North Anna Power Station
Updated Final Safety Analysis Report

Chapter 6

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CHAPTER 6 ENGINEERED SAFETY FEATURES

6.1 GENERAL

Note: As required by the Renewed Operating Licenses for North Anna Units 1 and 2, issued March 20, 2003, various systems, structures, and components discussed within this chapter are subject to aging management. The programs and activities necessary to manage the aging of these systems, structures, and components are discussed in Chapter 18.

In the unlikely event of a loss-of-coolant accident (LOCA), the engineered safety features (ESF) will serve to mitigate the consequences of the accident and will protect the public by preventing or minimizing the release of fission products. The engineered safety features are designed to provide emergency coolant to ensure the structural integrity of the core and to maintain the integrity of the containment structure during accident conditions, thereby preventing or minimizing the release of fission products to the environment.

The following engineered safety features, each separate and independent, are provided to satisfy the functions indicated:

1. Containment Structure

   The containment structure is a cylindrical, carbon-steel-lined, reinforced concrete structure with a hemispherical dome including foundations, access openings, and penetrations (Chapter 3), which contains mechanical systems, components, and major piping within the reactor coolant pressure boundary.

   During normal operation and subsequent to a LOCA, the containment structure is maintained at a subatmospheric pressure to limit the peak pressure attained during an accident and to minimize outleakage after an accident.

   Assuming the proper operation of other ESF systems, the containment structure is designed to contain the release of radioactive fluids and fission products resulting from postulated accidents within the containment structure. The containment structure is described in detail in Sections 6.2.1 and 3.8.2.

2. Containment Depressurization System

   The integrity of the containment structure is ensured by the containment depressurization system, consisting of the following:

   a. Quench spray subsystem.

   b. Recirculation spray subsystem.

The combination of these subsystems is capable of cooling and depressurizing the containment structure to less than 2.0 psig in 1 hour and to subatmospheric pressure in less than 6 hours following a LOCA. The containment recirculation spray subsystem is capable of maintaining the subatmospheric pressure inside the containment structure following the LOCA.

Caustic (sodium hydroxide, NaOH) is added to the quench spray to reduce the concentration of radioiodine in the containment structure available for leakage.

The containment depressurization system is described in detail in Section 6.2.2.1.

3. Emergency Core Cooling System (ECCS)

   The ECCS will provide borated emergency cooling water to the reactor core for the entire spectrum of reactor coolant system (RCS) break sizes to limit core temperature, maintain core integrity, and provide negative reactivity for additional shutdown capability.

   The ECCS will automatically commence safety injection of water into the RCS on receipt of a safety injection actuation signal. During the injection mode (between start of the LOCA and attainment of low level in the refueling water storage tank (RWST), two charging pumps and two low head safety injection pumps will deliver chilled borated water from the RWST to the RCS. The charging pumps will discharge the water through the boron injection tank, which contains a concentrated boric acid solution for chemical shutdown. In addition, three nitrogen-pressurized accumulators, which require no initiation signal, will inject borated water into the RCS. When the RWST level reaches a low-level setpoint, the ECCS pumps automatically are aligned to take suction from the sump to provide long-term cooling for the reactor core. As a backup to the automatic function, the operator in the control room can manually complete the switchover prior to a minimum level in the RWST. The RWST will be isolated and the low head safety injection pumps will supply water from the containment sump to the RCS. The ECCS is described in detail in Section 6.3.

4. Containment Isolation System

   To ensure containment structure integrity following a LOCA, containment isolation valves are installed in the piping that penetrates the containment structure. Except as indicated in Section 6.2.4.2, the containment isolation valves are located inside and outside of the containment structure as close as possible to the penetrations and are either check valves, manual valves (normally closed), or control valves that will close automatically on receipt of a safety injection signal (SIS), containment isolation phase A signal (CIA), containment isolation phase B signal (CIB), or steam line isolation (SLI) signal.

   Section 6.2.4 describes the containment isolation system and Section 7.3 describes the actuation of the isolation valves.

A schematic of engineered safety features systems is shown in Figure 6.1-1.

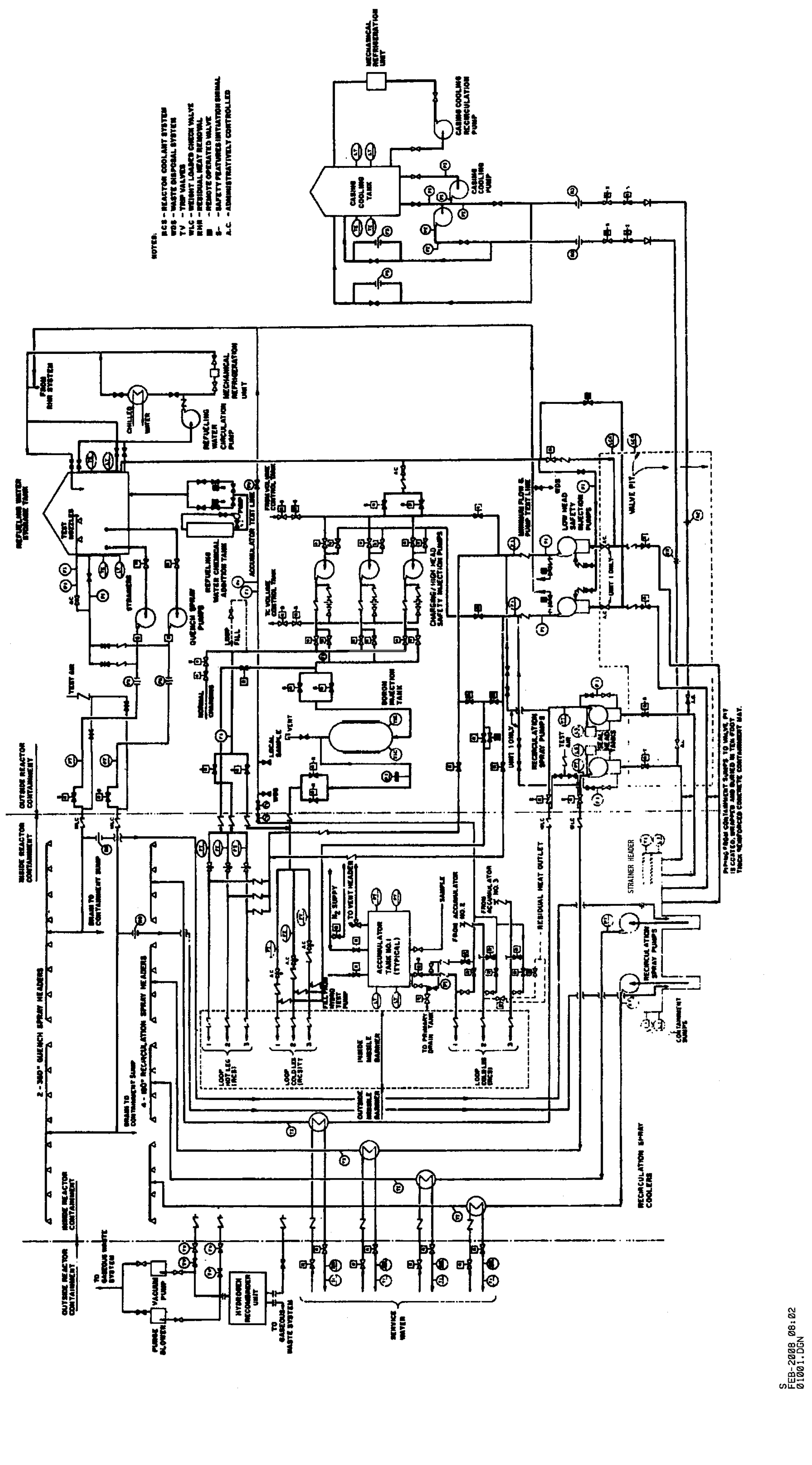
Each ESF system is designed with sufficient redundancy to provide the system safety function assuming a single failure (see Appendix 6A). Active components of the ESF system are powered from the emergency buses (Section 8.3.1). Emergency diesel generators are provided to ensure highly reliable power sources to the emergency buses should offsite power sources fail.

The operability of the ESF equipment is ensured in several ways: some of the equipment, such as the charging pumps, function during normal unit operation, thus providing a continuous check on operational status. The balance of the ESF equipment, such as the pumps in the containment depressurization system, will function only in the event of an accident. In this case, system and equipment design permits periodic testing. Testing is described in the applicable system testing sections.

To ensure that a high quality level was obtained in the ESF components and system, a program of quality assurance was in effect during the design, manufacture, installation, and testing of the ESF systems. The quality assurance program is described in Chapter 17.

The habitability systems for the control room are provided, as described in Section 6.4, to ensure continuous occupancy of the area during and after natural phenomena, fire, and missiles, as discussed in Chapter 3, as well as for all postulated accidents, discussed in Chapter 15, which may or may not release radioactivity to the environment. Protection against the effects of toxic materials, which could overcome control room operators, is also provided by the habitability systems in accordance with Regulatory Guides 1.78 and 1.95.

Figure 6.1-1
ENGINEERED SAFETY FEATURES SYSTEM



6.2 CONTAINMENT SYSTEMS

6.2.1 Containment Functional Design

6.2.1.1 Design Basis

6.2.1.1.1 Containment

A new analysis of containment peak pressure and depressurization following a LOCA was performed to support the installation of the GSI-191 strainer modifications at Unit 1 and Unit 2. The mass and energy releases for this analysis are based on the models described in Section 6.2.1.1.1.3. The new analysis includes the model 51F steam generator parameters.

6.2.1.1.1.1 *Design Criteria.* The design of the subatmospheric containment structure is based on the following criteria:

1. The peak calculated containment atmosphere pressure shall not exceed the design pressure of 45 psig.
2. The containment shall be depressurized following a loss-of-coolant accident (LOCA) to subatmospheric in less than 6 hours. During the period of 1-6 hours after a LOCA, containment pressure shall not exceed 2.0 psig.
3. Once depressurized, the containment shall be maintained at a pressure less than 1 atm. absolute for the duration of the accident.

The peak containment pressure is a function of the initial total pressure and average temperature of the containment atmosphere, the containment free volume, the passive heat sinks in the containment, the quench spray (QS) system design, and the rates of mass and energy released to the containment. The passive heat sinks in the containment are considered to be at the same initial temperature as the initial average containment atmosphere temperature. Maximizing the initial containment total pressure and average atmospheric temperature maximizes the calculated peak pressure.

The time required to depressurize the containment and the capability to maintain it depressurized below 1 atm. of pressure after a double ended pump suction guillotine (DEPSG) depends on the mass of air in the containment, on the design of the containment depressurization system (both QS and recirculation spray (RS) subsystems, see Section 6.2.2.1), and on the service water temperature. When the service water temperatures are elevated, it is more difficult to depressurize the containment to subatmospheric conditions (design criteria 2, above). Therefore, the containment air partial pressure must be reduced to values that are acceptable at high service water temperatures.

In summary, the containment structure is sized for the cold service water conditions (design criteria 1, above) and the containment depressurization system (see Section 6.2.2.1) is sized in accordance with design criteria 2 and 3, above, for the warm service water conditions. Table 6.2-1

is a summary of the range of operating conditions for which the containment meets the design criteria.

All three of the above design criteria are met by varying the containment air partial pressure as a function of the service water temperature. Permissible air partial pressure as a function of service water temperature is given in the Technical Specifications.

The postulated LOCA is defined as a double-ended rupture (DER) of a reactor coolant pipe. The reactor is assumed to be operating at the maximum licensed core thermal power of 2940 MWt plus 0.37% calorimetric error (total core power of 2951 MWt) and to have been operating at this power long enough to have reached its equilibrium concentration of fission products. Coincident with the DER is a complete loss of all offsite electric power. One emergency generator on the plant experiencing the LOCA (four generators are provided for two units) is started and operates to supply emergency power.

Minimum engineered safety features that are activated to limit the consequences of a LOCA are the following:

1. All of three nitrogen-pressurized accumulators discharge into the reactor coolant system (RCS).
2. Emergency core cooling by:
    a. One out of three charging pumps.
    b. One out of two low-head safety injection (LHSI) pumps.
3. Containment depressurization by:
    a. One out of two trains of the containment QS subsystem and
    b. One out of two trains of the containment RS subsystem (i.e., one inside and one outside recirculation spray pump, with the associated casing cooling pump).

The emergency diesel generator provides the power to operate the pumps. The accumulators are passive and discharge into the RCS when the RCS pressure drops below the accumulator pressure. The ECCS limits the extent of the zirconium-water reaction (see Section 6.3).

The amounts of mass (steam and/or water) and energy that are released to the containment structure are time-dependent variables depending upon the pipe break size. After the rupture occurs, the reactor coolant flows out of the break and flashes, raising the temperature and pressure inside the containment atmosphere. Sensible heat energy stored in the hot metal of the reactor vessel, piping, and core, the fission product decay heat, and power coastdown heat are transferred to the reactor coolant and hence into the containment atmosphere.

6.2.1.1.1.2 *GOTHIC Computer Code.* Analyses for the study of the effects on the containment of high energy line breaks were made using the GOTHIC computer program (Reference 51). This

program calculates the temperature and pressure of the containment atmosphere as a function of time following a LOCA or a main steam line break (MSLB) inside containment. The accident starts with a break in the line, and this event is used as the zero reference time for the accident analysis. The program considers the various heat sources and sinks as a function of time in a given containment configuration to calculate the temperature and pressure transients of the containment atmosphere.

The GOTHIC computer program which is used to model the containment system, the passive heat sinks, and the containment heat removal systems, was developed for the Electric Power Research Institute (EPRI) by Numerical Applications, Inc. A topical report, DOM-NAF-3-0.0-P-A (Reference 51), describes in detail the assumptions used and the mathematical formulations employed. The use of GOTHIC for containment analysis has been approved by the NRC as documented in Reference 51. The NRC approved the specific application of GOTHIC at North Anna for the containment design analyses that were performed for the NRC GSI-191 project in Reference 52.

GOTHIC solves the conservation equations for mass, momentum, and energy for multi-component, multi-phase flow in lumped parameter and/or multi-dimensional geometries. The phase balance equations are coupled by mechanistic models for interface mass, energy and momentum transfer that cover the entire flow regime from bubbly flow to film/drop flow, as well as single phase flows. The interface models allow for the possibility of thermal non-equilibrium between phases and unequal phase velocities, including countercurrent flow. GOTHIC includes full treatment of the momentum transport terms in multidimensional models, with optional models for turbulent shear and turbulent mass and energy diffusion. Other phenomena include models for commonly available safety equipment, heat transfer to structures, hydrogen burn and isotope transport.

Passive Heat Sinks

Thermal conductors are the primary heat sink for the blowdown energy. The conductors can be made up of any number of layers of different materials. One-dimensional conduction solutions are used to be consistent with the lumped modeling approach.

The thermal conductor is divided into regions, one for each material layer, with an appropriate thickness and material property for each region. GOTHIC accepts inputs for material density, thermal conductivity and specific heat. These values are obtained from published literature for the materials present in each conductor. Conductors with high heat flux at the surface and low thermal conductivity must have closely spaced nodes near the surface to adequately track the steep temperature profile. The node spacing is set so the Biot number for each node is less than 0.1. The Biot number is the ratio of external to internal conductance.

It is not practical or necessary to model each individual piece of equipment or structure in the containment with a separate conductor. Smaller conductors of similar material composition can be combined into a single effective conductor. In this combination, the total mass and the total

exposed surface area of the conductors is preserved. The thickness controls the response time for the conductors and is of secondary importance. The conductors are grouped by thickness and material type. The containment heat sinks are grouped into the following categories:

1. Containment structure shell below grade
2. Containment structure shell above grade
3. Containment structure dome and liner
4. Containment structure floor above floor liner
5. Containment structure mat below floor liner
6. Internal concrete slabs
7. Carbon steel inside the containment
8. Stainless steel inside the containment
9. Accumulator tanks filled with water (MSLB only)

The effective thickness for a group of wall conductors is calculated by the equation below. The heat sink material types, surface areas, and thickness are derived based on plant-specific inventories. Concrete, carbon steel, and stainless steel are the most common materials.

$$t_{eff} = \frac{\sum_{i \in group} t_i A_i}{\sum_{i \in group} A_i}$$

If there is a small air gap or a contact resistance between the containment liner and the concrete, it is modeled as a separate material layer at the nominal gap thickness with applicable material properties. This overestimates the contact resistance because convection and radiation effects will be ignored. A maximum gap conductance of 40 Btu/hr-ft^2-F is used. The gap width is determined by dividing the gap thermal conductivity by the gap conductance.

All containment passive heat sinks are included in the lumped containment volume. The primary system metal and SG secondary shells are included in the simplified RCS model that is used for the calculation of long-term mass and energy release; however, these conductors are not used for condensation or convection heat transfer with the containment atmosphere.

Conductor Surface Heat Transfer

The Direct heat transfer option with the Diffusion Layer Model (DLM) condensation option is used for all containment passive heat sinks except the sump floor. With the Direct option, all condensate goes directly to the liquid pool at the bottom of the volume. The effects of the condensate film on the heat and mass transfer are incorporated in the formulation of the DLM

option. Under the DLM option, the condensation rate is calculated using a heat and mass transfer analogy to account for the presence of non-condensing gases.

For a conductor representing the containment floor or sump walls that will eventually be covered with water from the break and condensate, the Split heat transfer option is used to switch the heat transfer from the vapor phase to the liquid phase as the liquid level in the containment builds. A quicker transition to liquid heat transfer is more conservative for containment analysis. The Split option is used with α_{lmax}, the maximum liquid fraction, set to

$$\alpha_{l_{max}} = \frac{d}{H}$$

where d is the transition water depth and H is the volume height. A reasonable value for d of 0.1 inch switches the heat transfer from the vapor phase to the liquid phase as the liquid level in the containment reaches 0.1 inch. Other values may be appropriate depending on the geometry of the floor and sump.

For conductors with both sides exposed to the containment atmosphere, the Direct option is applied to both sides. Alternatively, if the conductor is symmetric about the centerplane, a half-thickness conductor can be used with the total surface area of the two sides and an insulated back side heat transfer option. The conductor face that is not exposed to the atmosphere is assumed insulated. The Specified Heat Flux option is used with the nominal heat flux set to zero.

Containment walls above grade and the containment dome have a specified external temperature boundary condition with a heat transfer coefficient of 2.0 Btu/hr-ft^2-°F to model convective heat transfer to the outside atmosphere. The GOTHIC heat transfer solution scheme allows for accurate initialization of the temperature distribution in the containment wall and dome prior to the transient initiation.

A conservative containment liner response is obtained by adding a small conductor that has the same construction and properties as the liner conductor. A conductor surface area of 1 ft^2 is used to minimize impact on the lumped containment pressure and temperature response. The inside heat transfer option is the same as used for the actual liner conductor (Direct with DLM) with a multiplier of 1.2 for conservatism.

Spray Modeling

GOTHIC includes models that calculate the sensible heat transfer between the drops and the vapor and the evaporation or condensation at the drop surface. The efficiency, the actual temperature rise over the difference between the vapor temperature and the drop inlet temperature, cannot be directly specified in GOTHIC. The efficiency is primarily a function of the drop diameter. The GOTHIC models account for the effect of the diameter through the Reynolds number dependent fall velocity and heat transfer coefficients. A heat and mass transfer analogy is used to calculate the effective mass transfer coefficient, which is used to calculate the evaporation or condensation. Containment spray is modeled as described in Reference 51.

Containment Heat Removal

Heat exchangers that remove energy from the containment sump are modeled with the available heat exchanger options in GOTHIC. Use of a GOTHIC heat exchanger option dynamically couples the heat exchanger performance to the predicted primary and secondary fluid conditions. This can provide a small benefit compared to other codes (e.g., LOCTIC) that use bounding UA values to cover the fluid conditions predicted over the entire transient.

The GOTHIC heat exchanger type that closely matches the actual heat exchanger is selected. The inside and outside heat transfer areas are calculated from the heat exchanger geometry details. For tube and shell arrangements, the shell side flow area is set to the open area across the tubes at the mid-plane of the heat exchanger and the shell side hydraulic diameter is set to the tube outer diameter. The GOTHIC option for built-in heat transfer coefficients is used to determine heat transfer coefficients that depend on the primary and secondary side Reynolds and Prandtl numbers. The heat exchanger models in GOTHIC are for basic heat exchanger designs and may not account for the details of a particular heat exchanger (e.g., baffling in a tube-and-shell heat exchanger). A forcing function can be used on the primary and secondary side heat transfer coefficients to tune the heat exchanger performance to manufacturer or measured specifications. Alternatively, the heat transfer area can be adjusted to match the specified performance. Fouling factors and tube plugging are applied when conservative.

Break Release Methodology

The break release methodology in Section 3.5 of Reference 51 is applied. The GOTHIC model assumes a constant drop size of 100 microns for the liquid release from the break until after the LOCA blowdown phase, at which time a continuous liquid is assumed.

Containment Depressurization System

The containment depressurization system consists of the containment QS subsystem and the containment RS subsystem, with casing cooling. The containment depressurization system discharges water into the containment via the quench spray and recirculation spray headers and the casing cooling lines, with the system discharge rate being a function of the appropriate driving forces.

The QS subsystem sprays chilled water from the RWST into the containment via the quench spray headers located approximately 100 feet above the main operating floor. The GOTHIC model includes the QS pump flow as a function of the RWST level and containment pressure. Pump heat is added when conservative. Pipe fill time and pump start delays are incorporated into a delay time that passes before the QS pumps deliver flow to the spray headers. A fraction of each QS pump flow is diverted to the suction of the respective inside recirculation spray (IRS) pump to increase NPSH available (NPSHa).

The RS subsystem rejects heat from the containment through the recirculation spray heat exchangers to the service water reservoir. The RS headers are located approximately 85 feet

above the main operating floor. Constant RS pump flow rates are assumed to bound the minimum and maximum delivered flow rates calculated from system analyses. RS pump heat is added when conservative. The IRS and outside recirculation spray (ORS) pumps are started in accordance with the station setpoints. Delays are incorporated in the pump start times to include fill times for the RS pump discharge piping and time for the pumps to start and reach full flow. NPSHa is calculated at the first-stage impeller of the pump and includes allowances for suction friction and form losses in the flow paths. The casing cooling pumps inject chilled water from the casing cooling tank to the suction of the ORS pumps. The primary function of casing cooling is to provide adequate NPSHa for the ORS pumps. The chilled water added to the ORS pump suction decreases the recirculation spray temperature, which reduces containment depressurization time.

Each of the four recirculation spray lines contains a single-pass, shell-and-tube heat exchanger located inside containment between the RS pump and the spray header. Heat exchanger performance is modeled to ensure a conservative prediction of heat removal from the sump for long-term accident analysis. The RS heat exchanger model selections in GOTHIC were benchmarked to a detailed heat exchanger design code over the range of accident flow rates and temperatures in the RS and SW systems. The heat exchanger models include 2% tube plugging and fouling for analyses where it is conservative.

The average vertical fall height of the water droplets, considering spray header location and probable trajectories, is in excess of 80 feet. Calculations indicate that the small water droplets approach 100% of thermal equilibrium with the containment atmosphere, whereas the large water droplets approach 99% of equilibrium with the containment atmosphere.

Nozzle components are used for each spray line. The Sauter mean diameter was calculated for each spray system in accordance with Section 3.4.1 of Reference 51. For containment integrity analyses, the nozzle spray flow fractions are set to 1.0 and the containment height is reduced using the methodology in Section 3.4.1.2 of Reference 51. The floor area gives the correct drop volume and surface area exposed to the containment atmosphere. NPSHa is not sensitive to a reduction in containment height once the other assumptions that minimize NPSHa are implemented. Therefore, the containment height in the NPSH models is input from the containment free volume and the pool surface area.

6.2.1.1.1.3 *LOCA Mass and Energy Release.* This section presents the LOCA mass and energy releases that were generated in support of the steam generator replacement program.

The containment system receives mass and energy releases following a postulated rupture of the RCS. These releases continue through blowdown and post-blowdown.

The LOCA transient is typically divided into four phases:

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

2. Refill—the period of time when the lower plenum is being filled by accumulator and safety injection water. At the end of blowdown, a large amount of water remains in the cold legs, downcomer, and lower plenum. To conservatively consider the refill period for the purpose of containment mass and energy releases, 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 transient. For the pump suction break, a two-phase mixture exits the core, passes through the hot legs, and is superheated in the steam generators. After the broken loop steam generator cools, the break flow becomes two phase.

LOCA Mass and Energy Release Analysis

The evaluation model for the blowdown, refill and reflood phases for the double-ended pump suction and double-ended hot leg (blowdown phase) mass and energy release calculations was the March 1979 model described in Reference 33. This evaluation model has been reviewed and approved by the NRC, and has been used in the analysis of other dry containment plants.

During a LOCA event, the steam generated by flashing will displace most of the vessel water. The vessel is then refilled by the accumulators and the high and low head safety injection pumps. GOTHIC is not suitable for modeling the refill period because it involves quenching of the fuel rods where film boiling conditions may exist. Current versions of GOTHIC do not have models for quenching and film boiling. Therefore, for the blowdown, refill and reflood stages, the mass and energy release rates are obtained from Westinghouse LOCA analysis. The Westinghouse release data includes the water from the ECCS accumulators, but the nitrogen release to containment is modeled separately in the GOTHIC containment response.

During the post-reflood phase, the GOTHIC RCS system model is used to calculate the mass and energy release to the containment. The model was created using the guidelines in Section 3.5 of Reference 51. The end-of-reflood mass and energy distribution in the primary system and steam generator secondary side is acquired from the Westinghouse mass and energy release analysis. The mass and energy release accounts for the transfer of decay heat and the stored energy in the primary and secondary systems to the containment.

Break Size and Location

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

Three distinct locations in the reactor coolant system loop can be postulated for pipe rupture:

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

The breaks analyzed in support of the steam generator replacement are the double ended hot leg guillotine, DEHLG (9.17 ft^2), and the double ended pump suction guillotine, DEPSG (10.48 ft^2). Break releases have been calculated for the blowdown, reflood, and post-reflood phases of the LOCA for each case analyzed.

Westinghouse Mass and Energy Release Analysis Through the End of Reflood

The double-ended hot leg guillotine has been shown in previous studies to result in the highest blowdown mass and energy release rates. Although the core flooding rate would be highest for this break location, the amount of energy released from the steam generator secondary side is minimal because the majority of the fluid which exits the core bypasses the steam generators by venting directly to containment. As a result, the reflood mass and energy releases are reduced significantly as compared to either the pump suction or cold leg break locations where the core exit mixture must pass through the steam generators before venting through the break. For the hot leg break, 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). The double-ended hot leg reflood phase calculations are not required to determine peak containment pressure, but were calculated for use in the NPSH analysis of the recirculation spray system pumps using the 1975 mass and energy evaluation model described in Reference 36. The post reflood phase calculations are handled with GOTHIC (Reference 51).

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 containment peak pressure for a cold leg break occurs at the end of blowdown. The cold leg break is not usually performed since the hot leg break is expected to result in the highest blowdown peak pressure, and the pump suction break results in the highest post blowdown energy releases into containment.

The pump suction break combines the effects of the relatively high core flooding rate, as in the hot leg break, and the addition of the stored energy in the steam generators. As a result, the pump suction break yields the highest energy flow rates during the post-blowdown period.

Application of Single Failure Criteria

An analysis of the effects of the single failure criteria has been performed on the mass and energy release rates for 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 safety injection system.

Two cases have been analyzed for the effects of a single failure. In the case of minimum safeguards, the single failure postulated to occur is the loss of an emergency diesel generator. This results in the loss of one pumped safety injection train, thereby minimizing the safety injection flow. For the case of maximum safeguards, no failure is postulated to occur. The analysis of both maximum and minimum safeguards cases ensures that the effect of all credible single failures is bounded.

Significant Modeling Assumptions

The following items ensure that the mass and energy releases are conservatively calculated to maximize energy release to containment:

1. Maximum expected operating temperature of the reactor coolant system
2. Allowance in RCS temperature for instrument error and dead band (+4.2°F)
3. Margin in volume of 3% (which is composed of 1.6% allowance for thermal expansion, and 1.4% for uncertainty)

   Note: The evaluation of the reactor vessel head replacement on NSSS Accident Analyses utilized a small portion, less than 0.1% RCS volume, of the 1.4% RCS volume uncertainty.
4. Total core power level of 2956 MWt, which includes an allowance for calorimetric error
5. Net reactor coolant pump heat of 12 MWt
6. Conservative coefficients of heat transfer (i.e., steam generator primary/secondary heat transfer and reactor coolant system metal heat transfer)
7. Allowance in core stored energy for effect of fuel densification.
8. Allowance for RCS pressure uncertainty (+30 psi)
9. 0% steam generator tube plugging level
   - Maximizes reactor coolant volume
   - Maximizes heat transfer area across the SG tubes
   - Lower resistance in loop, therefore increased break flow, lower delta P upstream of break

   With respect to transient behavior of the limiting breaks analyzed, 0% steam generator tube plugging (SGTP) assumptions bound asymmetric SGTP because of the following:

- For the double-ended hot leg guillotine, which is the limiting break location for peak pressure, the amount of energy released from the steam generator secondary side is minimal because the majority of the fluid which exits the core bypasses the steam generator by venting to containment.
- The effects of asymmetric tube plugging on the DEPS case has been assessed and determined to be bounded by the assumption of no tube plugging. This is due to the effects described above as well as the insensitivity of total energy released to tube plugging levels.

10. A constant backpressure equal to the containment design pressure (45.0 psig) has been assumed in the mass and energy release analysis through the end of reflood.

Blowdown Mass and Energy Release Data

The SATAN-VI code was originally used for computing the blowdown transient and is the same as that used for the ECCS calculation in Reference 34. The methodology for the use of this model is described in Reference 33. Updated blowdown transient data for the DEPSG and DEHLG breaks have been performed using the SATAN-78 code version and methodology described in Reference 33. Tables 6.2-13 and 6.2-54 present the calculated mass and energy releases for the blowdown phase of the break analyzed for the double-ended pump suction and double-ended hot leg breaks, respectively.

Reflood Mass and Energy Release Data

The WREFLOOD code used for computing the reflood transient is a modified version of that used in the 1981 ECCS evaluation model (Reference 34). The methodology for the use of this model is described in Reference 33.

A complete thermal equilibrium mixing condition for the steam and emergency core cooling injection water during the reflood phase has been assumed for each loop receiving injection water. This is consistent with the usage and application of the Reference 33 mass and energy release evaluation model, in recent analyses, e.g., the D.C. Cook Nuclear Plant Docket (Reference 38). Even though the Reference 33 model credits steam/mixing only in the intact loop and not in the broken loop, justification, applicability, and NRC approval for using the mixing model in the broken loop has been documented (Reference 38). This assumption is justified and supported by test data, and is summarized as follows:

1. The most applicable steam/water mixing test data has been reviewed for validation of the containment integrity reflood steam/water mixing model. This data is that generated in 1/3 scale tests (Reference 35), which are the largest scale data available and thus most closely simulates the flow regimes and gravitational effects that would occur in a pressurized water reactor (PWR). These tests were designed specifically to study the steam/water interaction for PWR reflood conditions.

2. From the entire series of 1/3 scale tests, a group corresponds almost directly to containment integrity reflood conditions. The injection flow rates for this group cover all phases and mixing conditions calculated during the reflood transient. The data from these tests were reviewed and discussed in detail in Reference 33. For all of these tests, the data clearly indicates 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.
3. Additionally, the following justification is also noted. The limiting break for the containment integrity peak pressure and analysis during the post-blowdown phase is the double ended pump suction 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 steam generator, the other via reverse flow through the reactor coolant pump. Steam which is not condensed by ECC injection in the intact RCS loops passes around the downcomer and through the broken loop cold leg and pump in venting to containment. This steam also encounters ECC 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 ECC injection nozzle. A description of the test and test results is contained in References 33 and 35.
4. The methodology previously discussed and described in Reference 33 has been utilized and approved on the Dockets for Catawba Units 1 and 2, McGuire Units 1 and 2, Sequoyah Units 1 and 2, Watts Bar Units 1 and 2, Millstone Unit 3, and Beaver Valley Unit 2.

Table 6.2-14 presents the calculated mass and energy release for the reflood phase of the double-ended pump suction break, with minimum safety injection. A significantly higher discharge occurs during the period the accumulators are injecting (from 23.36 to 47.42 seconds) as illustrated in Table 6.2-14.

The transient of the principal parameters during reflood are given in Table 6.2-13.

GOTHIC Post-Reflood Mass and Energy Release Data

The GOTHIC code in Reference 51 is used for computing the post-reflood transient. The LOCA mass and energy release rates are input to GOTHIC for the blowdown and reflood periods of the design basis LOCAs, but GOTHIC calculates the transfer of decay heat and the stored energy in the primary and secondary systems to the containment. The calculation of these release rates is described in Section 6.2.1.1.1.3. The mass and energy release rates used in the containment peak pressure, containment depressurization, and NPSH analyses for the RS and LHSI pumps are provided in this section. The mass and energy release rates for the DEHLG through the end of reflood are tabulated in Table 6.2-54 for maximum two-train safety injection flow. The mass and energy release rates for the reactor coolant DEPSG through the end of reflood are provided in Tables 6.2-13 and 6.2-14 for maximum single-train safety injection flow.

At the end of reflood, the core has been recovered with water and the ECCS continues to supply water to the vessel. Residual stored energy and decay heat comes from the fuel rods. Stored energy in the vessel and primary system metal will also be gradually released to the injection water and released to the containment via steaming through the core or spillage into the containment sump. In addition, there may be some buoyancy-driven circulation through the intact steam generator loops that will remove stored energy from the steam generator metal and the water on the secondary side. Depending on the location of the break, the two-phase mixture in the vessel may pass through the steam generator on the broken loop and acquire heat from the stored energy in the secondary system. For these conditions, GOTHIC is capable of calculating the mass and energy release from the break into containment.

The GOTHIC long-term mass and energy release accounts for the transfer of the decay heat and the stored energy in the primary and secondary systems to the containment after the end of reflood. The energy for each source term is acquired at the end of reflood from the Westinghouse mass and energy release analysis. The rate of energy release is determined by a simplified GOTHIC RCS model that is coupled to the containment volume. Thus, the flow from the vessel to the containment is dependent on the GOTHIC-calculated containment pressure.

The 1979 ANS Standard-5.1 decay heat model is used in the calculation of mass and energy releases to the containment following a loss-of-coolant accident. Therefore, to more realistically model the RCS, the decay heat model in Reference 37 is utilized in the GOTHIC containment analysis.

Significant assumptions in the generation of the decay heat values:

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 Equation 11 of Reference 37 up to 10,000 seconds, and Table 10 of Reference 37 beyond 10,000 seconds.
5. The fuel has been assumed to be at full power for 10^8 seconds.
6. The number of atoms of U-239 produced per second has been assumed to be equal to 70% of the fission rate.
7. The total recoverable energy associated with one fission has been assumed to be 200 MeV/fission.
8. Two sigma uncertainty (two times the standard deviation) has been applied to the fission product decay.

Lumped volumes are used for the vessel, downcomer, cold legs, steam generator secondary side, up-flow steam generator tubes and down-flow steam generator tubes. Separate sets of loop and secondary system volumes are used for the intact and broken loops with the connections between the broken loop and containment as necessary for the modeled break location. The Westinghouse calculated mass and energy inventory at the end of reflood establishes the liquid volume fractions and the fluid temperatures in the primary and secondary systems.

The primary and secondary system geometries, including primary system resistances, are consistent with the models used for non-LOCA accident analyses. In order to predict the natural circulation through the intact loops and the correct water level in the vessel and downcomer, the volumes are modeled with the correct elevations and heights. The vessel height may be adjusted so that the water and steam inventory at the end of reflood matches the vendor's boundary conditions, but this correction does not affect the hydraulic analysis.

Safety injection fluid is added to the downcomer volume (for the intact cold legs) and the broken loop cold leg. In both locations, the SI fluid mixes with the resident fluid and any vapor from the intact SGs. The SI flow is taken from the RWST until a low-low level is reached, at which time the SI fluid is taken from the containment basement via the strainer header.

A thermal conductor is used to model the transfer of energy stored in the shell side of the steam generator to the SG secondary fluid. The initial temperature is set to match the available stored energy specified at the end of reflood by the fuel vendor analysis. The up flow and down flow tubes on the steam generators are modeled separately with thermal conductors. This allows for the possibility of boiling in the up flow tubes and superheating of the steam in the down flow tubes. The heat transfer from the secondary side to the primary side is modeled using conductors with the inside connected to the primary system tube volumes. The Film heat transfer option is used on both sides of the tube. This option automatically accounts for heat transfer to the liquid or vapor phase as appropriate and includes boiling heat transfer modes.

Sources of Mass and Energy

The sources of mass considered in the LOCA mass and energy release analysis are the reactor coolant system, accumulators, and pumped safety injection.

The energy inventories considered in the LOCA mass and energy release analysis are:

1. Reactor Coolant System Water
2. Accumulator Water
3. Pumped Injection Water
4. Decay Heat
5. Core Stored Energy
6. Reactor Coolant System Metal

7. Steam Generator Metal
8. Steam Generator Secondary Energy
9. Secondary Transfer of Energy (feedwater into and steam out of the steam generator secondary)

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

The consideration of the various energy sources in the 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 mass and energy inventories are presented at the following times, as appropriate:

1. Time zero (initial conditions)
2. End of blowdown time
3. End of refill time
4. End of reflood time
5. Time of full depressurizations
6. End of analysis

6.2.1.1.2 Subcompartments

6.2.1.1.2.1 *Design Criteria.* The containment structure subcompartment walls are designed for the maximum differential pressure developed across the walls should there be a break of a high-energy pipe inside the compartment.

The computer programs THREED and RELAP are used for determining the design pressures for the interior compartments (or rooms) such as the reactor cavity and steam generator and pressurizer cubicles. In order to calculate the pressure transient within the compartment, THREED and RELAP numerically solve equations defining heat and mass flows into and out of the interior of the compartment. The programs are a mathematical description of the compartment and calculate the pressure effects of reactor coolant discharging into the compartment and the heat and mass flows from the compartment to the main volume of the containment atmosphere. The mass and energy flow rates from a DER of the primary coolant pipe into the compartment are obtained from SATAN V results.

Tables 6.2-5 through 6.2-9 present the limiting breaks and the mass and energy release rates used in the reactor cavity, pressurizer cubicle, and steam generator compartment analysis.

6.2.1.1.2.2 *Description of the THREED Code.* The THREED computer code is used to calculate pressure and temperature transients in various nuclear power plant subcompartments following a

postulated high-energy pipe break. THREED allows the user to subdivide all cubicles into nodes in order to take into account all known major flow obstructions. The compartments studied are typically the reactor cavity, the steam generator cubicles, and the pressurizer cubicle.

THREED has the capability of modeling up to 100 nodes (volumes). Each node can vent to as many as seven of the other nodes. Blowdown can be put into any combination of nodes. In addition, a revised vent flow model is used.

Assumptions

The derivation of the analytical model for THREED is based on the laws of conservation of mass, momentum, and energy, the equations of state for air, steam, and water, and the principles of two-phase flow. In order to approximate such a problem numerically the following simplifying assumptions are made:

1. *Adiabatic Process*—The system is defined as the compartment atmosphere at any given time. This includes any air, steam, and water droplets present, but not the walls, equipment, or internal structure of the compartment itself. Heat sinks that may exist within the compartment are not included.

2. *Quasi-Steady State*—Mass and energy flows are calculated on the basis of the node thermodynamic state, as determined at the end of the previous time interval. The thermodynamic state is determined from mass and energy flows during the time interval based on flow rates evaluated at the beginning of the interval.

3. *Complete Mixing in the Node*—The atmosphere in the node mixes instantaneously and homogeneously. At each point in time the atmosphere is in a state of thermodynamic equilibrium.

4. *Independent Inflow*—The mass and energy inflows are independent of the compartment back pressure. This assumption allows the mass and energy inflows to be specified as input to the program, and it is accurate for those time periods when the flow through the pipe break is sonic.

   For subsonic break flow, it is conservative with respect to pressure buildup in the compartment. The inflow may be divided among as many nodes as is appropriate to the configuration under investigation.

Computational Method

THREED numerically solves finite difference equations that account for mass and energy flows into and out of a node. The computation approach used in THREED is summarized in the conceptual flowchart shown on Figure 6.2-2.

Equations

1. *Mass and Energy Releases-Rates from a High-Energy Pipe Break*—The rates of mass and energy release from a high-energy pipe break are supplied as input to THREED. These blowdown rates are obtained from the nuclear steam supply system (NSSS) vendor's SATAN V computer program. Since these rates are calculated neglecting the effect of compartment back pressure, they are conservatively high.

2. *Calculation of the Thermodynamic State of a Node*—At the end of each time interval of the numerical calculation, the stagnation temperature and pressure in the node are determined based on new inventories of mass and internal energy.

3. *Internodal Flow Rates*—The THREED computer code includes three two-phase vent flow options. The user specifies the correct option for each vent. The first two options consider the vent flows to be homogeneous. The correct option is dictated by the vent geometry. For vents with a contraction at the inlet, an isentropic entrance effect is included in the homogeneous vent flow model Number 1 (HXFM-1). When there is no contraction the isentropic entrance effect is not appropriate and the correct option is homogeneous vent flow model Number 2 (HVFM-2).

   The third flow model is the frictionless Moody flow model (Reference 6). This model is used for design of compartments that do not contain high-energy lines, but that are adjacent to compartments that do. In this case, flow to the compartment under consideration from source nodes is considered to be frictionless Moody flow with a multiplier (discharge coefficient) of 1.

   The approach used in the derivation of the homogeneous vent flow models (HVFM) is a modification and extension of that presented in Reference 7.

The major assumptions used in deriving the models are as follows:

1. The flow is quasi-steady state.
2. The flow is one-dimensional.
3. The flow is homogeneous (no slip between phases).
4. The flow is adiabatic (no heat transfer between the vent and the fluid or between the fluid phases).
5. The quality is constant over the length of the vent.
6. The air-steam mixture is considered as a perfect gas.
7. The sonic velocity of the mixture is equal to the sonic velocity of the gaseous phase only. The basis for this assumption is experimental data that show that this is true for the high void fractions (> 0.95) that are expected to occur in subcompartment analyses (References 8, 9, & 10).

8. Pressure changes within the vent due to gravity effects are negligible.
9. The flow is accelerated isentropically from the source node to the vent inlet (HVFM-1 only).
10. The flow rate becomes critical when the exit Mach number of the gaseous phase becomes unity.
11. The vent pressure loss coefficient (K) is constant during the time interval.
12. The vent flow area is constant during the time interval.

Solving the equations of state, continuity, energy, and momentum consistent with the above assumptions leads to the following equations:

(6.2-4)

$$K = \frac{x}{\gamma}\left(\frac{M_2^2 - M_1^2}{M_1^2 M_2^2}\right) + \frac{\gamma+1}{2\gamma}\ ln\left[\frac{\left(1 + \frac{-1}{2X}M_2^2\right)M_1^2}{\left(1 + \frac{\gamma-1}{2X}M_1^2\right)M_2^2}\right]$$

$$\left.\begin{aligned}
\frac{P_{01}}{P_{02}} &= \frac{1}{\left(1 + \frac{\gamma-1}{2X}M_1^2\right)^{\frac{\gamma}{\gamma-1}}}\left[\frac{\left(2 + \frac{\gamma-1}{X}M_1^2\right)M_1^2}{\left(2 + \frac{\gamma-1}{X}M_2^2\right)M_2^2}\right]^{\frac{1}{2}} \quad (HVFM-1) \\
\frac{P_{02}}{P_{01}} &= \left[\frac{\left(2 + \frac{\gamma-1}{X}M_1^2\right)M_1^2}{\left(2 + \frac{\gamma-1}{X}M_2^2\right)M_2^2}\right]^{\frac{1}{2}} \quad (HVFM-2)
\end{aligned}\right\}$$

(6.2-5)

where:

K = vent pressure loss coefficient

X = quality of the mixture (mass ratio of air and steam to air, steam, and liquid)

γ = air-steam specific heat ratio

M_1 = Mach number of the gaseous phase at the inlet to the vent

M_2 = Mach number of the gaseous phase at the exit of the vent

P_{01} = stagnation pressure in the source node (psia)

P_{02} = stagnation pressure in the sink node (psia)

The K-factor used in THREED includes the contraction, bend, and friction losses that are encountered in flow through the vent. The vent exit loss is implicit in the scheme of THREED. In the calculation of the thermodynamic state, the flow velocity within each node is assumed to be zero. Thus, one full velocity head is lost at the exit of the vent. This is equivalent to a K-factor of 1 at the exit of the vent.

This system of nonlinear equations is solved for M_1 and M_2 using a Newton-Raphson iteration technique. If $M_2 \geq 1$, the flow is assumed to be choked. When $M_2 > 1$, Equation 6.2-1 is solved for M_1 with M_2 set equal to 1.

Once M_1 has been determined, the mass flow rate per unit area is determined by the following equation for both subcritical ($M_2 < 1$) and critical ($M_2 = 1$) flow:

$$\left.\begin{aligned} G &= \frac{12\,M_1}{X\left(1 + \frac{\gamma - 1}{2x} M_1^2\right)^{\frac{1}{\gamma - 1}}}\left[\frac{\gamma \rho_{01} P_{01} g_c}{1 + \frac{\gamma - 1}{2x} M_1^2}\right]^{\frac{1}{2}} \quad \underline{(HVFM-1)} \\ G &= \frac{12\,M_1}{X}\left[\gamma \rho_{01} P_{01} g_c\right]^{\frac{1}{2}} \quad \overline{(HVFM-2)} \end{aligned}\right\}$$

(6.2-6)

where:

G = mass flow rate per unit area (lbm/ft^2-sec)

P_{01} = density of vapor in the source node (lbm/ft^3)

g = 32.1739 lbm-ft/lb-sec^2

No discharge coefficient (less than unity) is applied to the HVFM critical flow rate.

Isentropic Exponent γ

The liquid effects are considered in the momentum equation for calculating acceleration and local irreversible losses (Reference 7). The specific heat ratio is based on an air-steam mixture and flashing is not allowed in the vents, since the resident time in the vent is usually too short to allow flashing. Thus, the composition in each phase remains constant, but the vapor phase increases in volume. The value of γ ranges from 1.1 to 1.4.

Conservatisms

The major conservatisms in the THREED computer program are as follows:

1. Two-phase homogeneous flow between nodes.
2. Complete mixing in each node:
   a. 100% (liquid) carryover fraction.
   b. Instantaneous flashing of liquid to steam.
3. Fluid brought to rest in each node.
4. Adiabatic system.

Justification of THREED

1. *Analytical*—As can be seen in Table 6.2-10, the THREED solutions to the NRC's Standard Subcompartment Problems agree quite favorably with results of the NRC's version of RELAP-3 and the COMPARE program. Exceptions are problems 4, 5, and 6, in which the effect of inertia is evidently significant (see inertial effect discussion below).
2. *Experimental*—Comparisons of THREED results with those obtained in the Swedish Marviken test number 14 are shown on Figures 6.2-3 through 6.2-5. These comparisons indicate that the peak pressure differential between compartments calculated with THREED is from 1.18 to 2.18 times the experimentally determined value. Thus, THREED results are conservative with respect to experiments.

Inertial Effect

The RELAP4 computer code (Reference 11) is used to investigate the effect of inertia on vent flow. Each subcompartment is analyzed with both the THREED and RELAP4 codes. The larger calculated differential pressure is selected for the design basis.

For the reactor cavity, mass and energy release rates are computed for a limited displacement rupture of the cold leg. The break area assumed for this calculation is 150 in^2.

For the steam generator subcompartment analysis, WCAP-8132A justifies break locations and sizes for both guillotine breaks and longitudinal splits of the primary reactor coolant loop piping for North Anna Units 1 and 2. The break size justification and subsequent subcompartment pressurization analysis utilized three primary reactor coolant loop pipe whip restraints (two beneath the steam generator and one beneath the reactor coolant pump). These pipe whip restraints restricted pipe displacement thereby limiting the break flow area of any guillotine break to a size significantly less than the longitudinal split break. With these pipe restraints in place, the limiting break area assumed is a 660 in^2 longitudinal split of the reactor coolant loop hot-leg pipe.

License Amendment Nos. 107 and 93 for North Anna Units 1 and 2, respectively, allows for the removal of these pipe whip restraints based on the amendment dated October 27, 1987 to

General Design Criteria 4 in Appendix A of 10 CFR 50 and the NRC approved LBB analysis for North Anna Units 1 and 2 provided in WCAPs 11163/11164. Accordingly, reactor coolant loop ruptures no longer need to be considered in the subcompartment pressurization analysis, that is, only reactor coolant loop branch line ruptures require evaluation. The largest possible branch line rupture corresponds to the 14" o.d. pressurized surge line which has a maximum break area of 98.3 in^2. Therefore, the original analysis based on a break area of 660 in^2 is conservative and bounds any pipe rupture in the steam generator subcompartment even with the pipe whip restraints removed.

Calculation of the total jet force on subcompartment walls from a postulated rupture is based on Moody's theoretical model (References 12, 13 & 14) and Fauske's experimental data (Reference 15). It is assumed that the retarding action of the surrounding air on the jet is negligible and the total jet force is constant at all axial locations. The jet impingement pressure on a distant object is computed by assuming that the jet stream expands conically at a solid angle of 20 degrees.

For normal impingement, the jet impingement force on a distant object is equal to the product of the jet impingement pressure and the intercepted jet area. If the object intercepts the jet stream with a curved or inclined surface area, then the drag force between the jet and the object is taken as the jet impingement force.

6.2.1.2 **System Design**

The containment structure consists of a steel-lined, reinforced concrete structure designed to withstand an internal pressure at 45 psig and temperature of 280°F. The free volume within the containment structure is given in Table 6.2-2. For further details of the containment structure design, see Section 3.8.2.

The containment structure size is primarily based on a study of equipment placement criteria, shielding, and expected maintenance procedures. This study yields minimum containment structure size.

The design basis accident leakage rate is 0.1 volume percent of the containment atmosphere per day at the design pressure within the containment structure of 45 psig.

All structures, systems, and components within the containment are designated Seismic Category I except those listed here. A failure of the non Seismic Category I items will not damage Seismic Category I structures, systems, or components.

6.2.1.2.1 Neutron Detector Carriage

This equipment, which is a hoist used to raise and lower the excore detector, is stored in the lower reactor cavity between the neutron shield skirt and the primary shield wall. This hoist and pully are secured to the wall to prevent excessive movement during a seismic event. The storage

location is not near the incore instrumentation tubes. A failure of this component will not damage Seismic Category I structures, systems, or components and is not required to be checked.

6.2.1.2.2 Reactor Upper Internals Storage Stand

The upper internals storage stand is permanently mounted on the floor of the refueling cavity as shown in Reference Drawing 8. Although not designated as Seismic Category I, this equipment is designed to retain its structural integrity under a seismic event or when loaded with the upper reactor internals.

6.2.1.2.3 Reactor Lower Internals Storage Stand

The reactor lower internals storage stand is permanently mounted on the refueling cavity floor. The location of the stand is shown in Reference Drawing 10. The stand consists of four individual support struts, each of which is constructed of three columns with appropriate mounting plates and brackets. This stand serves no function during normal plant operation and is used only when the lower internals are removed from the reactor vessel. Because of the relatively low mass of the support struts, a failure due to a seismic event, during normal plant operation, would not damage any Seismic Category I equipment. A failure while loaded with the lower internals could result in damage to the reactor vessel. However, since the reactor core has been removed at this time, no safety-related consequences would occur.

6.2.1.2.4 Reactor Vessel Head Storage Stand

The reactor vessel head storage stand is a portable cylindrical structure approximately 12 feet in diameter and 4 feet high. Two containment areas are designated for reactor vessel head storage. One, at the lowest containment elevation, is shown in Reference Drawing 8 and the other, at the operating level, is shown in Reference Drawing 10. The reactor head storage stand, although not designated as Seismic Category I, is seismically designed to maintain its structural integrity when loaded with the reactor head. The stand is restrained to prevent movement during a seismic event.

6.2.1.2.5 Neutron Shield Tank Cooling Water Subsystem

The neutron shield tank cooling water subsystem is described in detail in Section 9.2.2. All components that constitute this system are located in one area within the containment, shown on Reference Drawings 5 and 8. This system is not required for reactor shutdown. The neutron shield surge tank and associated supports will maintain structural integrity during a design-basis earthquake, and therefore will not affect the integrity of safety-related components.

6.2.1.2.6 Personnel Staging Basket

The personnel staging basket is used to gain access to the quench and recirculation spray heads within the containment. During normal plant operation the basket is removed from containment or stored on the polar crane, shown on Reference Drawing 9. The staging basket is secured or removed from containment to prevent movement during a seismic event.

6.2.1.2.7 Sump Pumps

The containment is provided with sump pumps in the incore instrumentation tunnel and immediately outside the primary shield wall as shown in Reference Drawing 7. The pumps are small in size and located on the lowest containment elevation at Elevation 216 ft. 11 in. Because of the low elevation, remoteness from Seismic Class I components, structures, and systems, the seismic failure of the sump pumps will not cause damage to Seismic Class I components.

6.2.1.2.8 Primary Vent Pot

The primary vent pot is a component of the vent and drain system described in Section 9.3.3. The location of this component is shown in Reference Drawing 7 and 10. This component is not above or near Seismic Class I equipment or systems. The seismic failure of the vent pot will not damage such equipment.

6.2.1.2.9 Iodine Filtration Fans

Location of the iodine filtration fans is shown in Reference Drawing 7 and 8. The fans are described in Section 9.4.9.2. Seismic Class I equipment near the fans are the primary drain transfer tank and cooler, the air recirculation cooling coils, a safety injection accumulator, and the recirculation air coolers. The fans are located at least 10 feet from this equipment. Due to this distance and the relatively low mass of the fans it is unlikely that the fans upon failure could travel this distance and damage the surrounding Seismic Class I equipment during a seismic event.

6.2.1.2.10 Containment Instrument Air Compressors and Receivers

6.2.1.2.10.1 *Containment Instrument Air Compressors.* These items are located in the lowest containment floor elevation, Elevation 216 ft. 11 in., and are shown in Reference Drawing 7. These items would not obtain significant amounts of energy from a seismic failure to cause damage to the Seismic Class I components.

6.2.1.2.10.2 *Containment Instrument Air Receivers.* Units 1 and 2 containment instrument air receivers are located in the lowest containment floor elevation, Elevation 216 ft. 11 in. The support legs and anchor bolts for these tanks have been designed to withstand the effects of a seismic event thereby eliminating the potential of these tanks to damage any nearby safety related equipment.

6.2.1.2.11 Radiation Monitoring Equipment

Area radiation monitors are provided in three locations within the containment. One monitor is mounted on a manipulator crane, and one in the personnel hatch area, and one in the incore instrumentation cubicle. This equipment is insignificant in size, and consequently failure due to a seismic event would not affect the integrity of Seismic Class I structures, systems, or components.

6.2.1.2.12 Supplementary Neutron Shielding

Supplementary neutron shielding is located in the reactor cavity between the reactor vessel and the neutron shield tank, inside periphery. Shielding is mounted to the neutron shield tank. The supports are designed considering seismic loads but are not considered Seismic Class I. Consequently, the shielding is adequately restrained to prevent damage to the Seismic Class I equipment.

6.2.1.2.13 Containment Annulus Hoist

The containment annulus hoist is mounted on a monorail located at Elevation 323 ft. 4 in., as shown in Reference Drawing 8. The hoist is used to maneuver equipment during shutdown. Seismic failure of the hoist is not expected; however, as a precaution against damage to other equipment during normal plant operation, the hoist is positioned above the concrete slab at the equipment hatch.

6.2.1.2.14 Pressurizer Relief Tank

The pressurizer relief tank is permanently mounted on the floor of the pressurizer cubicle as shown in Reference Drawing 6. The tank is designed with rupture disks to prevent failure should the tank become overpressurized. The tank is located such that the rupture disks are directed away from any Seismic Class I Equipment. Sufficient energy will not be developed to damage Seismic Class I structures.

6.2.1.2.15 Reactor Vessel Head Shielding

The reactor vessel head shielding reduces exposure during certain refueling operations. The reactor vessel head shielding is mounted on the reactor vessel head and is supported by the intermediate lift ring. The intermediate lift ring attaches to the three reactor vessel head lifting lugs. The head shielding is seismically designed in order to prevent possible damage to safety-related equipment in the area due to a seismic event.

6.2.1.2.16 Miscellaneous Equipment

Items such as ladders, doors, monorails, and containment elevator with enclosures are supported by Seismic Class I structures and will experience the same seismic motion as the structure. These items are low mass and consequently will not impose a sufficient force to be broken free from the supporting structure itself.

6.2.1.2.17 Steel Tool Box (Unit 1 only)

The steel tool box shown in Reference Drawing 5 may be accessed during subatmospheric entries to retrieve various tools or parts utilized during such entries. Although the tool box is not designated Seismic Category I, it is seismically restrained to prevent movement during a seismic event.

6.2.1.2.18 Storage Boxes for Lead Blanket Shielding

Steel storage boxes, containing lead blanket shielding, are located inside of Units 1 & 2 containments as shown in Reference Drawings 4, 5 and 6. Although the storage boxes are not designated Seismic Category I and are not anchored down, they are located sufficiently far away from neighboring plant equipment to prevent impact during a seismic event. As such, these storage boxes are considered “seismically restrained.”

6.2.1.2.19 Storage Boxes for Scaffolding

Steel storage boxes and drums, containing scaffolding components, are located inside of Units 1 & 2 containment annulus. Although the storage boxes are not designated Seismic Category I and are not anchored down, they are located sufficiently far away from neighboring plant equipment to prevent impact during a seismic event. Storage drums holding clamps, coupling pins, etc. are chained to the side of the storage boxes to prevent overturning. As such, these storage boxes are considered “seismically restrained.”

6.2.1.2.20 Manbasket

A Non-safety Related manbasket may be located on the Unit 1 & 2 Reactor Containment Operating deck (291 ft level) during power operations (unless it is stored in alternate areas outside containment). The manbasket is used for personnel access to the reactor head seismic restraints (prior to and after refueling) and other components that are accessible via the polar crane auxiliary hook. During power operations, Seismic Housekeeping administrative controls will be used to select the manbasket Operating deck storage location and determine appropriate seismic restraints/clearances.

6.2.1.3 **Design Evaluation**

6.2.1.3.1 Containment

The reactor containment is maintained at a subatmospheric pressure during reactor operation, during which time the air partial pressure is maintained as a function of service water temperature within the operating curve. The allowable variation is based upon long-term effects on the cooldown capability of the ESF from external conditions such as seasonal temperature changes in the service water. The containment will not rise above the design pressure of 45 psig following a LOCA or MSLB inside containment. The containment pressure is less than 2.0 psig during the period from 1 to 6 hours and subatmospheric within 6 hours from the occurrence of a LOCA, thus terminating any outleakage from the containment.

The subatmospheric containment feature limits that outleakage of fission products and satisfies 10 CFR 50.67 criteria for a LOCA.

At a containment leak rate of 0.1 volume percent of the containment atmosphere per day, air inleakage is not significant for a considerable length of time after a LOCA. Ultimately, several weeks to several months later, air inleakage could result in a containment pressure slightly above

atmospheric. To prevent this, the containment atmosphere cleanup system (Section 6.2.5) maintains the containment pressure at several inches of mercury below the lowest expected atmospheric pressure. When the hydrogen recombiner blowers are started after the accident and are operated in the containment purge mode, they discharge through the gaseous waste disposal system. Vacuum cannot be lost rapidly because of the inherent low-leakage design features of the containment structure.

Penetrations through the containment, including piping and electrical penetrations and access hatches, have been designed so that double barriers or seals exist between the containment atmosphere and the outside environment. Hence, there are no direct leakage paths between the containment and outside environment.

6.2.1.3.1.1 *Loss-of-Coolant Accident.* This section describes the loss-of-coolant accident (LOCA) containment transient analyses that are performed to confirm that the containment peak pressure is less than the design limit of 45 psig. The LOCA containment depressurization analyses are described in Section 6.2.2.6.3. Containment response analyses are performed using the GOTHIC computer code and the methodology described in Reference 51. Refer to Section 6.2.1.1.1.2 for the GOTHIC methodology description. Key input parameters for the analysis are shown in Table 6.2-2.

The peak containment pressure occurs during the blowdown phase and is a function of the initial total pressure and average temperature of the containment atmosphere, the containment free volume, the passive heat sinks in the containment, and the rates of mass and energy released to the containment. The passive heat sinks in the containment are assumed to be at the same initial temperature as the initial average containment atmosphere temperature. Maximizing the initial containment total pressure and average atmospheric temperature maximizes the calculated peak pressure. The peak containment pressure is independent of single failure and service water temperature since the peak occurs before containment depressurization systems start. The magnitude of the containment peak pressure is governed by the heat transfer to the containment passive heat sinks. The LOCA peak pressure analyses assume maximum initial containment pressure, maximum air temperature, 100% relative humidity, minimum containment free volume, and minimum heat sink surface area.

The double-ended hot leg guillotine (DEHLG) break causes a more limiting blowdown peak pressure than the double-ended pump suction break (DEPSG). The results from the limiting DEHLG break are presented in Table 6.2-11. Figures 6.2-60 and 6.2-61 illustrate the GOTHIC containment pressure and vapor temperature response. The containment peak pressure of 57.4 psia is less than the design limit of 59.7 psia. The LOCA peak containment temperature is obtained from the peak pressure case because the containment atmosphere is saturated. The containment vapor and liner temperatures remain below 280ºF.

During operation, the containment air partial pressure is varied to maintain the capability to depressurize to less than 2.0 psig during the period 1-6 hours and to subatmospheric within

6 hours after a LOCA. As discussed in Section 6.2.1.1, this capability is a function of the service water temperature. If air inleakage should occur, the containment vacuum system (Section 6.2.6) is used to maintain the containment atmosphere at the specified operating air partial pressure (see the Technical Specifications). The air removed from the containment structure is metered and provides a constant indication of the containment system integrity.

After the first 6 hours of a LOCA, the containment will remain at a subatmospheric pressure for at least 30 days following an accident with an inleakage allowance of 0.1 volume percent of the containment atmosphere per day. Depressurization by the vacuum relief system (containment vacuum system) would not be necessary.

6.2.1.3.1.2 *Main Steam Line Break (MSLB) Inside Containment.* The MSLB analysis is performed using the GOTHIC computer code to determine the containment pressure and temperature response. Key input parameters for this analysis are provided in Table 6.2-2.

6.2.1.3.1.2.1 *MSLB Mass and Energy Release.* The key reactor system variables are initial power level, RCS pressure, RCS temperature, and RCS loop flow. For this analysis, a maximum core power of 2956 MWt (100.54% of 2940 MWt) was used. The thermal design flow was assumed along with the nominal RCS pressure of 2250 psia. The RCS average temperature was assumed to be 4°F above the nominal value to account for measurement and control system uncertainties. Tables 6.2-17 and 6.2-18 summarize the RCS and secondary system initial conditions.

The core kinetic parameters were chosen to simulate end-of-cycle conditions with the most reactive rod stuck out of the core. These assumptions maximize the positive reactivity insertion due to moderator feedback during cooldown. Additionally, minimum safety injection was assumed to restrict the flow of borated water to a rate corresponding to the operation of one charging pump. The safety injection lines downstream of the boron injection tank are assumed to have a zero boron concentration. These assumptions minimize the magnitude of the negative reactivity inserted.

The assumptions regarding the secondary system are intended to produce conservative results. The main feedwater system is designed to maintain feedwater flow equal to steam flow. Therefore, the feedwater flow rate increases following a steam line break resulting in several effects. First, the steam pressure is lower due to the presence of subcooled water in the generator. Second, the heat transfer from primary to secondary is increased. Finally, for large break cases the increased feedwater flow increases the amount of entrained water in the steam exiting the break. Since these are competing conditions it is not possible to define the worst feedwater transient for all plant conditions. Therefore in order to insure conservative results each of the above parameters were defined at its least positive or most negative extreme.

The feedwater flow rate was conservatively modeled by assuming an increase in response to the steam line break. For split breaks and small double-ended ruptures feedwater flow was increased proportionally to the steam line flow increase. For the large double ended rupture cases

the feedwater flow was instantaneously ramped to a maximum of 220% of nominal full feedwater flow in response to the decreasing steam generator pressure.

There are many other less significant secondary parameters which were modeled conservatively including: initial steam generator fluid mass, critical flow model loss coefficient and steam line blowdown volume. These parameters are specified in Tables 6.2-17 and 6.2-18. The interface between the primary and the secondary system was also modeled conservatively. Reverse heat transfer from the intact steam generators to the primary loop was modeled which resulted in the release of more energy to the containment.

An auxiliary feedwater initiation 0 seconds after the accident with a rate of 900 gpm to the broken loop steam generator was also assumed in the analysis. The impact of increasing this auxiliary feedwater flow rate to 970 gpm was subsequently evaluated in Reference 41. That evaluation confirmed that this increase in flow would not impact the calculated containment peak pressure and temperature.

Various system component failures were evaluated to determine which failure results in the largest increase in releases to the containment. The failure of one safeguards train to operate was assumed in the analysis along with the failure of the non-return valve in the steam line with the faulted steam generator. The safeguards train failure reduces boron delivery to the core while the non-return valve failure allows the steam generators to blowdown until the main steam isolation valves on the intact loops are isolated. Since the main steam trip valves at the North Anna Power Station do not prevent reverse flow, the nonisolatable volume in the main steam system continues to blowdown even after steam line isolation occurs.

The steam line break analysis evaluates four break areas at each of four different power levels. The break areas are determined as follows:

1. A full DER downstream of the main steam flow restrictor equal to 1.4 ft^2.
2. A small DER having an area just larger than that for which moisture entrainment occurs.
3. A small DER having an area just smaller than that for which moisture entrainment occurs.
4. A small split rupture that will neither generate a steam line isolation signal from the Westinghouse Solid State Protection System nor result in water entrainment.

The four power levels used for development of the mass and energy release rates are 0, 30, 70, and 102% of 2898 MWt core power. These analyses are bounding for the measurement uncertainty recapture power uprate to 2940 MWt.

Mass and energy release rates were generated for each case. Generally, the transients are characterized by rapid increases in mass flow rate and energy flow rate lasting a few seconds and begin to exponentially decrease. The mass flow rate is largest for the 1.4 ft^2 DER breaks and the hot zero power cases. The energy release rate is largest for the 1.4 ft^2 break and for the 102% of 2898 MWt core power cases. The actual data are presented in WCAP-11431 (Reference 44).

6.2.1.3.1.2.2 *MSLB Containment Response.* The GOTHIC MSLB analyses do not credit RS system operation. With the assumption of an emergency bus failure, one QS pump is the only means of reducing containment pressure until the AFW flow to the faulted SG is isolated at 30 minutes. The long-term containment pressure and temperature are dependent on the boiloff rate from the maximum AFW flow rate and the capacity of the operating QS pump. The quench spray is assumed to reach the containment atmosphere 70 seconds after the containment depressurization actuation (CDA) setpoint is reached.

MSLB Peak Pressure Analysis

The maximum peak pressure of 57.65 psia occurs with an initial 1.4 ft^2 DER at 30% of 2898 MWt core power, shortly after AFW is terminated when the remaining SG liquid mass has been boiled to the containment. Figure 6.2-7 shows the containment pressure. The high AFW flows combined with the high initial mass in the SGs at low power result in the limiting case. Table 6.2-16 presents a summary of the peak containment pressure and temperature results. The containment peak pressure is less than the design limit of 59.7 psia.

The initial conditions for the peak containment pressure calculation assume a saturated atmosphere with an air partial pressure which results in a worst case peak pressure of approximately 57.65 psia. The maximum operating temperature is utilized since the containment heat sinks play a dominant role in the magnitude of the peak pressure and are initialized at the operating temperature at the start of the analysis.

MSLB Peak Temperature Analysis

The peak containment temperature of 308.4°F was achieved using minimum air partial pressure, maximum containment air temperature, and 0% humidity for the MSLB analyses. The maximum peak temperature occurs for the 0.6 ft^2 break at 102% of 2898 MWt core power very early in the transient. Figure 6.2-6 shows the containment vapor temperature response for the limiting case.

While the peak temperature exceeds the containment design temperature of 280°F, Dominion has examined the effect of this on nonelectrical equipment inside the containment structure itself. On the basis of this examination, it has been concluded that because of the large heat capacity of the structures and the relatively short period of time the containment atmosphere is above the containment design temperature, there would be no adverse effect on nonelectrical equipment inside the containment or on the containment structure. Appendix 3F discusses the effects of this transient on safety-related electrical equipment and concludes that the maximum calculated surface temperature is less than the qualification temperature.

6.2.1.3.2 Subcompartments

A pressure response analysis is performed on three subcompartments: the reactor cavity, the pressurizer cubicle, and a steam generator cubicle. Subcompartments downstream of the cubicle are also analyzed. The initial conditions used for subcompartment analysis are: pressure of

8.6 psia, relative humidity of 0%, and temperature of 120°F. A sensitivity study of these quantities is presented below.

The effect of initial pressure on the calculated peak differential pressure across the steam generator cubicle walls following a hot-leg DER is shown in Table 6.2-19. The use of low initial pressure is conservative.

The temperature and relative humidity over the range of permissible operating conditions have negligible effect on the peak differential pressure, as shown in Tables 6.2-20 and 6.2-21.

The subcompartment pressure response analyses, presented in Section 6.2.1, made use of the following initial conditions:

1. Initial pressure - 8.6 psia.
2. Initial relative humidity - 0%.
3. Initial temperature - 120°F.

The results presented in Tables 6.2-19 through 6.2-21 are for a typical plant, although the conclusions apply to North Anna.

6.2.1.3.2.1 *Reactor Cavity.* The reactor cavity has the following principal features:

1. The inside diameter of a hot-leg penetration through the reactor cavity wall is 50 inches.
2. The vent area of the annulus between a hot-leg pipe and the reactor cavity penetration at both the inner and outer face of the reactor cavity wall is 3.69 ft^2.
3. The location of the vents from the reactor cavity and the configuration of the reactor cavity can be found using Reference Drawing 4 through 10.
4. The free volume of the upper reactor cavity is 1537.7 ft^3. The free volume of the lower reactor cavity is 7526 ft^3. A restriction of 64.32 ft^2 in the lower reactor cavity separates the instrumentation tunnel from the remainder of the lower reactor cavity. The free volume of the annulus between the reactor pressure vessel and the neutron shield tank, bounded by the upper and lower reactor cavity, is 209 ft^3. The vent area between the upper reactor cavity and this annulus is 9.5 ft^2, while the vent area between the lower reactor cavity and this annulus is 11.24 ft^2.
5. A tabulation of all vents from the reactor cavity showing the vent area and the compartment to which the vent is discharged is shown in Table 6.2-22.

The following vent areas in the containment are covered by floor grating or similar obstructions:

1. At the 291 ft. 10 in. elevation, shown on Reference Drawing 4, the vent areas above the steam generator cubicle, pressurizer cubicle, and incore instrumentation cubicle are covered with grating.

2. At the 262 ft. 10 in. elevation, shown in Reference Drawing 5, the vent areas above the pressurizer relief tank cubicle are covered with grating, and there are also vent areas in the steam generator cubicles, pressurizer cubicle, and incore instrumentation cubicle, which are obstructed by swingout doors.
3. At the 241 ft. 0 in. elevation, shown in Reference Drawing 6, the vent areas from the steam generator cubicles and the pressurizer relief tank cubicle are obstructed by swingout doors. There are also vent areas from the steam generator cubicles to the lower level that are obstructed by grating, as shown in Reference Drawings 9 and 10.

Containment loss-of-coolant accident analyses, which include the effects of flow through vents with grating, as described in 1, 2, and 3 above, consider only the reduced effective flow areas after subtracting the grating area.

The doors, as described in 2 and 3 above, are locked during normal operation and flow calculations did not take credit for the door opening vent area.

6.2.1.3.2.2 *Upper Reactor Cavity.* Five nodal configurations of the upper reactor cavity (URC) are considered. These nodal models contain 1, 6, 12, 18, and 36 nodes in the URC. The reactor annulus (RA), lower reactor cavity (LRC), incore instrumentation tunnel (IIT), and containment are represented as additional nodes. A 150-in^2 cold-leg limited displacement rupture (LDR) is postulated.

The one-node model considers the entire URC as a single node. The six-node model considers the URC to be circumferentially divided with nodal boundaries formed by a vertical plane through the center line of each nozzle. The 12-node model is an extension of the six-node model, with the URC axially divided by a horizontal plane through the nozzle center lines. The 18-node model is an extension of the 12-node model, with the URC axially divided again by a horizontal plane through the bottom of the reactor vessel flange (57-3/16 inches above the nozzle center lines). The 36-node model is an extension of the 18-node model, with the URC circumferentially divided again by vertical planes bisecting the angles between each pair of adjacent nozzles.

The peak pressure differential between the URC and the containment is plotted as a function of the number of nodes in the URC on Figure 6.2-8. No significant increase in peak differential pressure results from dividing the URC beyond 12 nodes. Thus, the 12-node model is the design-basis model for the URC. Including downstream nodes, this model contains a total of 21 nodes.

The nodalization study is performed using the THREED computer code. The pressure transient is also calculated using the RELAP4 computer code. For the 12-node URC model, the THREED computer code calculates a higher peak differential pressure than the RELAP4 code.

A schematic drawing showing the nodalization of the reactor cavity and indicating the nodal net free volumes and interconnecting flow paths is given on Figure 6.2-9. Flow path vent areas are given in Table 6.2-23.

Plan and section drawings showing the general arrangement of the reactor cavity structures and piping and indicating the nodal configuration of the 12-node URC model are presented on Figures 6.2-10 and 6.2-11, respectively. The IIT (node 20) extends beyond what is shown on the figures and eventually vents to the containment (node 21).

The break area used in this analysis is a 150-in^2 LDR at the reactor pressure vessel (RPV) inlet nozzle safe end.

The vent loss coefficients used to calculate flow between nodal volumes are presented in Table 6.2-23. The vent loss coefficients are obtained from References 18 and 19.

Table 6.2-6 presents the mass and energy blowdown rates used in this analysis.

Figure 6.2-12 presents the transients of pressure differential across the shield wall for each node in the 12-node URC model.

6.2.1.3.2.3 *Subcompartments Located Downstream of the Upper Reactor Cavity.* Three subcompartments located downstream of the URC are considered: the RA, the LRC, and the IIT. Nodalization studies performed show that the pressure transient in each subcompartment is not very sensitive to the number of nodes. Nodalizing the RA axially and circumferentially does not significantly increase the pressure transient calculated with the entire RA modeled as a single node. Thus, the one-node RA model is the design-basis model. Nodalizing the LRC into two nodes at the catwalk at Elevation 223 ft. 4 in. results in a slightly higher pressure transient than that calculated treating the LRC as a single node. The two-node LRC model is the design-basis model. Nodalizing the IIT into three nodes (the walkway around the outside of the vessel support skirt, the IIT access hatch, and the IIT itself) results in a slightly higher pressure than that calculated treating the IIT as a single node. The three-node IIT model is the design-basis model.

For all three subcompartments downstream of the URC, the THREED program yielded higher differential pressures than RELAP4. Therefore, the THREED results are the design-basis values.

Schematic drawings showing the nodal configuration of the RA, LRC, and IIT and indicating nodal net free volumes and interconnecting flow areas are presented on Figures 6.2-13, 6.2-14, and 6.2-15, respectively. The flow area of 10.91 ft^2, between the containment node and the incore instrument tunnel nodes, has conservatively excluded the flow area of the 12-inch diameter Incore Sump Room (ISR) drain at centerline elevation 219'-6".

A section view of the RA is shown on Figure 6.2-11. Plan and section drawings showing the general arrangement of the LRC and IIT structures and indicating the nodal configuration are presented on Figures 6.2-16, 6.2-17, 6.2-18, and 6.2-19, respectively. On Figures 6.2-17

and 6.2-19 certain details not actually shown in the plan view at 228 ft. 6.25 in. are presented for clarity.

The vent loss coefficients used to calculate flow between nodal volumes for the one-node RA model, the two-node LRC model, and the three-node IIT model are presented in Tables 6.2-24, 6.2-25, and 6.2-26, respectively.

The IIT blowout panels are assumed to completely block flow out to the containment while initially in place. They are assumed to open when the differential pressure has increased to 0.25 psid. No other removable obstructions to vent flow are present in the IIT.

Table 6.2-6 gives the mass and energy blowdown rates used in these analyses. Since the peak differential pressure for the IIT occurs at 1.3 seconds after the accident, the mass and energy blowdown rates at 1 second are held constant out to 2 seconds for the IIT analysis.

Figure 6.2-20 presents the differential pressure between the RA and the containment versus time after the accident. Figure 6.2-21 presents the differential pressure between the LRC and the walkway around the vessel support skirt, i.e., across the reactor vessel support skirt versus time after the accident. Figure 6.2-22 presents the differential pressure between the IIT and the containment versus time after the accident.

The calculated differential pressures presented in this response have been used to verify the structural integrity of the support skirt, concrete walls, and floors. In addition, the structural integrity of the LRC has been verified for the NRC-calculated peak differential pressure of 6.6 psid. The verification analysis used equations and methods previously outlined in Chapter 3.

The assumptions used to maximize the pressure calculated for downstream nodes are consistent with maximizing the mass and energy flows into the downstream compartments and minimizing the mass and energy flows out of these compartments. The assumptions are:

1. Use of frictionless Moody flow with a multiplier of 1.0. This maximizes the flow rate to the downstream node.
2. In modeling inlet vent areas of such items as grating and snubbers, simplifying approximations are made in a manner that maximizes them.
3. In cases where the downstream node in which pressure is to be maximized is not adjacent to the break node, several nodes that are intermediate between that node and the break node may be combined to maximize the entrance pressure to the vent to the downstream node.

The assumptions used to maximize the pressure calculated in the break node are:

1. Vent flow from the break node is minimized by:
    a. In THREED computer code, a homogeneous vent flow model (HVFM-1 or HVFM-2) is used.

b. In the RELAP4 computer code, the minimum of a frictionless Moody flow model with a multiplier of 0.6 and an inertial flow model is selected.

2. In modeling outlet vent areas of such items as gratings and snubbers, simplifying assumptions are made in a manner that minimizes them.

Assumptions regarding the doors and blowout panels in the steam generator compartments, pressurizer cubicle, and the incore instrumentation tunnel are as follows:

1. The blowout panels in the subcompartments have vent areas for which an obstruction must be displaced to provide area for pressure relief. The blow-out panels are passive, not active, devices. The vent areas are assumed to have zero value until a specified pressure is reached. At this pressure the blowout panel is displaced and the vent areas and resistance coefficients assume values that remain constant with time.

2. The doors in the steam generator and pressurizer cubicles consist of blowout sheet metal panels fastened by sheet-metal screws to the frame of a locked wire grid door. A differential pressure of 5 psi creates sufficient force to ensure panel blowout. After a differential pressure of 5 psi is reached, the sheet-metal panel is blown out, and the locked wire grid door is conservatively assumed to remain in place. A sheet-metal panel is considered to blow out only when it is located on the far side of the wire grid door from the break node.

Expansion and contraction losses associated with flow through wire grid door gratings and screens are calculated as follows:

1. Determine the resistance coefficient (K-factor for contraction and expansion losses combined) by means of Reference 20, based on the velocity upstream of the grating.

2. Multiply the resistance coefficient calculated above by the square of the open area fraction so that it corresponds to the velocity through the grating.

6.2.1.3.2.4 *Grating.* Typical grating used in the cubicles in the reactor containment is as shown on Figure 6.2-23.

The analyses were performed with the assumption that 25% of the grate area was covered. This reflects the presence of structural members behind the grating. The open area fraction was thus conservatively assumed to be 75%.

The case of a thickened grid (perforated plate or laths) is selected from Reference 20 (Section VIII, Diagram 8-4), to determine the resistance coefficient associated with flow through gratings (see Figure 6.2-24).

The resistance coefficient for a grate used in the analyses for Section 6.2.1.3.2 was 0.33.

6.2.1.3.2.5 *Screens.* The resistance coefficient of the screen doors in the steam generator and pressurizer subcompartments is calculated using the same procedure and similar nomenclature as for the grating. The case of a screen is selected from Reference 20 (Section VIII, Diagram 8-6).

For each wire grid door considered, a 70% free open vent area (0.70×30.64 ft^2 = 21.4 ft^2) is considered, and the resistance coefficient is conservatively calculated as 0.33. The contraction coefficient for each blowout panel is 0.50. Therefore, the contraction plus friction coefficient is 0.83.

Two blowout panels are considered as vent areas in the pressurizer cubicle analysis; more than two different blowout panels are considered as vent areas in each steam generator cubicle.

Figures 6.2-25 and 6.2-26 amplify Reference Drawing 9 and illustrate two views of the area surrounding the air duct and blowout diaphragm under consideration. No credit is taken for the displacement of the air duct.

Table 6.2-27 presents the vent type, the method of calculation of the vent area, the vent area, and the K-factors as input into the THREED and RELAP codes. For conservatism, in the 14 node steam generator subcompartment analysis, credit is not taken for four small vent openings per steam generator cubicle (4.2 ft^2 per opening); two vent openings are on the air duct and two in the floor at the 242 ft. 6 in. elevation.

There are six blowout panels in the incore instrumentation tunnel access hatch. The two end blowout panel openings (37.25 inches x 21.75 inches) and four side blowout panels (39.25 inches x 21.75 inches) are closed to seal the access hatch during normal operation. The blowout panel doors are displaced open by a pressure differential of 0.25 psid. Since the blowout panel openings are an unsecured access point to a very high radiation area, a jail bar type barrier exists inside the opening to prevent unauthorized personnel access into the incore instrumentation tunnel. The six blowout panel openings provide a total effective vent area in excess of the minimum required vent area necessary of 28 ft^2. This includes the reduction in area due to the jail bar type barrier in each opening.

The function of the reactor flange-mounted ventilation seal, comprised of 48 panels, is to seal off the annular space between the reactor vessel and the shield wall in order to ensure that sufficient air is pumped through the nozzle penetrations to maintain concrete temperatures within acceptable limits during normal operation. After a LOCA, the reactor flange-mounted ventilation seal panels burst and thereby prevent pressure buildup in the reactor cavity from becoming excessive. If each ventilation panel is treated as a membrane in tension with a one-dimensional stress distribution, only hoop tension need be considered, and the following equation applies: pressure equals hoop tension divided by membrane radius of curvature. The reactor flange-mounted ventilation panels are perforated in such a way as to ensure they burst at a hoop tension of 400 lb/in. The reactor-flange-mounted ventilation seal panels are designed to fail at 18 psid; the total vent area thereby provided is approximately 147 ft^2.

There are no inservice inspection procedures needed to ensure that the blowout panels in the steam generator and in the pressurizer cubicles will open as required. The blowout panels in the incore instrumentation tunnel will be inservice inspected by pulling the doors open at regular

inservice inspection time intervals. The reactor flange-mounted ventilation seal panels will be visually inspected for cracks at regular inservice inspection time intervals.

The reactor cavity ventilation seal is designed to blow out at 18 psig. The analysis for subcompartment pressure assumes a value of 20 psig for blowout. Dynamic analysis of accelerations shows that the panels will have moved far enough in 0.005 seconds to provide full venting area. Peak pressures in the reactor cavity have been checked with the assumption that blowout panels remain in place up to 40 psid. These studies show that the peak pressure is insensitive to the pressure at which the panels blow out. Varying the blowout pressure from 10 psid to 40 psid results in only 1 psi difference in the reactor cavity peak pressure. If the panels rupture at 20 psid, the opening will be completely operative before 40 psid would have been reached. In addition, the opening will be partially venting from the instant the panels rupture.

Blowout panel strength was checked on the actual material of manufacturer prior to fabrication. Tests were tensile tests of partial sections of the panels pulled in a tensile tester to ensure that the actual breaking point corresponds to that computed by the calculation. Adjustments were made in the design to achieve proper breaking point prior to the start of fabrication.

The blowout panels in the steam generator and pressurizer cubicles are designed to blow out at 2.5 psid. The subcompartment analysis assumes them to be clear by 5 psid. Dynamic analysis shows that the opening will be completely venting in less than 0.018 seconds and would be completely out of the way before the 5 psid used in the analysis could have been reached. These openings also will be venting before full opening is achieved. The pressure rise rate in these cubicles is much slower than that in the reactor cavity.

With the exception of the incore tunnel blowout panels, all blowout panels are made of thin-gauge steel sheet metal. The incore panels are heavier but are equipped with hinges to retain them. The light sheet metal panels are not substantial enough to damage equipment, piping, or conduits, nor are they located in areas containing safety equipment. The only safety-related equipment likely to be hit are the cables to the control rod drive mechanisms (CRDMs), which deactivate in the event of an accident.

The structural capacities of the various subcompartments have been verified using the following calculated pressures:

Lower pressurizer cubicle - 12.2 psid

Upper pressurizer cubicle - 1.0 psid

Steam generator cubicle

Concrete shield wall at 291 ft. 10 in. - 12.7 psid

Subcompartment walls - 28.9 psid

Differential pressure across steam generator - 17.7 psid

Differential pressure across steam generator and reactor coolant pump supports - 8.3 psid

Vertical lift on steam generator - 33.9 psid

Upper reactor cavity - 104 psid

Reactor annulus - 42.9 psid

Lower reactor cavity - 4.4 psid

Incore instrumentation tunnel - 9.1 psid

Structural capacity check was accomplished in accordance with the design formulas in Section 3.8.2.2.

Blowout panels in the cubicles, with the exception of the reactor cavity blowout panels, will blow into areas where there is floor grating for floors. The grating openings are small enough to prevent the panels from reaching the containment sump. The blowout panels are designed to blow out in one piece. The gratings have sufficient excess area that the blowout panels could not cause blockage.

The blowout panels in the reactor cavity could potentially block the drain from the refueling canal since it is only a 6-inch opening. To prevent blowout panels from blocking the opening, a raised steel dome with holes was installed over the refueling canal drain. The remainder of the refueling cavity has numerous drains through the reactor cavity penetrations and incore tunnel openings to drain the water even if some of the openings were blocked.

Most of the insulation on the primary coolant lines and the reactor vessels is reflective-type metal insulation that would remain in large enough pieces that it will not find its way to the sump.

6.2.1.3.2.6 *Steam Generator Cubicle.* The steam generator subcompartment has been analyzed using both the THREED and RELAP4 computer codes for a hot-leg split equivalent in area to the cross section of the 29-inch hot-leg pipe. This is a conservative analysis since only reactor coolant loop branch lines need to be considered in the subcompartment pressurization analysis. The mass and energy release rates are tabulated in Table 6.2-9. A comparison of volumes and vent areas for the three steam generator subcompartments (A, B, and C) indicates that it is conservative to analyze subcompartment A since it has the least free volume and vent area. No credit is taken for failure of the air duct that surrounds the reactor cavity beneath the steam generator subcompartments. Permanent scaffold installed on the blowout panels (elevation approximately 244') in each of the three steam generator subcompartments were (1) analyzed to survive a 25 psi LOCA-generated pressure loading and were (2) configured to avoid reducing the subcompartment ventilation area.

Differential Pressure Across the Concrete Shield Walls (Around Steam Generator Above Operating Floor)

The nodalization study of the volume surrounded by the concrete shield walls considers one- and two-node models. The one-node model considers this volume as a single node. This two-node model considers the volume to be vertically divided by a horizontal boundary at Elevation 299 ft. 10 in., where there is a flare in the steam generator.

The peak differential pressure across the concrete shield walls is determined by:

1. Treating the steam generator subcompartment below the operating floor as a single node.
2. Maximizing the flow rate from the steam generator subcompartment to the volume surrounded by the concrete shield walls above the operating floor.
3. Minimizing the obstruction by the snubbers of the vent area leading to the volume surrounded by the concrete shield walls above the operating floor.

The use of the one-node and two-node models shows no significant difference in the peak differential pressure either across the steam generator subcompartment wall or across the concrete shield walls. Based on this observation, the removable block wall volume is treated as one node in the analysis of the steam generator subcompartment, which is analyzed below.

A block diagram showing the two-node model (five nodes total) is presented on Figure 6.2-27. Vent areas, pressure loss coefficients, and flow models for the two-node model are presented in Table 6.2-28. For this model the RELAP4 analysis yields higher differential pressures than the THREED analysis. Table 6.2-29 provides the length-to-area ratios used in RELAP4. The peak calculated differential pressure across the concrete shield walls is 12.7 psid. The design differential pressure is 14 psid. Figure 6.2-28 presents the transient of the pressure differential across the concrete shield walls for the two-node model. The removable block walls previously used for shielding on the operating floor level were replaced by reinforced concrete walls during construction.

Nodalization Study for the Steam Generator Subcompartment

The nodalization study of the steam generator subcompartment consists of vertical and horizontal nodes and is based on calculating the peak differential pressures across the steam generator subcompartment walls. In addition to the nodes within the steam generator subcompartment, three other nodes are considered. The volume bounded by the concrete shield walls between Elevation 291 ft. 10 in. and 301 ft. 10 in. is considered as one node; the volume above the air duct (Elevation 236 ft. 8 in.) and below the grating (approximate Elevation 243 ft. 0 in.) is treated as another node; and the third additional node is the remainder of the containment.

In the vertical nodalization study the models contain one, two, four, and five nodes within the steam generator subcompartment. In all of these models the total blowdown enters a break node at Elevation 256 ft. The one-node model considers the entire steam generator compartment

as a single node. The two-node model considers the steam generator subcompartment to be vertically divided by a horizontal plane at Elevation 259 ft. 0 in., where the refueling cavity wall projects inward into the steam generator subcompartment. The four-node model is an extension of the two-node model with horizontal planes at Elevation 262 ft. 6 in. (grating) and Elevation 271 ft. 6 in. (platforms). The five-node model is an extension of the four-node model with an imaginary horizontal boundary dividing the volume of the break node in half and with all the blowdown entering the upper half.

For the vertical nodalization study, the peak differential pressure between the steam generator subcompartment and the containment is plotted as a function of the number of nodes in the steam generator subcompartment (Figure 6.2-29). No significant increase in peak differential pressure results from dividing the steam generator subcompartment beyond four nodes.

Using the four-node (vertical) model as a base, two horizontal nodalization studies are considered as follows:

1. The steam generator subcompartment between Elevation 243 ft. 0 in. and 259 ft. 0 in. is divided into four nodes (three additional nodes) by three vertical planes, as follows: (a) a plane through the axis of the hot leg, (b) a plane through the axis of the cold leg, and (c) a plane through the axis of the cross-over pipe between the steam generator and the reactor coolant pump (see Figure 6.2-30). This yields a model with seven nodes in the steam generator subcompartment. The break nodes are one and two, and each receives 50% of the blowdown.
2. Nodes one, two, and three of part (a) (Figure 6.2-30) are subdivided by projecting a vertical plane from the outer surface of the refueling cavity wall from Elevation 259 ft. 0 in. downward to Elevation 243 ft. 0 in. The resulting additional nodes five, six, and seven are shown on Figure 6.2-31, and the edge of the vertical plane is shown on Figure 6.2-32. This model, therefore, has 10 nodes in the steam generator subcompartment.

For the horizontal nodalization study, the peak differential pressure across the subcompartment walls is plotted as a function of the number of nodes in the steam generator subcompartment in Figure 6.2-33. As can be seen in the figure, the seven-node model (10 nodes total) gives the greatest peak differential pressure and is, therefore, the design-basis case for the steam generator subcompartment walls. Figure 6.2-34 presents transients of pressure differential across the steam generator subcompartment walls for the seven-node case. The THREED code results in higher differential pressures across the steam generator subcompartment walls than the RELAP4 code. Vent areas, pressure loss coefficients, and flow models for the seven-node model (10 nodes total) are presented in Table 6.2-30. A block diagram showing the seven-node model is presented on Figure 6.2-35.

Asymmetric Pressure Results for Components and Supports

For the purpose of obtaining asymmetric pressure loads on the steam generator, three horizontal nodes between Elevation 259 ft. 0 in. and 262 ft. 6 in. are added to the seven-node model described above. The resulting model, therefore, has 10 nodes in the steam generator subcompartment (13 nodes total). A block diagram showing the nodal configuration of the model is presented on Figure 6.2-36.

Nodes five through eight above Elevation 259 ft. 0 in. (Figure 6.2-37) are formed by extending upward the same vertical planes that divide nodes one through four (Figure 6.2-30) below Elevation 259 ft. 0 in. No change in pressurization of node one results from the addition of the three horizontal nodes between Elevation 259 ft. 0 in. and 262 ft. 6 in.

Vent areas, pressure loss coefficients, and flow models for the 10-node model (13 nodes total) are presented in Table 6.2-31. Plan and section drawings showing the general arrangement of the steam generator subcompartment structures, piping, and nodal arrangement are shown in Figures 6.2-37, 6.2-38, and 6.2-39. Figure 6.2-38 is the same as Figure 5.5-9, but in addition includes elevation designations.

The 10-node model yields graphs of differential pressure across the supports and the components versus time after the accident, as shown on Figures 6.2-40 and 6.2-41, respectively. The peak calculated differential pressures and the computer code yielding the higher calculated differential pressures are presented in Table 6.2-32. Table 6.2-33 provides the length-to-area ratios used in RELAP4.

Vertical Pressurization of the Steam Generator (Uplift)

For the case of the vertical pressurization of the steam generator (steam generator uplift), it is conservatively considered that the volume within the steam generator lower support frame receives all the blowdown.

A node representing the volume within the support frame is added to the four-node (seven nodes total) and seven-node (10 nodes total) steam generator subcompartment models described above. The former model is the more limiting (yields higher uplift pressure) and is shown schematically on Figure 6.2-42. Node eight represents the volume within the support frame.

Table 6.2-34 presents the summary of vent areas, pressure loss coefficients, and vent flow models for the steam generator five-node model (8 nodes total). Figure 6.2-43 presents the transient of the differential pressure between nodes seven and eight, which acts upward across the steam generator. The THREED code results in higher differential pressures than the RELAP4 code for this case. The peak calculated differential pressure for the vertical pressurization across the steam generator is 33.9 psid. The vertical pressure across the reactor coolant pump is considered negligible due to the open area around the pump support columns.

These calculated pressures have been combined with other associated loads to verify the integrity of the components' supports and structures.

6.2.1.3.2.7 *Pressurizer Compartment.* The pressurizer subcompartment pressure response has been analyzed using both the THREED and RELAP4 computer codes for both spray-line DER and a surge-line DER. The THREED analysis yields higher differential pressure than the RELAP4 analysis. The mass and energy release rates for these breaks are tabulated in Tables 6.2-7 and 6.2-8.

The nodal configuration used for this analysis is shown on Figure 6.2-44. The corresponding vent flow parameters are tabulated in Table 6.2-35. The peak differential pressure in the lower pressurizer subcompartment (node two[1]) calculated by THREED is 12.2 psid for a surge-line DER postulated in that node. The peak calculated differential pressure in the pressurizer relief tank subcompartment (node one) is slightly less than 12.2 psid. The differential pressure transient for node two as determined with THREED is shown on Figure 6.2-45.

The peak differential pressure calculated in the upper pressurizer cubicle is that determined for node three. The upper pressurizer cubicle is modeled as nodes three, four, and five. Node three yields the maximum differential pressure because it has the smallest volume, least vent area, and is adjacent to node two, which contains the surge line. A spray-line DER in node three generates a peak differential pressure of 0.4 psid in node three.

A surge-line DER in node two yields a peak differential pressure in node three of 1.0 psid. The differential pressure transients determined with THREED for node three are shown in Figure 6.2-46. Table 6.2-36 provides the length-to-area ratios used in RELAP4 for the pressurizer cubicle analysis.

The design differential pressure equals 10 psid for nodes one and two, and 2.25 psid for nodes three, four, and five. The structural integrity of the lower pressurizer subcompartment has been verified for the new calculated pressure.

6.2.1.4 **Testing and Inspection**

The following types of tests will be performed on the containment:

1. Structural Acceptance Test: To verify the structural adequacy of the containment (described in Section 3.8).
2. Containment Leakage Test: To verify that the leakage of the containment is within allowable limits.

6.2.1.4.1 Containment Leakage Tests

A performance-based testing program is conducted to test containment leakage periodically through the operating life of the unit.

1. Node two is the only node of the pressurizer subcompartment through which the surge line passes.

The performance-based testing program will include Type A tests to measure the containment overall integrated leakage rate, Type B tests to detect and measure local leakage from certain containment components, and Type C tests to measure containment isolation valve leakage rates.

The containment leakage tests are performed as required by 10 CFR 50, Appendix J, *Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors*, Option B, as modified by approved exemptions, and in accordance with the guidelines contained in Regulatory Guide 1.163, *Performance-Based Containment Leak-Test Program*, dated September 1995.

6.2.1.4.2 Type A Tests

A performance-based test program for Type A testing is conducted in accordance with Appendix J, following preoperational Type B and C tests, except as noted. A preoperational test was performed at a pressure of ≥ 40.6 psig. Leakage characteristics yielded by this test were used to establish the preoperational measured containment leakage rate, L_{am}. Periodic Type A tests performed at or greater than P_a during unit shutdown will be conducted as required by Appendix J of 10 CFR 50, except as modified by NRC-approved exemptions and in accordance with the guidelines contained in Regulatory Guide 1.163, dated September 1995. Most equipment within the containment is designed to withstand periodic testing at P_a without affecting operational capabilities or operating life. Other equipment will be vented, vented and drained, drained or removed as required prior to the testing program. Tests will meet the acceptance criteria described in Appendix J. Type A tests will be performed with the containment leakage monitoring system (Section 6.2.7) or acceptable temporary test equipment, using the test methods described in ANSI/ANS 56.8-1994, *Containment System Leakage Testing Requirements.*

Sensitivities of the instrumentation used in Type A testing and overall Type A test accuracy are presented in Section 6.2.7.2.

6.2.1.4.3 Type B Tests

A performance-based Type B test program is conducted to detect and measure local leakage as required by 10 CFR 50, Appendix J, Option B, as modified by approved exemptions, and in accordance with the guidelines contained in Regulatory Guide 1.163, dated September 1995. Components and equipment subject to Type B testing will include:

1. Containment penetrations, the design of which incorporates resilient seals, gaskets, or sealant compounds.
2. Personnel access locks and equipment hatch doors with resilient seals or gaskets.

Components subject to Type B testing are equipped with test connections to allow pressurization with air or nitrogen. Several test methods are used to locate leaks. Soap bubble testing at a pressure $\geq P_a$ will provide a sensitive and rapid method for qualitative determination of leakage over large areas. Each penetration is configured to permit required testing. A bubble test

rig or other acceptable flow measuring device may be employed to check component leakage. Basically, the testing consists of a nitrogen or air source piped to the component test connection, through a “bubbler” containing glycerin or some other flow measuring device. Pressure P_a is applied to the test connection and penetration components.

The acceptance criteria of the Technical Specifications specify that the combined leakage rate of all components subject to Type B and C testing must be less than 60% of the design basis accident (DBA) leakage rate (L_a). Quantitative leakage rate measurements will be made by pressurizing the component to be tested with air or nitrogen to pressure P_a and measuring the amount of gas required to maintain that pressure.

Performance-based Type B testing, including containment air locks, is conducted as required by 10 CFR 50, Appendix J, as modified by approved exemptions, and in accordance with the guidelines contained in Regulatory Guide 1.163, dated September 1995.

6.2.1.4.4 Type C Tests

A performance-based Type C test is conducted to verify containment isolation valve leaktightness. Valves that are subject to Type C testing include the following:

1. Containment isolation valves that provide a direct connection between the atmosphere inside and outside the reactor containment during normal operation.
2. Valves that are required to close automatically upon receipt of the containment isolation signal.
3. Valves required to operate intermittently under post-accident conditions.

In addition, components other than those listed above that may develop leaks in service will be tested with Type B test methods and repaired so as to meet the acceptance criteria of Type A testing.

Containment isolation valves subject to Type C testing will be tested at a pressure $\geq P_a$. The containment side of each valve will be pressurized with air or nitrogen. A flow meter will measure any leakage through the valve. Redundant valves, evaluated to not be affected by test pressure direction, may be tested simultaneously by applying the pressure through the pipe volume between the isolation valves. A list of the containment isolation valves is provided in the Technical Requirements Manual. The associated containment penetrations and testing methods to be used are listed in Table 6.2-37.

Certain containment isolation valves receive an “open” signal on CDA signal, while others close or remain isolated, locking water within the penetration. Additionally, the isolated penetrations remain pressurized by the supply source. The associated penetrations are hence maintained in a “water-filled” configuration during accident conditions and are not considered potential containment atmosphere leakage paths. The specific valves for which this applies are identified in the list of containment isolation valves in the Technical Requirements Manual.

In accordance with the performance-based test program, Type C tests will be scheduled during each refueling shutdown. The combined leakage rate for all components subject to Type B and C tests will not exceed 60% of L_a.

6.2.1.4.5 Additional Tests

Prior to entering MODE 4 from MODE 5 after containment vacuum has been broken, containment purge valves with resilient seals will be leak rate tested in accordance with Technical Specifications. The pressure test consists of gas pressurization between the isolation valves. These tests are to identify excessive degradation of the valve seats. They are in addition to the Type C testing requirements for these valves.

6.2.1.4.6 Testing Results

A post outage report is prepared to summarize the results of the previous cycle's Type B and Type C tests and Type A, Type B, and Type C tests, if performed during the outage.

If the results of any Type A, B, or C test do not meet the applicable acceptance criteria of the Technical Specifications, those results are evaluated for reportability under 10 CFR 50.72 and 10 CFR 50.73.

6.2.1.5 **Instrumentation Application**

Redundant containment pressure transmitters indicate and continuously measure containment pressure within a range of minus 5 psig to 3 times the design pressure of the containment (180 psia), as required by NUREG-0578. One of the two wide-range transmitter loops is recorded in the control room, and the other is monitored by the SPDS computer. Loop power supplies are fed from diverse vital buses.

Descriptions of the instrumentation provided to monitor the integrity of the containment systems are included in the following sections:

Title	Section
Containment isolation system	6.2.4
Containment leakage monitoring system	6.2.7

6.2.2 Containment Heat Removal Systems - Containment Depressurization System

6.2.2.1 **Design Bases**

The containment depressurization system is used to return the containment atmosphere to subatmospheric pressure after a LOCA by removing heat from the containment structure. The containment depressurization system consists of two subsystems: (1) the quench spray (QS) subsystem and (2) the recirculation spray (RS) subsystem. The QS subsystem transfers heat from the containment atmosphere to the quench spray, which is collected in the containment sump. The RS subsystem transfers heat, via the recirculation spray coolers, from the water collected on the containment structure floor and from the containment atmosphere to the Service Water system.

These subsystems are the only engineered safety features (ESF) that are used after a LOCA or main steam line break (MSLB) inside containment to remove heat from the containment structure. The MSLB containment response analyses do not credit operation of the RS system.

Other containment heat removal systems, such as the containment air recirculation system and control rod drive mechanism (CRDM) cooling system, are used during normal unit operation and are not required but may be used after Condition II and Condition III accidents (see Sections 15.2 and 15.3) to remove heat from the containment structure. Since these systems are not used after Condition IV accidents, they are not considered ESF.

Other coolant leakage in excess of the capacity of the air recirculation coolers and static heat sinks to absorb and remove heat will result in actuation of the ESF. For leakage from a high-energy pipe, such as in the reactor coolant system (RCS), the size of the leak that would increase the containment pressure to the high-pressure setpoints is relatively small compared to the size of a complete rupture of a pipe. A 10-gpm leak of reactor coolant will not cause actuation of the ESF.

The containment total pressure cannot exceed the containment high-pressure setpoint (17 psia) without actuating ESF (Phase A). The containment high-pressure signal activates a partial containment isolation (see Figure 7.3-3), actuation of the safety injection system (SIS) if it is not the initiating signal, and reactor scram. If the operator takes no further action, sprays will be initiated at 27.75 psia, effectively terminating any pressure transient. The containment integrity analysis assumes a CDA setpoint of 30 psia for quench spray actuation. For a LOCA, there is no potential for exceeding the design temperature and pressure of the containment, 280°F and 45 psig, unless no sprays are actuated. The 280°F design temperature is exceeded for a short period during an MSLB inside containment.

All valve operators, instrumentation and control penetrations, and other equipment that are essential to ESF operation are designed to withstand the containment design temperature of 280°F.

The containment depressurization system has the following design bases:

1. Cool and depressurize the containment atmosphere to less than 2.0 psig in 1 hour and to subatmospheric pressure in less than 6 hours following a LOCA.
2. Reduce the concentration of radioactive iodine in the containment atmosphere quickly so that for any outleakage during the time the containment is above 1-atm. pressure, the resulting dose is within the limits specified in 10 CFR 50.67.
3. Provide the ECCS with water for effective core cooling on a long-term basis after a LOCA.

The sources and quantities of energy that would be removed from the containment structure to fulfill these design objectives are given in Section 6.2.1.3, which discusses the amount of heat removal from the primary system components.

As discussed in Section 6.2.1, the containment structure is designed for the maximum amount of energy that would be transferred from the RCS after a LOCA or MSLB. During the period prior to quench spray injection into the containment atmosphere, energy is absorbed by the static heat sinks, i.e., the concrete and steel within the containment, thereby attenuating the effect on the containment structure.

The RS subsystem is capable of maintaining the subatmospheric pressure in the containment following a LOCA. On a long-term basis, the containment atmosphere cleanup system (Section 6.2.5) is used to remove any subsequent air leakage into the containment structure.

6.2.2.2 **System Design**

The containment depressurization system consists of two separate but parallel QS subsystems, each of 100% capacity, and four separate but parallel RS subsystems, each of approximately 50% capacity.

The quench and recirculation spray subsystems are shown on Figure 6.2-47, Figure 6.2-48, and Reference Drawing 1.

Each of the QS subsystems draws water independently from the refueling water storage tank (RWST). Sodium hydroxide solution is added to the QS subsystem water by a balanced gravity feed from the chemical addition tank. The sodium hydroxide enhances iodine removal from the containment atmosphere. The RWST is a vertical cylinder with a flat bottom and a dome top and is secured to a reinforced concrete foundation. It is fabricated of ASTM A240, Type 304L stainless steel, in accordance with API Standard 650. The chemical addition tank is a vertical cylindrical vessel with flanged and dished heads mounted on a skirt and secured to a reinforced concrete foundation. This tank is fabricated of ASTM A240, Type 304 stainless steel, in accordance with Section VIII of the ASME Boiler and Pressure Vessel Code (1971).

Both tanks are designed as Seismic Class 1 components, as described in Section 3.2, to withstand design seismic loading in accordance with the design stress criteria of Section III, Figure N-414, entitled *Nuclear Vessels, ASME Boiler and Pressure Vessel Code* (1968). The connecting piping is designed to withstand seismic loading to ensure the functioning of the system.

The suction line for the Unit 1/2-QS-P-1B pump is equipped with a normally closed piping connection that can be used for refilling the RWST from other water sources or used as a suction connection for a portable pump during a Beyond Design Basis Event.

The maximum fill rate of the refueling water storage tank is approximately 2500 gpm using the residual heat removal (RHR) pump to pump down the reactor cavity after a refueling evolution. The volume of the tanks above the high-level alarm (Section 6.2.2.8.1) is approximately 12,500 gallons. An operator would have to ignore these alarms and level indicators for 5 minutes (worst case) in order for the tank to overflow.

The water in the RWST is maintained between 40°F and 50°F. The water can be cooled to a temperature of slightly below 45°F by circulating the water through heat exchangers that use chilled water from the chilled water system. Mechanical refrigeration units are capable of maintaining the tank water within the operating band. The tank is insulated to limit the average temperature rise of the water to approximately 0.5°F per 24-hour period when the refrigeration units are not operating.

The RWST also has a connection that supplies water to the ECCS (Section 6.3). The tank nozzle outlets connecting to the QS subsystem are located within an enclosure formed by a weir and the wall of the tank. The weir is shown on Figure 6.2-49.

The chemical addition tank has an operating volume of between 4800 and 5500 gallons and is located in close proximity to the RWST. The chemical addition tank and the RWST are connected by a pipe that conveys the sodium hydroxide solution from the bottom of the chemical addition tank through a 6-inch diameter opening to the volume within the weir in the RWST. There it mixes with the borated water flowing to the QS subsystems and flows through two 10-inch diameter openings located symmetrically on either side of the 6-inch inlet. The effect caused by the combination of various flow directions creates turbulence within the weir, which enhances the mixing operation. The mixture is then discharged under turbulent flow conditions to the quench spray pumps where the pump impeller will supply final mixing. The mixing process is not sensitive to any particular flow pattern. Both tanks are adequately vented to permit rapid drawdown.

Two parallel redundant motor-operated valves are located in the line between the chemical addition tank and the RWST. The valves are closed during normal unit operation to prevent mixing of the sodium hydroxide solution with the water in the RWST. Five minutes after receipt of a CDA signal, the motor-operated valves in the line between the RWST and the chemical addition tank open. This delay is to permit the operator to determine if the signal is authentic and to prevent sodium hydroxide injection to the quench sprays if the signal is spurious. As water is pumped out of the RWST, the sodium hydroxide solution flows under its hydrostatic head from the chemical addition tank to the RWST, keeping the liquid levels in the two tanks together once the connecting valves are opened. The height of the chemical addition tank has been chosen so that the column of sodium hydroxide in the chemical addition tank and the column of water in the RWST are in hydrostatic balance after the valves open. When 400,000 gallons of water have been withdrawn from the RWST, the chemical addition tank is empty. Sodium hydroxide addition initiated by a CDA signal cannot be manually terminated unless the CDA signal has been cleared.

The chemical addition tank is insulated and, if required, the fluid is recirculated to keep the tank contents at a temperature above the freezing point of the solution. The chemical addition tank has a low-temperature alarm and a low-level alarm.

The two electric motor-driven quench spray pumps are capable of supplying 1600 to 2000 gpm each of borated water to separate 360-degree quench spray ring headers located

approximately 100 feet above the operating floor in the dome of the containment structure. The quench spray pumps are located in the safeguards area, an enclosure adjacent to the containment structure and the RWST. The pumps have been constructed in accordance with Class II of the Draft ASME Code for Pumps and Valves for Nuclear Power (1968). Each quench spray supply line to the containment contains a weight-loaded check valve to prevent air inleakage to the containment when it is at a subatmospheric pressure. One-quarter-inch drain lines located downstream of the check valves inside the containment will drain the quench spray manifolds should any water enter the manifolds during periodic testing. A stainless steel strainer is provided in the discharge of each quench spray pump.

Figure 6.2-50 shows how the headers are supported and shows the physical orientation with respect to the adjacent structure.

Each of the two 360-degree quench spray headers covers approximately 40% of the entire containment atmosphere and 45% of the atmosphere above the operating floor for the period during which it is the only operating spray system. The two headers have a diameter of 67 feet and are located at Elevations 391 ft. 10 in. and 393 ft. 2 in. Once the four 180-degree recirculation sprays have started, the spray coverage is 86% of the entire containment atmosphere and 87% of the atmosphere above the operating floor. Estimates of volumes covered by each set of spray headers, quench and recirculation, are given as follows:

1. The total volume of the containment that is covered by the quench spray system is 721,000 ft^3.
2. The total volume of the containment that is covered by the recirculation spray system is 1,401,200 ft^3.
3. The upper containment volume (above the operating deck at 291 ft. 10 in.) that is not covered by spray is 141,000 ft^3.
4. The volume below the operating deck that is not covered by spray is 146,000 ft^3.

These figures represent the volumes of the containment atmosphere that are available for spraying and do not include any equipment or structural volumes. They are arrived at in the following way:

1. Calculating the total empty containment volume.
2. Subtracting from this volume all equipment, structural, and enclosed cubicle volumes, yielding a total containment atmosphere volume that is available to be sprayed.
3. Calculating the various volumes covered by the sprays.
4. Assuming that all of the equipment, etc., volumes are contained within the sprayed volumes, thereby maximizing the unsprayed volume.

5. Subtracting this corrected spray volume from the available air volume to find the unsprayed volume.

This process gives the maximum unsprayed volume and the minimum sprayed volume. The cubicle volumes below the operating floor are considered part of the unsprayed volume.

The pattern of the spray coverage, shown on Figure 6.2-51, was determined from the nozzle manufacturer's data. Non-normalized drop size density functions for the quench spray headers and recirculation spray headers are presented in Figures 6.2-52 and 6.2-53. These figures are based on nozzle manufacture's data. Figure 6.2-54 shows the recirculation of the containment caused by the sprays, which will ensure a high degree of containment mixing.

There are no tests that could be performed that would demonstrate the adequacy of the sprays to mix the containment atmosphere without initiating the sprays. Means have been provided to perform qualitative inplace air flow tests on the nozzles. The extremely large flow rates of air necessary to develop a measurable pressure drop across nozzles designed to spray water render quantitative air flow tests unfeasible.

Each RS subsystem, shown on Figure 6.2-48 and Reference Drawing 1, consists of a recirculation spray pump, a recirculation spray cooler, and a 180-degree spray ring header located approximately 85 feet above the operating floor of the containment structure.

Two of the recirculation spray pumps and motors are located inside the containment structure, and two pumps and motors are located outside the containment. The four pumps are of the vertical deep-well type, and essentially are of the same type of design. The outside recirculation spray pumps have shaft extensions long enough to permit locating the pump impellers and suctions at a level below the containment floor, with the motors some 45 feet higher at an elevation slightly below ground grade. The outside recirculation spray pumps are rated at 3700 gpm and the inside pumps at 3300 gpm. The LOCA containment analysis assumes 3050 gpm for the IRS pump and 3350 gpm for the ORS pump. The motors for the inside recirculation spray pumps are of totally enclosed, fan-cooled design. These motors are identical to one that was subjected to environmental testing for qualification for the Surry Power Station. The motors for the outside pumps were selected from a standard proven design. The pumps have been constructed in accordance with Class II of the Draft ASME Code for Pumps and Valves for Nuclear Power (1968).

Unit 1 has the capability to cross-connect the ORS pumps to the SIS (see Figure 6.1-1 and Section 6.3). This cross-connect is not required to mitigate the effects of a postulated LOCA and its use is not considered in the system design evaluation.

The two recirculation spray pumps located outside the containment are fitted with a tandem mechanical seal arrangement. The space between the seal faces is filled with primary grade water, which is maintained at a pressure slightly greater than the recirculation spray pump discharge pressure, thus preventing leakage of radioactive recirculation spray water.

Following a LOCA, water accumulates in the containment from the break in RCS, the RWST, the refueling water chemical addition tank, the casing cooling tank, and the three accumulators. This water accumulates on the containment floor and flows by gravity through strainers and into suction piping of the RS and LHSI pumps. See Figures 6.2-55 and 6.2-58 for the sump arrangement.

The containment sump is a depressed area in the floor to hold water and to provide suction points for the four recirculation spray pumps and, after commencement of the recirculation mode of safety injection, the two low head safety injection (LHSI) pumps. The water is continuously recirculated through the containment to remove heat from the reactor core and containment atmosphere and radioiodine from the containment atmosphere. The water in the containment sump is cooled when pumped through the RS heat exchangers.

One strainer assembly is provided for both the IRS pumps and the ORS pumps. The strainer assembly consists of a number of modules which channel water to the pump suction. Each module contains a number of fins which filter the water flowing into the modules. Each fin contains a number of holes 0.0625-inch (nominal) in diameter that prevents particles larger than 0.06875-inch (0.0625-inch plus 10 percent) from entering the system. Modules are connected to each other by flexible metal seals. Seal closure frames with Metex seals are installed over the existing flexible metal seals. The seal closure frame assemblies form the seal between adjacent strainer modules. The RS strainer assembly consists of two trains which traverse along the containment wall on both sides of the sump. Each suction opening is connected to the modules via the strainer header.

For the ORS pumps, the strainer header is connected to each suction opening by a flanged transition adapter. The OD of the strainer header is machine cut and slip-fitted into the pump suction inlet ensuring that the gaps between the strainer header and the pump suction inlet do not exceed 0.0625 inches.

Since the installation of the strainer assembly, inspections have identified gaps in the assembly larger than the allowable 0.0625-inch gap size. These gaps have been evaluated, and it has been determined that particles entering the strainer through the larger gaps have no adverse effect on components downstream of the strainer.

For the IRS pumps, the strainer header is connected to the pump well via a well housing extension.

The strainer assembly is designed and fabricated to the requirements of ASME Section III, Subsection NF, Class 3. All material used in the construction of the strainer assembly is austenitic stainless steel.

Water from the containment floor is filtered as it passes through perforated fins and into the modules. The filtered water flows through the modules to the pump suction inlets. Two separate strainer assemblies are provided, one for the four RS pumps and one for the two LHSI pumps.

The entire containment sump strainer assembly is raised off of the floor. The bottom of the RS strainer is six inches off the floor. The LHSI strainer is located on the top of the RS strainer, so it sits approximately 19 inches off the floor. Since the strainer is raised off the floor, heavy pieces of debris are prevented from reaching the fins and blocking them.

The fins filter the water as it flows through the strainer and to the pumps. The fins have holes that are smaller than the size of the smallest nozzle orifice in the recirculation spray header. The finned perforated area performs the same function as the original inner sump screens. The fins are hollow tubes, which are perforated with holes having a nominal diameter of 1/16 inch (0.0625 inches).

The strainer is located in an area outside the crane wall. There are no high-energy pipe lines overhead, so jet impingement or pipe whip from a high-energy line break (HELB) is not a concern. In addition, missiles resulting from a HELB accident, for which sump recirculation is required, would not occur close enough to the strainer to damage it.

The strainer assembly is designed to withstand the force of full debris loading in conjunction with all design basis accident conditions including seismic event.

Perforations on the strainer fins prevent particles larger than 0.06875 inches (0.0625 inches plus 10 percent) from entering the RS System. The strainer fins provide filtered water to the strainer header. The total perforation area is large enough to allow sufficient flow to the suctions of the RS pumps to meet NPSH requirements. In addition, particles larger than 0.06875 inches were evaluated in response to gaps identified in the strainer assembly. As part of the evaluation, it was assumed that 1% of the total generated particles between 0.06875 inches (0.0625 inches plus 10 percent) and 0.1375 inches (0.125 inches plus 10 percent) would pass through the strainer. It was determined that these particles would not impact the performance of downstream components. Each suction for the ORS pumps is fed directly from the strainer headers. The IRS pumps take suction from the bottom of a well located within the containment sump. This well is also provided with water directly from the strainer headers. The IRS pumps, piping, and strainer modules are configured such that only water coming directly from the strainer modules reaches the pump suction. Refer to Figures 6.2-55, 6.2-56, 6.2-58 and 6.2-59 for typical arrangement for the strainer headers and modules. From the strainer headers, the flow paths to each of the systems are as follows:

1. Inside Recirculation Spray (IRS) System. From the strainer header, the water travels into the IRS pump suction well, through the pump to a 10-inch discharge line, through the recirculation spray cooler shell side, up to the 8-inch diameter, 180-degree recirculation spray header, and out the spray nozzles into the containment. The internal restriction size of the spray nozzles is slightly larger than the maximum size particles the strainers will pass.
2. Outside Recirculation Spray (ORS) System. From the strainers, the water enters a 12-inch pipe via an 18- x 12-inch reducer. The water flows through containment isolation valves and into the ORS pump casing, through the pump, through containment isolation valves,

through the recirculation spray cooler shell side, and up to the 8-inch, 180-degree recirculation spray header, and out the spray nozzles into the containment. The internal restriction size of these spray nozzles is slightly larger than the maximum size particles the strainers will pass.

The pathway through the LHSI system is discussed in Section 6.3.

Only one strainer assembly has been provided for both LHSI System pumps and one for the RS System pumps. It does not include features that further separate the strainers from opposite pumps within the same system. The basis of the new design is such that the strainer can withstand the full debris loading and has sufficiently large perforated fin area available to compensate for debris blockage.

QS Injection into IRS Suction

The IRS pumps are directly connected to the strainer modules located outside the containment sump via the strainer header. A single bleed line has been hard piped directly to the suction header entering to the IRS pump casing.

The single bleed line allows for proper mixing of the cold water from the bleed line with the water from the IRS suction header as intended since the cold water is injected into the suction line rather than pump casing that allows time for cold water to mix prior to entering into the pump casing. Each bleed line has a flanged connection to the pipe flange provided on the strainer header.

Each 4-inch bleed line has been reduced in size to 2.5 inches near the sump and a 2-inch inline spring loaded flange insert type check valve is installed in each bleed line designed to close when a QS pump trips. Figure 6.2-57 is a simplified flow diagram showing the arrangement.

The check valves are designed to remain closed with minimal amount of leakage during operating conditions for 30 days post-LOCA. The check valves are designed not to begin to open until a nominal cracking pressure of 10 psig at the valve in the normal flow direction has been established.

The net positive suction head (NPSH) available to the IRS pumps is increased by reducing the temperature of the water at the pump suction. This is accomplished by diverting a flow rate that varies from approximately 135 to 155 gpm per pump from the QS Subsystem. The GOTHIC containment analyses assume 150 gpm bleed flow.

All piping and ring headers are Schedule 40 stainless steel pipe. The pipe and necessary fittings are procured and erected to the same code requirement, quality assurance, and seismic standards as the QS piping. The piping and ring headers are designed to accommodate the effects of water hammer.

The NPSH required by the ORS pumps is identical to that for the IRS pumps. However, the line loss in the ORS pump suction must be considered. The NPSH available to the ORS pumps is increased by means of cold water injection from the casing cooling subsystem to the suction piping. Approximately 700 gpm per pump is supplied from the casing cooling subsystem.

The ORS pump NPSH analysis assumes the flow of 3750 gpm per ORS pump. The assumed casing cooling water flow rate of 700 gpm is required to satisfy the NPSH requirement. This colder water provided to the spray nozzles also improves containment depressurization performance. The ORS pump NPSH analysis assumes a casing cooling flow of 700 gpm per pump.

The casing cooling subsystem consists of a storage tank with a nominal volume of 116,500 gallons of chilled (35°F to 50°F) and borated (2600 to 2800 ppm) water. The water is injected into the suction of each ORS pump at a rate of 700 gpm by a casing cooling pump (Reference Drawing 2). The CDA signal initiates operation of the casing cooling subsystem by starting both casing cooling pumps, opening each pump's normally closed discharge valve, and giving an "assure" open signal to each pump's normally open discharge valve. Refer to Figure 6.2-48 and Reference Drawing 2.

Two chiller units and two tank recirculation pumps maintain the chilled water in the casing cooling tank between 35°F and 50°F. Tank level and temperature are monitored by indicators located in the control room. High and low alarms are provided in the control room for both tank level and temperature. Two redundant channels of temperature and level instrumentation are provided.

Each tank level monitor channel automatically closes the respective train related pump discharge valve. Low casing cooling pump discharge flow, concurrent with a CDA signal, automatically closes the respective discharge valve.

The subsystem is normally in the automatic mode, but the casing cooling pumps, as well as the motor-operated valves, can be manually operated from the control room at the end of casing cooling injection.

The casing cooling pumps are powered from separate safety-related buses. The normally closed discharge motor-operated valve shares the power supply of the associated casing cooling pump. The two motor-operated valves in the same discharge line are fed from redundant safety-related motor control centers to ensure containment isolation when required. This arrangement meets the single failure criteria.

The two casing cooling pumps, the two tank recirculation pumps, and the two chiller units are housed in the casing cooling building in Unit 2. One chiller is located outside of the casing cooling building in Unit 1. Each building is a 27-foot by 28-foot steel-reinforced concrete structure, which is seismically designed.

The casing cooling buildings are provided with heating and ventilation systems comprised of electric unit heaters for freeze protection and power roof ventilators for general exhaust purposes.

Each casing cooling tank is a seismically designed, QA Category I welded stainless steel tank with a nominal available volume of 110,000 gallons between the minimum level required by Technical Specifications and the top of the pump suction nozzle. A volume of 90,000 gallons is assumed to be the usable volume for the accident analysis. The tank is set on a steel-reinforced concrete pad located adjacent to its corresponding casing cooling building. The casing cooling pumps are seismically designed, QA Category I, single-stage, centrifugal pumps. The design capacity of the Unit 1 casing cooling pumps is 3000 gpm and the design capacity of the Unit 2 casing cooling pumps is 1000 gpm.

The casing cooling subsystem is shown on Figure 6.2-48 and Reference Drawing 2. Piping is routed underground from the casing cooling building to the safeguards building. With the exception of 2-RS-MR-2, electric power for equipment within the casing cooling originates in 480V motor control centers in the auxiliary building. Electric power for cooler 2-RS-MR-2 originates in a 480V motor control center in the fuel building.

The recirculation spray water flows through recirculation spray coolers where it is cooled by service water. The containment depressurization analysis assumes a service water flowrate of 4410 gpm per recirculation spray cooler, which is 4500 gpm minimum flow reduced to account for 2% tube plugging. Since the recirculation spray water pressure in the coolers is greater than that of the service water, inleakage cannot occur. Therefore, dilution of the borated water in the containment by service water is not possible. This ensures that the necessary boron concentration is maintained. In addition, the temperature of service water discharged from the recirculation spray coolers is monitored as a Reg. Guide 1.97 variable.

Leakage of sump water from the recirculation spray side to the service water side of the cooler is detected by means of radiation monitors. If outleakage is detected, the defective subsystem can be identified and shut down. Chapter 11 describes the monitoring devices and techniques that are employed.

All spray headers have a combination of spray nozzle types that are oriented to obtain a wide distribution of varying size spray droplets. This provides maximum containment spray coverage.

The volume median diameter for the 1/2B40 nozzles in the quench spray header is approximately 660 microns at 40 psid. The volume median diameter for the 1HH30100 nozzles in the quench spray header is approximately 710 microns at 40 psid. The recirculation spray droplets volume median diameter is 775 microns at 20 psid for the 1HH30100 nozzles and 910 microns at 20 psid for the 1/2B60 nozzles.

The individual events that contribute to the time delays in making the quench spray pumps function after a DBA are given in Table 6.2-55.

The quench spray portion of the containment depressurization system has no additional specific delay to minimize the effects on emergency core cooling performance. The quench spray delay is inherent in the system operation because of the time required for pump start and system fill.

The casing cooling flow switch is equipped with a timer to prevent the associated discharge valve from closing on low pump discharge flow before the pump is in service.

The delay in starting the recirculation spray portion of the containment depressurization system allows enhanced core cooling during the reflooding portion of the transient that follows a cold-leg DER. With a higher containment back pressure, the density of steam flowing to the break in the cold leg is higher and the associated pressure drop between the core exit and the break is reduced. This reduces holdup of emergency core cooling water in the vessel reactor downcomer and increases the rate at which the core is reflooded.

The containment depressurization system actuates on a CDA signal, as described in Chapter 7. No additional monitoring system is used to start the containment depressurization system operation.

The entire containment depressurization system is constructed of corrosion-resistant materials, primarily stainless steel. The system has a 150-psig design pressure.

The two ORS pumps, the two low head safety injection pumps, and the associated suction line valves and motors installed outside the containment are designed and installed to account for the differential movement that may occur between these components and the supporting structure. Valve stem extensions and operators are provided to allow remote control of valves located in the valve pit. Restraints are placed where necessary, according to seismic analyses of the structures and piping systems.

6.2.2.2.1 Components

The containment depressurization system is designed, fabricated, inspected, and installed to prevent and/or mitigate the consequences of accidents that could affect the public health and safety.

6.2.2.2.2 Pumps and Valves

The pumps and valves are fabricated, welded, and inspected in accordance with requirements of applicable portions of ASME Codes, Sections VIII and IX. The quench spray, casing cooling, and recirculation spray pumps are also fabricated per the Draft ASME Code for Pumps and Valves for Nuclear Power (1968). Materials of construction are stainless steel or equivalent. The motor-operated and check valves of the casing cooling subsystem, from the casing cooling tank to the outside recirculation spray pumps, are designed to ASME III, Class 2.

Valve packings are selected to minimize or eliminate leakage. The operators for motor-operated valves are selected to ensure reliable operation under accident conditions.

6.2.2.2.3 Piping

Piping fabrication, installation, and testing is in accordance with the USA Standard Code for Pressure Piping - ANSI B31.7-1969 and Addenda through 1970 - Nuclear Piping Code. RWST cooling subsystem piping fabrication, installation, and testing is in accordance with the USA Standard Code for Pressure Piping - ANSI B31.1-1967 - Power Piping.

6.2.2.2.4 Motors

Motors are designed and tested in accordance with ANSI, IEEE, and NEMA standards. Stator temperature rise for the inside recirculation spray pump motors is 80°C, measured by resistance at 100% load. Stator temperature rise for the outside recirculation spray pump motors is 80°C, measured by resistance at 115% load. Stator temperature rise for the quench spray pump motors is 90°C, measured by resistance at 115% load. Electrical insulation resistance tests are performed during the lifetime of the motors to verify the integrity of the insulation. Periodic tests are also performed to ensure the motors remain in a reliable operating condition. In addition, the IRS pump motors are designed and tested for LOCA conditions.

6.2.2.2.5 Heat Exchangers and Vessels

The recirculation spray heat exchangers are designed to the ASME Code Section IIIC (1968). The pump casings are designed to the ASME Code Section IIIB (1968). The chemical addition tank is designed to ASME Code Section VIII (1971); the RWST is designed to API Standard 650. The heat exchangers and vessels are radiographically inspected to ensure their structural integrity. Welded construction is used to preclude leakage.

In order to ensure long-term reliability of the recirculation spray cooler, following each periodic test of the heat exchanger inlet and outlet valves, the heat exchangers are put in dry layup by first isolating the heat exchangers, and then draining water out through the heat exchanger drain valves. An air hose is then attached to each of the four heat exchanger vents, and the exchanger is purged with compressed air until there is no visual indication of moisture discharging from the drains. The vents and drains are then closed.

After this drain and purge operation, it is possible that a residual film of water may remain on the heat exchanger surfaces. It is not considered feasible that chloride stress corrosion would result from this condition due to the following reasons:

1. The source of service water contains less than 300 ppm chloride. Stress corrosion cracking is not expected in austenitic stainless steel with chloride content up to 300 ppm and water temperature below 110°F. Chloride content in the Service Water Reservoir is annually monitored to ensure 300 ppm is not exceeded.

2. The service water side of the coolers (tube side) is of a straight tube design. This type of design does not invite stress corrosion because there are no critical interfaces that contain crevices or other such places where water could become trapped.

During unit operation, the level of water in the service water supply and discharge lines to the recirculation spray heat exchangers is monitored through the performing of a periodic test. This is done to ensure that the level does not rise to a point that any leakby of the isolation valves would result in water entering the heat exchanger and cause fouling.

The effect of fouling of these heat exchangers on system performance is discussed in Section 6.2.2.3.

The design data for the containment depressurization system components are given in Table 6.2-39.

6.2.2.3 Design Evaluation

The Containment Depressurization System consists of two completely separate, 100% capacity Quench Spray Subsystems and four completely separate 50% capacity RS Subsystems. The use of a separate spray header connected to the discharge of each pump results in a fixed flow rate. In addition, this equipment arrangement also ensures that a failure of a component in any one subsystem does not affect the operational capability of the other subsystems.

6.2.2.4 Spray Nozzles

Spray nozzles have been selected in various sizes to give the optimum combination of small spray particles for maximum heat transfer and larger particles for better coverage toward the center and sides of the containment.

6.2.2.5 RS Strainer Assembly

NRC Generic Letter 2004-02 (Reference 47) required licensees to perform an evaluation of the emergency core cooling and containment spray systems functions in light of the potential impact of debris blockage on the originally installed containment sump screens. The potential debris blockage could have resulted in a debris-induced loss of NPSH margin during sump recirculation. The potential impact of the debris generation and transport was evaluated and this evaluation was provided to the NRC (Reference 48).

The response resulted in the completion of several evaluations and tests to determine the impacts of the new requirements in GL 2004-02 to the original containment sump screens. This resulted in the replacement of the original containment screen assembly with a passive sump strainer design.

Evaluations and tests were performed in accordance with the NEI 04-07 (Reference 49) and its associated SER (Reference 50) to determine the size of the strainer that would ensure that the

post-accident debris blockage will not impede the operation of the LHSI and RS systems in the recirculation mode.

The following evaluations and tests were performed:

- Evaluation of debris generation caused by a LOCA
- Evaluation of debris transport to the strainer
- Evaluation of downstream effects of blockage and wear on components
- Evaluation of downstream wear effects on system performance
- Evaluation of downstream effects of blockage and chemical precipitation on fuels
- Strainer hydraulic test to determine head loss due to debris and chemical effects
- Strainer fiber bypass test

The following types of materials were determined to become debris and chemical effects contributors in the event of a LOCA:

- Piping and Equipment Insulation
- Insulation Jacketing
- Missile Barrier Penetration Seals
- Qualified Coatings
- Unqualified/Damaged Coatings
- Latent Debris
- Fire Stop Materials
- Foreign Materials
- Aluminum Materials
- Coated and Uncoated Concrete

The amount of debris generated during a LOCA was determined such that it would maximize the head loss across the containment sump strainer during recirculation mode. A number of breaks in the reactor coolant system were considered to bound variations in debris generation by size, quantity, and type of debris. All LOCA and high energy line break generated debris are conservatively evaluated as falling to the containment elevation 216’-11”.

The steam generator cubicles contain the largest diameter high energy piping and the largest quantity of insulation that could be exposed to a high-energy coolant jet and/or whipping pipe. No other mechanism for insulation dislodgement has been identified. The area of influence of a high-energy coolant jet is also the largest in the steam generator cubicles due to the large pipe

diameters present. Break areas inside the steam generator cubicles are discussed in Section 6.2.1.1.2.

The volume of insulation contributing to the total debris inventory is calculated using the deterministic methodology described in NEI 04-07, Reference 49. This methodology utilizes a spherical Zone of Influence (ZOI) radiating from the pipe rupture. Insulation and protective coatings located within the ZOI are assumed to be damaged and removed from pipe or equipment. The ZOI radius varies depending on the type of material impacted. The ZOI radius is given in terms of the number of pipe diameters from the ruptured pipe.

Debris which falls in small particles poses the largest threat to RS (Recirculation Spray) System and ECCS pump performance. This particulate debris can be generated from latent debris and protective coatings. In addition to the particulate debris other potential threats to the RS System and ECCS are fibrous insulation and reflective metal insulation.

In the event that recirculation system problems such as pump vibration or loss of pump discharge pressure occur, instrumentation is provided in the main control room to help the operator recognize and contend with these problems. Instrumentation available to monitor recirculation is summarized in Table 6.2-38.

The replacement steam generators employ a removable encapsulated fiberglass insulation system. The insulation system consists of a light density fiberglass insulating material encapsulated in a tough woven fiberglass cloth to form a blanket or pillow. The pillows are attached together with Velcro and are covered with a protective and removable stainless steel sheathing. Encapsulation of the fiberglass results in a significant increase in insulation system strength and resistance to the impinging jet forces which emanate from a postulated pipe rupture.

Reflective metal insulation (RMI) consists of multiple layers of 0.002-inch thick type 304 stainless steel foil suitably supported by stainless steel type 304 clips and spacers, enclosed by a casing of 24-gauge type 304 stainless steel. The type 304 stainless steel foil does not break down when wet, but shredding and/or tearing my occur if it is directly impinged upon in the ZOI of a pipe rupture. Reflective metal debris, based on NEI 04-07 methodology, breaks down into small fines and large pieces. The small fines are determined to transport to the containment sump strainer. Large pieces are determined to not transport into the containment sump strainer due to the material's high density.

Fibrous insulation debris consists of materials such as Tempmat, Thermal Wrap, and Mineral Wool. Most fibrous insulation within containment is jacketed in stainless steel and will not deteriorate when wet, therefore most fibrous debris results from direct impingement of the insulation. If the fibrous insulation becomes debris it would fall to the containment floor. All fibrous debris except that which has been considered to be intact pieces of debris would transport to the containment sump strainer.

As described in Section 3.8.2.7.6, only protective coatings (paints) on exposed concrete and carbon steel surfaces remain intact if subjected to the environment associated with a postulated LOCA. All unqualified protective coatings will fail in a postulated LOCA environment. It should be noted that the paint on the steam generators and pressurizer is not qualified. The surfaces of these components are covered with metal jacketed insulation and therefore, normally not exposed.

No sand plugs, sand bags, or loose insulation are located inside the containment.

Some insulation fragments or small pieces are carried immediately to the floor by the pressure difference resulting from the rupture. Some fragments are directed away from the containment floor and sump by the pressure difference and some remain in the subcompartment on gratings and horizontal surfaces. After the blowdown terminates, the spray systems begin to wash down the small fragments of insulation. The most plausible path for debris to exit the steam generator cubicles is through the grating or blowout panel openings at Elevation 241 ft. 0 in. adjacent to the primary shield wall. Debris falling through these openings would fall to the containment floor no closer than 32 feet from the containment sump strainer.

Debris created by a LOCA inside the lower reactor cavity would be expected to reach the containment sump strainer through a 12-inch diameter drain hole core drilled through the primary shield wall plug in the lower portion of the reactor cavity (Incore Sump Room). The debris generated would have to pass through the narrow gap between the reactor vessel and the neutron shield tank before reaching the drain opening and being conveyed to the containment sump strainer.

Based on NEI 04-07, debris types were determined to not transport, completely transport, or partially transport depending on the debris characteristics. The maximum postulated debris transported to the strainer was used to determine its size. Based on the debris transported to the containment strainer, the head loss across the strainer was determined. The strainer head loss was evaluated for several different strainer debris inventories.

The strainer manufacturer has performed various hydraulic tests that simulated the actual debris loading and chemical conditions specific to the North Anna Power Station based on the debris generation, debris transport, and chemical effects evaluations. Fibrous, particulate, and chemical debris were added to a test rig to simulate the plant-specific chemical environment present in the water of the containment sump. Each test was performed for more than 30 days after the formation of the debris bed and initial chemical addition at specified temperatures and flow rates to assess chemical precipitate formation and head loss change. These tests sized the strainer and verified that adequate NPSH is available to support the operation of the LHSI and RS pumps during recirculation mode.

The downstream effects evaluation was performed in accordance with WCAP-16406-P (Reference 53) to determine whether RS System and ECCS components are susceptible to blockage and wear due to debris bypass post-LOCA. The evaluation determined that downstream

components have sufficient flow clearances in the RS System and ECCS flow paths that would allow debris to pass through openings without causing blockage.

The downstream effects evaluation determined the effects of erosive and abrasive wears on RS System and ECCS components, overall system hydraulic performance, and the system piping vibrations. A wear model was developed in accordance with methodology provided in WCAP-16406-P to assess the amount of wear in RS System and ECCS components based on the initial debris concentration in the pumped fluid, the debris concentration depletion, the hardness of the wear surfaces, and the mission time. The results for wear on the manually throttled valves, orifices, containment spray nozzles, and RS heat exchanger were determined acceptable in accordance with criteria set forth in WCAP-16406-P. The dynamic performance of RS System and ECCS pumps is evaluated for wear effects on pump components. The evaluation concluded that the RS System and ECCS pumps meet the acceptance criteria for vibrations specified in WCAP-16406-P and therefore, will operate satisfactorily without excessive vibrations for a period of 30 days post-LOCA. The degraded hydraulic performance curves for RS System and ECCS pumps were developed resulting from erosive and abrasive wears of pump internal components. Changes in system resistance due to wear of system components such as orifices and manually throttled valves and degraded pump hydraulic performance were used as input to system models to evaluate whether minimum system flow requirements would be met. Modeling was then used to establish whether the degraded system resistance would cause the pumps to operate in an unacceptable run out condition. The results of the evaluation indicated that RS System and ECCS pumps are acceptable with respect to run out flow and will meet the minimum flow requirement to depressurize the containment and cool the reactor core for a period of 30 days post-LOCA.

System vibration analysis was performed to evaluate the effects of wear of RS System and ECCS piping and components downstream of the containment sump strainer. The RS System and ECCS pumps were also evaluated for hydraulically induced vibrations using maximum flows derived in the hydraulic analysis portion of the evaluation and for pump rotor dynamic analysis affected by the erosive and abrasive wears of pump internal components due to debris in the pumped fluid. The effects of wear on net positive suction head (NPSH) requirements and availability were considered to ensure cavitation would not induce unacceptable pump vibration. The evaluation indicated that the expected erosive and abrasive wear of the RS System and ECCS piping would be negligible after 30 days post-LOCA and therefore, the structural characteristics of the systems considered are not impacted. The RS System and ECCS pumps are acceptable for hydraulically induced vibration and meet the acceptance criteria for rotor dynamic vibration documented in WCAP-16406-P (Reference 53). Pump cavitation will not occur since the available NPSH exceeds the required NPSH for all RS System and ECCS pumps. Therefore, based on the acceptability of downstream wear effects and pump vibration and cavitation analyses, the evaluation concluded that the RS System and ECCS piping and components are not susceptible to excessive vibrations due to post-LOCA downstream wear.

The downstream effects evaluation was performed for fuels in accordance with WCAP-16793-NP (Reference 54) to determine the impact of fibrous, particulate, and chemical precipitant debris on the fuel and long-term cooling. The evaluation demonstrated that all of the WCAP evaluations and conclusions are directly applicable to Units 1 and 2. This provided reasonable assurance that for both units long-term core cooling will be established and maintained post-LOCA considering the presence of debris in the Reactor Coolant System and core.

Based on the above evaluations and tests, the flow area of perforations in the strainer fins was determined to be sufficient such that under full debris loading conditions there would be adequate NPSH available to the RS and LHSI pumps during accident conditions.

The strainer modules and fins extend out from the sump area along the containment wall and are arranged so that no single failure could result in the clogging of all four suction points to the RS Subsystems. These modules and fins strain the fluid being drawn to the RS pumps to prevent spray nozzle plugging. The strainer design provides sufficient perforated fin area and ensures that the assembly is capable of withstanding the force of full debris loading, in conjunction with all design basis conditions, including seismic events. Therefore, no physical failure of the strainer assembly is considered which could limit the strainer's effectiveness in carrying out its intended design function during accident conditions.

The required heat removal capacity of the RS Subsystems continually decreases after the first several hours following a LOCA; therefore, excessive plugging of nozzles, which could only be considered on a long-term basis, would have no significant effect on the capability of the subsystems.

6.2.2.6 **Component Corrosion**

Following a LOCA, the water sources for the sumps feeding the recirculation spray pumps are the RCS, the RWST, the refueling water chemical addition tank, the casing cooling tank, and the three safety injection accumulators. The chloride content of the refueling water chemical addition tank is limited to a maximum of 2000 ppm and the RCS, RWST, the casing cooling tank, and the three safety injection accumulators are limited to a maximum of 0.15 ppm.

Credible leachable chloride addition to the sump water has been investigated. This investigation included such sources as thermal insulation, neutron shielding, electrical cables, paint, and concrete. From this investigation it has been concluded that a negligible amount of leachable chloride will be in the sump water. Each of the possible sources that have been considered is addressed below:

1. Thermal Insulation. Seven types of thermal insulation are used in the containment. Each one has been analyzed for leachable chlorides and found acceptable for use on stainless steel. Therefore, the amount of leachable chlorides available to the sump water will be negligible.

2. Neutron Shielding. The neutron shielding is a dimethyl polysilaxane silicone elastomer and is totally free from chlorides.

3. Electrical Cables. The insulating material will not leach chlorides or fluorides under a LOCA containment environment.
4. Containment Paint. Paints used in the containment are discussed in Section 3.8.2.7.6. Paint manufacturer's data indicate that the maximum halide concentration of the paint is 50 ppm. A calculation was performed to determine sump water halogen concentration, assuming leachable halides are credible from the top half of the paint layer. The results have shown that the sump water halide concentration would be approximately 0.17 ppm. This concentration is only 0.02 ppm higher than the allowable for the reactor coolant. The sump water temperature is less than 150°F approximately 3 hours after an accident. Considering the low sump water temperature and low halide concentration of this water, stress corrosion will be negligible.

Qualification testing of the inside recirculation spray pump motors has been conducted to prove that the motors will function without degradation of performance in the accident environment. The tests were performed on a Surry Power Station inside recirculation pump motor, and apply to the North Anna Units 1 and 2 inside recirculation spray pump motors because the motors are identical. The details of the test program may be found in the Surry Power Station topical report, *GE Vertical Induction Motors - Inside Containment Recirculation Spray Pump Motors*, filed under Dockets Nos. 50-280 and 50-281. The test program included exposure of the motor to a nuclear radiation dose exceeding that anticipated from normal and post-accident service, a vibration test simulating seismic disturbance, and a steam/chemical spray exposure simulating the containment environment following a LOCA. These tests were preceded by thermal aging of the motor.

Qualification testing of the recirculation spray pumps to determine stress corrosion of pump materials that come in contact with sump water has not been performed. A test of this nature is not considered necessary because of the strict chemical control of all sources of water entering the sump and because of the corrosion-resistant qualities of the recirculation spray pump materials.

The results of the test program verify that the inside recirculation spray pump motors are capable of functioning in an accident environment. The outside recirculation spray pump motors are not exposed to the spray solution, and therefore preclude such requirement.

Deposition of corrosion products or chemical precipitation so as to restrict lines or reduce heat transfer is not expected because the turbulent flow conditions within the system preclude precipitation. Also, because of the corrosion-resistant properties of the recirculation spray system, plating of corrosion products is negligible. Zinc will accumulate at a rate of less than 1 ppm/day. Turbulent flow will keep it in suspension in the recirculation spray system. The zinc oxides and hydroxides will deposit out in the sump areas where flow is low. The small amount will not result in any appreciable buildup. Consequently, qualification testing of the recirculation spray system to determine the effects of plating of corrosion products or chemical precipitation so as to plug lines or reduce heat transfer is not necessary.

Qualification testing on the recirculation spray nozzles to demonstrate that they will function without degradation of performance due to corrosion has not been performed. Because of the corrosion-resistant material chosen for the nozzles, degradation of the spray nozzles is not expected. However, an inspection or air/smoke test of the spray nozzles is conducted as described in the technical specifications.

The RS subsystem coolers are laid up with the tube side clean and dry. There may be some fouling of the tubes, for long-term operation, with resultant loss in heat transfer capability. This loss of heat transfer capability would be more than offset by the decrease in heat load due to decreasing decay heat production. The shell side of the RS subsystem coolers are maintained dry, though some water may accumulate due to condensation and/or valve leakby during pump testing. These sources of water would not challenge the ability of the RS subsystem coolers to meet their design function. The shell side of the RS subsystem coolers are periodically drained.

Borated water for quench spray and safety injection is stored in the refueling water storage tank which is 304 stainless steel and is subject to negligible corrosive attack from this solution. The sodium hydroxide is stored in solution in the chemical addition tank which is also 304 stainless steel and is subject to negligible corrosion. Borated water is stored in 304 stainless steel clad safety injection accumulators under a nitrogen blanket and stainless steel clad boron injection tank.

Recirculation pumps and sample points are provided on the refueling water storage tank and chemical addition tank to allow testing of the fluid to verify proper chemical concentration.

The recirculation spray heat exchangers are designed in accordance with Section III of the ASME Boiler and Pressure Vessel Code (1969) and have welded construction at all points where there could be a potential for leakage of radioactive recirculation spray into the service water. The pressure at all points on the recirculation spray side of the heat exchangers is greater than that of the service water side. This prevents any inleakage of nonborated water into the reactor containment, which would decrease the reactor shutdown margin. The service water lines from the recirculation spray heat exchangers are equipped with radiation monitors to detect leakage and any defective subsystems could be shut down if leakage above the allowable limit is detected.

Piping, valve packings, and pump seals in the RS subsystem outside the containment are designed to minimize the probability of leakage. However, should leaks occur, they would be controlled as follows:

1. Large leaks in the discharge piping of the outside recirculation spray pumps would be detected by variations in the recirculation spray pump discharge pressure readings in the main control room and by sump pump alarms. If such a pipe break should occur, the operator in the main control room can stop and remote-manually isolate the pump involved.

2. Large leaks in the suction piping in the valve pit occurring during the long-term recirculation period are detected by liquid level measuring devices. The valve pit is provided with a baffle,

dividing the pit into two sections. Thus, leakage from one set of recirculation spray or low head safety injection suction lines is detected by the increased liquid level on the affected side of the baffle. Upon detection, if the break is between the isolation valve and the pump, the operator in the main control room can remote-manually isolate the leaking pipe, leaving the other outside recirculation spray loop operable. If the break is in the short length of pipe between the containment and the isolation valve, the pump can be left in operation by permitting it to take suction directly from the flooded section of the valve pit. This does not endanger the public as the water cannot rise high enough in the safeguards structure to escape.

In the case of small leaks, immediate detection and subsequent isolation of a leak may not be possible. However, the atmosphere of the safeguards structure enclosing the piping is discharged to the atmosphere through high-efficiency particulate air (HEPA)/charcoal filters when the subsystem is operating. These leaks will eventually be indicated by a rise in valve pit or safeguards area sump levels in sufficient time to allow corrective action to prevent flooding of equipment.

6.2.2.6.1 System Lines Penetrating Containment

The systems with lines penetrating the containment that are provided with remote manual isolation valves that are open following a LOCA are safety injection (low head safety injection pump suction and high head discharge lines), quench spray (pump discharge lines), recirculation spray (outside recirculation spray pump suction and discharge lines and casing cooling discharge piping), feedwater (auxiliary feed pump discharge lines), chemical and volume control (reactor coolant pump seal injection lines), and service water (supply and return for the recirculation spray coolers).

The methods of detection of a break in the lines outside the containment for the above-listed systems are explained below. Also given is the method for ensuring the integrity of a containment isolation boundary against inleakage of air, following a break in the external piping.

A break within the quench spray pump discharge lines, auxiliary feed pump discharge line, or the reactor coolant pump seal injection lines is not discussed because these lines operate for only a short period of time after a LOCA, and a passive failure in the short term is not credible.

A break in the service water lines to or from the recirculation spray coolers results in flow into the quench spray pump house basement, where it is indicated by a high level alarm.

Containment pressure is subatmospheric within 6 hours after a LOCA. Therefore, consistent with the single failure design bases presented in Section 3.1, pipe breaks discussed are postulated and analyzed only for subatmospheric containment conditions.

The following methods assure that a containment isolation boundary to prevent air inleakage is maintained for the systems listed above. For the suction lines to the low head safety injection pumps or the outside recirculation spray pumps and to the discharge of the casing

cooling pumps from the containment sump, the water level in the containment gives sufficient head to prevent air inleakage in the event of a break in one of the pipes.

Each of the low and high head safety injection pumps discharge lines has check valves within the containment. If a break should occur external to the containment, the check valves in that line are sealed closed by the back pressure from the adjoining lines, maintaining isolation.

A break in the discharge lines of the outside recirculation spray pumps does not cause a loss of containment isolation because each line has a weight-loaded check valve designed to preclude inleakage as well as outleakage of air.

The service water line to the recirculation spray coolers is a closed loop inside the containment. In the event of a line break external to the containment, isolation is maintained.

Table 6.2-40 summarizes the potential leakage from RS subsystems. Leakage of pumped fluid from the containment recirculation spray pump shafts does not occur due to the manner in which the pump is sealed. Two mechanical seals are arranged in tandem with a fluid seal between them. The seal fluid is supplied from a reservoir arranged in such a manner that the pressure of the seal fluid is slightly above (approximately 1.0 psi) the pumped fluid pressure at the inboard side of the inboard seal. This arrangement is provided so that, assuming a single seal failure, seal fluid will leak through the failed seal and the other seal will remain available to prevent escape of pumped fluid to the atmosphere. A level alarm on the reservoir provides indication of seal failure. Makeup is through a hose connection on the seal reservoir.

Consistent with letters from the ACRS (References 23 & 24) concerning vital piping that must function during a LOCA, passive failure of the recirculation spray suction piping during a LOCA in the short term is not considered credible. If a large leak were to occur in the long term after depressurization of the containment atmosphere to subatmospheric pressure had been achieved, the valve pit would flood with recirculation water. The spilled recirculation water would provide a water seal to prevent inleakage of outside air into the containment and subsequent return of the containment to atmospheric pressure.

Flooding of the valve pit area has no effect on the operation of the LHSI system or the RS subsystem. The pumps are not affected by flooding since they are of the vertical shaft type with discharge head and motor drivers located some 45 feet above the valve pit.

Table 6.2-41 discusses the likelihood and consequences of various component malfunctions.

6.2.2.6.2 Recirculation Spray Pump Net Positive Suction Head

The containment recirculation spray pumps are capable of meeting NPSH requirements under accident conditions. Sufficient NPSH is available when the pumps are started (at the maximum sump water temperature) and after the containment structure has been returned to

subatmospheric conditions. A sufficient margin of NPSH is available over and above that required for satisfactory pump operation for all post-LOCA conditions.

A transient GOTHIC calculation is performed to demonstrate that the IRS and ORS pumps have adequate NPSH throughout a postulated design basis LOCA. The NPSH available (NPSHa) must be greater than the required NPSH at all times during the accident. The calculation of NPSHa with GOTHIC follows the methodology outlined in Section 3.8 of Reference 51. Section 6.2.1.3.1.3 demonstrates that the RS pumps are not needed for MSLB mitigation, so only LOCA events are analyzed for RS pump NPSHa. Key analysis parameters are shown in Table 6.2-2. The GOTHIC analyses start the RS pumps on RWST Level Low (60% RWST wide range level nominal setpoint) coincident with a CDA High High containment pressure. The delay in operation of the RS pumps allows sufficient water to accumulate on the containment structure floor to satisfy the submergence requirements for the RS strainer.

6.2.2.6.2.1 *NPSH Analyses.* The GOTHIC methodology ensures a conservative calculation of RS pump NPSHa as described in Topical Report DOM-NAF-3. Sensitivity studies were performed on break location, single failure, and initial conditions to identify the most limiting NPSHa for the IRS and ORS pumps. GOTHIC analyses of RS pump NPSHa are limiting at the Technical Specifications minimum containment air partial pressure of 10.3 psia and 35ºF service water (SW). The cold SW temperature produces cold RS spray that provides a fast containment depressurization. The DEPSG break produces a higher long-term energy release to the containment than the DEHLG break because of the available energy in the SG secondary side. For all single failure scenarios with RS pump start on RWST Level Low coincident with a CDA, the DEPSG break produces a lower NPSHa than the DEHLG break. Maximum RS pump flow rate is conservative for determining the NPSHa for that pump because it causes the highest suction friction loss and imposes the most restrictive NPSH required.

The minimum NPSHa for the IRS pump is 15.17 ft (5.13 ft of margin) for a DEPSG break with the single failure of an emergency diesel generator (loss of an emergency bus). The loss of QS bleed flow and the casing cooling pump on the failed bus provide the minimum subcooling and water level benefit. This more than offsets the lower spray flow rate compared to other cases. The IRS pump NPSHa analyses assume a maximum IRS pump flow rate of 3425 gpm and a QS bleed flow rate of 150 gpm. Figure 6.2-65 (available NPSH and water level), Figure 6.2-66 (containment and IRS pump suction vapor pressure), Figure 6.2-67 (containment vapor and liquid temperature), and Figure 6.2-68 (RS cooler heat rate) illustrate the performance of key variables.

The minimum NPSHa for the ORS pump is 18.29 ft, with 6.421 ft of margin, for a DEPSG break with the single failure of a casing cooling pump. The ORS pump NPSHa analyses assume a maximum ORS pump flow rate of 3750 gpm and a casing cooling pump flow rate of 700 gpm. The IRS pump has more NPSH margin than the ORS pump. Figures 6.2-69 through 6.2-72 show the behavior of key variables from the ORS pump NPSH limiting case.

For the RS pump NPSHa analyses, the minimum containment water level is 1.86 ft above 216'11" floor elevation (where the sump and the RS strainer are located) when the ORS pump starts for an assumed single failure of a casing cooling pump during a DEHLG break. This water level assumes a conservative holdup volume in containment of about 42,400 gallons, earliest pump start using 2.5% level uncertainty on the trip setpoint, and minimum initial RWST volume.

Pump	Flow Rate (gpm)	Minimum NPSHa [a] (ft)	NPSHr (ft)	Minimum Margin[a] (ft)
Outside RS	3750	18.29	11.86	6.421
Inside RS	3425	15.17	10.04	5.13

a. The minimum NPSHa from GOTHIC does not include the RS strainer clean and debris head loss. The total RS strainer head loss is less than the reported NPSH margin.

6.2.2.6.2.2 *Long-term NPSH Margin for RS Pumps.* When the RWST and casing cooling tanks are emptied, the injection flow to the RS pump suctions is lost. However, significant NPSH margin is available at these times and there is no adverse effect on long-term containment cooling. NPSH margin was analyzed for 10,000 seconds to show the effects of exhausting the tanks. When QS stops on empty RWST, the IRS pump NPSHa decreases a small amount initially but then continues to increase as sump temperature drops. When casing cooling stops, the ORS pump NPSHa decreases from 30.85 ft to 28.06 ft (for the limiting case) and remains level for the duration of the analysis. There is sufficient NPSH margin for long-term cooling from the RS system.

6.2.2.6.2.3 *Water Drainage in Refueling Cavity.* There are two paths by which any water drains from the refueling cavity:

1. Spray water drains through the 16 ft^2 of open annulus around the reactor vessel and floods the reactor cavity and the incore instrumentation tunnel. A 12” diameter core drilled hole provides a flowpath from the lower reactor cavity to the containment sump strainer with the centerline of the drain hole at El. 219’-6”. The drain is designed to convey the held up water, above the invert elevation of the Incore Sump Room drain, from the Incore Sump Room to the containment sump strainer. This additional water facilitates submergence of the containment sump strainer for RS and LHSI pumps during post-LOCA operation.

2. Water also flows from the refueling cavity to the containment basement through a 6-inch drain located in the fuel transfer portion of the refueling cavity. The drain is located in the bottom of the fuel transfer canal at an elevation of 251 ft. 4 in. and is shown on Reference Drawing 11. The GOTHIC analyses of NPSH available in Sections 6.2.2.6.2 and 6.3.2.2.6 do not credit this drain.

An analysis indicates that the refueling cavity represents 9.2% of the area exposed to the quench and recirculation sprays. The fuel transfer canal receives 11% of the spray falling into the refueling cavity. NPSH analyses for the RS and LHSI pumps account for time-dependent holdup of water that is sprayed into the refueling cavity and stored below the spillover elevation of

262'10". Water above this elevation will flow into the reactor cavity. The Incore Sump Room drains to the area outside the Primary Shield Wall (PSW) through the ISR Drain. The water volume, above the invert elevation of the Incore Sump Room drain, in the Incore Sump Room is considered available to the containment sump strainers.

The valves in the refueling cavity transfer canal drain lines are administratively controlled by procedure and by physically chaining and locking the valves in the open position during normal operation of the reactor.

Operators are required to verify all valve positions before starting or resuming normal operation when it is known that those systems have been used during the shutdown period.

Conversely, when the reactor is being refueled, these valves are locked closed under similar administrative control.

6.2.2.6.3 Containment Depressurization Time

This section describes the loss-of-coolant accident (LOCA) containment depressurization transient analyses that are performed to confirm that the containment pressure is less than the assumed pressure profile in the LOCA dose consequences analyses in Section 15.4.1.9. Containment response analyses are performed using the GOTHIC computer code and the methodology described in Reference 51. Refer to Section 6.2.1.1.1.2 for the GOTHIC methodology description. The analyses are performed for a rated core power of 2940 MWt plus 0.37% calorimetric uncertainty (total core power of 2951 MWt). Key analysis input parameters are shown in Table 6.2-2.

The time required to depressurize the containment and the capability to maintain it subatmospheric after a double-ended pump suction guillotine (DEPSG) break depends on the design of the containment depressurization systems, SW temperature, and the mass of air in the containment. The DEPSG break is limiting because it has the largest energy release to the containment due to the available energy removal from the steam generator secondary side. When SW temperature is elevated, it is more difficult to depressurize the containment and containment air partial pressure must be reduced to meet the depressurization limits. Thus, the containment air partial pressure is controlled as a function of service water temperature according to the Technical Specifications. This is required to ensure that the containment pressure following a LOCA will be less than 45 psig during the first hour, less than 2.0 psig during the period from 1 to 6 hours, and subatmospheric after 6 hours. The containment pressure limits are consistent with the containment leak rate assumptions in the LOCA dose consequences analysis in Section 15.4.1.9.

Once the operating QS pump is stopped after RWST depletion, only the RS system provides spray flow to the containment and at higher temperatures than the QS system. Once QS is terminated, the containment pressure increases until it reaches the depressurization peak pressure, which is limited by the heat removal capacity of the RS system and the air mass in containment. A minimum initial containment temperature is conservative for depressurization peak pressure

analyses, because a higher initial air mass makes it more difficult to maintain subatmospheric conditions after QS termination.

The loss of one emergency bus is the limiting single failure because it provides only one train of spray flow for containment atmosphere cooling.

Appropriate delay times for receipt of ESF signals, valve operation, and pump starts are inputs to the GOTHIC code.

Minimum engineered safety features are defined as those engineered safety features that operate during loss of all offsite electric power, and with one emergency diesel generator per unit available. These minimum features are as follows:

1. The ECCS safety injection flow, comprising discharges from one charging pump and one low head safety injection pump, becomes effective 30 seconds after the start of the accident.
2. Three nitrogen-pressurized accumulators discharge into the RCS when RCS pressure drops below accumulator pressure.
3. One quench spray pump becomes effective (i.e., spray flow leaves the nozzles) no later than 71.1 seconds after the loss of offsite power that occurs concurrent with the LOCA (see Table 6.2-55). The GOTHIC analyses assume that the QS spray is effective 70 seconds after the CDA setpoint of 30.0 psia is reached. From Table 6.2-12, the GOTHIC analyses take at least 2.6 seconds to reach the CDA setpoint and QS is effective no earlier than 72.6 seconds. The analysis is conservative with respect to the 71.1-second effective time.
4. The two recirculation spray pumps start when the RWST wide range level reaches the low level setpoint coincident with a CDA signal. The ORS pumps receive an immediate start signal and the IRS pumps start after a 2-minute delay time. The RWST Level Low plant setpoint is 60%, but the containment depressurization analyses assume a lower limit of 57.5%. The timing of the RS pump start depends on the initial RWST volume and the QS and SI pump flow rates.
5. The analyses assume that the casing cooling pump begins injected chilled water 55 seconds after the CDA signal. The casing cooling pump discharge water flows into the containment sump until the corresponding ORS pump starts. Then the casing cooling pump water goes entirely to the ORS pump suction.

Table 6.2-12 compares the accident chronology for the limiting containment depressurization analyses performed at service water temperatures of 55ºF and 95ºF and their corresponding maximum Technical Specifications containment air partial pressure limits. Table 6.2-12 provides the time to depressurize below 2.0 psig, the depressurization peak pressure, and the final time to depressurize to subatmospheric conditions. Figures 6.2-62 through 6.2-64 show results from the containment depressurization analysis at Technical Specification limits of 55ºF service water and 12.3 psia containment air partial pressure. Figure 6.2-62 shows the containment pressure profile. Figure 6.2-63 shows the containment vapor and sump water

temperature. Figure 6.2-64 shows the total recirculation spray cooler duty. The GOTHIC analyses demonstrate that the containment pressure profile is less than 45 psig during the first hour, less than 2.0 psig during the period from 1 to 6 hours, and subatmospheric after 6 hours. Thus, the containment analyses are bounded by the leak rate assumptions in the LOCA dose consequences analysis in Section 15.4.1.9.

6.2.2.7 **Testing and Inspection**

6.2.2.7.1 Quench Spray Subsystems

Two types of tests are performed on the QS subsystems. The first type of tests are performed after installation and prior to station operation to ensure that the subsystems meet the design criteria. The second type of tests provide for testing the subsystems throughout the life of the station to ensure the operability of the subsystems.

The quench spray headers are fitted with blind flanges in order to connect temporary drain lines needed for testing the subsystems. After the subsystems are completely installed, the temporary drain lines are connected to the blind flanges and pipe plugs are placed in the spray nozzle sockets. The quench spray pumps are started and operated over their entire range of flow, circulating water through the spray header supply lines to the spray headers and out the temporary drain connections. This provides a full system capability test to ensure that the subsystems meet both the flow and starting time requirements. At the completion of this test, the temporary drain lines are removed, blind flanges replaced, pipe plugs removed, and the spray nozzles installed. After installation of the nozzles, a nozzle air test is conducted.

The pre-operational nozzle air tests are performed by connecting an air supply to the spray header being tested. Comparison of flow for each nozzle is made to assure that free air flow exists. Should a nozzle appear to be clogged, the nozzle is removed and cleaned. Access to the spray header is provided by a specially designed personnel staging basket.

Verification that the nozzles are unobstructed is sufficient to demonstrate that the nozzles are capable of delivering the design flow to the containment because the nozzles are subjected to a rigorous quality control dimensional check to determine if the nozzles are dimensionally correct. The nozzles have been chosen because they provide the required flow rate and droplet size under the pressure delivered by the quench and recirculation spray pumps.

The slot opening size of the quench spray pump discharge strainer is 0.187 inches. There are two types of quench spray nozzles. One type has a flow diameter of 0.3594 inches, and the other, 0.2500 inches.

With a complete system flush to remove all particulate matter prior to the installation of spray nozzles and with the use of corrosion-resistant nozzles and piping and full flow strainers, it is not credible that a significant number of nozzles could become plugged during the life of the station and, therefore, reduce the effectiveness of the subsystems. However, an inspection or

smoke or air test of the quench spray nozzles will be performed following maintenance or an activity which could result in nozzle blockage as per Technical Specifications.

The quench spray pumps are tested periodically throughout the life of the station by opening the normally closed valves on the quench spray pump recirculation line returning water to the RWST. Initiation of the subsystem will allow the pumps to operate and recirculate a quantity of water back to the tank. The discharge into the RWST will be divided into two fractions, one for the major portion of the recirculation flow and the other to pass a small quantity of water through test nozzles that provide flow characteristics similar to those used on the quench spray headers. The purpose of recirculation through the test nozzles is to ensure that there is no particulate material in the RWST and the QS subsystem that could result in plugging of the spray header nozzles. The flow rate through the test nozzles will be monitored and compared to the previously established flow rate. The presence of any particulate material that could cause plugging will readily become apparent through a reduction in flow rate through the nozzles. This surveillance will also show any degradation of the spray nozzles when exposed to RWST water during periodic quench spray pump tests. The RWST may be filtered by means of the refueling purification pumps and filters (Section 9.1.3). The weight-loaded check valves inside the containment will be tested by pressurizing the pump discharge lines with air and checking for air flow.

Monitoring of particulate matter within the RWST is performed on a quarterly basis. Therefore, buildup of particulate matter within a quench spray pump discharge strainer that could adversely affect the QS system performance is not credible. As a consequence, the strainers do not require surveillance; but if the strainer were clogged, a reduction in the total flow and an increase in pump discharge pressure would be noted on the periodic quarterly pump test.

Each QS subsystem is maintained full of water from the RWST to the containment isolation valves to ensure that water is immediately available for the quench spray pumps in the event that a CDA signal activates the QS subsystems.

6.2.2.7.2 Recirculation Spray Subsystems

The following information is HISTORICAL and is not intended or expected to be updated for the life of the plant.

Prior to station start-up, the initial full flow system test with water was performed on the RS subsystems as follows:

1. With the spray nozzle sockets plugged, permanently installed spray header drain lines running between the spray nozzle headers and the containment sump were temporarily connected.
2. Sufficient water was then added to the containment sump, surrounded by a portable dike, so that each recirculation spray pump could recirculate water up through its respective heat exchanger and the spray nozzle headers.

The following information is HISTORICAL and is not intended or expected to be updated for the life of the plant.

3. The full flow test through the shell side of each recirculation spray heat exchanger ensured that the required flow and head for effective spray nozzle operation and system operation was achieved.
4. Upon completion of the above system test, the water was drained from each recirculation spray heat exchanger, the pump, the spray nozzle headers, and the containment sump. The temporary connections on the drain lines between the headers and the sump were removed, and the spray nozzles installed.

After the flow test, the nozzle air test was conducted utilizing methods employed for testing the quench spray nozzles. This completed preoperational testing.

The clean, dry, and ready condition of the recirculation spray heat exchangers after flow testing makes further flow testing of the exchangers unnecessary, since plugging or loss of capability is not considered credible.

The outside recirculation pumps were subjected to an extended run test. After the initial 6-day run, the pumps were inspected and subsequently returned to operation for 450 hours. Pump operation and subsequent inspections were successful in demonstrating the capability of the pump.

The ORS pumps are periodically flow tested. Closing of the suction line valve and the isolation valve between the pump discharge and the containment penetration will allow the pump casing to be filled with water and the pump to recirculate water through a test line from the pump discharge to the pump casing.

Full-flow testing of the inside recirculation spray pumps is performed in accordance with the ASME Code during each refueling outage. This testing is accomplished by rotating the installed elbow spool piece to allow the water to flow through a test line back to the containment sump (see Figure 6.2-73). A portable dike is installed around the sump during flow testing. When use of the dike is required, it is secured to permanently mounted support brackets around the sump. The dike will contain approximately 5000 gallons of water, which is adequate for filling the system and properly testing the pump. A minimum water level is required in the containment sump and portable dike in order to provide the pump with a minimum NPSH. The dike is suitably sealed to the walls and floor of the containment during testing and is readily removable and stored elsewhere during normal station operation.

The throttle valve in the test line is used to vary the pump discharge pressure so that flow readings at various discharge pressures are obtained during the flow test. These points are compared to the pump operating curve, which was established from previous data and checked during pre-operational testing, to ensure that the pump is performing properly.

Following completion of the test, strict administrative procedures ensure that the dike is removed and the elbow spool piece is rotated to provide the proper flow path from the pump through the coolers and out the spray headers.

Testing of system controls is discussed in Chapter 7.

6.2.2.8 **Instrumentation Application**

Instrumentation and associated analog and logic channels employed for initiation of containment depressurization are discussed in Section 7.3. This section describes the instrumentation provided to allow monitoring of the system parameters during operation and testing.

6.2.2.8.1 Quench Spray Subsystem

1. Redundant level and temperature instrumentation is provided for the RWST. Four channels of level and two of temperature indication are provided on the main control board. Low (within 6 inches of normal level) RWST water level and high (exceeding normal by 6 inches) RWST water level alarms are also provided on the main control board; these alarms are actuated by a fifth independent level circuit. RWST temperature instrumentation provides automatic start and stop signals to the refueling water recirculation pumps. The level transmitters on the RWST are protected from freezing by redundant Category 1 safety related heat tracing.
2. Temperature and level are measured in the refueling water chemical addition tank and indicated on the main control board. Low temperature and low level alarms are provided and displayed in the main control room. Refueling water chemical addition valve positions are monitored on the main control board.
3. Flow and temperature switches are provided to initiate automatic start and stop of the refueling water refrigeration units.
4. Pressure indication and low flow alarms are provided on the main control board to monitor the discharge pressure and flow of each quench spray pump.
5. Additional local flow and pressure indication is provided to measure the quench spray pump discharge during pump testing.
6. Motor-operated valves on the discharge lines of the quench spray pumps are opened on a CDA signal. Motor-operated valves on the suction lines of the quench spray pumps are normally open and also receive a CDA signal to open. Valve positions are indicated on the main control board.

6.2.2.8.2 Recirculation Spray Subsystem

1. The discharge pressure of each containment recirculation spray pump is measured and displayed on the main control board.

2. Temperature indication is provided on the main control board for the containment sump water and the discharge from each containment recirculation spray cooler.
3. A common "Hi/Lo" level alarm is provided in the main control room for each seal head tank on the outside recirculation spray pumps, and liquid level alarms are provided for the ESF valve pit area. Redundant level indication is provided on the main control board for the containment structure sump.
4. Vibration sensors are provided for each recirculation spray pump that initiates a high vibration alarm on the main control board. Each inside recirculation spray pump is also equipped with a pump shaft speed sensor that illuminates a blue light on the main control board when the pump shaft is rotating.
5. Connections for local pressure and flow measurements are provided to measure pump discharge during testing.
6. Normally open motor-operated valves in the suction and discharge lines of each outside recirculation spray pump receive a CDA signal to ensure valve opening. Valve positions are indicated on the main control board.
7. Instrumentation for the inside and outside recirculation pumps and their associated heat exchangers are grouped together on the main control board, and color-coded to agree with system power supplies to increase the control room operator's awareness of system operations.
8. Service Water temperature is provided as a Reg. Guide 1.97 variable in the monitoring of the discharge from each containment recirculation spray cooler.

6.2.2.8.3 Casing Cooling (See also Section 6.2.2.2)

1. Casing cooling supply valves are provided on the discharge lines of the casing cooling pump. One of the two in-series discharge motor-operated valves (MOV) on each train of the Casing Cooling pump flow path (normally closed MOV that opens on a CDA signal) will automatically close at the Casing Cooling tank low-low level setpoint. This provides vortex prevention in the tank. The timing of this function is important to prevent gas transport to the outside recirculation pumps.

   The other in-series discharge MOVs (normally open MOV) will automatically close upon low recirculation flow. Loss of the Casing Cooling pump or depletion of fluid in the Casing Cooling tank will eventually result in low recirculation flow and cause these MOVs to close.
2. Temperature and level high/low alarms and indicators are provided in the main control room for the casing cooling tank.
3. Local indicators for discharge flow, discharge pressure, and suction pressure are provided for the casing cooling pumps.

4. Local temperature indicators are provided for the chiller outlet temperature and chiller recirculation pump inlet temperature.
5. Indicators and switches for the Casing Cooling discharge valves are provided in the main control room.

6.2.3 Containment Air Purification and Cleanup Systems

The systems used for containment air purification and cleanup are discussed in the following sections of this report:

- Containment ventilation systems - Section 9.4.9
- Combustible gas control in containment - Section 6.2.5
- Iodine removal by spray system and filters - Section 6.2.3

6.2.3.1 **Design Bases**

The containment depressurization systems (Section 6.2.2) are designed to reduce post accident containment pressure by condensing steam, and to absorb iodine present in the containment atmosphere with chemical spray.

In the evaluations that follow, the containment atmosphere is assumed to be well mixed, and all the drops are assumed to contain an excess of chemical reagent to react with the iodine and convert it to a practically nonvolatile form. Based on the foregoing, the rate of removal of elemental iodine from the containment atmosphere can be calculated on the basis of an exponential removal as spray passes through the containment by the following relationship:

$$C_t = C_0 \exp(-\lambda t)$$

where,

C_t is the amount of iodine in the containment at time t (Ci),

C_0 is the amount of iodine in the initial containment atmosphere (Ci),

λ is the iodine removal coefficient (LAMBDA) (sec), and

t is the time (sec)

6.2.3.1.1 Iodine Removal Coefficient Evaluation

The iodine removal coefficient (λ) can be evaluated for parameters that represent the design conditions for the quench spray subsystems. In this system, caustic (NaOH) is added from the chemical addition tank to the flow to the quench spray pumps in proportion to the total flow from the refueling water storage tank (RWST) so that the pH of the borated water in the containment sump is between 7.0 and 8.5 when the RWST and chemical addition tank are empty.

The RWST initially, at its full capacity, has a minimum concentration of 2600 ppm boron (1.486% H_3BO_3, 0.24 M, pH 4.7). The flow from one or two quench spray pumps begins to be

discharged at 73 seconds after the accident through the top spray headers, which are approximately 100 feet above the operating floor, into the containment at an initial pH of 4.7. After a 5-minute delay, caustic addition begins and the pH rises to between 8.5 and 10.5, depending on the combination of pumps operating. In this analysis, one quench spray pump is considered to be in operation with spray header flow rates from 1400 to 1600 gpm. The volume median diameter for the 1/2B40 nozzles in the quench spray header is approximately 660 microns at 40 psid. The volume median diameter for the 1HH30100 nozzles in the quench spray header is approximately 710 microns at 40 psid.

In addition to the quench spray headers, there are recirculation spray headers located an average of approximately 85 feet above the operating floor. Two recirculation spray pumps take suction from the containment sump and pump water through the recirculation spray headers, beginning at approximately 40 minutes after the accident. In the dose analysis, it is assumed the inside RS pump delivers 3000 gpm and the outside RS pump delivers 3350 gpm to the spray headers. The borated water in the sump rises from a pH of about 4.7 to a pH between 7.0 and 8.5 when the RWST is empty. The atomized spray droplets from the 1HH30100 nozzles in the recirculation spray headers have a volume median diameter of 775 microns at 20 psid and the 1/2B60 nozzles in the recirculation spray headers have a volume median diameter of 910 microns at 20 psid, as determined from the spray nozzle manufacturer's experimental data.

The iodine removal coefficients confirmed in this section are used in calculating the offsite and control room dose resulting from a LOCA. The offsite and control room doses that result from the iodine and other isotopes released are presented in Section 15.4.1.9.

The loss-of-coolant accident (LOCA) dose consequence analysis, documented in Section 15.4.1.9, assumed an effective elemental iodine removal coefficient (spray lambda) of 10 per hour for the duration of the accident. This value is consistent with the value used by the NRC in their audit calculation of the radiological consequences of the North Anna LOCA and documented in NUREG-0053 (Reference 26).

To confirm that the value of 10 per hour used for spray lambda was valid, specific data for North Anna were input into the NUREG-0800 Standard Review Plan (SRP) 6.5.2 methodology. The SRP 6.5.2 methodology used to calculate the iodine removal coefficient is described below:

$$\lambda = \frac{6K_gTF}{VD}$$

where,

K_g is the gas-phase mass-transfer coefficient,

T is the time of fall of drops, which is estimated by the ratio of the average fall height to the terminal velocity of the mass-mean drop,

F is the volume flow rate of the spray pump,

V is the containment building net free volume, and

D is the mass-mean diameter of the spray drops

The expression $\frac{6F}{VD}$ represents the rate of solution surface created per unit gas volume in the containment atmosphere.

Using North Anna specific data in the above equation, a value greater than 20 per hour was calculated for the iodine removal coefficient.

In the LOCA radiological analysis, discussed in Section 15.4.1.9, the effective particulate (also referred to as aerosol) iodine removal coefficients were calculated using NUREG/CR-5966 methodology. The results of these calculations are shown in Table 15.4-11.

6.2.3.2 **System Design**

For a description of the containment spray system, see Section 6.2.2. The effective volume of the containment covered by the quench spray is 721,000 ft^3. The volume covered by the recirculation sprays is 1,401,200 ft^3. The volume covered by both spray systems operating together is 1,606,500 ft^3.

During non-accident shutdown operations, two auxiliary building filters are provided that can be used to filter radioiodine and radioactive particulates from the containment atmosphere.

Exhaust from the following areas may be directed through these HEPA/charcoal filter assemblies and hence to the station vent stack:

1. Auxiliary building central area.
2. Auxiliary building general area.
3. Unit 1 reactor containment.
4. Unit 1 safeguard area.
5. Unit 2 reactor containment.
6. Unit 2 safeguard area.
7. Fuel building.
8. Decontamination building.

No HEPA filters are supplied downstream of the charcoal adsorbers as dusting or fines release is expected to be considerably less than 0.1% of the charcoal in the adsorbers. Assuming a fines release from the charcoal of 0.1% with a postulated adsorber iodine efficiency of 90% would result in a 0.9% increase in the release of iodine and iodides. However, the limiting fuel handling accident analysis covered in Section 15.4.5 meets with the dose limits of 10 CFR 50.67 and Regulatory Guide 1.183 without taking credit for iodine removal by charcoal adsorption or HEPA filtration. Therefore, the limiting fuel handling accident dose is not affected by consideration of released fines.

No demisters are provided upstream of the HEPA filter and charcoal adsorber since moisture in air from the above-mentioned areas will only exist as vapor during the period when exhaust is directed through these filter assemblies.

The auxiliary building charcoal adsorber units require no moisture separators, since water will exist only as vapor in the air stream during non-accident modes of operation. Under these conditions automatically controlled electric heaters powered from normal bus sources are utilized to prevent moisture accumulation on these main filtration units by limiting the relative humidity to below 70%.

During accident conditions, the equipment in the engineered safeguard areas and auxiliary building central area, where ventilation exhaust systems are powered from emergency bus sources, will give off sufficient heat to assure that the relative humidity of the air entering the filter will be below 70%. The safeguards area contains four pumps, all of which may operate during accident conditions. The auxiliary building central area exhaust system will remove heat from the charging pump cubicles, three of which may have operating pumps during accident conditions.

The iodine inventory on the charcoal adsorber bank located in the auxiliary building filters is given in Table 6.2-42 for various times after a postulated fuel handling accident based on the TID-14844 source term and analyzed using the guidance in Regulatory Guide 1.25 and the following conservative assumptions:

1. The iodine released from the spent fuel pit is instantaneously deposited on the auxiliary building filters.
2. The filters are 100% efficient and collect all of the iodine released from the fuel pit.

The HEPA/charcoal filter bank in the auxiliary building is shielded by 18 inches of concrete. The dose rates at various times after the accident at the surface of the concrete shield are given in Table 6.2-43. The current fuel handling accident radiological analysis is based on the alternate source term and the guidance of Regulatory Guide 1.183 and is discussed in Section 15.4.5.

Table 6.2-44 lists the activity of iodine isotopes on the process vent filters for various times after the start of a post-accident hydrogen purge based on assumptions given in Section 6.2.5.

No shielding is provided around these filters for the following reasons:

1. No access is required in this area of the auxiliary building immediately following an accident.
2. The containment atmosphere cleanup system utilizes redundant hydrogen recombiners and, consequently, there is little likelihood that the purge system will be used.

Table 6.2-45 provides a listing of the compliance with each position in Regulatory Guide 1.52, Revisions 1 and 2, *Design, Testing, and Maintenance Criteria for Atmosphere Cleanup System Air Filtration and Absorption Units for Light-Water-Cooled Nuclear Power Plants*, for the auxiliary building filtration system and control room filtration system.

6.2.3.3 **Design Evaluation**

The containment spray system consists of quench spray and recirculation spray which are initiated at approximately 73 and 2400 seconds, respectively, after the start of the accident.

The model used to calculate the doses at offsite locations resulting from LOCA releases is given in Section 15.4.1.9 and is in accordance with the methods described in Regulatory Guide 1.183.

The dose reduction factor is dependent on:

1. The initiation of containment chemical sprays to remove radioiodine from the containment atmosphere before it leaks into the environment.
2. The initiation of the containment sprays to reduce containment pressure and, therefore, the leakage rate of radioiodine from the containment.
3. The filtering of radioiodines prior to the release of containment atmosphere during a post-accident hydrogen purge, if it should be necessary to purge, as discussed in Section 6.2.5. There is no filtering of radioiodine released through leaks from the containment into the environment.

The iodine removal coefficients used in Section 15.4.1.9 for calculating the offsite doses resulting from an accident are confirmed in Section 6.2.3.1.

The LOCA dose analysis documented in Section 15.4.1.9 conservatively assumed a quench spray coverage of 37.6% until the start of recirculation spray. Simultaneous operation of QS and RS results in a spray coverage of 83.8%. RS operating alone has a spray coverage of 73.1%. Note that after the first six hours no containment leakage was assumed in the analysis.

The strainer modules, as well as the physical arrangement of the containment sump and the recirculation sprays and low head safety injection pumps and piping, as discussed in Section 6.2.2, provide uniform mixing of fission products collected in the containment sump water.

The entire amount of water available in the containment sump, discussed in detail in Section 6.2.2.2, is available to the recirculation spray and low head safety injection pumps.

An analysis was performed on the quench spray (QS) subsystem as part of the FSAR development.

The following figures summarize the results for a typical hot-leg DER with winter conditions obtained with the LOCTIC code (Reference 1):

Figure 6.2-74 Quench Spray Flow Rate vs. Time Minimum ESF

Figure 6.2-75 Quench Spray Flow Rate vs. Time Normal ESF Except Minimum Quench Sprays

Figure 6.2-76 Static Head in RWST vs. Time Minimum ESF

Figure 6.2-77 Static Head in RWST vs. Time Normal ESF Except Minimum Quench Sprays

Figure 6.2-78 Static Head in CAT vs. Time Minimum ESF

Figure 6.2-79 Static Head in CAT vs. Time Normal ESF Except Minimum Quench Sprays

Figure 6.2-80 Concentration of NaOH vs. Time for Quench Spray and Containment Sump Solution Minimum ESF

Figure 6.2-81 Concentration of NaOH vs. Time for Quench Spray and Containment Sump Solutions Normal ESF Except Minimum Quench Sprays

6.2.3.4 **Tests and Inspections**

The tests and inspections of the containment depressurization system are described in Section 6.2.2.7.

The weir in the RWST permits mixing of the fluids from the chemical addition tank and the RWST, as described in Section 6.2.2.2. Reynolds numbers calculated for the weir and the quench spray piping show turbulent flow ensuring complete mixture. The pressure drop in the lines between the chemical addition tank and RWST is extremely low, so that hydrostatic balance is maintained between the two tanks. Because of this hydrostatic balance, the flow rate proportion is controlled only by the volume per foot of height ratio of the two tanks.

The NaOH concentration is maintained between 12 and 13% by weight in the chemical addition tank. Once mixed, the concentration of NaOH in the chemical addition tank remains constant. However, the chemical addition tank pumps are run to circulate the solution and take samples, as required by the technical specifications, thus ensuring correct NaOH concentration and preventing freezing.

Section 6.2.2.7 describes both initial and periodic tests to verify that the QS subsystem can provide adequate flow. Only the RWST water is involved in these tests. After an accident the NaOH accounts for only 1.75% of the quench spray flow and serves to increase the specific gravity of the water by 0.3%. Thus, the difference in the hydraulic properties of the test fluid and the postaccident fluid is negligible in regard to flow test results.

The pre-operational tests included a test on Unit 1 to demonstrate that the system as built would allow the RWST and the NaOH addition tank to draw down together as designed. This test was conducted with clean water in both tanks and the levels in both tanks were monitored until the automatic switchover setpoint (RWST low level that initiates change from injection mode to recirculation mode of ECCS) was reached to confirm that the tanks draw down together as designed.

6.2.3.5 Instrumentation Application

Instrument application for the containment depressurization system is given in Section 6.2.2.8.

6.2.3.6 Materials

It is current practice to enhance the ability of the sprays to remove iodine from the containment atmosphere by the addition of either sodium thiosulfate or sodium hydroxide, although experiments indicate that the low pH borate solution from the RWST is almost as effective in removing iodine from the containment atmosphere. Sodium thiosulfate quantitatively absorbs iodine in an irreversible action, but is more susceptible to radiation and may cause problems in long-term storage. Therefore, sodium hydroxide is the preferred spray additive to enhance the iodine removal capability of the sprays. The addition of sodium hydroxide to the spray water, which subsequently mixes with the water spilled from the reactor coolant system, results in a final pH in the containment sumps of approximately 8.0. The USNRC considers this adequate to obtain credit for iodine removal. Iodine removal reduction factors of about 5 have been allowed for this type of spray system (Reference 29).

The materials of construction for the containment depressurization components are given in Table 6.2-39.

Radiolytic decomposition and corrosion products do not interfere with the operation of any engineered safety feature. Extensive experimental studies (References 30 & 31) have been made to determine the corrosion rates and the effect on the materials in the containment from the use of base borate spray solutions. The hydrogen generated by corrosion and radiolysis in the containment is kept at a safe level by the containment atmosphere cleanup system, which is described in Section 6.2.5. Copper is relatively unaffected by base borate spray solutions. Concrete wetted by the spray solution does not seem to have its strength impaired. This has been substantiated by several experiments.

The possibility of long-term stress-corrosion cracking of the stainless steel piping due to the borated water from the RWST has been investigated (Reference 31). It was found that the higher pH solutions cause little or no short- or long- term stress-corrosion cracking.

6.2.4 Containment Isolation System

6.2.4.1 Design Bases

The containment isolation system has the following design bases:

1. For pipe penetrations through the containment, it provides, during accident conditions, at least two barriers between the atmosphere outside the containment structure and
   a. The atmosphere inside the containment structure, or
   b. The fluid inside the reactor coolant pressure boundary.
2. The design pressure of all piping and connecting components forming the isolation boundary is greater than the 45-psig design pressure of the containment. Piping forming the isolation boundary is designed to Class I or II of the American Standard Code for Pressure Piping - ANSI B31.7-1969 Nuclear Power Piping.
3. Failure of a single valve or barrier does not prevent isolation.
4. Operation of the containment isolation system is automatic.
5. All isolation valves and equipment are protected from missiles and water jets originating from the reactor coolant system (RCS).
6. All remotely actuated valves and automatically operated isolation valves have their positions indicated in, and can be operated from, the main control room.

All isolation valves located outside the containment in accordance with General Design Criteria 55, 56, and 57 are located as close to the penetration as possible without limiting the service accessibility of the valves or interfering with other valves, piping, or structural members. Approximately 70% of all outside isolation valves are located within 10 feet of the penetration. The six valves not within about 20 feet of their penetration are on 3/8-inch lines and are located 50 and 60 feet from their penetration. These six valves are located at this distance to maintain separation of components, as in the leakage monitoring system, or due to the physical size of the isolation valve, such as in the sampling system.

The pressure retaining integrity of the containment pipe penetrations will be maintained under an applicable pressure, temperature, and mechanical load combination, including SSE effects. The intent of Regulatory Guide 1.29 for these penetrations is met by the load combinations and elastic stress limits specified in Table 3.8-8. The plastic pipe loads M_P and T_P, which are far greater than the actual calculated pipe seismic loads, plus pipe design pressure and temperature effects, are each sufficient to fully yield the loaded pipe across its entire cross section at the penetration. The resulting penetration assembly stresses for these loads are limited to elastic stress limits such as 3S.

As part of the issues identified in NRC GL 96-06, isolated containment penetration piping with confined fluid was reviewed for susceptibility to thermal over-pressurization following a

DBA. The linear elastic analysis criteria stipulated in the 1989 version of the ASME Boiler and Pressure Vessel Code Section III, Appendix F, was used for structural integrity evaluation. The internal pressure in piping penetrations during a design basis accident (LOCA or MSLB) was calculated by taking into account the differences in the expansion of the fluid and the pipe, the temperature increase immediately following the DBA and credit for a limited amount of circumferential strain in the pipe. The analysis established that thermally induced over-pressurization of isolated water-filled piping sections in the containment boundary could not jeopardize the ability of the accident mitigating systems to perform their safety functions and could not lead to a breach of containment integrity (Reference 45).

All containment pipe penetrations are designed, built, inspected, and tested to the requirements of B31.7-1969, Class I or II. In 1971, these requirements were incorporated into ASME Boiler and Pressure Vessel Code, Section III, for Class 2 pipes without significant alterations. The penetrations are stamped NPT, Class I, or Class II. Consequently, these pipe penetrations meet the requirements for quality Class B as required by Regulatory Position C.1 and Table 1 of Regulatory Guide 1.26.

The component cooling lines to the residual heat removal heat exchangers are designed and built to ANSI-B31.7-1969 Class III, which corresponds to the requirements of quality Class C of Regulatory Guide 1.26.

All of the sealed systems used as isolation barriers in lieu of isolation valves meet the requirements of Regulatory Guide 1.29 for Class I seismic equipment.

6.2.4.2 **System Design**

Table 6.2-37 provides information concerning every penetration that is in service, as to the type of valves that are provided, their positions under various conditions, the fluids they contain, and the systems they connect. In addition, each containment is provided with spare piping penetrations. These serve no function but are available should design modifications require additional penetrations. The spare penetrations are welded closed to prevent leakage.

All Stone & Webster-procured remotely operated valves associated with piping penetrations through the containment are listed in Table 6.2-46. The automatic trip valves listed are spring-opposed diaphragm, piston-operated, or direct-acting solenoid, which fail closed upon loss of air or electrical power.

Westinghouse-procured remotely operated valves associated with piping penetrations are listed in Table 6.2-47. The valves listed as using air for motive force fail closed upon loss of air.

All motor-operated valves listed in the Tables fail in an “as is” condition upon loss of electrical power. Motor-operated valves used for containment isolation that are allowed to be open during normal conditions, are powered from the onsite emergency power system, and in addition, are equipped with a hand wheel that allows manual operation of the valves in case of a power failure. The Containment purge and exhaust valves may only be open during shutdown

conditions. These valves would be closed on a high-high signal from a radiation monitor in the event of a fuel handling accident in the containment, but no credit is taken for closing of these valves in the limiting fuel handling accident analysis. These valves are not powered from the onsite emergency power system.

The status of the valves during normal, shutdown, and accident conditions is given in Table 6.2-37.

All containment isolation valves purchased by Stone & Webster are factory tested and inspected. A written specification defines the specific requirements for valve procurement, which include the following:

1. Welding and NDT procedure qualification.
2. Welder and NDT operator qualifications.
3. Mill test reports.
4. Dye penetrant test (as required).
5. Magnetic particle test (as required).
6. Radiography (as required).
7. Body hydrostatic test.
8. Seat and valve steam leakage test.
9. Performance test.
10. Dimensional check.
11. Cleaning.
12. Preparation for shipment.
13. Seismic qualification.

The automatic trip containment isolation valves, with the exception of the main steam isolation valves, which are discussed in Chapter 10, are air-operated, globe, butterfly, or direct-acting solenoid valves. The piston or diaphragm operators are spring opposed, so that the valve fails closed upon loss of instrument air or loss of power to the solenoid pilot, or trips closed upon receipt of a safety signal. Factory tests of these valves include tests to ensure proper stroke action and operation of accessories.

The use of Limitorque operators has been specified for all motor-operated containment isolation valves. The specific requirements for motor-operated valves include a seismic analysis of the valve and operator as a combined unit. With the exception of the main steam isolation valves, which are required to close within 5 seconds, all containment isolation valves must be capable of closing within 60 seconds after receipt of a containment isolation signal.

The basis for the 60-second limit is that no fuel cladding is expected to melt or fail until after 60 seconds following a loss-of-coolant accident (LOCA). Thus, fission product release from the core to the containment atmosphere or to other portions of the RCS could not occur until at least 1 minute after the event.

This is verified for PWRs by the FLECHT experimental results (Reference 32), which indicate that peak temperatures occur 60 seconds or more after the start of reflooding (30 seconds more after the accident) for low reflooding rates like those that might lead to clad melting.

All containment isolation valves purchased by Stone & Webster satisfy the bases described below.

The bases for the selection of containment isolation valves and valve operators purchased by Westinghouse are primarily systems requirements. For those lines that penetrate the containment and are in part of a system that is not necessary following a LOCA and whose inadvertent closure is not detrimental to plant operation or equipment integrity, Westinghouse systems criteria specify the use of “fail-closed” valves with air operators. This is based upon the fact that the energy needed for closure of this type of valve/operator is stored in a spring. The failure of the solenoid valve, electric power, pneumatic device, or the loss of air pressure does not prevent the valve from moving to the desired position. This design ensures a high degree of reliability of the valve function. For those lines that penetrate the containment and are a part of a system that is required following a LOCA, or in which an inadvertent valve closure would jeopardize safe plant operation or component integrity, Westinghouse criteria specify the use of valves with motor-type operators, designed to fail as is. For these types of valves, the failure of the motor operator on loss of electrical power to the operator does not affect the valve movement and ensures that the valve remains in its desired position. This design ensures a high degree of reliability of the valve function.

Finally, Westinghouse criteria specify the use of check valves as the inside isolation valves where applicable. Since the only motive force necessary to provide closure of this valve is a pressure difference, this design ensures a high degree of reliability in the valve function.

In addition to these design requirements, all the valves are subjected to functional tests and shell hydrostatic tests prior to installation. These design and testing requirements are specified within the equipment specifications.

The original plant design for containment isolation required valves provided by Westinghouse that are not normally in the closed position to be capable of closing within 10 seconds. Current timing for fast-acting valves of the engineered safety features is based upon the accident analyses requirements.

All lines passing through the containment penetrations enter into the auxiliary building pipe tunnel, safeguards area, main steam valve house, or cable penetration (vault) area. All of the external isolation valves for these pipes are located in one of the above areas. The temperature of

the auxiliary building pipe tunnel and safeguards area is maintained at a minimum of 50°F, which precludes the freezing of the valves and piping in those areas. The main steam valve house is also maintained at a temperature greater than freezing.

These areas are all within Seismic Class I structures and are provided with tornado missile protection as indicated in Table 3.2-1.

Figure 6.2-83 shows schematic representations of typical containment isolation valve arrangements. Although the station is designed to the General Design Criteria published in 1966, most isolation arrangements conform to Criterion 55, 56, or 57 of Appendix A, *General Design Criteria for Nuclear Power Plants* to 10 CFR 50 published in 1971. A discussion of these criteria can be found in Section 3.1.

General Design Criteria 55, 56, and 57 had not been promulgated when four penetrations, which use check valves outside the containment as isolation valves, were designed. As explained in Section 6.2.4.2 and Table 6.2-37, these penetrations constitute exceptions taken to General Design Criteria 55, 56, and 57. This situation occurs only where there is a sealed Seismic Class I system inside the containment serving as a second isolation barrier. These penetrations are considered to meet the requirements of General Design Criterion 53 (July 10, 1967), in effect at the time of design. Table 6.2-37 also indicates the isolation criterion to which the penetration conforms, or references one of the following sections, which describe isolation arrangements that differ from those listed in the criteria:

1. Reactor containment leakage monitoring lines to open taps and containment vacuum pump suction lines.

   The leakage monitoring lines to the open taps have one manual, administratively controlled valve followed by two automatic trip valves in series outside the containment. There are four of these lines utilizing four penetrations. The two automatic valves in each line shut on receipt of a containment isolation Phase A signal.

   The containment vacuum pump suction lines have two normally open automatic trip valves in series. The containment atmosphere cleanup system takes its suction from these lines upstream of the two automatic trip valves. These valves receive a containment isolation Phase A signal to close.

   The arrangements of two trip valves in series outside the containment is necessary to provide accessibility to the valves in order to ensure operation of these systems following an accident.

2. Component cooling water supply to the residual heat removal system (RHRS), the excess letdown heat exchanger and the containment air recirculation cooling coils, feedwater lines, and chemical feed lines.

The penetrations for these lines have two barriers between fluid inside the reactor coolant pressure boundary or between the containment atmosphere and the atmosphere outside the containment. These two barriers are the various heat exchangers served by the lines and the check valves outside the containment. These check valves shut under a differential pressure when the higher pressure is on the containment side of the check valve. The piping inside the containment from the penetration to the component is run so that it is protected from potential missiles generated as a result of an accident.

Each feedwater line has the following connections between the isolation valve outside the containment and the steam generator inside the containment. The isolation arrangement for each of these connections is described:

a. A 3-inch auxiliary feed line, located outside the containment, with a check valve.

b. The chemical feed line connects to the main feedwater line inside the containment. In addition to the check valve in the chemical feed line, there is a normally open manual isolation valve on each side of the containment in this line.

3. Main steam line

Each 32-inch main steam line is isolated by an automatically tripped, normally open swing-check valve installed in the direction to prevent flow out of the containment. Flow into the containment is prevented by a motor-operated nonreturn valve installed adjacent to the trip valve. These valves are located outside the containment with a sealed system inside; this arrangement conforms to General Design Criterion 57. The following lines join each main steam line between the steam generator inside the containment and the isolation valve outside the containment with the exception of the flow element loop, which is completely within the containment. The isolation arrangement for each of the lines outside containment is described below:

a. Four 3/4-inch pressure instrument lines, each with two manually operated isolation valves.

b. A 4-inch steam line to the turbine-driven auxiliary feedwater pump, isolated by one manually operated valve. This line reduces to 3 inches upstream of the isolation valve.

c. A 32-inch riser to the safety and relief valves. This line leads to five parallel safety valves and one automatically controlled power-operated relief valve. There is a manually operated isolation valve in the line to the power-operated relief valve, and the safety valves can be manually shut.

d. A 3-inch decay heat release line outside the containment contains a check valve. After the junction of the three decay heat release lines from the three main steam lines, there is a remote, manually controlled isolation valve.

e. A 3-inch warm-up bypass line around the main steam trip valve contains an automatic trip valve for isolation.

f. Several test connections, each with two normally closed isolation valves.

g. A 1½-inch condensate drain line contains automatic trip valves for isolation.

4. Residual heat removal sample lines

   These 3/8-inch lines contain direct-acting solenoid isolation valves (automatic trip) inside and outside the containment. These isolation valves are normally shut during station operation and normally opened only when the reactor is shut down and at reduced pressure and temperature. During power operation, the RHRS is isolated from the RCS by motor-operated valves. The automatic trip receives a signal from safety injection system (SIS) train A for the valve located inside the containment and from SIS train B for the valve located outside the containment.

5. Safety injection discharge lines to the RCS

   The safety injection system is operated following a LOCA to keep the reactor core covered with water (Section 6.3). The valves affecting containment isolation in the boron injection path to the RCS cold legs are therefore designed to open upon receipt of a safety injection actuation signal. All other valves (except in the low head safety injection (LHSI) header to the RCS cold legs) are normally closed and opened as necessary by the control room operator after an accident has occurred.

   The high head safety injection line to the reactor coolant system cold legs (boron injection line) is provided with two normally closed, remotely controlled, motor-operated isolation valves in parallel outside the containment and one check valve inside the containment. A separate post-accident high head recirculation header feeding the cold-leg injection branch lines is provided with one normally closed, remotely controlled, motor-operated isolation valve outside the containment and one check valve inside the containment. In addition, there is a 1-inch line connecting to the boron injection line between the penetration and the isolation valve outside the containment bypassing the boron injection tank. This line is supplied with a normally locked, closed, manually operated isolation valve.

   The high head safety injection lines to the RCS hot legs are each provided with one normally closed, remotely controlled, motor-operated valve outside the containment and one check valve in each header inside the containment.

The LHSI lines consist of three headers outside the containment supplied by both low head safety injection pumps. One LHSI header to the RCS cold legs is provided with two parallel, normally open, remotely controlled, motor-operated isolation valves outside the containment and one check valve in each branch line to the cold legs inside the containment. Two LHSI headers to the RCS hot legs are provided, each with one normally closed, remotely controlled, motor-operated isolation valve outside the containment and one check valve in each header inside the containment.

These containment isolation arrangements conform with the single-failure criteria specified in Section 6.2.4.1 and also allow the SIS to perform its design function.

6. Reactor coolant pump seal water supply

These lines are each provided with a check valve and a normally open manual isolation valve inside the containment and normally open manual isolation valve outside the containment. In addition, there is an additional check valve inside the containment that is not missile protected.

The two isolation barriers are the check valve inside the containment and the closed portion of the Chemical and Volume Control System on the discharge side of the charging pumps. The piping from the check valve inside the containment to the manual isolation valve outside the containment is designed to Class I of ANSI B31.7, and the piping from the manual valve to the charging pump discharge is designed to Class II of ANSI B31.7.

Water is pumped through these lines, through the reactor coolant pump seals, and into the RCS during normal operation and safety injection. Thus these lines remain open after receipt of a safety injection signal and the flow contributes to the SIS flow to the RCS while protecting the reactor coolant pump seals.

7. Quench spray, recirculation spray, and casing cooling pump discharge lines

The containment depressurization system operates after an accident to depressurize the containment. The valves in the lines from the quench spray and outside recirculation spray pumps are therefore designed to be opened upon receipt of a containment depressurization (high-high containment pressure) signal if they are not already open.

The quench spray pump discharge lines are provided with a check valve inside the containment and one normally closed, remotely controlled, motor-operated valve outside the containment. This motor-operated valve opens upon receipt of a containment depressurization (high-high containment pressure) signal. The two isolation barriers are provided for these penetrations by the check valve and the motor-operated valve.

The outside recirculation spray pump discharge lines are provided with a check valve inside the containment and a normally open, remotely controlled, motor-operated valve outside the containment. The two isolation barriers are provided by the check valve inside the containment and the closed system outside the containment. This closed system includes the recirculation spray pumps and their casings. The system piping conforms to Class II of ANSI B31.7, the recirculation spray pumps conform to Class II of the Nuclear Pump and Valve Code, and the recirculation spray pump casings conform to ASME Section III B.

The casing cooling pump discharge lines terminate at the suction to the outside recirculation spray pumps. These lines are provided with a check valve, one normally open MOV, and one normally closed MOV outside containment The normally closed MOV opens on a CDA signal. Containment isolation is maintained when the Casing Cooling tank drops below the low-low level setpoint and the low-low level MOV (normally closed) closes automatically. If the low-low level MOV fails to close automatically, either the low-low level or the low-flow MOV can be closed manually. The low-flow MOV (normally open) will also close automatically on low pump discharge flow as measured from dp across the recirculation path. Low recirculation flow would occur upon loss of the Casing Cooling pump or depletion of the Casing Cooling tank volume. The isolation barriers are the check valve, two motor-operated valves, and the closed outside recirculation spray pump suction piping, as discussed in Section 6.2.4.2.

8. Low head safety injection pump and outside recirculation spray pump suction lines

   Special consideration given to the low head and recirculation spray pump inlet lines, which take suction from sumps inside the containment, results in a conservative design and use of highly reliable components in a single-valve arrangement that is enclosed in a special valve pit. The major portion of the piping is buried in the reinforced concrete basemat, and only a short length of piping exists between the mat and the isolation valve. The single valve is equipped with a highly reliable remote operator. If a failure occurs in this suction line, the valve pit becomes flooded. This provides a water seal between the containment and the outside atmosphere, which prevents leakage into or out of the containment. The design of this portion of the installation is compatible with letters from the Advisory Committee on Reactor Safeguards (References 23 & 24). Provisions for detecting leaks in these suction lines are described in Sections 6.2.2 and 6.3.

   The isolation valve at the suction of the outside recirculation spray pumps is a normally open, remotely controlled, motor-operated valve. The isolation valve for each LHSI pump suction penetration is a normally closed, remotely controlled, motor-operated valve.

9. Fuel transfer tube

   A 20-inch o.d. fuel transfer tube in the fuel transfer penetration between the refueling canal inside the containment and the spent fuel pit is fitted with a blind flange inside the containment which has two o-ring seals to prevent leakage through the transfer tube during accident conditions. A manual isolation valve located on the fuel building side of the transfer tube provides a mechanism of isolation in the event of a loss of spent fuel pool level or reactor cavity level during refueling operations. The manual isolation valve also provides isolation of the fuel pool from the reactor cavity so that the blind flange can be removed for refueling operations. The manual isolation valve is not required to be leak tested.

10. Dead weight pressure calibrator

    The line to the pressurizer dead weight pressure calibrator is provided with two normally closed, administratively-controlled, manual isolation valves outside the containment. This line is not normally used during station operation.

11. Containment atmosphere cleanup system suction and discharge lines

    The containment atmosphere cleanup system is designed to remove hydrogen from the containment atmosphere and maintain the containment subatmospheric during long-term recovery from a loss-of-coolant accident.

    The hydrogen analyzer suction lines have an administratively-controlled, remotely-operated solenoid valve inside and outside of the containment. The discharge lines do not have a separate penetration, but tap into the hydrogen recombiner discharge line between the hydrogen recombiner isolation valves and the containment penetration. The hydrogen analyzer discharge lines have two administratively-controlled, remotely-operated, solenoid valves, in series, outside of the containment and there is a check valve inside of the containment on the hydrogen recombiner discharge line. All of the isolation valves are normally closed and must be manually opened, under administrative control, after an accident.

    The hydrogen recombiner suction lines tap off of the containment vacuum pump suction lines between the containment penetration and the containment vacuum pump isolation valves. There are two administratively-controlled, remotely-operated, air-operated valves, in series, outside of the containment for isolation. The hydrogen recombiner discharge lines have two administratively-controlled, remotely-operated, air-operated valves, in series, outside of the containment and a check valve inside of the containment for isolation. All of the isolation valves are normally closed and must be manually opened, under administrative control, after an accident.

Branch lines intersecting between isolation barriers consist of leakage monitoring connections that are provided with normally closed valves and caps. Leakage monitoring connections are designed to the same criteria as their respective main lines.

The reactor coolant letdown line has a branch with a normally closed relief valve (Reference Drawing 12). A temperature element is provided to monitor leakage downstream from the relief valve, and the relief valve set pressure exceeds the test pressure of the containment. Therefore, no isolation barrier is required.

When internal closed loop systems represent a barrier for containment isolation, the containment penetrations, the piping inside the containment, and the piping up to and including the isolation valves outside the containment are designed in accordance with Seismic Category I criteria.

The definition of Seismic Category I criteria is in Section 3.2.1. Table 3.2-1 lists the seismic criteria for structures, systems, and components.

The containment isolation system valves are protected from the effects of pipe whip by separation, physical barriers, and the application of pipe whip restraints.

Design-basis breaks are postulated in the high-energy piping in accordance with Section 3.6 and NRC Regulatory Guide 1.46. For the main steam and feedwater lines, whip restraints are designed that eliminate any possibility of damage to an isolation valve from pipe whip. Additional restraints or barriers are supplied, as required, to prevent damage to the isolation valves and related piping from a break in any other high-energy line.

The following containment penetrations are identified as open to the containment with a check valve inside and a single valve outside subject to active failure:

1. Quench spray.
2. Outside recirculation spray discharge.

Note, however, that there is no penetration with a check valve inside and a single valve outside subject to single active failure that is connected to non-Seismic Class I piping or components.

Leakage into the containment for those containment penetrations identified above is prevented as described below.

The suction lines to the outside recirculation spray pumps are sealed against inleakage by a head of water in the containment sump.

The outside recirculation spray system is a closed loop outside the containment and is Seismic Class I throughout so that although it is open to the containment on the inside, failure of an isolation valve does not result in inleakage to the containment.

Additional automatic valves are provided for the air ejector vent, instrument air supply line, and containment radiation monitoring return line. These valves appear in Table 6.2-37, which lists all containment isolation valves.

The check valves used in the quench and recirculation spray systems are of the same design. They are soft-seated swing check valves. Closing force is provided by external weights of lever arms located on both sides of the valves. The weights are initially set at the factory to hold the disks closed with 2-psi differential pressure in the normal flow direction. Once open, reseating is expected to occur at about 0.5 psi.

The opening and reseating pressures are adjusted by moving the external weight along the weight arm. Opening and reseating pressures are not independently adjustable.

In response to Generic Letter 96-06, *Assurance of Equipment Operability and Containment Integrity During Design Basis Accident Conditions*, several containment penetrations which are isolated during normal power operation are partially drained (Reference 40).

6.2.4.3 Design Evaluation

Containment isolation is accomplished under the following conditions:

1. Containment isolation Phase A is initiated by a safety injection actuation signal. (This signal may be actuated by a high containment pressure or any of several other signals as described in Section 7.3). During this phase all normally open trip and motor-operated containment isolation valves in lines penetrating the containment are closed except as follows:

   a. Normally open isolation valves will remain open in component cooling water lines to and from the reactor coolant pump motors and thermal barrier, the control rod drive mechanism shroud cooling coils, and the containment recirculation air cooling coils.

   b. Isolation valves in the main steam lines are normally open and remain open. If a safety injection actuation signal causes containment isolation after a steam-line break, the main steam isolation valves are shut.

   c. Normally open valves in the safety injection system lines remain open. The normally closed valves in the SIS open to allow the SIS to operate.

   d. Isolation valves in containment depressurization system lines remain as is.

   e. Instrument air supply valves to the containment remain open.

   This allows an orderly reactor shutdown without actuation of the containment depressurization system.

2. A rise in containment pressure to a pressure between the high and the high-high containment pressure setpoints (Section 7.3) initiates main steam line isolation trip. The main steam line isolation valves may already be shut, as described in 1.b. above.

3. Containment isolation Phase B is initiated by a further rise in the containment pressure to the high-high containment pressure setpoint. At this point, all normally open trip valves in lines penetrating the containment that are not required for containment depressurization or safety

injection are closed. This phase occurs simultaneously with the actuation of the containment depressurization system (Section 6.2.2).

4. If the automatic signals fail to actuate the containment isolation trip valves, isolation can be accomplished remote-manually from the main control room. The solenoid valves that operate the automatic trip valves can be actuated by a pushbutton from the main control room.

Instrumentation and adjunct control circuits associated with automatic valve closure fail safe (initiate closure) upon loss of voltage and/or control air. Circuits that control redundant automatic valves are redundant in the sense that no single failure precludes isolation.

Several spare containment pipe penetrations of various sizes are provided. All pipes in these spare penetrations are sealed at both ends by welded pipe caps. On lines less than or equal to 2-inch nominal size, socket welded pipe caps are installed on both ends of each spare penetration. On piping larger than 2-inch nominal pipe size, a butt weld cap is positioned on both ends of the penetration and held in position by using a piping sleeve welded to the cap and the pipe. All pipe caps are equipped with test plugs.

All isolation valves and equipment are protected from missiles and water jets originating from the RCS. Missile protection for isolation valves, actuators, and controls is provided by locating isolation valves in the annulus between the crane wall and the containment wall or outside the containment structure. The devices that register containment pressure are located outside the containment and are connected to the leakage monitoring tubing, which is open to the containment.

The isolation valves are, to a large extent, protected by separation from the effects of jet impingement from a main steam or feedwater line break. The main steam lines are routed around the annulus at an elevation of 329 feet and the feedwater lines at an elevation of 302 feet, with the isolation valves located in the annulus below 262 feet. Therefore, for breaks at the steam-generator nozzles and for many of the intermediate break locations, there is adequate separation from the isolation valves.

However, the main steam lines drop vertically to Elevation 285 ft. and the feedwater lines to Elevation 278 ft. to the penetrations into the valve house. For longitudinal breaks at the lower elbows or at the penetrations, a jet could conceivably impinge on isolation valves at a distance of approximately 25 feet.

Breaks have been postulated for the main steam and feedwater lines in accordance with Section 3.6. The restraints that have been designed to prevent pipe whip also limit the possible trajectories of the jet by limiting movement of the ruptured segments of pipe.

Calculation of the total jet force from a postulated rupture is based on Moody's theoretical model (References 12, 13 & 14) and Fauske's experimental data (Reference 15). It is assumed that the retarding action of the surrounding air on the jet is negligible and the total jet force is

constant at all axial locations. The jet impingement pressure on a distant object is computed by assuming that the jet steam expands conically at an angle of 20 degrees.

For normal impingement, the jet impingement force on a distant object is equal to the product of the jet impingement pressure and the intercepted jet area. If the object intercepts the jet stream with a curved or inclined surface area, then the drag force between the jet and the object is taken as the jet impingement force.

In calculating impingement from a longitudinal rupture, the break area is considered to equal the effective flow area of the pipe with a break length of two diameters.

In accordance with these assumptions, the impingement loads are calculated on the isolation valves and associated piping with shielding added as required.

Missile protection is provided in accordance with Section 3.5. With regard to a main steam or feedwater break, or any other piping break, it is accepted that no missiles are formed by a whipping pipe. The main steam and feedwater lines are restrained from whipping and direct missile formation is not postulated to occur.

The following precautions apply to all lines penetrating the containment to prevent inadvertent opening of these lines to the atmosphere outside the containment:

1. Automatic isolation valves can be opened only upon manual reset of the actuating signal or signals and remote manual operation of the individual valve.
2. Automatic isolation valves are capable of remote-manual actuation from the main control room with the limitations for opening of the valve discussed in 1 above.
3. Manual valves used as isolation valves are opened only under administrative control.

For the items above, and for flanged closures, specific administrative procedures define their positioning in the containment isolation system during normal operation, shutdown, and accident conditions.

The maintenance of subatmospheric conditions is not jeopardized by check valve inleakage. Either the line with a check valve has double isolation boundaries, such that leakage into the containment atmosphere is possible only in the event of a break inside the containment combined with a break outside the containment in the same line, which is not considered credible, or the check valve is contained in a line that has another closed isolation valve as the primary isolation boundary following an accident.

This is the case for all lines with check valves that are used as containment isolation boundaries, with the exception of recirculation and quench spray pump discharge lines, which are open to the containment atmosphere, utilize weight-loaded check valves to preclude inleakage of air.

The following steps have been taken to minimize the potential for common mode failure of the main steam isolation valves:

1. The main steam isolation valve solenoid valves are located in an environment maintained between 75°F and 105°F and are environmentally isolated from any steam piping.
2. The instrument air system is supplied by oil-free compressors. Hence, oil contamination of the solenoid valves does not occur.
3. Additional assurance of solenoid valve coil integrity is provided by a coil continuity test performed on a refueling cycle interval.
4. Solenoid valve coils supplied are for high-temperature service, giving additional margin over operating conditions.

6.2.4.4 **Tests and Inspections**

Means are provided to test periodically the functioning of the automatic isolation equipment, such as the accuracy of sensors, bi-stable setpoint, speed of response, and operability of fail-safe features. The containment isolation instrumentation testing is discussed in more detail in Chapter 7.

Type C tests are performed on isolation valves to verify their leak-tightness. A performance-based test program of isolation valves is required by Technical Specifications.

The leakage monitoring arrangement provided to periodically test the leak-tightness of each containment isolation valve consists of monitoring taps on the main line upstream and downstream of each isolation valve. To test for tightness, the main piping section upstream of each valve is pressurized with a test gas or water, and evidence of fluid leakage is checked at the downstream tap or the rate of gas pressure decay measured. When not in use, the monitoring lines are plugged at the open end and the valves closed.

In order to establish conditions that, as nearly as practical, duplicate the post-accident status, system conditions are as described in Table 6.2-48.

Valve leakage testing, where pressure is applied between two isolation valves and measured by the makeup air method, is performed on penetrations which were identified as having a history of problems during Type C testing and when a downstream leakage method of testing across each valve is not possible. This method of testing is considered conservative because the leakage observed for that penetration is the total of the leakage of the two valves as if arranged in a parallel configuration. The actual penetration leakage must be less than or equal to this observed value since both valves are arranged in a series configuration, limiting the actual penetration leakage to the lesser of the that from the two valves.

Spring-loaded or weighted check valves are tested by introducing air into the lines and monitoring the pressure at which the valves open. The system design incorporates leakage

monitoring connections to which portable air pressure and flow measuring equipment is connected. These tests are conducted at the interval identified in the Technical Specifications.

During testing, the pressures at which the weighted check valves open are recorded.

The following information is HISTORICAL and is not intended or expected to be updated for the life of the plant.

Valves of similar design are in use at Surry 1 and 2 and Beaver Valley 1 power stations.

Airlocks are tested as required by 10 CFR 50, Appendix J, Option B, as modified by approved exemption, and in accordance with the guidelines contained in Regulatory Guide 1.163, dated September 1995.

Isolation valves with resilient seals in the containment purge lines will be tested in accordance with Technical Specifications.

6.2.5 Combustible Gas Control in Containment - Containment Atmosphere Cleanup System

6.2.5.1 Design Basis

The containment atmosphere cleanup system is designed to perform the following functions:

1. Provide, in the control room, a continuous indication of hydrogen concentration in the containment atmosphere. Per Reference 46, the hydrogen analyzers must be functional, reliable, and capable of continuously measuring the concentration of hydrogen in the containment atmosphere following a significant beyond design basis accident for accident management, including emergency planning (10 CFR 50.44, RG 1.7 Revision 3). The measurement range of the hydrogen analyzers shall cover 0% to 10% hydrogen concentration.
2. Purge the containment at a controlled rate through the gaseous waste disposal system charcoal and particulate filters (Section 11.3).
3. Maintain the containment subatmospheric during long-term recovery from a LOCA.

The containment atmosphere cleanup system is a Seismic Category I system, with the exception of the backup containment purge blower. As described in Reference 43, the hydrogen recombiners and purge blowers are not credited in the design basis or accident analysis. The hydrogen recombiners continue to be used in the Emergency Operating Procedures (EOPs). The hydrogen recombiners continue to be maintained and periodically tested.

6.2.5.2 System Description

The containment atmosphere cleanup system is shown on Figure 6.2-82 and Reference Drawing 3. The system consists of two identical portable skid-mounted hydrogen recombiners,

two hydrogen analyzers, two purge blowers, and associated piping systems. The system and its components are common to both reactor units. The skid-mounted recombiners are hooked up to the reactor containments, and the system is designed to allow either recombiner to be operational on either containment in 24 hours or less after a LOCA.

The recombiner design and performance specifications are discussed in detail in the generic reports, *Thermal Hydrogen Recombiner System for Water-Cooled Reactors*, AI-75-2, Rev. 3(P), dated January 10, 1975, and *Hydrogen Recombiners for Pressurized Water Reactor Post LOCA Application*, AI-73-27, dated April 20, 1973, and Rev. 1, dated October 26, 1973. The recombiner test program is described in the report, *Thermal Recombiner Demonstration Test*, AI-72-61, dated October 15, 1972. These reports have been approved by the USNRC.

The following information is HISTORICAL and is not intended or expected to be updated for the life of the plant.

The hydrogen recombiners purchased for use at the North Anna Power Station were identical to those purchased for use at the Beaver Valley Power Station, Units 1 and 2, by Duquesne Light, Docket Nos. 50-334 and 50-412, and were similar to the units installed at the Zion Power Station by Commonwealth Edison Company, Chicago, Illinois.

The containment atmosphere cleanup system draws the containment atmosphere through either of two open lines in the containment. These lines are also used as inlets for the containment vacuum pumps (Section 6.2.6). The gases pass through the hydrogen recombiner where any hydrogen present is converted to water vapor and returned to the containment. When the system is used to purge the containment, the gas is discharged to the atmosphere through the gaseous waste disposal system.

The hydrogen recombiner system can reduce the hydrogen concentration in the inlet air from 4% to 0.5% at a flow rate of 50 standard cubic feet per minute (scfm). Each recombiner consists of a blower, an electric preheater, a reaction chamber, a cooler, instrumentation, and piping, all of which are mounted on a skid.

The positive displacement blower draws suction on the containment and discharges the air through the recombiner. The blower is enclosed in a leaktight container preventing air leakage to the atmosphere. An electric heater raises the temperature of the inlet air to a minimum of 700°F to optimize the operation of the recombiner reaction chamber. An automatic, temperature-controlled operator controls the heater to maintain the proper temperature band in the reaction chamber. As the heated containment atmosphere passes through the reaction chamber, any hydrogen present is oxidized to water vapor. The oxidation of the hydrogen occurs as a result of the elevated temperatures in the reaction chamber. The heat generated by this process heats the gas and the water vapor leaving the chamber. A 3% hydrogen concentration in the inlet gas results in an outlet temperature of about 1300°F. After leaving the reaction chamber, the gas is cooled to 150°F or less in the self-contained air cooler.

Although the recombiner system is leaktight, provisions have been made to prevent the potential of having an unmonitored release of radioactivity from the recombiner to the atmosphere, following an accident, should the sealed and tested recombiner system nevertheless leak. The air discharging from the recombiner heat exchanger is ducted via a seismically supported system to the auxiliary building central area exhaust system and is discharged through ventilation vent stack A, which is monitored for radiation release.

All portions of the hydrogen recombiner system that are exposed to containment atmosphere are designed to the ASME Boiler and Pressure Vessel Code, Section III, Class 2, through the Summer 1973 Addendum. The remainder of the piping in the containment atmosphere cleanup system is designed to the ANSI B31.7, 1969, code for nuclear power piping.

The hydrogen analyzers can provide continuous indication of hydrogen concentration, over the range of 0 to 10% concentration, within 90 minutes after safety injection. The 90-minute timeframe is based upon the functional requirements provided in RG 1.7, Revision 3. Compliance with RG 1.7 ensures that indication of hydrogen concentration in the containment atmosphere is available in a timely manner to support the Emergency Plan (and related procedures) and related activities such as guidance for the severe accident management plan. In addition, the hydrogen analyzers are specified to the requirements of IEEE 323-1974, IEEE 344-1975, IEEE 334-1974, and IEEE 383-1974.

A transfer switch with control circuitry provides for the capability of Unit 1 to utilize both analyzers or for Unit 2 to utilize both analyzers. The same circuitry allows for the operation of Unit 1 containment isolation valves associated with the hydrogen analyzers, when Unit 1 is utilizing the analyzers, and for the operation of Unit 2 containment isolation valves associated with the hydrogen analyzers, when Unit 2 is utilizing the analyzers.

The Unit 1 hydrogen analyzer receives a transferable power supply from the “H” train of the two units’ emergency power, and the Unit 2 hydrogen analyzer receives a transferable power supply from the “J” train of the two units’ emergency power. This ensures redundancy of each unit. Each hydrogen analyzer is supplied with a remote control panel which is located in the instrument rack room in a seismically-mounted instrument rack. Functional controls for accident monitoring are remotely located in the control room on the post-accident monitoring and control panels for the two units. In addition, each analyzer is provided with an alarm located in the control room for trouble/high hydrogen content. Each hydrogen analyzer requires a supply of oxygen gas for recombining and a supply of hydrogen gas for calibrating.

A permanently installed, 50-scfm, positive-displacement, containment purge blower is installed in parallel with the containment vacuum pumps for each unit. This blower can draw air from the containment after a LOCA and discharge it to the gaseous waste disposal system. It can operate in parallel with the hydrogen recombiner system blowers when the containment is to be purged, ensuring that a failure of neither the permanently installed blower nor the recombiner system will leave a containment structure without purge capability.

The containment isolation valves for the hydrogen analyzers are direct acting solenoid valves, powered from vital electrical buses, which are backed up by the battery. The containment isolation valves for the hydrogen recombiners are air-operated valves also powered from vital power supplies which are backed up by the battery. A Seismic Category I pneumatic supply to these valves is provided by two redundant high-pressure nitrogen bottles located in the recombiner vault area within a steel structure. The valves fail shut on loss of electric power. Control switches for these valves are provided in the control room from the post-accident monitoring and control (PAMC) panel.

The hydrogen analyzer and hydrogen recombiner have a common containment penetration with the containment vacuum pumps; however, each has its own dedicated set of containment isolation valves.

As discussed in Section 6.2.5.1, containment purging is available, however, it is unlikely that the purge mode of post-accident hydrogen control would be used to control hydrogen. Purging is not credited in the analyses. The plant emergency procedures use the 100% capacity, redundant, post-accident hydrogen recombiners.

The internal design of the containment structure allows air to circulate freely. All cubicles and most compartments within the containment are provided with openings near the top as well as openings in the floor to allow air circulation. Convective mixing in conjunction with containment spray assures a uniform mixture of hydrogen in the containment.

Containment system experiment tests (Knudsen and Hilliard 1969 (Reference 42); Hilliard et al. 1970 (Reference 27)) have verified that adequate mixing of the containment atmosphere is achieved by the CSS.

6.2.5.3 Testing and Inspections

The system is tested while directly connected to the containment atmosphere. Containment isolation valves are operated as necessary under administrative control. Normal design function of the various components, together with a satisfactory temperature rise through the recombiner, indicate proper system performance. The hydrogen analyzer is calibrated and tested at regular intervals in accordance with the manufacturer’s instructions.

6.2.5.4 Instrumentation Application

All instrumentation and controls associated with the hydrogen recombiner systems are on the local control panel attached to the skids. The inlet and outlet temperatures of the heater and reaction chamber and the outlet pressure of the blower are indicated on the control panel. Start and stop controls for the blower and electric heaters are located on the panel. The hydrogen analyzer instrumentation and controls are located on the post-accident monitoring and control panel in the control room. The containment purge blowers are locally controlled. The suction pressure of the purge blower is containment pressure and is monitored by the leakage monitoring

system (Section 6.2.7). All electrical equipment in the containment atmosphere cleanup system is powered from the emergency buses to ensure power availability after a LOCA.

6.2.6 Containment Vacuum System

The containment vacuum system is used to obtain the initial subatmospheric pressure in the containment and to maintain that pressure during normal unit operation.

The system consists of a steam jet ejector, two vacuum pumps with coolers and moisture separators, and the required piping, valves, and instrumentation.

6.2.6.1 **Design Basis**

Prior to unit operation, the containment pressure is at atmospheric pressure. During the reactor coolant system (RCS) heatup, the containment pressure is reduced with a steam ejector.

A subatmospheric containment pressure is maintained whenever the average reactor coolant temperature is above 200°F. The vacuum system maintains the containment subatmospheric. The air pumped out is metered to provide a constant indication of containment system integrity. After reactor shutdown and RCS depressurization and prior to refueling or extended maintenance, the containment pressure is returned to atmospheric.

Thus, the containment vacuum system is designed to perform two functions:

1. Evacuation of the containment to subatmospheric pressure.
2. Maintenance of the subatmospheric pressure during normal operation.

6.2.6.2 **System Design**

The containment vacuum system is shown on Figure 6.2-84. Pump control and system instrumentation are discussed in Section 6.2.6.5.

The steam ejector removes air from the containment to create the initial vacuum prior to unit operation, using 150-psig steam provided by the auxiliary steam system, as discussed in Section 10.4.1. The steam ejector is sized to reduce the containment pressure from atmospheric pressure to 9.5 psia in about 4 hours.

Two mechanical vacuum pumps, each with a capacity of about 40 scfm at 8.3 psia and 90°F, are provided. Each pump has more than 100% of the required capacity. The pumps are powered from the emergency buses and discharge through the charcoal filters of the gaseous waste disposal system to the process vent.

Table 6.2-49 gives design data for the containment vacuum system.

6.2.6.3 **Design Evaluation**

The steam ejector, which is used only for the initial containment pressure reduction, and the vacuum pumps, which are used only during normal plant operation, are not considered part of the engineered safety features (ESF).

Each of the mechanical vacuum pumps is capable of removing containment inleakage and maintaining the required vacuum. The pumps are rated for intermittent or continuous operation. On a continuous operation basis, each has the capacity for removing inleakage at a rate of 40 scfm.

The containment is designed and demonstrated to have a leak rate when isolated not exceeding 0.1% of the contained volume per day at the design pressure of 45 psig. Conservatively assuming the same leak rate at near atmospheric pressures, the inleakage rate would be approximately 1.3 scfm, or less than 5% of the design capacity of a single vacuum pump. However, air lock entries, system testing, and other controlled operations introduce air into the containment during normal plant operation. The vacuum pumps are sized to accommodate this additional introduction of air, without limiting access to the containment during operation.

Should an accident occur, operation of the containment vacuum system is not required; however, the effect of long-term inleakage after the accident is a design consideration. Ultimately, air inleakage could result in the containment pressure increasing to atmospheric, with barometric fluctuations possibly causing the pressure to be slightly above atmospheric. The containment atmosphere cleanup system (Section 6.2.5) could then be used to maintain the containment at a subatmospheric pressure.

The establishment of subatmospheric pressure in the containment is governed by administrative procedures and is closely supervised by personnel responsible for unit start-up. Pressure indicators are located in the control room to provide the operator with continuous indication of the containment pressure. This close supervision and monitoring ensure that the normal operating pressure is not reduced below that permitted by the technical specifications. In the unlikely event containment pressure is reduced below the value defined by the technical specifications, a low-pressure alarm is annunciated in the main control room, notifying the operator that the low-pressure condition exists.

The vacuum pumps have a relatively small capacity when compared to the containment free volume. It would require a vacuum alarm system failure and uninterrupted operation of one vacuum pump for approximately 4 days to result in a 1-psi decrease in containment pressure.

The steam ejectors are valved off during normal operation by administrative control to preclude the possibility of excessive depressurization.

A containment vacuum breaker system is not necessary because it is not possible to exceed the vacuum loading on the containment structure by any credible means. The absence of a

vacuum breaker also eliminates a possible leakage path during the pressurized phase after a postulated LOCA and once the containment is depressurized after a postulated LOCA.

The minimum credible pressure that can be attained is caused by inadvertent operation of the quench spray system. Analysis of this accident with starting conditions of 10.0 psia (which is the lowest permissible air partial pressure allowable by the Technical Specifications minus 0.3 psia uncertainty) and 116.5°F (the maximum allowable bulk containment air temperature plus 1.5°F uncertainty) and assuming that the containment is instantaneously cooled to a low quench spray temperature (32°F) yields a conservatively low internal pressure (maximum external differential pressure).

Table 6.2-50 presents the initial conditions, method, and results of the analysis. The minimum total pressure possible in the containment from this worst case analysis is 8.62 psia.

Different portions of the containment liner can withstand different minimum pressures on the inside of the liner, as follows:

1. The shell and dome plate liners are capable of withstanding an internal pressure as low as 3 psia, which is considerably less than the calculated 8.62 psia minimum containment pressure.
2. That portion of the bottom mat liner that is covered by concrete (i.e., everywhere but the sump) can withstand an internal pressure of 5.5 psia, which is also less than the calculated 8.62 psia minimum containment pressure.
3. The bottom mat liner where exposed (i.e., the sump) due to its configuration, is capable of withstanding an internal pressure as low as 5.5 psia. This is less than the calculated minimum pressure of 8.62 psia.

6.2.6.4 **Testing and Inspections**

The steam ejector and the vacuum pumps are not considered part of the ESF. The steam air ejector is a simple mechanical device; therefore, periodic tests are not required. The mechanical vacuum pumps are operated during the initial containment leakage rate test to demonstrate their capacity to remove inleakage. During normal unit operation, they are alternated in service periodically, so their performance status is currently known. The system is designed to permit inspection and repair of one vacuum pump while the other vacuum pump is operating.

6.2.6.5 **Instrumentation Application**

The containment vacuum pumps and vacuum ejectors are operated manually as required. The following instrumentation is available in the main control room:

1. Selector switch with indicator lights for both vacuum pumps.
2. Hand setpoint station common for each containment vacuum control channel.

3. Alarms for containment partial air pressure high-high, high, low and vacuum pumps running, and low.
4. Flow indicator and flow integrator for the vacuum pump common discharge.
5. Containment absolute pressure indicators.
6. Containment air temperature indicators.

Design details and logic of the instrumentation are discussed in Chapter 7.

6.2.7 Leakage Monitoring System

The leakage monitoring system was used for pre-operational leak testing of the containment and also designed to periodically measure air leakage into the containment during normal operation. During pre-operational testing it was determined that the sealed pressure system portion of the containment leakage monitoring system did not function properly. The sealed pressure system portion of the leakage monitoring system is not used during normal operation.

The containment pressure and air temperature instrumentation in the leakage monitoring system is used during normal operation. In addition to the containment pressure and air temperature instrumentation, the containment air moisture and air flow instrumentation in the leakage monitoring system may be used for periodic leak testing of the containment in accordance with 10 CFR 50, Appendix J. Containment leakage testing is accomplished by using the absolute method and verified by the superimposed leak method.

6.2.7.1 Design Basis

The system provides for measurement of a containment leak rate consistent with the containment design objective of less than 0.1 weight percent of the containment free volume in 24 hours at accident pressure (P_a).

The system is designed and operated in accordance with ANS 7.60, the proposed standard for leakage-rate testing of containment structures, dated April 29, 1970, and meets the requirements for measurement of leakage in accordance with ANSI N45.4-1972, *Leakage-Rate Testing of Containment Structures for Nuclear Reactors*. Tests will meet the acceptance criteria described in Technical Specifications.

6.2.7.2 System Design

The permanently installed leakage monitoring system is shown on Figure 6.2-85.

The consequence of leakage into the containment during operation is an increase in the air mass. Limiting conditions of operation with respect to this air mass are specified in the Technical Specifications.

The leakage monitoring system containment pressure transmitters are used during normal operation to alert the operator to a change in containment pressure, which could indicate RCPB leakage or excessive containment inleakage.

Containment leakage is measured during Type A testing by the absolute method and verified by a superimposed leak verification test.

The absolute quartz manometers, that may be installed in the system during Type A testing, are sensitive to pressure changes of 0.001 psi. Temperature detectors are sensitive to changes of ±0.1°F, and dewpoint sensors are sensitive to dewpoint changes of ±0.5°F.

Combining the above sensitivities with inaccuracies involved in converting the electrical signals to numerical values in the computer yields an estimate of the accuracy of a single calculation of mass of air in the containment. Applying linear regression and considering previous nuclear power plant experience it can be shown that the containment structure leak rate can be measured to ±0.01% per day with 95% confidence.

The absolute method of leakage rate determination is a direct application of the ideal gas law. Containment pressure, dewpoint, and temperature measurements are taken throughout the test and the data are fitted by the method of linear regression to a linear equation relating air mass inside the containment structure to time. The slope of the line representing this equation is the containment structure leakage rate. Mean containment temperature is obtained through the use of resistance temperature detectors located throughout the interior of the containment structure. A precision manometer or pressure sensor is temporarily installed to measure the containment pressure. This installed manometer samples pressure from four open-ended pipes inside the containment structure. Dewpoint or humidity sensors are utilized to provide data to correct leakage rates for changes in containment humidity. The test is considered acceptable when the magnitude and trend of the data establishes allowable leakage rates. Methods for trending the data are described in ANSI/ANS-56.8-1994.

The superimposed leak method consists of establishing a measured leak from the containment and comparing the sum of the previously determined leakage and the superimposed leakage to the verification test calculated leakage using the absolute method. The superimposed leak test, using the gas meter measurements, serves to verify the leakage monitoring results of the absolute method. Details of the tests performed on the containment structure are discussed in Section 6.2.1.4.

6.2.7.3 **Design Evaluation**

Periodic leak rate measurements can be performed during Type A testing by the absolute method and verified by the superimposed leak method.

As part of the containment isolation system (Section 6.2.4), each leakage monitoring line penetrating the containment structure is provided with two automatic trip valves which are normally closed. Thus, in the event of an accident, no leakage to the environment occurs. The leakage monitoring system tubing, as an extension of the containment, is designed to withstand the pressure and temperature expected during an accident.

The Regulatory Guide 1.97 instrumentation associated with the leakage monitoring system is required to operate after an accident. This instrumentation will detect a rupture of a high-temperature fluid line within the containment.

Leakage from a rupture of the nitrogen supply lines or inleakage through the radiation monitoring system due to a break in the external piping may also result in a pressure increase within the containment. Such events are distinguished from a steam-line break in that they do not increase the containment humidity.

Instrument air lines within the containment are normally supplied by outside instrument air through trip valves. The containment instrument air compressors are normally kept in a standby mode. If instrument air system leakage within containment becomes excessive, the outside instrument air supply can be isolated to prevent such leakage from resulting in a containment pressure increase. The instrument air system is discussed in Section 9.3.1.

6.2.7.4 **Tests and Inspections**

All required instruments, including resistance thermometers and dewpoint sensors, are calibrated before the integrated leak rate tests.

The leakage monitoring system instruments are tested and calibrated on a regular basis during normal station operation.

The inspection methods that could be used to specifically identify the source of leakage into the containment are those specified by ANSI/ANS-56.8-1994, *Containment System Leakage Testing Requirements*, for Type B and Type C leak-test methods. In addition, soap bubble testing and visual observation could be used.

6.2.7.5 **Instrumentation Applications**

The permanently installed leakage monitoring system instrumentation, that may be used during Type A testing, is described in preceding Sections 6.2.7.1 and 6.2.7.2, and Table 6.2-51.

The pressure transmitters shown on Figure 6.2-85 are part of the engineered safety features and the containment isolation system and are discussed in Chapter 7.

The containment average temperature detector locations and method for calculating the weighted average containment temperature are shown on Table 6.2-52 for Unit 1 and Table 6.2-53 for Unit 2.

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6.2 REFERENCE DRAWINGS

The list of Station Drawings below is provided for information only. The referenced drawings are not part of the UFSAR. This is not intended to be a complete listing of all Station Drawings referenced from this section of the UFSAR. The contents of Station Drawings are controlled by station procedure.

	Drawing Number	Description
1.	11715-FM-091A	Flow/Valve Operating Numbers Diagram: Containment Quench and Recirculation Spray Subsystem, Unit 1
	12050-FM-091A	Flow/Valve Operating Numbers Diagram: Containment Quench and Recirculation Spray Subsystem, Unit 2
2.	11715-FM-091B	Flow/Valve Operating Numbers Diagram: Containment Quench and Recirculation Spray Subsystem, Unit 1
	12050-FM-091B	Flow/Valve Operating Numbers Diagram: Containment Quench and Recirculation Spray Subsystem, Unit 2
3.	11715-FM-106A	Flow/Valve Operating Numbers Diagram: Containment Atmosphere Cleanup System, Unit 1
4.	11715-FM-1A	Machine Location: Reactor Containment, Plan, Elevation 291'- 10", Unit 1
	12050-FM-1A	Machine Location: Reactor Containment, Plan, Elevation 291'- 10", Unit 2
5.	11715-FM-1B	Machine Location: Reactor Containment, Plan, Elevation 262'- 10", Unit 1
	12050-FM-1B	Machine Location: Reactor Containment, Plan, Elevation 262'- 10", Unit 2
6.	11715-FM-1C	Machine Location: Reactor Containment, Plan, Elevation 241'- 0", Unit 1
	12050-FM-1C	Machine Location: Reactor Containment, Plan, Elevation 241'- 0", Unit 2
7.	11715-FM-1D	Machine Location: Reactor Containment, Plan, Elevation 216'- 11", Unit 1
	12050-FM-1D	Machine Location: Reactor Containment, Plan, Elevation 216'- 11", Unit 2
8.	11715-FM-1E	Machine Location: Reactor Containment, Sections 1-1 & 5-5, Unit 1
	12050-FM-1E	Machine Location: Reactor Containment; Sections 1-1, 7-7, 8-8, & 9-9; Unit 2

9.	11715-FM-1F	Machine Location: Reactor Containment; Sections 2-2, 6-6, 7-7, & 10-10; Unit 1
	12050-FM-1F	Machine Location: Reactor Containment; Sections 2-2, 5-5, & 6-6; Unit 2
10.	11715-FM-1G	Machine Location: Reactor Containment, Sections 3-3 & 4-4, Unit 1
	12050-FM-1G	Machine Location: Reactor Containment, Sections 3-3 & 4-4, Unit 2
11.	11715-FM-088A	Flow/Valve Operating Numbers Diagram: Fuel Pit Cooling and Refueling Purification System, Unit 1
12.	11715-FM-095C	Flow/Valve Operating Numbers Diagram: Chemical and Volume Control System, Unit 1
	12050-FM-095C	Flow/Valve Operating Numbers Diagram: Chemical and Volume Control System, Unit 2

Table 6.2-1
REACTOR CONTAINMENT DESIGN CONDITIONS

Initial Conditions	Value[a]
Air Partial Pressure	b
Temperature	86-115°F
Relative Humidity	0-100%
RWST Temperature	40-50°F
Service Water Temperature	35-95°F

a. Instrumentation uncertainties for these parameters have been included in the safety analysis.
b. The primary containment air partial pressure is a function of the service water temperature. Permissible air partial pressure versus service water temperature is specified in the Technical Specifications.

Table 6.2-2
KEY INPUT PARAMETERS FOR THE GOTHIC CONTAINMENT ANALYSES

Parameter	Value
Maximum Core Power (100.37% x 2940 rated thermal power), MWt	2951
TS Containment Air Partial Pressure, psia	TS Figure 3.6.4-1
Containment Air Partial Pressure Uncertainty, psi	± 0.250
Containment Temperature, °F (includes 1.5ºF uncertainty)	84.5–116.5
Containment Relative Humidity, %	0–100
SW Temperature, °F (includes 3.0ºF uncertainty)	32–98
RWST Temperature, °F (includes 2.0ºF uncertainty)[a]	32–52
Accumulator Pressure, psia	590–705
Accumulator Temperature, °F	84.5–121.5
Accumulator Water Volume, ft^3 (includes uncertainty)	1007.3–1042.8
Accumulator Nitrogen Volume, ft^3	407.2–442.7
Minimum Service Water Flow Rate with 2% RSHX tube plugging, gpm	4410[b]
Maximum Service Water Flow Rate with 0% RSHX tube plugging, gpm	9935
LHSI Injection Mode Flow Rate (Single-Train), gpm	3066-4282
Maximum LHSI Recirculation Mode Flow Rate (Single-Train), gpm	4100
High-Head Safety Injection (HHSI) Injection Mode Flow Rate (Single-Train), gpm	588–660
ORS Pump Flow Rate, gpm	3350–3750
IRS Pump Flow Rate, gpm	3050–3425
Minimum Casing Cooling Flow Rate to ORS Pump Suction, gpm	700
Casing Cooling Tank Available Volume, gallons	90,000
Casing Cooling Tank Maximum Temperature, ºF (includes 3.0ºF uncertainty)	53

a. Minimum RWST temperature of 32ºF is assumed for evaluation of the inadvertent QS actuation event, but the GOTHIC analyses use 38ºF. Normal operating range for RWST temperature is 40-50ºF.

b. The minimum SW flow rate per RSHX is 4500 gpm with no tube plugging. The flow rate is reduced to 4410 gpm to account for 2% tube plugging.

c. The QS flow rate varies with the differential pressure between the containment and RWST water level.

d. The strainer head loss (clean and debris-laden) for the RS and LHSI strainers are not included in the GOTHIC model. Instead, the strainer head loss is shown to be less than the reported NPSH margin from GOTHIC.

e. The containment net free volume of 1,825,000 ft^3 is calculated by subtracting the calculated equipment volumes and the interior concrete volumes from the gross volume.
The gross volume is calculated to be 2,087,195 ft^3. Construction tolerances on the liner allow a maximum deviation of ±3 in. from the nominal dimensions. Actual field data from Surry Units 1 and 2 and North Anna Units 1 and 2 indicate that the average deviation is very close to zero (i.e., the positive deviations approximate the negative deviations). Nevertheless, for conservatism, it is assumed that the actual dimensions are smaller than the nominal by 1 in. on the radius.
The containment size is:
126 ft. 0 in. i.d.
125 ft. 1 in. basemat to bend line.
The 1-in. negative deviation used leads to a 6199 ft^3 deviation in the total gross volume.
The interior concrete volume is 181,592 ft^3, and the equipment volume is 52,837 ft^3, for a total occupied volume of 234,429 ft^3. To compensate for possible subsequent layout changes, the occupied volume has been increased by 5% (11,721 ft^3) for a total of 246,150 ft^3. Therefore, there is a total conservatism in the volume of 6199 plus 11,721 for a total of 17,920 ft^3.

Table 6.2-2 (continued)
KEY INPUT PARAMETERS FOR THE GOTHIC CONTAINMENT ANALYSES

Parameter	Value
Maximum Casing Cooling Delivery Delay from CDA signal, sec	55
QS Flow Rate, gpm	Variable[c]
QS Bleed Flow Rate to IRS Pump Suction, gpm	150
QS Spray Delivery Delay from CDA signal, sec	56–70
LHSI Pump Suction Friction Loss at maximum 1-pump flow[d], ft	8.8
ORS Pump Suction Friction Loss at maximum flow[d], ft	5.1
IRS Pump Suction Friction Loss at maximum flow[d], ft	0.42
CDA High High Containment Pressure, psia	30
RWST WR Level for RS Pump Start (60% ± 2.5% uncertainty), %	57.5–62.5
ORS Pump Start Time Delay after 60% RWST level + CDA, seconds (0 or 10 seconds for ramp to full flow depending on which is conservative)	0–10
ORS Piping Fill Time, seconds	46–62.8
IRS Pump Start Time Delay after 60% RWST level + CDA, seconds (± 12 second timer uncertainty + 0 or -10 seconds for ramp to full flow, depending on which is conservative)	108–142
IRS Piping Fill Time, seconds	52–55.4
RWST WR Level Setpoint for RMT (16% Plant Setpoint ± 2.5% uncertainty), %	13.5–18.5
Time to complete RMT function, seconds	95–210
Minimum RWST volume at accident initiation, gallons	462,640
Minimum containment free volume, ft^3	1,825,000[e]
Maximum containment free volume for NPSHa Analysis, ft^3	1,916,000

a. Minimum RWST temperature of 32ºF is assumed for evaluation of the inadvertent QS actuation event, but the GOTHIC analyses use 38ºF. Normal operating range for RWST temperature is 40-50ºF.

b. The minimum SW flow rate per RSHX is 4500 gpm with no tube plugging. The flow rate is reduced to 4410 gpm to account for 2% tube plugging.

c. The QS flow rate varies with the differential pressure between the containment and RWST water level.

d. The strainer head loss (clean and debris-laden) for the RS and LHSI strainers are not included in the GOTHIC model. Instead, the strainer head loss is shown to be less than the reported NPSH margin from GOTHIC.

e. The containment net free volume of 1,825,000 ft^3 is calculated by subtracting the calculated equipment volumes and the interior concrete volumes from the gross volume.
The gross volume is calculated to be 2,087,195 ft^3. Construction tolerances on the liner allow a maximum deviation of ±3 in. from the nominal dimensions. Actual field data from Surry Units 1 and 2 and North Anna Units 1 and 2 indicate that the average deviation is very close to zero (i.e., the positive deviations approximate the negative deviations). Nevertheless, for conservatism, it is assumed that the actual dimensions are smaller than the nominal by 1 in. on the radius.
The containment size is:
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125 ft. 1 in. basemat to bend line.
The 1-in. negative deviation used leads to a 6199 ft^3 deviation in the total gross volume.
The interior concrete volume is 181,592 ft^3, and the equipment volume is 52,837 ft^3, for a total occupied volume of 234,429 ft^3. To compensate for possible subsequent layout changes, the occupied volume has been increased by 5% (11,721 ft^3) for a total of 246,150 ft^3. Therefore, there is a total conservatism in the volume of 6199 plus 11,721 for a total of 17,920 ft^3.

Table 6.2-3
PHYSICAL CONSTANTS FOR CONTAINMENT STRUCTURE AND REACTOR COOLANT SYSTEM MATERIALS

Material	Temperature °F	Density lbm/ft^3	Thermal Conductivity Btu/hr-ft-°F	Specific Heat Btu/lbm-°F
Carbon steel	70	490	27	0.10
Stainless steel	70	501	9.4	0.12
Concrete	75	142	1.0	0.156
Paint	75	110	0.125	0.10

Table 6.2-4
GOTHIC CONTAINMENT EVALUATION PARAMETERS - PASSIVE HEAT SINK INVENTORY[a]

Thermal Conductor No.	Description	Face Surface Area[b] (ft^2)	Material	Material Thickness
1.	Interior walls, floors, etc.	7741	Painted concrete	0.5 ft
2.	Interior walls, floors, etc.	57,435	Painted concrete	1.0 ft
3.	Interior walls, floors, etc.	51,064	Painted concrete	1.5 ft
4.	Interior walls, floors, etc.	10,691	Painted concrete	2.0 ft
5.	Interior walls, floors, etc.	8674	Painted concrete	2.25 ft
6.	Interior walls, floors, etc.	3354	Painted concrete	3.0 ft
7.	Containment structure wall below grade	21,397	Painted carbon steel	0.375 in.
			Concrete	4.5 ft.
8.	Containment structure wall above grade	28,090	Painted carbon steel	0.375 in.
			Concrete	4.5 ft.
9.	Containment dome	24,925	Painted carbon steel	0.500 in.
			Concrete	2.5 ft.
10.	Containment mat and subfloor	11,757	Painted concrete	2.2 ft.
			Carbon steel	0.25 in.
			Concrete	10.00 ft.

a. The structural heat sinks that were used for the ECCS backpressure analysis are provided in Table 15.4-3. All heat sinks are slabs exposed on one face and insulated on the other face.
There is additional conservatism in the calculation of heat sink area. For other than flat slab geometries (for example, a curved wall exposed on both sides), the area used in the containment pressure calculation is twice the inner or smaller curved surface. Also, for the slab-type geometries, only the largest exposed areas are considered; the ends of the slab are not included.
Conservatism in containment heat sink mass is mainly in two parts. First, since a conservatively small area has been used, such as in a curved wall, the calculated mass is smaller than the actual value. Second, the value for density of concrete sinks does not consider embedded steel. This yields a conservatively small value for the mass in the sink.

b. Thermal Conductor Nos. 1-6 areas have been reduced 5% to bound floor plug removal.
*GOTHIC model thermal conductors 21-28 are assigned to the RCS model and are not classified as containment passive heat sinks.

Table 6.2-4 (continued)
GOTHIC CONTAINMENT EVALUATION PARAMETERS - PASSIVE HEAT SINK INVENTORY[a]

Thermal Conductor No.	Description	Face Surface Area[b] (ft^2)	Material	Material Thickness
11.	Stainless Steel 0.3"–0.7"	9378	Stainless steel	0.360 in.
12.	Stainless Steel > 0.7"	330	Stainless steel	1.490 in.
13.	Carbon Steel < 0.3"	74,920	Painted carbon steel	0.225 in.
14.	Carbon Steel 0.3"–0.6"	12,304	Painted carbon steel	0.333 in.
15.	Carbon Steel 0.6"–1.0"	1413	Painted carbon steel	0.921 in.
16.	Carbon Steel 1.0"–2.0"	17,749	Painted carbon steel	1.430 in.
17.	Carbon Steel > 2.0"	1969	Painted carbon steel	2.239 in.
18.	Galvanized Metal	95,667	Painted carbon steel	0.069 in.
19.	Containment Liner Temperature Response	1	Painted carbon steel Concrete	0.375 in. 4.5 ft.
20.	EQ Conductor	1	Painted carbon steel	1.0 in.
29*	RC Pipe Break Restraints and Uninsulated SG supports	21,054	Painted carbon steel	1.143 in.
30.	Sump Strainer	17,190	Stainless steel	0.114 in.

a. The structural heat sinks that were used for the ECCS backpressure analysis are provided in Table 15.4-3. All heat sinks are slabs exposed on one face and insulated on the other face.
There is additional conservatism in the calculation of heat sink area. For other than flat slab geometries (for example, a curved wall exposed on both sides), the area used in the containment pressure calculation is twice the inner or smaller curved surface. Also, for the slab-type geometries, only the largest exposed areas are considered; the ends of the slab are not included.
Conservatism in containment heat sink mass is mainly in two parts. First, since a conservatively small area has been used, such as in a curved wall, the calculated mass is smaller than the actual value. Second, the value for density of concrete sinks does not consider embedded steel. This yields a conservatively small value for the mass in the sink.

b. Thermal Conductor Nos. 1-6 areas have been reduced 5% to bound floor plug removal.
*GOTHIC model thermal conductors 21-28 are assigned to the RCS model and are not classified as containment passive heat sinks.

Table 6.2-5
SUBCOMPARTMENT LIMITING BREAKS

Subcompartment	Limiting Break	Tabulated Release Rates
Reactor cavity	150 in^2 cold leg-LDR	Table 6.2-6
Pressurizer cubicle	Surge line DER Spray line DER	Table 6.2-7 Table 6.2-8
Steam generator compartment	Hot leg single-ended split	Table 6.2-9

Table 6.2-6
SATAN V MASS AND ENERGY RELEASE RATES 150 IN^2
COLD LEG LIMITED DISPLACEMENT RUPTURE

Time (sec)	Mass Flow Rate (10^3 lbm/sec)	Energy Flow Rate (10^6 Btu/sec)
0.0	0.0	0.0
0.0025	12.48	7.004
0.0050	16.64	9.338
0.0075	19.18	10.76
0.0100	21.40	12.00
0.0201	25.46	14.26
0.0250	24.63	13.77
0.0326	25.52	14.25
0.0450	27.39	15.30
0.0551	26.84	14.98
0.0700	25.40	14.16
0.0775	25.94	14.47
0.0925	23.79	13.24
0.1100	22.97	12.78
0.1252	23.99	13.36
0.1401	23.18	12.90
0.1550	22.77	12.67
0.1701	22.14	12.32
0.1901	22.61	12.58
0.2125	23.10	12.86
0.2375	22.67	12.61
0.2501	22.92	12.76
0.2751	22.34	12.43
0.3127	22.91	12.75
0.3251	22.70	12.63
0.3502	22.76	12.67
0.3875	22.66	12.61
0.4376	22.86	12.72
0.4626	22.71	12.64
0.5000	22.89	12.74
0.5500	22.75	12.66
0.5751	22.87	12.73
0.6002	22.80	12.69
0.6252	22.91	12.75
0.6751	22.86	12.72
0.7501	22.97	12.78
0.7751	22.93	12.76
0.8500	22.99	12.80
0.9252	23.00	12.80
1.0000	23.02	12.81

Table 6.2-7
SATAN V MASS AND ENERGY RELEASE RATES
SURGE LINE DER

Time (sec)	Mass Flow Rate (10^3 lbm/sec)	Energy Flow Rate (10^6 Btu/sec)
0.0	0.0	0.0
0.0010	14.47	9.845
0.0050	14.42	9.801
0.0090	14.43	9.792
0.0100	18.53	12.48
0.0110	19.06	12.82
0.0150	18.48	12.42
0.0200	17.35	11.67
0.0250	17.81	11.96
0.0301	18.23	12.24
0.0341	18.63	12.49
0.0371	18.78	12.59
0.0390	18.94	12.69
0.0401	19.01	12.73
0.0431	19.00	12.73
0.0440	18.98	12.71
0.0471	18.98	12.71
0.0500	18.94	12.69
0.0550	18.73	12.55
0.0601	18.43	12.35
0.0650	18.23	12.22
0.0701	18.19	12.19
0.0751	18.19	12.19
0.0800	17.98	12.05
0.0881	17.52	11.76
0.0931	17.44	11.70
0.1001	17.79	11.90
0.1150	18.55	12.43
0.1300	17.85	11.96
0.1500	16.32	10.97
0.1751	15.02	10.13
0.2000	14.55	9.822
0.3003	14.21	9.600
0.3801	14.17	9.564
0.5002	14.09	9.508
0.6004	14.02	9.457
0.7001	13.97	9.418
0.8003	13.92	9.381
0.9003	13.86	9.333
1.0001	13.80	9.288

Table 6.2-8
SATAN V MASS AND ENERGY RELEASE RATES
SPRAY LINE DER

Time (sec)	Mass Flow Rate (10^3 lbm/sec)	Energy Flow Rate (10^6 Btu/sec)
0.0	0.0	0.0
0.05	4.422	2.702
0.10	4.527	2.760
0.15	4.309	2.637
0.20	4.296	2.629
0.25	4.307	2.633
0.30	4.222	2.587
0.35	4.243	2.597
0.40	4.227	2.588
0.45	4.245	2.597
0.50	4.263	2.607
0.55	4.246	2.597
0.60	4.252	2.599
0.65	4.246	2.596
0.70	4.243	2.593
0.75	4.241	2.591
0.80	4.227	2.583
0.85	4.223	2.580
0.90	4.214	2.574
0.95	4.202	2.567
1.00	4.197	2.564
1.05	4.181	2.554
1.10	4.172	2.549
1.15	4.163	2.543
1.20	4.150	2.534
1.25	4.142	2.530
1.30	4.130	2.523
1.35	4.118	2.516
1.40	4.111	2.511
1.45	4.098	2.503
1.50	4.088	2.498
1.55	4.079	2.492
1.60	4.067	2.485
1.65	4.057	2.479
1.70	4.048	2.473
1.75	4.036	2.467
1.80	4.026	2.461
1.85	4.016	2.454
1.95	3.994	2.442
2.00	3.984	2.436

Table 6.2-9
SATAN V MASS AND ENERGY RELEASE RATES
HOT LEG SINGLE-ENDED SPLIT

Time (sec)	Mass Flow Rate (10^4 lbm/sec)	Energy Flow Rate (10^6 Btu/sec)
0.0	0.0	0.0
0.001	3.540	23.20
0.002	4.354	28.48
0.0060	3.696	23.94
0.0130	4.187	27.03
0.0170	4.178	26.96
0.0260	4.313	27.83
0.0381	4.523	29.22
0.0440	4.611	29.82
0.0450	4.966	32.15
0.0470	4.698	30.38
0.0510	4.778	30.90
0.0540	4.781	30.93
0.0600	4.697	30.43
0.0700	4.618	30.03
0.0780	4.582	29.95
0.0840	4.618	30.27
0.0910	4.798	31.48
0.1000	4.606	30.19
0.1500	4.274	28.24
0.2001	4.342	28.49
0.2500	4.194	27.48
0.3002	4.234	27.70
0.3401	4.189	27.40
0.3701	4.205	27.50
0.4500	4.183	27.31
0.5002	4.140	27.07
0.6001	3.939	26.07
0.6703	3.861	25.56
0.7802	3.904	25.41
0.8501	3.865	25.21
0.8802	3.825	25.05
0.9501	3.729	24.62
1.0001	3.672	24.29
1.5002	3.222	21.53

Table 6.2-10
NRC STANDARD SUBCOMPARTMENT PROBLEMS
COMPARISON OF PEAK PRESSURE DIFFERENCES

Problem Number	Peak Pressure Differential (psid)			
	S&W RELAP4	THREED	NRC-RELAP3	Compare
1		287.2	268	268
2		102.8	105	92
3		51.9	64	45
4	6.07	2.2	5.8	2.2
5	3.4	1.2	3.5	1.1
6	8.43	2.8	8.7	2.8
7		82.6	73	69
8		48.6	47	46
9		17.6	20	18
10		31.1	33	32
11		15.6	17	16
12		8.2	7.5	8.2
13 V-1		495.2		
V-2		4.7		
V-3		4.7		

Table 6.2-11
SUMMARY OF RESULTS OF CONTAINMENT ANALYSIS

Break Location	Break Area $(ft)^2$	Initial Containment Pressure/Temperature (psia/°F)	Peak Containment Pressure (psig)	Time of Peak Pressure (sec)	Peak Containment Temperature (°F)
Hot leg	9.17	14.085/116.5	42.7	18.18	269.4

Notes:
1. Peak containment pressure and peak containment temperature occur concurrently.
2. Results are based on parameters listed in Tables 6.2-1 and 6.2-2.

Table 6.2-12
ACCIDENT CHRONOLOGY FOR DEPSG CONTAINMENT DEPRESSURIZATION ANALYSES WITH FAILURE OF A DIESEL GENERATOR

TS SW Temperature	55ºF	95ºF
TS Containment Air Partial Pressure	12.3 psia	10.4 psia
	Time (sec)	Time (sec)
Accident occurs	0.0	0.0
Containment depressurization actuation signal	2.6	3.1
Casing cooling flow starts to containment	57.6	58.1
Quench spray delivers to containment atmosphere	72.6	73.1
Core reflooding ends	216.4	216.4
Outside recirculation spray delivers to containment atmosphere	2180.6	2156.4
Inside recirculation spray delivers to containment atmosphere	2305.2	2281.0
Containment pressure less than 2.0 psig	3192	2984
Recirculation mode transfer complete	4420.2	4389.5
Quench spray pump stops	5833.7	5794.4
Depressurization peak containment pressure occurs (pressure)	6527 (0.65 psig)	6555 (0.47 psig)
Casing cooling pump stops	7769.9	7770.4
Containment pressure becomes subatmospheric permanently	11,590	13,670

Table 6.2-13
BLOWDOWN MASS AND ENERGY RELEASE
DOUBLE-ENDED PUMP SUCTION—MIN SI

	Break Path No. 1 (M&E Exiting the SG Side of the Break)		Break Path No. 2 (M&E Exiting the pump Side of the Break)	
Time	Flow	Energy Thousand	Flow	Energy Thousand
sec	lbm/sec	Btu/sec	lbm/sec	Btu/sec
0.00	0.00	0.00	0.00	0.00
0.0008	3068.70	1677.24	3021.18	1650.31
0.0014	5945.21	3249.64	5401.92	2939.30
0.07	36051.07	19794.81	23668.74	12734.87
0.13	45245.72	24615.00	22427.20	12145.57
0.20	46062.90	25164.65	22416.54	12139.47
0.26	46851.14	25733.14	23211.03	12576.53
0.33	46791.19	25856.82	23168.49	12563.13
0.39	47054.72	26181.69	22956.47	12458.21
0.46	47313.96	26521.70	22528.85	12234.09
0.52	47030.76	26573.10	22105.19	12010.11
0.59	46102.25	26255.75	21679.40	11783.65
0.65	43211.96	24798.05	21315.76	11589.07
0.72	41008.58	23693.79	21004.15	11421.66
0.78	41241.52	23977.32	20740.76	11279.43
0.85	41425.07	24220.65	20472.83	11135.61
0.91	41481.48	24386.18	20309.47	11048.36
0.98	41437.40	24483.76	20141.82	10959.15
1.04	41277.24	24509.69	20021.64	10895.17
1.11	41051.66	24493.59	19897.81	10828.63
1.17	40801.16	24462.05	19776.57	10763.05
1.24	40535.19	24420.78	19685.07	10713.38
1.30	40247.53	24366.42	19620.27	10678.16
1.37	39937.53	24298.10	19583.66	10658.31
1.43	39614.90	24221.12	19562.49	10646.80
1.50	39278.24	24139.80	19541.19	10635.13
1.56	38934.13	24053.27	19510.55	10618.13
1.63	38559.74	23951.32	19469.81	10595.48
1.69	38140.80	23824.14	19427.23	10571.76
1.76	37661.42	23664.26	19390.00	10551.00
1.82	37116.08	23462.84	19364.90	10536.98.
1.89	36212.81	23041.22	19346.24	10526.59
1.95	35345.42	22643.38	19324.25	10514.45
2.02	34488.33	22253.26	19273.96	10486.75
2.08	33625.60	21855.04	19195.05	10443.37
2.15	32656.22	21384.08	19094.03	10387.98

Table 6.2-13 (continued)
BLOWDOWN MASS AND ENERGY RELEASE
DOUBLE-ENDED PUMP SUCTION—MIN SI

	Break Path No. 1 (M&E Exiting the SG Side of the Break)		Break Path No. 2 (M&E Exiting the pump Side of the Break)	
Time sec	Flow lbm/sec	Energy Thousand Btu/sec	Flow lbm/sec	Energy Thousand Btu/sec
2.21	31750.47	20946.93	18989.28	10330.68
2.28	30826.07	20493.34	18882.06	10272.20
2.34	29917.40	20044.22	18772.24	10212.45
2.41	29012.04	19590.08	18642.44	10141.82
2.47	28103.92	19124.04	18437.30	10029.75
2.54	27155.56	18616.04	18134.13	9864.65
2.60	26165.22	18068.92	17922.26	9749.56
2.67	25099.69	17458.40	17715.88	9637.68
2.73	24100.05	16887.24	17527.03	9535.32
2.80	23188.28	16372.45	17336.25	9431.86
2.86	22306.50	15864.83	17133.01	9321.66
2.93	21469.74	15372.86	16925.56	9209.22
2.99	20709.86	14916.04	16685.79	9079.16
3.06	19832.30	14348.81	16470.45	8962.72
3.12	18892.11	13720.37	16288.38	8864.59
3.19	18172.53	13241.64	16112.27	8769.76
3.25	17641.00	12890.01	15936.68	8675.18
3.32	17282.09	12654.07	15755.92	8577.70
3.38	16861.60	12362.46	15560.55	8472.23
3.45	16523.19	12130.31	15370.17	8369.50
3.51	16219.84	11922.80	15198.12	8276.91
3.58	15986.46	11766.54	15053.75	8199.52
3.64	15735.82	11595.02	14903.31	8118.68
3.71	15507.98	11440.24	14765.21	8044.48
3.77	15293.20	11294.85	14628.23	7970.85
3.84	15101.31	11165.83	14503.06	7903.87
3.90	14923.09	11045.94	14534.14	7922.72
3.97	14751.22	10928.90	14459.33	7882.72
4.13	14246.44	10578.32	14092.42	7684.00
4.26	13983.89	10393.48	13908.36	7585.61
4.39	13716.86	10189.67	13676.82	7460.54
4.52	13493.67	10005.23	14595.16	7969.32
4.65	13299.85	9830.40	14726.87	8039.17
4.78	13149.01	9678.07	14463.02	7895.46
4.91	13072.11	9572.48	14259.30	7786.37
5.04	10368.84	9513.40	14159.69	7733.30

Table 6.2-13 (continued)
BLOWDOWN MASS AND ENERGY RELEASE
DOUBLE-ENDED PUMP SUCTION—MIN SI

	Break Path No. 1 (M&E Exiting the SG Side of the Break)		Break Path No. 2 (M&E Exiting the pump Side of the Break)	
Time sec	Flow lbm/sec	Energy Thousand Btu/sec	Flow lbm/sec	Energy Thousand Btu/sec
5.17	13112.12	9483.62	14023.41	7659.95
5.30	13172.92	9465.71	13888.23	7588.06
5.43	13244.69	9455.45	13788.26	7535.80
5.56	13235.82	9385.70	13695.82	7487.98
5.69	13505.40	9521.66	13652.61	7467.13
5.82	14755.94	10312.18	13576.66	7428.86
5.95	14055.55	10040.59	13544.25	7413.73
6.08	12588.21	9433.10	13402.23	7336.70
6.21	11785.68	9031.92	13205.17	7228.62
6.34	11686.56	8932.81	13054.17	7145.93
6.47	11874.49	8970.01	12928.10	7077.11
6.60	12180.33	9065.36	12800.38	7007.85
6.73	12457.60	9133.78	12680.93	6943.32
6.86	12743.26	9221.90	12563.93	6879.98
6.99	12882.16	9238.16	12427.38	6805.41
7.12	12962.08	9235.96	12270.90	6719.63
7.25	13009.76	9220.62	12112.03	6632.69
7.38	13041.82	9197.74	11966.06	6553.27
7.51	13057.74	9166.90	11838.74	6484.49
7.64	13019.38	9104.58	11714.91	6417.59
7.77	13078.91	9117.61	11573.10	6340.51
7.90	12992.90	9040.75	11412.00	6252.52
8.03	12802.43	8910.65	11262.92	6171.36
8.16	12555.74	8752.27	11133.65	6101.40
8.29	12289.94	8579.51	11007.04	6032.87
8.42	11977.99	8379.14	10865.57	5955.85
8.55	11616.65	8151.77	10719.07	5875.88
8.68	11223.64	7906.60	10589.35	5805.26
8.81	10820.32	7655.92	10476.60	5744.05
8.94	10411.19	7403.62	10363.19	5682.15
9.07	10002.33	7155.63	10245.34	5617.29
9.20	9607.48	6922.12	10135.47	5556.34
9.33	9233.67	6706.88	10033.24	5499.34
9.46	8878.85	6506.77	9924.62	5438.29
9.59	8547.32	6322.56	9822.28	5375.87
9.72	8235.94	6150.95	9763.61	5325.95

Table 6.2-13 (continued)
BLOWDOWN MASS AND ENERGY RELEASE
DOUBLE-ENDED PUMP SUCTION—MIN SI

	Break Path No. 1 (M&E Exiting the SG Side of the Break)		Break Path No. 2 (M&E Exiting the pump Side of the Break)	
Time sec	Flow lbm/sec	Energy Thousand Btu/sec	Flow lbm/sec	Energy Thousand Btu/sec
9.85	7943.18	5989.27	9710.35	5268.16
9.98	7657.98	5830.29	9635.83	5189.43
10.13	7346.88	5653.93	9571.26	5101.01
10.26	7097.06	5509.72	9562.42	5043.26
10.39	6863.80	5370.90	9567.88	4990.18
10.52	6642.23	5234.43	9550.27	4924.53
10.65	6419.55	5090.84	9443.75	4816.42
10.78	6194.11	4942.20	9244.17	4664.09
10.91	5994.11	4807.76	9113.45	4548.46
11.04	5826.03	4686.17	9065.00	4476.40
11.17	5684.29	4577.61	8991.85	4398.40
11.30	5569.96	4488.73	8945.26	4339.06
11.43	5470.26	4414.61	8837.35	4254.85
11.56	5364.52	4346.28	8647.94	4134.86
11.69	5237.83	4276.30	8468.21	4021.14
11.82	5092.75	4205.06	8317.07	3921.87
11.95	4938.42	4136.83	8205.76	3842.22
12.08	4778.14	4070.50	7994.58	3717.03
12.21	4610.68	4006.50	7575.38	3496.92
12.34	4434.46	3943.82	7313.12	3350.61
12.47	4249.97	3884.26	7384.00	3354.52
12.60	4066.33	3827.62	7649.77	3447.89
12.73	3864.74	3766.90	7837.56	3512.58
12.86	3639.05	3701.21	8176.61	3651.19
12.99	3400.72	3635.20	8198.22	3651.07
13.12	3159.70	3567.47	7537.06	3347.70
13.25	2939.80	3495.65	6738.66	2983.00
13.38	2783.54	3391.50	6243.97	2753.11
13.51	2569.80	3165.47	5758.90	2524.49
13.64	2393.16	2959.79	5353.36	2325.61
13.77	2227.34	2762.19	4963.51	2123.29
13.90	2072.02	2574.21	5041.32	2102.95
14.03	1933.35	2407.18	5238.05	2116.41
14.16	1806.61	2253.04	5813.70	2277.78
14.29	1687.43	2107.62	6769.94	2586.79
14.42	1573.59	1968.02	5844.52	2196.79

Table 6.2-13 (continued)
BLOWDOWN MASS AND ENERGY RELEASE
DOUBLE-ENDED PUMP SUCTION—MIN SI

	Break Path No. 1 (M&E Exiting the SG Side of the Break)		Break Path No. 2 (M&E Exiting the pump Side of the Break)	
Time sec	Flow lbm/sec	Energy Thousand Btu/sec	Flow lbm/sec	Energy Thousand Btu/sec
14.55	1466.68	1836.73	5966.20	2211.73
14.68	1370.53	1718.43	5671.84	2077.90
14.81	1280.03	1606.69	5016.91	1820.95
14.94	1194.57	1500.89	5077.47	1822.51
15.07	1112.72	1399.40	4857.25	1721.95
15.20	1181.51	1481.09	4628.29	1617.74
15.33	995.96	1256.51	4390.63	1510.42
15.46	916.57	1157.39	4161.57	1406.22
15.59	846.57	1069.48	3944.62	1307.08
15.72	782.39	988.87	3732.77	1210.80
15.85	728.82	921.62	3517.26	1115.45
15.98	675.64	854.54	3288.19	1019.23
16.11	625.25	791.39	3030.78	918.20
16.24	603.31	763.75	2740.83	812.00
16.37	569.66	721.37	2405.78	697.75
16.50	543.38	688.28	2004.42	570.35
16.63	513.40	650.48	1517.60	425.06
16.76	483.57	612.87	944.75	261.84
16.89	457.57	580.05	294.53	81.43
17.02	428.35	543.14	-34.65	-10.06
17.15	400.88	508.45	-21.15	-8.97
17.28	374.74	475.43	23.92	18.36
17.41	351.29	445.80	14.14	13.12
17.54	333.80	423.72	10.66	10.64
17.67	304.96	387.20	13.55	14.46
17.80	285.76	362.92	12.50	13.77
17.93	262.70	333.73	36.74	18.12
18.06	229.84	292.10	23.50	9.76
18.19	207.21	263.44	18.71	10.17
18.32	183.67	233.61	9.40	6.76
18.45	169.40	215.57	4.65	4.15
18.58	161.51	205.58	3.99	3.72
18.71	141.03	179.60	-1.53	-1.47
18.84	125.35	159.70	18.61	8.42
18.97	92.55	118.02	-14.24	-5.56
19.10	68.40	87.34	12.31	5.97

Table 6.2-13 (continued)
BLOWDOWN MASS AND ENERGY RELEASE
DOUBLE-ENDED PUMP SUCTION—MIN SI

	Break Path No. 1 (M&E Exiting the SG Side of the Break)		Break Path No. 2 (M&E Exiting the pump Side of the Break)	
Time	Flow	Energy Thousand	Flow	Energy Thousand
sec	lbm/sec	Btu/sec	lbm/sec	Btu/sec
19.23	25.20	32.28	0.00	0.00
19.36	0.00	0.00	0.00	0.00

Table 6.2-14
REFLOOD MASS AND ENERGY RELEASE
DOUBLE-ENDED PUMP SUCTION—MIN SI—4150 GPM

	Break Path No. 1 (M&E Exiting the SG Side of the Break)		Break Path No. 2 (M&E Exiting the pump Side of the Break)	
Time	Flow	Energy Thousand	Flow	Energy Thousand
sec	lbm/sec	Btu/sec	lbm/sec	Btu/sec
19.81	0.00	0.00	0.00	0.00
20.01	0.00	0.00	0.00	0.00
20.11	0.00	0.00	0.00	0.00
20.21	0.00	0.00	0.00	0.00
20.26	0.00	0.00	0.00	0.00
20.36	98.61	116.17	0.00	0.00
20.46	41.41	48.76	0.00	0.00
20.56	43.24	50.92	0.00	0.00
20.66	50.32	59.26	0.00	0.00
20.76	56.99	67.11	0.00	0.00
20.86	63.21	74.43	0.00	0.00
20.96	69.04	81.31	0.00	0.00
21.06	74.54	87.79	0.00	0.00
21.16	79.77	93.95	0.00	0.00
21.24	83.52	98.37	0.00	0.00
21.26	84.74	99.81	0.00	0.00
21.36	89.50	105.42	0.00	0.00
21.46	94.07	110.80	0.00	0.00
21.56	98.46	115.98	0.00	0.00
21.66	102.70	120.98	0.00	0.00
21.76	106.80	125.81	0.00	0.00
21.86	110.77	130.49	0.00	0.00
21.96	114.62	135.03	0.00	0.00
22.06	118.36	139.45	0.00	0.00
22.16	122.00	143.74	0.00	0.00
22.26	125.54	147.92	0.00	0.00
22.36	129.00	152.00	0.00	0.00
23.36	170.86	201.42	648.73	64.76
24.37	480.58	569.66	4689.55	520.40
24.67	479.48	568.35	4678.31	520.92
25.37	473.14	560.78	4621.79	516.75
26.37	462.42	547.96	4522.95	508.48
27.37	451.38	534.77	4418.86	499.48
28.37	440.56	521.84	4315.30	490.39
28.87	435.31	515.56	4264.46	485.90
29.37	430.16	509.42	4214.42	481.47

Table 6.2-14 (continued)
REFLOOD MASS AND ENERGY RELEASE
DOUBLE-ENDED PUMP SUCTION—MIN SI—4150 GPM

	Break Path No. 1 (M&E Exiting the SG Side of the Break)		Break Path No. 2 (M&E Exiting the pump Side of the Break)	
Time	Flow	Energy Thousand	Flow	Energy Thousand
sec	lbm/sec	Btu/sec	lbm/sec	Btu/sec
30.37	420.24	497.57	4117.03	472.83
31.37	410.80	486.31	4023.40	464.50
32.37	401.84	475.61	3933.52	456.50
33.37	393.32	465.46	3847.29	448.82
34.37	385.23	455.81	3764.52	441.45
34.47	384.44	454.87	3756.43	440.73
35.37	377.52	446.63	3685.05	434.36
36.37	370.18	437.88	3608.66	427.56
37.37	363.18	429.55	3535.17	421.01
38.37	356.50	421.58	3464.39	414.70
39.37	350.10	413.97	3396.17	408.61
40.37	343.97	406.68	3330.32	402.74
41.12	377.35	446.42	3718.08	407.57
41.42	375.68	444.43	3700.70	405.93
42.42	370.05	437.72	3641.83	400.62
43.42	364.63	431.27	3584.82	395.47
44.42	359.41	425.05	3529.55	390.48
45.42	354.37	419.05	3475.92	385.63
46.42	349.50	413.25	3423.85	380.91
47.42	212.48	250.70	1481.59	224.18
48.22	182.19	214.87	457.13	113.66
48.42	175.42	206.82	315.18	87.97
49.42	177.13	208.84	313.11	87.93
50.42	175.87	207.35	315.17	88.08
51.42	174.58	205.83	317.30	88.24
52.42	173.28	204.29	319.45	88.41
53.42	171.96	202.74	321.62	88.58
54.42	170.64	201.17	323.79	88.76
55.42	169.29	199.58	325.99	88.94
56.42	167.94	197.98	328.19	89.13
57.42	166.57	196.36	330.41	89.33
58.42	165.18	194.72	332.65	89.53
59.42	163.77	193.06	334.91	89.74
60.42	162.35	191.38	337.19	89.96
61.42	160.90	189.66	339.50	90.18
62.42	159.42	187.92	341.84	90.41

Table 6.2-14 (continued)
REFLOOD MASS AND ENERGY RELEASE
DOUBLE-ENDED PUMP SUCTION—MIN SI—4150 GPM

	Break Path No. 1 (M&E Exiting the SG Side of the Break)		Break Path No. 2 (M&E Exiting the pump Side of the Break)	
Time	Flow	Energy Thousand	Flow	Energy Thousand
sec	lbm/sec	Btu/sec	lbm/sec	Btu/sec
63.42	157.93	186.16	344.21	90.65
64.42	156.41	184.36	346.60	90.90
65.42	155.59	183.39	348.09	90.88
66.42	155.28	183.03	348.87	90.75
67.42	154.97	182.67	349.64	90.62
68.42	154.67	182.30	350.41	90.49
69.42	154.36	181.94	351.19	90.36
69.82	154.24	181.80	351.49	90.31
70.42	154.06	181.58	351.96	90.24
71.42	153.76	181.23	352.73	90.11
72.42	163.45	180.87	353.50	89.99
73.42	153.15	180.52	354.27	89.86
74.42	152.85	180.16	355.04	89.74
75.42	152.55	179.81	355.81	89.61
76.42	152.26	179.46	356.58	89.49
77.42	151.96	179.10	357.35	89.37
78.42	151.66	178.75	358.12	89.25
79.42	151.37	178.40	358.88	89.12
80.42	151.07	178.05	359.65	89.00
82.42	150.47	177.35	361.17	88.75
84.42	149.88	176.64	362.68	88.51
86.42	149.28	175.94	364.18	88.26
88.42	148.68	175.24	365.67	88.00
90.42	148.09	174.53	367.15	87.75
92.42	147.49	173.83	368.61	87.50
93.62	147.13	173.40	369.49	87.34
94.42	146.89	173.12	370.07	87.24
96.42	146.30	172.42	371.52	86.98
98.42	145.70	171.71	372.97	86.72
100.42	145.10	171.01	374.40	86.46
102.42	144.50	170.29	375.84	86.20
104.42	143.89	169.58	377.27	85.94
106.42	143.28	168.86	378.70	85.68
108.42	142.67	168.14	380.12	85.41
110.42	142.07	167.42	391.53	85.14
112.42	141.46	166.70	382.94	84.88

Table 6.2-14 (continued)
REFLOOD MASS AND ENERGY RELEASE
DOUBLE-ENDED PUMP SUCTION—MIN SI—4150 GPM

	Break Path No. 1 (M&E Exiting the SG Side of the Break)		Break Path No. 2 (M&E Exiting the pump Side of the Break)	
Time	Flow	Energy Thousand	Flow	Energy Thousand
sec	lbm/sec	Btu/sec	lbm/sec	Btu/sec
114.42	140.84	165.98	384.35	84.61
116.42	140.23	165.26	385.75	84.34
118.42	139.61	164.53	387.14	84.07
119.62	139.25	164.10	387.98	83.91
120.42	139.00	163.81	388.54	83.80
122.42	138.39	163.08	389.92	83.53
124.42	137.77	162.35	391.31	83.26
126.42	137.15	161.62	392.68	82.99
128.42	136.53	160.89	394.06	82.72
130.42	135.91	160.16	395.43	82.45
132.42	135.29	159.43	396.80	82.18
134.42	134.67	158.70	398.16	81.90
136.42	134.05	157.96	399.52	81.63
138.42	133.43	157.23	400.88	81.36
140.42	132.80	156.49	402.23	81.09
142.42	132.18	155.76	403.58	80.81
144.42	131.55	155.02	404.92	80.54
146.42	130.93	154.28	406.27	80.27
148.22	130.37	153.62	407.47	80.02
148.42	130.30	153.54	407.60	80.00
150.42	129.68	152.80	408.94	79.72
152.42	129.04	152.05	410.28	79.45
154.42	128.41	151.30	411.62	79.17
156.42	127.77	150.55	412.95	78.90
158.42	127.13	149.80	414.29	78.63
160.42	126.50	149.05	415.62	78.35
162.42	125.86	148.30	416.95	78.08
164.42	125.22	147.54	418.27	77.81
166.42	124.58	146.79	419.59	77.53
168.42	123.94	146.03	420.91	77.26
170.42	123.30	145.28	422.23	76.99
172.42	122.66	144.52	423.54	76.72
174.42	122.02	143.76	424.85	76.45
176.42	121.38	143.01	426.16	76.17
178.42	120.73	142.25	427.46	75.91
180.02	120.22	141.64	428.51	75.69

Table 6.2-14 (continued)
REFLOOD MASS AND ENERGY RELEASE
DOUBLE-ENDED PUMP SUCTION—MIN SI—4150 GPM

	Break Path No. 1 (M&E Exiting the SG Side of the Break)		Break Path No. 2 (M&E Exiting the pump Side of the Break)	
Time	Flow	Energy Thousand	Flow	Energy Thousand
sec	lbm/sec	Btu/sec	lbm/sec	Btu/sec
180.42	120.09	141.49	428.77	75.64
182.42	119.45	140.73	430.07	75.37
184.42	118.81	139.98	431.36	75.10
186.42	118.16	139.22	432.66	74.83
188.42	117.52	138.46	433.95	74.57
190.42	116.87	137.70	435.24	74.30
192.42	116.23	136.94	436.53	74.04
194.42	115.59	136.18	437.81	73.77
196.42	114.94	135.42	439.09	73.51
198.42	114.30	134.66	440.37	73.25
200.42	113.65	133.90	441.65	72.99
202.42	113.00	133.13	442.94	72.72
204.42	112.35	132.36	444.22	72.46
206.42	111.69	131.59	445.50	72.20
208.42	111.04	130.82	446.78	71.94
210.42	110.39	130.05	448.06	71.69
212.42	109.74	129.28	449.33	71.43
214.42	109.08	128.51	450.60	71.18
216.32	108.46	127.78	451.81	70.93

The following information is HISTORICAL and is not intended or expected to be updated for the life of the plant.

Table 6.2-15

PRINCIPAL PARAMETERS DURING REFLOOD

DOUBLE-ENDED PUMP SUCTION—MIN SI

Time Seconds	Flooding Temp Degree F	Flooding Rate In/Sec	Carryover Fraction	Core Height ft	Downcomer Height ft	Flow Fraction	Injection Total	Injection Accumulator (pounds mass per second)	Injection Spill (pounds mass per second)	Enthalpy btu/lbm
20.8	252.1	0.000	0.000	0.00	0.00	0.333	0.0	0.0	0.0	0.00
21.5	248.1	27.148	0.000	0.67	1.54	0.000	7453.1	7453.1	0.0	94.48
21.6	246.2	30.866	0.000	1.04	1.62	0.000	7380.0	7380.0	0.0	94.48
22.8	244.4	2.654	0.319	1.50	4.95	0.410	6944.3	6944.3	0.0	94.48
23.8	244.0	2.506	0.446	1.63	7.82	0.442	6654.9	6654.9	0.0	94.48
27.4	242.5	4.748	0.658	2.00	15.59	0.681	5290.0	5290.0	0.0	94.48
28.9	241.3	4.463	0.700	2.18	15.61	0.679	5013.8	5013.8	0.0	94.48
32.3	239.2	3.949	0.738	2.51	15.61	0.669	4565.3	4565.3	0.0	94.48
37.9	237.2	3.485	0.758	2.94	15.61	0.652	3997.9	3997.9	0.0	94.48
38.8	237.0	3.698	0.761	3.00	15.61	0.668	4411.2	3789.3	0.0	84.43
46.1	235.7	3.332	0.768	3.50	15.61	0.652	3891.7	3267.2	0.0	83.04
52.9	235.5	2.235	0.760	3.91	15.61	0.518	632.6	0.0	0.0	23.23
54.9	235.8	2.182	0.760	4.00	15.61	0.508	633.2	0.0	0.0	23.23
66.9	238.6	2.037	0.761	4.51	15.61	0.489	633.7	0.0	0.0	23.23
79.3	243.2	1.982	0.763	5.00	15.61	0.489	633.7	0.0	0.0	23.23
93.9	249.8	1.920	0.767	5.56	15.61	0.490	633.7	0.0	0.0	23.23
106.0	255.7	1.868	0.770	6.00	15.61	0.491	633.7	0.0	0.0	23.23
121.9	262.4	1.802	0.775	6.55	15.61	0.492	633.7	0.0	0.0	23.23
135.5	267.2	1.746	0.779	7.00	15.61	0.492	633.7	0.0	0.0	23.23
151.9	272.1	1.679	0.784	7.51	15.61	0.493	633.7	0.0	0.0	23.23
168.6	276.4	1.611	0.789	8.00	15.61	0.493	633.7	0.0	0.0	23.23
187.9	280.5	1.534	0.795	8.53	15.61	0.493	633.7	0.0	0.0	23.23
206.7	283.8	1.459	0.802	9.00	15.61	0.493	633.8	0.0	0.0	23.23
229.9	287.2	1.368	0.812	9.53	15.61	0.493	633.8	0.0	0.0	23.23
253.4	290.1	1.277	0.825	10.00	15.61	0.493	633.8	0.0	0.0	23.23

Table 6.2-16
LIMITING MSLB CONTAINMENT PEAK PRESSURE CASES & PEAK TEMPERATURE CASES

Peak Pressure Cases

Steam Line Break Size, ft^2	1.4	1.4	0.707	0.4
Break Type	DER	DER	Split	DER
Core Power, % of 2898 MWt	0	30	30	30
Peak containment pressure, psia	56.88	57.65	57.38	57.08
Time of peak pressure, sec	214.5	1812	1814	1825

Note: GOTHIC pressure is 14.135 psia from TS air maximum pressure of 12.3 psia + 0.30 psi uncertainty + 1.535 psia vapor pressure

Peak Temperature Cases

Steam Line Break Size, ft^2	0.7	0.6	0.4
Break Type	DER	DER	DER
Core Power, % of 2898 MWt	102	102	30
Peak containment temperature, ºF	296.2	308.4	291.9
Time of peak temperature, sec	26.4	30.8	50.6

Note: GOTHIC pressure is 10.0 psia from TS minimum air pressure of 10.3 psia - 0.30 psi uncertainty

Table 6.2-17
MAIN STEAM LINE BREAK MASS AND ENERGY RELEASE ANALYSIS
REACTOR COOLANT SYSTEM PARAMETERS

Parameter	Value[a]
NSSS Power	2910 MWt
Reactor Power	2898 MWt
Reactor Coolant Pump Heat	12 MWt
Vessel Flow	278,400 gpm
Pressurizer Pressure	2250 psia
Reactor Coolant Temperatures	
Core Outlet	624.0°F
Vessel Outlet	621.2°F
Core Average	590.4°F
Vessel Average	586.8°F
Vessel/Core Inlet	552.3°F
Steam Generator	
Steam Temperature	525.2°F
Steam Pressure	850 psia
Zero Load Temperature	547.0°F

a. These are the nominal plant conditions that were used to develop initial conditions for analyses at 0%, 30%, 70% and 102% of the stated power levels. The maximum analysis core power was 2956 MWt (102% of 2898 MWt) with a corresponding NSSS power of 2968 MWt.

Table 6.2-18
MAIN STEAM LINE BREAK MASS AND ENERGY RELEASE ANALYSIS
SECONDARY SYSTEM PARAMETERS

Parameter	Value
Auxiliary Feedwater Flow Rate[a]	
Intact Loops	350 gpm
Faulted Loop	900 gpm
Steam Generator Fluid Mass, % of prog lvl	+5
Unisolated volume per feedwater line	404 ft^3
Auxiliary feedwater isolation time	30 minutes
Auxiliary feedwater purge volume	40 ft^3
Auxiliary feedwater isolation delay	60 seconds
Moody contraction coefficient	0 fl/D
Unisolated steamline volume	7283 ft^3

a. The impact of increasing this auxiliary feedwater flow rate to 970 gpm was subsequently evaluated. That evaluation confirmed that with this increase in auxiliary feedwater flow, the results of the analysis for main steam line break in containment would still be within the acceptance criteria. The auxiliary feedwater flow rate to the intact steam generators is among the less significant secondary parameters. Expected variations in auxiliary feedwater flow to the intact steam generators do not invalidate the results of the analysis, so flow is conservatively modeled as a constant flow rate.

Table 6.2-19
EFFECT OF INITIAL TOTAL PRESSURE

Initial temperature = 120°F
Initial relative humidity = 14.8%

Initial Pressure (psia)	Peak Total Pressure (psia)	Peak Differential Pressure (psid)
9.25	49.84	40.59
10.25	49.85	39.60
11.25	49.86	38.61

Table 6.2-20
EFFECT OF RELATIVE HUMIDITY

Initial total pressure = 11.0 pisa
Initial temperature = 100°F

Initial Relative Humidity (%)	Peak Total Pressure (psia)	Peak Differential Pressure (psid)
40	49.77	38.77
60	49.77	38.77
100	49.76	38.76

Table 6.2-21
EFFECT OF INITIAL TEMPERATURE

Initial total pressure = 11.0 psia
Initial relative humidity = 100%

Initial Temperature (°F)	Peak Total Pressure (psia)	Peak Differential Pressure (psid)
80	49.77	38.77
100	49.76	38.76
120	49.75	38.75

Table 6.2-22
REACTOR CAVITY VENT AREAS

Vent	Vent Area (ft^2)	Compartment to Which Vent Discharges
Hot leg penetration	3.69	Steam generator cubicle (1-RC-E-1A or 2-RC-E-1A)
Hot leg penetration	3.69	Steam generator cubicle (1-RC-E-1B or 2-RC-E-1B)
Hot leg penetration	3.69	Steam generator cubicle (1-RC-E-1C or 2-RC-E-1C)
Cold leg penetration	3.44	Steam generator cubicle (1-RC-E-1A or 2-RC-E-1A)
Cold leg penetration	3.44	Steam generator cubicle (1-RC-E-1B or 2-RC-E-1B)
Cold leg penetration	3.44	Steam generator cubicle (1-RC-E-1C or 2-RC-E-1C)
Upper reactor cavity	144.7	Refueling cavity
Passageway in ceiling of incore instrumentation tunnel	60	Residual heat removal heat exchanger area
Ventilation line in incore instrumentation tunnel	8.7	Residual heat removal heat exchanger area
Ventilation line in reactor cavity wall	2.18	Lower level of containment (El. 228 ft. 3 in.)

All vent areas and volumes are calculated assuming that insulation remains in place.

Table 6.2-23
SUMMARY OF VENT AREAS, PRESSURE LOSS COEFFICIENTS, AND VENT FLOW MODELS USED IN THE 12-NODE UPPER REACTOR CAVITY MODEL

Between Nodes [a]	Vent Area (ft^2)	Contraction K	Friction K	Bend K	Input to THREED [b]	Expansion K	Input to RELAP4	THREED Flow Model
1-2	12.48	0.23	.003	—	0.23	0.30	0.53	HVFM-4
3-4	23.42	0.15	00.004	—	0.15	0.15	0.30	HVFM-4
5-6	12.48	0.23	0.003	—	0.23	0.30	0.53	HVFM-4
7-8	23.42	0.15	0.004	—	0.15	0.15	0.30	HVFM-4
9-10	12.48	0.23	0.003	—	0.23	0.30	0.53	HVFM-4
11-12	23.42	0.15	0.004	—	0.15	0.15	0.30	HVFM-4
1-3	13.11	0.12	0.075	0.112	0.31	0.43	0.74	HVFM-4
2-4	3.335	0.32	0.103	0.172	0.60	0.83	1.43	HVFM-4
3-5	12.11	0.15	0.075	0.112	0.34	0.31	0.65	HVFM-4
4-6	2.337	0.36	0.103	0.172	0.64,	0.84	1.48	HVFM-4
5-7	13.11	0.12	0.075	0.112	0.31	0.43	0.74	HVFM-4
6-8	3.335	0.32	0.103	0.172	0.60	0.83	1.43	HVFM-4
7-9	12.11	0.15	0.075	0.112	0.34	0.31	0.65	HVFM-4
8-10	2.337	0.36	0.103	0.172	0.64	0.84	1.48	HVFM-4
9-11	13.11	0.12	0.075	0.112	0.31	0.43	0.74	HVFM-4
10-12	3.335	0.32	0.103	0.172	0.60	0.83	1.43	HVFM-4
1-21	20.10	0.07	0.004	—	0.07	1.00	1.07	HVFM-4

a. Areas and losses between two nodes are the same in the forward and reverse direction, except where the nodes are listed in both directions, and except when one of the nodes is the containment (Node 21). It is assumed there is no backflow from the containment to any other node.
b. Input to THREED—an exit loss of 1.0 is built into THREED and is added to these values.
c. Venting through shield wall penetrations.
d. These vents are not included in the RELAP4 input due to code limitations. This will not have a significant effect on the peak calculated pressure differential.
e. Same as in forward direction.

Table 6.2-23 (continued)
SUMMARY OF VENT AREAS, PRESSURE LOSS COEFFICIENTS, AND VENT FLOW MODELS USED IN THE 12-NODE UPPER REACTOR CAVITY MODEL

Between Nodes [a]	Vent Area (ft^2)	Contraction K	Friction K	Bend K	Input to THREED [b]	Expansion K	Input to RELAP4	THREED Flow Model
3-21	28.14	0.07	0.005	—	0.08	1.00	1.08	HVFM-4
5-21	20.10	0.07	0.004	—	0.07	1.00	1.07	HVFM-4
7-21	28.14	0.07	0.005	—	0.08	1.00	1.08	HVFM-4
9-21	20.10	0.07	0.004	—	0.007	1.00	1.07	HVFM-4
11-21	28.14	0.07	0.005	—	0.08	1.00	1.08	HVFM-4
2-13	1.322	0.45	0.043	—	0.49	0.02	0.51	HVFM-4
4-14	1.851	0.45	0.061	—	0.51	0.02	0.53	HVFM-4
6-15	1.322	0.45	0.043	—	0.49	0.02	0.51	HVFM-4
8-16	1.851	0.45	0.061	—	0.51	0.02	0.53	HVFM-4
10-17	1.322	0.45	0.043	—	0.49	0.02	0.51	HVFM-4
12-18	1.851	0.45	0.061	—	0.51	0.02	0.53	HVFM-4
13-14	8.711	—	0.482	0.056	0.54	—	0.54	HVFM-2
14-15	8.711	—	0.482	0.056	0.54	—	0.54	HVFM-2
15-16	8.711	—	0.482	0.056	0.54	—	0.54	HVFM-2
17-18	8.711	—	0.482	0.056	0.54	—	0.54	HVFM-2
18-13	8.711	—	0.482	0.056	0.54	—	0.54	HVFM-2
13-19	1.561	—	0.040	—	0.04	0.98	1.02	HVFM-2

a. Areas and losses between two nodes are the same in the forward and reverse direction, except where the nodes are listed in both directions, and except when one of the nodes is the containment (Node 21). It is assumed there is no backflow from the containment to any other node.
b. Input to THREED—an exit loss of 1.0 is built into THREED and is added to these values.
c. Venting through shield wall penetrations.
d. These vents are not included in the RELAP4 input due to code limitations. This will not have a significant effect on the peak calculated pressure differential.
e. Same as in forward direction.

Table 6.2-23 (continued)
SUMMARY OF VENT AREAS, PRESSURE LOSS COEFFICIENTS, AND VENT FLOW MODELS USED IN THE 12-NODE UPPER REACTOR CAVITY MODEL

Between Nodes [a]	Vent Area (ft^2)	Contraction K	Friction K	Bend K	Input to THREED [b]	Expansion K	Input to RELAP4	THREED Flow Model
14-19	2.186	—	0.056	—	0.06	0.98	1.04	HVFM-2
15-19	1.561	—	0.040	—	0.04	0.98	1.02	HVFM-2
16-19	2.186	—	0.056	—	0.06	0.98	1.04	HVFM-2
17-19	1.561	—	0.040	—	0.04	0.98	1.02	HVFM-2
18-19	2.186	—	0.056	—	0.06	0.98	1.04	HVFM-2
19-20	64.32	0.5	—	—	0.5	0.30	0.80	HVFM-1
20-21	45.36	0.5	—	—	0.5	0.99	1.49	HVFM-1
1-11	12.11	0.15	0.075	0.112	0.34	0.47	0.81	HVFM-1
2-12	2.337	0.36	0.103	0.172	0.64	0.88	1.52	HVFM-1
1-21[c]	0.923	0.5	0.066	—	0.57	d	d	HVFM-1
2-21[c]	0.923	0.5	0.066	—	0.57	d	d	HVFM-1
3-21[c]	1.783	0.5	0.092	—	0.59	d	d	HVFM-1
4-21[c]	1.783	0.5	0.092	—	0.59	d	d	HVFM-1
5-21[c]	1.783	0.5	0.092	—	0.59	d	d	HVFM-1
6-21[c]	1.783	0.5	0.092	—	0.59	d	d	HVFM-1
7-21[c]	1.783	0.5	0.092	—	0.59	d	d	HVFM-1
8-21[c]	1.783	0.5	0.092	—	0.59	d	d	HVFM-1

a. Areas and losses between two nodes are the same in the forward and reverse direction, except where the nodes are listed in both directions, and except when one of the nodes is the containment (Node 21). It is assumed there is no backflow from the containment to any other node.
b. Input to THREED—an exit loss of 1.0 is built into THREED and is added to these values.
c. Venting through shield wall penetrations.
d. These vents are not included in the RELAP4 input due to code limitations. This will not have a significant effect on the peak calculated pressure differential.
e. Same as in forward direction.

Table 6.2-23 (continued)
SUMMARY OF VENT AREAS, PRESSURE LOSS COEFFICIENTS, AND VENT FLOW MODELS USED IN THE 12-NODE UPPER REACTOR CAVITY MODEL

Between Nodes [a]	Vent Area (ft^2)	Contraction K	Friction K	Bend K	Input to THREED [b]	Expansion K	Input to RELAP4	THREED Flow Model
9-21[c]	1.783	0.5	0.092	—	0.59	d	d	HVFM-1
10-21[c]	1.783	0.5	0.092	—	0.59	d	d	HVFM-1
11-21[c]	0.923	0.5	0.066	—	0.57	d	d	HVFM-1
12-21[c]	0.923	0.5	0.066	—	0.57	d	d	HVFM-1
13-2	1.322	0.03	0.043	—	0.07	e	e	HVFM-1
14-4	1.851	0.03	0.061	—	0.09	e	e	HVFM-1
15-6	1.322	0.03	0.043	—	0.07	e	e	HVFM-1
16-8	1.851	0.03	0.061	—	0.09	e	e	HVFM-1
17-10	1.322	0.03	0.043	—	0.07	e	e	HVFM-1
18-12	1.851	0.03	0.061	—	0.09	e	e	HVFM-1

a. Areas and losses between two nodes are the same in the forward and reverse direction, except where the nodes are listed in both directions, and except when one of the nodes is the containment (Node 21). It is assumed there is no backflow from the containment to any other node.
b. Input to THREED—an exit loss of 1.0 is built into THREED and is added to these values.
c. Venting through shield wall penetrations.
d. These vents are not included in the RELAP4 input due to code limitations. This will not have a significant effect on the peak calculated pressure differential.
e. Same as in forward direction.

Table 6.2-24
SUMMARY OF VENT AREAS, PRESSURE LOSS COEFFICIENTS,
AND VENT FLOW MODELS USED IN THE FIVE-NODE REACTOR ANNULUS MODEL

Between Nodes	Vent Area (ft^2)	Contraction K	Friction K	K Input to THREED[a]	THREED Flow Model	Expansion K	Input to RELAP
1-2	9.52	—	—	1.0	Moody	—	0
2-3	11.24	—	1.43	1.43	HVFM-2	1.0	2.43
3-4	64.32	0.5	—	0.5	HVFM-1	1.0	1.5
4-5	39.52	0.5	—	0.5	HVFM-1	1.0	1.5
1-5	144.7	0.07	0.03	0.1	HVFM-1	1.0	1.1

a. An exit loss of 1.0 is built into THREED and is added to these values. In those cases where the THREED flow model is Moody, the K input to THREED is the Moody multiplier.

Table 6.2-25

SUMMARY OF VENT AREAS, PRESSURE LOSS COEFFICIENTS, AND VENT FLOW MODELS USED IN THE SIX-NODE REACTOR ANNULUS MODEL

Between Nodes	Vent Area (ft^2)	Contraction K	Friction K	K Input to THREED[a]	THREED Flow Model	Expansion K	Input to RELAP
1-2	9.52	—	—	1.0	Moody	—	0
2-3	11.24	—	—	1.0	Moody	—	0
3-4	102.4	0.14	—	0.14	HVFM-1	1.0	1.14
3-5	8.05	0.5	—	0.5	HVFM-1	1.0	1.5
4-5	56.28	0.5	—	0.5	HVFM-1	1.0	1.5
5-6	39.52	0.5	—	0.5	HVFM-1	1.0	1.5
1-6	144.7	0.7	0.03	0.1	HVFM-1	1.0	1.1

a. An exit loss of 1.0 is built into THREED and is added to these values. In those cases where the THREED flow model is Moody, the K input to THREED is the Moody multiplier.

Table 6.2-26
SUMMARY OF VENT AREAS, PRESSURE LOSS COEFFICIENTS, AND VENT FLOW MODELS USED IN THE SEVEN-NODE INCORE INSTRUMENTATION TUNNEL MODEL

Between Nodes	Vent Area (ft^2)	Contraction K	Friction K	K Input to THREED[a]	THREED Flow Model	Expansion K	Input to RELAP
1-2	9.52	-	-	1.0	Moody	-	0
2-3	11.24	-	-	1.0	Moody	-	0
3-4	64.32	-	-	1.0	Moody	-	0
4-5	190.0	0.5	-	0.5	HVFM-1	1.0	1.5
4-6	60.0	0.5	-	0.5	HVFM-1	1.0	1.5
4-7	8.73	0.5	-	0.5	HVFM-1	1.0	1.5
5-7	2.18	0.5	-	0.5	HVFM-1	1.0	1.5
6-7	28.61	0.5	-	0.5	HVFM-1	1.0	1.5
1-7	144.7	0.07	0.03	0.10	HVFM-1	1.0	1.1

a. An exit loss of 1.0 is built into THREED and is added to these values. In those cases where the THREED flow model is Moody, the K input to THREED is the Moody multiplier.

Table 6.2-27
STEAM GENERATOR COMPARTMENT VENT AREAS AND PRESSURE LOSS COEFFICIENTS

Vent Type	Method of Calculation of Vent Area[a]	Vent Area (ft^2)	Contraction Plus Friction K	Expansion K[b]	K Input to THREED[c]	K Input to RELAP
Vent area a (blowdown diaphragm)	$\frac{\phi}{360}\ \ell\ \pi(R_3 + R_4)$ $\frac{65.68}{360}\ 3.42\ \pi(22.5 + 25)$	93.04	0.38	0.99	0.38	1.37
Vent area b (area between air duct and reactor cavity wall						
Vent area c (area above air duct venting horizontally)	$2\left\{\frac{1}{2}[(R_4 - R_1) + (R_3 - R_1)](239 - 236.67)\right\}$ $2\left\{\frac{1}{2}[(25 - 15.5) + (22.5 - 15.5)](2.33)\right\}$	38.4	0.50	0.99	0.50	1.49
Vent area d (area through grating)	See Section 6.2.1.3.2.4	180	0.83[d]	0.08	0.83	0.91

a. Geometric dimensions are illustrated on Figure 6.2-24.

b. An expansion loss at the vent exit, K_E, which is used as input into RELAP, is based on the equation: $K_E = \left[1 - \frac{Area\ Junction}{Area\ Downstream}\right]^2$

c. An expansion loss at the vent exit with a coefficient of 1.0 is built into THREED.

d. A grating resistance coefficient of 0.33 is included.

Table 6.2-28
VENT AREAS, K-FACTORS, AND VENT FLOW MODELS USED IN THE TWO-NODE CONCRETE SHIELD WALL ANALYSIS (FIVE NODES TOTAL)

Between Nodes	Vent Area (ft^2)	Contraction + Friction K	Expansion K	K Input to THREED	K Input to RELAP	THREED Flow Model
1-2	180	0.83	0.082	0.83	0.912	HVFM-1
1-5	64	0.5	1.0	0.05	1.5	HVFM-1
1-5	93	0.38	1.0	0.38	1.38	a
2-5	202.3	0.5	1.0	0.5	1.5	HVFM-1
2-5	21.4	0.83	1.83	0.83	1.83	a
2-5	21.4	0.83	1.83	0.83	1.83	a
2-3	140	-	-	-	-	1.0 Moody
3-4	256.2	0.1	0.0	0.1	0.1	HVFM-1
4-5	256.2	0.02	1.0	0.02	1.02	HVFM-2

a. Blowout panels designed to release before 5.0 psid.

Table 6.2-29
LENGTH-TO-AREA RATIOS STEAM GENERATOR SUBCOMPARTMENT ANALYSIS WITH RELAP4

Connecting Nodes	L/A (ft^{-1})
1-2	0.044
1-5	0.11
1-5	0.015
2-5	0.055
2-5	0.175
2-5	0.175
2-3	0.071
3-4	0.014
4-5	0.008

Table 6.2-30
VENT AREAS, K-FACTORS, AND VENT FLOW MODELS
USED IN THE SEVEN-NODE DIFFERENTIAL PRESSURE ANALYSIS
ACROSS THE STEAM GENERATOR SUBCOMPARTMENT WALLS

Between Nodes	Vent Area (ft^2)	Contraction + Friction K	Expansion K	K Input to THREED	K Input to RELAP	THREED Flow Model
1-2	189	0.1	0.18	0.1	0.28	HVFM-1
1-4	24	0.47	0.47	0.27	1.38	a
1-5	56	0.36	0.86	0.36	1.22	HVFM-1
1-9	66.1	0.50	0.53	0.50	1.03	HVFM-1
2-3	258	0.10	0.05	0.10	0.15	HVFM-1
2-4	24	0.48	0.91	0.48	1.39	a
2-5	119	0.20	0.72	0.20	0.92	HVFM-1
2-9	68.9	0.65	0.51	0.65	1.16	HVFM-1
3-4	24	0.45	0.91	0.45	1.36	a
3-5	72	0.21	0.82	0.21	1.03	HVFM-1
3-9	45	0.62	0.66	0.62	1.28	HVFM-1
4-5	237	0.02	0.48	0.02	0.50	HVFM-2
4-10	21.4	0.83	1.0	0.83	1.83	b
5-6	690	0.33	0.0	0.33	0.33	HVFM-2
6-7	565	0.13	0.05	0.13	0.18	HVFM-1
6-10	21.4	0.83	1.0	0.83	1.83	b
7-8	140	0.47	0.34	0.47	0.81	HVFM-1
7-10	202.3	0.50	1.0	0.50	1.50	HVFM-1
8-10	256.2	0.10	1.0	0.10	1.50	HVFM-1
9-10	64	0.50	1.0	0.50	1.50	HVFM-1
9-10	93	0.38	1.0	0.38	1.38	b

a. The HVFM-1 flow model is used. Note that no credit is taken for flow through the support structure.
b. Blowout panels are designed to release at or before 4.0 psid.

Table 6.2-31
VENT AREAS, K-FACTORS, AND VENT FLOW MODELS USED IN THE 10-NODE ASYMMETRIC PRESSURE ANALYSIS IN THE STEAM GENERATOR SUBCOMPARTMENT

Between Nodes	Vent Area (ft^2)	Contraction + Friction K	Expansion K	K Input to THREED	K Input to RELAP	THREED Flow Model
1-2	189	0.10	0.18	0.10	0.28	a
1-4	24	0.47	0.91	0.47	1.38	a
1-5	56	0.36	0.25	0.36	0.61	b
1-12	66.1	0.50	0.53	0.50	1.03	HVFM-1
2-3	258	0.10	0.05	0.10	0.15	a
2-4	24	0.48	0.91	0.48	1.39	a
2-6	119	0.20	0.19	0.20	0.39	b
2-12	68.9	0.65	0.51	0.65	1.16	HVFM-1
3-4	24	0.45	0.91	0.45	1.36	a
3-7	72	0.21	0.14	0.21	0.35	b
3-12	45	0.62	0.66	0.62	1.28	HVFM-1
4-8	237	0.02	0.0	0.02	0.02	c
4-13	21.4	0.83	1.0	0.83	1.83	d
5-6	30.6	0.30	0.32	0.30	0.62	HVFM-1
5-8	12.3	0.37	0.80	0.37	1.17	HVFM-1
5-9	94.4	0.33	0.77	0.33	1.1	HVFM-2
6-7	59.5	0.05	0.02	0.05	0.07	HVFM-1
6-8	25.4	0.33	0.60	0.33	0.93	HVFM-1
7-8	14.0	0.24	0.77	0.24	1.01	HVFM-1
7-9	98.1	0.33	0.76	0.33	1.09	HVFM-2
8-9	322.0	0.33	0.34	0.33	0.67	HVFM-2
9-10	565.0	0.13	0.05	0.13	0.18	HVFM-1
9-13	21.4	0.83	1.0	0.83	1.83	d
10-11	140.0	0.47	0.34	0.47	0.81	HVFM-1
10-13	202.3	0.50	1.0	0.50	1.5	HVFM-1
11-13	256.2	0.10	1.0	0.10	1.1	HVFM-1
6-9	179.1	0.33	0.59	0.33	0.92	HVFM-2
12-13	64.0	0.50	1.0	0.50	1.5	HVFM-1
12-13	93.0	0.38	1.0	0.38	1.38	d

a. The HVFM-1 flow model is used. Note that no credit is taken for flow through the steam generator support structure.
b. The HVFM-1 flow model is used to calculate asymmetric loads in Nodes 1 to 4. The Moody flow model with a 1.0 multiplier is used to calculate asymmetric loads in Nodes 5 to 8.
c. The HVFM-2 flow model is used to calculate asymmetric loads in Nodes 1 to 4. The Moody flow model with a 1.0 multiplier is used to calculate asymmetric loads in Nodes 5 to 8.
d. Blowout panels are designed to release at or before 5.0 psid.

Table 6.2-32
ASYMMETRIC PRESSURE ANALYSIS, STEAM GENERATOR SUBCOMPARTMENT PRESSURE DIFFERENTIALS FOR THE 10-NODE MODEL (13 NODES TOTAL)

Description	Peak Calculated Differential Pressure (psid)	Computer Code Yielding the Higher Calculated Differential Pressure
Differential pressure across steam generator subcompartment walls	28.9	THREED
Differential pressure across steam generator supports	17.7	RELAP4
Differential pressure across steam generator and reactor coolant pump	8.3	RELAP4

Table 6.2-33
LENGTH-TO-AREA RATIOS STEAM GENERATOR SUBCOMPARTMENT
ASYMMETRIC PRESSURE ANALYSIS WITH RELAP4
10-NODE CONNECTING MODEL (12 NODES TOTAL)

Connecting Nodes	L/A (ft^{-1})
1-2	0.041
1-4	0.095
1-5	0.075
1-12	0.071
2-3	0.03
2-4	0.57
2-6	0.05
2-12	0.053
3-4	0.125
3-7	0.082
3-12	0.079
4-8	0.039
4-13	0.239
5-6	0.117
5-8	0.313
5-9	0.022
6-7	0.145
6-8	0.222
6-9	0.015
7-8	0.476
7-9	0.021
8-9	0.011
9-10	0.017
9-13	0.199
10-11	0.057
10-13	0.036
11-13	0.021
12-13	0.11
12-13	0.017

Table 6.2-34
VENT AREAS, K-FACTORS, AND VENT FLOW MODELS USED IN THE STEAM GENERATOR UPLIFT ANALYSIS FIVE-NODE MODEL (8 NODES TOTAL)

Between Nodes	Vent Area (ft^2)	Contraction ± Friction K	Expansion K	K Input to THREED	K Input to RELAP	THREED Flow Model	Junction Inertia for RELAP (ft^{-1})
1-7	64.0	0.5	1.0	0.5	1.5	HVFM-1	0.111
1-7	93.0	0.38	1.0	0.38	1.38	a	0.017
2-1	180.0	0.83	0.09	0.83	0.92	HVFM-1	0.024
2-3	492.0	0.34	0.16	0.34	0.5	HVFM-1	0.013
2-7	21.4	0.83	1.0	0.83	1.83	a	0.156
3-4	690.0	0.33	0.0	0.33	0.33	HVFM-2	0.0075
4-5	565.0	0.13	0.09	0.13	0.22	HVFM-1	0.016
4-7	21.4	0.83	1.0	0.83	1.83	a	0.15
5-6	140.0	0.47	0.28	0.47	0.75	HVFM-1	0.054
5-7	202.3	0.5	1.0	0.5	1.5	HVFM-1	0.035
6-7	256.2	0.1	1.0	0.1	1.1	HVFM-1	0.02
8-2	416.0	0.5	1.0	0.5	1.5	HVFM-1	0.043

a. Blowout panels designed to release at or before 5.0 psid.

Table 6.2-35
VENT AREAS, K-FACTORS, AND VENT FLOW MODELS USED IN THE PRESSURIZER CUBICLE SUBCOMPARTMENT ANALYSIS

Between Nodes	Vent Area (ft^2)	Contraction + Friction K	Expansion K	K Input to THREED	K Input to RELAP	THREED Flow Model
1-2	12.5	0.50	0.98	0.50	1.48	HVFM-1[a]
1-2	99.5	0.83	0.82	0.83	1.65	HVFM-1
1-6	21.5	0.50	1.00	0.50	1.50	HVFM-1
1-6	21.4[b]	0.83	1.00	0.83	1.83	HVFM-1
2-3						
(spray)[c]	73.5	0.50	0.86	0.50	1.36	HVFM-1
(surge)[c]	73.5	0.50	0.71	0.50	1.21	a
2-6	108.4	0.50	0.98	0.50	1.48	HVFM-1
2-6	21.4[b]	0.83	1.00	0.83	1.83	HVFM-1
3-4	458.0	0.33	0.00	0.33	0.33	HVFM-2
3-6	37.0	0.50	0.99	0.50	1.49	HVFM-1
4-5	422.0	0.33	0.04	0.33	0.37	HVFM-2
4-6	42.0	0.50	0.99	0.50	1.49	HVFM-1
5-6	279.0	0.50	0.96	0.50	1.46	HVFM-1

a. HVFM-1 is used to maximize the differential pressure in the lower pressurizer subcompartment (surge line DER). Frictionless Moody flow is used to maximize differential pressure in the upper pressurizer sub-compartment (surge line DER).

b. Vent areas and K-factors for these blowout panels are values used after 5.0 psid is attained.

c. The loss coefficient is dependent on the direction of flow, which is determined by the break type, i.e., spray line or surge line.

Table 6.2-36
LENGTH-TO-AREA RATIOS PRESSURIZER CUBICLE ANALYSIS WITH RELAP4

Connecting Nodes	L/A (ft^{-1})
1-2	0.1792
1-2	0.0392
1-6	0.10522
1-6	0.10566
2-3	0.04501
2-6	0.03466
2-6	0.10967
3-4	0.01325
3-6	0.03786
4-5	0.01576
4-6	0.03547
5-6	0.01692

Table 6.2-37

MAJOR PIPING PENETRATIONS THROUGH THE REACTOR CONTAINMENT STRUCTURE

Service	No. of Penetrations	Line Size (in.)	Line Status			Isolation Valves Provided		Actuation Signal[a]	Fluid	Temp.[b]	Sealed System[c]		Design Criteria Met, or Exception	Type of Test
			Normal	Shutdown	Incident	Inside	Outside				Inside	Outside		
Service water from recirculating spray coolers[e]	4	16	Open	Open	Open	None	Remote manual	CDA	Liquid	Cold	Yes	No	57	C
High-head safety injection—boron injection to reactor coolant system	1	3	Closed	Closed	Open	Check	2-Remote manual	SI	Liquid	Cold	No	No	6.2.4.2 (5)	None
Reactor coolant pump seal water supply	3	2	Open	Open	Open	Check and manual	Manual (AC)	None	Liquid	Cold	No	No	6.2.4.2 (6)	None
Low head safety injection pump discharge to reactor coolant system hot legs	2	6	Closed	Closed	Open	Check	Remote manual	None	Liquid	Cold	No	No	6.2.4.2 (5)	None
Quench spray pump discharge	2	8	Closed	Closed	Open	Check	Remote manual	CDA	Liquid	Cold	No	No	6.2.4.2 (7)	C
Recirculation spray pump discharge	2	10	Open	Closed	Open	Check	Remote manual	CDA	Liquid	Cold	No	Yes	6.2.4.2 (7)	None
Service water to recirculation spray coolers	4	16	Closed	Closed	Open	Check	Remote manual	CDA	Liquid	Cold	Yes	No	57	C
Recirculation spray pump suction line[d]	2[d]	12	Open	Open	Open	None	Remote manual	CDA	Liquid	Cold	No	Yes	6.2.4.2 (8)	None
Low head safety injection pump suction from containment sump	2	12	Closed	Closed	Open	None	Remote manual	None	Liquid	Cold	No	No	6.2.4.2 (8)	None
High-head safety injection pump discharge to reactor coolant system except boron injection line	3	3	Closed	Closed	Open	Check	Remote manual	None	Liquid	Cold	No	No	6.2.4.2 (5)	None
Low-head safety injection pump discharge to reactor coolant system cold legs	1	6	Open	Open	Open	Check	Remote manual	None	Liquid	Cold	No	No	6.2.4.2 (5)	None

a. CIA - Containment isolation Phase A.
CIB - Containment isolation Phase B.
CDA - Containment depressurization actuation.
IHH - Intermediate high-high containment pressure (between high and high-high).
SI - Safety injection.

b. Cold - below 250°F, hot - above 250°F.

c. A system is considered sealed if:
1. It is not open to the atmosphere.
2. It is seismically designed.
3. It is not connected to the RCPB.

d. Recirculation spray pump suction line and casing cooling pump discharge line share the same penetration.

e. Type “C” leakage valves are not required to be added to the Type “A” leakage rate.

f. This modification was cancelled, only the containment penetration was installed.

g. Penetrations No. 103 and No. 104 are isolated and partially drained during normal power operation (Reference 40).

h. Fluid inside containment is isolated from fluid outside containment by a sensor bellow (inside) and hydraulic isolator (outside).

Table 6.2-37 (continued)
MAJOR PIPING PENETRATIONS THROUGH THE REACTOR CONTAINMENT STRUCTURE

Service	No. of Penetrations	Line Size (in.)	Line Status: Normal	Line Status: Shutdown	Line Status: Incident	Isolation Valves Provided: Inside	Isolation Valves Provided: Outside	Actuation Signal[a]	Fluid	Temp.[b]	Sealed System[c]: Inside	Sealed System[c]: Outside	Design Criteria Met, or Exception	Type of Test
Casing cooling pump discharge	2[d]	6	Closed	Closed	Open	None	2-Remote manual	CDA	Liquid	Cold	No	No	6.2.4.2 (7)	None
Component cooling water to residual heat removal system and excess letdown heat exchanger	2	18	Open	Open	Closed	None	Check	None	Liquid	Cold	Yes	No	6.2.4.2 (2)	C
Component cooling water to containment air recirculation cooling coils	3	6	Open	Open	Closed	None	Check	None	Liquid	Cold	Yes	No	6.2.4.2 (2)	C
Feedwater lines	3	16	Open	Closed	Open	None	Check	None	Liquid	Hot	Yes	No	6.2.4.2 (2)	None
Chemical feed lines	3	3/4	Open	Int.	Closed	None	Check	None	Liquid	Cold	Yes	No	6.2.4.2 (2)	None
Component cooling water from residual heat removal system and excess letdown heat exchanger	2	18	Open	Open	Closed	None	Auto-trip	CIB	Liquid	Cold	Yes	No	57	C
Chilled cooling water from containment air recirculating cooling coils	3	6	Open	Open	Closed	Auto-trip	Auto-trip	CIB	Liquid	Cold	Yes	No	56 (4)	C
Steam generator blowdown	3	3	Open	Closed	Closed	1-Auto-trip	1-Auto-trip	CIA, CIB aux. feed start (excess flow inside valve only)	Liquid	Hot	Yes	No	57	C
Main steam lines	3	32	Open	Closed	Closed	None	Throttle	Steam line isol. or IHH	Gas	Hot	Yes	No	6.2.4.2(3)	None
Residual heat removal sample lines	1	3/8	Closed	Open	Closed	Auto-trip	Auto-trip	SI	Liquid	Hot	No	No	6.2.4.2 (4)	C
Reactor containment leakage monitoring lines to reference system	2	1/4	Closed	Closed	Closed	None	2-Auto-trip	CIA	Gas	Cold	No	No	6.2.4.2 (2)	C
Steam generator surface sample	1	3/8	Int.	Int.	Closed	Auto-trip	Auto-trip	CIA	Liquid	Hot	Yes	No	56 (4)	C

a. CIA - Containment isolation Phase A.
   CIB - Containment isolation Phase B.
   CDA - Containment depressurization actuation.
   IHH - Intermediate high-high containment pressure (between high and high-high).
   SI - Safety injection.

b. Cold - below 250°F, hot - above 250°F.

c. A system is considered sealed if:
   1. It is not open to the atmosphere.
   2. It is seismically designed.
   3. It is not connected to the RCPB.

d. Recirculation spray pump suction line and casing cooling pump discharge line share the same penetration.

e. Type “C” leakage valves are not required to be added to the Type “A” leakage rate.

f. This modification was cancelled, only the containment penetration was installed.

g. Penetrations No. 103 and No. 104 are isolated and partially drained during normal power operation (Reference 40).

h. Fluid inside containment is isolated from fluid outside containment by a sensor bellow (inside) and hydraulic isolator (outside).

Table 6.2-37 (continued)
MAJOR PIPING PENETRATIONS THROUGH THE REACTOR CONTAINMENT STRUCTURE

Service	No. of Penetrations	Line Size (in.)	Line Status: Normal	Line Status: Shutdown	Line Status: Incident	Isolation Valves Provided: Inside	Isolation Valves Provided: Outside	Actuation Signal[a]	Fluid	Temp.[b]	Sealed System[c]: Inside	Sealed System[c]: Outside	Design Criteria Met, or Exception	Type of Test
Pressurizer liquid surface sample	1	3/8	Int.	Int.	Closed	Auto-trip	Auto-trip	CIA	Liquid	Hot	No	No	55 (4)	C
Pressurizer vapor space sample	1	3/8	Int.	Int.	Closed	Auto-trip	Auto-trip	CIA	Liquid	Hot	No	No	55 (4)	C
Primary drain transfer pump discharge	1	2	Int.	Int.	Closed	Auto-trip	Auto-trip	CIA	Liquid	Cold	No	No	56 (4)	C
Reactor containment sump pump discharge to waste drain tanks	1	2	Int.	Int.	Closed	Auto-trip	Auto-trip	CIA	Liquid	Cold	No	No	56 (4)	C
Air radiation containment gaseous activity and particle monitor supply	1	1	Open	Open	Closed	Auto-trip	Auto-trip	CIA	Gas	Cold	No	No	56 (4)	C
Primary vent header	1	1½	Int.	Closed	Closed	Auto-trip	Auto-trip	CIA	Gas	Cold	No	No	56 (4)	C
Safety injection accumulators to waste gas charcoal filters	1	1	Int.	Closed	Closed	Auto-trip	Auto-trip	CIA	Gas	Cold	No	No	55 (4)	C
Pressurizer relief tank sample	1	3/8	Int.	Int.	Closed	Auto-trip	Auto-trip	CIA	Gas	Cold	No	No	55 (4)	C
Reactor coolant samples	2	3/8	Open	Open	Closed	Auto-trip	Auto-trip	CIA	Liquid and gas	Hot	No	No	55 (4)	C
Reactor containment leakage monitoring lines to open taps	4	3/8	Closed	Closed	Closed	None	2-Auto-trip	CIA	Gas	Cold	No	No	6.2.4.2 (1)	C
Containment vacuum pump suction	2	2	Open	Open	Closed	None	2-Auto-trip	CIA	Gas	Cold	No	No	6.2.4.2 (1)	C
Charging line	1	3	Throttle	Closed	Closed	Check	Remote manual (AC)	SI	Liquid	Cold	No	No	55 (4)	None
Reactor coolant pump cooling water in	3	8	Open	Closed	Closed	Check	Auto-trip	CIB	Liquid	Cold	No	No	55 (4)	C
Air radiation monitor return	1	1	Open	Open	Closed	Check	2-Auto-trip	CIA	Gas	Cold	No	No	56 (4)	C
Primary grade water to pressurizer relief tank	1	3	Int.	Closed	Closed	Check	Auto-trip	CIA	Liquid	Cold	No	No	55 (4)	C
Nitrogen to pressurizer relief tank and safety injection accumulators	1	1	Int.	Closed	Closed	Check	Auto-trip	CIA	Gas	Cold	No	No	55 (4)	C
Condenser air ejector vent	1	6	Closed	Closed	Closed	Check	2-Auto-trip	CIA	Gas	Cold	No	No	56 (4)	C
Steam generator wet layup system	3	2	Closed	Open	Closed	Manual (AC)	Manual (AC)	None	Liquid	Cold	Yes	No	56 (1)	C

a. CIA - Containment isolation Phase A.
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CDA - Containment depressurization actuation.
IHH - Intermediate high-high containment pressure (between high and high-high).
SI - Safety injection.

b. Cold - below 250°F, hot - above 250°F.

c. A system is considered sealed if:
1. It is not open to the atmosphere.
2. It is seismically designed.
3. It is not connected to the RCPB.

d. Recirculation spray pump suction line and casing cooling pump discharge line share the same penetration.

e. Type “C” leakage valves are not required to be added to the Type “A” leakage rate.

f. This modification was cancelled, only the containment penetration was installed.

g. Penetrations No. 103 and No. 104 are isolated and partially drained during normal power operation (Reference 40).

h. Fluid inside containment is isolated from fluid outside containment by a sensor bellow (inside) and hydraulic isolator (outside).

Table 6.2-37 (continued)
MAJOR PIPING PENETRATIONS THROUGH THE REACTOR CONTAINMENT STRUCTURE

Service	No. of Penetrations	Line Size (in.)	Line Status: Normal	Line Status: Shutdown	Line Status: Incident	Isolation Valves Provided: Inside	Isolation Valves Provided: Outside	Actuation Signal[a]	Fluid	Temp.[b]	Sealed System[c]: Inside	Sealed System[c]: Outside	Design Criteria Met, or Exception	Type of Test
Reactor coolant pump thermal barrier cooling water out	1	4	Open	Open	Closed	Auto-trip	Auto-trip	CIB	Liquid	Cold	No	No	55 (4)	C
Reactor coolant pump seal water return	1	3	Open	Open	Closed	Auto-trip	Auto-trip	CIA	Liquid	Cold	No	No	56 (4)	C
Reactor coolant pump bearing and shroud cooling water out	3	8	Open	Open	Closed	Auto-trip	Auto-trip	CIB	Liquid	Cold	No	No	56 (4)	C
Reactor coolant letdown	1	2	Open	Closed	Closed	Auto-trip	Auto-trip	CIA	Liquid	Hot	No	No	55 (4)	C
Safety injection accumulator makeup	1	1	Int.	Int.	Closed	Check	Manual (AC)	None	Liquid	Cold	No	No	55 (2)	C
Residual heat removal system to refueling water storage tank	1	6	Closed	Int.	Closed	Manual (AC)	Manual (AC)	None	Liquid	Cold	No	No	55 (1)	C
Service air line	1	2	Closed	Open	Closed	Manual (AC)	Manual (AC)	None	Gas	Cold	No	No	56 (1)	C
Loop fill header	1	2	Closed	Int.	Closed	Check)	Remote (AC) manual	None	Liquid	Cold	No	No	55 (2)	None
Vent line from primary vent pot	1	2	Closed	Closed	Closed	Manual (AC)	Manual (AC)	None	Gas	Cold	No	No	56 (2)	C
Fuel transfer tube	1	20	Closed	Open	Closed	Blind flange		None	Liquid	Cold	No	No	6.2.4.2 (9)	B
Purge exhaust/supply/bypass	2	36	Closed	Open	Closed	Remote (AC) manual	Remote (AC) manual	None	Gas	Cold	No	No	56 (1)	C
Containment air ejector suction	1	8	Closed	Closed	Closed	Remote (AC) manual	Manual (AC)	None	Gas	Cold	No	No	56 (1)	C
Containment instrument air compressor suction (Unit 2)[f]	1	3	Open	Open	Closed	Auto-trip	Auto-trip	CIA	Gas	Cold	No	No	56 (4)	C
Containment instrument air supply	1	2	Open	Open	Closed	Check	2-Auto-trip	CIB	Gas	Cold	No	No	56 (4)	C
Fire protection system	1	4	Closed	Closed	Closed	Check	2-Manual (AC)	None	Liquid	Cold	No	No	56 (2)	C
Refueling purification lines[g]	2	6	Closed	Open	Closed	Manual (AC)	Manual (AC)	None	Liquid	Cold	No	No	56 (1)	C
Safety injection test line	1	2	Int.	Closed	Closed	Auto-trip	Auto-trip	SI	Liquid	Cold	No	No	55 (4)	C
Dead weight pressure calibrator	1	1/8	Closed	Int.	Closed	None	2-Manual (AC)	None	Liquid	Hot	No	No	6.2.4.2 (10)	C

a. CIA - Containment isolation Phase A.
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SI - Safety injection.

b. Cold - below 250°F, hot - above 250°F.

c. A system is considered sealed if:
1. It is not open to the atmosphere.
2. It is seismically designed.
3. It is not connected to the RCPB.

d. Recirculation spray pump suction line and casing cooling pump discharge line share the same penetration.

e. Type “C” leakage valves are not required to be added to the Type “A” leakage rate.

f. This modification was cancelled, only the containment penetration was installed.

g. Penetrations No. 103 and No. 104 are isolated and partially drained during normal power operation (Reference 40).

h. Fluid inside containment is isolated from fluid outside containment by a sensor bellow (inside) and hydraulic isolator (outside).

Table 6.2-37 (continued)

MAJOR PIPING PENETRATIONS THROUGH THE REACTOR CONTAINMENT STRUCTURE

Service	No. of Penetrations	Line Size (in.)	Line Status			Isolation Valves Provided		Actuation Signal[a]	Fluid	Temp.[b]	Sealed System[c]		Design Criteria Met, or Exception	Type of Test
			Normal	Shutdown	Incident	Inside	Outside				Inside	Outside		
Hydrogen analyzer suction	2	3/8	Closed	Closed	Open	Remote (AC) manual	1-Remote manual (AC)	None	Gas	Cold	No	Yes	6.2.4.2 (11)	C
Hydrogen analyzer discharge	2	3/8	Closed	Closed	Open	Check	2-Remote manual (AC)	None	Gas	Cold	No	Yes	6.2.4.2 (11)	C
Hydrogen recombiner suction	2	2	Closed	Closed	Open	None	2-Remote manual (AC)	None	Gas	Cold	No	Yes	6.2.4.2 (11)	C
Hydrogen recombiner discharge	2	2	Closed	Closed	Open	Check	2-Remote manual (AC)	None	Gas	Cold	No	Yes	6.2.4.2 (11)	C
Reactor vessel level indicator system	2	1/4	Closed	Closed	Closed	h	h	CIA	Liquid	Hot	No	No	55 (4)	None
Postaccident sample system	1	3/4	Closed	Closed	Open	None	2-Auto-trip	CIA	Liquid and gas	Hot	No	No	57	C

a. CIA - Containment isolation Phase A.
CIB - Containment isolation Phase B.
CDA - Containment depressurization actuation.
IHH - Intermediate high-high containment pressure (between high and high-high).
SI - Safety injection.

b. Cold - below 250°F, hot - above 250°F.

c. A system is considered sealed if:
1. It is not open to the atmosphere.
2. It is seismically designed.
3. It is not connected to the RCPB.

d. Recirculation spray pump suction line and casing cooling pump discharge line share the same penetration.

e. Type “C” leakage valves are not required to be added to the Type “A” leakage rate.

f. This modification was cancelled, only the containment penetration was installed.

g. Penetrations No. 103 and No. 104 are isolated and partially drained during normal power operation (Reference 40).

h. Fluid inside containment is isolated from fluid outside containment by a sensor bellow (inside) and hydraulic isolator (outside).

Table 6.2-38
INSTRUMENTATION AVAILABLE TO MONITOR RECIRCULATION STATUS

Pump	Reference Drawing[a]	Instrumentation Available	Sensor Location	Readout Location
IRS (2-RS-P-1A)	Reference Drawing 1	Pump/motor vibration (VT-RS-200A)	Containment (pump motor)	Main control room (alarm only)
	Reference Drawing 1	Pump discharge Pressure (PI-RS-252A)	Containment (line 10-RS-401-153A-Q2)	Main control room
IRS (2-RS-P-1B)	Reference Drawing 1	Pump/motor vibration (VT-RS-200B)	Containment (pump motor)	Main control room (alarm only)
	Reference Drawing 1	Pump discharge pressure PI-RS-252B)	Containment (line 10-RS-402-153A-Q2)	Main control room
All pumps	Reference Drawing 1	Sump level (LI-251A and B)	Containment sump	Main control room
ORS (2-RS-P-2A)	Reference Drawing 1	Pump/motor vibration (VT-RS-201A)	Valve pit building (motor)	Alarm in main control room
	Reference Drawing 1	Pump discharge pressure	Valve pit building (line 10-RS409-153A-Q2)	Main control room
ORS (2-RS-P-2B)	Reference Drawing 1	Pump/motor vibration	Valve pit building (motor)	Alarm in main control room
	Reference Drawing 1	Pump discharge pressure	Valve pit building (line10-RS410-153A-Q2)	Main control room

a. Drawings show labels for Unit 1 components; Unit 2 is similar. Unit 2 numbers are given.

Table 6.2-39
CONTAINMENT DEPRESSURIZATION SYSTEM DESIGN DATA

Quench Spray Pump	
Number (per unit)	2
Type	Horizontal centrifugal
Rated flow	2000 gpm each
Rated head	265 ft
Brake horsepower	185
Seal	Mechanical
Design pressure	150 psig
Material	
Pump casing	A296-CF8M
Shaft	303 stainless steel
Impeller	A296-CF8M
Quench Spray Pump Motor	
Number (per unit)	2
Horsepower	250 hp
Electrical characteristics	460V, 3 phase, 60 Hz
Service factor	1.15
Insulation	Class B
Refueling Water Storage Tank	
Number (per unit)	1
Usable volume	450,000 gal
Boron concentration	2600 to 2800 ppm
Design pressure	Hydraulic head
Design temperature	150°F
Operating pressure	Hydraulic head
Operating temperature	40-50°F
Material	A240-T304L
Design code	API STD-650
Recirculation Spray Pump (Inside Containment)	
Number (per unit)	2
Type	Vertical turbine
Rated flow	3300 gpm
Rated head	269 ft
Brake horsepower	279
Seal	Throttle bushing

Table 6.2-39 (continued)
CONTAINMENT DEPRESSURIZATION SYSTEM DESIGN DATA

Recirculation Spray Pump (Inside Containment) (continued)		
Material		
Shaft	A564-T630	
Pump casing	A351-CF8	
Impeller	A351-CF8	
Recirculation Spray Pump Motor (Inside Containment)		
Number (per unit)	2	
Horsepower	300	
Electrical characteristics	460V, 3 phase, 60 Hz	
Service factor	1.0	
Insulation	Class H	
Recirculation Spray Pump (Outside Containment)		
Number (per unit)	2	
Type	Vertical turbine	
Rated flow	3700 gpm	
Rated head	286.7 ft	
Brake horsepower	334	
Seal	Tandem mechanical	
Material		
Shaft	A564-T630	
Pump casing	A351-CF8	
Impeller	A351-CF8	
Recirculation Spray Pump Motor (Outside Containment)		
Number (per unit)	2	
Horsepower	400	
Electrical characteristics	4000V, 3 phase, 60 Hz	
Service factor	1.15	
Insulation	Class B	
Recirculation Spray Coolers		
Number (per unit)	4	
Design duty each	56,835,000 Btu/hr	
	Shell	Tube
Fluid flowing	Recirculation spray water	Service water
Design pressure	150 psig	150 psig
Design temperature	280°F	280°F
Operating pressure	100 psig	85 psig
Inlet temperature max	206°F	110°F

Table 6.2-39 (continued)
CONTAINMENT DEPRESSURIZATION SYSTEM DESIGN DATA

Recirculation Spray Coolers (continued)		
	Shell	Tube
Material	304 SS	304 SS
Refueling Water Chemical Addition Tank		
Number (per unit)	1	
Type	Vertical cylindrical	
Operating volume	4800-5500 gal	
Design pressure	Hydraulic head	
Design temperature	150°F	
Material	304 SS	
Design code	ASME Section VIII	
Operating pressure	Atmospheric	
Operating temperature	125°F	
NaOH concentration	12-13%	
Refueling Water Chemical Addition Tank Recirculating Pump		
Number (per unit)	1	
Type	Horizontal centrifugal	
Rated flow	100 gpm	
Rated head	64 ft	
Brake horsepower	3.96	
Design pressure	275 psig	
Material		
Pump casing	316 SS	
Impeller	316 SS	
Refueling Water Recirculating Pump		
Number (per unit)	2	
Type	2-speed vertical centrifugal	
	High Speed	Low Speed
Rated flow	520 gpm	72 gpm
Rated head	230 ft	118 ft
Brake horsepower	26.7	1.7
Seal	Mechanical	
Design pressure	185 psig	
Material		
Pump casing	316 SS	
Shaft	416 SS	
Impeller	316 SS	

Table 6.2-39 (continued)
CONTAINMENT DEPRESSURIZATION SYSTEM DESIGN DATA

Refueling Water Storage Tank Cooler		
Number (per unit)	2	
Design duty each	2,076,000 Btu/hr	
	Shell	Tube
Fluid flowing	Chilled water	Borated water
Design pressure	150 psig	150 psig
Design temperature	200°F	200°F
Operating pressure	120 psig	85 psig
Inlet temperature	60°F	113.4°F
Material	SA-53	304 SS
Refueling Water Refrigeration Unit		
Number (per unit)	2	
Type	Compression, air cooled	
Fluid	Borated water	
Flow	72 gpm	
Design pressure	200 psig	
Design temperature	300°F	
Inlet pressure max.	125 psig	
Material	316 SS	
Fan power	2.8 kw	
Compressor power	26.8 kw	
Quench Spray Pump Discharge Strainers		
Number (per unit)	2	
Fluid	Borated water	
Flow	2200 gpm	
Design differential pressure	150 psi	
Operating pressure at 2200 gpm	100 psig	
Operating temperature	45-120°F	
Operating differential pressure at 2200 gpm	1.0 psi	
Material	304 SS	
Piping		
Piping is designed to the USA Standard Code for Pressure Piping - ANSI B31.7 - Nuclear Power Piping. RWST Cooling Subsystem Piping is designed to ANSI B31.1 - Power Piping.		

Table 6.2-39 (continued)
CONTAINMENT DEPRESSURIZATION SYSTEM DESIGN DATA

Recirculation Spray System Strainer Assembly	
Number	1 (for both ORS and IRS Systems)
Material	SS 304 or 304L
Structural DP	9.0 psid
Perforation	0.0625 inches (nominal)
Operating Pressure	9.0-59.7 psia
Operating Temperature	75-280°F
Fluid Flowing	Borated Water

Table 6.2-40
RECIRCULATION SPRAY SUBSYSTEMS LEAKAGE OUTSIDE CONTAINMENT

Item	No. of Items	Type of Leakage Control and Unit Leakage Rate Used in the Analysis [a]	Uncollected Leakage (cc/hr)	Leakage to Vent and Drain System (cc/hr)
Recirculation spray pumps (outside)	2	No leakage of spray water due to tandem seal arrangement	0	0
Flanges:		Adjusted to zero leakage following any test - assumed 10 drops per minute per flange		
a. Pumps	4		120	0
b. Valves - bonnet to body (larger than 2 in.)	4		115	
Miscellaneous small valves	2	Flanged body, packed stems - 1 drop per minute	6	0
Total			241 cc/hr (0.064 gal/hr)	

a. Unit leakage rates are original design criteria. The actual allowable leakage for each leakage control component may exceed the original leakage rate indicated as long as the total ECCS recirculation loop leakage and the recirculation spray subsystem leakage outside of containment does not exceed the curve of allowable ECCS leakage which corresponds to the control room unfiltered inleakage. A curve of allowables ECCS leakage for 250 cfm of unfiltered control room inleakage is shown in Figure 15.4-78 and its use is discussed in Sections 15.4.1.9.6 and 15.4.1.9.8.

Table 6.2-41
CONSEQUENCE OF COMPONENT MALFUNCTIONS

	Components	Malfunction	Comments and Consequences
1.	Quench spray pumps	Pump casing ruptures	The casing is designed for 250°F temperature; design test pressure is 150 psig and maximum test pressure is 225 psig. These conditions exceed those that could occur during any operating condition. The casings are made from stainless steel (ASTM A296-CF8M); this metal has excellent corrosion-erosion resistance and produces sound castings. The pumps conform to Class II of the ASME Code for Pumps and Valves for Nuclear Power. Rupture of the pump casing is, therefore, not considered credible.
2.	Quench spray pumps	Pump fails to start	The quench spray system has two parallel 100%-capacity pumps. Sufficient capacity is provided by one pump in the case of failure of the other pump.
3.	Quench spray pump discharge valve	Valve fails to open	The quench spray system consists of two 100%-capacity subsystems. If the quench spray pump discharge valve in one subsystem fails to open, the remaining subsystem is available.
4.	Quench spray pump discharge valve	Rupture of valve body	Valve body is designed for 150 psig. The castings are made from stainless steel; this material has excellent corrosion-erosion resistance and produces sound castings. Rupture of valve body is not considered credible.
5.	Quench spray pump discharge valve	Weight-loaded valve in pump discharge sticks closed	Valve is checked each refueling shutdown. In addition, a parallel 100%-capacity quench spray subsystem is available.
6.	Quench spray piping	Pipe rupture	Pipe material has maximum permissible operating conditions of 100°F temperature and 275-psig pressure. These conditions exceed those that could occur during operation. The piping is fabricated of Type 304 stainless steel; this metal has corrosion-erosion resistance. Piping is designed for Seismic Class I. The piping is fabricated in accordance with ANSI B31.7 (ANSI B31.1-1967 for RWST cooling subsystem). Pipe rupture is not considered credible.

Table 6.2-41 (continued)
CONSEQUENCE OF COMPONENT MALFUNCTIONS

Components	Malfunction	Comments and Consequences
7. Recirculation spray pump	Pump fails to start	Four recirculation spray pumps of approximation 50% capacity are provided.
8. Recirculation spray cooler	Tube or shell rupture	Four 50%-capacity recirculation spray coolers are provided. The recirculation spray coolers are designed to the ASME Code Section III C and Seismic Class I. Rupture is considered unlikely. However, in the event of a rupture, motor-operated valves are provided in the service water system to isolate the cooler and prevent further leakage into the service water system.
9. Outside recirculation spray pumps	Rupture of pump can	The can is fabricated of ASME A-240 Type 304 stainless steel; this metal is corrosion resistant. The cans are missile-protected and set in concrete. Rupture of the pump can is not considered credible.
10. Recirculation spray piping	Rupture of piping	Piping is fabricated of Type 304 stainless steel and designed to Seismic Class I criteria, and is missile-protected. The piping is fabricated in accordance with ANSI B31.7. Rupture of the piping is not considered credible. Nevertheless, isolation valves are provided to isolate the major portions of recirculation piping outside the containment.
11. Motor-operated valves	Loss of power to one valve due to failure of electric bus	Redundant parallel valves are provided where valves are required to open on a CDA signal. Electric power to these valves is supplied from separate buses. Other valves are left open during normal plant operation to ensure against failure to open
12. Automatic electric and control instrumentation trains to actuate Engineered Safeguards equipment	Failure of one train	Redundant train will actuate redundant equipment.

Table 6.2-41 (continued)
CONSEQUENCE OF COMPONENT MALFUNCTIONS

Components	Malfunction	Comments and Consequences
13. Spray nozzles	Spray nozzles plugged	Filters are provided in the discharge of the quench spray pumps. Strainer modules are provided in the suction of recirculation spray pumps. The filters are small enough to prevent any material that could plug the spray nozzles from passing through.
14. Containment sump strainer modules and fins	Fin or strainer module failure	The fins and strainer modules are designed such that they can withstand full debris loading and have sufficiently large perforated fin area available to compensate for debris blockage. The strainers are capable of withstanding the force of full debris loading and other conditions including seismic events.

Table 6.2-42
IODINE ACTIVITY ON THE CHARCOAL ADSORBER BANK IN THE AUXILIARY BUILDING FILTERS FOLLOWING A POSTULATED FUEL HANDLING ACCIDENT

Time (hr)	0+	10	100	1000
		Activity (Ci)		
I-131	4.2+02	4.1+02	3.0+02	1.2+01
I-132	3.8+02	1.8+01	a	
I-133	4.9+01	3.6+01	1.82+00	
I-135	4.0-02	1.4-02	1.3-06	

a. Implies < 1.0-06 Ci.

Table 6.2-43
DOSE RATES AT SURFACE OF HEPA/CHARCOAL FILTER BANK SHIELDS IN THE AUXILIARY BUILDING FOLLOWING A FUEL HANDLING ACCIDENT

Time (hr)	0	10	100	1000
Dose rate (mrem/hr)	430	42	36	0.46

Table 6.2-44

POST-DBA HYDROGEN PURGE OF CONTAINMENT HALOGENS ACCUMULATED ON THE PROCESS VENT FILTER

(Time = 0 is start of purge)

T(J) (hr)		0	2	8	44	168	700	2040
	Activity at Time T(J) (Ci)							
I-131	0.0	0.0	0.3441E04	0.1341E05	0.3749E05	0.1397E06	0.7739E05	0.685E03
I-132	0.0	0.0	0.2734E04	0.1032E05	0.2650E05	0.4600E05	0.2574E04	0.1095E01
I-133	0.0	0.0	0.2128E03	0.6950E03	0.1214E04	0.6534E02	0.1091E03	0.3821E-24
I-134	0.0	0.0	0.1519E-37	0.5221E-39	0.4852E-44	0.0	0.0	0.0
I-135	0.0	0.0	0.3630E-01	0.7774E-01	0.4407E-01	0.9513E-07	0.1023E-24	0.0
Total			0.6386E04	0.2442E05	0.6520E05	0.1858E06	0.7996E05	0.6853E03

Table 6.2-45
COMPLIANCE WITH REGULATORY GUIDE 1.52

Regulatory Guide 1.52		Auxiliary Building Filtration System	Control Room Filtration System
Rev. No.	Position No.		
1	C.1.a	See explanatory note C.1.a	See explanatory note C.1.a
1	C.1.b	Complies	Complies
1	C.1.c	Complies	Complies
1	C.1.d	See explanatory note C.1.d	See explanatory note C.1.d
1	C.1.e	Complies	Complies
1	C.2.a	See explanatory note C.2.a	See explanatory note C.2.a
1	C.2.b	See explanatory note C.2.b	Complies
1	C.2.c	See explanatory note C.2.c	Complies
1	C.2.d	See explanatory note C.1.a	See explanatory note C.1.a
1	C.2.e	Complies	Complies
1	C.2.f	See explanatory note C.2.f	Complies
1	C.2.g	See explanatory note C.2.g	See explanatory note C.2.g
1	C.2.h	See explanatory note C.2.h	Complies
2	C.2.i	See explanatory note C.2.i	See explanatory note C.2.i
2	C.2.j	See explanatory note C.2.j	Complies
2	C.2.k	Complies	Complies
2	C.2.l	See explanatory note C.2.l	See explanatory note C.2.l
1	C.3.a	See explanatory note C.3.a	Complies
1	C.3.b	Complies	Complies
1	C.3.c	See explanatory note C.3.c	See explanatory note C.3.c
1	C.3.d	See explanatory note C.3.d	See explanatory note C.3.d
1	C.3.e	See explanatory note C.3.e	See explanatory note C.3.e
1	C.3.f	Complies	Complies
1	C.3.g	Complies	Complies
1	C.3.h	See explanatory note C.3.h	See explanatory note C.3.h
1	C.3.i	See explanatory note C.3.i	See explanatory note C.3.i
1	C.3.j	See explanatory note C.3.j	See explanatory note C.3.j
1	C.3.k	See explanatory note C.3.k	See explanatory note C.3.k
1	C.3.l	Complies	Complies
1	C.3.m	Complies	Complies
1	C.3.n	See explanatory note C.3.n	See explanatory note C.3.n

Table 6.2-45 (continued)
COMPLIANCE WITH REGULATORY GUIDE 1.52

Regulatory Guide 1.52		Auxiliary Building Filtration System	Control Room Filtration System
Rev. No.	Position No.		
1	C.3.o	Complies	Complies
2	C.3.p	See explanatory note C.3.p	See explanatory note C.3.p
1	C.4.a	See explanatory note C.4.a	See explanatory note C.4.a
1	C.4.b	See explanatory note C.4.a	See explanatory note C.4.a
1	C.4.c	Complies	See explanatory note C.4.c
1	C.4.d	See explanatory note C.4.d	See explanatory note C.4.d
1	C.4.e	See explanatory note C.4.e	See explanatory note C.4.e
1	C.4.f	Complies	Complies
2	C.5.a	Complies	Complies
1	C.5.b	Complies	Complies
2	C.5.c	See explanatory note C.5.c	See explanatory note C.5.c
2	C.5.d	See explanatory note C.5.d	See explanatory note C.5.d
2	C.6.a	See explanatory notes C.6.a and C.3.i	See explanatory note C.6.a and C.6.i
2	C.6.b	See explanatory note C.6.b	See explanatory note C.6.b

Table 6.2-45 (continued)
COMPLIANCE WITH REGULATORY GUIDE 1.52

Explanatory Notes:

C.1.a Since the auxiliary building and control room filter assemblies are not located inside the reactor containment, the maximum pressure differential across the filter housing will be due solely to the fans.

C.1.d The atmosphere cleanup systems are compatible with other engineered safety features. However, there is no need for them to be compatible with the containment spray system, as they are not located within the containment structure.

C.2.a The exceptions for the auxiliary building filtration system are the single dampers and ductwork to/from the filter headers, and the heaters and demisters associated with the charcoal filters. The dampers do contain redundant actuation signals and SOVs. Sections of ductwork associated with the redundant auxiliary building filters are common between filters, including common headers for filter inlet and outlet. The demisters and HEPA filters are utilized as stated in Section 6.2.3.2. Heaters are included in the auxiliary building filtration system to control humidity during normal operation. These heaters are not powered by safety related power. The exception for the control room filtration system is that no HEPA filters are supplied downstream of the charcoal adsorbers.

C.2.b The redundant air filtration systems are not protected against missiles, but are physically separated by a concrete block wall. The common ductwork and single isolation dampers to/from the filter headers are not protected. Filter inlet/outlet header isolation from the spaces is provided by single dampers. System ductwork between the Auxiliary Building and the Safeguards Area is not protected from missiles. However, there is no requirement that a missile be considered in conjunction with a design basis LOCA.

C.2.c Portions of the fuel building ventilation system which support filtration of the exhaust (including the fans) are not Seismic Category I.

C.2.f The volumetric air flow of a single filter train in the auxiliary building, as furnished, is 39,200 cfm vs. the 30,000-cfm limit recommended in the regulatory guide. The assembly is three HEPA filters high, which allows for ease of maintenance. The width is 11, instead of the recommended 10, HEPA filters, which is not detrimental to maintaining the assemblies or to testing in accordance with ANSI Standard N510-1975.

C.2.g The auxiliary building filtration system is instrumented to annunciate locally and in the main control room. There is no recording of these data. The control room filtration system is in a continuously occupied area. Only local visual pressure drop indication is provided. There is no alarm or recording of the data.

C.2.h The fans in the fuel building exhaust system, which draw air from the Fuel Building and through the filters, are not safety related and are not powered from safety related buses.

Table 6.2-45 (continued)
COMPLIANCE WITH REGULATORY GUIDE 1.52

C.2.i The auxiliary building filtration was designed at the same time that RG 1.52 was developed. It meets the requirements of RG 1.52, Rev. 2 except that the safeguards area exhaust system is automatically aligned to the auxiliary building filter banks upon a CDA signal and the auxiliary building central exhaust system is manually aligned to the filter banks by post-LOCA emergency procedures. To account for the manual realignment of auxiliary building central exhaust to the filter banks, a 60-minute delay in filtration of ECCS leakage is included in the analysis of doses resulting from a LOCA design basis accident.

The control room filtration was designed to RG 1.52 Rev. 1 requirements. The filtration system auto starts when the MCR bottle air system is actuated by Train A or B Safety Injection signal or Hi Hi Radiation signal during refueling operations.

C.2.j The enclosed auxiliary building filtration system is designed to be removed as a minimum number of segmented sections. Individual filter components will be removed prior to cutting the housing into segmented sections.

Cutting the auxiliary building filtration system into segmented sections for removal with the individual components intact exposes the local environment and personnel to unnecessary contamination. When components are removed and handled individually, each component is packaged, shielded, and shipped to minimize operator exposure. When all components are removed, the housing would be decontaminated. Removal of the decontaminated housing is completed by cutting into a minimum number of sections, which can be packaged, shielded, and shipped, minimizing the exposure.

C.2.l System housings and ductwork under negative pressure are not tested for leakage.

Exhaust fans for the auxiliary building and control room filtration systems are located downstream of the filters, assuring no exfiltration of airborne contamination.

If leakage occurs into the ductwork between the filters and the fan, the infiltration is expected to be small. In addition, radiation monitors are provided at the ventilation stacks to ensure that releases do not exceed site requirements.

Pressure-tight dampers are furnished to isolate each auxiliary building filter bank. These dampers have a leakage rate of less than 1% maximum flow as tested in accordance with AMCA Standard 500. This leakage is within design limits.

Ductwork under positive pressure is leak tested in accordance with SMACNA standards to exhibit a maximum leakage (in cfm) of less than 10% of ductwork volume (in ft^3).

C.3.a Demisters are not provided.

C.3.c Prefilters have been purchased as an integral part of the HEPA filters and are available only in 2-in.-thick sizes with efficiencies of approximately 10% NBS dust spot.

Table 6.2-45 (continued)
COMPLIANCE WITH REGULATORY GUIDE 1.52

C.3.d VEPCO is in disagreement with the requirement that each and every HEPA filter purchased must be sent to the appropriate Quality Assurance Station to be tested in accordance with the current ERDA Health and Safety Bulletin for Filter Unit Inspection and Testing Service. Presently, we specify that all HEPA filters furnished to nuclear power plants shall be in accordance with MIL-F-51068D and that documentation be furnished to VEPCO to prove that these filters have been qualified recently in accordance with Paragraph 4.2 of MIL-F-51068D and tested at the Filter Quality Assurance Station, Oak Ridge, Tennessee. When installed in the filter housing, the HEPA filters and housing are inspected for defects and tested for leaktightness in accordance with ANSI N510-1975.

We feel, at the present, that the requirement that the vendor qualifies his HEPA filters in accordance with Section 4.2 of MIL-F-50168D periodically and that the periodic in-place DOP smoke testing at 100% and 20% of rated flow and visual inspection of HEPA filters is sufficient to guarantee the integrity and performance of the HEPA filters used. Our experience is that the fewer times the HEPA filters are handled, the better. HEPA filters are easily damaged. Therefore, the requirement of sending every HEPA filter for additional testing seems to needlessly increase the chances of damaging the filter with little increase of quality assurance.

C.3.e Filter and adsorber mounting frames are constructed and designed in accordance with the intent of the recommendations of Section 4.2 of ORNL-NSIC-65, except for the frame tolerance guidelines in Table 4.2. The tolerances for HEPA and adsorber mounting are sufficient to satisfy the bank leak test criteria of the Ventilation Filter Testing Program.

Section 4.3 of ORNL-NSIC-65 lists fabrication tolerance recommendations for HEPA and adsorber mounting frames. Specifying these tolerance levels implies a costly testing program to ensure compliance. Vendor’s standard tolerance levels and acceptable results from inplace bank leak tests specified in the Ventilation Filter Testing Program ensure a satisfactory installation.

C.3.h Floor drains are not provided in either the control room or auxiliary building filter units. The control room units have no sprinkler protection. If washdown should be required for decontamination, filter housings will be dried out by using the fan and electric heating coils. The auxiliary building filter units are protected against fire by carbon dioxide, which precludes use of floor drains. Washdown and drying will be performed in a similar manner as the control room units.

C.3.i Laboratory testing of engineered safety features ventilation systems activated charcoal samples is performed as noted below:

Main Control Room/Emergency Switchgear Room Emergency Ventilation System charcoal adsorber samples are tested in accordance with the Ventilation Filter Testing Program.

Emergency Core Cooling System Pump Room Exhaust Air Cleanup System charcoal adsorber samples are tested in accordance with the Ventilation Filter Testing Program.

Table 6.2-45 (continued)
COMPLIANCE WITH REGULATORY GUIDE 1.52

C.3.j A qualification test on the prototype adsorber was not performed in accordance with paragraph 7.4.1 of AACC-CS-8T using the design-basis earthquake parameters particular to the North Anna 1 and 2 site, since the requirements to perform these tests did not exist at the time the equipment was procured.

C.3.k No cooling mechanisms are provided, as it is expected that the maximum decay heat generation from collected radioiodines is insufficient to raise the charcoal bed temperature above 250°F with no system airflow. Fire detectors are provided in the auxiliary building filtration system.

C.3.n Ductwork is not designed in accordance with Section 2.8 ORNL-NSIC-65. The design is based on High Velocity Duct Construction Standards, Second Edition, Sheet Metal and Air Conditioning, Contractors National Association, Inc. (SMACCNA), modified to suit seismic requirements. This design provides ductwork and supports of sufficient strength for the service intended.

C.3.p Damper requirements not included in Revision 1. The requirements of Revision 2 were added after the NAPS primary ventilation system was designed and installed. Only the testing requirements of the ANSI standard apply to the installed dampers. Therefore, they are not required to be designed or installed in accordance with Section 5.9 of ANSI N509-1976.

C.4.a See Chapter 12.

C.4.c Mounting frames in the control room filtration system are less than 3 feet apart. HEPA filters are of the side-loaded type. The charcoal trays are of the end-loading type, which requires the filter housing to be disconnected at the inlet and outlet flanges. Then the whole unit would be moved clear of duct transition pieces. The trays are then removed from the unit.

C.4.d Permanent test probes are not provided or manifolded. However, permanent (normally capped) injection parts and test probe connections are supplied.

C.4.e The atmosphere cleanup systems not normally in operation will be tested at least once a month to ensure proper operation. Local heaters within the filter train will maintain temperatures, and thus relative humidity, to reduce the buildup of moisture on adsorbers. However, VEPCO does not feel it is necessary to run these trains beyond the time required to ensure that components are functioning properly to reduce the amount of moisture on the adsorbers and the HEPA filters.

C.5.c Periodic testing will be in accordance with the Ventilation Filter Testing Program.

C.5.d Periodic testing will be in accordance with the Ventilation Filter Testing Program.

C.6.a Requirements shall be in accordance with the Ventilation Filter Testing Program.

C.6.b The auxiliary building filter assembly utilizes test canisters for laboratory test purposes. The control room filter assemblies are small units and test canisters are not furnished. Therefore, during the test for control room adsorber units, one out of the three charcoal trays will be removed for laboratory analysis. A previously tested and properly stored spare tray will be inserted in the assembly prior to conducting the leakage efficiency test. Testing will be in accordance with the Ventilation Filter Testing Program.

Table 6.2-46
STONE & WEBSTER-PROCURED REMOTELY OPERATED VALVES ASSOCIATED WITH PIPING PENETRATIONS

Penetration Number	Inside Valve No.	Service	Valve type
8	TV-CC101B	Reactor coolant pump thermal barrier cooling water	Auto-trip
12	TV-CC105B	Chilled water from containment air recirculating cooling coils	Auto-trip
13	TV-CC105C	Chilled water from containment air recirculating cooling coils	Auto-trip
14	TV-CC105A	Chilled water from containment air recirculating cooling coils	Auto-trip
25	TV-CC102F	Reactor coolant pump bearing cooling water	Auto-trip
26	TV-CC102B	Reactor coolant pump bearing cooling water	Auto-trip
27	TV-CC102D	Reactor coolant pump bearing cooling water	Auto-trip
33	TV-DG100B	Primary drain transfer pump discharge	Auto-trip
38	TV-DA100B	Containment sump pump discharge	Auto-trip
39	TV-BD100B	Steam generator blowdown	Auto-trip
40	TV-BD100F	Steam generator blowdown	Auto-trip
41	TV-BD100D	Steam generator blowdown	Auto-trip
44	TV-RM100C	Air radiation monitor supply	Auto-trip
48	TV-VG100B	Primary vent header	Auto-trip
56	TV-SS106A	Reactor primary coolant hot-leg sample line	Auto-trip
56	TV-SS102A	Reactor primary coolant cold-leg sample line	Auto-trip
	TV-SS100A	Pressurizer liquid space sample	Auto-trip
	TV-SS112A	Steam generator surface sample	Auto-trip
57	TV-SS101A	Pressurizer vapor space sample	Auto-trip
57	TV-SS104A	Pressurizer relief tank sample	Auto-trip
90	MOV-HV100C	Purge exhaust	Motor-operated
91	MOV-HV100A	Purge supply	Motor-operated
94	TV-CV100	Containment air ejector suction	Auto-trip

Table 6.2-46 (continued)
STONE & WEBSTER-PROCURED REMOTELY OPERATED VALVES ASSOCIATED WITH PIPING PENETRATIONS

Penetration Number	Inside Valve No.	Service	Valve type
97	TV-SS103A	RHR Sample Containment Isolation Valve	Auto-trip
98	TV-HC100A	Hydrogen analyzer suction	Manual
105	TV-HC102A	Hydrogen analyzer suction	Manual
112[a]	TV-IA201A	Containment instrument air compressor suction	Auto-trip
Penetration Number	**Outside Valve No.**	**Service**	**Valve Type**
1	TV-CC103B	Component cooling water from RHR system and excess letdown heat exchanger	Auto-trip
5	TV-CC103A	Component cooling water from RHR system and excess letdown heat exchanger	Auto-trip
8	TV-CC101A	Reactor coolant pump thermal barrier cooling water	Auto-trip
12	TV-CC100B	Chilled water from containment air recirculating cooling coils	Auto-trip
13	TV-CC100C	Chilled water from containment air recirculating cooling coils	Auto-trip
14	TV-CC100A	Chilled water from containment air recirculating cooling coils	Auto-trip
16	TV-CC104C	Reactor coolant pump cooling water - in	Auto-trip
17	TV-CC104B	Reactor coolant pump cooling water - in	Auto-trip
18	TV-CC104A	Reactor coolant pump cooling water - in	Auto-trip
25	TV-CC102E	Reactor coolant pump bearing cooling water	Auto-trip
26	TV-CC102A	Reactor coolant pump bearing cooling water	Auto-trip
27	TV-CC102C	Reactor coolant pump bearing cooling water	Auto-trip
31	TV-HC105A	Hydrogen recombiner	Manual

Table 6.2-46 (continued)
STONE & WEBSTER-PROCURED REMOTELY OPERATED
VALVES ASSOCIATED WITH PIPING PENETRATIONS

Penetration Number	Outside Valve No.	Service	Valve Type
31	TV-HC105B	Hydrogen recombiner discharge	Manual
31	TV-HC101A	Hydrogen analyzer	Manual
31	TV-HC101B	Hydrogen analyzer discharge	Manual
33	TV-DG100A	Primary drain transfer pump discharge	Auto-trip
38	TV-DA100A	Containment sump pump discharge	Auto-trip
39	TV-BD100A	Steam generator blowdown	Auto-trip
40	TV-BD100E	Steam generator blowdown	Auto-trip
41	TV-BD100C	Steam generator blowdown	Auto-trip
43	TV-RM100A TV-RM100D	Air radiation monitor return	Auto-trip Auto-trip
44	TV-RM100B	Air radiation monitor supply	Auto-trip
47	TV-IA102A TV-IA102B	Containment instrument air supply	Auto-trip Auto-trip
48	TV-VG100A	Primary vent header	Auto-trip
50	TV-SI101	Safety injection accumulators to waste gas filters	Auto-trip
53	TV-SI100	Nitrogen to pressurizer relief tank and safety injection accumulators	Auto-trip
55	TV-LM200E TV-LM200F	Reactor containment leakage monitoring lines	Auto-trip Auto-trip
56	TV-SS102B	Reactor coolant cold-leg sample line	Auto-trip
56	TV-SS106B	Reactor coolant hot-leg sample line	Auto-trip
56	TV-SS100B	Pressurizer liquid space sample	Auto-trip
56	TV-SS112B	Steam generator surface sample	Auto-trip
57	TV-SS101B	Pressurizer vapor space sample	Auto-trip
57	TV-LM100G	Reactor containment leakage monitoring lines	Auto-trip
57	TV-LM100H	Reactor containment leakage monitoring lines	Auto-trip

Table 6.2-46 (continued)
STONE & WEBSTER-PROCURED REMOTELY OPERATED VALVES ASSOCIATED WITH PIPING PENETRATIONS

Penetration Number	Outside Valve No.	Service	Valve Type
57	TV-SS104B	Pressure relief tank sample	Auto-trip
63	MOV-QS101B	Quench spray pump discharge	Motor-operated
64	MOV-QS101A	Quench spray pump discharge	Motor-operated
66	MOV-RS155A MOV-RS100A MOV-RS101A	Recirc. spray pump suction and casing cooling discharge	Motor-operated
67	MOV-RS155B MOV-RS100B MOV-RS101B	Recirc. spray pump suction and casing cooling discharge	Motor-operated
70	MOV-RS156B	Recirc. spray pump discharge	Motor-operated
71	MOV-RS156A	Recirc. spray pump discharge	Motor-operated
73	TV-MS101A	Main steam line	Auto-trip
	TV-MS113A	Main steam line bypass	Auto-trip
	TV-MS110	Main steam line to blowdown	Auto-trip
	TV-MS109	Main steam line to blowdown	Auto-trip
74	TV-MS101B TV-MS113B	Main steam line Main steam line bypass	Auto-trip Auto-trip
75	TV-MS101C TV-MS113C	Main steam line Main steam line bypass	Auto-trip Auto-trip
79	MOV-SW103D	Service water to recirc spray coolers	Motor-operated
80	MOV-SW103C	Service water to recirc. spray coolers	Motor-operated
81	MOV-SW103B	Service water to recirc. spray coolers	Motor-operated
82	MOV-SW103A	Service water to recirc. spray coolers	Motor-operated
83	MOV-SW104D	Service water from recirc. spray coolers	Motor-operated
84	MOV-SW104C	Service water from recirc. spray coolers	Motor-operated
85	MOV-SW104B	Service water from recirc. spray coolers	Motor-operated
86	MOV-SW104A	Service water from recirc. spray coolers	Motor-operated

Table 6.2-46 (continued)
STONE & WEBSTER-PROCURED REMOTELY OPERATED
VALVES ASSOCIATED WITH PIPING PENETRATIONS

Penetration Number	Outside Valve No.	Service	Valve Type
89	TV-SV102-1	Condenser air ejector vent	Auto-trip
	TV-SV103	Condenser air ejector vent	Auto-trip
90	MOV-HV100D	Purge exhaust	Motor-operated
	MOV-HV101	Bypass	Motor-operated
91	MOV-HV100B	Purge supply	Motor-operated
	MOV-HV102	Alternate supply	Motor-operated
92	TV-CV150C TV-CV150D	Containment vacuum pump suction	Auto-trip Auto-trip
93	TV-CV150A TV-CV150B	Containment vacuum pump suction	Auto-trip Auto-trip
97	TV-SS103A	RHR sample containment isolation valve	Auto-trip
98	TV-LM100A TV-LM100B	Containment leakage monitoring (open taps)	Auto-trip Auto-trip
	TV-HC100B	Hydrogen analyzer suction	Manual
105	TV-LM100C TV-LM100D	Containment leakage monitoring (open taps)	Auto-trip Auto-trip
	TV-LM101A TV-LM101B	Containment leakage monitoring (reference)	Auto-trip Auto-trip
	TV-LM101C TV-LM101D	Containment leakage monitoring (reference)	Auto-trip Auto-trip
	TV-HC102B	Hydrogen analyzer suction	Manual
109	TV-HC103A	Hydrogen analyzer	Manual
	TV-HC103B	Hydrogen analyzer discharge	Manual
	TV-HC107A	Hydrogen analyzer	Manual
	TV-HC107B	Hydrogen analyzer discharge	Manual
111	TV-DA103A TV-DA103B	Post accident sample system Containment return line	Auto-trip Auto-trip
112[a]	TV-IA201B	Containment instrument air compressor suction	Auto-trip

a. Applies to Unit 2 only.

Table 6.2-47
WESTINGHOUSE PROCURED REMOTELY OPERATED VALVES ASSOCIATED WITH PIPING PENETRATIONS

Penetration Number	Outside Valve No.	Inside Valve No.	Service	Motive Force
50		HCV-1936	Safety injection accumulators to waste gas filters	Air
7	MOV-1867C,D		High head SI pump discharge to RCS cold legs	Motor
15	MOV-1289A[b]		Charging line	Motor
19	MOV-1381	MOV-1380	Reactor coolant pump seal water return	Motor
22	MOV-1836		High head SI pump discharge to RCS cold legs	Motor
28	TV-1204B[a]	TV-1204A[b]	Reactor coolant letdown line	Air
45	TV-1519A		Primary grade water to pressurizer relief tank	Air
46	FCV-1160		Loop fill header	Air
60	MOV-1890B		Low head SI to RCS hot legs	Motor
61	MOV-1890A		Low head SI to RCS hot legs	Motor
62	MOV-1890C		Low head SI to RCS cold legs	
	MOV-1890D		Low head SI to RCS cold legs	Motor
68	MOV-1860B		Low head SI pump suction from containment sump	Motor
69	MOV-1860A		Low head SI pump suction from containment sump	Motor
106	TV-1859	TV-1842	Safety injection accumulator	Air
113	MOV-1869B		High head SI to RCS hot legs	Motor
114	MOV-1869A		High head SI to RCS hot legs	Motor

a. Originally TV-1204
b. VEPCO Procured

Table 6.2-48
MECHANICAL PENETRATIONS CONTAINMENT LEAK RATE TEST STATUS

Penetration Number	Description	Local Leak Rate	Type A Condition	Notes
001	CC Return from “B” RHR Heat Exchanger	Type “C”	Flooded/Operating	Note 3
002	CC Supply to “A” RHR Heat Exchanger	Type “C”	Flooded/Operating	Note 3
003	Spare (No pipe insert)	None	Sealed	—
004	CC Supply to “B” RHR Heat Exchanger	Type “C”	Flooded/Operating	Note 3
005	CC Return from “A” RHR Heat Exchanger	Type “C”	Flooded/Operating	Note 3
006	Spare (No pipe insert)	None	Sealed	—
007	High Head SI-Bit Outlet to RCS Cold Legs	None	Flooded/Operating	Note 10
008	CC Return from RC Pump Thermal Barriers	Type “C”	Flooded/Isolated	Note 8
009	CC Supply to “C” Recirc Air Cooling Coils	Type “C”	Flooded/Operating	Note 1
010	CC Supply to “B” Recirc Air Cooling Coils	Type “C”	Flooded/Operating	Note 1
011	CC Supply to “A” Recirc Air Cooling Coils	Type “C”	Flooded/Operating	Note 1
012	CC Return from “B” Recirc Air Cooling Coils	Type “C”	Flooded/Operating	Note 1
013	CC Return from “C” Recirc Air Cooling Coils	Type “C”	Flooded/Operating	Note 1
014	CC Return from “A” Recirc Air Cooling coils	Type “C”	Flooded/Operating	Note 1
015	Main Charging Header	None	Flooded/Operating	Note 10
016	CC Supply to “C” RC Pump Coolers	Type “C”	Flooded/Isolated	Note 8
017	CC Supply to “B” RC Pump Coolers	Type “C”	Flooded/Isolated	Note 8
018	CC Supply to “A” RC Pump Coolers	Type “C”	Flooded/Isolated	Note 8
019	RC Pump Seal Water Return	Type “C”	Flooded/Isolated	Note 9

* These systems are exposed to the test pressure.

— N/A

** These systems are operating during the performance of Type “A” Test. It is not vented to the outside of containment since its needed to measure containment pressure. Type “C” penalties will be applied to test results.

Table 6.2-48 (continued)
MECHANICAL PENETRATIONS CONTAINMENT LEAK RATE TEST STATUS

Penetration Number	Description	Local Leak Rate	Type A Condition	Notes
020	Accumulator Makeup	Type “C”	Isolated	Note 7
021	Spare (with pipe insert)	None	Sealed	—
022	High Head SI-Charging Header to RCS Cold Legs	None	Flooded/Isolated	Note 10
023	Spare (with pipe insert)	None	Sealed	—
024	RHR to Refueling Water Storage Tank	Type “C”	Flooded/Isolated	Note 3
025	CC Return from “A” RC Pump Coolers	Type “C”	Flooded/Isolated	Note 8
026	CC Return from “C” RC Pump Coolers	Type “C”	Flooded/Isolated	Note 8
027	CC Return from “B” RC Pump Coolers	Type “C”	Flooded/Isolated	Note 8
028	Main Letdown Header	Type “C”	Flooded/Operable	Note 3
029	Spare (no pipe insert)	None	Sealed	—
030	Spare (no pipe insert)	None	Sealed	—
031	Return to Containment from H_2Analyzer H_2 Recombiner and Air Sample Panel	Type “C”	Isolated	Note 7
032	Wet Lay-up for “A” Steam Generator	Type “C”	Vented/Isolated	*
033	Primary Drain Transfer Pump Discharge	Type “C”	Vented/Isolated	*
034	Fire Protection Supply to Containment	Type “C”	Vented/Isolated	*
035	Seal Water Supply to “C” RC Pump	None	Flooded/Isolated	Note 2
036	Seal Water Supply to “A” RC Pump	None	Flooded/Isolated	Note 2
037	Seal Water Supply to “B” RC Pump	None	Flooded/Isolated	Note 2

* These systems are exposed to the test pressure.
— N/A
** These systems are operating during the performance of Type “A” Test. It is not vented to the outside of containment since its needed to measure containment pressure. Type “C” penalties will be applied to test results.

Table 6.2-48 (continued)
MECHANICAL PENETRATIONS CONTAINMENT LEAK RATE TEST STATUS

Penetration Number	Description	Local Leak Rate	Type A Condition	Notes
038	Containment Sump Pump Discharge	Type “C”	Vented/Isolated	*
039	“A” Steam Generator Blowdown	Type “C”	Flooded/Isolated	Note 9
040	“C” Steam Generator Blowdown	Type “C”	Flooded/Isolated	Note 9
041	“B” Steam Generator Blowdown	Type “C”	Flooded/Isolated	Note 9
042	Service Air Supply to Containment	Type “C”	Vented/Isolated	*
043	Return to Containment from Rad Monitoring Cabinet	Type “C”	Vented/Isolated	*
044	Supply to Rad Monitoring Cabinet	Type “C”	Vented/Isolated	*
045	Primary Grade Water Supply to Containment	Type “C”	Vented/Isolated	*
046	Charging to Loop Fill Header	None	Flooded/Isolated	Note 10
047	Instrument Air Supply to Containment	Type “C”	Vented/Isolated	*
048	Primary Vent Header	Type “C”	Vented/Isolated	*
049	Spare (with pipe insert)	None	Sealed	—
050	Nitrogen to Waste Gas Charcoal Filters	Type “C”	Vented/Isolated	*
051	Spare (with sleeved pipe insert)	None	Sealed	—
052	Spare (with sleeved pipe insert)	None	Sealed	—
053	Nitrogen to SI Accumulators and PRT	Type “C”	Vented/Isolated	*
054	Primary Vent Pot Vent	Type “C”	Vented/Isolated	*
055A	RVLIS-Reactor Vessel Head (Train A)	None	Flooded/Operating	Note 5
055B	RVLIS-Hot Leg (Train A)	None	Flooded/Operating	Note 5

* These systems are exposed to the test pressure.

— N/A

** These systems are operating during the performance of Type “A” Test. It is not vented to the outside of containment since its needed to measure containment pressure. Type “C” penalties will be applied to test results.

Table 6.2-48 (continued)
MECHANICAL PENETRATIONS CONTAINMENT LEAK RATE TEST STATUS

Penetration Number	Description	Local Leak Rate	Type A Condition	Notes
055C	RVLIS-Seal Table (Train A)	None	Flooded/Operating	Note 5
055D	Containment Leakage Monitoring Open System	Type “C”	Vented (I)-Sealed (O) /Operating	**
056A	Pressurizer Liquid Space Sample Line	Type “C”	Flooded/Isolated	Note 3
056B	RC Hot Legs Sample Header	Type “C”	Flooded/Isolated	Note 3
056C	RC Cold Legs Sample Header	Type “C”	Flooded/Isolated	Note 3
056D	Steam Generator Blowdown Sample Header	Type “C”	Flooded/Isolated	Note 9
057A	Containment Leakage Monitoring Open System	Type “C”	Vented (I)-Sealed (O) /Operating	**
057B	Pressurizer Relief Tank Gas Space System	Type “C”	Vented/Isolated	*
057C	Pressurizer Vapor Space Sample Line	Type “C”	Vented/Isolated	*
057D	Spare (with pipe insert)	None	Sealed	—
058	Spare (with pipe insert)	None	Sealed	—
059	Spare (with pipe insert)	None	Sealed	—
060	Low Head SI to Hot Legs	None	Flooded/Isolated	Note 10
061	Low Head SI to Hot Legs	None	Flooded/Isolated	Note 10
062	Low Head SI to Cold Legs	None	Flooded/Isolated	Note 10

* These systems are exposed to the test pressure.
— N/A
** These systems are operating during the performance of Type “A” Test. It is not vented to the outside of containment since its needed to measure containment pressure. Type “C” penalties will be applied to test results.

Table 6.2-48 (continued)
MECHANICAL PENETRATIONS CONTAINMENT LEAK RATE TEST STATUS

Penetration Number	Description	Local Leak Rate	Type A Condition	Notes
063	“B” Quench Spray Header	Type “C”	Vented (I) Flooded (O) /Isolated	Note 9 Flooded to outside motor-operated valve
064	“A” Quench Spray Header	Type “C”	Vented (I) Flooded (O) /Isolated	Note 9 Flooded to outside motor-operated valve
065	Fuel Transfer Tube	Type “B”	Vented (I) Flooded (O)	*
066	Supply to “A” Outside Recirc Spray from Sump	None	Flooded/Open MOV	Note 10
	Casing Cooling to “A” Outside Recirc Spray Suction	None	Flooded/Isolated	Note 10
067	Supply to “B” Outside Recirc Spray from Sump	None	Flooded/Open MOV	Note 10
	Casing Cooling to “B” Outside Recirc Spray Suction	None	Flooded/Isolated	Note 10
068	Supply to “B” Low Head SI Pump from Sump	None	Flooded/Isolated	Note 10
069	Supply to “A” Low Head SI Pump for Sump	None	Flooded/Isolated	Note 10
070	“B” Outside Recirc Spray Pump Discharge	None	Vented(I)/Isolated (O)	Note 10
071	“A” Outside Recirc Spray Pump Discharge	None	Vented(I)/Isolated (O)	Note 10
072	Spare (no pipe insert)	None	Sealed	-
073	“A” Main Steam Header	None	Non-vented/Sealed (I) Isolated (O)	Note 4

* These systems are exposed to the test pressure.
— N/A
** These systems are operating during the performance of Type “A” Test. It is not vented to the outside of containment since its needed to measure containment pressure. Type “C” penalties will be applied to test results.

Table 6.2-48 (continued)
MECHANICAL PENETRATIONS CONTAINMENT LEAK RATE TEST STATUS

Penetration Number	Description	Local Leak Rate	Type A Condition	Notes
074	"B" Main Steam Header	None	Non-vented/Sealed (I) Isolated (O)	Note 4
075	"C" Main Steam Header	None	Non-vented/Sealed (I) Isolated (O)	Note 4
076	"A" Feedwater Header	None	Flooded/Sealed (I) Isolated (O)	Note 4
077	"C" Feedwater Header	None	Flooded/Sealed (I) Isolated (O)	Note 4
078	"B" Feedwater Header	None	Flooded/Sealed (I) Isolated (O)	Note 4
079	Service Water Supply to "D" Recirc Spray Heat Exchanger	Type "C"	Flooded/Isolated	Note 6
080	Service Water Supply to "C" Recirc Spray Heat Exchanger	Type "C"	Flooded/Isolated	Note 6
081	Service Water Supply to "B" Recirc Spray Heat Exchanger	Type "C"	Flooded/Isolated	Note 6
082	Service Water Supply to "A" Recirc Spray Heat Exchanger	Type "C"	Flooded/Isolated	Note 6
083	Service Water Return from "D" Recirc Spray Heat Exchanger	Type "C"	Flooded/Isolated	Note 6
084	Service Water Return from "C" Recirc Spray Heat Exchanger	Type "C"	Flooded/Isolated	Note 6
085	Service Water Return from "B" Recirc Spray Heat Exchanger	Type "C"	Flooded/Isolated	Note 6
086	Service Water Return from "A" Recirc Spray Heat Exchanger	Type "C"	Flooded/Isolated	Note 6
087	Spare (with pipe insert)	None	Sealed	—
088	Spare (with pipe insert)	None	Sealed	—

* These systems are exposed to the test pressure.
— N/A
** These systems are operating during the performance of Type "A" Test. It is not vented to the outside of containment since its needed to measure containment pressure. Type "C" penalties will be applied to test results.

Table 6.2-48 (continued)
MECHANICAL PENETRATIONS CONTAINMENT LEAK RATE TEST STATUS

Penetration Number	Description	Local Leak Rate	Type A Condition	Notes
089	Condenser Air Removal Discharge to Containment	Type “C”	Vented/Isolated	*
090	Containment Purge Exhaust	Type “C”	Vented/Isolated	*
091	Containment Purge Supply	Type “C”	Vented/Isolated	*
092	Supply to “B” Containment Vacuum Pump, HC Purge Blowers & Hydrogen Recombiners	Type “C”	Isolated	Note 7
093	Supply to “A” Containment Vacuum Pump, HC Purge Blowers & Hydrogen Recombiners	Type “C”	Isolated	Note 7
094	Suction to Containment Vacuum Ejector	Type “C”	Vented/Isolated	*
095	Spare (with pipe insert)	None	Sealed	—
096	Spare (with pipe insert)	None	Sealed	—
097A	RHR Sample Line	Type “C”	Flooded/Isolated	Note 3
097B	Containment Leakage Monitoring Open System	Type “C”	Vented (I) Sealed (O) /Operating	**
097C	Pressurizer Dead Weight Test Line	Type “C”	Vented/Isolated	*
097D	Spare (with pipe insert)	None	Sealed	—
098A	Supply to Hydrogen Analyzers	Type “C”	Isolated	Note 7
098B	Supply to Air Sample Panel	Type “C”	Isolated	Note 7
098C	Spare (with pipe insert)	None	Sealed	—
098D	Spare (with pipe insert)	None	Sealed	—
099	Spare (no pipe insert)	None	Sealed	—

* These systems are exposed to the test pressure.

— N/A

** These systems are operating during the performance of Type “A” Test. It is not vented to the outside of containment since its needed to measure containment pressure. Type “C” penalties will be applied to test results.

Table 6.2-48 (continued)
MECHANICAL PENETRATIONS CONTAINMENT LEAK RATE TEST STATUS

Penetration Number	Description	Local Leak Rate	Type A Condition	Notes
100	Wet Lay-up for “B” Steam Generator	Type “C”	Vented/Isolated	*
101	Spare (with pipe insert)	None	Sealed	—
102	Spare (with pipe insert)	None	Sealed	—
103	Refueling Purification Return to Reactor Cavity	Type “C”	Vented/Isolated	*
104	Refueling Purification From the Reactor Cavity	Type “C”	Vented/Isolated	*
105A	Containment Leakage Monitoring Open System	Type “C”	Vented (I) Sealed (O) /Operating	**
105B	Supply to Hydrogen Analyzers	Type “C”	Isolated	Note 7
105C	Containment Leakage Monitoring Sealed System	Type “C”	Sealed (I) Isolated	**
105D	Containment Leakage Monitoring Sealed System	Type “C”	Sealed (I) Isolated	**
106	Accumulator Test Line	Type “C”	Vented/Isolated	*
107	Spare (no pipe insert)	None	Sealed	—
108	Wet Lay-up for “C” Steam Generator	Type “C”	Vented/Isolated	*
109	Return to Containment from Hydrogen Analyzers	Type “C”	Isolated	Note 7
110	Spare (with pipe insert)	None	Sealed	—
111A	Chemical Feed to “A” Steam Generator	None	Flooded/Open (I) Isolated (O)	Note 4
111B	Chemical Feed to “B” Steam Generator	None	Flooded/Open (I) Isolated (O)	Note 4

* These systems are exposed to the test pressure.

— N/A

** These systems are operating during the performance of Type “A” Test. It is not vented to the outside of containment since its needed to measure containment pressure. Type “C” penalties will be applied to test results.

Table 6.2-48 (continued)
MECHANICAL PENETRATIONS CONTAINMENT LEAK RATE TEST STATUS

Penetration Number	Description	Local Leak Rate	Type A Condition	Notes
111C	Chemical Feed to “C” Steam Generator	None	Flooded/Open (I) Isolated (O)	Note 4
111D	Post Accident Sample System Return to Containment	Type “C”	Vented/Isolated	*
112 (a)	Instrument Air Compressor Suction	Type “C”	Vented/Isolated	*
113	High Head SI from Charging to RCS Hot Legs	None	Flooded/Isolated	Note 10
114	High Head SI from Charging to RCS Hot Legs	None	Flooded/Isolated	Note 10
115	Spare (With pipe insert)	None	Sealed	—
116	Spare (with pipe insert)	None	Sealed	—
117A	RVLIS-Reactor Vessel Head (Train B)	None	Flooded/Operating	Note 5
117B	Spare (With pipe insert)	None	Sealed	—
117C	RVLIS-Hot Leg (Train B)	None	Flooded/Operating	Note 5
117D	RVLIS-Seal Table (Train B)	None	Flooded/Operating	Note 5

(a) Unit #2 Only

* These systems are exposed to the test pressure.

— N/A

** These systems are operating during the performance of Type “A” Test. It is not vented to the outside of containment since its needed to measure containment pressure. Type “C” penalties will be applied to test results.

Table 6.2-48 (continued)
NOTES

Containment Leak Note Test Status

1. These systems are required to operate in support of the Type “A” Test. Type “C” penalties will be applied
2. These systems are designed to operate during a LOCA at greater than accident pressure. No Type “C” penalty will be applied.
3. These systems are needed for safe operation of the plant during the Type “A” Test. Type “C” penalty will be applied to test results.
4. These systems are part of the pressurized water reactor secondary system. As such they are considered extensions of the containment boundary. No Type “C” penalty will be applied.
5. These systems are designed to operate during a LOCA and are considered part of the containment boundary. No local leak rate penalty will be applied.
6. These systems will be operating during a LOCA. No Type “C” penalty will be applied.
7. System common to both units. Penetration isolated if system is needed by the other unit. Type “C” penalty will be applied to Type “A” test results.
8. Draining/Venting poses a personnel safety hazard due to chromated water. Type “C” penalties will be applied to these penetrations.
9. These systems are required to be flooded or isolated during the Type “A” Test. Type “C” penalty will be applied.
10. These penetrations are in systems that are water filled and/or normally operating under accident conditions (LOCA) at a pressure greater than peak accident pressure. Therefore, these penetrations are not considered credible leakage paths from containment.

Table 6.2-49
CONTAINMENT VACUUM SYSTEM DATA

Containment vacuum pumps	
Number	Two (one required) per unit
Type	Liquid ring, oil free
Power source	480V emergency bus
Capacity	40 scfm
Steam jet ejector	
Number	One per unit
Power source	150 psig auxiliary steam
Capacity	51,000 lb of air in 4 hours

Table 6.2-50
CONTAINMENT PRESSURE ANALYSIS FOR INADVERTENT OPERATION OF QUENCH SPRAY SYSTEM

Initial Conditions		
Minimum air partial pressure (P_1)	10.0 psia	TS limit of 10.3–0.3 psi uncertainty
Maximum bulk air temperature (T_1)	116.5°F	TS limit of 115.0°F + 1.5°F uncertainty
Minimum RWST temperature (T_2)	32°F	Bounding minimum value
Water vapor saturation pressure at T_2 (P_{sat})	0.09 psia	at 32°F

Using Charles' Law for the air partial pressure (temperatures converted to Rankine), the final pressure in containment is calculated:

$$P_{total} = P_{air} + P_{vapor} = \frac{T_2}{T_1}P_1 + P_{sat}(T_2) = \frac{(460 + 32)(10.3 - 0.3)}{(460 + 116.5)} + 0.09 = 8.62 \text{ psia}$$

Table 6.2-51
LEAKAGE MONITORING SYSTEM COMPONENT DESIGN DATA

Containment pressure	
Type	Electronic pressure sensing
Range	0-65 psi
Accuracy	± 2.3%
Containment air temperature	
Type	Resistance temperature detectors
Range	60-120°F
Accuracy	± 0.5°F
Containment air moisture	
Type	Dewcell resistance bulb
Range	30-100°F
Accuracy	± 2.0°F
Makeup air system	
Volumetric flow meter	
Type	Linear electric
Design pressure	125
Range	0-100 scfm

Table 6.2-52
UNIT 1 CONTAINMENT AVERAGE TEMPERATURE DETECTOR LOCATIONS
AND METHOD FOR CALCULATING
THE WEIGHTED AVERAGE CONTAINMENT TEMPERATURE

Location		Weight Factor (WF)	Min. No. of Temperature Detectors
a. Containment dome	Elev. ~390	0.09604	2
b. Inside crane wall	Elev. ~329	0.04846	3
c. Annulus	Elev. ~329	0.02256	3
d. Annulus	Elev. ~238	0.04972	2
e. Cubicles	Elev. ~268	0.06785 (.07513)[a]	3

The average containment air temperature shall be determined by the following relationship:

$$T_{containment} = \frac{1.0}{\left[\sum_{i=1}^{n} \frac{W_{Fi}}{T_i}\right]} \quad \text{where}$$

WF_i is the weight factor for the temperature T_i, of the i^{th} temperature measurement.

a. Weight factor to be used for pressurizer cubicle at Elev. 268.

Table 6.2-53
UNIT 2 CONTAINMENT AVERAGE TEMPERATURE DETECTOR LOCATIONS
AND METHOD FOR CALCULATING
THE WEIGHTED AVERAGE CONTAINMENT TEMPERATURE

Location		Weight Factor (WF)	Min. No. of Temperature Detectors
a. Containment dome	Elev. ~390	0.04789	2
b. Inside crane wall	Elev. ~329	0.09373	3
c. Annulus	Elev. ~329	0.02283 (0.02935)[a]	3
d. Annulus	Elev. ~238	0.08309	2
e. Cubicles	Elev. ~268	b	3

The average containment air temperature shall be determined by the following relationship:

$$T_{containment} = \frac{1.0}{\sum_{i=1}^{n} \frac{W_{Fi}}{T_i}}$$ where

WF_i is the weight factor for the temperature T_i, of the i^{th} temperature measurement.

a. Weight factor to be used for pressurizer cubicle at Elev. 268.
b. Weight factor to be used for cubicles A = 0.03932, B = 0.03597, C = 0.03619.

Table 6.2-54
BLOWDOWN MASS AND ENERGY RELEASE
DOUBLE-ENDED HOT LEG GUILLOTINE
Note:(Blowdown applicable for minimum and maximum ESF)

	Break Path No. 1 (M&E Exiting the Vessel Side of the Break)		Break Path No. 2 (M&E Exiting the SG Side of the Break)	
Time	Flow	Energy Thousand	Flow	Energy Thousand
sec	lbm/sec	Btu/sec	lbm/sec	Btu/sec
0.00	0.00	0.00	0.00	0.00
0.0008	1624.65	1045.33	1624.65	1045.33
0.0014	2839.06	1827.33	2808.28	1804.88
0.0020	4272.37	2753.03	3525.07	2260.63
0.07	42298.57	27419.26	25370.99	16210.35
0.13	40494.97	26446.71	26060.75	16660.68
0.20	33931.74	22321.20	24194.70	15413.44
0.26	34477.35	22682.14	21954.04	13892.07
0.33	34619.58	22766.79	21036.75	13196.07
0.39	33985.40	22355.96	20129.14	12499.45
0.46	33814.80	22251.92	19593.66	12039.42
0.52	33502.00	22063.79	19078.68	11603.85
0.59	3323562	21912.44	18730.99	11283.02
0.65	32997.46	21785.27	18389.83	10978.86
0.72	32731.78	21646.27	18087.86	10709.84
0.78	32429.78	21488.49	17859.74	10494.70
0.85	32055.79	21287.11	17647.62	10299.04
0.91	31639.05	21061.99	17433.19	10108.79
0.98	31200.23	20824.20	17287.30	9965.66
1.04	30903.83	20688.75	17133.94	9824.60
1.11	30606.10	20565.67	16990.28	9693.82
1.17	30379.41	20484.93	16881.12	9588.41
1.24	30114.52	20373.47	16779.12	9490.28
1.30	29828.83	20234.99	16723.36	9421.96
1.37	29510.94	20065.28	16690.57	9369.15
1.43	29195.80	19892.49	16681.84	9332.76
1.50	28875.91	19715.30	16692.55	9308.70
1.56	28554.90	19537.49	16715.57	9293.12
1.63	28223.23	19352.32	16747.86	9284.21
1.69	27903.32	19175.85	16785.83	9279.77
1.76	27553.86	18979.69	16828.39	9278.93
1.82	27207.43	18783.94	16872.04	9280.40
1.89	26826.57	18563.76	16918.89	9284.44
1.95	26439.21	18336.17	16966.10	9290.34
2.02	26042.56	18099.69	17014.75	9298.49

Table 6.2-54 (continued)
BLOWDOWN MASS AND ENERGY RELEASE
DOUBLE-ENDED HOT LEG GUILLOTINE
Note:(Blowdown applicable for minimum and maximum ESF)

	Break Path No. 1 (M&E Exiting the Vessel Side of the Break)		Break Path No. 2 (M&E Exiting the SG Side of the Break)	
Time	Flow	Energy Thousand	Flow	Energy Thousand
sec	lbm/sec	Btu/sec	lbm/sec	Btu/sec
2.08	25647.45	17862.22	17064.14	9308.34
2.15	25269.17	17635.62	17113.72	9319.57
2.21	24915.33	17424.20	17160.72	9330.98
2.28	24569.49	17215.71	17203.62	9341.51
2.34	24216.83	16999.22	17241.55	9350.45
2.41	23862.05	16776.23	17275.03	9358.20
2.47	23519.70	16557.06	17300.74	9362.98
2.54	23185.41	16339.75	17321.40	9366.03
2.60	22863.29	16126.64	17335.57	9366.65
2.67	22548.06	15914.32	17343.26	9364.64
2.73	22255.88	15714.64	17344.44	9360.09
2.80	21985.69	15527.98	17339.59	9353.04
2.86	21732.56	15350.17	17329.38	9343.86
2.93	21490.60	15176.10	17313.41	9332.34
2.99	21245.87	14995.82	17291.24	9318.08
3.06	21021.07	14826.51	17263.45	9301.42
3.12	20817.84	14669.64	17231.16	9282.93
3.19	20619.45	14512.89	17194.37	9262.55
3.25	20421.20	14352.95	17152.58	9239.99
3.32	20235.93	14199.27	17105.90	9215.28
3.38	20074.86	14060.34	17054.68	9188.60
3.45	19935.86	13935.13	17001.01	9161.01
3.51	19803.02	13811.84	16943.27	9131.64
3.58	19676.77	13691.37	16881.81	9100.64
3.64	19565.22	13579.00	16815.58	9067.46
3.71	19472.60	13478.58	16745.93	9032.78
3.77	19389.36	13382.45	16671.68	8995.96
3.84	19314.50	13291.84	16596.45	8958.76
3.90	19248.62	13205.29	16516.54	8919.38
3.97	19196.98	13128.38	16434.58	8879.07
4.13	19106.34	12959.20	16216.56	8772.17
4.26	19084.48	12858.34	16031.78	8681.68
4.39	19143.58	12812.99	15841.00	8588.45
4.52	19264.63	12808.89	15641.32	8490.74
4.65	19416.58	12814.44	15441.08	8393.29

Table 6.2-54 (continued)
BLOWDOWN MASS AND ENERGY RELEASE
DOUBLE-ENDED HOT LEG GUILLOTINE
Note:(Blowdown applicable for minimum and maximum ESF)

	Break Path No. 1 (M&E Exiting the Vessel Side of the Break)		Break Path No. 2 (M&E Exiting the SG Side of the Break)	
Time	Flow	Energy Thousand	Flow	Energy Thousand
sec	lbm/sec	Btu/sec	lbm/sec	Btu/sec
4.78	19617.34	12839.63	15261.39	8307.43
4.91	19869.90	12886.46	15058.02	8208.31
5.04	20275.71	13004.89	14797.45	8077.76
5.17	11808.14	8462.52	14526.39	7941.09
5.30	15651.24	10918.19	14268.30	7812.14
5.43	15508.04	10744.49	14019.74	7688.82
5.56	15629.49	10766.41	13783.36	7572.60
5.69	15742.28	10760.02	13562.42	7465.09
5.82	15845.41	10741.05	13356.18	7385.57
5.95	15951.98	10768.44	13149.25	7264.87
6.08	16065.09	10740.57	12931.58	7157.02
6.21	16126.16	10772.81	12701.63	7041.49
6.34	16226.19	10759.01	12468.61	6923.66
6.47	16291.24	10805.38	12248.16	6811.99
6.60	16289.59	10769.82	12029.65	6700.93
6.73	15900.57	10545.44	11814.31	6591.26
6.86	16101.86	10622.47	11603.28	6483.56
6.99	16200.18	1640.39	11387.81	6373.30
7.12	16283.42	10651.31	11176.57	6265.45
7.25	16294.45	10622.63	10970.63	6160.74
7.38	16265.92	10573.29	10769.84	6059.07
7.51	16210.94	10511.08	10571.04	5958.46
7.64	16097.75	10426.13	10367.96	5855.34
7.77	15942.47	10320.68	10170.03	5754.85
7.90	15761.69	10198.72	9975.13	5655.94
8.03	15541.59	10055.91	9778.72	5556.40
8.16	15265.96	9883.49	9588.63	5460.23
8.29	14970.68	9701.05	9397.99	5363.87
8.42	14664.35	9514.35	9210.46	5269.07
8.55	14350.51	9325.18	9025.94	5176.01
8.68	14031.43	9134.77	8842.06	5083.49
8.81	13714.57	8947.11	8661.52	4992.82
8.94	13402.96	8763.29	8482.65	4903.28
9.07	13098.57	8584.26	8305.39	4814.43
9.20	12792.40	8404.93	8127.13	4725.17

Table 6.2-54 (continued)
BLOWDOWN MASS AND ENERGY RELEASE
DOUBLE-ENDED HOT LEG GUILLOTINE
Note:(Blowdown applicable for minimum and maximum ESF)

	Break Path No. 1 (M&E Exiting the Vessel Side of the Break)		Break Path No. 2 (M&E Exiting the SG Side of the Break)	
		Energy		Energy
Time	Flow	Thousand	Flow	Thousand
sec	lbm/sec	Btu/sec	lbm/sec	Btu/sec
9.33	12487.98	8227.11	7950.35	4637.14
9.46	12191.53	8054.24	7773.58	4549.27
9.59	11898.73	7883.85	7596.61	4461.49
9.72	11603.27	7712.85	7420.41	4374.27
9.85	11288.77	7532.95	7242.09	4286.17
9.98	10956.27	7345.96	7055.43	4194.24
10.13	10567.60	7131.61	6832.48	4083.51
10.26	10227.37	6946.11	6624.93	3980.23
10.39	9879.17	6759.08	6402.02	3869.70
10.52	9534.54	6578.81	6170.16	3755.74
10.65	9186.09	6402.65	5923.51	3635.73
10.78	8835.30	6232.45	5666.69	3512.06
10.91	8490.38	6072.70	5407.53	3387.89
11.04	8152.17	5923.13	5156.89	3267.51
11.17	7810.84	5779.63	4915.73	3151.02
11.30	7470.64	5644.42	4689.68	3040.29
11.43	7127.19	5515.10	4481.15	2936.17
11.56	6788.96	5393.94	4295.84	2841.13
11.69	6449.36	5276.35	4131.32	2754.32
11.82	6104.19	5157.17	3983.47	2673.97
11.95	5761.24	5035.29	3852.30	2600.69
12.08	5428.54	4911.83	3735.61	2533.94
12.21	5108.41	4785.68	3629.49	2472.26
12.34	4805.62	4654.24	3532.50	2416.07
12.47	4518.13	4521.56	3441.28	2363.29
12.60	4247.14	4391.47	3355.80	2314.03
12.73	3970.29	4257.26	3272.93	2266.73
12.86	3690.64	4107.11	3185.59	2218.34
12.99	3439.69	3949.00	3102.27	2174.31
13.12	3222.05	3774.98	3018.05	2133.09
13.25	3030.64	3598.63	2932.56	2092.23
13.38	2856.42	3420.77	2844.20	2050.20
13.51	2717.78	3276.64	2755.72	2008.97
13.64	2571.42	3117.28	2664.74	1967.16
13.77	2437.71	2970.59	2567.74	1923.18

Table 6.2-54 (continued)
BLOWDOWN MASS AND ENERGY RELEASE
DOUBLE-ENDED HOT LEG GUILLOTINE
Note:(Blowdown applicable for minimum and maximum ESF)

	Break Path No. 1 (M&E Exiting the Vessel Side of the Break)		Break Path No. 2 (M&E Exiting the SG Side of the Break)	
Time	Flow	Energy Thousand	Flow	Energy Thousand
sec	lbm/sec	Btu/sec	lbm/sec	Btu/sec
13.90	2309.03	2829.01	2463.54	1878.07
14.03	2204.40	2710.56	2355.35	1836.22
14.16	2108.82	2595.24	2234.89	1791.52
14.29	2061.26	2530.96	2128.47	1755.42
14.42	2008.94	2458.29	2020.40	1719.82
14.55	1953.69	2379.56	1907.81	1679.95
14.68	1906.52	2305.73	1806.56	1642.73
14.81	1867.65	2246.20	1713.21	1609.07
14.94	1827.10	2192.60	1633.23	1580.24
15.07	1766.21	2110.81	1556.93	1549.14
15.20	1734.60	2046.00	1487.23	1524.95
15.33	1707.70	1983.03	1422.62	1498.07
15.46	1662.16	1920.39	1370.44	1474.68
15.59	1597.79	1866.06	1331.49	1452.10
15.72	1509.54	1803.40	1303.08	1431.50
15.85	1442.94	1752.91	1284.23	1411.49
15.98	1389.51	1697.31	1269.55	1392.59
16.11	1339.33	1642.18	1250.95	1374.06
16.24	1291.43	1588.18	1221.64	1355.01
16.37	1242.27	1535.43	1175.91	1334.09
16.50	1200.79	1486.86	1108.62	1301.00
16.63	1160.26	1438.68	1027.55	1242.38
16.76	1105.28	1379.87	992.10	1215.01
16.89	1036.39	1298.70	897.17	1105.47
17.02	983.98	1236.42	861.18	1062.99
17.15	1041.66	1312.07	846.28	1048.48
17.28	971.52	1225.55	830.57	1026.63
17.41	913.50	1148.72	771.65	954.05
17.54	864.38	1087.20	672.85	832.45
17.67	803.44	1014.24	551.69	683.94
17.80	749.37	947.13	534.50	665.61
17.93	701.15	886.69	492.68	613.02
18.06	654.05	828.06	369.31	459.25
18.19	612.29	775.91	306.73	383.37
18.32	574.14	728.56	286.88	359.52

Table 6.2-54 (continued)
BLOWDOWN MASS AND ENERGY RELEASE
DOUBLE-ENDED HOT LEG GUILLOTINE
Note:(Blowdown applicable for minimum and maximum ESF)

	Break Path No. 1 (M&E Exiting the Vessel Side of the Break)		Break Path No. 2 (M&E Exiting the SG Side of the Break)	
		Energy		Energy
Time	Flow	Thousand	Flow	Thousand
sec	lbm/sec	Btu/sec	lbm/sec	Btu/sec
18.45	535.68	680.16	202.31	253.53
18.58	502.52	638.32	150.71	189.77
18.71	472.62	601.06	58.92	74.64
18.84	238.60	306.01	0.00	0.00
18.97	55.75	71.94	0.00	0.00
19.10	0.00	0.00	0.00	0.00

Table 6.2-54 (continued)
REFLOOD MASS AND ENERGY RELEASE DOUBLE-ENDED HOT LEG GUILLOTINE - MAX SI/ LHSI - 3500 GPM[a]

Time	Steam Release		Water Release	
seconds	lbm/sec	1000 btu/sec	lbm/sec	1000 btu/sec
24.0	0.0	0.0	0.0	0.0
24.3	0.0	0.0	0.0	0.0
24.6	473.5	246.6	0.0	0.0
24.7	319.0	262.8	0.0	0.0
27.5	1087.7	456.5	0.0	0.0
31.5	1714.9	596.1	0.0	0.0
32.0	1738.6	601.4	0.0	0.0
41.3	1630.1	577.0	2342.0	200.6
46.7	1571.5	560.5	1943.3	164.3
50.0	1537.1	550.7	1757.4	147.3
54.0	1493.8	538.5	1524.5	126.5
55.3	1480.6	446.4	0.0	0.0
64.8	963.6	367.5	0.0	0.0
100.0	552.6	298.4	0.0	0.0
118.8	354.2	258.7	0.0	0.0
170.8	342.2	254.5	0.0	0.0

Entrainment ends at 170.82 seconds

a. The DEHLG reflood mass and energy data is maintained to demonstrate the post-blowdown containment response.

Table 6.2-55
MAXIMUM TIME DELAYS FOR EFFECTIVENESS OF QUENCH SPRAY SYSTEM

Event	Event Time (sec)	Cumulative Time (sec)
Accident & Loss of Station Power	0	0
CDA Signal[a]	3.0	3.0
Emergency Bus Undervoltage Response Time	3.1	3.1
Emergency Diesel Generator (EDG) Start[b]	10.0	13.1
EDG Breaker Closure Time	0.2	13.3
QS Pump Sequencing Timer Delay[c]	15.75	29.05
QS Pump to Rated Flow	3.0	32.05
Motor-Operated Valve Fully Opened[d]	40	53.0
QS Piping Fill Time and Flow Delivered to Containment Atmosphere	39.05	71.1[e]

a. CDA safety analysis limit of 30.0 psi is reached in less than 3.0 seconds in Section 6.2.2.6.3.
b. The emergency diesel generators are capable of reaching rated voltage and speed within 10 seconds after loss of offsite power, as described in Section 8.3.1.1.2.1.
c. The 15.75-second delay to initiate a start signal for the quench spray pumps accounts for the diesel load sequencer set-point of 15 seconds with 0.75 second uncertainty.
d. The motor-operated valves will begin to open on receipt of a CDA signal and have no diesel load sequencing delay. Ten of the 53.3 seconds are for the diesel generator to reach rated voltage and speed. The valves open concurrently with pump acceleration.
e. This result assumes a loss of offsite power concurrent with accident initiation. The containment response analyses and LOCA dose consequences analyses use a more conservative time for QS system effectiveness.

Figure 6.2-1
DECAY HEAT GENERATION AFTER SHUTDOWN ANS-5.1 - 1979

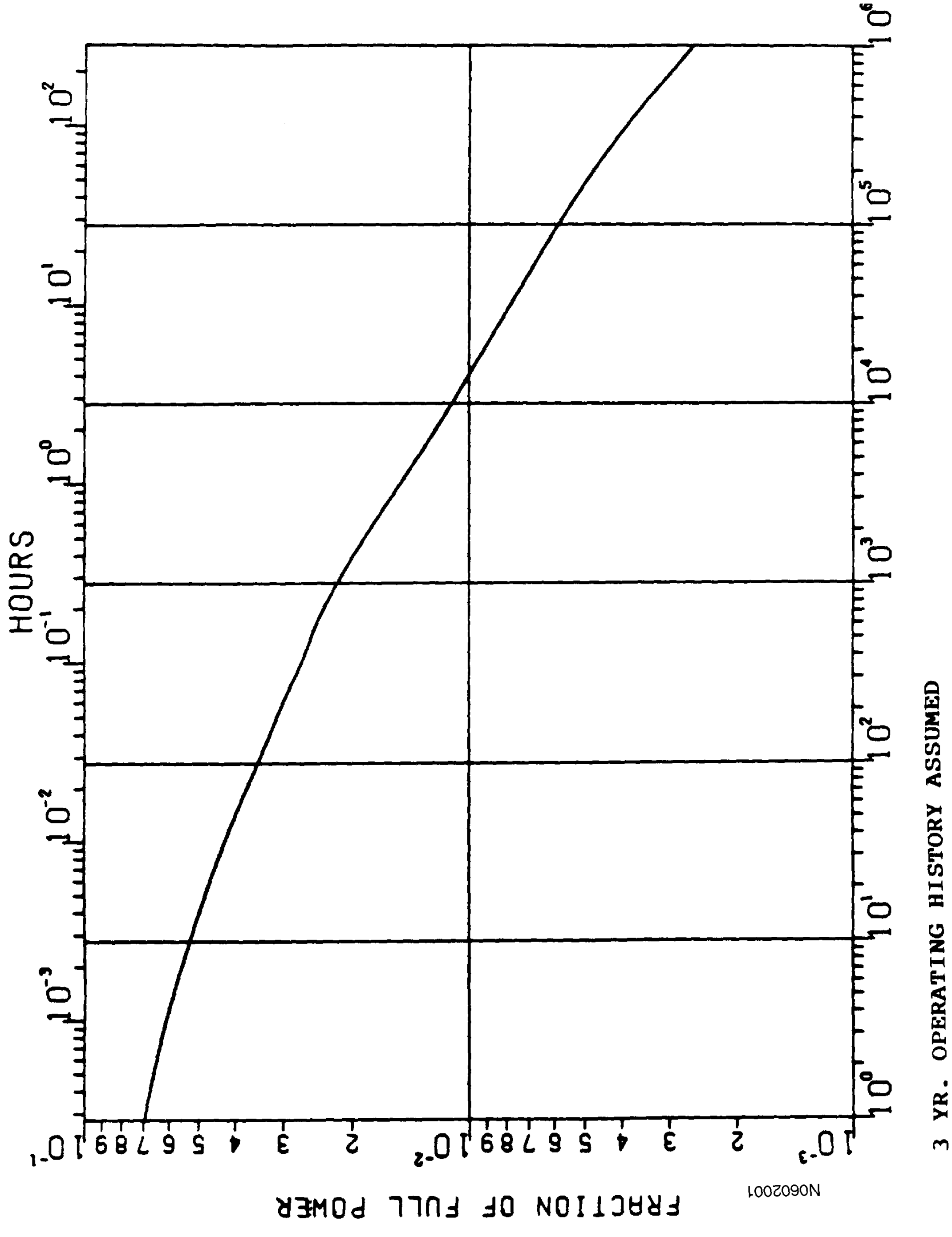


Figure 6.2-2
CONCEPTUAL FLOW CHART OF THREED COMPUTER CODE

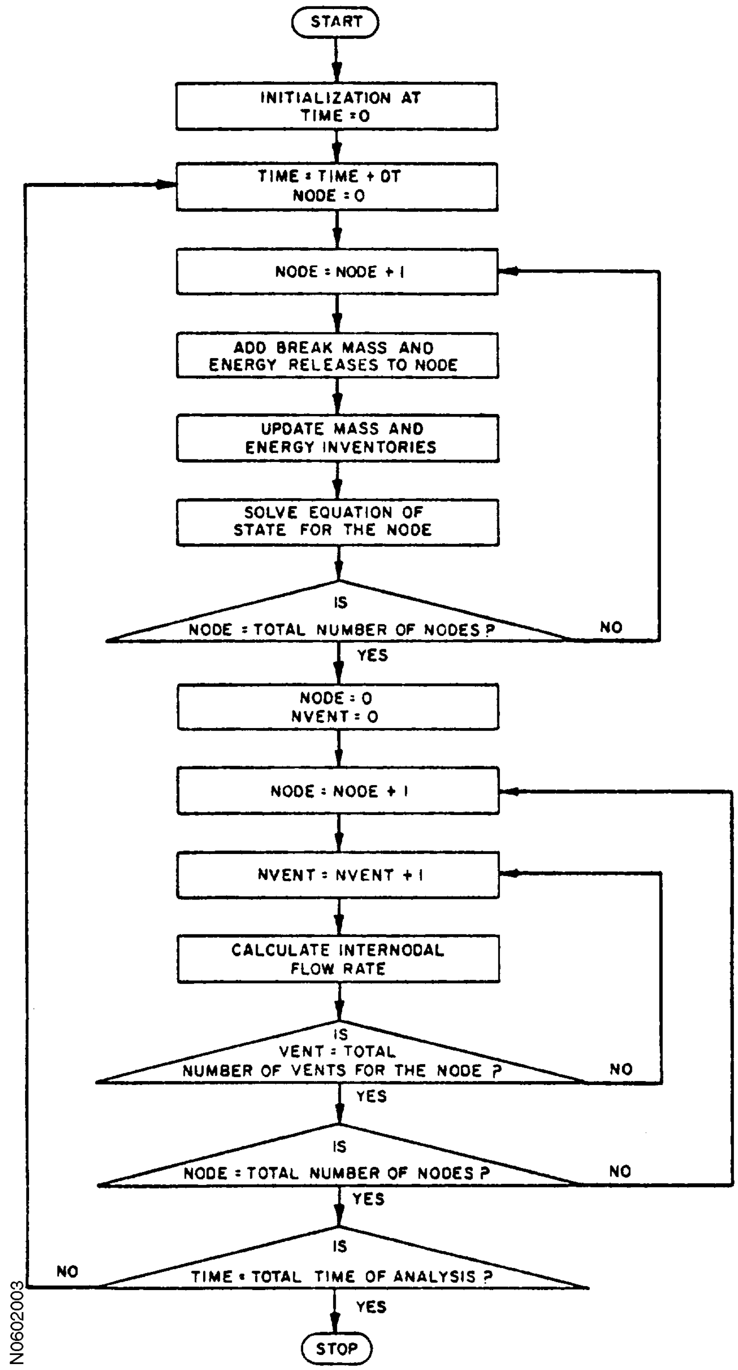


Figure 6.2-3
COMPARISON OF DIFFERENTIAL PRESSURES RMS 22-111.1 (NODES 1-2)

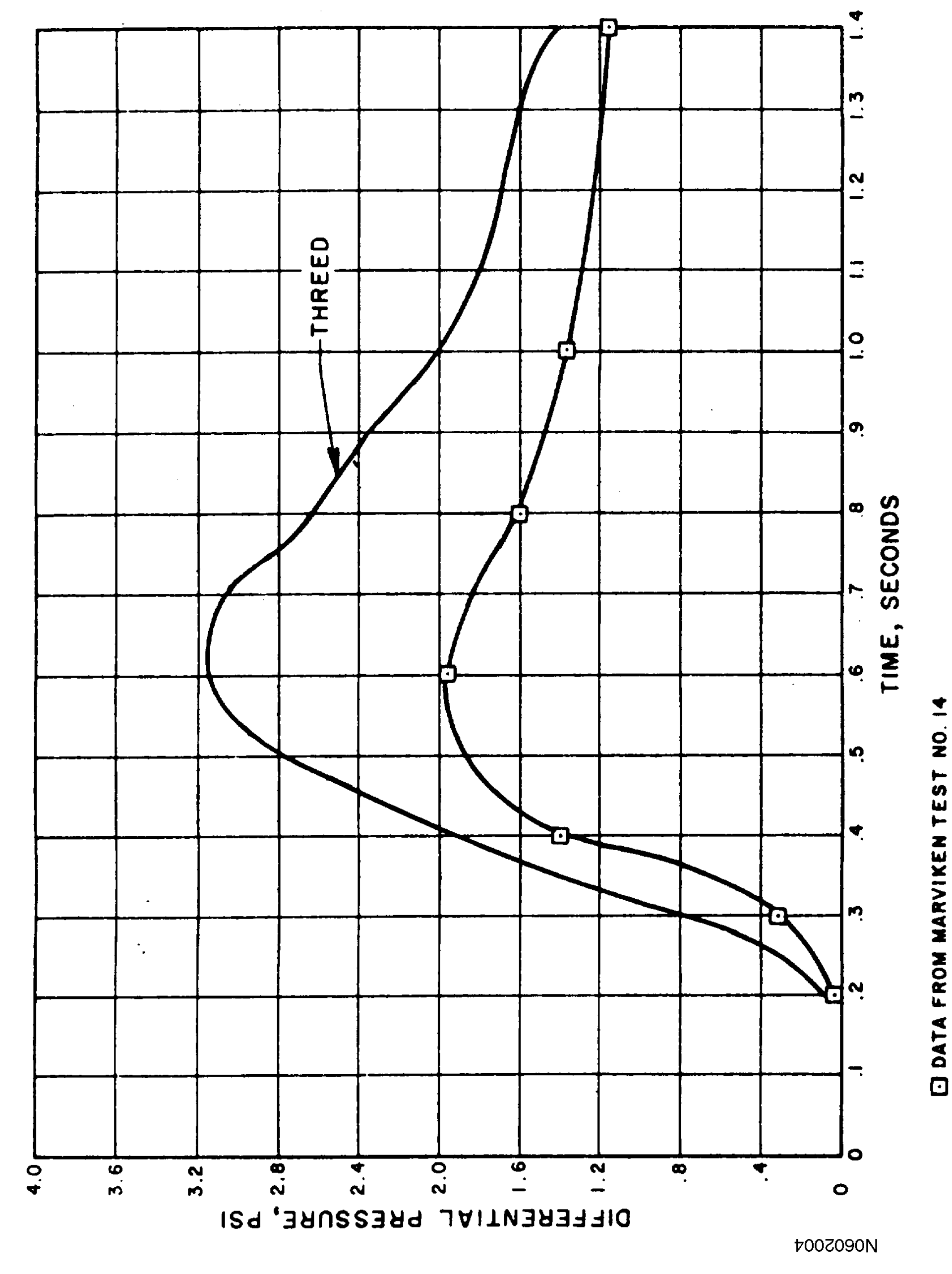


Figure 6.2-4
COMPARISON OF DIFFERENTIAL PRESSURES RMS 22-121 (NODES 1-4)

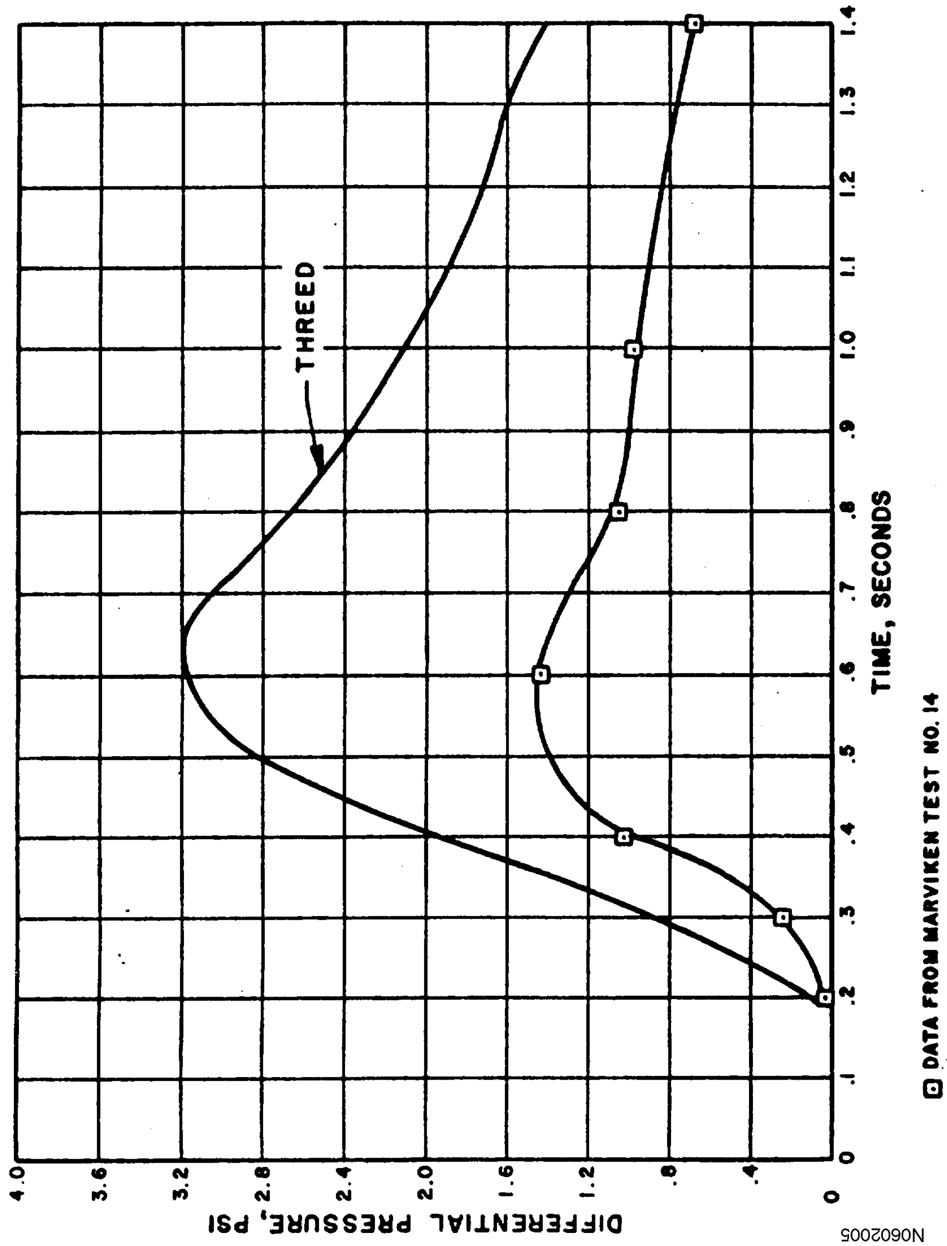


Figure 6.2-5
COMPARISON OF DIFFERENTIAL PRESSURES RMS 122-124 (NODES 1-5)

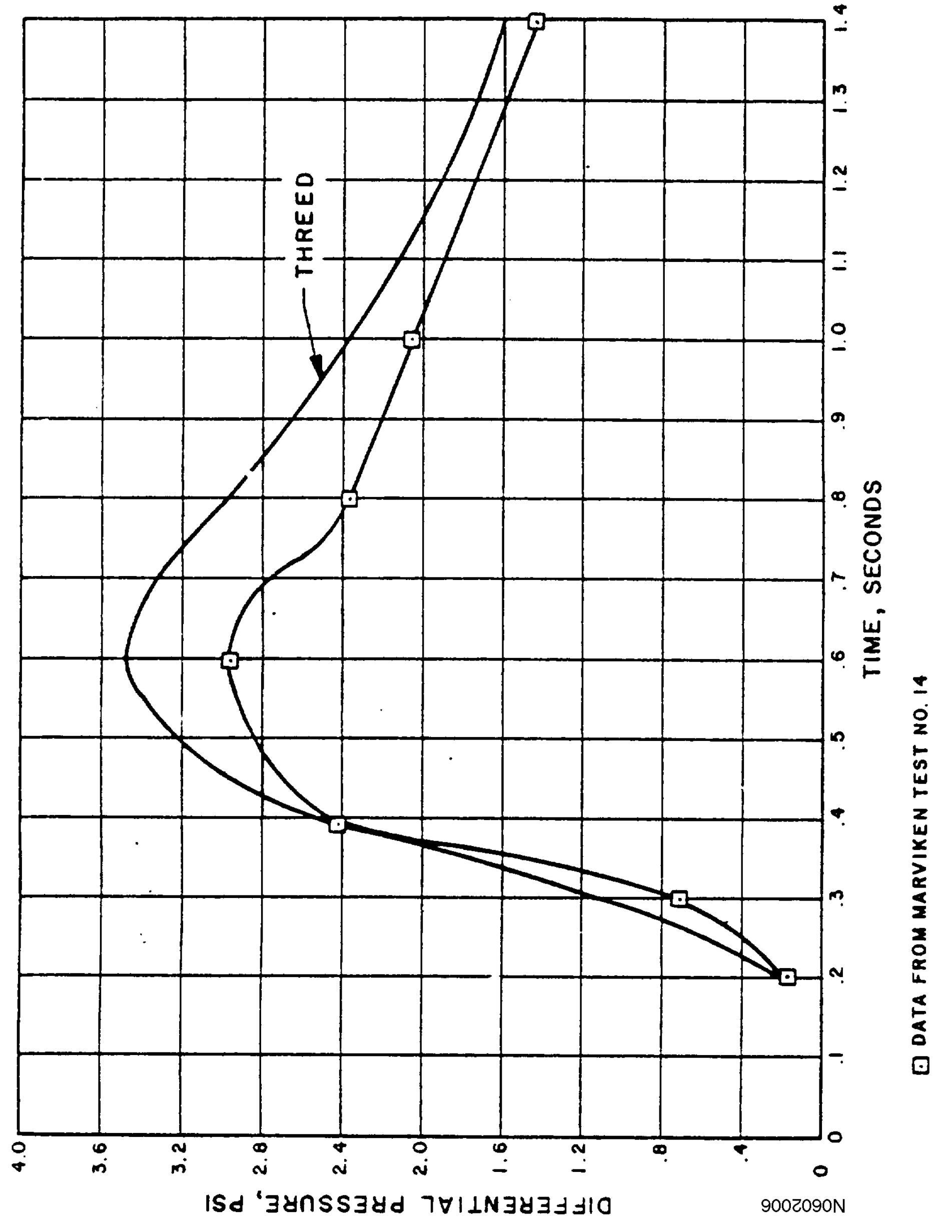


Figure 6.2-6
LIMITING CONTAINMENT TEMPERATURE FOR A MAIN STEAM LINE BREAK
102% OF 2898 MWT CORE POWER, 0.6 ft^2 DER

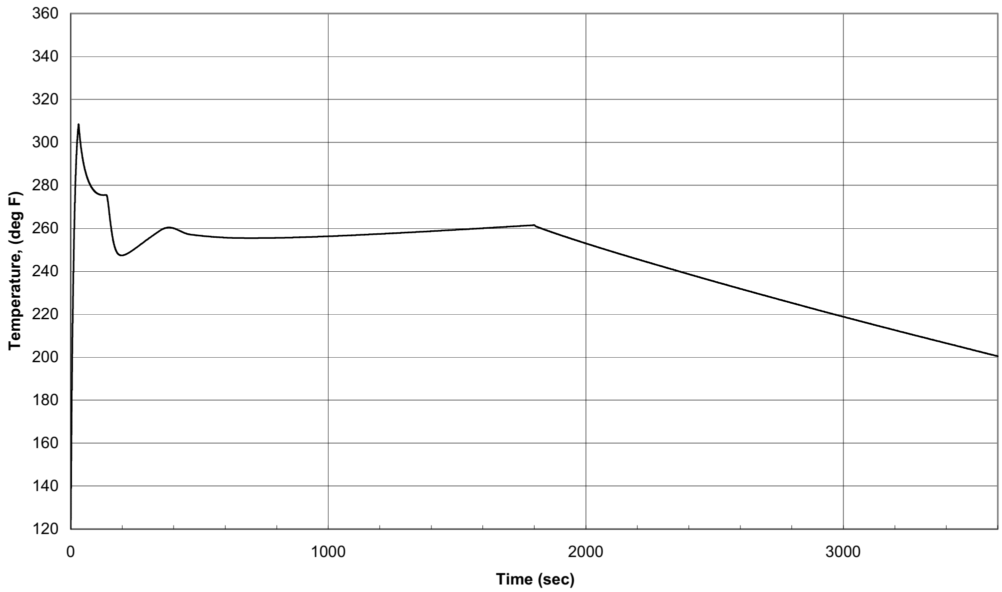


Figure 6.2-7
LIMITING CONTAINMENT PRESSURE FOR A MAIN STEAM LINE BREAK
30% OF 2898 MWT CORE POWER, 1.4 ft^2 DER

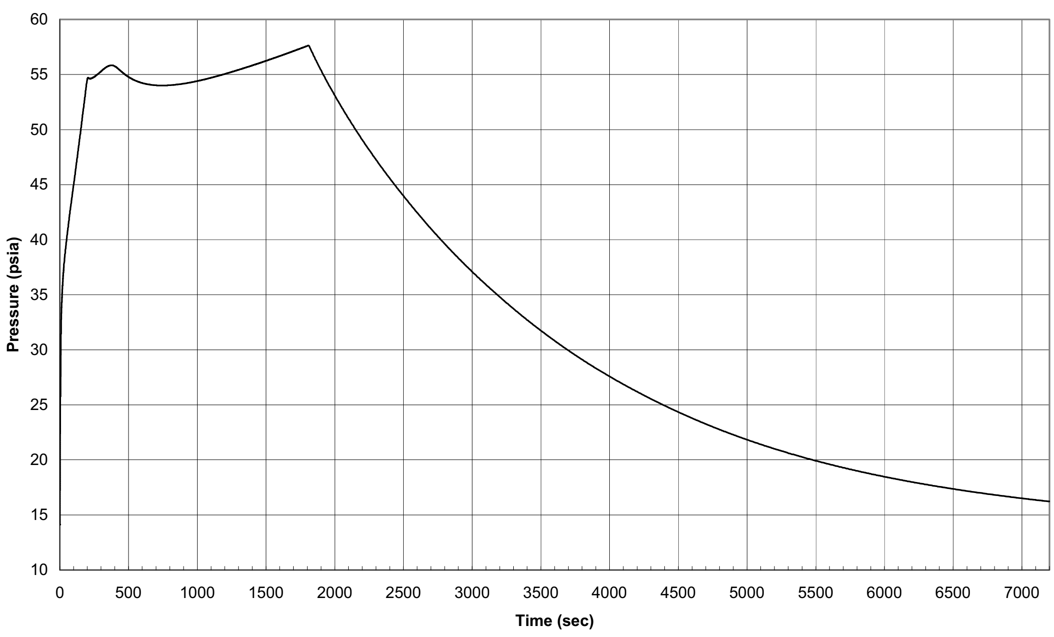


Figure 6.2-8
UPPER REACTOR CAVITY NODALIZATION STUDY
PEAK PRESSURE DIFF. VS. NUMBER OF NODES

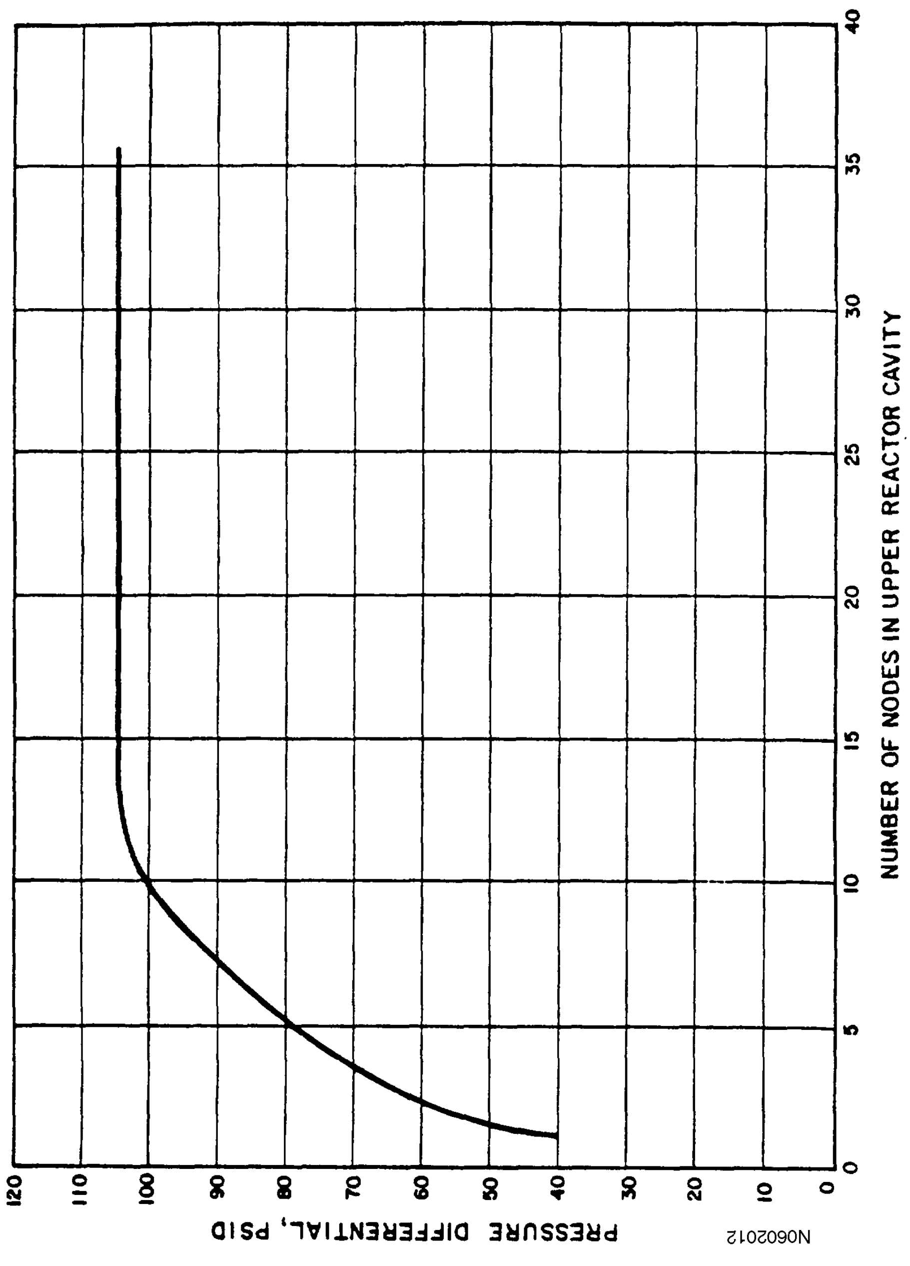
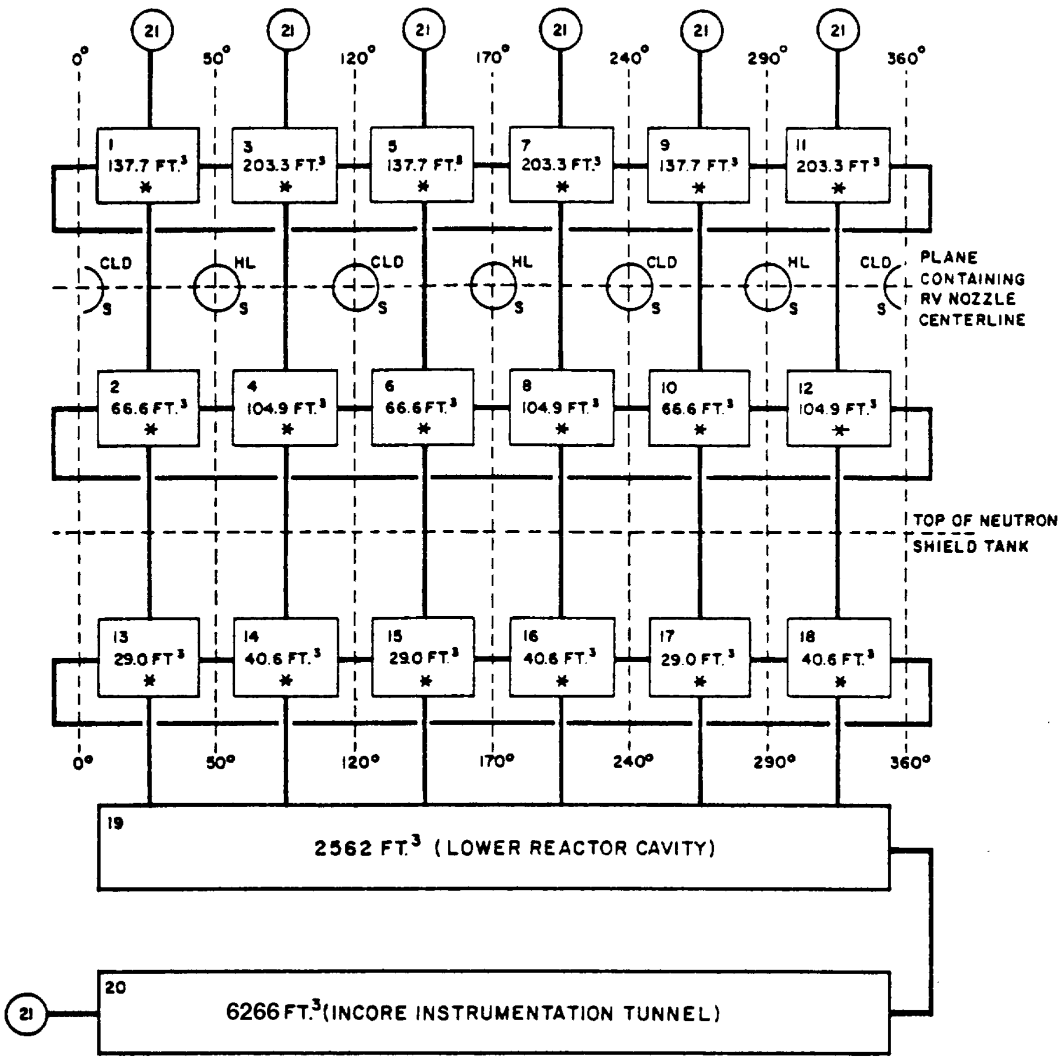


Figure 6.2-9
12-NODE UPPER REACTOR CAVITY MODEL (21 TOTAL NODES)



NOTES:

1. *-NODE RECEIVES 25% OF THE BLOWDOWN
2. THE VOLUME OF NODE 21 (CONTAINMENT) IS 1,814,000 FT.³
3. ANGLES SHOWN ABOVE NODES 1,3,5,7,9 & 11, BELOW NODES 13-18 ARE THE ANGLES FROM THE CENTERLINE OF THE BROKEN PIPE
4. TWELVE VENTS ARE NOT SHOWN - THESE ARE THE VENTS FROM NODES 1-12 TO NODE 21 (CONTAINMENT) THROUGH THE PIPING PENETRATIONS IN THE SHIELD WALL
5. THE FOLLOWING SYMBOL REPRESENTATIONS ARE USED:
   HL - HOT LEG
   CLD - COLD LEG DISCHARGE
   S - SUPPORTED NOZZLE

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Figure 6.2-10
12-NODE UPPER REACTOR CAVITY MODEL
PLAN VIEW—ELEVATION 256' 3-15/16"

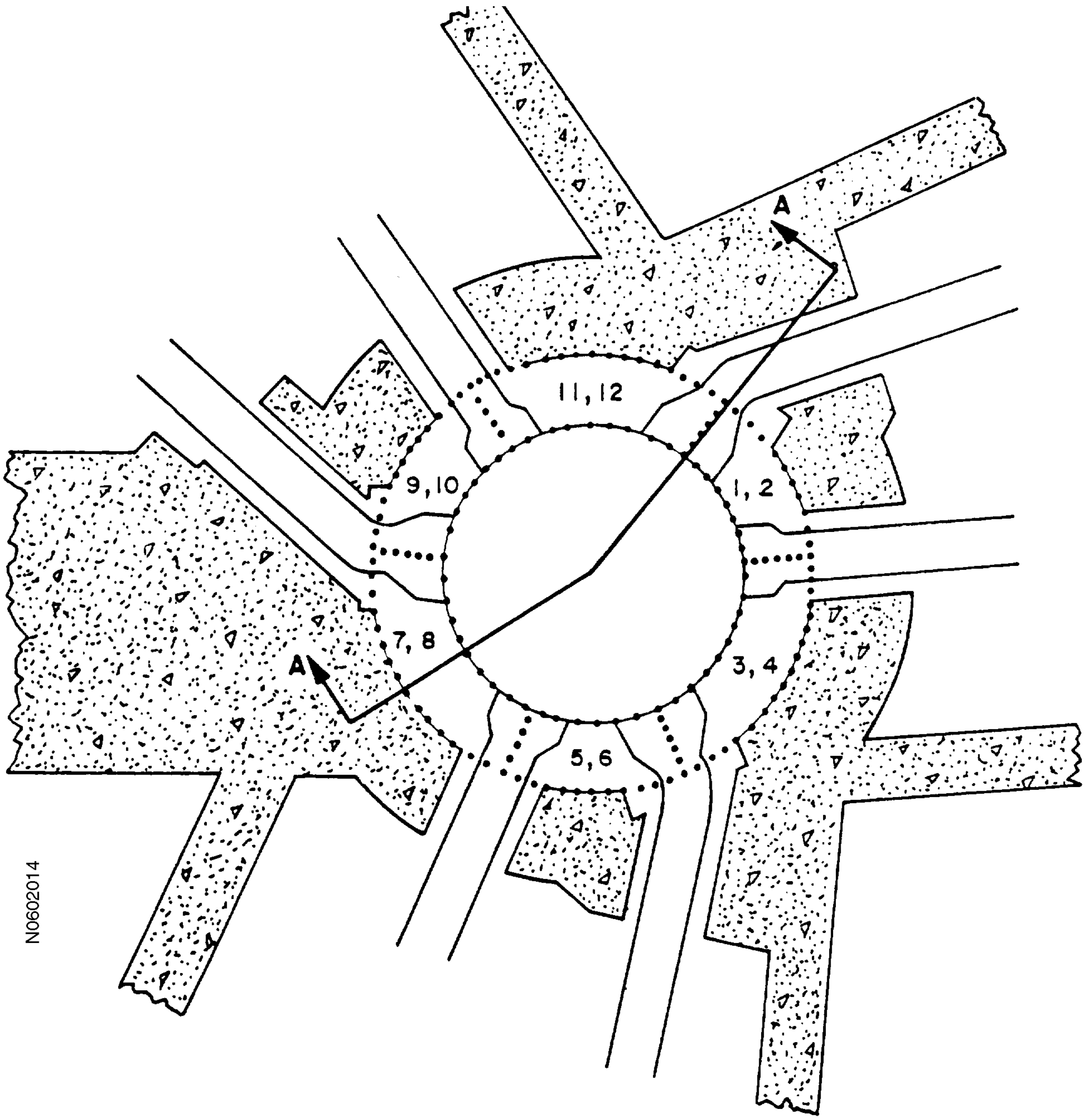


Figure 6.2-11
ELEVATION VIEW OF UPPER REACTOR CAVITY
NODALIZATION MODEL

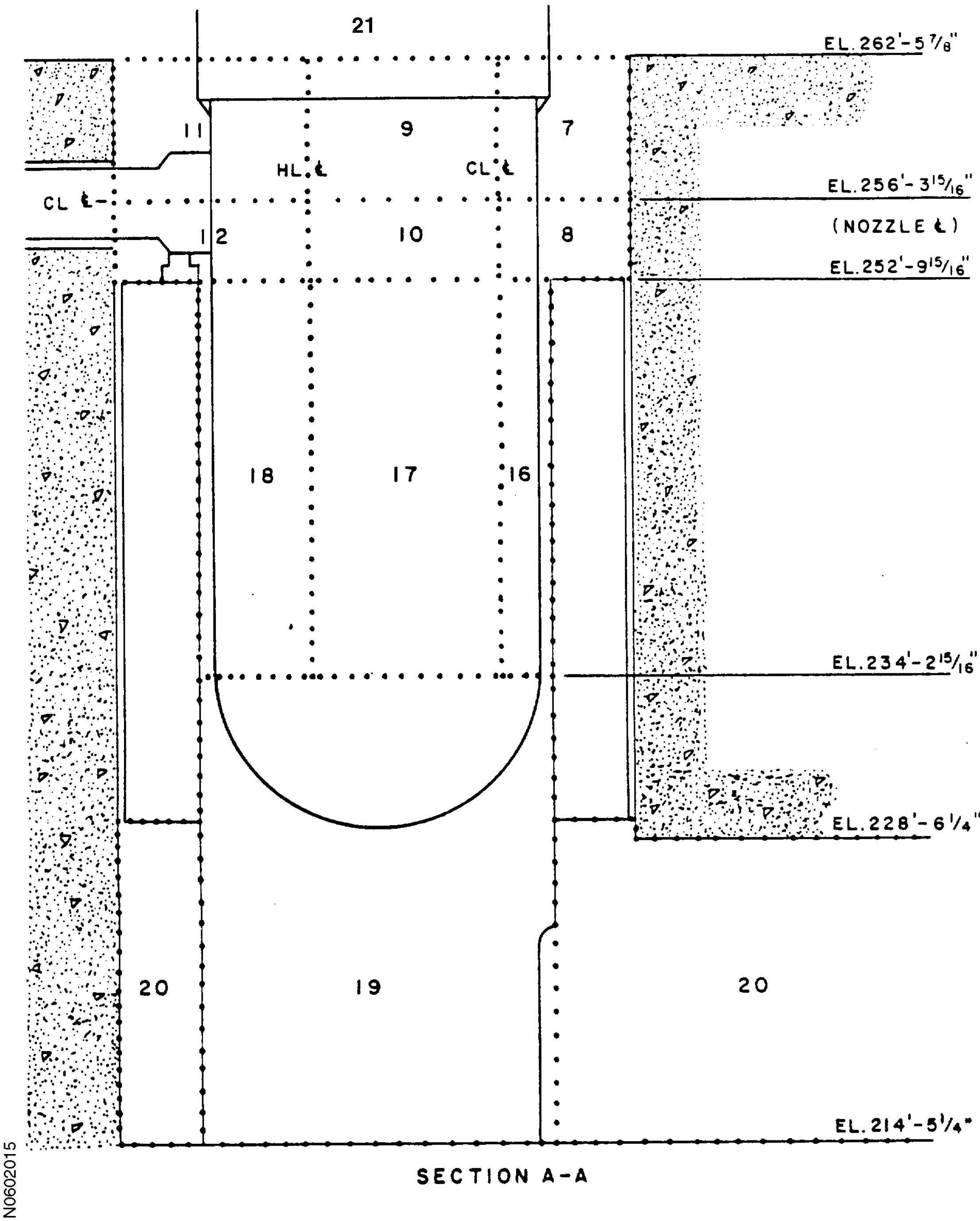


Figure 6.2-12
PRESSURE DIFFERENTIAL IN UPPER REACTOR CAVITY VS. TIME AFTER ACCIDENT
(12 NODE UPPER REACTOR CAVITY MODEL)

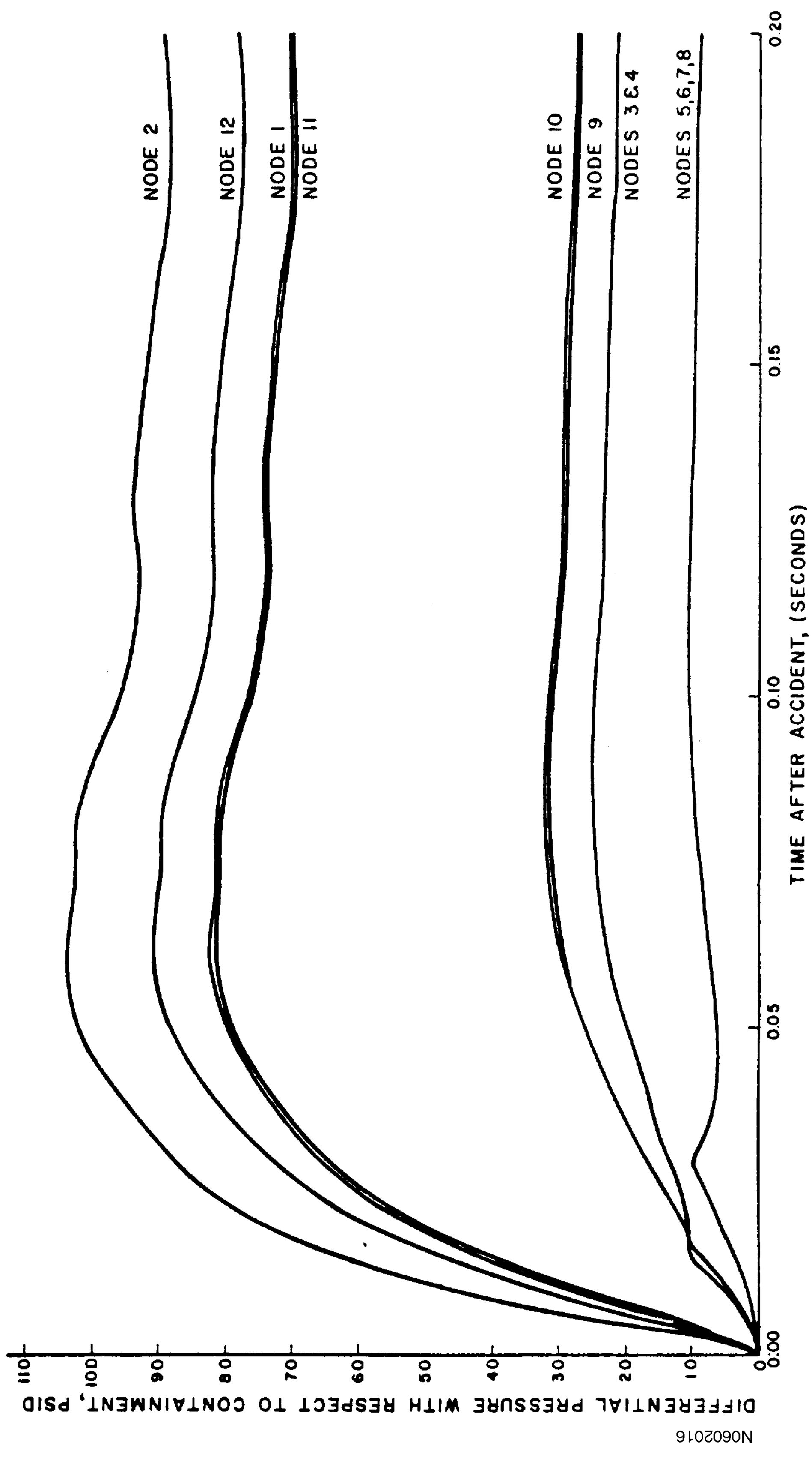


Figure 6.2-13
5-NODE MODEL FOR THE PEAK DIFFERENTIAL PRESSURE BETWEEN THE REACTOR ANNULUS AND THE CONTAINMENT

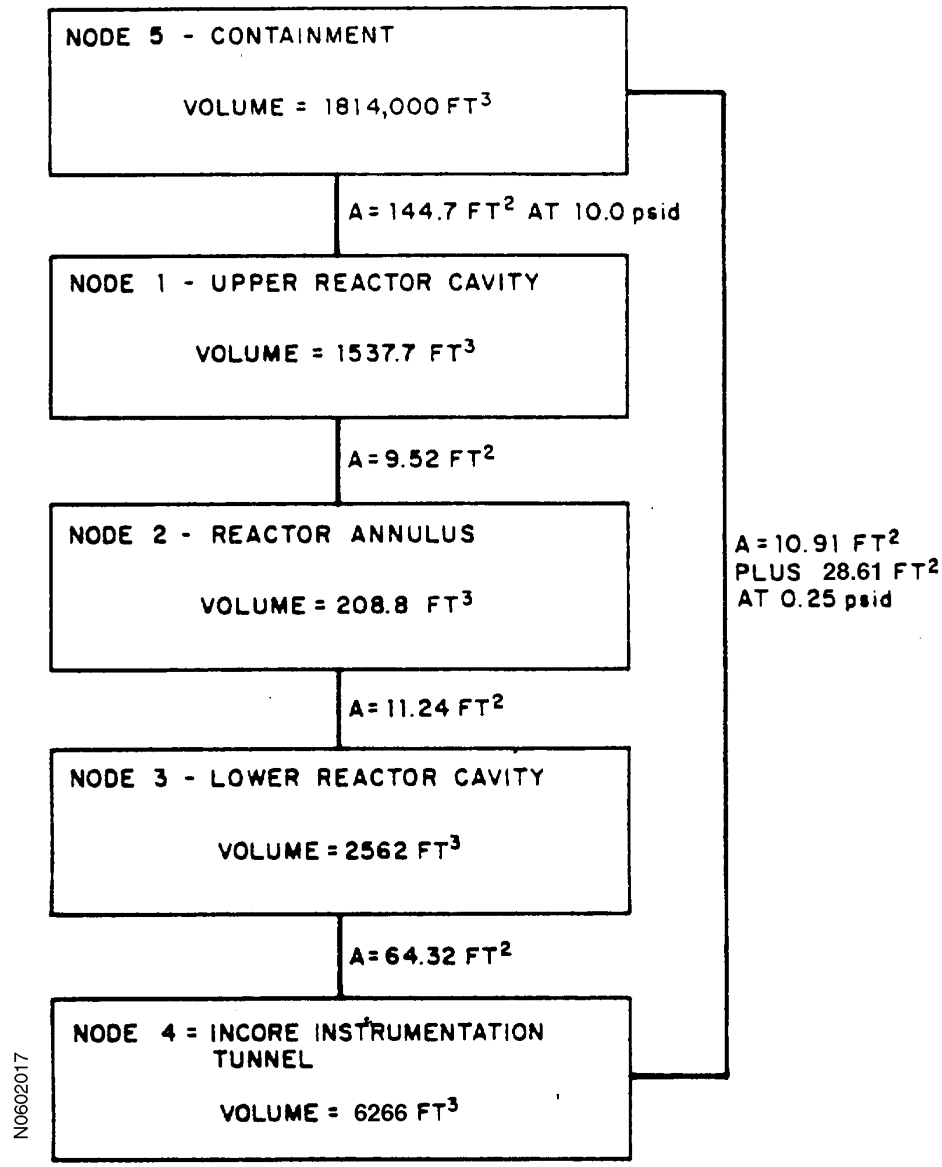


Figure 6.2-14
6-NODE MODEL FOR THE PEAK DIFFERENTIAL PRESSURE BETWEEN THE LOWER REACTOR CAVITY AND THE WALKWAY AROUND THE VESSEL SUPPORT SKIRT

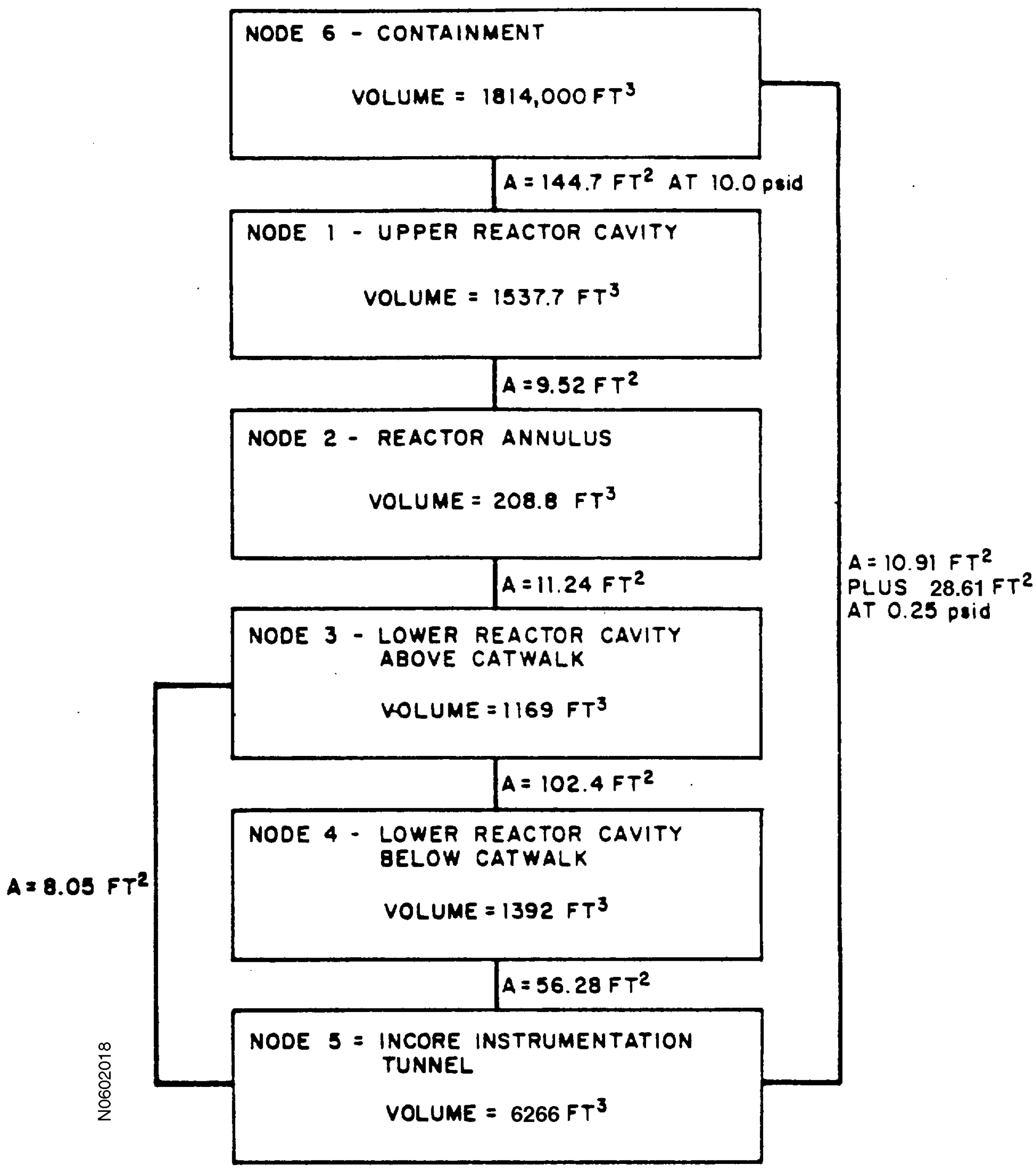


Figure 6.2-15
7-NODE MODEL FOR THE PEAK DIFFERENTIAL PRESSURE BETWEEN THE INCORE INSTRUMENTATION TUNNEL AND THIS CONTAINMENT

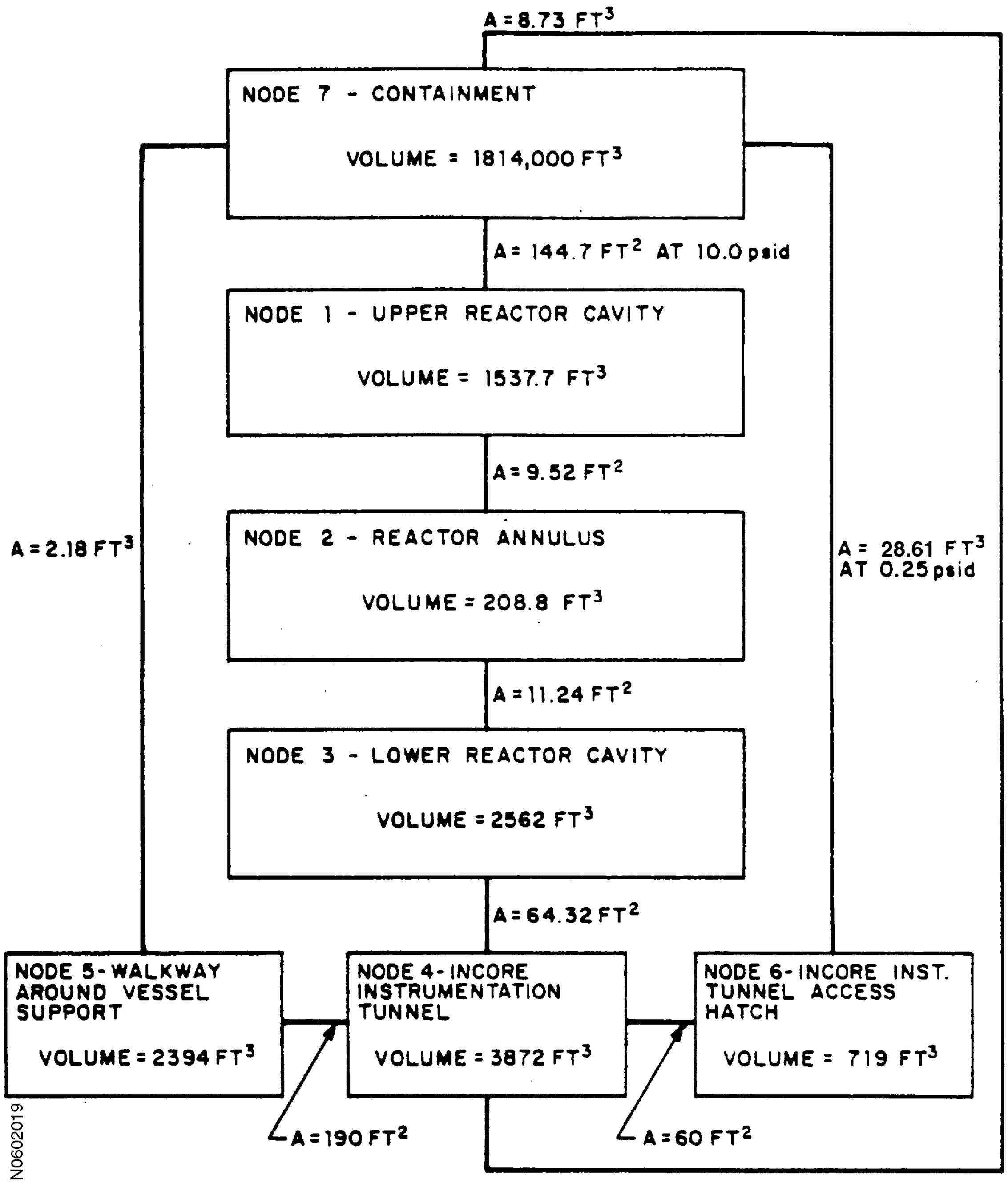
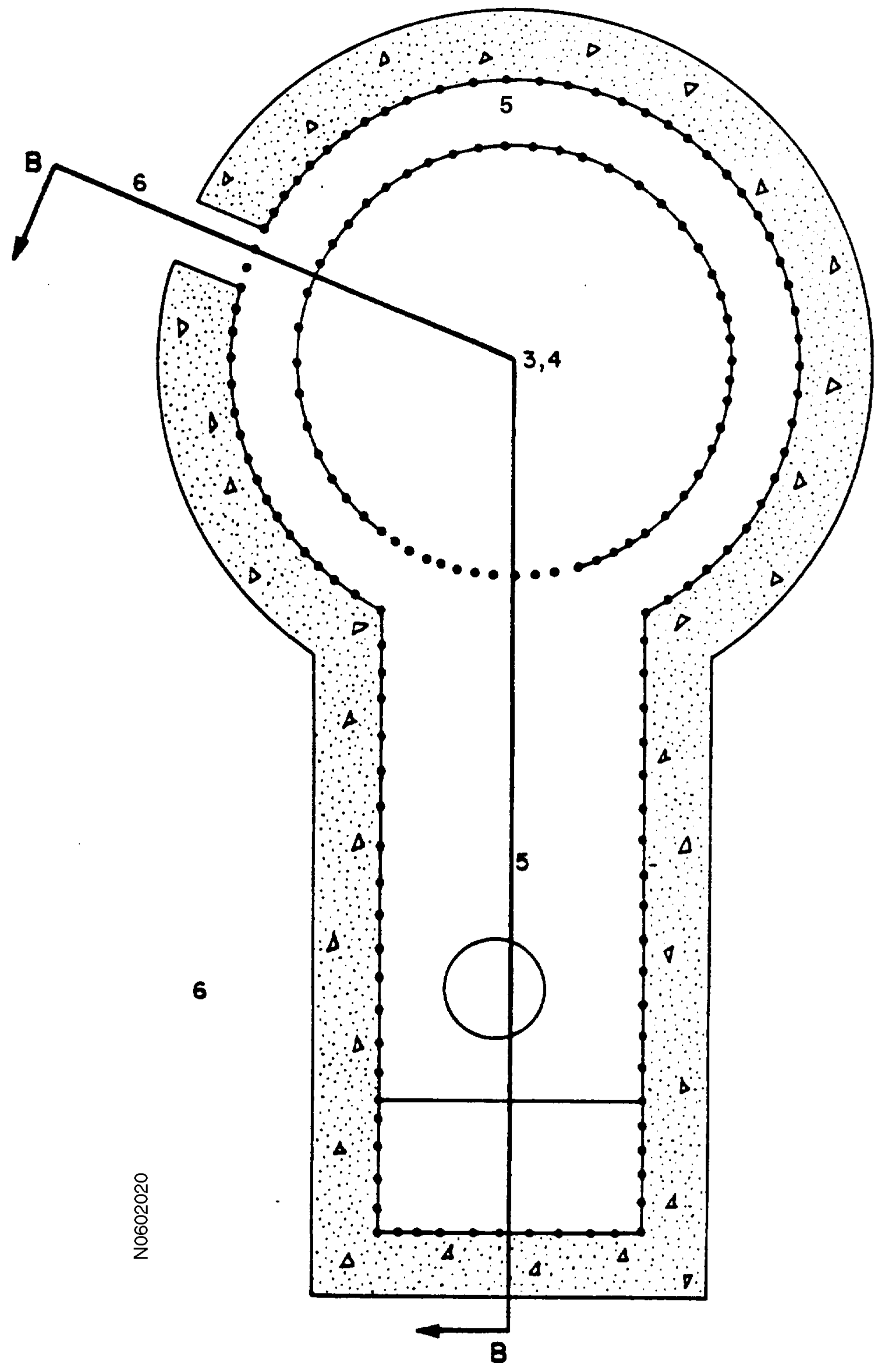


Figure 6.2-16
6-NODE MODEL (2 NODES IN THE LOWER REACTOR CAVITY)
PLAN VIEW—ELEVATION 228' 6-1/4"



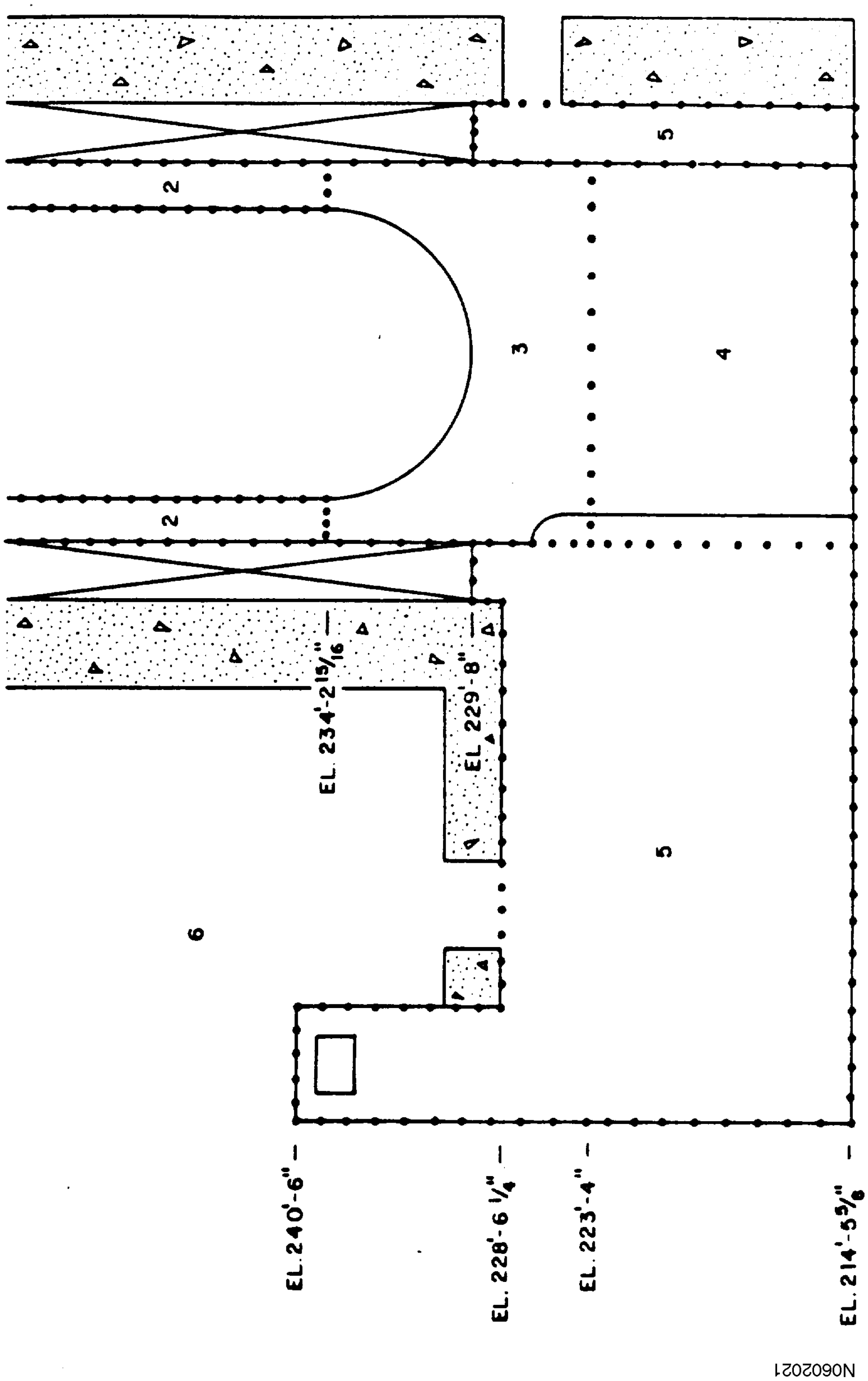
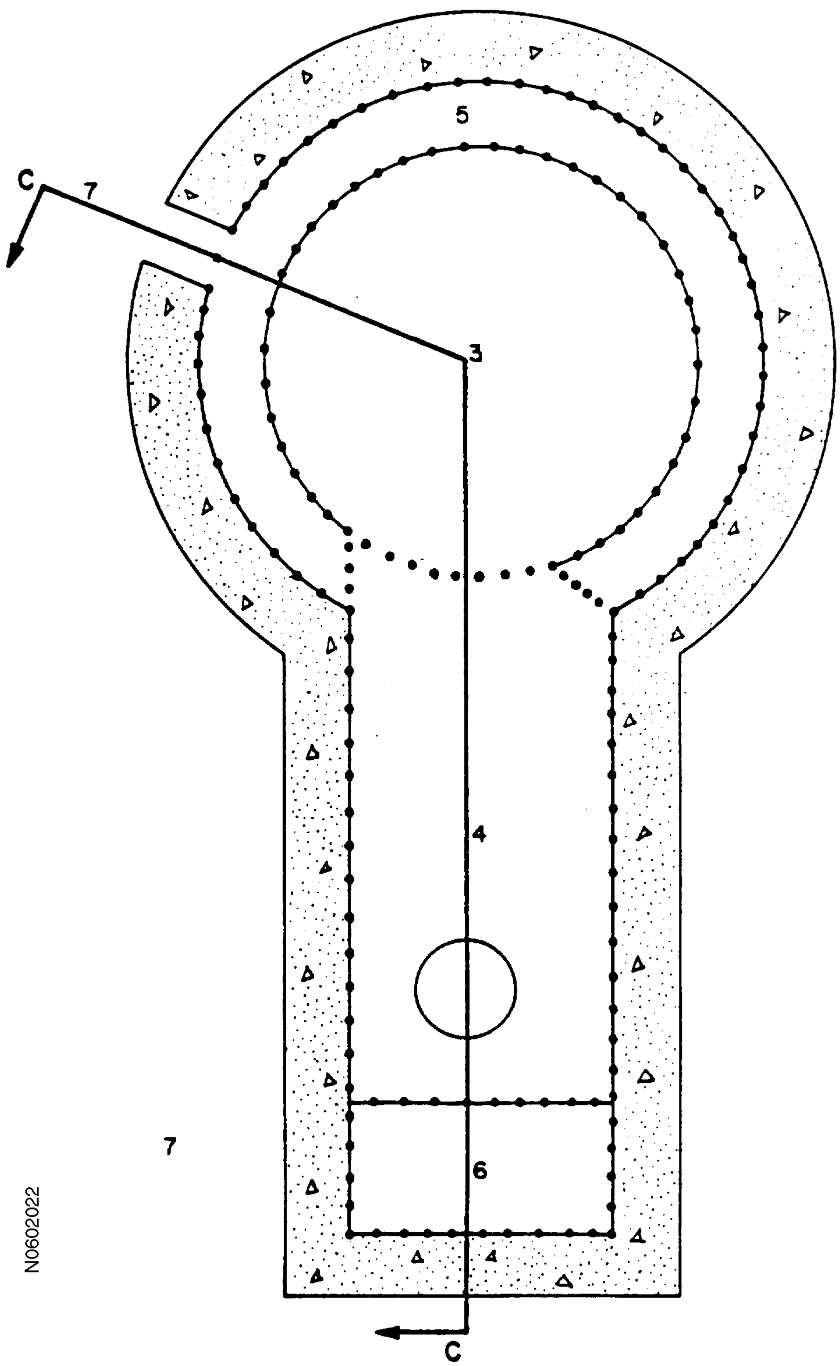


Figure 6.2-17
SECTION B-B OF FIGURE 6.2-23 6-NODE MODEL

Figure 6.2-18
7-NODE MODEL (3 NODES IN THE INCORE INSTRUMENTATION TUNNEL)
PLAN VIEW—ELEVATION 228' 6-1/4"



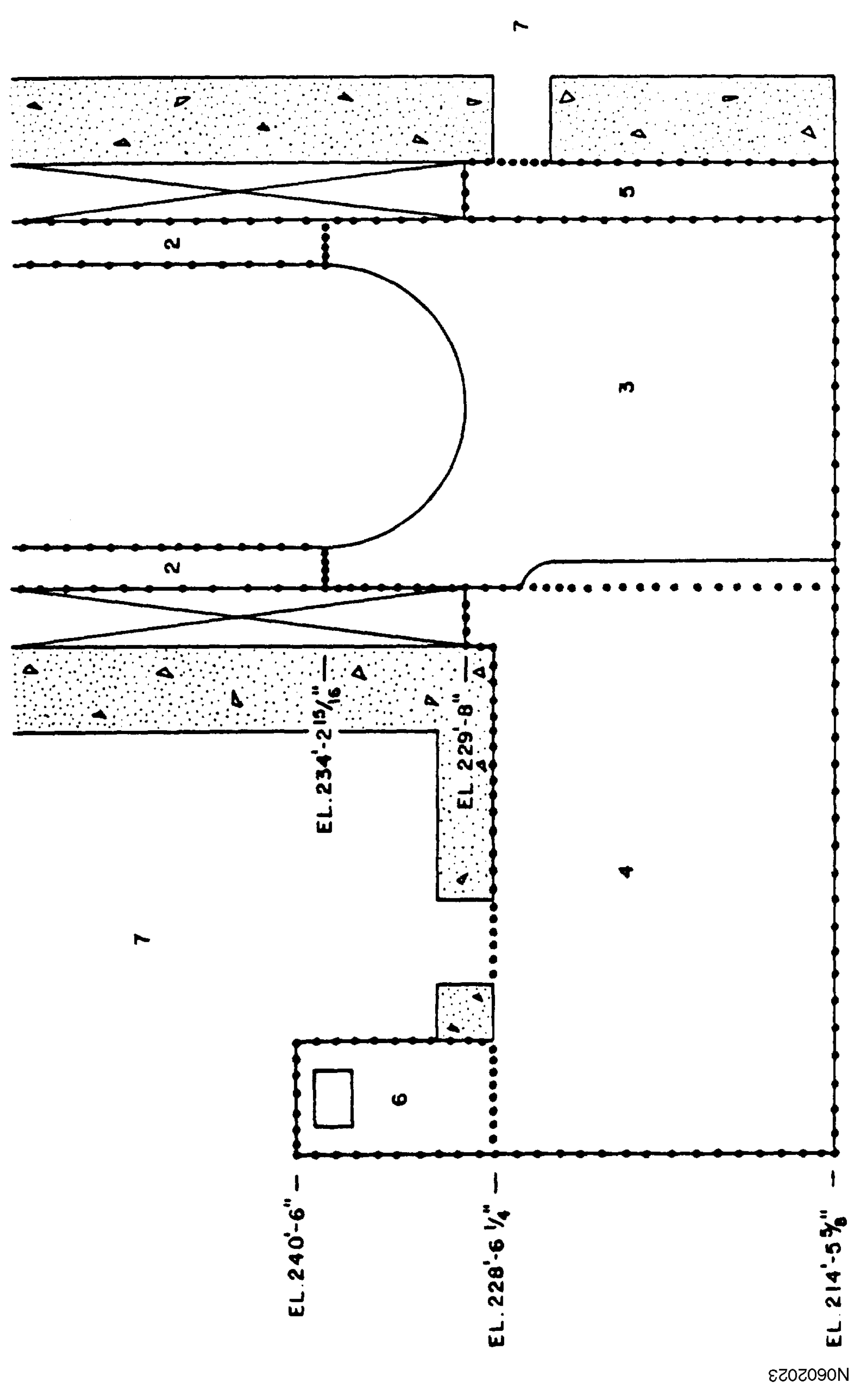


Figure 6.2-19
SECTION C-C OF FIGURE 6.2-18 7-NODE MODEL

Figure 6.2-20
PRESSURE DIFFERENTIAL IN REACTOR ANNULUS
FOLLOWING A 150 SQ. IN. CL LDR IN THE UPPER REACTOR CAVITY

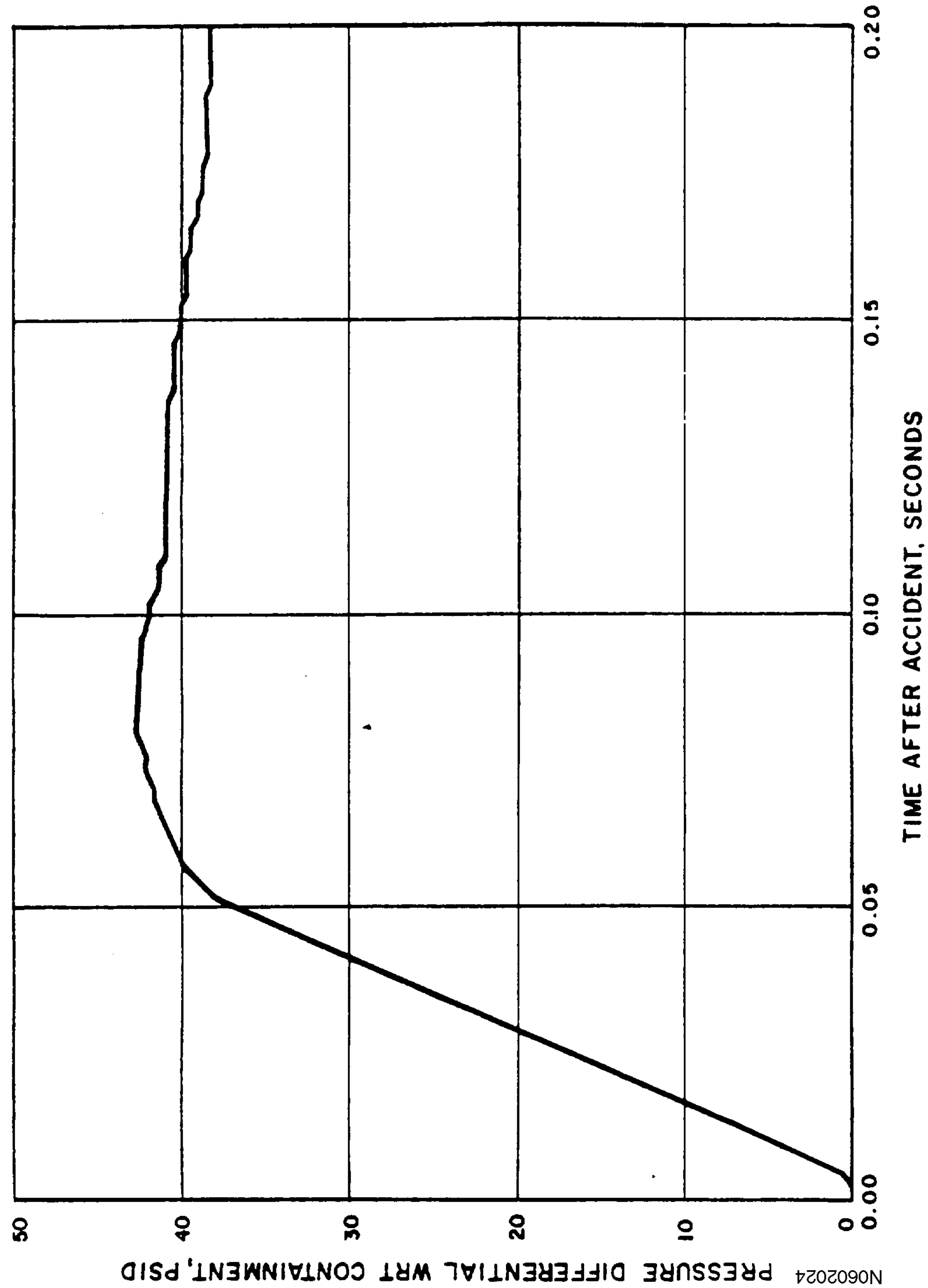


Figure 6.2-21
DIFFERENTIAL PRESSURE BETWEEN THE LOWER REACTOR CAVITY AND THE WALKWAY AROUND THE OUTSIDE OF THE VESSEL SUPPORT SKIRT

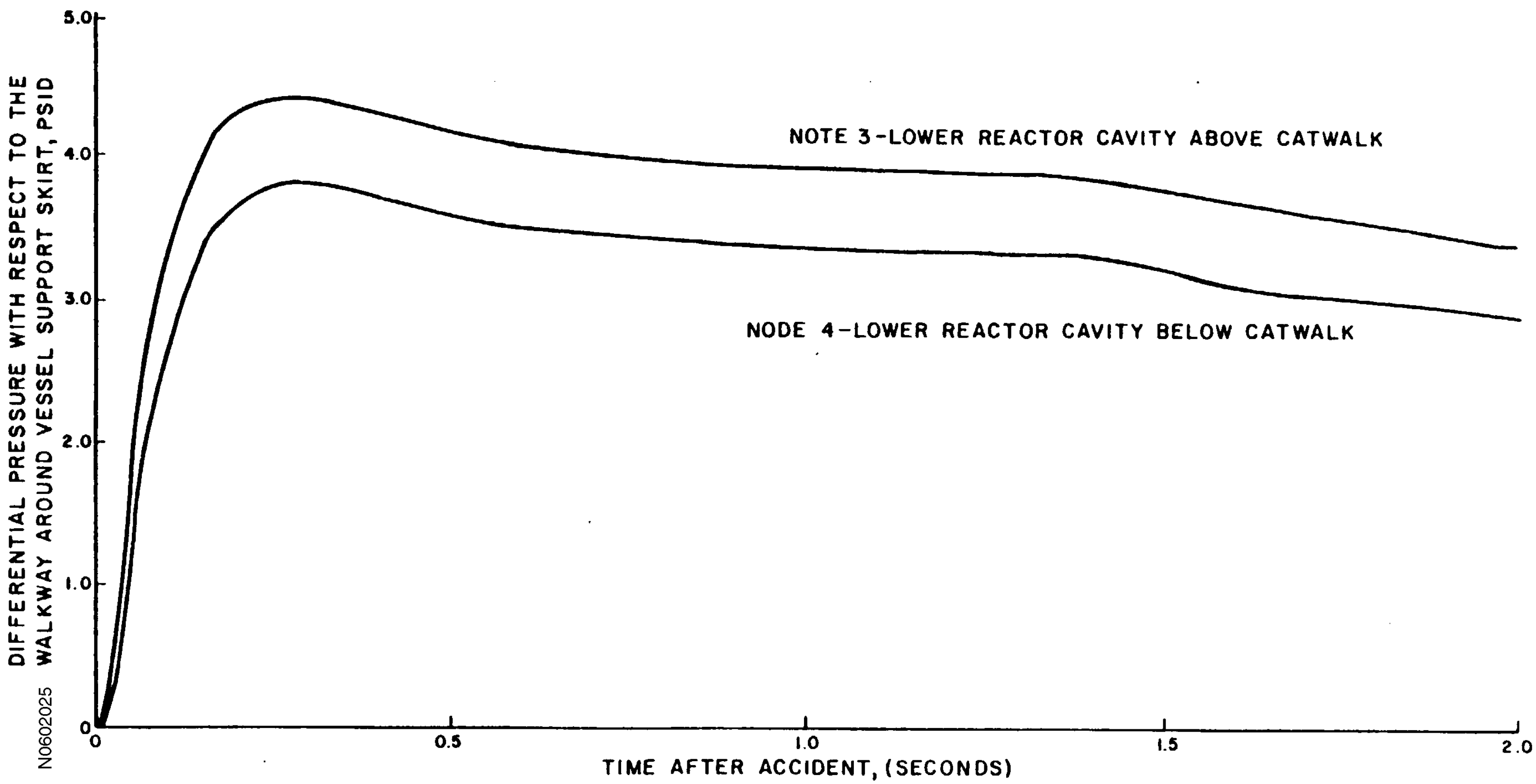


Figure 6.2-22
PRESSURE DIFFERENTIAL IN INCORE INST. TUNNEL
FOLLOWING A 150 SQ. IN. CL LDR IN THE UPPER REACTOR CAVITY

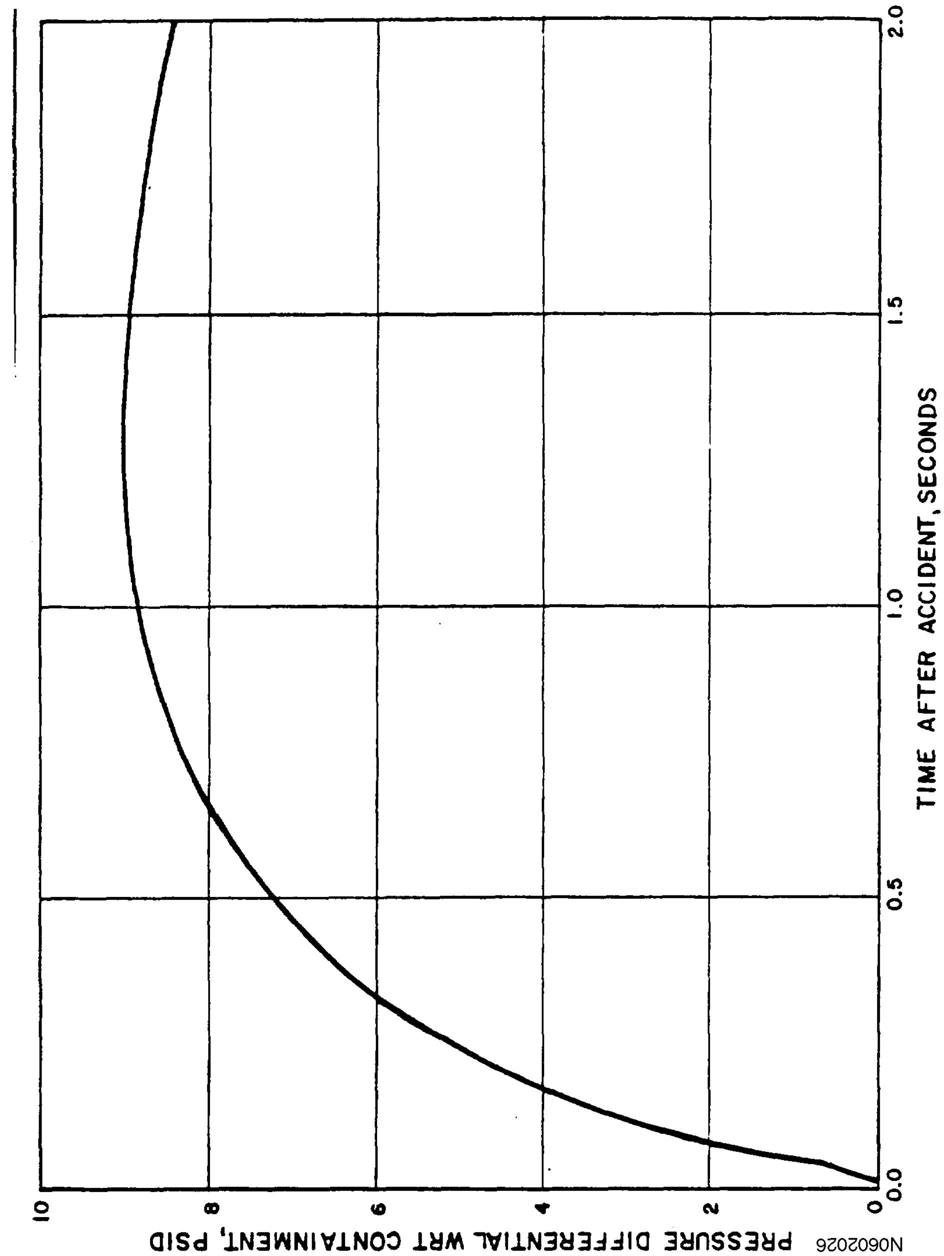


Figure 6.2-23
TYPICAL GRATING USED IN THE CUBICLES
IN THE REACTOR CONTAINMENT

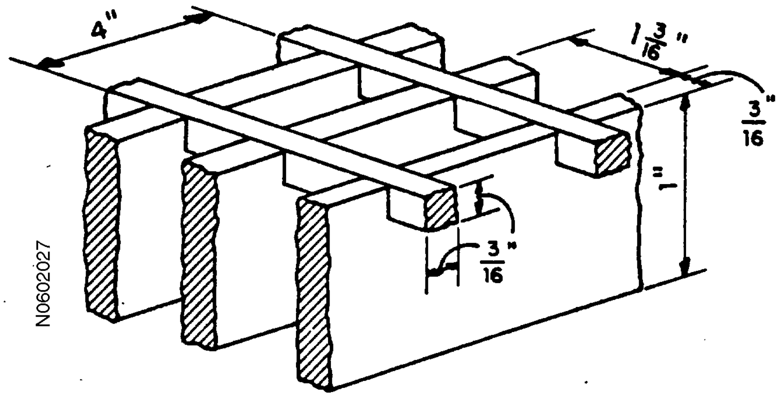


Figure 6.2-24

THICKENED GRID $\frac{l}{d_h} > 0.015$

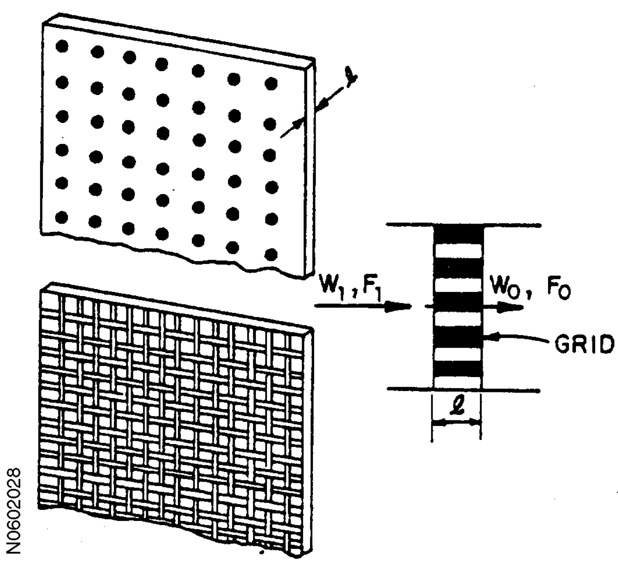
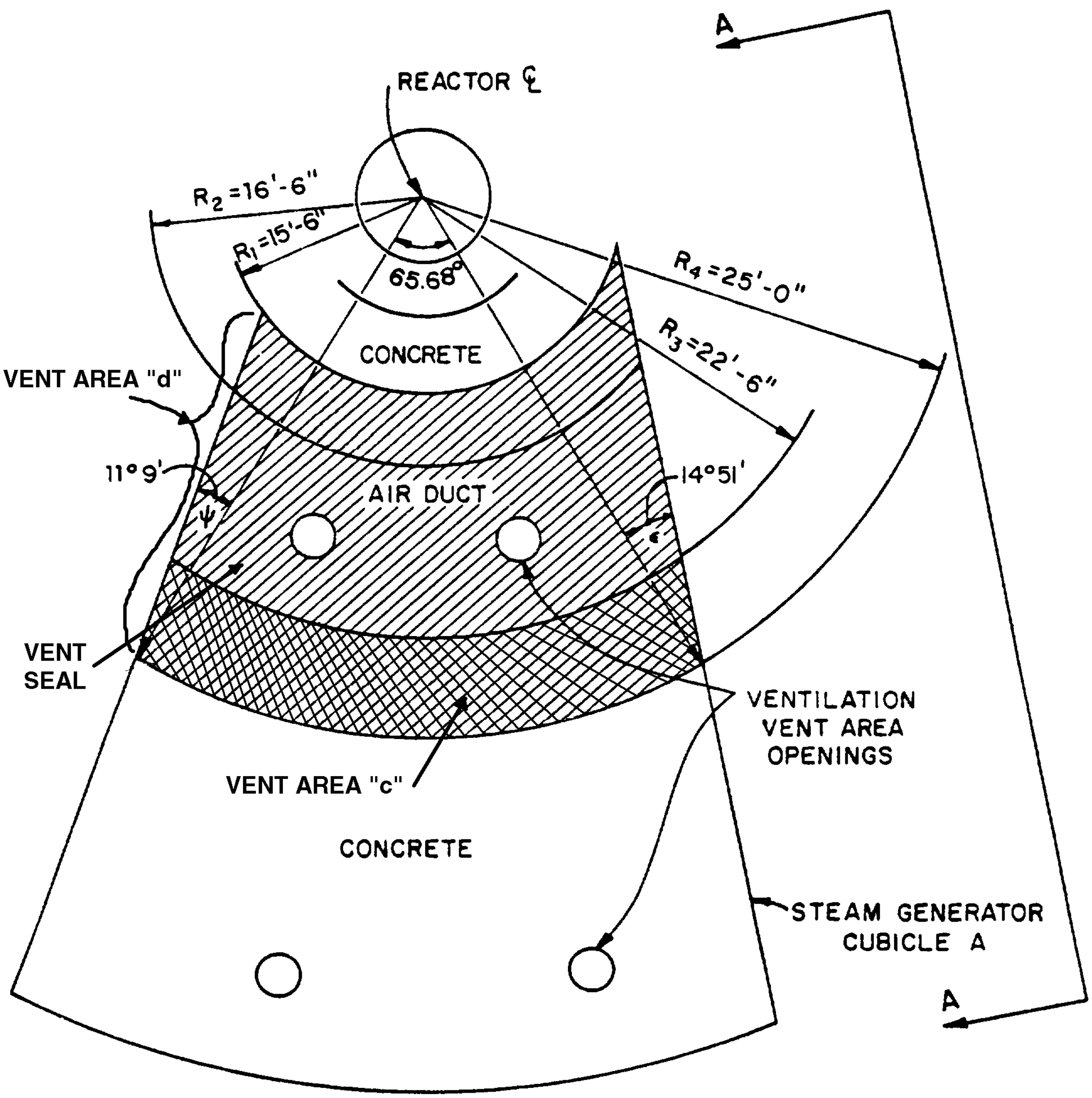


Figure 6.2-25
PLAN VIEW OF ELEVATION 242'-6"
INDICATING VENT AREAS AROUND AIR DUCT



NOTE:
THE VENT SEAL WHICH EXISTS AT ELEVATION 242'-6" IS INDICATED BY THE FOLLOWING SYMBOL: ▨. THERE IS AN AIR DUCT BELOW THE VENT SEAL AT ELEVATION 236'-8". THE AIR DUCT IS INCLUDED IN THIS VIEW TO PROVIDE A MORE ACCURATE DESCRIPTION OF THE VENT AREA.

N0602029

Figure 6.2-26
SECTION A-A OF FIGURE 6.2-25 INDICATING VENT AREAS AROUND AIR DUCT

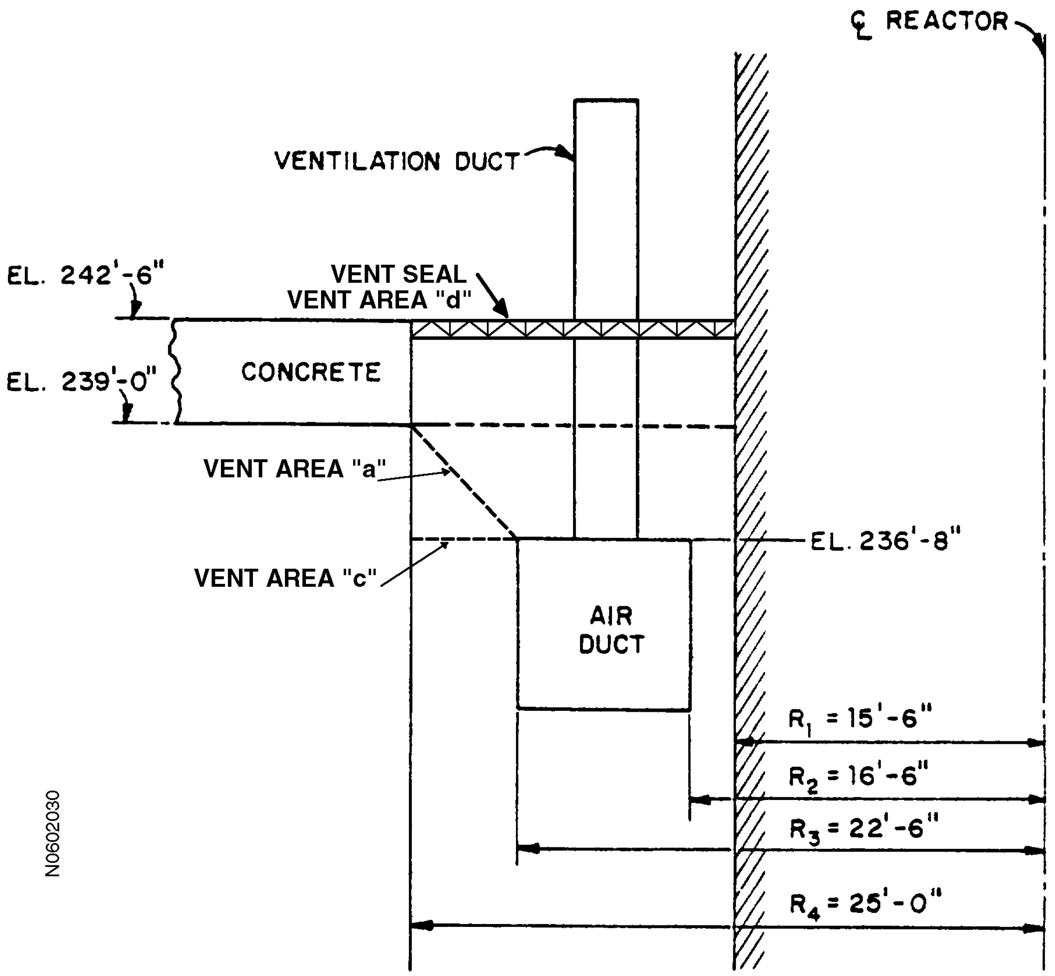
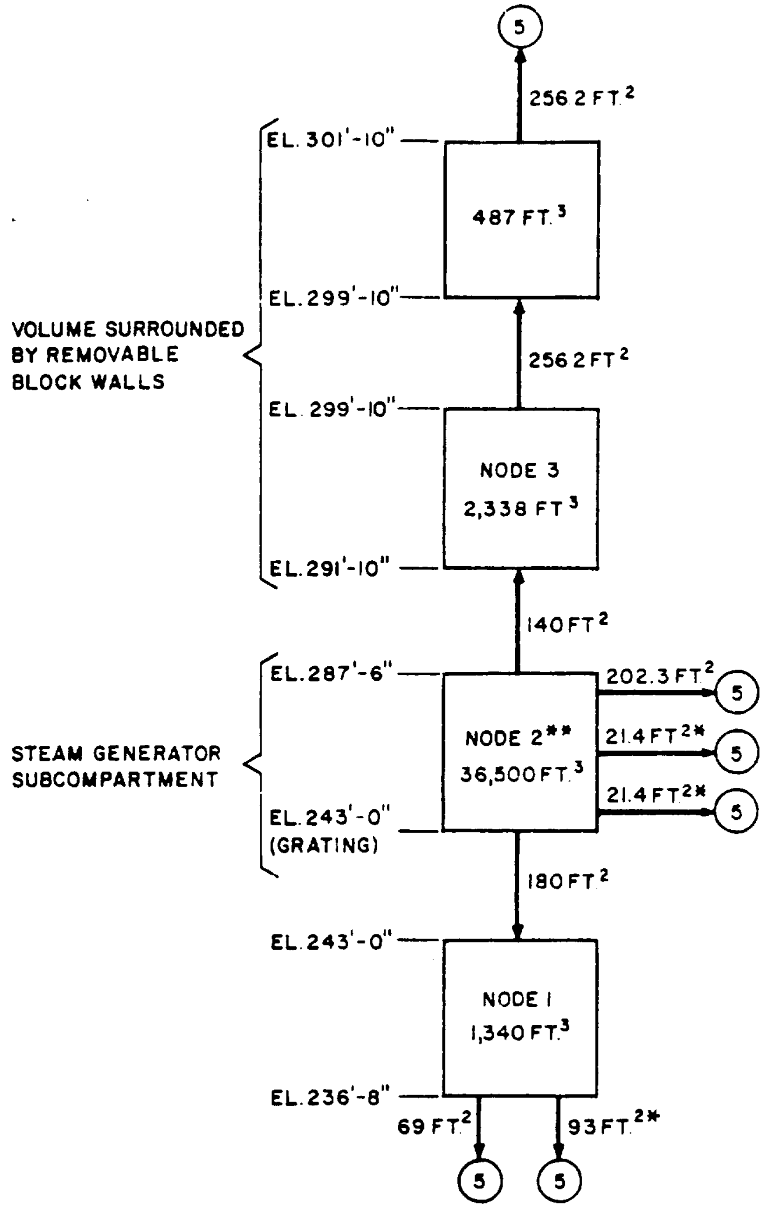


Figure 6.2-27
2-NODE MODEL OF VOLUME SURROUNDED BY
REMOVABLE BLOCK WALL FOR DIFFERENTIAL PRESSURE ANALYSIS



NOTES:
* - NO FLOW IS CONSIDERED BEFORE THE PANEL BLOWS OUT AT 5 PSID
** - NODE RECEIVES 100% OF THE BLOWDOWN

NODE No.

1	VOLUME BELOW THE GRATE FROM EL. 236'-8" TO EL 242'-6"
2	STEAM GENERATOR SUBCOMPARTMENT FROM EL. 243'-0" TO EL. 291'-10"
3	VOLUME SURROUNDED BY REMOVABLE BLOCK WALLS FROM EL. 291'-10" TO EL. 299'-10"
4	VOLUME SURROUNDED BY REMOVABLE BLOCK WALLS FROM EL. 299'-10" TO EL. 301'-10"
5	BULK CONTAINMENT

N0602031

Figure 6.2-28
2-NODE MODEL DIFFERENTIAL PRESSURE
ACROSS THE REMOVABLE BLOCK WALLS

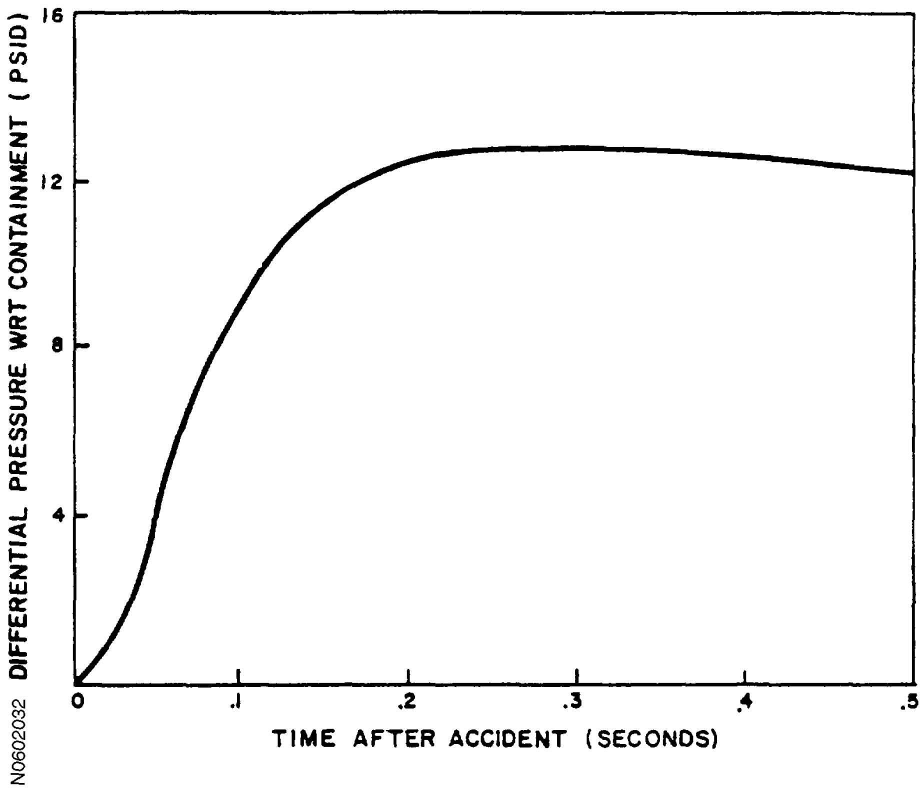


Figure 6.2-29
VERTICAL NODALIZATION STUDY
OF THE STEAM GENERATOR COMPARTMENT

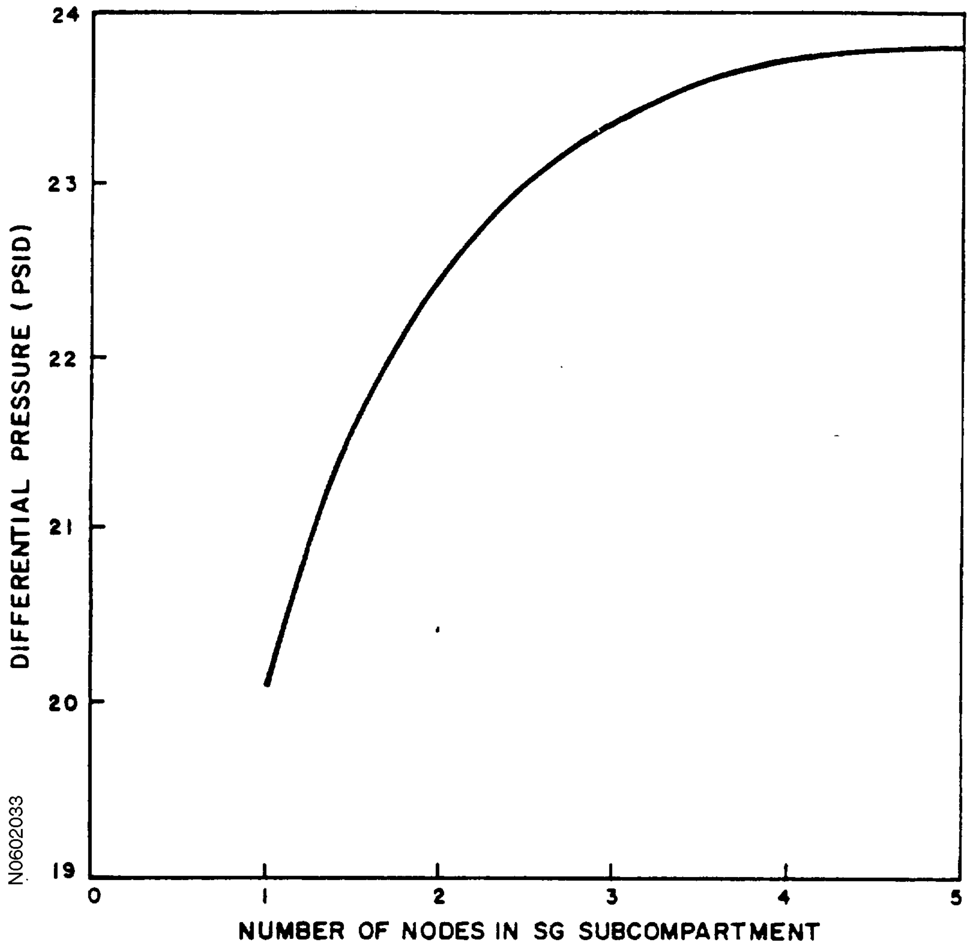
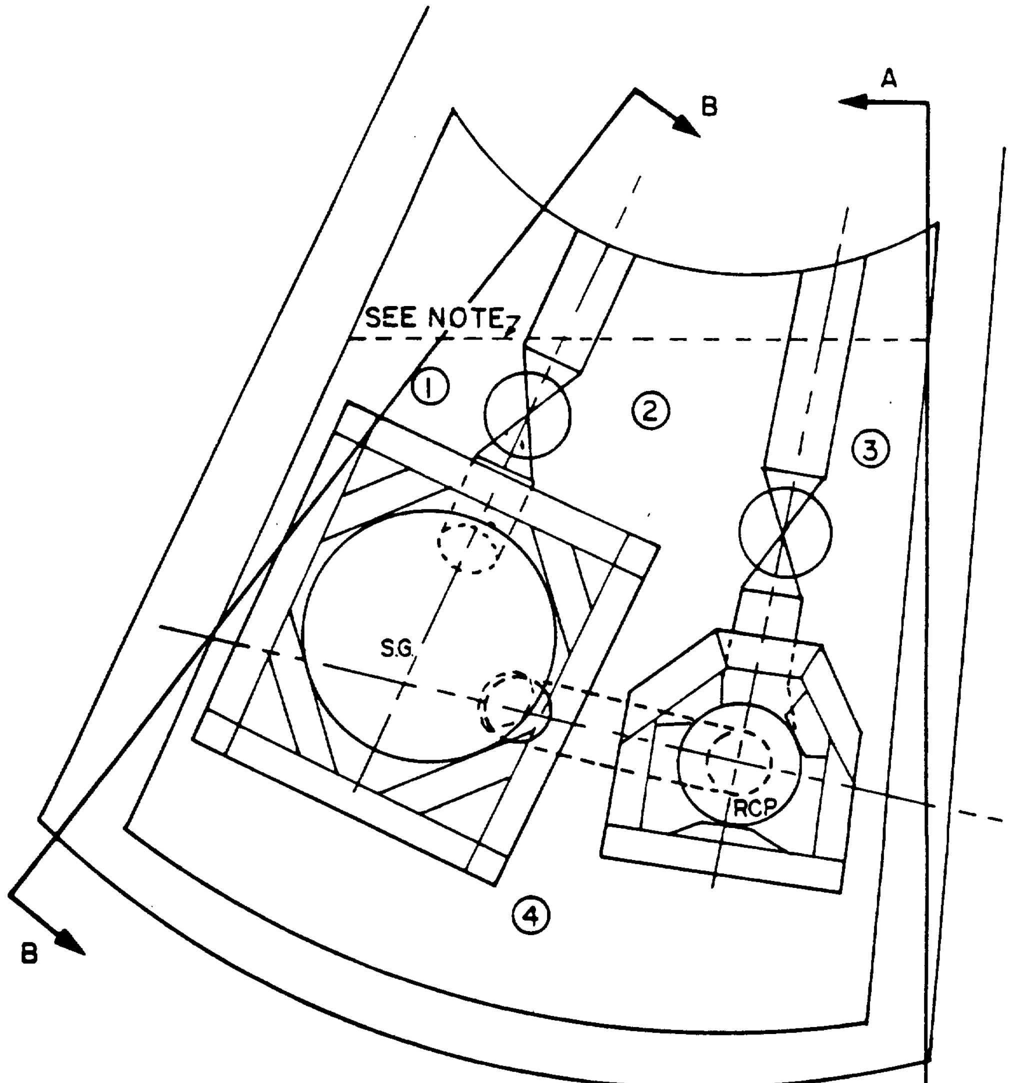


Figure 6.2-30
10-NODE (13 NODES TOTAL) STEAM GENERATOR SUBCOMPARTMENT MODEL PLAN VIEW—ELEVATION 243'-0"

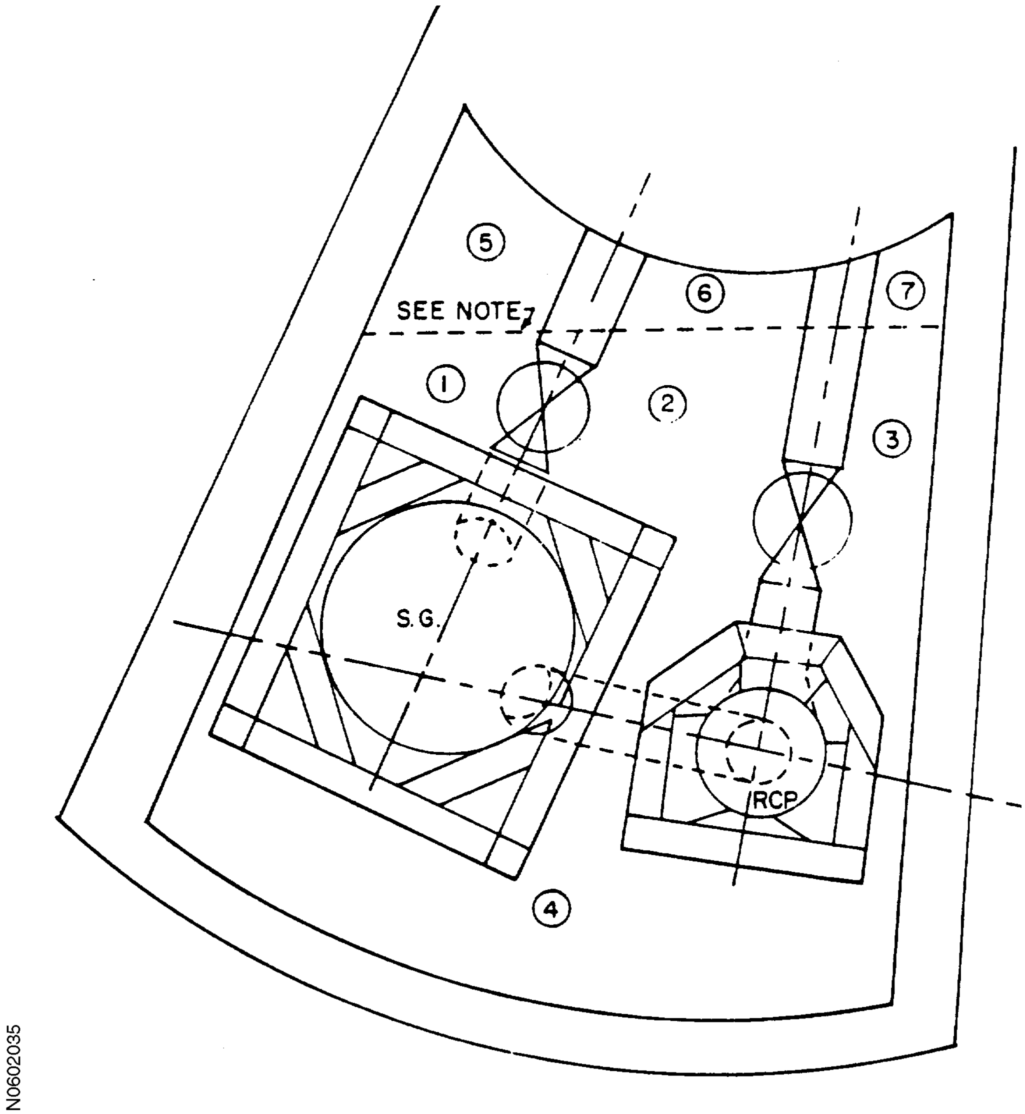


NOTES:

1. DRAWING ELEVATION-258 FT. LINE REPRESENTS THE WALL AT ELEVATION 259 FT.
2. THE ABREVIATIONS S.G. AND R.C.P. REFER TO THE STEAM GENERATOR AND THE REACTOR COOLANT PUMP RESPECTIVELY
3. THE DESIGNATION (1) REFERS TO NODE 1 ETC.

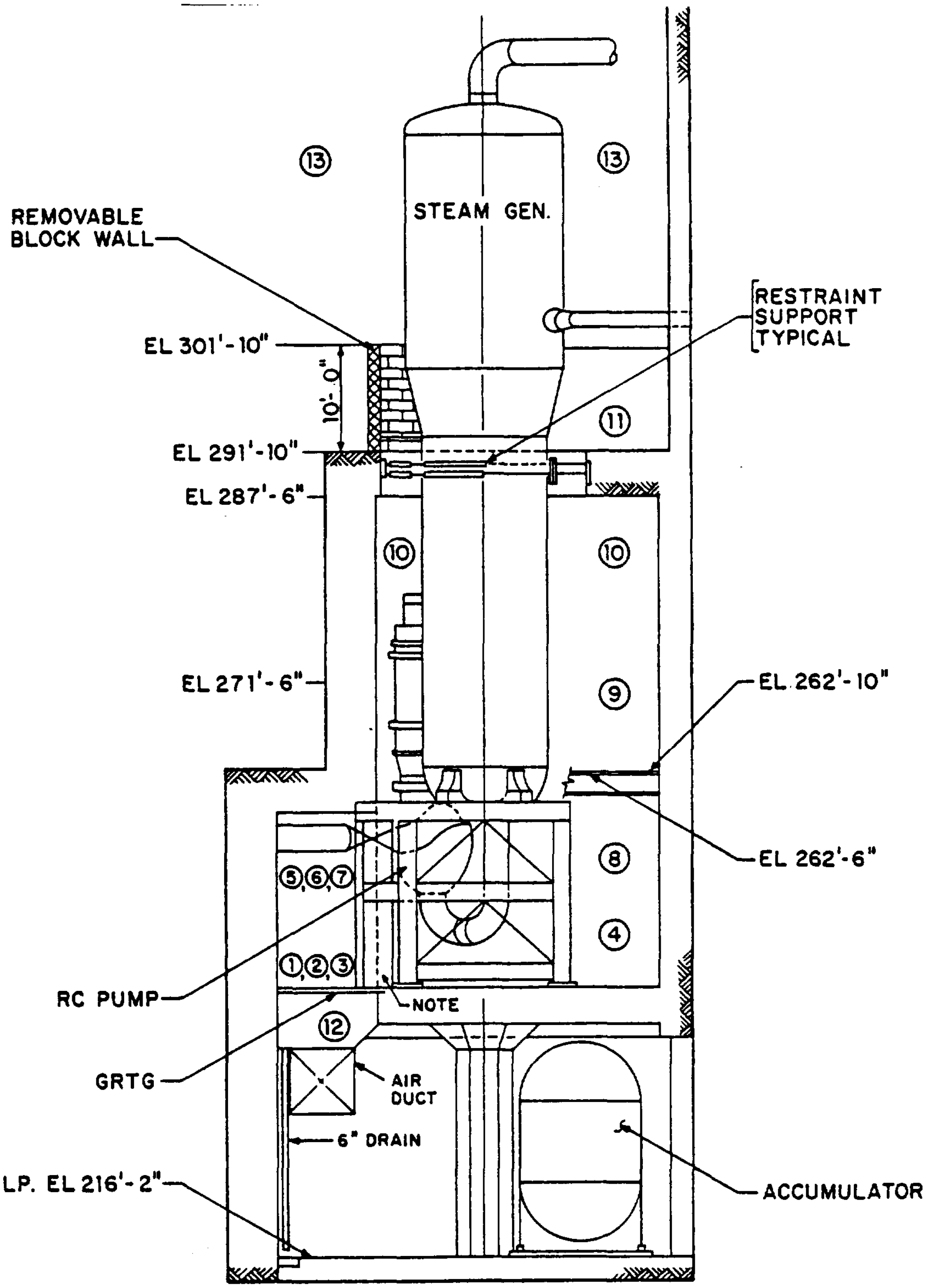
N0602034

Figure 6.2-31
STEAM GENERATOR SUBCOMPARTMENT MODEL
PLAN VIEW—ELEVATION 259'-0"



NOTES:
1. DRAWING ELEVATION - 258 FT LINE REPRESENTS THE WALL AT ELEVATION 254 FT.
2. THE ABBREVIATIONS S.G. AND R.C.P. REFER TO THE STEAM GENERATOR AND THE REACTOR COOLANT PUMP RESPECTIVELY
3. DESIGNATION (1) REFERS TO NODE 1. ETC.

Figure 6.2-32
SECTION B-B OF FIGURES 6.2-30 AND 6.2-37



NOTE:
THIS NOTE APPLIES ONLY TO THE HORIZONTAL NODALIZATION STUDY DESCRIBED IN PART B AND REFERS TO PROJECTING A VERTICAL PLANE FROM THE OUTER SURFACE

N0602036

Figure 6.2-33
HORIZONTAL NODALIZATION STUDY
OF THE MAIN STEAM GENERATOR SUBCOMPARTMENT

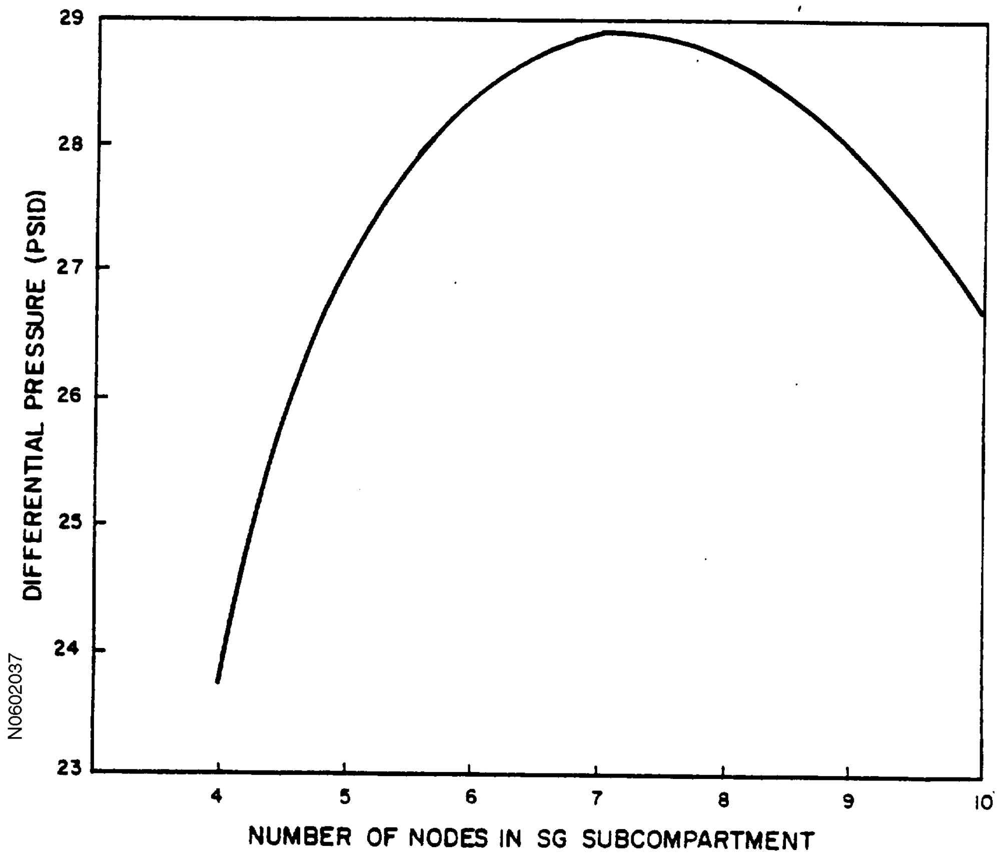


Figure 6.2-34
7-NODE MODEL DIFFERENTIAL PRESSURE BETWEEN THE STEAM GENERATOR SUBCOMPARTMENT (ELEVATION 243'- 0" TO ELEVATION 287'-6") AND THE CONTAINMENT VERSUS TIME AFTER ACCIDENT

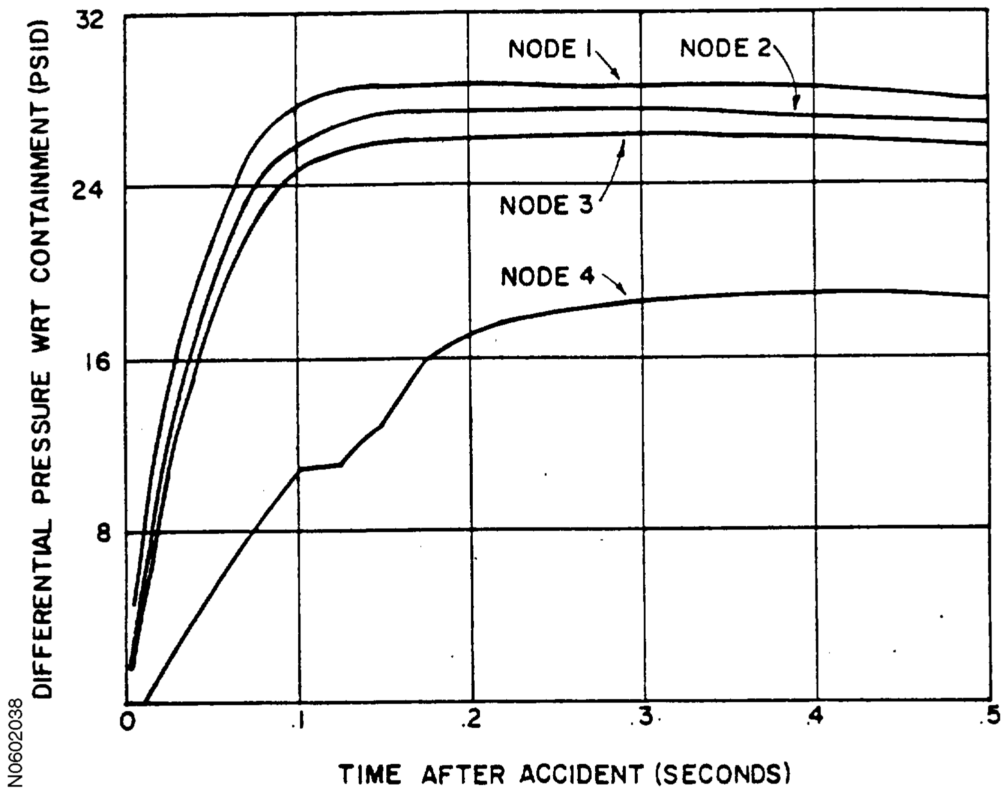
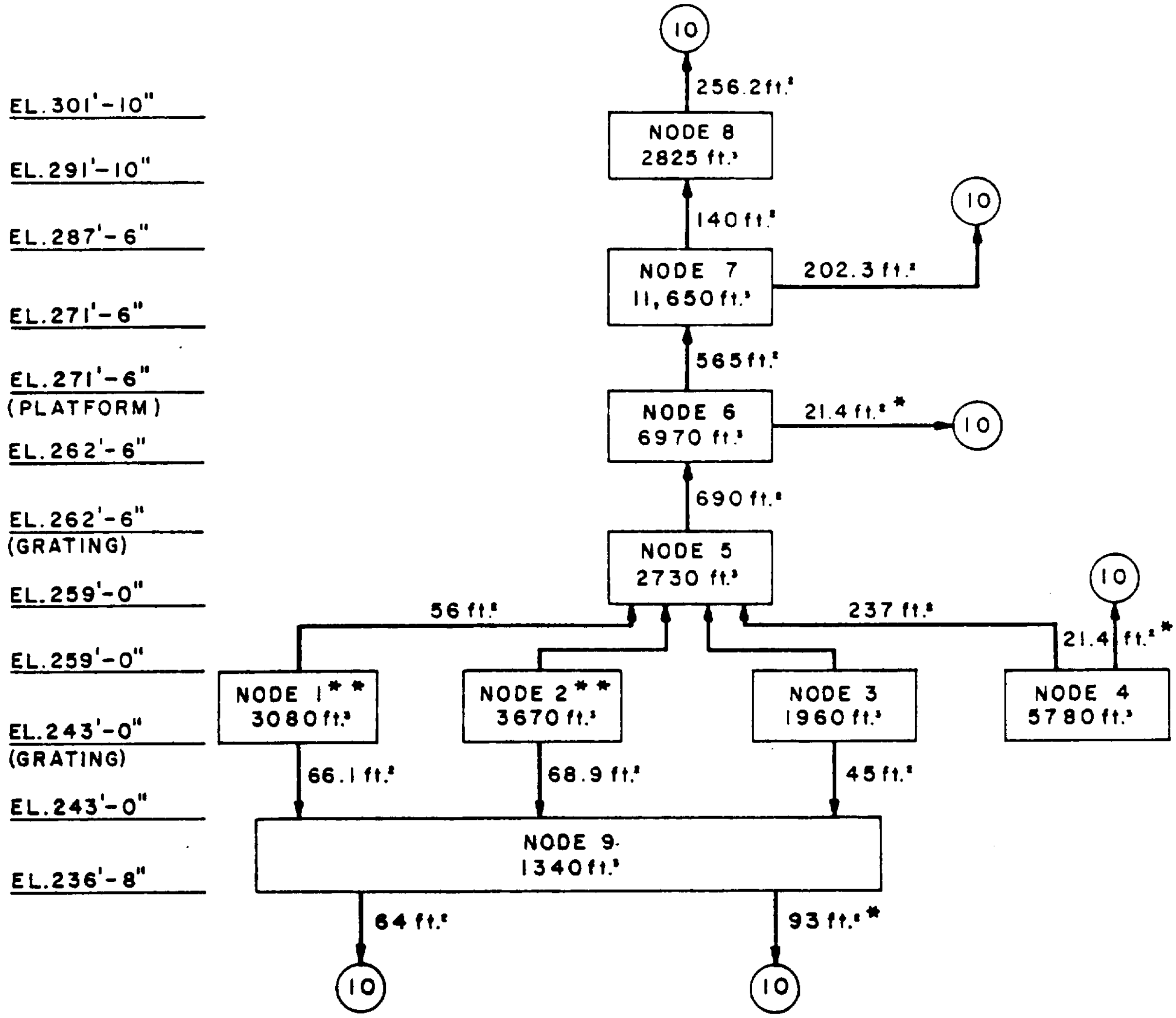


Figure 6.2-35
7-NODE MODEL (10 NODES TOTAL)
NODAL ARRANGEMENT FOR THE STEAM GENERATOR SUBCOMPARTMENT



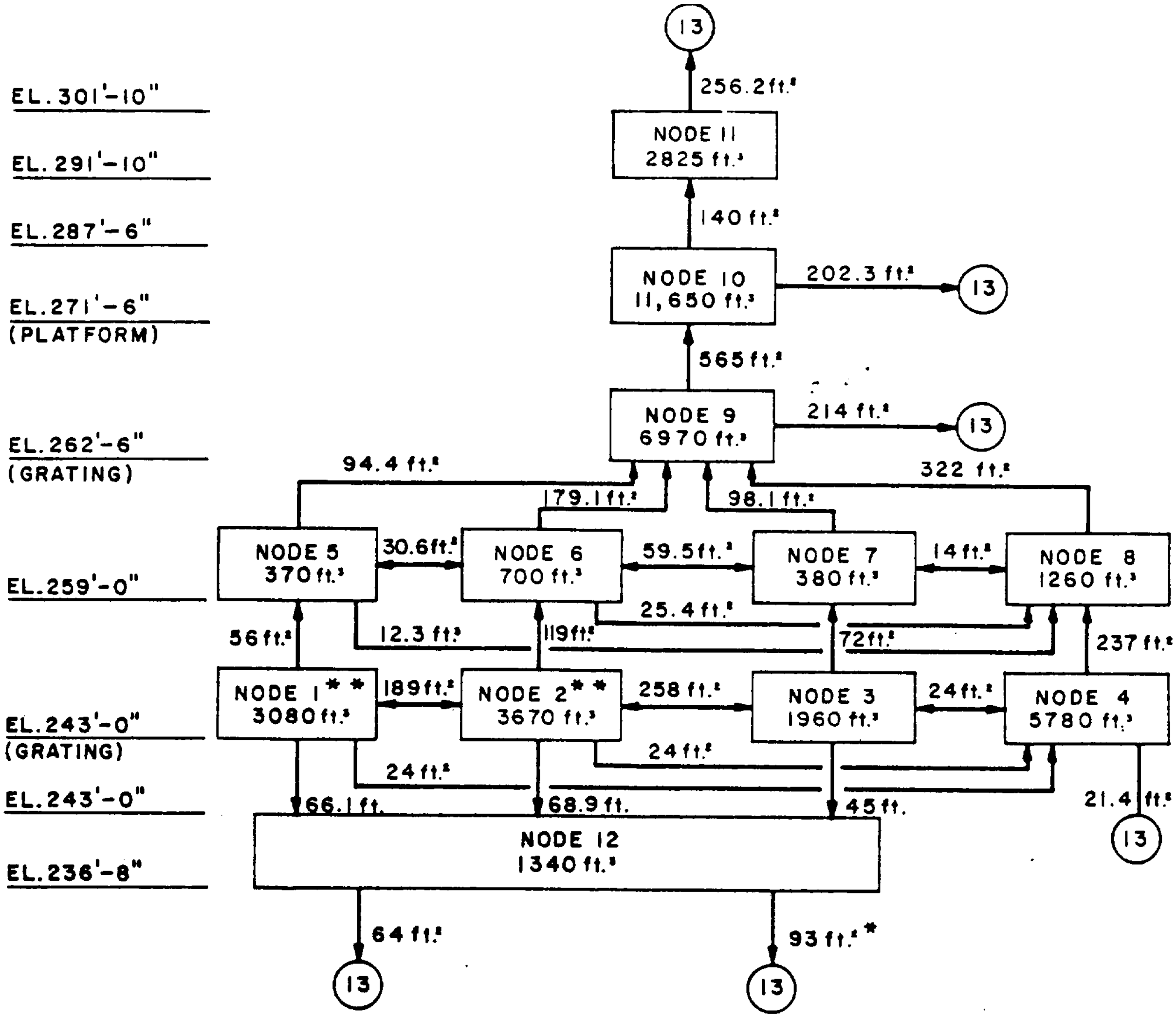
* NO FLOW IS CONSIDERED BEFORE THE PANEL BLOWS OUT AT 5 PSID

** NODE RECEIVES 50 % OF THE BLOWDOWN

NODE No.	DESCRIPTION
1, 2, 3, 4	STEAM GENERATOR SUBCOMPARTMENT FROM EL. 243'-0" TO EL. 259'-0"
5	STEAM GENERATOR SUBCOMPARTMENT FROM EL. 259'-0" TO EL. 262'-6"
6	STEAM GENERATOR SUBCOMPARTMENT FROM EL. 262'-6" TO EL. 271'-6"
7	STEAM GENERATOR SUBCOMPARTMENT FROM EL. 271'-6" TO EL. 287'-6"
8	VOLUME SURROUNDED BY REMOVABLE BLOCK WALLS FROM EL. 291'-10" TO EL. 301'-10"
9	VOLUME ABOVE THE AIR DUCT (EL. 236'-8") AND BELOW THE GRATE (EL. 243'-0")
10	BULK CONTAINMENT

N0602039

Figure 6.2-36
10-NODE MODEL (13 NODES TOTAL)
NODAL ARRANGEMENT FOR THE STEAM GENERATOR SUBCOMPARTMENT



* NO FLOW IS CONSIDERED BEFORE THIS PANEL BLOWS OUT AT 5 PSID

** NODE RECEIVES 50% OF THE BLOWDOWN

NODE No.	DESCRIPTION
1, 2, 3, 4	STEAM GENERATOR SUBCOMPARTMENT FROM EL. 243'-0" TO EL. 259'-0"
5, 6, 7, 8	STEAM GENERATOR SUBCOMPARTMENT FROM EL. 259'-0" TO EL. 262'-6"
9	STEAM GENERATOR SUBCOMPARTMENT FROM EL. 262'-6" TO EL. 271'-6"
10	STEAM GENERATOR SUBCOMPARTMENT FROM EL. 271'-6" TO EL. 287'-6"
11	VOLUME SURROUNDED BY THE BLOCK WALLS FROM EL. 291'-10" TO EL. 301'-10"
12	VOLUME ABOVE THE AIR DUCT (EL. 236'-8") AND BELOW THE GRATE (EL. 243'-0")
13	BULK CONTAINMENT

N0602040

Figure 6.2-37
10-NODE MODEL (13 NODES TOTAL) STEAM GENERATOR SUBCOMPARTMENT MODEL PLAN VIEW—ELEVATION 259'-0"

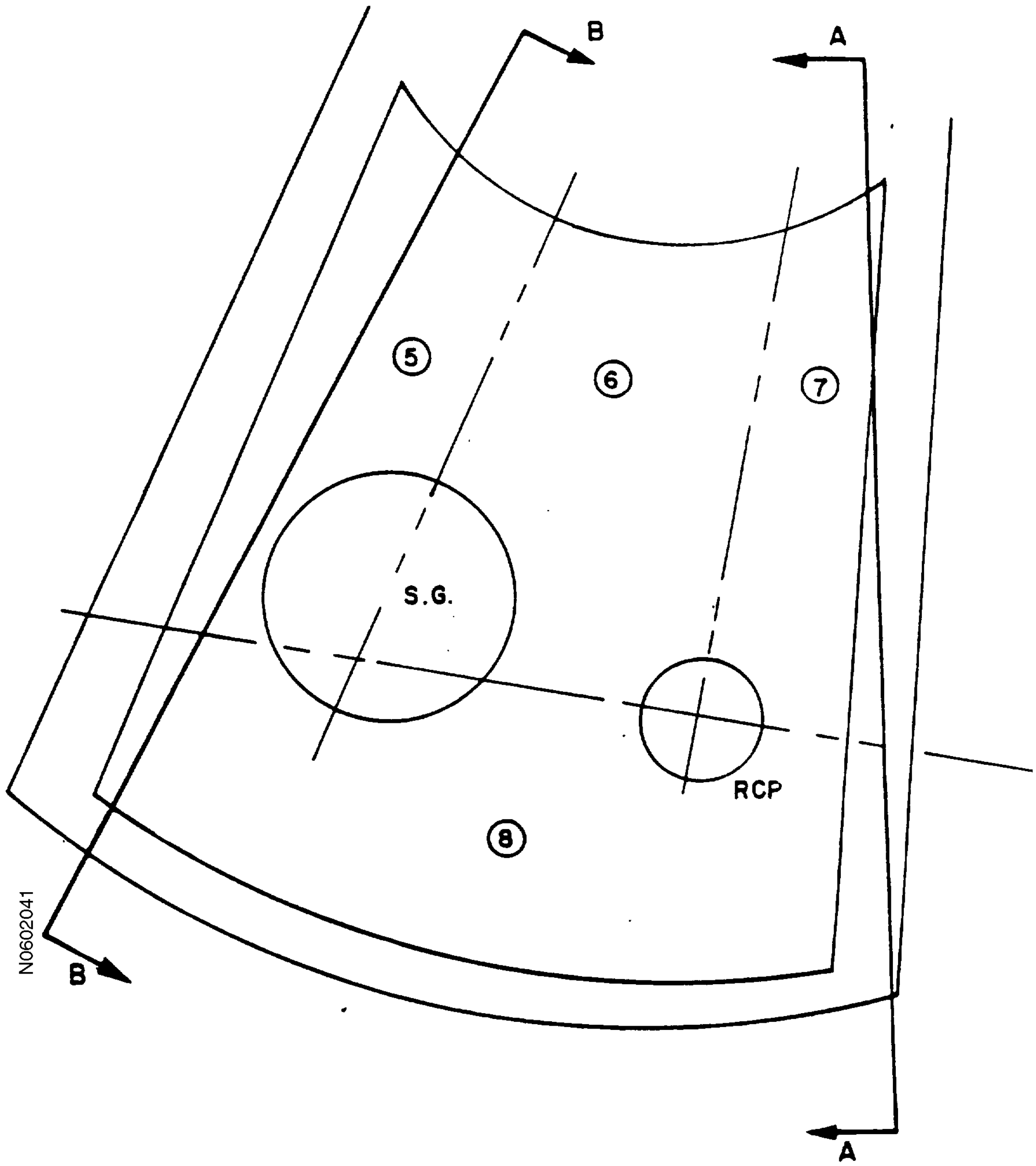


Figure 6.2-38
STEAM GENERATOR AND REACTOR COOLANT PUMP

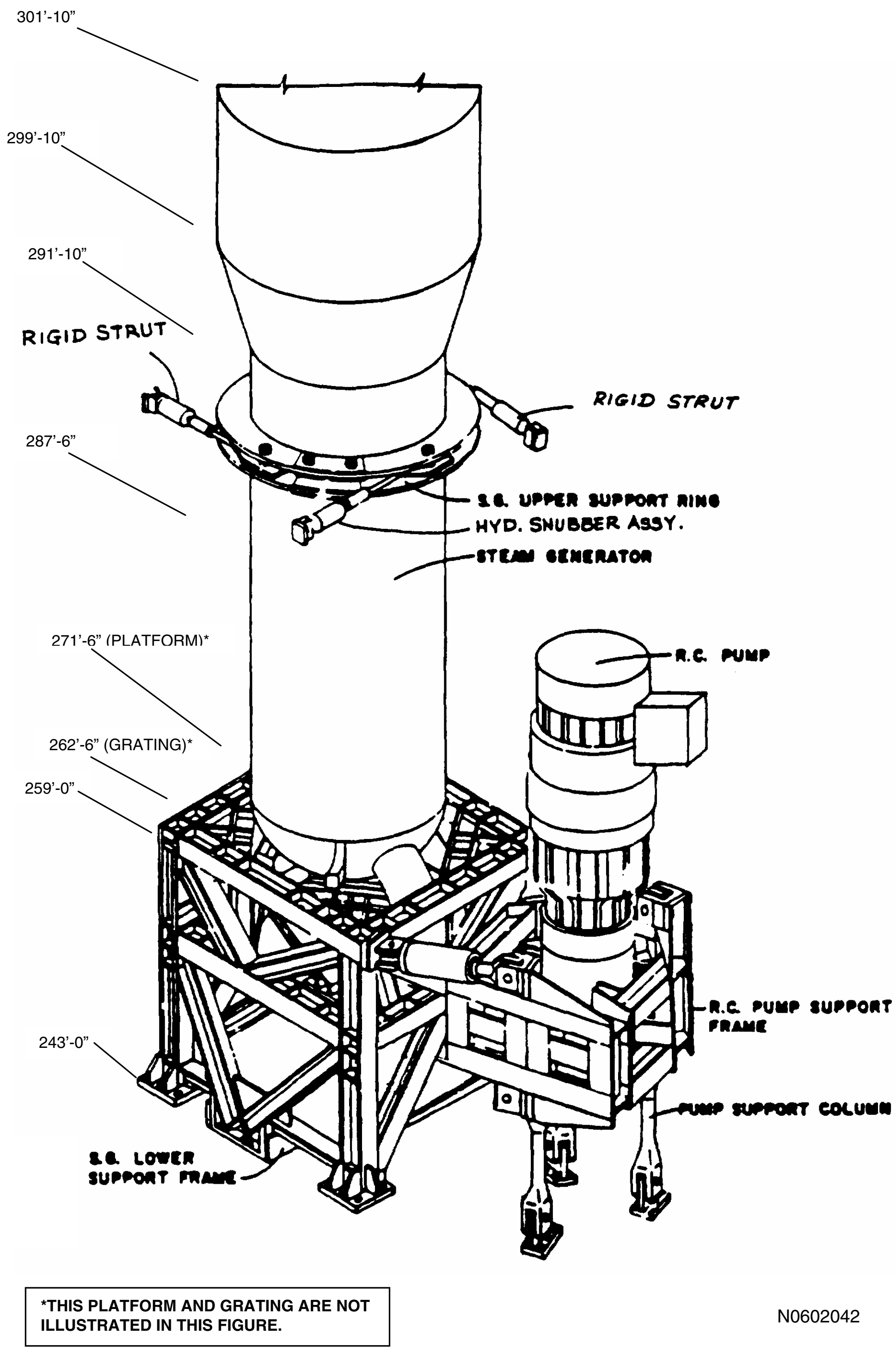


Figure 6.2-39
SECTION A-A OF FIGURES 6.2-30 AND 6.2-37

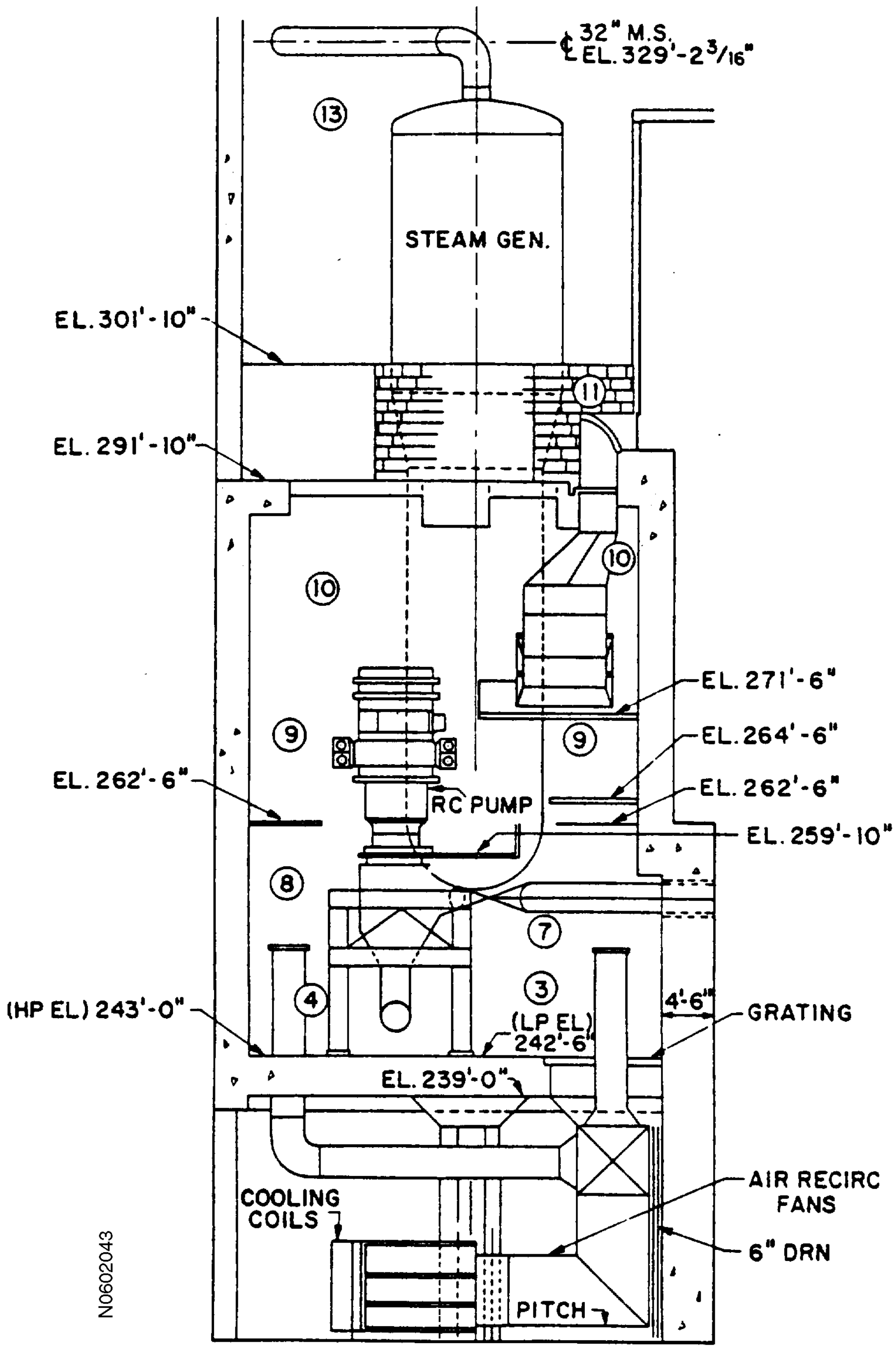


Figure 6.2-40
DIFFERENTIAL PRESSURE ACROSS THE STEAM GENERATOR SUPPORTS AND THE REACTOR COOLANT PUMP SUPPORTS

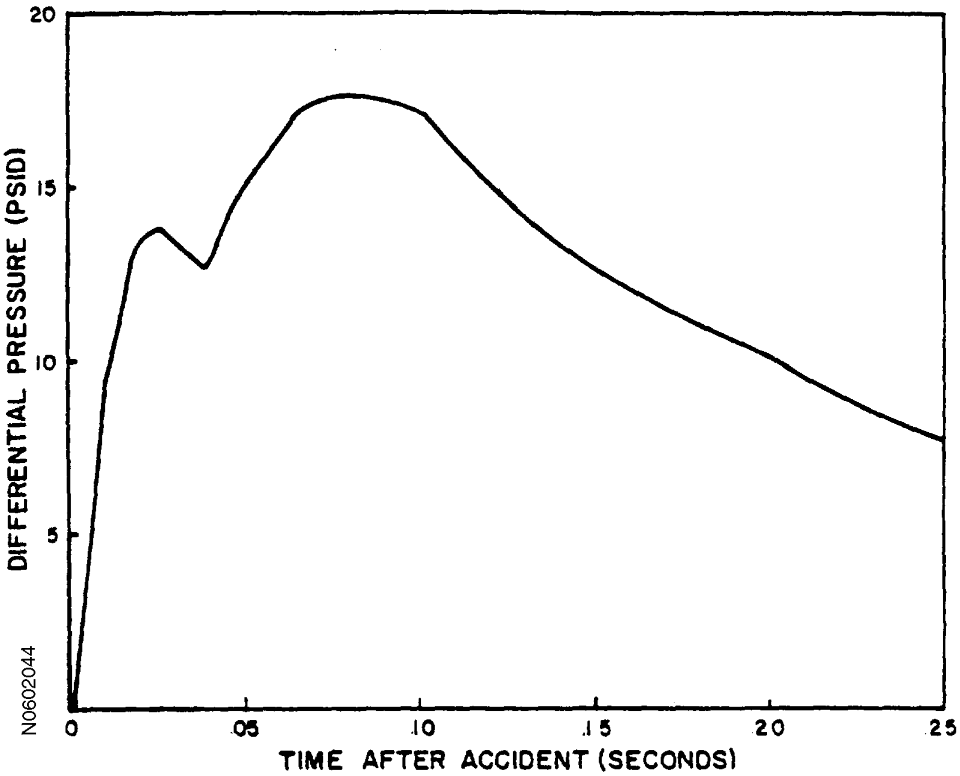


Figure 6.2-41
DIFFERENTIAL PRESSURE ACROSS THE STEAM GENERATOR
AND THE REACTOR COOLANT PUMP

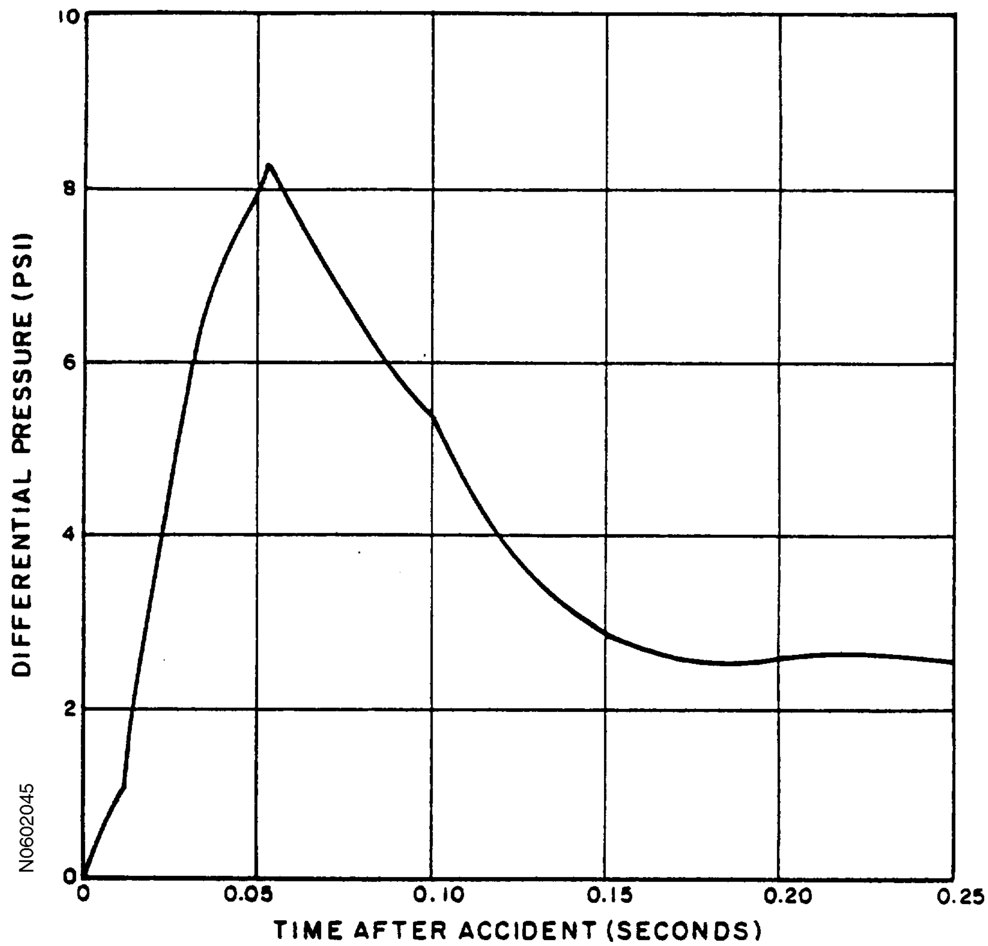
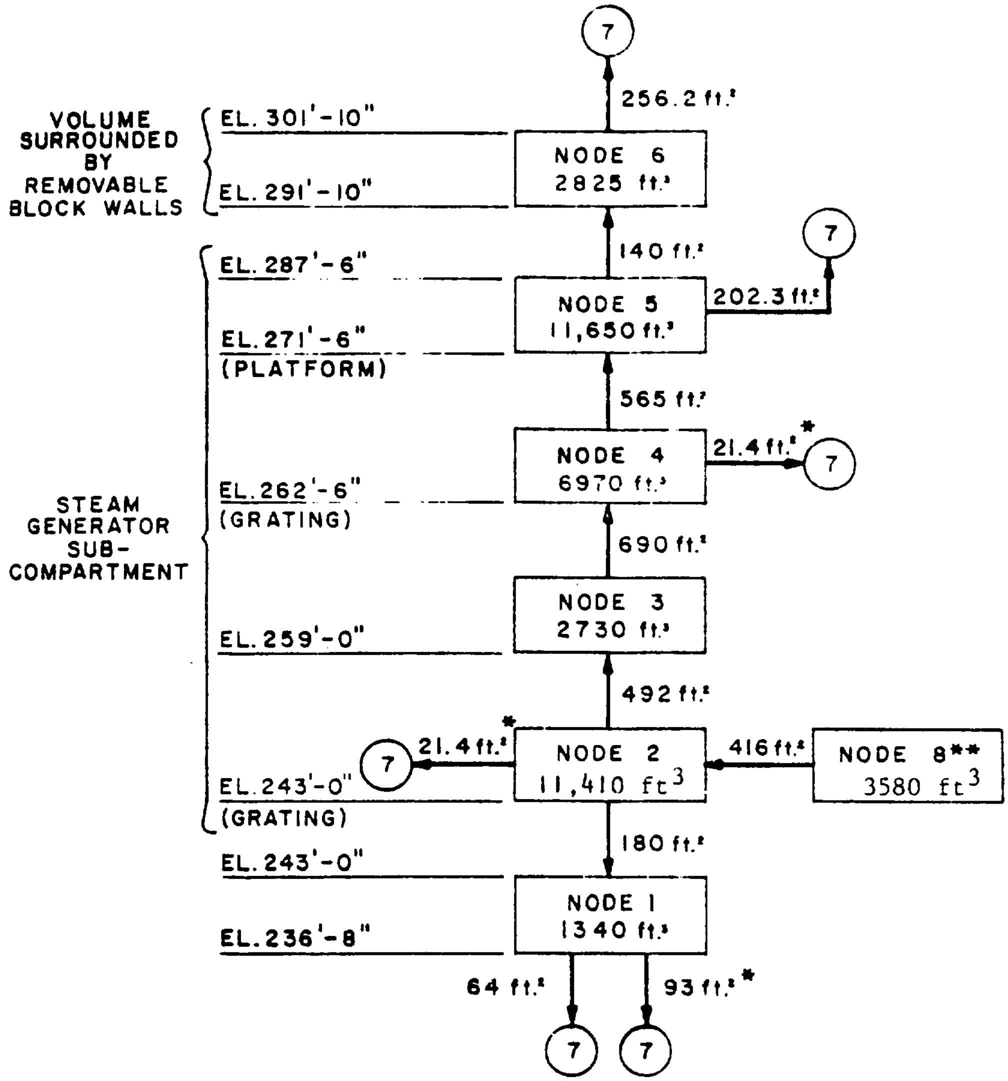


Figure 6.2-42
5-NODE MODEL (8 NODES TOTAL) PARAMETERS FOR THE ANALYSIS OF VERTICAL PRESSURIZATION OF THE STEAM GENERATOR (UPLIFT)



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* NO FLOW IS CONSIDERED BEFORE THE PANEL BLOWS OUT AT 5 PSID
** NODE RECEIVES 100% OF THE BLOWDOWN

NODE No.	DESCRIPTION
1	VOLUME ABOVE THE AIR DUCT (EL. 236'-8") AND BELOW THE GRATE (EL. 243'-0")
2	STEAM GENERATOR SUBCOMPARTMENT FROM EL 243'-0" TO EL. 259'-0"
3	STEAM GENERATOR SUBCOMPARTMENT FROM EL. 259'-0" TO EL. 262'-6"
4	STEAM GENERATOR SUBCOMPARTMENT FROM EL. 262'-6" TO EL. 271'-6"
5	STEAM GENERATOR SUBCOMPARTMENT FROM EL. 271'-6" TO EL. 287'-6"
6	VOLUME SURROUNDED BY THE REMOVABLE BLOCK WALLS FROM EL. 291'-10" TO EL. 301'-10"
7	BULK CONTAINMENT VOLUME SURROUNDED BY THE STEAM GENERATOR
8	LOWER SUPPORT FRAME FROM EL. 243'-0" TO EL. 259'-0"

N0602046

Figure 6.2-43
VERTICAL PRESSURIZATION OF THE STEAM GENERATOR (UPLIFT)

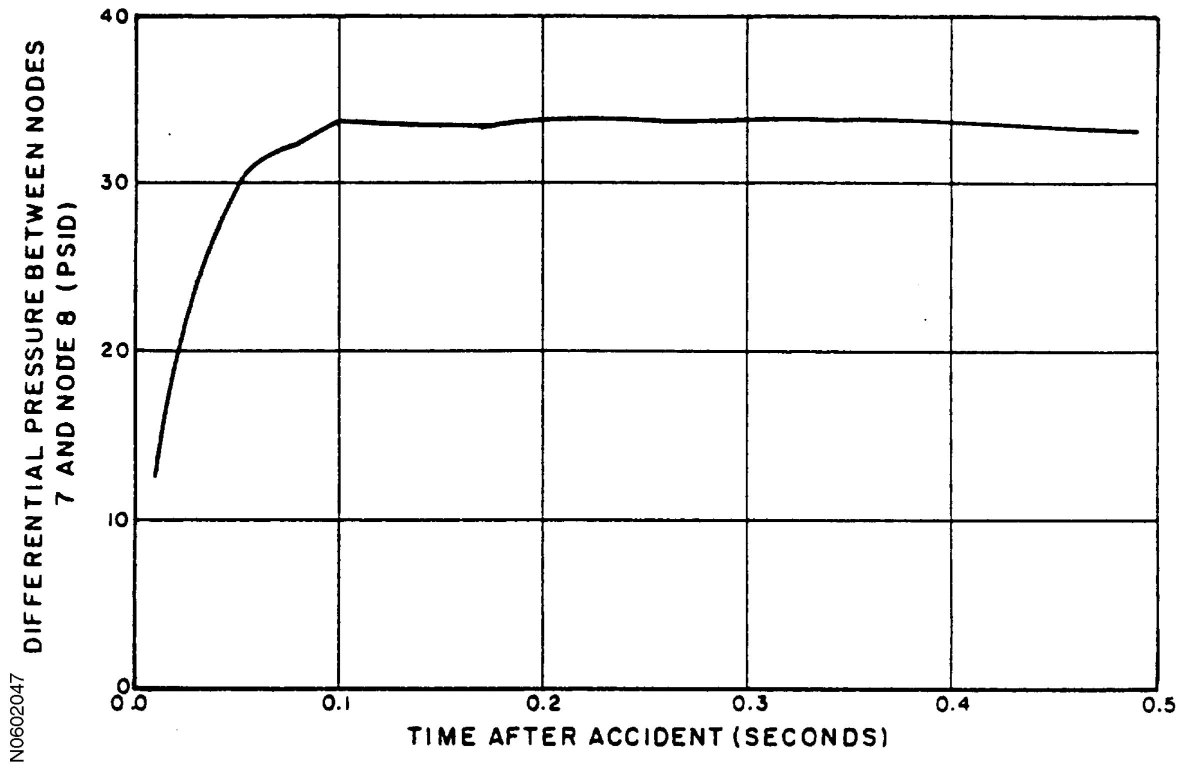
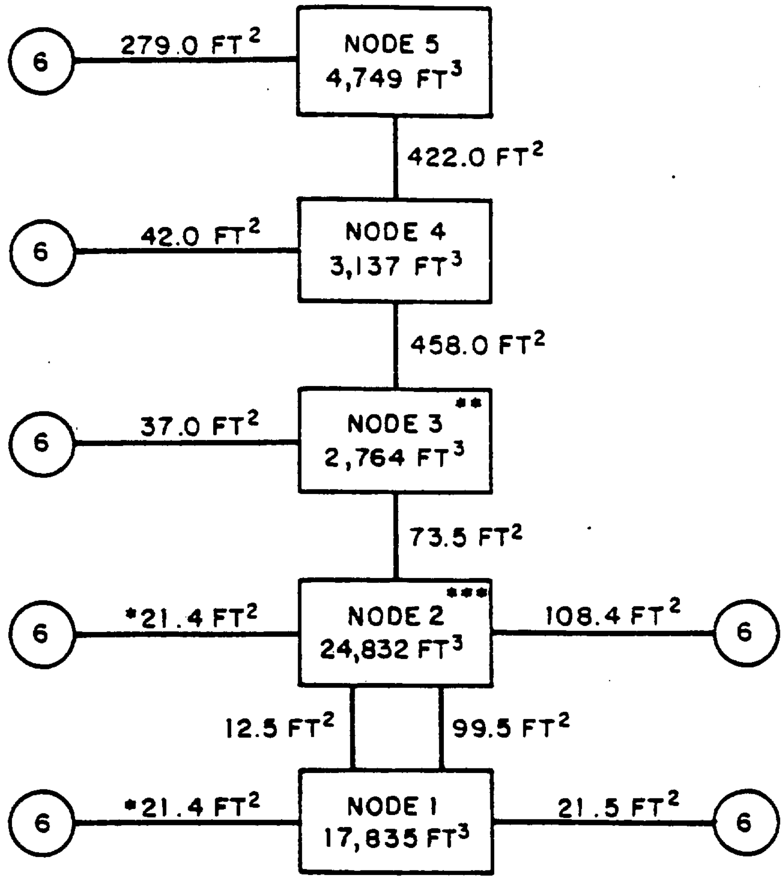


Figure 6.2-44
PRESSURIZER NODAL ARRANGEMENT



NODE NO.	NODE DESCRIPTION
1	PRESSURIZER RELIEF TANK SUBCOMPARTMENT
2	LOWER PRESSURIZER SUBCOMPARTMENT UP TO ELEVATION 291'-10"
3	UPPER PRESSURIZER SUBCOMPARTMENT UP TO GRATING AT ELEVATION 298'-0"
4	UPPER PRESSURIZER SUBCOMPARTMENT UP TO GRATING AT ELEVATION 305'-0"
5	UPPER PRESSURIZER SUBCOMPARTMENT UP TO ELEVATION 315'-2"
6	CONTAINMENT

N0602048

* BLOWOUT PANELS DESIGNED TO RELEASE AT 5.0 PSID
** SPRAY LINE DER POSTULATE IN THIS NODE.
*** SURGE LINE DER POSTULATE IN THIS NODE.

Figure 6.2-45
ABSOLUTE DIFFERENTIAL PRESSURE RESPONSE
FOR SURGE LINE DER (NODE 2 IS THE BREAK NODE)

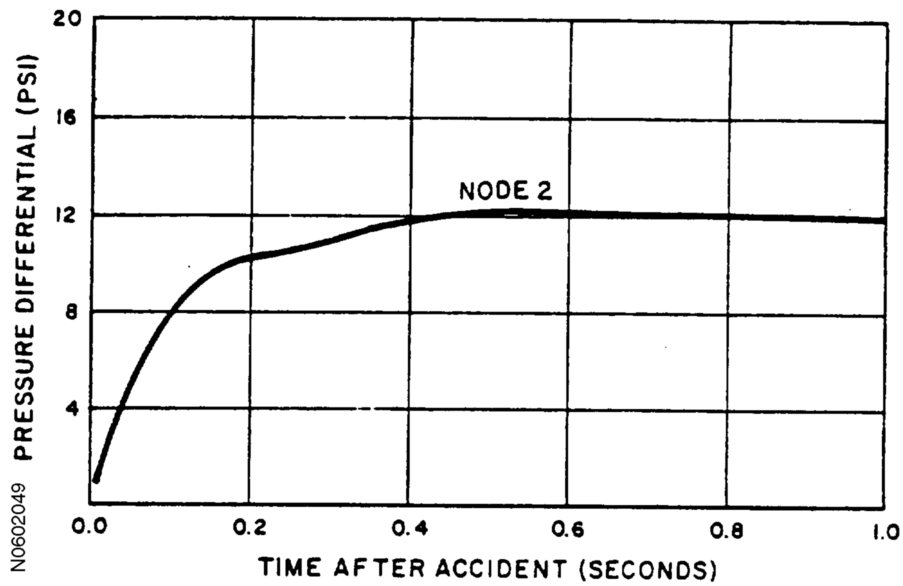
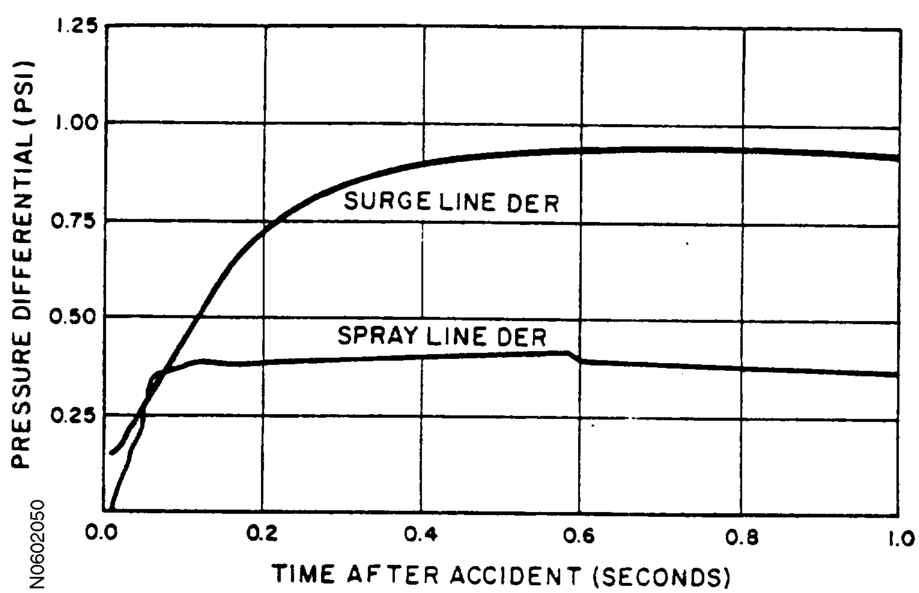


Figure 6.2-46
DIFFERENTIAL PRESSURE RESPONSE
FOR NODE 3 OF PRESSURIZER SUBCOMPARTMENT



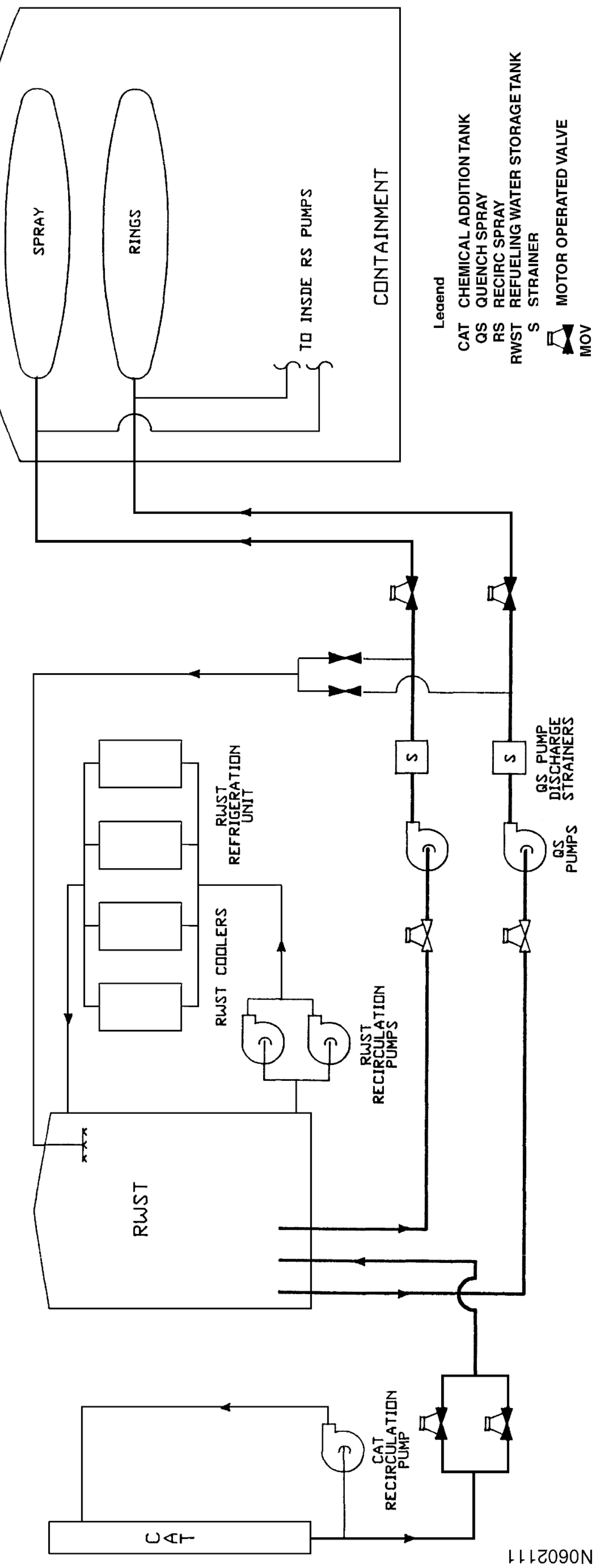


Figure 6.2-47
QUENCH SPRAY SUBSYSTEM

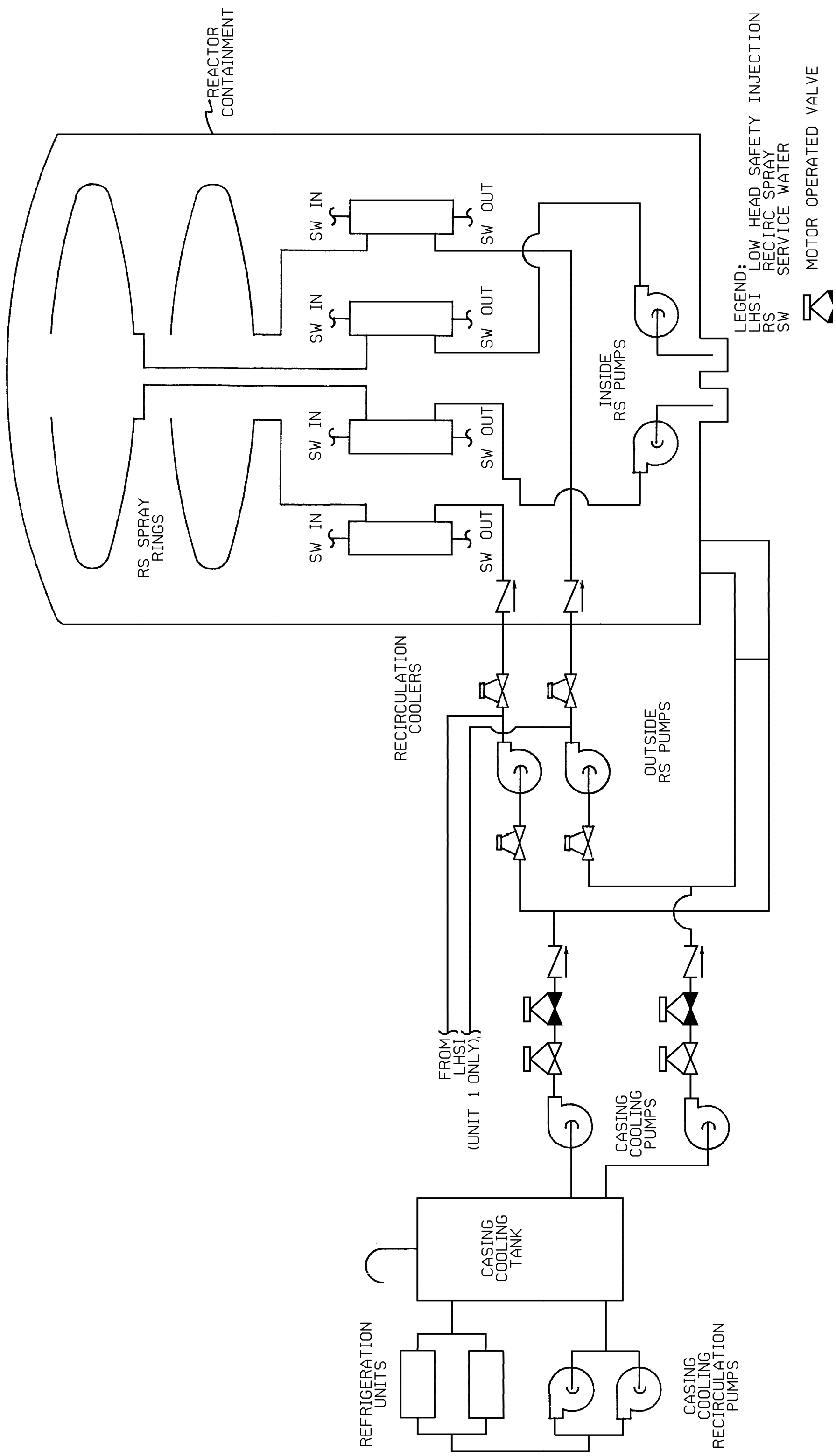


Figure 6.2-48
RECIRCULATION SPRAY SUBSYSTEM

Figure 6.2-49
RWST INTERNAL WEIR

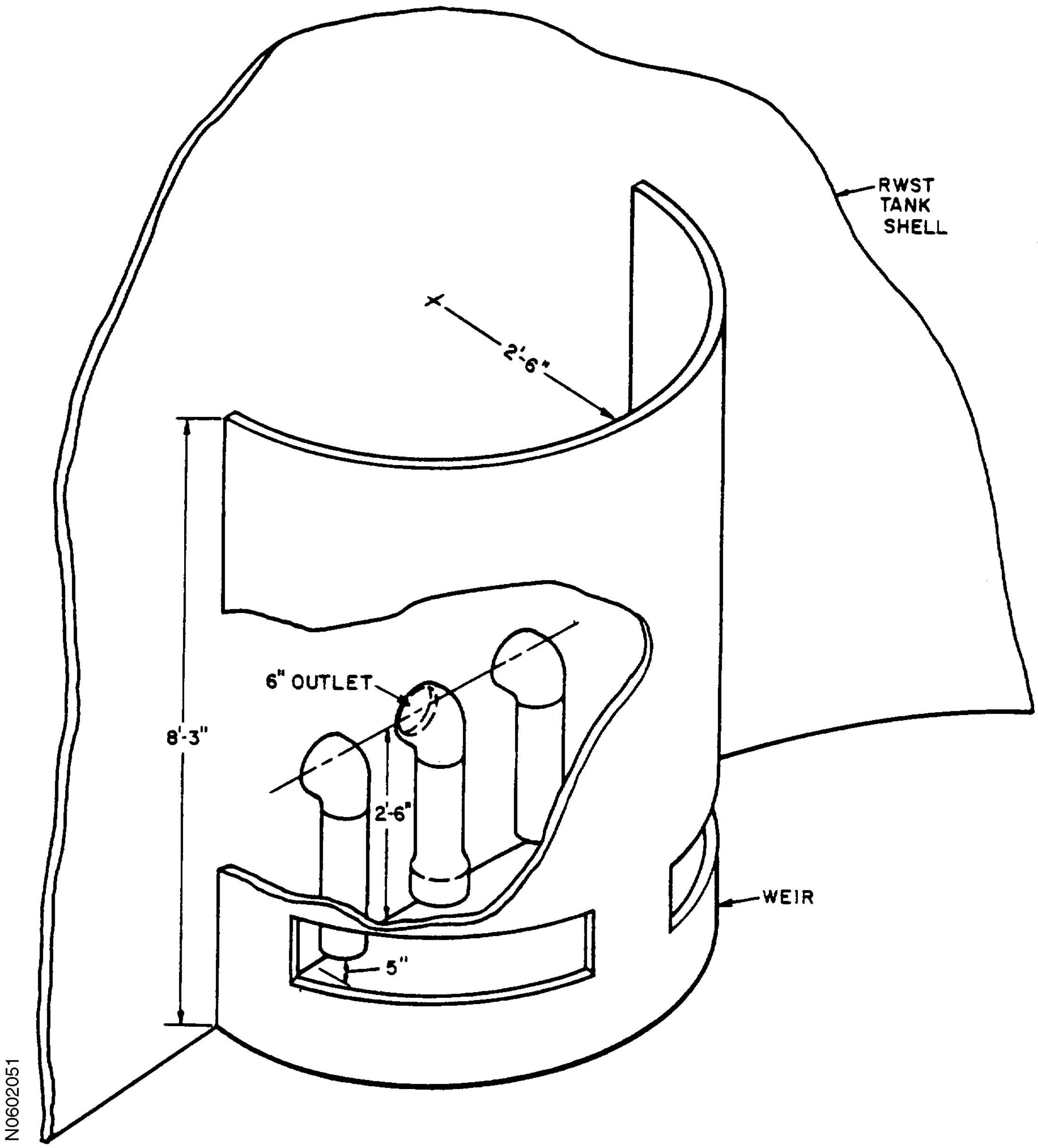
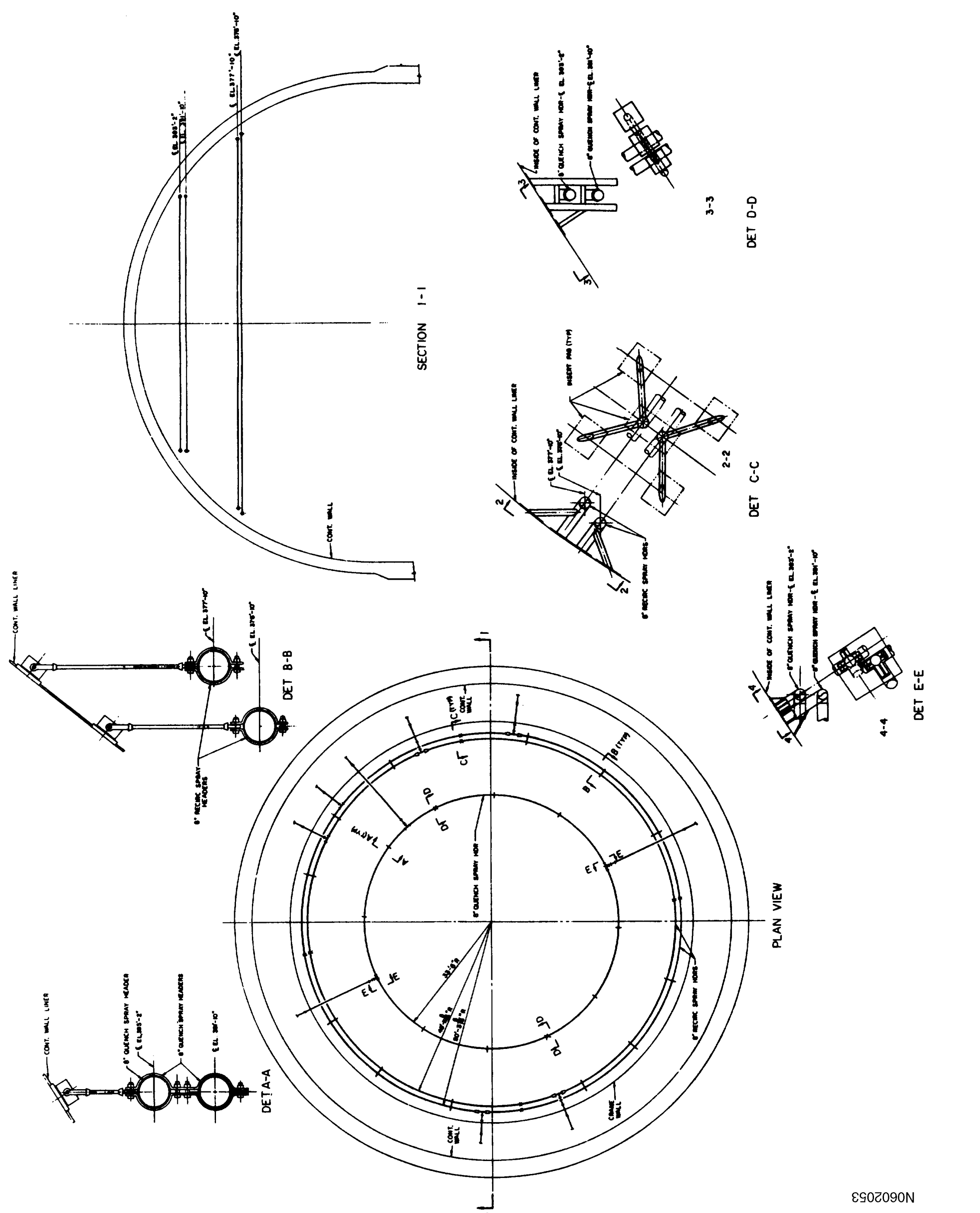


Figure 6.2-50
TYPICAL HANGER ARRANGEMENT RECIRCULATING & QUENCH SPRAY HEADERS



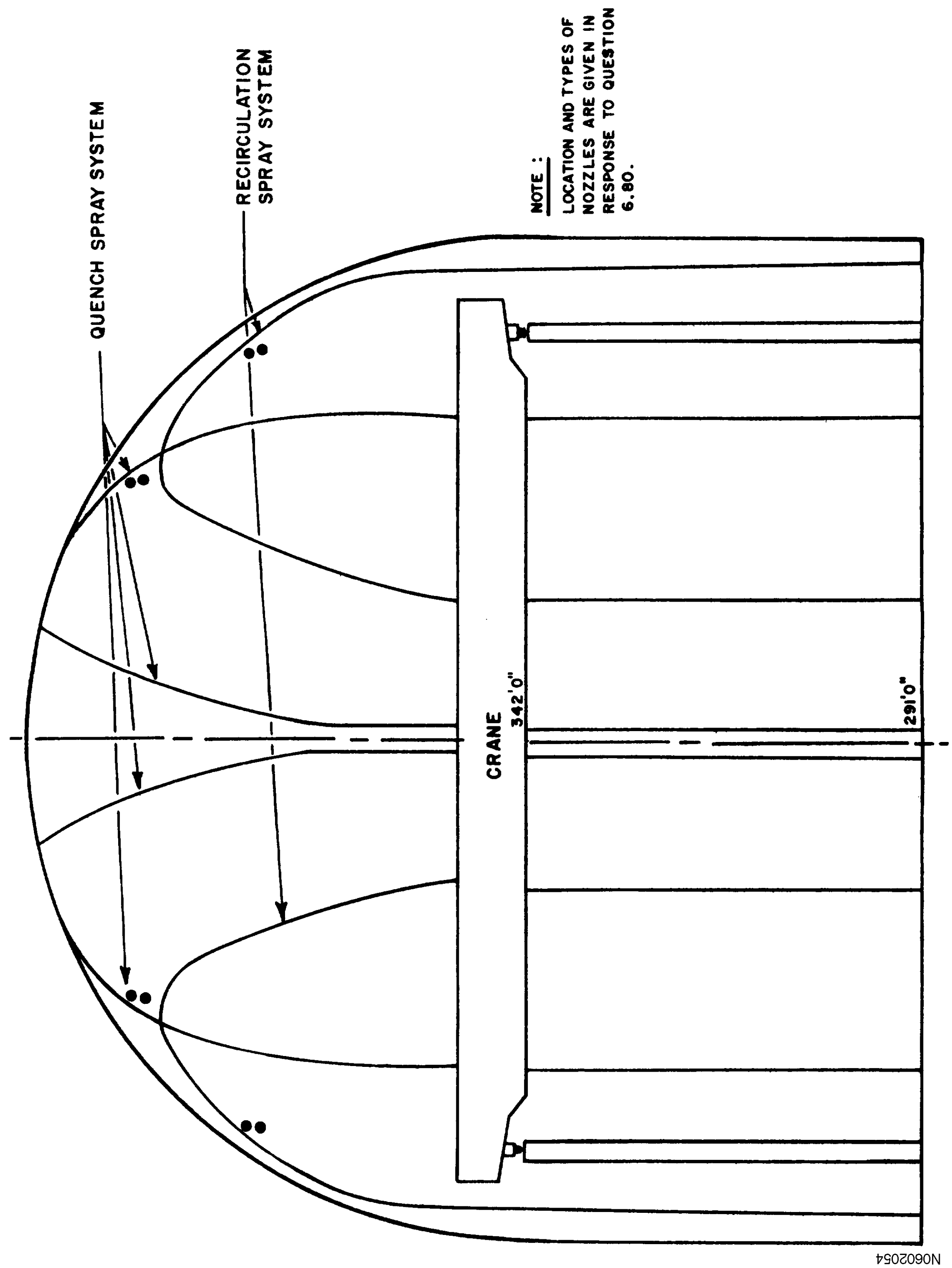


Figure 6.2-51
SPRAY PATTERNS IN CONTAINMENT

Figure 6.2-52
NON-NORMALIZED DENSITY FUNCTION f(D)
OF DROP DIAMETERS FOR QUENCH SPRAY HEADER AT 40 PSI

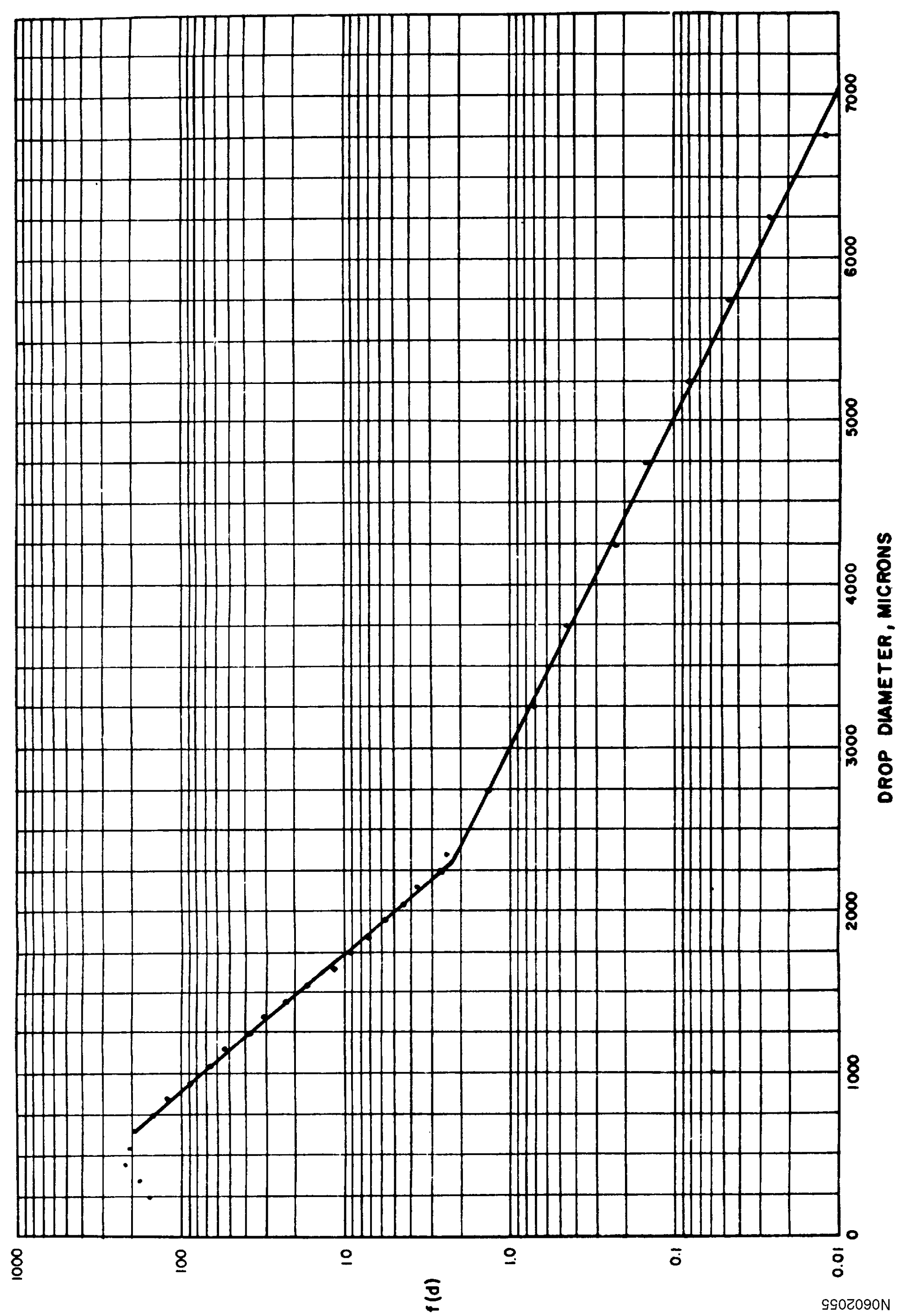


Figure 6.2-53
NON-NORMALIZED DENSITY FUNCTION f(d)
OF DROP DIAMETERS FOR RECIRCULATION SPRAY HEADER AT 25 PSI

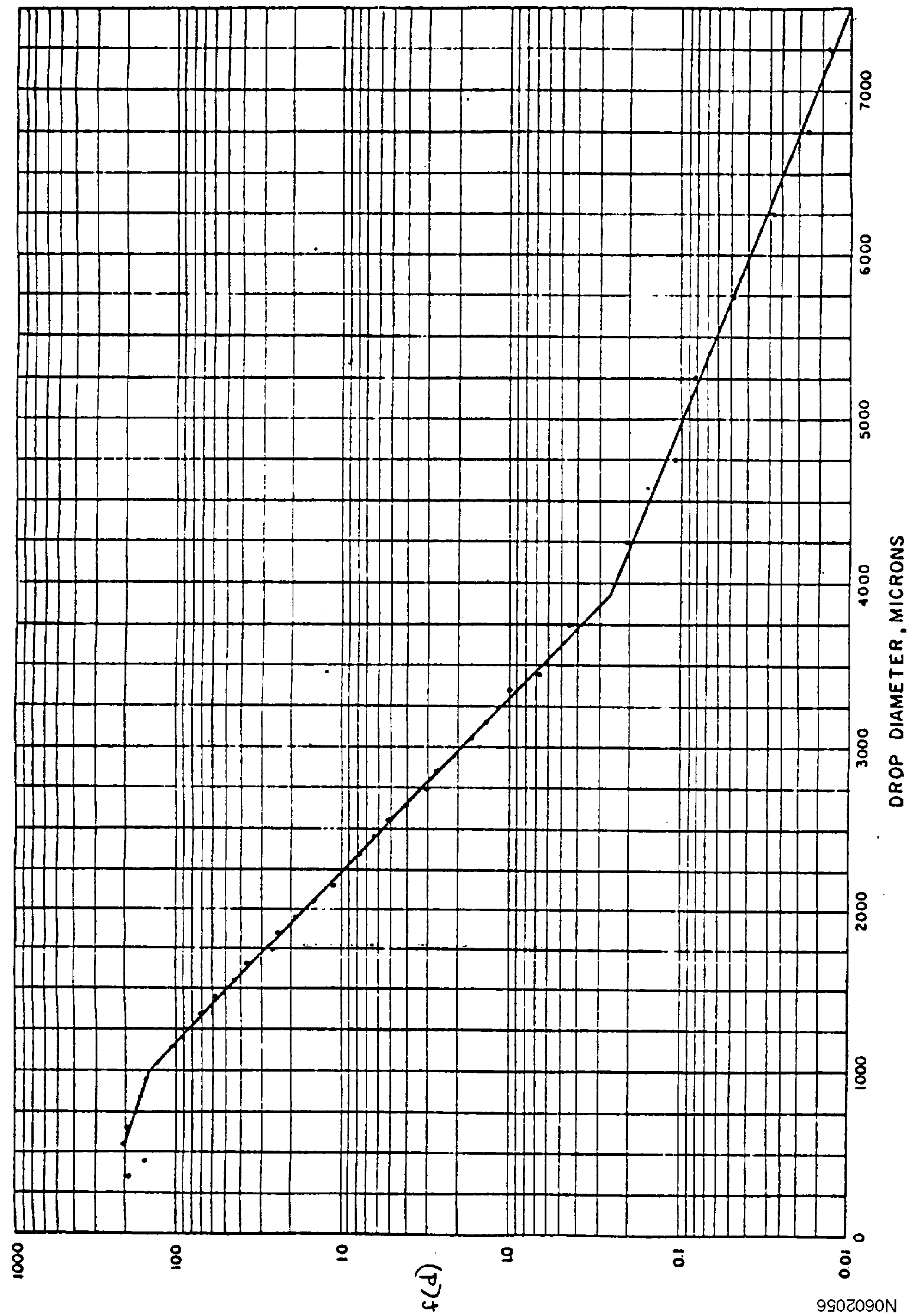


Figure 6.2-54
AIR MOTION INDUCED BY RECIRCULATION SPRAYS

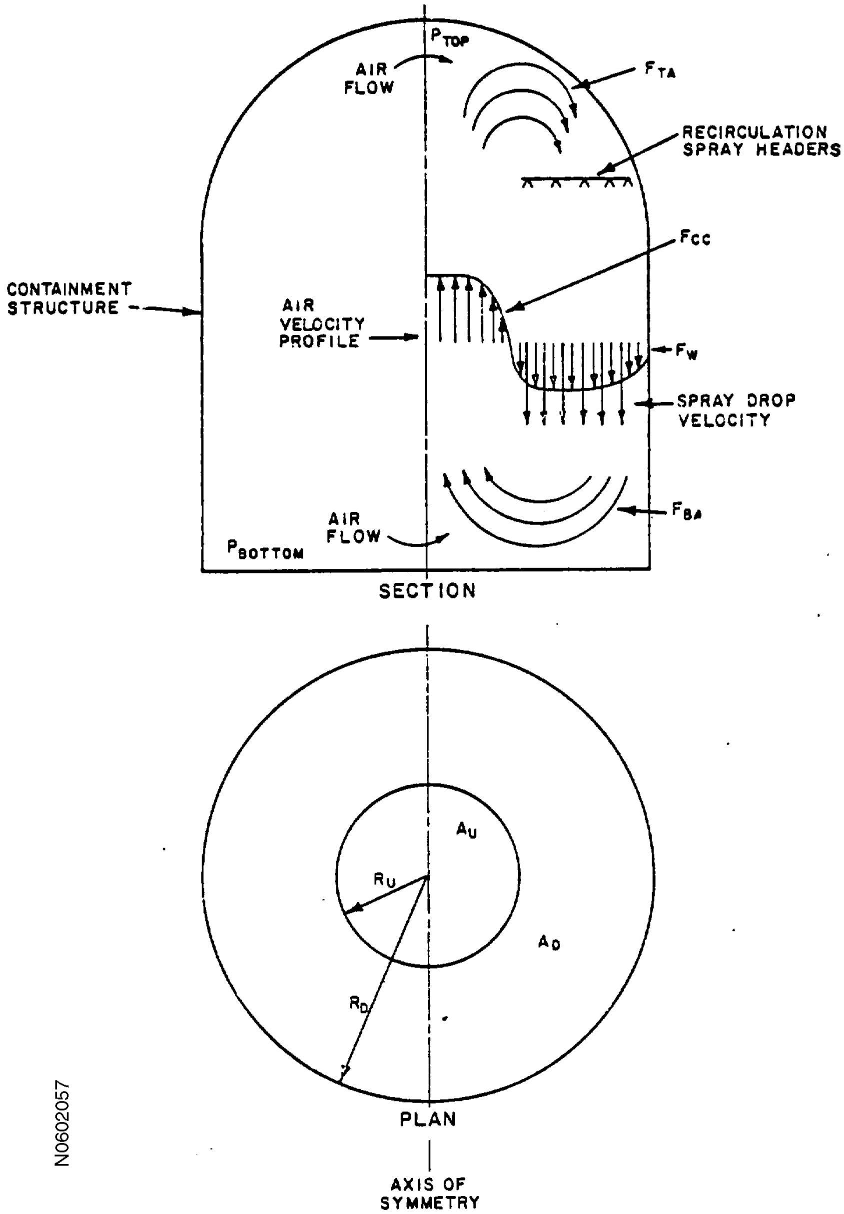
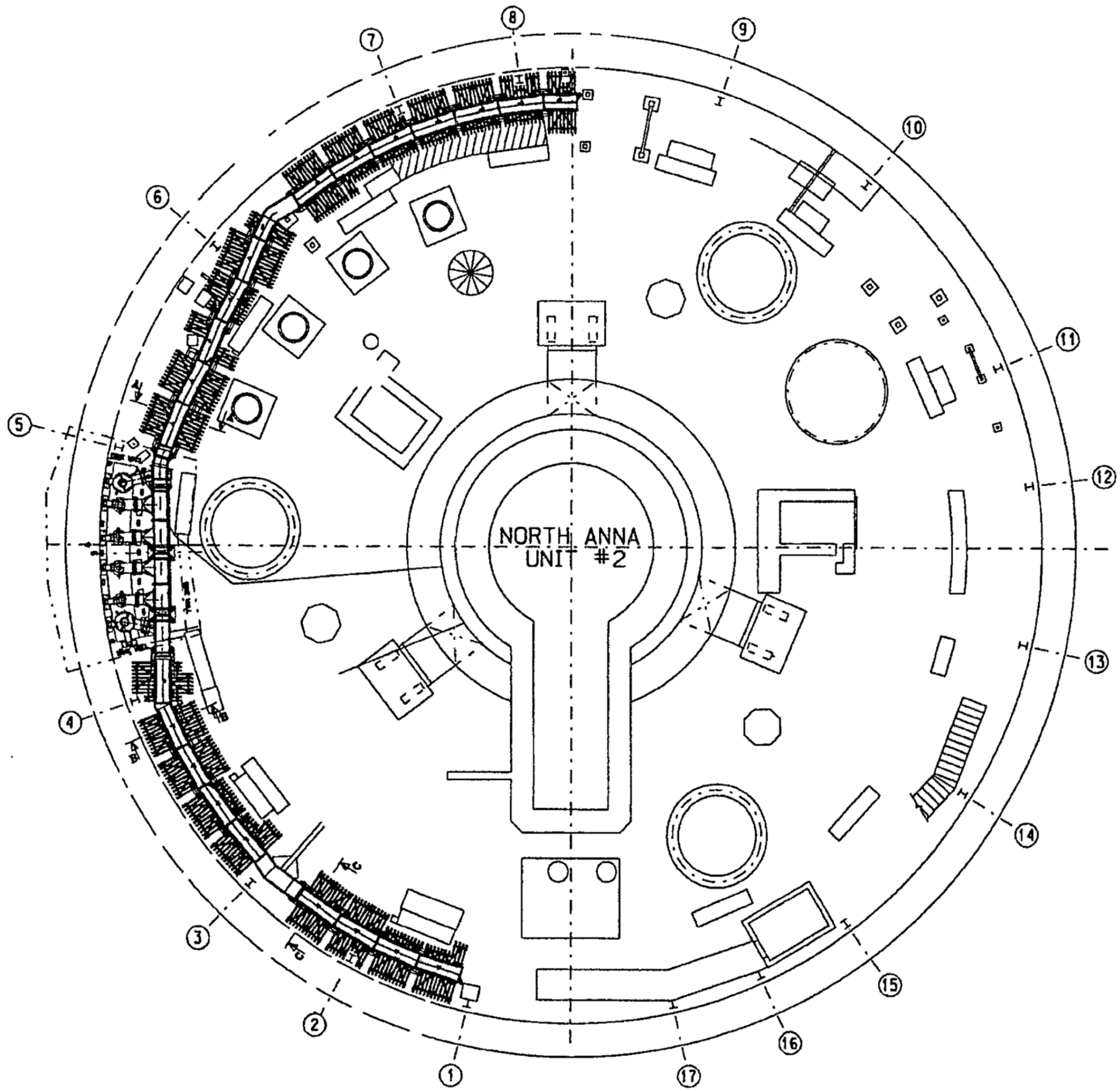


Figure 6.2-55
ARRANGEMENT CONTAINMENT SUMP STRAINER FOR UNIT 2



PLAN ELEVATION 216'-11"

Figure 6.2-56
ARRANGEMENT CONTAINMENT SUMP STRAINER HEADER FOR UNIT 2

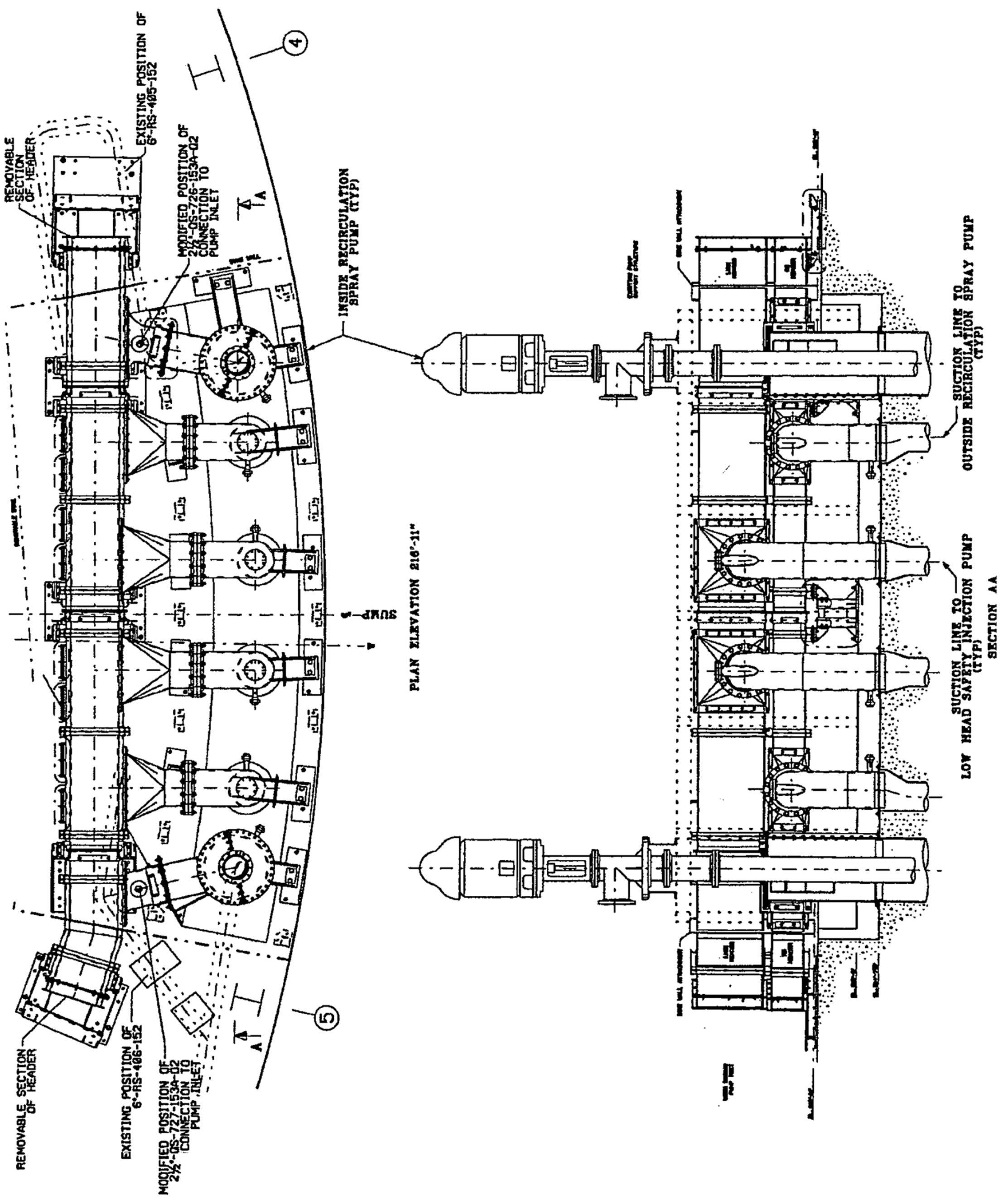


Figure 6.2-57
SIMPLIFIED RECIRCULATION SPRAY SYSTEM

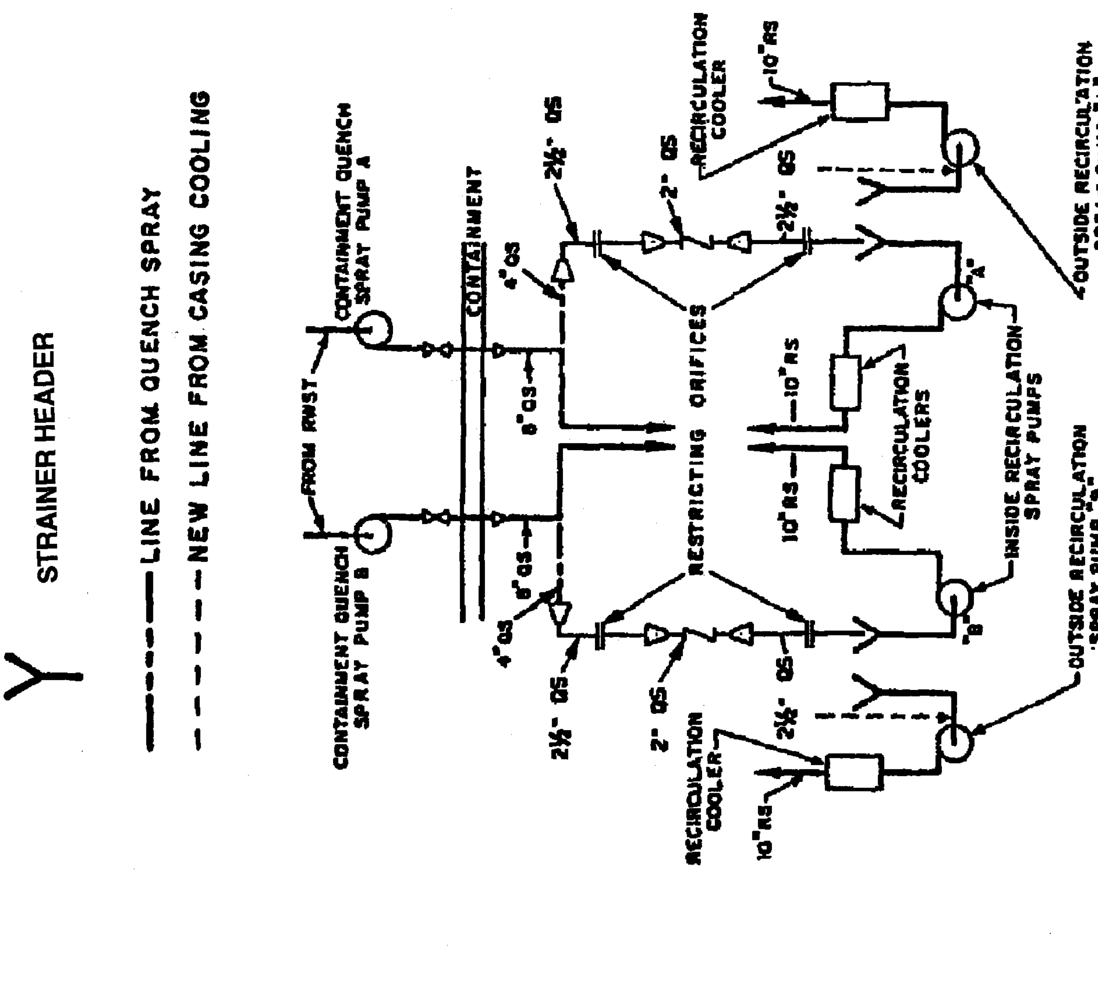


Figure 6.2-58
ARRANGEMENT CONTAINMENT SUMP STRAINER FOR UNIT 1

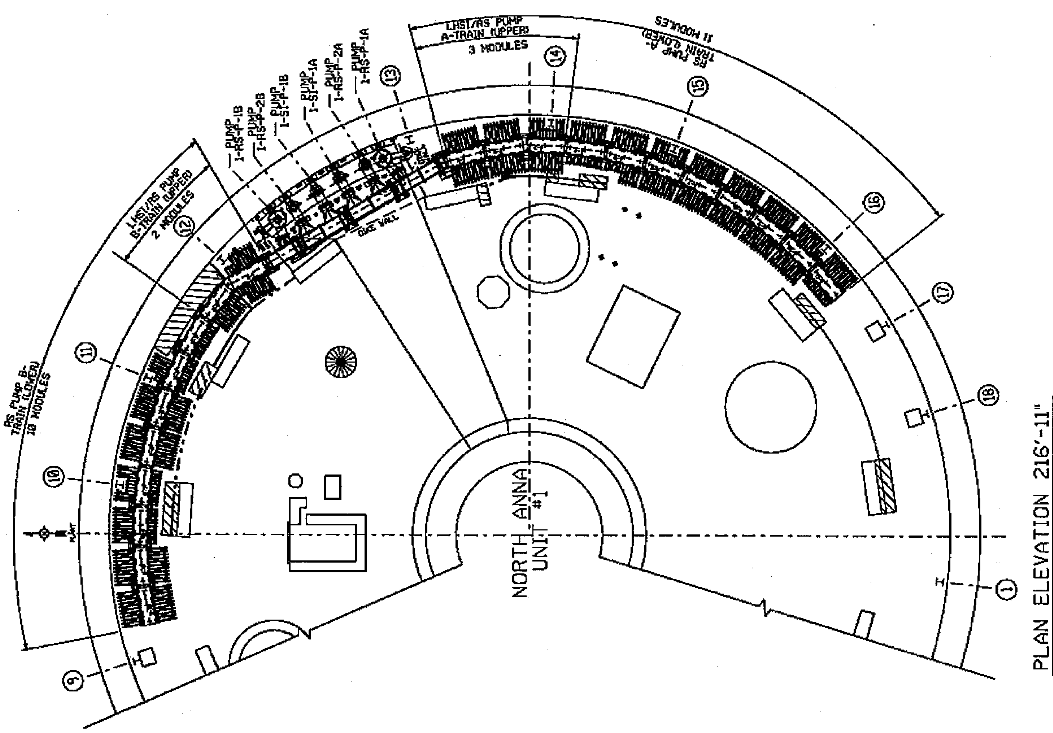


Figure 6.2-59
ARRANGEMENT CONTAINMENT SUMP STRAINER HEADER FOR UNIT 1

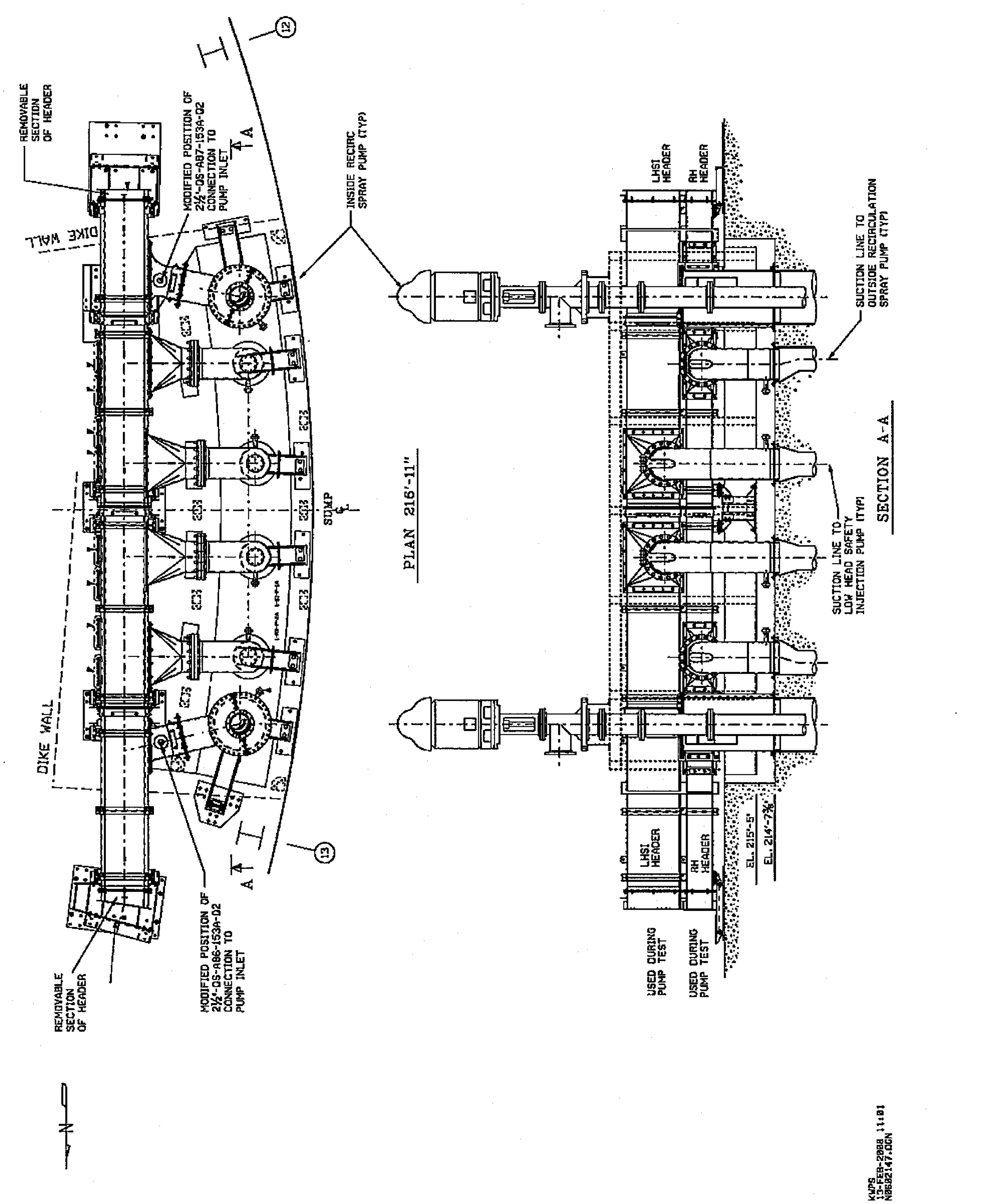


Figure 6.2-60
CONTAINMENT PRESSURE FROM DEHLG PEAK PRESSURE ANALYSIS

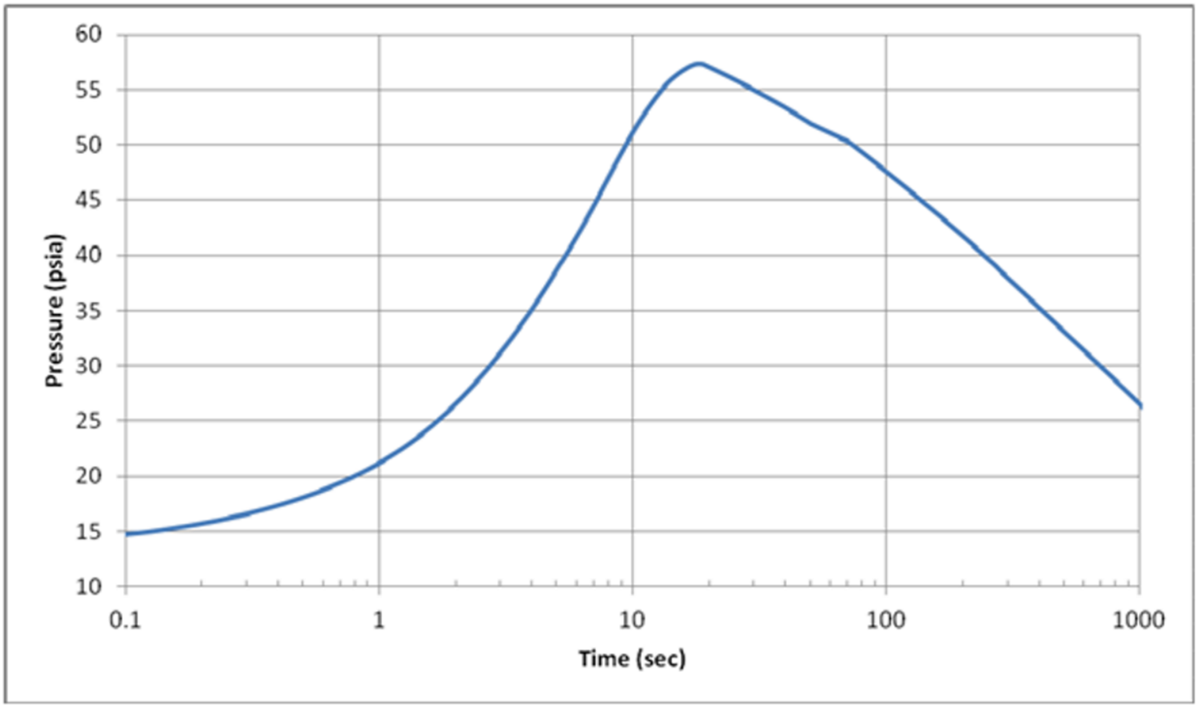


Figure 6.2-61
CONTAINMENT TEMPERATURE FROM DEHLG PEAK PRESSURE ANALYSIS

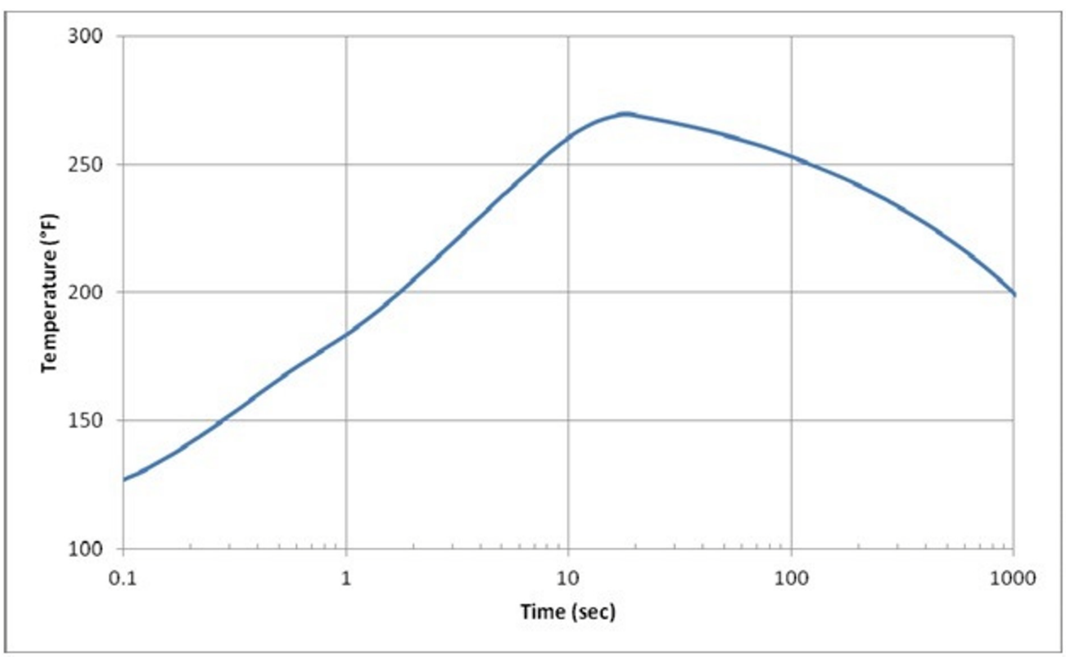


Figure 6.2-62
CONTAINMENT PRESSURE FROM DEPSG
DEPRESSURIZATION ANALYSIS AT 55°F SW

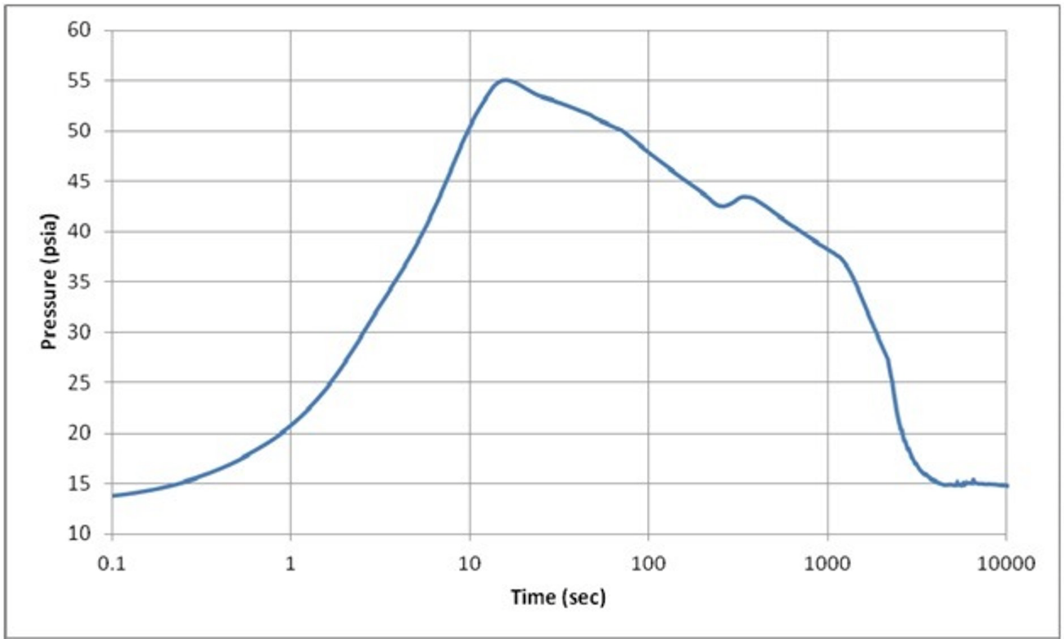


Figure 6.2-63
CONTAINMENT TEMPERATURE FROM DEPSG
DEPRESSURIZATION ANALYSIS AT 55°F SW

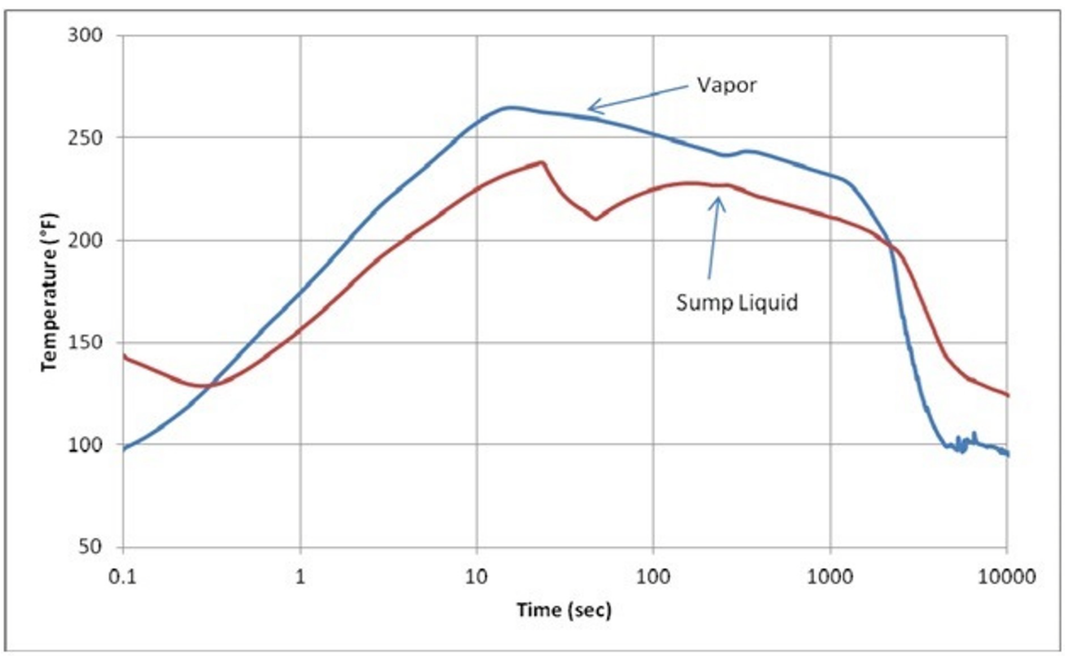


Figure 6.2-64
RS COOLER HEAT RATE FROM DEPSG DEPRESSURIZATION ANALYSIS AT 55°F SW

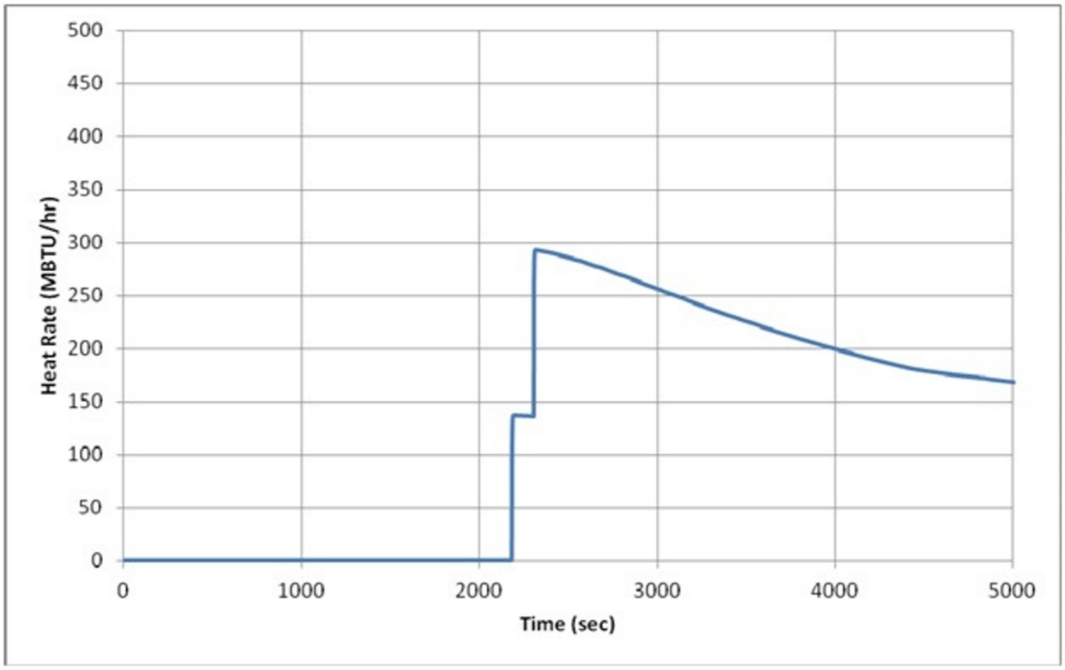


Figure 6.2-65
AVAILABLE NPSH INSIDE RS PUMP NPSH AVAILABLE ANALYSIS

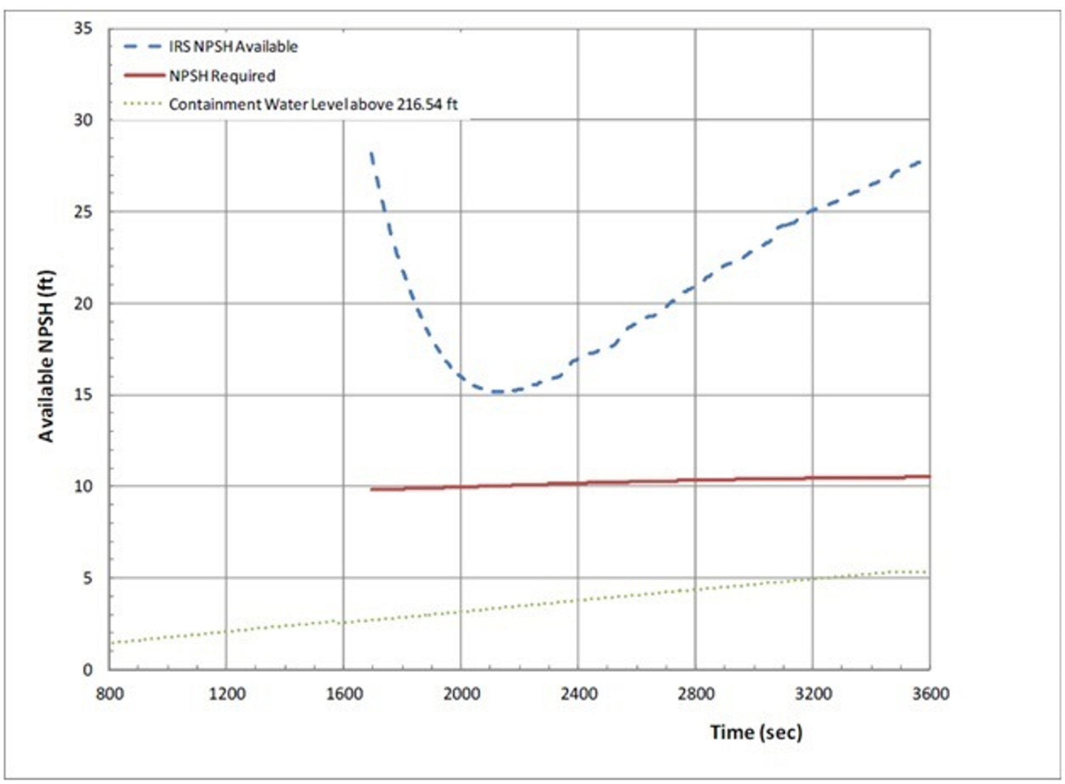


Figure 6.2-66
CONTAINMENT PRESSURE INSIDE RS PUMP NPSH AVAILABLE ANALYSIS

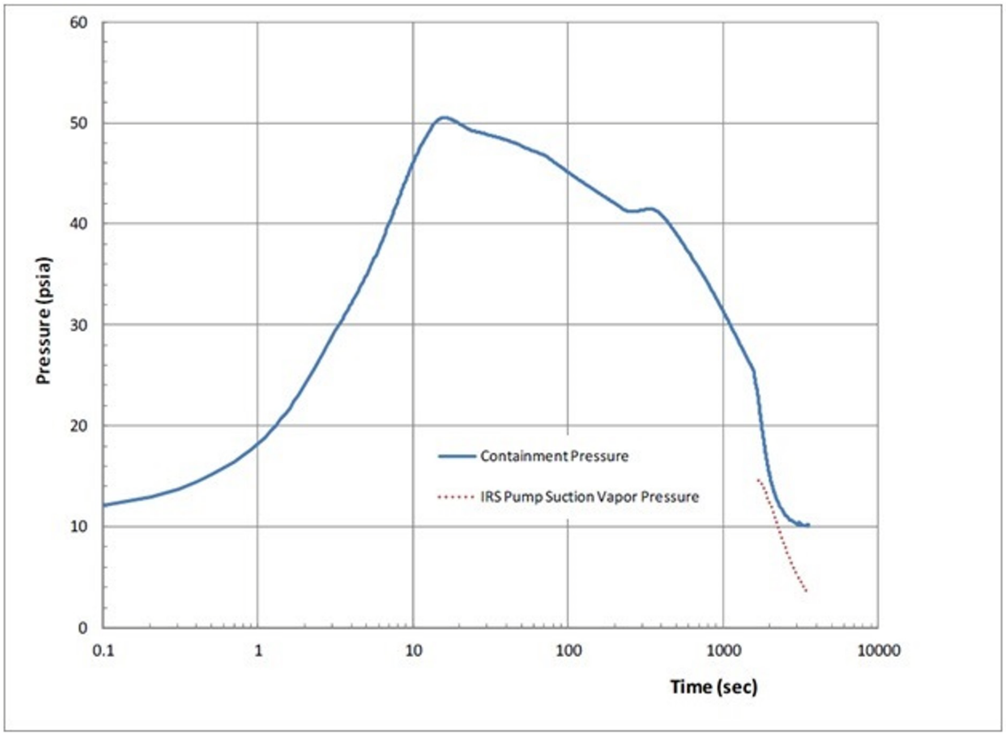


Figure 6.2-67
CONTAINMENT TEMPERATURE INSIDE RS PUMP NPSH AVAILABLE ANALYSIS

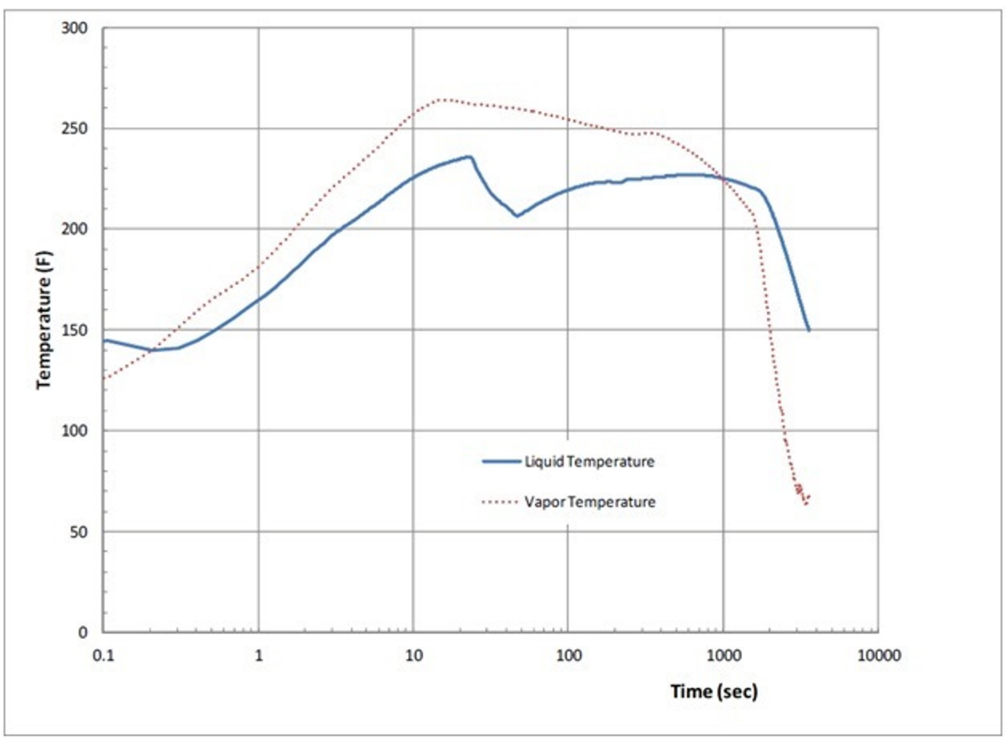


Figure 6.2-68
TOTAL RSHX HEAT RATE INSIDE RS PUMP NPSH AVAILABLE ANALYSIS

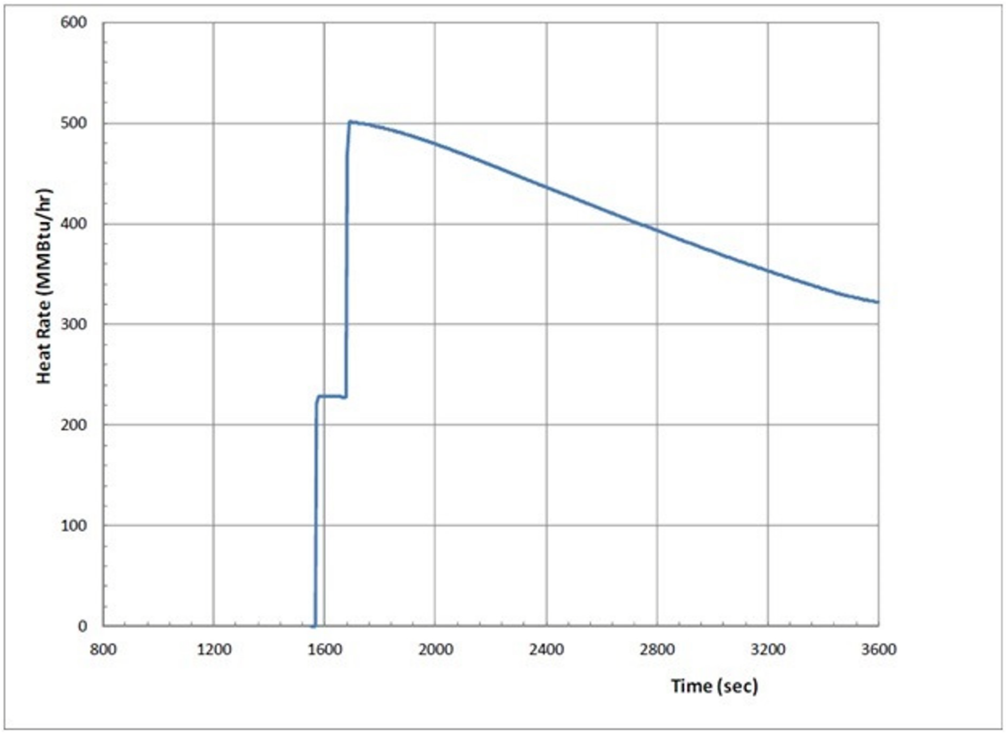


Figure 6.2-69
AVAILABLE NPSH OUTSIDE RS PUMP NPSH AVAILABLE ANALYSIS

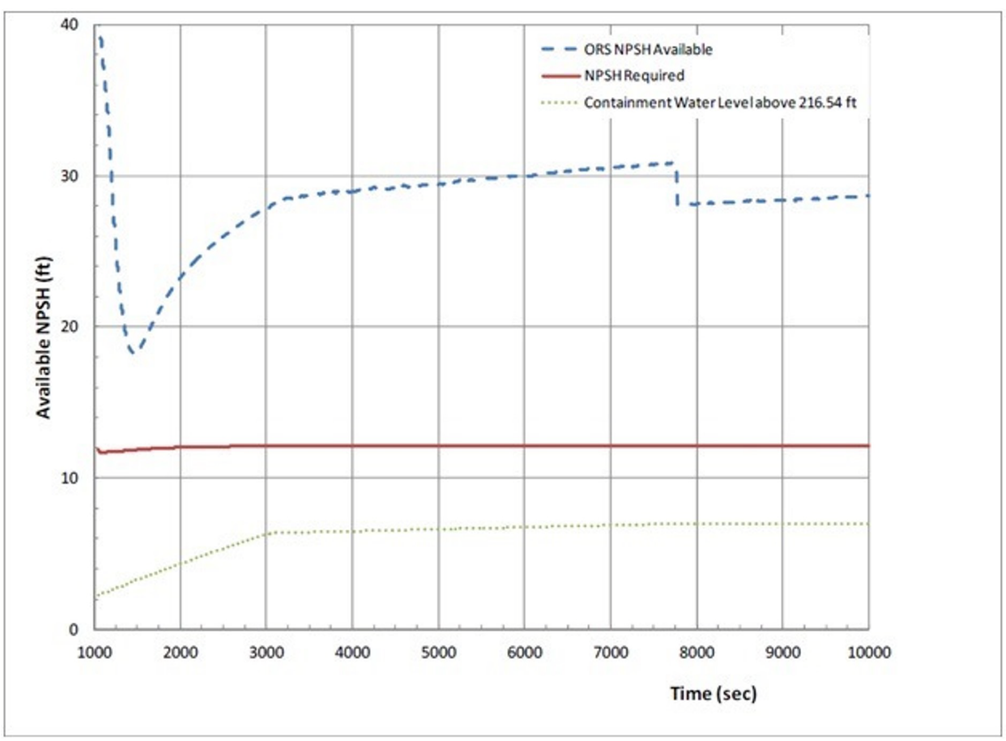


Figure 6.2-70
CONTAINMENT PRESSURE OUTSIDE RS PUMP NPSH AVAILABLE ANALYSIS

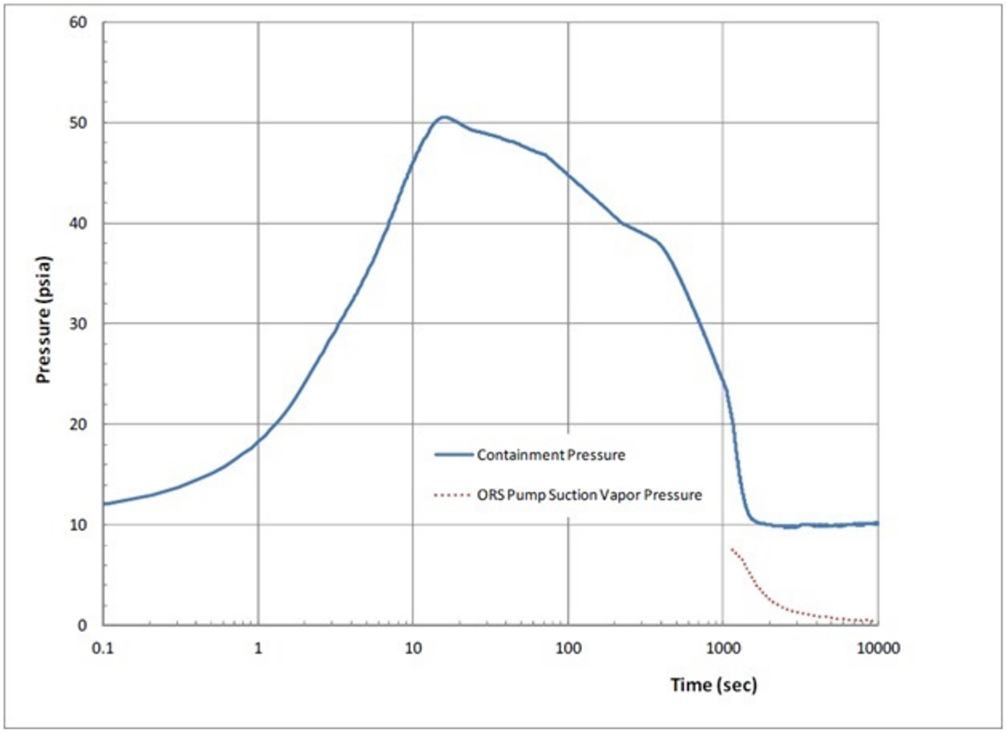


Figure 6.2-71
CONTAINMENT TEMPERATURE OUTSIDE RS PUMP NPSH AVAILABLE ANALYSIS

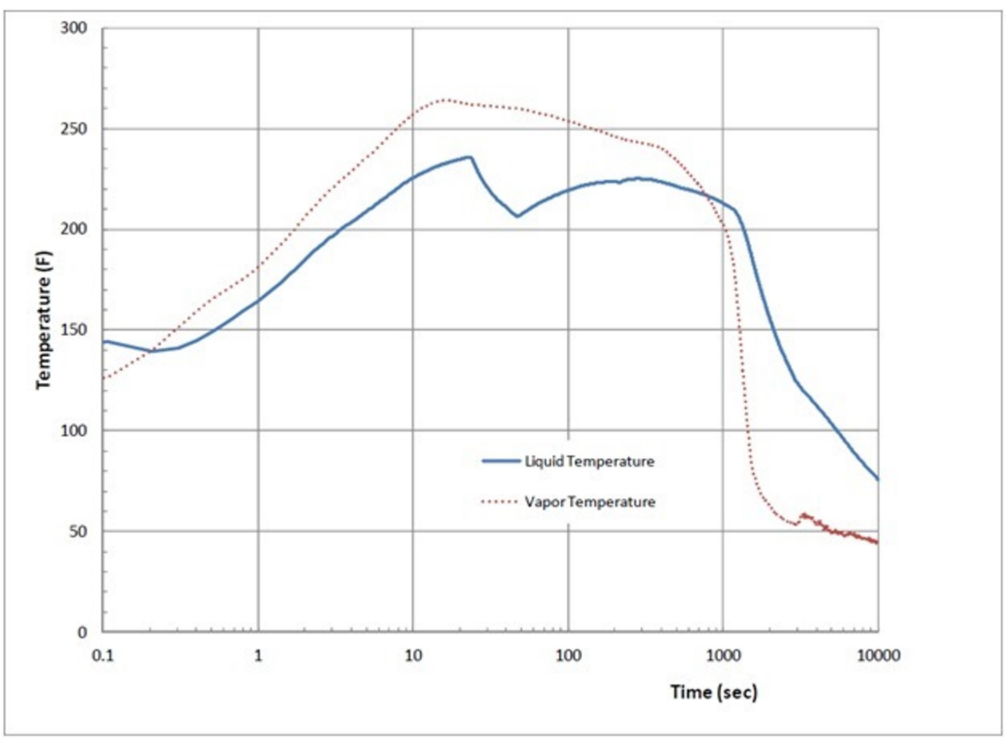


Figure 6.2-72
TOTAL RSHX HEAT RATE OUTSIDE RS PUMP NPSH AVAILABLE ANALYSIS

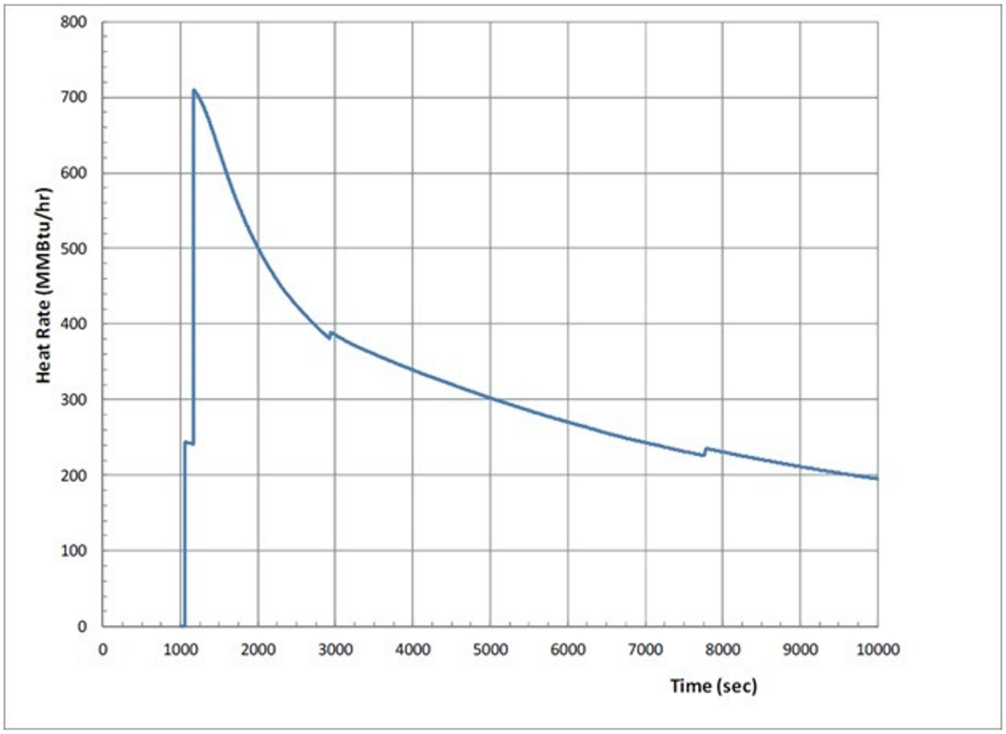


Figure 6.2-73
RECIRCULATION SPRAY SUBSYSTEM FLOW TESTING ARRANGEMENT

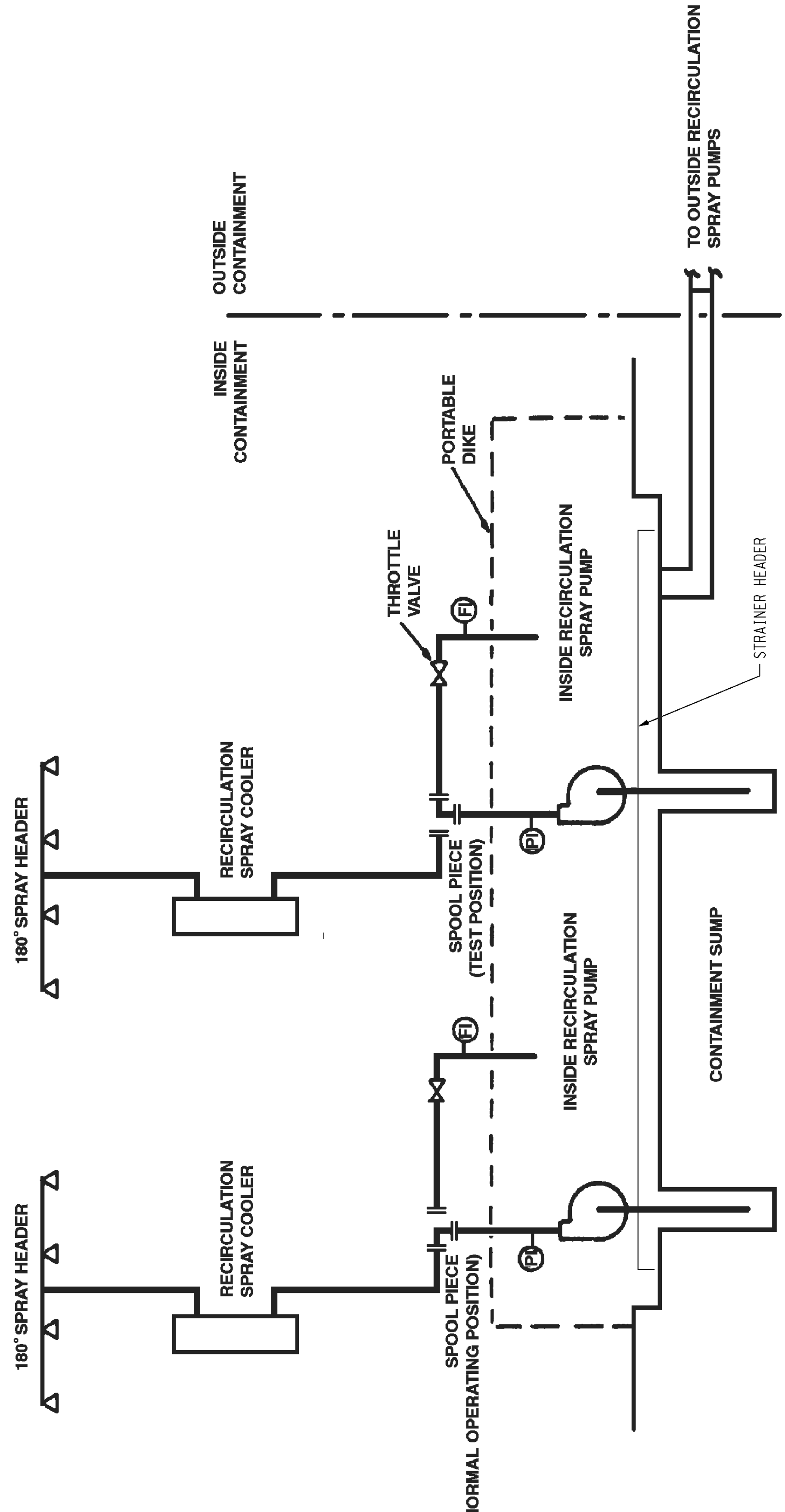


Figure 6.2-74
QUENCH SPRAY FLOW RATE VS. TIME:
DER OF A HOT LEG, WINTER CONDITIONS, MINIMUM ESF

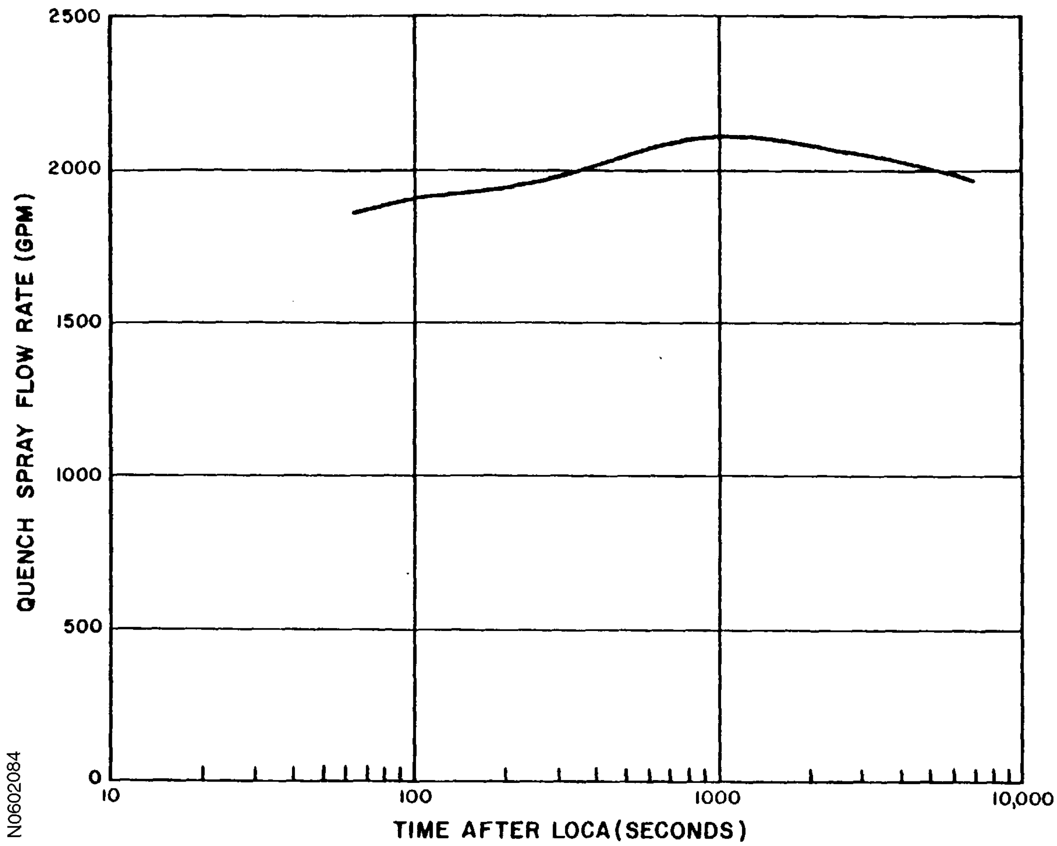


Figure 6.2-75
QUENCH SPRAY FLOW VS. TIME DER OF A HOT LEG, WINTER CONDITIONS, NORMAL ESF EXCEPT MINIMUM QUENCH SPRAYS

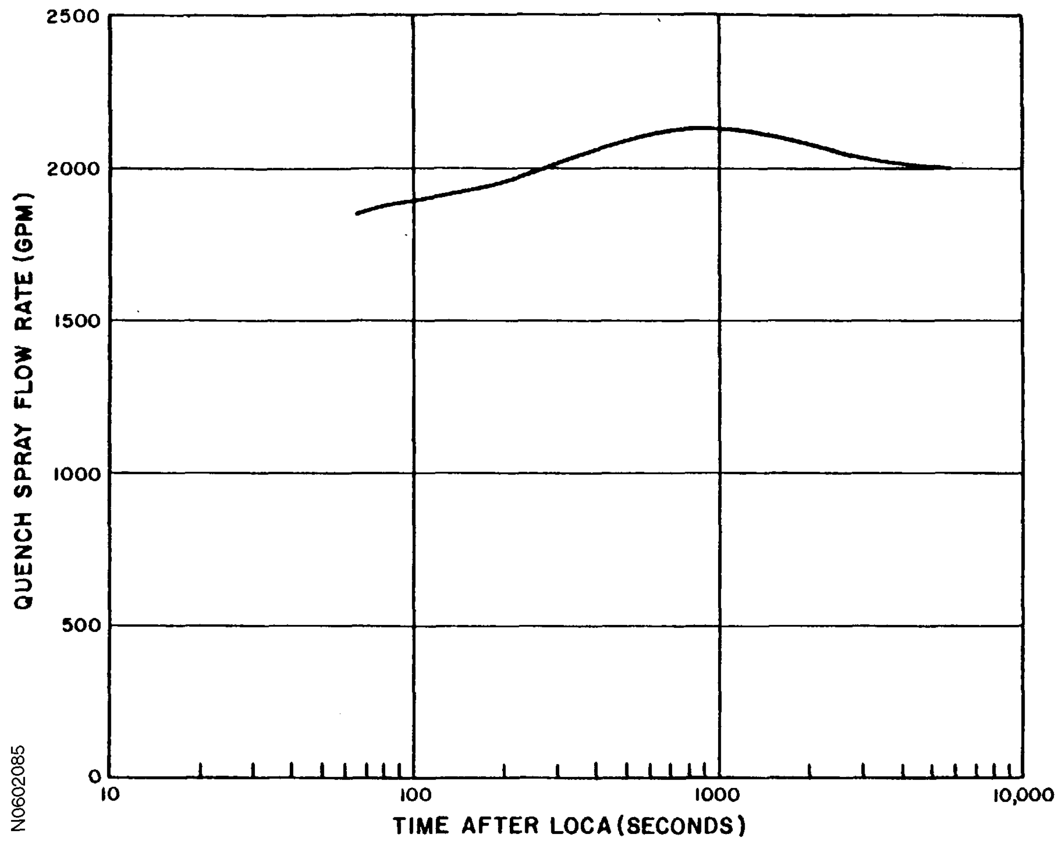


Figure 6.2-76
STATIC HEAD IN RWST VS. TIME DER OF HOT LEG,
WINTER CONDITIONS, MINIMUM ESF

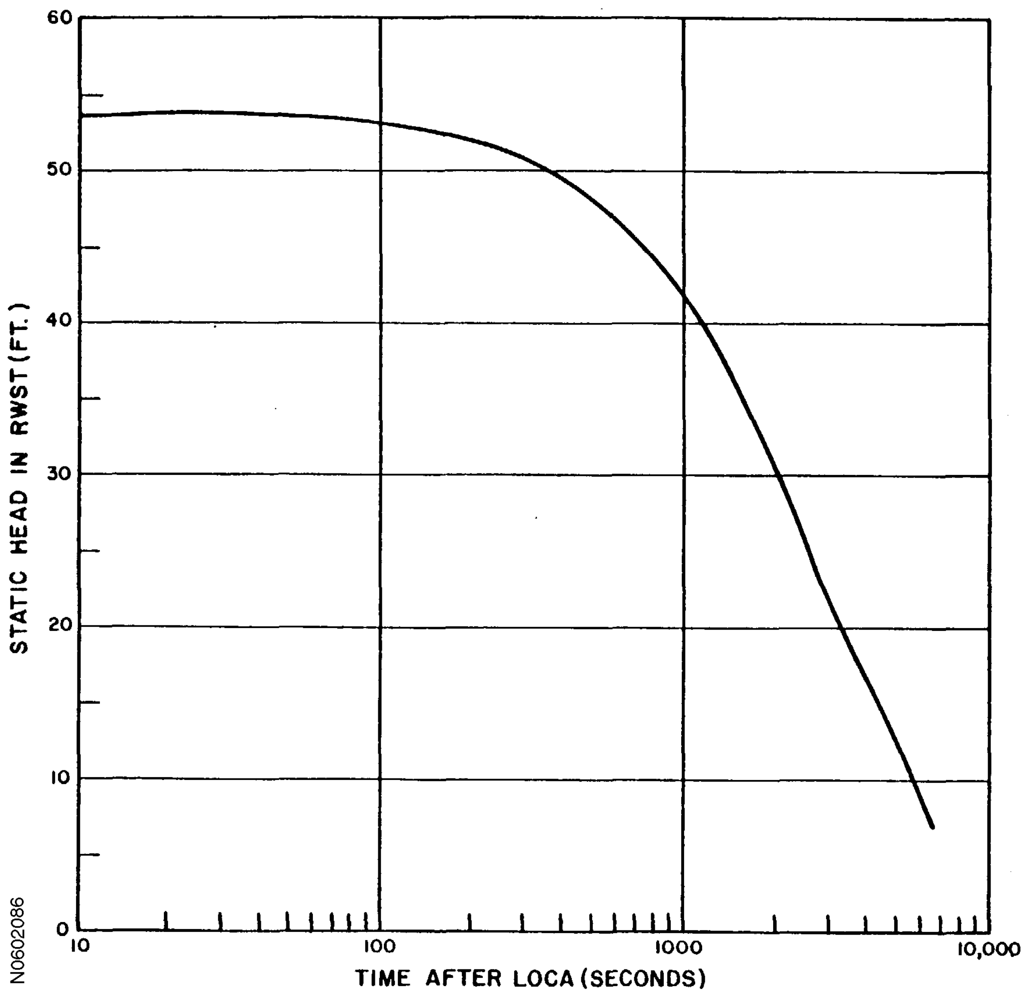


Figure 6.2-77
STATIC HEAD IN RWST VS. TIME DER OF HOT LEG, WINTER CONDITIONS, NORMAL ESF EXCEPT MINIMUM QUENCH SPRAYS

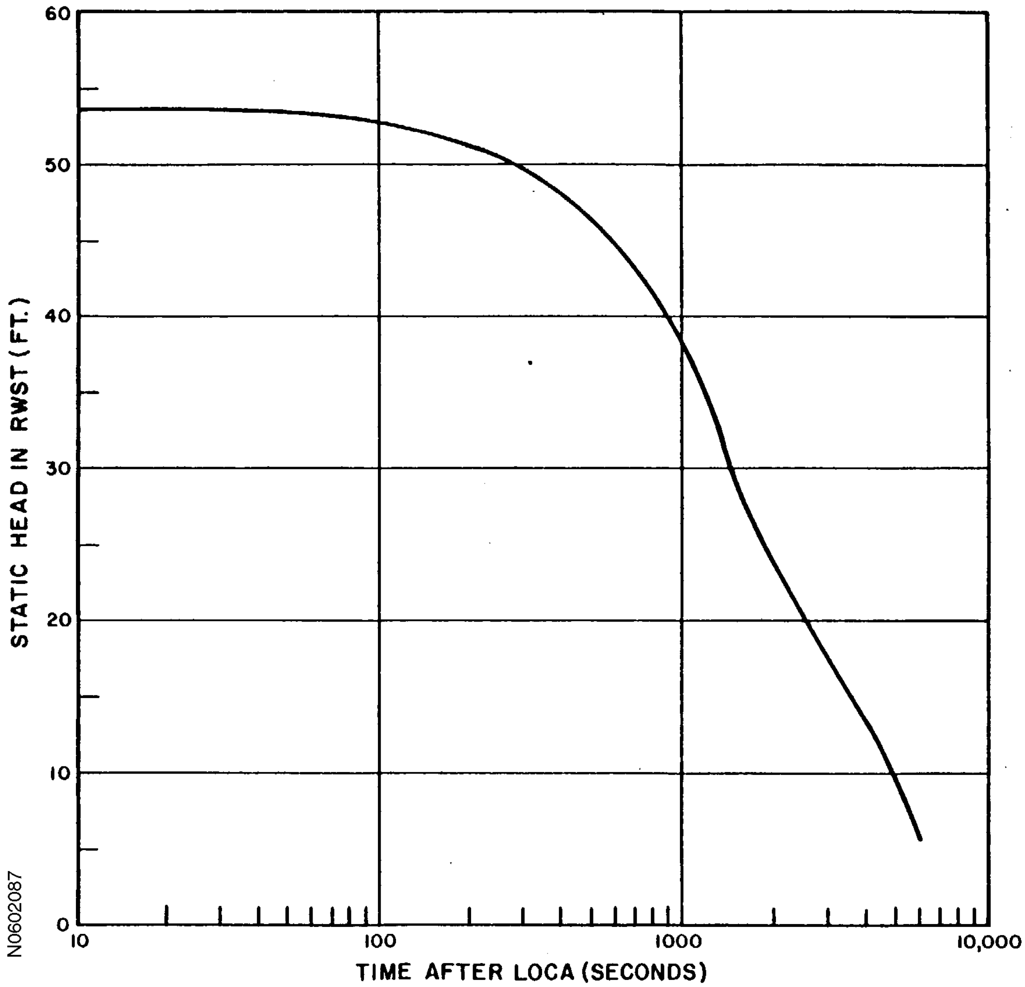


Figure 6.2-78
STATIC HEAD IN CAT
VS. TIME DER OF HOT LEG, WINTER CONDITIONS, MINIMUM ESF

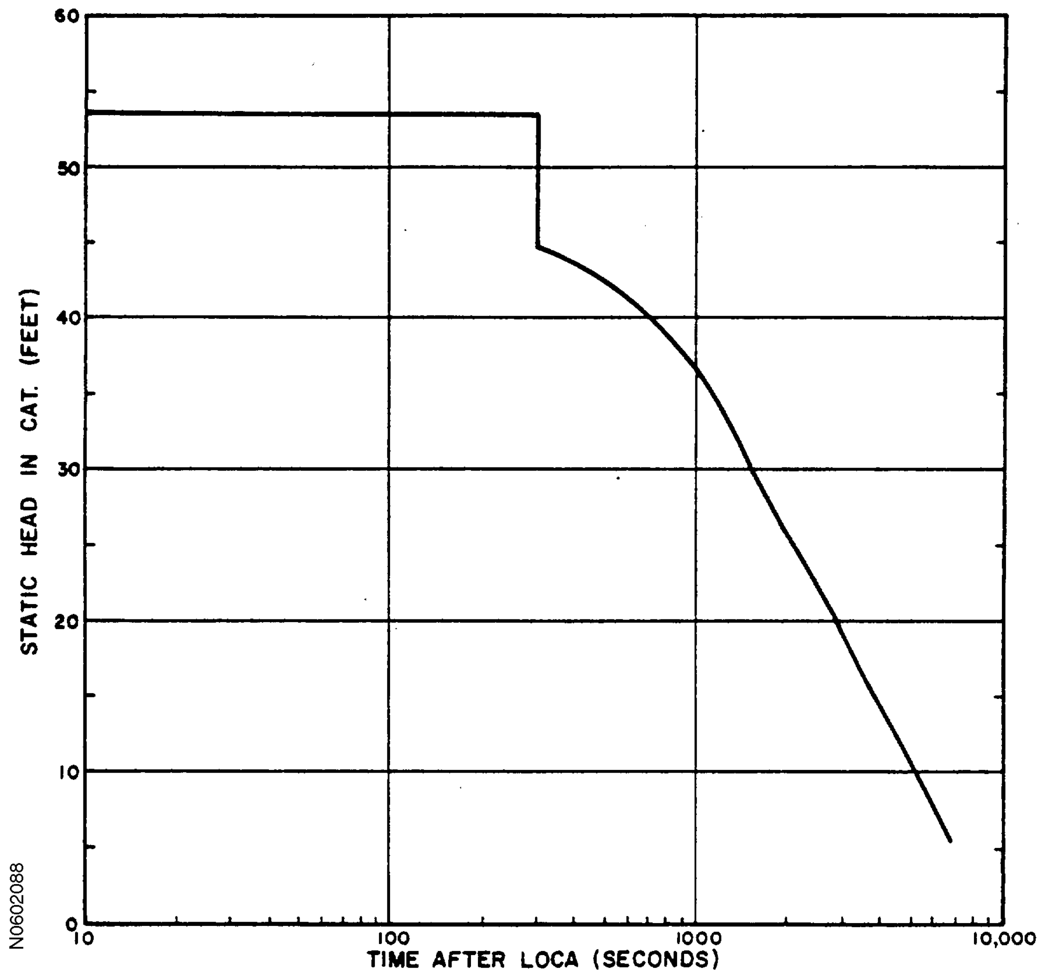


Figure 6.2-79
STATIC HEAD IN CAT VS. TIME DER OF A HOT LEG, WINTER CONDITIONS, NORMAL ESF EXCEPT MINIMUM QUENCH SPRAYS

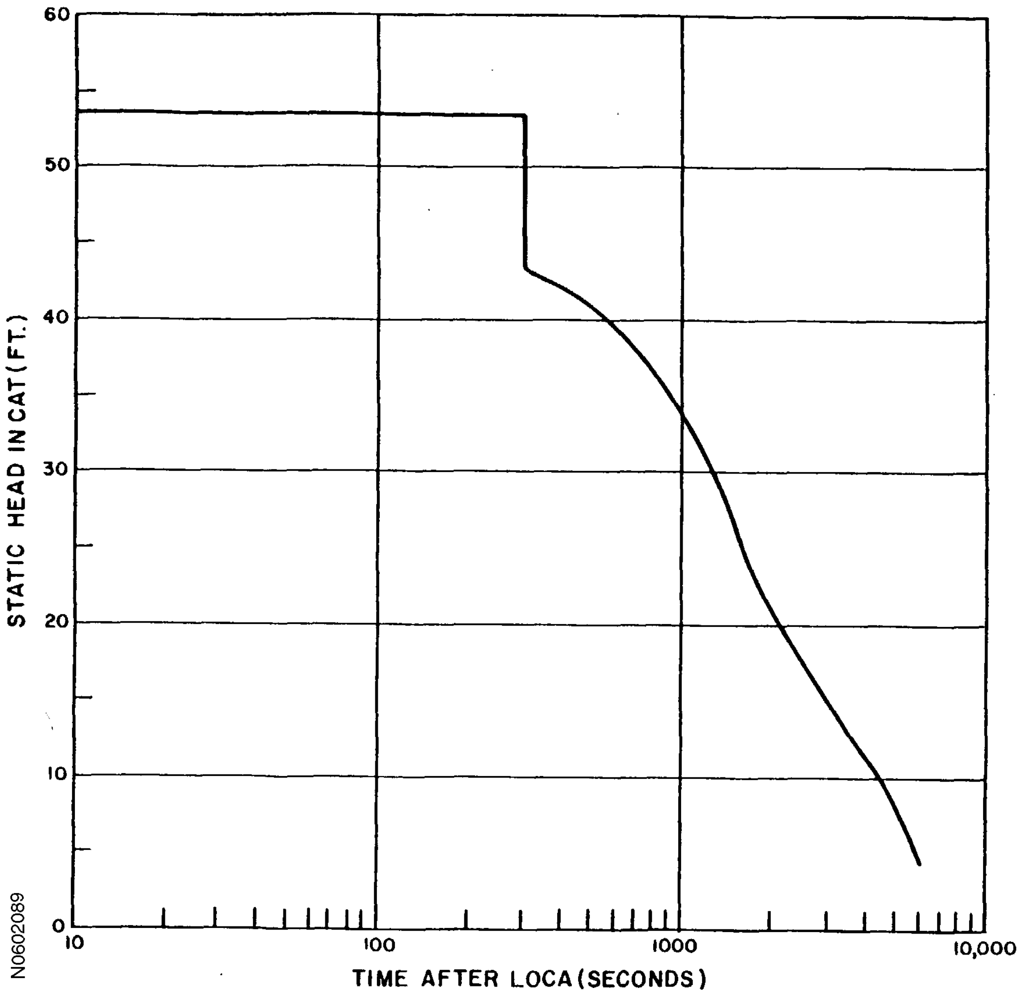


Figure 6.2-80
CONCENTRATION OF NaOH VS. TIME FOR QUENCH SPRAY AND CONTAINMENT AND SUMP SOLUTIONS HOT LEG DER, WINTER CONDITIONS, MINIMUM ESF

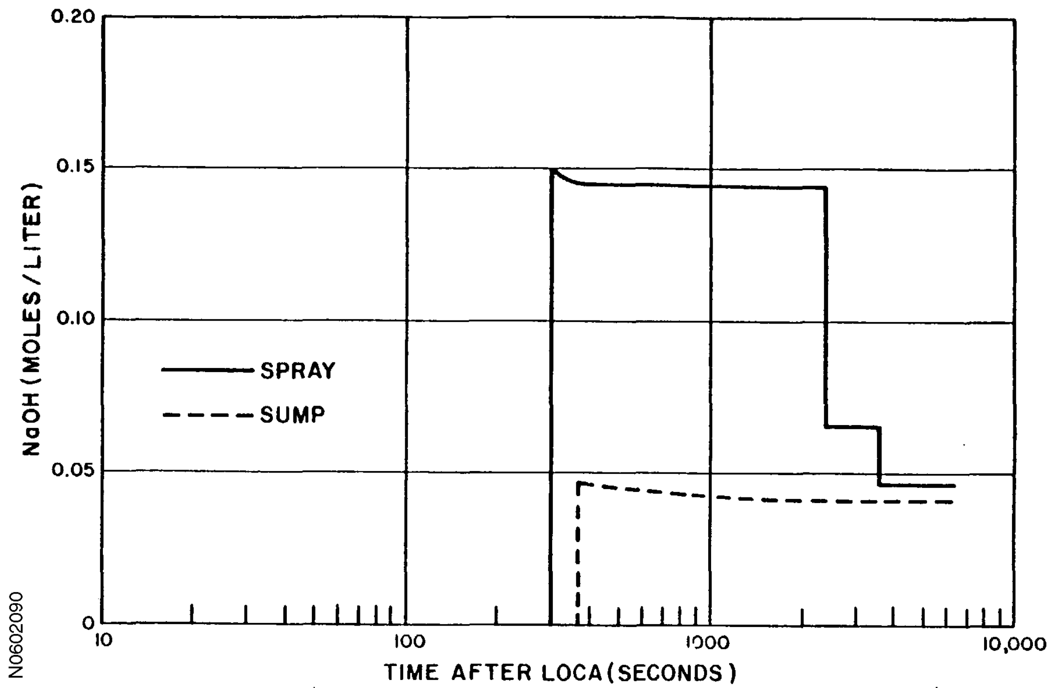


Figure 6.2-81
CONCENTRATION OF NaOH VS. TIME FOR QUENCH SPRAY AND CONTAINMENT SUMP SOLUTIONS HOT LEG DER, WINTER CONDITIONS, NORMAL ESF EXCEPT MINIMUM QUENCH SPRAYS

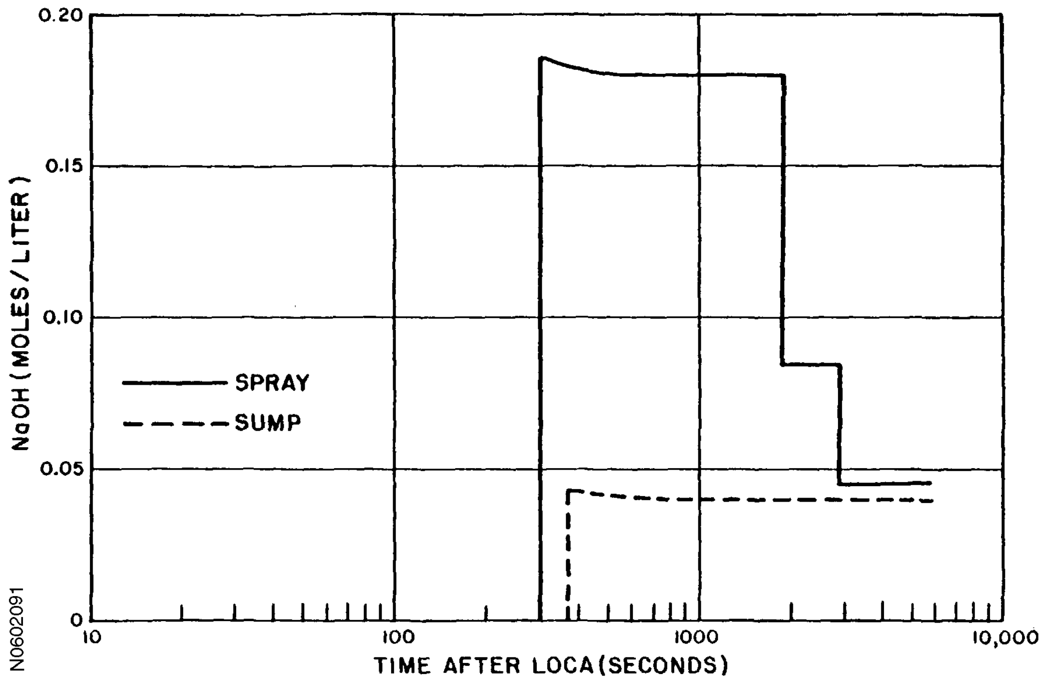


Figure 6.2-82
CONTAINMENT ATMOSPHERE CLEANUP SYSTEM

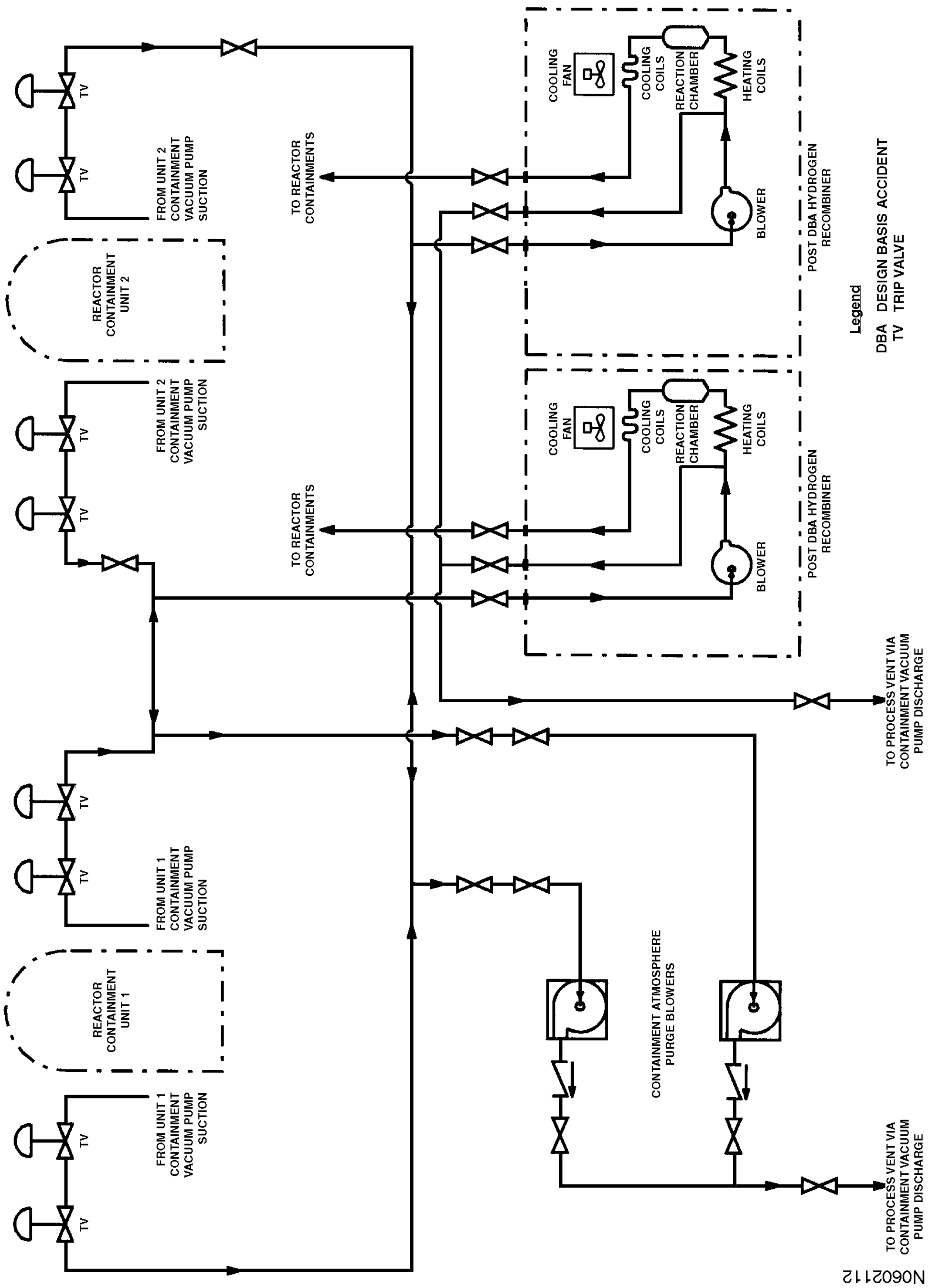


Figure 6.2-83
TYPICAL CONTAINMENT ISOLATION ARRANGEMENTS

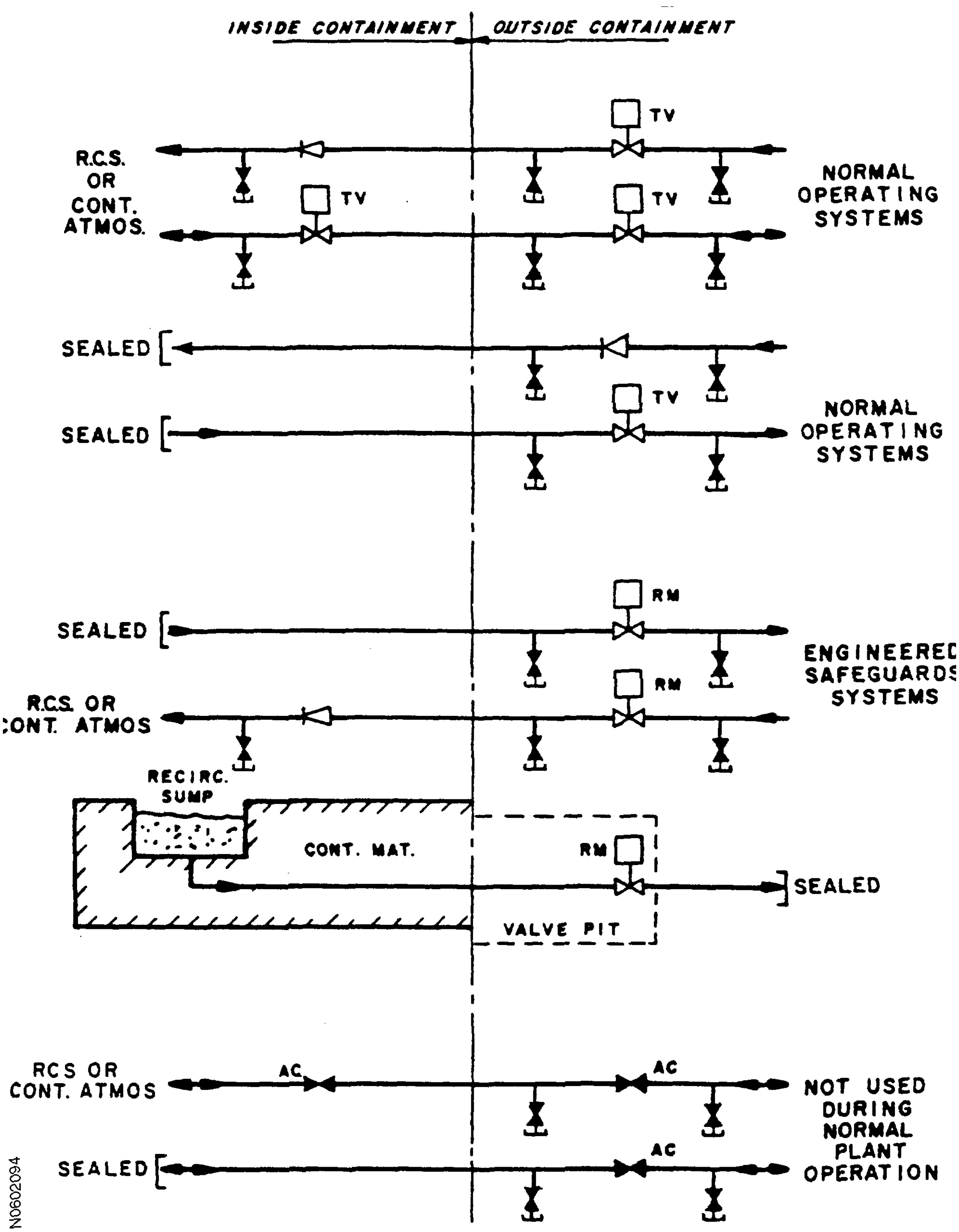


Figure 6.2-84
CONTAINMENT VACUUM SYSTEM

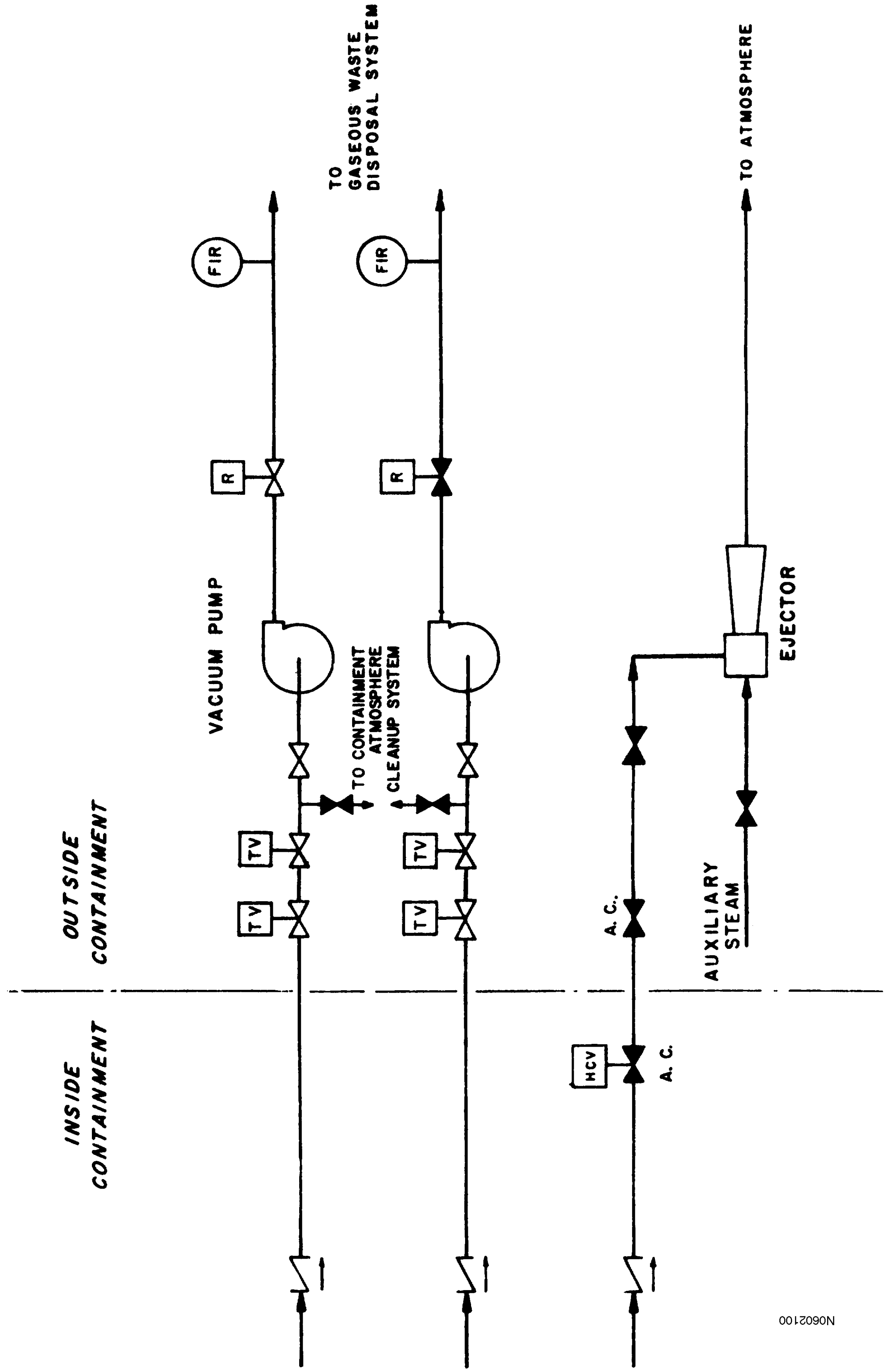
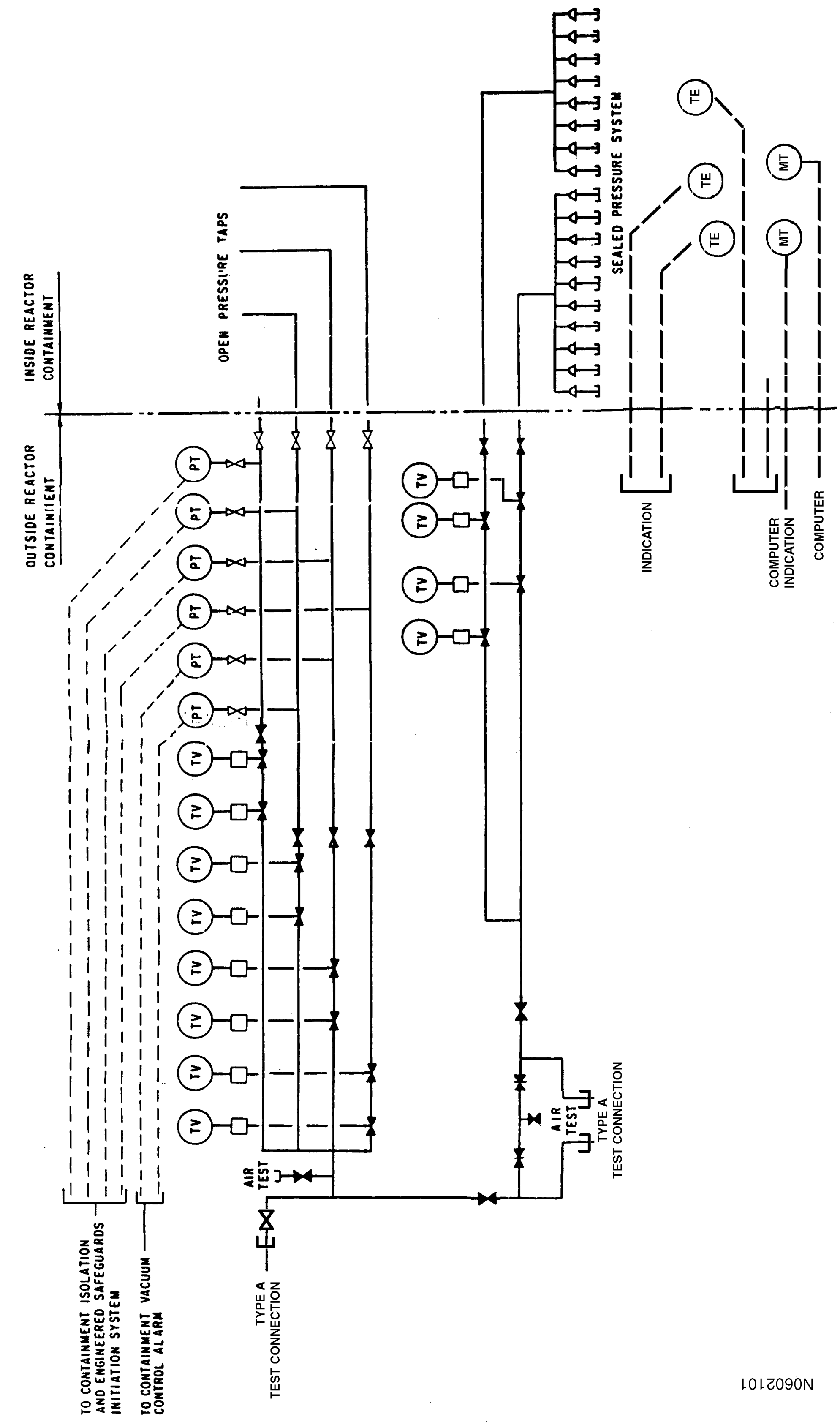


Figure 6.2-85
INSTALLED LEAKAGE MONITORING SYSTEM



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6.3 EMERGENCY CORE COOLING SYSTEM

6.3.1 Design Bases

The emergency core cooling system (ECCS) is designed to cool the reactor core as well as to provide additional shutdown capability following initiation of the following accident conditions:

1. Pipe breaks and spurious relief valve lifting in the reactor coolant system (RCS) that cause a discharge larger than that which can be made up by the normal makeup system, up to and including the instantaneous circumferential rupture of the largest pipe in the RCS.
2. Rupture of a control rod drive mechanism (CRDM) causing a rod cluster control assembly ejection accident.
3. Pipe breaks and spurious safety valve lifting in the main steam system, up to and including the instantaneous circumferential rupture of the largest pipe in the main steam system.
4. A steam generator tube rupture.

The acceptance criteria for the consequences of each of these accidents is described in Chapter 15 in the respective accident analyses sections.

6.3.1.1 Core Cooling Capability

The primary function of the ECCS following a loss-of-coolant accident (LOCA) is to remove the stored and fission product decay heat from the reactor core such that fuel rod damage, to the extent that it would impair effective cooling of the core, is prevented. The acceptance criteria for the accidents, as well as analyses of the accidents, are provided in Chapter 15.

6.3.1.2 Shutdown Capability

The ECCS provides shutdown capability for the accidents listed above by means of 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 Chapter 15.

6.3.1.3 Single-Failure Capability

In order to ensure that the ECCS will perform its desired function during the accidents listed above, 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 Appendix 6A, Section 6A.3.1.

6.3.1.4 Loss of Offsite Power

The ECCS is designed to meet its minimum required level of functional performance with onsite electrical power system operation (assuming offsite power is not available) or with offsite

electrical power system operation (assuming onsite power is not available) for any of the above abnormal occurrences, assuming a single failure as defined above.

6.3.1.5 **Seismic Requirements**

The ECCS is designed to perform its function of ensuring core cooling and providing shutdown capability following an accident under simultaneous design-basis earthquake loading. The seismic requirements are defined in Chapter 3.

6.3.2 System Design

The ECCS is shown in Figure 6.3-1, Reference Drawings 1 and 2, and Reference Drawings 6 and 8. Pertinent design and operating parameters for the components of the ECCS are given in Table 6.3-1. The codes and standards to which the individual components of the ECCS are designed are listed in Table 6.3-2.

Flow and pressure transients associated with various LOCA conditions are discussed in Section 15.4.1.

The analysis presented in Section 6.3.3.1.4 for Unit 1 demonstrates that check valves are not required in either branch line from the refueling water storage tank (RWST) to the suction of the low head safety injection (LHSI) pumps, and that ECCS flows will not be less than minimum requirements should the motor-operated valves misfunction.

The component design pressure and temperature conditions are specified as the most severe conditions to which each respective component is exposed during either normal plant operation or during operation of the ECCS. These conditions are considered for each component in relation to the code to which it is designed. The fundamental assurance of structural integrity of the ECCS components is maintained by designing the components in accordance with applicable codes, and with due consideration for the design and operating conditions. Components of the ECCS are designed to withstand the appropriate seismic loadings in accordance with their safety class as given in Table 6.3-2.

Materials employed for components of the ECCS are given in Table 6.3-3. Materials are selected to meet the applicable material requirements of the codes in Table 6.3-2 and the following additional requirements:

1. All parts of components in contact with borated water are fabricated of, or clad with, austenitic stainless steel or equivalent corrosion-resistant material.
2. All parts of components in contact (internally) with sump solution during recirculation are fabricated of austenitic stainless steel or equivalent corrosion-resistant material.
3. Valve seating surfaces are hard faced with Stellite Number 6 or equivalent to prevent galling and to reduce wear.

4. Valve stem materials are selected for their corrosion resistance, high tensile properties, and resistance to surface scoring by the packing.

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.

The following information is HISTORICAL and is not intended or expected to be updated for the life of the plant.

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.

6.3.2.1 **Component Description**

6.3.2.1.1 Accumulators

The accumulators are pressure vessels filled with borated water and pressurized with nitrogen gas. During normal operation each accumulator is isolated from the RCS by two check valves in series. Should the RCS pressure fall below the accumulator pressure, the check valves open and borated water is forced into the RCS. One accumulator is attached to each cold leg of the RCS. Mechanical operation of the swing-disk check valves is the only action required to open the injection path from the accumulators to the core via the cold leg.

Although the run of piping between the two SIA discharge check valves is credited in meeting the minimum SIA volume requirement, the minimum boron concentration requirement does not apply to this run of piping. Applicable accident analyses have explicitly considered in-leakage from the RCS, and the resulting reduction in boron concentration in this run of piping, which is not sampled.

Connections are provided for remotely adjusting the level and boron concentration of the water in each accumulator during normal plant operation as required. Accumulator water level may be lowered by draining to the primary drain transfer tank or to the refueling water storage tank (RWST), or raised by pumping borated water from the RWST. Accumulator water level may also be controlled by cross-connecting the liquid spaces of two accumulators for a duration of less than one hour. Samples of the solution in the accumulators are taken periodically to verify the boron concentration. One accumulator at a time may be recirculated to the RWST through the accumulator test line to control boron concentration.

Accumulator pressure is provided by a supply of nitrogen gas, and can be adjusted as required during normal plant operation. However, the accumulators are normally isolated from this nitrogen supply. Relief valves on the accumulators protect them from overpressure.

Since the accumulators are located within the containment, a release of the nitrogen gas in the accumulators would cause a slight increase in normal containment pressure. The containment pressure following release of the gas from all three accumulators would be increased less than 1 psi, which is well below the containment pressure setpoint for ECCS actuation.

In the event of an equipment failure or operator error that results in release of accumulator gas to the containment, this release would be detected by accumulator pressure indicators and alarms. This enables the operator to take the action required to maintain plant operation within the requirements of Technical Specifications covering accumulator operability.

Three accumulators are provided, each containing a minimum volume of 1013 ft^3 of borated water. Each accumulator is provided with a total of 10 penetrations. The outlet nozzle, the makeup connection, and the sample connection are provided in the bottom hemispherical head of the accumulator. In addition, a remote sample connection is provided in the accumulator test line for sampling for boron concentration from the Safeguards Building. The nitrogen gas inlet connection is provided in the upper hemispherical head of the accumulator. The four level transmitter penetrations, the relief valve connection, and a manway are provided in the cylindrical shell portion of the accumulator. In analyzing the LOCA, the contents of one accumulator are assumed to spill into the containment via the postulated break. Added conservatism is introduced into the analysis by assuming that all accumulator water injected prior to the end of blowdown is discarded.

Westinghouse criteria for the location of the accumulators require that they be placed outside of the missile barrier to protect them from missiles. The location of the accumulator above or below the RCS cold leg has no significant effect on accumulator injection during the accident. The accumulator injection following a large break is shown as a function of time in Figure 15.4-9.

A parametric analysis of accumulator pressure and water level has been made to assess the effect of accumulator inleakage on a LOCA. The analysis is broken down into two parts:

1. Inleakage results in an increase in water volume and tank pressure. Figure 6.3-2 shows flow delivery curves for a typical accumulator system during a typical double-ended cold-leg guillotine (DECLG) blowdown transient for various initial water volumes and initial tank pressures. The tank pressure increase was proportioned to the gas volume reduction (in all cases, both tanks were assumed to be at the same conditions).
2. Figure 6.3-3 shows the flow delivery curves resulting from accumulators that had inleakage, but the assumption was made that the operator adjusted the tank pressure to 600 psia. These two figures are the basis of the following discussion of the effect of accumulator inleakage on the consequences of a LOCA.

One hundred percent of the accumulator water injected during blowdown is assumed to go around the downcomer annulus and out the break. The accumulator water entering the RCS has small effect on the blowdown transient, so the only interesting parameters for evaluating the effects of inleakage are the accumulator water left at the end of blowdown and the time required to deliver this water to the RCS.

In the case where both the water level and pressure were increased (Figure 6.3-2), increasing leakage increased the amount of water expelled during blowdown, but there was still more water (than in the design case) left at the end of blowdown for lower plenum refill and core reflood.

Two factors are considered in determining the effect on peak clad temperature. Having more accumulator water left at the end of blowdown results in an increase in the downcomer core water level (when the accumulators empty), which forces water into the core. This would tend to decrease peak clad temperature. Having more water takes longer to empty the accumulators. Considering the "cold-leg plug" requirement during accumulator injection, venting of steam through the intact loops is restricted for longer times, which tends to increase peak clad temperature. From this study, however, differences calculated in accumulator empty time and the amount of accumulator water left at the end of blowdown would cause very little difference in the peak clad temperature calculations. Further, it should be noted that the "cold-leg plug" is an artificial interim acceptance criteria requirement and steam-water mixing tests have shown that ECCS injection does not block the cold-leg lines. In the case where the water volume increased but the pressure was reset to 600 psia (Figure 6.3-3), similar reasoning applies. One noticeable difference between the accumulator flow transients of this set of calculations and those in Figure 6.3-2 is that for the same inleakage, accumulator empty time is much longer. This, again, would tend to increase peak clad temperature. For all cases, however, any penalty associated with the increased water delivery time would be negligible if the latest steam-water mixing data were applied.

6.3.2.1.2 Boron Injection Tank

The boron injection tank contains a nominal 8 weight percent of concentrated boric acid solution and is connected to the discharge of the centrifugal charging pumps. Upon actuation of the safety injection signal, the charging pumps provide the pressure and flow capacity necessary to inject the boric acid solution into the RCS when the isolation valves open.

To prevent cold spots and stratification within the tank during normal operation, the contents of the boron injection tank are continuously recirculated with the boric acid storage tanks via a boric acid transfer pump. The boron injection tank incorporates a sparger type inlet that distributes the incoming boric acid solution in a 360 degree fan as it enters the tank. This prevents channeling and also ensures radial homogeneity of the boric acid solution.

Redundant tank heaters and line heat tracing ensure the temperature of the solution remains at or above 115°F. This ensures the solution temperature remains above the solubility limit (111°F) for the maximum allowable boric acid concentration (9 weight percent or 15,750 ppm).

6.3.2.1.3 Pumps

6.3.2.1.3.1 *Low Head Safety Injection.* Two LHSI pumps are provided to deliver water from the RWST to the RCS when the RCS pressure falls below their shutoff head. These pumps are also used to recirculate water from the containment sump to the RCS and to the suction of the charging/high head safety injection pumps during the recirculation phase of accident recovery. Each pump is a two-stage, deepwell-type centrifugal pump, with self-contained mechanical seals driven by an induction motor.

A minimum flow bypass line is provided for each pump to recirculate fluid to the RWST for test purposes and for operation during a LOCA when the RCS pressure is above their shutoff head. This minimum flow bypass line is isolated during the recirculation phase following a LOCA. This line prevents deadheading the pumps and permits pump testing during normal operation.

The LHSI pump performance curves used as the basis for the ECCS analyses in Chapter 15 are based on the actual measured plant data for the installed pumps in both units. The LHSI pump performance curve used in the analytical model for the LHSI delivered flow is developed from the lowest measured installed pump head curve data, reduced to account for pump degradation and instrument accuracy.

A test of a Unit 2 LHSI pump by Lasalle Hydraulic Institute, documented in test report LHL-716 (Reference 2), recommended installation of a turbulence limiter in the suction bell of each pump and installation of a false bottom in the pump can to reduce the distance between the bottom of the pump can and the bottom of the suction bell to 0.25 suction bell diameters. These modifications were subsequently installed. The addition of these devices resulted in a reduction in pump head at lower flows, when compared to the pump manufacturer's test, in which these devices were not installed.

All the original LHSI pumps, being of identical design, performed with similar results to the indicated curve. Since the same flow-straightening devices have been installed on all the pump cans, the performance of the pumps would be similar to that of the tested pump.

6.3.2.1.3.2 *Centrifugal Charging Pumps.* These pumps deliver water from the RWST through the boron injection tank to the RCS at the prevailing RCS pressure during the injection phase. These pumps also provide recirculation flow to the RCS by taking suction from the LHSI pumps. Each centrifugal charging pump (CCP) is a multistage, diffuser design, barrel-type casing with vertical suction and discharge nozzles. Each pump has a self-contained lubrication system.

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. Upon a safety injection signal, plant operating procedures require the operator to:

1. Close the CCP miniflow isolation valves when the actual RCS pressure drops to the calculated pressure for manual reactor coolant pump trip.
2. Reopen the CCP miniflow isolation valves should the wide range RCS pressure subsequently rise to greater than 2000 psig.

During normal plant operation, at least one charging pump is in use. The other charging pumps can be tested during normal operation through the use of the minimum flow bypass line.

CH pump performance curves used as the basis for ECCS performance analyses in Chapter 15 are based on the actual measured plant data for the installed pumps in both units. The CH pump performance curve used in the analytical model for high-head safety injection (HHSI) delivered flow is developed from the lowest measured installed pump head curve data, reduced to account for pump degradation and instrument accuracy.

There are no ECCS pumps outside the containment that are housed in compartments through which steam lines pass.

6.3.2.1.4 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 are:

1. Valves that are normally open, except check valves and those that perform a control function, are provided with backseats to limit stem leakage.
2. Normally closed globe valves are installed with recirculation fluid pressure under the seat to prevent stem leakage of recirculated (radioactive) water. The Charging Pump Recirculation MOVs are installed with pressure over the seat to improve MOV design margin when closing the valve. The reversed flow configuration exposes the packing to full charging pump pressure with the valve closed.
3. Relief valves are enclosed, i.e., they are provided with a closed bonnet, and discharge to the safeguards area.

6.3.2.1.4.1 *Motor-Operated Valves*. The seating design of all motor-operated valves is of the parallel disk, flexible wedge, solid wedge, split wedge, or globe type design. All of the MOV designs have been analyzed and tested to ensure that they will perform their intended safety function under the worst-case accident conditions. The disks are guided to prevent chattering and to provide ease of movement. The seating surfaces are hard faced to prevent galling and to reduce wear.

When a gasket is employed for the body-to-bonnet joint, it is either a fully trapped, controlled compression, spiral wound graphoil gasket with provisions for seal welding, or it is of the pressure seal design with provisions for seal welding. The valve stuffing boxes are designed to maintain leakage within acceptable limits established by the total ESF component leakage assumed for post-LOCA accident dose analyses.

The motor operator incorporates a "hammer blow" feature that allows the motor to build up speed under no load conditions prior to engaging the disc sleeve. 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 disk.

Valves 1-SI-MOV-1860A/B and 2-SI-MOV-2860A/B provide isolation of the containment sump from the suction of the LHSI pumps. Valves 1-SI-MOV-1860A/B and 2-SI-MOV-2860A/B have two independent means of protection, either of which may be used to prevent the potential effect of thermally induced valve-bonnet pressure locking. Each of the valves has a normally open bonnet pressure equalization line that precludes the potential effect of thermally induced pressure locking during a DBA. Another means of protection is provided by the physical arrangement of the sump piping. The wetted sump piping, located below the floor of the sump piping, provides a passive thermal barrier that mitigates the effects of thermally induced pressure locking during a DBA (Reference 6). It should be noted that only one of the two protective measures described above is required to avert thermally induced pressure locking of valves 1-SI-MOV-1860A/B and 2-SI-MOV-2860A/B.

6.3.2.1.4.2 *Manual Globe, Gate, and Check Valves.* Gate valves are either wedge design or parallel disk and have straight-through flow paths. The wedge is either split or solid. All gate valves have backseat and outside screw and yoke.

Globe valves, "T" and "Y" style, are full ported with outside screw and yoke construction.

Check valves are spring-loaded lift piston types for sizes 2-inch and smaller, swing type for size 2-1/2-inch and larger, except for valves 1-CH-240 which is a 1-inch spring-loaded piston lift check type and 2-CH-155 which is a 1-inch ball type lift check valve. Stainless steel check valves have no penetration welds other than the inlet, outlet, and bonnet except for valves 1-SI-9, 1-SI-26, 1-SI-195, 1-SI-197, 1-SI-199, 2-SI-9, 2-SI-32, 2-SI-91, 2-SI-99, and 2-SI-105 which have a bonnet penetration to facilitate venting of the check valve bonnets. The check hinge is serviced through the bonnet.

Limitations are not imposed on modes of check valves failures as applied to FSAR single-failure assumptions consistent with accepted passive component definition. The most probable failure mechanism is back leakage through a normally closed valve. Such leakage is conservatively estimated to be small compared to the 50-gpm failure criteria.

The stem packing and gasket of the stainless steel manual globe and gate valves, 2-inch and larger, are similar to those described above for motor-operated valves. Some small valves have

been of the packless design since original construction or have been replaced with packless valves. Carbon steel manual valves are employed to pass nonradioactive fluids only and, therefore, do not contain the double packing and seal weld provisions.

6.3.2.1.4.3 *Vent Valves*. High point vents have been installed at critical points in the suction lines of the charging (HHSI) pumps, and the discharge lines of the LHSI pumps where gases could collect during plant operation. These vents have been installed to allow venting during plant operations to minimize the possibility of gas binding of the HHSI pumps. These vents are also used to reduce the potential for pressure surges, which may challenge the thermal relief valves upon LHSI pump starts. Following maintenance, the vents can be used to ensure the piping is adequately filled. (Reference 7).

6.3.2.1.4.4 *Diaphragm Valves*. The diaphragm valves are of the Saunders patent-type, which uses the diaphragm member for shutoff with even weir bodies. These valves are used in systems not exceeding 200°F and 200 psig design temperature and pressure.

6.3.2.1.4.5 *Accumulator Check Valves (Swing-Disk)*. The accumulator check valve is designed with a low pressure drop configuration with all operating parts contained within the body.

Design considerations and analyses that ensure that leakage across the check valves located in each accumulator injection line will not impair accumulator availability are as follows:

1. During normal operation the check valves are in the closed position with a nominal differential pressure across the disk 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.

2. When the RCS is being pressurized during the normal plant heatup operation, the check valves closest to the cold legs 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 disk 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.

3. The experience derived from the check valves employed in the emergency 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 and subsequent plants where the usage of check valves is identical to this application.

Each accumulator is provided with redundant level instrumentation. Each channel has a high and low level alarm. These alarms enable the operator to periodically adjust the accumulator water level as required. Hence, back leakage from the RCS to the accumulators via the check valves is not a serious problem.

The alarm setpoints are established consistent with the accumulator parameters used in the LOCA analysis. The operator must correct any accumulator alarm condition and, thus, the accumulator will be maintained in a state consistent with that assumed in the ECCS analysis.

4. The accumulators can accept some inleakage from the RCS without affecting availability. Inleakage would require, however, that the accumulator water volume be adjusted according to Technical Specification requirements. Accumulator boron concentration is verified after a cumulative volume change of ≥ 50% of indicated level that is not the result of addition from the RWST, in addition to normal sample frequency, thus protecting against accumulator dilution from the RCS.

6.3.2.1.4.6 *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, but 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. Other relief valves are installed in various sections of the ECCS to protect lines that have a lower design pressure than the RCS. Relief valve discharge locations are listed in Table 6.3-4. The wetted parts of the valve stem and spring adjustment assembly are of corrosion resistant material or isolated from the system fluids by a bellows seal between the valve disk and spindle. The closed bonnet provides an additional barrier for enclosure of the relief valves. Table 6.3-4 lists the system's relief valves with their capacities and setpoints.

There are no ECCS valves located outside the containment that are housed in compartments through which steam lines pass.

6.3.2.1.4.7 *Single Failure of ECCS Valves.* In order to prevent spurious action of selected motor-operated valves, an additional switch has been installed on the control board for these valves, which de-energizes the control power to the motor starters. This essentially ensures that spurious action of these motor-operated valves will not occur, by requiring that two separate contacts be closed by operation of two switches in order to apply power to the motor operators. The valves for which this modification has been implemented do not require automatic actuation prior to or during the course of a LOCA. The valves modified as described above are

MOV-1890A; 2890A - LHSI injection to hot leg.

MOV-1890B; 2890B - LHSI injection to hot leg.

MOV-1869A; 2869A - Charging pump injection to hot leg.

MOV-1869B; 2869B - Charging pump injection to hot leg.

MOV-1836; 2836 - Charging pump injection to cold leg.

These valves are normally closed; therefore, a mechanical failure would not prevent them from performing their intended isolation function. Redundancy is provided to ensure that there is adequate injection flow during hot-leg injection even if a mechanical or spurious failure occurs; therefore, spurious or mechanical failure of one of these valves during the hot-leg injection is not detrimental to the performance of the ECCS.

Motor-operated valves MOV-1890C and MOV-2890C (LHSI injection to cold leg) are normally open to provide an injection flow path to the cold legs. A failure of this valve may prevent injection flow and eliminate isolation capability. Installation of a redundant parallel valve ensures that the single-failure criteria are satisfied. This valve is shown on Reference Drawing 1.

The single-failure criteria for MOV-1885C and MOV-2885C (LHSI miniflow line) are satisfied by modifying the piping so that MOV-1885C is in series with MOV-1885A and an additional valve is provided in series with MOV-1885B, thereby eliminating the valve in the common line to the RWST. This modification is shown on Reference Drawing 1.

With regard to the RWST branch lines, a simplified sketch of the original ECCS design is shown in Figure 6.3-4. As can be seen in the figure, a second motor-operated valve (MOV-1862A) in the common line from the RWST to the suction of the LHSI pumps was installed in parallel with MOV-1862 to eliminate a potential for a single-failure problem. Modifying the piping as shown in Figure 6.3-4 created a situation of having two branch lines, each with its own motor-operated valve; thus, failure of MOV-1862A or MOV-1862B will only eliminate the flow to one of two LHSI pumps. From this illustration, it can be seen that the check valve in the branch line to the suction of LHSI pump 1-SI-P-1B was originally the check valve in the common line to the suction of the LHSI pumps. Since this valve was already installed at the time it was decided to modify the ECCS system, the check valve was simply left in place.

The accumulator discharge valves listed below are normally open and are not required to be operated during or subsequent to a LOCA. The single-failure criteria for MOV-1865A, B, and C are satisfied when the reactor coolant pressure is above the SI unblock setpoint, by blocking the valves (in nonisolated loops) in the open position as directed by the Technical Specifications. Blocking the valves (in nonisolated loops) in the open position is done by ensuring that the valves are open and then locking the breakers in the open position at the motor control center.

In addition to the modifications described above, redundant position indication (in the control room) has been provided for valves MOV-1836, MOV-1890A and B, MOV-1869A and B, and MOV-1865A, B, and C, as required by Branch Technical Position 18, *Application for the Single Failure Criterion to Manually-Controlled Electrically Operated Valves*. The details of these changes, including electrical schematics and instrumentation drawings, were forwarded to the Commission by the VEPCO letter of April 15, 1976 (Serial No. 963). The appropriate safety-related electrical schematics have been revised to show these changes.

The modifications and procedures described above (1) meet the requirements of Branch Technical Position EICSB 18, and (2) ensure that the single-failure criteria are satisfied for the North Anna ECCS.

6.3.2.1.5 Piping

All piping joints are welded except for the flanged connections at the relief valves, flow elements, and safety injection pumps.

Weld connections for pipes sized 2-1/2 inches and larger are butt welded. Note that the 2-inch end connections for the HHSI flow venturis are butt welded. Reducing tees are normally used when 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 inch through 2-inch, are attached to the header by means of full penetration welds, using pre-engineered integrally reinforced branch connections.

Minimum wall thicknesses are determined by the USASI Code formula found in the ANSI B31.7, 1969, Code for Pressure Piping. This minimum thickness is increased to allow for a manufacturing tolerance of 12.5% on the nominal wall in addition to an allowance of up to 8% for bending. Purchased pipes and fittings have a specified nominal wall thickness that is no less than the sum of that required for pressure containment, pipe bending, mechanical strength, and manufacturing tolerance.

Heat tracing is installed on all piping, valves, flanges, and instrumentation lines normally carrying the nominal 8 weight percent concentrated boric acid solution. The heat tracing system is designed in accordance with the following criteria:

1. One hundred percent redundant and separate heat tracing systems are provided.
2. Each heat tracing system is designed to maintain the fluid temperature ≥115°F with an ambient air temperature of 40°F.
3. Each redundant heat tracing system is supplied from a separate bus capable of being connected to the redundant emergency diesel generators.
4. Only one heat tracing system is energized at a time. Should the energized system fail, the other redundant system will be energized. Low temperature resulting from failure of the energized heat tracing system and loss of power to the redundant system will be annunciated at the individual heat tracing annunciator cabinets.

Heat tracing (redundant, Category I) is also used to prevent the freezing of the RWST level transmitters.

With regard to RWST vent freezing, the screen on the vent of the RWST is made of 1/4-inch mesh with 0.047-inch diameter stainless steel wire. These large openings should preclude the possibility of blocking the vent due to freezing. In addition, the vent faces downward where the

screen is protected from direct rain or snow impingement and the tank level does not usually cycle, as it is kept full and normally used only for refueling; therefore, very little air actually passes through the vent. The tank contents are maintained cold, which further reduces the moisture passing the screen.

6.3.2.1.6 LHSI Strainer Assembly

The LHSI strainer assembly provides filtered borated water to both LHSI pumps during recirculation mode. The strainer assembly consists of a number of modules which channel water to the pump suction. Modules are connected to each other by flexible metal seals. Seal closure frames with Metex seals are installed over the existing flexible metal seals. The seal closure frame assemblies form the seal between adjacent strainer modules. Each module contains a number of fins which filter the water flowing into the modules. Each fin contains a number of holes 0.0625-inch (nominal) in diameter. Perforations on the strainer fins prevent particles larger than 0.06875-inch (0.0625-inch plus 10 percent) from entering the LHSI System. The total perforation area is large enough to allow sufficient flow to the suctions of the LHSI pumps to meet NPSH requirements. In addition, particles larger than 0.06875 inches were evaluated in response to gaps identified in the strainer assembly. As part of the evaluation, it was assumed that 1% of the total generated particles between 0.06875 inches (0.0625 inches plus 10 percent) and 0.1375 inches (0.125 inches plus 10 percent) would pass through the strainer. It was determined that these particles would not impact the performance of downstream components.

The LHSI strainer assembly consists of two trains which traverse along the containment wall on both sides of the sump, on top of the RS strainer assembly. Each suction opening is connected to the modules via the strainer header. The strainer header is connected to each suction opening by a flanged transition adapter. The OD of the new strainer header is machine cut and slip-fit into the pump suction inlet ensuring that the gaps between the header and the pump suction inlet do not exceed 0.0625 inches.

The strainer assembly is designed and fabricated to the requirements of ASME Section III, Subsection NF, Class 3. All material used in the construction of the strainer assembly is austenitic stainless steel.

The strainer assembly is capable of withstanding the full debris loading in conjunction with all design basis conditions without collapse or structural damage.

The design of the LHSI strainer assembly is similar to the design of the RS strainer assembly. Refer to Section 6.2.2.2 for further information.

6.3.2.2 System Operation

The operation of the ECCS following a LOCA can be divided into two distinct modes:

1. The injection mode in which any reactivity increase following the postulated accident is terminated, initial cooling of the core is accomplished, and coolant lost from the primary system is replenished.
2. The recirculation mode in which long-term core cooling is provided during the accident recovery period.

Figure 6.3-5 provides a graphical representation of the events following postulated accidents. This sequence is somewhat similar for both large and small breaks, the principal difference being one of time.

The assumed single failure is the failure of an onsite diesel generator to start upon receipt of the safety injection signal. As this failure may eliminate a maximum of two of the three charging pumps and one of the LHSI pumps from service, it represents the worst single failure of the ECCS and provides added conservatism for the associated analyses.

For the ESF identified in Figure 6.3-5, the following auxiliaries are required:

1. Centrifugal charging pump.
2. LHSI pump.
3. Portions of Chemical and Volume Control System.
4. Portions of safety injection system.
5. Service water system.
6. Recirculation spray system.

6.3.2.2.1 Injection Mode After Loss of Primary Coolant

The principal mechanical components of the ECCS that provide core cooling immediately following a LOCA are the accumulators (one for each loop), the LHSI pumps, the centrifugal charging pumps, and the associated valves, tanks, and piping.

For a large pipe rupture, the RCS would be depressurized and voided of coolant rapidly, and a high flow rate of emergency coolant is required to quickly cover the exposed fuel rods and limit possible core damage. This high flow injection is provided by the passive accumulators, followed by the charging pumps and LHSI pumps discharging into the cold legs of the RCS. The LHSI pumps, passive accumulators, and charging pumps deliver directly into the cold legs of the RCS during the injection mode.

Pumps with high head capability are needed for small break protection when the RCS is not promptly depressurized below the accumulator discharge pressure. The centrifugal charging pumps fill this need.

During the injection mode, the charging pumps take suction from the RWST and deliver borated water to the cold legs of the RCS at the prevailing pressure. The discharge from the pumps initially sweeps the concentrated boric acid in the boron injection tank into the RCS.

The LHSI pumps also take suction from the RWST and deliver borated water to the cold legs of the RCS. These pumps begin to deliver water to the RCS only after the pressure has fallen below the pump shutoff head.

The injection mode of the emergency core cooling is initiated by the safety injection signal. This signal is actuated by any of the following:

1. Low-low pressurizer pressure.
2. High containment pressure.
3. High differential pressure between any two steam generators.
4. High steam flow coincident with either low-low T_{avg} or low steam line pressure.
5. Manual actuation.

Operation of the ECCS during the injection mode is completely automatic. The safety injection signal automatically initiates the following actions:

1. Starts the diesel generators.
2. Starts the nonoperating charging pumps and the LHSI 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 pump suction lines from the RWST.
    c. Closing the valves in the charging pump normal suction line from the volume control tank.
    d. Closing the valves in the boron injection tank recirculation lines.
    e. Opening the boron injection tank inlet and discharge line isolation valves.

Remotely operated valves for the injection mode that are under manual control (i.e., valves that normally are in their ready position and do not require a safety injection signal) have their positions indicated on a common portion of the control board. If a component is out of its proper position, its monitor light will so indicate on the control panel. At any time during operation when one of these valves is not in the ready position for injection, this condition is shown visually on the board in the main control room. Section 6.3.5.5 discusses additional position indication

features provided for the accumulator isolation valves and the LHSI pump suction valves to the RWSTs. The injection mode continues until the low level is reached in the RWST at which time the operator initiates the system alignment change to the recirculation mode.

The LHSI hot leg loop injection line (penetration 61) is provided with a normally closed piping connection that can be used with a portable pump to supply high pressure and/or low pressure water to the RCS if needed during a Beyond Design Basis Event or other accident.

6.3.2.2.2 Changeover From the Injection Mode to Recirculation After Loss of Primary Coolant

Water level indication and alarms in the containment sump and on the RWST provide ample warning to terminate the injection mode while the operating pumps still have adequate NPSH. Since the injection mode of operation following a LOCA is terminated before the RWST is completely emptied, all pipes are kept filled with water before recirculation is initiated.

Manual switchover from injection to recirculation can be accomplished by the operator. The time required to complete the operation is the time for the switchgear to function. Controls for ECCS components are grouped together on the main control board. The component position lights verify when the function of a given switch has been completed.

Automatic switchover from injection to recirculation mode is credited in the accident analysis. The operator can perform manual actions as a backup to the automatic functions.

The Train A and Train B automatic switchovers are initiated when actuation signals are generated by the two-of-four RWST low-low level protection logic and the safeguards protection logic (safety injection signal). The automatic switchover sequence is shown below for Train A valves with Train B valves in parentheses.

1. Valve 1863A (1863B) opens.
2. Valves 1885 A and C (1885B and D) close.
3. Valve 1860A (1860B) opens.
4. Valve 1862A (1862B) closes.

The key values for the RWST assumed in the containment analysis are presented in Table 6.2-2. The analysis values are conservative with respect to plant operation. The volume expended during early switchover is used for the NPSH calculation and the volume expended at time of late switchover is used for the depressurization calculation.

The latest transfer completion is based on the following:

1. Automatic switchover sequence starts (emergency procedures prevent manual action before the automatic setpoint is reached).
2. The level instrument error is giving its minimum level reading.

3. Maximum time to complete switchover equals 210 seconds.

The above assumptions result in a maximum of water expended before switchover is completed. This value is used for all depressurization calculations in Section 6.2.2 since later switchover results in less RWST water available for the quench sprays in maintaining the containment pressure subatmospheric.

Provisions are included in the system to permit online testing of the switchover sequence without affecting normal plant operation.

6.3.2.2.3 Recirculation Mode After Loss of Primary Coolant

After the injection operation, water collected in the containment sump is returned to the reactor coolant system by the low head or low head/high head recirculation flow paths. Cooling of sump water is provided by the RS subsystems (Section 6.2).

During the recirculation mode, the water passes through the strainer fins, modules, and headers in route to the LHSI pump suction intake. The water then enters a 12-inch pipe and flows to the LHSI pumps through a containment isolation valve, a check valve, and, in the case of Unit 1, a gate valve. The discharge from the LHSI pumps takes one of the following two paths: through containment isolation valves to 6-inch lines into the individual reactor coolant loops, or through 8-inch headers to the suction of the high head safety injection/charging pumps and then from the charging pumps through containment isolation valves to 2-inch lines into individual reactor coolant loops. From the loops, in both cases, the injected fluid flows through the RCS.

The high head mode of operation ensures flow in the event of a small rupture when the depressurization proceeds more slowly such that the RCS pressure is still in excess of the LHSI pumps at the onset of recirculation. The initial recirculation mode will provide recirculation flow to the cold legs of the RCS. After a period of cold-leg recirculation, the ECCS will be realigned to provide hot-leg recirculation flow in order to assure termination of boiling.

The redundant features of the recirculation loop include one pump in each of two trains outside the containment with crossover capability at the discharge of each pump. All heat removal is through the RS subsystem. There are no heat exchangers in the ECCS.

6.3.2.2.4 Steam-Line Rupture

Following a steam-line rupture, the ECCS is automatically actuated to deliver borated water from the boron injection tank to the RCS. The response of the ECCS following a steam-line break is similar to its response during the injection mode of operation following a LOCA.

The safety injection signal initiates identical actions as described for the injection mode of the LOCA, even though not all of these actions are required following a steam-line rupture, e.g., the LHSI pumps are not required since the RCS pressure remains above their shutoff head.

The delivery of the concentrated boric acid from the boron injection tank provides negative reactivity to counteract the increase in reactivity caused by the system cooldown. After all the concentrated boric acid is swept through the boron injection tank, the charging pumps continue to deliver borated water from the RWST, until enough water has been added to the RCS to make up for the shrinkage due to cooldown. After pressurizer water level has been restored, the injection is manually terminated. The sequence of events following a postulated steam-line break is described in Section 15.4.

6.3.2.2.5 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:

1. The remaining equipment has been demonstrated to be in operable condition, ready to function just before the initiation of the maintenance.
2. 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 safety injection system is to provide backup equipment so that maintenance is possible during operation without impairment of the safety function of the system. Routine servicing and maintenance of equipment of this type could be scheduled to be performed on-line or during periods of refueling and maintenance outages. Testing requirements are delineated in the Technical Specifications.

6.3.2.2.6 Pump Net Positive Suction Head

The ECCS is designed so that adequate NPSH is provided to system pumps. The calculation of available NPSH (NPSHA) in the recirculation mode considers the static head and suction line pressure drop, the vapor pressure of the liquid in the sump, and the containment pressure. This calculation ensures that the NPSHA meets the pump requirements.

The calculation of NPSHA is as follows:

$$NPSHA = (h)_{\text{containment pressure}} - (h)_{\text{vapor pressure}} \pm (h)_{\text{static head}} - (h)_{\text{loss}}$$

Adequate NPSH is shown to be available for all pumps as follows:

1. Low Head Safety Injection - The NPSH of the LHSI pumps is evaluated for both the injection and recirculation modes of operation for various LOCAs. Recirculation mode operation gives the limiting NPSH requirement.

A transient GOTHIC calculation is performed to demonstrate that the LHSI pumps have adequate NPSH during the recirculation phase following the postulated LOCA. The NPSH available (NPSHa) must be greater than the NPSH required at all times during the accident. The difference between available and required NPSH is margin, which can be used to overcome the strainer pressure drop. The calculation of NPSHa with GOTHIC follows the methodology outlined in Section 3.8 of topical report DOM-NAF-3-0.0-P-A (Reference 8). The DEPSG break provides the limiting LHSI pump NPSH results because it causes the largest energy release to the containment before recirculation mode transfer (RMT). Key analysis parameters are shown in Table 6.2-2.

The LHSI recirculation flow rate is conservatively assumed to be 4100 gpm based on one emergency bus as the most limiting single failure. This single failure leaves one LHSI and one HHSI pump, maximizes the pump suction friction loss, maximizes the LHSI pump required NPSH, and minimizes NPSHa. The analyses assume minimum heat sink surface area, minimum RS flow rates, minimum service water (SW) flow rate, maximum QS flow rate, maximum SI flow rates, and maximum containment temperature. These analyses include the SI RMT setpoint of 16.0% RWST level plus 2.5% uncertainty. The TS range for SW temperature (35-95°F) was analyzed in 10°F SW temperature steps. The minimum NPSHa of 15.43 ft occurs at RMT for a TS limit of 10.3 psia air partial pressure and a service water temperature of 45°F. As the transient progresses beyond RMT, the RS system removes heat from the containment sump and the sump water level continues to increase until the RWST and casing cooling tanks are empty. Because of the higher water level and colder sump temperatures, the long-term NPSHa is much greater than the minimum value reported.

The system operating time before RMT is less than 1700 seconds. Lower SW temperature brings down the containment pressure quickly but the sump temperature holds up. Lower SW temperature is limiting because the short operation period of RS before RMT provides little cooling of the sump liquid while still generating low containment pressures. There is a tradeoff between reduced spray temperature and reduced sump temperature. Once SW temperature drops below 75°F, the minimum NPSHa has little variability. As SW temperature decreases, both the containment pressure and sump temperature decrease and the effect of each change on NPSHa is offset.

While several cases along the air partial pressure limit generate about the same minimum NPSHa, the analysis at 10.3 psia and 45°F SW temperature is selected as the limiting case for showing transient behavior. Figure 6.3-6 (LHSI pump NPSHa and water level), Figure 6.3-7 (containment and LHSI pump suction vapor pressure), Figure 6.3-8 (containment vapor and liquid temperature), and Figure 6.3-9 (RS cooler heat rate) illustrate the performance of key variables for the LHSI pump NPSHa analysis at 45°F SW.

For the 4100-gpm maximum recirculation flow calculated for minimum ESF, NPSHA analyses for the LHSI pumps show that NPSH is always above the required NPSH (NPSHR) of 13.7 feet.

Transient conditions encountered during pump start or during a switchover of suction expose the suction impeller, as well as other parts of the pump, to short-term mechanical and hydraulic changes that will not alter long-term capability or reliability. Extensive shop testing of vertical can-type pumps has shown that when exposed to suction conditions well below steady-state requirements for periods of time in excess of 300 seconds, there is no measurable impact on performance.

The analysis of the transient NPSH available to the LHSI pumps indicates that, conservatively, the available NPSH approaches the pumps' required NPSH. This condition occurs at the point of switchover from the RWST to the containment sump. Prior to the time of switchover, the NPSH margin is ample to support steady-state flow conditions. At the moment of transfer, the flow delivered by the pump drops slightly and there is a reduction of NPSHA to a level slightly above the NPSHR. For approximately 60 seconds the margin between the NPSHA and NPSHR is less than 10%. During this time period a slight, less than 1%, reduction in pump head-flow capacity may occur. After this transient period, a sufficient NPSH margin is restored as the pumped fluid temperature decreases thus precluding cavitation and its impact on performance and reliability. Because of its low energy density and rotor rigidity, short-term transient suction conditions do not affect the pump's capability to perform its intended function. As a consequence, the conditions present during the analyzed condition at North Anna will not degrade the pumps either mechanically or hydraulically.

Should one of the reactor containment sump valves (MOV-1860A, B) fail to open during the recirculation phase, there will be no flow path for water to the LHSI pump on that side of the failure. This pump will cavitate as a result. The pump on the opposite side will have the normal NPSH and flow. Since only one LHSI pump is required to operate during the recirculation phase, plant safety criteria are not compromised.

2. Centrifugal Charging Pumps - The NPSH for the centrifugal charging pump is evaluated for both the injection and recirculation modes of operation following a LOCA. The end of the injection mode of operation gives the limiting NPSHA. The NPSHA is determined from atmospheric pressure, the elevation head, the 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.

6.3.2.2.7 Accumulator Motor-Operated Valve Protection

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 and automatic opening of that valve. Although the valve is normally open, it receives the safety injection actuation signal and will open automatically upon receipt of this signal should the valve be closed. This safety injection signal overrides any bypass feature that allows the valve to be closed for short times

during normal operation for test purposes. A further discussion of these interlocks is found in Section 7.6.6. Valve position indication is discussed in Section 6.3.5.5.

6.3.3 Performance Evaluation

6.3.3.1 Evaluation of Core Cooling Capability Following a LOCA

The following RCS pipe ruptures are met by the ECCS operating with minimum design equipment:

1. Large pipe break analysis.
2. Small line break analysis.
3. Recirculation cooling.

The flow delivered to the RCS by the ECCS as a function of reactor coolant pressure with the operation of minimum design equipment is shown in Section 15.4.

The performance characteristic utilized in the accident analysis for the LHSI pumps is developed from the lowest measured installed pump head curve data, reduced to account for pump degradation and instrument accuracy. The reduced performance curve is used to conservatively model LHSI delivered flow rates used as an input to the accident analyses. The performance characteristic for the CH pump is as described in Section 15.3. The injection curve utilized in the analysis accounts for the loss of injection water through the broken loop.

The pressurizer pressure at the time of the accident is assumed to be equal to 2280 psia. However, the RCS pressure following a large LOCA decreases extremely rapidly in the first 0.1 second. Although this phenomenon is calculated, the scale on the figures in Chapter 15 is such that it cannot be clearly shown.

6.3.3.1.1 Large Break Analysis

The large pipe break analysis is used to evaluate the initial core thermal transient for a spectrum of pipe ruptures from a break size of 0.5 ft^2 up to the double-ended rupture (DER) of the largest pipe in the RCS.

The injection flow from active components is required to control the cladding temperature subsequent to accumulator injection, complete reactor vessel refill, and eventually return the core to a subcooled state. The results indicate that the maximum cladding temperature attained at any point in the core is such that the limits on core behavior as specified in Section 15.4 are met.

6.3.3.1.2 Small Pipe Break Analysis

The small pipe break analysis is used to evaluate the initial core thermal transient for a spectrum of pipe ruptures up to and including a 0.5-ft^2 rupture. For breaks 3/8 inch or smaller, the charging system can maintain the pressurizer level at the RCS operating pressure and the ECCS would not be automatically actuated and is not required.

Subsequent to a small break LOCA, diversion of the suction of the charging pumps from the volume control tank to the RWST is ensured because the safety injection signal would cause two valves, in series, in the volume control tank discharge line to close, and two valves, in parallel, in the line leading to the RWST to open. Thus, a single failure could not prevent realignment of charging pump suction from the volume control tank to the RWST.

The results of the small pipe break analysis indicate that the limits on core behavior are adequately met, as shown in Section 15.3.

6.3.3.1.3 Recirculation Cooling

Core cooling during recirculation can be maintained by the flow from one charging pump and one LHSI pump. If RCS pressure is low, one LHSI pump is sufficient to maintain adequate recirculation flow and cooling. However, the RCS pressure at the start of recirculation could remain elevated for small breaks and thus the additional head supplied by one charging pump, operating in series with one LHSI pump, is needed to maintain adequate flow and cooling.

The procedure for establishing recirculation flow requires that the LHSI and charging pumps be utilized. Thus, the operator is not required to determine the size of the break. The LHSI pump provides recirculation flow to the RCS as well as providing suction flow to the charging pump. The charging pump delivers recirculation flow to the RCS. Thus, for all breaks, recirculation flow will be maintained by the flow from one charging pump and/or one LHSI pump.

Initially, part of the recirculated water would be evaporated at the prevailing reactor coolant pressure when recirculated to the reactor. The remainder of the recirculated water spills from the break. The pressure at the time recirculation is initiated is not dependent on the initiating break size for large breaks. Heat removal is accomplished from the recirculated sump water via operation of the RS subsystem (see Section 6.2).

6.3.3.1.4 Effect of LHSI Valve Misfunction on ECCS Flow

Adequate emergency core cooling system flow is provided by the LHSI system even in the event a malfunction should cause either valve (MOV 1860A or 1860B) in the lines from the containment sump to open while the containment is pressurized following a LOCA.

As described in Section 6.3, the two low head safety injection pumps take suction from the refueling water storage tank during the injection phase. The pipe supplying this water runs from the RWST to the safeguards area, then down to the LHSI pump suction. For purposes of this discussion, a simplified schematic of the LHSI system is shown in Figure 6.3-10.

Upon receipt of a safety injection signal, the LHSI pumps start and establish flow from the RWST. The flow splits at Point D on Figure 6.3-10 and enters each LHSI pump suction through the branch lines. Flow rates from the RWST are about 3000 gpm per pump, a total of approximately 6000 gpm through the common line upstream of Point D on Figure 6.3-10.

In the event that either MOV 1860A or 1860B in the lines from the containment sump open, water would flow from the sump into the affected branch line, or the check valve in the line from the containment sump would prevent flow from the affected branch line to the sump, depending on relative pressure heads contributed by the RWST and the containment sump. For this analysis, MOV 1860B is assumed to open. The intersection of this affected branch line and the LHSI pump suction line is identified as Point B on Figure 6.3-10. Points A and C at the LHSI pumps have also been designated on this figure.

Assuming that the total flow to the LHSI pump in that branch line (Point A) then came from the sump, the flow from the RWST in the common line (Point D) would be reduced to 3000 gpm. Flow to the other LHSI pump (Point C) would, of course, continue from the RWST after the sump valve MOV-1860B opened, providing that the driving head from the RWST was greater than the head through the open line from the containment sump. The pressures from the RWST and containment sump at Point B for this assumed condition have been calculated and are shown in Figure 6.3-11. As can be seen from the figure, the head from the RWST at Point B is always greater than that from the containment sump, thereby assuring that flow will not come from the sump into the affected branch. The RWST head and the check valve in the line from the containment sump assure that the flow to both LHSI pumps is maintained from the RWST at approximately 3000 gpm per pump.

An analysis has also been performed to determine the relative pressure heads contributed by the RWST and the containment sump at Point B assuming that the LHSI pump at Point A becomes inoperative. It was further assumed that there would be zero flow to the inoperative LHSI pump from either the RWST or the containment sump and that there would be 3000-gpm flow to the other LHSI pump (Point C) from the RWST. The results of this analysis are plotted on Figure 6.3-12. As can be noted from the figure, the pressure at Point B from the RWST would always be greater than the pressure from the containment sump, thereby confirming that 100% flow would be maintained from the RWST to the operative LHSI pump (Point C).

By identical reasoning, since the branch lines from Point D to the pumps are redundant, it is shown that RWST flow will be maintained to the other pump (Point A), should MOV-1860A inadvertently open during the postulated accident conditions.

6.3.3.1.5 Required Operating Status of Emergency Core Cooling System Components

The analyses of Sections 15.3 and 15.4 show that the 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 6.3-5.

The ECCS components are available whenever the coolant energy is high and the reactor is critical. During low temperature physics tests there is a negligible amount of stored energy in the coolant and low decay heat. Therefore, an accident comparable in severity to accidents occurring at operating conditions is not possible and ECCS components are not required.

ECCS actuation from two out of three low-low pressurizer pressure channels automatically becomes available when the reactor coolant system pressure is raised above the safety injection unblock pressure (2000 psig). Technical Specifications have been established governing the operability of ECCS components and associated limitations on reactor operation.

6.3.3.1.6 Range of Core Protection

Core protection is afforded with the minimum ESF equipment. The minimum ESF equipment is defined by consideration of the single-failure criteria as discussed in Section 3.1, Section 6.3.1.3, and Appendix 6A. The minimum design case ensures that the entire break spectrum is accounted for and core cooling design bases of Section 6.3.1 are met. Analyses are presented in Sections 15.3 and 15.4.

For large RCS ruptures, the accumulators and the active high head and low head pumping components serve to complete the core refill. One LHSI pump is required for long-term recirculation in conjunction with components of the auxiliary heat removal systems, which are required to transfer heat from the recirculation water.

The accumulators and the active high head and low head pumps provide necessary injection flow for breaks, between 0.5 and 0.785 ft^2. Long-term recirculation requires one LHSI pump in conjunction with one charging pump and components of the auxiliary heat removal systems that are required to transfer heat from the recirculation water.

If the break is small (6-inch equivalent diameter or less) the accumulators and one charging pump ensure adequate cooling during the injection mode. Long-term recirculation requires one LHSI pump in conjunction with one charging pump and components of the service water system that are required to transfer heat from the recirculation spray water. The LOCA discussions are presented in Sections 15.3 and 15.4.

6.3.3.2 **System Response**

The minimum active components are capable of delivering full-rated flow within the specified time interval after process parameters reach the setpoints for the safety injection signal. 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 safety injection signal. In analyses of system performance, delays in reaching the programmed trip points and in actuation of components are established on the basis that only emergency onsite power is available. The emergency power system is discussed in detail in Chapter 8.

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.

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.

Accumulator injection occurs immediately when RCS pressure has decreased below the operating pressure of the accumulator.

6.3.3.3 **Coolant Storage Reserves**

The RWST is sized to provide the borated water necessary for injection to the RCS and for depressurization of the containment within the 60-minute requirement. This storage volume is also sufficient to ensure that, after a RCS break, adequate water is available within containment to permit recirculation cooling flow to the core, and to meet the NPSH requirement of the LHSI pumps. Thus, adequate storage volume of emergency coolant for ECCS operation is provided.

6.3.3.4 **Boron Concentration**

When water in the RWST at its minimum boron concentration is mixed with the contents of the RCS, the resulting boron concentration ensures that the reactor will remain subcritical in the cold condition with all control rods except the most reactive rod cluster control assembly inserted into the core.

The boron concentrations of the accumulator and the RWST are below the solubility limit of boric acid at their respective temperatures.

The RWST is maintained at a boron concentration between 2600 and 2800 ppm, well below the concentration that would precipitate at a temperature of 32°F. The RWST is an insulated tank and is capable of being recirculated. The normal temperature maintained in this tank is approximately 45°F. At this temperature and because of the recirculation, no precipitation will occur.

Figure 6.3-13 is a plot of the saturation temperature versus boric acid concentration.

The heating and recirculation provided is adequate to ensure that precipitation will not occur in the boron injection tank and piping.

6.3.3.5 **Emergency Core Cooling System 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 not only cause 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 that isolate the ECCS from the RCS are installed as close as possible to the reactor coolant piping.

Reactor pressure maintains a relatively uniform back pressure in all injection lines for a small break 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 contents of the accumulator associated with the broken leg are discharged from the break.

As stated in Section 6A.3, a failure of the shaft seal on one of the high head safety injection pumps (charging pumps) would result in a leak rate of less than 50 gpm. This leakage is directed to the auxiliary building sump, which is equipped with two 50-gpm capacity sump pumps and redundant high level sump alarms as stated in Appendix 6A.

The auxiliary building sump pump will start and will run continuously or on a very long cycle. A situation that results in long-term running is an indication of an abnormal situation. The control room operator has level indication and an alarm for the tank to which this pump discharges (high level waste tank). An increasing tank level and/or alarm coupled with a continuously running auxiliary building sump pump would cause the operator to initiate an inspection of the auxiliary building. This inspection would start with active components in systems under high pressure, as they present the greatest leak potential. Thus, the running charging pump would be among the first items checked. It is estimated that this procedure would take no more than 2 hours. Fluid from the auxiliary building sump will go to the waste disposal system, which has a storage capacity of 20,000 gallons.

6.3.3.6 **External Recirculation Loop**

The ECCS recirculation loop piping and components external to containment are surrounded by shielding. This shielding is designed to permit access for maintenance to a component such as a pump while the redundant component is recirculating sump fluid.

Any releases from pressure relieving devices located in portions of the ECCS outside the containment, which might contain radioactivity, are collected and discharged to the waste disposal system.

During recirculation, significant margin exists between the design and operating conditions (in terms of pressure and temperature) of the ECCS components. Since redundant flow paths are provided during recirculation, a leaking component outside the containment in one of the flow paths may be isolated. This action curtails any further leakage and renders the component available for corrective maintenance. Maximum potential leakage from components during normal operation is given in Table 6.3-6.

Analyses indicate that the offsite dose resulting from recirculation loop leakage is less than allowed by 10 CFR 50.67, assuming a maximum leakage as discussed in Appendix 6A. Leakage detection exterior to the containment is achieved through the use of sump level detection. An alarm in the control room indicates that water has accumulated in the sump. Valving is provided to permit the operator to isolate individually the LHSI pumps.

The injection line piping is arranged so that a water seal is provided upstream of the valves located outside the containment, and this piping can be isolated from the containment. Thus, outleakage of air from the containment to the RWST, and hence to the atmosphere, is prevented.

6.3.3.7 **Shared Components of the ECCS**

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:

1. Components of the ECCS that perform no other function are:
   a. One accumulator for each loop that discharges borated water into its respective cold leg of the reactor coolant loop piping.
   b. Two LHSI pumps that supply borated water for core cooling to the RCS.
   c. One boron injection tank.
   d. Associated piping, valves, and instrumentation.
2. Components that also have a normal operating function are as follows:
   a. The centrifugal charging pumps: these pumps are normally aligned for charging service. As a part of the Chemical and Volume Control System, the normal operation of these pumps is discussed in Section 9.3.4.
   b. The RWST: this tank is used to fill the refueling cavity and canal for refueling operations. However, during all other plant operating periods, it is aligned to the suction of the LHSI pumps. The charging pumps are automatically aligned to the RWST upon receipt of the safety injection signal or on VCT low-low level.

An evaluation of all components required for operation of the ECCS demonstrates that either:

1. The component is not shared with other systems, or:
2. If the component is shared with other systems, it is aligned during normal plant operation to perform its accident function; or if not aligned to its accident function, two valves in parallel are provided to align the system for injection, and two valves in series are provided to isolate portions of the system not utilized for injection. These valves are automatically actuated by the safety injection signal.

Table 6.3-7 indicates the alignment of components during normal operation, and the realignment required to perform the accident function.

Other systems that operate in conjunction with the ECCS are as follows:

1. The RS subsystem is used to remove heat from the containment following a LOCA. In performing this duty, it cools the water in the containment sump. By recirculating this water from the sump into the core via the LHSI pumps and the high head centrifugal charging pumps, the ECCS removes the core residual heat.

   On Unit 1, the ORS pumps are capable of recirculating water from the sump to the discharge of the LHSI pumps, providing an alternate source of water for the LHSI pumps 24 hours after the accident.

2. The electrical systems provide normal and emergency power sources for the ECCS.
3. The engineered safety features actuation system generates the initiation signal for emergency core cooling.
4. The auxiliary feedwater system supplies feedwater to the steam generators.

6.3.3.8 **Evaluation of Shutdown Reactivity Capability Following an Abnormal Release of Steam from the Main Steam System**

Analyses have been performed to ensure that the core limitations defined in Sections 15.2, 15.3, and 15.4 are met following a steam-line rupture or a single-active failure in the main steam system.

6.3.3.8.1 Main Steam System Single Active Failure

Analyses of reactor behavior following any single active failure in the main steam system that 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, thereby resulting 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 departure from nucleate boiling in the core. Thus, the ECCS provides adequate protection for this accident.

6.3.3.8.2 Steam-Line Rupture

This accident is discussed in detail in Sections 6.2.1.3.1.2 and 15.4. The limiting steam-line rupture is a complete line severance.

The results of the analysis in Sections 6.2.1.3.1.2 and 15.4 indicate that the design-basis criteria are met. Thus, the ECCS adequately fulfills its shutdown reactivity addition function.

A technical specification ensures the availability of the concentrated boric acid solution in the boron injection tank that provides the shutdown reactivity. Since each channel of tank heaters and each channel of line heat tracing is sized to maintain the temperature of the boric acid in the

tank and lines, respectively, well above solidification temperature, continued operation with only one of the duplicate systems is acceptable.

6.3.3.9 Evaluation of Loss of Offsite Power

The emergency power supply system supplies power to ECCS components in the event that all sources of offsite power become unavailable.

The supply of emergency power to the ECCS components is arranged such that as a minimum, one charging pump and one LHSI pump together with the associated valves will automatically receive adequate power in the event that a loss of offsite power occurs simultaneously with any one of the accidents described in Section 6.3.1, even with a single failure in the emergency power system, such as the failure of an emergency diesel to start.

6.3.3.10 Evaluation of the Capability to Withstand Postaccident Environment

A comprehensive testing program has been undertaken to demonstrate that ECCS components and associated instrumentation and electrical equipment that are located inside the containment will operate for the time period required in the combined post-LOCA conditions of temperature, pressure, humidity, radiation, and chemistry (Reference 1).

Components such as remote motor-operated valves and flow and pressure transmitters have been shown capable of operating for the required postaccident periods, when exposed to post-loss-of-coolant environmental conditions.

All motor-operated valves in the ECCS system required to operate following a LOCA located in an area of potential flooding have their motor operators above Elevation 267 ft. 6 in. The valves are connected to their motor operators by valve stem extensions such that no motor operators will be submerged.

A further review of the equipment, instrumentation, terminations, and cable routing located within the containment has ensured that no required safety-related systems or equipment will be adversely affected by flooding following the postulated LOCA. Equipment that would be subject to flooding falls into one of the following categories:

1. Safety-related equipment that is not required to function following a LOCA:
   a. The instrumentation for the primary coolant loop flow that is energized during normal operation. Protection for each instrument loop against adverse effects of flooding is provided by a series of Class 1E protective devices consisting of three breakers, three fuses, and two power supplies that have current limiting capabilities.
   b. The accumulator discharge valves that are de-energized during normal operation.

2. Nonsafety-related equipment that is not connected to an electrical source that supplies safety-related equipment, as follows: containment sump pumps, shroud cooling coils, temperature instrumentation, reactor coolant pumps (RCP) temperature instrumentation, and containment instrument air dryers.

3. Nonsafety-related equipment that is supplied from an electrical source that also supplies safety-related equipment:

   a. The containment recirculation fans that are de-energized directly by a containment depressurization actuation (CDA) signal.

   b. Protection of the nonsafety-related equipment against adverse effect of flooding is provided by redundant limiting and overcurrent protection devices to ensure that the loss of the nonsafety-related equipment will not result in the loss of the safety-related equipment source in the following manners.

      (1) Protection is provided by two Class 1E breakers in series with both thermal and magnetic trip settings for the containment instrument air compressors and Class 1E breaker for the following equipment: pressurizer relief tank valve, accumulator fill and drain line valves, RCP leakoff seal isolation valves, charging line stop valve, and auxiliary spray valve.

      (2) Protection for each instrument loop is provided by the following protective devices: one Class 1E breaker in series with a current limiting power supply containing both input and output fuses for the following equipment: excess letdown valve control, residual heat removal (RHR) system temperature and flow control.

      (3) Protection for each instrument loop is provided by a series of Class 1E protective devices consisting of three breakers, three fuses, and two power supplies that have current limiting capabilities for the following equipment: accumulator pressure, hot-leg safety injection flow, RCP seal and cooling water flow, reactor vessel flange leakoff temperature, excess letdown temperature, RHR heat exchanger bypass flow, neutron shield tank flow, primary drain transfer tank level, RHR system flow control, and containment air dryer pressure.

A review of the containment electrical penetrations for the above equipment was conducted to ensure that the integrity will be maintained for the postulated flooding conditions. The following results correlate numerically to the above categories:

1.

   a. Maximum capable current during a submerged condition is well within the normal operating range of the penetrations.

   b. De-energized condition creates no areas of concern.

2. The containment sump pumps and the instrument air dryers were analyzed for maximum fault current and the results were minimal compared to the actual test values of the penetrations. The temperature instrumentation is powered from the plant computer power supply, which is a low voltage and current source. Fault currents were verified to be well within the normal operating range of the penetrations.

3.

   a. Equipment automatically de-energized directly by a safety injection signal creates no areas of concern.

   b. The maximum fault currents were determined to be within the operating range of the penetrations.

   c. Maximum capable currents during a submerged condition are well within the normal operating range of penetrations.

Written emergency operating procedures require switchover from cold-leg to hot-leg recirculation 5 hours after a LOCA. Following the initial switchover, hot- and cold-leg injections will be alternated at intervals not to exceed 5 hours. This timing should preclude the possibility of exceeding the boron solubility limits. For additional information, refer to Westinghouse letter CLC-NS-309 (Reference 3).

Redundancy in equipment is provided by both charging and LHSI pumps during hot-leg recirculation. Each LHSI pump delivers to three hot legs during hot-leg recirculation. In addition, during hot-leg recirculation, the centrifugal charging pumps deliver to the hot legs. Therefore, assuming the worst single failure (train failure), one LHSI pump and one centrifugal charging pump are available for hot-leg recirculation.

Boric acid buildup considerations during long-term cooling have been addressed in the letter from C. Caso of Westinghouse Electric Corporation to T. Novak of NRC dated April 1, 1975. During cold-leg injection for a cold-leg pipe break the analysis shows that boric acid concentrations within the reactor vessel and core region remain at acceptable levels up to the time of the initiation of hot-leg injection. The safety injection system alignment during the hot-leg recirculation phase provides injection to the RCS hot legs from the low head pumps and the charging pumps. During the hot-leg recirculation phase for a cold-leg break, flow through the core and out the break will act as a mechanism to preclude any further buildup of boric acid concentration in the core region.

6.3.3.11 Evaluation of System Parameters

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 NPSH), 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 the minimum required level of functional performance.

6.3.3.12 Minimum Containment Pressure Analysis (Westinghouse Evaluation Model)

The containment backpressure used for the 10 CFR 50.46 ECCS analysis of Westinghouse Robust Fuel Assembly 2 (RFA-2) fuel is presented in Section 15.4.1.2. The containment backpressure is calculated using the methods and assumptions described in *Westinghouse Emergency Core Cooling System Evaluation Model - Summary*, WCAP-8339, Appendix A (Reference 4).

The following information is HISTORICAL and is not intended or expected to be updated for the life of the plant.

The containment initial conditions of 90°F and 9.6 psia are representatively low values anticipated during normal full-power operation. The initial relative humidity is conservatively assumed to be 98.8%. Refer to Table 15.4-2 for current analysis data.

A summary listing of the input parameters used in the containment backpressure analysis is provided in Table 15.4-2. The net free containment volume is calculated based on containment dimensions, the volume of interior concrete, and the volume of installed equipment. The containment initial conditions of 84.5°F and 10.0 psia are representatively low values anticipated during normal full-power operation. The containment quench spray actuation time of 38 seconds, used for the design-basis LOCA, is the conservative minimum time based on the assumption that offsite power is available. The RWST temperature of 38°F is less than the minimum allowed by the Technical Specifications. The quench spray flow rate of 2500 gpm per pump is conservative. The RS subsystem and service water temperature are not considered in this analysis since the large break LOCA transient ends prior to the initiation of the RS subsystem (See Figure 15.4-1).

The detail listing of the actual containment heat sinks is provided in Table 6.3-8. A summary listing of the assumed heat sinks used in the input for the containment backpressure analysis is provided in Table 15.4-3.

The containment pressure response is provided in Figure 15.4-13. The condensing heat transfer coefficients used for heat transfer to the containment structures are given in Figure 6.3-15 for the limiting PCT case. The energy releases used in the containment backpressure calculation for the limiting PCT case are presented in Figure 6.3-17.

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. Performance tests of the components were performed in the manufacturer's shop. An initial flow test was performed to demonstrate the proper functioning of all of the components.

6.3.4.1 **Quality Control**

Tests and inspections are carried out during fabrication of each of the ECCS components. These tests are conducted and documented in accordance with the quality assurance program discussed in Chapter 17.

The following information is HISTORICAL and is not intended or expected to be updated for the life of the plant.

6.3.4.2 **Preoperational System Tests**

These tests were intended to evaluate the hydraulic and mechanical performance of the passive and active components involved in the injection mode by demonstrating that they had been installed and adjusted so they would operate in accordance with the design intent. These tests were divided into three individual sections that were performed as plant conditions allowed without compromising the integrity of the tests.

One of these individual sections consisted of system actuation tests that verified: the operability of all emergency core cooling system valves initiated by the safety injection signal, the containment isolation Phase A signal (CIA), and the containment isolation phase B signal (CIB); 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 was the accumulator injection test. The objective of this section was to check the accumulator injection line to verify that the lines were free from obstructions and that the accumulator check valves operated correctly. The test objectives were met by a low pressure blowdown of each accumulator. The test was performed with the reactor head and internals removed.

The last of the individual sections consisted of operational tests of the major pumps (i.e., the charging pumps and the LHSI pumps). The purpose of these tests was to evaluate the hydraulic and mechanical performance of the pumps delivering through the flow paths required for emergency core cooling. These tests were divided into two parts: pump operation under recirculation conditions and pump operation at full flow conditions.

The criteria for testing the ECCS pumps during preoperational testing were established to ensure that the minimum required ECCS flow rates are obtainable and maximum flow rates, based on NPSH considerations, are not exceeded.

The charging and low head injection pump flows for the original accident analyses were based upon reducing the original performance curves by 5%. By reducing the head versus flow pump curves, conservatism in the ECCS injection flow calculation was gained. The conservative ECCS injection flow rates were then used as an input to the original accident analyses. Preoperational tests were performed to show that the actual pumped flow rates were greater than those used in that accident analysis.

The following information is HISTORICAL and is not intended or expected to be updated for the life of the plant.

During preoperational testing, the pumps were operated singularly and drew suction from a full RWST. Each charging pump was required to deliver a minimum total flow of at least 627 gpm when aligned for safety injection. Flow distribution during this test had to indicate that a minimum flow of 580 gpm was directed to the cold legs and a minimum flow of 47 gpm was directed to the reactor coolant pumps via the seal injection lines. The charging pump miniflow line was isolated during this test. Each LHSI pump had to deliver a minimum flow of 3770 gpm to the cold leg when aligned for safety injection. The LHSI pump miniflow line was open during this test. Miniflow was not included in the LHSI pump minimum required flow rate of 3770 gpm.

The basis for establishing the minimum level to prevent vortexing during the test was the minimum water level in the sump while the pumps are operating, which is approximately 219 feet in elevation or about 4 ft. 5 in. above the inlet of the LHSI pumps, which are taking suction from the containment. This level corresponds to the minimum level in the containment during the ECCS recirculation phase following a LOCA. Vortexing will not occur because of the dampening effect of the circular pump screens as shown in Figure 6.3-14. Without the sump screens in place, vortexing would not occur unless the water level fell below Elevation 218 ft. 1 in., with the pump at runout flow, based on data in Reference 5.

The maximum flow during the test is determined by the line resistance of the temporary test piping. This flow rate is at about the operating point of 3166 gpm but not less than 3000 gpm, which is the nominal design flow.

With the water in the sump kept below 120°F (for personnel protection during the test), the LHSI pumps have adequate NPSHA with any water level at runout flow.

The test configuration for the sump is shown in Figure 6.3-14.

The temporary dike was filled above the level required to prevent the formation of vortexing without installation of the sump screen. However, the determining factors in establishing the level were the volume of water required to fill the test loop and the heating of the water as a result of the running of the pump. The sumps were filled to approximately a 220-foot elevation prior to the tests, so that sufficient water was available to perform the tests while keeping the water temperature below 120°F.

The information in this section is applicable only to the testing of the actual containment sump.

Scale model tests were also performed by Alden Research Laboratories and delineated in Appendix 3A, Section 3A.40.

The following information is HISTORICAL and is not intended or expected to be updated for the life of the plant.

Where applicable, the system resistance was established by measuring the flow in each piping branch, as the pump delivered from the RWST to the open reactor vessel, and adjustments were made where necessary so that no one branch had an unacceptably low or high resistance. During this flow test, the system was set up or checked to ensure that there was 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 were compared with the minimum acceptable flows as determined for the accident analyses reported in Sections 15.3 and 15.4.

The systems were accepted only after demonstration of proper actuation of all components and after demonstration of flow delivery of all components within design requirements.

6.3.4.3 **Start-up Testing**

For the purpose of conducting start-up testing prior to evaluation of LHSI pump reliability, an engineering review of the accident analyses given in Chapter 15 was conducted to ascertain whether the consequences of any of the applicable accidents could be more severe when postulated to occur with the LHSI pumps assumed inoperable, under the following conditions:

1. None of the fuel in the core has produced any sensible heat.
2. The unit is in hot standby with $k_{eff} \leq 0.98$ and $T_{avg} \leq 550°F$.

The conclusion of this review was that the consequences of the applicable accidents, under the conditions postulated, are not made more severe than those given in Chapter 15. The LHSI system is only necessary to provide cooling of residual decay heat after a postulated rupture of the reactor coolant system. Since none of the fuel in the core would have produced any sensible heat, and since the core would not become critical and no sensible heat would be produced as a result of any accident involving a rupture of the reactor coolant system when postulated to occur at hot standby conditions, there would be no significant residual decay heat produced, and hence no need for the LHSI system to operate under the assumed conditions.

LHSI pump reliability evaluations were conducted in accordance with NRC staff requirements. The long-term test involved a successful 23-day, continuous-operation, LHSI pump operation at a nominal flow 4000 gpm. In order to provide additional assurance of core cooling (in the event of a postulated accident) prior to completion of the required long-term pump testing, cross-connect piping was installed between the ORS and LHSI systems on Unit 1. These cross connects are described in Sections 6.2 and 6.3. See also Figure 6.1-1.

6.3.4.4 **Periodic System Tests**

Periodically, ECCS system tests are performed to verify that the system performance will meet the requirements specified in 10 CFR 50.46.

The maximum flow criteria for the charging and LHSI pumps are based upon NPSH considerations. In order to maintain proper NPSH margin, the maximum flow (or runout) is limited to an acceptable pump value. For the charging pumps the maximum continuous runout

flow rate per pump is 675 gpm. To ensure the flow rate limit is observed during design basis conditions, the flow rate is set by throttling manual globe valves in each individual injection line to limit total pumped flow to less than the maximum flow test acceptance criteria for full flow testing. The charging pump full flow test configuration simulates injection mode operation. However, the charging pump maximum flow test acceptance criterion also bounds the maximum charging pump flow during SI recirculation mode operation which includes the LHSI pump boost. Additionally, minimum HHSI flow acceptance criteria for charging pump full flow testing verifies that minimum delivered HHSI flow satisfies accident analysis assumptions.

Acceptance criteria for LHSI pump full flow testing, which simulates the SI injection mode system configuration, verifies the LHSI minimum delivered flow assumed in the accident analysis. For the LHSI pumps, the anticipated maximum flow per pump is less than 4200 gpm during the injection mode of operation following a LOCA and less than 4050 gpm during the recirculation mode of operation following a LOCA. This flow rate is limited by the actual line resistance due to the piping system (pipe, valves, etc.).

The maximum LHSI pump flow rate of 4500 gpm in Table 6.3-1 is a pump design and shop test performance point specified by Westinghouse to cover the application of this pump to a number of installations and does not necessarily reflect an actual operating point in the North Anna design. This point is selected to exceed the maximum anticipated operating condition for all applications to ensure that certified performance for all parameters required, including NPSH, is available for the full performance range.

6.3.4.5 **Periodic Component Testing**

Routine periodic testing of the ECCS components and all necessary support systems at power is performed. The associated pumps of the ECCS are tested as required by the Technical Specifications. Valves that operate after a LOCA are operated through a complete cycle, and pumps are operated individually in this test on their recirculation lines except the charging pumps, which 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 valves just downstream of the stop valve and check valve, respectively. Flow through the test line can be observed on instruments and the opening and closing of the discharge line stop valve can be sensed on this instrumentation.

Provisions are made to permit 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. Periodic ECCS component testing requirements are detailed

in the Technical Specifications. Inservice inspection provides further confirmation that no significant deterioration is occurring in the ECCS fluid boundary.

The design ensures that the following testing can be performed:

1. Active components may be tested periodically for operability (e.g., pumps on recirculation, certain valves, etc.).
2. An integrated system actuation test can be performed when the plant is cooled down and the residual heat removal system (RHRS) is in operation. The ECCS is then arranged so that no flow is introduced into the RCS for this test. Details of the testing of the sensors and logic circuits associated with the generation of a safety injection signal together with the application of this signal to the operation of each active component are given in Sections 7.2 and 7.3.
3. An initial flow test of the full operational sequence can be performed. See Section 6.3.4.2.

The design features that assure this test capability are specifically:

1. Power sources are provided to permit individual actuation of each active component of the ECCS.
2. The LHSI pumps can be tested periodically during plant operation using the minimum flow recirculation lines provided.
3. The centrifugal charging pumps are either normally in use for charging service or can be tested periodically on recirculation.
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. A flow indicator is provided in the charging safety injection pump header, and in the LHSI pump headers. Pressure instrumentation is also provided in these lines.
7. An integrated system test can be performed when the plant is cooled down and the RHRS is in operation. This test does not introduce flow into the RCS but does demonstrate the operation of the valves, pump circuit breakers, and automatic circuitry including diesel starting and the automatic loading of ECCS components on the diesels (by simultaneously simulating a loss of offsite power to the emergency electrical buses).

6.3.5 Instrumentation Application

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 main control room.

6.3.5.1 **Temperature Indication**

6.3.5.1.1 Boron Injection Tank Temperature

Duplicate temperature control channels are provided for the boron injection tank electric strip heaters. Both actuate high and low temperature alarms and both channels provide local temperature indication.

6.3.5.1.2 Heat Tracing Temperature

Separate thermostatic controls are provided for each section of the heat tracing in the boron recirculation loop to maintain the temperature within the specified range. High and low temperature alarms are provided to warn of failure to maintain the temperature within the control band.

6.3.5.2 **Pressure Indication**

6.3.5.2.1 Accumulator Pressure

Duplicate pressure channels are installed on each accumulator. Pressure indication in the main control room and high and low pressure alarms are provided by each channel.

6.3.5.2.2 Test Line Pressure

Local pressure test connections used to check for proper seating of the accumulator check valves between the injection lines and the RCS are installed on the leakage test lines.

6.3.5.2.3 LHSI Pump Discharge Pressure

LHSI pump discharge pressure for each pump is indicated locally.

6.3.5.3 **Flow Indication**

6.3.5.3.1 Boric Acid Recirculation Flow

Boric acid recirculation flow through the boron injection tank is indicated locally.

6.3.5.3.2 Charging Pump Injection Flow

Injection flow through the common header and each branch line to the reactor cold legs is indicated in the main control room. In addition, HHSI flow venturis provide means for local flow measurement of the HHSI branch lines to the RCS cold legs during unit outages.

6.3.5.3.3 Charging Pump Recirculation Flow

Recirculation flow through the common header and each branch line to the reactor hot legs is indicated in the main control room.

6.3.5.3.4 LHSI Pump Injection Flow

Flow through each LHSI pump discharge header is indicated in the main control room.

6.3.5.3.5 Test Line Flow

Local indication of the leakage test line flow is provided to check for proper seating of the accumulator check valves between the injection lines and the RCS.

6.3.5.3.6 LHSI Pump Minimum Flow

A flow indicator is installed in the LHSI pump minimum flow line.

6.3.5.4 **Level Indication**

6.3.5.4.1 Refueling Water Storage Tank Level

See Section 6.2.2.8.

6.3.5.4.2 Accumulator Water Level

Duplicate water level channels are provided for each accumulator. Both channels provide indication in the control room and actuate high and low water level alarms.

6.3.5.4.3 Containment Sump Water Level

See Section 6.2.2.8.

6.3.5.5 **Valve Position Indication**

Valve positions are indicated on the control board by lights. When the valve is open, a red light is lit; when the valve is closed, a green light is lit. Thus, a highly visible indication is available to the operator.

6.3.5.5.1 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. When the valve is in an intermediate position, both red and green lights are lit. The accumulator valves are also provided with a single red indicating light, located near the A and B valve switches. This light is powered from the vital bus, and actuated by valve stem switches when all three of the accumulator isolation valves are fully open. A common annunciator is also activated whenever any valve is fully closed. Three separate annunciator points are activated by valve position limit switches whenever the appropriate valve is not fully open with the system at pressure (the pressure at which the safety injection block is removed).

6.3.5.5.2 Low Head Safety Injection Pump RWST Suction Valve

The control and indications provided for this valve are identical to those provided for the accumulator isolation valves, with the exceptions that a safety injection actuation signal is not applied to this valve and automatic opening above a preset pressure is not provided.

6.3 REFERENCES

1. E. G. Igne and J. Locante, *Environment Testing of Engineered Safety Features Related Equipment (NSSS - Standard Scope)*, WCAP-7744, Vol. 1, August 1971.
2. *Hydraulic Model Studies—Low Head Safety Injection Pump and Recirculation Spray Pump*, LHL-716, January 1978.
3. Westinghouse letter CLC-NS-309, transmitted to T. Novak of the NRC Staff, April 1, 1975.
4. *Westinghouse Emergency Core Cooling System, Evaluation Model - Summary*, WCAP-8339, June 1974.
5. J. L. Gordon, *Vortices at Intake*, *Water Power*, April 1970.
6. Generic Letter 95-07, *Pressure Locking and Thermal Binding of Safety-Related Power-Operated Gate Valves*, Virginia Power letters to the NRC, Serial Number 95-566A, February 7, 1996; and Serial Number 96-315, July 3, 1996.
7. Information Notice 88-23, *Potential for Gas Binding of High Pressure Safety Injection Pumps During a Loss-of-Coolant Accident*.
8. Topical Report DOM-NAF-3-0.0-P-A, *GOTHIC Methodology For Analyzing the Response to Postulated Pipe Ruptures Inside Containment,* September 2006.
9. NRC Generic Letter GL 2004-02: *Potential Impact of Debris Blockage on Emergency Recirculation During Design Basis Accidents at Pressurized Water Reactors*, dated September 13, 2004.
10. Letter from Dominion Resources Inc. to the NRC, dated September 1, 2005, Serial No. 05-212, Response to NRC Generic Letter 2004-02.
11. Nuclear Energy Institute (NEI) Document NEI 04-07, *Pressurized Water Reactor Sump Performance Evaluation Methodology*, dated December 2004.
12. Safety Evaluation by the Office of Nuclear Reactor Regulation Related to NRC Generic Letter 2004-02, *Nuclear Energy Institute Guidance Report Pressurized Water Reactor Sump Performance Evaluation Methodology.*
13. Westinghouse Document WCAP-16406-P, Revision 1, *Downstream Wear Evaluation Methodology for Containment Sump Screens in Pressurized Water Reactors*.
14. Westinghouse Document WCAP-16793-NP, Revision 0, *Evaluation of Long-Term Cooling Considering Particulate, Fibrous and Chemical Debris in the Recirculating Fluid*.

6.3 REFERENCE DRAWINGS

The list of Station Drawings below is provided for information only. The referenced drawings are not part of the UFSAR. This is not intended to be a complete listing of all Station Drawings referenced from this section of the UFSAR. The contents of Station Drawings are controlled by station procedure.

	Drawing Number	Description
1.	11715-FM-096A	Flow/Valve Operating Numbers Diagram: Safety Injection System, Unit 1
	12050-FM-096A	Flow/Valve Operating Numbers Diagram: Safety Injection System, Unit 2
2.	11715-FM-096B	Flow/Valve Operating Numbers Diagram: Safety Injection System, Unit 1
	12050-FM-096B	Flow/Valve Operating Numbers Diagram: Safety Injection System, Unit 2
3.	11715-FM-091A	Flow/Valve Operating Numbers Diagram: Containment Quench and Recirculation Spray Subsystem, Unit 1
4.	11715-FM-093A	Flow/Valve Operating Numbers Diagram: Reactor Coolant System; Loops 1, 2, & 3; Unit 1
5.	11715-FM-094A	Flow/Valve Operating Numbers Diagram: Residual Heat Removal System, Unit 1
6.	11715-FM-095A	Flow/Valve Operating Numbers Diagram: Chemical and Volume Control System, Unit 1
	12050-FM-095A	Flow/Valve Operating Numbers Diagram: Chemical and Volume Control System, Unit 2
7.	11715-FM-095B	Flow/Valve Operating Numbers Diagram: Chemical and Volume Control System, Unit 1
	12050-FM-095B	Flow/Valve Operating Numbers Diagram: Chemical and Volume Control System, Unit 2
8.	11715-FM-095C	Flow/Valve Operating Numbers Diagram: Chemical and Volume Control System, Unit 1
	12050-FM-095C	Flow/Valve Operating Numbers Diagram: Chemical and Volume Control System, Unit 2

Table 6.3-1
EMERGENCY CORE COOLING SYSTEM COMPONENT PARAMETERS

Component	Parameters	
Accumulators	Number	3
	Design pressure	700 psig
	Design temperature	300°F
	Operating temperature	60-150°F
	Normal operating pressure	650 psig
	Minimum operating pressure	599 psig
	Total volume	1450 ft^3 each
	Minimum water volume	1013 ft^3 each
	Volume N gas	425 ft^3
	Boric acid concentration, minimum	2500 ppm
	maximum	2800 ppm
	Relief valve setpoint	700 psig
Centrifugal charging pumps	Number	3
	Design pressure	2735 psig
	Design temperature	250°F
	Design flow rate	150 gpm
	Design head	5800 ft
	Max flow rate	675 gpm
	Head at max flow rate	2400 ft
	Discharge head at shutoff	6000 ft
	Motor rating (BHP)[a]	900
Low head safety injection pumps	Number	2
	Design pressure	300 psig
	Design temperature	300°F
	Design flow rate	3000 gpm
	Design head	250 ft
	Max flow rate	4500 gpm
	Head at max flow rate	150 ft
	Motor rating (BHP)[a]	250
Low head safety injection strainer assembly	Number	1
	Material	SS 304 or 304L
	Structural DP	9.0 psid
	Perforation	0.0625 inches (nominal)
	Operating Pressure	9.0-59.7 psia
	Operating Temperature	75-280°F
	Fluid Flowing	Borated Water
Boron injection tank	Number	1
	Total volume	900 gal

a. 1.15 service factor not included.

Table 6.3-1 (continued)
EMERGENCY CORE COOLING SYSTEM COMPONENT PARAMETERS

Component	Parameters	
	Usable volume at operating conditions (solution)	900 gal
	Boron concentration, nominal	14,000 ppm
	Design pressure	2735 psig
	Operating pressure	100 psig
	Design temperature	300°F
	Operating temperature	120-150°F
Heaters	Number of channels	2
	Capacity, each channel	6 kW
	Type	Strip
Valves		
1. All motor-operated valves that must function on safety injection signal (SIS)	Maximum opening or closing time (Note 1)	
a. Up to and including 8 inches	Time	10 sec
b. Over 8 inches	Speed	49 in/min
2. All other motor-operated valves	Maximum opening or closing time (Note 1)	
a. Up to and including 8 inches	Speed	12 in/min
b. Over 8 inches	Time	120 sec
3. Leakage		
a. Conventional globe valves	Disk leakage (of nominal pipe size)	3 cc/hr/in
	Backseat leakage (when open) (stem diameter)	1 cc/hr/in
b. Gate valves	Disk leakage (of nominal pipe size)	3 cc/hr/in
c. Check valves	Disk leakage (of nominal pipe size)	3 cc/hr/in
d. Diaphragm valves	Disk leakage	None
e. Pressure relief valves	Disk leakage (of nominal pipe size)	3 cc/hr/in
f. Accumulator check valves	Disk leakage (of nominal pipe size)	3 cc/hr/in

NOTES:

1. The maximum stroke times in this table are the stroke time requirements that originally applied. Current allowable valve stroke times may exceed original requirements. The current allowable valve stroke times are based on safety system response requirements and/or accident analysis and are considered when developing acceptance criteria to ensure that the valve performance does not degrade below the performance required by the safety system response requirements and/or accident analysis. The acceptance criteria are developed and the tests performed in accordance with the ASME Inservice Testing Program as required by 10 CFR 50.55(a)(f)4.

a. 1.15 service factor not included.

Table 6.3-2
EMERGENCY CORE COOLING SYSTEM CODE REQUIREMENTS

Component	Code
Accumulators	ASME III, Class C, 1968
Boron injection tank	ASME III, Class C, 1968
Valves	ANSI B16.5
	MSS SP 66, ASME, III
Piping	ANSI B31.7-1969 and Addenda through 1970 - Nuclear Power Piping Code (ANSI B31.7)
Pumps (charging, low head safety injection)	ASME III, Class C, 1968

Table 6.3-3

MATERIALS EMPLOYED FOR EMERGENCY CORE COOLING SYSTEM COMPONENTS

Component	Material
Accumulators	Carbon steel, clad with austenitic stainless steel
Boron injection tank	Carbon steel, clad with austenitic stainless steel
Pumps	
Centrifugal charging	Austenitic stainless steel or equivalent corrosion-resistant material
Low head safety injection	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
Motor-operated valves containing nonradioactive, 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	17-4PH stainless
Relief valves	
Stainless steel bodies	Stainless steel
Carbon steel bodies	Carbon steel
All nozzles, disks, 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	Austenitic stainless steel

Table 6.3-4
ECCS RELIEF VALVE DATA

Description	Fluid Discharged	Fluid Inlet Temp. Normal	Set Pressure PSIG	Back Pressure Constant	PSIG Buildup	Capacity
N_2 supply to accumulators	N_2	120	700	0	0	1500 scfm
Boron injection tank relief	8% B.A.	135	2735	3	12	20 gpm
Low head pumps SI line	Water	120	257/264[a]	3	50	20 gpm
Accumulator	Water or N_2 gas	120	700	0	0	1500 scfm
Hydrotest pump	Water	120	900	0	0	25 gpm
Accumulator test line relief	Water	150	1000	0	0	66 gpm

a. 257 Unit 1
264 Unit 2

Table 6.3-5
NORMAL OPERATING STATUS OF EMERGENCY CORE COOLING SYSTEM COMPONENTS FOR CORE COOLING

Number of low head safety injection pumps operable	2
Number of charging pumps operable	2
Refueling water storage tank volume	466,200 - 487,000 gal
Boron concentration in refueling water storage tank	2600 - 2800 ppm
Boron concentration in accumulator	2500 - 2800 ppm
Number of accumulators	3
Minimum accumulator pressure	599 psig
Minimum accumulator water volume	1013 ft^3

Table 6.3-6
MAXIMUM POTENTIAL RECIRCULATION LOOP LEAKAGE EXTERNAL TO CONTAINMENT

Items	No. of Units	Type of Leakage Control and Unit Leakage Rate Used in the Analysis[a]	Leakage to Atmosphere (cc/hr)	Leakage to Vent and Drain System (cc/hr)
Low head safety injection pumps	2	Mechanical seal with leakoff - three drops per min.	0[b]	10
Safety injection charging	3	Mechanical seal with leakoff - three drops per min.	0	20
Flanges:				
1. Pump	10	Gasket - adjusted to zero leakage following any test. Valves - ten drops per min. per flange used in analysis. Pump - due to leaktight flanges on pumps, no leakage assumed.	0	0
2. Valves bonnet to body (larger than 2 in.)	54		560	0
Valves - steam leakoffs	27	Backseated, double packing with leakoff - one cc/hr per in. stem diameter	0	27
Misc. valves	33	Flanged body packed stems - one drop per min. used	99	0
Totals			659	57

a. Unit leakage rates are original design criteria. The actual allowable leakage for each leakage control component may exceed the original leakage rate indicated as long as the total ECCS recirculation loop leakage and the recirculation spray subsystem leakage outside of containment does not exceed the curve of allowable ECCS leakage which corresponds to the control room unfiltered inleakage. A curve of allowables ECCS leakage for 250 cfm of unfiltered control room inleakage is shown in Figure 15.4-78 and its use is discussed in Sections 15.4.1.9.6 and 15.4.1.9.8.

b. Due to the tandem double seal arrangement and the use of water from the RWST as a buffer between the seals, no radioactive leakage from the low head safety injection pumps to the atmosphere is expected.

Table 6.3-7
EMERGENCY CORE COOLING SYSTEM SHARED FUNCTIONS EVALUATION

Component	Normal Operating Arrangement	Accident Arrangement
Boron injection tank	Lined up for recirculation to the Chemical and Volume Control System boric acid storage tanks	Lined up to discharge of charging pumps and cold legs of reactor coolant system. Valves for realignment meet single-failure criteria.
Refueling water storage	Lined up to suction of low head safety injection pumps	Lined up to suction of centrifugal charging and low head safety injection pumps. Valves for realignment to suction of centrifugal charging pumps meet single-failure criteria
Centrifugal charging pumps	Lined up for charging service	Lined up to inlet of boron injection tank. Valves for realignment meet single-failure criteria.

Table 6.3-8
LISTING OF DETAILED CONTAINMENT HEAT SINKS

No.	Item Description	Weight (lb)	Surface Area (ft^2)	Thickness (ft)	Material	Surface Coating
A.	Nonconcrete Listing[a]					
1	Structural steel[b]	664,000	74,700	0.036	Carbon steel	Paint
2	Grating[c]	138,700	41,600	0.016	Carbon steel	Galvanized
3	Pipe supports, linings, piping, and valves	150,600	11,600	0.026	Carbon steel	Paint
4	Pressurizer relief tank	22,880	627	0.052	Carbon steel	Paint
5	Safety injection accumulator	189,000	2073	0.145	Carbon steel	Paint
6	Containment polar crane[a]	522,800	8970	0.236	Carbon steel	Paint
7	Reactor coolant pump support[b]	165,000	3147	0.210	Carbon steel	Paint
8	Steam generator lower support[b]	660,000	12,807	0.206	Carbon steel	Paint
9	Containment ductwork	38,000	25,300	0.003	Carbon steel	Galvanized
10	Heat shield surge tank	2900	206	0.028	Carbon steel	Paint
11	Containment liner	1,270,000	74,700	0.034	Carbon steel	Paint
12	Containment vent duct fans	4760	311	0.031	Carbon steel	Paint
13	CRDM coolant duct fans	4170	273	0.031	Carbon steel	Paint
14	Refueling cavity liner	106,000	7566	0.021	Stainless steel	None
15	Reactor head storage stand	12,650	360	0.083	Carbon steel	Paint
16	Containment condenser coils	57,350	784	0.131	Copper	None
17	Main steam and feedwater pipe break restraints	246,000	3940	0.250	Carbon steel	Paint

a. Density of steel = 490 lb/ft^3.
b. Indicates heat sinks with surface exposed on *both* sides to the containment atmosphere.
c. Indicates heat sinks with surface exposed on *four* sides to the containment atmosphere.

Table 6.3-8 (continued)
LISTING OF DETAILED CONTAINMENT HEAT SINKS

No.	Item Description	Weight (lb)	Surface Area (ft^2)	Thickness (ft)	Material	Surface Coating
A.	Nonconcrete Listing[a] (continued)					
18	Instrument air compressors	1960	96	0.041	Carbon steel and cast iron	Paint
19	Primary drain transfer tank	3000	125	0.048	Stainless steel	None
20	Head lift gear w/intermediate lift ring	34,200	610	0.080	Carbon steel	Paint
21	Refueling cavity water seal	17,000	226	0.167	Carbon steel	Paint
22	Residual heat removal pumps	11,120	172	0.131	Stainless steel	None
23	Electrical conduit and EMT	47,550	12,482	0.008	Carbon steel	Galvanized
24	Manipulator crane	40,000	4000	0.020	Carbon steel	Paint
25	Reactor coolant pumps	393,000	1050	0.691	Carbon steel	Paint
26	Recirculation spray pumps	10,600	176	0.124	Stainless steel	None
27	Reactor upper and lower internal storage assembly	12,480	443	0.056	Stainless steel	None
28	Cable trays	57,560	21,320	0.005	Carbon steel	Galvanized
29	Instrument air tanks	1450	163	0.018	Carbon steel	Paint
30	Residual heat exchanger	8300	466	0.036	Carbon steel	Paint
31	Recirculation spray cooler	26,600	1738	0.031	Stainless steel	None
32	Pipe break restraints	346,000	9606	0.074	Carbon steel	
33	Loop room permanent scaffold	3,800	960	0.008	Carbon steel	Galvanized

TOTAL WEIGHT = 5,269,400 lb
TOTAL SURFACE AREA = 322,597 ft^2

a. Density of steel = 490 lb/ft^3.
b. Indicates heat sinks with surface exposed on *both* sides to the containment atmosphere.
c. Indicates heat sinks with surface exposed on *four* sides to the containment atmosphere.

Table 6.3-8 (continued)
LISTING OF DETAILED CONTAINMENT HEAT SINKS

No.	Item Description	Area (ft^2)	Thickness (ft)
B.	Concrete Listing[d]		
1	Interior concrete walls and floor	8149	0.5
2	Interior concrete walls and floor	60,457	1.0
3	Interior concrete walls and floor	53,752	1.5
4	Interior concrete walls and floor	11,253	2.0
5	Interior concrete walls and floor	9130	2.25
6	Interior concrete walls and floor	3530	3.0
7	Containment concrete shell - below grade	21,397	4.5
8	Containment concrete shell - above grade	28,090	4.5
9	Containment concrete dome	24,925	2.5
10	Containment mat and subfloor	11,757	2.2/10.0
C.	Supplemental Information		

1. Item No. 2 and 4 thickness, surface area, and weight are based on manufacturer specifications.
2. Item No. 5 surface area underneath the accumulator skirt (i.e., not exposed to the containment atmosphere) is not included.
3. Item No. 14 thickness and weight are based on manufacturer specifications. Weight includes the steel I-beams located behind the liner. However, the surface area of the I-beams is not exposed to the containment atmosphere.
4. Item No. 15 and 21 top and sides surface area only are exposed to containment atmosphere.
5. The concrete listings (i.e., Part B of this table) are incorporated into the first 10 categories of the structural heat sinks listed in Table 15.4-11.
6. The nonconcrete listings (i.e., Part A of this table) of stainless steel material are incorporated into Category 11 of the structural heat sinks listed in Table 15.4-11.
7. The nonconcrete listings (i.e., Part A of this table) of nonstainless steel material are incorporated into Category 12 of the structural heat sinks listed in Table 15.4-11.
8. Item No. 32 of the nonconcrete listings (i.e., Part A of this table) is incorporated into Category 13 of the structural heat sinks listed in Table 15.4-11.
9. A conservative metal surface area and mass were used to model the AECL containment sump strainer in the ECCS containment backpressure analysis. Refer to Sections 6.3.3.12 and Table 15.4-11.

d. Density of concrete = 145 lb/ft^3.

Figure 6.3-1 (SHEET 1 OF 2)
SAFETY INJECTION SYSTEM

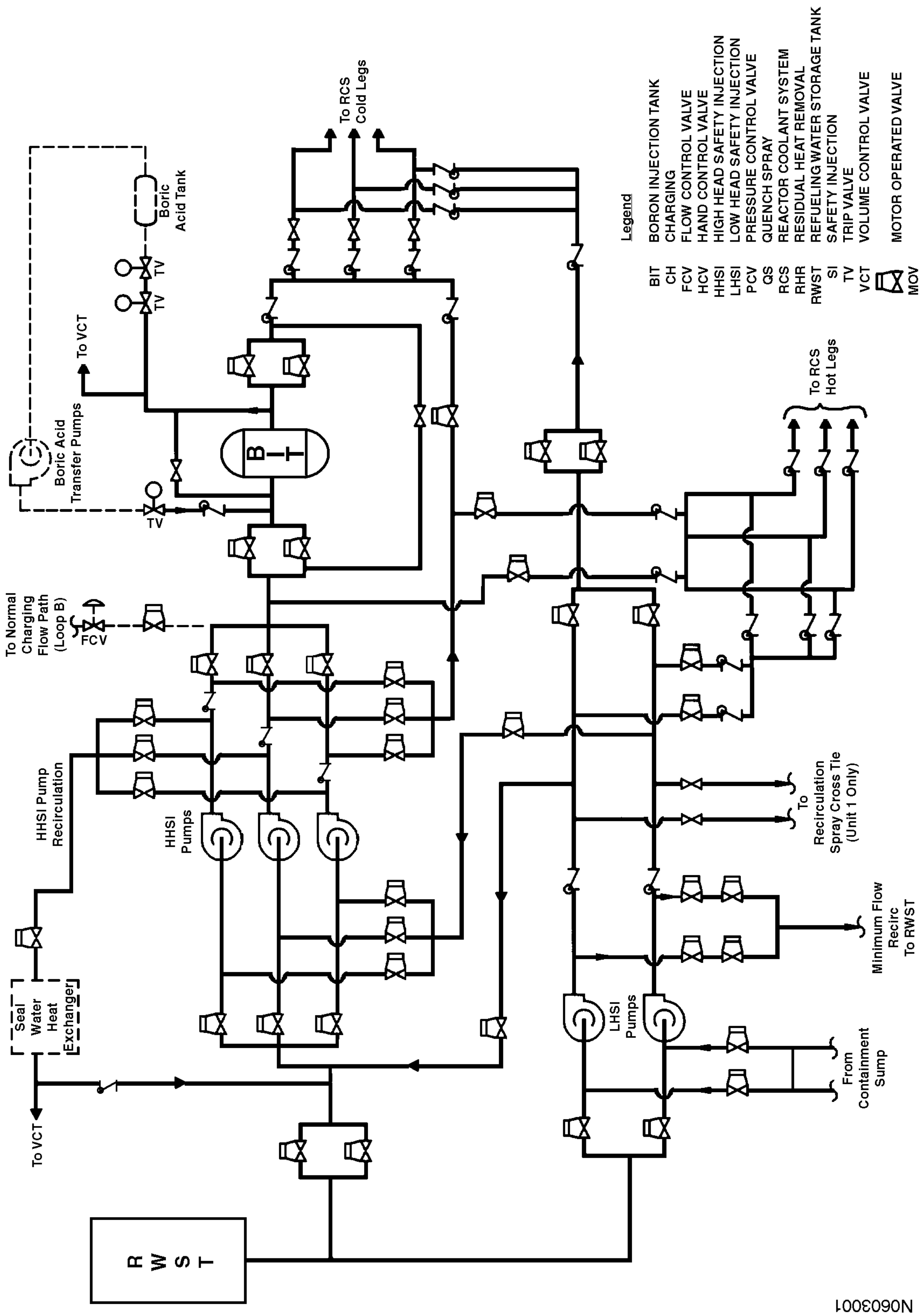


Figure 6.3-1 (SHEET 2 OF 2)
SAFETY INJECTION SYSTEM

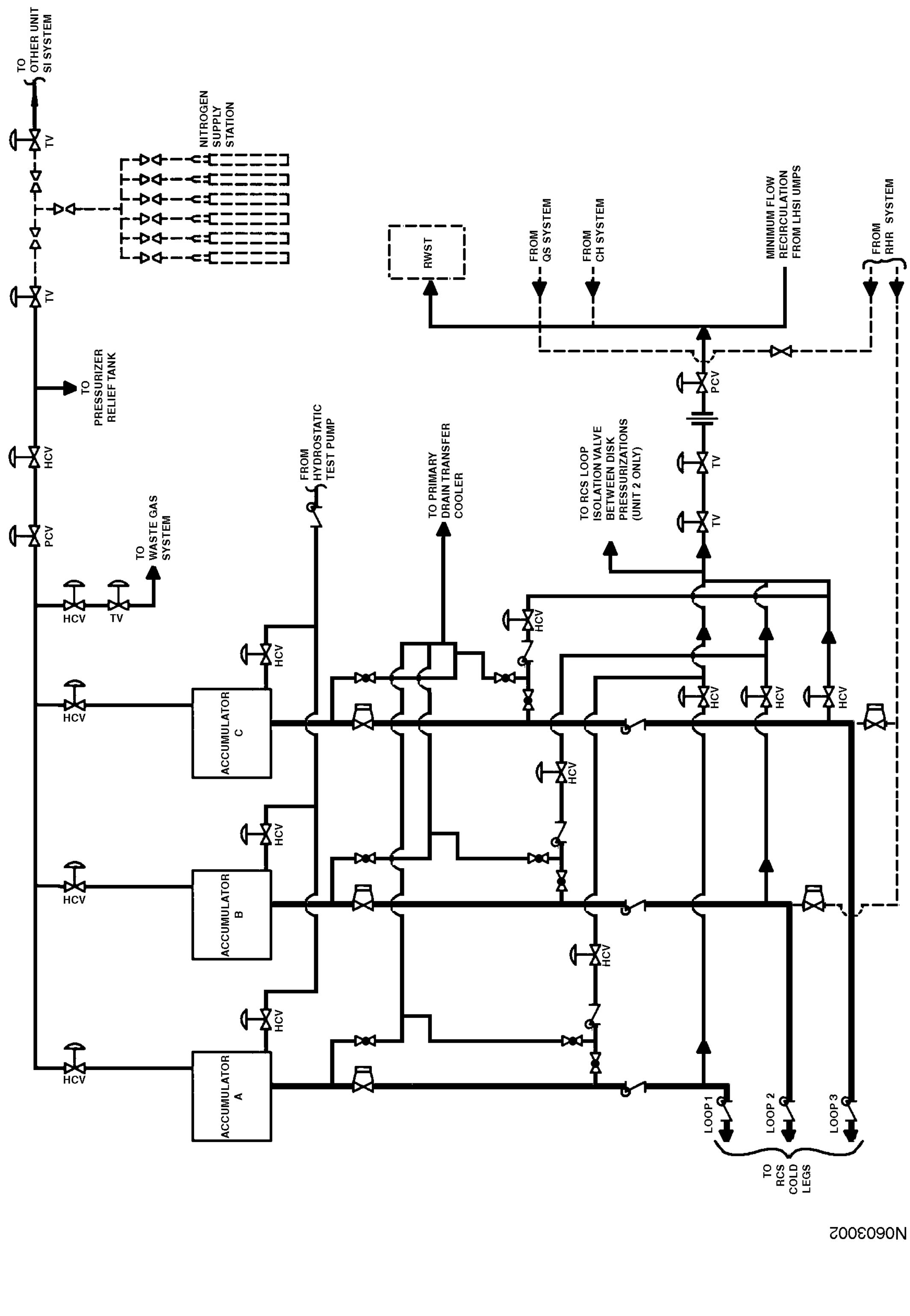
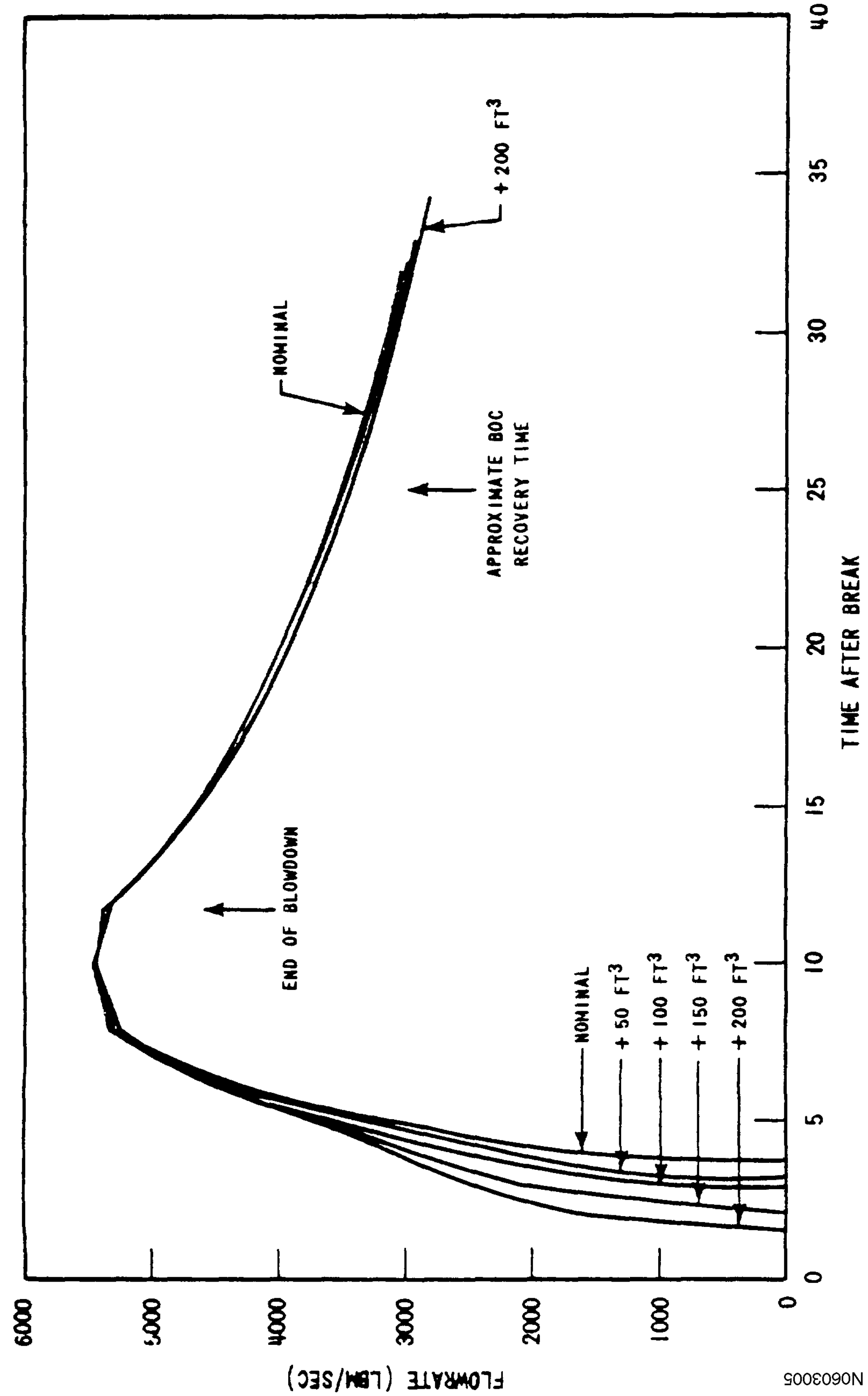


Figure 6.3-2
(VARY INITIAL PRESSURE & VOLUME)
INITIAL ACCUMULATOR PRESSURE INCREASED PROPORTIONALLY TO GAS VOLUME REDUCTION AFTER IN-LEAKAGE
(Volume Increase Indicated is for Two Tanks)



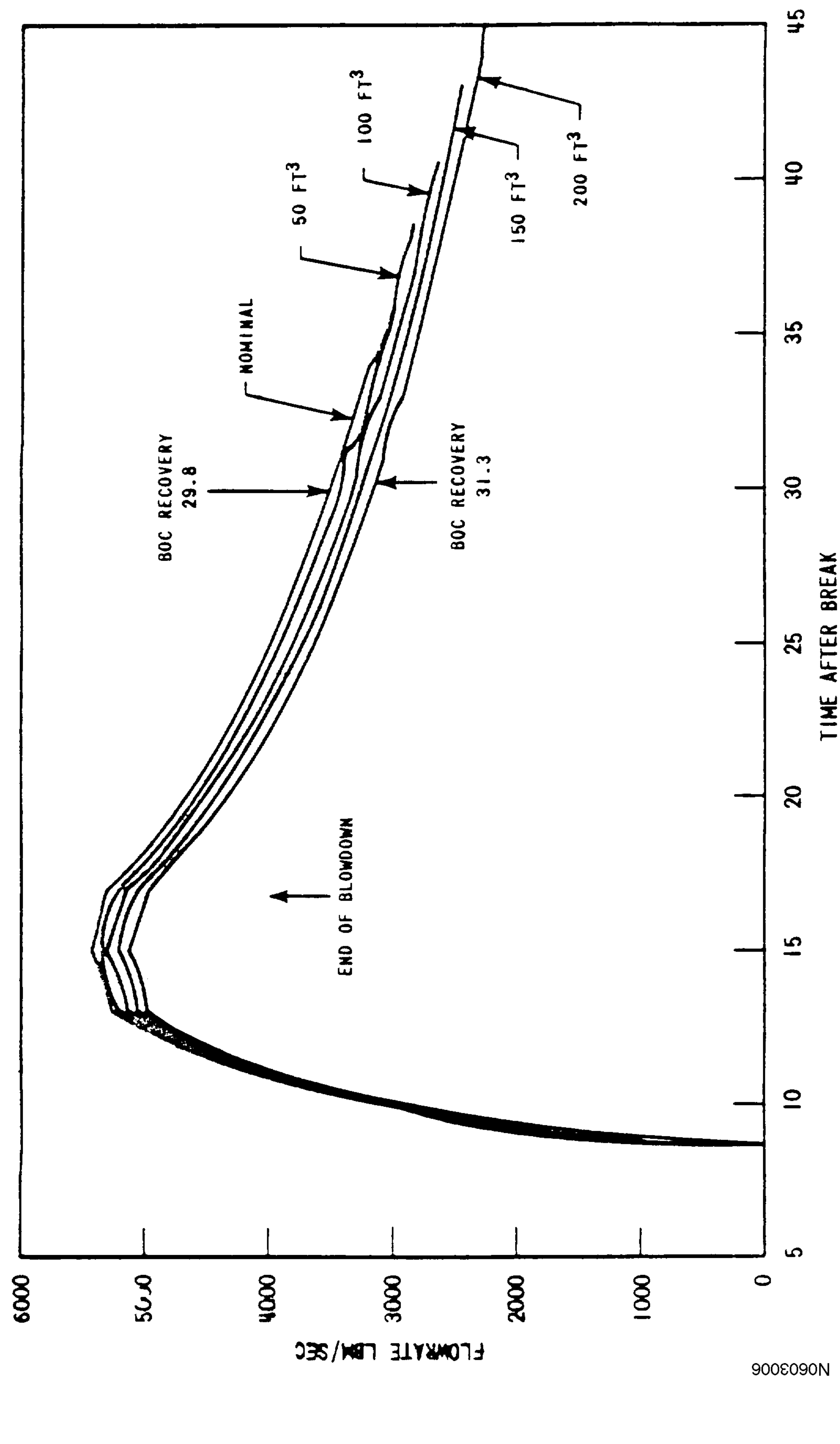


Figure 6.3-3
(VARY INITIAL VOLUME ONLY)
TANK PRESSURE RESET TO 600 PSIA AFTER IN-LEAKAGE
(Volume Increase Indicated is for Two Tanks)

Figure 6.3-4
SIMPLIFIED SKETCH OF THE ORIGINAL
ECCS DESIGN SHOWING PRESENT DESIGN MODIFICATIONS

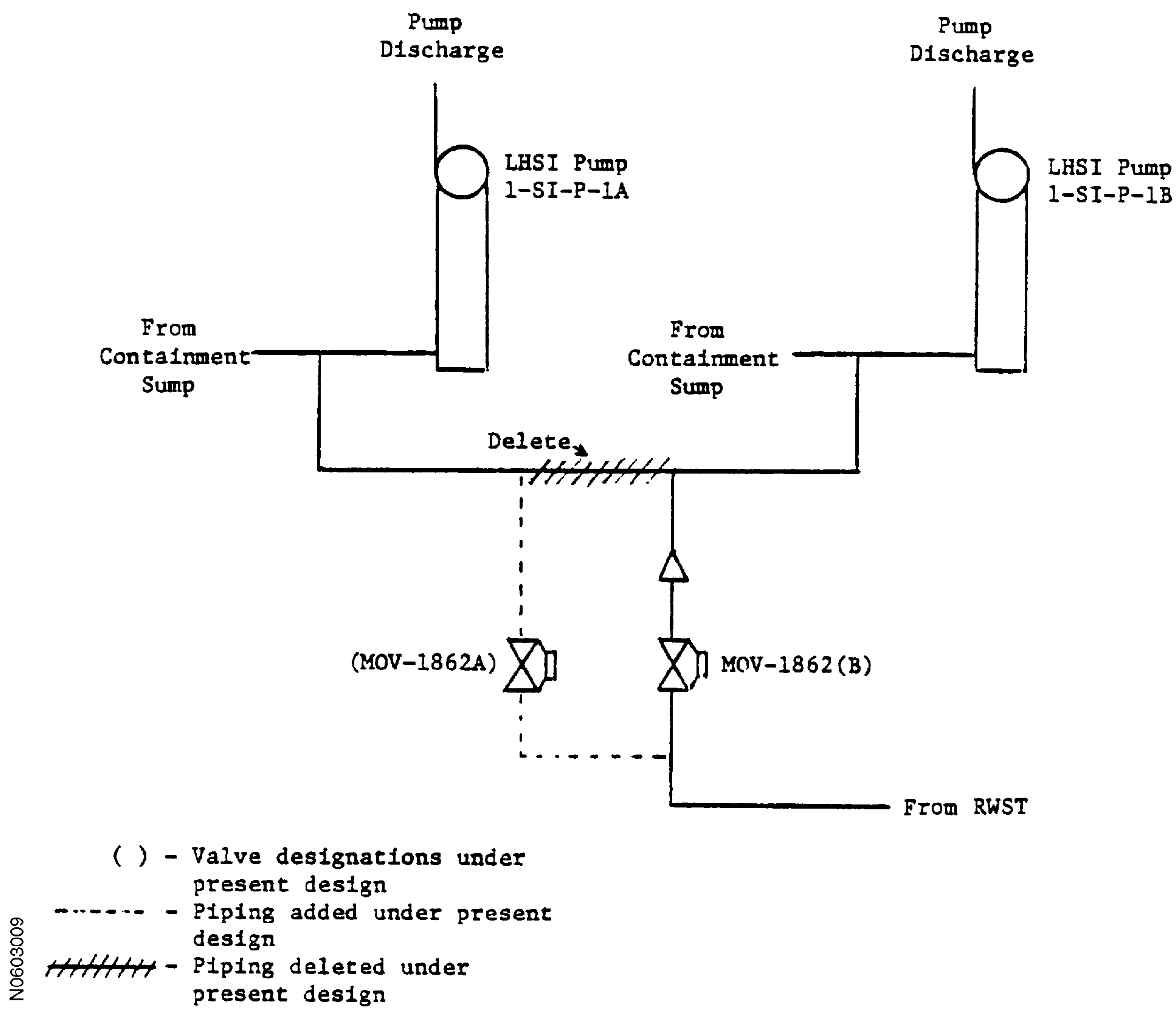
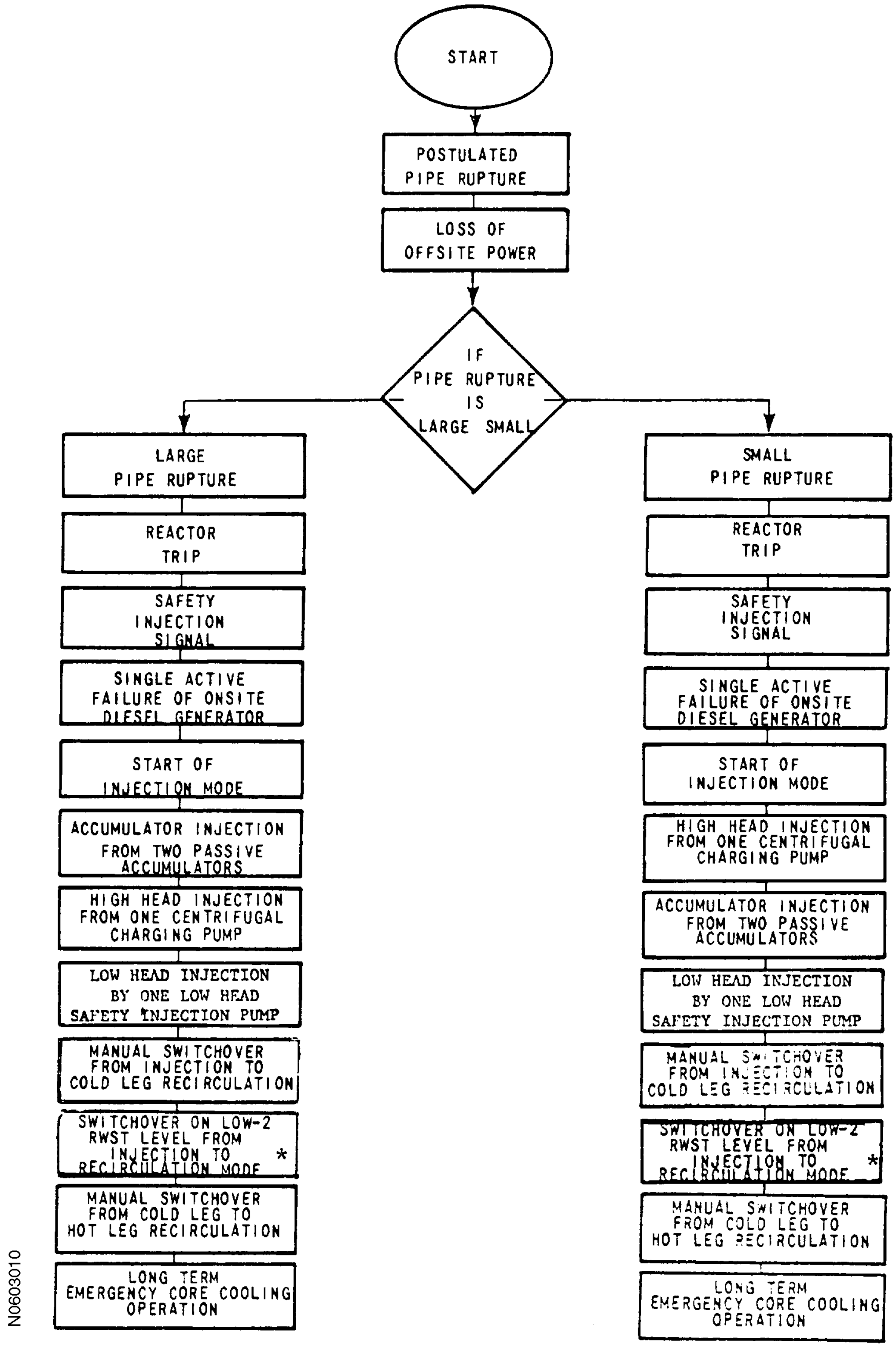


Figure 6.3-5
SEQUENCE OF EVENTS FOLLOWING POSTULATED ACCIDENTS



*If manual switchover is not performed.

Figure 6.3-6
AVAILABLE NPSH LHSI PUMP NPSH AVAILABLE ANALYSIS

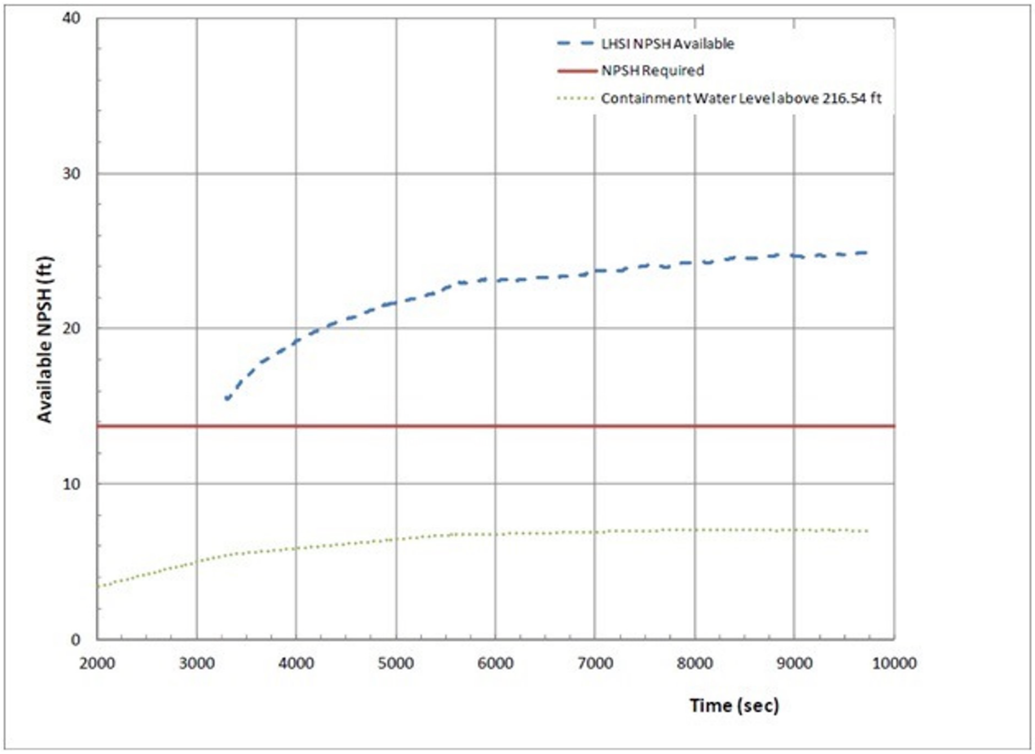


Figure 6.3-7
CONTAINMENT PRESSURE LHSI PUMP NPSH AVAILABLE ANALYSIS

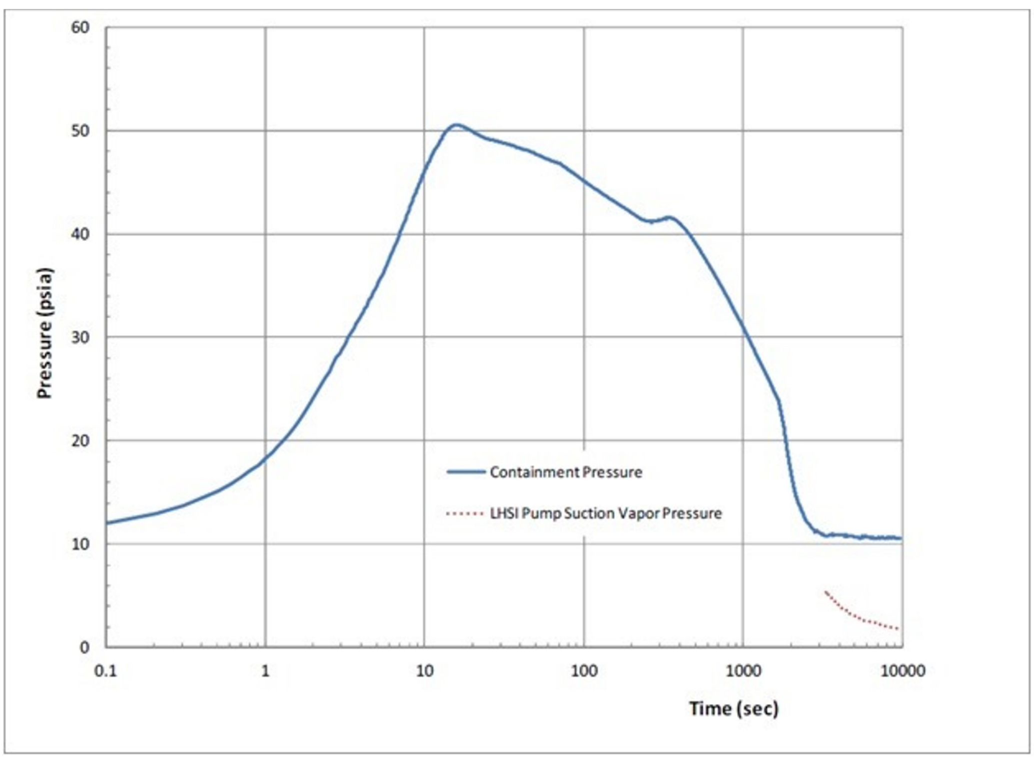


Figure 6.3-8
CONTAINMENT TEMPERATURE FROM LHSI PUMP NPSH AVAILABLE ANALYSIS

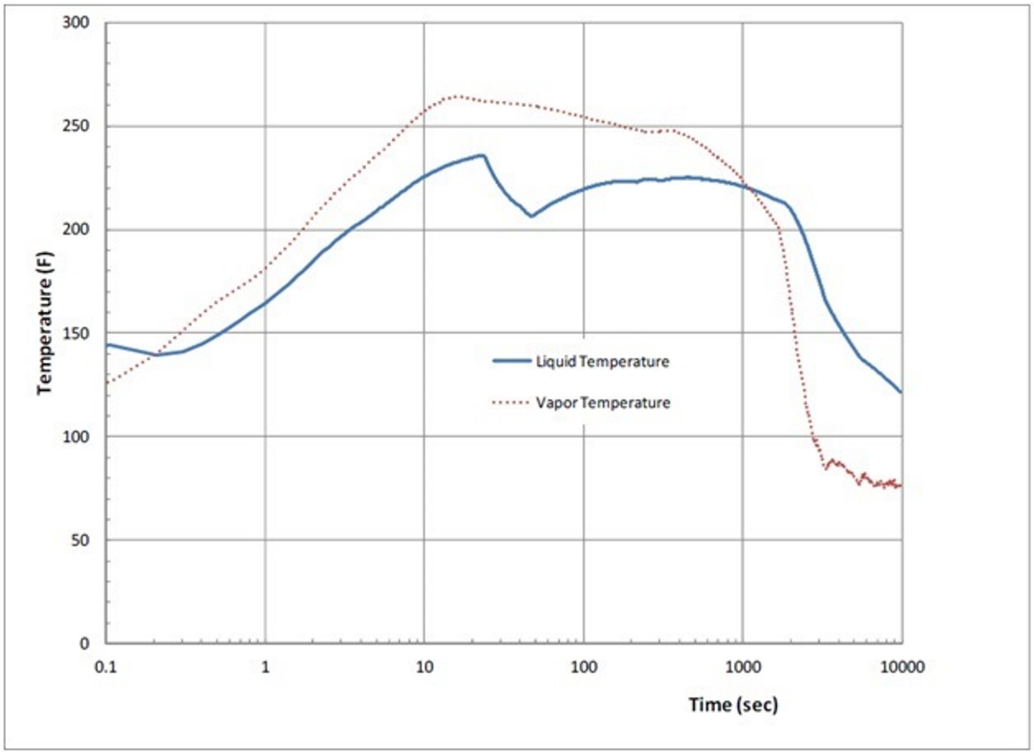


Figure 6.3-9
TOTAL RSHX HEAT RATE LHSI PUMP NPSH AVAILABLE ANALYSIS

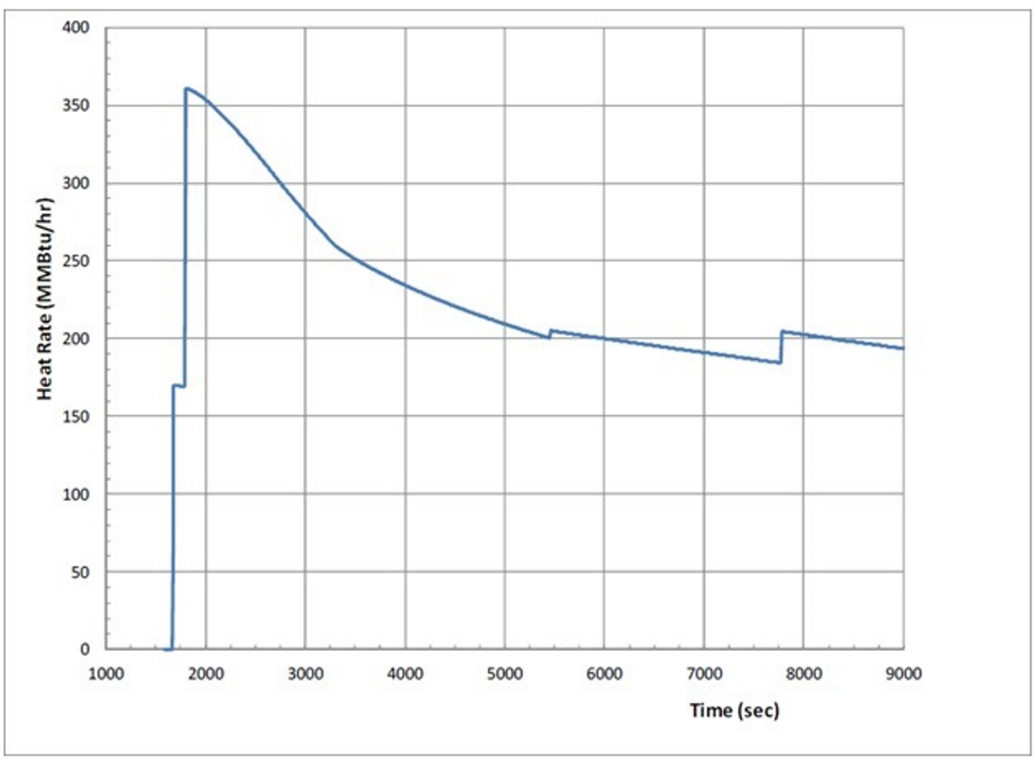
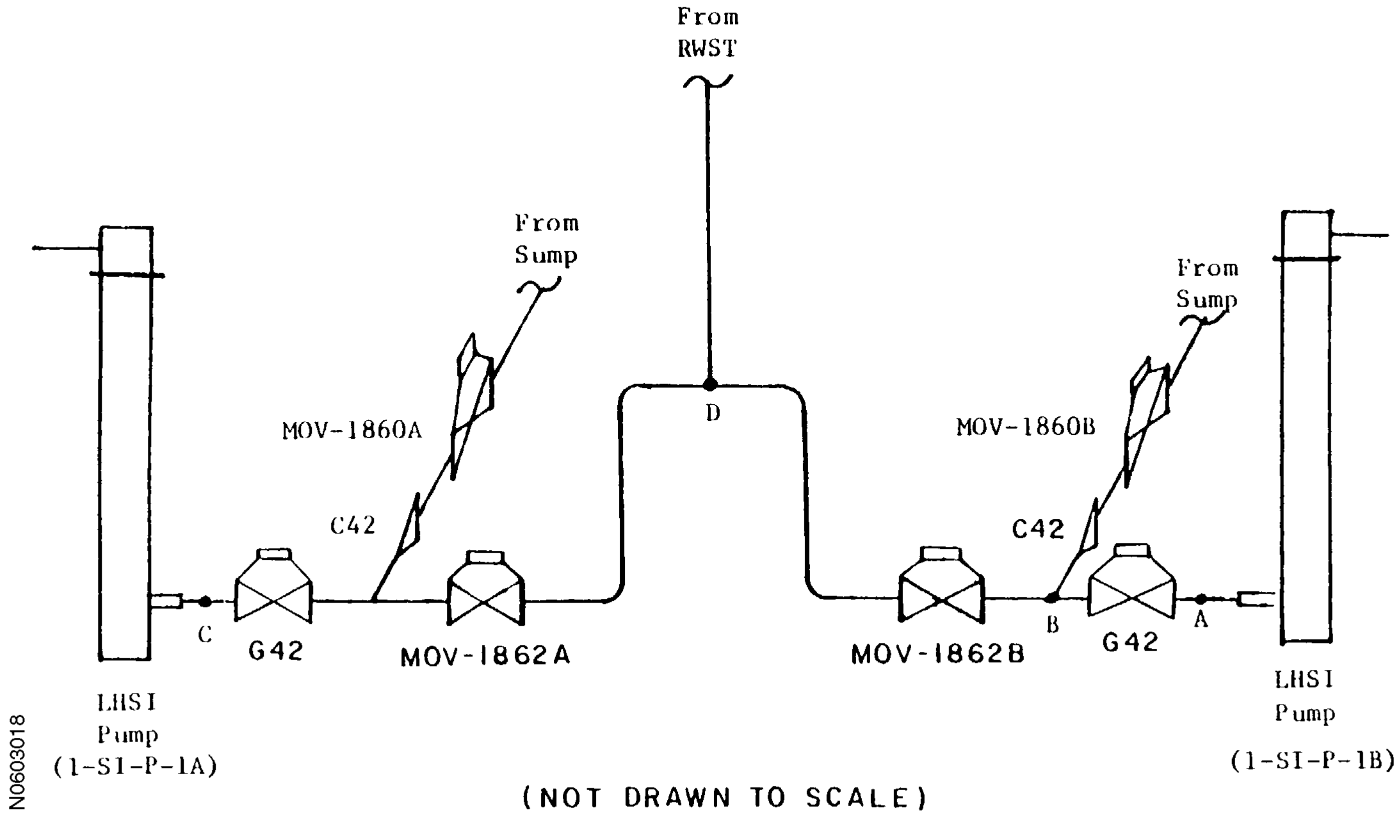
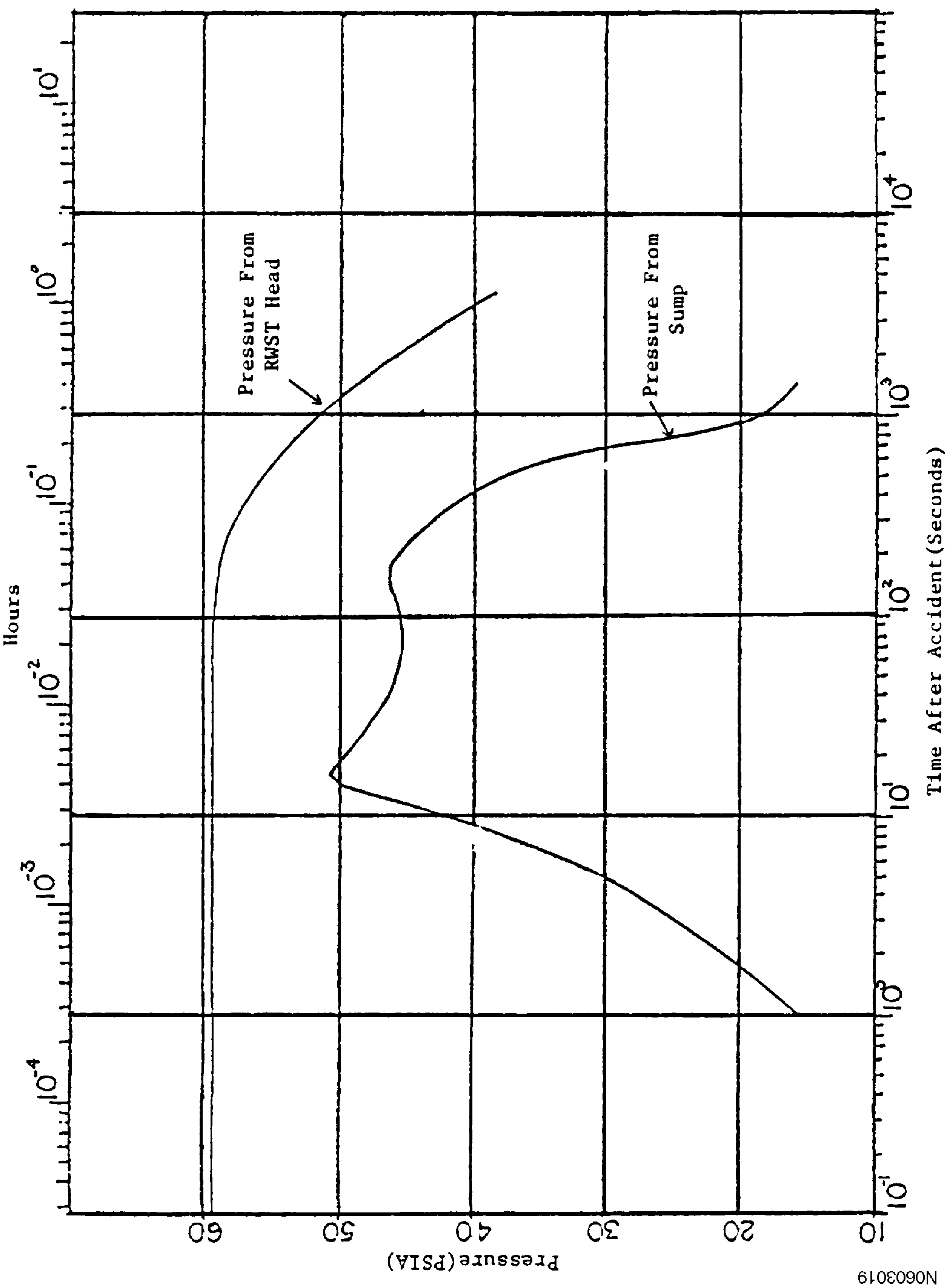


Figure 6.3-10
SIMPLIFIED SCHEMATIC LOW HEAD SAFETY INJECTION SYSTEM



(NOT DRAWN TO SCALE)

Figure 6.3-11
PRESSURE AT POINT B AS SHOWN ON FIGURE 6.3-10
ASSUMING 3000-GPM FLOW FROM SUMP TO LHSI PUMP AT POINT C



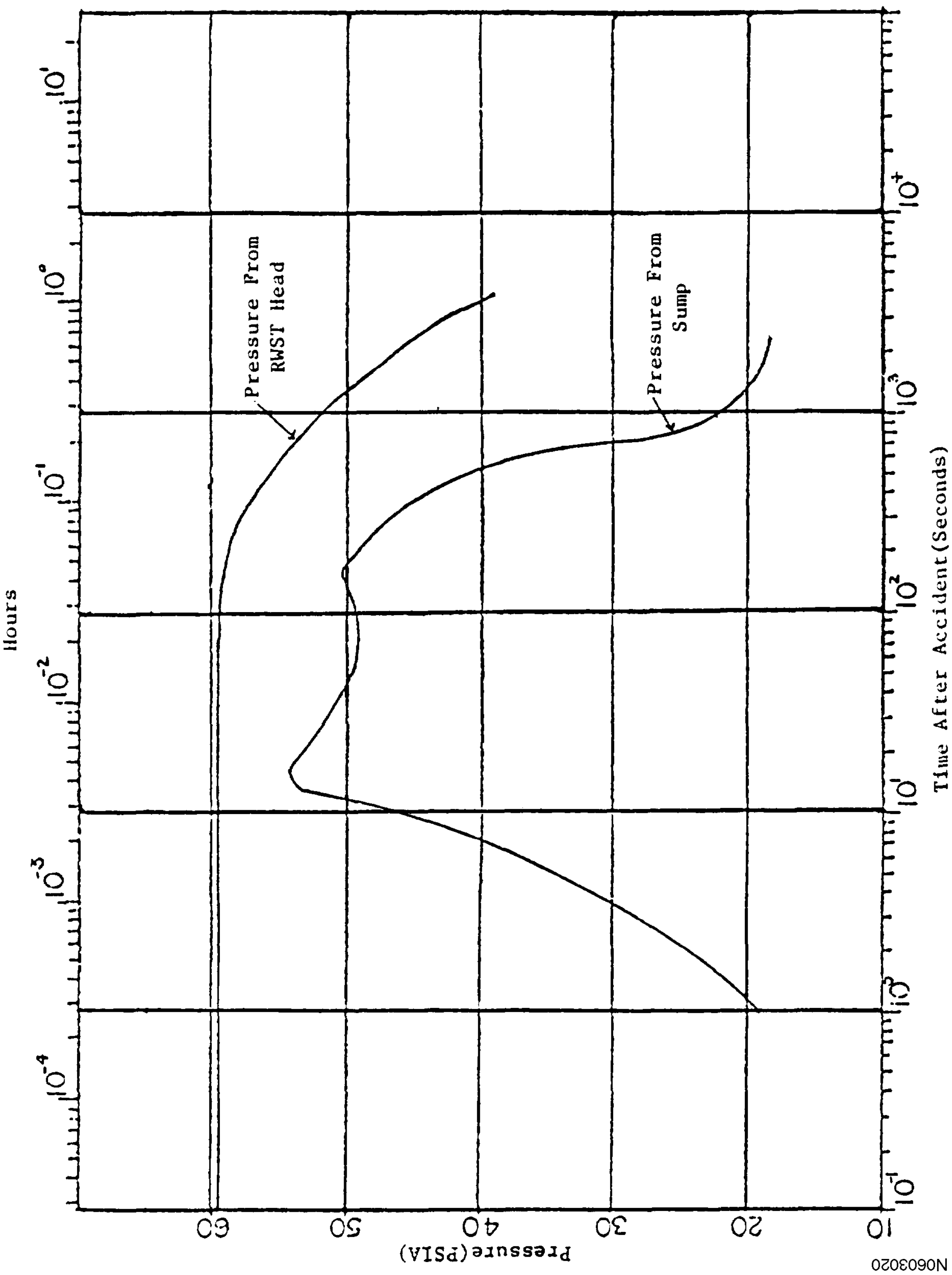


Figure 6.3-12
PRESSURE AT POINT B AS SHOWN ON FIGURE 6.3-10
ASSUMING NO FLOW TO LHSI PUMP AT POINT C

Figure 6.3-13
BORIC ACID SOLUBILITY VS. TEMPERATURE

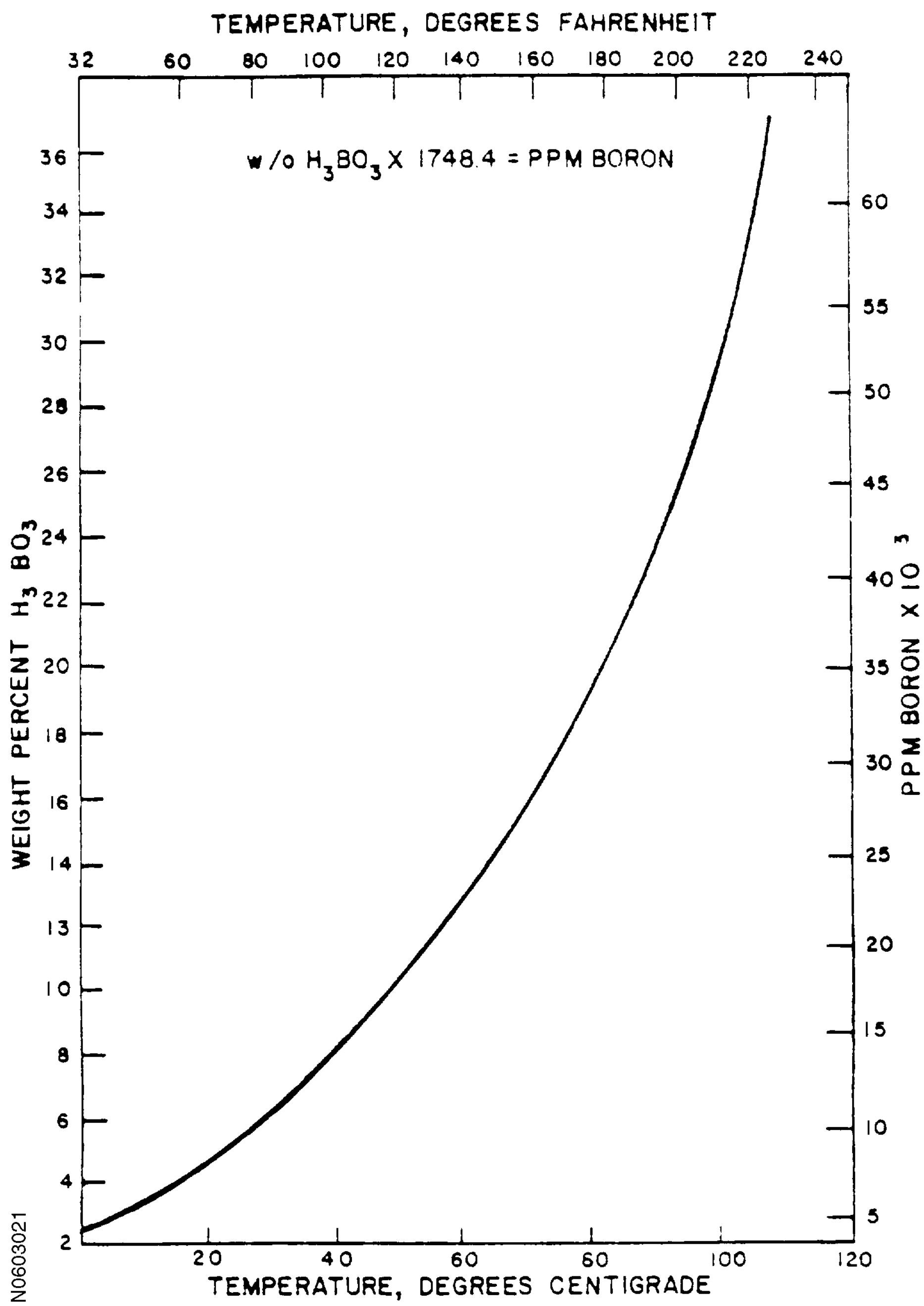
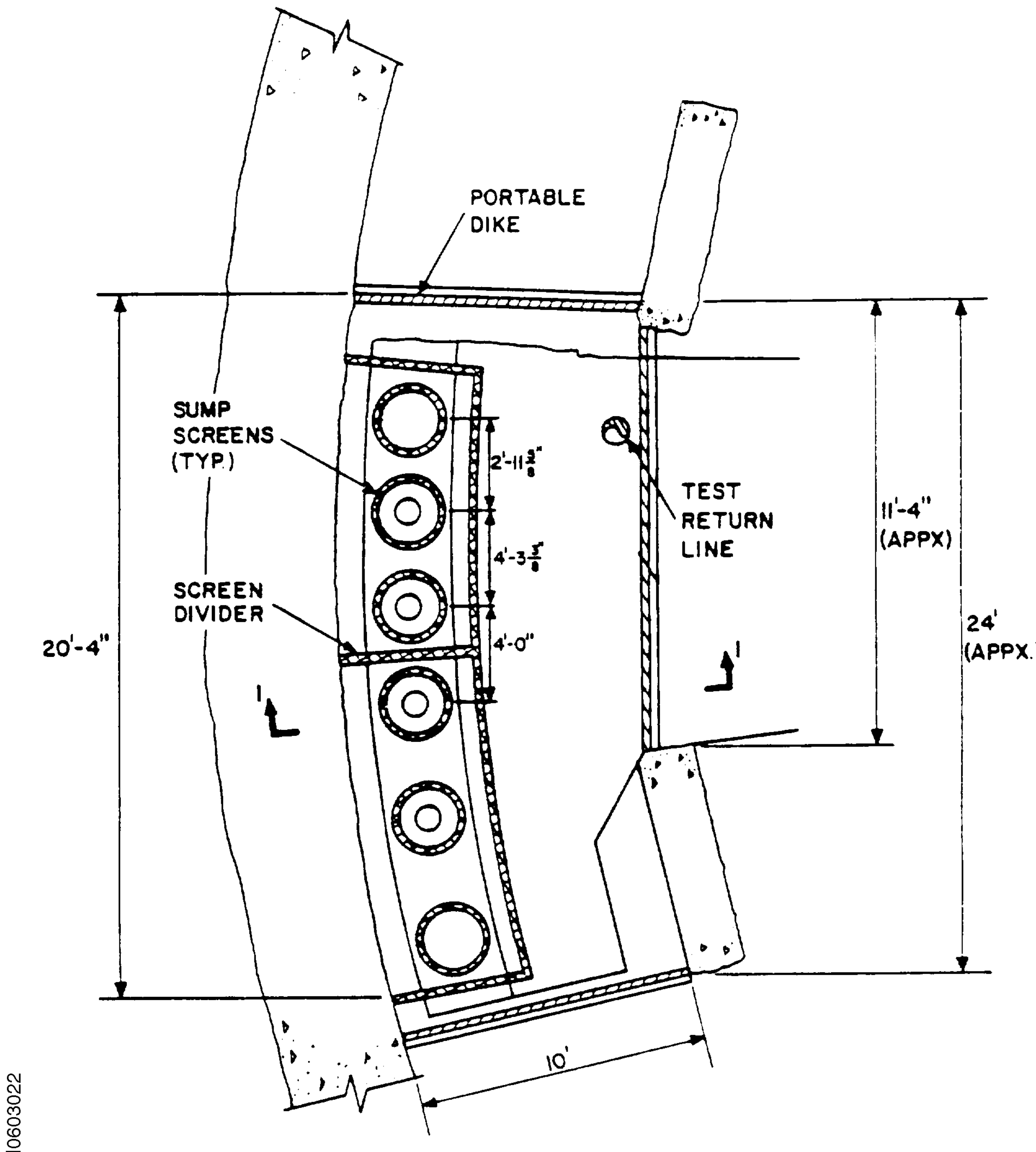


Figure 6.3-14 (SHEET 1 OF 2)
LOW HEAD SI PUMP TEST ARRANGEMENT



NOTE:
* 4'-0" WIDTH WILL GIVE 120% OF MAXIMUM APPROACH VELOCITY (.41 FT/SEC) = .49 FT. SEC.

Figure 6.3-14 (SHEET 2 OF 2)
LOW HEAD SI PUMP TEST ARRANGEMENT

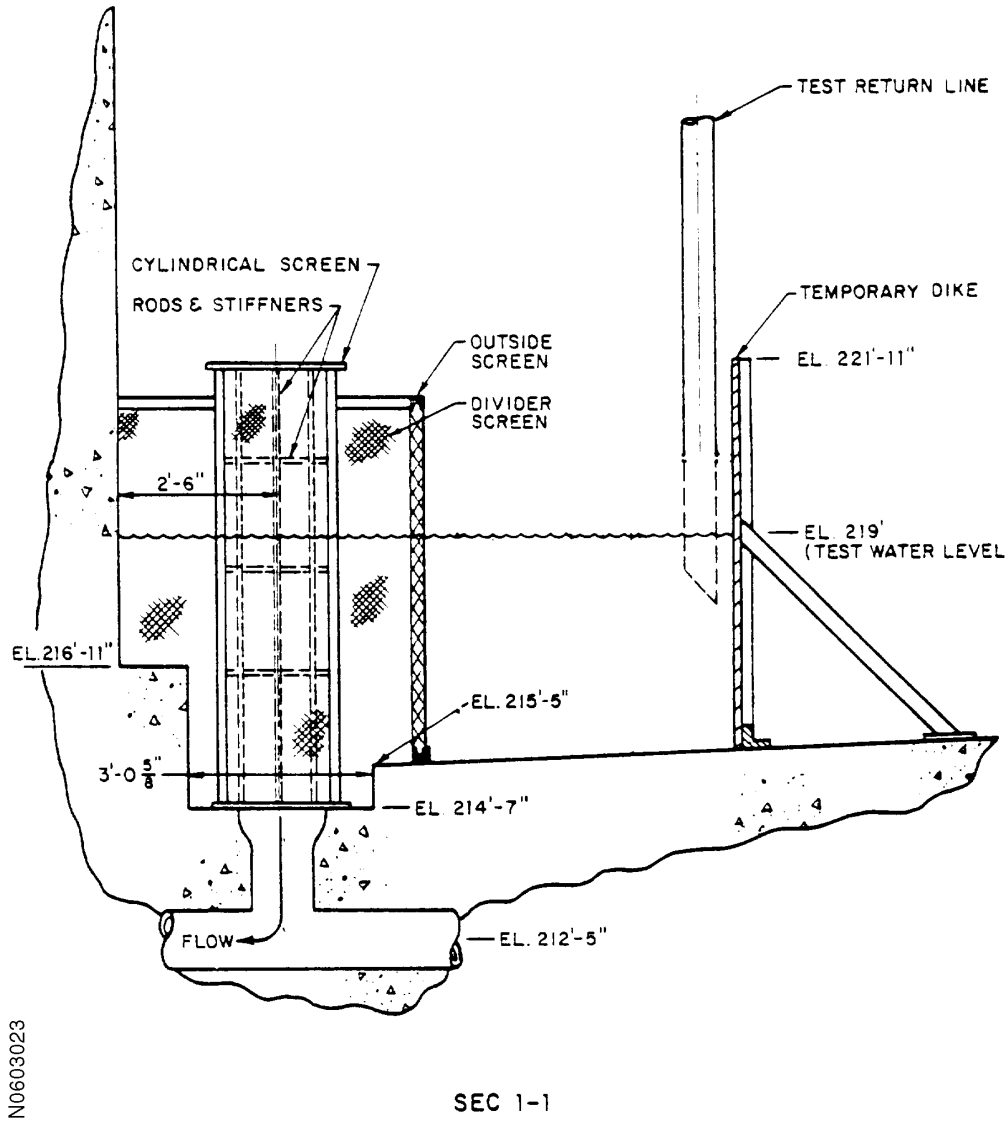


Figure 6.3-15
NORTH ANNA UNITS 1 AND 2 CONTAINMENT WALL HEAT TRANSFER COEFFICIENT

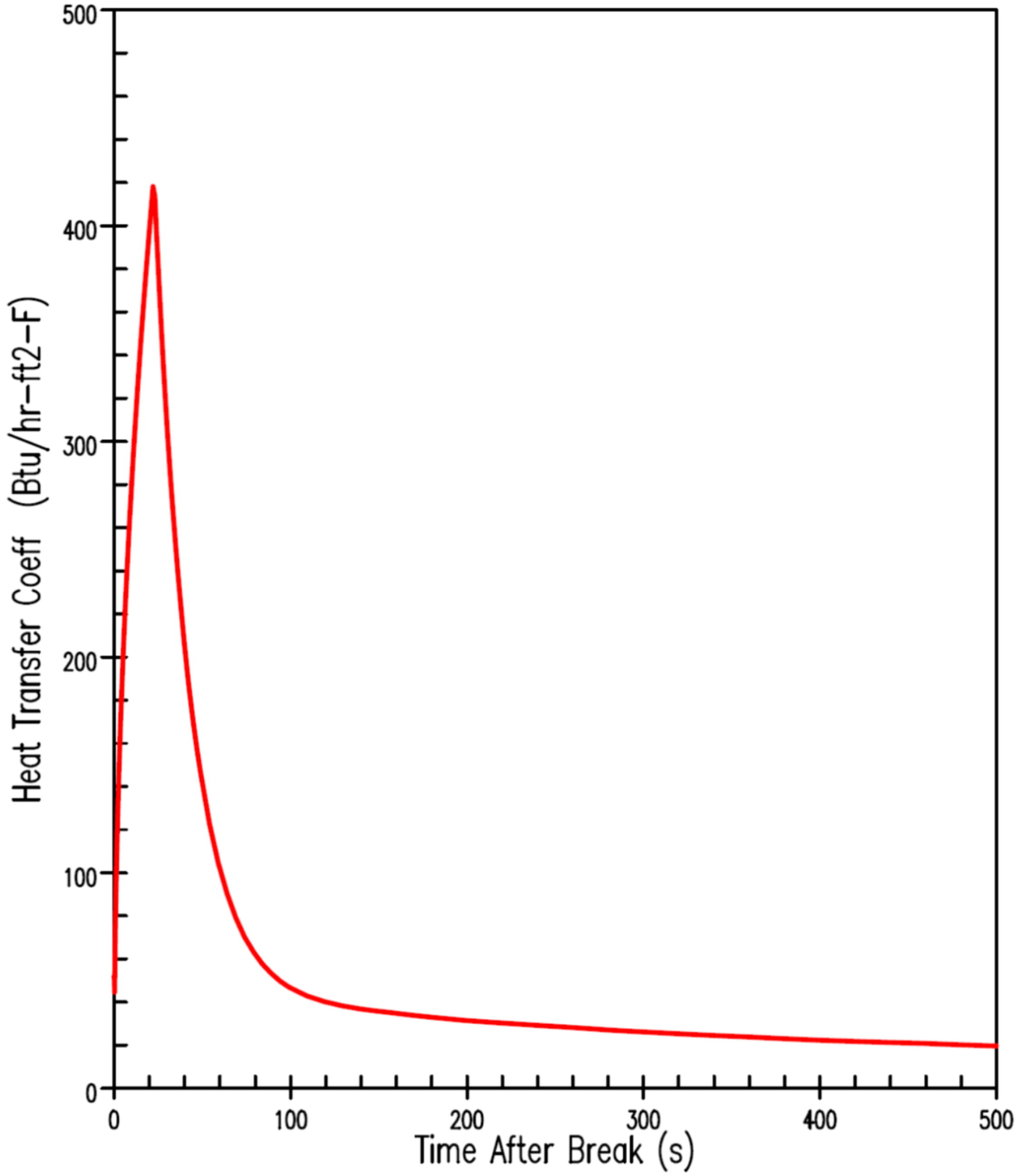


Figure 6.3-16
[DELETED]

Figure 6.3-17
NORTH ANNA UNITS 1 AND 2 BREAK ENERGY RELEASE

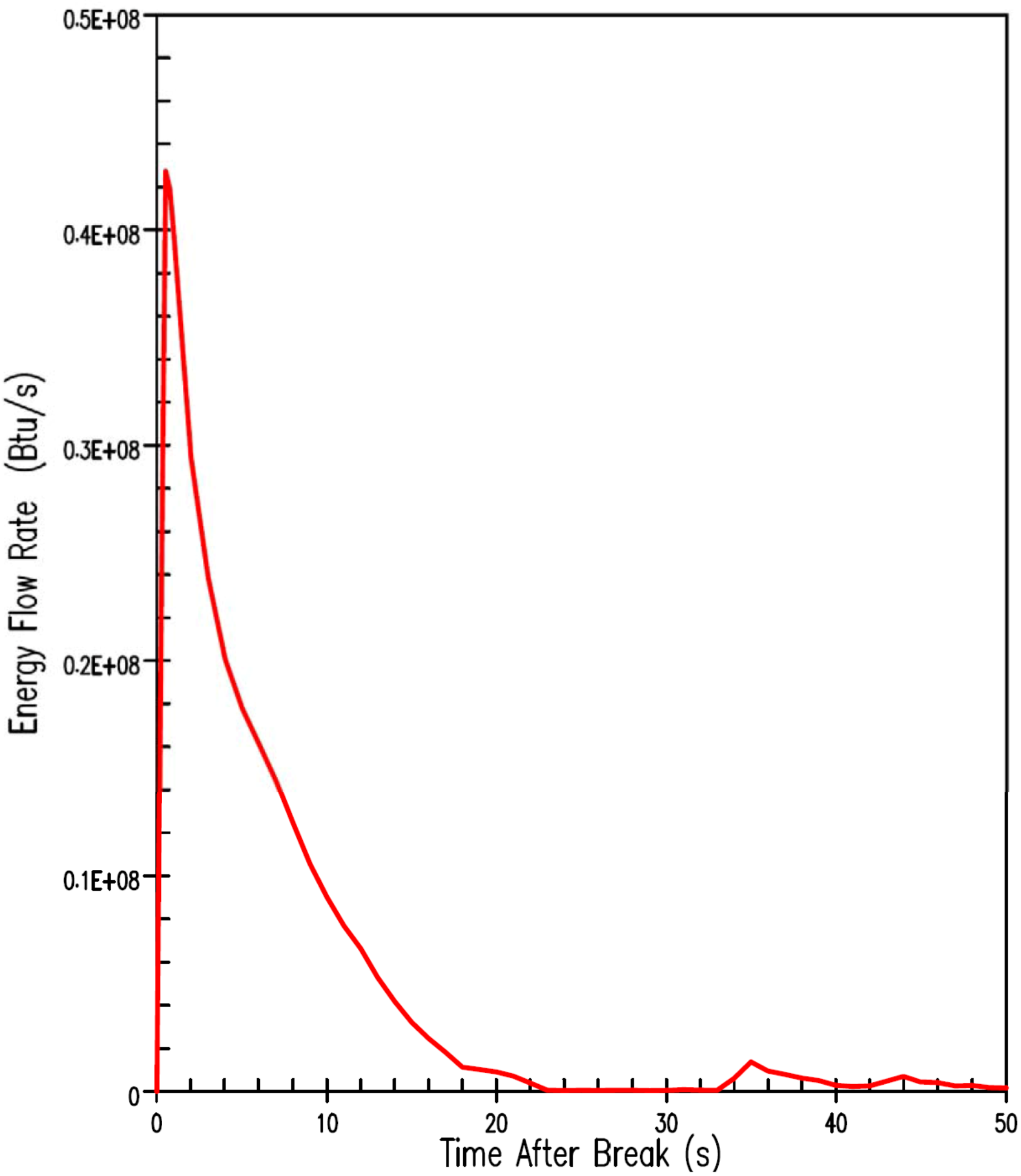


Figure 6.3-18
[DELETED]

6.4 HABITABILITY SYSTEMS

6.4.1 Habitability Systems Functional Design

6.4.1.1 Design Bases

The habitability systems for the control room are provided to ensure that continuous occupancy of the area is possible during and after natural phenomena, fire, and missiles, as discussed in Chapter 3, as well as for all postulated accidents, discussed in Chapter 15, that may or may not release radioactivity to the environs.

In accordance with GDC 19 as revised for alternative source terms (AST), personnel exposure is limited to 5 rem TEDE for the duration of an accident. All accidents, postulated in Chapter 15, that could release radioactivity to the atmosphere have been examined for their effect on the control room habitability.

Provisions are made to ensure that a continuous supply of breathable quality air is available in the MCR/ESGR envelope during accident conditions, as discussed in Section 9.4.1.

Redundant air-conditioning equipment maintains the required climatic conditions within the control room during normal and accident conditions. This equipment has been constructed and installed to Seismic Class I requirements and is tornado protected (Section 9.4.1).

Emergency diesel generators provide emergency power to all electrically powered motors and controls associated with the air-conditioning and ventilation systems in the event of a loss of offsite power under either normal operating or accident conditions.

The assumptions regarding the sources and amounts of radioactivity that leak into the control room following a steam-line break accident are discussed in Section 15.4.2.1.

Emergency lighting is provided in the control room, as discussed in Section 9.5.3.

The control room portion of the service building is a Seismic Class I structure and tornado missile protected.

When the MCR/ESGR envelope must be isolated in an emergency, supplemental breathing air is provided from a compressed air supply. Concurrently, the emergency ventilation starts in the recirculation configuration. Within one hour after an MCR/ESGR envelope isolation, the emergency ventilation system (EVS) is manually aligned to supply fresh, filtered air into the MCR/ESGR envelope. This ensures an adequate long-term supply of breathable air. Although neither the bottled air system, nor the pressurization of the MCR/ESGR envelope is credited in the accident or dose analysis, the EVS operation in this alignment and the bottled air system may provide a positive pressure and limit in-leakage in the envelope. The air-conditioning system is designed to provide uninterrupted service under normal and accident conditions (Section 9.4.1).

Fire hazards in the control room are minimized by the following measures:

1. The control room construction utilizes noncombustible materials.
2. Control cables and switchboard wiring are insulated with flame-resistant insulation. In addition, all control cables have an overall flame-resistant jacket with nonflammable fillers.
3. Furniture used in the control room is primarily of metal construction.
4. Combustible supplies, such as logs, records, procedures, drawings, and manuals, are limited to the amounts required for operation.
5. All areas of the control room are readily accessible in case of a fire, except the underfloor cable area, which is provided with a Halon 1301 automatic fire suppression system.
6. Adequate fire extinguishers and self-contained breathing equipment are provided.
7. The control room is occupied at all times.

6.4.1.2 System Design

The control room air-conditioning system arrangement with emergency outside air makeup through special high-efficiency particulate filters and charcoal adsorbers is shown on Figure 9.4-1 and Reference Drawing 1. Flow diagrams for the control room air-conditioning, chilled water, and condenser water systems are shown on Reference Drawings 2 and 3.

Ventilation system components, ducting, and dampers for the main control room and relay rooms are shown on Figures 6.4-1, 6.4-2, and 6.4-3. The locations of the emergency control room intakes are given below and on Figure 6.4-1.

A normal fresh air supply for the main control room is from Unit 1-HV-AC-4. This inlet is located near column line E, approximately 8 feet east of column line 4 at an elevation of 291 ft. 6 in.

The two emergency air inlets are 12-inch diameter pipes, located in the turbine building wall at column line C with the centerlines approximately 92 feet east of column line 9 (Unit 1) and 22 feet west of column line 9 (Unit 2), both at a centerline elevation of 290 ft. 11 in. Two additional inlets are located in the service building; one in each unit's air cond. chiller room. The manually selected suction path determines whether the emergency ventilation fans operate in the outside supply or recirculation configuration. Due to the location of the air inlet for 1-HV-F-41 with respect to vent stack B, it cannot be used to provide outside air for filtered supply.

Four (two per unit) compressed breathable air cylinder storage systems (bottled air) are available to discharge breathable quality air into the MCR/ESGR envelope. Any of the four systems are capable of providing a source of breathing quality air to limit the in-leakage of outside air into the MCR/ESGR envelope. Flow diagrams for the control room and relay room compressed dry air bottle system are shown on Reference Drawing 4.

The air-conditioning air-handling equipment, including the emergency air filters for the control room, is within the control room envelope at Elevation 276 ft. 9 in.

The air-conditioning air-handling equipment for the relay room is within the control room envelope at Elevation 254 ft. 0 in.

Upon receipt of a safety injection signal, a high-high radiation condition in the fuel building during fuel handling operations or manual actuation, the control and relay room normal outside air supply and exhaust fans are automatically shut down, and supply and exhaust dampers are closed, thus isolating the spaces; the bottled air supply system is initiated, and the emergency ventilation system fans start in the recirculation configuration. If the fuel pit bridge radiation monitor, 1-RM-RMS-153, is out of service; operating procedures require a set of compensatory actions to be in place prior to fuel handling operations.

Within the first hour after an accident the EVS is manually aligned to provide outside filtered air to the MCR/ESGR for an indefinite period. Each filtration system consists of a high-efficiency particulate air filter and a charcoal adsorber.

The high-efficiency particulate air (HEPA) filter bank is designed so that air flow through the standard 24 inches x 24 inches x 11 inches cell does not exceed approximately 1100 ft^3/min.

Air velocity through the charcoal adsorber at rated capacity provides a minimum gas residence time in the charcoal bed of 0.25 second. Activated carbon material, in accordance with ASTM D-3803, is provided to meet these gas flow and minimum residence time requirements.

The filter assembly components are fully accessible with ample space between components to permit access for filter inspection, testing, and maintenance.

Manufacturer testing demonstrates that HEPA filters are capable of removing a minimum of 99.97% thermally generated dioctylphthalate particulates at the design flow rate.

The charcoal adsorber section in each HEPA/charcoal filter assembly is tested as described in the Ventilation Filter Testing Program.

The control room floor is at Elevation 276 ft. 9 in. with the structural support floor depressed 18 inches to provide the control room underfloor cable area, except at the stairway, toilet, and the two air-conditioning equipment rooms. As the underfloor cable area is not readily accessible, it is protected by a two-zone Halon 1301 fire suppression system with supervised thermal and combustion product detection for alarm and annunciation with manual and automatic actuation.

Portable carbon dioxide extinguishers are provided for extinguishment of fire in the accessible control panels, and stored pressure water-type portable extinguishers are provided for Class A fire in solid waste or stored materials.

6.4.1.3 **Design Evaluation**

The control room air-conditioning system is designed to maintain a suitable environment for personnel and equipment during normal and emergency conditions. Two carbon dioxide monitors have been installed to verify carbon dioxide levels in the control rooms are at accepted habitability limits. One monitor is installed in Unit 1 control room and one is installed in Unit 2 control room. All components of the air-conditioning system are constructed and installed to Seismic Class I criteria, and they are housed in a building designed and constructed to satisfy Seismic Class I and tornado criteria, as listed in Table 3.2-1.

Redundant Seismic Class I chilled water systems are provided, as described in Section 9.4.1.2. All intake and exhaust openings are tornado missile protected. All outside air that enters the control room through the emergency air supply system will pass through an emergency HEPA/charcoal filter assembly. Cables and pipes entering the control room envelope are sealed to aid in maintaining the integrity of the control room envelope.

Initiation of the bottled air system will provide breathing quality air and assist in limiting the in-leakage of ambient area smoke or airborne radiation from entering the MCR/ESGR envelope after an accident. Within one hour after an accident, supply air is introduced through one of two emergency makeup air systems, each with HEPA/charcoal filter assembly and fan.

The design for the control room permits access to and occupancy of the control room under accident conditions and for the duration of the maximum credible accident without receiving radiation exposures in excess of the 10 CFR 50 Appendix A GDC 19 limit as revised for AST.

Auxiliary shutdown panels for both units are located in their respective relay rooms at Elevation 254 near column 8C. The panels (as discussed in Section 7.4) are utilized to bring the plant to hot standby condition in the unlikely event that the control room may need to be evacuated at a time when no additional accident conditions simultaneously occur, other than loss of external power, and that all automatic systems continue to function.

Exfiltration from the isolated control room is shown on Figure 6.4-1.

The 12,500 ft^3/min air-conditioning unit is part of the 100% recirculating air-handling system and has no connecting duct work to the outside. Therefore, no leakage occurs from the outside to this system.

During normal operation, the control room redundant emergency air supply systems, each with a HEPA/charcoal filter assembly, are on standby. These air supply systems have provisions to be manually configured in either a recirculation mode or an outside supply mode.

During an accident, emergency fans start automatically and concurrently with the discharge of the bottled air systems in the recirculation configuration. Within one hour after an accident, one emergency fan is manually placed in the outside supply mode, with outside air supplied only from

the turbine building through special filter assemblies for removal of particulates, elemental iodine, and iodine compounds. The filtered outside air is supplied to the return air plenum room upstream of the filters integral with each of the air-conditioning units.

As insignificant “dusting” of carbon fines are released from the charcoal adsorbers, no downstream HEPA filters have been provided.

Control room filtration systems in the outside supply configuration take suction from the turbine building. In the event of a main steam line break in the turbine building, water droplets and vapor may be encountered even 1 hour after the accident. To preclude degrading adsorber units with moisture, water separators are installed in the intake of the control room ventilation systems. Electric heaters are also installed to maintain relative humidity below 70%. These heaters are powered from emergency buses.

Table 6.2-45 compares each engineered safety feature air filtration system to each position in Regulatory Guide 1.52.

6.4.1.3.1 Evaluation of Radiological Protection

The following postulated accidents, described in Chapter 15, could result in some radioactivity release to the environs:

1. Waste gas decay tank rupture (Section 15.3.5)
2. Volume control tank rupture (Section 15.3.6)
3. Loss-of-coolant accident (LOCA) (Section 15.4.1)
4. Major secondary pipe rupture (Section 15.4.2)
5. Steam generator tube rupture (Section 15.4.3)
6. Fuel-handling accidents (Section 15.4.5)

For each of the above accidents, the dose to control room personnel in 30 days does not exceed the maximum dose limits in 10 CFR 50, Appendix A, GDC 19 as revised for AST.

Control room isolation and initiation of the bottled air system is designed to automatically actuate by a safety injection signal (SIS) or a high-high radiation condition in the fuel building during fuel handling operations. Most potential incidents that could cause high radiation in the fresh air inlet to the control room would result in the initiation of an SIS, thereby isolating the control room before the potentially contaminated air reaches the fresh air inlet. Actuation by a signal from the fresh air inlet would in effect be isolation after the contaminated air has entered the control room. If the control room should become contaminated for some unknown reason the area monitor will alarm and the operator can manually isolate the MCR/ESGR envelope to further reduce infiltration and initiate the EVS to provide filtered breathing air to the MCR/ESGR envelope.

Although a fuel handling incident or operation of the main steam safety valves would not cause an SIS to be generated, during all fuel handling the radiation levels in the fuel building will be monitored and the exhaust air can be filtered before being discharged. A high-high level fuel building radiation condition will automatically initiate the control room isolation and discharge of the bottled air system. Continuous communication is required between the control room operator and fuel handling coordinator during fuel movement inside containment. In the event of a fuel handling accident inside containment with the potential for release of radioactive material from the reactor cavity, the control room operator will be required to manually isolate the control room within two minutes to ensure that control room doses remain within GDC-19 as revised for AST criteria even if the containment purge and exhaust valves fail to close. Main steam safety valve operation with highly contaminated steam would more than likely be accompanied by a large primary to secondary leak, which would initiate a safety injection and automatic isolation of the control room. Again as a backup, should automatic isolation not take place, the area monitor in the control room would alarm and indicate to the operator to manually isolate the control room.

The design basis for the main control room shielding and the results of the calculated dose to control room personnel for the LOCA are described in Section 12.1.2.10. The analytical model equations and input parameters used in computing the control room doses due to direct radiation from airborne radioactivity outside the control room and the containment is described below.

The dose rate transmitted through shielding by gamma radiation emanating from a cloud of finite or infinite dimensions is calculated by the program SHLDCLD.

The cloud is assumed to be hemispherical and composed of radioisotopes that have leaked from a reactor containment after a hypothetical LOCA. The gamma rays from the isotopes contained in the cloud are grouped according to energy, and the dose rate due to each energy group is calculated, the total dose rate being the sum of these values. The method of calculation is to consider an element of volume of the cloud as an isotropic point source. A dose rate equation for this elemental source and dose point is developed using point-kernel techniques and buildup factors and is then integrated over the solid angle of the cloud using Simpson's Rule to give the dose rate from the entire cloud at the point of interest. This point is separated from the cloud by a flat shield whose radius is at least as large as the cloud radius. The dose point is on the cloud centerline and on the shield surface.

Input data are the number of energies present, and their values in MeV; the source term in MeV/cc-sec; the cloud radius; and the shield material and its thickness.

The computer prints out all input data, as well as absorption and attenuation coefficients, buildup factors, slant penetration factors, and time after the accident, and the dose rate due to each gamma energy and the total dose rate.

The direct gamma radiation contribution from the containment building to an external receptor is calculated by the program CONTAINMENT SHIELD. This program models the containment building in the following manner:

1. The reactor vessel and its internals is represented as an “object shield” for this calculation.
2. The circular concrete crane wall inside the containment is modeled in close resemblance to the actual wall.
3. The cylindrical containment wall is divided into 10 vertical belts of different thickness to model effects of ground, external walls, or internal structure.
4. The top section is hemispherical and of constant thickness.
5. The steel liner is included in both the cylindrical and hemispherical regions, by an equivalent amount of concrete.

The source strength spectral information is input to the program (10 energy groups). The program assumes uniform dispersion of the source in the containment space.

A point-kernel solution is used, having a series of equivalent point sources distributed throughout the containment. Attenuation in the containment atmosphere and in the air outside is considered. Buildup is also considered, using the Berget form of the buildup factor, best fit to 20 mean free paths, as provided in DASA-1892-3, *Weapons Radiation Shielding Handbook*.

For the specific case of control room doses due to direct radiation, the control room wall thickness has been added to the containment wall thickness, and all radial partitions and equipment in the containment have been neglected.

6.4.1.3.2 Evaluation of Fire Protection

The Halon 1301 suppression system provided for the control room underfloor area does not create a hazard to personnel committed to permanent occupancy in the control room. Halon 1301 is described in NFPA 12A-1973 as being classified by Underwriters’ Laboratories in Group 6 (least toxic).

Should both Halon 1301 zones be released into the control room underfloor area, the maximum concentration for a sealed control room would not exceed 1.5%. When outside air is introduced through the emergency air filter system at the design rate, the control room Halon concentration can be reduced by approximately one-half in 1 hour.

It is expected that only a small fraction of the extinguishing gas would actually enter the control room proper from the underfloor area. The effect of oxygen depletion and anesthesia on the personnel would, therefore, be inconsequential. In addition, portable breathing apparatus is available for use in the control and relay rooms.

A fire in the underfloor area would create toxic products of combustion if flame or surface temperature exceeds 900°F, which is the decomposition temperature of Halon 1301; however, these quantities would be very limited by the short response time of the detection and extinguishing system. The response time is expected to be relatively short due to the small volume of the area and sensitivity of the installed detection system; and any potential escape of toxic products is also limited by the small area of leakage in the false floor to the control room.

6.4.1.3.3 Evaluation of Toxic Chemical Protection

The design of the North Anna Units 1 and 2 control room complies with existing Regulatory Guides and Standard Review Plan providing requirements for protection of control room personnel from the hazards associated with toxic material. These requirements are outlined by Regulatory Guide 1.78, *Assumptions for Evaluating the Habitability of a Nuclear Power Plant Control Room During a Postulated Hazardous Chemical Release*; Regulatory Guide 1.95, *Protection of Nuclear Power Plant Control Room Operators Against an Accidental Chlorine Release*; and Standard Review Plan Sections 2.2.1, 2.2.2, 2.2.3, and 6.4. The compliance of North Anna Units 1 and 2 with these documents is described in the following paragraphs.

Standard Review Plan Sections 2.2.1, 2.2.2, 2.2.3, and 6.4 establish certain criteria to be used in reviewing compliance with Regulatory Guides 1.78 and 1.95.

Regulatory Guide 1.95 describes design features and procedures that would mitigate hazards to control room operators from an accidental chlorine release. No gaseous chlorine is currently used on site. Liquefied chlorine is not stored on site except small quantities for laboratory use, 20 lb or less, as allowed by the Regulatory Guide. Therefore, North Anna Units 1 and 2 comply with Regulatory Guide 1.95.

Regulatory Guide 1.78 requires that the habitability of the control room be evaluated for a postulated release of chemicals located within a 5-mile radius of the reactor facility. As stated in Section 2.2, there are no manufacturing plants, chemical plants and storage facilities, major water transportation routes, or oil and gas pipelines within 5 miles of the plant site. This leaves the following conditions to be evaluated: (1) chemicals shipped on roads within 5 miles of the plant at a frequency of 10 or more per year, and in weights as outlined by the Regulatory Guide, and (2) chemicals stored on site in a quantity greater than 100 lb.

The following roads pass within 5 miles of the plant site:

Road	Distance (miles)	Direction from Site
Secondary State Road 652	1-1/2	S
Secondary State Road 601	2	NE
Primary State Road 208	2	NW
U. S. Route 522	5	WNW

There are no specific data available on the types, quantities, and frequency of chemical shipments along these routes; however, considering the lack of chemical and industrial facilities along Routes 652, 601, and 208, and considering the distance between Route 522 and the plant site, it is expected that there are no chemicals shipped along these routes at a frequency and weight great enough to require evaluation in accordance with the Regulatory Guide, and certainly none that would pose a hazard to the habitability of the control room.

Table 6.4-1 lists potentially hazardous chemicals stored on the plant site in quantities greater than 100 lb. The table also presents results of the evaluations for the worst case accidental release of each type of chemical.

The evaluations for all chemicals stored onsite, as listed in Table 6.4-1, indicate that the worst-case concentration of the chemical within the control room is less than the toxicity limit for that chemical. All cases are evaluated on the basis of no action being taken by the control room operator.

The evaluation for ammonium hydroxide is not based on the equations from NUREG-0570 because they are inaccurate for vapor clouds of small quantity. These equations fail to reduce the concentration of the chemical in the vapor cloud at the air intake by the amount of the chemical that is drawn into the control room.

The only hydrazine storage container having a quantity greater than 100 lb is the 345 gallon bin of 35% hydrazine solution located in the Unit 2 Turbine Building, Elevation 303 ft. 0 in. The hydrazine bin is a closed refillable container. A nitrogen blanket caps the hydrazine within the bin. When the hydrazine is depleted, the bin is removed from the site and replaced with a new full bin. The hydrazine solution flows by gravity from the hydrazine bin to the 5-gallon measuring tank located in the vicinity of the chemical addition tanks and pumps. This case is evaluated in Table 6.4-1.

The 345 gallons of pH control solution up to 85% ethanolamine is also stored within a closed, refillable bin on the 303'-0" elevation of the Unit 2 Turbine Building. The bin is removed from the site and replaced with a full bin when the pH control solution ethanolamine is depleted.

The pH control solution ethanolamine flows by gravity from the bin to the 5-gallon measuring tank located in the vicinity of the chemical addition tanks and pumps. The effect of 345 gallons of pH control solution ethanolamine on Control Room Habitability is evaluated in Table 6.4-1.

One 55 gallon drum of hydrazine and one 55 gallon drum of pH control solution ethanolamine may also be stored on the 254'-0" elevation of the Unit 2 Turbine Building. These cases are bounded by the cases evaluated in Table 6.4-1.

A safety shower and an eye wash station are located in the immediate area of the chemical addition tanks for use if needed. The tanks and bins are located in an open area of the turbine building thereby providing adequate ventilation.

Based on the evaluations described above, North Anna Units 1 and 2 comply with the requirements for control room habitability as outlined by Regulatory Guides 1.78 and 1.95.

6.4.1.4 **Testing and Inspection**

The major items of equipment that maintain the habitability of the control room are the emergency filter assemblies, mechanical refrigeration, water chillers, fans, air-conditioning units, chilled water pumps, and compressed breathing air storage bottles.

The regular operating equipment for air conditioning of the control room is redundant for Unit 1, as well as for Unit 2. In addition, as described in Section 9.4.1.2, there is a third packaged water chiller for Units 1 and 2, which may be connected into either of the two respective redundant chilled water piping systems for the control room air-conditioning units.

The emergency systems, which do not normally operate, are tested as required by the Technical Specifications. The tests include

1. Isolation of the control room by closing all outside air dampers and valves.
2. EVS fans start upon isolation signal.

The filter assembly (including HEPA filters and charcoal (iodine) adsorbers of each emergency air system) is periodically tested as described by the Ventilation Filter Testing Program.

During integrity testing of the control room, all doors to the control room will be closed. All fans and other doors in areas surrounding the control room will be treated in a manner that will not bias the results of the test, unless such treatment can be justified.

Test and inspection of the Halon 1301 fire extinguishing system is conducted in accordance with the requirements of the American Nuclear Insurers (formerly Nuclear Energy Property Insurance Association) and the National Fire Protection Association (NFPA), as described in the North Anna Technical Requirements Manual (Reference Section 16.2).

6.4.1.5 **Instrumentation Requirement**

The control equipment for the control room ventilation and air-conditioning control room system is designed to Seismic Class I requirements and described in Sections 7.6.4, 7.7.1.12, and 9.4.1.5. The changeover to redundant ventilation and air-conditioning equipment is manual. The operator is alerted to manually switch to the emergency ventilation air supply on indication of low pressure of bottled air supply.

The radiation level in the main control room is measured by a fixed monitor to verify safe operating conditions. Portable monitors are also available to provide backup to the fixed monitors.

Local pressure differential gauges are provided to monitor the pressure differential between the MCR/ESGR envelope and adjoining areas. These gauges are used as an indicator of MCR/ESGR envelope integrity.

6.4 REFERENCE DRAWINGS

The list of Station Drawings below is provided for information only. The referenced drawings are not part of the UFSAR. This is not intended to be a complete listing of all Station Drawings referenced from this section of the UFSAR. The contents of Station Drawings are controlled by station procedure.

	Drawing Number	Description
1.	11715-FB-23A	Arrangement: Service Building, Ventilation, Sheet 1
2.	11715-FB-40A	Flow Diagram: Air Conditioning, Chilled Water Systems
3.	11715-FB-040D	Flow/Valve Operating Numbers Diagram: Air Conditioning Condenser Water Systems, Unit 1
4.	11715-FB-34F	Flow/Valve Operating Numbers Diagram: Compressed Dry Air Bottle System

Table 6.4-1

CHEMICAL STORAGE AND CONTROL ROOM HABITABILITY EVALUATIONS[a]

Chemical	Normal Storage Areas		Worst Case Exposure to Control Room			
	Quantity	Location	Distance From C.R. Air Intake	Type of Release	Maximum Concentration in C.R.	Toxicity Limit
Hydrogen	700 lb	Storage tank, N. of Turbine Bldg.	N/A	N/A	Hydrogen from any potential release would be dispersed to the atmosphere before reaching the control room air intake.	
Carbon dioxide	34,000 lb	1-FP-TK-5, N. of Turbine Bldg.	200 ft N. of intake	34,000 lb instantaneous	16,500 ppm[b, c]	10,000 ppm or asphyxiant [d]
Nitrogen	4500 lb	1-GN-TK-2, S. of Unit 1 Containment	420 ft S. of intake	4500 lb instantaneous	10,000 ppm[b, c]	Asphyxiant
Sulfuric acid	60 gal	Warehouse 7	N/A	N/A	c	
Ammonium hydroxide (30% solution by weight)	55 gal	Col. 15-C, Turbine Bldg.	40 ft below intake	55 gal instantaneous	e	100 ppm
	Mult. 55 gal	Warehouse 7	N/A	N/A		
Acetone	55 gal	Drum storage area adjacent to Warehouse 2	420 ft E. of intake	55 gal evaporation	180 ppm[b, c]	2000 ppm

a. Quantities less than 100 lb and chemicals stored and used in solid form are not listed.

b. Evaluations are based on control room normal ventilation intake and air exchange rate of 0.92 volume changes per hour for those chemicals so noted.

c. Evaluations based on the equations in NUREG-0570, *Toxic Vapor Concentrations in the Control Room Following a Postulated Accidental Release*. These equations and the assumptions used in the individual evaluations are very conservative. The actual concentration in the control room from a postulated chemical release is expected to be much less than the values indicated.

d. Carbon dioxide is an asphyxiant according to Sax's *Dangerous Properties of Industrial Materials-Fifth Edition* and as such would have a toxicity limit of 33%.

e. The worst case exposure to the control room from ammonium hydroxide would be the rupture of a 55-gallon drum at the chemical addition area in the turbine building concurrent with turbine building ventilation system failure and with the control room emergency ventilation system operating. In order to achieve an ammonia concentration of 100 ppm in the control room in this case, greater than 17% of the ammonia release would have to be brought into the control room through the air intake. Considering the density of ammonia, the turbulent air flow that would exist in the turbine building, and the distance from the point of release to the air intake, it is not believed possible to achieve a concentration of 100 ppm in the control room.

f. The distance from the Control Room air intake is not a factor in the calculation. It was assumed that the entire quantity of the chemical evaporates directly into the air intake.

g. A toxicity limit is not given in Regulatory Guide 1.78, but is defined as being the maximum concentration that can be tolerated for 2 min without physical incapacitation of the average human (i.e., severe coughing, eye burn, or severe skin irritation). The best indicator of this is the "immediately dangerous to life and health" (IDLH) limit as given in the table.

h. The 55 gal spill of Hydrazine is bounded by the 345 gal evaluation.

i. These chemicals pose a minimal inhalation hazard in their liquid or vapor forms. Dust particles or fumes generated from burning are hazardous; however, a fire concurrent with catastrophic spill is not required to be postulated.

j. All chemical properties and toxicity limits for H-130 microbiocide are not available; however, the inhalation hazard is low based on the MSDS for this chemical. Ethanol, a component of H-130, was evaluated and the resulting MCR concentration following a spill was found to be less than the toxicity limit for Ethanol.

k. Existing quantity is bounded by the original calculation.

l. While the maximum concentration of liquid alum (ref. to aluminum sulfate) is slightly higher than its respective toxicity limit, this is acceptable. The permissible exposure limit (PEL) used, for the most part, is the same as the threshold limit value (TLV). The TLV of a chemical substance is a level to which it is believed a worker can be exposed day after day for a working lifetime without adverse health effects. This would be significantly lower than a 2-minute toxicity limit.

m. Reference Calc. Number ME-0567 Rev. 1 Add. F for all additional chemical assumptions and conclusions.

Table 6.4-1 (continued)

CHEMICAL STORAGE AND CONTROL ROOM HABITABILITY EVALUATIONS[a]

Chemical	Normal Storage Areas		Worst Case Exposure to Control Room			
	Quantity	Location	Distance From C.R. Air Intake	Type of Release	Maximum Concentration in C.R.	Toxicity Limit
Hydrazine (35% solution by weight)	345 gal	Column 14-C, El. 303', Unit 2 Turbine Bldg.	N/A [f]	345 gal evaporation	5.33 ppm[c]	50.0 ppm[g]
	55 gal	Column 15-C, Ground Floor, Unit 2 Turbine Bldg.	40 ft below intake	55 gal evaporation	h	g
	345 gal	Warehouse 7	N/A	N/A		
Ethanolamine (100% solution by weight)	345 gal	Column 14-C, El. 303', Unit 2 Turbine Bldg.	N/A [f]	345 gal evaporation	0.72 ppm[c]	30 ppm[g]
	55 gal	Col. 15-C, Unit 2 Turbine Bldg.	40 ft below intake	55 gal evaporation	2.04 ppm[c]	30 ppm[g]
	345 gal	Warehouse 7	N/A	N/A		
Boric acid (12,950 - 15,750 ppm)	7500 gal	1-CH-TK-1A, Col. 9-H, Aux. Bldg.	N/A	N/A	Insignificant vapor given off at ambient temperature	
	7500 gal	1-CH-TK-1B, Col. 9-H, Aux. Bldg.				
	7500 gal	1-CH-TK-1C, Col. 9-H, Aux. Bldg.				

a. Quantities less than 100 lb and chemicals stored and used in solid form are not listed.

b. Evaluations are based on control room normal ventilation intake and air exchange rate of 0.92 volume changes per hour for those chemicals so noted.

c. Evaluations based on the equations in NUREG-0570, *Toxic Vapor Concentrations in the Control Room Following a Postulated Accidental Release*. These equations and the assumptions used in the individual evaluations are very conservative. The actual concentration in the control room from a postulated chemical release is expected to be much less than the values indicated.

d. Carbon dioxide is an asphyxiant according to Sax's *Dangerous Properties of Industrial Materials-Fifth Edition* and as such would have a toxicity limit of 33%.

e. The worst case exposure to the control room from ammonium hydroxide would be the rupture of a 55-gallon drum at the chemical addition area in the turbine building concurrent with turbine building ventilation system failure and with the control room emergency ventilation system operating. In order to achieve an ammonia concentration of 100 ppm in the control room in this case, greater than 17% of the ammonia release would have to be brought into the control room through the air intake. Considering the density of ammonia, the turbulent air flow that would exist in the turbine building, and the distance from the point of release to the air intake, it is not believed possible to achieve a concentration of 100 ppm in the control room.

f. The distance from the Control Room air intake is not a factor in the calculation. It was assumed that the entire quantity of the chemical evaporates directly into the air intake.

g. A toxicity limit is not given in Regulatory Guide 1.78, but is defined as being the maximum concentration that can be tolerated for 2 min without physical incapacitation of the average human (i.e., severe coughing, eye burn, or severe skin irritation). The best indicator of this is the "immediately dangerous to life and health" (IDLH) limit as given in the table.

h. The 55 gal spill of Hydrazine is bounded by the 345 gal evaluation.

i. These chemicals pose a minimal inhalation hazard in their liquid or vapor forms. Dust particles or fumes generated from burning are hazardous; however, a fire concurrent with catastrophic spill is not required to be postulated.

j. All chemical properties and toxicity limits for H-130 microbiocide are not available; however, the inhalation hazard is low based on the MSDS for this chemical. Ethanol, a component of H-130, was evaluated and the resulting MCR concentration following a spill was found to be less than the toxicity limit for Ethanol.

k. Existing quantity is bounded by the original calculation.

l. While the maximum concentration of liquid alum (ref. to aluminum sulfate) is slightly higher than its respective toxicity limit, this is acceptable. The permissible exposure limit (PEL) used, for the most part, is the same as the threshold limit value (TLV). The TLV of a chemical substance is a level to which it is believed a worker can be exposed day after day for a working lifetime without adverse health effects. This would be significantly lower than a 2-minute toxicity limit.

m. Reference Calc. Number ME-0567 Rev. 1 Add. F for all additional chemical assumptions and conclusions.

Table 6.4-1 (continued)

CHEMICAL STORAGE AND CONTROL ROOM HABITABILITY EVALUATIONS[a]

Chemical	Normal Storage Areas		Worst Case Exposure to Control Room			
	Quantity	Location	Distance From C.R. Air Intake	Type of Release	Maximum Concentration in C.R.	Toxicity Limit
Sodium hydroxide (50% solution by weight)	600 gal	GE Trailer	N/A	600 gal evaporation	2 mg/m^3	10 mg/m^3 [g]
Sodium hypochlorite (15% solution by weight)	Mult. 55 gal	Warehouse 7	N/A	N/A	Insignificant vapor given off at ambient temperature	
	400 gal	Bearing Cooling Tower	N/A	N/A	Insignificant vapor given off at ambient temperature	
	Mult. 400 gal	Warehouse 7	N/A	N/A	Insignificant vapor given off at ambient temperature	
Sodium bromide (30-60% solution by weight)	400 gal	Bearing Cooling Tower	N/A	N/A	Insignificant vapor given off at ambient temperature	
	Mult. 400 gal	Warehouse 7	N/A	N/A	Insignificant vapor given off at ambient temperature	
H-130 Microbiocide	2000 gal	1-SW-TK-4, SW Chemical Addition Bldg.	N/A	2000 gal evaporation	c, j	1000 ppm

a. Quantities less than 100 lb and chemicals stored and used in solid form are not listed.

b. Evaluations are based on control room normal ventilation intake and air exchange rate of 0.92 volume changes per hour for those chemicals so noted.

c. Evaluations based on the equations in NUREG-0570, *Toxic Vapor Concentrations in the Control Room Following a Postulated Accidental Release.* These equations and the assumptions used in the individual evaluations are very conservative. The actual concentration in the control room from a postulated chemical release is expected to be much less than the values indicated.

d. Carbon dioxide is an asphyxiant according to Sax's *Dangerous Properties of Industrial Materials-Fifth Edition* and as such would have a toxicity limit of 33%.

e. The worst case exposure to the control room from ammonium hydroxide would be the rupture of a 55-gallon drum at the chemical addition area in the turbine building concurrent with turbine building ventilation system failure and with the control room emergency ventilation system operating. In order to achieve an ammonia concentration of 100 ppm in the control room in this case, greater than 17% of the ammonia release would have to be brought into the control room through the air intake. Considering the density of ammonia, the turbulent air flow that would exist in the turbine building, and the distance from the point of release to the air intake, it is not believed possible to achieve a concentration of 100 ppm in the control room.

f. The distance from the Control Room air intake is not a factor in the calculation. It was assumed that the entire quantity of the chemical evaporates directly into the air intake.

g. A toxicity limit is not given in Regulatory Guide 1.78, but is defined as being the maximum concentration that can be tolerated for 2 min without physical incapacitation of the average human (i.e., severe coughing, eye burn, or severe skin irritation). The best indicator of this is the "immediately dangerous to life and health" (IDLH) limit as given in the table.

h. The 55 gal spill of Hydrazine is bounded by the 345 gal evaluation.

i. These chemicals pose a minimal inhalation hazard in their liquid or vapor forms. Dust particles or fumes generated from burning are hazardous; however, a fire concurrent with catastrophic spill is not required to be postulated.

j. All chemical properties and toxicity limits for H-130 microbiocide are not available; however, the inhalation hazard is low based on the MSDS for this chemical. Ethanol, a component of H-130, was evaluated and the resulting MCR concentration following a spill was found to be less than the toxicity limit for Ethanol.

k. Existing quantity is bounded by the original calculation.

l. While the maximum concentration of liquid alum (ref. to aluminum sulfate) is slightly higher than its respective toxicity limit, this is acceptable. The permissible exposure limit (PEL) used, for the most part, is the same as the threshold limit value (TLV). The TLV of a chemical substance is a level to which it is believed a worker can be exposed day after day for a working lifetime without adverse health effects. This would be significantly lower than a 2-minute toxicity limit.

m. Reference Calc. Number ME-0567 Rev. 1 Add. F for all additional chemical assumptions and conclusions.

Table 6.4-1 (continued)
CHEMICAL STORAGE AND CONTROL ROOM HABITABILITY EVALUATIONS[a]

Chemical	Normal Storage Areas		Worst Case Exposure to Control Room			
	Quantity	Location	Distance From C.R. Air Intake	Type of Release	Maximum Concentration in C.R.	Toxicity Limit
Isothiazolin Biocide (4.5% solution by weight)	400 gal[k]	Bearing Cooling Tower	N/A[f]	1000 gal[k] evaporation	0.0072 mg/m^3 [c]	0.1 mg/m^3
	Mult. 55 gal	Warehouse 7	N/A	N/A	N/A	0.1 mg/m^3
Hydrochloric Acid (31% solution by weight)	55 gal	GE Trailer	N/A[f]	55 gal evaporation	7.63 ppm	50 ppm[g]
Hydrogen peroxide (35% solution by weight)	250 gal	GE Trailer	N/A	250 gal evaporation	46.6 ppm	75 ppm[g]
Dianodic DN2472	≤ 3000 gal	Bearing Cooling Tower	N/A	N/A	minimal inhalation hazard	
	3000 gal	1-BC-TK-1, Ground Floor Unit 1 Turbine Bldg	N/A	N/A	minimal inhalation hazard	
	Mult. containers ≤ 3000 gal	Warehouse 7	N/A	N/A	minimal inhalation hazard	
Stoddard Solvent	55 gal	Paint Shop	N/A	55 gal evaporation	123.69 mg/m^3	20,000 mg/m^3 [m]

a. Quantities less than 100 lb and chemicals stored and used in solid form are not listed.
b. Evaluations are based on control room normal ventilation intake and air exchange rate of 0.92 volume changes per hour for those chemicals so noted.
c. Evaluations based on the equations in NUREG-0570, *Toxic Vapor Concentrations in the Control Room Following a Postulated Accidental Release*. These equations and the assumptions used in the individual evaluations are very conservative. The actual concentration in the control room from a postulated chemical release is expected to be much less than the values indicated.
d. Carbon dioxide is an asphyxiant according to Sax's *Dangerous Properties of Industrial Materials-Fifth Edition* and as such would have a toxicity limit of 33%.
e. The worst case exposure to the control room from ammonium hydroxide would be the rupture of a 55-gallon drum at the chemical addition area in the turbine building concurrent with turbine building ventilation system failure and with the control room emergency ventilation system operating. In order to achieve an ammonia concentration of 100 ppm in the control room in this case, greater than 17% of the ammonia release would have to be brought into the control room through the air intake. Considering the density of ammonia, the turbulent air flow that would exist in the turbine building, and the distance from the point of release to the air intake, it is not believed possible to achieve a concentration of 100 ppm in the control room.
f. The distance from the Control Room air intake is not a factor in the calculation. It was assumed that the entire quantity of the chemical evaporates directly into the air intake.
g. A toxicity limit is not given in Regulatory Guide 1.78, but is defined as being the maximum concentration that can be tolerated for 2 min without physical incapacitation of the average human (i.e., severe coughing, eye burn, or severe skin irritation). The best indicator of this is the "immediately dangerous to life and health" (IDLH) limit as given in the table.
h. The 55 gal spill of Hydrazine is bounded by the 345 gal evaluation.
i. These chemicals pose a minimal inhalation hazard in their liquid or vapor forms. Dust particles or fumes generated from burning are hazardous; however, a fire concurrent with catastrophic spill is not required to be postulated.
j. All chemical properties and toxicity limits for H-130 microbiocide are not available; however, the inhalation hazard is low based on the MSDS for this chemical. Ethanol, a component of H-130, was evaluated and the resulting MCR concentration following a spill was found to be less than the toxicity limit for Ethanol.
k. Existing quantity is bounded by the original calculation.
l. While the maximum concentration of liquid alum (ref. to aluminum sulfate) is slightly higher than its respective toxicity limit, this is acceptable. The permissible exposure limit (PEL) used, for the most part, is the same as the threshold limit value (TLV). The TLV of a chemical substance is a level to which it is believed a worker can be exposed day after day for a working lifetime without adverse health effects. This would be significantly lower than a 2-minute toxicity limit.
m. Reference Calc. Number ME-0567 Rev. 1 Add. F for all additional chemical assumptions and conclusions.

Table 6.4-1 (continued)

CHEMICAL STORAGE AND CONTROL ROOM HABITABILITY EVALUATIONS[a]

Chemical	Normal Storage Areas		Worst Case Exposure to Control Room			
	Quantity	Location	Distance From C.R. Air Intake	Type of Release	Maximum Concentration in C.R.	Toxicity Limit
Solvent Naphtha (Aromatic Hydrocarbon)	55 gal	Paint Shop	N/A	55 gal evaporation	2.66 ppm	19 ppm[m]
1,2,4-Trimethylbenzene	55 gal	Paint Shop	N/A	55 gal evaporation	8.67 ppm	25 ppm[m]
Xylene	55 gal	Paint Shop	N/A	55 gal evaporation	38.84 ppm	900 ppm[m]
Ethylbenzene	55 gal	Paint Shop	N/A	55 gal evaporation	28.77 ppm	800 ppm[m]
1-Methoxy-2-Propanol	55 gal	Paint Shop	N/A	55 gal evaporation	50.35 ppm	150 ppm[m]
n-Butyl Acetate	55 gal	Paint Shop	N/A	55 gal evaporation	63.87 ppm	1,700 ppm[m]
Butanone	55 gal	Paint Shop	N/A	55 gal evaporation	428.63 ppm	3,000 ppm[m]
Cumene	55 gal	Paint Shop	N/A	55 gal evaporation	34.13 ppm	900 ppm[m]

a. Quantities less than 100 lb and chemicals stored and used in solid form are not listed.

b. Evaluations are based on control room normal ventilation intake and air exchange rate of 0.92 volume changes per hour for those chemicals so noted.

c. Evaluations based on the equations in NUREG-0570, *Toxic Vapor Concentrations in the Control Room Following a Postulated Accidental Release.* These equations and the assumptions used in the individual evaluations are very conservative. The actual concentration in the control room from a postulated chemical release is expected to be much less than the values indicated.

d. Carbon dioxide is an asphyxiant according to Sax's *Dangerous Properties of Industrial Materials-Fifth Edition* and as such would have a toxicity limit of 33%.

e. The worst case exposure to the control room from ammonium hydroxide would be the rupture of a 55-gallon drum at the chemical addition area in the turbine building concurrent with turbine building ventilation system failure and with the control room emergency ventilation system operating. In order to achieve an ammonia concentration of 100 ppm in the control room in this case, greater than 17% of the ammonia release would have to be brought into the control room through the air intake. Considering the density of ammonia, the turbulent air flow that would exist in the turbine building, and the distance from the point of release to the air intake, it is not believed possible to achieve a concentration of 100 ppm in the control room.

f. The distance from the Control Room air intake is not a factor in the calculation. It was assumed that the entire quantity of the chemical evaporates directly into the air intake.

g. A toxicity limit is not given in Regulatory Guide 1.78, but is defined as being the maximum concentration that can be tolerated for 2 min without physical incapacitation of the average human (i.e., severe coughing, eye burn, or severe skin irritation). The best indicator of this is the "immediately dangerous to life and health" (IDLH) limit as given in the table.

h. The 55 gal spill of Hydrazine is bounded by the 345 gal evaluation.

i. These chemicals pose a minimal inhalation hazard in their liquid or vapor forms. Dust particles or fumes generated from burning are hazardous; however, a fire concurrent with catastrophic spill is not required to be postulated.

j. All chemical properties and toxicity limits for H-130 microbiocide are not available; however, the inhalation hazard is low based on the MSDS for this chemical. Ethanol, a component of H-130, was evaluated and the resulting MCR concentration following a spill was found to be less than the toxicity limit for Ethanol.

k. Existing quantity is bounded by the original calculation.

l. While the maximum concentration of liquid alum (ref. to aluminum sulfate) is slightly higher than its respective toxicity limit, this is acceptable. The permissible exposure limit (PEL) used, for the most part, is the same as the threshold limit value (TLV). The TLV of a chemical substance is a level to which it is believed a worker can be exposed day after day for a working lifetime without adverse health effects. This would be significantly lower than a 2-minute toxicity limit.

m. Reference Calc. Number ME-0567 Rev. 1 Add. F for all additional chemical assumptions and conclusions.

Table 6.4-1 (continued)
CHEMICAL STORAGE AND CONTROL ROOM HABITABILITY EVALUATIONS[a]

Chemical	Normal Storage Areas		Worst Case Exposure to Control Room			
	Quantity	Location	Distance From C.R. Air Intake	Type of Release	Maximum Concentration in C.R.	Toxicity Limit
Methyl Pyrrolidone	1	Paint Shop	N/A	1 gal evaporation	0.03 ppm	10 ppm[m]
Propylene Glycol Monomethyl Ether	1	Paint Shop	N/A	1 gal evaporation	0.97 ppm	150 ppm[m]
2-Butoxyethanol	1	Paint Shop	N/A	1 gal evaporation	0.06 ppm	700 ppm[m]
Methyl Isobutyl Ketone	1	Paint Shop	N/A	1 gal evaporation	1.26 ppm	500 ppm[m]
Sodium Hydroxide	600	RO System Warehouse 7	N/A	600 gal evaporation	1.39 mg/m^3	10 mg/m^3[m]
Aluminum Sulfate	1100	RO System Warehouse 7	N/A	1100 gal evaporation	2.25 mg/m^3	2 mg/m^3[l,m]
Hydrochloric (Muriatic) Acid	25	RO System	N/A	25 gal evaporation	14.23 ppm	50 ppm[m]
Hydrogen Peroxide	600	RO System Warehouse 7	N/A	600 gal evaporation	6.49 ppm	75 ppm[m]

a. Quantities less than 100 lb and chemicals stored and used in solid form are not listed.
b. Evaluations are based on control room normal ventilation intake and air exchange rate of 0.92 volume changes per hour for those chemicals so noted.
c. Evaluations based on the equations in NUREG-0570, *Toxic Vapor Concentrations in the Control Room Following a Postulated Accidental Release.* These equations and the assumptions used in the individual evaluations are very conservative. The actual concentration in the control room from a postulated chemical release is expected to be much less than the values indicated.
d. Carbon dioxide is an asphyxiant according to Sax's *Dangerous Properties of Industrial Materials-Fifth Edition* and as such would have a toxicity limit of 33%.
e. The worst case exposure to the control room from ammonium hydroxide would be the rupture of a 55-gallon drum at the chemical addition area in the turbine building concurrent with turbine building ventilation system failure and with the control room emergency ventilation system operating. In order to achieve an ammonia concentration of 100 ppm in the control room in this case, greater than 17% of the ammonia release would have to be brought into the control room through the air intake. Considering the density of ammonia, the turbulent air flow that would exist in the turbine building, and the distance from the point of release to the air intake, it is not believed possible to achieve a concentration of 100 ppm in the control room.
f. The distance from the Control Room air intake is not a factor in the calculation. It was assumed that the entire quantity of the chemical evaporates directly into the air intake.
g. A toxicity limit is not given in Regulatory Guide 1.78, but is defined as being the maximum concentration that can be tolerated for 2 min without physical incapacitation of the average human (i.e., severe coughing, eye burn, or severe skin irritation). The best indicator of this is the "immediately dangerous to life and health" (IDLH) limit as given in the table.
h. The 55 gal spill of Hydrazine is bounded by the 345 gal evaluation.
i. These chemicals pose a minimal inhalation hazard in their liquid or vapor forms. Dust particles or fumes generated from burning are hazardous; however, a fire concurrent with catastrophic spill is not required to be postulated.
j. All chemical properties and toxicity limits for H-130 microbiocide are not available; however, the inhalation hazard is low based on the MSDS for this chemical. Ethanol, a component of H-130, was evaluated and the resulting MCR concentration following a spill was found to be less than the toxicity limit for Ethanol.
k. Existing quantity is bounded by the original calculation.
l. While the maximum concentration of liquid alum (ref. to aluminum sulfate) is slightly higher than its respective toxicity limit, this is acceptable. The permissible exposure limit (PEL) used, for the most part, is the same as the threshold limit value (TLV). The TLV of a chemical substance is a level to which it is believed a worker can be exposed day after day for a working lifetime without adverse health effects. This would be significantly lower than a 2-minute toxicity limit.
m. Reference Calc. Number ME-0567 Rev. 1 Add. F for all additional chemical assumptions and conclusions.

Withheld under 10 CFR 2.390 (d) (1)

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Withheld under 10 CFR 2.390 (d) (1)

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Withheld under 10 CFR 2.390 (d) (1)

[redacted]

Appendix 6A
Single Failure Capability[1]

1. This Appendix was Appendix 6A in the original North Anna FSAR.

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6A.1 DEFINITIONS OF TERMS

Definitions of terms used in this appendix are located in Section 3.1.

6A.2 ACTIVE-FAILURE CRITERIA

The emergency core cooling system (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.A-1, 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 similar to that following a loss-of-coolant accident (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.

6A.3 PASSIVE-FAILURE CRITERIA

The following philosophy provides for necessary redundancy in component and system arrangement to meet the intent of the Atomic Energy Commission (AEC) 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 providing an active failure has not occurred during the short term.

6A.3.1 Redundancy of Flow Paths and Components for Long-Term Emergency Core Cooling

In design of the ECCS, Westinghouse utilizes the following criteria. During the long-term cooling period following a LOCA, the ECCS has sufficient redundancy to deliver adequate flow to the core in the event of any postulated passive failure. Should a passive failure in the ECCS occur in the ECCS outside the containment, means are available to remotely terminate that leakage while maintaining the minimum core cooling function via alternate recirculation paths.

6A.3.2 Subsequent Leakage from Components in Engineered Safety Features System

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 that flows less than 50 gpm would result. Piping leaks, valve packing leaks, or flange gasket leaks have tended 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:

1. The system piping is located within a controlled area on the plant site.
2. The piping system receives periodic pressure tests and is accessible for periodic visual inspection.
3. 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 was based upon handling of leaks up to a maximum of 50 gpm.

With these design ground rules, continued function of the ECCS will meet minimum core cooling requirements.

Per NUREG-800 Section 15.6.5, Appendix B, the dose consequences of a 50 gpm of passive components failure was not considered in Section 15.4.1.9 analysis since the failure would occur in the area where ESF filtration system exists.

A single-passive-failure analysis is presented in Table 6.A-2. 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 that failed.

Table 6.A-1
SINGLE ACTIVE FAILURE ANALYSIS FOR EMERGENCY CORE COOLING SYSTEM COMPONENTS

Component	Malfunction	Comments
	I. Short-Term Phase	
A. Accumulator	Deliver to broken loop	Totally passive system with one accumulator per loop. Evaluation based on one spilling accumulator
B. Pump		
1. Centrifugal charging	Fails to start	Three provided. Evaluation based on operation of one.
2. Low-head safety injection	Fails to start	Two provided. Evaluation based on operation of one.
C. Automatically operated valves		
1. Boron injection tank isolation		
a. Inlet	Fails to open	Two parallel lines; one valve in either line is required to open.
b. Outlet	Fails to open	Two parallel lines; one valve in either line is required to open.
2. 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 is required to close
c. Bypass line	Fails to close	Two valves in series; only one valve is required to close.
d. Suction from volume control tank	Fails to close	Two valves in series; only one valve is required to close.

Table 6.A-1 (continued)
SINGLE ACTIVE FAILURE ANALYSIS FOR EMERGENCY CORE COOLING SYSTEM COMPONENTS

Component	Malfunction	Comments
II. Long-Term Phase		
A. Valves operated from control room for recirculation		
1. Containment sump recirculation isolation	Fails to open	Two lines parallel; only one valve in either line is required to open.
2. Low-head safety injection pump suction line to refueling water storage tank	Fails to close	Check valve in series with gate valve, operation of only one valve required.
3. 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.
4. Centrifugal charging pump suction line from discharge of low-head safety injection pump	Fails to open	Separate and independent paths from discharge of low-head safety injection pumps. One valve in either path is required to open.
5. Low-head safety injection pump cold-leg injection header isolation valve	Fails to close	Separate and redundant paths from discharge of low-head safety injection pumps. One valve in either path is required to close.
6. Low-head safety injection pump hot-leg recirculation header isolation valve	Fails to open	Separate and redundant paths from discharge of low-head safety injection pumps. One valve in either path is required to open.

Table 6.A-1 (continued)
SINGLE ACTIVE FAILURE ANALYSIS FOR EMERGENCY CORE COOLING SYSTEM COMPONENTS

Component	Malfunction	Comments
III. Long-Term Phase (continued)		
A. Valves operated from control room for recirculation (continued)		
7. Centrifugal charging pump cold-leg recirculation isolation valves	Fails to open	Separate and redundant paths from discharge of charging/safety injection pumps. One path opened by parallel valves by “S” signal and redundant path opened for recirculation.
8. Centrifugal charging pump hot-leg recirculation isolation valves	Fails to open	Separate and redundant paths from discharge of charging/safety injection pumps. One valve in either path is required to open.
B. Pumps		
1. Charging pump	Fails to operate	Same as short-term phase.
2. Low-head safety injection pumps	Fails to operate	Two provided, evaluation based on operation of one.

Table 6.A-2
EMERGENCY COOLING SYSTEM RECIRCULATION PIPING PASSIVE-FAILURE ANALYSIS, LONG-TERM PHASE

Flow Path	Indication of Loss of Flow Path	Alternate Flow Path
Low-head recirculation		
From containment sump to low-head injection header via the low-head safety injection pumps	Accumulation of water in a low-head safety injection pump compartment or auxiliary building sump	Via the independent low-head flow path utilizing the second low-head pump
High-head recirculation		
From containment sump to the high-head injection header via low-head safety injection pump, and the high-head injection pumps	Accumulation of water in a low-head safety injection pump compartment or the auxiliary building sump	From containment sump to the high-head injection headers via alternate low-head safety injection pump and the alternate high-head charging pump