	0.12	0.20	0.06	0.12	0.20	
17	5	6	HEM	431.25	81.0	0.15	0.03	-	-	0.23	0.26	0.21	-	0.24	
18	5	8	HEM	436.0	209.6	0.04	0.01	-	0.06	0.12	0.19	0.06	0.12	0.19	
19	6	2	Inertial	426.5	119.2	0.05	0.01	-	-	-	0.01	-	-	0.01	
20	6	7	HEM	431.25	71.6	0.16	0.02	-	0.0	0.18	0.20	0.13	0.03	0.18	

TABLE 6.2-14a (Continued)

14 NODE LOOP C STEAM GENERATOR COMPARTMENT MODEL
HOT AND COLD LEG BREAKS, FLOW PATH DATA

Junction	Control	Control	Choking	Elevation	Flow	Inertia	Head Loss, K [1]								
							Friction		Forward Flow			Reverse Flow			02-01
No.	Volume 1	Volume 2	Model	(ft)	Area (ft^2)	(ft^{-1})	fL/D	Bends	Expansion	Contraction	Total	Expansion	Contraction	Total	
21	6	8	HEM	436.0	53.7	0.09	0.01	-	0.19	0.25	0.45	0.05	0.21	0.27	02-01
22	7	2	HEM	426.5	110.8	0.12	0.02	-	0.58	-	0.60	-	-	0.02	
23	7	8	HEM	436.0	59.0	0.10	0.02	-	0.05	0.11	0.18	0.05	0.11	0.18	
24	8	9	HEM	459.5	225.5	0.05	0.02	-	0.08	0.64	0.74	0.08	0.64	0.74	
25	8	14	HEM	459.5	128.4	0.05	-	-	1.00	0.50	1.50	1.00	0.50	1.50	
26	9	14	HEM	475.4	317.6	0.02	0.01	-	1.00	-	1.01	1.00	-	1.01	
27	10	11	HEM	417.5	107.7	0.12	-	1.2	0.57	0.23	2.0	0.20	0.38	1.78	
28	11	12	HEM	419.5	90.9	0.04	-	1.2	1.00	0.33	2.53	0.44	0.50	2.14	
29	10	14	HEM	415.9	26.8	0.75	-	0.80	0.71	0.19	1.7	0.71	0.19	1.7	
30	10	14	HEM	420.0	33.0	0.24	-	1.2	1.00	0.50	2.7	1.00	0.50	2.7	
31	10	12	HEM	419.5	78.8	0.12	-	-	1.00	0.29	1.29	0.25	0.50	0.75	
32	12	13	HEM	459.5	160..2	0.12	0.02	-	0.16	0.70	0.88	0.16	0.70	0.88	
33	12	14	HEM	423.2	178.7	0.11	-	-	1.00	0.50	1.50	1.00	0.50	1.50	
34	12	14	HEM	415.9	26.8	0.28	-	0.68	0.51	0.38	1.57	0.51	0.38	1.57	
35	12	14	HEM	459.5	184.6	0.04	-	-	1.00	0.38	1.38	1.00	0.38	1.38	
36	13	14	HEM	475.4	269.1	0.03	0.01	-	1.00	-	1.01	1.00	-	1.01	
37	0	3	NA [4]	430.75	0.5	Fill system; hot and cold leg breaks									02-01
38	0	4	NA	430.75	0.5	Fill system; cold leg break									
39	0	7	NA	430.75	0.5	Fill system; hot leg break									

NOTES:

1. With respect to minimum flow area.
2. No choking allowed.
3. Homogeneous equilibrium model. 02-01
4. NA, not applicable.

TABLE 6.2-15

SUMMARY OF LOOP C GENERATOR COMPARTMENT MODELS

Model	Model Arrangement Figure No.	Model Description/Results Listed in Table No.	Results Illustrated by Fig. No.	Forces and Moments Data	
				Fig. No.	Table No.
20 Node Model, Loop C Steam Generator Compartment, Cold Leg Break, Reactor Coolant Pump Loading	6.2-23	6.2-15a, 5.2-15b	6.2-23a, 6.2-23b, 6.2-23c	6.2-24	6.2-15c
25 Node Model, Loop C Steam Generator Compartment, Hot Leg Break, Steam Generator Loading	6.2-25	6.2-15d, 16.2-15e	6.2-25a, 6.2-25b, 6.2-25c	6.2-26	6.2-15f

TABLE 6.2-15a

20 NODE LOOP C STEAM GENERATOR COMPARTMENT MODEL
COLD LEG BREAK, VOLUME DESCRIPTION

Control Volume No.	Description	Height (ft)	Volume (ft^3)	Volume Flow Area (ft^2)	Bottom Elev. (ft)	Initial Conditions Temp. (°F)	Initial Conditions Pressure (psia)	Initial Conditions Quality	Calc. Peak Diff. Press [(1)] (psi)	Design Diff. Press [(2)] (psi)	Design Margin (%)	02-01
1	Lower 1/2 Compartment below Reactor Coolant Pump	14.0	5,519.	442.	412.5	212	14.696	0.573	23.0	41.2	79	
2	Lower 1/2 Compartment below Steam Generator	14.0	6,232.	462.	412.5	212	14.696	0.573	23.0	41.2	79	
3	Between Reactor Coolant Pump, Primary Shield and Steam Generator	9.5	1,528.	586.	426.5	212	14.696	0.573	26.5	41.2	55	
4	Between Cold Legs, Reactor Coolant Pump and Secondary Shield	9.5	609.	203.	426.5	212	14.696	0.573	47.9	41.2	-14 [(4)]	02-01
5	Between Reactor Coolant Pump, Steam Generator and Secondary Shield	9.5	2,640.	300.	426.5	212	14.696	0.573	23.0	41.2	79	
6	Between Steam Generator and Secondary Shield	9.5	1,022.	163.	426.5	212	14.696	0.573	23.1	41.2	78	
7	Between Steam Generator, Hot Leg and Secondary Shield	9.5	864.	158.	426.5	212	14.696	0.573	23.2	41.2	78	
8	Between Reactor Coolant Pump, Steam Generator and Primary Shield above Elevation 436'	15.0	2,300.	277.	436.0	212	14.696	0.573	22.8	41.2	81	
9	Around Steam Generator above Elevation 459.5'	15.9	5,461.	315.	459.5	212	14.696	0.573	6.4	16.9	164	02-01
10	Gallery between Loops A and C	11.5	6,900.	200.	412.0	212	14.696	0.573	8.9	41.2	363	
11	Pressurizer Relief Tank Compartment	22.5	6,200.	100.	412.5	212	14.696	0.573	NA [(3)]			02-01
12	Steam Generator A Compartment below Elevation 459.5'	47.0	35,254.	858.	412.5	212	14.696	0.573	NA			
13	Steam Generator A Compartment above Elevation 459.5'	15.9	4,826.	260.	459.5	212	14.696	0.573	NA			02-01

Reformatted Per
Amendment 02-01

TABLE 6.2-15a (Continued)

20 NODE LOOP C STEAM GENERATOR COMPARTMENT MODEL
COLD LEG BREAK, VOLUME DESCRIPTION

Control Volume No.	Description	Height (ft)	Volume (ft^3)	Volume Flow Area (ft^2)	Bottom Elev. (ft)	Initial Conditions Temp. (°F)	Initial Conditions Pressure (psia)	Initial Conditions Quality	Calc. Peak Diff. Press (1) (psi)	Design Diff. Press (2) (psi)	Design Margin (%)
14	Containment	200.	1.74+6	12,500.	412.0	212	14.696	0.573	NA		
15	Between Reactor Coolant Pump and Secondary Shield above Elevation 436’	15.	966.	126.	436.0	212	14.696	0.573	22.8	41.2	81
16	Between Reactor Coolant Pump, Steam Generator and Secondary Shield above Elevation 436’	15.	4,161.	382.	436.0	212	14.696	0.573	22.8	41.2	81
17	Between Steam Generator and Secondary Shield above Elevation 436’	15.	1,431.	191.	436.0	212	14.696	0.573	22.8	41.2	81
18	Between Steam Generator and Primary Shield above Elevation 436’	15.	1,144.	176.	436.0	212	14.696	0.573	22.8	41.2	81
19	Between Steam Generator and Secondary Shield above Elevation 451’	8.5	1,457.	237.	451.0	212	14.696	0.573	22.8	41.2	81
20	Above Reactor Coolant Pump	8.5	4,208.	495.	451.0	212	14.696	0.573	22.8	41.2	81

02-01

Notes:

1. With respect to containment.
2. Values to be multiplied by dynamic load factor of 1.1 for purposes of structural design.
3. NA, not applicable.
4. For discussion of results see Section 6.2.1.3.9.2.3.

TABLE 6.2-15b

20 NODE LOOP C STEAM GENERATOR COMPARTMENT MODEL
COLD LEG BREAK, FLOW PATH DATA

02-01

Junction No.	Control Volume 1	Control Volume 2	Choking Model	Elevation (ft)	Flow Area (ft^2)	Inertia (ft^{-1})	Head Loss, K [(1)] Friction fL/D	Bends	Forward Flow Expansion	Forward Flow Contraction	Forward Flow Total	Reverse Flow Expansion	Reverse Flow Contraction	Reverse Flow Total
1	1	2	Inertial[(2)]	419.5	304.0	0.05	0.01	-	-	-	0.01	-	-	0.01
2	1	10	HEM[(3)]	417.8	161.8	0.09	0.02	-	0.21	0.22	0.45	0.27	-	0.29
3	1	14	HEM	415.9	26.8	0.28	-	0.68	0.51	0.38	1.57	0.51	0.38	1.57
4	2	14	HEM	415.9	53.6	0.28	-	0.68	0.51	0.38	1.57	0.51	0.38	1.57
5	2	14	HEM	418.0	53.8	0.17	-	1.8	0.59	0.50	2.89	0.59	0.50	2.89
6	3	1	HEM	426.5	63.0	0.08	0.01	-	0.72	-	0.73	-	0.06	0.07
7	3	2	HEM	426.5	116.4	0.05	0.01	-	0.56	-	0.57	-	0.37	0.38
8	3	4	HEM	431.25	44.3	0.28	0.02	-	1.86	1.13	3.01	0.41	-	0.43
9	3	5	HEM	431.25	54.8	0.25	0.02	-	0.50	0.28	0.80	0.33	0.33	0.68
10	3	7	HEM	431.25	74.8	0.14	0.03	-	0.04	0.20	0.27	0.15	0.11	0.29
11	3	8	HEM	436.0	108.8	0.07	0.02	-	0.08	0.19	0.29	0.14	0.01	0.17
12	4	1	HEM	426.5	64.4	0.08	0.02	-	0.71	-	0.73	0.10	-	0.12
13	4	5	HEM	431.25	40.0	0.20	0.02	-	0.22	0.24	0.48	0.22	0.24	0.48
14	4	15	HEM	436.0	60.5	0.16	0.02	-	-	0.03	0.05	-	0.03	0.05
15	5	1	HEM	426.5	144.4	0.03	0.01	-	0.12	0.15	0.28	0.09	0.17	0.27
16	5	2	HEM	426.5	80.7	0.15	0.02	-	0.06	0.12	0.20	0.06	0.12	0.20
17	5	6	HEM	431.25	81.0	0.04	0.03	-	-	0.23	0.26	0.22	-	0.25
18	5	16	HEM	436.0	209.6	0.04	0.01	-	0.06	0.12	0.19	0.06	0.12	0.19
19	6	2	Inertial	426.5	119.2	0.05	0.01	-	-	-	0.01	-	-	0.01
20	6	7	HEM	431.25	71.6	0.16	0.02	-	-	0.18	0.20	0.13	0.03	0.18

TABLE 6.2-15b (Continued)

20 NODE LOOP C STEAM GENERATOR COMPARTMENT MODEL
COLD LEG BREAK, FLOW PATH DATA

02-01

Junction No.	Control Volume 1	Control Volume 2	Choking Model	Elevation (ft)	Flow Area (ft^2)	Inertia (ft^{-1})	Head Loss, K (1): Friction fL/D	Bends	Forward Flow: Expansion	Forward Flow: Contraction	Forward Flow: Total	Reverse Flow: Expansion	Reverse Flow: Contraction	Reverse Flow: Total
21	6	17	HEM	436.0	53.7	0.11	0.01	-	0.19	0.25	0.45	0.05	0.21	0.27
22	7	2	HEM	426.5	110.8	0.12	0.02	-	0.58	-	0.60	-	-	0.02
23	7	18	HEM	436.0	59.0	0.14	0.02	-	0.05	0.11	0.18	0.05	0.11	0.18
24	19	9	HEM	459.5	140.8	0.07	0.02	-	0.32	0.09	0.43	0.32	0.09	0.43
25	20	14	HEM	459.5	128.4	0.04	0.01	-	1.00	0.37	1.38	1.00	0.37	1.38
26	9	14	HEM	475.4	317.6	0.02	0.01	-	1.00	-	1.01	1.00	-	1.01
27	10	11	HEM	417.5	107.7	0.12	-	1.2	0.57	0.23	2.0	0.20	0.38	1.78
28	11	12	HEM	419.5	90.9	0.04	-	1.2	1.00	0.33	2.53	0.44	0.50	2.14
29	10	14	HEM	415.9	26.8	0.75	-	0.8	0.71	0.19	1.7	0.71	0.19	1.7
30	10	14	HEM	420.0	33.0	0.24	-	1.2	1.00	0.5	2.7	1.00	0.50	2.7
31	10	12	HEM	419.5	78.8	0.12	-	-	1.00	0.29	1.29	0.25	0.50	0.75
32	12	13	HEM	459.5	160.2	0.12	0.02	-	0.16	0.70	0.88	0.16	0.70	0.88
33	12	14	HEM	423.2	178.7	0.11	-	-	1.00	0.50	1.50	1.00	0.50	1.50
34	12	14	HEM	415.9	26.8	0.28	-	0.68	0.51	0.38	1.57	0.51	0.38	1.57
35	12	14	HEM	459.5	184.6	0.04	-	-	1.00	0..38	1.38	1.00	0.38	1.38
36	13	14	HEM	475.4	269.1	0.03	0.01	-	1.00	-	1.01	1.00	-	1.01
37	8	15	HEM	443.5	45.0	0.16	-	-	0.25	0.30	0.55	0.41	0.25	0.66
38	8	16	HEM	443.5	112.5	0.15	0.01	-	0.39	0.21	0.61	0.18	0.31	0.50
39	8	18	HEM	443.5	135.0	0.08	0.02	-	0.01	0.15	0.18	0.09	0.05	0.16
40	8	20	Inertial	451.0	153.3	0.08	0.02	-	-	-	0.02	-	-	0.02

TABLE 6.2-15b (Continued)

20 NODE LOOP C STEAM GENERATOR COMPARTMENT MODEL
COLD LEG BREAK, FLOW PATH DATA

02-01

Junction No.	Control Volume 1	Control Volume 2	Choking Model	Elevation (ft)	Flow Area (ft^2)	Inertia (ft^{-1})	Head Loss, K [1]								
							Friction		Forward Flow			Reverse Flow			
							fL/D	Bends	Expansion	Contraction	Total	Expansion	Contraction	Total	
41	15	16	HEM	443.5	82.5	0.13	0.01	-	0.10	0.16	0.27	0.10	0.16	0.27	
42	15	20	Inertial	451.0	64.4	0.17	0.02	-	0.04	-	0.06	0.04	-	0.06	
43	16	17	HEM	443.5	150.0	0.18	0.02	-	-	0.19	0.21	0.14	-	0.16	
44	16	20	Inertial	451.0	209.6	0.06	0.02	-	0.0	-	0.02	-	-	0.02	
45	17	18	HEM	443.5	82.5	0.11	0.02	-	0.10	0.27	0.39	0.29	0.16	0.47	
46	17	19	Inertial	451.0	95.4	0.10	0.02	-	-	-	0.02	-	-	0.02	
47	18	19	Inertial	451.0	76.0	0.13	0.02	-	-	-	0.02	-	-	0.02	
48	19	20	HEM	455.2	85.0	0.11	0.02	-	0.14	-	0.16	-	0.19	0.21	
49	19	20	HEM	455.2	76.5	0.14	0.02	-	0.19	-	0.21	-	0.21	0.23	
50	20	9	HEM	459.5	84.7	0.07	0.02	-	0.55	0.41	0.98	0.55	0.41	0.98	
51	0	3	NA [4]	430.75	0.5	Fill syst.									02-01
52	0	4	NA [4]	430.75	0.5	Fill syst.									02-01

NOTES:

1. With respect to minimum flow area.
2. No choking allowed.
3. Homogeneous equilibrium model. 02-01
4. NA, not applicable.

TABLE 6.2-15c

20 NODE LOOP C STEAM GENERATOR COMPARTMENT MODEL
FORCES AND MOMENTS ON REACTOR COOLANT PUMP

02-01

Time (Sec)	F_x (lbf)	F_y (lbf)	F_z (lbf)	M_x (ft-lbf)	M_y (ft-lbf)	Node	A_x (ft^2)	A_y (ft^2)	A_z (ft^2)	AM_x (ft^3)	AM_y (ft^3)
0	0	0	0	0	0	1	0.0	0.0	0.0	0.0	0.0
2.50-3	-9.49+4	-6.26+4	2.86+4	-3.38+5	5.13+5	2	0.0	0.0	0.0	0.0	0.0
5.10-3	-2.19+5	-1.46+5	6.60+4	-7.48+5	1.12+6	3	3.133+1	-3.603+1	7.560+0	-1.982+2	-1.723+2
7.50-3	-2.94+5	-1.98+5	8.78+4	-9.14+5	1.37+6	4	-6.957+1	-2.286+1	1.412+1	-1.257+2	3.826+2
1.00-2	-3.22+5	-2.17+5	9.40+4	-8.28+5	1.25+6	5	3.824+1	5.889+1	1.412+1	3.239+2	-2.103+2
1.48-2	-3.16+5	-2.01+5	8.26+4	-3.09+5	6.23+5	6	0.0	0.0	0.0	0.0	0.0
2.96-2	-1.62+5	-7.95+4	5.23+4	-3.29+5	8.14+5	7	0.0	0.0	0.0	0.0	0.0
3.52-2	-1.05+5	-3.64+4	5.01+4	-5.72+5	1.15+6	8	3.838+1	-4.913+1	0.0	3.685+2	2.879+2
4.00-2	-8.26+4	-1.29+4	5.00+4	-6.79+5	1.26+6	9	0.0	0.0	0.0	0.0	0.0
5.00-2	-1.18+5	-3.20+4	4.81+4	-5.84+5	9.83+5	10	0.0	0.0	0.0	0.0	0.0
6.00-2	-2.54+5	-1.43+5	6.48+4	-4.64+5	7.86+5	11	0.0	0.0	0.0	0.0	0.0
7.00-2	-3.20+5	-2.03+5	7.88+4	-3.04+5	7.84+5	12	0.0	0.0	0.0	0.0	0.0
8.00-2	-2.85+5	-1.73+5	8.49+4	-5.62+5	1.25+6	13	0.0	0.0	0.0	0.0	0.0
9.00-2	-2.40+5	-1.20+5	8.60+4	-1.00+6	1.73+6	14	0.0	0.0	0.0	0.0	0.0
1.00-1	-2.44+5	-9.97+4	8.25+4	-1.11+6	1.12+6	15	-9.053+1	-3.117+1	0.0	2.338+2	-6.790+2
1.50-1	-2.64+5	-1.16+5	7.19+4	-7.92+5	1.54+6	16	5.215+1	8.030+1	0.0	-6.023+2	3.911+2
2.00-1	-2.58+5	-1.20+5	6.54+4	-5.87+5	1.42+6	17	0.0	0.0	0.0	0.0	0.0
3.00-1	-2.33+5	-1.04+5	5.60+4	-5.34+5	1.24+6	18	0.0	0.0	0.0	0.0	0.0
4.00-1	-2.10+5	-9.11+4	4.93+4	-4.90+5	1.12+6	19	0.0	0.0	0.0	0.0	0.0
5.00-1	-1.95+5	-8.55+4	4.74+4	-4.50+5	1.03+6	20	0.0	0.0	-3.58+1	0.0	0.0
1.00	-1.69+5	-7.32+4	4.03+4	-3.89+5	8.96+5						

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TABLE 6.2-15c (Continued)

20 NODE LOOP C STEAM GENERATOR COMPARTMENT MODEL
FORCES AND MOMENTS ON REACTOR COOLANT PUMP

Maximum Forces and Moments and Corresponding Times

Max. F_x	=	-3.240+5	at time	=	0.0116
Max F_y	=	-2.177+5	at time	=	0.0104
Max. F_z	=	9.403+4	at time	=	0.0099
Max. M_x	=	-1.124+6	at time	=	0.0970
Max. M_y	=	1.795+6	at time	=	0.0940

TABLE 6.2-15d

25 NODE LOOP C STEAM GENERATOR COMPARTMENT MODEL 02-01
HOT LEG BREAK, VOLUME DESCRIPTION

Control Volume No	Description	Height (ft)	Volume (ft^3)	Volume Flow Area (ft^2)	Bottom Elev. (ft)	Initial Conditions Temp (°F)	Initial Conditions Pressure (psia)	Initial Conditions Quality	Clac. Peak Diff. press[1] (psi)	Design Diff. press [2] (psi)	Design Margin (%)	
1	Lower 1/2 Compartment below Reactor Coolant Pump	14.0	5,519.	442.	412.5	212	14.696	0.573	20.5	41.2	101	
2	Lower 1/2 Compartment below Steam Generator	14.0	6,232.	462.	412.5	212	14.696	0.573	20.4	41.2	102	
3	Between Reactor Coolant Pump, Primary Shield and Steam Generator	9.5	1,528.	586.	426.5	212	14.696	0.573	27.2	41.2	51	
4	Between Cold Leg, Reactor Coolant Pump and Secondary Shield	9.5	609.	203.	426.5	212	14.696	0.573	20.3	41.2	103	
5	Between Reactor Coolant Pump, Steam Generator and Secondary Shield	9.5	2,640.	300.	426.5	212	14.696	0.573	20.5	41.2	101	
6	Between Steam Generator and Secondary Shield	9.5	1,022.	163.	426.5	212	14.696	0.573	20.8	41.2	98	
7	Between Steam Generator, Hot Leg and Secondary Shield	9.5	864.	158.	426.5	212	14.696	0.573	45.9	41.2	-10 [4]	02-01
8	Between Reactor Coolant Pump, Steam Generator and Primary Shield above Elevation 436’	15.0	2,300.	277.	436.0	212	14.696	0.573	21.6	41.2	91	
9	Around Steam Generator between Elevations 475.4’ and 513.5’	38.1	4,314.	259.	475.4	212	14.696	0.573	0.9	12.1	1244	
10	Gallery between Loops A and C	11.5	6,900.	200.	412.0	212	14.696	0.573	11.2	41.2	268	02-01
11	Pressurizer Relief Tank Compartment	22.5	6,200.	100.	412.5	212	14.696	0.573	NA [3]			02-01
12	Steam Generator A Compartment below Elevation 459.5’	47.0	35,254.	858.	412.5	212	14.696	0.573	NA			02-01
13	Steam Generator A Compartment above Elevation 459.5’	15.9	4,826.	260.	459.5	212	14.696	0.573	NA			02-01
14	Containment	200.	1.74+6	12,500.	412.0	212	14.696	0.573	NA			02-01

TABLE 6.2-15d (Continued)

25 NODE LOOP C STEAM GENERATOR COMPARTMENT MODEL
HOT LEG BREAK, VOLUME DESCRIPTION

02-01

Control Volume No	Description	Height (ft)	Volume (ft^3)	Volume Flow Area (ft^2)	Bottom Elev. (ft)	Initial Conditions Temp (°F)	Initial Conditions Pressure (psia)	Initial Conditions Quality	Clac. Peak Diff. press[(1)] (psi)	Design Diff. press [(2)] (psi)	Design Margin (%)
15	Between Reactor Coolant Pump and Secondary Shield above Elevation 436'	15.	966.	126.	436.0	212	14.696	0.573	20.8	41.2	98
16	Between Reactor Coolant Pump, Steam Generator and Secondary Shield above Elevation 436'	15.	4,161.	382.	436.0	212	14.696	0.573	21.0	41.2	96
17	Between Steam Generator and Secondary Shield above Elevation 436'	15.	1,431.	191.	436.0	212	14.696	0.573	20.4	41.2	102
18	Between Steam Generator and Primary Shield above Elevation 436'	15.	1,144.	176.	436.0	212	14.696	0.573	20.5	41.2	101
19	Between Steam Generator and Secondary Shield above Elevation 451'	8.5	1,457.	237.	451.0	212	14.696	0.573	20.5	41.2	101
20	Above Reactor Coolant Pump	8.5	4,208.	495.	451.0	212	14.696	0.573	20.4	41.2	102
21	Around Steam Generator between Elevations 459.5' and 475.4'	15.9	378.	87.	459.5	212	14.696	0.573	10.4	16.9	63
22	Around Steam Generator between Elevations 459.5' and 475.4'	15.9	1,277.	166.	459.5	212	14.696	0.573	10.3	16.9	64
23	Around Steam Generator between Elevations 459.5' and 475.4'	15.9	2,094.	236.	459.5	212	14.696	0.573	10.4	16.9	63
24	Around Steam Generator between Elevations 459.5' and 475.4'	15.9	1,687.	162.	459.5	212	14.696	0.573	10.2	16.9	66
25	Around Steam Generator between Elevations 475.4' and 513.5'	38.1	5,156.	313.	475.4	212	14.696	0.573	1.1	12.1	1,000

Notes:

1. With respect to containment. 02-01
2. Values to be multiplied by dynamic load factor of 1.1 for purposes of structural design.
3. NA, not applicable. 02-01
4. For discussion of results see Section 6.2.1.3.9.2.3.

TABLE 6.2-15e

25 NODE LOOP C STEAM GENERATOR COMPARTMENT MODEL
HOT LEG BREAK, FLOW PATH DATA

							Head Loss, K (1)								02-01
									Forward Flow			Reverse Flow			
Junction No.	Control Volume 1	Control Volume 2	Choking Model	Elevation (ft)	Flow Area (ft^2)	Inertia (ft^{-1})	Friction fL/D	Bends	Expansion	Contraction	Total	Expansion	Contraction	Total	
1	1	2	Inertial(2)	419.5	304.0	0.05	0.01	-	-	-	0.01	-	-	0.01	
2	1	10	HEM(3)	417.8	161.8	0.09	0.02	-	0.21	0.22	0.45	0.27	-	0.29	02-01
3	1	14	HEM	415.9	26.8	0.28	-	0.68	0.51	0.38	1.57	0.51	0.38	1.57	
4	2	14	HEM	415.9	53.6	0.28	-	0.68	0.51	0.38	1.57	0.51	0.38	1.57	02-01
5	2	14	HEM	418.0	53.8	0.17	-	1.8	0.59	0.50	2.89	0.59	0.50	2.89	
6	3	1	HEM	426.5	63.0	0.08	0.01	-	0.72	-	0.73	-	0.06	0.07	
7	3	2	HEM	426.5	116.4	0.05	0.01	-	0.56	-	0.57	-	0.37	0.38	
8	3	4	HEM	431.25	44.3	0.28	0.02	-	1.86	1.13	3.01	0.41	-	0.43	
9	3	5	HEM	431.25	54.8	0.25	0.02	-	0.50	0.28	0.80	0.33	0.33	0.68	
10	3	7	HEM	431.25	74.8	0.14	0.03	-	0.04	0.20	0.27	0.15	0.11	0.29	
11	3	8	HEM	436.0	108.8	0.07	0.02	-	0.08	0.19	0.29	0.14	0.01	0.17	
12	4	1	HEM	426.5	64.4	0.08	0.02	-	0.71	-	0.73	0.10	-	0.12	
13	4	5	HEM	431.25	40.0	0.20	0.02	-	0.22	0.24	0.48	0.22	0.24	0.48	
14	4	15	HEM	436.0	60.5	0.16	0.02	-	-	0.03	0.05	-	0.03	0.05	
15	5	1	HEM	426.5	144.4	0.03	0.01	-	0.12	0.15	0.28	0.09	0.17	0.27	
16	5	2	HEM	426.5	80.7	0.15	0.02	-	0.06	0.12	0.20	0.06	0.12	0.20	
17	5	6	HEM	431.25	81.0	0.04	0.03	-	-	0.23	0.26	0.22	-	0.25	
18	5	16	HEM	436.0	209.6	0.04	0.01	-	0.06	0.12	0.19	0.06	0.12	0.19	
19	6	2	Inertial	426.5	119.2	0.05	0.01	-	-	-	0.01	-	-	0.01	
20	6	7	HEM	431.25	71.6	0.16	0.02	-	0.0	0.18	0.20	0.13	0.03	0.18	

TABLE 6.2-15e (Continued)

25 NODE LOOP C STEAM GENERATOR COMPARTMENT MODEL
HOT LEG BREAK, FLOW PATH DATA

02-01

Junction No.	Control Volume 1	Control Volume 2	Choking Model	Elevation (ft)	Flow Area (ft^2)	Inertia (ft^{-1})	Head Loss, K (1): Friction fL/D	Bends	Forward Flow: Expansion	Forward Flow: Contraction	Forward Flow: Total	Reverse Flow: Expansion	Reverse Flow: Contraction	Reverse Flow: Total
21	6	17	HEM	436.0	53.7	0.11	0.01	-	0.19	0.25	0.45	0.05	0.21	0.27
22	7	2	HEM	426.5	110.8	0.12	0.02	-	0.58	-	0.60	-	-	0.02
23	7	18	HEM	436.0	59.0	0.14	0.02	-	0.05	0.11	0.18	0.05	0.11	0.18
24	20	21	HEM	459.5	12.0	0.20	0.01	-	0.55	0.43	0.99	0.86	0.33	1.20
25	20	14	HEM	459.5	128.4	0.04	0.01	-	1.00	0.37	1.38	1.00	0.37	1.38
26	20	22	HEM	459.5	60.9	0.10	0.01	-	0.17	0.28	0.46	0.40	0.17	0.58
27	10	11	HEM	417.5	107.7	0.12	-	1.2	0.57	0.23	2.0	0.20	0.38	1.78
28	11	12	HEM	419.5	90.9	0.04	-	1.2	1.00	0.33	2.53	0.44	0.50	2.14
29	10	14	HEM	415.9	26.8	0.75	-	0.8	0.71	0.19	1.7	0.71	0.19	1.7
30	10	14	HEM	420.0	33.0	0.24	-	1.2	1.00	0.5	2.7	1.00	0.50	2.7
31	10	12	HEM	419.5	78.8	0.12	-	-	1.00	0.29	1.29	0.25	0.50	0.75
32	12	13	HEM	459.5	160.2	0.12	0.02	-	0.16	0.70	0.88	0.16	0.70	0.88
33	12	14	HEM	423.2	178.7	0.11	-	-	1.00	0.50	1.50	1.00	0.50	1.50
34	12	14	HEM	415.9	26.8	0.28	-	0.68	0.51	0.38	1.57	0.51	0.38	1.57
35	12	14	HEM	459.5	184.6	0.04	-	-	1.00	0.38	1.38	1.00	0.38	1.38
36	13	14	HEM	475.4	269.1	0.03	0.01	-	1.00	-	1.01	1.00	-	1.01
37	8	15	HEM	443.5	45.0	0.16	-	-	0.25	0.30	0.55	0.41	0.25	0.66
38	8	16	HEM	443.5	112.5	0.15	0.01	-	0.39	0.21	0.61	0.18	0.31	0.50
39	8	18	HEM	443.5	135.0	0.08	0.02	-	0.01	0.15	0.18	0.09	0.05	0.16
40	8	20	Inertial	451.0	153.3	0.08	0.02	-	-	-	0.02	-	-	0.02

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TABLE 6.2-15e (Continued)

25 NODE LOOP C STEAM GENERATOR COMPARTMENT MODEL
HOT LEG BREAK, FLOW PATH DATA

02-01

Junction No.	Control Volume 1	Control Volume 2	Choking Model	Elevation (ft)	Flow Area (ft^2)	Inertia (ft^{-1})	Head Loss, K (1): Friction fL/D	Bends	Forward Flow: Expansion	Forward Flow: Contraction	Forward Flow: Total	Reverse Flow: Expansion	Reverse Flow: Contraction	Reverse Flow: Total
41	15	16	HEM	443.5	82.5	0.13	0.01	-	0.10	0.16	0.27	0.10	0.16	0.27
42	15	20	Inertial	451.0	64.4	0.17	0.02	-	0.04	-	0.06	0.04	-	0.06
43	16	17	HEM	443.5	150.0	0.18	0.02	-	-	0.19	0.21	0.14	-	0.16
44	16	20	Inertial	451.0	209.6	0.06	0.02	-	0.0	-	0.02	-	-	0.02
45	17	18	HEM	443.5	82.5	0.11	0.02	-	0.10	0.27	0.39	0.29	0.16	0.47
46	17	19	Inertial	451.0	95.4	0.10	0.02	-	-	-	0.02	-	-	0.02
47	18	19	Inertial	451.0	76.0	0.13	0.02	-	-	-	0.02	-	-	0.02
48	19	20	HEM	455.2	85.0	0.11	0.02	-	0.14	-	0.16	-	0.19	0.21
49	19	20	HEM	455.2	76.5	0.14	0.02	-	0.19	-	0.21	-	0.21	0.23
50	19	23	HEM	459.5	85.4	0.06	0.01	-	0.25	0.33	0.59	0.55	0.22	0.78
51	20	23	HEM	459.5	11.8	0.10	0.01	-	0.17	0.28	0.46	0.40	0.17	0.58
52	19	24	HEM	459.5	55.4	0.06	0.01	-	0.39	0.37	0.77	0.68	0.28	0.97
53	21	22	HEM	467.5	34.6	0.31	0.01	-	0.32	0.25	0.58	0.32	0.25	0.58
54	21	24	HEM	467.5	33.7	0.30	0.01	-	0.40	0.08	0.49	0.08	0.28	0.37
55	22	23	HEM	467.5	121.2	0.12	0.01	-	0.04	0.21	0.26	0.24	0.04	0.29
56	23	24	Inertial	467.5	75.4	0.16	0.01	-	0.02	0.02	0.05	0.02	0.02	0.05
57	21	14	HEM	475.4	18.3	0.48	0.01	-	1.00	-	1.01	1.00	-	1.01
58	22	14	HEM	475.4	78.7	0.10	0.01	-	1.00	-	1.01	1.00	-	1.01
59	23	25	HEM	475.4	114.5	0.24	0.01	-	-	0.09	0.10	0.09	-	0.10
60	24	9	HEM	475.4	87.4	0.31	0.01	-	0.01	0.12	0.14	0.12	0.01	0.14

Reformatted Per
Amendment 02-01

TABLE 6.2-15e (Continued)

25 NODE LOOP C STEAM GENERATOR COMPARTMENT MODEL
HOT LEG BREAK, FLOW PATH DATA

02-01

Junction No.	Control Volume 1	Control Volume 2	Choking Model	Elevation (ft)	Flow Area (ft^2)	Inertia (ft^{-1})	Head Loss, K [1] Friction fL/D	Bends	Forward Flow Expansion	Forward Flow Contraction	Forward Flow Total	Reverse Flow Expansion	Reverse Flow Contraction	Reverse Flow Total
61	25	14	HEM	513.5	114.5	0.22	0.01	-	1.00	-	1.01	1.00	-	1.01
62	25	14	HEM	494.5	275.5	0.03	0.01	-	1.00	-	1.01	1.00	-	1.01
63	25	9	Inertial	494.5	122.0	0.12	0.01	-	0.02	0.02	0.05	0.02	0.02	0.05
64	9	14	HEM	513.5	87.4	0.22	0.01	-	1.00	-	1.01	1.00	-	1.01
65	9	14	HEM	494.5	152.0	0.07	0.01	-	1.00	-	1.01	1.00	-	1.01
66	0	3	NA [4]	430.75	0.5	Fill syst.								
67	0	7	NA	430.75	0.5	Fill syst.								

02-01

NOTES:

1. With respect to minimum flow area.
2. No choking allowed.
3. Homogeneous equilibrium model. 02-01
4. NA, not applicable.

TABLE 6.2-15f

25 NODE LOOP C STEAM GENERATOR COMPARTMENT MODEL
FORCES AND MOMENTS ON STEAM GENERATOR 02-01

Time (Sec)	F_x (lbf)	F_y (lbf)	F_z (lbf)	M_x (ft-lbf)	M_y (ft-lbf)	Node	A_x (ft^2)	A_y (ft^2)	A_z (ft^2)	AM_x (ft^3)	AM_y (ft^3)	
												02-01
0	0	0	0	0	0	1	0.0	0.0	0.0	0.0	0.0	
2.50-3	3.49+4	-7.71+4	9.36+4	-1.40+5	-6.37+4	2	0.0	0.0	0.0	0.0	0.0	
5.10-3	8.83+5	-1.90+5	2.21+5	-2.28+5	-1.12+5	3	-2.552+1	-2.338+1	2.602+1	-4.676+1	5.104+1	02-01
7.50-3	1.32+5	-2.69+5	2.91+5	-5.53+4	-5.07+4	4	0.0	0.0	0.0	0.0	0.0	02-01
1.00-2	1.70+5	-3.23+5	3.15+5	3.75+5	1.41+5	5	-2.464+1	2.338+1	2.781+1	4.676+1	4.928+1	
1.48-2	1.96+5	-3.54+5	2.98+5	1.27+6	6.90+5	6	1.935+1	2.306+1	2.393+1	4.612+1	-3.870+1	
2.96-2	-2.24+4	-3.03+4	2.66+5	-3.38+5	5.85+5	7	3.081+1	-2.306+1	2.991+1	-4.612+1	-6.162+1	
3.52-2	-7.36+4	5.41+4	2.95+5	-9.72+5	1.60+5	8	-8.322+1	-8.769+1	0.0	6.577+2	-6.241+2	
4.00-2	-7.52+4	8.98+4	3.28+5	-1.22+6	1.16+5	9	1.380+2	-2.830+2	0.0	1.449+4	7.069+3	02-01
5.00-2	1.43+4	6.85+4	3.76+5	-8.36+5	1.55+6	10	0.0	0.0	0.0	0.0	0.0	
6.00-2	1.25+5	-9.64+4	3.86+5	8.74+5	3.80+6	11	0.0	0.0	0.0	0.0	0.0	
7.00-2	1.40+5	-2.33+5	4.16+5	2.38+6	4.32+6	12	0.0	0.0	0.0	0.0	0.0	
8.00-2	6.28+4	-1.47+5	4.47+5	1.74+6	3.13+6	13	0.0	0.0	0.0	0.0	0.0	
9.00-2	-1.32+4	1.61+4	4.56+5	-1.11+5	1.70+6	14	-3.165+2	5.867+1	-1.791+2	-3.004+3	-1.621+4	02-01
1.00-1	-7.26+3	3.12+4	4.57+5	-9.05+5	1.39+6	15	0.0	0.0	0.0	0.0	0.0	02-01
1.50-1	1.85+4	-9.15+3	4.58+5	7.40+5	1.64+6	16	-9.241+1	8.769+1	0.0	-6.577+2	-6.931+2	02-01
2.00-1	4.44+4	-1.20+4	4.59+5	3.43+5	2.21+6	17	7.257+1	8.648+1	0.0	-6.486+2	5.443+2	
3.00-1	4.84+4	-2.76+4	4.13+5	3.04+5	2.07+6	18	1.031+2	-8.648+1	0.0	6.486+2	7.729+2	
4.00-1	3.84+4	-3.17+4	3.55+5	2.91+5	1.68+6	19	9.952+1	0.0	0.0	0.0	1.916+3	
5.00-1	3.37+4	-2.96+4	3.05+5	2.37+5	1.48+6	20	-9.952+1	0.0	0.0	0.0	-1.916+3	02-01
1.00	2.65+4	-2.28+4	1.62+5	1.74+5	1.24+6	21	-8.394+1	-4.095+1	1.741+1	1.288+3	-2.641+3	02-01

TABLE 6.2-15f (Continued)

25 NODE LOOP C STEAM GENERATOR COMPARTMENT MODEL
FORCES AND MOMENTS ON STEAM GENERATOR 02-01

Time (Sec)	F_x (lbf)	F_y (lbf)	F_z (lbf)	M_x (ft-lbf)	M_y (ft-lbf)	Node	A_x (ft^2)	A_y (ft^2)	A_z (ft^2)	AM_x (ft^3)	AM_y (ft^3) 02-01
						22	-9.262+1	7.368+1	1.860+1	-2.318+3	-2.695+3
						23	9.956+1	1.252+2	1.600+1	-3.938+3	2.914+3
						24	7.700+1	-1.579+2	2.000+1	4.968+3	2.422+3
						25	1.785+2	2.244+2	0.0	-1.149+4	9.139+3

Maximum Forces and Moments and Corresponding Times

Max. F_x = 1.992+5 at time = 0.0136
Max F_y = -3.594+5 at time = 0.0136
Max. F_z = 4.602+5 at time = 0.1680
Max. M_x = 2.446+6 at time = 0.0720
Max. M_y = 4.381+6 at time = 0.0680

TABLE 6.2-16

REACTOR CAVITY MODEL CROSS REFERENCES

	22 Node Model	31 Node Model	33 Node Model
Penetration Nodes	Figure 6.2-30	Figure 6.2-30	Figure 6.2-30a
Reactor Vessel Annulus Nodes	Figure 6.2-31	Figure 6.2-31a	Figure 6.2-31a
Overall Model Schematic	Figure 6.2-32	Figure 6.2-32a	Figure 6.2-32a
Control Volumes	Table 6.2-16a	Table 6.2-16b	Table 6.2-16c
Flow Paths	Table 6.2-17	Table 6.2-17a	Table 6.2-17b
Force/Moment Areas	Table 6.2-18	Table 6.2-18a	Table 6.2-18b

TABLE 6.2-16a

22 NODE REACTOR CAVITY MODEL
CONTROL VOLUMES

Control Volume No	Description	Height (ft)	Volume (ft^3)	Volume Flow Area (ft^2)	Bottom Elev. (ft)	Initial Conditions Temp (°F)	Pressure (psia)	Quality	Calc. Peak Diff. press(1) (psi)	Design Diff. press.(2) (psi)	Design Margin (%)	
												02-01
1	Reactor Vessel Annulus -	2.56	14.4	6.5	434.67	212	14.7	0.573	5.4	22.4	310	
2	See Figure 6.2-31								18.3	43.4	140	
3									5.4	22.4	310	
4	Reactor Vessel Annulus -	3.92	23.3	7.3	430.75	212	14.7	0.573	5.3	22.4	320	
5	See Figure 6.2-31								18.7	105.0	460	
6									5.3	22.4	320	
7	Reactor Vessel Annulus -	4.08	24.2	7.3	426.67	212	14.7	0.573	4.2	22.4	430	
8	See Figure 6.2-31								15.1	105.0	590	
9									4.2	22.4	430	
10	Reactor Vessel Annulus -	7.42	35.8	7.0	419.25	212	14.7	0.573	4.2	30.0	610	
11	See Figure 6.2-31								8.0	198.0	2370	
12									4.2	30.0	610	02-01
13	Reactor Vessel Annulus -	7.42	35.8	7.0	411.83	212	14.7	0.573	3.7	30.0	710	
14	See Figure 6.2-31								4.0	48.0	1100	
15									3.7	30.0	710	
16	Under Reactor Vessel	24.16	5528.	270.	386.67	212	14.7	0.573	0.9	30.0	3230	02-01
17	Instrument Chase 1	8.5	4439.	130.5	387.5	212	14.7	0.573	0.8	-	-	
18	Instrument Chase 2	32.	3472.	108.5	396.	212	14.7	0.573	0.3	-	-	
19	Penetration - Break Node - See Figure 6.2-30	7.57	163.4	61.9	427.76	212	14.7	0.573	237.7	483.0	100	
20	Penetration - Pipe Sleeve - See Figure 6.2-30	5.83	46.0	15.5	427.83	212	14.7	0.573	207.2	280.	30	
21	Penetration - Inspection Port - See Figure 6.2-30	1.88	25.9	13.8	435.33	212	14.7	0.573	146.1	483.0	230	
22	Containment	200.	1.82+6	9100.	412.	212	14.7	0.545	-	-	-	

Notes:

1. With respect to containment.
2. Values to be multiplied by dynamic load factor of 1.5 for purposes of structural design.

TABLE 6.2-16b

31 NODE REACTOR CAVITY MODEL
CONTROL VOLUMES

Control Volume No	Description	Height (ft)	Volume (ft^3)	Volume Flow Area (ft^2)	Bottom Elev. (ft)	Initial Conditions			Calc. Peak Diff. press[1] (psi)	Design Diff. press.[2] (psi)	Design Margin (%)	
						Temp (°F)	Pressure (psia)	Quality				02-01
1	Reactor Vessel Annulus -	2.56	8.4	4.1	434.67	212	14.7	0.573	4.9	22.4	360	
3	See Figure 6.2-31a								16.0	43.4	170	02-01
5									6.3	22.4	250	
2	Reactor Vessel Annulus -	2.56	6.0	3.2	434.67	212	14.7	0.573	7.7	22.4	190	
4	See Figure 6.2-31a								15.9	77.0	380	
6									4.9	22.4	360	
7	Reactor Vessel Annulus -	3.92	13.6	4.8	430.75	212	14.7	0.573	4.8	22.4	370	
9	See Figure 6.2-31a								16.4	105.0	540	
11									6.2	22.4	260	
8	Reactor Vessel Annulus -	3.92	9.7	3.9	430.75	212	14.7	0.573	7.6	23.1	200	
10	See Figure 6.2-31a								16.8	77.0	360	
12									4.9	22.4	360	
13	Reactor Vessel Annulus -	4.08	14.1	4.9	426.67	212	14.7	0.573	4.2	22.4	430	
15	See Figure 6.2-31a								14.1	105.0	640	
17									4.3	22.4	420	
14	Reactor Vessel Annulus -	4.08	10.1	3.9	426.67	212	14.7	0.573	4.2	23.1	450	
16	See Figure 6.2-31a								13.9	77.0	450	
18									4.2	22.4	430	
19	Reactor Vessel Annulus -	7.42	35.8	4.8	419.25	212	14.7	0.573	4.1	30.0	630	
20	See Figure 6.2-31a								6.8	198.0	2810	
21									4.1	30.0	630	
22	Reactor Vessel Annulus -	7.42	35.8	4.8	411.83	212	14.7	0.573	3.7	30.0	710	
23	See Figure 6.2-31a								3.8	48.0	1160	
24									3.7	30.0	710	
25	Under Reactor Vessel	24.16	5528.	270.	387.67	212	14.7	0.573	0.9	30.0	3230	
26	Instrument Chase 1	8.5	4439.	130.5	387.5	212	14.7	0.573	0.8	-	-	
27	Instrument Chase 2	32.	3472.	108.5	396.	212	14.7	0.573	0.3	-	-	

TABLE 6.2-16b (Continued)

31 NODE REACTOR CAVITY MODEL
CONTROL VOLUMES

Control Volume No	Description	Height (ft)	Volume (ft^3)	Volume Flow Area (ft^2)	Bottom Elev. (ft)	Initial Conditions Temp (°F)	Pressure (psia)	Quality	Calc. Peak Diff. press[(1)] (psi)	Design Diff. press.[(2)] (psi)	Design Margin (%)	
28	Penetration - Break Node -See Figure 6.2-30	7.57	163.4	61.9	427.76	212	14.7	0.573	237.7	483.0	100	02-01
29	Penetration - Pipe Sleeve - See Figure 6.2-30	5.83	46.0	15.5	427.83	212	14.7	0.573	207.2	280.	30	
30	Penetration - Inspection Port -See Figure 6.2-30	1.88	25.9	13.8	435.33	212	14.7	0.573	146.1	483.0	230	02-01
31	Containment	200.	1.82+6	9100.	412.	212	14.7	0.545	-	-	-	

Notes:

1. With respect to containment.
2. Values to be multiplied by dynamic load factor of 1.5 for purposes of structural design.

TABLE 6.2-16c

33 NODE REACTOR CAVITY MODEL
CONTROL VOLUMES

Control Volume No	Description	Height (ft)	Volume (ft^3)	Volume Flow Area (ft^2)	Bottom Elev. (ft)	Initial Conditions Temp (°F)	Initial Conditions Pressure (psia)	Initial Conditions Quality	Calc. Peak Diff. press(1) (psi)	Design Diff. press.(2) (psi)	Design Margin (%)
1	Reactor Vessel Annulus -	2.56	8.4	4.1	434.67	212	14.7	0.573	4.9	22.4	350
3	See Figure 6.2-31a								16.1	43.4	170
5									6.3	22.4	250
2	Reactor Vessel Annulus -	2.56	6.0	3.2	434.67	212	14.7	0.573	7.7	22.4	190
4	See Figure 6.2-31a								16.0	77.0	380
6									5.0	22.4	350
7	Reactor Vessel Annulus -	3.92	13.6	4.8	430.75	212	14.7	0.573	4.9	22.4	360
9	See Figure 6.2-31a								16.5	105.0	530
11									6.2	22.4	260
8	Reactor Vessel Annulus -	3.92	9.7	3.9	430.75	212	14.7	0.573	7.7	23.1	250
10	See Figure 6.2-31a								16.7	77.0	360
12									4.9	22.4	350
13	Reactor Vessel Annulus -	4.08	14.1	4.9	426.67	212	14.7	0.573	4.2	22.4	430
15	See Figure 6.2-31a								14.3	105.0	630
17									4.4	22.4	410
14	Reactor Vessel Annulus -	4.08	10.1	3.9	426.67	212	14.7	0.573	4.3	23.1	440
16	See Figure 6.2-31a								14.1	77.0	440
18									4.2	22.4	430
19	Reactor Vessel Annulus -	7.42	35.8	7.0	419.25	212	14.7	0.573	4.2	30.0	610
20	See Figure 6.2-31a								6.5	198.0	2930
21									4.2	30.0	610
22	Reactor Vessel Annulus -	7.42	35.8	7.0	411.83	212	14.7	0.573	3.7	30.0	700
23	See Figure 6.2-31a								3.8	48.0	1360
24									3.7	30.0	710
25	Under Reactor Vessel	24.16	5528.	270.	386.67	212	14.7	0.573	0.9	30.0	3270
26	Instrument Chase 1	8.5	4439.	130.5	387.5	212	14.7	0.573	0.8	-	-
27	Instrument Chase 2	32.	3472.	108.5	396.	212	14.7	0.573	0.3	-	-

02-01

TABLE 6.2-16c (Continued)

33 NODE REACTOR CAVITY MODEL
CONTROL VOLUMES

Control Volume No	Description	Height (ft)	Volume (ft^3)	Volume Flow Area (ft^2)	Bottom Elev. (ft)	Initial Conditions Temp (°F)	Pressure (psia)	Quality	Calc. Peak Diff. press(1) (psi)	Design Diff. press.(2) (psi)	Design Margin (%)	02-01
28	Penetration - Break Node 1 - See Figure 6.2-30a	2.99	63.7	45.3	427.76	212	14.7	0.573	242.3 (2)(3) 195.2 (2)(4)	483.0 195.2	100 0	02-01
29	Penetration - Break Node 2 - See Figure 6.2-30a	4.58	99.7	45.3	430.75	212	14.7	0.573	236.3 (2)(3) 188.5 (2)(4)	483.0 188.5	100 0	02-01
30	Penetration - Pipe Sleeve 1 -See Figure 6.2-30a	2.92	23.0	7.8	427.83	212	14.7	0.573	211.7	280.	30	
31	Penetration - Pipe Sleeve 2 -See Figure 6.2-30a	2.92	23.0	7.8	430.75	212	14.7	0.573	206.5	280.	30	
32	Penetration - Inspection Port -See Figure 6.2-30a	1.88	25.9	13.8	435.33	212	14.7	0.573	145.2 (2)(3) 109.2 (2)(4)	483.0 109.2	230 0	02-01
33	Containment	200.	1.82+6	9100.	412.	212	14.7	0.545	-	-	-	

Notes:

1. With respect to containment.
2. The following notes 3 and 4 are applicable to the design of the reactor baffle system.
3. These values are for 150 sq. ins. break and apply to original design of the baffles before the deign details were changed in 1978 to accommodate a revised construction sequence. These loads were increased by a dynamic load factor of 1.5 for structural design.
4. For Nodes 28, 29 and 32 the peak differential pressures were also evaluated for a break size of 100 sq. ins. in 1978 and 127 sq. ins. in 1988. The 127 sq. in. break values were obtained by interpolation of the pressure/temperature analyses for the 100 sq. in. and 150 sq. in. breaks. The calculated peak and design differential pressures (with respect to annulus) shown are for the 127 sq. in. break which is the maximum expected break size that can occur. These values were increased by a dynamic load factor of 1.2 for structural design.

TABLE 6.2-17

22 NODE REACTOR CAVITY MODEL JUNCTIONS

02-01

Junction No.	Description	Control Volume 1	Control Volume 2	Choking Model	Elevation (ft)	Flow Area (ft^2)	Inertia (ft^{-1})	Head Loss, K[(1)] Friction fL/D	Bends	Expansion	Contraction	Total
1	Reactor Vessel Annulus -	1	2	Inertia	435.95	0.85	20.0	0.62	0.0	0.0	0.0	0.62
2	See Figure 6.2-31	2	3									
3		3	1									
4	Reactor Vessel Annulus -	1	22	HEM[(2)]	437.23	0.72	0.69	0.001	1.2	1.0	0.0	2.2
5	See Figure 6.2-31	2	22									
6		3	22									
7	Reactor Vessel Annulus -	1	4	HEM	434.67	5.8	0.56	0.092	0.0	0.0	0.0	0.092
8	See Figure 6.2-31	2	5									
9		3	6									
10	Reactor Vessel Annulus -	4	5	HEM	432.71	0.60	19.5	0.20	0.0	1.0	0.69	1.9
11	See Figure 6.2-31	5	6									
12		6	4									
13	Reactor Vessel Annulus -	4	7	HEM	430.75	2.4	1.4	0.11	0.0	0.36	0.26	0.73
14	See Figure 6.2-31	5	8									
15		6	9									
16	Reactor Vessel Annulus -	7	8	HEM	428.71	0.65	17.6	0.20	0.0	1.09	0.70	2.0
17	See Figure 6.2-31	8	9									
18		9	7									
19	Reactor Vessel Annulus -	7	10	HEM	426.67	4.8	1.1	0.079	0.0	0.0	0.0	0.079
20	See Figure 6.2-31	8	11									
21		9	12									
22	Reactor Vessel Annulus -	10	11	Inertia	422.96	2.2	7.3	0.53	0.0	0.0	0.0	0.53
23	See Figure 6.2-31	11	12									
24		12	10									
25	Reactor Vessel Annulus -	10	13	Inertia	419.25	4.8	1.5	0.25	0.0	0.0	0.0	0.25
26	See Figure 6.2-31	11	14									
27		12	15									
28	Reactor Vessel Annulus -	13	14	Inertia	415.54	2.2	7.3	0.53	0.0	0.0	0.0	0.53
29	See Figure 6.2-31	14	15									
30		15	13									

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Amendment 02-01

TABLE 6.2-17 (Continued)

22 NODE REACTOR CAVITY MODEL JUNCTIONS

Junction No.	Description	Control Volume 1	Control Volume 2	Choking Model	Elevation (ft)	Flow Area (ft^2)	Inertia (ft^{-1})	Head Loss, K(1) Friction fL/D	Bends	Expansion	Contraction	Total
31	Reactor Vessel Annulus -	13	16	HEM	411.83	4.8	0.85	0.12	0.0	1.0	0.0	1.12
32	See Figure 6.2-31	14	16									
33		15	16									
34	Instrument Chase	16	17	HEM	391.83	130.5	0.21	0.02	1.2	0.0	0.05	1.27
35	Instrument Chase	17	18	HEM	396.0	81.4	0.31	0.03	(3)	(3)	0.02	0.35
36	Instrument Chase	18	22	HEM	420.0	29.0	0.089	0.0	0.0	1.0	0.5	1.5
37	Instrument Chase	18	22	HEM	420.0	41.2	0.27	0.0	0.0	1.0	0.5	1.5
38	From Break Node - See Figure 6.2-30	19	20	HEM	430.75	11.7	0.21	0.02	0.0	0.0	0.28	0.30
39	From Break Node - See Figure 6.2-30	19	21	HEM	435.33	13.8	0.23	0.01	0.0	0.04	0.22	0.27
40 41	From Break Node - See Figure 6.2-30	19	5	HEM	430.75	0.43	2.8	0.16	0.0	0.13	0.5	0.79
42 43	From Break Node - See Figure 6.2-30	19	8	HEM	430.75	0.25	1.6	0.16	0.0	0.13	0.5	0.79
44	Penetration to Contain.	20	22	HEM	430.75	11.7	0.14	0.02	0.0	1.0	0.0	1.02
45	Penetration to Contain.	21	22	HEM	437.21	13.8	0.07	0.01	0.0	1.0	0.0	1.01
46 47	From Reactor Vessel Annulus to Contain. through Penetr.	4	22	HEM	430.75	0.73	3.1	0.17	0.0	1.0	0.05	1.67
48 49	From Reactor Vessel Annulus to Contain. through Penetr.	5	22	HEM	430.75	0.30	3.5	0.17	0.0	1.0	0.05	1.67
50 51	From Reactor Vessel Annulus to Contain. through Penetr.	6	22	HEM	430.75	0.73	3.1	0.17	0.0	1.0	0.05	1.67

02-01

TABLE 6.2-17 (Continued)

22 NODE REACTOR CAVITY MODEL JUNCTIONS

Junction No.	Description	Control Volume 1	Control Volume 2	Choking Model	Elevation (ft)	Flow Area (ft^2)	Inertia (ft^{-1})	Head Loss, K(1) Friction fL/D	Bends	Expansion	Contraction	Total
52 53	From Reactor Vessel Annulus to Contain. through Penetr.	7	22	HEM	430.75	0.42	1.8	0.17	0.0	1.0	0.5	1.67
54 55	From Reactor Vessel Annulus to Contain. through Penetr.	8	22	HEM	430.75	0.17	2.0	0.17	0.0	1.0	0.5	1.67
56 57	From Reactor Vessel Annulus to Contain. through Penetr.	9	22	HEM	430.75	0.42	1.8	0.17	0.0	1.0	0.5	1.67
58	Break Path	19	0	None								

02-01

NOTES:

1. With respect to the minimum area.
2. Homogeneous equilibrium model.
3. K grating = 0.30.

02-01

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Amendment 02-01

TABLE 6.2-17a

31 NODE REACTOR CAVITY MODEL JUNCTIONS

Junction No.	Description	Control Volume 1	Control Volume 2	Choking Model	Elevation (ft)	Flow Area (ft^2)	Inertia (ft^{-1})	Head Loss, K[1]					
								Friction fL/D	Bends	Expansion	Contraction	Total	02-01
1	Reactor Vessel	1	2	Inertia	435.95	0.85	10.0	0.31	0.0	0.0	0.0	0.31	
2	Annulus - See	2	3										
3	Figure 6.2-31a	3	4										
4		4	5										
5		5	6										
6		6	1										
7	Reactor Vessel	1	31	HEM [2]	437.23	0.42	1.2	0.001	1.2	1.0	0.0	2.2	02-01
8	Annulus - See	3	31										
9	Figure 6.2-31a	5	31										
10	Reactor Vessel	2	31	HEM	437.23	0.30	1.7	0.001	1.2	1.0	0.0	2.2	
11	Annulus - See	4	31										
12	Figure 6.2-31a	6	31										
13	Reactor Vessel	1	7	HEM	434.67	3.4	0.96	0.092	0.0	0.0	0.0	0.092	
14	Annulus - See	3	9										
15	Figure 6.2-31a	5	11										
16	Reactor Vessel	2	8	HEM	434.67	2.4	1.3	0.092	0.0	0.0	0.0	0.092	
17	Annulus - See	4	10										
18	Figure 6.2-31a	6	12										
19	Reactor Vessel	7	8	HEM	432.71	0.48	10.6	0.10	0.0	0.42	0.28	0.80	
20	Annulus - See	9	10										
21	Figure 6.2-31a	11	12										
22	Reactor Vessel	8	9	HEM	432.71	0.60	8.9	0.11	0.0	0.34	0.25	0.70	
23	Annulus - See	10	11										
24	Figure 6.2-31a	12	7										
25	Reactor Vessel	7	13	HEM	430.75	1.68	2.1	0.11	0.0	0.25	0.22	0.58	
26	Annulus - See	9	15										
27	Figure 6.2-31a	11	17										
28	Reactor Vessel	8	14	HEM	430.75	0.69	4.3	0.11	0.0	0.52	0.32	0.95	
29	Annulus - See	10	16										
30	Figure 6.2-31a	12	18										

TABLE 6.2-17a (Continued)

31 NODE REACTOR CAVITY MODEL JUNCTIONS

02-01

Junction No.	Description	Control Volume 1	Control Volume 2	Choking Model	Elevation (ft)	Flow Area (ft^2)	Inertia (ft^{-1})	Head Loss, K[(1)] Friction fL/D	Bends	Expansion	Contraction	Total
31	Reactor Vessel	13	14	HEM	428.71	0.54	9.2	0.10	0.0	0.53	0.32	0.95
32	Annulus - See	15	16									
33	Figure 6.2-31a	17	18									
34	Reactor Vessel	14	15	HEM	428.71	0.65	8.4	0.11	0.0	0.32	0.24	0.67
35	Annulus - See	16	17									
36	Figure 6.2-31a	18	13									
37	Reactor Vessel	13	19	HEM	426.67	2.8	1.4	0.079	0.0	0.0	0.0	0.079
38	Annulus - See	15	20									
39	Figure 6.2-31a	17	21									
40	Reactor Vessel	14	19	HEM	426.67	2.0	1.6	0.079	0.0	0.0	0.0	0.079
41	Annulus - See	16	20									
42	Figure 6.2-31a	18	21									
43	Reactor Vessel	19	20	Inertia	422.96	2.2	7.3	0.53	0.0	0.0	0.0	0.53
44	Annulus - See	20	21									
45	Figure 6.2-31a	21	19									
46	Reactor Vessel	19	22	Inertia	419.25	4.8	1.5	0.25	0.0	0.0	0.0	0.25
47	Annulus - See	20	23									
48	Figure 6.2-31a	21	24									
49	Reactor Vessel	22	23	Inertia	415.54	2.2	7.3	0.53	0.0	0.0	0.0	0.53
50	Annulus - See	23	24									
51	Figure 6.2-31a	24	22									
52	Reactor Vessel	22	25	HEM	411.83	4.8	0.85	0.12	0.0	1.0	0.0	1.12
53	Annulus - See	23	25									
54	Figure 6.2-31a	24	25									
55	Instrument Chase	25	26	HEM	391.83	130.5	0.21	0.02	1.2	0.0	0.05	1.27
56	Instrument Chase	26	27	HEM	396.	81.4	0.31	0.03	(3)	(3)	0.02	0.35
57	Instrument Chase	27	31	HEM	420.0	29.0	0.089	0.0	0.0	1.0	0.5	1.5
58	Instrument Chase	27	31	HEM	420.0	41.2	0.27	0.0	0.0	1.0	0.5	1.5

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TABLE 6.2-17a (Continued)

31 NODE REACTOR CAVITY MODEL JUNCTIONS

Junction No.	Description	Control Volume 1	Control Volume 2	Choking Model	Elevation (ft)	Flow Area (ft^2)	Inertia (ft^{-1})	Head Loss, K[(1)] Friction fL/D	Bends	Expansion	Contraction	Total
59	From Reactor Vessel Annulus	7	31	HEM	430.75	0.73	3.1	0.17	0.0	1.0	0.5	1.67
60	to Contain. through Penetr.	8	31									
61	From Reactor Vessel Annulus	9	31	HEM	430.75	0.30	3.5	0.17	0.0	1.0	0.5	1.67
62	to Contain. through Penetr.	10	31									
63	From Reactor Vessel Annulus	11	31	HEM	430.75	0.73	3.1	0.17	0.0	1.0	0.5	1.67
64	to Contain. through Penetr.	12	31									
65	From Reactor Vessel Annulus	13	31	HEM	430.75	0.42	1.8	0.17	0.0	1.0	0.5	1.67
66	to Contain. through Penetr.	14	31									
67	From Reactor Vessel Annulus	15	31	HEM	430.75	0.17	2.0	0.17	0.0	1.0	0.5	1.67
68	to Contain. through Penetr.	16	31									
69	From Reactor Vessel Annulus	17	31	HEM	430.75	0.42	1.8	0.17	0.0	1.0	0.5	1.67
70	to Contain. through Penetr.	18	31									
71	From Break Node - See Figure 6.2-30	28	29	HEM	430.75	11.7	0.21	0.02	0.0	0.0	0.28	0.30
72	From Break Node - See Figure 6.2-30	28	30	HEM	435.33	13.8	0.23	0.01	0.0	0.04	0.22	0.27
73	From Break Node - See	28	9	HEM	430.75	0.43	2.8	0.16	0.0	0.13	0.5	0.79
74	Figure 6.2-30	28	10									
75	From Break Node - See	28	15	HEM	430.75	0.25	1.6	0.16	0.0	0.13	0.5	0.79
76	Figure 6.2-30	28	16									
77	Penetration to Contain.	29	31	HEM	430.75	11.7	0.14	0.02	0.0	1.0	0.0	1.02
78	Penetration to Contain.	30	31	HEM	437.21	13.8	0.07	0.01	0.0	1.0	0.0	1.01
79	Break Path	28	0	None								

02-01

02-01

NOTES:
1. With respect to the minimum area.
2. Homogeneous equilibrium model.
3. K grating = 0.30.

02-01

TABLE 6.2-17b

33 NODE REACTOR CAVITY MODEL JUNCTIONS

Junction No.	Description	Control Volume 1	Control Volume 2	Choking Model	Elevation (ft)	Flow Area (ft^2)	Inertia (ft^{-1})	Head Loss, K[1] Friction fL/D	Bends	Expansion	Contraction	Total	
													02-01
1 through 70	Same as 31 Node Model - See Table 6.2-17a - Except Containment is Node 33												
71	From Break Node -	28	15	HEM[2]	429.26	0.25	1.6	0.16	0.0	0.13	0.5	0.79	
72	See Figure 6.2-30a	28	16										
73	From Break Node -	29	9	HEM	433.04	0.43	2.8	0.16	0.0	0.13	0.5	0.79	
74	See Figure 6.2-30a	29	10										
75	From Break Node - See Figure 6.2-30a	28	29	HEM	430.75	13.3	0.34	0.01	0.0	0.28	0.23	0.52	
76	From Break Node - See Figure 6.2-30a	28	30	HEM	429.26	5.85	0.38	0.04	0.0	0.0	0.32	0.36	
77	From Break Node - See Figure 6.2-30a	29	31	HEM	432.25	5.85	0.38	0.04	0.0	0.0	0.32	0.36	
78	From Break Node - See Figure 6.2-30a	29	32	HEM	435.33	13.8	0.15	0.01	0.0	0.0	0.22	0.23	
79	Penetration to Contain.	30	33	HEM	429.26	5.85	0.28	0.02	0.0	1.0	0.0	1.02	02-01
80	Penetration to Contain.	31	33	HEM	432.25	5.85	0.28	0.02	0.0	1.0	0.0	1.02	02-01
81	Penetration to Contain.	32	33	HEM	437.21	13.8	0.07	0.01	0.0	1.0	0.0	1.01	
82	Break Path	28	0	None									
83	Break Path	29	0	None									

NOTES:

1. With respect to minimum area.
2. Homogeneous equilibrium area. 02-01

TABLE 6.2-18

22 NODE REACTOR CAVITY MODEL FORCE/MOMENT AREAS

Control Volume No.	Description	A_x (ft^2)[1]	A_y (ft^2)	A_z (ft^2) [2]	M_x(ft^3) [3]	M_y(ft^3)
1	Reactor Vessel Annulus	-21.48	-25.59	15.34	217.71	-170.95
2	See Figure 6.2-31	32.89	-5.80	15.34	39.14	273.93
3		-11.41	31.39	15.34	-256.85	-102.98
4		-26.28	-28.88	-12.08	56.61	-51.51
5		39.49	-7.92	-12.08	15.52	77.40
6		-13.21	36.80	-12.08	-72.13	-25.89
7		-27.57	-30.44	13.04	-57.14	52.78
8		41.47	-8.27	13.04	-16.34	-78.58
9		-13.90	38.71	13.04	73.48	25.80
10		-59.80	-71.23	0.	-554.9	465.8
11		91.58	-16.16	0.	-125.9	-713.4
12		-31.78	87.39	0.	680.8	247.6
13		-59.80	-71.23	0.	-1083.	910.
14		91.58	-16.16	0.	-246.	-1393.
15		-31.78	87.39	0.	1329.	483.
16	Under Reactor Vessel	0.	0.	164.42	0.	0.
17	Instrument Chase - 1	0.	0.	0.	0.	0.
18	Instrument Chase - 2	0.	0.	0.	0.	0.
19	Penetration - Break Node - See Figure 6.2-30	14.93	0.	0.	0.	3.02
20	Penetration - Pipe Sleeve - See Figure 6.2-30	0.	0.	0.	0.	0.
21	Penetration - Inspection Port See Figure 6.2-30	0.	0.	0.	0.	0.
22	Containment	-14.93	0.	0.	0.	-3.02

02-01

NOTES:
1. Net area for force calculation.
2. F_Z calculations use nodal pressure minus containment pressure.
3. Net area times moment arm for moment calculations.

02-01

TABLE 6.2-18a

31 NODE REACTOR CAVITY MODEL FORCE/MOMENT AREAS

Control Volume No.	Description	A_x (ft^2)$^{(1)}$	A_y (ft^2)	A_z (ft^2) $^{(2)}$	M_x(ft^3) $^{(3)}$	M_y(ft^3)
1	Reactor Vessel Annulus	-20.05	-9.35	7.67	73.84	-158.31
2	See Figure 6.2-31a	-1.43	-16.24	7.67	143.87	-12.64
3		18.12	-12.68	7.67	100.14	143.08
4		14.77	6.88	7.67	-61.00	130.85
5		1.93	22.04	7.67	-174.03	15.24
6		-13.34	9.35	7.67	-82.82	-118.22
7		-24.75	-11.54	-6.04	22.62	-48.51
8		-1.53	-17.34	-6.04	33.99	-3.00
9		22.58	-15.80	-6.04	30.97	44.26
10		16.91	7.88	-6.04	-15.45	33.14
11		2.27	25.95	-6.04	-50.86	4.45
12		-15.48	10.85	-6.04	-21.27	-30.34
13		-25.96	-12.11	6.52	-23.23	49.80
14		-1.61	-18.33	6.52	-33.91	2.98
15		23.67	-16.57	6.52	-31.80	-45.53
16		17.80	8.30	6.52	15.46	-33.15
17		2.39	27.29	6.52	52.19	-4.57
18		-16.29	11.42	6.52	21.29	30.37
19		-59.80	-71.23	0.	-554.9	465.8
20		91.58	-16.16	0.	-125.9	-713.4

02-01

TABLE 6.2-18a (Continued)

31 NODE REACTOR CAVITY MODEL FORCE/MOMENT AREAS

Control Volume No.	Description	A_x (ft^2)[(1)]	A_y (ft^2)	A_z (ft^2) [(2)]	M_x(ft^3) [(3)]	M_y(ft^3)
21		-31.78	87.39	0.	680.8	247.6
22		-59.80	-71.23	0.	-1083.	910.
23		91.58	-16.16	0.	-246.	-1393.
24		-31.78	87.39	0.	1329.	483.
25	Under Reactor Vessel	0.	0.	164.42	0.	0.
26	Instrument Chase - 1	0.	0.	0.	0.	0.
27	Instrument Chase - 2	0.	0.	0.	0.	0.
28	Penetration - Break Node See Figure 6.2-30	14.93	0.	0.	0.	3.02
29	Penetration - Pipe Sleeve See Figure 6.2-30	0.	0.	0.	0.	0.
30	Penetration - Inspection Port See Figure 6.2-30	0.	0.	0.	0.	0.
31	Containment	-14.93	0.	0.	0.	-3.02

NOTES:

1. Net area for force calculation. 02-01
2. F_z calculations use nodal pressure minus containment pressure.
3. Net area times moment arm for moment calculations.

TABLE 6.2-18b

33 NODE REACTOR CAVITY MODEL FORCE/MOMENT AREAS

Control Volume No.	Description	A_x (ft^2)$^{(1)}$	A_y (ft^2)	A_z (ft^2) $^{(2)}$	M_x(ft^3) $^{(3)}$	M_y(ft^3)
1 through 24	Reactor Vessel Annulus - Same as 31 Node Model See Table 6.2-18a					
25	Under Reactor Vessel	0.	0.	164.42	0.	0.
26	Instrument Chase - 1	0.	0.	0.	0.	0.
27	Instrument Chase - 2	0.	0.	0.	0.	0.
28	Penetration - Break Node 1 - See Figure 6.2-30a	6.81	0.	15.45	0.	-7.70
29	Penetration - Break Node 2 - See Figure 6.2-30a	8.12	0.	-15.45	0.	10.72
30	Penetration - Pipe Sleeve 1 - See Figure 6.2-30a	0.	0.	0.	0.	0.
31	Penetration - Pipe Sleeve 2 - See Figure 6.2-30a	0.	0.	0.	0.	0.
32	Penetration - Inspection Port - See Figure 6.2-30a	0.	0.	0.	0.	0.
33	Containment	-14.93	0.	0.	0.	-3.02

02-01

NOTES:

1. Net area for force calculation.
2. F_z calculations use nodal pressure minus containment pressure.
3. Net area times moment arm for moment calculations.

02-01

TABLE 6.2-18c

REACTOR CAVITY MODEL RESULTS
CROSS REFERENCE

	22 Node Model	31 Node Model	33 Node Model	
Peak Differential Pressure	Table 6.2-16a	Table 6.2-16b	Table 6.2-16c	
Transient Differential Pressure for Selected Nodes	Figures 6.2-33 6.2-33a	Figures 6.2-33b 6.2-33c	Figures 6.2-33d 6.2-33e	02-01
Forces and Moments	Table 6.2-18d	Table 6.2-18e	Table 6.2-18f; Figures 6.2-34, 6.2-34a, 6.2-34b 6.2-35, 6.2-35a	

TABLE 6.2-18d

22 NODE REACTOR CAVITY MODEL FORCES AND MOMENTS

Time (sec)	F_x (lbf)	F_y (lbf)	F_z (lbf)	M_x (ft-lbf)	M_y (ft-lbf)	
						02-01
0.0	0.0	0.0	0.0	0.0	0.0	
0.0051	1.252+5	-5.636+3	1.328+2	2.598+3	3.774+4	
0.01	2.785+5	-1.851+4	9.343+3	4.779+3	9.869+4	
0.0148	4.335+5	-3.352+4	1.800+4	-2.219+4	-1.030+4	02-01
0.02	5.802+5	-4.879+4	2.382+4	-6.805+4	-2.386+5	
0.0256	6.755+5	-5.823+4	3.557+4	-8.64+4 (1)	-3.098+5	
0.0296	7.108+5	-5.993+4 (1)	4.054+4	-8.327+4	-2.821+5	
0.032	7.236+5	-5.963+4	4.204+4	-8.122+4	-2.693+5	
0.04	7.436+5 (1)	-5.592+4	5.016+4	-7.764+4	-2.443+5	
0.05	7.227+5	-4.764+4	5.948+4	-2.425+4	5.758+4	
0.06	7.034+5	-4.161+4	6.548+4	5.131+4	4.884+5	02-01
0.067	7.056+5	-4.082+4	6.753+4	6.777+4	5.811+5 (1)	
0.07	7.094+5	-4.110+4	6.838+4	6.615+4	5.724+5	
0.08	7.23+5	-4.258+4	7.033+4	4.719+4	4.679+5	
0.09	7.253+5	-4.236+4	7.159+4	3.905+4	4.210+5	
0.10	7.166+5	-4.04+4	7.27+4	4.73+4	4.655+5	
0.11	7.074+5	-3.849+4	7.36+4	5.774+4	5.23+5	
0.12	7.048+5	-3.78+4	7.43+4	6.027+4	5.35+5	
0.13	7.074+5	-3.815+4	7.491+4	5.506+4	5.083+5	
0.14	7.101+5	-3.855+4	7.557+4	5.059+4	4.818+5	
0.15	7.108+5	-3.86+4	7.626+4	4.917+4	4.739+5	
0.16	7.102+5	-3.842+4	7.677+4	4.974+4	4.766+5	
0.167	7.098+5	-3.829+4	7.689+4 (1)	5.026+4	4.794+5	02-01
0.18	7.097+5	-3.822+4	7.638+4	5.024+4	4.795+5	
0.19	7.103+5	-3.827+4	7.527+4	4.974+4	4.767+5	02-01
0.20	7.108+5	-3.832+4	7.361+4	4.948+4	4.754+5	

NOTE:

1. Maximum absolute value.

TABLE 6.2-18e

31 NODE REACTOR CAVITY MODEL FORCES AND MOMENTS

Time (sec)	F_x (lbf)	F_y (lbf)	F_z (lbf)	M_x (ft-lbf)	M_y (ft-lbf)	
0.0	0.0	0.0	0.0	0.0	0.0	
0.0051	1.194+5	-2.882+3	-1.201+2	-8.95+2	3.119+4	
0.01	2.734+5	-1.187+4	7.719+3	-1.731+4	6.836+4	
0.0152	4.329+5	-2.448+4	1.663+4	-5.663+4	-8.696+4	
0.02	5.492+5	-3.386+4	2.168+4	-8.78+4	-2.779+5	
0.0296	6.4+5	-3.852+4	3.667+4	-5.224+4	-1.84+5	
0.04	6.76+5	-3.167+4	4.637+4	-3.138+4	-6.723+4	
0.05	6.899+5	-3.046+4	5.484+4	-7.078+3	1.035+5	
0.06	6.917+5	-3.367+4	6.059+4	6.952+4	3.651+5	
0.07	6.959+5	-3.751+4	6.422+4	1.005+5 (1)	4.269+5	
0.077	6.974+5 (1)	-3.898+4	6.622+4	8.824+4	4.153+5	
0.082	6.968+5	-3.930+4 (1)	6.721+4	7.552+4	4.084+5	
0.09	6.934+5	-3.834+4	6.861+4	6.864+4	4.164+5	
0.10	6.874+5	-3.583+4	6.978+4	8.091+4	4.486+5	
0.11	6.835+5	-3.446+4	7.063+4	8.913+4	4.737+5	02-01
0.115	6.83+5	-3.465+4	7.098+4	8.50+4	4.778+5 (1)	
0.13	6.854+5	-3.608+4	7.209+4	7.159+4	4.551+5	
0.14	6.869+5	-3.657+4	7.290+4	6.757+4	4.414+5	
0.15	6.87+5	-3.653+4	7.365+4	6.613+4	4.396+5	
0.16	6.866+5	-3.625+4	7.416+4	6.642+4	4.424+5	
0.166	6.864+5	-3.611+4	7.426+4 (1)	6.648+4	4.431+5	
0.18	6.866+5	-3.601+4	7.37+4	6.60+4	4.415+5	
0.19	6.87+5	-3.604+4	7.258+4	6.572+4	4.395+5	
0.20	6.874+5	-3.606+4	7.092+4	6.594+4	4.399+5	

NOTE:
1. Maximum absolute value.

TABLE 6.2-18f

33 NODE REACTOR CAVITY MODEL FORCES AND MOMENTS

Time (sec)	F_x (lbf)	F_y (lbf)	F_z (lbf)	M_x (ft-lbf)	M_y (ft-lbf)
0.0	0.0	0.0	0.0	0.0	0.0
0.0051	1.255+5	-3.274+3	4.615+4	-2.510+3	-4.645+3
0.01	2.803+5	-1.264+4	8.109+4	-2.483+4	-3.587+4
0.0152	4.418+5	-2.563+4	7.694+4	-6.682+4	-2.205+5
0.020	5.565+5	-3.476+4	6.092+4	-9.087+4	-3.438+5
0.0296	6.403+5	-3.781+4	4.712+4	-4.364+4	-1.23+5 02-01
0.04	6.823+5	-3.114+4	6.923+4	-3.485+4	-8.853+4
0.05	6.988+5	-3.088+4	7.566+4	-9.219+3	4.629+4
0.06	6.982+5	-3.479+4	7.401+4	7.053+4	3.483+5
0.069	6.99+5	-3.808+4	7.738+4	1.005+5 (1)	4.360+5
0.078	7.00+5 (1)	-3.950+4	8.179+4	8.584+4	4.211+5
0.081	6.999+5	-3.960+4 (1)	8.256+4	7.889+4	4.127+5
0.10	6.921+5	-3.614+4	8.348+4	8.241+4	4.365+5
0.11	6.877+5	-3.500+4	8.456+4	8.717+4	4.649+5
0.115	6.869+5	-3.516+4	8.510+4	8.386+4	4.69+5(1)
0.12	6.871+5	-3.556+4	8.552+4	7.956+4	4.652+5
0.13	6.889+5	-3.646+4	8.611+4	7.247+4	4.475+5
0.14	6.905+5	-3.694+4	8.678+4	6.797+4	4.344+5
0.15	6.908+5	-3.692+4	8.756+4	6.608+4	4.302+5
0.16	6.905+5	-3.665+4	8.807+4	6.610+4	4.311+5
0.165	6.903+5	-3.653+4	8.815+4 (1)	6.622+4	4.320+5
0.17	6.902+5	-3.645+4	8.809+4	6.621+4	4.325+5
0.18	6.903+5	-3.640+4	8.755+4	6.59+4	4.318+5
0.19	6.907+5	-3.642+4	8.642+4	6.560+4	4.301+5
0.20	6.911+5	-3.645+4	8.475+4	6.577+4	4.301+5

NOTE:

1. Maximum absolute value.

TABLE 6.2-18g

REACTOR CAVITY NODILIZATION STUDY
PEAK FORCES AND MOMENTS

	22 Node Model	31 Node Model	33 Node Model	
F_x (lbf) (1)	7.436+5	6.974+5	7.00+5	
F_z (lbf)	7.689+4	7.426+4	8.82+4	02-01
M_y (ft-lbf) (1)	5.811+5	4.778+5	4.69+5	

1. F_i, force in the i direction.

   M_y, Moment around the y axis.

TABLE 6.2-19

BLOWDOWN MASS AND ENERGY RELEASES
DOUBLE ENDED PUMP SUCTION GUILLOTINE

02-01

TIME SECONDS	BREAK PATH LBM/SEC	NO. 1 FLOW THOUSAND BTU/SEC	BREAK PATH LBM/SEC	NO. 2 FLOW THOUSAND BTU/SEC
0.000	0.0	0.0	0.0	0.0
0.101	41226.3	22798.3	22509.4	12407.1
0.201	41151.8	22857.9	24598.8	13572.4
0.600	42021.0	24036.7	21813.2	12091.2
0.901	41596.5	24453.0	20262.5	11265.6
1.40	38613.1	23653.5	19519.5	10841.8
2.00	33824.8	21885.0	19227.8	10682.1
2.50	26966.8	18578.2	17874.7	9954.5
2.70	20342.8	14290.5	17235.4	9609.8
2.90	17530.7	12472.6	16664.7	9302.6
3.20	14967.0	10742.3	15945.2	8913.6
4.20	11851.4	8639. 7	14257.3	7960.9
4.80	11211.8	8445.6	13362.6	7447.3
5.20	8680.8	7660.3	13609.6	7581.3
5.40	7650.9	7142.5	13426.4	7477.7
6.20	7694.5	6610.6	12890.6	7182.4
7.20	8353.3	6306.3	12178.7	6784.8
8.00	8031.9	6080.8	11699.2	6511.9
9.80	6443.7	5402.5	10478.1	5829.6
11.6	5529.9	4696.0	9229.6	5146.6
14.2	4260.9	3693.6	7415.4	4158.4
14.6	4128.9	3611.2	7237.4	3944.6
14.8	4059.6	3585.0	8616.4	4612.5
15.0	3961.4	3556.0	6301.6	3367.6
15.2	3886.7	3551.5	11251.0	5853.2
15.4	3704.1	3494.3	11196.1	5851.6
15.6	3677.5	3611.6	4829.9	2511.8
15.8	3477.4	3525.7	9022.3	4356.0
16.0	3270.5	3501.4	8927.5	4374.1
16.2	3090.6	3493.0	4650.6	2292.0
16.6	2467.6	3023.5	8768.5	3941.9
16.8	2123.2	2624.3	5556.8	2540.9
17.2	1618.6	2015.9	3758.2	1743.9
18.4	571.0	720.5	1256.4	759.7
19.6	0.0	0.0	0.0	0.0

Reformatted Per
Amendment 02-01

TABLE 6.2-20

BLOWDOWN MASS AND ENERGY RELEASES
DOUBLE ENDED HOT LEG GUILLOTINE

02-01

TIME SECONDS	BREAK PATH LBM/SEC	NO. 1 FLOW THOUSAND BTU/SEC	BREAK PATH LBM/SEC	NO. 2 FLOW THOUSAND BTU/SEC
0.000	0.0	0.0	0.0	0.0
0.100	40707.9	26964.9	28641.3	18613.5
0.200	37093.4	24611.9	23845.5	15400.7
0.300	35362.1	23347.9	21758.7	13867.2
0.601	34739.7	22929.6	19114.6	11631.3
1.20	31342.0	21403.5	17380.0	10059.9
2.20	23965.5	17302.9	17132.6	9580.2
2.80	21069.8	15272.0	17270.0	9571.5
3.30	19800.2	14176.8	16976.1	9390.0
4.00	19119.2	13341.5	15537.5	8628.8
4.40	19570.4	13333.2	14463.4	8077.3
5.00	21265.5	13802.9	12433.7	7023.0
5.20	14974.3	11011.2	11792.4	6691.5
6.00	16239.9	11358.0	9807.0	5662.3
6.20	17613.8	11934.2	9465.8	5482.3
6.60	26967.4	17319.2	8901.8	5180.9
7.80	26423.9	16211.8	7318.3	4332.4
8.60	24989.3	15306.5	6187.5	3768.5
8.80	15237.9	9142.2	5918.2	3639.9
9.20	15813.8	9546.9	5406.1	3402.4
9.40	10036.9	7348.0	5187.3	3306.9
9.60	9692.6	7209.3	4991.7	3222.3
10.4	11042.5	7754.2	4447.3	2969.2
10.8	13961.6	9556.2	4243.7	2864.6
11.4	12073.9	8363.3	3891.7	2698.3
11.8	5710.9	4926.3	3559.5	2557.1
12.4	4768.5	4341.9	2870.3	2325.5
13.2	2690.8	3036.4	1845.3	1999.9
13.8	1959.9	2320.3	1118.0	1385.5
15.6	882.5	1101.1	565.1	711.9
16.2	921.1	961.9	320.5	407.1
16.4	329.8	405.2	412.8	524.3
17.0	1043.3	749.3	188.9	240.7
17.4	0.0	0.0	119.2	152.8
18.2	0.0	0.0	0.0	0.0

TABLE 6.2-21

BLOWDOWN MASS AND ENERGY RELEASES
DOUBLE ENDED PUMP SUCTION - MIN SI

TIME SECONDS	BREAK PATH LBM/SEC	NO. 1 FLOW THOUSAND BTU/SEC	BREAK PATH LBM/SEC	NO. 2 FLOW THOUSAND BTU/SEC
19.6	0.0	0.0	0.0	0.0
20.3	0.0	0.0	0.0	0.0
20.4	68.0	80.4	0.0	0.0
20.6	64.8	76.6	0.0	0.0
22.1	137.6	162.6	0.0	0.0
22.2	148.9	176.0	514.1	56.1
22.3	369.6	438.4	3994.2	446.3
22.4	537.9	640.0	5659.3	660.0
22.5	597.1	711.3	6181.8	745.3
22.8	641.4	764.5	6577.3	804.3
23.7	625.9	746.1	6420.0	808.9
25.7	570.8	679.8	5943.3	755.2
26.7	545.9	649.8	5718.8	730.4
27.7	523.0	622.3	5508.6	707.2
29.7	482.6	573.9	5127.6	665.5
31.7	448.1	532.5	4791.0	628.7
33.7	418.2	496.7	4490.3	596.1
35.7	392.0	465.3	4218.8	566.7
36.8	285.1	337.8	2990.3	441.1
37.0	283.5	335.9	2971.2	439.0
38.8	270.2	320.0	2807.5	421.2
39.8	285.1	337.8	3019.6	431.3
41.8	272.1	322.3	2856.9	414.0
43.8	260.7	308.8	2716.4	398.5
44.8	265.5	314.4	255.2	151.0
51.8	239.5	283.5	244.2	135.5
61.8	209.6	248.0	231.6	118.0
76.8	176.3	208.5	217.9	99.0
92.8	152.6	180.4	208.3	86.1
138.8	126.7	149.7	197.6	72.1
174.8	125.2	148.0	196.6	70.7
190.8	127.2	150.4	199.1	71.8
198.8	128.8	152.2	206.1	73.8
214.8	129.4	152.9	225.4	77.7
231.5	125.2	147.9	247.2	80.7

02-01

02-01

TABLE 6.2-22

BLOWDOWN MASS AND ENERGY RELEASES
DOUBLE ENDED PUMP SUCTION - MAX SI

02-01

TIME SECONDS	BREAK PATH LBM/SEC	NO. 1 FLOW THOUSAND BTU/SEC	BREAK PATH LBM/SEC	NO. 2 FLOW THOUSAND BTU/SEC
19.6	0.0	0.0	0.0	0.0
20.3	0.0	0.0	0.0	0.0
20.4	68.0	80.4	0.0	0.0
20.6	64.8	76.6	0.0	0.0
22.1	137.6	162.6	0.0	0.0
22.2	148.9	176.0	514.1	56.1
22.3	369.6	438.4	3994.2	446.3
22.4	537.9	640.0	5659.3	660.0
22.5	597.1	711.3	6181.8	745.3
22.8	641.4	764.5	6557.3	804.3
23.7	625.9	746.1	6420.0	808.9
25.7	570.8	679.8	5943.3	755.2
26.7	545.9	649.8	5718.8	730.4
27.7	523.0	622.3	5508.6	707.2
29.7	482.6	573.9	5127.6	665.5
31.7	448.1	532.5	4791.0	628.7
33.7	418.2	496.7	4490.3	596.1
35.7	392.0	465.3	4218.8	566.7
36.8	285.1	337.8	2990.3	441.1
37.0	283.5	335.9	2971.2	439.0
38.8	270.2	320.0	2807.5	421.2
39.8	314.9	373.2	3412.5	454.9
40.8	307.4	364.4	3308.0	447.2
41.8	301.3	357.0	3236.5	439.2
43.8	289.7	343.3	3101.5	424.2
44.8	151.5	179.1	733.7	152.2
46.8	150.7	178.2	735.2	151.8
62.8	145.0	171.3	746.0	149.0
64.8	144.3	170.5	747.3	148.7
80.8	138.9	164.2	757.6	146.1
102.8	132.0	156.0	771.5	142.9
162.8	120.2	142.1	794.3	139.6
172.8	118.4	139.9	797.8	138.9
238.8	106.9	126.5	820.2	133.7
239.5	106.8	126.2	820.5	133.6

TABLE 6.2-23

PRINCIPLE PARAMETERS DURING REFLOOD
DOUBLE-ENDED PUMP SUCTION - MIN SI

TIME SEC	TEMP DEG F	FLOODING RATE IN/SEC	CARRYOVER FRACTION	CORE HEIGHT FT	DOWN HEIGHT FT	FLOW FRACT	INJECTION LBM/SEC TOTAL	INJECTION LBM/SEC ACCU	INJECTION LBM/SEC SPILL	ENTHALPHY BTU/LBM
19.6	269.3	0.000	0.000	0.00	0.00	0.333	0.0	0.0	0.0	0.00
20.2	264.3	31.931	0.000	0.63	3.07	0.000	12601.6	12601.6	0.0	89.57
20.3	261.2	40.513	0.000	1.08	3.44	0.000	12386.3	12386.3	0.0	89.57
21.2	258.5	3.371	0.338	1.52	8.95	0.397	11417.8	11417.8	0.0	89.57
22.1	257.6	3.134	0.471	1.66	14.33	0.423	10709.1	10709.1	0.0	89.57
22.7	256.4	7.122	0.553	1.79	15.59	0.719	8259.1	8259.1	0.0	89.57
23.7	254.0	6.137	0.645	2.00	15.60	0.708	7555.4	7555.4	0.0	89.57
24.7	252.0	5.573	0.684	2.16	15.60	0.706	7190.6	7190.6	0.0	89.57
27.7	247.6	4.686	0.728	2.52	15.60	0.693	6330.7	6330.7	0.0	89.57
33.7	242.4	3.801	0.748	3.07	15.60	0.666	5132.1	5132.1	0.0	89.57
38.8	240.4	2.783	0.748	3.41	15.60	0.588	3237.7	3237.7	0.0	89.57
39.8	240.2	2.876	0.750	3.47	15.60	0.600	3469.3	3070.8	0.0	86.52
40.8	240.0	2.825	0.750	3.53	15.60	0.595	3371.5	2971.8	0.0	86.42
43.8	239.5	2.697	0.750	3.70	15.60	0.585	3131.0	2725.6	0.0	86.13
44.8	239.4	2.731	0.751	3.76	15.49	0.587	403.6	0.0	0.0	63.01
49.8	239.5	2.564	0.750	4.03	14.83	0.582	408.5	0.0	0.0	63.01
59.8	241.8	2.286	0.748	4.54	13.82	0.570	416.3	0.0	0.0	63.01
70.8	246.7	2.051	0.746	5.04	13.10	0.558	422.3	0.0	0.0	63.01
82.8	253.4	1.857	0.746	5.54	12.68	0.544	426.8	0.0	0.0	63.01
96.8	261.1	1.697	0.747	6.06	12.53	0.531	430.1	0.0	0.0	63.01
110.8	267.4	1.594	0.749	6.55	12.62	0.521	431.6	0.0	0.0	63.01
124.8	272.7	1.529	0.751	7.00	12.87	0.514	432.4	0.0	0.0	63.01
142.8	278.5	1.483	0.756	7.56	13.32	0.511	433.0	0.0	0.0	63.01
158.8	282.9	1.462	0.761	8.03	13.80	0.510	433.2	0.0	0.0	63.01
176.8	287.1	1.451	0.768	8.54	14.36	0.511	433.2	0.0	0.0	63.01
182.8	288.4	1.450	0.770	8.71	14.56	0.511	433.2	0.0	0.0	63.01
194.8	290.8	1.457	0.775	9.04	14.93	0.515	433.0	0.0	0.0	63.01
198.8	291.5	1.458	0.776	9.15	15.03	0.517	432.9	0.0	0.0	63.01
212.8	293.9	1.445	0.782	9.53	15.32	0.521	432.7	0.0	0.0	63.01
231.5		1.388	0.788	10.00	15.52	0.520	433.1	0.0	0.0	63.01

02-01

TABLE 6.2-24

PRINCIPLE PARAMETERS DURING REFLOOD
DOUBLE-ENDED PUMP SUCTION - MAX SI

TIME SEC	TEMP DEG F	FLOODING RATE IN/SEC	CARRYOVER FRACTION	CORE HEIGHT FT	DOWN HEIGHT FT	FLOW FRACT	INJECTION LBM/SEC TOTAL	INJECTION LBM/SEC ACCU	INJECTION LBM/SEC SPILL	ENTHALPHY BTU/LBM
19.6	269.3	0.000	0.000	0.00	0.00	0.333	0.0	0.0	0.0	0.00
19.6	269.3	0.000	0.000	0.00	0.00	0.333	0.0	0.0	0.0	0.00
20.2	264.3	31.931	0.000	0.63	3.07	0.000	12601.6	12601.6	0.0	89.57
20.3	261.2	40.513	0.000	1.08	3.44	0.000	12386.3	12386.3	0.0	89.57
21.2	258.5	3.371	0.338	1.52	8.95	0.397	11417.8	11417.8	0.0	89.57
22.1	257.6	3.134	0.471	1.66	14.33	0.423	10709.1	10709.1	0.0	89.57
22.7	256.4	7.122	0.553	1.79	15.59	0.719	8259.1	8259.1	0.0	89.57
23.7	254.0	6.137	0.645	2.00	15.60	0.708	7555.4	7555.4	0.0	89.57
24.7	252.0	5.573	0.684	2.16	15.60	0.706	7190.6	7190.6	0.0	89.57
27.7	247.6	4.686	0.728	2.52	15.60	0.693	6330.7	6330.7	0.0	89.57
33.7	242.4	3.801	0.748	3.07	15.60	0.666	5132.1	5132.1	0.0	89.57
38.8	240.4	2.783	0.748	3.41	15.60	0.588	3237.7	3237.7	0.0	89.57
39.8	240.2	3.069	0.752	3.47	15.60	0.624	3902.0	2979.5	0.0	83.29
40.8	239.9	3.018	0.751	3.53	15.60	0.615	3788.5	2866.7	0.0	83.11
44.8	239.3	2.003	0.739	3.77	15.60	0.465	979.5	0.0	0.0	63.01
50.8	239.9	1.970	0.740	4.02	15.60	0.465	979.6	0.0	0.0	63.01
62.8	243.2	1.910	0.742	4.53	15.60	0.464	979.7	0.0	0.0	63.01
74.8	248.4	1.852	0.744	5.01	15.60	0.464	979.8	0.0	0.0	63.01
88.8	255.8	1.786	0.747	5.55	15.60	0.464	979.9	0.0	0.0	63.01
102.8	263.1	1.721	0.751	6.06	15.60	0.464	980.0.	0.0	0.0	63.01
116.8	269.2	1.670	0.754	6.55	15.60	0.465	980.0	0.0	0.0	63.01
130.8	274.4	1.628	0.758	7.02	15.60	0.467	979.9	0.0	0.0	63.01
146.8	279.5	1.566	0.762	7.53	15.60	0.469	979.9	0.0	0.0	63.01
162.8	283.8	1.512	0.766	8.01	15.60	0.471	979.8	0.0	0.0	63.01
180.8	288.0	1.453	0.771	8.53	15.60	0.474	979.8	0.0	0.0	63.01
198.8	291.5	1.395	0.776	9.01	15.60	0.477	979.7	0.0	0.0	63.01
218.8	294.8	1.333	0.781	9.52	15.60	0.480	979.7	0.0	0.0	63.01
239.5	297.7	1.270	0.788	1.00	15.60	0.484	979.6	0.0	0.0	63.01

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TABLE 6.2-25

POST-REFLOOD MASS AND ENERGY RELEASES
DOUBLE-ENDED PUMP SUCTION - MIN SI 02-01

TIME	BREAK PATH	NO. 1 FLOW	BREAK PATH	NO. 2 FLOW
SECONDS	LBM/SEC	THOUSAND BTU/SEC	LBM/SEC	THOUSAND BTU/SEC
231.6	118.5	149.3	319.4	91.8
266.6	116.2	146.4	321.7	91.4
401.6	106.0	133.6	331.8	90.4
406.6	107.2	135.1	330.7	89.9
446.6	105.5	133.0	332.4	89.2
451.6	106.6	134.4	331.3	88.8
486.6	105.2	132.5	332.7	88.2
491.6	106.2	133.9	331.6	87.8
526.6	104.7	132.0	333.1	87.2
531.6	105.8	133.3	332.1	86.7
561.6	104.5	131.7	333.4	86.2
566.6	105.5	133.0	332.4	85.8
596.6	104.1	131.2	333.7	85.2
601.6	105.2	132.5	332.7	84.8
636.6	103.8	130.9	334.0	87.8
641.6	104.9	132.1	333.0	87.3
676.6	103.5	130.4	334.4	86.5
746.6	104.1	131.1	333.8	84.0
806.6	102.5	129.2	335.4	85.7
836.6	103.4	130.3	334.5	84.3
861.6	102.2	128.8	335.7	83.6
911.6	102.9	129.6	335.0	81.4
931.6	101.8	128.3	336.1	84.1
971.6	102.4	129.1	335.5	82.1
1106.6	100.9	127.2	337.0	81.6
1436.6	101.0	127.3	336.9	80.7
1436.7	72.6	90.0	365.3	85.8
1579.0	72.6	90.0	365.3	85.5
1579.1	70.0	80.5	367.9	32.0
2393.9	70.0	80.5	367.9	32.0
2394.0	74.0	85.0	397.1	73.3
3599.9	74.0	85.0	397.1	73.3
3600.0	57.9	66.6	413.2	76.3
3600.1	43.3	49.8	427.8	63.3

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TABLE 6.2-26

POST-REFLOOD MASS AND ENERGY RELEASES
DOUBLE-ENDED PUMP SUCTION - MAX SI 02-01

TIME SECONDS	BREAK PATH LBM/SEC	NO. 1 FLOW THOUSAND BTU/SEC	BREAK PATH LBM/SEC	NO. 2 FLOW THOUSAND BTU/SEC
239.6	118.3	149.5	864.8	128.7
259.6	117.0	147.9	866.1	128.5
264.6	118.1	149.3	864.9	128.0
289.6	116.5	147.2	866.6	127.7
294.6	117.6	148.6	865.5	127.3
314.6	116.3	146.9	866.8	127.0
344.6	117.1	148.0	866.0	121.9
369.6	115.4	145.8	867.7	121.7
374.6	116.4	147.2	866.6	121.3
429.6	114.7	145.0	868.4	124.1
434.6	115.9	146.4	867.2	123.6
464.6	114.4	144.6	868.6	123.1
499.6	115.4	145.9	867.6	121.8
529.6	113.9	144.0	869.1	121.2
534.6	115.0	145.3	868.1	120.8
589.6	113.4	143.3	869.7	119.5
624.6	114.2	144.3	868.9	118.1
649.6	113.0	142.8	870.1	117.6
704.6	113.8	143.8	869.3	119.1
749.6	112.4	142.0	870.7	117.9
774.6	113.1	143.0	869.9	116.7
809.6	112.0	141.6	871.1	115.7
829.6	112.9	142.6	870.2	114.7
889.6	111.5	141.0	871.5	115.9
914.6	112.3	141.9	870.7	114.6
984.6	111.1	140.4	872.0	114.8
1349.6	111.2	140.6	871.8	113.5
1349.7	73.2	91.1	1012.7	220.1
1569.3	72.1	89.8	1013.8	220.4
1569.4	69.6	80.0	1016.4	165.1
3600.0	57.4	66.0	1028.5	167.3
3600.1	43.3	49.8	1042.6	154.3

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TABLE 6.2-27

MASS BALANCE
DOUBLE-ENDED PUMP SUCTION - MIN SI

	TIME(SECONDS)	0.0	19.60	19.60	231.53	1441.60	1578.99	3600.00
		MASS (THOUSAND LBM)						
INITIAL MASS	IN RCS AND ACC	609.42	609.42	609.42	609.42	609.42	609.42	609.42
ADDED MASS								
	PUMPED INJECTION	0.00	0.00	0.00	82.62	612.11	676.83	1628.90
	TOTAL ADDED	0.00	0.00	0.00	82.62	612.11	676.83	1628.90
TOTAL AVAILABLE		609.42	609.42	609.42	691.68	1221.54	1286.26	2238.32
DISTRIBUTION								
	REACTOR COOLANT	421.29	47.69	54.59	105.15	105.15	105.15	105.15
	ACCUMULATOR	188.13	146.19	139.29	0.00	0.00	0.00	0.00
	TOTAL CONTENTS	609.42	193.88	193.88	105.15	105.15	105.15	105.15
EFFLUENT								
	BREAK FLOW	0.00	415.53	415.53	586.53	1116.38	1181.10	2133.17
	ECCS SPILL	0.00	0.00	0.00	0.00	0.00	0.00	0.00
	TOTAL EFFLUENT	0.00	415.53	415.53	586.53	1116.38	1181.10	2133.17
TOTAL ACCOUNTABLE		609.42	609.42	609.42	691.68	1221.53	1286.25	2238.32

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TABLE 6.2-28

MASS BALANCE
DOUBLE-ENDED PUMP SUCTION - MAX SI

	TIME(SECONDS)	0.0	19.60	19.60	239.50	1354.60	1569.33	3600.00
		MASS (THOUSAND LBM)						
INITIAL MASS	IN RCS AND ACC	609.42	609.42	609.42	609.42	609.42	609.42	609.42
ADDED MASS								
	PUMPED INJECTION	0.00	0.00	0.00	195.63	1292.77	1525.96	3731.14
	TOTAL ADDED	0.00	0.00	0.00	195.63	1292.77	1525.96	3731.14
TOTAL AVAILABLE		609.42	609.42	609.42	805.05	1902.20	2135.38	4340.56
DISTRIBUTION								
	REACTOR COOLANT	421.29	47.69	54.59	105.82	105.82	105.82	105.82
	ACCUMULATOR	188.13	146.19	139.29	0.00	0.00	0.00	0.00
	TOTAL CONTENTS	609.42	193.88	193.88	105.82	105. 82	105.82	105.82
EFFLUENT								
	BREAK FLOW	0.00	415.53	415.53	699.23	1796.37	2029.56	4234.74
	ECCS SPILL	0.00	0.00	0.00	0.00	0.00	0.00	0.00
	TOTAL EFFLUENT	0.00	415.53	415.53	586.53	1796.37	2029.56	4234.74
TOTAL ACCOUNTABLE		609.42	609.42	609.42	805.05	1902.19	2135.38	4340.56

TABLE 6.2-29

MASS BALANCE
DOUBLE-ENDED HOT LEG GUILLOTINE

	TIME(SECONDS)	0.0	18.20	18.20
	MASS (THOUSAND LBM)			
INITIAL MASS	IN RCS AND ACC	609.42	609.42	609.42
ADDED MASS				
	PUMPED INJECTION	0.00	0.00	0.00
	TOTAL ADDED	0.00	0.00	0.00
TOTAL AVAILABLE		609.42	609.42	609.42
DISTRIBUTION				
	REACTOR COOLANT	421.29	95.75	102.64
	ACCUMULATOR	188.13	116.21	109.31
	TOTAL CONTENTS	609.42	211.96	211.96
EFFLUENT				
	BREAK FLOW	0.00	397.46	397.46
	ECCS SPILL	0.00	0.00	0.00
	TOTAL EFFLUENT	0.00	397.46	397.46
TOTAL ACCOUNTABLE		609.42	609.42	609.42

TABLE 6.2-30

ENERGY BALANCE
DOUBLE-ENDED PUMP SUCTION - MIN SI

	TIME(SECONDS)	0.0	19.60	19.60	231.53	1441.60	1578.99	3600.00
	ENERGY (MILLION LBM)							
INITIAL ENERGY	IN RCS AND ACC, S GEN	720.39	720.39	720.39	720.39	720.39	720.39	720.39
ADDED ENERGY								
	PUMPED INJECTION	0.00	0.00	0.00	5.18	38.57	48.15	189.05
	DECAY HEAT	0.00	5.54	5.54	26.56	107.39	115.06	212.59
	HEAT FROM SECONDARY	0.00	-2.33	-2.33	-2.33	1.67	1.67	1.67
	TOTAL ADDED	0.00	3.21	3.21	29.41	147.62	164.88	403.31
TOTAL AVAILABLE		720.39	723.60	723.60	749.80	868.01	885.26	1123.69
DISTRIBUTION								
	REACTOR COOLANT	251.33	13.54	14.61	29.77	29.77	29.77	29.77
	ACCUMULATOR	16.85	13.09	12.48	0.00	0.00	0.00	0.00
	CORE STORED	21.91	10.73	10.73	4.10	3.95	3.86	2.71
	PRIMARY METAL	126.07	119.58	119.58	98.57	60.50	57.80	41.62
	SECONDARY METAL	80.76	79.64	79.64	73.31	45.45	42.89	31.31
	STEAM GENERATOR	223.47	224.89	224.89	203.60	123.41	116.70	85.56
	TOTAL CONTENTS	720.39	461.48	461.48	409.35	263.07	251.03	190.97
EFFLUENT								
	BREAK FLOW	0.00	262.11	262.11	334.25	598.74	628.04	926.53
	ECCS SPILL	0.00	0.00	0.00	0.00	0.00	0.00	0.00
	TOTAL EFFLUENT	0.00	262.11	262.11	334.25	598.74	628.04	926.53
TOTAL ACCOUNTABLE		720.39	723.59	723.59	743.60	861.81	879.07	1117.50

TABLE 6.2-31

ENERGY BALANCE
DOUBLE-ENDED PUMP SUCTION - MAX SI

	TIME(SECONDS)	0.00	19.60	19.60	239.50	1354.60	1569.33	3600.00
	ENERGY (MILLION BTU)							
INITIAL ENERGY	IN RCS, ACC, S GEN	720.39	720.39	720.39	720.39	720.39	720.39	720.39
ADDED ENERGY								
	PUMPED INJECTION	0.00	0.00	0.00	12.33	82.38	116.89	443.26
	DECAY HEAT	0.00	5.54	5.54	27.24	102.41	114.53	212.59
	HEAT FROM SECONDARY	0.00	-2.33	-2.33	-2.33	2.99	3.37	3.37
	TOTAL ADDED	0.00	3.21	3.21	37.23	187.78	234.79	659.22
TOTAL AVAILABLE		720.39	723.60	723.60	757.62	908.16	955.18	1379.61
DISTRIBUTION								
	REACTOR COOLANT	251.33	13.54	14.61	29.98	29.98	29.98	29.98
	ACCUMULATOR	16.85	13.09	12.48	0.00	0.00	0.00	0.00
	CORE STORED	21.91	10.73	10.73	4.10	3.95	3.82	2.71
	PRIMARY METAL	126.07	119.58	119.58	97.37	61.29	57.13	41.28
	SECONDARY METAL	80.76	79.64	79.64	73.43	46.34	42.36	30.92
	STEAM GENERATOR	223.47	224.89	224.89	203.88	127.25	116.94	86.16
	TOTAL CONTENTS	720.39	461.48	461.48	408.75	268.82	250.25	191.05
EFFLUENT								
	BREAK FLOW	0.00	262.11	262.11	342.67	633.15	698.74	1182.37
	ECCS SPILL	0.00	0.00	0.00	0.00	0.00	0.00	0.00
	TOTAL EFFLUENT	0.00	262.11	262.11	342.67	633.15	698.74	1182.37
TOTAL ACCOUNTABLE		720.39	723.59	723.59	751.42	901.97	948.98	1373.41

TABLE 6.2-32

ENERGY BALANCE
DOUBLE-ENDED HOT LEG GUILLOTINE

	TIME(SECONDS)	0.00	18.20	18.20
	ENERGY (MILLION BTU)			
INITIAL ENERGY	IN RCS, ACC, S GEN	720.39	720.39	720.39
ADDED ENERGY				
	PUMPED INJECTION	0.00	0.00	0.00
	DECAY HEAT	0.00	5.87	5.87
	HEAT FROM SECONDARY	0.00	-3.48	-3.48
	TOTAL ADDED	0.00	2.39	2.39
** TOTAL AVAILABLE **		720.39	722.77	722.77
DISTRIBUTION				
	REACTOR COOLANT	251.33	23.93	24.55
	ACCUMULATOR	16.85	10.41	9.79
	CORE STORED	21.91	8.56	8.56
	PRIMARY METAL	126.07	118.22	118.22
	SECONDARY METAL	80.76	78.42	78.42
	STEAM GENERATOR	233.47	220.64	220.64
	TOTAL CONTENTS	720.39	460.18	460.18
EFFLUENT				
	BREAK FLOW	0.00	262.58	262.58
	ECCS SPILL	0.00	0.00	0.00
	TOTAL EFFLUENT	0.00	262.58	262.58
** TOTAL ACCOUNTABLE **		720.39	722.76	722.76

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TABLE 6.2-33

SYSTEM INITIAL CONDITIONS

Power Level (Includes +2% Allowance for Instrument Error and Deadband	2958 MWt
Vessel Average Temperature (Includes +5.3°F Allowance for Instrument Error and Deadband)	592.7°F
Core Inlet Temperature (Includes +5.3°F Allowance for Instrument Error and Deadband)	558.2°F
Mass of Reactor Coolant	421.3×10^3 lbm
Reactor Coolant Pressures (Includes +50 psi Allowance for Instrument Error and Deadband)	2300 psia
Initial Steam Generator Steam Pressure	966 psia
Assumed Maximum Containment Back Pressure	71.7 psia

TABLE 6.2-34

WESTINGHOUSE MODEL DECAY HEAT CURVE

(1979 ANS PLUS 2-SIGMA UNCERTAINTY)

TIME (SEC)	DECAY HEAT GENERATION RATE (BTU/BTU)
1.00E+01	0.053876
1.50E+01	0.050401
2.00E+01	0.048018
4.00E+01	0.042401
6.00E+01	0.039244
8.00E+01	0.037065
1.00E+02	0.035466
1.50E+02	0.032724
2.00E+02	0.030936
4.00E+02	0.027078
6.00E+02	0.024931
8.00E+02	0.023389
1.00E+03	0.022156
1.50E+03	0.019921
2.00E+03	0.018315
4.00E+03	0.014781
6.00E+03	0.013040
8.00E+03	0.012000
1.00E+04	0.011262
1.50E+04	0.010097
4.00E+04	0.007778
1.00E+05	0.006021
4.00E+05	0.003770
6.00E+05	0.003201
8.00E+05	0.002834
1.00E+06	0.002580
1.50E+06	0.002168
2.00E+06	0.001909
4.00E+06	0.001355
6.00E+06	0.001091
8.00E+06	0.000927
1.00E+07	0.000808
2.00E+07	0.000477
4.00E+07	0.000269
6.00E+07	0.000188
8.00E+07	0.000140
1.00E+08	0.000111

TABLE 6.2-41

BLOWDOWN TABLE (WITHOUT MARGIN) FOR REACTOR CAVITY PRESSURE RESPONSE ANALYSIS, 150 SQUARE INCH BREAK AREA, COLD LEG BREAK

Time (sec)	Mass Flow (10^3 lbm/sec)	Enthalpy (Btu/lbm)
0.0	0.00	557.61
0.001	8.83	557.61
0.002	12.46	557.40
0.003	12.86	557.52
0.010	20.53	556.35
0.017	24.70	555.57
0.022	23.82	554.11
0.036	26.72	553.97
0.042	26.23	553.43
0.049	26.99	553.61

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TABLE 6.2-42

BLOWDOWN TABLE FOR STEAM GENERATOR
COMPARTMENT DIFFERENTIAL PRESSURE ANALYSIS
COLD LEG DOUBLE ENDED GUILLOTINE BREAK

Time (sec)	Mass Flow (10^3 lbm/sec)	Enthalpy (Btu/lbm)
0.	10.022	553.81
0.001	40.519	549.06
0.004	60.438	548.88
0.010	51.753	547.88
0.015	51.461	548.64
0.025	54.794	548.97
0.030	56.260	549.08
0.035	57.373	549.14
0.040	58.238	549.26
0.045	58.870	549.42
0.050	59.569	549.76
0.055	69.678	554.68
0.058	79.314	550.76
0.060	81.633	551.21
0.062	80.386	550.82
0.065	81.351	550.81
0.070	83.484	550.83
0.075	84.880	550.82
0.080	87.451	551.00
0.085	88.606	550.95
0.090	89.752	550.96
0.095	89.695	550.83
0.100	90.318	550.85
0.125	92.418	550.87
0.150	94.527	551.23
0.175	95.191	551.17
0.200	95.876	551.15
0.250	96.582	551.24
0.300	96.209	551.18
0.350	94.955	551.09
0.400	93.945	551.14
0.450	92.873	551.14
0.500	91.364	551.21
0.600	89.085	551.63
0.750	87.717	552.71
0.900	85.512	554.42
1.000	82.637	555.98
1.100	80.511	557.86

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TABLE 6.2-43

BLOWDOWN TABLE FOR STEAM GENERATOR COMPARTMENT DIFFERENTIAL PRESSURE ANALYSIS HOT LEG DOUBLE ENDED GUILLOTINE BREAK

Time (sec)	Mass Flow (10^3 lbm/sec)	Enthalpy (Btu/lbm)
0.	10.022	647.51
0.001	54.859	645.59
0.003	80.761	645.01
0.008	63.883	647.45
0.010	65.471	648.08
0.015	66.879	649.87
0.020	68.935	650.91
0.030	70.920	652.36
0.039	71.408	653.24
0.050	71.948	655.22
0.060	75.985	656.17
0.070	80.768	654.92
0.080	83.236	653.78
0.085	83.747	653.39
0.089	83.859	653.15
0.100	83.421	652.72
0.125	80.990	651.45
0.150	76.694	650.73
0.175	74.661	649.64
0.200	73.906	647.85
0.250	71.393	644.44
0.300	69.196	641.27
0.350	67.446	638.54
0.400	66.152	636.18
0.450	65.167	634.16
0.500	64.153	632.66
0.600	61.847	630.68
0.700	59.740	630.19
0.800	59.165	631.82
0.900	57.600	634.08
1.000	55.551	636.03
1.100	53.998	638.07

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TABLE 6.2-44

DOUBLE ENDED PRESSURIZER SURGE LINE GUILLOTINE BREAK
SHORT TERM MASS AND ENERGY RELEASE RATES INTO PRESSURIZER CAVITY

Time (Seconds)	Mass Flow (10^3 lbm/sec)	Enthalpy (BTU/lbm)
0.00	21.76	668.6
0.02	22.52	662.9
0.04	23.40	670.3
0.05	23.71	671.2
0.06	23.91	659.0
0.08	24.16	633.5
0.10	24.27	599.5
0.12	23.78	579.0
0.14	23.02	570.4
0.16	20.87	611.0
0.18	18.40	677.5
0.20	18.11	674.3
0.22	18.08	674.3
0.24	18.06	674.5
0.26	18.04	674.3
0.28	18.01	675.0
0.30	17.99	675.2
0.35	17.93	675.4
0.40	17.87	676.3
0.45	17.83	675.5
0.50	17.77	675.7
0.55	17.72	676.2
0.60	17.70	675.1
0.65	17.69	674.2
0.70	17.66	673.2
0.75	17.62	673.6
0.80	17.58	673.6
0.85	17.56	672.6
0.90	17.52	673.0
1.00	17.37	676.2

TABLE 6.2-45

BLOWDOWN TABLE FOR PRESSURIZER SPRAY LINE BREAK

Time (Seconds)	Mass Flow (10^3 lbm/sec)	Enthalpy (BTU/lbm)
0.00000	5.350	611.05
0.05003	5.350	611.05
0.10011	5.478	609.82
0.15001	5.214	612.01
0.20062	5.199	612.03
0.25024	5.207	611.80
0.30016	5.108	612.62
0.35000	5.134	612.22
0.40023	5.115	612.26
0.45080	5.137	611.88
0.50011	5.158	611.52
0.55038	5.138	611.56
0.60015	5.145	611.34
0.65025	5.138	611.25
0.70026	5.134	611.14
0.75041	5.132	611.02
0.80005	5.115	611.05
0.85031	5110	610.95
0.90033	5.099	610.93
0.95038	5.084	610.93
1.00004	5.079	610.86
1.05002	5.059	610.93
1.10003	5.048	610.90
1.15004	5.037	610.89
1.20016	5.022	610.95
1.25006	5.012	610.91
1.30023	4.997	610.94
1.35011	4.983	610.97
1.40013	4.974	610.95
1.45020	4.959	611.00
1.50020	4.947	611.02
1.55112	4.936	611.02
1.60021	4.921	611.07
1.65033	4.909	611.09
1.70032	4.898	611.11
1.75012	4.884	611.17
1.80021	4.872	611.20
1.85040	4.860	611.24
1.90010	4.844	611.30
1.95032	4.832	611.34
2.00001	4.820	611.40
4.0	4.820	611.40

02-01

TABLE 6.2-46a

Deleted per RN 01-073 02-01

TABLE 6.2-46b

STEAM LINE BREAK MASS & ENERGY RELEASE
INSIDE CONTAINMENT SYSTEM NOMINAL PARAMETERS [(1)]

NSSS Power, Mwt	2912
Core Power Mwt	2900
Reactor Coolant Pump Heat, Mwt	12
Reactor Coolant Flow, Total, gpm	277800
Pressurizer Pressure, psia	2250
Reactor Coolant Temperature, °F	
Core Outlet	627.7
Vessel Outlet	621.9
Core Average	592.8
Vessel Average	587.4
Vessel/Core Inlet	552.9
Steam Generator	
Steam Temperature, °F	540.4
Steam Pressure, psia	966

1. Parameters for the partial power cases are adjusted accordingly

TABLE 6.2-46c

STEAM LINE BREAK MASS & ENERGY RELEASE INSIDE CONTAINMENT
ANALYSIS INITIAL CONDITION ASSUMPTIONS

02-01

PARAMETER Power Level (%)	102	75	50	25	0
RCS Average Temperature (°F)	591.4	583.2	575.6	568.0	557.0
RCS Flowrate (gpm)	277800	277800	277800	277800	277800
RCS Pressure (psia)	2250	2250	2250	2250	2250
Pressurizer Water Volume (ft^3)	861.5	739.5	626.5	513.5	409.6
Feedwater Enthalpy (btu/lbm)	419.4	385.6	349.3	297.2	220.8
Faulted SG Water Level (% span)	65	65	65	65	65
Intact SG Water Level (% span)	55	55	55	55	55

TABLE 6.2-46d

STEAM LINE BREAK MASS & ENERGY RELEASE
INSIDE CONTAINMENT SAFETY SYSTEM ASSUMPTIONS

PARAMETER	VALUE
Trip Setpoints	
Low Pressurizer Pressure Reactor Trip (psia)	1775
Low Pressurizer Pressure Safety Injection (psia)	1715
Low Steam Line Pressure (psia)	430
Overpower ΔT	1.143
Time Constants	
Steam Line Pressure Lead Time Constant (sec)	50.0
Steam Line Pressure Lag Time Constant (sec)	5.0
Time Delays (sec)	
Low Pressurizer Pressure Reactor Trip Delay	2.0
Low Steam Line Pressure Reactor Trip Delay	2.0
Overpower ΔT Reactor Trip Delay	8.5
High-1 Containment Pressure Reactor Trip Delay	3.0
Steam Line Isolation Valve Closure	10.0
Feedwater Isolation Delay	10.0
SI System Reactor Trip Delay	3.0
SI Start Up Time (with offsite power)	27.0
Auxiliary Feedwater Delay	0.0

TABLE 6.2-47a

STEAM LINE BREAK MASS & ENERGY RELEASE
INSIDE CONTAINMENT CASE ANALYZED

Break Type/Case	Power (%)	Break Size (ft^2)	Failure(s)	Entrainment	Reactor Trip (sec)	FW Isolation (sec)	SL Isolation (sec)
DER/1G	102	1.4	FWIV, SIS	No	LSP (3.6)	LSP (10.6)	LSP (10.6)
DER/1B	102	1.2	FWIV, SIS, EFW	No	LSP (4.6)	LSP (11.6)	LSP (11.6)
DER/1C	102	1.1	FWIV, SIS, EFW	No	LSP (5.0)	LSP (12.0)	LSP (12.0)
DER/1H	102	1.4	SIS	No	LSP (3.6)	LSP (10.6)	LSP (10.6)
DER/1I	102	1.4	SIS, EFW	No	LSP (3.6)	LSP (10.6)	LSP (10.6)
DER/1J	102	1.4	DIESEL	No	LSP (3.6)	LSP (10.6)	LSP (10.6)
DER/1K	102	1.4	CH-A	No	LSP (3.6)	LOP (8.5)	LOP (7.0)
SPLIT/1	102	0.878	FWIV, SIS, EFW	No	Hi-1 (15.0)	Hi-1 (22.0)	Hi-2 (22.0)
DER/2E	75	1.4	CH-A	Yes	LSP (3.7)	LOP (8.5)	LOP (7.0)
DER/2B	75	1.1	FWIV, SIS, EFW	Yes	LSP (4.4)	LSP (11.4)	LSP (11.4)
DER/2C	75	1.0	FWIV, SIS, EFW	No	LSP (4.6)	LSP (11.6)	LSP (11.6)
SPLIT/2	75	0.871	FWIV, SIS, EFW	No	Hi-1 (14.7)	Hi-1 (21.7)	Hi-2 (21.7)
DER/3E	50	1.4	CH-A	Yes	LSP (3.8)	LOP (8.5)	LOP (7.0)
DER/3B	50	0.8	FWIV, SIS, EFW	Yes	LSP (4.6)	LSP (11.6)	LSP (11.6)
DER/3C	50	0.7	FWIV, SIS, EFW	No	LSP (4.9)	LSP (11.9)	LSP (11.9)
SPLIT/3	50	0.863	FWIV, SIS, EFW	No	Hi-1 (14.5)	Hi-1 (21.5)	Hi-2 (21.5)
DER/4E	25	1.4	CH-A	Yes	LSP (3.8)	LOP (8.5)	LOP (7.0)
DER/4G	25	1.4	FWIV, SIS	Yes	LSP (3.8)	LSP (10.9)	LSP (10.9)
DER/4B	25	0.6	FWIV, SIS, EFW	Yes	LSP (4.5)	LSP (11.5)	LSP (11.5)
DER/4C	25	0.5	FWIV, SIS, EFW	No	LSP (4.8	LSP (11.8)	LSP (11.5)
SPLIT/4	25	0.849	FWIV, SIS, EFW	No	Hi-1 (14.3)	Hi-1 (21.3)	Hi-2 (21.3)
DER/5F	0	1.4	CH-A	Yes	LSP (3.8)	LOP (8.5)	LOP (7.0)
DER/5B	0	0.2	FWIV, SIS, EFW	Yes	LSP (4.4)	LSP (11.4)	LSP (11.4)
DER/5C	0	0.1	FWIV, SIS, EFW	No	LSP (4.6)	LSP (11.6)	LSP (11.6)
SPLIT/5	0	0.772	FWIV, SIS, EFW	No	Hi-1 (15.4)	Hi-1-(22-4)	Hi-2 (22.4)

CH-A - Failure of Electrical Channel A
DER - Double-ended rupture
FWIV - Feedwater Isolation Valve Failure
EFW - Emergency Feedwater Runout Control Failure
SIS - Failure of 1 Train of the Safety Injection System
LSP - Low Steam Line Pressure Setpoint
Hi-1 - High-1 Containment Pressure Setpoint
Hi-2 - High-2 Containment Pressure Setpoint
LOP - Loss of Power

TABLE 6.2-47b

CONTAINMENT PRESSURE ENGINEERED SAFETY FEATURES ACTUATION TIMES FOR STEAM LINE BREAKS INSIDE CONTAINMENT

		Hi-1		Hi-2	Hi-3	
Reactor building pressure setpoint value used in analysis.		4.2 psig		6.95 psig	12.65 psig	02-01
Time after break to reach setpoint		t_1		t_2	t_3	
Signal processing delay and response time		1.5 seconds		1.5 seconds	1.5 seconds	
	Feedwater Isolation	Diesel	Fan Coolers			
Feedwater isolation valve closure time	8.5 seconds					
Main steam valve closure				8.5 seconds		
Diesel start time		10 sec.	10 sec.			
R.B. fan coolers start time, water side switchover, diesel loading sequence time			66.5 sec.			
R.B. spray pump start and header fill time					37 seconds	98-01
	t_1+10.0	t_1+11.5	t_1+76.5	t_2+10.0	t_3+38.5	

Hi-1 - Diesel Start, Feedwater Isolation Signal, Fan Cooler Actuation Signal

Hi-2 - Main Steam Isolation Signal

Hi-3 - Reactor Building Spray Actuation Signal

TABLE 6.2-47b (Continued)

CONTAINMENT PRESSURE
ENGINEERED SAFETY FEATURES ACTUATION TIMES
FOR STEAM LINE BREAKS

(Values Used For Analysis)

02-01

Break Type/Case	Power (%)	Break Size (ft^2)	t_1 (sec)	t_2 (sec)	t_3 (sec)	Main Feedwater Isolation (sec)	Main Steam Isolation (sec)	Fan Cooler Actuation (sec)	Spray Actuation (sec)
DER/1G	102	1.4	0.5	1	2.5	(1, 2)	(1, 2)	86.5	53.1
DER/1B	102	1.2	3	4.5	8.5	(1, 2)	(1, 2)	86.5	53.1
DER/1C	102	1.1	3	5	9	(1, 2)	(1, 2)	86.5	53.1
DER/1H	102	1.4	1	1.5	3	(1, 2)	(1, 2)	86.5	53.1
DER/1I	102	1.4	0.5	1	2.5	(1, 2)	(1, 2)	86.5	53.1
DER/1J	102	1.4	0.5	1	2.5	(1, 2)	(1, 2)	86.5	53.1
DER/1K	102	1.4	0.5	1	2.5	(1, 2)	(1, 2)	86.5	53.1
SPLIT/1	102	0.878	6.8	12	24	22	22	86.5	67.1
DER/2E	75	1.4	0.5	1	2.5	(1, 2)	(1, 2)	86.5	53.1
DER/2B	75	1.1	2	4	8	(1, 2)	(1, 2)	86.5	53.1
DER/2C	75	1.0	3	5.5	10	(1, 2)	(1, 2)	86.5	53.1
SPLIT/2	75	0.871	6.6	11.6	24	21.7	21.7	86.5	67.1
DER/3E	50	1.4	0.5	1	2.5	(1, 2)	(1, 2)	86.5	53.1
DER/3B	50	0.8	2.5	5	14	(1, 2)	(1, 2)	86.5	57.1
DER/3C	50	0.7	4.5	7.5	15	(1, 2)	(1, 2)	86.5	58.1
SPLIT/3	50	0.863	6.6	11.4	23	21.5	21.5	86.5	66.1
DER/4E	25	1.4	0.5	1	2.5	(1, 2)	(1, 2)	86.5	53.1
DER/4G	25	1.4	0.5	1	2.5	(1, 2)	(1, 2)	86.5	53.1
DER/4B	25	0.6	3.5	6.5		(1, 2)	(1, 2)	86.5	65.1
DER/4C	25	0.5	6	11	24	(1, 2)	(1, 2)	86.5	67.1
SPLIT/4	25	0.849	6.4	11.2	23	21.3	21.3	86.5	66.1
DER/5F	0	1.4	0.5	1	2.5	(1, 2)	(1, 2)	86.5	53.1
DER/5B	0	0.2	10	28	120	(1, 2)	(1, 2)	86.5	163.1
DER/5C	0	0.1	39	87	441	(1, 2)	(1, 2)	115.5	484.1
SPLIT/5	0	0.772	7	12.4	26	22.4	22.4	86.5	69.1

Notes:

(1) Main steam and feedwater isolation is initiated by other variable than containment pressure

(2) Analysis values are defined in Table 6.2-47a

TABLE 6.2-48a

SLB M&E INSIDE CONTAINMENT
CASE 4E: 25% POWER 1.4 FT^2 DER
FAILURE: ELECTRICAL CH-A

Time (sec)	Break Flow Rate (lbm/sec)	Break Energy (10^6 btu/sec)	Integrated Mass Release (10^3 lbm)	Integrated Energy Release (10^6 btus)
0.0	0.0	0.0	0.0	0.0
0.2	9,441.0	11.23	1.888	2.245
0.4	9,207.0	10.96	3.73	4.437
0.6	9,052.0	10.78	5.54	6.592
0.8	8,903.0	10.61	7.321	8.713
1	8,758.0	10.44	9.072	10.80
2	8,125.0	9.704	17.44	20.79
3	16,729.0	12.14	29.66	31.58
4	18,151.0	12.35	47.31	43.87
5	17,156.0	11.83	64.86	55.91
6	16,214.0	11.33	81.45	67.43
7	14,980.0	10.74	96.93	78.40
7.2	4,711.0	3.373	97.87	79.08
8	4,397.0	3.217	101.5	81.70
9	4,009.0	3.048	105.6	84.81
10	3,691.0	2.917	109.5	87.77
15	2,562.0	2.366	124.7	100.9
20	1,540.0	1.831	134.9	111.3
30	1,205.0	1.451	148.3	127.4
40	1,064.0	1.281	159.5	140.9
61.5	1,002.0	1.206	181.4	167.3
80	987.6	1.189	199.8	189.4
100	980.0	1.180	219.4	213.1
120	982.1	1.182	239.0	236.7
140	979.7	1.179	258.7	260.3
160	976.0	1.175	278.2	283.8
180	299.1	0.3558	291.0	299.2
200	297.1	0.3525	297.2	306.5
220	458.4	0.5478	303.6	314.2
240	236.6	0.2812	309.1	320.7
260	186.6	0.2205	313.8	326.3
280	143.9	0.1690	317.3	330.3
300	141.0	0.1653	320.1	333.7
340	137.5	0.1612	325.6	340.1
1800	137.5	0.1612	526.4	575.5
1802	138.0	0.1619	526.7	575.9
1828	0.0	0.0	527.5	576.8

02-01

02-01

Steamline isolation: 7.0 sec Feedline isolation: 8.5 sec

TABLE 6.2-48b

SLB M&E INSIDE CONTAINMENT
CASE 1H: 102% POWER 1.4 FT^2 DER
FAILURE: 1 Train SI

Time (sec)	Break Flow Rate (lbm/sec)	Break Energy (10^6 btu/sec)	Integrated Mass Release (10^3 lbm)	Integrated Energy Release (10^6 btus)
0.0	0.0	0.0	0.0	0.0
0.2	8,622.0	10.28	1.724	2.057
0.4	8,400.0	10.02	3.404	4.062
0.6	8,255.0	9.856	5.055	6.033
0.8	8,117.0	9.695	6.679	7.972
1	7,985.0	9.541	8.276	9.880
2	7,399.0	8.856	15.90	19.00
3	6,908.0	8.28	23.00	27.50
4	6,635.0	7.958	29.70	35.54
5	6,521.0	7.824	36.26	43.42
6	6,416.0	7.699	42.72	51.16
7	6,320.0	7.587	49.08	58.79
8	6,228.0	7.478	55.34	66.32
9	6,132.0	7.365	61.52	73.73
10	6,022.0	7.234	67.58	81.02
10.6	5,944.0	7.142	71.17	85.32
10.8	2,074.0	2.491	71.58	85.32
11	2,067.0	2.482	71.99	86.31
15	1,859.0	2.236	79.85	95.75
20	1,594.0	1.919	88.43	106.1
30	1,264.0	1.522	102.5	123.0
40	1,101.0	1.326	114.2	137.1
61.5	1,002.0	1.206	136.4	163.9
80	978.3	1.178	154.7	185.8
100	964.8	1.161	174.1	209.2
110	959.7	1.155	183.7	220.8
120	981.2	1.181	193.5	232.5
140	981.5	1.181	213.1	256.2
160	962.5	1.158	232.7	279.7
180	250.6	0.2964	244.8	294.3
200	29.23	0.337	246.7	296.5
206	0.0	0.0	246.8	296.6
252	48.42	0.0562	247.8	297.7
506	0.0	0.0	254.9	306.0
752	51.17	0.0594	261.5	313.6
1000	0.0	0.0	268.4	321.6
1250	49.56	0.0575	275.2	329.5
1502	0.0	0.0	282.4	337.7
1754	51.92	0.0603	289.1	345.5
1800	0.0	0.0	290.3	346.9
1808	13.43	0.0155	290.7	347.3
1810	0.0	0.0	290.7	347.3

02-01

Steamline isolation: 10.6 sec   Feedline isolation: 10.6 sec

TABLE 6.2-49

REACTOR BUILDING SPRAY SYSTEM COMPONENT DATA

Reactor Building Spray Pumps		
Number	2	
Rated Capacity, gpm	2500	
Rated Head, ft	450	
Material	Stainless steel (wetted parts) or equivalent corrosion resistant material	98-01
Design Pressure, psig	300	
Design Temperature, °F	250	
Code	ASME III, Class 2	99-01
Safety Class	2a	
Sodium Hydroxide Storage Tank		
Number	1	
Nominal Capacity, gal (minimum required usable volume)	3300 (3050)	99-01
Material	Carbon steel	
Design Pressure, ft	Static Head + 3 psig	
Code	ASME III, Class 3	
Safety Class	2b	
Valves Exposed to Recirculated Coolant and Valves in the Refueling Water Storage Tank Discharge Line		02-01
Material	Stainless steel	
Code	ASME III, Class 2	
Safety Class	2a	

TABLE 6.2-49 (Continued)

REACTOR BUILDING SPRAY SYSTEM COMPONENT DATA

Valves in the Sodium Hydroxide Storage Tank Discharge Line	
Material	Stainless steel
Code	ASME III, Class 3
Safety Class	2b
Piping in Sodium Hydroxide Storage Tank Discharge Line	
Material	Stainless steel
Code	ASME III, Class 3
Safety Class	2b
Other Piping, exposed to water recirculated from the Reactor Building Sump and water from the RWST	
Material	Stainless steel
Code	ASME III, Class 2
Safety Class	2a
Spray Nozzles	
Number	330
Material	Stainless steel
Type	Hollow cone
Flow, gpm/nozzle	15.2
Pressure Drop, psi	40

TABLE 6.2-50

SINGLE FAILURE ANALYSIS - REACTOR BUILDING SPRAY SYSTEM

	Component	Malfunction	Comments
1.	Reactor building spray pump	Fails to start.	Since each of the two subsystems of the reactor building spray system is equally sized, the remaining subsystem provides heat removal capability at a reduced rate. In combination with the reactor building emergency cooling system, heat removal capability in excess of the requirements is provided. Elemental iodine removal is adequate with one subsystem operating.
2.	Reactor building isolation valve.	Fails to open.	(Same as above.)
3.	Check valve in suction line.	Fails to open.	(Same as above.)
4.	Sodium hydroxide tank discharge valve.	Fails to open.	The redundant discharge valve admits adequate sodium hydroxide to the other subsystem to provide the required iodine removal.

TABLE 6.2-51

REACTOR BUILDING SPRAY SYSTEM
HEADER LOCATIONS AND SIZES

Subsystem A

Header	Ring Diameter (ft)	Pipe Size (in)	Elevation (ft-in)
Outer header	98	6	578-10
Middle header	58	6	590-4
Inner header	16	3	595-7

Subsystem B

Header	Ring Diameter (ft)	Pipe Size (in)	Elevation (ft-in)
Outer header	102	6	577-0
Middle header	62	6	589-6
Inner header	20	3	595-0

TABLE 6.2-52

SINGLE FAILURE ANALYSIS - REACTOR BUILDING HEAT REMOVAL AND FILTERING SYSTEM 00-01

	Component	Malfunction	Comments	
1.	Air Handling Unit Fan Motor	Fails to start or fails to start at low speed.	Cooling continues from the remaining operating unit with supplemental cooling from the spray system.	
2.	Cooling Coils	Rupture	The coils are designed for hydrostatic testing at 200 psig which is in excess of the design service water pressure. However, if rupture occurs, cooling continues as noted under item 1 above.	00-01
3.	Plenum and Ductwork	Rupture	The plenums and connected ductwork are designed to withstand the post accident pressure transient and the systems are sufficiently separated so that the effects of a pipe break or missile are confined to a single unit. Thus, rupture of a portion of the system or a single component does not prevent continued cooling as noted under item 1 above.	00-01
4.	Cooling Water Piping	Rupture	The piping for two of the units is physically separated from that of the remaining two units and is designed as Seismic Category I. Thus, rupture of a portion of the system does not affect all units and does not prevent cooling as noted under item 1 above.	

TABLE 6.2-52 (Continued)

SINGLE FAILURE ANALYSIS - REACTOR BUILDING HEAT REMOVAL AND FILTERING SYSTEM

	Component	Malfunction	Comments
5.	Water Control Valves	Fail to operate.	The valves are used for switching from the normal cooling water supply to the service water supply during a post accident period. The valves are redundant and are periodically tested. Failure of a single valve does not prevent cooling as noted under item 1 above.
6.	HEPA Filter Bypass Duct	Fails to operate.	Failure of the dampers in the normal position results in loss of filtering ability but not in loss of cooling for the affected units.

Reformatted Per
Amendment 02-01

TABLE 6.2-52a

MAJOR HEAT GENERATING COMPONENTS INSIDE THE REACTOR BUILDING

Component
Steam Generators
Pressurizer
Reactor Coolant Pumps and Motors
Reactor Vessel
Control Rod Drive Mechanisms
Pressurizer Relief Tank
Reactor Coolant Drain Tank Pumps
Regenerative Heat Exchanger
Excess Letdown Heat Exchangers
Ventilation Equipment Fan Motors
Piping
Equipment and Piping Supports and Restraints

TABLE 6.2-52b

MAXIMUM EXPECTED LEAKAGE TO THE AUXILIARY BUILDING FROM THE REACTOR BUILDING SPRAY SYSTEM

RN 03-019

Leakage Source	Leakage (cc/hr)
Reactor Building Spray Pump Seals	300
Flanges	120
Process Valves	12
Instrumentation Valves	18
Valve Seats as Boundaries	420
Total	870

TABLE 6.2-52c

REFUELING WATER STORAGE TANK DRAWDOWN ANALYSIS BRANCH DATA

No.	From Junction	To Junction	Pipe ID (Inch)	Straight Pipe (Feet)	90° Elbow	45° Elbow	Gate Valve	Check Valve	Tee Run	Tee Branch	Reducers	Total Equivalent Diameters (L/D)	Total Equivalent Diameters (Feet)	Elevation Change Feet	
1	3	4	19.25	37.10	2	1	1		1		1	99.30	196.39	-13.80	
2	4	5	11.94	75.60	2	1			1	1		94.30	169.43	5.00	
3	5	6	11.94	43.20	2	2	1	1	2		1	210.60	252.75	-7.00	
4	6	7	11.94	32.10	2	1	1			1		97.30	128.91	-21.70	
5	7	8	10.02	4.70	1						1	18.70	20.31	-1.90	
6	5	9	11.94	56.70	3	1	1	1	2	1		275.30	330.62	-7.00	02-01
7	9	10	11.94	38.40	4	2	1		1			91.60	129.54	-21.70	
8	10	11	10.02	4.70	1						1	19.00	20.56	-1.90	
9	4	12	19.25	3.00					2			20.00	35.08	0.00	
10	12	13	13.12	78.20	2	1			2	1		114.30	203.17	5.00	02-01
11	13	14	13.12	93.10	6	3	1	1	2	1		340.90	465.82	-27.80	
12	13	15	13.12	60.90	5		1	1	1	1		283.00	370.31	-27.80	
13	12	16	19.25	3.00					1			10.00	19.04	0.00	
14	16	17	7.98	10.90	1			1	1		1	185.10	133.99	0.00	
15	17	18	7.98	2.60			1		1	1		73.00	51.14	0.00	
16	18	19	7.98	37.90	2	1			1	2		144.30	133.86	-5.50	02-01
17	18	24	7.98	11.90	1		2		1	1		99.00	77.73	0.00	
18	19	20	6.07	24.50	3		1				1	63.70	56.72	-4.60	
19	17	21	7.98	3.60			1		2			33.00	25.54	0.00	02-01
20	21	22	7.98	48.00	4	1			2			90.30	108.05	-5.50	
21	22	23	6.07	23.40	3		1				1	63.70	55.62	-4.60	
22	21	24	7.98	11.20			2		1	1		86.00	68.39	0.00	
23	24	25	7.98	23.50	4	1			1			70.30	70.25	-5.50	
24	25	26	6.07	16.20	2	1	1				1	59.00	46.04	-4.60	
25	31	32	3.07	56.90	3	1	1		2	1	2	145.10	94.02	-7.00	02-01
26	32	9	3.07	57.60	8	2	1	1		2	1	368.60	151.90	-7.00	
27	32	6	3.07	32.00	3		1	1	1	1	1	247.00	95.19	-7.00	

TABLE 6.2-52d

PUMP HEAD DATA

Reactor Building Spray Pumps A and B

Flow (gpm)	0.0	1500.0	2000.0	2500.0	3000.0	3500.0	4000.0
Head (feet)	506.0	482.0	471.0	450.0	430.0	400.0	369.0

Charging Pump A

Flow (gpm)	0.0	200.0	300.0	400.0	500.0	600.0	700.0
Head (feet)	6100.0	6000.0	5780.0	5300.0	4690.0	3820.0	2900.0

Charging Pump B

Flow (gpm)	0.0	200.0	300.0	400.0	500.0	600.0	700.0
Head (feet)	6075.0	5975.0	5740.0	5260.0	4600.0	3750.0	2900.0

Charging Pump C

Flow (gpm)	0.0	200.0	300.0	400.0	500.0	600.0	700.0
Head (feet)	6020.0	5980.0	5700.0	5150.0	4470.0	3650.0	2700.0

Residual Heat Removal Pump A

Flow (gpm)	0.0	1000.0	2000.0	3000.0	3500.0	4000.0	5000.0
Head (feet)	347.0	325.0	310.0	280.0	261.0	240.0	193.0

Residual Heat Removal Pump B

Flow (gpm)	0.0	1000.0	2000.0	3000.0	3500.0	4000.0	5000.0
Head (feet)	350.0	325.0	310.0	279.0	260.0	238.0	195.0

TABLE 6.2-52e

TANK PARAMETERS

Tank	ID	* Outlet Elevation	Minimum Fill Level Elevation	Maximum Fill Level Elevation	
Refueling Water Storage Tank	40’	415.25’	464’ - 3”	466’ - 5”	
Sodium Hydroxide Storage Tank	45”	413.50’	450’ - 7 1/16”	451’ - 8 1/8”	99-01

* Center Line 02-01

TABLE 6.2-52f

SPECIFIC VOLUME AND VISCOSITY CURVES

	NaOH (20 w/o)		NaOH (22 w/o)	
	ρ^{-1}	ν	ρ^{-1}	ν
Temp. (°F)	(ft^3/lb)	(lbm/ft-sec)	(ft^3/lb)	(lbm/ft-sec)
40	1.306×10^{-2}	6.100×10^{-3}	1.282×10^{-2}	7.150×10^{-3}
50	1.309×10^{-2}	4.650×10^{-3}	1.285×10^{-2}	5.650×10^{-3}
60	1.312×10^{-2}	3.600×10^{-3}	1.288×10^{-2}	4.490×10^{-3}
70	1.315×10^{-2}	2.840×10^{-3}	1.292×10^{-2}	3.600×10^{-3}
80	1.318×10^{-2}	2.280×10^{-3}	1.295×10^{-2}	2.900×10^{-3}
90	1.322×10^{-2}	1.900×10^{-3}	1.299×10^{-2}	2.390×10^{-3}
100	1.326×10^{-2}	1.630×10^{-3}	1.302×10^{-2}	1.990×10^{-3}

TABLE 6.2-52g

REACTOR BUILDING SPRAY AND ECCS STORAGE TANKS DRAWDOWN TRANSIENT ANALYSIS
ANALYTICAL RESULTS TO MINIMUM SODIUM HYDROXIDE INITIAL CONDITIONS

	TEMP (°F)	CONCENTRATION (w/o)	INITIAL LEVEL (feet)	TIME (min)	FINAL LEVEL (feet)	VOLUME DRAWN FROM TANK (gallons)	MASS OF CHEMICAL (lbs)	MASS OF WATER (lbs)
Reactor Building Spray System								
Design Case				67.49				
Refueling Water Storage Tank	70	1.429	51.15		1.55	466,220	55,490	3,838,770
Sodium Hydroxide Storage Tank	70	20	37.00		3.47	2,770	5,630	22,540
Normal Case				39.06				
Refueling Water Storage Tank	70	1.429	51.15		1.55	466,220	55,490	3,838,770
Sodium Hydroxide Storage Tank	70	20	37.00		0.38	3,020	6,140	24,570
Normal Case With One RBSS Pump Inoperable				49.25				
Refueling Water Storage Tank	70	1.429	51.15		1.55	466,220	55,490	3,838,770
Sodium Hydroxide Storage Tank	70	20	37.00		9.63	2,260	4,600	18,380
Normal Case With One RHR Pump Inoperable				43.55				
Refueling Water Storage Tank	70	1.429	51.15		6.73	417,530	49,700	3,437,860
Sodium Hydroxide Storage Tank	70	20	37.00		0.00	3,060	6,220	24,900
				48.660				
Refueling Water Storage Tank					1.55	466,220	55,490	3,838,770
Sodium Hydroxide Storage Tank					0.00	3,060	6,220	24,900

02-01

TABLE 6.2-52h

REACTOR BUILDING SPRAY AND ECCS STORAGE TANKS DRAWDOWN TRANSIENT ANALYSIS
ANALYTICAL RESULTS TO MAXIMUM SODIUM HYDROXIDE INITIAL CONDITIONS

	TEMP (°F)	CONCENTRATION (w/o)	INITIAL LEVEL (feet)	TIME (min)	FINAL LEVEL (feet)	VOLUME DRAWN FROM TANK (gallons)	MASS OF CHEMICAL (lbs)	MASS OF WATER (lbs)
Reactor Building Spray System								
Design Case				64.60				
Refueling Water Storage Tank	70	1.314	49.00		1.55	446,010	48,810	3,672,360
Sodium Hydroxide Storage Tank	70	22	38.00		3.89	2,820	6,420	22,770
Normal Case				37.41				
Refueling Water Storage Tank	70	1.314	49.00		1.55	446,010	48,810	3,672,360
Sodium Hydroxide Storage Tank	70	22	38.00		1.07	3,050	6,940	24,630
Normal Case With One RBSS Pump Inoperable				47.16				
Refueling Water Storage Tank	70	1.314	49.00		1.55	446,010	48,810	3,672,360
Sodium Hydroxide Storage Tank	70	22	38.00		10.23	2,290	5,210	18,490
Normal Case With One RHR Pump Inoperable				42.22				
Refueling Water Storage Tank	70	1.314	49.00		6.00	404,180	44,230	3,327,940
Sodium Hydroxide Storage Tank	70	22	38.00		0.00	3,140	7,150	25,350
				46.55				
Refueling Water Storage Tank					1.55	446,010	48,810	3,672,360
Sodium Hydroxide Storage Tank					0.00	3,140	7,150	25,350

02-01

TABLE 6.2-53

ISOLATION SUMMARY

Penetration No. XRP- (Unless Noted)	Service	System (Note 1)	Fluid	General Design Criteria	Valve Arrang. Fig. 6.2-52	Essential (Note 2)	Line Size	Pipe Length to Vlv. (Note 3)	Type B Leak Test (Note 4)	Drawing Remarks	00-01
101	Reactor Building Purge Exhaust	AH	Air	56	1	No	36"	18"	No	D-912-103-G-11	
102	Reactor Building Cooling Unit B Return	SW	Water	57	12	Yes	16"	31’3"	No	D-302-222-G-03	
103	Post Accident Hydrogen Purge Line	HR	Air	56	1	No	6"	20’	No	D-302-861-G-11	
NOZZ 104	Spare	--	--	--	--	--	--	--	--	Sleeve size - 18"	00-01
105A	H_2 Analyzer Supply	HR	Air	56	1	Yes	3/8"	--	No	D-302-861-E-11	
105B	H_2 Analyzer Discharge	HR	Air	56	1	Yes	3/8"	--	No	D-302-861-E-11	
NOZZ 106	Spare	--	--	--	--	--	--	--	--	Sleeve size - 18"	
107	Fuel Transfer Tube	SF	Water	56	13	No	20"	--	Yes	D-302-651-H-10	00-01
108	Spare	--	--	--	--	--	--	--	--	Sleeve size - 18"	
201	Reactor Building Leak Rate Test Flow Test Line	LR	Air	56	21	No	2"	8’6"	Yes	D-302-811-J-08 Type B test for blind flanges in process line	00-01
202	Main Steam Loop C	MS	Steam	57	11A	Yes	32"	150’	No	D-302-011-F-10	
203	Feedwater Loop C	FW	Water	57	11B	No	18"	16’9"	No	D-302-083-G-10	
204	Component Cooling to Reactor Coolant Pumps	CC	Water	57	2	No	3"	21’3"	No	D-302-612-F-01	
205	Emergency Feedwater B	EF	Water	57	11D	Yes	4"	8’6"	No	D-302-083-H-10	
206	Feedwater Loop B	FW	Water	57	11B	No	18"	7’6"	No	D-302-083-E-10	
207	Main Steam Loop B	MS	Steam	57	11A	Yes	32"	88’9"	No	D-302-011-D-11	
208	CRDM Coolant Water inlet Line	AC	Water	56	7	No	6"	7’3"	No	D-302-852-E-11	

TABLE 6.2-53 (Continued)

ISOLATION SUMMARY

Penetration No. XRP- (Unless Noted)	Service	System (Note 1)	Fluid	General Design Criteria	Valve Arrang. Fig. 6.2-52	Essential (Note 2)	Line Size	Pipe Length to Vlv. (Note 3)	Type B Leak Test (Note 4)	Drawing Remarks	
209	CRDM Coolant Water Outlet Line	AC	Water	56	7	No	6"	13’6"	No	D-302-852-E-11	00-01
210	Reactor Building Leak Rate Test Pressure Sensing Line	LR	Air	56	21	No	2"	8’6"	Yes	D-302-811-J-7 Type B test for blind flanges in process line	00-01
211	Reactor Building Leak Rate Test Blowdown	LR	Air	56	21	No	8"	13’	Yes	D-302-811-J-7 Type B test for blind flanges in process line	
212	Reactor Building Leak Rate Test Blowdown	LR	Air	56	21	No	8"	9’9"	Yes	D-302-811-J-10	
213	Emergency Feedwater C	EF	Water	57	11D	Yes	4"	21’	No	D-302-083-H-9	
214	Spare	--	--	--	--	--	--	--	--	Sleeve size - 10"	
215	Spare	--	--	--	--	--	--	--	--	Sleeve size - 12"	
216	Reactor Building Leak Rate Test Pressurization Line	LR	Air	56	21	No	8"	5’	Yes	D-302-811-J-11 Type B test for blind flanges in process line	
217	Reactor Building Pressure Sensing	--	See remarks	56	14	Yes	1/4"	--	--	D-302-861-B-3 The capillary between bellows and transmitter is fluid filled.	RN 07-022
218	Spare	--	--	--	--	--	--	--	--	Sleeve size - 10"	
219	Steam Generator Blowdown Loop C	BD	Water	57	11C	No	3"	17’8"	No	D-302-781-H-9	00-01
220	Steam Generator Blowdown Loop C Sampling Line	SS	Water	57	11C	No	3/8"	15’	No	D-302-771-G-10	
221	Seal Injection to Reactor Coolant Pump C	CS	Water	55	2	No	1-1/2"	15’6"	No	E-302-673-H-15	
222	High Head Safety Injection to Reactor Loops	SI	Water	55	6	Yes	3"	15’6"	No	E-302-691-C-9	

TABLE 6.2-53 (Continued)

ISOLATION SUMMARY

Penetration No. XRP- (Unless Noted)	Service	System (Note 1)	Fluid	General Design Criteria	Valve Arrang. Fig. 6.2-52	Essential (Note 2)	Line Size	Pipe Length to Vlv. (Note 3)	Type B Leak Test (Note 4)	Drawing Remarks	00-01
223	Sampling Line from Reactor Coolant Loop C	SS	Water	55	1	No	3/8"	15’	No	D-302-771-E-10	
224	Steam Generator Blowdown Loop B	BD	Water	57	11C	No	3"	16’6"	No	D-302-781-F-9	00-01
225	Steam Generator Blowdown Loop B Sampling Line	SS	Water	57	11C	No	3/8"	15’	No	D-302-771-F-10	
226	Residual Heat Removal Pump Suction for Reactor Coolant Loop C	RH	Water	55	3	Yes	12"	--	No	E-302-641-F-13	
227	Low Head Safety Injection to Reactor Coolant Loops	SI	Water	55	2	Yes	10"	12’9"	No	E-302-693-G-14	00-01
228	Spare	--	--	--	--	--	--	--	--	Sleeve size 10"	
229	Seal Injection to Reactor Coolant Pump B	CS	Water	55	2	No	1-1/2"	11’6"	No	E-302-672-H-15	
230A	Reactor Vessel Level Sensing Line	RC	See remarks	57	25	Yes	1/4"	--	--	E-302-601-G-5 Fluid between bellows & transmitter is water	00-01
230B	Reactor Vessel Level Sensing Line	RC	See remarks	57	25	Yes	1/4"	--	--	E-302-601-G-5 Fluid between bellows & transmitter is water	00-01
230C	Reactor Vessel Level Sensing Line	RC	See remarks	57	25	Yes	1/4"	--	--	E-302-601-G-5 Fluid between bellows & transmitter is water	00-01
230D	Spare	--	--	--	--	--	3/8"	--	--	Capped spare	
230E	Spare	--	--	--	--	--	3/8"	--	--	Capped spare	
231	Demineralized Water	DN	Air	56	17	No	1"	3’	No	D-302-715-P-11	98-01
232	Spare	--	--	--	--	--	--	--	--	Sleeve size 12"	
301A	H2 Analyzer Supply	HR	Air	56	15	Yes	3/8"	--	No	D-302-861-C-11	

TABLE 6.2-53 (Continued)

ISOLATION SUMMARY

Penetration No. XRP- (Unless Noted)	Service	System (Note 1)	Fluid	General Design Criteria	Valve Arrang. Fig. 6.2-52	Essential (Note 2)	Line Size	Pipe Length to Vlv. (Note 3)	Type B Leak Test (Note 4)	Drawing Remarks
301B	H2 Analyzer Discharge	HR	Air	56	22	Yes	3/8"	--	No	D-302-861-C-11
302	Alternate R.B. Purge Line	HR	Air	56	1	No	6"	9'9"	No	D-302-861-K-10
303	Supply to Reactor Building Spray Nozzles - Train B	SP	Water	56	2	Yes	10"	22'6"	No	D-302-661-E-11
304	Service Water from Reactor Building Cooling Unit A	SW	Water	57	12	Yes	16"	27'3"	No	D-302-222-C-5
305	Service Water from Reactor Building Cooling Unit A	SW	Water	57	12	Yes	16"	24'3"	No	D-302-222-C-2
306	Feedwater Loop A	FW	Water	57	11B	No	18"	11'0"	No	D-302-083-B-9
307	Number Not Used	--	--	--	--	--	--	--	--	Not Used
308	Emergency Feedwater A	EF	Water	57	11D	Yes	4"	18'9"	No	D-302-083-C-9
309	Abandoned Rad Monitor and Check Source	RM	NA	--	23	No	--	Note 6	No	Sleeve size 10" welded shut inside containment
310	Reactor Building Station Service Air	SA	Air	56	16	No	2"	13'9"	No	D-302-241-B-10
311	Reactor Building Instrument Air	IA	Air	56	2	No	2"	13'9"	No	D-302-273-G-4
312	Component Cooling Water to Reactor Coolant Pump Bearings	CC	Water	57	2	No	8"	19'	No	D-302-612-H-11
313	Nitrogen Supply to Feedwater Lines	NG	N2	56	16	No	1"	1'	No	D-302-311-B-12
314	Sampling Line from Reactor Coolant Loop B	SS	Water	55	1	No	3/8"	16'	No	D-302-771-D-10
315	Spare	--	--	--	--	--	--	--	--	Sleeve size 18"

00-01

TABLE 6.2-53 (Continued)

ISOLATION SUMMARY

Penetration No. XRP- (Unless Noted)	Service	System (Note 1)	Fluid	General Design Criteria	Valve Arrang. Fig. 6.2-52	Essential (Note 2)	Line Size	Pipe Length to Vlv. (Note 3)	Type B Leak Test (Note 4)	Drawing Remarks	00-01
316	Residual Heat Removal Pump Suction from Reactor Coolant Loop A	RH	Water	55	3	Yes	12"	--	No	E-302-641-H-13	99-01
317	Fill Line to Accumulators	SI	Water	56	2	No	1"	9’9"	No	E-302-692-G-3	
318	Reactor Coolant to Letdown Heat Exchanger	CS	Water	55	20	No	3"	16’	No	E-302-673-A-3	
319	Reactor Building Instrument Air Compressor Suction Line	IA	Air	56	1	No	6"	14’3"	No	D-302-273-B-4	
320	Accumulator Nitrogen Supply	SI	N2	56	2	No	1"	10’6"	No	E-302-692-A-3	00-01
321	Accumulator Test Line	SI	Water	56	1	No	3/4"	11’3"	No	E-302-692-B-14	00-01
322	Low Head Safety Injection to Reactor Coolant Loops	SI	Water	55	2	Yes	10"	12’9"	No	E-302-693-E-14	
323	Accumulator Sampling Line	SS	Water	57	8	No	3/8"	11’3"	No	D-302-771-J-10	
324	Breathing Air	IA	Air	56	17	No	2"	8’0"	No	D-302-274-C-4	98-01
325	Low Head Safety Injection to Reactor Coolant Loop Hot Legs	SI	Water	55	4	Yes	10"	12’9"	No	E-302-693-C-14	
326	Steam Generator Blowdown Loop A	BD	Water	57	11C	No	3"	12’6"	No	D-302-781-C-9	
327	Spray Pump A Suction from Reactor Building Recirculation Sump	SP	Water	56	5	Yes	12"	53’ (Note 5)	Yes	D-302-661-F-11	
328	Spray Pump B Suction from Reactor Building Recirculation Sump	SP	Water	56	5	Yes	12"	34’ (Note 5)	Yes	D-302-661-F-11	02-01

TABLE 6.2-53 (Continued)										
ISOLATION SUMMARY										
Penetration No. XRP- (Unless Noted)	Service	System (Note 1)	Fluid	General Design Criteria	Valve Arrang. Fig. 6.2-52	Essential (Note 2)	Line Size	Pipe Length to Vlv. (Note 3)	Type B Leak Test (Note 4)	Drawing Remarks
329	RHR Pump A Suction from Reactor Building Recirculation Sump	SI	Water	56	5	Yes	14"	44’3" (Note 5)	Yes	E-302-693-J-14
330	Component Cooling Water from Reactor Coolant Pump Bearings	CC	Water	57	7	No	8"	13’6"	No	D-302-612-H-12
401	Supply to Reactor Building Spray Nozzles Train A	SP	Water	56	2	Yes	10"	27’9"	No	D-302-661-E-11
402	Reactor Building Purge Supply	AH	Air	56	1	No	36"	18"	No	D-912-103-D-5
403	Reactor Building Cooling Unit B Supply	SW	Water	57	12	Yes	16"	40’	No	D-302-222-K-6
404	Fire Service Hose Reel Supply	FS	Air	56	17	No	4"	14’	No	D-302-231-D-6
405	Sampling Line from Pressurizer	SS	Steam or Water	55	15	No	3/8"	14’	No	D-302-771-C-10
406A	Spare	--	--	--	--	--	1/4"	--	--	B-814-028 Spare - tube seal welded.
406B	Reactor Building Pressure Sensing Line	--	See remarks	56	24	Yes	1/4"	--	--	D-302-861-B-3 The capillary between bellows and transmitter is fluid filled.
407A	Radiation Monitor Supply	SS	Air	56	1	No	1"	2’-5"	No	D-302-771-E-14
407B	Radiation Monitor Return	SS	Air	56	1	No	1"	2’-5"	No	D-302-771-E-14
408	Seal Injection to Reactor Coolant Pump A	CS	Water	55	2	No	1-1/2"	17’3"	No	E-302-671-H-16
409	Charging Line to Regenerative Heat Exchanger	CS	Water	55	2	No	3"	16’6"	No	E-302-673-A-3

00-01

98-01

RN 07-022

TABLE 6.2-53 (Continued)

ISOLATION SUMMARY

Penetration No. XRP- (Unless Noted)	Service	System (Note 1)	Fluid	General Design Criteria	Valve Arrang. Fig. 6.2-52	Essential (Note 2)	Line Size	Pipe Length to Vlv. (Note 3)	Type B Leak Test (Note 4)	Drawing Remarks	00-01
410	Reactor Coolant Pump Seal Water Return	CS	Water	55	7	No	2"	12'9"	No	E-302-673-C-3	
411	Steam Generator Blowdown Loop A Sampling Line	SS	Water	57	11C	No	3/8"	12'6"	No	D-302-771-F-10	
412	High Head Safety Injection to Reactor Coolant Loops	SI	Water	55	6	Yes	3"	11'3"	No	E-302-691-C-8	
413	Spare	--	--	--	--	--	--	--	--	Sleeve size - 10"	
414	Spare	--	--	--	--	--	--	--	--	Sleeve size - 10"	
415	High Head Safety Injection to Reactor Coolant	SI	Water	55	6	Yes	3"	11'3"	No	E-302-691-C-6	
416	Spare	--	--	--	--	--	--	--	--	Sleeve size - 10"	
417A	Reactor Vessel Level Sensing Line	RC	See remarks	57	25	Yes	1/4"	--	--	E-302-601-J-5 Fluid between bellows & transmitter is water	02-01
417B	Reactor Vessel Level Sensing Line	RC	See remarks	57	25	Yes	1/4"	--	--	E-302-601-J-5 Fluid between bellows & transmitter is water	02-01
417C	Reactor Vessel Level Sensing Line	RC	See remarks	57	25	Yes	1/4"	--	--	E-302-601-J-5 Fluid between bellows & transmitter is water	02-01
417D	Sample Return to PRT	SS	Water	56	1	No	3/8"	2'0"	No	D-302-772-J-3	
417E	Spare	--	--	--	--	--	3/8"	--	--	Capped spare	
418	Reactor Coolant Drain Tank to Vent Header and H2	WL	H2	56	1	No	3/4"	20'6"	No	E-302-735-B-12	
419	Refueling Cavity Drain Line	SF	Air	56	17	No	3"	5'	No	D-302-651-H-12	98-01
420	Pressure Relief Tank	RC	N2	56	1	No	1"	15'6"	No	E-302-602-B-6	

TABLE 6.2-53 (Continued)

ISOLATION SUMMARY

Penetration No. XRP- (Unless Noted)	Service	System (Note 1)	Fluid	General Design Criteria	Valve Arrang. Fig. 6.2-52	Essential (Note 2)	Line Size	Pipe Length to Vlv. (Note 3)	Type B Leak Test (Note 4)	Drawing Remarks	
											00-01
421	Refueling Cavity Fill Line	SF	Air	56	17	No	3"	3’3"	No	D-302-651-H-12	98-01
422	Pressure Relief Tank Makeup	RC	Water	56	2	No	3"	16’	No	E-302-602-B-3	
423	Reactor Coolant Drain Tank	WL	Water	56	1	No	3"	11’3"	No	E-302-735-C-5	
424	Reactor Building Sump Drain	ND	Water	56	1	No	3"	14’6"	No	D-302-821-A-9	
425	RHR Pump B Suction from Reactor Building Recirculation Sump	SI	Water	56	5	Yes	14"	24’ (Note 5)	Yes	E-302-693-H-14	
426	High Head Safety Injection to Reactor Coolant Loops	SI	Water	55	18	Yes	3"	16’	No	E-302-691-C-15	
427	Fire Service Deluge	FS	Water	56	2	No	4"	11’9"	No	D-302-231-D-6	
428	Main Steam Loop A	MS	Steam	57	11A	Yes	32"	176’9"	No	D-302-011-B-10	
500	Reactor Building Pressure Sensing	--	See remarks	56	14	Yes	1/4"	--	No	D-302-861-A-3 The capillary between bellows and transmitter is fluid filled.	RN 07-022
505	Outage Penetration	--	Air	56	21	No	18"	--	Yes	E-511-106-F-2 Type B Test for Blind Flanges	
600	Outage Penetration	--	Air	56	21	No	12"	--	Yes	E-511-106-E-4 Type B Test for Blind Flanges	
602	Outage Penetration	--	Air	56	21	No	18"	--	Yes	E-511-106-D-2 Type B Test for Blind Flanges	
703	Reactor Building Pressure Sensing	--	See remarks	56	24	Yes	1/4"	--	No	D-302-861-A-3 The capillary between bellows and transmitter is fluid filled.	RN 07-022

TABLE 6.2-53 (Continued)

ISOLATION SUMMARY

NOTES TO TABLE 6.2-53

1. System abbreviations are as follows:

AC	-	CRDM Cooling Water System
AH	-	Air Handling System
BD	-	Steam Generator Blowdown System
CC	-	Component Cooling Water System
CS	-	Chemical and Volume Control System
DN	-	Demineralized Water, Nuclear Services System
EF	-	Emergency Feedwater
FS	-	Fire Service System
FW	-	Feedwater System
HR	-	Post Accident Hydrogen Removal System
IA	-	Instrument Air System
LR	-	Leak Rate Test System
MS	-	Main Steam System
ND	-	Reactor Building and Auxiliary Building Sump Pumps System
NG	-	Nitrogen Blanketing System
RC	-	Reactor Coolant System
RM	-	Radiation Monitoring System
SA	-	Service Air System
SF	-	Spent Fuel Cooling System
SI	-	Safety Injection System
SP	-	Reactor Building Spray System
SS	-	Nuclear Sampling System
SW	-	Service Water System
WL	-	Liquid Waste Processing System

TABLE 6.2-53 (Continued)

ISOLATION SUMMARY

NOTES TO TABLE 6.2-53 (Continued)

2. “Yes" indicates that the penetration serves a system which is essential for safe shutdown or accident mitigation. Penetrations which may provide some post accident services (e.g., sampling) are not indicated as essential. 99-01

3. This is the distance, along the pipe, from the outer edge of the concrete of the reactor building to the nearest edge of the outside containment isolation valve. Where there are one or more branches from the pipe, the distance is to the largest valve to the farthest of more than one equal size valves.

4. “No" indicates that these penetrations are sealed inside containment by a flat plate of 1/2 inch thickness or thicker, welded to both the penetration sleeve and process pipe. Since this isolation barrier does not utilize a resilient or flexible seal, these penetrations do not require Type B leakage tests.

5. This distance is measured as in Note 3 but it should be noted that the valve is within the valve compartment discussed in Section 6.2.4.2.2.

6. Isolation valve arrangement and pipe length to valve are not applicable since penetration is sealed inside containment by a welded cap.

TABLE 6.2-53a

SUMMARY OF VALVES SUBJECTED TO TYPE C LEAKAGE TESTS

Penetration Number	Service	System	Type C Leakage Test (1) Inside Containment	Type C Leakage Test (1) Outside Containment	
101	Reactor Building Purge Exhaust	AH	Yes (6)	Yes	RN 03-035
102 (7)	Reactor Building Cooling Unit B Return	SW	N/A	No (10)	RN 14-009
103	Post Accident Hydrogen Purge Line	HR	Yes	Yes	RN 03-035
104	Spare	-	N/A	N/A	
105A	H_2 Analyzer Supply	HR	Yes	Yes	RN 03-035
105B	H_2 Analyzer Discharge	HR	Yes	Yes	
106	Spare	-	N/A	N/A	
107	Fuel Transfer Tube	SF	No (2)	No (2)	RN 03-035
108	Spare	-	N/A	N/A	
201 (7)	Reactor Building Leak Rate Test Flow Test Line	LR	No (2)	No (2)	
202	Main Steam Loop C	MS	N/A	No (3)	RN 03-035
203	Feedwater Loop C	FW	N/A	No (3)	RN 98-098
204 (7)	Component Cooling to Reactor Coolant Pumps	CC	Yes	Yes	
205	Emergency Feedwater B	EF	N/A	No (3)	
206	Feedwater Loop B	FW	N/A	No (3)	
207	Main Steam Loop B	MS	N/A	No (3)	RN 03-035
208	CRDM Coolant Water Inlet Line	AC	Yes	Yes	
209	CRDM Coolant Water Outlet Line	AC	Yes	Yes	
210 (7)	Reactor Building Leak Rate Test Pressure Sensing Line	LR	No (2)	No (2)	
211 (7)	Reactor Building Leak Rate Test Blowdown	LR	No (2)	No (2)	

TABLE 6.2-53a (Continued)

SUMMARY OF VALVES SUBJECTED TO TYPE C LEAKAGE TESTS

Penetration Number	Service	System	Type C Leakage Test [1] Inside Containment	Type C Leakage Test [1] Outside Containment	
212 [7]	Reactor Building Leak Rate Test Blowdown	LR	No [2]	No [2]	
213	Emergency Feedwater C	EF	N/A	No [3]	
214	Spare	-	N/A	N/A	
215	Spare	-	N/A	N/A	
216 [7]	Reactor Building Leak Rate Test Pressurization Line	LR	No [2]	No [2]	
217	Reactor Building Pressure Sensing	-	N/A	N/A	
218	Spare	-	N/A	N/A	
219	Steam Generator Blowdown Loop C	BD	N/A	No [3]	RN 03-035
220	Steam Generator Blowdown Loop C Sampling Line	SS	N/A	No [3]	
221	Seal Injection to Reactor Coolant Pump C	CS	Yes	Yes	
					RN 11-034
223	Sampling Line from Reactor Coolant Loop C	SS	Yes	Yes	
224	Steam Generator Blowdown Loop B	BD	N/A	No [3]	RN 03-035
225	Steam Generator Blowdown Loop B Sampling Line	SS	N/A	No [3]	
226 [7]	Residual Heat Removal Pump Suction from Reactor Coolant Loop C	RH	Yes [8]	N/A	
227 [7]	Low Head Safety Injection to Reactor Coolant Loops	SI	No [5]	No [10]	RN 14-009
228	Spare	-	N/A	N/A	

TABLE 6.2-53a (Continued)

SUMMARY OF VALVES SUBJECTED TO TYPE C LEAKAGE TESTS

Penetration Number	Service	System	Type C Leakage Test [1] Inside Containment	Type C Leakage Test [1] Outside Containment	
229	Seal Injection to Reactor Coolant Pump B	CS	Yes	Yes	
230A	Reactor Vessel Level Sensing Line	RC	N/A	N/A	
230B	Reactor Vessel Level Sensing Line	RC	N/A	N/A	
230C	Reactor Vessel Level Sensing Line	RC	N/A	N/A	
230D&E	Spare	-	N/A	N/A	
231	Demineralized Water	DN	Yes	Yes	RN 03-035
232	Spare	-	N/A	N/A	
301A	H_2 Analyzer Supply	HR	Yes	Yes	RN 03-035
301B	H_2 Analyzer Discharge	HR	Yes	Yes	RN 03-035
302	Alternate R.B. Purge Line	HR	Yes	Yes	RN 03-035
303	Supply to Reactor Building Spray Nozzles - Train B	SP	Yes	Yes	RN 03-035
304 [7]	Service Water to Reactor Building Cooling Unit A	SW	N/A	No [10]	RN 14-009
305 [7]	Service Water from Reactor Building Cooling Unit A	SW	N/A	No [10]	RN 14-009
306	Feedwater Loop A	FW	N/A	No [3]	
307	Number Not Used	-	N/A	N/A	
308	Emergency Feedwater A	EF	N/A	No [3]	
309	Abandoned Rad Monitor and Check Source	RM	N/A	N/A	
310	Reactor Building Station Service Air	SA	Yes	Yes	RN 03-035
311	Reactor Building Instrument Air	IA	Yes	Yes	RN 03-035

TABLE 6.2-53a (Continued)

SUMMARY OF VALVES SUBJECTED TO TYPE C LEAKAGE TESTS

Penetration Number	Service	System	Type C Leakage Test [1] Inside Containment	Type C Leakage Test [1] Outside Containment	
312 [7]	Component Cooling Water to Reactor Coolant Pump Bearings	CC	Yes	Yes	
313	Nitrogen Supply to Feedwater Lines	NG	Yes	Yes	RN 03-035
314	Sampling Line from Reactor Coolant Loop B	SS	Yes	Yes	RN 03-035
315	Spare	-	N/A	N/A	
316 [7]	Residual Heat Removal Pump Suction from Reactor Coolant Loop A	RH	Yes [8]	N/A	RN 01-113
317	Fill Line to Accumulators	SI	Yes	Yes	RN 03-035
318	Reactor Coolant to Letdown Heat Exchanger	CS	Yes	Yes	
319	Reactor Building Instrument Air Compressor Suction Line	IA	Yes	Yes	RN 03-035
320	Accumulator Nitrogen Supply	SI	Yes	Yes	RN 03-035
321	Accumulator Test Line	SI	Yes	Yes	RN 03-035
322 [7]	Low Heat Safety Injection to Reactor Coolant Loops	SI	No [5]	No [10]	RN 14-009
323	Accumulator Sampling Line	SS	N/A	Yes	RN 03-035
324	Breathing Air	IA	Yes	Yes	RN 03-035
325 [7]	Low Head Safety Injection to Reactor Coolant Loop Hot Legs	SI	No [5]	Yes [8]	RN 99-105 14-009
326	Steam Generator Blowdown Loop A	BD	N/A	No [3]	RN 03-035
327	Spray Pump A Suction from Reactor Building Recirculation Sump	SP	N/A	No [10]	RN 14-009
328	Spray Pump B Suction from Reactor Building Recirculation Sump	SP	N/A	No [10]	RN 14-009
329	RHR Pump A Suction from Reactor Building Recirculation Sump	SI	N/A	No [10]	RN 14-009

TABLE 6.2-53a (Continued)

SUMMARY OF VALVES SUBJECTED TO TYPE C LEAKAGE TESTS

Penetration Number	Service	System	Type C Leakage Test [1] Inside Containment	Type C Leakage Test [1] Outside Containment	
330 [7]	Component Cooling Water from Reactor Coolant Pump Bearings	CC	Yes	Yes	
401 [4-I]	Supply to Reactor Building Spray Nozzles Train A	SP	Yes	Yes	
402	Reactor Building Purge Supply	AH	Yes [6]	Yes	RN 03-035
403 [7]	Reactor Building Cooling Unit B Supply	SW	N/A	No [10]	RN 01-080
404	Fire Service Hose Reel Supply	FS	Yes	Yes	RN 03-035
405	Sampling Line from Pressurizer	SS	Yes	Yes	
406A	Spare	-	N/A	N/A	
406B	Reactor Building Pressure Sensing Line	-	N/A	N/A	
407A	Radiation Monitor Supply	SS	Yes	Yes	RN 03-035
407B	Radiation Monitor Return	SS	Yes	Yes	
408	Seal Injection to Reactor Coolant Pump A	CS	Yes	Yes	
409	Charging Line to Regenerative Heat Exchanger	CS	Yes	Yes	
410	Reactor Coolant Pump Seal Water Return	CS	Yes	Yes	
411	Steam Generator Blowdown Loop A Sampling	SS	N/A	No [3]	RN 03-035
					RN 11-034
413	Spare	-	N/A	N/A	
414	Spare	-	N/A	N/A	
					RN 11-034

TABLE 6.2-53a (Continued)

SUMMARY OF VALVES SUBJECTED TO TYPE C LEAKAGE TESTS

Penetration Number	Service	System	Type C Leakage Test (1): Inside Containment	Type C Leakage Test (1): Outside Containment	
416	Spare	-	N/A	N/A	
417A,B,C	Reactor Vessel Level Sensing	RC	N/A	N/A	
417D	Sample Return to PRT	SS	Yes	Yes	RN 03-035
417E	Spare	-	N/A	N/A	
418	Reactor Coolant Drain Tank to Vent Header and H_2	WL	Yes	Yes	RN 03-035
419	Refueling Cavity Drain Line	SF	Yes	Yes	
420	Pressure Relief Tank	RC	Yes	Yes	
421	Refueling Cavity Fill Line	SF	Yes	Yes	
422	Pressure Relief Tank Makeup	RC	Yes	Yes	
423	Reactor Coolant Drain Tank	WL	Yes	Yes	
424	Reactor Building Sump Drain	ND	Yes	Yes	
425	RHR Pump B Suction from Reactor Building Recirculation Sump	SI	N/A	No (10)	RN 14-009
					RN 11-034
427	Fire Service Deluge	SF	Yes	Yes	RN 03-035
428	Main Steam Loop A	MS	N/A	No (3)	
500	Reactor Building Pressure Sensing	-	N/A	N/A	
505	Outage Penetration	-	No (2)	No (2)	
600	Outage Penetration	-	No (2)	No (2)	
602	Outage Penetration	-	No (2)	No (2)	
703	Reactor Building Pressure Sensing	-	N/A	N/A	

TABLE 6.2-53a (Continued)

SUMMARY OF VALVES SUBJECTED TO TYPE C LEAKAGE TESTS

NOTES:

1. N/A - not applicable.

2. These piping penetrations, which utilize blind flanges as isolation barriers, are subject to Type B leakage testing as indicated in Table 6.2-53.

3. Isolation valves located in secondary side piping that penetrates containment are exempt from Type C leakage testing since the requirements of Appendix J to 10CFR 50 for leak testing of these valves applies only to direct cycle boiling water power reactors. These valves are called Containment Boundary Valves (CBV) on the system flow diagrams.

4. Deleted RN 03-035, 10-21-2003 RN 03-035

5. These check valves are not testable due to their location and orientation within their systems.

6. These valves are tested in the reverse direction due to system design and their orientation.,

7. In operation during Integrated Leak Rate Test or during LOCA.

8. Meets 10CFR50 Appendix J Type C leak test requirements by Type C air leak test or by taking credit for 30 day water seal capability upon passing a water leak test.

9. XVG08889-SI is equipped with a bonnet relief system that allows the test medium to be introduced between the valves disc. This allows the leakage to be measured in both directions. RN 99-105

10. Per ANS/ANSI-56.8, the associated valve is exempt from Type C leakage testing since the boundary does not constitue a potential primary containment atmospheric pathway during or following a LOCA. This valve is a Containment Isolation Valve as indicated on the system flow diagram. RN 01-080

TABLE 6.2-54

ISOLATION VALVE SUMMARY

Pene-tration No.	Valve Arrang Fig 6.2-52	Location Referred to RB	Valve No.	Valve Type	Valve Size (in.)	Primary Method of Actuation (Note 1)	Backup Method of Actuation (Note 1)	Signal (Note 2)	Position Indication (Note 3)	Normal Valve Position	Shutdown Valve Position (Note 4)	Post Accident Position	Power Failure Position	Power Source (Note 5)	Closure Time (Note 6)	Remarks	
101	1	Inside	XVB-2B	Butterfly	36	AO/FC	-	H(B)	Yes	Closed	Closed	Closed	Closed	IA-B	3 Sec Max		RN 02-036
		Outside	XVB-2A	Butterfly	36	AO/FC	-	H(A)	Yes	Closed	Closed	Closed	Closed	IA-A	3 Sec Max		
102	12	Inside	-	-	-	-	-	-	-	-	-	-	-	-	-	Closed System inside	
		Outside	3103B	Gate	16	EMO	HW	J(B)	Yes	Open	Open	Open	As Is	EP-B	45 Sec		
103	1	Inside	6056	Gate	6	AO/FC	-	K(A)	Yes	Closed	Closed	(See Remarks)	Closed	IA-A	5 Sec (Note 9)	Post accident position is initially "closed." Valves may be opened during accident if H^2 sample/purge is required.	02-01
		Outside	6057	Gate	6	AO/FC	-	K(B)	Yes	Closed	Closed	(See Remarks)	Closed	IA-B	5 Sec (Note 9)		
104	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	Capped Spare	
105A	1	Inside	6051B	Solenoid	3/8	EO	-	-	Yes	Closed	Closed	(See Remarks)	Closed	EP-B	NA (Note 11)	Post accident position is initially "closed." Valves may be opened during accident if H^2 sample/purge is required.	02-01
		Outside	6053B	Solenoid	3/8	EO	-	-	Yes	Closed	Closed	(See Remarks)	Closed	EP-B	NA (Note 11)		
105B	1	Inside	6050B	Solenoid	3/8	EO	-	-	Yes	Closed	Closed	(See Remarks)	Closed	EP-B	NA (Note 11)	Post accident position is initially "closed." Valves may be opened during accident if H^2 sample/purge is required.	
		Outside	6052B	Solenoid	3/8	EO	-	-	Yes	Closed	Closed	(See Remarks)	Closed	EP-B	NA (Note 11)		
106	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	Capped Spare	
107	13	Inside	-	Bl Flng	-	-	-	-	-	Closed	Closed	Closed	Closed	-	-	Double O-ring seal provides redundant isolation	
108	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	Capped Spare	
201	21	Inside	-	Bl Flng	-	-	-	-	-	Closed	Closed	Closed	Closed	-	-		02-01
		Outside	-	Bl Flng	-	-	-	-	-	Closed	Closed	Closed	Closed	-	-		
202	11a	Inside	-	-	-	-	-	-	-	-	-	-	-	-	-	Closed system inside	
		Outside	2877B	Globe	1-1/2	AO/FC	HW	J(A, B)	Yes	Open	Closed	Open	Closed	IA-A/B	N/A	Dual Solenoids	00-01
		Outside	2020	Globe	8	AO/FC	HW	J(A, B)	Yes	Closed	Closed	Closed	Closed (note 8)	IA-A/B	N/A	Six Solenoids	
		Outside	2802B	Gate	4	EMO	HW	J(B)	Yes	Open	Closed	Open	As Is	EP-B	N/A		
		Outside	2843C	Globe	1-1/2	AO/FC	HW	J(A, B)	Yes	Open	Closed	Open	Closed	IA-A/B	N/A	Dual Solenoids	
		Outside	2869C	Globe	4	AO/FC	-	C(A, B)	Yes	Closed	Closed	Closed	Closed	IA-A/B	10 Sec	Dual Solenoids	
		Outside	2801C	Globe	32	AO/FC	-	C(A, B)	Yes	Open	Closed	Closed	Closed	IA-A/B	7 Sec	Dual Solenoids	
		Outside	2806K, L, M, N, P	Safety	6x10	PO/FC	-	None	No	Closed	Closed	Closed	Closed			Five Valves	

TABLE 6.2-54 (Continued)

ISOLATION VALVE SUMMARY

Pene-tration No.	Valve Arrang Fig 6.2-52	Location Referred to RB	Valve No.	Valve Type	Valve Size (in.)	Primary Method of Actuation (Note 1)	Backup Method of Actuation (Note 1)	Signal (Note 2)	Position Indication (Note 3)	Normal Valve Position	Shutdown Valve Position (Note 4)	Post Accident Position	Power Failure Position	Power Source (Note 5)	Closure Time (Note 6)	Remarks	
203	11B	Outside	11672	Gate	3	HW	-	-	No	Closed	Closed	Closed	-	-	-	Locked Closed	RN
203	11C	Inside	-	-	-	-	-	-	-	-	-	-	-	-	-	Closed System Inside	07-028
		Outside	1611C	Gate	18	AO/AC	-	M(A	Yes	Open	Closed	Closed	Closed	IA-A	5 Sec (Note 12)	Dual A Train Solenoids	02-01
			1611C	Gate				M(B)	Yes	Open	Closed	Closed	As Is	IA-B		B Train Solenoid	
204	2	Inside	9602	Check	3	-	-	-	-	-	-	-	-	None			
		Outside	9600	Gate	3	EMO	HW	P(A)	Yes	Open	Open	Closed	As Is	EP-A	60 Sec (Note 9)		
205	11D	Inside	-	-	-	-	-	-	-	-	-	-	-	-	-	Closed System inside	
		Outside	1009B	Spl Chk	4	AO/FC	-	J(B)	Yes	Closed	Closed	Open	Closed	IA-B	N/A		
		Outside	1633B	Stp Chk	1-1/2	EMO	HW	P(B)	Yes	Open	Open	Closed	As Is	EP-B	60 Sec (Note 9)		
206	11B	Outside	11671	Gate	3	HW	-	-	No	Closed	Closed	Closed	-	-	-	Locked Closed	RN
206	11C	Inside	-	-	-	-	-	-	-	-	-	-	-	-	-	Closed System Inside	07-028
		Outside	1611B	Gate	18	AO/AC	-	M(A)	Yes	Open	Closed	Closed	Closed	IA-A	5 Sec (Note 12)	Dual Train A Solenoids	02-01
			1611B	Gate				M(B)	Yes	Open	Closed	Closed	As Is	IA-B		Train B Solenoid	
207	11A	Inside	-	-	-	-	-	-	-	-	-	-	-	-	-	Closed System Inside	
		Outside	2010	Globe	8	AO/FC	HW	J(A/B)	Yes	Closed	Closed	Closed	Closed (note 8)	IA-A/B	N/A	Six Solenoids	
		Outside	2802A	Gate	4	EMO	HW	J(A)	Yes	Open	Closed	Open	As Is	EP-A	N/A		00-01
		Outside	2843B	Globe	1-1/2	AO/FC	HW	J(A/B)	Yes	Open	Closed	Open	Closed	IA-A/B	N/A	Dual Solenoids	
		Outside	2801B	Globe	32	AO/FC	-	C(A/B)	Yes	Open	Closed	Closed	Closed	IA-A/B	7 Sec	Dual Solenoids	
		Outside	2869B	Globe	4	AO/FC	-	C(A, B)	Yes	Closed	Closed	Closed	Closed	IA-A/B	10 Sec	Dual Solenoids	
		Outside	2806F, G, H, I, J	Safety	6x10	PO/FC	-	None	No	Closed	Closed	Closed	Closed	-	-	Five Valves	
208	7	Inside	7502	Gate	6	EMO	HW	T(B)	Yes	Open	Open	Closed	As-Is	EP-B	40 Sec (Note 9)		
		Inside	7541	Check	3/4	-	-	-	No	-	-	-	-	-	-		
		Outside	7501	Gate	6	EMO	HW	T(A)	Yes	Open	Open	Closed	As-Is	EP-A	40 Sec (Note 9)		
209	7	Inside	7503	Gate	6	EMO	HW	T(A)	Yes	Open	Open	Closed	As-Is	EP-A	40 Sec (Note 9)		
		Inside	7544	Check	3/4	-	-	-	No	-	-	-	-	-	-		
		Outside	7504	Gate	6	EMO	HW	T(B)	Yes	Open	Open	Closed	As-Is	EP-B	40 Sec (Note 9)		
210	21	Inside		Bl Flng	-	-	-	-	-	Closed	Closed	Closed	Closed	-	-		
		Outside	-	Bl Flng	-	-	-	-	-	Closed	Closed	Closed	Closed	-	-		

TABLE 6.2-54 (Continued)

ISOLATION VALVE SUMMARY

Pene-tration No.	Valve Arrang Fig 6.2-52	Location Referred to RB	Valve No.	Valve Type	Valve Size (in.)	Primary Method of Actuation (Note 1)	Backup Method of Actuation (Note 1)	Signal (Note 2)	Position Indication (Note 3)	Normal Valve Position	Shutdown Valve Position (Note 4)	Post Accident Position	Power Failure Position	Power Source (Note 5)	Closure Time (Note 6)	Remarks
211	21	Inside Outside	- -	Bl Flng Bl Flng	- -	- -	- -	- -	-	Closed Closed	Closed Closed	Closed Closed	Closed Closed	- -	- -	
212	21	Inside Outside	- -	Bl Flng Bl Flng	- -	- -	- -	- -	-	Closed Closed	Closed Closed	Closed Closed	Closed Closed	- -	- -	
213	11D	Inside Outside Outside	- 1009C 1633C	- Spl Chk Stp Chk	- 4 1-1/2	- AO/FC EMO	- - HW	- J(A) P(A)	- Yes Yes	- Closed Open	- Closed Open	- Open Closed	- Closed As Is	- IA-A EP-A	- N/A 60 Sec (Note 9)	Closed system inside
214	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	Capped spare
215	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	Capped spare
216	21	Inside Outside	- -	Bl Flng Bl Flng	- -	- -	- -	- -	-	Closed Closed	Closed Closed	Closed Closed	Closed Closed	- -	- -	
217	14	Inside Outside	- -	- -	- -	- -	- -	- -	- -	- -	- -	- -	- -	- -	- -	Bellows inside Transmitter Outside
218	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	Capped spare
219	11C	Inside Outside	- 503C	- Gate	- 3	- AO/FC	- HW	- F(A/B)	- Yes	- Open	- Closed	- Closed	- Closed	- IA-A/B	- 40 Sec (Note 9)	Closed system inside Two solenoids in series
220	11C	Inside Outside	- 9398C	- Globe	- 1	- EO	- -	- F(B)	- Yes	- Open	- Open	- Closed	- Closed	- EP-B	- 40 Sec (Note 9)	Closed system inside
221	2	Inside Outside	8368C 8102C	Check Globe	1-1/2 1-1/2	- EMO	- HW	- J(B)	No Yes	- Open	- Open	- Open	- As Is	- EP-B	- N/A	
222	6	Inside Inside Inside Outside	8995A 8995B 8995C 8885	Check Check Check Gate	2 2 2 3	- - - EMO	- - - HW	- - - J(A)	No No No Yes	- - - Closed	- - - Closed	- - - Closed	- - - As Is	- - - EP-A	- - - 10 Sec	Open for post accident recirc phase

00-01

TABLE 6.2-54 (Continued)

ISOLATION VALVE SUMMARY

Pene-tration No.	Valve Arrang Fig 6.2-52	Location Referred to RB	Valve No.	Valve Type	Valve Size (in.)	Primary Method of Actuation (Note 1)	Backup Method of Actuation (Note 1)	Signal (Note 2)	Position Indication (Note 3)	Normal Valve Position	Shutdown Valve Position (Note 4)	Post Accident Position	Power Failure Position	Power Source (Note 5)	Closure Time (Note 6)	Remarks
223	1	Inside Outside	9364C 9365C	Globe Globe	3/8 3/8	EO EO	- -	T(A) T(B)	Yes Yes	Closed Closed	Closed Closed	Closed Closed	Closed Closed	EP-A EP-B	40 Sec (Note 9) 40 Sec (Note 9)	Intermittent operation post Accident Intermittent operation post Accident
224	11C	Inside Outside	- 503B	- Gate	- 3	- AO/FC	- HW	- F(A/B)	- Yes	- Open	- Closed	- Closed	- Closed	- IA-A/B	- 40 Sec (Note 9)	Closed system inside Two solenoids in series
225	11C	Inside Outside	- 9398B	- Globe	- 1	- EO	- -	- F(B)	- Yes	- Open	- Open	- Closed	- Closed	- EP-B	- 40 Sec (Note 9)	Closed system inside
226	3	Inside Outside	8701B -	Gate -	12 -	EMO -	HW -	J(B) -	Yes -	Closed -	Open -	Closed -	As Is -	EP-A -	120 Sec -	 Closed system outside
227	2	Inside Outside	8974B 8888B	Check Gate	10 10	- EMO	 HW	- J(B)	No Yes	- Open	- Open	- Open	- As Is	- EP-B	- 15 Sec	 Operated during post accident recirc phase
228	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	Capped spare
229	2	Inside Outside	8368B 8102B	Check Globe	1-1/2 1-1/2	- EMO	- HW	- J(B)	No Yes	- Open	- Open	- Open	- As Is	- EP-B	- N/A	
230A	25	Inside Outside	- -	- -		- -	- -	- -	- -	- -	- -	- -	- -	- -	- -	Bellows inside Transmitter Outside
230B	25	Inside Outside	- -	- -		- -	- -	- -	- -	- -	- -	- -	- -	- -	- -	Bellows inside Transmitter Outside
230C	25	Inside Outside	- -	- -		- -	- -	- -	- -	- -	- -	- -	- -	- -	- -	Bellows inside Transmitter Outside
230D	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	Capped 3/8 spare
230E	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	Capped 3/8 spare
231	17	Inside Outside	8768 8767	Diaph Diaph	1 1	HW HW	- -	- -	No No	Closed Closed	Closed Closed	Closed Closed	- -	- -	- -	Locked Closed Locked Closed
232	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	Capped Spare

99-01

TABLE 6.2-54 (Continued)

ISOLATION VALVE SUMMARY

Pene-tration No.	Valve Arrang Fig 6.2-52	Location Referred to RB	Valve No.	Valve Type	Valve Size (in.)	Primary Method of Actuation (Note 1)	Backup Method of Actuation (Note 1)	Signal (Note 2)	Position Indication (Note 3)	Normal Valve Position	Shutdown Valve Position (Note 4)	Post Accident Position	Power Failure Position	Power Source (Note 5)	Closure Time (Note 6)	Remarks	
301A	15	Inside Inside Outside	6051A 6051C 6053A	Solenoid Solenoid Solenoid	3/8 3/8 3/8	EO EO EO	- - -	- - -	Yes Yes Yes	Closed Closed Closed	Closed Closed Closed	(See Remarks) (See Remarks) (See Remarks)	Closed Closed Closed	EP-A EP-A EP-A	N/A (Note 11) N/A (Note 11) N/A (Note 11)	Post accident position is initially "closed." Valves may be opened during accident if H^2 sample/purge is required.	
301B	22	Inside Outside Outside	6050A 6052A 6054	Solenoid Solenoid Solenoid	3/8 3/8 3/8	EO EO EO	- - -	T(A) - T(B)	Yes Yes Yes	Open Closed Open	Closed Closed Closed	(See Remarks) (See Remarks) (See Remarks)	Closed Closed Closed	EP-A EP-A EP-B	40 Sec (Note 9) N/A (Note 11) 40 Sec (Note 9)	Post accident position is initially "closed." Valves may be opened during accident if H^2 sample/purge is required.	02-01
302	1	Inside Outside	6066 6067	Gate Gate	6 6	AO/FC AO/FC	- -	K(A) K(B)	Yes Yes	Closed Closed	Closed Closed	(See Remarks) (See Remarks)	Closed Closed	IA-A IA-B	5 Sec (Note 9) 5 Sec (Note 9)	Post accident position is initially "closed." Valves may be opened during accident if H^2 sample/purge is required.	
303	2	Inside Outside	3009B 3003B	Check Gate	10 10	- EMO	- HW	- T(B)	No Yes	- Closed	- Closed	- Open	- As Is	- EP-B	- 30 Sec (Note 13)	Signal opens valve	98-01
304	12	Inside Outside Outside	- 3106A 3110A	- Butterfly Butterfly	- 16 12	- EMO EMO	- HW HW	- I(A) I(A)	- Yes Yes	- Closed Open	- Closed Open	- Open Closed	- As Is As Is	- EP-A EP-A	- 45 Sec 70 Sec	Closed system inside	
305	12	Inside Outside	- 3103A	- Gate	- 16	- EMO	- HW	- J(A)	- Yes	- Open	- Open	- Open	- As Is	- EP-A	- 45 Sec	Closed system inside	
306	11B	Outside	11670	Gate	3	HW	-	-	No	Closed	Closed	Closed	-	-	-	Locked Closed	RN
306	11C	Inside Outside	- 1611A 1611A	- Gate Gate	- 18	- AO/AC	- -	- M(A) M(B)	- Yes Yes	- Open Open	- Closed Closed	- Closed Closed	- Closed As Is	- IA-A IA-B	- 5 Sec (Note 12)	Closed system inside Dual A train Solenoids B Train Solenoids	07-028 02-01
307	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	Not Used	99-01
308	11D	Inside Outside Outside	- 1633A 1009A	- Stp Chk Spl Chk	- 1-1/2 4	- EMO AO/FC	- HW -	- P(A) J(A)	- Yes Yes	- Open Closed	- Open Closed	- Closed Open	- As Is Closed	- EP-A IA-A	- 60 Sec (Note 9) N/A	Closed system inside	
309	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	Capped rad mon abandoned	
310	16	Inside Outside	2913 2912	Check Globe	2 2	- HW	-		No No	- Closed	- Closed	- Closed	- Closed	- -	- -	Locked closed	

TABLE 6.2-54 (Continued)

ISOLATION VALVE SUMMARY

Pene-tration No.	Valve Arrang Fig 6.2-52	Location Referred to RB	Valve No.	Valve Type	Valve Size (in.)	Primary Method of Actuation (Note 1)	Backup Method of Actuation (Note 1)	Signal (Note 2)	Position Indication (Note 3)	Normal Valve Position	Shutdown Valve Position (Note 4)	Post Accident Position	Power Failure Position	Power Source (Note 5)	Closure Time (Note 6)	Remarks
311	2	Inside	2661	Check	2	-	-		No	-	-	-	-	-	-	
		Outside	2660	Globe	2	AO/FC	-	T(A)	Yes	Open	Open	Closed	Open	IA-A	40 Sec (Note 9)	
312	2	Inside	9570	Check	8	-	-	-	No	-	-	-	-	-	-	
		Outside	9568	Gate	8	EMO	HW	P(A)	Yes	Open	Open	Closed	As-Is	EP-A	60 Sec (Note 9)	
313	16	Inside	6588	Check	1	-	-	-	No	-	-	-	-	-	-	
		Outside	6587	Globe	1	HW	-	-	No	Closed	Closed	Closed	Closed	-	-	Locked Closed
314	1	Inside	9364B	Globe	3/8	EO	-	T(A)	Yes	Closed	Closed	Closed	Closed	EP-A	40 Sec (Note 9)	Intermittent operation
		Outside	9365B	Globe	3/8	EO	-	T(B)	Yes	Closed	Closed	Closed	Closed	EP-B	40 Sec (Note 9)	Post Accident
315	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	Capped spare
316	3	Inside	8701A	Gate	12	EMO	HW	J(A)	Yes	Closed	Open	Closed	As Is	EP-A	120 Sec	
		Outside	-	-	-	-	-	-	-	-	-	-	-	-	-	Closed system outside
317	2	Inside	8861	Check	1	-	-	-	No	-	-	-	-	-	-	
		Outside	8860	Globe	1	AO/FC	-	T(B)	Yes	Closed	Closed	Closed	Closed	IA-B	40 Sec (Note 9)	
318	20	Inside	8149A	Globe	2	AO/FC	-	T(A)	Yes	Note 14	Note 14	Closed	Closed	IA-A	40 Sec (Note 9)	
		Inside	8149B	Globe	2	AO/FC	-	T(A)	Yes	Note 14	Note 14	Closed	Closed	IA-A	40 Sec (Note 9)	
		Inside	8149C	Globe	2	AO/FC	-	T(A)	Yes	Note 14	Note 14	Closed	Closed	IA-A	40 Sec (Note 9)	99-01
		Inside	8117	Relief	2	-	-	-	No	Closed	Closed	Closed	Closed	-	-	
		Outside	8152	Globe	3	AO/FC	-	T(B)	Yes	Open	Open	Closed	Closed	IA-B	40 Sec (Note 9)	
319	1	Inside	2662B	Globe	6	AO/FC	-	T(B)	Yes	Open	Open	Closed	Open	IA-B	40 Sec (Note 9)	
		Outside	2662A	Globe	6	AO/FC	-	T(A)	Yes	Open	Open	Closed	Open	IA-A	40 Sec (Note 9)	
320	2	Inside	8947	Check	1	-	-	-	No	-	-	-	-	-	-	
		Outside	8880	Globe	1	AO/FC	-	T(B)	Yes	Closed	Closed	Closed	Closed	IA-B	40 Sec (Note 9)	
321	1	Inside	8871	Globe	3/4	AO/FC	-	T(A)	Yes	Closed	Closed	Closed	Closed	IA-A	40 Sec (Note 9)	
		Outside	8961	Globe	3/4	AO/FC	-	T(B)	Yes	Closed	Closed	Closed	Closed	IA-B	40 Sec (Note 9)	

TABLE 6.2-54 (Continued)

ISOLATION VALVE SUMMARY

Pene-tration No.	Valve Arrang Fig 6.2-52	Location Referred to RB	Valve No.	Valve Type	Valve Size (in.)	Primary Method of Actuation (Note 1)	Backup Method of Actuation (Note 1)	Signal (Note 2)	Position Indication (Note 3)	Normal Valve Position	Shutdown Valve Position (Note 4)	Post Accident Position	Power Failure Position	Power Source (Note 5)	Closure Time (Note 6)	Remarks	
322	2	Inside	8974A	Check	10	-	-		No	-	-	-	-	-	-		
		Outside	8888A	Gate	10	EMO	HW	J(A)	Yes	Open	Open	Open	As Is	EP-A	15 Sec	Operated during post accident recirc phase	
323	8	Inside	-	-	-	-	-	-	-	-	-	-	-	-	-	Closed system inside	
		Outside	9387	Globe	3/8	EO	-	T(B)	Yes	Closed	Closed	Closed	Closed	EP-B	40 Sec (Note 9)		
324	17	Inside	2680	Globe	2	HW	-	-	No	Closed	Closed	Closed	-	-	-	Locked Closed	
		Outside	2679	Globe	2	HW	-	-	No	Closed	Closed	Closed	-	-	-	Locked Closed	
325	4	Inside	8988A	Check	6	-	-	-	No	-	-	-	-	-	-		
		Inside	8988B	Check	6	-	-	-	No	-	-	-	-	-	-		
		Outside	8889	Gate	10	EMO	HW	J(A)	Yes	Closed	Closed	Closed	As Is	EP-A	15 Sec	Open for post accident recirc phase	00-01
326	11C	Inside	-	-	-	-	-	-	-	-	-	-	-	-	-	Closed system inside	
		Outside	503A	Gate	3	AOFC	HW	F(A/B)	Yes	Open	Closed	Closed	Closed	IA-A/B	40 Sec (Note 9)	Two solenoids in series	
327	5	Outside	3004A	Gate	12	EMO	HW	J(A)	Yes	Closed	Closed	Closed	As Is	EP-A	60 Sec Max	Open for post accident recirc phase	RN 11-010
		Outside	-	-	-	-	-	-	-	-	-	-	-	-	-	Guard Pipe and chamber	
328	5	Outside	3004B	Gate	12	EMO	HW	J(B)	Yes	Closed	Closed	Closed	As Is	EP-B	60 Sec Max	Open for post accident recirc phase	RN 11-010
		Outside	-	-	-	-	-	-	-	-	-	-	-	-	-	Guard Pipe and chamber	
329	5	Outside	8811A	Gate	14	EMO	HW	A(A)	Yes	Closed	Closed	Closed	As Is	EP-A	20 Sec	Open for post accident recirc phase	
		Outside	-	-	-	-	-	-	-	-	-	-	-	-	-	Guard Pipe and chamber	
330	7	Inside	9605	Gate	8	EMO	HW	P(A)	Yes	Open	Open	Closed	As Is	EP-A	60 Sec (Note 9)		
		Inside	9689	Check	3/4	-	-	-	No	-	-	-	-	-	-		
		Outside	9606	Gate	8	EMO	HW	P(B)	Yes	Open	Open	Closed	As Is	EP-B	60 Sec (Note 9)		
401	2	Inside	3009A	Check	10	-	-	-	No	-	-	-	-	-	-		
		Outside	3003A	Gate	10	EMO	HW	T(A)	Yes	Closed	Closed	Open	As Is	EP-A	30 Sec (Note 13)	Signal opens valve	98-01

TABLE 6.2-54 (Continued)

ISOLATION VALVE SUMMARY

Pene-tration No.	Valve Arrang Fig 6.2-52	Location Referred to RB	Valve No.	Valve Type	Valve Size (in.)	Primary Method of Actuation (Note 1)	Backup Method of Actuation (Note 1)	Signal (Note 2)	Position Indication (Note 3)	Normal Valve Position	Shutdown Valve Position (Note 4)	Post Accident Position	Power Failure Position	Power Source (Note 5)	Closure Time (Note 6)	Remarks	
402	1	Inside	000-1B	Butterfly	36	AO/FC	-	H(B)	Yes	Closed	Closed	Closed	Closed	IA-B	3 Sec Max		RN 02-036
		Outside	000-1A	Butterfly	36	AO/FC	-	H(A)	Yes	Closed	Closed	Closed	Closed	IA-A	3 Sec Max		
403	12	Inside	-	-	-	-	-	-	-	-	-	-	-	-	-	Closed system inside	
		Outside	3106B	Butterfly	16	EMO	HW	I(B)	Yes	Closed	Closed	Open	As Is	EP-B	45 Sec		
		Outside	3110B	Butterfly	12	EMO	HW	I(B)	Yes	Open	Open	Closed	As Is	EP-B	70 Sec Max		
404	17	Inside	6773	Gate	4	HW	-	-	No	Closed	Closed	Closed	Closed	-	-	Locked closed	
		Outside	6772	Gate	4	HW	-	-	No	Closed	Closed	Closed	Closed	-	-	Locked closed	
405	15	Inside	9356A	Globe	3/8	EO	-	T(A)	Yes	Closed	Closed	Closed	Closed	EP-A	40 Sec (Note 9)	Intermittent operation post accident	98-01
		Inside	9356B	Globe	3/8	EO	-	T(A)	Yes	Closed	Closed	Closed	Closed	EP-A	40 Sec (Note 9)	Intermittent operation post accident	
		Outside	9357	Globe	3/8	EO	-	T(B)	Yes	Closed	Closed	Closed	Closed	EP-B	40 Sec (Note 9)	Intermittent operation post accident	
406A	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	Seal welded spare	
406B	24	Inside	-	-	-	-	-	-	-	-	-	-	-	-	-	Bellows inside	
		Outside	-	-	-	-	-	-	-	-	-	-	-	-	-	Transmitter outside	
407A	1	Outside	9311B	Ball	1	AO/FC	-	T(B)	Yes	Open	Open	Closed	Closed	IA-B	40 Sec (Note 9)	Intermittent operation post accident	
		Inside	9311A	Ball	1	AO/FC	-	T(A)	Yes	Open	Open	Closed	Closed	IA-A	40 Sec (Note 9)	Intermittent operation post accident	
407B	1	Outside	9312B	Ball	1	AO/FC	-	T(B)	Yes	Open	Open	Closed	Closed	IA-B	40 Sec (Note 9)	Intermittent operation post accident	
		Inside	9312A	Ball	1	AO/FC	-	T(A)	Yes	Open	Open	Closed	Closed	IA-A	40 Sec (Note 9)	Intermittent operation post accident	
408	2	Inside	8368A	Check	1-1/2	-	-	-	No	-	-	-	-	-	-		
		Outside	8102A	Globe	1-1/2	EMO	HW	J(B)	Yes	Open	Open	Open	As Is	EP-B	N/A		
409	2	Inside	8381	Check	3	-	-	-	No	-	-	-	-	-	-		RN 11-004
		Outside	8107	Gate	3	EMO	HW	S(A)	Yes	Open	Closed	Closed	As Is	EP-A	10 Sec (Note 10)		

TABLE 6.2-54 (Continued)

ISOLATION VALVE SUMMARY

Pene-tration No.	Valve Arrang Fig 6.2-52	Location Referred to RB	Valve No.	Valve Type	Valve Size (in.)	Primary Method of Actuation (Note 1)	Backup Method of Actuation (Note 1)	Signal (Note 2)	Position Indication (Note 3)	Normal Valve Position	Shutdown Valve Position (Note 4)	Post Accident Position	Power Failure Position	Power Source (Note 5)	Closure Time (Note 6)	Remarks
410	7	Inside Inside Outside	8112 8103 8100	Globe Check Globe	2 3/4 2	EMO - EMO	HW - HW	T(A) - T(B)	Yes No Yes	Open - Open	Open - Open	Closed - Closed	As Is - As Is	EP-A - EP-B	40 Sec (Note 9) - 40 Sec (Note 9)	
411	11C	Inside Outside	- 9398A	- Globe	- 1	- EO	- -	- F(B)	- Yes	- Open	- Open	- Closed	- Closed	- EP-B-	- 40 Sec (Note 9)	Closed system inside
412	6	Inside Inside Inside Outside	8990A 8990B 8990C 8886	Check Check Check Gate	2 2 2 3	- - - EMO	- - - HW	- - - J(B)	No No No Yes	- - - Closed	- - - Closed	- - - Closed	- - - As Is	- - - EP-B	- - - 10 Sec	 Open for post accident recirc phase
413	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	Capped spare
414	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	Capped spare
415	6	Inside Inside Inside Outside	8992A 8992B 8992C 8884	Check Check Check Gate	2 2 2 3	- - - EMO	- - - HW	- - - J(A)	No No No Yes	- - - Closed	- - - Closed	- - - Closed	- - - As Is	- - - EP-A	- - - 10 Sec	 Open for post accident recirc phase
416	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	Capped spare
417A	25	Inside Outside	- -	- -	- -	- -	- -	- -	- -	- -	- -	- -	- -	- -	- -	Bellows inside Transmitter outside
417B	25	Inside Outside	- -	- -	- -	- -	- -	- -	- -	- -	- -	- -	- -	- -	- -	Bellows inside Transmitter outside
417C	25	Inside Outside	- -	- -	- -	- -	- -	- -	- -	- -	- -	- -	- -	- -	-	Bellows inside Transmitter outside
417D	1	Inside Outside	9339 9341	Globe Globe	3/8 3/8	EO EO	- -	T(B) T(A)	Yes Yes	Closed Closed	Closed Closed	Closed Closed	Closed Closed	EP-B EP-A	40 Sec (Note 9) 40 Sec (Note 9)	
417E	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	Capped 3/8 spare

TABLE 6.2-54 (Continued)

ISOLATION VALVE SUMMARY

Pene-tration No.	Valve Arrang Fig 6.2-52	Location Referred to RB	Valve No.	Valve Type	Valve Size (in.)	Primary Method of Actuation (Note 1)	Backup Method of Actuation (Note 1)	Signal (Note 2)	Position Indication (Note 3)	Normal Valve Position	Shutdown Valve Position (Note 4)	Post Accident Position	Power Failure Position	Power Source (Note 5)	Closure Time (Note 6)	Remarks
418	1	Inside	7126	Diaph	3/4	AO/FC	-	T(A)	Yes	Open	Open	Closed	Closed	IA-A	40 Sec (Note 9)	
		Outside	7150	Diaph	3/4	AO/FC	-	T(B)	Yes	Open	Open	Closed	Closed	IA-B	40 Sec (Note 9)	
419	17	Inside	6671	Diaph	3	HW	-	-	No	Closed	Closed	Closed	-	-	-	Locked Closed
		Outside	6672	Diaph	3	HW	-	-	No	Closed	Closed	Closed	-	-	-	Locked Closed
420	1	Inside	8047	Diaph	1	AO/FC	-	T(A)	Yes	Open	Open	Closed	Closed	IA-A	40 Sec(Note 9)	
		Outside	8033	Diaph	1	AO/FC	-	T(B)	Yes	Open	Open	Closed	Closed	IA-B	40 Sec (Note 9)	
421	17	Inside	6698	Diaph	3	HW	-	-	No	Closed	Closed	Closed	-		-	Locked Closed
		Outside	6697	Diaph	3	HW	-	-	No	Closed	Closed	Closed	-		-	Locked Closed
422	2	Inside	8046	Check	3	-	-	-	No	-	-	-	-	-	--	
		Outside	8028	Diaph	3	AO/FC	-	T(B)	Yes	Closed	Closed	Closed	Closed	IA-B	40 Sec (Note 9)	
423	1	Inside	7170	Diaph	3	AO/FC	-	T(A)	Yes	Open	Open	Closed	Closed	IA-A	40 Sec (Note 9)	
		Outside	7136	Diaph	3	AO/FC	-	T(B)	Yes	Open	Open	Closed	Closed	IA-B	40 Sec (Note 9)	
424	1	Inside	6242A	Diaph	3	AO/FC	-	T(A)	Yes	Closed	Closed	Closed	Closed	IA-A	40 Sec (Note 9)	
		Outside	6242B	Diaph	3	AO/FC	-	T(B)	Yes	Closed	Closed	Closed	Closed	IA-B	40 Sec (Note 9)	
425	5	Outside	8811B	Gate	14	EMO	HW	A(B)	Yes	Closed	Closed	Closed	As Is	EP-B	20 Sec	Open for post accident recirc phase
		Outside	-	-	-	-	-	-	-	-	-	-	-	-	-	Guard Pipe and chamber
426	18	Inside	8997A	Check	2	-	-	-	No	-	-	-	-	-	-	
		Inside	8997B	Check	2	-	-	-	No	-	-	-	-	-	-	
		Inside	8997C	Check	2	-	-	-	No	-	-	-	-	-	-	
		Outside	8801A	Gate	3	EMO	HW	S(A)	Yes	Closed	Closed	Open	As Is	EP-A	10 Sec (Note 10)	Signal opens valve, Operated during Postaccident Recirc Phase
		Outside	8801B	Gate	3	EMO	HW	S(B)	Yes	Closed	Closed	Open	As Is	EP-B	10 Sec (Note 10)	Signal opens valve, Operated during Postaccident Recirc Phase
427	2	Inside	6799	Check	4	-	-	-	No	-	-	-	-	-	-	
		Outside	6797	Gate	4	EMO	HW	T(B)	Yes	Open	Open	Closed	As Is	EP-B	40 Sec (Note 9)	

00-01

TABLE 6.2-54 (Continued)

ISOLATION VALVE SUMMARY

Pene-tration No.	Valve Arrang Fig 6.2-52	Location Referred to RB	Valve No.	Valve Type	Valve Size (in.)	Primary Method of Actuation (Note 1)	Backup Method of Actuation (Note 1)	Signal (Note 2)	Position Indication (Note 3)	Normal Valve Position	Shutdown Valve Position (Note 4)	Post Accident Position	Power Failure Position	Power Source (Note 5)	Closure Time (Note 6)	Remarks
428	11A	Inside	-	-	-	-	-	-	-	-	-	-	-	-	-	Closed system
		Outside	2877A	Globe	1-1/2	AO/FC	HW	J(A/B)	Yes	Open	Closed	Open	Closed	IA-A/B	N/A	Dual solenoids
		Outside	2000	Globe	8	AO/FC	HW	J(A/B)	Yes	Closed	Closed	Closed	Closed (Note 8)	IA-A/B	N/A	Six solenoids
		Outside	2843A	Globe	1-1/2	AO/FC	HW	J(A/B)	Yes	Open	Closed	Open	Closed	IA-A/B	N/A	Dual solenoids
		Outside	2869A	Globe	4	AO/FE	-	C(A/B)	Yes	Closed	Closed	Closed	Closed	IA-A/B	10 Sec	Dual solenoids
		Outside	2801A	Globe	32	AO/FC	-	C(A/B)	Yes	Open	Closed	Closed	Closed	IA-A/B	7 Sec	Dual solenoids
		Outside	2806A, B, C, D, E	Safety	6X10	PO/FC	-	-	No	Closed	Closed	Closed	Closed	-	-	Five valves
500	14	Inside	-	-	-	-	-	-	-	-	-	-	-	-	-	Bellows inside
		Outside	-	-	-	-	-	-	-	-	-	-	-	-	-	Transmitter outside
505	21	Inside	-	Bl Flng	-	-	-	-	-	-	-	-	-	-	-	Outage penetration
		Outside	-	Bl Flng	-	-	-	-	-	-	-	-	-	-	-	
600	21	Inside	-	Bl Flng	-	-	-	-	-	-	-	-	-	-	-	Outage penetration
		Outside	-	Bl Flng	-	-	-	-	-	-	-	-	-	-	-	
602	21	Inside	-	Bl Flng	-	-	-	-	-	-	-	-	-	-	-	Outage penetration
		Outside	-	Bl Flng	-	-	-	-	-	-	-	-	-	-	-	
703	24	Inside	-	-	-	-	-	-	-	-	-	-	-	-	-	Bellows inside
		Outside	-	-	-	-	-	-	-	-	-	-	-	-	-	Transmitter outside

00-01

98-01

TABLE 6.2-54 (Continued)

ISOLATION VALVE SUMMARY

NOTES TO TABLE 6.2-54

General - This table does not include small manual valves on test connections or test vents used for Type C leak tests.

1. Method of Actuation

EMO - Electric motor operated

AO/FC - Air pressure to open; spring return, fail close on loss of air

AO/FO - Air pressure to close; spring return, fail open on loss of air

PO/FC - Process pressure to open; spring return, fail close on loss of process pressure 02-01

HW - Hand wheel, manual operation

EO - Energize solenoid coil to open

AO/AC - Air to open; and air to close, fail closed on loss of electric signal 02-01

2. Actuation Signal and Channel

S - Safety Injection Signal

(1) Low pressurizer pressure (2/3), or

(2) High containment pressure (2/3), or

(3) Differential steam line pressure (2/3 on two loops), or

(4) Low steam line pressure (2/3), or

(5) Manual (1/2).

T - Containment Isolation Signal (Phase A)

(1) Safety injection signal or

(2) Manual (1/2)

TABLE 6.2-54 (Continued)

ISOLATION VALVE SUMMARY

NOTES TO TABLE 6.2-54 (Continued)

C - Main Steam Isolation Signal

(1) High reactor building pressure (2/3) (note, different from S(2), above), or

(2) High steam flow (2/3) in coincidence with or low-low T_{avg} (2/3), or

(3) Low Steam pressure (2/3), or

(4) Manual (1/2)

D - Feedwater Isolation Signal

(1) Safety injection signal, or

(2) Steam generator "hi-hi" level (2/3), or

(3) Low T_{avg} coincident with reactor trip

P - Containment Isolation (Phase B)

(1) High-high reactor building pressure (2/4), or

(2) Manual (2/2)

F - Steam Generator Blowdown Isolation Signal

(1) Containment isolation (Phase A), or

(2) Start of emergency feedwater system.

G - Valve opens on pump start signal (ESF Loading sequence)

H - Reactor Building purge supply and exhaust isolation

(1) Containment ventilation isolation signal, or

(2) High radiation in reactor building purge exhaust vent (A train valves) or containment atmosphere high radiation (B train valves), or

(3) Manual

TABLE 6.2-54 (Continued)

ISOLATION VALVE SUMMARY

NOTES TO TABLE 6.2-54 (Continued)

I - Engineered safety features load sequencer (ESFLS)

(1) Safety injection signal, or
(2) Blackout, or
(3) Manual

J - Remote Manual

K - Reactor Building 6 inch purge supply and exhaust vent

(1) Containment isolation (Phase A), or
(2) High radiation in reactor building purge exhaust vent (A train valves) or containment atmosphere high radiation (B train valves), or
(3) Manual

L - (Intentionally left Blank by Amendment 96-03).

M - Feedwater isolation valve isolation signal

(1) Feedwater isolation (comprised of D above, Train A only), or
(2) (Intentionally left blank by Amendment 02-01) 02-01
(3) (Intentionally left blank by Amendment 96-03)
(4) High-high intermediate sump level (2 of 3), or
(5) (Intentionally left blank by Amendment 02-01) 02-01
(6) Manual

A - Automatic actuation from refueling water storage tank Lo-Lo level coincident with safety injection signal.

CHANNEL

(A) Signal is from the A channel of the actuation system. 99-01
(B) Signal is from the B channel of the actuation system
(A/B) Signal is from both channels of the actuation system (e.g., dual solenoid valves for an air operated valve - one from channel A, the other from channel B)

TABLE 6.2-54 (Continued)

ISOLATION VALVE SUMMARY

NOTES TO TABLE 6.2-54 (Continued)

3. "Yes" indicates remote position indication

4. "Shutdown" means normal shutdown on residual heat removal system. It does not include testing or refueling operations.

5. Power Source:

   IA-A - Valve is operated by the nonsafety-related instrument air system. The associated solenoid valve is supplied from safety-related battery A.

   IA-B - Valve is operated by the nonsafety-related instrument air system. The associated solenoid valve is supplied from safety-related battery B.

   EP-A - Valve is supplied from the safety-related emergency power bus A.

   EP-B - Valve is supplied from the safety-related emergency power bus B.

6. Closure time is only for electric motor operated valves and air operated valves larger than 2-1/2 inches. Stroke times listed are not necessarily representative of actual stroke times. Tabulated stroke times reflect Technical Specifications, accident analysis assumptions, and design specified values.

7. Valve closure times are verified during preoperational testing.

8. Valve is controlled by non-class 1E control signal or loss of Class 1E power to the Train A and B Solenoid valves.

9. This is the maximum closure time originally supplied by Technical Specification, Table 3.6-1, for satisfying system isolation requirements. Tech. Spec. Table 3.6-1 has since been deleted by Amendment No. 110 in accordance with Generic Letter 91-08.

10. This is maximum stroke time assumed in Accident Analysis.

11. Technical Specification 3.6.1 listed NA for maximum stroke time of these valves. Tech. Spec. Table 3.6.1 has since been deleted by Amendment No. 110 in accordance with Generic Letter 91-08.

TABLE 6.2-54 (Continued)

ISOLATION VALVE SUMMARY

NOTES TO TABLE 6.2-54 (Continued)

12. For more information about FIV stroke time see FSAR section 6.2.1.3.10.3.1 and Technical Specification 4.7.1.6 and Bases.

13. The stoke time for XVG3003A/B-SP is greater than 30 seconds as a result of re-gearing the actuators per MRF-21745 (40 seconds). This increased stroke time is acceptable on the basis that 100% flow is achieved at 75% of the valve's stroke time. Therefore the 30 seconds required stroke is maintained (.75 x 40 second = 30 seconds) 98-01

14. During normal or shutdown operation, either one letdown isolation valve (8149B or C) or any combination of two letdown isolation valves inside containment may be open. During shutdown operation, all three letdown isolation valves may be open when RCS pressure is less than normal. 99-01

TABLE 6.2-54a

REACTOR BUILDING ELECTRICAL PENETRATIONS

Penetration Tag Number	Nozzle Number and Size	Location		Type of Service	Penetration Designation	Voltage Rating	Safety Class	Separation Group and Tray Class	Wiring Drawing Reference	Remarks	Flange Installation	
		Azimuth	Elevation									
XRP0001	504-18”	14°-10’	441’-0”	Med Volt Pwr	Reactor Coolant Pump “C”	15KV	2a/NNS	X-2	B-210-304	Junction Box Both Sides	Outside	
XRP0003	601-18”	143°-50’	441’-0”	Med Volt Pwr	Reactor Coolant Pump “B”	15KV	2a/NNS	X-2	B-210-305	Junction Box Both Sides	Outside	
XRP0006	710-18”	267°-10’	441’-0”	Med Volt Pwr	Reactor Coolant Pump “A”	15KV	2a/NNS	X-2	B-210-317	Junction Box Both Sides	Inside	
XRP0007	501-12”	14°-10’	473’-0”	Low Volt Pwr	Reactor Building Cooling Unit Fan (Train “B”)	1000V	2a/1e	B-4	B-210-302		Outside	
XRP0010	726-12”	245°-10’	473’-0”	Low Volt Pwr	Reactor Building Cooling Unit Fan (Train “A”)	1000V	2a/1e	A-4	B-210-329		Outside	
XRP0011	727-12”	248°-50’	473’-0”	Low Volt Pwr	Reactor Building Cooling Unit Fan (Train “A”)	1000V	2a/1e	A-4	B-210-330		Outside	
XRP0014	808-12”	337°-30’	473’-0”	Low Volt Pwr	Reactor Building Cooling Unit Fan (Train “B”)	1000V	2a/1e	A-4	B-210-333		Outside	
XRP0015	805-12”	289°-10’	469’-0”	Low Volt Pwr	Reactor Building Polar Crane	1000V	2a/NNS	X-4	B-210-332		Outside	
XRP0016	503-12”	17°-50’	469’-0”	Low Volt Pwr	Miscellaneous 480V Power	1000V	2a/NNS	X-5	B-210-303		Outside	
XRP0017	704-12”	263°-30’	469’-0”	Control/ Low Volt Pwr	Control Train "A"	1000V	2a/1e	A-6	B-210-313	Note 6 (inside) FCR B:6668E	Outside	
XRP0018	722-12”	259°-50’	469’-0”	Low Volt Pwr	Miscellaneous 480V Power	1000V	2a/NNS	X-5	B-210-326		Outside	
XRP0019	717-12”	226°-50’	441’-0”	Low Volt Pwr	Miscellaneous 480V Power	1000V	2a/NNS	X-5	B-210-323		Inside	
XRP0020	809-12”	274°-30’	441’-0”	Low Volt Pwr	Pressurizer Heaters (Control Group)	1000V	2a/NNS	X-4	B-210-334		Outside	
XRP0021	810-12”	278°-10’	441’-0”	Low Volt Pwr	Pressurizer Heaters (Backup Group II)	1000V	2a/1e	X-4	B-210-335		Outside	02-01
XRP0022	815-12”	270°-50’	441’-0”	Low Volt Pwr	Pressurizer Heaters (Backup Group I)	1000V	2a/1e	X-4	B-210-338		Outside	
XRP0023	700-12”	252°-30’	473’-0”	Low Volt Pwr	Control Rod Drive Power	1000V	2a/NNS	X-5	B-210-310		Outside	
XRP0024	701-12”	256°-10’	473’-0”	Low Volt Pwr	Control Rod Drive Power	1000V	2a/NNS	X-5	B-210-311		Outside	00-01
XRP0025	707-12”	252°-30’	469’-0”	Low Volt Pwr	Control Rod Drive Power	1000V	2a/NNS	X-5	B-210-315		Outside	
XRP0026	708-12”	256°-10’	469’-0”	Low Volt Pwr	Control Rod Drive Power	1000V	2a/NNS	X-5	B-210-316		Outside	
XRP0028	705-12”	241°-30’	469’-0”	Control/ Low Volt Pwr	Control Train “A”/Channel I	1000V	2a/1e	A-6	B-210-314	Note 6 (inside) FCR B:6616E	Outside	
XRP0030	723-12”	223°-10’	425’-0”	Control/ Low Volt Pwr	Control Train “B”/Channel III	1000V	2a/1e	B-6	B-210-327	Note 6 (inside) FCR B:6670E	Outside	
XRP0031	712-12”	215°-50’	441’-0”	Control/ Low Volt Pwr	Moveable Incore Detector Drive Control Power Circuits	1000V	2a/NNS	X-6	B-210-319	Note 6 (outside) NNSRCR 1E-1703	Inside	
XRP0032	714-12”	223°-10’	441’-0”	Instrument	Incore Detector Position Indication	-	2a/NNS	X-9	B-210-320	Note 6 (outside) NNSRCR 1E-1704	Inside	
XRP0033	725-12”	241°-30’	441’-0”	Control	Miscellaneous Control	1000V	2a/NNS	X-6	B-210-328	Note 6 (outside) NNSRCR 1E-1701	Inside	
XRP0034	812-12”	344°-50’	449'-0"	Control	Miscellaneous Control	1000V	2a/NNS	X-6	B-210-336	Note 6 (inside) NNSRCR 1E-1605	Outside	
XRP0035	800-12”	270°-50’	469'-0”	Instrument	Process (Control) Instrumentation Group 2 (D)	-	2a/1e	D-9	B-210-331	Note 6 (inside) FCR B:6741E	Outside	
XRP0036	715-12”	215°-50’	425’-0”	Instrument	Process (Control) Instrumentation Group 4 (E)	-	2a/1e	E-9	B-210-321	Note 6 (inside) FCR B:6893E	Outside	
XRP0037	711-12”	212°-10’	441’-0”	Instrument	Control Rod Position Indication and Incore Temp THCPLES	-	2a/NNS	X-9	B-210-318	Note 6 (outside) NNSRCR 1E-1706	Inside	
XRP0039	716-12”	230°-30’	441’-0”	Instrument	Miscellaneous Instrumentation	-	2a/NNS	X-9	B-210-322	Note 6 (outside) NNSRCR 1E-1706	Inside	
XRP0040	718-12”	234°-10’	441’-0”	Instrument	Miscellaneous Instrumentation	-	2a/NNS	X-9	B-210-324	Note 6 (outside) NNSRCR 1E-1709	Inside	
XRP0042	607-12”	136°-30’	429’-0”	Instrument	Miscellaneous Instrumentation	-	2a/NNS	X-9	B-210-309	Note 6 (Inside) NNSRCR 1E-1702	Outside	
XRP0043	605-12”	125°-30’	425’-0”	Instrument	Nuclear and Protection Instrumentation Channel I	-	2a/1e	A-8&10	B-210-307	Junction Box Both Sides	Outside	
XRP0044	606-12”	169°-30’	425’-0”	Instrument	Nuclear and Protection Instrumentation Channel III	-	2a/1e	B-8&10	B-210-308	Junction Box Both Sides	Outside	
XRP0045	721-12”	212°-10’	425’-0”	Instrument	Nuclear and Protection Instrumentation Channel II	-	2a/1e	D-8&10	B-210-325	Junction Box Both Sides	Outside	
XRP0046	814-12”	289°-10’	425’-0”	Instrument	Nuclear and Protection Instrumentation Channel IV	-	2a/1e	E-8&10	B-210-337	Junction Box Both Sides	Outside	
XRP0048	604-12”	169°-30’	429’-0”	Instrument	Process (Control) INST Group 3/Train “B” INST	-	2a/1e	B-9	B-210-306	Note 6 (inside) FCR B-7000E	Outside	

TABLE 6.2-54a (Continued)

REACTOR BUILDING ELECTRICAL PENETRATIONS

Penetration Tag Number	Nozzle Number and Size	Location		Type of Service	Penetration Designation	Voltage Rating	Safety Class	Separation Group and Tray Class	Wiring Drawing Reference	Remarks	Flange Installation	
XRP0051	702-12”	237°-50’	469’-0”	Instrument	Process (Control) INST Group 1/Train “A” INST	-	2a/1e	A-9	B-210-312	Note 6 (inside) FCR B-6630E	Outside	
XRP0052	502-12”	14°-10’	469’-0”	Low Volt Pwr	Miscellaneous 480V Power	1000V	2a/NNS	X-5	B-210-339		Outside	
XRP0053	802-12”	278°-10’	469’-0”	Low Volt Pwr	Miscellaneous 480V Power	1000V	2a/NNS	X-5	B-210-342		Outside	
XRP0054	706-12”	245°-10’	469’-0”	Low Volt Pwr	Control Rod Drive Power	1000V	2a/NNS	X-5	B-210-340		Outside	
XRP0055	728-12”	248°-50’	469’-0”	Low Volt Pwr	Control Rod Drive Power	1000V	2a/NNS	X-5	B-210-341		Outside	
XRP0056	803-12”	281°-50’	469’-0”	Low Volt Pwr	Control Rod Drive Power	1000V	2a/NNS	X-5	B-210-343		Outside	
XRP0057	804-12”	285°-30’	469’-0”	Low Volt Pwr	Control Rod Drive Power	1000V	2a/NNS	X-5	B-210-344		Outside	
	603-12”	125°-30’	429’-0”		Spare Nozzle							
	709-18”	263°-30’	441’-0”		Spare Nozzle							02-01
	713-12”	219°-30’	441’-0”		Spare Nozzle							
	719-12”	237°-50’	441’-0”		Spare Nozzle							02-01
XRP0104	720-12”	212°-10’	429’-0”	Instrument	Instrument Group 3/Train “B” Instrument	-	2a/1e	B-9	B-210-346		Outside	
	724-12”	226°-50’	425’-0”		Spare Nozzle							
	801-12”	274°-30’	469’-0”		Spare Nozzle							00-01
XRP0106	806-12”	330°-10’	469’-0”	Instrument	Instrument Group 1/Train “A” Instrument	-	2a/1e	A-9	B-210-345		Outside	
	809-12”	337°-30’	469’-0”		Spare Nozzle							
	811-12”	285°-30’	425’-0”		Spare Nozzle							
	813-12”	289°-10’	429’-0”		Spare Nozzle							
XRP100A-ES		Aux Bldg	397’-0”		Penetration Junction Box for XVG3004A-SP			A	B-210-361	1-18C-#4 AWG Module		
XRP100B-ES		Aux Bldg	397’-0”		Penetration Junction Box for XVG3004B-SP			B	B-210-362	1-18C-#4 AWG Module		
XRP101A-ES		Aux Bldg	397’-0”		Penetration Junction Box for XVG8811A-SI			A	B-210-363	1-18C-#4 AWG Module		
XRP101B-ES		Aux Bldg	397’-0”		Penetration Junction Box for XVG8811B-SI			B	B-210-364	1-18C-#4 AWG Module		

Notes:

1. Cable Bill of Material is the same both inside and outside containment, unless otherwise noted.
2. Field to install spare pigtails (through nozzle side only) and tag with pin and plug assignment. (See Note 9)
3. For convenience spare pigtail cable B/M number is called out in B/M number column of field cable. Should spares be used in the future, information will be added.
4. Spare plug pin position on junction box side of penetration flange to have blinding pins per vendor instructions. (See Note 9)
5. The individual conductor’s coming from the plug shall be identified by wire marks as to which pin in the plug it comes from. When making the splice in the splice box, the field cable shall be spliced to the plug cable using the field cable term sheet to show which pin (plug conductor) it goes to.
6. Field installed junction box. 02-01
7. Spare cables should be coiled and put in a field mounted junction box or a suitable cable tray, on XRP0043,44,45, and 46. The spare cable should be coiled in the penetration junction box.
8. All spare cables to use binding pins as specified c32c2103-pin #. (See Note 9)
9. As required to facilitate construction/maintenance, the field shall replace all blinding pins on the junction box side. At the option of SCE&G or whenever the contractor enters a j-box side plug, all spare pins shall be wired out with 25’ long pigtails of the type cable specified. Tag all spare pigtails, per Note 5.

TABLE 6.2-55

ELECTRIC HYDROGEN RECOMBINER TYPICAL PARAMETERS

Power (Maximum), kW	75
Capacity (Minimum), scfm	100
Heaters	
Number	5
Heater Surface Area/Heater, ft^2	35
Maximum Heat Flux, BTU/hr-ft^2	2850
Maximum Sheath Temperature, °F	1550
Gas Temperature	
Inlet, °F	80 to 155
In Heater Section, °F	1150 to 1400
Materials	
Outer Structure	300-Series Stainless Steel
Inner Structure	Inconel-600
Heater Element Sheath	Incoloy-800

TABLE 6.2-60

PASSIVE HEAT SINK DATA FOR
MINIMUM POST LOCA CONTAINMENT PRESSURE

RN 06-040

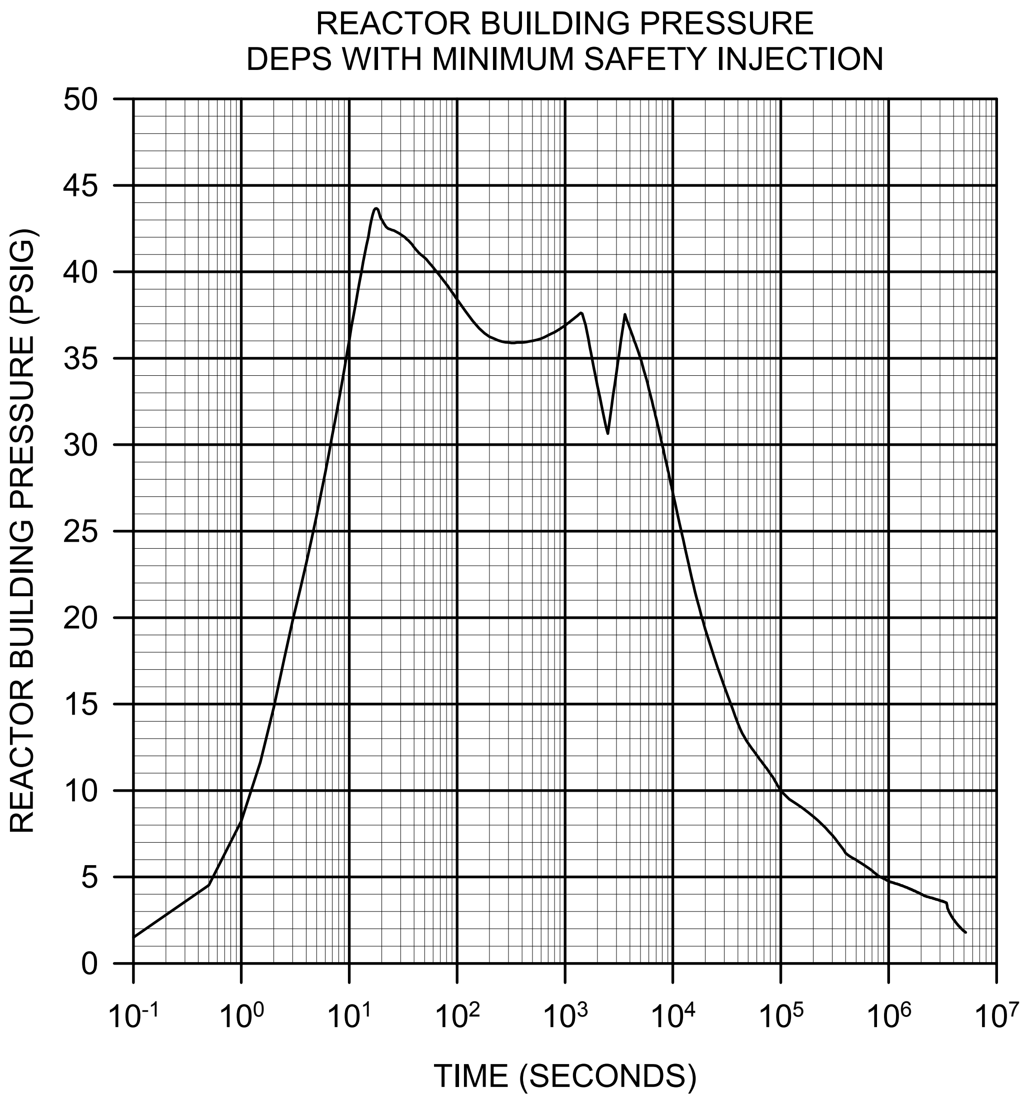
STRUCTURAL HEAT SINKS

Thickness (in)	Area (ft^2)
	57,397
0.25 Carbon Steel, 48.0 Concrete	
0.264 Carbon Steel, 36.0 Concrete	20,241
24.0 Concrete, 0.125 Carbon Steel	11,694
18.0 Concrete	315
21.96 Concrete	43,537
12.0 Concrete	10,811
48.0 Concrete	19,020
0.06 Stainless Steel	91,038
0.06 Carbon Steel	1,187,924
1.512 Stainless Steel	454
1.128 Stainless Steel	606
0.6 Stainless Steel	924
0.336 Stainless Steel	2,194
6.672 Carbon Steel	3,300
3.504 Carbon Steel	130
1.9 Carbon Steel	4,647
0.87 Carbon Steel	8,787
0.672 Carbon Steel	26,393
0.504 Carbon Steel	5,873
0.324 Carbon Steel	7,793

02-01

THERMOPHYSICAL PROPERTIES

	Volumetric Heat Capacity BTU/ft^3-°F	Thermal Conductivity BTU/hr-ft-°F
Concrete	22.5	1.0
Carbon Steel	55.4	36.0
Stainless Steel	55.1	9.1



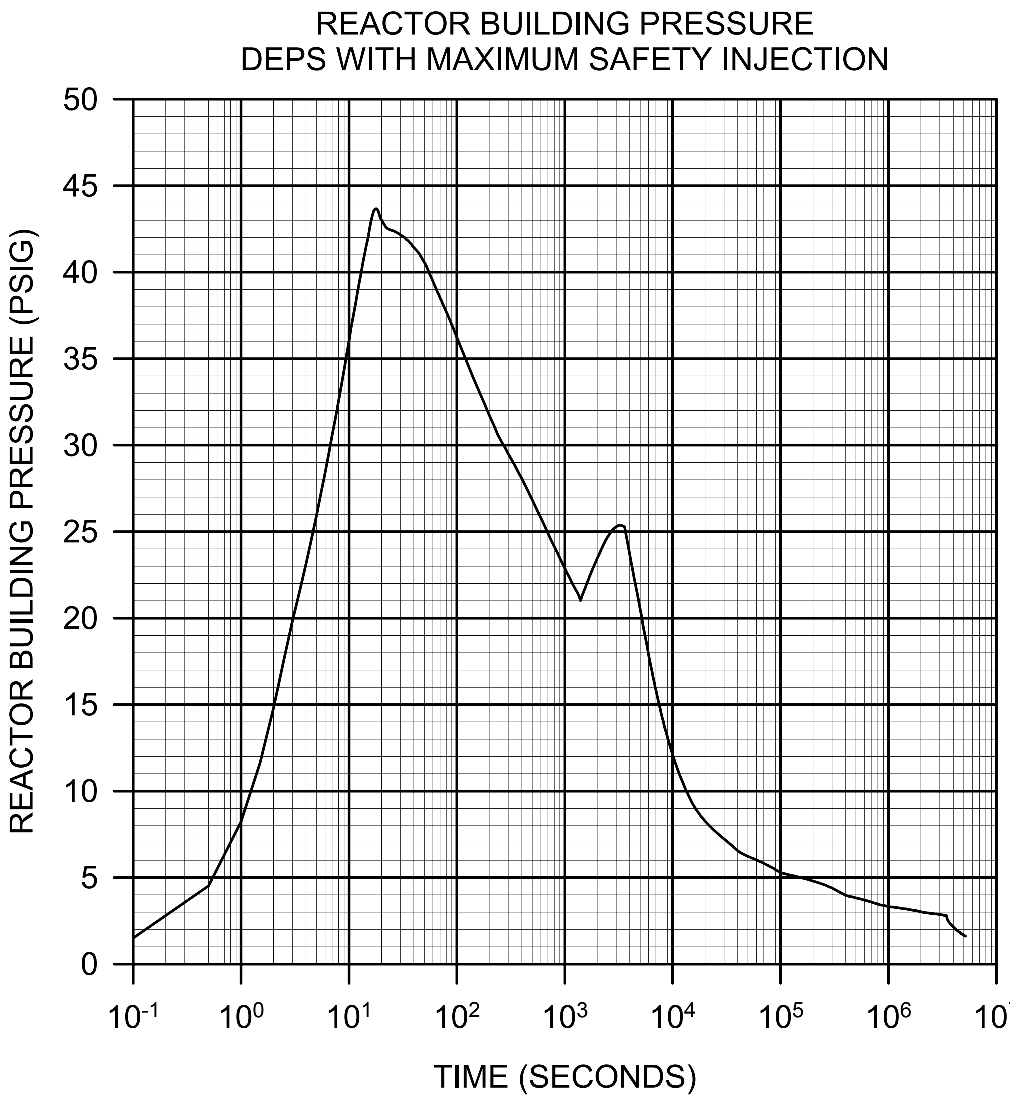
RN
03-003

RN 03-003
June 2003

SOUTH CAROLINA ELECTRIC & GAS CO
VIRGIL C. SUMMER NUCLEAR STATION

Double Ended Pump Suction Break
Minimum Safety Injection
Reactor Building Pressure vs. Time

Figure 6.2-1



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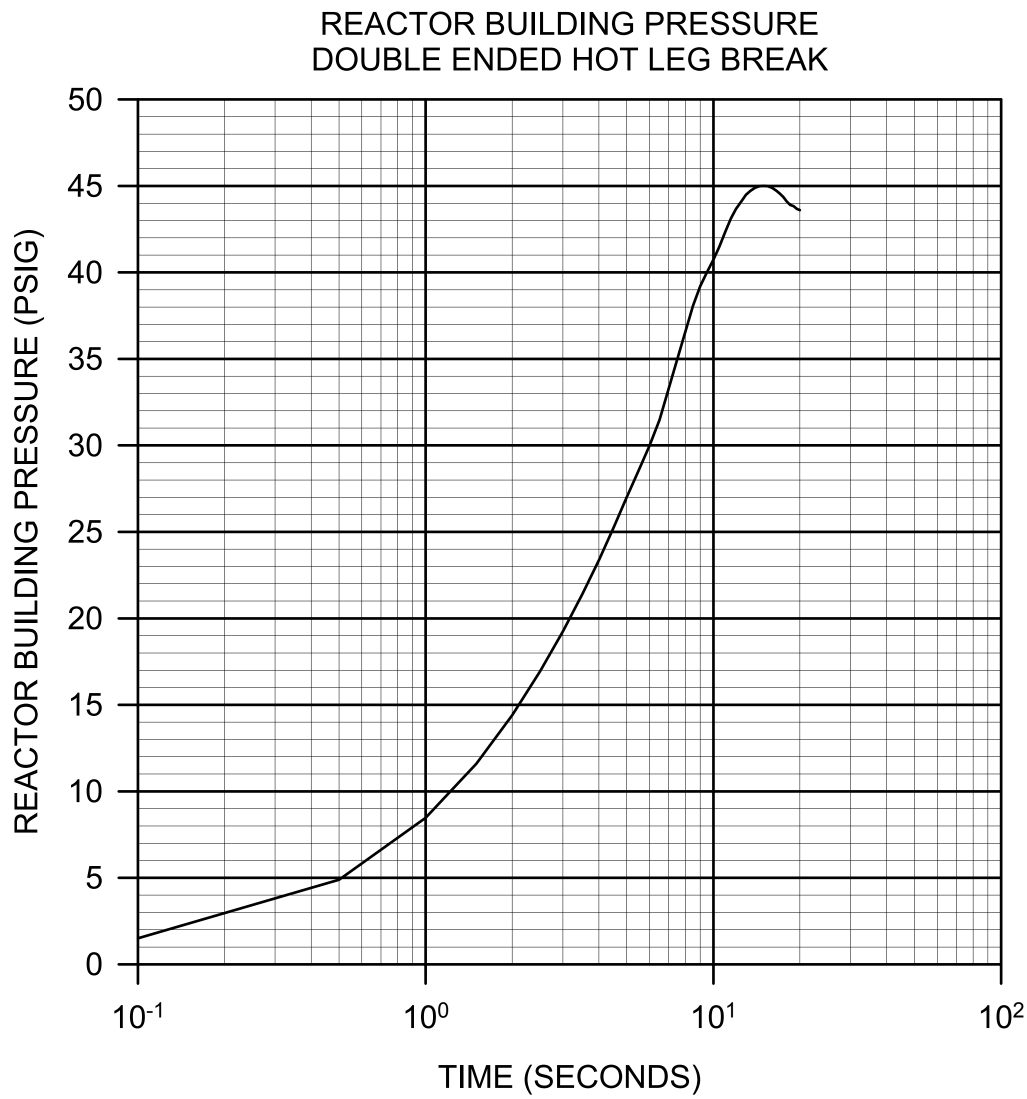
SOUTH CAROLINA ELECTRIC & GAS CO
VIRGIL C. SUMMER NUCLEAR STATION

Double Ended Pump Suction Break
Maximum Safety Injection
Reactor Building Pressure vs. Time

Figure 6.2-2



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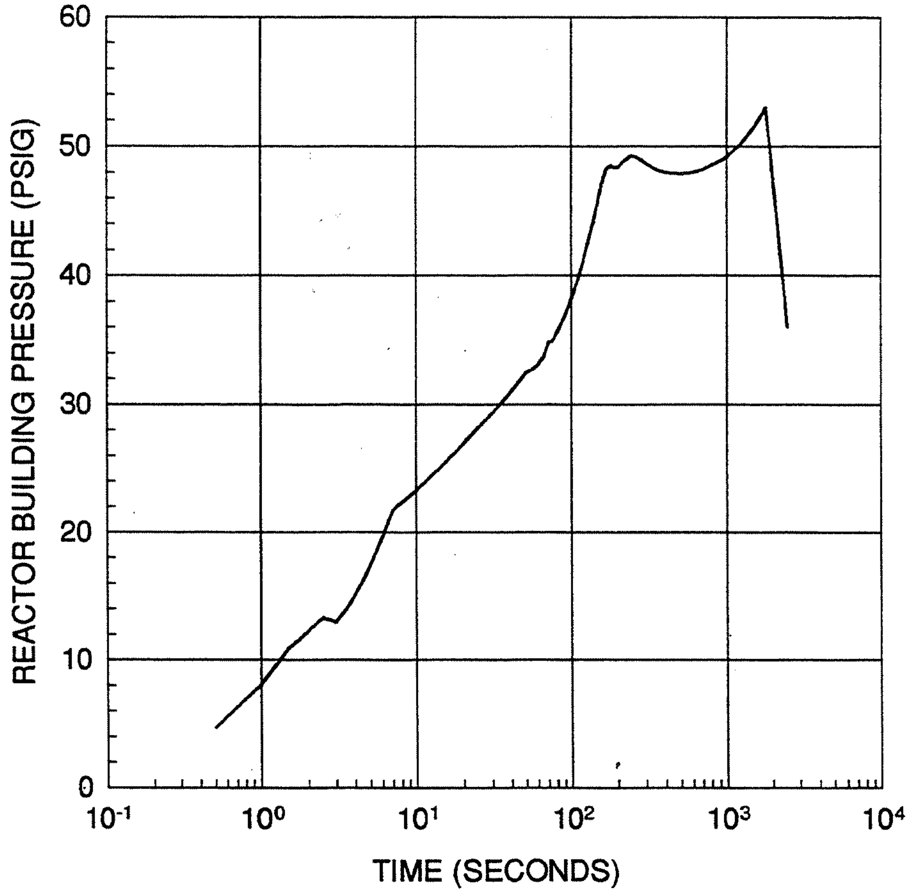
SOUTH CAROLINA ELECTRIC & GAS CO
VIRGIL C. SUMMER NUCLEAR STATION

Double Ended Hot Leg Break
Reactor Building Pressure vs. Time

Figure 6.2-3

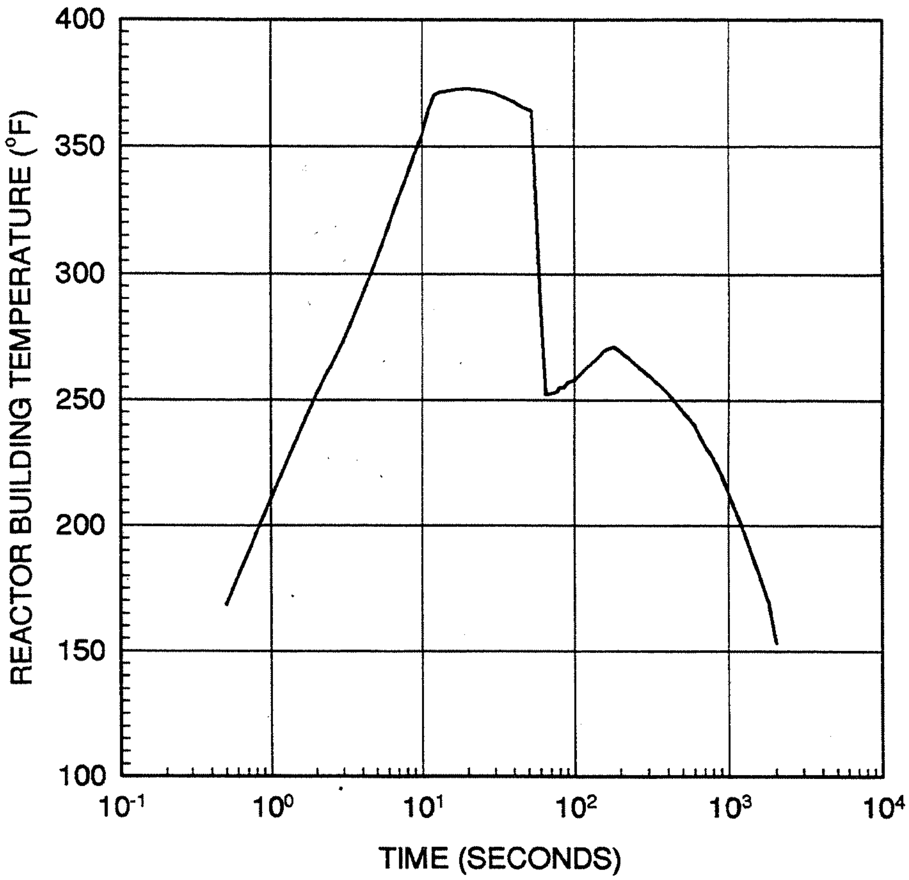


SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Main Steam Line Break
(Design Basis Accident)
1.4 ft^2 DER at 25% Power
Reactor Building Pressure Vs. Time

Figure 6.2-4

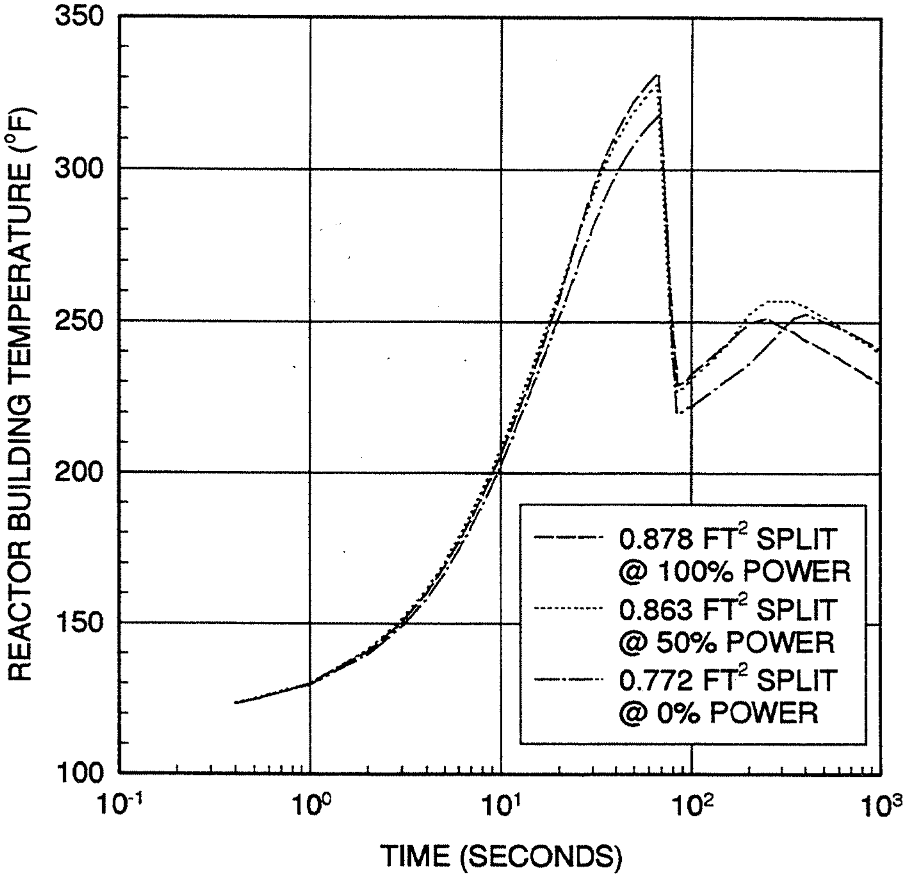
FIGURE 6.2-5 Deleted By Amendment 96-02



SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Worst Case (Temperature)
Main Steam Line Break
1.4 ft^2 DER at 102% Power
Reactor Building Temperature Vs. Time

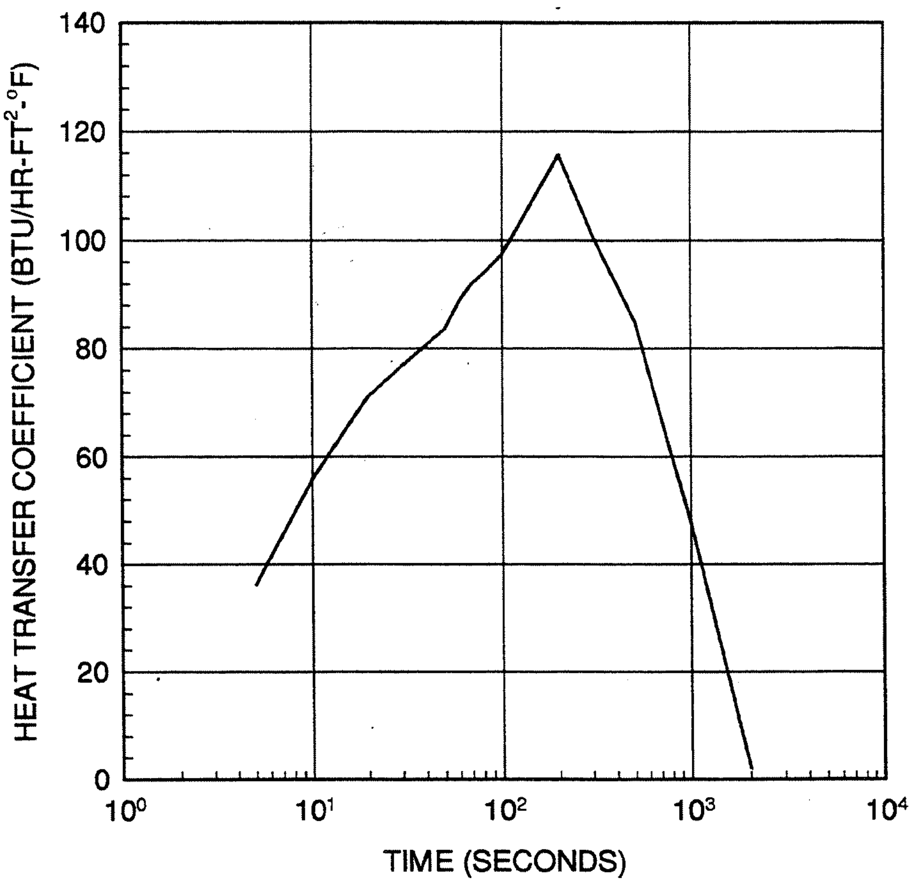
Figure 6.2-5a



SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Reactor Building Vapor Temperature
For A Spectrum of Small
Main Steam Line Breaks

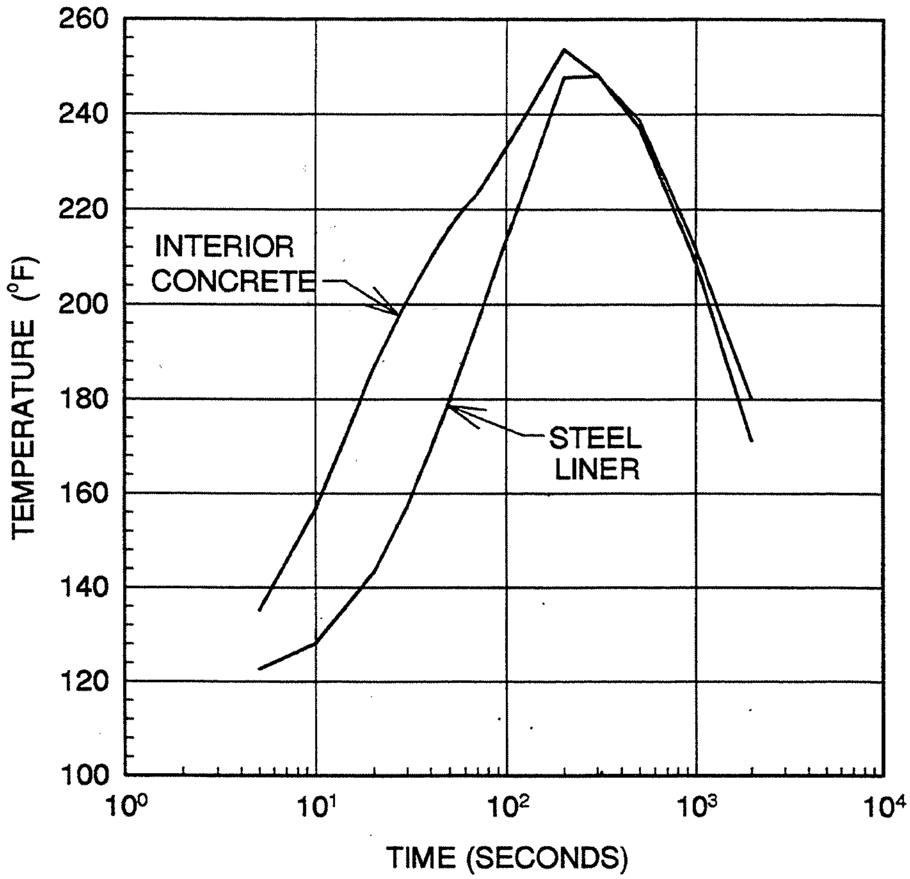
Figure 6.2-5b



SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Worst Case (Temperature)
Main Steam Line Break
1.4 ft^2 DER at 102% Power
Liner Condensing Film Heat Transfer
Coefficient (UCHIDA) Vs. Time

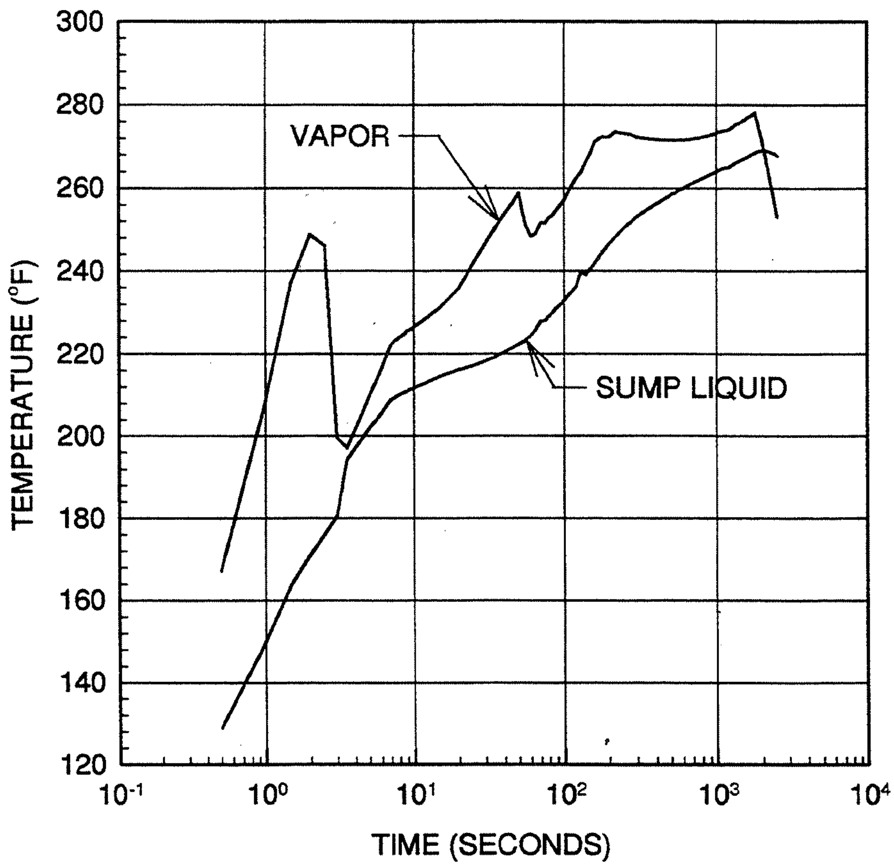
Figure 6.2-5c



SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Worst Case (Temperature)
Main Steam Line Break
1.4 ft^2 DER at 102% Power
Reactor Building Liner and Concrete
Temperature Vs. Time

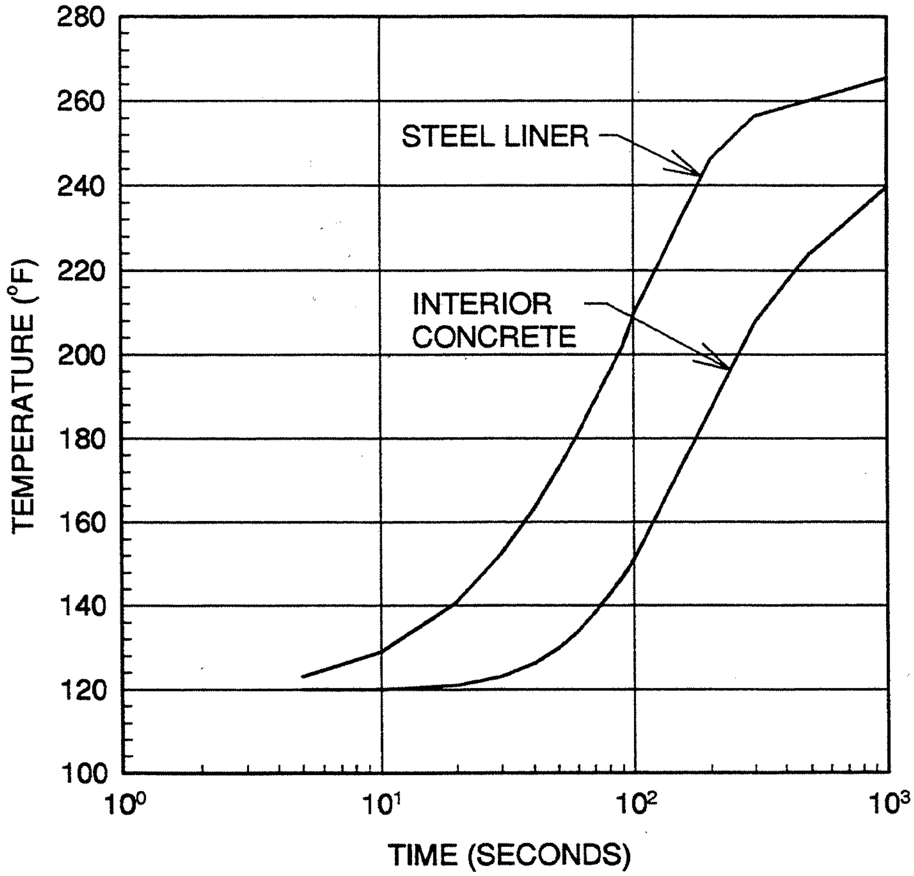
Figure 6.2-5d



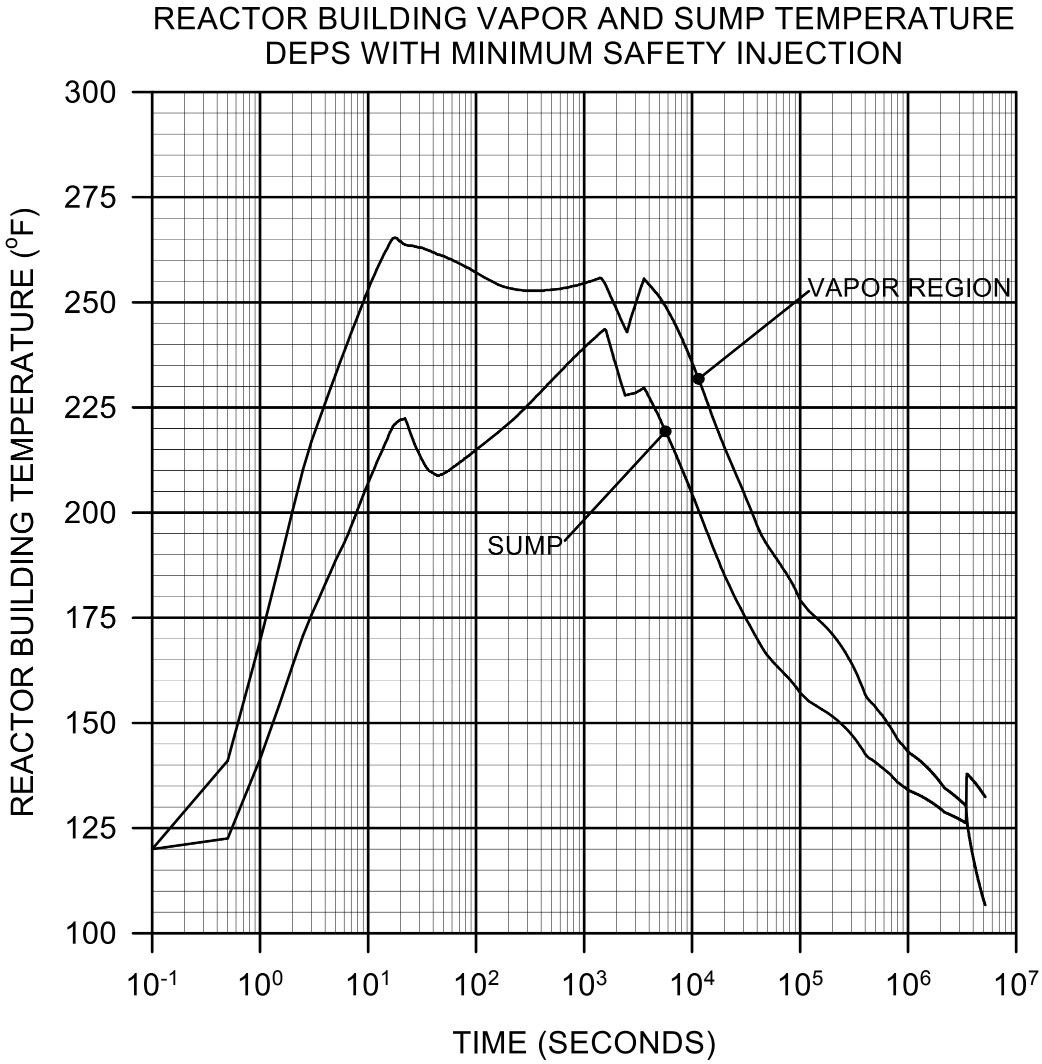
SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Main Steam Line Break
(Design Basis Accident)
1.4 ft^2 DER at 25% Power
Reactor Building Vapor and Sump
Temperature Vs. Time

Figure 6.2-6



SOUTH CAROLINA ELECTRIC & GAS CO. VIRGIL C. SUMMER NUCLEAR STATION
Main Steam Line Break (Design Basis Accident) 1.4 ft^2 DER at 25% Power Reactor Building Liner and Concrete Temperature Vs. Time Figure 6.2-6a



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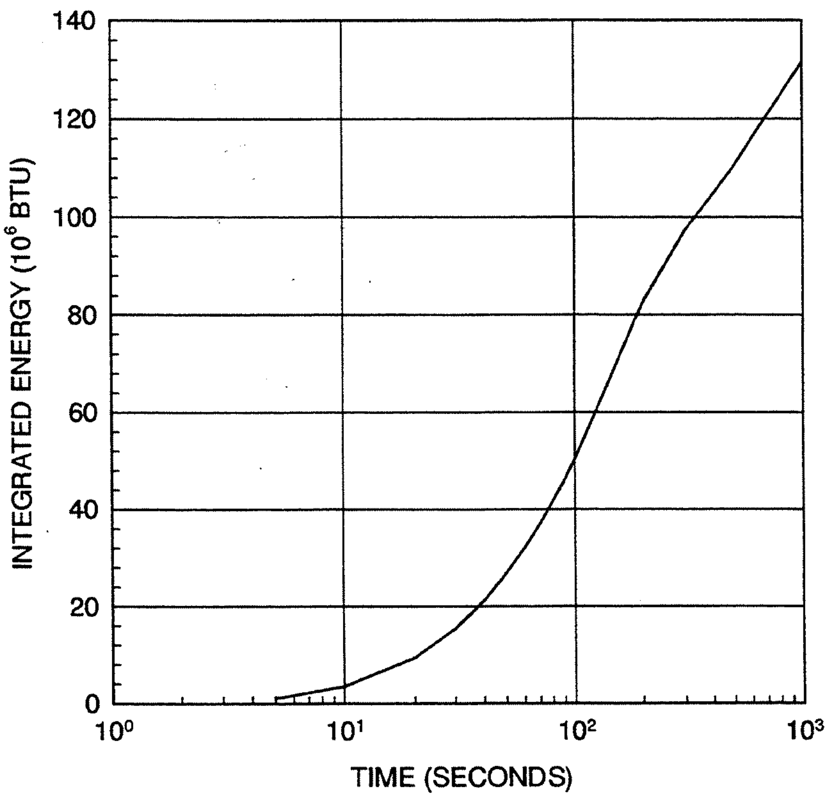
RN 03-003
June 2003

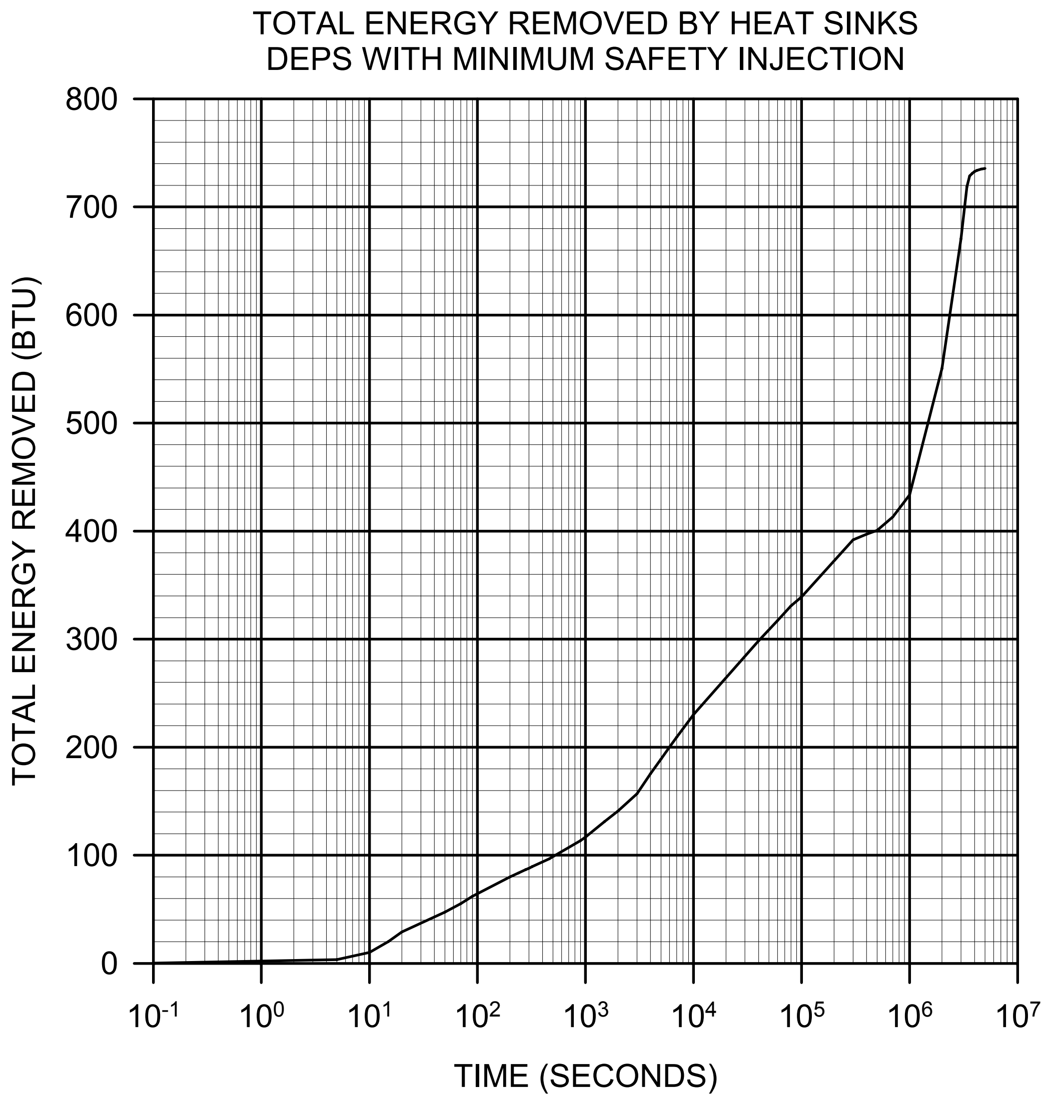
SOUTH CAROLINA ELECTRIC & GAS CO VIRGIL C. SUMMER NUCLEAR STATION
Double Ended Pump Suction Break Minimum Safety Injection Reactor Building Vapor and Sump Temperature vs. Time Figure 6.2-7



SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Main Steam Line Break
(Design Basis Accident)
1.4 ft^2 DER at 25% Power
Total Energy In Heat Sinks Vs. Time

Figure 6.2-8



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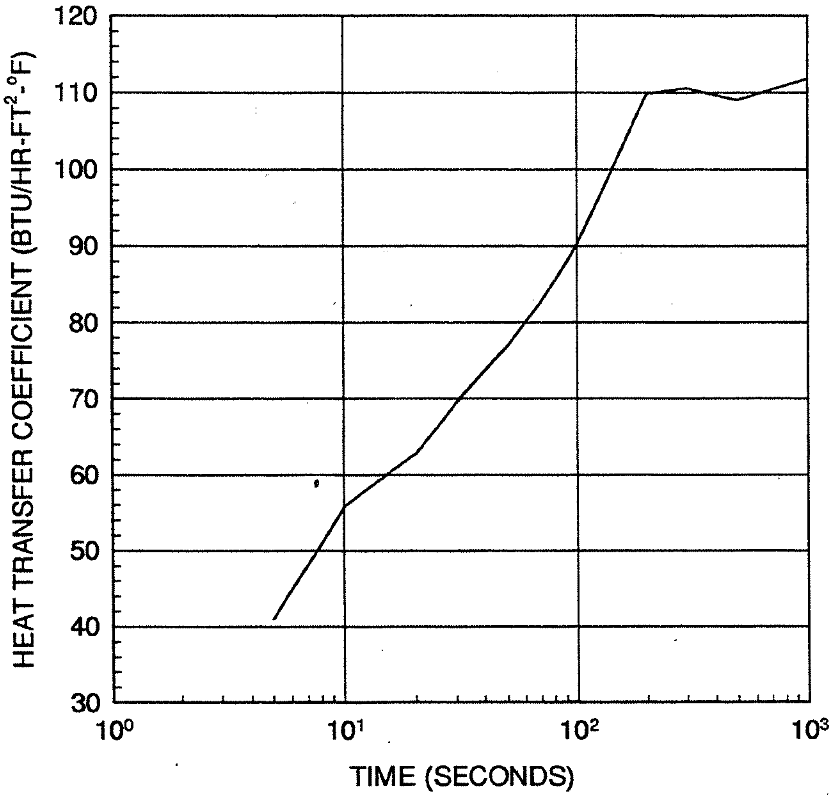
SOUTH CAROLINA ELECTRIC & GAS CO
VIRGIL C. SUMMER NUCLEAR STATION

Double Ended Pump Suction Break
Minimum Safety Injection
Total Energy Removed by Heat Sinks

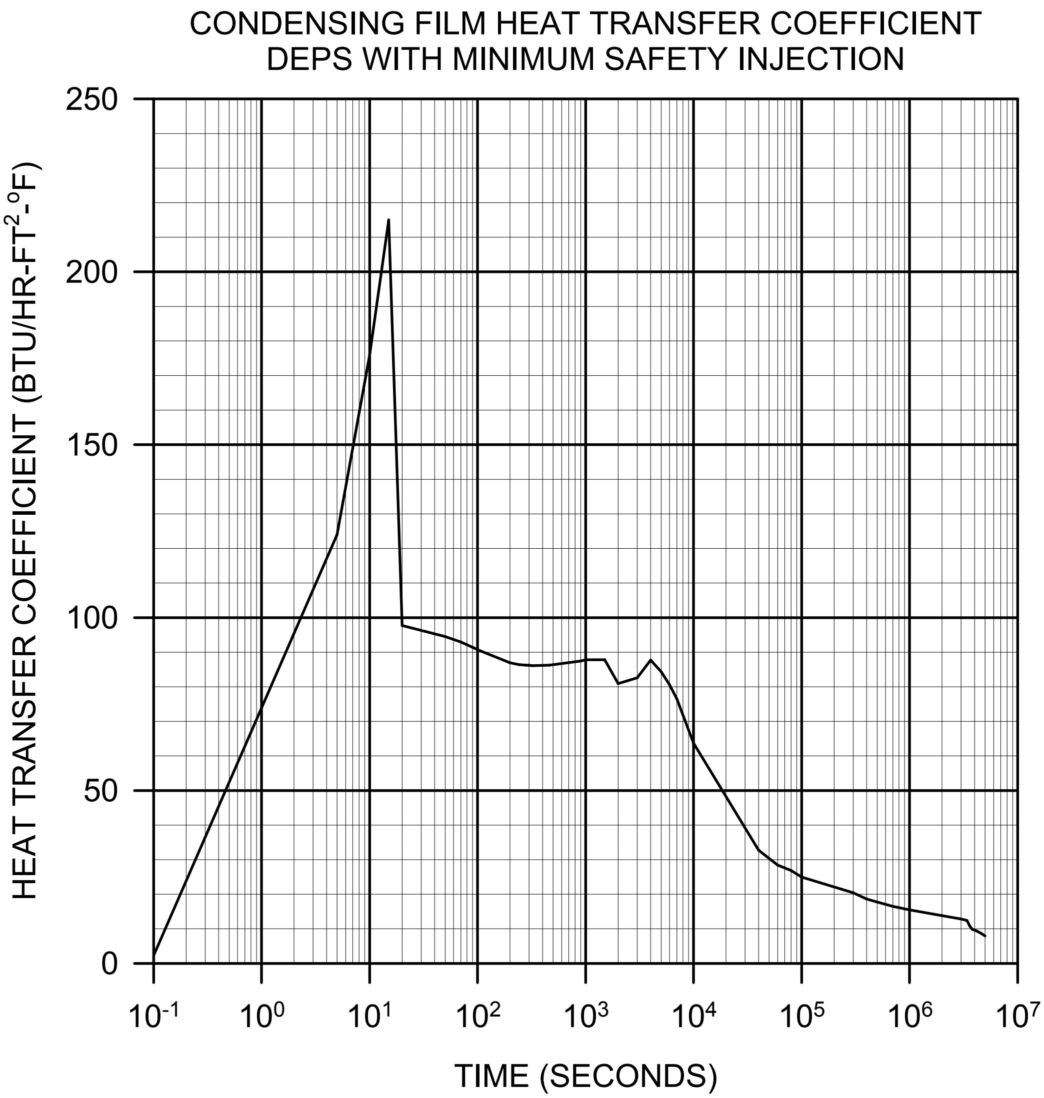
Figure 6.2-9



SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Main Steam Line Break
(Design Basis Accident)
1.4 ft^2 DER at 25% Power
Condensing Film Heat Transfer Coefficient
(UCHIDA) Vs. Time

Figure 6.2-10



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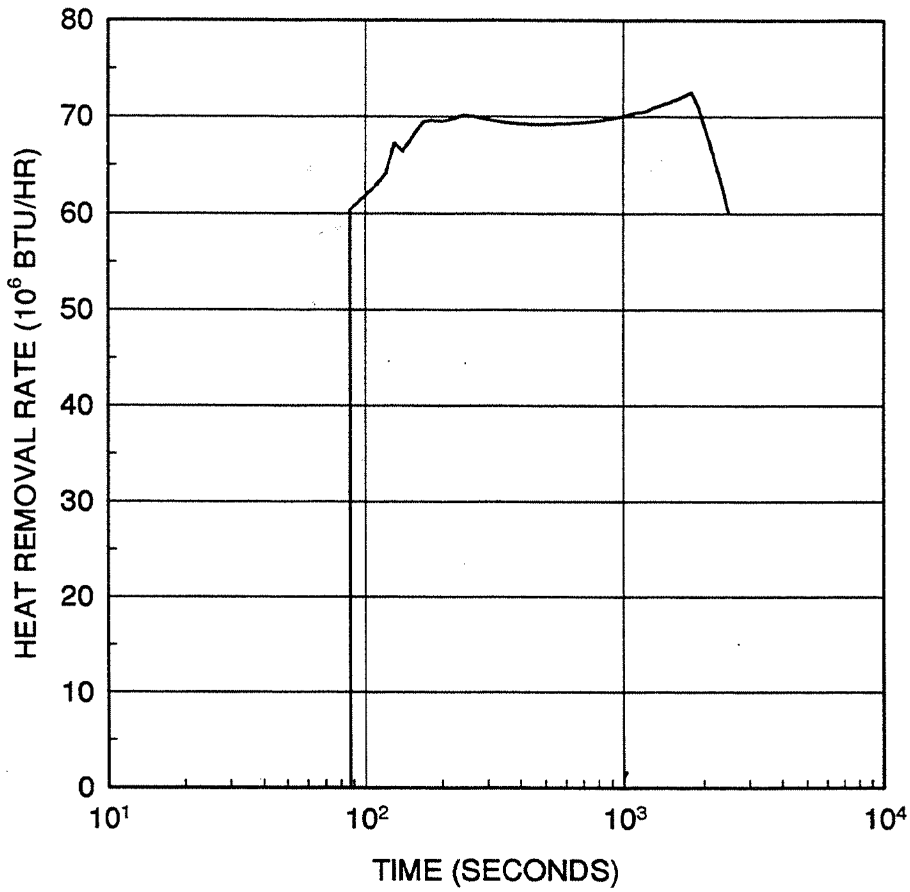
SOUTH CAROLINA ELECTRIC & GAS CO
VIRGIL C. SUMMER NUCLEAR STATION

Double Ended Pump Suction Break
Minimum Safety Injection
Film Heat Transfer Coefficient

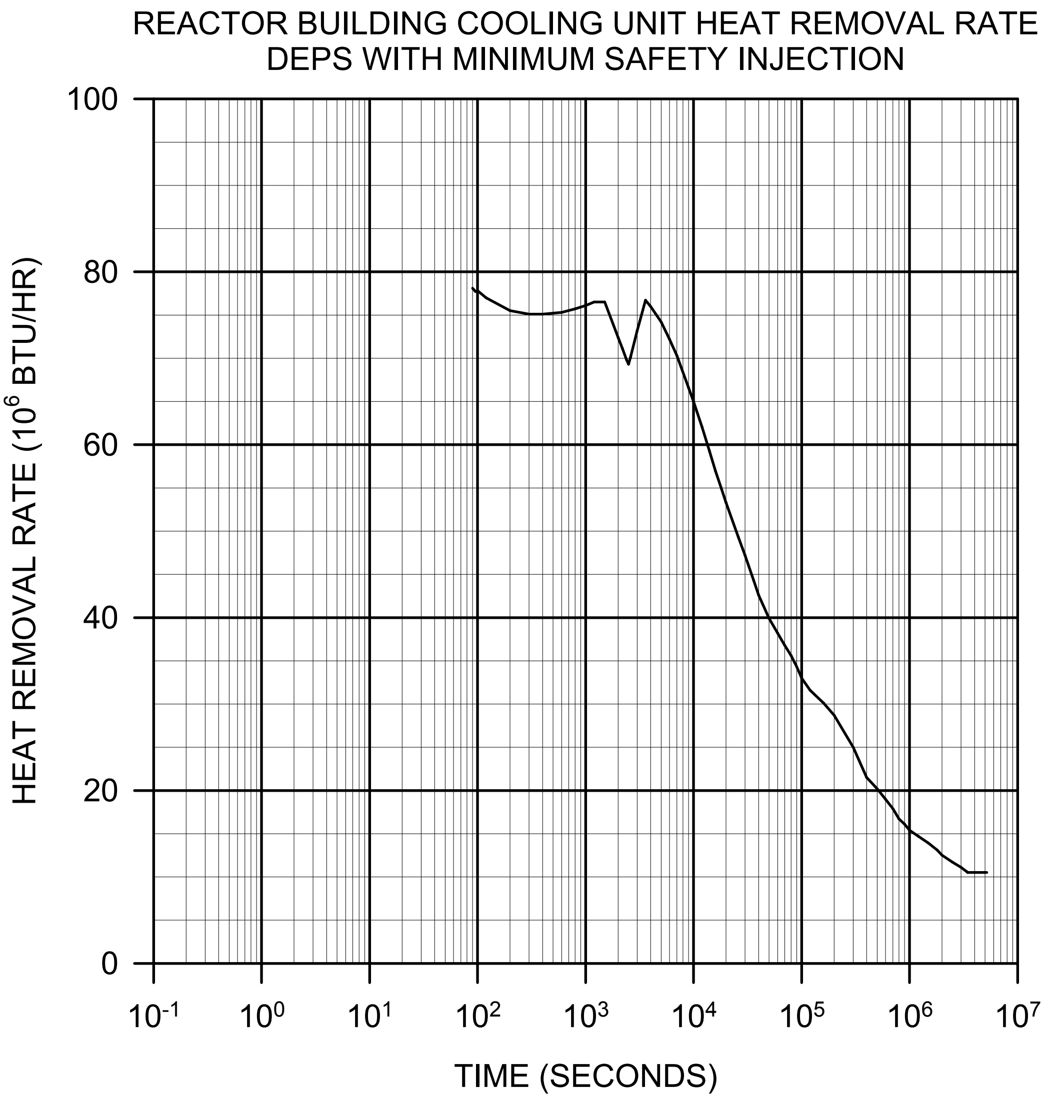
Figure 6.2-11



SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Main Steam Line Break
(Design Basis Accident)
1.4 ft^2 DER at 25% Power
Reactor Building Cooling Unit
Heat Removal Rate Vs. Time

Figure 6.2-12



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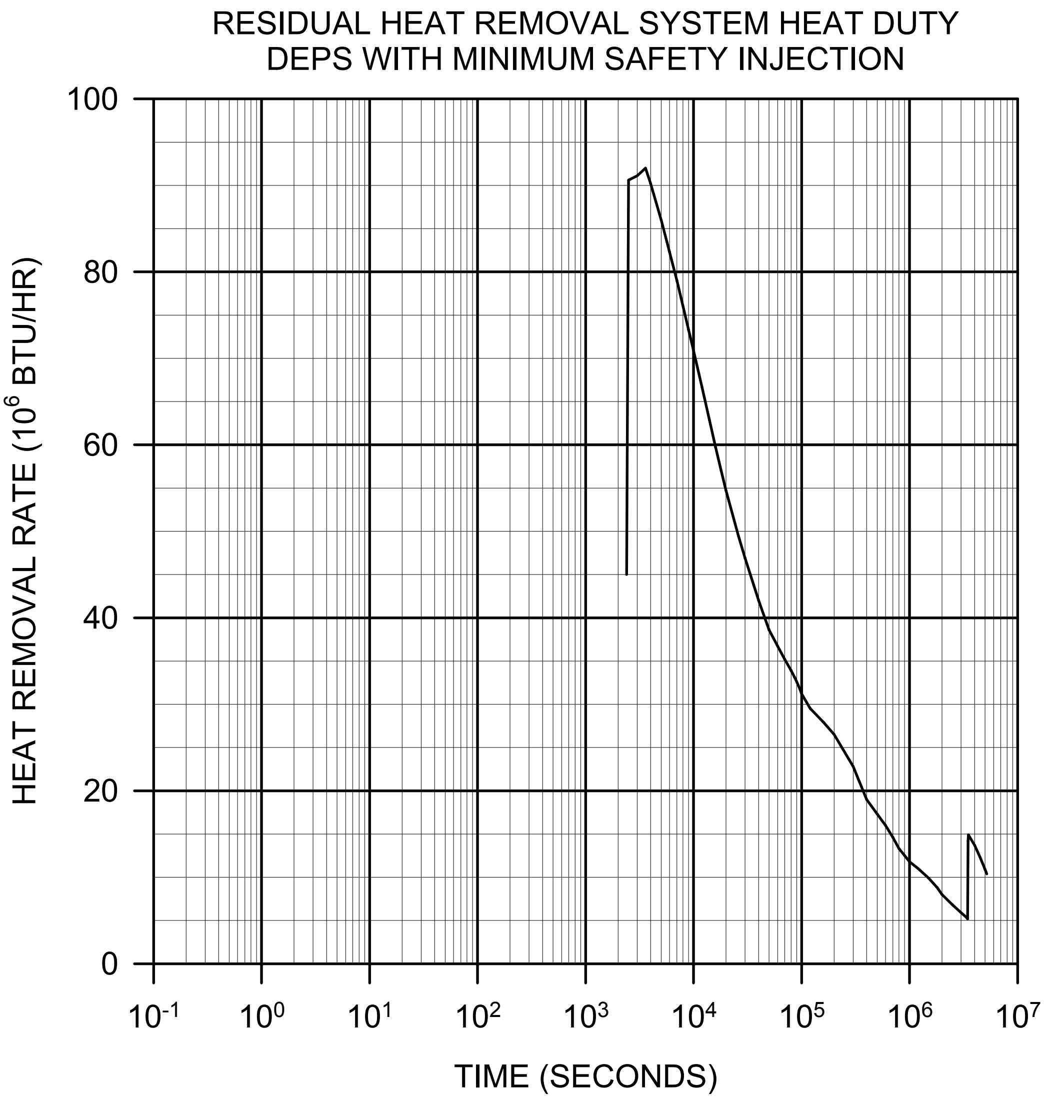
SOUTH CAROLINA ELECTRIC & GAS CO
VIRGIL C. SUMMER NUCLEAR STATION

Double Ended Pump Suction Break
Minimum Safety Injection
Reactor Building Cooling Unit
Heat Removal Rate vs. Time

Figure 6.2-13



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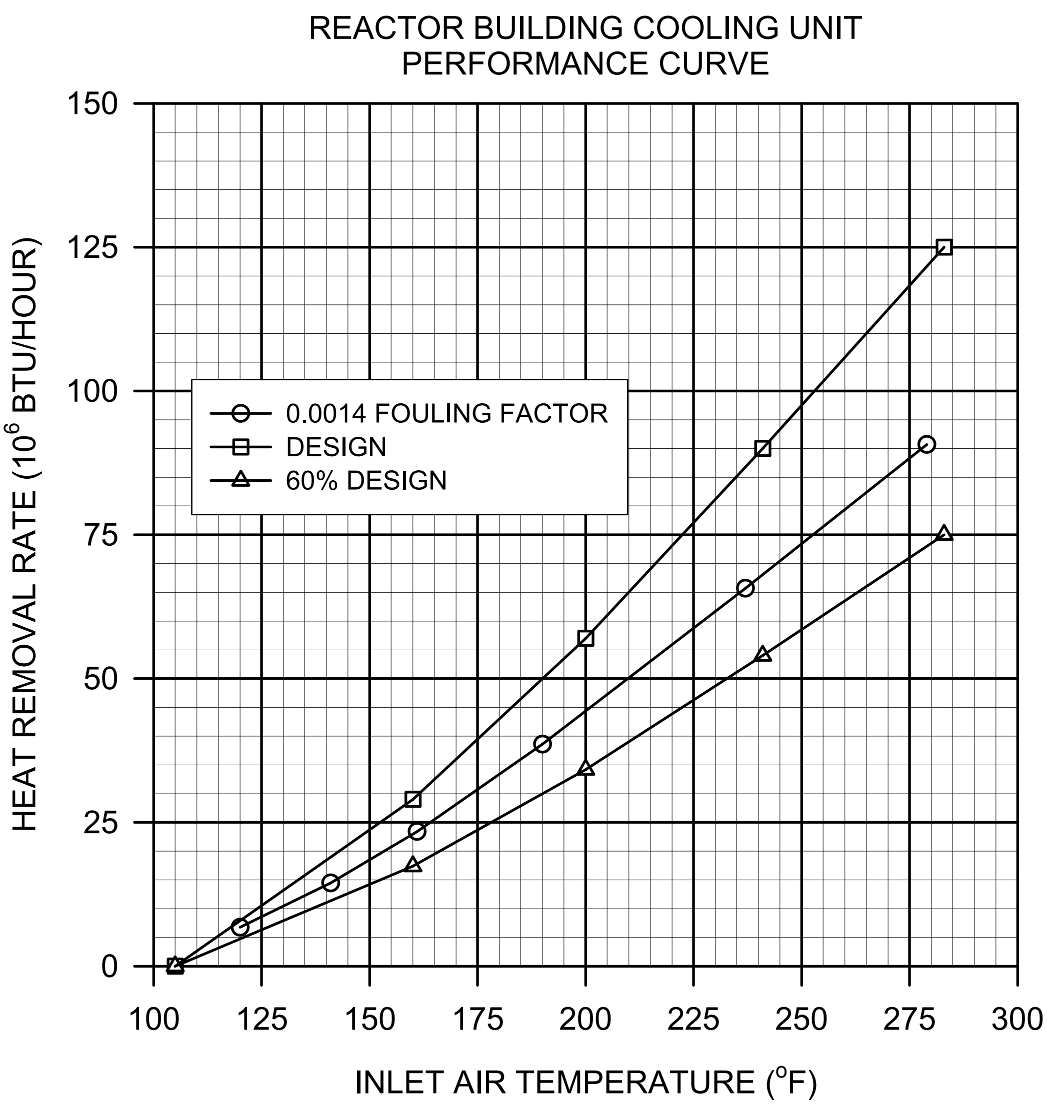
SOUTH CAROLINA ELECTRIC & GAS CO
VIRGIL C. SUMMER NUCLEAR STATION

Double Ended Pump Suction Break
Minimum Safety Injection
Residual Heat Removal System
Heat Duty vs. Time

Figure 6.2-14



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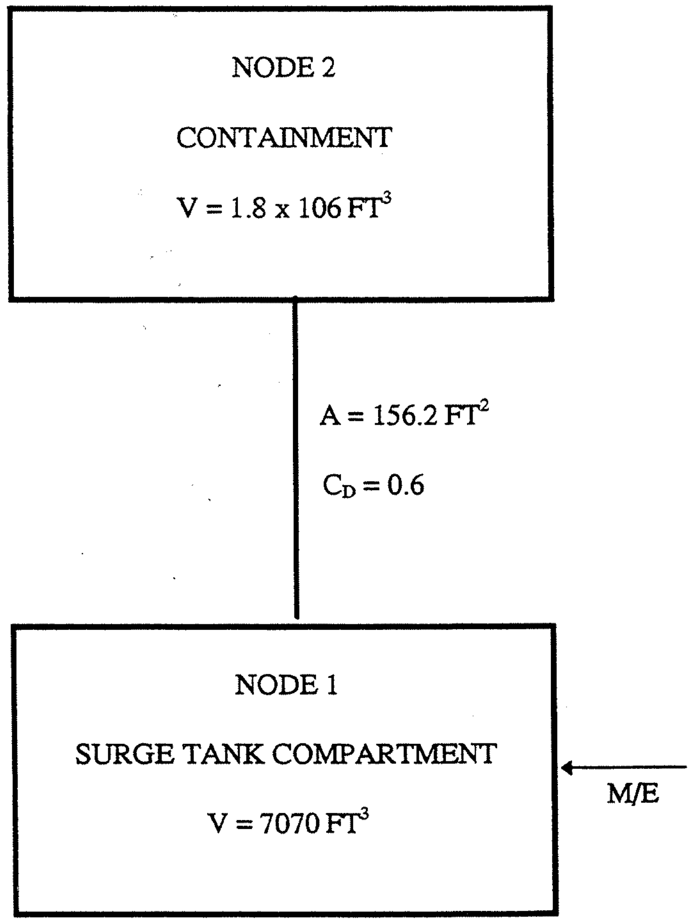
Reactor Building Cooling Unit
Design Heat Removal Performance

Figure 6.2-15



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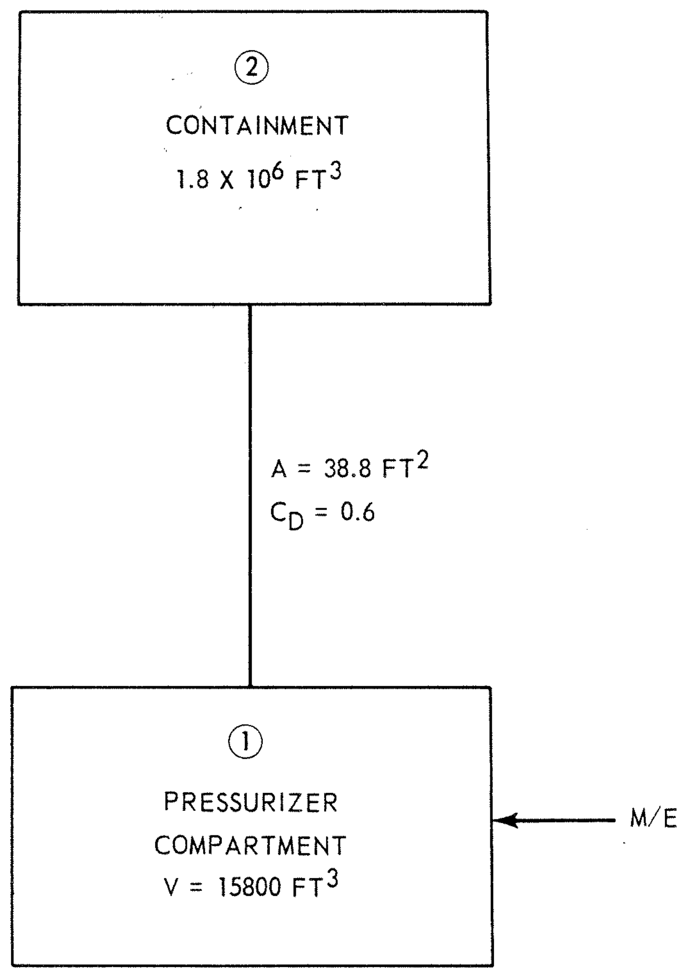
VIRGIL C. SUMMER NUCLEAR STATION

Surge Line Rupture In Surge Tank Compartment

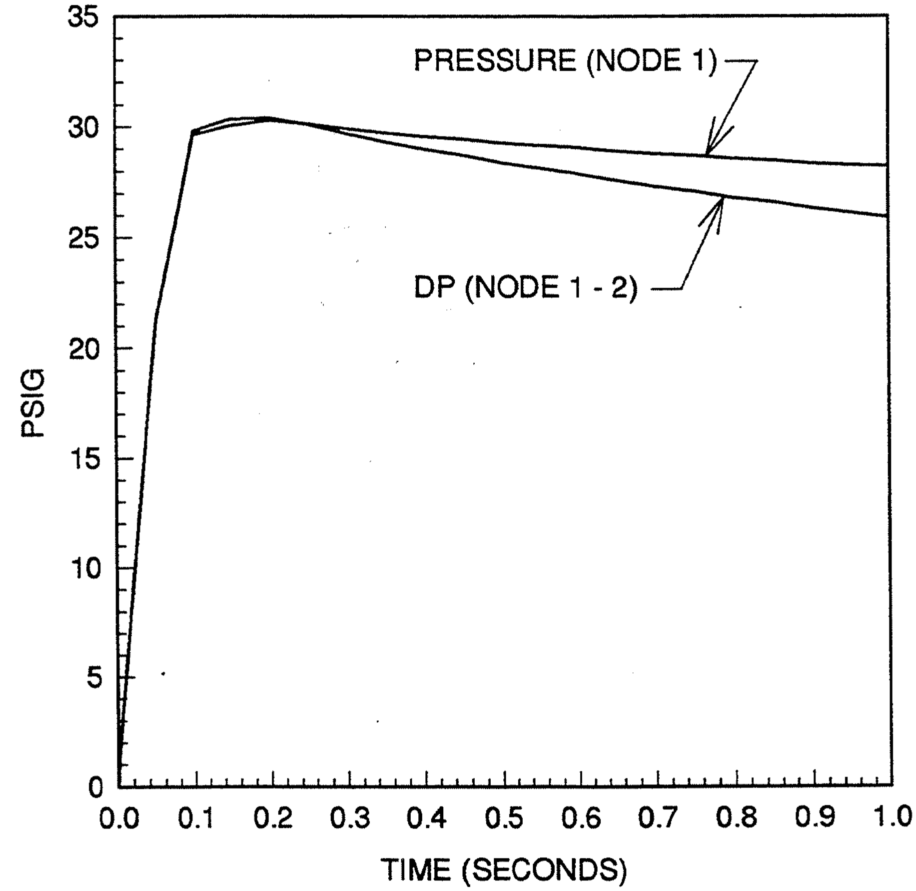
Figure 6.2-18



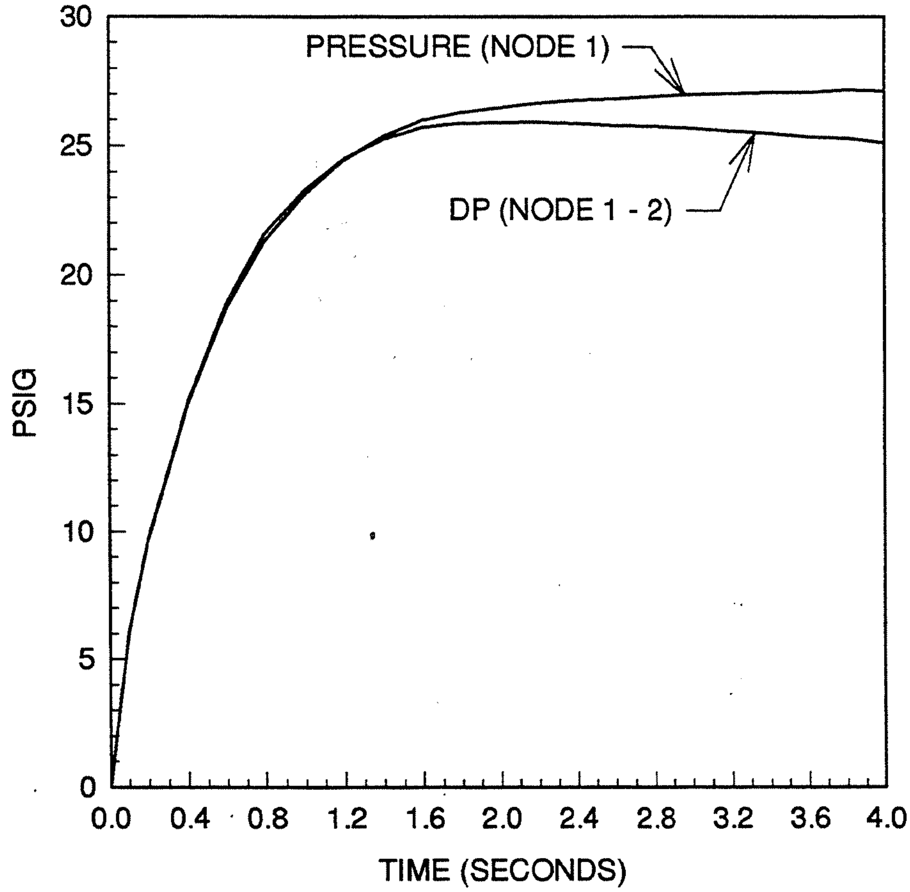
SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Spray Line Rupture In Pressurizer Compartment

Figure 6.2-19



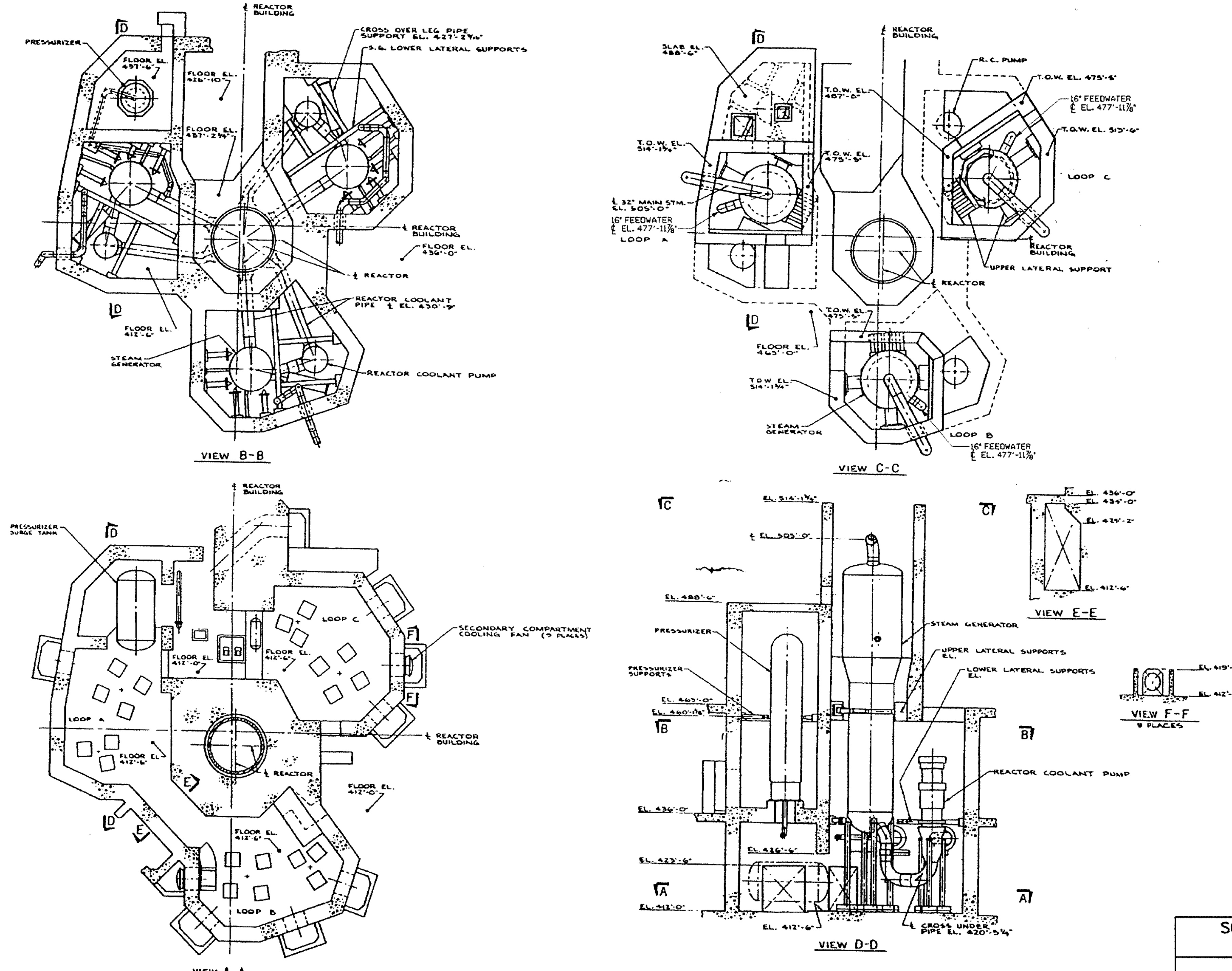
SOUTH CAROLINA ELECTRIC & GAS CO. VIRGIL C. SUMMER NUCLEAR STATION
Surge Tank Compartment Pressurization Figure 6.2-20



SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Pressurizer Compartment
Pressurization

Figure 6.2-21



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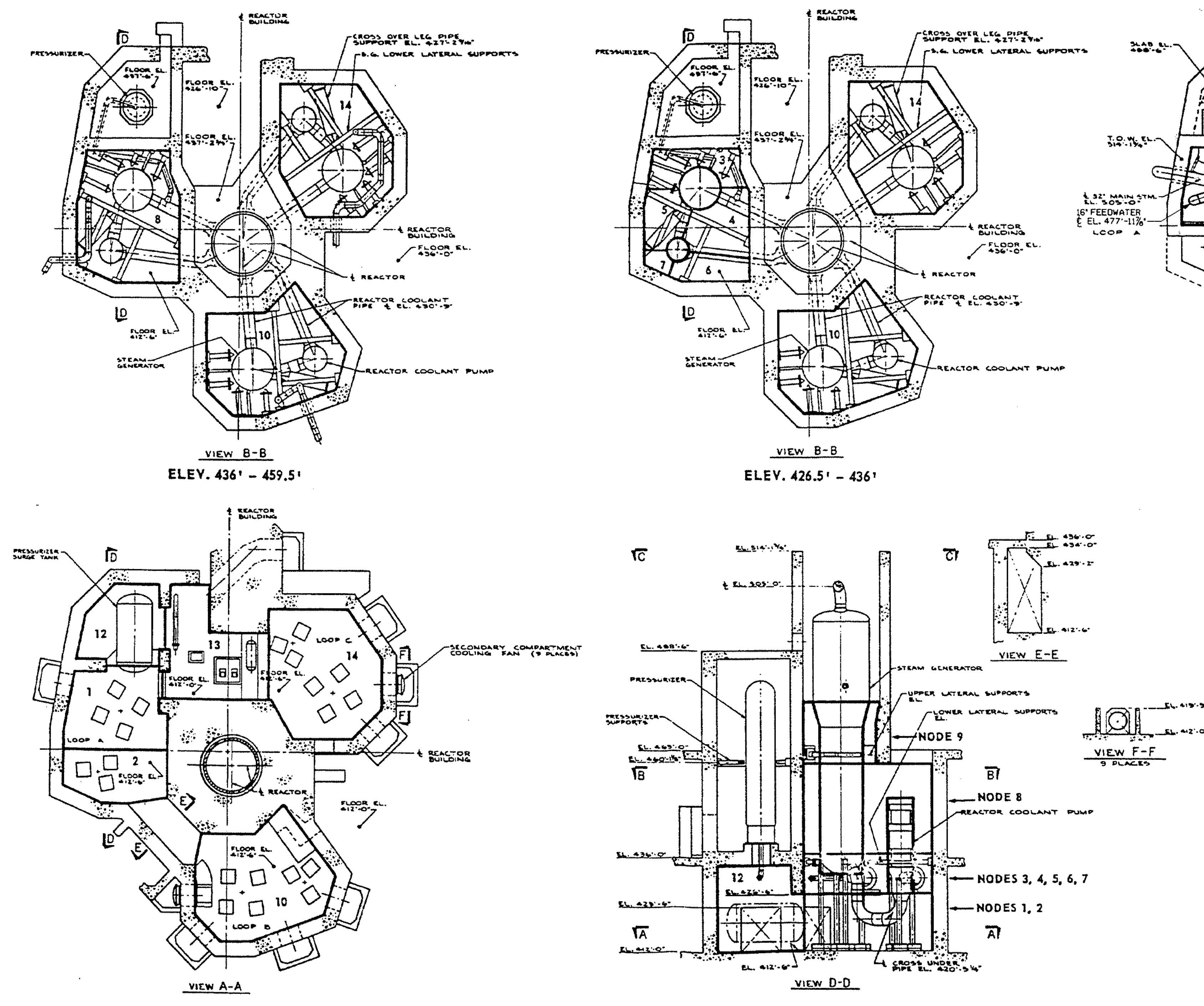
VIRGIL C. SUMMER NUCLEAR STATION

GENERAL LAYOUT - STEAM GENERATOR COMPARTMENTS

Figure 6.2-22



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SOUTH CAROLINA ELECTRIC & GAS CO.
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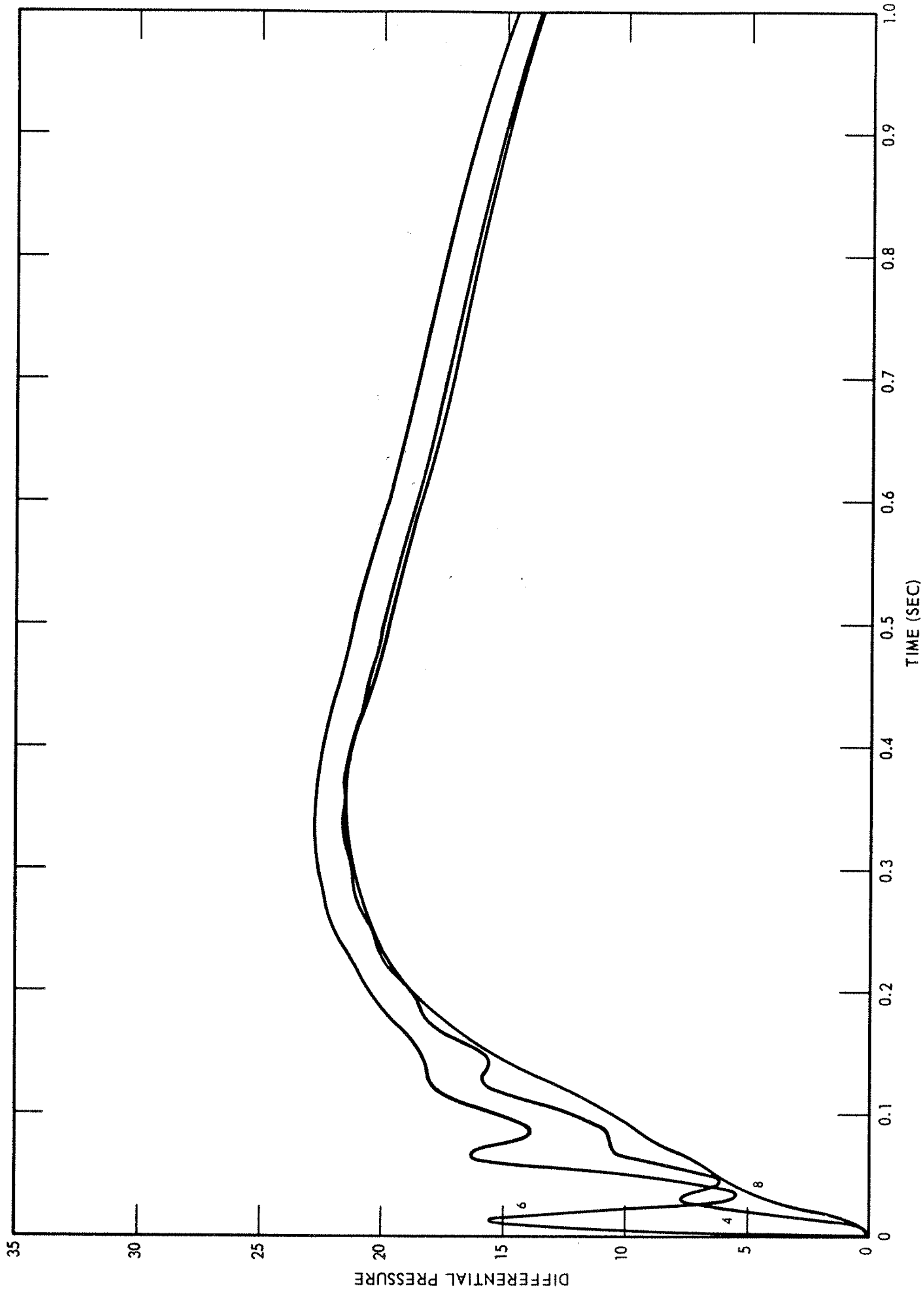
Loop A Steam Generator Compartment Nodal Arrangement 16 Node Model, Cold Leg Break

Figure 6.2-22a



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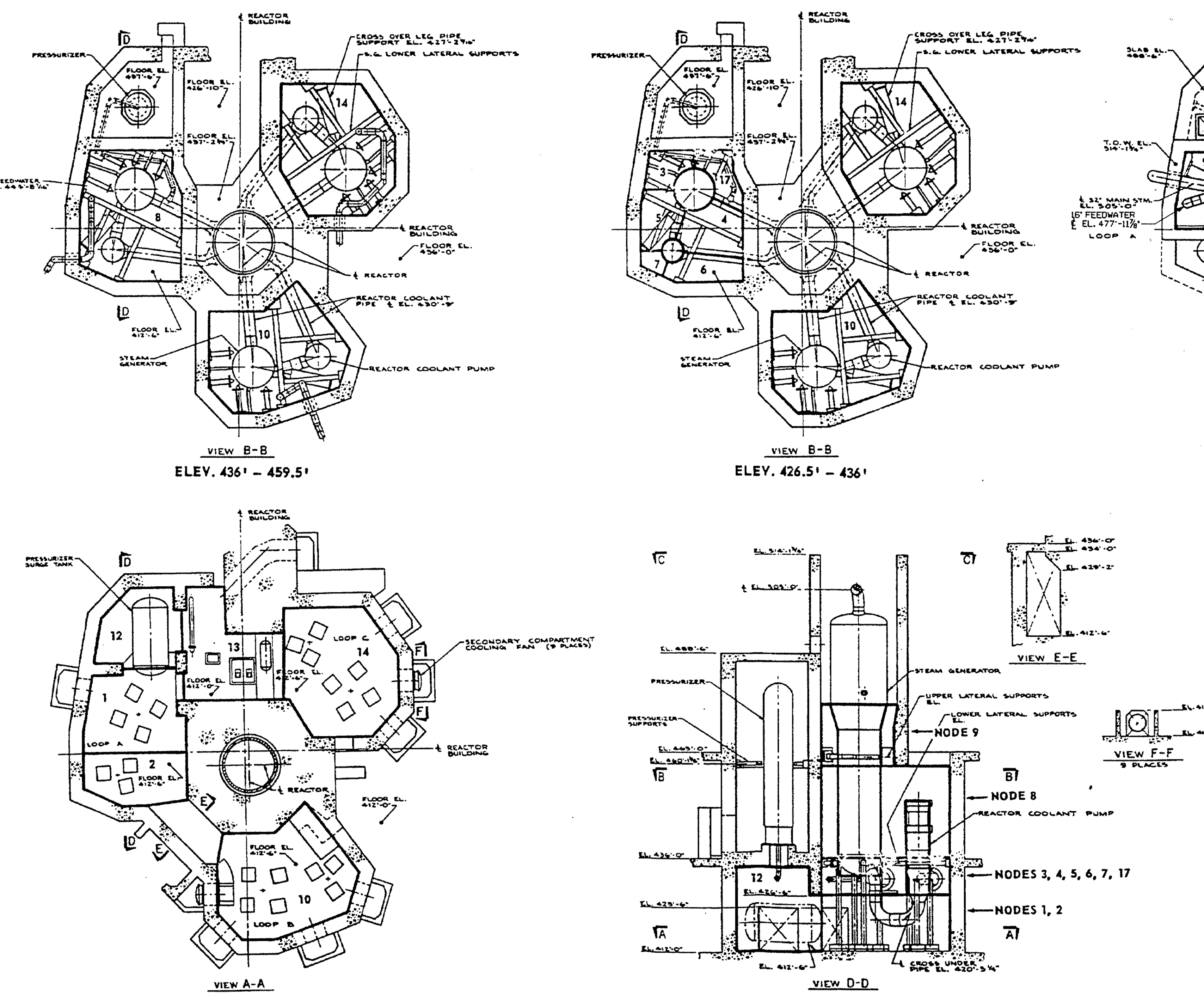
VIRGIL C. SUMMER NUCLEAR STATION

Loop A Steam Generator Compartment
Nodal Pressures Relative to Containment
16 Node Cold Leg Break Model

Figure 6.2-22b



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SOUTH CAROLINA ELECTRIC & GAS CO.
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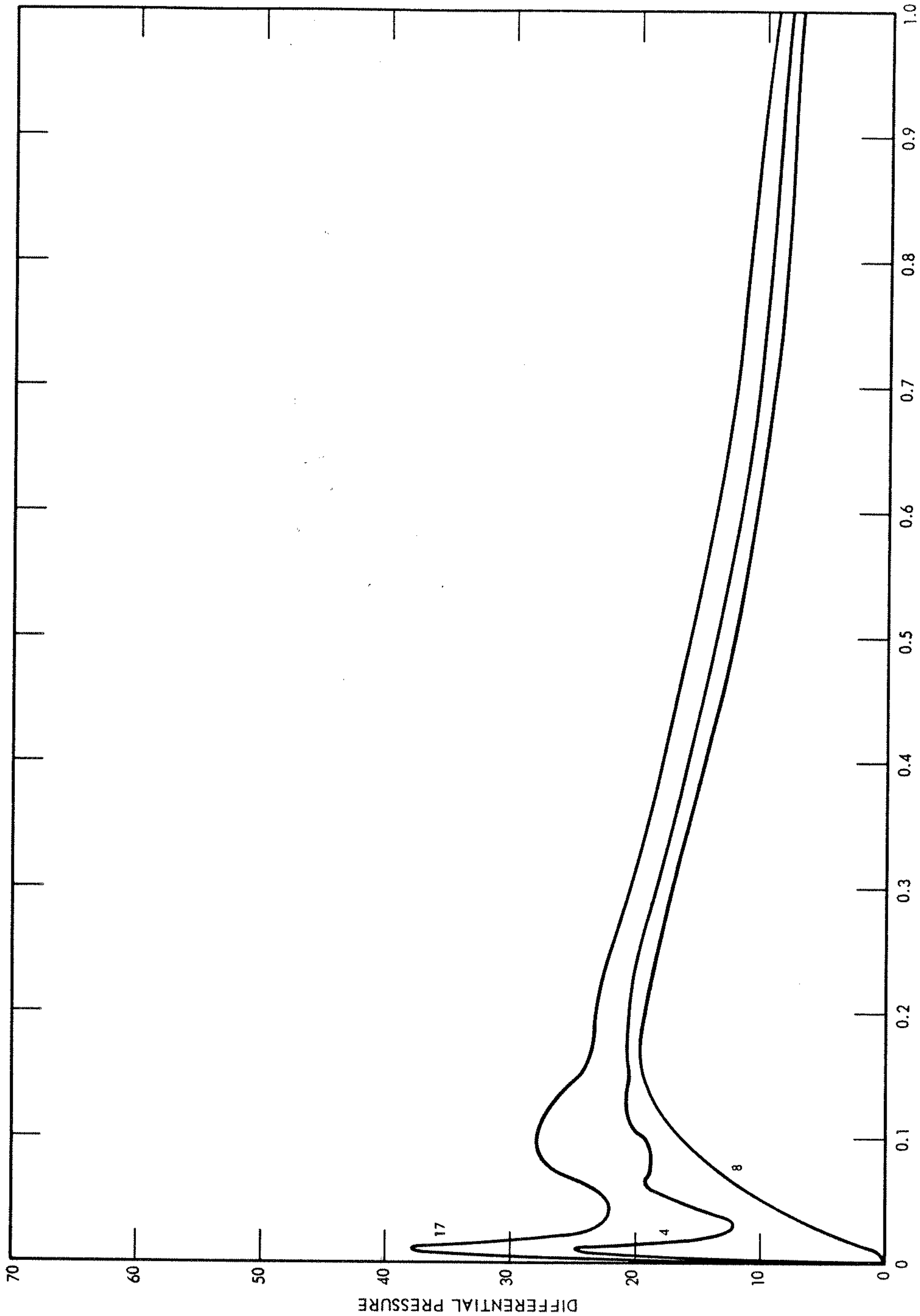
Loop A Steam Generator Compartment Nodal Arrangement 17 Node Model, Hot Leg Break

Figure 6.2-22c



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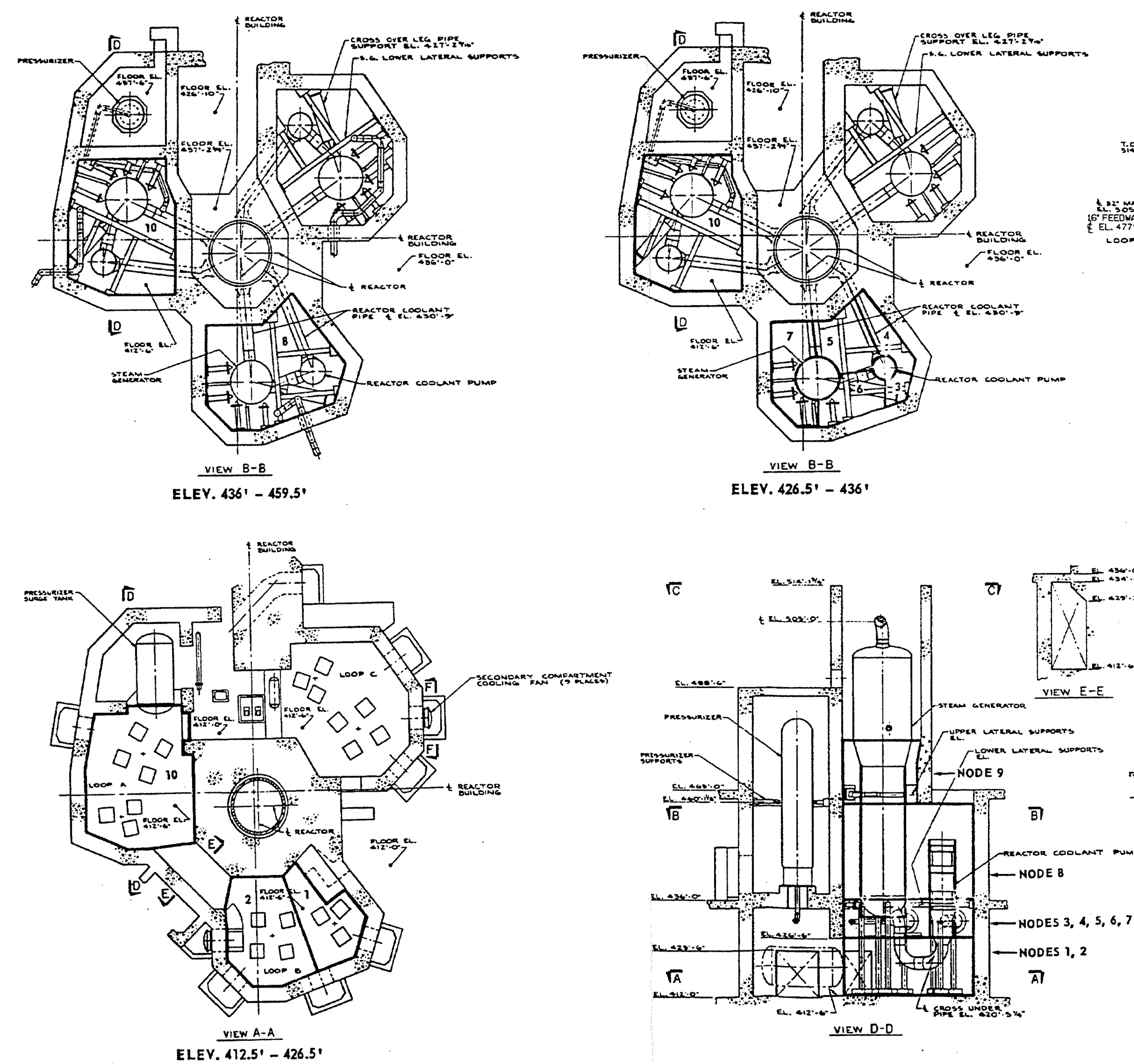
Loop A Steam Generator Compartment
Nodal Pressures Relative to Containment
17 Node Hot Leg Break Model

Figure 6.2-22d



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SOUTH CAROLINA ELECTRIC & GAS CO.
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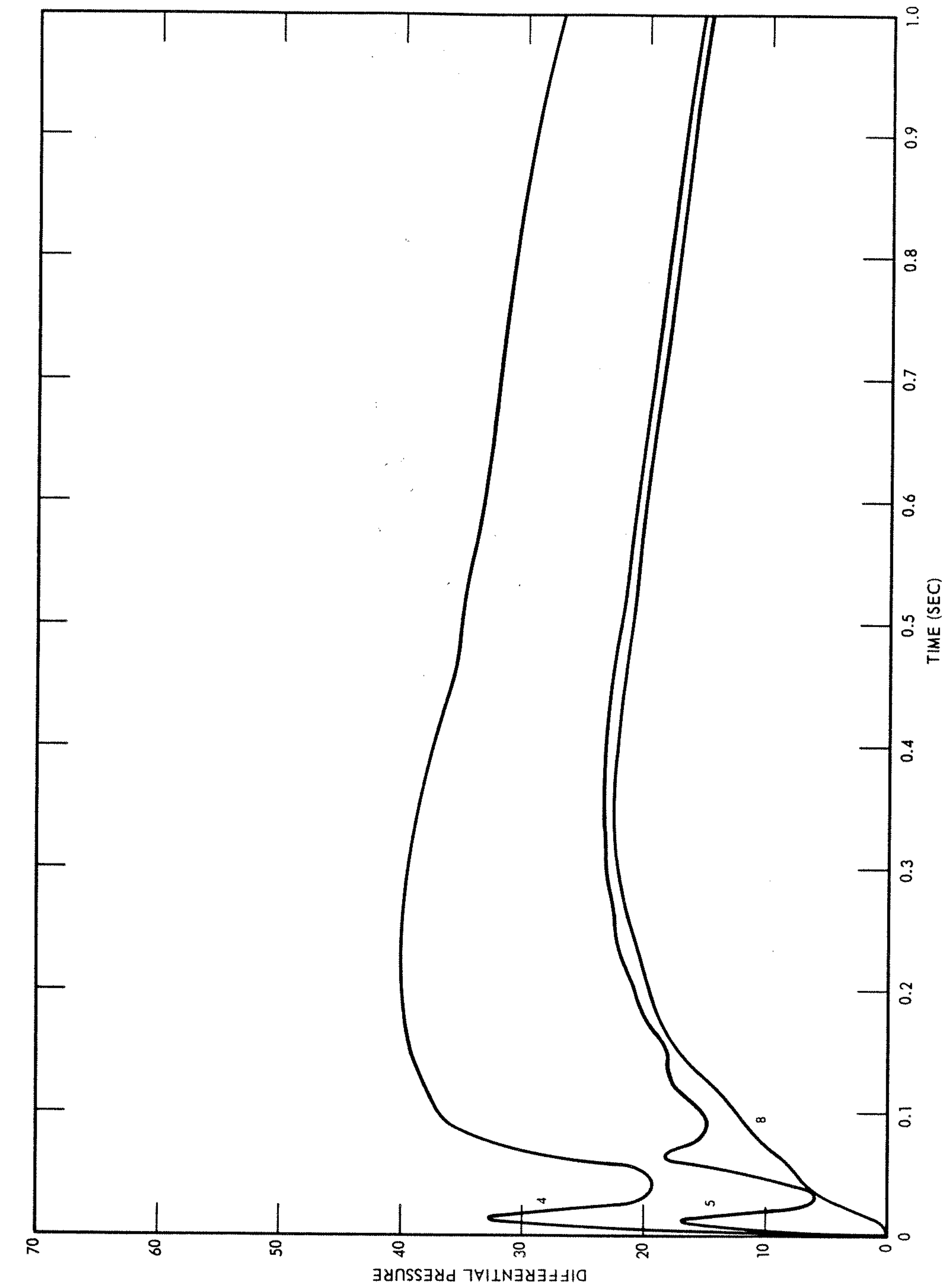
Loop B Steam Generator Compartment
Nodal Arrangement
12 Node Model, Cold Leg Break

Figure 6.2-22e



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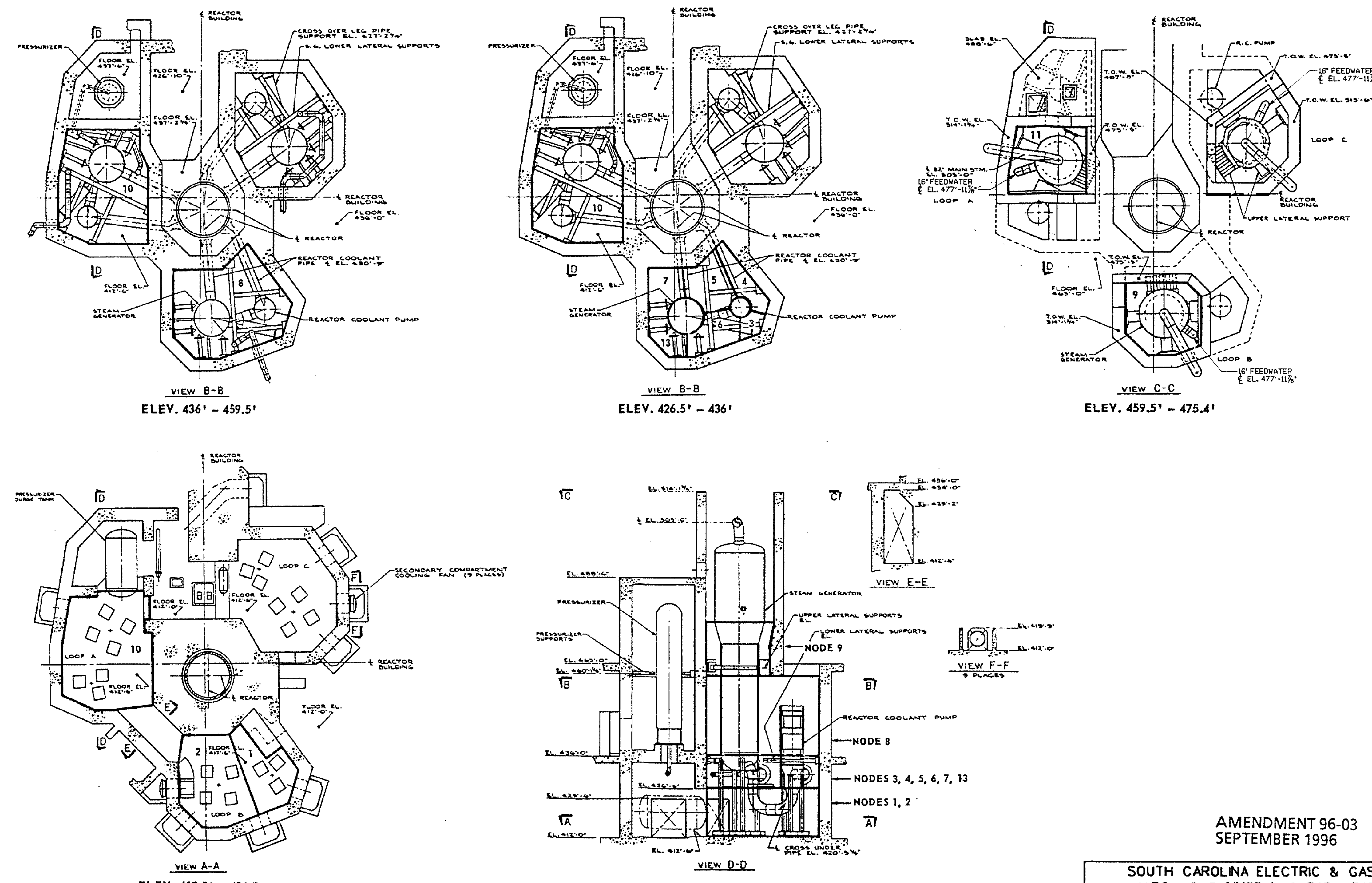
August 1984

SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Loop B Steam Generator Compartment
Nodal Pressures Relative to Containment
12 Node Cold Leg Break Model

Figure 6.2-22f



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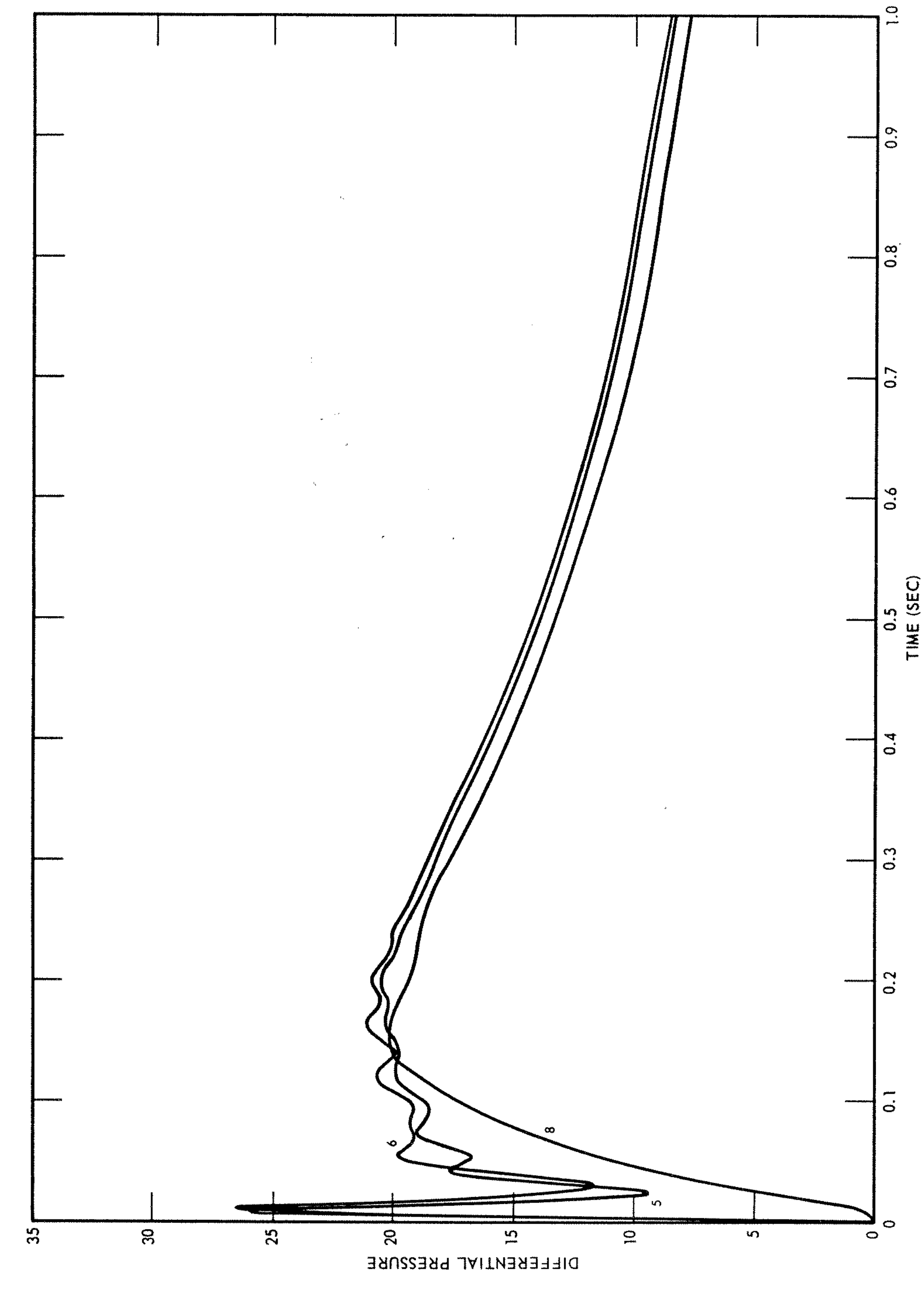
SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Loop B Steam Generator Compartment
Nodal Arrangement
13 Node Model, Hot Leg Break

Figure 6.2-22g



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SOUTH CAROLINA ELECTRIC & GAS CO.
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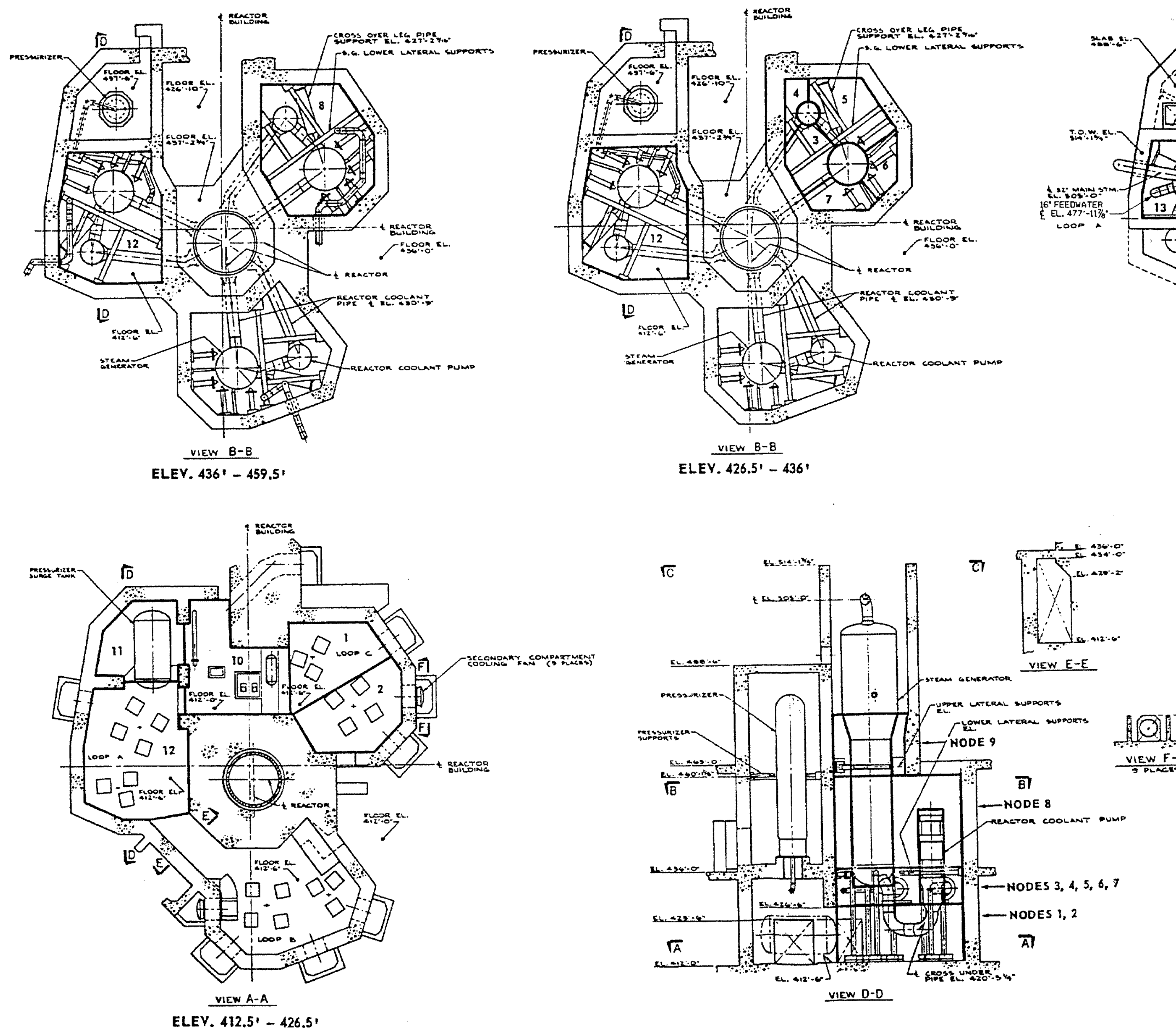
Loop B Steam Generator Compartment
Nodal Pressures Relative to Containment
13 Node Hot Leg Break Model

Figure 6.2-22h



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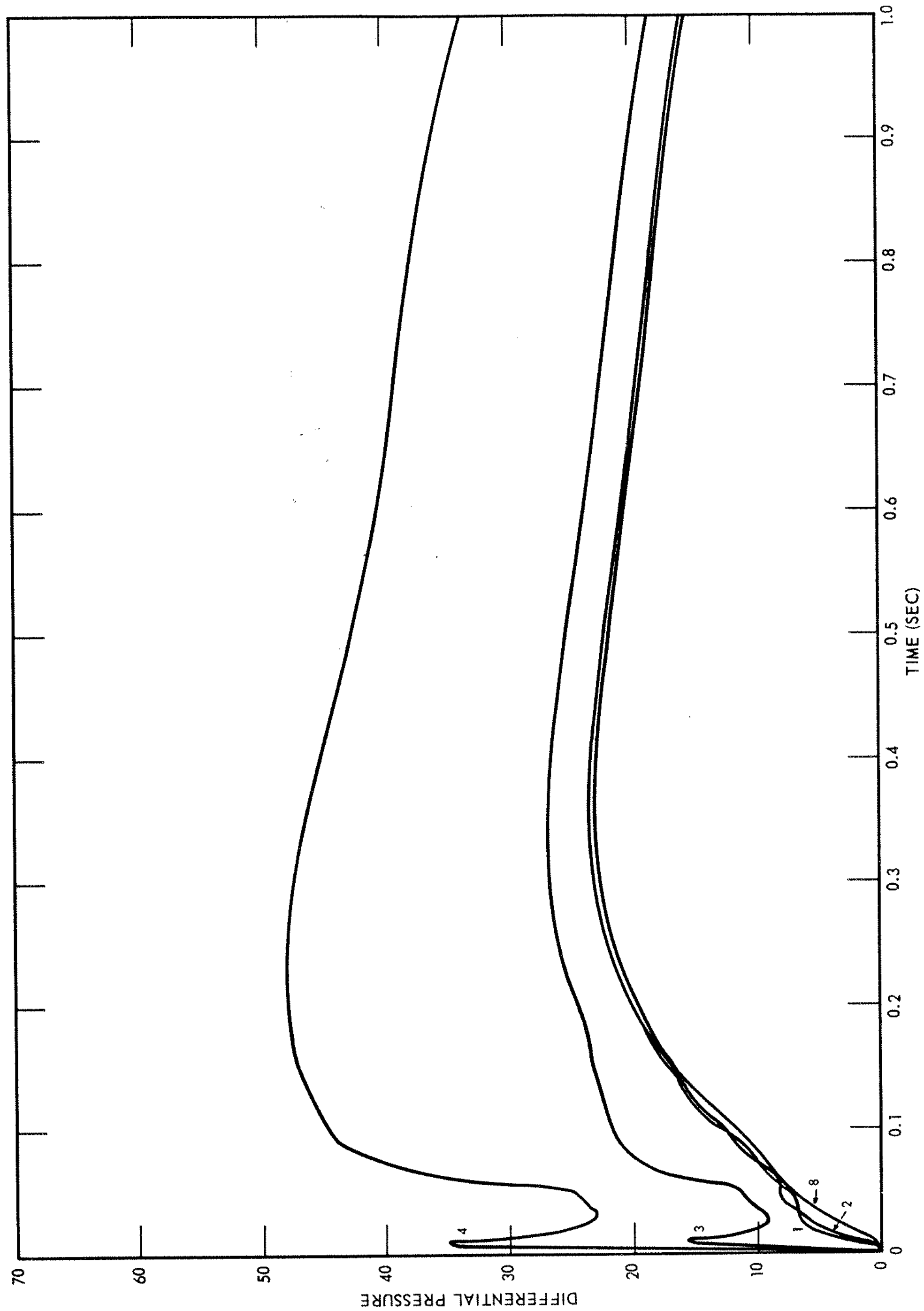
SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Loop C Steam Generator Compartment
Nodal Arrangement
14 Node Model, Hot and Cold Leg Breaks

Figure 6.2-22i



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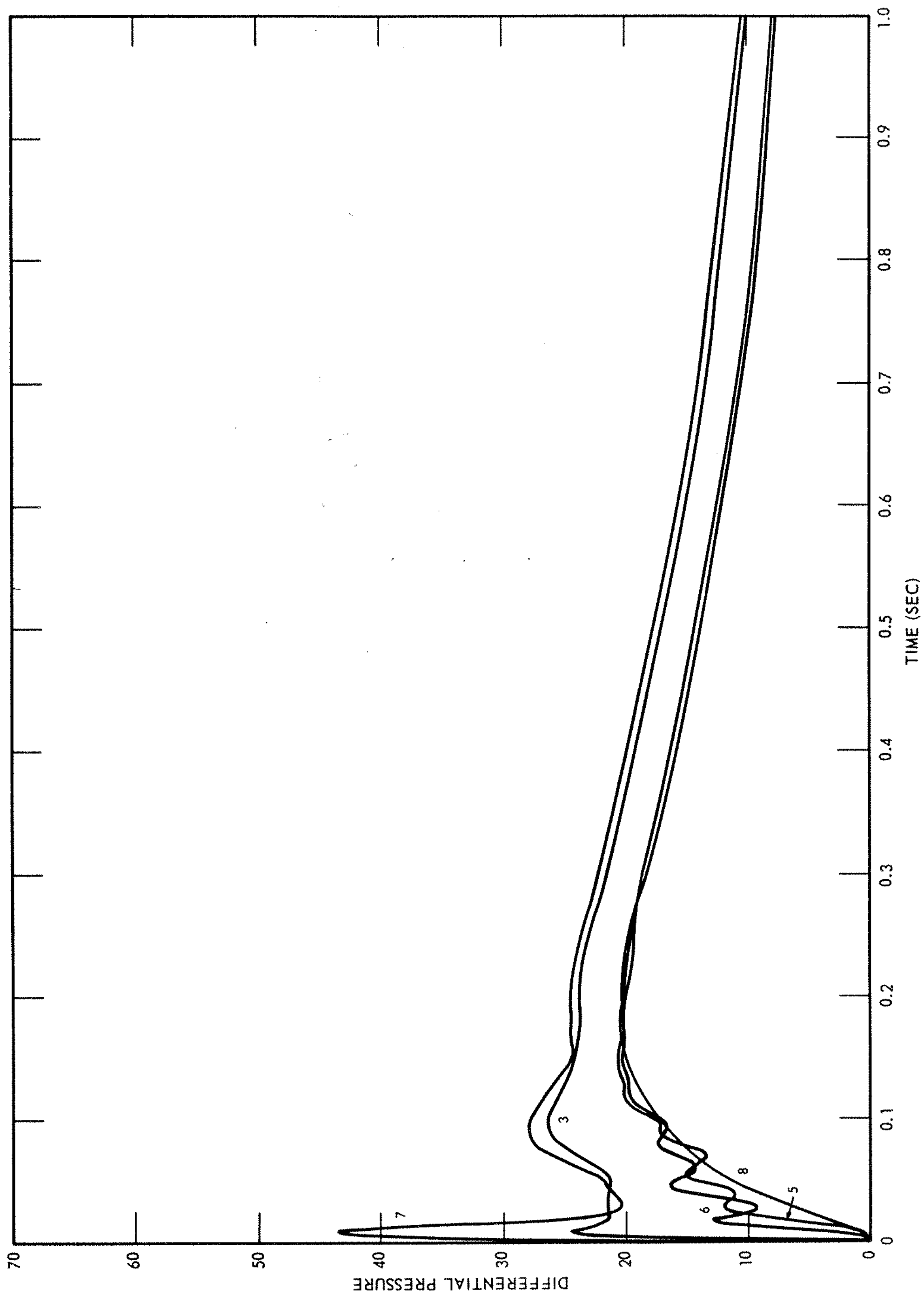
VIRGIL C. SUMMER NUCLEAR STATION

Loop C Steam Generator Compartment
Nodal Pressures Relative to Containment
14 Node Cold Leg Break Model

Figure 6.2-22j



SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Loop C Steam Generator Compartment
Nodal Pressures Relative to Containment
14 Node Hot Leg Break Model

Figure 6.2-22k

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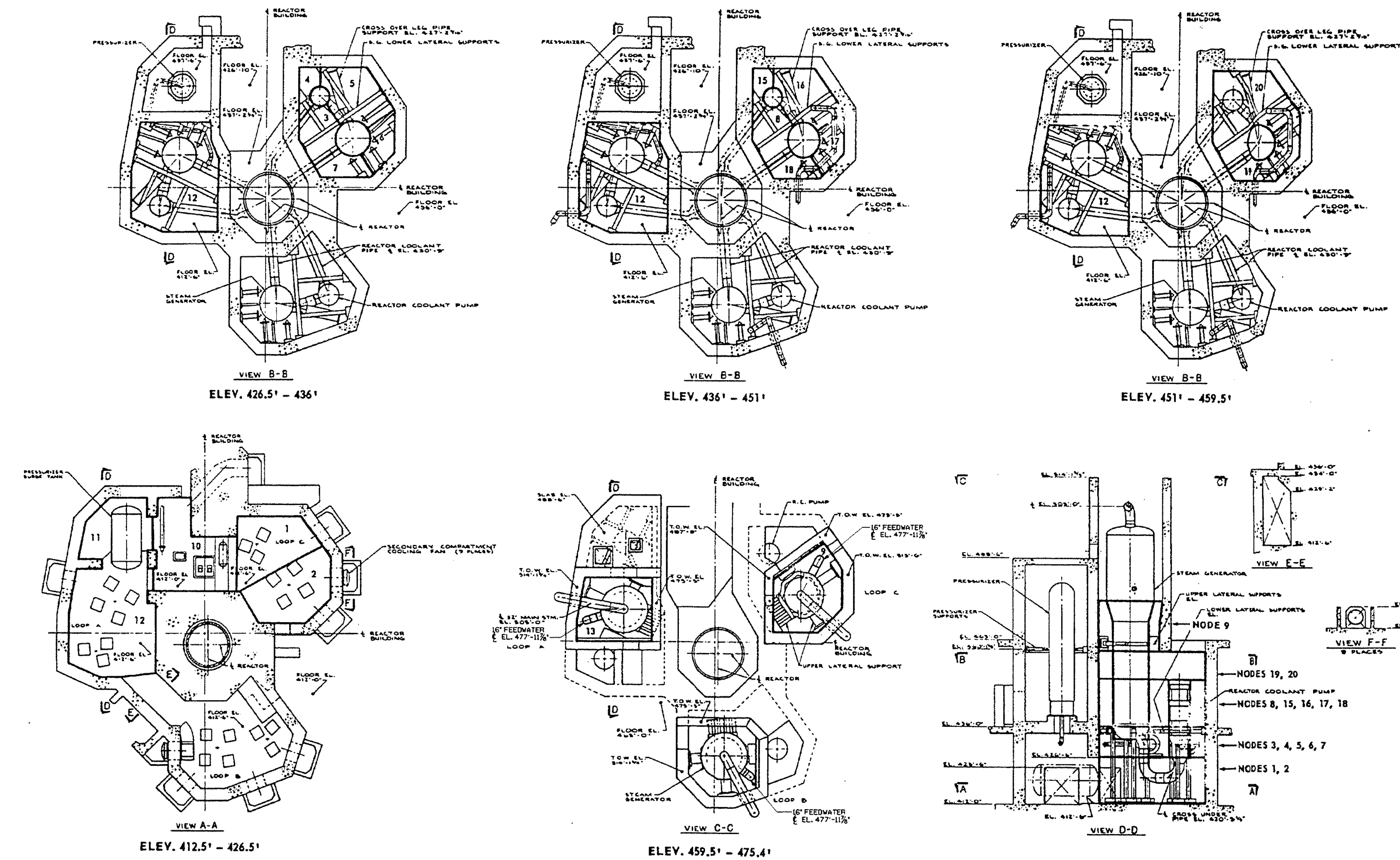

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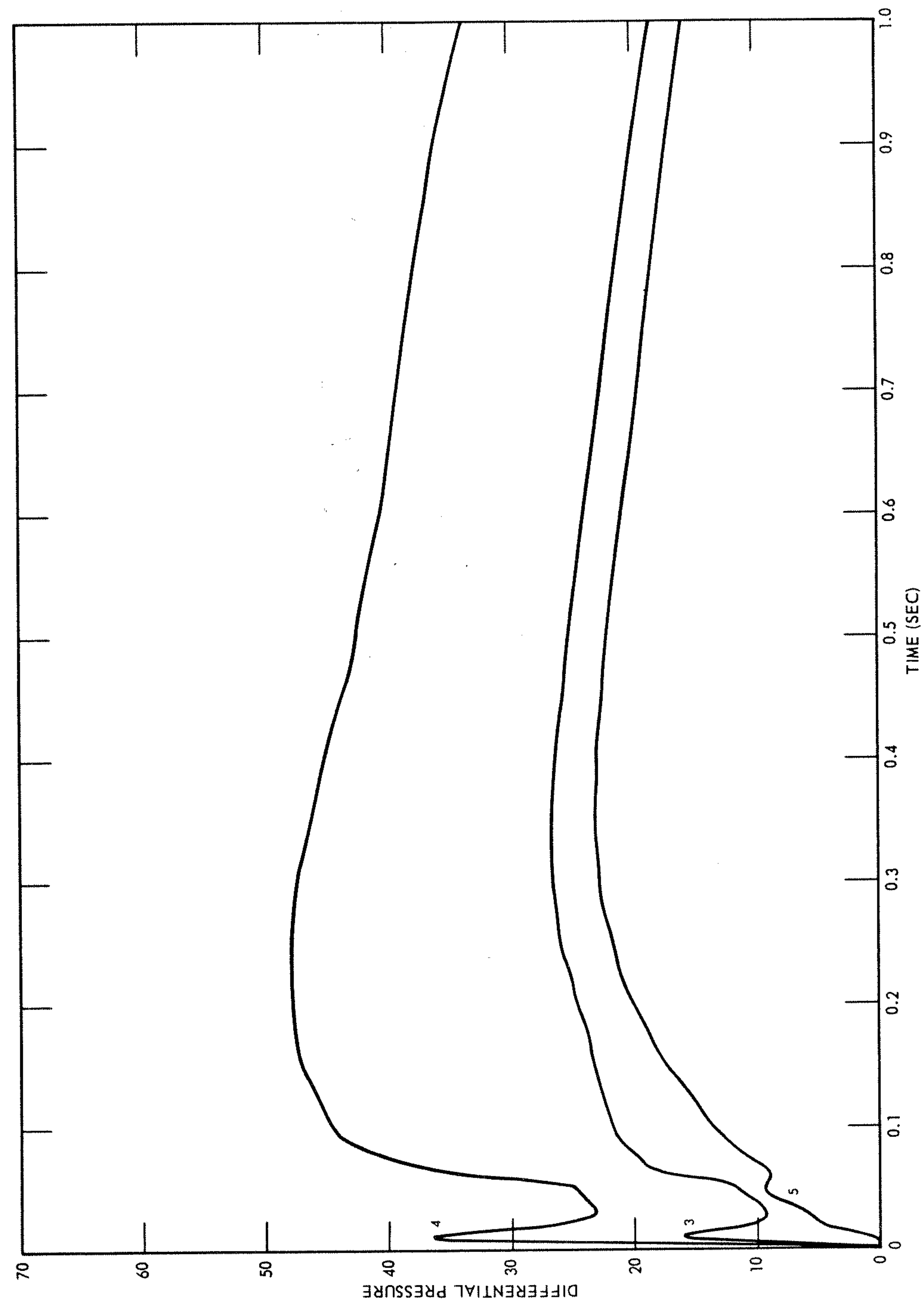
VIRGIL C. SUMMER NUCLEAR STATION

Loop C Steam Generator Compartment Nodal Arrangement 20 Node Model, Cold Leg Break

Figure 6.2-23



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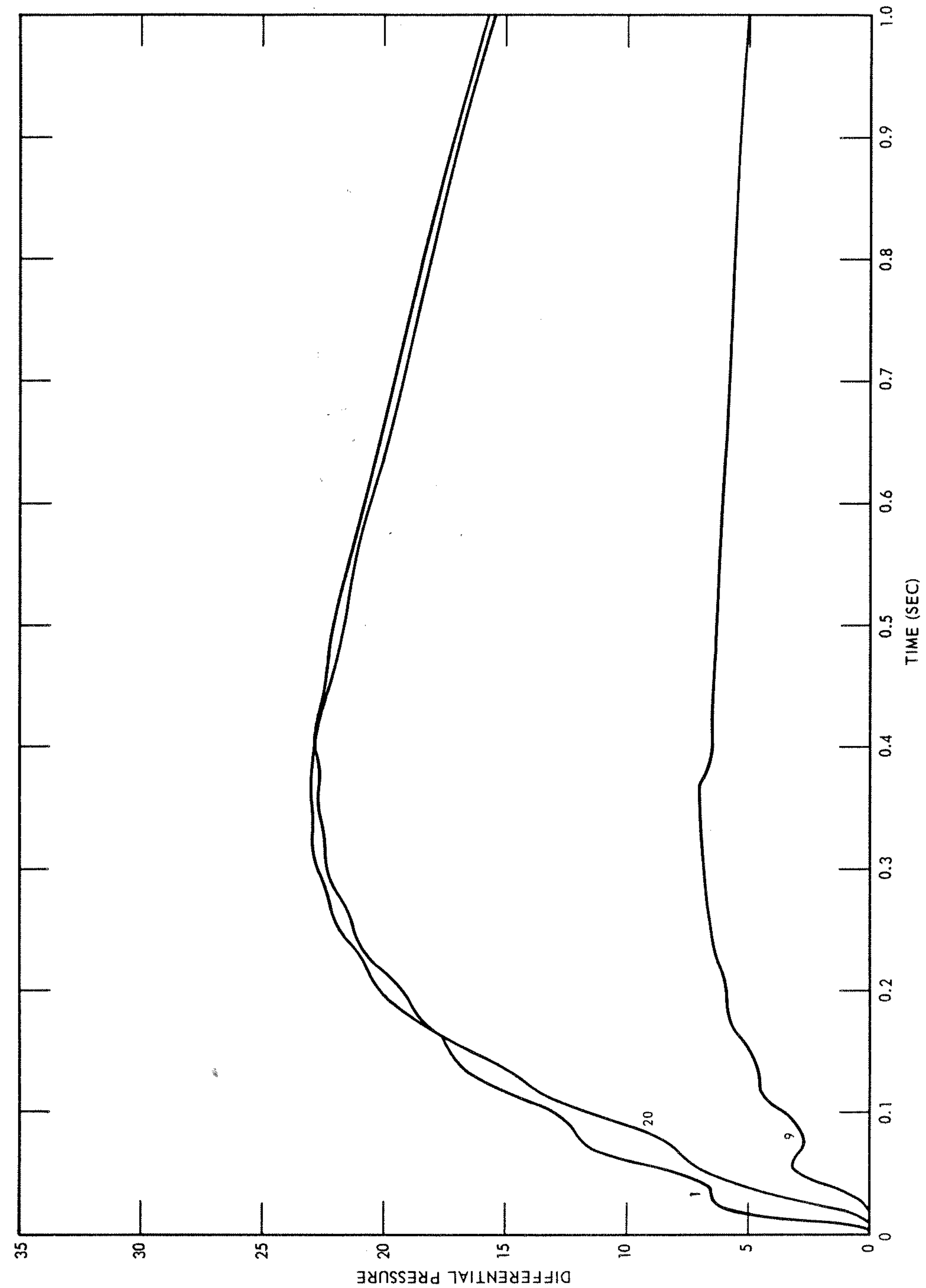
VIRGIL C. SUMMER NUCLEAR STATION

Loop C Steam Generator Compartment
Nodal Pressures Relative to Containment
20 Node Cold Leg Break Model

Figure 6.2-23a



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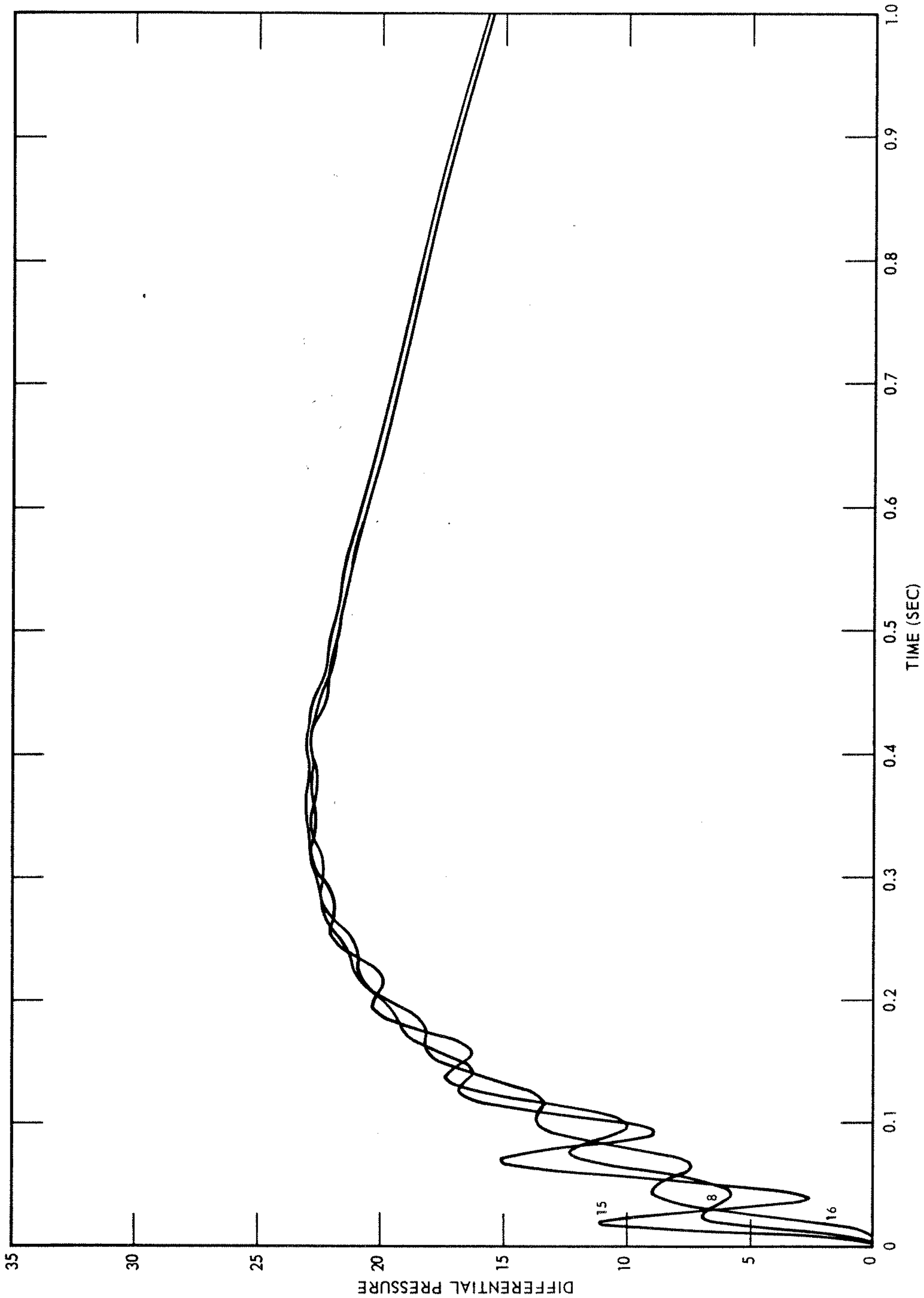
SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Loop C Steam Generator Compartment
Nodal Pressures Relative to Containment
20 Node Cold Leg Break Model

Figure 6.2-23b



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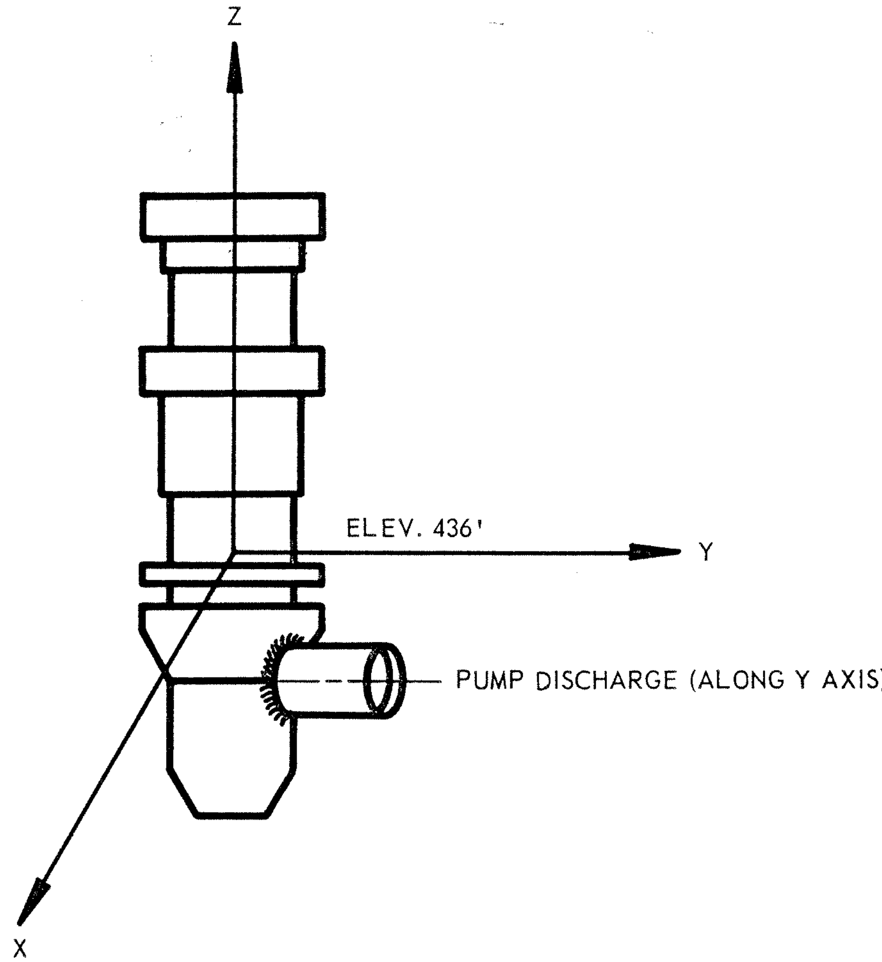
Loop C Steam Generator Compartment
Nodal Pressures Relative to Containment
20 Node Cold Leg Break Model

Figure 6.2-23c



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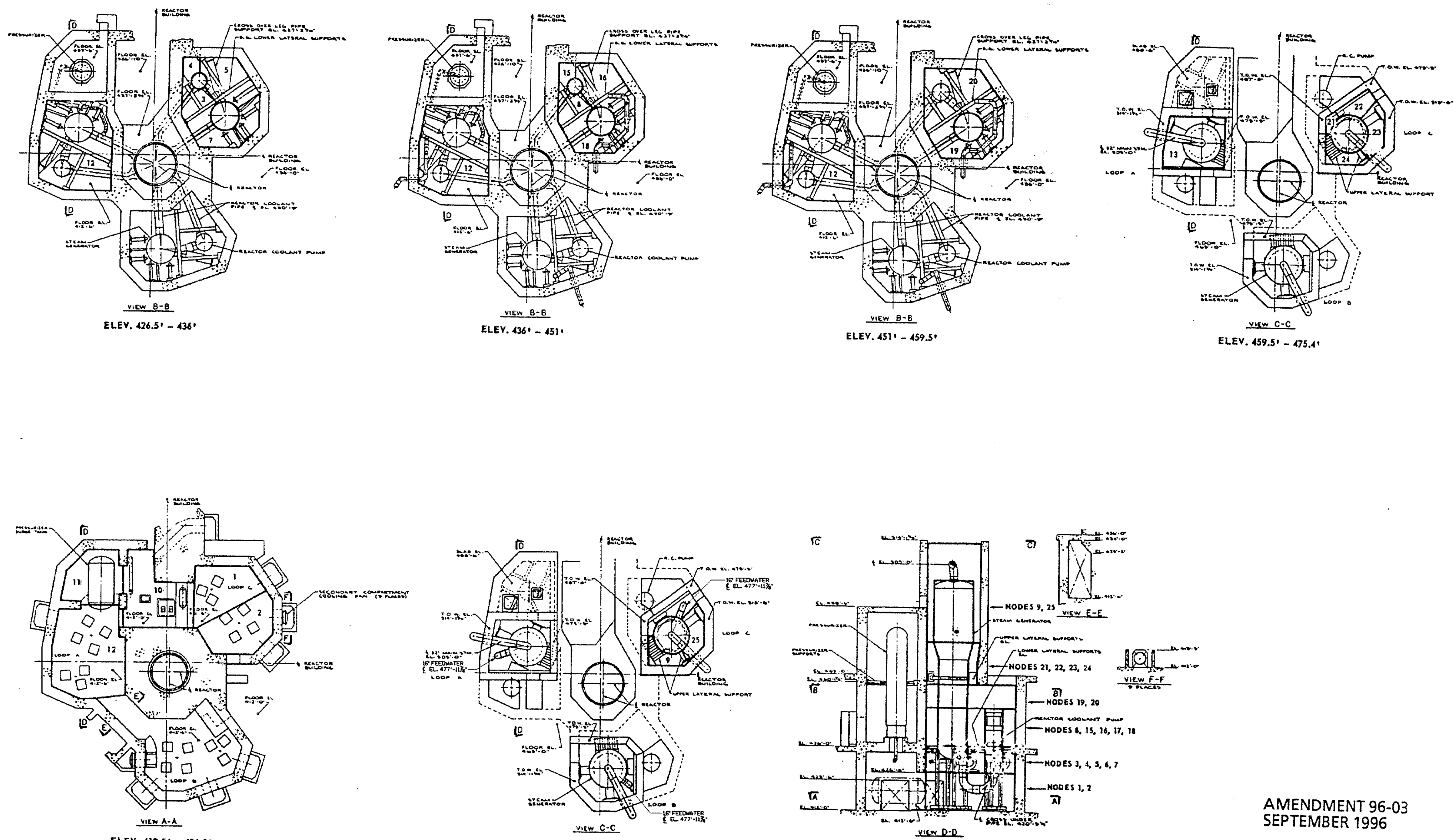
VIRGIL C. SUMMER NUCLEAR STATION

Forces and Moments on Reactor Coolant Pump Coordinate System

Figure 6.2-24



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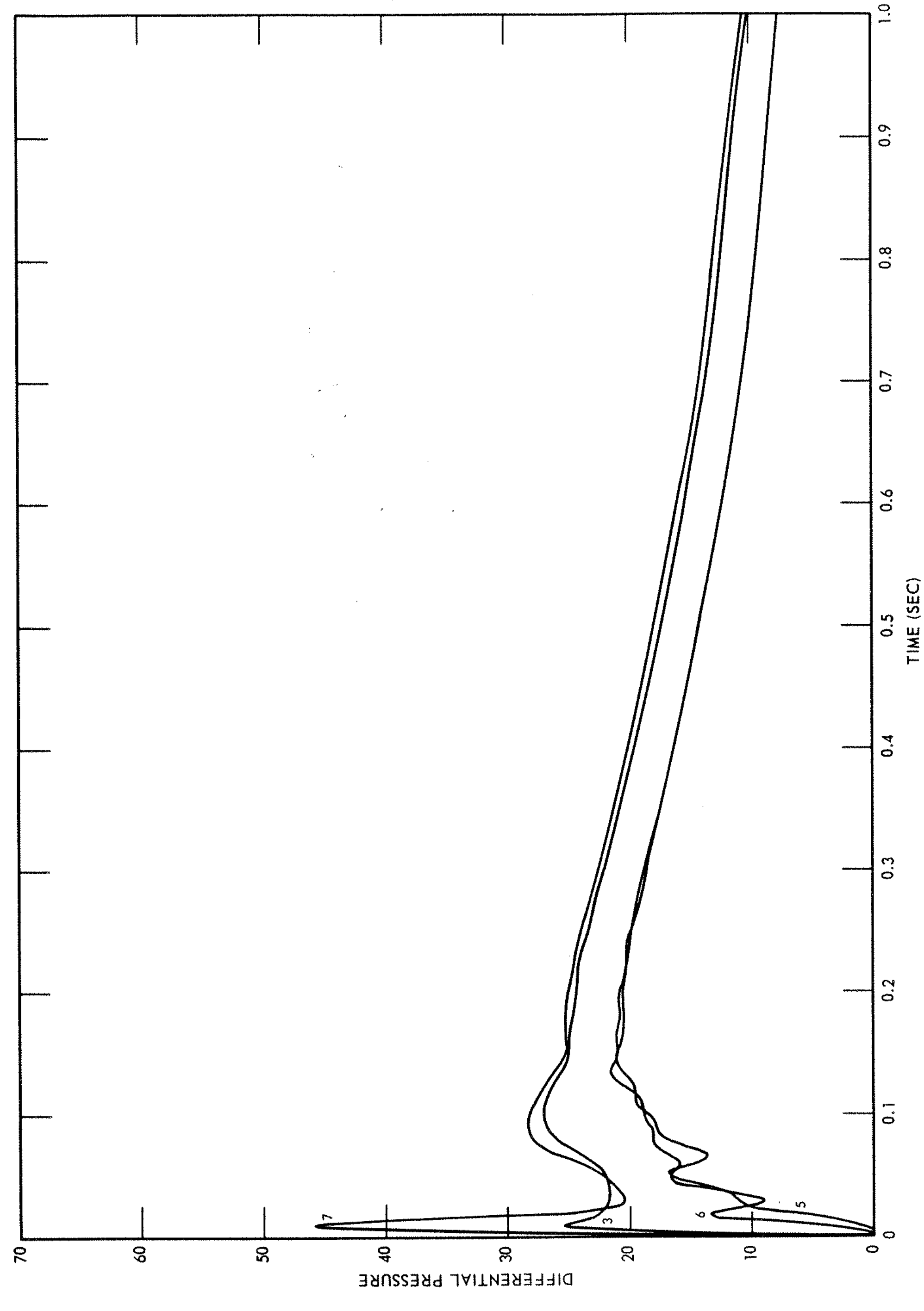
SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Loop C Steam Generator Compartment Nodal Arrangement 25 Node Model, Hot Leg Break

Figure 6.2-25



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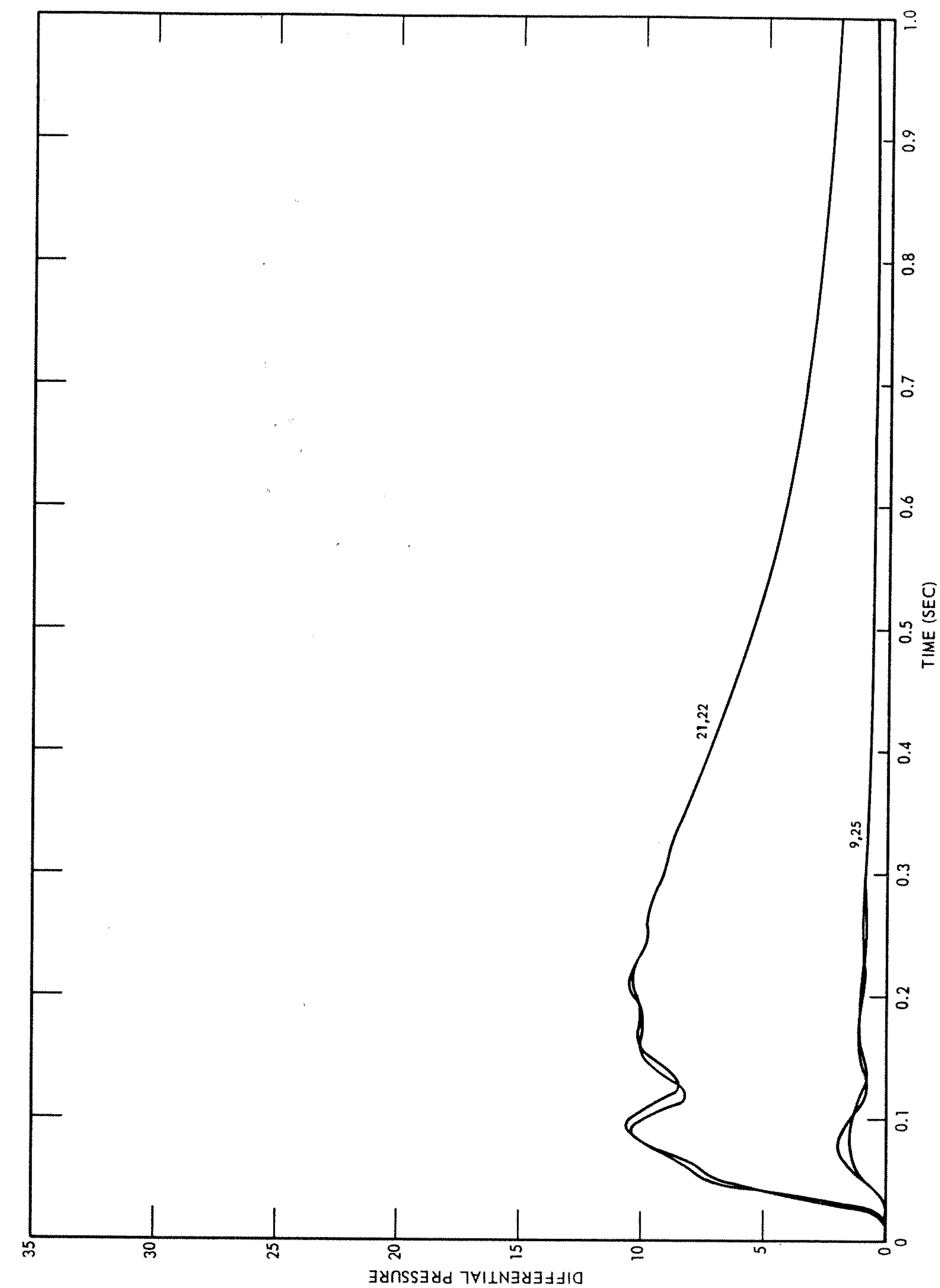
SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Loop C Steam Generator Compartment
Nodal Pressures Relative to Containment
25 Node Hot Leg Break Model

Figure 6.2-25a



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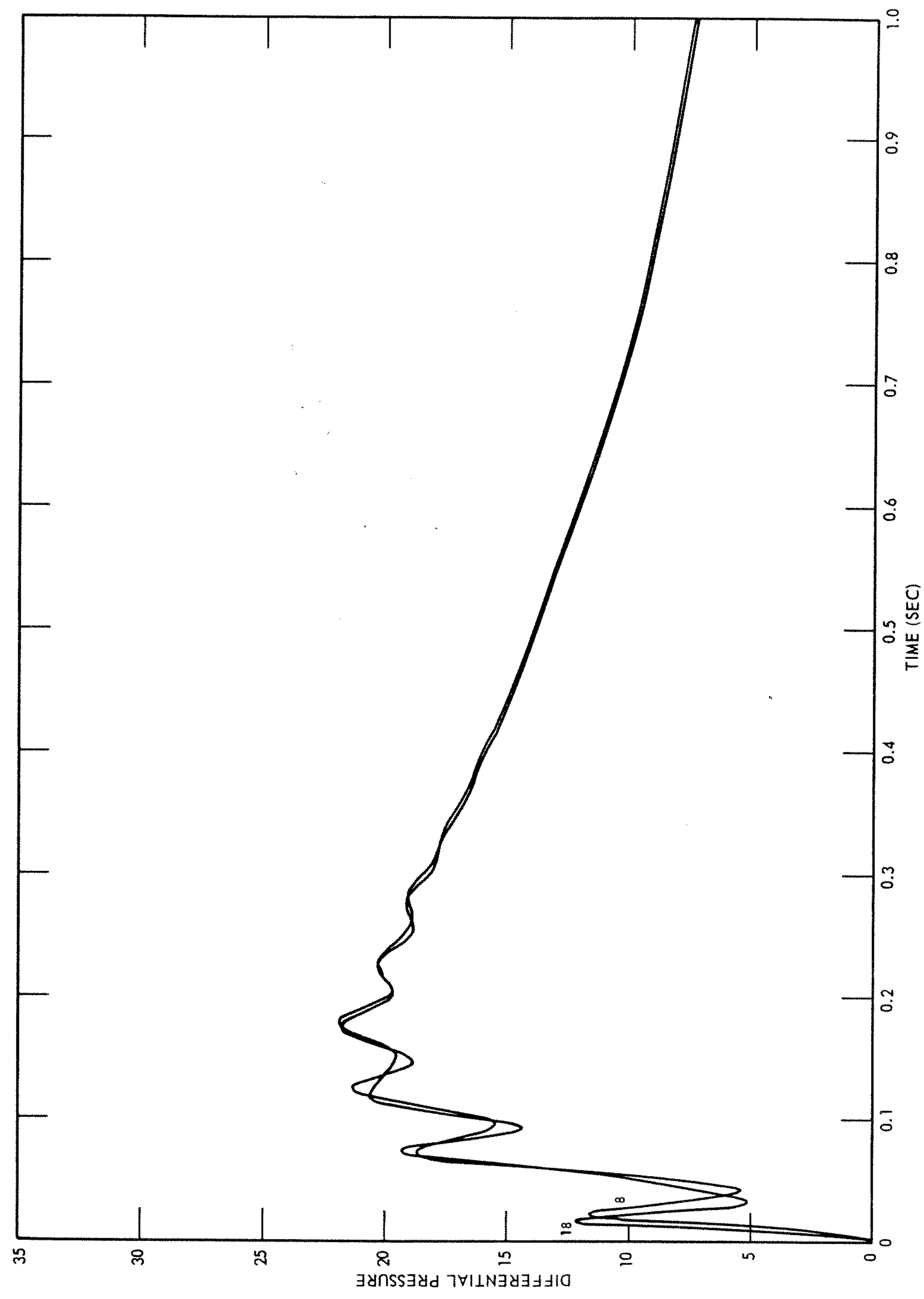
SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Loop C Steam Generator Compartment
Nodal Pressures Relative to Containment
25 Node Hot Leg Break Model

Figure 6.2-25b



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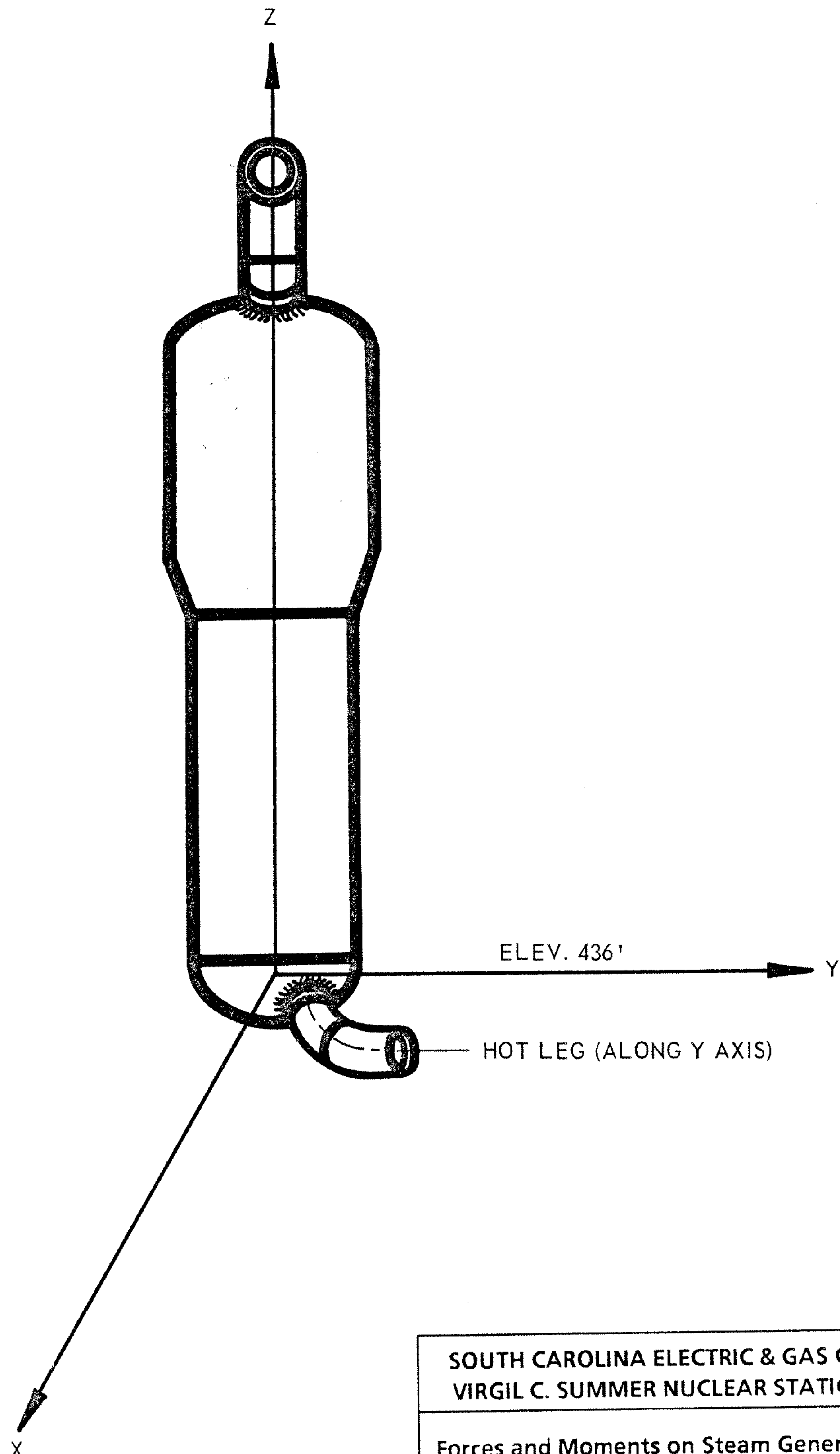
SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Loop C Steam Generator Compartment
Nodal Pressures Relative to Containment
25 Node Hot Leg Break Model

Figure 6.2-25c



SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Forces and Moments on Steam Generator Coordinate System

Figure 6.2-26

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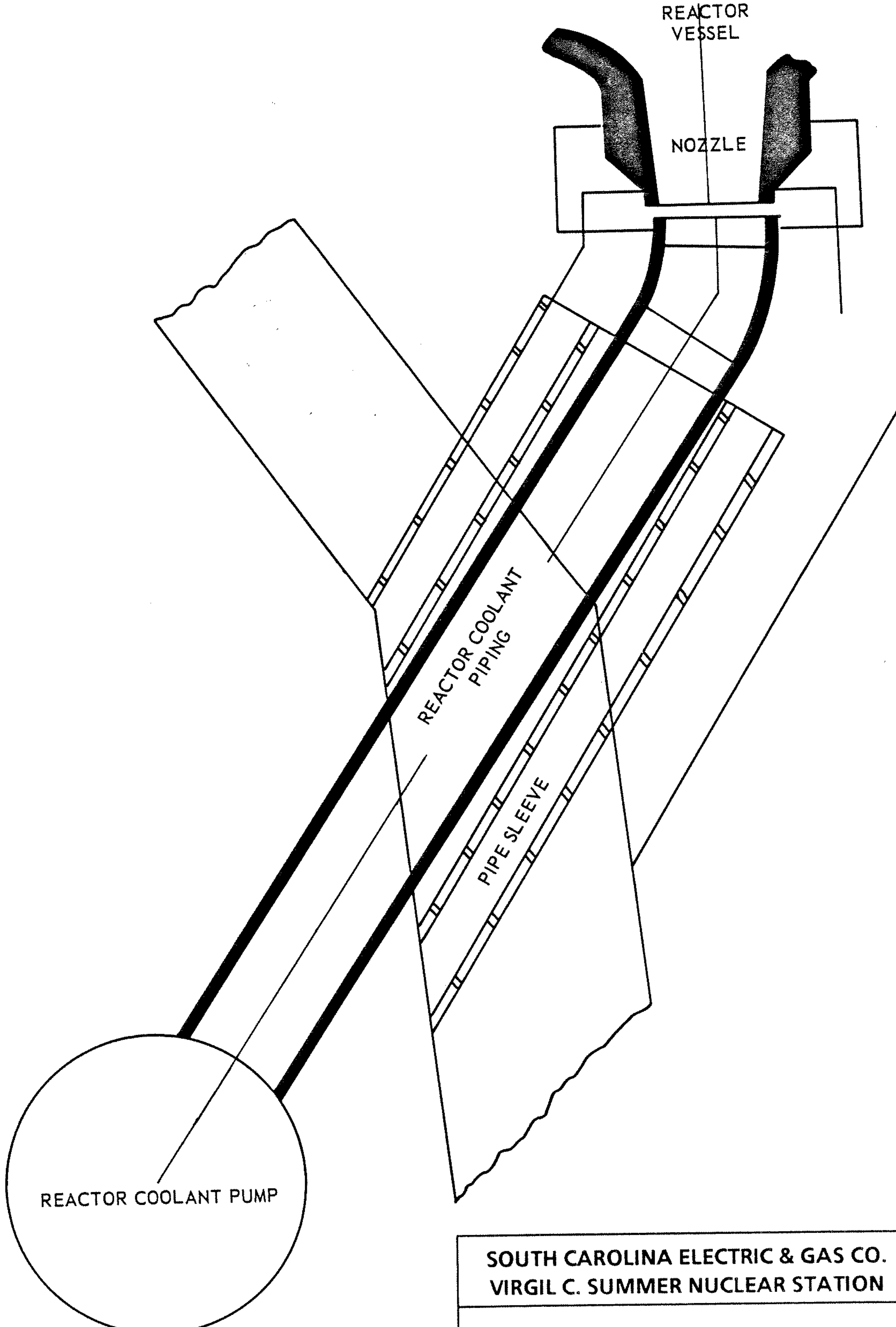
August 1984



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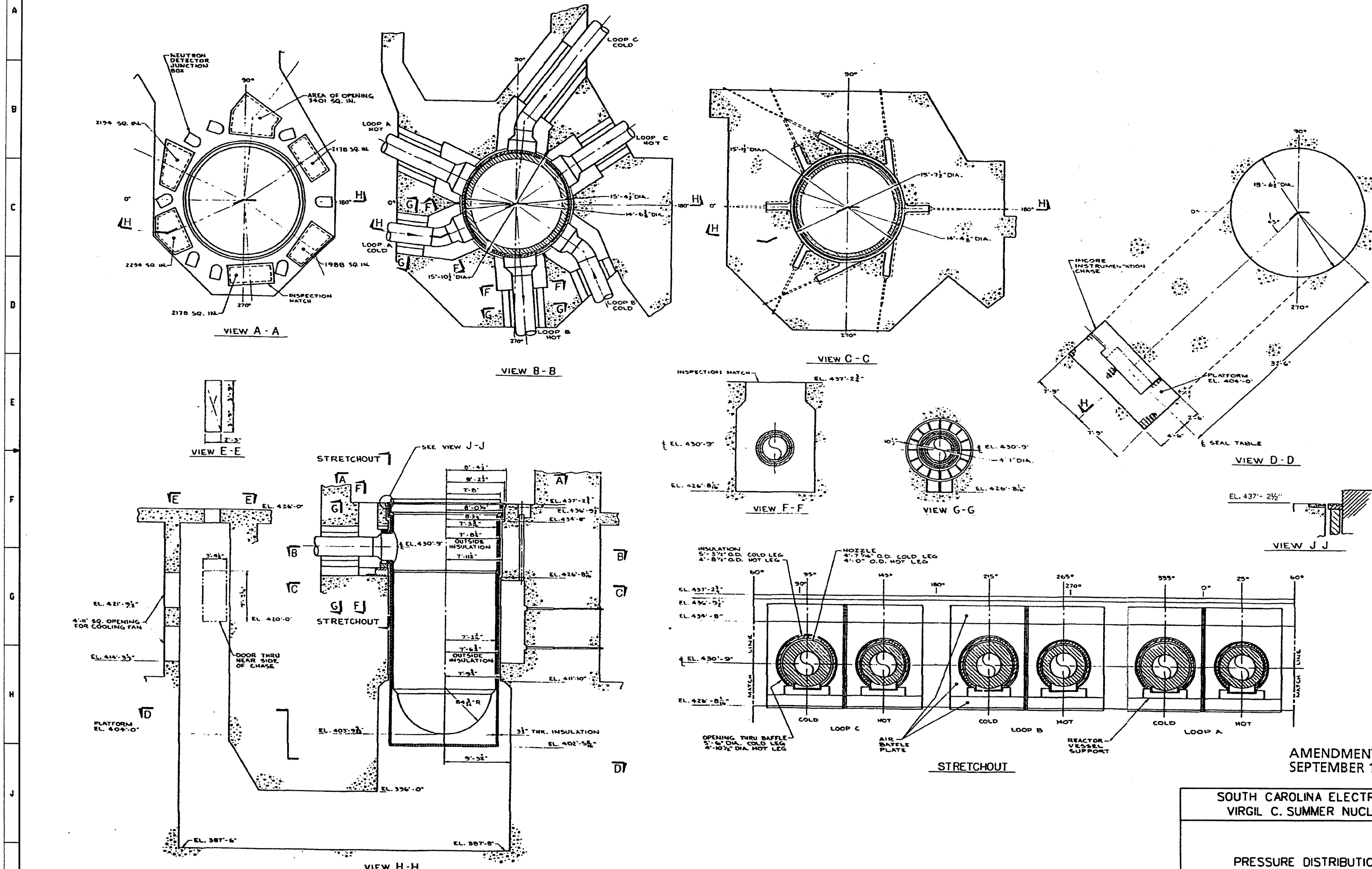
VIRGIL C. SUMMER NUCLEAR STATION

Break Geometry

Figure 6.2-27



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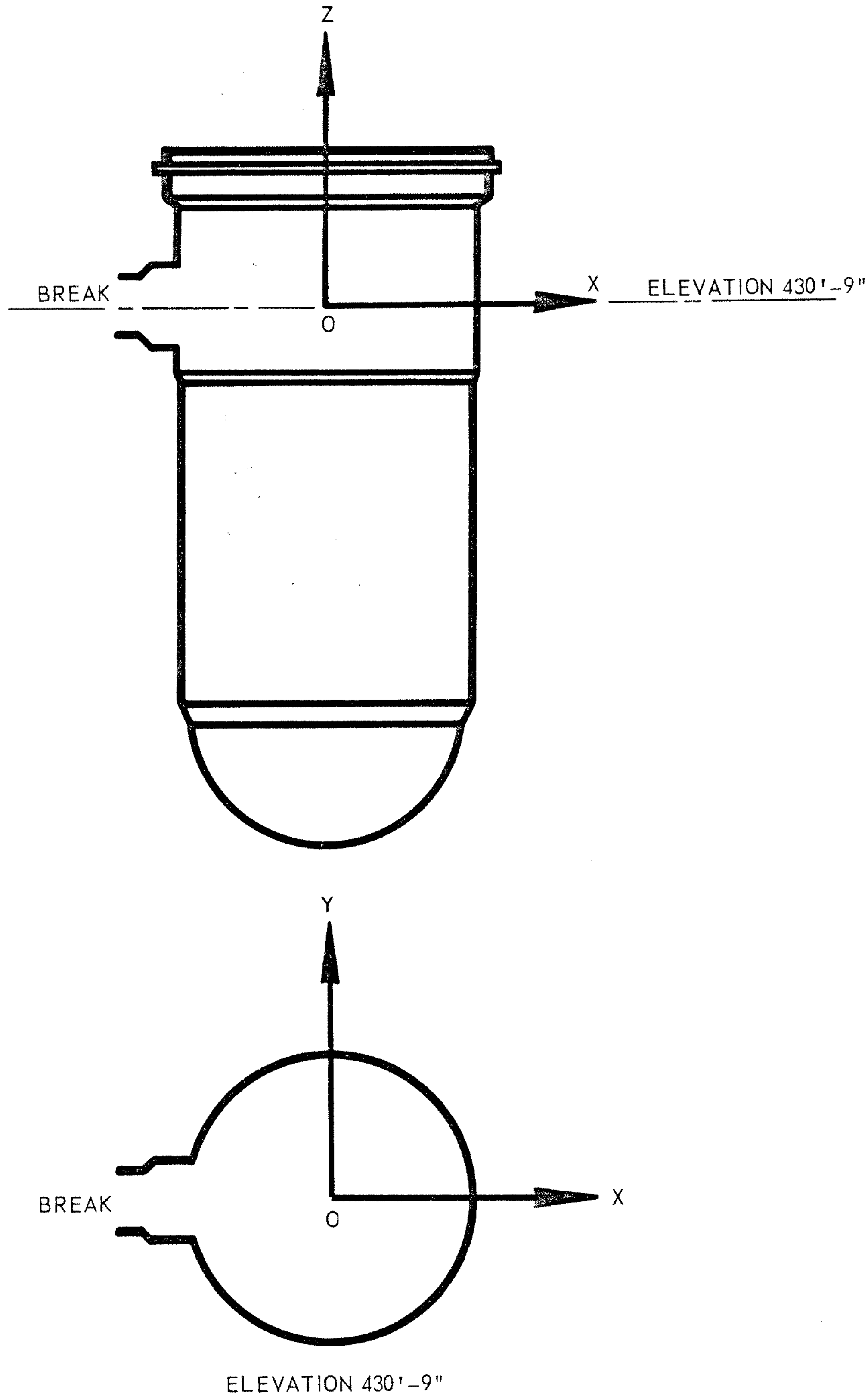
VIRGIL C. SUMMER NUCLEAR STATION

PRESSURE DISTRIBUTION REACTOR VESSEL CAVITY

Figure 6.2-28



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VIRGIL C. SUMMER NUCLEAR STATION

Reactor Vessel Axes for Forces and Moments

Figure 6.2-29

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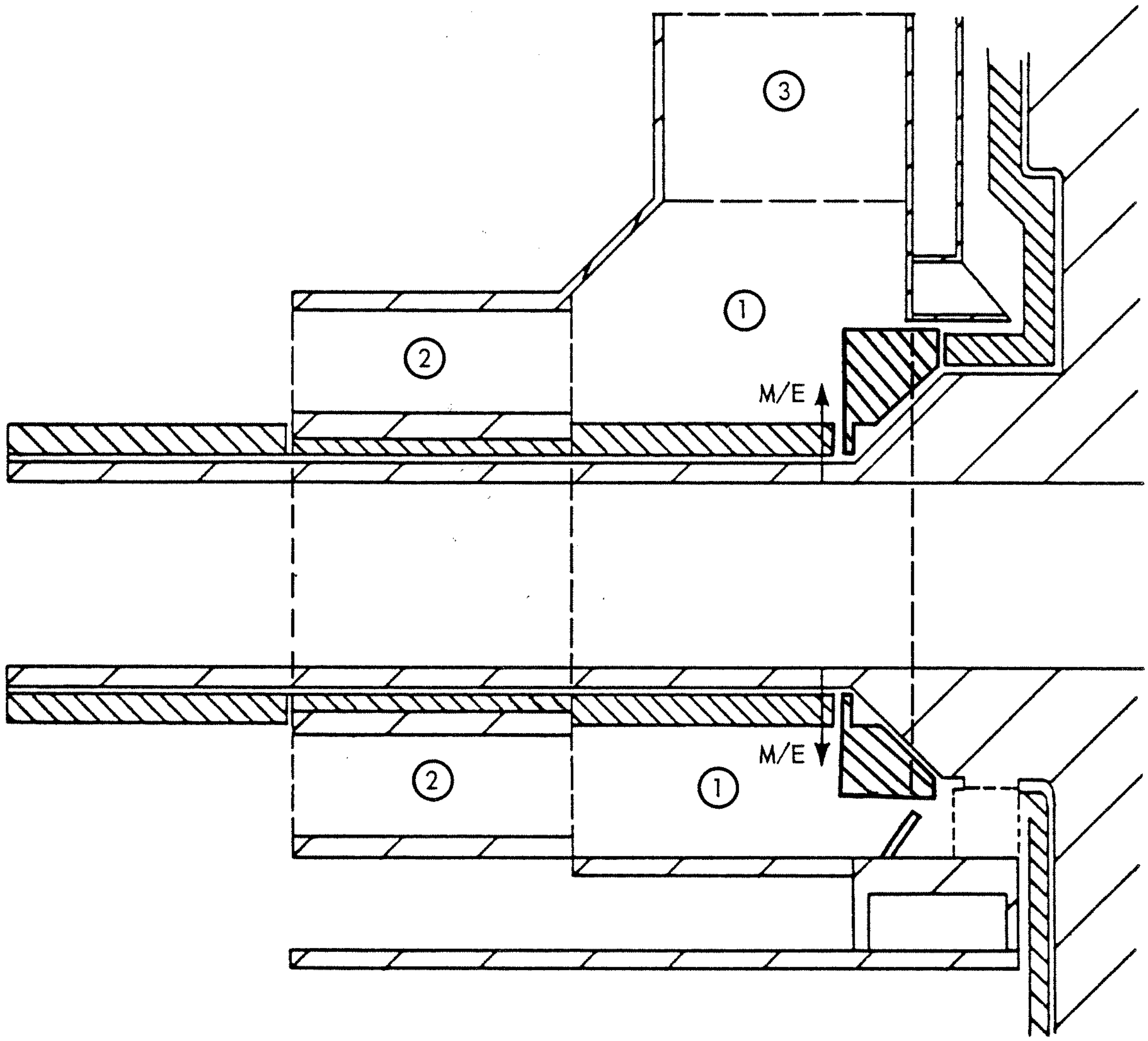

- CONTROL VOLUME NUMBER

- STEEL

- INSULATION

M/E - MASS/ENERGY RELEASES

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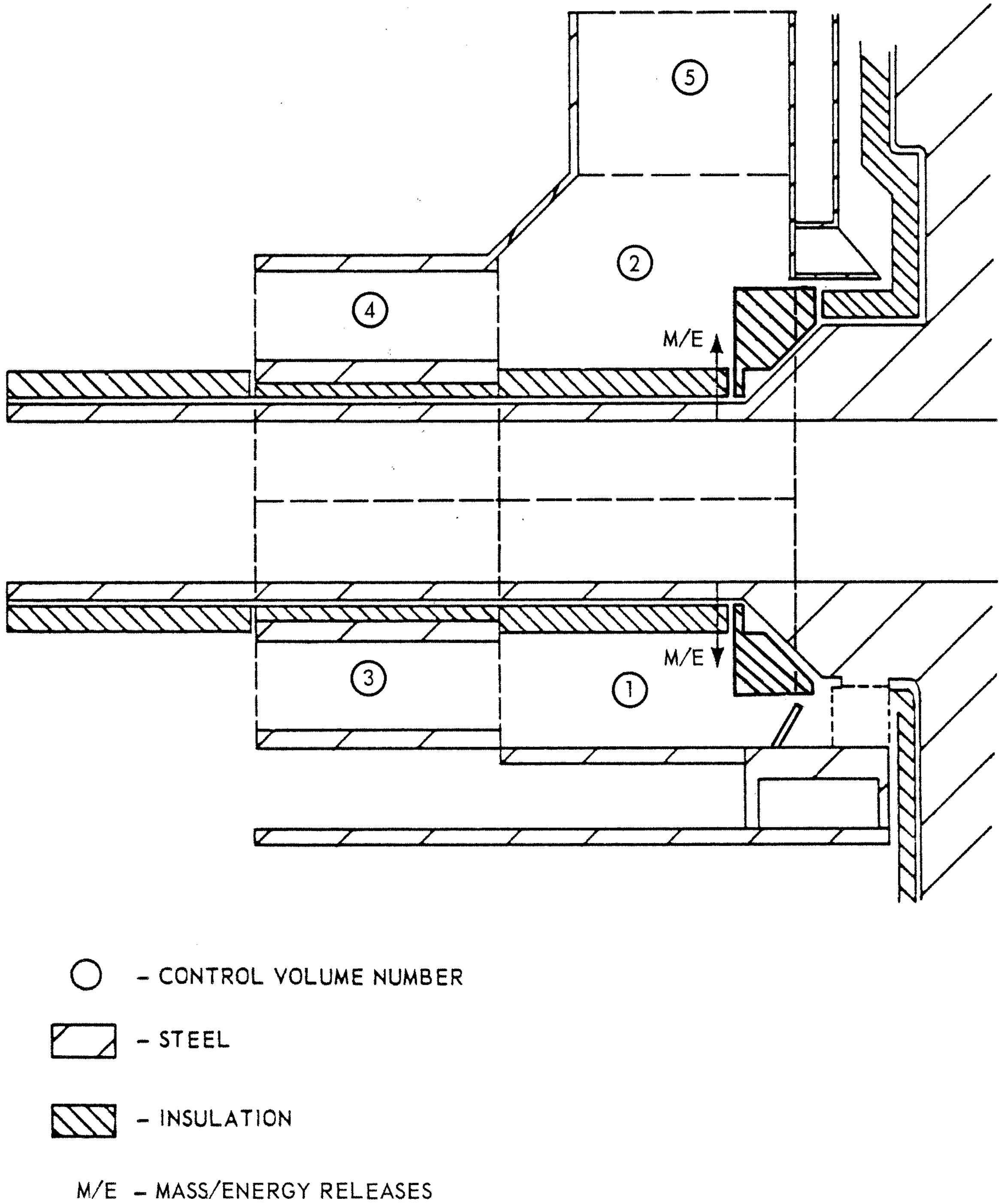
VIRGIL C. SUMMER NUCLEAR STATION

Penetration Nodes for 22 and 31 Node Reactor Cavity Models

Figure 6.2-30



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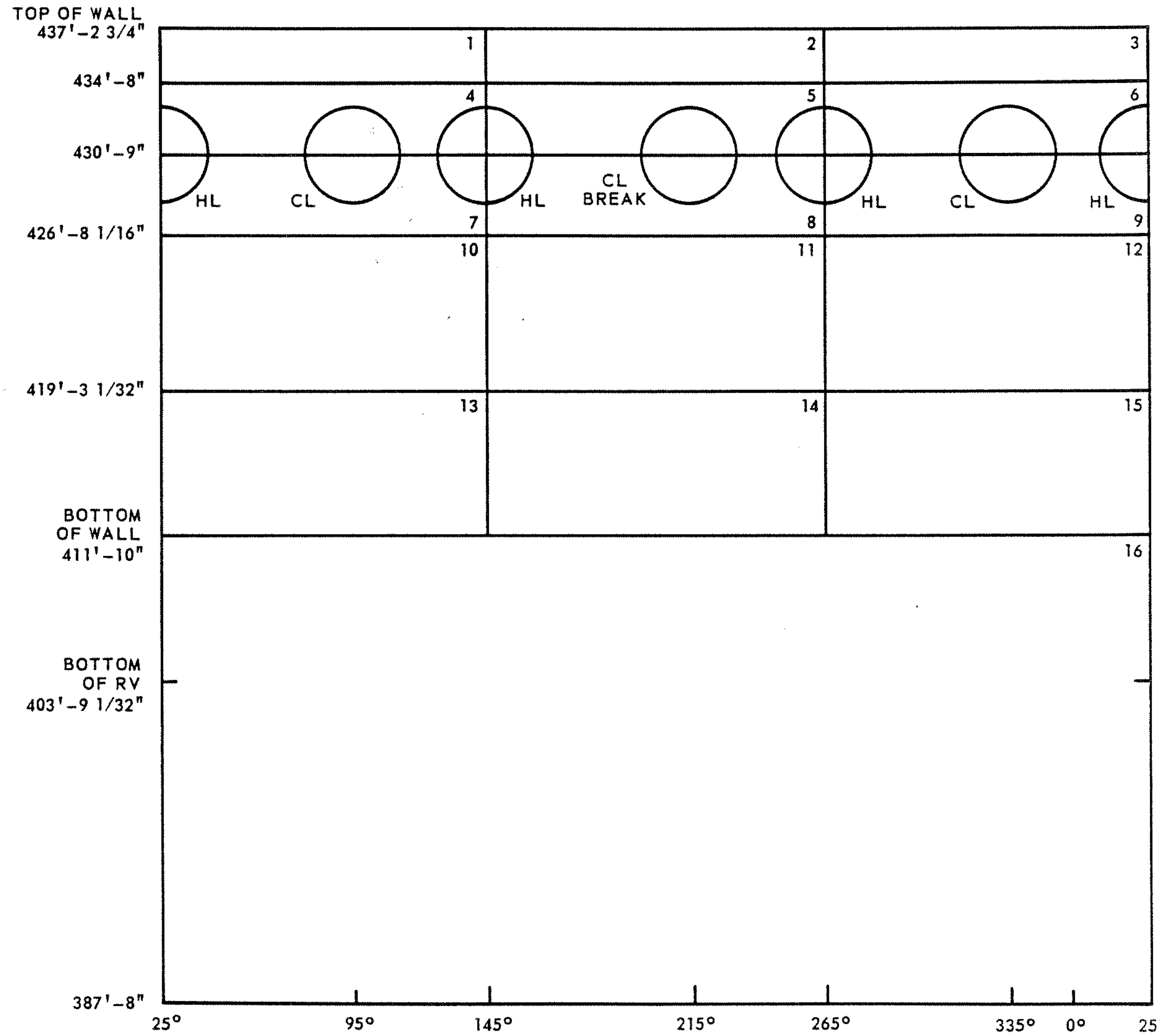
Penetration Nodes for 33 Node Reactor Cavity Modell

Figure 6.2-30a

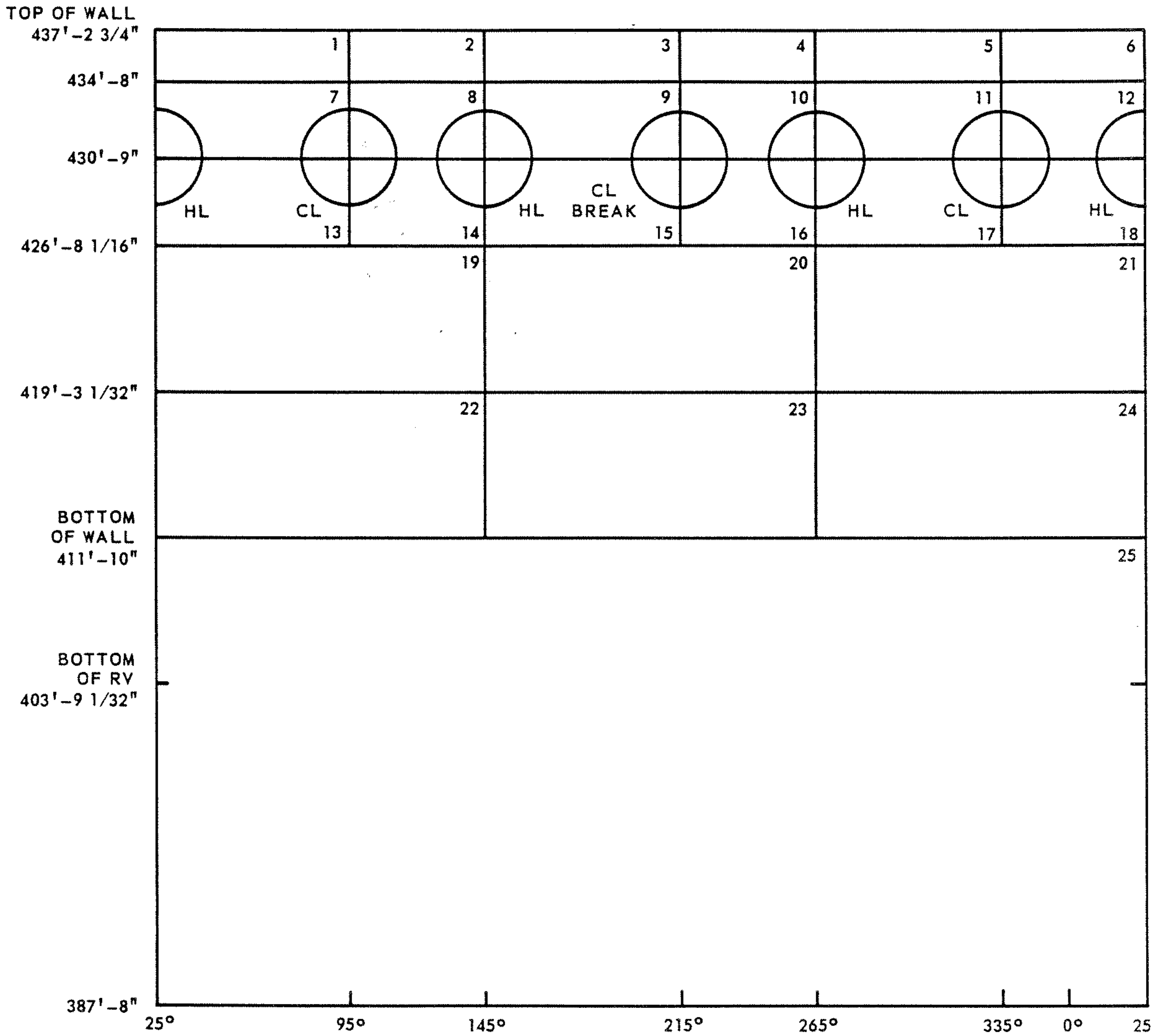


SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Reactor Vessel Annulus Nodes for
22 Node Reactor Cavity Model

Figure 6.2-31

Amendment 0
August 1984

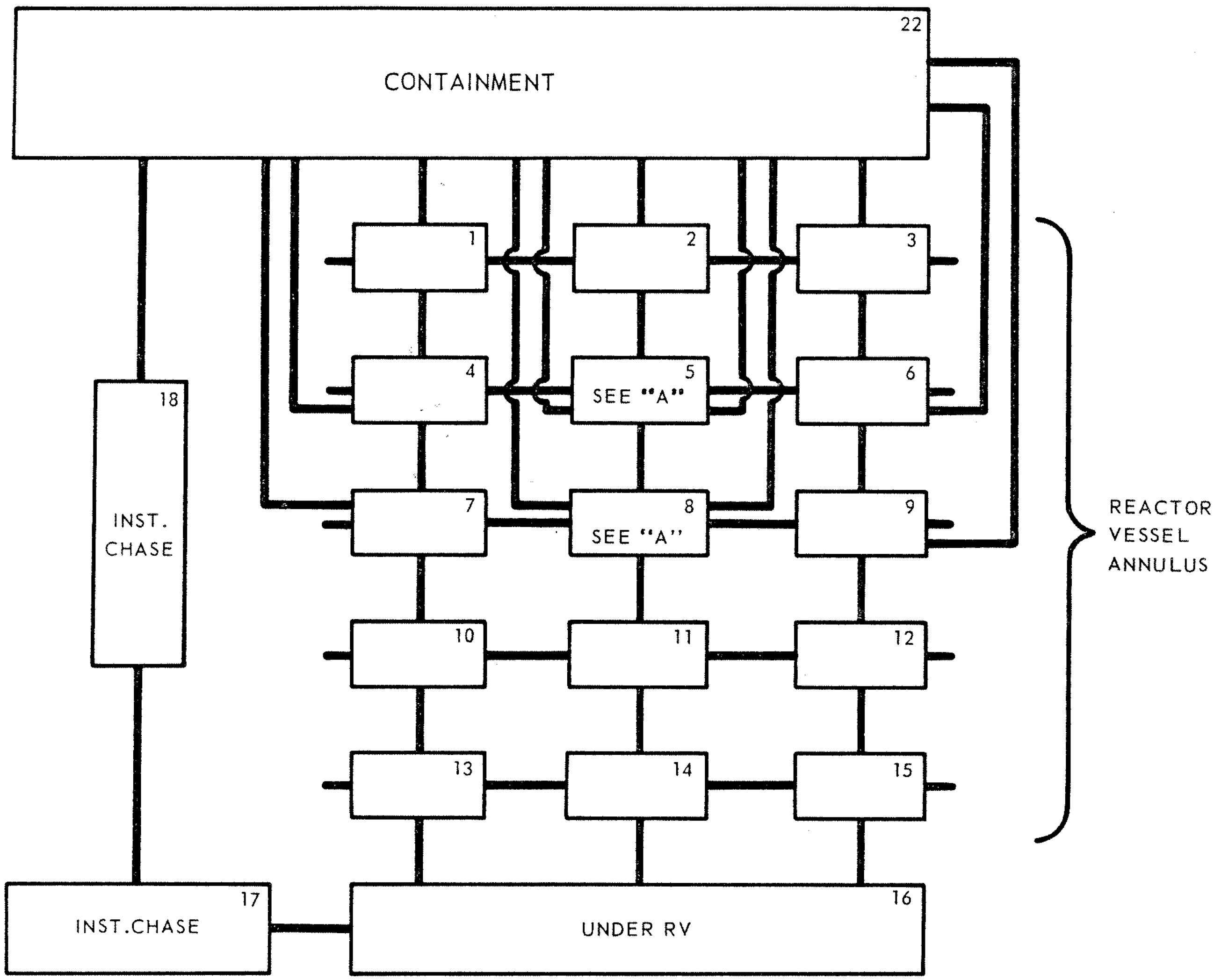


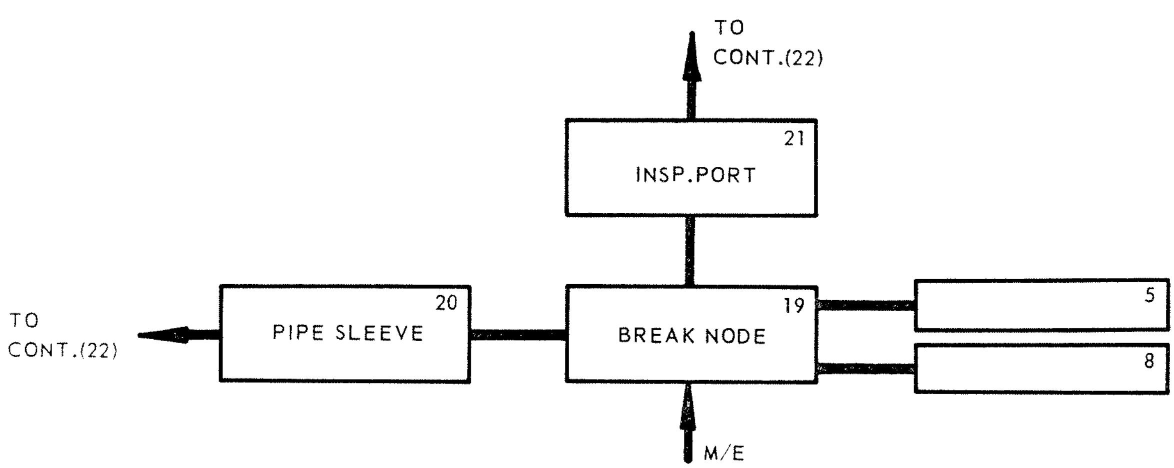
Amendment 0
August 1984

SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Reactor Vessel Annulus Nodes for 31 and 33 Node Reactor Cavity Model

Figure 6.2-31a





DETAIL "A"

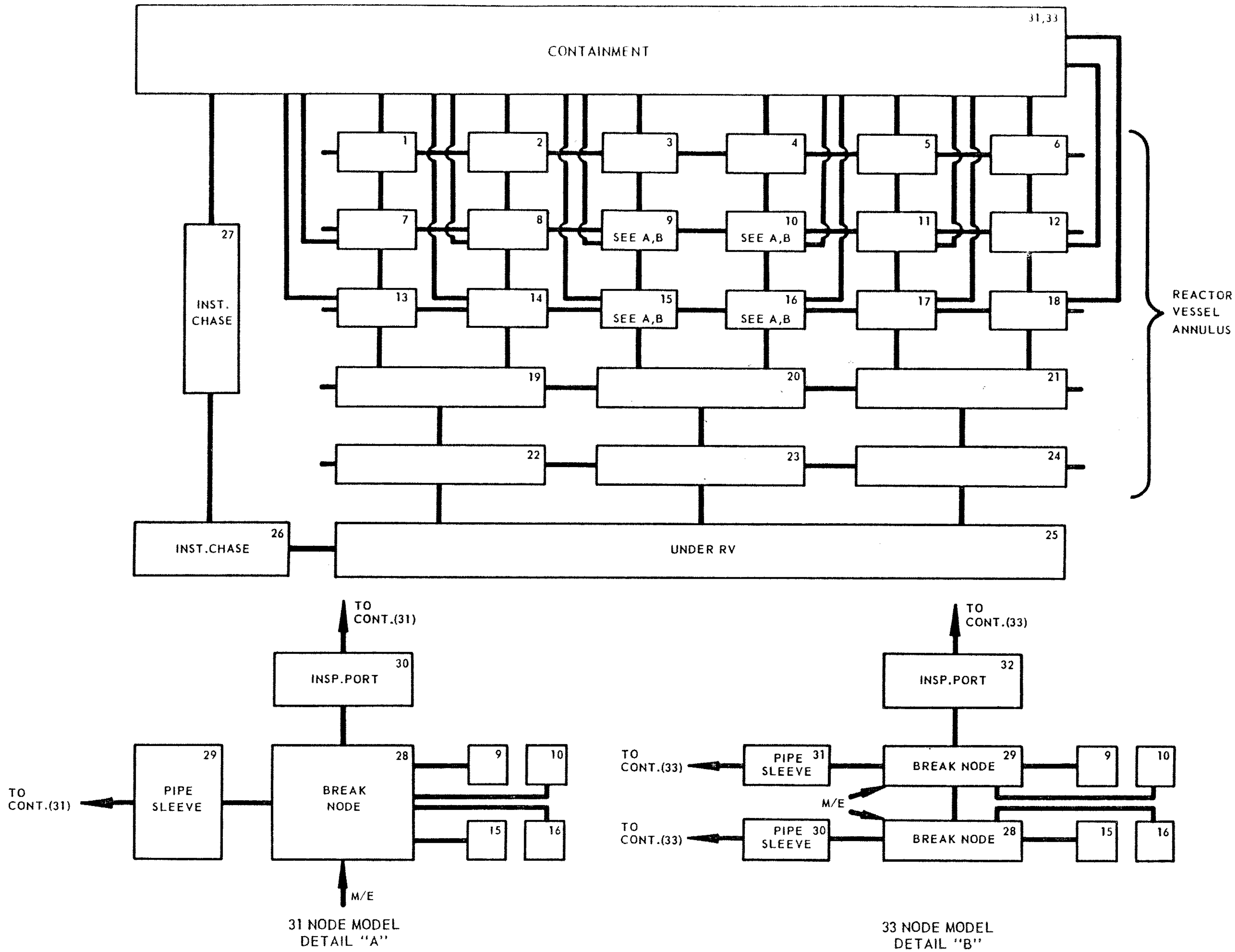
SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

22 Node Reactor Cavity Model

Figure 6.2-32

Amendment 0
August 1984

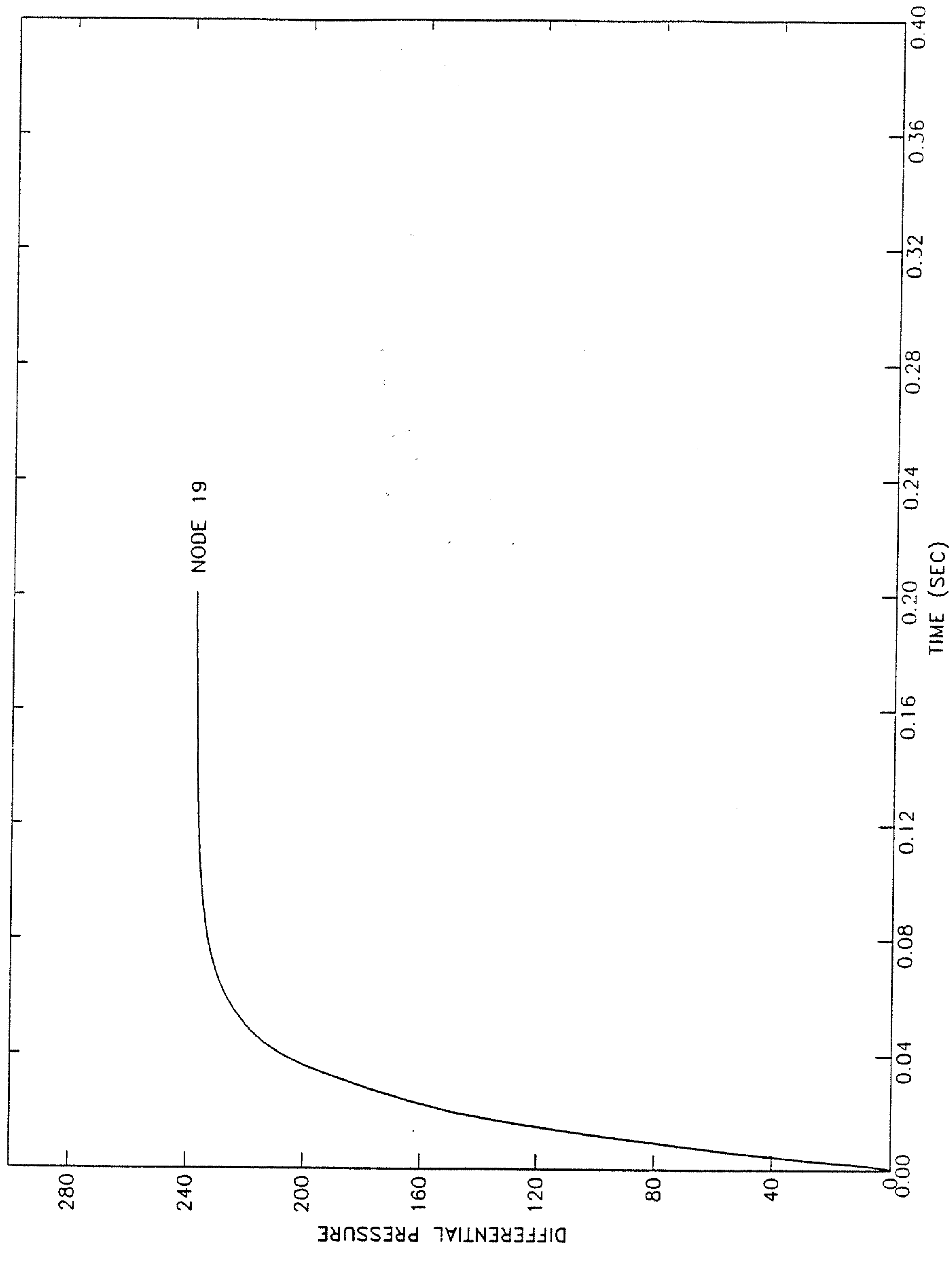
Amendment 0
August 1984



SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

31 and 32 Node Reactor
Cavity Model

Figure 6.2-32a

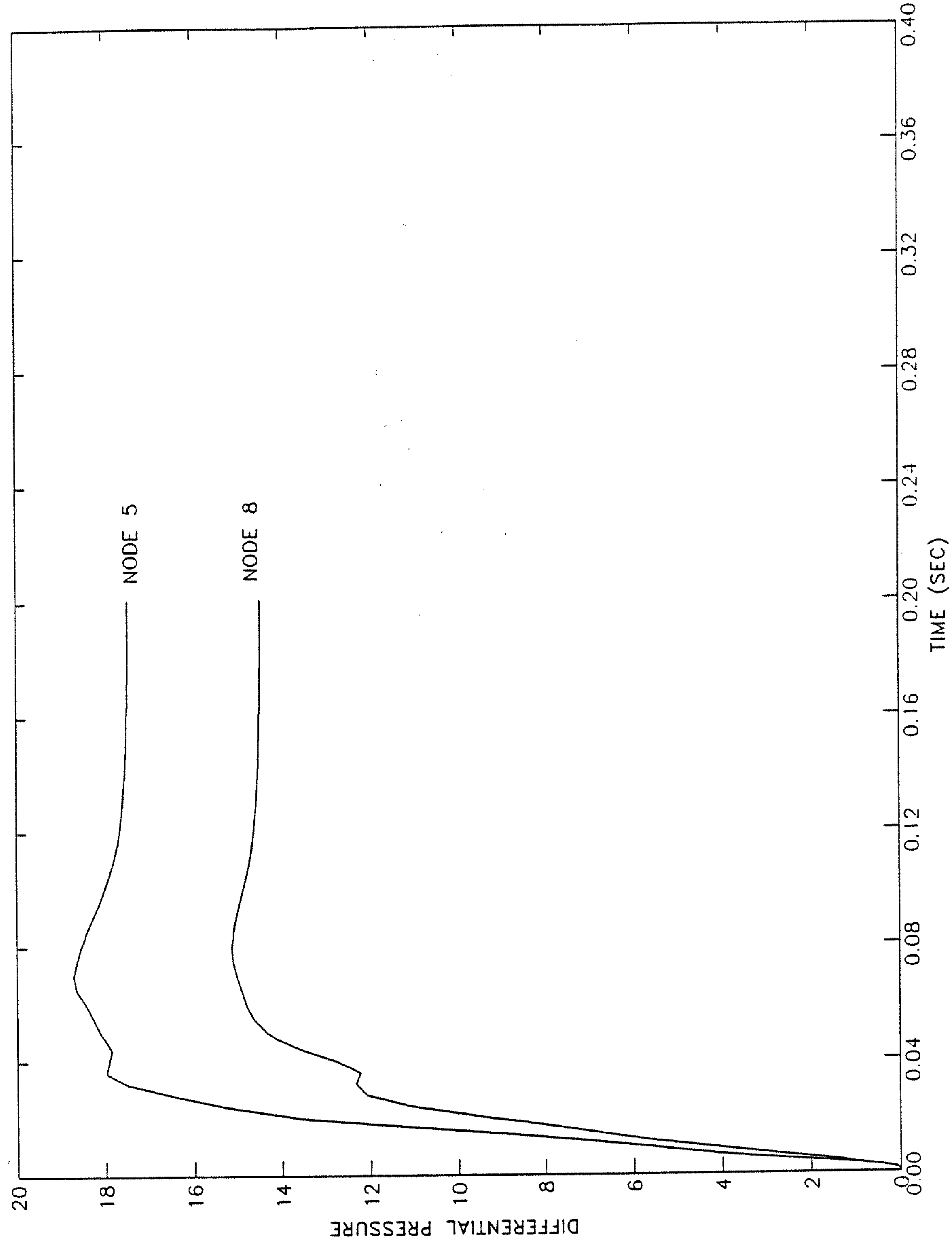


AMENDMENT 5
AUGUST, 1989

SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

22 Node Reactor Cavity Model - Break Node Differential Pressure Versus Time

Figure 6.2-33

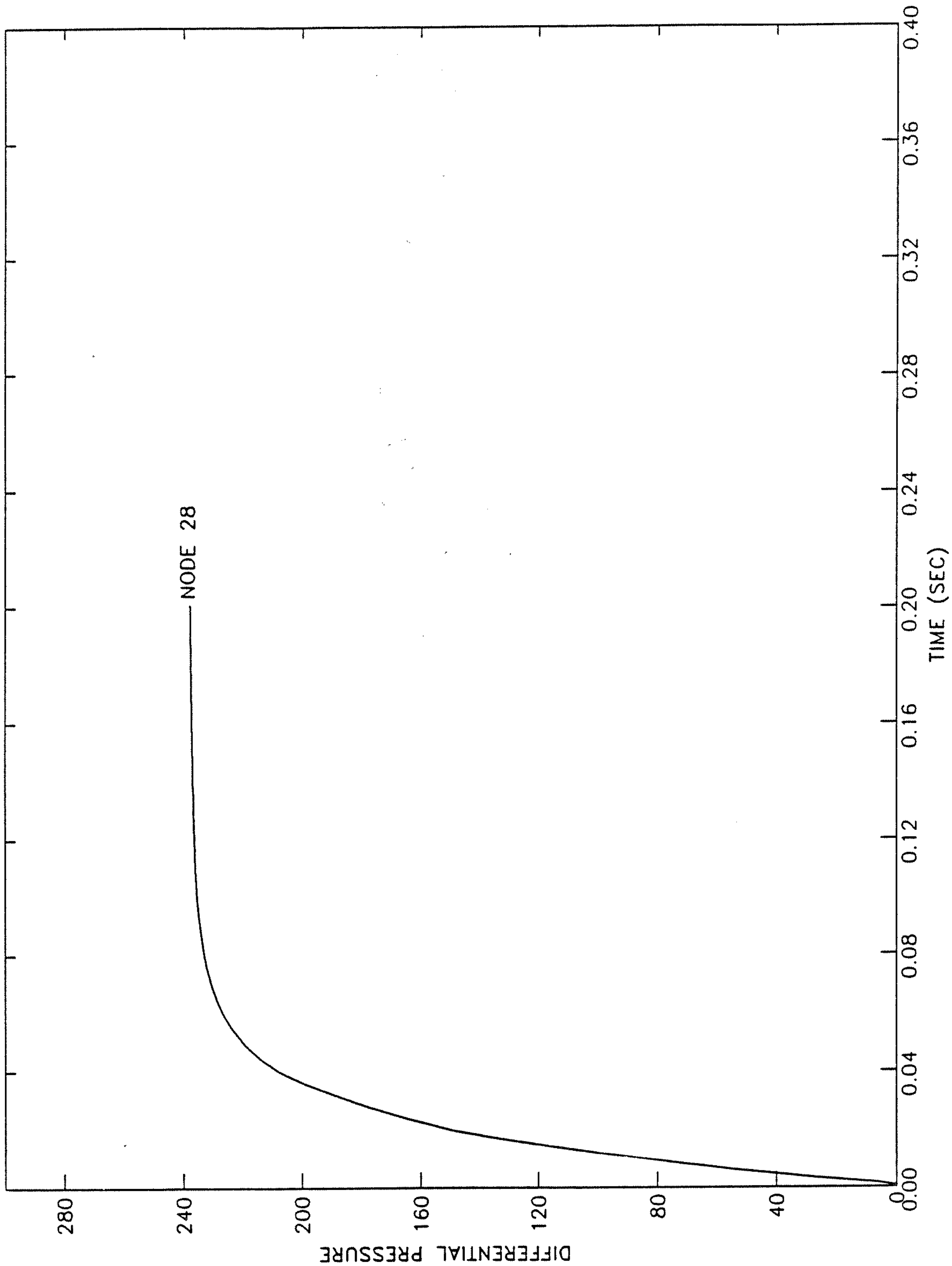


AMENDMENT 5
AUGUST, 1989

SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

22 Node Reactor Cavity Model - Differential Pressure at Selected Nodes Versus Time

Figure 6.2-33a

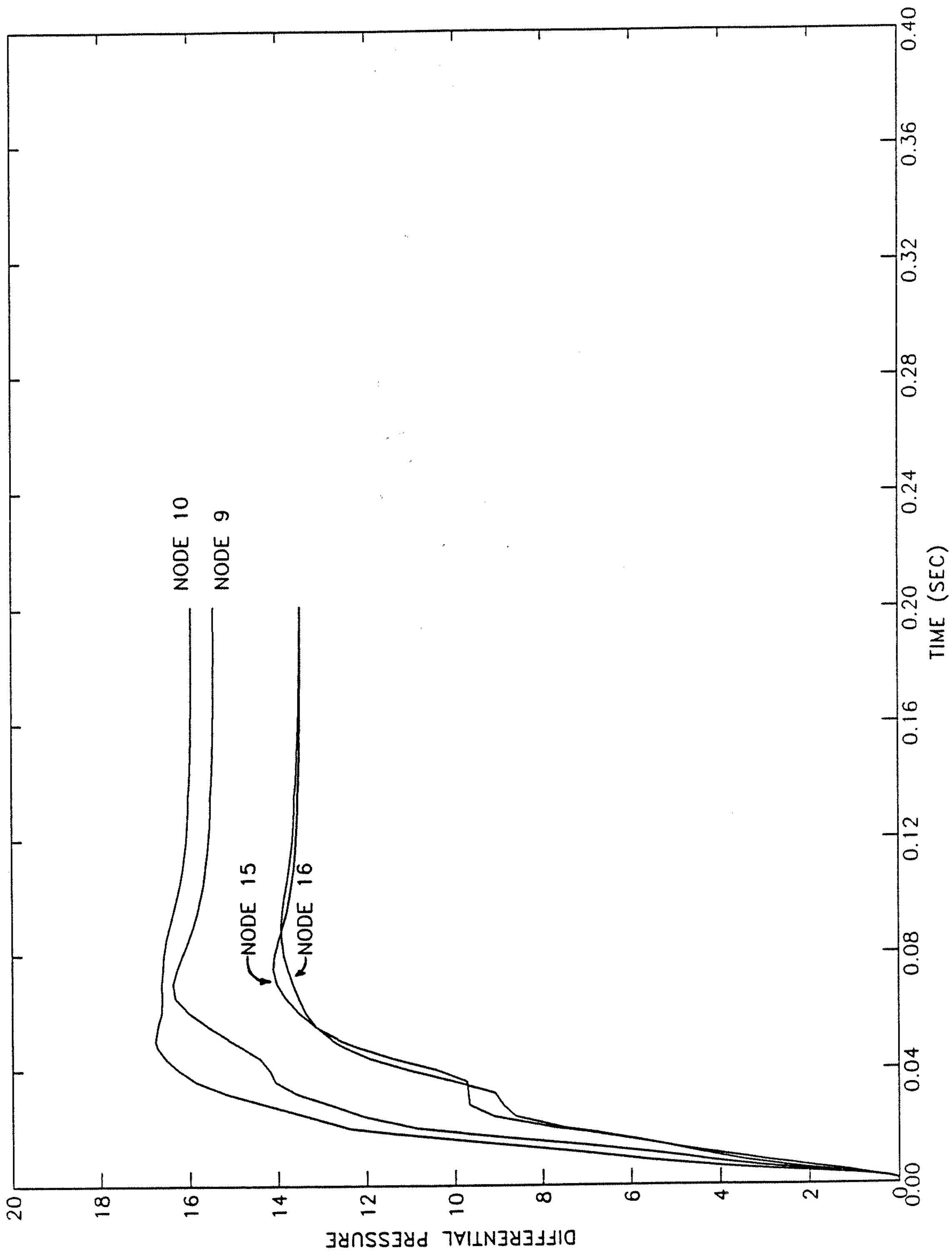


AMENDMENT 5
AUGUST, 1989

SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

31 Node Reactor Cavity Model - Break Node Differential Pressure Versus Time

Figure 6.2-33b

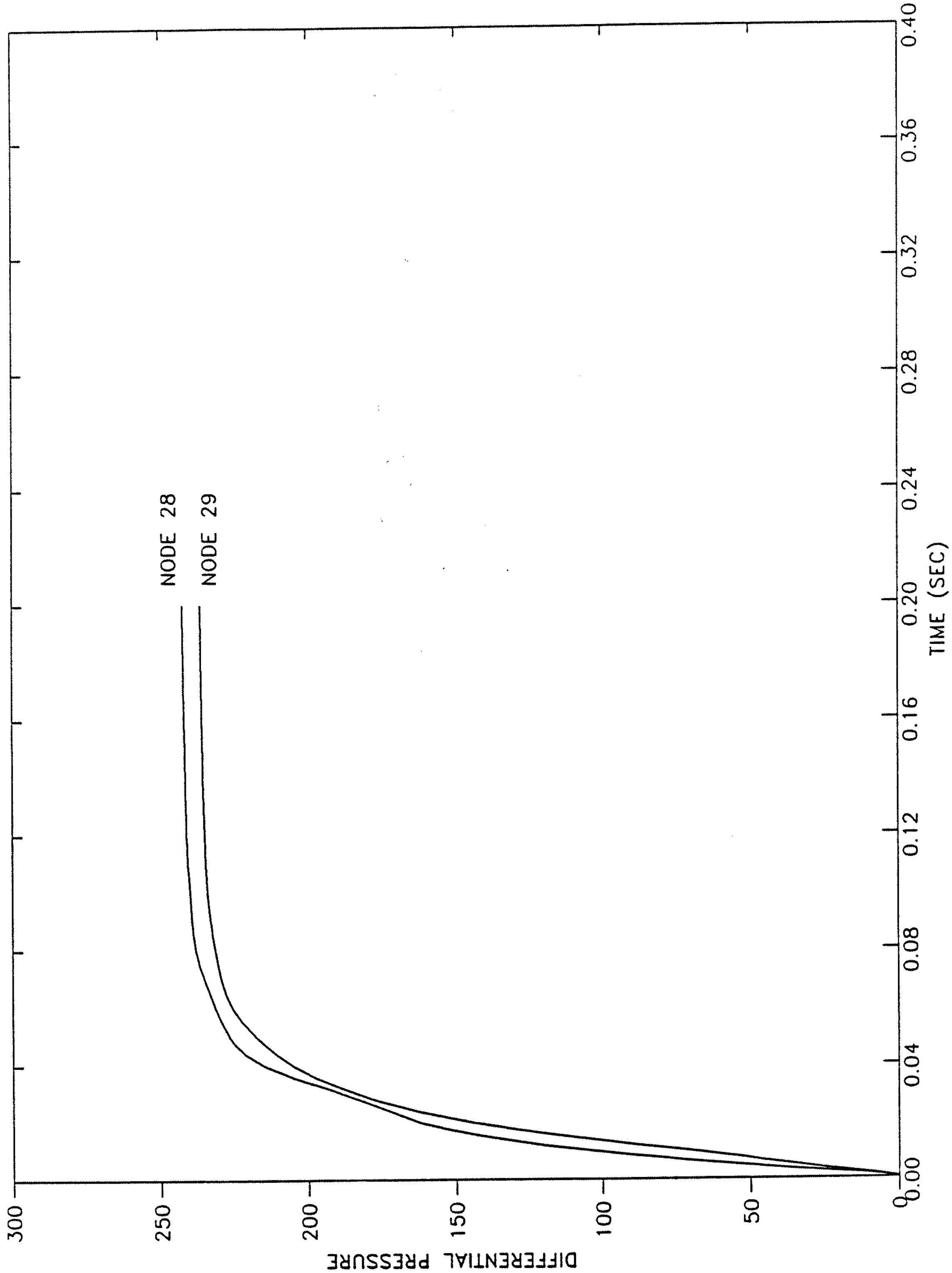


AMENDMENT 5
AUGUST, 1989

SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

31 Node Reactor Cavity Model - Differential Pressure at Selected Nodes Versus Time

Figure 6.2-33c

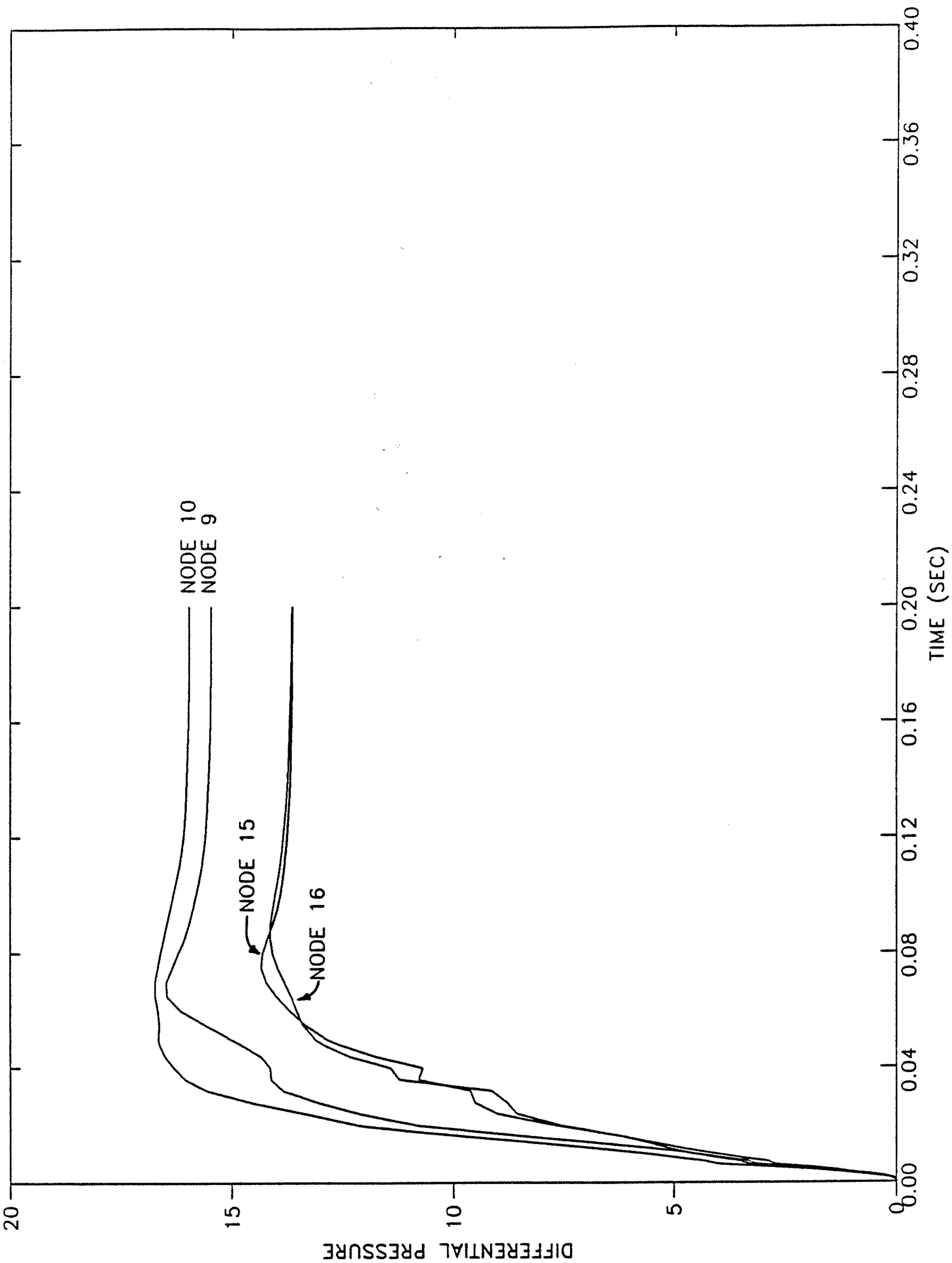


AMENDMENT 5
AUGUST, 1989

SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

33 Node Reactor Cavity Model -
Break Node Differential
Pressure Versus Time

Figure 6.2-33d

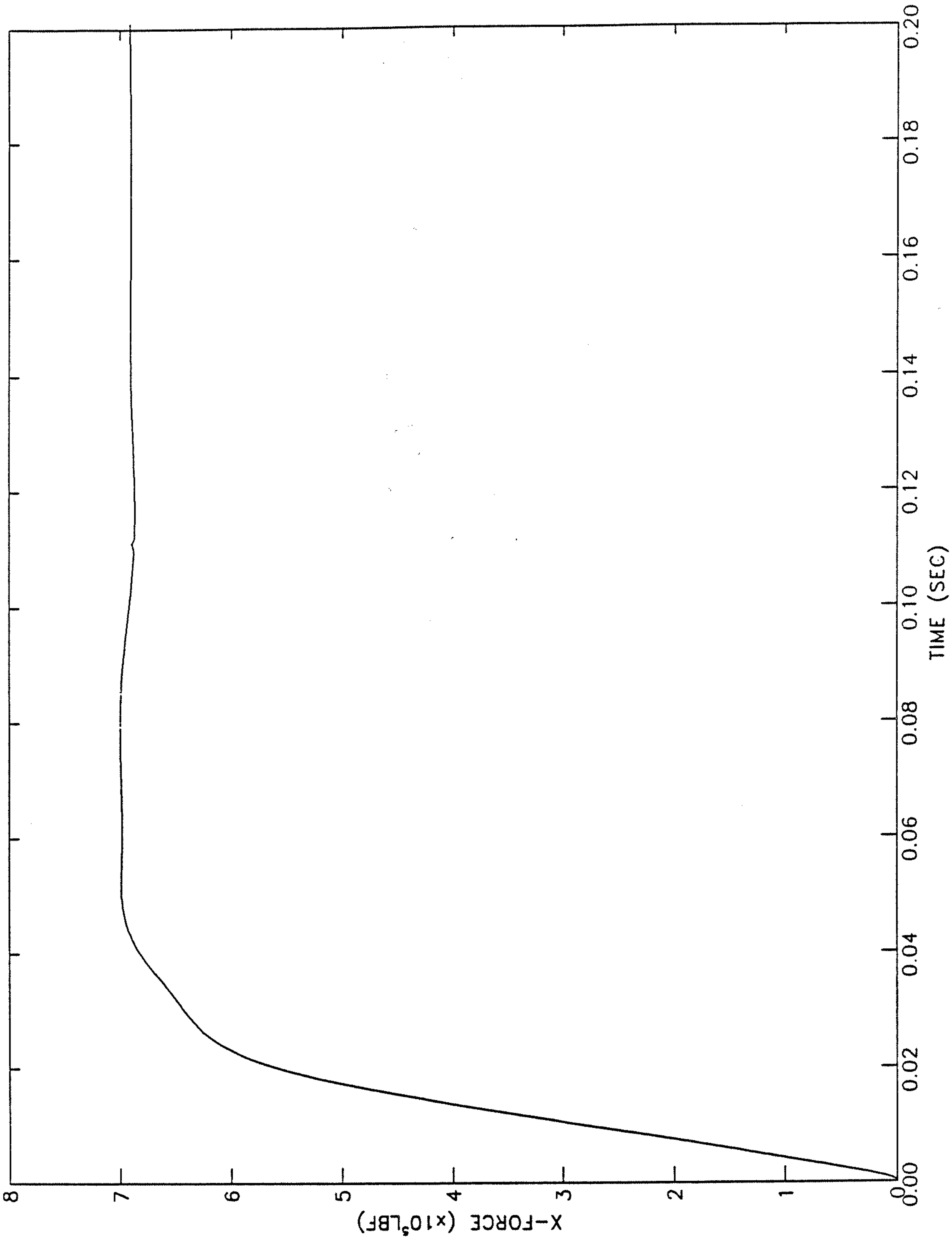


AMENDMENT 5
AUGUST, 1989

SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

33 Node Reactor Cavity Model - Differential Pressure at Selected Nodes Versus Time

Figure 6.2-33e

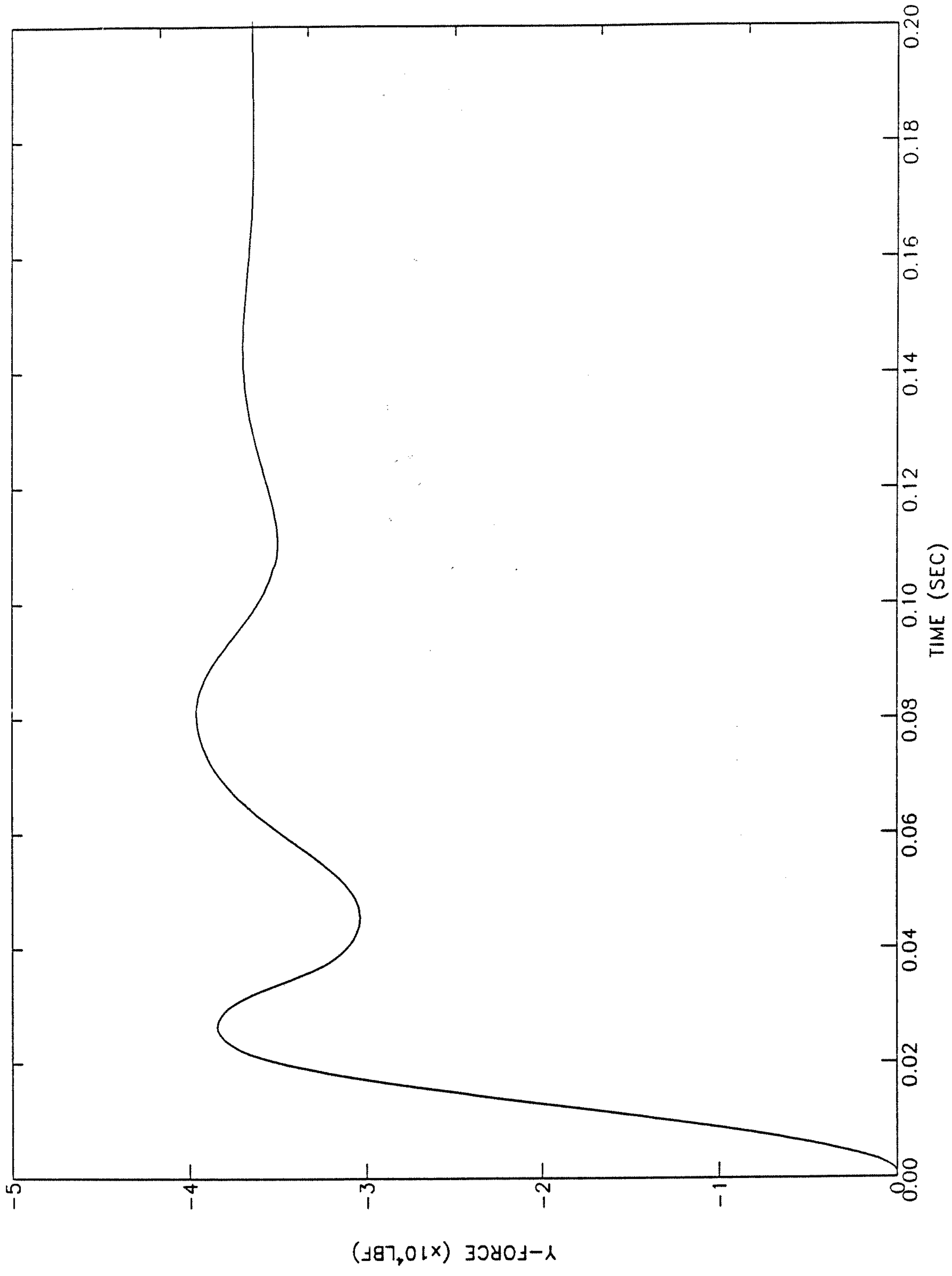


AMENDMENT 5
AUGUST, 1989

SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

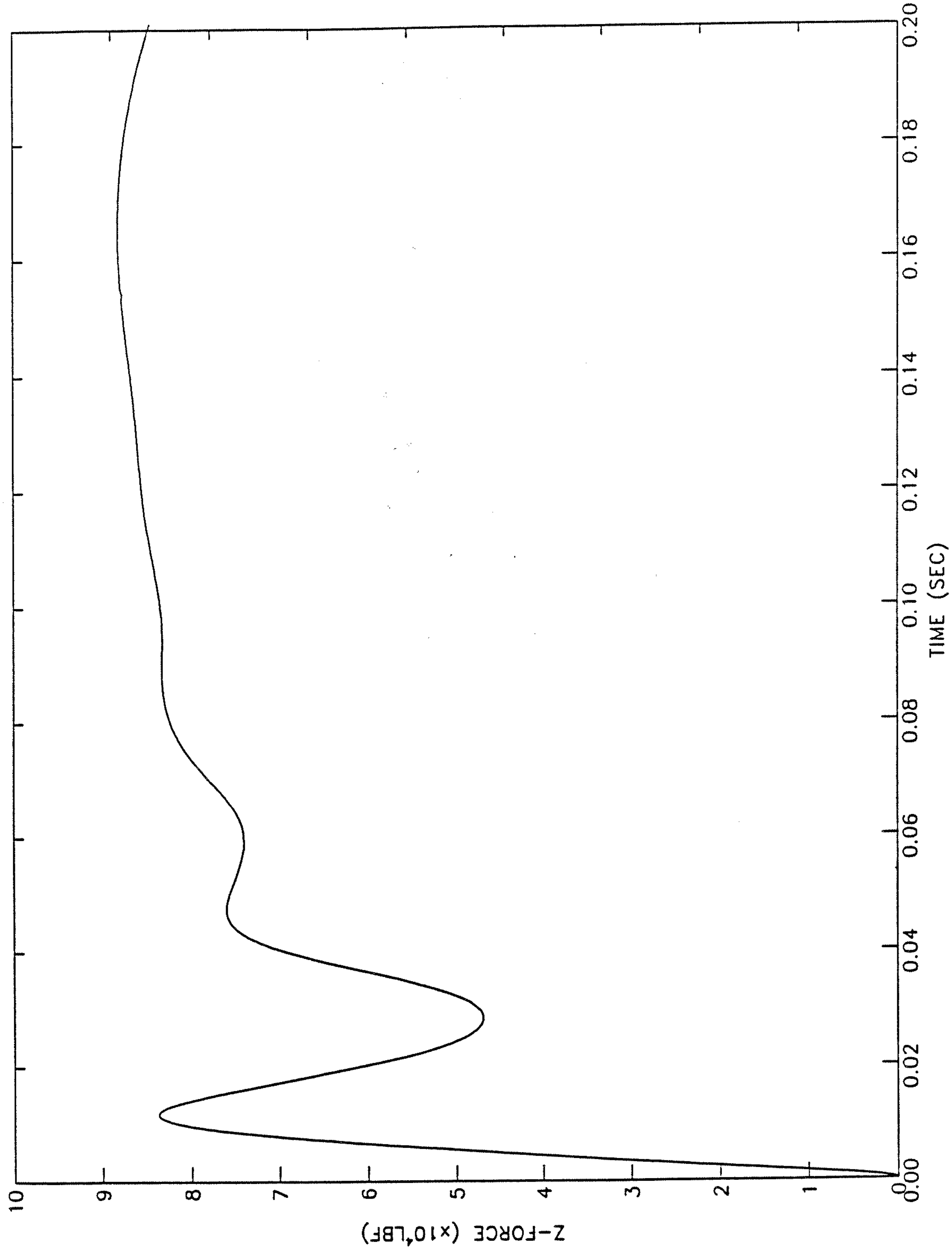
33 Node Reactor Cavity Model - Force in the X Direction

Figure 6.2-34



AMENDMENT 5
AUGUST, 1989

SOUTH CAROLINA ELECTRIC & GAS CO. **VIRGIL C. SUMMER NUCLEAR STATION**
33 Node Reactor Cavity Model - Force in the Y Direction **Figure 6.2-34a**

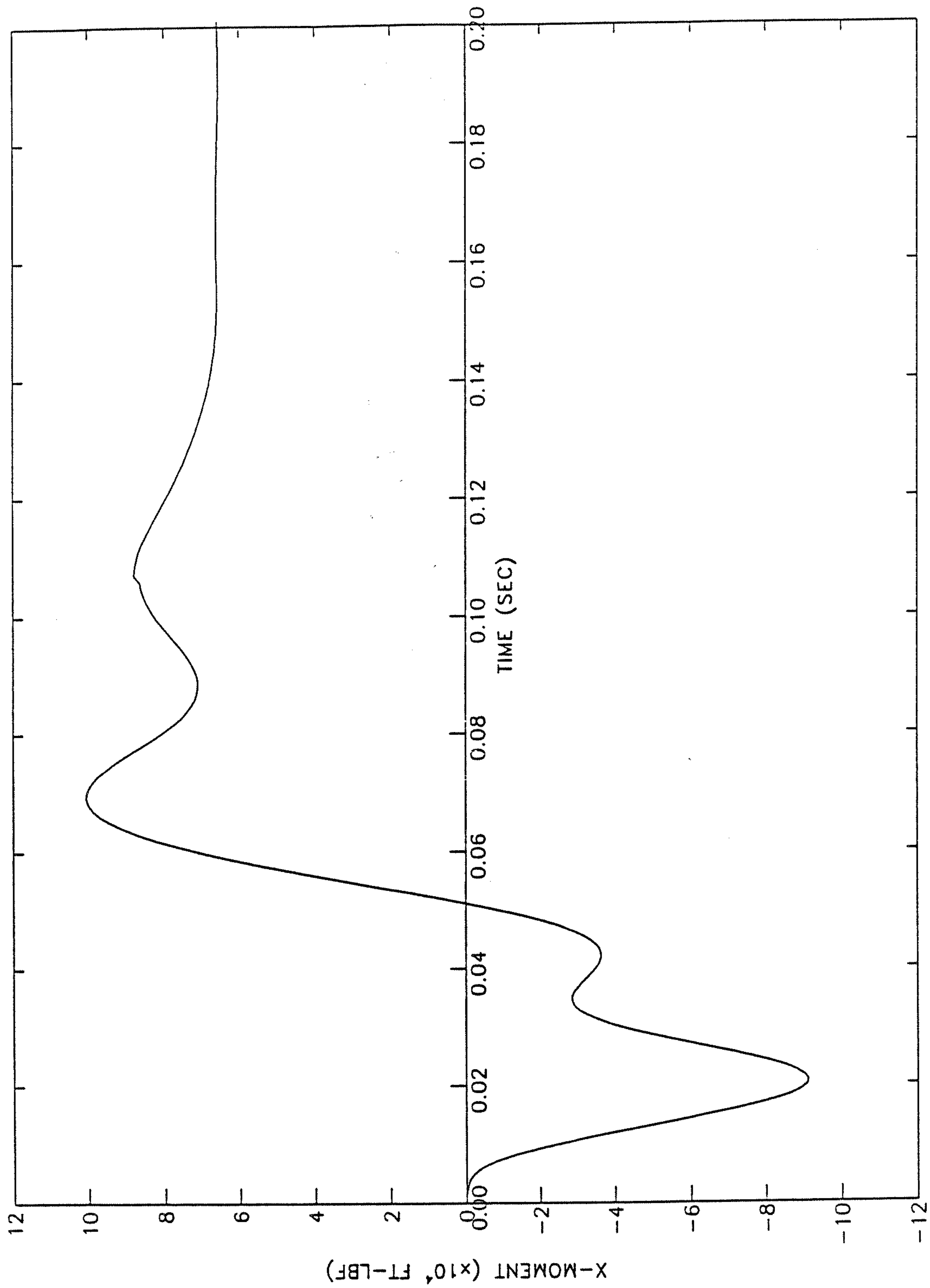


AMENDMENT 5
AUGUST, 1989

SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

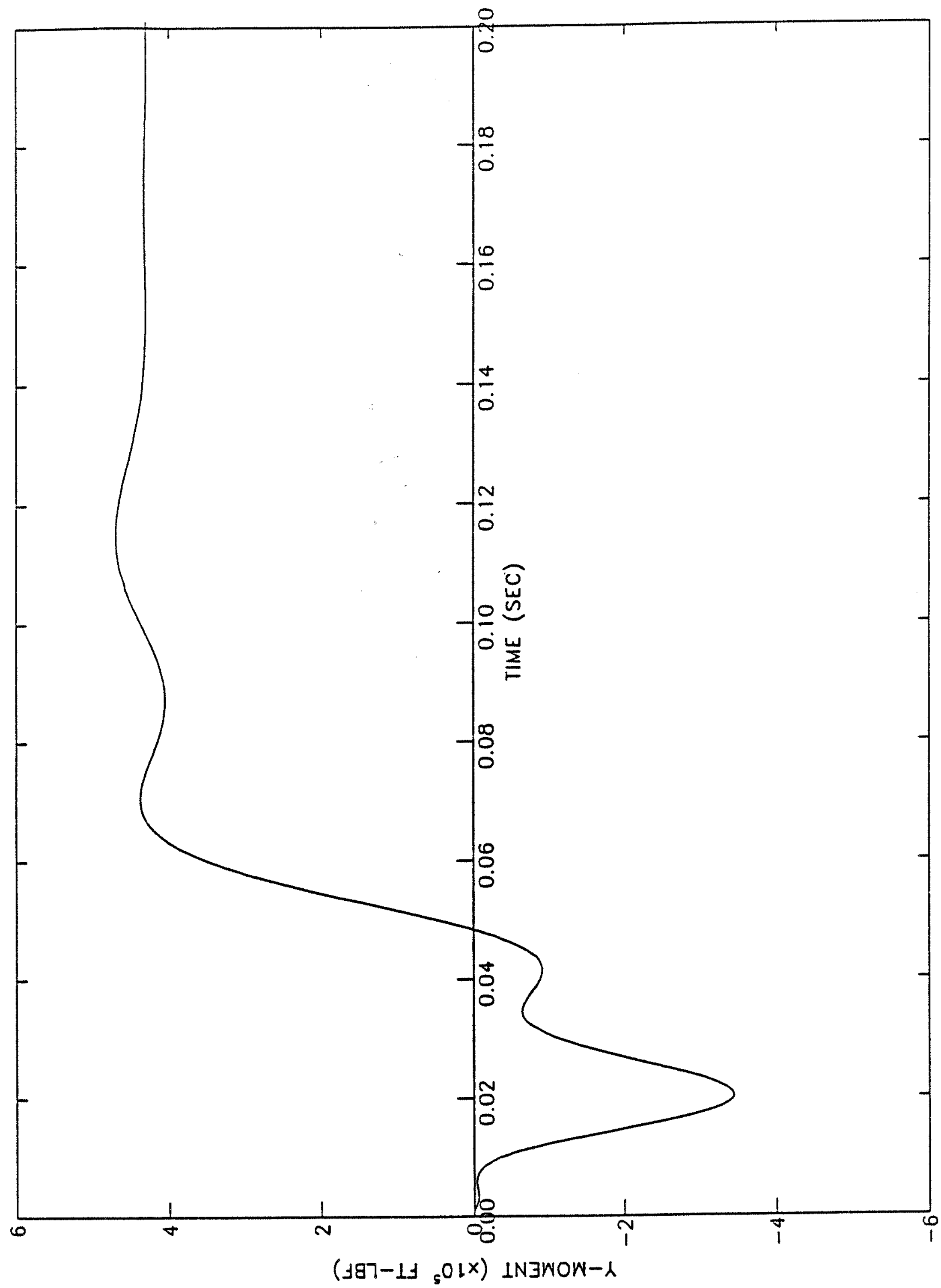
33 Node Reactor Cavity Model - Force in the Z Direction

Figure 6.2-34b



AMENDMENT 5
AUGUST, 1989

SOUTH CAROLINA ELECTRIC & GAS CO. VIRGIL C. SUMMER NUCLEAR STATION
33 Node Reactor Cavity Model - Moment About the X Axis Figure 6.2-35



AMENDMENT 5
AUGUST, 1989

SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

33 Node Reactor Cavity Model - Moment About the Y Axis

Figure 6.2-35a

SYSTEM DATA					
#	GPM	PSIG	°F	BY	REMARKS
1	4900	26	120	KRC	
2	2450	31	120	KRC	
3	2500	44	120	KRC	INJECTION
	2500	44	196	KRC	RECIRC.
4	2500	230	120	KRC	INJECTION
	2500	230	196	KRC	RECIRC.
5	1214	97	120	KRC	INJECTION
	1214	97	196	KRC	RECIRC.
6	76	97	120	KRC	INJECTION
	76	97	196	KRC	RECIRC.
7	2500	35	196	KRC	
8	90	27	105	KRC	*
9	65	31	105	KRC	**
10	2500	50	120	KRC	
11	100	7	100		
12	100	42	100		
13	50	75	100		
14	0.667	50	100		

* MAXIMUN FLOW WITH BOTH PUMPS OPERATING.

** MAXIMUN FLOW WITH ONE PUMP OPERATING.

NOTES:

1. FOR PUMP CASING DRAINS FOR BOTH PUMPS, SEE DETAIL 'A'.
2. SPECTACLE ORIFICE PLATES FOR TESTING ISOLATION CHECK VALVES.
4. RECORDER SHARED WITH SERVICE WATER FLOW INSTRUMENTATION FOR R.B. COOLING UNIT FLOW.
5. THE FLANGE HAS BEEN DRILLED AND TAPPED TO ACCEPT A 3/4" COMPRESSION FITTING TO FACILITATE LLRT WORK. THE FITTING SHOULD BE CAPPED AT OTHER TIMES.

(2 PLACES) (A) CONTAINMENT SPRAY ACTUATION TRAIN 'A' REF. 1MS-41-011-8 (W DWG 108D837 SH. 8) REF. 208-097 SP03

(2 PLACES) (B) CONTAINMENT SPRAY ACTUATION TRAIN 'B' REF. 1MS-41-011-8 (W DWG 108D837 SH. 8) REF. 208-097 SP04

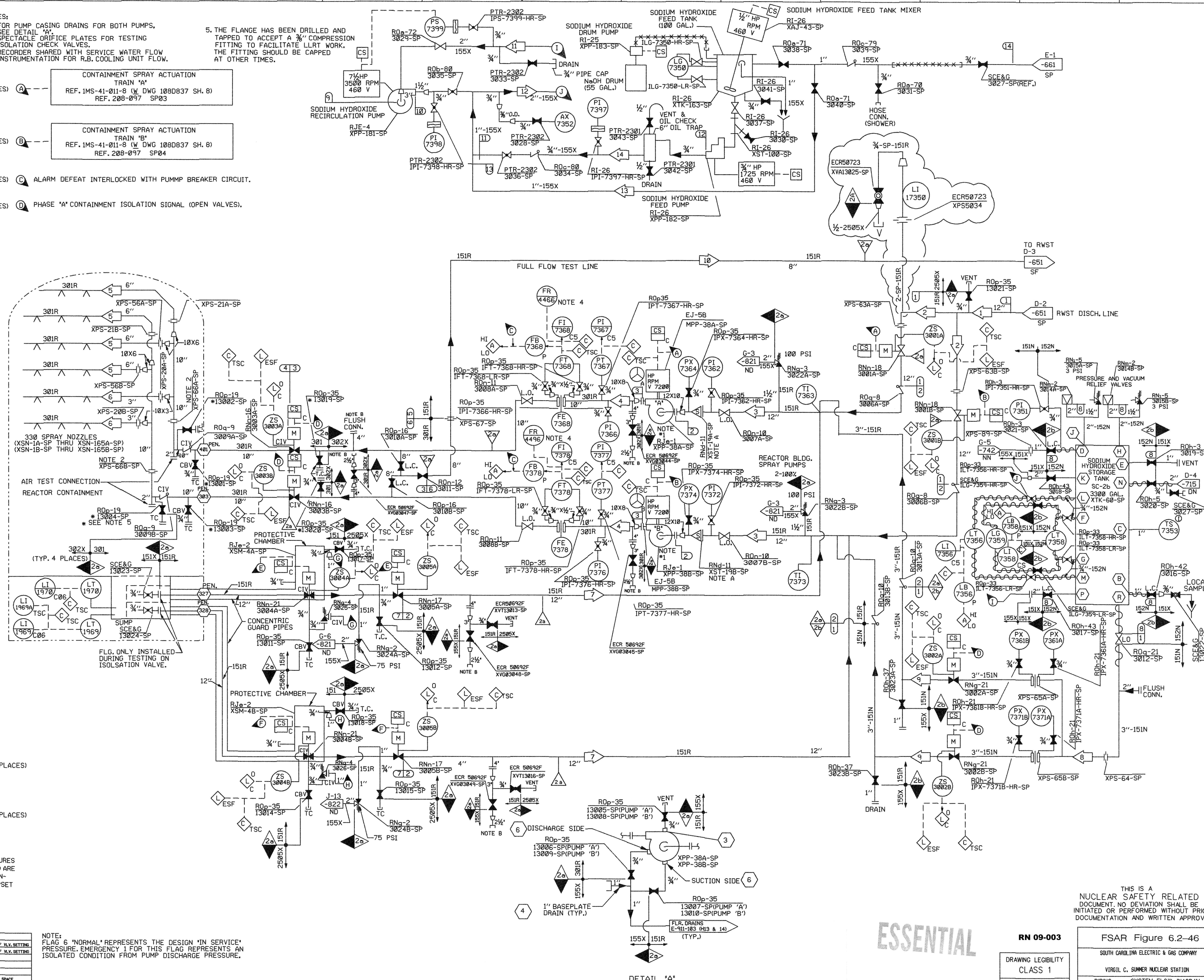
(2 PLACES) (C) ALARM DEFEAT INTERLOCKED WITH PUMMP BREAKER CIRCUIT.

(4 PLACES) (D) PHASE 'A' CONTAINMENT ISOLATION SIGNAL (OPEN VALVES).

NOTE:

A. TEMPORARY STRAINER FOR SYS. CLEAN UP TO BE REMOVED AFTER FINAL FLUSH.

B. HOSE/FLUSH CONNECTION PROVIDED OR AVAILABLE TO SATISFY OPERATING LICENSE CONDITION 2.C.(34).



RECIRCULATION SUMP AUTO SWITCHOVER LOGIC 'TRAIN A' REF.208-097 SP09, SP11 (E) (2 PLACES)

RECIRCULATION SUMP AUTO SWITCHOVER LOGIC 'TRAIN B' REF.208-097 SP10, SP12 (F) (2 PLACES)

NOTE:
WHEN APPLICABLE, THE NORMAL DESIGN PRESSURES & TEMPERATURES (ESPECIALLY TEMPERATURES) ARE THE WORST CONDITIONS THAT EXIST DURING INJECTION PHASE OF SAFETY INJECTION. THE UPSET CONDITIONS ARE THE WORST CONDITIONS FOR RECIRCULATION.

#	NORMAL PSIG	NORMAL °F	EMERGENCY 1 PSIG	EMERGENCY 1 °F	EMERGENCY 1 DURA.	PSIG HY-DRO	BY	CHKD	REMARKS
14	100	100	110	150		165			
13	100	100	110	150		165			*NaOH FEED PUMP RELIEF VLV. SETTING
12	125*	100	150	100		225			*NaOH FEED PUMP RELIEF VLV. SETTING
11	125*	100	150	100		225			
10	35	120	45	120		157.5			
9	7	120	3	120					
8	3*	120	3*	120		4*	PLL		*APPLIED TO TK. VAPOR SPACE
7	ATM	120	70	250		88	DTK		
6	225	120	45	120		282	DTK		
5	45	120	45	120		57	ECD		
4	ATM	120	165	250		207	DTK		
3	225	120	245	250		307	ECD		
2	45	120	80	250		100	ECD		
1	30	120	35	120		44	ECD		

DESIGN DATA

NOTE:
FLAG 6 'NORMAL' REPRESENTS THE DESIGN 'IN SERVICE' PRESSURE. EMERGENCY 1 FOR THIS FLAG REPRESENTS AN ISOLATED CONDITION FROM PUMP DISCHARGE PRESSURE.

NOTE:
HYDROTEST TEMP. TO BE AT OR ABOVE 60°F. FOR CARBON STEEL SUBSYSTEM, AND AT OR ABOVE 40°F. FOR STAINLESS STEEL SUB-SYSTEMS. HYDROSTATIC TESTING OF RING HEADERS & NOZZLES TO BE ACCOMPLISHED PER CODE N-58, PAR. 1614.

1/4"=1'-0" 0' 5' 10' 15' 20'

ESSENTIAL

RN 09-003

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DRAWING LEGIBILITY CLASS 1

SCE&G CAD ENHANCED

NO.	DATE	BY	REVISION	CKD. BY	LE APPROVAL
30	12/4/97	RHM	REVISED PER CGSS-97-0636	MGR	WTW
35	11/6/09	JTS	REVISED PER ECR-50723	DDJ	WTW
34	5/20/08	DDJ	REVISED PER ECR-50692F	MGR	RAT
33	1/22/08	JMR	ENHANCED PER ECR-50239	MGR	DJ
32	5/4/05	JTS	REVISED PER ECR-50316	JMW	RLR
31	5/3/99	JMR	REVISED PER CGSS-99-0261	RHM	TOC

FSAR Figure 6.2–46

SOUTH CAROLINA ELECTRIC & GAS COMPANY

VIRGIL C. SUMMER NUCLEAR STATION

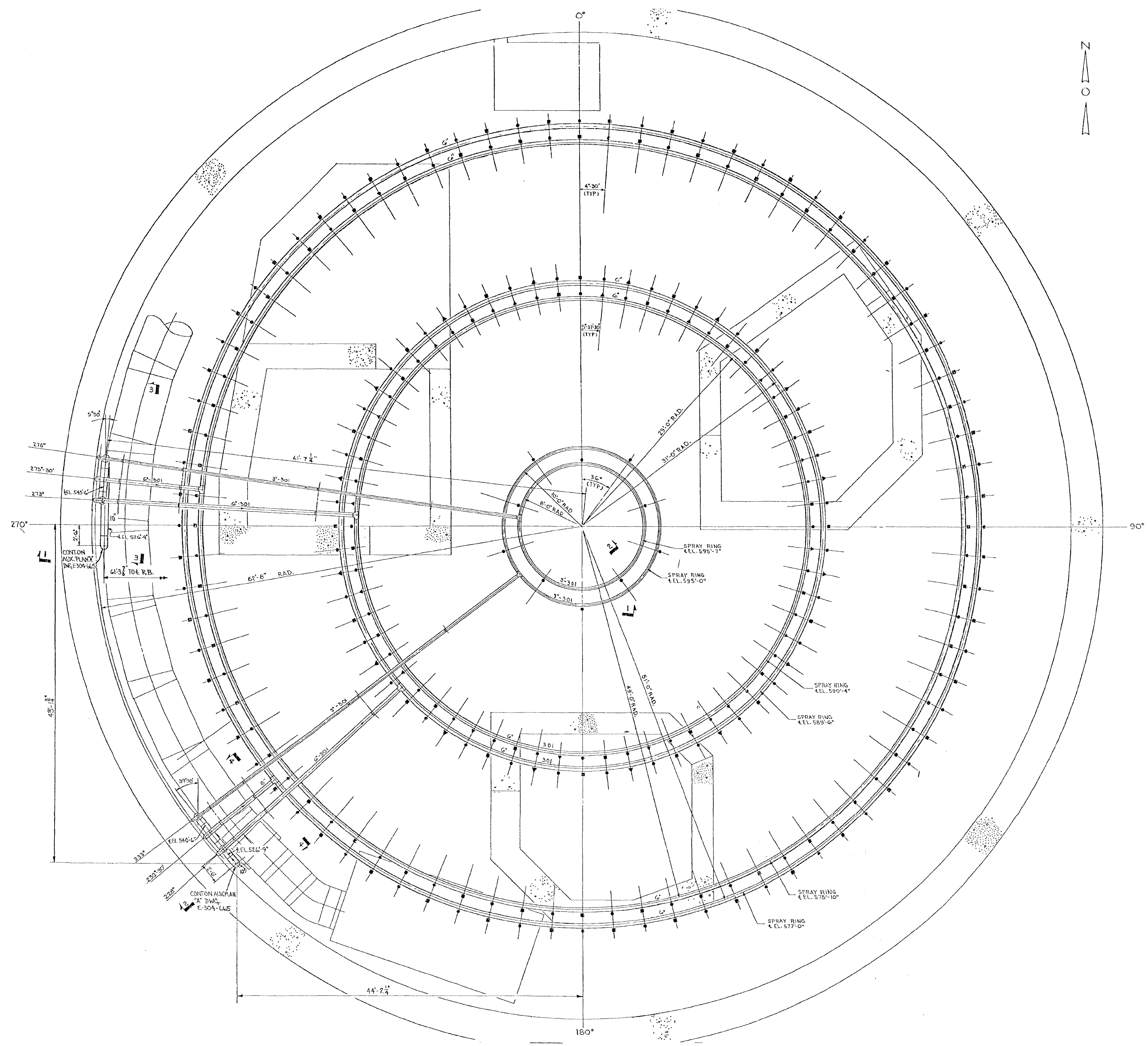
PIPING SYSTEM FLOW DIAGRAM

REACTOR BUILDING SPRAY SYSTEM

SCE&G A SCANA COMPANY — DESIGN ENGINEERING — V. C. SUMMER NUCLEAR STATION JENKINSVILLE, S. C.

MADE	CHECKED	LE APPROVAL
1. DDJ	2. MGR	3. GAL

SCALE: NO SCALE — DRAWING NUMBER: D-302-661 — SHT. NUMBER: — REV: 35



N

LEGEND

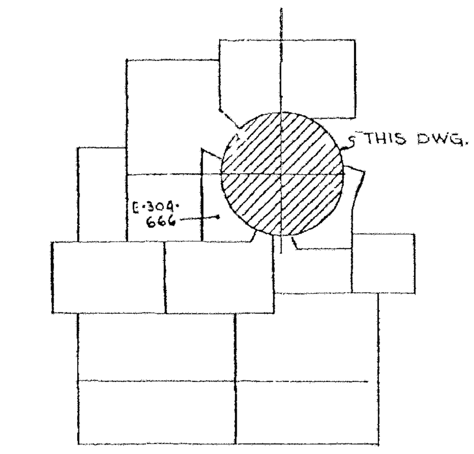
- INDICATES SPRAY NOZZLE AS PER DETAIL-A DWG. E-304-664
- INDICATES SPRAY NOZZLE AS PER DETAIL-B DWG. E-304-664
- INDICATES SPRAY NOZZLE AS PER DETAIL-C DWG. E-304-664

NOTES

1. FOR PIPE MATERIAL SHOWN ON THIS DRAWING, REFER TO PIPING SPECIFICATION SP-337-4461-00 FOR NON-NUCLEAR SAFETY CLASS, LINE SPEC 302X. SP-545-044461-000 FOR ANS SAFETY CLASS, LINE SPEC 301
2. PIPING TO BE ANS SAFETY CLASSIFICATION 2A UNLESS NOTED.
3. NO ALLOWANCE MADE FOR WELD GAPS FOR WELD END DETAIL SEE DWG C-301-002.
4. ALL ATTACHMENTS WELDED TO ALLOY PIPE SHALL BE SAME MATERIAL AS PIPE.
5. ALL CONNECTIONS TO PENETRATIONS TO BE FABRICATED 6" LONGER THAN SHOWN FOR PENETRATION LOCATION TOLERANCES.
6. WELDING WILL BE PERMITTED IN THE BENT SECTIONS OF THE PIPING

DRAWING REFERENCES:-

DWG. NO.	TITLE
E-304-664	REACTOR BUILDING SPRAY SYSTEM – SECTIONS & DETAILS
E-304-665	REACTOR BUILDING SPRAY SYSTEM – SECTIONS & DETAILS
D-302-661	REACTOR BUILDING SPRAY SYSTEM – FLOW DIAGRAM



KEY PLAN

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FSAR Figure 6.2-47

SOUTH CAROLINA ELECTRIC & GAS COMPANY

VIRGIL C. SUMMER NUCLEAR STATION

PIPING SYSTEM

REACTOR BUILDING SPRAY SYSTEM REACTOR BUILDING ABOVE ELEVATION 524'-0"

DESIGN ENGINEERING

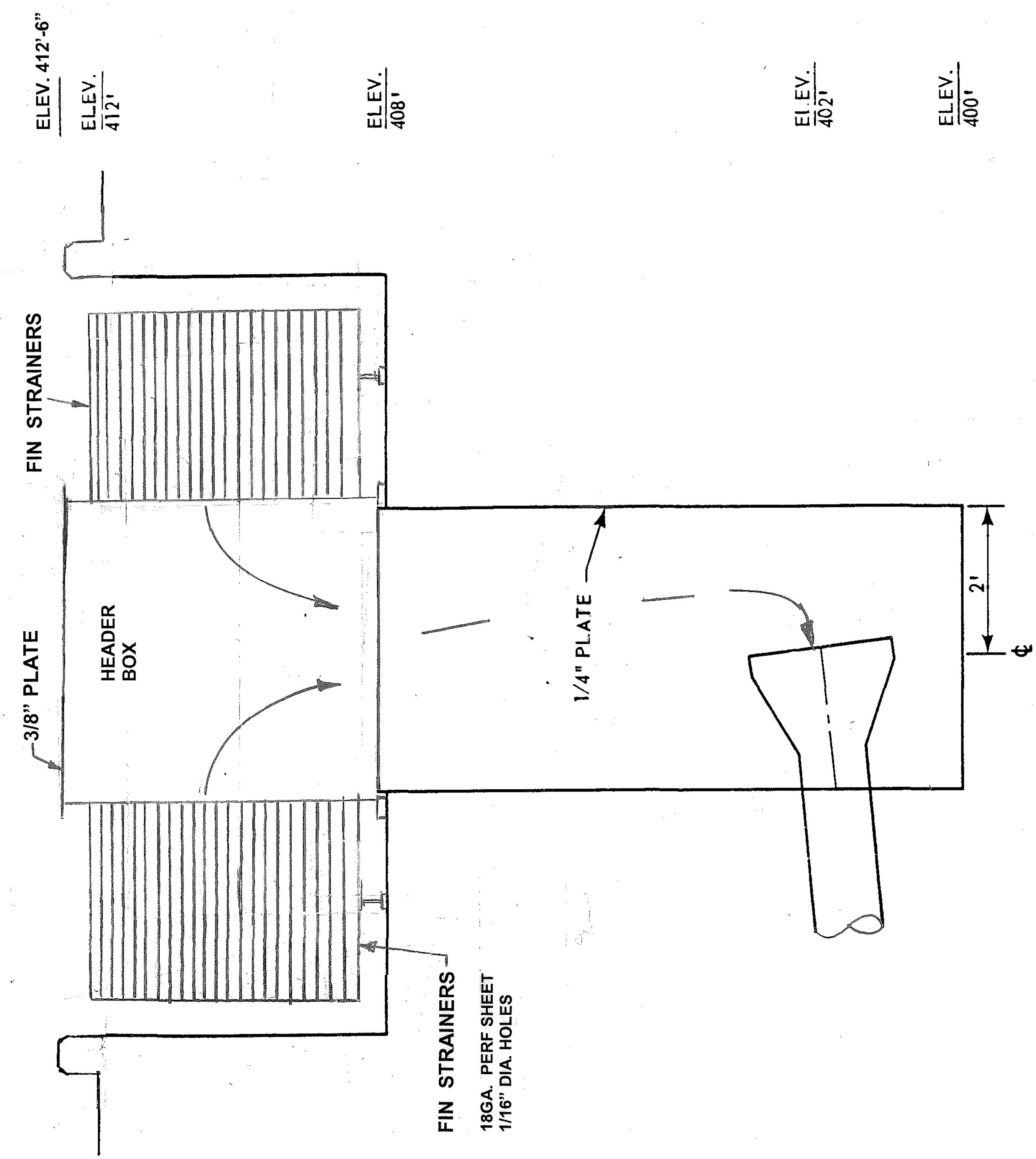
V. C. SUMMER NUCLEAR STATION JENKINSVILLE, S. C.

Amendment 02-01
May 2002

NO.	DATE	BY	REVISION	CKD. BY	LE APPROVAL
3	9/25/01	DDJ	REVISED PER ECR-70028	MGR	DDJ

MADE	CHECKED	LE APPROVAL
1. DDJ	2. MGR	3. DDJ

SCALE	DRAWING NUMBER	SHT. NUMBER	REV
NO SCALE	E-304-663		3



RN 13-022

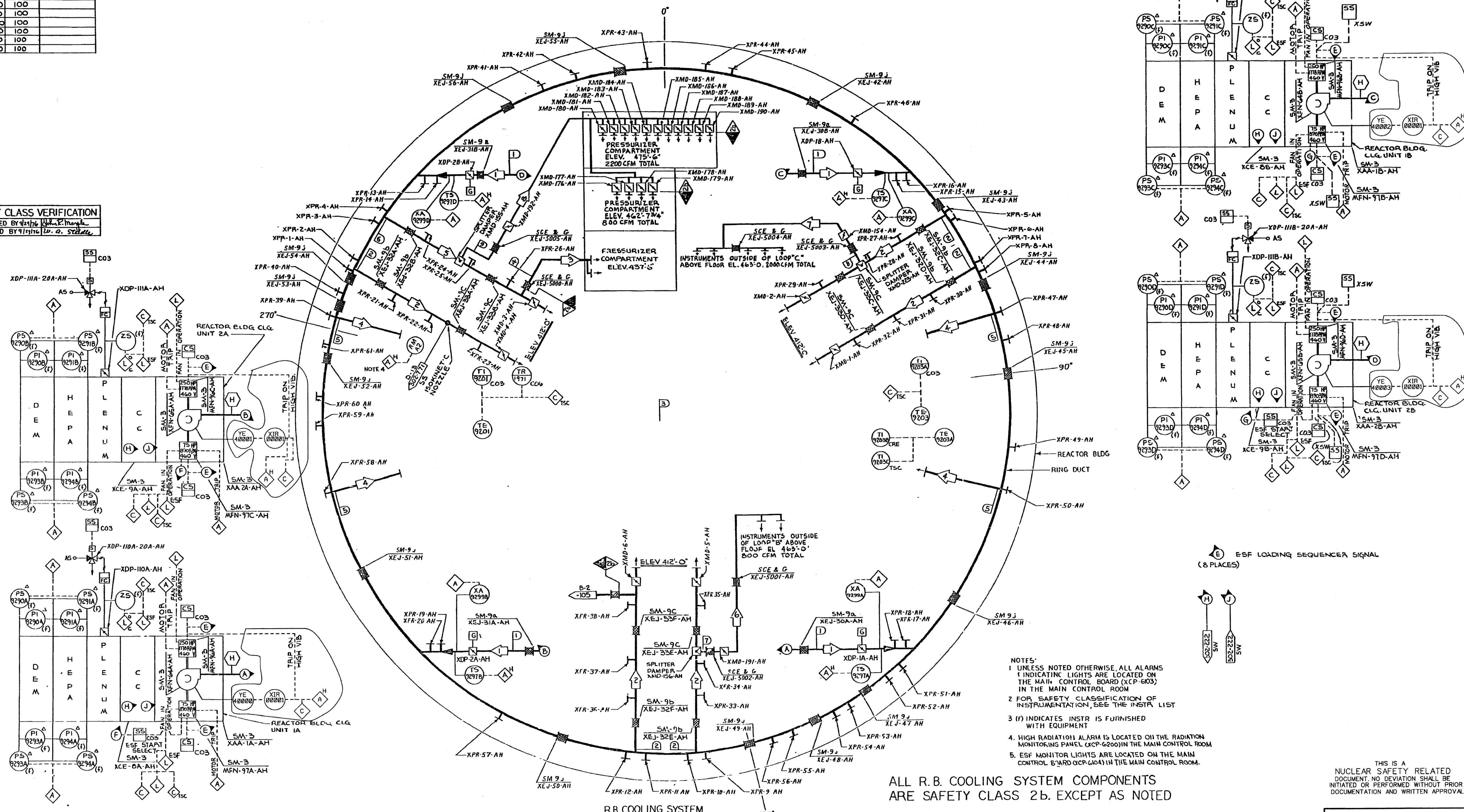
SOUTH CAROLINA ELECTRIC & GAS CO. VIRGIL C. SUMMER NUCLEAR STATION
Section Through Typical Sump Figure 6.2-48

Page 1 of 1

\\Vcs01lp\cadd\Working\Interims\50683\912102_0001.dgn
rml5043 20-NOV-2009 22:48

SYSTEM OPERATING DATA			
⇨	AIR FLOW ACFM	AIR TEMP °F	
1	123,000	100	
2	58,160	100	
3	5,200	100	
4	5,000	100	
5	58,200	100	
6	800	100	
7	2,000	100	
8	3,000	100	

SAFETY CLASS VERIFICATION
ORIGINATED BY 4/21/76 John R. Marple
REVIEWED BY 9/13/76 W. G. Stibbe



NOTES:
1. UNLESS NOTED OTHERWISE, ALL ALARMS & INDICATING LIGHTS ARE LOCATED ON THE MAIN CONTROL BOARD (XCP-6103) IN THE MAIN CONTROL ROOM
2. FOR SAFETY CLASSIFICATION OF INSTRUMENTATION, SEE THE INSTR. LIST
3. (f) INDICATES INSTR IS FURNISHED WITH EQUIPMENT
4. HIGH RADIATION ALARM IS LOCATED ON THE RADIATION MONITORING PANEL (XCP-6200) IN THE MAIN CONTROL ROOM
5. ESF MONITOR LIGHTS ARE LOCATED ON THE MAIN CONTROL BOARD (XCP-6104) IN THE MAIN CONTROL ROOM.

ALL R.B. COOLING SYSTEM COMPONENTS ARE SAFETY CLASS 2b. EXCEPT AS NOTED

R.B. COOLING SYSTEM

9	3.000	100		810	—	
8	2,000	100		540	—	
7	800	100		216	—	
6	58,200	100		19,140		
5	5000	100		1600	—	
4	5200	100		1700		
3	367,000	120		20,540		
2	58160	100		19,[illegible]	—	
1	123,000	100		[illegible]	—	
⇨	AIR FLOW ACFM	AIR TEMP °F		AIR FLOW ACFM	AIR TEMP °F	
	NORMAL			EMERGENCY		
SYSTEM DESIGN DATA						

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RN 09-023
June 2010

NO.	DATE	BY	REVISION	CKD. BY	LE APPROVAL
21	11/20/09	RHM	REVISED PER ECR-50683	RHM	MM
20	10/02/01	DDJ	REVISED PER ECR-70028	MGR	DDJ

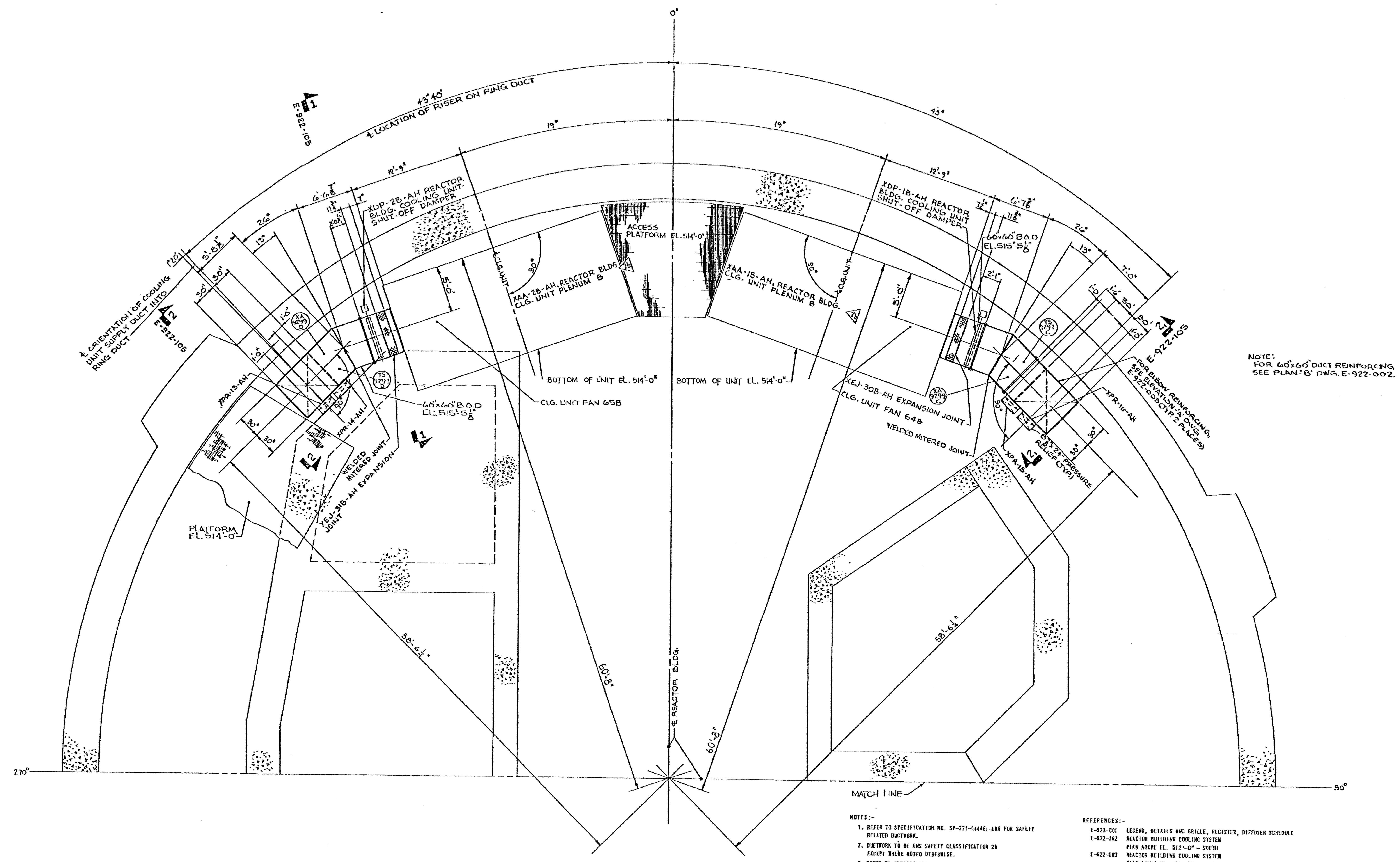
FSAR Figure 6.2–49

SOUTH CAROLINA ELECTRIC & GAS COMPANY
VIRGIL C. SUMMER NUCLEAR STATION
BUILDING SERVICE SYSTEM FLOW DIAGRAM
R.B. COOLING SYSTEM

SCE&G A SCANA COMPANY — DESIGN ENGINEERING
V. C. SUMMER NUCLEAR STATION JENKINSVILLE, S. C.

MADE	CHECKED	LE APPROVAL
1. DDJ	2. MGR	3. DDJ

SCALE: NO SCALE — DRAWING NUMBER: D-912-102 — SHT. NUMBER: — REV: 21



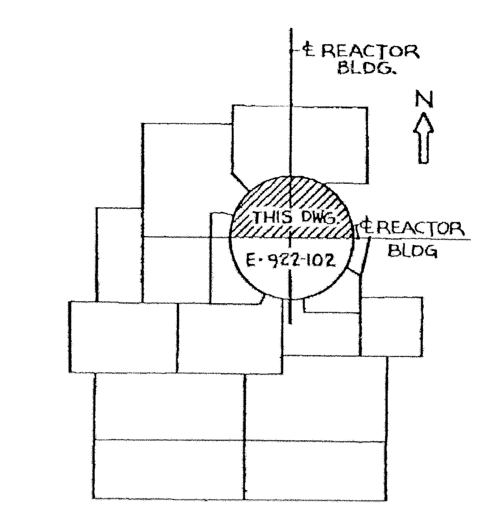
FOR CONTINUATION SEE DRAWING E-922-102

NOTES:-

1. REFER TO SPECIFICATION NO. SP-221-044461-000 FOR SAFETY RELATED DUCTWORK.
2. DUCTWORK TO BE ANS SAFETY CLASSIFICATION 2b EXCEPT WHERE NOTED OTHERWISE.
3. REFER TO SPECIFICATION NO. SP-223-044461-000 FOR INSTALLATION OF EQUIPMENT.
4. DUCT CONSTRUCTION SCHEDULE S-903-001 AND S-903-002 SHOWN IN SPEC. SP-221-044461-000 WILL NOT BE USED FOR THE RING DUCT, RISERS OR 60" X 60" DUCT FROM UNITS TO RING DUCT.
5. ALL TRANSVERSE LONGITUDINAL OR CIRCUMFERENTIAL MITERED SEAMS THAT DO NOT OCCUR AT STIFFENER LOCATIONS ARE TO BE BUTT WELDED CONNECTIONS.
6. DUCT FOR PRESSURE RELIEFS TO BE WELDED CONSTRUCTION.

REFERENCES:-

E-922-001	LEGEND, DETAILS AND GRILLE, REGISTER, DIFFUSER SCHEDULE
E-922-102	REACTOR BUILDING COOLING SYSTEM PLAN ABOVE EL. 512'-0" - SOUTH
E-922-103	REACTOR BUILDING COOLING SYSTEM PLAN ABOVE EL. 463'-0" - NORTH
E-922-104	REACTOR BUILDING COOLING SYSTEM PLAN ABOVE EL. 463'-0" - SOUTH
E-922-105	REACTOR BUILDING COOLING SYSTEM - SECTIONS
D-912-102	REACTOR BUILDING COOLING SYSTEM - FLOW DIAGRAM
D-912-103	REACTOR BUILDING PURGE SUPPLY AND PURGE EXHAUST SYSTEMS - FLOW DIAGRAM
E-936-101	REACTOR BUILDING COOLING SYSTEM PLAN ABOVE EL. 512'-0" NORTH - DUCT HANGERS
E-936-102	REACTOR BUILDING COOLING SYSTEM PLAN ABOVE EL. 512'-0" SOUTH - DUCT HANGERS
E-936-103	REACTOR BUILDING COOLING SYSTEM PLAN ABOVE EL. 463'-0" NORTH - DUCT HANGERS
E-936-104	REACTOR BUILDING COOLING SYSTEM PLAN ABOVE EL. 463'-0" SOUTH - DUCT HANGERS
E-936-105	REACTOR BUILDING COOLING SYSTEM SECTIONS - DUCT HANGERS
E-922-002	DUCT REINFORCING DETAILS - REACTOR BUILDING
E-922-003	DUCT REINFORCING DETAILS - REACTOR BUILDING



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Amendment 02-01
May 2002

DRAWING LEGIBILITY CLASS 1
SCE&G CAD ENHANCED

FSAR Figure 6.2-50, Sht. 1

SOUTH CAROLINA ELECTRIC & GAS COMPANY
VIRGIL C. SUMMER NUCLEAR STATION
BUILDING SERVICE — HVAC
REACTOR BUILDING COOLING SYSTEM
PLAN ABOVE EL. 512'-0" - NORTH
DESIGN ENGINEERING
V. C. SUMMER NUCLEAR STATION JENKINSVILLE, S. C.

NO.	DATE	BY	REVISION	CKD. BY	LE APPROVAL
4	10/16/01	DDJ	REVISED PER ECR-70028	MGR	DDJ

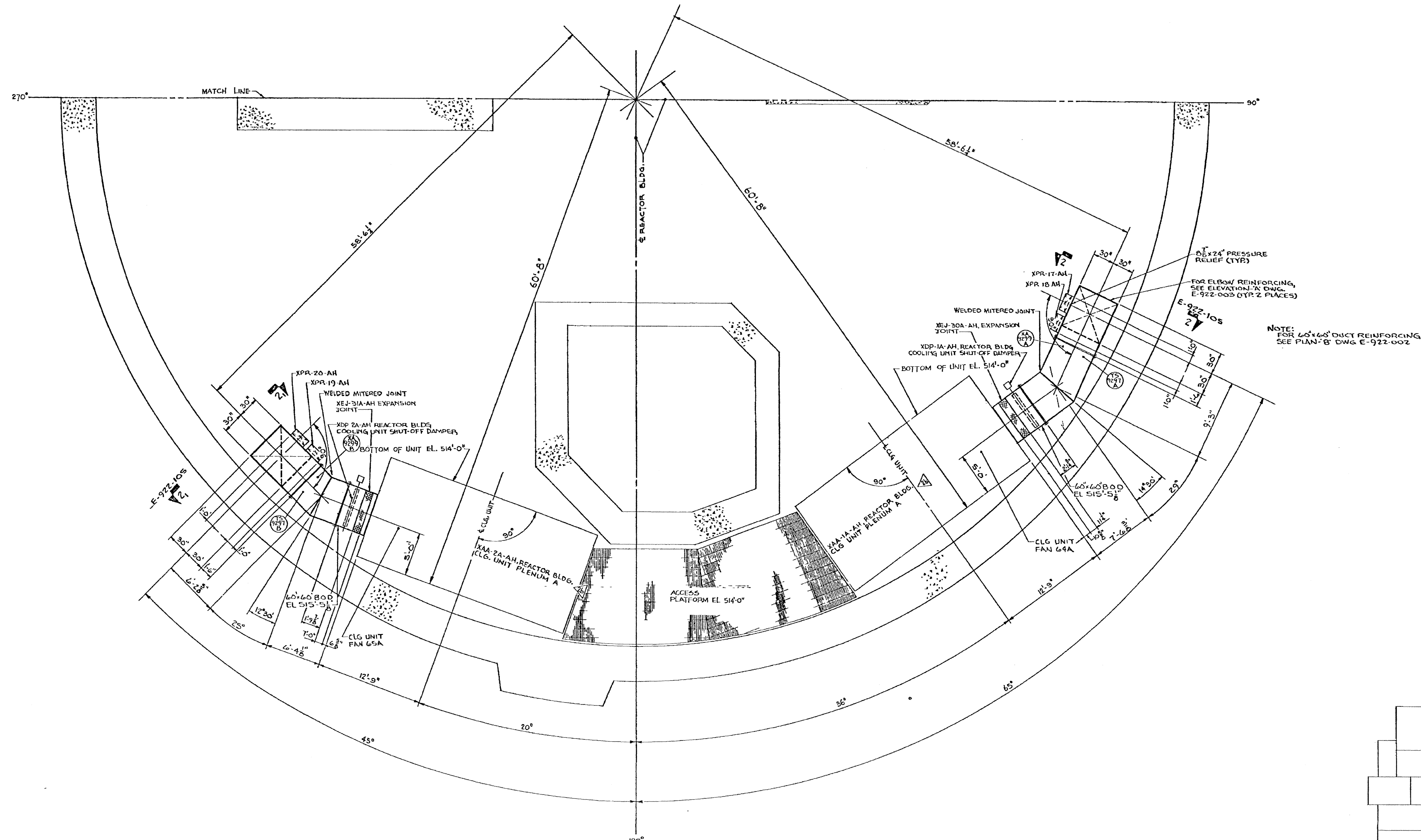
MADE	CHECKED	LE APPROVAL
DDJ	MGR	DDJ

SCALE: NO SCALE — DRAWING NUMBER: E-922-101 — REV. 4

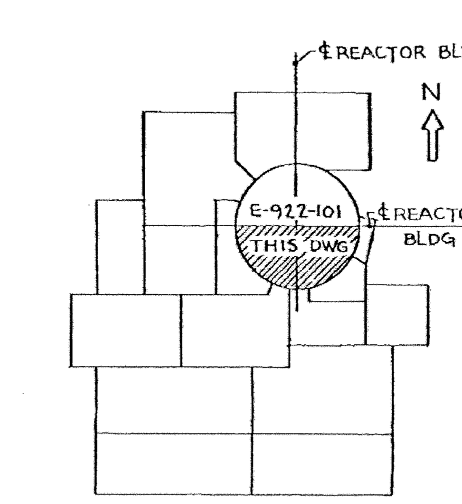
1/4"=1'-0" 0' 5' 10' 15' 20'

...\ondeck\2601\922101_0001.dgn Oct. 09, 2001 15:49:06

FOR CONTINUATION SEE DRAWING E-922-101



NOTES.
1. FOR NOTES AND REFERENCES SEE DWG. E-922-101.



KEY PLAN

1/4"=1'-0" 0' 5' 10' 15' 20'

THIS IS A NUCLEAR SAFETY RELATED DOCUMENT. NO DEVIATION SHALL BE INITIATED OR PERFORMED WITHOUT PRIOR DOCUMENTATION AND WRITTEN APPROVAL

Amendment 02-01
May 2002

DRAWING LEGIBILITY CLASS 1
SCE&G CAD ENHANCED

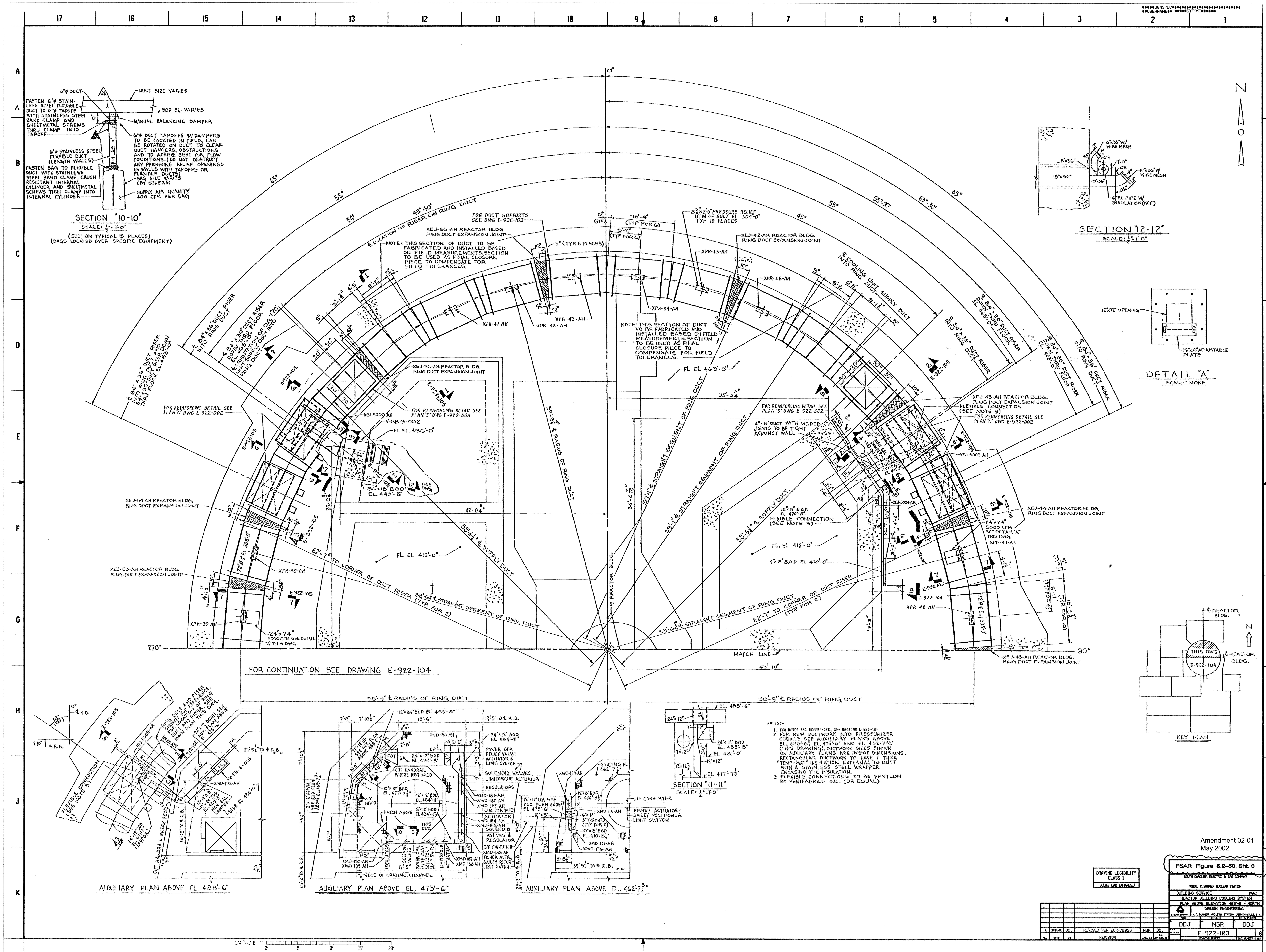
FSAR Figure 6.2-50, Sht. 2

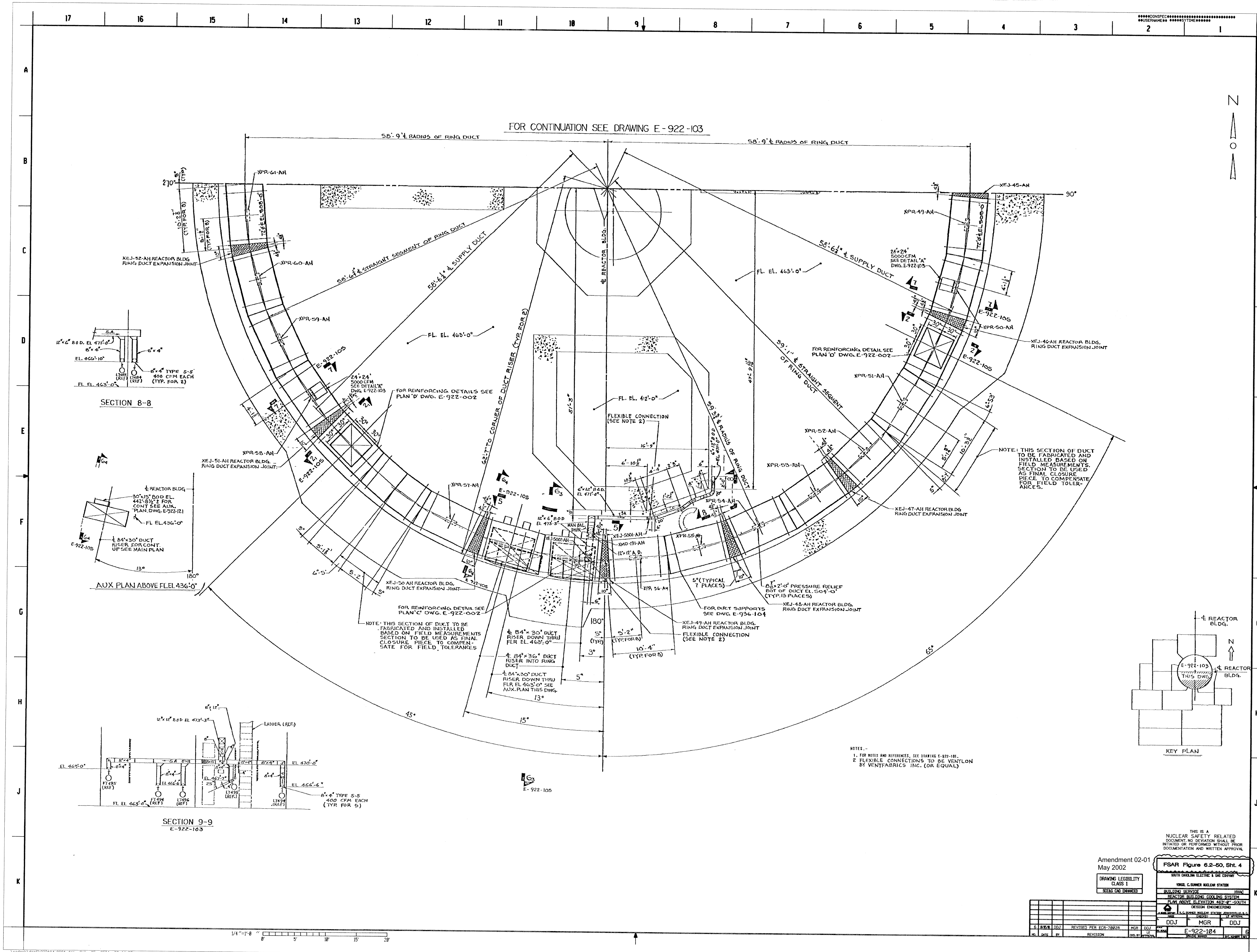
SOUTH CAROLINA ELECTRIC & GAS COMPANY
VIRGIL C. SUMMER NUCLEAR STATION
BUILDING SERVICE — HVAC
REACTOR BUILDING COOLING SYSTEM
PLAN ABOVE ELEVATION 512'-0" - SOUTH
DESIGN ENGINEERING
V. C. SUMMER NUCLEAR STATION JENKINSVILLE, S. C.

MADE	CHECKED	LE APPROVAL
DDJ	MGR	DDJ

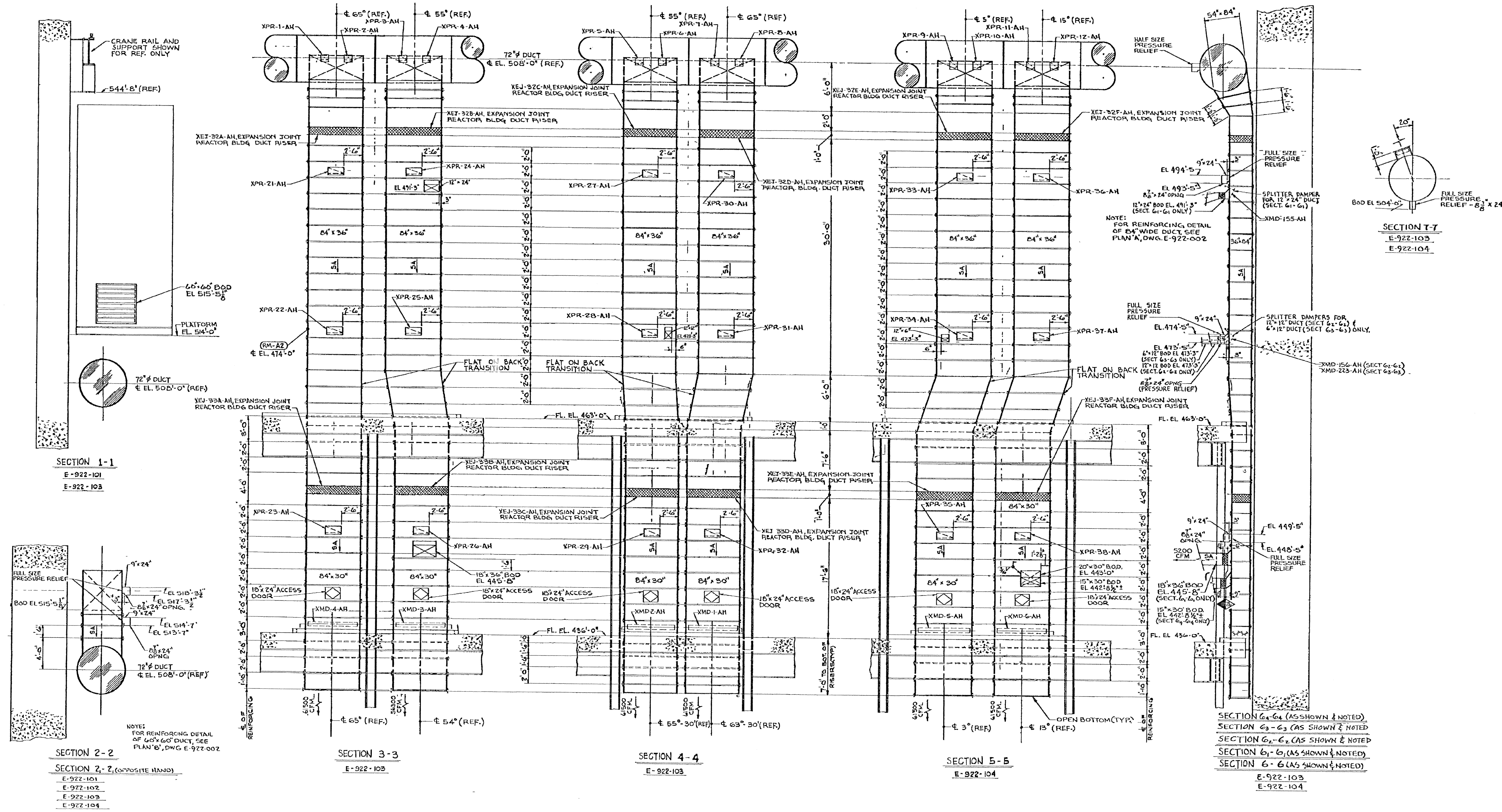
NO.	DATE	BY	REVISION	CKD. BY	LE APPROVAL
3	[illegible]	DDJ	REVISED PER ECR-70028	MGR	DDJ

DRAWING NUMBER: E-922-102 REV: 3


SECTION "10-10"
SCALE: 1/2" = 1'-0"
(SECTION TYPICAL 15 PLACES)
(BAGS LOCATED OVER SPECIFIC EQUIPMENT)
SECTION "12-12"
DETAIL "A"
SCALE: NONE
FOR CONTINUATION SEE DRAWING E-922-104
SECTION "11-11"
KEY PLAN
AUXILIARY PLAN ABOVE EL. 488'-6"
AUXILIARY PLAN ABOVE EL. 475'-6"
AUXILIARY PLAN ABOVE EL. 462'-7 3/4"
NOTES:-
1. FOR NOTES AND REFERENCES, SEE DRAWING E-922-101
2. FOR NEW DUCTWORK INTO PRESSURIZER CUBICLE SEE AUXILIARY PLANS ABOVE EL. 488'-6", EL. 475'-6" AND EL. 462'-7 3/4" (THIS DRAWING). DUCTWORK SIZES SHOWN ON AUXILIARY PLANS ARE INSIDE DIMENSIONS. RECTANGULAR DUCTWORK TO HAVE 1" THICK "TEMP-MAT" INSULATION EXTERNAL TO DUCT WITH A STAINLESS STEEL WRAPPER ENCASING THE INSULATION.
3. FLEXIBLE CONNECTIONS TO BE VENTLON BY VENTFABRICS INC. (OR EQUAL)
Amendment 02-01
May 2002
FSAR Figure 6.2-50, Sht. 3
SOUTH CAROLINA ELECTRIC & GAS COMPANY
VIRGIL C. SUMMER NUCLEAR STATION
BUILDING SERVICE HVAC
REACTOR BUILDING COOLING SYSTEM
PLAN ABOVE ELEVATION 463'-0" - NORTH
DESIGN ENGINEERING
DRAWING LEGIBILITY CLASS 1
E-922-103
6 REVISED PER ECR-70026



FOR CONTINUATION SEE DRAWING E-922-103
58'-9" ℄ RADIUS OF RING DUCT
58'-6¼" ℄ STRAIGHT SEGMENT OF RING DUCT
58'-6¼" ℄ SUPPLY DUCT
59'-1" ℄ STRAIGHT SEGMENT OF RING DUCT
59'-3" ℄ RADIUS OF RING DUCT
62'-7" TO CORNER OF DUCT RISER (TYP. FOR 2)
℄ REACTOR BLDG.
FL. EL. 463'-0"
FL. EL. 412'-0"
FLEXIBLE CONNECTION (SEE NOTE 2)
XPR-61-AH
XPR-60-AH
XPR-59-AH
XPR-58-AH
XPR-57-AH
XPR-56-AH
XPR-55-AH
XPR-54-AH
XPR-53-AH
XPR-52-AH
XPR-51-AH
XPR-50-AH
XPR-49-AH
XEJ-45-AH
XEJ-46-AH REACTOR BLDG. RING DUCT EXPANSION JOINT
XEJ-47-AH REACTOR BLDG. RING DUCT EXPANSION JOINT
XEJ-48-AH REACTOR BLDG. RING DUCT EXPANSION JOINT
XEJ-49-AH REACTOR BLDG. RING DUCT EXPANSION JOINT
XEJ-50-AH REACTOR BLDG. RING DUCT EXPANSION JOINT
XEJ-51-AH REACTOR BLDG. RING DUCT EXPANSION JOINT
XEJ-52-AH REACTOR BLDG RING DUCT EXPANSION JOINT
XEJ-5001-AH
XMD-191-AH
FOR REINFORCING DETAILS SEE PLAN 'D' DWG. E-922-002
FOR REINFORCING DETAIL SEE PLAN 'C' DWG. E-922-002
24"x24" 5000 CFM SEE DETAIL "A" DWG. E-922-103
NOTE: THIS SECTION OF DUCT TO BE FABRICATED AND INSTALLED BASED ON FIELD MEASUREMENTS. SECTION TO BE USED AS FINAL CLOSURE PIECE TO COMPENSATE FOR FIELD TOLERANCES.
8"x2'-0" PRESSURE RELIEF BOT OF DUCT EL. 504'-0" (TYP. 13 PLACES)
FOR DUCT SUPPORTS SEE DWG. E-936-104
5° (TYPICAL 7 PLACES)
℄ 84"x30" DUCT RISER DOWN THRU FLR EL. 463'-0"
℄ 84"x36" DUCT RISER INTO RING DUCT
℄ 84"x30" DUCT RISER DOWN THRU FLR EL. 463'-0" SEE AUX. PLAN THIS DWG.
270°
90°
180°
45°
65°
SECTION 8-8
AUX PLAN ABOVE FL. EL. 436'-0"
SECTION 9-9
E-922-103
KEY PLAN
THIS DWG.
NOTES:-
1. FOR NOTES AND REFERENCES, SEE DRAWING E-922-101.
2. FLEXIBLE CONNECTIONS TO BE VENTLON BY VENTFABRICS INC. (OR EQUAL)
THIS IS A NUCLEAR SAFETY RELATED DOCUMENT. NO DEVIATION SHALL BE INITIATED OR PERFORMED WITHOUT PRIOR DOCUMENTATION AND WRITTEN APPROVAL
Amendment 02-01 May 2002
FSAR Figure 6.2-50, Sht. 4
DRAWING LEGIBILITY CLASS 1
SOUTH CAROLINA ELECTRIC & GAS COMPANY
VIRGIL C. SUMMER NUCLEAR STATION
BUILDING SERVICE HVAC
REACTOR BUILDING COOLING SYSTEM
PLAN ABOVE ELEVATION 463'-0" - SOUTH
DESIGN ENGINEERING
REVISED PER ECR-70026
REVISION
DDJ MGR DDJ
E-922-104



NOTES
1 FOR NOTES AND REFERENCES SEE DRAWING E-922-101.

THIS IS A NUCLEAR SAFETY RELATED DOCUMENT. NO DEVIATION SHALL BE INITIATED OR PERFORMED WITHOUT PRIOR DOCUMENTATION AND WRITTEN APPROVAL.

Amendment 02-01
May 2002

FSAR Figure 6.2-51

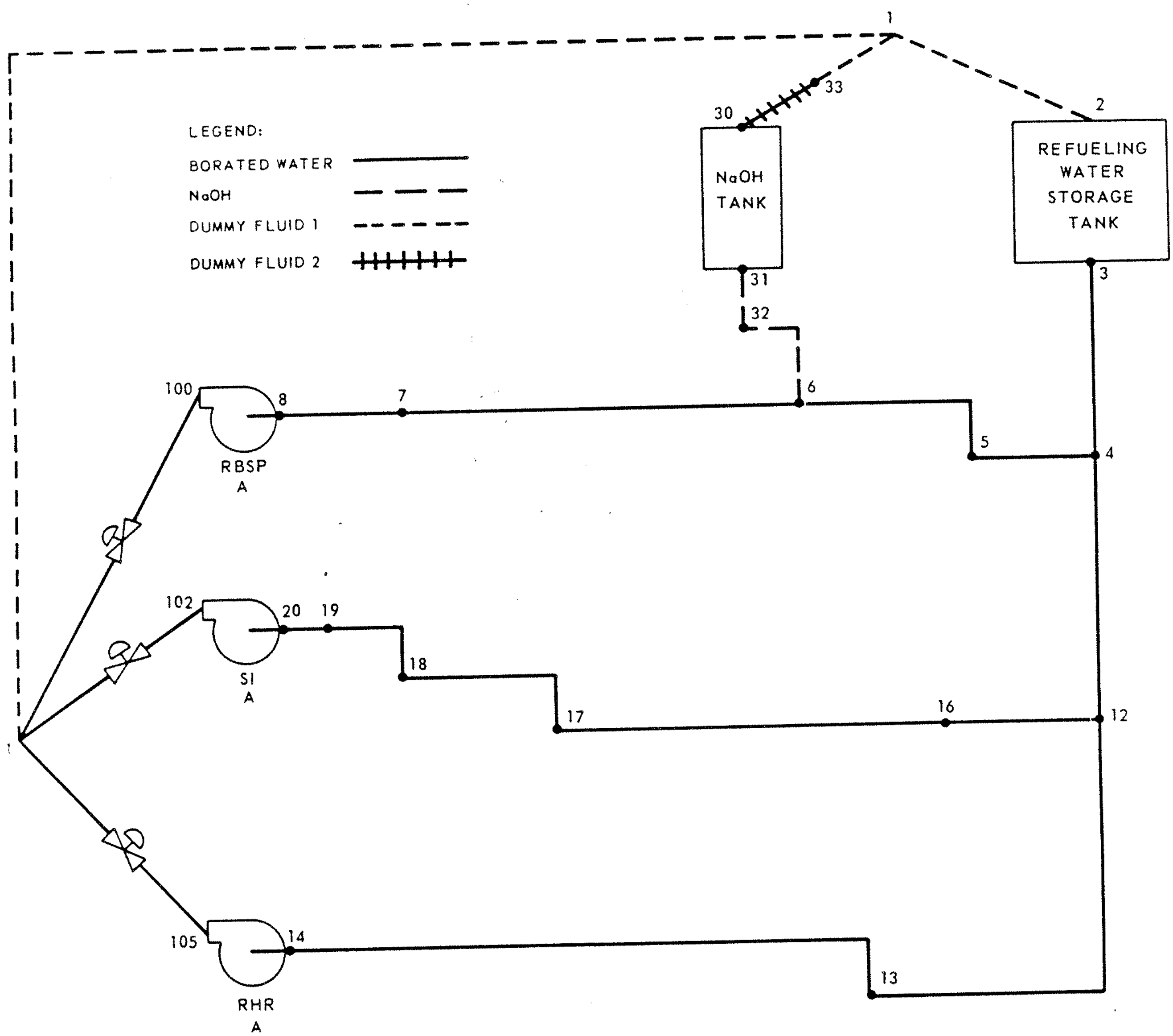
DRAWING LEGIBILITY CLASS I

SOUTH CAROLINA ELECTRIC & GAS COMPANY
VIRGIL C. SUMMER NUCLEAR STATION
BUILDING SERVICE HVAC
REACTOR BUILDING COOLING SYSTEM
SECTIONS
DESIGN ENGINEERING

5		DDJ	REVISED PER ECR-70028	MGR	DDJ	
NO.	DATE	BY	REVISION	CHD. BY	APPROVAL	

DDJ | MGR | DDJ
E-922-105 | 5

1/4"=1'-0" 0 5' 10' 15' 20'

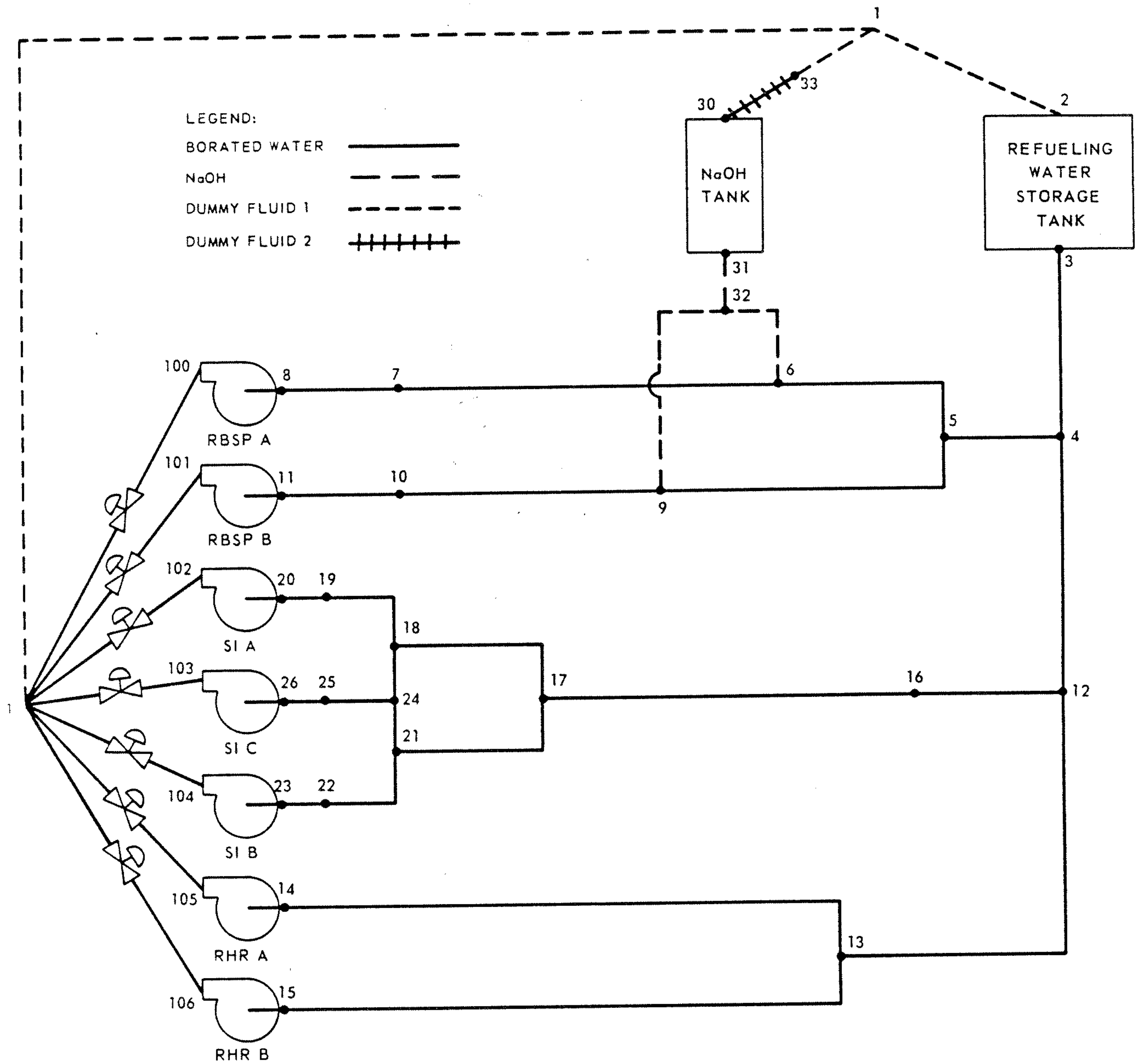


Amendment 0
August 1984

SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Model of ECCS and Reactor Building Spray System for Reactor Building Spray System Design Case

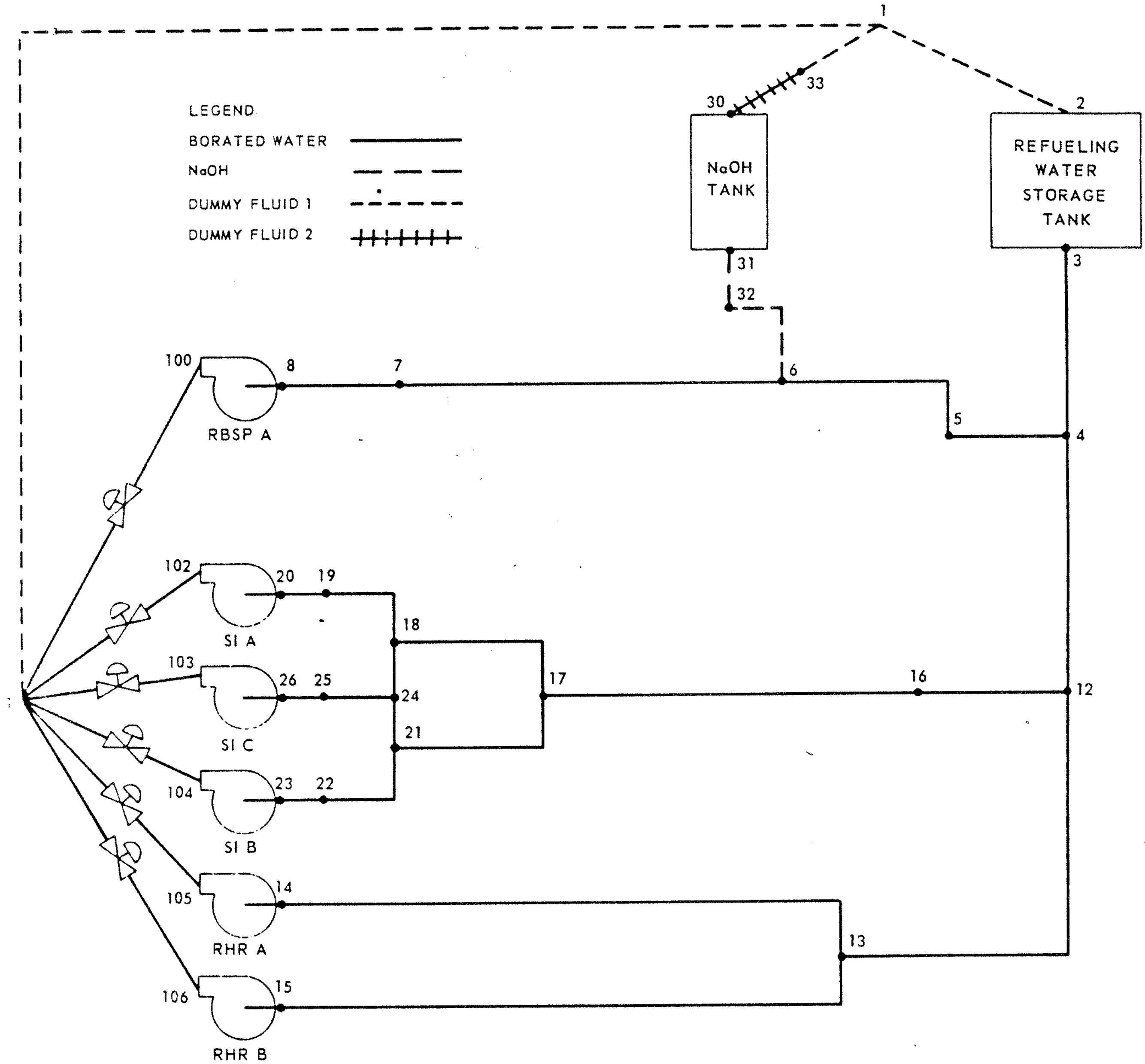
Figure 6.2-51a



SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Model of ECCS and Reactor Building Spray System for Normal Case

Figure 6.2-51b

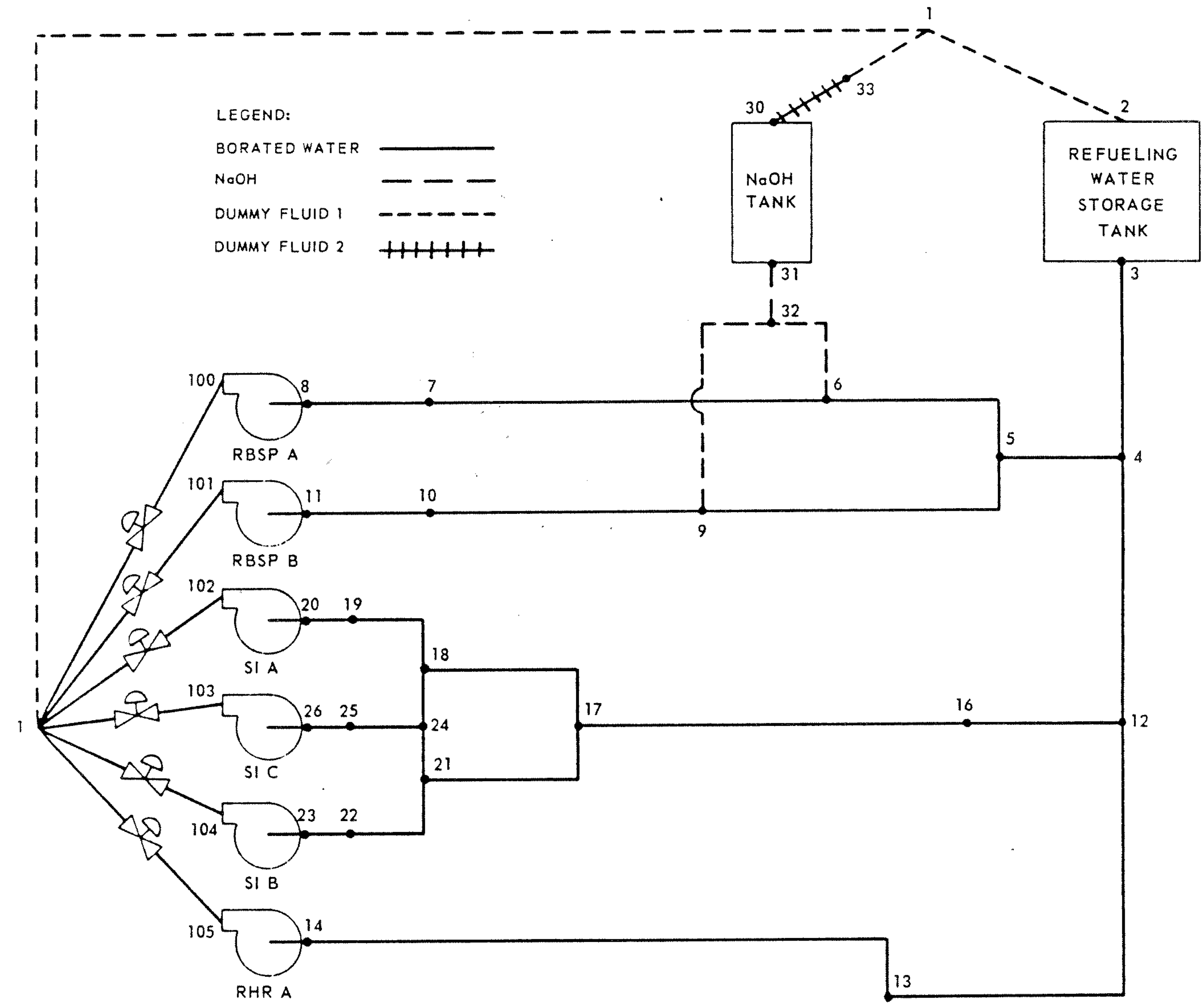


Amendment 0
August 1984

SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Model of ECCS and Reactor Building Spray System for Normal Case with One Reactor Building Spray Pump Inoperable

Figure 6.2-51c

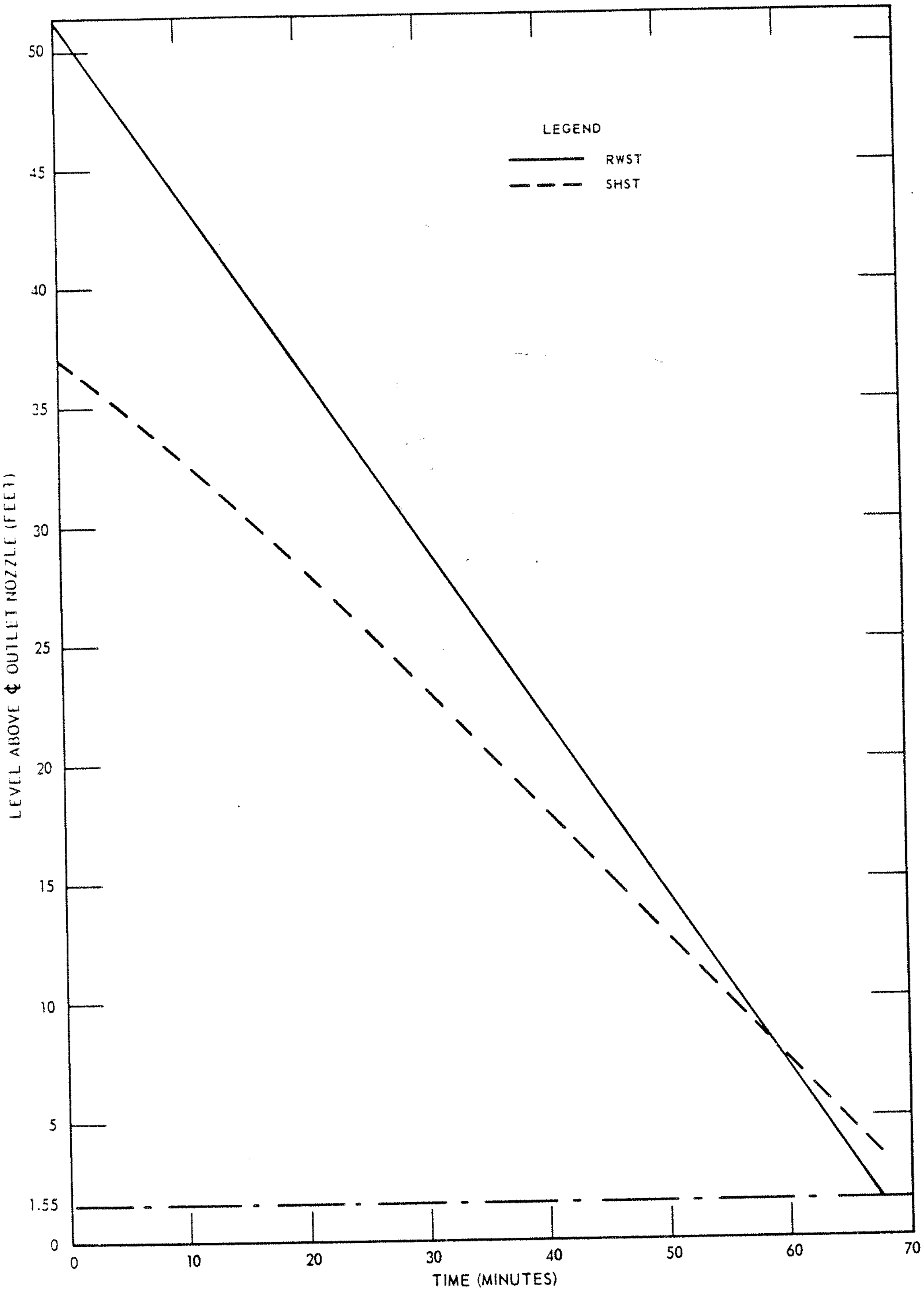


Amendment 0
August 1984

SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Model of ECCS and Reactor Building Spray System for Normal Case with One Residual Heat Removal Pump Inoperable

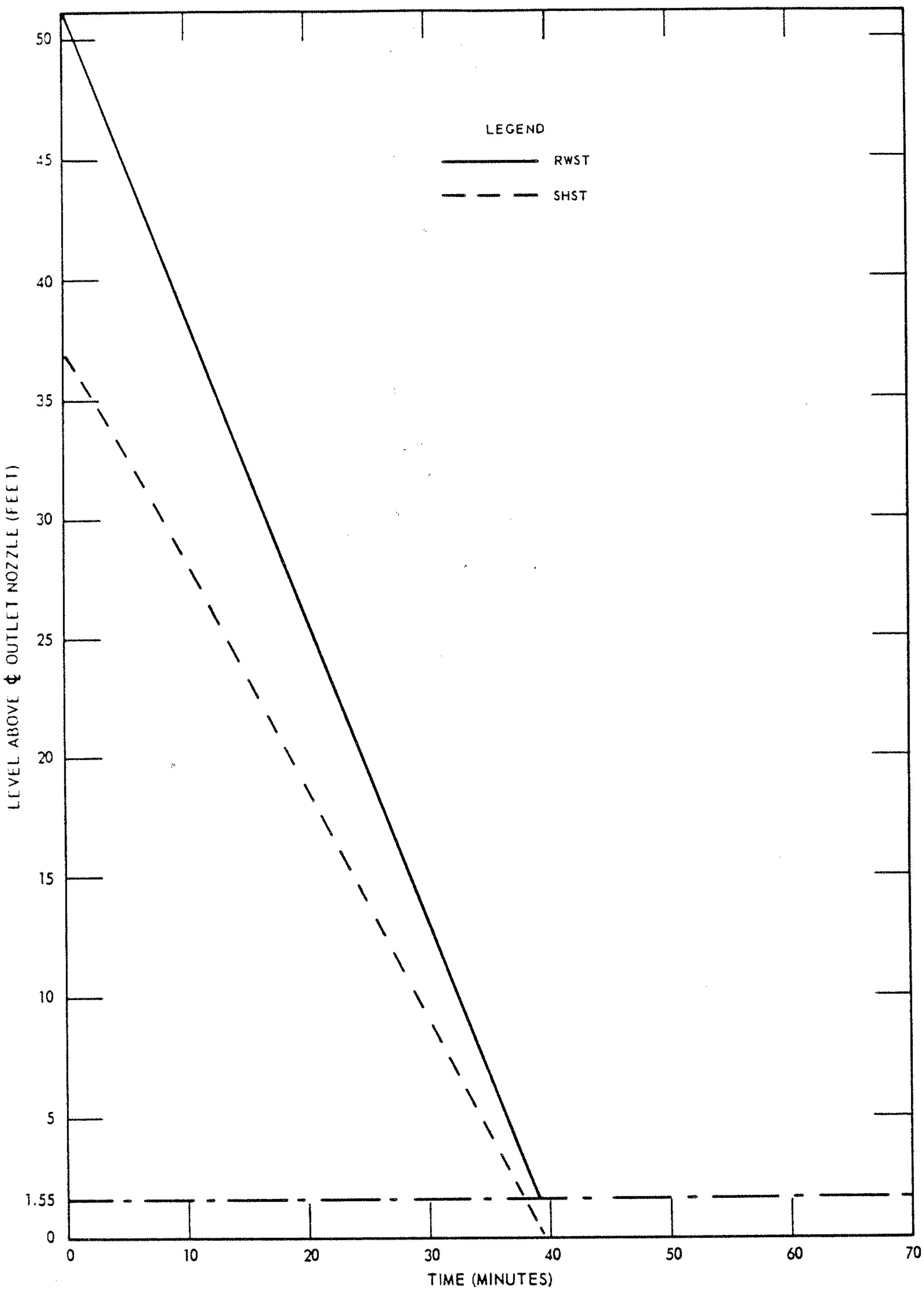
Figure 6.2-51d



SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Analytical Results - Chemical Drawdown - Reactor Building Spray System Design Case - Minimum Sodium Hydroxide Initial Conditions
Figure 6.2-51e

Amendment 0
August 1984

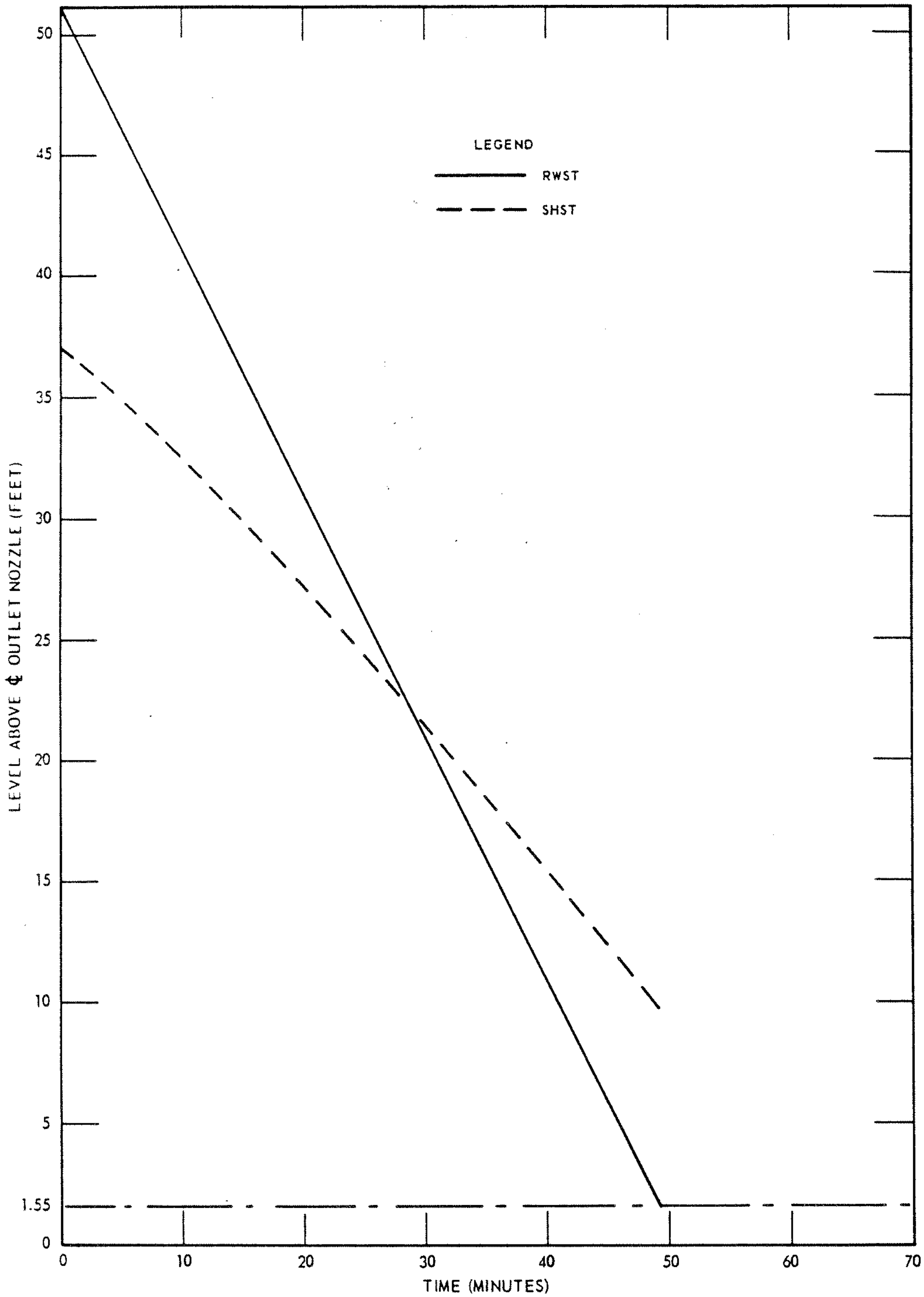


Amendment 0
August 1984

SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Analytical Results - Chemical Drawdown - Normal Case - Minimum Sodium Hydroxide Initial Conditions

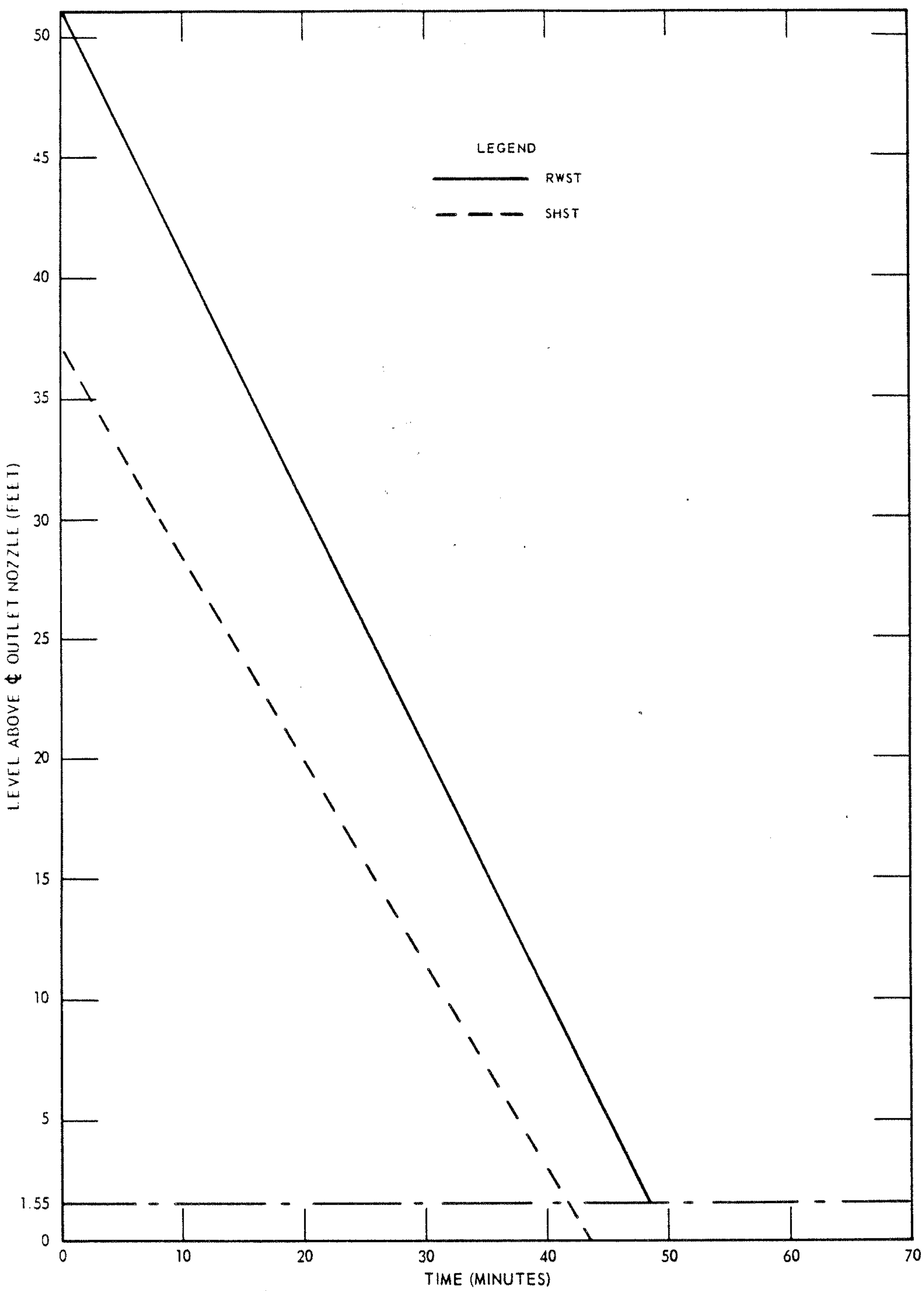
Figure 6.2-51f



Amendment 0
August 1984

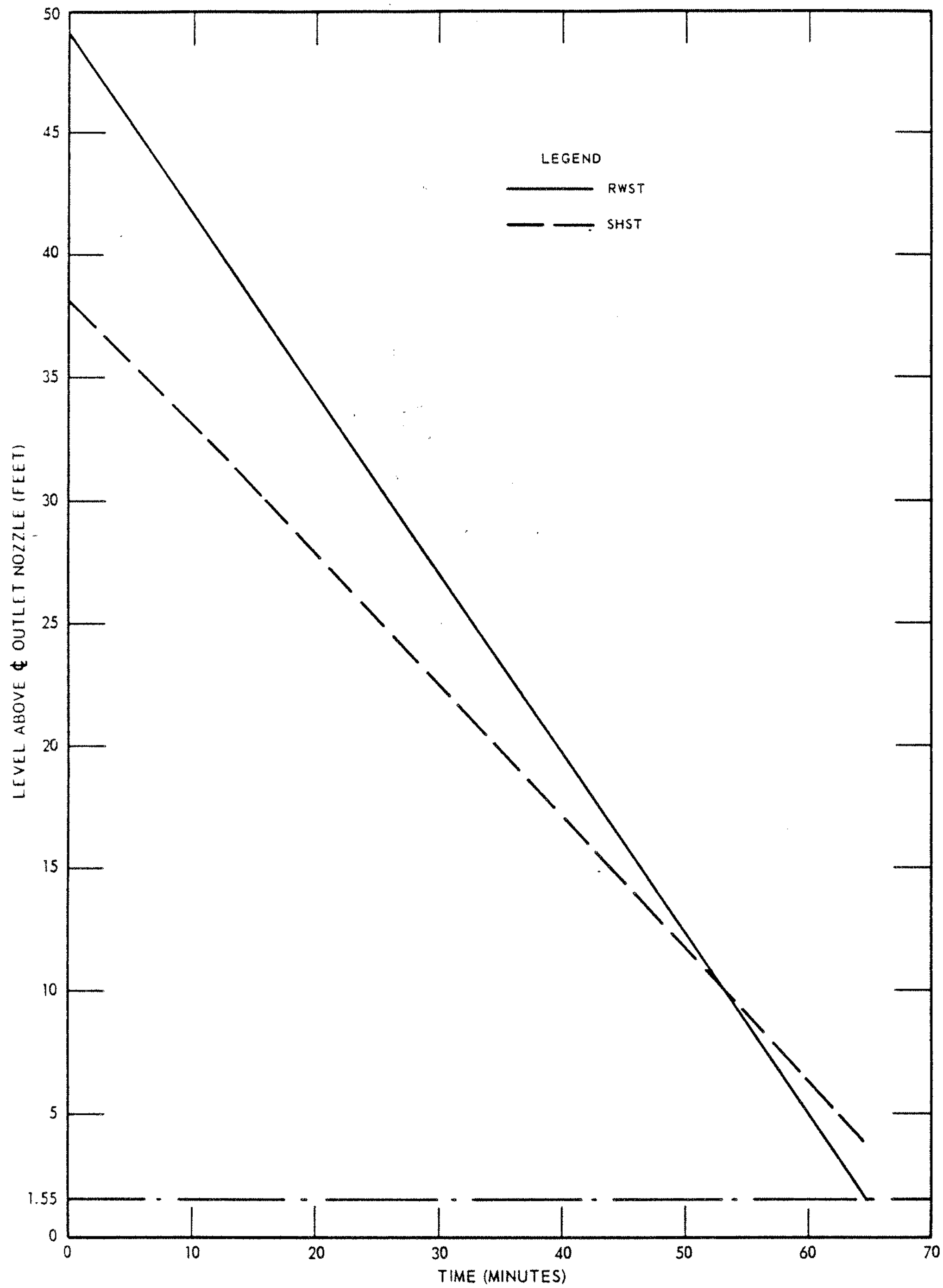
SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Analytical Results - Chemical Drawdown - Normal Case - With One Reactor Building Spray Pump Inoperable - Minimum Sodium Hydroxide Initial Conditions
Figure 6.2-51g



SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

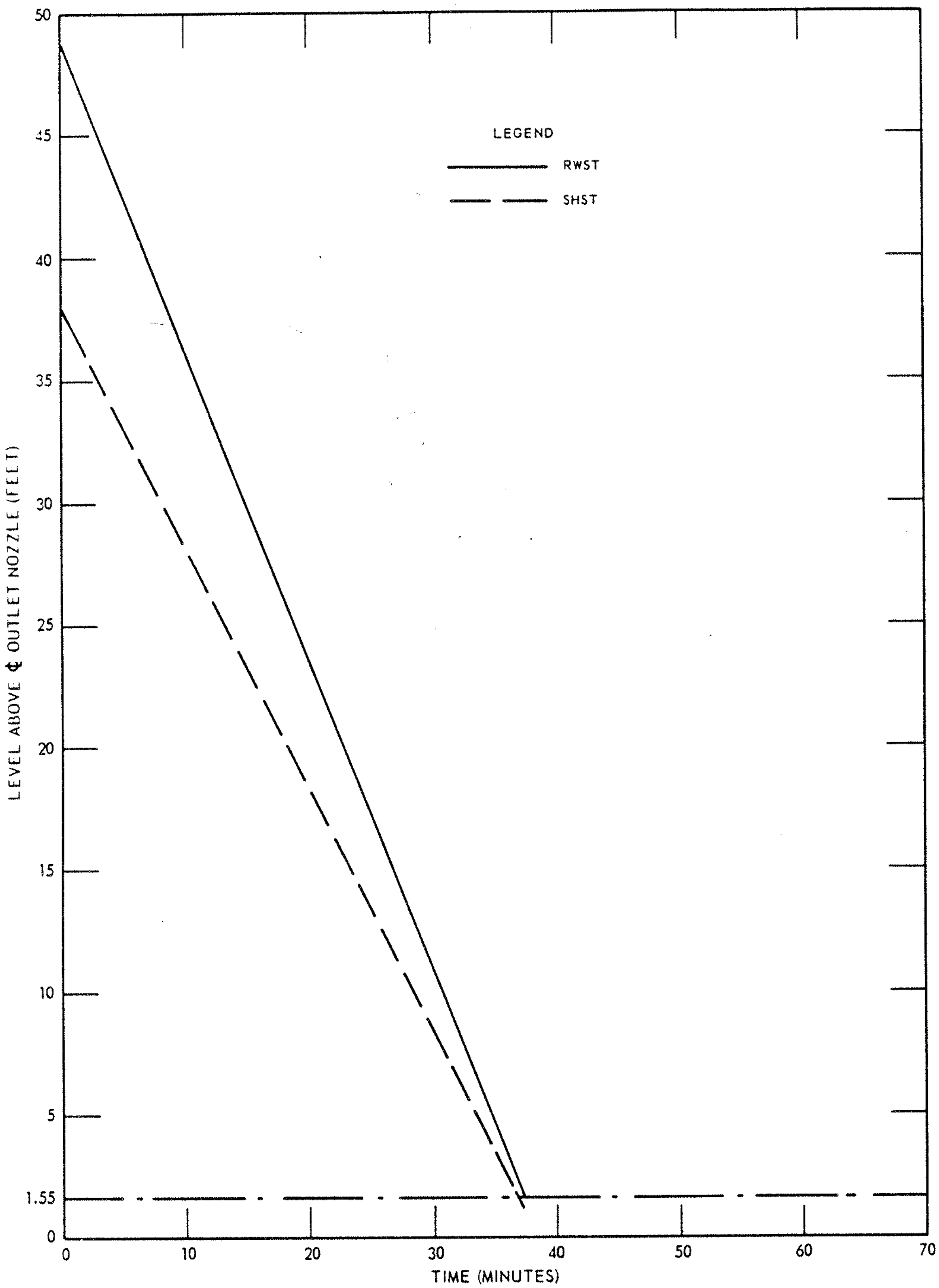
Analytical Results - Chemical Drawdown - Normal Case - With One Residual Heat Removal Pump Inoperable - Minimum Sodium Hydroxide Initial Conditions
Figure 6.2-51h



SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Analytical Results - Chemical Drawdown - Reactor Building Spray System Design Case - Maximum Sodium Hydroxide Initial Conditions

Figure 6.2-51i

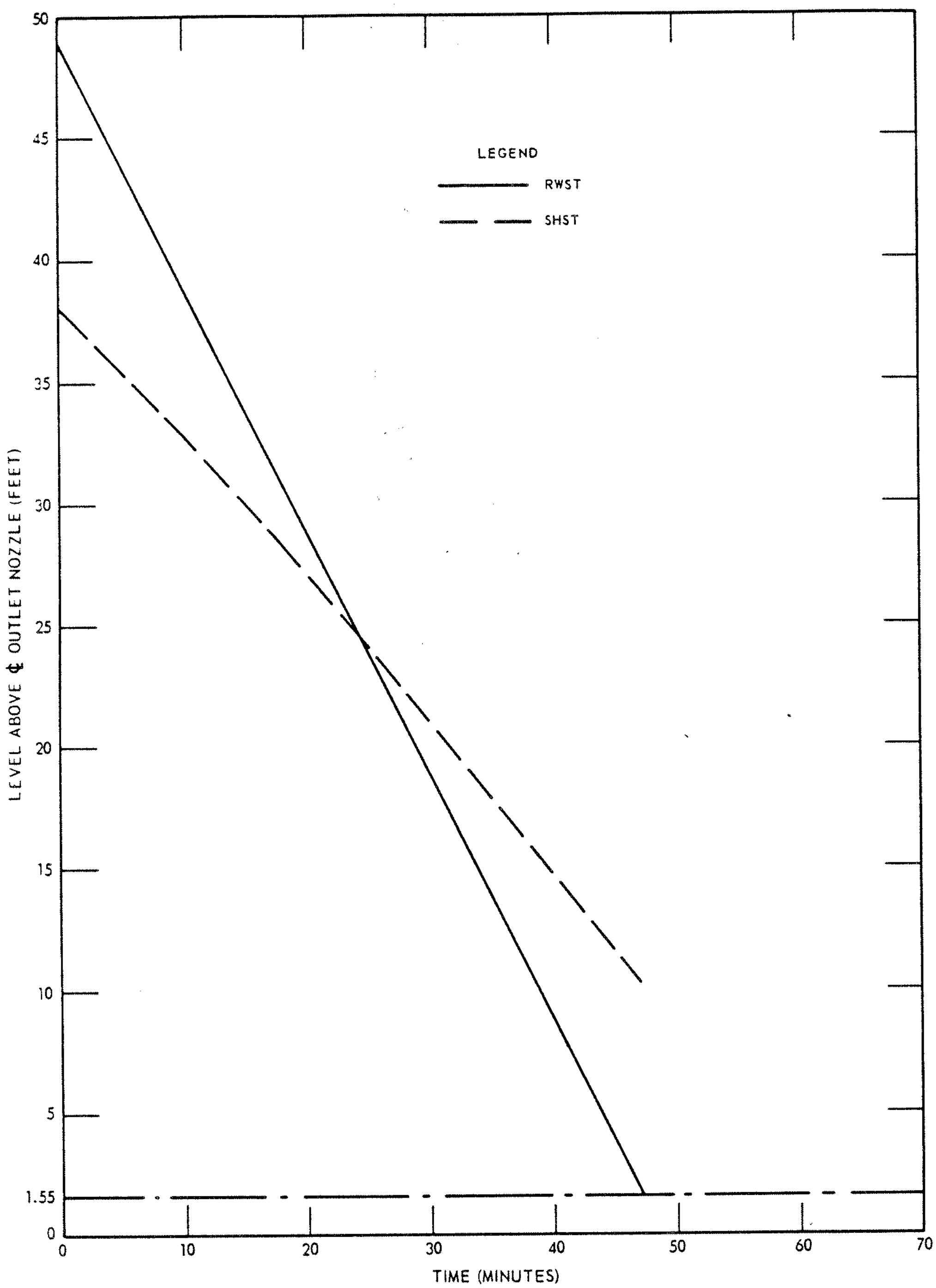


SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Analytical Results - Chemical Drawdown - Normal Case - Maximum Sodium Hydroxide Initial Conditions

Figure 6.2-51j

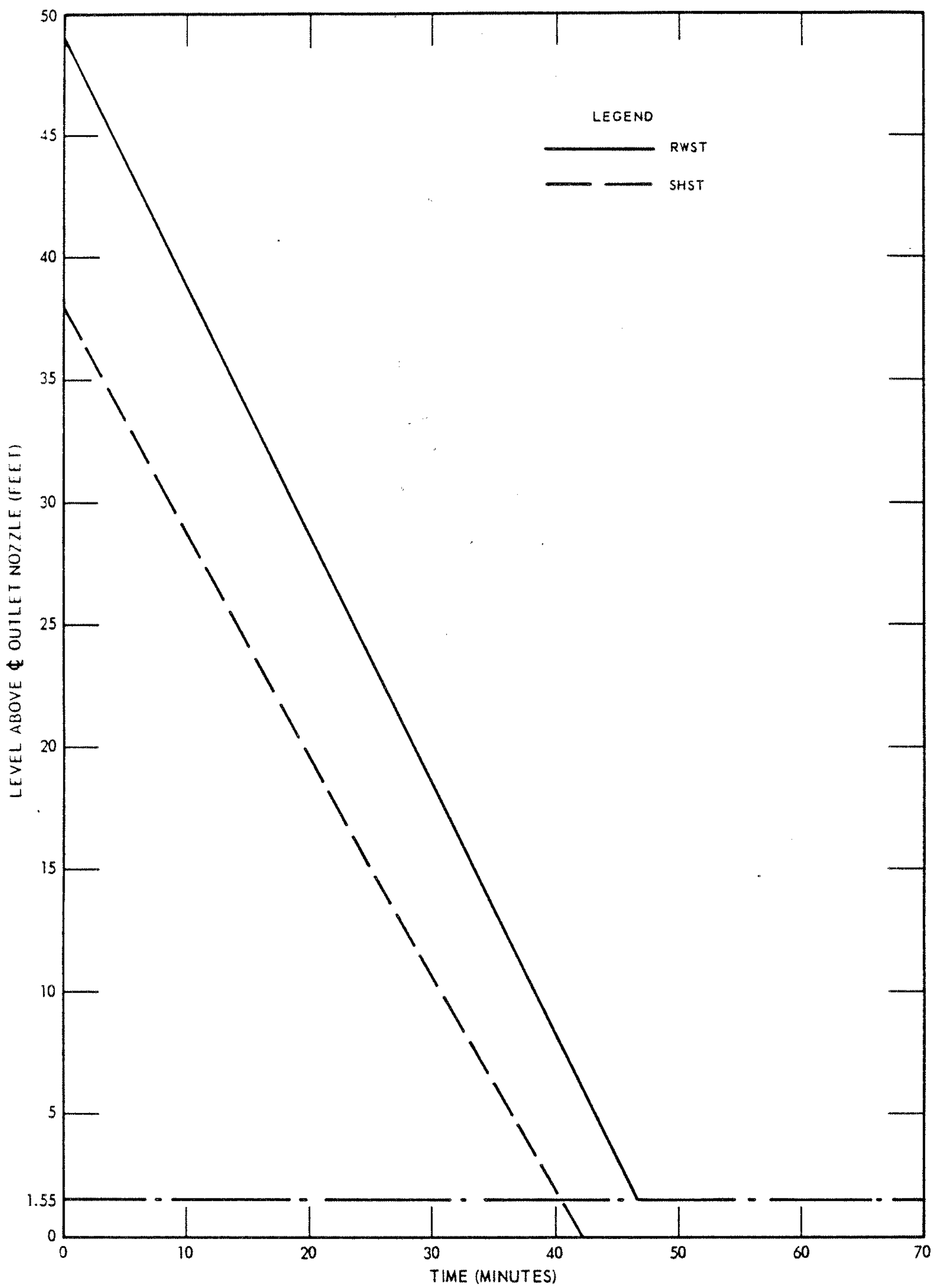
Amendment 0
August 1984



Amendment 0
August 1984

SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

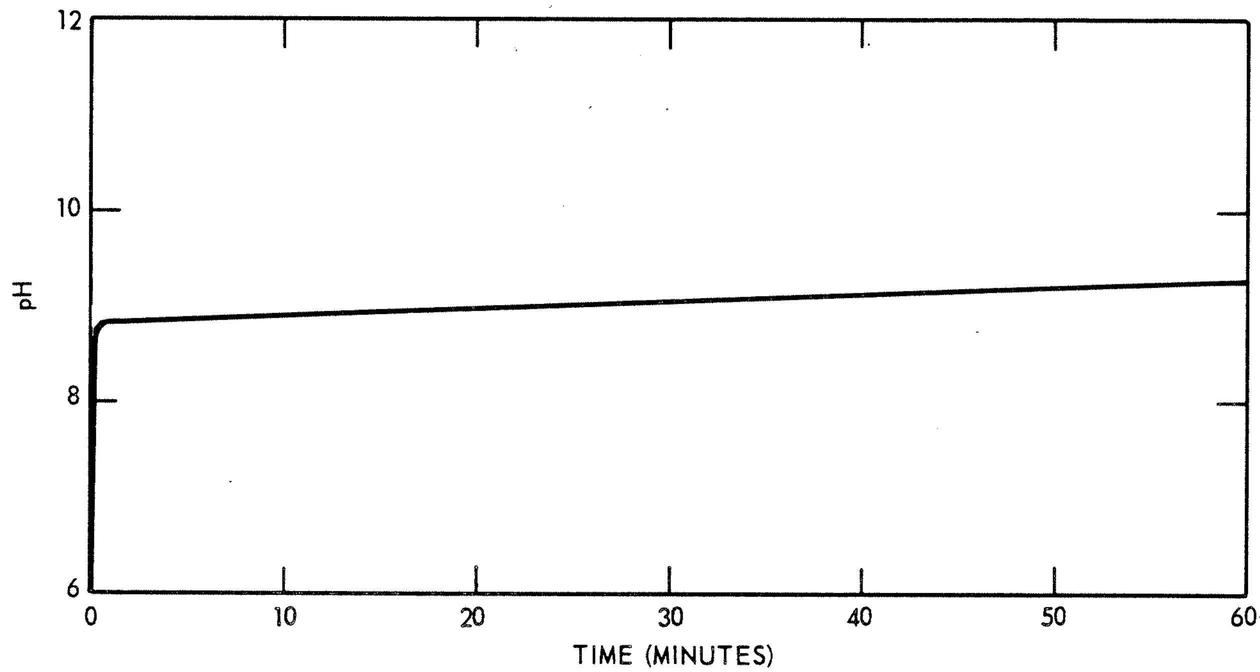
Analytical Results - Chemical Drawdown - Normal Case With One Reactor Building Spray Pump Inoperable - Maximum Sodium Hydroxide Initial Conditions
Figure 6.2-51k



Amendment 0
August 1984

SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

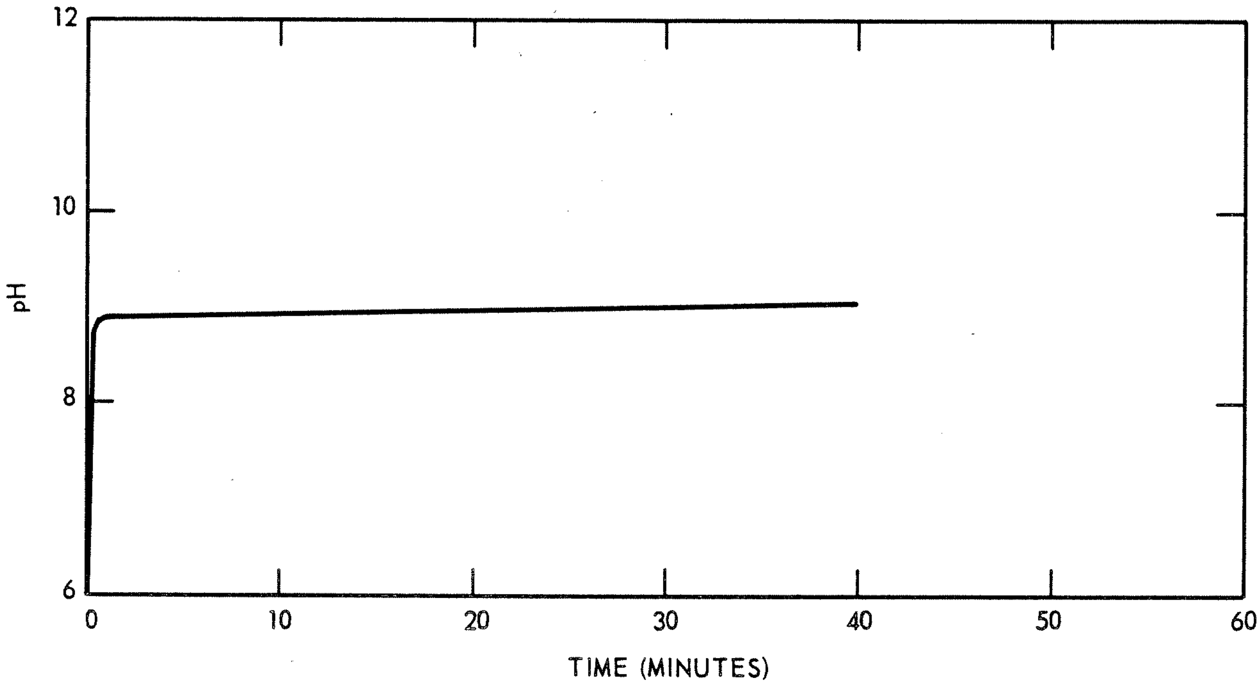
Analytical Results - Chemical Drawdown - Normal Case With One Residual Heat Removal Pump Inoperable - Maximum Sodium Hydroxide Initial Conditions
Figure 6.2-51I



Amendment 6
August 1990

SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Spray Header pH During Initial Injection - Minimum Sodium Hydroxide Initial Conditions - Reactor Building Spray System Design Case
Figure 6.2-51m

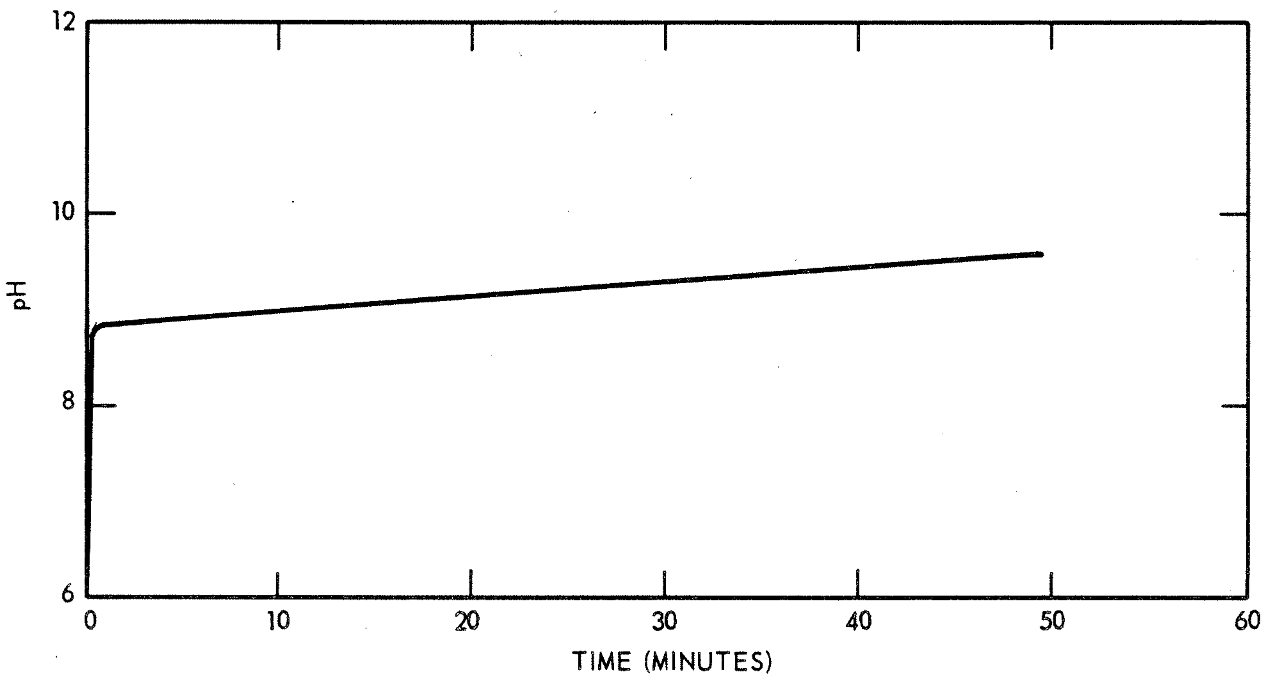


Amendment 6
August 1990

SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Spray Header pH During Initial Injection - Minimum Sodium Hydroxide Initial Conditions - Normal Case

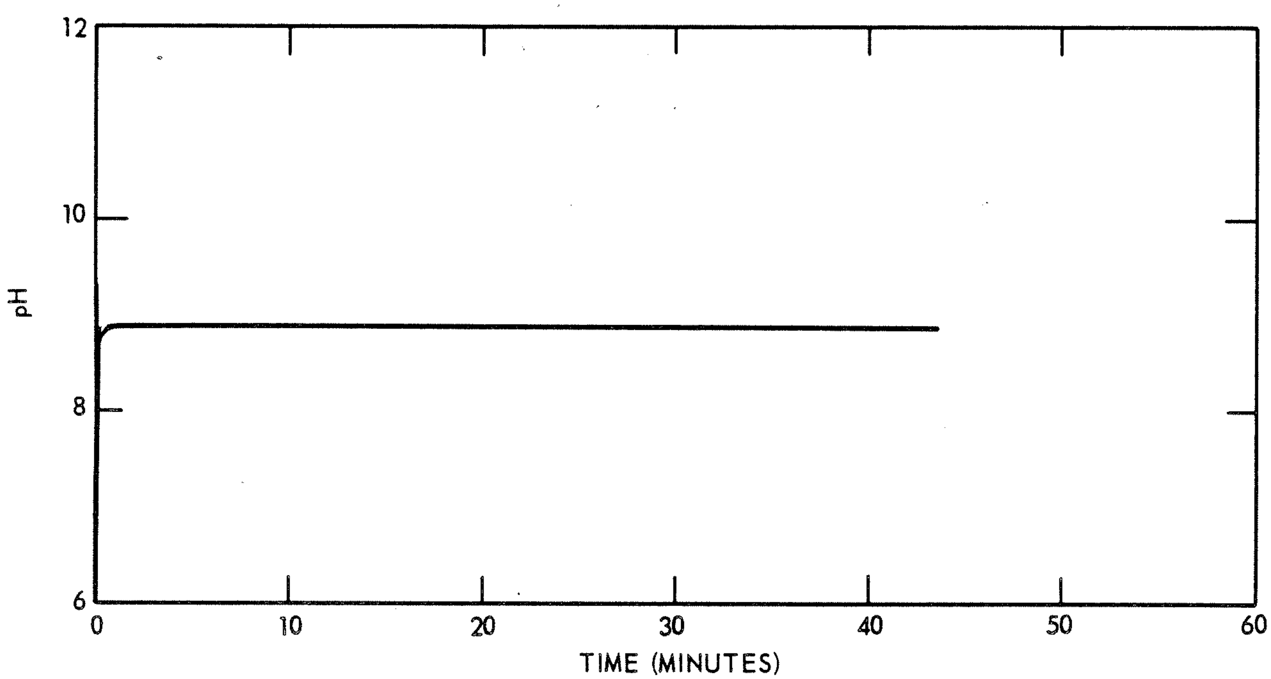
Figure 6.2-51n



Amendment 6
August 1990

SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

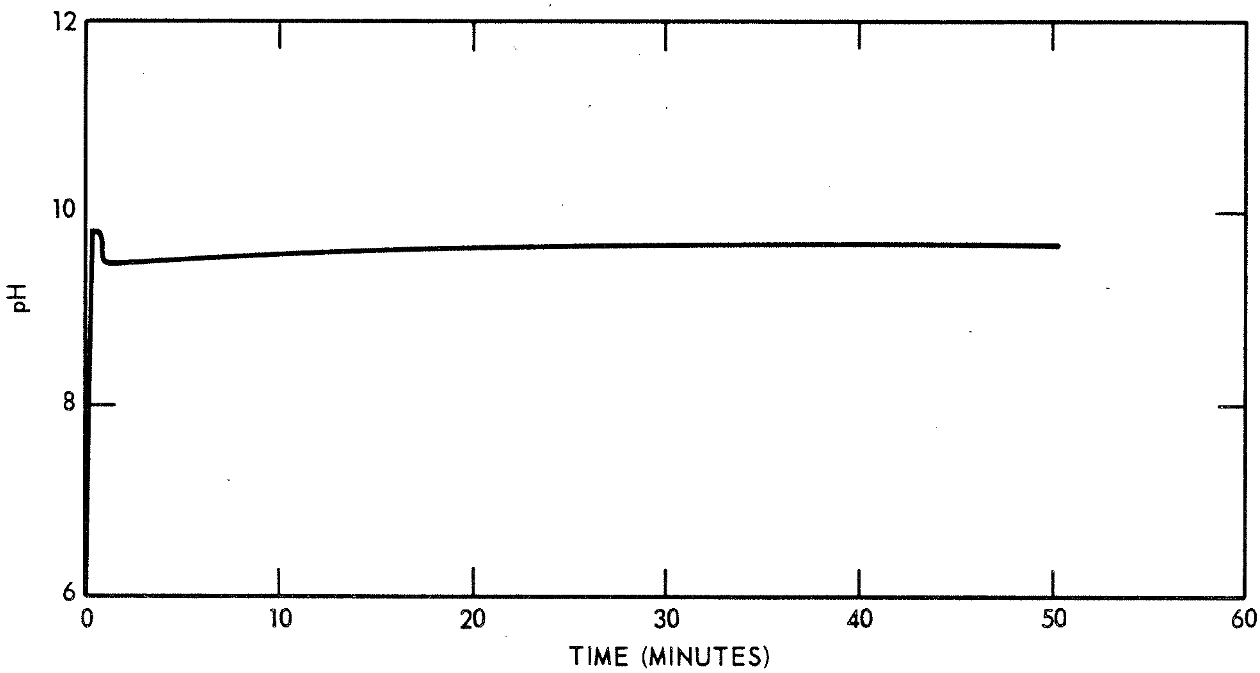
Spray Header pH During Initial Injection - Minimum Sodium Hydroxide Initial Conditions - Normal Case with One Spray Pump Inoperable
Figure 6.2-51o



Amendment 6
August 1990

SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

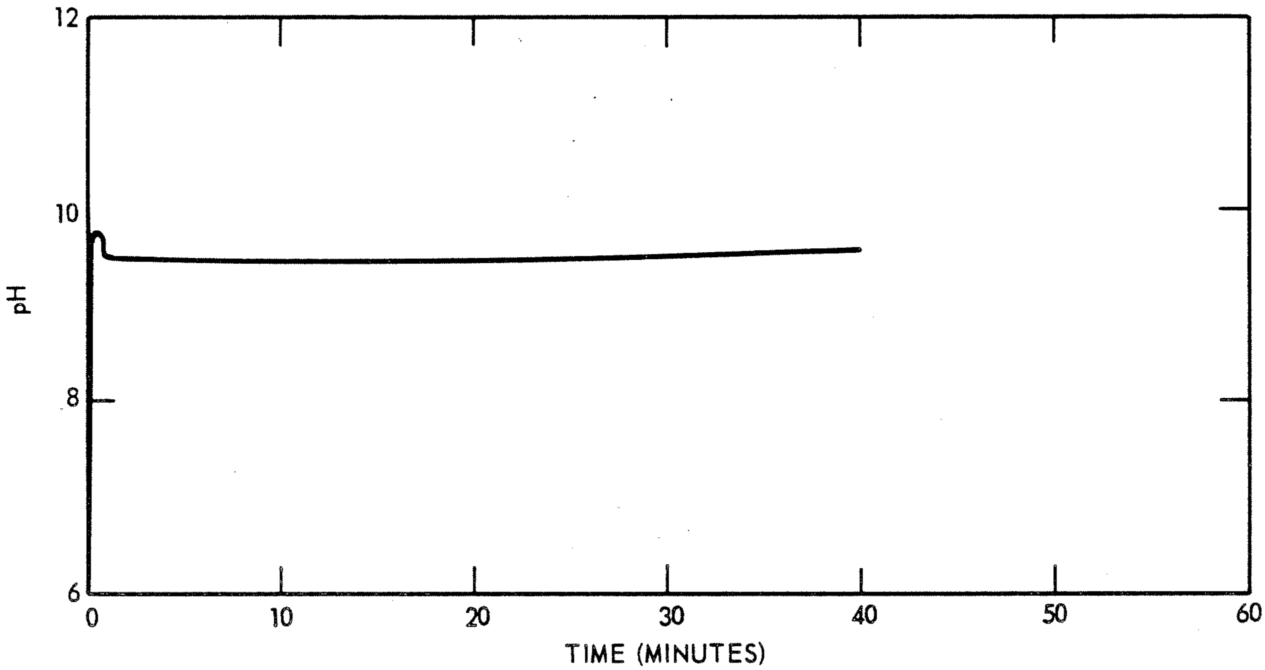
Spray Header pH During Initial Injection - Minimum Sodium Hydroxide Initial Conditions - Normal Case with One RHR Pump Inoperable
Figure 6.2-51p



Amendment 6
August 1990

SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Spray Header pH During Initial Injection - Maximum Sodium Hydroxide Initial Conditions - Reactor Building Spray System Design Case
Figure 6.2-51q

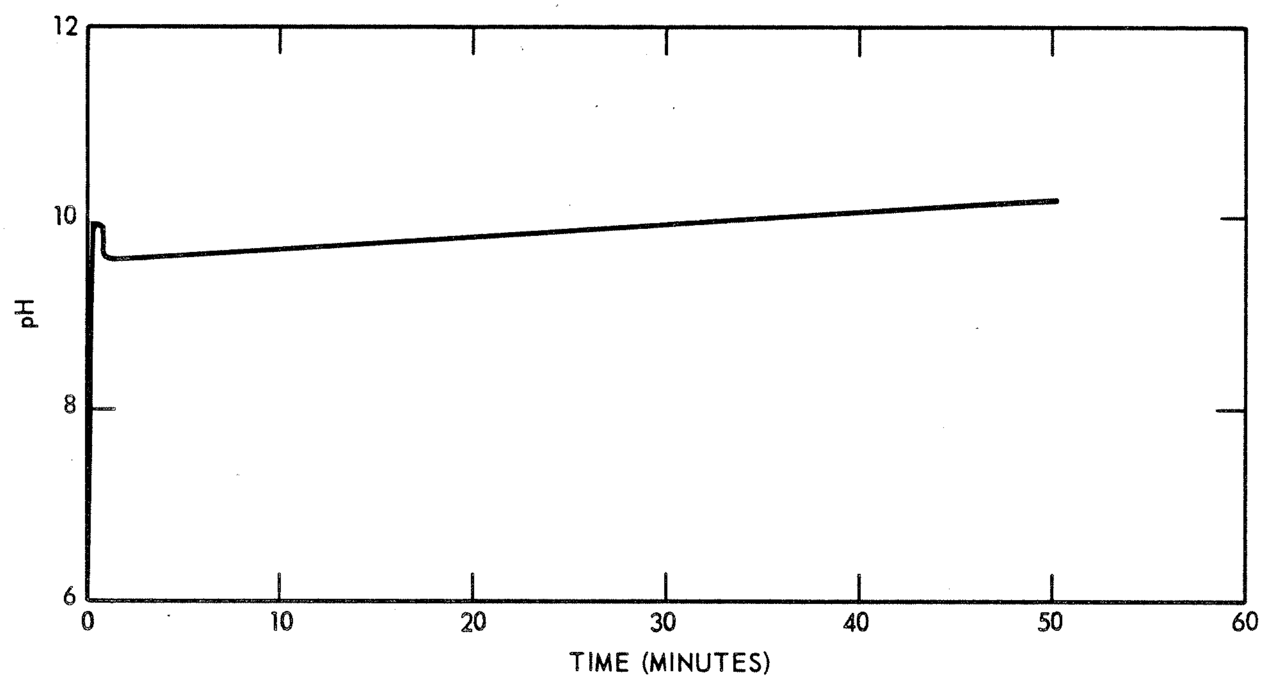


Amendment 6
August 1990

SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Spray Header pH During Initial Injection - Maximum Sodium Hydroxide Initial Conditions - Normal Case

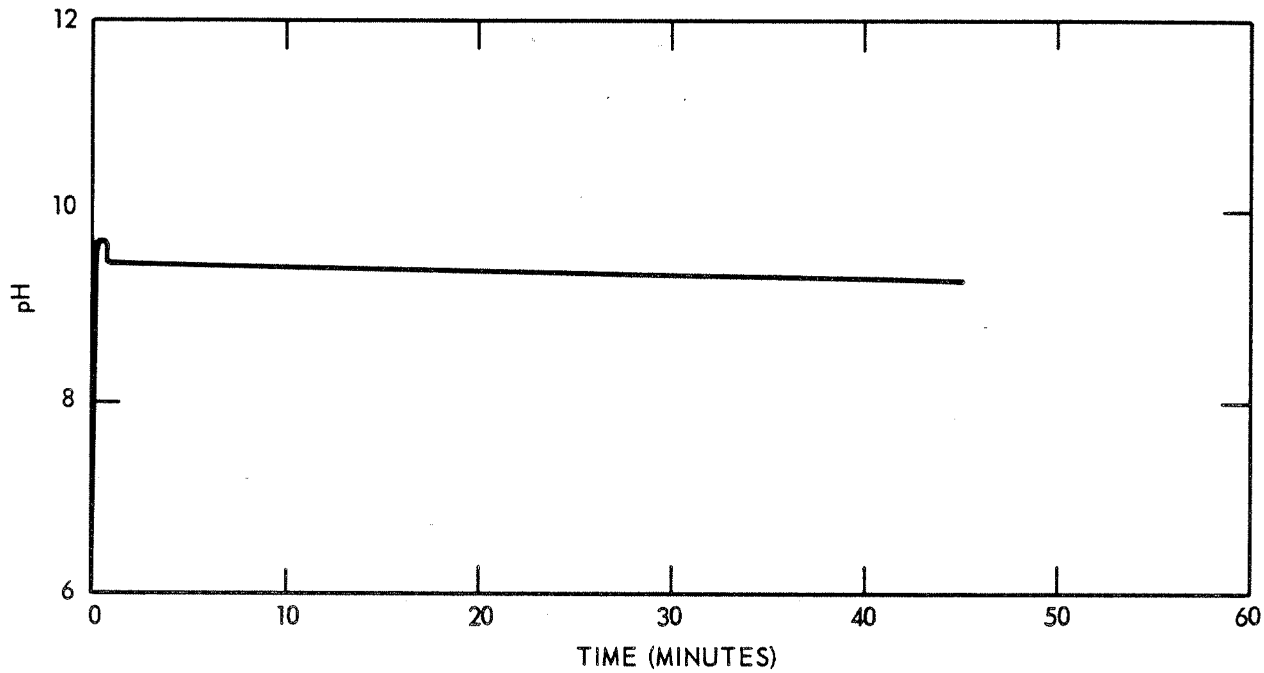
Figure 6.2-51r



Amendment 6
August 1990

SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

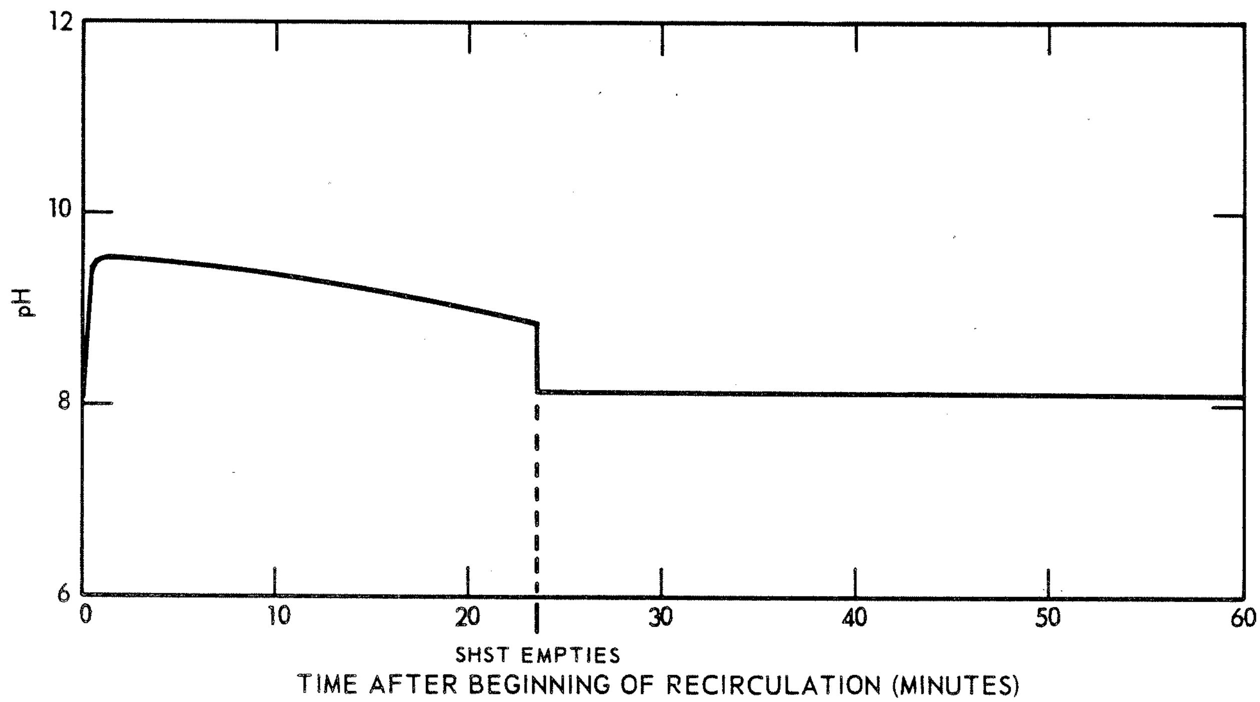
Spray Header pH During Initial Injection - Maximum Sodium Hydroxide Initial Conditions - Normal Case with One Spray Pump Inoperable
Figure 6.2-51s



Amendment 6
August 1990

SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

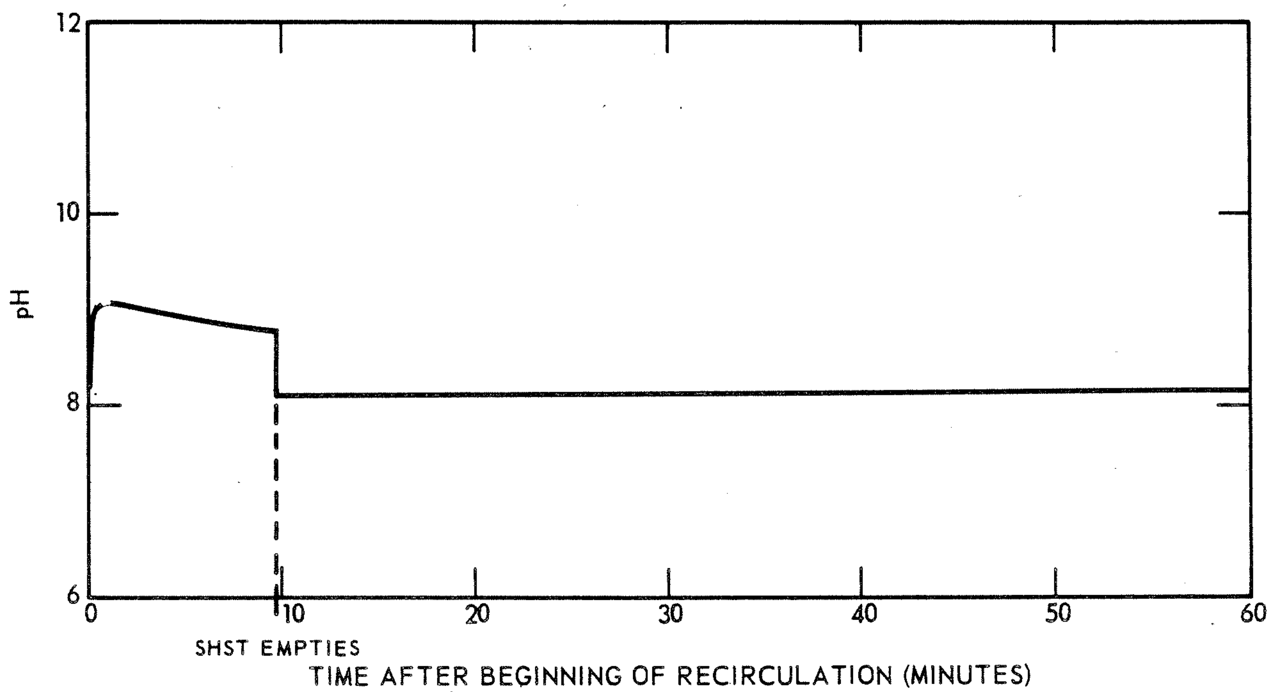
Spray Header pH During Initial Injection - Maximum Sodium Hydroxide Initial Conditions - Normal Case with One RHR Pump Inoperable
Figure 6.2-51t



Amendment 6
August 1990

SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Spray Header pH During Recirculation - Minimum Sodium Hydroxide Initial Conditions - Reactor Building Spray System Design Case
Figure 6.2-51u

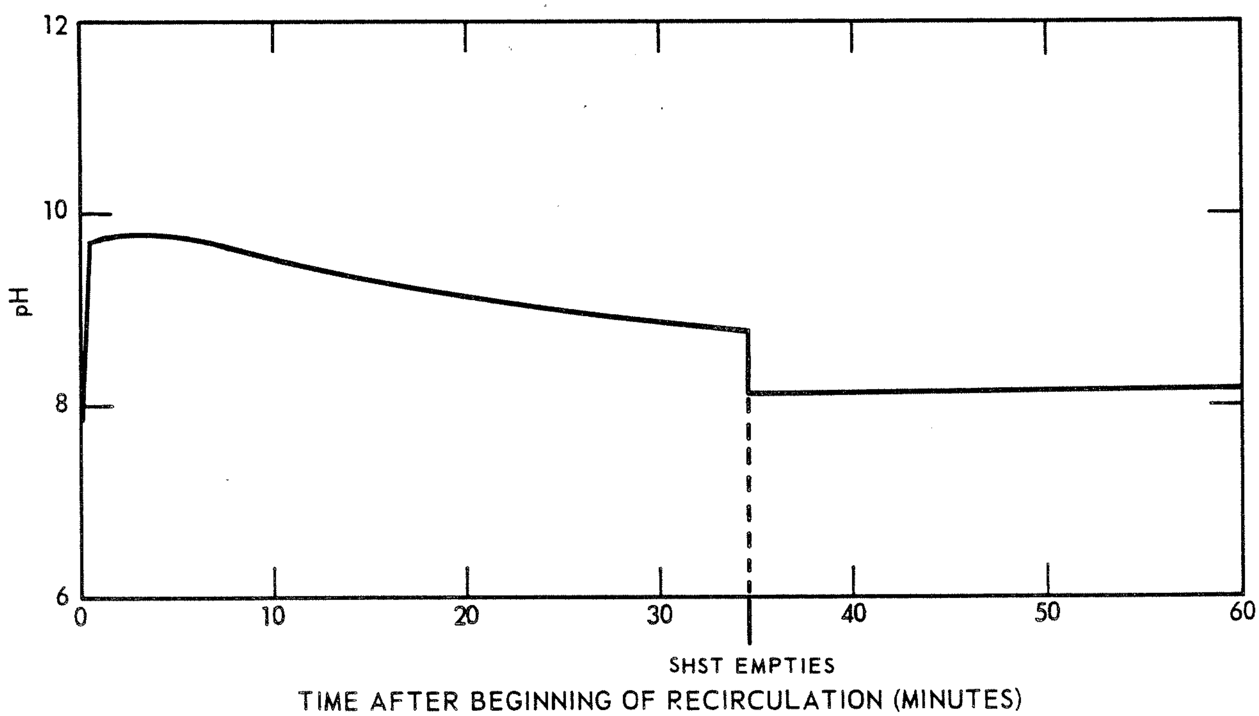


Amendment 6
August 1990

SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Spray Header pH During Recirculation - Minimum Sodium Hydroxide Initial Conditions - Normal Case

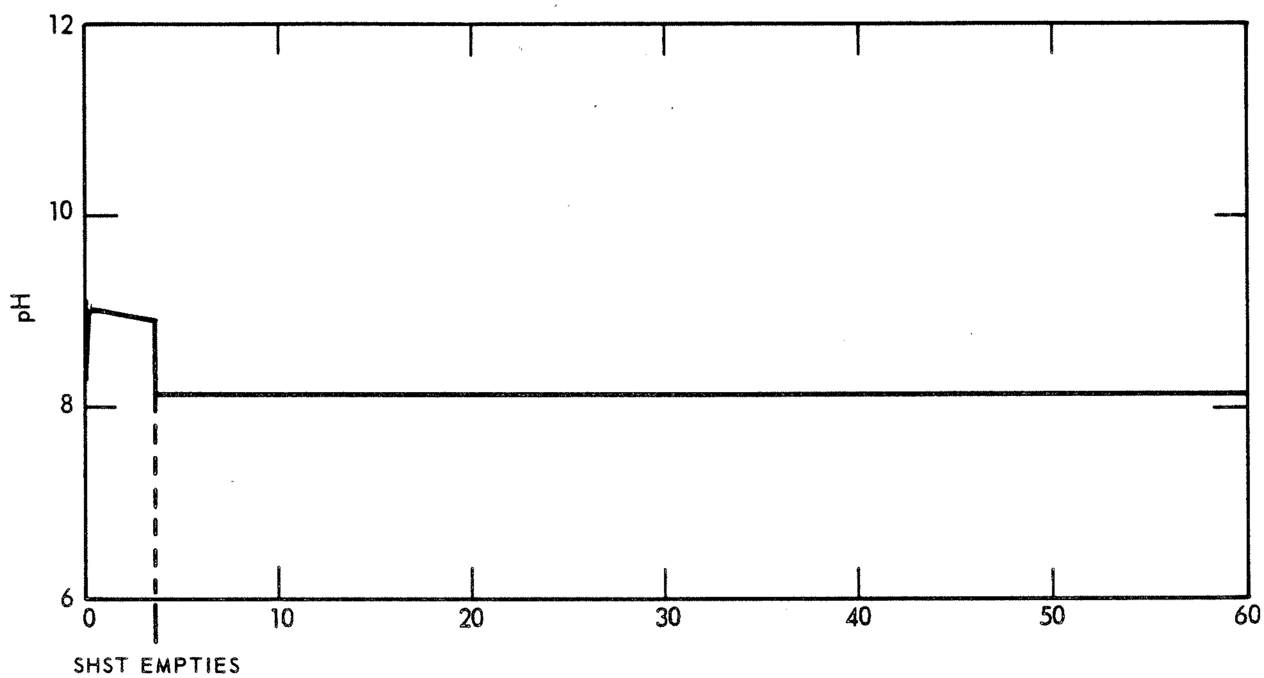
Figure 6.2-51v



Amendment 6
August 1990

SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

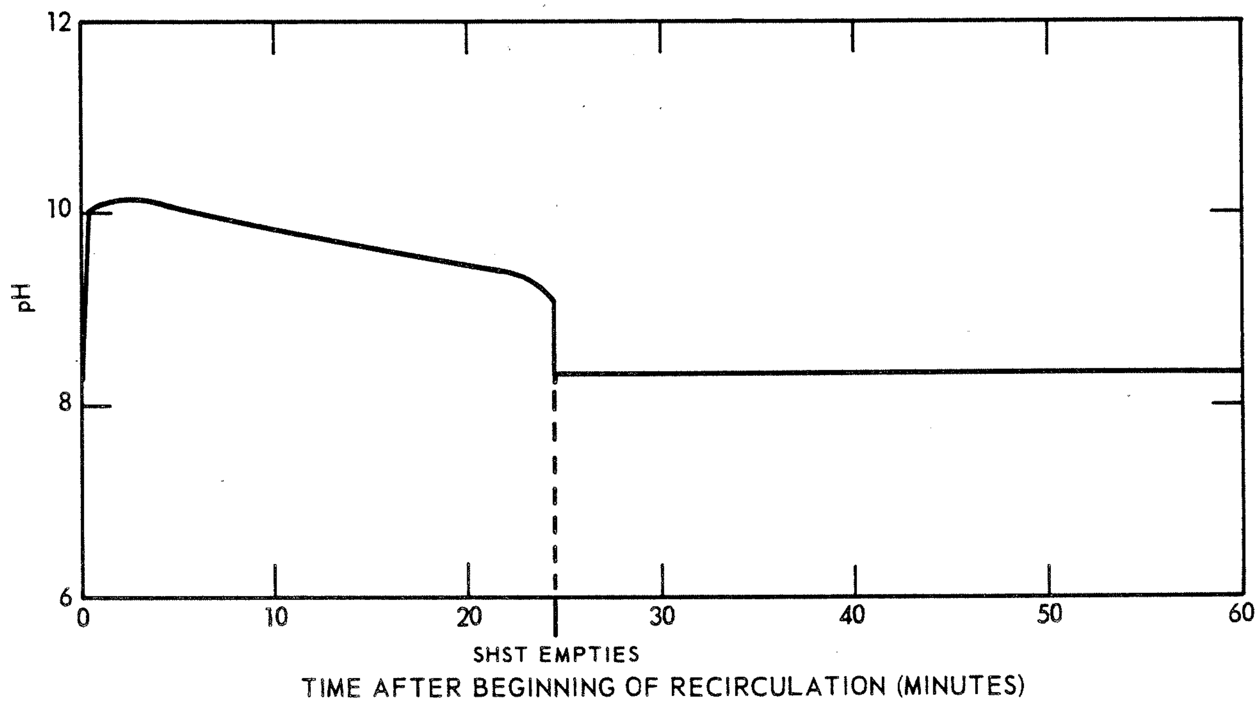
Spray Header pH During Recirculation - Minimum Sodium Hydroxide Initial Conditions - Normal Case with One Spray Pump Inoperable
Figure 6.2-51w



Amendment 6
August 1990

SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

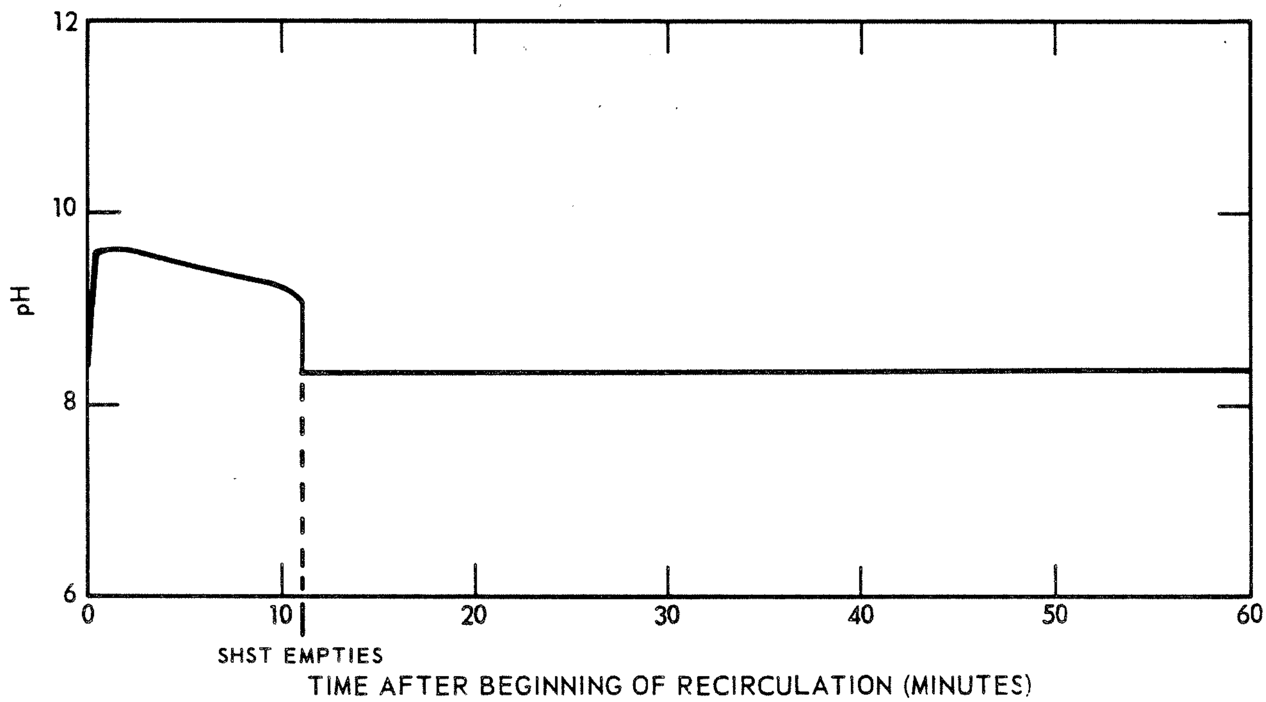
Spray Header pH During Recirculation - Minimum Sodium Hydroxide Initial Conditions - Normal Case with One RHR Pump Inoperable
Figure 6.2-51x



Amendment 6
August 1990

SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Spray Header pH During Recirculation - Maximum Sodium Hydroxide Initial Conditions - Reactor Building Spray System Design Case
Figure 6.2-51y

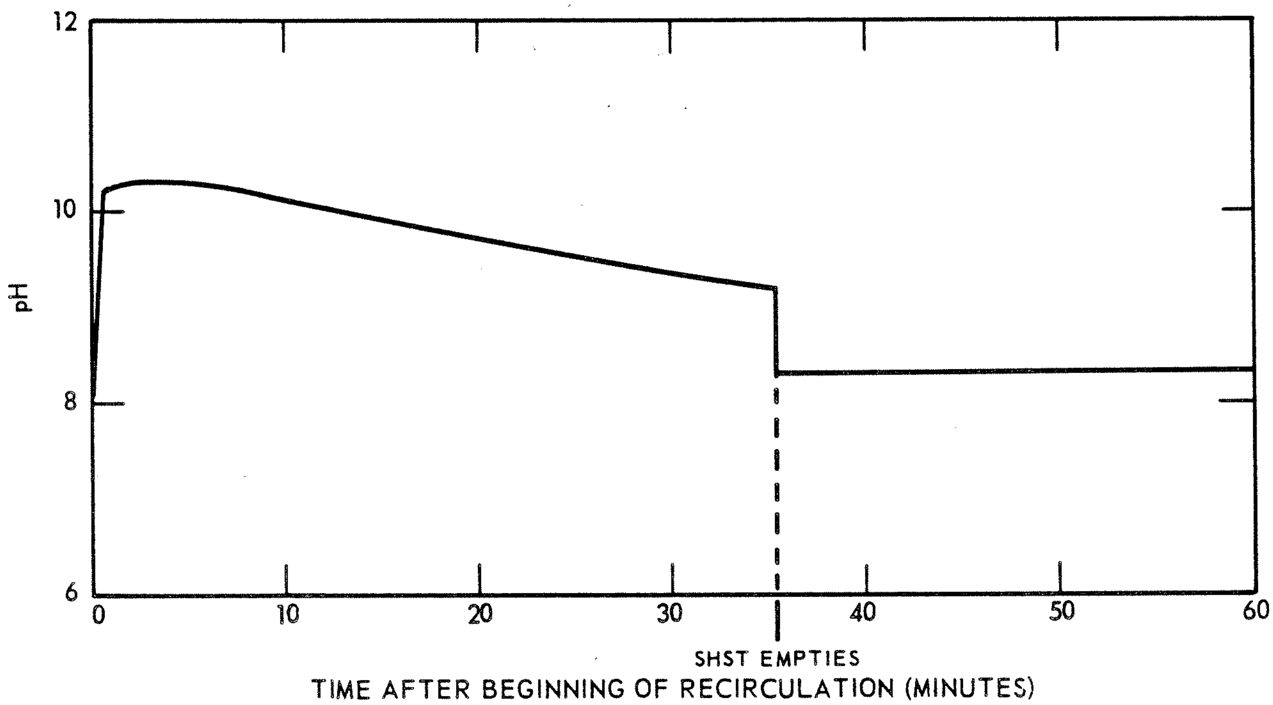


Amendment 6
August 1990

SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Spray Header pH During Recirculation - Maximum Sodium Hydroxide Initial Conditions - Normal Case

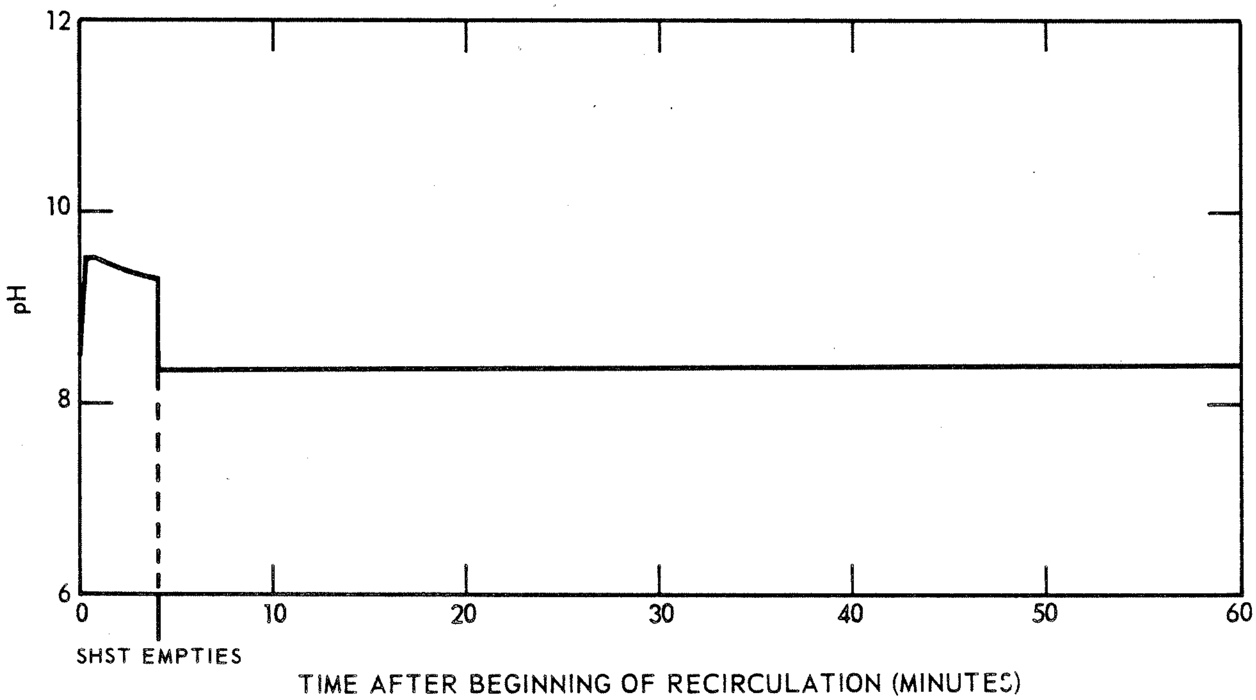
Figure 6.2-51z



Amendment 6
August 1990

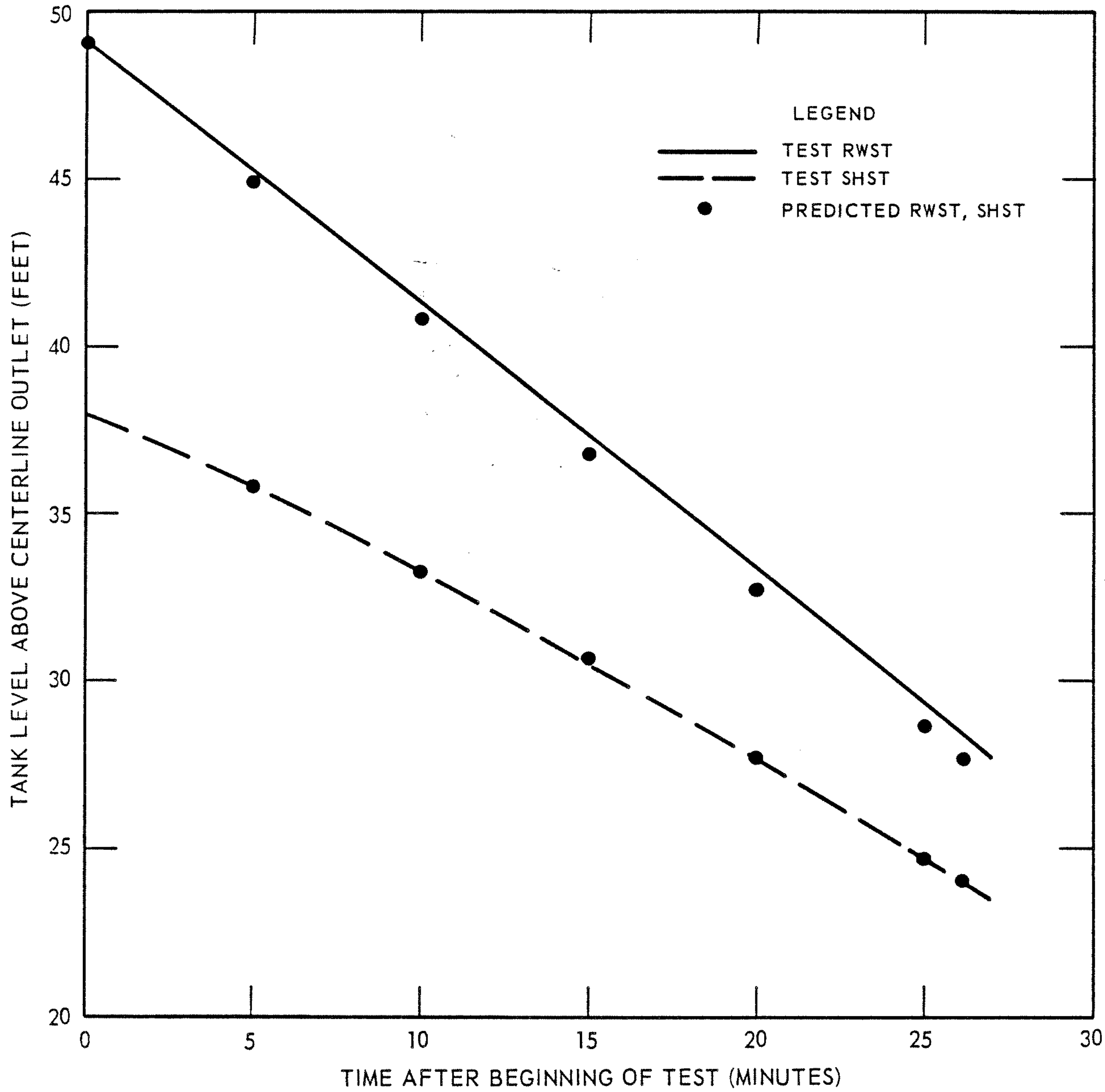
SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Spray Header pH During Recirculation - Maximum Sodium Hydroxide Initial Conditions - Normal Case with One Spray Pump Inoperable
Figure 6.2-51aa



Amendment 6
August 1990

SOUTH CAROLINA ELECTRIC & GAS CO. **VIRGIL C. SUMMER NUCLEAR STATION**
Spray Header pH During Recirculation - Maximum Sodium Hydroxide Initial Conditions - Normal Case with One RHR Pump Inoperable **Figure 6.2-51bb**

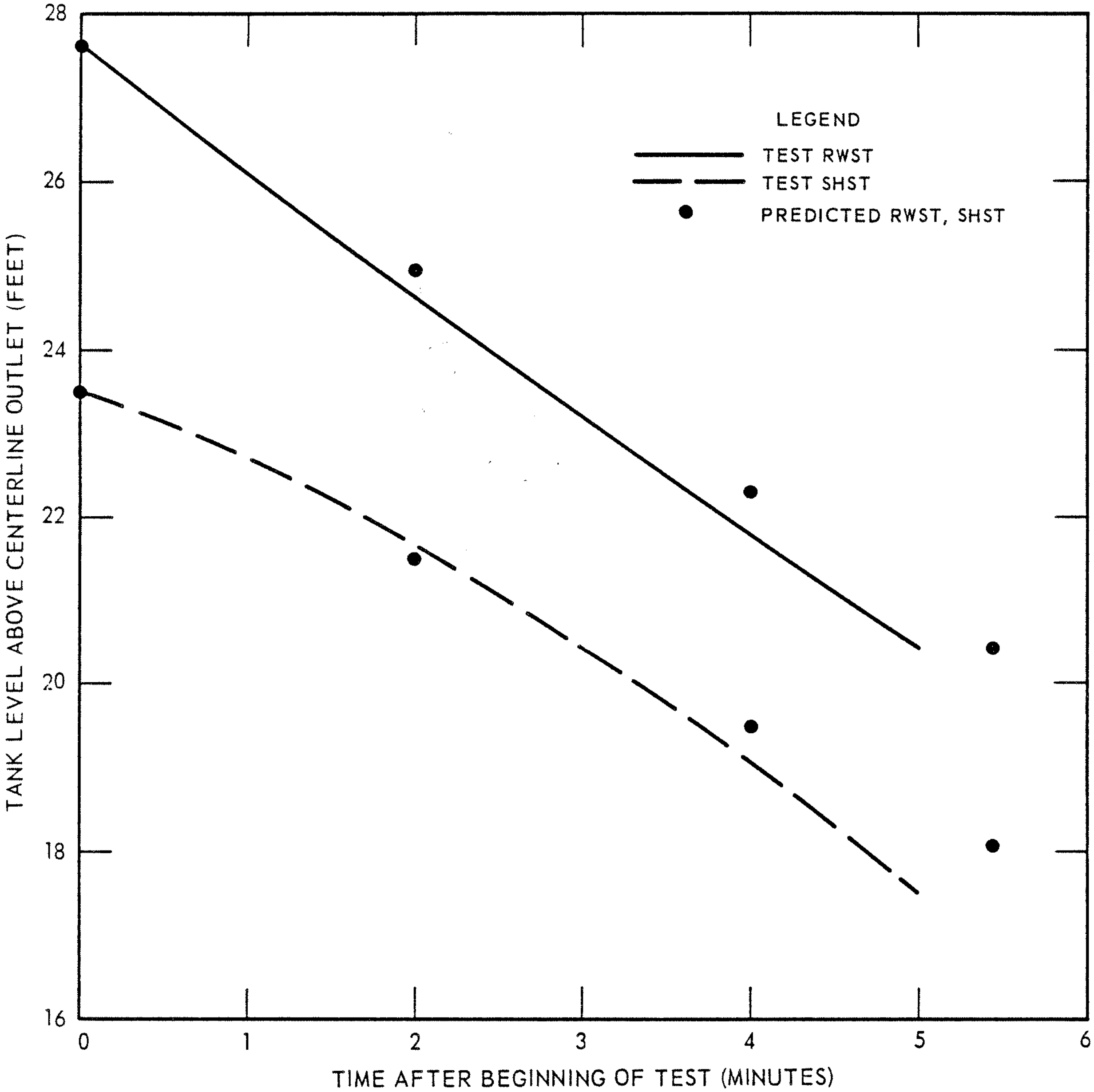


Amendment 0
August 1984

SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Analytical Prediction to Water Test
Case 1: Reactor Building Spray System
Design Case

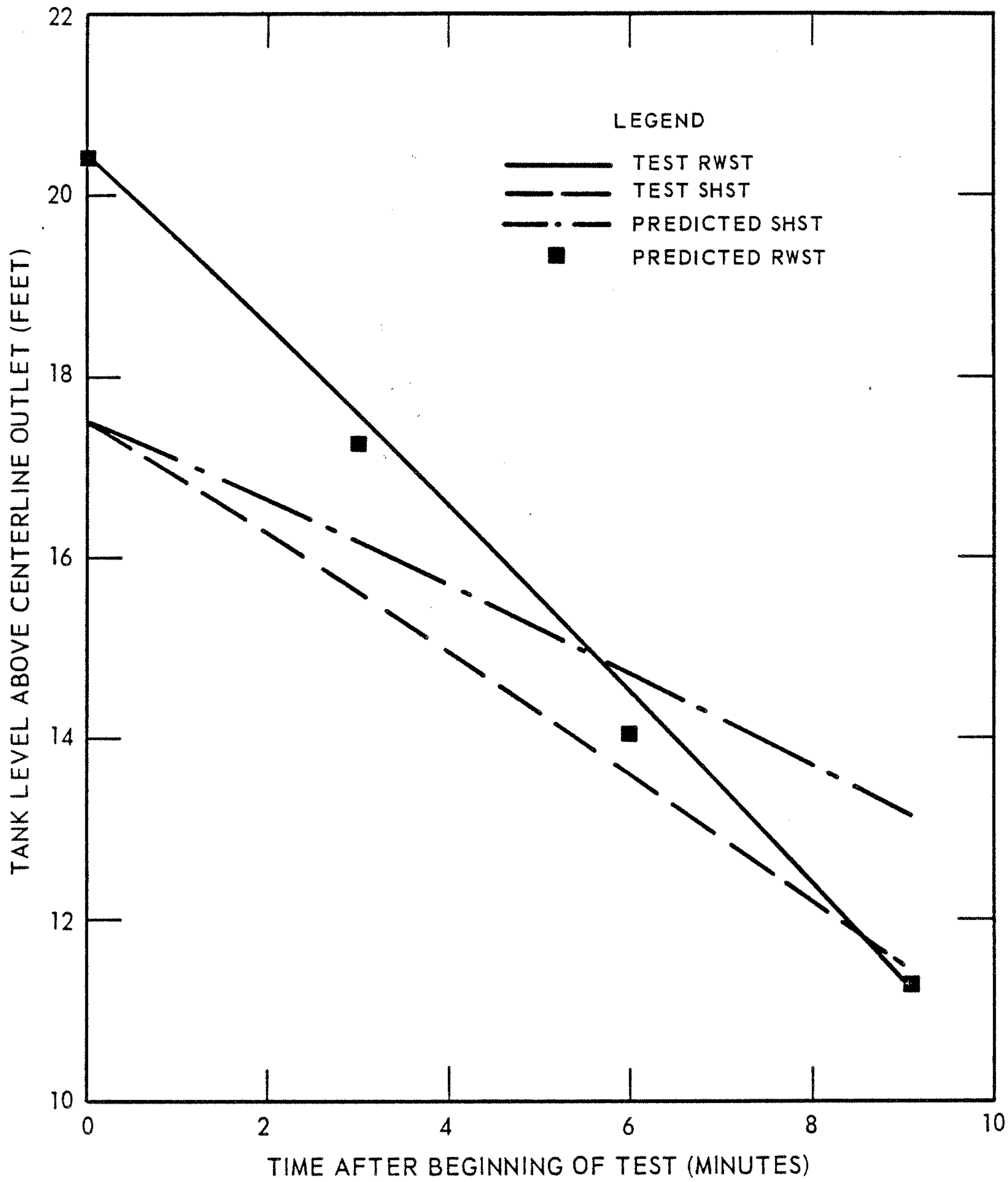
Figure 6.2-51cc



SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Analytical Prediction to Water Test
Case 2: Normal Case

Figure 6.2-51dd

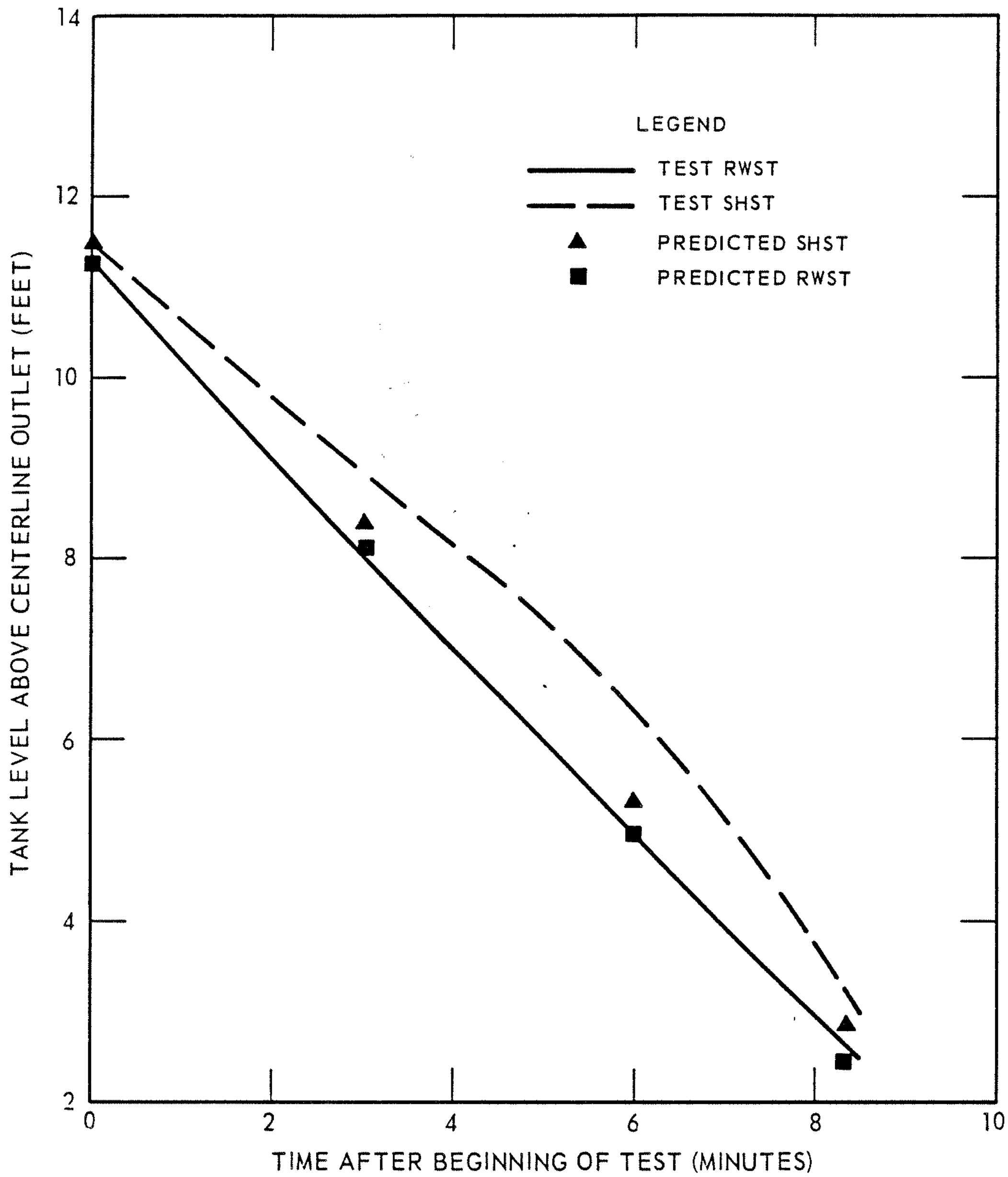


Amendment 0
August 1984

SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Analytical Prediction to Water Test Case 3: Normal Case with One Reactor Building Spray Pump Inoperable

Figure 6.2-51ee

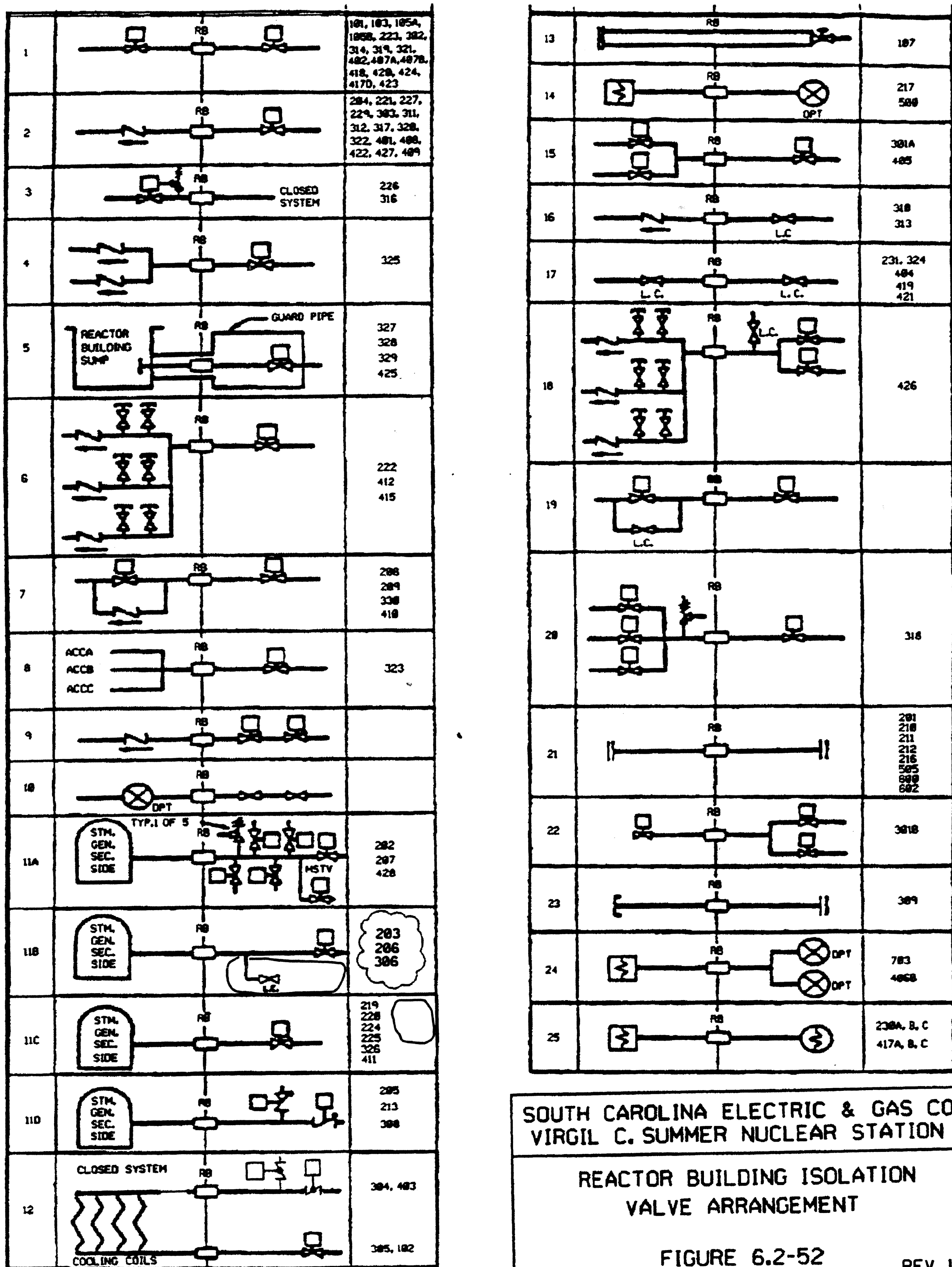


Amendment 0
August 1984

SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Analytical Prediction to Water Test Case 4: Normal Case with One RHR Pump Inoperable

Figure 6.2-51ff

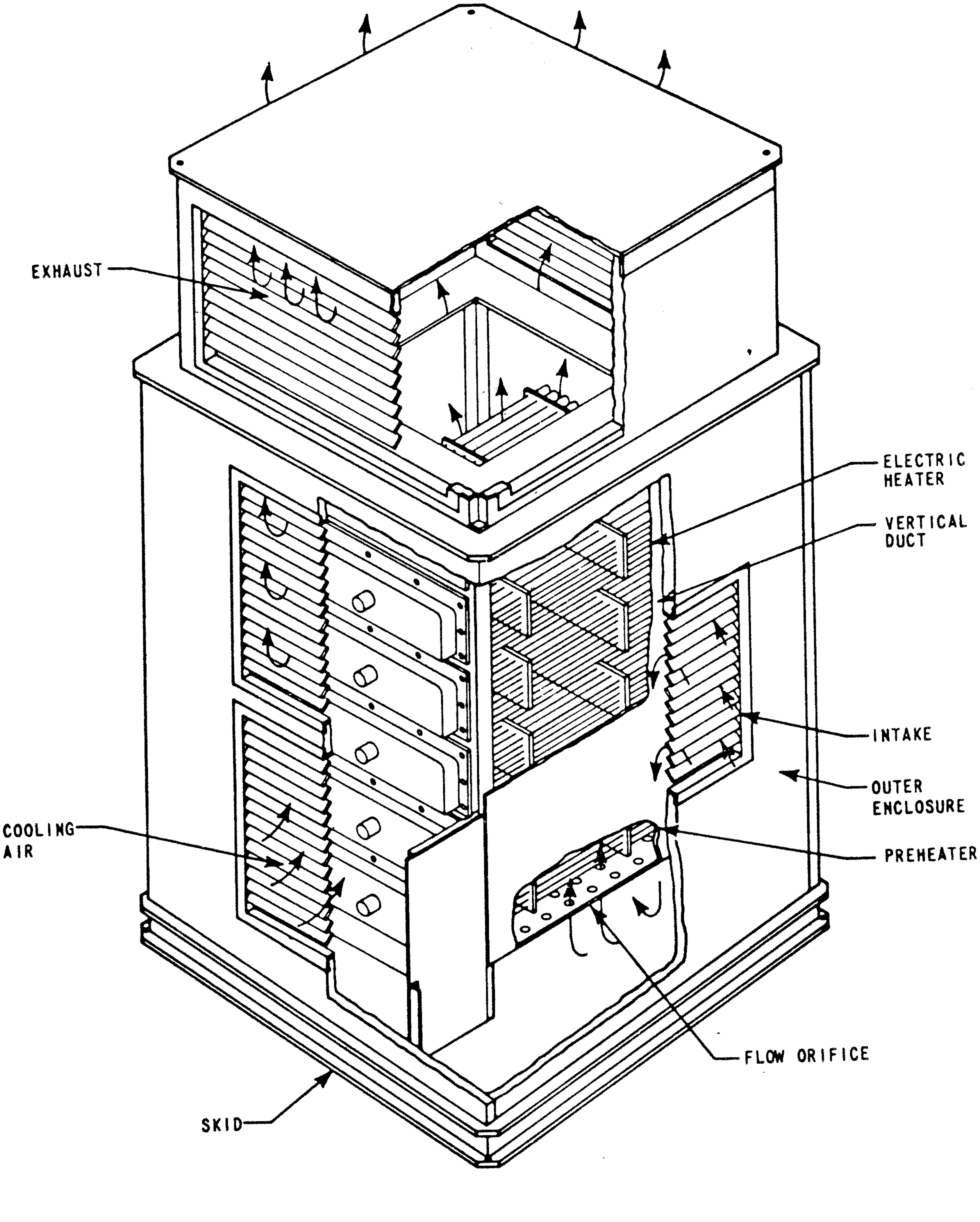


SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

REACTOR BUILDING ISOLATION VALVE ARRANGEMENT

FIGURE 6.2-52

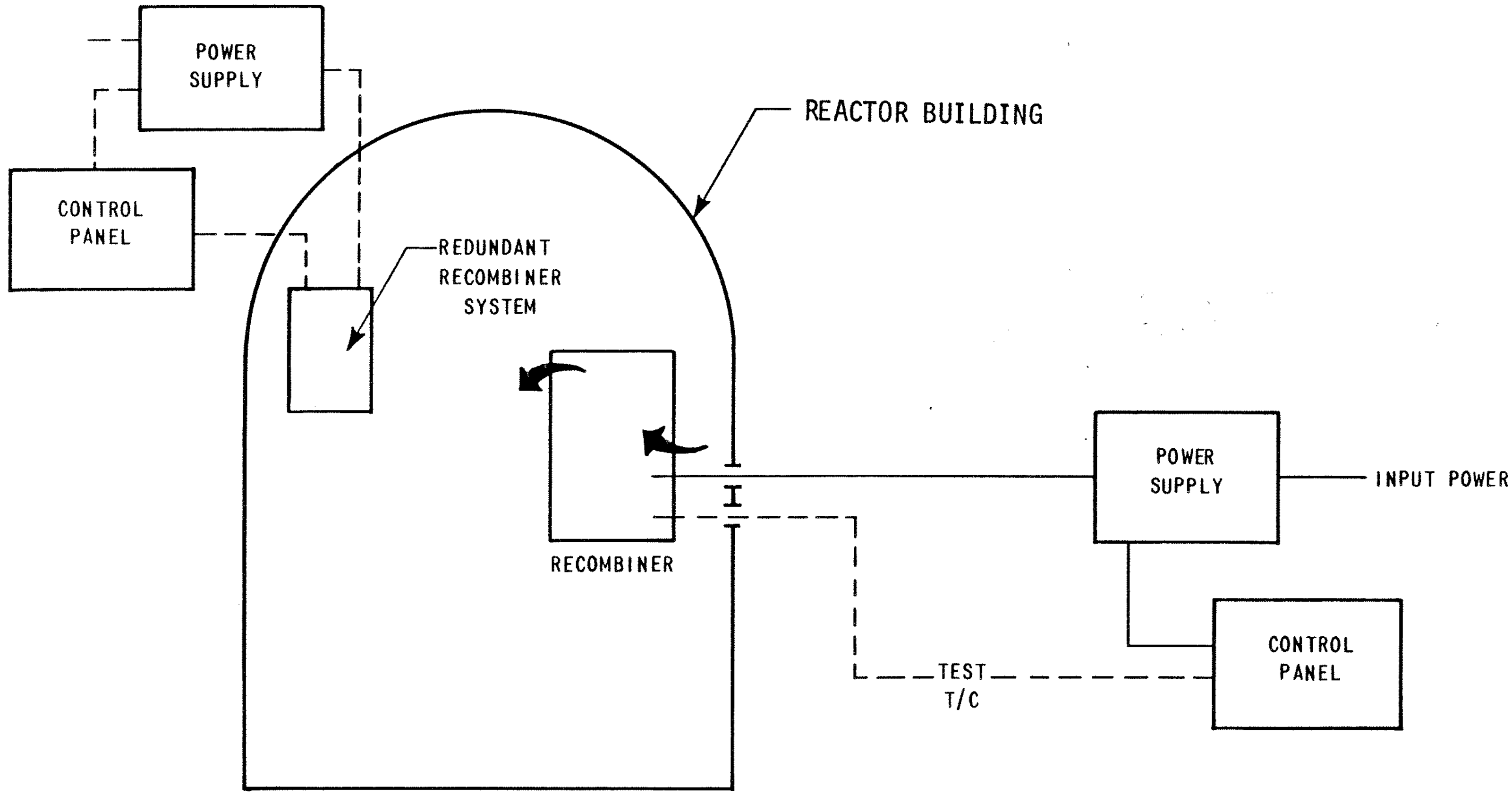
REV. 1



SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Electric Hydrogen Recombiner

Figure 6.2-53



BOTH RECOMBINERS ARE LOCATED
ON THE OPERATING FLOOR
ELEVATION 463'-0"

Amendment 0
August 1984

SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Schematic Electric Recombiner System

Figure 6.2-54

SYSTEM DATA					
#	CFM	PSIG	°F	BY	REMARKS
1	0.5	3	90	DJK	
2	100	3	90	DJK	
3	600	3	90	DJK	
4					

CONTAINMENT ISOLATION PHASE A (TRAIN A) — A (3 PLCS)

CONTAINMENT ISOLATION PHASE A (TRAIN B) — B (3 PLCS)

PURGE EXHAUST RADIATION RM-A4 — C (2 PLCS)

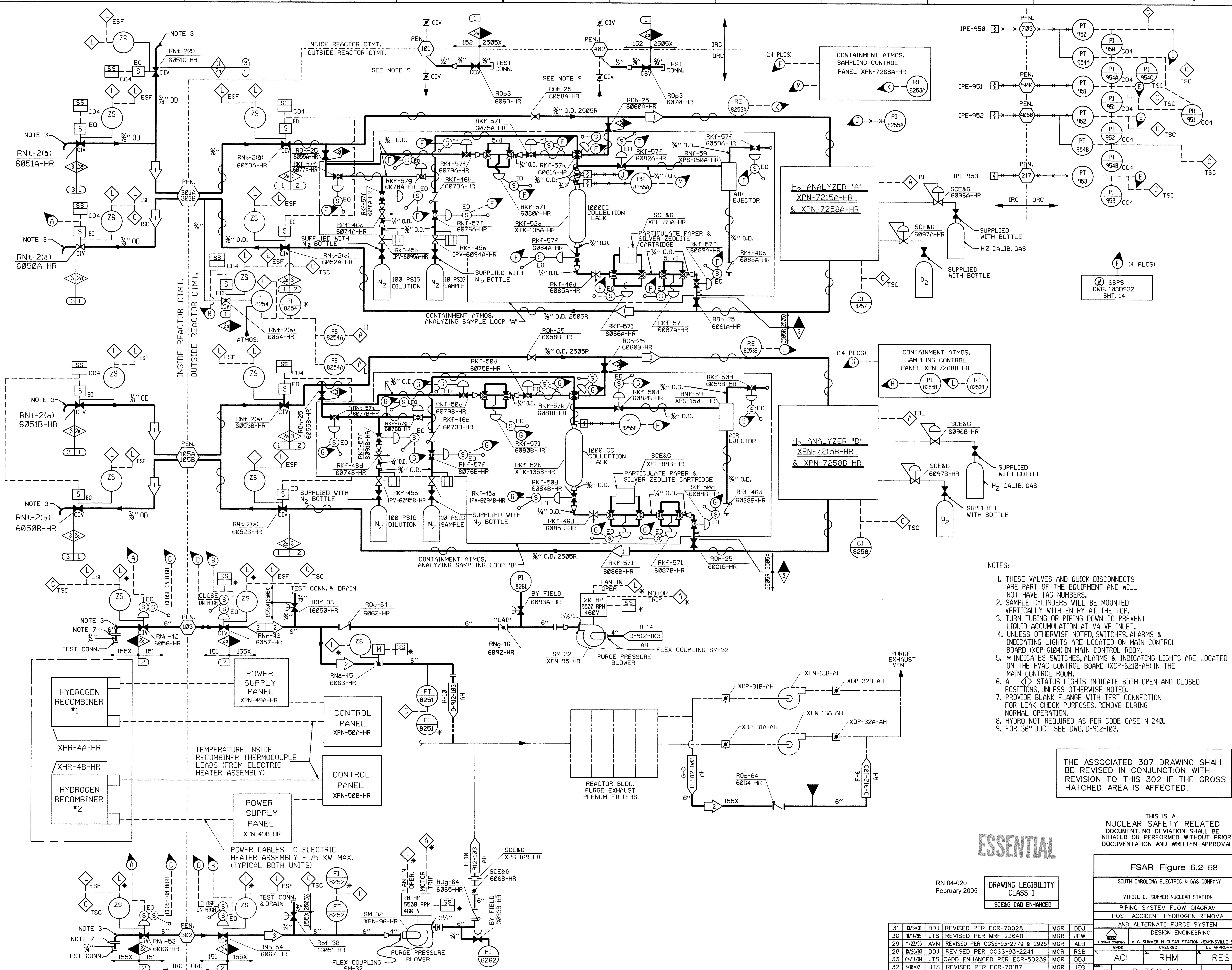
CONTAINMENT RADIATION RM-A2 — D (2 PLCS)

SAFETY CLASS VERIFICATION	
ORIGINATED BY	
REVIEWED BY	

HR	16050	16051
HR	6050	6097
SYS. SUFFIX	FIRST NO.	LAST NO.

VALVE NUMBERING

NEXT NO. USED P



NOTES:

1. THESE VALVES AND QUICK-DISCONNECTS ARE PART OF THE EQUIPMENT AND WILL NOT HAVE TAG NUMBERS.
2. SAMPLE CYLINDERS WILL BE MOUNTED VERTICALLY WITH ENTRY AT THE TOP.
3. TURN TUBING OR PIPING DOWN TO PREVENT LIQUID ACCUMULATION AT VALVE INLET.
4. UNLESS OTHERWISE NOTED, SWITCHES, ALARMS & INDICATING LIGHTS ARE LOCATED ON MAIN CONTROL BOARD (XCP-6104) IN MAIN CONTROL ROOM.
5. * INDICATES SWITCHES, ALARMS & INDICATING LIGHTS ARE LOCATED ON THE HVAC CONTROL BOARD (XCP-6210-AH) IN THE MAIN CONTROL ROOM.
6. ALL L STATUS LIGHTS INDICATE BOTH OPEN AND CLOSED POSITIONS, UNLESS OTHERWISE NOTED.
7. PROVIDE BLANK FLANGE WITH TEST CONNECTION FOR LEAK CHECK PURPOSES. REMOVE DURING NORMAL OPERATION.
8. HYDRO NOT REQUIRED AS PER CODE CASE N-240.
9. FOR 36" DUCT SEE DWG. D-912-103.

THE ASSOCIATED 307 DRAWING SHALL BE REVISED IN CONJUNCTION WITH REVISION TO THIS 302 IF THE CROSS HATCHED AREA IS AFFECTED.

HYDROTEST TEMP. 40°F MIN

#	NORMAL PSIG	NORMAL °F	UPSET PSIG	UPSET °F	UPSET DURATION	HYDRO	CHKD	BY	REMARKS
5									
4									
3	ATM	120	66	283	< 1%	NOTE 8		PLL	
2	ATM	120	66	190	< 1%	85		PLL	
1	ATM	120	66	283	< 1%	85		PLL	

DESIGN DATA

ESSENTIAL

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RN 04-020
February 2005

DRAWING LEGIBILITY CLASS 1
SCE&G CAD ENHANCED

NO.	DATE	BY	REVISION	CKD. BY	LE APPROVAL
31	10/19/01	DDJ	REVISED PER ECR-70028	MGR	DDJ
30	11/14/95	JTS	REVISED PER MRF-22640	MGR	JEW
29	11/23/93	AVN	REVISED PER CGSS-93-2779 & 2925	MGR	ALB
28	10/26/93	DDJ	REVISED PER CGSS-93-2241	MGR	RSB
33	04/14/04	JTS	CADD ENHANCED PER ECR-50239	MGR	DDJ
32	6/18/02	JTS	REVISED PER ECR-70187	MGR	JEG

FSAR Figure 6.2–58

SOUTH CAROLINA ELECTRIC & GAS COMPANY

VIRGIL C. SUMMER NUCLEAR STATION

PIPING SYSTEM FLOW DIAGRAM
POST ACCIDENT HYDROGEN REMOVAL
AND ALTERNATE PURGE SYSTEM

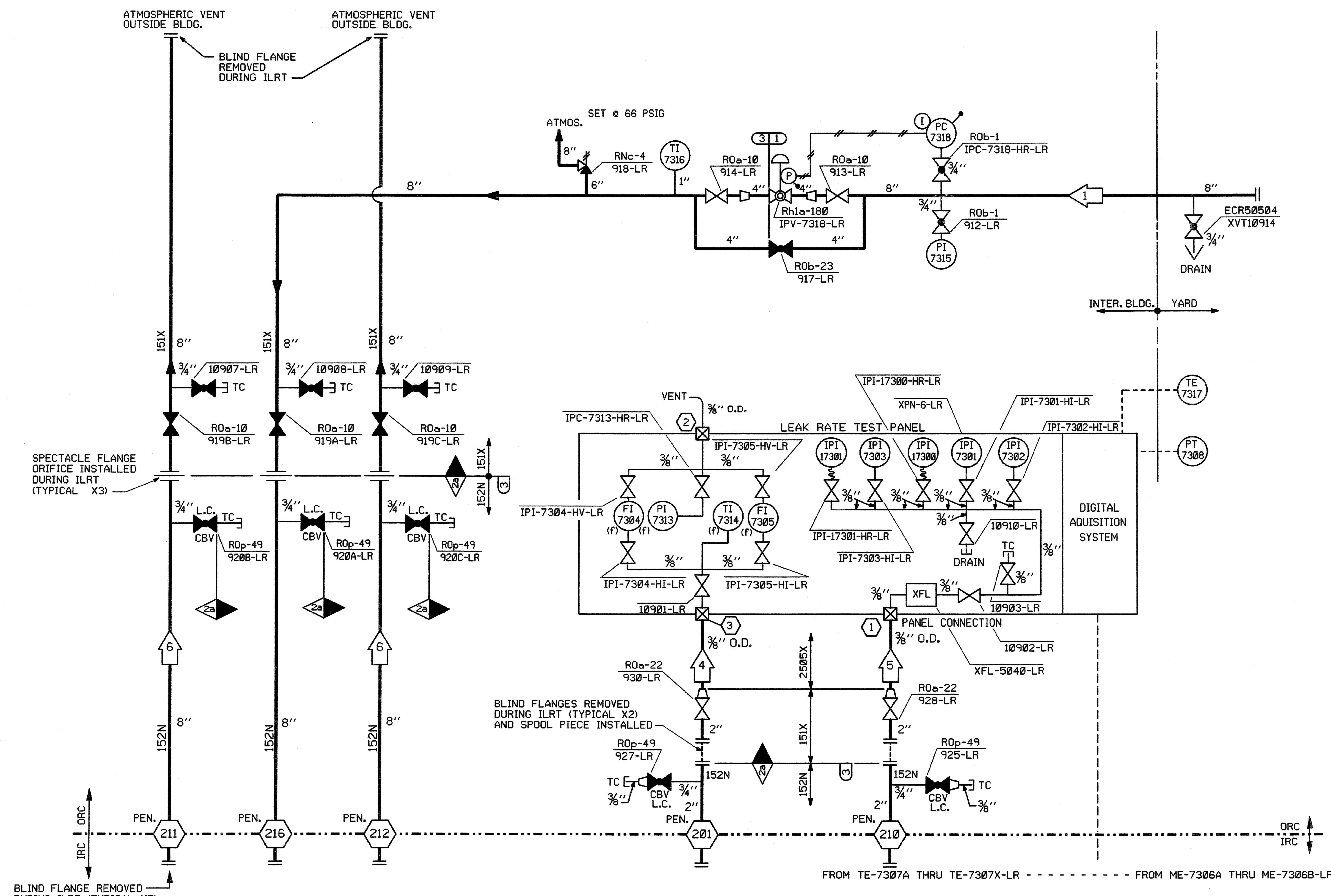
SCE&G A SCANA COMPANY — DESIGN ENGINEERING — V. C. SUMMER NUCLEAR STATION JENKINSVILLE S. C.

MADE	CHECKED	LE APPROVAL
1. ACI	2. RHM	3. RES

SCALE	DRAWING NUMBER	SHT. NUMBER	REV
NO SCALE	D-302-861		33

SYSTEM DATA					
NO.	SCFM	PSIG	°F	BY	REMARKS
1	25000	100	100	PAR	
4	6.5	57	100	PAR	
5	6.5	57	100	PAR	
6	37200	65.6	100	JGS	

LR	10901	10909
LR	900	939
SYSTEM SUFFIX	FIRST NO.	LAST NO.
VALVE NUMBERING		



MINIMUM PNEU. TEST TEMP IS 60°F									
3	66	100	66	100	< 1%	85	JGS		
1	110	100	125	100	< 1%	125	JGS		
D	PSIG	°F	PSIG	°F	DURATION	PNEU.	BY	CHK'D.	REMARKS
	NORMAL		UPSET						
DESIGN DATA									

NOTES:

1. ITEMS DESIGNATED WITH (f) ARE FURNISHED BY EQUIPMENT SUPPLIER.
2. ILRT - REACTOR CONTAINMENT INTEGRATED LEAK RATE TEST.

ESSENTIAL

THIS IS A NUCLEAR SAFETY RELATED DOCUMENT. NO DEVIATION SHALL BE INITIATED OR PERFORMED WITHOUT PRIOR DOCUMENTATION AND WRITTEN APPROVAL

RN 05-041
October 2005

DRAWING LEGIBILITY CLASS 1

SCE&G CAD ENHANCED

FSAR Figure 6.2-59

SOUTH CAROLINA ELECTRIC & GAS COMPANY

VIRGIL C. SUMMER NUCLEAR STATION

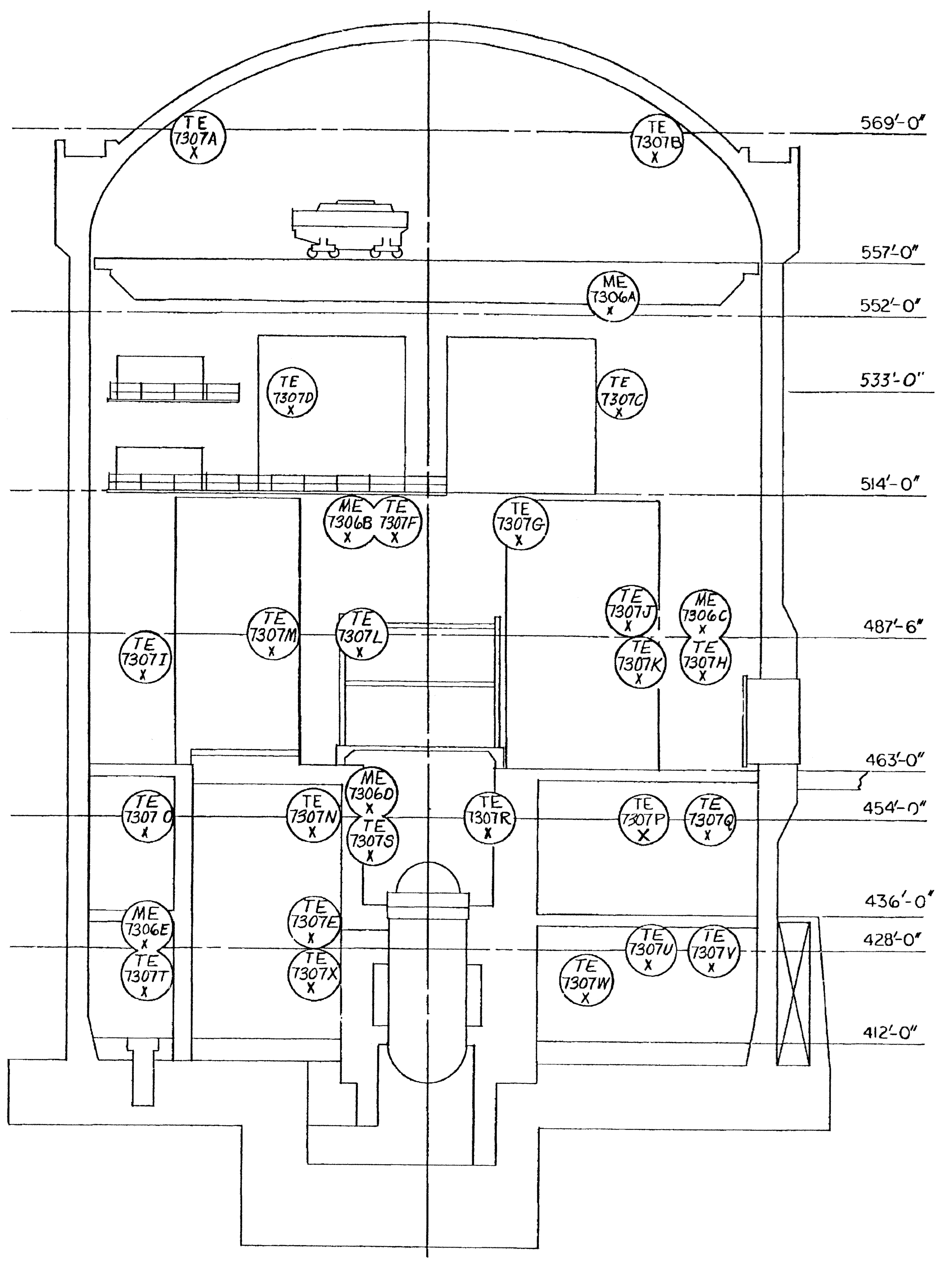
PIPING — SYSTEM FLOW DIAGRAM
REACTOR BUILDING LEAK RATE TESTING SYSTEM
DESIGN ENGINEERING
V. C. SUMMER NUCLEAR STATION JENKINSVILLE, S.C.

MADE	CHECKED	LE APPROVAL
A.V.N.	M.G.R.	R.S.B.

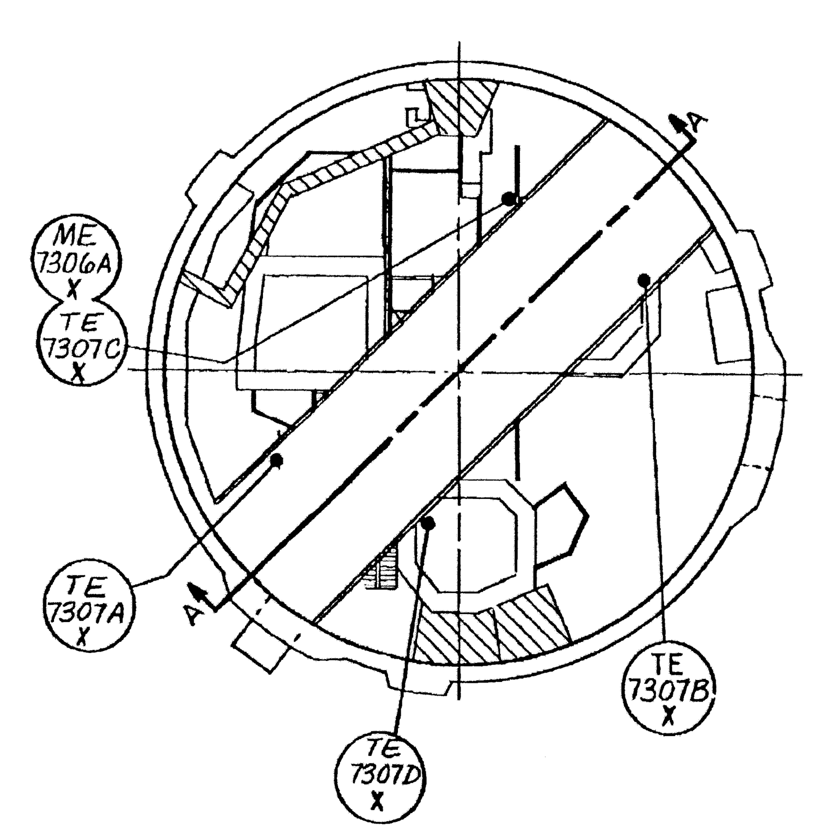
D-302-811 — SHT. NUMBER — REV. 11

NO.	DATE	BY	REVISION	CKD. BY	LE APPROVAL
11	4/4/05	JTS	CADD ENHANCED PER ECR-50239	RHM	DDJ
10	10/1/03	RHM	REVISED PER ECR-50504	RHM	MH
9	3/5/99	TGB	REVISED PER CGSS-99-0093	MGR	RR
8	11/15/95	JTS	REVISED PER MRF-22640	MGR	JEW

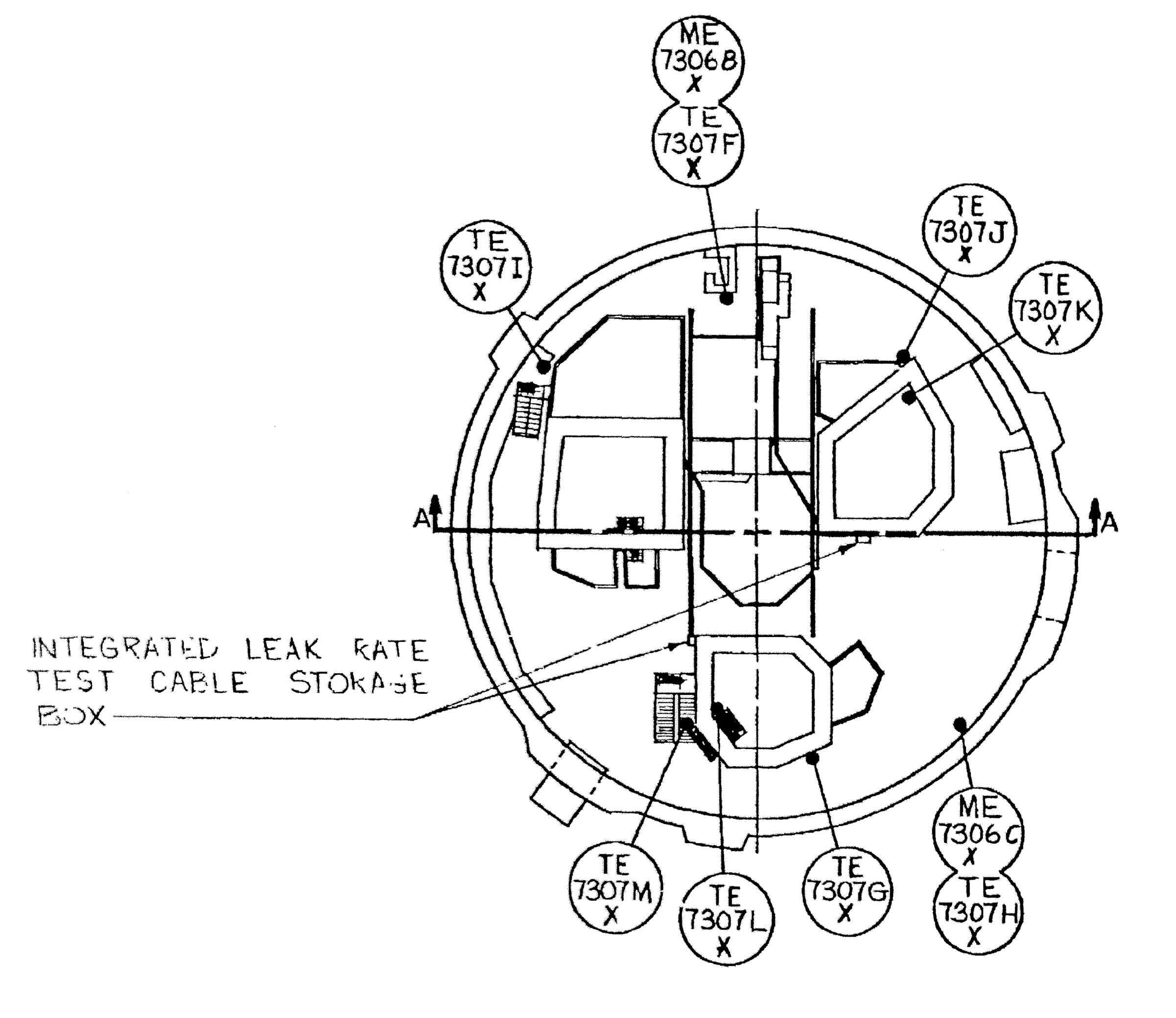
$$$$$DGNSPEC$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$$
$$USERNAME$$ $$$$$SYTIME$$$$$$



VIEW AA



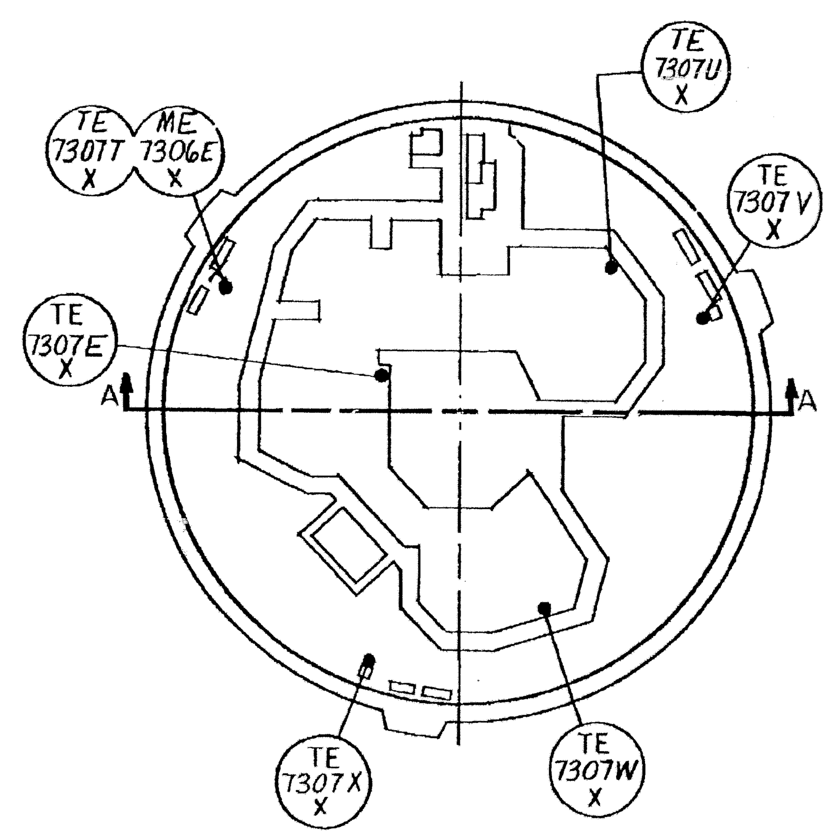
ABOVE FL. EL. 514'-0"



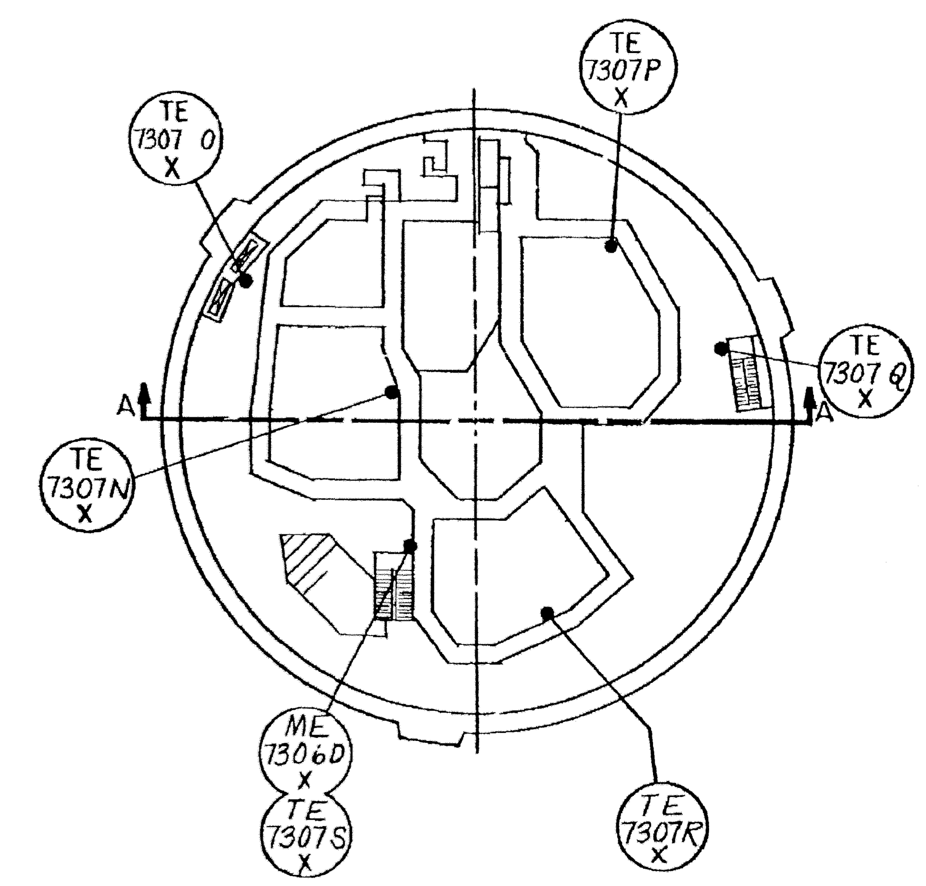
ABOVE FL. EL. 463'-0"

N

SENSOR LOCATION		DETAIL
INSTRUMENT TAG	ELEVATION	
TE-7307-A	569'-0"	
TE-7307-B	569'-0"	
ME-7306-A	552'-0"	
TE-7307-C	533'-0"	
TE-7307-D	533'-0"	
ME-7306-B	514'-0"	
TE-7307-F	514'-0"	
TE-7307-G	514'-0"	
ME-7306-C	487'-6"	
TE-7307-H	487'-6"	
TE-7307-I	487'-6"	
TE-7307-J	487'-6"	
TE-7307-K	487'-6"	
TE-7307-L	487'-6"	
TE-7307-M	487'-6"	
TE-7307-N	459'-0"	
ME-7306-D	459'-0"	
TE-7307-O	459'-0"	
TE-7307-P	459'-0"	
TE-7307-Q	459'-0"	
TE-7307-R	459'-0"	
TE-7307-S	459'-0"	
TE-7307-T	428'-0"	
ME-7306-E	428'-0"	
TE-7307-U	428'-0"	
TE-7307-V	428'-0"	
TE-7307-W	428'-0"	
TE-7307-X	428'-0"	
TE-7307-E	428'-0"	



ABOVE FL. EL. 412'-0"



ABOVE FL. EL. 436'-0"

REFERENCES:-

D-302-811	REACTOR BUILDING LEAK RATE TESTING SYSTEM
B-814-202	LEAK RATE TEST INSTRUMENT HOOKUP ISOMETRIC
B-809-451	TYPICAL TEMPERATURE ELEMENT MOUNTING BRACKET
E-803-001	GENERAL NOTES, REFERENCES AND SYMBOLS
E-811-003	INSTRUMENT LOCATION LAYOUT
E-811-004	INSTRUMENT LOCATION LAYOUT
E-811-005	INSTRUMENT LOCATION LAYOUT
E-215-101	ELECTRICAL CONDUIT LAYOUT
E-215-102	ELECTRICAL CONDUIT LAYOUT
E-215-103	ELECTRICAL CONDUIT LAYOUT
E-215-105	ELECTRICAL CONDUIT LAYOUT
B-809-476	R.B. HUMIDITY ELEMENT "A" ME-7306A

FSAR Figure 6.2-60

SOUTH CAROLINA ELECTRIC & GAS COMPANY

VIRGIL C. SUMMER NUCLEAR STATION

INSTRUMENT LOCATION LAYOUT
INTEGRATED LEAK RATE TEST
SENSOR LOCATIONS

SCE&G A SCANA COMPANY — DESIGN ENGINEERING — V. C. SUMMER NUCLEAR STATION JENKINSVILLE, S. C.

Amendment 02-01
May 2002

NO.	DATE	BY	REVISION	CKD. BY	LE APPROVAL
3	10/01/01	DDJ	REVISED PER ECR-70028	MGR	DDJ

MADE	CHECKED	LE APPROVAL
1. DDJ	2. MGR	3. DDJ

SCALE	DRAWING NUMBER	SHT. NUMBER	REV
NO SCALE	C-811-002		3

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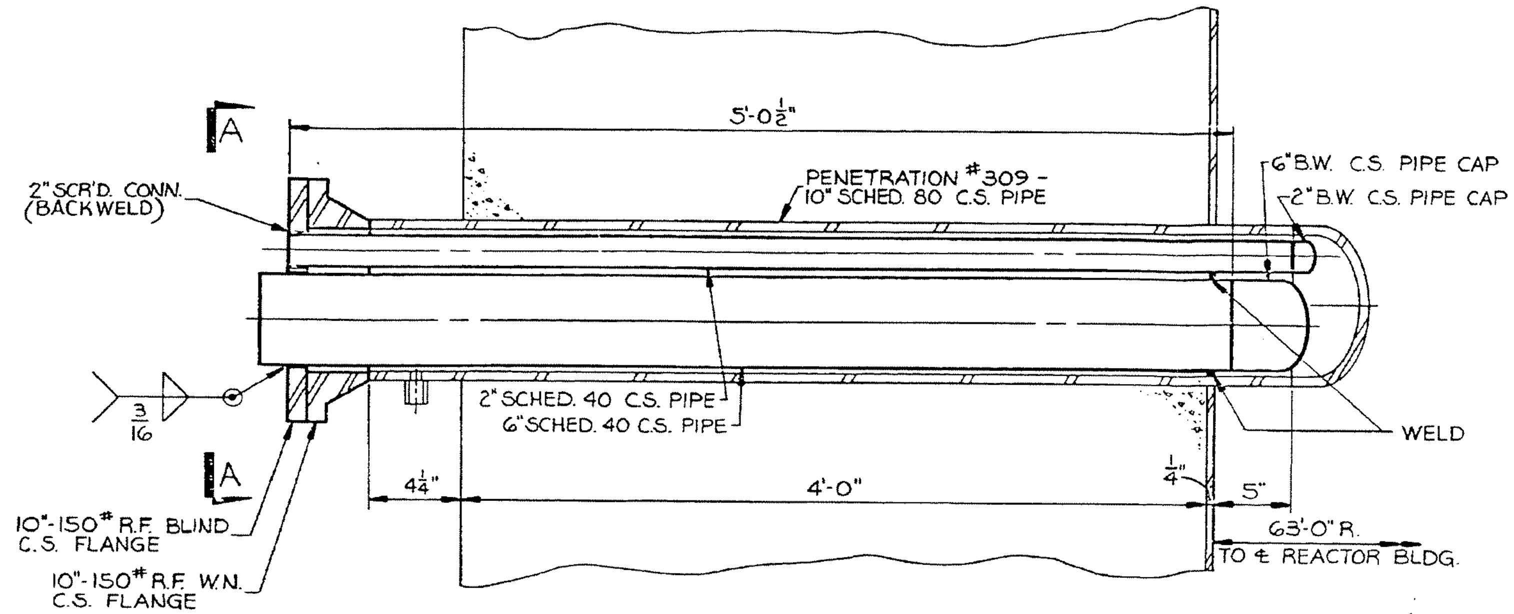
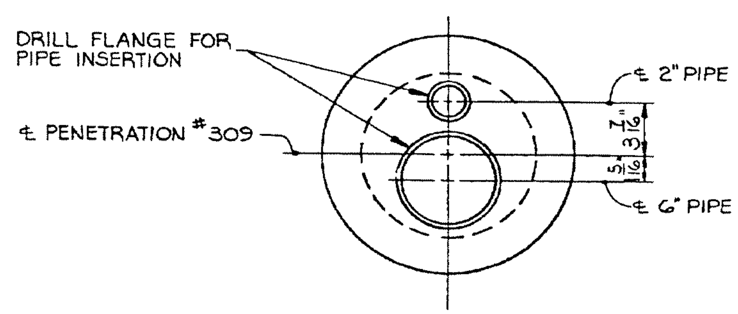


PENETRATION #309



SECTION "A-A"

NOTES:

1. FOR PIPE MATERIAL SHOWN ON THIS DRAWING, REFER TO PIPING SPECIFICATION SP-545-044461-000, LINE SPEC. 152.
2. PIPING TO BE ANS SAFETY CLASSIFICATION 2A. INSTALLATION TO BE IN ACCORDANCE WITH SP-220-044461-000 AND ASME CODE SECTION III, SUBSECTION NE.

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AMENDMENT 02-01
MAY 2002

FSAR Figure 6.2-60a

SOUTH CAROLINA ELECTRIC & GAS COMPANY

VIRGIL C. SUMMER NUCLEAR STATION

I & C — DETAIL

PENTRATION #309 DETAILS

DESIGN ENGINEERING

V. C. SUMMER NUCLEAR STATION JENKINSVILLE, S. C.

MADE	CHECKED	LE APPROVAL
JTS	MGR	RAT

NO SCALE — B-809-466 — DRAWING NUMBER — SHT. NUMBER — REV 3

NO.	DATE	BY	REVISION	CKD. BY	LE APPROVAL
3	02/18/00	JTS	REVISED PER CGSS-00-0042	MGR	RAT

RN 00-015

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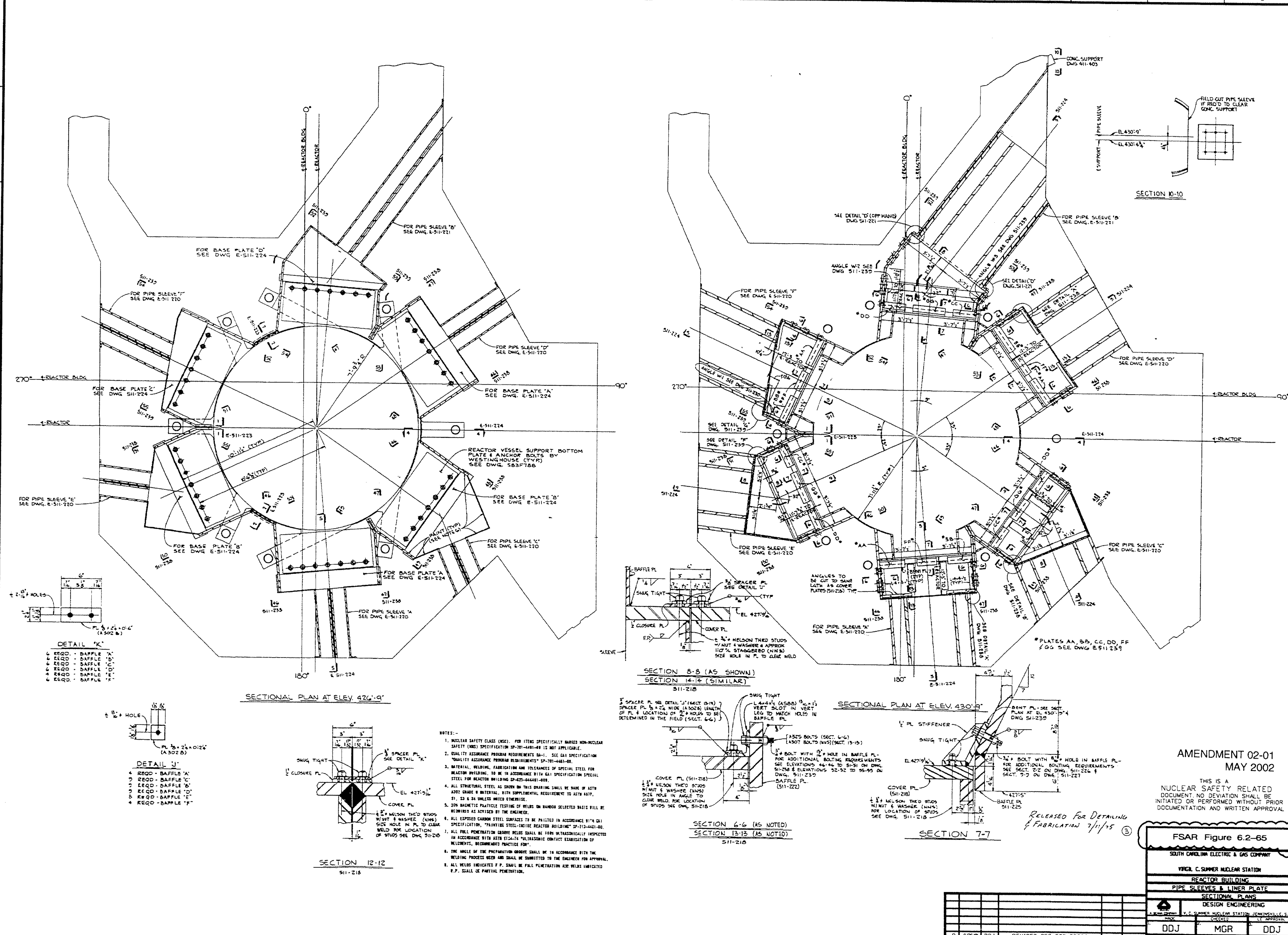
Figure 6.2-61
Deleted per RN 01-073 02-01

Figure 6.2-62 Through 6.2-64
Deleted per RN 06-040 RN 06-040



NOTES:-

1. NUCLEAR SAFETY CLASS (NSC). FOR ITEMS SPECIFICALLY MARKED NON-NUCLEAR SAFETY (NNS) SPECIFICATION SP-701-4461-00 IS NOT APPLICABLE.
2. QUALITY ASSURANCE PROGRAM REQUIREMENTS QA-1. SEE GAI SPECIFICATION "QUALITY ASSURANCE PROGRAM REQUIREMENTS" SP-701-4461-00.
3. MATERIAL, WELDING, FABRICATION AND TOLERANCES OF SPECIAL STEEL FOR REACTOR BUILDING, TO BE IN ACCORDANCE WITH GAI SPECIFICATION SPECIAL STEEL FOR REACTOR BUILDING SP-625-044461-000.
4. ALL STRUCTURAL STEEL AS SHOWN ON THIS DRAWING SHALL BE MADE OF ASTM A302 GRADE B MATERIAL, WITH SUPPLEMENTAL REQUIREMENT TO ASTM A87P, S1, S3 & S4 UNLESS NOTED OTHERWISE.
5. 20% MAGNETIC PARTICLE TESTING OF WELDS ON RANDOM SELECTED BASIS WILL BE REQUIRED AS ADVISED BY THE ENGINEER.
6. ALL EXPOSED CARBON STEEL SURFACES TO BE PAINTED IN ACCORDANCE WITH GAI SPECIFICATION, "PAINTING STEEL-INSIDE REACTOR BUILDING" SP-713-4461-00.
7. ALL FULL PENETRATION GROOVE WELDS SHALL BE 100% ULTRASONICALLY INSPECTED IN ACCORDANCE WITH ASTM E164-74 "ULTRASONIC CONTACT EXAMINATION OF WELDMENTS, RECOMMENDED PRACTICE FOR".
8. THE ANGLE OF THE PREPARATION GROOVE SHALL BE IN ACCORDANCE WITH THE WELDING PROCESS USED AND SHALL BE SUBMITTED TO THE ENGINEER FOR APPROVAL.
9. ALL WELDS INDICATED F.P. SHALL BE FULL PENETRATION AND WELDS INDICATED P.P. SHALL BE PARTIAL PENETRATION.

NO.	DATE	BY	REVISION	CKD. BY	LE APPROVAL
9	9/26/01	DDJ	REVISED PER ECR-70028	MGR	DDJ

AMENDMENT 02-01
MAY 2002

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FSAR Figure 6.2–65

SOUTH CAROLINA ELECTRIC & GAS COMPANY

VIRGIL C. SUMMER NUCLEAR STATION

REACTOR BUILDING
PIPE SLEEVES & LINER PLATE
SECTIONAL PLANS

DESIGN ENGINEERING
V. C. SUMMER NUCLEAR STATION JENKINSVILLE, S. C.

MADE	CHECKED	LE APPROVAL
DDJ	MGR	DDJ

SCALE: NO SCALE — DRAWING NUMBER: E-511-219 — REV: 9

RN 01-095

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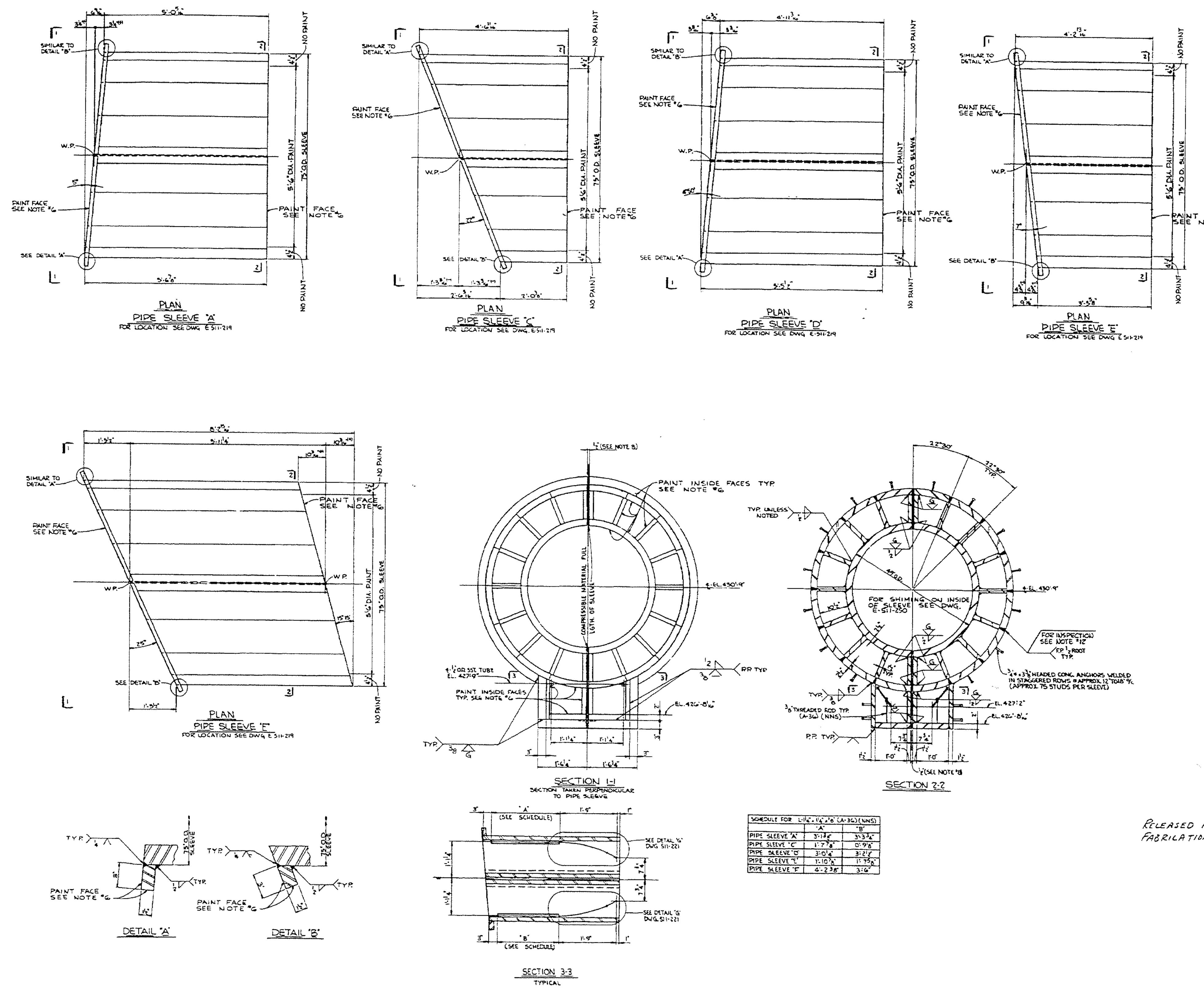

NOTES:

1. NUCLEAR SAFETY CLASS (NSC) FOR ITEMS SPECIFICALLY MARKED NON-NUCLEAR SAFETY (NNS), SPECIFICATION SP-701-4481-00 IS NOT APPLICABLE.
2. QUALITY ASSURANCE PROGRAM REQUIREMENTS QA-1 SEE GAI SPECIFICATION "QUALITY ASSURANCE PROGRAM REQUIREMENTS" SP-701-4481-00
3. MATERIAL, WELDING, FABRICATION AND TOLERANCES OF SPECIAL STEEL FOR REACTOR BUILDING, TO BE IN ACCORDANCE WITH GAI SPECIFICATION SPECIAL STEEL FOR REACTOR BUILDING SPECIFICATION SP-623-444481-000
4. ALL STRUCTURAL STEEL AS SHOWN ON THIS DRAWING SHALL BE MADE OF ASTM A302 GRADE B MATERIAL, WITH SUPPLEMENTAL REQUIREMENT TO ASTM A578, S1, S3, AND S4, UNLESS NOTED OTHERWISE.
5. 20% MAGNETIC PARTICLE TESTING OF WELDS ON RANDOM SELECTED BASIS WILL BE REQUIRED AS ADVISED BY THE ENGINEER.
6. ALL EXPOSED CARBON STEEL SURFACES TO BE PAINTED IN ACCORDANCE WITH GAI SPECIFICATION, "PAINTING STEEL-INSIDE REACTOR BUILDING" SP-713-4481-00.
7. TOLERANCES ON THE PIPE SLEEVES AS A COMPLETED ASSEMBLY WILL BE AS FOLLOWS:
   A. INNER RADIUS -1/16 +3/16
   B. OUTER RADIUS ±1/4
   C. WARPING ±1/8
   D. CENTER LINE STRAIGHTNESS ±1/8
8. THE 1/2 INCH CLEARANCE AS NOTED ON SECTION 2-2 IS A REQUIRED FIELD GAP. A 1/8 INCH MAXIMUM GAP IS ALLOWED DURING SHOP ASSEMBLY AND TOLERANCE VERIFICATION.
9. UNLESS INDICATED OTHERWISE ON THE DRAWING, ALL FULL PENETRATION GROOVE WELDS SHALL BE 100% ULTRASONICALLY INSPECTED IN ACCORDANCE WITH ASTM E164-74 "ULTRASONIC CONTACT EXAMINATION OF WELDMENTS, RECOMMENDED PRACTICE FOR".
10. THE ANGLE OF THE PREPARATION GROOVE SHALL BE IN ACCORDANCE WITH THE WELDING PROCESS USED AND SHALL BE SUBMITTED TO THE ENGINEER FOR APPROVAL
11. ALL WELDS INDICATED F.P. SHALL BE FULL PENETRATION AND WELDS INDICATED P.P. SHALL BE PARTIAL PENETRATION
12. VOLUMETRIC MAGNETIC PARTICLE INSPECTION OF FIRST TWO ROOT WELD PASSES, WITH ULTRASONIC INSPECTION OF REMAINING PASSES

SCHEDULE FOR L-1¼"×1¼"×⅛" (A-36)(NNS)		
	"A"	"B"
PIPE SLEEVE "A"	3'-1⅜"	3'-3¾"
PIPE SLEEVE "C"	1'-7⅞"	0'-9⅛"
PIPE SLEEVE "D"	3'-0¼"	3'-2½"
PIPE SLEEVE "E"	1'-10⅞"	1'-7⅝"
PIPE SLEEVE "F"	4'-2⅜"	3'-6"

RELEASED FOR DETAILING & FABRICATION 7/3/75 ②
SP625 Rev. 6-75

AMENDMENT 02-01
MAY 2002

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FSAR Figure 6.2-66

SOUTH CAROLINA ELECTRIC & GAS COMPANY
VIRGIL C. SUMMER NUCLEAR STATION
REACTOR BUILDING
PIPE SLEEVES
SECTIONS AND DETAILS
DESIGN ENGINEERING
A SCANA COMPANY
V. C. SUMMER NUCLEAR STATION JENKINSVILLE, S.C.

MADE	CHECKED	LE APPROVAL
DDJ	MGR	DDJ

NO.	DATE	BY	REVISION	CKD. BY	LE APPROVAL
9	9/26/01	DDJ	REVISED PER ECR-70028	MGR	DDJ

DRAWING NUMBER: E-511-220 REV: 9

RN 01-095

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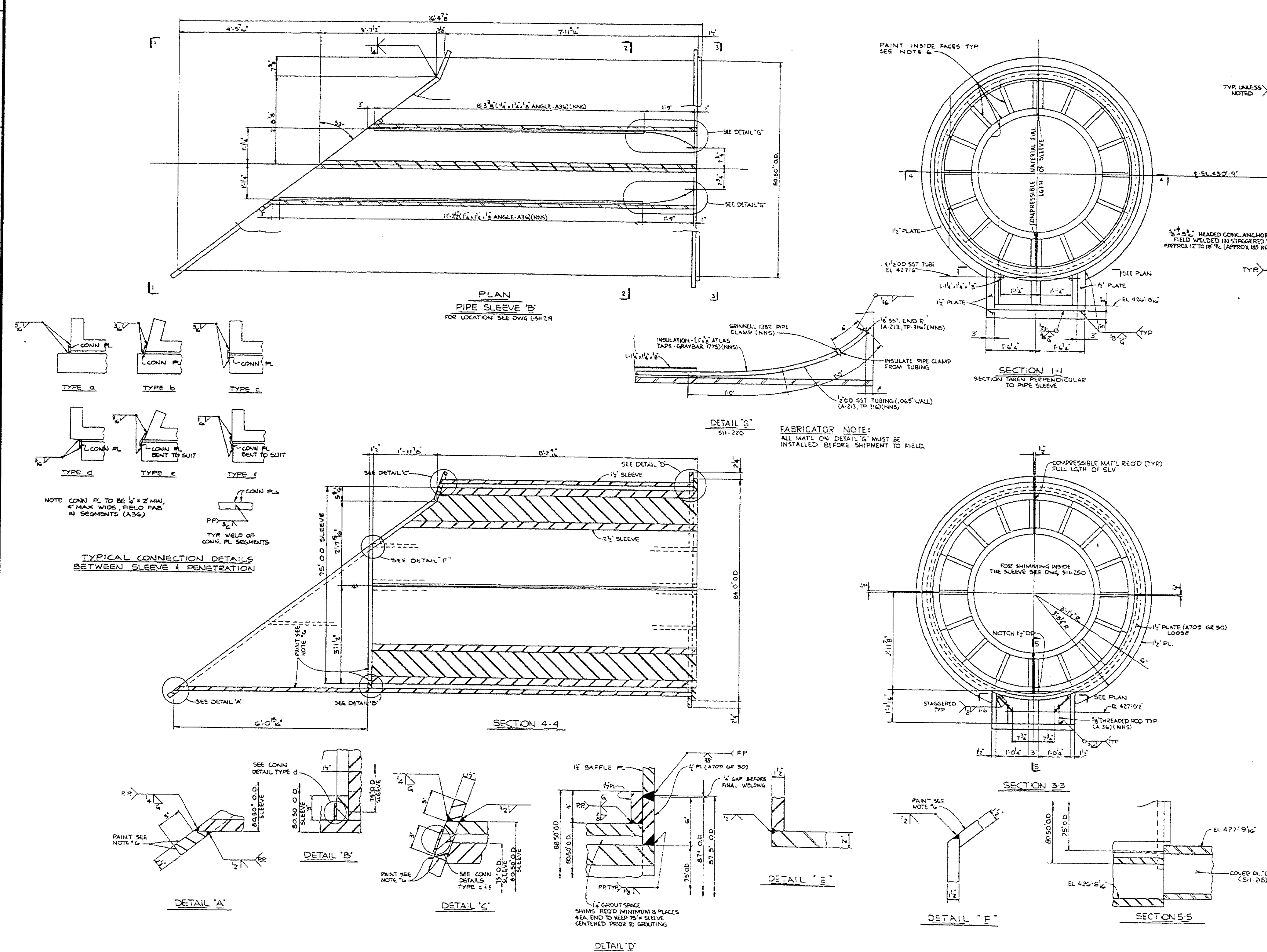

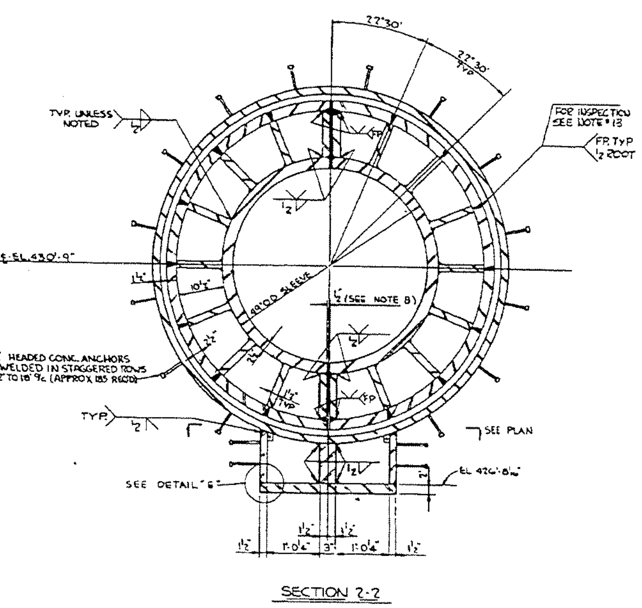

NOTES

1. NUCLEAR SAFETY CLASS (NSC) FOR ITEMS SPECIFICALLY MARKED NON-NUCLEAR SAFETY (NNS), SPECIFICATION SP-701-4461-00 IS NOT APPLICABLE
2. QUALITY ASSURANCE PROGRAM REQUIREMENTS QA-1 SEE GAI SPECIFICATION "QUALITY ASSURANCE PROGRAM REQUIREMENTS" SP-701-4461-00.
3. MATERIAL, WELDING, FABRICATION AND TOLERANCES OF SPECIAL STEEL FOR REACTOR BUILDING, TO BE IN ACCORDANCE WITH GAI SPECIFICATION SPECIAL STEEL FOR REACTOR BUILDING SPECIFICATION SP-023-044461-000.
4. ALL STRUCTURAL STEEL AS SHOWN ON THIS DRAWING SHALL BE MADE OF ASTM A302 GRADE B MATERIAL, WITH SUPPLEMENTAL REQUIREMENT TO ASTM A378, S1, S3, AND S4, UNLESS NOTED OTHERWISE
5. 20% MAGNETIC PARTICLE TESTING OF WELDS ON RANDOM SELECTED BASIS WILL BE REQUIRED AS ADVISED BY THE ENGINEER
6. ALL EXPOSED CARBON STEEL SURFACES TO BE PAINTED IN ACCORDANCE WITH GAI SPECIFICATION, "PAINTING STEEL-INSIDE REACTOR BUILDING" SP-713-4461-00
7. TOLERANCES ON THE PIPE SLEEVES AS A COMPLETE ASSEMBLY WILL BE AS FOLLOWS
   A. INNER RADIUS -1/16 +3/16
   B. OUTER RADIUS ±1/4
   C. WARPING ±1/8
   D. CENTER LINE STRAIGHTNESS ±1/8
8. THE 1/2 INCH CLEARANCE AS NOTED ON SECTION 2-2 IS A REQUIRED FIELD GAP. A 1/2 INCH MAXIMUM GAP IS REQUIRED DURING SHOP ASSEMBLY AND TOLERANCE VERIFICATION
9. CENTER LINE STRAIGHTNESS TOLERANCE ±1/8 INCH FOR THE FULL OUTSIDE PIPE
10. UNLESS INDICATED OTHERWISE ON THE DRAWING, ALL FULL PENETRATION GROOVE WELDS SHALL BE 100% ULTRASONICALLY INSPECTED IN ACCORDANCE WITH ASTM E164-74 "ULTRASONIC CONTACT EXAMINATION OF WELDMENTS, RECOMMENDED PRACTICE FOR"
11. THE ANGLE OF THE PREPARATION GROOVE SHALL BE IN ACCORDANCE WITH THE WELDING PROCESS USED AND SHALL BE SUBMITTED TO THE ENGINEER FOR APPROVAL
12. ALL WELDS INDICATED F.P. SHALL BE FULL PENETRATION AND WELDS INDICATED P.P. SHALL BE PARTIAL PENETRATION.
13. VOLUMETRIC MAGNETIC PARTICLE INSPECTION OF FIRST TWO ROOT WELD PASSES, WITH ULTRASONIC INSPECTION OF REMAINING PASSES.

RELEASED FOR DETAILING & FABRICATION 7/3/75

SP625

AMENDMENT 02-01
MAY 2002

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NO.	DATE	BY	REVISION	CKD. BY	LE APPROVAL
15	9/26/01	DDJ	REVISED PER ECR-70028	MGR	DDJ

FSAR Figure 6.2-67

SOUTH CAROLINA ELECTRIC & GAS COMPANY
VIRGIL C. SUMMER NUCLEAR STATION
REACTOR BUILDING
PIPE SLEEVES
SECTIONS AND DETAILS
DESIGN ENGINEERING
V. C. SUMMER NUCLEAR STATION JENKINSVILLE, S. C.

MADE	CHECKED	LE APPROVAL
DDJ	MGR	DDJ

NO SCALE — DRAWING NUMBER E-511-221 — REV 15

RN 01-095

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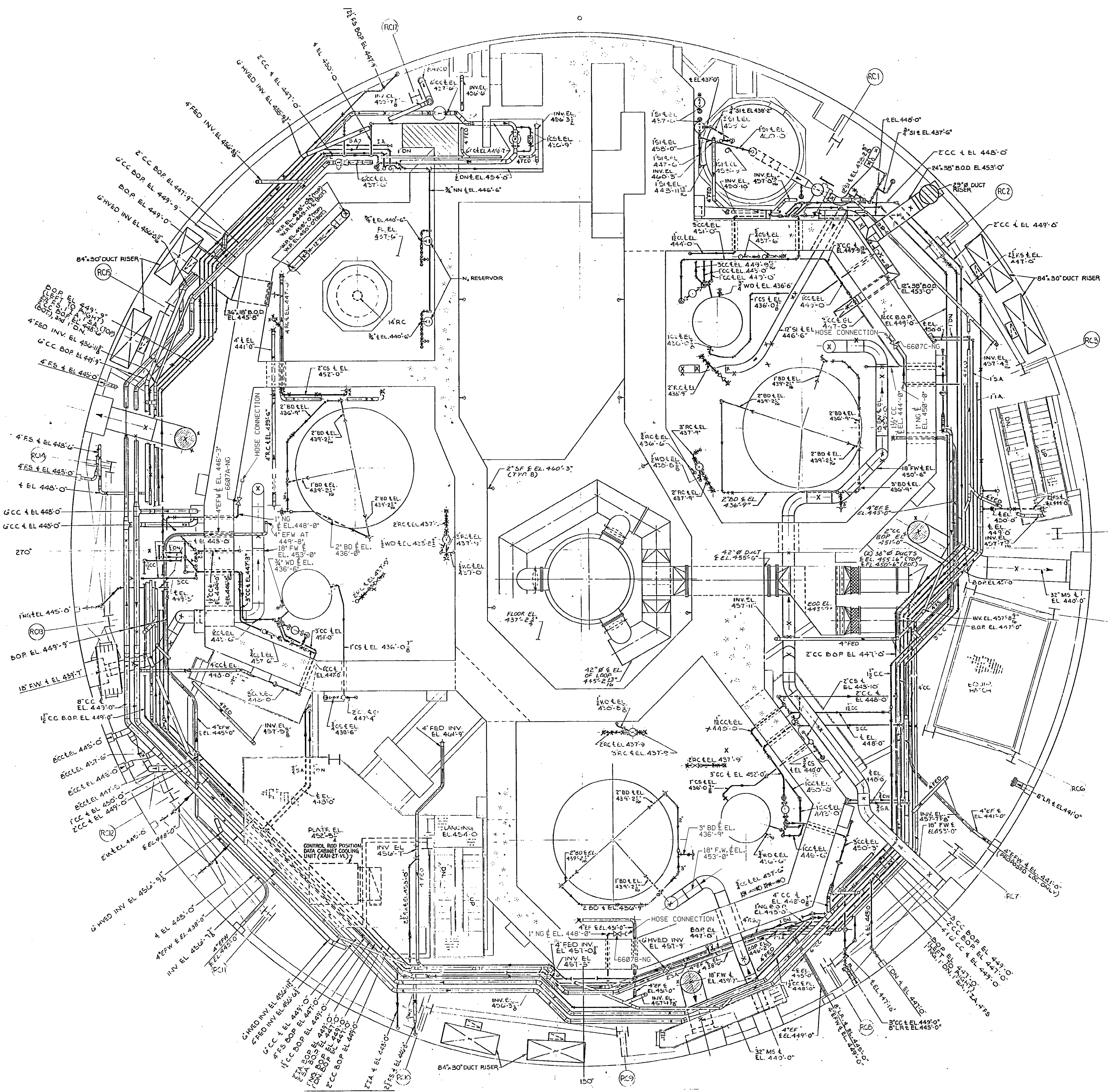
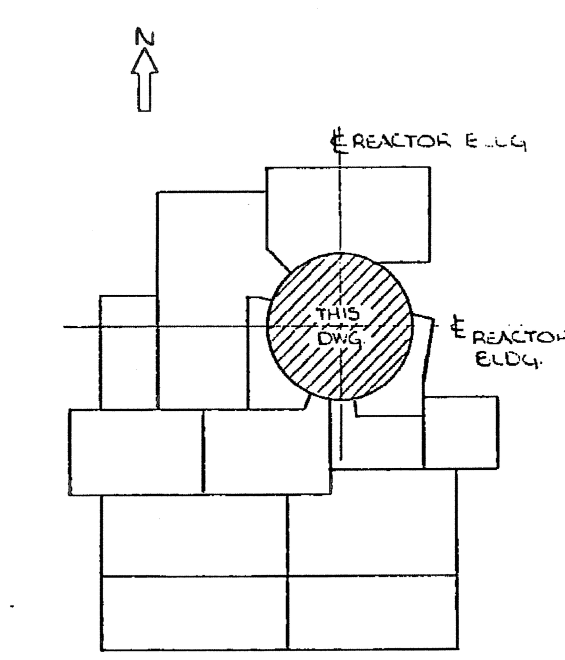




KEY PLAN

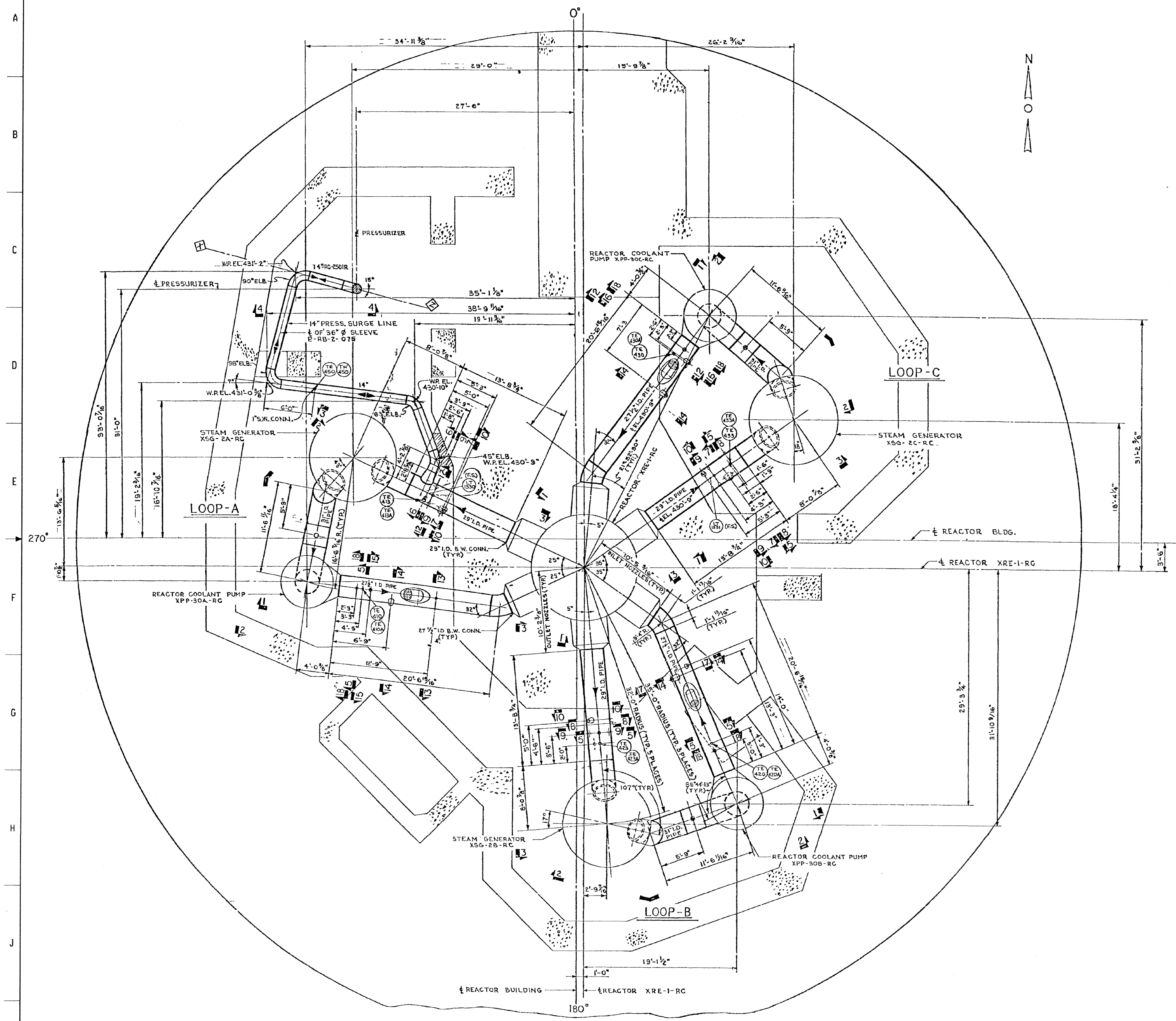
FOR INFORMATION ONLY
NOT CHECKED

Amendment 96-03
September 1996

SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Reactor Building Composite Plan
Below Elevation 463'-0"
(GAI Dwg. No. E-303-131)

Figure 6.2-68



PLAN

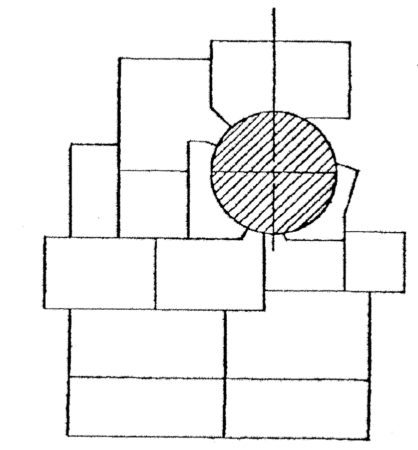
SCALE: 1/4"=1'-0"

NOTES:-

1. FOR PIPE MATERIAL SHOWN ON THIS DRAWING, REFER TO WESTINGHOUSE SPECIFICATIONS 679148, 673188 & 678204 AND LINE SPEC. 2501R.
2. FOR INSULATION, REFER TO SPECIFICATION SP-436-044461-000
3. PIPING TO BE ANS SAFETY CLASSIFICATION 1, EXCEPT AS NOTED.
4. NO ALLOWANCE MADE FOR WELD GAPS. - FOR MAIN LOOP WELD END DETAILS, SEE WESTINGHOUSE DRAWINGS AA-42332, 42333, 42828, 42834, 43098, 43233 & 43234. FOR 14" SURGE LINE SEE WESTINGHOUSE DWG. 271C800.
5. ALL ATTACHMENTS WELDED TO ALLOY PIPE SHALL BE SAME MATERIAL AS PIPE.
6. CONTRACTOR TO BE SURE INSTRUMENTS, VALVES AND ALL APPENDAGES CLEAR INSULATION REQUIREMENTS AT TIME OF INSTALLATION AND ARE READILY ACCESSIBLE AND READABLE.
7. ALL CONNECTING FIELD RUN PIPE LINES TO BE SUFFICIENTLY FLEXIBLE TO HAVE NO RESTRAINING EFFECT ON FREE THERMAL MOVEMENT OF MAIN PIPE LINE.

DRAWING REFERENCES:

E-304-602	REACTOR COOLANT MAIN LOOP SECTIONS & DETAILS
114E072	SHEETS 1 & 2 WESTINGHOUSE REACTOR COOLANT SYSTEM FLOW DIAGRAM
E-302-601	REACTOR COOLANT - FLOW DIAGRAM
E-302-602	REACTOR COOLANT - FLOW DIAGRAM
E-302-603	REACTOR COOLANT - FLOW DIAGRAM
E-302-604	REACTOR COOLANT - FLOW DIAGRAM
E-302-605	REACTOR COOLANT - FLOW DIAGRAM



KEY PLAN

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FSAR Figure 6.2-69

SOUTH CAROLINA ELECTRIC & GAS COMPANY

VIRGIL C. SUMMER NUCLEAR STATION

PIPING SYSTEMS

REACTOR COOLANT MAIN LOOP

PLAN

DESIGN ENGINEERING

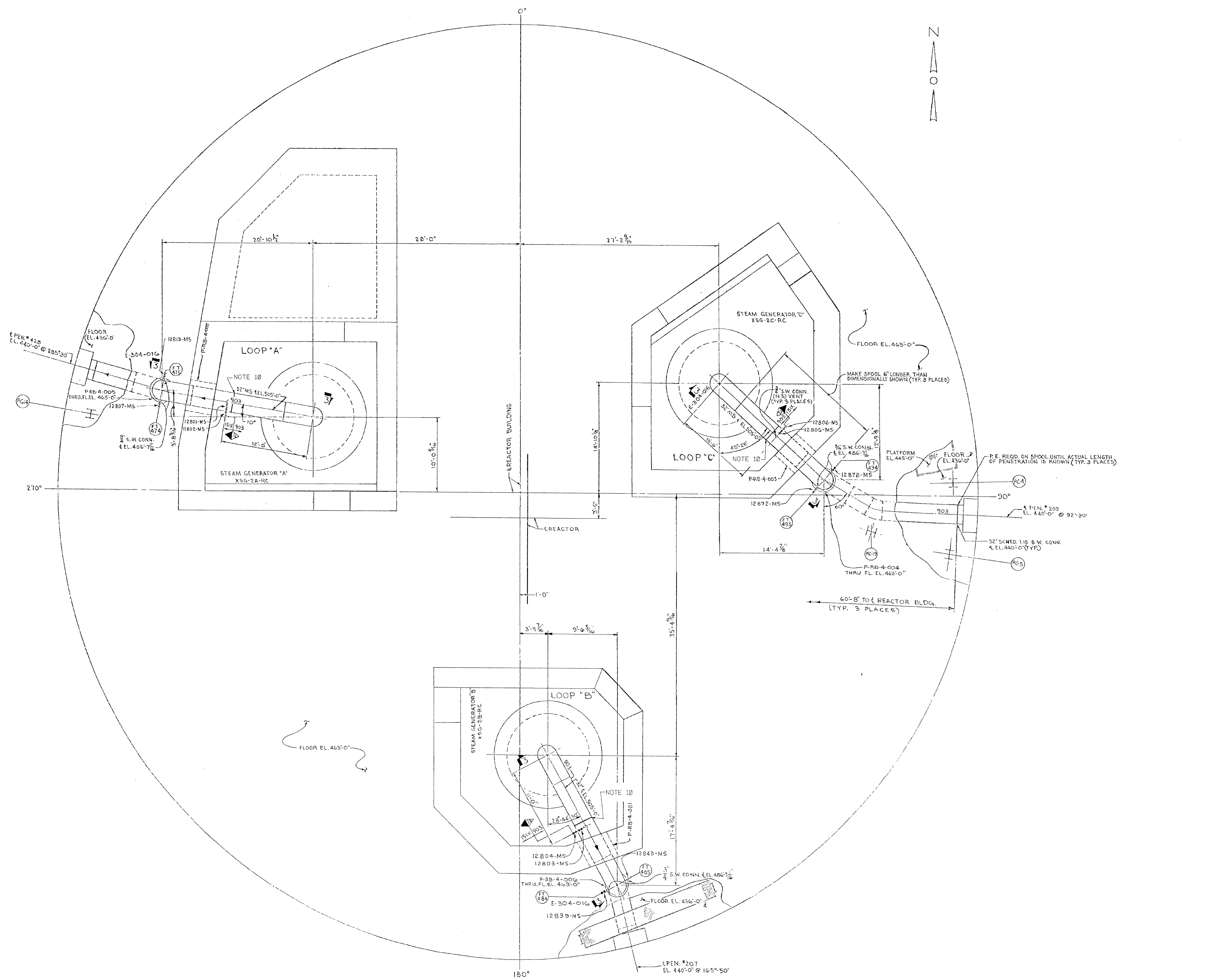
SCE&G A SCANA COMPANY — V. C. SUMMER NUCLEAR STATION JENKINSVILLE, S. C.

MADE	CHECKED	LE APPROVAL
1. DDJ	2. MGR	3. DDJ

SCALE	DRAWING NUMBER	SHT. NUMBER	REV
NO SCALE	E-304-601		9

NO.	DATE	BY	REVISION	CKD. BY	LE APPROVAL
9	9/25/01	DDJ	REVISED PER ECR-70028	MGR	DDJ

Amendment 02-01
May 2002

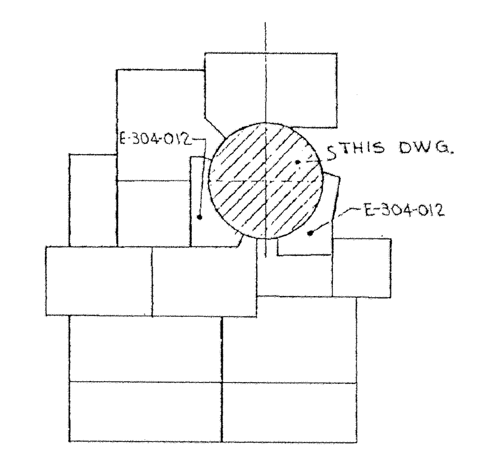


NOTES:

1. PIPING WITHIN THESE FLAGS IS TO MEET ALL THE REQUIREMENTS, EXCEPT FOR STAMPING, OF ASME CODE SECTION III CLASS 2 AND IS SPECIFICALLY INCLUDED IN THE SCOPE OF SPECIFICATION SP-544-044461-000.
2. FOR PIPE MATERIAL SHOWN ON THIS DRAWING, REFER TO PIPING SPECIFICATION SP-545-044461-000 LINE SPEC. AS NOTED.
3. PIPING TO BE ANS SAFETY CLASSIFICATION 2A.
4. NO ALLOWANCE MADE FOR WELD GAPS. FOR WELD END DETAILS, SEE DWG. C-301-00[illegible]
5. CONTRACTOR TO BE SURE INSTRUMENTS, VALVES AND ALL APPENDAGES CLEAR INSULATION REQUIREMENTS AT TIME OF INSTALLATION AND ARE READILY ACCESSIBLE AND READABLE.
6. ALL CONNECTING FIELD RUN PIPE LINES TO BE SUFFICIENTLY FLEXIBLE TO HAVE NO RESTRAINING EFFECT ON FREE THERMAL MOVEMENT OF MAIN PIPE LINE.
7. THE EMERGENCY FEEDWATER PUMP DRIVE TURBINE EXHAUST PIPING SHALL BE DESIGNED, FABRICATED AND INSTALLED IN ACCORDANCE WITH ASME B&PV CODE SECTION III CLASS 3 RULES, HOWEVER, THIS PIPING DOES NOT REQUIRE HYDROSTATIC TESTING AND CODE STAMPING.
8. FOR INSULATION, REFER TO SPECIFICATION SP-136-044461-000.
9. TEMPORARY STRAINER FOR SYSTEM CLEAN UP, TO BE REMOVED AFTER FINAL FLUSH.
10. EXISTING MAIN STEAM PIPE (HDR PIPE AT BIOWALL PEN. AND OUTSIDE BIOWALL) WAS REALIGNED LATERALLY AND AXIALLY 0.50" MAX FOR FIT-UP WITH M.S. SPOOL PIPE (SG NOZZLE TO FIELD WELD).

DRAWING REFERENCES:

E-304-012	MAIN STEAM (NUCLEAR) - INTERMEDIATE BUILDING & PENETRATION ROOM PLAN ABOVE EL. 436'-0"
E-304-016	MAIN STEAM (NUCLEAR) - INTERMEDIATE BUILDING PLAN BELOW EL. 436'-0" AND SECTIONS
D-302-011	MAIN STEAM (NUCLEAR) FLOW DIAGRAM



KEY PLAN

CONSTRUCTION COMPLETE & RSG AS-BUILT DWG.

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FSAR Figure 6.2-70

Amendment 02-01
May 2002

NO.	DATE	BY	REVISION	CKD. BY	LE APPROVAL
10	9/25/01	DDJ	REVISED PER ECR-70028	MGR	DDJ
9	03/27/95	DDJ	REVISED PER MRF-90001	MGR	GGW

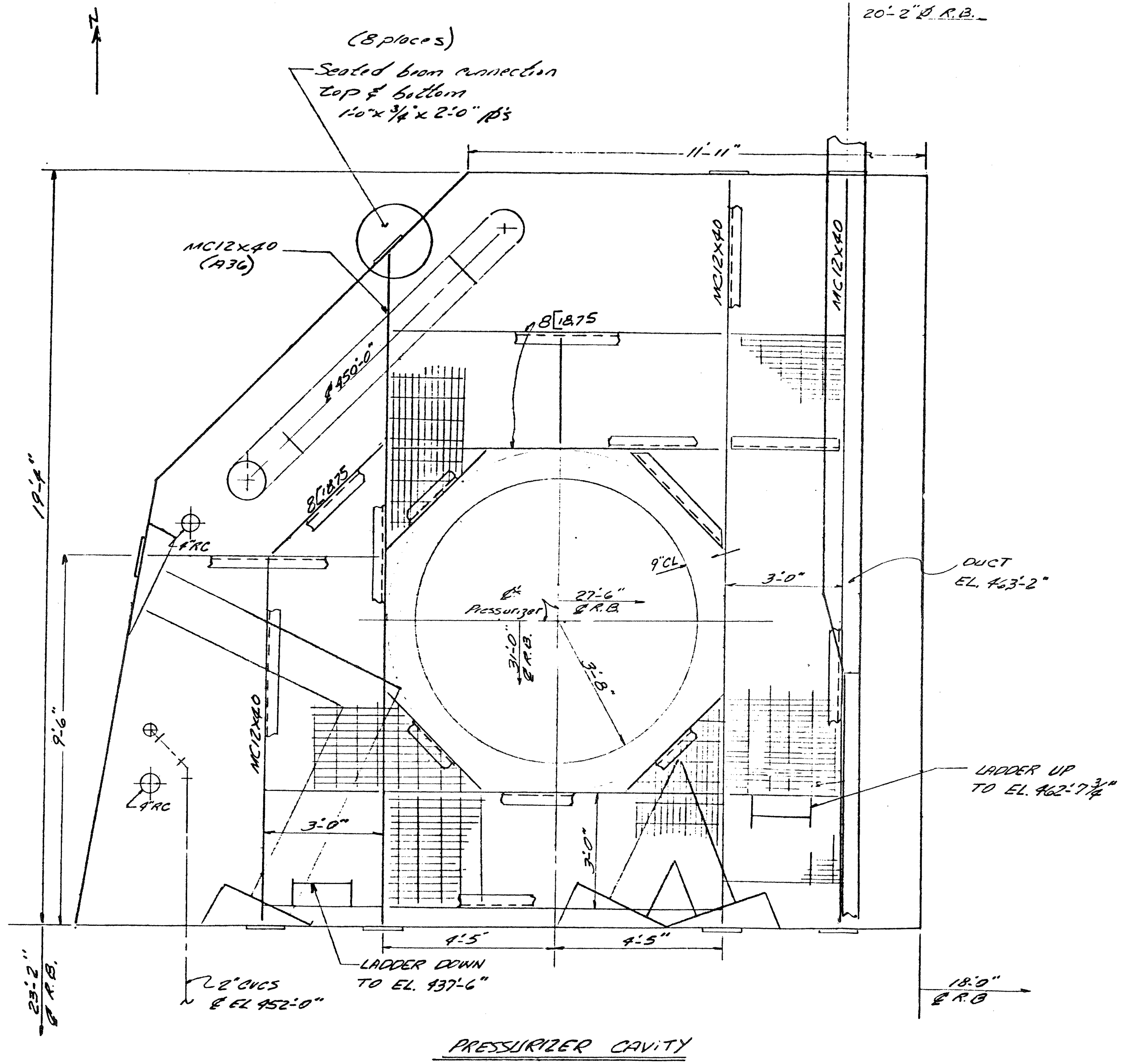
SOUTH CAROLINA ELECTRIC & GAS COMPANY
VIRGIL C. SUMMER NUCLEAR STATION
PIPING SYSTEM
MAIN STEAM (NUCLEAR)- REACTOR BUILDING
PLAN & SECTIONS ABOVE ELEV. 436'-0"
SCE&G A SCANA COMPANY
DESIGN ENGINEERING
V. C. SUMMER NUCLEAR STATION JENKINSVILLE, S. C.

MADE	CHECKED	LE APPROVAL
1. DDJ	2. MGR	3. GGW

SCALE: NO SCALE | DRAWING NUMBER: E-304-011 | SHT. NUMBER: | REV: 10

1/4" = 1'-0" 0' 5' 10' 15' 20'

304011_0001.m DJ14998 Tuesday September 25 2001 09:31:27 AM EDT

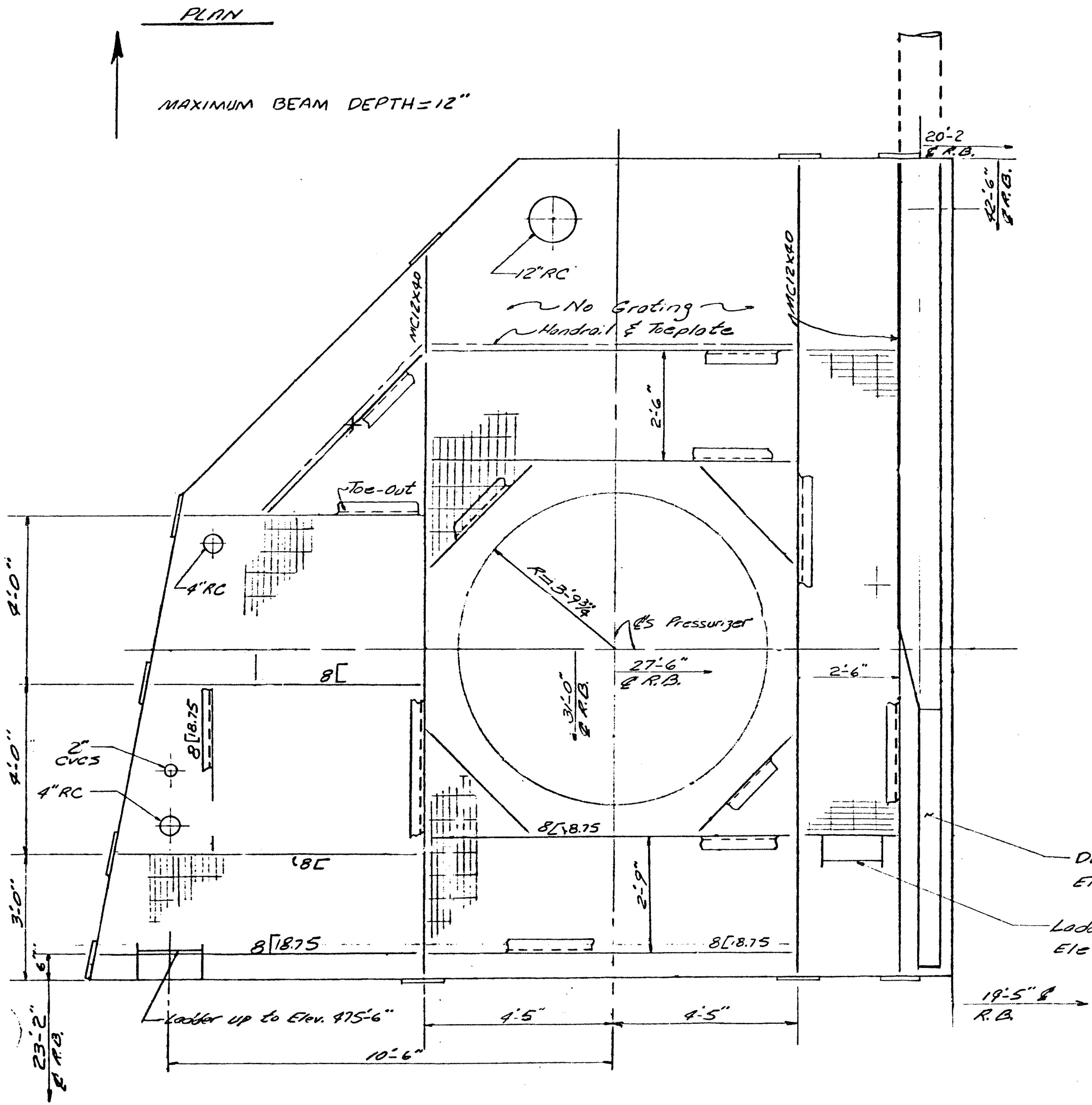


Amendment 0
August 1984

SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Reactor Building - ISI Platform
Pressurizer Cavity Platform
Elevation 452' - 6"

Figure 6.2-71

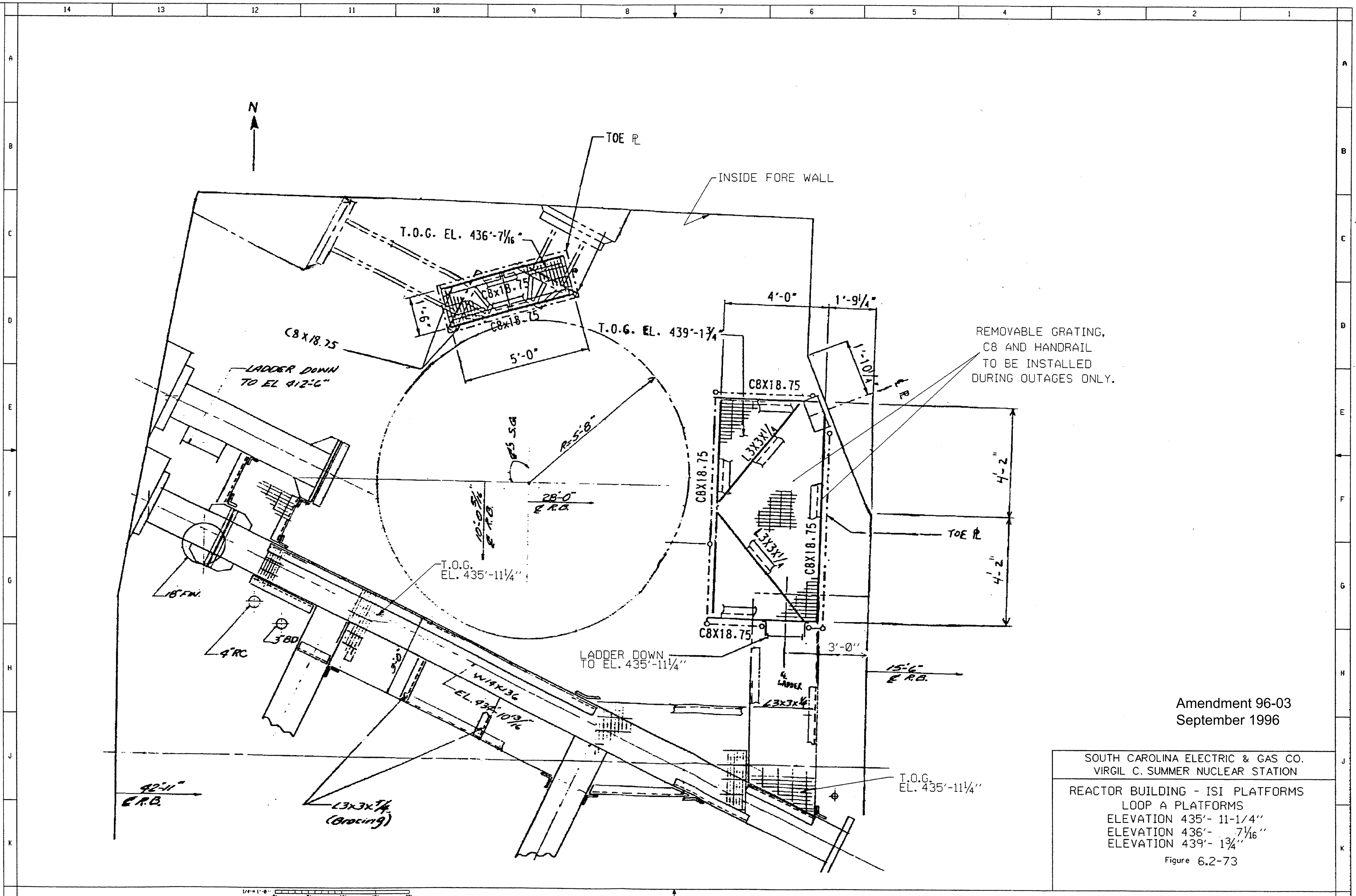


Amendment 0
August 1984

SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Reactor Building - ISI Platform
Pressurizer Cavity Platform
Elevation 462' - 7-3/4"

Figure 6.2-72

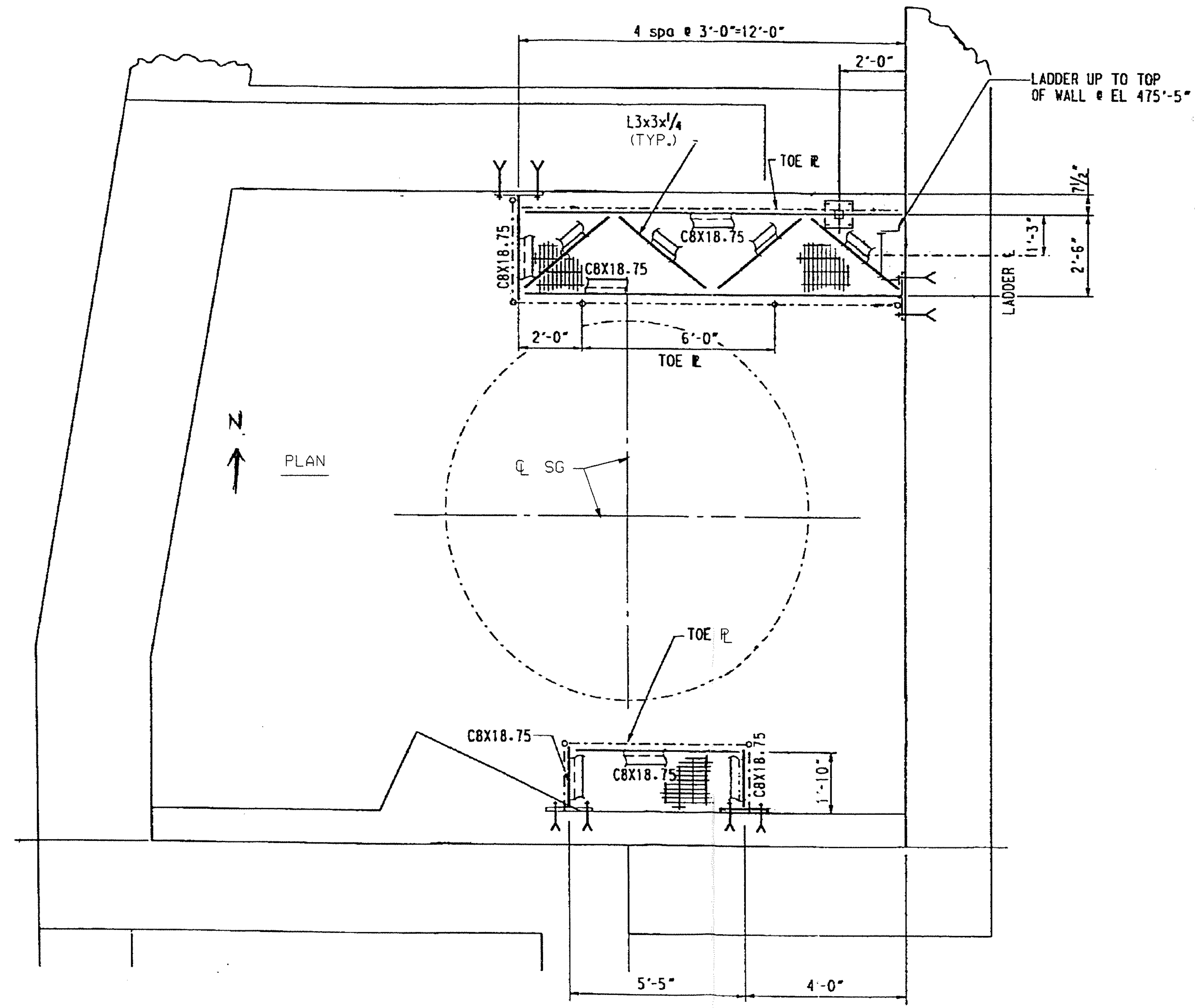


Amendment 96-03
September 1996

SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

REACTOR BUILDING - ISI PLATFORMS
LOOP A PLATFORMS
ELEVATION 435'- 11-1/4"
ELEVATION 436'- 7 1/16"
ELEVATION 439'- 1 3/4"

Figure 6.2-73

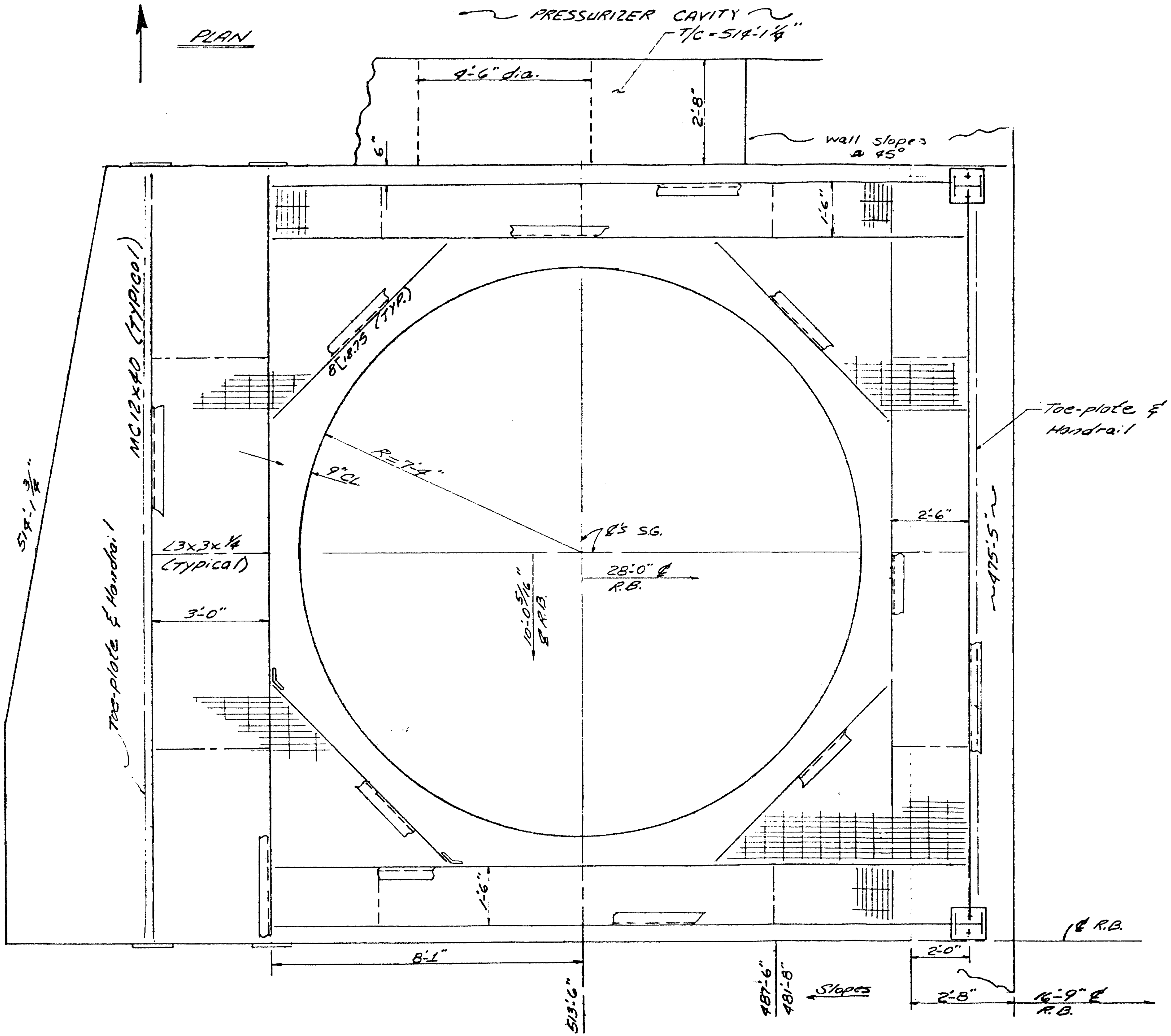


Amendment 96-03
September 1996

SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

REACTOR BUILDING - ISI PLATFORMS
LOOP A PLATFORM
ELEVATION 466'-11"

Figure 6.2-73a

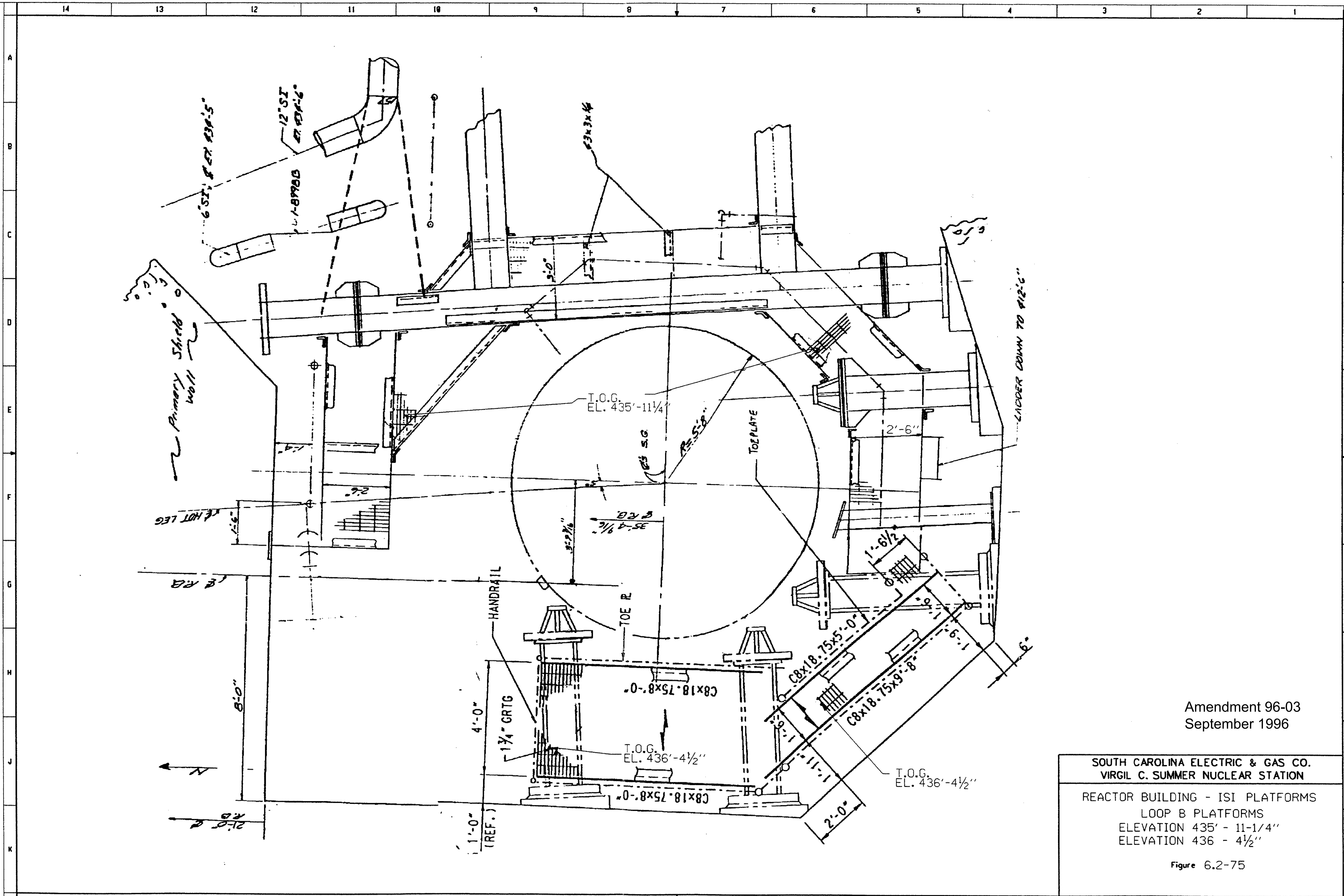


Amendment 0
August 1984

SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Reactor Building - ISI Platforms
Loop A Platform
Elevation 488' - 6"

Figure 6.2-74

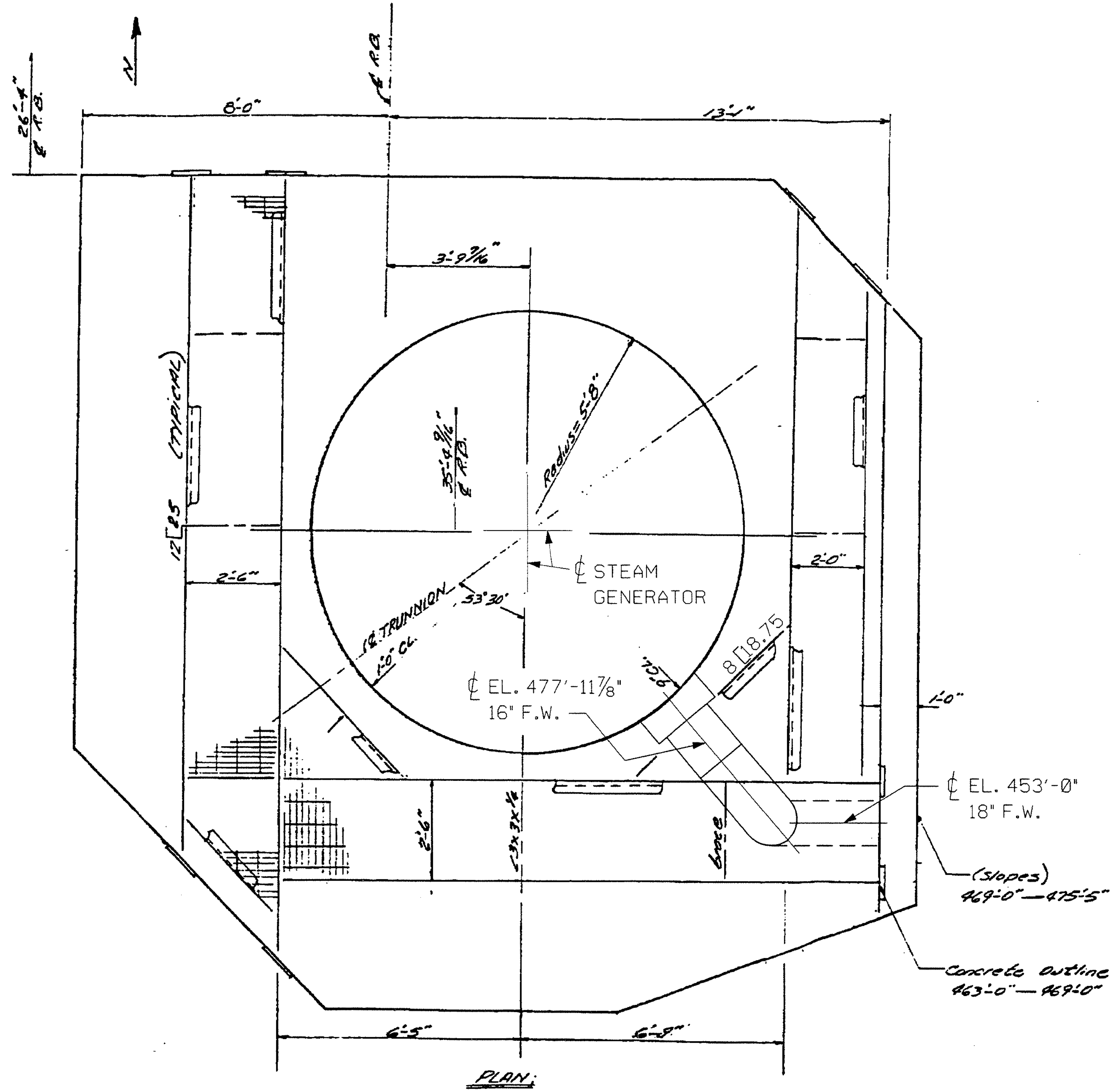


Amendment 96-03
September 1996

SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

REACTOR BUILDING - ISI PLATFORMS
LOOP B PLATFORMS
ELEVATION 435' - 11-1/4"
ELEVATION 436 - 4½"

Figure 6.2-75

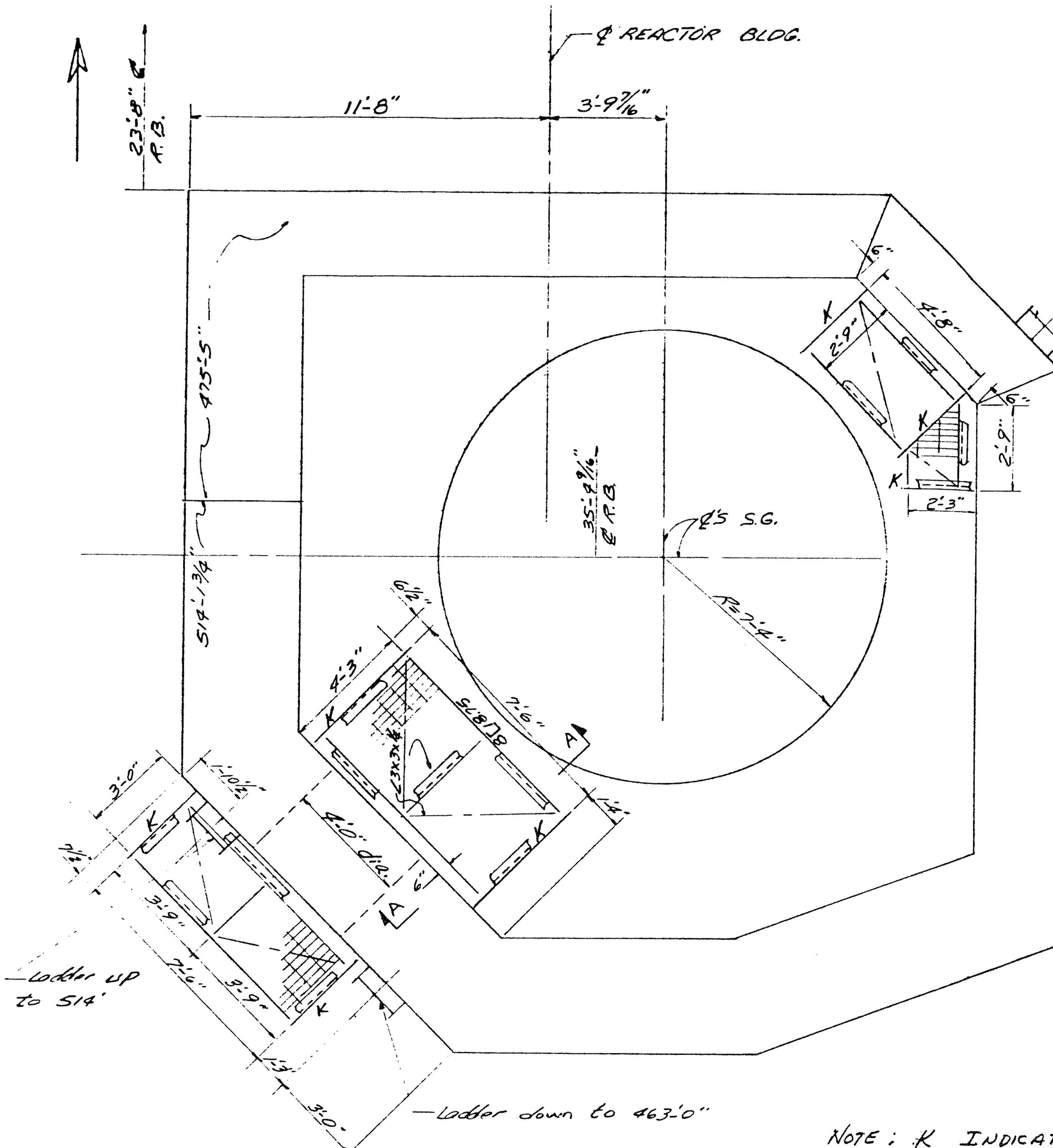


Amendment 96-03
September 1996

SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

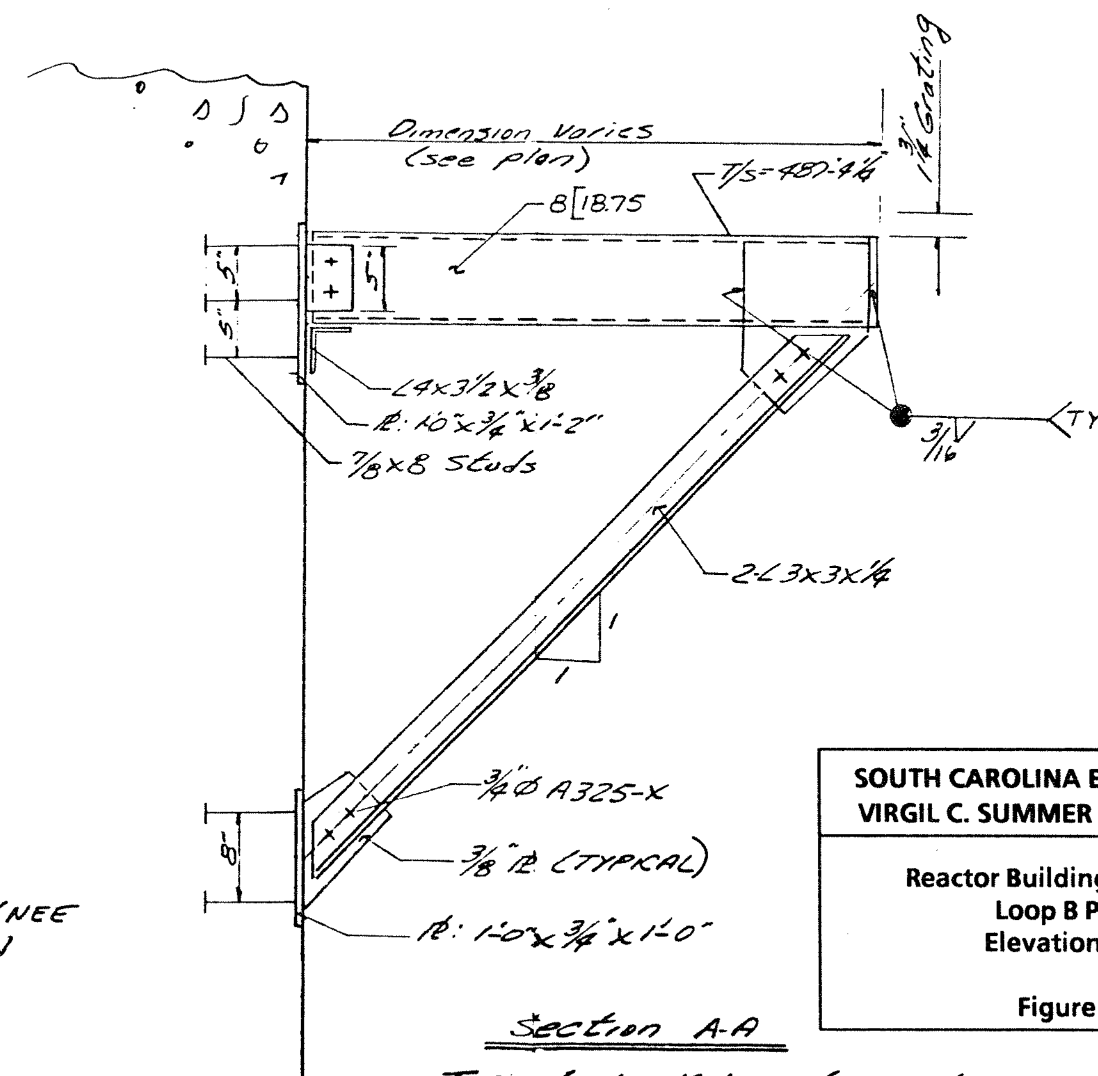
Reactor Building - ISI Platform
Loop B Platform
Elevation 467' - 9"

Figure **6.2-76**



Note: K Indicates Knee Brace Location

PLAN: LOOP B
I.S.I. PLATFORMS



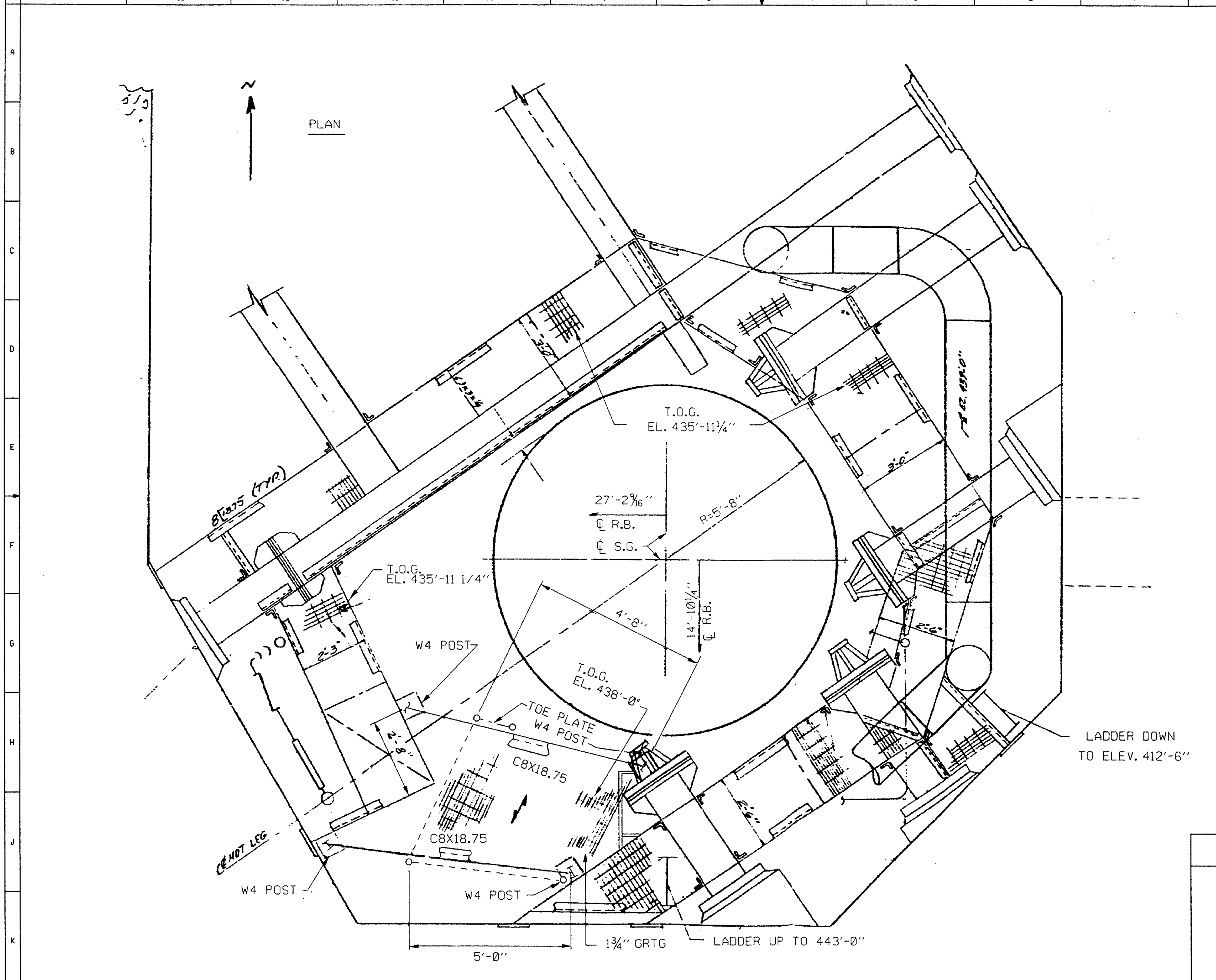
Section A-A
Typical for K-braced members

Amendment 0
August 1984

SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Reactor Building - ISI Platform
Loop B Platform
Elevation 487' - 6"

Figure 6.2-77

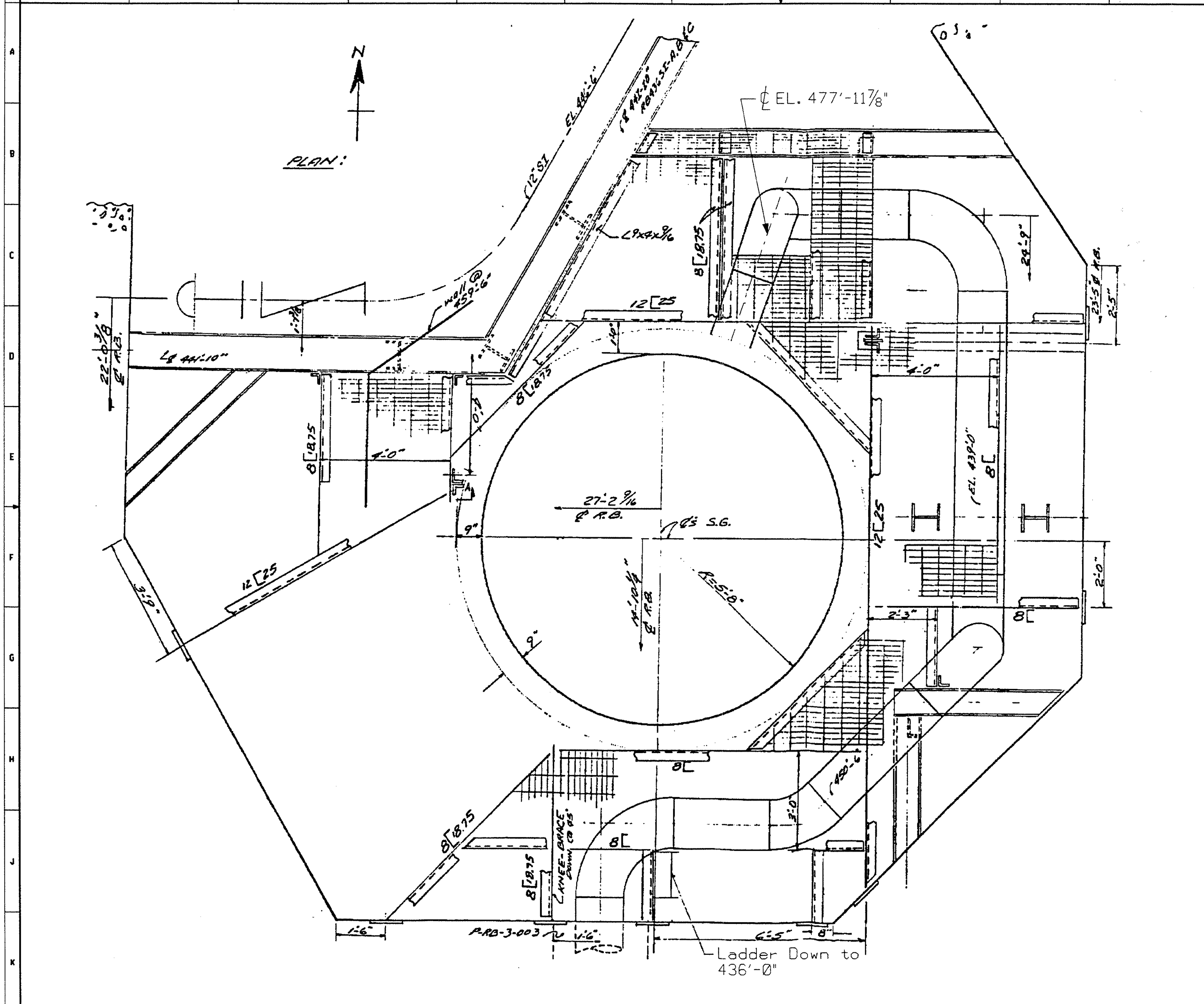


Amendment 96-03
September 1996

SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

REACTOR BUILDING - ISI PLATFORMS
LOOP C PLATFORMS
ELEVATION 435' - 11-1/4"
ELEVATION 438' - 0"

Figure 6.2-78

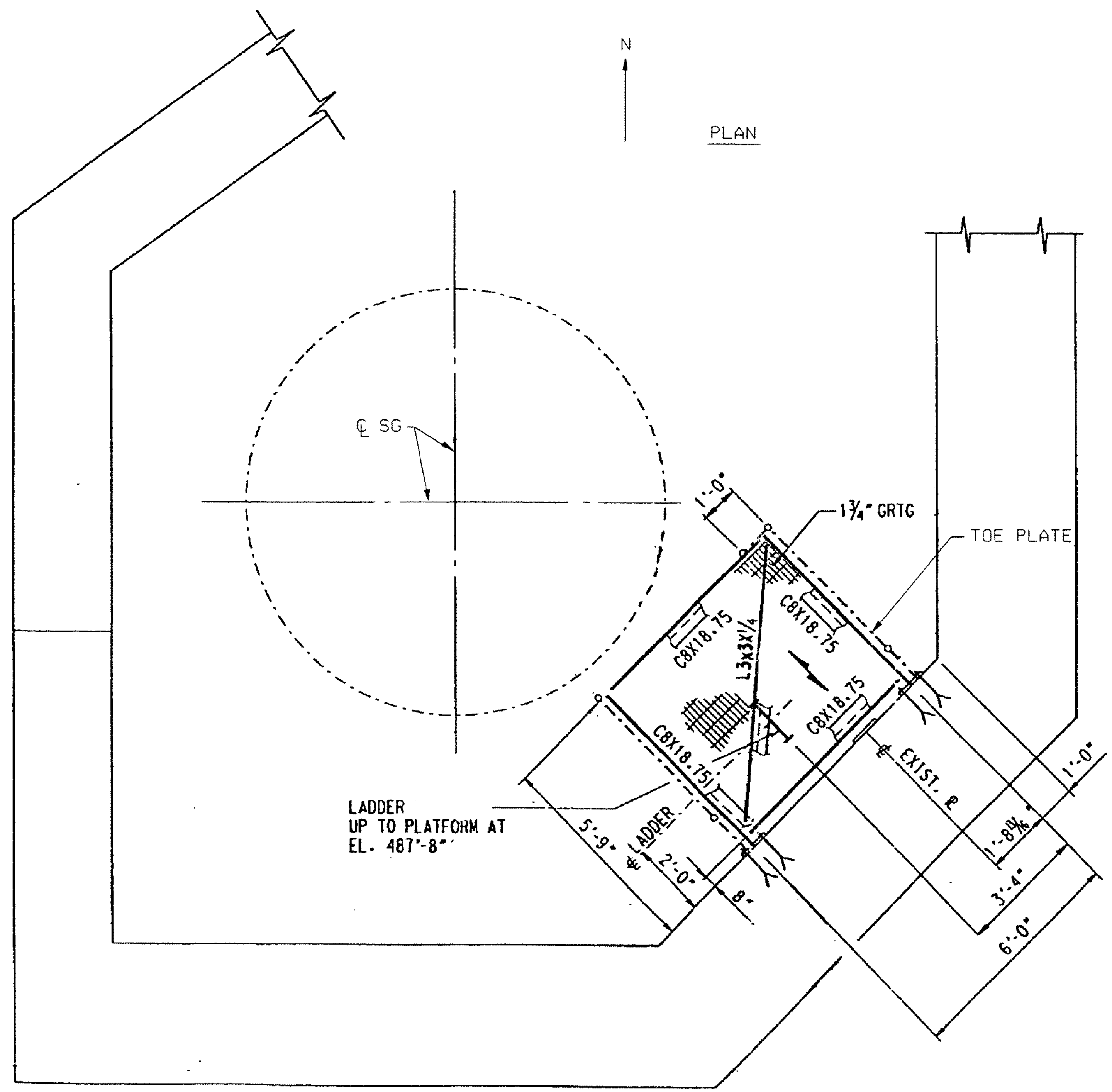


Amendment 96-03
September 1996

SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Reactor Building - ISI Platforms
Loop C Platform
Elevation 443' - 0"

Figure **6.2-79**

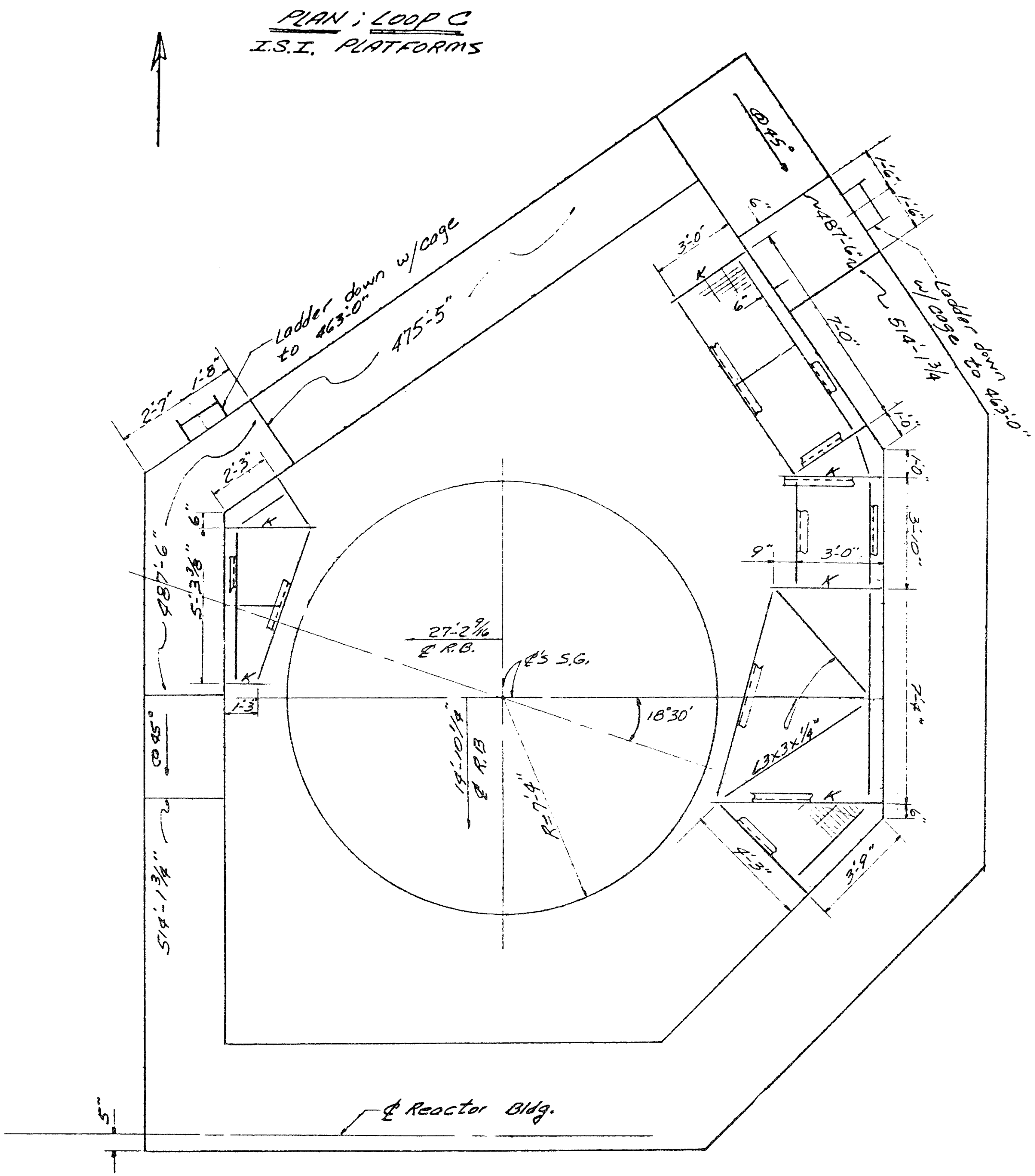


Amendment 96-03
September 1996

SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

REACTOR BUILDING - ISI PLATFORMS
LOOP C PLATFORMS
ELEVATION 467' - 0"

Figure 6.2-79a

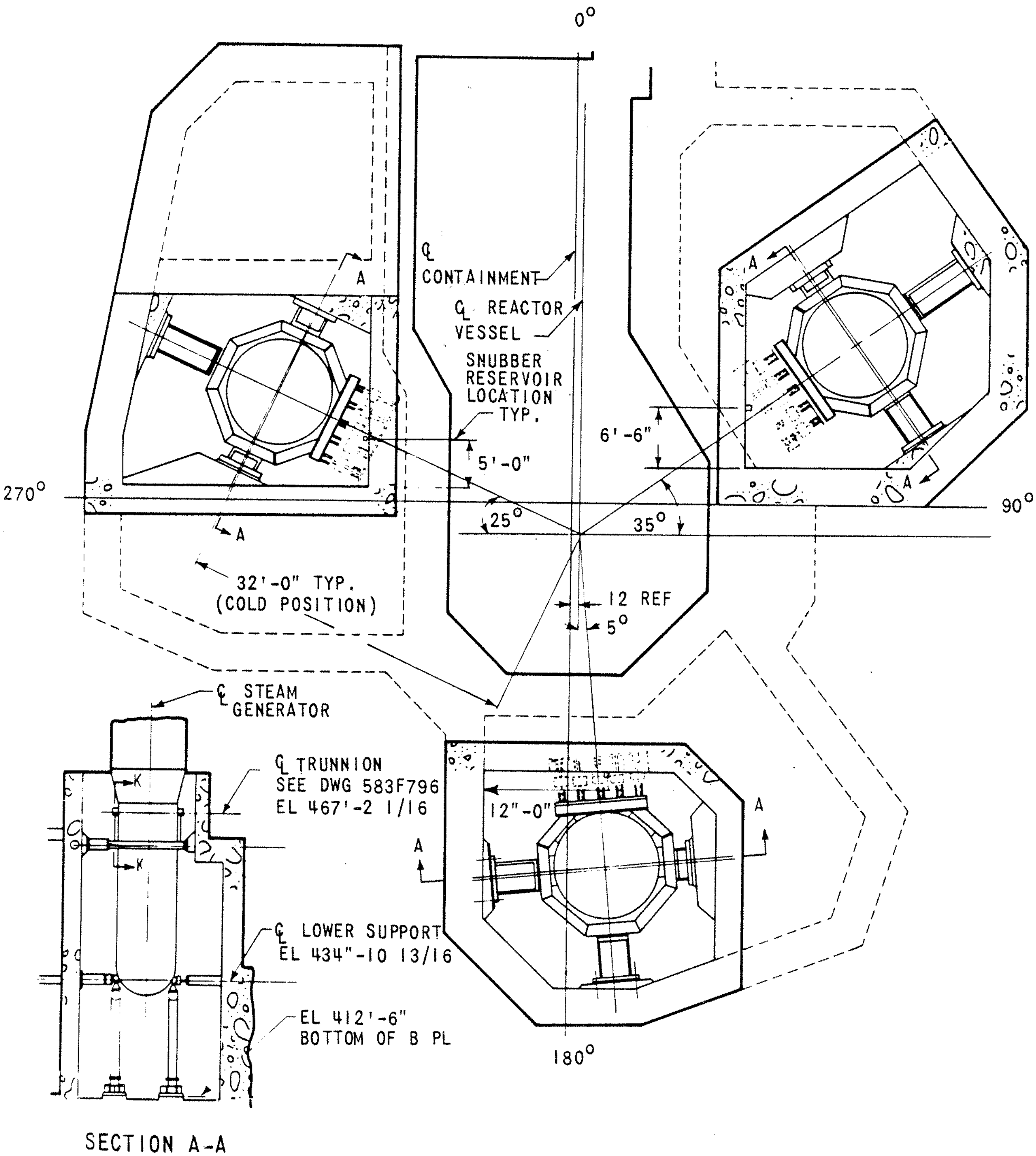


Amendment 0
August 1984

SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Reactor Building - ISI Platforms
Loop C Platform
Elevations 487' - 6", 487' - 8"

Figure 6.2-80

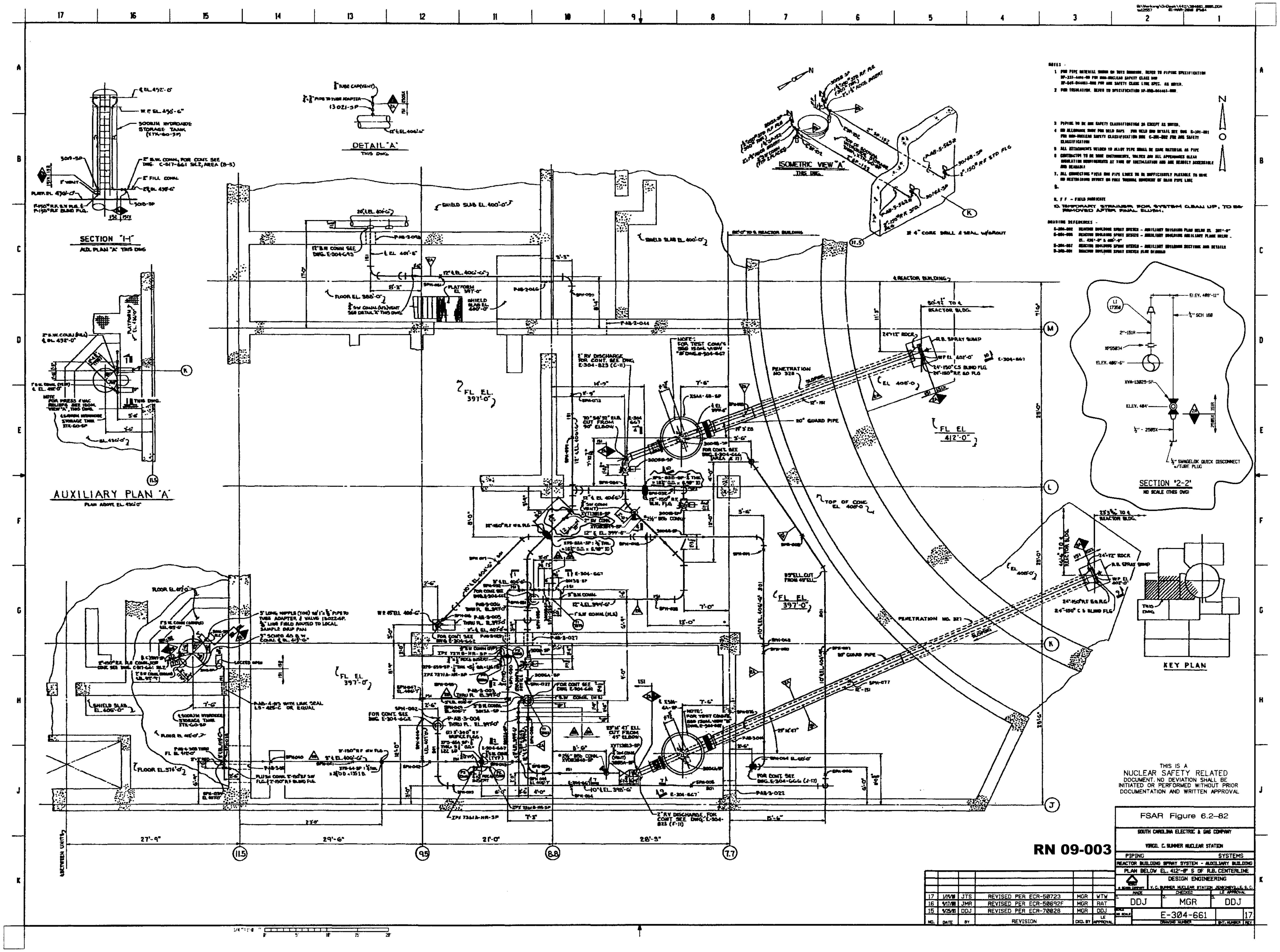


Amendment 0
August 1984

SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Steam Generator Upper Lateral Supports - General Arrangement

Figure 6.2-81



SECTION "1-1"
AUXILIARY PLAN "A"
DETAIL "A"
ISOMETRIC VIEW "A"
SECTION "2-2"
KEY PLAN
FL EL 397'-0"
FL EL 412'-0"
PENETRATION NO. 327
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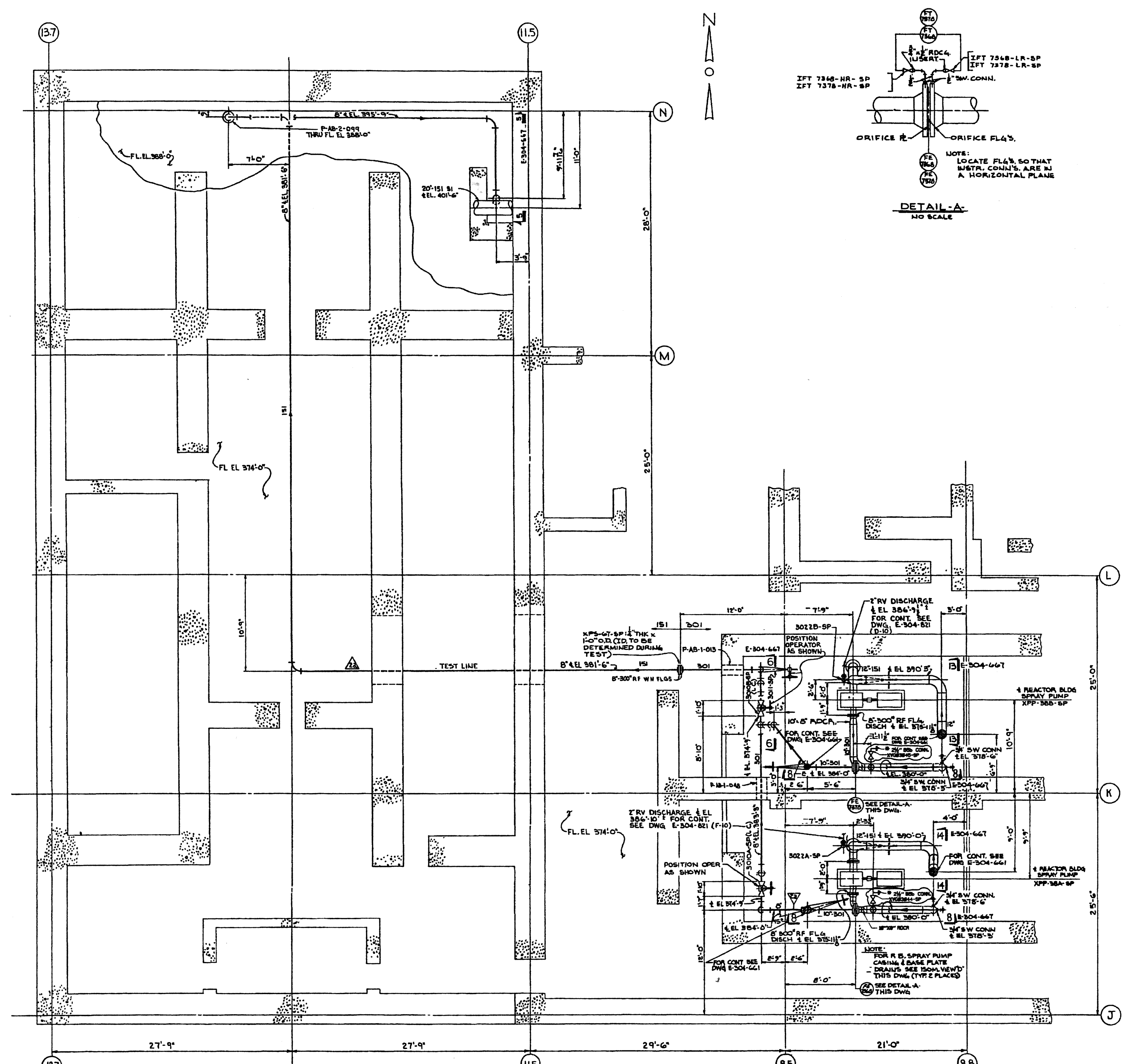
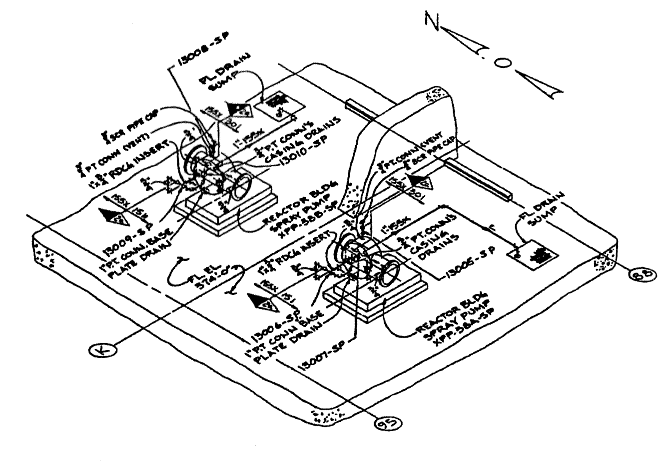
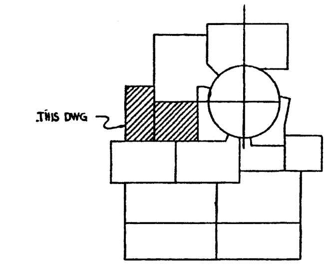
RN 09-003
FSAR Figure 6.2-82
SOUTH CAROLINA ELECTRIC & GAS COMPANY
VIRGIL C. SUMMER NUCLEAR STATION
PIPING SYSTEMS
REACTOR BUILDING SPRAY SYSTEM - AUXILIARY BUILDING
PLAN BELOW EL. 412'-0" S OF R.B. CENTERLINE
DESIGN ENGINEERING
17 REVISED PER ECR-50723
16 REVISED PER ECR-50892F
15 REVISED PER ECR-70828
REVISION
DDJ MGR DDJ
E-304-661
17



DETAIL -A-
NO SCALE



ISOMETRIC VIEW "D"
NO SCALE



KEY PLAN

NOTES:
1. FOR NOTES & REFERENCES SEE DWG. E-304-661.

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RN 07-028
November 2008

FSAR Figure 6.2–83

SOUTH CAROLINA ELECTRIC & GAS COMPANY

VIRGIL C. SUMMER NUCLEAR STATION

PIPING SYSTEM

REACTOR BUILDING SPRAY SYSTEM -
AUXILIARY BLDG PLAN BELOW EL. 397'-0"

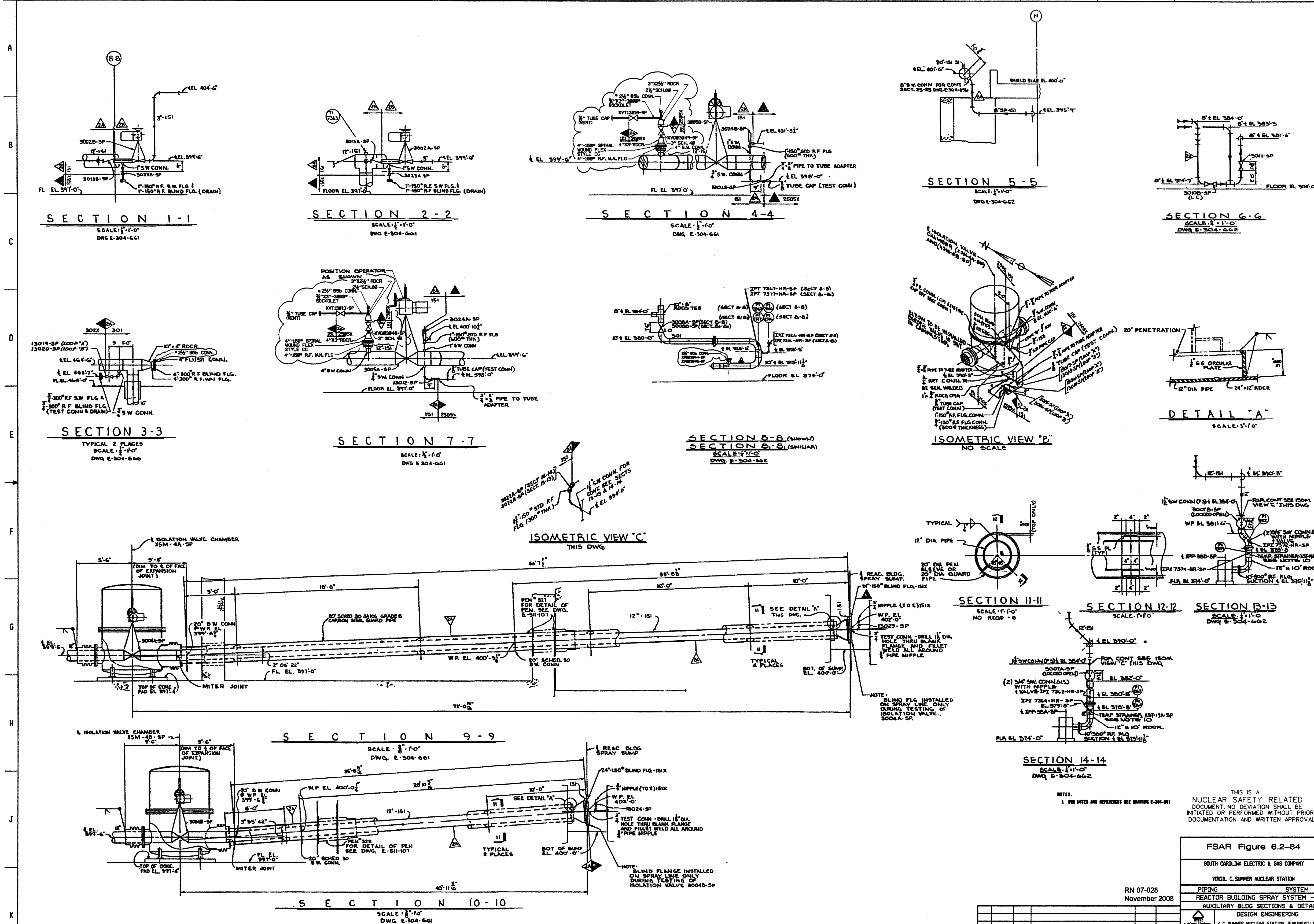
DESIGN ENGINEERING
SCE&G A SCANA COMPANY
V. C. SUMMER NUCLEAR STATION JENKINSVILLE, S. C.

MADE	CHECKED	LE APPROVAL
1. DDJ	2. MGR	3. DDJ

SCALE	DRAWING NUMBER	SHT. NUMBER	REV
NO SCALE	E-304-662		9

NO.	DATE	BY	REVISION	CKD. BY	LE APPROVAL
9	9/19/08	JMR	REVISED PER ECR-50692F	MGR	RAT
8	9/25/01	DDJ	REVISED PER ECR-70028	MGR	DDJ

1/4"=1'-0"



NOTES.
1 FOR NOTES AND REFERENCES SEE DRAWING E-304-661

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RN 07-028
November 2008

FSAR Figure 6.2–84

SOUTH CAROLINA ELECTRIC & GAS COMPANY

VIRGIL C. SUMMER NUCLEAR STATION

PIPING SYSTEM
REACTOR BUILDING SPRAY SYSTEM -
AUXILIARY BLDG SECTIONS & DETAILS

SCE&G A SCANA COMPANY
DESIGN ENGINEERING
V. C. SUMMER NUCLEAR STATION JENKINSVILLE, S. C.

MADE	CHECKED	LE APPROVAL
1. DDJ	2. MGR	3. DDJ

SCALE	DRAWING NUMBER	SHT. NUMBER	REV
NO SCALE	E-304-667		23

NO.	DATE	BY	REVISION	CKD. BY	LE APPROVAL
23	9/17/08	JMR	REVISED PER ECR-50692F	MGR	RAT
22	9/25/01	DDJ	REVISED PER ECR-70028	MGR	DDJ

6.3 EMERGENCY CORE COOLING SYSTEM

6.3.1 DESIGN BASES

6.3.1.1 Range of Coolant Ruptures and Leaks

The emergency core cooling system (ECCS) is designed to cool the reactor core as well as to provide additional shutdown capability following initiation of the following accident conditions:

1. Pipe breaks in the reactor coolant system (RCS) which cause a discharge larger than that which can be made up by the normal makeup system, up to and including the instantaneous circumferential rupture of the largest pipe in the RCS.

2. Rupture of a control rod drive mechanism causing a rod cluster control assembly ejection accident.

3. Pipe breaks in the steam system, up to and including the instantaneous circumferential rupture of the largest pipe in the steam system.

4. A steam generator tube rupture.

The acceptance criteria for the consequences of each of these accidents are described in Chapter 15 in the respective accident analyses sections.

6.3.1.2 Fission Product Decay Heat

The primary function of the ECCS following a loss of coolant accident is to remove the stored and fission product decay heat from the reactor core such that fuel rod damage, to the extent that it would impair effective cooling of the core, is prevented. The acceptance criteria for the accidents, as well as analyses of the accidents are provided in Chapter 15.

6.3.1.3 Reactivity Required for Cold Shutdown

The ECCS provides shutdown capability following the accidents listed in Section 6.3.1.1 by means of injecting borated water. The most critical accident for shutdown capability, relative to this system, is the steam line break and for this the ECCS meets the criteria defined in Chapter 15.

6.3.1.4 Capability to Meet Functional Requirements

To ensure that the ECCS will perform its desired function during the accidents listed in Section 6.3.1.1, it is designed to tolerate a single active failure during the short term immediately following an accident, or to tolerate a single active or passive failure during the long term following an accident. This subject is detailed in Sections 3.1 and 6.3.2.11. 00-01

6.3.1.5 Loss of Offsite Power

The ECCS is designed to meet its minimum required level of functional performance with onsite electrical power system operation (assuming offsite power is not available) or with offsite electrical power system operation for any of the above abnormal occurrences assuming a single failure as defined above.

6.3.1.6 Seismic Requirements

The ECCS is designed to perform its function of ensuring core cooling and providing shutdown capability following an accident under simultaneous safe shutdown earthquake (SSE) loading. The seismic requirements are defined in Chapter 3.

6.3.2 SYSTEM DESIGN

The ECCS components are designed such that a minimum of two accumulators, one charging pump and one residual heat removal pump together with their associated valves and piping will assure adequate core cooling in the event of a Design Basis Accident. The redundant onsite emergency diesels assure adequate emergency power to all electrically operated components in the event that a loss of offsite power occurs simultaneously with a loss of coolant accident, even assuming a single failure in the emergency power system such as the failure of one diesel to start. Also, at strategically identified local high points, the ECCS is provided with indicating air-traps to ensure the ECCS remains void free (reference NRC GL 2008-01). RN 09-003

6.3.2.1 Schematic Piping and Instrumentation Diagrams

Schematic piping and instrumentation and process flow diagrams of the ECCS are shown in Figure 6.3-1 (Sheets 1 through 3) and Figure 6.3-2, respectively. Pertinent design and operating parameters for the components of the ECCS are given in Table designed are listed in Table 3.2-1.

6.3.2.2 Equipment and Component Descriptions

The component design and operating conditions are specified as the most severe conditions to which each respective component is exposed. For each component, these conditions are considered in relation to the Code to which it is designed. By designing the components in accordance with applicable codes, and with due consideration for the design and operating conditions, the fundamental assurance of integrity of the ECCS components is maintained. Components of the ECCS are designed to withstand the appropriate seismic loadings in accordance with their safety class as given in Table 3.2-1.

The major mechanical components of the ECCS are discussed in Sections 6.3.2.2.1 through 6.3.2.2.6.

6.3.2.2.1 Accumulators

The accumulators are pressure vessels partially filled with borated water and pressurized with nitrogen gas. During normal operation each accumulator is isolated from the RCS by two check valves in series. Should the RCS pressure fall below the Accumulator pressure, the check valves open and borated water is forced into the RCS. One accumulator is attached to each of the cold legs of the RCS. Mechanical operation of the swing-disc check valves is the only action required to open the injection path from the accumulators to the core via the cold leg.

Connections are provided for remotely adjusting the level and boron concentration of the borated water in each accumulator during normal operation as required. Accumulator water level may be adjusted either by draining to the refueling water storage tank or by pumping borated water from the refueling water storage tank to the accumulator. Samples of the solution in the accumulators are taken periodically for checks of boron concentration.

Accumulator pressure is provided by a supply of nitrogen gas, and can be adjusted as required during normal plant operation; however, the accumulators are normally isolated from this nitrogen supply. Gas relief valves on the accumulators protect them from pressures in excess of design pressure.

The accumulators are located within the reactor building but outside of the secondary shield wall which protects them from missiles. Since the accumulators are located within the reactor building, a release of the nitrogen gas in the accumulators would cause an increase in normal containment pressure. Containment pressure increase following release of the gas from all accumulators has been calculated and is well below the containment pressure setpoint for ECCS actuation.

Release of accumulator gas is indicated by the accumulator pressure indicators and alarms. Thus, the operator takes action promptly as required to maintain plant operation within the requirements of the Technical Specification covering accumulator operability.

6.3.2.2.2 Boron Injection Tank

Deleted by Amendment 3.

6.3.2.2.3 Boron Injection Surge Tank

Deleted by Amendment 3.

6.3.2.2.4 Pumps

6.3.2.2.4.1 Residual Heat Removal Pumps

In the event of a loss of coolant accident the residual heat removal pumps are started automatically on receipt of a safety injection signal. The residual heat removal pumps deliver water to the RCS from the refueling water storage tank during the injection phase and from the reactor building sump during the recirculation phase. Each residual heat removal pump is a single stage, vertical position, centrifugal pump. It has an integral motor-pump shaft driven by an induction motor. The unit has a self-contained mechanical seal cooled by component cooling water.

A minimum flow bypass line is provided for the pumps to recirculate and return the pump discharge fluid to the pump suction should the pumps be started with their normal flow paths blocked. Once flow is established to the RCS, the bypass line is automatically closed. This line prevents deadheading of the pumps and permits pump testing during normal operation.

The residual heat removal pumps are discussed further in Section 5.5.7. A pump performance curve is given in Figure 6.3-3.

6.3.2.2.4.2 Centrifugal Charging Pumps

In the event of an accident. the two operable charging pumps are started automatically on receipt of a safety injection signal and are automatically aligned to take suction from the refueling water storage tank during injection. During recirculation, suction is provided by the residual heat removal pumps.

These pumps deliver flow to the RCS at the prevailing RCS pressure. Each centrifugal charging pump is a multistage diffuser design, barrel-type casing with vertical suction and discharge nozzles. The pump is driven through a speed increaser connected to an induction motor. The unit has a self-contained lubrication system which is cooled by component cooling water and a self-contained mechanical seal cooling system. 98-01

A minimum flow bypass line is provided on each pump discharge to recirculate flow to the volume control tank after cooling via the seal water heat exchanger during normal plant operation. Each minimum flow bypass line contains a valve which may be closed during the switchover from injection to recirculation. The common minimum flow bypass line also contains a valve which may be closed during the switchover to recirculation. A safety injection signal closes the valves to isolate the normal charging line and volume control tank and opens the charging pump/refueling water storage tank suction valves to align the high head portion of the ECCS for injection. During normal plant operation, at least one charging pump is continuously in service. The second pump is a non-running pump on the inactive loop and the third, if available, is designated as a spare and its breaker(s) are racked out. The other charging pumps may be tested during power operation via the minimum flow bypass lines.

Pump performance curves are given in Figures 6.3-4a, b, and c. 00-01

6.3.2.2.4.3 Boron Injection Recirculation Pumps

Deleted by Amendment 3.

6.3.2.2.4.4 Positive Displacement Hydrostatic Test Pump

This pump serves two functions, neither of which is safety-related. Permanent connections are provided to the accumulators to allow addition of borated water from the refueling water storage tank. Temporary connections permit the use of this pump in hydro-testing the plant piping systems and leak testing check valves as described in Section 5.2.2.4. 00-01

6.3.2.2.5 Residual Heat Exchangers

The residual heat exchangers are conventional shell and U-tube type units. The tubes are seal welded to the tubesheet. During normal cooldown operation, the residual heat removal pumps circulate reactor coolant through the tube side while component cooling water flows through the shell side. During emergency core cooling recirculation operation, water from the reactor building sump flows through the tube side.

A further discussion of the residual heat exchangers is found in Section 5.5.7.

6.3.2.2.6 Valves

Design parameters for all types of valves used in the ECCS are given in Table 6.3-1. Operability requirements are discussed in Section 3.9.

Design features employed to minimize valve leakage include:

1. Where possible, packless valves are used.

2. Other valves which are normally open, except check valves and those which perform a control function, are provided with backseats to limit stem leakage.

3. Normally closed globe valves are installed with recirculation fluid pressure under the seat to prevent stem leakage of recirculated (radioactive) water.

4. Relief valves are enclosed, i.e., they are provided with closed bonnet.

6.3.2.2.6.1 Motor Operated Valves

The seating design of all motor operated gate valves is of the Crane flexible wedge design. These designs release the mechanical holding force during the first increment of travel so that the motor operator works only against the frictional component of the hydraulic unbalance on the disc and the packing box friction. The discs are guided throughout the full disc travel to prevent chattering and to provide ease of gate movement. The seating surfaces are hard faced to prevent galling and to reduce wear. The discs for the SI/RHR sump recirc valves, XVG08811A(B)-SI and XVG08812A(B)-SI have been drilled with a 1/8 inch hole to provide assurance against pressure locking. This hole provides a drain path from the bonnet to the bearing block void space then through the hole to one side of the disc. A bonnet relief system has been installed on the bonnet of XVG08889-SI to eliminate postulated pressure locking conditions.

RN 99-010 99-01 RN 99-105

Where a gasket is employed for the body to bonnet joint, it is either a fully trapped, controlled compression, spiral wound gasket with provisions for seal welding, or it is of the pressure seal design with provisions for seal welding. The valve stuffing boxes are designed with a lantern ring leakoff connection with a minimum of a full set of packing below the lantern ring and a minimum of one-half of a set of packing above the lantern ring. A full set of packing is defined as a depth of packing equal to 1-1/2 times the stem diameter. The motor operator incorporates a "hammer blow" feature that allows the motor to impact the discs away from the seat upon opening or backseat upon closing. This "hammer blow" feature not only impacts the disc but allows the motor to attain its operational speed prior to impact. Valves which must function against system pressure are designed such that they function with a pressure drop equal to full system pressure across the valve disc.

RN 98-056 RN 99-010

Each of the three centrifugal charging pump minimum flow bypass lines is equipped with a motor operated globe valve. The valves are of the bolted bonnet design utilizing a spiral wound asbestos gasket with provisions for seal welding. The seating surfaces are hard faced to prevent galling and reduce wear. The valve stuffing box contains six rings of packing. During normal operation with the valves open, fluid pressure at the packing glands is 40 psig due to the presence of restricting orifices upstream of the valves. Installation of the valves is such that the charging pump discharge pressure is under the valve seat to prevent stem leakage when the valve is closed.

RN 99-010

6.3.2.2.6.2 Manual Globe, Gate, and Check Valves

Gate valves are either wedge design or parallel disc and are straight through. The wedge is either split or solid. All gate valves have backseat and outside screw and yoke.

Globe valves, "T" and "Y" style are full ported with outside screw and yoke construction.

Check valves are spring loaded lift piston types for sizes 2 inches and smaller, and swing type for sizes greater than 2 inches. Stainless steel check valves have no penetration welds other than the inlet, outlet, and bonnet. The check hinge is serviced through the bonnet.

RN 99-013

The stem packing and gasket of the stainless steel manual globe and gate valves are similar to those described above for motor operated valves. Carbon steel manual valves are employed to pass nonradioactive fluids only and therefore do not contain the double packing and seal weld provisions.

The High Head Safety Injection Throttle valve(s) is a 2-inch y-pattern globe valve that does contain valve packing. The valve design provides the required stem travel for allowing sump debris to pass through. RN 06-019

6.3.2.2.6.3 Accumulator Check Valves (Swing-disc)

The accumulator check valve is designed with a low pressure drop configuration with all operating parts contained within the body.

Design considerations and analyses which assure that leakage across the check valves located in each accumulator injection line does not impair accumulator availability are as follows:

1. During normal operation the check valves are in the closed position with a nominal differential pressure across the disc of approximately 1650 psi. Since the valves remain in this position except for testing or when called upon to open following an accident, and are therefore not subject to the abuse of flowing operation or impact loads caused by sudden flow reversal and seating, they do not experience significant wear of the moving parts, and are expected to function with minimal backleakage. This backleakage can be checked via the test connection as described in Section 6.3.4.

2. When the RCS is being pressurized during the normal plant heatup operation, the check valves are tested for leakage as soon as reactor system pressure reaches about 2000 psi. This test confirms the seating of the disc and whether or not there has been an increase in the leakage since the last test. When this test is completed the accumulator discharge line motor operated isolation valves are re-opened. There should be no increase in leakage from this point on since increasing reactor coolant pressure increases the seating force and decreases the leakage. 98-01

3. The experience derived from the check valves employed in the emergency injection systems indicate that the system is reliable and workable; check valve leakage has not been a problem. This is substantiated by the satisfactory experience obtained from operation of the Robert Emmett Ginna Station and subsequent plants where the usage of check valves is identical to this application.

4. The accumulators can accept some inleakage from the RCS without affecting availability. Inleakage would require, however, that the accumulator water volume be adjusted in accordance with Technical Specification requirements.

6.3.2.2.6.4 Relief Valves

Relief valves are installed in various sections of the ECCS to protect lines which have a lower design pressure than the RCS. Valves that are, or potentially could be, in contact with borated water are constructed with stainless steel bodies. The bonnets of valves equipped with a balancing bellows are carbon steel. Bonnets of valves which are not equipped with a balancing bellows are stainless steel. The accumulator nitrogen supply relief valve does not have a potential to contact borated water, and the body and bonnet are constructed of carbon steel. Table 6.3-2 lists the systems relief valves with their capacities and setpoints. RN 99-013

6.3.2.2.6.5 Butterfly Valves

Each main residual heat removal line has an air operated butterfly valve which is normally open and is designed to fail in the open position. The actuator is arranged such that air pressure on the diaphragm overcomes the spring force, causing the linkage to move the butterfly to the closed position. Upon loss of air pressure, the spring returns the butterfly to the open position. These valves are left in the full open position during normal operation to maximize flow from this system to the RCS during the injection mode of the ECCS operation. These valves are used during normal residual heat removal system (RHRS) operation to control cooldown flowrate.

Each residual heat removal heat exchanger bypass line has an air operated butterfly valve which is normally closed and is designed to fail closed. These valves are used during normal cooldown to avoid thermal shock to the residual heat exchanger.

Monitor lights on the control board and alarms via the Bypass Inoperable Status Indication (BISI) CRT displays in the Control Room provide indication that the valves are and remain in the correct (open) position.

6.3.2.2.7 System Operation

The operation of the ECCS, following a loss of coolant accident, can be divided into two distinct modes:

1. The injection mode in which any reactivity increase following the postulated accident is terminated; initial cooling of the core is accomplished; and coolant lost from the primary system is replenished, and

2. The recirculation mode in which long term core cooling is provided during the accident recovery period.

A discussion of these modes follows.

The principal mechanical components of the ECCS which provide core cooling immediately following a loss of coolant accident are the accumulators, the centrifugal charging pumps, and the residual heat removal pumps.

For large pipe ruptures, the RCS is depressurized and voided of coolant rapidly, and a high flowrate of emergency coolant is required to quickly cover the exposed fuel rods and limit possible core damage. This high flow is provided by the passive accumulators, followed by the charging pumps and the residual heat removal pumps discharging into the cold legs of the RCS.

Emergency cooling is provided for small ruptures primarily by high head Injection.[1]
Small ruptures are those which do not immediately depressurize the RCS below the
accumulator discharge pressure. The centrifugal charging pumps deliver borated water
at the prevailing RCS pressure to the cold legs of the RCS. During the injection mode, 00-01
the charging pumps take suction from the refueling water storage tank. Evaluation of
these small ruptures is also discussed in Sections 3.6.1.1.3 and 15.3.1.

The residual heat removal pumps take suction from the refueling water storage tank and deliver borated water to the RCS. These pumps begin to deliver water to the RCS only after the pressure has fallen below the pump shutoff head.

Core protection is afforded with the minimum engineered safety feature equipment. The minimum engineered safety feature equipment is defined by consideration of the single failure criteria as discussed in Sections 3.1, 6.3.1.4, and 6.3.2.11. The minimum design case will ensure the entire break spectrum is accounted for and core cooling design bases of Section 6.3.1 are met. The analyses for this case are presented in Sections 15.3 and 15.4.

The Technical Specifications permit various pumps of the ECCS to be inoperable during 00-01
power operation and for an additional time period while the reactor is at hot shutdown, provided that the remaining pump has been demonstrated to be operable prior to the repair. With regard to these repair times:

1. In most cases, repairs will be completed within the allowable time period.

2. If it is determined that this time period is not adequate to restore the component to the operable condition, putting the reactor in the hot shutdown condition significantly reduces the cooling requirements following a postulated loss of coolant accident.

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(1) The centrifugal charging pumps are commonly referred to as "high head pumps" and the residual heat removal pumps as "low head pumps." Likewise, the term "high head injection" is used to denote charging pump injection and "low head injection" refers to residual heat removal pump injection.

3. Failure to complete repairs within the designated time interval after going to hot shutdown is considered indicative of a requirement for major maintenance and therefore the reactor is put in the cold shutdown condition.

The injection mode of emergency core cooling is initiated by the safety injection signal. This signal is actuated by any of the following:

1. Low pressurizer pressure

2. High Reactor Building pressure

3. High differential pressure between any two steam lines

4. Low Steam line pressure

5. Manual actuation

Operation of the ECCS during the injection mode is completely automatic. The safety injection signal automatically initiates the following actions:

1. Starts the emergency diesel generators, which if all other sources of power are lost, supplies the engineered safety feature buses.

2. Starts the charging pumps and the residual heat removal pumps.

3. Aligns the charging pumps for injection by:

    a. Closing the valves in the charging pump discharge line to the normal charging line.

    b. Opening the high head injection isolation valves.

    c. Opening the valves in the charging pumps suction line from the refueling water storage tank. 00-01

    d. Closing the valves in the charging pump normal suction line from the volume control tank.

Remotely operated valves for the injection mode which are under manual control (i.e., valves which normally are in their ready position and do not require a safety injection signal) have their positions indicated on a common portion of the control board. If a component is out of its proper position, its monitor light will indicate this on the control board. At any time during operation when one of these valves is not in the ready position for injection, this condition is alarmed and shown visually on the board.

Process instrumentation available to the operator in the control room to provide him with appropriate and useful information during the automatic injection mode of operation are listed in Section 7.5.

The injection mode continues until the residual heat removal pumps have been realigned to the recirculation mode. During the injection mode all pumps take suction from the refueling water storage tank until a lo-lo level signal from the refueling water storage tank aligns the residual heat removal pumps to take suction from the reactor building sump. This signal also actuates an alarm to alert the operator to complete the realignment of the ECCS to the recirculation mode. The switchover procedure that must be performed by the operator is given in Table 6.3-3.

After the injection operation, water collected in the reactor building sump is cooled and returned to the RCS via the low/high head recirculation flow paths. The residual heat removal pumps are aligned to take suction from the reactor building sump and to deliver directly to the RCS and to supply suction to the charging pumps. The charging pumps deliver flow directly to the RCS cold legs. This latter mode of operation assures flow in the event of a small rupture where the depressurization proceeds more slowly such that the RCS pressure is still in excess of the shutoff head of the residual heat removal pumps at the onset of recirculation.

The maximum predicted water level elevation inside the Reactor Building following a postulated large break LOCA is elevation 419.524'. This height was calculated based upon the following: 00-01

1. The total amount of water available to flood the bottom of the Reactor Building is conservatively assumed to include:

   a. Water in the refueling water storage tank, from the maximum full level to the empty level alarm, plus an instrument accuracy allowance. It allows 96.7% of the volume of the RWST to drain to the Reactor Building basement floor. 00-01

   b. The entire water inventory of the safety injection accumulators.

   c. The entire water inventory of the sodium hydroxide storage tank.

   d. The inventory of the reactor coolant system above the level of the reactor vessel nozzles.

2. The effective height of the Reactor Building basement floor was determined by weighing the heights of the various portions of the floor by the area of each portion.

3. The volume of all significant structures and equipment that reduces the free volume above the basement floor is explicitly accounted for in the calculations.

4. The volumes of the sumps and the incore instrumentation pit, which would fill with water following a LOCA and, thus reduce the volume of the water above the basement floor, are explicitly accounted for in the calculations.

As part of a Cycle 12 specific evaluation, an additional flood elevation was determined for LOCA with no single failures. The Reactor Building flood elevation was recalculated at 418.883' (418' 10.956") for this case. This flood level was calculated using operator time lines for swap-over. This no single failure case is also analyzed with the RWST filled to the overflow as an initial condition. 00-01

A calculation was also performed for a double ended main steam line break which resulted in a predicted flood level below the LOCA level. This calculation was performed using the same methodology and assumptions used for the LOCA calculation described above, except that the inventory of the safety injection accumulators and the inventory of the reactor coolant system above the reactor vessel nozzles was not included. 00-01

The location of equipment inside the Reactor Building has been reviewed to identify safety grade and associated non-safety grade equipment that would be submerged as a result of an accident.

The review resulted in the identification of valves and instrumentation that would be below a design value for the flood level of 420'0". Submerged instrumentation identified as a result of the review is in one of the following categories: 02-01

1. Instrumentation is non-safety grade but is associated with safety grade equipment. Fuses and/or circuit breakers provide sufficient protection to prevent adverse consequences to the safety grade equipment. Instruments in this category include level, pressure, and temperature monitoring devices that are used for normal control purposes and do not have any protection system functions.

2. The instrumentation is safety grade and is submerged while required to operate. Instrumentation in this category has been qualified as discussed in revised Section 3.11.1.2. Instruments in this category include post accident monitoring instrumentation associated with the reactor building RHR pump sumps and reactor building spray pump sumps (recirculation sumps).

Flooding outside the Reactor Building as a result of an accident will not affect safety grade equipment since protection is provided as follows:

1. Curbs are located at entrances to cubicles housing safety grade equipment.

2. Leakage detection systems, described in Section 7.6.5, are provided.

For a discussion of flooding outside the Reactor Building as a result of site flooding, see Section 3.4.

The location of valves inside the Reactor Building has been reviewed to identify safety grade and associated nonsafety grade valves flooded as a result of an accident. The review resulted in identification of the valves discussed below which have some portion of the electrical circuits associated with the valve operators below the design flood level. 02-01
Features that preclude adverse safety consequences or adverse consequences follows:

1. Valve, Tag No. HCV00137 00-01

   This valve, which is associated nonsafety grade equipment, is used for letdown flow control. Fuses provide sufficient protection to prevent adverse consequences to safety grade equipment. The electrical circuitry associated with this valve consists of a 4 to 20 ma signal transmitted from a Westinghouse Remote/Manual Setpoint Station which is connected to a 45 volt d-c power supply. This 45 volt d-c power is protected by a 1/2 ampere fuse on the load side of the power supply. The circuitry is not "protection grade" and serves only a "control function." Loss of this circuit does not jeopardize plant safety.

2. Valve, Tag No. LCV01003 00-01

   This valve, which is associated with nonsafety grade equipment, is used for level control in the reactor coolant drain tank. It is provided with a positioning signal from the waste processing panel. The electrical circuitry associated with this valve 98-01
   is not "protection grade" and serves only a "control function." Loss of this circuit does not jeopardize plant safety.

3. Valve, Tag No. XVM08143 00-01

   This valve, which is associated with nonsafety grade equipment, is used to direct excess letdown flow to either the volume control tank or the reactor coolant drain tank. Fuses provide sufficient protection to prevent adverse consequences to safety grade equipment. Electrical circuits for this valve are illustrated by GAI Drawing B-208-021, Sheet CS79, submitted as part of the Wiring and Schematic Package (see also Section 1.7).

4. Valves, Tag Nos. XVT08153 and 8154 00-01

   These valves, which are located in series, are used for redundant isolation of the reactor coolant system from the excess letdown heat exchangers. These valves are closed during normal plant operation, except as an alternate Letdown Flow path if normal Letdown has been isolated. These valves are open only during startup. Fuses provide sufficient protection to prevent spurious operation after flooding since limit switches are not used for seal-in or interlock functions. Electrical circuits for these valves are illustrated by GAI Drawings B-208-021,

Sheet CS87 (valve 8153) and Sheet CS88 (valve 8154) submitted as part of the Wiring and Schematic Package (see also Section 1.7). 00-01

5. Valve, Tag No. XVG08701A 00-01

Safety functions of this valve are isolation of residual heat removal system from the reactor coolant system, containment isolation, and interlock of external limit switch to valve 8706A. Performance of these isolation functions are assured by the institution of administrative controls to close the valve, and open the circuit breaker 00-01 at the motor control center during power operation. The external limit switch, which is interlocked to valve 8706A, is above the design flood level. 02-01

6. Valve, Tag No. XVG08701B 00-01

Safety function of this valve are isolation of residual heat removal system from the reactor coolant system, containment isolation, and interlock of external limit switch to valve 8706B. Performance of the isolation functions are assured by the institution of administrative controls to close the valve and open the circuit breaker at the motor control center during power operation. The external limit switch, which is interlocked to valve 8706B, is above the design flood level. 02-01

7. Valve, Tag No. XVG08702A 00-01

Safety functions of this valve are isolation of residual heat removal to reactor coolant system and interlock of external limit switch to valve 8706A. Performance of the isolation function is assured by the institution of administrative controls to close the valve and open the circuit breaker at the motor control center during power operation. The external limit switch, which is interlocked to valve 8706A, is moved above the design flood level. 02-01

8. Valve, Tag No. XVG08702B

Safety functions of this valve are isolation of residual heat removal system from the reactor coolant system and interlock of external limit switch to valve 8706B. 00-01 Performance of the isolation function is assured by the institution of administrative controls to close the valve and open the circuit breaker at the motor control center during power operation. The external limit switch, which is interlocked to valve 8706B, is above the design flood level. 02-01

9. Valves, Tag No. LCV00459 and 460

These valves, which are located in series, are used for redundant isolation of the reactor coolant system letdown. These valves are open during normal plant operation. They do not get an automatic closure signal from the reactor protection system and are not required to close in an accident condition. They are fail closed valves and are not required to function in accident conditions. Isolation of the RCS

is provided by the Letdown Isolation valves, XVT08149A, B, and C and XVT08152. Fuses provide sufficient protection to prevent the valves' controls from adversely affecting any other components.

Within 8 hours following the loss of coolant accident, simultaneous hot leg and cold leg recirculation will be initiated to avoid boron precipitation in the core and to terminate boiling. RN 02-039

Core cooling during recirculation can be maintained by the flow from one residual heat removal pump if RCS pressure is low. If RCS pressure remains high, one charging pump operating in series with one residual heat removal pump provides sufficient additional head.

Initially the recirculated water is at a slightly subcooled condition. The residual heat removal pumps deliver this sump water to the RCS via the residual heat exchangers which subcool the water further. When the water is delivered to the reactor vessel, the majority of the subcooled water spills back to the sump through the broken RCS loop. The system pressure at the time recirculation is initiated is dependent on the initiating break size. Heat removal is accomplished from the recirculated sump water by operation of one or both of the residual heat exchangers.

The ECCS recirculation loop piping and components external to the Reactor Building are surrounded by shielding. This shielding is designed to permit access for maintenance to a component such as a pump while the redundant component is recirculating sump fluid.

Pressure relieving devices, from portions of the ECCS located outside the Reactor Building which might contain radioactivity, discharge to the recycle holdup tanks.

An analysis has been performed to evaluate the radiological effects of recirculation loop leakage. A discussion of the analysis and releases is provided in Section 15.4.1.4.2. A 00-01
loop is assumed to include a centrifugal charging pump, a residual heat removal pump, a residual heat exchanger, and the associated piping. Thus, two separate loops are provided, each of which is adequate for core cooling.

Since redundant flow paths are provided during recirculation, a leaking component in one of the flow paths may be isolated. This action curtails any further leakage and renders the component available for corrective maintenance. Maximum potential external recirculation loop leakage is given in Table 6.3-4.

During recirculation, significant margin exists between the design and operating conditions (in terms of pressure and temperature) of the ECCS components.

6.3.2.3 Applicable Codes and Classifications

The codes and standards to which the individual components of the ECCS are designed are listed in Table 3.2-1.

6.3.2.4 Materials Specifications and Compatibility

Materials employed for components of the ECCS are given in Table 6.3-5. Materials are selected to meet the applicable material requirements of the codes in Table 3.2-1 and the following additional requirements:

1. All parts of components in contact with borated water are fabricated of or clad with austenitic stainless steel or equivalent corrosion resistant material.

2. All parts of components in contact (internal) with sump solution during recirculation are fabricated of austenitic stainless steel or equivalent corrosion resistant material. A soft seat material that is approved for system chemistry and design requirements is used in selected manual valves to minimize potential leakage. RN 10-010

3. Valve seating surfaces are hard faced with Stellite Number 6 or equivalent to prevent galling and to reduce wear.

4. Valve stem materials are selected for their corrosion resistance, high tensile properties, and resistance to surface scoring by the packing.

No cold worked austenitic stainless steel having a yield strength greater than 90,000 psi is used in the ECCS.

The elevated temperature of the sump solution during recirculation is well within the design temperature of all ECCS components. In addition, consideration has been given to the potential for corrosion of various types of metals exposed to the fluid conditions prevalent immediately after the accident or during long term recirculation operations.

Environmental testing of ECCS equipment inside the Reactor Building, which is required to operate following a loss of coolant accident, is discussed in Section 3.11. These components were qualified to function following exposure to a chemical spray solution equivalent to the Post LOCA spray described in Section 3.11.5.1.2. The environmental qualification test results of the engineered safety feature equipment indicate that this equipment will operate satisfactorily during and following exposure to the combined Reactor Building post accident environments of temperature, pressure, chemistry and radiation. 00-01

In addition to environmental testing of ECCS equipment, discussed in Section 3.11, the integrity of the materials of construction of engineered safety features equipment when exposed to post-design basis accident (DBA) conditions has been evaluated. Post-DBA conditions were conservatively represented by test conditions. The test program

(Reference [2]) performed by Westinghouse considered spray and core cooling solutions of the design chemical compositions, as well as the design chemical compositions contaminated with corrosion and deterioration products which may be transferred to the solution during recirculation. The effects of sodium (free caustic), chlorine (chloride), and fluorine (fluoride) on austenitic stainless steels were considered. Based on the results of this investigation, as well as testing by Oak Ridge National Laboratory (ORNL) and others, the behavior of austenitic stainless steels in the post-DBA environment will be acceptable. No cracking is anticipated on any equipment even in the presence of postulated levels of contaminants, provided the core cooling and spray solution pH is maintained at an adequate level. The inhibitive properties of alkalinity (hydroxyl ion) against chloride cracking and the inhibitive characteristic of boric acid on fluoride cracking have been demonstrated.

6.3.2.5 Design Pressures and Temperatures

The component design pressure and temperature conditions are given in Table 6.3-1. These pressure and temperature conditions are specified as the most severe conditions to which each respective component (including the piping) is exposed. For each component, these conditions are considered in relation to the code to which it is designed. By designing the components in accordance with applicable codes (see Section 3.2) and with due consideration for the design and operation conditions, the fundamental assurance of structural integrity of the ECCS components is maintained.

6.3.2.6 Coolant Quantity

A minimum volume is required to be stored in the refueling water storage tank (see Table 6.3-6) to ensure that after RCS pipe ruptures, sufficient water will be injected to satisfy LOCA requirements and then be available within the Reactor Building to permit recirculation of cooling flow to the core, and to meet the net positive suction head requirements of the charging, residual heat removal pumps, and RB spray pumps. Also, sufficient water must be available for the prevention of RWST vortex formation, especially during the transition from cold leg injection to cold leg recirculation.

The sizing design bases for the refueling water storage tank (RWST) include the following design allowances: 02-01

a. A minimum volume of water (350,000 gallons) is maintained above the nominal lo-lo level semi-automatic switchover setpoint. This amount is more than sufficient to satisfy the injection requirements for LOCA. The RWST is designed to be kept at the "normal full" level during normal power operation. The volume of water in the RWST between the nominal "normal full" level and the nominal "lo-lo" level semi-automatic switchover setpoint is 387,245 gallons. This provides a margin of 37,245 gallons as compared to the 350,000 gallons.

b. The normal and accident errors for the RWST level instrumentation are defined in the referenced design calculation DC00040-077 - Design Basis Timelines for Completing Transfer From Injection to Re-circulation Phase During LOCA. RN 06-029

c. For normal working allowance, the operator has at least 5,320 gallons between the maximum full level and the minimum full level (lo level), with normal instrument error conservatively combined. 02-01

d. The volume of water available in the RWST below the lo-lo level semi-automatic switchover setpoint, with the instrument error is defined in the referenced design calculation DC00040-077.

e. Refer to design calculation DC00040-077 for a discussion of the volume of water that will be removed from the tank during automatic and manual switchover and manual operations with no failures and with the worst single failure (failure of 8809A(B) to close on demand) is the limiting single failure when appropriate operator actions to mitigate RWST outflow are considered. RN 06-029

f. The unusable volume in the RWST, from the tank bottom to the critical vortex level is 38,671 gallons. However, 10,266 gallons of the 38,671 gallons are above the top of the outlet nozzle.

In addition to the design allowances outlined above, the design of RWST includes an anti-vortex baffle assembly (see Figure 6.3-7) which consists of a horizontal plate supported over the outlet nozzle. Welded to the end of the horizontal plate is a vertical plate extending downward to the tank bottom. The water must enter the two open sides of the "box" formed by the two plates and the tank, and make a 90-degree turn before exiting into the outlet pipe. The baffle assembly greatly reduces the tank level where air-inducing vortices begin to form. To show the effectiveness of the baffle, tests were performed on a ¼ scale model of the RWST. The results of the tests show that critical vortex levels are a linear function with total RWST outflow. At the maximum RWST outflow of 14,575 gpm, the critical vortex level is 5% RWST level, which is ~ 1 foot above the top of the outlet pipe and is coincident with the top of the baffle assembly. 02-01

Maximum total RWST outflow during injection mode operation, for LBLOCA, is 14,575 gpm and, for SBLOCA is 6975 gpm (975 gpm for two CHG pumps and 3000 gpm/pump for two RB Spray pumps). Given the minimum volume of 350,000 gallons above the lo-lo setpoint, the shortest times from LOCA initiation to receipt of the lo-lo semi-automatic switchover signal are approximately 24 minutes and 50 minutes for LB and SBLOCA respectively.

The switchover portion of the ECCS system features a semi-automatic scheme. The "lo-lo" level semi-automatic switchover signal in the RWST, coincident with a safety injection signal causes the RHR and Reactor Building spray suction valves from the Reactor Building recirculation sumps to automatically open (valves 3004A(B), 3005A(B), 8811A(B), and 8812A(B)). The RWST suction isolation valves (8809A(B) and 3001A(B) auto close on full open signals from the corresponding recirculation sump suction valves. To ensure that ECCS pumps and RB spray pumps have a suction source, the operator must complete actions to accomplish the transition from injection to recirculation mode prior to the depletion of the transfer allowance. The required operator actions are detailed in Table 6.3-3. The time available for switchover is dependent upon the flow rate out of the RWST as the switchover actions are performed. In order to provide conservative time allowances for the completion of these actions, the following are assumed: 02-01 RN 06-029 02-01

1. The RWST volume available for switchover is defined as the transfer allowance.

2. The operator will secure all pumps taking suction from the RWST at 6% RWST level plus uncertainty. RN 06-029

3. Containment and RCS pressures for large break conditions are assumed to be 0 psig. Thus, no credit is taken for the reduction in RWST outflow that will result with higher containment and RCS pressures following a large break.

   The same conservative assumption is made for the small break conditions (except that RCS pressure is assumed to be greater than RHR pump shutoff head resulting in no RHR pump flow to the RCS for small break conditions). Because the RHR pump is deadheaded during the small break, backflow from the RWST to the sump will occur when the RHR sump suction valves open at the lo-lo semi-automatic switchover setpoint when 0 psig RB pressure is assumed. This effect is considered in the timeline development. 02-01

4. Flow out of the RWST during switchover includes allowances for both pumped flow to the RCS and containment and backflow to the Reactor Building recirculation sumps based upon the 0 psig containment pressure assumption. Pump flow rates are assumed to be constant during switchover and include the following conservative flow rates:

   (a) Charging pump - 975 gpm for two pumps and 688 gpm for one pump

   (b) RHR pump - 7600 gpm for two pumps and 4500 gpm for one pump

(c) Reactor Building Spray Pump – 3000 gpm per pump

Backflow to the Reactor Building recirculation sumps will occur during ECCS switchover based upon the 0 psig containment pressure assumption if the RWST suction and Sump suction valves are both open and the corresponding RHR or RB Spray pump is secured (or RHR pump is deadheaded in a SBLOCA). Backflow will vary as the switchover proceeds, depending upon ECCS alignment and postulated failures. 02-01

5. Flow rates out of the RWST for both the no failure condition and the worst single failure condition vary as the switchover proceeds. The RWST outflow vs. time is calculated and used to determine limiting allowed timelines for the completion of switchover before the transfer allowance is depleted.

Based upon the above criteria, the flow rates out of the RWST as a function of switchover completion are detailed in the referenced calculation DC00040-077. The large break cases are limiting with respect to the targeted timelines. The timelines presented provide sufficient time for the operator to successfully complete the switchover. It may be necessary to secure pump(s) when it is determined that the suction(s) cannot be transferred from the RWST. This conserves RWST water for the completion of the switchover in the event of a failure. RN 06-029

6.3.2.6.1 Containment Sump and its Effect on Long Term Cooling Following a LOCA

The Virgil C. Summer Nuclear Station Technical Specifications require that each Emergency Core Cooling System (ECCS) subsystem be demonstrated operable by a visual inspection. This visual inspection verifies that no loose debris is present in the Reactor Building which could be transported to the RHR and RB Spray sumps to cause restriction of their pump suctions. This inspection is performed immediately: 00-01

1. For all accessible areas of the Reactor Building prior to establishing CONTAINMENT INTEGRITY, and

2. In the areas affected within the Reactor Building at the completion of each Reactor Building entry when CONTAINMENT INTEGRITY is established. 00-01

Station Surveillance Test Procedures cover these requirements.

The Virgil C. Summer Nuclear Station Technical Specifications require a visual inspection of each containment sump and subsystem suction inlet at least once every 18 months. Station Surveillance Test Procedures have been written to satisfy these requirements. The suction inlets and sump components are inspected to ensure that debris is not present and that no evidence exists of structural distress or abnormal corrosion. 00-01

An instruction dealing with possible ECCS sump blockage is contained in the Virgil C. Summer Nuclear Station Emergency Operating Procedures (EOP-2.2). This procedure addressed long term cooling and requires the operator to monitor pump discharge pressure and motor amps when in the sump recirculation mode. It also cautions the operator that if pump discharge pressure or amps decrease, the cause may be sump blockage. If there is an indication of sump blockage, the operator is instructed to close the associated bypass valve and throttle the outlet valve to stabilize pump amps. If suction is completely lost, the operator is instructed to stop the affected pump. If at least one flow path from RHR sumps to RCS can not be established or maintained, the operator is directed to EOP-2.4 Loss of Emergency Coolant Recirculation. 00-01

Two separate redundant ECCS Sumps, Sump A and Sump B, are provided for the Virgil C. Summer Nuclear Station. Sump A and Sump B each includes a separate deep sump pit for RHR and for RB spray. The strainer assemblies located over each deep sump pit and the cross duct between the RHR and RB Spray header boxes within each sump are described in Section 6.2.2.2.1.1. FSAR Figure 1.2-4 shows the physical separation between these sumps and the Reactor Building drain sump. The Reactor Building drainage system as well as the Refueling Canal drain are located such that this drainage water flow does not directly impinge in the sump areas. There are no high energy lines in the areas which will subject the sumps to postulated pipe whip or jet impingement loads and cause subsequent damage.

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There is also a 6 inch high toe plate around the perimeter of the mezzanine and operating floors that prevents water from cascading down through the space between the floor perimeter and the containment wall to the sumps below. Water on the operating floor at 463’ will tend to go down through the stair wells for Stairways A and B. At the Stairway A on mezzanine floor at 436’, a 6 inch high toe plate incorporated into a hinged gate is located across the head of Stairway A. The gate is in the closed position during operation and the toe plate on the bottom of the gate diverts water flowing across the elevation 436’ floor from entering Stairway A and falling near Sump A at the 412’ basement floor level. Water flowing across the 436’ floor falls to the sump level at the 412’ floor either through the Stairway B or through two 8 foot long openings provided through the floor perimeter toe plate and which are located around the containment perimeter away from the ECCS Sumps A and B.

Sump A is surrounded on four sides and Sump B on three sides by a continuous 6 inch high curb at the basement floor level 412'. The fourth side of Sump B is bounded by a concrete secondary compartment wall. The floor level within the sumps is at elevation 408'. The sump strainer assemblies located at the 408' level preclude debris entry down into the deep sump pits and into the recirculation piping. The LOCA condition water level when recirculation begins is approximately 6 to 7 feet above the general sump floor at elevation 408'. A horizontal grating with top elevation 412'-9" is provided over the sumps. The 6 inch curb and grating serve to limit debris and trash from transport down to the sump strainers. An engineered opening is provided through the grating to provide sufficient flow into the sumps assuming a postulated worst case where the horizontal grating surface is covered by debris following a postulated LOCA. The suction strainer assembly design is described in Section 6.2.2.2.1.1. The strainer openings are limited to 1/16 inch diameter holes to limit the size and quantity of debris permitted to pass through the strainers. The strainer fins and header box are designed with openings (maximum 1/16 inch hole size) that provide venting for air in the system as the sump fills. A hinged and bolted hatch cover is provided on the top of each of the four sump strainer header boxes located over the deep sump pits to permit personnel entry to the deep sump pit for inspection purposes. Components of the ECCS sumps are seismically designed for Seismic Category 1 conditions and are constructed of stainless steel.

In addition to the curbing around the sumps and grating over the sumps which serve to limit trash and debris transport to the sump strainers at the 408' level, two partitions with integral gates consisting of welded stainless steel bars are provided at the 412' floor level, one on either side of the area in which Sump A and Sump B are located. These partitions have bar spacing approximately 8 inches by 8 inches maximum and serve to stop postulated large trash items inadvertently left in the Reactor Building following a refueling outage from transporting to the sump strainers. The partitions are a minimum of approximately 7 feet high and span across the perimeter circumferential corridor between the secondary compartment concrete wall and the Reactor Building inside surface. A separation space is provided between the partitions and the Reactor Building inside surface to ensure no interaction under design base conditions including seismic. The welded stainless steel bar gates and partitions are designed for Seismic Category 1 conditions. The gates are maintained in the closed position when the plant is in operation. RN 13-022

Analysis and testing have been completed to evaluate sump strainer blockage and head loss in response to Generic Letter 2004-02. The analysis is comprised of five steps:

1. Debris Generation covers LOCA jet impact on insulation and Level 1 coatings, latent debris (dust and dirt on containment surfaces), coatings, labeling and placards which may disbond during design basis events and containment Foreign Material Exclusion which are identified during containment closeout inspection.

2. Debris Transport determines the fraction of debris (based on type, size and location) that may transport to the sump strainer. The analysis uses Computational Fluid Dynamics (CFD) modeling to determine post-LOCA sump flow velocities and experimental data on debris transport velocities.

3. Debris Accumulation combines the Generation and Transport analysis to develop debris loading cases for the strainer which includes type and quantity of debris.

4. Chemical Effects analysis which determines the quantity and time-line for aluminum corrosion precipitant entering the sump fluid and adding to the debris load.

5. Debris Headloss determines the pressure drop across the strainer screen using the strainer surface area, debris load and expected fluid velocity. For V. C. Summer Nuclear Station (VCSNS) the head loss is based on small scale chemical effects testing covered in Technical Report TR04560-004 (Reference [3]).

Three (3) types of insulation are used inside the Reactor Building: all stainless steel reflective insulation, encapsulated Marinite XL, and TempMat. The Marinite XL and TempMat are used in limited applications and are encased in stainless steel.

The stainless steel reflective insulation is used on most piping and major components within the Reactor Building such as the steam generators, pressurizer and reactor vessel. If impacted by a LOCA jet, the stainless steel insulation cassettes may detach and break apart when the cassette hits the floor. If a cassette breaks open, the inner stainless foil is released. The foil and cassette body sink to the bottom of the containment pool. Testing and analysis have demonstrated that the stainless steel insulation debris does not readily transport (tumble along the floor) for the recirculation flow velocities calculated at VCSNS. Only a small amount is conservatively predicted to reach the sump pit and would not present a sump blockage concern. Large scale testing of the sump strainer was also completed with the test strainer assembly completely covered in stainless steel insulation debris. The strainer head loss was well within acceptable limits and did not provide a design limiting sump strainer blockage case.

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The Marinite XL is a mass type insulation used only around the reactor pressure vessel loop nozzles and portions of the reactor coolant pipe that penetrate the primary shield wall. The Marinite XL insulation cassettes around the nozzles serve a dual purpose for the VCSNS design. The first is thermal insulation. The second is a robust barrier around the nozzle to limit LOCA jet expansion into the reactor vessel cavity and pressurization of the primary shield wall. The insulation cassettes are Marinite XL machined to fit the nozzle taper and encased in 1/4 inch stainless steel. The cassettes are specifically designed to withstand the LOCA jet forces from a double ended guillotine break at the reactor vessel nozzle safe end. Due to this robust design, Marinite XL around the reactor vessel nozzles is not postulated to become LOCA generated debris.

A 1 inch thickness of Marinite XL is also used around the primary loop piping as it passes through the primary shield wall. For a break within the primary shield wall, the Marinite XL insulation is assumed to be fully particulated by the LOCA jet and mechanical damage from pipe whip within the wall. In particulate form the Marinite XL is 100 percent transportable to the sump strainer. Only the Marinite XL insulation around one pipe is postulated to become debris since each pipe is physically separated within the primary shield wall. The limiting break location for Marinite XL debris load is the loop C cold leg nozzle which has the largest volume of Marinite XL.

TempMat fibrous insulation is used on pressurizer ventilation ducts and steam generator level instrument lines in the loop A and loop C compartments. The TempMat insulating material is encased with stainless steel in each location. The three locations are in separate rooms such that only one location can be impacted by a LOCA jet. Debris Generation and Debris Transport calculations determine the limiting TempMat debris load is from the A loop compartment with a break at the steam generator outlet on the 31 inch cross-over leg.

In response to an NRC RAI, the limiting debris load cases for Marinite XL and for TempMat were tested to determine the limiting debris load for strainer head loss. Tests were carried out on a small scale test loop and demonstrated that the Marinite XL debris load is limiting by a substantial margin.

Other postulated debris sources include: RN 13-022

1. Rubber boots for reactor nozzle and support feet ventilation (six total).

2. Fifteen (15) fibrous reinforced silicon rubber enclosures which provide forced air cooling of equipment inside the pressurizer cubicle.

3. Rubber expansion joints in the ring header duct.

4. Kaowool wrapping (with stainless steel encasement sealed on edges with an elastomer sealant) for electrical conduit to meet separation criteria of electrical circuits for fire protection.

5. M-Board (with stainless steel encasement sealed on edges with an elastomer sealant) for electrical conduit and cable trays to meet separation criteria of electrical circuits for fire protection.

6. Unqualified labels and placards.

7. Failed and Unqualified coatings.

8. Latent Debris.

Two rubber boots on the reactor nozzle and support feet ventilation are located in each of the three loop compartments. The boots are bolted to the primary shield wall and clamped to the primary loop piping. The sump strainer analysis assumes two of the boots are dislodged by the LOCA jet and transported (100 percent) to the sump strainer. The analysis models the rubber boot debris load as a loss of surface area of the strainer. The rubber enclosures in the pressurizer cubicle are within the pressurizer cubicle and not subject to LOCA jet impingement from a loop compartment. The surface area of the rubber enclosures in the pressurizer compartment are bounded by the two rubber boots within a loop compartment, so the pressurizer rubber boots are not a limiting debris source. The ductwork expansion joints are located high in the Reactor Building along the perimeter. These are not subject to LOCA jet impingement and are not a sump strainer debris source.

The Kaowool is totally enclosed by a stainless steel corrugated tube which is split along the axial direction and installed over the Kaowool. The split tube is held in place by half inch wide banding straps. Strap spacing is 4 inches in the axial direction except at conduit supports. The ends of the encasement are sealed by an elastomer sealant to preclude entrance of any spray fluid into the encasement. The encasement assembly will also withstand seismic inputs to the installation without loss of structural integrity or sealing function. All Kaowool is located outside the Secondary Shield wall and are not subject to LOCA jet impingement. Kaowool does not enter the sump debris pool.

The M-Board installations are designed with total encasement of each M-Board by stainless steel sheets with the edges sealed by overlap of the sheets or angle clamps to preclude entrance of spray fluid. The assemblies of the various M-Board shapes are joined by angles and channels by bolting for structural strength and rigidity. The various sections which make up particular fire barrier are seismically qualified to remain in place. All of the M-Board installations are above post-LOCA sump fluid levels. One M-Board installation is located within the secondary shield wall, directly under the pressurizer. This location is only subject to LOCA jet impingement from a pressurizer surge line break. The M-Board fire barrier at this location is relative small and bounded by the Marinite XL debris load case.

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A small amount of unqualified labels and placards were identified in response to Generic Letter 2004-02. These include items such as fire zone floor labeling, communication device labeling and similar material not qualified for design basis LOCA conditions. This material is assumed to disbond and transport to the sump strainer. The material is modeled as a loss of sump strainer surface area.

Failed coating assumptions for debris loading were made to provide operating margin for the Level 1 coating program. Unqualified coatings (non-Level 1 coatings) were also identified as a part of the Generic Letter 2004-02 response. All particulate is assumed to be 100 percent transportable to the sump strainer. Paint chips were a part of the Debris Transport calculation. Paint chips that arrived at the sump strainer were modeled as a loss of strainer surface area in the analysis.

Latent debris (dust and dirt) is based on containment housekeeping. Walk downs and sampling were completed in response to Generic Letter 2004-02 to develop a latent debris load. This debris is 100 percent transportable to the containment sump strainer.

Debris which washes down to the sump and passes through the strainer is evaluated for both blockage and erosion of downstream systems and components. The size of debris is limited by the 1/16 inch round holes in the perforated plate strainer. System and component downstream effects are analyzed based on methodology in WCAP-16406, Rev. 1 (Reference [4]). The downstream effect on potential core blockage is a continuing industry effort with Virgil C. Summer Nuclear Station resolution plans specifically identified to the NRC in SCE&G letter RC-13-0006 from T. D. Gatlin to the Document Control Desk, Virgil C. Summer Nuclear Station (VCSNS) Unit 1, Path Forward for Resolution of Generic Safety Issue (GSI)-191.

The design basis head loss through the sump strainers was determined via testing on the small scale chemical effects test loops by the strainer vendor, Atomic Energy of Canada, Ltd. The test featured small scale corregated fins representative of those installed, debris addition tank and chemical additions. The test report is presented in Technical Report TR04650-004.

The analysis and testing was summarized and submitted to the NRC for review and approval. The submittals, NRC response and Request for Additional Information (RAI) are documented in the following correspondence: RN 13-022

1. SCE&G Letter RC-08-0031 (ADAMS Accession No. ML080640545) from Jeffrey B. Archie to Document Control Desk dated February 29, 2008, "Supplemental Response to NRC Generic Letter 2004-02: Potential Impact of Debris Blockage on Emergency Recirculation during Design Basis Accidents at Pressurized-Water Reactors."

2. NRC Letter (ADAMS Accession No. ML090270927) from Robert E. Martin to Jeffrey B. Archie dated February 3, 2009, "V. C. Summer Nuclear Station - Request for Additional Information for Generic Letter 2004-02 (TAC NO. MC4721)."

3. SCE&G Letter RC-09-0134 (ADAMS Accession No. ML093360336) from Jeffrey B. Archie to Document Control Desk dated November 29, 2009, "Response to Request for Additional Information for Generic Letter 2004-02."

4. SCE&G Letter RC-10-0165 (ADAMS Accession No. ML103610171) from T. D. Gatlin to Document Control Desk dated December 17, 2010, "Follow-up Response to Request for Additional Information for Generic Letter 2004-02."

5. NRC Meeting Notes - "Summary of September 14, 2009 Public Conference Call to Discuss Responses to Generic Letter 2004-02 Requests for Additional Information (TAC NO. MC4721) (ADAMS Accession No. ML093000573).

6. NRC Memorandum E. L. Geiger to M. L. Scott, "Staff Review of WCAP-16571-P Referenced in Virgil C. Summer's GL 2004-02 Supplemental Response for Downstream Effects Evaluation of Components," May 17, 2010 (ADAMS Accession No. ML100920035). RN 13-022

RHR and RB Spray pump flow beyond rated runout is another condition which requires evaluation. Due to the different known and unknown conditions of operation, this evaluation is more important for the RHR system. Therefore, full scale tests of the RHR pumps were performed at the Virgil C. Summer Nuclear Station for the different modes of injection and recirculation. As a result of these tests, flow restricting orifices have been installed at the outlet of each heat exchanger. A retest of the RHR system will be performed to insure the adequacy of these orifices.

A full scale test of the RB Spray discharge ring headers and spray nozzles is not feasible. However, this system is only subjected to two worst case operating conditions, injection and recirculation, as its design basis. Full scale tests have been performed using the suction and recirculation test returns to the RWST. Test data on the suction piping has been compared to the results of the design calculations as a measure of their accuracy. A detailed analysis was then performed of the Reactor Building Spray System's calculations to ensure that flow rates beyond runout are not possible with subsequent pump damage.

6.3.2.7 Pump Characteristics

Design parameters for the ECCS pumps are given in Table 6.3-1. Pump characteristic curves are shown in Figures 6.3-3 and 6.3-4a to 6.3-4c, see Section 5.5.7 for residual heat removal component data.

6.3.2.8 Heat Exchanger Characteristics

The design basis for the residual heat exchanger is discussed in Section 5.5.7. Design parameters appear in Table 5.5-8. The design is based on a normal cooldown rather than the recirculation phase of ECCS operation following a loss of coolant accident. This is due to the relatively small ΔT that exists on the tube side of the heat exchanger during the latter part of normal cooldown. Moreover, no credit is taken for the residual heat removal heat exchanger with respect to core cooling. RN 02-020

6.3.2.9 Emergency Core Cooling System Flow Diagrams

Piping and instrumentation and process flow diagrams of the ECCS are given in Figure 6.3-1 (Sheets 1 through 3) and Figure 6.3-2, respectively.

6.3.2.10 Relief Valves

The ECCS relief valves, their capacities, and settings are given in Table 6.3-2.

6.3.2.11 System Reliability

1. Definitions of Terms

   a. Period of Recovery

      The time necessary to bring the plant to a cold shutdown and regain access to faulted equipment. The recovery period is the sum of the short and long term periods defined below.

   b. Incident

      Any natural or accidental event of infrequent occurrence and its related consequences which affect the plant operation and require the use of engineered safeguards systems. Such events, which are analyzed independently and are not assumed to occur simultaneously, include the loss of coolant accident, steam line ruptures, steam generator tube ruptures, etc. A loss of offsite power may be an isolated occurrence or may be concurrent with any event requiring engineered safeguards systems use.

   c. Short Term

      The time immediately following the incident during which automatic actions are performed, system responses are checked, type of incident is identified, and preparations for long term recovery operation are made. The short term is the injection phase for loss of coolant accidents and is the first 24 hours following initiation of the event for all others.

   d. Long Term

      The remainder of the recovery period following the short term. In comparison with the short term where the main concern is to prevent or limit site release, the long term period of operation involves bringing the plant to cold shutdown conditions where access to the Reactor Building can be gained and repair effected.

   e. Active Failure

      The failure of a powered component such as a piece of mechanical equipment, component of the electrical supply system or instrumentation and control equipment to act on command to perform its design function. Examples include the failure of a motor operated valve to move to its correct position, the failure of an electrical breaker or relay to respond, the failure of a pump, fan or diesel generator to start, etc.

f. Passive Failure

The structural failure of a static component which limits the component's effectiveness in carrying out its design function. When applied to a fluid system, this means a break in the pressure boundary resulting in abnormal leakage not exceeding 50 gpm for 30 minutes. Such leak rates are consistent with limited cracks in pipes, sprung flanges, valve packing leaks, or pump seal failures.

2. Active Failure Criteria

The ECCS is designed to accept a single failure following the incident without loss of its protective function. The system is designed to tolerate the failure of any single active component in the ECCS itself or in the necessary associated service systems at any time during the period of required system operations following the incident.

A single active failure analysis is presented in Table 6.3-7, and demonstrates that the ECCS can sustain the failure of any single active component in either the short or long term and still meet the level of performance for core cooling.

Since the operation of the active components of the ECCS following a steam line rupture is identical to that following a loss of coolant accident, the same analysis is applicable and the ECCS can sustain the failure of any single active component and still meet the level of performance for the addition of shutdown reactivity.

3. Passive Failure Criteria

The following philosophy provides for necessary redundancy in component and system arrangement to meet the intent of the General Design Criteria on single failure as it specifically applies to failure of passive components in the ECCS. Thus, for the long term, the system design is based on accepting either a passive or an active failure, assuming no failures in the short term.

6.3.2.11.1 Redundancy of Flow Paths and Components for Long Term Emergency Core Cooling

In design of the ECCS, Westinghouse utilizes the following criteria.

1. During the long term cooling period following a loss of coolant, the emergency core cooling flow paths shall be separable into two subsystems, either of which can provide minimum core cooling functions and return spilled water from the floor of the Reactor Building back to the RCS.

2. Either of the two subsystems can be isolated and removed from service in the event of a leak outside the Reactor Building.

3. Adequate redundancy of check valves is provided to tolerate failure of a check valve during the long term as a passive component.

4. Should one of these two subsystems be isolated in this long term period, the other subsystem remains operable.

5. Provisions are also made in the design to detect leakage from components outside
the Reactor Building, collect this leakage, and to provide for maintenance of the
affected equipment. Level switches are located in alarm drain sumps and certain 98-01
floor drain sumps. Excessive leakage actuates an alarm in the control room,
alerting the operator to take action to isolate the leak.

Thus, for the long term emergency core cooling function, adequate core cooling capacity exists with one flow path removed from service.

6.3.2.11.2 Leakage from Engineered Safety Features Systems Outside Containment

Leakage from the engineered safety feature systems is collected by the Auxiliary
Building drain system. Excessive leakage is detected by level switches located in pump 98-01
room sumps and specially provided alarm drains. These alarm drains are strategically
located in areas housing ECCS and reactor building spray system equipment. Leakage
source identification is facilitated by zoning of the alarm drains with respect to ECCS
and reactor building spray system equipment trains.

When leakage exceeds a flowrate of 25 gpm for the floor drains or 45 gpm for the sump
drains, an alarm is activated in the Control Room. Alarm response time for leakage in 00-01
excess of the maximum expected ranges from a few seconds for the alarm drains to
less than four (4) minutes for the Residual Heat Removal pump room B sump. Upon
actuation of this alarm, the operator can determine which level switch caused the alarm 98-01
and, thus, identify which area housing ECCS or reactor building spray system
equipment is affected. The operator then takes the required action to isolate the leak.

Components of the leakage detection system are seismically qualified, redundant, and can be tested during any phase of plant operation.

The largest sudden leak potential would be the sudden failure of an RHR pump shaft
seal with a conservatively calculated leakage rate of 7.5 gpm. The results of an RN
evaluation to determine RHR pump shaft leak rates are shown by Figure 6.3-5. Figure 12-034
6.3-6 illustrates the leakage rates after severe operation.

A single passive failure analysis is presented by Table 6.3-8. This analysis
demonstrates that the ECCS can sustain a single passive failure during the long term
phase and still retain an intact flow path to the core to supply sufficient flow to maintain
the core covered and effect the removal of decay heat. The procedure followed to 00-01
establish the alternate flow path also isolates the component which failed.

6.3.2.11.3 Integrity of Systems Outside Containment Likely to Contain Radioactive Materials

The design of systems that normally contain radioactive fluids includes a number of features for minimizing leakage paths from these systems to the surrounding environment. These features are outlined as follows:

1. Minimize the Use of Flanged Connections

   As standard practice for the V. C. Summer Project, piping system connections are socket-welded for 2" and smaller piping, and butt-welded for 2-1/2" and larger piping. 00-01

2. Use of Valve Leakoff Connections

   All 2-1/2" or greater air-operated or motor-operated valves in radioactive fluid systems, incorporate lantern-ring leakoff connections which are hard-piped to waste liquid holdup tanks. Therefore, leakage by the valve packings is directed to holdup tanks to prevent contamination of the area around the valves.

3. Use of "Zero-Leakage" Valves

   Manually-operated valves in systems that contain radioactive fluids are "zero-leakage" type valves, i.e., designed to prevent leakage within the environment of the valves. These valves include, but are not limited to, 00-01 bellows-stem-seal type, metal-diaphragm seal, and diaphragm valves. An example of the metal-diaphragm seal type is the Rockwell-Edward's "Hermavalve" (hermetically-sealed). Drain and vent valves are discussed as a separate feature. 00-01

4. Dual Barrier Design for Drain and Vent Valve Connections

   Drain and vent connections in radioactive fluid systems are provided with either (a) two (2) normally-closed isolation valves or (b) one (1) normally-closed isolation valve with a pipe or tube extension that is capped or incorporated a fluid flange. These schemes provide a dual barrier against potential leakage paths from radioactive fluid systems. RN 10-010

5. Use of Mechanical Shaft Seals for Critical Pumps

   The use of mechanical shaft seals for critical pumps limits the amount of radioactive fluid leakage for both normal and adverse operating conditions. For further discussion of pump shaft seals, see FSAR Section 6.3.2.11.2.

6. Use of "Alarm Drains" for Monitoring Leakage from the ECCS Systems

The "Alarm Drains" are strategically located to monitor and alarm leakage from the ECCS Systems during the injection and recirculation modes of operation following a LOCA. For further discussion of the Alarm Drain System, See FSAR Section 6.3.2.11.2.

A single passive failure analysis is presented in Table 6.3-8. This analysis is discussed in Section 6.3.2.11.2. 00-01

Those systems and components outside the containment whose function may be required during serious transient or accident conditions which could or would contain highly radioactive fluids have been identified. In addition, these systems and component interfaces with other systems and components which would not be required to function to achieve cold stable conditions following an accident have been identified.

These systems and components are incorporated into leak reduction programs including, but not limited to, Inservice Inspection Hydrostatic and Leak Detection Testing, Preventative Maintenance Programs, Integrated System Leakage Testing as provided for by Technical Specification Surveillance Requirements, and existing Leak Detection systems. 00-01

6.3.2.12 Protection Provisions

The provisions taken to protect the system from damage that might result from dynamic effects are discussed in Section 3.6. The provisions taken to protect the system from missiles are discussed in Section 3.5. The provisions to protect the system from seismic damage are discussed in Sections 3.7, 3.9, and 3.10. The provisions to maintain the refueling water storage tank at or above 40°F and for freeze protection of outdoor piping connected to the tank are discussed in Sections 9.1.3.2 and 9.1.3.5.5. Thermal stresses on the RCS are discussed in Section 5.2.

6.3.2.13 Provisions for Performance Testing

The provisions incorporated to facilitate performance testing are discussed in Section 6.3.4.

6.3.2.14 Pump Net Positive Suction Head

The ECCS is designed so that adequate net positive suction head (NPSH) is provided to system pumps. The NPSH available is a function of Reactor Building pressure, water temperature, static head of water, friction losses in the pump suction piping and pressure drop across the sump strainer as follows:

$$NPSH_a = P_c - P_s + \Delta H_z - \Delta H_f - \Delta H_s$$

Where:

$NPSH_a$ = NPSH available at the pump suction.

P_c = Reactor Building pressure.

P_s = Saturation pressure of the pumped water.

ΔH_Z = Elevation pressure due to the height of water above the pump nozzle.

ΔH_f = Pressure loss due to friction in the suction line.

ΔH_s = Pressure loss across the sump strainer (in the recirculation mode).

RN 13-022

In the recirculation mode of operation, the pressure in the Reactor Building (P_c) will always equal or exceed the vapor pressure of the water in the sump because water will evaporate from the sump to increase the vapor pressure of the atmosphere if P_c starts to decrease below P_s. Generally, the partial pressure of the air in the atmosphere will maintain P_c greater than P_s. For conservatism, no containment overpressure is assumed so $P_c = P_s$. This is consistent with the guidance provided in Regulatory Guide 1.1.

Adequate net positive suction head is shown to be available for all pumps as follows:

1. Residual Heat Removal Pumps

   The net positive suction head of the residual heat removal pumps is evaluated for normal plant cooldown operation, and for both the injection and recirculation modes of operation for the Design Basis Accident. Recirculation operation gives the limiting net positive suction head requirement, and the net positive suction head available is determined from the Reactor Building sump level relative to the pump elevation and the pressure drop in the suction piping from the sump to the pumps. The net positive suction head evaluation is based on one residual heat removal pump delivering to three RCS loops and the suction of one charging pump.

A summary of the results for the RHR pump in the recirdulation mode is as follows: RN 13-022

	Flow (gpm)	NPSH Available (ft)	NPSH Required (ft)	Margin (ft)
Pump A	4300	20.2	17	3.2
Pump B	4200	20.8	16	4.8

2. Centrifugal Charging Pumps

The net positive suction head for the centrifugal charging pumps is evaluated for both the injection and recirculation modes of operation for the Design Basis Accident. The end of the injection mode of operation gives the limiting net positive suction head available. The net positive suction head available is determined from the elevation head and vapor pressure of the water in the refueling water storage tank, which is at atmospheric pressure, and the pressure drop in the suction piping from the tank to the pumps. When a predetermined low refueling water storage tank level is reached the charging pumps are manually aligned to take suction from the residual heat removal pump discharge headers. The net positive suction head requirements of these pumps are therefore satisfied by the discharge head of the residual heat removal pumps.

6.3.2.15 Accumulator Motor Operated Valve Controls

As part of the plant shutdown administrative procedures, the operator is required to close these valves. This prevents a loss of accumulator water inventory to the RCS and is done shortly after the RCS has been depressurized to between 900 and 950 psig. 00-01
The redundant pressure and level alarms on each accumulator would remind the operator to close these valves, if any were inadvertently left open. Power is disconnected after the valves are closed.

During plant startup, the operator is instructed via procedures to energize and open these valves before the RCS pressure is between 900 and 950 psig. Monitor lights in 00-01
conjunction with an audible alarm will alert the operator should any of these valves be left inadvertently closed once the RCS pressure increases above the safety injection block setpoint (P-11). 00-01

For each accumulator isolation valve (8808A, 8808B, or 8808C) position indicating design provisions include the following:

1. Visual indicating in the main control room by means of an ESF monitor light that will be "bright" when the valve is not open. Section 7.5.4 provides a more detailed explanation of the operation of the ESF monitor lights. An alarm will be annunciated via a common monitor light alarm if an accumulator valve is not open. This grouping highlights a valve not properly lined up. This light is energized from an ESF monitor light supply different from the valve control power and actuated by

a valve motor operator switch. In addition to this visual position information, there are also red (open) and green (closed) position indicating lights at the control switch for each valve. These lights are actuated by valve motor operator limit switches and are powered by valve control power which is distributed through the same accumulator valve breaker that feeds power to the motor operator.

2. Audible and visual alarm annunciator points will be activated by both a valve motor operator limit switch and by a valve position limit switch activated by stem travel whenever an accumulator valve is not fully open for any reason with the system at pressure (the pressure at which the safety injection block is unblocked). The alarm activated by the motor operator limit switch will be recycled by a timer at approximately 60 minute intervals to remind the operator of the improper valve line up. Both the alarm reflash annunciator point and the timer will be energized separately from the valve control power.

The accumulator isolation valves are considered to be "operating bypasses" in the context of IEEE 279-1971, which requires that bypasses of protective functions be removed automatically whenever permissive conditions are not met. The automatic unblock is derived from P-11 (1985 psig) which is the Safety Injection setpoint. 00-01

In that these valves are required by Technical Specifications to be open and incapable of being closed during power operation and do not act as active valves during any accident condition, the requirements of IEEE 279-1971 are not applicable for this situation in that an operating bypass function is not required for these valves. For a discussion of limiting conditions for operation and surveillance requirements of these valves, refer to the Technical Specifications. 00-01

The accumulator isolation valves receive a safety injection signal to ensure that they are open in the event of an accident which initiates safety injection.

For further discussions of the instrumentation associated with these valves refer to Sections 6.3.5, 7.3.1.1.2, and 7.6.4.

6.3.2.16 Motor Operated Valves and Controls

Remotely operated valves for the injection mode which are under manual control (i.e., valves which normally are in their ready position and do not require a safety injection signal) have their positions indicated on a common portion of the control board. If a component is out of its proper position, its monitor light will indicate this on the control panel. At any time during operation when one of these valves is not in the ready position for injection, this condition is shown visually on the board, and an audible alarm is sounded in the control room.

6.3.2.17 Manual Actions

No manual actions are required of the operator for proper operation of the ECCS during the injection mode of operation. Only limited manual actions are required by the operator to complete realignment of the system for the cold leg recirculation mode of operation, and, within 8 hours after event initiation, for the simultaneous hot leg and cold leg recirculation mode of operation. See Table 6.3-3 for the cold and hot leg recirculation switchover procedures. RN 02-039

6.3.2.18 Process Instrumentation

Process instrumentation available to the operator in the control room to assist in assessing post loss of coolant accident conditions are tabulated in Section 7.5.

6.3.2.19 Materials

Materials employed for components of the ECCS are given in Table 6.3-5. These materials are chosen based upon their ability to resist radiolytic and pyrolytic decomposition (see Section 6.3.2.4).

6.3.2.20 Power Lockout of Manually Controlled Electrically Operated Valves

Manually controlled electrically operated valves in the ECCS which require power lockout in accordance with Branch Technical Position EICSB 18 are 8808A, B, and C (accumulator isolation valves); 8884 and 8886 (high head safety injection hot leg injection isolation valves); 8888A and B (low head, safety injection cold leg isolation valves); 8889 (low head safety injection hot leg injection isolation valve); and 8133A and B (high head safety injection discharge cross-connect valves). 98-01

Power is locked out to accumulator isolation valves 8808A, B, and C (see Technical Specification 3/4.5.1 and to reactor head vent system isolation valves 8095A, B and 8096A, B by padlocking the breaker operating mechanism of the motor control center.

The power, control, and indication circuits for motor operated valves 8884, 8886, 8888A and B, 8889, and 8133A and B will satisfy the main control board "removal and restoration" requirements identified below: 98-01

1. Removal and restoration of electrical power for the valve power circuit will be accomplished from the main control board. A power contactor will be installed in XMCIDA2Y and wired in series to reversing starters for valves 8884, 8888A and 8889. When the contactor is open, it will break power to the starters. A similar situation exists in XMCIDB2X for valves 8886 and 8888B. Power lockouts are also installed in XMC1DA2Y for valve 8133A and XMC1DB2Y for valve 8133B. 98-01

Before the valve position can be changed, the power lockout contactor must be closed from the main control board to provide power to the starters. Then, using a separate switch on the main control board, the starter must be actuated to open or close the valve. Using these normally de-energized power lockout contactors in series with the starters, no single electrical fault could cause valve motion.

2. Valve position indications, not defeated by the control circuit electrical power removal and restoration, are provided at the main control board. Valve position sensing for these position indicators is provided by the cam operated switch within the motor operator. AC Power for the indicators is supplied from the power lockout contactor control transformer in the motor control center (MCC).

3. Redundant valve position indicators, powered from the DC train opposite to the valve motive power train, are provided at the main control board. Valve position sensing for these position indicators is by separate, stem mounted limit switches.

6.3.3 PERFORMANCE EVALUATION

6.3.3.1 Evaluation Model

The following analyses are performed to ensure that the limits on core behavior following a pipe rupture are met by the ECCS operating with minimum design equipment:

1. Large pipe rupture analysis

2. Small pipe rupture analysis

3. Steam line rupture analysis

The performance characteristic is derived from the performance characteristic for each pump with system piping resistance, based upon final piping layouts.

The pump performance characteristics utilized in the accident analyses includes a 3.5% head reduction for the high head pumps and a 7% head reduction for the low head pumps for margin. The loss of coolant accident analysis is also based on the spilling of one injection line through the postulated break. This accounts for the case where the initiating incident is assumed to be the severance of an injection line. RN 06-019 00-01

6.3.3.2 Emergency Core Cooling System Performance

The large pipe rupture analysis is used to evaluate the initial thermal transient for a spectrum of pipe ruptures up to and including the instantaneous, double ended, circumferential rupture of the largest pipe in the RCS. The results of this analysis are presented in Section 15.4.

The flow delivered to the RCS by the ECCS as a function of the RCS pressure is also shown in Section 15.4. This flowrate is based on one injection path spilling out the postulated break, the loss of offsite power, and the assumed single failure of an onsite emergency diesel generator to start; leaving two accumulators, one charging pump, and one residual heat removal pump available for core cooling, with the injection path of least resistance assumed to be spilling.

RCS loop-to-loop flow imbalance of 5% is considered, to incorporate the effects of the RCS Flow Asymmetry, per Westinghouse Nuclear Safety Advisory Letter (NSAL) 00-008. RN 01-125

Initial core cooling is provided by the passive accumulators. Injection flow from the ECCS pumps is required subsequent to accumulator injection to complete vessel refill, reflood the core, and eventually return the core to a subcooled state. The results of the analysis presented in Section 15.4 indicate the peak clad temperature attained at any point in the core is such that the limits on core behavior as specified in Section 15.4 are met.

During shutdown the following operator actions pertain to the isolation of ECCS equipment and would affect a LOCA during the time accumulator isolation valves are closed with power locked out. (Startup is not addressed since shutdown is more limiting due to the higher core decay heat generation): 98-01

1. The operator is instructed to manually block the pressurizer safety injection signal when pressurizer pressure has decreased below 1985 psig and the steamline safety injection signal when T_{avg} decreases below 552°F. All other safety injection signals including containment high pressure and high steam line differential pressure are armed and will actuate safety injection if their setpoints are exceeded. Manual safety injection actuation is also available. 98-01

2. With the RCS pressure maintained between 900 and 950 psig, the operator closes and locks out the safety injection accumulator isolation valves. At this time, two residual heat removal pumps (low head safety injection) would be available from either automatic or manual safety injection actuation. 98-01

3. At less than 425 psig and 350°F, the operator aligns the suction of both trains of RHRS to the RCS. The valves in the line from the refueling water storage tank (RWST) are closed. Prior to dropping below 300°F, the operator also locks out all but one high head charging pump. 00-01

The significance of these actions on the mitigation of a LOCA when power is locked out to the isolation valves is that:

1. Between 950 psig and 425 psig, a portion of the ECCS may be actuated automatically on containment high pressure or high steam line differential pressure signals or manually by the operator. The equipment that can be energized are two residual heat removal and one high head charging pumps. Subsequently, the operator would reinstitute power at the motor control centers to the other high head charging pump as necessary to maintain water level in the reactor vessel. 98-01 00-01

2. Below 425 psig and 350° F, the system is in the residual heat removal mode with both trains of RHRS aligned to the RCS and only one train in service until Mode 5. The opposite train is protected such that in the event of a LOCA, it is capable of performing its Mode 4 ECCS function. If required, the operator would manually realign the protected train per abnormal operating procedure for mitigation of a LOCA. In addition, the procedures direct the operator to start additional SI pumps, as necessary, to maintain water level in the reactor vessel. RN 09-031

6.3.3.3 Alternate Methods of Analysis

6.3.3.3.1 Small Pipe Break

The small pipe break analysis is used to evaluate the initial core thermal transient for a spectrum of pipe ruptures greater than 3/8 inches in diameter. For a rupture 3/8 inches in diameter or smaller, the charging system can maintain the pressurizer level. The ECCS would not be required and would not be automatically actuated in the short term, although it is actuated by low pressurizer pressure. A range of reactor coolant system small pipe breaks have been evaluated, as discussed in Section 15.3.1. The results of the small break analysis indicate that the limits on core behavior are adequately met as shown in Section 15.3. 00-01

6.3.3.3.2 Steam Line Rupture

Following a steam line rupture the ECCS is automatically actuated to deliver borated water to the RCS. The response of the ECCS following a steam line break is identical to its response during the injection mode of operation following a loss of coolant accident.

The safety injection signal initiates identical actions as described for the injection mode of the loss of coolant accident, even though not all of these actions are required following a steam line rupture, e.g., the residual heat removal pumps are not required since the RCS pressure will remain above their shutoff head.

The charging pumps deliver borated water from the refueling water storage tank, until enough water has been added to the RCS to make up for the shrinkage due to cooldown. After pressurizer water level has been restored, the injection is manually terminated.

The sequence of events following a postulated steam line break is described in Section 15.4.

6.3.3.4 Fuel Rod Perforation

Discussions of peak clad temperatures and metal-water reactions appear in Sections 15.3.1 and 15.4.1. Analyses of the radiological consequences of a fission product release due to the rupture of a pipe in the RCS are presented in Section 15.4.

6.3.3.5 through 6.3.3.10 Are not applicable to Pressurized Water Reactors.

6.3.3.11 Effects of ECCS Operation on the Core

The effects of ECCS operation on the reactor core are presented in Sections 15.3 and 15.4.

6.3.3.12 Use of Dual Function Components

The ECCS contains components which have no other operating function as well as components which are shared with other systems. Components in each category are as follows:

1. Components of the ECCS which perform no other function are:

    a. One accumulator for each loop which discharges borated water into its respective cold leg of the reactor coolant loop piping.

    b. Piping, valves and instrumentation associated with the accumulators or high head injection.

    c. Valves and piping associated with RB sump recirculation. 00-01

2. Components which also have a normal operating function are as follows:

    a. The residual heat removal pumps and the residual heat exchangers are normally used during the latter stages of normal reactor cooldown and when the reactor is held at cold shutdown for core decay heat removal. However, during all other plant operating periods, they are aligned to perform the low head injection function.

    b. The centrifugal charging pumps are normally aligned for charging service. As a part of the chemical and volume control system, the normal operation of these pumps is discussed in Chapter 9.

c. The refueling water storage tank is used to fill the refueling canal for refueling operations. However, during other plant operating periods it is aligned to the suction of the residual heat removal pumps. The charging pumps are automatically aligned to the suction of the refueling water storage tank upon receipt of the safety injection signal.

An evaluation of all components required for operation of the ECCS demonstrates that either:

1. The component is not shared with other systems, or

2. If the component is shared with other systems, it is aligned during normal plant operation to perform its accident function; or if not aligned to its accident function, two valves in parallel are provided to align the system for injection, and two valves in series are provided to isolate portions of the system not utilized for injection. These valves are automatically actuated by the safety injection signal. During the 00-01
recirculation mode of operation, manual valve alignments are required. (Reference 6.3.2.2.7)

Table 6.3-9 indicates the alignment of components during normal operation, and the realignment required to perform the accident function.

6.3.3.13 <u>Lag Times</u>

The minimum active components will be capable of delivering full rated flow within a specified time interval after process parameters reach the setpoints for the safety injection signal. Response of the system is automatic, with appropriate allowances for 00-01
delays in actuation of circuitry and active components. The active portions of the system are actuated by the safety injection signal. In analyses of system performance, delays in reaching the programmed trip points and in actuation of components are established on the basis that only emergency onsite power is available.

In the loss of coolant accident analyses presented in Sections 15.3 and 15.4, no credit is assumed for partial flow prior to the establishment of full flow and no credit is assumed for the availability of normal offsite power sources. After the initiation parameter achieves the setpoint, a delay is assumed for signal transmission, diesel startup, load sequencing, and pump startup, as discussed in Sections 15.3.1.2 00-01
and 15.4.2.1.

For smaller loss of coolant accidents, there is some additional delay before the process variables reach their respective programmed trip setpoints since this is a function of the severity of the transient imposed by the accident. This is allowed for in the analyses of the range of loss of coolant accidents.

Accumulator injection occurs immediately when RCS pressure has decreased below the operating pressure of the accumulator.

6.3.3.14 Thermal Shock Considerations

Thermal shock considerations are discussed in Section 5.2.

6.3.3.15 Limits on System Parameters

The analyses show that the Performance characteristic of the ECCS is adequate to meet the requirements for core cooling following a loss of coolant accident with the minimum engineered safety feature equipment operating. In order to ensure this capability in the event of the simultaneous failure to operate any single active component, Technical Specifications are established for reactor operation.

Normal operating status of ECCS components is given in Table 6.3-6.

The ECCS components are available whenever the coolant energy is high and the reactor is critical. During low temperature physics tests there is a negligible amount of stored energy in the coolant and low decay heat; therefore, an accident comparable in severity to accidents occurring at operating conditions is not possible and ECCS components are not required.

A comprehensive testing program has been undertaken to demonstrate that ECCS components and associated instrumentation and electrical equipment which are located inside the Reactor Building will operate for the time period required in the combined post loss of coolant accident conditions of temperature, pressure, humidity, radiation, chemistry, and seismic phenomena (see Section 3.11).

Components such as remote motor operated valves have been shown capable of operating for the required post accident periods, when exposed to post loss of coolant environmental conditions. All other ECCS components are located outside of the Reactor Building. 00-01

The specification of individual parameters as given in Table 6.3-1 considers allowances for margin over and above the required performance value (e.g., pump flow and net positive suction head), and the most severe conditions to which the component could be subjected (e.g., pressure, temperature, and flow).

This consideration ensures that the ECCS is capable of meeting its minimum required level of functional performance.

6.3.4 TESTS AND INSPECTIONS

To demonstrate the readiness and operability of the ECCS, the components are subjected to periodic tests and inspections. Performance tests of the components are performed in the manufacturer’s shop. An initial system flow test is performed to demonstrate the proper functioning of the components.

6.3.4.1 Quality Control

Tests and inspections are carried out during fabrication of each of the ECCS components. These tests are conducted and documented in accordance with the Quality Assurance Program discussed in Chapter 17.

6.3.4.2 Preoperational Tests

These tests are intended to evaluate the hydraulic and mechanical performance of the passive and active components involved in the injection mode by demonstrating that they have been installed and adjusted so they will operate in accordance with the design intent. These tests are divided into three individual sections that may be performed as plant conditions allow without compromising the integrity of the tests.

One of these individual sections consists of system actuation tests to verify: the operability of ECCS valves initiated by the safety injection signal (S), the phase A containment isolation signal (T), and the phase B containment isolation signal (P); the operability of safeguard pump circuitry down through the pump breaker control circuits; and the proper operation of valve interlocks.

Another of the individual sections is the accumulator injection test. The objective of this section is to check the accumulator injection line to verify that the lines are free from obstructions and that the accumulator check valves operate correctly. The test objectives will be met by a low pressure blowdown of each accumulator. The test will be performed with the reactor head and internals removed.

The last of the individual sections consists of operational tests of the major pumps (i.e., the charging pumps and the residual heat removal pumps). The purpose of these tests is to evaluate the hydraulic and mechanical performance of the pumps delivering through the flow paths required for emergency core cooling. These tests will be divided into two parts: pump operation under minimum flow conditions and pump operation at full flow conditions.

The predicted system resistance will be verified by measuring the flow in various piping branches, as each pump delivers from the refueling water storage tank to the open reactor vessel, and adjustments made where necessary to assure that no one branch has an unacceptably low or high resistance. During this flow test, the system will also be checked to assure there is sufficient total line resistance to prevent excessive runout of the pump. At the completion of the flow test, the total pump flow and relative flow between the branch lines will be compared with the performance characteristics used in the accident analysis of Chapter 15.

The systems are accepted only after demonstration of proper actuation of components and after demonstration of flow delivery of components within design requirements.

6.3.4.3 Periodic Component Testing

Routine periodic testing of the ECCS components and necessary support systems at power is planned. Valves which operate after a loss of coolant accident are operated through a complete cycle, and pumps are operated individually. The testing is conducted on minimum flow and at full flow for the charging pumps and at full flow for the RHR pumps. If such testing indicates a need for corrective maintenance, the redundancy of equipment in these systems permits such maintenance, to be performed without shutting down or reducing load under certain conditions. These conditions include considerations such as the period within which the component should be restored to service and the capability of the remaining equipment to provide the minimum required level of performance during such a period.

The condition of the remote stop valve and the check valve in each accumulator tank discharge line may be tested by opening the remote test line valves just downstream of the stop valve and check valve respectively. Flow through the test line can be observed on instruments and the opening and closing of the discharge line stop valve can be sensed on this instrumentation.

Where series pairs of check valves form the high pressure to low pressure isolation barrier between the RCS and ECCS piping outside the Reactor Building, testing of these check valves must be performed to provide assurance that certain postulated failure modes will not result in a loss of coolant from the low pressure system outside the Reactor Building with a simultaneous loss of safety injection pumping capacity. The tests performed verify that each of the series check valves can independently sustain differential pressure across its disc, and also verify that the valve is in its closed position. The required periodic tests are to be performed after each refueling just prior to plant startup, after the RCS has been pressurized.

Lines in which the series check valves are to be tested are the residual heat removal pump cold leg and hot leg injection lines.

To implement the periodic component testing requirements, Technical Specifications are established. During periodic system testing, a visual inspection of pump seals, valve packings, flanged connections, and relief valves is made to detect leakage. Inservice inspection provides further confirmation that no significant deterioration is occurring in the ECCS fluid boundary. 00-01

Design measures have been taken to assure that the following testing can be performed:

1. Active components are tested periodically for operability (e.g., pumps, certain valves, etc.).

2. An integrated system test [1] can be performed when the plant is cooled down and the RHRS is in operation. This test does not introduce flow into the RCS but does demonstrate the operation of the valves, pump circuit breakers, and automatic circuitry including diesel starting and the automatic loading of ECCS components on the diesels (by simultaneously simulating a loss of offsite power).

3. An initial flow test of the full operational sequence can be performed.

The design features which assure this test capability are specifically:

1. Power sources are provided to permit individual actuation of each active pump and valve component of the ECCS.

2. The residual heat removal pumps are used every time the RHRS is put into operation. They can also be tested periodically when the plant is at power.

3. The centrifugal charging pumps are either normally in use for charging service or can be tested periodically on minimum flow, or full flow during plant shutdown.

4. Remote operated valves can be exercised during routing plant maintenance. These valves are tested both quarterly and/or during plant shutdowns based on operating and test requirements.

5. Level and pressure instrumentation is provided for each accumulator tank, for continuous monitoring of these parameters during plant operation.

6. Flow from each accumulator discharge tank can be directed at any time through a test line to determine check valve leakage.

7. Flow indicators are provided in the charging pump headers and in the residual heat removal pump headers. Pressure instrumentation is also provided in the residual heat removal pump headers and in the charging pump headers.

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(1) Details of the testing of the sensors and logic circuits associated with the generation of a safety injection signal together with the application of this signal to the operation of each active component are given in Section 7.2.

6.3.4.4 Inservice Tests and Inspections

The ECCS components are designed and fabricated to permit inspection and inservice tests in accordance with the ASME Code, prescribed under 10CFR50.55a. Inservice inspection is discussed in Section 5.7. RN 04-012

6.3.5 INSTRUMENTATION REQUIREMENTS

Instrumentation and associated analog and logic channels employed for initiation of ECCS operation is discussed in Section 7.3 This section describes the instrumentation employed for monitoring ECCS components during normal plant operation and also ECCS post accident operation. Alarms are annunciated in the control room.

6.3.5.1 Temperature Instrumentation

6.3.5.1.1 Boron Injection Tank Temperature

Deleted by Amendment 3.

6.3.5.1.2 Residual Heat Exchanger Inlet and Outlet Temperature

The fluid temperature at the inlet and outlet of each residual heat exchanger is recorded in the control room.

6.3.5.1.3 Boron Injection Surge Tank Temperature

Deleted by Amendment 3.

6.3.5.2 Pressure Instrumentation

6.3.5.2.1 High Head Injection Pressure

High head injection pressure is indicated in the control room. A high pressure alarm is provided.

6.3.5.2.2 Accumulator Pressure

Duplicate pressure channels are installed on each accumulator. Pressure indication in the control room and high and low pressure alarms are provided by each channel.

6.3.5.2.3 Hydrotest Pump Discharge Pressure

This instrument provides local indication of hydrotest pump discharge pressure.

6.3.5.2.4 Test Line Pressure

A local pressure indicator used to check for proper seating of the accumulator check valves is installed on the leakage test line.

6.3.5.2.5 Residual Heat Removal Pump Discharge Pressure

Residual heat removal discharge pressure for each pump is indicated in the control room. A high pressure alarm is actuated by each channel.

6.3.5.3 Flow Instrumentation

6.3.5.3.1 Boric Acid Recirculation Flow

Deleted by Amendment 3.

6.3.5.3.2 Charging Pump Injection Flow

These flow transmitters located in the hot leg and cold leg injection headers provide flow indication in the control room.

6.3.5.3.3 Residual Heat Removal Pump Injection Flow

Flow through each residual heat removal pump header is indicated in the control room. A low flow alarm is provided.

6.3.5.3.4 Test Line Flow

Local indication of the leakage test line flow is provided to check for proper seating of the accumulator check valves between the injection lines and the RCS.

6.3.5.3.5 Residual Heat Removal Pump Minimum Flow

A flowmeter installed in each residual heat removal pump discharge header provides control for the valve located in the pump minimum flow line.

6.3.5.4 Level Instrumentation

6.3.5.4.1 Refueling Water Storage Tank Level

Four independent refueling water storage tank level channels are provided for automatic actuation of the residual heat removal pump valves to initiate recirculation upon receipt of two out of four (2/4) lo-lo level signals coincident with the receipt of a safety injection signal (the residual heat removal pumps continue in the operating mode).

The refueling water storage tank water level is indicated in the control room. Two of the four channels also provide a high, low, and an empty level alarm. The high level (maximum fill) alarm is provided to protect against possible overflow of the refueling 00-01

water storage tank. The low level (minimum fill) alarm is provided to assure that a sufficient volume of water is always available in the refueling water storage tank in conformance with the Technical Specifications.

6.3.5.4.2 Accumulator Water Level

Duplicate water level channels are provided for each accumulator. Both channels provide indication in the control room and actuate high and low water level alarms.

6.3.5.4.3 Boron Injection Surge Tank Level

Deleted by Amendment 3.

6.3.5.5 Valve Position Indication

Valve positions for those valves provided with engineered safety features monitoring lights (see Section 7.5) are indicated on the control board by a "bright-dim" system; i.e., should the valve not be in its proper position, a bright white light will be lit and thus give 00-01
a highly visible indication to the operator. Valve position for remote manual ECCS valves is also indicated on the control board by red and green indicator lights.

The accumulator motor operated valves are provided with red (open) and green (closed) position indicating lights located at the control switch for each valve. These lights are powered by valve control power and actuated by valve motor operator limit switches.

A monitor light that is bright when the valve is not fully open (i.e., mispositioned) is provided in an array of monitor lights that are all dim when their respective valves are in 00-01
proper position enabling engineered safety features operation. This light is energized from a separate monitor light supply and actuated by a valve motor operated limit switch.

An alarm annunciator point is activated by both a valve motor operator limit switch and by a valve position limit switch activated by stem travel whenever an accumulator valve is not fully open for any reason with the system at pressure (the pressure at which the safety injection block is unblocked approximately 1900 psig). A separate annunciator point is used for each accumulator valve.

The manual valve (XVG-6700-SF) in the common header from the refueling water storage tank is provided with a limit switch which alarms the main control room if the valve is not full open.

6.3.6 REFERENCES

1. Deleted (RN 99-069) RN 99-069

2. "Behavior of Austenitic Stainless Steel in Post Hypothetical Loss of Coolant Accident Environment," WCAP-7798-L (Proprietary) and WCAP-7803 (Non-Proprietary), January 1972.

3. Technical Report TR04650-004, Rev. 0, “Chemical Effects Testing for V. C. Summer Replacement Sump Strainers.” RN 13-022

4. WCAP-16406-P-A, Rev. 1, “Evaluation of Downstream Sump Debris Effects in Support of GSI-191.”

TABLE 6.3-1

EMERGENCY CORE COOLING SYSTEM
COMPONENT PARAMETERS

Accumulators

Number	3	
Design Pressure, psig	700	
Design Temperature, °F	300	
Operating Temperature, °F	60-150	00-01
Nominal Operating Pressure, psig	628 psig	
Total Volume, ft^3	1450 each	
Nominal Water Volume, ft^3	1014 each	99-01
Nominal Volume N_2 Gas ft^3	436 each	
Boric Acid Concentration, minimum, ppm	2200	
Relief Valve Setpoint, psig	700	

Centrifugal Charging Pumps

Number	3
Design Pressure, psig	2900
Design Temperature, °F	300
Design Flowrate(1), gpm	150
Design Head, ft	5800
Maximum Flowrate, gpm	688
Design Discharge Head at Shutoff, ft	6200
Motor Rating(2), bhp	900

(1) Includes minimum flow
(2) 1.15 service factor not included

TABLE 6.3-1 (Continued)

EMERGENCY CORE COOLING SYSTEM
COMPONENT PARAMETERS

Hydrotest Pump

Number	1
Design Pressure, psig	3300
Design Temperature, °F	300
Normal Operating Temperature, °F	Ambient
Design Flowrate, gpm	24.5
Developed Head of Design Flowrate, ft	7100

Residual Heat Removal Pumps

(See Section 5.5.7 for design parameters)

Residual Heat Exchangers

(See Section 5.5.7 for design parameters)

Boron Injection Tank

Deleted by Amendment 3.

Boron Injection Surge Tank

Deleted by Amendment 3.

Heaters

Deleted by Amendment 3.

Boron Injection Tank Recirculation Pump

Deleted by Amendment 3.

TABLE 6.3-1 (Continued)

EMERGENCY CORE COOLING SYSTEM COMPONENT PARAMETERS

Component Valves		Parameter		
1. Motor Operated Valves		Design Opening or Closing Time		00-01
	a. 4 inches and under	Time, sec	≤ 10	
	b. > 4 inches, ≤ 10 inches	Time, sec	≤ 15*	
	c. Over 10 inches	Time, sec	≤ 20	
2. Leakage				
	a. Conventional Globe Valves	Disc Leakage, cc/hr/in of nominal pipe size	3	
		Backseat Leakage (when open), cc/hr/in of stem diameter	1	
	b. Gate Valves	Disc Leakage, cc/hr/in of nominal pipe size	3	
		Backseat Leakage (when open), cc/hr/in of stem diameter	1	
	c. Check Valves	Disc Leakage, cc/hr/in of nominal pipe size	3	
	d. Diaphragm Valves	Disc Leakage	none	
	e. Accumulator Check Valves	Disc Leakage, cc/hr/in of nominal pipe size	3	00-01
	f. Seal Water Injection Filter Vent and Drain Ball Valves	Seat and External Leakage, cc/hr	3	RN 10-010

* 8 inch valves 8130A/B and 8131A/B are currently tested to ≤ 30 seconds. 00-01

TABLE 6.3-2

EMERGENCY CORE COOLING SYSTEM RELIEF VALVE DATA

Description	Fluid Discharged	Fluid Inlet Temperature Normal	Set Pressure Psig	Back Pressure Constant	Psig Buildup	Capacity	
N_2 Supply to Accumulators (See Note)	N_2	120	700	0	0	1500 scfm	RN 99-013
Residual Heat Removal Pump Safety Injection Lines	Water	350	600	3	50	20 gpm	RN 99-010
Accumulator to Reactor Building	N_2 Gas	120	700	0	0	1500 scfm	
Hydrotest Pump Discharge (See Note)	Water	120	800	0	0	30 gpm	RN 99-013
Safety Injection Pumps Suction Lines	Water	115	220	3	50	25 gpm	00-01

Note: The N_2 supply header and hydrostatic test pump relief valves are included in the SI system but are outside the boundary of the ECCS. The N_2 supply relief valve protects the N_2 supply line from overpressurization due to failure of the pressure regulator. The hydrostatic test pump relief valve protects the pump discharge piping during accumulator makeup operations. RN 99-013

TABLE 6.3-3

SEQUENCE OF SWITCHOVER OPERATION
FROM INJECTION TO RECIRCULATION

The following manual operator actions are required to complete the switchover operation from the injection mode to the recirculation mode in accordance with the provisions of Branch Technical Position (BTP) EICSB20. During the injection mode, the operator verifies that all ECCS pumps (not including the spare or out of service charging pump) are operating and monitors the RWST and reactor building recirculation sump levels in anticipation of switchover. The operator opens, or verifies open, the component cooling water inlet isolation valves to the residual heat removal heat exchanger prior to switchover initiation. Upon receipt of the RWST "lo-lo" level signal in conjunction with the safety injection signal, the reactor building recirculation sump isolation valves (3004A & B, 3005A & B, 8811A & B, and 8812A & B) automatically open. Following these automatic actions, the operator is required to initiate the following manual actions to complete switchover. 00-01 RN 06-029

The following manual actions must be completed in a timely manner following switchover initiation to align the charging pumps suction to the residual heat removal pumps discharge. RN 06-029

1. Verify that the reactor building recirculation sump isolation valves (3004A & B, 3005A & B, 8811A & B, and 8812A & B) are open, the residual heat removal pump suction valves (8809A & B) and the reactor building spray pump suction valves (3001A & B) from the refueling water storage tank are closed. RN 06-029
2. Close the valves (8887A & B) in the crossover line downstream of the residual heat removal heat exchanger.
3. Close the valves (8106, 8109A, B, & C) in the charging pump miniflow lines.
4. Open residual heat removal pump discharge valves (8706A & B) to the suction of the charging pumps.

Note that the four basic steps above are followed even if a single failure occurs; however, it may be necessary to secure pump(s) when it is determined that the suction(s) cannot be transferred from the RWST to conserve RWST water for the completion of the switchover. RN 06-029 02-01

All ECCS pumps are now aligned with suction flow from the reactor building recirculation sumps. The operator verifies proper operation and alignment of all ECCS components and proceeds to complete the following manual actions to align the ECCS in redundant flow paths for long term recirculation operation:

1. Close charging pump suction valves (LCV-115B & D) from the refueling water storage tank.
2. Close valves (8130A & B or 8131A & B) (depending on operating charging pumps) in the suction headers to establish two separate high head recirculation suction systems.

RN 02-039

3. Open valve (8885) in the alternate high head cold leg recirculation line.
4. Close valves (8132A & B or 8133A & B) (depending on operating charging pumps) in the discharge header to establish two separate high head recirculation discharge systems.

SEQUENCE OF SWITCHOVER OPERATION FROM COLD LEG TO HOT LEG RECIRCULATION
(Procedure is executed one train at a time)

The following manual operator actions are required to perform the changeover operation from the cold leg recirculation mode to the hot leg and cold leg recirculation mode.

1. Stop charging pump no. 1.
2. Close the alternate high head cold leg header isolation valve (8885) and open the corresponding high head hot leg header isolation valve (8884). RN 02-039
3. Restart charging pump no. 1.
4. Stop charging pump no. 2.
5. Close the high head injection isolation valves (8801A/B) and open the corresponding high head hot leg header isolation valve (8886).
6. Restart charging pump no. 2.

TABLE 6.3-4

MAXIMUM EXPECTED EXTERNAL RECIRCULATION LOOP LEAKAGE

	Items	Number of Units	Type of Leakage Control and Unit Leakage Rate	Leakage to Atmosphere cc/hr	Leakage to Drain Tank cc/hr	
1.	Residual Heat Removal Pumps (Low Head Safety Injection)	2	Mechanical seal with leakoff 50 cc/hr	100	0	RN 03-019
2.	Charging Pumps (High Head Safety Injection)	3	Mechanical seal with leakoff 180 cc/hr	540	0	
3.	Flanges:					
	a. Pumps	6	Gasket - adjusted to zero	180	0	
	b. Valves Bonnet to Body (Larger than 2 inches)	40	leakage following any test - 10 drops/min/flange used	1200	0	
	c. Control Valves (Butterfly)	8	30 cc/hr	240	0	
	d. Heat Exchangers (2)	4	-	120	0	
4.	Valves - Steam Leakoffs	25	Backseated double packing with leakoff - 1 cc/hr/in stem diameter used	0	25	
5.	Miscellaneous Small Valves	108	Flanged body packed stems - 1 drop/min used (3 cc/hr)	324	0	RN 10-010
6.	Miscellaneous Large Valves (Larger than 2 inches)	20	Double packing - 1 cc/hr/in stem diameter used	20	0	
	Total			2724	25	RN 10-010

TABLE 6.3-5

MATERIALS EMPLOYED FOR
EMERGENCY CORE COOLING SYSTEM COMPONENTS

Component	Material
Accumulators	Carbon Steel, Clad with Austenitic Stainless Steel
Pumps	
Centrifugal Charging	Austenitic Stainless Steel
Residual Heat Removal	Austenitic Stainless Steel
Residual Heat Exchangers	
Shell	Carbon Steel
Shell End Cap	Carbon Steel
Tubes	Austenitic Stainless Steel
Channel	Austenitic Stainless Steel
Channel Cover	Austenitic Stainless Steel
Tubesheet	Austenitic Stainless Steel
Valves	
Motor Operated Valves Containing Radioactive Fluids	
Pressure Containing Parts	Austenitic Stainless Steel or Equivalent
Body-to-Bonnet Bolting and Nuts	Stainless Steel or Alloy Steel
Seating Surfaces	Stellite Number 6 or Equivalent
Stems	17-4 PH Stainless Steel

RN
99-013

TABLE 6.3-5 (Continued)

MATERIALS EMPLOYED FOR
EMERGENCY CORE COOLING SYSTEM COMPONENTS

Component	Material
Accumulator Check Valves	
Parts Contacting Borated Water	Austenitic Stainless Steel or Alloy Steel
Clapper Arm Shaft	PH Nickel Alloy
Seating Surfaces	Stellite GB
Relief Valves	
Stainless Steel Bodies	Stainless Steel
Carbon Steel Bodies (XVR-8857-SI Only)	Carbon Steel
All Nozzles, Discs, Spindles and Guides	Stainless Steel or Alloy Steel
Bonnets for Stainless Steel Valves Without a Balancing Bellows	Stainless Steel
All Other Bonnets	Carbon Steel
Air Operated Valves	
Parts Contacting Borated Water	Stainless Steel
Seating Surfaces	Stainless Steel or Stellite
Stems	Stainless Steel
Body to Bonnet Bolting and Nuts	Stainless Steel or Alloy Steel
Piping	
All Piping in Contact with Borated Water	Austenitic Stainless Steel

RN
99-013

TABLE 6.3-6

NORMAL OPERATING STATUS OF EMERGENCY CORE COOLING SYSTEM COMPONENTS FOR CORE COOLING

Number of Charging Pumps Operable	2*	
Number of Residual Heat Removal Pumps Operable	2	
Number of Residual Heat Exchangers Operable	2	
Refueling Water Storage Tank Volume		
Minimum Contained Volume, gal	453,800	98-01
Maximum Contained Volume, gal	514,665	98-01
Boron Concentration in Refueling Water Storage Tanks, Minimum, ppm	2,300	
Boron Concentration in Accumulator, Minimum, ppm	2,200	
Number of Accumulators	3	
Nominal Accumulator Pressure, psig	628	00-01
Nominal Accumulator Water Volume, ft^3	1,014	99-01
System Valves, Interlocks, and Piping Required for the Above Components which are Operable	All	

* The third pump is a designated spare and has its circuit breaker(s) racked out.

TABLE 6.3-7

SINGLE ACTIVE FAILURE ANALYSIS FOR EMERGENCY CORE COOLING SYSTEM COMPONENTS

INJECTION MODE 00-01

Component	Malfunction	Comments
1. Pumps		
a. Centrifugal charging	Fails to start	Three provided, evaluation based on operation of one.
b. Residual heat removal	Fails to start	Two provided, evaluation based on operation of one.
2. Automatically Operated Valves		
a. High head injection isolation	Fails to open	Two parallel lines; one valve in either line required to open.
b. Centrifugal Charging Pumps		
i. Suction line from refueling water storage tank	Fails to open	Two parallel lines; only one valve in either line required to open.
ii. Discharge line to the normal charging path	Fails to close	Two valves in series; only one valve required to close.
iii. Suction from volume control tank	Fails to close	Two valves in series; only one valve required to close.
c. Residual Heat Removal Pumps		
i. Minimum flow bypass line	Fails to close	Partial loss of maximum flow from one of two pumps. Evaluation based on flow from only one pump.

TABLE 6.3-7 (Continued)

SINGLE ACTIVE FAILURE ANALYSIS FOR EMERGENCY CORE COOLING SYSTEM COMPONENTS

RECIRCULATION MODE

Component Valves	Malfunction	Comments
1. Residual heat removal pumps suction line to refueling water storage tank	Fails to close	Two parallel lines; each with a gate and check valve. Operation of a gate or check in each line is required.
2. Centrifugal Charging Pump		
a. Suction line from refueling water storage tank	Fails to close	Check valve in series with two parallel gate valves. Operation of either the check valve or the gate valves required.
b. Minimum flow bypass line	Fails to close	One valve in each of three parallel lines backed up by one valve in common line. Only three parallel valves or one common valve required to close. 00-01
3. Residual heat removal pumps suction line from reactor building sump	Fails to open	Two parallel lines, with two valves in series in each line, one pair of valves in either line is required to open.
4. Centrifugal charging/high head pump suction line at discharge of residual heat exchanger	Fails to open	One line provided from discharge of each residual heat removal pump, valve in only one required to open.
5. Residual heat removal cross-connect line	Fails to close	Two valves in series; operation of one required.

TABLE 6.3-7 (Continued)

SINGLE ACTIVE FAILURE ANALYSIS FOR EMERGENCY CORE COOLING SYSTEM COMPONENTS

RECIRCULATION MODE

Component	Malfunction	Comments
6. Residual heat removal cross-connect line	Fails to open	Two valves provided: Hot leg recirculation flow from one RHR pump via one valve adequate.
7. Charging/high head safety injection pump suction header and discharge header cross-connect isolation valves	Fails to close	Two valves in series. One required to close.
8. High head cold leg recirculation header isolation valves	Fails to open	Header containing redundant parallel valves open during safety injection. Other header is redundant and contains one valve that opens. One header required. 98-01 00-01
9. High head hot leg recirculation header isolation valves	Fails to open	Redundant headers in parallel with one valve in each header. One header required.
10. RHR cold leg recirculation/injection isolation valves	Fails to close	One cold leg RHR recirculation isolation valve provided per RHR pump. Hot leg recirculation flow from one pump adequate.
11. RHR hot leg recirculation isolation valve	Fails to open	Sufficient hot leg recirculation flow provided via two charging high head pumps.
12. High head cold leg recirculation header isolation valves	Fails to close	Two headers; one header contains redundant parallel valves that close during hot leg recirculation, the other header contains one valve that closes. One header is required to be isolated.

TABLE 6.3-8

EMERGENCY CORE COOLING SYSTEM RECIRCULATION PIPING PASSIVE FAILURE ANALYSIS

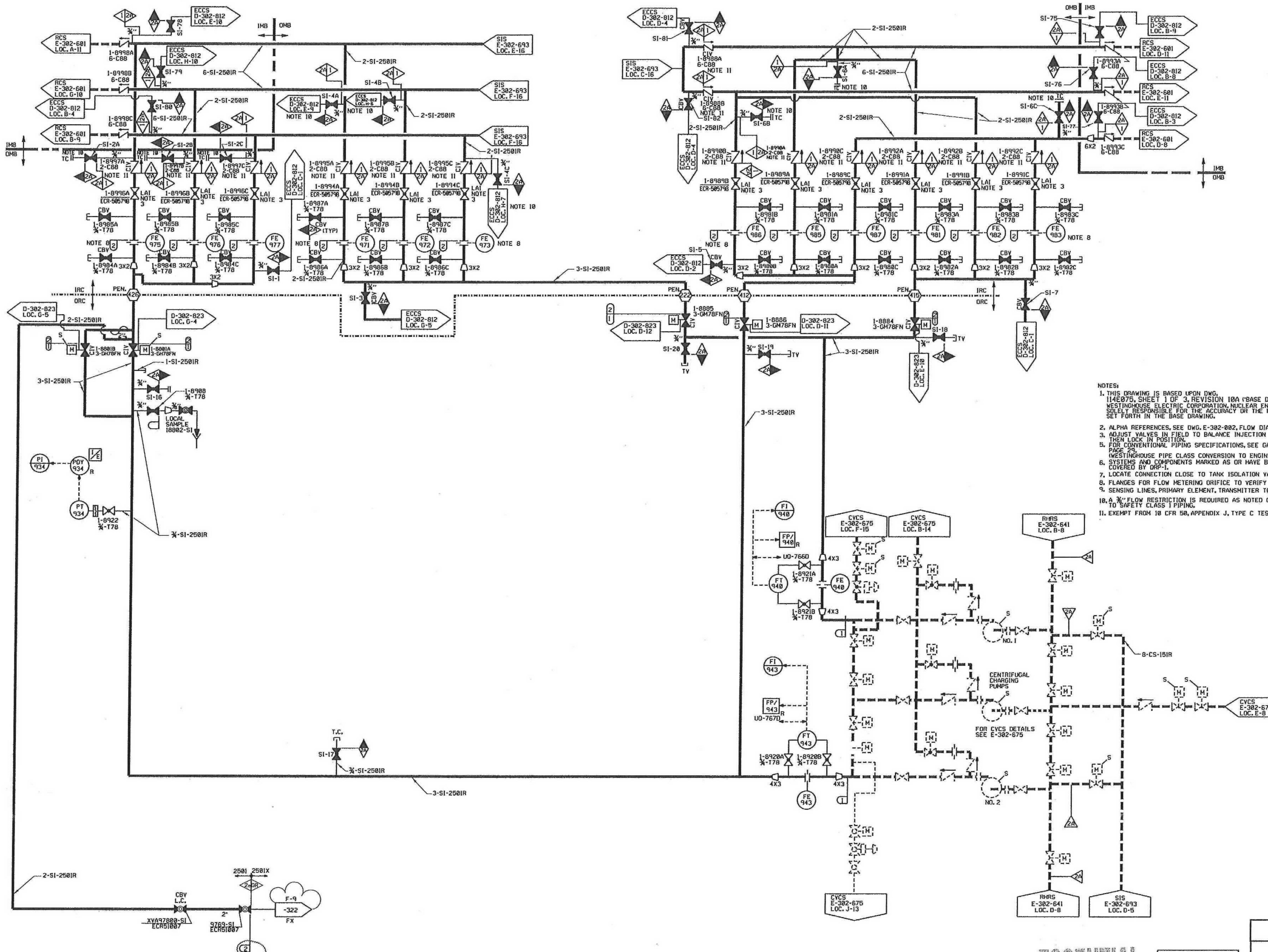
LONG TERM PHASE

Flow Path	Indication of Loss of Flow Path	Alternate Flow Path
Low Head Recirculation		
From reactor building sump to low head injection header via the residual heat removal pumps and the residual heat exchangers	Accumulation of water in a residual heat removal pump compartment or auxiliary building sump actuates an alarm in the control room	Via the independent, identical low head flow path utilizing the second residual heat exchanger and residual heat removal pump
High Head Recirculation		
From reactor building sump to the high head injection header via residual heat removal pump, residual heat exchanger and the high head injection pumps	Accumulation of water in a residual heat removal pump compartment or the auxiliary building sump or charging pump compartments actuates an alarm in the control room	From reactor building sump to the high head injection headers via alternate residual heat removal pump, residual heat exchanger, or charging pump

TABLE 6.3-9

EMERGENCY CORE COOLING SYSTEM SHARED FUNCTIONS EVALUATION

Component	Normal Operating Arrangement	Accident Arrangement	
Refueling Water Storage Tank	Lined up to suction of residual heat removal pumps	Lined up to suction centrifugal charging and residual heat removal pumps. Valves for realignment meet single failure criteria.	
Centrifugal Charging Pumps	Lined up for charging service	Lined up to cold leg injection. Valves for realignment meet single failure criteria.	98-01
Residual Heat Removal Pumps	Lined up to cold legs of reactor coolant piping	Lined up to cold legs of reactor coolant piping	
Residual Heat Exchangers	Lined up to cold legs of reactor coolant piping	Lined up to cold legs of reactor coolant piping	



NOTES:

1. THIS DRAWING IS BASED UPON DWG. 114E075, SHEET 1 OF 3, REVISION 10A ("BASE DRAWING") OF WESTINGHOUSE ELECTRIC CORPORATION, NUCLEAR ENERGY SYSTEMS, PITTSBURGH, PS WHO IS SOLELY RESPONSIBLE FOR THE ACCURACY OR THE RELIABILITY OF THE DESIGN INFORMATION SET FORTH IN THE BASE DRAWING.
2. ALPHA REFERENCES, SEE DWG. E-302-002, FLOW DIAGRAM LEGEND.
3. ADJUST VALVES IN FIELD TO BALANCE INJECTION LINE FLOW AND TO LIMIT PUMP RUNOUT, THEN LOCK IN POSITION.
5. FOR CONVENTIONAL PIPING SPECIFICATIONS, SEE GAI SPECIFICATION SP-329-4461-00, PAGE 29. (WESTINGHOUSE PIPE CLASS CONVERSION TO ENGINEER'S PIPE LINE SPECIFICATION).
6. SYSTEMS AND COMPONENTS MARKED AS OR HAVE BEEN DECLASSIFIED TO QUALITY RELATED AS COVERED BY QRP-1.
7. LOCATE CONNECTION CLOSE TO TANK ISOLATION VALVES.
8. FLANGES FOR FLOW METERING ORIFICE TO VERIFY FLOW DURING PRE-OPERATIONAL TESTING.
9. SENSING LINES, PRIMARY ELEMENT, TRANSMITTER TO BE ENCLOSED IN HEATED CUBICLE.
10. A 3/8" FLOW RESTRICTION IS REQUIRED AS NOTED ON LEGEND FOR ALL TEST CONNECTIONS TO SAFETY CLASS I PIPING.
11. EXEMPT FROM 10 CFR 50, APPENDIX J, TYPE C TESTING BASED ON FSAR TABLE 6.2-53a.

THE ASSOCIATED 307 DRAWING SHALL BE REVISED IN CONJUNCTION WITH REVISION TO THIS 302 IF THE CROSS HATCHED AREA IS AFFECTED.

RN14-022

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1	2890	130	2900	250	3625	60(MIN)
2	2500	130	2735	250	3420	60(MIN)
#	PSIG	°F	PSIG	°F	PSIG	°F
	NORMAL		UPSET		•HYDROTEST	

DESIGN DATA

• HYDROTEST PRESSURE FOR NEW CONSTRUCTION PER ASME CODE SECTION III

1/4"=1'-0"

ESSENTIAL

DRAWING LEGIBILITY CLASS 1

SCE&G CAD ENHANCED

NO.	DATE	BY	REVISION	CKD. BY	LE APPROVAL
11	6/14/2001	TGB	REVISED PER ECR-70001	MGR	TOC
16	3/29/16	JMR	REVISED PER ECR-51007	RHM	NG
15	12/07/15	KO	REVISED PER ECR-51007	MGR	AF
14	11/2/15	RHM	REVISED PER ECR-51007	MGR	AF
13	10/24/06	RHM	REVISED PER ECR-50579	RHM	BB
12	2/17/04	JMR	CADD ENHANCED PER ECR-50239	MGR	DDJ

FSAR Figure 6.3-1, SH. 1

SOUTH CAROLINA ELECTRIC & GAS COMPANY

VIRGIL C. SUMMER NUCLEAR STATION

PIPING SYSTEM FLOW DIAGRAM

SAFETY INJECTION

SCE&G A SCANA COMPANY

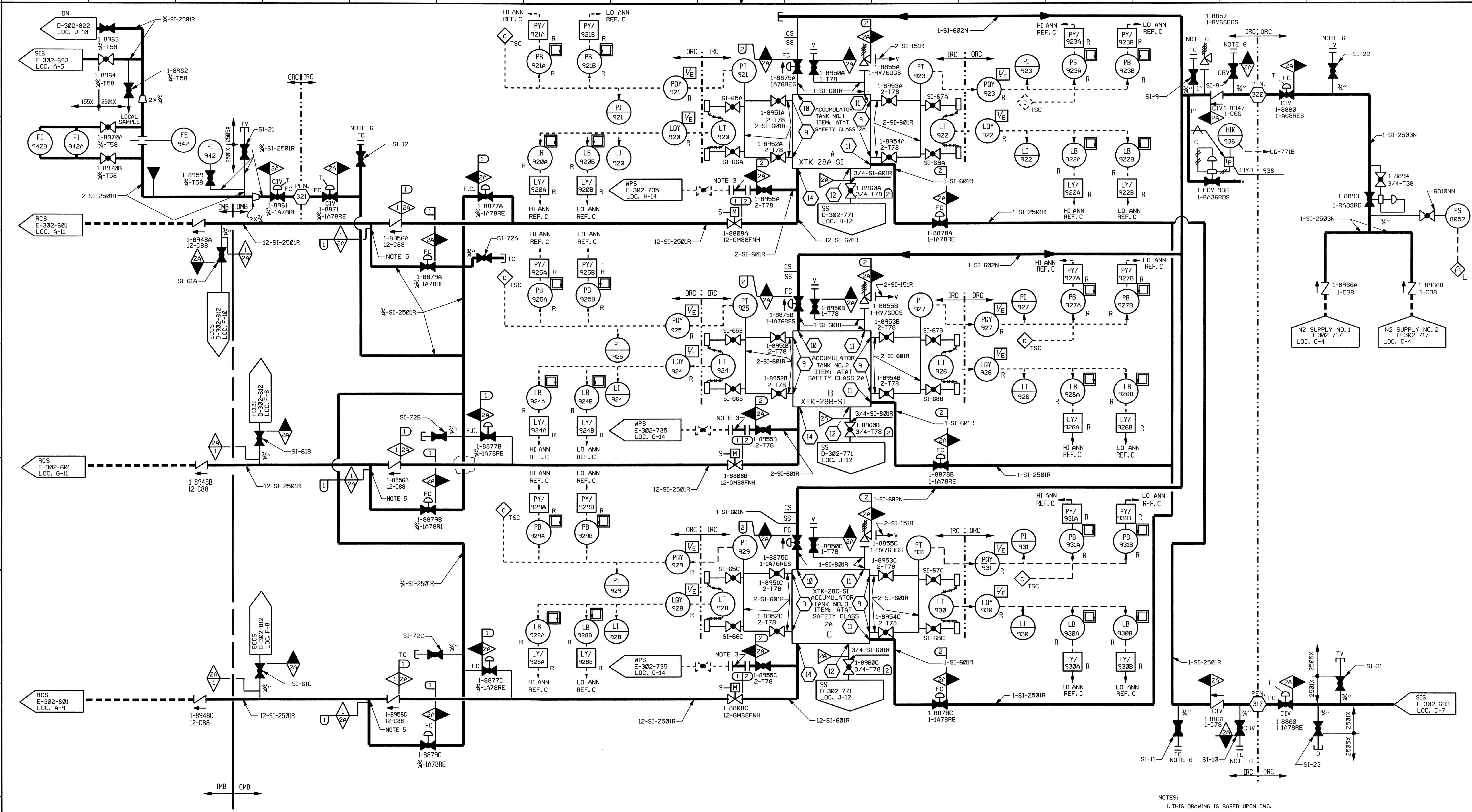
DESIGN ENGINEERING

V. C. SUMMER NUCLEAR STATION JENKINSVILLE, S. C.

MADE	CHECKED	LE APPROVAL
1. A.V.N.	2. M.G.R.	3. R.S.B.

SCALE NO SCALE

DRAWING NUMBER E-302-691 | SHT. NUMBER | REV 16



NOTES:

1. THIS DRAWING IS BASED UPON DWG. 114E075, SHEET 2 OF 3, REVISION 10A ("BASE DRAWING") OF WESTINGHOUSE ELECTRIC CORPORATION, NUCLEAR ENERGY SYSTEMS, PITTSBURGH, PA. WHO IS SOLELY RESPONSIBLE FOR THE ACCURACY OR THE RELIABILITY OF THE DESIGN INFORMATION SET FORTH IN THE BASE DRAWING.
2. FOR ALPHA REFERENCES, SEE DWG. E-302-002, FLOW DIAGRAM LEGEND.
3. SPOOL PIECE TO BE REPLACED WITH BLANK FLANGES, EXCEPT DURING ACCUMULATOR DRAINING.
4. FOR CONVENTIONAL PIPING SPECIFICATIONS, SEE GAI SPECIFICATION SP-329-4461-00, PAGE 29, (WESTINGHOUSE PIPE CLASS CONVERSION TO ENGINEER'S PIPE LINE SPECIFICATION).
5. A 3/8" FLOW RESTRICTION IS REQUIRED AS NOTED ON LEGEND.
6. FLANGE HAS BEEN DRILLED AND TAPPED TO ACCEPT A 3/8" COMPRESSION FITTING TO FACILITATE LLRT WORK. THE FITTING SHOULD BE CAPPED AT OTHER TIMES.

SI	SI-18801	SI-18804
SI	SI-1	SI-87
SYSTEM SUFFIX	FIRST NO.	LAST NO.

VALVE NUMBERING

2	660	120	700	120	875	35(MIN)
1	660	120	2485	120	3107	35(MIN)
#	PSIG	F	PSIG	F	PSIG	F
	NORMAL		UPSET		HYDROTEST	

DESIGN. DATA

ESSENTIAL

RN 11-017

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FSAR Figure 6.3-1, SH. 2

DRAWING LEGIBILITY CLASS 1
SCE&G CAD ENHANCED

SOUTH CAROLINA ELECTRIC & GAS COMPANY
VIRGIL C. SUMMER NUCLEAR STATION

PIPING SYSTEM FLOW DIAGRAM
SAFETY INJECTION

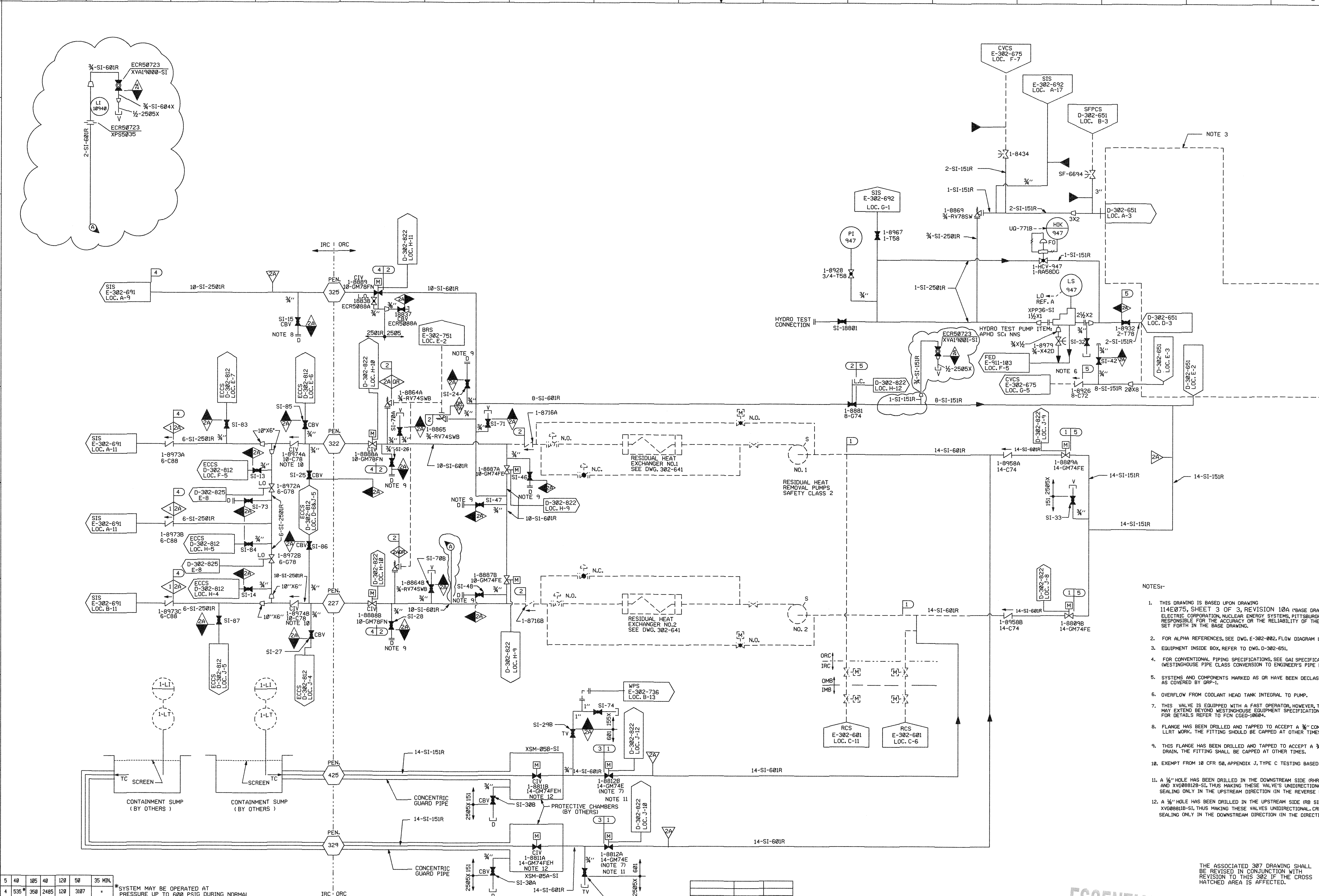
SCE&G A SCANA COMPANY
DESIGN ENGINEERING
V. C. SUMMER NUCLEAR STATION JENKINSVILLE, S. C.

MADE	CHECKED	LE APPROVAL
1. DDJ	2. MGR	3. GAL

SCALE NO SCALE | DRAWING NUMBER E-302-692 | SHT. NUMBER | REV 14

NO.	DATE	BY	REVISION	CKD. BY	LE APPROVAL
11	11/12/97	AVN	REVISED PER CGSS-97-0579	LEK	MGR
10	12/14/95	AVN	REVISED PER MRF-22640	MGR	JEW
9	11/16/94	JMH	REVISED BORDER PER CGSS-94-1680	RHM	MGR
14	6/14/11	RHM	REVISED PER ECR-71578	MGR	GK
13	4/23/09	JTS	CADD ENHANCED PER ECR-50239	MGR	DDJ
12	6/27/05	JMR	REVISED PER NCN-05-2217	MGR	DD

1/4" = 1'-0" 0' 5' 10' 15' 20'



NOTES:-

1. THIS DRAWING IS BASED UPON DRAWING 114E075, SHEET 3 OF 3, REVISION 10A ("BASE DRAWING") OF WESTINGHOUSE ELECTRIC CORPORATION, NUCLEAR ENERGY SYSTEMS, PITTSBURGH, PA. WHO IS SOLELY RESPONSIBLE FOR THE ACCURACY OR THE RELIABILITY OF THE DESIGN INFORMATION SET FORTH IN THE BASE DRAWING.
2. FOR ALPHA REFERENCES, SEE DWG. E-302-002, FLOW DIAGRAM LEGEND.
3. EQUIPMENT INSIDE BOX, REFER TO DWG. D-302-651.
4. FOR CONVENTIONAL PIPING SPECIFICATIONS, SEE GAI SPECIFICATION SP-329-4461-00, PAGE 29, (WESTINGHOUSE PIPE CLASS CONVERSION TO ENGINEER'S PIPE LINE SPECIFICATION).
5. SYSTEMS AND COMPONENTS MARKED AS QR HAVE BEEN DECLASSIFIED TO QUALITY RELATED AS COVERED BY QRP-1.
6. OVERFLOW FROM COOLANT HEAD TANK INTEGRAL TO PUMP.
7. THIS VALVE IS EQUIPPED WITH A FAST OPERATOR, HOWEVER, THE STROKE TIME MAY EXTEND BEYOND WESTINGHOUSE EQUIPMENT SPECIFICATION 677473. FOR DETAILS REFER TO FCN CGEO-10604.
8. FLANGE HAS BEEN DRILLED AND TAPPED TO ACCEPT A 3/8" COMPRESSION FITTING TO FACILITATE LLRT WORK. THE FITTING SHOULD BE CAPPED AT OTHER TIMES.
9. THIS FLANGE HAS BEEN DRILLED AND TAPPED TO ACCEPT A 3/4" COMPRESSION FITTING TO FACILITATE DRAIN. THE FITTING SHALL BE CAPPED AT OTHER TIMES.
10. EXEMPT FROM 10 CFR 50, APPENDIX J, TYPE C TESTING BASED ON FSAR TABLE 6.2-53a.
11. A 1/8" HOLE HAS BEEN DRILLED IN THE DOWNSTREAM SIDE (RHR SIDE) OF THE DISC'S FOR XVG08812A-SI AND XVG08812B-SI, THUS MAKING THESE VALVE'S UNIDIRECTIONAL. CREDIT CAN BE TAKEN FOR THESE VALVE'S SEALING ONLY IN THE UPSTREAM DIRECTION (IN THE REVERSE DIRECTION OF THE SYSTEM FLOW ARROW).
12. A 1/8" HOLE HAS BEEN DRILLED IN THE UPSTREAM SIDE (RB SIDE) OF THE DISC'S FOR XVG08811A-SI AND XVG08811B-SI, THUS MAKING THESE VALVES UNIDIRECTIONAL. CREDIT CAN BE TAKEN FOR THESE VALVES SEALING ONLY IN THE DOWNSTREAM DIRECTION (IN THE DIRECTION OF THE SYSTEM FLOW ARROW).

RN 09-003

THE ASSOCIATED 307 DRAWING SHALL BE REVISED IN CONJUNCTION WITH REVISION TO THIS 302 IF THE CROSS HATCHED AREA IS AFFECTED.

THIS IS A NUCLEAR SAFETY RELATED DOCUMENT. NO DEVIATION SHALL BE INITIATED OR PERFORMED WITHOUT PRIOR DOCUMENTATION AND WRITTEN APPROVAL

5	40	105	40	120	50	35 MIN.
4	535*	350	2485	120	3107	"
3	50	250	50	250	63	"
2	535*	350	600	350	750	"
1	400	350	450	350	563	"
P	PSIG	°F	PSIG	°F	PSIG	°F
	NORMAL		UPSET		HYDRO-TEST	
DESIGN DATA						

*SYSTEM MAY BE OPERATED AT PRESSURE UP TO 600 PSIG DURING NORMAL PLANT OPERATIONS. REFER TO DESIGN SPECIFICATION DSP-544EC, TABLE 3.

1/4" = 1'-0"   0'   5'   10'   15'   20'

SI	SI-18801	SI-18838
SI	SI-1	SI-87
SYSTEM SUFFIX	FIRST NO.	LAST NO.

VALVE NUMBERING

ESSENTIAL

DRAWING LEGIBILITY CLASS 1

NO.	DATE	BY	REVISION	CKD. BY	LE APPROVAL
20	2/26/04	JMR	CADD ENHANCED PER ECR-50239	MGR	DDJ
19	6/13/2001	TGB	REVISED PER ECR-70001	MGR	TOC
18	01/11/00	JTS	REVISED PER NCN 99-0071	MGR	CCB
17	4/23/99	DDJ	REVISED PER ECR-50088A	RHM	AME
22	11/6/09	JTS	REVISED PER ECR-50723	DDJ	WTW
21	7/20/05	TGB	REVISED PER ECR-70514	MGR	LWH

FSAR Figure 6.3-1, SH. 3

SOUTH CAROLINA ELECTRIC & GAS COMPANY

VIRGIL C. SUMMER NUCLEAR STATION

PIPING SYSTEM FLOW DIAGRAM

SAFETY INJECTION

SCE&G A SCANA COMPANY

DESIGN ENGINEERING

V. C. SUMMER NUCLEAR STATION JENKINSVILLE, S. C.

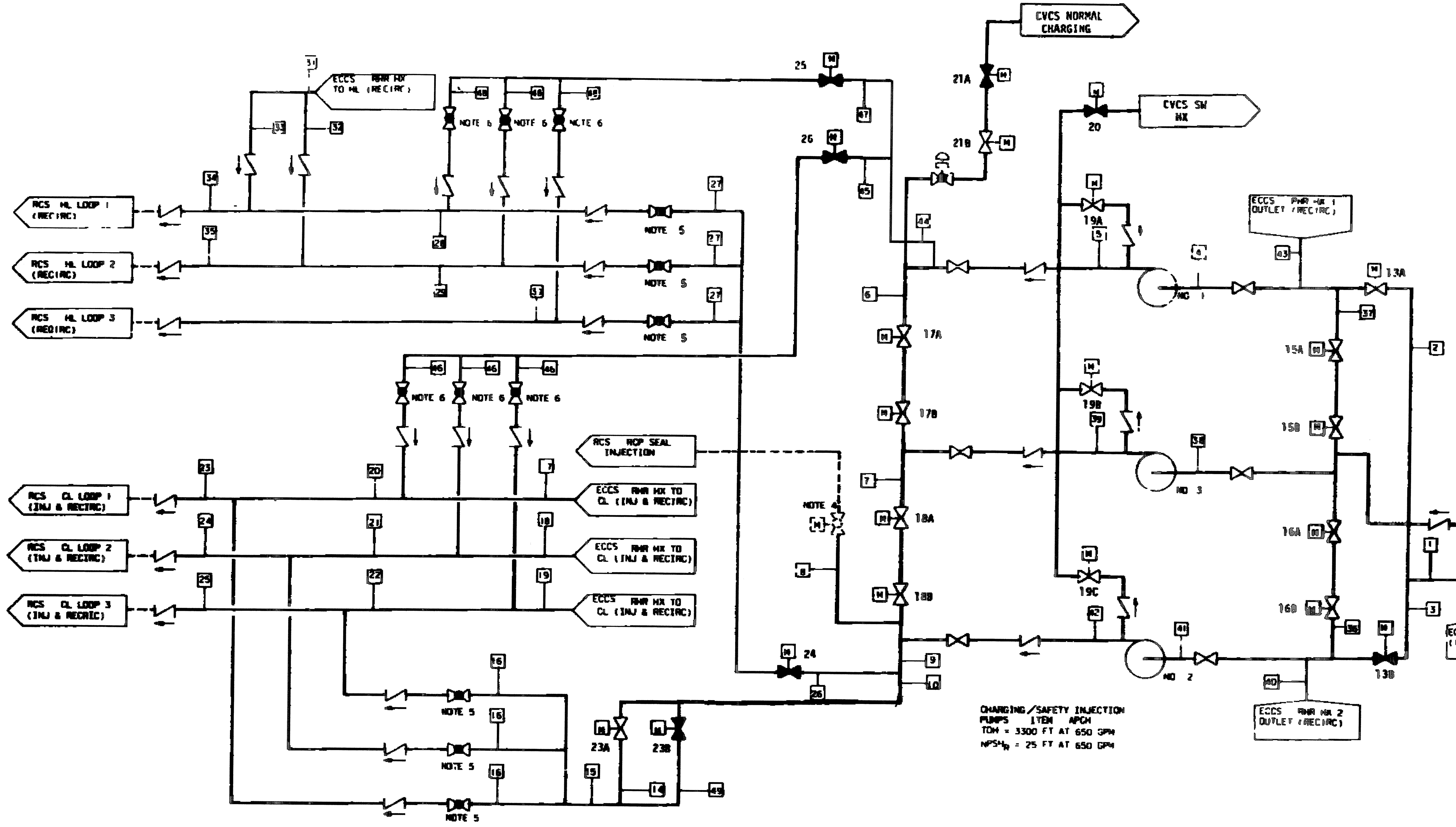
MADE	CHECKED	LE APPROVAL
1. SRM	2. DVW	3. DJC

SCALE NO SCALE

E-302-693 (DRAWING NUMBER)

SHT. NUMBER

REV 22



NOTES

1. PIPING SIZE AND CLASS HAVE NOT BEEN INDICATED REFER TO (R) PIPING AND INSTRUMENTATION DIAGRAM
2. MINIMUM SAFEGUARDS CONSISTS OF ONE OF TWO REDUNDANT SAFEGUARDS TRAINS OPERABLE
3. ITEMIZED FLOWS ARE PUMP RUNOUT FLOWS BASED ON 0 PSIG REACTOR COOLANT SYSTEM PRESSURE. ACTUAL ECCS FLOWS VARY DURING ECCS OPERATION DEPENDENT ON THE REACTOR COOLANT SYSTEM AND CONTAINMENT PRESSURE TRANSIENT
4. SEAL INJECTION PATH REMAINS OPEN DURING THE INJECTION AND RECIRCULATION MODES OF OPERATION.
5. THROTTLE VALVES ADJUSTED TO EQUALIZE PATH FLOWS WHILE RESTRICTING ONE CHARGING/SAFETY INJECTION PUMP RUNOUT TO 688 GPM WITH SEAL INJECTION PATH OPEN
6. EACH SET OF THROTTLE VALVES ADJUSTED TO EQUALIZE PATH FLOWS WHILE RESTRICTING ONE CHARGING/SAFETY INJECTION PUMP RUNOUT TO 688 GPM WITH SEAL INJECTION PATH OPEN.
7. THESE PATHS CONSTITUTE SECTIONS OF THE REDUNDANT SAFEGUARDS TRAIN ASSUMED TO BE INOPERABLE DURING MINIMUM SAFEGUARDS OPERATION
8. NIS - NOT IN SERVICE
9. VALVE ALIGNMENT SHOWN IS FOR THE COLD LEG SAFETY INJECTION OPERATIONAL MODE
10. VALVE ALIGNMENTS ARE PROVIDED IN "NOTES TO FIGURE 6.3-2" FOR PRINCIPLE MODES OF ECCS OPERATION
11. INFORMATION IN THIS TABLE IS HISTORICAL & DOES NOT REPRESENT CURRENT DESIGN.

COLD LEG SAFETY INJECTION (MINIMUM SAFEGUARDS - NOTES 2,11)

LOCATION	1	2	3	4	5	6	7	8	9	10
FLOW - GPM	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]
PRESSURE - PSIG	[illegible]			[illegible]	[illegible]					
TEMPERATURE - °F	[illegible]									

COLD LEG RECIRCULATION (MINIMUM SAFEGUARDS - NOTES 2,11)

LOCATION	[illegible]
FLOW - GPM	[illegible]
PRESSURE - PSIG	[illegible]
TEMPERATURE - °F	[illegible]

HOT LEG RECIRCULATION (MINIMUM SAFEGUARDS - NOTES 2,11)

LOCATION	[illegible]
FLOW - GPM	[illegible]
PRESSURE - PSIG	[illegible]
TEMPERATURE - °F	[illegible]

[illegible]	[illegible]

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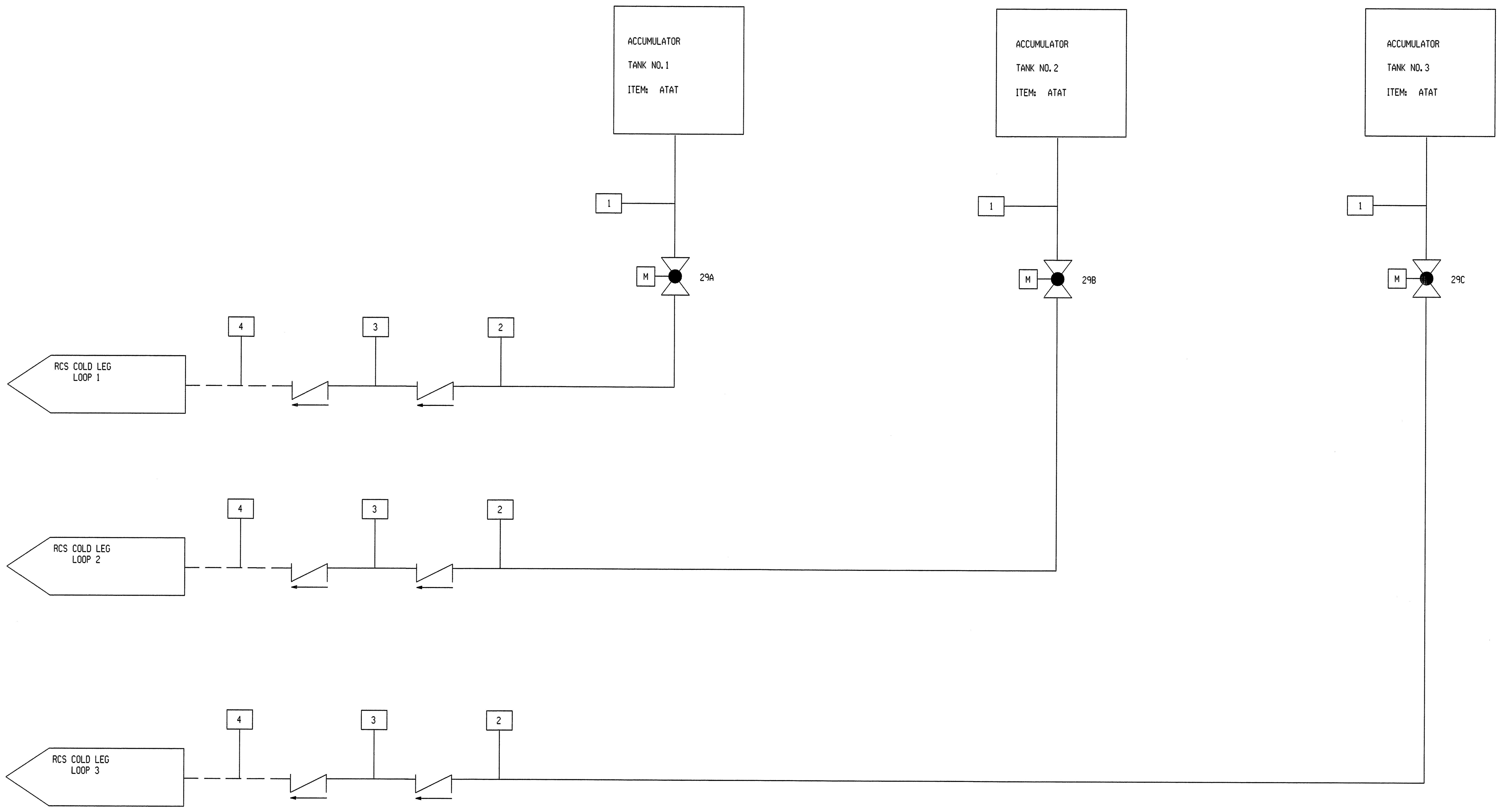
SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Emergency Core Cooling System Process Flow Diagram (Sheet 1 of 3)

Figure 6.3-2

REV. 1

ACCUMULATOR TANKS
NOMINAL TANK VOLUME - 1450 $FT.^3$
NOMINAL WATER VOLUME - 1014 $FT.^3$
NOMINAL GAS PRESSURE - 640 PSIG
NOMINAL TEMPERATURE - 120°F



NOTES:

1. PIPING SIZE AND CLASS HAVE NOT BEEN INDICATED. REFER TO N PIPING AND INSTRUMENTATION DIAGRAM.
2. ACCUMULATORS DISCHARGE TO THE REACTOR COOLANT SYSTEM DURING THE REACTOR COOLANT SYSTEM BLOWDOWN PRESSURE TRANSIENT. ACTUAL FLOW RATES DEPEND ON TYPE OF LOSS OF COOLANT ACCIDENT AND SUBSEQUENT PRESSURE TRANSIENT. FLOW DATA PRESENTED IS THE MAXIMUM FLOW THAT OCCURS FOR DOUBLE ENDED. GUILLOTINE COLD LEG LOSS OF COOLANT ACCIDENT.
3. VALVE ALIGNMENTS ARE PROVIDED IN "NOTES TO FIGURE 6.3-2" FOR PRINCIPLE MODES OF ECCS OPERATION.
4. INFORMATION IN THIS TABLE IS HISTORICAL & DOES NOT REPRESENT CURRENT DESIGN.

TYPICAL ACCUMULATOR BLOWDOWN (DOUBLE-ENDED GUILLOTINE COLD LEG LOSS OF COOLANT ACCIDENT - NOTES 2,4)

LOCATION	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16	17	18	19	20	21	22
FLOW - RPM	19000	19000	19000	19000																		
PRESSURE - PSIG	NOTE 2	NOTE 2	NOTE 2	NOTE 2																		
TEMPERATURE - °F	120	120	120	120																		

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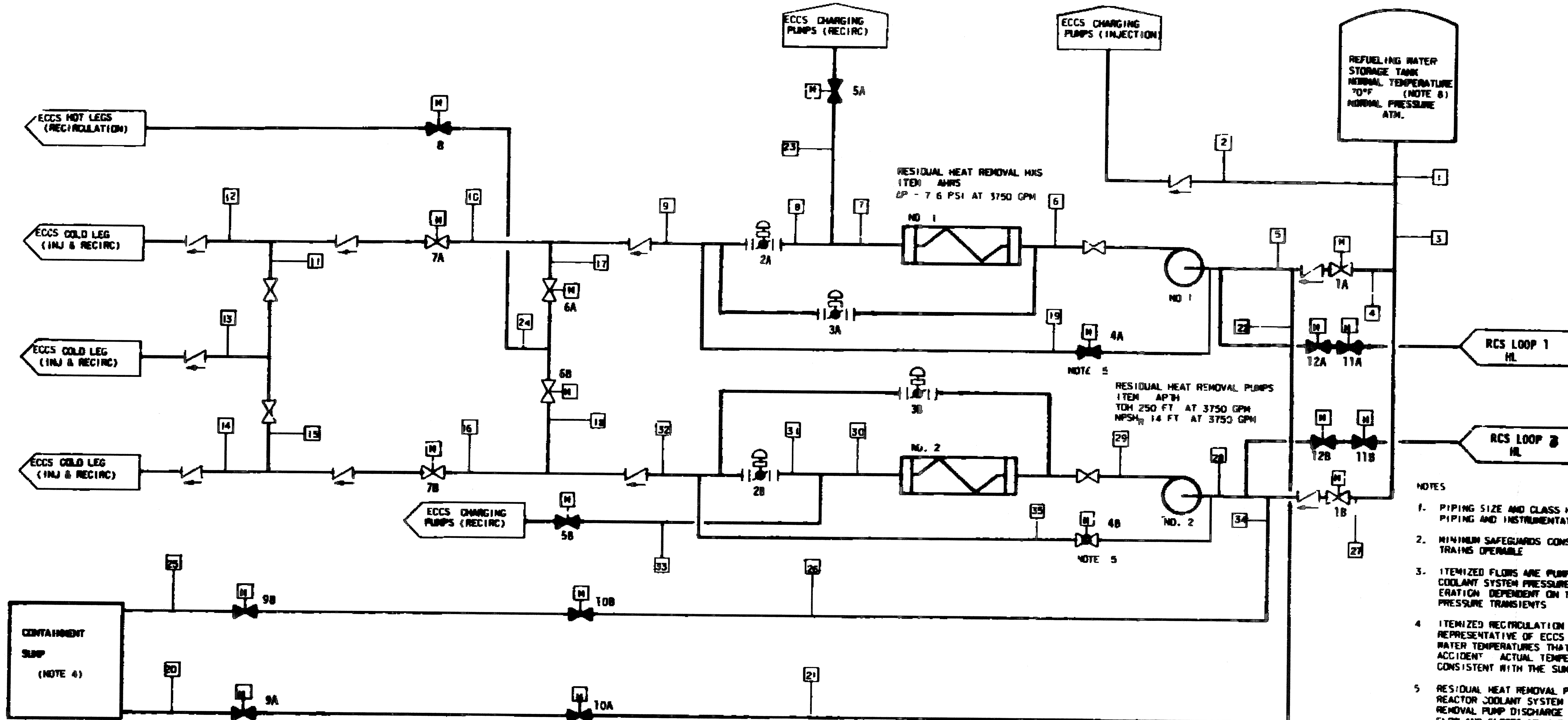
SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

EMERGENCY CORE COOLING SYSTEM
PROCESS FLOW DIAGRAM
(Sheet 2 OF 3)

Figure 6.3-2

REV.0



COLD LEG SAFETY INJECTION (MINIMUM SAFEGUARDS - NOTES 2,13)

LOCATION	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16	17	18	19	20	21	22	23	24	25	26	27	28	29	30	31	32	33	34	35
FLOW - GPM	[illegible]	[illegible]	3750	3750	3750				3750	[illegible]	[illegible]	1267	1275	1197	576	1773	1773	1773	0 NOTE 5	N/S						N/S	N/S NOTE 6					N/S NOTE 6	N/S	N/S	[illegible]
PRESSURE-PSIG	0 NOTE 9				0	106				[illegible]		[illegible]	42	[illegible]		[illegible]																			
TEMPERATURE-°F	70																	70																	

COLD LEG RECIRCULATION (MINIMUM SAFEGUARDS - NOTES 2,13)

LOCATION	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16	17	18	19	20	21	22	23	24	25	26	27	28	9	30	31	32	33	34	35
FLOW - GPM	N/S			N/S	3019	3019	3019	[illegible]	3161	3161	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	N/S NOTE 6	N/S	N/S	0 NOTE 5	3019		3019	[illegible]	[illegible]	N/S NOTE 6	N/S NOTE 6	N/S	N/S NOTE 6							N/S NOTE 6
PRESSURE - PSIG					12	118				57		[illegible]	23	18						0 NOTE 10															
TEMPERATURE - °F					[illegible] NOTE 4	[illegible] NOTE 4	[illegible] NOTE 4								146 NOTE 4					[illegible] NOTE 4		[illegible] NOTE 4	145 NOTE 4												

HOT LEG RECIRCULATION (MINIMUM SAFEGURADS - NOTES 2,13)

LOCATION	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16	17	18	19	20	21	22	23	24	25	26	27	28	29	30	31	32	33	34	35
FLOW - GPM	N/S			N/S	3710	3710	3710	[illegible]	[illegible]	N/S						N/S	3952	N/S NOTE 6	0 NOTE 5	3710	3710	3710	[illegible]	3952	N/S NOTE 6	N/S NOTE 6	N/S	N/S NOTE 6							N/S NOTE 6
PRESSURE - PSIG					12	[illegible]														0 NOTE 10				42											
TEMPERATURE-°F					146 NOTE 4	146 NOTE 4	125 NOTE 4	125 NOTE 4	125 NOTE 4								125 NOTE 4			146 NOTE 4		146 NOTE 4	125 NOTE 4	125 NOTE 4											

NOTES

1. PIPING SIZE AND CLASS HAVE NOT BEEN INDICATED. REFER TO PIPING AND INSTRUMENTATION DIAGRAM
2. MINIMUM SAFEGUARDS CONSISTS OF ONE OF TWO REDUNDANT SAFEGUARDS TRAINS OPERABLE
3. ITEMIZED FLOWS ARE PUMP RUNOUT FLOWS BASED ON 0 PSIG REACTOR COOLANT SYSTEM PRESSURE. ACTUAL ECCS FLOWS VARY DURING ECCS OPERATION DEPENDENT ON THE REACTOR COOLANT SYSTEM AND CONTAINMENT PRESSURE TRANSIENTS
4. ITEMIZED RECIRCULATION OPERATIONAL MODE TEMPERATURES ARE REPRESENTATIVE OF ECCS OPERATION AND ARE BASED ON CONTAINMENT SUMP WATER TEMPERATURES THAT CAN EXIST FOLLOWING A LOSS OF COOLANT ACCIDENT. ACTUAL TEMPERATURES VARY DURING THE RECIRCULATION MODE CONSISTENT WITH THE SUMP WATER TEMPERATURE TRANSIENT
5. RESIDUAL HEAT REMOVAL PUMP MINIFLOW PROVIDES A FLOW PATH WHEN REACTOR COOLANT SYSTEM PRESSURE IS HIGHER THAN RESIDUAL HEAT REMOVAL PUMP DISCHARGE HEAD. ISOLATION VALVE OPENS AT LOW PUMP FLOW AND CLOSES AT HIGH PUMP FLOW
6. THESE PATHS CONSTITUTE SECTIONS OF THE REDUNDANT SAFEGUARDS TRAIN ASSUMED TO BE INOPERABLE DURING MINIMUM SAFEGUARDS OPERATION
7. VALVE ALIGNMENT SHOWN IS FOR THE COLD LEG SAFETY INJECTION OPERATIONAL MODE
8. NORMAL RWST TEMPERATURE RANGE IS 40°F 120°F
9. ITEMIZED PRESSURES ARE BASED ON 0 PSIG AT THE REFUELING WATER STORAGE TANK CONNECTION
10. ITEMIZED PRESSURES ARE BASED ON 0 PSIG AT THE CONTAINMENT SUMP RECIRCULATION CONNECTION. ACTUAL PRESSURE VARIES WITH THE CONTAINMENT PRESSURE TRANSIENT
11. N/S NOT IN SERVICE
12. VALVE ALIGNMENTS ARE PROVIDED IN "NOTES TO FIGURE 6.3-2" FOR PRINCIPLE MODES OF ECCS OPERATION
13. INFORMATION IN THIS TABLE IS HISTORICAL & DOES NOT REPRESENT CURRENT DESIGN.

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June 2003

SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Emergency Core Cooling System
Process Flow Diagram
(Sheet 3 of 3)

Figure 6.3-2

REV. 1

NOTES TO FIGURE 6.3-2 (Sheet 1 of 4)

1. Westinghouse process flow diagrams are provided for illustrative purposes only and are not intended to represent the flow rates used by Westinghouse in various accident analyses. Flow rates to the RCS are provided in Chapter 15, where appropriate. The process flow diagrams are developed to provide representative best estimate system performance data based on minimum safeguards systems alignment. The flow rates in the FSAR accident analyses are developed based on pump test curves degraded by 5 - 7 percent (see Section 6.3.3.1) and worst case assumptions pertaining to spilling line system resistances (e.g., maximum allowable resistances in lines connected to unbroken loops and minimum allowable resistances in the line connected to the broken loop) and RCS pressures. 00-01

2. The minimum safeguards alignments are for Train A components only. Minimum alignments assume a loss of power to Train B at the beginning of the event.

3. Charging pump suction and discharge valves are listed as Closed during the recirculation mode. In actuality, two suction and two discharge header valves will be open, depending on which charging pumps are in operation to establish two separate high head recirculation discharge flowpaths. (Reference Table 6.3-3) 00-01

4. RHR minimum recirculation valves are controlled automatically. During periods of low RHR injection flow rates, these valves open to maintain pump minimum flow requirements. (Reference Section 6.3.2.2.4.1)

5. In order to minimize procedure complexity, the preferred lineup for hot leg recirculation is to align the Charging SI pumps to the RCS cold legs and the RHR SI pumps to the hot legs. Adequate core cooling is ensured with either the preferred lineup or by aligning Charging SI pumps to the hot legs and the RHR SI pumps to the cold legs. RN 03-006

NOTES TO FIGURE 6.3-2 (Sheet 2 of 4)

VALVE ALIGNMENT FOR PRINCIPLE MODES OF ECCS OPERATION

Valve No.	A Normal Standby	B Injection Maximum Safeguards	C Injection Minimum Safeguards (Train A Only - See Note 2)	D Cold Leg Recirculation Maximum Safeguards	E Cold Leg Recirculation Minimum Safeguards (Train A Only - See Note 2)	F Hot Leg Recirculation Maximum Safeguards (See Note 5)	G Hot Leg Recirculation Minimum Safeguards (Train A Only - See Note 2 and Note 5)	
								RN 03-006
1A	O	O	O	C	C	C	C	
1B	O	O	O	C	O	C	O	
2A	O	O	O	O	O	O	O	
2B	O	O	O	O	O	O	O	
3A	C	C	C	C	C	C	C	
3B	C	C	C	C	C	C	C	
4A (Note 4)	O	C	C	C	C	C	C	00-01
4B (Note 4)	O	C	O	C	O	C	O	
5A	C	C	C	O	O	O	O	
5B	C	C	C	O	C	O	C	
6A	O	O	O	C	C	C	C	
6B	O	O	O	C	O	C	C	
7A	O	O	O	O	O	O	C	RN 03-006
7B	O	O	O	O	O	O	O	
8	C	C	C	C	C	C	C	
9A	C	C	C	O	O	O	O	
9B	C	C	C	O	C	O	C	

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Amendment 00-01

NOTES TO FIGURE 6.3-2 (Sheet 3 of 4)

VALVE ALIGNMENT FOR PRINCIPLE MODES OF ECCS OPERATION

Valve No.	A Normal Standby	B Injection Maximum Safeguards	C Injection Minimum Safeguards (Train A Only - See Note 2)	D Cold Leg Recirculation Maximum Safeguards	E Cold Leg Recirculation Minimum Safeguards (Train A Only - See Note 2)	F Hot Leg Recirculation Maximum Safeguards (See Note 5)	G Hot Leg Recirculation Minimum Safeguards (Train A Only - See Note 2 and Note 5)	
								RN 03-006
10A	C	C	C	O	O	O	O	
10B	C	C	C	O	C	O	C	
11A	C	C	C	C	C	C	C	
11B	C	C	C	C	C	C	C	
12A	C	C	C	C	C	C	C	
12B	C	C	C	C	C	C	C	
13A	C	O	O	C	C	C	C	
13B	C	O	C	C	C	C	C	
14A	O	C	O	C	O	C	O	
14B	O	C	C	C	C	C	C	
15A (Note 3)	O	O	O	C	C	C	C	
15B (Note 3)	O	O	O	C	O	C	O	
16A (Note 3)	O	O	O	C	C	C	C	
16B (Note 3)	O	O	O	C	O	C	O	00-01
17A (Note 3)	O	O	O	C	C	C	C	
17B (Note 3)	O	O	O	C	O	C	O	

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Amendment 00-01

NOTES TO FIGURE 6.3-2 (Sheet 4 of 4)

VALVE ALIGNMENT FOR PRINCIPLE MODES OF ECCS OPERATION

Valve No.	A Normal Standby	B Injection Maximum Safeguards	C Injection Minimum Safeguards (Train A Only - See Note 2)	D Cold Leg Recirculation Maximum Safeguards	E Cold Leg Recirculation Minimum Safeguards (Train A Only - See Note 2)	F Hot Leg Recirculation Maximum Safeguards (See Note 5)	G Hot Leg Recirculation Minimum Safeguards (Train A Only - See Note 2 and Note 5)	RN 03-006
18A (Note 3)	O	O	O	C	C	C	C	
18B (Note 3)	O	O	O	C	O	C	O	
19A	O	O	O	C	O	C	O	
19B	O	O	O	C	O	C	O	
19C	O	O	O	C	O	C	O	
20	O	O	O	C	C	C	C	
21A	O	C	C	C	C	C	C	
21B	O	C	O	C	O	C	O	Amend 3
23A	C	O	O	O	O	C	C	
23B	C	O	C	O	C	C	C	
24	C	C	C	C	C	O	C	
25	C	C	C	C	C	O	O	
26	C	C	C	O	C	C	C	Amend 3
29A	O	O	O	O	O	O	O	
29B	O	O	O	O	O	O	O	
29C	O	O	O	O	O	O	O	

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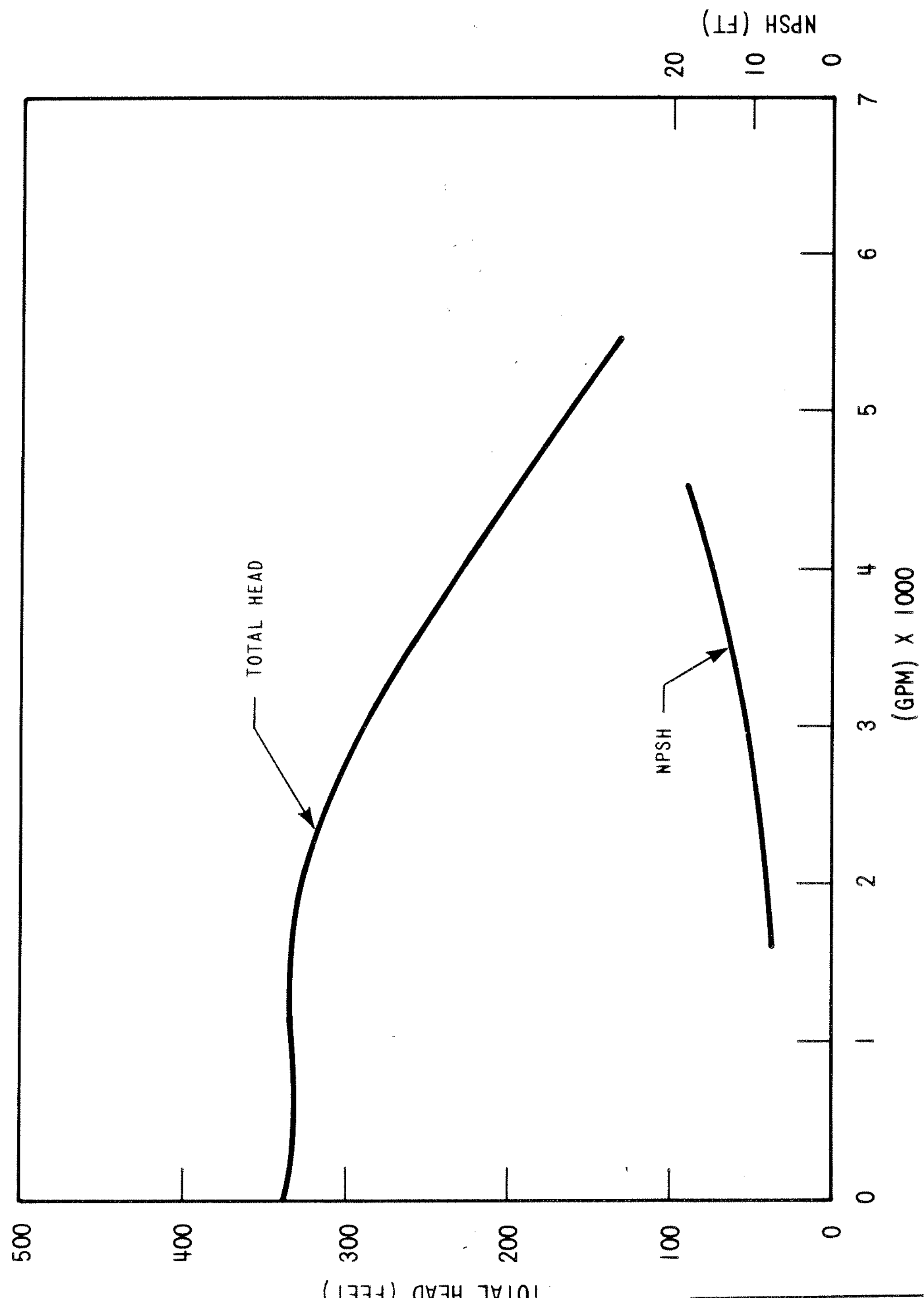
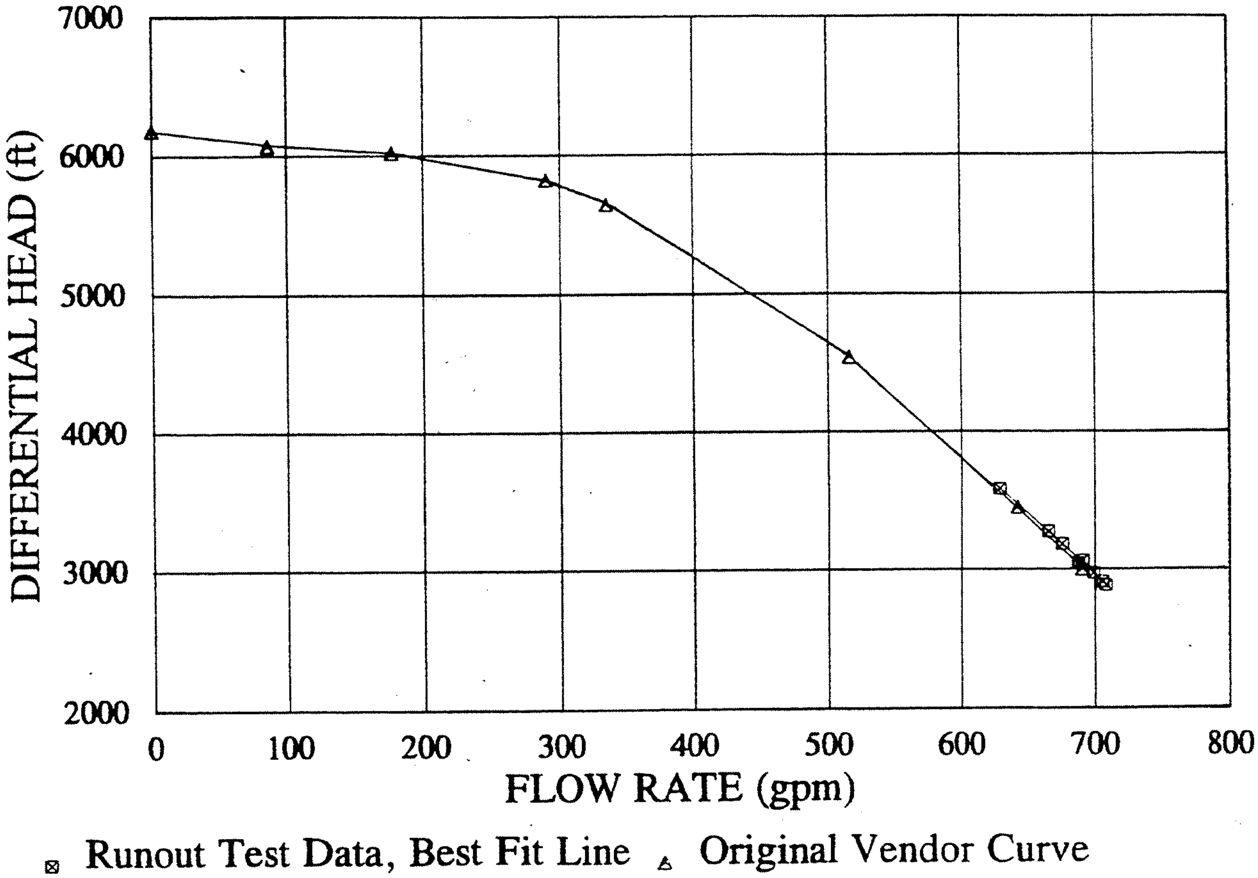
Amendment 00-01



SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Performance Curve for Residual Heat Removal Pump

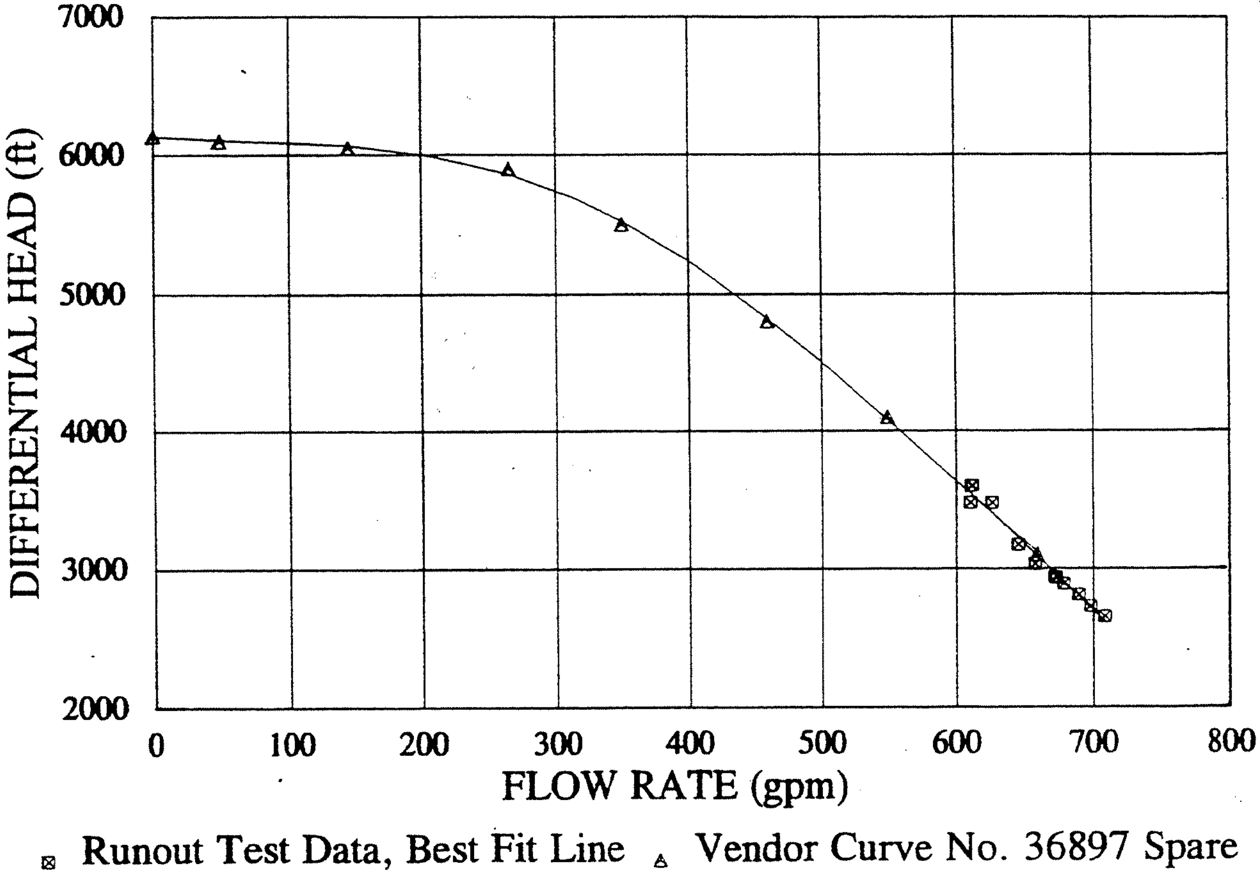
Figure 6.3-3



SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

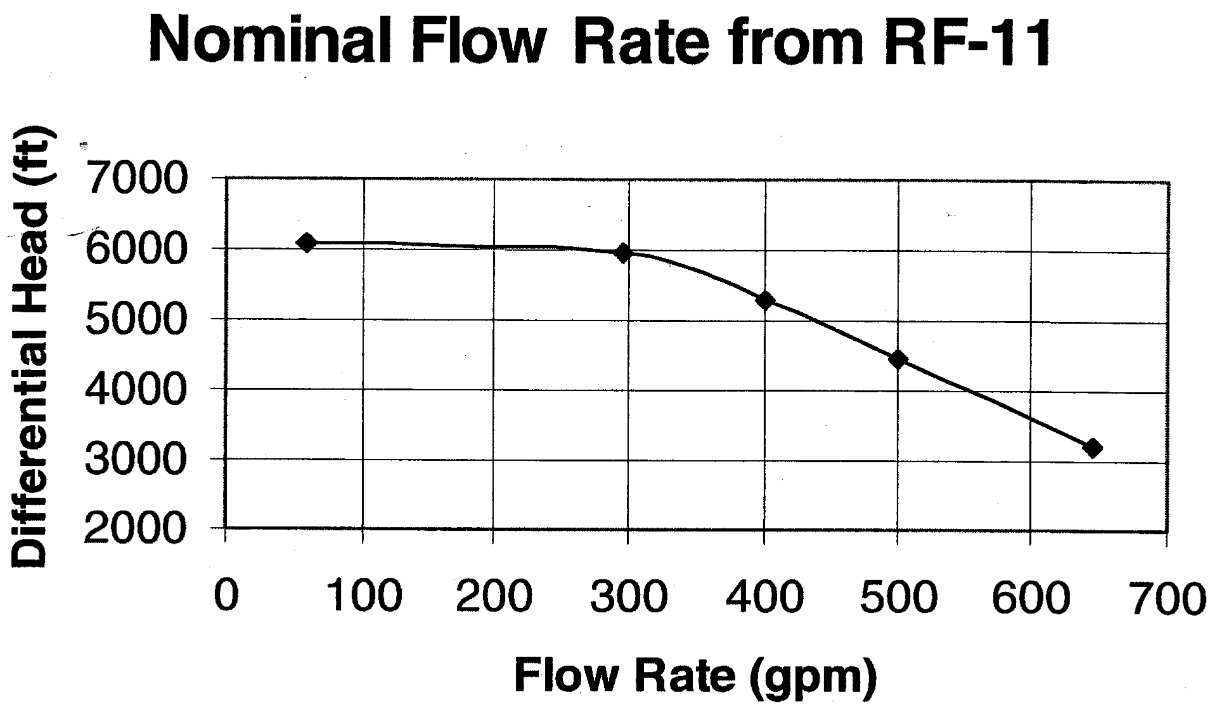
Performance Curve For Centrifugal
Charging Pump A

Figure 6.3-4a



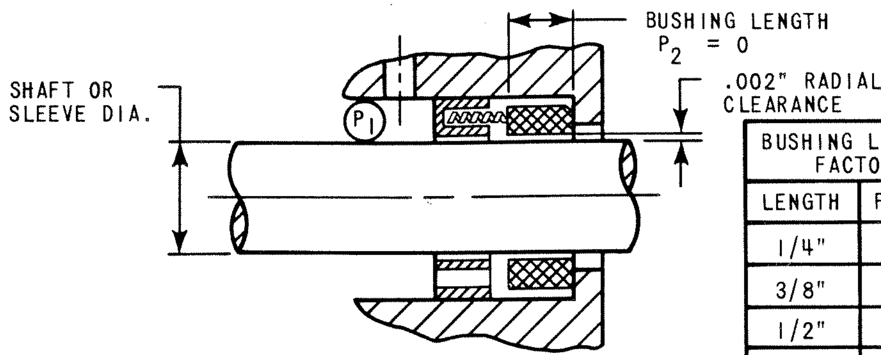
SOUTH CAROLINA ELECTRIC & GAS CO. VIRGIL C. SUMMER NUCLEAR STATION
Performance Curve For Centrifugal Charging Pump B Figure 6.3-4b

Amendment 96-02
July 1996



AMENDMENT 02-01
MAY 2002

SOUTH CAROLINA ELECTRIC AND GAS CO. VIRGIL C. SUMMER NUCLEAR STATION
Performance Curve For Centrifugal Charging Pump C Figure 6.3-4c

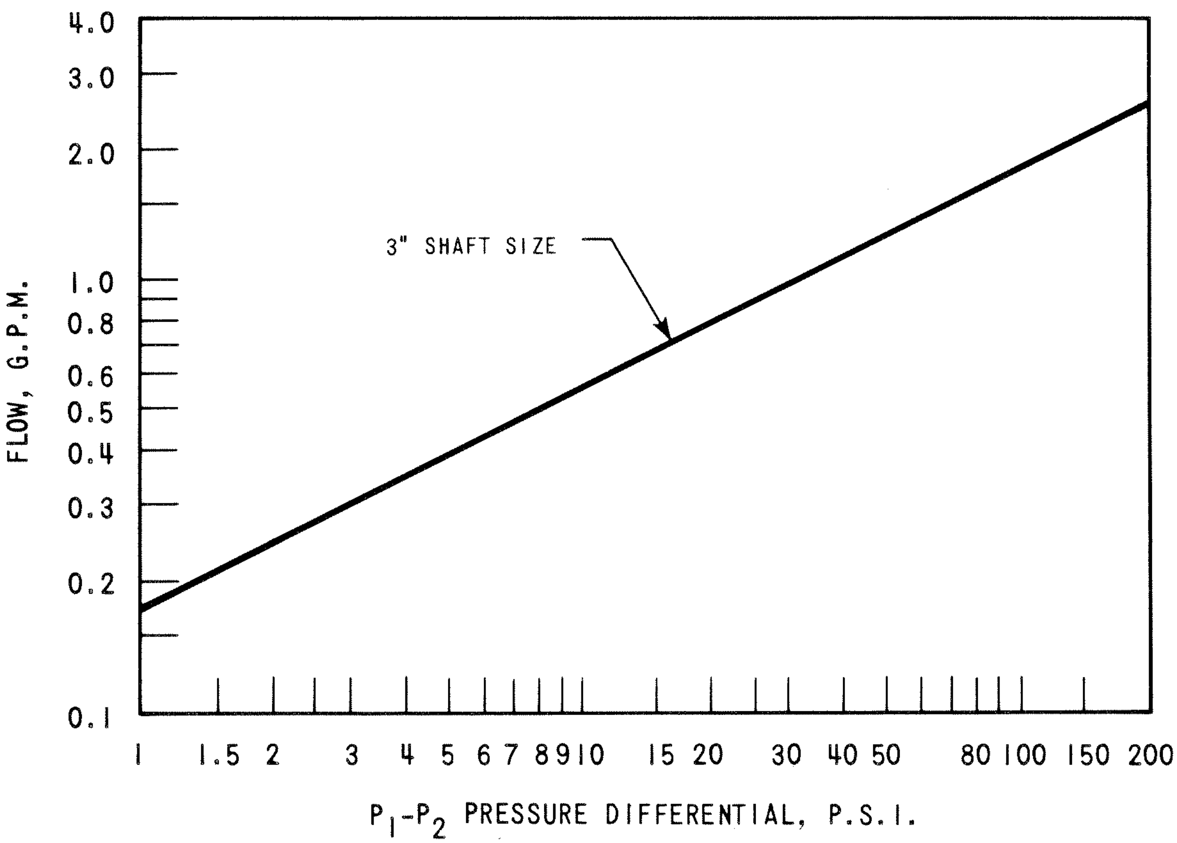


FLOATING BUSHING AS THROTTLE BUSHING

BUSHING LENGTH FACTOR	
LENGTH	FACTOR
1/4"	1.0
3/8"	0.85
1/2"	0.75
3/4"	0.60

EXAMPLE - FLOATING BUSHING AS THROAT BUSHING

BOX PRESSURE - P_2 = 25 P.S.I.
FLUSHING FLUID PRESSURE - P_1 = 50 P.S.I.
SHAFT DIA = 2" BUSHING LENGTH = 1/2"
PRESSURE DIFFERENTIAL = P_1-P_2 = 50-25 = 25 P.S.I.
FROM CHART FOR 25 P.S.I. AND 2" SHAFT DIA. FLOW IS 0.64"G.P.M. BUSHING LENGTH FACTOR FOR 1/2" BUSHING IS .75. FLOW = 0.64 G.P.M. X .75 = 0.48 G.P.M.

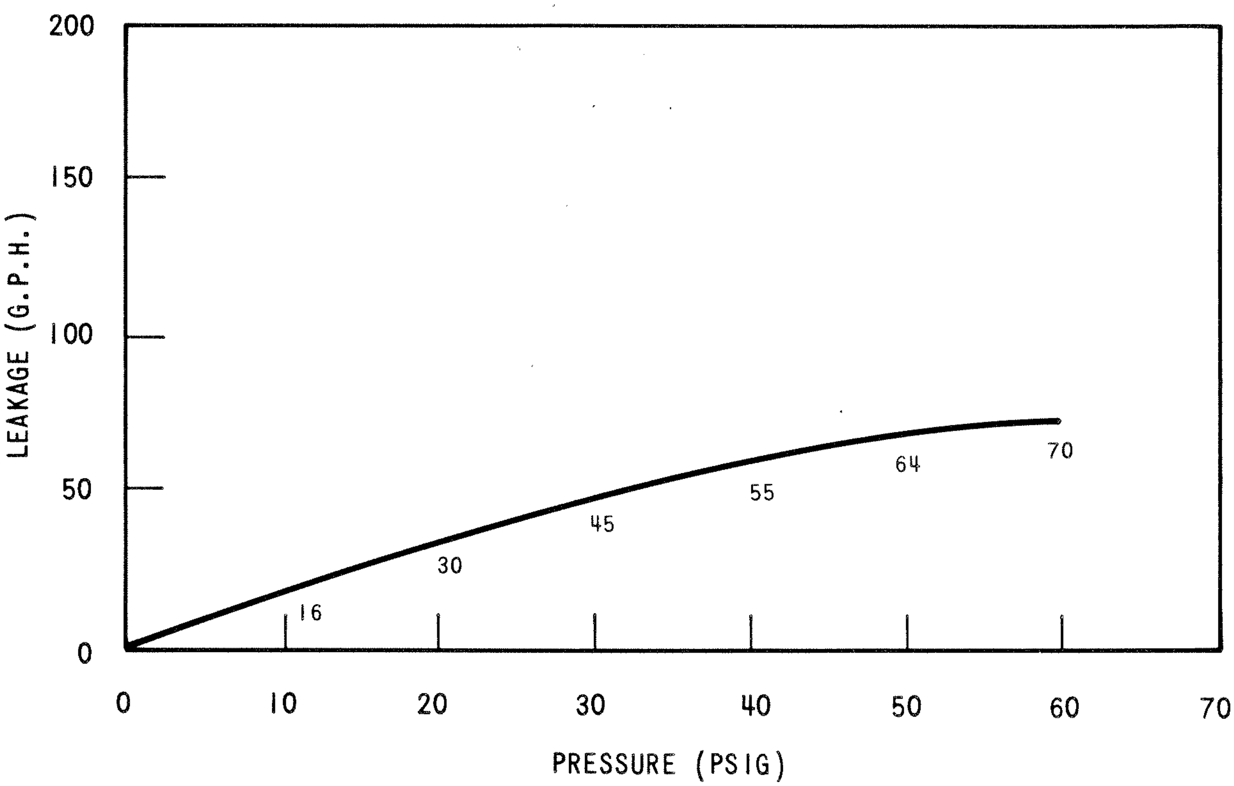


Amendment 0
August 1984

SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Flow Rates for Floating Bushing

Figure 6.3-5

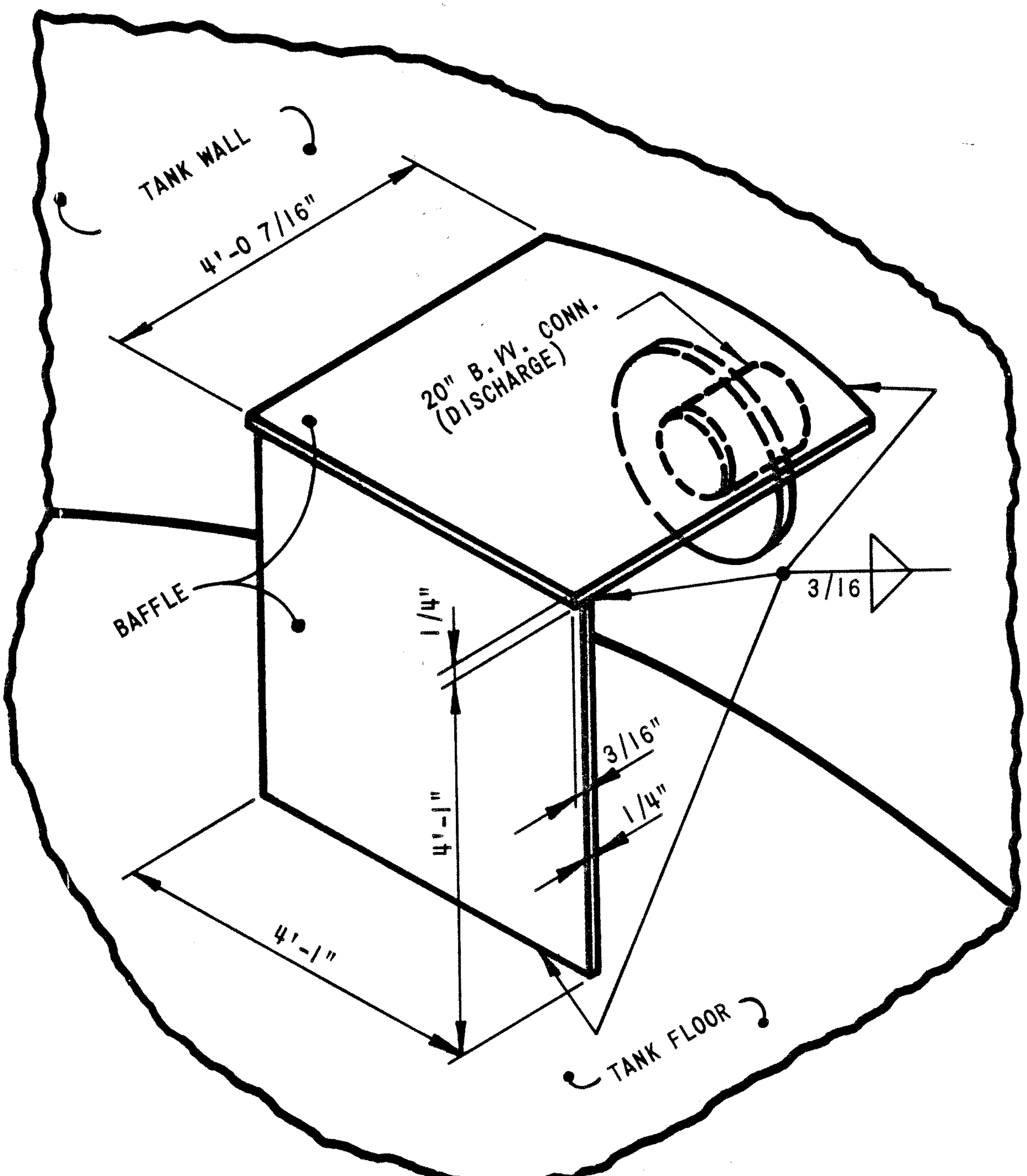


SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Bushing Leak Rate After Severe Operation

Figure 6.3-6

Amendment 0
August 1984



SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Refueling Water Storage Tank
20" Discharge Baffle

Figure 6.3-7

Amendment 0
August 1984

6.4 HABITABILITY SYSTEMS

Control room systems are designed in accordance with the design bases described in Section 6.4.1 so that habitability of the control room can be maintained under normal and accident conditions. The general guidance contained in Regulatory Guide 1.78 and the specific guidance contained in Regulatory Guide 1.95 is reflected throughout this section. RN 12-034

6.4.1 HABITABILITY SYSTEMS FUNCTIONAL DESIGN

6.4.1.1 Design Bases

The design bases for control room habitability systems are as follows:

1. Control Room Envelope

   The control room envelope includes most areas located on the 448' and 463' elevations of the Control Building. Housed within this control room envelope are the monitoring equipment, instrumentation, and control panels required for safe operation and shutdown of the plant. The control room envelope is provided with fire protection equipment, adequate lighting, communications equipment and kitchen, sanitary, administrative and storage facilities, and spaces necessary for normal plant operation and required to maintain the plant in a safe condition following an accident. The control room envelope ambient atmosphere is normally maintained at 75°F ± 2°F, drybulb and 50 percent relative humidity. RN 98-028

2. Period of Habitability

   The control room envelope is equipped to sustain seven people for a period of seven days following an accident.

3. Capacity

   The normal occupancy level of the control room is six people.

4. Food, Water, Medical Supplies, and Sanitary Facilities

   First aid equipment, food, and water are provided to sustain seven people for seven days following an accident. Chemical toilet facilities are provided for use in the event that normal sanitary facilities become inoperative.

5. Radiation Protection

   Radiation protection, as required by 10 CFR 50, Appendix A, Criterion 19, is provided by shield walls on the four exposures, shield slabs at floor and ceiling, radiation monitoring equipment, and emergency filtering systems. The control room atmosphere is monitored for radiation. When required, the control room atmosphere can be recirculated through the emergency filter system to remove contaminants. This filter system consists of roughing, high efficiency particulate air (HEPA), and charcoal filters. Assumptions and analyses regarding sources and amounts of radioactivity which may surround or leak into the control room and related shielding requirements are discussed in Chapters 12 and 15. The Radiation Monitoring System is discussed in Sections 11.4, 12.1.4 and 12.2.4.

6. Noxious Gas Protection

   Smoke detectors located in the control room air supply duct and in the emergency filter system discharge duct actuate alarms to indicate the presence of smoke in these locations. Additionally, the control room can be purged with outside air if required. Regulatory Guides 1.78 and 1.95 are discussed in Appendix 3A.

7. Respiratory, Eye, and Facial Protection for Emergencies

   Self-contained breathing apparatus are provided for control room occupants.

8. Habitability System Operation during Emergencies

   Operation of the habitability system during emergencies is discussed in Section 9.4.1.2.1.

9. Emergency Monitors and Control Equipment

   Emergency monitors and control equipment are discussed in Section 13.3.

6.4.1.2 <u>System Design</u>

6.4.1.2.1 Definition of Control Room Envelope

The areas to which the control room operator could require access during an emergency are the following:

1. Control room main control board and monitoring panel area - continuous occupancy required.

2. Chart and storage room - infrequent access required.

3. Shift Manager’s office - infrequent access required. RN 15-011

4. Kitchen facility - infrequent access required.

5. Toilet room - infrequent access required.

The equipment to which the control room operator could require access during an emergency is listed in Table 6.4-1.

6.4.1.2.2 Ventilation System Design

Normal and emergency ventilation of the control room are discussed in Section 9.4.1.2.1. The system diagram, Figure 9.4-1, indicates major components, ducts, dampers, instrumentation, and normal and emergency air flows.

System isolation dampers are pneumatically operated, spring opposed. Damper leakage does not exceed 20 cfm/ft^2 of damper area against a static pressure of 4 inches, water gage. Outside air inlet valves are 14 inch butterfly valves which have negligible leakage. Table 6.4-2 provides a summary of system isolation damper/valve data. RN 98-103

Components, essential instrumentation, ducting and outside air intake, and relief vents of the Control Building Ventilation System are designed in accordance with Seismic Category I requirements.

The components are not subject to the effects of floods, catastrophic weather, internal or external missiles, pipe whip, or jet impingement.

Figures 1.2-15 and 1.2-16 present layout drawings of the control room indicating doors, corridors, stairwells, shielded walls, ventilation equipment, and the location of outside air intakes.

The location of potential radioactive gas releases and their effect upon control room operation and the monitoring instrumentation and controls located therein are discussed in Chapter 15.

A description of the emergency filter system and a discussion of Regulatory Guide 1.52 is presented in Section 6.5.1 and Appendix 3A.

6.4.1.2.3 Leak Tightness

The control room system is designed so that, when operating in a normal or an emergency mode (admitting outside air), positive differential pressure is maintained between the control room and adjacent spaces. In the emergency mode, outside air is filtered by the emergency filter system and is used to pressurize the control room.

The potential paths of air leakage into the control room include: outside air valves in the non-operating redundant system and relief air dampers; openings around supply and return ducts in the control room walls and in duct chase floors; openings for electrical conduit and cables in the control room and chase walls and floors; doors; and piping. A review of these paths, as summarized below, indicates that air leakage through these paths during the emergency mode is minimal:

1. In the non-operating redundant system, the outside air is sealed from the control room by two 14 inch butterfly valves in series. Both valves are the low leakage type and fail closed upon loss of control air or power. The maximum leakage through a single closed 14 inch butterfly valve is negligible.

2. The relief dampers are two 36 inch by 36 inch dampers, one in each train. Both dampers are gasketed and blanked closed, except during control room purge, to ensure negligible air leakage.

3. The purge dampers are two 42 inch by 42 inch dampers in series. The inlet plenum to these dampers is gasketed and blanked closed, except during control room purge to ensure negligible air leakage. These dampers are fixed in the full open position.

4. Openings around supply and return ducts in the walls and floors of the control room and chases are sealed with air tight expanded silicone foam with a fire resistance rating of three hours. Negligible air leakage is present in these openings.

5. Openings for electrical conduit and cables in the walls and floors of the control room and chases are sealed in the same manner as the openings noted in item 4, above.

6. Doors from the control room to the chase area are three hour fire rated doors with closures. The maximum anticipated total air gap around each door is 0.21 ft^2. The doors leading from the turbine building to the control room are gasketed, pressure tight doors.

7. Piping to plumbing fixtures, drains, and potable water leaving the control room are sealed in the manner noted in item 4, above.

None of the control room doors lead directly to the outside. Doors lead to closed chase spaces, stairwells, or corridor spaces. Thus, neither outside wind conditions or other ventilation system cause infiltration or leakage into the control room.

6.4.1.2.4 Interaction With Other Zones and Pressure Containing Equipment

The Control Room Ventilation System is not interconnected with other building areas where potential for radioactivity exists. The system interconnects with cable spreading areas and with chase spaces through closed fire doors. There are no pressure containing pipes or equipment containing hazardous materials in the control room, cable spreading areas or duct chases.

The Turbine Building contains sources of carbon dioxide and steam but releases of these substances are diluted by the large Turbine Building volume and would be unlikely to reach the control room in any significant quantity since entry is protected by a pressure retaining door, a corridor, and a second door.

The normal and emergency operating modes of the Control Room Ventilation System maintain a positive pressure differential between the control room and adjacent spaces. Thus, the likelihood of infiltration is reduced. A radiation monitor sensing control room atmospheric radioactivity immediately causes the ventilation system to shift into the emergency mode upon detection of high gaseous activity. Thus, the potential for entry of outside airborne material into the control room is minimal. The Control Room Ventilation System can also be manually controlled to purge the control room with outside air. The outside air purge rate is variable up to 100 percent of the system flow rate.

6.4.1.2.5 Shielding Design

The control room shielding design is discussed in Section 12.1.

6.4.1.3 <u>Design Evaluation</u>

Each of the operating systems which ensures control room habitability is discussed in detail in other sections. These systems and the FSAR section in which they are discussed are as follows:

1. Control Room Ventilation System, 9.4.1.

2. Fire Protection System, 9.5.1.

3. Communications System, 9.5.2.

4. Lighting System, 9.5.3.

5. Offsite power system, 8.2.

6. Onsite power system, 8.3.

7. Radiation Monitoring System, 11.4, 12.1.4, 12.2.4.

A summary evaluation of control room habitability based upon selected considerations is presented in Sections 6.4.1.3.1 through 6.4.1.3.6.

6.4.1.3.1 Radiological Protection

The evaluation of radiological exposures to control room operators for postulated accident conditions is presented in Section 15.4.

6.4.1.3.2 Toxic Gas Protection

The primary hazardous chemical onsite in sufficient quantities which may under accidental release approach toxicity limits in the control room is anticipated to be bottled chlorine.

In accordance with the recommendations of Item C2 of Regulatory Guide 1.95 chlorine is stored onsite in containers having a single container inventory of 150 pounds or less. It is stored in the chlorine shed adjacent to the Water Treatment Building and is greater than 150 meters from the control room intake. Indication of a chlorine leak is provided by a chlorine leak detector located in the chlorine shed which annunciates an alarm locally and in the control room. Capability for manual isolation of the Control Room Ventilation System is described in Section 9.4.1. For discussion of postulated offsite hazardous chemical releases see Section 2.2.

6.4.1.3.3 Control Room Filtering System

The general arrangement and control of the Control Room Filtering Systems are as described in Section 9.4.1. RN 98-064

6.4.1.3.4 Control of the Control Room Thermal Environment

The Control Room Air Handling System operates during normal and emergency periods to maintain an environment suitable for personnel and equipment. The conditions maintained and general system description are presented in Section 9.4.1. RN 98-064

6.4.1.3.5 Fire Protection

Evaluation of the Fire Protection System is presented in Section 9.5.1. RN 98-064

6.4.1.3.6 Food, Water and Sanitation

A seven day supply of food is provided in the control room for emergency use. If the emergency requires confinement for periods longer than seven days, additional food will be brought onsite in protected containers. Site accessibility will be determined by the plant Health Physicist.

Potable water is normally available from the Potable Water System described in Section 9.2.4. Should this system become unavailable during an emergency period, stored, bottled water is available in an area immediately adjacent to the control room. Additional bottled water could be brought into the control room if required.

Normal sanitation facilities are available as described in Section 9.2.4. Chemical toilets are also available in these areas for use in the event that the normal facilities become inoperative following an emergency.

6.4.1.4 Testing and Inspection

The equipment which maintains control room habitability includes the Emergency Filter System, the Control Room Air Handling System, and the Chilled Water System.

The testing and inspection is further described in Section 9.4.1.4. RN 98-064

6.4.1.5 Instrumentation Requirements

Habitability systems instrumentation and control equipment provides for control and monitoring of performance and status during system operation and testing. The instrumentation and control provisions for each of the systems used to ensure control room habitability are discussed in other sections of the FSAR. A list of these systems and the sections in which they are discussed is presented in Section 6.4.1.3. RN 98-064

6.4.2 SYSTEM OPERATIONAL PROCEDURES

The normal and emergency operating modes of normal and emergency control room habitability equipment, instrumentation, and the associated controls are described in Section 9.4.1.2.1.

TABLE 6.4-1

EQUIPMENT TO WHICH THE CONTROL ROOM OPERATOR COULD REQUIRE ACCESS DURING AN EMERGENCY

Item or Equipment	Location Within Control Room Envelope
Control and Monitoring Panels	Identified on Figure 1.2-15
Portable Radiation Measuring Instruments	Store Room
Emergency Procedures, Manuals and Drawings	Operator's desk
Self-Contained Breathing Apparatus	Store Room
Communications Equipment	Operator's desk
Fire Extinguishing Equipment	On control room wall outside Shift Manager's office RN 15-011
Food Supplies	Kitchen

TABLE 6.4-2

CONTROL ROOM VENTILATION SYSTEM
ISOLATION DAMPER/VALVE SUMMARY

Damper Function	Size (in)	Identification	Failure Position
Purge Air Inlets	42 by 42	XDP-18A, 19A-AH	Open*
Purge Air Inlets	42 by 42	XDP-18B, 19B-AH	Open*
System Relief Air	36 by 36	XDP-21A, 21B-AH	Closed**
System Return Air	42 by 42	XDP-22A, 22B-AH	Closed
Equipment Room Relief Air	72 by 48	XDP-133A, 133B-AH	Closed
Equipment Room Relief Air	72 by 48	XDP-234A, 234B-AH	Closed
Emergency Filter Inlet Damper	42 by 42	XDP-23A, 23B-AH	Open
Emergency Filter Fan Discharge Damper	42 by 42	XDP-24A, 24B-AH	Open
Outside Air Inlets	14 (in) Butterfly Valve	XVB-3A, 4A-AH	Closed
Outside Air Inlets	14 (in) Butterfly Valve	XVB-3B, 4B-AH	Closed

* Inlet plenum blanked to ensure no air ingress

** Outlet plenum blanked to ensure no air leakage

TABLE 6.4-3

INFORMATION REQUIRED FOR CONTROL ROOM HABITABILITY EVALUATION

Reference: NUREG-0737, Section III.D.3.4, Attachment 1

	Subject	FSAR Source	Response	
(1)	Control room mode of operation	6.4.1.2.4 9.4.1.2.1 15.4.1.4.3	Zone isolation with filtered air recirculation and positive zone air pressurization	
(2)	Control room characteristics:			
(a)	Control room volume	Table 15.4-17	226,040 ft^3 (free volume)	
(b)	Control room emergency zone	6.4.1.1 6.4.1.2.1	Most areas located on elevation 463'0" of the Control Building	RN 98-103
(c)	Control room ventilation system schematic		FSAR Figure 9.4-1 FSAR Table 6.4-2	
(d)	Infiltration leakage rate	Table 15.4-17	1508 cfm (Total)	RN 12-034
(e)	Filter efficiencies	9.4.1.2.1		
1.	Charcoal filters		95% for all species of iodine	
(f)	Closest distance between release points and control room air intake	Table 2.3-122	See Table 2.3-122	
(g)	Layout drawing	1.2	FSAR Figures 1.2-1 and 1.2-26	RN 98-103
(h)	Control room shielding	6.4.1.1 15.4.1.4.3	2 ft. concrete	
(i)	Automatic isolation capability			
1.	Valve closing time		N/A – Valves remain open for Control Room pressurization See 6.5.1.5.2	RN 98-103
2.	Valve leakage	Table 6.4-2	Negligible	
3.	Valve area		113 in^2	

TABLE 6.4-3 (Continued)

INFORMATION REQUIRED FOR CONTROL ROOM HABITABILITY EVALUATION

	Subject	FSAR Source	Response	
(j)	Chlorine detectors or tube gas (local or remote)	6.4.1.3.2	Detector in chlorine shed with annunciation both local and in control room.	
(k)	Self-contained breathing apparatus availability	6.4.1.1 Table 12.3-4	1 unit/control room operator	
(l)	Bottled air supply	Appendix 3A RG 1.78	3.5 hours (30 min/person)	
(m)	Emergency food and potable water supply	6.4.1.1 6.4.1.3.6	7 persons for 7 days Resupply considered for periods longer than 7 days	RN 01-113
(n)	Control room personnel capacity (normal and emergency)	6.4.1.1 7.7.3	Normal = 6 person (control room) Emergency = 7 persons (control room) and ≥ 25 persons (technical support center)	RN 01-113
(o)	Potassium iodide drug supply		Will be available	
(3)	Onsite storage of chlorine and other hazardous chemicals			
(a)	Total amount and size of containers	2.2.3.1.1 6.4.1.3.2	2 containers - with 150 lb chlorine/container	
(b)	Closest distance from control room intake	2.2.3.1.1 6.4.1.3.2	Approximately 507 feet southeast of control building	

TABLE 6.4-3 (Continued)

INFORMATION REQUIRED FOR CONTROL ROOM HABITABILITY EVALUATION

Subject	FSAR Source	Response
(4) Offsite manufacturing, storage, or transportation facilities of hazardous materials		
(a) Identify facilities within a 5 mile radius	2.2 Table 2.2-1	1. Nyline Corporation 2. Farm Milling Service 3. Interstate Materials Inc., Div. of Clement Bros. Co. 4. Winnsboro Granite Corp.
(b) Distance from control room	Table 2.2-1	1. 2.6 miles southeast 2. 3.6 miles northeast 3. 3.0 miles northeast 4. 4.8 miles northeast
(c) Quantity of hazardous chemicals in one container	Table 2.2-1	Gross quantities are: 1. 42,000 lbs caprolactan 2. 150 - 200 tons ammonium nitrate 3. 25,000 lbs ammonium nitrate and torpex 50,000 lbs ammonium nitrate and fuel oil (ANFOPRILL) RN 01-113 4. 100 lbs black powder 200 blasting caps RN 01-113
(d) Frequency of hazardous chemical transportation traffic (truck, rail, and barge)	2.2.3	Refer to FSAR Section 2.2.3
(5) Technical specifications		
(a) Chlorine detection system	2.2 6.4 Appendix 3A	Not required in technical specification
(b) Control room emergency filtration system	6.4 9.4	Technical Specification Section 3.7.6

6.5 FISSION PRODUCT REMOVAL AND CONTROL SYSTEMS

6.5.1 ENGINEERED SAFETY FEATURE FILTER SYSTEMS

The engineered safety features (ESF) filter systems include:

1. The high efficiency particulate air (HEPA) filters in the Reactor Building Cooling System.

2. The control room emergency filter plenums.

3. The Fuel Handling Building charcoal exhaust system.

6.5.1.1 Design Bases

The ESF filter systems are designed and sized to perform the following functions. Non ESF filter systems are discussed in Section 9.4.

1. Satisfy the radiation control requirements of 10 CFR 20.

2. Satisfy requirements of General Design Criteria 19, 41, 42 and 43.

3. Remove particulate fission products from the post accident Reactor Building atmosphere (Reactor Building Cooling System only).

4. Remove gaseous and particulate fission products from the spent fuel pool surface and from general areas of the Fuel Handling Building.

6.5.1.2 System Design

Descriptive information concerning the ESF filter systems is presented in the following sections:

1. Reactor Building cooling system HEPA filters:

    a. Design bases - Sections 6.5.1.1 through 6.5.1.3

    b. General description - Section 6.2.2.2.2

    c. System design features - Section 6.2.2.2.2.1

2. Control room emergency filter plenums:

    a. Design bases - Section 9.4.1.1

    b. General description - Section 9.4.1.2

    c. System design features - Section 9.4.1.2.1

3. Fuel Handling Building charcoal exhaust system.

    a. Design bases - Section 9.4.3.1

    b. General description - Section 9.4.3.2

    c. System design features - Section 9.4.3.2.1

Table 6.5-1 provides a comparison of the emergency filter systems with the requirements of Regulatory Guide 1.52. General compliance with Regulatory Guide 1.52 is discussed in Appendix 3A.

6.5.1.3 Design Evaluation

The general arrangement and control of the ESF filter systems are described in Sections 6.2.2, 9.4.1, and 9.4.3. System components are separated, redundant, powered by Class 1E electric systems, and are housed in Seismic Category 1 structures. The control room system is not subject to jet impingement, pipe whip or flooding. The HEPA filter systems in the Reactor Building are physically separated so that no more than 1 unit is subject to these occurrences. The Fuel Handling Building charcoal exhaust system plenums are housed in shielded and separated cubicles. The exhaust fans of this system are adjacent to each other. Neither the exhaust fans nor the filter plenums are subject to jet impingement, pipe whip or flooding. The ESF filter systems can sustain a single component failure without impairing their full functional capability. Additionally, provision is made for suitable maintenance and change-out space and adequate instrumentation and lighting, all of which reduce personnel exposure by reducing time exposure.

The control room emergency filter system is required in the event of radioactive leakage from the outside into the Control Building following a LOCA. Post LOCA operation of the control room filter systems is ensured by automatic actuation as a result of a safety injection signal or high radiation in the control room signal from the radiation monitor (RM-A1).

The Reactor Building cooling system and control room emergency filter trains provide a minimum of 99.97 percent removal efficiency for particulates. The control room emergency filters also provide a minimum of 95 percent removal efficiency for methyl iodine at 85 percent relative humidity and 70°F (see Reference [1]).

6.5.1.4 Tests and Inspections

Details of the tests and inspections for the ESF filter systems are provided in the following sections:

6.2.2.4.2 - Reactor Building Cooling System HEPA Filters 99-01

9.4.1.4 - Control Room Emergency Filters

9.4.3.4 - Fuel Handling Building Charcoal Exhaust System

Laboratory tests, inplace tests, and test acceptance criteria are in accordance with the requirements of Regulatory Guide 1.52 (see Table 6.5-1).

6.5.1.5 Instrumentation Requirements

Each of the ESF filter systems include instrumentation for system performance and status monitoring during normal plant operation, system tests, and system operation following a design basis accident. The instrumentation and control features of the ESF filter systems are described in Sections 6.5.1.5.1 and 6.5.1.5.2.

6.5.1.5.1 Reactor Building Cooling System HEPA Filter Instrumentation

Activation of this filtering system is initiated automatically by a safety injection signal (from the ESF actuation system) or by manual action at the main control board. A separate automatic initiation signal is supplied to each redundant train. Under accident conditions, the filter bypass dampers of both trains (4 units) receive a signal to automatically close, 1 preselected cooling unit in each train is automatically started in low speed.

Main control board switches are provided for each fan motor (high and low) and for the filter bypass damper of each reactor building cooling unit.

Instrumentation used for monitoring the performance of this filter system including bypass damper position indication, is described in Section 6.2.2.5.2.

6.5.1.5.2 Control Room Emergency Filter Instrumentation

Activation of the control room emergency filter system is initiated automatically by a safety injection signal, a high control room radiation signal (RM-A1), (see Section 12.2.4) or by manual action at the HVAC control board in the control room. Other elements of the control room ventilation system are operated in conjunction with this emergency filter system to carry out the air cleaning function. In particular, the relief dampers and outside air supply dampers of the control room ventilation system are automatically tripped closed by the signals listed above. HVAC control board switches are provided for each emergency filtering fan. The fan inlet and discharge dampers operate in conjunction with their respective fans.

The overall operation of the control room ventilation system is described in Section 9.4.1. Instrumentation used for monitoring the performance of this filter system, including fan and damper status indication, is described in Section 9.4.1.2.1.

6.5.1.5.3 Fuel Handling Building Charcoal Exhaust System Instrumentation

The Fuel Handling Building charcoal exhaust system components (2 filter plenums, one exhaust fan) are in continuous operation during normal, shutdown and refueling periods. The ESF function of the fuel building charcoal exhaust system is only required during the movement of fuel in the spent fuel pool or during crane operation with loads over the pool. Instrumentation used for monitoring system performance, including fan and damper status indication, is described in Section 9.4.3.2.1. 98-01

6.5.1.6 <u>Materials</u>

Filter materials employed in the ESF filter systems are discussed in Section 6.5.1.6.1 and 6.5.1.6.2.

6.5.1.6.1 Reactor Building Cooling System HEPA Filter Materials

Each of four Reactor Building cooling system filter plenums includes 120 HEPA cells. Cells are nominally 24 inches square by 12 inches deep. Cells include media of high strength, waterproof glass fiber without separators, high temperature silicon adhesive, metal cell sides and silicon rubber closed cell sponge gasketing. RN 08-009

6.5.1.6.2 Control Room Emergency Filter and Fuel Handling Building Charcoal Exhaust System Filter Materials

Both of the control room emergency filter plenums contain 1 prefilter bank, each with 15 prefilter cells. Each of the 3 Fuel Handling Building charcoal exhaust system filter plenums contains 1 prefilter bank, each with 12 prefilter cells. The prefilters satisfy the requirements of and are marked for Underwriters Standard UL-900 [2], Class I; and therefore, the radiolytic or pyrolytic decomposition products will not interfere with safe ESF operation.

Both of the control room emergency filter plenums contain 2 banks of HEPA filters with 15 HEPA filters per bank. Each of the Fuel Handling Building charcoal exhaust system filter plenums includes 2 banks of HEPA filters with 12 HEPA filters per bank. Each filter is 24 inches square by 12 inches deep and includes glass media. The HEPA filters satisfy the requirements of MIL-C-51068C, MIL-F-51079 and are qualified to Underwriters Standard UL-586 [3]; and therefore, the radiolytic or pyrolytic decomposition products will not interfere with safe ESF operation. 00-01

The 2 control room emergency filter plenums and the 3 Fuel Handling Building charcoal exhaust system filter plenums are of the vertical bed design. The control room plenums contain 11 sections and the Fuel Handling Building charcoal exhaust system plenums contain 8 sections. Media is impregnated, activated coconut shell charcoal which meets the requirements of Regulatory Guide 1.52, Revision 2.

6.5.2 <u>REFERENCES</u>

1. American Air Filter Report, TR-7102 Impregnated Activated Carbon for Removal of Radioiodine Compounds from Reactor Containment Atmospheres.

2. “Air Filter Units,” Underwriters Laboratories, UL-900.

3. “High Efficiency, Particulate, Air Filter Units,” Underwriters Laboratories, UL-586.

COMPARISON OF ENGINEERED SAFETY FEATURES FILTER SYSTEMS WITH REGULATORY GUIDE 1.52 (REVISION 1, 7/76) REQUIREMENTS

1. Reactor Building Cooling System HEPA Filters

Regulatory Guide 1.52 Item	
1-a	The system complies with this requirement.
1-b, c	The filters comply with the integrated radiation dose criteria as required by items 1-b, c.
1-d	The system complies with this requirement.
1-e	Not applicable.
2-a	Filter plenums include only demisters and HEPA filters. Systems are redundant. Charcoal adsorbers are not included since the reactor building spray system serves this function.
2-b	The system complies with this requirement.
2-c	The system complies with this requirement.
2-d	The system complies with this requirement.
2-e	The system complies with this requirement.
2-f	The capacity of each plenum is 120,000 cfm. The plenum is arranged as three filter modules, each four filters high and ten filters wide. Space limitations make use of multiple 30,000 cfm plenums for this system physically impractical.
2-g	The system is instrumented to indicate differential pressure drop across demisters and HEPA filters by means of alarms in the control room and to indicate filter damper position. Since the filters are used only following a LOCA or during test, the additional instrumentation specified would provide minimum benefit and, therefore, was not added.
2-h	See Chapters 7.0 and 8.0 for IEEE compliance.

COMPARISON OF ENGINEERED SAFETY FEATURES FILTER SYSTEMS WITH REGULATORY GUIDE 1.52 (REVISION 1, 7/76) REQUIREMENTS

1. Reactor Building Cooling System HEPA Filters (Continued)

Regulatory Guide 1.52 Item	
2-i	The system complies with Regulatory Guide 8.8 through provision of service platforms and arrangement of HEPA filters for "bagging" during the changing procedure. As noted for item 2-f, above, capacity of the unit made change as a single unit physically impractical. Also, high radiation deposits are not expected to accumulate on these filters from normal operation. Therefore, filter replacement as a single unit loses importance.
2-j	Not applicable
2-k	Duct and housing leakage is not of primary importance since this system is a recirculation type system housed inside the reactor building
3-a	The system complies with this requirement.
3-b	Not applicable.
3-c	The system complies with this requirement.
3-d	The system complies with this requirement, except that filter tests were performed by the manufacturer.
3-e	The system complies with this requirement.
3-f	The system complies with this requirement.
3-g	The system complies with this requirement.
3-h	Not applicable since there is no fire spray system to this plenum.
3-i	Not applicable since there are no adsorption filters in this system.
3-j	Not applicable since there are no adsorption filters in this system.
3-k	Not applicable since there are no adsorption filters in this system.

COMPARISON OF ENGINEERED SAFETY FEATURES FILTER SYSTEMS WITH REGULATORY GUIDE 1.52 (REVISION 1, 7/76) REQUIREMENTS

1. Reactor Building Cooling System HEPA Filters (Continued)

Regulatory Guide 1.52 Item	
3-l	The system complies with this requirement.
3-m	The system complies with this requirement.
3-n	The system complies with this requirement.
3-o	The system complies with this requirement. 02-01
4-a	The system complies with this requirement.
4-b	The system complies with this requirement.
4-c	The capacity of this system makes this requirement physically impractical.
4-d	External connections have been provided for field insertion of test probes and manifolds.
4-e	Operating procedures satisfy this request.
4-f	System procedures comply with this request.
5-a	Startup and test and operating procedures comply with this requirement except testing will be per-formed every 18 months in lieu of once per operating cycle and after 720 hours of system operation. The visual test will be performed before each in place air flow distribution test, or DOP test (Note 1).
5-b	Filter plenum testing procedures comply with this requirement.
5-c	Operating procedures and filter plenum testing procedures comply with this requirement except testing will be performed every 18 months in lieu of once per operating cycle and after 720 hours of system operation. (Note 1).
5-d	Not applicable since there are no adsorption filters in this system.
6-a	Not applicable since there are no adsorption filters in this system.
6-b	Not applicable since there are no adsorption filters in this system.

COMPARISON OF ENGINEERED SAFETY FEATURES FILTER SYSTEMS WITH REGULATORY GUIDE 1.52 (REVISION 1, 7/76) REQUIREMENTS

2. Control Room Emergency Filter

Regulatory Guide 1.52 Item	
1-a	The listed criteria, except radiation dose rate, relative to this system are not affected by the DBA. The dose rate following DBA was considered in the filter design.
1-b	The filters comply with the integrated radiation dose criteria as required by items 1-b, c.
1-d	Not applicable.
1-3	Not applicable.
2-a	Filter plenums include all sequential components listed, except demisters and heating/cooling coils. Demisters are not included because there is no source of water spray coincident with continuing filter operation. Heating/cooling coils are not included because relative humidity is not expected to exceed 50 percent. Systems are redundant.
2-b	The system complies with this requirement.
2-c	The system complies with this requirement.
2-d	Not applicable since filter plenums are not in an area where pressure surges could occur.
2-e	The system complies with this requirement.
2-f	The system complies with this requirement.
2-g	The system complies with this requirement through provision of control room indication of isolation damper position, high temperature and smoke. Recorders are not considered pertinent since this system is operated only for test or under accident conditions. Provisions have been made for local indication of pressure drop across filter sections.
2-h	See Chapters 7.0 and 8.0 for IEEE compliance.
2-i	The system complies with this requirement. 02-01

COMPARISON OF ENGINEERED SAFETY FEATURES FILTER SYSTEMS WITH REGULATORY GUIDE 1.52 (REVISION 1, 7/76) REQUIREMENTS

2. Control Room Emergency Filter (Continued)

Regulatory Guide 1.52 Item	
2-j	Not applicable.
2-k	Ducts were specified to satisfy leakage testing requirements of SMACNA, Chapter 8, 1967. Plenums were specified to indicate no leakage through soap bubble testing with ducts at 2 psig. Neither the plenum nor the attached ducts are in any contaminated area between the plenum equipment area and the control room.
3-a	Not applicable since this system will not be exposed to a water spray.
3-b	No heaters are included in the design. The filters process room return air and room relative humidity should not exceed 50 percent. Therefore, heaters are not required.
3-c	The system complies with this requirement.
3-d	The system complies with this requirement.
3-e	The system complies with this requirement.
3-f	The system complies with this requirement.
3-g	The system complies with this requirement.
3-h	The system complies with this requirement.
3-i	The system complies with this requirement. New activated carbon meets the physical property specifications given in Table 5.1 of ANSI N509-1976. (Note 1) New and used activated carbon meet the laboratory testing criteria of ASTM D3803-1989, at a relative humidity of 70% and a temperature of 30° C (86° F) with a methyl iodide penetration of less than 2.5%. 00-01 RN 02-034
3-j	The system complies with this requirement.
3-k	The system complies with this requirement.
3-l	The system complies with this requirement.
3-m	The system complies with this requirement.
3-n	The system complies with this requirement.
3-o	The system complies with this requirement.

COMPARISON OF ENGINEERED SAFETY FEATURES FILTER SYSTEMS WITH REGULATORY GUIDE 1.52 (REVISION 1, 7/76) REQUIREMENTS

2. Control Room Emergency Filter (Continued)

Regulatory Guide 1.52 Item		
		02-01
4-a	The system complies with this requirement.	
4-b	The system complies with this requirement.	
4-c	The system complies with this requirement.	
4-d	External connections have been provided for field insertion of test probes and manifolds.	
4-e	Operating procedures satisfy this request.	
4-f	System procedures comply with this request.	
5-a	Startup and test and operating procedures comply with this requirement except testing will be performed every 18 months in lieu of once per generating cycle and after 720 hours of system operation. The visual test will be performed before each in place air flow distribution test, DOP test or activated carbon adsorber section leak test. (Note 1)	02-01
5-b	Filter plenum testing procedures comply with this requirement.	
5-c	Operating procedures and filter plenum testing procedures comply with this requirement except testing will be performed every 18 months in lieu of once per operating cycle and after 720 hours of system operation. (Note 1)	
5-d	Filter plenum testing procedures comply with this requirement except adsorber testing will be done with then DOP test is performed and following removal of an adsorber sample for laboratory testing if the integrity of the adsorber section is affected. (Note 1)	02-01 02-01
6-a	The system analysis complies with this requirement. New activated carbon meets the physical property specifications given in Table 5.1 of ANSI N 509-1976 (Note 1)	

COMPARISON OF ENGINEERED SAFETY FEATURES FILTER SYSTEMS WITH REGULATORY GUIDE 1.52 (REVISION 1, 7/76) REQUIREMENTS

2. Control Room Emergency Filter (Continued)

Regulatory Guide 1.52 Item	
6-b	The system analysis complies with this requirement except laboratory tests will be performed every 18 months in lieu of once per operating cycle. New unused activated carbon meeting the physical requirements of Table 5.1 of ANSI N 509-1976 will be used if (1) Testing is in accordance with footnote c of Table 3 (except as noted above) results in a representative sample failing to pass the applicable test in Table 3 or (2) no representative sample is available for testing. (Note 1) New and used activated carbon meet the laboratory testing criteria of ASTM D3803-1989, at a relative humidity of 70% and a temperature of 30° C (86° F) with a methyl iodide penetration of less than 2.5%. RN 02-034

3. Fuel Handling Building Charcoal Exhaust System

Regulatory Guide 1.52 Item	
1-a	The system complies with this requirement.
1-b, c	The filter plenums comply with the integrated radiation dose criteria as required by items 1-b, c.
1-d	The system complies with this requirement.
1-e	Not applicable.
2-a	Filter plenums include all sequential components listed, except demisters and heating/cooling coils. Demisters are not included because there is no source of water spray coincident with continuing filter operation. Heating/cooling coils are not provided because the relative humidity is not expected to exceed 70 percent. Plenums and fans are redundant.
2-b	The plenums are physically separated. The fans are adjacent to each other but are not in an area exposed to missiles.

COMPARISON OF ENGINEERED SAFETY FEATURES FILTER SYSTEMS WITH REGULATORY GUIDE 1.52 (REVISION 1, 7/76) REQUIREMENTS

3. Fuel Handling Building Charcoal Exhaust System (Continued)

Regulatory Guide 1.52 Item	
2-c	The system complies with this requirement.
2-d	Not applicable since the filter plenums are not in an area where pressure surges could occur.
2-e	The system complies with this requirement.
2-f	The system complies with this requirement.
2-g	Temperature of the charcoal filter sections of the plenums is recorded in the control room and detection of high temperature actuates alarms in the control room. Fan operation and isolation damper position are indicated in the control room. Filter bank pressure drop is indicated locally.
2-h	See Chapters 7.0 and 8.0 for IEEE compliance.
2-i	The system complies with this requirement.
2-j	Not applicable.
2-k	Ducts were specified to satisfy leakage testing requirements of SMACNA Chapter 8, 1967. Plenums were specified to indicate no leakage through soap bubble testing with ducts at 2 psig.
3-a	Not applicable since this system will not be exposed to a water spray.
3-b	No heaters are included in the design. The filters process room air and room relative humidity should not exceed 70 percent. Therefore, a heater is not required.
3-c	The system complies with this requirement.
3-d	The system complies with this requirement.
3-e	The system complies with this requirement.
3-f	The system complies with this requirement.
3-g	The system complies with this requirement.
3-h	The system complies with this requirement.

COMPARISON OF ENGINEERED SAFETY FEATURES FILTER SYSTEMS WITH REGULATORY GUIDE 1.52 (REVISION 1, 7/76) REQUIREMENTS

3. Fuel Handling Building Charcoal Exhaust System (Continued)

Regulatory Guide 1.52 Item		
3-i	The system complies with this requirement. New activated carbon meets the physical property specifications given in Table 5.1 of ANSI N509-1976. (Note 1) New and used activated carbon meet the laboratory testing criteria of ASTM D3803-1989, at a test media temperature of 30° C (86° F).	RN 02-034
3-j	The system complies with this requirement.	
3-k	The system complies with this requirement.	
3-l	The system complies with this requirement.	
3-m	The system complies with this requirement.	
3-n	The system complies with this requirement.	
3-o	The system complies with this requirement.	
4-a	The system complies with this requirement.	
4-b	The system complies with this requirement.	
4-c	The system complies with this requirement.	
4-d	External connections have been provided for field insertion of test probes and manifolds.	
4-e	Operating procedures comply with this request.	
4-f	System procedures comply with this request.	
5-a	Startup and test and operating procedures comply with this requirement except testing will be performed every 18 months in lieu of once per operating cycle and after 720 hours of system operation. The visual test will be performed before each in place air flow distribution test, DOP test or activated carbon adsorber leak test. (Note 1)	02-01
5-b	Filter plenum testing procedures comply with this requirement.	

COMPARISON OF ENGINEERED SAFETY FEATURES FILTER SYSTEMS WITH REGULATORY GUIDE 1.52 (REVISION 1, 7/76) REQUIREMENTS

3. Fuel Handling Building Charcoal Exhaust System (Continued)

Regulatory Guide 1.52 Item		
5-c	Operating procedures and filter plenum testing procedures comply with this requirement except testing will be performed every 18 months in lieu of once per operating cycle and after 720 hours of system operation. (Note 1)	
5-d	Filter plenum testing procedures comply with this requirement except adsorber testing will be done when the DOP test is performed and following removal of an adsorber sample for laboratory testing if the integrity of the adsorber section is affected. (Note 1)	02-01
6-a	The system analysis complies with this requirement. New activated carbon meets the physical property specifications given in Table 5-1 of ANSI N 509-1976. (Note 1)	
6-b	The system complies with this requirement except laboratory tests will be performed every 18 months in lieu of once per operating cycle (Note 1) and every 720 hours (since units run continuously) during the period of time there is fuel or crane movement with loads over the pool. New unused activated carbon meets the physical requirements of Table 5.1 of ANSI N 509-1976 will be used if (1) testing is in accordance with footnote C of Table 3 (except as noted above) results in a representative sample failing to pass the applicable test in Table 3 or (2) no representative sample is available for testing. (Note 1) New and used activated carbon meet the laboratory testing criteria of ASTM D3803-1989, at a media temperature of 30° C (86° F).	RN 02-034 RN 02-034

Note 1 Exceptions and clarifications are consistent with revisions made to these sections in Revision 2 of Regulatory Guide 1.52.