6.0 ENGINEERED SAFETY FEATURES

TABLE OF CONTENTS

TABLE OF CONTENTS

TABLE OF CONTENTS

TABLE OF CONTENTS

Page

LIST OF TABLES

LIST OF TABLES

LIST OF TABLES

LIST OF FIGURES

LIST OF FIGURES

LIST OF FIGURES

LIST OF FIGURES

LIST OF FIGURES

6.0 ENGINEERED SAFETY FEATURES

6.1 GENERAL

Engineered safety features are structures and equipment required to mitigate design basis accidents including the loss of coolant accident and high energy pipe breaks such as a steam pipe break and a main feedwater pipe break. Engineered safety features are designed to Seismic Category I requirements. They are designed to perform their safety function with complete loss of offsite power. Such equipment is provided with sufficient redundancy that failure of a single component will not result in the loss of the safety function. Engineered safety features fulfill the following safety functions under accident conditions:

A. Protect the fuel cladding.

B. Ensure containment integrity.

C. Minimize containment leakage.

D. Remove fission products from the containment atmosphere.

The operator action times assumed in this chapter include conservative actions to provide an adequate safety margin for the purpose of nuclear safety system design and nuclear safety analysis of the design basis events. However, they are not intended to serve as a basis for actual operator action times in procedures or training. The assumed time periods are considered in the basis of plant design to permit credit for operator actions. The Westinghouse Owners Group (WOG) Emergency Response Guidelines (ERGs) provide a basis for operatior action in response to design basis accidents.

6.1.1 SAFETY FEATURES SYSTEMS

The safety features systems provided to satisfy the functions listed above are as follows:

- Containment isolation system (subsection 6.2.4).
- Containment spray system (subsection 6.2.2).
- Containment fan cooler system (subsection 6.2.2).
- Containment air purification and cleanup system (subsection 6.2.3).
- Emergency core cooling system (section 6.3).
- Residual heat removal system (section 5.5).
- Combustible gas control in containment (subsection 6.2.5).
- Penetration room filtration system (section 6.2).

- Auxiliary feedwater system (section 6.5).

The fuel cladding is protected by the timely, continuous, and adequate supply of borated water to the reactor coolant system (RCS) and, ultimately, the reactor core. This supply of water is provided by the emergency core cooling system (ECCS). These systems provide high head (centrifugal charging pumps), low head (residual heat removal pumps) injection, and accumulator injection immediately following an incident, and low head/high head recirculation in the long term recovery period.

The containment integrity is ensured and the containment leakage is minimized by the provision of means for condensing the steam inside the containment, depressurizing the containment following an incident, and maintaining the containment at near atmospheric conditions for an extended period of time. The containment isolation system, spray system, fan cooler system, and the electric hydrogen recombiners provide the means for satisfying these requirements.

The fission products are removed from the containment atmosphere by the chemical spray additive which enhances the removal of radioactive iodine from the containment atmosphere following an incident. The containment air purification and cleanup systems are provided to meet this function.

The safety features systems are designed with sufficient redundancy to meet the general design criteria as discussed in sections 3.1, 3.2, and subsection 6.3.2.11. Electrical power for all safety features systems is provided both from offsite sources and from emergency onsite sources as described in sections 8.2 and 8.3, respectively.

Safety features are separated into two independent trains of equal capability. Either train can handle the entire emergency coolant injection and emergency cooling loads; either train can provide the entire containment isolation, containment cleanup, and containment leakage minimization functions. Each train has an independent onsite and offsite power source. Failure of either train cannot affect the other.

Some of high and low pressure emergency injection systems use equipment that serves normal functions during normal plant operation or shutdown. Observation of their normal functioning provides monitoring of equipment availability and condition. In cases where equipment is used for emergencies only, systems are designed to permit periodic inspection and tests.

6.1.2 OPERATIONAL RELIABILITY

Operational reliability is achieved by using proven components and by conducting tests required by the quality control requirements presented in chapter 17.0. All safety features systems are quality items meeting the requirements of 10 CFR 50, Appendix B, and seismically designed as discussed in chapter 3.0. Those safety features essential for post-tornado safety are designed to survive without loss of function the design tornado described in section 3.3.

Other sections of this report contain additional information on the safety features systems. Information on seismic requirements is provided in chapters 2.0 and 3.0. Information on the actuation instrumentation of the safety features system is provided in chapter 7.0.

Information on functions performed by components of the safety features systems during normal plant operation is provided in chapters 9.0 and 5.0. The safety analysis and demonstration of the ability of the safety features systems to provide adequate protection during accident conditions as provided in chapter 15.0.

The design bases, design description and evaluation, tests, inspections, and instrumentation for the safety features systems are presented in this chapter.

6.2 CONTAINMENT SYSTEMS

6.2.1 CONTAINMENT FUNCTIONAL DESIGN

6.2.1.1 Design Bases

6.2.1.1.1 Postulated Accident Conditions

The containment, in conjunction with engineered safety features, is designed to withstand the internal pressure and coincident temperature resulting from the energy release of the loss-of-coolant accident (LOCA) associated with 2831 MWt and to limit the site boundary radiation dose to within the guidelines set forth in 10 CFR 100. The containment system functional design meets the NRC acceptance criteria contained in General Design Criteria 16 and 50 of 10 CFR Part 50.

For the original containment analysis, the LOCA was assumed to occur for a range of reactor coolant pipe breaks, up to and including a double-ended break of the largest reactor coolant pipe. For power uprate and steam generator replacement (PU/RSG) only the limiting breaks were reanalyzed. The design bases also include a simultaneous loss of offsite electrical power and a failure of a single engineered safety feature (ESF).

The postulated accidents considered are as follows:

A. Double-ended pump suction guillotine (DEPSG), maximum ESF on safety injection (SI) flow. (Nonlimiting break not reanalyzed for PU/RSG.)

B. DEPSG, minimum ESF. (RSG analysis for initial pressure, P_O = +3 psig.)

C. 0.6 DEPSG, maximum ESF. (Nonlimiting break not reanalyzed for PU/RSG.)

D. 3 ft^2 pump suction split (PSS), maximum ESF. (Nonlimiting break not reanalyzed for PU/RSG.)

E. Double-ended cold leg guillotine (DECLG), maximum ESF. (Nonlimiting break not reanalyzed for PU/RSG.)

F. Double-ended hot leg guillotine (DEHLG), maximum ESF. (Nonlimiting break not reanalyzed for PU/RSG.)

G. Double-ended hot let guillotine (DEHLG), blowdown phase only. (RSG analysis for initial pressure, P_O = +3 psig.)

H. Spectrum of main steam line breaks.

Experience gained by the reactor manufacturer in analyzing several other dry containments has led to the conclusion that the hot leg break results in the highest containment peak pressure because it produces the highest blowdown mass and energy release rates. Since studies have confirmed that there is no reflood peak for the double-ended hot leg (DEHL) break, the analysis of the break extends only out to the end of blowdown. 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 flowrates during the post-blowdown period by including all of the available energy of the reactor coolant system (RCS) in calculating the releases to the containment.

The spectrum of break sizes together with minimum and maximum ESF have been studied to establish the upper bounds of containment pressure and temperature following the LOCA. The LOCA that results in the peak containment temperature at the end of blowdown is the DEHL break. The LOCA which produces the highest pressure in the post-blowdown period is the DEPSG. A composite of these two LOCA conditions is defined as the design basis accident (DBA). The containment response analyses assume that plant offsite power is lost and a single failure of any active containment or safety-related component occurs simultaneously with the hypothesized pipe rupture. Other postulated simultaneous occurrences such as a seismic event or local pipe break effects are not explicitly evaluated in these sections, except in the event that such occurrences might affect the mass and energy release to the containment. The effects of these other simultaneous occurrences upon the containment structural design are evaluated in section 3.8.

6.2.1.1.2 Post-Accident Energy Sources

In order to predict the peak containment pressure following an accident, energy sources are determined by the reactor manufacturer in the calculations of energy and mass release during postulated pipe break events. For reactor coolant pipe ruptures, energy sources include the stored heat energy contained within the primary coolant, SI water, fuel, cladding, reactor vessel internals, reactor vessel, reactor coolant piping, and steam generator secondary coolant and metal. Also considered as an energy source is the fission product decay energy generated within the reactor core. The available energy from the above source is added to the reactor coolant during the course of the event.

6.2.1.1.3 Contribution of Other Engineered Safety Features

After an accident, the ESFs, in conjunction with the containment and plant cooling water systems, provide protection for the public and plant personnel from the accidental release of radioactive products from the reactor system. The engineered safeguards function to localize, control, mitigate, and terminate all postulated accidents to ensure that the offsite radiation dose is within the guidelines of 10 CFR 100.

The ESFs consist of the following systems:

A. High-head SI system.

B. Low-head SI system.

C. Containment spray system.

D. Containment cooling system.

E. Penetration room filtration system.

Consistent with criteria concerning loss of plant offsite power, the containment heat removal systems and emergency core cooling systems (ECCSs) are assumed to operate in their minimum heat removal modes. However, due to the redundancy of these systems, the reactor core is adequately cooled so that core integrity is constantly maintained and heat is continuously removed from the core in an orderly and predictable manner. The functional performance of the containment and the ECCSs relies upon the operation of the containment isolation systems as described in subsection 6.2.4. Required isolation operations are assumed for purposes of the containment design evaluation in paragraph 6.2.1.3.

The high- and low-head SI systems inject borated water into the RCS. This provides cooling to limit fuel cladding and core damage and thus minimize fission product release. An adequate shutdown margin is assured regardless of the reactor coolant temperature. The SI system also provides continuous long-term post-accident cooling of the reactor core by recirculating borated water from the containment sump through the RCS.

The containment for each unit is equipped, as follows, with two independent, full capacity systems for cooling the containment atmosphere after the postulated LOCA:

A. The containment spray system supplies borated water to cool the containment atmosphere. The spray system in combination with one of the containment air coolers operating at reduced speed is sized to provide adequate cooling with one of the two containment spray pumps in service on emergency power. The pumps take suction from the refueling water storage tank (RWST). When the RWST reaches low-low level, the suction of the containment spray pumps is aligned to pump water directly from the containment sump back into the containment atmosphere by means of the containment spray nozzles.

   Trisodium phosphate is added to the recirculation sump to help reduce airborne iodine activity levels inside containment and to retain the removed iodine in solution in the sump

B. The containment cooling system is designed to provide containment atmosphere mixing and cooling. The system design basis is to provide adequate containment cooling from the operation of one train of containment spray and one containment cooler. The system limits the pressure transient inside the containment following a LOCA.

The penetration room filtration system collects and processes ECCS recirculation leakages. The system limits the environmental activity levels following a LOCA.

6.2.1.1.4 Subcompartment Differential Pressure

The original differential pressure analyses for the steam generator compartments were based on a double-ended circumferential rupture of the reactor coolant cold leg. The original differential pressure analysis for the reactor cavity was based on a restrained guillotine break of 100 in.^{2} in the cold-leg pipe. Currently, Leak-Before-Break (LBB) is applicable to the steam generator subcompartments; therefore, main RCS loop pipes are not postulated to break. The next largest line is the pressurizer surge line (LBB is also approved for the pressurizer surge line but is not credited in this analysis). Differential pressures are maintained below design limits providing adequate venting to the containment.

6.2.1.1.5 Post-Accident Pressure Reduction

The parameters affecting the assumed capability for post-accident pressure reduction are discussed in paragraph 6.2.1.3.

6.2.1.1.6 Rejection of Energy to the Outside Environment

Parameters affecting the assumed capability to reject energy to the outside environment are discussed in paragraph 6.2.1.1.3.

6.2.1.2 System Design

The design of the containment is based upon the mass and energy absorption capacity of the volume contained within the structure. The principal design parameters for the containment are given in table 6.2-1. These values are used for the pressure temperature analyses given in paragraph 6.2.1.3.3.

The design of the reactor and steam generator subcompartments considers applicable thermal, static, seismic, impingement force, and pressure loadings during a LOCA as described in section 3.8.

Materials compatibility considerations are discussed in appendix 6A.

6.2.1.3 Design Evaluation

6.2.1.3.1 Assurance of Containment Leaktightness

The containment leakage surveillance system, used to assure containment leaktightness during plant operation, is described in paragraph 6.2.1.4.

6.2.1.3.2 System Capability Analysis

A discussion of system capability is given in paragraph 6.2.1.4.

6.2.1.3.3 Containment Pressure Transient Analysis

A. Pipe Break Spectrum

In the event of a hypothetical LOCA, or main steam line break, the release of the coolant from the rupture area will cause the high pressure, high temperature fluid to rapidly flash to steam and water within the containment. The release of this mass and energy will result in a rise in the pressure and temperature of the containment atmosphere. The rate and magnitude of the pressure increase depend upon the nature, location, and size of the rupture. In order to establish the controlling rupture, a spectrum of primary and secondary coolant breaks is considered. The reactor coolant breaks examined are from a condition of full rated power. Secondary coolant system break analysis considers a spectrum of breaks at different power levels. All of these main steam line breaks allow complete blowdown of one steam generator. These postulated accidents are evaluated to determine their significance in selecting a containment design basis. The most severe of these accidents is selected as the controlling containment DBA.

B. Initial Conditions and Input Data

The containment pressure analysis input data have been based upon the final plant design. A conservative prediction of LOCA consequences has been assured by determining expected values of containment initial conditions and geometric and thermodynamic parameters. A thorough discussion of the input data is given below.

The containment design parameters which determine the net free internal volume, the containment surface areas, and the design pressure and temperature are given in tables 6.2-1 and 6.2-2. As an additional conservatism, the volume occupied by the reactor coolant prior to the LOCA is included as occupied volume rather than free volume.

The initial conditions within the RCS and the containment system prior to accident initiation are given in table 6.2-3. The containment pressure and temperature response analyses are conducted assuming a minimum of available ECCS is in operation. For the LOCA, the containment system is assumed to be at ambient pressure or the maximum pressure permitted under the Technical Specifications, and maximum inside and maximum outside design operating air temperatures to minimize heat transfer during a LOCA. Main steam line break (MSLB) cases were analyzed at varying initial pressure conditions as described below.

The containment heat sink data used in the LOCA analysis are fully described in tables 6.2-2 and 6.2-4. Table 6.2-2 lists the geometry of each heat sink and the way it is modeled for the analysis. An air gap equivalent to a thermal resistance of 0.01 ft/h/°F/Btu is postulated for the interface between the containment liner and wall and for the interface between the stainless steel refueling canal liner and the concrete.

Table 6.2-4 lists the material properties and heat transfer coefficients used in the analysis. The coating properties were supplied by the manufacturer. Metal and concrete properties are typical for the temperature range expected. The steel imbedded in the concrete was not considered in the concrete conductivity. Containment air cooler unit duty, Btu/h, as a function of containment saturation temperature, is given in figure 6.2-42. The RHR heat exchanger duty is given in figure 6.2-44 as a function of sump liquid temperature.

C. Accident Identification and Results

Containment pressure/temperature vs. time responses for the various breaks are shown on figures 6.2-1 through 6.2-41. The peak pressures, times of peak pressure, peak temperatures, times of peak temperature, and blowdown energy releases at the times of peak pressure are given in table 6.2-6 for the spectrum of breaks for LOCA. Based on the results presented, the DEHL with minimum ESF produces the maximum containment temperature over the short term while the DEPSG LOCA produces the maximum pressure. Both form the design basis for the short-term. The DEPSG LOCA produces limiting conditions over the post-blowdown period and is considered the design basis for the post-blowdown period. The limiting LOCA was reanalyzed as described in paragraph 6.2.1.3.3, and the resultant peak pressure is below the design pressure of 54 psig, as shown in table 6.2-6.

The blowdown mass and energy release rates as a function of time for the LOCA cases are presented in tables 6.2-10 and 6.2-14.

For the time of peak pressure, a detailed mass and energy balance has been performed on the RCS and containment. These data are given in tables 6.2-19 and 6.2-20. Table 6.2-19 lists the calculated containment pressures, temperatures, and masses for the time of pipe ruptures and the time of peak pressure with the limiting LOCA. Table 6.2-20 gives the energy distribution in the RCS and in the containment at the time of the break and at the time of peak pressure. This table verifies the energy balance during the containment pressurization period since the total RCS energy release equals the net gain in the containment system energy.

The containment condensing heat transfer coefficient vs. time for the DEPSG LOCA case is shown on figure 6.2-48. The initial portion of the curve is the Modified Tagami value with a maximum of 218 Btu/h-ft^2-°F at 21.6 s. After

21.6 s, the value decays to the Uchida value, which is dependent upon the slowly changing steam air mass ratio in the containment atmosphere.

6.2.1.3.4 Method of Analysis

6.2.1.3.4.1 LOCA Mass and Energy Releases

The uncontrolled release of pressurized high temperature reactor coolant, termed a LOCA, will result in release of steam and water into the containment. This, in turn, will result in increases in the local subcompartment pressures and an increase in the global containment pressure and temperature. Therefore, there are both long- and short-term issues reviewed relative to a postulated LOCA that must be considered at the PU/RSG conditions for Farley Units 1 and 2.

The long-term LOCA mass and energy releases are analyzed to 3600 seconds and are utilized as input in the containment integrity analysis, which demonstrates the acceptability of the containment safeguards systems to mitigate the consequences of a hypothetical large break LOCA. The containment safeguards systems must be capable of limiting the peak containment pressure to less than the design pressure and to limit the temperature excursion to less than the Environmental Qualification (EQ) acceptance limits. For this program, Westinghouse generated the mass and energy releases using the March 1979 model, described in reference 7. The NRC review and approval letter is included with references 7 and 27. Even though this is a first time application of this methodology for Farley Units 1 and 2, it has also been utilized and approved on many plant-specific dockets. This section discusses the long-term LOCA mass and energy releases analysis. The results of this analysis were provided for use in the containment integrity analysis and EQ reviews.

The short-term LOCA-related mass and energy releases are used as input to the subcompartment analyses, which are performed to ensure that the walls of a subcompartment can maintain their structural integrity during the short pressure pulse (generally < 3 s) accompanying a high-energy line pipe rupture within that subcompartment. The subcompartments evaluated included the steam generator compartment, the reactor cavity region, and the pressurizer compartment. For the reactor cavity region, the fact that Farley is approved for LBB was used to qualitatively demonstrate that any changes associated with PU/RSG are offset by the LBB benefit of using the smaller RCS nozzle breaks, thus demonstrating that the current licensing bases for these subcompartments remain bounding. LBB is also applicable to the steam generator subcompartments; therefore, main RCS loop pipes are not postulated to break. The next largest line is the pressurizer surge line, and mass and energy releases as discussed below are used in the PU/RSG analysis of the steam generator subcompartments. For the pressurizer compartment, the critical mass flux correlation utilized in the SATAN computer program (reference 10) was used to conservatively estimate the impact of the changes in RCS temperatures on the short-term releases. The power uprate replacement steam generator (PU/RSG) program evaluation showed that the releases would increase by 18% from the original design basis. The measurement uncertainty recapture (MUR) program evaluation showed that it would be bounded by PU/RSG program. This section discusses the short-term evaluation. The results of this evaluation were provided for use in the pressurizer subcompartment evaluation.

Long-Term LOCA Mass and Energy Releases

The mass and energy release rates described in this section form the basis of further computations by Southern Company Services (SCS) to evaluate the containment following the postulated accident. Discussed in this section are the long-term LOCA mass and energy releases for the hypothetical double-ended pump suction (DEPS) rupture and DEHL rupture break cases.

Input Parameters and Assumptions

The mass and energy release analysis is sensitive to the assumed characteristics of various plant systems, in addition to other key modeling assumptions. Where appropriate, bounding inputs are utilized and instrumentation uncertainties are included. For example, the RCS operating temperatures are chosen to bound the highest average coolant temperature range of all operating cases, and a temperature uncertainty allowance of (+6.0°F) is then added. Nominal parameters are used in certain instances. For example, the RCS pressure in this analysis is based on a nominal value of 2250 psia plus an uncertainty allowance (+50 psi). All input parameters are chosen consistent with accepted analysis methodology.

Some of the most critical items are the RCS initial conditions, core decay heat, SI flow, and primary and secondary metal mass and steam generator heat release modeling. Specific assumptions concerning each of these items are discussed next. Tables 6.2-7 through 6.2-9 present key data assumed in the analysis.

A core power of 2820.5 MWt representing rated thermal power (RTP) adjusted for calorimetric error was used in the analysis. As previously noted, the use of RCS operating temperatures to bound the highest average coolant temperature range were used as bounding analysis conditions. 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 to account for instrument error and deadband is reflected in the initial RCS temperatures. The selection of 2250 psia as the limiting pressure is considered to affect the blowdown phase results only, 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.

The rate at which the RCS blows down is initially more severe at the higher RCS pressure. Additionally, the RCS has a higher fluid density at the higher pressure (assuming a constant temperature) and subsequently has a higher RCS mass available for releases. Thus, 2250 psia plus uncertainty was selected for the initial pressure as the limiting case for the long-term mass and energy release calculations.

The selection of the fuel design features for the long-term mass and energy release calculation is based on the need to conservatively maximize the energy stored in the fuel at the beginning of the postulated accident (i.e., to maximize the core stored energy). The core stored energy was selected to bound the 17 x 17 optimized fuel assembly (OFA) fuel product loaded at Farley Units 1 and 2. The margins in the core stored energy address the thermal fuel model and the associated manufacturing uncertainties and the time in the fuel cycle for the maximum fuel densification. Thus, the analysis very conservatively accounts for the stored energy in the core.

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

A uniform steam generator (SG) tube plugging level of 0% is modeled. This assumption maximizes the reactor coolant volume and fluid release by virtue of consideration of the RCS fluid in all SG tubes. During the post-blowdown period the SGs are active heat sources since significant energy remains in the secondary metal and secondary mass that has the potential to be transferred to the primary side. The 0% tube plugging assumption maximizes heat transfer area and, therefore, the transfer of secondary heat across the SG tubes. Additionally, this assumption reduces the reactor coolant loop resistance, which reduces the ΔP upstream of the break for the pump suction breaks and increases break flow. Thus, the analysis very conservatively accounts for the level of SG tube plugging.

Regarding SI flow, the mass and energy release calculation considered configurations/failures to conservatively bound respective alignments. The cases include (a) a Minimum Safeguards case (1 CH/SI and 1 LHSI pump); and (b) a Maximum Safeguards case (2 CH/SI and 2 LHSI pumps).

The following assumptions were employed to ensure that the mass and energy releases are conservatively calculated, thereby maximizing energy release to containment.

1. Maximum expected operating temperature of the RCS (100% full-power conditions).
2. Allowance for RCS temperature uncertainty (+6.0°F).
3. Margin in RCS volume of 3% (which is composed of 1.6% allowance for thermal expansion and 1.4% for uncertainty).
4. Analyzed core power of 2830.5-MWt.
5. Item #4 includes an allowance for a plant specific calorimetric error.
6. Conservative heat transfer coefficients (i.e., steam generator primary/secondary heat transfer and RCS metal heat transfer).
7. Allowance in core stored energy for effect of fuel densification.
8. A margin in core stored energy to bound OFA fuel.
9. An allowance for RCS initial pressure uncertainty (+50 psi).
10. A maximum containment backpressure equal to design pressure (54 psig).
11. Allowance for RCS flow uncertainty (-2.4%).

12. Steam generator tube plugging leveling (0% uniform).
    - Maximizes reactor coolant volume and fluid release.
    - Maximizes heat transfer area across the SG tubes.
    - Reduces coolant loop resistance, which reduces the ΔP upstream of the break for the pump suction breaks and increases break flow.

Limiting releases bounding for both units are provided herein.

Thus, based on the previously discussed conditions and assumptions, a bounding analysis of Farley Units 1 and 2 was made for the release of mass and energy from the RCS in the event of a LOCA.

Description of Analyses

The evaluation model used for the long-term LOCA mass and energy release calculations is the March 1979 model described in reference 7. This evaluation model has been reviewed and approved generically by the NRC. The approval letter is included with reference 7 and reference 27. Even though this is a first time application for Farley Units 1 and 2, it has also been utilized and approved on the plant-specific dockets for other Westinghouse PWRs.

This report section presents the long-term LOCA mass and energy releases generated in support of the Farley Units 1 and 2 uprating program. These mass and energy releases are then subsequently used in the containment integrity analysis.

LOCA M&E Release Phases

The containment system receives mass and energy releases following a postulated rupture in the RCS. These releases continue over a time period which, for the LOCA mass and energy analysis, is typically divided into four phases.

1. Blowdown - the period of time from accident initiation (when the reactor is at steady-state operation) to the time that the RCS and containment reach an equilibrium state.

2. Refill - the period of time when the lower plenum is being filled by accumulator and ECCS water. At the end of blowdown, a large amount of water remains in the cold legs, downcomer, and lower plenum. To conservatively consider the refill period for the purpose of containment mass and energy releases, it is assumed that this water is instantaneously transferred to the lower plenum along with sufficient accumulator water to completely fill 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 phase. For the pump suction break, a two-phase mixture exits the core, passes through the hot legs, and is superheated in the SGs prior to exiting the break as steam. After the broken loop SG cools, the break flow becomes two-phase.

Computer Codes

The reference 7 mass and energy release evaluation model is comprised of mass and energy release versions of the following codes: SATAN VI, WREFLOOD, FROTH, AND EPITOME. These codes were used to calculate the long-term LOCA mass and energy releases for Farley Units 1 and 2.

SATAN VI calculates blowdown, the first portion of the thermal-hydraulic transient following break initiation including pressure, enthalpy, density, mass and energy flowrates, and energy transfer between primary and secondary systems as a function of time.

•

The WREFLOOD code addresses the portion of the LOCA transient where the core reflooding phase occurs after the primary coolant system has depressurized (blowdown) due to the loss of water through the break and when water supplied by the ECCS refills the reactor vessel and provides cooling to the core. The most important feature of WREFLOOD is the steam/water mixing model.

FROTH models the post-reflood portion of the transient. The FROTH code is used for the SG heat addition calculation from the broken and intact loop SGs.

EPITOME continues the FROTH post-reflood portion of the transient from the time at which the secondary equilibrates to containment design pressure to the end of the transient. It also compiles a summary of data on the entire transient, including formal instantaneous mass and energy release tables and mass and energy balance tables with data at critical times.

Break Size and Location

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

Three distinct locations in the RCS loop can be postulated for pipe rupture for any release purposes:

- Hot leg (between vessel and steam generator).
- Cold leg (between pump and vessel).
- Pump suction (between steam generator and pump).

The break locations analyzed for this program are the double-ended pump suction (DEPS) rupture (10.48 ft^2) and the DEHL rupture (9.154 ft^2). Break mass and energy releases have been calculated for the blowdown, reflood, and post-reflood phases of the LOCA for the DEPS cases. For the DEHL case, the releases were calculated only for the blowdown. The following information provides a discussion on each break location.

The DEHL rupture has been shown in previous studies to result in the highest blowdown mass and energy release rates. Although the core flooding rates would be the highest for this break location, the amount of energy released from the SG secondary is minimal because the majority of the fluid which exits the core vents directly to containment bypassing the SGs. 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 SGs before venting through the break. For the hot leg break, generic studies have confirmed that there is no reflood peak (i.e., from the end of the blowdown period the containment pressure would continually decrease). Therefore, only the mass and energy releases for the hot leg break blowdown phase are calculated and presented in this section of the report.

The cold leg break location has also been found in previous studies to be much less limiting in terms of the overall containment energy releases. 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 reduced. Therefore, the cold leg break is bounded by other breaks and no further evaluation is necessary.

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 flowrates during the post-blowdown period by including all of the available energy of the RCS in calculating the releases to containment.

Application of Single-Failure Criterion

An analysis of the effects of the single-failure criterion has been performed on the mass and energy release rates for each break analyzed. 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 SI system. This is not an issue for the blowdown period which is limited by the DEHL break.

Two cases have been analyzed to assess the effects of a single failure. The first case assumes minimum safeguards SI flow based on the postulated single failure of an emergency diesel generator. This results in the loss of one train of safeguards equipment. The other case assumes maximum safeguards SI flow based on no postulated failures that would impact the amount of ECCS flow. The analysis of the cases described provides confidence that the effect of credible single failures is bounded.

Acceptance Criteria for Analyses

A large break LOCA is classified as an ANS Condition IV event, an infrequent fault. To satisfy the Nuclear Regulatory Commission (NRC) acceptance criteria presented in the Standard Review Plan section 6.2.1.3, the relevant requirements are as follows:

- 10 CFR 50, Appendix A.
- 10 CFR 50, Appendix K, paragraph I.A.

In order to meet these requirements, the following must be addressed:

- Sources of energy.
- Break size and location.
- Calculation of each phase of the accident.

M&E Release Data

Blowdown Mass and Energy Release Data

The SATAN-VI code is used for computing the blowdown transient. The code utilizes the control volume (element) approach with the capability for modeling a large variety of thermal fluid system configurations. The fluid properties are considered uniform and thermodynamic equilibrium is assumed in each element. A point kinetics model is used with weighted feedback effects. The major feedback effects include moderator density, moderator temperature, and Doppler broadening. A critical flow calculation for subcooled (modified Zaloudek), two-phase (Moody), or superheated break flow is incorporated into the analysis. The methodology for the use of this model is described in reference 7.

Table 6.2-10 presents the calculated mass and energy release for the blowdown phase of the DEHL break. For the hot leg break mass and energy release tables, break path 1 refers to the mass and energy exiting from the reactor vessel side of the break; break path 2 refers to the mass and energy exiting from the SG side of the break.

Table 6.2-14 presents the calculated mass and energy releases for the blowdown phase of the DEPS break. For the pump suction breaks, break path 1 in the mass and energy release tables refers to the mass and energy exiting from the SG side of the break; break path 2 refers to the mass and energy exiting from the pump side of the break.

Reflood Mass and Energy Release Data

The WREFLOOD code is used for computing the reflood transient. The WREFLOOD code consists of two basic hydraulic models - one for the contents of the reactor vessel and one for the coolant loops. The two models are coupled through the interchange of the boundary conditions applied at the vessel outlet nozzles and at the top of the downcomer. Additional transient phenomena such as pumped SI and accumulators, reactor coolant pump performance, and SG release are included as auxiliary equations which interact with the basic models as required. The WREFLOOD code permits the capability to calculate variations during the core reflooding transient of basic parameters such as core flooding rate, core and downcomer water levels, fluid thermodynamic conditions (pressure, enthalpy, density) throughout the primary system, and mass flowrates through the primary system. The code permits hydraulic modeling of the two flow paths available for discharging steam and entrained water from the core to the break, i.e., the path through the broken loop and the path through the unbroken loops.

A complete thermal equilibrium mixing condition for the steam and ECCS injection water during the reflood phase has been assumed for each loop receiving ECCS water. This is consistent with the usage and application of the reference 7 mass and energy release evaluation model in the initial docketed analysis for this methodology, e.g., D.C. Cook Docket (reference 8). Even though the reference 7 model credits steam/water mixing only in the intact loop and not in the broken loop, the justification, applicability, and NRC approval for using the mixing model in the broken loop has been documented (reference 8). Moreover, this assumption is supported by test data and is further discussed below.

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 two distinct physical processes. The first is a two-phase interaction with condensation of steam by cold ECCS water. The second is a single-phase mixing of condensate and ECCS water. Since the steam release 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/water mixing test data have been reviewed for validation of the containment integrity reflood steam/water mixing model. These data were generated in 1/3-scale tests (reference 9), which are the largest scale data available and, thus, most clearly simulate 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.

A group of 1/3-scale tests corresponds directly to containment integrity reflood conditions. The injection flowrates 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 7. For all of these tests, the data clearly indicate the occurrence of very effective mixing with rapid steam condensation. The mixing model used in the 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 post-blowdown limiting break for the containment integrity peak pressure analysis is the pump suction double-ended rupture break. For this break, there are two flow paths available in the RCS by which mass and energy may be released to containment. One is through the outlet of the SG, 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 and then vents 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 7 and 9.

Tables 6.2-20 and 6.2-46 present the calculated mass and energy releases for the reflood phase of the pump suction double-ended rupture, minimum safeguards, and maximum safeguards cases, respectively.

The transient response of the principal parameters during reflood are given in tables 6.2-42 and 6.2-47 for the DEPS cases.

Post-Reflood Mass and Energy Release Data

The FROTH code (reference 10) is used for computing the post-reflood transient. The FROTH code calculates the heat release rates resulting from a two-phase mixture present in the SG tubes. The mass and energy releases that occur during this phase are typically superheated due to the depressurization and equilibration of the broken loop and intact loop SGs. During this phase of the transient, the RCS has equilibrated with the containment pressure, but the SGs contain a secondary inventory at an enthalpy that is much higher than the primary side. Therefore, there is a significant amount of reverse heat transfer that occurs. Steam is produced in the core due to core decay heat. For a pump suction break, a two-phase fluid exits the core, flows through the hot legs, and becomes superheated as it passes through the SG. Once the broken loop cools, the break flow becomes two-phase. During the FROTH calculation, ECCS injection is addressed for both the injection phase and the recirculation phase. The FROTH code calculation stops when the secondary side equilibrates to the saturation temperature (T_{sat}) at the containment design pressure; after this point the EPITOME code completes the SG depressurization. (See the subsection “Steam Generator Equilibration and Depressurization” for additional information.)

The methodology for the use of this model is described in reference 7. The mass and energy release rates are calculated by FROTH and EPITOME until the time of containment depressurization. After containment depressurization (14.7 psia), the mass and energy release available to containment is generated directly from core boiloff/decay heat.

Tables 6.2-43 and 6.2-48 present the two-phase post-reflood mass and energy release data for the pump suction double-ended cases.

Decay Heat Model

On November 2, 1978, the Nuclear Power Plant Standards Committee (NUPPSCO) of the American Nuclear Society approved ANS Standard 5.1 (reference 11) for the determination of decay heat. This standard was used in the mass and energy release model for Farley Nuclear Plant Units 1 and 2. Table 6.2-54 lists the decay heat curve used in the Farley SG replacement mass and energy release analysis.

Significant assumptions in the generation of the decay heat curve for use in the LOCA mass and energy releases analysis include the following:

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 has been taken from Table 10 of reference 11.

5. The fuel has been assumed to be at full power for 10^8 s.

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

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

Based upon NRC staff review, Safety Evaluation Report (SER) of the March 1979 evaluation model (reference 7), use of the ANS Standard 5.1, November 1979 heat model was approved for the calculation of mass and energy releases to the containment following a LOCA.

Steam Generator Equilibration and Depressurization

SG equilibration and depressurization is the process by which secondary side energy is removed from the SGs in stages. The FROTH computer code calculates the heat removal from the secondary mass until the secondary temperature is the saturation temperature (T_{sat}) at the containment design pressure. After the FROTH calculations, the EPITOME code continues the FROTH calculation for SG cooldown removing SG secondary energy at different rates (i.e., first- and second-stage rates). The first-stage rate is applied until the SG reaches T_{sat} at the user specified intermediate equilibration pressure, when the secondary pressure is assumed to reach the actual containment pressure. Then the second-stage rate is used until the final depressurization, when the secondary reaches the reference temperature of T_{sat} at 14.7 psia, or 212°F. The heat removal of the broken loop and intact loop SGs is calculated separately.

During the FROTH calculations, SG heat removal rates are calculated using the secondary side temperature, primary side temperature, and a secondary side heat transfer coefficient determined using a modified McAdam's correlation (reference 12). SG energy is removed during the FROTH transient until the secondary side temperature reaches saturation temperature at the containment design pressure. The constant heat removal rate used during the first heat removal stage is based on the final heat removal rate calculated by FROTH. The SG energy available to be released during the first-stage interval is determined by calculating the difference in secondary energy available at the containment design pressure and that at the (lower) user specified intermediate equilibration pressure, assuming saturated conditions. This energy is then divided by the first-stage energy removal rate, resulting in an intermediate equilibration time. At this time, the rate of energy release drops substantially to the second stage rate. The second stage rate is determined as the fraction of the difference in secondary energy available between the intermediate equilibration and final depressurization at 212°F, and the time difference from the time of the intermediate equilibration to the user specified time of the final depressurization at 212°F. With current methodology, all of the secondary energy remaining after the intermediate equilibration is conservatively assumed to be released by imposing a mandatory cooldown and subsequent depressurization down to atmospheric pressure at 3600 s, i.e., 14.7 psia and 212°F.

Sources of Mass and Energy

The sources of mass considered in the LOCA mass and energy release analysis are given in tables 6.2-12, 6.2-44, and 6.2-49. These sources are the RCS, accumulators, and pumped SI.

The energy inventories considered in the LOCA mass and energy release analysis are given in tables 6.2-13, 6.2-45, and 6.2-50. The energy sources are listed below.

1. RCS water.
2. Accumulator water (all three inject).
3. Pumped SI water.
4. Decay heat.
5. Core stored energy.
6. RCS metal (includes SG tubes).
7. SG metal (includes transition cone, shell, wrapper, and other internals).
8. SG secondary energy (includes fluid mass and steam mass).
9. Secondary transfer of energy (feedwater into and steam out of the SG secondary).

The energy reference points are as follows:

1. Available Energy: 212°F; 14.7 psia.
2. Total Energy Content: 32°F; 14.7 psia.

The mass and energy inventories are presented in these balance tables 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 broken loop SG equilibration to pressure setpoint.
6. Time of intact loop SG equilibration to pressure setpoint.

7. Time of full depressurization (3600 s).

In the mass and energy release data presented, no Zirc-water reaction heat is specifically presented because the clad temperature does not rise high enough for the rate of the Zirc-water reaction heat to be of any significance.

The sequence of events for the LOCA mass and energy release analysis are shown in tables 6.2-51 through 6.2-53.

Conclusions

The consideration of the various energy sources in the long-term mass and energy release analysis provides assurance that all available sources of energy have been included in this analysis. Thus, the review guidelines presented in Standard Review Plan Section 6.2.1.3 have been satisfied. The results of this analysis were provided for use in the containment integrity analysis.

Short-Term LOCA Mass and Energy Releases

Purpose

An evaluation was conducted to determine the effect of a MUR program on the PU/RSG program results for short-term LOCA-related mass and energy releases that support subcompartment analyses discussed in section 6.2 of the Farley FSAR. From the FSAR, a double-ended circumferential rupture of the reactor coolant cold leg forms the basis for the steam generator compartments, a 100 in^2 reactor vessel inlet break forms the basis for the reactor cavity region, and both a spray line break and a surge line break were considered for the pressurizer compartment. This evaluation addressed the impact of MUR and other relevant issues on the current licensing basis for these four breaks.

Discussion and Evaluation

The subcompartment analysis is performed to ensure that the walls of a subcompartment can maintain their structural integrity during the short pressure pulse (generally < 3 s) which accompanies a high-energy line pipe rupture within the subcompartment. The magnitude of the pressure differential across the walls is a function of several parameters, which include the blowdown M&E release rates, the subcompartment volume, vent areas, and vent flow behavior. The blowdown M&E release rates are affected by the initial RCS temperature conditions. Since short-term releases are linked directly to the critical mass flux, which increases with decreasing temperatures, the short-term LOCA releases would be expected to increase due to any reductions in RCS coolant temperature conditions. Short-term blowdown transients are characterized by a peak mass and energy release rate that occurs during a subcooled condition; thus the Zaloudek correlation, which models this condition, is currently used in the short-term LOCA mass and energy release analyses with the SATAN computer program.

This correlation was used to conservatively evaluate the impact of the changes in RCS temperature conditions due to MUR on the short-term releases. The evaluation concluded that the RCS temperature conditions due to the PU/RSG program were bounding so the current analysis of record short-term LOCA mass and energy releases are also bounding.

Any changes in RCS volume, SG liquid/steam mass and volume, and differences in units have no effect on the releases because of the short duration of the postulated accident. Any volumetric changes are small and have no impact on the subcompartment model. Therefore, the only change that needs to be addressed for this program is the decreased RCS coolant temperatures.

For this MUR, a RCS pressure of 2250 psia, a vessel outlet temperature of 605.0°F, and a vessel/core inlet temperature of 530.6°F were considered. Considering the temperature uncertainty of 6°F and the current licensing basis initial conditions, the following RCS temperature ranges were used for the evaluations herein:

- Vessel Outlet 616.9°F to 597.8°F
- Vessel/Core Inlet 563.1°F to 524.6°F

A pressure uncertainty of 50 psi was also included in the evaluation.

Based upon the results of the MUR evaluation, the current design basis LOCA-related mass and energy releases, including the spray line and the surge line releases, would be bounded by the 1.18 factor that was determined for the PU/RSG program over the original design basis mass and energy releases due to RCS temperature effects.

Per reference 13, Farley is approved for LBB. LBB eliminates the dynamic effects of postulated primary loop pipe ruptures from the design basis. This means that the current breaks (a double-ended circumferential rupture of the reactor coolant cold leg break for the SG compartments and a 144-in^2 reactor vessel inlet break for the reactor cavity region) no longer have to be considered for the short-term effects. Since the RCS piping has been eliminated from consideration, the large branch nozzles must be considered for design verification. This includes the surge line, accumulator line, and the RHR line. These smaller breaks, which are outside the cavity region, would result in minimal asymmetric pressurization in the reactor cavity region. Additionally, compared to the large RCS double-ended ruptures, the differential loadings are significantly reduced. For example, the peak break compartment pressure can be reduced by a factor of > 2, and the peak differential across an adjacent wall can be reduced by a factor of > 3 if the nozzle breaks are considered. Therefore, since Farley is approved for LBB, the decrease in mass and energy releases associated with the small RCS nozzle breaks, as compared to the larger RCS pipe breaks, more than offsets the increased releases associated with decreased RCS initial coolant temperatures. The SG and pressurizer subcompartments were re-analyzed for PU/RSG using the pressurizer surge line increased mass and energy blowdown data. The current licensing basis subcompartment analyses that consider breaks in the RCS remain bounding.

Results and Conclusions

The short-term LOCA-related mass and energy releases discussed in section 6.2 of the Farley FSAR have been reviewed to assess the effects associated with the Farley Steam Generator Replacement Project. Results show that the current design basis spray line and surge line releases would increase 18% from the original design basis analysis due to RCS temperature effects. Therefore, the mass rates (lb/s) in FSAR tables 6.2-16 and 6.2-17 have been multiplied by 1.18. The results of this evaluation were used in the steam generator and pressurizer subcompartment structural analyses and are shown in table 6.2-22. Since Farley is approved for LBB, the decrease in mass and energy releases associated with the smaller RCS nozzle breaks, as compared to the larger RCS pipe breaks, more than offsets the increased releases associated with decreased RCS initial coolant temperatures. The current licensing basis subcompartment analyses that consider breaks in the primary loop RLS piping (i.e., SG subcompartment and reactor cavity region), therefore, remain bounding.

Evaluation of Impact on LOCA Mass and Energy Release Analysis of Closing MOV8887A/ MOV8887B and MOV8888B/MOV8888A During Quarterly RHR Pump Inservice Testing

During the performance of quarterly RHR pump inservice testing, MOV8887B and MOV8888A are closed when the A RHR pump is tested, and MOV8887A and MOV8888B are closed when the B RHR pump is tested. The closure MOV8887A/MOV8887B and MOV8888B/MOV8888A affects both trains of RHR, since the ECCS flow analysis assumes that this flow path is open to provide an even flow distribution to all three cold legs regardless of which RHR pump is operating.

The LOCA mass and energy (M&E) release analysis assumes the failure of an emergency diesel generator, which results in the loss of one train of ECCS (one charging/HHSI pump and one RHR/LHSI pump), one containment spray train, and one containment cooling train. The LOCA M&E release analysis assumes the minimum flow provided by one train of ECCS (one charging/HHSI pump and one RHR/LHSI pump). The RHR pump that is tested is declared inoperable during quarterly RHR pump testing and the Technical Specification Actions are entered. No additional failures are required to be considered when the Actions are entered, since the Actions only allow operation to continue for the duration of the Completion Time, prior to restoring the affected equipment to operable status. Therefore, the reduced ECCS flows available during the testing of an RHR pump are based on two charging pumps and one RHR pump.

The ECCS flow that would be provided during RHR pump testing is approximately 2.6 lbm/s less than the ECCS flow that is assumed in the current design basis LOCA M&E analysis. The ECCS flow reduction results in a penalty (increase) of approximately 0.03% in the integrated energy released during the first hour of the transient. An increase of this magnitude is judged to be insignificant with regard to the containment response during the transient. Additionally, both trains of containment spray and containment cooling systems would be available to further reduce the containment response during the transient.

Therefore, it can be concluded that there is no adverse impact on the long-term LOCA M&E releases and containment integrity analyses. The short-term LOCA M&E releases are not impacted since the ECCS is not actuated during the short-term transients.

6.2.1.3.4.2 **Containment Pressure Analysis.** The containment pressure analyses to determine the limiting LOCA were performed using the GOTHIC computer program that was developed for the purpose of transient analysis of atmospheres in multicompartment containments of water-cooled nuclear power plants. The use of GOTHIC for FNP containment pressure analyses was reviewed and approved by the NRC (reference 26).

The GOTHIC model predicts both the pressure and temperature within the containment regions and the temperatures in the containment structures. It is assumed that separate blowdown and core thermal behavior studies have been made by the nuclear steam supply system (NSSS) manufacturer to determine mass and/or energy input rates from sources such as: the release of reactor coolant, chemical reactions, and decay energy and sensible heat release which may cause heating or boiloff of residual water in the reactor vessel or superheating of steam as it passes through the reactor system and enters the containment through the postulated point of RCS rupture.

The GOTHIC model treats the containment and the heat transfer surfaces following a LOCA. Included in this model are ESF and analytical techniques to enable calculation of their effects upon the containment. Several options have been incorporated in the model to facilitate use of these features.

GOTHIC calculates a pressure-time transient with stepwise iteration between the thermodynamic state points. The iterations are based on the laws of the conservation of mass and energy together with their thermodynamic relationships. Superposition of heat input functions is assumed so that any combination of coolant release, decay heat generation, and sensible heat release can be used with appropriate ESF features to determine the containment pressure-time history associated with a LOCA.

The program uses a three-phase containment model consisting of the containment atmosphere (vapor phase), the sump (liquid phase), and falling drops (drop phase). Mass and energy are transferred between the regions by boiling, condensation, or liquid dropout. Evaporation is limited to 8% of the condensed steam. Heat transfer between the sump liquid and atmosphere vapor regions is modeled using the GOTHIC internal interfacial heat transfer model. Heat transfer in this mode is very small, so the inclusion of this mode of heat transfer is of no significance to the results of the analysis. Each region is assumed homogeneous, but a temperature difference can exist between regions. Moisture condensed in the vapor region during a time increment will agglomorate into drops which will fall into the liquid region. Noncondensible gases are included in the vapor region. Thermodynamic state points of steam in the saturated and superheated state are based on models developed for the COBRA-NC and COBRA-TRAC codes. The models are described in reference 22.

6.2.1.3.5 Special Pressure Reduction Containment Concepts

This section is not applicable.

6.2.1.3.6 Long-Term Containment Performance

The results of the long-term limiting LOCA case (DEPSG) have been evaluated to verify the ability of the ECCS and containment heat removal system (CHRS) to keep the reactor vessel flooded and maintain the containment below the design conditions following a LOCA, as described in paragraphs 6.2.1.3.12, 6.2.1.3.13, and 6.2.1.3.14. This evaluation has been based upon conservatively assumed performance of the ESFs. The containment fan cooling units are actuated at 21.7 psia (Containment High-1 Safety Analysis Limit). Since High-1 occurs prior to diesel start, a 92-s delay is assumed from the start of the accident. This 92-s delay includes 15 s for LOSP signal generation and diesel start, a 2-s High-1 signal delay, and 75 s for the containment cooler service water inlet motor-operated valce (MOV) to stroke fully open. The containment spray system is actuated at 44.7 psia (Containment High-3 Safety Analysis Limit) with a 62-s delay from the start of the accident since High-3 also occurs before the fastest expected diesel loading. This time includes 15 s for LOSP signal generation and diesel start, 12 s for the spray discharge MOV to stroke open, and 35 s to fill the spray discharge header. The ESF sequencer and spray pump start times are bounded by the MOV stroke time.

The residual decay heat rate is shown in figure 15.1-6 as watts per watt at maximum NSSS power level, 2785 MWt. Sensible heat remaining in the primary and secondary systems at the end of the post reflood failure is added to the reactor vessel water, as listed in table 6.2-14. These criteria aid in assuring a conservative prediction of the third containment pressure peak, which occurs during sump water recirculation, and demonstrate the ability of the CHRS to maintain pressure effectively at a fraction of that value occurring during blowdown.

The containment pressure time response for the limiting LOCA case out to 10^5 s is shown in figure 6.2-2 for the minimum safeguards performance mode outlined in table 6.2-5. The maximum pressure of 43.8 psig occurs at 552 s for the DEPSG break. A DEPSG break initial peak pressure is followed by a decrease in containment pressure during refill. At 2139 s, SI water from the sump begins to recirculate as the RWST reaches low level. The containment continues to depressurize until 4316 s, when containment spray water begins to recirculate from the sump as the RWST reaches low-low level. During sump water recirculation, a third pressure peak occurs due to steam evolution from the reactor because of boiloff of the hotter core injection water. The containment atmosphere and sump temperatures versus time are given in figure 6.2-41 for the DEPSG. The peak atmosphere temperature of 263°F occurs at 552 s. The maximum sump water temperature of 260°F occurs at 1252 s. The DEHL break is slightly higher for atmosphere temperature with a peak of 264°F at 18.7 s.

The temperature response of the containment structure is shown in figure 6.2-78 at several points in time for the limiting LOCA. The containment wall liner plate reaches a maximum temperature of approximately 250°F at 1000 s.

6.2.1.3.7 Accident Chronology

A chronology of events with time for the limiting LOCA case (DEPSG) is given in table 6.2-21 from the time of pipe rupture to 2.6 x 10^6 s when accident calculations were terminated. At that time, the containment pressure is 4.4 psig. It is assumed that time equals zero at the start of the LOCA.

6.2.1.3.8 Energy Balance

An energy balance for the limiting LOCA is given in table 6.2-20.

6.2.1.3.9 Post-LOCA Parameters

This section contains plots of various post-LOCA parameters as a function of time. The heat generation rate from core decay heat is shown in figure 15.1-6. The heat removal rate from the RHR heat exchanger and from the containment air cooler is shown in figures 6.2-79 and 6.2-80, respectively. The containment pressure/temperature vs. time profiles are shown in figures 6.2-1 through 6.2-5 and figure 6.2-40. Refer to paragraph 6.2.1.3.14 for subsequent evaluations regarding post-accident containment performance.

6.2.1.3.10 Containment Subcompartment Analysis

The NRC acceptance criteria associated with compartment design are based on their acceptance of the analysis techniques described in this section which demonstrate that compartment calculated peak pressures for postulated pipe ruptures do not exceed compartment design pressures. Postulated breaks in the reactor coolant loop (RCL), except for branch line connections, have been eliminated from the structural design basis for both Unit 1 and 2, as allowed by the revision of GDC-4. The elimination of these breaks is a result of application of LBB technology.

A. <u>Steam Generator Compartments</u>

Short-term differential pressurization across the secondary shield walls following a LOCA in the SG compartments was calculated with the Bechtel computer program COPDA. This program is described in attachment D of appendix 3K. The compressible flow option was used.

The block diagram for the break in SG C compartment is shown in figure 6.2-53. All major obstructions such as columns, tanks, and the SGs are accounted for in the volume and vent calculations, and the results are reduced by 10 percent to allow for minor obstructions such as cable trays and small pipes. Flow through the reactor cavity is neglected. Flow coefficients are calculated as for long passageways, since in each case L/D > 2. Entrance, exit, head, and friction losses are included. Each loss factor at cross-sectional areas greater than the minimum are scaled to the minimum by the square of the area ratio. Thus,

$$K_{total} = \Sigma_i K_i \left(\frac{A_{min}}{Ai} \right)^2$$

Then the flow coefficient is calculated by

$$C = \frac{1}{\sqrt{K_{total}}}$$

The peak differential pressures for the cold leg breaks are listed in table 6.2-22. The highest pressure (33.9 psid) is in the C compartment, since the refueling canal support and the incore instrumentation tunnel extension impede the outflow. The differential pressure for this case is plotted in figure 6.2-54. The blowdown mass and energy flowrates are listed in table 6.2-10. All pressures are below the allowable, as shown in table 6.2-22.

The hot leg break in the C compartment was also analyzed. Since the resulting peak is substantially lower than that for the cold leg break, the other compartments were not analyzed. The pump suction break yields lower flowrates than the cold leg breaks, so these were not analyzed either.

Application of LBB to the SG subcompartments eliminates the large RCS loop breaks. SG subcompartment C was reanalyzed for the RSG conditions with the pressure surge line mass and energy blowdown data provided in table 6.2-17. The results of this reanalysis confirm that the original analyses remain bounding, as shown in table 6.2-22.

B. Reactor Cavity

The breaking of the RCL at the reactor nozzle will result in differential loadings on the reactor vessel caused by a discharge of steam and liquid at the postulated breakpoint. The mass rate of flow of the discharge fluid will be a function of:

1. The opening area of the break.
2. The internal energy of the discharging fluid.

The loading imposed on the vessel as a result of the break will be influenced by:

1. Vent openings from the break compartment to the reactor cavity compartments.
2. Thrust loads imparted by the fluid discharge from the nozzle at the postulated breakpoint
3. Vent areas from the break compartments to the containment in a direction away from the reactor cavity
4. The transient movement of vapor from the break as it circumvents the reactor vessel.

Opening area of the postulated break:

The double-ended guillotine break occurs at the weld connecting the cold leg pipe to the nozzle. Pipe movement is restrained so that the break area is < 100 in.^{2} A baffle plate is attached to the cavity annulus. Thus, most of the flow is through the inspection holes, which are open under normal operating conditions.

The blowdown area is then a function of pipe movement, vessel movement, and reactor coolant pump movement. Blowdown area is effectively minimized by providing a pipe restraint on all the legs of the RCL, located at a point as close as possible to the postulated breakpoint. The calculated discharge area accounting for vessel movement, pump movement, and restrained pipe whip is 100 in^2 or less. The calculated discharge area accounting for the relative motion of the broken pipe end as determined from the reactor pressure vessel and RCL structural analyses is 85 in^2. (See appendix 3M.)

See drawings D-176275, D-176277, D-176278, D-176279, D-206275, D-206277, D-206278, and D-206279 for a physical layout describing the cavity area. For the physical layout of the area under the reactor, see drawings D-176107 and D-206107.

Internal energy of the fluid:

The internal energy of the fluid was determined by analyzing the thermodynamic properties of the entire reactor loop as it exists at the time of the postulated break. A cold leg break was found to be the most severe case for fluid discharge and was therefore used in the pressurization analysis. In addition, a break area of 100 in.^{2} was used to determine the blowdown data, thereby making the data conservative when compared to the actual calculated opening of 85 in.^{2}. The blowdown data for a 100 in.^{2} break are listed in table 6.2-15.

Thrust loads imparted by the fluid discharge from the nozzle at the postulated break point:

These loads were determined by the rate of fluid discharge and the energy released during discharge.

Vent areas from the break area to the containment:

Venting from the break compartment to the inspection plug opening was considered as a factor in the final calculations. The inlet nozzle at 215° was selected as the break location because that penetration has the smallest inspection plug opening, approximately 2033 in.^{2} Selection of this location, when coupled with cold leg blowdown data, presents the most conservative case for analysis of reactor cavity pressure response.

Transient movement of vapor from the break compartment as it circumvents the vessel:

Loads are imparted on the reactor vessel by virtue of the fact that vapor movement around the vessel is not instantaneous.

Cavity Model:

The reactor cavity was subdivided into compartments at all significant flow restrictions, such as hot and cold leg nozzles. Compartment 1 is below the nozzle centerline in the penetration at 215° and is outside the blowdown restrictor plate. Compartment 2 is directly above the nozzle centerline and includes the inspection plug opening to the containment. Blowdown is assumed to split evenly between compartments 1 and 2.

The annulus compartmentalization is illustrated in figure 6.2-50. In this model, insulation in the broken leg penetration is assumed to blow off and completely plug the penetration at the pipe restraint as well as the vessel support ventilation duct. All gaps in the broken leg blowdown restrictor plate remain unobstructed throughout the transient to allow maximum pressurization of the cavity. In all other areas, insulation is assumed to remain in place and uncrushed, and the insulation volume is deducted from compartment volumes and vent areas. A summary of all compartment volumes is provided in table 6.2-22.

Flow coefficients are calculated as for the SG compartments (paragraph A. above). Table 6.2-18 summarizes individual loss components for each flowpath as well as effective vent areas and calculated flow coefficients. Figure 6.2-51 is a block diagram of the reactor cavity model. It illustrates all compartment net free volumes, flow paths, and related vent areas and flow coefficients.

The Bechtel computer code COPDA was used for the analysis. (See attachment D, appendix 3K.) The program uses a Moody multiplier of 0.6. Moody flow formulations were used except when flow was subcritical, in which case homogeneous frozen flow equations were applied.

Results:

Maximum and design pressures and time to peak are listed for all compartments in table 6.2-22. A two-dimensional axisymmetric finite element analysis was done to check the acceptability of the reactor cavity under the nonuniform differential pressure loading, and the results verify that allowable loads are not exceeded. All other subcompartments are also below the design pressure limit. The time-dependent pressure histories for all nodes are shown graphically in figures 6.2-58 through 6.2-65. The net pressure vessel loading history is illustrated graphically in figure 6.2-52. The asymmetric pressure results in a maximum horizontal loading of 1.2×10^6 lb and a maximum vertical loading of 2.12×10^4 lb on the reactor pressure vessel.

Nodalization Sensitivity Study:

A nodalization sensitivity study was performed to determine the minimum number of nodes required to conservatively predict the maximum pressure in the reactor cavity. A base case and two additional models were analyzed. The mass and energy release for all three cases of the sensitivity study correspond to a 144 in.^{2} break in the loop 3 cold leg.

The nodalization in the base case, which is a 31-compartment model, was indentical to the model shown in figure 6.2-50 except that compartment 31 was combined with compartment 9, compartment 32 was combined with compartment 10, and compartment 33 was combined with compartment 8.

In model B, a 27-compartment model, nodes were combined directly above and below the nozzles between 25° and 145° to form single compartments. This decreased the number of nodes both circumferentially and axially. In model C, a 38-compartment model, this region was modeled with twice the number of nodes used in the base case model. In addition, compartment 3 (adjacent to the blowdown restrictor plate inside the cavity) was divided into four subcompartments. Blowdown from compartments 1 and 2 enters the annulus through these nodes.

For the base case, the maximum horizontal force was calculated to be 1.4 x 10^6 lbs and the maximum uplift force was calculated to be 5.9 x 10^4 lb. For both models B and C, the results show the maximum horizontal force to be 1.42 x 10^6 lb, or within 1.5% of the base case. Maximum uplift force for both cases was approximately 6.0 x 10^4 lb. Based on these results it is concluded that the compartment boundaries should properly be placed at flow restrictions and the addition of arbitrary compartments will not significantly affect the results.

C. Pressurizer

The pressurizer compartment has been analyzed for the double ended rupture of any line in this compartment. The pressurizer spray line break is the controlling one, the rupture of which results in a peak compartment pressure differential of 9.4 psid. The blowdown data for the spray line break are listed in table 6.2-16.

The blowdown model and flow model are shown in figures 6.2-55 and 6.2-56. The results are shown on figure 6.2-57.

The pressurizer surge line extends through an open penetration in the floor of the pressurizer compartment to the bottom of the pressurizer, and is enclosed by the pressurizer compartment volume.

The blowdown from a break in the pressurizer surge line, given by table 6.2-17, at the pressurizer nozzle will be vented directly into the lower compartment. This lower compartment is designed to withstand the differential pressures resulting from the DBA.

Postulated breaks in the pressurizer surge line have been eliminated from the structural design basis for both Unit 1 and 2 through the application of LBB technology.

D. Secondary Shield Annulus

The steam lines penetrate the containment vessel at el 138 ft, which is below the top of the secondary shield walls. An analysis was made to determine if local high pressures could occur in the relatively confined space. Steam flows around the annulus, upward and directly to the containment upper volume.

At no time do differential pressures across the walls exceed 0.25 psid.

6.2.1.3.11 Main Steam Line Ruptures Inside Containment

A. Introduction

Steam line ruptures occurring inside the containment structure may result in significant pressure and temperature transients. In order to determine the rupture which results in the worst case, a complete spectrum of ruptures was analyzed for power uprate. This extensive analysis was necessary due to the number of variables on the determination of the blowdown data. The five most limiting power uprate containment pressure and temperature response cases were reanalyzed for RSG and the measurement uncertainty recapture (MUR) power uprate (up to 2841 MWt, which includes 10 MWt RCP heat). These five cases span 0 to 100% nominal full-load MUR uprated power and three break sizes as noted below. A summary of the plant particular data utilized for this analysis is given in table 6.2-11.

After the blowdown was determined, a case-by-case containment analysis was performed using the GOTHIC computer code. As described in paragraph 6.2.1.3.12, GOTHIC is a containment pressure/temperature transient analysis code. A more detailed description is contained in reference 22.

The LOCA model heat sinks were used in the containment analyses of steam line ruptures. The pertinent information and data for these are given in tables 6.2-2 and 6.2-4.

The following list of ruptures was analyzed:

*Case 1: Full double-ended rupture at 100% MUR uprated power. (Reanalyzed for RSG at -1.5- and +3.0-psig initial pressure.)

Case 2: 0.7 ft^2 double-ended rupture at 102% power.

Case 3:	0.6 ft^2 double-ended rupture at 102% power, at both 0 and -1.5-psig initial pressure.
Case 4:	0.528 ft^2 split rupture at 102% power.
Case 5:	Full double-ended rupture at 70% power.
Case 6:	0.6 ft^2 double-ended rupture at 70% power.
Case 7:	0.5 ft^2 double-ended rupture at 70% power.
*Case 8:	0.22 ft^2 split rupture at 70% MUR uprated power. (Reanalyzed for RSG with a 0.22 ft^2 at -1.5- and +3.0-psig initial pressure.)
*Case 9:	Full double-ended rupture at 30% MUR uprated power. (Reanalyzed for RSG at -1.5- and +3.0-psig initial pressure.)
Case 10:	0.5 ft^2 double-ended rupture at 30% power.
Case 11:	0.4 ft^2 double-ended rupture at 30% power.
*Case 12:	0.33 ft^2 split rupture at 30% MUR uprated power. (Reanalyzed for RSG with a 0.33 ft^2 rupture at -1.5- and +3.0-psig initial pressure.)
*Case 13:	Full double-ended rupture at hot standby. (Reanalyzed for RSG at -1.5- and +3.0-psig initial pressure.)
Case 14:	0.2 ft^2 double-ended rupture at hot standby.
Case 15:	0.1 ft^2 double-ended rupture at hot standby.
Case 16:	0.3 ft^2 split rupture at hot standby.

All steam line rupture cases indicated with an asterisk (*) have been analyzed in support of the SG replacement and the MUR power uprate at both Farley Nuclear Plant units. These represent the limiting cases with respect to the containment pressure and temperature responses. The other 11 cases were analyzed in support of the power uprate for Farley Nuclear Plant. A discussion of the steam line rupture mass and energy releases is presented in FSAR paragraph 6.2.1.3.11, item B.

B. Mass and Energy Releases Following a Main Steam Line Rupture Inside Containment

The mass and energy release data for each case are determined using the methodology documented in reference 23. The LOFTRAN code (reference 24) was used in the analysis documented for the Farley Nuclear Plant power uprate. In support of the SG replacement and the MUR uprated power, the analysis methodology for the limiting 5 cases (see FSAR paragraph 6.2.1.3.11, item A) was supplemented using the model documented in reference 25.

C. Evaluation of Effects of Various Single Failures

The method of determining the blowdown assumes no failure in steam or feedwater isolation. This blowdown is used in conjunction with minimum containment spray and fan coolers to allow for failure of a diesel generator. Steam line isolation failure is not postulated, since there are two redundant swing disc trip valves in each steam line. However, since these valves stop flow only in the forward direction, the mass/energy release to containment as calculated by the LOFTRAN code (reference 24) was modified to include the entire steam piping volume downstream of the isolation valves for the other SGs, including the steam line header and steam dump piping. For the steam line rupture cases analyzed in support of the SG replacement, those indicated with an asterisk (*) in FSAR paragraph 6.1.2.3.11, item A, the mass/energy released to containment as calculated by the RETRAN model (reference 25) includes the contents from the main steam header piping downstream of the postulated break location.

For a complete description of the functions that provide the necessary protection against a steam pipe rupture refer to FSAR paragraph 15.4.2.1.1, items A through D.

D. Pressure Temperature Results

The pressure/temperature results of the analysis are illustrated in figures 6.2-6 through 6.2-37. The highest pressure obtained was 53.4 psig for a full double-ended rupture at 0-percent power. The highest temperature reached was 367°F for a double-ended rupture at 102-percent power.

Equipment Temperature Transient

A transient main steam line break (MSLB) containment thermal analysis was performed for the worst case of the spectrum of breaks for each type of Class 1E component inside containment to determine the peak component surface temperatures. The methodology used to calculate the equipment surface temperatures was based on the NRC staff's approved assumptions discussed in Appendix B of NUREG-0588. These analyses show that the peak surface temperature resulting from the MSLB environment do not exceed the qualification temperature for each type of component. The methodology and results of the calculations of the equipment surface temperatures are reported in the docketed Farley response to NUREG-0588 and I&E Bulletin 79-01B.

The above results show that the maximum pressure in the containment is below the containment design pressure at 54 psig. In addition, the components covered by I&E Bulletin 79-01B and NUREG-0588 and required for safe shutdown and accident mitigation maintain their environmental qualification for the resulting temperature and pressure profiles inside the containment as determined by the above analysis.

E. The containment design meets the NRC acceptance criteria contained in IE Bulletin 80-04 related to the issue of containment overpressurization resulting from a main steam line rupture with continued feedwater addition. Considering all possible sources of water, there is no potential for containment overpressurization because the main feedwater system is isolated and auxiliary feedwater system flow restrictors limit flow to the affected SG. Also, the auxiliary feedwater system pumps are protected from the effects of runout flow and, therefore, can be expected to carry out their intended function during a main steam line rupture event.

6.2.1.3.12 Reduced Service Water Flow Containment Response

Evaluations of the service water system have indicated that under certain conditions for component failures and loss of offsite power (LOSP), the service water flow to the containment air coolers may be reduced below the original design basis flow. The effect of this reduction in service water supply to the containment air coolers was included in the LOCA analysis previously described.

6.2.1.3.13 Containment Pressure/Temperature (P/T) Evaluations Subsequent to Power Uprate

As a result of calculating new RWST level uncertainties, evaluating net positive suction head for the ECCS/CSS pumps, and evaluating changes to the ECCS/CSS switchover sequence (from injection to sump recirculation), several parameter changes have been evaluated to determine the impact on the P/T analysis. The parameter changes which were evaluated include revised RWST deliverable volumes, RHR interruption times, time of ECCS/CSS switchover to sump recirculation, and increased CS flowrate.

The changes to RWST delivered volumes, RHR interruption times, and time of ECCS/CSS switchover affect only the long-term portion of the LOCA cases (MSLB is not affected). Additionally, increased spray flowrates do not adversely affect the containment P/T analysis since minimum spray flow is conservative. Since the LOCA peak pressure occurs at approximately 20 s and the parameter changes only affect the long-term portion of the transient, the LOCA peak pressure, as given in the power uprate containment P/T analysis (table 6.2-6), remains valid. However, for the long-term portion of the LOCA cases, increases in the pressure and temperature response have been evaluated. The evaluation demonstrates the acceptability of the results. This evaluation affects figures 6.2-1, and 6.2-40.

6.2.1.3.14 Containment Pressure/Temperature Evaluations Subsequent to Steam Generator Replacement

An evaluation was performed for the revised heat sinks in table 6.2-2. More recent information from the vendor also resulted in minor changes to coating thermodynamic properties as reflected in table 6.2-4. Additional changes involve the actuation delays associated with the containment spray and fan cooler ESFs. Table 7.3-16 has been appropriately revised to reflect the reassessed spray response time. The fan cooler delay of 92 s assumed in the containment pressure/temperature analysis is conservative with respect to the 87-s service water delay in table 7.3-16. The DEHL and DEPS LOCA cases, the limiting main steam line break cases (Cases 1 and 13), and a nonlimiting MSLB case (Case 9) were reanalyzed with these revised inputs. In order to offset adverse impacts from these changes, minimum spray flow was recalculated and increased from 2175 gal/min during injection and recirculation modes to 2480 gal/min and to 2290 gal/min in injection and recirculation modes, respectively. LOCA peak temperature and MSLB and LOCA peak pressures remained bounded by prior results. The peak MSLB temperature increased 0.2°F which is indiscernible in the transient figures. MSLB pressures were slightly higher early in the transients (~0.3 psi) due to revised spray and fan cooler response times but remained bounded by the LOCA cases at this point in the transient. The revised profiles were evaluated for impact on the Environmental Qualification program and it was determined that the existing P/T profiles remain valid. Figures for the reanalyzed breaks were revised. Based on the observed sensitivities, the remaining figures remain bounding or are negligibly affected.

6.2.1.4 <u>Containment Testing and Inspection</u>

6.2.1.4.1 Preoperation Testing

6.2.1.4.1.1 <u>Integrated Test</u>. Upon completion of the containment and installation of all penetrations, an integrated leakage rate test was performed to verify that the potential leakage rate from the containment is maintained within acceptable values.

The integrated leakage rate tests consist initially of a preoperational test at the peak calculated accident pressure of 48.0 psig, as well as at least one at a lower pressure.

The total allowable leak rate is not more than 0.15 percent by weight of the contained atmosphere per day at 48.0 psig. It has been demonstrated that with good quality control during construction, this is a reasonable requirement. The basis for the performance of the integrated leakage rate test is "Containment Structures for Nuclear Power Plants," Revision 1, November 1, 1972.

The initial leak rate test of the containment and its penetrations were conducted at 100 percent of peak calculated accident pressure and 50 percent of peak calculated accident pressure. Values of containment ambient dry bulb temperature and relative humidity were recorded during the test period for correction of data as required. The test establishes the capability of the containment to contain the pressure for which it was designed, at a leak rate not exceeding that specified.

The test measurement system utilized for the initial leak rate test of the containment is a packaged portable unit designed for use on Units 1 and 2. This portable unit is also used for the periodic leak rate test. The instruments are calibrated prior to each periodic leak rate test. It is anticipated that these instruments will be available for the lifetime of the plant; however, spare instruments of each type are provided.
For further details, see the Technical Specifications.

6.2.1.4.1.2 **Local Tests.** Prior to initial startup, penetrations and isolation valves were leak tested to verify that the potential leakage is within acceptable values.

The local tests will be performed at a pressure of 48.0 psig and in accordance with the Technical Specifications.

6.2.1.4.2 Postoperational Leakage Tests

Periodic leak rate tests of the containment, penetrations, and isolation valves are conducted to verify their continued leaktight integrity.

The first postoperational integrated leakage rate tests were conducted at 50 percent of peak calculated accident pressure while the penetrations and isolation valves postoperational tests were conducted at 48.0 psig.

Postoperational leakage tests are currently conducted in accordance with the Containment Leakage Rate Testing Program. The test frequencies and acceptance criteria are specified in this program.

The Containment Leakage Rate Testing Program complies with the NRC acceptance criteria contained in GDC 52, 53, and 54 and implements the requirements of 10 CFR 50 Appendix J, Option B; the guidance in Nuclear Energy Institute (NEI) 94-01, Revisions 2A and 3A with their associated limitations and conditions; and the technical methods and techniques for performing containment leakage tests in ANSI/ANS 56.8-2002.

The ILRT performance criterion (L_a) is 0.15 weight % per day. The NRC acceptance criteria regarding airlock leakage tests are also in the Containment Leakage Rate Testing Program.

6.2.1.4.3 Containment Materials Inspection, Testing, and Surveillance

A. Tests to Ensure Liner Integrity

The following tests were/are performed:

1. Construction tests during the erection of the containment liner.

2. Preoperational tests after the erection of the containment complete with liner, electrical and piping penetrations, equipment hatch, and personnel lock, but before reactor operation.

3. Postoperational leakage tests will be performed at periodic intervals for the life of the plant.

4. Inservice inspection of the metallic liner and the pressure retaining concrete structure of the containment will be performed at periodic intervals for the life of the plant as discussed in paragraph 3.8.1.7, Testing and Inservice Surveillance Requirements.

[HISTORICAL] [<u>*Tests on Liner During Construction*</u>

Inspection procedures employed during construction for the liner seam welds, liner fastening, and around penetrations consist of visual inspection, vacuum box soap bubble testing, radiography, dye-penetrant testing, and magnetic particle inspection.

A. <u>*Visual Inspection of Welds*</u>

All of the welding is visually examined by a technician responsible for welding quality control. The basis for visual quality of welds is as follows:

1. Each weld is uniform in width and size throughout its full length. Each layer of welding shall be smooth and free of slag, cracks, pinholes, and undercut and shall be completely fused to the adjacent weld beads and base metal. In addition, the cover pass is free of coarse ripples, irregular surface, nonuniform head pattern, high crown, and deep ridges or valleys between beads. Peening of welds is not permitted, except for light peening for cleaning purposes.

2. Butt welds are of multipass construction, slightly convex, of uniform height, and have full penetration.

3. Fillet welds are of the specified size, with full throat and legs of uniform length.

B. <u>*Soap Bubble Tests*</u>

All of the welding required for containment integrity is vacuum box soap bubble tested except where the structural configuration or space limitation does not allow. In this test a vacuum box containing a window is placed over the area to be tested and is evacuated *to produce at least 5 psi pressure differential. Before the vacuum box is placed over the test area, a soap solution is applied to the weld and any leaks will be indicated by bubbles observed through the window in the box.*

C. *Radiography*

Radiography is used as an aid to quality control. The primary purpose of the liner plate and the welds therein is to provide leaktightness integrity to the posttensioned concrete containment. Structural integrity of the containment will be provided by the posttensioned concrete and not by the liner plate.

Radiography is not recognized as a completely effective method for examining welds to assure leaktightness. Therefore, the maximum benefit expected from radiography in connection with obtaining leaktight welds will be as an aid to quality control. Random radiography of each welder's work will provide verification that the welding is under control and being done in accordance with the previously established and qualified procedures. Additionally, employing random radiography to inspect each welder's work has been proved by past experience to have a positive psychological effect on the improving overall welding workmanship.

For quality control purposes, at least one spot radiograph 12 inches long was taken in the first ten feet of welding completed in the flat, vertical, horizontal, and overhead positions by each welder on liner plate welds. No further welding was permitted until initial radiographic inspection has been satisfactorily completed and the welding found to be acceptable by the Inspector.

Thereafter, a minimum of 2 percent of the welding was progressively spot examined as welding is performed, using film 12 in. long, on a random basis to be specified by the inspector, in such a manner that an approximately equal number of spot radiographs was taken from the work of each welder. In addition to the 2 percent radiograph, 18 percent of the welding was nondestructively examined. Under conditions where two or more welders make weld layers in a joint or on the two sides of a double-welded butt joint, one spot examination represented the work for both welders. Where a radiograph discloses welding which did not comply with the minimum quality requirements, as defined in paragraph UW52, Section VIII of ASME code, two additional spots, each 12 inches long, were examined in the same weld seam at locations away from the original spot. The locations of these additional spots were determined by the Inspector as provided for the original spot examination. If two additional spots examined showed welding which met the minimum quality requirements, the entire weld represented by the three radiographs was acceptable. The defective welding disclosed by the first of the three radiographs was removed and repaired by welding.

If either of the two additional spots examined showed welding which did not comply with minimum quality requirements, the entire portion of the seam represented was rejected; or, at the fabricator's option, the entire weld represented was completely radiographed, and defective welding corrected.

The rewelded joints or weld-repaired areas were completely reradiographed and met the weld quality requirements cited above.

D. *Dye-Penetrant and Magnetic-Particle Inspection*

Dye-penetrant and magnetic-particle inspection were used as an aid to quality control. The field welding inspectors used dye-penetrant or magnetic-particle inspection to closely examine welds judged to be of questionable quality on the basis of the initial visual inspection. Dye-penetrant or magnetic-particle inspection of liner plate welds were in accordance with Section VIII of the ASME Boiler and Pressure Vessel Code.]

6.2.1.5 Instrumentation Requirements

The containment atmosphere is continuously monitored by four pressure transmitters outside the containment, which are connected to the containment atmosphere. The ESF actuation design details and logics associated with these pressure transmitters are discussed in section 7.3. Instrumentation also monitors the containment to identify any pressure boundary leakage in the RCS. The leakage detection system is discussed in subsection 5.2.7.

Airborne radiation inside the containment is monitored by the containment monitoring system, which is discussed in section 11.4.

Liquid level indication is provided in the containment sump, as discussed in section 7.3.

6.2.1.6 Materials

Materials used in or on ESF systems were chosen so that any decomposition products of the materials will not interfere with safe operation of ESFs.

6.2.2 CONTAINMENT HEAT REMOVAL SYSTEMS

Three systems are provided to reduce containment atmosphere temperature and pressure and/or to remove heat from the containment under post-accident conditions. These are the low-head SI/RHRS, the containment spray system, and the containment cooling system. The design, operation, and reliability of the low-head SI/RHRS are discussed in section 6.3; the remaining two systems are discussed below.

6.2.2.1 Design Bases

Design environmental conditions for the containment heat removal systems are given in table 3.11-1. Design parameters for system components are discussed in paragraph 6.2.2.2. The sources and amounts of energy that were considered in sizing the heat removal systems are listed in table 6.2-20. The containment heat removal systems are in conformance with the

NRC acceptance criteria contained in General Design Criteria 38, 39, and 40 and Regulatory Guide 1.1.

6.2.2.1.1 Containment Spray System

The function of the containment spray system is to spray water into the containment atmosphere, when appropriate, in the event of a LOCA to ensure that containment peak pressure is below its design value. This protection is afforded for pipe breaks up to and including the hypothetical instantaneous double-ended circumferential rupture of an RCL as analyzed in subsection 6.2.1. Although the water in the reactor vessel after a LOCA is quickly subcooled by the ECCS, the containment spray system design was based on the conservative assumption that the core residual heat would be released to the containment as steam.

The two redundant trains of the containment spray system have been designed to provide sufficient heat removal capacity to prevent exceeding containment design pressure for all piping breaks. Assuming that the water in the RWST is at a temperature of 110°F, the containment atmospheric heat removal capability associated with the spray from one containment spray train will initially be 2×10^8 Btu/h at initial spray actuation during the injection phase.

The containment spray system is designed to operate over an extended period of time and under environmental conditions existing following a RCS failure.

6.2.2.1.2 Containment Cooling System

The containment cooling system has been designed to remove heat which will be released to the containment atmosphere during any MSLB or LOCA up to and including the double-ended rupture of the largest system pipe. This is accomplished by one of four containment air coolers.

The experimental heat transfer data for the containment air coolers are based on the following documents which have been submitted to the NRC:

A. "Cooling Coil Thermal and Structural Capacity Evaluation for the Palisades Plant of Consumers Power Company," American Air Filter Company, Inc., Report R-1003 (Palisades Docket Number 50-255).

B. Topical Report on "Design and Testing of Fan Cooler/Filter Systems For Nuclear Applications," AAF-TR-7101 February 20, 1972.

Test requirements are certified by the cooler manufacturer.

6.2.2.2 System Design

Design parameters for the heat removal system components are presented in table 6.2-24. Individual system designs are discussed below.

6.2.2.2.1 Containment Spray System

The containment spray system P&ID shown in D-175038, sheet 3 and D-205038, sheet 3 consists of two pumps, spray ring headers and nozzles, valves, and piping. During the initial (injection) phase of operation, water from the RWST is used for containment spraying. During the later (recirculation) phase of operation, water for containment spraying is recirculated from the containment sump.

Detection of leakage from the containment spray system involves the use of sump level instrumentation and floor drain tank level instrumentation. The spray pumps are in rooms that contain a sump. If the alarm in the control room indicates a high sump level, the sump contents would be pumped by duplex sump pumps to the waste holdup tank. Any leak in one of the pump rooms would be detected and the leak isolated.

A leak in the spray system piping in a corridor, pipe chase, or penetration room is indicated by an increase in the floor drain tank level.

Whenever one of the spray headers is isolated during the process of system leak detection and isolation, flow indication in the operating header assures the operator that sufficient spray flow is being returned to the containment.

The two redundant and independent containment spray trains that comprise the containment spray system, including required auxiliary systems, are designed so that a single active failure during the injection phase or a single active or passive failure during the recirculation phase following a RCS failure does not result in loss of the protective function.

The containment spray system is designed to accommodate the 1/2 safe shutdown earthquake (1/2 SSE) within applicable codes stress limits and to withstand the SSE without rupture or loss of function.

Between the edge of the containment mat and the auxiliary building, each containment spray suction line is surrounded by a concentric guard pipe. Three annular seal rings are installed around the guard pipe at each end to minimize any postulated bypass leakage along the outside of the guard pipe. The concentric guard pipes enter the auxiliary building in a trench within the pipe penetration room. Therefore, any postulated airborne leakage that escapes outside the concentric guard pipes is processed by the penetration room filtration system, and any postulated water leakage is processed by the waste disposal system. Within the auxiliary building, each suction line is located within a protective chamber. This chamber is designed to withstand the containment design pressure in addition to the head of water present in the containment sump at the end of the injection phase. From the protective chamber to the first motor-operated isolation valve inside the auxiliary building, each suction line is surrounded by a concentric guard pipe. The first motor-operated isolation valve is surrounded by a watertight closure. This arrangement, which is the same as that used with the suction lines in the low-head SI system, ensures that, in the unlikely event of leakage from the suction pipe during long term recirculation, the integrity of the recirculation system is not impaired and public safety is not hazarded.

Adequate net positive suction head (NPSH) is available to the containment spray pump suctions at all times during both the injection and the recirculation phases of operation, assuming the most adverse combination of flowrate, sump water temperature, and containment pressure. The margin between available and required NPSH during the recirculation phase is based upon conservative calculations that account for the head loss due to debris accumulation on the sump strainers. These calculations neglect any containment pressure in excess of the TS minimum operating containment pressure prior to the LOCA accident when the containment sump temperature is lower than the saturation temperature corresponding to the TS minimum operating containment pressure. For sump temperatures above the saturation temperature at the TS minimum operating containment pressure, containment pressure in excess of the saturation pressure corresponding to the sump water temperature is conservatively neglected. The available NPSH for the containment spray pumps during the recirculation phase is provided in subsection 6.3.2.14.

Descriptions of the containment spray system components are presented below.

Refueling Water Storage Tank

The RWST serves as a source of emergency borated cooling water for injection. It is normally used to fill the refueling canal for refueling operations. However, during all other plant operating periods, it is aligned to the suction of the residual heat removal pumps and the containment spray pumps. The capacity of the tank is 66,850 ft^3. The tank is fabricated from stainless steel and is designed and constructed in accordance with Code ASME III, Class 2. Water in the tank is borated to a concentration which assures reactor shutdown by at least 10% $\Delta k/k$, when all RCC assemblies are inserted and the core cooled down for refueling.

Containment Spray Pumps

The containment spray pumps are horizontal centrifugal type driven by electric motors. Each motor is powered from an individual emergency bus.

The pumps are designed to perform at rated capacity against a total head composed of containment design pressure, nozzle elevation head, and the line and nozzle pressure losses. Pump design parameters are presented in table 6.2-24.

Pump room coolers are used to maintain air temperatures in the pump rooms at or below 104°F during normal operation. Refer to table 9.4-6A for post-DBA room temperatures. The cooler fans are interlocked to start with the respective pump motors. The Seismic Category I portion of the service water system is used as the cooling medium for the pump room coolers. Containment spray pump room coolers are discussed in paragraph 9.4.2.1.9.

A gas accumulation monitoring and trending process for the Farley Unit 1 and 2 ECCS (CVCS and RHR) and containment spray systems has been established to meet the requirements of NRC Generic Letter 2008-01.

Spray Nozzles

The spray nozzles, which are of the hollow cone design, are not subject to clogging by particles < 1/4 in. in size, and produce a drop-size spectrum with a mean diameter of < 700 microns when operating at the design pressure differential of 40 psi. The spray nozzles are stainless steel and have a 3/8-in. diameter orifice, which is larger than the 3/32-in. openings in the Unit 1 and 2 post-LOCA containment sump strainers.

[HISTORICAL][*For Unit 1 there are three 3/8-in. diameter holes between the solid cover plate on the top of the sump screen and the bioshield wall for venting of air during the initial phase of the LOCA when the water level in the sump rises. The slot size varies from approximately 1/4 in. to 1 in. across its length of approximately 3 ft. The potential for debris to enter through this path has been evaluated. The location of the slot near the shield wall was specifically selected to minimize the potential for debris to enter the sump. Since this slot and the vent holes will be under water during the recirculation phase of a LOCA, the debris entering through this path will sink to the sump floor due to low approach velocities near the bioshield wall and will not be swept into the opening of the intake pipe.]* Therefore, the hydraulic performance of the CS system will not be adversely affected.

There is a vent slot and, for Unit 1, three 3/8-in.-diameter holes between the solid cover plate on the top of the sump screen and the bioshield wall for venting of air during the initial phase of the LOCA when the water level in the sump rises. The slot size varies from approximately 1/4 in. to 1 in. across its length of approximately 3 ft. The potential for debris to enter through this path has been evaluated. The location of the slot near the shield wall was specifically selected to minimize the potential for debris to enter the sump. Since this slot and the vent holes will be under water during the recirculation phase of a LOCA, the debris entering through this path will sink to the sump floor due to low approach velocities near the bioshield wall and will not be swept into the opening of the intake pipe. Therefore, the hydraulic performance of the CS system will not be adversely affected.

Containment Recirculation Sump

See appendix 6D.

Motors, Valves, and Piping

Motors, valves, and piping for the containment spray system are designed in accordance with the specifications for these components in the ECCS, as discussed in section 6.3.

Electrical Supply

Details of the emergency bus power source are discussed in subsection 8.3.1.

6.2.2.2.2 Containment Cooling System

The containment cooling system consists of four containment air coolers, each with a one-third cooling capacity during normal operation, up to four units will be operating. Each air cooler consists of a fan and finned tube coil supplied by water from the service water system. During outage periods, auxiliary cooling may be supplied to one of the four containment air coolers in Unit 2 to enhance containment heat removal.

As the post-accident containment atmosphere, which consists of a steam-air mixture, is circulated through the bank of cooling coils, it is cooled and a portion of the steam is condensed. The capacity of one cooler in conjunction with one containment spray train is adequate to maintain pressure and temperature below peak calculated LOCA conditions.

Cooling surfaces are constructed in accordance with TEMA guidelines. Headers for coolers are designed and fabricated to the requirements of the ANSI B31.7. The service water system piping inside the containment is designed to ASME Section III.

Fan motors conform to standards of NEMA, IEEE, and ANSI. A supply prototype fan motor has demonstrated capability to supply design flow of steam air mixture through the cooling coils. Dropout plates with release mechanisms actuated by fusible links are provided at the discharge of the containment coolers. These plates fall away to uncouple the cooler discharge from the distribution ductwork after a LOCA, thereby reducing discharge backpressure and allowing less restricted flow through the coolers. The fusible links used with the dropout plate release mechanisms are constructed from suitable materials and are of a proven design shown to be reliable by extensive past commercial usage in critical service for fire protection. They are designed, manufactured, and tested in accordance with the applicable provisions of the following codes and standards:

- Fire Protection Handbook.
- Underwriters Laboratories' Standard for Fusible Links for Fire Protection Service, UL33-1968.
- Underwriters Laboratories' Standard of Safety, UL555-1970.
- Factory Mutual Approval Guide, 1971.

6.2.2.3 <u>Design Valuation</u>

Descriptions of the analytical methods and models used to assess the performance capability of the containment heat removal systems are provided in subsection 6.2.1. Curves describing the calculated response of pertinent system and containment parameters following a LOCA are presented in subsection 6.2.1. Additional design evaluations are presented below.

6.2.2.3.1 Containment Spray System

The containment spray system is designed to limit the effects of postblowdown energy additions to the containment during the injection phase in the event of a LOCA assuming no credit for core cooling by the ECCS. The accident analysis presented in subsection 6.2.1 assumes that containment cooling capability is reduced to one of two containment spray pumps and one out of four fan coolers. This is the minimum equipment available considering the single-failure criterion in the emergency power, the containment spray, and the containment cooling system.

The minimum fall path of water droplets is approximately 110 ft, which is the distance from the lowest spray ring to the operating deck. Detailed calculation of the heatup of spray drops in post-accident containment atmospheres by Parsly[2] show that drops of all sizes encountered in the containment spray reach equilibrium in a fraction of their residence time.

A failure analysis has been made on all active components of the system during the injection phase, and on all active and passive components during the recirculation phase, to show that the failure of any single component will not prevent fulfilling the design function. This analysis is summarized in table 6.2-26.

Concerning the containment sumps, there are several different flow paths which a particle might follow in circulating through the different systems that must use the sumps as a source. The maximum particle size that could be passed through the post-LOCA sump strainer screen is < 3/32 in. The different flow paths are:

A. A particle of sufficiently small size which has passed through one of the fine mesh screens may flow into one of the containment spray pump intakes, through the spray pump, and be discharged through one of the spray nozzles into the containment. The particle may or may not pass through one of the containment spray eductors; however, the eductor size is sufficiently large for the largest particle to pass through. See paragraph 6.2.2.2.1 for a further description of the spray system components.

B. A particle of sufficiently small size which has passed through one of the fine mesh sump screens may flow into one of the low-head SIS pump intakes, through the low-head SIS pump, and through one of the residual heat exchangers; from there it may follow one of the following paths:

    1. It may flow directly into the RCL (hot leg or cold leg).

    2. It may flow into the suction of the high-head SIS pumps and through one of these pumps. From there, it may flow into the seal injection filter, which supplies seal water to the reactor coolant pumps and is considered an alternate SI path post LOCA; or it may flow directly into the RCL (hot leg or cold leg).

The effects of debris ingested through the containment sump strainer during operation of the containment spray system in recirculation mode were evaluated for Generic Letter 2004-02, “Potential Impact of Debris on Emergency Recirculation during Design Basis Accidents at Pressurized Water Reactors”. The Generic Letter Response (including RAI responses and Supplemental Content Guide) includes evaluations of potential erosive wear, abrasion, and blockage of flow paths within containment spray system components. The Generic Letter 2004-02 evaluations conclude that system design is adequate to maintain containment spray function during recirculation mode operation.

Details of the design evaluations for the effects of debris on the function of the containment spray system are provided in Appendix 6D.

6.2.2.3.2 Containment Cooling System

The air coolers, with their associated piping, valves, and instrumentation are located outside the secondary shield walls and above the maximum possible post-accident water level to provide protection against missiles and flooding.

Each containment cooler is equipped with a dropout plate, held shut during normal operation by fusible links, which disconnects the cooler discharge from the containment ductwork after a LOCA. During operation, the load on the fusible links is supplied by springs in order to minimize the effect upon the load of thermal growth of the release mechanism and any elongation of the fusible link just prior to separation. Positive means are provided to assure that each dropout plate is released after separation of the fusible links. The fusible links are specified to release at a temperature of 135°F. Following a LOCA, the actuation of the dropout plates and the reduction to low speed decreases the fan motor capacity to approximately 40,000 sft^3/min at 7.9-in. WG static pressure with a power requirement of approximately 105 hp.

Following a LOCA, the SIS water in the core rapidly falls below saturation temperature. Nevertheless, the containment cooling system design is based on the assumption that core residual heat appears as steam in the containment.

The criteria for the selection of the design service lifetime for the air coolers in the accident environment is based primarily on the service lifetime for the fan motor. The motor service lifetime is based on the criteria in IEEE Report NSG/TCS/SC2-A, Proposed Guide for Qualification Tests for Class IE Motors Installed Within the Containment of Nuclear Fueled Generating Stations.

Coil design and arrangement provide for rapid drainage of large quantities of condensed steam, thereby preventing loss of cooling capacity and maintaining cooling water temperatures below the boiling point.

Service water flow to the containment air cooling coils is unmodulated, reducing the probability of a failure due to a modulating valve or controller malfunction. Each of the four cooling coils has a separate branch supply and return run through the containment wall with appropriate isolation valves as discussed in subsection 6.2.4.

A radiation monitor is provided in the service water discharge from the containment air coolers to detect and prevent the release of radioactive service water to the environment.

High reliability is maintained through careful quality control and assurance procedures and by general arrangement of equipment to provide access for inspection and maintenance. Components are designed to operate and withstand the post-accident environment.
Failure of a cooling unit is more than compensated for by the operation of either spray pump. If a containment air cooling unit coil should leak excessively, adequate alternate systems for containment heat removal are provided to meet the design requirements.

A coil rupture is detected by flow monitoring equipment and, as shown in the service water P&ID, drawings D-175003, sheet 1; D-175003, sheet 2; D-175003, sheet 3; D-175003, sheet 4;

D-205003, sheet 1; D-205003, sheet 2; D-205003, sheet 3; and D-205003, sheet 4, isolated. Leakage into the containment would have a negligible effect upon water boron concentration. A relief valve is provided on each coil to prevent excessive tube pressure due to heat buildup in the isolated portion after isolation. Because service water is being constantly circulated through the cooling coils, tube clogging during an accident is highly unlikely.

The heat sink for the containment air cooling units is the service water system. Adequate redundancy is provided in this system to assure continued availability in the event of a single failure. Failure of an inlet or outlet valve to a containment air cooling coil is detected by flow reduction. The containment sprays provide adequate backup cooling capacity.

6.2.2.4 <u>Testing and Inspection</u>

6.2.2.4.1 Containment Spray System

Components of the spray system are inspected periodically to demonstrate system readiness, as discussed in response to GDC 39 in section 3.1. The pressure containing systems are inspected for leaks from pump seals, valve packings, flanged joints, and safety valves during system testing. The components of the system outside the containment are accessible for leaktightness inspection during periodic flow tests.

Proper functioning of the containment spray system will be demonstrated by periodic tests, as discussed in the response to GDC 40 in section 3.1.

Preoperational testing is performed to:

A. Demonstrate that the system is adequate to meet the design pressure conditions. Outside the containment, the piping welds are subjected to radiographic inspection and/or part hydrostatic testing. Inside the containment, the spray header welds are subjected to 100% radiographic inspection.

B. Demonstrate that the spray nozzles in the containment spray header are clear of obstructions by passing air through the test connections.

C. Verify that the proper sequencing of valves and pumps occurs on initiation of the containment spray signal, and demonstrate the proper operation of remotely operated valves.

D. Verify the operation of the spray pumps. Each pump is run at shutoff and the miniflow directed through the normal path back to the RWST. During this time, the pump flow and discharge pressure should be recorded.

Periodic postoperational testing is performed to:

A. Verify that each automatic containment spray valve in the flow path that is not locked, sealed, or otherwise secured in position actuates to the correct position and each containment spray pump starts automatically on an actual or simulated actuation signal.

B. Verify each containment spray pump's developed head at the flow test point is greater than or equal to the required developed head.

With the procedures used to clean the containment spray system prior to completion of installation of the spray nozzles - system flush to remove particulate matter - and with the use of corrosion resistant materials of construction, it is not considered credible that a significant clogging of any of the nozzles would occur that would reduce the effectiveness of the system. For this reason, the airflow through the nozzles has been tested during the preoperational testing program and will be verified every 20 years to indicate that plugging of the nozzles has not occurred.

Airflow is delivered to the nozzles, through the headers, via test lines. Either balloons or streamers are deployed at the nozzle openings or hot air with an infrared camera is used to verify that sufficient airflow is established to the nozzles.

The sequence for the transfer of ESF loads to alternate power sources is discussed in subsection 8.3.1.

6.2.2.4.2 Containment Cooling System

The containment cooling system is periodically tested and inspected as discussed in the response to GDC 39 and 40 in section 3.1.

Fans are tested and rated in accordance with the standards of the Air Moving Conditioning Association (AMCA).

Containment air cooling unit coil performance has been verified by factory test of a small coil section tested under the normal and LOCA conditions. Containment air cooling unit fan motors are certified by the manufacturer to operate during and/or subsequent to a LOCA. The certification is based on a qualification test on a prototype full size Class IE motor higher in horsepower and equal in design and construction to that of the actual purchased fan motor. Test results taken from the prototype motor are interpolated into the size and capacity of the actual purchased motor. The test sequence simulating the design service conditions is aging, seismic forces, and containment environment. The motor manufacturer has conformed with detailed test procedures based on environmental conditions shown in table 3.11-1, Category A.

The fusible links are performance rated by Underwriters Laboratories. Two hundred fifty tests of actuation and tests to verify speed of operation demonstrate the acceptability of the link design in accordance with the provisions of Underwriters Laboratories' Standard of Safety UL 555-1970.

The load in the fusible links after they have been installed in the containment air cooling unit housing is supplied by adjustable compression springs designed to minimize the effect of thermal growth in the links which could possibly elongate them, thereby affecting their ability to permit fast opening of the plates at the moment of separation.

Provisions are made to allow inplace testing of fusible links to demonstrate that they perform their function following a LOCA.

The testing of the containment cooling system as well as its components demonstrates the initial capability of equipment and systems. Written test procedures establish minimum acceptance values for all tests. A recording of test results is useful in enabling detection of subsequent faulty performance.

Periodic tests of the activation circuitry and the system components can be conducted during normal plant operation assuring, in this way, a reliable performance upon demand throughout the plant lifetime.

Component qualification tests have been performed that demonstrate the characteristics of materials to be incorporated into components for the system by the manufacturer and ensure that the requirements of the procurement specification have been met.

Component acceptance tests have been performed at the factory that demonstrate the capability of the components incorporated in the various subsystems in which they are to operate. For example, fans are tested in the manufacturer's shop to determine their characteristic curves. System valves are tested in the shop to verify effectiveness of seal, opening and closing periods, and the ability of the valve operator to actuate the valve at the maximum anticipated differential pressure.

Systems acceptance tests consist of deenergized and energized tests which demonstrate the proper mounting of components, proper hookup of circuits and connections, setting of instrumentation, and operation of interlocks. Equipment and system performance are monitored and rated.

6.2.2.5 Instrumentation Requirements

6.2.2.5.1 Containment Spray System

Instrumentation and associated analog and logic channels employed for initiation of the containment spray system are discussed in section 7.3.

The injection phase of operation of the spray system is actuated either manually from the control room (2/2 logic) or on coincidence of two sets out of four high-high-high containment pressure signals. This signal starts the containment spray pumps and opens the discharge valves to the spray headers. When the RWST is exhausted, the recirculation phase of operation is manually initiated by the operator.

The following describes the instrumentation that is used for monitoring the containment spray system during normal or post-LOCA operation:

RWST level - Two channels of level instrumentation are provided on the RWST. Both channels additionally provide a high, low, low-low, and Technical Specification minimum level alarm which annunciate in the control room.

Containment sump water level - Two level indicators provide control room indication of containment sump level.

Containment pressure - Four channels provide containment pressure indication and high, high-high, and high-high-high containment pressure annunciations in the control room.

Pump discharge pressure - A pressure indicator is located in each containment spray pump discharge line. Readout is local.

Pump discharge flow - A flow indicator is located in each containment spray pump discharge line. Readout is provided in the control room.

6.2.2.5.2 Containment Cooling System

Instrumentation and associated analog and logic channels employed for initiation of the containment cooling system are discussed in section 7.3.

The high-speed winding for each dual-speed fan motor receives power from a normal bus, and the low-speed winding receives power from an emergency bus. Electrical interlocks prevent the connection of both power supplies simultaneously. During normal operation, up to four units will be operating. On receipt of an SI signal, with offsite power available, the operating fans will be electrically switched from high to low speed. One unit per train could then be placed in the standby mode by the operator. On receipt of an SI signal with a loss of offsite power (LOSP/SI), any running fans will trip (load shed signal or bus UV signal), and only two fans, as selected on the MCB train selector switches, will automatically start in slow speed. Additional fans may be manually started as desired, if diesel capacity and plant conditions permit. If failure of the electrical interlocks were to allow the fan motor to operate at high speed during an accident, the high-speed winding power supply would be disconnected by the overload trip due to the excessive current required to handle the high density steam air mixture. The fan motor would then be operated at low speed if required.

The following describes the instrumentation that is used to monitor the containment air cooling system during normal or accident conditions.

- Cooling water exit temperature - Cooling water exit temperature is monitored. Remote indication is provided in the control room.

- Cooling water flow - Remote indication of cooling water supply and return flow is provided and low return flow is annunciated in the control room.

- Cooling water radiation level - Cooling water return flow is monitored for radiation level. Indication and high-level annunciation are provided in the control room.

6.2.2.6 Materials

A. Decomposition Products

Materials used in or on ESF systems have been chosen so that the decomposition products, if any, of each material will not interfere with safe operation of any ESF. The estimated amount of material or chemicals that would interfere with the safe operation of this system is zero. There are no commercial name materials included in this system which would cause degradation of system performance.

B. Materials Compatibility

Materials compatibility of the containment spray system is discussed in paragraph 6.2.3.6.

6.2.3 CONTAINMENT AIR PURIFICATION AND CLEANUP SYSTEMS

Post-accident ECCS recirculation fluid pH control system operates in conjunction with the penetration room filtration system. The penetration room filtration system is provided to limit the radiological offsite consequences resulting from a LOCA. The containment preaccess filter system and the containment purge system provide ventilation and filtration to allow access to the containment after shutdown. The containment minipurge system provides continuous ventilation and filtration of the containment atmosphere so as to allow periodic occupation of the containment during normal power operation. The control rod drive mechanism (CRDM) cooling system, the reactor cavity cooling system, the refueling water surface ventilation system, and the containment air cooling system provide ventilation and cooling for equipment located inside the containment.

6.2.3.1 Design Bases

6.2.3.1.1 ECCS Recirculation Sump pH Control System

The ECCS recirculation sump pH control system is designed to ensure that offsite doses from an accident are within the limits of 10 CFR 100 by providing the following:

A. Scrubbing action in the containment atmosphere to reduce airborne iodine activity levels.

B. Retention of the iodine in solution.

C. The ECCS recirculation sump pH control system is designed to operate over a short period of time and under the environmental conditions existing after an accident as presented in table 3.11-1.

6.2.3.1.2 Penetration Room Filtration System

The primary purpose of the penetration room filtration system is to limit release to the environment of radioisotopes from ECCS leakage into the penetration room under accident conditions. Although not credited in accident analyses, the penetration room filtration system also filters radioisotopes from containment leaks into the penetration area following a DBA. The system performs the following safety functions:

A. Exhausting of the penetration room atmosphere to the environment through particulate and charcoal filters which remove airborne radioactivity and maintain a slightly negative pressure within the penetration room area. This negative pressure ensures inleakage to the penetration room area, which helps to minimize the release of radioactivity to the environment.

B. Processing the air from the fuel handling area following a fuel handling accident.

Post-LOCA, the flow through each filter train may either be exhausted to the plant vent or partially recirculated to the penetration room area and ECCS pump rooms depending on the alignment of the PRF system. Recirculation is provided as a feature that provides multipass filtration of long-term ECCS recirculation and containment leakage which may reduce post-LOCA releases to the environment through fission product cleanup. The recirculated flow when ducted through the penetration area boundary will provide dilution and mixing of airborne radioactivity within the boundary. Recirculation cleanup of air through the penetration rooms is not credited in the accident analysis.

The penetration rooms are formed adjacent to the outside surface of the containment by enclosing the area around all penetrations, except as noted below. The penetration room boundary is shown in figures 1.2-2 through 1.2-7 and encloses a volume of 328,000 ft^3. The only penetrations which do not pass within the penetration room are the following:

- Containment leak rate test connections.

- Refueling tube.
- Equipment hatch, which contains a doubled gasketed closure.
- Personnel access lock.
- Auxiliary access lock.
- Main steam and feedwater penetrations.

The containment leak rate test connections and the refueling tubes are equipped with blind flanges which are opened only during shutdown. The equipment hatch, personnel access lock, and auxiliary access lock can be tested during normal operation and are not considered sources of significant leakage. The main steam and feedwater penetrations are not considered a source of significant leakage because they are welded to the containment liner plate.

In analyzing the possibility of containment isolation valve leakage bypassing the penetration room filtration system, the penetrations can be divided into the following groups (the numbers in parentheses refer to the item numbers listed in table 6.2-31):

A. Penetrations for systems open to the containment atmosphere after a LOCA but which have components within the penetration room boundary.

   1. RHR loop-in/low-head SI (10).
   2. Reactor coolant pump seal water supply (14).
   3. Containment spray lines (36).
   4. High-head SI lines (35).
   5. Containment sump recirculation lines (9, 37, and 38).
   6. High-head to RCS cold-leg injection line (43).
   7. Low-head SI line (44).

   Potential leakage from the components of these systems is contained within the penetration room boundary.

B. Penetrations for systems which operate after a LOCA and are not open to the containment atmosphere, but which extend beyond the penetration room boundary.

   1. Service water lines to and from the containment air coolers (27 and 28). Potential leakage from these lines can be analyzed as follows:

a. The pressure in the supply line to the coolers is approximately 50 psig. Any leakage, even for the short-term pressure transient, is into the containment. Isolation valves can be closed to isolate the supply line if necessary.

b. The pressure in the return lines from the coolers is approximately 15 psig. Any leakage into the service water system is detected by the radiation monitor discussed in chapter 11, and isolation valves are closed to isolate the return line.

C. Penetrations for systems which are completely isolated by containment isolation valves after a LOCA and which form closed loops within the penetration room boundary.

1. Charging pump suction relief valve discharge to pressurizer relief tank (6).

2. Containment differential pressure instrument (8).

3. RHR loop-out (9).

4. Containment pressure instrument (15).

5. Containment sump pumps discharge (26).

6. Containment sump pump sample recirculation line (47).

7. Post-LOCA containment sampling system (48 and 49).

Potential leakage from these systems is contained within penetration room boundary, with the exception of backleakage to the reactor water storage tank vent past the isolation valves of the containment sump pumps discharge. This leakage pathway is included in the evaluation of LOCA dose consequences in section 15.4.

D. Penetrations for systems which are completely isolated by containment isolation valves after a LOCA but pass through the penetration room into other portions of the auxiliary building. They are connected to Seismic Category I systems and are water filled in the post-accident condition.

1. Pressurizer relief tank makeup (5).

2. Normal charging line (13).

3. Pressurizer steam sample line (16).

4. Pressurizer liquid sample line (17).

5. Hot leg sample line (18).

6. Reactor coolant pump cooling water supply and return (29 and 31).

7. Reactor coolant pump thermal barrier cooling water return (32).

8. Excess letdown heat exchanger and RC drain tank heat exchanger component cooling water supply and return (33 and 34).

9. Accumulator sample line (40).

10. Pressurizer pressure deadweight generator (41).

11. Service water to and from reactor coolant pump motor air coolers (45 and 46).

12. Normal letdown line (11).

13. Excess letdown and seal water (12).

Potential leakage from these systems is considered in the same manner as potential leakage from the penetrations in group E below.

E. Penetrations for systems which are completely isolated by containment isolation valves after a LOCA but pass through portions of the auxiliary building other than the penetration room and are either connected to nonseismic Category I systems or are not water filled in the post-accident condition.

1. Accumulator test line (1).

2. Refueling cavity supply (2).

3. Nitrogen supply to accumulators (3).

4. Nitrogen supply to pressurizer relief tank (4).

5. Reactor coolant drain tank drain (7).

6. Deleted.

7. Deleted.

8. Fuel transfer tube (19).

9. Service air (20).

10. Instrument air (21).

11. Containment air sampling system (22 and 23).

12. Containment purge supply and exhaust (24 and 25).[a]

13. Leak rate test lines (30).

14. Accumulator makeup line (39).

15. Reactor coolant drain tank vent (42).

16. Post-LOCA containment venting (50).

17. Demineralized water (51).

As stated in subsection 6.2.4, all containment isolation valves which are closed in the post-LOCA condition are individually tested for leaktightness as required by the Containment Leakage Rate Testing Program. The allowable rate of isolation valve leakage is defined and limited by the Containment Leakage Rate Testing Program. Any leakage past the isolation valves of the systems in groups D and E above has the potential to leak into the auxiliary building beyond the penetration room boundary.

The design environmental conditions for the penetration room filtration system components are presented in table 3.11-1.

6.2.3.1.3 Containment Preaccess Filtration and Purge Systems

The containment preaccess filtration and purge systems are designed to do the following:

A. Provide sufficient air circulation and filtering throughout all containment areas to permit safe and continuous access to the containment within 2 h after reactor shutdown, assuming defects exist in no more than 1% of the fuel rods.

B. The minipurge system will provide sufficient purge air circulation so as to allow up to 6 h per week occupancy of the containment during power operation.

C. The containment purge system will normally provide purging of the containment after cold shutdown (prohibited in Modes 1 through 4). The minipurge system may be in use during reactor operation, reactor shutdown, and defueled. Interlocks prevent simultaneous operation of the minipurge system and the containment purge system.

---

a. Containment purge supply and exhaust lines contain a third isolation valve and a normally open bleed valve. As shown in drawings D-175010, sheet 1 and D-175010, sheet 2, any leakage past the first two isolation valves is vented into the penetration room. Similarly, any leakage past the minipurge supply and exhaust isolation valves is vented into the penetration room.

D. The containment purge system and the minipurge system provide for purging of the containment to the vent stack for dispersion to the environment.

E. The isolation valves on the minipurge system are capable of containment purge isolation within 5 s in the event an accident is detected in the containment. The containment purge system isolation valves are closed and deactivated during Modes 1 through 4. During modes 5 and 6, these valves are designed to close within 5 s upon receipt of a high radiation signal and containment ventilation isolation signal.

Total releases due to containment purging are in accordance with requirements of the applicable portions of the Code of Federal Regulations, Title 10. For a discussion of plant releases due to containment purging, refer to subsection 11.3.6.

6.2.3.1.4 Containment Ventilation Systems

The containment ventilation systems, consisting of the containment air cooling system, the CRDM cooling system, the reactor cavity cooling system, and the refueling water surface ventilation system, are designed to perform the following functions:

A. Remove the normal heat loss from all equipment and piping in the containment during plant operation and to limit the maximum average ambient temperature to 120°F.

B. Provide for depressurization of the containment following an accident. The post-accident design and operating criteria are detailed in subsection 6.2.2.

C. Provide a reliable supply of cooling air to the CRDMs.

D. Remove water vapor above the reactor cavity pool during refueling operations to improve visibility of fuel elements within the pool, reduce the heat stress on personnel, and minimize personnel inhalation exposure during shutdown.

E. Provide a minimum containment ambient temperature of 60°F during reactor shutdown.

F. Provide cooling air for the primary shield, the enclosed nuclear instrumentation detectors, and the reactor vessel supports.

The systems are designed to satisfy the containment ventilation and cooling requirements during normal operation. In certain cases where ESF functions are also served by the systems, component sizing is determined from the heavier duty specifications based upon accident conditions and requirements.

6.2.3.2 System Design

6.2.3.2.1 ECCS Recirculation Sump pH Control System

The ECCS recirculation fluid pH control system consists of trisodium phosphate baskets located in the recirculation sump area of containment. The initial spray water consists of borated water from the RWST. As the containment is sprayed during injection, the recirculation sump water level rises, eventually immersing the baskets and dissolving the trisodium phosphate contents. Once the contents are dissolved through the fine mesh basket screen sides and bottom, the recirculation sump water pH is raised to an equilibrium value of 7.0 or greater. The containment spray pumps will circulate approximately 5 percent of the flow through their bypass line. Refer to D-175038, sheet 3 and D-205038, sheet 3 for configuration.

When the RWST reaches low-low level, recirculation spray flow is initiated for iodine removal. The operator can remotely initiate recirculation flow by use of either of the spray pumps. This mode of operation is continued for a period of at least 2 h following the accident in order to continue iodine removal from the containment atmosphere.

Coatings used in the containment meet ANSI 101.2-1972. This requirement is not applicable to paints procured for the touch-up or coating of off-the-shelf items or equipment having manufacturers' standard finishes. Most types of paint used in the containment are listed by manufacturers' designation in paragraph 3.8.1.6.6. Table 6.2-37 provides a listing of most containment coatings including dry specific gravity and surface area covered. The quality assurance program for paint applications is discussed in chapter 17.

6.2.3.2.2 Penetration Room Filtration System

The penetration room filtration system is shown in D-175022, while the conformance of the penetration room filtration system to NRC acceptance criteria contained in Regulatory Guide 1.52 is presented in table 6.2-25.

The system consists of two full-capacity redundant fans and filter systems which share a common vent. The fans and filters are located in the penetration room filtration system equipment room.

The design flowrate through each filter train is 5000 sft^3/min. The exhaust fan has a capacity of 500 sft^3/min., and the recirculation fan capacity is 4500 sft^3/min. These design flowrates are based on the following sizing criteria:

(**Note:** The assumed inleakage used in sizing the fan capacities was not achieved during plant startup. Higher inleakage rates are acceptable due to conservative system sizing and the system's ability to maintain a slightly negative pressure.)

A. The exhaust flowrate is equivalent to the penetration room boundary inleakage; i.e., the sum of all possible inleakages when a pressure of -1.5-in. WG is maintained within the penetration room boundary. This is an important parameter because it determines the relationship between the amount of filter effluent discharged to the environment and the amount recirculated to the penetration room.

Minimizing the penetration room inleakage increases the system effectiveness. Evaluation of the proposed structure joints and pipe and cable tray seals indicates a potential inleakage rate of < 10 percent of the penetration room boundary volume per day. However, for estimating the exhaust fan capacity, it has been conservatively assumed that, with a -1.5-in. WG pressure, the inleakage is 100% of the penetration room volume per day. This inleakage is equivalent to 250 sft^3/min. Each exhaust fan has been conservatively designed to provide twice this flowrate.

B. The design ratio of recirculation flow to exhaust flow is 9 to 1. No recirculation credit has been taken in calculating the consequences of an accident.

Each system consists of a prefilter, high efficiency particulate air (HEPA) filter, charcoal filter, heating coil, and two fans.

Both the recirculation and exhaust fans may be used to hold the penetration room area and ECCS pump rooms or the spent fuel pool area at a negative pressure with respect to the penetration room filtration equipment room following a LOCA or FHA, respectively.

During normal operations, the spent fuel pool area is served by its normal heating, ventilating, and filtration system. Whenever irradiated fuel is in the spent fuel storage pool, one penetration room filtration system train is aligned to the spent fuel pool area. Both penetration room filtration system trains are aligned to the spent fuel pool area during movement of fuel or heavy loads over the fuel. During maintenance activities, the penetration room filtration system may be aligned to allow the spent fuel ventilation system to exhaust penetration room areas in order to provide temporary ventilation. Under accident conditions, the penetration room filtration system can be aligned to operate under either of two modes, namely the fuel handling accident (FHA) mode or the post-LOCA operating mode. While the penetration room filtration system is not credited in the FHA dose analysis (subsection 15.4.5), its operation in FHA mode is described below.

Upon receiving an actuation signal initiated by either high radiation or low flow in the normal spent fuel pool ventilation system, the penetration room filtration system is automatically placed in its fuel handling accident alignment by starting the penetration room filtration system fans and isolating the normal spent fuel pool ventilation system. The spent fuel pool room is automatically isolated by the closure of redundant isolation dampers located in the normal spent fuel pool HVAC supply and exhaust ductwork. Movement of new fuel over spent fuel with the spent fuel area roof new fuel access hatch open creates the potential for a fuel handling accident with a release pathway which bypasses the radiation monitors in the exhaust duct, and consequently a bypass of the PRF. This configuration is specifically evaluated in subsection 15.4.5 as being bounded by the design basis accident FHA. Both Train A and B penetration room filtration systems start based on input from their respective instruments. Operators may manually shut down one train after the autostart of both penetration room filtration system trains. Train A and B penetration room filtration systems share a common duct from the spent fuel pool area to the penetration room filtration system mechanical equipment room.

Upon receiving a containment isolation actuation system (CIAS) phase B signal from the protection system, both Train A and B penetration room filtration system fans energize and the suction dampers to the penetration room area open automatically. The flow through each filter train may either be exhausted to the plant vent or partially recirculated to the penetration room area with the remainder exhausted to the plant vent stack. The recirculation damper may be positioned automatically or manually to maintain negative pressure control of the penetration room area. The penetration room filtration system recirculation and exhaust discharge dampers to the plant vent stack are interlocked with their respective fans to open and close automatically whenever the fan starts or stops. A backdraft damper is in series with the discharge dampers to prevent backflow through the filtration unit. Before post-LOCA emergency core cooling recirculation, the penetration room filtration system is placed in its post-LOCA alignment by manually securing spent fuel pool area suction line valves closed.

<u>Fans</u> - The exhaust and recirculation fans are closed impeller centrifugal fans. The fan motor sizes are 1.75 hp for the exhaust fan and 20 hp for the recirculation fan. Fan motors conform to standards of NEMA, IEEE, and ANSI.

<u>Filters</u> - The filters are composite units consisting of prefilter section, absolute filter section, and impregnated charcoal bed filter section. Each section is designed as follows:

A. The prefilters are designed to have a mean efficiency of 85% when tested in accordance with the National Institute of Standards and Technology Discoloration Test Method.

B. The HEPA filters are designed to be capable of removing 99.97% minimum of particulate matter 0.3 micron or larger in size. This particulate filter is water and fire resistant.

C. The charcoal filter is an impregnated, activated carbon bed that is designed to be capable of removing elemental iodine with an efficiency of 99.0% and, at relative humidities below 70%, all iodines, including organic, with an efficiency of 99.0 %.

The prefilter, absolute, and charcoal filters are designed for a nominal flowrate of 1000 ft^3/min per 4 ft^2 face area and are sized with the assumption that 100% iodines from ECCS recirculation leakage will pass through the penetration room filtration system and will also be held up on the filters. The total amount of carbon in each penetration room filtration train is approximately 774 lb.

The HEPA filters used in the penetration room filtration system are designed and manufactured in accordance with the requirements of USNRC Health and Safety Bulletin No. 306 (Military Specification MIL-F-51068C, June 8, 1970, Filter, Particulate, High-Efficiency, Fire-Resistant), and USNRC Health and Safety Bulletin No. 297. The charcoal is activated, coconut base charcoal, impregnated and retained in horizontal trays of nominal 2-in. bed depth in accordance with AACC-CS-IT, 1968. Humidity control consists of a heater which energizes when the penetration room filtration system fans are operating.

<u>Ducting</u> - The ducting upstream of the fans takes suction at the higher elevation of the penetration room boundary, and the recirculation ducts discharge air into the lower elevation of the boundary.

The ducts upstream of the fans are designed to operate under a pressure of -10-in. WG. The ducts from the discharge of the fans to the vent stack are designed for 2-in. WG positive pressure. The recirculation ducts are designed to operate under a pressure of -6-in. WG.

<u>Valves</u> - The valves at the discharge of all fans are pneumatically operated, two-position, fail-open butterfly valves. The valves in the ducting of the filter trains and in the suction and recirculation lines are motor-operated valves.

The penetration room filtration system instrumentation is in accordance with the requirements of IEEE 279.

Three differential pressure transmitters measure the differential pressure between the penetration rooms and pressure in the penetration room filtration system equipment room. The recirculation fan continues to operate in the exhaust mode until the pressure in the penetration room boundary reaches a predetermined set point. The analysis of the combined system (fans vs. inleakage) indicates a setpoint of -2-in. WG pressure to be used for switching to recirculation operation. A two-out-of-three arrangement of differential pressure switch signals actuates the recirculation line butterfly valve to open when the setpoint is reached. A two-out-of-three differential pressure signal of -2 in. and a recirculation line valve open signal is indicated in the control room. A permit switch allows the recirculation line valve to open only if the recirculation fan is operating.

After automatic post-LOCA valve alignment, the system valves may be closed or throttled by the operators to affect ALARA as long as negative pressure is maintained. The flow from the exhaust and recirculation fans may be exhausted to the plant vent or aligned to partially recirculate flow to the penetration room boundary with the remaining exhausted by the plant vent.

In the event of control air failure, the penetration room filtration system air operated valves fail open (in the exhaust mode), except the spent fuel pool suction valves to the penetration room filtration system, which fail closed on loss of control air. In the event of control air failure during fuel handling operation, air accumulators provide backup air supply. The suction valves are manually closed remotely during post-LOCA operation.

If both systems start following an accident, one can be manually shut down and placed in the standby mode. The standby system is manually restarted if the operating system should fail.

The following conservative assumptions were used to calculate an upperbound value of heat generation rate inside the penetration room filtration system charcoal filters versus time after a LOCA:

A. Regulatory Guide 1.4 source terms.

B. Leak rate from the containment into the penetration room volume of 0.15 containment volume percent per day.

C. Instantaneous deposition of 100% leaked iodines onto the charcoal filters.

D. Heat deposition in the filters from 100% of all betas and 50% of all gammas.

E. No spray removal of contaminants inside containment.

Using these assumptions, a peak filter heat generation rate of 163 W is calculated to occur at 11.5 days after a LOCA. Further calculations indicate that natural convection is more than sufficient to remove this heat load from an isolated filter, while maintaining filter temperatures acceptably low.

The penetration room filtration system mechanical room is heated by area heaters located in the filter area to ensure that warm, moist air cannot be processed by relatively cooler filters, thereby resulting in moisture condensation and accumulation inside the charcoal filters. Electric heating coils are also provided at the upstream side of each unit to maintain the entering air below 70 percent relative humidity. The heating coils are provided for defense-in-depth since no credit is taken for reduction of relative humidity in the accident analysis. Temperatures in the emergency pump rooms within the penetration room boundary are controlled by pump room coolers in each room.

Details of the emergency power sources are discussed in subsection 8.3.1.

6.2.3.2.3 Containment Preaccess Filtration and Purge Systems

The containment preaccess filtration system is designed to reduce activity levels in the containment atmosphere prior to routine personnel access at power or in advance of a scheduled reactor shutdown. The system, as shown in drawings D-175010, sheet 1 and D-175010, sheet 2 draws contaminated air from the containment across a filter assembly which consists of a prefilter, HEPA filter, and a charcoal filter. The air then passes through the system fan and is discharged back into the containment. Design system flowrate is 10,000 ft^3/min through each of two filter subsystems.

The containment purge system is independent of any other system and includes provisions to supply and exhaust air from the containment. The system is shown in drawings D-175010, sheet 1 and D-175010, sheet 2; D-205010, sheet 1 and D-205010, sheet 2. The supply system includes an outside air connection to prefilters, heating coils, a fan, a duct system, and a supply penetration with three butterfly valves in series for tight shutoff. The exhaust system includes an exhaust penetration with three butterfly valves in series, a duct system, a filter bank with prefilters, HEPA and charcoal filters, and an exhaust fan. The exhaust system includes a two-speed fan so that purging can be performed at half or full flowrate. The full flowrate is 48,500 ft^3/min. The quick closing purge isolation valves are capable of closing within 5 s, during Modes 5 and 6, upon receipt of a high radiation signal from the purge discharge radiation monitors, and upon receipt of a containment ventilation isolation signal.

NRC acceptance criteria associated with the design of the containment purge system include the capability to detect high radioactivity conditions in containment prior to opening any of the purge valves. High radioactive conditions inside containment would be detected prior to opening the purge valves by means of the containment atmosphere particulate radioactivity monitor (R-11) and the containment atmosphere gaseous radioactivity monitor (R-12). Should prescribed high radiation levels exist, an audible annunciation would be provided in the control room.

The containment minipurge system is designed to maintain radioactivity levels in the containment consistent with occupancy requirements with continuous system operation. The preaccess filtration system is not required for this occupancy but is available for use in minimizing the need for containment purging.

The operation of the minipurge system is independent of the operation of the containment purge system, although there are common ductwork and common filters. The system is shown in drawing D-175010, sheet 1 and D-175010, sheet 2. The supply system uses the containment purge supply filter and ducting. A separate minipurge supply fan, located in the penetration room, operates in an 18-in. duct system which reduces to 8 in. and bypasses the first purge isolation valve (48 in.) outside the containment. The outside air is discharged through an 18-in. duct which reduces to 8 in. and is connected to the 48-in. purge duct. The minipurge supply system has two isolation valves in series for tight shutoff. The exhaust system has an exhaust connection inside the containment and an exhaust fan similar to arrangement to the supply system. Exhaust air passes through the containment purge exhaust filter. The exhaust system has two isolation valves in series.

The operation of the exhaust fan is dependent upon the operation of the supply fan. An interlock from the supply fan is installed in the control circuit of the exhaust fan such that the exhaust fan runs automatically when the supply fan is running. However, the exhaust fan may be operated independent of the supply fan, when the main purge supply valves or the minipurge supply fan is out of service by defeating the interlock between the two fans. During this mode of operation the plant administrative controls will be in place to ensure that the containment pressure is maintained within the Technical Specification limits.

The minipurge supply fan provides a flowrate of approximately 2850 ft^3/min and the exhaust fan provides a flowrate of approximately 2650 ft^3/min. The 8-in. isolation valves are similar in design to the 48-in. containment purge isolation valves and they are capable of closing in 5 s after receipt of a containment ventilation isolation signal or a high radiation signal from the purge discharge radiation monitors.

The plant can operate either in the minipurge mode, with the 48-in. containment isolation valves closed in the supply and exhaust ducts during plant Modes 1 through 6 and defueled, or in the full-purge mode, with the 48-in. containment isolation valves open and the 8-in. minipurge isolation valves closed during plant Modes 5 and 6 or defueled. Interlocks prevent operation in both modes simultaneously.

The containment purge system and minipurge system are designed to meet the NRC acceptance criteria for containment isolation requirements found in Standard Review Plan (SRP) Section 6.2.4, Revision 1; Branch Technical Position CSB 6-4, Revision 1; SRP Section 3.9.3, Revision 1; and NUREG-0737, item II.E.4.2. In addition, the safety signals to all purge and ventilation isolation valves meet the following criteria:

A. The overriding (i.e., the signal is still present but is blocked in order to perform a function contrary to the signal) of one type of safety actuation signal (e.g., radiation) must not cause the blocking of any other type of safety actuation signal (e.g., pressure) to the isolation valves.

B. Sufficient physical features (e.g., key lock switches) are provided to facilitate adequate administrative controls.

C. The system-level annunciation of the overridden status is provided for every safety system impacted when any override is active.

D. Diverse signals should be provided to initiate isolation of the containment ventilation system. Specifically, containment high radiation, SI actuation, and containment high pressure should automatically initiate containment ventilation isolation.

E. The instrumentation and control systems provided to initiate containment ventilation isolation should be designed and qualified as safety-grade equipment.

F. The overriding or resetting (i.e., the signal has come and gone, and the circuit is being cleared in order to return it to the normal condition) of the isolation actuation signal should not cause the automatic reopening of any isolation/purge valve.

Other NRC acceptance criteria include:

G. The radioactivity released during normal operation will be within the limits of column 1, table II, Appendix B to 10 CFR 20.1 - 20.601; the total radioactivity released following an accident results in calculated offsite doses less than the guideline values in 10 CFR 100; and the systems meet the applicable requirements of General Design Criterion 56.

H. The resilient material valve seals of the 48-in. and the 8-in. containment purge supply and exhaust isolation valves shall be replaced at least once per 9 years, in accordance with the Technical Requirements Manual.

The temperature and the humidity of the air stream through the preaccess and the purge systems are the same as for the containment atmosphere and are maintained by the containment cooling and ventilating systems.

Fans - The fan used in each half-capacity subsystem of the containment preaccess filtration system is of the vaneaxial type, with a design flowrate of 10,000 sf^3/min each. Fan motors are 25 hp each. Both the exhaust fan and the supply fan in the containment purge system are two-speed centrifugal fans with design flowrates of 50,000/25,000 sf^3/min each.

Fan motors are 60 and 125 hp for the supply and exhaust fans, respectively. Both the minipurge supply and exhaust fans are centrifugal fans with rated flowrates of 5000 sf^3/min each; however, the supply fan will operate at approximately 2850 ft^3/min and the exhaust fan will operate at approximately 2650 ft^3/min with the 8-in. minipurge valves. Motors for these fans are

15 hp each. They are designed in accordance with the applicable portions of AMCA 99-67, Standards Handbook; AMCA 210-67, Test Codes for Air Handling Devices; and AMCA 211A-65, Certified Rating Program for Air Moving Devices.

Filters - The filters are composite units consisting of prefilter sections, absolute filter section, and impregnated charcoal bed filter section. The prefilter section and the absolute (HEPA) filter section are as described in paragraph 6.2.3.2.2. The charcoal filters are impregnated, activated carbon beds that are designed to be capable of removing, at relative humidities below 70%, all iodines with an efficiency of at least 95%. The prefilters and the absolute and charcoal filters are designed for a nominal flowrate of 1000 ft^3/min per 4-ft^2 face area. Carbon weights are approximately 1875 lb and 7400 lb for the preaccess filtration and purge systems, respectively.

The containment purge exhaust filters are seismically qualified. Details of the emergency power sources are discussed in chapter 8.

6.2.3.2.4 Containment Ventilation Systems

The containment ventilation systems flow diagram is shown in drawings D-175010, sheet 1; D-175010, sheet 2; D-205010, sheet 1; and D-205010, sheet 2. The containment air cooling fans, CRDM cooling fans, and reactor vessel support cooling fans are direct driven units, each with standby units for redundancy. Each system is provided with pressure differential switches to verify the existence of airflow in the associated ducting.

The emergency functioning of the containment air coolers is discussed in subsection 6.2.2. During normal operation, the four coolers take suction from the operating level at el 155 ft and discharge into a common header which distributes the cooled air to the lower regions of the containment through distribution ductwork. Each air cooling unit consists of the following equipment arranged so that, during normal operation, air flows through the assembly in the following sequence: inlet screen, cooling coil, fan, and a discharge header which is common to all units. Containment cooler and cooler fan design parameters are presented in table 6.2-28.

The containment air cooling units are located at the 155-ft elevation outside the secondary shield area. The shielded location makes inspection of the equipment possible at power under controlled access conditions and immediately after a hot shutdown. The containment cooling and ventilating functions are augmented by the containment recirculation fans, which take suction from the containment dome and discharge downward toward el 155 ft. Design parameters for these four fans are presented in table 6.2-28.

The containment air cooling units, in conjunction with the containment recirculation fans, provide mixing of the containment atmosphere during normal operation to augment heat removal and maintain uniform temperature distributions throughout the containment volume. A portion of the containment air cooler discharge is ducted directly to the reactor cavity to provide cooling to the cavity and to the instrumentation and equipment within.

The CRDM cooling system consists of fans and ducting to draw air through the CRDM shroud and eject it to the main containment atmosphere. One hundred-percent redundancy is provided by a standby fan.

In the event of a failure of the CRDM cooling system, the airflow to the CRDMs may be lost. Loss of airflow to the CRDM would increase the operating coil temperatures. In the event of high CRDM temperatures, the control rods fall to the safe position, i.e., inserted into the core.

Design parameters for the CRDM cooling system fans are presented in table 6.2-28.

The reactor vessel support cooling system, consisting of two 100% capacity fans and ducting, is arranged to cool the reactor vessel supports by drawing air through the supports. One hundred percent redundancy of all active components is provided.

Reactor cavity cooling system fan design parameters are presented in table 6.2-28.

The refueling water surface ventilation system is utilized when necessary during refueling operations to remove water vapor above the refueling canal, thereby improving the visibility of the fuel elements within the pool and reducing the heat stress on personnel working in the vicinity of the refueling canal. This is accomplished by having a supply fan draw air from the containment atmosphere and supply it above the water surface. This air mixes with the water vapor emanating from the canal. The exhaust fan draws air from the opposite side of the canal and exhausts to the containment where it is diluted and filtered before being discharged through the plant vent. Refueling water surface ventilation system supply and exhaust fan design parameters are presented in table 6.2-28.

6.2.3.3 <u>Design Evaluation</u>

6.2.3.3.1 ECCS Recirculation Sump pH Control System

The spray system, by virtue of the large surface area provided between the droplets and the containment atmosphere, affords an excellent removal mechanism for fission products postulated to be dispersed in the containment atmosphere. Radioiodine, in its various forms, is the fission product of primary concern in the evaluation of a LOCA. The major benefit of the containment spray is its capacity to absorb molecular and particulate iodine from the containment atmosphere. To enhance the capacity to retain this iodine, the recirculation solution is adjusted to an alkaline pH which promotes iodine hydrolysis to nonvolatile forms.

Values for the spray removal half-life of iodine in a typical containment are on the order of minutes. This makes the spray system a very efficient fission product removal system in comparison to such alternatives as charcoal filtration systems.

6.2.3.3.1.1 **<u>Containment Spray Iodine Removal Model</u>.** The spray iodine removal model is based on data obtained from measurements of the performance of Spray Engineering Company nozzle 1713 for the range of pressures expected to be encountered in the containment for a LOCA. A conservative elemental iodine spray removal coefficient, as shown in table 6.2-29, is used based on the data provided in references 19 through 21. The limiting decontamination factor (DF), calculated in accordance with reference 21 using a partition coefficient based on NUREG-0800, is reached at about 24 min.

A particulate iodine spray removal coefficient is calculated by the method outlined in reference 20. Removal efficiency for particulate iodine is assumed to decrease by a factor of 10 upon reaching a DF of 50, and to cease at 8 h.

6.2.3.3.1.2 Deposition Iodine Removal Model. The deposition removal coefficient for elemental iodine is calculated based on the guidance of reference 20. Deposition is assumed to occur only on galvanized surfaces and surfaces coated with zinc based paint or epoxy paint, as listed in table 6.2-2. The deposition removal rate of iodine is assumed to continue at the initial rate until a DF of 100 is achieved for the containment atmosphere, and then at a reduced rate until a DF of 1000 is achieved, as shown in table 6.2-29.

6.2.3.3.1.3 Spray Performance Evaluation.

Injection Phase Operation

The spray iodine removal analysis is based on the assumptions that:

A. Only one of two spray pumps is operating.

B. The iodine removal constants are calculated based on conservative containment parameters and test data from references 19 and 21.

The performance of the spray system was conservatively evaluated at the peak temperature and pressure resulting from a double-ended rupture of the RCS with no credit taken for the subcooling of the ECCS. These pressure and temperature conditions were used resulting in a minimum spray flowrate of 2480 gal/min during injection and 2240 gal/min during recirculation for the single spray pump assumed to be operating.

Iodine removal constants for the spray system, calculated with the model described and with the above mentioned assumptions, are shown in table 6.2-29.

Recirculation Phase

The containment spray system is operated in the injection mode until the RWST low-low level setpoint is reached. The spray system is then operated in the recirculation mode, taking suction from the containment sump.

Reevolution of Iodine

Any reevolution of dissolved iodine from the sump to the containment atmosphere is dependent on the concentration gradient between the liquid and vapor phases, pH, and temperature. The assumptions used minimize the amount of TSP dissolved in the sump, and maximize the temperature and volume of water delivered to the sump.

With this set of assumptions, the containment recirculation sump conditions (temperature and pH) are such that the required partition coefficient to maintain the calculated DF in the vapor phase is exceeded in the spray recirculation phase.

6.2.3.3.2 Penetration Room Filtration System

The penetration room filtration system consists of two complete and separate fan and filter systems, each with 100-percent capacity. Although the system actuation signal automatically starts both fan and filter systems, one system can be placed in the standby mode by the operator. Motor or fan failure in the operating system is annunciated in the control room. The standby system is then manually started. Upon loss of offsite power, the entire system is automatically connected to the emergency diesel generators.

The ducting conveying unfiltered air upstream of the filter trains is at negative pressure, barring the possibility of outleakages of contaminants.

Filter components receive factory and field tests to ensure against bypass and confirm specified efficiencies.

The penetration room filtration system produces a slightly negative pressure gradient inside the penetration room boundary by exhausting air from the highest point within the boundary, the electrical penetration room. Communicative paths between the various rooms within the penetration room boundary are connected by a series of pipe sleeves of various sizes to serve as flow paths from the penetration room boundary lower levels. Pressure differentials between the highest and lowest points within the penetration room boundary are small and allow for boundary pressure to be sensed at a single point and displayed in the control room, with high and low boundary pressure alarmed. Pressure detection within the penetration room boundary is at the location of least negative pressure, the RHR heat exchanger room, which is located at the lowest level within the boundary.

The RHR heat exchanger room is maintained at a slightly negative pressure with the penetration room filtration system partially recirculating to the penetration room boundary with the remainder exhausted to the plant vent stack. If the penetration room filtration system exhausts all air from the boundary to the vent stack, much lower negative boundary pressures can be maintained. The penetration room boundary is located primarily below grade and is completely enclosed within the auxiliary building; therefore, no unfiltered leakage from wind effects are considered.

The penetration room filtration system ensures that offsite radiation exposures resulting from post-LOCA ECCS recirculation leakage are within the requirements of 10 CFR 50.67.

A single-failure analysis for the penetration room filtration system is given in table 6.2-30.

The penetration room model used in the LOCA analysis is given in paragraph 15.4.1.10.

6.2.3.3.3 Containment Preaccess Filtration and Purge Systems

An evaluation of the capability of the containment preaccess filtration, minipurge, and purge systems to maintain offsite effluent concentrations during normal operation within established guidelines is included in subsection 11.3.6. Radiation monitors isolate the minipurge and full purge system supply and discharge in the event of high discharge activity levels, such as might be experienced following a fuel handling accident inside the containment.

6.2.3.3.4 Containment Ventilation Systems

The post-accident operation of the containment air coolers is evaluated in paragraph 6.2.2.3. The normal operation of the containment air coolers and the operation of the remaining containment ventilation systems are not required to reduce accident doses or maintain offsite effluent concentrations during normal operation within established guidelines.

6.2.3.4 Tests and Inspections

6.2.3.4.1 ECCS Recirculation Sump pH Control System

The functional and periodic post-operational testing of the containment spray system is described in paragraph 6.2.2.4.1. A test signal simulating the containment spray signal is used to demonstrate the operation of the spray system up to the isolation valves on the pump discharge. The isolation valves are closed for the test. These isolation valves are checked separately.

ECCS recirculation sump pH control is a passive system which consists of three baskets located in the recirculation sump area of containment. Each basket has level marks visible from the outside which indicate the acceptable range in trisodium phosphate volume for the basket for the trisodium phosphate compound being used (currently $Na_3PO_4 \bullet 12H_2O \bullet 1/4NaOH$). An equivalent amount of trisodium phosphate compound with a different chemical formula may be used. When equivalent compounds are used, the allowable weights/volumes may be different; however, the equivalent amount of trisodium phosphate compound must raise the pH of the recirculating solution into the range of 7.0 to 10.5. A visual inspection of the baskets is performed each refueling outage to verify structural integrity and level for the baskets. The intent of the associated surveillance requirements is to verify containment of the trisodium phosphate. Therefore, broken, crimped, or oxidized screen mesh is acceptable as long as the contents are contained. Also, lumps/caking is an analyzed condition.

6.2.3.4.2 Penetration Room Filtration System

Fans are tested and rated in accordance with the standards of the Air Moving Conditioning Association (AMCA).

The penetration room filtration system, as well as its components, are tested prior to startup and periodically during operation. Written test procedures establish minimum acceptance values for all tests. A recording of test results and instrumentation that alarms at the main control board enables early detection of faulty performance.

The penetration room filtration system is provided with testing facilities to demonstrate system operability.

Periodic tests of the activation circuitry and the system components can be conducted during normal plant operation ensuring a reliable performance upon demand for the life of the plant. Periodic testing of the penetration room filtration system actuation instrumentation is conducted in accordance with the Technical Specifications.

[HISTORICAL] *[Initial tests and the purpose of each test are listed as follows:*

A. *Component qualification tests - These tests demonstrate the characteristics of materials to be incorporated by the manufacturer into components for a system and ensure that they meet the requirements of procurement specification. The design conditions, which form the basis for these component qualification tests, are presented in table 3.11-1.*

B. *Component acceptance tests - These tests are factory tests which demonstrate the capability of the components incorporated in the various systems in which they are to operate. For example, fans associated with safeguards systems are tested in the manufacturer's shop to determine their characteristic curves. System valves are tested in the shop to verify effectiveness of seal, opening and closing periods, and the ability of the valve operator to actuate the valve at the maximum anticipated differential pressure.*

Test results on actual or similar types of filter assemblies demonstrate their adequacy for this application. The following demonstrative tests are performed:

A. *Radioactive iodine removal efficiency - a charcoal sample 2 in. in diameter by 2 in. deep is exposed to air flow at 40 ft/min face velocity. The air stream contains concentrations of elemental iodine and methyl iodide, similar to those predicted to occur in the penetration room filters during faulted conditions. Air stream temperature is 150°F, relative humidity 70 percent, and test duration is 12 hours. The efficiency is determined by measuring the activity of iodines upstream and downstream of the sample. Minimum acceptable efficiency is 99.0 percent at the end of 12 hours.*

B. *Flow resistance test - A module consisting of three absorbine units (six trays), stacked vertically, is capable of filtering 100 ft^3/min of air at a pressure drop not exceeding 1.0 in. wg. The actual resistance is recorded and kept available.*

C. *Leak test - Each filter element is tested for 5 minutes in an air flow of 330 ft^3/min containing approximately 20 ppm of Freon 112. Instrumentation is provided to measure the relative upstream and downstream concentrations of Freon 112. A downstream concentration in excess of 0.2 percent of the upstream concentration shall cause rejection of the filter.*

D. *Carbon lot tests - A sample from each lot of carbon after impregnation will have been subjected to the following tests by the manufacturer and results made a matter of permanent record:*

1. *Gas life - A bed of carbon 2 in. deep and 2 in. in diameter is tested for iodine collection at a velocity of 40 ft per minute (air at standard conditions). The iodine concentration upstream of the bed is 1000 mg/m^3 and the penetration does not exceed 1.0 percent for a period of no less than 12 hours.*

2. *Wash test - 250 ml of demineralized water is brought to a minimum boil. Twenty five grams of impregnated carbon is added to the demineralized water and the minimum boil is maintained for 1 minute. After 1 minute the water is decanted from the carbon and analyzed for the impregnate. With a knowledge of initial impregnate loading in the carbon and quantity of impregnate removed by the boiling water, results are reported as percentages of impregnate retained.*

3. *Ignition temperature - A sample from each lot of carbon is tested for ignition temperature in accordance with the procedure described in USNRC Report DP-1075, "High Temperature Adsorbents for Iodine," by R.C. Milhans.*

4. *Carbon tetrachloride test - Samples of carbon are tested for carbon tetrachloride adsorption capacity. Testing follows the procedures described in paragraph 6.2 of Military Specification MIL-C-17605.*

Systems acceptance tests - Deenergized and energized tests demonstrate the proper mounting of components, proper hookup of circuits and connection, setting of instrumentation and operation of interlocks. Equipment and system performance are monitored and rated.

For the penetration room filtration system, all ducting is given a pneumatic pressure test prior to the installation of the filter elements to assure leak-tight construction. Dimensional tolerances on filter assemblies and frame assemblies are checked to ensure that suitable gasket compression is uniformly achieved on the filter sealing faces.

A test program is performed after construction tests are completed to demonstrate the following:

A. *Proper actuation of control circuitry in both modes.*

B. *Proper flow path alignment in both modes.*

C. *Leaktightness of each filter assembly.*

D. *Verification that a negative pressure is maintained in the spent fuel area with the penetration room filtration system operating in the fuel handling area.*

The following tests are performed prior to installation of the filter elements and charcoal bed. A test assembly is installed to simulate filter pressure drop.

A. *Simulate an actuation signal and observe the performance of the system in the LOCA mode.*

B. *In the LOCA mode, measure the discharge flow from the exhaust fan. At steady state conditions with the penetration room sealed, this corresponds to the penetration room leak rate.*

C. *In the LOCA mode, verify that the recirculation fan recirculation valve opens on receipt of a differential pressure signal from two out of three differential pressure instruments between the penetration rooms and pressure in the filtration system equipment room.*

D. With the system operating, verify circulation of air within the penetration rooms in the LOCA mode.

E. Install the roughing filter and high-efficiency filter. With the systems operating, test leaktightness and performance, using DOP smoke of 0.3 micron mass median diameter. Penetration should not exceed 0.1 percent.

F. Install the charcoal beds. With the system operating, test the performance, using Freon 112. The test is performed using similar portable equipment described in USNRC Report ORNL-NSIC-65, 1970, by C. A. Burchsted and A. B. Fuller, entitled "Design, Construction, and Testing of High Efficiency Filtration Systems for Nuclear Application" (paragraph 7.5.1, pages 7.8 - 7.9). The testing procedure is in accordance with a paper by D. R. Muhlbaier, "Standardized Non-Destructive Test of Carbon Beds for Reactor Containment Applications," DP-1082, July 1967. Test results must demonstrate removal of 99.5 percent of the Freon 112. The pressure drop is also measured.

G. Simulate a spent fuel pool high radiation signal and observe system performance in the fuel handling mode.

H. Simulate a spent fuel pool low differential pressure and observe system performance in the fuel handling mode.

I. With the system operating in the fuel handling mode, verify that there is a vacuum in the spent fuel pool.

In addition, all instruments are calibrated, alarms, controls, and interlocks checked, and each remotely operated valve is individually stroked to determine its operability and correct performance of indicating lights.

The inleakage characteristics of the penetration boundary are determined by means of a flowmeter in the supply ducting to the penetration room filtration system filters and a vacuum gauge in the penetration room. With all normally operating ventilation systems in the auxiliary building secured, the internal pressure in the penetration rooms and the exhaust air flow provides the data necessary to ascertain the leaktightness of the joints, partitions, and seals.]

Periodic tests following acceptance - These tests demonstrate that the system performance capability is in accordance with the Technical Specifications. Operability of the penetration room filtration system is checked by periodically starting the fans and exercising the valves. For purposes of periodically testing the retentive capability of the charcoal filter, test canisters are placed in the filter housing in locations which allow the canisters to be subjected to the same air currents as the beds. These are periodically removed and tested at the charcoal supplier's laboratory.

Provision is made for periodic inspection and testing of the penetration room filtration system. Methods of testing the system are per guidance from ASME N510-1989. Reference FSAR table 9.4-18 for conformance positions to ASME N510-1989. Inspection of the filters for gasket deterioration is performed periodically. Routine testing of the filters and adsorbers can be performed during scheduled refuelings. Permanently installed inspection and sampling ports are provided and adequate space is provided adjacent to the filter housings for test equipment.

Testing of the inleakage characteristics of the penetration rooms is performed following major modifications or repair to the penetration rooms' boundary.

6.2.3.4.3 Containment Preaccess Filtration and Purge Systems

The radiation monitors and actuation circuits associated with isolation of the containment purge supply and discharge in the event of high discharge activity levels are tested and inspected in accordance with the provisions of IEEE Standard 279-1971. Periodic testing of the radiation monitors and actuation circuits is discussed in the Technical Specifications. The testing and inspection of the isolation valves associated with this function are discussed in paragraph 6.2.4.4. The remaining portions of the containment preaccess filtration system are not required to reduce the radiological consequences of an accident. The portion of the purge exhaust up to the discharge of the filter is Seismic Category I to assure the pressure boundary integrity of the penetration room filtration system, containment isolation, and radiation detection.

6.2.3.4.4 Containment Ventilation Systems

The containment ventilation systems are not required to reduce the radiological consequences of an accident.

6.2.3.5 Instrumentation Requirements

6.2.3.5.1 ECCS Recirculation Sump pH Control System

The system is passive and requires no instrumentation.

6.2.3.5.2 Penetration Room Filtration System

When aligned in the fuel handling mode, both trains will be started automatically by an actuation signal from the fuel handling area, initiated by either high radiation or low flow in the spent fuel pool exhaust system. When aligned in the normal operating mode, both trains of the penetration room filtration system are automatically started upon receipt of a containment phase "B" isolation actuation signal.

The following is displayed in the control room:

A. Position indication of all fan discharge valves and the recirculation line valves.

B. Temperature indication of the charcoal filter beds. High temperature is annunciated.

C. Differential pressure across each filter train.

D. Differential pressure across each fan.

E. Fan status for all fans.

F. Radiation level of exhaust air.

G. Differential pressure between the penetration rooms and the PRF equipment room. High penetration room pressure is annunciated.

H. Penetration room temperature.

In addition, all alarms are annunciated in the control room.

The actuating circuits and instrumentation for the penetration room filtration system will meet the criteria for a protection system as set forth in IEEE Standard, Criteria for Protection Systems for Nuclear Power Generating Stations, IEEE Standard 279-1971.

6.2.3.5.3 Containment Preaccess Filtration and Purge Systems

The containment minipurge system operates during power operation to continuously purge the containment atmosphere. Particulate and gas monitor indications of the purge exhaust activity levels are used to determine routine releases from the containment. Radiation monitors in the purge exhaust duct are designed to IEEE Standard 279-1971.

During power operation, the containment particulate and gas monitor indications help determine the desirability of using the preaccess filtration system in addition to the minipurge system for preaccess cleanup.

The full containment purge system operates only during reactor shutdown. Releases from the plant vent are continuously monitored during this period with a radioactive particulate and gas monitor.

6.2.3.5.4 Containment Ventilation Systems

Instrumentation applications associated with post-accident operation of the containment air coolers are discussed in paragraph 6.2.2.5.2. The remaining aspects of the operation of the containment ventilation systems are not safety related and are manually actuated. During normal operation, temperature indications verify the proper operation of the containment air cooling system, the CRDM cooling system, and the reactor vessel support cooling system. The refueling water surface ventilation system is used only during shutdown.

6.2.3.6 <u>Materials</u>

Materials information regarding paints used within containment, including type and manufacturer's designation as well as surface areas covered, are provided in table 6.2-37. Materials considerations are further described below:

A. Decomposition Products

Materials used in or on ESFs have been chosen so that the decomposition products of the materials will not interfere with safe operation of engineered safety features.

B. Materials Compatibility

Spray pumps initially provide borated water from the RWST with an approximate pH of 4.5. After dissolving the trisodium phosphate compound in the baskets, an equilibrium sump pH between 7.0 and 10.5 is attained. Studies have indicated no adverse effects on austenitic stainless steel stress corrosion crack initiation as a result of this short duration exposure to low pH spray. Sprays will be maintained after the equilibrium sump pH is reached to wash the low-pH solution from the containment materials. The carbon steel surfaces of the trisodium phosphate baskets are coated in accordance with the guidance in ANSI 101.2 (1972), "Protective Coatings (Paints) for Light Water Reactor Containment Facilities."

6.2.4 CONTAINMENT ISOLATION SYSTEM

6.2.4.1 Design Bases

The containment isolation system is in conformance with the NRC acceptance criteria contained in General Design Criteria 54, 55, 56, and 57 and Regulatory Guide 1.11. The general design basis governing isolation valve requirements is as follows:

Leakage through all fluid penetrations not serving accident consequence limiting systems is minimized by a double barrier so that no single credible failure or malfunction of an active component results in loss of isolation or intolerable leakage. The installed double barriers take the form of closed piping systems, both inside and outside the containment, and various types of isolation valves.

Containment isolation in nonessential process lines occurs coincident with SI actuation signal. Valves which isolate penetrations that are directly open to the containment, such as the purge valves and sump drain valves, are included in this group. Isolation of valves in essential process lines, such as reactor coolant pump cooling, occurs coincident with containment spray actuation signal. The closure signal for each containment isolation valve is shown in table 6.2-31. The closure signal for each SG isolation valve is shown in table 6.2-32. The conditions required for generation of containment isolation signals are outlined in section 7.3.

In accordance with GDC 54, fluid lines penetrating the containment are provided with isolation capability as follows:

Type I Each line connecting directly to the RCSs has two containment isolation valves. One valve is located inside the containment and is either an automatic valve, locked closed, or a check valve depending on the direction of normal flow. The second valve is located outside the containment and is either an automatic valve or is locked closed.

Type II — Each line connecting directly to the containment atmosphere has two containment isolation valves. One valve is located inside the containment and is either an automatic valve, is locked closed, or is a check valve, depending on the direction of normal flow. The second valve is located outside the containment and is either an automatic valve or is locked closed.

Type III — Each line that is not directly connected to the RCS nor is open to the containment atmosphere has at least one containment isolation valve located outside the containment. This valve is either an automatic valve or is locked closed, or is capable of remote manual operation.

The above isolation valve arrangements have been established to conform with GDC 55, 56, and 57. Fluid instrument lines have been classified in accordance with the design basis given above. The specific classification of each isolation valve arrangement is given in table 6.2-31. The classification of each SG isolation valve is shown in table 6.2-32. Table 6.2-31 (sheet 6) delineates special valve arrangements for meeting the General Design Criteria.

Simple check valves are not used as isolation valves outside containment.

6.2.4.2 <u>System Design</u>

Tables 6.2-31 and 6.2-32 list each fluid penetration and its isolation valves.

In addition to the containment isolation signals shown in tables 6.2-31 and 6.2-32, all automatic valves are provided with hand switches in the main control room for remote manual operation. All valves outside the containment are provided with means for manual closure.

Inside the containment, containment isolation valves are located outside the missile barrier. Outside containment, containment isolation valves are located in the penetration rooms as close as practical to the containment wall. (Any exceptions to this are specified in table 6.2-31.)

The valve closure times for all containment isolation valves listed in table 6.2-31 are shown in the table. Motor operators for valves installed within the containment have been designed and qualified for the post-LOCA environment as described in subsection 3.1.1.

Each containment penetration is part of a particular system which is listed in tables 6.2-31 and 6.2-32. The design conditions and applicable codes for the containment and SG isolation valves are described in their respective system sections.

The radiation dosage design requirements for the containment isolation valves are given in table 3.11-1.

The piping and valves forming the isolation system associated with each penetration are designed as Seismic Category I and have been analyzed for the dynamic effects of the SSE in addition to the normal operating pressure and temperature conditions. The main steam piping system has also been analyzed for the case in which one of the isolation valves is inadvertently closed when the line is at full flow; this condition does not adversely affect the integrity of the isolation system.

6.2.4.3 Design Evaluation

The containment isolation system is designed for leaktightness and reliability of operation. The isolation scheme and valve types for each penetration have been selected in consideration of normal plant operating requirements as well as the isolation function.

Where two electric motor-operated or solenoid-operated valves are provided for isolation of a Type I or Type II line, these valves are actuated from separate power supplies. All solenoid actuated valves will close if electrical power to the solenoid is lost. Where air-operated valves are provided as containment isolation valves, loss of instrument air pressure from the valve operator closes the valve.

The motor-operated valves listed in table 6.2-31 will remain in the as-is position upon power failure. The use of motor operated or manual valves which fail as is upon loss of actuating power in lines penetrating the containment are based upon the consideration of what valve position ensures the greatest plant safety. Furthermore, each of these valves that fail as is are provided with redundant backup valves to ensure that no single failure will prevent the system as a whole from performing its isolation function, (e.g., a check valve inside the containment and motor-operated valve outside the containment or two motor-operated valves in series, each powered from a separate safeguards bus).

All piping and valves relied upon to perform a containment function are classified as Safety Class 2a or better. No single failure, active or passive, will prevent the system as a whole from performing its isolation function.

6.2.4.4 Tests and Inspections

All isolation valves were shop tested for leakage and reliability of operation by the manufacturer. Gate, globe, and check valve leakage did not exceed 3 cm^3/in. of nominal valve size. Diaphragm valves were tested to zero leakage requirements. The butterfly valves for the containment purge supply and return penetration were shop tested to zero leakage in accordance with MSS-SP-67 requirements for Type I valves.

Each valve may be tested after installation to ensure its operability and leaktightness in the piping system. All piping systems penetrating the containment have been provided with test connections and test vents, or have other provisions to allow periodic leak testing as required by Appendix J to 10 CFR 50. Throughout plant life, the containment isolation valves can be tested periodically. Valves which cannot be tested during plant operation are tested during scheduled shutdowns.

Table 6.2-38 lists the containment penetrations and the type of tests required.

Type C testing of the residual heat removal sump suctions and the containment spray sump suctions is not required. The justification for this is that these valves are required to be open at some time after the accident to effect immediate and long-term core cooling. Type C testing of the residual heat removal suctions from the reactor coolant system hot legs shares a common suction line with the residual heat removal sump suctions. These systems are closed systems outside containment, and are water filled during normal operation and following a LOCA. These systems are designed and constructed to ASME III, Class 2 and Seismic Category I requirements and, as such, they do not constitute a potential containment atmosphere leak path during or following a LOCA with a single-active failure of a system component. Should the valves leak slightly when closed, the fluid seal within the pipe or the closed piping system outside/inside containment would preclude release of containment atmosphere to the environs.

Table 6.2-39 lists the containment isolation valves, their location with respect to the containment, appropriate referenced drawings and/or FSAR figures, and the valve identification number. Table 6.2-40 lists the same information for the SG isolation valves. For the valves listed in table 6.2-39 and associated with table 6.2-38, which must be Type C tested to comply with Appendix J to 10 CFR 50, the fluid to be used will be in accordance with the Containment Leakage Rate Testing Program.

Testing requirements associated with containment isolation valves are discussed in the Technical Specifications and the Technical Requirements Manual.

6.2.4.5 Materials

Material compatibility is discussed in appendix 6A.

6.2.5 COMBUSTIBLE GAS CONTROL IN CONTAINMENT

Following a DBA, hydrogen gas may be generated inside the containment by reactions such as zirconium metal with water, corrosion of materials of construction, and radiolysis of aqueous solution in the core and sump. To ensure that the hydrogen concentration is maintained at a safe level, redundant recombiners are provided along with a backup post-accident venting system. A mixing system is provided to maintain accumulation below the flammability limit and a sampling system is provided to monitor the containment atmosphere.

The combustible gas control system is in conformance with the NRC acceptance criteria contained in General Design Criteria 41, 42, and 43 and Regulatory Guide 1.7. Post-LOCA hydrogen concentration in the containment would not reach the lower flammability limits of 4.0 volume percent, assuming that one electric hydrogen recombiner is started 1 day after the start of LOCA (see figure 6.2-94). However, no credit is taken for venting of the containment atmosphere.

6.2.5.1 Design Bases

6.2.5.1.1 Electric Hydrogen Recombiners

The following design bases apply to the electric hydrogen recombiners:

A. The recombiners are designed to sustain all normal loads as well as accident loads including seismic loads and temperature and pressure transients from a design basis LOCA.

B. The recombiners are protected from damage by missiles or jet impingement from broken lines.

C. The recombiners are located away from high velocity air streams such as could emanate from fan cooler exhaust ports, or they are protected from direct impingement of such high velocity air streams by suitable barriers such as walls or floors.

D. The recombiners are designed for a lifetime of 40 years, consistent with that of the plant.[a]

E. All materials used in the recombiners are selected to be compatible with the environmental conditions inside the reactor containment during normal operation or during accident conditions.

F. The recombiners are powered from separate electrical power trains and can be powered from an emergency power source if necessary.

G. Process capacity is such that the containment hydrogen concentration will not exceed 4 volume percent based on the NRC TID release model as indicated in NRC Regulatory Guide 1.7: "Control of Combustible Gas Concentrations in Containment Following a Loss-of-Coolant Accident."

---

a. The renewed operating licenses authorize an additional 20-year period of extended operation for both FNP units, resulting in a plant operating life of 60 years. In accordance with 10 CFR Part 54, appropriate aging management programs and activities have been initiated to manage the detrimental effects of aging to maintain functionality during the period of extended operation (see chapter 18).

C. 6.2.5.1.2 Post-Accident Venting System

The following design bases apply to the post-accident venting system:

A. The venting system is a Seismic Category I system, and is designed to sustain all normal loads as well as temperature and pressure transients from the design basis LOCA.

B. The venting system is protected from damage by missiles or jet impingement from broken pipes.

C. The venting inlet ducts are located in a well ventilated part of the containment.

D. The system is designed for a lifetime consistent with that of the reactor plant.

E. All materials in the venting system are selected to be compatible with the environmental conditions inside the reactor containment during normal operation or during accident conditions.

F. Process capacity such that the containment hydrogen concentration will not exceed 4 volume percent based on the NRC TID release model as indicated in NRC Regulatory Guide 1.7, "Control of Combustible Gas Concentrations in Containment Following a Loss-of-Coolant Accident."

6.2.5.1.3 Post-Accident Combustible Gas Sampling System

The following design bases apply to the post-accident combustible gas sampling system:

A. The sampling system is a Seismic Category I system and is designed to sustain all normal loads as well as temperature and pressure transients from the design basis LOCA.

B. The sampling system is protected from damage by missiles or jet impingement from broken pipes.

C. The sampling system will provide samples to determine representative hydrogen concentrations inside the containment from at least two independent locations. Hydrogen analyzers will be provided to continuously monitor the hydrogen concentration of the containment atmosphere following a LOCA. The system will have the capability to provide grab samples.

D. The combustible gas monitoring system is described in subsection 7.6.4.

E. All materials in the sampling system are selected to be compatible with the environmental conditions inside the reactor containment during normal operation or during accident conditions.

6.2.5.1.4 Post-Accident Mixing System

The following design bases apply to the post-accident mixing system:

A. The mixing system is a Seismic Category I system and is designed to sustain all normal loads as well as temperature and pressure transients from a design basis LOCA.

B. The mixing system components are protected from damage by missiles or jet impingement from broken pipes.

C. The system is designed for maintenance free operation for a period of 100 days following a LOCA.

D. All materials in the post-accident mixing system are selected to be compatible with the environmental conditions inside the containment during normal operation or after an accident.

E. The active components are redundant, and the electric cables and instrument lines are separated so that no single failure can incapacitate the entire system.

F. The active components are powered from separate power supply trains and can be powered from an emergency power source if necessary.

6.2.5.2 System Design

The total system for control of combustible H_2 concentrations in the containment following a LOCA consists of a sampling system that provides containment atmosphere samples, electric hydrogen recombiners as the primary means of reducing H_2 containment concentrations, a venting system which is used as a backup system to the recombiners, and a mixing system to maintain concentrations in the lower containment compartments below the lower flammability limit.

6.2.5.2.1 Electric Hydrogen Recombiners

Redundant electrical recombiners, as shown on figure 6.2-90 and drawings D-175019 and D-205019, are located inside the containment upper compartment for controlling hydrogen concentrations following a DBA. The recombiners meet safety features requirements and the controls are located outside the containment. The recombiner units are located in the containment such that they process a flow of containment air containing hydrogen at a concentration which is generally typical of the average concentration throughout the containment.

To meet the requirements for redundancy and independence, two recombiners are provided. Each recombiner is provided with a separate power panel and control panel and each is powered from a separate bus. Each can be switched to the emergency power source if necessary. There is no interdependency between this system and the other ESFs.

Containment atmosphere is circulated by natural convection through a recombiner where hydrogen is removed by heating to a temperature sufficient to cause recombination with the containment oxygen.

The recombiner consists of a thermally insulated vertical metal duct with electric resistance metal sheathed heaters provided to heat a continuous flow of containment 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 water coming from the containment spray system. The recombiner consists of an inlet preheater section, a heater recombination section, and a mixing chamber.

The unit is manufactured primarily of corrosion resistant, high temperature material for major structural components, except for the base which is carbon steel. The electric hydrogen recombiner uses conventional type electric resistance heaters sheathed with Incoloy-800, which is an excellent corrosion resistant material for this service. These heaters are designed to operate with sheath temperatures equal to those used in certain commercial heaters; however, these recombiner heaters operate at significantly lower power densities than is commercial practice.

Air is drawn into the recombiner by natural convection and passes first through the preheater section. This section consists of a shroud placed around the central 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 an orifice plate and then enters the electric heater section where it is heated to between 1150°F - 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 five assemblies of electric heaters stacked vertically. Each assembly contains individual heating elements. Operation of the unit is virtually unaffected should a few individual heating elements fail to function properly. Table 6.2-33 gives the recombiner design parameters.

The recombiner, power supply panel, and control panel are shown schematically in figure 6.2-91. The power panel for the recombiner is located in the auxiliary building, and contains an isolation transformer plus an SCR controller to regulate power into the recombiner. This equipment is not exposed to the post-LOCA environment.

The control panel is located in the control room. 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 controller setting will be set at the control panel to maintain the recombiner air temperature between 1150°F and 1400°F. This power setting and recombiner air temperature range will cover variations in containment temperature, pressure, and hydrogen concentration in the post-LOCA environment. For an equipment test and periodic checkout, a thermocouple readout instrument is also provided in the control panel for monitoring temperatures in the recombiner.

Results of testing a prototype of the electric hydrogen recombiner are given in WCAP-7820, Supplement 3, "Electrical Hydrogen Recombiner for Water Reactor Containments," March 1974. A power setting of approximately 48.9 kW for the prototype recombiners was determined to be satisfactory for PWR containments. However, production recombiners and power supplies have been designed to achieve up to 75 kW to provide operational flexibility for containments and design conditions which differ from the prototype testing.

6.2.5.2.2 Post-Accident Venting System

The post-accident containment venting system, as shown on drawings D-175019 and D-205019, consists of a supply line through which hydrogen free air can be admitted to the containment, and an exhaust line through which hydrogen bearing gases may be vented from the containment. The gases are filtered through HEPA and charcoal filters to limit discharge of particulates and iodine. Piping and valving in the exhaust line are Safety Class 2A starting inside the containment, proceeding up to and including the venting filtration unit outside the containment. Equipment and piping beyond the filter unit are Safety Class 2B. Design parameters are given in table 6.2-34.

The supply line in the containment is located in a missile protected area and is terminated so as to prevent either spray or sump water from entering the pipe. The exhaust line in the containment is located in a missile protected area and is terminated in well ventilated areas in a manner which prevents either spray or sump water from entering the pipe.

The containment hydrogen concentration and hydrogen generation rate are determined every 24 h after the initiation of the accident. When the projected hydrogen concentration for the next 24-h period has increased to 3.5 volume percent, containment pressurization is initiated. Using plant air compressors, hydrogen free air will be pumped into the containment until the required containment pressure is reached. The air supply will then be stopped and the supply line isolated. The addition of air to pressurize the containment will lower the hydrogen concentration and permit the containment to be isolated until sampling indicates that the next-day projected hydrogen concentration will be above 3.5 volume percent when venting is necessary.

Venting is then started by opening the motor-operated valves in the line and adjusting the manually controlled flow control valve to obtain the required flow. The flowrate required to maintain the hydrogen concentration less than 4 volume percent of the containment volume by venting the containment is determined from the containment hydrogen concentration and the hydrogen generation rate. When the operation is complete, the vent line is isolated by the valve outside the containment. The motor-operated valve inside the containment will also be closed. This process of containment pressurization followed by venting may be repeated as necessary to maintain the hydrogen concentration below 4 volume percent.

6.2.5.2.3 Post-Accident Combustible Gas Sampling System

The post-accident sampling system is shown schematically on drawings D-175019 and D-205019 and comprises two independent trains consisting of sample inlet lines, a removable sample container, a hydrogen analyzer, and a return line routed to the containment. Piping and equipment from the innermost containment isolation valves outward are Safety Class 2A. Piping and equipment inside the innermost containment isolation valves are Safety Class 2B. Design parameters are given in table 6.2-35.

The supply lines in the containment are located in missile protected areas and are terminated so as to prevent either spray or sump water from entering the pipe. The sample exhaust line in the containment is located in a missile protected area and is terminated in a well ventilated area in a manner which prevents either spray or sump water from obstructing the pipe.

A containment air sample can be taken from either of four independent locations within the containment. Sample point locations are as noted on drawings D-175019 and D-205019. Each of the sample lines is routed independently inside the containment up to and including motor operated globe valves, which serve as containment isolation barriers. Immediately downstream of these valves the four lines are joined into two and are routed through two separate containment penetrations, one penetration for each train. Outside the containment, the sample flows through a motor-operated isolation valve, through a line which includes pressure indication, and through a hydrogen analyzer. Upstream of the analyzer there is a removable sample vessel, located in parallel with the sample discharge line, which can be used to provide grab samples. The discharge line passes through a motor-operated isolation valve located outside the containment. The line is routed through a containment penetration and discharges into the containment atmosphere through a motor-operated globe valve serving as a containment isolation valve.

6.2.5.2.4 Post-Accident Containment Mixing System

Drawings D-175019 and D-205019 include the P&ID for the post-LOCA containment mixing system.

Four fans are provided to circulate containment atmosphere among the lower containment compartments, the upper containment volume above the operating floor, and the reactor cavity. These fans are placed on the hatch covers above the three reactor coolant pumps.

Of these four fans, two are powered from each of two separate and redundant emergency power trains. These fans take suction from the equipment compartments and the lower plenum-like volume over the containment sump and discharge upward, thereby establishing flow downward around the periphery of the containment, through the lower containment volume, and upward out of the lower containment volume through the SG compartments. A labyrinth at each fan suction will protect the fans and motors from missiles that might be generated within the secondary missile shield inside containment.

Two additional fans ventilate the reactor cavity to ensure that this volume is available for the dilution of containment hydrogen and to maintain hydrogen concentrations in this volume in equilibrium with that of the remainder of the containment. These fans discharge into the reactor cavity through a circular header embedded in the cavity wall at an elevation approximately coincident with that of the lower reactor vessel head. This flow is discharged from the cavity upward around the reactor vessel and outward through the incore instrument chase. The two fans, one each of which will be powered from one of two separate and redundant emergency power trains, will take suction from the periphery of the containment just below the operating floor. These fans and motors are located outside the secondary missile shield inside the containment.

The post-LOCA mixing fans and the reactor cavity hydrogen dilution fans are designed to Seismic Category I requirements. The design capacities of these fans are given in table 6.2-36.

The internal structures of the containment building have been designed to provide vertical cylindrical compartments around each SG which project upward from a lower plenum covering much of the containment sump. (See figures 6.2-92 and 6.2-93.) Following a LOCA, the SGs and reactor coolant piping (short term) and the sump fluid (long term) represent heat sources that establish and maintain natural convective flows upward out of the lower containment volume through the SG compartments. This, coupled with the generally lower air temperatures provided by the containment air coolers around the periphery of the containment, will establish downward flow from the upper containment volume through grating located around the periphery of the containment. This convective flow into and out of the lower containment, coupled with the mixing action of the sprays and containment air coolers above the operating floor, will complement the action of the post-LOCA mixing fans. The undersides of all floors inside the containment have been sloped to augment this natural convection. Inverted pockets that might tend to encourage stagnant accumulations of containment atmosphere have been avoided by design.

To determine the design capacities of the post-LOCA mixing fans, the containment was sectionalized into specific compartments as shown in figures 6.2-92 and 6.2-93. The lower compartment was defined as the summation of compartments 2, 3, 4, 5, 6, and 8. Figures 6.2-96 and 6.2-97 give the calculated volume percent of hydrogen in each compartment as a function of time assuming no intercompartmental mixing. Figures 6.2-98 and 6.2-83 give the hydrogen generation rates as a function of time. For each compartment, the ventilation sweep rate required to maintain the compartmental hydrogen concentration below 3.5 volume percent was calculated. The post-LOCA mixing fans were selected so that they have a design flow that exceeds the calculated required sweep rate for the lower compartments.

6.2.5.3 <u>Design Evaluation</u>

6.2.5.3.1 Electric Hydrogen Recombiners

The prediction of hydrogen generation following a LOCA is shown in figures 6.2-83 and 6.2-98, which demonstrate that the hydrogen production rate decreases with time after the accident. As discussed in paragraph 15.4.1.6.5, and as can be determined from these figures, the total hydrogen accumulation can exceed the lower flammability limit of 4 volume percent and control measures are necessary to prevent hydrogen accumulation to this limit. The electric recombiner provides the means to prevent unsafe levels of hydrogen concentration from being reached in the containment following a LOCA.

For the purpose of showing that the electric recombiner is capable of maintaining safe hydrogen concentrations, analysis was performed using the NRC Regulatory Guide No. 1.7 Model. The result is shown in figure 6.2-94. The Regulatory Guide No. 1.7 Model is based upon assuming a fission product activity release specified in TID-14844 and the values for post-accident hydrogen generation specified in this guide. The results using the Westinghouse model are also shown on this figure.

Each electric recombiner is capable of continually processing a minimum of 100 sft^3/min (standard conditions of 68°F and 1 ATM) of containment atmosphere. All of the hydrogen contained in the processed atmosphere is converted to steam, thus reducing the overall containment hydrogen concentration. The hydrogen concentration in the containment was calculated for the models described above based on a recombiner capability of processing 93 sft^3/min of containment atmosphere at modeled conditions of 32°F and 1 ATM. This calculation shows that the maximum hydrogen concentration will be much less than the lower flammability limit of 4 volume percent if the recombiner is started 1 day following the accident. Therefore, one of these units 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 is indicative of that time when the amount of hydrogen being generated is equal to the amount of hydrogen being reprocessed. Since the production rate of hydrogen decreases with increasing time following the accident, 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 containment. Thus, once the peak has been reached, the electric recombiner provides a continually increasing margin between the containment hydrogen concentration and the lower flammability limit of 4 volume percent.

The recombiners are protected from damage by missiles. The unit 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 recombiners are manually initiated devices and, although IEEE-279 is not a design requirement, they do meet the single failure, separation testability, qualification, and manual initiation criteria of IEEE-279.

6.2.5.3.2 Post-Accident Venting System

As a backup system to the electric recombiners for controlling combustible gas concentrations in the containment following a LOCA, the venting system has a process capacity such that H_2 concentration will not exceed 4 volume percent. This process capacity permits intermittent operation of the system as deemed necessary by the operator. A curve of hydrogen concentration as a function of time for the containment purge mode of operation is provided in figure 6.2-95.

The venting system is designed to be protected from missiles, to sustain all normal as well as accident loads without loss of function.

An analysis of the offsite doses due to operation of the venting system is presented in section 15.4.

6.2.5.3.3 Post-Accident Combustible Gas Sampling System

The post-accident containment sampling system is designed to obtain post-accident containment atmosphere grab samples for analysis, to monitor the hydrogen concentration in the containment atmosphere following a LOCA, and to provide a method of monitoring hydrogen recombiner operation. Samples may be obtained from any of four independent locations within the containment in the containment dome, above the containment cooler units, below hydrogen recombiner No. 1B, and in a lower compartment. Locating sample points at different elevations in all quadrants of the containment provides the capability to obtain a representative sample of the containment atmosphere.

The post-accident containment sampling system is protected from missiles and is designed to sustain all normal and accident loads without loss of function. All sample lines are terminated so as to prevent spray or sump water from entering the line. All sample lines within the containment are sloped to enhance drainage of any accumulated moisture and all potential water pocket areas are avoided by design.

6.2.5.3.4 Post-Accident Mixing System

In order to maintain compartment hydrogen accumulations below the lower flammability limit, the units are sized to limit the hydrogen accumulation to 3.5 volume percent based on NRC Regulatory Guide No. 1.7 assumptions.

The system can be operated continuously or intermittently depending on containment conditions and is designed to be protected from missiles and to sustain all normal as well as accident loads without loss of function.

6.2.5.4 Tests and Inspections

6.2.5.4.1 Electric Hydrogen Recombiners

The electric hydrogen recombiners have undergone many tests as part of the Westinghouse development program. These tests encompassed the initial analytical studies, laboratory proof of principle tests, and full scale prototype testing. The full scale prototype tests included the effects of:

- Varying hydrogen concentrations.

- Alkaline spray atmosphere.
- Steam effects.
- Convection currents.

A detailed discussion of these tests is given in WCAP-7820, Supplement 1.

Post-operational tests and inspections are performed in accordance with the Technical Specifications and the Technical Requirements Manual. Testing will be performed by inputting power to the recombiner and observing the thermocouple readout instruments on the control panel. Inspections will be performed to verify the integrity of all electrical connections to assure the capability of the recombiner to perform its function.

6.2.5.4.2 Post-Accident Venting and Sampling Systems

Test connections are provided to allow testing of the systems. Testing will periodically demonstrate that the systems will perform their intended functions.

Periodic testing of the post-accident venting system is based on testing guidance from ASME N510-1989. Field surveillance testing, as provided in ASME N510-1989, is based on nuclear filtration units designed and installed to the requirements of ASME N509-1989. Since the post-accident venting filter units were designed to earlier standards, these filter units do not possess all of the design features required to provide for field testing as described in ASME N510-1989. Conformance to ASME N510-1989 is clarified in table 9.4-20.

6.2.5.4.3 Post-Accident Containment Mixing System

To demonstrate that this system will perform as required, tests will be performed in accordance with the Technical Specifications. Pressure instrumentation is provided to demonstrate that fan developed head has not deteriorated. Inspections will be performed to verify the integrity of all electrical connections to assure the capability of the fans to perform their function.

Post-accident mixing system fans' and motors' capability to operate during LOCA is certified by the manufacturer. The qualification test is the same as the containment air cooling units qualification test which is described under paragraph 6.2.2.4.2.

6.2.5.5 Instrumentation Requirements

The recombiners will be started manually after a LOCA, and the mixing system is started automatically upon receipt of an SI signal. The sampling system will be used in obtaining containment atmosphere samples that will indicate when the recombiners or the venting system should be actuated. This sample can be taken from any of four well ventilated locations within the containment.

The recombiners do not require any instrumentation inside the containment for proper operation after a LOCA.

6.2.5.6 Materials

6.2.5.6.1 Electric Recombiner

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 operates 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 Coatings (Paints) for Light Water Nuclear Reactor Containment Facilities."

Testing of the electric recombiner in the post-LOCA spray environments is reported in WCAP-7820 (Supplement 1).

6.2.5.6.2 Post-Accident Venting and Sampling System

The materials of construction of these systems are chosen as to their location relative to the reactor containment. All components inside containment, and up to and including the containment isolation valves outside are fabricated of stainless steel. There are no pyrolytic or radiolytic decomposition products from this material; thus, these systems have no effect on any other safety feature system.

6.2.5.6.3 Post-Accident Containment Mixing System

All components of this system are fabricated from stainless steel. There are no pyrolytic or radiolytic decomposition products of this material, and this system has no effect on any other safety feature system.

REFERENCES

1. Strikovich, M. A., et al., Atomnoya Energiya, Volume 17, No. 1, pp. 45-49, (Translation in UDE - 621.039.562.5), July 1964.

2. Parsly, L. F., Jr., "Design Considerations of Reactor Containment Spray Systems - Part VI," ORNL-TM-2412, Part 6, 1969.

3. Parsly, L. F., Jr., "Design Considerations of Reactor Containment Spray Systems - Part VII," ORNL-TM-2412, Part 7, 1970.

4. Eggleton, A. E. J., "A Theoretical Examination of Iodine Water Partition Coefficient," AERE (R) - 4887, 1967.

5. Deleted

6. Burchsted, C. A., and Fuller, A. B., Design, Construction, and Testing of High Efficiency Air Filtration Systems for Nuclear Application," ORNL-NSIC-65.

7. “Westinghouse LOCA Mass and Energy Release Model for Containment Design - March 1979 Version,” WCAP-10325-P-A, May 1983 (Proprietary), WCAP-10326-A (Nonproprietary).

8. Docket No. 50-315, “Amendment No. 126, Facility Operating License No. DPR-58 (TAC No. 7106), for D. C. Cook Nuclear Plant Unit 1,” June 9, 1989.

9. EPRI 294-2, “Mixing of Emergency Core Cooling Water with Steam: 1/3-Scale Test and Summary,” WCAP-8423, Final Report, June 1975.

10. “Westinghouse Mass and Energy Release Data for Containment Design,” WCAP-8264-P-A, Rev. 1, August 1975 (Proprietary), WCAP-8312-A, (Nonproprietary).

11. ANSI/ANS-5.1 1979, “American National Standard for Decay Heat Power in Light Water Reactors,” August 1979.

12. McAdam, W. H., Heat Transmission, McGraw-Hill, 3rd Edition, 1954, p. 172.

13. NL-91-0873, “Joseph M. Farley Nuclear Plant Leak-Before Break Approval,” 9/13/91.

14. Deleted

15. Schmitt, R. C., Binham, G. E., and Norberg, J. A., "Simulated Design Basis Accident Tests of the Carolinas Virginia Tube Reactor Containment," Idaho Nuclear Corp., NTIS TID-4500 1970.

16. "Performance and Sizing of Dry Pressure Containment," BN-TOP-3, Rev. 3, Bechtel Power Corporation, August 1975.

17. Chapman, A. J., "Heat Transfer," MacMillan, 1967.

18. EPRI/NAI 8907-02, Rev. 6, "GOTHIC Containment Analysis Package User Manual," Version 5.

19. Rubin, K., Grover, J. L., Henninger, W. A., and Miller, T. A., "Methodology for Elimination of the Containment Spray Additive," WCAP-11611, Rev. 0, March 1988.

20. U. S. Nuclear Regulatory Commission, "Technological Bases for Models of Spray Washout of Airborne Contaminants in Containment Vessels," NUREG/CR-0009, 10/78.

21. U. S. Atomic Energy Commission, "A Review of Mathematical Models for Predicting Spray Removal of Fission Products in Reactor Containment Vessels," WASH-1329, June 15, 1974.

22. EPRI/NAI 8907-06, Rev. 5, "GOTHIC Containment Analysis Package Technical Manual," Version 5.

23. "Mass and Energy Releases Following a Steam Line Rupture," WCAP-8822 (Proprietary), WCAP-8860 (Nonproprietary), September 1976; "Supplement 1 - Calculations of Steam Superheat in Mass/Energy Releases Following a Steamline Rupture," WCAP-8822-S1-P-A (Proprietary), WCAP-8860-S1-A (Nonproprietary), September 1986; "Supplement 2 - Impact of Steam Superheat in Mass/Energy Releases Following a Steamline Rupture for Dry and Sub-Atmospheric Containment Designs," WCAP-8822-S2-P-A (Proprietary), WCAP-8860-S2-A (Nonproprietary), September 1986.

24. Burnett, T. W. T., et al, "LOFTRAN Code Description," WCAP-7907-P-A (Proprietary), WCAP-7907-A (Non-Proprietary), April 1984.

25. Huegel, D. S., et al., "RETRAN-02 Modeling and Qualification for Westinghouse Pressurized Water Reactor Non-LOCA Safety Analyses," WCAP-14882 (Proprietary), June 1997.

26. Safety Evaluation for Amendment No. 137 to NPF-2 and Amendment No. 129 to NPF-8, April 29, 1998 (Power Uprate).

27. Letter from Herbert N. Berkow, Director (NRC) to James A. Gresham (Westinghouse), "Acceptance of Clarifications of Tropical Report WCAP-10325-P-A, 'Westinghouse LOCA Mass and Energy Release Model for Containment Design -March 1979 Version' (TAC No. MC7980)," October 18, 2005 [ADAMS Accession No. ML052660242].

TABLE 6.2-1

PRINCIPAL CONTAINMENT DESIGN PARAMETERS

Characteristics	Data
Containment design pressure (psig)	54
Containment design temperature (°F)	280
Internal dimensions	
Cylindrical wall diameter (ft)	130
Cylindrical wall height (ft)	139
Curved dome height (ft)	43.5
Volumes	
Gross internal volume (ft^3)	2.35×10^6
Net free internal volume (ft^3)	2.0×10^6
Containment design leak rate	
First 24 h, percent of containment free volume per day	0.15
After first day, percent per day	0.075

TABLE 6.2-2 (SHEET 1 OF 5)

HEAT SINK GEOMETRIC DATA[(a)]

Heat Sink 1 - Containment Cylinder and Dome **74,908 ft^2**

Exposure

1. Containment Atmosphere
2. Outside Atmosphere

Material	Thickness (in.)
Paint/Primer	0.0084
Carbon Steel	0.25
Air Gap	0.00204
Concrete	45.0

Heat Sink 2 - Penetration Plates & Liner Stiffners **3,802 ft^2**

Exposure

1. Containment Atmosphere
2. Outside Atmosphere

Material	Thickness (in.)
Paint/Primer	0.0084
Carbon Steel	0.51
Air Gap	0.00204
Concrete	45.0

Heat Sink 3 - Unlined Concrete (excluding reactor support) **60,375 ft^2**

Exposure

1. Containment Atmosphere
2. Insulated

Material	Thickness (in.)
Paint	0.019
Surfacer	0.125
Concrete	18.0

TABLE 6.2-2 (SHEET 2 OF 5)

Heat Sink 4 - Galvanized Steel (excluding cable trays) **43,320 ft^2**

Exposure

1. Containment Atmosphere
2. Insulated

Material	Thickness (in.)
Zinc	0.0034
Carbon Steel	0.07

Heat Sink 5 - Painted Carbon Steel ≤ 0.5-in. Thickness **95,210 ft^2**

Exposure

1. Containment Atmosphere
2. Insulated

Material	Thickness (in.)
Paint/Primer	0.0084
Carbon Steel	0.18

Heat Sink 6 - Painted Carbon Steel ≤ 1.0-in. Thickness **25,681 ft^2**

Exposure

1. Containment Atmosphere
2. Insulated

Material	Thickness (in.)
Paint/Primer	0.0084
Carbon Steel	0.59

TABLE 6.2-2 (SHEET 3 OF 5)

Heat Sink 7 - Painted Carbon Steel ≤ 2.0-in. Thickness **8,802 ft^2**

Exposure

1. Containment Atmosphere
2. Insulated

Material	Thickness (in.)
Paint/Primer	0.0084
Carbon Steel	1.35

Heat Sink 8 - Painted Carbon Steel ≥ 2.0-in. Thickness **3,353 ft^2**

Exposure

1. Containment Atmosphere
2. Insulated

Material	Thickness (in.)
Paint/Primer	0.0084
Carbon Steel	3.59

Heat Sink 9 - Containment Floor **5,402 ft^2**

Exposure

1. Containment Atmosphere
2. Insulated

Material	Thickness (in.)
Concrete	108.0

TABLE 6.2-2 (SHEET 4 OF 5)

Heat Sink 10 - Refuel Canal Liner **7,894 ft2**

Exposure

1. Containment Atmosphere
2. Insulated

Material	Thickness (in.)
Stainless Steel	0.25
Air Gap	0.00204
Concrete	18.0

Heat Sink 11 - Unpainted Stainless Steel **10,116 ft^2**

Exposure

1. Containment Atmosphere
2. Insulated

Material	Thickness (in.)
Stainless Steel	0.12

Heat Sink 12 - Galvanized Steel Cable Trays **22,164 ft^2**

Exposure

1. Containment Atmosphere
2. Insulated

Material	Thickness (in.)
Zinc	0.0034
Carbon Steel	0.05

TABLE 6.2-2 (SHEET 5 OF 5)

Heat Sink 13 - Reactor Support	**2,182 ft^2**

Exposure

1. Containment Atmosphere - A 150°F source to account for the higher reactor cavity operating temperature
2. Insulated

Material	Thickness (in.)
Paint	0.019
Surfacer	0.125
Concrete	86.0

(a) An evaluation for these parameters was performed as described in Section 6.2.1.3.13.

TABLE 6.2-3

INITIAL CONDITIONS FOR PRESSURE ANALYSIS

Characteristics	Data
Containment System	
Pressure (psia)	13.2 - 17.7
Relative humidity (percent)	50
Inside temperature (°F)	120[a]
Outside temperature (°F)	95
Refueling water storage tank water temperature (°F)	110
Accumulator tank water temperature (°F)	120
Service water temperature (°F)	95[b]
Stored Water	
Refueling water storage tank (gal)	471,000[c]
Three accumulators (ft^3)	240

a. 120°F is the Technical Specifications limit, 127°F was used in the analysis.

b. Service water temperature of 97.3°F was used in the analysis.

c. A refueling water storage tank delivery capacity of 390,000 gallons was used in the analysis.

TABLE 6.2-4

HEAT SINK THERMODYNAMIC DATA

MATERIAL PROPERTIES[a]

Material	Density (lbm/ft^3)	Thermal Conductivity Btu/h-ft-°F)	Heat Capacity (Btu/lbm-°F)
Paint (Ameron 66)	162.3	0.50/0.25[b]	0.29
Paint (Ameron 90, 90HS)	160.8	0.38/0.25[b]	0.31
Primer (Dimetcote 6)	196.8	0.63	0.11
Carbon steel	489.0	29.6	0.1096
Concrete	144.0	1.0	0.2292
Surfacer (Ameron 110 AA, 3366/3367)	121.2	0.39	0.23
Zinc	446.0	62.2	0.0942
Stainless steel	488.0	8.6	0.1232
Air	0.069	0.017	0.2095

HEAT TRANSFER COEFFICIENTS

Surface	Value
Sink surfaces exposed to containment atmosphere	Modified Tagami (LOCA Blowdown) UCHIDA (LOCA Reflood & MSLB)
Sump liquid to containment atmosphere	Conduction
Containment sump and floor to sump liquid	Conduction
Sink surfaces exposed to outside atmosphere	2.0 $Btu/h\text{-}ft^2\text{-}°F$

---

a. An evaluation was performed for these parameters as described in Section 6.2.1.3.13.

b. Value for Paint/Primer in combination.

TABLE 6.2-5 (SHEET 1 OF 2)

ENGINEERED SAFETY FEATURES PERFORMANCE FOR CONTAINMENT PRESSURE TRANSIENT ANALYSIS

System	Operation	Values Used for Containment Analysis Maximum ESF	Minimum ESF
Containment spray			
Water sources	Borated water from RWST or sump		
Initiation	Initiated by Containment Press. High-High-High		
Number of lines and headers		2	1
Number of pumps		2	1
Flowrate, gal/min per pump		2175	2480 (Injection) 2290 (Recirculation)
Containment air coolers			
Initiation	Initiated by SIS		
Number of units		4	1[b]
Flowrate (air side), ft^3/min per unit		40000	40000
Total design heat removal at containment design temperature, (Btu/h) per unit		80×10^6	80×10^6[a]
Service water temperature (°F)		97.3	97.3
RHR/Low pressure safety injection heat exchangers			
Type	Horizontal shell U-tube		
Cooling water supply	Component cooling water		
Number of units		2	1
Heat transfer area, ft^2 per unit		4070	3500
Overall heat transfer coefficient, Btu/h-ft^2-°F		383	383

TABLE 6.2-5 (SHEET 2 OF 2)

System	Operation	Values Used for Containment Analysis Maximum ESF	Values Used for Containment Analysis Minimum ESF
Flowrate:	Injection	3000	3000
Sump water side, gal/min per unit	Recirculation	3750	3750
Component cooling water side, gal/min per unit		4755	4755
Return water point		Primary loop	Primary loop
Passive safety injection system			
Capacity, gal each accumulator	600		
Number of accumu-lators		3	3
Pressure setpoint, psig		600	600
Active safety injection system			
Initiation	Initiated by SIS		
High pressure safety injection:			
Number of lines		3	3
Number of pumps		2	1
Flowrate, gal/min per pump		511	511
Low pressure safety injection:			
Number of lines		3	3
Number of pumps		2	1
Flowrate, gal/min per pump	Injection	3000	3000
	Recirculation	3750	3750

a. Value for 600-gal/min service water flow for paragraph 6.2.1.3.12 analysis is 31.2×10^6 at 275 °F.

b. Having fewer than 12 coils per containment cooler is acceptable provided that each cooler can adequately remove the containment analysis heat load described in note “a”.

TABLE 6.2-6

CONTAINMENT PRESSURE ANALYSIS RESULTS FOR THE SPECTRUM OF RCS BREAK SIZES[a]

	DEPSG[b] MIN ESF P_0 = 0 psig	DEPSG MIN ESF P_0 = +3 psig	**0.6** DEPSG[b] MAX ESF 4.95 ft^2	PSS[b] MAX ESF 3 ft^2	DECLG[b] MAX ESF 8.25 ft^2	DEHLG MIN ESF P_0 = +3 psig
Peak pressure (psig)	38.0	43.8	40.1	40.9	37.6	43.6
Time of peak pressure (s)	19.4	552	191.9	194.3	22.3	18.8
Peak temperature (°F)	260	263	264	265	260	264
Time of peak temperature (s)	19.4	552	191.9	194.3	22.3	18.7

a. See Table 6.2-41 for MSLB results.

b. Non-limiting cases, not reanalyzed for power uprate/steam generator replacement, maintained for historical purposes

TABLE 6.2-7

SYSTEM PARAMETERS
INITIAL CONDITIONS FOR THERMAL UPRATE

PARAMETERS	VALUE
Core Thermal Power (MWt)	2830.5
Reactor Coolant System Total Flowrate (lbm/sec)	27250.0
Vessel Outlet Temperature (°F)	619.3
Core Inlet Temperature (°F)	547.1
Vessel Average Temperature (°F)	583.2
Initial Steam Generator Steam Pressure (psia)	817
Steam Generator Design	Model 54F
Steam Generator Tube Plugging (%)	0
Initial Steam Generator Secondary Side Mass (lbm)	121826.1
Assumed Maximum Containment Backpressure (psia)	68.7
Accumulator	
Water Volume (ft^3) per accumulator	1040
N_2 Cover Gas Pressure (psia)	600
Temperature (°F)	120
Safety Injection Delay, total (sec) (from beginning of event)	30.9

Note: Core Thermal Power, RCS Total Flowrate, RCS Coolant Temperatures, and Steam Generator Secondary Side Mass include appropriate uncertainty and/or allowance.

TABLE 6.2-8

SAFETY INJECTION FLOW
MINIMUM SAFEGUARDS

RCS PRESSURE (psig)	TOTAL FLOW (gpm)
INJECTION MODE (REFLOOD PHASE)	
0	4411.2
20	4163.4
40	3897.1
60	3603.8
80	3275.0
100	2900.8
120	2190.7
140	1619.5
160	482.7
180	480.0
COLD LEG RECIRCULATION MODE	
0	3997.8

TABLE 6.2-9

SAFETY INJECTION FLOW
MAXIMUM SAFEGUARDS

RCS PRESSURE (psig)	TOTAL FLOW (gpm)
INJECTION MODE (REFLOOD PHASE)	
0	8575.0
20	8094.4
40	7581.5
60	7028.8
80	6425.3
100	5752.0
120	4976.6
140	4327.8
160	3530.3
180	2376.1
COLD LEG RECIRCULATION MODE	
0	8575.0

TABLE 6.2-10 (SHEET 1 OF 4)

DOUBLE-ENDED HOT LEG BREAK
BLOWDOWN MASS AND ENERGY RELEASES

TIME (seconds)	BREAK PATH NO.1 FLOW* (lbm/sec)	BREAK PATH NO.1 FLOW* THOUSAND (Btu/sec)	BREAK PATH NO.2 FLOW** (lbm/sec)	BREAK PATH NO.2 FLOW** THOUSAND (Btu/sec)
.00000	.0	.0	.0	.0
.00113	46198.2	29516.1	46195.4	29512.8
.101	40189.0	26064.4	26821.5	17100.0
.201	34372.1	22298.5	23762.2	15065.7
.301	33846.7	21888.6	21200.8	13289.5
.401	32820.3	21206.0	19842.3	12249.4
.501	32023.9	20690.1	18991.0	11540.4
.601	31901.2	20605.1	18375.5	11003.2
.702	31874.4	20599.8	17849.2	10548.9
.801	31502.1	20403.4	17484.0	10215.0
.901	30897.3	20080.6	17160.3	9923.1
1.00	30486.2	19905.6	16866.9	9666.7
1.10	30168.9	19810.0	16638.3	9459.5
1.20	29888.3	19739.6	16448.8	9285.5
1.30	29539.0	19615.6	16339.9	9165.2
1.40	29120.5	19445.2	16294.4	9086.2
1.50	28623.3	19222.2	16299.4	9039.9
1.60	28060.0	18952.1	16335.9	9015.3
1.70	27484.3	18671.5	16392.6	9004.8
1.80	26923.5	18398.8	16463.9	9004.9
1.90	26365.0	18122.1	16540.5	9011.4
2.00	25754.5	17800.6	16619.8	9022.6
2.10	25092.6	17432.7	16698.9	9036.8
2.20	24448.5	17068.0	16775.4	9053.0
2.30	23848.9	16728.9	16847.1	9069.6
2.40	23275.6	16398.4	16908.6	9083.8
2.50	22699.0	16052.7	16957.3	9093.9
2.60	22152.3	15716.1	16990.9	9098.6
2.70	21633.8	15387.8	17010.2	9098.1
2.80	21137.9	15064.3	17014.2	9091.2
2.90	20673.7	14755.2	17004.2	9078.8
3.00	20244.4	14460.2	16980.5	9060.7
3.10	19840.5	14170.6	16943.2	9036.8
3.20	19487.3	13909.9	16893.0	9007.4
3.30	19173.3	13668.3	16831.6	8973.1
3.40	18885.6	13436.2	16758.0	8933.5
3.50	18645.8	13233.3	16673.0	8888.7

TABLE 6.2-10 (SHEET 2 OF 4)

TIME (seconds)	BREAK PATH NO.1 FLOW* (lbm/sec)	BREAK PATH NO.1 FLOW* THOUSAND (Btu/sec)	BREAK PATH NO.2 FLOW** (lbm/sec)	BREAK PATH NO.2 FLOW** THOUSAND (Btu/sec)
3.60	18435.9	13045.7	16577.9	8839.6
3.70	18249.1	12868.5	16472.4	8785.7
3.80	18099.7	12715.2	16358.8	8728.4
3.90	17969.9	12572.2	16235.2	8666.6
4.00	17855.3	12436.7	16100.7	8599.8
4.20	17697.1	12216.4	15802.5	8452.7
4.40	17639.5	12063.2	15468.3	8288.4
4.60	17745.0	12020.9	15116.7	8115.9
4.80	17952.2	12017.3	14771.0	7947.5
5.00	18303.0	12077.3	14352.5	7739.9
5.20	18790.2	12199.5	13896.4	7511.5
5.40	13503.2	9607.3	13445.3	7286.1
5.60	14722.7	10243.4	13012.2	7070.8
5.80	14884.5	10185.1	12576.0	6853.4
6.00	14975.8	10228.1	12111.8	6619.8
6.20	15007.9	10189.6	11650.1	6386.0
6.40	15046.6	10177.1	11197.6	6155.7
6.60	15138.4	10141.8	10763.7	5933.6
6.80	15178.5	10074.0	10344.3	5717.9
7.00	15231.5	9998.6	9941.0	5509.8
7.20	14935.3	9813.0	9565.7	5315.9
7.40	15073.7	9819.4	9219.0	5136.9
7.60	15146.3	9794.5	8891.8	4968.1
7.80	15183.5	9756.1	8590.5	4813.0
8.00	15159.8	9685.1	8304.6	4665.8
8.20	15111.2	9601.5	8034.1	4526.8
8.40	15027.5	9502.2	7780.3	4396.7
8.60	14901.2	9383.0	7532.9	4270.1
8.80	14729.3	9243.8	7296.7	4149.5
9.00	14509.7	9083.3	7068.2	4033.5
9.20	14245.5	8903.4	6846.7	3921.6
9.40	13945.5	8708.8	6632.5	3814.1
9.60	13620.4	8505.1	6424.8	3710.5
9.80	13277.4	8295.5	6222.2	3610.3
10.0	12927.4	8085.7	6026.5	3514.2
10.2	12570.6	7875.5	5834.9	3420.9
10.2	12567.5	7873.8	5833.4	3420.2

TABLE 6.2-10 (SHEET 3 OF 4)

TIME (seconds)	BREAK PATH NO.1 FLOW* (lbm/sec)	BREAK PATH NO.1 FLOW* THOUSAND (Btu/sec)	BREAK PATH NO.2 FLOW** (lbm/sec)	BREAK PATH NO.2 FLOW** THOUSAND (Btu/sec)
10.4	12210.5	7666.6	5649.8	3331.5
10.6	11856.3	7463.8	5471.9	3246.4
10.8	11500.2	7262.7	5298.1	3164.0
11.0	11151.5	7068.5	5130.8	3085.2
11.2	10805.2	6878.1	4967.8	3009.2
11.4	10465.5	6694.1	4811.0	2936.6
11.6	10130.0	6514.7	4658.5	2866.6
11.8	9791.2	6336.2	4509.6	2798.5
12.0	9437.7	6153.3	4362.5	2731.4
12.2	9059.5	5961.8	4210.8	2661.9
12.4	8654.2	5762.0	4048.8	2588.1
12.6	8228.7	5559.3	3873.8	2509.9
12.8	7800.9	5364.4	3685.7	2428.4
13.0	7369.9	5178.2	3479.8	2341.9
13.2	6952.2	5007.3	3269.6	2255.5
13.4	6533.3	4844.8	3054.5	2167.4
13.6	6122.1	4691.4	2851.2	2082.8
13.8	5711.4	4540.4	2663.8	2000.8
14.0	5304.8	4389.3	2502.3	1925.0
14.2	4906.4	4239.9	2368.1	1856.7
14.4	4513.5	4085.7	2259.7	1796.5
14.6	4070.0	3880.6	2173.8	1744.8
14.8	3665.2	3570.9	2101.9	1697.9
15.0	3386.8	3341.6	2039.7	1656.0
15.2	3100.1	3136.6	1982.4	1618.9
15.4	2792.5	2926.1	1923.6	1584.8
15.6	2470.0	2714.0	1860.2	1553.4
15.8	2150.8	2494.5	1787.6	1521.7
16.0	1957.2	2348.3	1703.4	1489.5
16.2	1810.1	2197.2	1605.9	1458.1
16.4	1699.4	2072.4	1498.2	1431.7
16.6	1574.1	1930.3	1380.6	1403.9
16.8	1458.7	1798.1	1265.4	1374.7
17.0	1362.8	1681.6	1169.2	1338.6
17.2	1258.7	1560.0	1102.7	1295.5
17.4	1160.9	1446.1	1041.3	1242.4
17.6	1081.9	1347.9	984.2	1183.1
17.8	1007.4	1260.6	929.9	1126.9

TABLE 6.2-10 (SHEET 4 OF 4)

TIME (seconds)	BREAK PATH NO.1 FLOW* (lbm/sec)	BREAK PATH NO.1 FLOW* THOUSAND (Btu/sec)	BREAK PATH NO.2 FLOW** (lbm/sec)	BREAK PATH NO.2 FLOW** THOUSAND (Btu/sec)
18.0	910.8	1141.2	836.0	1023.3
18.2	814.1	1024.0	707.5	870.1
18.4	715.7	900.0	641.0	791.0
18.6	629.5	793.5	529.7	653.6
18.8	547.0	690.0	403.6	499.9
19.0	458.5	579.1	267.2	332.2
19.2	372.8	471.7	159.1	198.9
19.4	282.1	357.9	97.1	122.4
19.6	207.7	264.6	84.2	107.3
19.8	89.1	114.1	.0	.0
20.0	.0	.0	.0	.0

*mass and energy exiting from the reactor vessel side of the break.

**mass and energy exiting from the SG side of the break.

TABLE 6.2-11

PLANT DATA FOR BLOWDOWN

Reactor coolant loops	3
Minimum steam line internal diameter	14 inches
Main feedwater isolation valve closing time	30 s
Main feedwater control valve closing time	5 s
Main steam line isolation valve closing time	8 s
Main steam line bypass line isolation valve closing time	10 s
Maximum steam line volume between the steam generator and the nearest steam line stop valve	1180 ft^3
Maximum steam line volume between the faulted steam generator stop valves and the steam line stop valves in the other steam generator loops, including 2 seconds of bypass line reverse flow	4381 ft^3
Maximum unisolated feed line volume	202 ft^3
Maximum auxiliary feedwater flow to a depressurized steam generator	Varies with steam generator pressure
Time to auxiliary feedwater isolation	1800 s
Main feedwater flow	Varies
Containment pressure setpoint for main steam line isolation signal	19.2 psig
Air cooler initiation pressure	7.0 psig
Air cooler delay from start of accident	92 s

TABLE 6.2-12

DOUBLE-ENDED HOT LEG BREAK MASS BALANCE

Time (Seconds)		.00	20.00	20.0
			Mass (Thousand lbm)	
Initial	In RCS & ACC	620.08	620.08	620.08
Added Mass	Pumped Injection	.00	.00	.00
	Total Added	.00	.00	.00
TOTAL AVAILABLE		620.08	620.08	620.08
Distribution	Reactor Coolant	416.79	65.09	84.67
	Accumulator	203.30	152.90	133.32
	Total Contents	620.08	217.99	217.99
Effluent	Break Flow	.00	402.08	402.08
	ECCS Spill	.00	.00	.00
	Total Effluent	.00	402.08	402.08
TOTAL ACCOUNTABLE		620.08	620.07	620.07

TABLE 6.2-13

DOUBLE-ENDED HOT LEG BREAK
ENERGY BALANCE

Time (Seconds)		.00	20.00	20.0
			Energy (Million BTU)	
Initial Energy	In RCS, ACC, S. Gen	673.30	673.30	673.30
Added Energy	Pumped Injection	.00	.00	.00
	Decay Heat	.00	5.79	5.79
	Heat From Secondary	.00	-6.91	-6.91
	Total Added	.00	-1.12	-1.12
TOTAL AVAILABLE		673.30	672.19	672.19
Distribution	Reactor Coolant	244.82	14.44	16.19
	Accumulator	18.20	13.68	11.93
	Core Stored	18.93	7.36	7.36
	Primary Metal	118. 16	110.31	110.31
	Secondary Metal	76.01	74.48	74.48
	Steam Generator	197.20	192.31	192.31
	Total Contents	673.30	412.59	412.59
Effluent	Break Flow	.00	259.11	259.11
	ECCS Spill	.00	.00	.00
	Total Effluent	.00	259. 11	259.11
TOTAL ACCOUNTABLE		673.30	671.70	671.70

TABLE 6.2-14 (SHEET 1 OF 4)

DOUBLE-ENDED PUMP SUCTION BREAK
BLOWDOWN MASS AND ENERGY RELEASES

	BREAK PATH NO.1 FLOW*		BREAK PATH NO.2 FLOW**	
TIME (seconds)	(lbm/sec)	THOUSAND (Btu/sec)	(lbm/sec)	THOUSAND (Btu/sec)
.00000	.0	.0	.0	.0
.00108	90598.7	48888.7	40349.1	21718.9
.101	40353.3	21793.7	20648.3	11108.4
.202	46482.0	25280.7	22386.6	12050.6
.302	46307.3	25410.3	22661.8	12210.0
.402	46761.9	25935.4	22249.9	12001.2
.502	46296.0	25990.1	21549.4	11632.8
.602	44228.9	25131.5	20917.5	11297.9
.702	44745.5	25709.3	20392.8	11019.3
.801	44635.7	25899.4	19916.9	10765.7
.901	43950.0	25731.1	19498.4	10541.9
1.00	42962.6	25369.1	19151.6	10356.4
1.10	41980.3	24996.5	18887.2	10214.8
1.20	41034.1	24632.9	18699.2	10114.0
1.30	40163.4	24298.2	18560.7	10039.4
1.40	39364.0	23993.8	18435.7	9971.6
1.50	38594.0	23697.1	18316.5	9906.5
1.60	37802.5	23382.0	18214.6	9850.7
1.70	36954.7	23040.0	18126.9	9802.8
1.80	36079.1	22697.6	18027.0	9748.3
1.90	35106.4	22318.3	17885.7	9671.2
2.00	33872.9	21794.1	17715.2	9578.1
2.10	32344.2	21088.9	17549.4	9488.1
2.20	30820.7	20391.9	17349.3	9379.4
2.30	29075.9	19528.2	17091.6	9239.6
2.40	25410.4	17280.8	16797.9	9080.3
2.50	22059.0	15206.0	16471.4	8903.4
2.60	19923.2	13913.5	16092.2	8698.8
2.70	18262.9	12868.1	15821.6	8553.9
2.80	16954.1	12014.8	15567.8	8418.1
2.90	15973.2	11368.2	15302.8	8276.2
3.00	15192.2	10851.6	15017.0	8123.1
3.10	14591.9	10460.8	14790.8	8002.9
3.20	14100.8	10144.9	14596.6	7899.9
3.30	13665.0	9865.9	14406.5	7798.9
3.40	13257.3	9607.6	14230.0	7705.3
3.50	12888.2	9378.1	14119.3	7647.8
3.60	12567.0	9181.8	14054.6	7614.6
3.70	12247.2	8981.6	13891.7	7527.7

TABLE 6.2-14 (SHEET 2 OF 4)

TIME	BREAK PATH NO.1 FLOW*		BREAK PATH NO.2 FLOW**	
(seconds)	(lbm/sec)	THOUSAND (Btu/sec)	(lbm/sec)	THOUSAND (Btu/sec)
3.80	11934.8	8783.5	13724.4	7438.6
3.90	11648.7	8602.6	13561.2	7351.8
4.00	11396.4	8441.0	13396.3	7263.5
4.20	10974.9	8155.0	13058.2	7082.8
4.40	10630.1	7903.0	12794.2	6942.2
4.60	10367.3	7695.6	12572.7	6823.8
4.80	10144.5	7507.1	12312.0	6684.3
5.00	9969.6	7348.6	12100.0	6571.6
5.20	9816.7	7199.9	13057.1	7093.8
5.40	9710.3	7083.2	12743.2	6924.8
5.60	9660.3	7002.2	12615.2	6858.1
5.80	9648.9	6946.2	12420.1	6754.2
6.00	9649.2	6898.6	12299.5	6691.7
6.20	9661.8	6859.0	12185.5	6632.5
6.40	9882.2	6962.3	12043.3	6557.5
6.60	10191.9	7145.0	11990.7	6530.9
6.80	9940.3	7237.9	11911.0	6487.0
7.00	8907.6	6929.3	11760.1	6402.5
7.20	8293.4	6649.5	11602.2	6314.2
7.40	8116.3	6514.4	11455.6	6233.1
7.60	8051.8	6425.4	11314.4	6155.5
7.80	7991.2	6330.6	11160.2	6070.3
8.00	7966.4	6232.7	10991.6	5976.9
8.20	8002.8	6153.3	10830.6	5887.7
8.40	8065.0	6087.9	10677.9	5803.2
8.60	8120.8	6030.9	10524.0	5718.0
8.80	8149.6	5972.1	10369.5	5632.5
9.00	8135.7	5901.3	10219.6	5549.6
9.20	8089.0	5826.5	10074.9	5469.7
9.40	8007.8	5745.1	9929.7	5389.5
9.60	7894.2	5656.3	9786.3	5310.5
9.80	7753.1	5562.2	9647.8	5234.3
10.0	7605.4	5477.1	9509.1	5158.1
10.2	7437.8	5385.7	9364.9	5078.9
10.4	7264.6	5293.5	9226.4	5003.1
10.6	7093.4	5203.6	9086.0	4926.4
10.8	6923.6	5115.3	8946.7	4850.4
11.0	6758.0	5029.1	8809.1	4775.3
11.2	6595.2	4943.5	8671.7	4700.4
11.4	6439.1	4859.7	8536.6	4626.8
11.6	6289.0	4777.2	8402.2	4553.7

TABLE 6.2-14 (SHEET 3 OF 4)

TIME	BREAK PATH NO.1 FLOW*		BREAK PATH NO.2 FLOW**	
(seconds)	(lbm/sec)	THOUSAND (Btu/sec)	(lbm/sec)	THOUSAND (Btu/sec)
11.8	6145.2	4696.4	8269.5	4481.8
12.0	6008.1	4617.9	8138.9	4411.0
12.2	5876.2	4541.4	8010.2	4341.3
12.4	5748.2	4467.9	7882.8	4272.2
12.6	5624.9	4396.3	7759.7	4205.4
12.8	5504.5	4327.1	7637.5	4139.2
13.0	5386.5	4260.6	7517.0	4073.9
13.2	5271.4	4197.3	7398.9	4009.9
13.4	5154.4	4134.4	7213.8	3908.8
13.6	5029.6	4068.0	7067.1	3827.5
13.8	4888.2	3990.5	6878.9	3700.7
14.0	4735.1	3903.5	6840.4	3626.7
14.2	4567.2	3794.9	6612.0	3434.9
14.4	4414.2	3692.8	6562.6	3323.2
14.6	4279.4	3593.3	6813.5	3376.9
14.8	4176.4	3518.4	5778.5	2808.3
15.0	4087.4	3461.3	7270.0	3434.7
15.2	3953.2	3382.4	11154.0	5315.1
15.4	3773.3	3296.0	7340.3	3538.0
15.6	3761.2	3363.0	4436.2	2138.9
15.8	3693.6	3364.8	6754.0	3020.2
16.0	3470.4	3296.9	9836.8	4401.6
16.2	3281.2	3275.4	5828.1	2647.8
16.4	3221.5	3322.6	4708.7	2159.7
16.6	3092.4	3317.4	4129.3	1838.1
16.8	2770.1	3155.6	4691.2	1973.3
17.0	2504.2	3008.1	4954.4	2033.4
17.2	2210.4	2709.2	4419.5	1791.5
17.4	2001.1	2467.9	4204.6	1677.7
17.6	1821.6	2254.3	4260.6	1652.9
17.8	1651.8	2049.3	4635.3	1729.8
18.0	1497.6	1862.4	4545.8	1641.5
18.2	1353.9	1687.2	4397.7	1546.9
18.4	1219.1	1521.5	4220.7	1449.9
18.6	1091.9	1365.7	3948.7	1325.8
18.8	961.6	1204.7	3568.6	1171.0
19.0	840.8	1054.7	3109.8	997.2
19.2	735.9	924.0	2739.8	858.6
19.4	638.6	802.6	2316.7	710.0
19.6	566.7	712.9	1886.5	566.3
19.8	495.0	623.3	1443.6	425.6

TABLE 6.2-14 (SHEET 4 OF 4)

TIME	BREAK PATH NO.1 FLOW*		BREAK PATH NO.2 FLOW**	
		THOUSAND		THOUSAND
(seconds)	(lbm/sec)	(Btu/sec)	(lbm/sec)	(Btu/sec)
20.0	449.1	565.7	984.8	286.2
20.2	405.0	510.5	529.0	152.3
20.4	351.5	443.3	105.7	30.4
20.6	293.9	370.8	.0	.0
20.8	225.7	285.0	.0	.0
21.0	147.8	186.9	126.9	36.8
21.2	88.6	112.3	87.3	25.3
21.4	29.2	37.2	.0	.0
21.6	.0	.0	.0	.0

*mass and energy exiting the SG side of the break.

** mass and energy exiting the pump side of the break.

TABLE 6.2-15 (SHEET 1 OF 2)

REACTOR CAVITY RELEASE

Time (s)	Flow (lb/s)	Enthalpy (Btu/lb)
0.0	0.0	0.0
1.00×10^{-3}	7.16×10^{3}	5.56×10^{2}
3.04×10^{-3}	1.09×10^{4}	5.55×10^{2}
5.04×10^{-3}	1.33×10^{4}	5.54×10^{2}
7.04×10^{-3}	1.47×10^{4}	5.52×10^{2}
9.08×10^{-3}	1.69×10^{4}	5.51×10^{2}
1.01×10^{-2}	1.72×10^{4}	5.51×10^{2}
1.10×10^{-2}	1.71×10^{4}	5.50×10^{2}
1.31×10^{-2}	1.63×10^{4}	5.48×10^{2}
1.40×10^{-2}	1.72×10^{4}	5.48×10^{2}
1.51×10^{-2}	1.85×10^{4}	5.48×10^{2}
1.70×10^{-2}	1.99×10^{4}	5.47×10^{2}
1.90×10^{-2}	2.01×10^{4}	5.46×10^{2}
2.01×10^{-2}	2.02×10^{4}	5.45×10^{2}
2.11×10^{-2}	2.00×10^{4}	5.44×10^{2}
2.31×10^{-2}	1.99×10^{4}	5.43×10^{2}
2.51×10^{-2}	1.97×10^{4}	5.42×10^{2}
2.70×10^{-2}	1.97×10^{4}	5.41×10^{2}
2.91×10^{-2}	1.98×10^{4}	5.40×10^{2}
3.11×10^{-2}	2.01×10^{4}	5.40×10^{2}
3.31×10^{-2}	2.04×10^{4}	5.39×10^{2}
3.50×10^{-2}	2.08×10^{4}	5.39×10^{2}
3.70×10^{-2}	2.10×10^{4}	5.39×10^{2}
3.91×10^{-2}	2.13×10^{4}	5.38×10^{2}
4.11×10^{-2}	2.14×10^{4}	5.38×10^{2}
4.21×10^{-2}	2.14×10^{4}	5.38×10^{2}
4.31×10^{-2}	2.14×10^{4}	5.37×10^{2}
4.51×10^{-2}	2.11×10^{4}	5.37×10^{2}
4.71×10^{-2}	2.09×10^{4}	5.36×10^{2}
4.92×10^{-2}	2.06×10^{4}	5.36×10^{2}
5.10×10^{-2}	2.04×10^{4}	5.35×10^{2}
5.31×10^{-2}	2.03×10^{4}	5.35×10^{2}
5.50×10^{-2}	2.03×10^{4}	5.35×10^{2}
6.01×10^{-2}	2.04×10^{4}	5.35×10^{2}
6.50×10^{-2}	2.03×10^{4}	5.34×10^{2}
7.01×10^{-2}	2.03×10^{4}	5.34×10^{2}
7.51×10^{-2}	2.02×10^{4}	5.34×10^{2}
8.01×10^{-2}	1.97×10^{4}	5.34×10^{2}
8.50×10^{-2}	1.92×10^{4}	5.33×10^{2}

TABLE 6.2-15 (SHEET 2 OF 2)

Time (s)	Flow (lb/s)	Enthalpy (Btu/lb)
9.00×10^{-2}	1.89×10^{4}	5.33×10^{2}
9.50×10^{-2}	1.87×10^{4}	5.33×10^{2}
1.00×10^{-1}	1.85×10^{4}	5.33×10^{2}
1.20×10^{-1}	1.91×10^{4}	5.33×10^{2}
1.25×10^{-1}	1.90×10^{4}	5.33×10^{2}
1.50×10^{-1}	1.71×10^{4}	5.32×10^{2}
1.75×10^{-1}	1.74×10^{4}	5.32×10^{2}
1.81×10^{-1}	1.75×10^{4}	5.32×10^{2}
2.00×10^{-1}	1.72×10^{4}	5.32×10^{2}
2.50×10^{-1}	1.78×10^{4}	5.32×10^{2}
3.00×10^{-1}	1.74×10^{4}	5.32×10^{2}
3.50×10^{-1}	1.78×10^{4}	5.32×10^{2}
4.00×10^{-1}	1.78×10^{4}	5.32×10^{2}
4.50×10^{-1}	1.80×10^{4}	5.32×10^{2}
4.60×10^{-1}	1.80×10^{4}	5.32×10^{2}
4.70×10^{-1}	1.80×10^{4}	5.32×10^{2}
5.00×10^{-1}	1.78×10^{4}	5.32×10^{2}
5.50×10^{-1}	1.78×10^{4}	5.32×10^{2}
6.00×10^{-1}	1.78×10^{4}	5.32×10^{2}
6.50×10^{-1}	1.76×10^{4}	5.32×10^{2}
7.00×10^{-1}	1.74×10^{4}	5.32×10^{2}
7.50×10^{-1}	1.73×10^{4}	5.32×10^{2}
8.00×10^{-1}	1.75×10^{4}	5.32×10^{2}
8.50×10^{-1}	1.76×10^{4}	5.32×10^{2}
9.00×10^{-1}	1.76×10^{4}	5.32×10^{2}
9.50×10^{-1}	1.77×10^{4}	5.32×10^{2}
1.00	1.78×10^{4}	5.32×10^{2}
1.50	1.81×10^{4}	5.32×10^{2}
1.90	1.82×10^{4}	5.33×10^{2}
2.40	1.82×10^{4}	5.32×10^{2}
2.80	1.80×10^{4}	5.32×10^{2}
3.00	1.79×10^{4}	5.32×10^{2}

TABLE 6.2-16

SPRAY LINE BREAK RELEASE

Time (s)	Flow (lb/s)	Enthalpy (Btu/lb)
0.	0.	6.42×10^2
0.025	3269	6.41×10^2
0.1	3245	6.39×10^2
0.15	3233	6.39×10^2
0.225	3210	6.39×10^2
0.3	3198	6.39×10^2
0.4	3186	6.39×10^2
0.75	3186	6.38×10^2
0.875	3174	6.38×10^2
1.0	3151	6.38×10^2
1.2	3127	6.38×10^2
1.4	3103	6.38×10^2
1.6	3080	6.38×10^2
1.8	3056	6.38×10^2
2.0	3033	6.38×10^1
2.2	3009	6.38×10^2
2.4	2985	6.38×10^2
2.6	2962	6.38×10^2
2.8	2938	6.38×10^2
3.0	2915	6.38×10^2

TABLE 6.2-17

SURGE LINE BREAK RELEASE

Time (s)	Flow (lb/s)	Enthalpy (Btu/lb)
0.	0.	692.8
0.025	6463	692.8
0.1	8585	700.9
0.15	8562	701.2
0.2	8569	700.2
0.3	8592	697.5
0.4	8600	695.0
0.5	8581	693.3
0.6	8533	692.7
0.7	8454	693.2
0.8	8352	694.4
0.9	8241	696.0
1.0	8133	697.3
1.2	7923	698.2
1.4	7841	696.3
1.6	7812	692.5
1.8	7789	688.8
2.0	7720	686.1
2.2	7619	684.4
2.4	7501	683.2
2.6	7381	681.1
2.8	7269	679.9
3.0	7167	677.5

TABLE 6.2-18 (SHEET 1 OF 3)

REACTOR CAVITY SUBCOMPARTMENT PRESSURE ANALYSIS SUMMARY OF FLOWPATHS AND VENT LOSS COEFFICIENTS

Flowpath (from to)	Vent Area (ft^2)	k Contraction	k Expansion	k Bend + Friction	Σk	C
1 →2	14.4	0.04	1.0	0.429	1.47	0.83
1 →9	0.3	0.42	1.0	---	1.42	0.84
1 → 32	0.3	0.42	1.0	---	1.42	0.84
2→9	0.4	0.42	1.0	---	1.42	0.84
2→32	0.4	0.42	1.0	---	1.42	0.84
2→34	14.118	0.067	1.0	---	1.07	0.97
3→4	2.03	0.34	1.0	0.301	1.64	0.78
3→6	2.03	0.34	1.0	0.301	1.64	0.78
3→12	1.52	0.32	1.0	0.363	1.68	0.77
3→19	1.52	0.32	1.0	0.264	1.58	0.79
3→34	1.14	0.42	1.0	---	1.42	0.84
4→5	2.03	---	1.0	0.214	1.21	0.91
4→13	0.98	---	1.0	0.371	1.37	0.85
4→20	0.98	---	1.0	0.523	1.52	0.81
5→7	1.03	0.27	1.0	0.250	1.52	0.81
5→13	0.80	---	1.0	0.377	1.38	0.85
5→20	0.80	---	1.0	0.510	1.51	0.81
5→34	0.55	0.42	1.0	---	1.42	0.84
6→31	1.03	0.27	1.0	0.330	1.60	0.79
6→11	1.05	---	1.0	0.371	1.37	0.85
6→18	1.05	---	1.0	0.522	1.52	0.81
6→34	0.55	0.42	1.0	---	1.42	0.84
7→8	0.88	0.37	1.0	0.471	1.84	0.74
7→14	1.81	---	1.0	0.360	1.36	0.86
7→21	1.81	---	1.0	0.508	1.51	0.81
7→34	1.12	0.42	1.0	---	1.42	0.84
8→15	1.27	---	1.0	0.366	1.37	0.856
8→22	1.27	---	1.0	0.515	1.52	0.812
8→34	0.57	0.42	1.0	---	1.42	0.84
8→33	2.03	---	1.0	0.310	1.31	0.874
9→32	0.88	0.37	1.0	0.460	1.83	0.74
9→16	1.27	---	1.0	0.366	1.36	0.856
9→23	1.27	---	1.0	0.515	1.52	0.812
10→17	0.905	---	1.0	0.374	1.37	0.853
10→24	0.905	---	1.0	0.527	1.53	0.809
10→34	0.55	0.42	1.0	---	1.42	0.84

TABLE 6.2-18 (SHEET 2 OF 3)

Flowpath (from to)	Vent Area (ft^2)	k Contraction	k Expansion	k Bend + Friction	Σk	C
11→12	0.95	---	1.0	0.321	1.32	0.87
11→16	0.95	---	1.0	0.447	1.45	0.83
11→34	0.65	---	1.0	0.187	1.19	0.92
12→13	0.95	---	1.0	0.412	1.41	0.84
12→34	0.95	---	1.0	0.183	1.18	0.92
13→14	0.95	---	1.0	0.447	1.45	0.83
13→34	1.10	---	1.0	0.184	1.18	0.92
14→15	0.95	---	1.0	0.542	1.54	0.81
14→34	1.13	---	1.0	0.183	1.18	0.92
15→17	0.95	---	1.0	0.542	1.54	0.81
15→34	1.58	---	1.0	0.181	1.18	0.92
16→17	0.95	---	1.0	0.542	1.54	0.81
16→34	1.58	---	1.0	0.181	1.18	0.92
17→34	1.13	---	1.0	0.183	1.18	0.92
18→19	2.17	---	1.0	0.308	1.31	0.87
18→23	2.17	---	1.0	0.430	1.43	0.84
18→25	1.05	---	1.0	0.534	1.53	0.81
19→20	2.17	---	1.0	0.395	1.39	0.85
19→25	1.52	---	1.0	0.524	1.52	0.81
20→21	2.17	---	1.0	0.430	1.43	0.84
20→25	1.78	---	1.0	0.519	1.52	0.81
21→22	2.17	---	1.0	0.521	1.52	0.81
21→26	1.81	---	1.0	0.520	1.52	0.81
22→24	2.17	---	1.0	0.521	1.52	0.81
22→26	2.54	---	1.0	0.513	1.51	0.81
23→24	2.17	---	1.0	0.521	1.52	0.81
23→27	2.54	---	1.0	0.513	1.51	0.81
24→27	1.81	---	1.0	0.520	1.52	0.81
25→26	2.13	---	1.0	1.010	2.01	0.71
25→27	2.13	---	1.0	1.010	2.01	0.71
25→28	4.35	---	1.0	0.759	1.76	0.75
26→27	2.13	---	1.0	1.010	2.01	0.71
26→28	4.35	---	1.0	0.759	1.76	0.75
27→28	4.35	---	1.0	0.759	1.76	0.75
28→29	106.70	---	1.0	0.350	1.35	0.86
29→30	93.31	0.05	1.0	0.070	1.12	0.94
30→34	56.66	0.08	1.0	1.198	2.28	0.66

TABLE 6.2-18 (SHEET 3 OF 3)

Flowpath (from to)	Vent Area (ft^2)	k Contraction	k Expansion	k Bend + Friction	Σk	C
31→34	0.55	0.42	1.0	---	1.42	0.84
31→9	2.03	---	1.0	0.310	1.31	0.874
31→16	1.27	---	1.0	0.366	1.37	0.856
31→23	1.27	---	1.0	0.515	1.52	0.812
32→10	2.03	---	1.0	0.218	1.22	0.906
32→17	0.905	---	1.0	0.374	1.37	0.853
32→24	0.905	---	1.0	0.527	1.53	0.809
33→10	1.03	0.34	1.0	0.440	1.78	0.75
33→34	0.55	0.42	1.0	---	1.42	0.84
33→15	1.27	---	1.0	0.366	1.37	0.856
33→22	1.27	---	1.0	0.515	1.5	0.812

TABLE 6.2-19

CONTAINMENT RESULTS FOR THE DESIGN BASIS LOCA

Parameter	Prior to LOCA	DEPSG At Peak	DEHL At Peak
Pressures			
Time (s)		552	18.8
Steam (psia)	1.03	37.1	37.7
Air (psia)	16.67	21.4	20.6
Total psia	17.70	58.5	58.3
Total gauge (psig)	3.0	43.8	43.6
Temperatures			
Time (s)		1252	20.0
Steam and air (°F)	127	263	264
Water in sump (°F)	-	260	256
Heat transfer coefficient (Btu/h-ft^2-°F)[a]	0	218	231

a. Between containment atmosphere and structure.

TABLE 6.2-20 (SHEET 1 OF 4)

DOUBLE-ENDED PUMP SUCTION BREAK - MINIMUM SAFEGUARDS REFLOOD MASS AND ENERGY RELEASES

	BREAK PATH NO.1 FLOW*		BREAK PATH NO.2 FLOW**	
TIME		THOUSAND		THOUSAND
(seconds)	(lbm/sec)	(Btu/sec)	(lbm/sec)	(Btu/sec)
21.6	.0	.0	.0	.0
22.1	.0	.0	.0	.0
22.2	.0	.0	.0	.0
22.3	.0	.0	.0	.0
22.4	.0	.0	.0	.0
22.5	.0	.0	.0	.0
22.6	55.3	65.3	.0	.0
22.7	31.1	36.7	.0	.0
22.8	33.2	39.2	.0	.0
22.9	39.1	46.1	.0	.0
23.0	45.9	54.2	.0	.0
23.1	49.4	58.3	.0	.0
23.2	55.4	65.3	.0	.0
23.3	59.8	70.6	.0	.0
23.4	64.0	75.5	.0	.0
23.5	68.0	80.2	.0	.0
23.6	71.8	84.7	.0	.0
23.7	75.4	89.0	.0	.0
23.8	79.0	93.2	.0	.0
23.9	82.4	97.2	.0	.0
24.0	85.7	101.1	.0	.0
24.1	88.8	104.9	.0	.0
24.2	91.9	108.5	.0	.0
24.3	94.9	112.0	.0	.0
24.4	97.8	115.5	.0	.0
24.5	100.7	118.8	.0	.0
24.6	103.4	122.1	.0	.0
25.6	127.9	151.0	.0	.0
26.6	148.3	175.1	.0	.0
27.6	165.9	195.9	.0	.0
28.2	321.3	380.3	2733.5	349.6
28.6	421.0	499.0	3803.6	500.4
29.7	453.5	537.9	4080.9	557.9
30.7	443.1	525.5	3984.9	548.9
31.7	460.0	545.7	4164.0	565.5
32.5	451.4	535.3	4085.8	556.9
32.7	449.2	532.8	4066.2	554.7

TABLE 6.2-20 (SHEET 2 OF 4)

TIME (seconds)	BREAK PATH NO.1 FLOW* (lbm/sec)	BREAK PATH NO.1 FLOW* THOUSAND (Btu/sec)	BREAK PATH NO.2 FLOW** (lbm/sec)	BREAK PATH NO.2 FLOW** THOUSAND (Btu/sec)
33.7	438.8	520.4	3970.3	544.3
34.7	428.9	508.5	3877.7	534.2
35.7	419.4	497.2	3788.3	524.4
36.7	410.3	486.4	3702.2	514.9
37.7	401.7	476.0	3619.1	505.8
37.9	400.0	474.0	3602.8	504.0
38.7	393.4	466.1	3538.9	497.0
39.7	385.4	456.7	3461.4	488.5
40.7	377.8	447.6	3386.5	480.3
41.7	370.5	438.9	3314.1	472.3
42.7	363.5	430.5	3243.9	464.5
43.7	356.7	422.4	3175.9	457.0
44.2	353.4	418.5	3142.6	453.4
44.7	350.2	414.7	3109.8	449.8
45.7	343.9	407.2	3045.7	442.7
46.7	337.8	400.0	2983.3	435.8
47.7	332.0	393.0	2922.7	429.1
48.7	326.3	386.3	2863.6	422.5
49.7	320.8	379.7	2806.0	416.1
50.7	315.5	373.4	2749.8	409.9
51.3	312.4	369.7	2716.8	406.2
51.7	310.3	367.3	2695.0	403.8
52.7	253.5	299.8	2021.5	332.6
53.7	324.4	383.8	280.0	153.3
54.7	341.3	404.0	285.3	162.0
55.7	336.8	398.8	283.7	159.7
56.7	332.4	393.5	282.0	157.5
57.7	328.1	388.3	280.4	155.3
58.7	323.7	383.2	278.8	153.2
59.7	319.5	378.1	277.2	151.1
60.7	315.2	373.1	275.6	149.0
61.7	311.0	368.0	274.1	146.9
62.7	306.7	362.9	272.5	144.8
63.7	302.7	358.2	271.0	142.8
64.7	298.7	353.5	269.6	140.9
65.7	294.9	348.9	268.2	139.0
66.5	291.8	345.2	267.1	137.5
66.7	291.0	344.3	266.8	137.1
67.7	287.2	339.8	265.4	135.3
68.7	283.5	335.4	264.1	133.5
69.7	279.8	331.0	262.8	131.8

TABLE 6.2-20 (SHEET 3 OF 4)

TIME	BREAK PATH NO.1 FLOW*		BREAK PATH NO.2 FLOW**	
(seconds)	(lbm/sec)	THOUSAND (Btu/sec)	(lbm/sec)	THOUSAND (Btu/sec)
70.7	276.2	326.6	261.5	130.0
71.7	272.6	322.4	260.2	128.3
72.7	269.0	318.2	258.9	126.7
73.7	265.6	314.1	257.7	125.0
74.7	262.1	310.0	256.5	123.4
75.7	258.7	306.0	255.3	121.9
76.7	255.4	302.0	254.1	120.3
77.7	252.1	298.1	253.0	118.8
78.7	248.9	294.3	251.9	117.3
79.7	245.8	290.6	250.8	115.9
80.7	242.6	286.9	249.7	114.5
81.7	239.6	283.2	248.6	113.1
82.7	236.6	279.6	247.6	111.7
84.2	232.2	274.4	246.1	109.8
84.7	230.7	272.7	245.6	109.1
86.7	225.1	266.0	243.7	106.6
88.7	219.7	259.6	241.8	104.2
90.7	214.5	253.5	240.1	102.0
92.7	209.5	247.5	238.4	99.8
94.7	204.7	241.9	236.8	97.8
96.7	200.1	236.5	235.3	95.8
98.7	195.7	231.3	233.9	94.0
100.7	191.6	226.3	232.5	92.2
102.7	187.6	221.6	231.2	90.5
104.7	183.8	217.1	230.0	89.0
105.3	182.7	215.8	229.7	88.5
106.7	180.2	212.9	228.9	87.5
108.7	176.8	208.8	227.8	86.1
110.7	173.5	205.0	226.8	84.8
112.7	170.5	201.4	225.8	83.6
114.7	167.6	198.0	224.9	82.4
116.7	164.9	194.8	224.1	81.4
118.7	162.4	191.8	223.3	80.3
120.7	160.0	189.0	222.5	79.4
122.7	157.8	186.3	221.9	78.5
124.7	155.7	183.9	221.2	77.7
126.7	153.7	181.6	220.6	77.0
128.7	151.9	179.4	220.1	76.3

TABLE 6.2-20 (SHEET 4 OF 4)

TIME (seconds)	BREAK PATH NO.1 FLOW* (lbm/sec)	BREAK PATH NO.1 FLOW* THOUSAND (Btu/sec)	BREAK PATH NO.2 FLOW** (lbm/sec)	BREAK PATH NO.2 FLOW** THOUSAND (Btu/sec)
130.1	150.8	178.0	219.7	75.8
130.7	150.3	177.5	219.6	75.6
132.7	148.7	175.6	219.1	75.0
134.7	147.3	173.9	218.7	74.5
136.7	146.0	172.4	218.3	73.9
138.7	144.8	171.0	217.9	73.5
140.7	143.7	169.7	217.6	73.0
142.7	142.6	168.5	217.2	72.6
144.7	141.7	167.4	217.0	72.3
146.7	140.9	166.4	216.7	71.9
148.7	140.1	165.5	216.5	71.6
150.7	139.4	164.7	216.2	71.4
152.7	138.8	163.9	216.0	71.1
154.7	138.2	163.2	215.8	70.9
156.7	137.7	162.6	215.7	70.6
158.0	137.4	162.2	215.6	70.5
158.7	137.2	162.0	215.5	70.4
160.7	136.8	161.5	215.4	70.3
162.7	136.4	161.1	215.2	70.1
164.7	136.1	160.7	215.1	70.0
166.7	135.8	160.4	215.0	69.8
168.7	135.6	160.1	214.9	69.7
170.7	135.4	159.9	214.9	69.6
172.7	135.3	159.8	214.8	69.5
174.7	135.2	159.6	214.7	69.5
176.7	135.1	159.5	214.7	69.4
178.7	135.0	159.5	214.6	69.3
180.7	135.0	159.4	214.6	69.3
182.7	135.2	159.6	214.6	69.3
184.7	135.6	160.2	215.3	69.6
186.7	136.1	160.7	216.8	70.0
187.7	136.3	161.0	217.8	70.3

* mass and energy exiting the SG side of the break.

** mass and energy exiting the pump side of the break.

TABLE 6.2-21

LOCA CHRONOLOGY OF EVENTS

Time (s)	Event
0.0	Pipe ruptures (DEPSG), reactor depressurization begins.
(a)	Mass and energy release modeling.
62.0	Containment sprays begin operation.
92.0	Air coolers begin operation.
552.0	Containment reaches maximum peak pressure
1252	Sump reaches maximum temperature.
2139	Safety injection water recirculation from the sump begins as RWST reaches low level.
4256	Containment spray water recirculation from the sump begins as RWST reaches low-low level.
10^7	Containment reaches atmospheric pressure (estimate).

a. See table 6.2-52

TABLE 6.2-22 (SHEET 1 OF 2)

SUBCOMPARTMENT DIFFERENTIAL PRESSURE RESULTS

Steam Generator Compartment

Compartment	Cold Leg Break (psid)	Time (s)	Design (psid)
1, SG-C	33.9	0.42	35
2, SG-A	22.6	0.42	35
3, SG-B	19.3	0.60	35

Pressurizer Compartment

Compartment	Spray Line Break (psid)	Time (s)	Design (psid)
1	11.5	0.3	35
2	4.5	0.07	35

Reactor Cavity

Cold Leg Break

Node No.	Volume (ft^3)	Pressure (psia)	Time (s)	Design Pressure (psid)
1[a]	67.70	305.48	0.129	667
2[a]	104.12	288.02	0.129	667
3	17.17	19.15	0.600	150
4	7.36	18.83	0.600	150
5	9.46	18.26	0.600	150
6	11.37	21.30	0.600	150
7	14.47	18.40	0.600	150
8	12.41	19.80	0.600	150
9	12.41	57.26	0.135	150
10	10.27	41.65	0.140	150
11	3.68	21.64	0.600	150
12	5.33	18.78	0.600	150
13	6.22	18.21	0.600	150
14	6.35	18.02	0.600	150
15	8.89	19.79	0.600	150
16	8.89	36.50	0.141	150
17	6.35	36.59	0.141	150

a. Inside penetration at inspection opening.

TABLE 6.2-22 (SHEET 2 OF 2)

Reactor Cavity (continued)

Cold Leg Break

Node No.	Volume (ft^3)	Pressure (psia)	Time (s)	Design Pressure (psid)
18	8.42	21.36	0.600	150
19	12.19	19.32	0.600	150
20	14.23	18.83	0.600	150
21	14.52	18.86	0.600	150
22	20.32	19.98	0.600	150
23	20.32	26.36	0.148	150
24	14.52	26.29	0.148	150
25	34.25	18.58	0.600	150
26	34.25	18.59	0.600	150
27	34.25	19.69	0.600	150
28	1055.70	16.07	0.598	150
29	2190.60	16.02	0.600	150
30	3603.10	15.98	0.600	150
31	13.00	37.62	0.141	150
32	9.68	58.82	0.135	150
33	13.00	20.20	0.600	150
34	2.0×10^6	15.88	0.600	54

Net Vessel Side Load[b]

Lbf	Time (s)
1.184×10^6	0.12

b. Reactor vessel support stresses not to exceed design criteria presented in tables 5.2-6 and 5.2-7.

TABLE 6.2-23

(This page has been intentionally deleted.)

TABLE 6.2-24 (SHEET 1 OF 2)

COMPONENT DESIGN PARAMETERS FOR CONTAINMENT SPRAY SYSTEM AND CONTAINMENT COOLING SYSTEM

<u>Containment Spray Pumps</u>	
Type	Horizontal Centrifugal
Number	2
Pressure (psig)	300
Temperature (°F)	250
Flowrate (each) (gal/min)	2600
Head (ft)	450
<u>Containment Coolers</u>	
Number	4[(a)]
Pressure (psig)	200
Temperature (°F)	300
Water inlet temperature (°F)	95
Flowrate (normal - high reactor coolant leakage) (gal/min)	800
Heat removal rate (normal) (Btu/h)	2.36×10^6
Flowrate (post-LOCA) (gal/min)	2000 (600 for containment analysis)
Heat removal rate (post-LOCA) (Btu/h)	80.0×10^6 (31.2×10^6 for containment analysis)
<u>Containment Cooler Fans</u>	
Type	Vaneaxial
Number	4
Flowrate (high speed) (sf^3/min)	80,000
Static head (high speed) (in. wg)	4.75
Horsepower (high speed) (hp)	80
Flowrate (low speed) (sf^3/min)	40,000
Static head (low speed) (in. wg)	7.90
Horsepower (low speed) (hp)	105

TABLE 6.2-24 (SHEET 2 OF 2)

COMPONENT DESIGN PARAMETERS FOR CONTAINMENT SPRAY SYSTEM AND CONTAINMENT COOLING SYSTEM

<u>Refueling Water Storage Tank</u>	
Quantity	1
Volume (gal)	500,000
Design pressure (psig)	atmosphere
Design temperature (°F)	ambient
Material	stainless steel
<u>Piping</u>	
Pressure (psig)	210
Temperature (°F)	300
<u>Valves</u>	
Pressure (psig)	210
Temperature (°F)	300

a. Having fewer than 12 coils per containment cooler is acceptable, provided that each cooler can adequately remove the containment analysis heat load.

TABLE 6.2-25 (SHEET 1 OF 3)

REGULATORY GUIDE 1.52, REV. 0
SECTION APPLICABILITY FOR THE PENETRATION ROOM FILTRATION SYSTEM

Regulatory Guide Section	Applicability to This System	Note Index
C.1.a	Yes	1
C.1.b	Yes	-
C.1.c	Yes	-
C.1.d	Yes	-
C.1.e	Yes	-
C.2.a	No	2
C.2.b	No	3
C.2.c	Yes	-
C.2.d	Yes	-
C.2.e	Yes	16
C.2.f	Yes	-
C.2.g	Yes	4
C.2.h	Yes	-
C.2.i	Yes	-
C.2.j	No	6
C.2.k	Yes	-
C.2.l	Yes	-
C.2.m	Yes	-
C.3.a	No	7
C.3.b	Yes	8
C.3.c	Yes	-
C.3.d	Yes	-
C.3.e	Yes	9
C.3.f	Yes	-
C.3.g	Yes	-
C.3.h	Yes	10
C.3.i	Yes	-
C.3.j	No	11
C.3.k	Yes	-
C.3.l	Yes	12
C.3.m	Yes	13
C.3.n	Yes	-

TABLE 6.2-25 (SHEET 2 OF 3)

Regulatory Guide Section	Applicability to This System	Note Index
C.4.a	Yes	-
C.4.b	Yes	-
C.4.c	Yes	14
C.4.d	Yes	-
C.4.e	Yes	-
C.4.f	Yes	-
C.4.g	Yes	-
C.4.h	Yes	15
C.4.i	Yes	-
C.4.j	Yes	-
C.4.k	Yes	-
C.4.l	Yes	-
C.4.m	Yes	-
C.5.a	Yes	-
C.5.b	Yes	17
C.5.c	Yes	17
C.6.a	Yes	-
C.6.b	Yes	-

NOTES

1. The design basis accident is the postulated 30-day LOCA.

2. No demister is provided because the unit is located outside the containment and no entrained water droplets are anticipated. No HEPA filters are provided downstream of the charcoals, since radioactive fines carryover is very unlikely. This is true because the charcoal trays are pressure tested at high velocity in the manufacturer's shop prior to delivery, thereby removing fines. Also, during system operation, air is passing through the charcoal at a very low velocity.

3. No physical separation is provided since these units are located in a room where no missiles are postulated.

4. Pressure drops across the prefilters, HEPA, and charcoal filters are instrumented to indicate in the control room. Pressure drops across the HEPA and charcoal filters are instrumented to alarm in the control room. No recording of these signals is provided. Fan loss of flow is also instrumented to signal and alarm in the control room.

TABLE 6.2-25 (SHEET 3 OF 3)

5. Deleted

6. The size of the engineered safety feature filtration units precludes replacement as a single unit. The unit components are replaced individually.

7. Demisters are not provided.

8. Electric heaters are used to reduce the relative humidity to 70 percent or less. The use of heating coils to control the relative humidity during DBAs is not credited in the respective DBA dose assessment.

9. Mounting frames for filter and charcoals are constructed of carbon steel coated with an inorganic nuclear grade paint.

10. Internal welds are carbon steel coated with an inorganic nuclear grade paint.

11. The deluge and drain system has been eliminated due to recurring problems experienced at other facilities associated with inadvertent wetting of the absorber. Temperature gauges have been installed to monitor any heat rise in the filter housing.

12. Environmental conditions for systems considered are those specified under outside containment and radioactive area.

13. Duct construction guidelines follow SMACNA in addition to ORNL-NSIC-65.

14. Vacuum breakers are not used. This presents the probability of system leakage from pressure-relieving device leakage or failure.

15. Test probes are not manifolded and are located in readily accessible locations with minimum piping.

16. The accident analyses do not credit the heaters for humidity control.

17. Periodic testing to confirm a penetration of less than 0.5% at rated flow.

TABLE 6.2-26

SINGLE-FAILURE ANALYSIS - CONTAINMENT SPRAY SYSTEM

Component	Malfunction	Comments and Consequences
Spray Nozzles	Clogged	Large number of nozzles precludes clogging of a significant number.
Pumps		
Containment Spray pump	Fails to start	Two pumps provided. Operation of one required.
Automatically operated valves (open on coincidence of two out of four high-high-high containment pressure signals or 2/2 manual initiation of spray system operation from the control room):		
Containment spray pump discharge isolation valve	Fails to open	Two valves provided. Operation of one required.
Valves operated from control room for recirculation		
Containment sump recirculation isolation	Fails to open	Two lines in parallel, one to each spray pump. Operation of one required.

TABLE 6.2-27 (SHEET 1 OF 4)

DOUBLE-ENDED PUMP SUCTION BREAK - MINIMUM SAFEGUARDS BLOWDOWN MASS AND ENERGY RELEASES

	BREAK PATH NO.1 FLOW		BREAK PATH NO.2 FLOW	
TIME		THOUSAND		THOUSAND
(seconds)	(lbm/sec)	(Btu/sec)	(lbm/sec)	(Btu/sec)
.00000	.0	.0	.0	.0
.00107	91449.9	49351.4	40349.8	21719.3
.101	40359.3	21797.0	20658.3	11113.7
.202	46617.0	25354.7	22408.0	12062.5
.302	46432.7	25480.6	22666.0	12212.7
.401	46869.1	25996.8	22243.1	11998.0
.501	46368.9	26033.7	21470.2	11589.9
.602	44240.1	25142.7	20750.4	11207.0
.702	44736.7	25706.9	20197.6	10911.5
.801	44554.7	25851.2	19705.5	10648.8
.902	43717.2	25588.5	19292.6	10429.1
1.00	42587.0	25134.1	18975.1	10260.1
1.10	41472.4	24675.8	18706.9	10116.8
1.20	40418.9	24239.3	18519.1	10016.3
1.30	39437.5	23830.9	18414.8	9960.8
1.40	38576.2	23476.8	18363.9	9933.9
1.50	37835.2	23180.4	18319.5	9910.0
1.60	37185.8	22928.6	18249.3	9871.5
1.70	36527.6	22670.7	18165.3	9825.3
1.80	35825.3	22391.2	18102.7	9790.9
1.90	35103.7	22112.9	18055.6	9765.3
2.00	34331.7	21820.3	17973.8	9720.8
2.10	33391.3	21435.8	17816.8	9635.2
2.20	32232.2	20918.3	17637.2	9537.5
2.30	30888.1	20287.8	17468.2	9446.2
2.40	29609.4	19693.0	17287.5	9348.6
2.50	28145.1	18960.7	16966.5	9174.7
2.60	24508.1	16681.4	16630.6	8993.2
2.70	21807.8	15032.8	16330.2	8831.3
2.80	20011.8	13965.9	16044.1	8677.3
2.90	18317.8	12903.2	15779.2	8535.2
3.00	16928.0	12017.7	15521.1	8397.0
3.10	15822.1	11306.9	15270.3	8262.8
3.20	14944.1	10739.5	15046.0	8143.3
3.30	14284.0	10314.6	14844.4	8036.1
3.40	13777.3	9987.8	14652.3	7933.9
3.50	13361.8	9717.3	14467.2	7835.4
3.60	12989.6	9474.7	14205.4	7694.8
3.70	12659.4	9261.3	14028.4	7600.9
3.80	12361.3	9070.4	13847.8	7504.7
3.90	12082.6	8891.7	13720.8	7438.2

TABLE 6.2-27 (SHEET 2 OF 4)

TIME (seconds)	BREAK PATH NO.1 FLOW (lbm/sec)	BREAK PATH NO.1 FLOW THOUSAND (Btu/sec)	BREAK PATH NO.2 FLOW (lbm/sec)	BREAK PATH NO.2 FLOW THOUSAND (Btu/sec)
4.00	11825.7	8727.7	13627.0	7388.7
4.20	11325.7	8401.6	13270.0	7197.5
4.40	10904.1	8121.2	12960.2	7032.1
4.60	10543.4	7866.4	12763.1	6927.4
4.80	10264.3	7653.1	12479.4	6774.8
5.00	10031.8	7461.6	12247.6	6650.9
5.20	9851.1	7298.9	13134.9	7138.0
5.40	9697.9	7150.0	12990.5	7057.9
5.60	9603.4	7039.6	12669.4	6886.1
5.80	9570.6	6967.1	12529.0	6811.9
6.00	9564.8	6913.1	12367.6	6726.8
6.20	9565.1	6864.3	12249.6	6665.6
6.40	9723.7	6927.8	12117.7	6596.4
6.60	10072.7	7127.9	12039.2	6556.2
6.80	9924.2	7261.9	11949.9	6508.5
7.00	8909.4	6974.7	11798.0	6424.8
7.20	8275.4	6686.1	11627.9	6330.5
7.40	8101.4	6554.2	11465.6	6241.2
7.60	8034.1	6477.0	11318.3	6160.3
7.80	7931.3	6382.2	11155.7	6070.8
8.00	7834.6	6267.4	10977.1	5972.0
8.20	7798.2	6161.1	10806.4	5877.7
8.40	7803.0	6069.0	10644.3	5788.2
8.60	7821.2	5991.1	10481.8	5698.3
8.80	7826.8	5914.7	10319.3	5608.5
9.00	7810.2	5836.0	10164.0	5522.5
9.20	7769.0	5755.2	10011.2	5438.0
9.40	7703.3	5672.7	9860.5	5354.7
9.60	7607.9	5582.5	9713.2	5273.3
9.80	7499.5	5497.8	9572.5	5195.7
10.0	7373.2	5411.9	9424.5	5114.1
10.2	7231.6	5320.2	9282.5	5036.1
10.4	7088.2	5230.3	9141.4	4958.7
10.4	7087.2	5229.7	9140.4	4958.1
10.4	7086.1	5229.1	9139.3	4957.5
10.6	6943.4	5142.6	9001.7	4882.1
10.8	6796.7	5056.6	8864.8	4807.2
11.0	6649.1	4971.8	8728.5	4732.7
11.2	6502.4	4888.3	8594.3	4659.5
11.4	6357.8	4806.1	8460.0	4586.5
11.6	6217.3	4725.9	8329.0	4515.4
11.8	6080.3	4647.4	8198.6	4444.7

TABLE 6.2-27 (SHEET 3 OF 4)

TIME (seconds)	BREAK PATH NO.1 FLOW (lbm/sec)	BREAK PATH NO.1 FLOW THOUSAND (Btu/sec)	BREAK PATH NO.2 FLOW (lbm/sec)	BREAK PATH NO.2 FLOW THOUSAND (Btu/sec)
12.0	5947.1	4570.8	8070.2	4375.1
12.2	5817.7	4496.2	7945.7	4307.6
12.4	5689.8	4423.9	7820.6	4239.8
12.6	5566.8	4354.1	7699.8	4174.3
12.8	5445.5	4285.7	7578.7	4108.8
13.0	5327.3	4219.9	7460.5	4044.9
13.2	5207.8	4153.8	7308.6	3962.2
13.4	5077.7	4081.2	7096.6	3847.0
13.6	4933.2	3998.6	6962.0	3763.2
13.8	4774.0	3901.6	6743.9	3609.9
14.0	4615.1	3797.8	6729.5	3543.7
14.2	4468.4	3694.9	6471.6	3337.2
14.4	4349.0	3606.0	6586.0	3319.5
14.6	4246.7	3531.2	6091.1	3005.9
14.8	4153.8	3473.2	6258.5	3014.6
15.0	4048.6	3417.8	6256.0	2966.8
15.2	3947.5	3378.7	5573.0	2607.2
15.4	3844.1	3346.3	5805.5	2657.7
15.6	3719.0	3305.7	5945.1	2683.1
15.8	3608.1	3289.6	5479.0	2453.2
16.0	3486.7	3275.8	5194.7	2300.8
16.2	3362.9	3270.5	5095.5	2229.6
16.4	3220.0	3261.8	5073.6	2193.1
16.6	3025.9	3222.9	4973.3	2127.3
16.8	2725.4	3101.7	4688.8	1986.2
17.0	2481.9	2985.0	4330.8	1817.9
17.2	2199.8	2697.7	4020.6	1669.3
17.4	1987.4	2451.5	3709.4	1515.2
17.6	1804.7	2233.5	3547.4	1413.2
17.8	1636.2	2030.1	3734.9	1439.5
18.0	1478.6	1838.6	4080.5	1520.7
18.2	1330.6	1658.1	4340.7	1567.9
18.4	1196.0	1492.6	4064.5	1432.5
18.6	1065.0	1332.3	3763.4	1297.4
18.8	937.4	1174.4	3425.7	1154.3
19.0	823.9	1033.5	3022.7	994.5
19.2	731.6	918.6	2674.9	858.6
19.4	651.6	818.8	2261.5	708.0
19.6	594.6	748.0	1812.0	554.0
19.8	538.3	677.5	1332.1	398.9

TABLE 6.2-27 (SHEET 4 OF 4)

TIME (seconds)	BREAK PATH NO.1 FLOW (lbm/sec)	BREAK PATH NO.1 FLOW THOUSAND (Btu/sec)	BREAK PATH NO.2 FLOW (lbm/sec)	BREAK PATH NO.2 FLOW THOUSAND (Btu/sec)
	20.0	483.3	608.7	816.3
	240.7			
20.2	426.2	537.0	306.4	89.7
20.4	366.0	461.5	.0	.0
20.6	305.4	385.3	.0	.0
20.8	248.8	314.1	.0	.0
21.0	193.1	244.0	.0	.0
21.2	107.4	136.0	.0	.0
21.4	12.3	15.7	.0	.0
21.6	.0	.0	.0	.0

TABLE 6.2-28 (SHEET 1 OF 2)

CONTAINMENT VENTILATION SYSTEMS COMPONENT DESIGN PARAMETERS

Containment Coolers (normal)

Number	4
Pressure (psig)	200
Temperature (°F)	300
Water inlet temperature (°F)	95
Flowrate (each) (gal/min)	800
Heat removal rate (each) (btu/h)	2.36×10^6

Containment Cooler Fans (normal)

Type	Vaneaxial
Number	4
Flowrate (each) (sft^3/min)	80,000
Static head (in. WG)	4.75
Motor horsepower (each) (hp)	80

Containment Recirculation Fans

Type	Vaneaxial
Number	4
Flowrate (each) (sft^3/min)	25,000
Static head (in. WG)	0.32
Motor horsepower (each) (hp)	7.5

Control-Rod Mechanism Cooling Fans

Type	Vaneaxial
Number	2
Flowrate (each) (sft^3/min)	40,000
Static head (in. WG)	9.0
Motor horsepower (each) (hp)	100

Reactor Cavity Cooling Fans

Type	Vaneaxial
Number	2
Flowrate (each) (sft^3/min)	17,000
Static head (in. WG)	2.46
Motor horsepower (each) (hp)	15

TABLE 6.2-28 (SHEET 2 OF 2)

<u>Refueling Water Surface Ventilation Supply Fan</u>

Type	Vaneaxial
Number	1
Flowrate (each) (sft^3/min)	7,500
Static head (in. WG)	4.5
Motor horsepower (each) (hp)	15

<u>Refueling Water Surface Ventilation Exhaust Fan</u>

Type	Vaneaxial
Number	1
Flowrate (each) (sft^3 min)	22,000
Static head (in. WG)	2.0
Motor horsepower (each) (hp)	15

TABLE 6.2-29

SPRAY EVALUATION PARAMETERS

Spray flowrate (gal/min)	2480 (injection) 2290 (recirculation)
Containment sump volume (ft^3)	4.92×10^4
Containment sprayed volume (ft^3)	1.67×10^6
Minimum spray fall height (ft)	110
Elemental λ_s (h^{-1})	10.0 (DF < 21) 0.0 (DF > 21)
Methyl λ_s (h^{-1})	0.0
Particulate λ_s (h^{-1})	5.4 (injection) 5.0 (recirculation, DF < 50) 0.0 ($\geq$ 8 h) 0.5 (DF > 50 until 8 h)
pH (Spray injection)	4.5
pH (Spray recirculation)	7.7

TABLE 6.2-30

SINGLE FAILURE ANALYSIS - PENETRATION ROOM FILTRATION SYSTEM

Component	Malfunction	Comments
Fan	Fails	The other fan and filter system will be available.
Fan discharge valve	Fails to open	Same as above.
Fan discharge valve	Fails to close	Check valve will prevent back flow.
Recirculation line valve	Fails to open	Recirculation fan will operate in the exhaust mode.
Recirculation line valve	Fails to close	The other system will be available.

TABLE 6.2-31 (SHEET 1 OF 7)

CONTAINMENT ISOLATION VALVE INFORMATION[a]

Item No.	Service (No. of Penetrations)	System	Penetration No.	Penetration Type	Penetration Line Size (in.)	Valve Arrangement	Flow Direction	Location Relative to Containment	Valve Type	Actuator	Signal	Normal Valve Position	Valve Pos. with Power Fail.	Pos. Ind.	Post LOCA Position	Valve Closure Time (s)
1	Accumulator Test Line (1)	SIS	29	II	3/4 3/4	24	OUT	Inside Outside	Globe Globe	Air Air	T T	Closed Closed	Closed Closed	Yes Yes	Closed Closed	≤10 ≤10
2	Refueling Cav. Supply (1)	FHS	95	II	2 2	35	IN	Inside Outside	Check Diaphram	-------- --------	-------- --------	-------- Lkd Closed	-------- As Is	--- ---	-------- Closed	--- ---
3	Nitro. Supply to Accumulators (1)	SIS	63	II	1	23	IN	Inside Outside	Check Globe	-------- Air	-------- T	-------- Closed	-------- Closed	--- Yes	-------- Closed	--- ≤10
4	Nitro. Supply-Press. Relief Tank (1)	RCS	64a	II	1 1	38	IN	Inside Outside	Diaphragm Diaphragm	Air Air	T T	Open Open	Closed Closed	Yes Yes	Closed Closed	≤10 ≤10
5	Press. Relief Tank Makeup (1)	RCS	30	II	3 3 3	27	IN	Inside Outside Outside	Check Diaphragm Relief	-------- Air --------	-------- T --------	-------- Closed Closed	-------- Closed --------	--- Yes ---	------- Closed -------	--- ≤10 ---
6	Charging Pump Suct. Relief Valve Discharge to Pressurizer Relief Tank(1)	CVCS	59	II	2	34	IN	Inside Outside	Check See Note 1	--------	--------	--------	--------	---	--------	---
7	Reactor Coolant Drain Tank Drain(1)	WPS	31	II	3 3 3 3	1	OUT	Inside Inside Outside Outside	Globe Diaphragm Relief Diaphragm	Air Manual -------- Air	T -------- -------- T	Open Locked Closed Open	Closed Closed -------- Closed	Yes No --- Yes	Closed Closed ------- Closed	≤10 --- --- ≤10
8	Containment Differential Pressure Instrument (1)	H&V	70	II	1 1	16	---	Inside Outside	Globe Globe	Elec.Mtr Elec.Mtr	T T	Open Open	As Is As Is	Yes Yes	Closed Closed	≤15 ≤15
9	Residual Heat Removal A & B Loop Pump Suct(2)	RHRS	16,18	I	12	5	OUT	Inside Outside	Gate See Note 2	Elec.Mtr	Remote Manual	Closed	As Is	Yes	Closed	≤120
10	Residual Heat Removal A & B Pump Disc.(2)	RHRS SIS	15,17	I	10 10	30	IN	Inside Outside	Check Gate	-------- Elec.Mtr	-------- Remote Manual	-------- Open	-------- As Is	--- Yes	-------- Open	N/A
11	Normal Letdown Line (1)	CVCS	23	I	2 3	33	OUT	Inside Inside Outside	Globe Relief Globe	Air -------- Air	T -------- T	Open Closed Open	Closed -------- Closed	Yes --- Yes	Closed -------- Closed	≤10 --- ≤10
12	Excess Letdown and RCP Seal Water Return (1)	CVCS	28	I	3 3 3/4	6	OUT	Inside Outside Inside	Gate Gate Check	Elec.Mtr Elec.Mtr --------	T T ---	Open Open --------	As Is As Is --------	Yes Yes ---	Closed Closed --------	≤15 ≤15 ---

a. Reference table 6.2-32, sheet 2; table 6.3-38; 6.2-39

TABLE 6.2-31 (SHEET 2 OF 7)(a)

Item No.	Service (No. of Penetrations)	System	Penetration No.	Penetration Type	Penetration Line Size (in.)	Valve Arrangement	Flow Direction	Location Relative to Containment	Valve Type	Actuator	Signal	Normal Valve Position	Valve Pos. with Power Failure	Pos. Ind.	Post LOCA Position	Valve Closure Time (sec)
13	Normal Charging Line (1)	CVCS	24	I	3" 3" 3"	7 See Note 3	IN	Inside Outside Outside	Check Gate Gate	-------- Elec. Mtr Elec. Mtr	---- S S	-------- Open Open	-------- As Is As Is	---- Yes Yes	-------- Closed Closed	 <10 <10
14	Reactor Coolant Pump Seal Water Supply (3)	CVCS	25,26,27	I	2"	15	IN	Inside Outside	Check See Note 4	--------	----	--------	--------	---	--------	---
15	Containment Pressure Pressure Instrument (6)	----	73,74,75, 76,65,97A	II		See Note 5										
16	Pressurizer Steam Sample Line (1)	SS	56	I	3/8" 3/8"	24	OUT	Inside Outside	Globe Globe	Solenoid Solenoid	T T	Closed Closed	Closed Closed	Yes Yes	Closed Closed	≤10 ≤10
17	Pressurizer Liquid Sample Line (1)	SS	57	I	3/8" 3/8"	24	OUT	Inside Outside	Globe Globe	Solenoid Solenoid	T T	Closed Closed	Closed Closed	Yes Yes	Closed Closed	≤10 ≤10
18	RCS Hot Leg Sample Line (1)	SS	58	I	3/8" 3/8"	24	OUT	Inside Outside	Globe Globe	Solenoid Solenoid	T T	Closed Closed	Closed Closed	Yes Yes	Closed Closed	≤10 ≤10
19	Fuel Transfer Tube (1)	FHS	14	II	36"	26	---	Inside	Blind Flange	----	----	Closed	------	No	Closed	---
20	Service Air & Breathing Air (1)	SA	47	II	2" 2"	10	IN	Inside Outside	Globe Globe	Manual Manual	---- ----	Locked Closed Locked Closed	As Is As Is	No No	Closed Closed	
20a	Breathing Air (1)	SA	79	II	2" 2"	10 See Note 9	IN	Inside Outside	Globe Globe	Manual Manual	---- ----	Locked Closed Locked Closed	As Is As Is	No No	Closed Closed	
21	Instrument Air (1)	IA	48	II	2" 2"	23	IN	Inside Outside	Check Globe	---- Air	---- P	-------- Open	---- Closed	--- Yes	------ Closed	 <10
22	Containment Air Sample Out (1)	SS	55	II	1" 1"	23	IN	Inside Outside	Check Globe	---- Air	---- T	----- Open	------ Closed	--- Yes	------- Closed	 <10
23	Containment Air Sample In (1)	SS	54	II	1" 1"	11	OUT	Inside Outside	Globe Globe	Elec. Mtr Air	T T	Open Open	As Is Closed	Yes Yes	Closed Closed	≤15 ≤10
24	Containment Main Purge/Supply (1)	H&V	12	II	48" 48"	12	IN	Inside Outside	Butterfly Butterfly	Air Air	T T	Closed Closed	Closed Closed	Yes Yes	Closed Closed	5 5
24a	Containment Mini Purge Supply (1)	H&V	12a	See Note 8	8" 8"	12	IN	Inside Outside	Butterfly Butterfly	Air Air	T T	Open Open	Closed Closed	Yes Yes	Closed Closed	5 5

a. Reference table 6.2-32, Sheet 2; table 6.2-38; 6.2-39

TABLE 6.2-31 (SHEET 3 OF 7)[a]

Item No.	Service (No. of Penetrations)	System	Penetration No.	Penetration Type	Penetration Line Size (in.)	Valve Arrangement	Flow Direction	Location Relative to Containment	Valve Type	Actuator	Signal	Normal Valve Position	Valve Pos. with Power Failure	Pos. Ind.	Post LOCA Position	Valve Closure Time (s)
25	Containment Main Purge Exhaust (1)	H&V	13	II	48” 48”	13	OUT	Inside Outside	Butterfly Butterfly	Air Air	T T	Closed Closed	Closed Closed	Yes Yes	Closed Closed	5 5
25a	Containment Mini Purge Exhaust (1)	H&V	13a	See Note 8	8” 8”	13	OUT	Inside Outside	Butterfly Butterfly	Air Air	T T	Open Open	Closed Closed	Yes Yes	Closed Closed	5 5
26	Containment Sump Pump Discharge (1)	WPS	78	II	3” 3” 3/4”	39	OUT	Inside Outside Inside	Globe Globe Check	Air Air -------	T T -----	Open Open ------	Closed Closed ------	Yes Yes ---	Closed Closed -------	≤10 ≤10 -----
27	Service Wtr Supply to Containment Coolers(4)	SWS	34,35,36,37	III	12” 12”	30	IN	Inside Outside Outside	Check Gate Relief	------- Elec Motor	----- S	------ Open	------ As Is	--- Yes	------- Open	----- 75
28	Service Wtr Return From Containment Coolers (4)	SWS	38,39,40,41	III	10” 10” 6”	9	OUT	Inside Outside Outside Outside	Gate Gate Gate Relief	Elec Motor Elec Motor Elec Motor	S S Remote Manual	Open Closed Open	As Is As Is As Is	Yes Yes Yes	Open Open Open	65 65 34
29	CCW Supply To RCPThermal Barriers and Oil Coolers (1)	CCS	42	II	6” 6”	30	IN	Inside Outside Outside	Check Gate Relief	------- Elec Motor	----- P	------ Open	------ As Is	--- Yes	------- Closed	----- ≤15
30	Leak Rate Test (2)	----	71,72	II	8” 8”	21	IN	Inside Outside	Blind Flange Globe	------- Elec Motor	------- Remote Manual	Closed Closed	Closed As Is	No Yes	Closed Closed	 <10
31	CCW Return From RCP Oil Coolers (1)	CCS	44	II	6” 6”	29	OUT	Inside Outside Outside	Gate Gate Relief	Elec Motor Elec Motor	P P	Open Open	As Is As Is	Yes Yes	Closed Closed	<36[b] ≤15
32	Reactor Coolant Pmp Thermal Barrier Cooling Water Return (1)	CCS	43	II	3” 3”	24	OUT	Inside Outside	Globe Globe	Air Air	P P	Open Open	NOTE 11 NOTE 11	Yes Yes	Closed Closed	≤10 ≤10
33	Excess Letdown Heat Exchanger and RC Drain Tank Heat Exchanger Component Cooling Water Supply (1)	CCS	45	III	6” 6”	8	IN	Inside Outside Outside	Check Globe Relief	------- Air	----- T	------ Open	------ Closed	--- Yes	------- Closed	 <10

a. Reference table 6.2-32, Sheet 2; table 6.2-38; 6.2-39.

b. In accordance with MDC 1053056701, Q1P17MOV3046 isolation time has been changed to < 36 s.

TABLE 6.2-31 (SHEET 4 OF 7)[a]

Item No.	Service (No. of Penetrations)	System	Penetration No.	Penetration Type	Penetration Line Size (in.)	Valve Arrangement	Flow Direction	Location Relative to Containment	Valve Type	Actuator	Signal	Normal Valve Position	Valve Pos. with Power Failure	Pos. Ind.	Post LOCA Position	Valve Closure Time (sec)
34	Excess Letdown Heat Exchanger and RC Drain Tank Heat Exchanger, Component Cooling Water Return (1)	CCS	46	III	6" 6"	24	OUT	Inside Outisde Outside	Globe Globe Relief	Air Air	T T	Open Open	Closed Closed	Yes Yes	Closed Closed	≤10 <10
35	High Head Safety InjectionLine (2) To Hot Legs	SIS	80, 81	I	2" 3"	14	IN	Inside Outside	Check Gate	---------- Elec. Mtr.	--------- Remote Manual	-------- Closed	------- As Is	--- Yes	------- Open	---- N/A
35A	Alternate HHSI Cold Leg Injection Line (1)	SIS	20	I	2" 1"	14"	IN	Inside Outside	Check Gate	---------- Elec. Mtr.	--------- Remote Manual	-------- Closed	------- As Is		------- Open	 ≤20
36	Containment Spray Line (2)	SIS	21, 22	II	8" 8"	30	IN	Inside Outside	Check Gate	---------- Elec. Mtr.	--------- CSAS	-------- Closed	------- As Is	--- Yes	------- Open	----- ≤12
37	Containment Sump Recirculation Line to Low Head Injection Pumps (2)	SIS	10, 11	II See Note 6	14" 14"	18	OUT	Outside Outside	Gate Gate	Elec. Mtr. Elec. Mtr.	Remote Manual Remote Manual	Closed Closed	As Is As Is	Yes Yes	Open Open	<17 <17
38	Containment Sump Recirculation Lines to Ctmt Spray Pumps (2)	SIS	93, 94	II See Note 6	12" 12"	18	OUT	Outside Outside	Gate Gate	Elec. Mtr. Elec. Mtr.	Remote Manual Remote Manual	Closed Closed	As Is As Is	Yes Yes	Open Open	<17 <17
39	Accumulator Makeup Line (1)	SIS	49	II	1" 1"	8	IN	Inside Outside	Check Globe	--------- Air	---------- T	-------- Closed	-------- Closed	--- Yes	--------- Closed	----- ≤10
40	Accumulator Sample Line (1)	SS	50	II	3/8" 3/8"	24	OUT	Inside Outside	Globe Globe	Air Air	T T	Closed Closed	Closed Closed	Yes Yes	Closed Closed	≤10 ≤10
41	Pressurizer Pressure Tester (Deadweight) (1)	RCS	64(b)	III	 3/8"	19 See Note 10	---	Inside Outside	See Note 7 Needle	 Manual	 ---------	 Locked Closed	 As Is	 No	 Closed	 ------
42	Reactor Coolant Drain Tank Hydrogen Supply & Vent Header (1)	WDS	62	II	3/4" 3/4"	28	OUT	Inside Outside	Diaphragm Diaphragm	Air Air	T T	Open Open	Closed Closed	Yes Yes	Closed Closed	≤10 ≤10

a. Reference table 6.2-32, Sheet 2; table 6.2-38; 6.2-39

TABLE 6.2-31 (SHEET 5 OF 7)[a]

Item No.	Service (No. of Penetrations)	System	Penetration No.	Penetration Type	Penetration Line Size (in.)	Valve Arrangement	Flow Direction	Location Relative to Containment	Valve Type	Actuator	Signal	Normal Valve Position	Valve Pos. with Power Fail.	Pos. Ind.	Post LOCA Position	Valve Closure Time (s)
43	High-head injection to RCS cold legs(1)	SIS	19	I	2 3	2	In	Inside Outside	Check Gate	------- Elec.mtr	-------- S	--------- Closed	---------- As is	---- Yes	-------- Open	----- ≤10
44	RHR injection to RCS hot legs(1)	SIS	101	I	6 10	17	In	Inside Outside	Check Gate	------- Elec.mtr	-------- Remote manual	--------- Closed	---------- As is	---- Yes	-------- Open	----- NA
45	Service water to reactor coolant coolers(1)	SWS	60	II	6 6	30	In	Inside Outside Outside	Check Butterfly Relief	------- Elec.mtr	-------- S	--------- Open	---------- As is	----- Yes	-------- Closed	----- <36
46	Service water from reactor coolant pump motor air coolers(1)	SWS	32	II	6 6	29	Out	Inside Outside Outside	Butterfly Butterfly Relief	Elec.mtr Elec.mtr	S S	Open Open	As is As is	Yes Yes	Closed Closed	<36[b] <36
47	Containment sump pump sample recirculation line(1)	WPS	33	II	2 2	23	In	Inside Outside	Check Globe	-------- Air	--------- T	--------- Open	---------- Closed	----- Yes	--------- Closed	----- <10
48	Post LOCA containment sample out (2)	SS	61A,67	II	3/4 3/4	37	Out	Inside Outside	Globe Globe	Elec.mtr Elec.mtr	Remote manual Remote manual	Locked closed Locked closed	As is As is	Yes Yes	Closed Closed	
49	Post LOCA containment sample in (2)	SS	61B,66	II	3/4 3/4	20	In	Inside Outside	Globe Globe	Elec.mtr Elec.mtr	Remote manual Remote manual	Locked closed Locked closed	As is As is	Yes Yes	Closed Closed	
50	Post LOCA containment venting (1)	SS	103	II	6 6	16	Out	Inside Outside	Globe Globe	Elec. mtr Elec.mtr	Remote manual Remote manual	Locked closed Locked closed	As is As is	Yes Yes	Closed Closed	
51	Demineralized water (1)	DWS	82	II	3 3	23	In	Inside Outside	Check Globe	-------- Air	------- T	------------ Closed	------ Closed	----- Yes	------- Closed	----- <10
52	Backup air supply to pressurizer PORV & Bypass (1)	IA	97B	II	1/2 3/4 3/4	23A	In	Inside Outside Outside	Check Globe Ball	-------- Air Manual	-------- P Manual	------------ Open Locked closed	------- Closed N/A	----- Yes N/A	------- Closed Closed	----- <10
53	Spare (Used with refueling module during refueling outages) (2)	----	90,92	II	10	40	NA	Inside Outside	Blind Flanges	--------	---------	------------	--------	-----	--------	------

a. Reference table 6.2-32, sheet 2; table 6.2-38; 6.2-39.

b. In accordance with MDCs 1052455201 and 1052455301, Q1P16MOV3134 and Q1P16MOV3135 isolation times have been changed to < 36 s.

TABLE 6.2-31 (SHEET 6 OF 7)

The "Valve Arrangement" number refers to figures 6.2-84 through 6.2-89.

NOTES

1. Item No. 6 - The closed relief valves act as the containment isolation barrier outside the containment.

2. Item No. 9 - Outside containment, the residual heat removal system is also the low head safety injection system. These lines are filled with sump water and at least one of these is in operation post-accident. It is the water seal in the piping outside containment, and not any valves in the RHR system outside containment, which provides the second barrier. This water seal is assured by the physical layout of the piping connected to the RHR pump suction.

3. Item No. 13 - Two motor-operated valves are provided for isolation of the charging line from the high head safety injection system post-LOCA. Only one valve is required for containment isolation.

4. Item No. 14 - The seal water injection lines are considered as open flow paths post-LOCA. The high pressure in-flow through these lines during the injection and recirculation phase precludes containment to atmosphere leakage. In the event of a loss of seal water flow through these lines, a water seal in the charging pump suction and discharge piping precludes containment to atmosphere leakage. In the event that maintenance requires interrupting flow through these lines, isolation can be achieved by closure of the manually operated needle valves outside containment.

5. Item No. 15 - A sealed, fluid-filled pressure instrument forms the isolation barrier both inside and outside the containment.

6. Item Nos. 37, 38 - The first isolation valve shown in Valve Arrangement No. 18 is encapsulated in a leaktight compartment designed for accident conditions. See Section 6.3.

TABLE 6.2-31 (SHEET 7 OF 7)

NOTES

7.	Item No. 41 -	Inside the containment, the tubing is separated from the pressurizer by the diaphragm of the pressurizer pressure sensor, which is designed to withstand full reactor coolant system pressure from either side.
8.	Item No. 24a -	The containment minipurge system uses the same penetration and 25as the penetration used by the main purge system. (See items 24 and 25.)
9.	Item No. 20a -	The arrangement of these two valves is for Unit 2 only. Unit 1 does not have these valves.
10.	Item No. 41 -	The single containment isolation valve required for this line is located outside of the piping penetration room in Unit 1. This is considered acceptable because of the small size of the line (primarily 1/8-in. OD tubing) and the fact that the system is closed inside the containment.
11.	Item No. 32 -	Valve fails open on loss of electrical power with instrument air available.

*The indicated valve response time requirement is based on performing an ESF function, not containment isolation. The ESF function requires the valve to open within the indicated response time. The valve closure time requirement for containment isolation is "N/A."

TABLE 6.2-32 (SHEET 1 OF 2)

STEAM GENERATOR ISOLATION VALVE INFORMATION

Item No.	Service (No. of Penetrations)	System	Penetration Type	Valve Arrangement	Flow Direction	Location Relative to Containment	Valve Type	Penetration Line Size (in.)	Actuator	Signal	Normal Valve Position	Valve Position with Power Failure	Position Indicator	Post LOCA Position	Valve Closure Time(s)
1	Main Steam (3)	MS	III	3	Out	Outside	Power operated Check	32	Air to open Spring closed	SLIAS	Open	As Is	Yes	Closed	7
2	Main Steam Isolation Valve Bypass (3)	MS	III	3	Out	Outside	Gate	3	Air	SLIAS	Closed	Closed	Yes	Closed	< 7 (Note 2)
3	Steam to Auxiliary Feedwater Pump Turbine Drive (2)	MS	III	3	Out	Outside	Stop check	3	Air	Remote manual	Closed	Closed	Yes	Closed	---
4	Steam to Aux. Feedwater Pump Drive Warming Line (2)	MS	III	3	Out	Outside	Globe	1	Air	T	Open	Closed	Yes	Closed	NA
5	Main Steam Atmospheric Relief (3)	MS	III	3	Out	Outside	Globe	6	Air	Remote manual	Closed	Closed	Yes	Closed	< 35
6	Feedwater (3)	FW	III	4	In	Outside	Stop check	14	Electric motor	Remote manual	Open	As is	Yes	Open	30
7	Auxiliary Feedwater (3)	AFW	III	4	In	Outside	Stop check	4	Electric motor	Remote manual	Open	As is	Yes	Open	14
8	Steam Generator Blowdown (3)	MS	III	25	Out	Outside	Globe	2	Air	AFPSS	Open	Closed	Yes	Closed	≤ 60
9	Steam Generator Blowdown Sample (3)	SS	III	32	Out	Outside	Globe	3/8	Air	Remote manual See Note 1 below	Open	Closed	Yes	Closed	< 5
10	Chemical Injection (3)	FW	III	4	In	Outside	Globe	1/2	Air	T	Open	Closed	Yes	Closed	< 5

1. Flow is isolated on AFPSS by valves inside containment.
2. Design requirement only, not operability requirement.

TABLE 6.2-32 (SHEET 2 OF 2)

The "Valve Arrangement" number refers to figures 6.2-84 through 6.2-89.

The abbreviations used in table 6.2-31 and 6.2-32 are as follows:

SYSTEM

DWS	-	Demineralized Water System
CCS	-	Component Cooling System
SIS	-	Safety Injection System
RCS	-	Reactor Coolant System
WPS	-	Waste Processing System
MS	-	Main Steam System
FW	-	Feedwater System
RHRS	-	Residual Heat Removal System
CVCS	-	Chemical and Volume Control System
SS	-	Sampling System
FHS	-	Fuel Handling System
RMS-		Radiation Monitoring System
H&V	-	Heating and Ventilation
SA	-	Service Air System
IA	-	Instrument Air System
FWS-		Fire Water System
SWS	-	Service Water System
AFW-	-	Auxiliary Feedwater System

SIGNALS

T	-	Containment Isolation Actuation Signal, Phase A
S	-	Safety Injection Signal
P	-	Containment Isolation Actuation Signal, Phase B
CSAS	-	Containment Spray Actuation Signal
SLIAS	-	Steam Line Isolation Actuation Signal
AFPSS	-	Auxiliary Feedwater Pump Start Signal

TABLE 6.2-33

ELECTRIC HYDROGEN RECOMBINER TYPICAL PARAMETERS

Parameter	Value
Power (maximum)	75 kW[a]
Capacity (minimum)	100 sft^3/min
Heaters	
-Number	5
-Heater surface area/heater	35 ft^2
-Maximum heat flux	2850 Btu/h-ft^2 or 5.8 Watts/$in.^2$
-Maximum sheath temperature	1550°F
Gas Temperature	
-Inlet	80 to 155°F
-In heater section	1150 to 1400°F
Materials	
-Outer structure	300-Series S.S.
-Inner structure	Inconel-600
-Heater element sheath	Incoloy-800
Dimensions	
-Height	9 ft
-Width	4.5 ft
-Depth	5.5 ft
Weight	4500 lb

a. Power can be controlled by SCR. Normal operating power for typical PWR containments is 48.9.

TABLE 6.2-34

POSTACCIDENT VENTING SYSTEM DESIGN PARAMETERS

Parameters		Value
Valves		
-Design pressure (psig)		150[a]
-Design temperature (°F)	Inside containment	300[a]
	Outside containment	300[a]
Piping		
-Design pressure (psig)		150[a]
-Design temperature (°F)	Inside containment	300[a]
	Outside containment	300[a]
HEPA Filter		
-Number		1
-Air Flow (sft^3/min)		500
-Approximate differential pressure (wg)		1.5
-Maximum differential pressure (loaded) (wg)		4.0
-Design temperature (°F)		180
-Particulate removal efficiency (0.3 micron)		99.97
Charcoal		
-Number		1
-Air flow (sft^3/min)		500
-Differential pressure (wg)		2.7
-Design temperature (°F)		180
-Charcoal type		iodine impregnated
-Elemental I_2 removal efficiency		99.9
-Organic I_2 removal efficiency		99.0

a. Represents as installed ratings of system piping and valves. Actual design requirements may be substantially lower and may vary throughout the system.

TABLE 6.2-35

POSTACCIDENT SAMPLING SYSTEM DESIGN PARAMETERS

<u>Sample Vessel</u>	
Number	2
Number required for operation	1
Design pressure (psig)	150
Design temperature (°F)	300
Material of construction	Stainless steel
<u>Valves</u>	
Design pressure (psig)	150
Design temperature (°F)	300
<u>Piping</u>	
Design pressure (psig)	150
Design temperature (°F)	300

TABLE 6.2-36

POSTACCIDENT MIXING SYSTEM DESIGN PARAMETERS

Post-LOCA Containment Mixing Fans:

	Type	Vaneaxial
	Number required	4
	Flow (ft^3/min) each	7500
	Static pressure (in. w.g.)	2.3

Reactor Cavity Hydrogen Dilution Fans:

Unit 1	Type	Centrifugal
	Number required	2
	Flow (ft^3/min) each	270
	Static pressure (in. w.g.)	126.7
Unit 2	Type	Vaneaxial
	Number required	2
	Flow (ft^3/min) each	1570
	Static pressure (in. w.g.)	3.26

TABLE 6.2-37

CONTAINMENT INTERIOR COATINGS SUMMARY

Surface Type	Manufacturer	Primer/Surfacer Product No.	Generic Type	Dry Specific Gravity	Ext. Top Coat Recoat Product No.	Ext. Top Coat Recoat Generic Type	Dry Specific Gravity	Surface Area (ft²)[a]
Carbon Steel	Ameron[b]	Dimetcote No. 6 (D-6)	Inorganic zinc	3.15	Amercoat 66 Amercoat 90	Epoxy polyamide Modified phenolic	2.60 2.58	213,750
		Amercoat 90	Modified phenolic	2.58	Amercoat 90	Modified phenolic	2.58	
	Carboline	Carbozinc 11 SG (CZ-11)	Inorganic zinc	4.61	Phenoline 305 Amercoat 90	Epoxy phenolic Modified phenolic	1.73 2.58	17,500
		4674 (Black)[c]	Modified silicone with low chloride content	1.345	---	---	---	2,300
		4700	Aluminum free paint	1.28	---	---	---	< 2
		4674 (Aluminum)[c]	Modified silicone aluminum with low chloride content	1.54	---	---	---	12,000
	Sterling	U-475 ERN	Epoxy varnish	1.001[d]	---	---	---	28
	---	Galvanized	Hot dipped zinc	7.15	---	---	---	62,932
Concrete	Ameron[b]	NU-KLAD 110AA	Epoxy polyamide - solid filled	1.95	Amercoat 66 Amercoat 90	Epoxy polyamide Modified phenolic	2.60 2.58	80,000
		Amercoat 90	Modified phenolic	2.58	Amercoat 90	Modified phenolic	2.58	
		Amercoat 3366	Epoxy surfacer	2.12	Amercoat 90HS	Epoxy phenolic	1.72	
		Amercoat 3367	Epoxy filler	1.80	Amercoat 90HS	Epoxy phenolic	1.72	

a. For coating requirements see NMP-MA-011, Nuclear Coatings Program.
b. Either system is acceptable for use as original system.
c. Generally covered by insulation.
d. Wet specific gravity at 75°F.

TABLE 6.2-38 (SHEET 1 OF 4)

CONTAINMENT PENETRATIONS

Penetration No.	Type Test	Purpose
1	A	Main steam from SG A [SG C][(b)]
2	A	Main steam from SG B [SG B]
3	A	Main steam from SG C [SG A]
4	A	Feedwater from SG A [SG C]
5	A	Feedwater from SG B [SG B]
6	A	Feedwater from SG C [SG A]
7	A	Steam generator blowdown from SG C [SG A]
8	A	Steam generator blowdown from SG B [SG B]
9	A	Steam generator blowdown from SG A [SG C]
10	A	Containment sump recirculation line to low-head injection pump B
11	A	Containment sump recirculation line to low-head injection pump A
12	C	Containment main purge supply
12a	C	Containment mini purge supply
13	C	Containment main purge exhaust
13a	C	Containment mini purge exhaust
14	B	Fuel transfer tube
15	A	Residual heat removal A loop pump disc
16	A	Residual heat removal A loop pump suction
17	A	Residual heat removal B loop pump disc
18	A	Residual heat removal B loop pump suction
19	A	High-head injection to RCS cold legs
20	A	High-head injection to RCS cold legs
21	A	Containment spray A
22	A	Containment spray B
23	A	Letdown line
24	A	Charging line
25	A	Reactor coolant pump seal water-in RCP B
26	A	Reactor coolant pump seal water-in RCP C
27	A	Reactor coolant pump seal water-in RCP A
28	A	Reactor coolant pump seal water return and excess letdown line
29	C	Accumulators test line
30	C	Pressurizer relief tank make-up
31	C	Reactor coolant drain tank drain
32	C	Service water return from reactor coolant pump motor air coolers
33	C	Containment sump pump sample recirculation

TABLE 6.2-38 (SHEET 2 OF 4)

Penetration No.	Type Test	Purpose
34	A	Service water supply to containment cooler A
35	A	Service water supply to containment cooler B
36	A	Service water supply to containment cooler C
37	A	Service water supply to containment cooler D
38	A	Service water return from containment cooler A
39	A	Service water return from containment cooler B
40	A	Service water return from containment cooler D
41	A	Service water return from containment cooler C
42	A	CCW supply to RCP thermal barriers and oil coolers
43	A	CCW return from RCP thermal barriers
44	A	CCW return from RCP oil coolers
45	A	CCW supply to excess letdown HX and RCDT HX
46	A	CCW return from excess letdown HX and RCDT HX
47[c]	C	Service air and breathing air
48	C	Instrument air
49	C	Accumulator make-up line
50	C[a]	Accumulator sample line
51	A	Steam generator blowdown sample SG A
52	A	Steam generator blowdown sample SG B
53	A	Steam generator blowdown sample SG C
54	C	Containment air sample-out
55	C	Containment air sample-in
56	C	Pressurizer steam sample line
57	C[a]	Pressurizer liquid sample line
58	C	RCS hot leg sample line
59	C	Relief valve discharge to pressurizer relief tank
60	C	Service water supply to reactor coolant pump motor air coolers
61A	A	Post-accident containment sample-out
61B	A	Post-accident containment sample-in

TABLE 6.2-38 (SHEET 3 OF 4)

Penetration No.	Type Test	Purpose
62	C	Hydrogen supply and vent header reactor coolant drain tank
63	C	Nitrogen supply-accumulator tanks
64A	C[(a)]	Nitrogen supply-pressurizer relief tank
64B	A	Pressurizer deadweight pressure tester
65	A	Containment pressure instrument
66	A	Post-accident containment sample in
67	A	Post-accident containment sample out
68	A	Spare
69	A	Spare
70	C	Containment differential pressure inst.
71	C	Leak rate test
72	C	Leak rate test
73	A	Containment pressure instrument
74	A	Containment pressure instrument
75	A	Containment pressure instrument
76	A	Containment pressure instrument
77	A	Spare
78	C	Containment sump pumps discharge
79[(d)]	A	Spare
80	A	High-head injection to RCS hot legs
81	A	High-head injection to RCS hot legs
82	C	Demineralized water
83	A	Spare
84	B	Equipment hatch
85	B	Electrical
86	B	Personnel lock
87	B	Auxiliary access lock
88	A	Electrical ground
89	A	Spare
90	B	Spare[(e)]
91	A	Spare
92	B	Spare[(e)]
93	A	Containment sump-recirculating line to containment spray pump B
94	A	Containment sump-recirculating line to containment spray pump A
95	C	Refueling cavity supply
96	A	Spare
97A	A	CTMT pressure instrument
97B	C	Backup air supply to pressurizer PORV
98	A	Spare
99	A	Spare

TABLE 6.2-38 (SHEET 4 OF 4)

Penetration No.	Type Test	Purpose
100	A	Spare
101	A	RHR injection to RCS hot legs
102	A	Spare
103	C	Post-accident containment venting
104	A	Spare

a. Piping system may remain water filled during the Type B or Type C test. All other penetrations which are local leak-rate tested must be drained.

b. [ ] indicates Unit 2 service.

c. Unit 2 - Type C test, service air.

d. Unit 2 - Type C test, breathing air.

e. To be used with refueling module during refueling outages.

TABLE 6.2-39 (SHEET 1 OF 7)

CONTAINMENT ISOLATION VALVES
(Reference: Table 6.2-31 [See Note])

No.	Location	FSAR Figure	System P&ID	Valve Identification Number
1	ic	6.2-87 (24)	D-175038, Sh. 2	Q1E21V049
			D-205038, Sh. 2	Q2E21V049
	oc	6.2-87 (24)	D-175038, Sh. 2	Q1E21V050
			D-205038, Sh. 2	Q2E21V050
2	ic	6.2-89 (35)	D-175043[(a)]	Q1G31V013
			D-205043	Q2G31V013
	oc	6.2-89 (35)	D-175043[(a)]	Q1G31V012
			D-205043	Q2G31V012
3	ic	6.2-87 (23)	D-175038, Sh. 2	Q1E21V058
			D-205038, Sh. 2	Q2E21V058
	oc	6.2-87 (23)	D-175038, Sh. 2	Q1E21V059
			D-205038, Sh. 2	Q2E21V059
4	ic	6.2-89 (38)	D-175037, Sh. 2	Q1B13V037
			D-205037, Sh. 2	Q2B13V037
	oc	6.2-89 (38)	D-175037, Sh. 2	Q1B13V039
			D-205037, Sh. 2	Q2B13V039
5	ic	6.2-87 (27)	D-175037, Sh. 2	Q1B13V038
			D-205037, Sh. 2	Q2B13V038
	oc	6.2-87 (27)	D-175037, Sh. 2	Q1B13V040
			D-205037, Sh. 2	Q2B13V040
			D-175037, Sh. 2	Q1B13V110
			D-205037, Sh. 2	Q2B13V110
6	ic	6.2-88 (34)	D-175037, Sh. 2	Q1B13V054
			D-205037, Sh. 2	Q2B13V054
	oc	6.2-88 (34)	D-175039, Sh. 6	Q1E21V263A,B
			D-205039, Sh. 2	Q2E21V263A,B
	oc	6.2-88 (34)	D-175038, Sh. 2	Q1E11V039A,B
			D-175038, Sh. 2	Q1E11V040
			D-205038, Sh. 2	Q2E11V039A,B
			D-205038, Sh. 2	Q2E11V040
7	ic	6.2-84 (1)	D-175042, Sh. 1	Q1G21V064
			D-175042, Sh. 1	Q1G21V005
			D-205042, Sh. 1[(a)]	Q2G21V064
			D-205042, Sh. 1[(a)]	Q2G21V005
	oc	6.2-84 (1)	D-175042, Sh. 1	Q1G21V006
			D-205042, Sh. 1[(a)]	Q2G21V006
			D-175042, Sh. 1	Q1G21V950
			D-205042, Sh. 1[(a)]	Q2G21V950

a. This drawing is not presented in the FSAR because the corresponding drawing is applicable to both units.
Note: Item numbers correlate with those on Table 6.2-31.

TABLE 6.2-39 (SHEET 2 OF 7)

No.	Location	FSAR Figure	System P&ID	Valve Identification Number
8	ic	6.2-86 (16)	D-175010, Sh. 2	Q1E14V004
			D-205010, Sh. 2	Q2E14V004
	oc	6.2-86 (16)	D-175010, Sh. 2	Q1E14V003
			D-205010, Sh. 2	Q2E14V003
9	ic	6.2-85 (5)	D-175041	Q1E11V001A,B
			D-205041	Q2E11V001A,B
	oc	See note 2 of FSAR table 6.2-31.		
10	ic	6.2-88 (30)	D-175038, Sh. 2	Q1E11V042A,B
			D-205038, Sh. 2	Q2E11V042A,B
	oc	6.2-88 (30)	D-175038, Sh. 2	Q1E11V023A,B
			D-205038, Sh. 2	Q2E11V023A,B
11	ic	6.2-88 (33)	D-175039, Sh. 1	Q1E21V253A,B,C
			D-205039, Sh. 1	Q2E21V253A,B,C
			D-175039, Sh. 1	Q1E21V255
			D-205039, Sh. 1	Q2E21V255
	oc	6.2-88 (33)	D-175039, Sh. 1	Q1E21V254
			D-205039, Sh. 1	Q2E21V254
12	ic	6.2-85 (6)	D-175039, Sh. 1	Q1E21V249A
			D-175039, Sh. 1	Q1E21V213
			D-205039, Sh. 1	Q2E21V249A
			D-205039, Sh. 1	Q2E21V213
	oc	6.2-85 (6)	D-175039, Sh. 1	Q1E21V249B
			D-205039, Sh. 1	Q2E21V249B
13	ic	6.2-85 (7)	D-175039, Sh. 1	Q1E21V119
			D-205039, Sh. 1	Q2E21V119
	oc	6.2-85 (7)	D-175039, Sh. 6	Q1E21V257
			D-175039, Sh. 6	Q1E21V258
			D-205039, Sh. 2	Q2E21V257
			D-205039, Sh. 2	Q2E21V258
14	ic	6.2-86 (15)	D-175039, Sh. 1	Q1E21V115A,B,C
			D-205039, Sh. 1	Q2E21V115A,B,C
		See note 4 of FSAR table 6.2-31.		
	oc	6.2-86 (15)	D-175039, Sh. 1	-
			D-205039, Sh. 2	-
		See note 4 of FSAR table 6.2-31.		
15	ic		D-175038, Sh. 3	-
			D-205038, Sh. 3	-
		See note 5 of FSAR table 6.2-31.		
	oc		D-175038, Sh. 3	-
			D-205038, Sh. 3	-
		See note 5 of FSAR table 6.2-31.		

TABLE 6.2-39 (SHEET 3 OF 7)

No.	Location	FSAR Figure	System P&ID	Valve Identification Number
16	ic	6.2-87 (24)	D-175009, Sh. 2	Q1P15SV3104
			D-205009, Sh. 2	Q2P15SV3104
	oc	6.2-87 (24)	D-175009, Sh. 2	Q1P15SV3331
			D-205009, Sh. 2	Q2P15SV3331
17	ic	6.2-87 (24)	D-175009, Sh. 1	Q1P15SV3103
			D-205009, Sh. 1	Q2P15SV3103
	oc	6.2-87 (24)	D-175009, Sh. 1	Q1P15SV3332
			D-205009, Sh. 1	Q2P15SV3332
18	ic	6.2-87 (24)	D-175009, Sh. 1	Q1P15SV3765
			D-205009, Sh. 1	Q2P15SV3765
			D-175009, Sh. 1	Q1P15SV3333
			D-205009, Sh. 1	Q2P15SV3333
19	ic	6.2-87 (26)	D-175067[b]	-
			D-205067[a][b]	-
20	ic	6.2-87 (10)	D-175035, Sh. 1	Q1P18V002
			D-205035, Sh. 1	Q2P18V002
	oc	6.2-87 (10)	D-175035, Sh. 1	Q1P18V001
			D-205035, Sh. 1	Q2P18V001
20a	ic	6.2-87 (10)	D-205035, Sh. 1	Q2P18V005
	oc	6.2-87 (10)	D-205035, Sh. 1	Q2P18V004
21	ic	6.2-87 (23)	D-175034, Sh. 3	Q1P19V002
			D-205034, Sh. 4	Q2P19V002
	oc	6.2-87 (23)	D-175034, Sh. 2	Q1P19HV3611
			D-205034, Sh. 2	Q2P19HV3611
22	ic	6.2-87 (23)	D-175010, Sh. 2	Q1E14V001
			D-205010, Sh. 2	Q2E14V001
	oc	6.2-87 (23)	D-175010, Sh. 2	Q1E14HV3657
			D-205010, Sh. 2	Q2E14HV3657
23	ic	6.2-85 (11)	D-175010, Sh. 2	Q1E14V002
			D-205010, Sh. 2	Q2E14V002
	oc	6.2-85 (11)	D-175010, Sh. 2	Q1E14HV3658
			D-175010, Sh. 2	Q2E14HV3658
24	ic	6.2-85 (12)	D-175010, Sh. 1	Q1P13V282
			D-205010, Sh. 1	Q2P13V282
	oc	6.2-85 (12)	D-175010, Sh. 2	Q1P13V281
			D-205010, Sh. 2	Q2P13V281

b. This is only a general arrangement drawing.

TABLE 6.2-39 (SHEET 4 OF 7)

No.	Location	FSAR Figure	System P&ID	Valve Identification Number
24a	ic	6.2-85 (12)	D-175010, Sh. 1	Q1P13V302
			D-205010, Sh. 1	Q2P13V302
	oc	6.2-85 (12)	D-175010, Sh. 2	Q1P13V301
			D-205010, Sh. 2	Q2P13V301
25	ic	6.2-85 (13)	D-175010, Sh. 1	Q1P13V283
			D-205010, Sh. 1	Q2P13V283
	oc	6.2-85 (13)	D-175010, Sh. 2	Q1P13V284
			D-205010, Sh. 2	Q2P13V284
25a	ic	6.2-85 (13)	D-175010, Sh. 1	Q1P13V304
			D-205010, Sh. 1	Q2P13V304
	oc	6.2-85 (13)	D-175010, Sh. 2	Q1P13V303
			D-205010, Sh. 2	Q2P13V303
26	ic	6.2-89 (39)	D-175004, Sh. 1	Q1G21V291
			D-175004, Sh. 1	Q1G21HV3376
			D-205004, Sh. 1[(a)]	Q2G21V291
			D-205004, Sh. 1[(a)]	Q2G21HV3376
	oc	6.2-89 (39)	D-175004, Sh. 1	Q1G21HV3377
			D-205004, Sh. 1[(a)]	Q2G21HV3377
27	ic	6.2-88 (30)	D-175003, Sh. 1	Q1P16V206A,B,C,D
			D-205003, Sh. 1[(a)]	Q2P16V206A,B,C,D
	oc	6.2-88 (30)	D-175003, Sh. 1	Q1P16V010A,B,C,D
			D-205003, Sh. 1[(a)]	Q2P16V010A,B,C,D
			D-175003, Sh. 1	Q1P16V205A,B,C,D
			D-205003, Sh. 1[(a)]	Q2P16V205A,B,C,D
28	ic	6.2-85 (9)	D-175003, Sh. 1	Q1P16V207A,B,C,D
			D-205003, Sh. 1[(a)]	Q2P16V207A,B,C,D
	oc	6.2-85 (9)	D-175003, Sh. 1	Q1P16V043A,B,C,D
			D-205003, Sh. 1[(a)]	Q2P16V043A,B,C,D
			D-175003, Sh. 1	Q1P16V044A,B,C,D
			D-205003, Sh. 1[(a)]	Q2P16V044A,B,C,D
			D-175003, Sh. 1	Q1P16V208A,B,C,D
			D-205003, Sh. 1[(a)]	Q2P16V208A,B,C,D
29	ic	6.2-88 (30)	D-175002, Sh. 2	Q1P17V083
			D-205002, Sh. 2	Q2P17V083
	oc	6.2-88 (30)	D-175002, Sh. 2	Q1P17V082
			D-205002, Sh. 2	Q2P17V082
			D-175002, Sh. 2	Q1P17V158
			D-205002, Sh. 2	Q2P17V158
30	oc	6.2-86 (21)	D-175010, Sh. 1	Q1P23V002A,B
			D-205010, Sh. 1	Q2P23V002A,B

TABLE 6.2-39 (SHEET 5 OF 7)

No.	Location	FSAR Figure	System P&ID	Valve Identification Number
31	ic	6.2-88 (29)	D-175002, Sh. 2	Q1P17V097
			D-205002, Sh. 2	Q2P17V097
	oc	6.2-88 (29)	D-175002, Sh. 2	Q1P17V099
			D-205002, Sh. 2	Q2P17V099
			D-175002, Sh. 2	Q1P17V155
			D-205002, Sh. 2	Q2P17V155
32	ic	6.2-87 (24)	D-175002, Sh. 2	Q1P17HV3184
			D-205002, Sh. 2	Q2P17HV3184
	oc	6.2-87 (24)	D-175002, Sh. 2	Q1P17HV3045
			D-205002, Sh. 2	Q2P17HV3045
33	ic	6.2-85 (8)	D-175002, Sh. 2	Q1P17V159
			D-205002, Sh. 2	Q2P17V159
	oc	6.2-85 (8)	D-175002, Sh. 2	Q1P17HV3095
			D-205002, Sh. 2	Q2P17HV3095
			D-175002, Sh. 2	Q1P17V153
			D-205002, Sh. 2	Q2P17V153
34	ic	6.2-87 (24)	D-175002, Sh. 2	Q1P17HV3443
			D-205002, Sh. 2	Q2P17HV3443
	oc	6.2-87 (24)	D-175002, Sh. 2	Q1P17HV3067
			D-205002, Sh. 2	Q2P17HV3067
			D-175002, Sh. 2	Q1P17V154
			D-205002, Sh. 2	Q2P17V154
35	ic	6.2-86 (14)	D-175038, Sh. 1	Q1E21V078A,B,C
			D-205038, Sh. 1	Q2E21V078A,B,C
			D-175038, Sh. 1	Q1E21V079A,B,C
			D-205038, Sh. 1	Q2E21V079A,B,C
			D-175038, Sh. 1	Q1E21V066A,B,C
			D-205038, Sh. 1	Q2E21V066A,B,C
	oc	6.2-86 (14)	D-175038, Sh. 1	Q1E21V068
			D-175038, Sh. 1	Q1E21V072
			D-175038, Sh. 1	Q1E21V063
			D-205038, Sh. 1	Q2E21V068
			D-205038, Sh. 1	Q2E21V072
			D-205038, Sh. 1	Q2E21V063
36	ic	6.2-88 (30)	D-175038, Sh. 3	Q1E13V002A,B
			D-205038, Sh. 3	Q2E13V002A,B
	oc	6.2-88 (30)	D-175038, Sh. 3	Q1E13V005A,B
			D-205038, Sh. 3	Q2E13V005A,B
37	oc	6.2-86 (18)	D-175038, Sh. 2	Q1E11V025A,B
			D-205038, Sh. 2	Q2E11V025A,B
			D-175038, Sh. 2	Q1E11V026A,B
			D-205038, Sh. 2	Q2E11V026A,B

TABLE 6.2-39 (SHEET 6 OF 7)

No.	Location	FSAR Figure	System P&ID	Valve Identification Number
38	oc	6.2-86 (18)	D-175038, Sh. 3	Q1E13V003A,B
			D-205038, Sh. 3	Q2E13V003A,B
			D-175038, Sh. 3	Q1E13V004A,B
			D-205038, Sh. 3	Q2E13V004A,B
39	ic	6.2-85 (39)	D-175038, Sh. 2	Q1E21V052
			D-205038, Sh. 2	Q2E21V052
	oc	6.2-85 (39)	D-175038, Sh. 2	Q1E21V091
			D-205038, Sh. 2	Q2E21V091
40	ic	6.2-87 (24)	D-175009, Sh. 1	Q1P15HV3766
			D-205009, Sh. 1	Q2P15HV3766
	oc	6.2-87 (24)	D-175009, Sh. 1	Q1P15HV3334
			D-205009, Sh. 1	Q2P15HV3334
41	ic	See note 7 of FSAR table 6.2-31.		
	oc	6.2-86 (19)	D-175037, Sh. 2	Q1B13V026B
			D-205037, Sh. 2	Q2B13V026B
42	ic	6.2-87 (28)	D-175042, Sh. 1	Q1G21V082
			D-205042, Sh. 1[(a)]	Q2G21V082
	oc	6.2-87 (28)	D-175042, Sh. 1	Q1G21V001
			D-205042, Sh. 2[(a)]	Q2G21V001
43	ic	6.2-84 (2)	D-175038, Sh. 1	Q1E21V062A,B,C
			D-205038, Sh. 1	Q2E21V062A,B,C
	oc	6.2-84 (2)	D-175038, Sh. 1	Q1E21V016A,B
			D-205038, Sh. 1	Q2E21V016A,B
44	ic	6.2-86 (17)	D-175038, Sh. 1	Q1E21V076A,B
			D-205038, Sh. 1	Q2E21V076A,B
	oc	6.2-86 (17)	D-175038, Sh. 2	Q1E11V044
			D-205038, Sh. 2	Q2E11V044
45	ic	6.2-88 (30)	D-175003, Sh. 2	Q1P16V075
			D-205003, Sh. 2[(a)]	Q2P16V075
	oc	6.2-88 (30)	D-175003, Sh. 2	Q1P16V071
			D-205003, Sh. 2[(a)]	Q2P16V071
			D-175003, Sh. 2	Q1P16V204
			D-205003, Sh. 2[(a)]	Q2P16V204
46	ic	6.2-88 (29)	D-175003, Sh. 2	Q1P16V081
			D-205003, Sh. 2[(a)]	Q2P16V081
	oc	6.2-88 (29)	D-175003, Sh. 2	Q1P16V072
			D-205003, Sh. 2[(a)]	Q2P16V072
			D-175003, Sh. 2	Q1P16V203
			D-205003, Sh. 2[(a)]	Q2P16V203
47	ic	6.2-87 (23)	D-175004, Sh. 1	Q1G21V204
			D-205004, Sh. 1[(a)]	Q2G21V204
	oc	6.2-87 (23)	D-175004, Sh. 1	Q1G21HV3380
			D-205004, Sh. 1[(a)]	Q2G21HV3380

TABLE 6.2-39 (SHEET 7 OF 7)

No.	Location	FSAR Figure	System P&ID	Valve Identification Number
48	ic	6.2-89 (37)	D-175019	Q1E23V022A,B,C,D
			D-205019	Q2E23V022A,B,C,D
	oc	6.2-89 (37)	D-175019	Q1E23V023A,B
			D-205019	Q2E23V023A,B
49	ic	6.2-86 (20)	D-175019	Q1E23V025A,B
			D-205019	Q2E23V025A,B
	oc	6.2-86 (20)	D-175019	Q1E23V024A,B
			D-205019	Q2E23V024A,B
50	ic	6.2-86 (16)	D-175019	Q1E23V003
			D-205019	Q2E23V003
	oc	6.2-86 (16)	D-175019	Q1E23V002
			D-205019	Q2E23V002
51	ic	6.2-87 (23)	D-175047	Q1P11V002
			D-205047	Q2P11V002
	oc	6.2-87 (23)	D-175047	Q1P11V001
			D-205047	Q2P11V001
52	ic	6.2-87 (23)	D-175034, Sh. 1	Q1P19V004
			D-205034, Sh. 4[a]	Q2P19V004
	oc	6.2-87 (23)	D-175034, Sh. 1	Q1P19HV2228
		6.2-87 (23A)	D-205034, Sh. 4[a]	Q1P19V1099
				Q2P19V006
				Q2P19V1099
53	ic	6.2-89 (40)	D-206164[b]	---
	oc	6.2-89 (40)	D-206164[b]	---

a. This drawing is not presented in the FSAR because the corresponding drawing is applicable to both units.

b. This is only a general arrangement drawing.

TABLE 6.2-40

STEAM GENERATOR ISOLATION VALVES

(Reference: TABLE 6.2-32 [See Note])

Item No.	Location	FSAR Figure	System P&ID	Valve Identification Number
1	oc	6.2-84 (3)	D-175033 Sh. 1	QV001A,B,C QV002A,B,C
2	oc	6.2-84 (3)	D-175033 Sh. 1	QV003A,B,C,D,E,F
3	oc	6.2-84 (3)	D-175033 Sh. 2	Q-N12V001A-A,B-B
4	oc	6.2-84 (3)	D-175033 Sh. 2	HV3234A,B
5	oc	6.2-84 (3)	D-175033 Sh. 1	PV3371A,B,C
6	oc	6.2-84 (4)	D-170117 Sh. 4	Q-N21V001A-B, B-B,C-B
7	oc	6.2-84 (4)	D-175007	V0011A,B,C
8	oc	6.2-87 (25)	D-175071 Sh. 1 D-205071 Sh. 1	7614A,B,C
9	oc	6.2-88 (32)	D-175009 Sh. 2 D-205009 Sh. 2	HV3328, HV3329, HV3330
10	oc	6.2-84 (4)	D-175000 Sh. 1	QV001A,B,C

Note: Item numbers correlate with those on Table 6.2-32.

TABLE 6.2-41

CONTAINMENT PRESSURE/TEMPERATURE FOR 600 gal/min SERVICE WATER FLOW, 0.003 FOULING FACTOR

Power Uprate Case	Peak Pressure (psia)	Time (s)	Peak Temp. (°F)	Time (s)
MSLB CASE 1, $P_0 = 0.0$	58.3	1811	368	60.1
MSLB CASE 1, $P_0 = -1.5$	56.6	1811	383	100.1
MSLB CASE 2	55.7	1811	355	150.1
MSLB CASE 3, $P_0 = 0.0$	55.4	1811	362	170.1
MSLB CASE 3, $P_0 = -1.5$	53.7	1811	370	195.1
MSLB CASE 4	59.9	1821	365	205.1
MSLB CASE 5	59.9	1811	324	70.1
MSLB CASE 6	57.3	1811	331	195.1
MSLB CASE 7	56.7	1811	354	215.1
MSLB CASE 8	61.6	1801	363	200.4
MSLB CASE 9, $P_0 = 0.0$	63.0	400.7	294	75.1
MSLB CASE 9, $P_0 = 3.0$	67.0	400.7	288	80.0
MSLB CASE 10	58.6	1891	313	260.8
MSLB CASE 11, $P_0 = 0.0$	57.3	1831	342	300.8
MSLB CASE 11, $P_0 = -1.5$	56.0	1832	347	340.8
MSLB CASE 12, $P_0 = 0.0$	63.3	1811	359	180.1
MSLB CASE 12, $P_0 = 3.0$	67.1	1811	347	165.1
MSLB CASE 13	61.0	380.7	273	380.7
MSLB CASE 14	43.1	1801	262	760.8
MSLB CASE 15	33.9	2001	302	290.8
MSLB CASE 16	45.3	1331	324	260.8
<u>RSG Case</u>				
MSLB CASE 1, $P_0 = 3.0$	59.4	1832	351	87.2
MSLB CASE 1, $P_0 = -1.5$	52.6	1828	367	92.2
MSLB CASE 8, $P_0 = 3.0$	62.2	1498	330	192
MSLB CASE 8, $P_0 = -1.5$	57.2	1503	347	157
MSLB CASE 9, $P_0 = 3.0$	64.5	482	347	87.2
MSLB CASE 9, $P_0 = -1.5$	59.2	1828	363	57.2
MSLB CASE 12, $P_0 = 3.0$	65.4	1518	331	162
MSLB CASE 12, $P_0 = -1.5$	60.3	1518	347	132
MSLB CASE 13, $P_0 = 3.0$	66.7	572	342	87.2
MSLB CASE 13, $P_0 = -1.5$	61.3	573	359	57.2

TABLE 6.2-42

DOUBLE-ENDED PUMP SUCTION BREAK - MINIMUM SAFEGUARDS PRINCIPLE PARAMETERS DURING REFLOOD

TIME (seconds)	FLOODING TEMP (degree F)	FLOODING RATE (in/sec)	CARRYOVER FRACTION	CORE HEIGHT (ft)	DOWNCOMER HEIGHT (ft)	FLOW FRACTION	INJECTION TOTAL (Pounds Mass per Second)	INJECTION ACCUMULATOR (Pounds Mass per Second)	INJECTION SPILL (Pounds Mass per Second)	ENTHALPY (Btu/lbm)
21.6	181.8	.000	.000	.00	.00	.333	.0	.0	.0	.00
22.3	179.9	26.292	.000	.67	1.47	.000	7724.4	7724.4	.0	89.50
22.5	179.0	29.720	.000	1.02	1.55	.000	7645.9	7645.9	.0	89.50
23.6	178.4	2.675	.300	1.50	4.42	.404	7191.8	7191.8	.0	89.50
24.6	178.5	2.516	.424	1.64	7.02	.440	6874.3	6874.3	.0	89.50
28.2	178.9	3.819	.628	2.00	15.37	.612	5612.0	5612.0	.0	89.50
29.7	178.9	4.533	.672	2.19	15.62	.673	4898.1	4898.1	.0	89.50
30.7	178.9	4.340	.689	2.30	15.62	.670	4739.6	4739.6	.0	89.50
31.7	179.0	4.378	.702	2.42	15.62	.677	4924.6	4469.4	.0	89.36
32.5	179.1	4.269	.709	2.51	15.62	.675	4823.5	4366.1	.0	89.36
37.9	180.0	3.765	.730	3.00	15.62	.659	4236.1	3767.1	.0	89.33
44.2	181.8	3.391	.737	3.50	15.62	.641	3699.8	3221.1	.0	89.31
51.3	184.3	3.083	.739	4.00	15.62	.622	3212.0	2725.5	.0	89.27
52.7	184.8	2.705	.734	4.09	15.62	.579	2434.2	1937.4	.0	89.19
53.7	185.2	3.222	.742	4.15	15.59	.634	487.1	.0	.0	88.00
54.7	185.6	3.274	.742	4.22	15.47	.637	480.7	.0	.0	88.00
59.7	188.0	3.086	.742	4.57	14.88	.633	484.3	.0	.0	88.00
66.5	191.9	2.851	.740	5.00	14.22	.627	488.5	.0	.0	88.00
75.7	198.1	2.575	.738	5.54	13.55	.618	493.2	.0	.0	88.00
84.2	204.2	2.354	.736	6.00	13.13	.608	496.5	.0	.0	88.00
94.7	212.0	2.128	.733	6.52	12.83	.595	499.4	.0	.0	88.00
105.3	219.9	1.947	.732	7.00	12.72	.582	501.5	.0	.0	88.00
118.7	229.5	1.780	.731	7.56	12.80	.567	503.2	.0	.0	88.00
130.1	236.6	1.683	.731	8.00	13.00	.556	504.1	.0	.0	88.00
144.7	244.5	1.605	.733	8.54	13.37	.547	504.9	.0	.0	88.00
158.0	250.7	1.563	.735	9.00	13.79	.543	505.2	.0	.0	88.00
172.7	256.8	1.538	.738	9.50	14.29	.541	505.4	.0	.0	88.00
180.7	259.9	1.531	.741	9.77	14.57	.541	505.5	.0	.0	88.00
187.7	262.4	1.535	.743	10.00	14.82	.543	505.4	.0	.0	88.00

TABLE 6.2-43 (SHEET 1 of 2)

DOUBLE-ENDED PUMP SUCTION BREAK
MINIMUM SAFEGUARDS
POST-REFLOOD MASS AND ENERGY RELEASES

	Break Path No. 1 Flow		Break Path No. 2 Flow	
Time (seconds)	(lbm/sec)	Thousand (Btu/sec)	(lbm/sec)	Thousand (Btu/sec)
187.8	174.1	216.5	334.1	104.8
192.8	173.9	216.3	334.2	104.6
197.8	172.9	215.1	335.2	104.7
202.8	172.9	215.0	335.2	104.5
207.8	172.1	214.1	336.0	104.5
212.8	171.4	213.2	336.7	104.5
217.8	171.4	213.2	336.7	104.3
222.8	170.7	212.3	337.4	104.3
227.8	169.9	211.4	338.2	104.3
232.8	169.9	211.3	338.2	104.2
237.8	169.2	210.4	339.0	104.2
242.8	168.4	209.4	339.7	104.2
247.8	168.3	209.4	339.8	104.0
252.8	167.6	208.4	340.6	104.0
257.8	166.8	207.4	341.3	104.0
262.8	166.7	207.3	341.4	103.9
267.8	165.9	206.3	342.2	103.9
272.8	165.8	206.2	342.3	103.7
277.8	165.0	205.2	343.1	103.7
282.8	164.1	204.2	344.0	103.8
287.8	164.0	204.0	344.1	103.6
292.8	163.2	202.9	345.0	103.6
297.8	163.0	202.7	345.1	103.5
302.8	162.1	201.7	346.0	103.5
307.8	161.9	201.4	346.2	103.3
312.8	161.1	200.3	347.1	103.4
317.8	160.2	199.2	347.9	103.4
322.8	159.9	198.9	348.2	103.3
327.8	159.0	197.8	349.1	103.3
332.8	158.7	197.4	349.4	103.2
337.8	158.4	197.1	349.7	103.1
342.8	157.5	195.9	350.6	103.1
347.8	157.2	195.5	350.9	103.0
352.8	162.9	202.6	345.2	103.8
357.8	161.8	201.3	346.3	103.9
362.8	161.3	200.6	346.8	103.8
367.8	160.8	200.0	347.3	103.8
372.8	160.2	199.3	347.9	103.7
377.8	159.6	198.5	348.5	103.6
382.8	159.0	197.7	349.1	103.6
387.8	158.3	196.9	349.8	103.6
392.8	157.6	196.1	350.5	103.5
397.8	156.9	195.2	351.2	103.5

TABLE 6.2-43 (SHEET 2 of 2)

	Break Path No. 1 Flow		Break Path No. 2 Flow	
Time (seconds)	(lbm/sec)	Thousand (Btu/sec)	(lbm/sec)	Thousand (Btu/sec)
402.8	156.2	194.3	351.9	103.5
407.8	156.1	194.1	352.0	103.3
412.8	155.4	193.2	352.8	103.3
417.8	154.6	192.3	353.5	103.3
422.8	154.3	191.9	353.8	103.1
427.8	153.4	190.8	354.7	103.1
432.8	153.0	190.3	355.1	103.0
437.8	152.0	189.1	356.1	103.1
442.8	151.5	188.4	356.7	103.0
447.8	150.8	187.6	357.3	103.0
452.8	150.1	186.7	358.0	102.9
457.8	149.7	186.2	358.4	102.8
462.8	148.8	185.1	359.3	102.8
467.8	148.2	184.3	359.9	102.8
472.8	147.5	183.4	360.7	102.7
477.8	147.0	182.8	361.1	102.6
482.8	145.9	181.4	362.2	102.7
487.8	152.0	189.0	356.2	103.3
492.8	151.4	188.3	356.7	103.2
497.8	150.8	187.6	357.3	103.1
502.8	149.9	186.5	358.2	103.1
507.8	149.3	185.7	358.8	103.1
512.8	148.4	184.6	359.7	103.0
517.8	147.6	183.6	360.5	103.0
522.8	72.3	90.0	435.8	123.2
711.2	72.3	90.0	435.8	123.2
711.3	76.6	94.4	431.5	119.3
712.8	76.6	94.4	431.5	119.2
1243.3	76.6	94.4	431.5	119.2
1243.4	67.3	77.4	440.8	45.3
2139.0	59.0	67.9	449.1	46.8
2139.1	59.0	67.9	9.8	8.2
2319.0	58.1	66.9	10.7	8.3
2319.1	58.1	66.9	475.8	95.8
3600.0	51.8	59.7	482.1	96.9

TABLE 6.2-44

**DOUBLE-ENDED PUMP SUCTION BREAK
MASS BALANCE
MINIMUM SAFEGUARDS**

	TIME (SECONDS)	.00	21.60	21.60	187.74	711.31	1243.32	3600.00
					MASS (THOUSAND LBM)			
INITIAL	IN RCS AND ACC	620.40	620.40	620.40	620.40	620.40	620.40	620.40
ADDED MASS	PUMPED INJECTION	.00	.00	.00	77.84	343.84	614.17	1765.59
	TOTAL ADDED	.00	.00	.00	77.84	343.84	614.17	1765.59
*** TOTAL AVAILABLE***		620.40	620.40	620.40	698.24	964.24	1234.57	2385.99
DISTRIBUTION	REACTOR CO0LANT	417.47	48.24	68.95	130.25	130.25	130.25	130.25
	ACCUMULATOR	202.93	155.44	134.73	.00	.00	.00	.00
	TOTAL CONTENTS	620.40	203.68	203.68	130.25	130.25	130.25	130.25
EFFLUENT	BREAK FLOW	.00	416.71	416.71	567.97	833.98	1104.30	2255.73
	ECCS SPILL	.00	.00	.00	.00	.00	.00	.00
	TOTAL EFFLUENT	.00	416.71	416.71	567.97	833.98	1104.30	2255.73
*** TOTAL ACCOUNTABLE ***		620.40	620.39	620.39	698.22	964.23	1234.55	2385.97

TABLE 6.2-45

DOUBLE-ENDED PUMP SUCTION BREAK ENERGY BALANCE
MINIMUM SAFEGUARDS

	Time (Seconds)	.00	21.60	21.60	187.74	711.31	1243.32	3600.00
					Energy (Million Btu)			
Initial Energy	In RCS, ACC, S. Gen	675.98	675.98	675.98	675.98	675.98	675.98	675.98
Added Energy	Pumped Injection	.00	.00	.00	6.85	30.26	54.05	214.96
	Decay Heat	.00	5.39	5.39	21.62	59.57	91.60	203.05
	Heat from Secondary	.00	-5.74	-5.74	-5.74	-3.77	-2.20	-2.20
	Total Added	.00	-.35	-.35	22.74	86.06	143.45	415.81
TOTAL AVAILABLE		675.98	675.63	675.63	698.72	762.04	819.43	1091.80
Distribution	Reactor Coolant	244.80	10.56	12.41	33.81	33.81	33.81	33.81
	Accummulator	18.16	13.91	12.06	.00	.00	.00	.00
	Core Stored	18.93	9.53	9.53	4.05	3.90	3.68	2.71
	Primary Metal	120.89	114.27	114.27	91.33	64.17	53.37	39.48
	Secondary Metal	76.01	75.71	75.71	68.26	51.06	39.88	29.57
	Steam Generator	197.20	196.23	196.23	173.56	127.03	99.38	73.89
	Total Contents	675.98	420.21	420.21	371.01	279.98	230.12	179.45
Effluent	Break Flow	.00	254.95	254.95	321.10	475.47	575.84	900.23
	ECCS Spill	.00	.00	.00	.00	.00	.00	.00
	Total Effluent	.00	254.95	254.95	321.10	475.47	575.84	900.23
TOTAL ACCOUNTABLE		675.98	675.16	675.16	692.11	755.44	805.95	1079.69

TABLE 6.2-46 (SHEET 1 OF 5)

DOUBLE-ENDED PUMP SUCTION BREAK - MAXIMUM SAFEGUARDS REFLOOD MASS AND ENERGY RELEASES

	Break Path No. 1 Flow		Break Path No. 2 Flow	
Time (seconds)	(lbm/sec)	Thousand (Btu/sec)	(lbm/sec)	Thousand (Btu/sec)
21.6	.0	.0	.0	.0
22.1	.0	.0	.0	.0
22.3	.0	.0	.0	.0
22.4	.0	.0	.0	.0
22.5	.0	.0	.0	.0
22.5	.0	.0	.0	.0
22.6	114.5	135.1	.0	.0
22.7	47.0	55.5	.0	.0
22.8	41.3	48.8	.0	.0
22.9	47.3	55.8	.0	.0
23.0	53.8	63.5	.0	.0
23.1	60.0	70.8	.0	.0
23.2	65.7	77.6	.0	.0
23.3	71.2	84.0	.0	.0
23.4	76.3	90.1	.0	.0
23.5	81.2	95.8	.0	.0
23.5	82.4	97.2	.0	.0
23.6	85.9	101.4	.0	.0
23.7	90.3	106.6	.0	.0
23.8	94.7	111.7	.0	.0
23.9	98.8	116.6	.0	.0
24.0	102.8	121.4	.0	.0
24.1	106.7	126.0	.0	.0
24.2	110.5	130.4	.0	.0
24.3	114.1	134.7	.0	.0
24.4	117.7	138.9	.0	.0
24.5	121.1	143.0	.0	.0
24.6	124.5	147.0	.0	.0
25.6	154.3	182.2	.0	.0
26.6	394.1	466.9	3497.1	447.9
27.4	478.5	567.7	4297.3	574.8
27.7	477.9	567.0	4288.2	576.4
28.7	466.9	553.9	4191.1	567.4
29.7	454.6	539.3	4080.7	556.0

TABLE 6.2-46 (SHEET 2 OF 5)

	Break Path No. 1 Flow		Break Path No. 2 Flow	
Time (seconds)	(lbm/sec)	Thousand (Btu/sec)	(lbm/sec)	Thousand (Btu/sec)
30.7	442.5	524.8	3969.8	544.3
31.4	491.3	583.0	4465.5	589.1
31.7	486.5	577.4	4413.9	586.9
32.7	476.0	564.8	4321.5	576.4
33.7	465.8	552.6	4229.1	566.3
34.7	456.0	540.9	4139.8	556.5
35.7	446.6	529.7	4053.7	547.0
36.5	439.4	521.1	3987.0	539.6
36.7	437.7	519.0	3970.6	537.8
37.7	429.1	508.7	3890.6	529.0
38.7	420.9	499.0	3813.4	520.5
39.7	413.0	489.6	3738.8	512.2
40.7	405.5	480.6	3666.7	504.3
41.7	398.3	471.9	3597.1	496.5
42.5	392.7	465.3	3542.9	490.5
42.7	391.3	463.6	3529.6	489.0
43.7	384.6	455.6	3464.2	481.8
44.7	378.1	447.9	3400.8	474.7
45.7	371.9	440.5	3339.2	467.9
46.7	365.8	433.3	3279.4	461.2
47.7	360.0	426.4	3221.3	454.7
48.7	354.4	419.6	3164.7	448.4
49.1	352.2	417.0	3142.5	445.9
49.7	348.9	413.1	3109.6	442.2
50.7	343.6	406.8	3055.8	436.2
51.7	338.4	400.7	3003.4	430.3
52.7	333.4	394.7	2952.3	424.6
53.7	328.6	388.9	2902.4	418.9
54.7	323.8	383.3	2853.5	413.4
55.7	183.9	217.2	647.8	157.9
56.7	183.4	216.7	648.5	157.7
57.7	183.0	216.2	649.4	157.6
58.7	182.5	215.7	650.3	157.4
59.7	182.1	215.2	651.2	157.2
60.7	181.7	214.6	652.1	157.1
61.7	181.3	214.1	653.1	156.9

TABLE 6.2-46 (SHEET 3 OF 5)

	Break Path No. 1 Flow		Break Path No. 2 Flow	
Time (seconds)	(lbm/sec)	Thousand (Btu/sec)	(lbm/sec)	Thousand (Btu/sec)
62.7	180.8	213.6	654.0	156.7
63.7	180.4	213.1	654.9	156.6
64.7	180.0	212.6	655.8	156.4
65.7	179.5	212.1	656.7	156.3
66.6	179.2	211.7	657.6	156.1
66.7	179.1	211.6	657.7	156.1
67.7	178.7	211.1	658.6	155.9
68.7	178.3	210.6	659.5	155.8
69.7	177.9	210.1	660.4	155.6
70.7	177.4	209.6	661.3	155.5
71.7	177.0	209.1	662.2	155.3
72.7	176.6	208.6	663.1	155.2
73.7	176.2	208.1	664.0	155.0
74.7	175.7	207.6	665.0	154.8
75.7	175.3	207.1	665.9	154.7
76.7	174.9	206.6	666.8	154.5
77.7	174.5	206.1	667.7	154.4
78.7	174.1	205.6	668.6	154.2
79.7	173.6	205.1	669.5	154.1
80.7	173.2	204.6	670.5	153.9
81.7	172.8	204.1	671.4	153.8
82.7	172.4	203.6	672.3	153.6
84.7	171.5	202.6	674.2	153.3
86.7	170.7	201.6	676.0	153.0
87.5	170.3	201.2	676.8	152.9
88.7	169.8	200.6	677.9	152.7
90.7	169.0	199.6	679.8	152.4
92.7	168.1	198.6	681.6	152.1
94.7	167.3	197.6	683.5	151.8
96.7	166.4	196.6	685.4	151.5
98.7	165.5	195.5	687.3	151.2
100.7	164.7	194.5	689.2	150.9
102.7	163.8	193.5	691.1	150.6
104.7	162.9	192.4	693.0	150.3
106.7	162.1	191.4	694.9	150.0
108.7	161.2	190.4	696.8	149.7

TABLE 6.2-46 (SHEET 4 OF 5)

	Break Path No. 1 Flow		Break Path No. 2 Flow	
Time (seconds)	(lbm/sec)	Thousand (Btu/sec)	(lbm/sec)	Thousand (Btu/sec)
110.1	160.6	189.6	698.1	149.5
110.7	160.3	189.3	698.7	149.4
112.7	159.4	188.3	700.6	149.1
114.7	158.5	187.2	702.5	148.8
116.7	157.6	186.1	704.3	148.5
118.7	156.7	185.1	706.2	148.2
120.7	155.8	184.0	708.1	147.9
122.7	154.9	183.0	709.9	147.6
124.7	154.0	181.9	711.8	147.3
126.7	153.1	180.8	713.7	147.0
128.7	152.2	179.7	715.5	146.7
130.7	151.3	178.7	717.4	146.4
132.7	150.4	177.6	719.2	146.1
134.7	149.4	176.5	721.1	145.8
136.7	148.5	175.4	722.9	145.4
138.7	147.6	174.3	724.7	145.1
140.7	146.7	173.2	726.6	144.8
142.7	145.8	172.1	728.4	144.5
144.7	144.8	171.0	730.2	144.2
146.7	143.9	169.9	732.0	143.9
148.7	143.0	168.8	733.9	143.6
150.7	142.0	167.7	735.7	143.3
152.7	141.1	166.6	737.5	143.0
154.7	140.2	165.5	739.3	142.7
156.7	139.2	164.4	741.2	142.4
158.7	138.3	163.3	743.0	142.0
160.7	137.4	162.3	744.9	141.9
161.9	137.1	161.9	745.6	141.9
162.7	136.9	161.6	746.0	141.9
164.7	136.3	160.9	747.2	141.8
166.7	135.7	160.2	748.3	141.7
168.7	135.1	159.5	749.4	141.7
170.7	134.5	158.9	750.5	141.6
172.7	134.0	158.2	751.6	141.5
174.7	133.4	157.5	752.7	141.4
176.7	132.8	156.8	753.7	141.3
178.7	132.3	156.2	754.8	141.3

TABLE 6.2-46 (SHEET 5 OF 5)

	Break Path No. 1 Flow		Break Path No. 2 Flow	
Time (seconds)	(lbm/sec)	Thousand (Btu/sec)	(lbm/sec)	Thousand (Btu/sec)
180.7	131.7	155.5	755.9	141.2
182.7	131.1	154.8	756.9	141.1
184.7	130.6	154.2	758.0	141.0
186.7	130.0	153.5	759.1	140.9
188.7	129.5	152.9	760.1	140.8
190.7	128.9	152.2	761.1	140.7
192.1	128.6	151.8	761.9	140.6

TABLE 6.2-47

DOUBLE-ENDED PUMP SUCTION BREAK - MAXIMUM SAFEGUARDS PRINCIPLE PARAMETERS DURING REFLOOD

Time Seconds	Flooding Temp °F	Flooding Rate in/sec	Carryover Fraction	Core Height (ft)	Downcomer Height (ft)	Flow Frac	Injection Total (Pounds Mass per Second)	Injection Accum (Pounds Mass per Second)	Injection Spill (Pounds Mass per Second)	Enthalpy Btu/lbm
21.6	183.1	.000	.000	.00	.00	.333	.0	.0	.0	.00
22.3	181.1	22.834	.000	.53	1.85	.000	7698.9	7698.9	.0	89.50
22.5	179.3	28.240	.000	1.06	1.85	.000	7582.0	7582.0	.0	89.50
23.5	178.4	2.939	.303	1.50	5.23	.418	7175.6	7175.6	.0	89.50
24.5	178.4	2.797	.436	1.65	8.77	.451	6865.9	6865.9	.0	89.50
27.4	178.3	5.018	.634	2.01	15.62	.679	5382.1	5382.1	.0	89.50
28.7	178.3	4.633	.673	2.19	15.62	.676	5120.7	5120.7	.0	89.50
30.7	178.4	4.259	.702	2.42	15.62	.670	4800.0	4800.0	.0	89.50
31.4	178.5	4.556	.710	2.50	15.62	.691	5360.3	4479.6	.0	89.25
36.5	179.7	4.036	.732	3.00	15.62	.674	4756.1	3854.9	.0	89.22
42.5	181.9	3.653	.739	3.51	15.62	.659	4224.7	3304.3	.0	89.17
49.1	184.9	3.345	.742	4.01	15.62	.644	3755.9	2820.2	.0	89.13
55.7	188.1	2.267	.729	4.45	15.62	.506	986.7	.0	.0	88.00
56.7	188.6	2.261	.729	4.50	15.62	.506	986.7	.0	.0	88.00
66.6	194.5	2.202	.730	5.00	15.62	.506	986.7	.0	.0	88.00
77.7	202.9	2.137	.732	5.54	15.62	.507	986.7	.0	.0	88.00
87.5	211.2	2.080	.733	6.00	15.62	.507	986.7	.0	.0	88.00
98.7	221.1	2.013	.736	6.51	15.62	.508	986.8	.0	.0	88.00
110.1	230.4	1.945	.738	7.00	15.62	.508	986.8	.0	0	88.00
122.7	239.2	1.870	.740	7.53	15.62	.508	986.9	.0	.0	88.00
134.7	246.4	1.799	.742	8.00	15.62	.509	987.0	.0	.0	88.00
148.7	253.6	1.717	.744	8.53	15.62	.508	987.1	.0	.0	88.00
161.9	259.5	1.642	.745	9.00	15.62	.509	987.2	.0	.0	88.00
176.7	265.1	1.577	.748	9.50	15.62	.511	987.1	.0	.0	88.00
192.1	270.2	1.512	.750	10.00	15.62	.515	987.0	.0	.0	88.00

TABLE 6.2-48 (SHEET 1 OF 3)

DOUBLE-ENDED PUMP SUCTION BREAK - MAXIMUM SAFEGUARDS POST-REFLOOD MASS AND ENERGY RELEASES

	Break Path No. 1 Flow		Break Path No. 2 Flow	
Time (seconds)	(lbm/sec)	Thousand (Btu/sec)	(lbm/sec)	Thousand (Btu/sec)
192.2	148.6	185.4	841.4	154.6
197.2	148.7	185.5	841.3	154.3
202.2	148.9	185.8	841.1	154.1
207.2	148.4	185.2	841.6	154.0
212.2	148.7	185.6	841.3	153.7
217.2	148.2	185.0	841.8	153.7
222.2	148.5	185.3	841.5	153.4
227.2	148.0	184.7	842.0	153.3
232.2	148.3	185.1	841.7	153.0
237.2	147.8	184.4	842.2	153.0
242.2	148.1	184.8	841.9	152.7
247.2	147.6	184.1	842.4	152.6
252.2	147.8	184.4	842.2	152.4
257.2	148.1	184.7	841.9	152.1
262.2	147.5	184.1	842.5	152.0
267.2	147.8	184.4	842.2	151.8
272.2	147.2	183.7	842.8	151.7
277.2	147.4	184.0	842.6	151.5
282.2	147.6	184.2	842.4	151.2
287.2	147.1	183.5	842.9	151.1
292.2	147.3	183.7	842.7	150.9
297.2	146.7	183.0	843.3	150.8
302.2	146.9	183.2	843.1	150.6
307.2	147.0	183.4	843.0	152.9
312.2	146.4	182.7	843.6	152.9
317.2	146.6	182.9	843.4	152.6
322.2	146.7	183.0	843.3	152.4
327.2	146.8	183.2	843.2	152.1
332.2	146.2	182.4	843.8	152.1
337.2	146.3	182.5	843.7	151.8
342.2	146.4	182.6	843.6	151.6
347.2	146.4	182.7	843.6	151.3
352.2	145.8	181.9	844.2	151.3
357.2	145.8	181.9	844.2	151.1
362.2	145.8	182.0	844.2	150.8
367.2	145.9	182.0	844.1	150.6

TABLE 6.2-48 (SHEET 2 OF 3)

Time (seconds)	Break Path No. 1 Flow (lbm/sec)	Break Path No. 1 Flow Thousand (Btu/sec)	Break Path No. 2 Flow (lbm/sec)	Break Path No. 2 Flow Thousand (Btu/sec)
372.2	145.9	182.0	844.1	150.4
377.2	145.8	182.0	844.2	150.2
382.2	145.1	181.1	844.9	150.1
387.2	145.1	181.0	844.9	149.9
392.2	145.0	180.9	845.0	149.7
397.2	144.9	180.8	845.1	149.5
402.2	144.9	180.8	845.1	149.3
407.2	144.9	180.8	845.1	149.1
412.2	144.9	180.8	845.1	148.8
417.2	144.9	180.7	845.1	148.6
422.2	144.8	180.7	845.2	148.4
427.2	144.7	180.6	845.3	148.2
432.2	144.6	180.5	845.4	148.0
437.2	144.5	180.3	845.5	147.8
442.2	144.4	180.1	845.6	147.6
447.2	144.8	180.7	845.2	147.3
452.2	144.6	180.5	845.4	147.1
457.2	144.4	180.2	845.6	146.9
462.2	144.2	179.9	845.8	149.2
467.2	144.5	180.3	845.5	148.9
472.2	144.2	179.9	845.8	148.7
477.2	144.4	180.2	845.6	148.4
482.2	144.0	179.7	846.0	148.2
487.2	144.2	179.9	845.8	148.0
492.2	144.3	180.0	845.7	147.7
497.2	143.8	179.4	846.2	147.6
502.2	143.8	179.4	846.2	147.3
507.2	143.7	179.3	846.3	147.1
512.2	143.6	179.2	846.4	146.9
517.2	143.9	179.6	846.1	146.5
522.2	143.7	179.2	846.3	146.4
527.2	143.8	179.4	846.2	146.1
532.2	143.4	178.9	846.6	145.9
537.2	143.3	178.9	846.7	145.7
542.2	143.7	179.3	846.3	145.3
547.2	143.4	178.9	846.6	147.5
552.2	143.4	178.9	846.6	147.2
557.2	143.3	178.7	846.7	147.0
562.2	143.4	178.9	846.6	146.7
567.2	143.2	178.7	846.8	146.5

TABLE 6.2-48 (SHEET 3 OF 3)

	Break Path No. 1 Flow		Break Path No. 2 Flow	
Time (seconds)	(lbm/sec)	Thousand (Btu/sec)	(lbm/sec)	Thousand (Btu/sec)
572.2	143.2	178.7	846.8	146.2
577.2	142.8	178.2	847.2	146.0
582.2	142.8	178.2	847.2	145.7
587.2	143.0	178.4	847.0	145.4
592.2	142.8	178.2	847.2	145.2
597.2	142.6	177.9	847.4	145.0
602.2	142.5	177.8	847.5	144.7
607.2	69.9	87.2	920.2	164.2
792.8	69.9	87.2	920.2	164.2
792.9	75.0	92.6	915.0	161.7
797.2	74.9	92.5	915.1	161.4
1221.1	74.9	92.5	915.1	161.4
1221.2	67.7	77.9	922.3	87.5
1311.6	66.5	76.6	923.5	87.7
1311.7	66.5	76.6	68.4	12.5
1491.6	64.3	74.0	70.6	12.9
1491.7	64.3	74.0	1095.0	166.0
3600.0	52.0	59.8	1107.4	168.2

TABLE 6.2-49

DOUBLE-ENDED PUMP SUCTION BREAK MASS BALANCE
MAXIMUM SAFEGUARDS

	Time (Seconds)	.00	21.60	21.60	192.12	792.85	1221.12	3600.00
					Mass (Thousand lbm)			
Initial	In RCS and ACC	620.08	620.08	620.08	620.08	620.08	620.08	620.08
Added Mass	Pumped Injection	.00	.00	.00	157.40	752.04	1176.03	3734.26
	Total Added	.00	.00	.00	157.40	752.04	1176.03	3734.26
TOTAL AVAILABLE		620.08	620.08	620.08	777.48	1372.13	1796.12	4354.35
Distribution	Reactor Coolant	416.79	47.97	67.66	118.91	118.91	118.91	118.91
	Accummulator	203.30	156.65	136.96	.00	.00	.00	.00
	Total Contents	620.08	204.61	204.61	118.91	118.91	118.91	118.91
Effluent	Break Flow	.00	415.46	415.46	649.72	1244.37	1668.36	4226.58
	ECCS Spill	.00	.00	.00	.00	.00	.00	.00
	Total Effluent	.00	415.46	415.46	649.72	1244.37	1668.36	4226.58
TOTAL ACCOUNTABLE		620.08	620.07	620.07	768.63	1363.28	1787.27	4345.49

TABLE 6.2-50

DOUBLE-ENDED PUMP SUCTION BREAK ENERGY BALANCE
MAXIMUM SAFEGUARDS

	Time (Seconds)	.00	21.60	21.60	192.12	792.85	1221.12	3600.00
					Energy (Million Btu)			
Initial Energy	In RCS, ACC, S Gen	673.30	673.30	673.30	673.30	673.30	673.30	673.30
Added Energy	Pumped Injection	.00	.00	.00	13.85	66.18	103.49	461.34
	Decay Heat	.00	5.73	5.73	22.33	65.12	90.67	203.33
	Heat From Secondary	.00	-5.70	-5.70	-5.70	-3.44	-2.25	-2.25
	Total Added	.00	.03	.03	30.48	127.86	191.91	662.43
TOTAL AVAILABLE		673.30	673.34	673.34	703.79	801.17	865.22	1335.73
Distribution	Reactor Coolant	244.82	10.46	12.22	31.25	31.25	31.25	31.25
	Accummulator	18.20	14.02	12.26	.00	.00	.00	.00
	Core Stored	18.93	9.68	9.68	4.05	3.90	3.72	2.71
	Primary Metal	118.16	111.56	111.56	89.39	61.11	52.35	38.51
	Secondary Metal	76.01	75.75	75.75	68.35	49.12	40.00	29.56
	Steam Generator	197.20	196.34	196.34	173.69	121.93	99.57	73.76
	Total Contents	673.30	417.81	417.81	366.73	267.30	226.88	175.78
Effluent	Break Flow	.00	255.05	255.05	328.89	525.69	621.47	1145.63
	ECCS Spill	.00	.00	.00	.00	.00	.00	.00
	Total Effluent	.00	255.05	255.05	328.89	525.69	621.47	1145.63
TOTAL ACCOUNTABLE		673.30	672.86	672.86	695.62	792.99	848.35	1321.40

TABLE 6.2-51

DOUBLE-ENDED HOT LEG BREAK SEQUENCE OF EVENTS

Time (sec)	Event Description
0.0	Break Occurs, Reactor Trip and Loss of Offsite Power are assumed
3.0	Low Pressurizer Pressure SI Setpoint - 1715 psia reached by SATAN
11.4	Broken Loop Accumulator Begins Injecting Water
11.6	Intact Loop Accumulator Begins Injecting Water
20.0	End of Blowdown Phase

TABLE 6.2-52

DOUBLE-ENDED PUMP SUCTION BREAK
MINIMUM SAFEGUARDS
SEQUENCE OF EVENTS

Time (sec)	Event Description
0.0	Break Occurs, Reactor Trip and Loss of Offsite Power are assumed
3.9	Low Pressurizer Pressure SI Setpoint - 1715 psia reached by SATAN
13.4	Broken Loop Accumulator Begins Injecting Water
13.6	Intact Loop Accumulator Begins Injecting Water
21.6	End of Blowdown Phase
30.9	Safety Injection Begins
52.1	Broken Loop Accumulator Water Injection Ends
53.3	Intact Loop Accumulator Water Injection Ends
187.7	End of Reflood Phase
2139.0	Cold Leg Recirculation Begins
1.0E+06	Transient Modeling Terminated

TABLE 6.2-53

DOUBLE-ENDED PUMP SUCTION BREAK
MAXIMUM SAFEGUARDS
SEQUENCE OF EVENTS

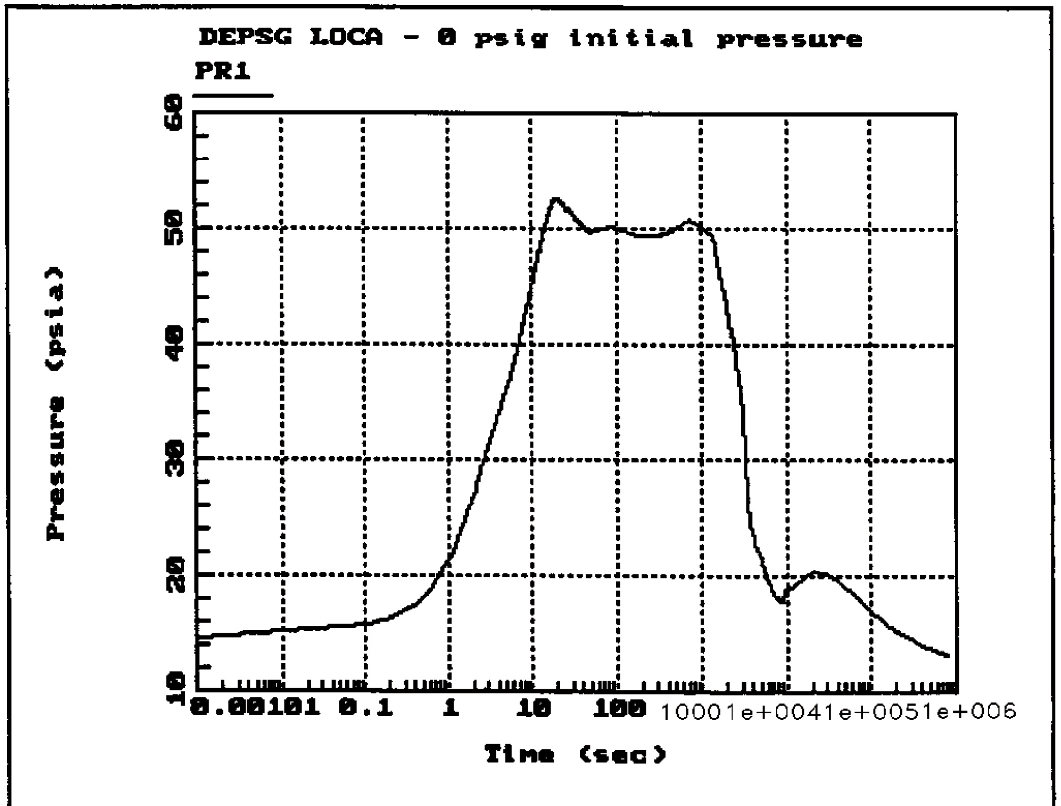
Time (sec)	Event Description
0.0	Break Occurs, Reactor Trip and Loss of Offsite Power are assumed
3.9	Low Pressurizer Pressure SI Setpoint - 1715 psia reached by SATAN
13.3	Broken Loop Accumulator Begins Injecting Water
13.5	Intact Loop Accumulator Begins Injecting Water
21.6	End of Blowdown Phase
30.9	Safety Injection Begins
54.7	Broken Loop Accumulator Water Injection Ends
54.9	Intact Loop Accumulator Water Injection Ends
192.1	End of Reflood Phase
1311.6	Cold Leg Recirculation Begins
1.0E+06	Transient Modeling Terminated

TABLE 6.2-54

LOCA MASS AND ENERGY RELEASE ANALYSIS
CORE DECAY HEAT FRACTION

Time (sec)	Decay Heat Generation Rate (Btu/Btu)
10	0.053876
15	0.050401
20	0.048018
40	0.042401
60	0.039244
80	0.037065
100	0.035466
150	0.032724
200	0.030936
400	0.027078
600	0.024931
800	0.023389
1000	0.022156
1500	0.019921
2000	0.018315
4000	0.014781
6000	0.013040
8000	0.012000
10000	0.011262
15000	0.010097
20000	0.009350
40000	0.007778
60000	0.006958
80000	0.006424
100000	0.006021
150000	0.005323
200000	0.004847
400000	0.003770
600000	0.003201
800000	0.002834
1000000	0.002580

LOCA Pressure Response
(Double-Ended Pump Suction Break)



An evaluation for changes to the long-term response has been performed as described in section 6.2.1.3.13.

16

REV 21 5/08

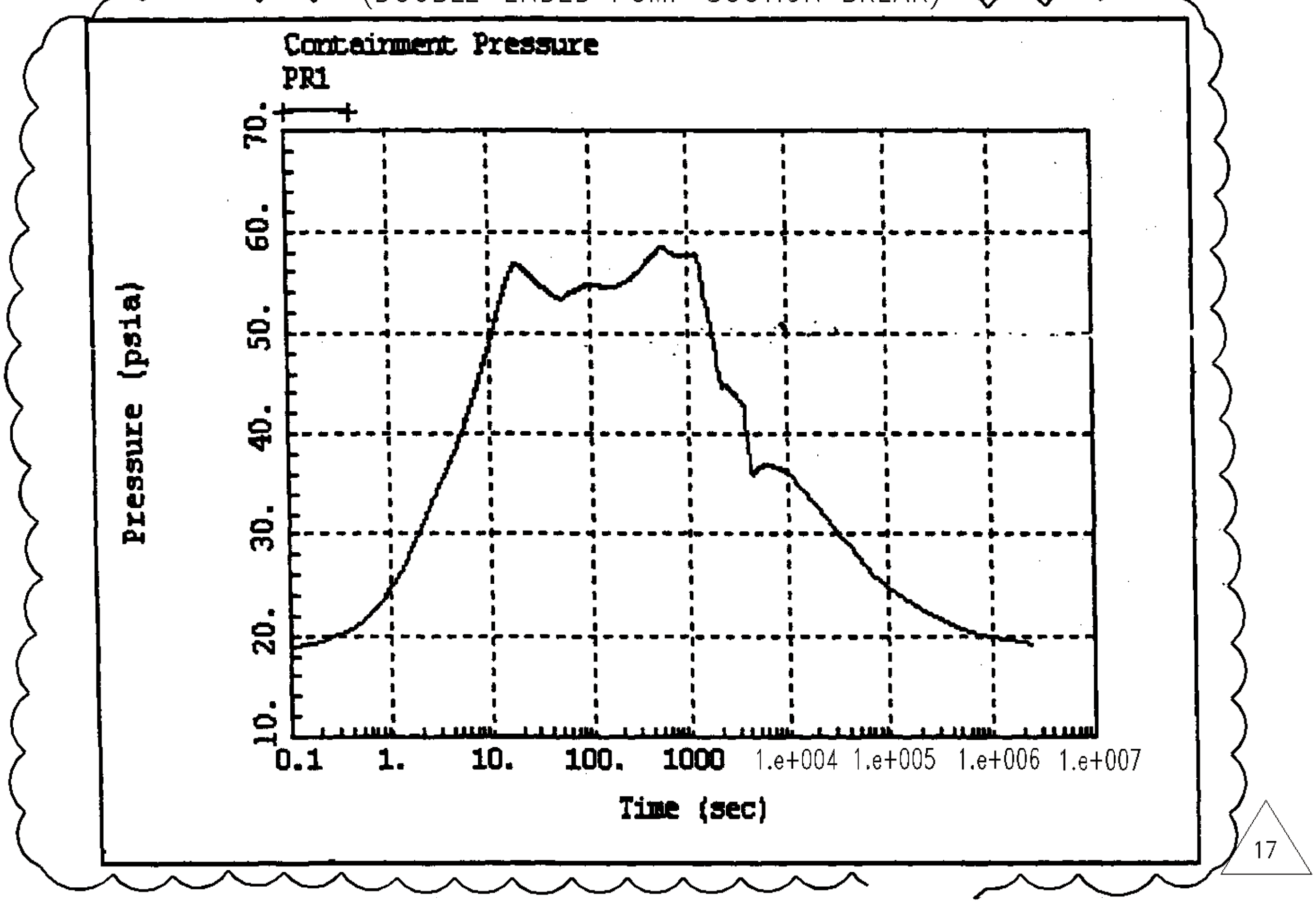
SOUTHERN COMPANY
Energy to Serve Your World®

JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

DEPSGB MINIMUM ESF 1 AC PRESSURE VS. TIME
P_O = 0 PSIG

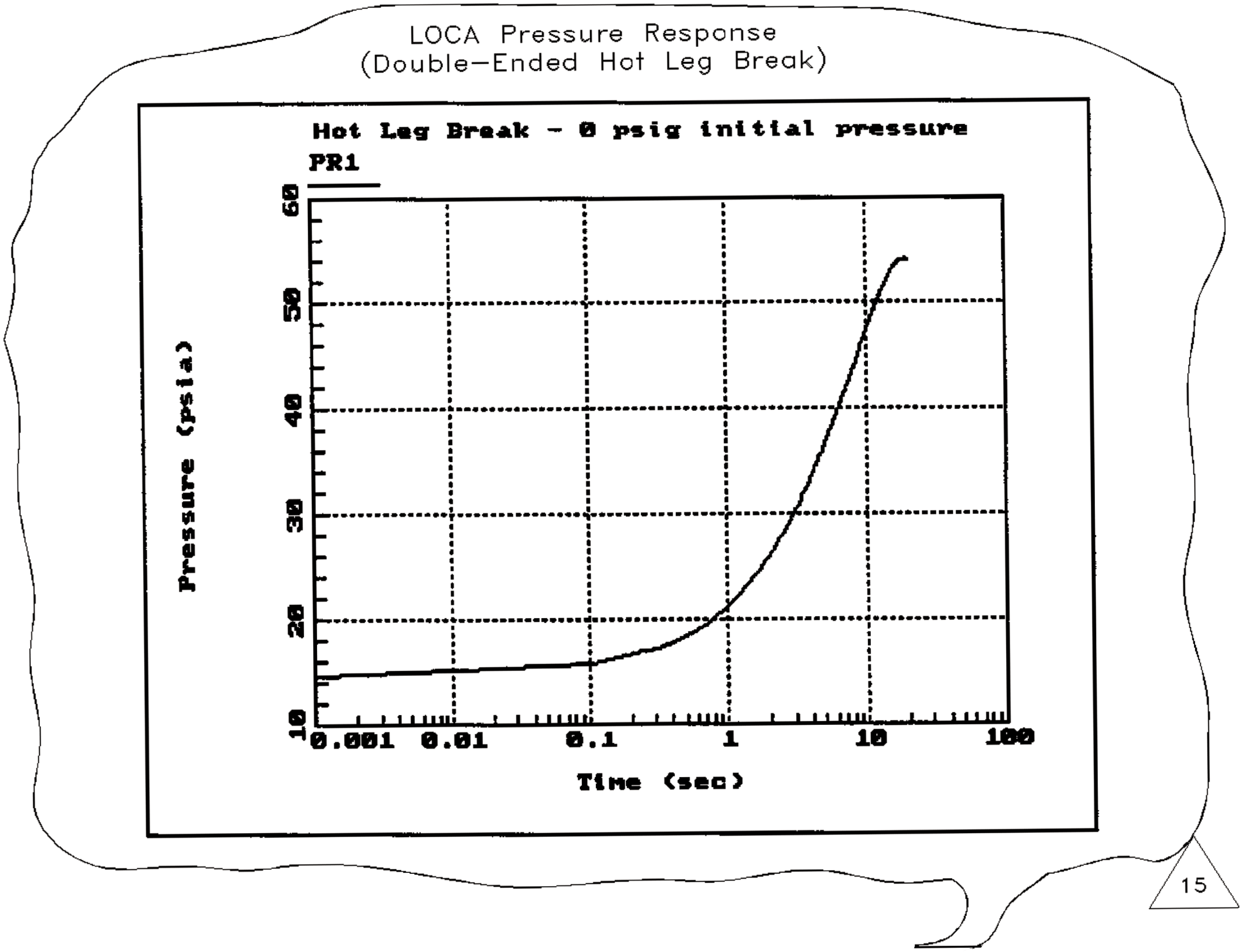
FIGURE 6.2-1

LOCA PRESSURE RESPONSE
(DOUBLE-ENDED PUMP SUCTION BREAK)



REV 21 5/08

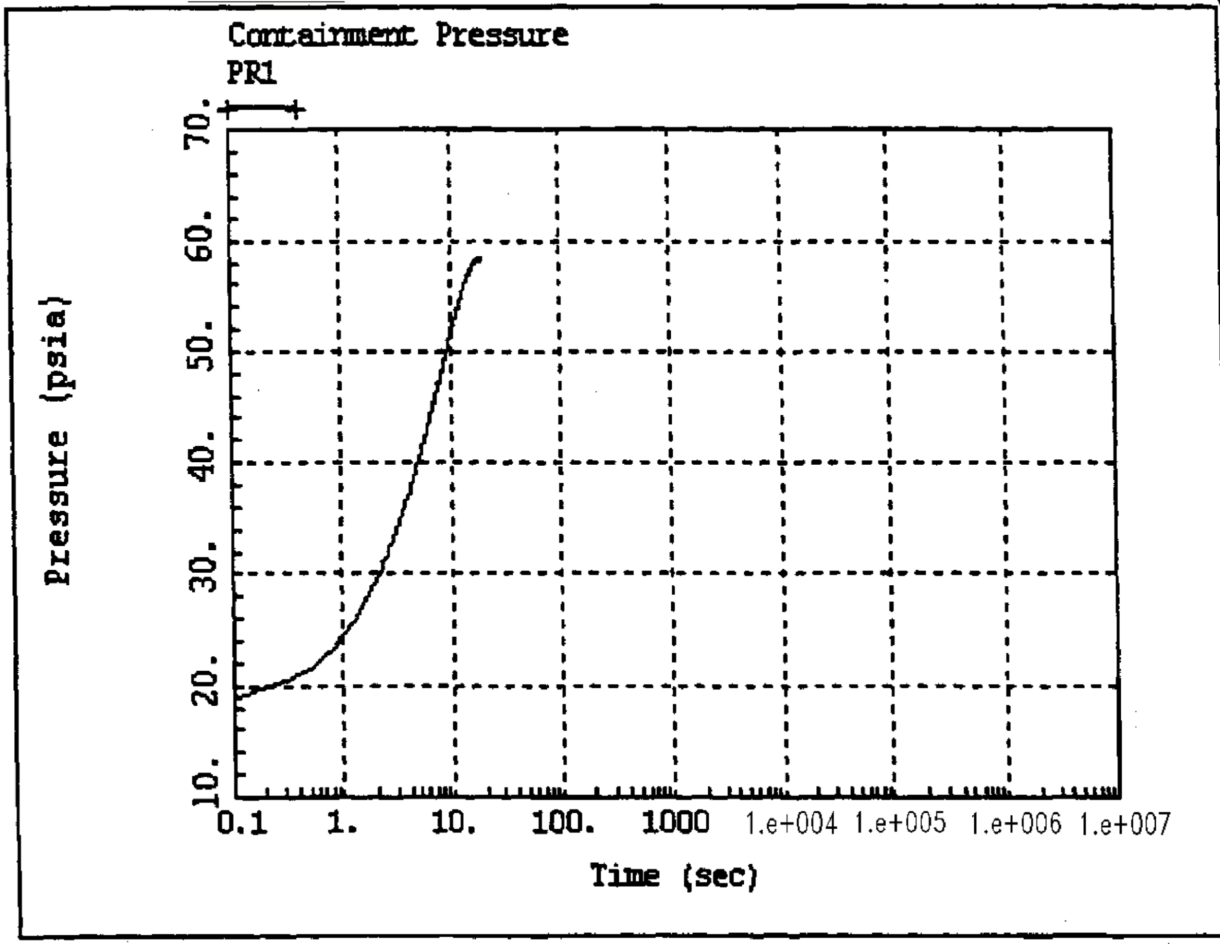
SOUTHERN COMPANY *Energy to Serve Your World®*	JOSEPH M. FARLEY NUCLEAR PLANT UNIT 1 AND UNIT 2	RSG DEPSG MINIMUM ESF 1 AC PRESSURE VS. TIME, P_O = 3 PSIG FIGURE 6.2-2



REV 21 5/08

SOUTHERN COMPANY *Energy to Serve Your World®*	**JOSEPH M. FARLEY NUCLEAR PLANT UNIT 1 AND UNIT 2**	DEHL MINIMUM ESF, DBA SHORT TERM PRESSURE VS. TIME, P_O = 0 PSIG
		FIGURE 6.2-3

LOCA PRESSURE RESPONSE
(DOUBLE-ENDED HOT LEG BREAK)



17

REV 21 5/08

SOUTHERN COMPANY
Energy to Serve Your World®

JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

RSG DEHLG MINIMUM ESF, DBA SHORT PRESSURE VS. TIME, P_O = +3 PSIG

FIGURE 6.2-4

THIS FIGURE HAS BEEN DELETED.

15

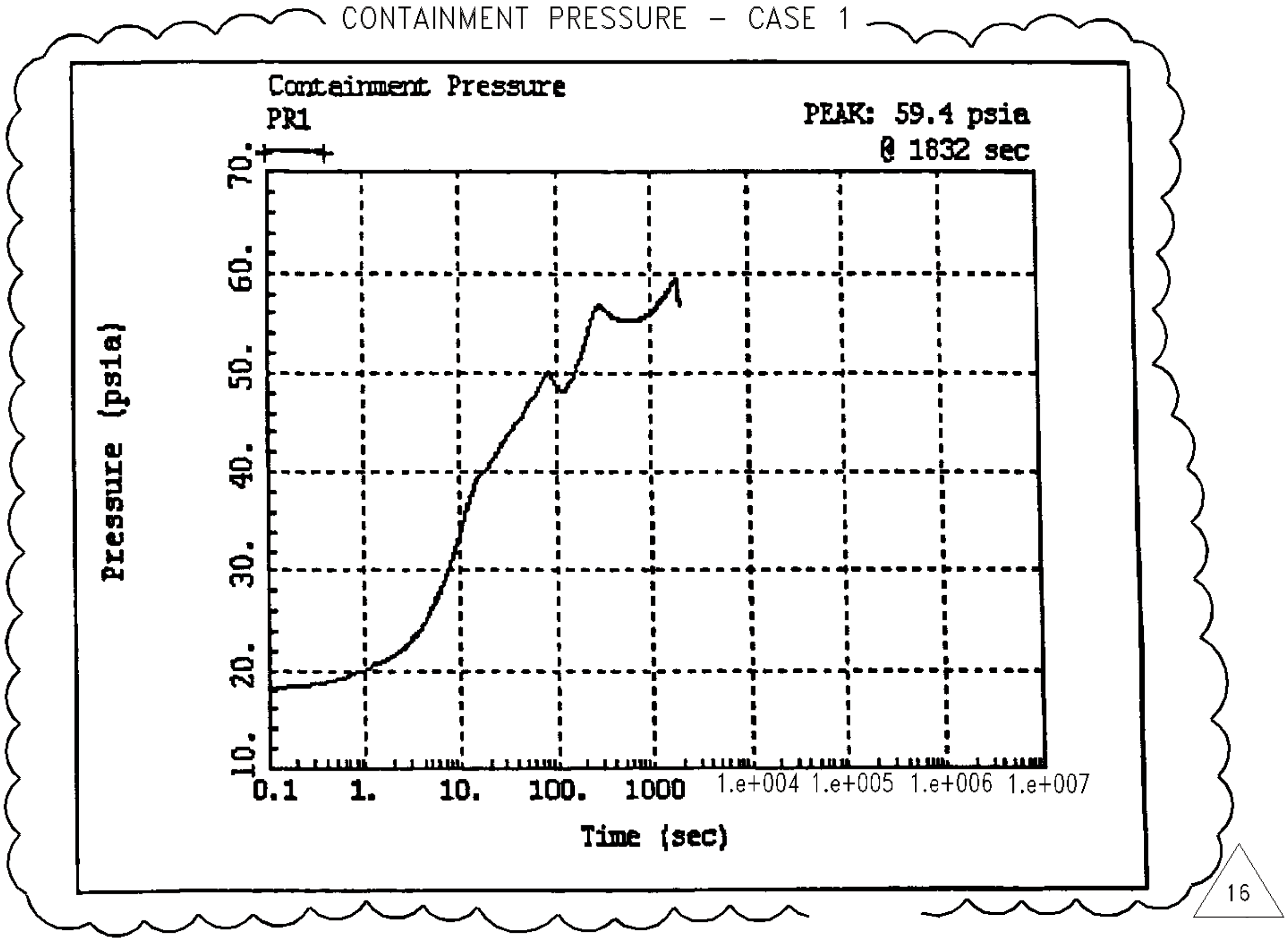
REV 21 5/08

SOUTHERN COMPANY
Energy to Serve Your World®

JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

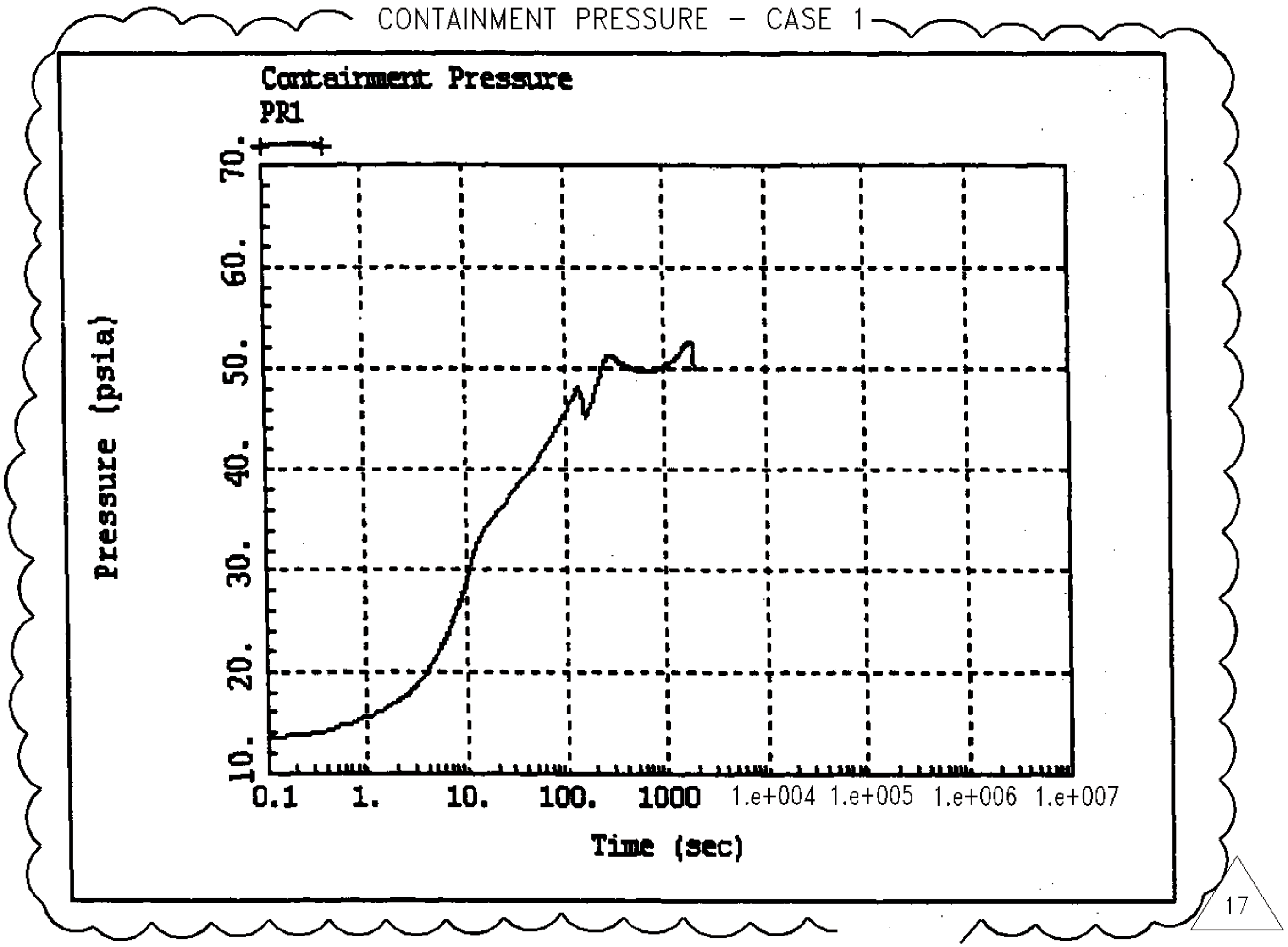
DECLG MAXIMUM ESF PRESSURE VS. TIME

FIGURE 6.2-5



REV 21 5/08

SOUTHERN COMPANY *Energy to Serve Your World®*	**JOSEPH M. FARLEY NUCLEAR PLANT UNIT 1 AND UNIT 2**	RSG PRESSURE VERSUS TIME STEAM LINE FULL D.E. BREAK 102% POWER, P_O = +3 PSIG
		FIGURE 6.2-6



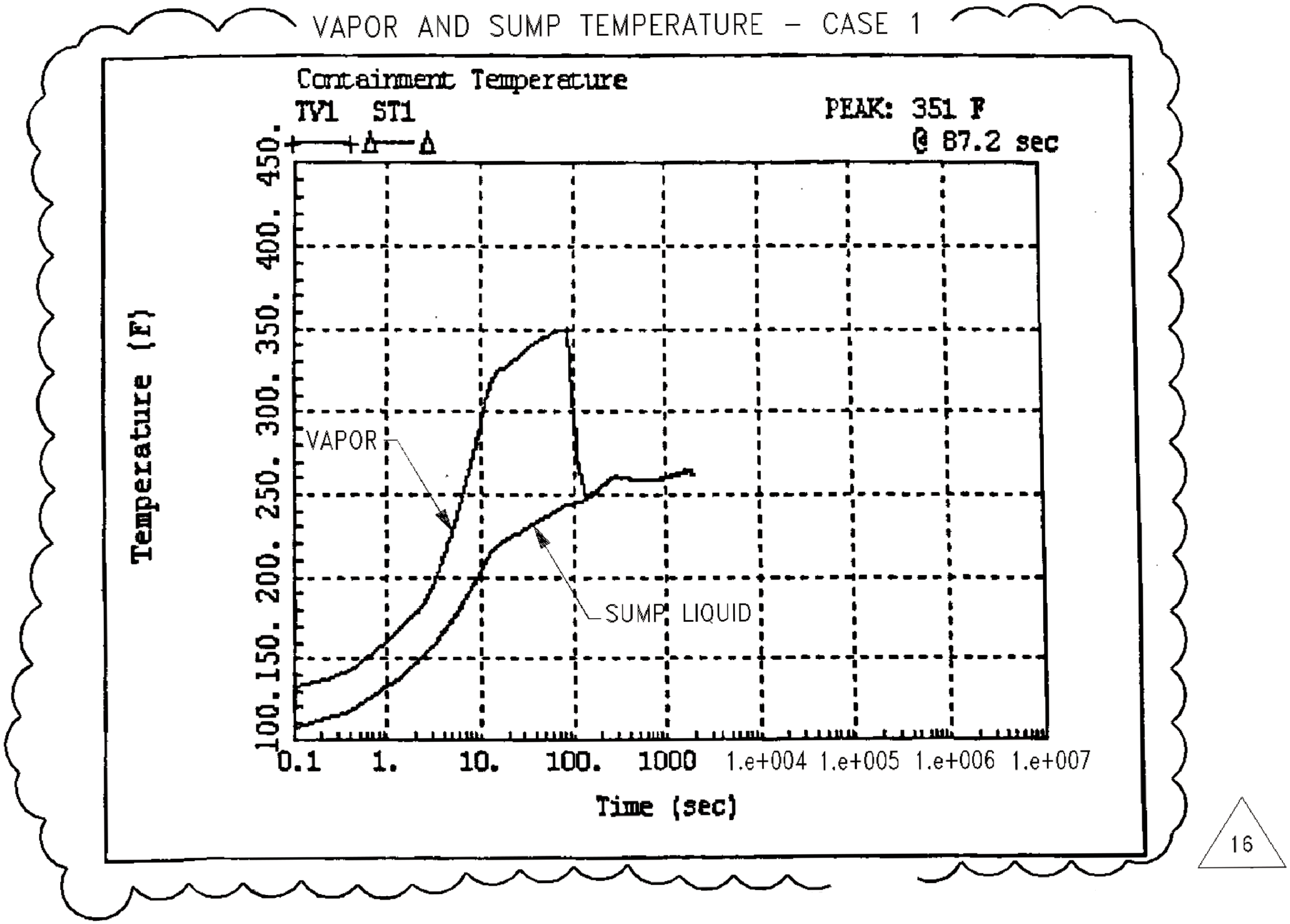
REV 21 5/08

SOUTHERN COMPANY
Energy to Serve Your World®

JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

RSG PRESSURE VERSUS TIME
STEAM LINE FULL D.E. BREAK
102% POWER, P_O = -1.5 PSIG

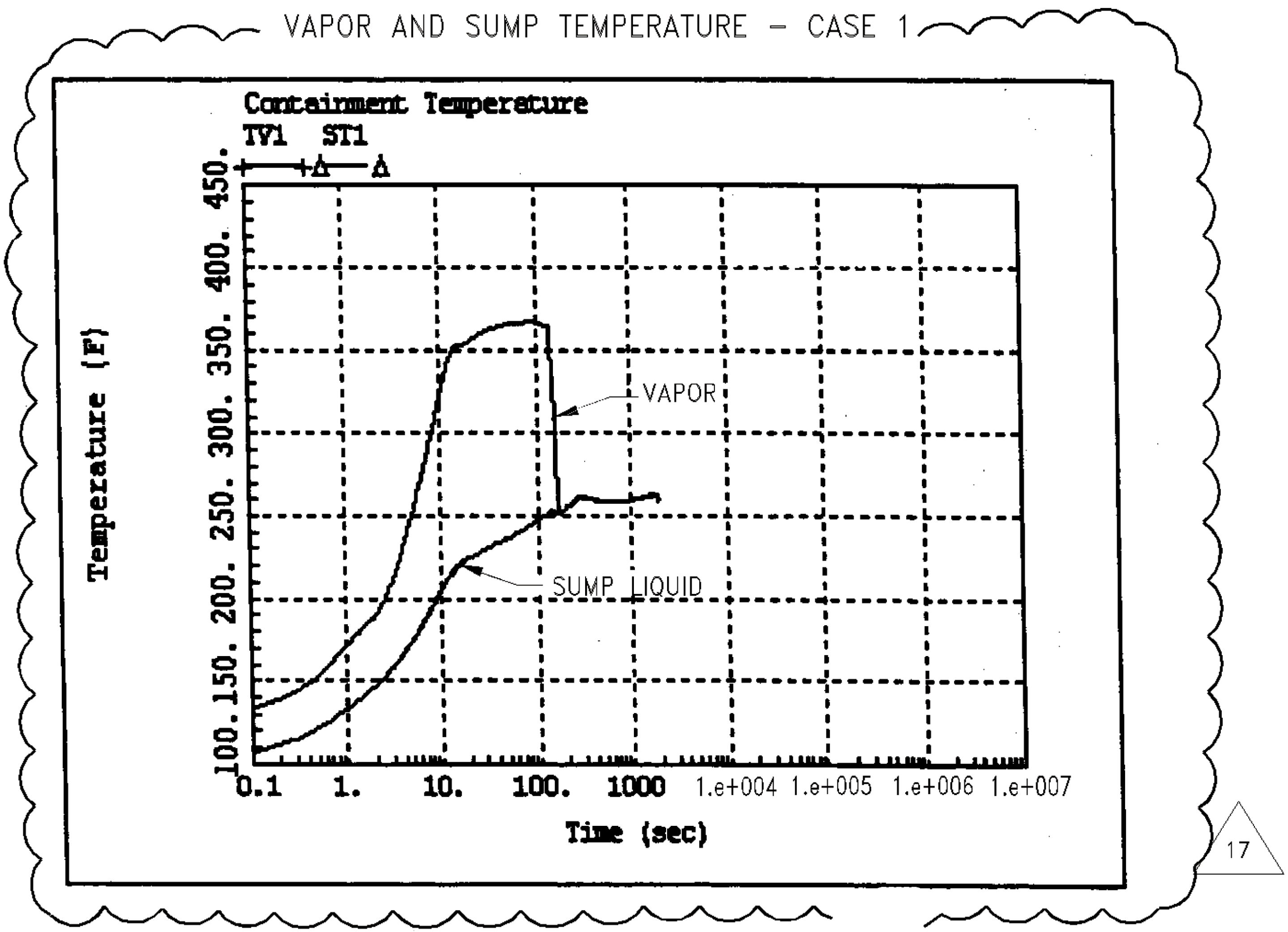
FIGURE 6.2-6A



REV 21 5/08

RSG TEMPERATURE VERSUS TIME
STEAM LINE FULL D.E. BREAK
102% POWER, P_O = +3 PSIG

FIGURE 6.2-7



REV 21 5/08

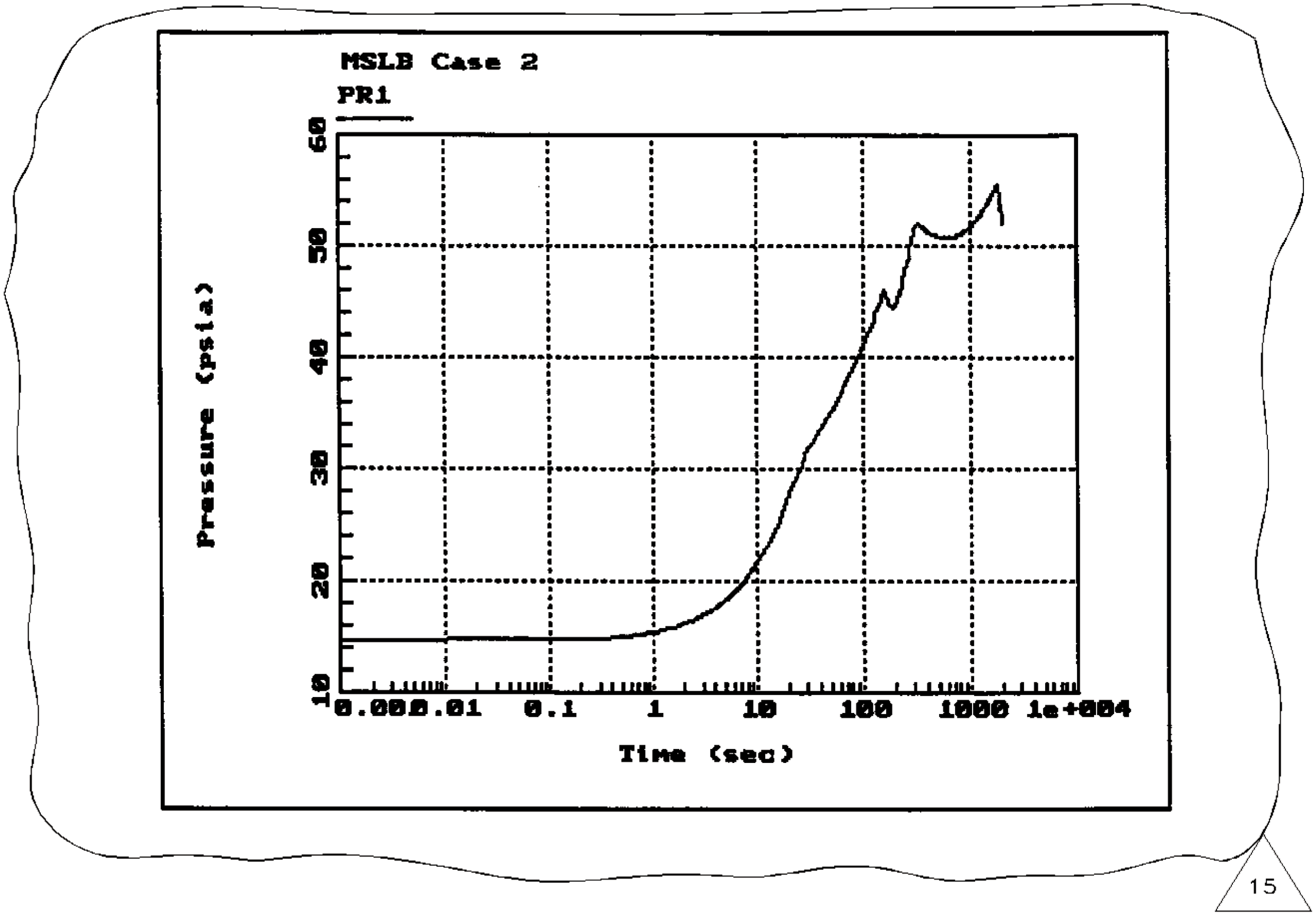
SOUTHERN COMPANY
Energy to Serve Your World®

JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

RSG TEMPERATURE VERSUS TIME
STEAM LINE FULL D.E. BREAK
102% POWER, P_O = -1.5 PSIG

FIGURE 6.2-7A

Containment Pressure — Case 2



REV 21 5/08

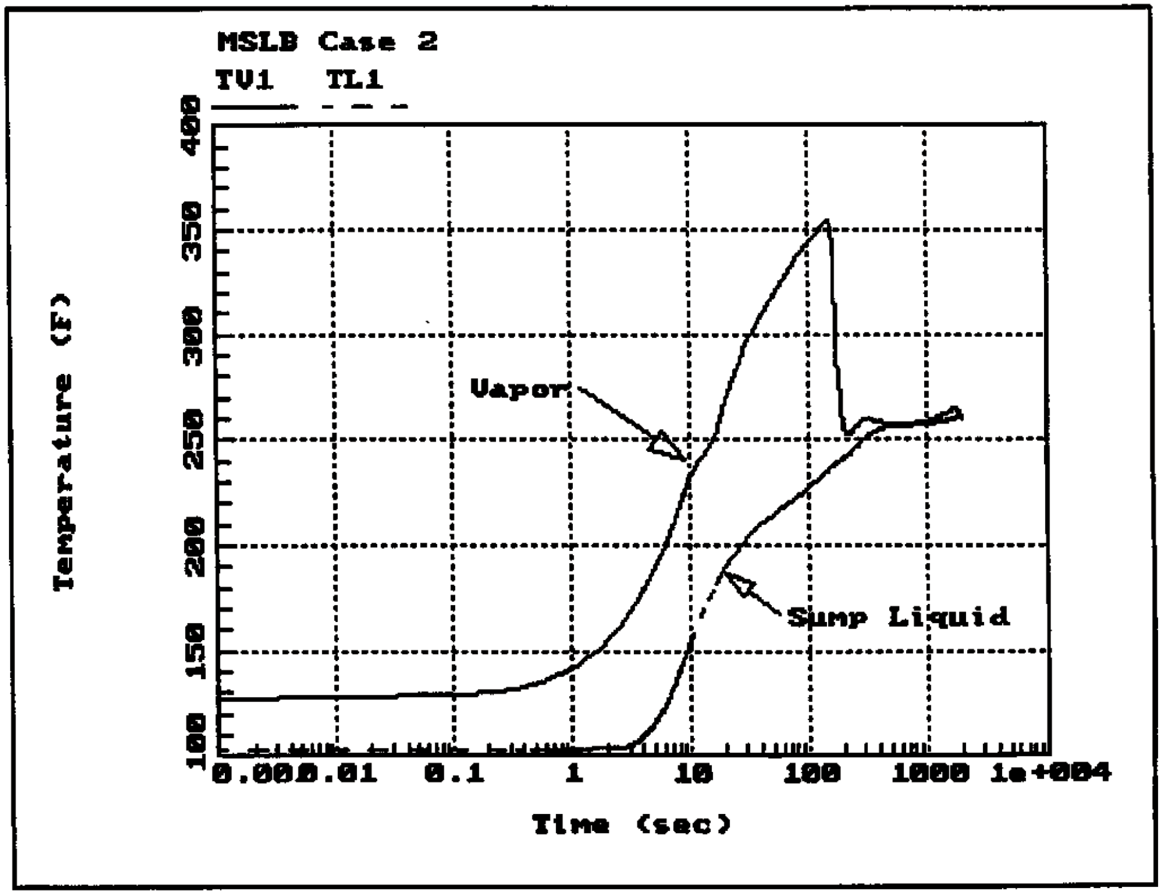
SOUTHERN COMPANY
Energy to Serve Your World®

JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

PRESSURE VERSUS TIME
STEAM LINE 0.7 ft^2 D.E. BREAK
102% POWER

FIGURE 6.2-8

Vapor and Sump Temperature — Case 2

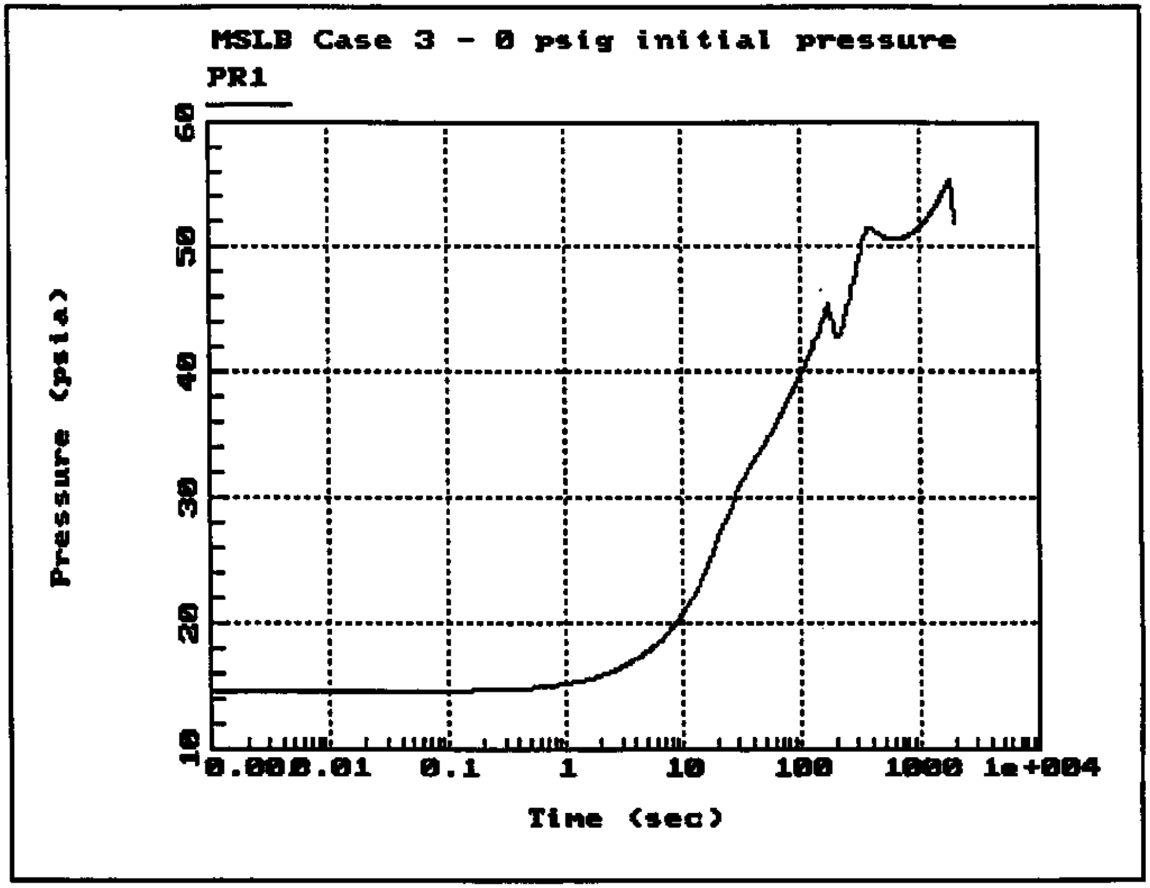


15

REV 21 5/08

SOUTHERN COMPANY *Energy to Serve Your World®*	JOSEPH M. FARLEY NUCLEAR PLANT UNIT 1 AND UNIT 2	TEMPERATURE VERSUS TIME STEAM LINE 0.7 ft^2 D.E. BREAK 102% POWER
		FIGURE 6.2-9

Containment Pressure — Case 3



15

REV 21 5/08

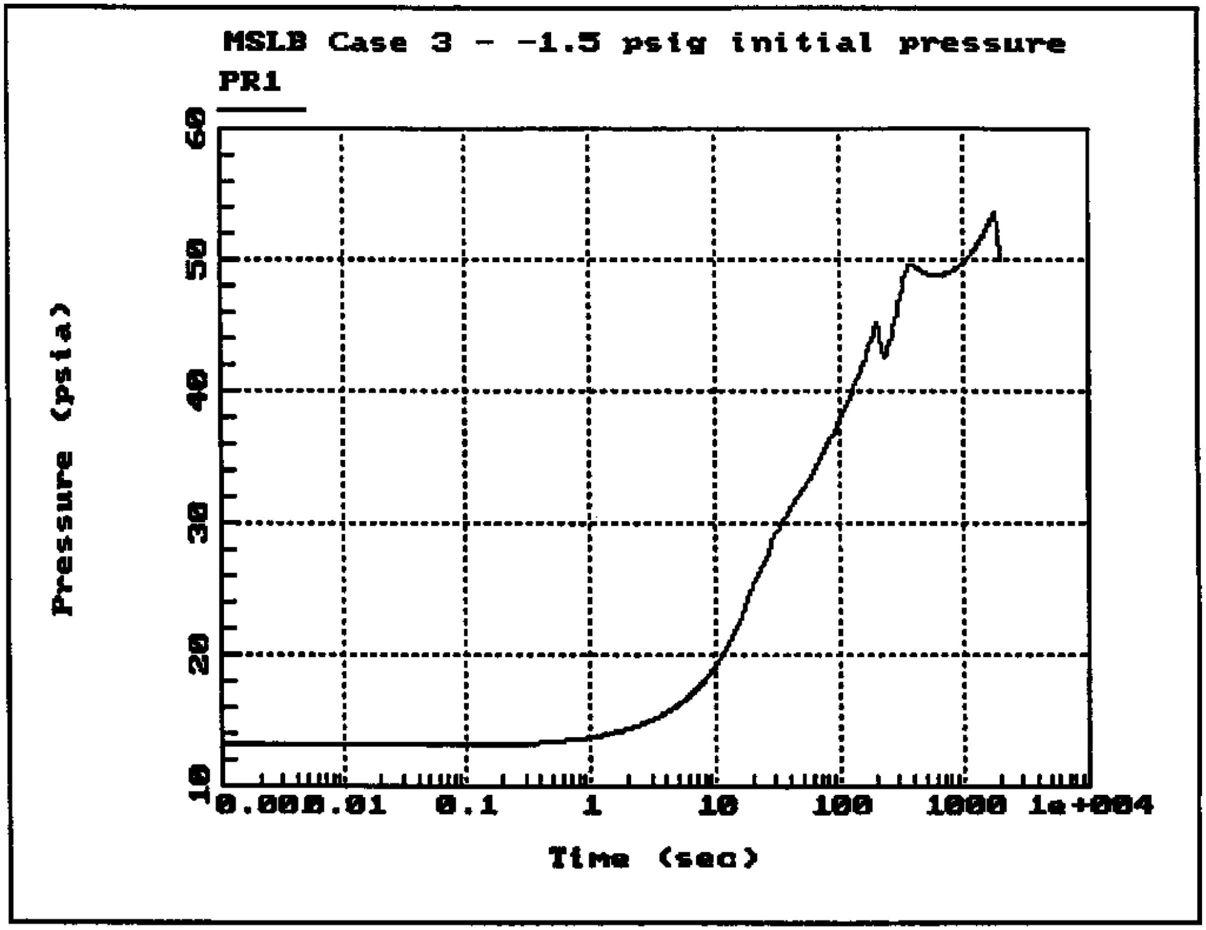
SOUTHERN COMPANY
Energy to Serve Your World®

JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

PRESSURE VERSUS TIME
STEAM LINE 0.6 ft^2 D.E. BREAK
102% POWER, P_O = 0 PSIG

FIGURE 6.2-10

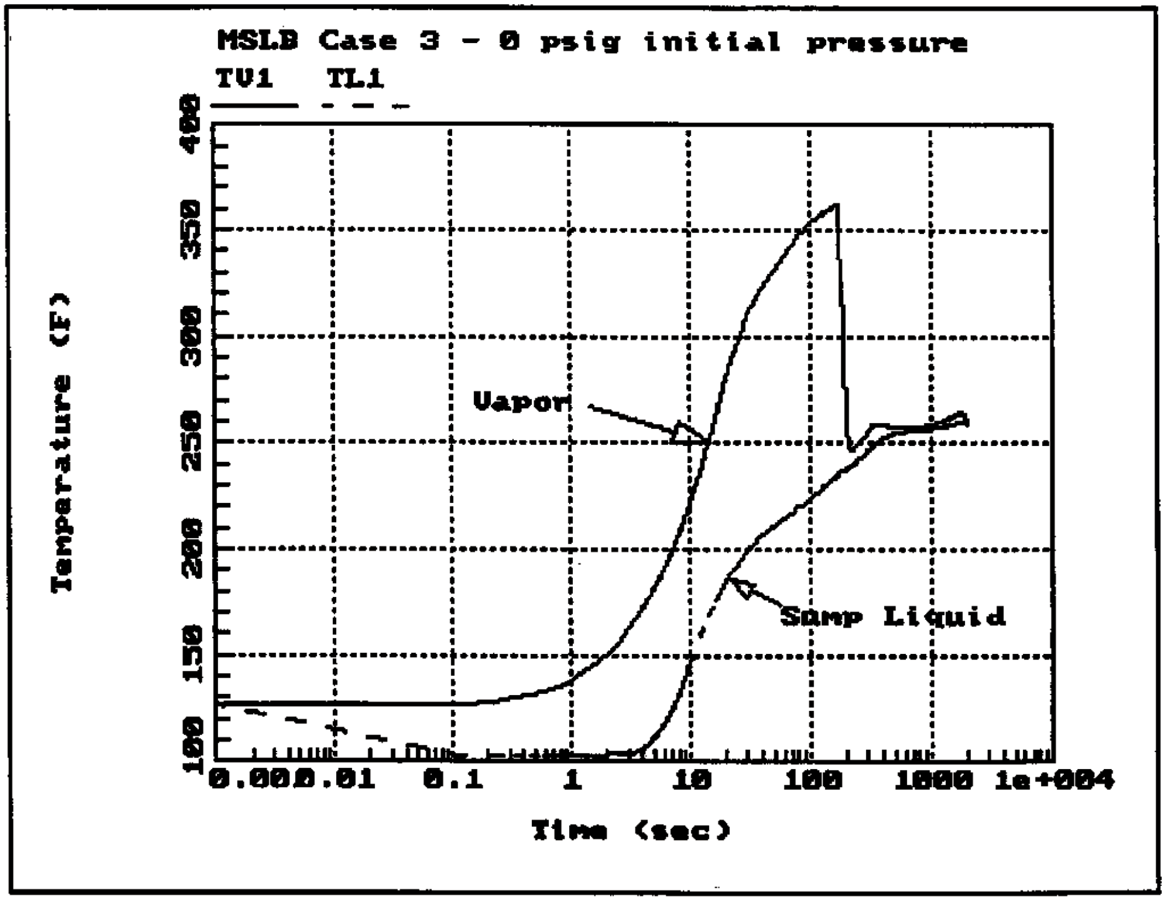
Containment Pressure — Case 3



REV 21 5/08

SOUTHERN COMPANY Energy to Serve Your World®	JOSEPH M. FARLEY NUCLEAR PLANT UNIT 1 AND UNIT 2	PRESSURE VERSUS TIME STEAM LINE 0.6 ft^2 D.E. BREAK 102% POWER, P_O = -1.5 PSIG
		FIGURE 6.2-10A

Vapor and Sump Temperature — Case 3



15

REV 21 5/08

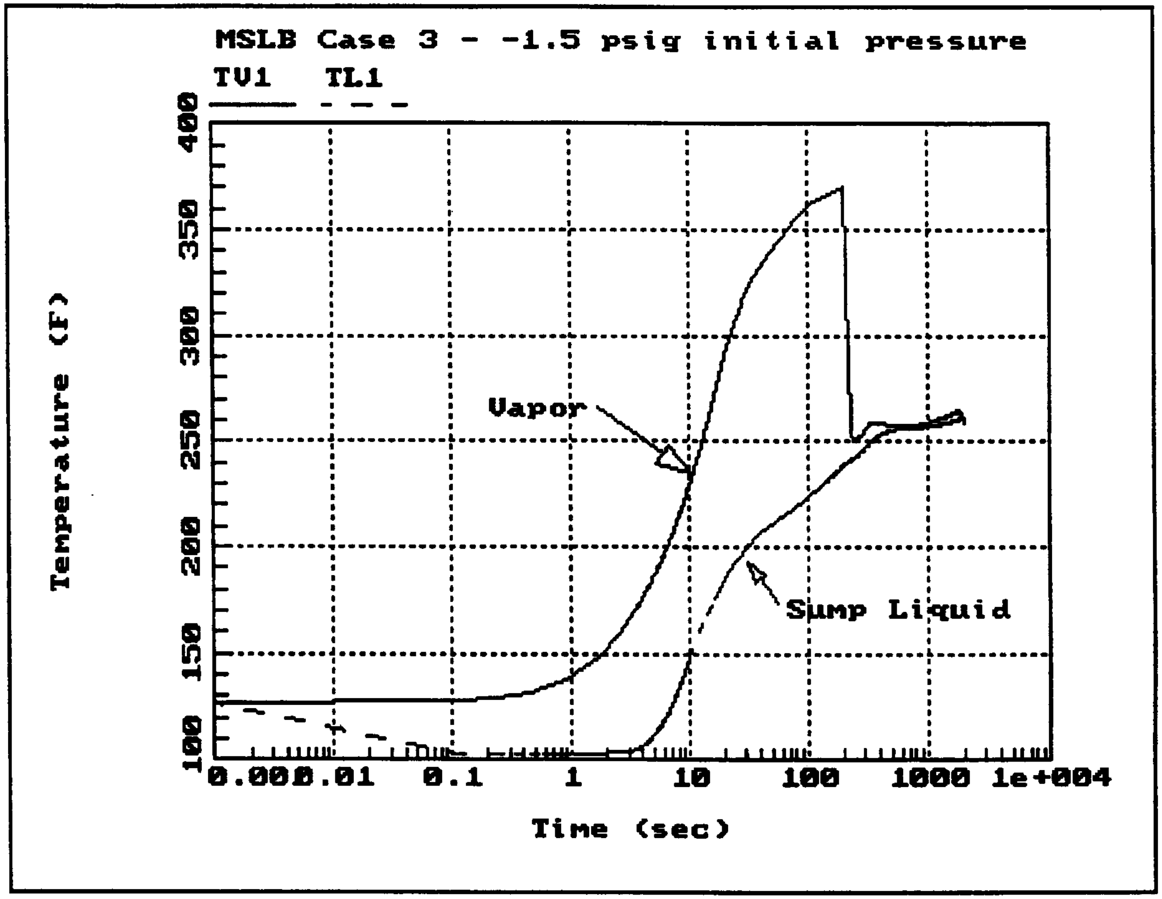
SOUTHERN COMPANY
Energy to Serve Your World®

JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

TEMPERATURE VERSUS TIME
STEAM LINE 0.6 ft^2 D.E. BREAK
102% POWER, P_O = - PSIG

FIGURE 6.2-11

Vapor and Sump Temperature — Case 3



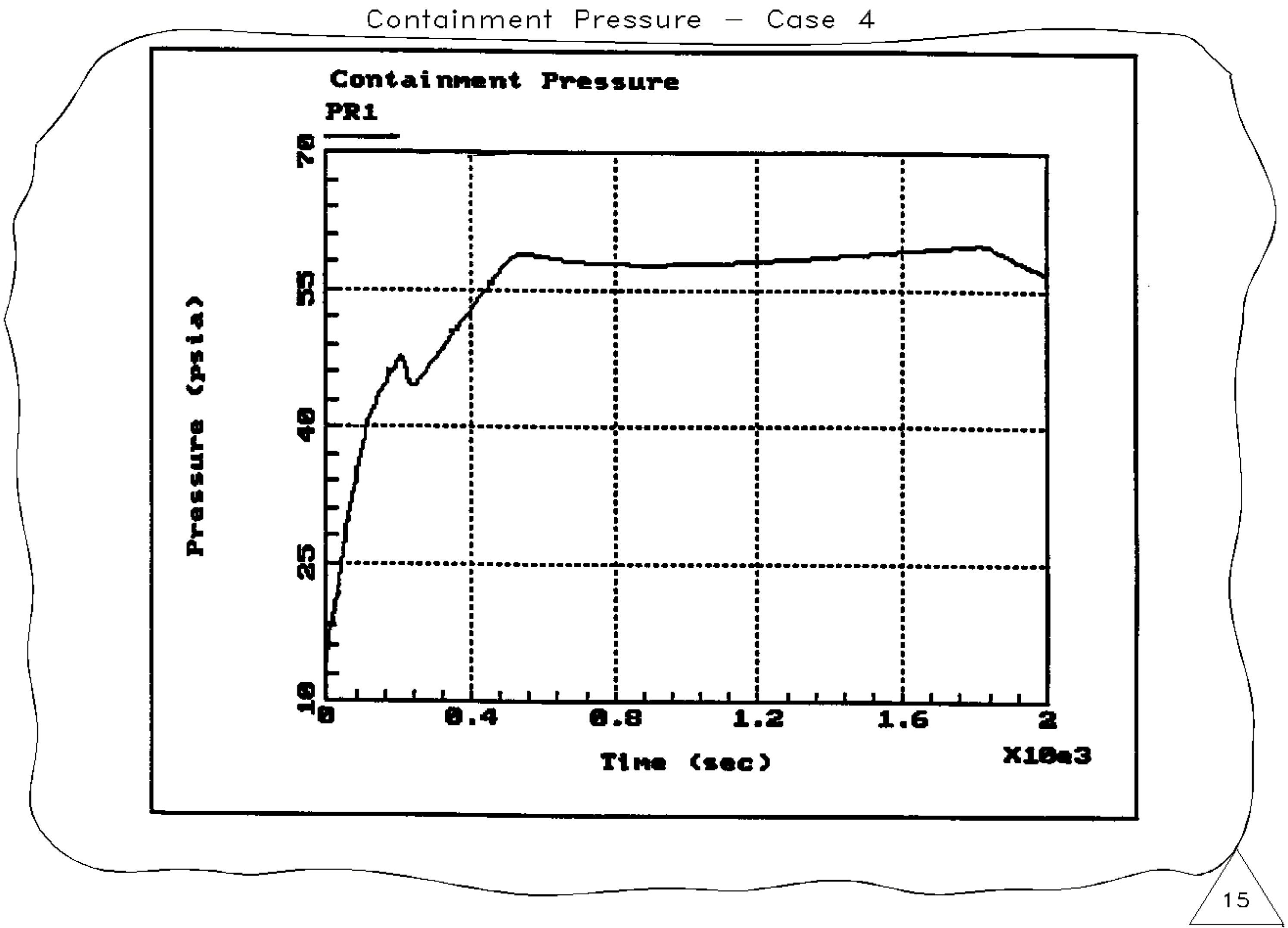
REV 21 5/08

SOUTHERN COMPANY
Energy to Serve Your World®

JOSEPH M. FARLEY NUCLEAR PLANT UNIT 1 AND UNIT 2

TEMPERATURE VERSUS TIME
STEAM LINE 0.6 ft^2 D.E. BREAK
102% POWER, P_O = -1.5 PSIG

FIGURE 6.2-11A



REV 21 5/08

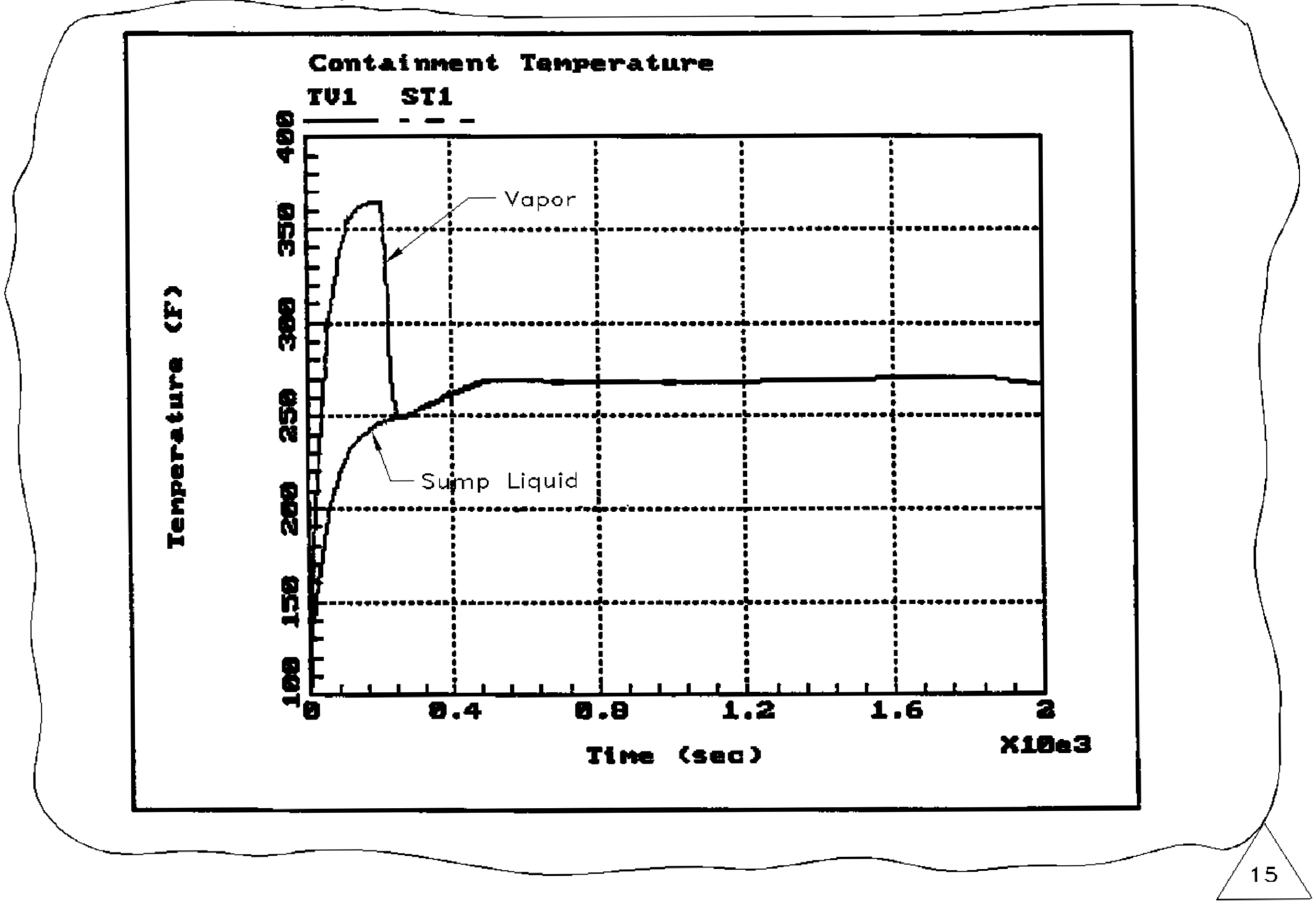
SOUTHERN COMPANY
Energy to Serve Your World®

JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

PRESSURE VERSUS TIME
STEAM LINE 0.528 ft^2 SPLIT
102% POWER

FIGURE 6.2-12

Vapor and Sump Temperature — Case 4



REV 21 5/08

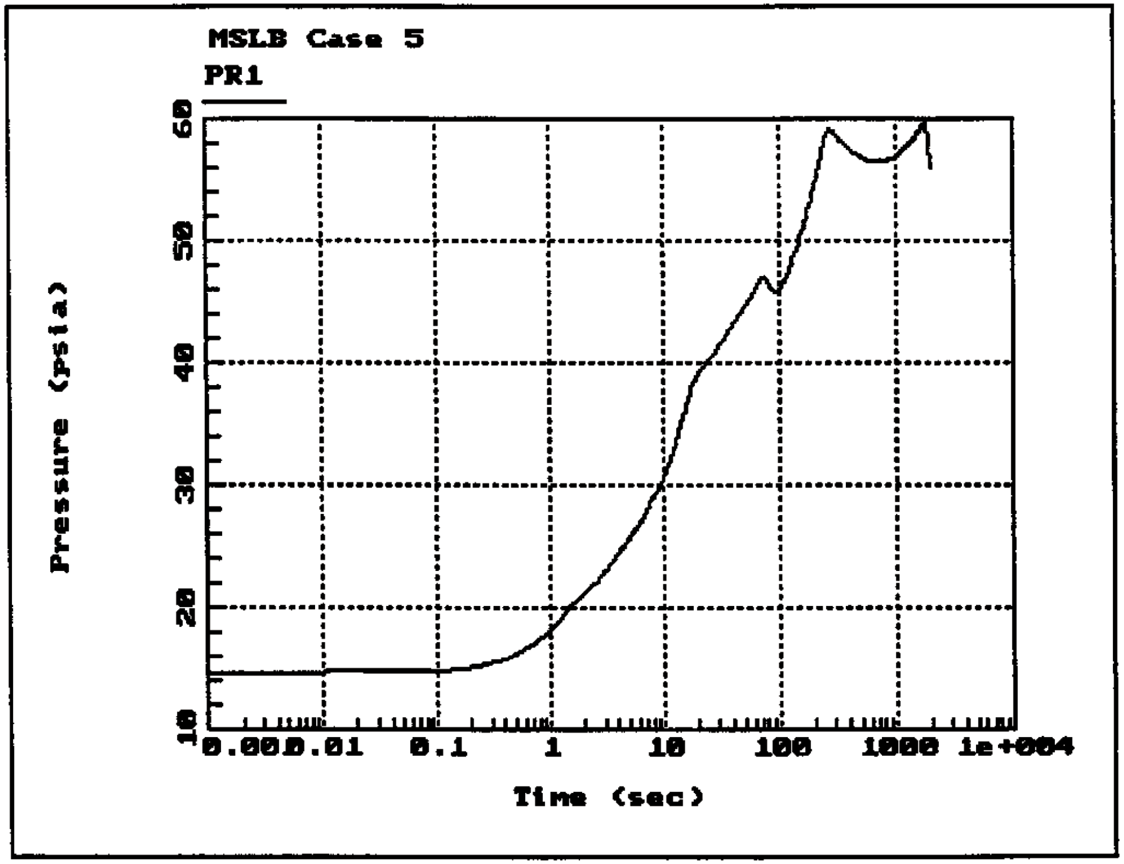
SOUTHERN COMPANY
Energy to Serve Your World®

JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

TEMPERATURE VERSUS TIME
STEAM LINE 0.528 ft^2 SPLIT
102% POWER

FIGURE 6.2-13

Containment Pressure — Case 5



15

REV 21 5/08

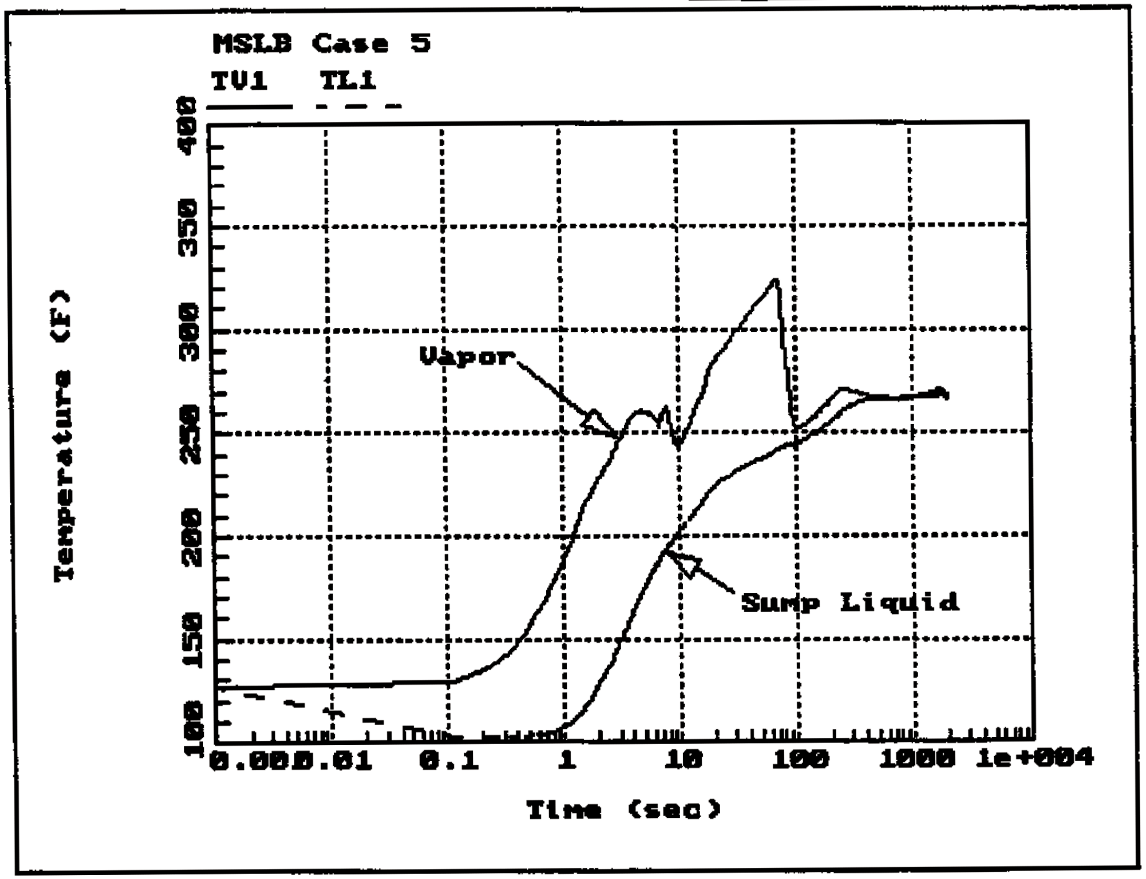
SOUTHERN COMPANY
Energy to Serve Your World®

JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

PRESSURE VERSUS TIME
STEAM LINE FULL D.E. BREAK
70% POWER

FIGURE 6.2-14

Vapor and Sump Temperature — Case 5



15

REV 21 5/08

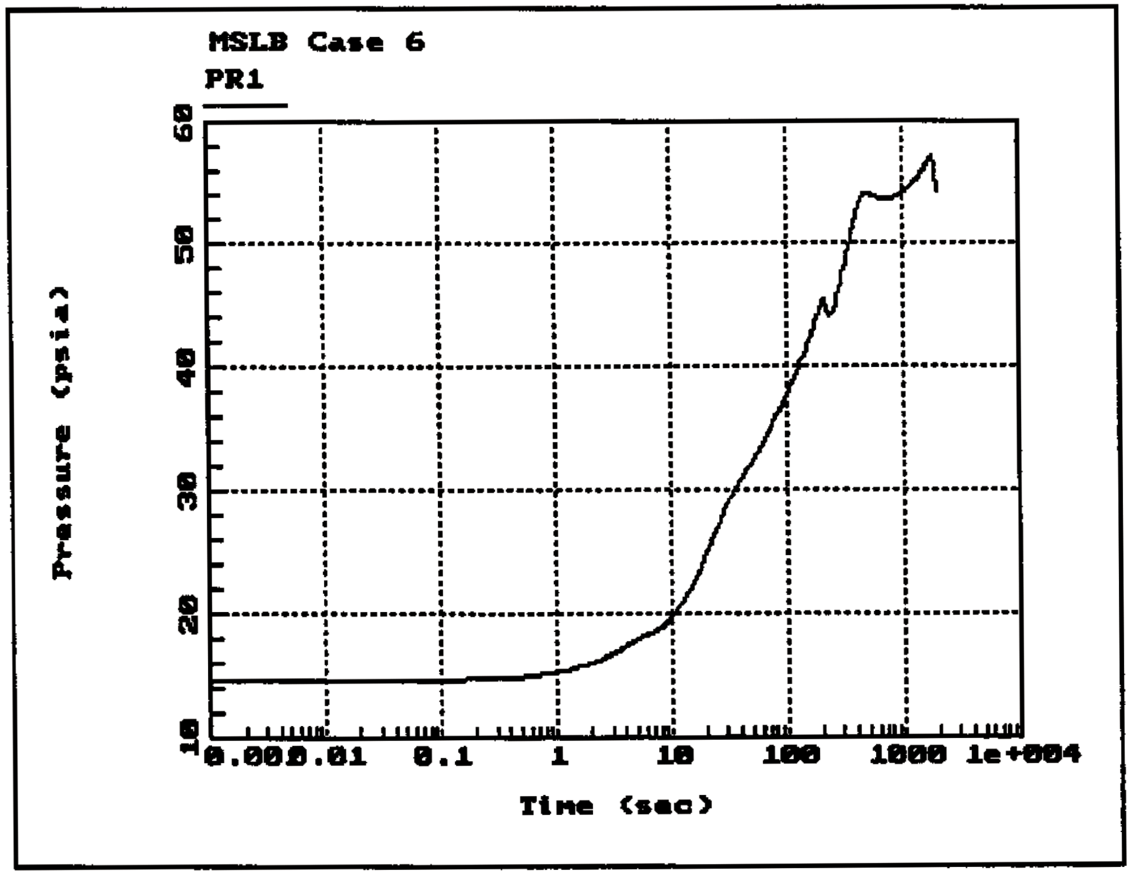
SOUTHERN COMPANY
Energy to Serve Your World®

JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

TEMPERATURE VERSUS TIME
STEAM LINE FULL D.E. BREAK
70% POWER

FIGURE 6.2-15

Containment Pressure — Case 6



15

REV 21 5/08

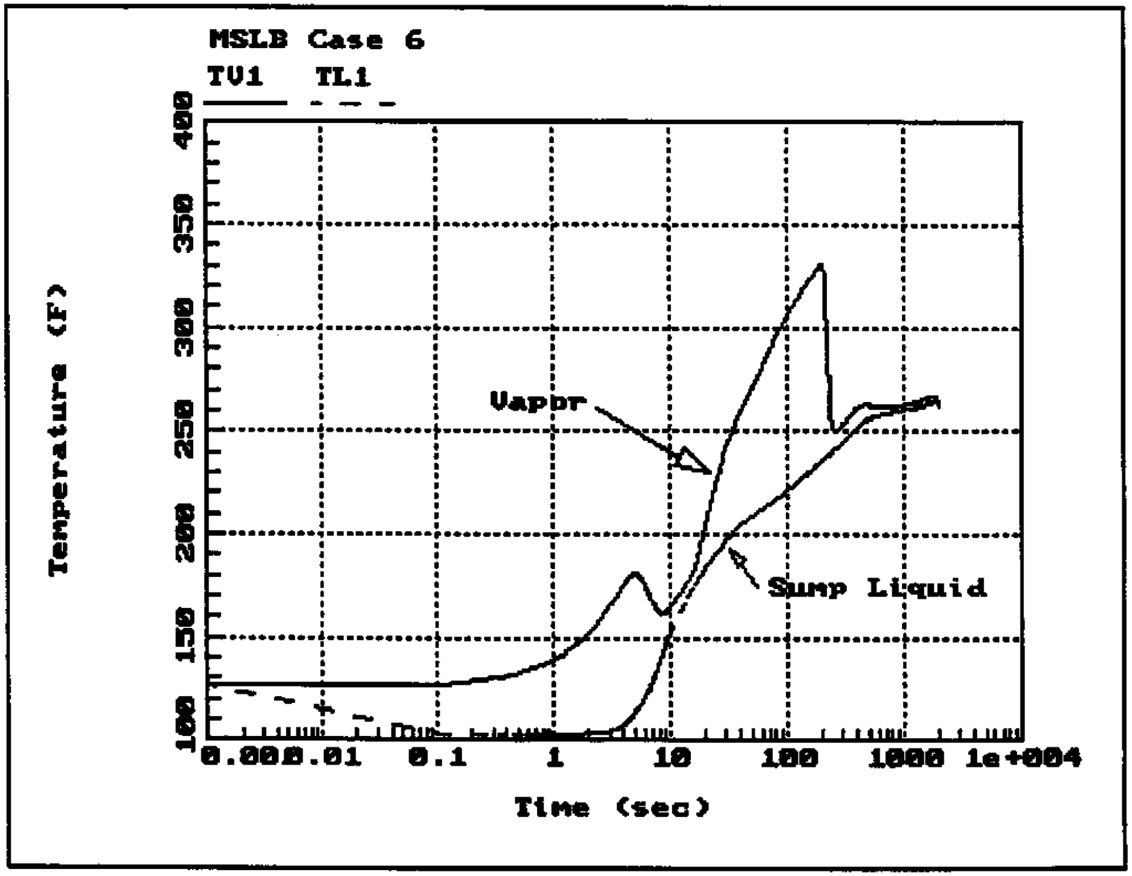
SOUTHERN COMPANY
Energy to Serve Your World®

JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

PRESSURE VERSUS TIME
STEAM LINE 0.6 ft^2 D.E. BREAK
70% POWER

FIGURE 6.2-16

Vapor and Sump Temperature — Case 6



15

REV 21 5/08

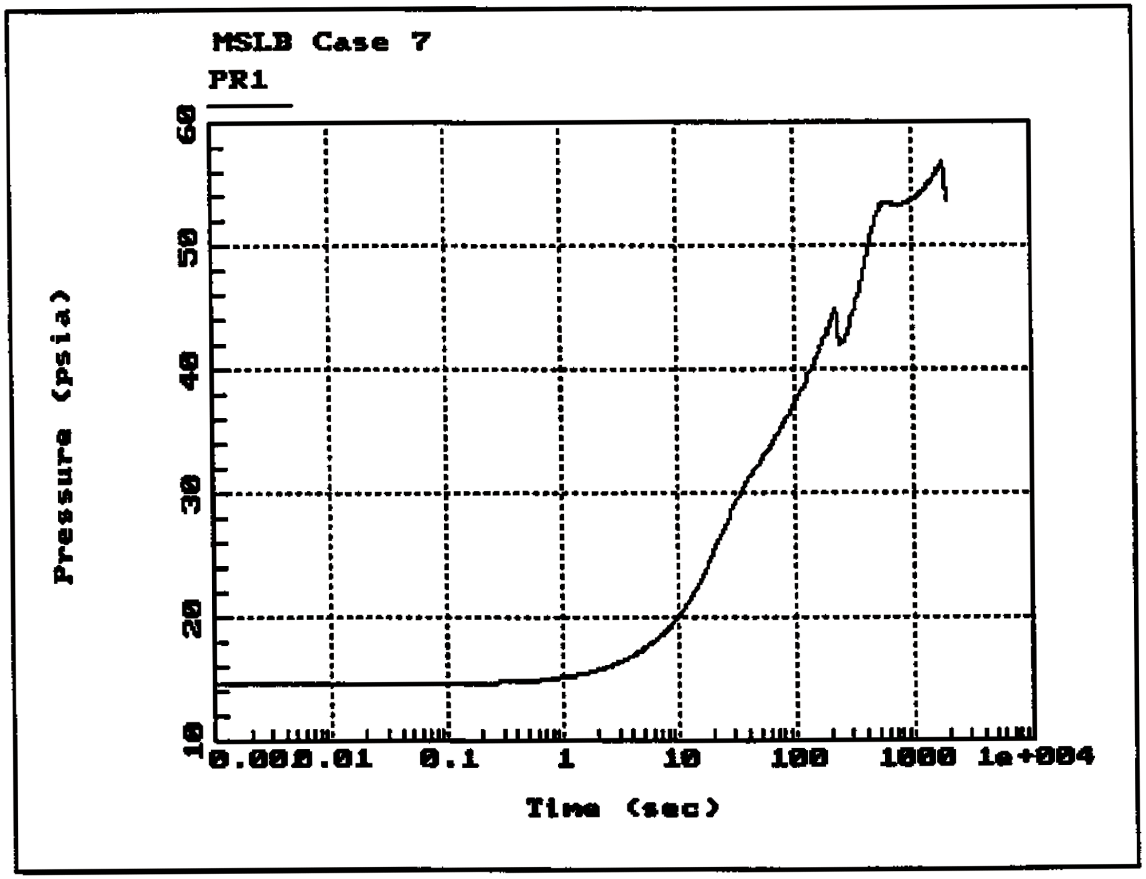
SOUTHERN COMPANY
Energy to Serve Your World®

JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

TEMPERATURE VERSUS TIME
STEAM LINE 0.6 ft^2 D.E. BREAK
70% POWER

FIGURE 6.2-17

Containment Pressure — Case 7



15

REV 21 5/08

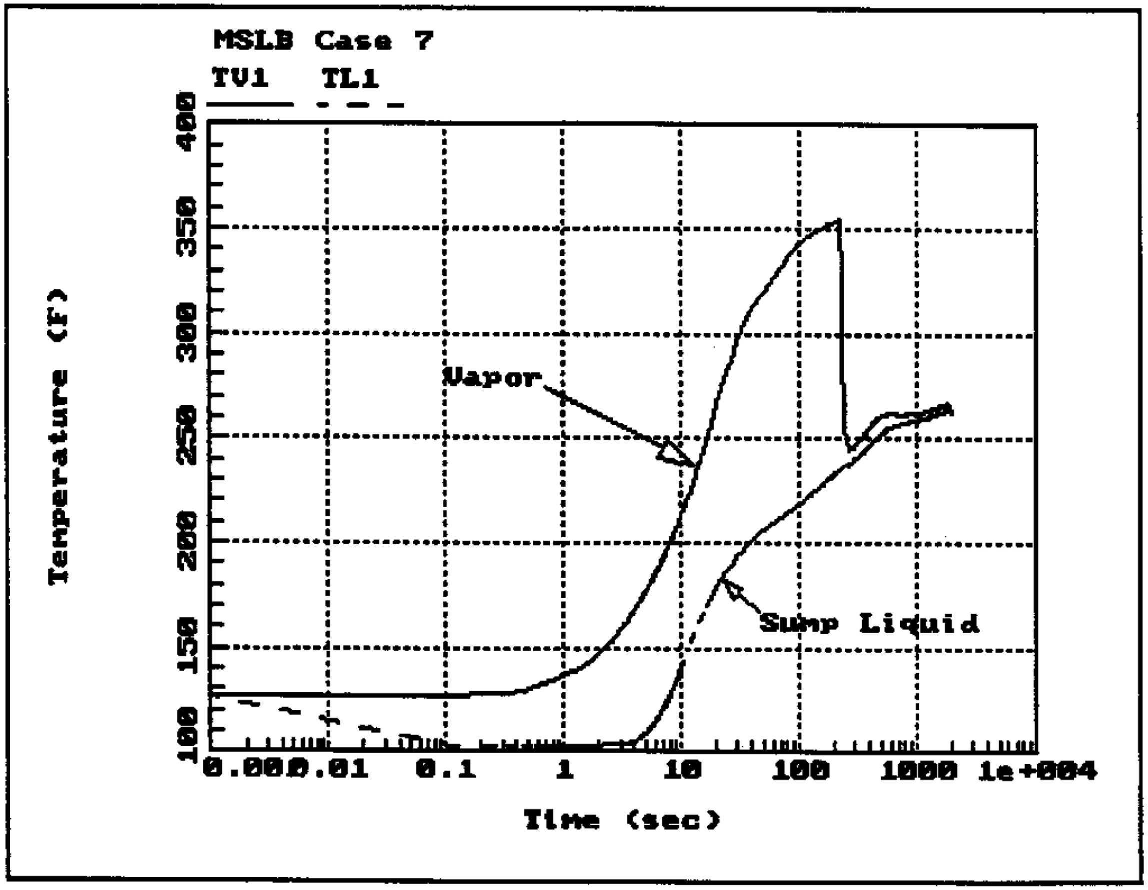
SOUTHERN COMPANY
Energy to Serve Your World®

JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

PRESSURE VERSUS TIME
STEAM LINE 0.5 ft^2 D.E. BREAK
70% POWER

FIGURE 6.2-18

Vapor and Sump Temperature — Case 7



15

REV 21 5/08

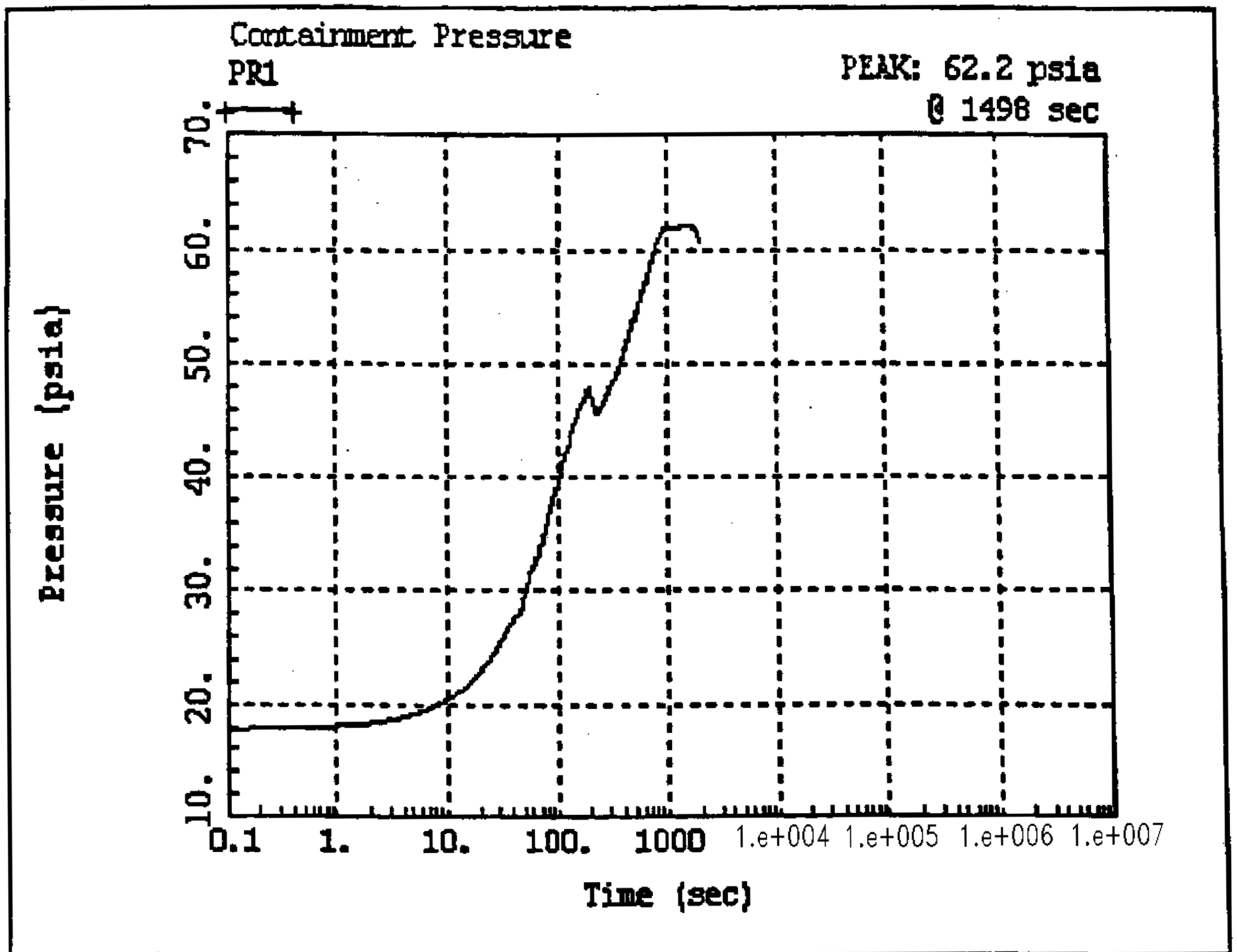
SOUTHERN COMPANY
Energy to Serve Your World®

JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

TEMPERATURE VERSUS TIME
STEAM LINE 0.5 ft^2 D.E. BREAK
70% POWER

FIGURE 6.2-19

CONTAINMENT PRESSURE – CASE 8



16

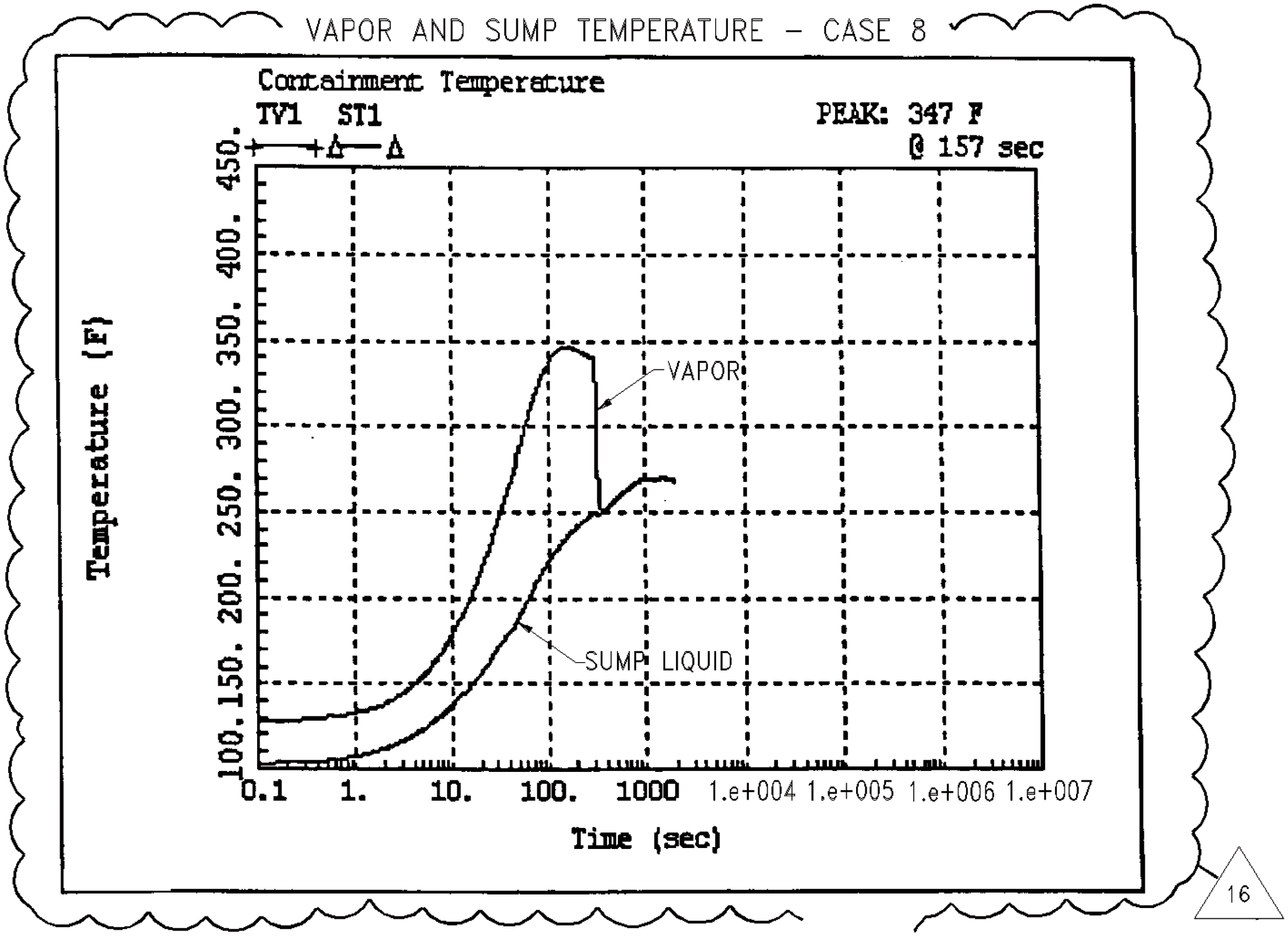
REV 21 5/08

SOUTHERN COMPANY
Energy to Serve Your World®

JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

RSG PRESSURE VERSUS TIME
STEAM LINE 0.47 ft^2 SPLIT
70% POWER, P_O = +3 PSIG

FIGURE 6.2-20



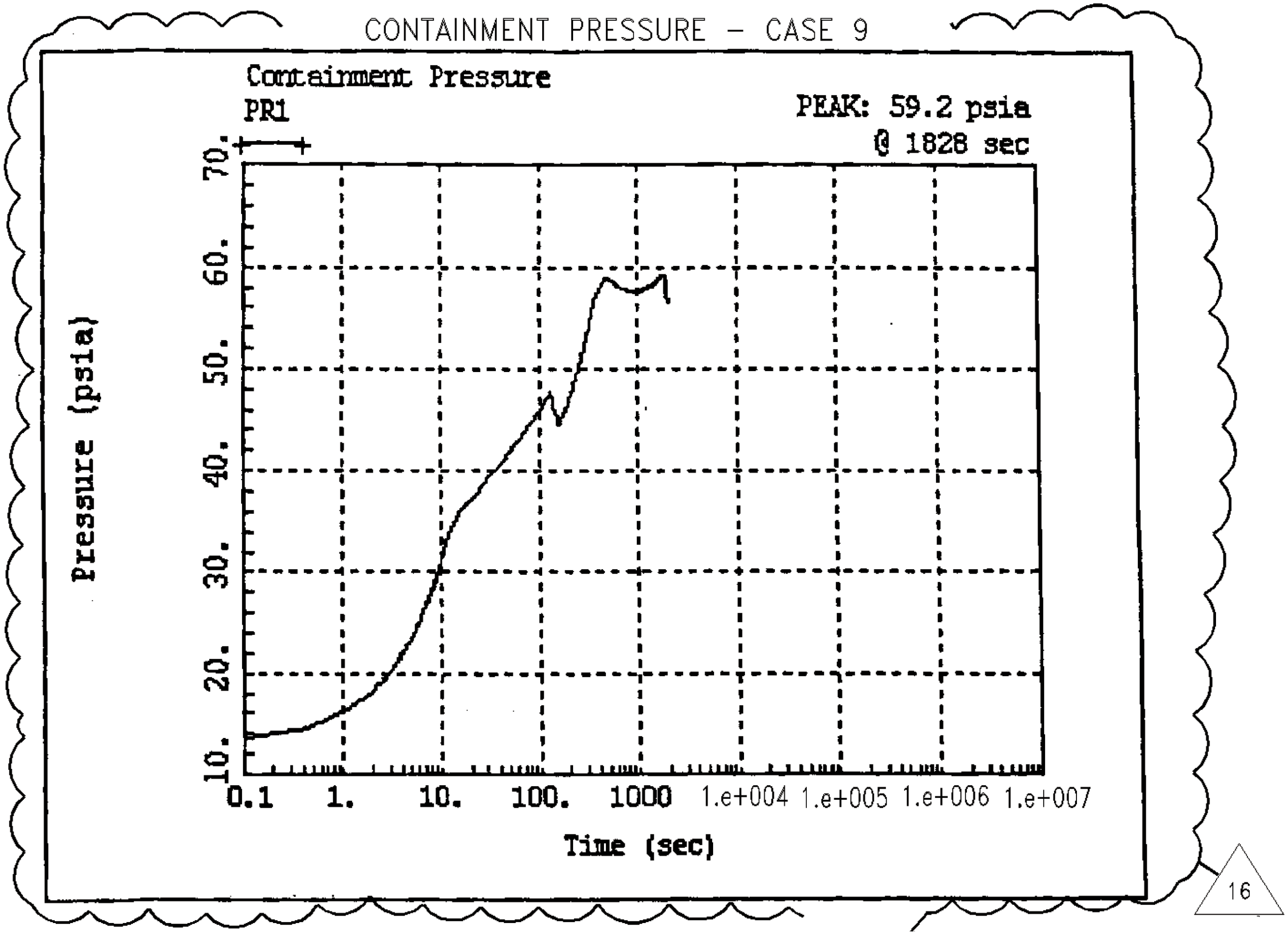
REV 21 5/08

SOUTHERN COMPANY
Energy to Serve Your World®

JOSEPH M. FARLEY NUCLEAR PLANT UNIT 1 AND UNIT 2

RSG TEMPERATURE VERSUS TIME
STEAM LINE 0.47 ft^2 SPLIT
70% POWER, P_O = -1.5 PSIG

FIGURE 6.2-21



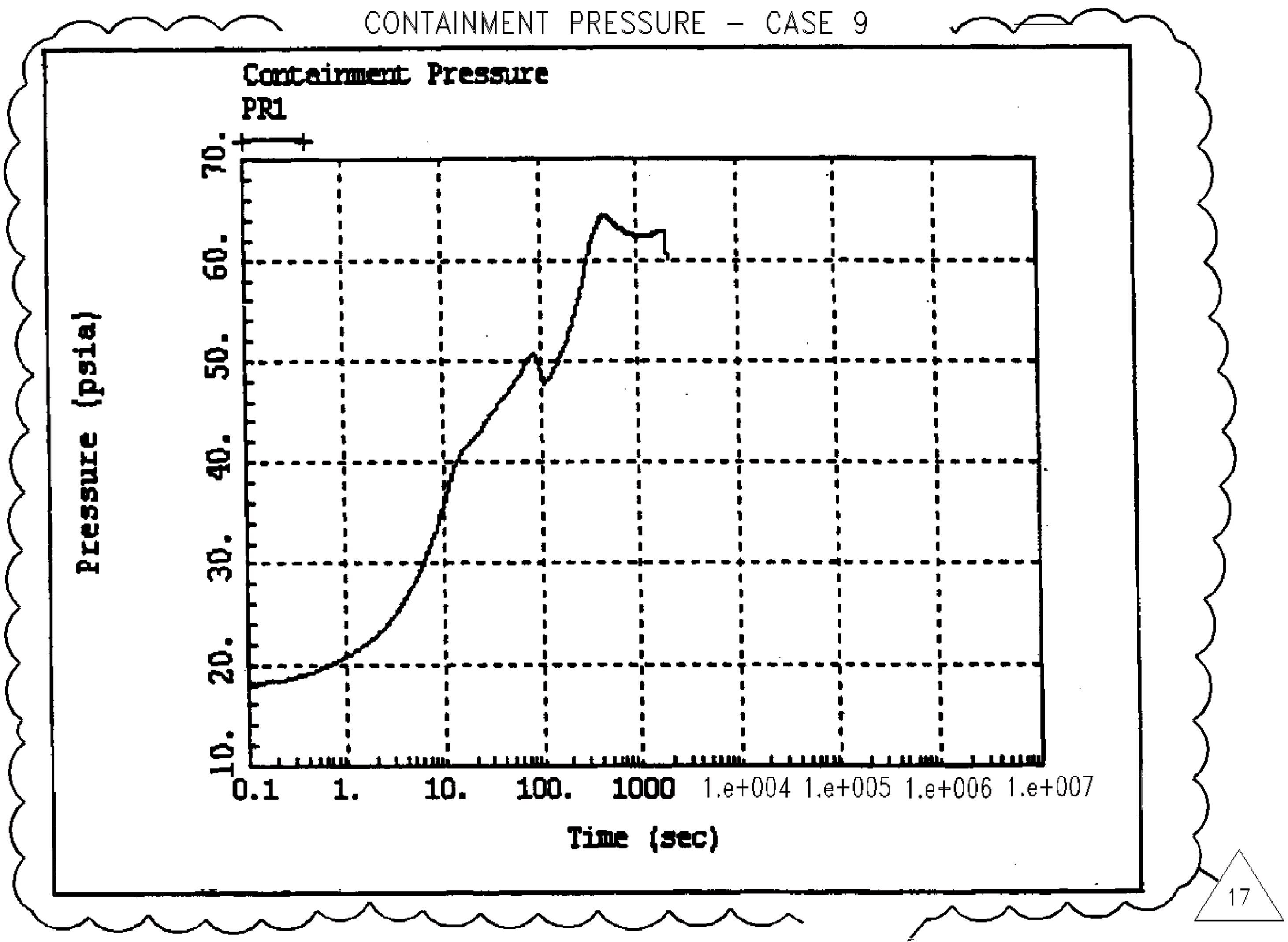
REV 21 5/08

SOUTHERN COMPANY
Energy to Serve Your World®

JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

RSG PRESSURE VERSUS TIME
STEAM LINE FULL D.E. BREAK
30% POWER, P_O = -1.5 PSIG

FIGURE 6.2-22



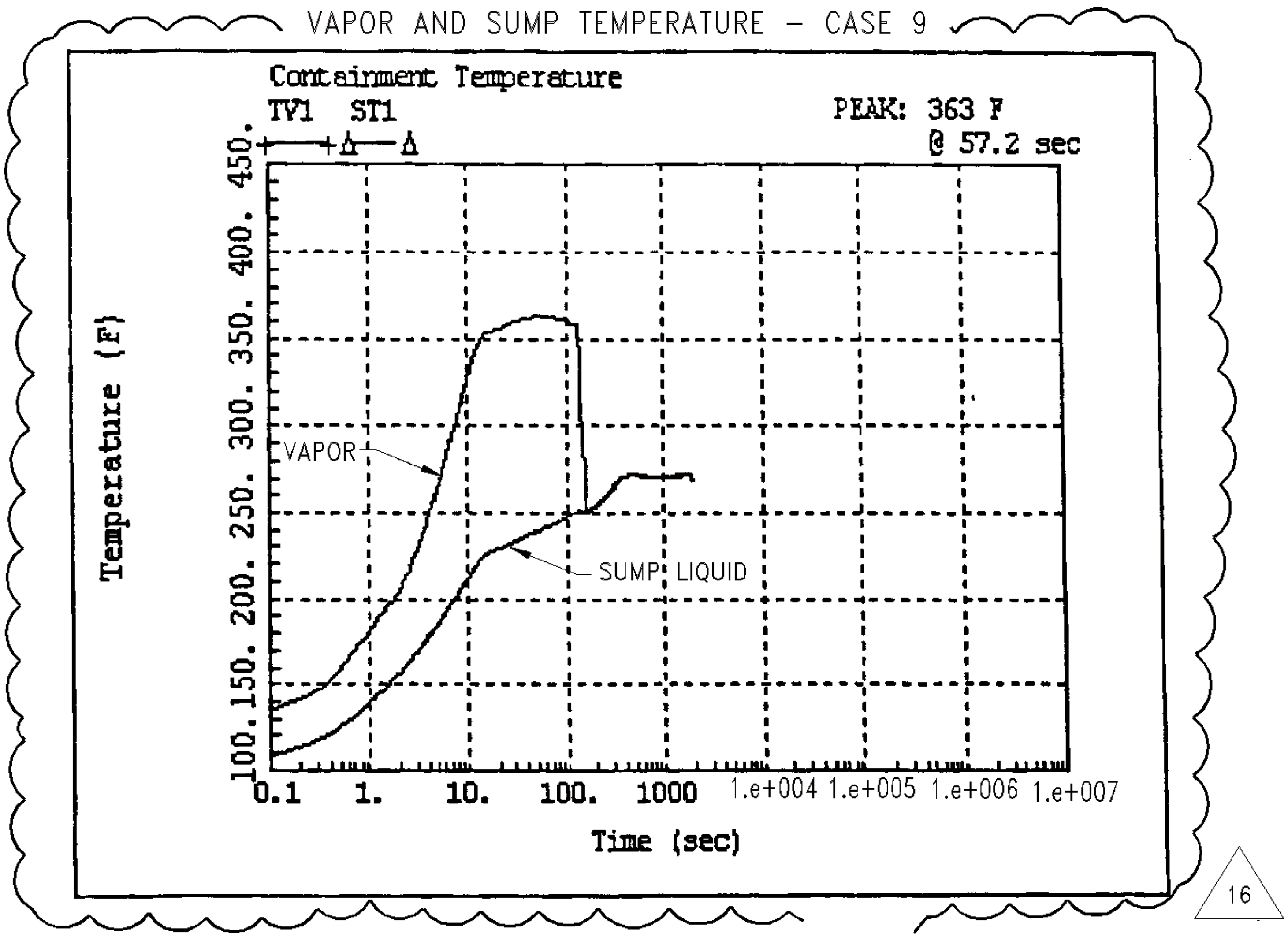
REV 21 5/08

SOUTHERN COMPANY
Energy to Serve Your World®

JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

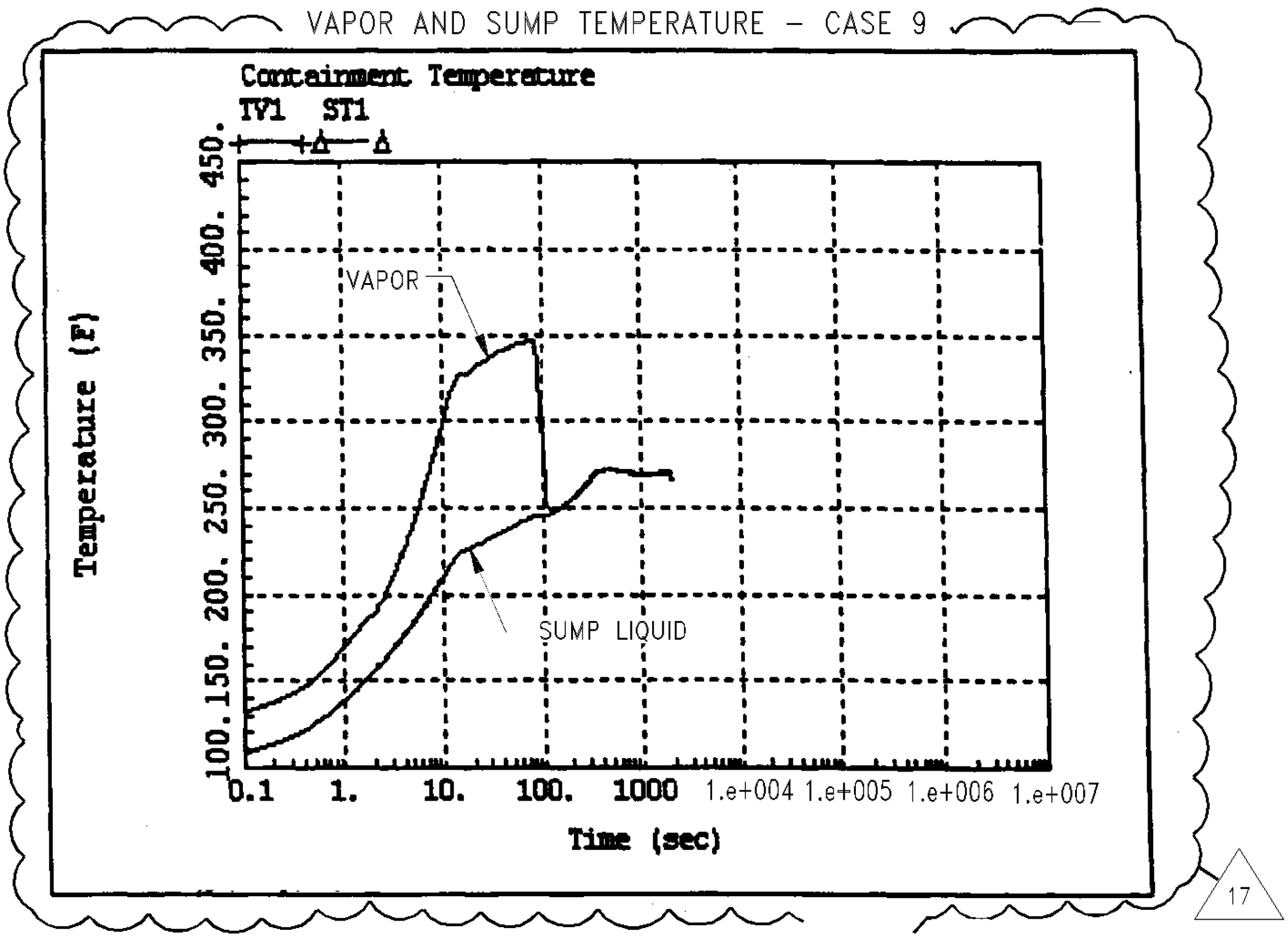
RSG PRESSURE VERSUS TIME
STEAM LINE FULL D.E. BREAK
30% POWER, P_O = +3 PSIG

FIGURE 6.2-22A



REV 21 5/08

SOUTHERN COMPANY *Energy to Serve Your World®*	**JOSEPH M. FARLEY NUCLEAR PLANT UNIT 1 AND UNIT 2**	RSG TEMPERATURE VERSUS TIME STEAM LINE FULL D.E. BREAK 30% POWER, P_O = -1.5 PSIG
		FIGURE 6.2-23



REV 21 5/08

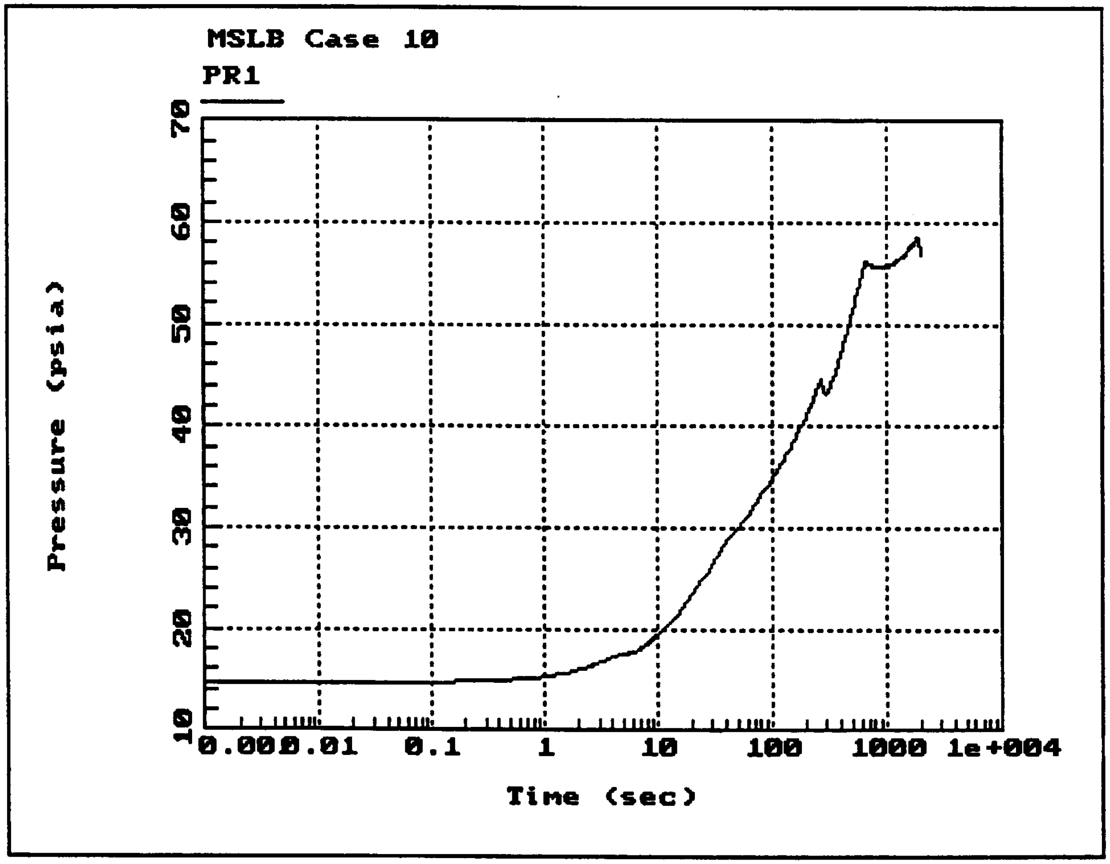
SOUTHERN COMPANY
Energy to Serve Your World®

JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

RSG TEMPERATURE VERSUS TIME
STEAM LINE FULL D.E. BREAK
30% POWER, P_O = +3 PSIG

FIGURE 6.2-23A

Containment Pressure — Case 10

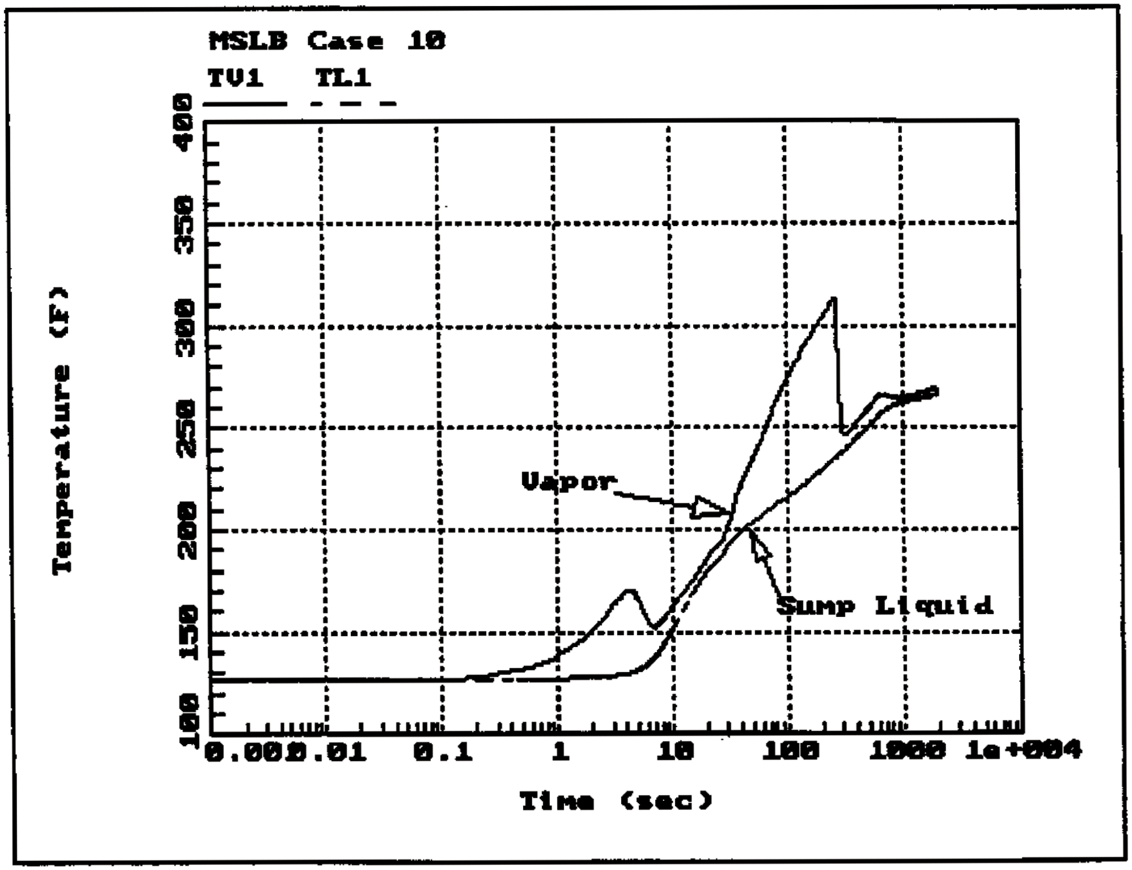


15

REV 21 5/08

SOUTHERN COMPANY *Energy to Serve Your World®* JOSEPH M. FARLEY NUCLEAR PLANT UNIT 1 AND UNIT 2	PRESSURE VERSUS TIME STEAM LINE 0.5 ft^2 D.E. BREAK 30% POWER
	FIGURE 6.2-24

Vapor and Sump Temperature — Case 10



15

REV 21 5/08

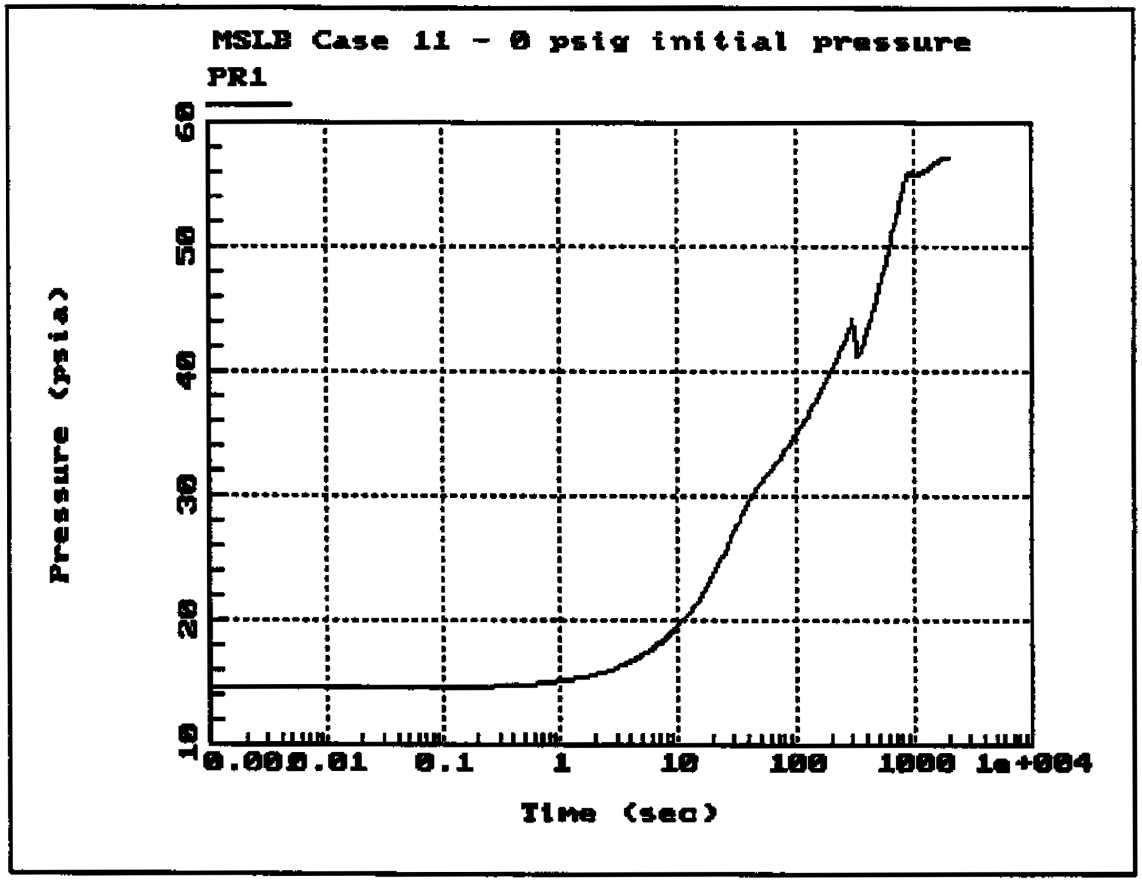
SOUTHERN COMPANY
Energy to Serve Your World®

JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

TEMPERATURE VERSUS TIME
STEAM LINE 0.5 ft^2 D.E. BREAK
30% POWER

FIGURE 6.2-25

Containment Pressure — Case 11



15

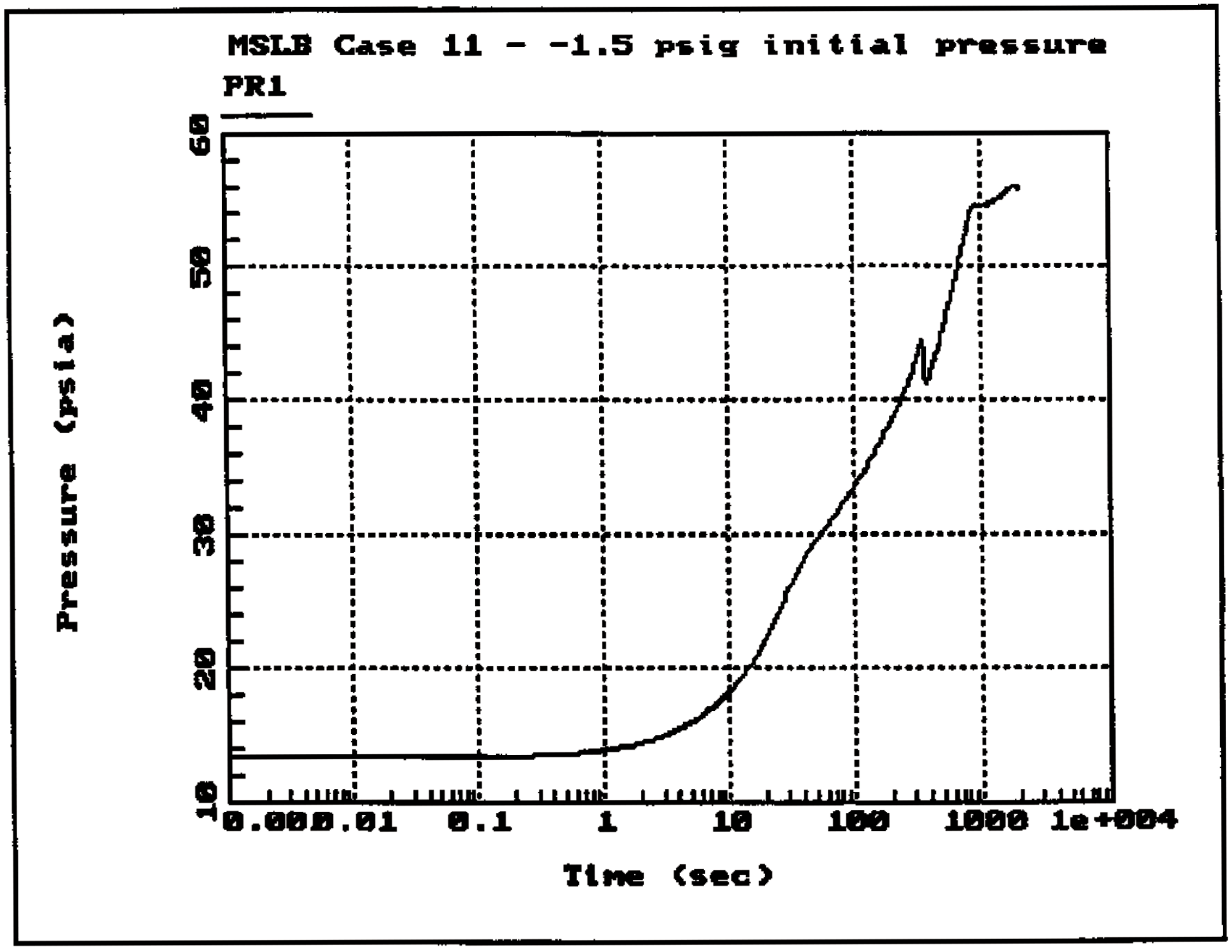
REV 21 5/08

JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

PRESSURE VERSUS TIME
STEAM LINE 0.4 ft^2 D.E. BREAK
30% POWER, P_O = 0 PSIG

FIGURE 6.2-26

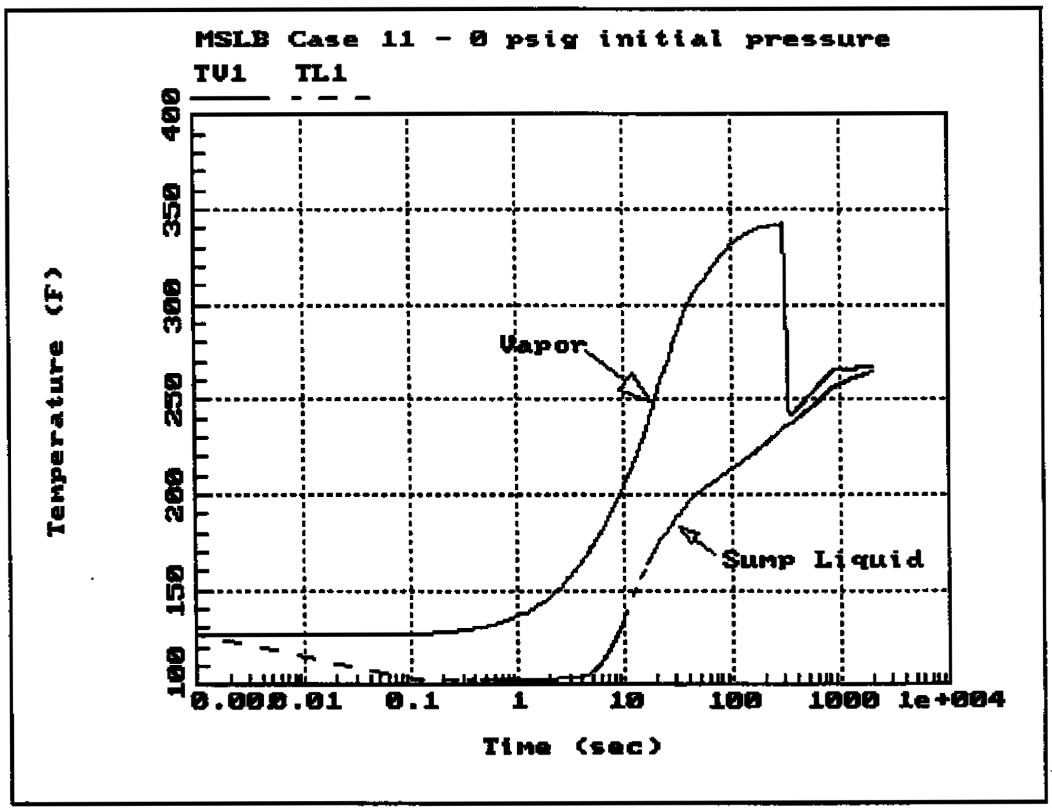
Containment Pressure — Case 11



REV 21 5/08

SOUTHERN COMPANY *Energy to Serve Your World®*	JOSEPH M. FARLEY NUCLEAR PLANT UNIT 1 AND UNIT 2	PRESSURE VERSUS TIME STEAM LINE 0.4 ft^2 D.E. BREAK 30% POWER, P_O = -1.5 PSIG FIGURE 6.2-26A

Vapor and Sump Temperature — Case 11



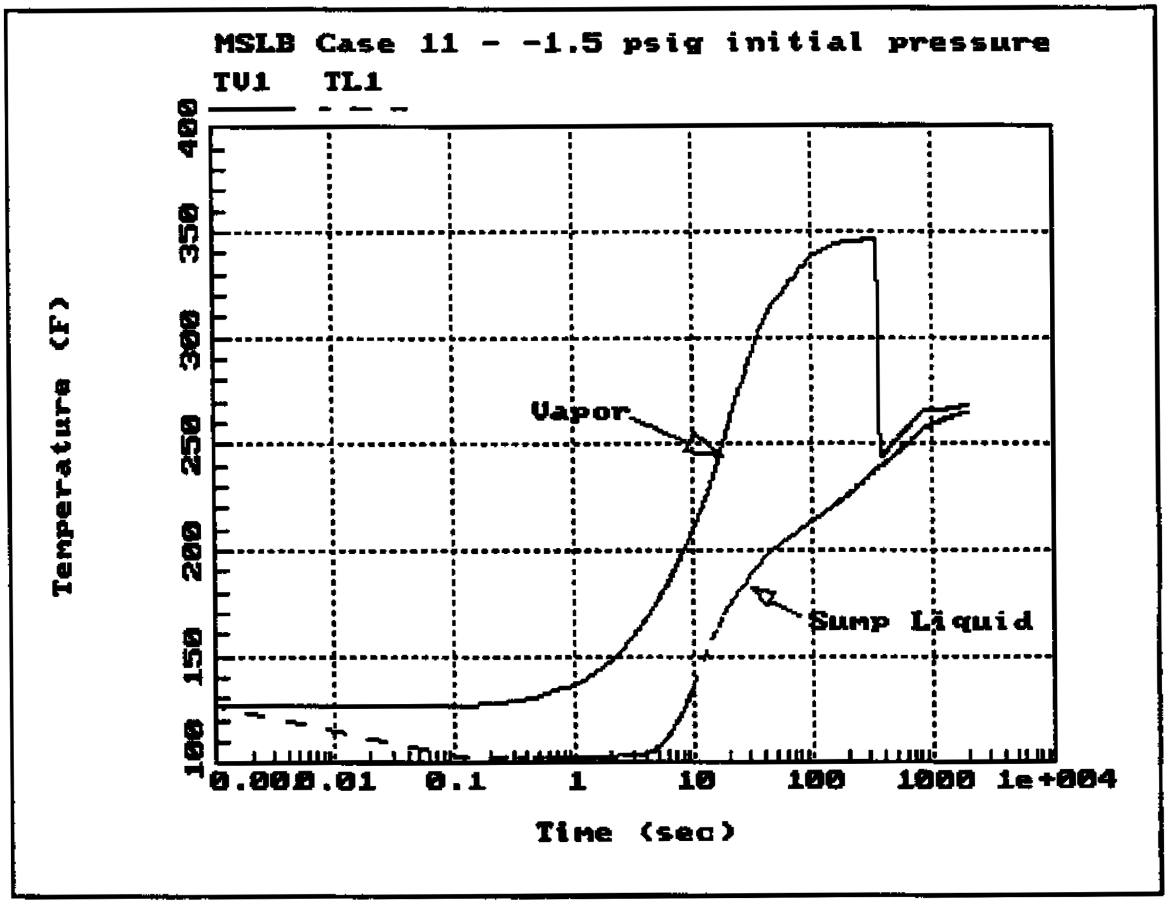
15

REV 21 5/08

TEMPERATURE VERSUS TIME
STEAM LINE 0.4 ft^2 D.E. BREAK
30% POWER, P_O = 0 PSIG

FIGURE 6.2-27

Vapor and Sump Temperature — Case 11



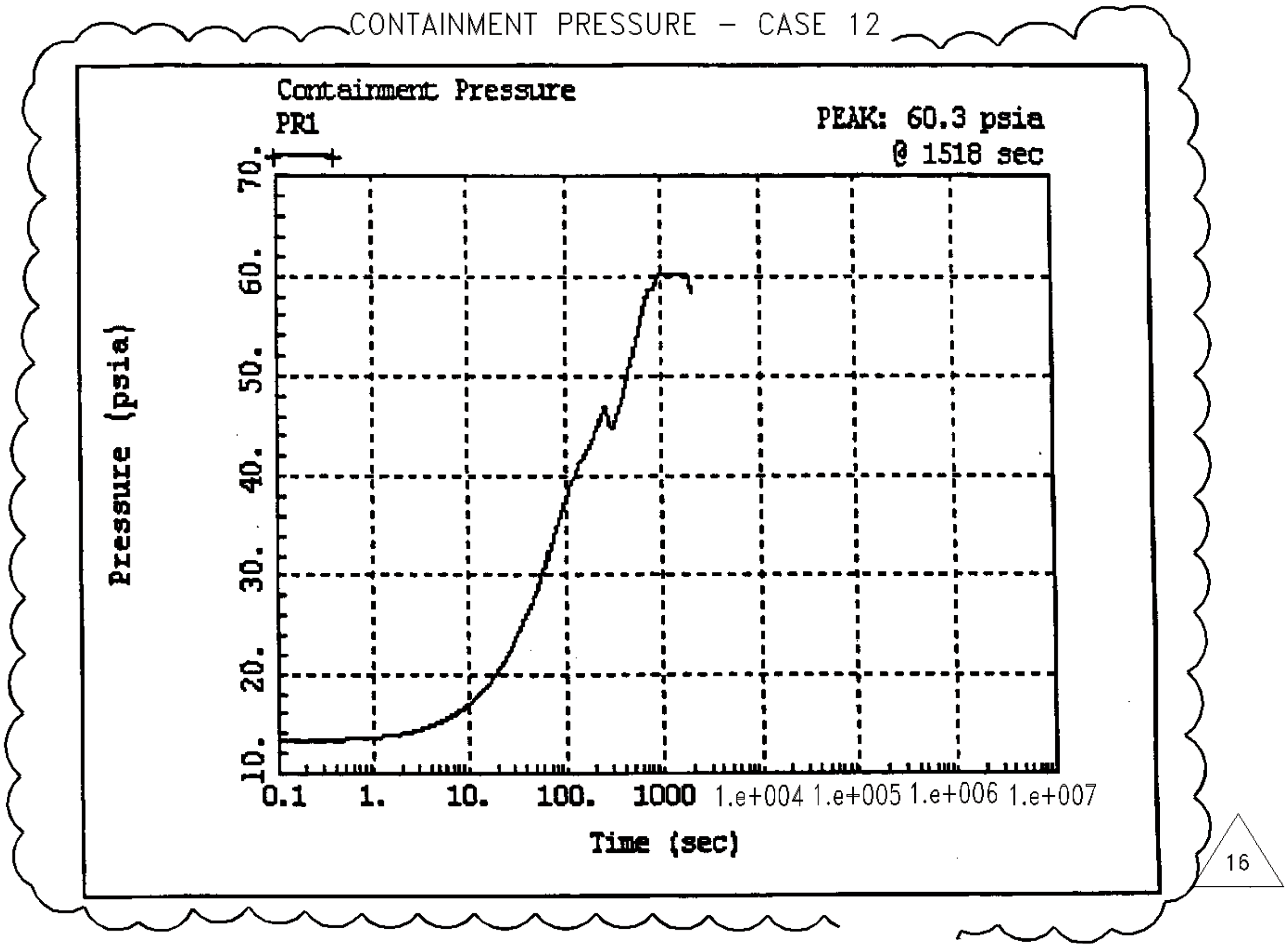
REV 21 5/08

SOUTHERN COMPANY
Energy to Serve Your World®

JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

TEMPERATURE VERSUS TIME
STEAM LINE 0.4 ft^2 D.E. BREAK
30% POWER, P_O = -1.5 PSIG

FIGURE 6.2-27A



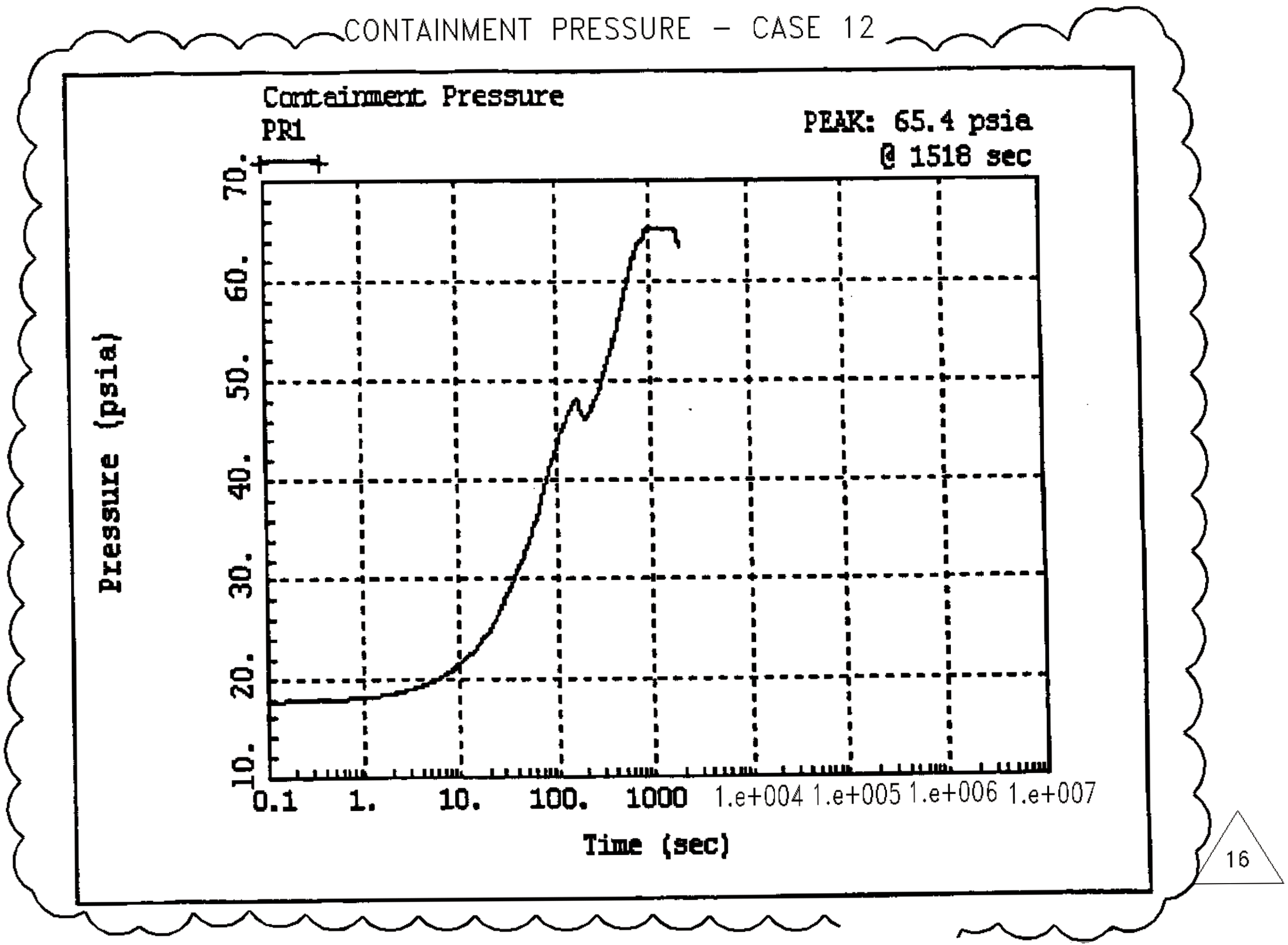
REV 21 5/08

SOUTHERN COMPANY
Energy to Serve Your World®

JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

RSG PRESSURE VERSUS TIME
STEAM LINE 0.60 ft^2 SPLIT
30% POWER, P_O = -1.5 PSIG

FIGURE 6.2-28



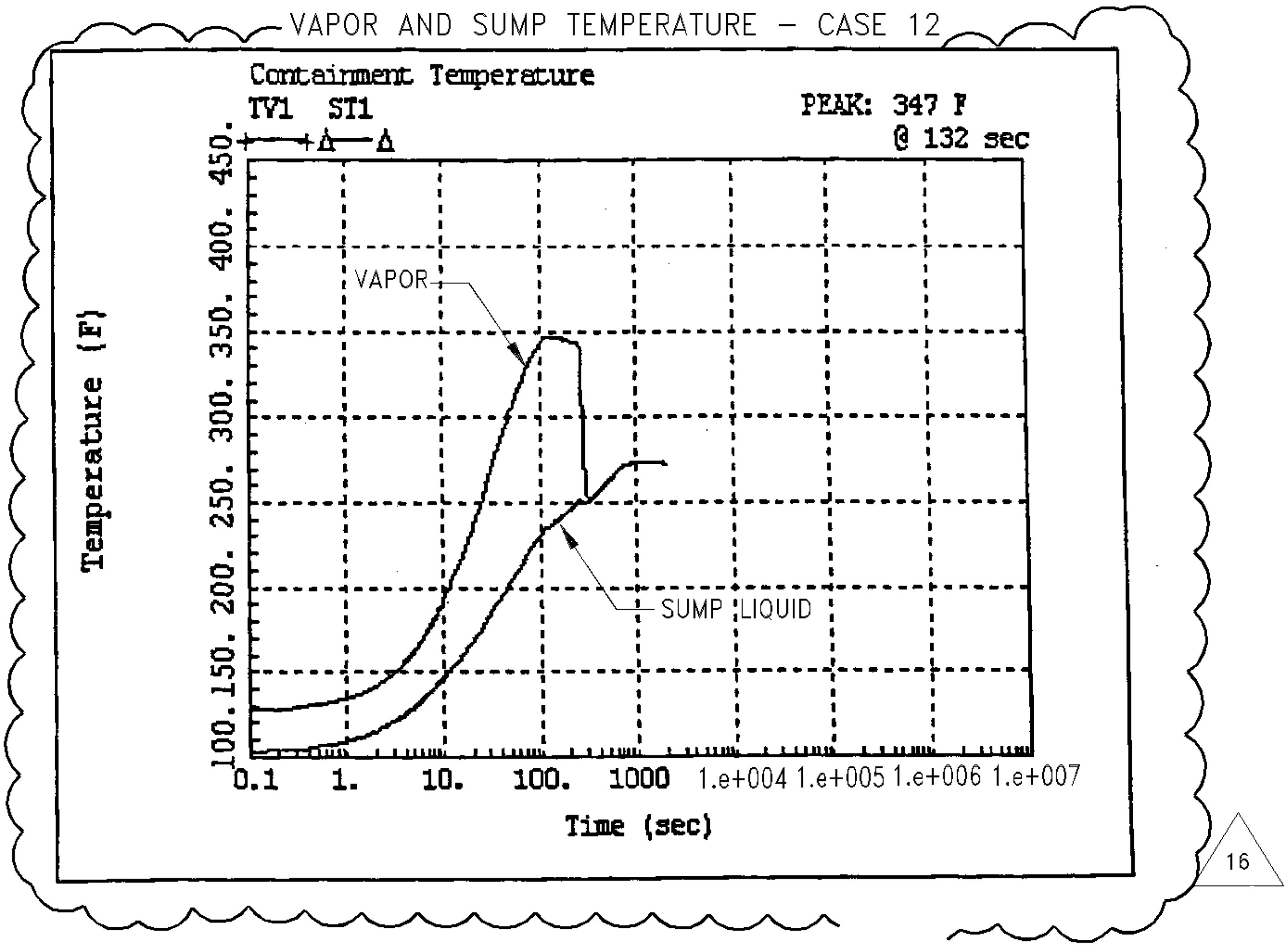
REV 21 5/08

SOUTHERN COMPANY
Energy to Serve Your World®

JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

RSG PRESSURE VERSUS TIME
STEAM LINE 0.60 ft^2 SPLIT
30% POWER, P_O = +3 PSIG

FIGURE 6.2-28A



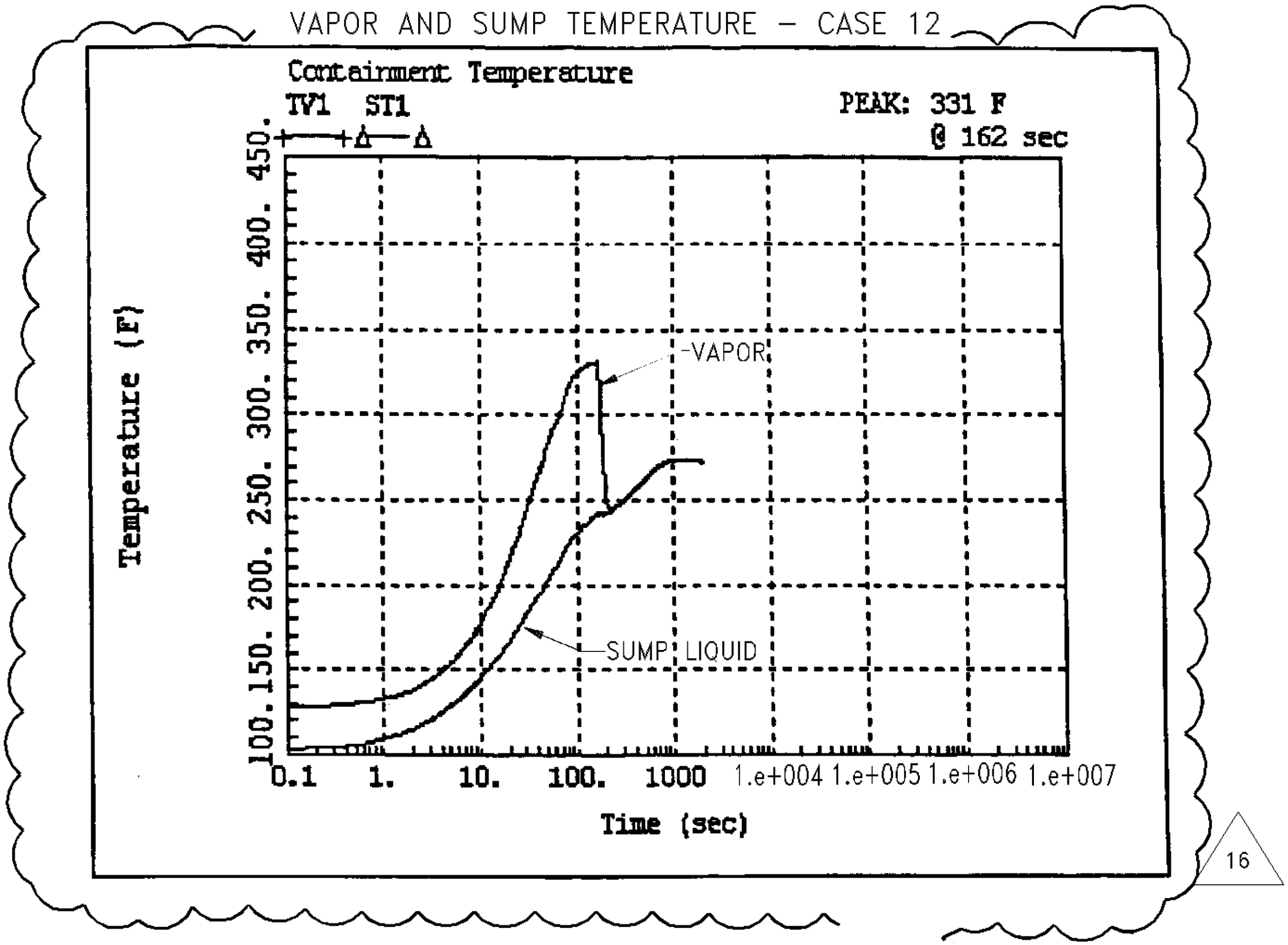
REV 21 5/08

SOUTHERN COMPANY
Energy to Serve Your World®

JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

RSG TEMPERATURE VERSUS TIME
STEAM LINE 0.60 ft^2 SPLIT
30% POWER, P_O = -1.5 PSIG

FIGURE 6.2-29



REV 21 5/08

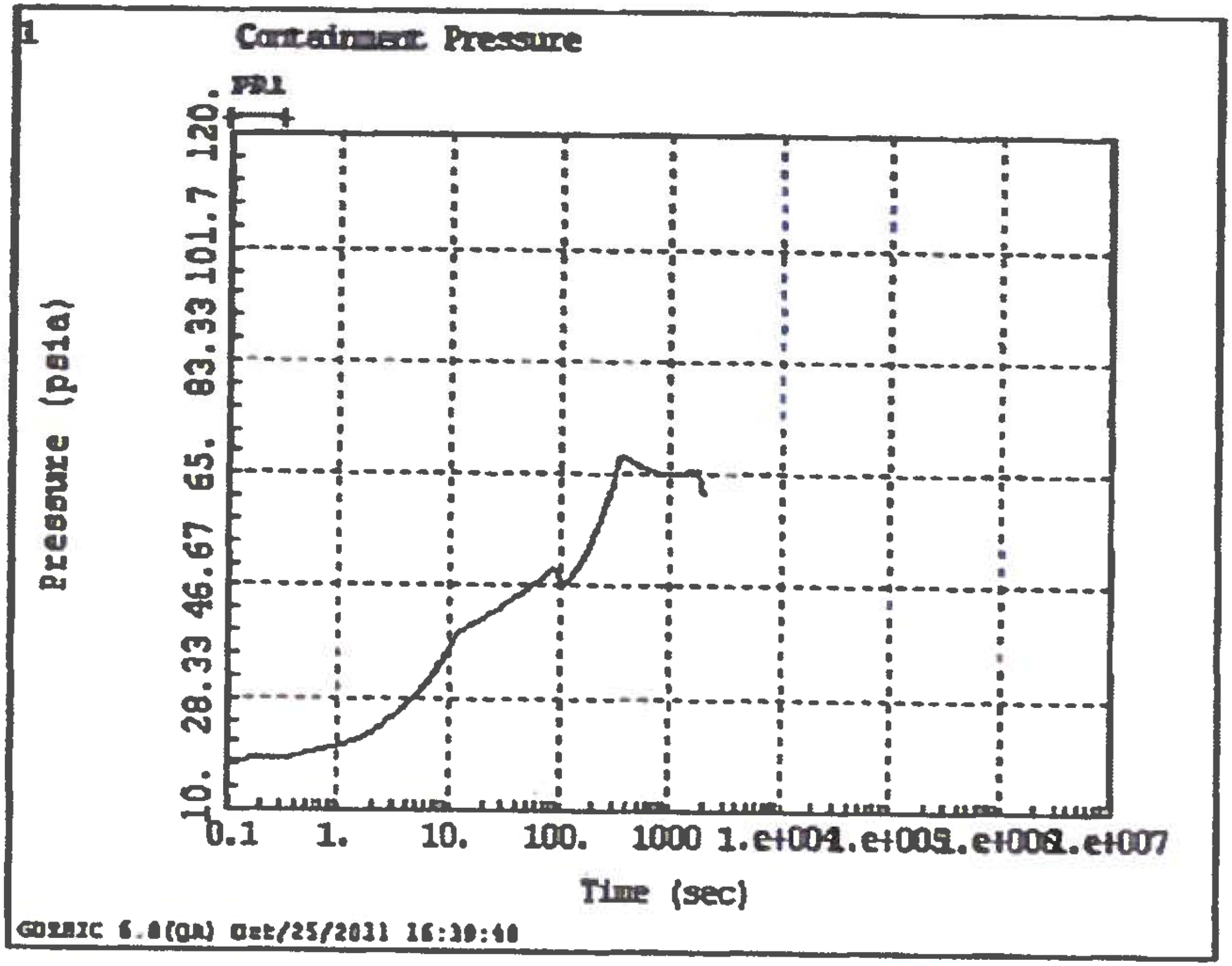
SOUTHERN COMPANY
Energy to Serve Your World®

JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

RSG TEMPERATURE VERSUS TIME
STEAM LINE 0.60 ft^2 SPLIT
30% POWER, P_O = +3 PSIG

FIGURE 6.2-29A

CONTAINMENT PRESSURE – CASE 13



REV 28 10/18

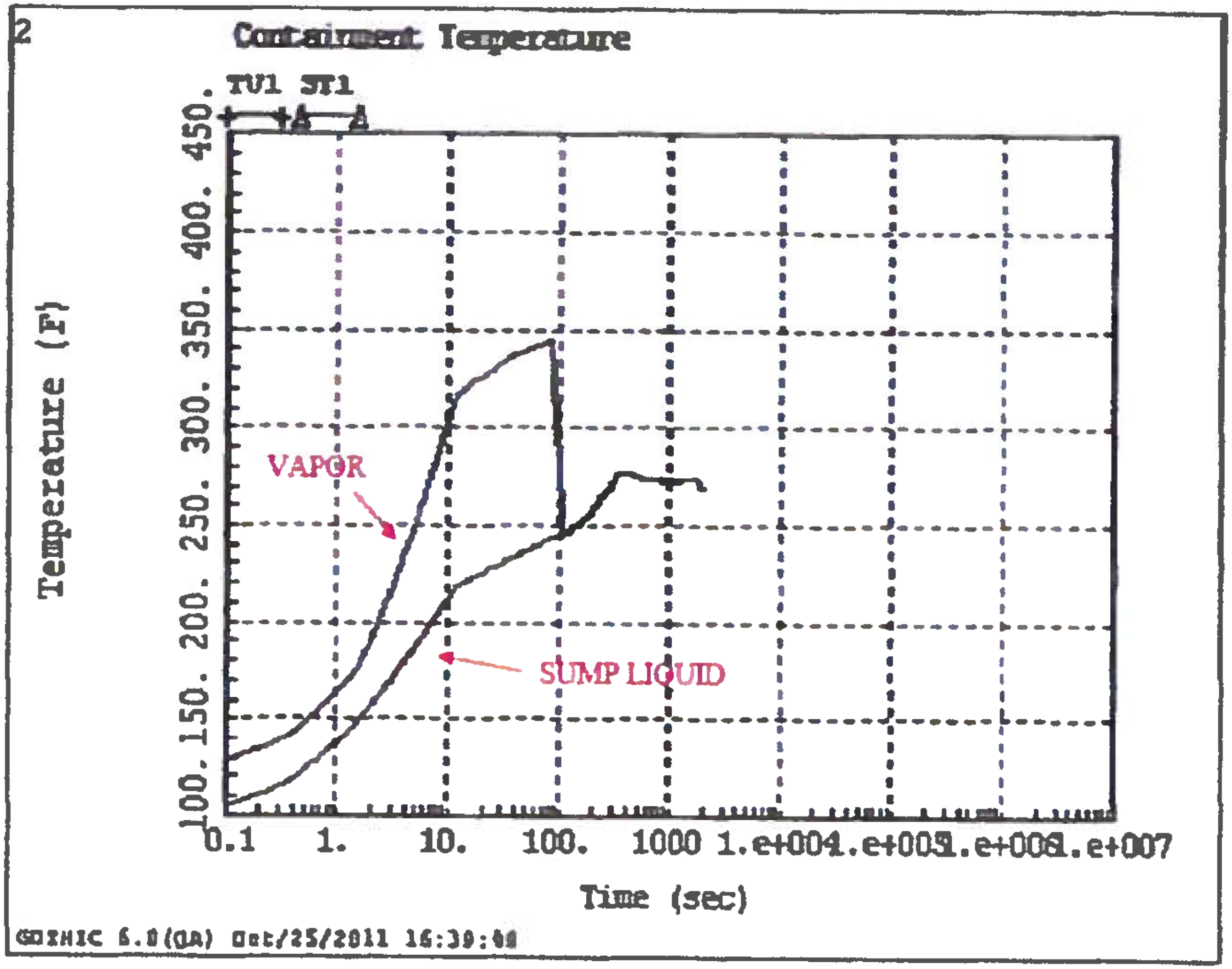
SOUTHERN COMPANY
Energy to Serve Your World®

JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

RSG PRESSURE VERSUS TIME
STEAM LINE FULL D.E. BREAK
HOT STANDBY P_O = +3 PSIG

FIGURE 6.2-30

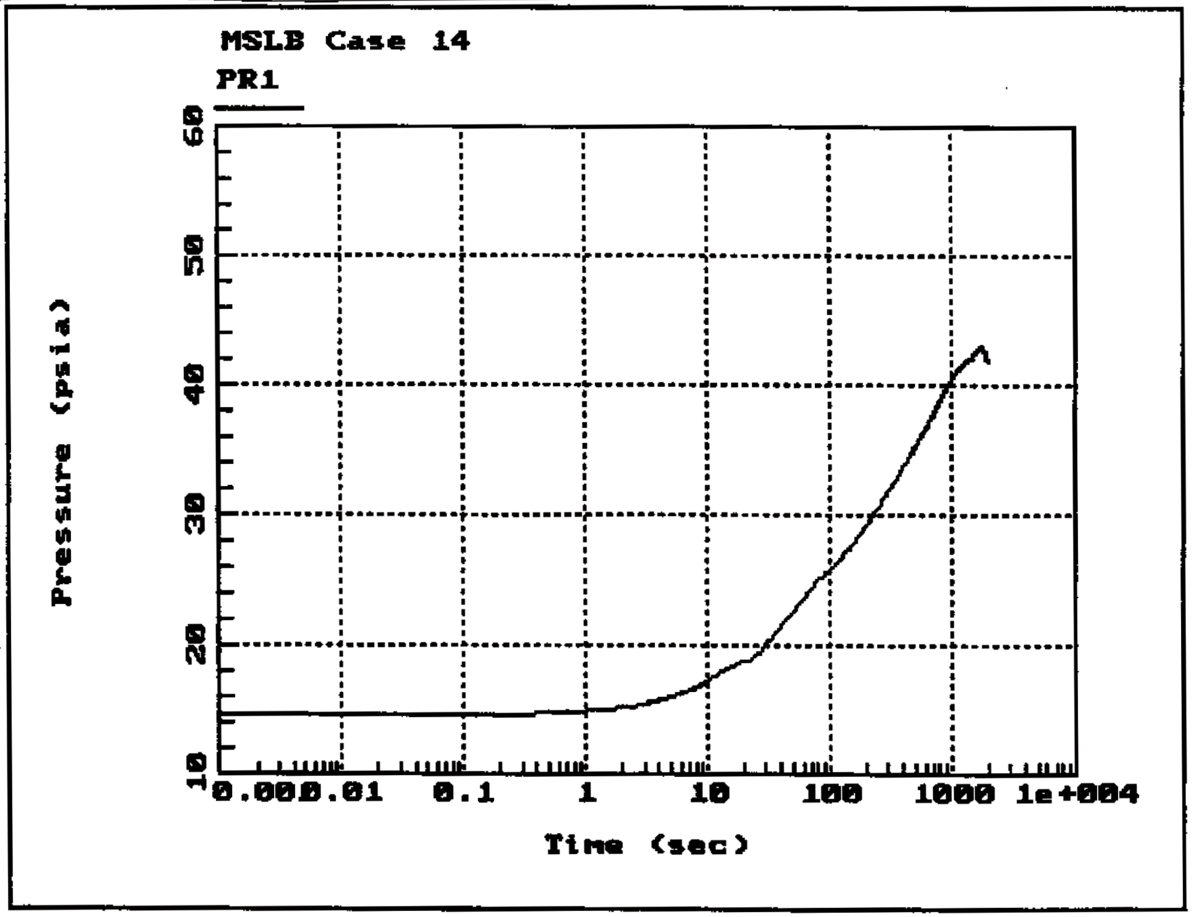
VAPOR AND SUMP TEMPERATURE – CASE 13



REV 28 10/18

SOUTHERN COMPANY *Energy to Serve Your World®*	JOSEPH M. FARLEY NUCLEAR PLANT UNIT 1 AND UNIT 2	RSG TEMPERATURE VERSUS TIME STEAM LINE FULL D.E. BREAK HOT STANDBY, P_O =-1.5 PSIG FIGURE 6.2-31

Containment Pressure — Case 14



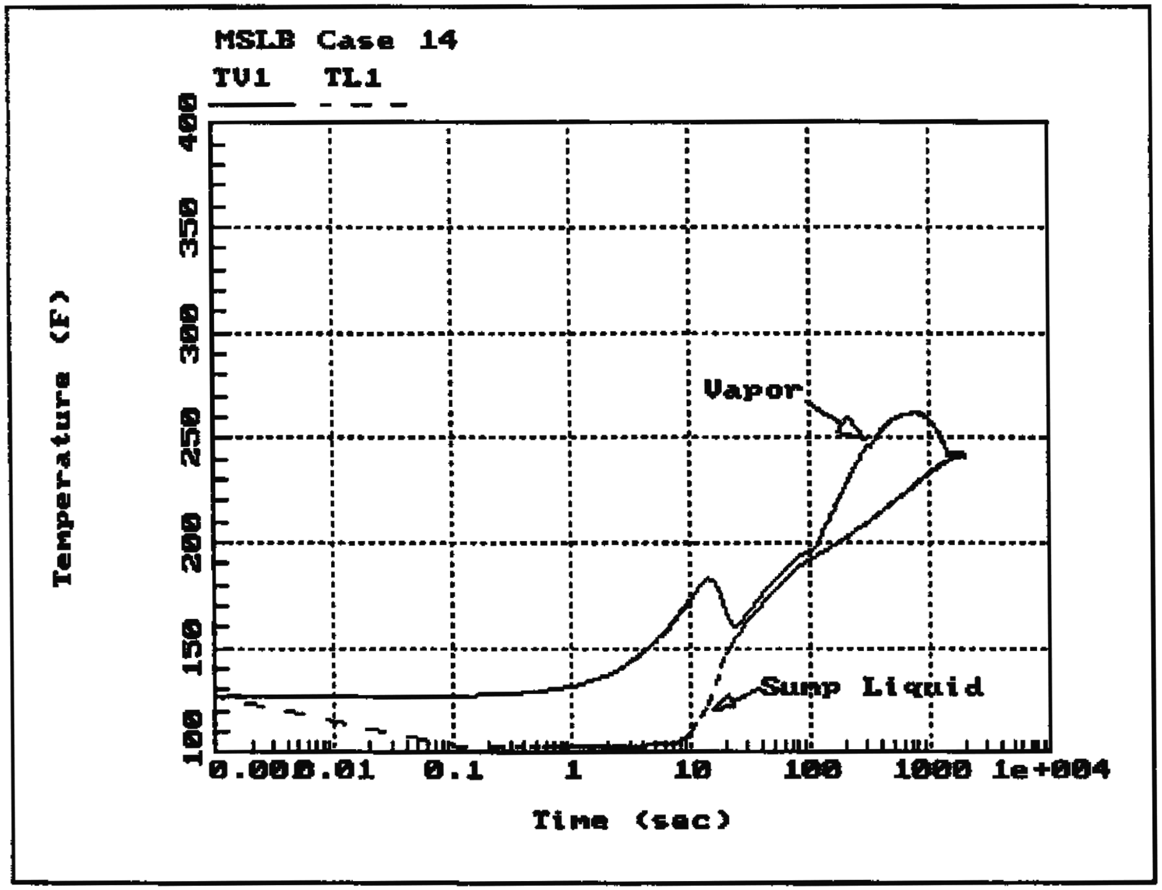
15

REV 21 5/08

PRESSURE VERSUS TIME
STEAM LINE 0.2 ft^2 D.E. BREAK
HOT STANDBY

FIGURE 6.2-32

Vapor and Sump Temperature — Case 14



15

REV 21 5/08

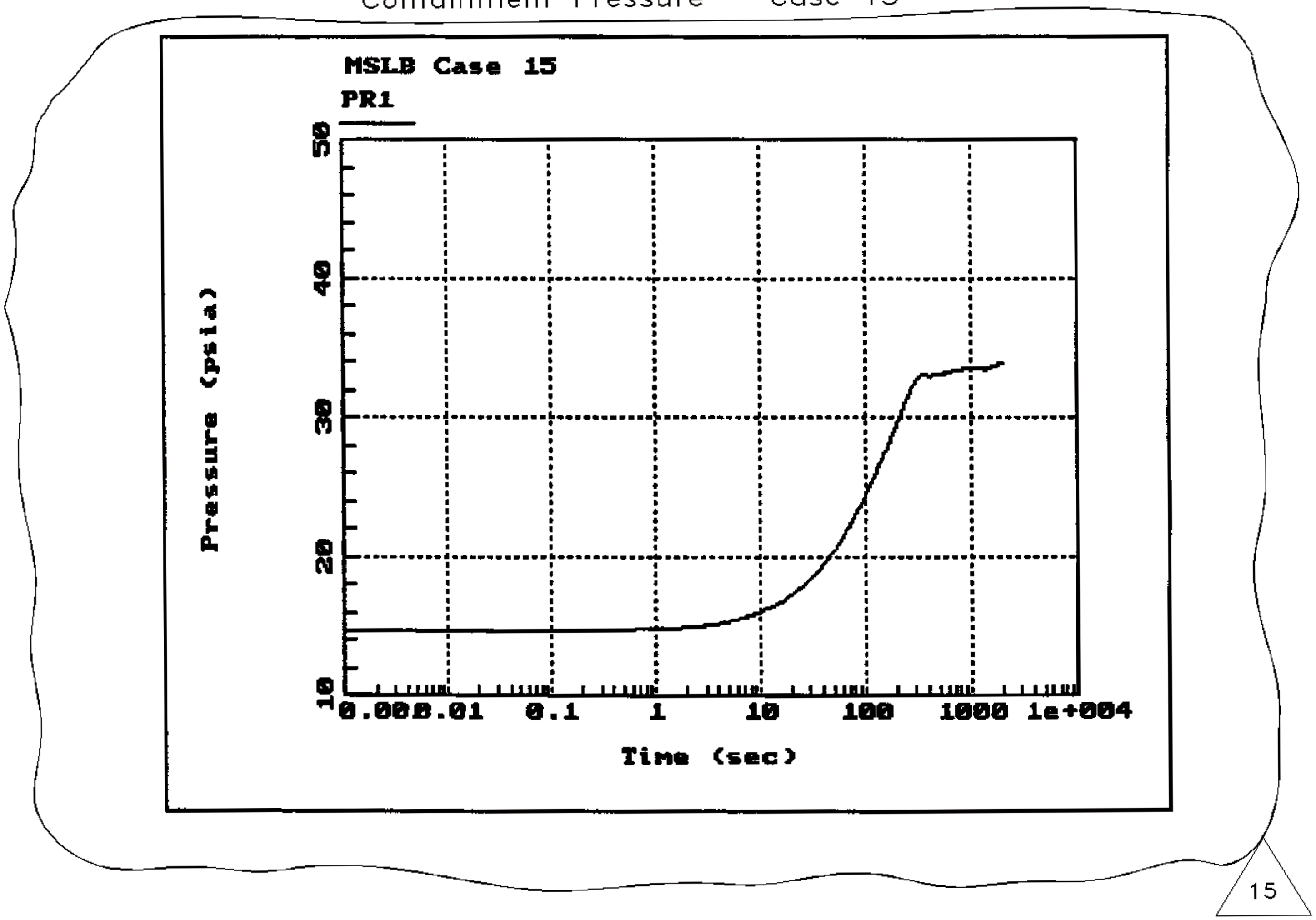
SOUTHERN COMPANY
Energy to Serve Your World®

JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

TEMPERATURE VERSUS TIME
STEAM LINE 0.2 ft^2 D.E. BREAK
HOT STANDBY

FIGURE 6.2-33

Containment Pressure — Case 15



REV 21 5/08

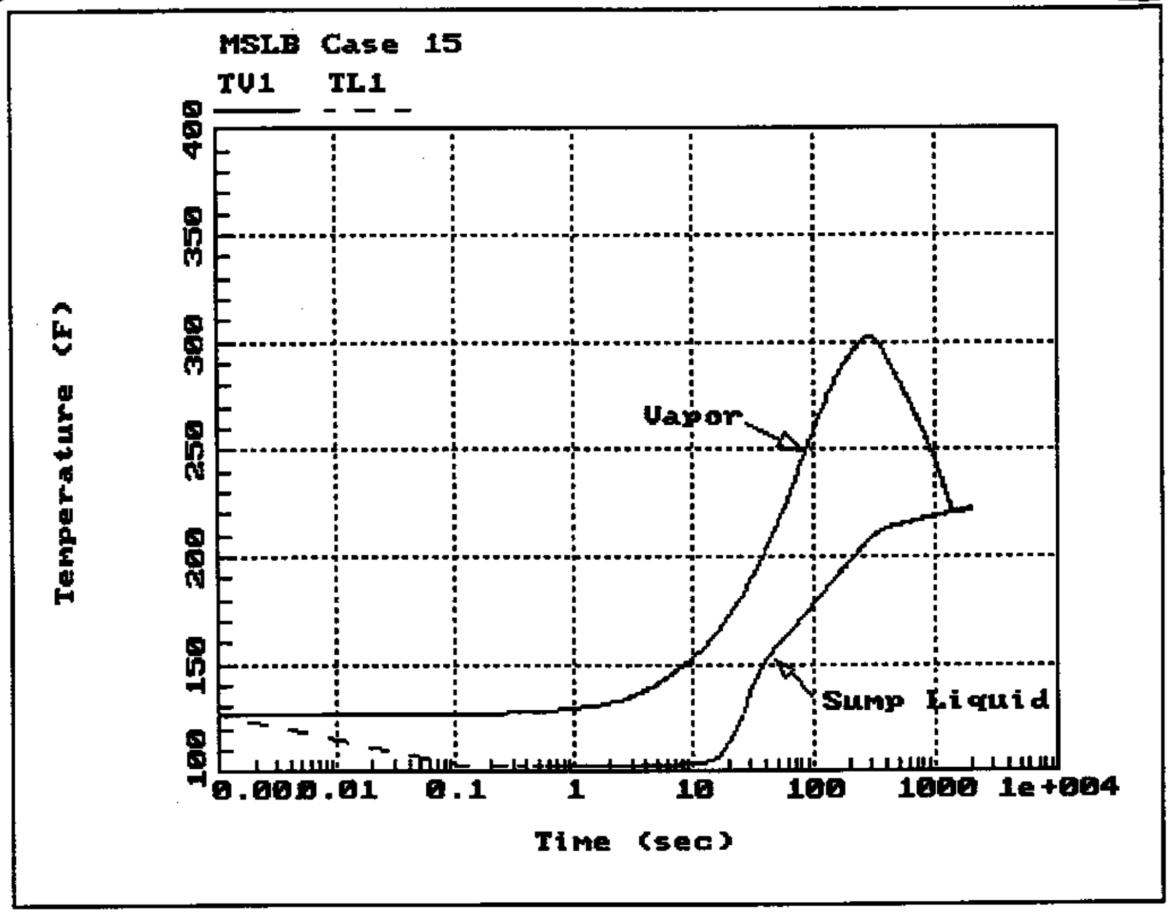
SOUTHERN COMPANY
Energy to Serve Your World®

JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

PRESSURE VERSUS TIME
STEAM LINE 0.1 ft^2 D.E. BREAK
HOT STANDBY

FIGURE 6.2-34

Vapor and Sump Temperature – Case 15

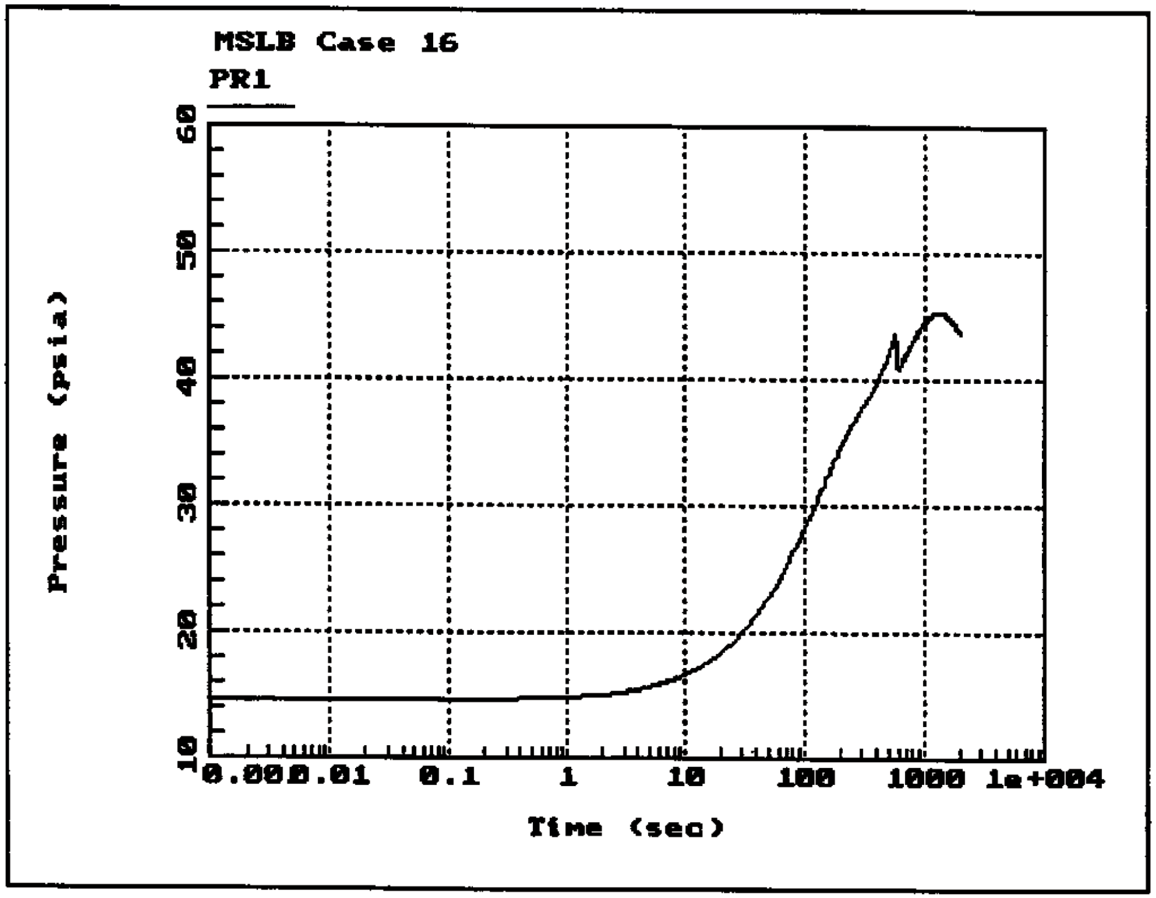


15

REV 21 5/08

SOUTHERN COMPANY *Energy to Serve Your World®*	**JOSEPH M. FARLEY NUCLEAR PLANT UNIT 1 AND UNIT 2**	TEMPERATURE VERSUS TIME STEAM LINE 0.1 ft^2 D.E. BREAK HOT STANDBY
		FIGURE 6.2-35

Containment Pressure — Case 16

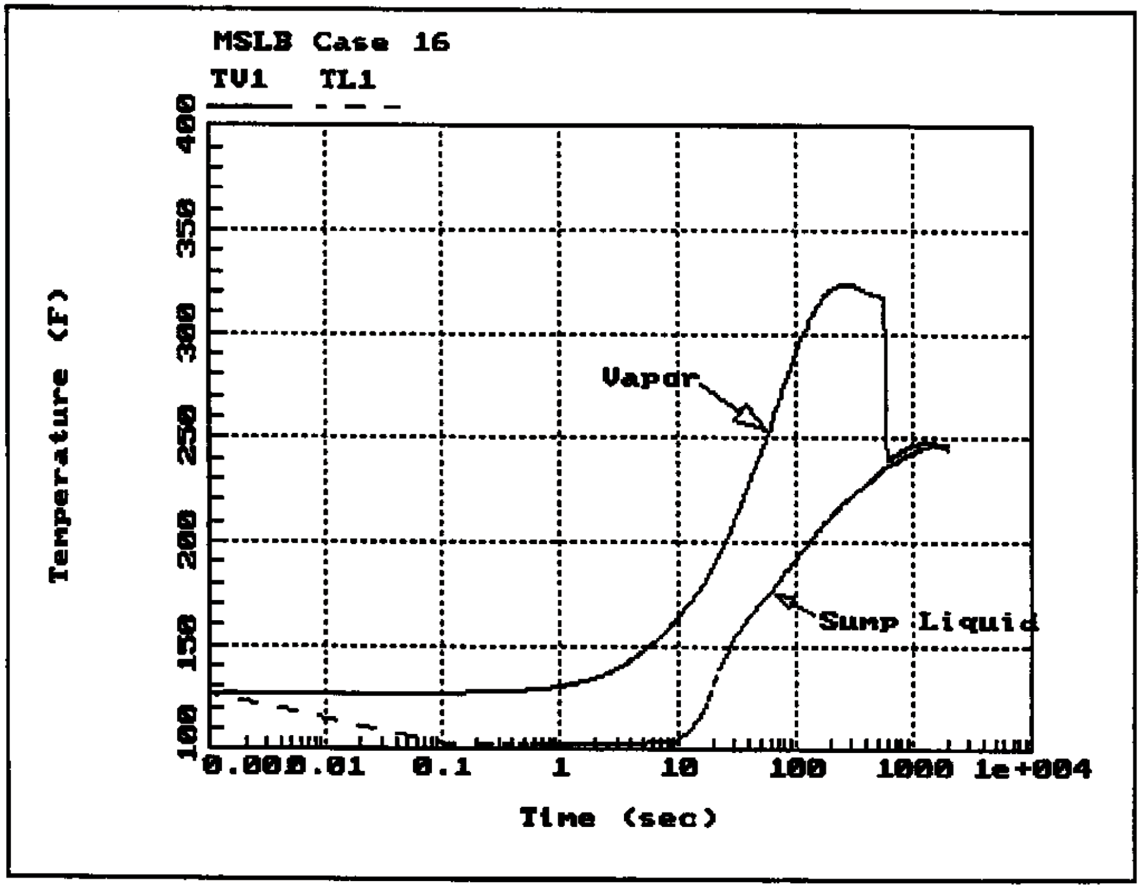


15

REV 21 5/08

SOUTHERN COMPANY *Energy to Serve Your World®*	JOSEPH M. FARLEY NUCLEAR PLANT UNIT 1 AND UNIT 2	PRESSURE VERSUS TIME STEAM LINE 0.30 ft^2 SPLIT HOT STANDBY
		FIGURE 6.2-36

Vapor and Sump Temperature — Case 16



15

REV 21 5/08

SOUTHERN COMPANY
Energy to Serve Your World®

JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

TEMPERATURE VERSUS TIME
STEAM LINE 0.30 ft^2 SPLIT
HOT STANDBY

FIGURE 6.2-37

THIS FIGURE HAS BEEN DELETED.

15

REV 21 5/08

SOUTHERN COMPANY

Energy to Serve Your World®

JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

TS, EQUIPMENT SURFACE TEMPERATURE
WITH UCHIDA CONDENSING HEAT
TRANSFER AND CONVECTIVE HEAT
TRANSFER COEFFICIENT OF 2 BTU/HR-ft^q

FIGURE 6.2-38

THIS FIGURE HAS BEEN DELETED.

12

REV 21 5/08

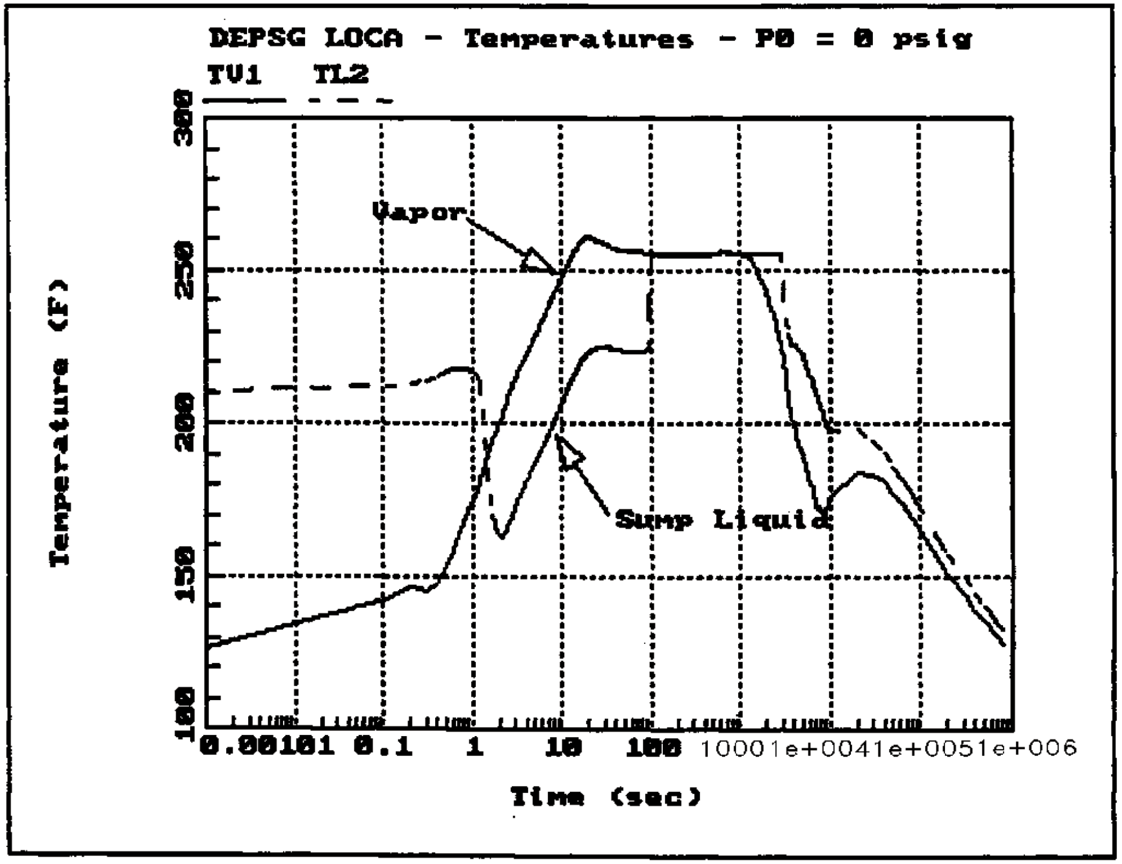
SOUTHERN COMPANY
Energy to Serve Your World®

JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

DEPSGB MINIMUM ESF 1 AC P/T
ANALYSIS LONG-TERM CONTAINMENT
PRESSURE VS. TIME

FIGURE 6.2-39

Vapor and Sump Temperature



An evaluation for changes to the long-term response has been performed as described in section 6.2.1.3.13.

16

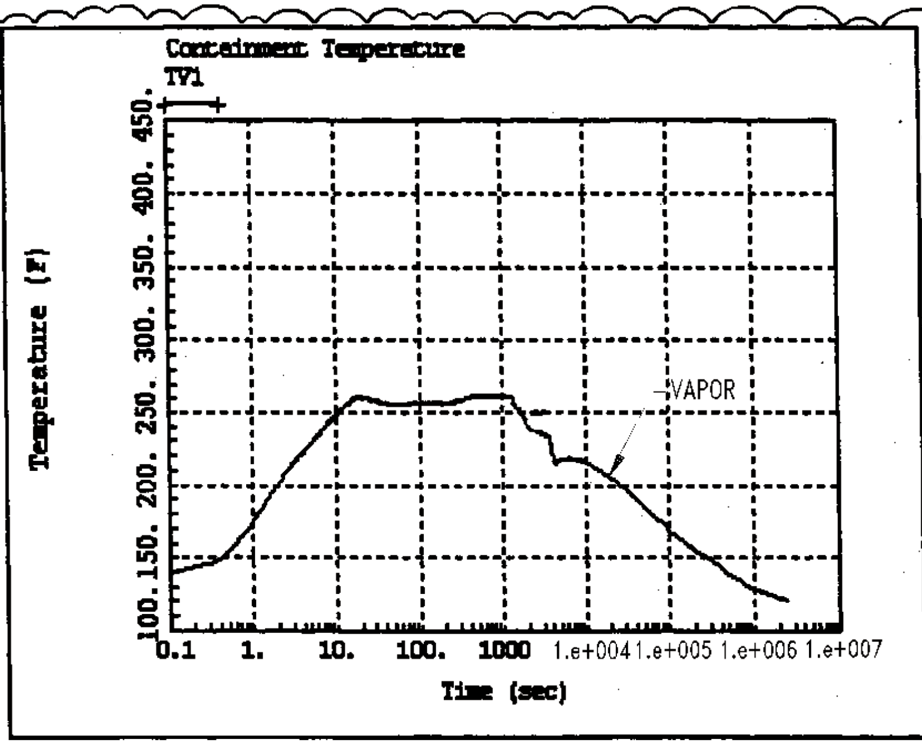
REV 21 5/08

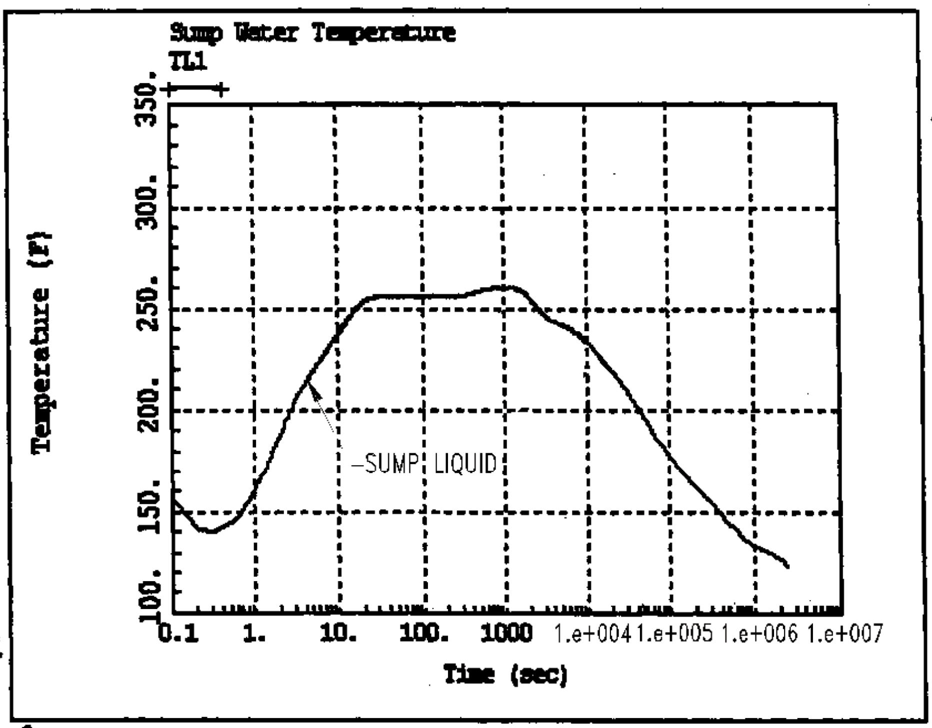
SOUTHERN COMPANY
Energy to Serve Your World®

JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

DEPSGB MINIMUM ESF DBA
TEMPERATURE VS. TIME
P_O = 0 PSIG

FIGURE 6.2-40





17

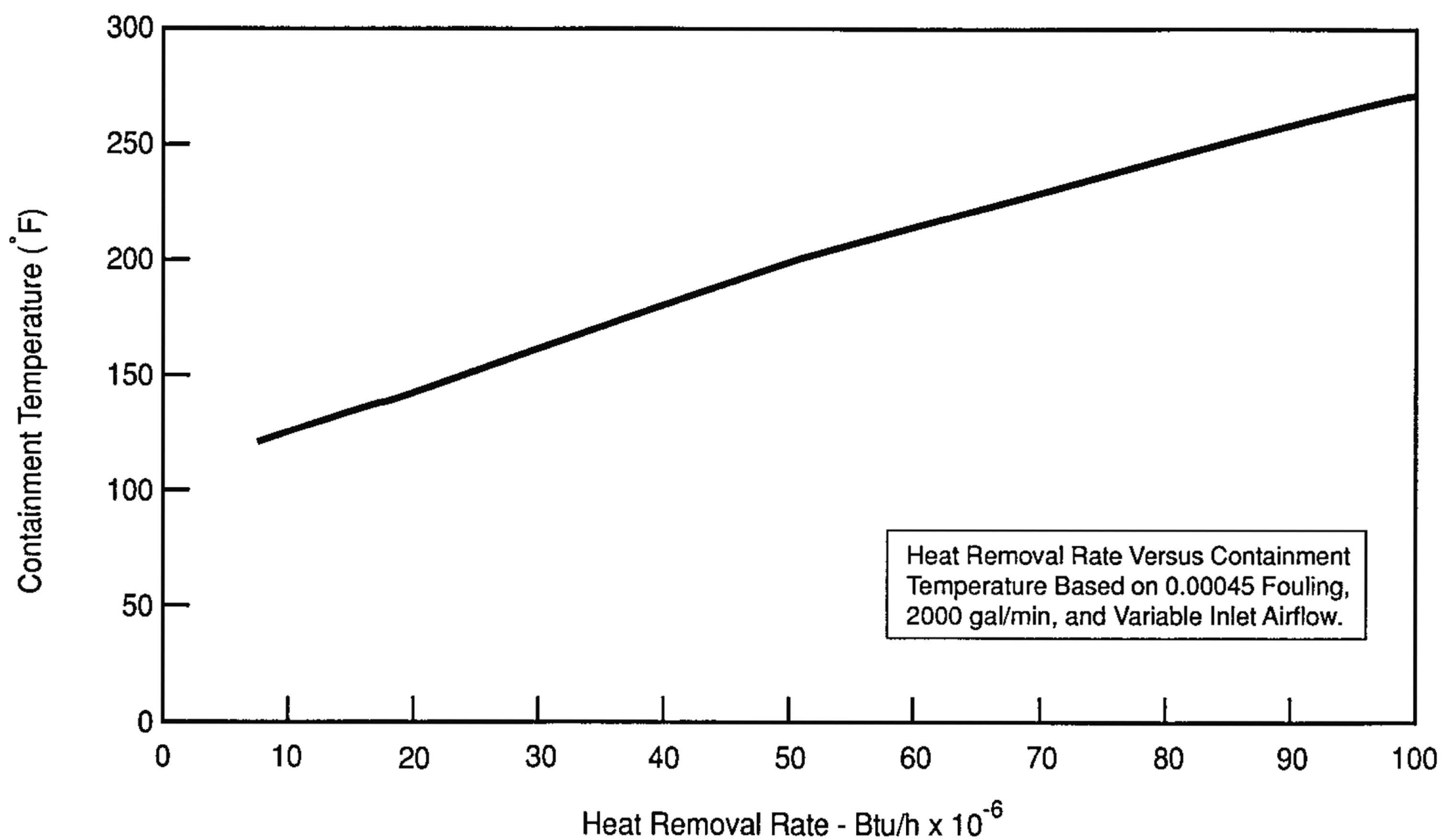
REV 21 5/08

SOUTHERN COMPANY
Energy to Serve Your World®

JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

RSG DEPSG MIN ESF DBA
TEMPERATURE VS. TIME
P_O = 3 PSIG

FIGURE 6.2-41



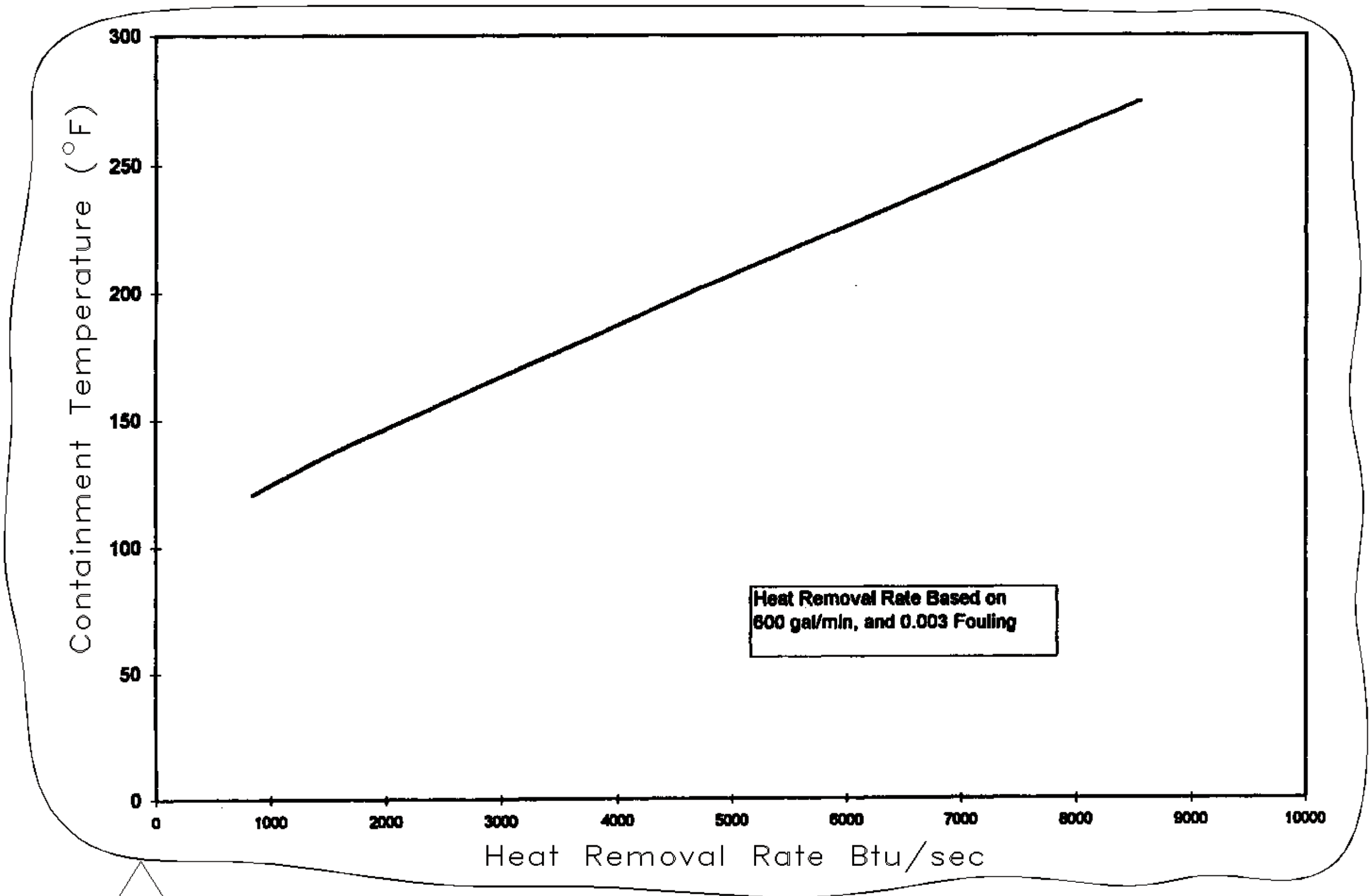
REV 21 5/08

SOUTHERN COMPANY
Energy to Serve Your World®

JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

CONTAINMENT AIR COOLER DUTY VS. TEMPERATURE

FIGURE 6.2-42 (SHEET 1 OF 2)



15

REV 21 5/08

SOUTHERN COMPANY
Energy to Serve Your World®

JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

CONTAINMENT AIR COOLER DUTY VS. TEMPERATURE

FIGURE 6.2-42 (SHEET 2 OF 2)

12

THIS FIGURE HAS BEEN DELETED.

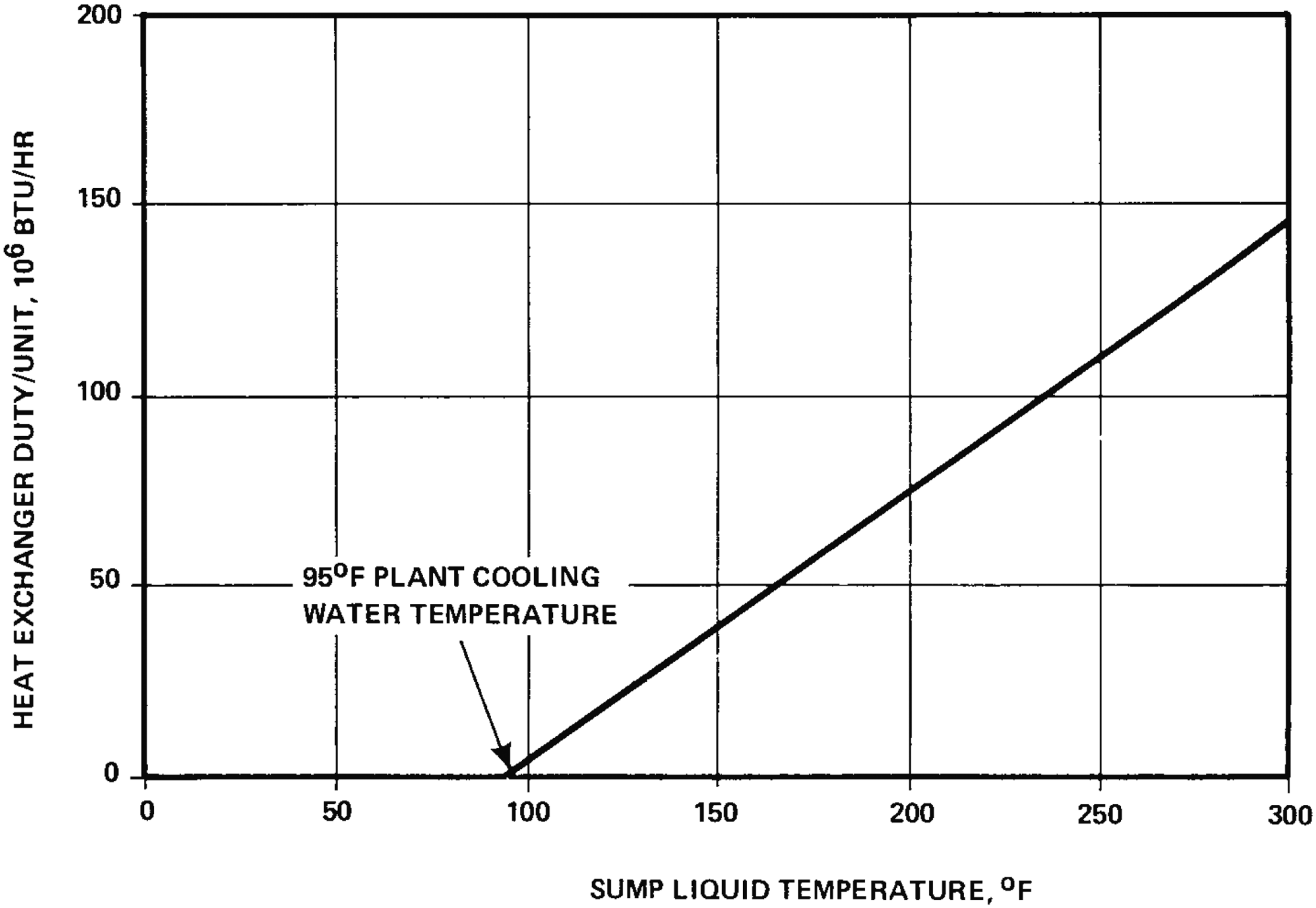
REV 21 5/08

SOUTHERN COMPANY
Energy to Serve Your World®

JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

THERMAL HEAT REMOVAL EFFICIENCY
OF CONTAINMENT ATMOSPHERE SPRAY

FIGURE 6.2-43



REV 21 5/08

SOUTHERN COMPANY
Energy to Serve Your World®

JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

RESIDUAL HEAT EXCHANGER DESIGN
DUTY ACCIDENT MODE

FIGURE 6.2-44

THIS FIGURE HAS BEEN DELETED.

15

REV 21 5/08

SOUTHERN COMPANY
Energy to Serve Your World®

JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

MASS &N ENERGY RATE VS TIME FOR DBA

FIGURE 6.2-45

THIS FIGURE HAS BEEN DELETED.

15

REV 21 5/08

SOUTHERN COMPANY
Energy to Serve Your World®

JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

LOCA BLOWDOWN MASS AND ENERGY
RELEASE RATES VS. TIME

FIGURE 6.2-46

THIS FIGURE HAS BEEN DELETED.

15

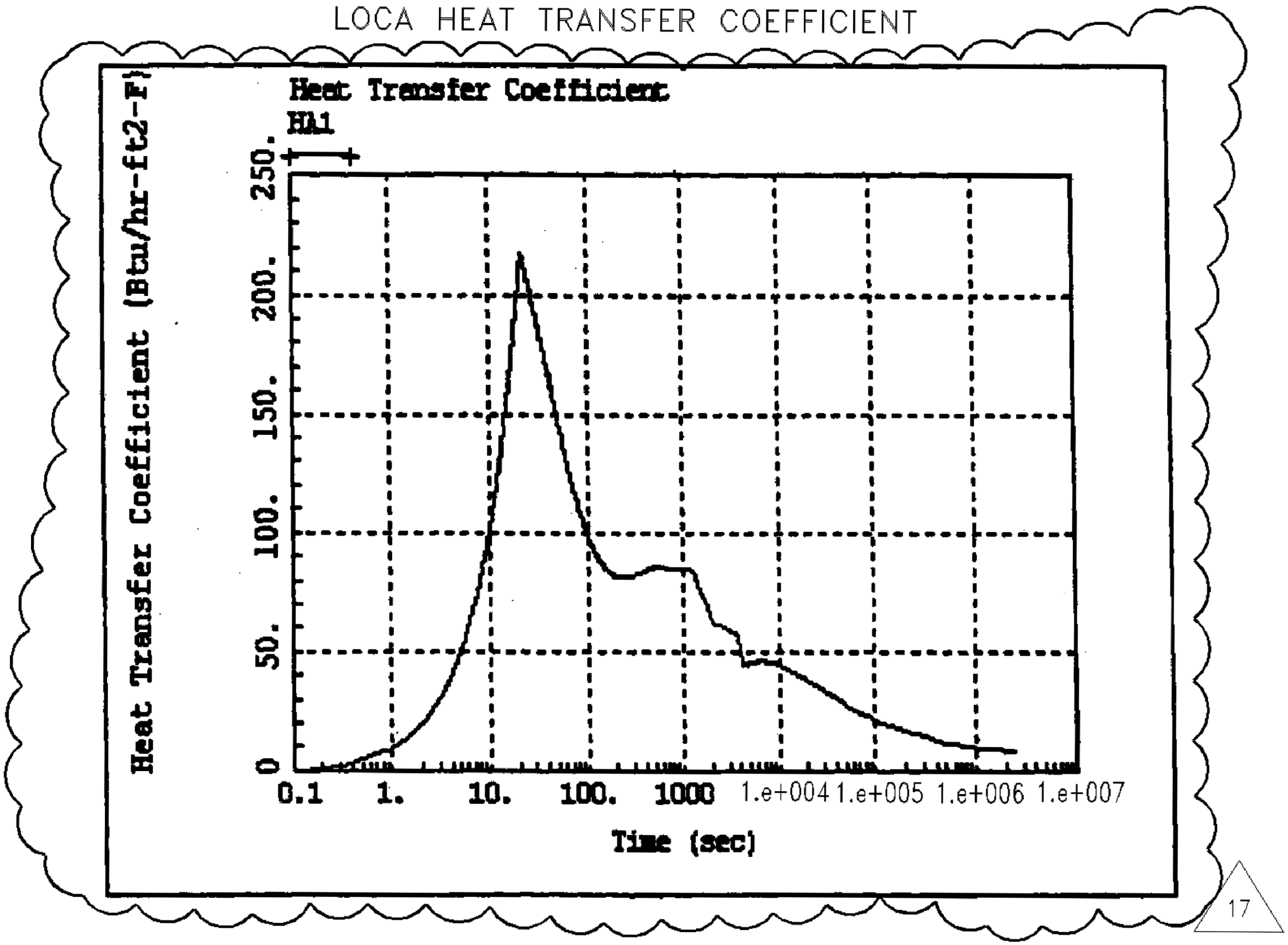
REV 21 5/08

SOUTHERN COMPANY
Energy to Serve Your World®

JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

LOCA POST-BLOWDOWN MASS AND ENERGY
RELEASE RATES VS. TIME

FIGURE 6.2-47



REV 21 5/08

SOUTHERN COMPANY
Energy to Serve Your World®

JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

DEPSG MIN ESF 1 AD P/T ANALYSIS
LONG-TERM CONDENSING HEAT
TRANSFER COEFFICIENT (RSG)

FIGURE 6.2-48

12

THIS FIGURE HAS BEEN DELETED.

REV 21 5/08

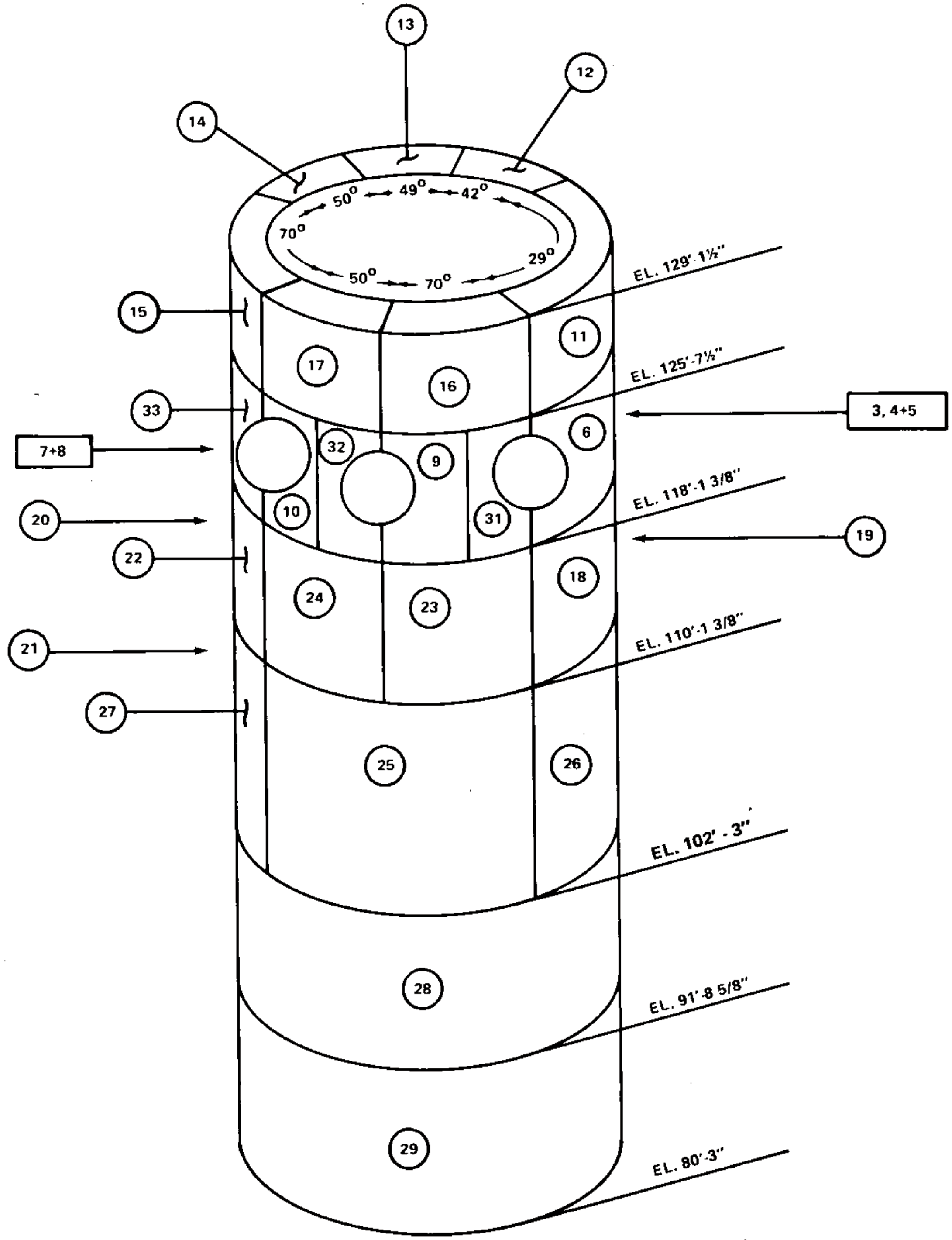
SOUTHERN COMPANY

Energy to Serve Your World®

JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

SHORT TERM CONDENSING HEAT
TRANSFER COEFFICIENT FOR DBA

FIGURE 6.2-49



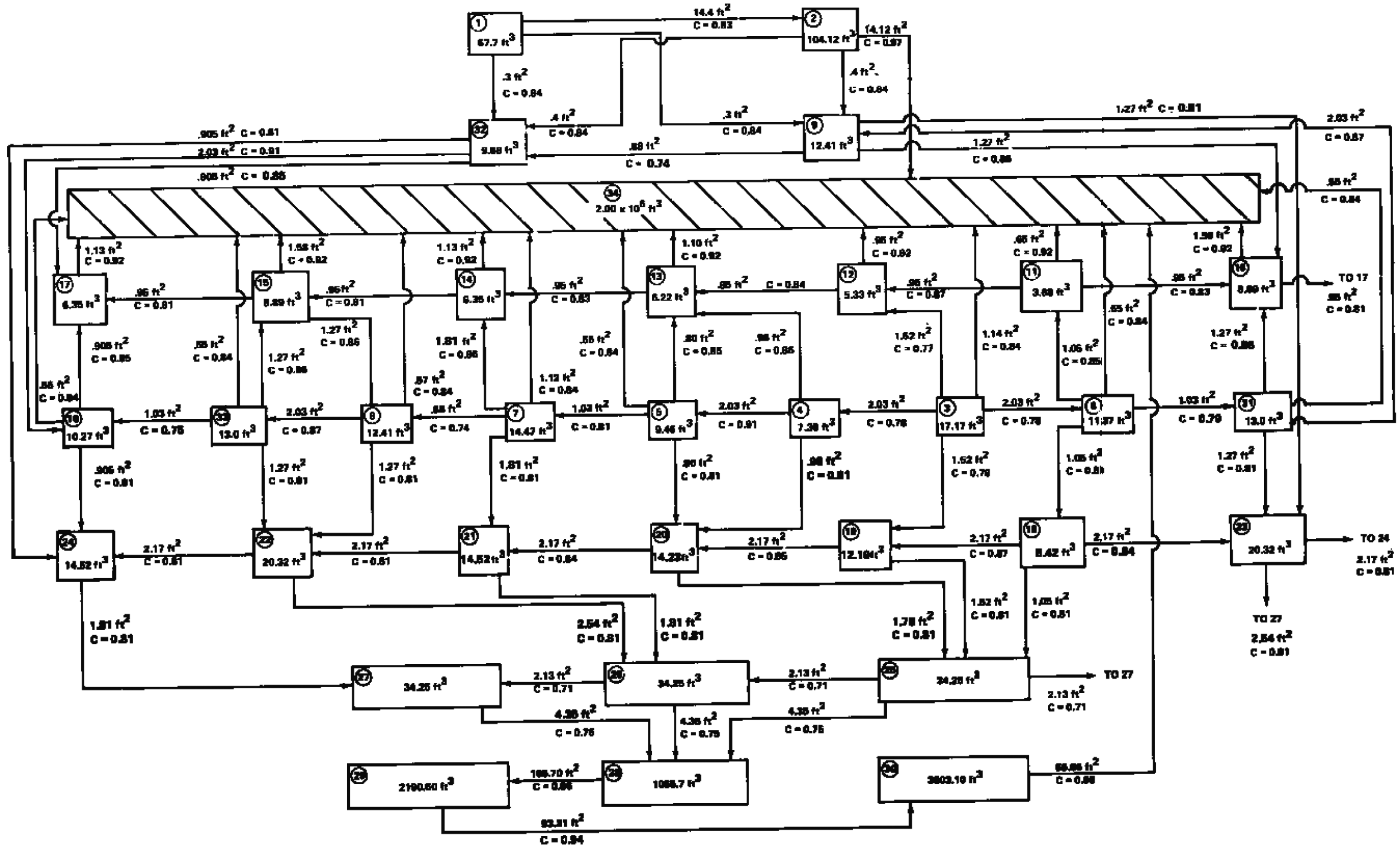
REV 21 5/08

SOUTHERN COMPANY
Energy to Serve Your World®

JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

REACTOR CAVITY MODEL

FIGURE 6.2-50



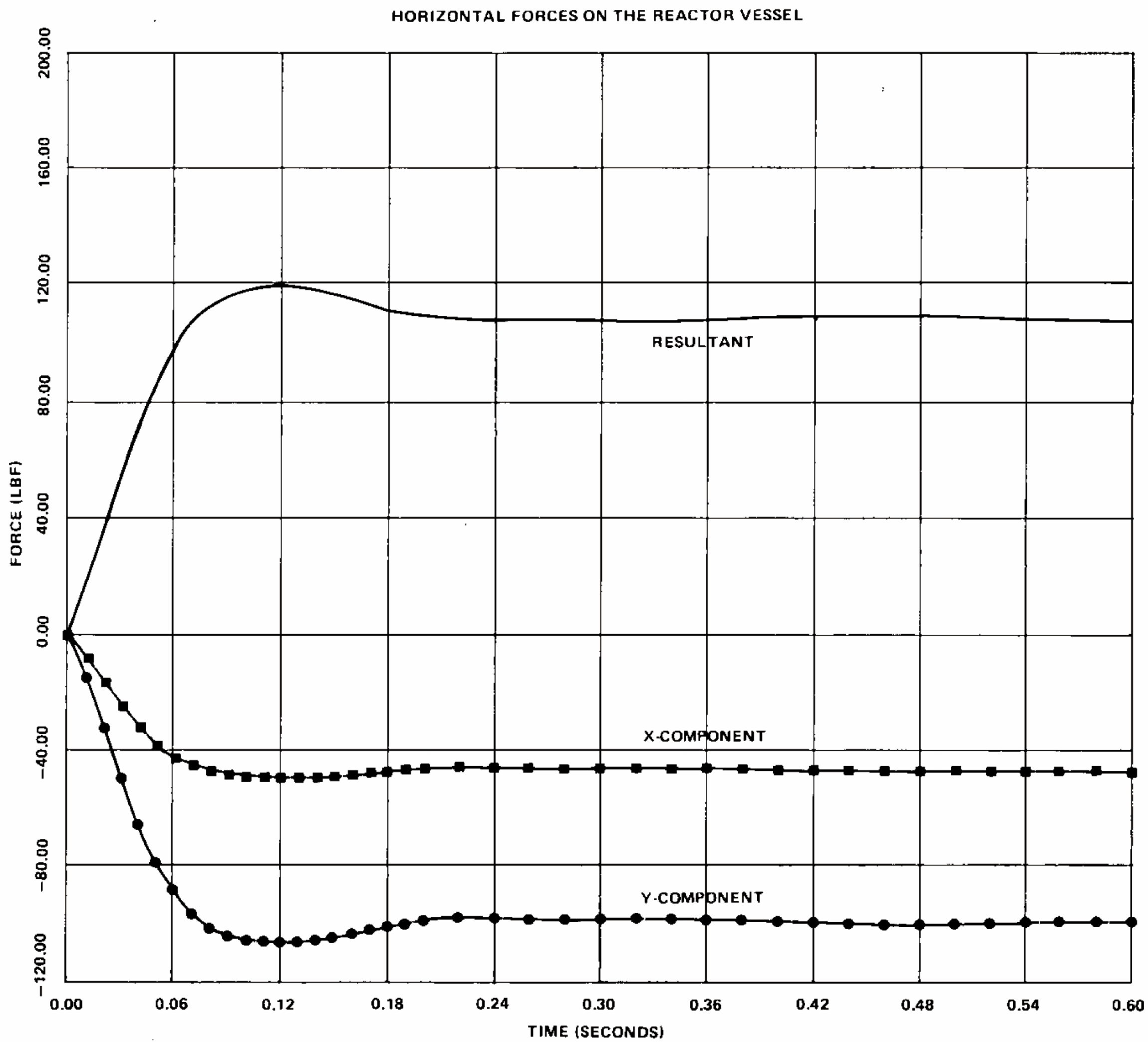
REV 21 5/08

SOUTHERN COMPANY
Energy to Serve Your World®

JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

REACTOR CAVITY BLOCK DIAGRAM

FIGURE 6.2-51



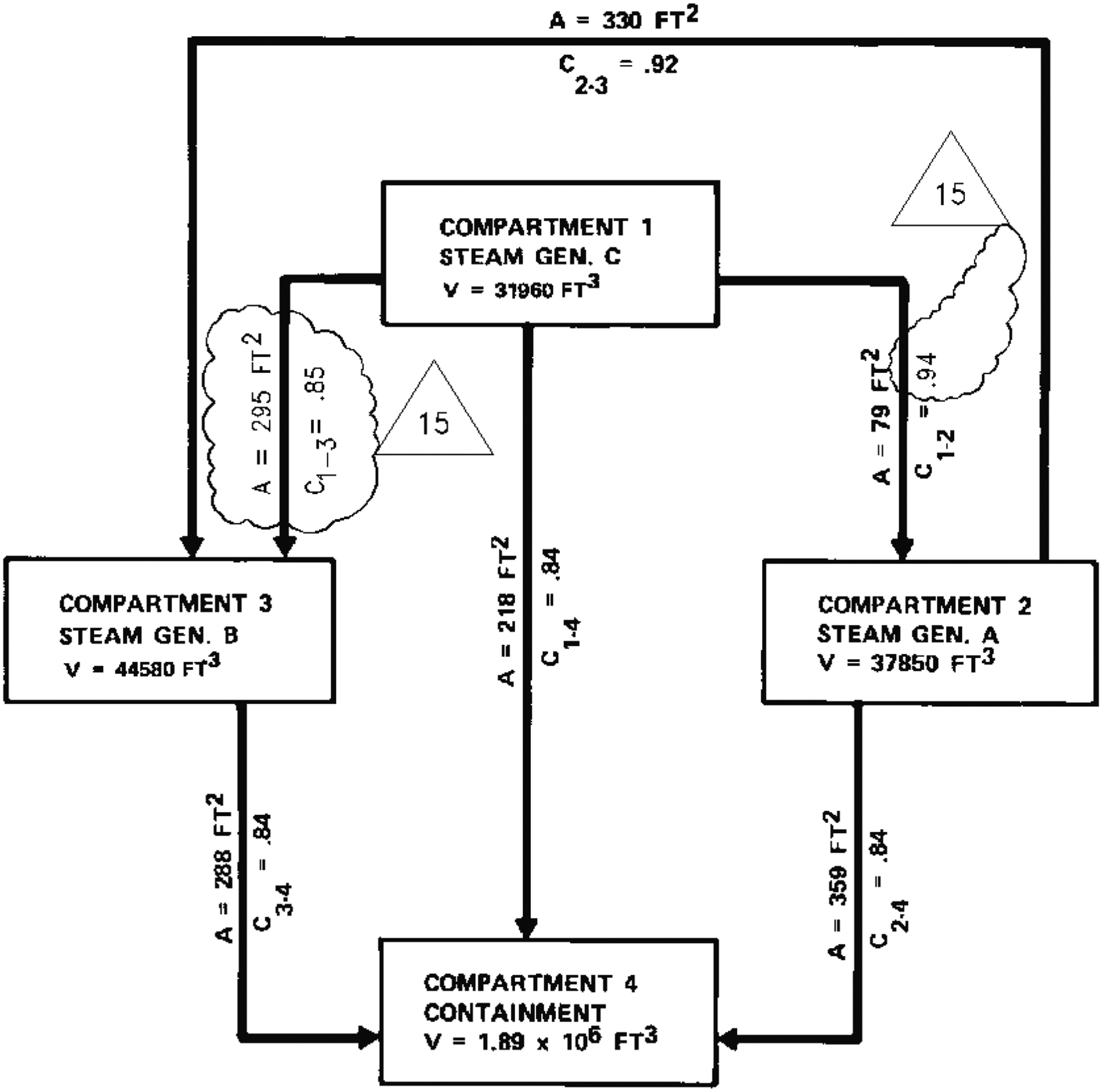
REV 21 5/08

SOUTHERN COMPANY
Energy to Serve Your World®

JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

TOTAL HORIZONTAL FORCE VERSUS TIME

FIGURE 6.2-52



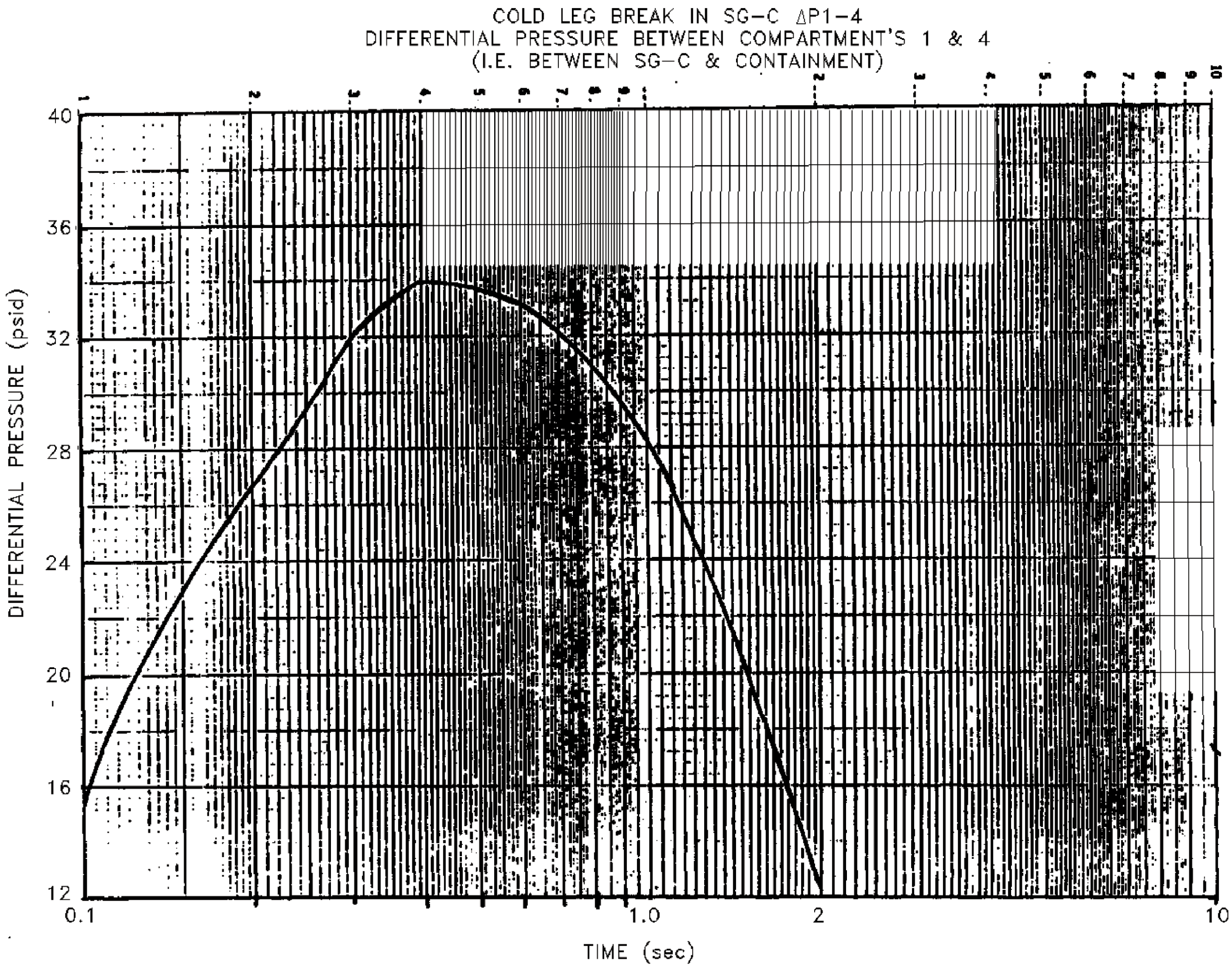
REV 21 5/08

SOUTHERN COMPANY
Energy to Serve Your World®

JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

STEAM GENERATOR BLOCK DIAGRAM

FIGURE 6.2-53



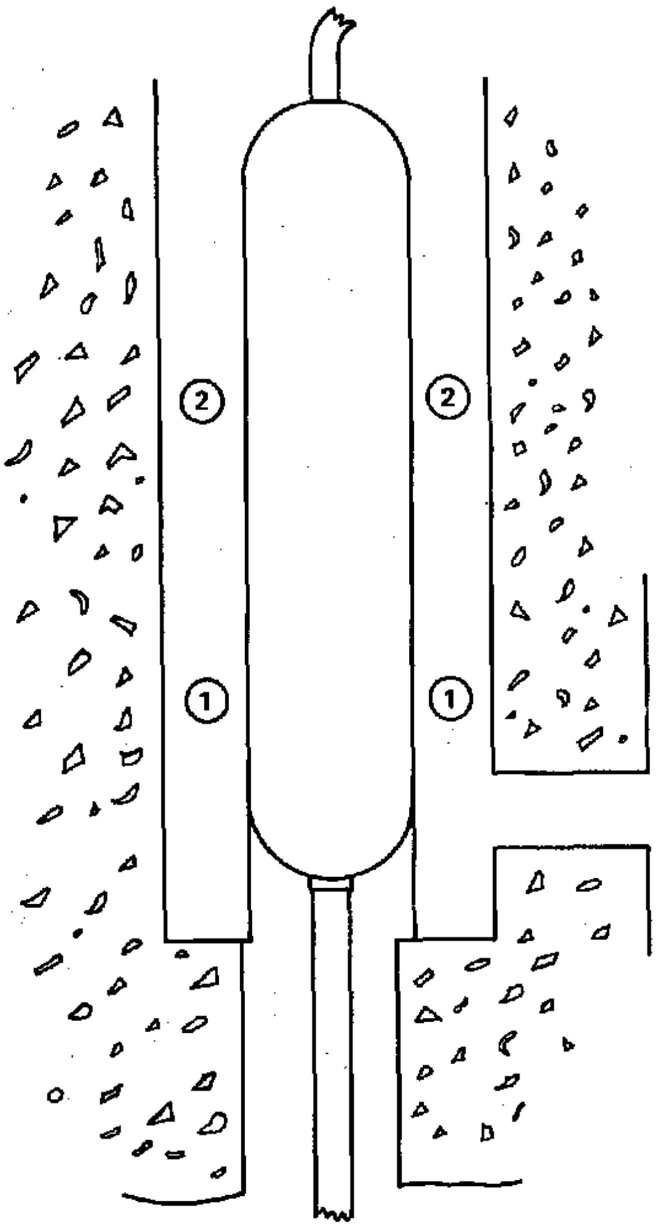
REV 21 5/08

SOUTHERN COMPANY
Energy to Serve Your World®

**JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2**

STEAM GENERATOR COMPARTMENT C
DIFFERENTIAL PRESSURE VS. TIME

FIGURE 6.2-54



13

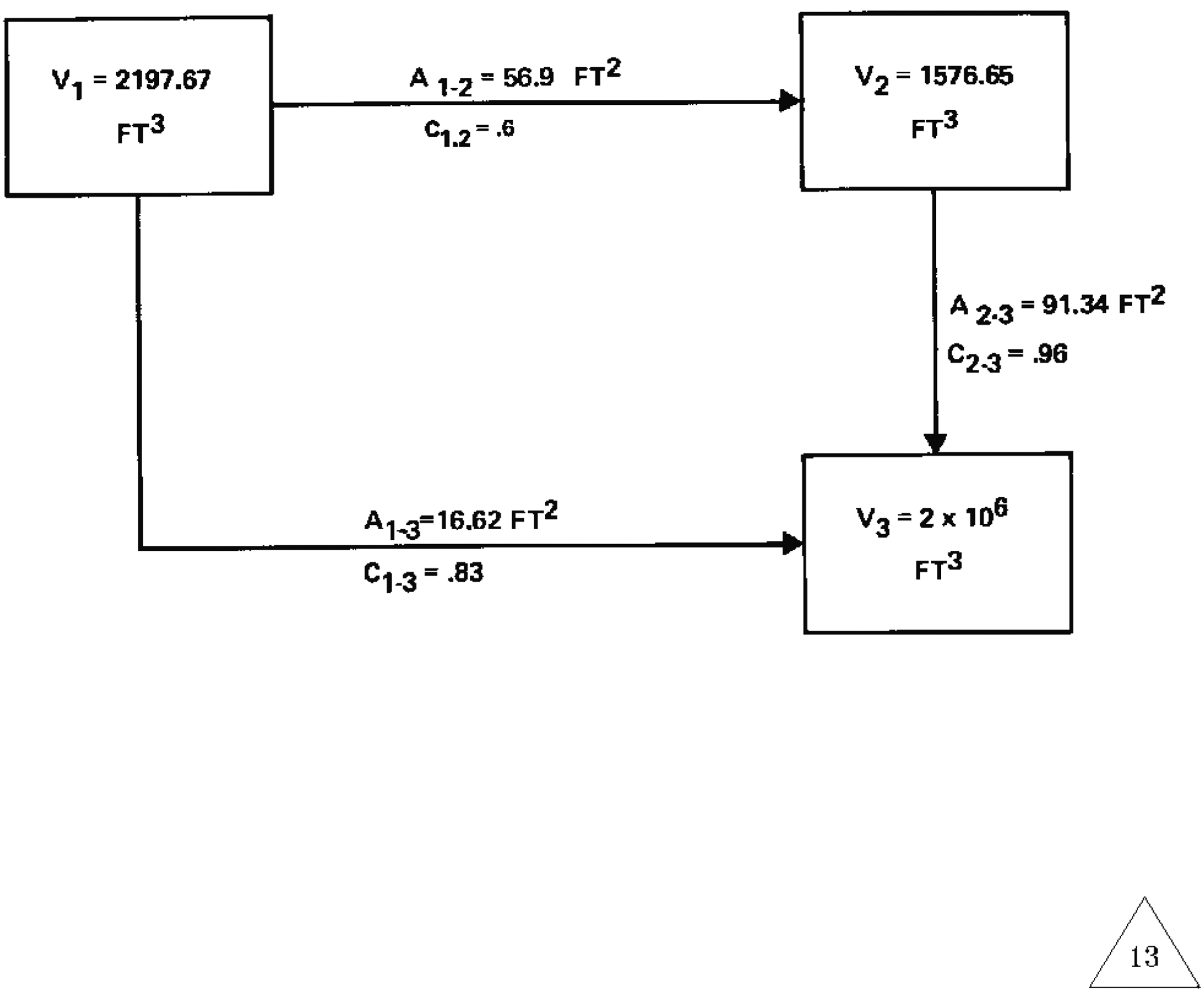
REV 21 5/08

SOUTHERN COMPANY
Energy to Serve Your World®

JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

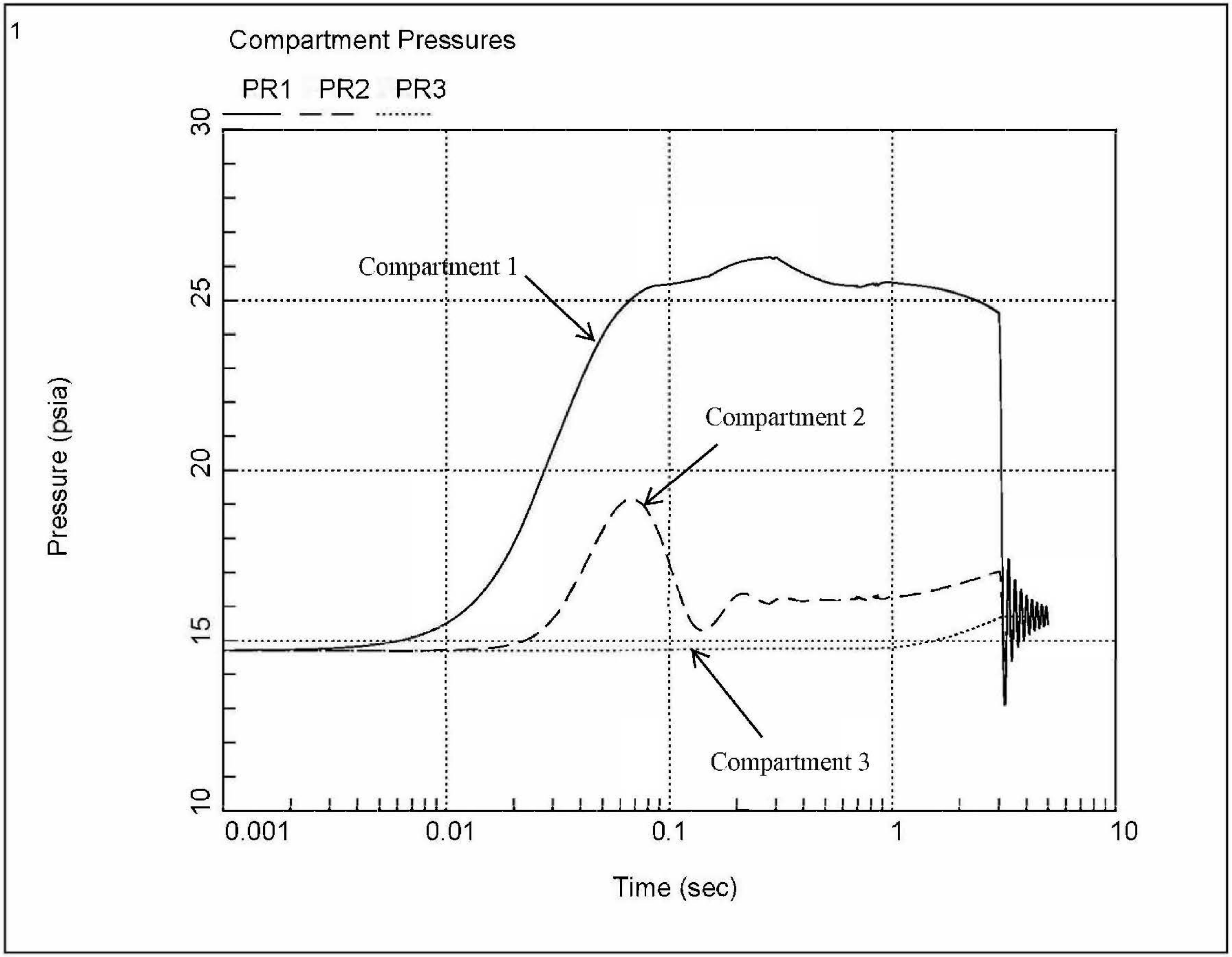
PRESSURIZER COMPARTMENT PRESSURE
MODEL (SPRAY LINE BREAK IN LOWER COMPARTMENT)

FIGURE 6.2-55



REV 21 5/08

SOUTHERN COMPANY *Energy to Serve Your World®*	JOSEPH M. FARLEY NUCLEAR PLANT UNIT 1 AND UNIT 2	PRESSURIZER COMPARTMENT FLOW MODEL
		FIGURE 6.2-56



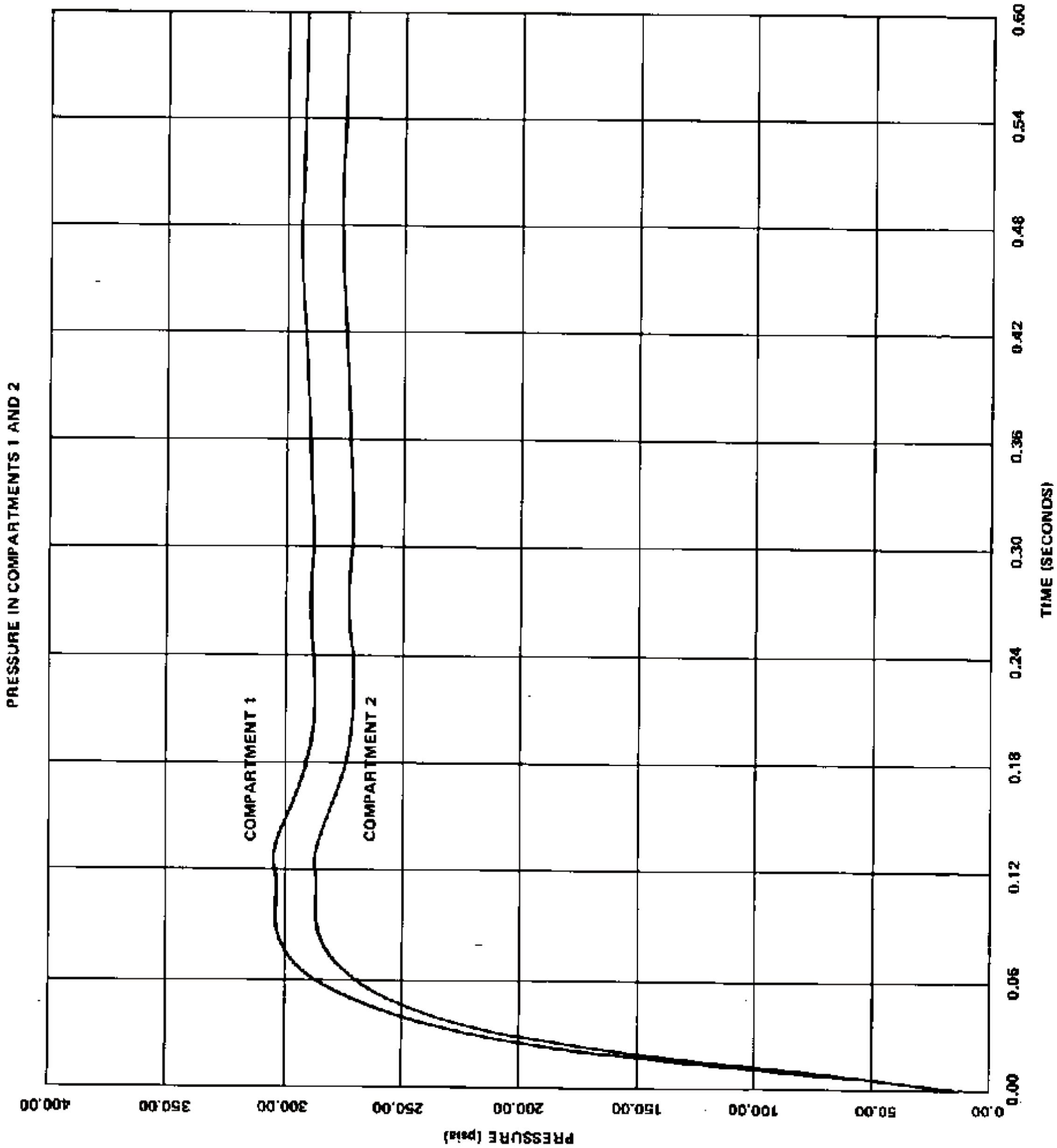
REV 29 4/20

SOUTHERN COMPANY
Energy to Serve Your World®

JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

PRESSURIZER COMPARTMENT SPRAY LINE RESULTS

FIGURE 6.2-57



13

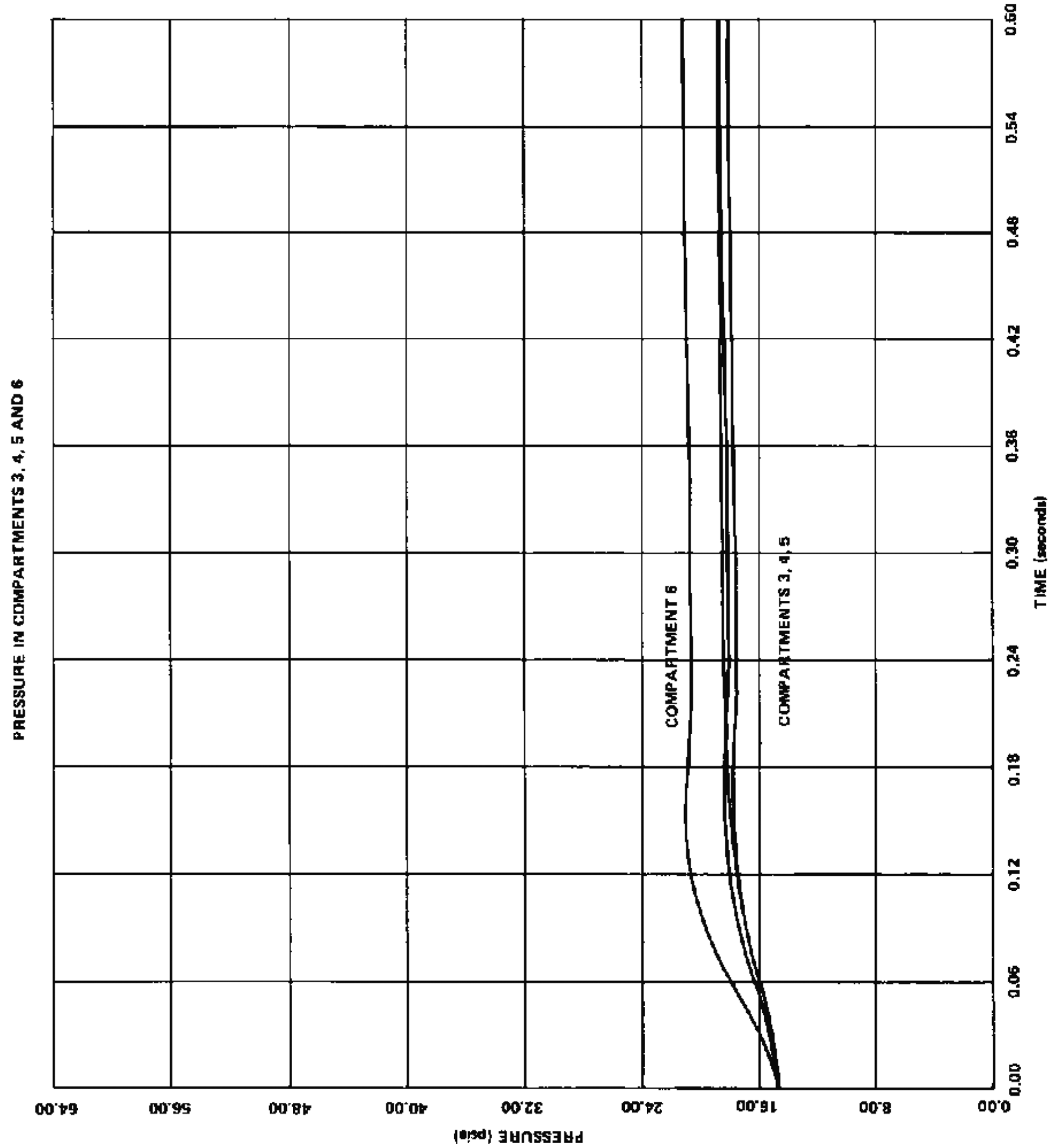
REV 21 5/08

SOUTHERN COMPANY
Energy to Serve Your World®

JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

NODE PRESSURES IN COMPARTMENTS
1 AND 2 VERSUS TIME

FIGURE 6.2-58



13

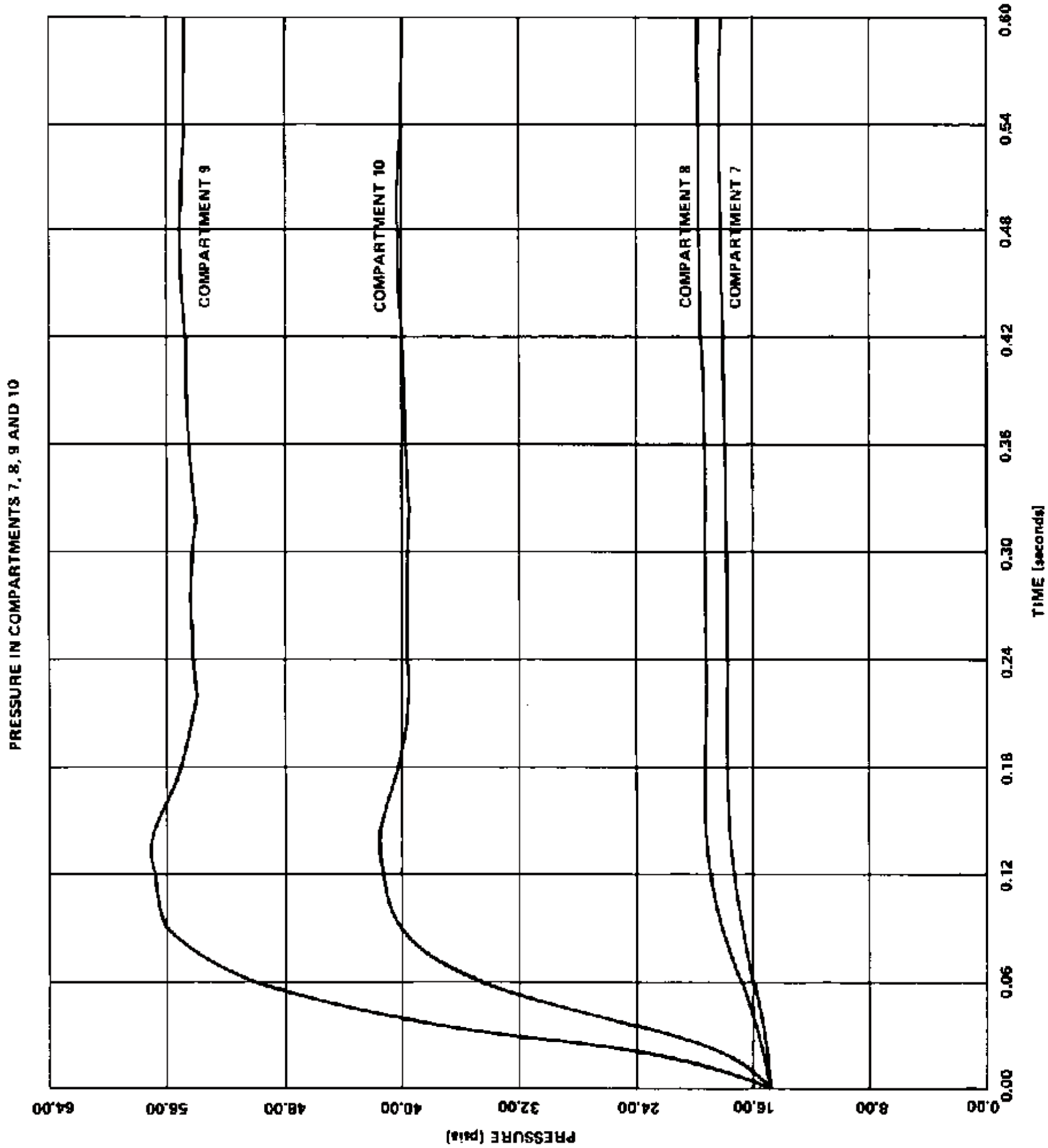
REV 21 5/08

SOUTHERN COMPANY
Energy to Serve Your World®

JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

NODE PRESSURES IN COMPARTMENTS
3, 4, 5, AND 6 VERSUS TIME

FIGURE 6.2-59



13

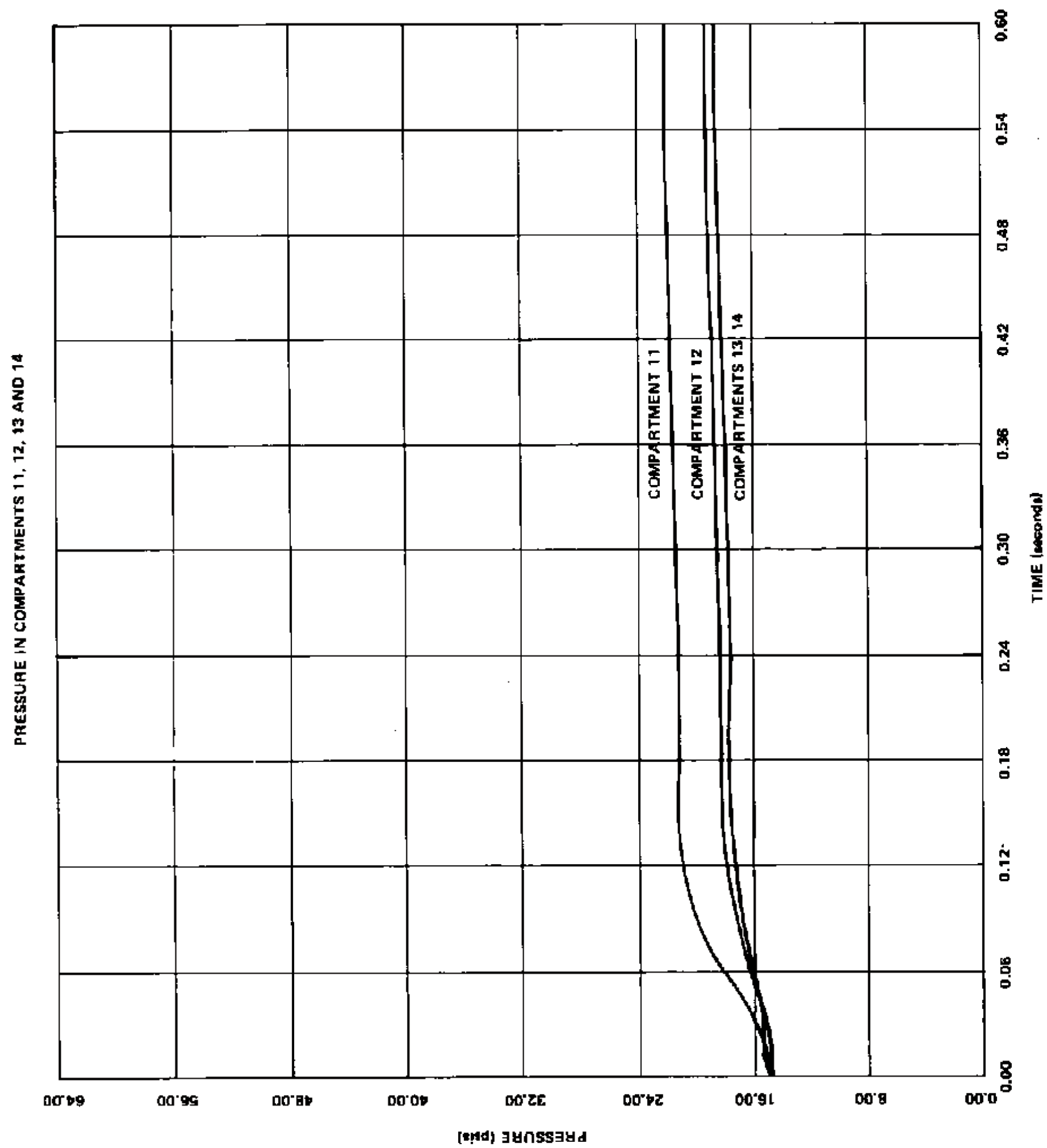
REV 21 5/08

SOUTHERN COMPANY
Energy to Serve Your World®

JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

NODE PRESSURES IN COMPARTMENTS
7, 8, 9, AND 10 VERSUS TIME

FIGURE 6.2-60



13

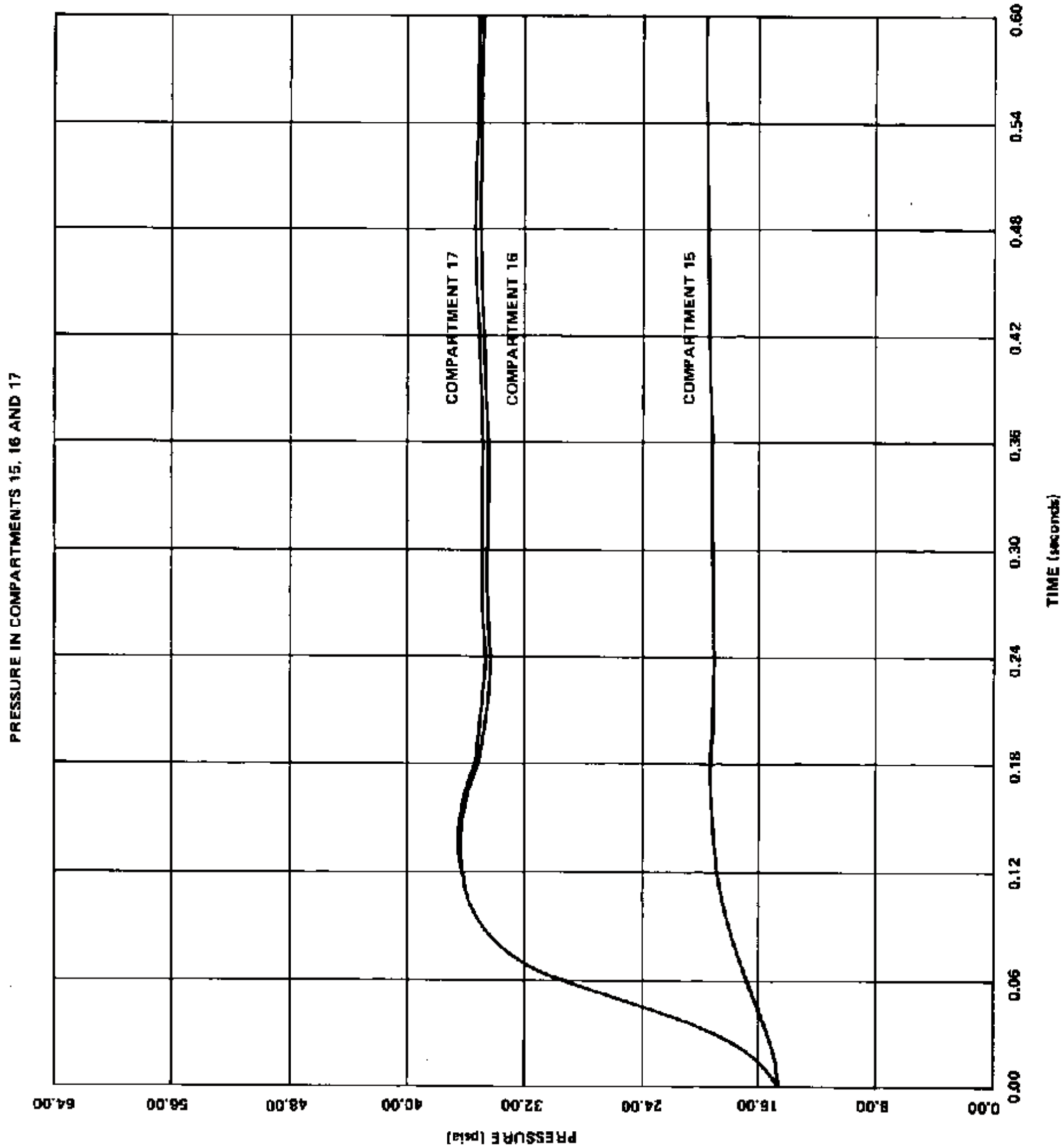
REV 21 5/08

SOUTHERN COMPANY
Energy to Serve Your World®

JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

NODE PRESSURES IN COMPARTMENTS
11, 12, 13, AND 14 VERSUS TIME

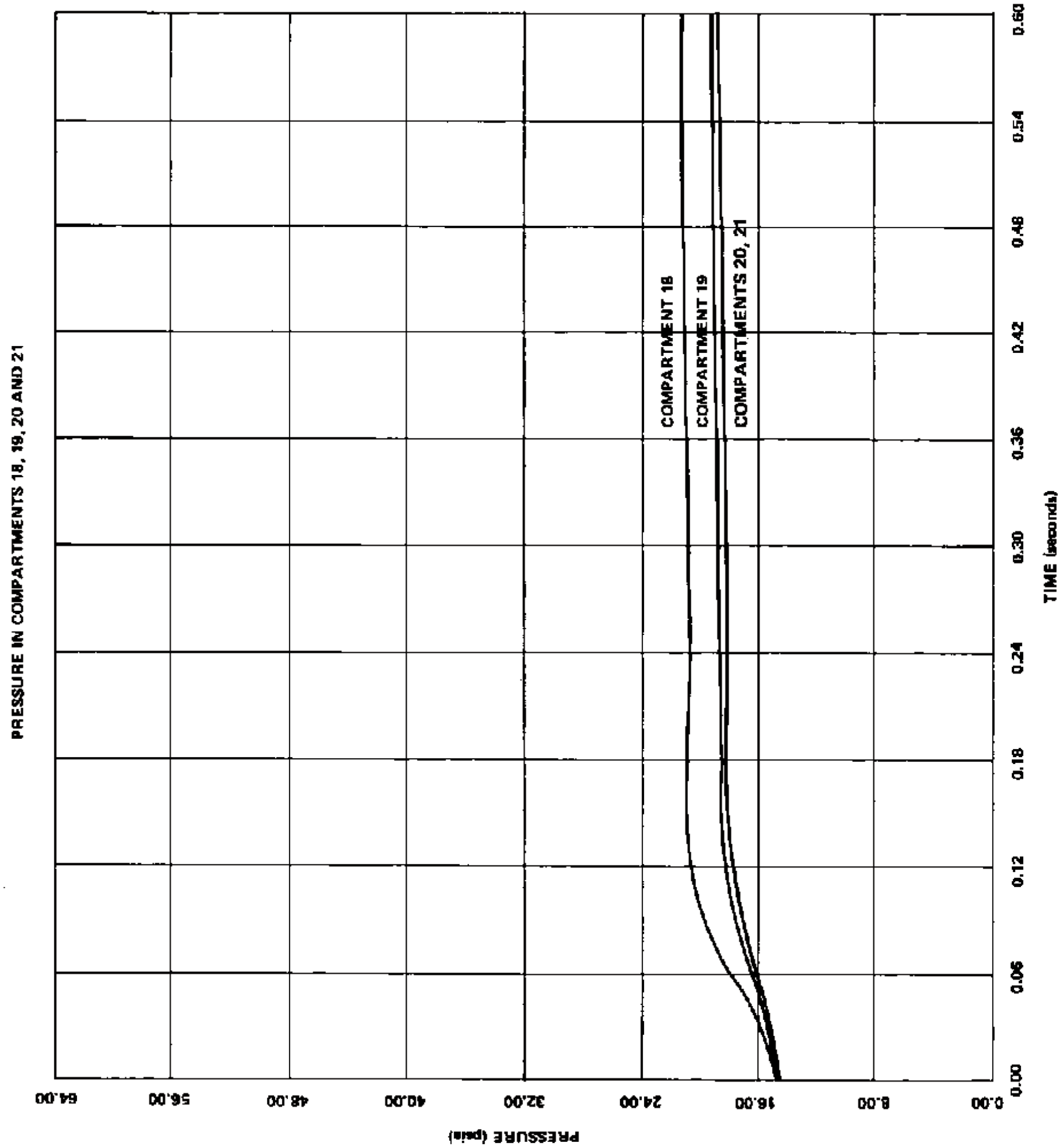
FIGURE 6.2-61



13

REV 21 5/08

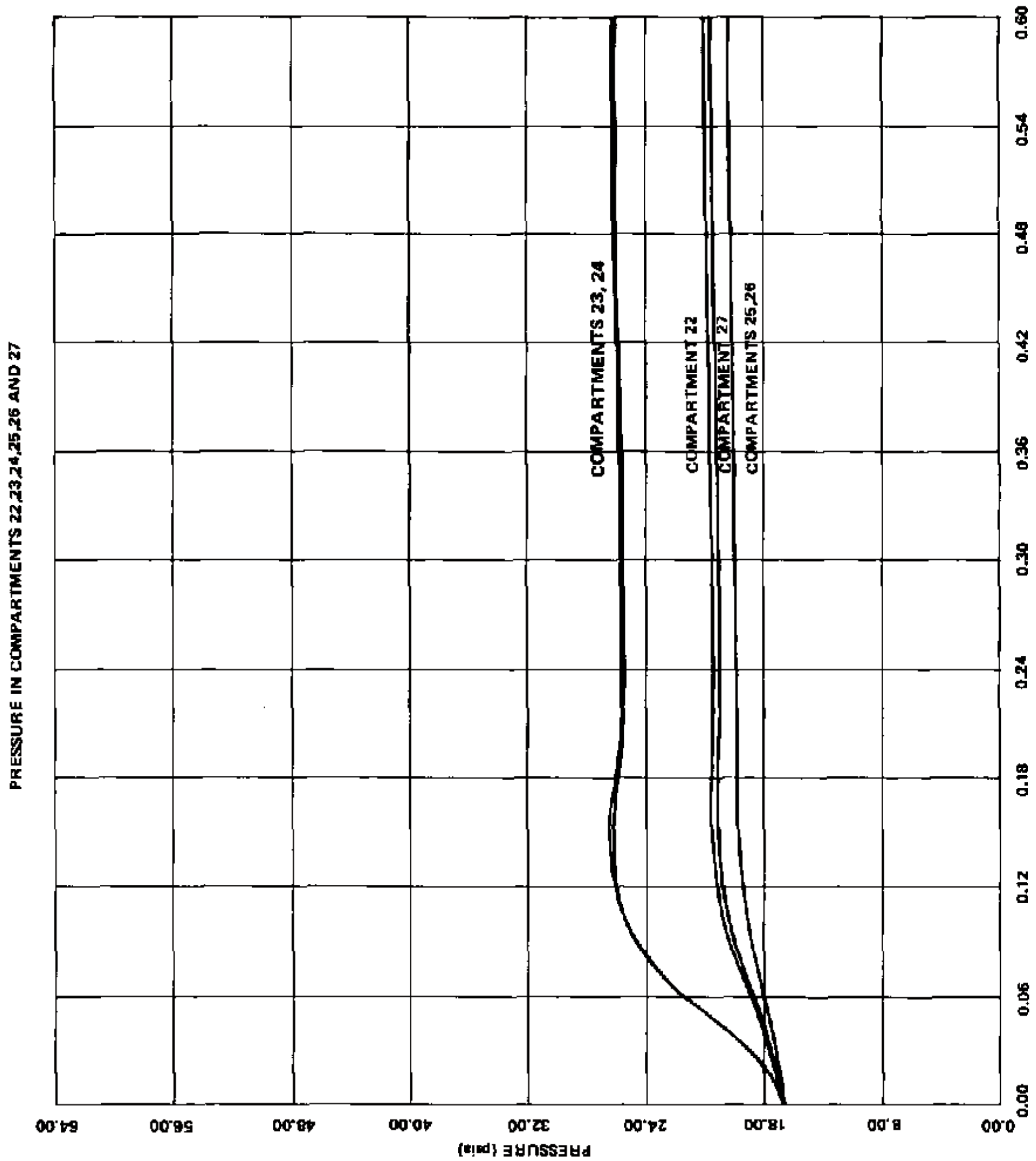
SOUTHERN COMPANY *Energy to Serve Your World®*	JOSEPH M. FARLEY NUCLEAR PLANT UNIT 1 AND UNIT 2	NODE PRESSURES IN COMPARTMENTS 15, 16, AND 17 VERSUS TIME FIGURE 6.2-62



13

REV 21 5/08

SOUTHERN COMPANY Energy to Serve Your World®	JOSEPH M. FARLEY NUCLEAR PLANT UNIT 1 AND UNIT 2	NODE PRESSURES IN COMPARTMENTS 18, 19, 20, AND 21 VERSUS TIME FIGURE 6.2-63



13

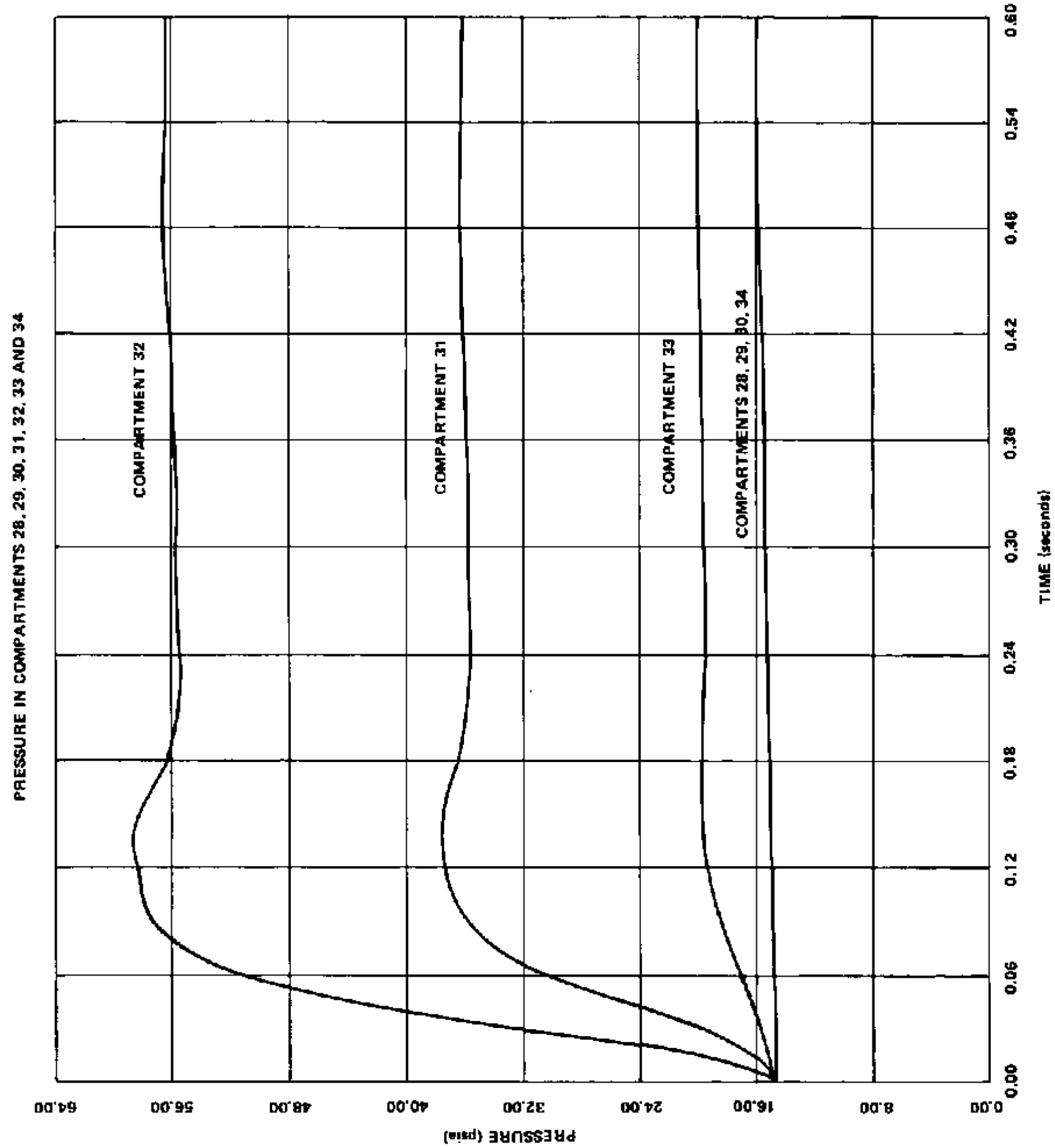
REV 21 5/08

SOUTHERN COMPANY
Energy to Serve Your World®

JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

NODE PRESSURES IN COMPARTMENTS
22, 23, 24, 25, 26, AND 27 VERSUS TIME

FIGURE 6.2-64



13

REV 21 5/08

SOUTHERN COMPANY
Energy to Serve Your World®

JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

NODE PRESSURES IN COMPARTMENTS
28, 29, 30, 31, 32, 33, AND 34 VERSUS TIME

FIGURE 6.2-65

THIS FIGURE HAS BEEN DELETED.

15

REV 21 5/08

SOUTHERN COMPANY
Energy to Serve Your World®

JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

SCHEMATIC OF REFLOOD CODE 19
ELEMENT LOOP MODEL FOR A PUMP SUCTION BREAK

FIGURE 6.2-66

THIS FIGURE HAS BEEN DELETED.

15

REV 21 5/08

SOUTHERN COMPANY
Energy to Serve Your World®

JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

CORE REFLOOD CORRELATION

FIGURE 6.2-67

THIS FIGURE HAS BEEN DELETED PER REV 15.

REV 21 5/08

SOUTHERN COMPANY *Energy to Serve Your World®*	JOSEPH M. FARLEY NUCLEAR PLANT UNIT 1 AND UNIT 2	COMPARISON OF MEASURE AND PREDICTED CARRY OVER RATE FRACTIONS FIGURE 6.2-68

THIS FIGURE HAS BEEN DELETED PER REV 15.

REV 21 5/08

THIS FIGURE HAS BEEN DELETED.

15

REV 21 5/08

SOUTHERN COMPANY
Energy to Serve Your World®

JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

VARIATION IN TEMPERATURE RISE,
TURNAROUND TIME AND QUENCH TIME
WITH RESPECT TO CORE ELEVATION

FIGURE 6.2-70

THIS FIGURE HAS BEEN DELETED.

15

REV 21 5/08

SOUTHERN COMPANY
Energy to Serve Your World®

JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

ENERGY BALANCE MODEL

FIGURE 6.2-71

THIS FIGURE HAS BEEN DELETED.

15

REV 21 5/08

SOUTHERN COMPANY
Energy to Serve Your World®

JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

REFLOOD RATE AND
CARRYOVER FRACTIONS VS.
TIME AFTER END OF BLOWDOWN

FIGURE 6.2-72

THIS FIGURE HAS BEEN DELETED.

15

REV 21 5/08

SOUTHERN COMPANY
Energy to Serve Your World®

JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

FLOW THROUGH BREAK VS.
TIME AFTER END OF BLOWDOWN

FIGURE 6.2-73

THIS FIGURE HAS BEEN DELETED.

15

REV 21 5/08

SOUTHERN COMPANY
Energy to Serve Your World®

JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

WATER HEIGHT VS. TIME AFTER END OF BLOWDOWN

FIGURE 6.2-74

THIS FIGURE HAS BEEN DELETED.

15

REV 21 5/08

SOUTHERN COMPANY
Energy to Serve Your World®

JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

POST-REFLOOD LOOP RESISTANCE MODEL

FIGURE 6.2-75

THIS FIGURE HAS BEEN DELETED.

15

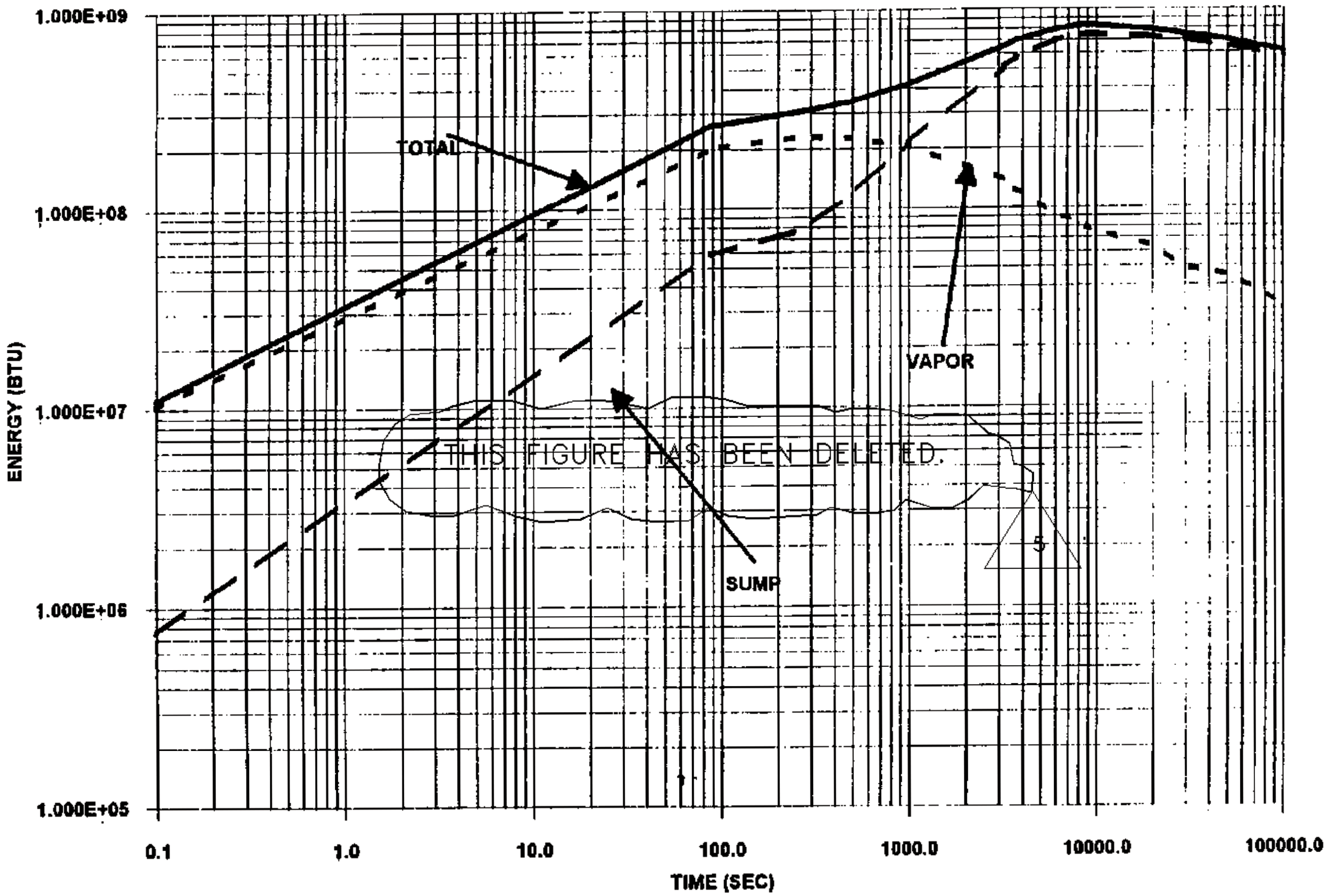
REV 21 5/08

SOUTHERN COMPANY
Energy to Serve Your World®

JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

S/G INTERNAL ENERGY VS. TIME AFTER BREAK

FIGURE 6.2-76



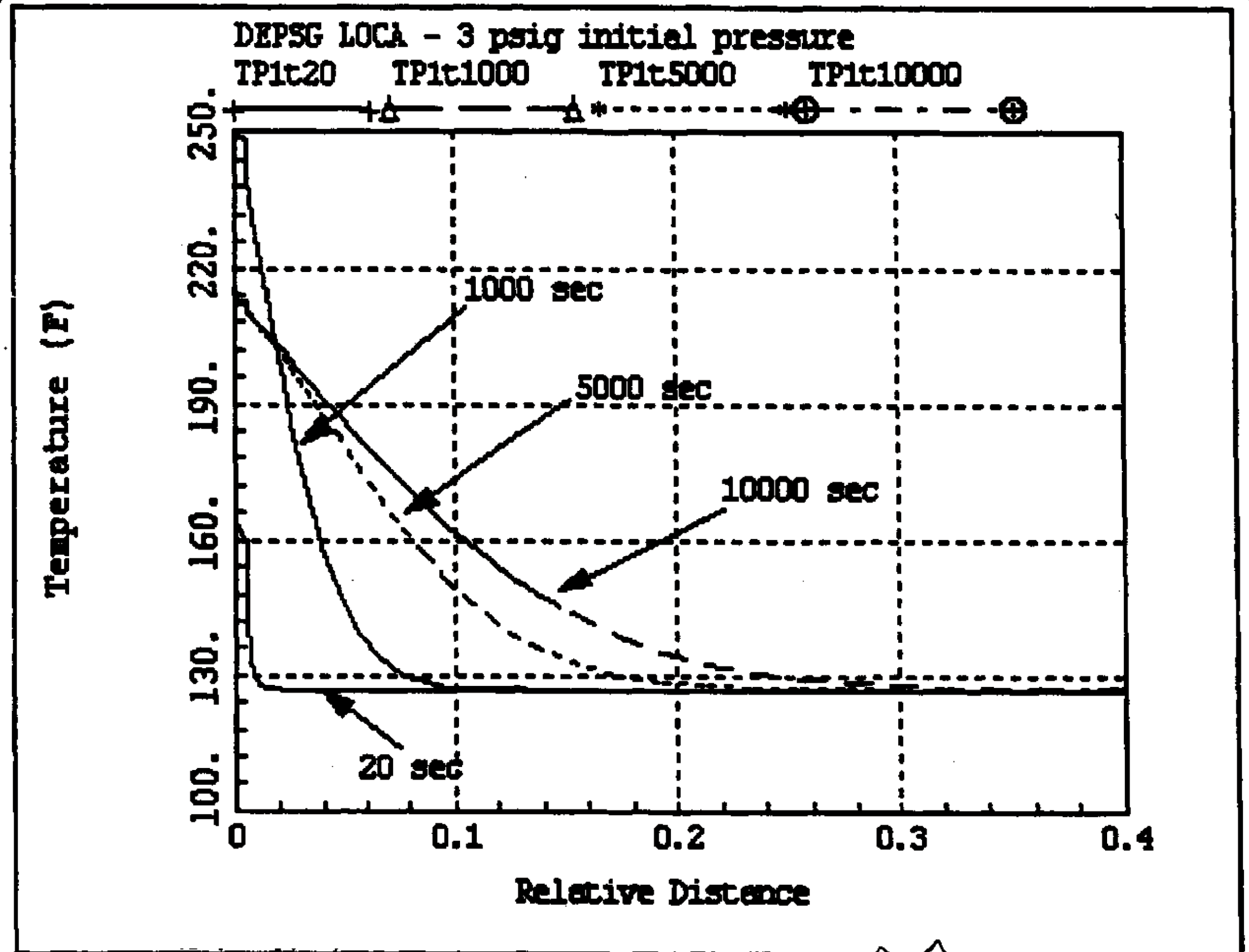
REV 21 5/08

SOUTHERN COMPANY
Energy to Serve Your World®

JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

ENERGY DISTRIBUTION VS. TIME

FIGURE 6.2-77



17

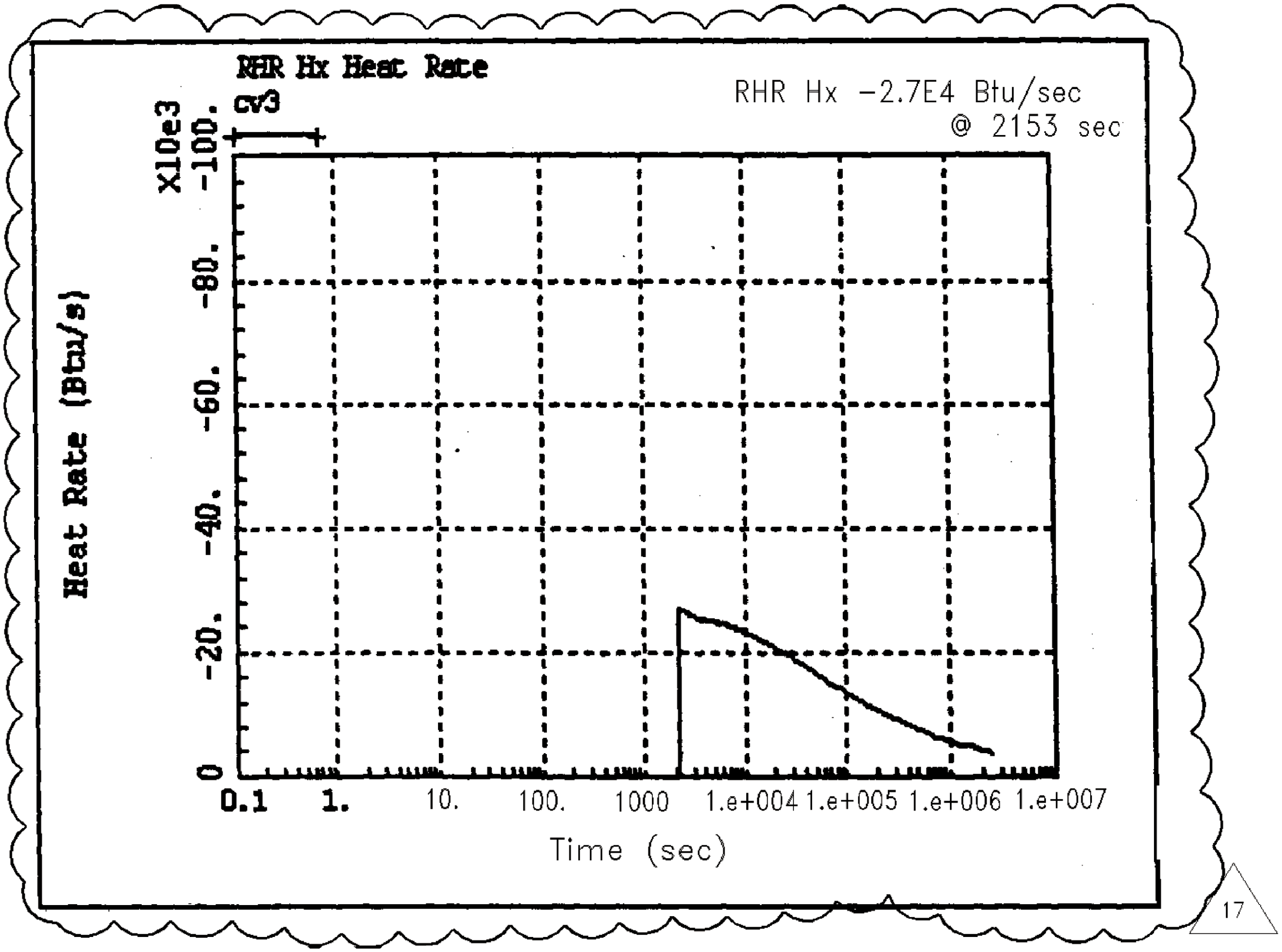
REV 21 5/08

SOUTHERN COMPANY
Energy to Serve Your World®

JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

RSG TEMPERATURE PROFILE
THROUGH CONTAINMENT WALL
P_O = +3 PSIG

FIGURE 6.2-78



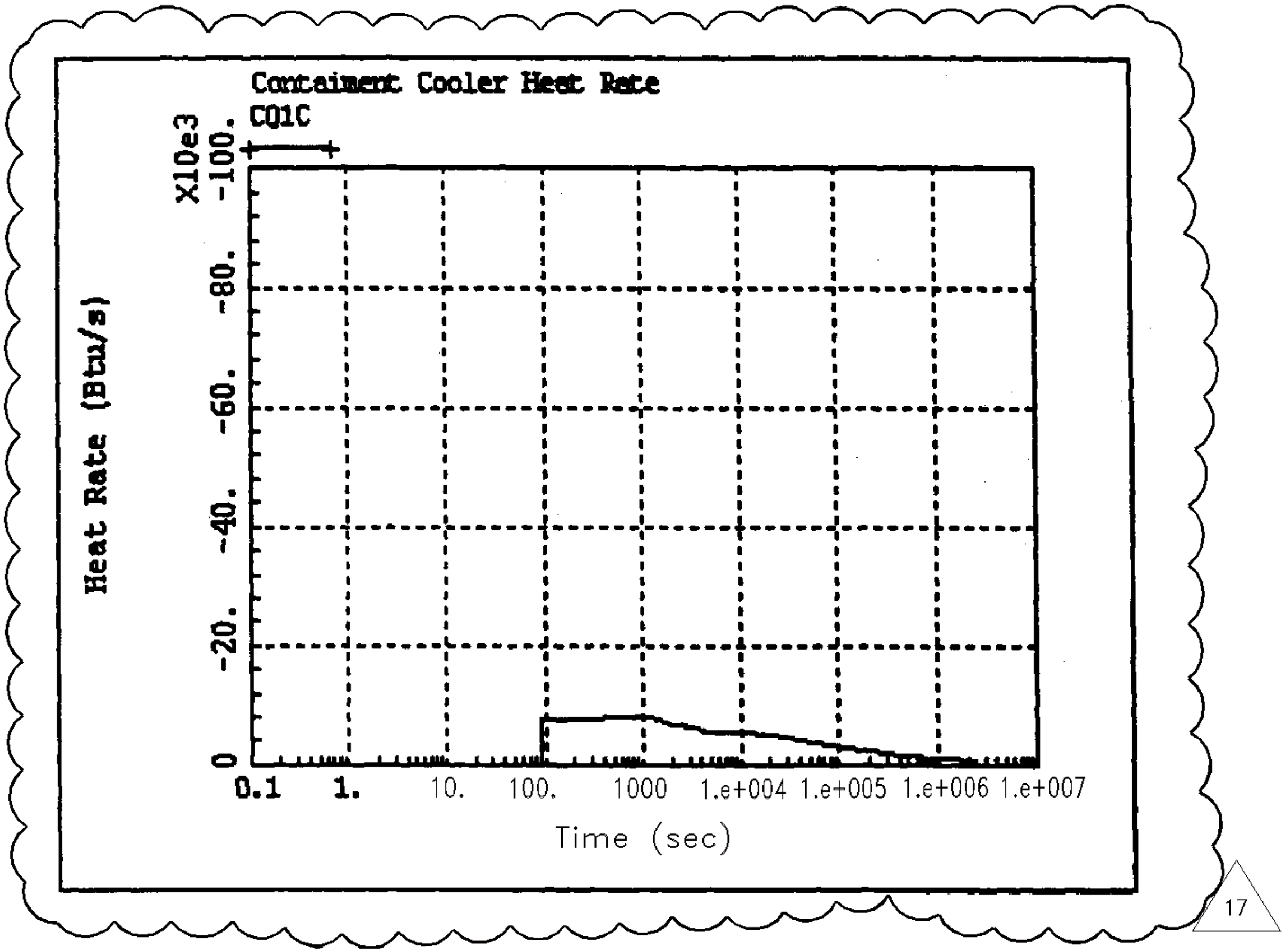
REV 21 5/08

SOUTHERN COMPANY
Energy to Serve Your World®

JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

RHR HX DUTY VS. TIME
RSG, P_O = +3 PSIG

FIGURE 6.2-79



REV 21 5/08

SOUTHERN COMPANY
Energy to Serve Your World®

JOSEPH M. FARLEY NUCLEAR PLANT UNIT 1 AND UNIT 2

CONTAINMENT AIR COOLING DUTY VS. TIME
RSG, P_O = +3 PSIG

FIGURE 6.2-80

THIS FIGURE HAS BEEN DELETED

15

REV 21 5/08

SOUTHERN COMPANY
Energy to Serve Your World®

JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

MINIMUM SUMP pH FOLLOWING LOCA VERSUS TIME

FIGURE 6.2-81

THIS FIGURE HAS BEEN DELETED

15

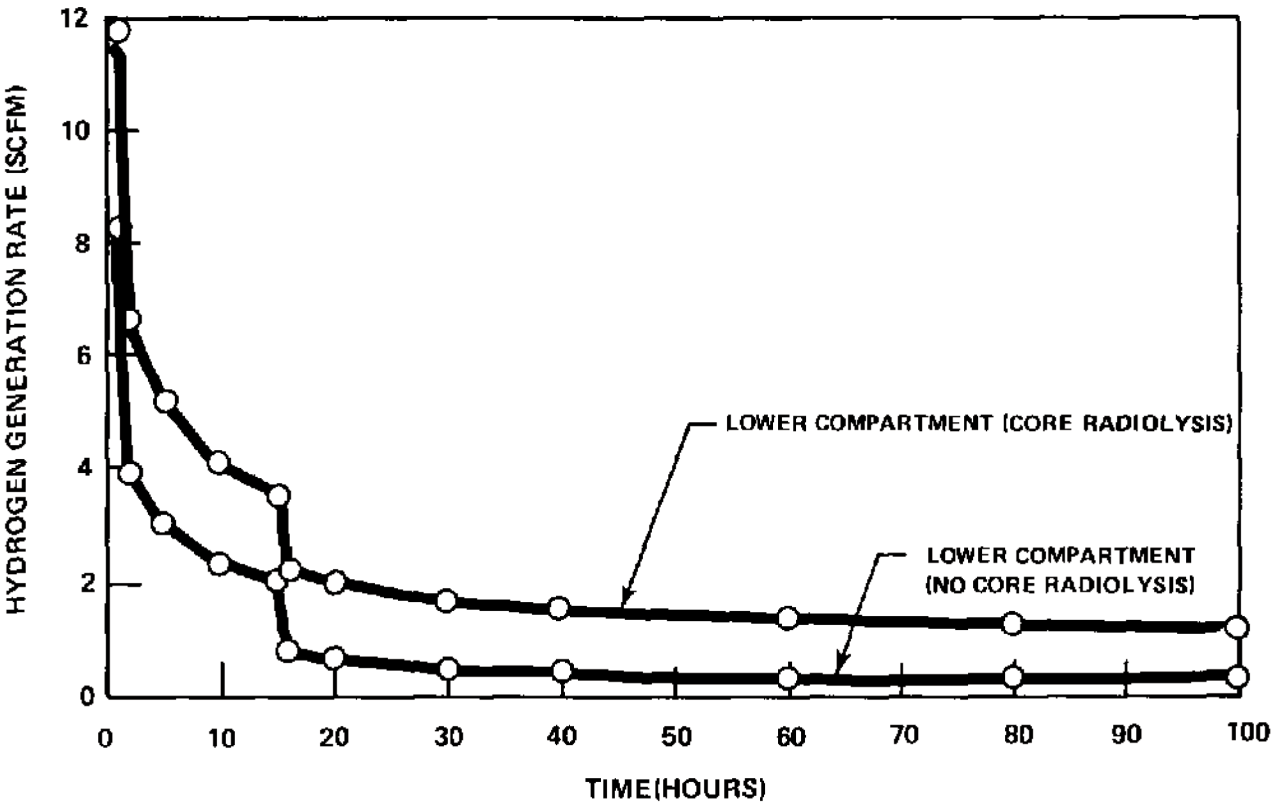
REV 21 5/08

SOUTHERN COMPANY
Energy to Serve Your World®

JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

MINIMUM PARTITION COEFFICIENT IN
THE SUMP VERSUS SOLUTION pH

FIGURE 6.2-82

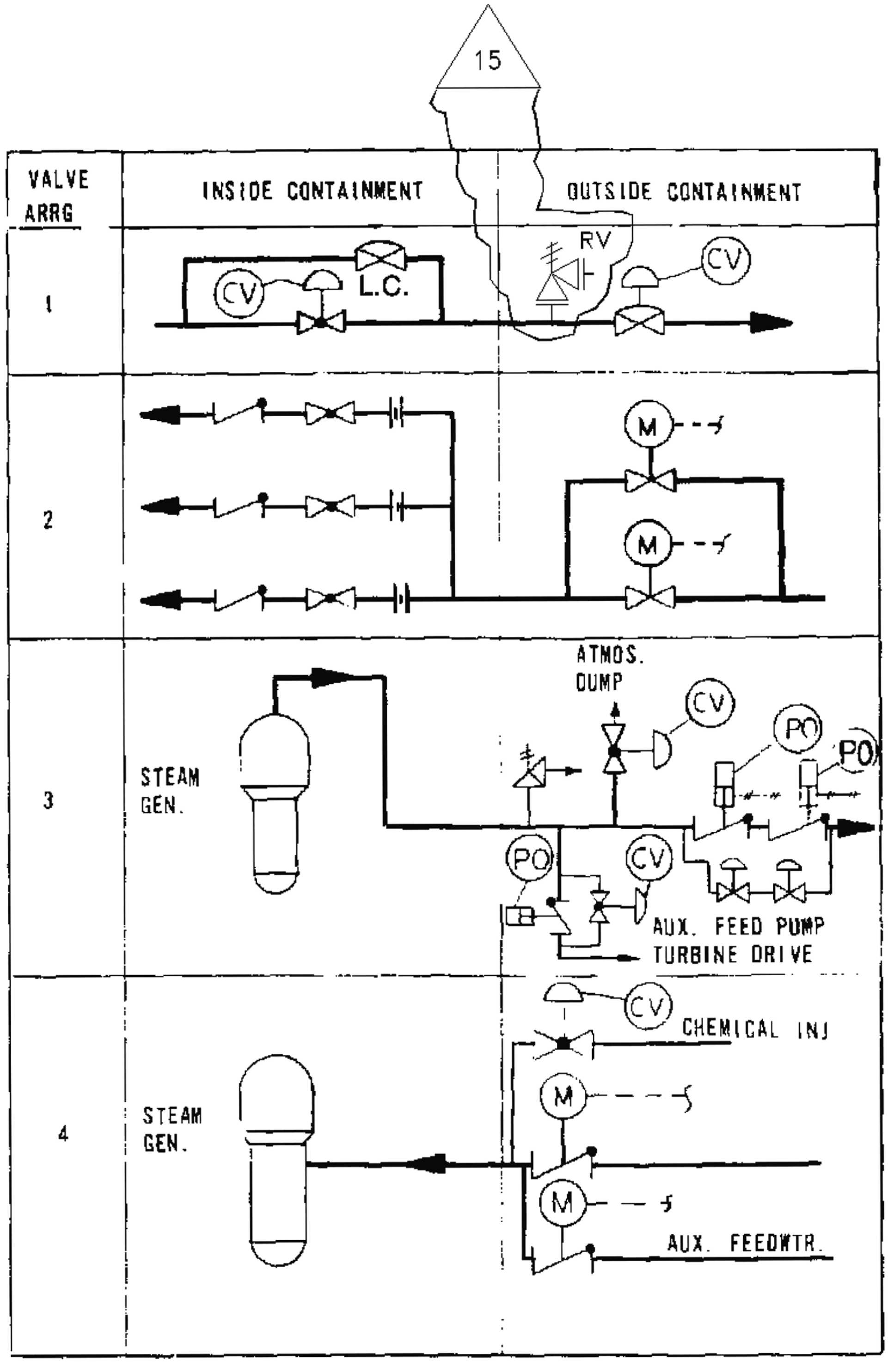


13

REV 21 5/08

HYDROGEN GENERATION RATE VS. TIME IN THE LOWER COMPARTMENT

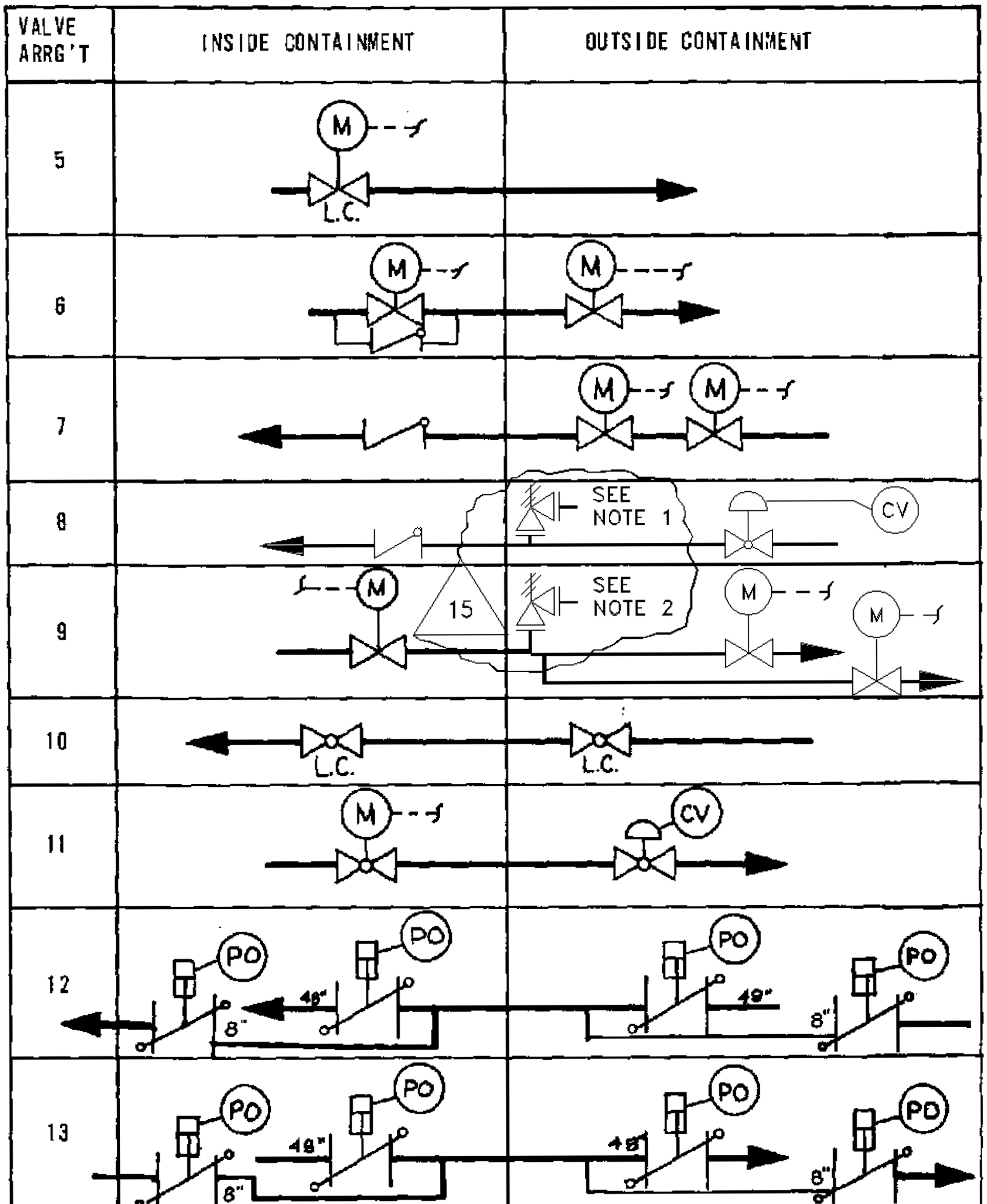
FIGURE 6.2-83



REV 21 5/08

ISOLATION VALVE ARRANGEMENT

FIGURE 6.2-84



NOTES:

1. RELIEF VALVE SHOWN OUTSIDE CONTAINMENT FOR ARRANGEMENT 8 IS APPLICABLE TO PENETRATION 45 ONLY. THE RELIEF VALVE IS CLASSIFIED AS A CONTAINMENT ISOLATION VALVE.
2. DETAIL APPLICABLE TO PENETRATIONS 38, 39, 40, AND 41 ONLY.

15

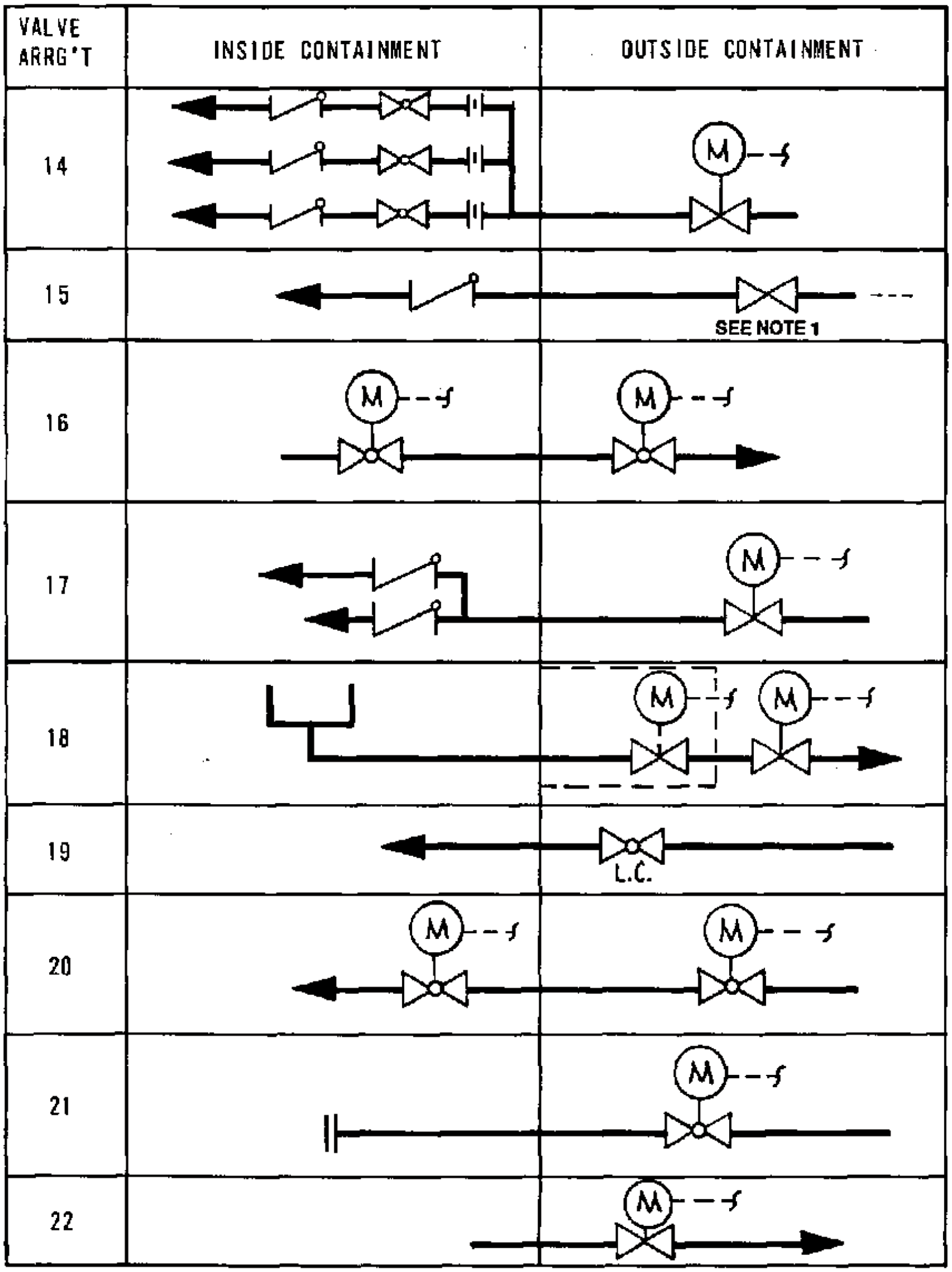
REV 21 5/08

SOUTHERN COMPANY
Energy to Serve Your World®

JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

ISOLATION VALVE ARRANGEMENT

FIGURE 6.2-85



Note 1: Containment Isolation Protection Is Provided By The High Pressure Seal Water In-Flow And The Check Valve Inside Containment. The Valve Outside Containment For Arrangement 15 Is Shown For Completeness Only And Is Not A Containment Isolation Valve.

13

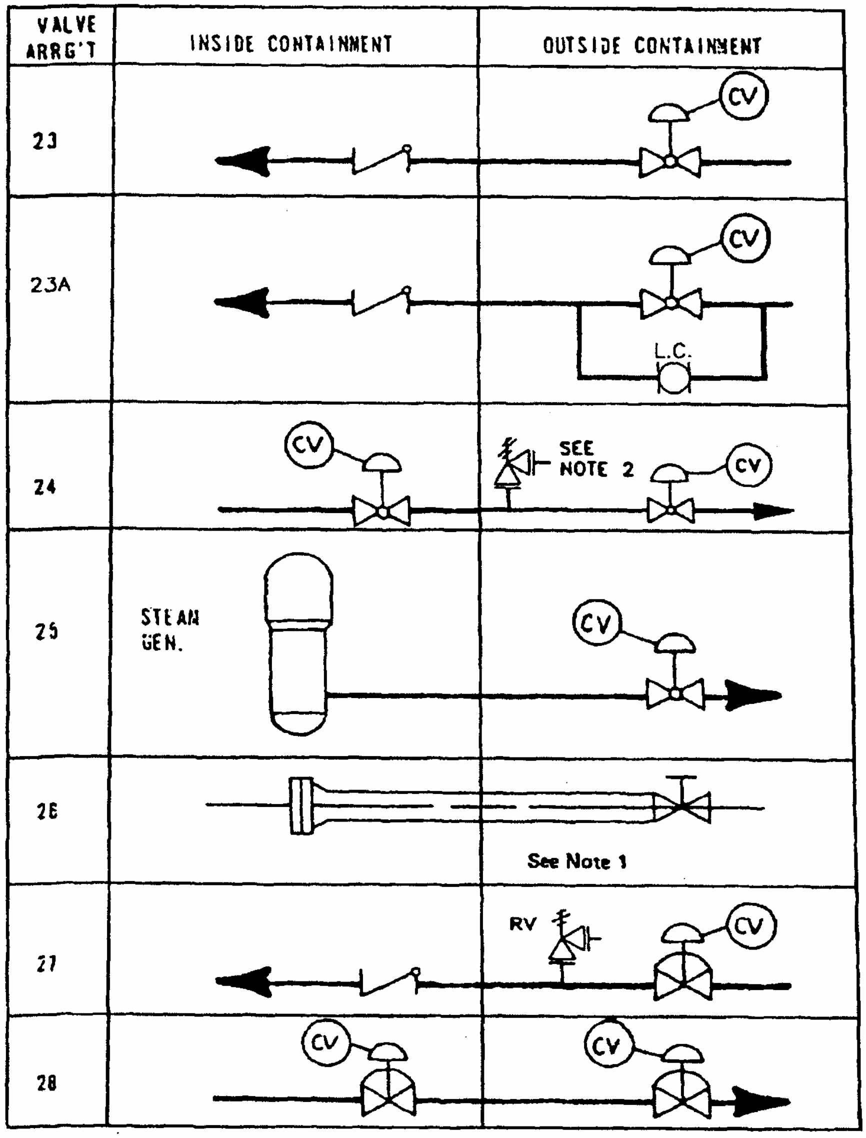
REV 21 5/08

SOUTHERN COMPANY
Energy to Serve Your World®

JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

ISOLATION VALVE ARRANGEMENT

FIGURE 6.2-86



Note 1: Containment isolation is provided by the blind flange inside containment. Valve outside containment for arrangement 26 is shown for completeness only and is not a containment isolation valve.

Note 2: Relief valve shown outside containment for arrangement 24 is applicable to penetration 46 only. The relief valve is classified as a containment isolation valve.

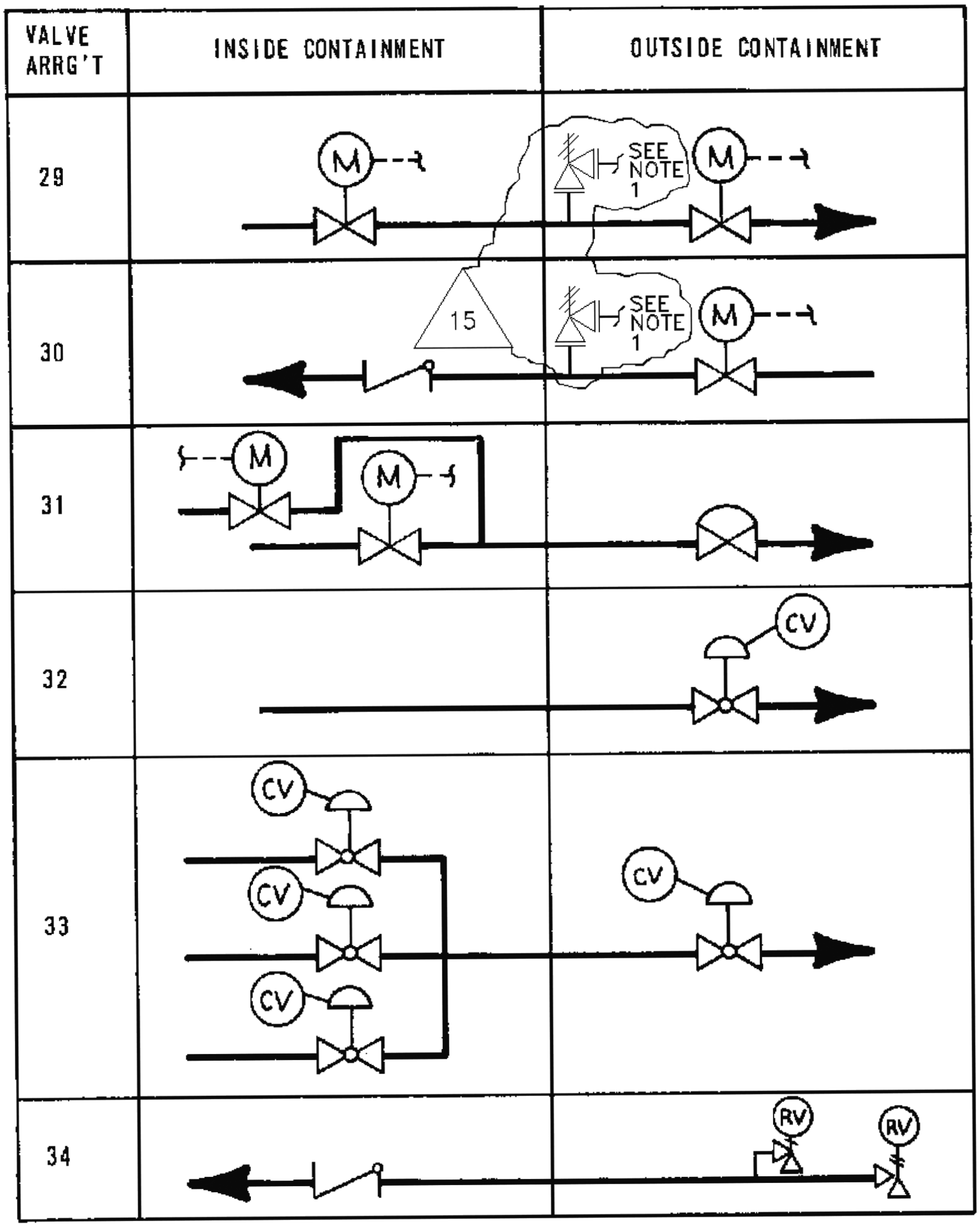
REV 21 5/08

SOUTHERN COMPANY
Energy to Serve Your World®

JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

ISOLATION VALVE ARRANGEMENT

FIGURE 6.2-87



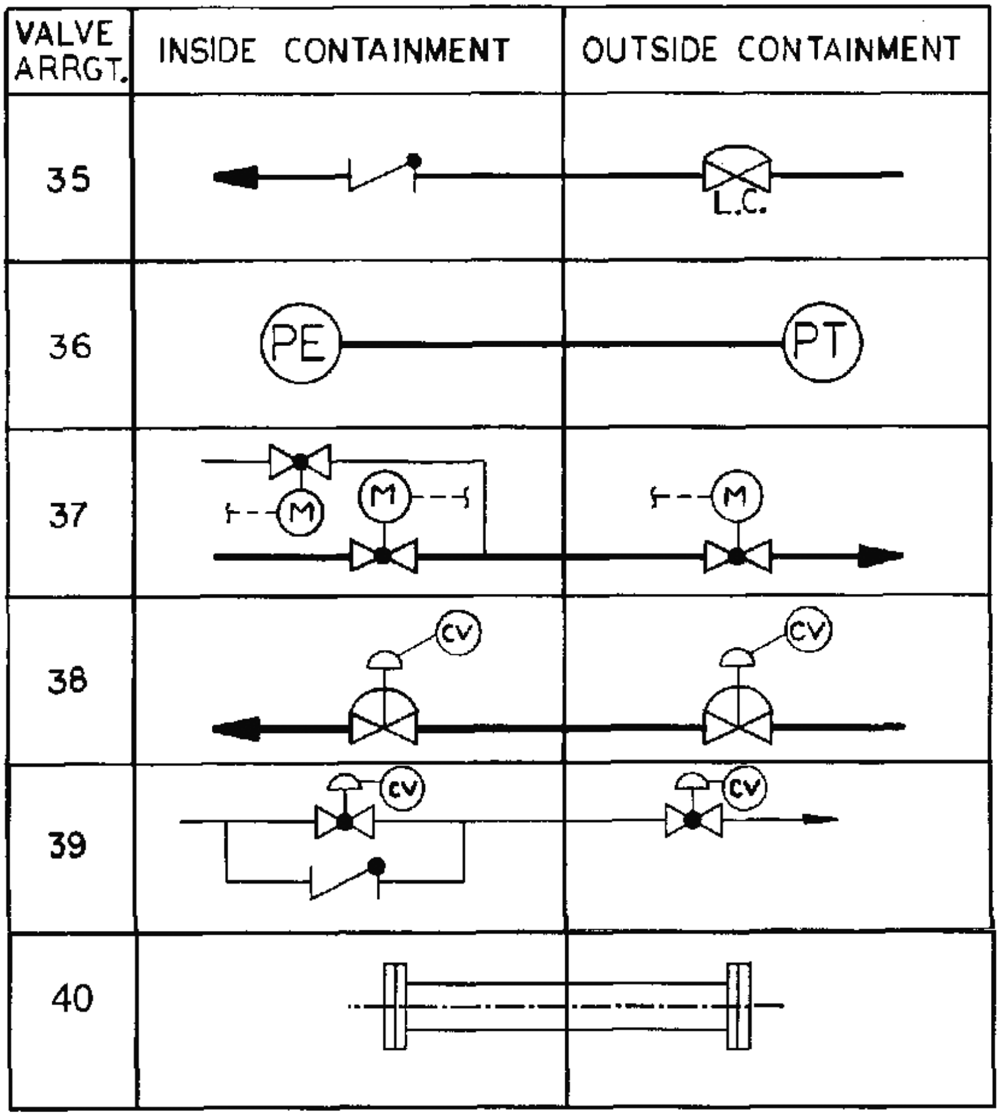
NOTE:

1. RELIEF VALVE SHOWN OUTSIDE CONTAINMENT FOR ARRANGEMENT 29 IS APPLICABLE TO PENETRATIONS 32 AND 44 ONLY. ARRANGEMENT 30 IS APPLICABLE TO PENETRATIONS 34, 35, 36, 37, 42, AND 60 ONLY. THESE RELIEF VALVES ARE CLASSIFIED AS CONTAINMENT ISOLATION VALVES.

15

REV 21 5/08

SOUTHERN COMPANY *Energy to Serve Your World®*	JOSEPH M. FARLEY NUCLEAR PLANT UNIT 1 AND UNIT 2	ISOLATION VALVE ARRANGEMENT FIGURE 6.2-88

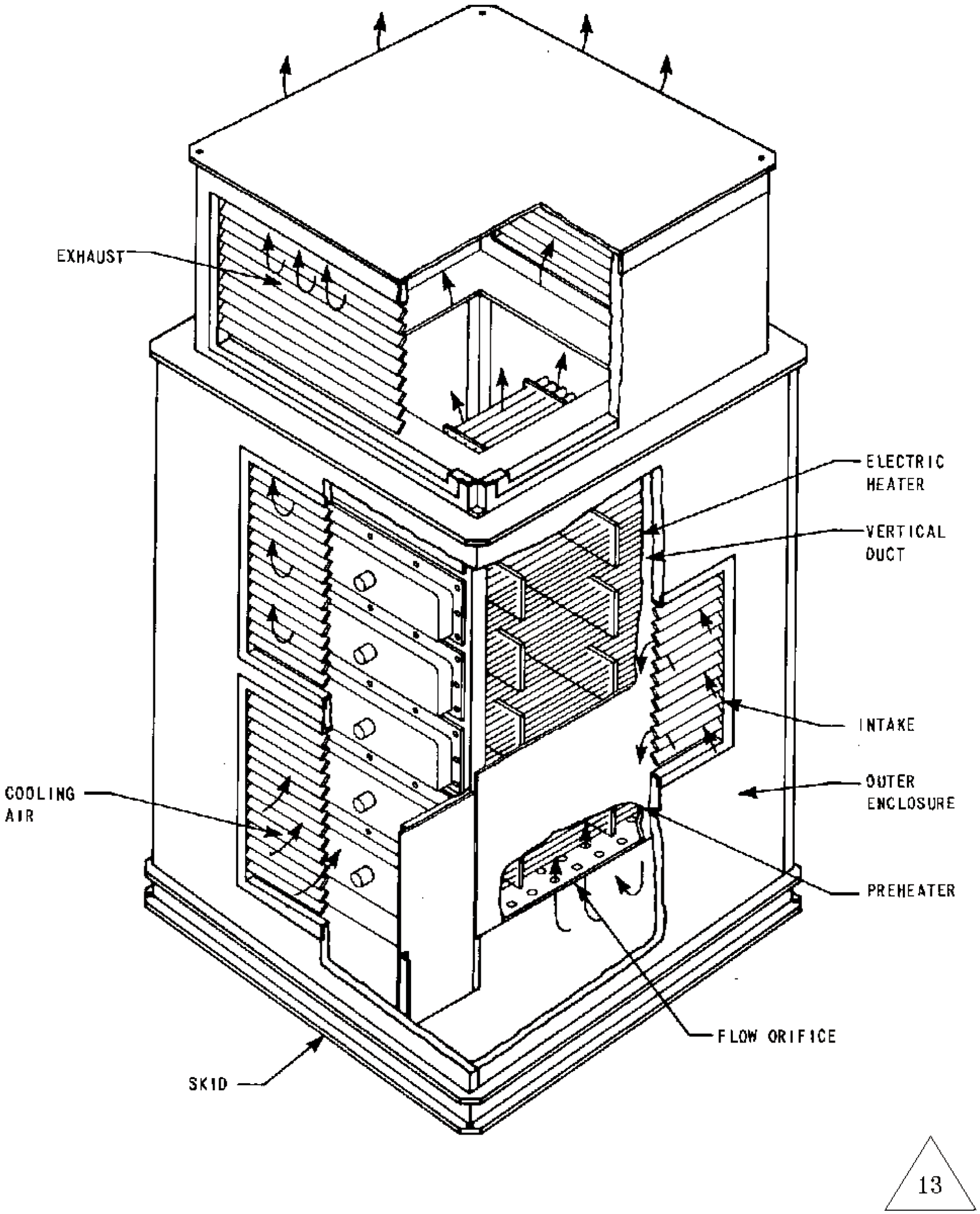


13

REV 21 5/08

ISOLATION VALVE ARRANGEMENT

FIGURE 6.2-89



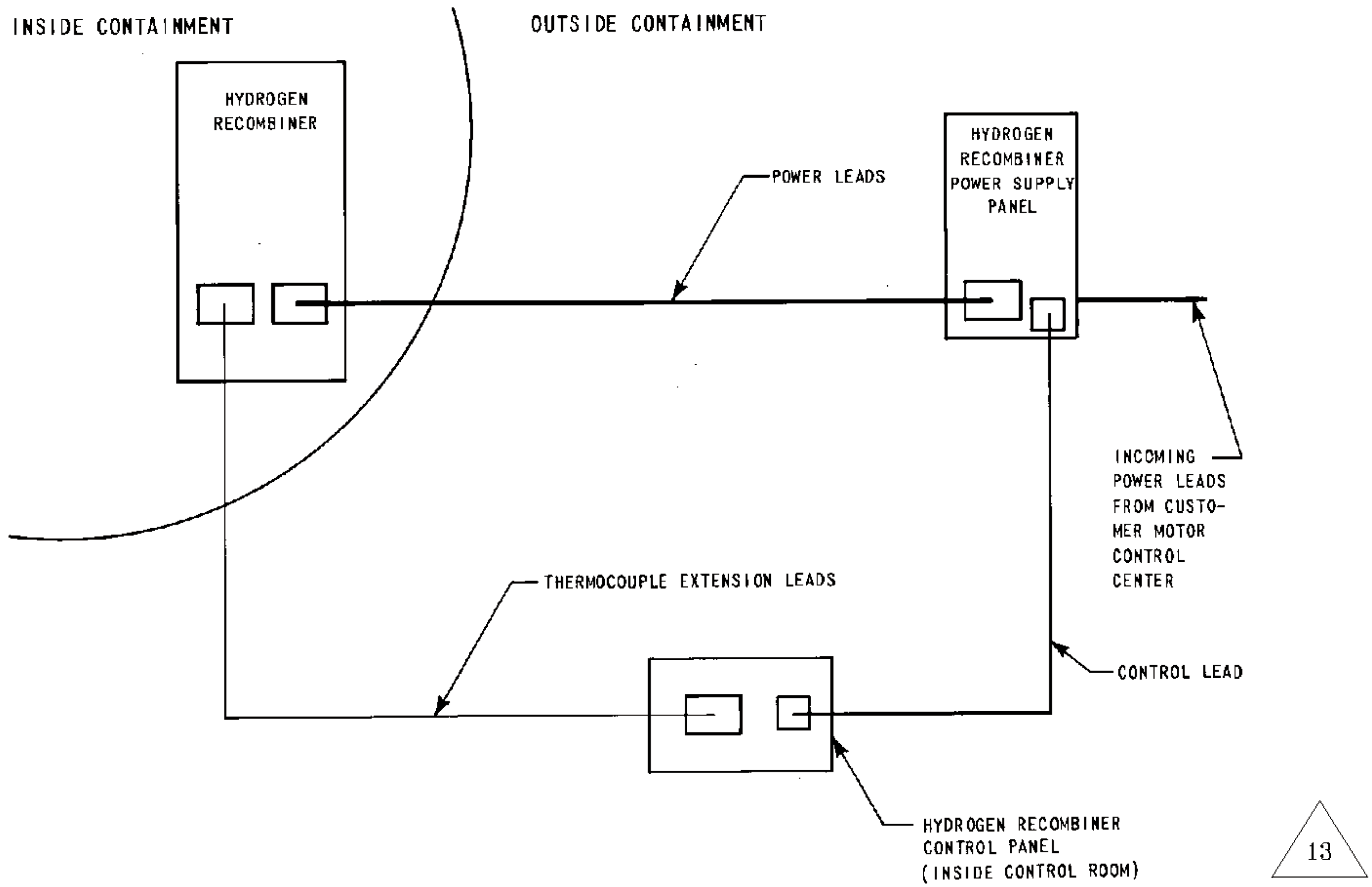
REV 21 5/08

SOUTHERN COMPANY
Energy to Serve Your World®

JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

ELECTRIC HYDROGEN RECOMBINER

FIGURE 6.2-90



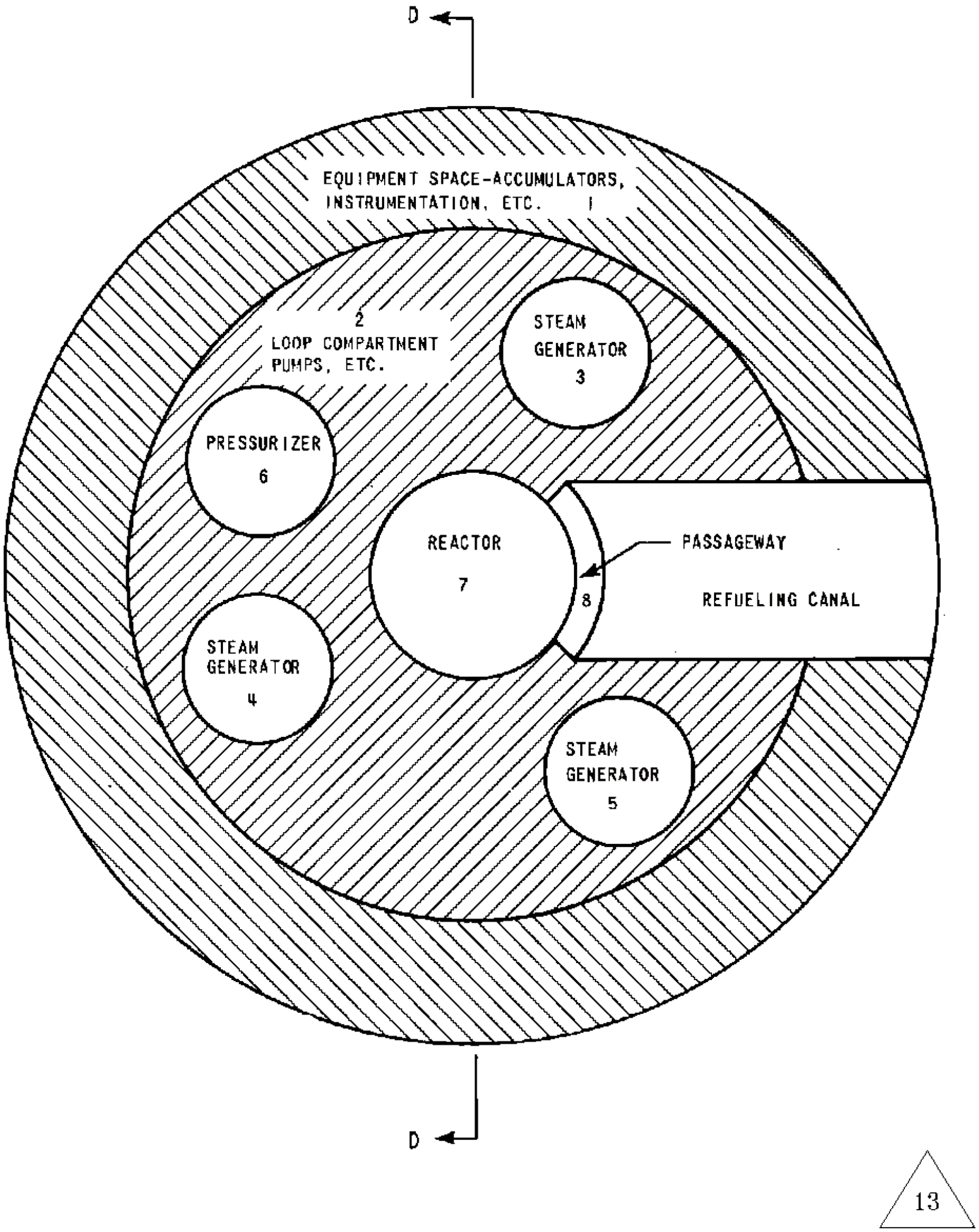
REV 21 5/08

SOUTHERN COMPANY
Energy to Serve Your World®

JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

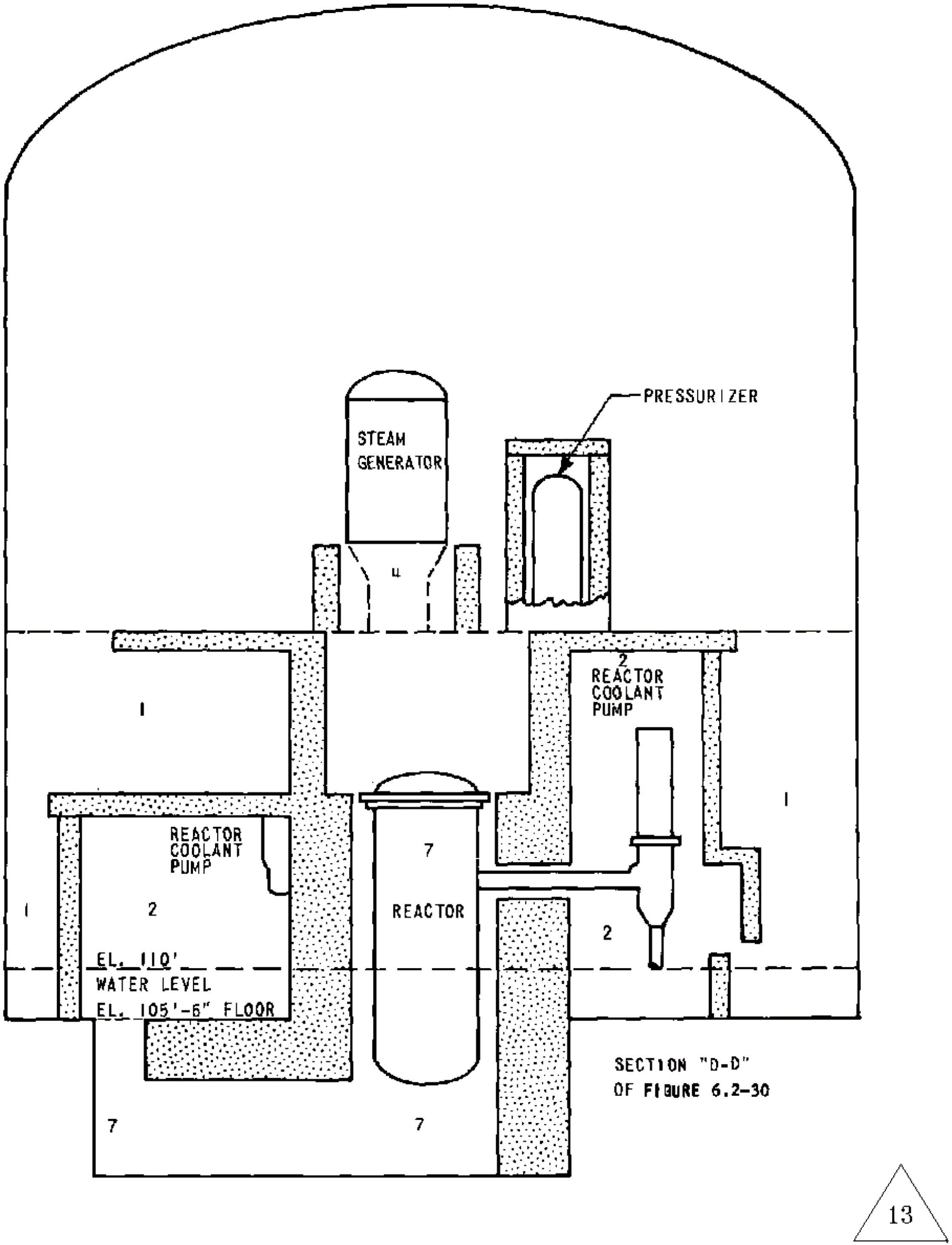
ELECTRIC HYDROGEN RECOMBINER
SCHEMATIC DIAGRAM (TYPICAL OF ONE RECOMBINER)

FIGURE 6.2-91



REV 21 5/08

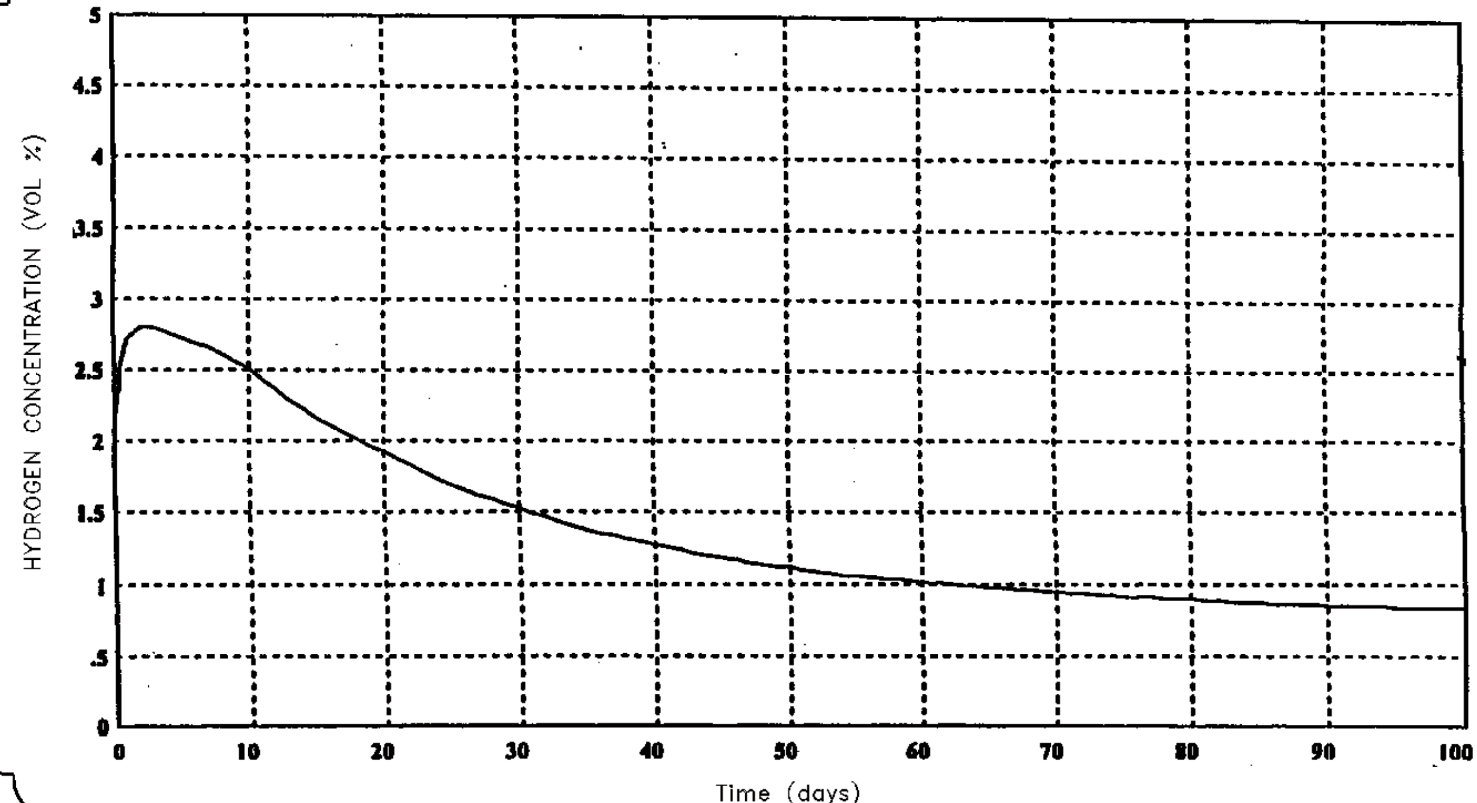
SOUTHERN COMPANY *Energy to Serve Your World®*	**JOSEPH M. FARLEY NUCLEAR PLANT UNIT 1 AND UNIT 2**	LOWER COMPARTMENT PLAN FIGURE 6.2-92



REV 21 5/08

SECTION OF LOWER REACTOR COMPARTMENT

FIGURE 6.2-93



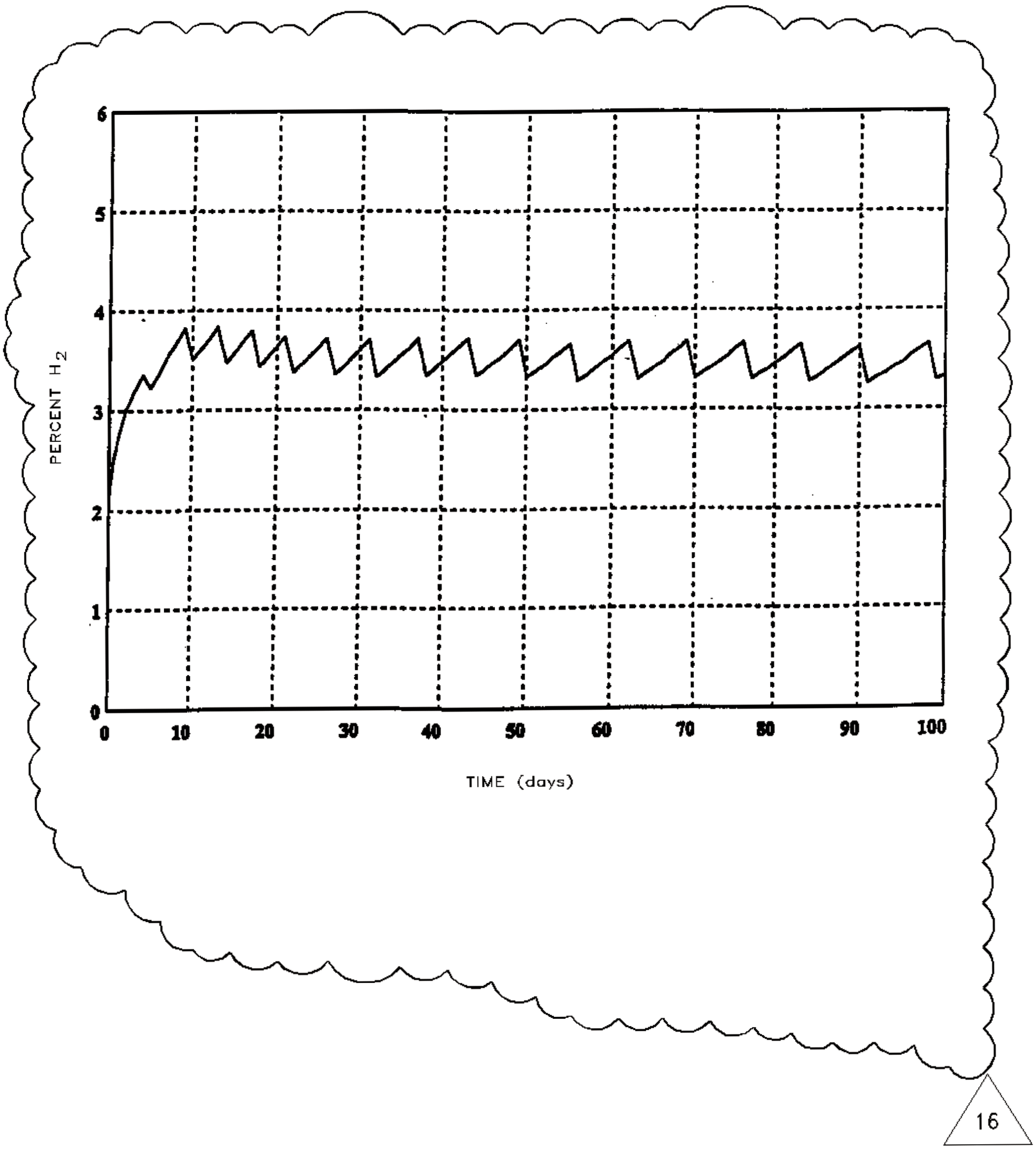
16

REV 21 5/08

JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

CONTAINMENT HYDROGEN CONCENTRATION
WITH ONE ELECTRIC RECOMBINER STARTED
ONE DAY AFTER A LOCA

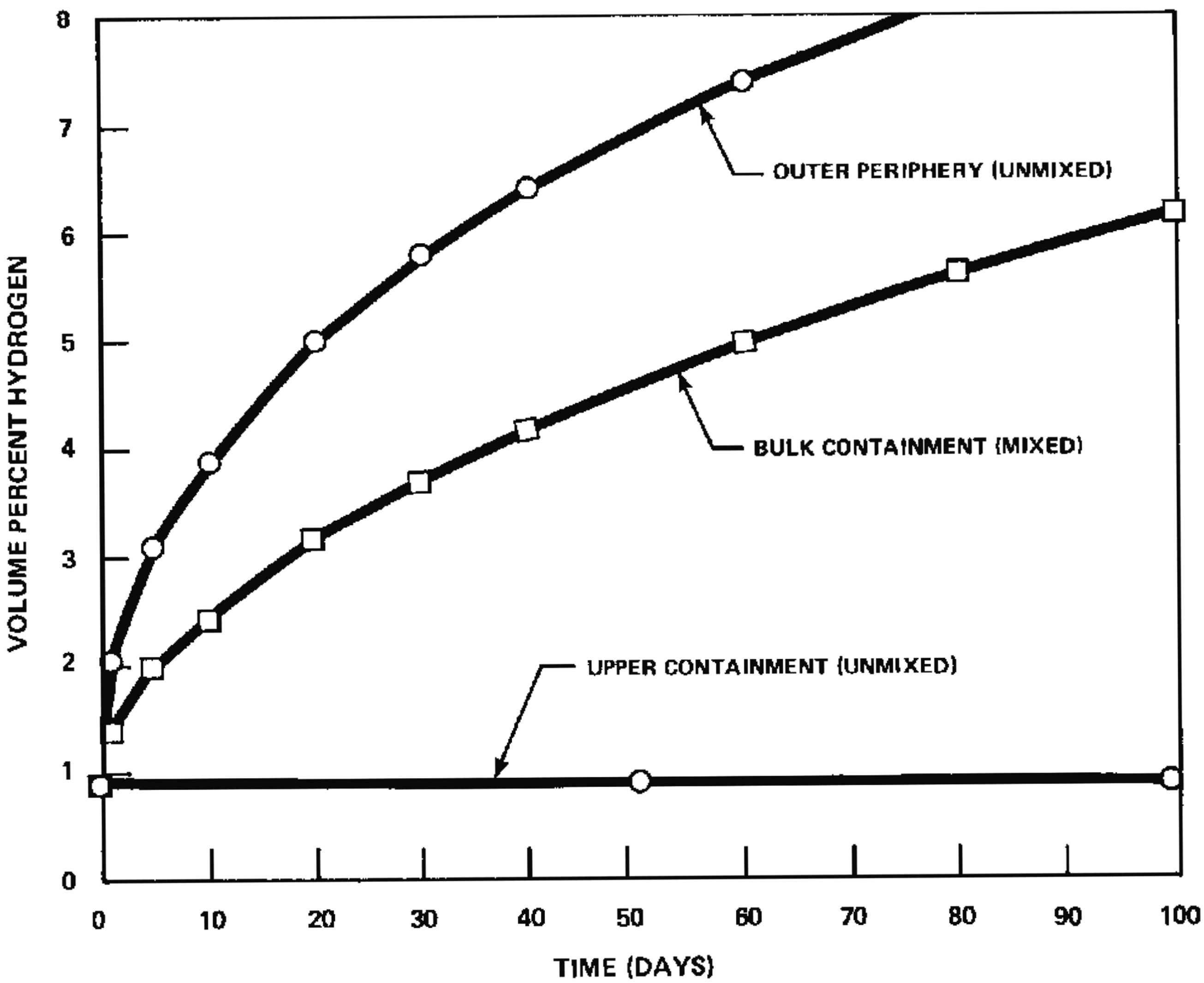
FIGURE 6.2-94



REV 21 5/08

HYDROGEN CONCENTRATION AS A FUNCTION OF TIME IN CONTAINMENT PURGE MODE

FIGURE 6.2-95



13

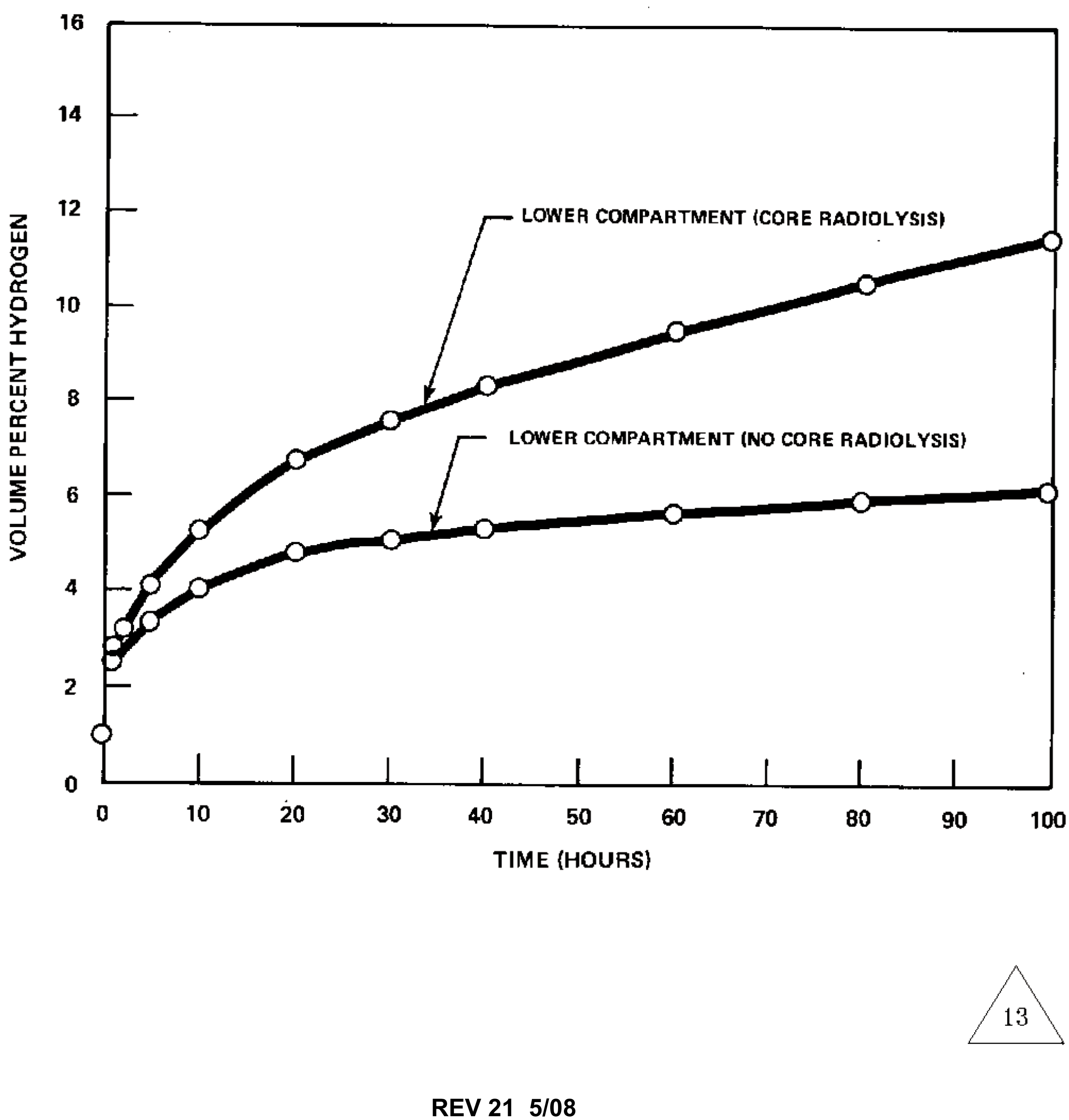
REV 21 5/08

SOUTHERN COMPANY
Energy to Serve Your World®

JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

VOLUME PERCENT HYDROGEN VS. TIME IN THE UPPER CONTAINMENT (UNMIXED), OUTER PERIPHERY (UNMIXED) AND BULK CONTAINMENT (MIXED)

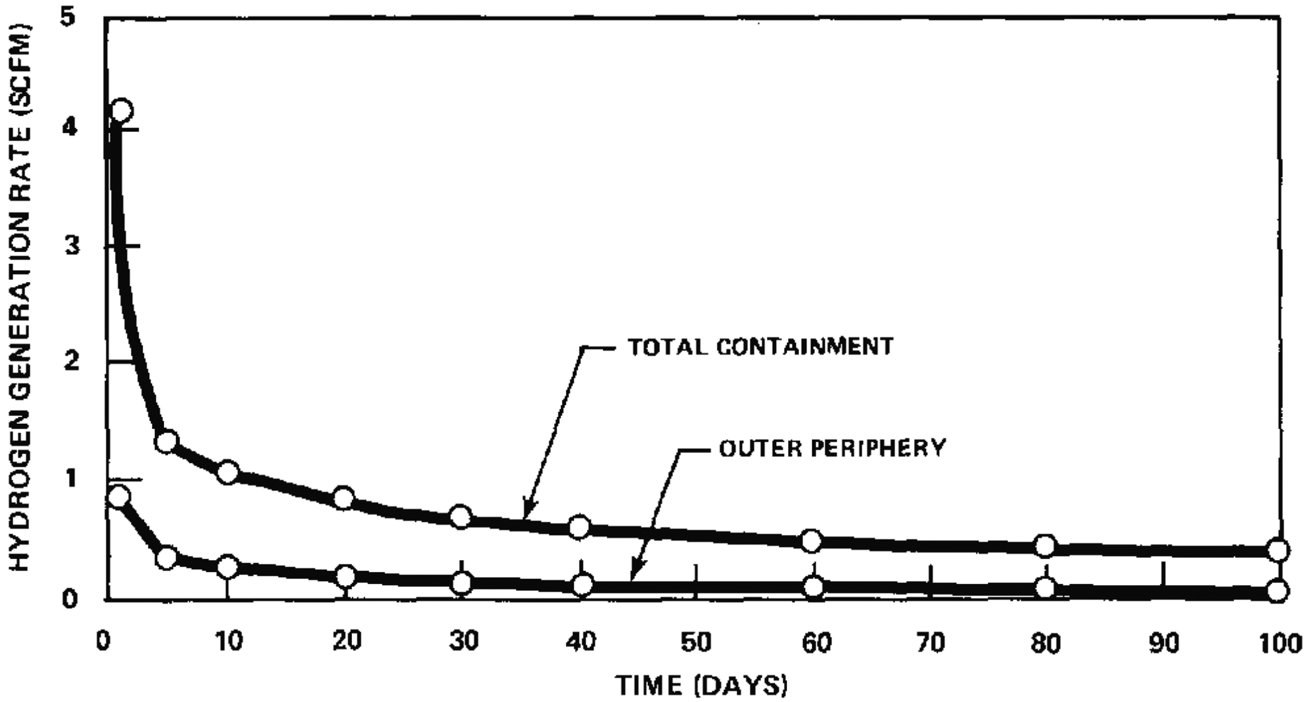
FIGURE 6.2-96



REV 21 5/08

VOLUME PERCENT HYDROGEN VS. TIME IN THE LOWER COMPARTMENT

FIGURE 6.2-97



13

REV 21 5/08

SOUTHERN COMPANY
Energy to Serve Your World®

JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

HYDROGEN GENERATION RATE VS. TIME IN
OUTER PERIPHERY AND OVERALL CONTAINMENT

FIGURE 6.2-98

6.3 EMERGENCY CORE COOLING SYSTEM

6.3.1 DESIGN BASES

The emergency core cooling system (ECCS) is discussed in detail in this section. For additional information on the ECCS, see the following sections:

A. Compliance with the acceptance criteria of 10 CFR 50.46 is discussed in subsection 15.4.1.

B. Components which are necessary following a postulated loss-of-coolant accident (LOCA), over the entire range of break sizes, are discussed in sections 15.3 and 15.4.

C. External forces and their effect on the operation of the ECCS are treated in sections 3.7 and 3.9.

D. Preoperational system testing is discussed in detail in chapter 14.

E. The initiation of the ECCS following a LOCA is discussed in section 7.3.

F. Instrumentation available to the operator to monitor conditions after a LOCA is found in section 7.5. Duration of time that these instruments need to be operable is also discussed in section 7.5.

G. Testing intervals are discussed in the Technical Specifications and the Technical Requirements Manual.

H. The containment sump description and ECCS recirculation intakes model test program are described in appendix 6D.

I. Details for the performance evaluation for the effects of debris on functions of the ECCS are provided in Appendix 6D.

6.3.1.1 Range of Coolant Ruptures and Leaks

The ECCS is designed to cool the reactor core as well as to provide additional shutdown capability following initiation of the following accident conditions:

A. Pipe breaks and spurious valve lifting 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. The analyses supporting the acceptance criteria for the consequences of RCS breaks are discussed in subsections 15.3.1, 15.4.1, and 15.2.12.

B. Rupture of a control rod drive mechanism causing a rod cluster control assembly (RCCA) ejection accident. The acceptance criteria for the consequences of a RCCA ejection accident are discussed in subsection 15.4.6.

C. Pipe breaks and spurious valve lifting in the steam system, up to and including the instantaneous circumferential rupture of the largest pipe in the steam system. The acceptance criteria for the consequences of steam system breaks are discussed in subsections 15.2.13, 15.3.2, and 15.4.2.

D. A steam generator tube rupture. The acceptance criteria for the consequences of a steam generator tube rupture are discussed in subsection 15.4.3.

6.3.1.2 Fission Product Decay Heat

The primary function of the ECCS following a LOCA is to remove the stored and fission product decay heat from the reactor core in such a way that fuel rod damage, to the extent that it would impair effective cooling of the core, is prevented. A discussion of the stored and fission product decay heat removal is found in section 15.1 and subsection 15.4.1.

6.3.1.3 Reactivity Required for Cold Shutdown

The ECCS provides shutdown capability for the accidents listed above by means of shutdown chemical (boron) injection. The most critical accident for shutdown capability is the steam line break; and for this accident the ECCS meets the criteria defined in subsection 15.4.2. Reactivity required for cold shutdown is also discussed in section 4.3.

6.3.1.4 Capability to Meet Functional Requirements

In order to ensure that the ECCS will perform its desired function during the accidents listed in paragraph 6.3.1.l, 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 section 3.1 and paragraph 6.3.2.11.

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) and with offsite electrical power system operation for any of the abnormal occurrences of paragraph 6.3.1.1, assuming a single failure as defined above.

The ECCS is designed to perform its function of ensuring core cooling and providing shutdown capability following an accident under simultaneous safe shutdown earthquake loading. The seismic requirements, missile protection, and protection from dynamic effects are discussed in sections 3.5, 3.6, and 3.7.

6.3.2 SYSTEM DESIGN

6.3.2.1 Schematic Piping and Instrumentation Diagrams

The flow diagram of the ECCS is shown in drawings D-175038, sheet 1; D-205038, sheet 1; D-175038, sheet 2; D-205038, sheet 2; D-175038, sheet 3; and D-205038, sheet 3. The residual heat removal (RHR) system, which also acts as part of the ECCS, is shown in drawings D-175041 and D-205041. Portions of the chemical and volume control system (CVCS), which also act as part of the ECCS, are shown in figure 9.3-1 and drawings D-175005; D-175004, sheet 1; D-175004, sheet 2; D-205004, sheet 1; D-205004, sheet 2; D-175039, sheet 1; D-205039, sheet 1; D-175039, sheet 2; and D-205039, sheet 2.

6.3.2.2 System Components

6.3.2.2.1 Pumps

A. Residual Heat Removal Pumps

Residual heat removal pumps are utilized to deliver water from the refueling water storage tank to the reactor coolant system should the RCS pressure fall below their shutoff head during an accident. Each RHR pump is a single-stage, vertical position, centrifugal pump. It has a coupled motor pump shaft, driven by an induction motor. The unit has a self-contained mechanical seal which is cooled by pump discharge fluid circulating through a small heat exchanger which is cooled by component cooling water.

A minimum flow bypass line is provided for the pumps to recirculate through the residual heat exchangers and return the cooled fluid to the pump suction should these pumps be started with RCS pressure above their shutoff head. Once flow sufficient for pump protection is established to the RCS, the bypass line is automatically closed. This line prevents deadheading the pumps and permits pump testing during normal operation. Performance curves for these pumps are given on figures 6.3-1 and 6.3-3.

These performance curves reflect the change in pump impeller diameters (increased from 17 1/4 in. to18 1/8 in.), which was needed to provide the required safety analysis injection flow.

Pump room coolers are used to maintain air temperature in the pump rooms at or below 104°F during normal operation. Refer to table 9.4-6A for post-DBA room temperatures. RHR pump room coolers are discussed in paragraph 9.4.2.1.9. The RHR pumps are also discussed in subsection 5.5.7.

A gas accumulation monitoring and trending process for the Farley Unit 1 and 2 ECCS (CVCS and RHR) and containment spray systems has been established to meet the requirements of NRC Generic Letter 2008-01.

B. Centrifugal Charging Pumps

These pumps deliver water from the refueling water storage tank (RWST) 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 cooled by component cooling water. Performance curves for these pumps are given on figure 6.3-2.

A minimum flow bypass line is provided on each pump discharge to recirculate flow to the pump suction after cooling in the seal water heat exchanger during normal operation. The minimum flow bypass line contains two valves in series which can be closed manually after receiving the SIS. During normal plant operation, at least one centrifugal charging pump is in use. The charging pumps are also discussed in subsection 9.3.4.

Pump room coolers are used to maintain air temperature in the pump rooms at or below 104°F during normal operation. Refer to table 9.4-6A for post-DBA room temperatures. Charging/high-head safety injection pump room coolers are discussed in paragraph 9.4.2.1.9.

A gas accumulation monitoring and trending process for the Farley Unit 1 and 2 ECCS (CVCS and RHR) and containment spray systems has been established to meet the requirements of NRC Generic Letter 2008-01.

6.3.2.2.4 Residual Heat Exchangers

The residual heat exchangers are conventional shell and U-tube type units. During normal operation of the RHR system, reactor coolant flows through the tube side while component cooling water flows through the shell side. During emergency core cooling recirculation operation, water from the containment sump flows through the tube side. The tubes are seal welded to the tube sheet.

A further discussion of the residual heat exchangers is found in subsection 5.5.7.

6.3.2.2.5 Valves

Design parameters for all types of valves used in the ECCS are given in table 6.3-1.

Design features employed to minimize valve leakage include:

A. Where possible, packless valves are used.

B. Other valves which are normally open, except check valves and those which perform a control function, are provided with backseats to limit stem leakage.

C. Relief valves are enclosed, i.e., they are provided with a closed bonnet, and discharge to a closed system.

D. Manual and motor-operated gate and globe valves (2-1/2 in. and above) exposed to recirculation flow have double packed stuffing boxes and stem leakoff connections to the waste processing system (WPS) consistent with applicable specifications and guidance from Regulatory Guide 8.8, Revision 3.

E. All modulating and three-way control valves in normal radioactive service have leakoff connections to the WPS, or the boron recycle holdup tanks.

F. Ball valve stems are of the backseated type (“blowout proof”).

Motor-Operated Valves

The seating design of motor-operated gate valves can be a flexible or solid wedge design. Gate valve design releases 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 disc is 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.

Where a gasket is employed for the body-to-bonnet joint, it is either a fully trapped, controlled compression, spiral wound asbestos (or equivalent) gasket with provisions for seal welding, or it is of the pressure seal design with provisions for seal welding. Some of the valve stuffing boxes are designed with a lantern ring leakoff connection, with a set of packing below and above the lantern ring.

Some motor operators incorporate a "hammer blow" feature that allows the motor to attain its operational speed prior to impact. Valves that must function against system pressure are designed so that they will function with a pressure differential equal to full system pressure across the valve disc.

Manual Globes, Gates 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 connection.

Globe valves, "T" and "Y" style, are full ported with outside screw and yoke construction.

Check valves are springloaded lift piston types for sizes 2 in. and smaller, swing type for size 2-1/2 in., and tilting disc type for size 4 in. and larger. Stainless steel check valves have no penetration welds other than the inlet, outlet, and bonnet. The check hinge is serviced through the bonnet.

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.

Diaphragm Valves

The diaphragm valves are of the Saunders patent type which used the diaphragm member for shutoff with even-weir bodies. These valves are used in service not exceeding 200°F and 200 psig design temperature and pressure.

Accumulator Check Valves (Swing-disc)

The accumulator check valve is designed with a low pressure drop configuration and all operating parts contained within the body.

Design considerations and analyses, to ensure that leakage across the check valves located in each accumulator injection line will not impair accumulator availability, are as follows:

A. 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 function 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 leakage.

B. When the RCS is being pressurized during the normal plant heatup operation, the check valves are tested for leakage as soon as there is a stable differential pressure of about 100 psi or more across the valve. 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 discharge line motor-operated isolation valves are opened and the RCS pressure increase is continued. There should be no increase in leakage from this point on, since increasing reactor coolant pressure increases the seating force and decreases the probability of leakage.

C. Experience derived from the check valves employed in the safety injection systems indicates 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 Ginna plant and more recently built plants where the usage of check valves is identical to this application.

D. The accumulators can accept some inleakage from the RCS without affecting availability. Inleakage would require, however, that the accumulator water volume and boron concentration be adjusted accordingly to stay within Technical Specification requirements.

Relief Valves

Relief valves are installed in various sections of the ECCS to protect lines that have a lower design pressure than the RCS. The valve stem and spring adjustment assembly are isolated from the system fluids by a bellows seal between the valve disc and spindle. The closed bonnet provides an additional barrier for enclosure of the relief valves. The accumulator relief valves are sized to pass nitrogen gas at a rate in excess of the accumulator gas fill line delivery rate. The relief valves will also pass water in excess of the expected accumulator inleakage rate; however, this is not considered to be necessary, because the time required to fill the gas space gives the operator ample opportunity to correct the situation. Table 6.3-2 lists the system's relief valves with their capacities and setpoints.

Butterfly Valves

Each main RHR line has an air-operated butterfly valve downstream of the heat exchanger, which is normally open and is designed to fail in 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 RHR system operation to adjust flow through the heat exchangers.

6.3.2.2.6 Piping

All piping joints are welded except for the pump, butterfly valve, post-LOCA strainer to pump suction interface flanged, and Bell-mouth connections.

Weld connections for pipes sized 2 1/2 in. and larger are butt welded. Reducing tees are used where the branch size exceeds one half of the header size. Branch connections of sizes that are equal to or less than one half of the header size conform to the ANSI code. Branch connections 1/2 in. through 2 in. are attached to the header by means of full penetration welds, using preengineered integrally reinforced branch connections.

Minimum piping and fitting wall thicknesses, as determined by ASME III Code formula, are increased to account for the manufacturer's permissible tolerance of -12 1/2 percent on the nominal wall and an appropriate allowance for wall thinning on the external radius during any pipe bending operations in the shop fabrication of the subassemblies.

The initial ECCS analysis results indicated that the originally proposed low-head safety injection piping system design was such that flow imbalances within the system would occur following postulated breaks at certain locations. These flow imbalances would result in injection flowrates lower than the design values following a LOCA. As a result of these analyses, the delivery capacity of this system was increased by a combination of changing pipe size at selected locations and modifying the low-head safety injection pump design. The portion of the low-head safety injection system piping located inside the containment was revised by changing pipe sizes to provide acceptable flow balancing to the three primary coolant loops. Following these design revisions the calculated pressure drops through the three individual injection lines inside containment, and through the header cross tie between these three lines, were evaluated in conjunction with the results of the low-head safety injection pump modifications. These analyses verified that the revised design results in system flowrates under normal and post-LOCA conditions will meet or exceed the values used in the safety analyses.

The revised piping arrangement is shown in drawings D-175038, sheet 2 and D-205038, sheet 2, locations E-1, E-2, E-3, F-1, F-2, and F-3.

6.3.2.2.7 System Operation

The operation of the ECCS can be divided into two distinct modes:

A. The injection mode in which any reactivity increase following the postulated accidents is terminated, initial cooling of the core is accomplished, and coolant lost from the primary system in the case of a LOCA is replenished, and

B. The recirculation mode in which long-term core cooling is provided during the accident recovery period.

A discussion of these modes follows.

A. Injection Mode

The injection mode of emergency core cooling is initiated by the SIS "S" signal. This signal is actuated by any of the following:

- Low pressurizer pressure.
- High containment pressure.
- High steam line differential pressure.
- Low steam line pressure.
- Manual actuation.

Operation of the ECCS during the injection mode is completely automatic. The SIS automatically initiates the following actions:

1. Starts the diesel generators and, if all other sources of power are lost, aligns them to the engineered safety features buses.

2. Starts the charging pumps and the RHR 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 valves in the charging pumps suction line from the RWST.

    c. Closing the valves in the charging pump normal suction line from the volume control tank.

d. Opening the injection line isolation valves.

The injection mode continues until the low level is reached in the RWST, at which time the operator alters system alignment to the cold leg recirculation mode.

B. Recirculation Mode

After the injection operation, water collected in the containment sump is cooled and returned to the RCS by the low-head/high-head recirculation flow paths. The changeover from the injection mode to recirculation mode is performed manually by operator action from the main control room. A level signal is provided as an automatic backup to open the containment sump valves when the RWST level channels indicate the low-low level setpoint has been reached. The charging pumps would continue to take suction from the RWST following the shutdown of the two RHR pumps and deliver to the RCS until manual operator action is taken to align them in series with the RHR pumps. The automatic backup to the manual switchover feature represents the NRC acceptance criteria to preclude loss of RHR cooling to the core when proceeding from the injection to recirculation mode of ECCS operation.

The low-level signal is alarmed to inform the operator to initiate the manual action required to realign safeguards pumps for the recirculation mode. The switchover sequence performed by the operator is outlined in table 6.3-4. As shown in table 6.3-4, operators will establish one charging pump in each train. Therefore, for RWST draindown purposes, this sequence is predicated on the operability of two charging pumps. Should any pump not be operable, suction and discharge valve alignment may be altered. Plant procedures dictate the proper operator actions for all postulated circumstances. When the switchover sequence is complete, the RHR pumps would take suction from the containment sump and deliver borated water directly to the RCS cold legs. A portion of each one of the RHR pump's discharge flow would be used to provide suction to two operating charging pumps which would also deliver directly to the RCS cold legs.

Assuming maximum safeguards (two RHR pumps, two charging pumps, two spray pumps), the low-level setpoint is reached approximately 20 min after initiation of safety injection. At this point, the RHR pumps are manually tripped when indicated level reaches the RWST low level. To evaluate whether there is sufficient time to perform the required manual actions, an analysis has been performed to determine the amount of time available for the case of maximum safeguards flow from the RWST.

Maximum safeguards flow from the RWST was calculated for Unit 1 and Unit 2 for each step of the switchover from injection to recirculation. The maximum safeguards flow at the start of the sequence is 13,400 gal/min (with both trains of ECCS and containment spray), and the maximum at the end of the sequence is 6,700 gal/min (containment spray only). In addition, the longest design time required to open or close the largest valve in the ECCS is 17 s. However, a 20-s time interval for each valve cycle was assumed.

Based upon these maximum flowrates, conservative operator action times and valve cycle times, the ECCS switchover is completed approximately 10 min after the RWST low-level switchover setpoint is reached with a water volume left in the RWST sufficient for containment spray pump switchover (see table 6.3-4).

The RCS can be supplied simultaneously from the RHR pumps and from a portion of the discharge from the RHR pumps which is directed to the charging pump suction and then to the RCS. The latter mode of operation assures flow in the event the depressurization proceeds more slowly so that the RCS pressure is still in excess of the shutoff head of the RHR pumps at the onset of recirculation.

The initial recirculation mode will provide recirculation flow to the cold leg of the RCS. Approximately 7.5 h after the initiation of the LOCA, simultaneous cold- and hot-leg recirculation will be initiated to assure termination of boiling. The simultaneous cold- and hot-leg recirculation may consist of high-head pump flow to the cold legs and low-head pump flow to the hot leg or high-head pump flow to the hot legs and low-head pump flow to the cold legs. A manual Train A/Train B power transfer switch is provided to ensure the simultaneous hot leg and cold leg recirculation phase of LOCA recovery can be achieved in the unlikely event that Train B power is lost.

Each train of the low-head safety injection (LHSI) and containment spray systems is provided with a motor-operated valve to isolate the pump suction from the RWST during post-LOCA recirculation. In the event that one of these valves fails to close during the transition from the injection to the recirculation phase of ECCS operation, a check valve is provided to prevent backflow of sump fluid into the RWST discharge header. Also, in the event that one of these motor-operated valves fails to close (steps 8 or 13, table 6.3-4), the charging pump in the same train as the failed valve would be stopped, and the train of equipment associated with the failed valve would not be operated. These motor-operated and check valves act to isolate the RWST from the respective pump suction only, and do not affect the flow paths to the other pumps taking suction from the RWST.

As an additional assurance that no potential leak path exists from the containment sump to the RWST during post-LOCA recirculation, the low-head safety injection or containment spray train associated with the postulated valve failure can be shut down following the failure. The redundant valves in the affected sump line would then be closed to provide an added measure of isolation between the containment sump and the RWST. These actions would not require shutdown or reduction in capacity of any portion of the remaining safety injection or spray systems.

Each end of the suction header for the high-head safety injection (HHSI) pumps is provided with a motor-operated valve to isolate this header from the RWST during post-LOCA recirculation. In the event that one of these two parallel valves fails to close during the transition from the injection to the recirculation phase of ECCS operation, a check valve is provided to prevent backflow of sump fluid into the RWST discharge header. These motor-operated and check valves act to isolate the RWST from the HHSI pump suctions only, and do not affect the flow paths to the other pumps taking suction from the RWST.

As an additional assurance that no potential leak path exists from the containment sump to the RWST during post-LOCA recirculation, the HHSI pump taking suction from the end of the suction header associated with the postulated valve failure could be shut down following the failure. The ends of the HHSI suction and discharge headers connected to this shutdown pump could then be isolated, using the redundant motor-operated valves provided in each header for this purpose. In like manner, the supply piping from the SIS residual heat exchanger to the isolated portion of the HHSI pump suction header and the discharge piping to the containment from the isolated portion of the HHSI pumps discharge header would be isolated using the motor operated valves provided. This will provide an added measure of isolation between the containment sump and the RWST. These actions would not require shutdown or reduction in capacity of any portion of the remaining SISs.

6.3.2.2.8 Break Spectrum Coverage

For large pipe ruptures, the RCS would be depressurized and voided of coolant rapidly, and a high flow rate of emergency coolant is required to cover the exposed fuel rods quickly and limit possible core damage. This high flow is provided by the passive accumulators, followed by the charging pumps, and then the RHR pumps discharging into the cold legs of the RCS.

Emergency cooling is provided for small ruptures primarily by high-head injection.[a] Small ruptures are those, with an equivalent diameter of 9.6 in. (0.5 ft^2 area) or less, 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, the charging pumps take suction from the RWST. A further discussion of ECCS performance over the entire range of RCS break sizes is contained in subsections 15.3.1 and 15.4.1.

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:

---

a. The charging pumps are commonly referred to as "high-head pumps" and the RHR pumps as "low-head pumps." Likewise, the term "high-head injection" is used to denote charging pump injection and "low-head injection" refers to RHR pump injection.

A. All parts of components in contact with borated water are fabricated of, or clad with, austenitic stainless steel or equivalent corrosion resistant material.

B. All parts of components in contact (internal) with sump solution during recirculation are fabricated of austenitic stainless steel or equivalent corrosion resistant material except for a short carbon steel spool piece at the transition of the pump suction line inside containment to the post-LOCA strainer assemblies (U2).

C. Valve seating surfaces are hard faced with Stellite number 6 or equivalent to prevent galling and to reduce wear.

D. Valve stem materials are selected for their corrosion resistance, high tensile properties, and resistance to surface scoring by the packing.

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. Refer to appendix 6A for a further discussion of materials compatibility.

Environmental testing of ECCS equipment inside the containment, which is required to operate following a LOCA, is discussed in reference 1. The chemistry environment used in the test program was obtained by using a spray solution of 1.5 weight percent boric acid in water and adjusting the pH to a value of 9.25 with sodium hydroxide. This solution is typical of that expected in the postaccident environment. The results of the test program indicate that the safety feature equipment will operate satisfactorily during and following exposure to the combined containment postaccident environments of temperature, pressure, chemistry and radiation.

6.3.2.5 <u>Design Pressures and Temperatures</u>

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 during either normal plant operation, or during operation of the ECCS. 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 operating conditions, the fundamental assurance of structural integrity of the ECCS components is maintained.

6.3.2.6 <u>Coolant Quantity</u>

The minimum storage volume for the RWST and the nominal storage volume for the accumulators is given in table 6.3-6. The minimum storage volume for the condensate storage tank is given in subsection 9.2.6. The minimum and nominal storage volume in the RWSTs and accumulators is sufficient to ensure that, after a RCS break, sufficient water is injected and available within containment to permit recirculation cooling flow to the core, and to meet the net positive suction head requirements of the RHR pumps. A further discussion of coolant requirements is contained in sections 15.3 and 15.4.

6.3.2.7 Pump Characteristics

Performance curves for the RHR pumps are given in figure 6.3-1. Performance curves for the centrifugal charging pumps are given in figure 6.3-2. The power requirements for these pumps are given in section 8.3.

6.3.2.8 Heat Exchanger Characteristics

The characteristics of the residual heat exchangers are given in subsection 5.5.7.

6.3.2.9 ECCS Flow Diagrams

Flow diagrams of the ECCS are given as drawings D-175038, sheet 1; D-205038, sheet 1; D-175038, sheet 2; D-205038, sheet 2; D-175038, sheet 3; and D-205038, sheet 3. See paragraph 6.3.2.1.

6.3.2.10 Relief Valves

The system's relief valves and their capacities and settings are given in table 6.3-2.

6.3.2.11 System Reliability

A. Definitions of Terms

Definitions of terms used in this section are located in subsection 3.1.1.

B. Active Failure Criteria

The ECCS is designed to accept a single failure following the incident without loss of its protective function. The system design will 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 LOCA, 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.

C. Passive Failure Criteria

The following philosophy provides for necessary redundancy in component and system arrangement to meet the intent of the NRC General Design Criterion 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 prior failure during the short term.

Redundancy of Flow Paths and Components for Long-Term Emergency Core Cooling

In design of the ECCS, Westinghouse utilizes the following criteria:

A. During the long-term cooling period following a loss-of-coolant, the emergency core cooling flow paths are separable into two subsystems, either of which can provide minimum core cooling functions and return spilled water from the floor of the containment back to the RCS.

B. Either of the two subsystems can be isolated and removed from service in the event of a leak in that subsystem outside the containment. Maximum potential leakage from components is given in table 6.3-8. For leakage used in evaluating dose consequences from a LOCA, see table 15.4-14.

C. Should one of these two subsystems be isolated in this long-term period, the other subsystem remains operable.

D. Provisions are also made in the design to detect leakage from components outside the containment, to collect this leakage, and to provide for maintenance of the affected equipment.

Thus, for the long-term emergency core cooling function, adequate core cooling capacity exists with one flow path removed from service whether isolated because of a leak, because of blocking of one flow path, or because failure of a line inside the containment results in a spill of the delivery of one subsystem.

Subsequent Leakage From Components In Safeguards Systems

With respect to piping and mechanical equipment outside the containment, considering the provisions for visual inspection and leak detection, leaks will be detected before they propagate to major proportions. A review of the equipment in the system indicates that the largest sudden leak potential would be the sudden failure of a pump shaft seal. Evaluation of leak rate assuming only the presence of a seal retention ring around the pump shaft showed flows less than 50 gal/min would result. Piping leaks, valve packing leaks, or flange gasket leaks have been of a nature to build up slowly with time and are considered less severe than the pump seal failure.

Larger leaks in the ECCS are prevented by the following:

A. The piping is classified in accordance with ANS Safety Class 2a and receives the ASME Class 2 quality assurance program associated with this safety class.

B. The piping, equipment, and supports are designed to seismic Category I classification permitting no loss of function for the safe shutdown earthquake.

C. The system piping is located within a controlled area on the plant site.

D. The piping system receives periodic pressure tests and is accessible for periodic visual inspection.

E. The piping is austenitic stainless steel which, due to its ductility, can withstand severe distortion without failure.

Based on this review, the design of the auxiliary building and related equipment is based upon handling of leaks up to a maximum of 50 gal/min. Means are also provided to detect and isolate such leaks in the emergency core cooling flow path within 30 minutes.

With these design ground rules, continued function of the ECCS will meet minimum core cooling requirements, and offsite doses resulting from the leak will be within 10 CFR 100 limits.

A single-passive failure analysis is presented in table 6.3-9. It 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 establish the alternate flow path also isolates the component which failed.

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. Thermal stresses on the RCS are discussed in section 5.2.

ECCS Piping Failures

The rupture of the portion of an injection line from the last check valve to the connection of the line to the RCS can cause not only a loss of coolant but impair the injection as well. To reduce the probability of an emergency core cooling line rupture causing a LOCA, the check valves which isolate the ECCS from the RCS are installed immediately adjacent to the reactor coolant piping.

For a small break, the reactor pressure maintains a relatively uniform back pressure in all injection lines so that a significant flow imbalance does not occur. A rupture in an accumulator injection line is accounted for in the analyses by assuming that for cold leg breaks the entire flowrate through this line is discharged from the break.

6.3.2.13 Provisions for Performance Testing

The provisions incorporated to facilitate performance testing of components are discussed in subsection 6.3.4.

6.3.2.14 Net Positive Suction Head (NPSH)

The ECCS is designed so that adequate net positive suction head is provided to system pumps. In addition to considering static head, suction line pressure drop, and debris head loss, the calculation of available NPSH in the recirculation mode assumes that the containment pressure is equal to the TS minimum operating containment pressure prior to the accident (-1.5 psig) for sump temperatures below 206.6 ºF (saturation temperature at the minimum TS containment pressure prior to the accident, -1.5 psig, or 13.2 psia). Above 206.6 ºF, the containment pressure is assumed to be equal to the vapor pressure of the liquid in the sump. These assumptions assure that the actual available NPSH is always greater than the pump required NPSH.

Testing and analyses have been performed to assure that flow-reducing or air-entraining vortices, which could limit available NPSH, do not occur at the containment sumps.

The calculation of available NPSH is as follows:

$$(\text{Net positive suction head})_{\text{available}} = (h)_{\text{containment pressure}} - (h)_{\text{vapor pressure}} + (h)_{\text{static head}} - (h)_{\text{loss}}$$

The minimum water level available at the containment sump during recirculation following the LBLOCA event which generates bounding debris head losses is used in calculating the static head.

Adequate NPSH is shown to be available for all pumps as follows:

A. RHR Pumps

The NPSH of the RHR pumps is evaluated for normal plant shutdown operation, and for both the injection and recirculation modes of operation for the design basis accident. Recirculation operation gives the limiting NPSH requirement. The NPSH available during recirculation is determined from the static head (containment sump water level relative to the pump inlet centerline elevation), the cumulative pressure drop across the sump strainer debris bed and in the suction piping from the sump to the pumps, and the difference in the assumed containment pressure and the vapor pressure of the sump liquid. The NPSH evaluation is based on all pumps operating at the maximum design flowrates. The NPSH required for the RHR pumps is 18 ft at 4500 gal/min for Unit 1 and Unit 2.

B. Centrifugal Charging Pumps

The NPSH for the centrifugal charging pump is calculated 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 NPSH available. The NPSH available is determined from the elevation head and vapor pressure of the water in the RWST, which is at atmospheric pressure, and the pressure drop in the suction piping from the tank to the pumps. At the end of the injection mode when suction from the RWST is terminated (low RWST level), adequate NPSH is supplied from the containment sump by the booster action of the low-head pumps. The NPSH evaluation for each pump is based on the pump operating at maximum system flowrates. The recommended NPSH required for the centrifugal charging pumps is 120 ft at flows between 705 and 715 gal/min.

Testing has determined that strainer head loss (including debris) decreases as sump temperature increases. The sump fluid is assumed saturated for sump temperatures above the saturation temperature at the TS minimum containment pressure prior to the accident (-1.5 psig). The vapor pressure of the sump inventory decreases significantly as the sump inventory cools below 206.6 ºF, while the containment pressure remains constant resulting in increased available NPSH. Therefore, the case which results in the least margin between available and required NPSH for Farley occurs at a sump temperature of approximately 212 ºF. There is a negligible difference in NPSH margin at sump temperatures ranging from 206.6 ºF to 212 ºF. Increased sump temperatures above 212 ºF will result in slightly greater NPSH margins based on the strainer head loss testing and constant containment pressure. Strainer head loss is reduced due to the decreased viscosity of water as temperature increases. Above 206.6 ºF NPSH margin will increase due to increased containment pressure and decreased water viscosity.

For RHR pumps at 212 ºF (see Notes a and b):

	Unit 1	Unit 2
Maximum runout flow	4500 gal/min	4500 gal/min
Available NPSH	20.7 ft (Alpha)	19.2 ft (Alpha)
	19.6 ft (Bravo)	19.3 ft (Bravo)
Required NPSH	18.0 ft	18.0 ft
Static head	28.4 ft	28.4 ft
Head losses (Note c)	7.7 ft (Alpha)	9.2 ft (Alpha)
	8.8 ft (Bravo)	9.1 ft (Bravo)

For Containment Spray Pumps at 212 ºF:

Maximum runout flow	3400 gal/min	3400 gal/min
Available NPSH	20.8 ft (Alpha)	20.7 ft (Alpha)
	21.2 ft (Bravo)	19.1 ft (Bravo)
Required NPSH	18.0 ft	18.0 ft
Static head	29.6 ft	29.6 ft
Head losses (Note c)	8.8 ft (Alpha)	9.0 ft (Alpha)
	8.4 ft (Bravo)	10.6 ft (Bravo)

Notes:

a. The NPSH available for the RHR pumps has been determined utilizing simulated recirculation mode plant test data for the sump entrance losses and plant test data for the RHR pump suction piping from the sump friction losses.

b. Data in this table are based upon RHR pump impeller vane diameter of 18.125 in.

c. The head losses include intake, velocity head, elbow, and debris losses.

Testing has determined that head loss due to the accumulation of chemical precipitants on the sump screen (i.e., chemical effects) is negligible for temperatures > 140 ºF. The case which results in the lowest NPSH margin between available and required NPSH for temperatures at which head loss due to chemical effects could begin to have an impact is at 140 ºF.

For RHR pumps at 140 ºF (see Notes a, b, and c):

Maximum runout flow	4500 gal/min	4500 gal/min
Available NPSH	43.8 ft (Alpha)	42.2 ft (Alpha)
	42.7 ft (Bravo)	42.4 ft (Bravo)
Required NPSH	18.0 ft	18.0 ft
Static head	28.4 ft	28.4 ft
Head losses (Note d)	8.8 ft (Alpha)	10.3 ft (Alpha)
	9.9 ft (Bravo)	10.2 ft (Bravo)

For Containment Spray pumps at 140 ºF (see Note c):

Maximum runout flow	3400 gal/min	3400 gal/min
Available NPSH	43.9 ft (Alpha)	43.7 ft (Alpha)
	39.2 ft (Bravo)	37.8 ft (Bravo)
Required NPSH	18.0 ft	18.0 ft
Static head	29.6 ft	29.6 ft
Head losses (Note d)	9.9 ft (Alpha)	10.1 ft (Alpha)
	14.6 ft (Bravo)	16.0 ft (Bravo)

Notes:

a. The NPSH available for the RHR pumps has been determined utilizing simulated recirculation mode plant test data for the sump debris and entrance losses and plant test data for the RHR pump suction piping from the sump friction losses.

b. Data are based upon RHR pump impeller vane diameter of 18.125 in.

c. When temperature is ≤ 206.6 ºF, the design basis assumes no increase in containment pressure from that present prior to the postulated loss-of-coolant accident which is the TS minimum allowable containment pressure of -1.5 psig.

d. The head losses include intake, velocity head, elbow, debris, and chemical effects losses.

6.3.2.15 Control of Motor-Operated Isolation Valves

The design of the control circuit for a motor-operated isolation valve in a line connecting an accumulator to the RCS provides protection against inadvertent closure of that valve. Although the valve is normally open, it receives the SIS and will open automatically upon receipt of this signal should the valve be closed. This SIS overrides any bypass feature that allows the valve to be closed for short times during normal operation for test purposes. Additionally, the Technical Specifications require that power is removed from each accumulator isolation valve operator when the RCS pressure is ≥ 2000 psig. A further discussion of the position indication for these valves is found in subsection 6.3.5. The valve will also open automatically when system pressure exceeds the SIS unblock pressure signal setpoint.

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 an SIS) have their positions indicated on the main control board via indicating lights at the valve handswitch. 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.

The motor-operated isolation valves located between the high-pressure RCS and the relatively low-pressure RHR system are discussed in subsections 5.5.7 and 7.6.2.

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. The only manual actions required to be taken by the operator are those necessary to realign the system for its cold-leg recirculation mode of operation, and, after approximately 7.5 h, its simultaneous cold- and hot-leg recirculation mode of operation.

6.3.2.18 Process Instrumentation

Process instrumentation available to the operator in the control room to assist in assessing post LOCA conditions is 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 paragraph 6.3.2.4). Coatings specified for use on the ECCS components (mainly, the accumulators) are designated to meet the requirements of ANSI 101.2; 1972, "Protective Coatings (Paints) For Light Water Nuclear Reactor Containment Facilities," as a minimum.

6.3.3 PERFORMANCE EVALUATION

6.3.3.1 Evaluation Model

The evaluation model used in the analyses of ECCS performance following a LOCA is discussed in detail in subsection 15.4.1, and is in accordance with the Interim Acceptance Criteria.

6.3.3.2 ECCS Performance

Using the evaluation model in subsection 15.4.1 and referred to in paragraph 6.3.3.1, analyses are performed to ensure that the limits on core behavior following a large RCS pipe rupture are met by the ECCS operating with minimum design equipment. The results of these analyses are presented as a series of figures in subsection 15.4.1.

6.3.3.3 Alternate Analysis Methods

In evaluating small RCS breaks, those less than 1.0 ft^2, an alternate evaluation model is used. The alternate evaluation model and the results of the analyses are presented in subsection 15.3.1. For breaks 3/8 of an inch or smaller, the charging system can maintain the pressurizer level at the RCS operating pressure, and ECCS actuation is not required.

6.3.3.3.1 Main Steam System Single-Active Failure

Analyses of reactor behavior following any single-active failure in the main steam system which results in an uncontrolled release of steam are included in section 15.2. The analyses assume that a single valve (largest of the safety, relief, or bypass valves) opens and fails to close, which results in an uncontrolled cooldown of the RCS.

Results indicate that if the incident is initiated at the hot shutdown condition, which results in the worst reactivity transient, there is no return to criticality. Thus, the ECCS provides adequate protection for this incident.

Steam Line Rupture

This accident is discussed in detail in sections 15.3 and 15.4. The limiting steam line rupture is a complete line severance. The results of the analysis in subsection 15.4.2 indicate that the design basis criteria are met. Thus, the ECCS adequately fulfills its shutdown reactivity addition function.

6.3.3.4 Fuel Rod Perforations

Discussions of peak clad temperature and metal water reactions appear in subsections 15.3.1 and 15.4.1. Analyses of the radiological consequences of a fission product release due to a rupture of a pipe in the RCS are presented in subsection 15.4.1.

6.3.3.5 Evaluation Model

Does not apply to the Farley Nuclear Plant.

6.3.3.6 Fuel Clad Effects

Does not apply to the Farley Nuclear Plant.

6.3.3.7 ECCS Performance

Does not apply to the Farley Nuclear Plant.

6.3.3.8 Peaking Factors

Does not apply to the Farley Nuclear Plant

6.3.3.9 Fuel Rod Perforations

Does not apply to the Farley Nuclear Plant.

6.3.3.10 Conformance with Interim Acceptance Criteria

Does not apply to the Farley Nuclear Plant.

6.3.3.11 Effects of ECCS Operation on the Core

The effects of ECCS operation on the reactor core are discussed in subsections 5.2.1 and 14.2.13 and sections 15.3 and 15.4.

6.3.3.12 Use of Dual Function Components

The ECCS contains components that have no other operating function as well as components that are shared with other systems. Components in each category are as follows:

A. Components of the ECCS which perform no other function are:

1. One accumulator for each loop which discharges borated water into each cold leg of the reactor coolant loop piping.

2. Associated piping, valves, and instrumentation.

B. Components which also have a normal operating function are as follows:

1. The RHR pumps and the residual heat exchangers: These components 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.

2. The centrifugal charging pumps: These pumps are normally aligned for charging service. As a part of the CVCS, the normal operation of these pumps is discussed in subsection 9.3.4.

3. The RWST: This tank is used to fill the refueling canal for refueling operations. However, during all other plant operating periods it is aligned to the suction of the RHR pumps. The charging pumps are aligned to the suction of the RWST upon receipt of the SIS.

An evaluation of all components required for operation of the ECCS demonstrates that either:

A. The component is not shared with other systems.

B. 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 SIS.

Table 6.3-10 indicates the alignment of components during normal operation, and the realignment required to perform the accident function.

6.3.3.13 <u>Dependence on Other Systems</u>

Other systems which operate in conjunction with the ECCS are as follows:

A. The component cooling system cools the residual heat exchangers during the recirculation mode of operation. It also supplies cooling water to the charging pumps and the RHR pumps during the injection and recirculation modes of operation.

B. The service water system provides cooling water to the component cooling heat exchangers and room coolers of the ECCS pump rooms.

C. The electrical systems provide normal and emergency power sources for the ECCS.

D. The engineered safety features actuation system generates the initiation signal for emergency core cooling.

E. The auxiliary feedwater system supplies feedwater to the steam generators.

Limiting Conditions for Maintenance During Operation

Maintenance on an active component will be permitted if the remaining components meet the minimum conditions for operation and the following conditions are also met:

A. The remaining equipment has been demonstrated to be in operable condition, ready to function just before the initiation of maintenance operating.

B. A suitable time limit is placed on the total time span of successful maintenance which returns the components to an operable condition, ready to function.

The design philosophy with respect to active components in the high-head/low-head injection system is to provide backup equipment so that maintenance is possible during operation without impairment of the safety function of the system. Due to the redundancy of trains and the diversity of subsystems, the inoperability of different ECCS components in different trains may not result in a loss of function for the ECCS. This allows increased flexibility in plant operations under circumstances when components in redundant subsystems may be inoperable; however, the ECCS remains capable of delivering 100% of the required flow equivalent. An evaluation was performed to determine the specific components required to be operable to ensure that 100% of the ECCS flow equivalent to a single operable ECCS train remains available. Routine servicing and maintenance of equipment of this type would generally be scheduled for periods of refueling and maintenance outages. The Technical Specifications give the exact periods of time during which any component of the ECCS may be out of service.

6.3.3.14 Lag Times

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 SIS. Response of the system is automatic, with appropriate allowances for delays in actuation of circuitry and active components. The active portions of the system are actuated by the SIS. 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. A further discussion of the starting sequence is given in subsection 8.3.1.

In the LOCA 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 except for the large break LOCA (subsection 15.4.1) where offsite power available has been determined to result in a more severe accident scenario. For smaller LOCAs, 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 LOCAs.

The time sequence of events for the accident analyses used to evaluate the ECCS performance is tabulated in sections 15.3 and 15.4.

6.3.3.15 Thermal Shock Considerations

Thermal shock considerations are discussed in section 5.2.

6.3.3.16 Limits on System Parameters

The analyses of sections 15.3 and 15.4 show that the design basis performance characteristic of the ECCS is adequate to meet the requirements for core cooling following a LOCA 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 and low decay heat in the coolant; therefore, an accident comparable in severity to accidents occurring at operating conditions is not possible and ECCS components are not required.

The specification of individual parameters as given in table 6.3-1 includes due consideration of allowances for margin over and above the required performance value (e.g., pump flow and tank capacity), 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 function performance.

6.3.4 TESTS AND INSPECTIONS

In order to demonstrate the readiness and operability of the ECCS, all of the components are subjected to periodic tests and inspections. Preoperational 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 all of the components.

In conformance with the NRC acceptance criteria in General Design Criterion 18, (1) electrical systems important to safety will be tested periodically for the operability and functional performance of their components, and (2) the ECCS integrated system test will be performed periodically to demonstrate the operability of the system as a whole and its supporting electrical systems.

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.

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 all ECCS valves initiated by the SIS (S), the phase A containment isolation signal (T), and the phase B containment isolation signal (P); the operability of all safeguard pump circuitry down through the pump breaker control circuits; and the proper operation of all 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 all of the major pumps, i.e., the charging pumps and the RHR 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 miniflow conditions and pump operation at full-flow conditions.

The predicted system resistance will be verified by measuring the flow in each piping branch, as each pump delivers from the RWST to the open reactor vessel, and adjustments made where necessary to assure that 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 minimum acceptable flows as determined for the analyses in sections 15.3 and 15.4.

The systems are accepted only after demonstration of proper actuation of all components and after demonstration of flow delivery of all components within design requirements. Other aspects of preoperational testing are discussed in appendix 3A under conformance to Regulatory Guide 1.79.

Periodic Component Testing

Routine periodic testing of the ECCS components and all necessary support systems at power is performed. Valves which operate after a LOCA are operated through a complete cycle and pumps are operated individually in this test on their miniflow lines, except that the charging pumps are tested by their normal charging function. 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 operation of the remote stop valve and the check valve in each accumulator tank discharge line may be tested by opening the remote test line valve 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.

Test lines are provided for periodic checks of the leakage of reactor coolant back through the accumulator discharge line check valves and to ascertain that these valves seat whenever the RCS pressure is raised. It is expected that this test will be routinely performed when the reactor is being returned to power after an outage and the reactor pressure is raised above the accumulator pressure. To implement the periodic component testing requirements, the Technical Specifications and Technical Requirements Manual requirements have been 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.

Design measures have been taken to ensure that the following testing can be performed:

A. Active components may be tested periodically for operability (e.g., pumps on miniflow, certain valves, etc.).

B. An integrated system actuation test[a] can be performed when the plant is cooled down.

C. An initial flow test of the full operational sequence can be performed.

D. Testing can be performed by aligning the RHR suction from the RCS loops.

The design features that ensure this test capability are:

1. Power sources are provided to permit individual actuation of each active component of the ECCS.

2. The RHR pumps are used every time the RHR system is put into operation. They can also be tested periodically when the plant is at power using the miniflow recirculation lines.

3. The centrifugal charging pumps are normally in use for charging service.

---

a. Details of the testing of the sensors and logic circuits associated with the generation of a SIS together with the application of this signal to the operation of each active component are given in sections 7.2 and 7.3.

4. Remote operated valves can be exercised during routine plant maintenance.

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 tank can be directed at any time through a test line to demonstrate operation of the accumulator motor-operated valves.

7. A flow indicator is provided in the RHR pump headers. Pressure instrumentation is also provided in these lines.

8. An integrated system test can be performed when the plant is cooled down. This test may or may 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 LOSP to the vital electrical buses).

9. Simulated ECCS recirc alignment may be obtained with the RHR suction aligned to the RCS loops.

10. The accumulator check valves can be tested by performing a low pressure accumulator blowdown test. The MOV in the accumulator discharge line is opened and the accumulator is allowed to discharge into the reactor vessel. The test can only be performed with the reactor defueled.

11. The branch line flow balance test may be conducted with the plant shut down using charging pumps to verify proper flow distribution in the high head SI flow paths.

6.3.5 INSTRUMENTATION REQUIREMENTS

Instrumentation and associated analog and logic channels employed for initiation of ECCS operation are discussed in section 7.3. This section describes the instrumentation employed for monitoring ECCS components during normal plant operation and also ECCS postaccident operation. All alarms are annunciated in the control room.

A. Temperature Indication

<u>Residual Heat Exchanger Outlet Temperature</u>

The fluid temperature at the outlet of each residual heat exchanger is recorded in the control room.

B. Pressure Indication

Injection Line Pressure

Injection line pressure is indicated in the control room. A high-pressure alarm is provided.

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.

Test Line Pressure

The test circuit available to check for proper seating of the check valves between the injection lines and the RCS includes the valves and piping installed on the leakage test line and may utilize the local pressure indicator, a calibrated temporary pressure indicator, the accumulator pressure indicators or other test methods described in the applicable ASME Codes as needed to conduct the appropriate tests.

RHR Pump Discharge Pressure

RHR pump discharge pressure for each pump is indicated in the control room. A high pressure alarm is actuated by each channel.

C. Flow Indication

Charging Pump Injection Flow

Injection flow into the reactor coolant loops is indicated in the control room.

RHR Pump Injection Flow

Flow through each RHR injection and recirculation header leading to the reactor coolant loops is indicated in the control room.

Test Line Flow

The test circuit available to check for proper seating of the check valves between the injection lines and the RCS includes the valves and piping installed on the leakage test line and may utilize the local flow indicator, the accumulator level indicators, a graduated poly bottle, or other test methods described in the applicable ASME Codes as needed to conduct the appropriate tests.

RHR Pump Minimum Flow

A flowmeter installed in each RHR pump discharge header provides control for the valve located in the pump minimum flow line.

D. Level Indication

RWST Level

Two water level indicator channels, which indicate in the control room, are provided for the RWST. Each channel alarms on high, Technical Specification minimum, lo, and lo-lo water levels, and is indicated on the main control board.

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.

Containment Sump Water Level

Two containment sump water level indicator channels are provided. Both indicate in the control room.

E. Valve Position Indication

Accumulator Isolation Valve Position Indication

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 on when the valve is fully open is provided in an array of monitor lights that are all on when their respective valves are in proper position following an automatic safety signal initiation. This light is energized from a separate monitor light supply and actuated by a valve motor operator limit switch. An annunciator is also activated whenever the valve is not fully open.

Another separate and redundant alarm annunciator point is activated 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). A separate annunciator point is used for each accumulator valve. This alarm will be recycled at approximately 1-h intervals to remind the operator of the improper valve lineup.

RWST Isolation Valve

The indications provided at the control switches for these motor-operated isolations valves are identical to those provided for the accumulator isolation valves. An annunciator is activated whenever one of these valves is not in its expected position. Like the accumulator isolation valves, the RWST to charging pump suction isolation valves are provided with a monitor light that is energized when the valve is fully open.

REFERENCE

1. Igne, E. G. and Locante, J., "Environmental Testing of Engineered Safety Features Related Equipment (NSSS-Standard Scope)," WCAP-7744, Volume I, August 1971.

TABLE 6.3-1 (SHEET 1 OF 2)

**EMERGENCY CORE COOLING SYSTEM
COMPONENT PARAMETERS**

Accumulators		Data
Number		3
Design pressure (psig)		700
Design temperature (°F)		300
Operating temperature (°F)		60-150
Normal operating pressure (psig)		601 to 649
Total volume (ft^3) (tank only)		1450 each
Nominal water volume (ft^3)		1025[(a)] each
Nominal volume N_2 gas (ft^3)		470 each
Boric acid concentration	(nominal ppm)	2300
	(minimum ppm)	2200
Relief valve setpoint (psig)		700
Inleakage alarm sounds (ft^3)		11

Centrifugal Charging Pumps	
Number	3
Design pressure (psig)	3000
Design temperature (°F)	300
Design flowrate (gal/min)	150
Head at design flowrate (ft)	5800
Maximum flowrate (gal/min)	708
Head at maximum flowrate (ft)	2400
Discharge head at shutoff (ft)	6000
Motor nameplate rating (bhp)	900

Parameter

Residual Heat Removal Pumps

Refer to Table 5.5-8

a. This value includes the liquid volume in the tank plus the liquid volume in the piping measured from the tank to the second check valve. The second check valve is defined as the second check valve from the tank or the first check valve from the reactor coolant system (RCS) loop.

b. The 2A charging pump (Q2E21P002A), the 2B charging pump (Q2E21P002B), 1C charging pump (Q1E21P002C), the 1A charging pump (Q1E21P002A), and the 2C charging pump (Q2E21P002C) design pressure is 3000 psig.

TABLE 6.3-1 (SHEET 2 OF 2)

Residual Heat Exchangers

Refer to Table 5.5-8

Valves

1.	All motor-operated valves which must function on safety injection ("S") signal up to and including 8 in. Maximum opening or closing time (s)		10
2.	Leakage		
a.	Conventional globe valves	Disc leakage, $cm^3/h/in.$ of nominal pipe size	3
		Backseat leakage (when open), $cm^3/h/in.$ of stem diameter	1
b.	Gate valves	Disc leakage, $cm^3/h/in.$ of nominal pipe size	3
		Backseat leakage (when open), $cm^3/h/in.$ of stem diameter	1
c.	Check valves	Disc leakage, $cm^3/h/in.$ of nominal pipe size	3
d.	Diaphragm valves	Disc leakage	none
e.	Pressure relief valves	Disc leakage, $cm^3/h/in.$ of nominal pipe size	3
f.	Accumulator check valves	Disc leakage, $cm^3/h/in.$ of nominal pipe size	3
g.	Ball valves	Seat leakage, $cm^3/h/in.$ of seat diameter	2

TABLE 6.3-2

ECCS RELIEF VALVE DATA

Description	Fluid Discharged	Fluid Inlet Temp. Normal	°F Relieving	Set Pressure (psig)	Back Pressure Constant	Psig Build-Up	Capacity
N_2 supply to accumulators	N_2 gas	120	120	700	atm.	0	1500 sf^3/min
RHR pumps discharge SI line	Water	250	350	600	3	50	20 gal/min
Accumulator to containment	N_2 gas	120	120	700	0	0	1500 sf^3/min

TABLE 6.3-3

SEQUENCE OF CHANGEOVER OPERATION FROM INJECTION TO RECIRCULATION

(This table has been deleted.)

TABLE 6.3-4 (SHEET 1 OF 3)

TIME ANALYSIS FOR ECCS INJECTION/RECIRCULATION SWITCHOVER

Step	Time (s)[a]	Flow Rate From RWST (gal/min) During Step	Time Step (s) for Constant RWST Flow Rate[a]	Volume Remaining In RWST (gal)
1. Low-level switchover setpoint	0		0	147,630
2. Verify SI reset	10	13,400		
3. Direct verification of PRF status	20	13,400		
4. Verify CCW flow to RHR heat exchangers	60	13,400		
5. Establish only one charging pump in each train	70	13,400		
6. Direct verification of recirculation disconnects	80	13,400	80	129,763
7. Stop both RHR pumps	90	9,000		
8. Close RWST supply to ‘A’ RHR pump suction	110	9,000		
9. Align containment sump to ‘A’ RHR pump suction	150	9,000		
10. Close RHR to RCS hot legs cross-connect	170	9,000	90	116,263
11. Start ‘A’ RHR pump	180	7,600		
12. Verify ‘A’ Train LHSI flow	185	7,600		
13. Close RWST supply to ‘B’ RHR pump suction	205	7,600		
14. Align containment sump to ‘B’ RHR pump	245	7,600		
15. Close RHR to RCS hot legs cross-connect	265	7,600		

TABLE 6.3-4 (SHEET 2 OF 3)

Step (continued)	Time (s)	Flow Rate From RWST (gal/min) During Step	Time Step (s) for Constant RWST Flow Rate[a]	Volume Remaining In RWST (gal)
16. Start 'B' RHR pump	275	7,600		
17. Verify 'B' Train LHSI flow	280	7,600		
18. IF 'A' RHR pump started, THEN align charging pump suction header isolation valves based on 'B' charging pump status	360	7,600	190	92,197
19. Open RHR supply to 'A' train charging pump suction	380	6,700		
20. Verify VCT level	385	6,700	25	89,405
21. Close 'A' train RWST to charging pump header valve	405	7,150		
22. IF 'B' RHR pump started, THEN align charging pump suction header isolation valves based on 'B' charging pump status	410	7,150	25	86,426
23. Open RHR supply to 'B' train charging pump suction	430	6,700		
24. Verify VCT level	435	6,700		
25. Close 'B' train RWST to charging pump header valve[b]	455	6,700		
26. Check one charging pump in each train	460	6,700		
27. Open charging pump recirculation to RCS cold legs valve	480	6,700		
28. Align charging pump discharge header isolation valves based on 'B' charging pump status	560	6,700		
29. Verify SI flow	565	6,700	155	69,118
Approximate time to low-low alarm (highest flow/nominal flow)[c]	658 / 752.55	13,400 / 6,700	93 / 187.55	48,348 / 48,174[d]

TABLE 6.3-4 (SHEET 3 OF 3)

Notes:

(a) Overall time and time steps are for reference only. Time Critical Operator Action (TCOA) time is controlled by plant procedure.

(b) Validation of TCOA actions ends after completion of step 25 since switchover is complete at that point.

(c) "Highest flow" completion time conservatively assumes a step or steps consuming the highest flow take longer. This is demonstrated by the RWST volume that would nominally be available after completion of step 29 being consumed at the highest flow of 13,400 gal/min, leading to a low-low alarm at about 658 s from the low-level switchover setpoint (i.e., step 1).

(d) Sufficient volume remaining in the RWST for containment spray switchover. Difference in final volume between highest flow and nominal scenarios due to rounding.

TABLE 6.3-5 (SHEET 1 OF 2)

**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
Tube sheet	Austenitic stainless steel
Valves	
Motor-operated valves	
Containing radioactive fluids Pressure containing parts	Austenitic stainless steel or equivalent
Body-to-bonnet bolting and nuts	Low alloy steel
Seating surfaces	Stellite No. 6 or equivalent
Stems	Austenitic stainless steel or 17-4PH stainless or B637 UNS N07718 (Inconel 718)
Motor-operated valves containing non-radioactive boron-free fluids	
Body, bonnet and flange	Carbon steel
Stems	Corrosion resistance steel
Diaphragm valves	Austenitic stainless steel
Accumulator check valves	
Parts contacting borated water	Austenitic stainless steel
Clapper arm shaft	

TABLE 6.3-5 (SHEET 2 OF 2)

Component	Material
Relief valves	17-4 pH stainless
Stainless steel bodies	Stainless steel
Carbon steel bodies	Carbon steel
All nozzles, discs, spindles, and guides	Austenitic stainless steel
Bonnets for stainless steel valves without a balancing bellows	Stainless steel or plated carbon steel
All other bonnets	Carbon steel
Piping	
All piping in contact with borated water except for a	Austenitic stainless steel
short carbon steel spool piece at the transition of the pump suction line inside containment to the post-LOCA strainer assemblies	Carbon steel

TABLE 6.3-6

NORMAL OPERATING STATUS OF EMERGENCY CORE COOLING

Number of charging pumps operable	2
Number of residual heat removal pumps operable	2
Number of residual heat exchangers operable	2
Minimum refueling water storage tank volume (gal)	471,000
Boron concentration in refueling water storage tanks (ppm)	2,300 to 2,500
Boron concentration in accumulator (ppm)	2,200 to 2,500
Number of accumulators	3
Normal operating accumulator pressure (psig) band	601 to 649
Nominal accumulator water volume (ft^3)	1025[a]

a. This value includes the liquid volume in the tank plus the liquid volume in the piping measured from the tank to the second check valve. The second check valve is defined as the second check valve from the tank or the first check valve from the reactor coolant system (RCS) loop.

TABLE 6.3-7 (SHEET 1 OF 2)

SINGLE ACTIVE FAILURE ANALYSIS FOR EMERGENCY CORE COOLING SYSTEM COMPONENTS SHORT TERM PHASE

Component	Malfunction	Comments
Accumulator	Deliver to broken loop	Totally passive system with one accumulator per loop. Evaluation based on one spilling accumulator
Pump		
Centrifugal charging	Fails to start	Three provided. Evaluation based on operation of one
Residual heat removal	Fails to start	Two provided. Evaluation based on operation of one
Automatically Operated Valves		
Injection line isolation	Fails to open	Two parallel lines; one valve in either line required to open
Residual heat removal pumps suction line to refueling water storage tank	Fails to close	Check valve in series with one gate valve; operation of only one valve required
Centrifugal charging pumps		
a. Suction line to refueling water storage tank	Fails to open	Two parallel lines; only one valve in either line is required to open
b. Discharge line to the normal charging path	Fails to close	Two valves in series; only one valve required to close
c. Miniflow line	Fails to close	Two valves in series; only one valve required to close
d. Suction from volume control tank	Fails to close	Two valves in series; only one valve required to close

TABLE 6.3-7 (SHEET 2 OF 2)

LONG TERM PHASE

Component	Malfunction	Comments
Valves operated from control room for recirculation		
Containment sump recirculation isolation	Fails to open	Two lines parallel; two valves in either lines are required to open
Residual heat removal pumps suction line to refueling water storage tank	Fails to close	Check valve in series with one gate valve; operation of either the check or the gate valve required
Centrifugal charging pump suction line to 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
Centrifugal charging pump suction line at discharge of residual heat exchanger	Fails to open	Separate and independent high head injection path taking suction from discharge of the other residual heat exchanger
Pumps		
Residual heat removal pump	Fails to start	Two provided. Evaluation based on operation of one
Centrifugal charging pump	Fails to operate	Same as short term phase
Failure of Train B power during switchover from cold leg recirculation to simultaneous hot and cold leg recirculation results in:		
• Residual heat removal discharge valve to hot legs (MOV 8889)	Fails to open	Align RHR pumps to cold legs, Train A charging pump to hot legs, and use Train A/Train B Power
• Centrifugal charging pump discharge valve to cold legs (MOV 8803B)	Fails to close	Transfer Switch (Q1/2R18B037) to apply Train A power to close MOV 8803B

TABLE 6.3-8

MAXIMUM POTENTIAL RECIRCULATION LOOP LEAKAGE EXTERNAL TO CONTAINMENT

Item	Type of Leakage Control and Unit Leakage Rate Used in the Analysis	Leakage to Atmosphere (cm^3/h)	Leakage to Drain Tank (cm^3/h)
Residual heat removal (low head safety injection)	Mechanical seal with leakoff - 10 cc/hr/seal	0	20
Charging pumps	Same as residual heat removal pump[(a)]	0	60
Flanges:			
Pumps	Gasket - adjusted to zero leakage following any test	0	0
Valves bonnet to body (larger than 2 in.)	10 drops/min/gauge used (30 cc/hr). Due to leak tight flanges on pumps, no	2400	0
Control valves	leakage is assumed to atmosphere	480	0
Heat exchangers		240	0
Valves - stem leakoffs	Back seated double packing with leakoff - 1 cc/hr in. stem diameter used. (See table 6.3-1.)	0	50
Miscellaneous small valves	Flanged body packed stems - 1 drop/min used (3 cm^3/h).	600	0
Miscellaneous large valves (larger than 2 in.)	Double packing 1 $cm^3/h/in.$ stem diameter used	40	0

a. Seals are acceptance tested to essentially zero leakage. Due to tandem double seal arrangement and the use of water from the refueling water storage tank as a buffer between the seals, no radioactive leakage from the pumps to the atmosphere is expected.

TABLE 6.3-9

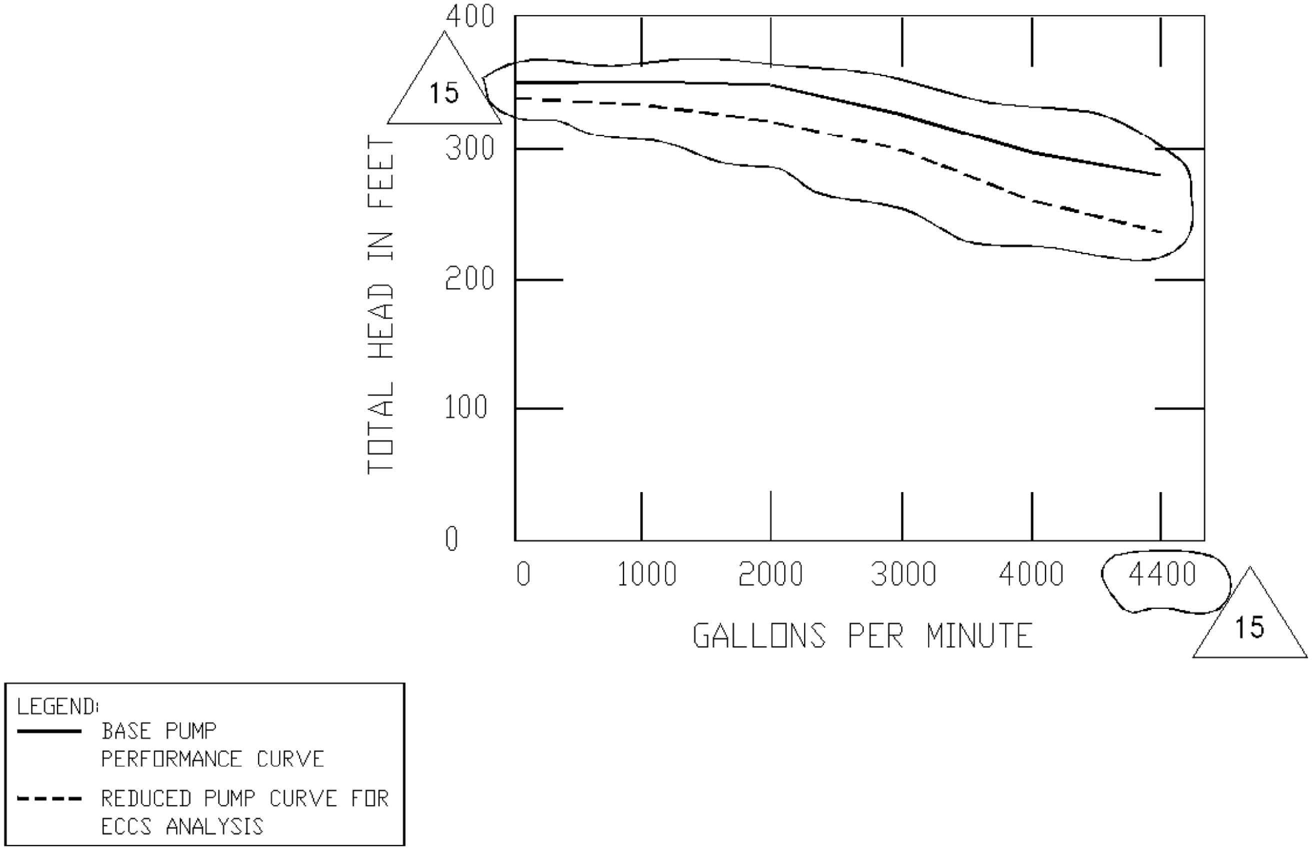
EMERGENCY CORE COOLING SYSTEM RECIRCULATION PIPING PASSIVE FAILURE ANALYSIS

Flow Path	Indication of Loss of Flow Path	Alternate Flow Path
Low head recirculation		During cold-leg recirculation:
From containment 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	Via the independent, identical low head flow path utilizing the second residual heat exchanger
High-head recirculation		During hot-leg recirculation:
		The high head pumps provide the required redundancy during this period
From containment 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	From containment sump to the high head injection headers via alternate residual heat removal pump, residual heat exchanger and the alternate high head charging pump

TABLE 6.3-10

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 of centrifugal charging and residual heat removal pumps. Valves for realignment of RWST to charging pumps meet the single failure criteria
Centrifugal charging pumps	Lined up for charging service	Lined up to high head safety injection header. Valves for realignment meet single failure criteria
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 for residual heat removal pump operation	Lined up for residual heat removal pump operation



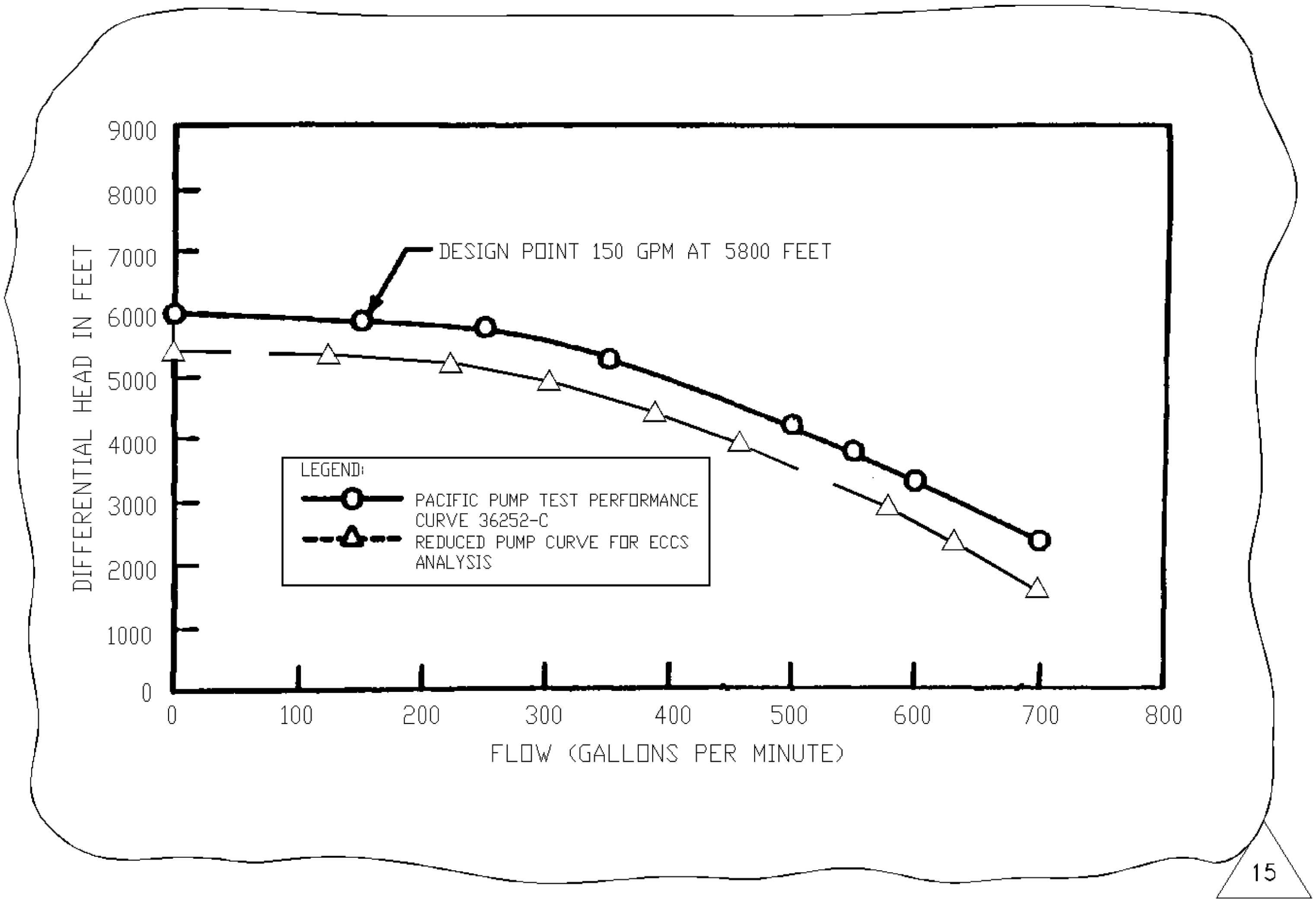
REV 21 5/08

SOUTHERN COMPANY
Energy to Serve Your World®

JOSEPH M. FARLEY NUCLEAR PLANT UNIT 1 AND UNIT 2

RESIDUAL HEAT REMOVAL PUMP PERFORMANCE CURVES

FIGURE 6.3-1

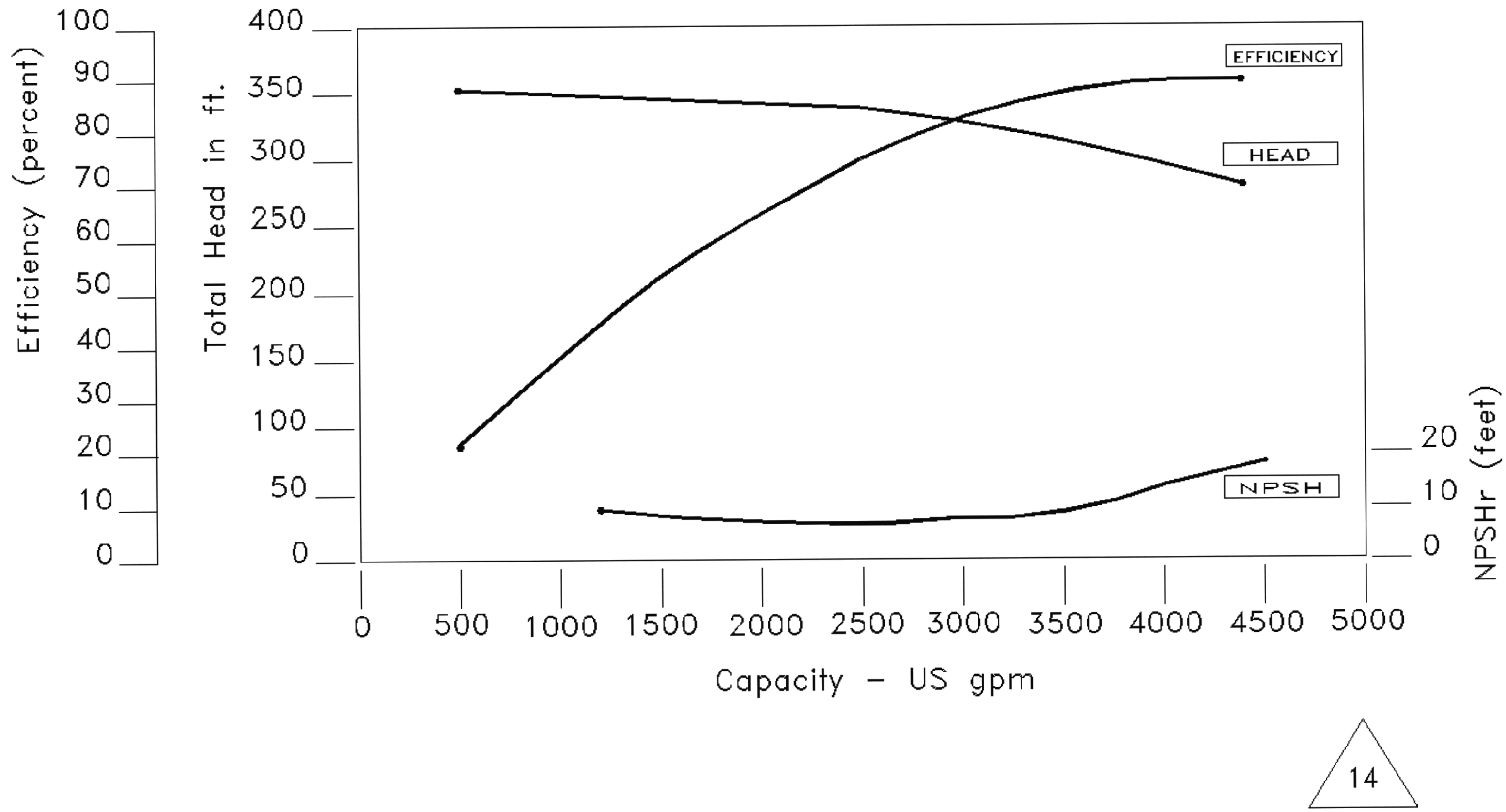


REV 21 5/08

JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

CHARGING PUMP PERFORMANCE CURVES

FIGURE 6.3-2



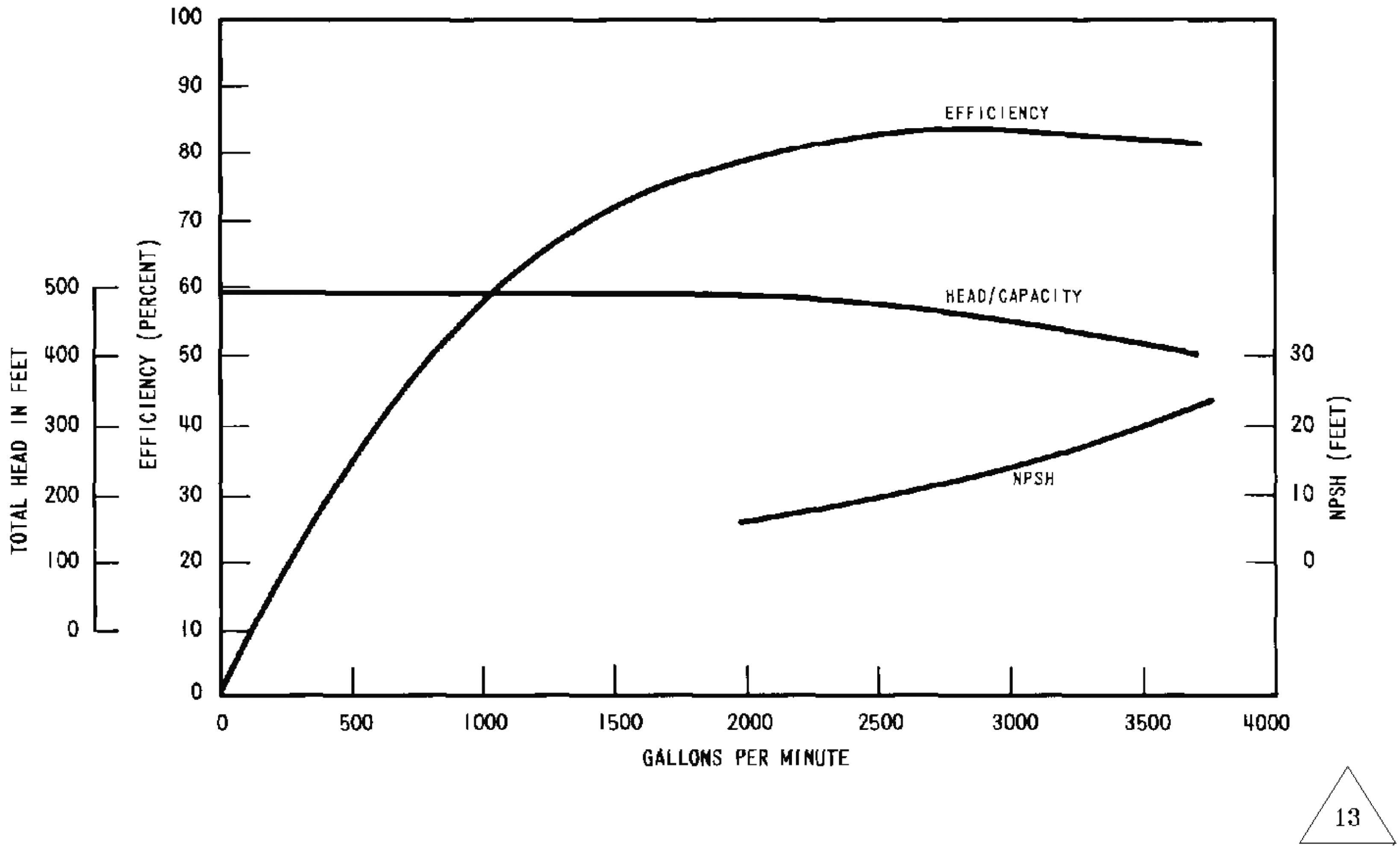
REV 21 5/08

SOUTHERN COMPANY
Energy to Serve Your World®

JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

TYPICAL RHR PUMP CHARACTERISTIC CURVES

FIGURE 6.3-3



REV 21 5/08

SOUTHERN COMPANY
Energy to Serve Your World®

JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

CONTAINMENT SPRAY PUMP CHARACTERISTIC CURVES

FIGURE 6.3-4

6.4 HABITABILITY SYSTEMS

The control room habitability systems are designed to provide maximum safety and comfort for operating personnel during normal operations and during postulated accident conditions. These habitability systems for the control room include control room envelope (CRE), shielding, charcoal filter systems, heating, ventilation and air conditioning, storage capacity of food and water, kitchen, sanitary facilities, and fire protection.

The control room habitability systems are designed to meet NRC acceptance criteria contained in 10 CFR 50.67, which is discussed in subsection 3.1.15. Sufficient shielding and ventilation are provided to permit occupancy of the control room for a period of 30 days following a design basis accident (DBA) without receiving more than 5 rem total effective dose equivalent (TEDE). Figure 12.1-1 shows the layout of the control room and its location with respect to the rest of the plant.

6.4.1 HABITABILITY SYSTEMS FUNCTIONAL DESIGN

6.4.1.1 Design Bases

The following design bases were used to determine the functional design of the habitability system:

A. The postulated accident that determines the habitability design requirements is the design basis LOCA. Postulated accidents are discussed in chapter 15.0.

B. The assumptions regarding the sources and amounts of radioactivity that could pose a hazard to the control room are discussed in subsection 12.1.3.

C. In the event of an accident, the ventilation system in the control room will be triggered by the containment isolation actuation system (CIAS) signal or detection of high radiation levels entering the control room. The CIAS signal automatically isolates the normal air systems and starts both trains of the control room ac system, pressurization system, and filtration system. Detection of high radiation levels automatically isolates the normal air systems. However, the pressurization and filtration systems must be manually initiated. The outside air used to pressurize the control room is filtered through a charcoal filter system which is capable of removing an allowed 99.0 percent of both the inorganic and organic iodine. An efficiency of 95 percent is allowed for recirculation.

   The filter system and the control room shielding are capable of keeping the dose to the operators less than 5 rem whole body dose or its equivalent to any part of the body for the duration of the accident. The control room shielding is discussed in section 12.1 and the control room air conditioning, heating, cooling, and ventilation systems are discussed in subsection 9.4.1.

D. Following postulated accidents, the limitations on control room pressure, temperature, and doses are as follows:

Parameter	Allowable
Control Room Pressure	> adjacent areas
Control Room Temperature	≤ 120°F
Doses (whole body equivalent)	5 rem

E. The CRE provides the pressure boundary for control room habitability. The CRE boundary is made up of walls, floor, roof, ducting, valves or dampers, and ESF HVAC equipment housings.

F. The fire protection system in the control room consists of an early warning ionization type detection system with hand portable H_2O extinguishers located in the control room itself. The ionization detectors used will rapidly detect products of combustion. Detectors are located on the false ceiling to detect smoke in the control room itself. Another set of detectors is located in the space above the false ceiling for detection in this area. Each detector is equipped with a light to indicate which detector has operated. All detectors will operate the visible and audible alarm on the main fire protection annunciator panel located in the control room. In addition, two fixed carbon dioxide hose reels with 100-ft hose are located in the immediate vicinity outside the control room and can be used to back up the hand extinguishers.

Noncombustible materials are used in construction and equipment as much as possible. The quantity of combustible material such as paper and other flammable supplies is kept to a minimum. A person trained in fire fighting is on duty in the control room at all times. For control room responses to a fire event impacting habitability see A-181805, NFPA 805 Fire Protection Program Design Basis Document.

6.4.1.2 <u>System Design</u>

6.4.1.2.1 Piping and Instrumentation Diagrams

The piping and instrumentation diagram of the control room ventilation and cleanup system is provided in drawings D-175012 and D-205012.

6.4.1.2.2 Performance Objectives

The performance objectives to be maintained are as follows:

Air condition flowrate	21,000 ft^3/min
HEPA and charcoal filter unit flowrate	3,000 ft^3/min
Pressure	> adjacent areas
Temperature	78°F (db)
Humidity	50%
Limits of radioactivity for normal plant operation	≤ values stated in Table 1 of Appendix B to 10 CFR 20.1001-20.2401.

6.4.1.2.3 Provisions to Intake, Exhaust, Monitor, and Filter

A. Intake, Exhaust, and Monitoring

During normal plant operation one of the two 100-percent-capacity air handling units recirculates 21,000 ft^3/min of cooled filtered air to the control room. Makeup air is supplied to the control room through a supply duct from the computer room air conditioning unit. A smoke detector near the return air duct to each recirculation fan will sound an alarm in the control room on high smoke level. If necessary, the operator can exhaust air from the control room by manually opening the pneumatic operated exhaust dampers and starting one of the two 100-percent-capacity exhaust fans.

The CIAS signal will automatically switch the control room ventilation system to emergency pressurization and activate the charcoal filter system. In addition, the radiation monitoring system in the control room will detect a high radiation level in the control room, isolate the normal control room ventilation system, and alert the operator to switch to the emergency recirculation mode with the charcoal filter systems, to thus reduce the radiation level.

B. Control Room Charcoal Filter System

The control room filtration system is designed to minimize the activity level in the control room resulting from high airborne radiation. In addition, the filtration system, along with the exhaust system, minimizes the hazards from any noxious gases. The control room filtration system consists of two parallel fully redundant full capacity fan and filter systems. The system is not used during normal operations but is to be tested periodically. The tests consist of fan operation to measure filter pressure drops and radioactive testing of charcoal samples at intervals to ensure efficiency. The fan and filter system are Seismic Category I and are designed to withstand, without exceeding the yield stresses and without loss of function, the forces resulting from the safe shutdown earthquake (SSE).

<u>Description of the Charcoal Filter System</u>

A. Charcoal Type

New, commercially pure, activated coconut shell, impregnated 5 percent by weight with iodine compounds. The granule size is 8-16 mesh.

B. Charcoal Weight, Tray Type Units

A minimum of 43 lb of charcoal will fill each unit (2 elements per unit). There are 3 units in each tray type charcoal filter system.

C. Charcoal Bed Configuration

1. Tray Type

The charcoal filter unit is standard manufactured size, having a frontal face size of 8 in. by 24 in. and having two horizontal, flat charcoal beds, each approximately 24 in. by 28 in. by 2 in. in depth, arranged in parallel fashion with an air space between beds.

2. Bed Type

The bed type filters are of the high efficiency carbon adsorber (HECA) type. The 2-inch (recirculation filter) and 6-inch (pressurization filter) layers of carbon are in modules of vertically oriented bed-welded construction. The filter beds are permanently installed by welding to the supporting frame and housing.

D. Charcoal Test Specifications

Samples of activated charcoal used in the filters are tested per testing practices required by Technical Specification surveillances. **[*HISTORICAL]* [***The following describes pre-operational testing requirements:*

1. Tray Type

a. Removal of all iodines with an efficiency of 95.0 percent for a 12 hour continuous flow at 150°F, 70 percent relative humidity, and 40 ft/min face velocity.

b. Each adsorbing unit (2 elements) is capable of filtering 333 ft^3/min of air at a pressure drop not exceeding 1.2 in. wg.

c. Each assembled filter unit will be leak tested by the manufacturer.

d. Each filter will be tested for 5 minutes in airflow of 330 ft^3/min containing 20 ppm refrigerant 112. A downstream concentration in excess of 0.2 percent of the upstream concentration will cause rejection of the filter.

2. Bed Type

a. Retaining 99.0 percent minimum of elemental iodine. At relative humidities below 70 percent at 150°F all organic iodines with an *efficiency of 95 percent for the recirculation filter and 99 percent for the pressurization filter.*

b. *Each assembled filter unit will be leak tested by the manufacturer.*

c. *Charcoal adsorbers remove ≥ 99 percent of a halogenated hydrocarbon refrigerant test gas.]*

E. Acceptance Criterion

In addition to meeting the above specifications and tests, the filter units shall be subject to in-place testing to ensure that the design requirements are met. The necessary modifications will be made if the filter units do not meet the design standards.

F. HEPA Filter Type

The HEPA filters conform to MIL-F-51079 (MIL-F51068), except for paragraph 6.3. Media shall be made principally of inorganic fibers and any organic fibers and any organic content shall not be more than 5.0 percent. The media shall be at least 0.015 in. thick. The filters are individually tested and certified by the manufacturer to have an efficiency of not less than 99.97 percent when tested with 0.3 micron smoke. Filters carry UL labels indicating full compliance with requirements of UL Standard UL-586.

G. Humidity Controls

A heater is provided to keep the relative humidity of the air entering the control room pressurization charcoal filter unit below 70 percent.

H. Test and Surveillance Requirements

Each charcoal filter unit is provided with charcoal canisters containing a sample of charcoal identical to that used in each filter element or a grain thief representative sample is taken. These test canisters are exposed to the airflow just as the main cells are; then they are used to determine the remaining life of the charcoal elements. In addition, access is provided so that visual inspections may be made periodically.

I. Expected Efficiencies for Iodine Removal

Control Room Pressurization Filters	Recirculation Filter	Filtration Filter
Elemental Iodine 99.0 percent	95.0 percent	95.0 percent
Organic Iodine 99.0 percent	95.0 percent	95.0 percent

6.4.1.3 Design Evaluations

6.4.1.3.1 Shielding

The shielding in the control room consists of the concrete walls, floor, and ceiling, as discussed in section 12.1. The shielding is designed for continuous occupancy during a LOCA and meets 10 CFR 50.67.

6.4.1.3.2 Heating, Ventilation, Air Conditioning, and Air Purification Systems

The control room heating, ventilation, air conditioning, and air purification systems are designed as two parallel, redundant, full capacity, seismic Category I systems. Redundant onsite power is provided in the event of a loss of offsite power. Only one of the two redundant systems is needed for the control room. A complete discussion of the air conditioning, heating, cooling, and ventilation systems is found in subsection 9.4.1.

6.4.1.3.3 Storage Capacity of Food and Water

There will be sufficient storage capacity (including a food freezer) for two shifts of operators for 30 days. Water is drawn from the potable and sanitary water system.

6.4.1.3.4 Kitchen

Kitchen facilities are available for refrigeration, cooking, and the cleaning of cooking and eating utensils.

6.4.1.3.5 Sanitary Facilities

Bathroom facilities are provided adjoining the control room.

6.4.1.3.6 Control Room Envelope (CRE)

The CRE airtight boundary is designed for continuous occupancy during normal and emergency conditions.

6.4.1.4 Testing and Inspection

6.4.1.4.1 Preoperational Testing

The preoperational testing programs are discussed in the following sections and subsections:

A.	Ventilation system	12.2
B.	Charcoal filter system	6.4.1.2.3
C.	Smoke detection system	6.4.1.5.1
D.	Radiation monitoring system	12.1.4

6.4.1.4.2 Inservice Surveillance

Programs of inservice surveillance are discussed below or in the following sections and subsections:

A.	Ventilation system	12.2
B.	Charcoal filter system	9.4.1.4
C.	Smoke detection system	6.4.1.5.1
D.	Radiation monitoring system	12.1.4

CRE inleakage testing, preventative maintenance, localized leak testing post modification or maintenance of boundary, boundary administrative controls, and related inleakage monitoring are conducted per the Control Room Integrity Program.

6.4.1.5 <u>Instrumentation Requirement</u>

The instrumentation employed for monitoring and actuation of the habitability systems consists of the smoke detectors, the high radiation level alarm system, and CIAS.

6.4.1.5.1 Smoke Detectors

The smoke detection system is an early warning ionization type. Detectors are located on the false ceiling to detect smoke in the control room itself. Another set of detectors is located in the space above the false ceiling to serve that area. Each detector is equipped with a light to indicate which detector has operated. All detectors will operate the visible and audible alarm on the main fire protection annunciator panel located in the control room. In addition, a smoke detector near the return air duct to each recirculation fan will sound an alarm in the control room on high smoke level.

Inservice surveillance of the control room smoke detection system and its components is performed to ensure the necessary reliability and integrity of this system.

6.4.1.5.2 High Radiation Level Alarm

An area radiation monitor located in the control room alarms on high radiation level and alerts the operator to the possible need for filtration of recirculated air. The monitor is capable of reading in the range of 10^{-4} R/h to 10 R/h.

6.5 AUXILIARY FEEDWATER SYSTEM

6.5.1 DESIGN BASES

The auxiliary feedwater system is designed to supply feedwater to the steam generator during plant startup, cooldown, and emergency conditions when the normal supply is not available.

The system contains two motor-driven pumps and one turbine-driven pump. Each of the motor-driven pumps is sized to supply the steam generators with 100 percent of the required feedwater flow for a normal safe cooldown of the reactor coolant system. The turbine-driven pump is capable of providing 200 percent of the required feedwater flow for a normal safe cooldown of the reactor coolant system. In the event of a main feedwater line break and assuming the worst single active failure (loss of the turbine-driven auxiliary feedwater pump), the two motor-driven auxiliary feedwater pumps are required to provide sufficient flow to the intact steam generators to achieve a safe shutdown of the plant.

The auxiliary feedwater system design is based on providing sufficient flow to prevent the loss of pressurizer vapor space during a feedwater line break with loss of offsite power. The turbine-driven pump is designed to operate with steam produced in the steam generators and to deliver sufficient feedwater flow to safely cool down the reactor coolant system. No ac power is required for 2 hours for operation of the turbine-driven auxiliary feedwater pump. Valves Q1N12V001A-A and 1B-B, as shown in drawing D-175033, sheet 2, have been provided with air reservoirs with sufficient capacity to open the valves and allow turbine operation for 2 hours.

The auxiliary feedwater system P&ID is shown on drawing D-175007. Parameters for the auxiliary feedwater pumps and drives are shown in Table 6.5-1. The steam supply to the auxiliary feedwater pump turbine drive is shown on drawings D-175033, sheet 1; D-175033, sheet 2; D-170114, sheet 1; D-170114, sheet 2; D-205033, sheet 1; D-205033, sheet 2; and D-200007.

The auxiliary feedwater system is an engineered safety feature designed to meet the single failure criterion as defined in subsection 3.1.17. The entire system is designed to meet Seismic Category I requirements. Since the portion of the AFW pumps' minimum flow recirculation line located outdoors is exposed to a potential tornado missile, the AFW system and condensate storage tank were designed to perform their function with a rupture in the recirculation line. Because the auxiliary feedwater system was designed to accomplish its required function with failure of these minimum flow recirculation lines, the NRC has determined that the auxiliary feedwater system conforms to the acceptance criteria in GL 81-14. In addition to the minimum flow recirculation lines, four flow instrumentation lines attached to the AFW pump suction lines are located outdoors and exposed to a potential tornado missile. An analysis was performed to verify that adequate reserve margin in the CST water is available even when considering rupture of these instrumentation lines from missile impact. The result of the analysis showed that the reserve water margin available in the protected volume (164,000 gal) in the CST after the volume used for decay and sensible heat removal, still exceeds the total water volume lost from all the ruptured lines. Further discussion of the CST water volume is given in subsection 9.2.6. The quality group classifications and the design codes that apply to the components for the auxiliary feedwater system are listed in subsection 3.2.2. The design life of the equipment is 40 years.[a]

6.5.2 SYSTEM DESCRIPTION

6.5.2.1 General Description

The system consists of two motor-driven pumps, one steam turbine-driven pump, associated piping, valves, instruments, and controls.

The pumps are normally aligned to take suction from the condensate storage tank. One 8-in. suction header supplies condensate to the two motor-driven pumps and a separate 8-in. suction line supplies condensate to the turbine-driven pump. Each pump's individual suction line contains a locked open isolation valve, a nonreturn valve, and a low pressure switch that annunciates in the main control room.

A backup source of water for the pumps is provided from the safety-related portion of the service water system. The service water is isolated from the normal suction piping by two closed motor-operated gate valves. These valves can be operated remote manually from the control room or by using the manual handwheel at the valve. Each of the three pumps can be supplied with water from either of the two redundant service water headers. The service water system is described in subsection 9.2.1.

Each of the two motor-driven pumps discharges through a nonreturn valve and an isolation valve into a common header. From this header, individual lines feed each steam generator through two normally opened motor-operated valves, a flow restriction orifice, and a control valve station, consisting of an air operated control valve, locked open manual block valves, and a nonreturn valve.

The breakers supplying power to the motor-operated stop check valves MOV-3350A, B, and C (drawing D-175007) will be racked out during normal plant operation, so that no power is supplied to the valve operators. In the event that actuation of these valves from the control room is required, power will be reconnected to these valve operators by closing the associated power breakers.

The turbine-driven pump discharges through a nonreturn valve and branches into three lines, each containing a flow restriction orifice and a control valve station. Downstream of the control valve station, each of these three lines joins with the corresponding line from the motor-driven pumps. A single supply line then connects to the main feedwater line downstream of the main

---

a. The renewed operating licenses authorize an additional 20-year period of extended operation for both FNP units, resulting in a plant operating life of 60 years. In accordance with 10 CFR Part 54, appropriate aging management programs and activities have been initiated to manage the detrimental effects of aging to maintain functionality during the period of extended operation (see chapter 18).

feedwater stop valve. The single auxiliary feedwater line for each steam generator contains a flow orifice for local and control room indication of the auxiliary feedwater flow. Downstream of this flow orifice, a check valve is provided which normally functions to prevent backflow of main feedwater into the auxiliary feedwater system. In addition, normally open motor-operated isolation valves can be operated from the control room to isolate failures in the steam and feedwater systems.

Each pump has a minimum flow recirculation line with a pressure reducing orifice or a locked manually operated anticavitation pressure reducing flow control valve, a nonreturn valve, and a manual locked open block valve. In addition to the minimum flow recirculation line, each pump has a manual lock closed recirculation system and a breakdown orifice for testing the pump at the design point. The minimum flow recirculation line and the test line for the three pumps are joined together and routed to the condensate storage tank.

Connections are provided on two of the plant's three main steam lines for steam supply to the auxiliary feedwater pump turbine drive. The isolation valves in these lines are normally closed; however, a normally open bypass warming line is provided to keep the supply piping at main steam temperature. Downstream of the isolation valves, the two steam lines penetrate the elevation 127-ft main steam and feedwater valve room floor and join into a common header, which contains an air- operated normally closed isolation valve and the turbine trip and throttle valve. Between the elevation 127-ft floor and the point where the two lines form a single header, each line contains a check valve. This valve prevents reverse steam flow and subsequent loss of steam supply to the turbine drive, in the event that one of the steam supply lines has been damaged due to a high energy line break in the main steam and feedwater valve room. The electronic governor is integral with the turbine. A turbine-driven, auxiliary feed pump start signal opens the steam supply valves at the main steam line connection and the isolation valve at the turbine. This signal is described in section 7.3.

6.5.2.2 <u>Component Description</u>

6.5.2.2.1 Auxiliary Feedwater Pumps

The auxiliary feedwater pumps are multistage, horizontal, centrifugal units designed to deliver the required flow to the steam generators under the highest head requirements that occur when the main steam safety valves are discharging to the atmosphere. Design data for the pumps are given in table 6.5-1.

Pump room coolers are used to maintain air temperature in the motor-driven auxiliary feedwater pump rooms at or below 104°F during normal operation. Refer to Table 9.4-6A for post-DBA room temperatures. Auxiliary feedwater pump room coolers are discussed in paragraph 9.4.2.1.9.

6.5.2.2.2 Auxiliary Feedwater Pump Turbine Drive

The auxiliary feedwater pump turbine drive is a steam driven, horizontal single stage noncondensing unit, utilizing an electronic governor, overspeed trip mechanism, and an integral trip and throttle valve. Turbine speed can be controlled locally or remote-manually from the control room. Steam for the turbine is taken from two of the three main steam lines upstream of the main steam stop valves. Bypass warming lines are provided to keep the steam supply piping up to the turbine inlet isolation valve at main steam temperature. Exhaust from the turbine is routed to the atmosphere.

Turbine bearings are lubricated by a forced feed lube oil system driven from the turbine shaft. Lube oil cooling water is supplied from the first stage of the auxiliary feedwater pump discharge and returned to the pump suction via the pump balancing line. This arrangement ensures a supply of cooled lube oil whenever the turbine is operating.

6.5.2.2.3 Piping

Auxiliary feedwater suction and discharge piping is seamless carbon steel. Welded joints are used throughout the system except for flanged connections at the pumps, flow orifices, and flow restriction orifices.

6.5.2.2.4 Valves

All valves in the auxiliary feedwater flowpath from the condensate storage tank to the steam generators are normally open, with the exception of the fail open auxiliary feedwater control valves when the steam generator level is being controlled by auxiliary feedwater or the auxiliary feedwater system is being tested.

With the exception of the suction piping from the condensate storage tank, the entire auxiliary feedwater system is designed for full feedwater pressure. A check valve is provided at the connection to the feedwater header and two additional check valves in series are provided to prevent backflow into the auxiliary feedwater system. Table 6.5-3 lists the electrical power supply and the failure position of all motor operated valves in the auxiliary feedwater system.

6.5.2.2.5 Controls

The auxiliary feedwater system can be operated locally from the hot shutdown panel or remotely from the control room. In addition, certain plant conditions, as described in subsection 6.5.2.3.3, will automatically initiate auxiliary feedwater flow to the steam generators. Instrumentation and controls for the system are shown on drawing D-175007. Instrumentation and controls for the steam supply to the turbine drive are shown on drawings D-175033, sheet 1, D-175033, sheet 2, D-170114, sheet 1, D-170114, sheet 2, D-205033, sheet 1, D-205033, sheet 2 and D-200007.

Controls and control signals for the two motor driven pumps are train oriented. The turbine driven pump can operate with a loss of all ac power. A 3-kVA uninterruptible power system

(UPS) has been uniquely assigned to provide a reliable source of control power for the turbine-driven auxiliary feedwater pump and its associated steam admission and discharge valves. (See subsection 8.3.3)

A selector switch for each of the six auxiliary feedwater flow control valves is provided on the hot shutdown panel. This switch places control of the valves either in the main control room or on the panel. Annunciation is provided in the main control room when any one of hot shutdown valves is under local control.

6.5.2.3 System Operation

The auxiliary feedwater system is not required to operate during normal power operation. The system is on standby to deliver auxiliary feedwater flow on receipt of any of the emergency signals given below in subsection 6.5.2.3.3.

6.5.2.3.1 Plant Startup

During a plant startup the auxiliary feedwater system is placed under manual control to supply feedwater to the steam generators. This in turn maintains steam generator water level until sufficient steam pressure is generated to allow startup of the turbine driven steam generator feedwater pumps. Suction is taken from the condensate storage tank.

6.5.2.3.2 Normal Cooldown

During normal plant cooldown, the auxiliary feedwater system is placed under manual control to supply feedwater to the steam generators for removal of decay and sensible heat from the reactor coolant system. The rate of auxiliary feedwater flow is remote-manually regulated from the control room to maintain the steam generator level while steam is dumped to the condenser or atmospheric relief valves depending on the mode of operation. After approximately 4 hours, the residual heat removal system is placed in operation and the auxiliary feedwater system is no longer required for heat removal. During a normal cooldown, suction is taken from the condensate storage tank.

The reactor plant cooldown rate is controlled by the operator using the steam dump valves or power operated relief valves. The operator controls the cooldown rate within the specified limits. However, operation of two auxiliary feedwater pumps (two at 350 gal/min each) delivering feedwater at 100°F will permit a maximum initial cooldown rate of up to about 200°F/h and operation of one auxiliary feedwater pump will permit a maximum initial cooldown rate of up to about 100°F/h.

6.5.2.3.3 Emergency Operation

With the control switches in the auto position, the motor driven auxiliary feedwater pumps automatically starts on any of the following signals:

A. Tripping of both steam generator feed pumps.

B. Low-low water level signals from two out of three level transmitters on any one steam generator.

C. Any of the conditions as defined in section 7.3 that cause a safety injection signal.

D. Blackout signal (loss of offsite power).

Operation of the turbine driven auxiliary feedwater pump is initiated by the opening of the steam supply valves to the turbine drive. Steam from the main steam header is automatically admitted to the turbine drive on either of the following signals:

A. Loss of power signal (2/3 reactor coolant pump bus undervoltage).

B. Low-low water level signals from two out of three of the level transmitters of any two out of three steam generators.

Cooling water to the turbine bearing oil cooler is automatically supplied, as described in subsection 6.5.2.2.2.

The flow control valves are normally open. However, if the flow control valves are closed, they will automatically open on an emergency pump start signal. The motor-driven auxiliary feedwater pump discharge valves open fully in response to all of the auxiliary feedwater system automatic initiation signals shown in drawing U-166244, regardless of the position of the valve control switches. No other valve operators are required to function in the auxiliary feedwater piping in order to establish flow when an automatic start signal is received.

The motor-driven auxiliary feedwater pumps are powered from redundant emergency buses.

Under emergency conditions, suction for the pumps is provided from the condensate storage tank. A backup source of water is available from the service water system.

In the unlikely event of the loss of the condensate storage tank, the operator must actuate the backup water source by remote-manual opening of the normally closed motor-operated valves, which separate the service water system from the auxiliary feedwater pump suction lines. The valve hand control switches are located in the main control room and the valves can be opened remote-manually with a minimum of operator action. After actuation of the valve hand control switches, 10 seconds are required for the valves to open fully.

The air-operated auxiliary feedwater flow control valves, which are located in the elevation 127-ft main steam room, have been provided with manual operators so that the valves may be modulated locally in the event of a loss of all power sources (electric and air). In the event of a high energy line break which prohibits personnel access to the manual handwheel located on each auxiliary feedwater flow control valve, and the simultaneous loss of valve air and power supplies, auxiliary feedwater flow may be regulated by the manual operation of globe valves that are located in each discharge line between the auxiliary feedwater pumps and the air-operated flow control valves. These manually operated globe valves are located in the area below the elevation 127-ft main steam room and are accessible after a high energy line break in the main steam room.

6.5.3 DESIGN EVALUATION

The auxiliary feedwater system is designed to function for the normal startup and shutdown of the plant during periods where the required flowrate is very small, relative to power level feed rates, and during startup when steam for the steam generator feed pump turbines is not available. In addition, the system is designed as an engineered safety feature to provide redundant means of removing decay and sensible heat from the reactor coolant system via the steam generators during emergency conditions. The auxiliary feedwater system design also conforms to the NRC acceptance criteria contained in (1) General Design Criterion 44 and (2) Branch Technical Position APCSB 10-1 regarding diversity of power sources, system flexibility, and redundancy including the combination single active failure and high energy line break.

The system is designed to meet the single failure criteria so that no single failure will prevent the supply of sufficient feedwater to at least two of the three steam generators.

In addition to the normal feedwater source from the condensate storage tank, a redundant backup source is provided from the Safety Class 2b portion of the service water system. Sufficient instrumentation is provided, as described in subsection 6.5.5, to alert plant operators of malfunctions or failures affecting auxiliary feedwater flow.

Both local and remote means are provided for system operation.

A failure analysis of the auxiliary feedwater system is provided in table 6.5-2.

Flow restriction orifices are installed in each auxiliary feedwater pump injection line upstream of the air-operated flow control valve, as shown on drawing D-175007. In the event of a main steam or feedwater line break, the auxiliary feedwater pumps will start, and within 60 seconds will pump auxiliary feedwater through the flow restriction orifices to the three steam generators. The flow restriction orifices limit flow to the faulted steam generator and establish flow to the two intact steam generators. Within 30 min after the main feedwater isolation signals, valves MOV 3764A, B, C, D, E, or F (drawing D-175007) will be manually activated from the control room, as required, to isolate flow from the motor-driven pumps to the faulted steam generator. Sufficient redundancy is provided so that the isolation function can be accomplished assuming a single failure, including loss of a power train. Isolation of the motor-driven pump flow to the faulted steam generator increases the flow to the intact steam generators and thus allows an orderly cooldown to the cold shutdown condition. With this case of no operator action being taken for 30 min, reactor coolant pressure will not exceed the pressurizer safety valve setpoint nor will the reactor coolant water level fall below the top of the core. For related additional details, see paragraphs 3K.4.1.4.7, 3K.4.1.4.9, and 15.4.2.2.

Steam can be supplied to the auxiliary feedwater pump turbine drive from two of the three main steam lines. Each of the connections to the main steam lines is sized to supply 100 percent of the required steam flow to operate the turbine.

6.5.4 TESTS AND INSPECTION

Each of the auxiliary feedwater pumps was hydrostatically tested in the manufacturer's shop in accordance with code requirements. Each pump was also performance tested in accordance with the ASME Performance Test Code PTC 8.2. The turbine drive was given a hydrostatic test, a mechanical running test, and an overspeed trip test in the manufacturer's shop prior to shipment. The results of these tests were acceptable. Subsequent shop tests for the pump rotating assemblies may be performed in accordance with either PTC 8.2 or the Hydraulic Institute Standards to verify acceptable pump performance.

The wall thicknesses of pressure boundary castings of the pumps, turbine, trip and throttle valve, and governor valve were checked and recorded by the manufacturer. Nondestructive testing of each component was performed in accordance with the requirements of the applicable codes.

The entire auxiliary feedwater system was hydrostatically tested after completion of field erection. The system is periodically tested on line by closing isolation valves and recirculating the auxiliary feedwater flow to the condensate storage tank. The full flow/preservice test is performed either off line or at reduced power by direct injection of auxiliary feedwater flow into the steam generators.

6.5.5 INSTRUMENTATION

Local and control room indication of auxiliary feedwater flow to each of the steam generators is provided by flow orifices in the auxiliary feedwater supply lines, located just upstream of the auxiliary feedwater stop check valves. The flow transmitters are seismically and environmentally qualified. The safety grade steam generator level instrumentation provides a qualified backup for the flow instrumentation. Redundant control room indicators and low level alarms are provided for condensate storage tank levels. The low level alarm setpoint gives at least 20 min for operator action, assuming that the turbine-driven pump is operating.

Flow indication is provided in the control room for each of the two 8-in. suction lines from the condensate storage tank. A high flow alarm is provided to indicate the failure of a suction line in the auxiliary building. In addition, each pump's suction pressure is indicated and low suction pressure is alarmed in the main control room. Upon a loss of normal feedwater supply from the condensate storage tank, the operator can remote-manually open the valves in the suction piping from the service water system. The discharge pressure of each pump is indicated in the main control room. Valve position lights are provided in the main control room for each of the air- and/or motor-operated valves in the system. Manual valves are locked in the safe position.

The auxiliary feedwater pump turbine drive speed is indicated and controlled in the main control room. Low steam inlet pressure is alarmed in the main control room.

A temperature monitoring system is provided to detect auxiliary feedwater fluid heatup due to backleakage from the main feedwater system through the auxiliary feedwater check valves. This fluid heatup could cause steam binding of the auxiliary feedwater pumps if this system did not detect the check valve leakage before the leakage became significant.

The temperature element locations are shown on drawing D-175007. A control room alarm is actuated if the temperature at any point exceeds a preset limit. Local indication is provided for the system.

TABLE 6.5-1

AUXILIARY FEEDWATER SYSTEM
AUXILIARY FEEDWATER PUMP DATA

		MOTOR-DRIVEN PUMPS (DATA PER PUMP)		TURBINE-DRIVEN PUMP
Type		Horizontal-Centrifugal		Horizontal-Centrifugal
No. of stages		10		7
Design pressure (psig)		1600		1600
Pumping temperature (°F)		95		95
Design flowrate (gal/min)		350		700
Design head (ft)		2845		2835
NPSH required at design (ft)		17		21
Minimum available NPSH (ft)		60		60
Suction pressure range (ft)		45-75		45-75
Shutoff head (ft)		3480		3380
RPM		3600		3960
Bhp required		366		687
Driver horsepower (max)		450		693
Materials:				
Casing		SA-217 Gr. WC 9		SA-217 Gr. WE 9
Impeller		SA-296 Gr. CA 15		SA-296 Gr. CA 15
	or	A-217 Gr. CA 15	or	A-217 Gr. CA 15[a]
Shaft		A-276 Tp 410 HT		SA-276 Tp 410 HT

TURBINE DRIVE

Type	Vertical-Single Stage
Design pressure (psig)	1250
Design temperature (°F)	572
Steam inlet pressure (psig)	
Minimum	90
Maximum	1148
Back pressure (psig)	0-10
RPM design/turbine trip	3960/4554
Rated Bhp	687
Governor	NEMA Class D
Lubrication	Forced feed
Cooling water	Pumped liquid

a. Changes are applicable to Unit 2 only.

TABLE 6.5-2 (SHEET 1 OF 4)

FAILURE ANALYSIS OF AUXILIARY FEEDWATER SYSTEM

Component	Failure	Comments and Consequences
Motor-driven auxiliary feedwater pump	Fails to start on automatic signal	Two motor-driven pumps are provided. One motor-driven pump in conjunction with the turbine-driven pump is sufficient to meet cooldown requirements for all emergency conditions. One motor-driven pump is sufficient to meet all normal cooldown requirements
Turbine-driven auxiliary feedwater pump	Fails to start on automatic signal	Operation of the two motor-driven pumps will provide sufficient flow to meet cooldown requirements for all conditions
Turbine-driven pump steam inlet isolation valve from main steam header	Fails to open on Blackout signal	Parallel connections are provided to two main steam lines. One of the two valves must open to supply 100 percent of the turbine steam requirements
Steam supply lines to turbine driven pump	One parallel supply line broken downstream of inlet isolation valve in main steam and feedwater valve room	Check valves installed in each parallel line, upstream of the common header connection and below the floor of the main steam and feedwater valve room, prevent blowdown through the broken line and subsequent loss of steam supply to the turbine drive
Condensate supply	Loss of normal supply from condensate storage tank	Water can be supplied to all pumps from the service water system. Service water supply is separate and redundant
Auxiliary feedwater pump discharge line	Failure of pressure boundary resulting in abnormal leakage	No single failure can prevent the auxiliary feedwater system from providing the minimum required flow. Both manual and motor-operated valves are provided for isolating potential breaks
Electrical power supply	Failure of power supply bus to components associated with one motor driven pump	Motor-driven pumps are separate and redundant including power supplies. One motor-driven pump in conjunction with the turbine-driven pump will supply the minimum required flow for all emergency conditions
Motor operated valves in pump discharge piping	Loss of power	All motor operated valves are manually open, fail "as is" on loss of power, and are closed remote manually
Air operated flow control valves in pump discharge	Loss of air or loss of 125-V dc power	Failure modes presented in table 7.3-10, sheet 2

TABLE 6.5-2 (SHEET 2 OF 4)

Component	Failure	Comments and Consequences
Isolation valves MOV 3350A, B, C	Spurious closure of motor-operated valve	During normal plant operation, these valves are in the open position and the breakers, which supply power to the valves, are opened and locked so that no power is supplied to the valve's motor operator.
Main feedwater line between the containment isolation valve and the steam generator	Failure of a main feedwater line with a simultaneous loss of Train A electrical power	Case 1 - Failure of main feedwater line to steam generator 1A.--The Train B motor-driven pump and the turbine-driven pump start and delivery flow through the restriction orifices which limit auxiliary feedwater flow to the faulted steam generator, thus establishing the minimum required flow to two intact steam generators. Closing valve MOV 3764E, which is powered from a Train B electrical power supply, isolates auxiliary feedwater flow from the motor-driven pump to the faulted steam generator. This increases flow to the two intact steam generators, allowing an orderly cooldown to the cold shutdown condition Case 2 - Failure of main feedwater line to steam generator 1B.--Identical to Case 1 above except that valve MOV 3764B, which is powered from a Train B electrical power supply, is closed to isolate auxiliary feedwater flow from the motor-driven pump to the faulted steam generator Case 3 - Failure of main feedwater line to steam generator 1C.--Identical to Case 1 above except that valve MOV 3764C, which is powered from a Train B electrical power supply, is closed to isolate auxiliary feedwater flow from the motor-driven pump to the faulted steam generator
Main feedwater line between the containment isolation valve and the steam generator	Failure of a main feedwater line with a simultaneous loss of Train B electrical power	Case 1 - Failure of main feedwater line to steam generator 1A.--The Train A motor-driven pump and the turbine-driven pump start and deliver flow through the restriction orifices which limit auxiliary feedwater flow to the faulted steam generator, thus establishing the minimum required flow to the two intact steam generators. Closing valve MOV 3764A, which is powered from a Train A electrical power supply, isolates auxiliary feedwater flow from the motor-driven pump to the faulted steam generator. This increases flow to the two intact steam generators, allowing an orderly cooldown to the cold shutdown condition.

TABLE 6.5-2 (SHEET 3 OF 4)

Component	Failure	Comments and Consequences
		Case 2 - Failure of main feedwater line to steam generator 1B.--Identical to Case 1 above except that valve MOV 3764D, which is powered from a Train A electrical power supply, is closed to isolate auxiliary feedwater flow from the motor-driven pump to the faulted steam generator. Case 3 - Failure of main feedwater line to steam generator 1C.--Identical to Case 1 above except that valve MOV 3764F, which is powered from a Train A electrical power supply, is closed to isolate auxiliary feedwater flow from the motor-driven pump to the faulted steam generator.
Main feedwater line between the containment isolation valve and the steam generator	Failure of a main feedwater line with a simultaneous loss of the turbine-driven auxiliary feedwater pump	Case 1 - Failure of main feedwater line to steam generator 1A.--The Train A and Train B motor-driven auxiliary feedwater pumps start and deliver flow through the restriction orifices which limit auxiliary feedwater flow to the faulted steam generator, thus establishing the the minimum required flow to the two intact steam generators. Closing either valve MOV 3764A or MOV 3764E isolates motor-driven auxiliary feedwater pump flow to the faulted steam generator. This increases flow to the two intact steam generators, allowing an orderly cooldown to the cold shutdown condition. Case 2 - Failure of main feedwater line to steam generator 1B.--Identical to Case 1 above except that either valve MOV 3764B or MOV 3764D is closed to isolate auxiliary feedwater flow to the faulted steam generator. Case 3 - Failure of main feedwater line to steam generator 1C.--Identical to Case 1 above except that either valve MOV 3764C or MOV 3764F is closed to isolate auxiliary feedwater flow to the faulted steam generator.

TABLE 6.5-2 (SHEET 4 OF 4)

Component	Failure	Comments and Consequences
Main feedwater line between the containment isolation valve and the steam generator	Failure of a main feedwater line with a simultaneous spurious closure of a motor operated valve in the pump discharge flow path	Case 1 - Failure of main feedwater line to steam generator 1A with a simultaneous spurious closure of either motor-operated valve located in the motor-driven pump discharge line to steam generator 1B.--The Train A and Train B motor-driven pumps and the turbine-driven pump start and deliver flow through the restriction orifices. The restriction orifices limit flow to the faulted steam generator, thus establishing the minimum required flow to the two intact steam generators. Closing either valve MOV 3764A or MOV 3764E isolates motor-driven pump flow to the faulted steam generator and increases motor-driven pump flow through the open flow path to steam generator 1C, thus allowing an orderly cooldown to the cold shutdown condition. Case 2 - Failure of main feedwater line to steam generator 1A with a simultaneous spurious closure of either motor operated valve located in the motor-driven pump discharge line to steam generator 1C.--Identical to Case 1 above except isolation of the faulted steam generator increases motor-driven pump flow to steam generator 1B. Note - For all possible combinations of a faulted steam generator and a spurious closure of any one of valves MOV 3764A, B, C, D, E or F, the operator can remote manually isolate the motor-driven pump flow to the faulted steam generator, which increases motor-driven pump flow through the open flow path(s) to the intact steam generators, thus allowing an orderly cooldown to the cold shutdown condition.

TABLE 6.5-3

AUXILIARY FEEDWATER SYSTEM MOTOR OPERATED VALVE DATA

Valve Motor Number (Ref. drawing D-175007)	Motor Control Center Supplying Electricity to Valve (Ref. drawing D-177001)	Valve Position After Loss of Power
MOV 3209A	MCC 1U	As is
MOV 3209B	MCC 1V	As is
MOV 3210A	MCC 1U	As is
MOV 3210B	MCC 1V	As is
MOV 3216	MCC 1U	As is
MOV 3350A	MCC 1U	As is
MOV 3350B	MCC 1U	As is
MOV 3350C	MCC 1U	As is
MOV 3764A	MCC 1U	As is
MOV 3764B	MCC 1V	As is
MOV 3764C	MCC 1V	As is
MOV 3764D	MCC 1U	As is
MOV 3764E	MCC 1V	As is
MOV 3764F	MCC 1U	As is

APPENDIX 6A

MATERIALS COMPATIBILITY REVIEW

TABLE OF CONTENTS

LIST OF TABLES

APPENDIX 6A

MATERIALS COMPATIBILITY REVIEW

6A.1 DEFINITION OF POSTACCIDENT CONTAINMENT ENVIRONMENTAL CONDITIONS

An evaluation of the suitability of materials of construction for use in the containment has been performed considering the following:

A. The integrity of the materials of construction of engineered safety features equipment when exposed to postdesign basis accident (DBA) conditions.

B. The effects of corrosion and deterioration products from both engineered safety features (vital equipment) and other (nonvital) equipment, on the integrity and operability of the engineered safety features equipment.

The post DBA environment conditions of temperature, pressure, radiation, and chemical composition are described in the following sections. The time temperature pressure cycle used in the materials evaluation is most conservative, since it considers only partial safeguards operation during the DBA. The spray and core cooling solutions considered herein include both the design chemical compositions and the design chemical compositions contaminated with deterioration products and fission products, which may conceivably be transferred to the solution during recirculation through the various containment safety features systems.

6A.1.1 DESIGN BASIS ACCIDENT TEMPERATURE-PRESSURE CYCLE

Containment pressure/temperature versus time responses for the various analyzed breaks are shown in figures 6.2-1 through 6.2-41. These figures represent containment environment conditions during and after a postulated accident considering partial safety features operation: that is, operation with 1 of the 2 spray pumps, 1 of the 4 containment fans, 1 of the 2 residual heat removal pumps, and 1 of the 3 safety injection pumps.

Table 6A-1 presents the evaluation conditions for Westinghouse supplied material subjected to the containment and the core environment, respectively. For equipment specified by Bechtel and Southern Company Services, Inc., refer to table 3.11-1.

Material evaluations, to be described, were performed, in general, for the time temperature conditions of table 6A-1 or conservatively considering high temperature conditions for longer periods. The basis for each material evaluation is described with the discussion of its particular suitability.

6A.1.2 DESIGN BASIS ACCIDENT RADIATION ENVIRONMENT

Evaluation of materials for use in containment included a consideration of the radiation stability requirements for the particular materials application. This evaluation utilized data that were calculated on the basis of a core meltdown and, assuming the following fission product fractional releases, consistent with TID 14844 model:

Noble gases	Fractional release	1.0
Halogens	Fractional release	0.5
Other isotopes	Fractional release	0.01

6A.1.3 DESIGN CHEMICAL COMPOSITION OF THE EMERGENCY CORE COOLING SOLUTION

Farley system designs provide for use of alkaline adjusted boric acid solution as the spray and core cooling fluid.

Alkaline Sodium Borate

Plant designs that utilize the spray solution for fission product iodine removal, as well as containment cooling include provisions for chemical addition to control pH. For Farley trisodium phosphate (TSP) is added to the containment sump. Boric acid solution, containing 2300 to 2500 ppm boron, is pumped from the refueling water storage tank into the core and to the containment by means of the safety injection system pumps, residual heat removal pumps, and spray pumps. The initial pH of the spilled RCS water and containment spray will be approximately 4.5. Three baskets are located on elevation 105'-6" which contain sufficient TSP so that when their contents dissolve in the water from the RWST, RCS, and accumulators, the resulting containment sump and recirculation (ECCS and spray) systems pH will be between 7.0 and 9.1.

For the purpose of materials evaluation in the design chemistry solution, the following concentration/time relationship was considered:

0	8 hours	pH	4.5	Boron 2500 ppm
8 hours	12 months	pH	10	Boron 2500 ppm

The solutions are considered aerated through the entire exposure period as in the case of pure boric acid spray solution.

6A.1.4 TRACE COMPOSITION OF EMERGENCY CORE COOLING SOLUTION

During spraying and recirculation, the emergency core cooling (ECC) solution will wash over virtually all the exposed components and structures in the reactor containment. The ECC solution is recirculated through a common sump; hence, any contamination deposited in or leached by the solution from the exposed components and structures will be uniformly mixed in the solution.

The materials compatibility discussion includes consideration of the effects of trace elements which are identified as conceivably being present in the ECC solution during recirculation.

To identify the trace elements in containment which may have a deleterious effect on engineered safety features equipment, one must first establish which elements are potentially harmful to the materials of construction of the safety features equipment and second, ascertain the presence of these elements in forms which can be released to the ECC solution following a design basis accident. Table 6A-2 presents a listing of the major periodic group of elements. Elements known to be harmful to various metals are noted and potential sources of these elements are identified.

The concentration of the trace contaminants in the ECC solution will vary with individual plant construction as well as with the chemical composition of the ECC solution itself.

6A.2 MATERIALS OF CONSTRUCTION IN CONTAINMENT

All materials in containment are reviewed from the standpoint of insuring the integrity of equipment of which they are constructed and to insure that deterioration products of some materials do not aggravate the accident condition. In essence, therefore, all materials of construction in the containment must exhibit resistance to the postaccident environment or, at worst, contribute only insignificant quantities of trace contaminants which have been identified as potentially harmful to vital safeguards equipment. Table 6A-3 lists typical material of construction used in the containment. Examples of equipment containing these materials are included in the table.

Corrosion testing, described in section 6A.3, showed that of all the metals tested only aluminum alloys and zinc were found incompatible with the alkaline sodium borate solutions. Aluminum and zinc were observed to corrode at a significant rate, with the generation of hydrogen gas. Since hydrogen generation can be hazardous to containment integrity a detailed survey was conducted to identify all aluminum and zinc components in containment.

The as-built aluminum inventory present inside the containment is described in drawing A-508597 (Farley Unit 1) and A-508928 (Farley Unit 2). The drawings also include the mass of metal and exposed surface area of each component used in the calculation of hydrogen generated post-LOCA. The 1100- and the 6000-series aluminum alloys are the major types found in containment. This inventory provides some insight into the range of components which are often fabricated from aluminum. All metals of construction in containment, including aluminum, are compatible with unadjusted boric acid solution under DBA conditions.

The total analyzed value of zinc inventory considered in the analysis of post-LOCA hydrogen generation is described below. Ample margin was included for each source of zinc in the analysis with respect to the zinc inventory for future addition of zinc inside containment.

Zinc Inventory:

Item	Surface Area (ft^2)
Zinc Based Paint	298,216
Galvanized Carbon Steel	125,864
Cable Trays	44,328

Since the corrosion rate of zinc is considerably lower than the aluminum, the rate of mass depletion of zinc due to corrosion is lower. Therefore, the thickness and mass of the zinc inventory is not considered in the post-LOCA hydrogen generation analysis.

6A.3 CORROSION OF METALS OF CONSTRUCTION IN DESIGN BASIS ECC SOLUTION

Emergency core cooling components are austenitic stainless steel and, hence, are quite corrosion resistant to the alkaline sodium borate solution as demonstrated by corrosion tests reported in WCAP-7153(1). The general corrosion rate, for Type 304 and 316 stainless steels, was found to be 0.01 mils/months in pH 10 solution at 200°F. Data on corrosion rates of these materials in the alkaline sodium borate solution have been reported by ORNL(2, 3) to confirm the low values.

Extensive testing was also performed on other metals of construction found in the reactor containment. Testing was performed on these materials to ascertain their compatibility with the spray solution at design post-accident conditions and to evaluate the extent of deterioration product formation, if any, from these materials.

Metals tested included zircaloy, Inconel, aluminum alloys, cupronickel alloys, carbon steel, galvanized carbon steel and copper. The results of the corrosion testing of these materials are reported in detail in reference 1. Of the materials tested, only aluminum and zinc were found to be incompatible with the alkaline sodium borate solution. Aluminum corrosion is discussed in section 6A.5. The following is a summary of the corrosion data obtained on various materials of construction exposed for several weeks in aerated alkaline (pH 9.0-9.3) sodium borate solution at 200°F. The exposure condition is considered conservative since the test temperature (200°F) is considerably higher than the long term design basis accident temperature (152°F), and the pH bounds the long term design basis accident pH. Corrosion of zinc in post-LOCA environment is discussed in section 15.4.1.6.2.

Material	**Maximum Observed Corrosion Rate mil/month**
Carbon Steel	0.003
Zr-4	0.004
Inconel 718	0.003
Copper	0.015
90 - 10 Cu-Ni	0.02
70 - 30 Cu-Ni	0.006
Galvanized carbon steel	0.051
Brass	0.01

Tests conducted at ORNL[2, 3] also have verified the compatibility of various materials of construction with alkaline sodium borate solution. In tests conducted at 284°F, 212°F, and 130°F, stainless steel, Inconel, cupronickel, Monel and zircaloy-2 experienced negligible changes in appearance and negligible weight loss.

Corrosion tests at both the Westinghouse Pressurized Water Reactor Division and ORNL have shown copper and copper nickel alloys suffer only slight attack when exposed to the alkaline sodium borate solution at DBA conditions. The corrosion rate of copper, for example, in alkaline sodium borate solution at 200°F is ~0.015 mil/month[1]. The corrosion of copper in an alkaline sodium borate environment under spray conditions at 264° and 212°F have been reported by ORNL. Corrosion penetrations of less than 0.02 mil was observed after 24-hour exposure at 284°F (reference 3, table 3-13) and a corrosion rate of less than 0.3 mil per month was observed at 212°C. (See reference 2, table 3-6.)

It can be seen therefore that the corrosion of copper in the postaccident environment will have a negligible effect on the integrity of the material. Further, the corrosion product formed during exposure to the solution appears tightly bound to the metal surface and hence will not be released to the ECC solution.

Consideration was given to possible caustic corrosion of austenitic steels by the alkaline solution. Data presented by Swandby[4] shows that these steels are not subject to caustic stress cracking at the temperature (285°F and below) and 6A-6 caustic concentration (less than 1 weight percent) of interest. The stress cracking boundary temperature as defined by Swandby is considerably above (~80°F) the long term, postaccident design temperature of 152°F.

It should be noted when considering the possibility of caustic cracking of stainless that the sodium hydroxide boric acid solution is a buffer mixture wherein no free caustic exists at the temperatures of interest, even should the solution be concentrated locally through evaporation of water; hence the above consideration is somewhat hypothetical with regard to the Farley postaccident environment.

6A.4 CORROSION OF METALS OF CONSTRUCTION BY TRACE CONTAMINANTS IN ECC SOLUTION

Of the various trace elements that could occur in the emergency core cooling solution in significant quantities, only chlorine (as chloride) and mercury are adjudged potentially harmful to the materials of construction of the safeguards equipment.

The use of mercury or mercury bearing items, however, has been restricted in the Farley containment. Most mercury vapor lamps, fluorescent lighting, and instruments that employ mercury for pressure and temperature measurements and for electrical equipment have been prohibited in the containment building. Contamination due to exposure to mercury is possible if one or more temporary underwater lights used in the refueling cavity, transfer canal, and the spent-fuel pool were to fail catastrophically. The lights approved for use in these areas are manufactured by ROS, model HPS-1000, and contain up to 3 mg of mercury each in double encapsulated bulbs. The use of up to twelve of these lights at any one time has been evaluated as acceptable.

The possibility of chloride stress corrosion of austenitic stainless steels has also been considered. It is believed that corrosion by this mechanism will not be significant during the postaccident period for the following reasons:

6A.4.1 LOW TEMPERATURE OF ECC SOLUTION

The temperature of the ECC solution is reduced after a relatively short period of time (i.e. a few hours) to about 150°F. While the influence of temperature on stress corrosion cracking of stainless steel has not been unequivocally defined, significant laboratory work and field experience indicate that lowering the temperature of the solution decreases the probability of failure. Hoar and Hines[5] observed this trend with austenitic stainless steel in 42 weight percent solutions of $MgCl_2$ with temperature decrease from 310° to 272°F. Staehle and Latanision[6] present data which also shows a decreased probability of failure with decreasing solution temperature from about 392°F to 302°F. Staehler and Latanision[6] also report the data of Warren[7] which showed the significant change with decrease in temperature from 212°F to 104°F. The work of Warren, while pertinent to the present consideration in that it shows the general relationship of temperature to time to failure, is not directly applicable in that the chloride concentration (1800 ppm Cl) believed to have effected the failure was far in excess of reasonable chloride contamination that may occur in the ECC solution.

6A.4.2 LOW CHLORIDE CONCENTRATION OF ECC SOLUTION

It is anticipated that the chloride concentration of the ECC solution during the postaccident period will be low.

Restrictions in the chloride content of the water used in the postaccident period will not impair system operability. The environment of low chloride concentration, low temperature, and high pH, which will be experienced during the long-term postaccident period, will not be conducive to chloride cracking. ***$$[HISTORICAL]$$*** ***$$[****Surveillance has been maintained throughout plant construction to ensure that the chloride inventory is maintained at a minimum.****]$$***

6A.5 CORROSION OF ALUMINUM ALLOYS

Corrosion testing showed that aluminum alloys are not compatible with alkaline borate solution. The alloys generally corrode fairly rapidly, at the post-accident condition temperatures, with the liberation of hydrogen gas. A number of corrosion tests were conducted in the Westinghouse Pressurized Water Reactor Division laboratories and at ORNL facilities. A review of applicable aluminum corrosion data is given in Table 6A-5. The corrosion rates at the various temperature steps were determined from the aluminum corrosion rate design curve which was chosen to include essentially all available corrosion data.

6A.6 THE NATURE AND BEHAVIOR OF ALUMINUM CORROSION PRODUCTS IN ALKALINE SOLUTION

The corrosion of aluminum in alkaline solution, expected following a design basis accident (DBA), has been shown to proceed with the formation of aluminum hydroxide[12,13,14] and the aluminate ion, as well as with the production of hydrogen gas.

The expected DBA conditions include the establishment of an alkaline ECC solution having a total volume of liquid of 4.5×10^5 gal after actuation of the engineered safety features.

As mentioned above, aluminum is known to corrode in alkaline solutions to give a precipitate of $Al(OH)_3$, which in turn can redissolve in an excess of alkali to form a complex aluminate. Van Horn[12] noted that the precipitation of $Al(OH)_3$ begins about pH 4 and is essentially complete at pH 7. A further increase in pH to about 9 causes dissolution of the hydroxide with the formation of the aluminate.

It can be seen, therefore, that the solubility of aluminum corrosion product is a function of the pH of the environment. Consistent with this, the corrosion of aluminum is also strongly dependent on the solution pH, since when the corrosion products are dissolved from the metal surface, corrosion of the base metal can proceed more freely.

Aluminum corrosion rate data had been reported in WCAP-7153[1], Table 8. The corrosion rate of aluminum is seen to decrease by a factor of 21 (1/.048) as the pH decreases from 9.3 to 8.3, and by a factor of 83 (1/.032) as the pH decreases from 9.3 to 7.0. Therefore, one must consider both corrosion and the dissolution of the corrosion products at specific reference conditions since the two are directly related. The corrosion reactions that are of interest in the DBA condition here would include the reaction of aluminum in alkaline solution to form aluminum hydroxide: i.e.,

$$2Al + 6H_2O \rightleftarrows 2Al(OH)_3 + H_2\uparrow \tag{1}$$

and dissolution of the hydroxide to form the aluminate, i.e.,

(2)

A knowledge of the solubility product of the aluminum hydroxide in an alkaline solution allows the determination of the solubility expected for the hydroxide in the DBA environment.

Deltombe and Pourbaix[15] have determined the solubility product of aluminum hydroxide. Using the value of 2.28 x 10^{-11} for K_{sp}, as reported by Deltombe and Pourbaix, the following calculation can be made.

The solubility of $Al(OH)_3$ is determined from equation 2

$$Al(OH)_3 \rightleftarrows AlO_2^- + H^+ + H_2O$$

$$K_{sp} = [AlO_2^-][H^+]$$

$$2.28x10^{-11} = [AlO_2^-][H^+]$$

at pH = 9.3

$$[AlO_2^-] = \frac{2.28x10^{-11}}{5x10^{-10}} = 4.6x10^{-2}\text{moles/liter}$$

Therefore, the solubility of $Al(OH)_3$ in a pH 9.3 solution at 25°C (77°F) is 4.6 x 10^{-2} moles/liter or 3.0 x 10^{-2} lb/gal. Expressed as aluminum, the solubility at these conditions is 1.05 x 10^{-2} lb/gal.

The solubility of the aluminum corrosion products in the post-accident environment is a function of both solution pH and temperature. Plots of the corrosion product solubility are expressed in terms of aluminum versus solution pH for temperatures of 77°F and 150°F. The change in solubility with temperature is found utilizing the relationship of the free energy of formation, temperature, and the solubility product.

With the knowledge of the reference aluminum corrosion behavior for any specific plant, one can calculate the expected solubility limits for the corrosion reaction.

For the Farley plant, 4.5 x 10^5 gal of ECC solution will be present in the containment after actuation of the safety features. The as-built aluminum inventory present inside the containment is described in drawing A-508597 (Farley Unit 1) and A-508928 (Farley Unit 2).

Table 6A-7 presents a summary of the applicable solubility and corrosion parameters for various conditions. The table lists the applicable solubility products (K_{sp}) and solubilities at the various temperatures and solution pHs together with the soluble aluminum limit for the Farley system at the specific conditions. The last values in the table give the aluminum solubility margin after 100 days corrosion; that is, the soluble Al limit divided by the aluminum corroded. It can be seen that in all cases, including the low temperature and low pH conditions, the ECC solution is not expected to be saturated with aluminum corrosion products. Further, within the expected design conditions for temperature and pH, adequate aluminum solubility margin is available as shown on table 6A-7.

It is concluded therefore, that the corrosion products of aluminum will be in the soluble form during the post accident period considered and, hence, there is no potential for deposition on flow orifices, spray nozzles or other equipment.

6A.6.1 BEHAVIOR OF CIRCULATING ALUMINUM CORROSION PRODUCTS

The solubility of aluminum corrosion products as shown that for the Farley plant, the entire inventory produced after 100 days exposure to the post-DBA condition would remain in solution. The review also indicates that the ECC solution is only approximately 5.5 percent saturated at 77°F and less than 3 percent saturated at 150°F.

It is of interest, however, to review the experience of facilities which have operated with insoluble aluminum corrosion products and to relate their conditions with those expected in the post accident environment.

The most significant experience available to date is that of Griess[16] who operated a recirculating test facility to measure the corrosion resistance of a variety of materials in alkaline sodium borate spray solution.

Tests were conducted on 1100, 3003, 5052, and 6061 aluminum alloys exposed at 100°C in pH 9.3 sodium borate solution (0.15 M NaOH - 0.28 M H_3B0_3). It was reported that even though the solution contained copious amounts of flocculent aluminum hydroxide, it has no effect on flow through the spray nozzle (0.093-in. orifice). The pH of the solution did not change because of the increase in the corrosion products.

Griess[a] in describing his observations with regards to aluminum corrosion product deposition potential stated that:

A. No significant deposition was observed on the cooling coil installed in the solution.

B. No significant deposition was observed on the heated surfaces of the facility.

C. No significant deposition was observed on isothermal facility surfaces.

a. Private communication.

The amounts of aluminum corroded to the solution in the tests conducted by Griess at 55°C and 100°C were approximately 4.0 and 18.6 grams, respectively. The concentration of aluminum present in the recirculation stream, therefore, was approximately 0.2 and 1 gram/liter, respectively. This value is about a factor of about 5 above the aluminum concentration expected in the postaccident ECC solution at the Indian Point plant in a pH 9.3 solution after 100 days.

Hatcher and Rae(17) describe the appearance of turbidity in the Canadian National Research Experimental Reactor Unit (NRU) reactor and "propose" that deposition of aluminum corrosion products may have occurred on heat exchanger surfaces, although they do not report any specific examination results. Moreover, Hatcher and Rae report no operations problems associated with the presence of aluminum corrosion product turbidity in the NRU reactor. The overall heat transfer coefficient for each NRU reactor heat exchanger was measured after 2 years of full power operation on several occasions and within the limit of accuracy of the measurements, reported at approximately 5 percent, no change in the thermal resistance had been observed.

It is concluded, therefore, from the work of Griess and Hatcher and Rae, that the deposition of aluminum corrosion products on heat exchangers, surfaces will not be significant in the postaccident environments even for the circumstances of insoluble product formation.

6A.7 EFFECT OF POSSIBLE CHEMICAL REACTIONS ON IODINE REMOVAL CAPABILITY OF THE CONTAINMENT SPRAY SOLUTION

In evaluating the effect of possible chemical reactions on the iodine removal capability of the spray solution, it has been determined that the reaction of aluminum with an alkaline ECC solution is the only reaction occurring in the containment system during a design basis accident (DBA) which has the potential for influencing the chemistry of the ECC solution. The corrosion rate of aluminum and the solubility of the aluminum corrosion products is dependent on the pH and temperature of the alkaline spray solution. Calculations are presented in this review which estimate the mass of aluminum which would be corroded in the Farley containment following a DBA, the mass of aluminum corrosion products which would be formed, and the solubility of these corrosion products in the emergency core cooling solution. As the values in table 6A-7 indicate, there is a conservative aluminum solubility margin in the ECC solution during DBA conditions.

In the operation of a test facility to measure the corrosion resistance of a variety of materials in alkaline sodium borate spray solution, the experience of Griess(16) was that the pH of the solution did not change as a result of the buildup of aluminum corrosion products. At concentrations of 0.2 - 1.0 g of aluminum per liter, the test facility experience is representative of the Farley post accident environment, assuming that all of the aluminum in the containment had corroded away and was present in the sump solution. Although no reduction in the sump solution pH is anticipated, the equilibrium sump solution pH of 7.0 exceeds the pH required to assure that iodine is retained in the sump solution.

6A.8 COMPATIBILITY OF PROTECTIVE COATINGS WITH POSTACCIDENT ENVIRONMENT

The investigation of materials compatibility in the postaccident design basis environment also includes an evaluation of protective coatings for use in containment.

The results of the protective coatings evaluation presented in WCAP-7198[11] showed that several inorganic zinc, modified phenolics, and epoxy coatings are resistant to an environment of high temperature (320°F maximum test temperature) and alkaline sodium borate. Long term tests included exposure to spray solution at 150°F - 175°F for 60 days, after initially being subjected to the conservative containment temperature transient shown in table 6A-1. The protective coating found to be resistant to the test conditions, that is, exhibited no significant loss of adhesion to the substrate nor formation of deterioration products, comprises virtually all of the protective coatings recommended for use in the containment. Hence, the protective coatings will not add deleterious products to the core cooling solution.

It should be pointed out that several test panels of the recommended types of protective coatings were exposed for two DBA cycles and showed no deterioration or loss of adhesion with the substrate. In addition, the protective coatings applied to the components of the containment do not function as an integral part of the engineered safeguard features during DBA conditions. Although the protective coatings are selected for use on the basis of their performance during a DBA, they do not serve as an engineered safety feature to inhibit corrosive attack following a loss-of-coolant accident on the substrates on which they are applied.

6A.9 EVALUATION OF THE COMPATIBILITY OF CONCRETE ECC SOLUTION IN THE POSTACCIDENT ENVIRONMENT

Concrete specimens were tested in boric acid and alkaline sodium borate solutions at conditions conservatively (320°F maximum and 200°F steady state) simulating the post-DBA environment.

The purpose of this study was to establish:

A. The extent of debris formation by solution attack of the concrete surfaces.

B. The extent and rate of boron removal from the ECC solution through boron concrete reaction.

Tests were conducted in an atmospheric pressure, reflux apparatus to simulate long term exposure conditions and in a high pressure autoclave facility to simulate the DBA short term, high temperature transient.

Table 6A-8 presents a summary of the data obtained from the concrete boron test series.

Testing of uncoated concrete specimens in the post accident environment showed that attack by both boric acid and the alkaline boric acid solution is negligible and the amount of deterioration product formation is insignificant. In addition, the boron removal rate from the ECC solution is low.

REFERENCES

1. Bell, M.J., Bulkowski, J.E. and Picone, L.F., "Investigation of Chemical Additives for Reactor Containment Sprays," WCAP-7153, March 1968. (Westinghouse Proprietary)

2. ORNL Nuclear Safety Research & Development Program Bimonthly Report for July-August 1968, ORNL TM-2368, p. 78.

3. ORNL Nuclear Safety Research & Development Program Bimonthly Report for September-October 1968, ORNL TM-2455, p. 53

4. Swandby, R.K., Chemical Engineer 69, 186 (November 12, 1962).

5. Hoar, T.P., and Hines, J.G., "Stress Corrosion Cracking of Austenitic Stainless Steel in Aqueous Chloride Solutions," Stress Corrosion Cracking and Embrittlement (ed. W.D. Robertson) John Wieley and Sons, 1956.

6. Latanision, R.M., and Staehle, R.W., Stress Corrosion Cracking of Iron - Nickel Chromium Alloys, Dept. of Metallurgical Engineering, The Ohio State University

7. Warren, D., Proceeding of Fifteenth Annual Industrial Work Conference, Purdue University, May 1960.

8. Edeleanu, C., JISI 173, 1963, 140.

9. Thomas, K.C., et al., "Stress Corrosion of Type 304 Stainless Steel in Chloride Environment," Corrosion, Vol. 20, 1964, p. 89t.

10. Sharfstein, L.R., and Brindley, W.F., "Chloride Stress Corrosion Cracking of Austenitic Stainless Steel - Effect of Temperature and pH," Corrosion, Vol. 14, 1958, p. 588t.

11. Picone, L.F., "Evaluation of Protective Coatings for Use in Reactor Containment," WCAP-7198, April 1968. (Westinghouse Proprietary)

12. Van Horn, K.C., Aluminum, Vol. I, American Society of Metals, (1967).

13. Sundararajan, J., and Rama Char, T.C., Corrosion 17, 39t, (1961).

14. Cotton, F.A., and Wilkinson, G., Advanced Inorganic Chemistry, Interscience Publishers, (1962).

15. Deltombe, E., and Pourbaix, M., Corrosion 14, 496t, (1958).

16. Griess, J.C., et al., "Corrosion Studies," pp. 76-81, ORNL Nuclear Safety Research and Development Program Bimonthly, July - August 1968, USNRC Report ORNL TM-2368.

17. Hatcher, S.R., and Rae, H.K., Nuclear Sci. and Eng., 10, 316, (1961).

18. Rubin, K., Grover, J. L., Henninger, W. A., and Miller, T. A., “Methodology for Elimination of the Containment Spray Additive,” WCAP-11611, Rev. 0, March 1988, (Westinghouse Proprietary)

TABLE 6A-1

POSTACCIDENT CONTAINMENT TEMPERATURE TRANSIENT USED IN THE MATERIAL COMPATIBILITY REVIEW

Time Interval (s)	Temperature (°F)
0 - 300	285
300 - 1000	266
1000 - 2000	234
2000 - 4000	190
>4000	147

TABLE 6A-2 (SHEET 1 OF 2)

REVIEW OF SOURCES OF VARIOUS ELEMENTS IN CONTAINMENT AND THEIR EFFECTS ON MATERIALS OF CONSTRUCTION

Group	Representative Elements	Corrosivity of Elements	Sources of Elements
0	H3, Ne, K, Xe	No effect on any materials of construction	Fission product release
I a	Li, Na, K	Generally corrosion inhibitive properties for steels, and copper alloys - harmful to aluminum	Li - coolant pH adjusting agent Na - spray additive solution, concrete leach product K - concrete leach product
II a	Mg, Ca, Sr, Ba	Generally not harmful to steel or copper base alloys	Concrete leach products - deteriorated insulation
III a	Y, La, Ac	Not considered harmful in low concentrations	Fission product release
IV a	Ti, Zr, Hf	Not considered harmful to any materials	Fuel rod cladding, control rod material, alloying constituent
V a	V, Nb, Ta	Not considered harmful to any materials	Alloying constituents in low concentration
VI a	Cr, Mo, W	Not considered harmful to any materials	Alloying constituents in equipment
VII a	Mn, Tc, Re	Not considered harmful	Mn - alloy constituent
VIII	Fe, Ni, Cr, Os	Fe, Ni, Cr - not harmful to any materials	Fe, Ni, Cr - alloying constituents. Others have no identifiable sources

TABLE 6A-2 (SHEET 2 OF 2)

REVIEW OF SOURCES OF VARIOUS ELEMENTS IN CONTAINMENT AND THEIR EFFECTS ON MATERIALS OF CONSTRUCTION

Group	Representative Elements	Corrosivity of Elements	Sources of Elements
I b	Cu, Ag, Au	Not harmful to any materials	Cu present as material of construction and alloying constituent
II b	Zn, Cd, Hg	Hg - harmful to stainless steel, Cu alloys, aluminum Zn - unknown Cd - unknown	Hg has been entirely excluded from use in the containment. Cd finish plating on components. Zn galvanizing and alloying constituent
III b	B, A1, Ga, In	Not harmful to material	B - neutron poison additive A1 - materials of construction
IV b	C, Si, Sn, Pb	C, Si, Sn not harmful to materials. Pb considered harmful to nickel alloys	Si - concrete leach product Pb - alloy constituent in some brazes
V b	N, P, As, Sb, Bi	No effect from N unless ammonia is formed. Others unknown	N - containment air. Others not identified in significant materials
VI b	O, S, Se, Te	S possibly harmful to nickel alloys	Te - fission product S - oils, greases, insulating materials
VII b	F, C1, Br, I	F considered potentially harmful to zircaloy. C1 potentially harmful to stainless steel Br and I, not generally harmful	C1 - concrete leach product general contamination F - organic materials I and Br - fission products low concentration

TABLE 6A-3

TYPICAL MATERIALS OF CONSTRUCTION IN THE FARLEY CONTAINMENT

Material	Equipment Application
300 series stainless steel	Reactor coolant system, residual heat removal loop, spray system, fan cooler material
400 series stainless steel	Valve materials
Inconel (600, 718)	Steam generator tubing, reactor vessel nozzles, core supports, and fuel rod grids
Galvanized steel	Ventilation duct work, CRDM shroud material, I & C conduit
Aluminum	Refer to drawing A-508597 for Farley Unit 1 and A-508928 for Farley Unit 2
Copper	Service water piping, fan cooler material
70-30 Cu Ni	Fan cooler material
90-10 Cu Ni	Fan cooler material
Carbon steel	Component cooling loop, structural steel, main steam piping, etc
Monel	Possibly instrument housings
Brass	Possibly instrument housings
Protective coatings Inorganic zincs Epoxy Modified phenolics	General use on carbon steel structures and equipment, concrete

TABLE 6A-4

This table has been deleted.

TABLE 6A-5

CORROSION OF ALUMINUM ALLOYS IN ALKALINE SODIUM BORATE SOLUTION

Data Point	Temperature (°F)	Alloy Type	Corrosion Test Duration	Rate ($mg/dm^2/h$)	pH	Exposure Condition	Reference
1	275	5025	3 hours	96.2	9	Solution	WCAP-7153, Table 9
2	275	5005	3 hours	840	9	Solution	WCAP-7153, Table 9
3	200	6061	320 hours	15.4	9.3	Solution	WCAP-7153, Table 8 WCAP-7153, Figure 9
4	210	5052	7 days	53.0	9	Solution	WCAP-7153, Table 7 WCAP-7153, Figure 8
5	210	5052	2 days	14.0	9	Solution	WCAP-7153, Table 5
6	210	5005	2 days	27.1	9	Solution	WCAP-7153, Table 5
7	284	5052	1 day	54	9.3	Spray	ORNL-TM-2425, Table 3.1
8	284	5052	1 day	31.5	9.3	Solution	ORNL-TM-2425, Table 3.1
9	212	6061	3 days	126	9.3	Spray	ORNL-TM-2368, Table 3.6
10	212	6061	3 days	110	9.3	Solution	ORNL-TM-2368, Table 3.6
11	150	6061	7 days	2.9	9.3	Solution	Westinghouse Pressurized Water Reactor Division recent data
12	150	5052	7 days	4.2	9.3	Solution	Westinghouse Pressurized recent data

TABLE 6A-6

This table has been deleted.

TABLE 6A-7

SUMMARY OF ALUMINUM CORROSION PRODUCT SOLUBILITY DATA

	Solution Temperature			
	77°F		150°F	
Parameter	pH 9.3	pH 8.3	pH 9.3	pH 8.3
Solubility product K_{sp}	2.28×10^{-11}	2.28×10^{-11}	4.16×10^{-10}	4.16×10^{-10}
Al solubility (lb Al/gal)	1.05×10^{-2}	1.05×10^{-3}	1.9×10^{-1}	1.9×10^{-2}
Soluble Al limit[a] for ECCS (lb)	4.73×10^{3}	4.73×10^{2}	8.55×10^{4}	8.55×10^{3}
Al corrosion rate (normalized)	(Not used)	(Not used)	1	0.048
Al corroded after 100 days (lb)	(Not used)	(Not used)	1800	1077
Al solubility margin at 100 days	18	3	47.5	7.9

a. Solution volume 4.5×10^{5} gal.

TABLE 6A-8

CONCRETE SPECIMEN TEST DATA

Concrete - Boron Test No.	Total Exposure Period (days)	Surface Volume (in. /gal)	Exposed Weight Change (grams)	Initial Specimen Weight (grams)	Visual examination
1	24	28	- 22.4	560.0	No apparent change
3	28	20	+ 21.5	404.0	Light, yellowish, deposit on specimen
4	72	38	0	641.2	No apparent change - coating adhesion excellent
5	72	43	- 0.2	769.5	Light, hard deposit on specimen
6	~4[a]	54	-	601.4	No apparent change - small amount of sand particles in test can
7	175	23	+ 11.0	457.0	No apparent change
8	175	38	+ 26.5	751.0	No apparent change - coating adhesion excellent
9	~5[a]	78	+ 4.0	702.0	No apparent change - coating adhesion excellent

a. These tests were at high temperature DBA transient conditions. All others at 195 - 205°F.

APPENDIX 6B

CONTAINMENT PRESSURE ANALYSIS

TABLE OF CONTENTS

LIST OF TABLES

LIST OF FIGURES

APPENDIX 6B

CONTAINMENT PRESSURE ANALYSIS

6B.1 CONTAINMENT PRESSURE RESPONSE

The containment pressure response to a loss-of-coolant accident (LOCA) has been analyzed using the heat sinks as presently designed. The methods and assumptions used in this analysis are described in paragraph 6.2.1. The double-ended pump suction break was originally determined to be the worst case. The analysis for the break showed a peak pressure of 48 psig at 276 s and a maximum temperature of 313°F, at 55 s after the break. Current results are provided in paragraph 6.2.1.3.6.

A summary of the current heat sinks is given in Table 6.2-2. Table 6B-1 provides a table of the original node spacings for original heat sinks. Node spacings for power uprate analyses are generally more fine or comparable to those shown in Table 6B-1. Detailed conservative calculations were performed to determine each heat sink surface area. For additional conservatism, some heat sinks (e.g., all piping in the containment and miscellaneous steel such as some support brackets and rails) were not included in the analysis.

6B.2 CONTAINMENT SUBCOMPARTMENT ANALYSIS

The following section provides a discussion of the original design prior to application of leak-before-break exclusion of RCS main loop breaks. Current analyses and results are provided in paragraph 6.2.1.3.4.1.

The containment subcompartments analyzed for the pressure response following a LOCA were the reactor cavity and the steam generator annulus (the volume below the steam generator compartments). The pressure transient analysis was performed using a Bechtel computer code which calculates short term pressure and temperature responses. The code conservatively neglects heat transfer and all engineered safety features. A detailed description of the code is provided in appendix 3K, attachment D.

The model used for the reactor cavity analysis is shown in figures 6B-1, 6B-2, and drawing D-176277. Volumes, vent area, and flow coefficients are also shown in figure 6B-1. Blowdown data was supplied by Westinghouse for the 1 ft^2 cold leg break (at 95° az. in drawing D-176277) which is the limiting case for reactor cavity design. The blowdown is split equally between volumes 1 and 2. Insulation in the break region (compartments 1 and 2) is assumed to blow off and completely plug the cold leg penetration at the wagon wheel restraint, as well as the support shoe area ventilation duct. All gaps in the broken leg blowdown restrictor/baffle plate remain completely unobstructed by insulation throughout the transient. In all other places (i.e., reactor vessel, nozzle, and pipes for all intact legs) insulation is assumed to remain in place and not crush, leaving the seal ring gap and unbroken leg baffle plate gaps open for ventilation to the containment. The maximum horizontal force was calculated to be 1.4×10^6 lbf. The maximum uplift force was 5.9×10^4 lbf. The force-time history results are shown in figures 6B-3 and 6B-4.

The flow models for the steam generator compartment pressurization analyses are shown in figure 6B-2. Blowdown data were supplied by Westinghouse for a double ended cold leg break in the steam generator compartment C, which is the limiting case. The maximum differential pressure between steam generator compartment C and the containment was found to be 33.9 psia at 0.42 seconds.

TABLE 6B-1 (SHEET 1 OF 5)

NODE SPACINGS

Heat Sink No. 1 - Containment Cylinder and Dome

Material	Node Spacing (in.)	Thickness (in.)
Paint	1×10^{-3}	2.0×10^{-2}
Primer[(a)]	1×10^{-3}	3.0×10^{-3}
Carbon steel	6.25×10^{-2}	2.5×10^{-1}
Concrete region 1	5.0×10^{-2}	3.0
Concrete region 2	4.0×10^{-1}	6.0
Concrete region 3	1.2	6.0
Concrete region 4	10.0	30.0

Heat Sink No. 2 - Unlined Concrete

Material	Node Spacing (in.)	Thickness (in.)
Paint	1.0×10^{-3}	18.0×10^{-3}
Surfacer[(a)]	1.0×10^{-2}	1.25×10^{-1}
Concrete region 1	5.0×10^{-2}	3.0
Concrete region 2	1.76×10^{-1}	3.0
Concrete region 3	6.0×10^{-1}	3.0

Heat Sink No. 3 - Outside Reactor Cavity

Material	Node Spacing (in.)	Thickness (in.)
Paint	1.0×10^{-3}	18.0×10^{-3}
Surfacer[(a)]	1.0×10^{-2}	1.25×10^{-1}
Concrete	5.0×10^{-2}	3.0

TABLE 6B-1 (SHEET 2 OF 5)

Heat Sink No. 4 - Galvanized Steel

Material	Node Spacing (in.)	Thickness (in.)
Zinc	6.7×10^{-4}	3.35×10^{-3}
Carbon steel	6.5×10^{-3}	6.56×10^{-2}

Heat Sink No. 5 - Miscellaneous Steel Less than 0.12 in. Thick

Material	Node Spacing (in.)	Thickness (in.)
Paint	1.0×10^{-3}	2.0×10^{-2}
Primer[(a)]	1.0×10^{-3}	3.0×10^{-3}
Steel	5.0×10^{-3}	7.64×10^{-2}

Heat Sink No. 6 - Miscellaneous Steel 0.12 to 0.16 in. Thick

Material	Node Spacing (in.)	Thickness (in.)
Paint	1.0×10^{-3}	2.0×10^{-2}
Primer[(a)]	1.0×10^{-3}	3.0×10^{-3}
Steel	5.0×10^{-3}	1.32×10^{-1}

Heat Sink No. 7 - Miscellaneous Steel 0.16 to 0.24 in. Thick

Material	Node Spacing (in.)	Thickness (in.)
Paint	1.0×10^{-3}	2.0×10^{-2}
Primer[(a)]	1.0×10^{-3}	3.0×10^{-3}
Steel	5.0×10^{-3}	1.91×10^{-1}

TABLE 6B-1 (SHEET 3 OF 5)

Heat Sink No. 8 - Miscellaneous Steel 0.24 to 0.30 in. Thick

Material	Node Spacing (in.)	Thickness (in.)
Paint	1.0×10^{-3}	2.0×10^{-2}
Primer[(a)]	1.0×10^{-3}	3.0×10^{-3}
Steel	5.0×10^{-3}	2.55×10^{-1}

Heat Sink No. 9 - Miscellaneous Steel 0.30 to 0.40 in. Thick

Material	Node Spacing (in.)	Thickness (in.)
Paint	1.0×10^{-3}	2.0×10^{-2}
Primer[(a)]	1.0×10^{-3}	3.0×10^{-3}
Steel	5.0×10^{-3}	3.38×10^{-1}

Heat Sink No. 10 - Miscellaneous Steel 0.40 to 0.50 in. Thick

Material	Node Spacing (in.)	Thickness (in.)
Paint	1.0×10^{-3}	2.0×10^{-2}
Primer[(a)]	1.0×10^{-3}	3.0×10^{-3}
Steel	1.0×10^{-2}	4.92×10^{-1}

Heat Sink No. 11 - Miscellaneous Steel 0.50 to 0.625 in. Thick

Material	Node Spacing (in.)	Thickness (in.)
Paint	1.0×10^{-3}	2.0×10^{-2}
Primer[(a)]	1.0×10^{-3}	3.0×10^{-3}
Steel	1.0×10^{-2}	5.76×10^{-1}

TABLE 6B-1 (SHEET 4 OF 5)

Heat Sink No. 12 - Miscellaneous Steel 0.625 to 0.75 in. Thick

Material	Node Spacing (in.)	Thickness (in.)
Paint	1.0×10^{-3}	2.0×10^{-2}
Primer[(a)]	1.0×10^{-3}	3.0×10^{-3}
Steel	1.0×10^{-2}	7.24×10^{-1}

Heat Sink No. 13 - Miscellaneous Steel 0.75 to 1.0 in. Thick

Material	Node Spacing (in.)	Thickness (in.)
Paint	1.0×10^{-3}	2.0×10^{-2}
Primer[(a)]	1.0×10^{-3}	3.0×10^{-3}
Steel	1.5×10^{-2}	9.35×10^{-1}

Heat Sink No. 14 - Miscellaneous Steel 1.0 to 1.5 in. Thick

Material	Node Spacing (in.)	Thickness (in.)
Paint	1.0×10^{-3}	2.0×10^{-2}
Primer[(a)]	1.0×10^{-3}	3.0×10^{-3}
Steel	2.0×10^{-2}	1.43

Heat Sink No. 15 - Miscellaneous Steel Greater than 1.5 in. Thick

Material	Node Spacing (in.)	Thickness (in.)
Paint	1.0×10^{-3}	2.0×10^{-2}
Primer[(a)]	1.0×10^{-3}	3.0×10^{-3}
Steel	3.5×10^{-2}	2.85

TABLE 6B-1 (SHEET 5 OF 5)

Heat Sink No. 16 - Stainless Steel

Material	Node Spacing (in.)	Thickness (in.)
Stainless steel	5.0×10^{-3}	1.68×10^{-1}

a. When Amercoat 90 is used as the primer, the average primer thickness will be 5.0 mils. However, the total thickness of primer plus finish coat will not exceed the total thickness of finish coat plus primer (surfacer) listed in the table.

TABLE 6B-2

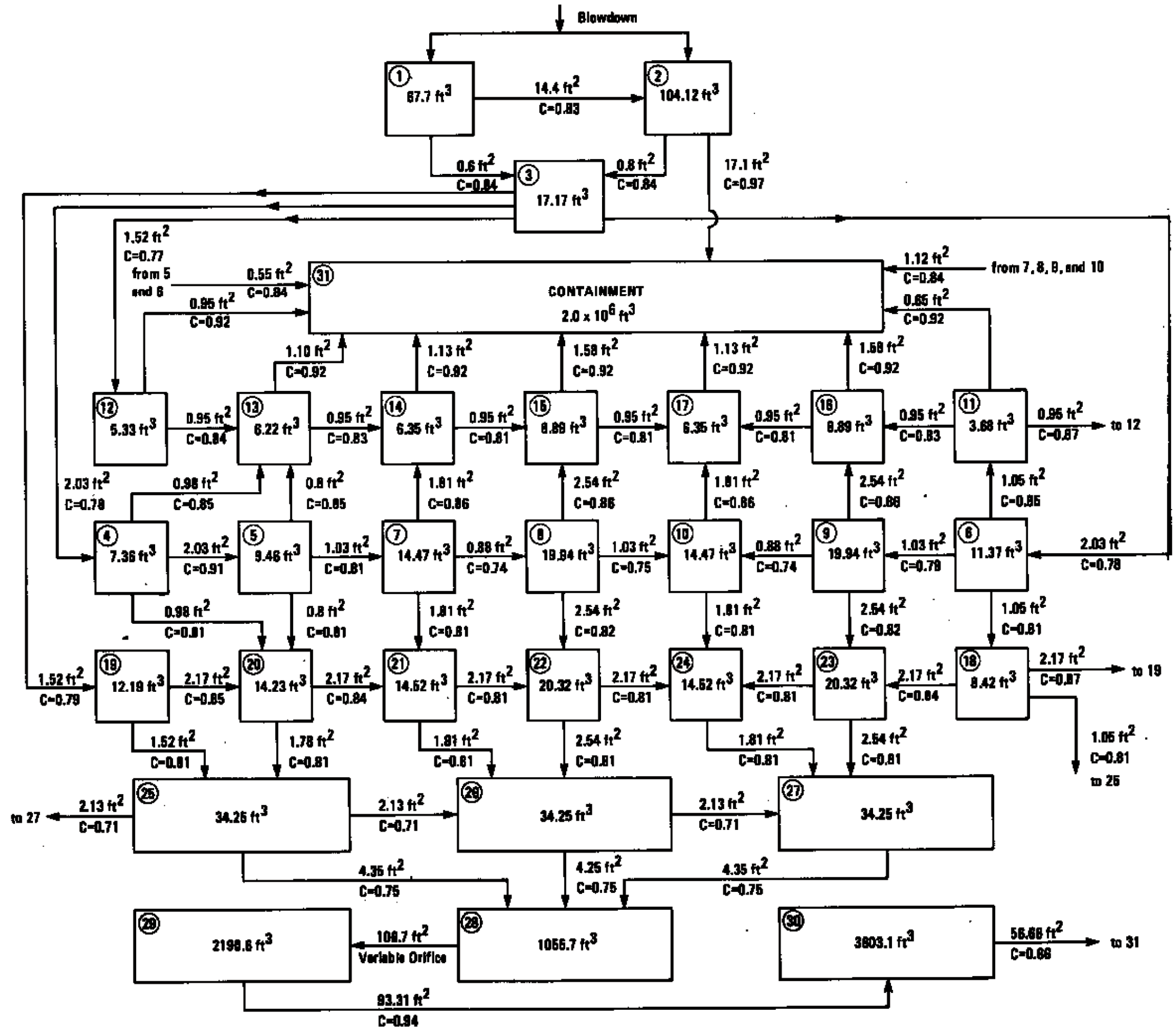
THICKNESS OF SENSITIVE HEAT CONDUCTION LAYER

Typical Materials	Heat Conduction Time 20 s	100 s	200 s	400 s
Concrete A K = 1.0 ρCp = 25.2	0.054 ft	0.121 ft	0.170 ft	0.243 ft
Steel K = 29.6 ρCp = 53.6	0.200 ft	0.450 ft	0.640 ft	0.906 ft
Inorganic Zinc Primer K = 1.24 ρCp = 27.36	0.058 ft	0.130 ft	0.184 ft	0.260 ft

TABLE 6B-3

MESH SPACING IN SENSITIVE LAYER TO ACHIEVE 0.5 PERCENT ACCURACY

Typical Materials	Accuracy Crossover Time 20 s	40 s	100 s	200 s	400 s
Concrete A K = 1.0 ρCP = 25.2	224 mesh ft	158 mesh ft	100 mesh ft	71 mesh ft	50 mesh ft
Steel K = 29.6 ρCp = 53.6	60 mesh ft	42 mesh ft	27 mesh ft	19 mesh ft	13 mesh ft
Inorganic zinc primer K = 1.24 ρCp = 27.36	210 mesh ft	148 mesh ft	94 mesh ft	66 mesh ft	47 mesh ft



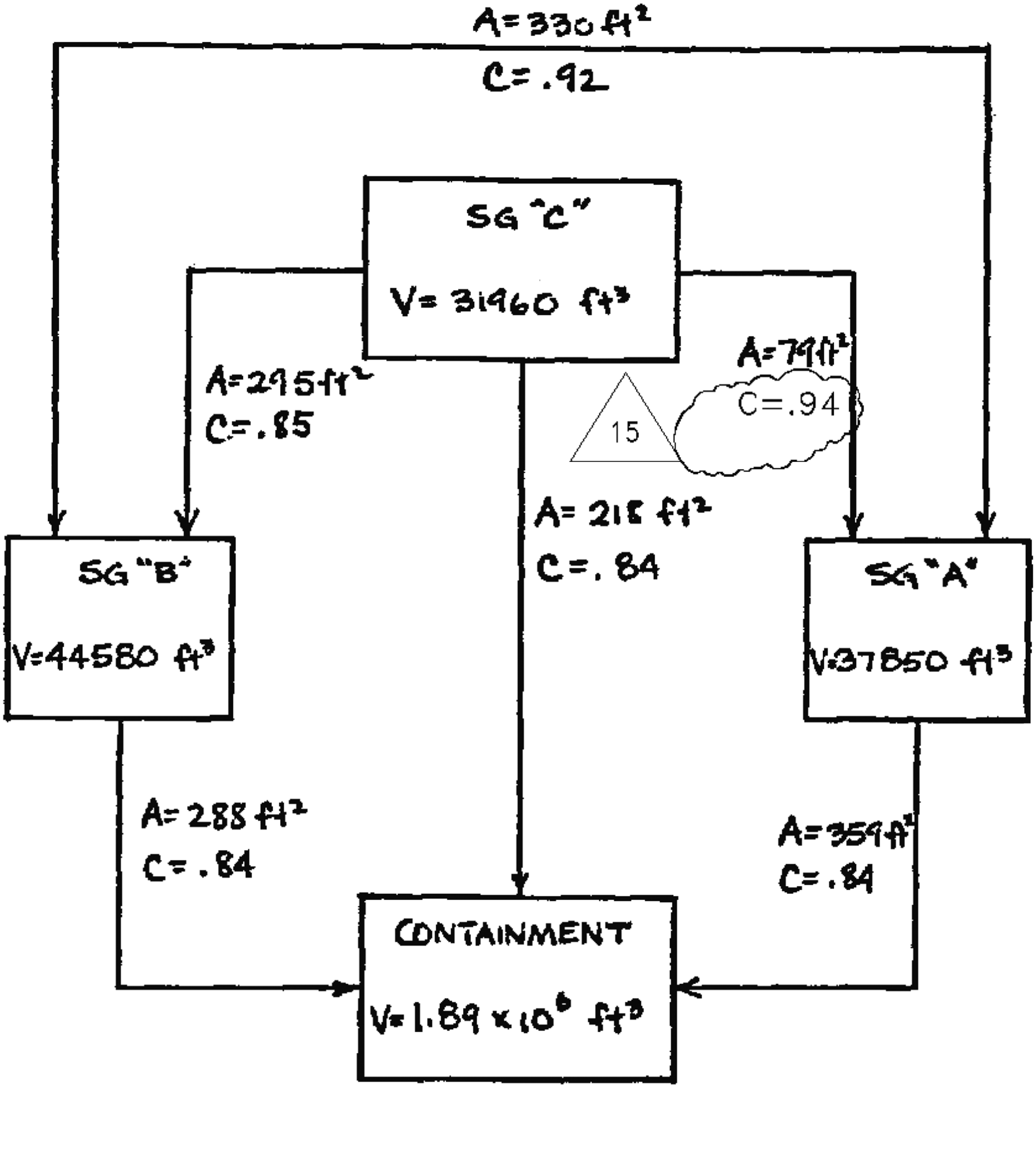
REV 21 5/08

SOUTHERN COMPANY
Energy to Serve Your World®

JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

REACTOR CAVITY BLOCK DIAGRAM

FIGURE 6B-1



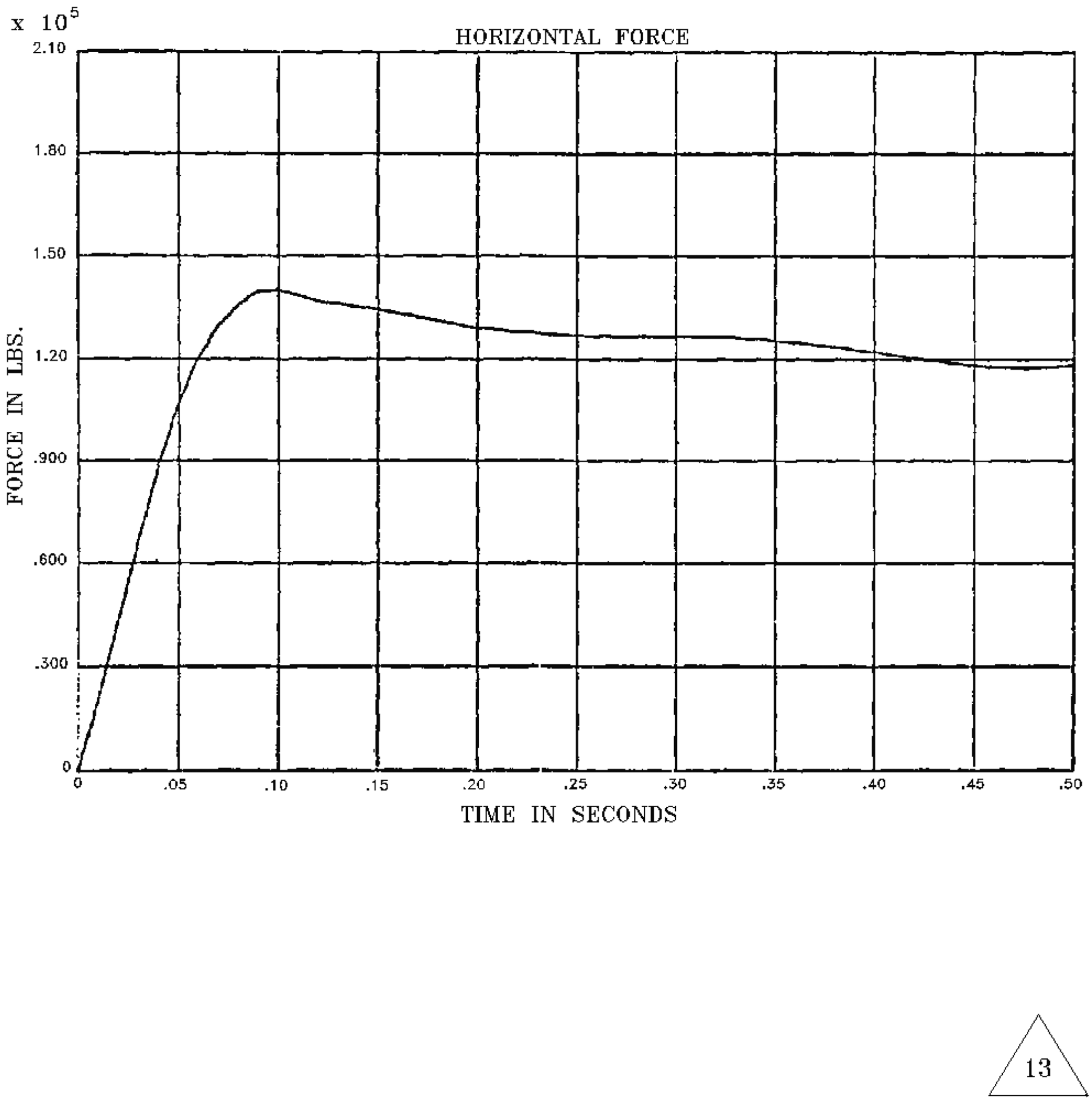
REV 21 5/08

SOUTHERN COMPANY
Energy to Serve Your World®

JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

STEAM GENERATOR CAVITY PRESSURIZATION ANALYSIS

FIGURE 6B-2

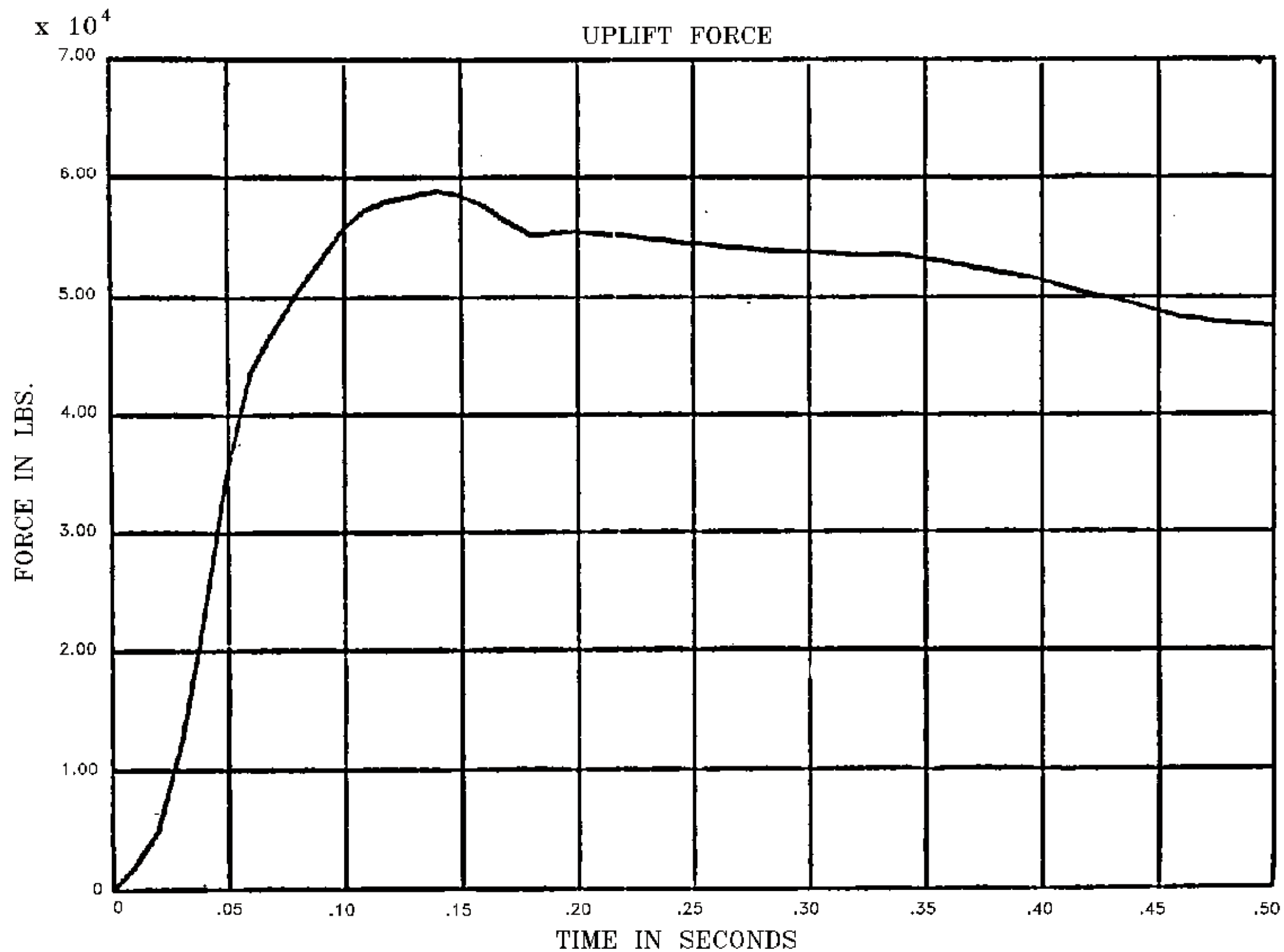


REV 21 5/08

JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

TOTAL HORIZONTAL FORCE VERSUS TIME

FIGURE 6B-3



13

REV 21 5/08

SOUTHERN COMPANY
Energy to Serve Your World®

JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

REACTOR CAVITY ANALYSIS

FIGURE 6B-4

[HISTORICAL (Prior to December 2007)] [APPENDIX 6C CONTAINMENT SUMP DESCRIPTION AND EMERGENCY CORE COOLING SYSTEM RECIRCULATION MODE TEST PROGRAM

TABLE OF CONTENTS

LIST OF TABLES

LIST OF FIGURES

[HISTORICAL (Prior to December 2007][APPENDIX 6C

CONTAINMENT SUMP DESCRIPTION AND EMERGENCY CORE COOLING SYSTEM RECIRCULATION MODE TEST PROGRAM]

Appendix C was made historical in December 2007 following the installation of new containment sump strainers for RHR and CS suction inlets. This was required by Generic Letter (GL) 2004-02, "Potential Impact for Debris Blockage on Emergency Recirculation During Design Basis Accidents at Pressurized-Water Reactors." Appendix 6D has been created to provide a description of the containment sump and the new suction strainers.

Appendix 6C contains the design bases for the original containment sumps and is being maintained for historical reference.

[HISTORICAL] [I. <u>CONTAINMENT SUMP DESCRIPTION</u>

The containment recirculation sump is a collecting reservoir designed to provide an adequate supply of water, with a minimum amount of particulate matter, to the containment spray system (CSS) and the residual heat removal system (RHRS). The containment sump performance meets the NRC acceptance criteria contained in General Design Criteria 35, 36, and 37, and the five NRC acceptance criteria listed below.

A. The net positive suction head (NPSH) available to each safety system pump has been shown to provide adequate margin over the required NPSH at limiting runout conditions (see FSAR paragraph 6.3.2.14).

B. Housekeeping requirements specified in the quality assurance program and the Technical Requirements Manual.

C. The avoidance of materials likely to form debris small enough to pass through sump screens.

D. The lack of an apparent mechanism for generating debris large enough to block more than 50 percent of the screen area.

E. The ability to monitor and control RHRS status.

The design criteria for the containment sumps and sump screens are the following:

A. Separate sumps are provided to serve each of the redundant halves of the ECCS and CSS. The redundant sumps are physically separated from each other and are located outside the missile barrier. The sumps are located on the lowest floor elevation in the containment, exclusive of the reactor vessel cavity.

B. The Unit 1 sump intakes are protected by an outer trash rack and a fine mesh inner screen with a steel grating support. The size of the openings in the fine screen take into account the overall operability of the system served.

C. A solid plate covers most of the top of each screen structure. This plate can be removed to facilitate inspection of the structure and pump suction intake. The top deck will be fully submerged after a LOCA and completion of safety injection.

D. Materials for the grating and screens were selected to avoid degradation during periods of inactivity and operation and have a low sensitivity to adverse effects, such as stress corrosion that may be induced by the chemically reactive spray during loss-of-coolant accident (LOCA) conditions.

E. A vortex breaker is provided at the sump intake end of each of the pump suction pipes.

F. The sumps are designed to yield low velocities of approach in the vicinity of the sumps to promote the settling out of debris, and to yield negligible pressure drops through the sump screens. Materials inside containment which could cause sump screen blockage post-LOCA have been eliminated or minimized by design.

G. The screens and associated structures have been designed to withstand the vibratory motion of seismic events without loss of structural integrity.

H. Each pump suction line is installed with a continuous slope from the sump to the pump to assure free venting of air. (See figure 6C-1.) There is a sufficient time interval before start of the recirculation phase to allow complete venting of the suction lines (approximately 30 min).

I. Field tests have been performed on the pump suction lines for two purposes: to flush the lines to remove any possible obstructions, and to verify pressure drop calculations made for pump NPSH requirements. The tests were run with the pump startup strainers in place.

A typical sump detail drawing prior to modification is shown on figure 6C-8.]

In each of the four pump suction lines from the containment sump, there are two motor-operated gate valves. There is no interdependency between systems or between the redundant portions of the same system.

The motor-operated gate valves in the lines from the containment sump to the various pumps are normally closed and remain closed during the injection phase of emergency core cooling system (ECCS) operation. The protective screened structures in the containment sump will be completely submerged at the end of the injection phase and will remain submerged during the recirculation phase.
The various parameters (e.g., flowrates, pressure drops, sump levels, etc.) listed in the following sections are from the original ECCS and CSS recirculation mode testing. The ECCS and CSS flowrates and sump levels utilized in the current pump NPSH calculations are within the range of flowrates and sump levels tested in the original sump recirculation tests. The pressure drop across the sump screen, vortex breaker, sump inlet, and suction piping utilized in the current NPSH calculations have been developed from the original sump recirculation test program and the ECCS field tests based on the calculated ECCS and CSS flowrates. Since the current parameters utilized in the NPSH calculations are bounded by those in the original sump recirculation tests, the ECCS and CSS sump intake design will not develop

II. ECCS RECIRCULATION MODE TEST PROGRAM

A. PURPOSE

The purpose of this hydraulic model study is to document that the ECCS intakes, of the J. M. Farley Nuclear Plant Units 1 and 2 will not develop unacceptable flow reducing or air entraining vortices.

The Unit 1 intakes were tested first. The model boundaries were placed remotely from the screen grating structures around the intakes and selected so as to be able to reproduce the flows in the area external to the intakes. Based on the findings from these tests it was concluded that it was not necessary to model the area outside the screen-grating structure for Unit 2. A description of the intakes, the test program and the results and conclusions for each unit are presented in the following sections.

III. UNIT I TESTS

A. INTRODUCTION

The emergency core cooling system intakes of Unit 1 are comprised of two 14 in. and two 10 in. vertical inlets located in three intake areas and are designated as intakes 1, 2, 3 and 4, as shown in figure 6C-2.

This section presents the results of testing the 14-inch nominal diameter intakes 1 and 2 and the 10-inch nominal diameter intakes 3 and 4 of Unit 1.

The tests were conducted to examine NRC's concern relative to the potential occurrence of vortices near or in the intake areas, which could result in loss of pumping capacity or pump failure due to vibration. Such occurrences could reduce pumping capacity by air entrainment and/or by unacceptably high intake head losses. Air entrainment could also produce unbalanced pressures on the pump impeller and cause pump failure because vibration. Therefore, a satisfactory intake design should be free of air entraining vortices and have acceptable intake loss coefficients.

Lack of published and documented information relative to effects of the complex flow patterns approaching the intakes, the grating and screens, and the low viscosity of the heated water precluded analytical or empirical predictions as to whether the intake configuration would be free from objectionable vortex action. The plant conditions do not permit inplace testing. Therefore, a hydraulic model was selected to evaluate the adequacy of the intake design with respect to vortices.

Drawing D-175200 shows the general features of the containment sump which could affect the flow of water to the sump area.

The elevator shaft in the area of the emergency cooling intakes, figure 6C-2, provided a natural model boundary and facilitated the examination of Intake 1 separately from Intakes 2, 3 and 4.

Intakes 1 and 2 design flows range from 3000 to 5900 gal/min. The 5900 gal/min corresponds to two residual heat removal (RHR) pumps taking suction through a single sump line. Intakes 3 and 4 have a design flow rate of 3050 gal/min each. The accident condition postulates that under certain conditions flow could approach the intake from both sides. However, for the majority of cases flow from the left (Q1) would exceed the intake 1 flow rate resulting in flow passing this intake toward Intakes 2, 3, and 4. (See figure 6C-2.) The calculated minimum and maximum water levels in the containment are 58.3 and 77.1 inches, respectively, above the floor. The maximum containment sump water temperature during recirculation following a postulated LOCA is 212°F at subcooled pressures. A maximum water temperature of 240°F was assumed for the model study.

B. THE MODEL

1. General

First, Intake 1 was modeled at a 1:1 undistorted scale within a 25 ft wide, 60 ft long, 12 ft deep concrete tank. Then Intakes 2, 3, and 4 were modeled in the same concrete tank. All columns, restraints, and piping greater than 2 in. diameter were represented in the model. (See figures 6C-3, 7, 10, and 11.) The protective screen and grating structure was constructed in accordance with figure 6C-5 and was modified as shown in figures 6C-5, 6, and 12. The screen cloth consisted of 0.120 in. wire with an effective opening of 51.6 percent. The screen was sandwiched between grating of 1-1/4 in. by 3/16 in. bars on 1-3/16 in. centers.

Flow baffles were placed at the extremities of the modeled area to insure uniform flow at the model boundaries. Viewing ports were incorporated in the tank to permit observation of flow conditions within the screen area around the intake. Piezometers were installed to measure static pressures inside and outside of the intake screens. Piezometer taps were installed initially at 5 pipe diameters downstream in the Intake 1 pipe and later at 29 pipe diameters downstream in the same intake pipe. They were also installed at 39.6, 36.7, and 25.7 pipe diameters downstream in the Intake 2, 3, and 4 pipes, respectively. Later, an additional tap was installed at 25.6 pipe diameters downstream of Intake 3.

The model was capable of being operated at 50 percent above prototype velocities and up to temperatures of 180°-190°F.

2. Scale Selection

The 1:1 scale was chosen in order to test the intakes under conditions which were as close to postulated LOCA conditions as practically possible.

The study of fluid dynamics has shown that the parameters which affect vortex formations may be represented by the following dimensionless numbers:

a. *Weber number,* $\frac{V^2D}{\sigma / \rho}$*, which is the ratio of surface tension to inertia forces.*

b. *Froude number* $\frac{V}{\sqrt{gD}}$*, which is a ratio of gravity to inertia forces.*

c. *Reynolds number,* $\frac{VD}{\upsilon}$ *, which is the ratio of viscous to inertia forces.*

d. *Circulation number,* $\frac{2\pi RVr}{Q}$ *, or the similar Kolf number, which characterizes circulation.*

e. *Strouhal number,* $\frac{f_eD}{V}$*, which characterizes the frequency of eddy shedding.*

The parameters identified in the preceding dimensionless numbers are:

r - *radius of inlet, ft.*
D - *characteristic length, ft., e.g., depth or diameter*
R - *radius of tank or perhaps flow streamline, ft.*
Q - *discharge,* ft^3/s
V - *characteristic velocity, ft/s*
fe - *frequency of eddy shedding,* s^{-1}
g - *gravitational acceleration,* ft/s^2
σ - *surface tension, lb/ft.*
υ - *kinematic viscosity,* ft^2/s
ρ- *Density,* $slugs/ft^3$

To reproduce exact dynamic and kinematic similarity on a geometrically similar model would require the value of all dimensionless numbers to be the same in model and prototype. The 1:1 scale model, and the test program which followed, permitted tests to be conducted at prototype values of all numbers, but not simultaneously. Conducting tests at prototype discharges and 170°F - 190°F temperatures reproduced the Froude, Circulation and Strouhal numbers with Reynolds and Weber numbers being lower than prototype values. Augmenting the discharges to reproduce prototype Reynolds number yielded Froude, Circulation, Strouhal and Weber numbers in the model which were higher than prototype values.

Since the Froude number involves the principal parameters related to surface flow phenomena, conducting the tests at prototype discharges establishes the surface flow characteristics outside the screen area concurrent with correctly simulated circulation and eddy shedding (Strouhal number) effects. At *equivalent Froude numbers, the model Weber number was less than the prototype value.*

Flow conditions within the screen area are independent of the Froude number and primarily dependent upon the Reynolds and Circulation numbers. Hence, conducting tests at prototype Reynolds numbers permitted examination of conditions within the screen area, concurrent with the Circulation and Weber numbers being greater than prototype values. Based upon the work of Dagget and Keulegan (reference 18), increasing the Circulation number for a constant Reynolds number increases vortex action. Hence it was considered conservative to conduct tests at prototype Reynolds numbers.

Furthermore with Reynolds number equivalence, the model Weber number was greater than the prototype value, which together with the unaugmented flow tests bracketed the prototype Weber number.

Therefore the 1:1 scale model, with tests conducted at and above prototype discharges, reproduced or exceeded the prototype values of the relevant dimensionless numbers. Exceeding prototype values of the dimensionless numbers was considered to produce conservative results.

C. THE MODEL TESTING PROGRAM

The tests examined the performance of Intake 1 over the range of flow conditions and water levels given in table 6C-1, for an unblocked condition and for the five postulated blockage conditions shown on figure 6C-4. These conditions were postulated by considering the nature of debris that could reach the screen, and the paths of the flow approaching the screens. Flow directions for Q_1 and Q_2 are indicated on figure 6C-2.

Tests 1 to 6, table 6C-1, were conducted with and without discharges augmented to develop Reynolds numbers equal to, or larger than, prototype values. A preliminary set of runs was also made on Tests 1 to 6 at prototype discharges, without blockage, to:

1. *Establish the general performance characteristics of the intake.*

2. *Observe surface flow conditions at Froude number equivalence between model and prototype.*

3. *Establish a basis for comparison of surface flow conditions with conditions at augmented discharges.*

Tests 7 and 8 were to be conducted with and without blockage at prototype discharges and water temperatures of 170°F. There was full prototype equivalence for these two tests.

The tests also examined the performance of Intakes 2, 3, and 4 over the range of flow conditions and water levels given in table 6C-2, for an unblocked condition, and for the five postulated blockage conditions shown on figure 6C-13. Flow directions for Q_1 and Q_2 are indicated on figure 6C-2.

Tests 1 to 6 and 8 to 10, table 6C-2, were conducted with and without, blockage and with water temperatures of 180°F or greater and prototype discharges augmented to develop Reynolds numbers equal to or larger than prototype values.

Test 7, table 6C-2, was conducted with and without blockage at prototype discharges and water temperatures of 180°F. There was full prototype equivalence for this test.

D. MODEL TEST RESULTS

INTAKE 1

1. General

Preliminary tests with the screen grating and Intake 1 design shown in figure 6C-8 indicated that air became trapped underneath the cover plate either during filling or upon coming out of solution due to heating. With an air pocket present, a vortex tended to form beneath the cover plate which immediately withdrew the air into the intake. To minimize the accumulation of trapped air under the plate, the modifications shown in figure 6C-5 were made. The solid cover plate of the screen structure was given a slope of 2 inches over its length and the plate was shortened 1/2 inch to provide a vent slot next to the secondary shield wall.

The initial test documentation was made with the intake design of figure 6C-5, for test conditions shown in table 6C-1. These tests indicated that blockage condition 5 created flow conditions within the screen area which generated a horizontally oriented vortex which originated at the secondary shield wall inside the screen and which entered the nearest quadrant of the inlet cruciform. A further modification consisting of the grating skirt shown in Figure 6C-6 was developed to eliminate the penetration of this vortex into the intake. A final series of tests was conducted for the configuration shown in figure 6C-6. The results of the preliminary, initial and final tests are presented below.

2. Preliminary Tests

The preliminary tests were run with unblocked screens at prototype velocities and at velocities increased to produce prototype Reynolds numbers.

These tests established that:

a. The proposed design could trap air under the solid cover plate which would lead to the formation of an air core vortex within the screen area that very quickly exhausted the trapped air.

b. There was no vortex formed outside of the screen structure.

c. There was no observable difference between flow patterns at prototype and augmented velocities. Hence, there was no distortion of flow patterns when departing from Froude similitude at augmented flows.

3. *Initial Documentation Tests*

The results from the initial documentation tests are as follows:

a. *There were no vortices under any test condition which established an air core from the free surface within the containment area to the screen grating around the intake. Surface depressions in the eye of any eddies or organized circulation did not exceed 1/2 inch in depth.*

b. *Air introduced artificially under the cover plate above the intake, which represented air trapped during flooding of the containment of coming out of solution, was able to escape through the 1/2 inch vent slot in the solid cover plate for all prototype test conditions. Air could remain trapped below the plate for short periods of time. This air swirled above the intake but no air was drawn into the intake irrespective of the quantity of air forced beneath the plate.*

c. *No organized circulation or vortices were observed within the screen area around the intake for an unblocked screen, nor for blockage conditions 1 through 4.*

Organized circulation did develop for blockage condition 5 with the strength of circulation being a function of the intake flow. The axis of circulation was horizontal and originated near the shield wall, approximately 9 inches to the left of the screen cage center. It curved into the intake quadrant nearest the left side and the shield wall. This condition could first be noticed at intake flows of about 4,000 gal/min. As the flow increased (and thus the pressure in the core of rotation decreased) an intermittent vapor core developed. Above discharges of about 4,300 gal/min a continuous vapor core 1/16 to 1/8 inch in diameter was present. This condition did not result in a measurable increase in intake head loss.

d. *The maximum measured head loss across the screen and grating corrected to prototype discharge was 0.09 ft.*

The intake loss coefficient was computed from the equation

$$K = \frac{\Delta h - V^2 / 2g}{V^2 / 2g}$$

Δh = pressure drop in feet of water from inside the screen to a pressure tap down stream from the pipe inlet (5.7 feet for Intake 1)

V = average flow velocity in the 14 inch diameter pipe.

The intake loss coefficient varied between 0.34 and 0.39 with no trend in the values with blockage cases or flow rates. The consistency of the loss *coefficients indicated that no discernible flow reduction developed because of circulation or vortex action within the screen area.*

4. *Final Documentation Tests*

An octagonally shaped grating skirt of 1 1/4-in. by 3/16-in. bars on 1 3/16-in. centers was placed around the intake within the screened area to eliminate the horizontal vortex which developed inside of the screen structure during initial tests. The intake design is shown on figure 6C-6.

Since the objectionable vortex action only occurred for blockage condition 5, final tests were only conducted for this case. The full range of test conditions, tests 1 through 8, table 1, were documented at water temperatures of 170°F, or greater.

The test results were as follows:

a. *A weak circulation with a horizontal axis was observed at the location where the vapor core developed during initial tests. However, no vapor core formed for any test condition. As noted previously, there were no vortices formed within the screen area for any other blockage condition.*

b. *The intake loss coefficient, as previously defined, remained between 0.34 and 0.39.*

Additional measurements were made utilizing piezometric taps at 29 diameters downstream of the intake to further isolate the full loss of the intake and the elbow, which has a centerline radius of about 1.5 times the pipe diameter. The following formula, which includes the correction for pipe friction, was used:

$$K = \frac{\Delta h - h_f - \frac{V^2}{2g}}{\frac{V^2}{2g}}$$

where h_f is the computed pressure drop in feet of water due to pipe friction above, based on the estimated pipe surface roughness, height, and other terms as defined before. For intake discharges from 3060 to 7915 gal/min, the loss coefficient for the intake and the elbow ranged from 0.46 to 0.48.

<u>Intakes 2, 3, and 4</u>

The documented tests were conducted with the improved screen and grating and grating cages placed over Intakes 2, 3, and 4 as shown in figures 6C-12 and 14.

The full range of test conditions, tests 1 through 10, table 6C-2, with unblocked, and blocked screens were run and documented at water temperatures of 180°F or greater.

The initial test results indicated a substantially higher loss coefficient for intake plus bend for Intake 3 than for Intakes 2 and 4. The loss coefficient was computed from the equation given in subsection D.3.d, above.

An inspection revealed that the model pipe walls of Intake 3 had been severely corroded by the hot water, with the tuberculated pipe indicating protrusions measuring 1/32 to 1/16 inch. An additional pressure tap was installed to record the actual pressure loss over a 9.51-foot section of pipe. The h value of Intake 3 was then calculated from the measured pressure drop values in the pipe.

The test results were as follows:

a. *There were no vortices under any test condition which established an air core from the free surface within the containment area to the screen grating around the intakes. Surface depressions in the eye of any eddies or organized circulation did not exceed 1/2 inch in depth.*

b. *No vortices were observed inside or outside the screen structure.*

c. *Air introduced artificially under the cover plate above the intake, which represented air trapped during flooding of the containment or coming out of solution, was able to escape through the 1/2-inch vent slot in the cover plate. At augmented discharges, pockets of air would remain trapped below the plate. This air swirled above the intake, but no air was drawn into the intake irrespective of the quantity of air forced beneath the plate. At prototype discharges, this air was able to escape through the 1/2-inch slot, and only a few small bubbles remained.*

d. *The maximum measured head loss across the screen and grating corrected to prototype discharge of 5900 gal/min for Intake 2 and 3050 gal/min for Intake 3 was 0.14 foot for Intakes 2 and 3 and 0.04 foot for Intake 4 with a discharge of 3050 gal/min.*

   The intake plus bend loss coefficients varied from

   0.36 to 0.46 for Intake 2

   0.38 to 0.45 for Intake 3

   0.33 to 0.40 for Intake 4,

   with no trend in the values with blockage cases or flow rates. The model indicated maximum combined prototype losses due to screen, intake, and bend of

1.50 ft for Intake 2

1.25 ft for Intake 3

1.01 ft for Intake 4.

The higher combined losses associated with Intakes 2 and 3 were attributed to larger flow per unit area approaching the intakes and the more turbulent approach condition resulting from the proximity of these intakes to the elevator shaft.

E. FIELD TEST

Several preoperational tests were performed at the Joseph M. Farley Nuclear Plant, Unit 1 to determine the actual piping resistance of the residual heat removal (RHR) pump sump suction lines. The testing revealed that the maximum expected RHR pump flow during the post-LOCA cold leg recirculation mode with only one RHR pump in operation was 5000 gal/min for Pump A and 4875 gal/min for Pump B. The actual net positive suction head (NPSH) available to each RHR pump from the containment sump was determined to be 18.4 feet at 5000 gal/min without taking credit for subcooling of the water in the containment sump and based upon the most resistive sump piping (25.2 feet elevation head from the sump and 3.9 feet of water above the sump line inlet less 10.7 feet of losses). The NPSH required for the RHR pump is 18.5 feet at 5000 gal/min and 18.0 feet at 4875 gal/min. This indicated that the NPSH available to the RHR pumps under worst case conditions would be marginal during the post-LOCA recirculation phase.

Upon the completion of additional tests confirming the resistance of the installed piping system, the RHR system resistance was increased to assure that adequate NPSH is available and that system performance is satisfactory during all operating modes. The system resistance was increased by physically restricting the maximum opening of valves HCV-603A and B on the outlet piping of the RHR heat exchangers and by addition of flow restriction orifices in each of the three cold leg low head safety injection lines. System tests conducted after these modifications show that the maximum flowrate with one pump operating during the cold leg recirculation mode of operation would be approximately 4200 gal/min. The NPSH available for RHR Pumps A and B utilizing simulated recirculation mode plant test data, at this flowrate, is 17.7 feet (25.2 feet elevation head from the sump less 7.5. feet of losses) and 19.2 feet (25.2 feet elevation head from the sump less 6.0 feet of losses) respectively. The NPSH required for the RHR pump is 15.0 feet at 4200 gal/min. Thus, adequate NPSH is assured. These calculations take no credit for water above the containment sump line inlet or for any subcooling of water in the containment sump. Evaluation of the postmodification tests also confirmed that ECCS flows would meet or exceed system requirements during all operating modes.

F. SUMMARY AND CONCLUSIONS

The 1:1 scale model of Intake 1, (figures 6C-5 and 6), which was tested at Reynolds numbers equal to or greater than prototype and with circulations which were greater *than prototype, indicated that the intake will operate without air entraining or flow reducing vortices.*

The maximum screen grating and intake losses computed from the model test results were 0.09 foot and 0.85 foot respectively at 5900 gal/min. These values were combined with field test data and compared with calculated data used in the NPSH evaluation.

The 1:1 scale model of Intakes 2, 3, and 4, (figure 6C-7), which was tested at Reynolds numbers equal to or greater than prototype and with circulations which were greater than prototype, indicated that the intake will operate without air entraining or flow reducing vortices.

The maximum losses determined from the model and field tests for each intake are:

	Pressure Drop (feet) ***Intake***			
Effect	*1*	*2*	*3*	*4*
Piping (from field data)(1, 2)	*5.32*	*3.86*	*6.28*	*6.89*
Inlet (from test data)(1)	*0.43*	*0.37*	*0.36*	*0.43*
(from test data)(1)	*1.49*	*1.49*	*2.39*	*2.39*
Screen (from test data)	*0.04*	*0.09*	*0.09*	*0.04*
Total	*7.28*	*5.81*	*9.12*	*9.75*

NOTES:

1. *Converted to 4200 gal/min base for Intakes 1 and 2 and to 3050 gal/min base for Intakes 3 and 4.*

2. *Includes additional losses due to 8 feet of test piping for Intakes 1 and 2 and additional losses due to 6 feet of test piping for Intakes 3 and 4.*

The measured head losses are less than the calculated losses of 8.4 feet for Intakes 1 and 2 and 9.9 feet for Intakes 3 and 4. (See subsection 6.3.2.14.)

Based on the results of Intake 1 tests, together with Intakes 2, 3, and 4 tests of Unit 1 and on similar work undertaken for other projects, it is the definite opinion of Western Canada Hydraulic Laboratories Ltd. and Bechtel that incorporating a grating cage similar to the above design, (figures 6C-6 and 12), will result in an intake design for all Units 1 and 2 intakes which will operate free from air entraining or flow reducing vortices.

IV. UNIT 2 TESTS

A. INTRODUCTION

This section presents the model test program undertaken and the results of these tests to ensure that Joseph M. Farley Unit 2 emergency core cooling and containment spray system recirculation intakes from the containment sump (floor) will operate without effects which could degrade the performance of the pumps in these systems.

Similar tests were conducted for the Farley Unit 1 containment sump intakes in which the containment geometry in the sump areas was modeled at a scale of 1:1 together with flow obstructions such as pipes, supports and valves, etc. around these intakes. These tests revealed that there were no air entraining vortices or flow reducing conditions at these intakes when these intakes were protected with the inner grating cage and the outer screen grating cage structure combination. This is discussed in detail in section II of this appendix.

The tests performed on containment sump intakes of Unit 1 and on other facilities provided strong evidence that the inner grating cage and the outer screen grating cage structure combination employed on Unit 1 was totally effective in destroying vortices ranging from pencil lead size to 1 inch in diameter or greater.

Based on these results, it was concluded that the tests on Unit 2 containment sump intakes can be effectively performed without modeling the containment geometry around the intakes as was done for Unit 1. A more detailed rationale for this approach along with a description of Unit 2 containment sump intakes, discussion of effects which could degrade pump performance, description of test facility and program, and the test result and conclusions are presented in this report.

B. DESCRIPTION OF UNIT 2 CONTAINMENT SUMP RECIRCULATION INTAKES

The emergency core cooling and containment spray system recirculation intakes for Unit 2 are comprised of two 14-inch and two 10-inch vertical inlets located in four separate intake areas on the containment floor and are designated as Intakes 1, 2, 3 and 4 as shown in figure 6C-19. Each intake is surrounded by a protective screen grating and grating cage structure as shown in figure 6C-17. The design flows for Intakes 1 and 2, which supply water to the RHR pumps, range from 3000 to 5900 gal/min each. The design flow rates for Intakes 3 and 4, which supply water to containment spray pumps, are 3050 gal/min each.

The calculated minimum and maximum water levels in the containment are 58.3 and 77.1 inches, respectively, above the floor.

The maximum expected containment sump water temperature during recirculation following a postulated LOCA is about 212°F at subcooled pressures.

The flowrates, water depths, water temperature, and the protective screen structure for Unit 2 are identical to Unit 1, except that four separate intake areas are provided for Unit 2 as compared to three intake areas for Unit 1. The elevator shaft in Unit 2 is *located outside the flow paths approaching the intakes. This will lead to a more uniform flow in the containment sump intake areas than that expected in Unit 1, where the elevator shaft is located in the containment sump intake area. Furthermore, the equipment layout at the Unit 2 containment floor elevation, as shown in figure 6C-20, is not expected to be significantly different from Unit 1, shown in figure 6C-21.*

C. PROBLEM DEFINITION

Regulatory Guide 1.82 states the position that "Pump intake locations in the sump should be carefully considered to prevent degrading effects, such as vortexing, on the pump performance." Two degrading actions are possible: ingestion of air (a vortex phenomenon), and/or intake entrance losses which are larger than design values used in establishing the required NPSH of the pumps.

Increased entrance loss can develop due to adverse flow approach conditions or free surface and internal vortex action.

1. Factors Causing Increased Entrance Losses

Intake losses are incurred due to contraction and expansion of the flow at the intake. The intake entrance losses are accounted for in the design of pumping systems by calculating the entrance loss based on established intake loss coefficients. Such coefficients are normally based upon measurements taken with uniform flow approaching the intake.

Intake head losses can be increased by high approach velocities, especially at an angle to the pipe axis and/or by strong circulation in the approach flow which results in an increased contraction of the flow at the intake.

Strong circulation can lead to vortex formation with a marked reduction in flow.

A full scale model is capable of indicating any head loss degrading effects for all conditions simulated and tested.

2. Factors Affecting Vortex Creation

Studies of vortex formation have been carried on by several investigators (see references in part II, G). The majority present test results as functions of the intake head loss coefficient, the depth of water at which the air core just penetrates the intake, the circulation numbers at which the air core just penetrates the intake, the Reynolds Number, or some variation of these parameters.

The performance of an intake, as represented by the head loss coefficient K, is usually described (Anwar 1968, Amphlett 1976, Chang 1976) as:

$K = F$ (local geometry, r_{max} RR, Γ_N, W)

where

local geometry = f (D, h, b)

$$R_R = \text{Radial Reynolds No. } \frac{Q}{\upsilon H}$$

$$\Gamma_N = \text{Circulation No.} = \frac{D\Gamma}{Q}$$

$$W = \text{Weber No. } \frac{Q^2}{\pi^2 D^4} h \frac{\rho}{\sigma}$$

r_{max}	=	*radius of the tank (or sump) in which the intake is located = maximum radius of circulation in the vicinity of the intake*
Q	=	*discharge*
D	=	*intake diameter*
b	=	*height of intake above sump floor*
ρ	=	*density of water*
ν	=	*surface tension of water*
σ	=	*surface tension of water*
h	=	*depth of submergence of intake*

Γ = *circulation strength = $2\pi Vtr$ where V is tangential velocity at radius r.*

Work by Daggett and Kuelegan (1974) and others have shown that for high Reynolds numbers (R_R >104) and moderate values of circulation ($\Gamma N \leq 2$), typical operation ranges for the Farley recirculation intakes, the effects of surface tension and viscosity are relatively small; i.e., W and RR are not important. In this case, the intake performance, and hence the formation of vortices, is a function of three parameters: the local geometry, the maximum circulation radius, and the strength of circulation of the approaching flow. Each of these factors is discussed in the following sections.

D. TEST PROGRAM

1. Rationale

As discussed in subsection C, the intake head losses may be increased by nonuniform flow and/or circulation in the approach flow into the intakes.

It is significant to note that ultimately it is the flow condition in the immediate vicinity of the intake pipe that establishes the intake head loss. This condition, and associated head losses, may or may not be affected by the flow conditions removed from the immediate vicinity of the intake pipe.

With an intake that is not protected by a screen grating cage, the flow in the immediate vicinity of the intake pipe will be established by the structural configuration of the containment and affected by the presence of flow obstructions such as valves, piping and restraints. The effect of these flow obstructions will increase with their increased proximity to the intake. Structural members may channelize the approach flow, affecting the approach flow directions and velocities. Channelization can also lead to a general circulation in the vicinity of the intakes, being bounded by the surrounding structures. Eddy-shedding will induce vorticity in the flow which can add to circulation in the vicinity of the intake.

Unquestionably, an intake that is not protected by a screen grating cage can only be tested with full representation of the structural configuration of the containment and valves, piping and restraints. However, the Farley Unit 2 containment sump recirculation intakes are to be covered and protected by a screen grating cage, comprised of a 0.047 inch screen wire with an effective opening of 51.6 percent, sandwiched between two layers of grating. The grating bars will be 1-1/4 inch by 3/16 inch on 1-3/16 inch centers, giving a total effective grating width in the direction of flow of 2-1/2 inch. (See figure 6C-17.) The inside grating bars are approximately 2 feet from the intake pipe. Furthermore, an inner grating cage will be placed over the intake pipe.

Due to the proximity of the Farley screen grating cage to the intake pipe, it was concluded that this structure would strongly influence, if not dominate, the approach flow into the intake pipes within the cage. This dominance was observed on full scale mockup tests of the Farley Unit 1 containment sump recirculation intakes where:

a. *The grating bars acted as flow straighteners and no angularity or circulation of flow approaching the screen cage, which could lead to the formation of a vortex, was transmitted through the structure, regardless of the angle of approach flow. Flow downstream from the grating exited at right angles to the plane of the grating.*

b. *The most nonuniform, rotational approach flow to the intake pipe, as evidenced by an air core vortex inside the cage, was developed by a partial screen blockage configuration. No vortex developed inside the screen cage without blockage.*

Hence, it was apparent from the Unit 1 tests that the approach conditions in the immediate vicinity of the intakes (within the screen grating cage) were established by the flow distribution through the screen grating cage. Angularity of flow approaching the outside of the screen grating was removed and any swirl or circulation inside the screen grating cage was due to a nonuniform flow *distribution through and normal to the plane of the grating. Furthermore, the most adverse velocity distribution inside the screen grating cage could be established by the blockage configuration imposed. These observations lead to the following considerations:*

a. *In the case of an intake not protected by a screen grating cage, the flow can be channelized at any angle by structural members. In the case of an intake covered by a screen grating cage, the screen grating and the blockage configuration impose the ultimate channelization, and establish the direction of flow normal to the plane of the grating.*

b. *Irrespective of the structural configuration external to the screen grating cage, and hence irrespective of the approach flow conditions this configuration imposes on an unblocked screen grating structure, there will be a blockage condition which will develop as adverse or a more adverse and potentially a more degrading effect on the intake performance. Hence, this proves that the grating gage will eliminate any vortex potential and would be proof that the potential developed by the external structural arrangement will be eliminated.*

c. *Because blockage conditions could establish potentially degrading conditions inside the screen grating cage, a grating cage must be incorporated inside the screen grating cage to remove circulation generated within the screen grating cage which could lead to the formation of vortices and/or increased intake head losses.*

Thus, based on the experience gained on the full scale mockup tests for Unit 1, the following rationale was applied to the test program for the Unit 2 containment sump recirculation intakes:

a. *Ultimately it is the flow condition in the immediate vicinity of the intake pipe that can lead to degrading effects of pump performance.*

b. *The immediate vicinity of the intakes will be covered by a screen grating cage.*

c. *If the screen grating cage does not transmit the angularity of circulation of flow outside of the cage, then flow conditions and air core vortex potential within the screen grating cage are established by the blockage conditions imposed (flow distribution), water depth (pressure inside the cage), intake discharge (velocities), and viscosity (fluid shear energy dissipation).*

d. *The fact the containment may be pressurized does not affect flow conditions. The flow field is established by pressure differentials which would be the same in a closed system irrespective of the air pressure on the water surface.*

e. *If angularity of approach flow is not transmitted through the screen grating cage, then the uniqueness of the flow distribution through an open screen established by the structural configuration and flow obstructions surrounding the screens represents one potential blockage condition.*

f. *Based on the above, it is necessary to model only the screen grating cage, and all features inside the cage, and demonstrate for postulated flow depths and flow rates that:*

   i. *The screen grating cage will not transmit the angularity or circulation of flows outside the cage.*

   ii. *Under adverse conditions generated by screen blockage, the grating cage over the intake inside the screen grating cage will preclude degrading effects on the performance of the recirculation pumps.*

g. *Furthermore, since circulation is an essential and necessary feature of a vortex, then irrespective of the strength of circulation, if flow circulation associated with a potential vortex is not transmitted through the screen grating cage, then the vortex formed outside of the cage cannot enter the intake pipe (as discussed in the Final Report on the Davis-Besse Nuclear Power Station ECCS Emergency Pumps and Pump Suction Line Testing, December 15, 1976).*

As discussed in subsection C, the formation of vortices is a function of three parameters: the local geometry, the maximum circulation radius, and the strength of circulation of the approaching flow.

Since full scale tests were to be conducted, the local geometry in the immediate vicinity of the intake would be correctly simulated. In addition, since all screen and grating characteristics would be correctly represented, all vortex and flow parameters from the screen grating structure inward to the intake would be correctly simulated, and the intake entrance losses would be correctly measured.

Swirls in the approach flow may vary with respect to the absolute size of the system, strength of circulation, velocity of translation, and travel path. The latter two parameters are of significance since, for a swirl to initiate an intake vortex, it must remain in the vicinity of the intake long enough to organize the circulation in the vicinity of the intake. Hence a stationary circulation directly above the intake becomes the critical case. The system size is of no concern when a 1:1 scale model is used. Thus there are two parameters which must be properly addressed: the maximum circulation radius (rmax), and the strength of circulation. Experimental evidence indicates that the critical submergence of the intake required to preclude the formation of air entraining vortices increases with both the maximum swirl radius, rmax, (Haindl, 1959) and strength of the initiating swirl (Amphlett, 1976), Daggett and Kuelegan, 1974; Springer & Peterson, 1969; Anwar, 1965).

From these experimental data, it can be concluded that a test procedure should include:

a. A stationary circulation directly over the intake.

b. A circulation strength equal to or greater than the largest reasonable values due to the expected prototype approach flow configuration.

c. A maximum circulation radius, rmax, equal to the largest reasonable value in the prototype, must be said to "bound" the effects developed y the plant geometry and structural members, etc., in the vicinity of the intakes which could lead to a vortex.

2. Objective

The prime objective of the test program was to demonstrate that the Farley Unit 2 containment sump recirculation intakes will not be subjected to degrading effects on pump performance, such as air ingestion or high intake head losses.

Achieving the following fulfilled the prime objective:

a. Documentation of the effectiveness of the grating cage over the intake in straightening the approach flow and removing imposed angularity or circulation which, without the grating cage present, could lead to an air entraining vortex.

b. Documenting that the screen grating cage removed angularity and circulation of approach flow outside of the cage.

Documentation of the effectiveness of the grating cage was achieved by:

i. Imposing on the grating cage, without the screen grating cage over it, a range of circulations, the largest of which was more massive than any circulation that could be developed by the geometry or the structural members of the containment or the presence of flow obstruction such as valves, piping and restraints.

ii. Imposing blockage conditions on the screen grating cage which generated potentially degrading flow conditions within the screen cage, and documenting that those conditions were eliminated by the grating cage.

Documentation of the effectiveness of the screen grating cage was achieved by:

i. Demonstrating that the single layer of grating on the grating cage was effective in removing angularity in the approach flow in the high velocity region close to the intake.

ii. *Demonstrating that no vortex, angularity or circulation of approach flow passed through the screen grating cage or grating cage when the screen grating cage was subjected to a range of circulation, the largest of which was more massive than any circulation that could be developed by the geometry or the structural members of the containment, or by the presence of flow obstructions such as valves, piping and restraints.*

E. TEST FACILITY

1. General

The plan view and a section of the experimental facility are shown in figures 6C-15 and 6C-16. Two source sumps, each containing a diffuser, provided the approach flow to the intake area within the concrete tank, (figure 6C-15). A sump floor, which was of 1/8 in. steel plate, was placed 4.5 feet above the tank floor to provide space for the 14 in. diameter intake piping and for an observation tunnel below the steel plate floor. (See figure 6C-16.)

The flows were distributed and controlled by means of two centrifugal pumps and a flow transmitting network of steel pipes, orifice meters with differential mercury manometers, and control valves. The direction and circulation of the approach flow was controlled by a system of 18-in.-wide vertical directional vanes, which extended over the full depth of the flow. (See figure 6C-16.)

Two 2,900,000 Btu/h gas heaters were used to heat the water to temperatures in excess of 180°F.

2. Intake Description

A cruciform and reducer section, which was shipped from the project site for use in the experimental facility, was mounted on the intake pipe 6 in. above the steel plate floor. The octagonal grating cage used for the Unit 1 model tests was modified to include a horizontal grating inside the grating gage, 3 in. above the floor, to eliminate potential floor vortices. (See figure 6C-18.) The grating cage thus totally encapsulated the intake pipe.

The whole assembly was enclosed by a steel screen grating cage with inside dimensions of 5-ft. x 5-ft. x 2-ft. 5-in. depth. (See figure 6C-17.) The floor below the screen grating was of acrylic plastic construction which, together with the portholes in the observation tunnel, permitted observation and lighting of the area inside the screen grating.

3. Test Cases and Procedures

a. Test Cases

The postulated post-LOCA condition and the condition for which the intake was tested are compared below:

	Postulated for Containment Sump post-LOCA	*Tested*
Minimum water depth (in.)	*58.3*	*24.0 to 58.3*
Maximum flow (gal/min)	*5900*	*6574 to 8524*
Water temperature (°F)	*212*	*61 to 184*
Maximum circulation (ft^2/s)	*3./for 58.3 in. water depth*	*8.5 to 10.7 for 58.3 in. water depth*
Maximum size of circulation cell (ft)	*17*	*18*
Pipe Reynolds Number	*4.7×10^6*	*1.5 to 5.7×10^6*

Considerable conservatism was incorporated in the test by:

i. *Conducting tests at greater than postulated flow rates.*

ii. *Conducting tests with screen blockage greater than 50 percent.*

iii. *Conducting tests at less than the minimum postulated water depths.*

iv. *Conducting tests with a circulation appreciably greater than the maximum value calculated for the plant during LOCA conditions.*

v. *Augmenting the postulated flows to develop Reynolds numbers in the test facility greater than postulated in the containment.*

Furthermore, model scale effects were reduced or eliminated by:

i. *Constructing the intake, grating cage, and screen grating cage at a 1:1 scale, thereby eliminating all scale effects introduced by modeling the screen and grating components.*

ii. *Conducting the tests with water heated to 180°F or greater.*

b. *Observations and Measurements*

All surface flow phenomena were observed from two platform decks. The lower deck was used to make all surface flow observations and to take velocity and temperature measurements. Observations could be made of flow phenomena inside the screen grating through the acrylic plastic cover plate. Video records were also made from this deck. Overhead photos were made from the upper platform deck.

Flow phenomena within the screen grating cage could be observed and recorded on video tape through the portholes in the observation tunnel.

Use was made of air bubbles injected into the screen grating area through the acrylic plastic floor for flow visualization. Dyes were used sparingly to preserve water clarity.

Flow measurements were obtained with the calibrated orifices and U-tube mercury manometers. Local velocities were measured with a Gurley propeller meter while surface velocities were obtained with the Gurley meter or from overhead photos of confetti traces.

Pipe intake and screen grating losses were determined from static pressure taps. Taps 1 and 2 were located in the supply sumps and indicated the water surface elevation. Tap 3 consisted of two interconnected taps in the floor inside the screen grating cage to produce an average pressure within the screened area. The mean static head indicated by Tap 3 therefore gave an indication of screen grating losses when compared to the mean water surface elevation from Taps 1 and 2. Two interconnected taps, each on the horizontal diameter, determined the average static pressure inside the intake pipe at each of four locations, at distances of 5.11, 13.11, 21.11, and 30.00 diameters downstream of the intake.

c. *Test Procedure*

The tests were conducted in two phases.

Phase 1 tests were related to documenting the effect of grating on approach flow conditions external to the screen grating cage.

For this series, without any screen or grating over the intake, given flows were set and the vanes surrounding the intake were adjusted to produce the maximum size vortex. The circulation, vortex size and pressure measurements were taken, together with observations of flow conditions in the immediate vicinity of the intake. This was done for both ambient and heated water. Tests at ambient water temperatures were conducted to facilitate the making of video records of the free surface flow conditions.

Without changing the vane angle, the tests were rerun and the results were documented with the addition of:

i. *Cruciform only.*

ii. *Grating cage and cruciform only.*

iii. *Screen grating cage and cruciform only.*

iv. *Screen grating cage, grating cage and cruciform.*

Phase 2 tests were related to documenting the effect of the grating cage on adverse flow conditions generated within the screen grating cage.

With the screen grating cage and cruciform in place, blockage was placed on the screen to produce the largest internal vortices achievable. Pressure measurements were then taken and observations made. The grating cage was then installed and data were recorded for the identical conditions which previously had produced internal vortices.

In summary, test procedures were developed for the Farley Unit 2 recirculation takes which:

i. *Modeled all effects of the screen grating cage and grating cage in the immediate vicinity of the intake on a 1:1 basis.*

ii. *Allowed testing for the effects of the containment geometry and structural members, etc., by subjecting the intake to a range of circulation, the largest of which was greater than will occur in the prototype approach flow.*

iii. *Demonstrated satisfactory intake performance under unrealistically severe conditions of water depth and circulation.*

F. TEST RESULTS

1. Phase 1 Test Results

a. Unprotected Intake

An air entraining vortex was easily formed over the unprotected intake pipe. With a water depth of 58 in. and an intake flow of approximately 7400 gal/min, the air entraining vortex was present intermittently when the flow vanes were aligned radially to the intake. The vortex increased in strength and became stable as the vanes were turned from the radial direction. The maximum vortex occurred with the vanes turned 48 degrees in either direction. The air core diameter of the vortex at the intake was 1.5 to 2 in. with a circulation of 8.5 $ft.^2/s$ as compared to a maximum calculated prototype value of 5.4 $ft.^2/s$.

With the vane angle set at 48 degrees, reducing the intake discharge from approximately 7400 gal/min to approximately 5300 gal/min reduced the diameter of the air core at the intake to 1 to 1.25 in.

b. Intake with Cruciform

The cruciform, by itself, did not eliminate air entraining vortices with a water depth of 58 in. The vortices were not as stable as without the *cruciform; nevertheless, the following air core sizes were observed at the intake:*

Flow gal/min	*Water Temp. °F*	*Vane Angle °*	*Air Core Diameter in.*
7344 to 8457	*65*	*48*	*1/2 to 3/4*
7412 to 8088	*65*	*0*	*1/8 to 3/8*
7018 to 8446	*180*	*48*	*3/4 to 1-1/2*

The circulation for a flow of 8456 gal/min was 9.1 ft^2/s.

The intake loss coefficient, K, was 0.69.

The maximum intake loss coefficient, K, for the heated water was 0.73 and the average intake loss coefficient was 0.72.

c. *Intake and Cruciform Protected by the Grating Cage*

No air entraining vortex penetrated the grating cage with the vanes set at 48 degrees and a water depth of 58 in. The intake flows tested were 7420 gal/min to 8513 gal/min with a water temperature of 119°F to 181°F and 8487 gal/min with a water temperature of 64°F. The flow circulation established by the vanes remained around the grating cage with the water surface depressed approximately 1 in. at the center. Bubbles or particulates in the flow surrounding the cage, which approached at an angle to the cage, were observed to exit at right angles to the plane of the grating.

The average intake loss coefficient was reduced from 0.72 with only the cruciform to 0.65 with the grating cage. The maximum intake loss coefficient was 0.66.

With an intake flow of 8400 gal/min, no air entraining vortex was produced when the water level was lowered from 58 inches to 24 inches.

d. *Intake and Cruciform Protected by Screen Grating Cage*

No air entraining vortex penetrated the screen grating cage for flows of 6574 to 8487 gal/min, vane angles of 0° and 48°, a water depth of 58 in., and water temperatures of 61°F to 64°F, and 173°F to 180°F. The maximum circulation was 10.7 $ft.^2$/s.

The circulation outside of the screen-grating cage was not transmitted through the structure.

e. *Intake and Cruciform Protected by a Grating Cage and Screen Grating Cage*

No air entraining vortex penetrated the screen grating cage for flows of 7741 gal/min to 8460 gal/min, vane angle of 48°, a water depth of 58 in., and water temperatures of 64° F and 184°F.

There was no organized circulation inside the unblocked screen grating cage.

The maximum intake loss coefficient was 0.67 and the average coefficient was 0.66.

The maximum screen head loss coefficient K_s *with or without blockage was 10.2.*

The K_s *values indicated a decreasing trend with increasing screen Reynolds number.*

2. *Phase 2 Tests*

The following summarizes the results of the Phase 2 tests:

a. *Intake and Cruciform Protected by Screen Grating Cage by Without Grating Cage*

Organized circulation could be established within the screen grating cage by selective blockage of the screen.

Air core vortices were established by screen blockage of 61 to 71 percent for intake flows of 7461 gal/min to 8420 gal/min and water temperatures of 150°F to 177°F. The water depth was 58 in.

Internal vortices could be formed from the floor, inside blockage plates (i.e., simulated walls), and the cover plate on the screen grating cage. One to five vortices could be generated simultaneously depending upon the blockage condition. A smooth surface within the screen grating cage was required to form an internal vortex.

The air core diameter of the internal vortices varied from 1/8 in. to 1/4 in.

The average intake loss coefficient was 0.69 and the maximum coefficient was 0.78. Internal vortices did not increase intake losses.

b. *Intake and Cruciform Protected by Screen Grating Cage and Grating Cage*

Installation of the grating cage over the intake and cruciform completely eliminated all the internal vortices previously generated by the screen blockage and flow conditions discussed in subsection F.2-a.

Flow circulation between the screen grating cage and grating cage, generated by the blockage, was not transmitted through the grating cage as evidenced by observing particulates in the flow.

The average intake loss coefficient with the screen cage, grating cage and cruciform in place was 0.66 and the maximum coefficient was 0.67.

G. SUMMARY AND CONCLUSIONS

The recirculation intake designs to be used for Farley Unit 2 were tested under flow and vortex producing conditions which were potentially more degrading on pump performance than any condition possible in the prototype. The screen grating and inner grating cage were modeled at a 1:1 scale. The following results were obtained:

1. *Vortex Action*

The screen grating cage will not permit any free surface air entraining vortices to form through which air can be ingested by the intake. Circulation (which is an essential feature of a vortex), or approach flow angularity, were not transmitted through the screen grating cage. The grating used in the screen grating cage was totally effective in eliminating any vortex with air core diameters of 1/8 in. to 2 in., which would have otherwise formed without the presence of the screen grating cage.

Without the inner grating cage, internal vortices could be developed by selective screen blockage. These vortices, which were formed only from smooth surfaces, did not increase intake entrance losses. However, with the grating cage in place as proposed for the Farley Unit 2 design, no internal vortices will develop. Circulation developed within the screen grating cage, which could lead to internal vortices, was not transmitted inside of the grating cage.

2. *Head Loss Coefficients*

The screen grating cage, grating cage, and cruciform protective design will have a head loss coefficient for the combined grating cage, intake and 90° pipe bend of 0.67, even with screen blockages in excess of 50 percent.

The maximum measured intake loss coefficients were as follows:

Cruciform	*0.73*

Grating cage and cruciform	*0.66*
Screen grating cage, grating cage and cruciform	*0.67*

Screen grating cage head losses are small with the maximum measured loss coefficient in the model being 10.2.

3. *Losses*

The maximum losses determined from the model test and calculations for each intake are:

Effect	*Pressure Drop (feet) Intake*			
	1	*2*	*3*	*4*
Piping (calculated)[1, 2]	*5.89*	*4.27*	*6.47*	*7.09*
Inlet (from test data)[1]	*0.75*	*0.75*	*1.02*	*1.02*
(From test data)[1]	*1.48*	*1.48*	*2.40*	*2.40*
Screen (from test data)	*0.09*	*0.03*	*0.03*	*0.05*
Total	*8.21*	*6.53*	*9.92*	*10.56*

NOTES:

1. *Converted to 4200 gal/min base for Intakes 1 and 2 and to 3050 gal/min base for Intakes 3 and 4.*

2. *These are calculated numbers and will be verified by a field test. However, comparison of the calculated values with the field test data for Unit 1 indicates that the calculated values are conservative (See section 6C.III.F).*

REFERENCES

1. *Addison, H., 1948; "Centrifugal and Other Rotodynamic Pumps," (Chapman and Hall, London.)*

2. *Akers and Crump, "The Vortex Drip," Journal, Institution of Civil Engineers, p 443, August 1960.*

3. *Al'Tshul, A. D. and Margonlin, M. S., "Effect of Vortices on the Discharge Coefficient for Flow of a Liquid Through an Orifice" (Translation), Gidrotekhnieheskoe Stroitel'stvi No. 6, p 32, June 1968.*

4. *Amphlett, M. B., "Air Entraining Vortices at a Horizontal Intake," Report No. OD/7, Hydraulic Research Station, Wallingford, April 1976.*

5. *Anwar, H. O., "Flow in a Free Vortex", Water Power, April 1965.*

6. *Anwar, H. O., "Formation of a Weak Vortex", Journal of Hydraulic Research, Vol. 4, No. 1, 1966.*

7. *Anwar, H. O., "Vortices at Low Head Intakes", Water Power, pp 455-457, November 1967.*

8. *Anwar, H. O., "Prevention of Vortices at Intakes," Water Power, p 393, October 1968.*

9. *Anwar, H. O., Discussion of "Effect of Viscosity on Vortex-Orifice Flow," by Paul B. Zielinski and James R. Villemonte, Journal of the Hydraulics Division, ASCE Vol. 95, No. HY 1, Proc. Paper 6323, pp 568-570, January 1969.*

10. *Berge, J. P., "Enquete sur la Formation de Vortex et Autres Anomalies d'ecoulement dans une enceinte avec ou sans Surface Libre," Societe Hydrotechnique de France-Section Machines - Group de travail No. 10, November 1964.*

11. *Berge, J. P., "A Study of Vortex Formation and Other Abnormal Flow in a Tank With and Without a Free Surface," La Huille Blanche, Grenoble, France, No. 1, pp 13-40, 1966.*

12. *Binnie, A. M., and Hockings, G. G., "Laboratory Experiments on Whirlpools," Proceedings, Royal Society, London, Series A. Vol. 194, pp 398-415, September 1948.*

13. *Brewer, D. "Vortices in Pumps Sumps," The Allen Engineering Review, March 1957.*

14. *"Review of Literature on Drain Vortices in Cylindrical Tanks," Report TN1342, BHRA, March 1976*

15. *Denny, D. F., 1953 British Hydromechanics Research Assoc. Report, R. R. 430, Preliminary Report on the Formation of Air entraining Vortices in Pump Section Wells.*

16. *Denny, D. F. "Experiments with Air in Centrifugal Pumps", British Hydromechanics Research Assoc. Research Report R. R. 465, 1953.*

17. *Denny, D. F., and Young, G. A. J. "The Prevention of Vortices and Swirl and Intakes", Proceedings, IAHR 7th Congress, Libson, 1957.*

18. *Denny, D. F., "An Experimental Study of Air Entraining Vortices in Pump Sumps", Proceedings of Inst. of Mechanical Engineers, Vol. 170, No.2, 1956.*

19. *Daggett, L. L. and Keulegan, G. H., "Simulitude Conditions in Free Surface Vortex Formations", Journal of the Hydraulics Division, ASCE, Vol. 100, No. Hy11, Nov. 1974, pp 1565-1581.*

20. *Donaldson, C. du p., and Sullivan, R. D., "Examination of the Solutions of the Navier-Stokes Equations for a Class of Three-dimensional Vortices, Part I: Velocity Distribution for Steady Motion", Proceedings, Heat Transfer and Fluid Mechanics Institute, Stanford University Press, Calif., 1960, p. 16-30.*

21. *Einstein, H. A., and Li, H., "Steady Vortex Flow in a Real Fluid," La Huille Blanche, Vol. 10, No. 4, Aug.-Sept., 1955, p. 483-496.*

22. *Folsom, R. G., "Some Performance Characteristics of Deepwell Turbine Pumps," Technical Memo. No. 6, HP-14, University of California, Pump Testing Laboratories, 1940.*

23. *Fraser, W. H., "Hydraulic Problems Encountered in Intake Structures of Vertical Wet-Pit Pumps and Methods Leading to Their Solution," Trans. ASME, vol. 75, No. 4, p. 643, 1953.*

24. *Gordon, J. L., "Vortices at Intakes," Water Power, April 1970, p. 137-138.*

25. *Cuiton, P., "Cavitation dans les Pompes", La Huille Blanche, Nov., 1962, No. 6.*

26. *Haindl, K., "Contribution to Air-Entrainment by a Vortex", Paper 16-D International Association for Hydraulic Research, Montrel, 1959.*

27. *Hattersley, R. T., "Hydraulic Design of Pump Intakes", HY 2, March, 1965, P. 223-249.*

28. *Hattersley, R. T., "Factors of Inlet Channel Flow affecting the Performance of a Pumping Plant", Report No.23, Water Research Lab., University of New South Wales, Australia, Sept., 1960.*

29. *Holtorff, G., "The Free Surface and the Conditions of Similitude for a Vortex", La Houille Blache, vol. 19, No. 3, 1964, P. 377-384.*

30. *Iversen, H. W.,; "Studies of Submergency Requirements of High Specific Speed Pumps", Transactions, ASME, Vol. 75, 1953.*

31. *Kaufman, Fluid Mechanics, McGraw-Hill, p. 265 and 279.*

32. *Keulegan, G. H., and Daggett, L. L., "A Note on Gravity Head Viscometer", Miscellaneous Paper H-74-3, United States Army Engineer Waterways Experiment Station, Corps of Engineers, Vicksburg, Miss., Mar., 1974.*

33. *Kolf, R. C., "Vortex Flow from Horizontal Thin Plate Orifices", thesis presented to the University of Wisconsin, at Madison, Wis., in 1956, in partial fulfillment of the requirements for the degree of Doctor of Philosophy.*

34. *Kolf, R. C., and Zielinski, P. B., "The Vortex Chamber as an Automatic Flow Control Device", Journal of the Hydraulics Division, ASCE, vol. 85, No. HY12, Proc. Paper 2310, Dec. 1959.*

35. *Lawton, F. L. "Factors Influencing Flow in Large Conduits", Report of the Task Force on Flow in Large Conduits of the Committee on Hydraulic Structures. Transactions, ASCE, Paper 4543, vol. 91 HY 6, November, 1965.*

36. *Lennart, R., "Flow Problems with Respect to Intakes and tunnels of Swedish Hydro Electric Power Plants", Transactions, of the Royal Institute of technology, Stockholm, Sweden, NR 71, 1953.*

37. *Lewellen, W. S. "A Solution for Three-Dimensional Vortex Flow with Strong Circulation," J. Fluid Mechanics, vol. 14, 1962.*

38. *Long, R. R. "A Vortex in an Infinite Fluid," Journal of Fluid Mechanics, vol. 11.*

39. *Marklund, E. and Pipe, J. A. "Experiments on a Small Pump Suction Well, with Particular Reference to Vortex Formations". Proceedings, The Institution of Mechanical Engineers, vol. 170, 1956.*

40. *Marklund, E., discussion of "Effect of Viscosity on Vortex-Orifice Flow", by Paul B. Zielinski and James R. Villemonte, Journal of the Hydraulic Division, ASCE, vol. 95, No. HY1, Proc. paper 6323, Jan., 1969, p. 567-568.*

41. *Messina, J. P., "Periodic Noise in Circulating Water Pumps", Power, Sept. 1971, p. 70-71.*

42. *McCorquodale, J. A., discussion of "Effect of Viscosity on Vortex-Orifice Flow," By Paul B. Zielinski and James R. Villemonte, Journal of the Hydraulics Division, ASCE, vol. 95, No. HY1, Proc. Paper 7323, Jan., 1969, p. 567-568.*

43. *McCorquodale, J. A., "Scale Effects in Swirling Flow", Journal of the Hydraulics Division, ASCE, vol. 94, HY1, Disc. by Marco Pica, HY1, Jan. 1969.*

44. *Pickford, J. A., and Reddy, Y. R. "Vortex Suppression in a Stilling Pond Over Flow", Journal of the Hydraulics Division, ASCE, vol. 100, No. HY11, Nov. 1974, pp 1685-1697.*

45. *Quick, M. C., "A Study of the Free Spiral Vortex", thesis presented to the University of Bristol, England, in 1961, in partial fulfillment of the requirements for the degree of Doctor of Philosophy.*

46. *Quick, M. C., "Scale Relationships between Geometrically Similar Free Spiral Vortices", Civil Engineering and Public Works Review, Part 1, September, 1962, Part II, October, 1962, p. 1319.*

47. *Quick, M. C., "Efficiency of Air Entraining Vortex Formation at Water Intake", Journal of the Hydraulics Div., ASCE, vol., No. 96, HY7, July 1970, p. 1403-1416.*

48. *Reddy, Y. R., and Pcikford, J. A., "Vortices at Intakes in Conventional Sumps", Water Power, March 1972,p. 108-109*

49. *Rouse, H., and Hsu, H., "On the Growth and Decay of a Vortex Filament", Proceedings, 1st National Congress of Applied Mechanics, 1952, p. 741-746.*

50. *Richardson, C. A., "Submergence and Spacing of Suction Bells," Water Works and Sewerage Reference and Data, Part 1, Water Supply, p. 25, 1941.*

51. *Springer, E. K., and Patterson, I. M., "Experimental Investigation of Critical Submergency for Vortexing in a Vertical Cylindrical Tank," ASME Paper 69-FF-49, June 1969.*

52. *Stepanoff, A. J., 1946 Centrifugal and Axial Flow Pumps, p. 963 (Chapman and Hall, London).*

53. *Stevens, J. C., discussion of "The Vortex Chamber as an Automatic Flow Control Device", by R. C. Kolf and P. B. Zielinski, Journal of the Hydraulics Division, ASCE, vol. 86, No. HY6, Proc. Paper 2525, June, 1960.*

54. *Stevens, J. C., and Kolf, R. C., "Vortex Flow Through Horizontal Orifices", Journal of the Sanitary Engineering Division, ASCE, Vol. 83, No. SA6, Proc. Paper 1461, Dec., 1957.*

55. *Weltmer, W. W. "Proper Suction Intakes Vital for Vertical Circulating Pumps," Power Engineering, vol. 54, No. 6, p. 74, 1950.*

56. *Young, C.A.H., "Swirl and Vortices at Intakes", Report No. SP 726, British Hydro-Mechanics Research Association, April, 1962.*

57. *Zelinski, P. B. and Vellimont, J. R., "Effect of Viscosity on Vortex Orifice Flow", vol. 94, HY3, May, 1968, p. 745-751. Disc. on above in Jan. 1969, by Marklund, E., McCorquodale, J. A. and Anwar, H.O.]*

[Historical] [TABLE 6C-1

TEST CONDITIONS FOR UNIT 1 INTAKE 1

Test No.	Water Depth In.	Discharges - gal/min Q_1 Prototype	Q_1 Model	Intake 1 Prototype	Intake 1 Model	Q_2 Prototype	Q_2 Model	Operating Pumps	Water Temperature Prototype	Water Temperature Model
1	58.3	3715	5500	4150	6150	-435	-644	1RHR	240	170+
2	77.1	3540	5240	4150	6150	-610	-905	1RHR	240	170+
3	58.3	5500	8140	3000	4440	2500	3700	2 RHR	240	170+
4	77.1	5500	8140	3000	4440	2500	3700	2RHR	240	170+
5	77.1	8750	12,450	4150	6150	4600	6808	1RHR,2S	240	170+
6	77.1	10,500	14,300	5900	8000	4600	6200	(a) 2RHR,2S	240	183+
7	77.1	10,600	10,600	4150	4150	6450	6450	2RHR,1S	170	170+
8	77.1	12,900	12,900	4150	4150	8750	8750	2RHR,2S	170	170+

RHR = Residual Heat

S = Spray

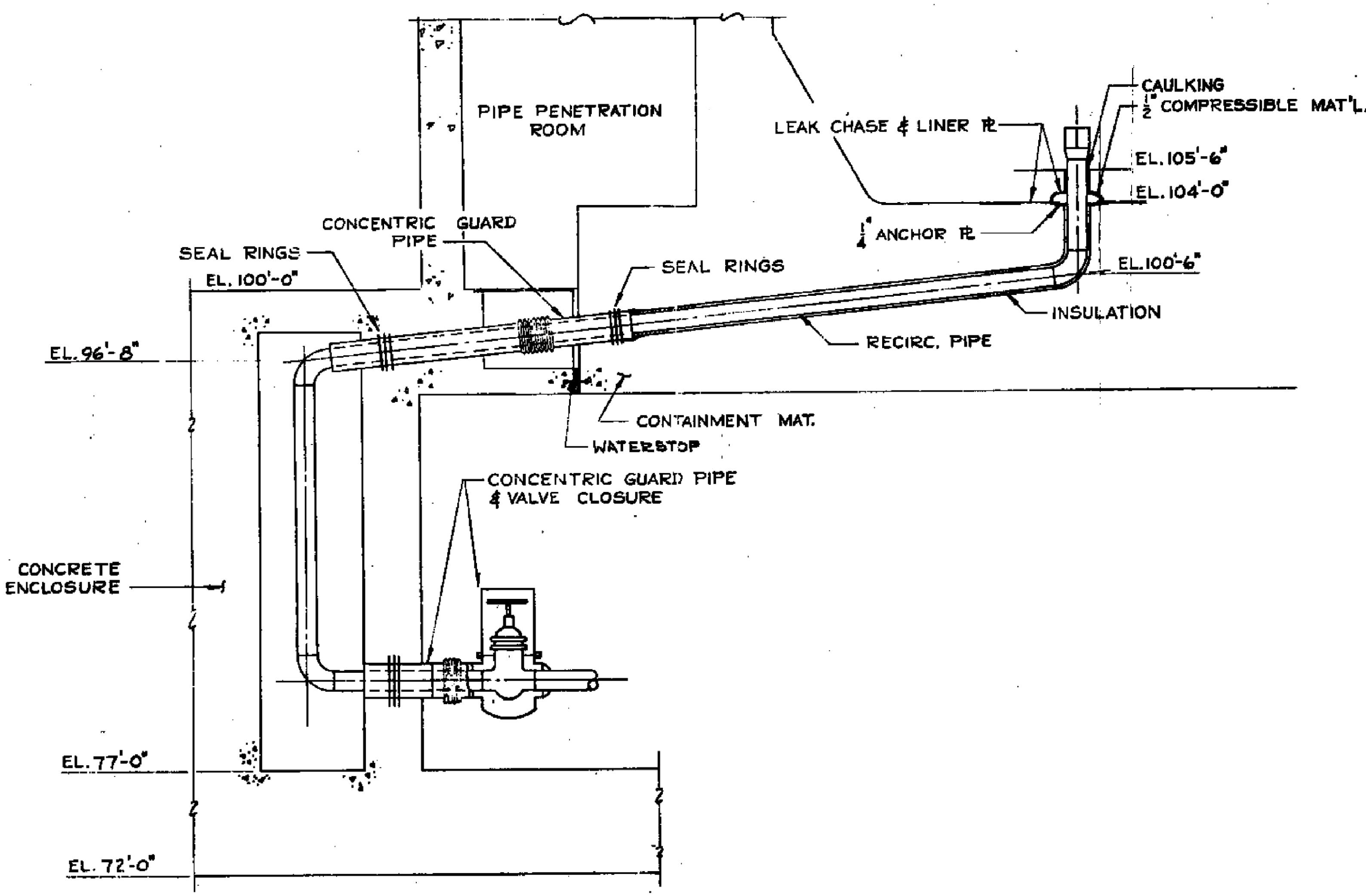
a. The two RHR pumps taking suction from one inlet]

[Historical] TABLE 6C-2

TEST CONDITIONS FOR
UNIT 1 INTAKES 2, 3, AND 4

Test No.	Water Depth in.	Q_2		Intake Discharges - gal/min						Q_3		Operating Pumps	Water Temperature, F	
				Intake 2		Intake 3		Intake 4						
		Prototype	Model	Prototype	Model	Prototype	Model	Prototype	Model	Prototype	Model		Prototype	Model
1	58.3	3,715	5,201	4,150	5,810	-	-	-	-	435	609	1RHR	240	180
2	77.1	6,450	9,030	4,150	5,810	3,050	4,270	-	-	750	1,050	1RHR, 1S	240	180
3	77.1	2,300	3,220	-	-	--	-	3,050	4,270	750	1,050	1RHR, 1S	240	180
4	77.1	4,600	6,440	-	-	3,050	4,270	3,050	4,270	1,500	2,100	1RHR, 2S	240	180
5	77.1	8,750	12,250	4,150	5,810	3,050	4,270	3,050	4,270	1,500	2,100	1RHR, 2S	240	180
6	77.1	5,300	7,420	3,000	4,200	3,050	4,270	-	-	750	1,050	2RHR, 1S	240	180
7	77.1	7,450	6,450	4,150	4,150	-	-	3,050	3,050	750	750	2RHR, 1S	170	180
8	77.1	10,500	14,700	5,900	8,260	3,050	4,270	3,050	4,270	1,500	2,100	2RHR*,2S	240	180
9	77.1	8,200	11,480	5,900	8,260	3,050	4,270	-	-	750	1,050	2RHR*,1S	240	180
10	77.1	8,200	11,480	5,900	8,260	-	-	3,050	4,270	750	1,050	2RHR*,1S	240	180

]



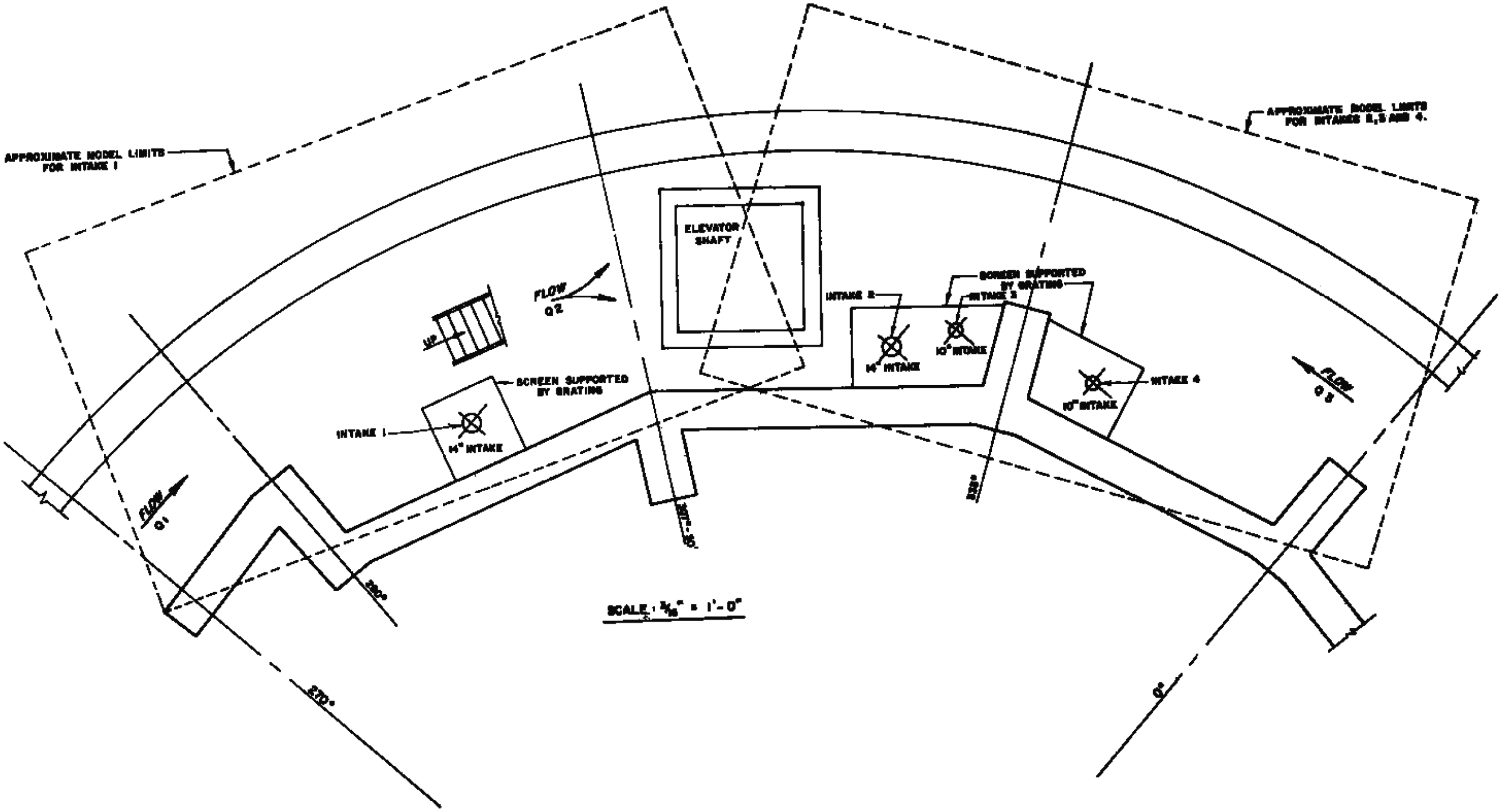
REV 21 5/08

SOUTHERN COMPANY
Energy to Serve Your World®

JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

[TYPICAL ARRANGEMENT OF CONTAINMENT SUMP SUCTION LINE

FIGURE 6C-1]



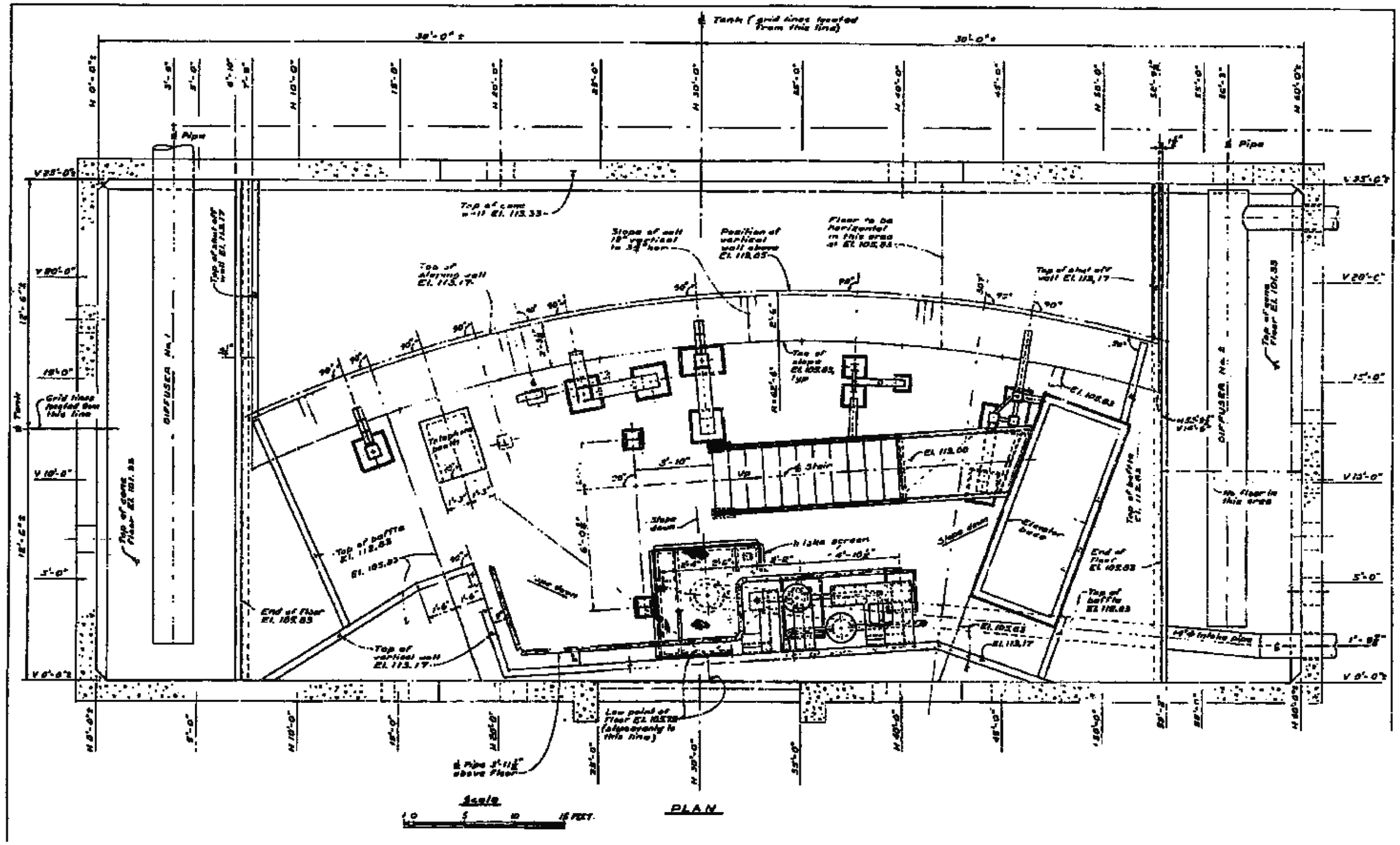
REV 21 5/08

SOUTHERN COMPANY
Energy to Serve Your World®

JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

[MODELED AREAS OF ECCS INTAKES

FIGURE 6C-2]



REV 21 5/08

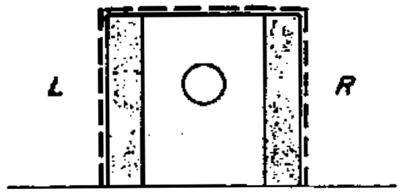
SOUTHERN COMPANY
Energy to Serve Your World®

JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

[HYDRAULIC MODEL PLAN FOR INTAKE NO. 1 TESTS

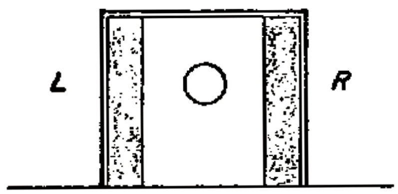
FIGURE 6C-3]

1.



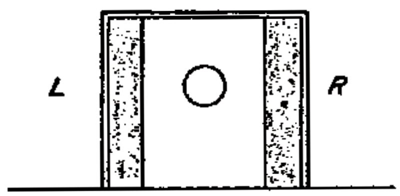
Top blocked 100%
Vertical screen blocked 50%
over full height by alternate
3" wide open and blocked strips

2.



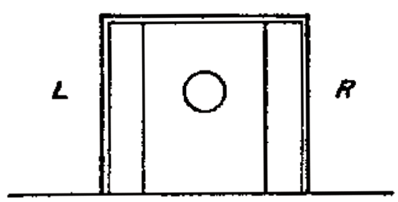
Top blocked 100%
Vertical screen blocked over
lower half around periphery

3.



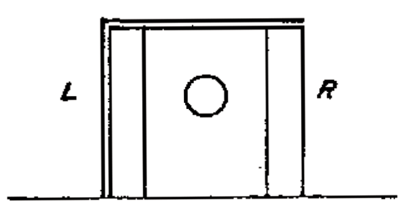
Top blocked 100%
Vertical screen blocked over
upper half around periphery

4.



Top open (apart from cover plate)
Vertical screen blocked over
lower half around periphery

5.



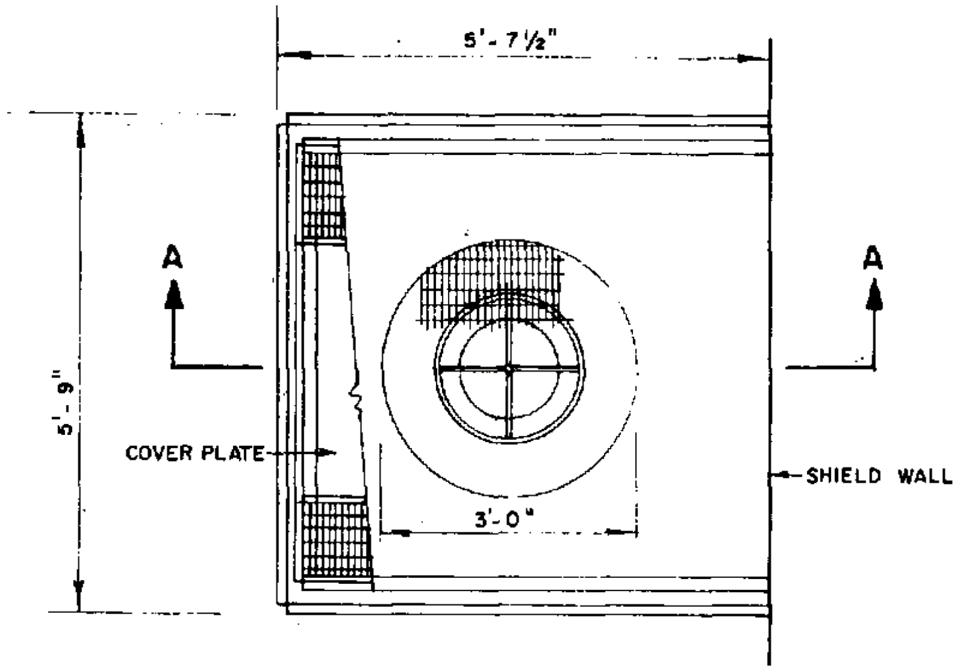
Top open (apart from cover plate)
Vertical screen completely blocked
over left and front sides for a total
of 67% of sides area.

13

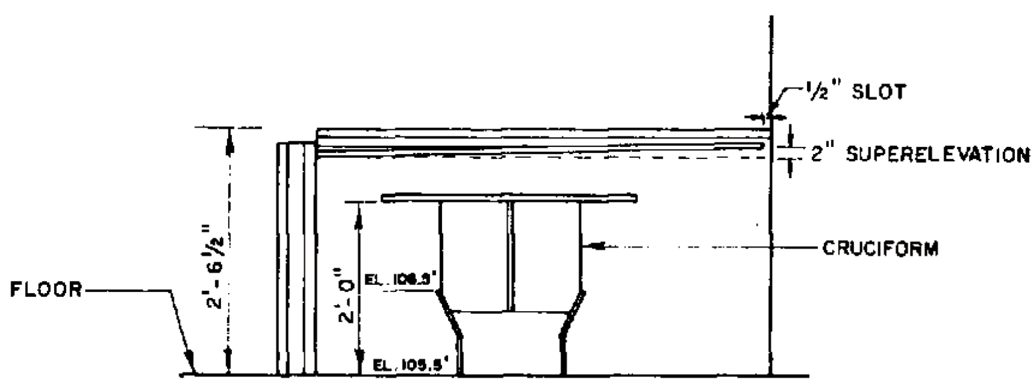
REV 21 5/08

[BLOCKAGE TEST CONDITIONS

FIGURE 6C-4]



PLAN



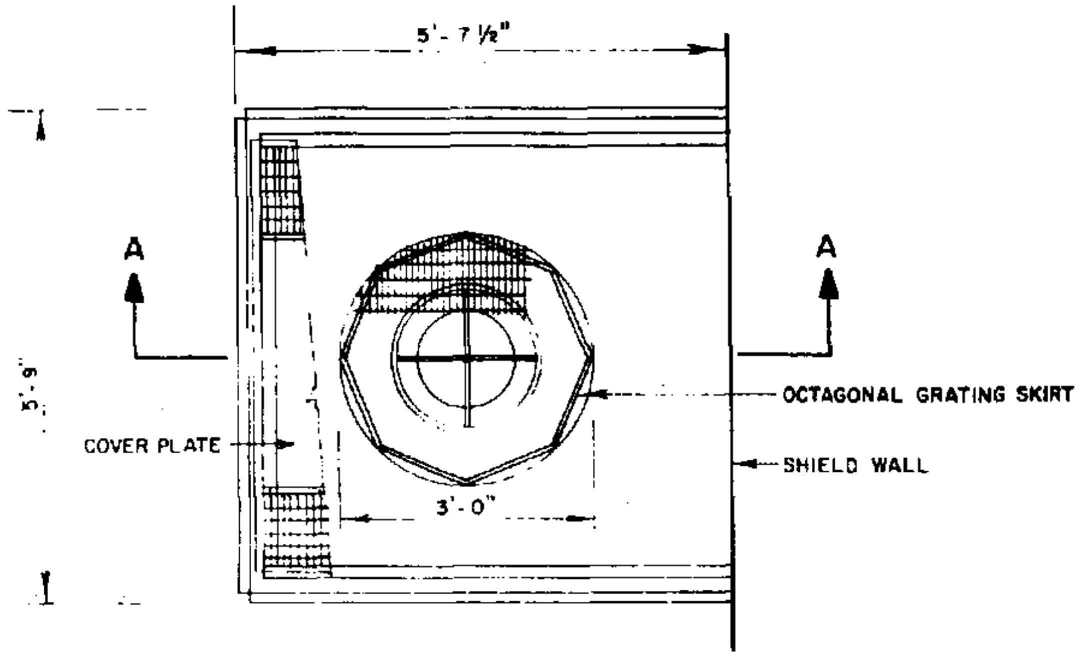
SECTION A - A

SCALE : 1/2" = 1'- 0"

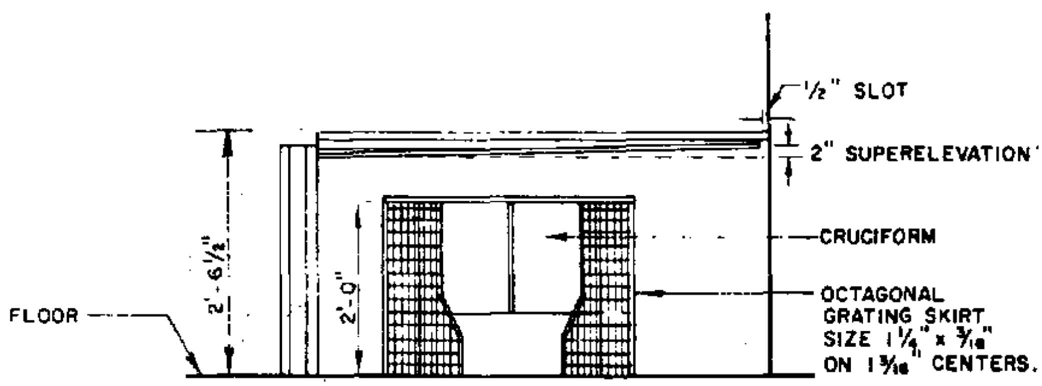
13

REV 21 5/08

SOUTHERN COMPANY *Energy to Serve Your World®*	JOSEPH M. FARLEY NUCLEAR PLANT UNIT 1 AND UNIT 2	*[NO. 1 INTAKE CONFIGURATION FOR INITIAL TESTS* *FIGURE 6C-5]*



PLAN



SECTION A - A

SCALE : ½" = 1'- 0"

13

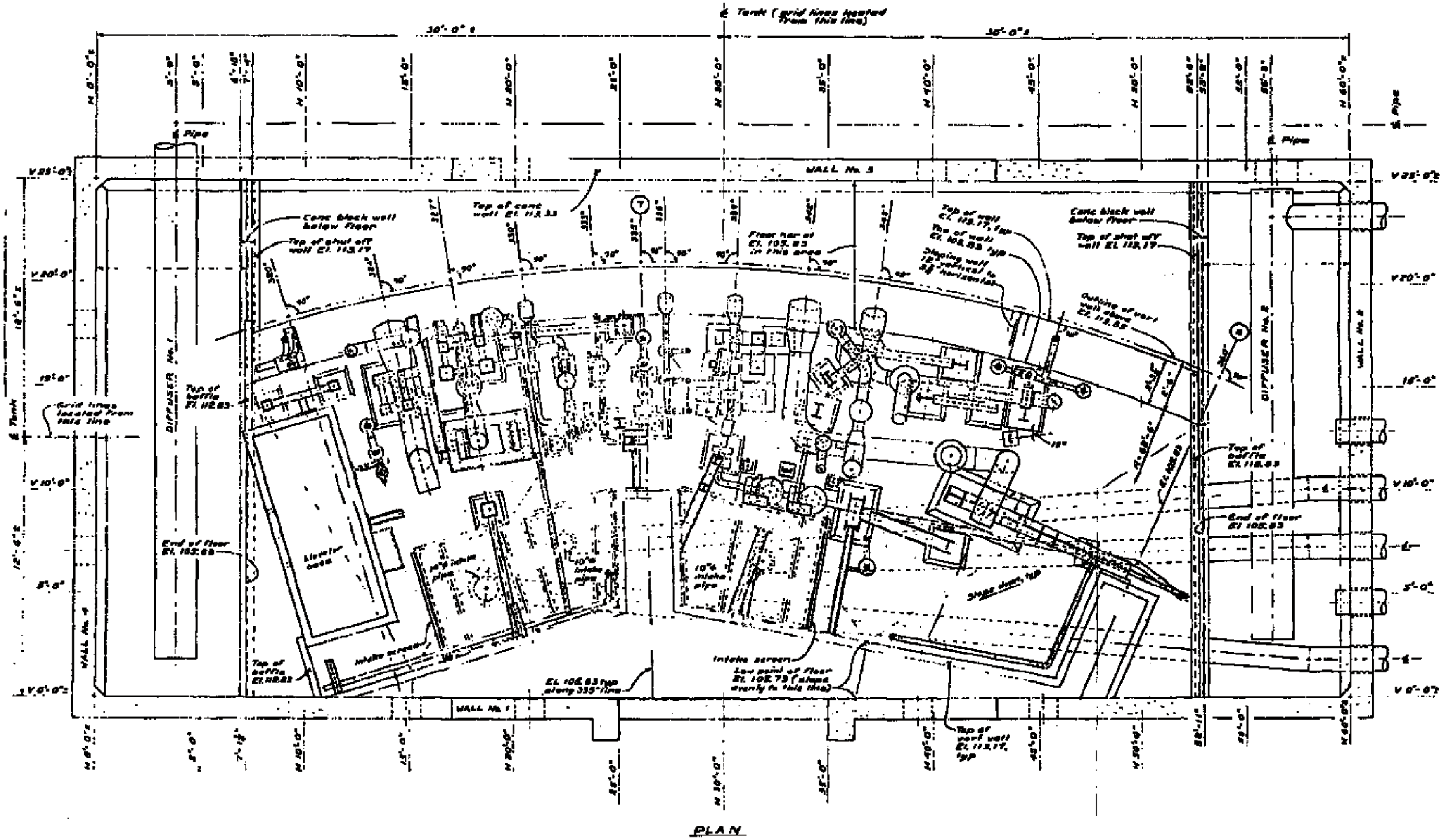
REV 21 5/08

SOUTHERN COMPANY
Energy to Serve Your World®

JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

[IMPROVED DESIGN INTAKE NO. 1

FIGURE 6C-6]



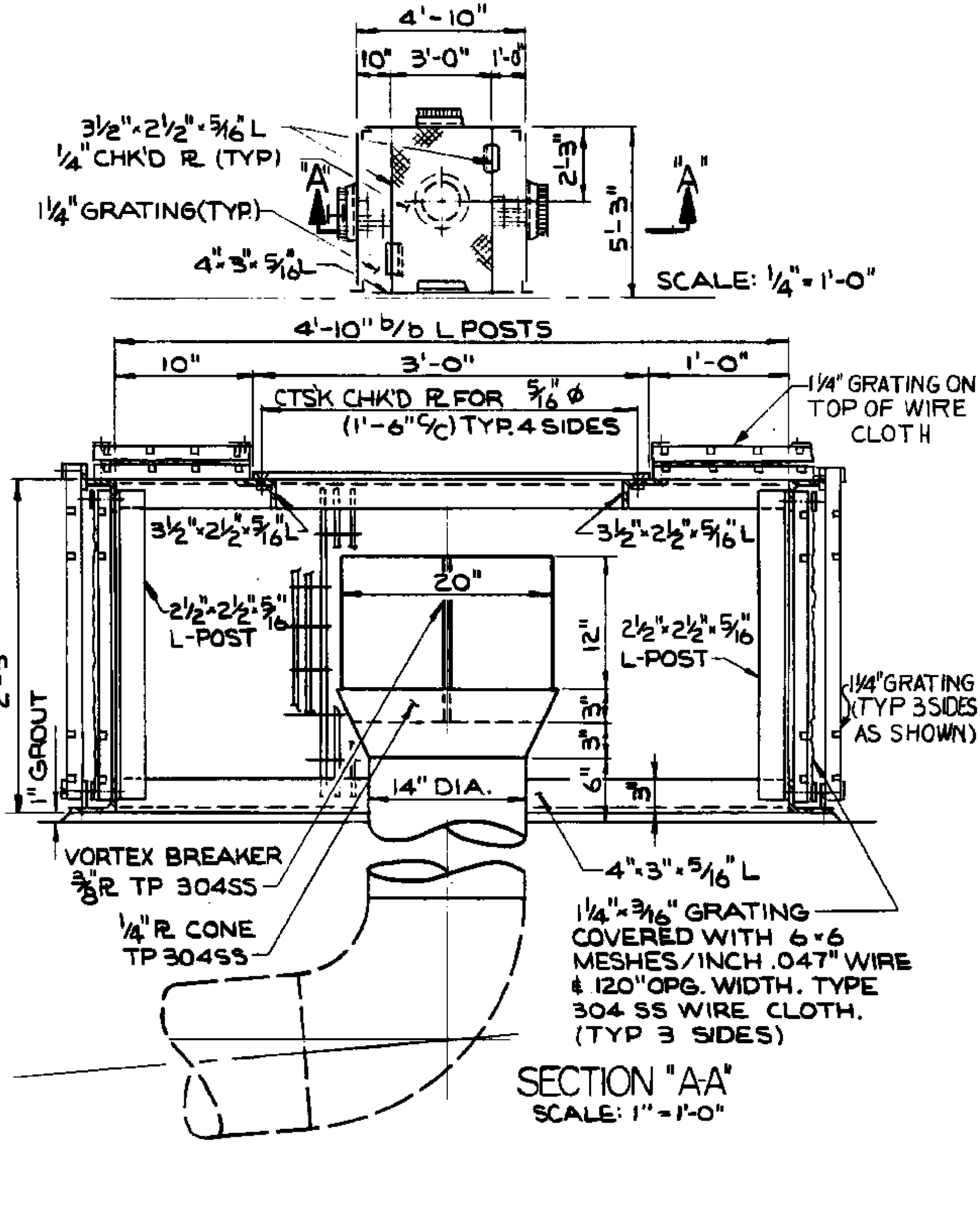
REV 21 5/08

SOUTHERN COMPANY
Energy to Serve Your World®

JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

PLAN OF MODELED AREA CONTAINING INTAKES 2, 3, AND 4

FIGURE 6C-7



REV 21 5/08

SOUTHERN COMPANY
Energy to Serve Your World®

JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

[CONTAINMENT SUMP

FIGURE 6C-8]

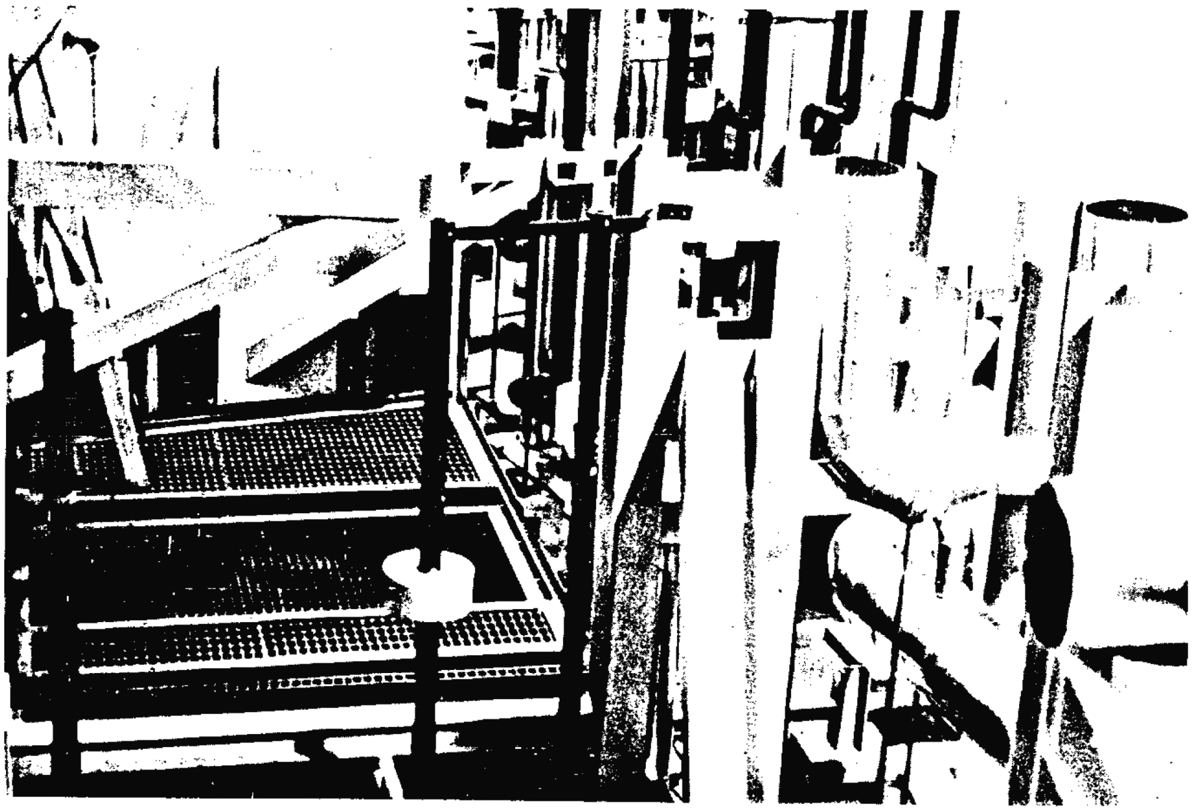
REV 21 5/08

SOUTHERN COMPANY
Energy to Serve Your World®

JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

[PHOTOGRAPH OF MODEL

FIGURE 6C-10]



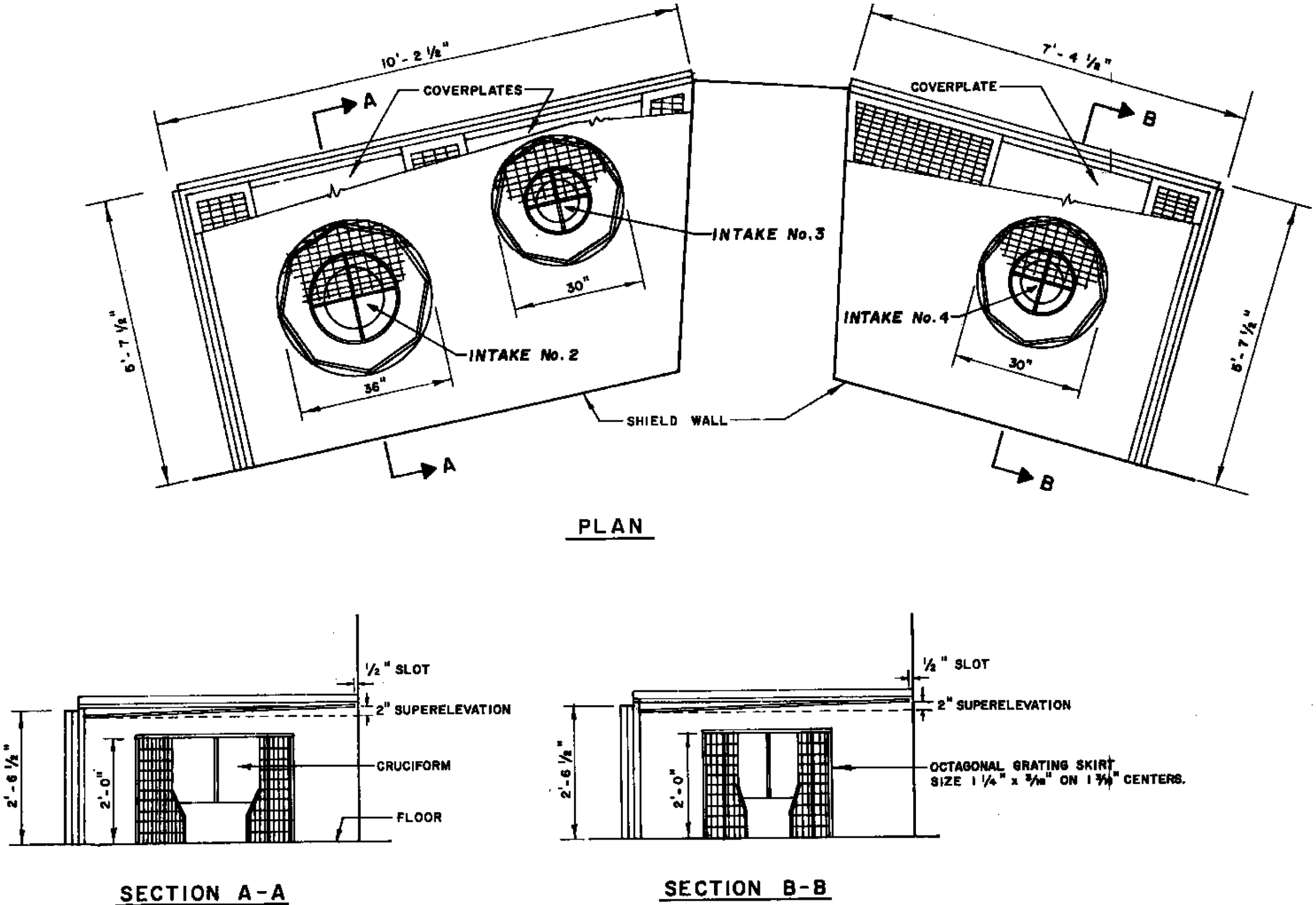
REV 21 5/08

SOUTHERN COMPANY
Energy to Serve Your World®

JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

[PHOTOGRAPH OF MODEL

FIGURE 6C-11]



REV 21 5/08

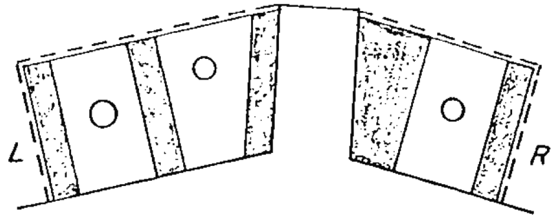
SOUTHERN COMPANY
Energy to Serve Your World®

JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

[INTAKES 2, 3, AND 4 IMPROVED DESIGN

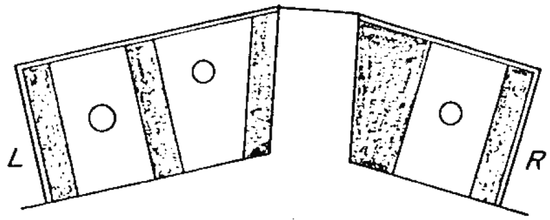
FIGURE 6C-12]

1.



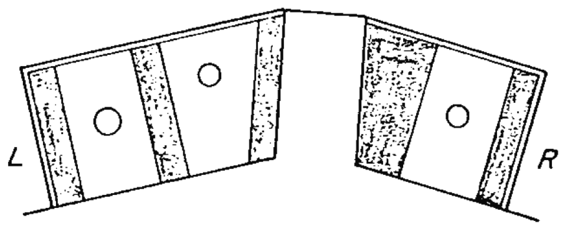
Top blocked 100%
Vertical screen blocked 50% over full height by alternate 3" wide open and blocked strips

2.



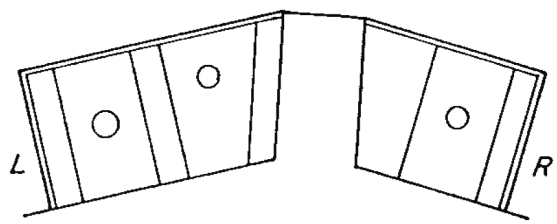
Top blocked 100%
Vertical screen blocked over lower half around periphery

3.



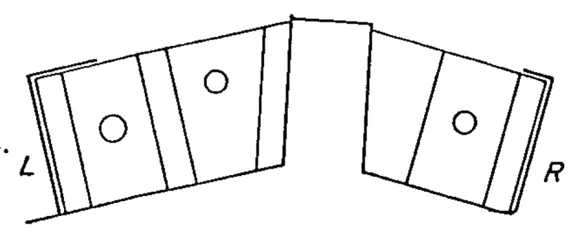
Top blocked 100%
Vertical screen blocked over upper half around periphery

4.



Top open (apart from cover plate) Vertical screen blocked over lower half around periphery

5.

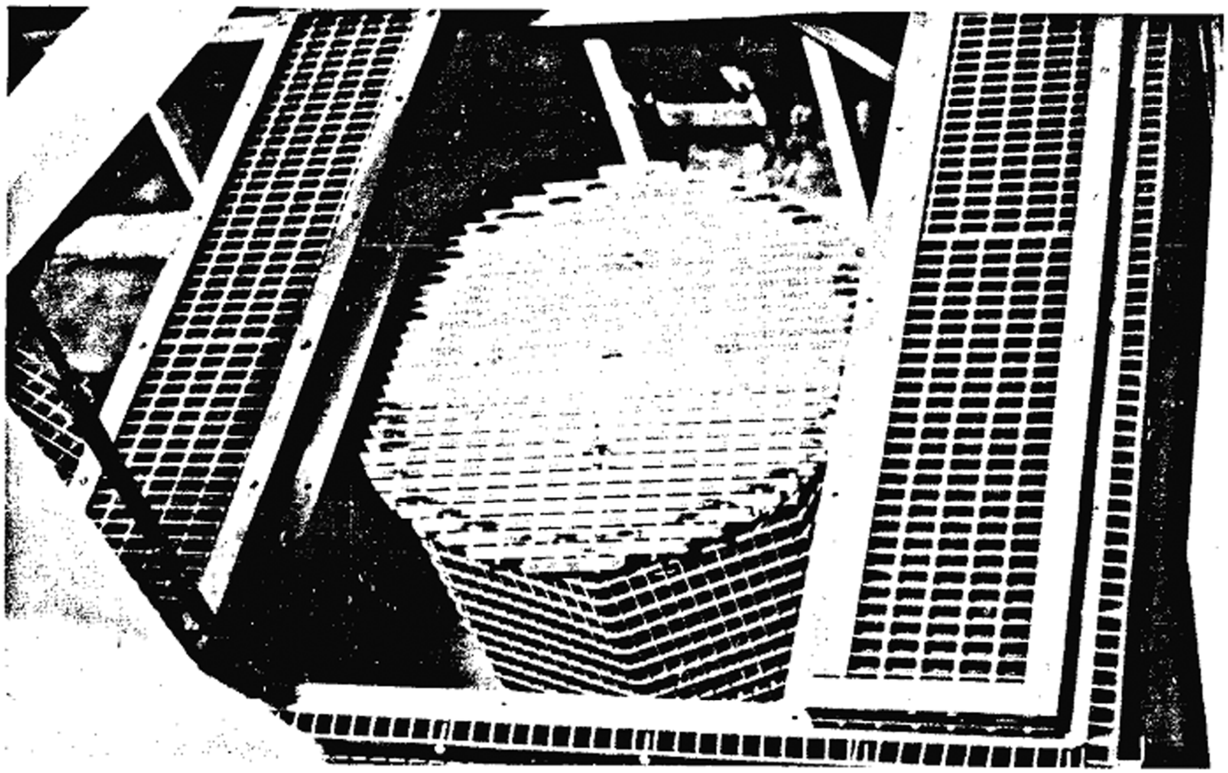


Top open (apart from cover plate) Upstream side of vertical screen completely blocked for 50% of total side area

REV 21 5/08

[BLOCKAGE TEST CONDITIONS FOR INTAKES 2, 3, 4

FIGURE 6C-13]



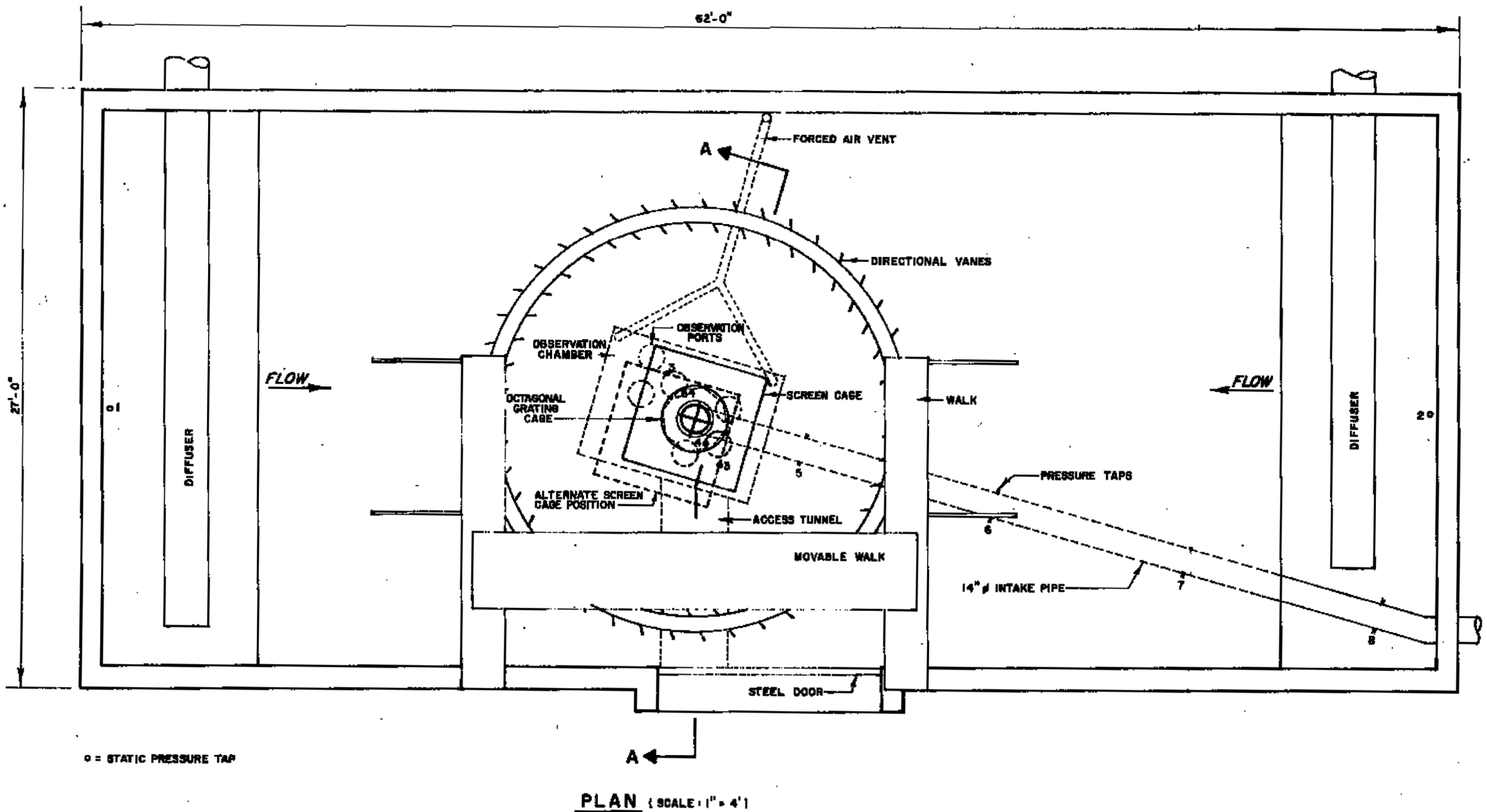
REV 21 5/08

SOUTHERN COMPANY
Energy to Serve Your World®

JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

[PHOTOGRAPH OF GRATING CAGE OVER INTAKE 2

FIGURE 6C-14]



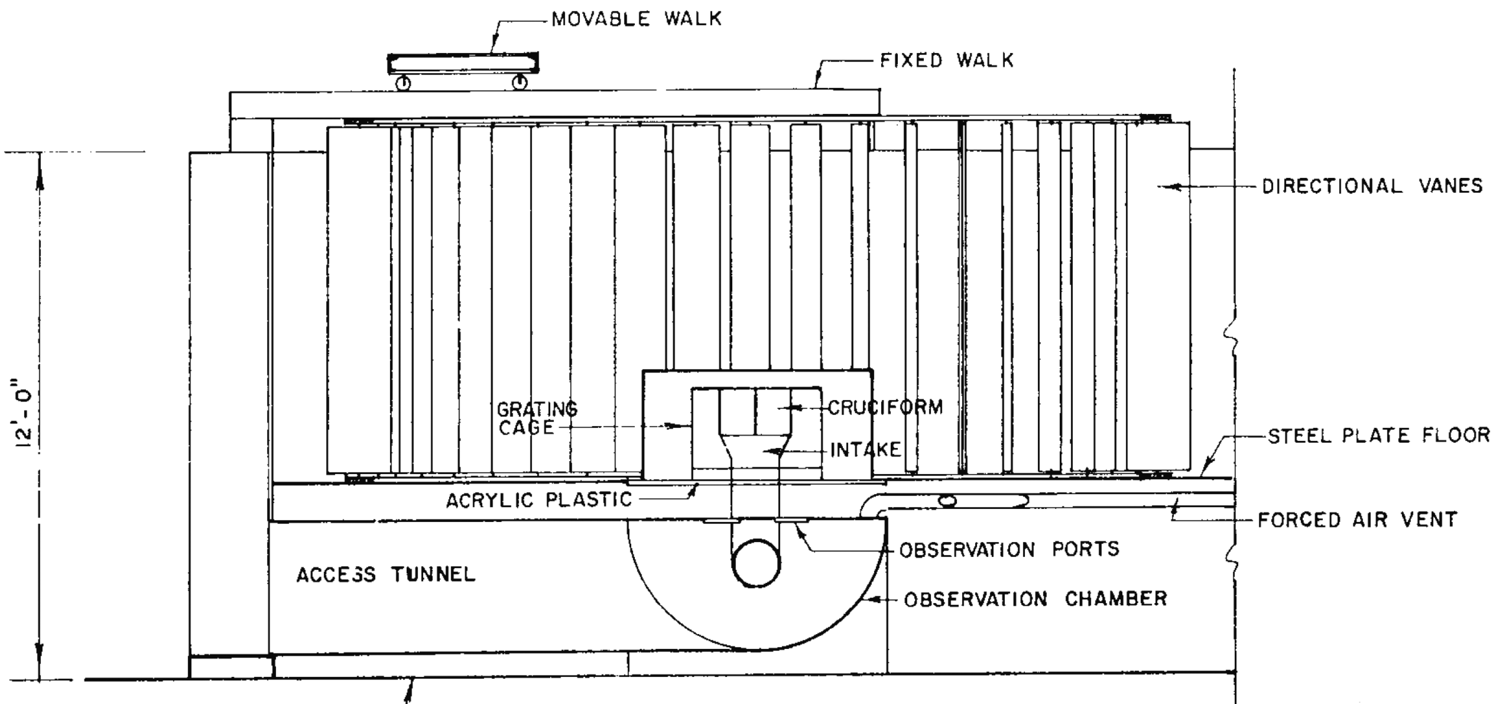
REV 21 5/08

SOUTHERN COMPANY
Energy to Serve Your World®

JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

[PLAN OF UNIT 2 TEST FACILITY

FIGURE 6C-15]



SECTION A - A (SCALE: 1" = 4')

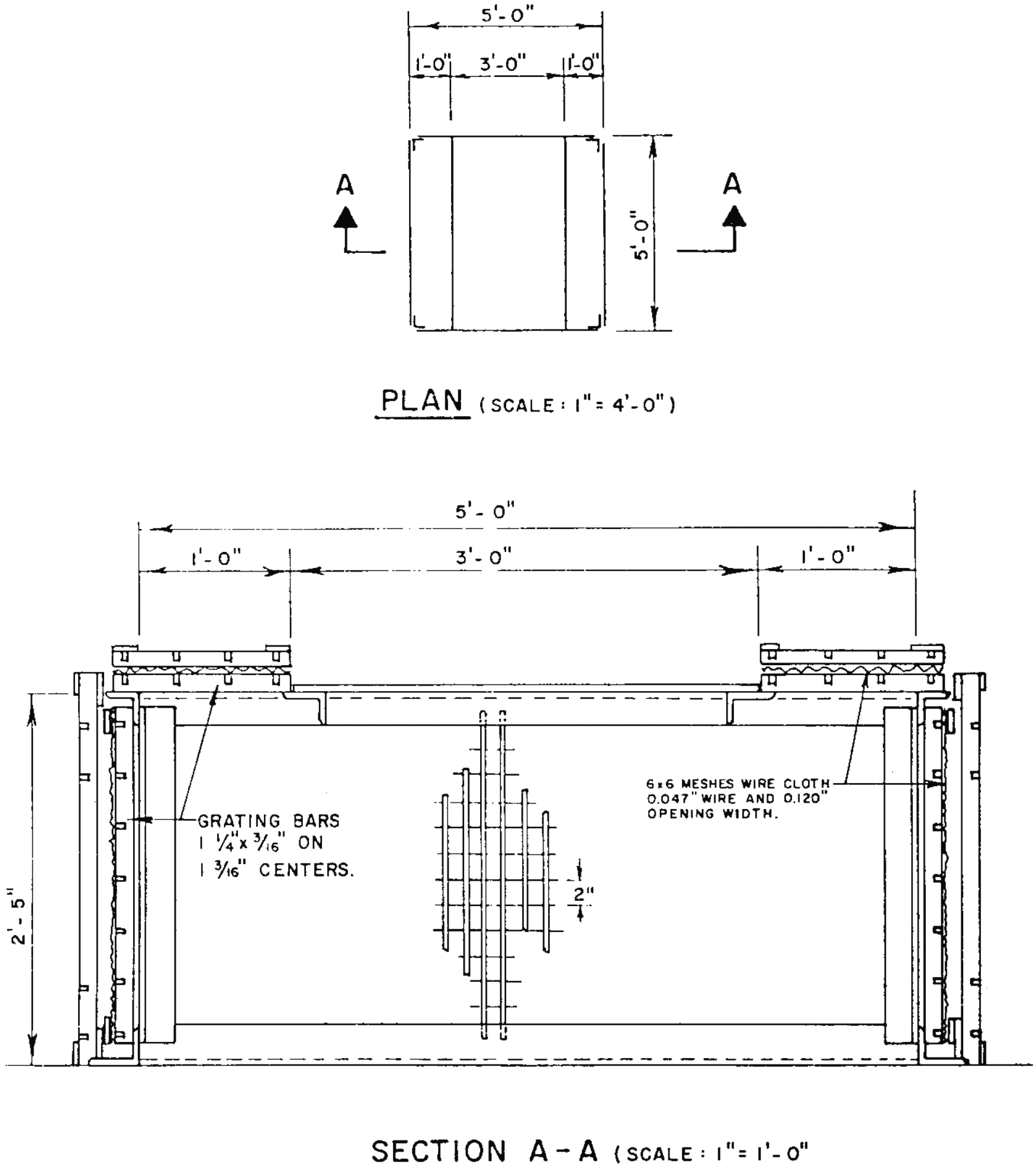
REV 21 5/08

SOUTHERN COMPANY
Energy to Serve Your World®

JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

[SECTION OF UNIT 2 TEST FACILITY

FIGURE 6C-16]



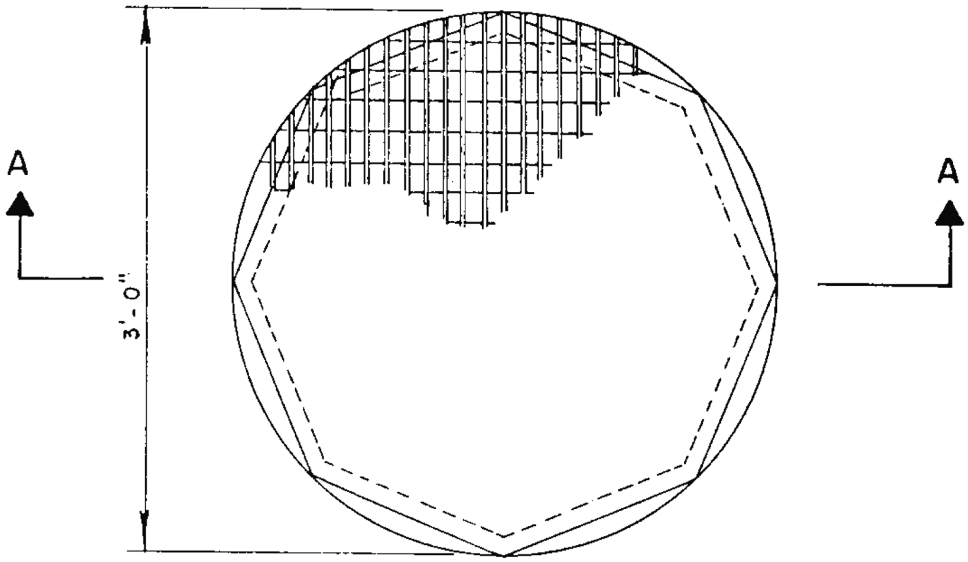
REV 21 5/08

SOUTHERN COMPANY
Energy to Serve Your World®

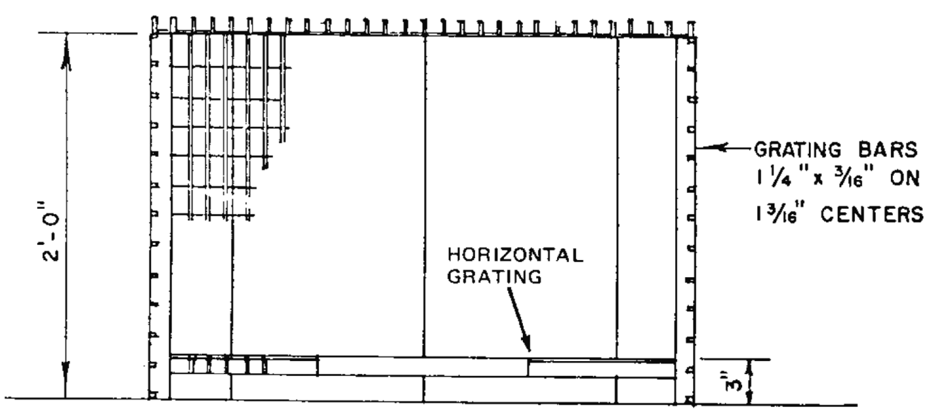
JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

[REPRESENTATIVE SCREEN-GRATING STRUCTURE

FIGURE 6C-17]



PLAN (SCALE: 1" = 1'-0")

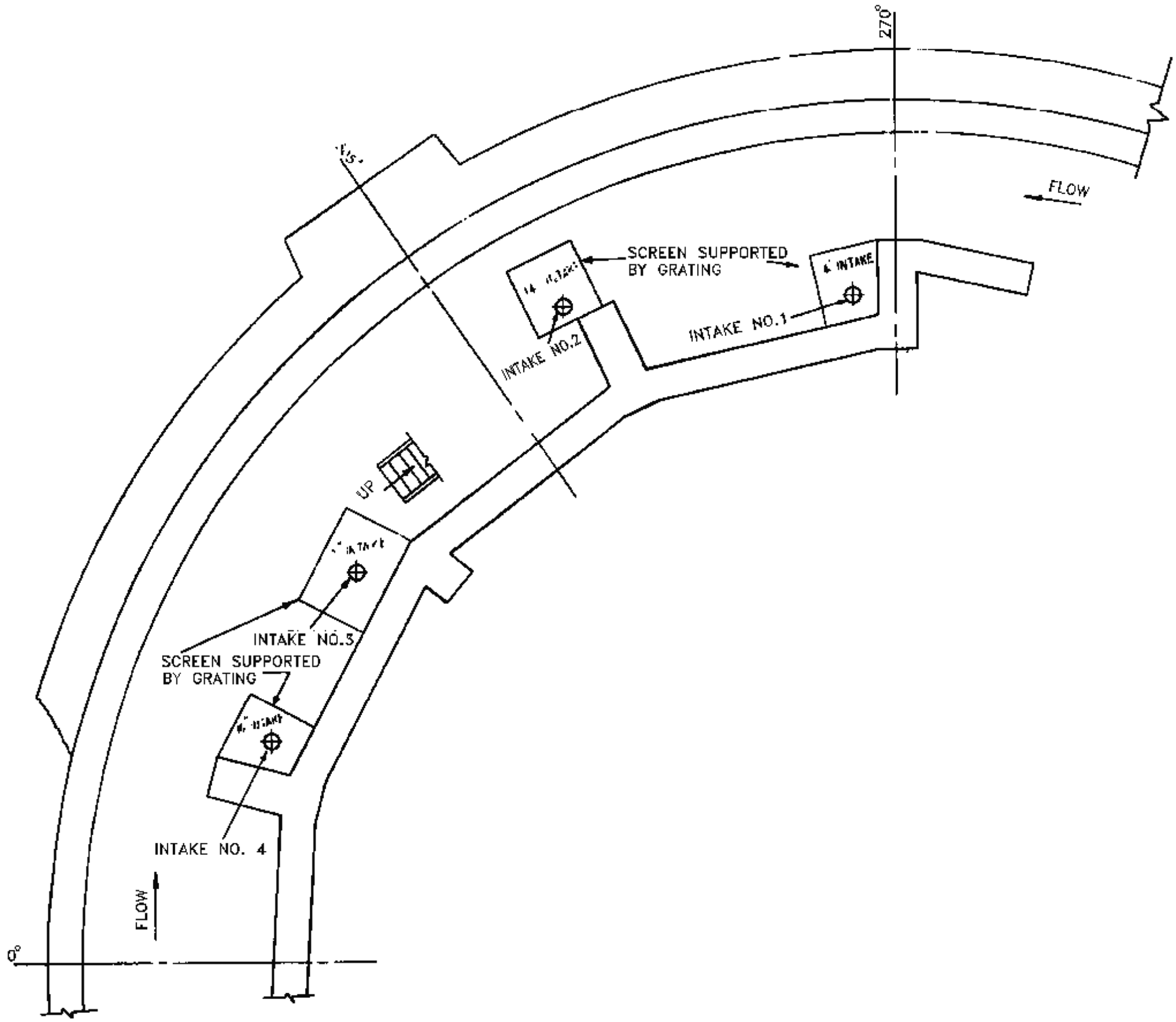


SECTION A - A (SCALE: 1" = 1'-0")

REV 21 5/08

[GRATING CAGE – FINAL DESIGN

FIGURE 6C-18]



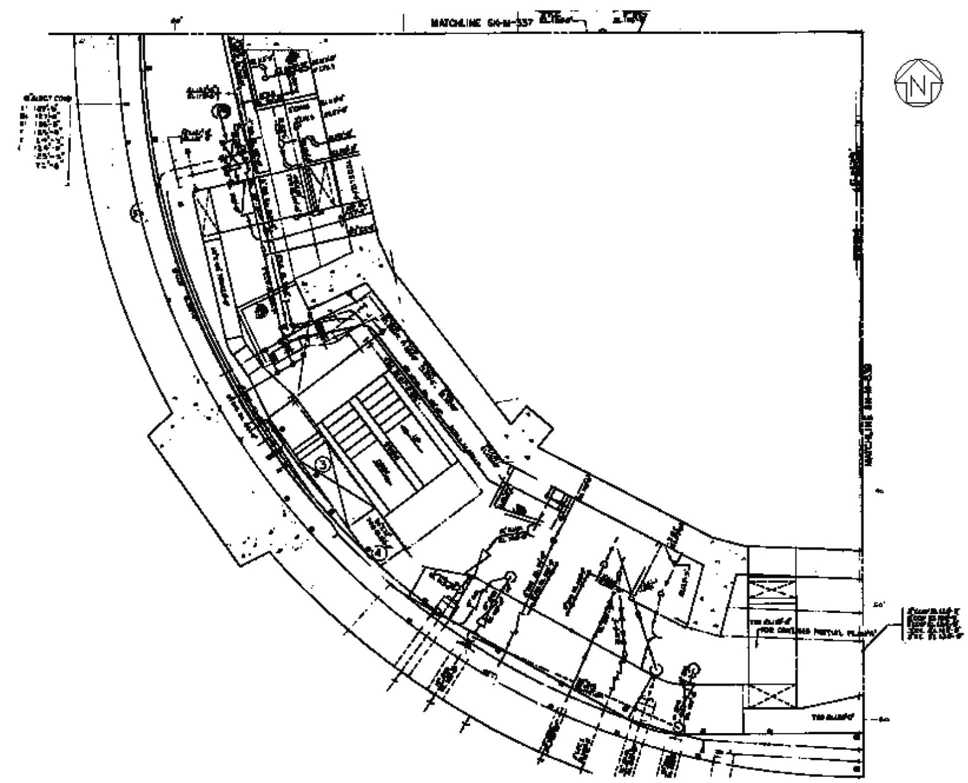
REV 21 5/08

SOUTHERN COMPANY
Energy to Serve Your World®

JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

[SUMP AREA OF UNIT 2

FIGURE 6C-19]



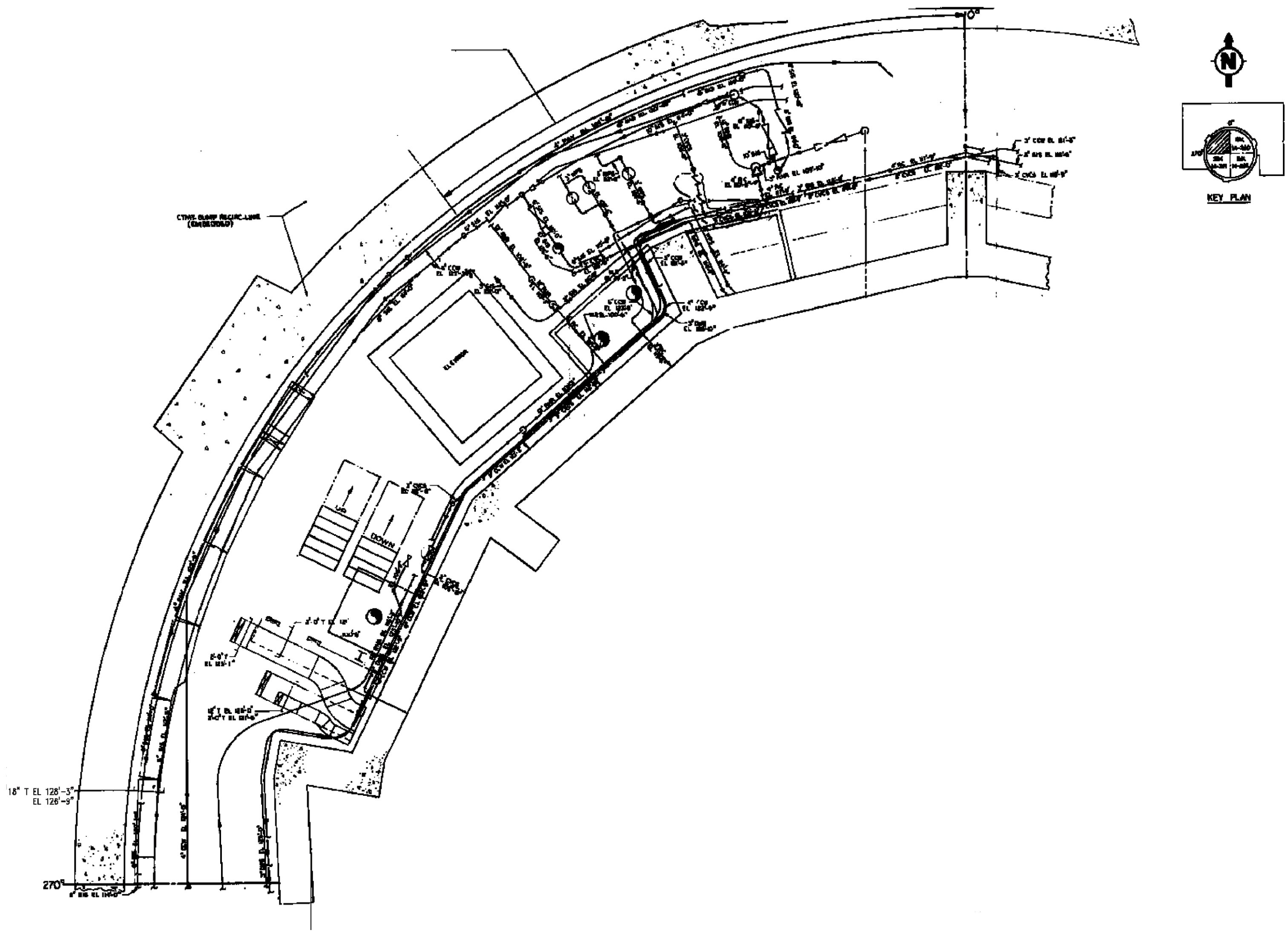
REV 21 5/08

SOUTHERN COMPANY
Energy to Serve Your World®

JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

[COMPOSITE DRAWING OF UNIT 2 SUMP

FIGURE 6C-20]



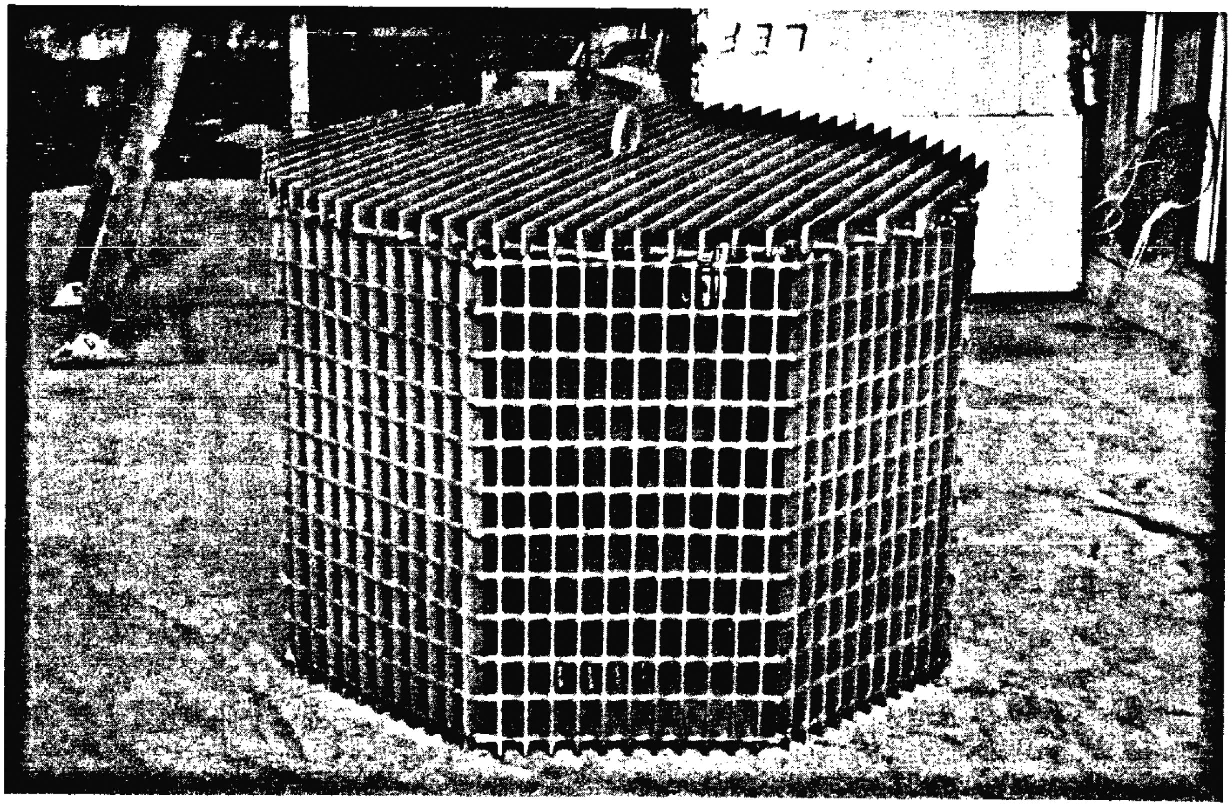
REV 21 5/08

SOUTHERN COMPANY
Energy to Serve Your World®

JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

[COMPOSITE DRAWING OF UNIT 1 SUMP

FIGURE 6C-21]



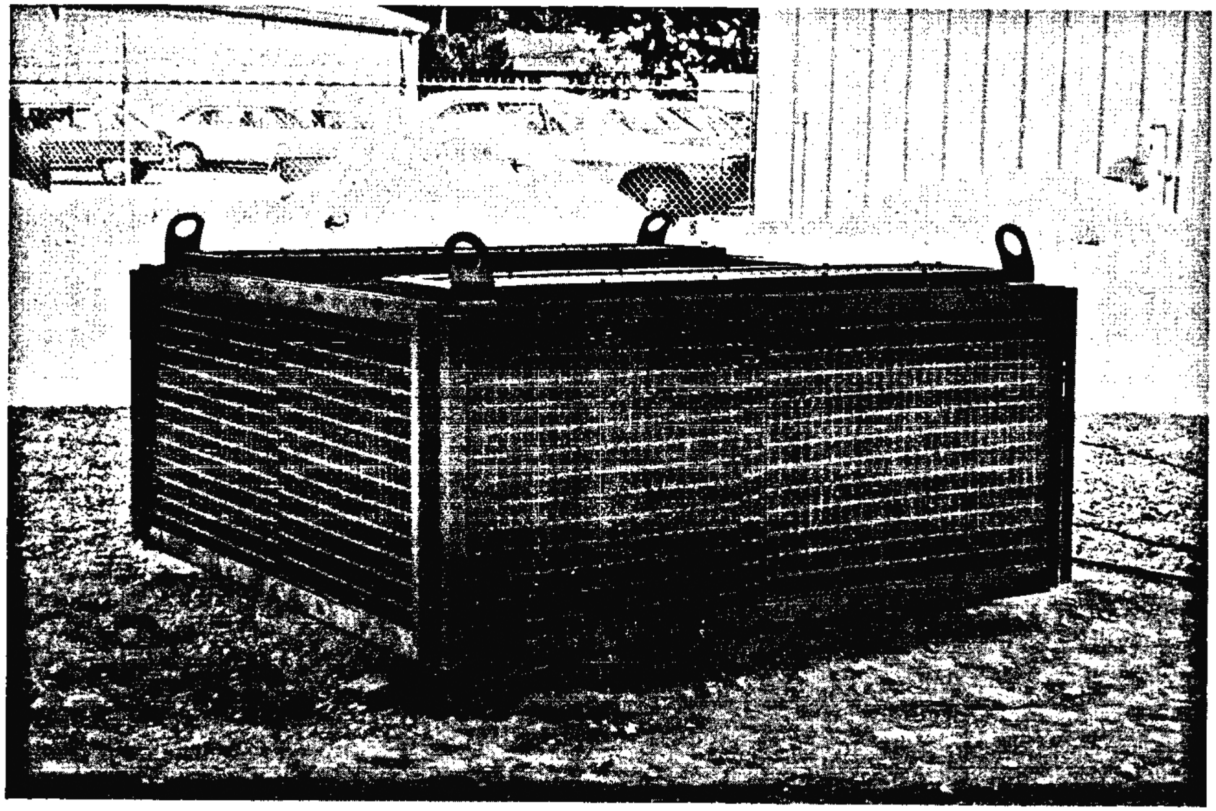
REV 21 5/08

SOUTHERN COMPANY
Energy to Serve Your World®

JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

[PHOTO OF UNIT 2 GRATING CAGE

FIGURE 6C-22]



REV 21 5/08

SOUTHERN COMPANY
Energy to Serve Your World®

JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

[PHOTO OF REPRESENTATIVE SCREEN – GRATING CAGE

FIGURE 6C-23]

APPENDIX 6D

CONTAINMENT SUMP DESCRIPTION AND EMERGENCY CORE COOLING SYSTEM RECIRCULATION SUMP STRAINER DESIGN

TABLE OF CONTENTS

Page

LIST OF TABLES

LIST OF FIGURES

APPENDIX 6D

CONTAINMENT SUMP DESCRIPTION AND EMERGENCY CORE COOLING SYSTEM RECIRCULATION SUMP STRAINER DESIGN

6D.1 CONTAINMENT SUMP DESCRIPTION

6D.1.1 GENERAL PLANT SYSTEM DESCRIPTION

Farley Nuclear Plant (FNP) Units 1 and 2 are Westinghouse three loop Pressurized Water Reactor (PWR) design. The residual heat removal system (RHRS) (low head safety injection), centrifugal charging system (CVCS) (high head safety injection), and containment spray system (CSS) pumps are started following a loss of coolant accident (LOCA). Initially, two RHR, two CVCS, and two CCS pumps take suction from the refueling water storage tank (RWST). When the RWST level reaches the low level setpoint, the RHR pumps are manually stopped and are realigned to take suction from the post-LOCA containment sump. Once the RHR switchover to recirculation is complete, the CVCS pumps take suction from the RHR pump discharge.

When the RWST level reaches low-low level, the CSS pumps are realigned to take suction from the containment sump. There are four independent suctions (two for RHR and two for CSS) located at el 105 ft-6 in. in the containment, the lowest floor elevation in the containment exclusive of the reactor cavity, and they are located outside the secondary shield wall.

The FNP nuclear steam supply system (NSSS) is a three-loop pressurized water reactor (PWR). The system consists of one reactor pressure vessel (RPV), three steam generators (SGs), three reactor coolant pumps (RCPs), one pressurizer (PZR), and the reactor coolant system (RCS) piping. The NSSS is located inside a bioshield and the reactor cavity. The area inside the bioshield is mostly open at the lowest levels, with the exception of the reactor cavity and surrounding walls in the center, and a concrete wall between the A and C loops. The concrete wall between loops A and C has a walkway against the reactor cavity wall that allows an opening between loops A and C. The outer bioshield walls extend from the containment base elevation of 105 ft-6 in. to el 129 ft-0 in. There are areas of the bioshield walls that are partially open; an inner wall extends from el 105 ft-6 in. to 116 ft-3 in., and an outer wall extends down from el. 129 ft-0 in. to el 115 ft-3 in. at some locations. Above el 129 ft-0 in. smaller “vaults” or “coffins” surround each loop and the associated steam generator and reactor coolant pump. These vaults further narrow around the steam generator at el 155 ft-0 in. and extend up to el 166 ft-6 in.. A separate vault for the pressurizer begins at el 129 ft-0 in. and extends up to el 181 ft-0 in.

The containment recirculation sump is a collecting reservoir designed to provide an adequate supply of water, with a minimum amount of particulate matter, to the CSS and the RHRS. The containment sump performance meets the NRC acceptance criteria contained in General Design Criteria 35, 36, and 37, and the NRC acceptance criteria listed below.

A. The net positive suction head (NPSH) available to each safety system pump has been shown to provide adequate margin over the required NPSH at limiting runout conditions (see FSAR paragraph 6.3.2.14).

B. Housekeeping requirements specified in the quality assurance program and the Technical Requirements Manual.

C. The ability to monitor and control RHRS status.

In each of the four pumps suction lines from the containment sump there are two motor-operated gate valves. There is no interdependency between systems or between the redundant portions of the same system.

The motor-operated gate valves in the lines from the containment sump to the various pumps are normally closed and remain closed during the injection phase of emergency core cooling system (ECCS) operation. The protective screened structures in the containment sump will be completely submerged at the end of the injection phase and will remain submerged during the recirculation phase.

6D.1.2 GENERAL DESCRIPTION OF CONTAINMENT SUMP STRAINERS

FNP contracted with General Electric Company (GE) to provide sump strainers that meet the requirements of GL 2004-02. GE provided FNP with seven horizontal stacked disk strainers (see figure 6D-4) and one vertical stacked disk strainer (see figure 6D-3). The strainers were installed in both Unit 1 and Unit 2. Unit 1 only has the vertical stacked strainer installed on the B-train containment spray pump suction. The strainer plate nominal hole size is 3/32 in.

The strainers for FNP Unit 1 and Unit 2 are located outside the biowall between the biowall and CTMT outside wall (see figures 6D-1 and 6D-2). This location protects the strainers from missile impacts.

6D.1.3 SIZE OF CONTAINMENT SUMP STRAINERS

For Unit 1 the passive strainer solution is shown on figure 6D-1. Each strainer assembly for both RHR strainers and CS Alpha strainer consists of two modular horizontal stacked disk strainer subunits connected to the post-LOCA pump suction through piping. The CS Bravo strainer assembly consists of three modular vertical stacked disk strainer subunits connected to a plenum that assists in directing flow to the post-LOCA pump suction inlet located within the plenum boundary. The RHR strainer assembly, either Alpha or Bravo, is composed of two strainer subunits per sump, each consisting of 22 stacked disks that are 40 in. X 40 in. and provide a total of approximately 878 ft^2 of perforated plate surface area. The CS Alpha strainer assembly consists of one strainer subunit with twenty two 40 in. X 40 in. stacked disks and the other with ten 40 in. X 40 in. stacked disks, providing a total of approximately 638 ft^2 of perforated plate surface area. The CS Bravo strainer assembly is composed of three strainer subunits, each with thirteen 30 in. X 30 in. vertical stacked disks, and provides a total of approximately 389 ft^2 of perforated plate surface area.

For Unit 2 the passive strainer solution is shown on figure 6D-2. Each strainer assembly for RHR and CS consists of two modular horizontal stacked disk strainers connected to the sump through piping. The RHR strainer assemblies, both Alpha and Bravo, are composed of two strainers per sump, each consisting of 22 stacked disks that are 40 in. X 40 in. and provide a total of approximately 878 ft^2 of perforated plate surface area. The CS Alpha strainer assembly consists of one strainer with twenty two 40 in. X 40 in. stacked disks and the other with ten 40 in. X 40 in. stacked disks, providing a total of approximately 638 ft² of perforated plate surface area. The CS Bravo strainer assembly is composed of two strainers, one with ten 40 in. X 40 in. stacked disks and the other with twenty two 30 in. X 30 in. disks, and provides a total of approximately 433 ft² of perforated plate surface area.

6D.2 SUMMARY DESCRIPTION OF APPROACH USED TO SIZE SUMP STRAINERS

SNC has performed analysis to determine the susceptibility of the ECCS and CSS recirculation functions for Farley Nuclear Plant to the adverse effects of post-accident debris blockage and operation with debris-laden fluids. These analyses conform to the greatest extent practicable to the NEI 04-07 methodology as approved by the NRC safety evaluation report dated December 6, 2004. Following is a summary description of the analysis areas performed:

6D.2.1 CONTAINMENT WALKDOWN

Walkdown of containment was performed by SNC personnel using the guidance of NEI 02-01. The information obtained from the walkdown confirmed the insulation that was installed in containment matched the design documentation. Containment walkdowns confirmed the general housekeeping condition of containment was being maintained per plant procedures.

6D.2.2 PIPE BREAK CHARACTERIZATION

Pipe break characterization was performed by Sargent and Lundy of Chicago. The piping runs considered for breaks are the RCS hot legs, the RCS cold legs, RCS interim legs, and all RCS attached energized piping. Breaks in these lines could decrease RCS inventory and result in the ECCS and/or CSS operating in recirculation mode, in which the system pumps would take suction from the containment sumps.

Regulatory position 1.3.2.3 of Regulatory Guide 1.82, "Water Sources for Long-Term Recirculation Cooling Following a Loss-of-Coolant Accident," Revision 3, was used to select the spectrum of breaks for evaluation. A summary of the break locations is provided in figure 6D-5.

6D.2.3 DEBRIS GENERATION

The debris generation analysis was performed by Sargent and Lundy of Chicago. The analysis determined the debris generated based on the NEI guidance and NRC SER of the NEI guidance. The analysis determined the ZOI for each type of material identified inside containment. See table 6D-1 for basis of ZOIs.

Insulation found inside containment that is adversely affected during a LOCA event, was determined to consist of a very small quantity of Tempmat fiber, Transco RMI, and Mirror insulations. Most of the insulation is Transco RMI and Mirror RMI. The amount of Tempmat fiber is very small. See table 6D-2 for summary of debris generated by each break.

The limiting break for coatings evaluated for a 4.0D ZOI is also on the intermediate leg of loop B, but at the RCP side of the pipe. Therefore, in order to conservatively maximize the debris available for transport, the maximum insulation debris location (break S2) is combined with the maximum coating debris location. See table 6D-3 for coating debris. Unqualified coatings are also identified in containment walkdown and plant condition reports.

6.D.2.4 LATENT DEBRIS ACCUMULATION WITHIN CONTAINMENT

Programmatic controls are in place at FNP that give bases for the amounts of foreign material and latent debris inside containment remaining below the amounts assumed in the sump analysis. See table 6D-4 for latent and foreign material debris used in the analysis.

6D.2.5 DEBRIS TRANSPORT TO THE SUMP

A debris transport analysis estimated the fraction of debris that is transported from debris sources (break locations) to the sump screen. The transport analysis is in accordance with the guidance of NEI 04-07 and the applicable NRC SER. The computational fluid dynamics (CFD) analysis was performed by RWDI Consulting Engineers and Scientists for Sargent and Lundy of Chicago. The CFD modeling techniques used are consistent with the SER, NEI Document number 04-07, and NUREG/CR-6773.

CFD analyses of the post-LOCA recirculation flow patterns within the FNP containments were performed to quantify the flow velocities expected inside the secondary shield wall, through the secondary shield wall, outside the secondary shield wall, and near the CS and RHR sumps. CFD analysis of the post-LOCA recirculation containment flows indicates velocities that will transport debris to the suction strainers.

The debris quantities transported to the sump strainers for the worst-case break are listed in Table 6D-5. The debris quantities listed in the table were derived using debris transport fractions for a single train failure case. Note that the fiber quantity listed in the table includes fines, small pieces, large pieces, and latent fiber debris.

Table 6D-6 lists the quantities of chemical products generated and transported to the sump strainers. See Section 6D.12 for additional details regarding the generation and transport of chemical products.

6D.2.6 HEAD LOSS AS A RESULT OF DEBRIS ACCUMULATION

The engineered sump screens installed at FNP are designed to operate in such a way that the thin bed effect does not occur on the sump screen surface. This is due to the small amount of fiber present in the FNP containment. Parametric analyses were performed to estimate the surface area of the engineered screen that meets the FNP head loss criterion for the identified debris inventory.

For the limiting break for screen head loss as selected in accordance with NEI 04-07, screens would be fully submerged at the minimum calculated sump levels. The RHR screen height is 44.75 in. above the floor. With leveling shims the height may be increased at points on the screens less than an inch. The minimum calculated water level is 54 in. above the floor elevation which is calculated to occur for the long term and not at the initiation of recirculation. This is largely due to gradual refilling of the area under the reactor vessel and due to conservatively postulated refilling of the SG tubes and the pressurizer. The tallest CS screen is 46.2 in. high; therefore, it may have slightly less submergence. Under this scenario the screens will be fully submerged by no less than 6 in.

A small break LOCA that results in minimum sump level would be one that occurs on top of the pressurizer. This level was not calculated as it is not a limiting break location that results in the highest screen head losses. The connections on the top of the pressurizer are 6 in. in diameter. Therefore, a break in this location would produce very small amounts of debris. In addition, as compared to the limiting large break location, a small break would result in lower sump flowrates and, therefore, reduced sump debris transport. The resultant reduced RHR flowrates would result in a reduction in both debris bed head loss and a reduction in the NPSH required for the RHR pumps. An SBLOCA clearly does not present a significant challenge to the ECCS sump performance and is bounded by a LBLOCA. Since this is not a limiting break location the screen submergence was not calculated for this break.

As the screens are well covered for the limiting breaks the potential for air injection due to buoyant debris accumulation on top of the strainer is not considered to be plausible. For breaks that may result in some transient uncoverage, RHR flowrates would be reduced. CS screens would be fully covered as the RWST level is drawn down further before CS is placed on recirculation.

A vortexing analysis was done for the Farley strainers assuming maximum RHR and CS flowrates. Vortexing was not indicated using the assumption that the strainer has the geometry of an open ended submerged pipe. This conservatively does not account for the complex stacked disc geometry of the strainer which would in effect act as vortex breakers.

6D.2.7 DEBRIS SOURCE TERM REDUCTION

Foreign material (i.e., tags, labels, etc., not qualified for LOCA environmental conditions) may fail following a LOCA and, therefore, can be transported to the sump. Actions have been taken by SNC to ensure that the quantity of foreign material is minimized.

6D.2.8 SUMP STRUCTURAL ANALYSIS

Structural analysis of the engineered passive screen has been completed. SNC has installed an engineered passive strainer on each RHR and CSS containment sump inlet pipe. The screens are located outside the secondary shield wall between the shield wall and the containment wall and, as such, are not exposed to jet impingement or postulated missiles generated from a LOCA event. The screens are of a robust design that support structural and hydraulic load created by the accumulation of debris during the post-LOCA environment. This robust design provides the strength of trash racks and is adequate to protect the screen during a LOCA event.

6D.2.9 UPSTREAM EFFECTS OF DEBRIS ACCUMULATION

Evaluations of containment along with review of the CFD model indicate no significant areas will become blocked with debris and hold up water during the sump recirculation phase. As a precautionary measure, SNC modified the reactor cavity drain covers to further reduce the possibility of the drain becoming clogged and trapping a volume of water in the reactor cavity.

6D.2.10 DOWNSTREAM EFFECTS - COMPONENTS AND SYSTEMS

The downstream impact of sump debris on the performance of the ECCS and containment spray system following a LOCA at FNP Units 1 and 2 has been evaluated. The FNP downstream effects evaluation uses the methodology presented in WCAP-16406-P-A, Revision 1. The effects of debris ingested through the containment sump strainers during the recirculation mode include erosive wear, abrasion, and potential blockage of flow paths. The smallest clearance found for the FNP Units 1 and 2 heat exchangers, orifices, and spray nozzles in the recirculation flow paths is 0.375 inches (3/8") for the containment spray nozzles. No blockage of the containment spray flow paths is expected with a sump strainer hole size of 0.09375 inch (3/32").

The instrumentation tubing is also evaluated for potential blockage of the sensing lines. The transverse velocity past this tubing is determined to be sufficient to prevent debris settlement into these lines, so no blockage will occur. The reactor vessel level instrumentation system (RVLIS) is also evaluated, and no effect on its performance is expected by debris.

For pumps, the effect of debris ingestion through the sump strainer on three aspects of operability, including hydraulic performance, mechanical shaft seal assembly performance, and the mechanical performance (vibration) of the pump, were evaluated. The hydraulic and mechanical performances of the pump were determined to not be affected by the recirculating sump debris. The mechanical shaft seal assembly performance evaluation resulted in the one action item with suggested replacement of the RHR pumps' carbon/graphite backup seal bushings with a more wear resistant material, such as bronze. However, FNP has an Engineered Safety Feature (ESF) atmospheric filtration system in its auxiliary building and this action is not required.

Evaluations of the system valves showed that the minimum recirculation flow rates are adequate to preclude debris sedimentation in all cases. All of the valves that are subject to being blocked pass the plugging criteria at their current positions, since the strainer mesh size is smaller than the minimum valve clearance. In order to pass the valve plugging criteria, the safety injection throttle valves were modified on Units 1 and 2. A flow reducing orifice was installed and the valves were replaced. All of the valves that are subject to erosion pass the acceptable criteria for the mission time of 30 days.

6D.2.11 DOWNSTREAM EFFECTS – FUEL AND VESSEL

Methods and results contained in WCAP-17788-P, Revision 1, were used to evaluate the accumulation of fiber inside the reactor vessel (Reference 6). During the post-LOCA sump recirculation phase, debris ingested by the ECCS could accumulate at the reactor core inlet or inside the reactor vessel, potentially challenging long-term core cooling. The quantity of fiber accumulation inside the reactor vessel was calculated for the worst-case hot leg break scenario and compared to the in-vessel debris limits defined in WCAP-17788-P, Revision 1. The calculation used containment accident generated and transported debris quantities from the break location that generated the largest quantity of fibrous debris. A conservative debris bypass fraction of 45% based on the NEI clean plant criteria was used in the evaluation. The calculated quantity of fiber accumulation in the reactor vessel for FNP meets the limit defined by WCAP-17788-P, Revision 1. Thus, the accumulation of fibrous debris in the reactor vessel will not challenge the ability to maintain adequate long-term core cooling at FNP.

The NRC has not generically approved WCAP-17788-P, Revision 1 for use and an evaluation was performed to demonstrate applicability of the methods and results to FNP. The applicability evaluation compares the values of key parameters assumed in the WCAP-17788 analysis to FNP specific values. The key parameter comparison is summarized in Table 6D-7. The evaluation concludes that the WCAP-17788-P, Revision 1 methods and results are applicable to FNP.

The effects of in-vessel downstream chemical effects are discussed in Section 6D.2.12.

6D.2.12 CHEMICAL EFFECTS

The new strainers installed at FNP have been sized to account for some increase in head loss across the strainer as a result of interaction of the sump water with the debris material as it approached the strainers during recirculation phase. The methodologies of the base model WCAP-16530-NP, "Evaluation of Post-Accident Chemical Effects in Containment Sump Fluids to Support GSI-191," Revision 0 (reference 10), as modified by NRC safety evaluation report dated December 21, 2007 (reference 11), were used to evaluate the impact of chemical precipitants on the containment sump screens during post-accident recirculation and the resulting effect on available NPSH for the ECCS and CSS pumps. SNC supplemented the chemical effects results with plant-specific test data that demonstrated that the aluminum precipitants do not form until the containment sump temperature drops below 140 ºF (see reference 15). Calculations using the chemical effects testing results and other inputs demonstrated the available NPSH margin for the ECCS and CSS pumps was adequate for the conditions expected during post-accident recirculation. The details of the chemical effects testing results are documented in GE Report 0000-0056-2976, Containment Sump Passive RHR & CS Strainer System S0100 Hydraulic Sizing Report, Revision 3 (reference 15).

Autoclave chemical effects testing documented in WCAP-17788-P Volume 5, Revision 1 investigated post-accident corrosion, dissolution, and precipitation reactions to determine the earliest time that chemical products are expected to be generated inside containment. FNP has demonstrated that the generation of chemical products will be sufficiently delayed by comparing the FNP prototypic plant conditions to the autoclave test conditions. The comparison concluded that the generation of chemical products will be delayed until after 24 hours, which is greater than the time that complete core inlet blockage can be tolerated (2.38 hours) and before the completion of transfer to hot leg recirculation.

6D.2.13 ANALYZED DEBRIS LIMITS

Containment accident generated and transported debris is defined as the quantity of debris calculated to arrive at the containment sump strainers. The limiting debris quantities considered in the sump strainer head loss, downstream ex-vessel and in-vessel effects evaluations described above are used to define the analyzed debris limits shown in Table 6D-8.

REFERENCES

1. NRC Generic Letter (GL) 2004-02, "Potential Impact of Debris Blockage on Emergency Recirculation during Design Basis Accidents for Pressurized-Water Reactors," September 13, 2004.

2. Nuclear Energy Institute (NEI) document NEI 04-07 Revision 0, "Pressurized Water Reactor Sump Performance Evaluation Methodology," December 2004.

3. Safety Evaluation by the Office of Nuclear Reactor Regulation Related to NRC Generic Letter 2004-02, Nuclear Energy Institute Guidance Report (Proposed Document Number NEI 04-07), "Pressurized Water Reactor Sump Performance Evaluation Methodology," December 6, 2004.

4. Regulatory Guide 1.82, "Water Sources for Long Term Recirculation Cooling Following a Loss of Coolant Accident," Revision 3, November 2003.

5. WCAP-16568-P, "Jet Impingement Testing to Determine the Zone of Influence (ZOI) for DBA-Qualified / Acceptable Coatings," Revision 0.

6. Westinghouse letter ALA-20-118 from E. Husser to C. Kharrl (SNC), “Farley Unit 1 and Unit 2 GSI-191 In-Vessel Debris Evaluation Final Deliverable,” December 17, 2020.

7. Deleted.

8. WCAP-16406-P, Evaluation of Downstream Sump Debris Effects in Support of GSI-191,” Revision 1.

9. NRC SER dated December 20, 2007, Safety Evaluation by the Office of Nuclear Reactor Regulation Topical Report (TR) WCAP-16406-P, Revision 1, “Evaluation of Downstream Sump Debris Effects in Support of GSI-191,” Pressurized Water Reactor Owners Group.

10. WCAP-16530-NP, “Evaluation of Post-Accident Chemical Effects in Containment Sump Fluids to Support GSI-191.”

11. NRC SER dated December 21, 2007, Final Safety Evaluation by the Office of Nuclear Reactor Regulation Topical Report WCAP-16530-NP, “Evaluation of Post-Accident Chemical Effects in Containment Sump Fluids to Support GSI-191.”

12. WCAP 16793-NP, Evaluation of Long-Term Cooling Considering Particulate, Fibrous and Chemical Debris in the Re-circulating Fluid, Revision 0.

13. SNC Letter NL-08-2173 dated February 28, 2008, “Joseph M. Farley Nuclear Plant Supplemental Response to NRC Generic Letter 2004-02.”

14. SNC Letter NL-08-0551 dated April 29, 2008, “Joseph M. Farley Nuclear Plant Final Supplemental Response to NRC Generic Letter 2004-02.”

15. GE Report 0000-0056-2976 [U-732504], “Containment Sump Passive RHR & CS Strainer System S0100 Hydraulic Sizing Report,” Revision 3.

TABLE 6D-1

CONTAINMENT SUMP DEBRIS GENERATION ZONE OF INFLUENCE (ZOI)

Debris Constituent	ZOI (Pipe Diameter)	Basis
Transco RMI	2.0D	NRC SER
Mirror RMI	28.6D	NRC SER
Temp-Mat Fiber	NA	All assumed as debris in analysis
Qualified Coatings	4.0D	WCAP-16568-P
Unqualified Coatings	NA	NRC SER – All assumed as debris in analysis
Latent Debris	NA	NRC SER – Conservative value based on plant walkdown
Foreign Materials (Labels, etc.)	NA	NRC SER – Conservative value based on plant walkdown

TABLE 6D-2

SUMMARY OF LOCA GENERATED INSULATION DEBRIS INSIDE ZOI

Break ID	Location	Transco RMI Foils (ft^2)	Mirror RMI Foils (ft^2)	RMI Jacketing (ft^2)	Temp-Mat (ft^3)
S1	Loop C Interim Leg	2054	25527	5795	1
S2*	Loop B Interim Leg	2383	35714	8022	1
S3	Loop A Cold Leg	0	34368	7522	1
S4 (alternate)	Loop B Interim Leg	1226	23258	5223	0

* S2 is the limiting location.

TABLE 6D-3

DEBRIS GENERATED FROM COATING BASED ON ZOI = 4D

Break	Coating Areas (ft^2)		Coating Volumes (ft^3)	
	Concrete	Steel	Concrete	Steel
Interim Leg at SG	200	1332	0.31	1.66
Interim Leg at Mid-span	218	1320	0.34	1.65
*Interim Leg at RCP	523	1091	0.81	1.36
Hot Leg at Primary Wall	294	758	0.46	0.95
Hot Leg at SG	0	1196	0	1.49
Unqualified Coatings	NA	1,070	NA	0.535

* Limiting location for coatings

TABLE 6D-4

LATENT AND FOREIGN MATERIAL DEBRIS USED IN ANALYSIS

Latent Debris Total	(lb_m)	200
Fiber	(lb_m)	30
Particulate	(lb_m)	170
Foreign Material Debris	(ft^2)	36.4

TABLE 6D-5

SUMMARY OF DEBRIS GENERATED AND TRANSPORTED TO STRAINER MODULES

Debris Type		Units	Quantity Generated	Transport Fraction	Quantity at Strainer Modules
			Fibrous Insulation Debris		
Temp-Mat		[ft^3]	1	1.0	1
			Qualified Coating Debris		
Concrete Coatings		[ft^2; ft^3]	523 ; 0.81	0.871	456 ; 0.71
Steel Coatings		[ft^2; ft^3]	1091 ; 1.36	0.704	768 ; 0.96
	Sum	[ft^2; ft^3]	1614 ; 2.18	——	1224 ; 1.67
			Unqualified Coating Debris Modeled		
Unqualified Coatings (Tested)		[ft^2; ft^3]	1070 ; 0.535	1.0	1070 ; 0.535
			Latent Debris		
Latent Fiber (Walkdown)		[ft^3]	7.8	1.0	7.8
Latent Fiber (30 lb_m)*		[ft^3]	12.5	1.0	12.5
Latent Particulate (Walkdown)		[ft^3]	0.63	1.0	0.63
Latent Particulate (170 lb_m)*		[ft^3]	1.01	1.0	1.01
			Reflective Metal Insulation Debris		
Transco Foil		[ft^2]	2383	0.799	1904
Mirror Foil		[ft^2]	35714	0.769	27464
	Foil Sum	[ft^2]	38097	——	29368
RMI Jacketing		[ft^2]	8022	0.338	2711
			Foreign Material		
Foreign Material[1] (labels, stickers, etc.)		[ft^2]	36.4	1.0	36.4

* Used for latent debris evaluation

TABLE 6D-6

SUMMARY OF MATERIAL QUANTITIES USED TO DETERMINE CHEMICAL PRODUCT GENERATION

Material	Units	Quantity
Aluminum Submerged	[ft^2]	1,741*
Aluminum Non-Submerged	[ft^2]	15,667*
Temp-Mat	[ft^3]	1
Concrete	[ft^2]	523
Trisodium Phosphate	[lbm]	13,133

Chemical Precipitates

Debris Type	Units	Quantity Generated	Transport Fraction	Quantity at Strainer Modules
Calcium Phosphate	[lbm]	0.70	1.0	0.70
Sodium Aluminum Silicate	[lbm]	7.22	1.0	7.22
Aluminum Oxy Hydroxide	[lbm]	729.67	1.0	729.67

*Aluminum values shown on this table represent those used for strainer chemical effects analysis; different values were used for in-vessel chemical effects.

TABLE 6D-7

IN-VESSEL DEBRIS EFFECTS KEY PARAMETER EVALUATION

Parameter	WCAP-17788 Value	FNP Value	Evaluation
Minimum Sump Switchover Time (min)	20	21	Later switchover time results in a lower decay heat at the time of debris arrival, reducing the potential for debris induced core uncovery and heatup.
Maximum Hot Leg Switchover Time (hr)	24(t_{chem})	7.5	Latest hot leg switchover occurs well before earliest potential chemical product generation.
Rated Thermal Power (MWt)	3658	2821	Lower thermal power results in lower decay heat.
Maximum AFP Resistance	WCAP-17788 Volume 4 Table 6-1	WCAP-17788 Volume 4 Table RAI-4.2-24	AFP resistance is less than the analyzed value, which increases the effectiveness of the AFP.
Minimum ECCS Recirculation Flow (gpm/FA)	8	8.9	Maximum debris bed resistance at the core inlet occurs at lower flow rates.

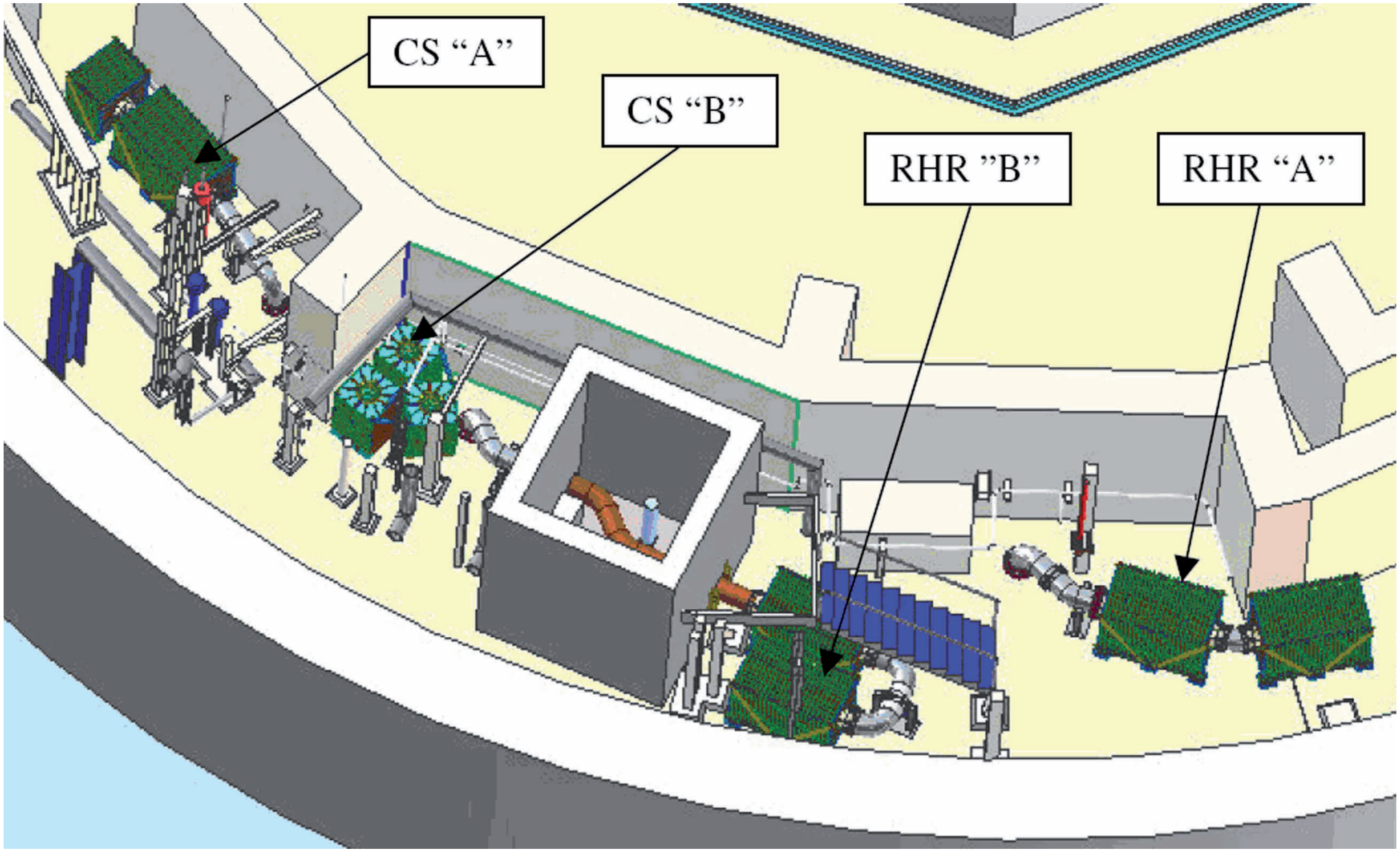
TABLE 6D-8

SUMMARY TABLE OF ANALYZED DEBRIS LIMITS

Debris Type	Units	Analyzed Limit
Temp-Mat	[ft^3]	1
RMI Foils	[ft^3]	6.12*
Qualified Coatings	[ft^3]	1.67
Unqualified Coatings	[ft^3]	0.535
Latent Debris	[lbm]	200**
Foreign Materials (labels, stickers, etc.)	[ft^2]	72.8
Aluminum	[ft^2]	17,049

*The RMI foil volume was determined by multiplying the total area of RMI destroyed by a foil thickness of 2.5 mils. RMI Foil was not used in the head loss test since the presence of RMI in the debris bed tended to decrease overall head loss. The value in this table is the RMI foil transported, not tested.

**Using a split of 85% dirt/dust and 15% fiber, along with the accepted fiber density of 2.4 lbm/ft^3 and dirt/dust density of 169 lbm/ft^3, this is equivalent to 12.5 ft^3 fiber and 1.0 ft^3 dirt/dust.



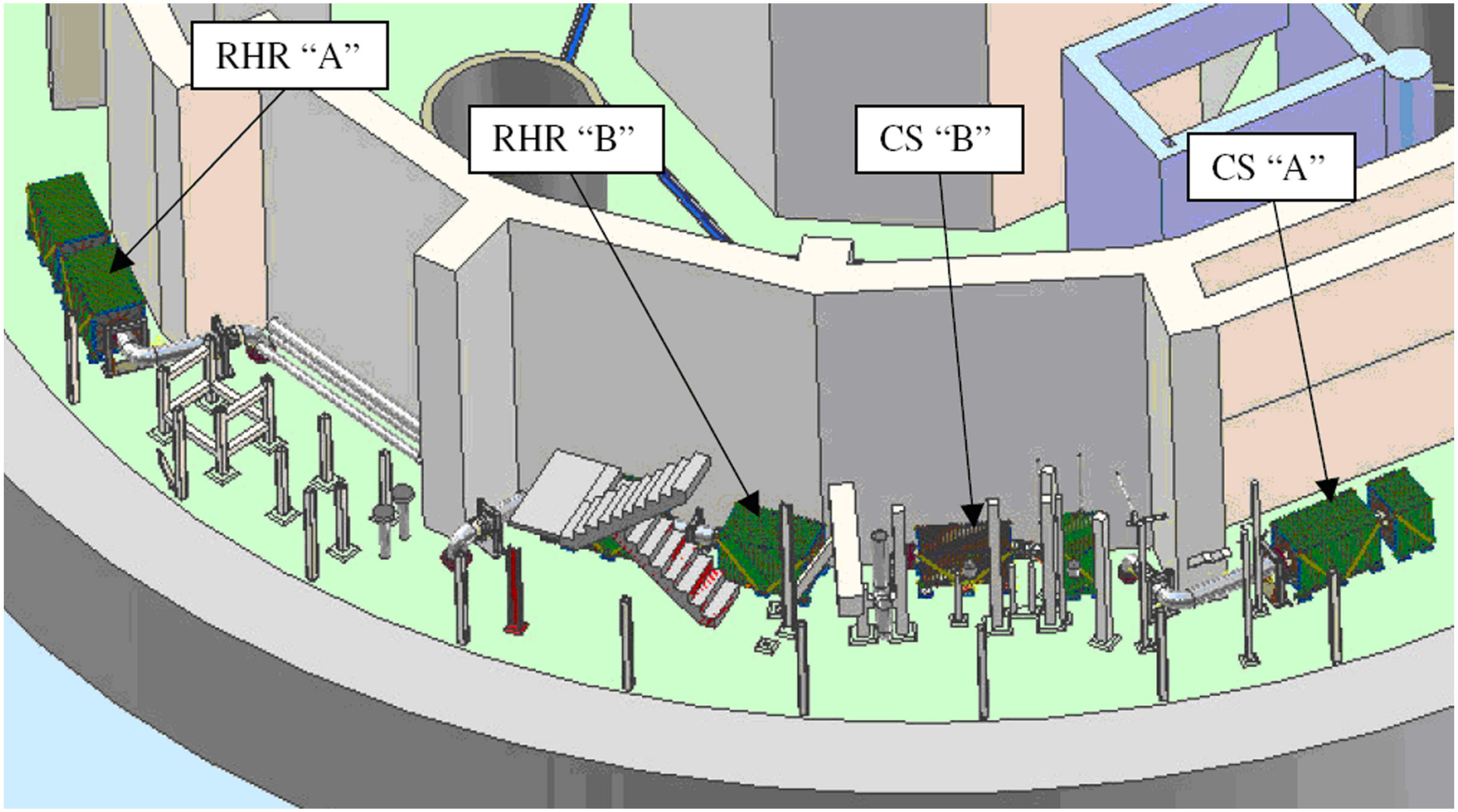
REV 21 5/08

SOUTHERN COMPANY
Energy to Serve Your World®

JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

FARLEY UNIT 1 STRAINER LAYOUT

FIGURE 6D-1



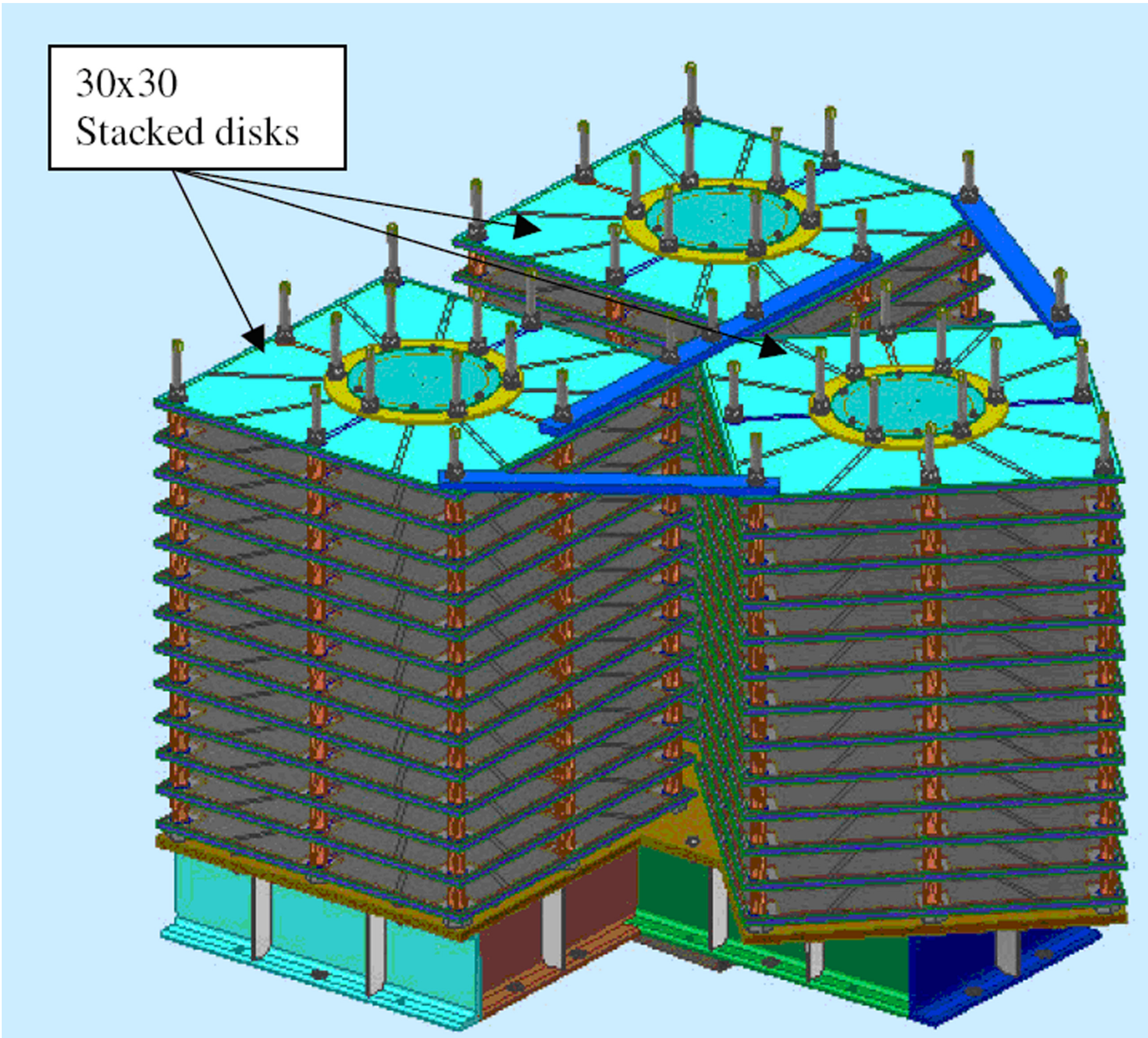
REV 21 5/08

SOUTHERN COMPANY
Energy to Serve Your World®

JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

FARLEY UNIT 2 STRAINER LAYOUT

FIGURE 6D-2



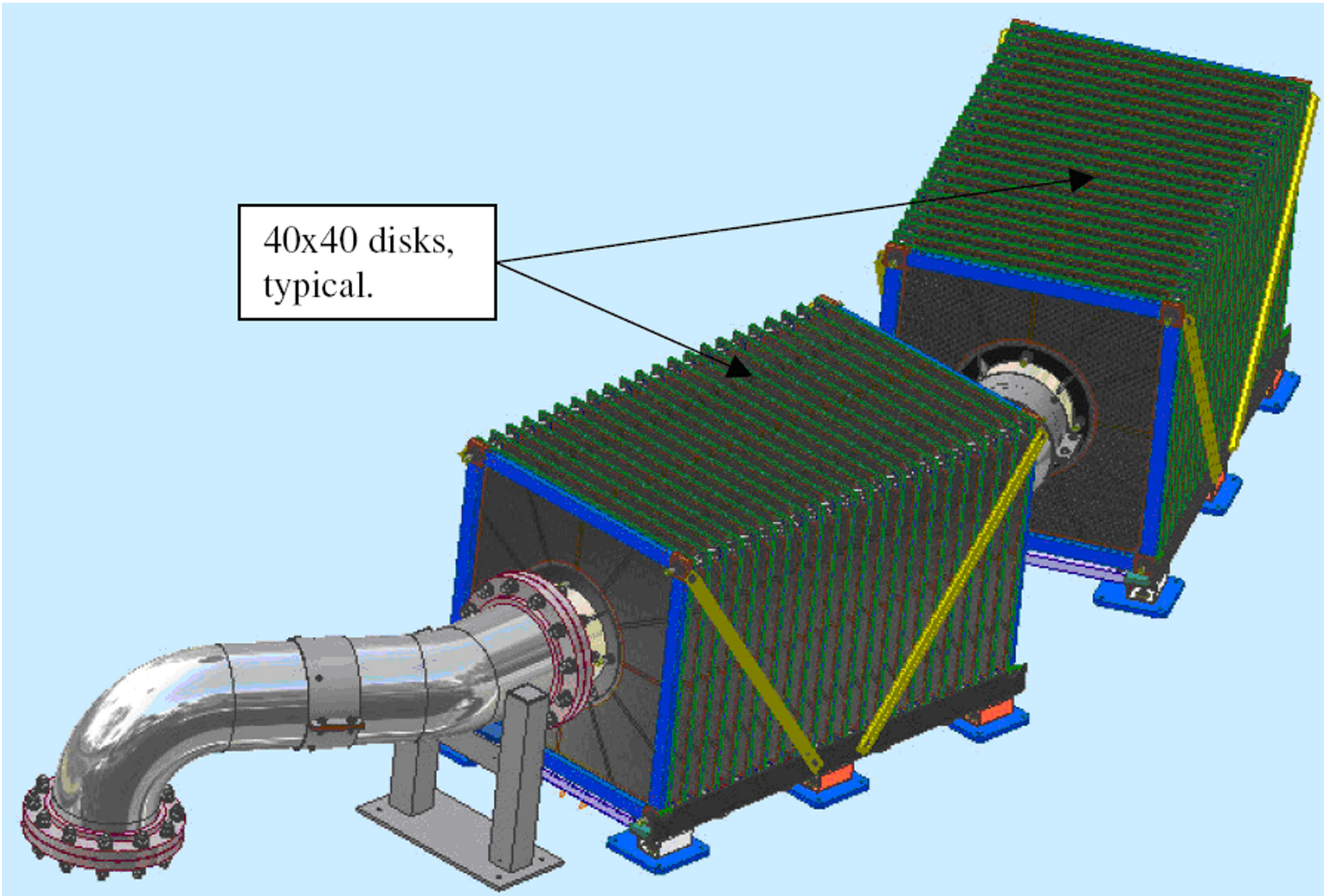
REV 21 5/08

SOUTHERN COMPANY
Energy to Serve Your World®

JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

VERTICAL STRAINER TYPE

FIGURE 6D-3



REV 21 5/08

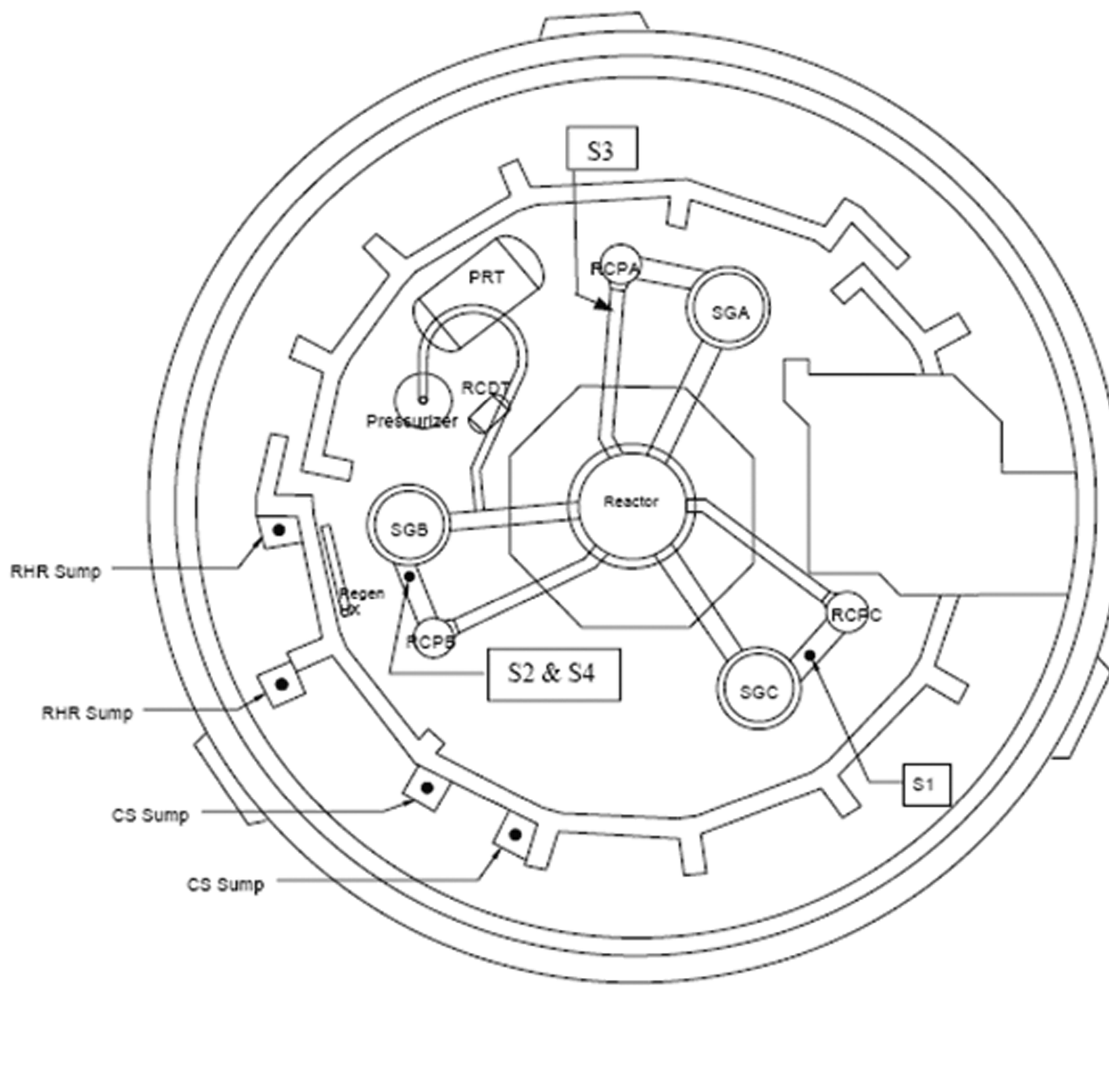
SOUTHERN COMPANY
Energy to Serve Your World®

JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

HORIZONTAL STRAINER TYPE

FIGURE 6D-4

Break Name	Break ID	Elevation	Piping
S1	31-inch	118'-0"	Interim Leg – Loop C
S2	31-inch	118'-0"	Interim Leg – Loop B
S3	27.5-inch	122-9"	Cold Leg – Loop A
S4	11.19-inch	118'-0"	Alternate Break (Interim Leg –Loop B)



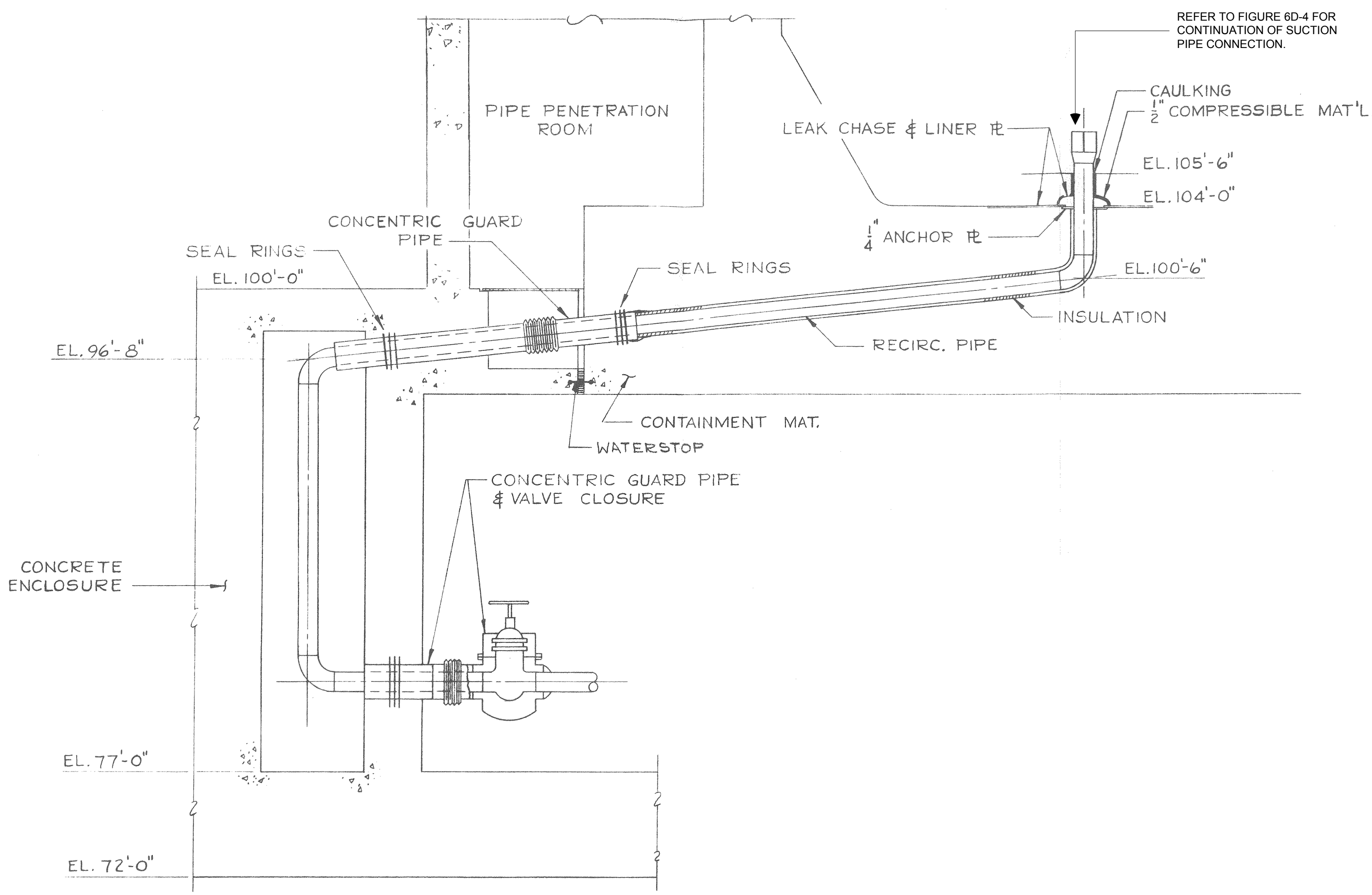
REV 21 5/08

SOUTHERN COMPANY
Energy to Serve Your World®

JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

POSTULATED BREAK LOCATIONS

FIGURE 6D-5



REV 22 8/09

SOUTHERN COMPANY
Energy to Serve Your World®

JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

TYPICAL ARRANGEMENT OF
CONTAINMENT SUMP SUCTION LINE

FIGURE 6D-6