6.2 CONTAINMENT SYSTEMS

6.2.1 CONTAINMENT FUNCTIONAL DESIGN

6.2.1.1 Containment Structure

6.2.1.1.1 Design Bases

The containment systems protect the public from the consequences of any credible break in the Reactor Coolant System. The containment systems consist of the steel containment vessel surrounded by the Shield Building, and the engineered safety feature systems which include the Safety Injection System, the Containment Heat Removal System (Containment Spray System and Containment Cooling System), the Shield Building Ventilation System, the Containment Isolation System, the Combustible Gas Control System and the Controlled Ventilation Area System.

The Shield Building provides biological shielding and controlled release of the annulus atmosphere under accident conditions, and environmental missile protection for the containment vessel and the Nuclear Steam Supply System. A physical description of the containment and the design criteria relating to construction techniques, static loads, and seismic loads are provided or referenced in Section 3.8. This section pertains to those aspects of containment design, testing, and evaluation that relate to the accident mitigation function.

The containment vessel is designed to withstand the pressure and temperature transients calculated to exist after a design basis accident (DBA). Post accident conditions are determined by evaluating the combined influence of the energy sources, heat sinks and engineered safety features (ESF) operation.

The capability of the containment vessel to maintain design leaktight integrity and to provide a predictable environment for operation of ESF systems is ensured by a comprehensive design, analysis, and testing program. This program considers the results of both the peak containment pressures and temperatures coincident with the safe shutdown earthquake (SSE), and the maximum containment external pressure resulting from inadvertent Containment Heat Removal System operations that reduce containment internal pressure below the Shield Building Annulus pressure.

→(DRN 01-230, R12; 04-705, R14)

The containment systems are designed to provide protection to the public from the consequences of a loss of coolant accident (LOCA) up to and including a break size equal to twice the area of the largest reactor coolant pipe coincident with the SSE, loss of offsite power, and any single component failure. The containment structure and the engineered safety features ensure that the radiological exposure to the public resulting from such an occurrence is below the guidelines established in 10CFR50.67.

←(DRN 01-230, R12; 04-705, R14)

→(DRN 01-230, R12)
The spectrum of postulated accidents considered in determining the design containment peak pressure and temperature, the subcompartment peak pressure, the containment external pressure, and the ECCS minimum containment pressure are summarized in Table 6.2-1. For postulated subcompartment pipe break accidents, a discussion of the criteria for selecting break locations is given in Subsection 6.2.1.2.
←(DRN 01-230, R12)
The accident controlling design for each of the categories of containment peak pressure, containment peak temperature, subcompartment peak pressure, containment external pressure, and containment minimum pressure is defined as a DBA and is that case which produces the most severe loadings for the spectrum of accidents postulated. The margin between values calculated for a DBA and design values is established by comparing Tables 6.2-2 and 6.2-3. Table 6.2-2 defines the DBA for each design category and Table 6.2-3 gives the margin and the formula for calculating margin.

For the containment peak pressure analysis, subcompartment peak pressure analyses, the containment peak temperature analysis, and the ECCS minimum containment pressure analysis, it is assumed that each accident is concurrent with the most limiting single failure. No two accidents are assumed to occur simultaneously or consecutively.
→(DRN 01-230, R12; 04-631, R13-B; 03-2060, R14; EC-8458, R307)
For LOCA cases, it is assumed that each accident is concurrent with a loss of offsite power and the most limiting single active failure loss of a containment heat removal train. For the MSLB cases, the offsite power is assumed to be available since availability of offsite power is more limiting. Single active failures assumed for MSLB cases include: (1) failure of one main feedwater isolation valve (MFIV) to close, (2) failure of one train of containment heat removal system to operate and (3) failure of one main steam isolation valve to close.
←(DRN 04-631, R13-B)

The containment pressure and temperature response for replacement steam generator (RSG) project are analyzed for limiting LOCA cases and a complete set of MSLB event from various initial power levels and three different single failures. The analyses resulted in the following limiting LOCA and MSLB cases as described in Table 6.2-1:
←(EC-8458, R307)

- LOCA peak pressure case - Double ended hot leg slot break (DEHLSB) blowdown pressure at 3735 MWt (3716 MWt + 0.5% uncertainty)

- LOCA worst case 24 hour pressure - Double ended discharge leg slot break with minimum safety injection flow at 3735 MWt (3716 MWt + 0.5% uncertainty)

→(EC-8458, R307)

- MSLB peak pressure case – MSLB event from hot zero power (HZP) conditions with the failure of one main steam isolation valve (MSIV) to close.

- MSLB peak temperature case – MSLB event from hot full power (HFP) conditions with failure of one MSIV to close.

←(DRN 03-2060, R14; EC-8458, R307)
For both LOCA and MSLB events only one containment fan cooler per train is assumed operable.
→(EC-8458, R307)
The tables containing the time dependent mass and energy release data for the above limiting accidents under the categories of containment peak pressure and temperature analyses are referenced in Table 6.2-1. The computer codes and assumptions used in deriving the mass and energy release data are discussed in Subsections 6.2.1.3 and 6.2.1.4.
←(EC-8458, R307)

Energy released to the containment atmosphere as a result of the postulated accidents is removed by the Containment Heat Removal System discussed in Subsection 6.2.2 and the passive heat sinks in the containment provided in Tables 6.2-7 and 6.2-8. Each train of the Containment Heat Removal System (consisting of one spray pump, one set of spray headers, one shutdown heat exchanger and two fan coolers (one CFC per train is assumed operable)) provides the required heat removal rate to maintain the
←(DRN 01-230, R12)

→(DRN 01-230)
containment peak pressure below the design pressure of 44 psig and is capable of reducing post-LOCA pressure to less than 50 percent of the containment peak calculated pressure within 24 hours following the postulated accident.
←(DRN 01-230)

→(DRN 01-230)
←(DRN 01-230)

The analysis of containment minimum pressure is based on confirming the ECCS core reflood capability under the conservative set of assumptions that maximize the heat removal effectiveness of ESF systems, structural heat sinks, and other potential heat removal processes. These assumptions are discussed in Subsection 6.2.1.5.

6.2.1.1.2 Design Features

The design bases and design measures taken to assure that the containment structure is adequately protected against the dynamic effects of postulated missiles and pipe ruptures are discussed in Sections 3.5 and 3.6 respectively. The codes, standards, and guides applied in the design of the containment structure and internal structures are described in Section 3.8.

Two redundant containment vacuum breakers have been provided for protection against loss of containment integrity under external loading conditions. Calculations of containment pressure following an inadvertent operation of the Containment Spray System have resulted in pressures within the containment design external allowable pressure. Details of this evaluation are provided in Subsection 6.2.1.1.3. The margin between calculated and design pressure differentials is shown in Table 6.2-3.
→(DRN 01-230)
The Containment Cooling System, discussed in Subsection 6.2.2, maintains the containment and subcompartment atmospheres within required pressure and temperature limits during normal plant operation. This system recirculates air in the containment through fan coolers (with air to water heat exchangers). The Containment Cooling System operates continuously during normal plant operations and is functionally capable of maintaining the pressure, temperature, and humidity within the limits used for equipment design and assumed for DBA analyses.
←(DRN 01-230)
The systems used for normal containment ventilation include the Containment Atmosphere Purge System and the Containment Cooling System. These systems are fully discussed in Section 9.4 and Subsection 6.2.2, respectively. No dependence is placed on the large containment purge maintaining containment temperature and humidity during normal operation. The limiting containment conditions for normal plant operation are contained in the plant technical specifications.

A Safety Injection System sump is provided to collect water for the recirculation mode. Refer to Subsection 6.2.2.2. The bottom of the SIS sump is at elevation -16 ft. MSL, which is below the lowest floor elevation inside containment except for the reactor vessel cavity and the containment sump.

6.2.1.1.3 Design Evaluation - Containment Pressure - Temperature Analysis
→(DRN 01-230)
In the event of a postulated loss-of-coolant accident (LOCA) or main steam line break (MSLB), the release of coolant from the rupture area causes the high temperature, high pressure fluid to flash to steam. This release of mass and energy raises the temperature and pressure of the containment atmosphere. The severity of the resulting containment peak temperature and pressure depends upon the nature, locations, and size of the postulated rupture.
←(DRN 01-230)

→(DRN 01-230, R12; 03-2060, R14; EC-8458, R307)
In order to establish the controlling LOCA and MSLB for containment design, a series of primary and secondary breaks described in Table 6.2-1 are analyzed to determine their significance in selecting the containment design basis accidents. A complete MSLB analysis was done for the RSG. A range of operating power levels (0%, 25%, 50%, 75% and 100%) were evaluated. Three different single failures were analyzed: (1) failure of one MFIV to close, (2) failure of one train of containment heat removal system to operate, and (3) failure of one MSIV to close. Table 6.2-4 presents the results for the limiting LOCA and MSLB cases, i.e., LOCA peak pressure, LOCA peak pressure at 24 hours, MSLB peak pressure, and MSLB peak temperature. Table 6.2-4 also provides the calculated peak pressure, peak temperature, time of peak pressure, time to end of blowdown, and total LOCA energy release during the blowdown phase.
←(DRN 01-230, R12; 03-2060, R14; EC-8458, R307)

The calculated transients following a postulated accident are a direct consequence of the energy balance within the containment. Of particular importance are the initial conditions postulated at the start of the accident, the ability of the heat sinks within the containment to absorb energy during the accident, and the capability of the Containment Heat Removal System to reduce the total energy within the containment thus bringing the containment heat sinks, sump water, and atmosphere into thermal equilibrium.

→(DRN 01-230, R12)
The containment pressure and temperature response analyses following LOCA and MSLB events are performed using GOTHIC (Generation of Thermal-Hydraulic Information for Containments) computer program developed by Numerical Applications, Inc. (References 28 and 29).

→(DRN 03-2060, R14; EC-8458, R307)
GOTHIC is a general purpose thermal-hydraulic computer program for design, licensing, safety and operating analysis of nuclear power plant containments and other confinement buildings. GOTHIC predicts both the pressure and temperature within the containment regions and the temperatures in the containment structures. The mass and energy release data from the break for all three LOCA phases (Blowdown, Reflood/Post-reflood and Long term boil-off) and MSLB events were generated by Westinghouse (formerly ABB CE) as described in Subsections 6.2.1.3 and 6.2.1.4. The mass and energy data were used as input to GOTHIC for containment pressure and temperature response analyses.
←(EC-8458, R307)

The following provides the values for some important parameters used in the limiting LOCA and MSLB analyses:

→(EC-8458, R307)

Initial containment temperature (°F)	120.0
Initial containment pressure (psia)	15.7
Initial containment spray riser level (ft MSL)	149.5
Component Cooling Water flow to containment fan coolers (gpm)	1100.0
Component Cooling Water temperature (°F)	115.0
RWSP water temperature (°F)	100.0

←(EC-8458, R307)

→(DRN 05-1769, R14-A)
An extra delay time of approximately 1.0 seconds is incorporated into the limiting LOCA and MSLB analyses to account for (1) existence of non-condensable gases in the containment spray (CS) piping and (2) the uncertainty associated with water level measurement in the CS header riser.
←(DRN 01-230, R12; 03-2060, R14; 05-1769, R14-A)

LOCA Analysis

➔(DRN 01-230, R12; 03-2060, R14; EC-8458, R307)

Containment pressure and temperature response for the limiting LOCA peak pressure (Double Ended Hot Leg Slot (DEHLS) break) and limiting 24 hour pressure (Double Ended Discharge Leg Slot (DEDLS) break with minimum safety injection) were analyzed for the RSG.

←(EC-8458, R307)

Initial Conditions and Input Data

➔(EC-8458, R307)

The initial conditions within the containment and Reactor Coolant System prior to LOCA initiation are given in Table 6.2-5. For the peak pressure DBA, the minimum containment volume and the maximum allowed initial pressure (one psig) are assumed. The initial amount of energy present in the containment is maximized by assuming the maximum containment operating temperature and lowest possible humidity. The maximum operating temperature is assumed to exist in all heat sinks. The containment vessel walls are assumed to be insulated at their outside surface to minimize heat transfer during the postulated accident. Also no leakage out of the containment is assumed.

←(DRN 03-2060, R14; EC-8458, R307)

Experience has shown that the highest-peak pressure is reached in the containment under conditions where the amount of energy initially within the containment is maximized and the ability of the containment atmosphere to absorb energy is minimized. For this reason the conservative initial conditions of fifty percent humidity and maximum initial temperature were chosen for the analyses. A lower initial temperature or a higher initial humidity results in increased energy absorption by the containment atmosphere since an increase in the amount of water vapor in the containment atmosphere increases the amount of energy the atmosphere could absorb.

←(DRN 01-230, R12)

➔(DRN 01-230, R12; 03-2060, R14)

For the purpose of the LOCA analyses, the ECCS (Emergency Core Cooling System) and the Containment Heat Removal Systems (CHRS) were assumed to operate in the mode that maximized the containment pressure response. For the Safety Injection system, both maximum and minimum safety injection flows were assumed for calculating the containment peak pressure response. For the CHRS, minimum system capacity is conservative for calculating containment peak pressure response.

➔((DRN 01-230, R12)

Therefore, the loss of one CHRS train was assumed to be the most conservative single failure for the LOCA analyses assuming either minimum or maximum safety injection flows.

The following describes the conservative assumptions made with respect to ESF (Engineered Safety Feature) system operations and parameters (see Table 6.2-6) for the LOCA maximum pressure and temperature analyses:

➔(EC-8458, R307)

a) One containment spray pump operates and sprays 1,750 gpm of water from the Refueling Water Storage Pool (RWSP) into the containment until the start of recirculation.

←(EC-8458, R307)

b) One shutdown cooling heat exchanger operates during the recirculation mode of operation to cool the containment spray. The heat exchanger is assumed to be supplied with cooling water flowing at 2550 gpm with a component cooling water temperature of (115°F).

←(DRN 01-230, R12; 03-2060, R14)

→(DRN 01-230, R12; EC-8458, R307)

c) The time until initiation of the recirculation mode is calculated on the basis of a minimum usable refueling water storage pool volume (383,000 gallons) transferred to the containment. This volume is assumed to be used during an accident before the recirculation actuation signal (RAS) is generated. At this time the low pressure safety injection pump is deenergized and the spray water is taken from the sump and is cooled by the shutdown cooling heat exchanger.

←(EC-8458, R307)

The timing of RAS depends on the break location and the assumption of maximum or minimum safety injection. The following provides the RAS time for different LOCA scenarios:

→(EC-8458, R307)

	Timing of RAS (Sec)	
	Maximum SI Flow	Minimum SI Flow
Hot Leg Break	1823.58	3171.25
Suction Leg Break	1839.15	3179.19
Discharge Leg Break	1881.39	3253.19

←(EC-8458, R307)

→(DRN 03-2060, R14)

d) The Containment Spray System starts spraying at 33.60 seconds (includes 1 second signal processing time) and 1.03 seconds extra delay for possible existence of noncondensible gas in the CS piping) and reaches full flow at 41.25 seconds after the containment reaches 19.7 psia (5 psig).

←(DRN 03-2060, R14)

e) Similarly the containment fan cooler will start 32.0 seconds (includes 1 second signal processing time) after the containment reaches the CIAS set point pressure (5 psig assumed).

f) Table 6.2-12C provides the nitrogen mass release and temperature from the Safety Injection tanks into the containment used in all LOCA analyses. The temperature of the expanded nitrogen provided in Table 6.2-12C is very low, for conservatism a constant temperature of 120°F is assumed.

The sizing of the Containment Heat Removal System was based on the heat removal rate necessary to keep the peak pressure reached during a LOCA less than the design pressure of the containment. The limiting LOCA peak pressure occurs during the blowdown phase of a DEHLS break.

←(DRN 01-230, R12)

The containment heat sink and heat source data used in accident analyses are described in Tables 6.2-7 and 6.2-8. Table 6.2-7 is a detailed list of the geometry of each heat sink and Table 6.2-8 describes the resulting simplified heat sink models used for computer input. Node spacing used for concrete, steel, and steel-lined concrete heat sinks is fine enough to ensure an accurate representation of the thermal gradient in each slab.

The given values for surface area and thickness reflect the total areas and surface-area weighted thickness for all steel exposed to the containment atmosphere from all sources. These sources include structural steel, polar crane and moving platform structures, instrumentation and control equipment (cabinet, tubes, etc.), hydrogen recombiners, HVAC equipment (ducts, fan coolers, valves), refueling machine, miscellaneous piping, and containment penetration nozzles.

→(DRN 01-230, R12)

All steel is coated with a layer of paint and finisher. All steel is assumed to be insulated on one side and in contact with the containment atmosphere (by a condensing heat transfer correlation) on the other.

←(DRN 01-230, R12)

The given values of the concrete for surface area and thickness reflect total areas and surface area weighted thickness of all concrete within the containment. This concrete includes the unlined pool wall, primary and secondary shield walls, pressurizer room, reheat exchanger room, valve room, pipe tunnel, containment sump pump wall, and the steam generator foundation.

All concrete is coated with paint. The concrete is assumed to have a zero temperature gradient at the center and is exposed to the containment atmosphere with a condensing heat transfer coefficient or to the sump water with a free convection heat transfer coefficient.

A complete list of the thermophysical properties used in the analysis is given on Table 6.2-8. No credit is taken for heat transfer to reinforcing steel in the internal concrete structures and a low value of thermal conductivity is used for these structures.

➔(DRN 01-230, R12)
←(DRN 01-230, R12)

➔(DRN 01-230, R12; 03-2060, R14; EC-8458, R307)
LOCA Results

Figures 6.2-1m and 6.2-1n provide the containment pressure and temperature for the limiting peak pressure LOCA case (Double Ended Hot Leg Slot Break). After the LOCA, the containment pressure rapidly increases and reaches its peak (54.9 psia, 40.2 psig at about 18.5 seconds). Continued heat removal by the passive heat sinks in the containment and the Containment Heat Removal System (CHRS) result in a decrease in pressure. Unlike a cold leg break, for the hot leg break most of the reflood fluid does not pass through a steam generator prior to being released to the containment; hence, there is no physical mechanism to rapidly remove the residual steam generator secondary energy during or after the reflood. Therefore, the pressure continues to decrease until the recirculation mode. At the start of the recirculation mode, water is drawn from the SIS sump by the high pressure safety injection pump and returned to the core. Simultaneously the spray pump takes suction from the sump and sprays the water back into the containment after passing it though the shutdown cooling heat exchanger (SDCHX). As a result of the higher HPSI pump inlet temperature, steam continues to be generated in the reactor core at a high rate due primarily to the release of decay heat and stored energy in the system internals. This causes the containment pressure to rise. The higher temperature of the recirculated containment spray water contributes to this pressure rise by reducing the ability of the sprays to remove heat from the containment atmosphere. This rise in pressure is reversed when the combined SDCHX, containment fan cooler and structural heat removal rate become greater than the net heat addition to the containment.

Figures 6.2-1p and 6.2-1q provide the containment pressure and containment vapor and sump water temperature for the limiting 24 hour pressure LOCA case (Double Ended Discharge Leg Slot break with minimum safety injection flow) respectively. Similar to the hot leg break, the pressure rises rapidly after the break and reaches its first peak value of 51.3 psia (36.6 psig) near end of the blowdown phase. Continued heat removal by the passive heat sinks and the CHRS result in a decrease of the first pressure peak. However, unlike the hot let break, the break flow passes through the steam generator and rapidly removes the steam generator secondary residual heat during the reflood/post-reflood phase prior to being released into the containment. This high energy release causes the pressure to rise to a maximum peak value of 51.8 psia (37.1 psig) at about 661 seconds. Continued heat absorption by the heat sinks and heat removal systems result in a decrease of this second pressure peak. The rise in containment pressure after about 3000 seconds is due to the higher spray water temperature from the safety injection sump inside containment (and therefore lower heat removal by containment spray) which occurs with Recirculation. The containment pressure behaves similar to hot leg break after the start of the recirculation mode. The containment temperature behavior is similar to pressure.
←(DRN 01-230, R12; 03-2060, R14; EC-8458, R307)

➔(DRN 01-230, R12)
←(DRN 01-230, R12)

A discussion of the computer codes and assumptions used in generating the LOCA mass and energy release rates are included in Subsection 6.2.1.3.

Accident chronologies for the most severe Reactor Coolant System breaks are provided in Table 6.2-10. It is assumed that time equals zero at the start of each accident.

→(DRN 01-230, R12)
←(DRN 01-230, R12)

→(DRN 01-230, R12; 03-2060, R14; EC-8458, R307)

Main Steam Line Breaks

To determine the limiting main steam line break (MSLB) event that results in peak containment pressure and peak containment temperature, the mass and energy release data for the MSLB events were generated for MSLB events from various initial power levels and three different single failures. The power levels included: (1) Hot Full Power (HFP), (2) seventy five percent power (75pp), (3) fifty percent power (50pp), (4) twenty five percent power (25pp), (6) Hot Zero Power (HZP) and (7) an equipment qualification (EQ) case (peak containment temperature case) from HFP conditions.

The three single failures assumed are:

- Failure of one Main Feedwater Isolation Valve (MFIV) to close,
- Failure of one Main Steam Isolation Valve (MSIV) to close, and
- Loss of One Containment Cooling Train (OCT)

The limiting peak containment pressure was determined to be the MSLB from hot zero power (HZP) with failure of one MSIV to close. The limiting peak containment temperature (Equipment Qualification (EQ) case) case was determined to be the MSLB from hot full power (HFP) with failure of one MSIV to close. The mass and energy release data for the EQ case assumes the lowest allowed initial containment pressure, steam superheating upon uncovering of the steam generator tubes and 8% re-evaporation of condensate from the passive heat sinks in the containment. For conservatism the 8% pre-evaporation was not considered in the containment response analysis.
←(EC-8458, R307)

Initial Conditions and Input Data

The initial conditions, e.g., initial containment pressure and temperature, initial heat sink temperatures and other input parameters such as CS flow rate, RWSP water temperature and containment fan cooler heat removal rate as a function of containment temperature are the same as the data used in the LOCA analyses. The following describes the specific input and assumptions used in the MSLB analyses:

- 8 seconds is assumed for the MSIV and MFIV closure time. The 8 seconds include 7 seconds valve closure time plus one second signal processing delay time.

→(EC-8458, R307)

- For the case where one MSIV is assumed to fail to close, the volume of the steam in the main steam line from MSIV to MSIV is assumed to be released to the containment.

←(EC-8458, R307)

- Maximum RCS flow rate is assumed, because it allows the maximum possible heat transfer between the primary and secondary system.

→(EC-8458, R307)

- The break size used is the largest possible for a given power level that produces a pure steam (dry) blowdown.

←(EC-8458, R307)

- As in the LOCA cases, one CFC per train is assumed operable.

←(DRN 01-230, R12; 03-2060, R14)

→(DRN 01-230, R12; 03-2060, R14)

- For MSIV and MFIV failure cases, both trains of the containment heat removal system are credited.

→(EC-8458, R307)

- Since the most severe MSLB containment transients are developed assuming the availability of offsite power, the containment sprays (CS) start spraying water into the containment at approximately 23.00 seconds (this includes 1 second signal processing time and 1.03 seconds extra delay for possible existence of noncondensible gas in the CS piping). Full containment spray flow is reached at 32 seconds after containment spray actuation signal (CSAS) conditions are reached in the containment.
- Consistent with the assumption of availability of offsite power, the containment fan coolers are assumed to be in full operation 20.5 seconds after the containment is calculated to reach the safety injection actuation signal (SIAS) setpoint (conservatively assumed to be 19.7 psia).

MSLB Results

The peak containment pressure of 56.3 psia (41.6 psig) is calculated to occur following a MSLB from HZP conditions with failure of one MSIV to close. The peak containment temperature (Equipment Qualification case) of 382.4°F is calculated to occur following a MSLB form HFP conditions with failure of one MSIV to close. The peak temperature case (EQ case) assumes the lowest allowed initial containment pressure of 14.275 psia. The mass and energy release data calculation for EQ cases assumes, steam superheating upon uncovering of the steam generator tubes and 8% re-evaporation of condensate from the passive heat sinks in the containment. For conservatism the 8% re-evaporation was not considered in the containment response analysis.

The containment pressure and temperature response for the limiting MSLB pressure and temperature cases are shown on Figures 6.2-1r and 6.2-1s respectively. Table 6.2-2 provides the peak pressure and temperature results for the limiting MSLB events.
←(DRN 01-230, R12; 03-2060; EC-8458, R307)

→(DRN 01-230, R12)
←(DRN 01-230, R12)

For all MSLBs analyzed, following blowdown of the ruptured steam generator unit, the RCS decay heat is transferred to the intact unit which, in turn, vents to the atmosphere when its safety relief valves open. Therefore, there is no physical mechanism for the release of significant amounts of mass or energy to the containment after the end of blowdown. Main feedwater line breaks (MFWLB) are not analyzed since such breaks result in a blowdown less limiting than the MSLB because the pipe break mass flow for the MFWLB is limited by the steam generator internals design. Fluid enthalpy for the MFWLB is also less than the enthalpy of the fluid in the MSLB.

A discussion of the computer codes, and the assumptions, including all assumed single active failures, used in deriving the MSLB mass and energy releases are discussed in Subsection 6.2.1.4.

→(DRN 01-230, R12)
Accident chronologies for the most severe secondary system breaks are provided in Table 6.2-10. It is assumed that time equals zero at the start of the accident.
←(DRN 01-230, R12)

→(EC-8458, R307)
←(EC-8458, R307)

➔(DRN 01-230, R12; 03-2060, R14; EC-8458, R307)

Environmental Design of Mechanical and Electrical Equipment

A list of equipment required to function during and subsequent to any design basis accident is provided in Table 3.11-2. The 3735 MWt composite LOCA & MSLB profile is not bounded by the original composite profile for the time period between 60 seconds and 110 seconds into the transient. However, using the Arrhenius method (Arrhenius Methodology is an NRC accepted method of equating one set of time and temperature parameters to another), the original composite temperature curve has been shown to bound the 3735 MWt composite curve. Since the peak calculated temperature for the MSLB used in the equipment qualification study (413.3°F) bounds the new MSLB peak temperature, the equipment qualification evaluation is not updated with the new MSLB mass and energy release data. The mass and energy release data for the MSLB analysis used in the equipment qualification study is retained in Table 6.2-12F (original mass and energy release).

🡐(DRN 03-2060, R14; EC-8458, R307)

It is necessary to estimate the velocity currents on the surface of the vital equipment in order to calculate the convection heat transfer coefficient for equipment qualification studies. To calculate this velocity on the containment vessel, the SOLA [19] code was used. This code numerically solves the Navier-Stokes equations for two-dimensional incompressible flow. A model of the containment vessel and the relevant internal components was set up as shown in Figure 6.2.32d. The scaling of the containment interior was five ft. for each node. The steam line break flow direction, with time, is indicated on Figures 6.2-32e through 32h. The velocity at the nozzle corresponds to a choked flow at approximately 950 ft/sec as indicated by MSLB blowdown data contained in Table 6.2-12.

🡐(DRN 01-230, R12)

The following conservatisms are incorporated in the analysis:

a) The angle of flow from the steam line break is perpendicular to the containment cylinder.

b) The flow obstruction caused by the steel support structure around the upper half of the steam generator was ignored.

c) Flow obstruction caused by gratings at elevations +46 ft. MSL and +21 ft. MSL were ignored.

d) Flow obstruction caused by the containment structures (polar crane, piping, etc.) were ignored.

e) Containment Cooling System mixing was ignored.

The flow velocity at the surface of the cylindrical containment vessel is time and spatially dependent (see Figure 6.2-32a). The maximum velocity at the containment vessel surface occurred in the SOLA model at 0.38 sec. This value is used to calculate the average velocity at the containment vessel surface during the MSLB blowdown. This average is multiplied by the maximum contact area of the jet on the containment cylinder surface divided by the total upper containment cylindrical area where the SOLA code calculates the containment surface velocities to be negligible. To be more conservative, ten ft. of the dome region is included in the height of the containment cylinder for the calculation of the jet and total cylindrical areas. The surface area of the remainder of the containment dome is conservatively ignored in the calculation of surface velocity. The velocity is calculated as follows:

Elevation +150 ft. MSL - assumed top of containment cylinder

Elevation +80 ft. MSL - point where MSLB flow is perpendicular to cylinder

Elevation +50 ft. MSL - lowest elevation of upper containment (actual location of grating)

R = (EL +150 ft. MSL) - (EL +80 FT. MSL) = 70 ft.

$$A_{eff} = \pi R^2 - \frac{1}{2} R^2 \left(\pi \frac{\alpha}{180} - Sin\alpha \right)$$

$$A_{eff} = 11{,}767\, ft^2$$

H = (EL +150 ft. MSL) - (EL +50 ft. MSL) = 100 ft

$$A_T = 2\pi RH = 43{,}982\, ft^2$$

→

since: $\overline{V_M} = 590\, ft / \sec$

←

$$\overline{V_A} = \int V dA / dA$$

→

$$\overline{V_A} \cong \frac{V_M A_{eff}}{A_T} = \frac{590(11{,}767)}{43{,}982}$$

since: $V_A \cong 160\, ft / \sec$

←

H	=	Height of upper containment cylinder
A_{eff}	=	Maximum effective jet area (upper containment cylinder)
A_T	=	Total upper containment cylindrical area
$\overline{V_M}$	=	Maximum average instantaneous velocity profile
$\overline{V_A}$	=	Area weighted velocity average

α = Angle used to calculate the area subtracted from the total jet area on the containment cylinder due to the presence of the grating at El +50 ft. MSL (approximately 129°).

The heat transfer coefficient used for the safety related component's surfaces exposed to the containment vapor region is the mass diffusion heat transfer coefficient. [20] It is a realistic method of calculating the heat sink condensing heat and mass transfer rates which include the effects of steam mass diffusion and of heat convection to the heat sink surface. Following the methodology discussed in Reference 20, the equation for the total heat removal rate to a heat sink surface can be written as:

$$\dot{q}_{total} = \dot{q}_{convection} + \dot{q}_{condensate}$$

where:

$$\dot{q}_{convection} = h_{conv} A \left(T_{vapor} - T_{sur} \right)$$

$$\dot{q}_{condensation} = \dot{m}_{conden} \left(h_g - h_f \right)$$

T_{vapor} = containment atmosphere temperature

T_{sur} = heat sink surface temperature

h_g = saturated steam ethalpy at vapor-film interface temperature

h_f = saturated liquid ethalpy at vapor-film interface temperature

$\dot{m}_{conden}$ = steam condensation rate

The convective heat transfer coefficient used for the analysis is estimated from the following relation for turbulent forced convection on a surface:

$Nu = 0.036\ Pr^{1/3}\ Re^{0.8}$

To evaluate the heat transfer correlations it is necessary to obtain a reference temperature which is defined as: [21]

$$T_{ref} = T_{sur} + (T_{vapor} - T_{sur})/3$$

The safety related components presently modeled include:

a) Electronic pressure transmitter (above El +46 ft. MSL)

b) Electric instrumentation cable (above El +46 ft. MSL)

These components (See Figure 6.2-34a) are modeled as unit area or unit length (such as cable) one-dimensional slab type heat sinks. The initial bulk temperature is assumed to be 120°F. Figure 6.2-32b shows the geometry and thermal properties of the safety related components analyzed. The instrumentation cable is considered to be the worst case from the point of view of heat absorption as it has the thinnest insulation and no jacket on individual conductors. The printed circuit board of the pressure transmitter is considered to be the most sensitive component. Therefore, the heat transfer model analyzes that part of the pressure transmitter.

For comparison, the surface temperatures of the electronic transmitter and the electric instrumentation cable have been recalculated using the USNRC regulatory position methodology outlined in reference (A). The surface temperature of the fan cooler motor has also been calculated using the USNRC methodology. This component is modeled as a one slab type heat sink with an initial bulk temperature equal to 120°F. Figure 6.2-32i shows the geometry and thermal properties of the fan cooler motor.

→(DRN 01-230, R12; 05-1004, R14)

The Ebasco modified version of CONTEMPT-LT Mod 26 is used to simultaneously calculate the containment pressure and temperature transients and the safety related components temperature gradients using both the mass diffusion heat transfer coefficient methodology [20] and the USNRC regulatory position methodology [22]. A description of the CONTEMPT computer code and the Ebasco modifications are contained in Appendix 6.2B. The equipment surface velocity using the mass diffusion method is assumed to be a constant 160 ft/sec until the end of blowdown when free convection is assumed to exist. The equipment surface velocity used in the heat transfer coefficient calculations using the USNRC methodology is calculated using the empirical formula listed in reference (22). The two velocity transients are shown on Figure 6.2-32j. All other assumptions used in these computer runs are identical to those assumed in the MSLB analysis. A summary of the peak calculated results are in Table 6.2-45.

←(DRN 01-230, R12; 05-1004, R14)

The graphs of the CONTEMPT-LT Mod 26 computer run results showing the temperature transients as a function of time are shown on Figure 6.2-33a. The heat transfer coefficient values versus time for the various cases are shown on Figure 6.2-32c. The temperature at the cable surface exceeds the design temperature for approximately 46 seconds for the mass diffusion case and 44 seconds using the USNRC methodology. However, the surface of the inner insulation (0.045 in. below the surface) never reaches the qualification temperature for either case. Therefore, it is safe to conclude that a high surface temperature will not impede the performance of the safety-related instrumentation cable because the temperature of the insulation does not reach the qualification temperature.

The difference in heat absorbed, and thus temperature, of a component calculated by the two heat transfer coefficient calculation methods is primarily due to the forced convection effect. The time dependent surface velocity assumed on a component using the USNRC empirical formulation is shown on Figure 6.2-32j. The mass diffusion method for calculating the heat transfer coefficient assumed a constant component surface velocity of 160 ft/sec until the end of blowdown at 55 sec. Using the cable as an example, since the heat transfer to this component has been determined to be due entirely to forced convection, a graph of the heat transfer coefficient and temperature versus time for the two methods have been plotted on Figures 6.2-32c and 6.2-33a. These figures illustrate that more heat is absorbed by the component using the mass diffusion heat transfer correlation as compared with the USNRC method during the period of time from approximately 15 seconds, when the surface velocity calculated using the USNRC method falls below 160 ft/sec (Figure 6.2-32j), to approximately 55 seconds when free convection is assumed to exist using both methodologies. Thus, the constant velocity assumption results in a higher total heat absorption. Additionally, the value of the heat transfer coefficient calculated for the transmitter using the USNRC methodology when referenced to the containment vapor temperature (Figure 6.2-32c) results in a higher heat transfer coefficient than the mass diffusion method after free convection is assumed to begin. This is due to the fact that the surface temperature has not reached saturation temperatures using the USNRC methodology. The USNRC method for calculating heat transfer coefficient using the vapor temperature as a reference also results in an unrealistic rise in magnitude as the containment vapor temperature approaches the saturation temperature even though free convection exists. It is expected, that using the USNRC methodology, this value would eventually approach the transmitter heat transfer coefficient value calculated using the saturation temperature as a reference temperature. This incongruity in heat transfer coefficient value calculated using the USNRC method indicates that the heat transfer coefficient calculated using the mass diffusion method results in more realistic and more technically defensible results, even though the assumed transient nature of the surface velocity is not totally realistic.

Containment External Pressure Analysis

→(EC-706, R304)

An analysis was performed to determine the maximum design basis differential pressure across the primary containment vessel and across the shield building following an inadvertent actuation of one train of the Containment Spray (CS) System during normal plant operation. The analysis was performed using the GOTHIC computer code. GOTHIC has been used to analyze the post-accident containment pressure and temperature response as discussed in sub-section 6.2.1.1.3.

The inputs and assumptions used to analyze the maximum pressure differentials across the containment vessel and containment shield building are provided in Table 6.2-11.

The calculated pressure differentials across containment vessel and containment shield wall are shown in Figures 6.2-13 Sheets 1 and 2 respectively. The maximum pressure differentials across containment vessel and containment shield wall are provided in Table 6.2-2. The containment vessel and shield building external pressure design values and margin are shown in Table 6.2-3. There is no single failure which could result in the operation of both spray trains and therefore, inadvertent actuation of one train of containment spray system is assumed. The containment fan coolers are assumed to remove normal plant operation heat load from the containment for the duration of the event. This assumption is conservative since the fan cooler heat removal rate decreases with reduction in containment temperature due to operation of the containment spray.

←(EC-706, R304)

→(EC-706, R304)

←(EC-706, R304)

6.2.1.2 Containment Subcompartments

6.2.1.2.1 Design Bases

The containment subcompartments are subject to pressure transients and jet impingement forces caused by the mass and energy releases from postulated high energy pipe ruptures within their boundaries. Subcompartments within which high energy ruptures are postulated include the reactor cavity, the pressurizer, and the steam generator.

Analyses were made to determine the peak pressure that could be produced by a line break discharging into the subcompartments. Venting of these chambers is employed to keep the differential pressures within structural limits. In addition, restraints on the pipes, reactor vessel, steam generators, and other pressurized equipment are designed so that neither pipe whip nor other forces transmitted through component supports threaten the integrity of the containment structures (see Section 3.6 and Appendix 5.4A).

→(EC-29816, R306)
Break locations for the pressurization analyses were chosen to maximize the pressures. In the original analyses inherent stiffness of the systems, together with the pipe whip restraints, limited the break openings to no more than the break sizes considered. The spectrum of pipe breaks analyzed for each subcompartment is listed in Table 6.2-1.

With the application of Leak-Before-Break methodologies to the main coolant loop (MCL) and surge line, breaks in these lines are no longer the controlling breaks for subcompartment pressurization.

Restraints were used to limit the break area of pipe ruptures and these Reactor Coolant System restraints were designed so that the numerical results of Reference 5 are applicable. The restraint function has been removed for some of the restraints previously credited with limiting the break area of the pipe ruptures along with the requirement to consider these breaks for subcompartment pressurization.
←(EC-29816, R306)

Pipe break analyses were performed for the break spectrum given in Table 1-1 of Reference 5. The characteristics of the primary coolant pipe ruptures were determined in accordance with the methods and criteria of Section 3.6.

The accident that results in the maximum differential pressure across the walls of a respective compartment is designated as the subcompartment design basis accident (DBA). Calculated DBA differential pressures are compared to the design differential pressure values used in the structural design of subcompartment walls and equipment to ensure that calculated values are less than design values.

6.2.1.2.2 Design Features

Plan and elevation drawings showing detailed design, nodes, and component and equipment locations are shown in Figures 6.2-13e through j for each subcompartment.

The reactor cavity is a heavily reinforced concrete structure that performs the dual function of providing reactor vessel support and radiation shielding. Figures 6.2-14 and 6.2-15 show the reactor cavity model. The two major pressure relief paths are the annular space around the upper vessel flange and the annular space around the six piping penetrations through the primary shield wall. The net free volume of the reactor cavity above the neutron shield and the total vent area from the cavity are shown on Figure 6.2-14. No blowout panels are provided.

The walls of the steam generator compartment are constructed of reinforced concrete that serves to support the equipment enclosed and provides radiation shielding. Figures 6.2-16 to 6.2-18 presents the steam generator compartment model. One major pressure relief path leads from the lower portion of the steam generator subcompartment to the containment annular area surrounding the secondary shield wall. The other pressure relief path leads from the upper portion of the steam generator subcompartment to the free volume of the upper containment. The flow area through the primary shield wall which leads to the reactor cavity is ignored for conservatism. The total vent area from the steam generator compartment, and the free volume are shown on Figure 6.2-17. No blowout panels are provided.

The pressurizer compartment is separate from the steam generator and reactor cavity subcompartments. Figure 6.2-19 shows the pressurizer compartment model. The major pressure relief paths exist in the upper portion of the pressurizer compartment and exit to the upper containment volume; venting also occurs from the volume beneath the pressurizer support to the containment subcompartment below. The total vent area from the pressurizer compartment and the net free volume are shown on Figure 6.2-19. No blowout panels are provided.

6.2.1.2.3 Design Evaluation

The CEFLASH-4A computer program was used to calculate the mass and energy release rates from postulated Reactor Coolant System pipe ruptures. This program is described in Section 6.2.1.1.4 of CESSAR (Reference 4). The Reactor Coolant System nodalization scheme is shown in Figure 6.2-20.

The magnitude of the subcompartment pressure response is proportional to the blowdown energy release rates. In order to provide a conservative design evaluation by maximizing the short term blowdown rates, the following methodology is employed:

a) subcooled and low quality break flow computed using the Henry/Fauske critical flow correlation with a discharge coefficient of 1.0;

b) break flow with quality above approximately 6.0 percent computed using the Moody critical flow correlation with a discharge coefficient of 1.0;

c) the momentum flux terms were omitted when solving the conservation of momentum equation for flows within the Reactor Coolant System;

→(DRN 03-2060, R14)

d) an initial reactor power level of 3735 MWt;

←(DRN 03-2060, R14)

e) Reactor Coolant System volumes increased over nominal values;

f) pressurizer water level above normal full power operating level.

Table 6.2-17 provides a summary of the Reactor Coolant System postulated pipe ruptures. Each pipe rupture was postulated for the evaluation of both the compartment structural design and the component supports design. There were no flow restrictions within postulated ruptured lines that were assumed to fail. The reactor coolant piping does not contain flow restrictions.

Mass and energy release data for the Reactor Coolant System postulated pipe ruptures of Table 6.2-17 are given in the tables indicated in Table 6.2-1. These ruptures are listed according to the compartment in which they occur. All of the breaks listed for each compartment were considered for the component supports evaluation. The mass flow rate and energy release rate as a function of time are contained in the tables indicated opposite the break description in Table 6.2-1.

→(DRN 03-2060, R14)
A re-analysis for the 3716 MWt extended power uprate generated LOCA mass and energy release data for the subcompartment analysis using CEFLASH-4A/FII. The mass and energy release data was generated for limiting or nearly limiting Reactor Coolant System (RCS) pipe breaks in several locations selected from Table 6.2-1, Item B. The re-analysis employed the same methodology as the original analysis with one exception. The original analysis generated the mass and energy release for the circumferential breaks by assuming a slot break in the piping at one end and then at the other end of the pipe. The re-analysis modeled the circumferential break as a double ended guillotine break in the piping. The re-analysis determined that the original data is bounding for all breaks with the exception of the 350 Sq. In. Discharge Leg Circumferential Break, which is shown in Table 6.2-13 Item C.
←(DRN 03-2060, R14)

The CEFLASH-4A computer program was also used to calculate the mass and energy release rates from the postulated feedwater line full double-ended circumferential rupture at the steam generator inlet nozzle. The steam generator and feedwater line nodal model is shown on Figure 6.2-20b and nodal description given on Figure 6.2-20c. Release rates are in Table 6.2-13(S,T).

Note that where two tables are given for the circumferential break release rates the total blowdown is equal to the sum of the similar parameters in the two tables.

Analysis of the pressure transients in the reactor cavity and, the steam generator subcompartment were performed using the RELAP-3 Mod 68 computer code (Reference 1). The code was modified to allow the use of a multiplier in the Moody choked flow calculation. A coefficient of 0.6 is used for all physical junctions in the three subcompartment models. The following subcompartment initial conditions whose conservatism in maximizing the pressure transient was discussed previously, are assumed in all three models:

a) each subcompartment is entirely filled with air;

b) no water vapor or liquid is initially present; and
→(DRN 01-230, R12)
c) the initial pressure and temperature are 14.7 psia and 120°F.
←(DRN 01-230, R12)
Analysis of the pressure transients in the pressurizer subcompartment were performed using the RELAP-4 Mod 5 computer code, Reference 6. The options used in running the code are:

a) The RELAP-4 - CONTAINMENT option.

b) The compressible single-stream form of the momentum equation with momentum flux.

c) The thermal homogeneous equilibrium critical flow correlation for air-stream-water mixtures.

→(DRN 01-230, R12)
The initial pressure and temperature in the pressurizer subcompartment are 14.7 psia and 120°F respectively.
←(DRN 01-230, R12)
The junction effective inertia (I/A) was calculated in a manner consistent with the methods described in RELAP-3. For a pair of volumes v_i and v_k, with cross-sectional areas, A_i and A_k, and lengths in the direction of flow, l_i and l_k and a junction with area A_j and length l_j,

where $A_j \neq A_i, A_k$ and $l_j < l_i, l_k$, the inertia coefficient 1/A was computed as:

$$\frac{1}{A} = \frac{l_i}{2A_i} + \frac{l_j}{A_j}\frac{l_k}{2A_k} \qquad (1)$$

→(DRN 03-2060, R14)
Flow coefficients for the subcompartment analysis were computed in a manner consistent with the calculations done by the RELAP-4 code (Reference 6). The junction friction coefficient utilized in the analyses is a combination of the wall friction losses (K_F), and any irreversible friction losses (K_T) such as area changes, flow obstructions due to turns and gratings. The wall friction loss is computed as:
←(DRN 03-2060, R14)

$$K_{Fi} = \frac{fl_i}{2D_{Hi}}\left(\frac{A_j}{A_i}\right)^2 \qquad (2)$$

$$K_{Fj} = \frac{fl_j}{D_{Hj}} \qquad (3)$$

$$K_{Fk} = \frac{fl_k}{2D_{Hk}}\left(\frac{A_j}{A_k}\right)^2 \qquad (4)$$

Where D_{Hn} are the hydraulic diameters of the system and f is conservatively assumed to be 0.02 K_T (see References 2 and 3) and is chosen to account for all friction loss within the associated volumes as well as loss within the junction itself.

The total friction loss coefficient at a junction (K_{RELAP}) is then represented as:

$$K_{RELAP} = K_{Fi} + K_{Fj} = K_{Fk} + K_T \left(\frac{A_j}{A_T}\right)^2$$

where A_T represents the reference area to which K_T applies.

Subcompartment Modeling

Subcompartment nodalization models are determined principally by physical flow restrictions within each compartment. These flow restrictions consider the presence of steel and concrete obstructions, doorways, vent pressurizer, the reactor vessel, and the reactor cavity missile and neutron shields. By choosing node boundaries at the various physical flow restrictions in a manner consistent with the flow model used by RELAP-3, calculated differential pressures and consequent support loads are realistically maximized. The nodalization sensitivity study performed in the Shearon Harris PSAR (docket 50-400, 401, 402 and 403) showed that the peak calculated differential pressure is very sensitive to an increasing number of nodes until that number equals the number defined by physical restrictions to flow. Increasing the number of nodes beyond that defined by the number of physical restrictions will not result in increased pressure differentials. It is therefore concluded that further arbitrary increase in the number of subcompartment nodes modeled is neither sensible nor realistic unless additional physical flow obstructions exist. The subcompartment models, discussed below, take into account all physical flow obstructions present.

a) Reactor Vessel Cavity

For the analyses of the pressure transient in the reactor cavity following a line break, the flow models consisted of 19 control volumes. The flow models used are illustrated in Figures 6.2-14 and 6.2-15. The control volume and vent path descriptions are given in Table 6.2-14.

Insulation in the reactor cavity was assumed to remain in place and was included in the volume and vent area calculations. The affected piping system, location, size and mass and energy release rates for each break is given in Table 6.2-13.

The reactor cavity is essentially a cylindrical annular air space between the reactor vessel and the primary shield wall. The cavity is bounded at the top by the reactor vessel flange and at the bottom by a neutron shield which is designed to prevent flow from a break into the lower cavity.

This air space contains two hot legs and four discharge legs, including nozzles and support structures. The air space is open to the refueling pool at the reactor vessel flange and to the steam generator compartments through six pipe penetrations, three penetrations to each compartment.

These structural components define the model to be used for subcompartment analysis. A typical node is thus bounded (in a cylindrical coordinate system) in r by the reactor vessel and primary shield wall, in θ by a hot or discharge leg on either side at 60° intervals, and in z by the upper surface of the neutron shield and the reactor vessel flange. For conservatism, it has also been assumed that there is no free volume beneath the legs within the region of the vessel supports.

At all interfaces between volumes, a 0.95 multiplier was used on the calculation of the minimum flow area. The minimum flow area in θ was defined by the maximum cross-sectional area of the nozzles. All volumes were multiplied by 0.9 to account for potential as-built additions or modifications.

The sensitivity study performed for the Carolina Power & Light Co. Shearon Harris plant demonstrated that for the upper reactor cavity (the region above the reactor vessel supports) a proper subdivision accounted for the obstruction of flow by the legs and supports and by the reactor vessel flange. In the Shearon Harris cavity, there is a narrow gap that vents to the lower cavity regions, but only a small percentage of the flow is ever directed to this region, and it forms a small percentage of the vessel forces.

Aside from this, the Shearon Harris and Waterford 3 cavity designs are essentially the same shape, i.e., cylindrical annuli containing six legs and nozzles. The height of the annulus is 9.37 ft. in the Shearon Harris plant as compared to the nominal height of 10.26 ft. used in this analysis. The Shearon Harris cavity is 25 ft. in diameter at the legs as compared to 24 ft. four in. in the Waterford 3 cavity. As the Shearon Harris cavity is then proportionately larger, the free volume was calculated at 375 ft^3 for a typical one-sixth of the upper cavity. A typical one-sixth of the Waterford 3 cavity was calculated to have 269 ft^3 free volume (or 299 ft^3 x 0.9 uncertainty). The Waterford 3 region is then about 20 percent smaller overall than the Shearon Harris cavity.

b) Steam Generator Compartments

For the analysis of the pressure transient in a steam generator compartment following a LOCA, a flow model was used consisting of 23 control volumes as shown in Figures 6.2-16 to 6.2-18. Each control volume and vent path is shown on Table 6.2-15. For conservatism, no flow from the steam generator into the reactor cavity was assumed in the analysis. The affected piping system, location, size, and mass and energy release rates for each break is given in Table 6.2-13. The steam generator pressure transient for a feedwater circumferential break in volume number 22 was also calculated. A pressure transient was not performed for a steam line break because the routing of this line does not pass through the steam generator subcompartment.

c) Pressurizer Compartment

→(EC-19087, R305)

For analysis of the pressurizer subcompartment pressure transient and uplift force, the pressurizer compartment was modeled as 10 control volumes. The nodalization of the pressurizer subcompartment is based upon the guidelines outlined under Subcompartment Modelling. All the junction areas represent flow restrictions (such as gratings (junction 3), blowout area (junction 5)). The specific symmetrical geometry of this subcompartment along with the symmetric location of the postulated break locations indicate that, using the previously outlined guidelines, a further nodalization would be unnecessary. The resulting flow model is depicted in Figure 6.2-19. Control volume and vent path descriptions are given in Table 6.2-16. The breaks considered for the analyses were a double ended pressurizer surge line guillotine break within the pressurizer skirt area, a pressurizer spray line guillotine and a pressurizer safety relief line guillotine. The mass and energy release rates for these breaks are listed on Table 6.2-13. Surge line breaks are conservatively assumed for the subcompartment pressurization analysis. The surge line has been removed as a possible piping rupture location under LBB as discussed in Section 3.6.3 including affects for subcompartment pressurization (NUREG-1061, Volume 3).

←(EC-19087, R305)

Due to the small mass and energy releases associated with the pressurizer safety relief and spray line guillotines (see Table 6.2-1) and due to the routing of these lines and location of their connections to the pressurizer vessel, the break is not capable of producing significant pressure differentials across the pressurizer or the pressurizer subcompartment walls.

Results

The design value for pressure in the subcompartments is at least 140 percent of the peak calculated values as shown in Tables 6.2-2 and 6.2-3.

Graphs of the subcompartment pressure response versus time for each break analyzed are given in Figures 6.2-21 through 6.2-29. The peak calculated differential pressure is limited to a small portion of the total wall area and is less than the design pressures.

In the design of primary shield wall of the reactor cavity, the design differential pressure is spatially varied as given in Figures 6.2-21 to 6.2-22h, and is utilized accordingly. The spatial variation around the reactor cavity is centered around each hot and cold leg respectively and the primary shield wall is analyzed separately for each of these six structural cases. In the design of the steam generator and pressurizer rooms, the maximum differential pressure for each room is uniformly applied to the corresponding subcompartment structure.

Using the assumptions specified for the RELAP-3 Mod 68 computer code, the pressurization analysis of the steam generator subcompartment for the DBA, the 592 in^2 SLG, indicates that all the flow in the junctions connected to the break volume (15) and the adjacent volume (16) remain choked until after the peak pressures are calculated in these volumes. Therefore, the RELAP-3 Mod 68 computer code is applicable for the steam generator subcompartment pressurization transients. The following table summarizes the choking conditions for the DBA steam generator pressurization analysis:

Time	Junction with Choked Flow
0.1 sec.	9, 10, 13, 14, 17, 19, 20, 21, 22, 23, 24, 25, 30, 31, 36
0.35 sec.	5, 7, 9, 10, 17, 20, 21, 22, 23, 30, 31, 36
0.40 sec.	5, 7, 9, 10, 17, 20, 21, 22, 23, 30, 31, 36

➔(EC-8458, R307)

6.2.1.3 Mass and Energy Release Analysis for Postulated Loss of Coolant Accidents

➔(DRN 01-230, R12)

This subsection discusses the mass and energy release analysis for postulated LOCAs for the Containment functional design. The accident is divided into four time frames which chronologically are: blowdown, reflood, post reflood, and long term. The blowdown phase starts at the initiation of the postulated pipe break. During the blowdown phase the primary coolant is being rapidly injected into the containment; the blowdown period ends when essentially all of the coolant has been injected into the containment. The reflood period is when the core is being recovered (or flooded) with Safety Injection System water. Reflood ends when the water level in the core region reaches a height which is sufficient to quench the previously hot core. Post reflood is when the steam generator secondary energy remaining at the end of reflood is used in conjunction with wall heat sources and decay heat to boil off part of a two phase flow now passing through the primary system. Post reflood ends when the steam generator secondary temperature has essentially reached equilibrium with the primary side temperature so that there is no longer a significant driving potential for secondary to primary heat transfer. Since during a hot leg break most of the break flow does not pass through any steam generator, the reflood and post-reflood phases do not apply to hot leg breaks. During the long term, Safety Injection System water boils at the containment pressure as a result of containment depressurization, decay heat generation, and residual thick metal and steam generator cooldown.

🡐(DRN 01-230, R12; EC-8458, R307)

→(DRN 01-230, R12)

Subsection 6.2.1.3.1 discusses the LOCA mass and energy release data. Subsection 6.2.1.3.2 discusses the energy sources used for the analysis. Subsections 6.2.1.3.3, 6.2.1.3.4, 6.2.1.3.5, and 6.2.1.3.6 discuss the models and assumptions for the blowdown, reflood, post-reflood and long term periods, respectively. Subsection 6.2.1.3.7 discusses LOCA single active failure analyses.

6.2.1.3.1 Mass and Energy Release Data

→(DRN 03-2060, R14)

Mass and energy release data for the limiting LOCA cases (double ended hot leg slot break, with maximum safety injection flow, and double ended discharge leg slot break with minimum safety injection flow) are provided in Table 6.2-12A and
6.2-12B respectively.

←(DRN 01-230, R12; 03-2060, R14)

6.2.1.3.2 Energy Sources

The following sources of generated and stored energy in the Reactor Coolant System and secondary system are considered: primary coolant, primary walls, secondary coolant, secondary walls, safety injection water, core power transient and decay heat, and steam generator forward and reverse heat transfer. The initial Reactor Coolant System water volumes were conservatively calculated based on maximum manufacturing tolerances for the reactor vessel and steam generator tubes. Expansion of the loop components from cold to hot operating conditions was also considered. The pressurizer water volume included an allowance for level instrumentation error.

→(DRN 02-525, R12; 01-230, R12)

←(DRN 02-525, R12; 01-230, R12)

6.2.1.3.3 Description of Blowdown Model

→(DRN 01-230, R12)

Blowdown mass and energy release rates were calculated using the CEFLASH-4A computer code as described in Reference 7. The following provide a list of key assumptions used in the CEFLASH-4A analyses:

- A two phase heat transfer coefficient, Jen Lottes correlation, is used to calculate the core to coolant heat transfer in the nucleate boiling regime (whenever the flow through the core is not pure steam).

- The core is modeled as 1 radial zone and 5 axial zones (5 axial nodes).

- Steam generator tube plugging is not assumed.

- It is assumed the steam flow goes to zero instantaneously at time zero.

- It is assumed that the termination of the main feed flow is by the closure of the Main Feedwater Isolation Valves (MFIVs) on high containment pressure.

- Emergency feedwater flow is conservatively omitted since it, too, would tend to cool the secondary sides.

←(DRN 01-230, R12)

→(DRN 01-230, R12)

←(DRN 01-230, R12)

→(DRN 01-230, R12)

6.2.1.3.4 Description of Core Reflood Model

The FLOOD3 code continues the generation of the break mass and energy release from the end-of-blowdown (EOB) to the end-of-post-reflood for cold leg break cases. The EOB is considered to be the time at which the original reactor coolant system mass has emptied into the containment. The end-of-reflood is taken to be the time when the core has been quenched (when the reflood water level reaches a point two feet below the top of active fuel). The refill period (the time it takes the safety injection tank water entering the reactor vessel to reach the bottom of the active fuel region), is omitted since it is a relatively short period during which there would be no cooling. This is conservative since including the core refill period in the analysis would decrease the energy release to the containment. The end-of-post-reflood time is defined to be the point at which the system inventory and metal have reached an equilibrium which the rate of energy (or steam) released is essentially equal to that produced by decay heat from the core. The primary and secondary conditions at the end of the blowdown are used as input to the FLOOD3 code for the calculation of mass and energy release during the reflood and post-reflood phases (Reference 8).

←(DRN 01-230, R12)

→(DRN 01-230, R12)

←(DRN 01-230, R12)

6.2.1.3.5 Description of Post Reflood Model

The methodology for generating Post Reflood mass and energy release rates is described in Sub-section 6.2.1.3.4.

→(DRN 01-230, R12)

←(DRN 01-230, R12)

6.2.1.3.6 Description of Long-Term Cooling Model

→(DRN 01-230, R12; 03-2060, R14)

Westinghouse (formerly ABB CE) containment analysis code, CONTRANS (Reference 26), has been used to calculate the post-LOCA long term mass and energy release to the containment. The following provide some of the significant input and assumptions used in the long term analyses:

←(DRN 03-2060, R14)

- For the double ended discharge leg break cases, ¼ of the safety injection tank and safety injection pump flows are directly spilled to the containment sump. This is because the safety injection flows are injected on the discharge side of the cold leg. This is not applicable to hot leg and suction leg breaks.

- The minimum usable Refueling Water Storage Pool (RWSP) volume assumed for the calculation of the time of Recirculation Actuation Signal (RAS) is 383,000 gallons.

- A bounding decay heat curve based on ANSI/ANS 5.1 1979 standard which accounts for long term actinides and heavy elements decay was used.

←(DRN 01-230, R12)

→(DRN 01-230, R12)

←(DRN 01-230, R12)

→(DRN 01-230, R12)

6.2.1.3.7 Single Active Failure Analysis

→(DRN 04-631, R13-B; EC-8458, R307)

For LOCA cases it is assumed that non-emergency power (offsite power) is not available at the time of the accident and the worst single active failure is a failure of a containment cooling train.

←(DRN 04-631, R13-B; EC-8458, R307)

Assuming the availability of non-emergency power (offsite power) is conservative for the MSLB events since it allows the continuation of reactor coolant pump operation. This maximizes the rate of heat transfer to the affected steam generator which maximizes the rate of mass/energy release.

The single active failures considered for the MSLB events are:

- failure of one main feedwater isolation valve (MFIV) to close,
- failure of one train of containment heat removal system to operate and
- failure of one main steam isolation valve (MSIV) to close

←(DRN 01-230, R12)

→(DRN 01-230, R12)
←(DRN 01-230, R12)

6.2.1.3.8 Metal-Water Reaction

→(EC-8458, R307)

Energy addition to the containment atmosphere due to zirconium water reaction is not included in the LOCA containment pressure-temperature analyses. This is because all the assumptions are biased to maximize the heat removal from the core, therefore, the fuel clad temperature will not reach high enough temperatures to cause metal-water reaction.

←(EC-8458, R307)

6 2.1.3.9 Energy Inventories

This sub-section has been deleted.

6.2.1.3.10 Additional Information Required for Confirmatory Analysis

This sub-section has been deleted.

←(DRN 01-230, R12)

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

Following a postulated main steam line break (MSLB) or a main feedwater line break (MFWLB) inside the containment, the contents of one steam generator (ruptured) will be released to the containment. Most of the contents of the other steam generator (intact) will be isolated by the main steam isolation valves (MSIV) and main feedwater isolation valves. Containment pressurization following a secondary side rupture depends on how much of the break fluid enters the containment atmosphere as steam. MSLB flows can be pure steam or two-phase. Part of any liquid in the break flow boils off in the containment and is also added to the atmosphere while the rest falls to the SIS sump and contributes nothing to containment pressurization.

→(DRN 01-230, R12)

For MSLB cases with large break areas, steam cannot escape fast enough from the two-phase region of the ruptured steam generator and the two-phase level in the steam generator rises rapidly to the steam line nozzle. A two-phase blowdown results. The duration of this blowdown is short; therefore, little primary to secondary heat transfer takes place and the break flow is largely liquid. For MSLB cases with small break areas, steam can escape fast enough from the two-phase region of the ruptured steam generator so that the

→(DRN 01-230, R12)

break. Because of the pressure reducing effects of active and passive containment heat sinks, the highest peak containment pressure resulting from a MSLB for a given set of initial steam generator conditions occurs for that case where the break area is the maximum at which a pure steam blowdown can occur. The potential for steam generator two-phase level swell following a MSLB increases as power level decreases; therefore, a spectrum of power levels must be analyzed to determine which one results in the peak MSLB containment pressure.
←(DRN 01-230, R12; 03-2060, R14)

The feedwater ring is below the steam generator water level; therefore, MFWLB cases always result in two-phase blowdowns and do not produce peak containment pressures as severe as MSLB cases.

→(DRN 01-230, R12; 03-2060, R14; EC-8458, R307)
To determine the limiting MSLB for the containment peak pressure and peak temperature, the mass and energy release data for the replacement steam generator conditions were analyzed from 100.5, 75, 50, 25, and 0 % power conditions assuming three different single failures using SGNIII (Reference 27) code. The break is assumed to be in the steam line near the nozzle of one of the steam generators. Unrestricted critical flow from the rupture is assumed. The RSG flow restrictor allowed full guillotine breakers to be analyzed down to 25% power. The hot zero power case (MSLB from HZP conditions) is a slot break with flow area reduced to 3.9 ft^2 to ensure an all steam blowdown.
←(DRN 03-2060, R14; EC-8458, R307)

In the plant, main steam isolation and main or emergency feedwater isolation are initiated simultaneously by signals from the Engineered Safety Features Actuation System. The main steam line isolation valves (MSIV) and the main feedwater isolation valves (MFIV) are assumed to close in seven seconds after receipt of the main steam isolation signal (MSIS). For the MSLB cases with MFIV failure to close, feedwater regulating valves are credited to close in five seconds after receipt of the MSIS. Main steam line isolation is discussed in Section 10.3. Main feedwater line isolation is discussed in Section 10.4. Emergency feedwater line isolation is also discussed in Section 10.4.
←(DRN 01-230, R12)

The Emergency Feedwater System functions automatically during MSLB to ensure that a heat sink is always available to the Reactor Coolant System by supplying cold feedwater to maintain an adequate water inventory in the intact steam generator. The ruptured steam generator is identified and isolated while a controlled flow path is provided to the intact steam generator. No credit for emergency feedwater flow to the intact unit is taken in the MSLB analysis.

→(DRN 01-230, R12; 03-2060, R14)
The volume of fluid in the line between the main feedwater isolation valve and the ruptured steam generator is 274 cubic ft. The flashing of this fluid into the ruptured steam generator and then into the containment is considered in the analysis.

Subsection 6.2.1.4.1 discusses the mass/energy release data for the most severe secondary system pipe rupture. Subsection 6.2.1.4.2 discusses the failure mode and effects analysis performed to determine the most severe single active failure for the purpose of maximizing the mass/energy release into the containment and the containment pressure response. Subsection 6.2.1.4.3 discusses the initial conditions for each case.

6.2.1.4.1 Mass and Energy Release Data

Mass and energy release data for the most limiting peak pressure and peak temperature cases are provided in Tables 6.2-12D and 6.2-12E respectively.
←(DRN 01-230, R12; 03-2060, R14)

→(DRN 01-230, R12)
←(DRN 01-230, R12)

6.2.1.4.2 Single-Failure Analysis

→(DRN 01-230, R12)
Main steam isolation and main feedwater isolation are initiated simultaneously by a signal from the Engineered Safety Features Actuation System. The plant has been configured such that MSIS (i.e., SG low pressure or containment high pressure) is used to isolate the steam generators. Feedwater flow is automatically terminated by the feedwater isolation valves seven seconds after receipt of the MSIS signal. The main steam line isolation valve closure time is also seven seconds.
←(DRN 01-230, R12)

A single failure of one MSIV during a postulated main steam line break accident will not cause uncontrolled blowdown of more than one steam generator. Closure of the operational MSIV will provide isolation of the intact steam generator.

→(DRN 01-230, R12; EC-8458, R307)
A single failure of one MFIV during a postulated main steam line break accident is accommodated by closure of the main feedwater regulating valve upon receipt of a MSIS signal. Thus assuming a single failure, feedwater flow to the affected steam generator is terminated by the closure of the main feedwater regulating valves within five seconds after receipt of signal. The MSLB events are analyzed assuming three different single active failures: (1) loss of one train of containment heat removal system; (2) failure of one MSIV to close; (3) failure of one MFIV to close.

→(DRN 03-2060, R14)
For the loss of one train of containment heat removal system only one containment spray train and only one containment fan cooler is assumed to operate. The MSIVs and MFIVs perform their function as designed. For the cases with one MSIV failure (1) the MFIVs and both Containment Heat Removal trains are postulated to perform as designed and (2) the volume of steam in the steam lines between the two MSIVs are assumed to be released into the containment. For the cases with one MFIV failure: (1) the MSIVs and both trains of containment heat removal system are postulated to perform as designed, and (2) the main feedwater regulating valve is assumed to close and isolate the feedwater flow to the broken steam generator and (3) the feedwater volume between the MFIV and the main feedwater regulating valve is assumed to be released in the containment.
←(DRN 03-2060, R14; EC-8458, R307)

Credit is not taken, however, for the coastdown of the main feedwater pumps or condensate pumps in the MSLB analysis. Therefore, no degradation of the feedwater flow occurs until the closure of the main feedwater isolation or regulating valve. Accordingly, trips of the feedwater or condensate pumps were not considered. All of the feedwater flow is diverted to the ruptured steam generator after the initiation of the MSLB.
←(DRN 01-230, R12)

Offsite power is assumed to be available for the analysis. Availability of offsite power allows the continuation of reactor coolant pump and feedwater pump flow. Maintaining reactor coolant and feedwater pump flow maximizes the rate of primary to secondary heat transfer which maximizes the rate of mass/energy release.

→(DRN 01-230, R12; 03-2060, R14; EC-8458, R307)

←(EC-8458, R307)

6.2.1.4.3 Input Parameters and Assumptions

→(DRN 02-525, R12)

The following provides the initial values for the key parameters and key assumptions used in the generation of mass and energy release data for the MSLB events.

- Initial Core Inlet Coolant Temperature, °F = 552°F
- Initial Reactor Coolant System Pressure (psia) = 2310.00
- Core thermal power = 3735 MWt (3716 MWt + 0.5% uncertainty)
- 8 seconds is assumed for the MSIV and MFIV closure time. The 8 seconds include 7 seconds valve closure time plus one second signal processing delay time.
- 6 seconds is assumed for the main feedwater regulating valve closure time. The 6 seconds include 5 seconds closure time plus 1 second signal processing delay time. The feedwater regulating valve closure is credited for the case of main feedwater isolation valve failure.
- All the feedwater is aligned to the affected SG until the feedwater is isolated.

→(EC-8458, R307)
←(EC-8458, R307)

- Maximum RCS flow rate is assumed, because it allows the maximum possible heat transfer between the primary and secondary system.
- No credit is taken for SG tube plugging.

→(EC-8458, R307)

- The break size used is the largest possible for a given power level that produces a pure steam (dry) blowdown. For the hot zero power MSLB, the break area was reduced to $3.9 ft^2$.

←(DRN 02-525, R12; 01-230, R12; 03-2060, R14; EC-8458, R307)

→(DRN 01-230, R12)
←(DRN 01-230, R12)

6.2.1.4.4 Description of Blowdown Model

→(DRN 01-230, R12)

The SGNIII digital computer code (Reference 27) is used for the secondary system pipe break analysis.

Following closure of the MFIV'S, there is an inventory of feedwater between the MFIV and the ruptured steam generator. As the ruptured steam generator depressurizes, this inventory starts to boil. As steam in the line expands, this feedwater inventory is pushed into the steam generator and is boiled off by primary to secondary heat transfer. The expansion of the feedwater inventory into the ruptured steam generator has been considered in the analysis.

6.2.1.4.5 Energy Inventories

This sub-section has been deleted.

6.2.1.4.6 Additional Information Required for Confirmatory Analyses

This sub-section has been deleted.

←(DRN 01-230, R12)

→(DRN 01-230, R12)

←(DRN 01-230, R12)

6.2.1.5 Minimum Containment Pressure Analysis for Performance Capability Studies on the Emergency Core Cooling System

6.2.1.5.1 Introduction and Summary

→(DRN 03-2060, R14)

Appendix K to 10CFR50 provides the required and acceptable features of Emergency Core Cooling System (ECCS) evaluation models.(11) Included in this list is the requirement that the containment pressure assumed in the evaluation of ECCS performance not exceed a pressure calculated conservatively for that purpose. The ECCS performance analysis for Waterford 3 which is presented in Subsection 6.3.3, meets the minimum containment pressure requirement of Reference 11, Appendix K, Paragraph I.D.2. The Reference analysis was performed during the extended power uprate to 3716 MWt.

6.2.1.5.2 Method of Calculation

The calculations reported in this section are performed using the large break evaluation model described in Reference 12, which was approved by the NRC in References 17, 23, and 30. The CEFLASH-4A (13) computer program is used to determine the mass and energy released to the containment during the blowdown phase of a postulated LOCA, and the COMPERC-II(14) computer program is used to determine both the mass and energy released to the containment during the reflood phase and the minimum containment pressure response to be used in the evaluation of the effectiveness of the Emergency Core Cooling System.

6.2.1.5.3 Input Parameters

→(DRN 06-1061, R15; EC-9533, R302; EC-8458, R307)

Input parameters given in this section represent the ECCS analysis performed for the extended power uprate to 3716 MWt with replacement steam generators and up to 10% SG tubes plugged for the full core implementation of CE 16x16 NGF assemblies.

←(DRN 03-2060, R14; 06-1061, R15; EC-9533, R302; EC-8458, R307)

6.2.1.5.3.1 Mass and Energy Release Data

→(DRN 06-1061, R15; EC-9533, R302)

The mass and energy released to the containment for the most severe LOCA, the 1.0xDEG/PD break, is listed as a function of time in Table 6.2-19. The mass and energy release during blowdown and reflood has been calculated specifically for Waterford 3. Based on results from Waterford 3 SIT blowdown tests, K-factors specific to the Waterford 3 SIT blowdown lines have been conservatively used for the CEFLASH-4A and COMPERC-II computer program calculations. The quantity of safety injection fluid assumed to spill from the break is discussed in Subsection 6.2.1.5.3.5.

←(DRN 06-1061, R15; EC-9533, R302)

6.2.1.5.3.2 Initial Containment Internal Conditions

The initial containment internal conditions which have been assumed for this analysis are:

→(DRN 03-2060, R14; EC-9533, R302)

Temperature - 95°F (Minimum)

←(EC-9533, R302)

Pressure – 14.025 psia (Minimum)

Relative Humidity - 100 percent (Maximum)

←(DRN 03-2060, R14)

For each parameter, the conservative direction with respect to minimizing the containment pressure appears in parentheses.

6.2.1.5.3.3 Containment Volume

The net free containment volume assumed for this analysis is 2,684,000 ft^3.

6.2.1.5.3.4 Active Heat Sinks

For this analysis, it is conservative to maximize the heat removal capacity of the containment active heat sinks; thus, both the containment sprays and all four containment fan coolers are assumed to actuate in the shortest possible time following the break and to operate at their maximum capacity, assuming the minimum temperature of both the stored water and cooling water. To minimize the actuation time, offsite power is assumed to be available for all active heat sinks. (It should be noted that offsite power is assumed to be unavailable for the SIS.)

The Safety Injection System equipment assumed to be operable for this analysis is discussed in Subsection 6.3.3.2.1.

The heat removal rate of each containment fan cooler is shown as a function of containment temperature in Figure 6.2-30a. The operating parameters assumed for the containment sprays are as follows:

→(DRN 03-2060, R14)

Flow - 4500 gpm (Maximum)

Temperature - 50°F (Minimum)

←(DRN 03-2060, R14)

6.2.1.5.3.5 Steam-Water Mixing

The effect of mixing and condensation of containment steam with spilled ECCS water upon the containment pressure is calculated in the manner described in Section III.D.2 of Reference 12. The effective ECCS spillage rate is shown as a function of time in Figure 6.2-30b.

6.2.1.5.3.6 Passive Heat Sinks

The surface areas and thickness of all exposed containment passive heat sinks used in the design basis analysis are listed in Table 6.2-7. To conservatively maximize the heat transfer to these passive sinks, the surface areas have been assumed to be at the maximum of their uncertainty ranges, and their thermal properties (conductivity and heat capacity) have been maximized. The thermal properties assumed for this analysis are:

→(DRN 03-2060, R14)

Material	Thermal Conductivity (BTU/hr-ft-F)	Volumetric Heat Capacity (BTU/ft^3 -F)
Zinc Coating	64	40.6
Carbon Steel	25.9	53.57
Stainless Steel	9.8	54.0
Concrete	1.0	31.9

←(DRN 03-2060, R14)

6.2.1.5.3.7 Heat Transfer to Passive Heat Sinks

The condensing heat transfer coefficients between the containment atmosphere and the passive heat sinks have been calculated in the manner described in Section III.D.2 and Figure III.D.2-2 of Reference 12. The variation of the condensing heat transfer coefficients as a function of time is shown quantitatively in Figure 6.2-30c.

6.2.1.5.3.8 Containment Purge System

Usage of the 48" Containment Purge System during normal operation will occur only during a small percentage of the calendar year. In addition, the purge system valves are mechanically prevented from opening more than part way during Modes 1-4 in accordance with the plant's Technical Specifications. The chance of a very low probability accident such as the postulated LOCA occurring during purging operations would, therefore, be even less than a LOCA occurring alone. Hence, the analysis presented in this subsection does not include the effects of operation of the containment purge system.

6.2.1.5.4 Results

→(DRN 03-2060, R14; 06-1061, R15; EC-9533, R302; EC-8458, R307)

For the most severe LOCA of the extended power uprate break spectrum analysis with replacement steam generators and up to 10% of the SG tubes plugged for full core implementation of CE 16x16 NGF assemblies, the 1.0xDEG/PD break, the minimum containment pressure response is shown in Figure 6.2-31a. The ECCS performance analysis used the same containment pressure; therefore, as required by 10CFR50 Appendix K, the containment pressure used in the ECCS performance evaluation does not exceed this pressure. The responses of the containment atmosphere and SIS (recirculation) sump temperatures are shown in Figures 6.2-31b and 6.2-31c, respectively.

←(DRN 03-2060, R14; 06-1061, R15; EC-9533, R302; EC-8458, R307)

6.2.1.6 Testing and Inspection

Preoperational and periodic tests are conducted to ensure the functional capability of the containment and associated structures, systems, and components.

Included are the Integrated Leak Rate Test, (see Subsection 6.2.6), Shield Building Leak Rate Test, and operational test on mechanical equipment that are required to operate following a pipe break.

A preoperational visual inspection and test shall be conducted to verify the structural integrity of the Shield Building. The test is conducted by ventilating the Shield Building at a flow rate less than or equal to 10,000 acfm and ensuring that the differential pressure between the Shield Building and the outside atmosphere is -1/4 in. wg. or greater. The results of the visual inspection, any repairs made thereof, and of the test shall be recorded in accordance with the test procedure. Technical Specifications shall determine the post preoperational test frequency and the action to be taken in the event acceptability requirements are not met.

Visual inspection, preoperational and periodic tests are performed at the frequency described in and to the acceptability and requirements of Technical Specifications for mechanical systems and components and in accordance with the preoperational or periodic test procedure. Pumps shall be visually inspected for signs of degradation, operated to ensure performance, and the following data recorded.

a) Pump total dynamic head vs flow.

b) Motor and pump bearing temperature.

c) Unusual noise or vibration.

Systems such as the Containment Spray System are visually inspected for signs of degradation and tested to ensure operability. For example, the containment spray nozzles are tested by pressurizing the system with air and verifying flow through each nozzle.

Fans, gaskets, etc. are inspected for signs of degradation. Fans are operated to ensure performance and system flow in accordance with the plant's Technical Specifications.

Containment isolation valves are visually inspected for signs of degradation, cycled to ensure performance and leak rate measurement performed.

6.2.1.7 Instrumentation Application

Pressure sensing instruments monitor the containment atmosphere and initiate CIAS, SIAS, MSIS and CSAS according to the logic discussed in Section 7.3. Radiation monitors which monitor containment atmosphere and isolate containment purge are discussed in Subsection 12.3.4. Instrumentation applications for the various engineered safety features associated with the containment, such as the Containment Heat Removal System and the Combustible Gas Control System, are discussed in Subsections 6.2.2 and 6.2.5, respectively.

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6.2.2 CONTAINMENT HEAT REMOVAL SYSTEMS

The function of the Containment Heat Removal Systems under accident conditions is to remove heat from the containment atmosphere, thus maintaining the containment pressure and temperature at acceptably low levels. The Containment Heat Removal Systems also serve to limit offsite radiation levels by reducing the pressure differential between the containment atmosphere and the external environment, thereby decreasing the driving force for fission product leakage across the containment. The two Containment Heat Removal Systems are the Containment Cooling System (CCS), and the Containment Spray System (CSS).

The CCS fan coolers are designed to operate during both normal plant operations and under LOCA or MSLB conditions. The outlets of the containment fan coolers are connected to a ductwork system which, during normal operation, transports cooled air to various regions of the containment for the cooling of the containment atmosphere. Figure 9.4-7 is an airflow diagram of the CCS.

The CSS is designed to operate during accident conditions only. A piping and instrumentation diagram of the CSS is given in Drawing G163.

The fission product removal effectiveness of the CSS is discussed in Subsection 6.5.2.

6.2.2.1 Design Bases

a) The CCS, and the CSS, are designed to remove heat from the Containment atmosphere following a LOCA (or Secondary System pipe rupture inside containment) as required by General Design Criterion 38.

b) The sources and amounts of energy released to the containment as a function of time which were used as the basis for sizing the Containment Heat Removal Systems are given in Subsections 6.2.1.3 and 6.2.1.4. The CCS and CSS are designed to remove 234 million Btu/hr under post accident steam-air mixture conditions of 269°F, 44 psig and 100 percent relative humidity while the

Component Cooling Water System (CCWS) is providing cooling water to CCS and CSS at 115°F. The systems are designed to reduce the containment pressure from the peak value to one-half peak value in 24 hours.

c) The heat removal capacity of the CCS and CSS is sufficient to keep the containment temperature and pressure below design conditions (see "b" above) for any size break up to and including double ended break of the largest reactor coolant pipe. The system is also designed to mitigate the consequences of any size break, up to and including a double ended break of a main steam line. The systems continue to reduce containment pressure and temperature and maintain them at acceptable levels post accident.

d) The CCS and CSS each consists of two redundant loops, and are designed such that a single failure does not degrade the systems' ability to provide the required heat removal capability. Two of four containment fan coolers and one CSS loop are powered from an independent safety-related bus. The other two containment fan coolers and CSS loop are powered from another independent safety related bus. The loss of one bus does not affect the ability of the Containment Heat Removal Systems to maintain containment temperatures and pressures below the design values.

e) The CCS and CSS are designed to safety class 2, seismic Category I requirements.Similarly the portion of the CCS ductwork required to remain functional post accident is also safety class 2, seismic Category I. However, nonsafety-related ductwork is seismically supported to prevent collapse following a Safe Shutdown Earthquake (SSE).

f) The CCS and CSS are protected against dynamic effects associated with the postulated rupture of piping as discussed in Section 3.6.

g) Protection of the CCS and CSS from missiles is described in Section 3.5.

h) Sections 3.4 and 3.5 discuss wind/tornado and flood protection for the Containment Heat Removal Systems.

i) Principle design codes and standards for CCS and CSS are given in Tables 6.2-21 and 6.2-22 respectively.

j) Both the CCS and CSS are designed to permit periodic inspection and testing as described in Subsection 6.2.2.4 and the Technical Specifications.

k) The safety-related portions of the CCS and the CSS located inside the containment are designed to withstand the containment environment associated with LOCA or MSLB. These environmental conditions are described in Section 3.11.

→(EC-999, R302)

l) Evaluations have been performed to document CSS compliance with the requirements of GSI-191 and Generic Letter 2004-02. See Section 6.2.2.2.2.1.

←(EC-999, R302)

6.2.2.2 System Design

6.2.2.2.1 Containment Cooling System

The CCS consists of four containment fan coolers designated AH-1 (3A-SA), AH-1 (3C-SA), AH-1 (3B-SB) and AH-1 (3D-SB) and a ducted air distribution system with associated instrumentation and controls. Figures 6.2-36, 37, 38 and 39 show the routing of the CCS supply ductwork inside the containment. Each fan cooler consists of two banks of cooling coils, casing, vane axial two speed fan and motor. The design data for the containment fan coolers is given in Table 6.2-21.

→(EC-40195, R307)
Each CCS loop consists of two fan coolers both of which discharge into a common duct. The duct from each loop then are interconnected into a common ring header and ductwork system which distributes the discharge of the fan coolers to different areas of the containment. The cooling units are located on two levels in the containment outside of the secondary shield wall. The A fan coolers are at elevation +21 ft. MSL and the B fan coolers are at elevation -4 ft. MSL. The A fan coolers are located in adjacent north quadrants of the containment. Similarly, the B fan coolers are located in the same adjacent quadrants. The accident analysis results are acceptable crediting only one operable fan cooler.
←(EC-40195, R307)

The two cooling units on one level are supplied with cooling water from one loop of the CCWS, the remaining two cooling units receive water from the other CCWS loop.

The cooling coils are horizontal tube with continuous plate type copper fins mechanically bonded to the tubes. They are air tested under water at 400 psig. Cooling coils are selected to satisfy the maximum heat removal requirements of the fan cooler under the worst surface fouling conditions (i.e. 0.001) likely to occur in the water side of the coil. The coil design provides for rapid drainage of the large quantities of condensed steam thus preventing loss of capacity.

The casing of each fan cooler is a rugged heavily reinforced steel unit designed to withstand a minimum pressure differential of three psig with no permanent deflection. Each cooler fan is vane axial type with airfoil design type blades which are adjustable to vary the blade pitch. Fans were selected on the performance curve for an intermediate blade setting to allow a variation of 15 percent static pressure from design at rated flow by resetting blade pitch. Fan operation is stable over the range of this adjustment. The vane axial fan readily passes transient pressure waves without sustaining damage to fan rotor, casing or motor and without loss of function.

Fan anti-friction bearings have a B-10 rating life as defined by AntiFriction Bearing Manufacturer's Association (AFBMA) of 200,000 hours under normal operating conditions. Bearings are suitable for the containment environmental conditions given in Section 3.11.

Each fan cooler has a back draft damper at the fan discharge which prevents backflow through the fan cooler if it is not operating.

The safety-related portion of the CCS distribution duct work is designed for functional integrity in the post accident environments. The safety-related portion of the duct work and housing is identified in Figure 6.2-37.

Pressure relief dampers are not required. The safety-related portion of the CCS ductwork is reinforced to withstand internal and external pressure surges caused by LOCA or MSLB conditions.

Each containment fan cooler is provided with an access door and vaportight lights to facilitate periodic inspections. Plugged connections are provided in the casing for differential pressure testing with portable manometers.

During normal operation three of the four fan coolers are manually started from the main control room and operate at the higher of two speeds.

As described in Section 7.3, each one of the four fan coolers receive a safety injection actuation signal (SIAS) following a LOCA or MSLB. Upon receipt of a SIAS, the fan coolers are automatically energized and placed in operation. The fans of all units operate at low speed and dampers D69 (SA) and D70 (SB) on the Safety Class 2 ducts automatically open. The dampers are pneumatically operated, Safety Class 2, and seismically qualified.

Emergency operation of the containment fan coolers can not be manually initiated from the main control room with a control switch during normal operation of the plant. However, it can be put into emergency (low speed) operation manually after actuation of SIAS.

The fan coolers will remain operational in the emergency mode until the effects of the accident have mitigated to the extent that the operator determines that the fan coolers are no longer required.

The sequence of events and the associated time delays for starting and bringing the CCS to full operation, with and without offsite power is given in Tables 6.2-23 and 24.

The capability of the fan coolers to remove the post accident heat load is verified by the Manufacturer by tests conducted on prototype fan cooler cooling coils.

The environmental tests are based on the conditions described in Subsection 3.11.1.

Tests on cooling coil performance under simulated design basis accident conditions were performed at the American Air Filter Test Facility. The tests indicate that the calculated heat transfer capability of full-size finned coils is valid for conditions ranging up to 289°F and 68.3 psia. These qualification tests are described in Topical Report AAF-TR-7101, "Design and Testing of Fan Cooler Filter Systems for Nuclear Applications".

6.2.2.2.2 Containment Spray System

The CSS consists of two independent and redundant loops each containing a spray pump, shutdown heat exchanger, piping, valves, spray headers and spray nozzles. The system has two modes of operation which are:

a) the injection mode, during which the system sprays borated water from the refueling water storage pool (RWSP) into the containment, and

b) the recirculation mode, which is automatically initiated by the Recirculation Actuation Signal (RAS) after low level is reached in the RWSP. During this mode of operation, suction for the spray pumps is from the safety injection system sump.

The equipment in the CSS is designed to the codes and standards listed in Table 6.2-22 and is Safety Class 2, seismic Category I.

Containment spray is automatically initiated by the containment spray actuation signal (CSAS) which is a coincidence of safety injection actuation signal (SIAS) and the high-high containment pressure signal. If required, the operator can manually actuate the system from the main control room. Refer to Section 7.3 for a discussion of the CSAS circuitry.

→(DRN 01-230, R12)
←(DRN 01-230, R12)

→(DRN 03-1355, R13)
The two containment spray pumps are horizontal centrifugal pumps driven by direct coupled induction motors. Each pump is rated for 1810 gpm flow (including 60 gpm recirculation requirement) at a head of 485 ft. Each pump together with a CCS loop provides the flow necessary to remove the heat generated inside the containment following a LOCA or MSLB. The flow characteristics of the pumps are shown on Figure 6.2-40, and pump design data is summarized in Table 6.2-22. The pumps are located at elevation -35 ft. MSL in the Reactor Auxiliary Building. The containment spray pumps are capable of continuous operation for a period of not less than one year without loss of design performance.

The sequence of events and the associated time delays for starting and bringing the CSS to full operation, with and without offsite power is given in Table 6.2-23 and 24.

→(DRN 05-12, R14)
Accident analyses typically assume 1.0 second for ESFAS response times (TRM) to account for sensor and processing time delays and inaccuracies. The CSS pump starting time is modified to include an additional time delay to enhance spray isolation valve performance. Due to the delayed pump start, a more realistic signal processing time has been accounted for in assessing the impact on the MSLB peak containment pressure analysis (shown in Table 6.2-23). The ESFAS signal time of 0.4 seconds is based on actual values from testing results and includes conservatism to bound expected performance.
←(DRN 05-12, R14)

The values in Table 6.2-23 [4.6 seconds (spray pumps up to speed) and 10.4 seconds (spray isolation valves fully open) respectively] are based on the 0.4 second processing time. CSS response times are acceptable as long as the spray pumps come up to speed (reach discharge pressure) in no more than 4.6 seconds and the CS-125A(B) valves open fully in no more than 10.4 seconds. These values are based upon a total acceptable CSS pump start delay of ≤ 2.5 seconds after CSAS.

Flow restriction orifices are installed in the spray pump minimum flow recirculation lines between pump discharge and RWSP. This alternative flow path ensures that the pumps are not damaged by inadvertently running against a closed system.

Upon system activation the containment spray pumps are started and the borated water flows into the containment spray headers. The spray headers are located to maximize heat removal. Each header conforms to the shape of the containment dome. Figures 6.2-41, 42, 43 and 44 show details of the spray headers. The headers are located outside of and above the movable missile shield and contain 116 spray nozzles each. During normal plant operation, CSS piping is maintained full of water from the RWSP to elevation +149.5 ft. MSL, in the 10 in. diameter risers within the containment. This reduces the system response delay time after actuation.

The spray nozzles, which are of open throat design, break the flow into small droplets, which enhances the cooling effect on the containment atmosphere. As these droplets fall through the containment atmosphere they absorb heat until they reach the temperature of the containment airsteam mixture. In order that the spray droplets attain thermal equilibrium with the containment atmosphere during the fall, approximately 120ft has been provided between the spray nozzles and the top of the steam generators. When the water reaches containment floors, it drains toward the SIS sump through a number of flow paths. Each spray nozzle is designed for a flow of 15.2 gpm with a 40 psi pressure drop across the nozzle. These nozzles have 3/8 in. spray orifices and are not subject to clogging. The nozzles are designed to produce droplets of approximately 700 microns mean diameter at rated system conditions.
←(DRN 03-1355, R13)

→(DRN 03-1355, R13)

The RWSP, located in the Reactor Auxiliary Building at elevation -4 ft. MSL, is a 600,000 gallon capacity pool containing borated water with a boron concentration of between 2050 and 2900 ppm. The stainless steel liner reaches to +16 ft. MSL elevation with a total volume in the RWSP to the top of the liner of 603,299 gallons. The RWSP is built to seismic Category I requirements. The pool is vented to atmosphere and has Safety Class 3 vacuum breakers to prevent pool damage during discharge. The pool contains vortex breakers at the two ECCS pumps suction headers to prevent air entrainment to the ECCS pumps when the RWSP is at its lowest levels.

When low level is reached in the RWSP sufficient water has been transferred to the containment, at least 383,000 gallons, to allow for the recirculation mode of operation. Spray pump suction is automatically realigned to the SIS sump upon RAS, which is described in Section 7.3. Automatic realignment of suction requires opening the valves in the recirculation lines. The operator then closes the RWSP isolation valves and the valves in the miniflow recirculation lines.

To assure adequate supply of water for the pumps during suction transfer, the sump valves are designed to be fully open within a nominal time of 25 seconds and a maximum time of 35 seconds.

During the recirculation mode, the spray water is cooled by the shutdown heat exchangers prior to discharge into the containment. The shutdown heat exchangers are cooled by the CCWS which is described in Subsection 9.2.2.

All components of the system which come in contact with the system fluid are made of stainless steel to minimize corrosion.

The operability of spray pumps will be tested by using a recirculation line which returns the flow to the RWSP. The recirculation line is located upstream of the shutdown heat exchangers.

6.2.2.2.2.1 SIS Sump Design

The SIS sump is a large collecting reservoir designed to provide an adequate supply of water to the CSS and SIS during the recirculation mode. Design of the SIS sump meets Regulatory Guide 1.82 with the exceptions and/or clarification's as described in the following paragraphs.

→(DRN 06-1026, R15)

One seismic Category SIS sump is provided. Two separate inlets one for each redundant half of the ECCS and CSS are provided. The inlets are separated from each other by a partition screen. The design of the sump has been completed prior to issuance of Regulatory Guide 1.82 and therefore, two separate sumps have not been provided.

←(DRN 06-1026, R15)

There are no high energy systems in the vicinity of the SIS sump, and therefore damage to the sump intake screens by whipping pipes or high velocity jets is prevented by separation.

The bottom of the sump is at elevation -16 ft. MSL, which is below the lowest floor elevation inside the containment exclusive of the reactor vessel cavity and the containment sump.

→(DRN 06-1026, R15)

The sump intake which is at the floor elevation of -11 ft. MSL is enclosed by strainer assembly. Figure 6.2-45 provides plan elevations of SIS sump strainer assembly. Figure 6.2-45a provides details of the grating cage installed over each sump intake for vortex suppression.

←(DRN 03-1355, R13; 06-1026, R15)

→(DRN 03-1355, R13)

A separate trash rack as recommended by Regulatory Guide 1.82 is not provided, because the design of the sump was completed prior to issue of the Regulatory Guide. To meet the Regulatory Guide 1.82, Position C-3 grating has been installed outside the steam generator compartments at elevation -4.0 ft. to prevent large debris entering the sump area. All insulated piping and equipment from which large debris can be emanated in this area is located above elevation -4.0 ft. The only major openings in the grating at elevation -4.0 feet are the stairs and penetration for the Reactor Drain Tank as shown on Figure 1.2-19. The sump area is protected by concrete shield walls from high energy systems from which the missiles can be generated.

→(DRN 06-1026, R15)

The strainer assembly and components are designed, fabricated and installed as Seismic Category 1, Nuclear Safety Related Components. The design ensures that the strainers are capable of withstanding the force of full debris loading, in conjunction with all design basis conditions, without collapse or structural damage. The strainer is designed to withstand the hydrodynamic loads and inertial effects of water in the containment basement, at full debris loading, without loss of structural integrity. The strainer assembly is made of stainless steel materials. The strainers are fully submerged after a LOCA and completion of safety injection.

The sump strainer consists of 11 strainer modules mounted on top of the plenum which is bolted to the floor. The modules are 40-inch square, each composed of 17 disks with the top disk having a solid plate. Perforated plates have 3/32-inch diameter holes staggered at 5/32-inch. The strainers prevent larger sized suspended particles from entering the sump. The 3/32-inch diameter hole was selected to avoid entrapment of particles in the fuel assembly spacer grids and to maximize the NPSH margin for HPSI and CS pumps.

The sump is partitioned in half by a Stainless steel grating plate attached to the sump floor and sides extending to the bottom of the plenum. The Sump partition separates Train A and Train B with a single piece of stainless steel grating. The grated partition will allow fluid mixing (including fine debris and particles) between the two (2) sump halves and not create completely separate screened in pump intakes. The partition maintains a single strainer assembly/single sump design that is available to both trains. Two separate 100% strainer assemblies (one per train) for single failure is not required and do not provide increased safety or protection in a common sump.

←(DRN 06-1026, R15)

→(DRN 00-314, R11)

SIS sump outlets are sized for less than 2.5 fps maximum velocity, and a minimum of 7.5 ft. submergence is provided above the centerline of the outlet pipes to prevent degrading effects such as vortexing. The sump outlet piping is enclosed by a grating cage which eliminates circular motion and precludes the possibility of vortex formation.

←(DRN 00-314, R11)

The floor level in the vicinity of the sump slopes down away from the SIS sump and towards the containment sump floor drains. This prevents minor leakage's (during plant operation) from entering the SIS sump and also prevents high density particles from moving towards the SIS sump.

→(DRN 00-204, R11; 03-2060, R14; 06-1026, R15; EC-1002, R302)

The sump intake will have a low approach velocity (0.025 ft/sec at design flow rate). The perforated velocity is calculated to be 0.004 ft/sec. At the start of recirculation, the water level inside the containment is above the top of the sump strainer assembly.

←(DRN 00-204, R11; 03-2060, R14; 06-1026, R15; EC-1002, R302)

←(DRN 03-1355, R13)

→(DRN 03-1355, R13)

Two carbon steel guard pipes extend from the SIS sump inside containment to outside of the Reactor Building. Each pipe is directly welded to a steel containment nozzle and is an extension of the nozzle in both directions. Passing through each carbon steel pipe is a stainless steel suction line which connects the SIS sump to the suction of the pumps. The stainless steel pipes are welded to the carbon steel pipes so that water cannot enter the annulus formed by the concentric pipes. Outside the containment the stainless steel pipes are sealed to the guard pipes by means of stainless steel bellows. The bellows seal allows for anticipated differential movement due to thermal or seismic forces.

→(EC-1002-, R302)

A full scale hydraulic model study has been performed to demonstrate that the design of the SIS sump will permit operation of the Containment Spray System and Safety Injection System without vortex formation. Grating cages have been installed at each intake to eliminate any rotational flow that is required to produce vortices. The hydraulic model study has been performed using these grating cages, and it has been demonstrated that the CSS and SIS can be operated in the recirculation mode without the formation of vortices. During testing of the sump strainers, no vortexing or air entrainment was observed.

←(EC-1002, R302)

→(EC-999, R302)

In response to GSI-191, Waterford 3 performed a detailed evaluation of the LOCA debris effects in accordance with guidance provided by the NRC Generic Letter 2004-02 and NEI 04-07. The updated design basis includes the following:

1. Debris quantities generated by a LOCA and transported in the sump are determined by analysis that established the distance from a break; at which various insulation, coatings, and other detrimental materials are released. A limiting break is determined from this analysis based on the debris mixture that produces the most detrimental strainer head loss.
2. Chemical effects associated with the precipitation of chemical compounds are evaluated for their impact on sump strainer head loss.
3. Strainer head loss due to debris and chemical precipitates was determined via scaled testing of strainer modules. The head loss test results were compared against available NPSH margins or strainer structural limits for acceptability.
4. Downstream effects are evaluated for components in the sump recirculation flow path to ensure debris generated by a LOCA that can pass through the strainer openings will not block flow or provide unacceptable wear.
5. The reactor vessel internals, including fuel assemblies, are evaluated for potential detrimental effects from debris generated by a LOCA, including chemical effects precipitates, which can pass through the strainer.

←(EC-999, R302)

6.2.2.2.2.2 Insulation

Piping and equipment insulation is considered to be the primary source of post accident debris inside containment which could potentially clog the SIS sump screening.

Thermal insulation used inside containment consists primarily of metallic reflective insulation, jacketed nonmetallic insulation or thermal wrap blanket insulation. The quantities and locations of each type of insulation are given in Table 6.2-25. The materials of construction of each type of insulation are identified and discussed in Table 6.2-26.

All insulation assemblies are designed to be self supporting from the associated piping and equipment or from adjacent removable or permanent coverings. Permanent insulation assemblies are attached by stainless steel straps and fasteners of the expansion type which prevent overstressing of the bands or damage to the insulation coverings due to thermal expansion of the equipment surface. Removable assemblies are attached by means of stainless steel buckles or other fasteners of the quick release type which vary depending upon installation requirements.

←(DRN 03-1355, R13)

➔(DRN 03-1355, R13; EC-8437, R307)
The only insulation related debris which after a LOCA might pass through safety sump fine intake screens are individual glass fibers. Containment Spray System model tests have demonstrated that the fibers so ingested will not affect pump or nozzle operation (Reference 3). As part of the changeout of the steam generators, new metal reflective insulation was installed to insulate the Replacement Steam Generators (RSGs) and RCS piping locations, reducing the fiber loading on the sump.
🡐(EC-8437, R307)

6.2.2.3 Design Evaluation

6.2.2.3.1 Containment Cooling System

The heat removal rate of each fan cooler as a function of the containment atmosphere temperature under LOCA conditions is shown graphically on Figure 6.2-12a.

The containment fan cooler heat removal rate for all containment pressure, temperature transients is based on the containment saturation temperature. The effect of the containment superheat is conservatively neglected in the calculation of fan cooler heat removal rate following a MSLB.

A fan cooler heat transfer test program was conducted by American Air Filter. The temperature, pressure and humidity were varied for each test. This was done by varying the steam-to-air ratio and velocity of saturated air entering the cooling coil. Heat removal capacity of the coil was measured for specified cooling water flow rates.

Test data is documented in reports furnished by American Air Filter. The results are analogous to those described in Topical Report AAF-TR-7101.

Surface fouling on the cooling coil interior surface by the component cooling water depends on water quality. The water is chemically treated to remove hardness and a corrosion inhibitor is added. The fouling factor may increase slightly as a surface film builds up on the interior surface over an extended time. However, this does not significantly reduce the heat removal capacity of the cooling coil.

A failure mode and effects analysis for the CCS is given in Table 6.2-27.

6.2.2.3.2 Containment Spray System

All the components of the CSS except headers and spray nozzles are located outside the containment and therefore do not have to withstand the post accident environment. Spray nozzles and piping inside the containment are qualified for the containment accident environment discussed in Subsection 3.11.1.

The CSS pumps are located in separate rooms at elevation -35 ft. MSL in the Reactor Auxiliary Building. A separate suction line is provided to each pump from the SIS sump. Both lines are enclosed separately in steel guard pipes from the SIS sump to but not including the containment isolation valves. Missiles can not impair both lines and common flooding is not expected in both rooms.

The CSS design requirements are such that the possibility of spray nozzle failures are remote and if encountered are inconsequential to the system's overall performance. This is made possible by the system's two redundant headers with 116 spray nozzles per header and the spray nozzles' unobstructed passage design. The spray orifice is sized larger (3/8 in. diameter) than the largest particle expected in the spray water. However, the spray orifice is still small enough to provide droplets that satisfy containment cooling requirements.

A random sample of typical spray nozzles were tested (1) and the results indicated that a test pressure drop of 40 psig produced an average flow rate of 15.1 gpm and a spray angle of 57 degrees. Previous testing (2) has confirmed the satisfactory performance of the spray nozzles in regard to droplet size and distribution. The average droplet size was determined by using high speed photographic sampling of a cross section of the spray cone. The average droplet size for the sample tested was 307 microns.
🡐(DRN 03-1355, R13)

→(DRN 03-1355, R13)

Actual testing of a sample of spray nozzles and previous performance testing indicate that the spray nozzles will perform satisfactorily during any postulated accident. Each CSS loop can cover over 85 percent of the total containment net free volume. The spray patterns are shown in Figures 6.2-41 through 44.

The RWSP capacity is based on the requirement for filling the refueling cavity and transfer tube to a depth of 24 ft. and is in excess of the requirements for containment spray and safety injection during any design basis accidents.

→(DRN 06-1026, R15)

Post-LOCA pH control is provided. Trisodium phosphate dodecahydrate (TSP) is stored in stainless steel baskets located in the containment near the SIS sump pump intake and in the northeast and northwest corners of containment elevation -11.0 feet.

←(DRN 06-1026, R15)

The baskets are constructed of stainless steel with mesh screen sides. Borated water from the spray will dissolve the TSP. Mixing is achieved as the solution is continuously recirculated from the sump to the spray nozzles, thereby raising the pH to at least 7.0.

A failure mode and effects analysis of the CSS is presented in Table 6.2-27 and it shows that no single failure of any component will degrade the ability of the CSS to fulfill its design requirements.

6.2.2.3.2.1 NPSH Calculations

The CSS design ensures compliance with the pump net positive suction head (NPSH) requirements of Regulatory Guide 1.1 (11/2/70). The maximum NPSH required by the pumps under all operating conditions is less than the NPSH calculated to be available, as determined from the suction piping layout and minimum elevation head.

The available NPSH for the pumps is calculated using a saturated sump model. The containment is conservatively assumed to be at the saturation pressure corresponding to the containment sump temperature. The available NPSH is calculated as follows:

NPSH (available) = $P_e - P_i + (P_s + P_a - P_v)$

Where: P_e = elevation pressure

P_i = friction pressure loss (including inlet losses)

P_s = containment steam pressure

P_a = partial pressure of containment atmosphere

P_v = vapor pressure of pumped water

Since $P_a + P_s = P_v$ for the saturated sump model,

NSPH (available) = $P_e - P_i$

Calculations show that adequate NPSH exists for all expected fluid temperatures without reliance on increased containment pressure.

←(DRN 03-1355, R13)

→(DRN 03-1355, R13; 06-1026, R15)

The parameters are as follows:

a) The calculation of NPSH in the recirculation mode considers that the containment pressure must be at least as high as the vapor pressure of the liquid in the SIS Pump. Therefore, the conservative assumption $P_a + P_s = P_v$ is used. This assures that the actual available NPSH is always greater than or equal to the calculated available NPSH.

b) The elevation pressure is calculated using the following:

1) Pump elevations is measured from the center of the pump volutes.

2) The elevation of the SIS sump water in the containment is based on a minimum water depth. In addition, the Containment Sump and all compartments in the lower part of the containment are filled with water.

c) The pressure drops in the suction lines are calculated using the following:

1) The use of pump maximum operating flows to calculate suction line friction losses.

2) Suction pipe lengths based on actual piping layouts.

→(EC-1002, R302)

3) Intake losses measured during the hydraulic testing of the sump design.

←(DRN 06-1026, R15; EC-1002, R302)

The minimum available NPSH during recirculation mode is as follows:

→(DRN 00-314, R11; 01-3706, R11-B; EC-1002, R302)

Pump	Flow (gpm)	NPSH (avail.) (ft)	NPSH (req.) (ft)	Margin (ft)	Pe (ft)	Pi (ft)
→(DRN 06-1026, R15; EC-25199, R307)						
Containment Spray	2250	23.02	18.63	4.42	27.21	4.16

←(DRN 06-1026, R15; EC-1002, R302; EC-25199, R307)

→(DRN 05-1332, R15)
←(DRN 05-1332, R15)

←(DRN 00-314, R11; 01-3706, R11-B)

←(DRN 03-1355, R13)

→(DRN 03-1355, R13)

6.2.2.3.2.2 Effects of Debris on SIS Sump Performance

A discussion of these effects is given in Subsection 6.1.2.4.

6.2.2.4 Test and Inspections

6.2.2.4.1 Containment Cooling System

The performance of the containment fan coolers is demonstrated by shop and jobsite testing in accordance with specified procedures. Certified reports covering test procedures, execution, equipment used and results of all factory and field testing is furnished.

→(EC-40281, R307)

The suitability of bearings and fan motors to operate under simulated post-LOCA conditions are established by prototype tests. Tests and calculations are made to verify performance of cooling coils and motors. Motor tests are conducted in accordance with IEEE-334-74/94, "Guide for Type Tests of Continuous Duty Class I Motors Installed Inside the Containment of Nuclear Power Generating Stations."

←(EC-40281, R307)

Cooling coils are pressure tested to 400 psig to demonstrate leaktightness during preoperational testing. Tests of the heat transfer capability of fullsize finned coils are performed to verify performance under simulated design basis accident conditions ranging up to 289°F at 68.3 psia.

The operating conditions include steam air mixtures at elevated temperatures and pressures. Details of such tests are provided in the American Air Filter Topical Report AAF-TR-7101, Addendum dated June 25, 1973.

Vane axial fans are tested according to AMCA (Air-Moving and Conditioning Association) standards and test results for performance and noise level through design pressure range are furnished by the fan manufacturer.

Each containment fan cooler is started by a simulated SIAS during preoperational testing. Periodic testing and inspection of CCS is described in the Technical Specifications.

6.2.2.4.2 Containment Spray System

Each pump casing is hydrostatically tested at 150 percent of maximum operating pressure.

Performance tests are conducted in the shop to establish pump characteristics. Suction pressure was adjusted to give the required NPSH to the pumps. Ten head capacity points including the design point were recorded. Developed head, brake horsepower and efficiency were determined at no flow, design flow and runout flow.

Suction tests are performed for each pump to determine NPSH requirements at various capacities.

The temperature transient test was also conducted for each pump to determine their ability to withstand the required thermal transient.

To verify that the CSS will perform as required, tests will be conducted either during reactor shutdowns for refueling, or during scheduled maintenance while the plant is on-line. For a further description of periodic testing and inspection refer to the Technical Specifications.

Leak testing will be done on that portion of the CSS outside containment as part of a leak reduction program as required by NUREG-0737

←(DRN 03-1355, R13)

→(DRN 03-1355, R13)

6.2.2.5 Instrumentation Requirements

The instrumentation associated with the CCS and the CSS provides information to the operator in the main control room for monitoring all modes of operation.

Indication is provided for each fan cooler and spray pump motor including "OPEN and "CLOSE" position of each containment isolation valve, and "OPEN and "CLOSE" position indication for each control valve.

a) Containment Fan Coolers

The air temperature entering and leaving each containment fan cooler is indicated and recorded in the main control room. Temperature indication is provided in the main control room for the CCWS water entering and leaving each fan cooler. Flow indication of CCWS water, leaving each fan cooler, is provided in the main control room. During normal plant operation, three out of the four fan cooler units are manually started from the main control room and operate at their normal (high speed) mode.

The CCWS lines, serving each pair of fan coolers have a control valve with multiple settings to increase or decrease the flow. Control valves CC-835A and CC-835B are controlled by mode of operation of the fan coolers.

The containment isolation valves, CC-807A, CC-808A, CC-808B, CC-807B, CC-823A, CC-822A, CC-822B, and CC-823B (two per fan cooler) are operated from the main control room in association with each of the four fans. During the normal operation mode, valves are opened when fan is running and closed when fan is tripped.

←(DRN 03-1355, R13)

A SIAS, as described in Section 7.3 starts the fan cooler emergency mode of operation. All four fan coolers start operating at the low speed mode and all isolation valves and temperature control valves will fully open upon receipt of a SIAS. Operator has the capability to override SIAS with control switch in main control room and trip the fan, however, all CCWS valves will remain open. Provision is made for remote manual isolation of these valves. Initiation of a Containment Isolation Actuation Signal (CIAS) has no effect on the position of isolation valves.

The following alarms are provided in the main control room for each containment fan cooler:

1) High fan bearing vibration.

2) High air discharge temperature.

3) Override switch out of “NORMAL” position.

b) Containment Spray System

The CSS is automatically actuated by a Containment Spray Actuation Signal (CSAS) as described in Subsection 7.3.1.1.3.

The CSAS is initiated from two actuation channels A and B with the instrumentation and controls for the equipment in loop A physically and electrically separate from the instrumentation and controls for the equipment in loop B. On initiation, the CS pump starts and the outboard containment isolation valve opens. Provision is made for remote manual isolation of this valve.

The recirculation mode of operation is automatically initiated by a Recirculation Actuation Signal (RAS) as described in Subsection 7.3.1.1.2.

SECTION 6.2.2: REFERENCES

→(DRN 03-2060, R14)
1) SPRACO Test Results dated November 18, 1976.
←(DRN 03-2060, R14)

2) SPRACO Droplet size booklet/performance data transmitted on October 28, 1976.

3) Topical Report OCF-1, Nuclear Containment Insulation System, on file with U. S. Nuclear Regulatory Commission.

6.2.3 SECONDARY CONTAINMENT FUNCTIONAL DESIGN

The secondary containment or Shield Building functional design provides for biological shielding, controlled release of the annulus atmosphere following a design basis accident, and environmental protection of the primary containment vessel.

The Shield Building is a concrete structure constructed to provide a four (4) ft. wide annular space between the interior wall of the Shield Building and the outer surface of the steel containment vessel. The annular space permits collection of primary containment outleakage and periodic visual inspection of the steel containment vessel. The containment vessel is protected from the effects of wind or tornado by the Shield Building.
The Shield Building facilitates the consolidation of potential air leakage through penetrations of the steel containment vessel. The Shield Building Ventilation System (SBVS) controls the release and removal of fission products which are filtered from the annulus atmosphere following a design basis accident.

6.2.3.1 Design Bases

The Shield Building and Shield Building Ventilation System (SBVS) are designed to:

→(DRN 05-785, R14)
a) Control the annulus pressure transient following a LOCA consistent with the radiological analyses, assuming a single active or passive failure (SBVS).
←(DRN 05-785, R14)

→(DRN 04-705, R14)
b) Provide mixing and dilution of radioactive materials which leak to the annulus by processing the annulus atmosphere through the particulate and charcoal filters to meet the guidelines of 10CFR50.67 before exhausting it to the environment (SBVS).
←(DRN 04-705, R14)

c) Hold-up the outleakage from the containment vessel within the Shield Building annulus. This excludes the fraction of the outleakage which is bypass leakage to the outside of the Shield Building.

→(DRN 04-705, R14)
d) Provide fission product removal capacity based on Alternative Source Terms (Regulatory Guide 1.183) and a containment vessel design leak rate of 0.5 percent volume per day (SBVS).
←(DRN 04-705, R14)

e) Seismic Category I and safety class 2 requirements.

f) Withstand post accident design basis environmental conditions without loss of function.

g) Permit appropriate periodic inspection and periodic pressure and functional testing to assure system integrity and functional capability.

6.2.3.2 System Design

6.2.3.2.1 Shield Building

The Shield Building is designed as a Seismic Category I structure and its design features are further described in Subsection 3.8.4. Plan and Elevation drawings are provided in Section 1.2.

The design and performance data for the Shield Building is provided in Table 6.2-28. Industry codes and standards and Regulatory Guides applied in the design of the Shield Building are listed in Subsection 3.8.3.2.
The combined performance objectives of the Shield Building structure and the SBVS is to assure to the general public adequate protection from accidental release of radioactivity following a design basis accident. The primary containment outleakage which flows into the annular volume of the Shield Building is filtered by the SBVS for removal of fission products before it is released to the atmosphere.

The Containment Isolation System is discussed in Subsection 6.2.4 and isolation valve design features are described in Subsection 6.2.4.3. The penetration types are defined, and their design features are discussed in Subsection 3.8.2.1.

The Plant Protection System signals which activate the isolation valves are listed in Table 6.2-32, where containment penetrations and their isolation valves are listed according to penetration types and isolation class.

The Shield Building design objectives include the establishment of an essentially leaktight barrier against uncontrolled release of radioactivity. The barrier is inherent in the Shield Building which provides the annular space to consolidate the primary containment outleakage, and the SBVS which maintains a negative pressure inside the annulus and thus prevents the leakage from flowing directly from the annular space through the Shield Building structure to the atmosphere.

A fraction of the primary containment outleakage occurs through penetrations which includes isolation valves, seals, gaskets and welds as possible sources for potential bypass leakage paths. These penetrations pass through the primary containment steel vessel as well as the Shield Building and are identified as potential bypass leakage paths in Subsection 6.2.4.4.

An evaluation of potential bypass leakage paths considering realistic equipment design limitations and test sensitivities is provided in Subsection 6.2.4.3.1.

Isolation valves which are subject to potential leakage are tested according to "Type C" tests, and seals, gaskets and welds are tested according to "Type B" tests defined in Appendix J to 10CFR50. Testing of bypass leakage paths and determination of the bypass leakage fraction are provided in the Technical Specifications.

All high energy pipes penetrating the Shield Building annulus are provided with guard pipes. Utilization of guard pipes for specific penetrations is described in Subsection 3.8.2.1.

6.2.3.2.2 Shield Building Ventilation System

a) System Description

→(DRN 05-785, R14)

The SBVS primary function is to assure that annulus pressure following a LOCA is maintained within the value assumed in the radiological analyses. A positive annulus pressure would permit primary containment outleakage to escape unfiltered directly through the Shield Building wall to the outside atmosphere.

←(DRN 05-785, R14)

The SBVS provides dilution, mixing, high efficiency removal of fission products by filtration, and hold-up time by recirculation, (refer to Subsection 6.5.3). The Shield Building Ventilation System is shown in Figures 9.4-7 and 9.4-3 (for Figure 9.4-3, Sheet 5, refer to Drawing G853, Sheet 19). Each redundant unit of the SBVS consists of one full capacity fan and one engineered safety feature (ESF) filter train, ductwork, valves and associated instruments. The fans and ESF filters are located in the Reactor Auxiliary Building HVAC Equipment Room at El 46.0 ft. MSL.

Each fan can either discharge to the stack through its main butterfly valve or return air to the annulus through its recirculation butterfly valve.

The SBVS filter train consists of a demister, electric heating coil, medium efficiency prefilter, pre-HEPA filter, charcoal adsorber and after-HEPA filter. The SBVS filter system is described in Subsection 6.5.1. Refer to Table 6.2-28 for design data and materials of construction for SBVS components. A comparison of the SBVS with the positions of Regulatory Guide 1.52 is presented in Table 6.5-1.

Following actuation of the SBVS, air is drawn through circumferential headers located in the upper and lower annulus regions. Each of the two circumferential headers has air intake openings spaced in an approximately uniform pattern over its entire circumference. Air enters the upper region exhaust header (center line El. 175.00 ft. MSL) through air openings at the top of the header and enters the lower region exhaust header (centerline El. varies from 22.08 ft. to 35.5 ft. MSL) through openings at the bottom of the header.

The exhaust headers are connected by an exhaust riser, to which are attached two horizontal 30 in. diameter exhaust ducts, one at centerline elevation 57.50 ft. MSL and the other at centerline elevation 63.25 ft. MSL. These 30 in. diameter exhaust ducts penetrate the Shield Building and run in the Reactor Auxiliary Building to their termination points where one is connected to a SBVS unit intake plenum and the other is connected to the other SBVS unit intake plenum.

The filtered air that is drawn through the fan and flows through the discharge duct to plant stack or is recirculated back to annulus to be refiltered, is controlled by the butterfly valves during the two phases of operation, as described below:

Phase 1: This is the exhaust phase of operation when all flow is exhausted to the plant stack.

Phase 2: During the second or zero venting stage, all flow is recirculated back to the circumferential supply header located in the annulus at centerline elevation 100 ft. MSL. The air within the supply header is then dispersed through nozzles, intermittently located on the top and bottom of the ring header, and mixes within the annulus air volume.

A six in. diameter line with a manual isolating butterfly valve 2HV-Bl66 cross-connects the two SBVS filter trains downstream of the filter banks and upstream of the fans to maintain flow through the non-operating filter train in order to remove any decay heat generated in its charcoal adsorber. A gravity damper is located at the discharge of each fan to prevent backflow through the nonoperating filter train and the subsequent loss of capacity of the operating filter train.

b) System Operation

One or both of the SBVS trains operate through the two phases described in Subsection 6.2.3.2.2.a. The two safety-related exhaust fans, and filter trains, and their valves are redundant and operate independently. Electric motor operated valves for the operating train or trains function as shown in Table 6.2-30.

Both SBVS trains are started after the receipt of a Safety Injection Actuation Signal (SIAS) and the system begins "phase 1" operation, at which time the filter trains inlet, outlet, and main exhaust valves in each train are opened. The two exhaust fans are energized through limit switches on the inlet valve, outlet valve and main exhaust valves which act as permissives for the fans to start. After the fans are started, the operator has the option to deactivate one SBVS filter train system and place it on standby after confirming that the other system is operating.

An air flow failure of the operating train is sensed by low differential pressure across the filter train and automatically starts the standby system. When the standby train is started in place of the other train, its valves will be set in positions similar to the valves of the deactivated train so that it operates in the same phase. A differential pressure transmitter across the filter train will indicate when airflow is established.

Each SBVS filter train system has an electric heating coil which is energized when the system starts, to limit the relative humidity of the air entering the charcoal adsorber to a maximum of 70 percent in order to prevent degradation of the adsorber efficiency. When a SBVS filter train is stopped, all valves, (except for the inlet valve of the filter train) go to their fully closed positions. The inlet valve of the filter train of the non-operating unit remains open so that the adequate air flow will always be available for filter cooling as long as the other unit operates.

Following a LOCA and receipt of a SIAS, the Annulus Negative Pressure System is deactivated and a transient condition exists in the Shield Building annulus until the SBVS is in full stable operation. The resultant pressure and temperature of the annular space, which increases due to containment vessel expansion and heat transfer through the vessel wall, is reduced by the operation of the SBVS.

For a postulated accident, such as a LOCA coincident with loss of offsite power, a conservative time of 30 seconds for SBVS fans to reach rated speed after start of LOCA was used in the analysis. When the SBVS is in "phase 1" operation all air is filtered at a maximum initial exhaust rate of 10,000 acfm approximately, in order to reduce the annulus pressure after a draw-down period to a setpoint of -8 in. WG below atmospheric pressure which corresponds to an operation period equal to or less than four minutes.

At this setpoint, the SBVS operation will change to "phase 2" or the recirculation mode; that is, the following sequence of operation is initiated by controls measuring annulus to atmosphere differential pressure.

1) The exhaust valve is closed.

2) The valve in the recirculation line is opened.

→(DRN 02-9, R12)

During "phase 2", energy addition by heat transfer and equipment operation plus in-leakage from the containment and from the outside atmosphere slowly bring the annulus pressure back up to -3 in. WG.

←(DRN 02-9, R12)

The increase in annulus pressure is accelerated again after 1556 seconds when initiation of the sump recirculation mode in the containment causes an increase of containment pressure. This increased containment pressure causes a volume decrease in the annulus and this effect, combined with the additional heat transfer into the annulus resulting from the increased containment temperature, raises the annulus pressure at an increased rate.

→(DRN 02-9, R12)

When the increasing annulus pressure reaches -3 in. WG, the automatic control system changes the mode of operation of the SBVS to Phase I mode by closing the valves in the recirculation line and opening the exhaust valve. An exhaust flow beginning at approximately 10,000 acfm will cause the annulus pressure to decrease to -8 in. WG. At this point the system will change to "Phase 2" operation. A stable operation follows the transient, with the annulus pressure being maintained between -3 in. WG and -8 in. WG.

←(DRN 02-9, R12)

6.2.3.3 Design Evaluation

6.2.3.3.1 Performance Requirements and Capabilities

Each of the two full capacity redundant fan-filter trains of the Shield Building Ventilation System has been designed to fulfill the performance requirements seated in the design bases, Subsection 6.2.3.1.

→(DRN 03-2060, R14)

The analysis of the functional capability of the SBVS to depressurize and maintain a uniform negative pressure within the Shield Building annulus is performed for the worst long term LOCA, the 9.82 ft^2 double ended suction leg slot break with maximum safety injection using the WATEMPT computer code described in Appendix 6.2A. Note that the description and results of the WATEMPT analysis of the shield building annulus in this section and in Appendix 6.2A is based on pre-uprate plant conditions. These original results are conservative and bounding since, due to improved modeling of mass and energy releases, the current 3716 MWt containment pressure and temperature response analysis determined that the peak LOCA pressure and temperature have decreased thereby decreasing the resulting transient in the shield building annulus. It must be noted that the worst case long term LOCA is the double ended discharge leg break with minimum safety injection flow based on the current 3716 MWt containment pressure and temperature response analysis.

The description of the development of the pipe break mass and energy release rate, the containment initial conditions and heat sinks, and the Containment Heat Removal System failures are discussed in Subsection 6.2.1. In addition, only one train of the Shield Building Ventilation System is assumed to operate. These assumptions represent a very conservative case since maximum safety injection is postulated with an assumed loss of offsite power to maximize the long term mass and energy releases, in addition to a failure of one train of the Containment Heat Removal Systems, and a failure of one train of the SBVS. Any additional initial conditions or changes from those listed in Subsection 6.2.1 are contained on Table 6.2-29. The same heat transfer coefficients are applied whether the surface temperature exceeds the annulus atmosphere or the annulus atmosphere temperature exceeds the surface temperature. The analysis results include the following information:

←(DRN 03-2060, R14)

a) Containment pressure and temperature as function of time (Figures 6.2-46, 6.2-48)

→(DRN 05-785, R14)

b) Annulus pressure and temperature as function of time (Figures 6.2-47a,b, 6.2-48). Note that the annulus transient could become positive for a short period of time when stratification and instrument uncertainty are accounted for; this is not reflected in Figures 6.2-47a or 6.2-47b which assume a nominal starting value for annulus pressure.

←(DRN 05-785, R14)

c) Containment vessel wall temperature gradient as a function of time and distance (Figure 6.2-50)

d) Containment vessel surface temperature as a function of time (Figure 6.2-49)

e) Shield Building surface temperature as a function of time (Figure 6.2-49a)

f) SBVS purge and recirculation flow rate as a function of flow differential pressure (Figures 6.2-51 and 6.2-47c)

An analysis was performed using 50 percent humidity as an initial condition inside the annulus. There were no observable variations from the 100 percent humidity case. The dominant factors for the annulus transient analysis are vessel expansion and in-leakage rates. Therefore, the results of varying the initial condition humidity inside the annulus are negligible. Consequently the figures shown in this section are applicable.

6.2.3.3.2 Single Failure Analysis

Two redundant SBVS trains are provided, either of which is capable of meeting the requirements of the system design bases. Each train is actuated by a separate SIAS actuation channel. As explained in Section 8.3, all redundant active components are powered from separate ESF buses.

Consequently, the Shield Building Ventilation System design function is not compromised by any single failure. Refer to Table 6.2-31 for failure modes and effects analysis for SBVS.

→(EC-5000082406, R301)
An analysis has been conducted which shows that a single active failure will not result in an air flow through a recently shutdown filter train of less than 1020 acfm, which ensures the adsorber desorption temperature of 300°F is not exceeded.
←(EC-5000082406, R301)

The single failure criterion has not been applied to common ductwork serving redundant safety related equipment. No ductwork operates at pressures greater than 20 in. WG (about 0.72 psi) so that rupture of the ductwork could not occur due to the pressure of the contained fluid.

Common ductwork serving safety related equipment occurs at the following:

a) Duct which cross-connects the two SBVS filter trains.

b) Annulus exhaust and supply ring headers.

c) Exhaust riser connecting the two intake ring headers.

d) Supply riser connecting both SBVS filter trains to supply ring header.

In each case the ducting is located in areas where there is no high pressure piping that could result in pipe whip or jet impingement. Similarly, the ducting is not located in areas where internally generated missiles or failure of non-seismic structures or components could affect system operation. Leakage from cracks in the ductwork will not prevent the vent system from performing its safety function. All high energy piping passing through both the primary containment vessel and the Shield Building wall is provided with guard pipes.

Access doors are two door, airlock type with entry administratively controlled.

6.2.3.3.3 Effectiveness of Mixing

→(DRN 04-705, R14)
The SBVS provides for LOCA dose reduction by recirculation to clean up iodine isotopes and hold up decay of noble gas isotopes. Mixing occurs both within the SBVS ductwork and within the annulus.
←(DRN 04-705, R14)

6.2.3.4 Test and Inspections

Witnessed, certified factory performance tests are conducted on one of the centrifugal fans. Fan curves are supplied showing brake horsepower, air temperature, static pressure, and static efficiency. Curves are plotted against fan capacity in cfm. See Figure 6.2-51. Testing of components is described in Subsection 6.5.1.4. This includes Visual Inspection, Testing and Balancing of System, Duct and Annulus Tracer Mixing Test, Filter Casing Leak Test, Air-Aerosol Mixing Uniformity Test and Inplace Leak Test for HEPA filters and Charcoal Adsorbers. Preoperational testing of the Shield Building Ventilation System consists of verifying performance through actuation and operation of the system, including ability to maintain negative annulus pressure. Instrumentation is available to confirm design flow capabilities.

Technical Specifications provide additional information and procedures regarding periodic testing and inspection.

The preoperational tests conducted on the SBVS are discussed in FSAR Subsection 14.2.12.2.20.

6.2.3.5 Instrumentation Requirements

The instrumentation associated with the Shield Building Ventilation System provides the operator in the main control room with continuous system monitoring capability. Normally the air in the Shield Building is kept as negative pressure (outside pressure higher than inside pressure) by the Annulus Negative Pressure System (non-safety). During emergency conditions a rapid increase of the inside pressure will occur as a result of a Loss of Coolant Accident (LOCA). A SIAS will stop the operation of the Annulus Negative Pressure System (as described in Section 7.3) and start the Shield Building Ventilation System automatically. Class 1E instrumentation containing sensing devices which measure the differential pressure between the Shield Building and the outside atmosphere are provided in order to maintain the proper mode of operation by positioning motor operated valves serving each filter train as described in Subsection 6.2.3.2.2.

The following safety related indication is provided in the main control room:

a) Status of SBVS fans.

b) Position of filter train inlet and outlet valves, recirculation valves to annulus, and the main exhaust valve to stack.

Additionally non-1E local differential pressure indicators are provided for each SBVS filter train to measure the pressure drop across the demister, roughing filter, upstream HEPA filter, and downstream HEPA filter.

The control room operator is provided with the following safety related alarms for each Shield Building Ventilation System filter train:

a) Air flow failure for operating filter train indicated by low differential pressure across the filter train during any phase of operation, will, after a time delay, activate the alarm.

b) High differential pressure across the filter train, to alarm high filter loading.

→(DRN 02-9, R12)

c) Annulus high pressure, above 3 in. WG (VAC).

←(DRN 02-9, R12)

d) Low differential temperature measured between points upstream and downstream of the electric heating coil to indicate failure of this component and consequent termination of fan operation.

6.2.4 CONTAINMENT ISOLATION SYSTEM

The Containment Isolation System (CIS) provides the means for isolating fluid systems that pass through the containment in order to confine any radioactivity that may be released following a LOCA or main steam line break (MSLB) inside containment. The containment purge isolation signal (CPIS) also isolates the containment purge upon high radiation inside the containment (see Section 7.6).

6.2.4.1 Design Bases

6.2.4 1.1 Conditions Requiring Isolation

a) Automatic initiation of a containment isolation actuation signal (CIAS) occurs when containment high pressure or low pressure in pressurizer is detected. The CIAS closes all automatic isolation valves in the lines penetrating the containment except for:

→(DRN 03-2060, R14)

- those required for the operation of the engineered safety features systems

←(DRN 03-2060, R14)

- the Main Steam Isolation Valves and the Main Feedwater Isolation Valves

- the Component Cooling Water penetrations servicing the Reactor Coolant Pumps and CEDM coolers which automatically close on a Containment Spray Actuation Signal (CSAS).

b) Main steam line and main feedwater isolation valves are closed automatically following a secondary system line break or detection of high pressure inside the containment. MSIS is generated by low steam generator pressure or high containment pressure.

c) The containment purge is automatically isolated upon receipt of a high radiation inside containment signal and high activity in the plant stack, and the ESF signal CIAS.

6.2.4.1.2 Criteria for Isolation of Fluid Systems Penetrating the Containment

a) Valves isolating penetrating lines serving engineered safety feature systems are not closed automatically by the CIAS, but have the ability to be closed by remote manual operation from the main control room or auxiliary relay room, thereby isolating any engineered safety feature system which malfunctions.

b) All penetration assemblies and containment isolation valves are seismic Category I, Safety 2 Class (except for valves, inside containment, connected to the RCPB which are Safety Class 1) and are protected from the effects of missiles and pipe break.

c) All piping penetrating the containment vessel shell is designed to withstand at least a pressure and temperature equal to the containment vessel design internal pressure and temperature and to withstand the post-accident transient environment. In each of the four classes of penetrations listed below, at least two barriers are provided between the containment atmosphere and the outside atmosphere, so that failure of one valve to close does not prevent isolation.

Class A1

Penetrations in class A1 are for lines connected directly to the containment atmosphere* that are normally open, or may be open during normal operation. Each line is provided with two valves in series with either:

1) One automatic isolation valve inside and one locked closed isolation valve outside containment, or

2) One automatic isolation valve inside and one automatic isolation valve outside containment.

Class A2

Penetrations in class A2 are for lines connected directly to the containment atmosphere* that are normally closed during normal operation. Each line is provided with two valves in series with either:

1) One locked closed isolation valve inside and one locked closed isolation valve outside containment, or

2) One automatic isolation valve inside and one locked closed isolation valve outside containment, or

3) One locked closed isolation valve inside and one automatic isolation valve outside containment, or

4) One automatic isolation valve inside and one automatic isolation valve outside containment.

Class B1

Penetrations in this class are for lines that are connected to the reactor coolant pressure boundary* that are normally open during operation. Each line is provided with two valves in series, with either:

1) One automatic isolation valve inside and one locked closed isolation valve outside containment, or

2) One automatic isolation valve inside and one automatic isolation valve outside.

Class B2

Penetrations in this class are for lines that are connected to the reactor coolant pressure boundary* which are normally closed and never open during normal operation. Each line is provided with two valves in series, either:

1) One locked closed isolation valve inside and one locked closed isolation valve outside containment, or

2) One automatic isolation valve inside and one locked closed valve outside containment, or

3) One locked closed isolation valve inside and one automatic isolation valve outside containment, (A simple check valve may not be used as the automatic isolation valve outside containment) or

4) One automatic isolation valve inside and one automatic isolation valve outside containment. (A simple check valve may not be used as the automatic isolation valve outside containment.)

Class C

Penetrations in this class are for lines which are neither connected to the reactor coolant pressure boundary nor connected directly to the containment atmosphere but connected to a closed seismic Category I system inside containment. Each line shall be provided with at least one containment isolation valve outside the containment which shall be either automatic or locked closed, or capable of remote manual operation.

*excluding engineered safety features systems

Class D

Penetrations in this class are for lines with a pressure barrier which are either connected to the reactor coolant pressure boundary or connected directly to the containment atmosphere. A pressure barrier is said to exist when the line is part of a closed system outside the containment which is designed for pressure equal to or greater than containment design pressure. Each line is provided with at least one containment isolation valve outside the containment which shall be either automatic or locked closed or capable of remote manual operation.

A simple check valve is not used as the automatic isolation valve outside the containment. An automatic valve or a locked closed valve may be used as the isolation valve outside the containment.

d) Isolation valves outside the containment are located as close to containment as practicable and upon loss of actuating power, automatic isolation valves are designed to take the position that provides greater safety. Each valve can be periodically tested to insure its operability when needed. All remotely operated isolation valves are provided with position indicators in the main control room, and control switches in the control room or auxiliary relay room.

6.2.4.1.3 Criteria For Isolation of Fluid Instrument Lines Penetrating the Containment

All fluid instrument lines penetrating the containment are in compliance with ANS-56.2/ANSI N271-1976 except the containment vacuum relief non-essential line is equipped with redundant solenoid valves installed outside containment. The containment isolation provisions for the containment vacuum relief non-essential line was approved per Licensing Amendment 128.

6.2.4.1.4 Design Requirements For Containment Isolation Barriers

a) All containment isolation valves are designed to insure leaktightness and reliability of operation. All remotely operated valves are designed to fail as indicated in Table 6.2-32.

b) All manual valves which serve as containment isolation barriers shall be under administrative control and secured in the closed position.

c) The portions of the containment isolation system that are a part of the reactor coolant pressure boundary are designed and constructed in accordance with Quality Group A Classifications. The remainder of the system is Quality Group B, as defined in NRC Regulatory Guide 1.26.

d) The containment isolation shall not be affected by missiles, pipe whip and jet forces and earthquakes.

6.2.4.2 System Design

Table 6.2-32 provides design information regarding all containment isolation provisions.

6.2.4.2.1 Actuation Signal

The actuating signals to isolate the containment are itemized in Table 6.2-32. The development of these signals, including the quantity and setpoints of parameters sensed and actuation logic is discussed in Section 7.3. Containment isolation provisions comply with General Design Criteria 54, 55, 56, and 57 of 10CFR50, Appendix A.

6.2.4.2.2 Instrument Lines

The only fluid instrument lines penetrating primary containment are the containment vacuum relief pressure sensing lines through Penetratrations 53 and 65 for redundant systems SA and SB respectively, and two sensing lines for the wide range containment pressure instrumentation through penetration 54. Penetrations 53 and 65 each contain two instrument lines. One line senses differential pressure across the containment vessel and provides a signal to actuate the Vacuum Relief Valves; the other line monitors this differential pressure and provides an input to the plant monitoring computer. The actuation line is considered essential and is therefore provided with an excess flow check valve as recommended by Regulatory Guide 1.11. The monitoring line is provi ded with an excess flow check valve and a solenoid operated valve, closed on a containment isolation signal. The wide range containment pressure instrumentation, which utilizes Penetration 54 is a sealed fluid system.

6.2.4.2.3 Isolation Barriers Design Requirements

a) Regulatory Guide 1.26, "Quality Group Classifications and Standards for Water-, Steam-, and Radioactive Waste Containing Components of Nuclear Power Plants."

The CIS design is consistent with the recommendations of Regulatory Guide 1.26 as described in Subsection 3.2.2.

b) Regulatory Guide 1.29, "Seismic Design Requirements"

The CIS is designed to seismic Category I requirements. This is consistent with the recommendations of Regulatory Guide 1.29.

Further detail is given in Subsection 3.2.1.

c) Missile Protection

The CIS is protected from missiles, both internal and tornado generated, as discussed in Section 3.5.

d) Jet Forces and Pipe Whip

The CIS is protected from the effects of jet forces and pipe whip as discussed in Subsection 3.6.

e) Earthquakes

The CIS is designed to seismic Category I standards and is located in a seismic Category I structure. Further discussion can be found in Sections 3.7, 3.9 and 3.10.

f) Operability of Valves and Valve Operators

Electric Power to the containment isolation valve operators is provided from the ESF buses to ensure valve operation. The ability of manually and automatically operated valves to function during and after the SSE is proven by seismic analysis and/or testing. The operators are designed to open and close the valves under full differential pressure.

Qualification of the electrical components of the valve operator performing safety- related functions is in accordance with the recommendations of NRC Regulatory Guide 1.73 (Jan. 1974) "Qualification Tests of Electric Valve Operators Installed Inside the Containment of Nuclear Power Plants" by conformance to the following IEEE standards;

IEEE-323-71	"Draft General Guide for qualifying Class I Electrical Equipment".
IEEE-344-71	"Seismic Qualification of Class I Electric Equipment for Nuclear Power Generating Stations"
IEEE-382-72	Guide for Type Test of Class I Electric Valve Operators for Nuclear Power Generating Stations"

Further information on valve operability is provided in Section 3.9.

g) Qualification of Closed Systems and Valves Inside and Outside of the Containment as an Isolation Barrier

The design of penetrations for closed system are in accordance with General Design Criterion 57 of 10CFR50 Appendix A and are discussed in Subsection 6.2.4.1.2.c under Class C or D penetrations.

h) Qualification of a valve as an isolation barrier.

The containment isolation valves which may significantly affect radiological release to atmosphere, have been factory tested for leakage. to the "bubble tight" shutoff requirements. The remaining valves have been factory tested for leakage to the MSS-SP61 or the ANSI B16.104 standards as a minimum. The containment isolation valves will be field tested for leakage as indicated in Table 6.2-32 and described in Subsection 6.2.6.

i) Valve Closure Times

→(DRN 04-705, R14)

The closure times of the containment isolation valves are shown in Table 6.2-32.

←(DRN 04-705, R14)

In accordance with the requirements of Technical Specifications, all containment isolation valves are capable of being periodically tested to ensure that they continue to meet the required stroking time in accordance with the provisions of ASME Code, Section XI, Rules for In-service Testing of Valves in Nuclear Power Plants.

j) Mechanical and Electrical Redundancy

Redundant containment isolation valves as listed in Table 6.2-32 are provided. To avoid common mode failures the redundant automatic isolation valves which must operate following an accident are powered from independent sources.

Table 6.2-32 indicates the use of separate Class 1E power supplies to redundant isolation valve operators.

k) Primary and Secondary Modes of Actuation

The primary and secondary modes of actuation are listed in Table 6.2-32.

l) Provisions for Detecting Leakage from Remote Manually Controlled Systems.

Provisions are made to allow the operator in the main control room to know when to isolate, by remote manual means, fluid systems that have a post-accident safety function. These provisions include instrumentation to measure flow rates, pressure and sump water levels in the safety equipment area and the penetration area, and instrumentation to measure radiation levels.

m) Leakage Testing of Isolation Valves

Leakage testing of isolation valves is described in Subsection 6.2.6.

Appendix J "Type B" penetration tests are performed on the electrical penetrations, maintenance hatch, the fuel transfer tube flange and the personnel and emergency air locks. Electrical penetrations also meet the testing requirements of IEEE-317-1972.

Specific information on isolation valve testing, bypass leakage, penetration testing and acceptance criteria is presented in the Technical Specifications.

6.2.4.3 Design Evaluation

6.2.4.3.1 Performance Requirements and Capabilities

After a LOCA, the only potential paths of direct leakage through the isolation system would be from seismic Category I system penetrations which are directly connected to the Reactor Coolant System and from penetrations for nonseismic piping systems. The first group would be directly open to the containment atmosphere following a LOCA. The second group could be postulated to be directly open to the containment atmosphere and to the outside atmosphere following a SSE.

→(DRN 03-1543, R13-A)

As a minimum, all original containment isolation valves were designed and factory tested so that the maximum allowable leakage did not exceed 1/10 of a standard cubic ft. of air per hour per in. of nominal valve diameter when subjected to containment maximum internal pressure. The offsite doses are discussed in Chapter 15. The Limiting Conditions of Operation and surveillance requirements are discussed in the Technical Specifications.

***NOTE**: The information concerning the original design and factory testing of the containment isolation valves is historical in nature and is not required to be updated. Demonstration that the containment isolation valves are capable of performing their design function is accomplished through Appendix J testing as described in Section 6.2.6.

←(DRN 03-1543, R13-A)

Closure times for isolation valves required to change position upon initation of CIAS are listed in Table 6.2-32. Valves take the failure position which provides the greatest level of safety. Refer to Section 7.3 for a discussion of CIAS circuitry and logic.

6.2.4.3.2 Single Failure Analysis

A redundant valve has been provided for each Class A or B containment penetration to maintain isolation in the event of any single active component malfunction. Wherever two automatic isolation valves are in series, each valve operator is actuated from a separate and redundant CIAS channel, A or B, with power supplied to each valve from a separate channel bus, A or B, as explained in Section 8.3. The Containment Purge valves (exhaust and makeup) are an exception to this arrangement. Since these valves are "fail closed", defense against a single failure event is provided by the CIAS and a redundant signal generated by the Containment Ambient Differential pressure switch. Table 6.2-32 lists the CIAS actuation channel for redundant isolation valves.

6.2.4.4 Testing and Inspections

All isolation valves are tested at the vendor's shop as per SP-6l, "Hydrostatic Testing of Steel Valves," 1961 edition, developed by the Manufacturers Standardization Society. The EFW control and shutoff valves are tested in accordance with ANSI B16.104 Class V.

Type C local leakage rate tests are performed on containment isolation valves as designated in Table 6.2-32. Refer to Subsection 6.2.6 for test details and acceptance criteria.

In-service inspection and testing is discussed in Section 6.6 and the Technical Specifications.

6.2.5 COMBUSTIBLE GAS CONTROL IN CONTAINMENT

The Combustible Gas Control System is designed in accordance with General Design Criterion 41 of 10CFR50, Appendix A, and Regulatory Guide 1.7 (3/10/71).

The Combustible Gas Control System (CGCS) consists of three systems: a Hydrogen Analyzer System, a Hydrogen Recombiner System and a Containment Atmosphere Release System (CARS).

6.2.5.1 Design Bases

→(DRN 05-1135, R14-B)

Following a beyond-design-basis LOCA a potential hazard may result from the gradual buildup of hydrogen in the containment. In an accident more severe than the design-basis loss-of-coolant accident (LOCA), combustible gas is predominately generated within the containment as a result of: 1. Fuel clad-coolant reaction between the fuel cladding and the reactor coolant, and 2. Molten core-concrete interaction in a severe core melt sequence with a failed reactor vessel. If a sufficient amount of hydrogen is generated, it may react with oxygen present in the containment at a rate rapid enough to lead to the breaching of the containment or a leakage rate in excess of technical specification limits. Additionally, damage to systems and components essential to continued control of the post-accident conditions could occur.

In SECY-00-0198, "Status Report on Study of Risk-Informed changes to the Technical Requirements of 10 CFR Part 50 (Option 3) And Recommendations on Riskinformed Changes to 10 CFR 50.44 (Combustible Gas Control)," the NRC staff recommended 0hanges to 10 CFR 50.44 that reflect the position that only combustible gas generated by a beyond-design-basis accident is a risk significant threat to containment integrity. The subsequent revision to 10 CFR 50.44 eliminates requirements that pertain to only design basis LOCAs. The CGCS is designed to prevent excessive hydrogen buildup in the containment and meets the following design bases:

a) The Hydrogen Analyzer System has the capability of sampling and measuring the containment hydrogen concentration at various points, can alert the operators in the main control room of the need to activate the hydrogen recombiners or the CARS, and can operate with both positive and negative containment pressure.

b) The entire Hydrogen Recombiner, CARS and Hydrogen Analyzer Systems are designed to safety class 2, class 1E, and seismic Category I standards, as appropriate. However, the Hydrogen Analyzer System contains items such as recorders and indicators whose operation may be affected during a seismic event. Hence, the hydrogen analyzer is qualified to function following, but not necessarily during a seismic event.

c) The CGCS is designed for the dynamic effects associated with postulated pipe breaks as described in Section 3.6.

d) The CGCS is designed to remain operable under post accident conditions inside the containment given in Section 3.11.

e) The CGCS is designed for forty years service plus one year post accident.

f) The CGCS is available for periodic testing and inspection as described in the Technical Specifications.

g) The hydrogen recombiners do not perform a design basis safety function since the revision of 10CFR50.44, where the design basis hydrogen release was eliminated and are also not required to manage beyond-design-basis accidents.

←(DRN 05-1135, R14-B)

6.2.5.2 System Design

→(EC-6494, R302)

NOTE: With the implementation of Waterford 3 Operating License Amendment No. 192, Elimination of Design Basis Hydrogen Release and Requirements for Hydrogen Recombiners, the information concerning hydrogen generation and the description and design evaluation of the Hydrogen Recombiner portions of the Combustible Gas Control System is historical in nature and is not required to be updated.

←(EC-6494, R302)

Flow diagrams for the Hydrogen Analyzer and the Hydrogen Recombiner Systems are shown in Figures 6.2-52 and 6.2-53. The CARS is shown in Figure 9.4-7.

6.2.5.2.1 Hydrogen Analyzer System

The Hydrogen Analyzer System consists of sample and return lines, isolation valves, and a hydrogen analyzer which includes sample coolers, moisture separators, and sample pumps. The analyzer panels, excluding the isolation valves and piping to and from the containment, are located at elevation -4 ft. MSL, in the Reactor Auxiliary Building. The controls required for start-up and operation, recorder for hydrogen concentration, and malfunction alarm annunciation are located in the main control room at elevation +46 ft. MSL. Other controls required for calibration, indicators required for failure diagnosis, and switches for modifying sample valve sequence are located on a control panel in the RAB east wing area at elevation +21 ft. MSL. This area is continually accessible during accident conditions.

The Hydrogen Analyzer System consists of two identical units which are completely independent of each other and are powered from an independent onsite source. Therefore, assuming a single failure, process capability is available to monitor the hydrogen concentration in the containment. Table 6.2-33 lists the design data for the Hydrogen Analyzer System.

For each unit there are seven 3/8 in. sample lines which take six separate samples of the containment atmosphere and one sample of the annulus atmosphere. The sample point locations are as follows:

a) Top of the containment located at the containment centerline.

b) Above the pressurizer.

c) Above the regenerative heat exchanger.

d) Below the missile shield.

e) One at the top of steam generator compartment A.

f) One at the top of steam generator compartment B.

g) Reactor Building annulus dome near centerline.

→(LBDCR 15-030, R308A)

These points provide broad coverage of the containment for hydrogen monitoring. Each set of six containment sample lines (a-f) has a common header inside the containment and penetrates the containment in a separate penetration assembly. Each of these lines has a solenoid operated isolation valve located inside the containment upstream of the header. These lines, as well as the sample lines running from the Reactor Building annulus dome (g) for each unit, have solenoid operated isolation valves located outside the containment. System actuation, including the isolation valves, can be initiated from the remote control panel to permit accurate hydrogen concentration recording following a LOCA.

←(LBDCR 15-030, R308A)

The remote panel location (main control room) is accessible during the postulated LOCA.

In the Hydrogen Analyzer System, the sample is dehumidified to avoid condensation in the detector, analyzed, recorded and returned to the containment when sampling is completed at one point. An automatic sequencer closes the sampling valve and opens the next sampling valve to begin measuring hydrogen concentration at the next sample location.

The hydrogen analyzer determines the percentage of hydrogen in the containment and enables rate of change of hydrogen concentration calculations to be made.

Containment atmosphere hydrogen analysis can also be conducted via the containment atmospheric grab sample cylinder. The latter is a manual operation and not remote manual like the hydrogen analyzer. The hydrogen analyzer is relied upon as the primary means of sampling following a LOCA.

The hydrogen analyzer has a range of 0-10 percent hydrogen with an accuracy of ± 2.5 percent of full scale and a minimum sensitivity of 0.2 percent hydrogen by volume. The hydrogen concentration is recorded during sampling and an alarm is actuated in the main control room if the concentration at any sample point exceeds three percent by volume.

Sample pumps draw the sample through the cooler and sample cylinder or analyzer and return it to the containment. The pump has a rated capacity of 0.5 cfm and is capable of pumping the sample back into the containment following an accident. Actual sample flow rate is restricted to insure proper moisture condensation.

The isolation valves for the Hydrogen Analyzer System are normally locked closed. The Containment Isolation Actuation Signal (CIAS) can be overridden for analyzing after a LOCA.

The Hydrogen Analyzer System piping, from the sample points within the containment, and piping returning the sample to the containment up to and including all containment isolation valves are designed and fabricated in accordance with ASME Section III, Class 2 (1974) and N-Stamped. In accordance with Appendix B of NUREG-0737 and the intent of Regulatory Guide 1.97, the Hydrogen Analyzer System instrumentation and controls are Class 1E and conform to IEEE-323-1974 and IEEE-344-1975.

6.2.5.2.2 Hydrogen Recombiner Subsystem

→(EC-6494, R302)

NOTE: With the implementation of Waterford 3 Operating License Amendment No. 192, Elimination of Design Basis Hydrogen Release and Requirements for Hydrogen Recombiners, the information concerning hydrogen generation and the description and design evaluation of the Hydrogen Recombiner portions of the Combustible Gas Control System is historical in nature and is not required to be updated.

←(EC-6494, R302)

→(DRN 05-1135, R14-B)

The hydrogen recombiners do not perform a design basis safety function since the revision of 10CFR50.44, where the design basis hydrogen release was eliminated and are also not required to manage beyond-design-basis accidents.

The Hydrogen Recombiner System consists of two stationary thermal (electric) recombiners. Two recombiner units are located inside the containment on the operating floor at elevation +46 ft. MSL. Each recombiner unit is provided with a power supply located outside the containment in an area which is accessible following a LOCA. Operation of each unit is manually initiated at one day post LOCA from a control panel located in the main control room.

Class 1E transfer switches (Isolation Panels) route the normal power supply to each Recombiner. However, to support Containment activities, the switches can be manually operated to isolate the Recombiners and transfer power to non-safety distribution panels. These panels are used to energize various equipment on +46 elevation.

←(DRN 05-1135, R14-B)

The stationary thermal recombiner consists of an inlet preheater section, a heater-recombination section, and a louvered exhaust chamber. It is designed and operated on the principle of natural convection, requiring no moving parts. Containment air is drawn into the thermal unit through louvers into the preheater section, which is a shroud placed around the central heaters and utilizes conduction to preheat the air. The preheated feed passes through an orifice plate to the electrical heater section where recombination occurs at approximately 1150°F to 1400°F. The heater section consists of five banks of electric resistance heaters stacked vertically. Each assembly contains individual heating elements. Operation of the unit is virtually unaffected should heating elements, having a total capacity of 20 kw, fail to function properly. After recombination the effluent water vapor is mixed and cooled with containment air drawn into the exhaust chamber through louvers. Condensation is discharged into the containment at 100°F above ambient.

The manufacturer's quality assurance program for the recombiners satisfies the requirements of Appendix B to 10CFR50. Seismic testing is in accordance with Regulatory Guide 1.29 (August, 1973) and IEEE-344-1971 for seismic Category I equipment. The control panels, including the power supplies are Class 1E, and are in accordance with IEEE-323-1971 and IEEE-279-1971.

The manufacturer's qualification test program includes proof-of-principle tests and full-scale prototype recombiner unit test. In the proof-of-principle test, gas mixtures with varying concentrations of hydrogen (0.15 to 4.4 percent by volume) and oxygen are passed through an electric resistance heater. Chemical analyses of gas samples are taken before and after the heater to verify that essentially 100 percent recombination occurs in all cases. Prototype tests have been performed to verify natural convection flow characteristics of the recombiner, recombiner power requirements as affected by the concentration of steam and hydrogen in the containment atmosphere, recombination effectiveness as affected by heater temperature, and effects of containment sprays with various chemical additives in a simulated post-LOCA environment with varying hydrogen combinations.

The unit is manufactured primarily of corrosion resistant, high temperature materials (Inconel-600 for inner structure and 300 series stainless steel for outer structure), except for the base which is carbon steel. The hydrogen recombiner uses conventional type electric resistance heaters sheathed with Inconel-800 which is a corrosion resistant material for this service. These heaters are designed to operate with sheath temperatures equal to those used in certain commercial heaters; however, these recombiner heaters operate at significantly lower power densities than commercial heaters of the same type.

Each recombiner unit operates on 480 volts, three phase power, and requires 75 kW-maximum.

The type of hydrogen recombiner provided for Waterford 3, i.e. the Westinghouse electric hydrogen recombiner, is the same as that provided for Calvert Cliffs 1 and 2 (Dockets 50-317/318) and for St. Lucie 1 (Docket 50-335). The advisory committee on Reactor Safeguards in their interim Calvert Cliffs report of May 18, 1973 has stated that the post-LOCA qualification testing was acceptable (Reference 2).

The development program conducted by Westinghouse to demonstrate the performance capabilities of the system and the results of the testing program can be found in WCAP 7709-L dated July, 1971 (Reference 3).

Design and performance data for the Hydrogen Recombiners is given in Table 6.2-34.

6.2.5.2.3 Containment Atmosphere Release System (CARS)

The CARS consists of two 100 percent capacity redundant exhaust fans, designated E-18 (3A-SA) and E-18 (3B-SB), and associated ductwork and two 100 percent capacity redundant supply fans, designated S3 (3A-SA) and S3 (3B-SB).

When post LOCA containment pressure has reduced sufficiently, CARS transfers combustible gases from inside containment to the Reactor Building annulus. The gases are filtered to remove radioactive particulates and iodines by the operating Shield Building Ventilation System (SBVS) prior to being released. The SBVS is described in Subsection 6.2.3.

Each exhaust train of the CARS includes steel exhaust ductwork and valves which are routed through the containment and Shield Building walls and up to the Reactor Auxiliary Building HVAC Equipment Room at elevation +46 ft. MSL. A centrifugal exhaust fan draws air from the containment and discharges into the recirculation duct of the Shield Building Ventilation System. The CARS supply ductwork extends from the controlled ventilation area into the containment and includes a check valve in the discharge piping to prevent backflow from the containment.

The CARS is manually operated from a remote switch by an operator inside the main control room after manually opening the locked closed valves in the RAB HVAC equipment room at elevation +46 ft. MSL. Design and performance data can be found in Table 6.2-35.

The system is accessible for periodic inspection and operability testing and follows the recommendations of Regulatory Guide 1.7. The design of the system is in compliance with standards and codes listed in Table 6.2.35.

The CARS supply and exhaust fans are mathematically analyzed to predict performance during a seismic event. The CARS exhaust fans are Type B spark resistant construction as defined by AMCA Standard 401-1966.

This standard requires a nonferrous fan wheel or impeller, a nonferrous shaft seal, bearing location outside the air or gas stream, and all fan parts to be electrically grounded. The CARS exhaust fans are designed to handle hot gases being purged from the containment but are located outside the containment and are not subjected to environmental conditions prevailing inside containment following a LOCA. The CARS fan/motor assemblies are safety class 2 and seismic Category I, the valves are safety class 2 and seismic Category I, and the ductwork is safety class 3 and seismic Category I.

Mixing inside containment following an accident is promoted by the containment Heat Removal System. (See Subsections 6.2.2 and 6.2.5.3)

6.2.5.3 Design Evaluation

→(EC-6494, R302)

NOTE: With the implementation of Waterford 3 Operating License Amendment No. 192, Elimination of Design Basis Hydrogen Release and Requirements for Hydrogen Recombiners, the information concerning hydrogen generation and the description and design evaluation of the Hydrogen Recombiner portions of the Combustible Gas Control System is historical in nature and is not required to be updated.

←(EC-6494, R302)

→(DRN 05-1135, R14-B)

The CGCS provides the capability to monitor and keep containment hydrogen concentrations within safe limits following a beyond-design-basis LOCA.

←(DRN 05-1135, R14-B)

Waterford 3 complies with that portion of Regulatory Guide 1.7, regulatory position C.1, which requires a plant to have the capability to measure hydrogen concentration and to prevent stratification of the atmosphere in the containment following a LOCA. Hydrogen concentration measurement is provided by the Hydrogen Analyzer System. A containment atmosphere mixing system is not necessary to prevent hydrogen stratification in the Waterford 3 containment.

→(DRN 05-1135, R14-B)

The CGCS meets the design, quality assurance, redundancy, energy source and instrumentation requirements for an engineered safety feature in accordance with position C.2 of Regulatory Guide 1.7.

Following a beyond-design-basis LOCA, hydrogen gas is generated inside the containment by the following sources:

←(DRN 05-1135, R14-B)

- Release of hydrogen dissolved in the reactor coolant.

- Metal-water reaction involving the zirconium fuel cladding and the reactor coolant.

- Radiolytic decomposition of water.

- Corrosion of metals and paints by solutions used for emergency core cooling and containment spray.

a) Dissolved Hydrogen

During normal plant operation, some hydrogen is dissolved in the primary coolant system water. The concentration can range from 10 to 50 cc (STP) / Kg (H_20) (see Subsection 9.3.4). The RCS contains up to 388,500 lbm of water at the maximum hydrogen concentration 1.9 lbm of hydrogen are dissolved in the coolant. This is assumed to be released instantaneously to the containment, but is only a minor source of hydrogen. Similarly, the concentration of dissolved oxygen in the primary coolant is less than 0.1 ppm. This represents 0.04 lbm of dissolved oxygen, a very small amount.

b) Metal-Water Reaction

As a result of a LOCA, elevated fuel cladding temperatures cause the zirconium to react with steam according to the following reaction:

$$Zr + 2\ H_20 \rightarrow ZrO_2 + 2\ H_2$$

Regulatory Guide 1.7 conservatively stipulates the amount of hydrogen that should be assumed to result from this metal-water reaction, for determining the performance requirements for the CGCS. That amount is five times the maximum amount calculated in accordance with 10CFR50.46, but no less than the amount that would result from reaction of all the metal in the outside surfaces of the cladding cylinders surrounding the fuel (excluding the cladding surrounding the plenum volume) to a depth of 0.00023 in. The maximum amount of zirconium calculated to react during ECCS operation is 0.8%. With the required factor of five, the amount assumed to react for purposes of hydrogen generation analysis is four percent. This four percent reaction is the governing requirement, rather than the 0.00023 in. reaction depth. The Waterford 3 core contains 64,092 lbm of zircaloy, and for the hydrogen generation analysis it is assumed that four percent of this reacts instantaneously.

c) Radiolytic Hydrogen Generation

→(DRN 05-1135, R14-B)

Water is decomposed into free hydrogen and oxygen by the absorption of energy emitted by fission products contained in the fuel and fission products intimately mixed with the water. The process is 100 ev as specified in Regulatory Guide 1.7, Rev. 0. The assumed distribution of fission products is the distribution specified in Regulatory Guide 1.7, Rev. 0.

←(DRN 05-1135, R14-B)

This is that one percent of the solids and 50 percent of the halogens present in the core are intimately mixed with the coolant water. The noble gases are assumed to be released to the atmosphere. For fission products intimately mixed with the water, the fraction of the radiation absorbed by the water. is 100 percent for both betas and gammas. For fission products remaining in the core, the fraction absorbed is 10 percent for gammas and zero for betas. These assumptions are summarized in Table 6.2-36. The beta and gamma releases were combined according to this model in calculating the amount of hydrogen generated by radiolysis (see Figure 6.2-54). The radiolytic hydrogen generation calculations were done using the NRC's BTP ASB 9-2 model.

To illustrate the order of magnitude of hydrogen evolution due to radiolysis of cable insulation, a conservative analysis was performed. There are approximately 55,500 pounds of cable insulation and jacket material in the containment, consisting of the elastomers hypalon, neoprene and EPR. Gas yields from organic materials reported in the literature are very conservative since tests are performed on thin sections in a vacuum where diffusion is very quick. From Reference 8, elastomers and rubbers have G values ranging from 0.15 to 0.8 (G value = molecules gas evolved/100 ev). Using a conservative G value of 1, and a total dose of $1.4 \times 1O^8$ rads (Appendix D NUREG-0589), results in a total hydrogen evolution of 2,800 ft^3.

This amount is approximately 0.1% of the containment volume, and insignificant compared to metal-water reaction and water radiolysis sources. The effect is further reduced when the time scale for gas evolution is considered, and the fact that more realistic estimates would probably reduce the volume almost an order of magnitude.

d) Corrosion of Metals

The use of aluminum and zinc and zinc-base paint has been restricted inside the Waterford 3 containment.

The exposed zinc is found in galvanized steel mainly in ductwork, cable trays and conduits. The sources of zinc used for the hydrogen concentration analysis are itemized in Table 6.2-37. Other surfaces, including grating, platforms, decking and the polar crane are painted rather than galvanized. A list of painted surfaces as given in Table 6.1-3. The zinc-based primers are Dimetcote 6, Dimetcote EZ and Carbozinc 11. The other paints are non-zinc-base. All surfaces painted with a zinc-base primer are coated with a non-zinc finish coat.

The sources of aluminum used for the hydrogen concentration analysis are also listed in Table 6.2-37.

The hydrogen generation due to corrosion of metallic zinc was obtained by superimposing the containment temperature transient for the design basis LOCA on the zinc corrosion rate data. The zinc corrosion rate was determined from the results of tests reported in Reference 1 for the concentration of boric acid which is used in the Waterford 3 spray solution. The zinc corrosion rate is shown as a function of time in Figure 6.2-55.

For this and all other calculations of hydrogen generated due to corrosion, the long term containment temperature history used is given in Figure 6.2-55c. It is an extension of the DBA transient for the containment atmosphere, from Figure 6.2-4. It is assumed that late in the accident the curve levels off and the temperature remains constant.

→(DRN 05-1135, R14-B)

The hydrogen generation due to corrosion of aluminum was similarly determined from the containment temperature transient and the aluminum corrosion rate data. The aluminum corrosion rate was taken from Reference 4. It is for a spray solution which contains sodium hydroxide and hence is much more corrosive than the Waterford 3 spray solution. Thus the corrosion rate used here is extremely conservative for Waterford 3. The aluminum corrosion rate as a function of time is shown in Figure 6.2-55a.

←(DRN 05-1135, R14-B)

The protective coating systems consisting of an inorganic zinc primer and an organic topcoat have all been successfully tested to postulated DBA conditions. These systems are extensively used by the nuclear industry and have repeatedly been submitted to DBA test conditions under a variety of pressure time correlations and spray solutions concentrations. These tests did not have any report indicative that corrosion of the zinc primer underneath the organic topcoat occurred (Ref 7).

If it is assumed that attack of the zinc primer by the spray solution occurs, it then would result in failure of the coating system by blistering and spalling the topcoat. However, the laboratories involved in testing protective coatings for DBA performance report that blistering and spalling of the topcoat does not occur.

One of the requirements, for testing, in ANSI N101.2 "Protective Coatings (Paints) for Light Water Nuclear Reactor Containment Facilities" is that steel panels be scribed (exposing the primer and substrate) and subjected to dynamic tests. Then those panels are examined and evaluated for coating defects, blistering etc. Test reports for the coating systems in consideration have been successfully evaluated for blistering.

Waterford 3 hydrogen generation analyses account for the corrosion of paint. The paint corrosion rate used is taken from Figure 9 of the Van Rooyen Report (Ref 5), which is believed to be derived from the paper by Zittel (Ref 6), and which is essentially the only publicly available data giving a finite paint hydrogen production rate, whatever the spray solution. This corrosion correlation is combined with the DBA temperature transient to obtain the paint hydrogen generation rate given on Figure 6.2-55b. The surface area painted with zinc-base paint is given in Table 6.1-3. It consists of 271,825 ft^2 of steel painted with Dimetcote plus a topcoat, 8,000 ft^2 of ZRC organic zinc primer for touch up of galvanized surfaces, and 91,900 ft^2 of containment vessel surface painted with Carbozinc plus a topcoat (Total Area = 375,725 ft^2). The mass of zinc-base paint did not affect the present calculation because it was considered that all of the painted surfaces produced hydrogen throughout the accident analysis.

→(DRN 05-1135, R14-B)

Two recombiners are provided inside the Waterford 3 containment. The flow rate for each recombiner is 100 scfm.

Figure 6.2-54 shows the hydrogen concentration as a function of time with the CGCS in operation. The concentration history is shown for operation of one or two recombiners with an assumed Aluminum surface area of 464 ft^2 (weight 1040 lb) and an assumed metallic zinc surface area of 429,300 ft^2 (weight 17,252 lb).

e) Hydrogen Mixing in the Containment

The potential for pocketing of hydrogen in the containment following a LOCA has been evaluated, and the results of the analyses show that pockets of hydrogen having concentrations within the flammable limits would not be produced unless the bulk concentration within the containment is also at the lower flammability limit.

←(DRN 05-1135, R14-B)

The bases for this conclusion rests on the following:

1) Hydrogen generated by metal-water reaction is released to the containment within a relatively short time after a LOCA and is uniformly dispersed throughout the containment atmosphere due to the turbulence accompanying the blowdown and the initial temperature stabilization. Under these conditions, the initial bulk concentration of hydrogen is approximately one percent.

2) The hydrogen is assumed to be generated at the water-steam-air interface at a rate of N_A lbm moles/hr at a uniform generation rate over an area A. Neglecting any convective mixing induced by free convection, the number of moles of hydrogen is given by:

$$\frac{N_A}{A} = h_m(\rho_s - \rho_b)$$

where:

ρ_s = hydrogen concentration at the interface

ρ_b = bulk hydrogen concentration in containment

h_m = mass transfer coefficient

The mass transfer coefficient, h, can be expressed in terms of the product of the Grashof (Gr) and Schmidt (Sc) numbers given by:

$$GrSc = \frac{g(\rho_b - \rho_s)}{\mu D} A^{3/2}$$

where:

g = acceleration due to gravity

μ = viscosity

D = diffusivity

For laminar flow:

$$h_m = 0.54 \frac{D}{A^{1/2}} (GrSc)^{1/3}$$

For turbulent flow:

$$h_m = 0.14 \frac{D}{A^{1/2}} (GrSc)^{1/3}$$

The mass transport equations, solved both for laminar and turbulent flow cases for varying bulk concentration of hydrogen, show that the interface concentration does not differ significantly from the bulk concentration unless the area over which the hydrogen is generated is very small, i.e., a small cubicle filled with essentially most of the water in the containment. Clearly this is not possible since water will cover a major portion of the lowest containment floor. Therefore mass diffusion is sufficient to mix the hydrogen throughout the containment.

3) In addition, the model used in the analysis is conservative in the following respects:

(a) If the area for hydrogen generation is much less than the cross sectional area of the containment there would be a chimney effect to enhance H_2 transport in addition to the eddies covered by the mass transport equation.

(b) Any H_2 reaching the water-air-steam interface in the form of bubbles will be propelled into the containment atmosphere with considerable velocity thereby enhancing the turbulence and resultant mixing. This effect would be more pronounced for small areas since, in order to have the same H_2 generation rate in lbm moles/hr, this area would necessarily require a large depth of water.

(c) Hydrogen generated from radiolysis and corrosion are generated over a period of days following the initial blowdown.

(d) Radiolytic hydrogen is generated in the core or in the water on the containment floor. Radiolytic hydrogen generated in the steam is negligible (approximately 10^{-7} lbm moles/hr) compared with values ranging from 0.2 to 1.7 lbm moles/hr generated in the water during the initial 10 days.

(e) Action of the containment sprays and thermal gradients within the containment will promote mixing.

Containment spray action will provide substantial mixing of the upper and lower containment volume due to mass exchange and momentum considerations. A mass flow rate per ft^2 at the operating deck of one lbm/min is provided by the Containment Spray System. Floor grating is provided both above and below the operating deck to ensure that the spray effectively mixes within the subcompartment regions.

The Containment Spray System will effectively remove heat from the containment atmosphere in the upper containment volume region in the event of a LOCA.
Analyses show that thermal equilibrium is reached within the first 10 to 15 ft. of spread droplet fall and hence a layer of cooler air is formed high in the upper containment.

By virtue of its greater density, this cooler air will continuously be replaced by lighter warmer air from below. Thus the atmosphere in the upper containment mixes and reaches its equilibrium temperature and density.

Drawings showing the airflow and spray patterns inside the Waterford 3 containment are given in Subsection 6.2.2.

(f) The major portion of the fan cooler ductwork is assumed to be conservatively unavailable following a LOCA. Only the safety grade portion of the ductwork, which extends from the fan cooler outlets to the operating floor is assumed available. An analysis showing that the containment is still adequately mixed is presented in Appendix 6.2C.

In addition, an analysis has demonstrated the ability of the entire ductwork system to survive the most severe LOCA pressure transient in the ductwork, a 592 sq. in. pump suction leg circumferential break which yields the highest steam generator subcompartment pressure.

Results of these analyses show pressure in the transient state as a function of time immediately after the accident. Figures 6.2-56 and 57 indicate pressure differential between the fan coolers and surroundings. Figure 6.2-58 indicates the pressure differential between the safety grade ductwork and containment volume.

In consideration of the above results and the maximum pressure pulse caused by a LOCA or a MSLB, the safety related portion of the CCS ductwork is reinforced to withstand without loss of function, the external differential pressures and internal pressure surges that may be imposed on them following a hypothetical accident.

The failure mode and effects analysis for the subsystems of the CGCS are provided as follows: Hydrogen Analyzer System, Table 6.2-38; Hydrogen Recombiners, Table 6.2-39; CARS, Table 6.2-40.

6.2.5.4 Tests and Inspections

→(EC-6494, R302)

NOTE: With the implementation of Waterford 3 Operating License Amendment No. 192, Elimination of Design Basis Hydrogen Release and Requirements for Hydrogen Recombiners, the information concerning hydrogen generation and the description and design evaluation of the Hydrogen Recombiner portions of the Combustible Gas Control System is historical in nature and is not required to be updated.

←(EC-6494, R302)

Preoperational testing of the Combustible Gas Control System is conducted during the final stages of plant construction prior to initial startup. These tests assure correct functioning of all components such as controls, instrumentation, alarm setpoints, fans, pumps, recombiners, piping and valves. System reference characteristics, such as pressure differentials and flow rates, are documented during the Preoperational tests and are used as base points for measurements in subsequent operational tests.

→(DRN 05-1135, R14-B)
←(DRN 05-1135, R14-B)

Purchase specifications require that vendors submit type test, calculation or operating data which verifies the ability of the equipment to function following a safe shutdown earthquake.

Switches, solenoids, and relays of the Hydrogen Analyzer System are shop energized to check operation. Electrical interlocks are shop checked under simulated operating and emergency conditions. Prior to installation, the vendor performs a system operation test, using air to demonstrate the operability of the analyzer. The analyzer is operationally tested using zero and span gas samples introduced at the local control panel.

The hydrogen recombiners will be tested using Preoperational testing procedures. Three tests are performed to achieve the purpose of the Preoperational procedure: (1) a Preoperational test; (2) a recombiner heat up test, and (3) an air flow test. The Preoperational test is performed in order to determine the temperature-to-power relationship of the recombiner heaters and verify their operability. The heat up test is performed to calibrate the recombiner power setting with temperature for proper operation under post LOCA conditions. Intake air flow rate characteristics and proper recombination flow rate are verified in the air flow test.

The results of qualification tests performed on system components to demonstrate functional capability and operability in the containment accident environment have been presented in WCAP 7709-L, dated July, 1971. An extensive test program described in the above document demonstrated that the proper recombiner operation is assured by measuring the amount of power to the recombiner. The NRC has advised Westinghouse Electric Corporation that this electric recombiner design is acceptable (letter to Westinghouse Electric Corporation dated June 22, 1978). The recombiner components satisfy the following type tests prior to acceptance.

a) Recombiner Enclosures

Each hydrogen recombiner unit is tested for air flow rates and temperature at rated power to demonstrate that hydrogen recombination temperature is reached. Hydrogen recombination is demonstrated during post-LOCA conditions with a production unit or a prototype unit.

During the normal service life of the recombiner periodic heatup and cooldown tests are performed to demonstrate its availability. These tests will be performed as described in the technical specifications. This heatup and cooldown cycling produces thermal stresses in the heater frame and support structure. To prove the recombiner can sustain this repeated cycling, a prototype unit has been subjected to 80 heatup and cooldown cycles. During refueling, the recombiner is visually inspected for fatigue failure. Results must show the recombiner undamaged by this cyclic operation.

b) Seismic Tests

A prototype recombiner was vibration tested as a complete assembly. Tests were performed using the methods described in Section 3.10.

(IEEE-344-1971). The acceleration test level was determined by the containment floor response spectrum. To determine the enclosure reserve strength and to verify the stress analysis, the unit was subjected to higher level tests. The results of the test proved that the recombiner is rated for the value dictated by the floor response spectrum.

c) Electric Heaters

The following heater tests were performed on a prototype recombiner unit or on the production unit:

1) Normal service life was demonstrated with the recombiner enclosure test described above. This test was successfully completed with 80 cycles using prototype heaters.

2) Seismic tests were performed with the recombiner enclosure as described above.

3) Post LOCA operation in the containment environment is proved through the ability of the heaters to withstand containment spray plus steam during the prototype testing which was conducted in the simulated reactor containment at atmospheric pressure.

d) Control Panel and Power Supply

The following tests were performed:

1) Prototype or production control and power supply panels are operationally tested. The power supply panel is used to power a prototype recombiner in the 80-cycle test, thereby qualifying it for service. A power supply panel and control panel are full load tested at the Supplier's plant to make sure the temperature in the power supply panel are within allowable limits and the equipment operated satisfactorily.

2) Seismic tests, using the same method as used on the recombiner enclosure and heaters are applied to the power supply and control panels. The equipment is qualified by test for the acceleration (g) value taken from the Reactor Auxiliary Building floor response spectrum.

3) Transformer tests are conducted during the power supply tests since this component is incorporated in the power supply.

One CARS supply fan and one CARS exhaust fan are each given a certified factory performance test from which the vendor will prepare fan static pressure vs. volume flow rate and fan horsepower vs. volume flow rate curves. The duplicate supply and exhaust fans are not given this performance test as they are similar in all respects.

6.2.5.5 Instrumentation Requirements

→(EC-6494, R302)

NOTE: With the implementation of Waterford 3 Operating License Amendment No. 192, Elimination of Design Basis Hydrogen Release and Requirements for Hydrogen Recombiners, the information concerning hydrogen generation and the description and design evaluation of the Hydrogen Recombiner portions of the Combustible Gas Control System is historical in nature and is not required to be updated.

←(EC-6494, R302)

The Combustible Gas Control System can be activated remotely from the main control room.

The Hydrogen Analyzer System is a fully redundant subsystem. Each set of sample lines is routed through the containment to ensure that a single event cannot cause loss of both sampling systems. Furthermore, should instrumentation fail, means are provided for manually taking a sample from the containment atmospheric grab sample cylinder for laboratory analysis.

The hydrogen recombiners do not require any instrumentation inside the containment for proper operation after a LOCA. Thermocouples are provided for convenience in test and periodic checkout of the recombiner but are not necessary to assure proper operation of the recombiners. Proper recombiner operation after an accident is assured by measuring the amount of electric power to the recombiner from the control panel outside the containment. The proper amount of air flow is fixed by an orifice plate in the recombiner.

The percent hydrogen indicating recorders are located in the main control room to allow the operators to monitor the containment hydrogen concentration.

For the Hydrogen Analyzer System the following operating conditions are monitored at their respective remote control panels. The following indicators are provided:

a) Digital Indication of Percent Hydrogen
b) Digital Indication of Sample Inlet and Outlet Pressure
c) Hydrogen Concentration Safe Indication
d) Purge Indication
e) Sample Location Being Analyzed
f) Isolation Valves Open/Close Indication
g) Low Sample Pressure Alarm
h) High Sample Pressure Alarm
i) Low Sample Flow - Alarm
j) High Dryer Temperature - Alarm
k) Hydrogen Concentration Caution/High - Alarms
l) Hydrogen Analyzer Calibration/Test Switches
m) Sample Valve Sequencer Control Switches

The following indicators and controls are provided in the main control room:

a) Recording of Percent Hydrogen
b) Isolation Valves Open/Close Indication
c) Hydrogen Analyzer Off - Standby - On Control Switch with Indicating Lights
d) Hydrogen Concentration High - Alarm
e) Power Failure Alarm
f) System Trouble Alarm
g) Isolation Valves Control Switch

The low sample flow alarm sounds when the sample flow drops below that which is required for proper operation of the hydrogen analyzer. The high hydrogen concentration alarm sounds when the hydrogen concentration inside the containment exceeds three percent.

The three percent hydrogen concentration alarm setpoint is chosen to allow sufficient time (see Figure 6.2-54) for the operators to implement the hydrogen recombiners and ensure that the hydrogen concentrations do not exceed the four percent flammable limit.

The Hydrogen Recombiner System also provides indication of:

a) Power Status - Control Room indication

b) Power Usage (wattmeter) - Control Room indication

c) Temperature Inside the Recombiner.

The CARS has manual on-off switches with status indicating lights for the fans in the main control room. Automatic valves have position switches and position indication in the main control room.

SECTION 6.2.5: REFERENCES

1) Hutchinson Island Plant, Unit No. 1, PSAR, Docket No. 50-335, October 1969, (Amendment 4).

2) Calvert Cliffs, Unit No 1&2, PSAR Docket No. 50-317/318.

3) WCAP 7709-L dated July 1971.

4) "Design Considerations of Reactor Containment Spray Systems" Part III, "The Corrosion of Materials by Spray Solutions" ORNL-TM-2412, By J.C. Greiss and Al Bacarella, Dec 1969.

5) BNL-NUREG-24532, "Hydrogen Release Rates from Corrosion of Zinc and Aluminum" by Daniel Van Rooyen of Brookhaven National Laboratory, May 1978.

6) "Post-Accident Hydrogen Generation from Protective Coatings in Power Reactors", by H. E. Zittel, of Oak Ridge National Laboratory, Nuclear Technology, Vol. 17, Feb 1973.)

7) Test reports for simulated DBA on identical protective coating systems are listed as follows:

By Carboline Company By ORNL

4 - 1066 A	Report of DBA Testing of Coating Systems from Carboline Company, dated Dec 5, 1975
4 - 993	
4 - 897	
4 - 1180	
4 - 900	
4 - 993	

Testing Project 01377 (100 days)

Testing Project 01399B
Testing Project 01445
Testing Project 01388A
Testing Project 01377

8) Bolt, R and J Carroll, "Radiation Effects on Organic Materials", Table 6.3, Academic Press, New York, 1963.

→(DRN 05-1135, R14-B)

9) Waterford 3 Operating License Amendment No. 192, Elimination of Requirements for Hydrogen Recombiners and Hydrogen Monitors, Issued March 9, 2004.

←(DRN 05-1135, R14-B)

6.2.6 CONTAINMENT LEAKAGE TESTING

6.2.6.1 Containment Integrated Leakage Rate Test

On completion of fabrication and post-weld heat treatment of the containment vessel, and prior to the installation penetration internals, pneumatic tests were performed in accordance with the application requirements of the ASME Section III Code to demonstrate the integrity and leak tightness of the completed pressure vessel.

The structural integrity tests are described in Subsection 3.8.2.7. These tests include soap bubble tests, over pressure test, construction integrated leak rate test, and leak testing of personnel locks including operational testing of latching mechanisms and interlocks.

Upon completion of construction of the containment vessel including installation of all portions of mechanical, fluid, electrical, and instrumentation systems penetrating the primary containment pressure boundary, and prior to any reactor operating period, a Preoperational leakage rate test of the containment vessel and leak tests of the testable penetrations, double gasketed seals, and containment isolation valves (Types A, B and C tests), are conducted to verify their leak tight integrity.

Initial Type A tests were conducted in accordance with the provisions of the ANSI N45.4-1972, "Leakage Rate Testing of Containment Structures for Nuclear Reactors," March 16, 1972 and 10CFR50 Appendix J (Option A). Subsequent Type A, B and C leakage rate tests are performed on a periodic basis in accordance with Option B of 10CFR50 Appendix J to assure that the containment leak tight integrity is maintained. Subsection 6.2.6.4 describes the schedule and reporting requirements.

A steel shell pressure containment vessel, designed, fabricated, inspected and pressure tested in accordance with the ASME Boiler and Pressure Vessel Code Section III Class MC, 1971 edition, Summer 1971 Addenda and protected by the concrete Shield Building offers continued structural integrity over the life of the unit. The containment vessel received a code stamp and represents the most recent developments in the techniques of pressure vessel design and fabrication that are backed up by years of research, testing and successful in-service experience. Therefore, there will be no need for any special in-service surveillance program other than visual inspection of the exposed interior and exterior surfaces of the containment vessel and periodic leakage rate testing.

→(DRN 05-75, R14)

If there is evidence of structural deterioration, the containment integrated leak test (Type A) is not performed until corrective action is taken in accordance with repair procedures, nondestructive examinations, and tests as specified in the applicable codes. Such structural deterioration and corrective actions taken is reported as part of the test summary report described in Subsection 6.2.6.4.

←(DRN 05-75, R14)

During the period between the initiation of the containment visual inspection and the performance of the Type A test, no repairs or adjustments are made so that the containment can be tested in as close to the "as is" condition as practical. During the period between the completion of one Type A test and the initiation of the containment inspection for the subsequent Type A test, any necessary repairs or adjustments are made to components whose leakage exceeds that specified in the Technical Specification.

If, during the performance of a Type A test (ILRT) including the supplemental test (CLRT), excessive leakage occurs through locally testable penetrations or isolation valves to the extent that it would interfere with the satisfactory completion of the test, these leakage paths may be isolated and the Type A test continued until completion. A local leakage test shall be performed before and after the repair of each isolated leakage path. The sum of the post repaired local leakage rates and the UCL shall be less than 75 percent of the maximum allowable leakage rate, La. Local leakage rates shall not be subtracted from the Type A test results to determine the acceptability of a test. The test results shall be reported with both the pre and post repair local leakage rates, as if two Type A tests had been conducted. Local leak tests are accomplished as described in Subsections 6.2.6.2 and 6.2.6.3 for Type "B" and "C" tests.

→(DRN 05-75, R14)
Closure of containment isolation valves for the Type A test is accomplished by normal operation and without any preliminary exercising or adjustments (e.g., no tightening of valve after closure by valve motor). Repairs of maloperating or leaking valves are made as necessary. Information on any valve closure malfunction or valve leakage that requires corrective action before the test is included in the test summary report described in Subsection 6.2.6.4.
←(DRN 05-75, R14)

Those portions of the fluid systems that are part of the reactor coolant pressure boundary and are open directly to the containment atmosphere under post-accident conditions and become an extension of the boundary of the containment are opened or vented to the containment atmosphere prior to and during the test. Portions of closed systems inside containment that penetrate containment and rupture as a result of a loss of coolant accident are vented to the containment atmosphere. All vented systems are drained of water or other fluids to the extent necessary to assure exposure of the system containment isolation valves to containment air test pressure and to assure they are subjected to the post-accident differential pressures. Systems that are required for safe shutdown are operable in their normal mode during the test and need not be vented. Refer to Table 6.2-32 for a tabulated list of containment process piping penetrations and isolation valve information. Also, Table 6.2-32 identifies portions of the system that will be vented and others that will remain fluid filled with justification stated. As allowed under Option B of 10CFR50 Appendix J, pathways Type B or C tested within 24 months of the Type A test need not be vented or drained for planning, scheduling or ALARA considerations. The Type A test leakage rate UCL is adjusted accordingly to determine the overall leakage rate.

The method used to determine the containment leakage rate is the absolute method. The absolute method of leak rate determination depends on the measurement of the temperature and pressure of the containment atmosphere, with suitable correction for changes in humidity. It assumes that the temperature variations during the test are insufficient to effect significant changes in the containment net free volume. The containment is pressurized to the maximum calculated accident (LOCA) pressure for the Preoperational test.

Sensing devices are located at different parts of the containment to measure average temperature and humidity. Fans may be used to minimize temperature variations in regions of the containment. Location of temperature and humidity sensors is made with consideration of the temperature or humidity pattern. These patterns are employed in determination of the mean representative temperature and humidity for the absolute method of leakage rate testing.

The relative humidity, temperature and absolute pressure inside the containment are monitored during the course of the leakage rate test so vapor pressure corrections can be made. To minimize the effect of variation in the partial pressure of water vapor, it is desirable to maintain the containment structure air at a reasonably constant temperature. Vapor pressure due to moisture content in the containment atmosphere is determined by acceptable methods of vapor pressure determination.

The containment environmental conditions are stabilized for a period of not less than four hours prior to the start of a leakage rate test.

The leakage preoperational rate test period extends to 24 hours of retained internal pressure. Subsequent Type A test durations, performed in accordance with Option B of 10CFR50 Appendix J, are as specified in ANSI/ANS - 56.8 - 1994. In an SER dated December 1, 1987 for Waterford 3, the NRC approved use of testing methodology described in Bechtel Corporation topical report the BN-TOP-1, Revision 1, "Testing Criteria for Integrated Leak Rate Testing of Primary Containment Structures for Nuclear Power Plants," dated November 1, 1972, which supports a reduced duration containment leakage rate test period. The Waterford 3 Option B Containment Leakage Program allows the use of BN-TOP-1 for performance of a Type A test.

Leakage rate tests shall not be started until essential temperature equilibrium has been attained.

Pressure, temperature, and humidity observations are made within the containment and recorded during the course of the leakage rate test. Pressure and temperature measurements of the outside atmosphere shall also be made and recorded. The times of observations are noted in hours and minutes. A dated log of events and pertinent observations is also maintained during the test. The correctness of data is attested to by those responsible for the test and, where specified, by a qualified witness.

Instrumentation is provided to monitor the containment leak rate testing. This instrumentation is designed, calibrated, and tested to provide high accuracy to ensure that the containment atmospheric parameters can be precisely measured.

An error analysis of these instruments is made to help establish the validity of preoperational Type A tests.

The test instrumentation is specified in the test procedure. This includes instrumentation to measure containment pressure, temperature and humidity as well as local leakage.

The accuracy of preoperational and subsequent Type A tests are verified by a supplemental test which superimposes a known leakage. The supplemental test method selected is conducted for sufficient duration to establish accurately the change in leakage rate between the Type A and supplemental test. Results from this supplemental test are acceptable provided the difference between the supplemental test and the test Type A is within 25 percent of the allowable leak rate. If the results do not meet the acceptance criteria, the reason shall be determined, corrective action taken and a successful supplemental test performed.

A practical and simple arrangement for superimposing a controlled and measurable leakage on the containment vessel employs the orifice leak of a microadjustable instrument flow valve installed at a convenient penetration of the containment vessel. The flow through the valve is measured by means of a suitable flowmeter or rotameter.

The test procedure involves placing the calibrated leak system into operation after the leakage rate test is completed. The flowmeter readings are then recorded. Concurrently, readings of the containment vessel leakage measuring system, which now records the composite leakage of both the containment vessel leaks and the superimposed orifice leak, are resumed.

The peak pressure test measured leakage rate shall be less than 0.75 of the allowable leak rate. The containment vessel and its penetrations are designed to limit leakage to no more than 0.5 weight percent of the internal net free volume of approximately 2,677,000 ft^3 in 24 hours at a pressure 44 psig.

If the Type A test fails to meet the applicable acceptance criteria, the reporting requirements of Option B to 10CFR50 Appendix J are applicable.

→(LBDCR 15-029, R309)

If a Type A tests fail to meet the applicable acceptance criteria, a Type A test shall be performed within the default interval of 48 months. A successful retest will re-establish the fifteen-year interval of Option B to 10CFR50 Appendix J for Type A tests.

←(LBDCR 15-029, R309)

6.2.6.2 Containment Penetration Leakage Rate Test

Type B penetration leakage testing is required on electrical penetrations, hatches, airlocks and process piping penetrations that have expandable bellows to accommodate thermal growth. Other process piping penetrations have the process pipe welded directly to the steel containment vessel and do not require Type B penetration leakage testing.

Table 6.2-41 provides a listing of electrical penetration assemblies. Table 6.2-42 provides a listing of hatch and airlocks. The construction hatch is a seal welded assembly which is leak tested during Type A leak testing. No type B leak testing is required on this penetration. The plan and elevation drawings of the Personnel Lock are shown on Figure 6.2-69, 6.2-70 and 6.2-71. Pressure test taps are provided in the door frames for the interior and exterior door to allow the double edge gasketed seals to be leak tested as illustrated in detail "A" of Figure 6.2-69. The interior door is provided with mounting slots (Detail 1 Figure 6.2-70 and Detail "B-B" Figure 6.2-71) to allow the installation of required tie downs for preventing the interior door from being unseated during local leak testing of the Personnel Lock at P_a. Two one inch pipe test connections are provided on the exterior end bulkhead for pressurizing, depressurizing and monitoring the Personnel Lock air space during the performance of this test.

Table 6.2-32 provides a listing of process piping penetration. Of this listing, only the following penetrations require Type B leak testing due to the presence of expandable bellows:

Penetrations 1 through 4	Main Steam and Feedwater Systems
Penetrations 32 and 33	Containment Sump Suction
Penetration 43	Reactor Drain Tank Outlet
Penetration 25	Fuel Transfer Tube and Flange

Type B leakage test should be completed prior to the preoperational Type A test. In the event that Type B test is not completed prior to the preoperational Type A test, any Type B penetration path test leakage not accounted for in Type A test shall be added to the measured leakage determined by the Type A test.

The Type B test is performed utilizing a local leak rate monitor or equivalent and pressurizing each testable penetration to its required pressure of not less than the peak accident pressure of 44 psig. This pressure is maintained until the leakage rate of the penetration is determined. The current test schedule for Type B penetrations is in accordance with Option B of 10CFR50 Appendix J.

The combined leakage rate of penetrations and valves subject to Type B and C tests shall be less than 0.60 of the design leakage rate of the containment.

6.2.6.3 Containment Isolation Valve Leakage Rate Test

Table 6.2-32 also provides a listing of all containment isolation valves and those that are exempt from leak rate test Type C. Table 6.2-43 provides justification for the penetrations that are exempt from the Type C leakage testing.

Type C leakage test should be completed prior to the Preoperational Type A test. In the event that Type C test is not completed prior to the preoperational Type A test, any Type C penetration path test leakage not accounted for in the Type A test shall be added to the measured leakage determined by the Type A test.

Table 6.2-32 and Figures 6.2-65 through 6.2-68 show that the arrangements to be provided for the testing containment isolation valves subject to Type C testing along with the direction of pressurization to be used for the test. Type C tests are performed by local pressurization. Administrative controls over test, vent and drain (TVD) connections shall be implemented as an integral part of the operating test and administrative procedures will ensure proper valve position following the test. During plant operation, the TVD valve and a cap or blind flange will provide a double isolation barrier. Table 6.2-44 lists all the containment isolation valves for which the applied pressure is not in the same direction as the pressure existing when the valve is required to perform its safety function. The design and construction of these valves is such that pressurization in the opposite direction will result in equivalent or more conservative leakage. This is in accordance with ASME Section XI, Subsection IWV.

Each valve to be tested shall be closed by normal operation and without any preliminary exercising or adjustments (e.g., no tightening of valve after closure by valve motor), and pressurized to not less than the peak accident pressure of 44 psig. The current test schedule for Type C penetrations is in accordance with Option B of 10CFR50 Appendix J.

The combined leakage rate of penetrations and valve subject to Type B and C tests shall be less than 0.60 of the design leakage rate of the containment in accordance with Option B of 10CFR50 Appendix J.

Penetrations which are identified as sources of bypass leakage have an additional acceptance criteria given in Technical Specifications 6.15. Bypass leakage paths are identified in plant procedures and encompass isolation valve leakage paths that are not contained by filtered exhaust systems.

6.2.6.4 Scheduling and Reporting of Periodic Tests

→(DRN 02-385, R13, LBDCR 15-029, R309)

After the preoperational leakage rate tests, Type A tests shall be performed at fifteen-year intervals. Technical Specification Amendment 244 was approved by the NRC on August 24, 2015. This amendment postpones the next Type A test performed after May 21, 2005 to no later than May 20, 2020. This results in intervals to be extended from no longer than 10 years to no longer than 15 years on a permanent basis, provided acceptable performance history and the requirements of NEI 94-01, Rev. 2-A are met.

←(DRN 02-385, R13, LBDCR 15-029, R309)

The performance of Type A tests shall be limited to periods when Waterford 3 is nonoperational and secured in the shutdown condition under the administrative controls and in accordance with the safety procedures defined in the operating license.

Type B tests except tests for air locks, shall be performed during reactor shutdown for refueling, or other convenient intervals, but in no case at intervals greater than ten years. Air locks shall be tested at 30 month intervals. However, air locks which are opened during such intervals, shall have the door seals tested within 7 days of each opening.

Type C tests shall be performed during reactor shutdown for refueling but in no case at intervals greater than five years in accordance with the Waterford 3 Option B 10CFR50 Appendix J Containment Leak Testing Program.

The preoperational and periodic tests shall be documented in a readily available summary report that will be made available for inspection, upon request, at Waterford 3.

The summary report shall include a schematic arrangement of the leakage measuring system, the instrumentation used, the supplemental test method, and the test program selected as applicable to the preoperational test, and all subsequent periodic tests. The report contains an analysis and interpretation of the leakage rate test data for the Type A test results to the extent necessary to demonstrate the acceptability of the containment leakage rate in meeting the acceptance criteria.

For each periodic test, leakage test results from Type A, B, and C tests shall be included in the summary report. The report shall contain an analysis and interpretation of the Type A test results and a summary analysis of periodic Type B and Type C tests that were performed since the last Type A test.

Leakage test results from Type A, B and C tests that failed to meet the acceptance criteria will be included in a separate accompanying summary report that includes an analysis and interpretation of the test data, the least squares fit analysis of the test data, the instrumentation error analysis, and the structural conditions of the containment or components, if any, which contributed to the failure in meeting the acceptance criteria. Results and analyses of the supplemental verification test employed to demonstrate the validity of the leakage rate test measurements will also be included.

6.2.6.5 Special Testing Requirements

Any major modification, replacement of a component which is part of the primary reactor containment boundary, or resealing a seal-welded door, performed after the preoperational leakage rate test shall be followed by either a Type A, Type B, or Type C test, as applicable, for the area affected by the modification. The measured leakage from this test shall be included in the summary report. Minor modifications replacements, or resealing of seal-welded doors, performed directly prior to the conduct of a scheduled Type A test do not require a separate test.

The Shield Building that encloses the entire primary containment vessel or portions thereof shall be subject to individual tests in accordance with Technical Specifications and discussed in Subsection 6.2.1.6.