CHAPTER 6 - ENGINEERED SAFETY FEATURES

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6.1 ENGINEERED SAFETY FEATURE MATERIALS

The specific materials used in engineered safety feature components are identified in those sections where each component is discussed. Refer to the appropriate section which describes a specific Engineered Safety Feature System for information on materials, as follows:

System	Section
Primary Containment System	6.2
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6.2 CONTAINMENT SYSTEMS

6.2.1 Containment Functional Design

The function of the Primary Containment System is to accommodate, with a minimum of leakage, the pressures and temperatures resulting from the break of any enclosed process pipe; and, thereby, to limit the release of radioactive fission products to values which will insure offsite dose rates well below 10CFR100 guideline limits.

6.2.1.1 Containment Structure

6.2.1.1.1 Design Bases

The analyses/discussions described in this Section are based on a power level of 1860 MW (thermal). Amendment 65 reported to the NRC that the analyses are still valid for the current power level of 1930 MW (thermal). In addition, loads and design bases have changed as a result of the Mark I Torus Program (see Section 3.8). As a result of this program, major changes to the containment have been incorporated such as (1) the addition of Y quenchers on the electromagnetic relief valve (EMRV) discharge lines, (2) EMRV vacuum breaker replacement, (3) downcomer bracing, (4) downcomer truncation, (5) installation of mid-bay saddles, and (6) strengthening of the torus.

The basis for the design pressure, and the resultant dynamic response of the Primary Containment, is the loss of coolant following the sudden and complete severance of the largest non isolable line connected to the reactor vessel, while the reactor is operating at its steady state ultimate power level. The design criteria for the Containment are as follows:

a. To withstand the peak transient pressure (coincident with an earthquake) which could occur due to the postulated break of any pipe inside the drywell.

b. To channel the flows from postulated pipe breaks to the torus.

c. To withstand the force caused by the impingement of the fluid from a break in the largest local pipe or connection, without containment failure.

d. To limit primary containment leakage rate during and following a postulated break in the primary system to substantially less than that which would result in offsite doses approaching the limiting values in 10CFR100.

e. To include provisions for leak rate tests.

f. To be capable of being flooded following a Design Basis Accident to a height which permits unloading of the core.

The Primary Containment consists of a pressure suppression system with two large chambers. The drywell houses the reactor vessel, the reactor coolant recirculating loops, and other components associated with the reactor system. It is a 70 ft diameter spherical steel shell with a 33 ft diameter by 23 ft high cylindrical steel shell extending from the top.

The torus is a steel shell, located below and around the base of the drywell. It has a major diameter of 101 ft, a chamber diameter of 30 ft, and is filled to approximately a 12 ft depth with water.

The two chambers are interconnected through ten vent pipes, 6'-6" in diameter, equally spaced around the circumference of the torus, which feed into a common header inside the torus. This header takes the shape of a torus with a 101 ft major diameter by 4'-7" minor diameter. There are 120 downcomer pipes approximately 2 ft in diameter and uniformly spaced which have their open ends extending 3 ft below the minimum water level in the torus. Gas phase return lines with vacuum breaker valves feed back gas to the drywell in case its pressure is less than the torus air space. The drywell is designed for a pressure of 44 psig and the torus for a pressure of 35 psig, with a design integrated leak rate for the system no greater than 0.5% of its total volume per day at 35 psig. The drywell was originally designed for a pressure of 63 psig but was reduced to 44 psig by Reference 15. A 15% margin was applied to revise the drywell design pressure to 44 psig. The design leak rate is 0.5% per day at a pressure of 35 psig. The pressure response of the drywell and adsorption chamber following an accident would be the same after about 60 seconds. Based on the calculated primary containment pressure response and the adsorption chamber design pressure, primary containment pre-operational test pressures were chosen. Based on primary containment pressure response and the fact that drywell and adsorption chamber function as a unit, the primary containment will be tested as a unit, rather than testing the individual components separately.

In the event of a loss-of-coolant accident, the peak drywell pressure would be 38 psig, which would rapidly reduce to 20 psig with 100 seconds following the pipe break. The total time the pressure would be above 35 psig is calculated to be about 7 seconds. Following the pipe break, the adsorption chamber pressure rises to 20 psig within 8 seconds, equalizes with the drywell pressure at 25 psig with 60 seconds, and thereafter rapidly decays with the drywell pressure decay. The design basis loss-of-coolant accident was evaluated at the primary containment maximum allowable accident leak rate of 1.0% per day at 35 psig, which provides an adequate margin of safety to assure the health and safety of the general public. Leakage rate tests are subsequently conducted at frequencies based upon the Commission's guide for containment leak rate testing. The current technical specifications define the actual frequency and acceptance criteria.

In order to establish a set of criteria to be used in designing the Containment System, the following basic assumptions were made.

The reactor is operating at a thermal power level of 1860 MW, 1020 psig pressure, and has a fission product inventory equivalent to having operated at this level continuously for 1000 days.

The Primary Containment is normally kept near atmospheric pressure. As a basis for calculation, an upper limit of 2 psig internal pressure was assumed.

The full range of Loss-of-Coolant Accidents has been analyzed; from a small break, where the makeup flow rate capability is greater than coolant loss rate, to the largest break, a highly improbable break equal in area to twice the flow area of a recirculation line. Analysis has shown that the largest break results in the peak pressure in the drywell and torus.

Therefore, the following accident is assumed: A guillotine break of one of the 24 in ID reactor coolant recirculation lines takes place. Critical flow from both ends of the pipe occurs. During the course of reactor depressurization, nearly all of the primary coolant is expelled into the drywell. Extensive voiding in the core region halts neutron power production by reducing moderation. Furthermore, a scram is initiated by low water level in the reactor and by high pressure in the Primary Containment. The voiding effect on neutron power is thus reinforced by control rod insertion.

Reduced water inventory in the reactor vessel reduces the effective convective cooling of the core. As a result, the core temperature rises until the Core Spray System is activated by the low-low water level at 7'-2" Top-of-Active-Fuel (TAF) water level condition or by high drywell pressure, thus, with the reactor depressurized, re-establishing adequate cooling. Nevertheless, the cladding of a portion of the fuel rods perforates and releases the fission products stored in their fuel rod gas plenums. None of the UO_2 fuel melts or reaches redistribution temperature.

The reactor coolant discharged into the drywell raises the drywell's temperature and pressure, and is vented to the torus where it is effectively condensed. It is assumed that in the process all the gas originally contained in the drywell is swept into the torus and is collected in the air space above the water pool.

On sensing the occurrence of the accident, isolation valves located in process lines penetrating the containment wall are activated, rapidly closing the valves so as to prevent release of radioactive contamination via that route.

The effect of jet forces of the high velocity coolant escaping from the severed pipe upon the walls of the drywell was considered, as well as the localized temperature effect of the jet impinging on the vessel wall surface.

Following coolant blowdown, the long term pressure in the containment depends upon the rate of heat removal, decay energy, and any hydrogen recombination and chemical energy released to the containment.

In addition to the use of the Core Spray System for removal of the decay heat from the reactor fuel, a containment spray system is utilized to remove heat from the entire system. Water drawn from the suppression pool will be pumped through the Containment Spray heat exchangers in the Reactor Building and to Containment Spray System headers in the drywell and torus, thereby cooling the entire containment and its internal components. The water accumulating at the bottom of the drywell will drain back into the suppression pool, via the vent system, to complete the loop.

The blowdown and transport of insulation debris to the torus region will be impeded by plant design and layout. Direct blowdown to the torus from pipe breaks within the drywell will be impeded by baffles at the inlets to the torus downcomers, followed by low bulk fluid velocity transport to the suction strainers. Additionally, insulation is a mix of reflective metallic and "blanket" type insulation. Metallic debris will not likely be drawn to the intake structures due to elevation difference between the intake structures and the torus bottom.

The parameters selected in the design of the drywell and torus were based on experience gained from the Bodega Bay tests conducted by Pacific Gas and Electric Company, at Moss Landing, in 1962:

a. The drywell and connecting vent system tubes are designed for 44 psig internal pressure at 292°F and/or 35 psig at 281°F (the corresponding saturation temperature), and an external pressure of 2 psig at 205°F.

b. The torus is designed for an internal pressure of 35 psig at 150°F and an external pressure of 1 psig at 150°F.

c. The drywell is designed to withstand a local hot spot temperature of 300°F with a surrounding shell temperature of 150°F, concurrent with the design pressure of 44 psig.

d. The minimum vent tube area is equal to the total design accident break flow area (twice the recirculation pipe area) divided by 0.0194. The entrance area around the jet deflection baffles from the drywell to the vent tubes is a minimum of 1.4 times the vent tube area in order to minimize entrance losses.

e. ASME Code impact test requirements, for materials which are used for pressure containing parts of primary containment vessels, call for the establishment of the lowest service metal temperature which will be experienced in service while the plant is in operation. The lowest temperature to which the primary containment vessel pressure retaining parts are subjected to while the plant is in service is 40°F. This temperature has been selected on the basis that the drywell and torus are housed in a building which is maintained above this minimum temperature during reactor operation, and that pressure-retaining parts are maintained at this temperature or above while they are subjected to the design loadings during the post transient operating sequence. To provide an additional factor of safety, a temperature of 30°F was actually used for the design basis.

6.2.1.1.2 Design Features

The primary containment pressure suppression system and associated systems are illustrated in Figure 6.2-2, and consist of:

a. A drywell, which houses the reactor system and its branch connections, and which has the primary objective of containing and channeling reactor system coolant flow to the torus, under accident conditions.

b. A torus, which contains a suppression pool and gas space for the reduction of pressure by condensation and containment of reactor system coolant.

c. A pressure suppression vent system, which directs reactor coolant flow from the drywell to the suppression pool in the torus.

d. A torus-to-drywell vacuum relief system, which prevents suppression pool water from backing up into the drywell during various reactor coolant and suppression system condensation modes, and which limits negative pressure differentials on the drywell in conjunction with the torus vacuum relief system (see Item e below).

e. A Reactor Building-to-torus vacuum relief system, which limits the torus negative pressure relative to atmospheric pressure. This system, in series with the torus-to-drywell vacuum relief system (see Item d. above), also limits drywell negative pressures relative to atmospheric pressure, and permits gas flow only inward from the atmosphere to the containment.

f. A Containment Spray System, which removes core decay heat and metal-water reaction heat (if any) rejected into the drywell following a Loss-of-Coolant Accident. This system is also capable of cooling the water in the suppression pool during normal operation or under postaccident conditions.

g. A Core Spray System, which prevents fuel clad melting under LOCA conditions.

h. A containment gas inerting system, which maintains a non explosive atmosphere in the containment.

i. A containment venting and filtering system, which can be used for containment postaccident recovery operations and for emergency overpressure relief.

j. A drywell cooling system, which limits normal operating temperature in the containment and which is used to mix, cool and purge the gases in the containment during normal plant operation.

k. A Containment Isolation System, including leak testing provisions, to ensure that containment outleakage is minimized during accident conditions.

The Primary Containment was designed, fabricated, inspected, and tested in accordance with the requirements of the ASME Boiler and Pressure Vessel Code, Section VIII, and Nuclear Code Cases 1270N-5, 1271N and 1272N-5.

Special precautions, not specified by Codes, were taken in the fabrication of the steel drywell shell. The plate was preheated to a minimum temperature of 200°F prior to welding of all seams whose thickness exceeds 1 1/4 inches, regardless of surrounding air temperature. Preheat at a minimum of 100°F was applied prior to welding all seams which are 1 1/4 inches in thickness, regardless of surrounding air temperature. Preheat at a minimum of 100°F was applied prior to welding of all seams less than 1 1/4 inches in thickness if the ambient temperature fell below 40°F. Charpy V-notch specimens were used for impact testing of plate and forging material to give assurance of proper material properties.

6.2.1.1.3 Design Evaluation

Analysis has shown that, of the various possible Loss-of-Coolant accidents, a complete circumferential break of one of the main 24 inch ID recirculation loop pipes or other equivalent size breaks would result in peak containment pressure. The Design Basis Accident therefore assumes such a failure to take place when the reactor is operating at full power with normal water level in the reactor. It is assumed that coolant is discharged at critical flow from both open ends of the pipe.

Two sets of computer code calculational results are presented here. The first computer code calculations predict that, under the above postulated conditions, the blowdown would result in a maximum drywell pressure of 33 psig at 275°F and a maximum torus pressure of 20 psig at a temperature of 150°F. The time pressure and time temperature curves for the blowdown and decay are given in Figures 6.2-3 through 6.2-5.

The second calculation predicted that the blowdown would result in a maximum drywell pressure of 38.4 psig at 283°F and a maximum torus pressure of 26.6 psig at 121°F. The time pressure and time temperature curves for the blowdown are given in Figures 6.2-58 through 6.2-60.

The results of actual tests in support of the Bodega Bay Plant for Pacific Gas and Electric Co. (Reference 1) when interpreted for Oyster Creek conditions predict a maximum drywell pressure of 37 psig and a maximum torus pressure of 20 psig. This is considered good confirmation of the computer code technique.

Since the drywell is designed for 44 psig at 292°F and 35 psig at 281°F, and the torus is designed for 35 psig at 150°F, there is more than ample margin of safety.
The analysis is based on the following input data:

The drywell dimensions are a sphere of 70'-0" inside diameter with a cylindrical shell at the top of 33'-0" inside diameter by 23'-3" high. The total volume of the vessel, including connected vent tubes is:

Gross Volume	227,000 ft^3
Occupied Space	47,000
Net Gas Volume	180,000 ft^3

The total volume of the liquid in the Reactor Coolant System (RCS), which could be discharged into the drywell and carried over into the torus in case of an accident, at a temperature of 140°F, is calculated to be:

Reactor Coolant	6,190 ft^3
Recirculation Water	1,020
Main Steam System	20
Feedwater System	350
Cleanup System	10
Shutdown Cooling System	10
TOTAL	7,600 ft^3

Assuming a 50°F rise in the suppression pool water temperature, with the reactor operating at full design power, the amount of water required to absorb the RCS sensible heat plus the heat generated in 10 seconds of full design power operation is calculated to be:

Reactor Vessel and Internals	17,305 ft^3
Reactor Coolant	53,910
Recirculation Water	9,250
Main Steam System	390
Feedwater System	1,910
Cleanup System	60
Shutdown Cooling System	120
Isolation Condenser System	50
Heat Generated in 10 Seconds	4,890
TOTAL	87,885 ft^3

The size of the suppression pool is calculated from the gas law equations, and the following parameters:

Initial Pressure	16.0 psia
Final Pressure	46.8 psia
Initial Temperature	90°F
Final Temperature	140°F

Hence, the maximum water volume available for absorption is 87,900 ft^3, plus 3100 ft^3 for level control. The volume of the structural materials in the torus is 13,000 ft^3; which includes the volume of vent tubes connected to the drywell. The gas volume of the torus is 119,000 ft^3 making the gross volume of the torus 223,000 ft^3. This value is consistent with the 101 foot major torus diameter and 30 foot minor diameter.

The vent system size is consistent with the criterion of twice the recirculation pipe area divided by 0.0194, or 46,700 in^2. The actual area of the downcomers is the total area of one hundred and twenty 24 inch diameter Schedule 10 tubes (0.250 inches thick), or 52,000 in^2. The original vacuum breaker sizing was based on the minimum requirement of vent area divided by 16, or 2920 in^2. The actual vacuum breaker area for seven 24 inch Schedule 20 tubes is 2970 in^2. During the Mark I Containment Program, it was identified that this design basis requirement of a one-to-sixteen ratio may not be the minimum vacuum break area required. As a result, a task of the Mark I Containment Program was established to identify the functional requirements for the Mark I torus-to-drywell vacuum breakers. NEDE-24802 (Reference 17) is a product of this task, and includes a sizing code to be used to determine a true minimum required vacuum break area, setpoint, and opening time based on first principles of thermodynamics. This sizing code was used for Oyster Creek and was documented in Reference 18, and was the basis for technical specification amendment no. 230. The results showed that 8 vacuum breakers are required to provide the vacuum relief function. An additional vacuum breaker is included for single failure criteria, bringing the total required to 9.

The key geometric and thermal parameters were duplicated for the Oyster Creek containment, to insure that condensation would be complete during blowdown.

The Bodega Bay and Humboldt Bay (Reference 2) tests form the major basis for the Pressure Suppression System design and specifically the Bodega Bay tests program.

The basis for determining the maximum design pressure is an analytical model calibrated with the Bodega Bay Test and the key tests used in establishing the Oyster Creek plant containment design. This was accomplished by duplicating, in the Oyster Creek containment, the key geometric parameters of the tests, such as vent to break area ratio, vent resistance, vent submergence, and maximum pool temperatures after blowdown.

In the Bodega Bay tests, the reactor vessel was simulated by a 1250 psig, 27 in. diameter, 21 ft long tank having a volume of 80 ft^3 of which about 54 ft^3 was water. The pipe break accident was simulated by breaking rupture discs mounted on the flange at the bottom, and letting the water and steam discharge into the drywell through various sized orifices or nozzles.

The drywell was simulated by a 150 psig, 85 in ID, 29 ft long vessel connected to the pressure vessel through a 20 in nozzle. The simulated drywell volume was 1100 ft^3.

The torus was contained within a vessel, 12 ft ID and 49 ft long. The torus itself was a section of this vessel, extending across the diameter of the vessel and extending downward from the top 28 ft. The sides of the section were parallel and 3'-8" apart. The bottom was curved, with a

13 ft radius. The rest of the vessel was filled with concrete. The torus contained 670 ft^3 of air and 339 ft^3 of water.

Instrumentation was included to measure all pertinent parameters, including drywell and torus pressure versus time.

The following were the principal steps taken in performing a test with this facility:

a. The desired test orifice or nozzle, together with the rupture disc assembly, was installed on the discharge flange of the reactor vessel.

b. The torus and pressure vessel were filled with water to the desired level.

c. Saturated steam from a 1400 psig source was admitted through a nozzle in the bottom of the reactor vessel to heat the water. During this time the reactor vessel was vented to remove any non condensible gases and drained to maintain the desired water level.

d. Prepurged tests were run to determine the effect of prepurging. For these tests, steam was admitted to the drywell by opening a valve connected to the 1400 psig steam source. This steam was permitted to flow until the torus pressure stopped rising, indicating air purging had ceased. This valve was then closed and the test continued normally.

e. The pressure between the rupture discs was then vented, thereby allowing the discs to rupture and blow down the vessel.

For all tests, reactor vessel pressure was at or near 1250 psig initially. After the rupture discs broke, there was a sharp drop in pressure for all tests, the amount of the drop increasing with orifice size and the amount of initial reactor vessel water subcooling. This was followed by a short period of fairly steady pressure and then a gradual pressure decrease. This was in turn followed by a more rapid rate of decrease in pressure. The initial drop in pressure is believed to be caused by a brief delay between initiation of flow and the start of flashing of the water in the vessel.

The change from a gradual pressure decrease to a more rapid rate of decrease occurs at the time that all water has been expelled from the reactor vessel; the more rapid rate of decrease resulting from the fact there is no more water to flash and help maintain pressure. The time after start when this change occurs is assumed to be the duration of water flow for purpose of calculating flow rates. All these characteristics were also observed during the Humboldt tests. The torus pressure oscillograph trace shows an initial sharp rise, generally to about 12 or 13 psig, followed by a slight drop in pressure; then, pressure would rise gradually reaching a maximum at about the time all water and steam was expelled from the reactor vessel. This gradual pressure rise was smooth for the smaller orifice sizes and was accompanied by some pulsations for the larger orifice sizes.

The initial sharp rise in pressure followed by a slight drop was also observed during the Humboldt testing. Visual observations of suppression pool action were also made during the Humboldt testing and this pressure characteristic was attributed to the observed violent, but brief, upward surge of pool water caused by the large flow of air coming over from the drywell at the start of the test, and resulting in momentary compression of the air in the torus air space.

Visual observation was not considered practical during the Bodega testing, but it is considered that the similar pressure characteristic observed was due to the cause described above.

Following the initial sharp rise in pressure, the gradual rise was caused by continued purging of the remaining air from the drywell, and by heatup. At no time did the torus pressure exceed 30 psig, indicating that the condensation of steam under all conditions tested was rapid and complete. Variations of plus or minus one foot in pool water level had no significant effects on performance.

For prepurged tests, the torus pressure traces have similar shapes to those of tests without prepurging. However, the initial sharp rise is only about 6 psi above the prepurged pressure before the tests. The gradual rise following the initial sharp rise has a much lesser slope than tests without prepurging, because little air remains in the drywell to be purged during this period. The gradual rise in the torus pressure is essentially all due to heating up of the torus air and vapor, while energy was being absorbed in the suppression pool.

At the start of a test, drywell pressure increased very rapidly until the water in the vent line was blown out and venting started. Then pressure increased at a slower rate, governed by torus pressure and vent line pressure drop, and decreased toward the end of the test as flow rate and pressure drop decreased.

The following principal conclusions, drawn from the results of the test program and related to Oyster Creek, are:

a. Condensation of steam in the torus is rapid and complete for flows associated with break areas at least up to 250% of the design break and for pool temperatures as high as 170°F at the end of blowdown.

b. Variations in suppression pool level did not affect performance of the containment system.

c. Moderate subcooling of reactor water and drywell preheating did not affect performance significantly.

Vent design was intentionally made to be the same as in the Humboldt and Bodega test configurations from a flow resistance viewpoint. Both the Bodega Bay tests and Oyster Creek submergences were four feet (this submergence has since been changed to three feet as part of the Mark I Torus Program). Humboldt test submergences varied but do not affect this correlation, since maximum drywell pressure occurs after vent clearing.

Interpolation of the test data from Figures 6.2-6 and 6.2-7 show that for Oyster Creek, the maximum drywell pressure is 52.0 psia vs. a design pressure of 58.7 psia and the maximum torus pressure is 34.4 psia vs. a design pressure of 50 psia.

The principal conservatism between the OCNGS Design Basis Accident and the Bodega Bay tests is that the latter were conducted at a reactor vessel pressure of 1250 psig as opposed to the Oyster Creek operating pressure of 1020 psig. This would have a nearly linear effect on the pressure rise in the drywell. A second conservatism is that the effect of friction was not included in calculating the equivalent area for the double ended pipe break for Oyster Creek, thus resulting in a larger break area to vent ratio than is actually present. A third conservatism is that the design pressure of 44 psig for Oyster Creek is 7 psi greater than that obtained by interpolation of the data.

It is concluded that the Oyster Creek drywell and torus pressures would be well within the measured peak transient taken under more severe conditions than are present for Oyster Creek. This further supports the analytical results discussed later in this section.

6.2.1.2 Containment Subcompartments

Not applicable to OCNGS.

6.2.1.3 Analytical Response to Blowdown

6.2.1.3.1 Model Description

Maximum drywell and torus pressures resulting from a recirculation line break are predicted from an analytical basis. The models employed for predicting system states, blowdown, and venting rates tend to overpredict maximum containment pressures when compared with Bodega Bay and Humboldt Bay pressure suppression tests. Figures 6.2-8 and 6.2-9 are typical when compared to the various test runs.

The analytical model incorporates the following assumptions:

a. Steam and water within the drywell follow the saturation line and are in thermal equilibrium.

b. No heat is transferred to the walls or other structures inside the drywell.

c. No heat is removed by the Shutdown Cooling Water System during blowdown.

d. The system is initially saturated with water vapor.

e. The system is initially at atmospheric pressure.

f. The blowdown from the vessel occurs through an ideal nozzle with no friction, as described in the literature. (Reference 3)

6.2.1.3.2 System States and Flow Rate Considerations

The reactor vessel contains saturated steam and water. After an accident, the drywell and torus will contain various amounts of gases in addition to steam and water. It can be expected that the gases will be mixed to some degree with steam. A gas mixing analysis has shown that system pressure changes only about 2.0 psi, depending on whether gas and vapor are completely mixed or totally separated at any instant during blowdown. Therefore, the general equilibrium state relationship for continuous completely mixed steam and gas was used in the analysis.

System pressure is expressed by the sum of vapor and gas partial pressures. Mass and internal energy contained in a system at any time generally are expressed by integrals of flow rate over time, where inflows are considered positive and outflows negative.

Flow rate per unit area between the drywell and the torus vent area is taken as a function of flow path resistance, stagnation enthalpy, and source and receiver pressure. (References 4 and 5) Blowdown flow rates from the reactor vessel are independent of drywell pressure, whereas vent flows may strongly depend on the drywell and torus pressures.

When steam and air flow simultaneously, the fraction of either is determined by that current fraction existing in the drywell. Water expulsion inertia causes rapid drywell pressure rise until the vents are cleared. It was, therefore, necessary to consider the dynamics of vent water expulsion to determine if further correlation parameters had to be included.

The torus air temperature is found assuming a constant volume available for the air. It has been assumed that all the air leaving the drywell enters the torus air space.

The suppression pool temperature is based on an energy balance of the pool, assuming that all the steam and liquid water entering from the drywell remain in the pool.

6.2.1.3.3 Accident Conditions and Description

In accordance with the design criteria, the following conditions exist at the time of the accident, and are used as the basis for calculating the results of the pipe break:

a. The reactor is operating at a thermal power level of 1860 MW (thermal) and 1020 psig. The maximum pressure attained during the blowdown is not particularly sensitive to power level because the peak pressure is determined by the mass flow rate leaving the vessel during the first few seconds which is primarily a function of pressure and it does not change significantly during that time span.

b. The initial conditions in the drywell are 150°F, 15.0 psia, and 100 percent relative humidity. This temperature represents the average within the drywell as maintained by the air cooling system.

   Note: If the drywell bulk temperature exceeds 150°F, there is an 8 hour time limit to get the drywell bulk temperature below 150°F. (Reference #19)

c. The temperature in the torus air space is 90°F, and is assumed to be in thermal equilibrium with the suppression pool water. The water volume is assumed to be at its maximum value in order to minimize the torus air space, thereby maximizing the calculated pressure at the end of blowdown.

d. Scram occurs on high drywell pressure within the first 0.25 seconds. Reactor shutdown begins immediately after the break because of void formation within the core.

e. No external fluid losses or inputs to the containment are assumed to occur during blowdown.

f. No condensation within the drywell is assumed to occur during blowdown. The steam volume and associated energy stored in the drywell immediately after blowdown is not included as an energy storage. No credit is taken for the reactor vessel volume.

The design basis accident is the complete instantaneous circumferential break of one of the recirculation lines while the reactor is at rated power. Voids form within the core, shutting the power down, and this is followed within a fraction (0.25) of a second by a high drywell pressure scram signal and later by the low level scram signal. All the water and steam is assumed to leave the vessel.

Immediately following the pipe break, the air steam mixture is vented to the torus. Within the first few seconds, all the air is swept into the torus. Because of the high velocity steam within the vents, the air cannot diffuse back into the drywell and it is all effectively pumped into the torus. After blowdown is complete, (about 20 seconds) only steam is present in the drywell. As the steam condenses on the various surfaces and the drywell spray is activated, the drywell pressure drops. This allows the vacuum breakers to open admitting the gas/steam mixture from the torus into the drywell, thus equalizing the two pressures.

The Core Spray System reaches full flow within 35 seconds after the pipe breaks, spraying water over the core and removing decay heat. The energy is transferred to the drywell in the form of hot water and steam spilling into the vents. The initial transient during which the drywell reached maximum pressure is now over. The long term quasi-steady state response of the containment begins as it responds to decay heat addition and the heat removal system.

The blowdown is calculated by assuming a perfect nozzle. The energy release rate is a function of time, and was obtained from Figures 6.2-10, 6.2-11 and 6.2-12, showing the flow rates leaving the vessel, the mass in the vessel, as well as the saturation pressure as a function of time. The important geometric parameters of the venting system were made the same as the Bodega tests, including the break area to vent area ratio, loss factor, and submergence.

6.2.1.3.4 Calculation Procedure

The system of equations representing the pressure suppression system must be solved simultaneously by analog or digital methods. A digital computer program was used to arrive at a solution.

6.2.1.3.5 Results

The peak drywell pressure calculated during the blowdown phase was 33 psig at 275°F. The torus peak pressure was 20 psig at 150°F. The curves are shown in Figure 6.2-3. The preceding discussion applies to the first 60 second portion of the curves.

The peak drywell pressure of 33 psig is lower than the 37 psig pressure interpolated from the test +10 data, primarily because the reactor pressure in Oyster Creek is 1020 psig rather than 1250 psig which was used in the tests. The drywell is designed for a 44 psig internal pressure coincident with a 292°F temperature and a 35 psig at 281°F. The torus is designed for a 35 psig internal pressure and a coincident 150°F temperature. Therefore it is concluded that the design conditions for the containment are well within that calculated for the Design Basis Accident.

6.2.1.3.6 Operational Considerations

The OCNGS primary containment design considers loads and load combinations corresponding to normal operating temperatures up to 150°F at close to atmospheric pressure (Subsection 3.8.2.3.b.1). During normal operations, the calculated bulk drywell temperature is usually near or a few degrees above 135°F. However, the accident scenario presented in Subsection 6.2.1.3.3 assumes an initial drywell bulk temperature of 150°F at the time of the break.

A series of analyses was performed to determine primary containment response to the design basis LOCA from various initial drywell temperatures (Reference 12). The resulting peak temperatures and pressures are nearly identical irrespective of the initial drywell temperature.

6.2.1.4 Long Term Response After Blowdown

6.2.1.4.1 Events Following Blowdown

Decay heat is removed from the core by the core spray water and transported to the drywell, where the core spray water and containment spray water mix and flow down the vents to the suppression pool. The Containment Spray heat exchangers remove heat from the pool. Only one of two loops was assumed to be operating for this evaluation. The heat transfer for the Containment Spray heat exchangers is a function of the inlet temperatures as given in Table 6.2-15. Also, the flow from one of the two Core Spray Loops was used in the calculations.

Following the initial blowdown and activation of the Containment Spray System, the containment pressure decreases rapidly. Without containment spray the pressure decrease would be very slow but would occur in any event due to condensation. The torus temperature continues to increase slowly until the heat exchanger heat removal ability just equals heat added by fission product decay. Thereafter the pressure and temperature will slowly decrease with containment spray system operation.

The core temperatures following the pipe break increase to a peak (below 2200°F) and remain at this value under the influence of the core spray cooling water until 40 to 60 minutes after the break. At this time, the fuel rods are wetted by a water film, reducing their temperature to under 300°F, and the critical phase of the Loss-of-Coolant Accident is essentially terminated. The core can then remain at these conditions for an indefinite period.

6.2.1.4.2 Analytical Model Description

This section describes the analysis model used in calculating the long term pressure and temperature response following a Loss-of-Coolant Accident. The model applies only after the blowdown phase has terminated. Assumptions are as follows:

a. Drywell and torus are at the same pressure.

b. All gases are saturated with water vapor.

c. Gases in the torus are at suppression pool temperature.

d. The Suppression pool is the only body in the system which stores energy.

e. The mass of water in the suppression pool is constant after blowdown of primary vessel.

f. The heat capacity of water is constant.

g. The internal energy of the reactor vessel and associated piping is constant during the process.

The temperature of the suppression pool is calculated as a function of time, in a conservative manner, by considering the pool to be the only heat absorber in the system. The effects of decay heat energy, stored energy in the core, and any energy from the metal-water reaction on the pool temperature are included. Also, the effect of heat exchangers in one Containment Spray System loop is included.

The drywell temperature is calculated considering an energy balance on the containment spray and/or core spray. The containment spray water enters at the discharge temperature of the heat exchanger and the core spray water enters at the suppression pool temperature. The combined flows (containment spray and core spray) drain back to the suppression pool, having been heated by the decay heat energy, stored energy in the core, and any possible metal-water reaction chemical energy. The drywell temperature is then taken to be 5°F hotter than the exiting flow. Where it is assumed that the containment and core sprays are not operating, no credit for heat removal is taken.

The total number of moles of non-condensible gas in the entire system (drywell and torus) is determined from the amount of gas originally in the system, plus any gas generated from the metal-water reaction.

6.2.1.5 Containment Capability with Respect to Metal-Water Reactions

6.2.1.5.1 Nature of Requirements

If the zircaloy in the reactor core is heated above about 2000°F in the presence of steam due to an accidental loss of coolant, a chemical reaction occurs in which zirconium oxide and hydrogen are formed. This is accompanied with an energy release of about 2800 Btu per pound of zirconium reacted. The energy produced is absorbed in the suppression pool. The hydrogen formed, however, will result in an increased pool, due simply to the added moles of gas in the fixed volume (depending on the amount produced). Although very small quantities of hydrogen are produced with the core spray, the containment has the inherent ability to accommodate much larger amounts as discussed below.

6.2.1.5.2 Expected Metal-Water Reactions

The metal-water reactions during core heatup, and within the first 40 to 60 minutes during which portions of the core are at temperatures significant to metal-water reactions, have been calculated by a core heatup computer code. The core was subdivided into nodes consisting of five radial zones, five axial nodes, four relative rod powers within each bundle, and with four radial fuel nodes in each fuel rod. Heatup was calculated during the blowdown phase employing experimentally determined heat transfer coefficients. Under core spray conditions the coefficients from prototype tests were experimentally determined. The metal-water reaction was calculated at each node as determined by the parabolic law. (Reference 6) This was integrated over the entire core until the rods were finally wetted and cooled by the Core Spray System, about an hour after the accident. The extent of the metal-water reaction, thus calculated for the Oyster Creek core, is under 0.6% of all the zirconium in the core. This reaction produces an energy release of an additional 0.4% of the total energy release during the loss of coolant. The amount of hydrogen produced across the complete spectrum of breaks is less than 15 pound moles which, when mixed with all the air in the drywell, is less than a 2.5%

mixture; well under the 5% mixture required to reach the flammability limits. The effect on pressure of this added amount of gas is insignificant since it is such a small fraction of the total moles present. Thus the expected metal-water reaction is not a significant contributor to either pressure considerations or the energy load on the Primary Containment.

However, the Primary Containment should be capable of tolerating some arbitrary event-dependent amount of metal-water reaction greater than that calculated for the postulated accident. Without a specified actual event it is not possible to calculate precisely either the duration or quantity of the metal-water reaction. However, as an index of the containment's ability to tolerate postulated metal-water reactions the concept of "Containment Capability" is used. Since this capability depends on the time domain, the duration over which the metal-water reaction is postulated to occur is one of the parameters used. The model is described in the next subsection.

6.2.1.5.3 Inherent Capability of Containment

The basic approach to evaluating the containment capability is to assume that energy and gas are liberated from the core region in a uniform manner over an arbitrary time period. Capability is measured in terms of the maximum percent of fuel channels and fuel cladding material which can enter into a metal-water reaction without the containment design pressure being exceeded. Since the percent metal-water reaction capability varies with the duration of the uniform energy and gas release, the percent metal-water reaction capability is plotted against the duration of release. This constitutes the containment capability for a particular system.

The following assumptions are made:

a. The drywell and the torus are at the same pressure.

b. Both energy and gas are released uniformly from the core region over an arbitrary time duration.

c. All gases are saturated with water vapor.

d. The torus air space gases are at suppression pool temperature.

e. The suppression pool is the only body in the system which stores energy.

f. The mass of water in the suppression pool is constant after the blowdown of the primary vessel.

g. Heat capacity of water is constant.

h. The internal energy of the reactor vessel and associated piping is constant during the event.

It can be shown that if the drywell temperature is known the suppression pool temperature is known and the system pressure can be stipulated, then the number of moles of noncondensible gas in the system can be determined. The total number of gas moles in the system is directly related to the metal-water reaction.

To determine the temperature of the drywell and torus, it is assumed that the gases in the torus are at the temperature of the suppression pool. Since it is assumed that the suppression pool is the only body in the system which stores energy, as energy is released from the core region it is absorbed by the suppression pool. Energy is removed from the pool by heat exchangers rejecting heat to the Emergency Service Water System. If the energy release is conservatively assumed to be uniform, and the service water temperature and heat exchanger flow rate are constant, the temperature response of the pool can be found in closed form. The temperature of the drywell is determined from an energy balance on the water flows through the drywell.

Sufficient relationships can be established to allow a calculation by computer of the percent metal-water reaction allowable for any assumed duration of energy release. For each particular value of duration, the percent metal-water reaction is iterated upon until the design pressure of the containment is reached. The calculations are made at the end of the energy release duration, not transiently, because the number of moles in the system are at a maximum, and the pool temperature is higher at this time than at any other time during the energy release.

It has been assumed that none of the drywell spray vaporizes. This condition allows the noncondensables to occupy both the drywell and torus and not be swept over into the torus by steam flow from the drywell. This condition does not exist for all durations. For short durations of energy release, steam is produced. Therefore, a second condition is calculated based upon all of the non-condensable gases being forced into the torus due to vaporization of the drywell spray. The capability is then determined in the same manner, but assuming that all of the non-condensables are stored in the torus air space.

As the corresponding equation is solved for the percent metal-water reaction, an iterative procedure is used to find solutions for the reaction at each duration, as in the previous solution. Finally, the larger percent reaction from two solutions is used at any time, and this represents the capability of the containment.

6.2.1.5.4 <u>Results</u>

Any hydrogen produced from a metal-water reaction would leave the reactor pressure vessel at a sufficiently high temperature to be burned in the drywell. The burning of the hydrogen would reduce the total moles of oxygen, thereby providing a greater magnitude of allowable metal-water reaction. For each particular value of duration, the percent metal-water reaction is iterated upon until the design pressure of the containment is reached. For the case of no drywell spray, all of the non-condensable gases are conservatively assumed to be stored in the torus. The allowable containment capability as shown by the flat portion of the curve on Figure 6.2-13 would be approximately 5% for the case of no burning. The burning of hydrogen gas in the drywell, as evolved, would increase the allowable metal-water reaction to approximately 25%. This is a factor of over 25 greater than the metal-water reaction calculated over the entire spectrum of breaks (less than 1%).

Although the burning of the hydrogen increases the energy content of the containment, the total moles of noncondensible gases is substantially reduced and thus the capability for metal-water reaction actually increases. The curves with containment spray assume rated operation of only one of the two loops. If the hydrogen thus formed does not burn, it would mix with all the air in the containment, and the resulting hydrogen-air moles mixture would be under 2.5%. This is well below the 5% flammability limit.

Figure 6.2-13 shows the capability of the containment to tolerate a broad spectrum of postulated metal-water reactions associated with a Loss-of-Coolant Accident. The initial portion of the curves on Figure 6.2-13 covers the time span during which the uniform energy release rate is high enough to generate steam within the drywell. All the gases are thus transferred to the torus. When the duration is long enough, the containment spray is sufficient to absorb all the energy without steam generation. The containment capability then increases with time as energy is removed from the system.

Even without spray, the containment can tolerate a significant amount of metal-water reaction several times greater than the 0.6% actually calculated across the break spectrum, consistent with the core cooling systems provided.

6.2.1.5.5 Conclusions

It is concluded that, even without containment spray or inerting, the capability of the Oyster Creek containment to tolerate postulated metal-water reactions following a Loss-of-Coolant Accident, provides ample margin above the level estimated to occur.

With containment spray, the capability is changed to accommodate the reaction of major fractions of the core, depending on the time duration over which the metal-water reaction is postulated to occur.

6.2.1.6 Testing and Inspection

Information concerning the containment testing and inspection program is presented in the Technical Specifications.

6.2.1.7 Instrumentation Requirements

The information provided for monitoring the containment conditions and actuating those systems and components having a safety function is discussed in Chapter 7.

6.2.2 Containment Heat Removal Systems

The Containment Heat Removal Systems are designed to reduce containment pressure and temperature following a Design Basis Loss-of-Coolant Accident (LOCA), by removing thermal energy from the containment atmosphere. These systems also serve to limit offsite doses by reducing the pressure differential between the containment atmosphere and the external environment. The Containment Spray and Emergency Service Water Systems comprise the Containment Heat Removal Systems for the OCNGS.

During the 14R refueling outage, the containment spray system was modified from an automatically actuated to a manually actuated system. An evaluation of this change is contained in Reference 13 and in License Amendment No. 160, (Reference 14).

6.2.2.1 Design Bases

The design bases of the Containment Heat Removal Systems are as follows:

a. To remove heat from the Primary Containment and, in conjunction with the Core Spray System, to assure continuity of core cooling.

b. To provide redundancy in the event of a single active component failure.
c. To provide for manual initiation of the system.
d. To be able to test active components during normal operations.

6.2.2.2 System Design

The Containment Spray System consists of two redundant loops which deliver water from the suppression pool to the spray headers in the drywell and torus. Each loop consists of two 100% pumps in parallel, two 50% heat exchangers in parallel, and two 50% drywell spray headers and a torus spray header (torus spray sparger can use up to 5% of containment spray system flow). Both loops share a common suction header from the suppression pool and a common spray header in the torus air space. The Containment Spray heat exchangers are cooled by the Emergency Service Water (ESW) System. There are two 100% ESW pumps per Containment Spray loop. Both Containment Spray and ESW pumps are powered by the Emergency Diesel Generators on loss of offsite power. The system was designed in accordance with ANSI B31.1 piping code. The Containment Spray piping arrangement is shown on Drawing GE148F740. The Emergency Service Water system piping arrangement is shown on Drawing BR 2005, Sheet 4. Minimum performance requirements for the Containment Spray and Emergency Service Water Systems are provided in Table 6.2-15. Component data is provided on Tables 6.2-3 through 6.2-9.

Water is pumped from the suppression pool through the suction strainers to the heat exchangers, sprayed into the containment and flows by gravity back into the suppression pool via the vent headers. The water spray removes latent and sensible heat from the drywell. The heat is rejected to the Emergency Service Water System via the Containment Spray heat exchangers.

The Containment spray flow is indicated, and containment spray temperature recorded, on Panel 1F/2F in the Control Room. Low flow is alarmed on Panel 1F/2F. Two redundant drywell pressure measurement channels are provided, each with indication on Panel 1F/2F in the Main Control Room.

Containment Spray System safety-related motor-operated valves are included in the Generic Letter (GL) 89-10 Motor-Operated Valve (MOV) Program as noted in the OCNGS Program Description for NRC Generic Letter 89-10 Motor-Operated Valve Program. This program has reestablished the design bases for safety-related motor-operated valves. Critical design basis assumptions, such as design bases differential pressure, safety function - open vs. close, minimum available AC/DC voltages, actuator gearing, torque switch control logic, valve factors, stem friction coefficients, and valve stroke times have been established in assessing GL 89-10 design bases capability. Plant changes or activities which can affect these design bases assumptions must consider the affect on the capability of GL 89-10 motor-operated valves to perform their safety function and on safety margins established for these valves.

For corrosion protection, all Emergency Service Water piping, except the intake structure, is internally coated with Type B coal-tar primer and coal-tar enamel in accordance with AWWA C-203 or with an engineer approved alternate coating. External Surfaces of buried ESW lines are coated with a Type B coal-tar primer and coal tar enamel in accordance with AWWA C-203 or with an engineer approved alternate coating. Where the piping has not been coated, in accordance with the previous discussion, evaluations were made to verify the design life of the pipe. This piping was replaced by coating approved piping prior to its end of design life.

Sodium hypochlorite is injected into the ESW system via the 2" Service Water Keep Full Line downstream of the ESW pumps to minimize bio-fouling.

Buried ESW piping is protected from freezing by maintaining a minimum cover above the pipe. Where the grade of the site above the pipe did not provide the minimum cover a filled barrier arrangement is used to provide the minimum cover.

6.2.2.2.1 Containment Spray Pumps

The Containment Spray pumps are located on El. (-)19'-6" in the Reactor Building. Pumps 51A and 51B in Loop 1 are in the northeast corner room. Pumps 51C and 51D in Loop 2 are in the southeast corner room. The performance curve for these pumps is shown on Figure 6.2-15. Pump characteristics are presented in Table 6.2-3.

The Containment Spray pumps are operated from switches in the Control Room. There are no local control switches. Two independent mode selector switches are provided, one for each loop. Each switch has two modes, "Drywell Spray" and "Torus Cooling". Moving the selector switch to "Drywell Spray" aligns that respective loop's valves to spray the drywell. Moving the selector switch to "Torus Cooling" aligns that respective loop's valves to cool the torus. A loss of 125 VDC control power will cause the valves to line-up in a "torus cooling" configuration. The Containment Spray pumps can be started manually for containment spray service if the mode selector switch is in the "Drywell Spray" mode, and the drywell pressure is greater than 0.6 psig.

Starting of the pumps requires simultaneous positioning of two switches. For each pump there is a STOP-NORM-START switch with spring return to normal, and for each loop there is an A(C)-NORM-B(D) interlocked switch with spring return to normal. This two-handed, pressure interlocked manual start procedure minimizes the potential for inadvertently spraying containment.

Containment spray pumps are automatically tripped when drywell pressure decreases to less than 0.6 psig, and if the mode selector switches are in the "Drywell Spray" mode. There is no automatic trip of any containment spray pumps operating in the "Torus Cooling mode.

Although, initiation of containment spray is by procedure, an interlock is provided to prevent pump start if diesel generator load sequencing is in progress.

6.2.2.2.2 Drywell and Torus Headers and Spray Nozzles

Each loop has two connected ring headers (Table 6.2-8) at two elevations inside the drywell. The full cone fine spray provides 100 to 200 micron uniform droplet size. Each nozzle can pass 1/8 inch particles without plugging.

A common 4 inch spray header is located in the torus air space at El. 10'-2" (Table 6.2-4). The header has 10 nozzles evenly distributed through the torus. The torus nozzles are the same as the drywell spray nozzles.

6.2.2.2.3 Suction Strainers

Water for the Containment Spray pumps is drawn from the suppression pool through three suction strainers. The suction strainers are sized to accommodate debris associated with the design basis loss of coolant accident while passing flow to two core and containment spray systems. The torus penetration nozzles from the suction strainers are connected to a 20 inch suction header, which nearly surrounds the torus at El. (-)15'-4". The 12 inch suction lines for each Containment Spray pump connect to this 20 inch header. The Core Spray pumps also take suction from this header (see Section 6.3).

6.2.2.2.4 Emergency Service Water Pumps

The four Emergency Service Water pumps (Table 6.2-6) are located at the Intake Structure. The two pumps in a loop discharge into a common header which supplies both heat exchangers in that loop. There are no shutoff valves between the pumps and the heat exchangers in ESW system II. There is a cross-connect line from ESW system I to SW system. This line will be isolated by a spectacle flange during normal operation. The shutoff valve between the pumps and the heat exchangers in ESW system I will be locked open during normal operation. During abnormal plant conditions, both SW pumps may be removed from service provided that the heat removal function requirements are accommodated by another source that will be evaluated on a case-by-case basis. For example, ESW system I can be aligned to the SW System. Check valves prevent backflow through the standby pump. There are 2-inch connections from the Service Water System into the Emergency Service Water pump discharge header for each loop, which maintains the ESW lines full (see Section 9.2). There are, locked in position, handwheel operated valves (V-3-87,88) in the discharge line for each loop, which discharges to the discharge canal. These valves also provide secondary containment isolation when the cooling water side of a corresponding Containment Spray Heat Exchanger is opened to the containment atmosphere.

The Emergency Service Water pumps are controlled manually from switches located in the Control Room; these pumps may be manually operated at any time except when diesel-generator load sequencing is in progress. The pumps may be manually operated after diesel-generator load sequencing is complete. While operation of the ESW pumps is by procedure, a backup interlock is provided to prevent pump start if diesel-generator load sequencing is in progress.

6.2.2.2.5 Containment Spray Heat Exchangers

The Containment Spray heat exchanger design specifications are identified in Table 6.2-7.

The heat exchanger tubes are rolled into the tube sheets. Three quarter inch safety valves are installed on the channels and shells. The heat exchangers are protected against galvanic action by zinc anodes bolted to each pass partition plate. The heat exchangers are located on floor El. 23'-6" of the Reactor Building, above the Containment Spray pumps.

6.2.2.3 Design Evaluation

6.2.2.3.1 General Discussion

The Containment Spray System consists of two independent cooling loops, each capable of removing fission product decay heat from the primary containment. Each loop has a pair of spray headers in the drywell. Each pair of spray headers is fed by two pumps and through two heat exchangers as shown on Drawing GE148F740.

Cooling water to the tube side of the Containment Spray System heat exchangers is delivered by the ESW System from the ultimate heat sink. Demineralized water from the pressure suppression chamber (TORUS) is circulated through the shell side of the heat exchangers to the Drywell and/or Torus for post LOCA containment cooling. The ESW pumps are throttled as necessary to allow for a positive tube-to-shell-side pressure differential. This differential pressure in the heat exchangers minimizes the radioactive leakage to the environment subsequent to a LOCA in the event of a tube leak. ESW discharge valves V-3-87 and V-3-88 are locked in the proper throttled position to assure that the ESW pumps sustain adequate flow to the Containment Spray Heat Exchangers. Control Room indication of the valve position is maintained.

For reliability, each loop, under the postulated accident conditions, has the heat removal capacity to maintain the primary containment pressure below the torus design pressure of 35 psig. This is illustrated on Figure 6.2-3. During the reactor blowdown associated with the Loss-of-Coolant Accident, the pressure in the drywell rises to 33 psig, then decreases to approximately 20 psig within 30 seconds as energy is absorbed by the suppression pool. All pressures during and subsequent to the initial blowdown are substantially less than the design pressure of the torus. The capacity of each cooling system loop will ensure that the containment pressure continues to remain well below the design pressure following the accident. This is illustrated on Figure 6.2-3, which assumes that only one loop is in operation. The temperature of the containment is also maintained below the design values, as illustrated in Figures 6.2-4 and 6.2-5.

The torus pool temperature response subsequent to a design basis LOCA has also been analyzed assuming the minimum operability limits for the Containment Spray and the Emergency Service Water pumps. The minimum ESW flow to the heat exchangers was determined by assuming postulated breaks to the nonseismic attached piping. The analysis was performed for various containment spray pump flows to determine the effects on NPSH available to the Core Spray pumps. The results of this analysis are shown in Table 6.2-15.

The system is capable of being tested, up to a block valve at the drywell vessel, for design flow conditions with the reactor in operation. Motor operated test valves are provided, so that the system can be returned to normal immediately if conditions so require. The Emergency Service Water pumps can be started and tested under design flow conditions at any time during reactor operation.

6.2.2.3.2 Failure Mode and Effects Analysis

A failure mode and effects analysis (FMEA) was performed to identify single failures in the Containment Spray System. The scope of the FMEA included the manual controls and their supporting power supply system that had the potential for disabling system loops or for spurious actuation of either loop. The divisional power distribution systems (4.16KV, 460V, and 125VDC) to the Containment Spray Pump Motors, Emergency Service Water Pump Motors and Motor Operated Valves are addressed separately in Sections 8.3.1 and 8.3.2.

The specific instrumentation and electrical components reviewed for postulated failures are listed below:

- 125VDC logic power - Dist. Center A Breaker 7, Dist. Center B Breaker 7, Dist. Center C Breaker 4.

- Core Spray System Low-Low Reactor Water Level Relays 16K110A, B, C, D, and associated 125 VDC logic power supplies.
- SWGR 1C Pumps 52A, B Control
- SWGR 1D Pumps 52C, D Control
- USS 1A2 Pumps 51A, B Control
- USS 1B2 Pumps 51C, D Control
- MCC 1A21B MOV V-21-7,-9,-11,-17,-18 Control
- MCC 1B21B MOV V-21-1,-3,-5,-13,-15 Control
- MOV V-21-11,-17,-5,-13 limit switches
- Normal/Emergency interlock relays control power supply

Potential single failures were identified within the boundaries as noted above. These failures and their effects are listed as follows:

a. If either the 16K21, 16K25 or 16K26 relays (low drywell pressure) fail in the energized position, or 16K16 fails deenergized, auto trip of the containment spray pumps is disabled.

b. Failure of the pump breakers in the closed position can prevent termination of containment spray by tripping pumps; however, containment spray can be diverted to the torus by placing the mode selector switch in the "Torus Cooling" position.

For the single failure identified in (a), automatic trip function was disabled. However, manual trip was still available, and analyzed as acceptable to fulfill the intended system safety function.

6.2.2.3.3 Drywell Temperature Response

The response of the drywell to a temperature transient in the drywell fluid was evaluated with a short computer program, which uses a simple forward finite difference analysis and the following assumptions:

a. The drywell can be considered a flat plate with a uniform temperature.

b. There is no heat flow to the compressible material located between the drywell and the concrete.

c. The drywell temperature is 140°F (135°F is the normal condition).

d. The calculated drywell fluid temperature transient is satisfactorily represented by 50 linear sections.

The drywell ambient temperature following a Loss-of-Coolant Accident and the possible range of the drywell liner temperature response are shown on Figure 6.2-18.

During the 50 to 150 seconds immediately following the accident, the fluid temperature will be higher than the wall temperature; the range of heat transfer coefficients associated with this condensing situation is 200 to 1000 Btu/ft^{2-}hr-F. These two values were used in determining the lower and upper boundaries, respectively of the drywell temperature response.

When the temperature difference between the drywell and the drywell fluid reverses, the heat transfer mechanism becomes one of natural convection. Constant coefficients of 2 and 6 Btu/ft^{2-}hr-F were used to evaluate the upper and lower boundaries (respectively) of the liner temperature range.

The drywell and the closure head are designed and constructed to withstand jet forces of the following magnitudes in the locations indicated from any direction within the drywell:

Location	Jet Force (Max)	Interior Area Subjected to Jet Force
Spherical part of drywell	566,000 pounds	3.14 sq. ft.
Cylinder and sphere to cylinder transition	466,000 pounds	2.54 sq. ft.
Closure head	16,000 pounds	0.09 sq. ft.

The spherical and cylindrical parts of the drywell are backed up by reinforced concrete with space for expansion of the material between the outside of the drywell and the concrete.

The above listed jet forces consist of steam and/or water at 300°F, maximum. The jet forces listed above do not occur simultaneously. However, a jet force is considered to occur coincident with design internal pressure and a temperature of 150°F. Where the drywell is backed up by concrete it may be assumed that local yielding will take place but it has been determined that a rupture will not occur. Where the drywell is not backed up by concrete, the stresses resulting from this combination of loads do not exceed 1 1/2 times the allowable stress values given in paragraph UCS-23, Section VIII, of the ASME Boiler and Pressure Vessel Code.

6.2.2.4 Tests and Inspections

The Containment Spray System can be tested (up to the block valves just upstream of the containment penetration) for design flow conditions with the reactor in operation. For the test, the flow is from the heat exchangers to the torus, the spray headers are bypassed. The bypass also permits operation of the system for controlling the temperature of the suppression pool during normal and abnormal plant operation. Motor operated test valves and block valves are provided so that the Containment Spray System can be returned to spray service during testing if accident conditions so require. The Emergency Service Water pumps can be started and tested under design flow conditions at any time. The capability exists where the nozzles can be tested during reactor shutdown by air test connections for each loop located outside the containment.

The Containment Spray System is considered part of the Primary Containment, because the suction and discharge lines are connected to the Primary Containment. During normal plant operation the suction line valves are normally open, the d/w spray line valves are normally closed and the torus cooling valves are normally open. Containment leak-rate measurements are made with the pump suction and system discharge valves open.

6.2.2.5 Instrumentation Requirements

Drywell pressure monitoring is provided on main Control Room panel 1F/2F (PI-IP0008), with a range of 0-75 psig. Additional Class 1E drywell pressure measurement is provided on 1F/2F (PI-642-9A/B) to comply with the requirements of Regulatory Guide 1.97. This measurement has a range of 0-40 psig. Additionally, drywell pressure is sensed at a 90 foot elevation penetration (from the H_2O_2 System) (PT-666-1), with indication on Control Room Panel 12XR, for use during primary containment flooding.

Pressure switches are provided for activating normal/emergency power LOCA interlocks at a drywell pressure increase above 2.9 psig, and to provide containment spray pump automatic trip at a drywell pressure of less than 0.6 psig, when the mode selector switch is in the "Drywell Spray" mode.

Torus pressure is indicated on main Control Room panels 1F/2F and 4F.

Temperature indication is provided to inform the operator of the need to cool the suppression pool. During the Cycle 12R outage, the existing (four temperature sensors) system for monitoring Torus water temperature was supplemented by the addition of a Suppression Pool Temperature Monitoring System, consisting of two separate channels of six temperature sensors per channel, which has the capability to measure, compute, monitor, and record valid Torus bulk water temperature during all plant operating conditions, including accident and post accident, in conformance with the requirements of Appendix A to NUREG-0661 and Regulatory Guide 1.97, Revision 3.

Local flow indication for the Emergency Service Water (E.S.W.) System is provided upstream of the Containment Spray Heat Exchangers via local gages DPI-532-0005 and DPI-532-0006.

In addition to the above, the E.S.W. Flow Instrumentation was modified during Cycle 13R Outage to add Control Room indication.

One (1) instrumentation channel in each of the two Emergency Service Water System loops was added. Each of the instrument loops consists of a pressure transmitter, flow converter and signal conditioner which connect the local process flow sensors to the Control Room computer terminal CRT for flow display on demand.

This plant modification was implemented as a result of a GPUN commitment to the NRC as part of Regulatory Guide 1.97 which requires that certain plant parameters be monitored from the Control Room during plant postaccident condition.

Local pressure indication at each containment spray pump suction is provided.

Table 6.2-10 lists the instrumentation and controls available to the operator.

6.2.3 Secondary Containment Functional Design

6.2.3.1 Design Bases

The Reactor Building completely encloses the reactor and the Primary Containment. The structure and the SGTS system provides secondary containment function when the Primary Containment is in service, and primary containment function during periods when the Primary Containment is open, as is the case during refueling.

The Reactor Building houses the refueling and reactor servicing equipment, new and spent fuel storage facilities, and other reactor auxiliary and service equipment, including the Isolation Condenser System, Standby Liquid Control System, Control Rod Hydraulic System equipment, and components of electrical equipment. The primary objective of the Reactor Building (i.e., secondary containment) is to minimize ground level release of airborne radioactive materials, and to provide for controlled, elevated release through the Stack of the building's atmosphere under accident conditions.

Secondary containment integrity shall be maintained at all times unless all of the following conditions are satisfied.

a. The reactor is subcritical and Technical Specification 3.2.A is met.

b. The reactor is in the cold shutdown condition.

c. The reactor vessel head or the drywell head are in place.

d. No work is being performed on the reactor or its connected systems in the reactor building which could result in inadvertent releases of radioactive material.

e. No operations are being performed in, above, or around the spent fuel storage pool that could cause release of radioactive materials.

6.2.3.2 System Design

The Reactor Building location is shown in Drawing JC19702, and its general arrangement is shown in Drawings 3E-153-02-001 through 3E-153-02-009. The building substructures consist of poured-in-place reinforced concrete exterior walls up to the refueling floor (El. 119'). Above the refueling floor the structure is of steel frame construction with insulated metal siding. The siding is installed with sealed joints. Refer to Table 6.2-11 for the Secondary Containment principal design parameters.

Access openings, by original plant design, include openings for personnel and equipment. These access openings have been designed to assure reactor building leakage is maintained within the design maximum of 100% of the building free volume per day at 0.25 inches of water differential pressure.

Access openings to the Reactor Building, including personnel entrances, equipment entrances and rail entrances, are provided with interlocked double doors to minimize reactor building leakage. The Trunnion Room entrance has only a single door for entry from the Turbine Building at elevation 23'-6" by design. The Trunnion Room is located on the west side of the Reactor Building and is an integral part of the Reactor Building (Secondary Containment). It is not an access opening to the reactor building, by design, as it does not allow access to the entire reactor building. It permits access only to the Trunion room. The door design has been considered within the capability of the Standby Gas Treatment System and the Reactor Building

Ventilation System to maintain reactor building leakage within the designed maximum. Passage through any of the double door entrances to the Reactor Building can occur without loss of secondary containment integrity since only one door is open at a time. Passage through the Trunnion Room door occurs infrequently and is controlled administratively. (Ref 16)

The building is designed to withstand an internal pressure of 0.20 psi without jeopardizing the overall integrity of the building structure. Pressures in excess of 0.20 psi are relieved by torsional buckling of the unbraced sections of girts and the subsequent separation of siding from the major building frame. The separation of the siding relieves internal pressure and limits further structural damage.

The Reactor Building is designed to have a limited inleakage rate in the isolated condition. In this condition the Standby Gas Treatment System exhausts the building atmosphere, through filters, to the plant Stack, maintaining the building's pressure below atmospheric. The maximum inleakage rate is 100 percent of the Reactor Building volume per day under neutral wind conditions when the Standby Gas Treatment System is maintaining a negative pressure of 0.25 inch of water.

6.2.3.3 Design Evaluation

The Primary and Secondary Containments constitute the principal barriers for the mitigation of accident consequences. With the Reactor Building atmosphere being filtered and discharged through the Stack at an elevated release point, the offsite doses under postulated accidents are significantly reduced. The Reactor Building is designed, in accordance with applicable codes, to withstand the most severe natural phenomena and to minimize ground level release of the building contents under adverse conditions.

The offsite accident consequences are relatively insensitive to the Reactor Building inleakage rate as long as the Standby Gas Treatment System can maintain the building at a vacuum and prevent exfiltration at low wind speeds.

The design of the Reactor Building for a low inleakage rate at 0.25 inches of water vacuum results in a low exfiltration rate even at high wind conditions. Calculations have been made of exfiltration in high winds, based on building pressure gradients determined by Irminger and Nkkentved (Reference 7) in wind tunnel tests of building models. Data for seven different wind directions were used in the calculations, from which it was determined that the greatest exfiltration rates arise from those associated with the wind blowing normal to the longest side of the Reactor Building.

The models used for the tests in the above reference were similar in geometry to the Reactor Building except that the length-to-width ratio was somewhat greater (2:1 as compared to about 1.3:1 for the Reactor Building).

A longer length-to-width ratio results in higher calculated exfiltration, so use of the model for Oyster Creek is conservative.

The leakage rate tests at low pressure differentials indicate that leakage rates may be correlated by the following equation (Reference 8):

$$\text{Leakage rate} = a(DP) + b(DP)^{1/2}$$

where "a" and "b" are constants which are dependent upon the leakage characteristics of the building and DP is the pressure differential between the building atmosphere and the outside. To determine the potential range of exfiltration rates, two series of calculations were made; one assuming leakage rates to be proportional to DP to determine maximum exfiltration rates, and the second proportional to $(DP)^{1/2}$ to determine minimum rates. Other basic assumptions include the following:

a. Throughout all ranges of wind velocities, the Standby Gas Treatment System discharges at a rate equivalent to 100 percent of the building volume per day.

b. The building inleakage rate is 100 percent of building volume per day when the building is maintained at a negative pressure of 0.25 inch of water with respect to the outside, under zero wind velocity conditions.

c. Leakage sources are assumed to be distributed homogeneously on all four side walls of the building in proportion to wall length. The roof is assumed not to leak.

Maximum and minimum calculated Reactor Building exfiltration rates as a function of wind velocity are shown in Figure 6.2-20. These data indicate that at wind velocities less than about 35 to 65 mph, there would be little if any, exfiltration from the Reactor Building. The calculations also indicate that exfiltration rates are almost directly proportional to the initial inleakage rate for a given negative building pressure. Calculations show that the exfiltration rate could be many orders of magnitude larger and occur at much lower wind speeds without increasing the post accident doses above 10CFR100 guideline limits.

The Reactor Building leakage rate specifications and controlled release ventilation rate provide a more than adequate system for Secondary Containment during reactor operation and for Primary Containment during refueling operations. At the high wind velocities necessary for exfiltration, the high wind results in significant dilution. Although there is a potential for increased offsite dose rates to the thyroid and lungs under high wind exfiltration conditions for the accidents described in Chapter 15, these are far below the dose rates generally considered acceptable in accident situations. Further, on the basis of wind velocity data (see Section 2.3), it is expected that high wind exfiltration conditions would not exist for more than a few hours per year.

The Reactor Building is provided with two systems of ventilation. These two systems are Reactor Building H&V and Standby Gas Treatment System. During normal power operation, shutdown, or refueling, the normal ventilation system provides fresh, filtered air to all levels and equipment rooms of the Reactor Building (see Section 9.4). During shutdown and refueling, the normal capacity is increased, if necessary, to provide additional ventilation to the fuel handling and drywell areas. Heating and cooling units are installed within the ventilation system to maintain temperatures for personnel comfort and equipment protection. The normal ventilation system provides a minimum of one air change per hour. The air flow pattern is from the filtered supply to uncontaminated areas, to potentially contaminated areas, and thence to the Stack. The air is not filtered before being exhausted.

Two isolation valves in series are provided in the Reactor Building, on both the inlet and outlet ducts which will close automatically when high radiation levels in the Reactor Building are detected or upon receipt of a primary containment isolation signal (high drywell pressure or lo-lo reactor level). The inlet and outlet isolation valves are also closed automatically upon manual initiation of the Standby Gas Treatment System. The inlet isolation valves are also closed automatically if the Reactor Building pressure is greater than 1" WG, high ductwork temperature is detected, or valves V-28-21 or –22 are not full open. These valves can also be manually isolated. Diagrams of the Reactor Building H&V and Standby Gas Treatment Systems are shown on Drawings BR 2011 and 3E-822-21-1000.

The Standby Gas Treatment System (see Section 6.5) is utilized to treat and exhaust the atmosphere of the Reactor Building to the Stack during containment isolation conditions with a minimum release of radioactive material to the environs. The Standby Gas Treatment System can also be used during normal power operation when the Reactor Building H&V system is out of service or instead of the Reactor Building H&V system during cold weather without the heating system in service. Cold weather operation of the Reactor Building H&V system when the heating system is out of service may result in unacceptably cold temperature in the Reactor Building. Operation of the SGTS in this instance minimizes heat loss. Operation of the SGTS in lieu of the Reactor Building H&V system ensures the building is maintained at negative pressure to minimize ground level release of radioactivity. Provisions are also made to direct the drywell and torus atmospheres through the Standby Gas Treatment System, if desired.

The drywell air inlet purge valves (V-28-42 and V-28-43) are part of the secondary containment boundary and, as such, must be able to serve as the primary containment during periods when the primary containment is open. These valves are normally closed during plant operations, and are opened during plant shutdown conditions or drywell deinerting. Closure of the purge valves will automatically occur on Reactor Building vent exhaust, fuel pool area or equipment hatch high radiation, high drywell pressure or reactor water level LO-LO signal, thus isolating the secondary containment. The air inlet purge valves are also closed automatically upon manual initiation of the Standby Gas Treatment System. The valves are also closed automatically if Reactor Building pressure is greater than 1" WG, high ductwork temperature is detected, or valves V-28-21 or –22 are not full open. The isolation signal interlock to the drywell air inlet purge valves will cause the valves to close or be prevented from opening, which will minimize a ground level release in the event of an abnormal plant condition.

6.2.3.4 Tests and Inspections

The Reactor Building leakage rate can be tested through building isolation and operation of the Standby Gas Treatment System.

The Reactor Building inleakage is tested as follows:

a. Each refueling outage prior to refueling

b. When an operation or event brings Reactor Building leakage integrity under question

c. At least once per operating cycle not to exceed 20 months by maintaining a 1/4 inch of water vacuum under calm wind conditions.

Automatic operations of isolation valves and the Standby Gas Treatment System are checked during each scheduled refueling outage.

6.2.3.5 Instrumentation Requirements

A discussion of the SGTS initiation logic is contained in Section 7.3 and Subsection 6.5.1.

6.2.4 Containment Isolation System

6.2.4.1 Primary Containment Penetrations

The Primary Containment is penetrated at several locations by piping, instrument lines, ventilation ducts, and electrical leads, see Drawing GE237E726. Penetrations are also provided for personnel access and for refueling.

a. Pipe Penetrations
Two general types of pipe penetrations are provided - those which must accommodate thermal movement, and those which experience relatively little thermal stress.

The piping penetrations for which movement provisions are made are the high temperature lines, such as the steam lines and other reactor system lines. A penetration of this type is shown in Figure 6.2-22. The penetration sleeve passes through the concrete and is welded to the primary containment vessel. The process line which passes through the penetration is free to move axially, and a bellows expansion joint is provided to accommodate the movement. A guard pipe immediately surrounds the process line and is designed to protect the bellows and maintain the penetration seal should the process line fail within the penetration. Insulation and air gaps are provided to reduce thermal stress. A seal arrangement is also provided to permit periodic leakage testing of these penetrations at a pressure equal to the primary containment design pressure.

The design of the penetration takes into account the simultaneous stresses associated with normal thermal expansion, live and dead loads, seismic loads, and loads associated with a Loss-of-Coolant Accident within the drywell. For these conditions the resultant stresses in the pipe and penetration components do not exceed the code allowable design stress. For failures of the steam pipe taken at random, the design takes into account the loadings given above in addition to the jet force loadings resulting from the failure. The resultant stresses in the pipe and penetration for this condition shall not exceed the ASME Code Section III, Subsection NF for Class MC Vessels experiencing service level "D" operating conditions.

The cold piping and ventilation duct penetrations are welded directly to the sleeves (see Figures 6.2-23 and 6.2-24). Bellows and guard pipes are not necessary in this design, since the thermal stresses are small and are accounted for in the design of the weld joints.

b. Electrical Penetrations

The electrical penetrations include electrical power, signal and instrument leads. Typical penetrations are shown in Figures 6.2-25 and 6.2-26. Two types of electrical penetrations are shown, basically they are of the same design. The penetrating sleeve is welded to the primary containment vessel, and the flanged ends are bolted and sealed with a soft gasket material. A bonding resin is utilized in the seals where the cable emerges from the flange. This arrangement provides a leak-tight configuration which is leak tested after installation, and provides a means for periodic leakage testing at the drywell design pressure. A third type is used for some power cables. This type is comprised of a seal plate with bushings on both sides. The penetration is affixed to the Drywell Penetration by means of a flanged joint. The flange has an aperture seal that allows leak testing to be performed by pressurizing between the seals and allows pressurization of the seal at the bushings. As with the other two types, this arrangement provides a leak-tight configuration that is leak tested after installation, and provides a means for periodic leakage testing at the drywell design pressure.

c. Personnel and Equipment Access

One personnel lock is provided for access to the drywell. The lock has two gasketed doors in series, and the doors are designed and constructed to withstand the drywell design pressure. The doors are mechanically interlocked to ensure that at least one door is locked at times when primary containment is required. The locking mechanisms are designed so that a tight seal will be maintained when the doors are subjected to either internal or external pressure. The double seals on this access opening allow leakage testing without pressurizing the containment system.

The personnel lock is also the equipment access hatch. The drywell head is also sealed with double seals, which can be leakage tested without pressurizing the containment. These hatches are bolted in place.

d. Access to the Torus

Access to the torus is provided at two locations from the Reactor Building. These consist of three foot diameter shielded manhole entrances with double gasketed bolted covers connected to the chamber by 3 foot diameter steel pipes. The access ports are bolted closed when Primary Containment is required, and are opened only when the reactor coolant temperature is below 212°F and Primary Containment is not required.

e. Access for Refueling

The top portion of the drywell vessel is removed during refueling operations. The head is held in place by bolts and is sealed with a double seal arrangement. The head is bolted closed when primary containment is required, and will be opened only when the reactor coolant temperature is below 212°F and Primary Containment is not required.

The double seal on the head flange provides a method for determining the leak tightness after the drywell head has been replaced.

6.2.4.2 Design Evaluation of Containment Penetrations

In order to minimize postaccident containment leakage, the containment penetrations are designed to withstand the normal environmental conditions during plant operation, and to retain their integrity during and following postulated accidents.

Pipe lines which penetrate or open into the containment shell, and which are capable of exerting a reaction force due to line thermal expansion or containment movement which cannot be restrained by the containment shell are provided with a bellows expansion seal. If necessary these lines are anchored outside the Primary Containment to limit the movement of the line relative to the containment. The bellows accommodates the relative movement between the pipe and the containment shell. This design assures integrity of the flexing penetration during plant operation.

Pipe lines which penetrate the Primary Containment where the reactive forces can be restrained by the containment shell are provided with full strength attachment welds between the pipe and the containment shell. These penetrations are designed for long term integrity without the use of a bellows seal.

Electrical penetrations require a special design to achieve zero leakage because of the design restriction imposed by creepage of electrical insulation.

A personnel access lock is provided with interlocked double doors so that access may be made to the containment while the reactor primary system is pressurized. Double doors are provided to assure that containment integrity is effective during access.

Access hatches are sealed in place, using flexible double seals or gaskets to assure leaktightness. These openings are closed at all times when containment integrity is required.

Inspection and surveillance provides additional assurance of integrity and functional performance of the penetrations. For this reason, provisions are made to individually leak test all electrical penetrations (except spares), the personnel access lock, and the access hatches. This can be accomplished without pressurizing the entire containment system. Monitoring the gross leakage of the primary containment provides further indication of the leaktightness of the penetrations. This is accomplished by reviewing the usage of Nitrogen for containment inerting makeup during plant operation.

6.2.4.3 Tests and Inspections of Penetrations

With the exception of the pipe penetrations which are welded directly to the primary containment shell and the main steam penetrations, it is possible to leak test individual containment penetrations without pressurizing the entire Primary Containment system. Leak testing may be accomplished by pressurizing the penetration between the double seals utilizing the pressure tap. Testable penetrations and double "O" ring sealed penetrations will be leak rate tested at a frequency established by the Primary Containment Leakage Rate Testing Program established to implement 10 CFR 50 Appendix J, Option B. The double "O" ring sealed penetrations will also be tested whenever they are broken and remade. This test frequency, along with the containment leak rate tests and the continuous leakage monitoring, is adequate to allow detection of leakage trends in the bellows and sealing compound seals.

The criteria for the allowable leakage rates are specified in the Technical Specifications.

Leakage through valves installed in pipe lines which open into the containment can be detected by pressurizing between pairs of containment isolation valves. Leakage through valves installed in pipe lines that connect to the RCS may be determined when the RCS is pressurized and the containment isolation valves are closed.

6.2.4.4 Isolation Valves

Table 6.2-12 is a listing of the isolation valves used in the piping which penetrate the Primary Containment. The isolation valve arrangements are shown in Figures 6.2-39 through 6.2-57. The table indicates the number and service of the valves, the motive power which actuates the valve, and the closure time of the valve.

Instrumentation piping connecting to the RCS which leaves the primary containment is dead ended at instruments located in the Reactor Building. These lines are provided with excess flow check valves and manual isolation valves, operable from the Reactor Building.

Each motor operated valve is provided with limit switches which indicate whether the valves are either open or closed. Each motor operated valve is capable of being actuated from the Control Room. Failure modes are shown in Table 6.2-12.

Isolation valve closure time is such that for any design basis break coolant loss is restricted so that the core would not be uncovered.

The functional performance of valves, sensors, and other automatic devices essential to the isolation of the containment can be tested. Such tests include demonstration of proper working conditions, correct set point of sensors, proper speed of response, and operability of fail safe features. Test taps between double isolation valves allow leakage testing by pressurization of the enclosed space. The test tap also allows leakage monitoring of the inboard valve during reactor system hydrostatic tests. For lines which are open to the containment, the inboard valve leakage is monitored when the containment is pressurized.

6.2.4.5 Design Evaluation of Isolation Valves

Since a rupture of a large line penetrating the containment shell and connecting to the Reactor Coolant System may be postulated to take place outside the Primary Containment, isolation valves are required to close automatically on various indications of reactor coolant loss. Additional reliability is added by a second valve, located outboard of the containment and as close as practical to it. This second valve also closes automatically if the inboard valve is normally open during operation. The two valves in series are provided with independent power sources. The piping and valving arrangement specified above requires three simultaneous independent failures before the reactor coolant could discharge in an uncontrolled manner beyond the primary containment.

The exceptions to this arrangement for lines connecting directly to the Primary Containment or RCS, are made only in the cases where it leads to a less desirable situation because of required operation or maintenance of the system in which the valves are located. Where, for example, the two isolation valves are located outside the Primary Containment, special attention is given to assure that the piping to the isolation valves has an integrity at least equal to that of the containment.

It is not necessary nor desirable that every isolation valve close simultaneously with a common isolation signal. For example, if a process pipe were to rupture in the drywell, it would be important to close all lines which are open to the drywell, and some effluent process lines such as the main steam lines. However, under these conditions, it is essential that the Containment and Core Spray Systems be operable. For this reason, specific actuation signals are utilized for isolation of the various process and safeguards systems.

Isolation valves must be closed before significant amounts of fission products are released from the reactor core under Design Basis Accident conditions. Because the amount of radioactive materials in the reactor coolant is small, a sufficient limitation of fission product release will be accomplished if the isolation valves are closed before fuel cladding fails. The valve closure times listed in Table 6.2-12 are adequate to limit the water loss from any pipe break to less than that lost from a design basis steam line rupture.

Valves, sensors, and other automatic devices essential to the isolation of the Primary Containment are provided with means for periodically testing the functional performance of the equipment. Such tests are necessary to provide reasonable assurance that the containment isolation devices will perform as required when called upon to do so.

6.2.4.6 Tests and Inspections of Isolation Valves

The leak tightness of the valves was demonstrated at the time the RCS was initially hydrostatically tested. Periodic tests are made to determine the continued leak tightness of the valves. This is accomplished each time the primary system is retested hydrostatically.

A functional performance test is periodically performed on all isolation valves (see the Technical Specifications). The valves are actuated by manual control and by simulated signals induced through the valve signal sensors. The closing times for specific containment isolation valves are listed in Table 6.2-12. This test is to be performed during each scheduled refueling outage.

Provisions are also made for determining leak tightness of individual valves. The valves are tested by the Primary Containment Leakage Rate Testing Program and plant Technical Specifications, which implement 10 CFR 50, Appendix J, Option B, as modified by approved exemptions.

6.2.5 Combustible Gas Control in Containment

Containment Inerting System maintains an atmosphere of nitrogen gas within the drywell and torus air space during power operations. The oxygen level is normally monitored by the Drywell and Torus Oxygen Monitoring System, and is maintained below 4% by volume. This precludes any energy releases from a hydrogen-oxygen reaction following a postulated LOCA in which hydrogen is produced by a metal-water reaction. Figure 6.2-27 shows that the maximum metal-water reaction does not exceed about 0.6 percent, which is below the metal-water reaction flammability limit by a factor of about two.

During accident conditions the containment is monitored by the Hydrogen/Oxygen Monitoring System

6.2.5.1 Design Bases

The design bases for combustible gas control inside containment are as follows:

a. To monitor oxygen levels inside containment during normal operations.

b. To maintain oxygen levels inside containment below 4% by volume during normal operations.

c. To ensure that hydrogen-oxygen combustion following a LOCA is prevented.

d. To permit containment isolation when required.

6.2.5.2 System Design

6.2.5.2.1 Containment Inerting System

The Containment Inerting System (Figure 6.2-28) performs two general functions: purging and makeup. Following a break of containment integrity, nitrogen must be introduced to displace oxygen from the free volume in the Primary Containment. In addition, during reactor power operation, nitrogen is required to maintain a low oxygen concentration in the Primary Containment, as air is introduced as a result of leakage and temperature changes.

Basically, the inerting system consists of a liquid nitrogen supply, a vaporizer, throttling or pressure reducing valves and controllers, makeup and venting valves and controls, system isolation valves, oxygen analyzers, nitrogen flowmeters, and containment pressure and temperature instrumentation.

Nitrogen for purge and makeup is added to the drywell through an 18 inch line at El. 86'-0" (high up in the neck of the drywell), and to the torus through a 20 inch line at El. 10'-2". The same connections are used for purging the Primary Containment with air in preparation for refueling, or prior to entry for maintenance. All gases are exhausted to the Stack through the Primary Containment Vent System during the purge operations. The drywell exhausts through an 18 inch line at El. 21'-3"; the torus exhausts through a 12 inch line at El. 14'-0", about 180 degrees from the purge inlet line. Purge supply valves V-23-13 and V-23-15 are manual control valves for controlling the pressure in the containment vessel. In each purge supply line, the two valves in series are normally closed during reactor operation to isolate the containment vessel from the purge line. All valves are remotely operated, and are interlocked to prevent operation during containment isolation conditions.

The makeup to the drywell and torus is manual. In each makeup supply line the two valves in series are normally closed during reactor operation to isolate the containment vessel from the makeup line, except during makeup. All valves are remotely operated, and all valves are automatically closed by the Reactor Protection System upon indications of high drywell pressure or reactor low-low water level.

During reactor operation, the nitrogen atmosphere is isolated within the drywell and recirculated by the drywell cooler blowers. Design requirements of the system are listed in Table 6.2-13.

6.2.5.2.1.1 Nitrogen Supply

The permanent onsite liquid nitrogen station for makeup is a commercially standard 9000 gallon double shell steel tank unit with vacuum insulation to reduce the loss of nitrogen gas through heat leakage. The inner tank is constructed of Type 304 stainless steel according to the ASME Unfired Pressure Vessel Code.

The makeup station includes an integral ambient air vaporizer. Supplementary electric heaters start and stop automatically to maintain the outlet nitrogen gas temperature at not less than about 50°F in cold weather.

An adjustable line pressure regulator from the vaporizer maintains the outlet pressure at about 60 psig. Other instrumentation provided with the makeup nitrogen supply includes tank level indication and a low level annunciator, tank pressure and temperature, and pressure and flow in the makeup header.

6.2.5.2.1.2 Air Purge and Exhaust

Air for purging the drywell is supplied from the Reactor Building Ventilation Supply System, through valves V-27-3 and V-27-4. These are air operated valves, remotely controlled, and interlocked to close automatically on a containment isolation signal. Air accumulators are provided as a backup air source, to ensure closure of the valves on a loss of control air signal. These valves are normally closed during reactor operation with the drywell inerted. The air purge line joins the nitrogen supply line and enters the drywell through a common penetration.

Air for purging the torus is supplied directly from the Reactor Building atmosphere when a slight vacuum is established by exhausting the torus to the Stack. The supply is controlled by the torus vacuum breaker valves. There are two sets of vacuum breaker valves in parallel, and in each set there is a check valve and an air operated butterfly valve in series. Butterfly valves V-26-16 and V-26-18 are automatically opened by differential pressure switches to prevent excessive vacuum in the torus; and these valves will open on loss of air pressure. The two valves are normally closed with positive or atmospheric pressure in the torus; and effect isolation of the chamber when required. The nitrogen supply line to the torus joins this vacuum breaker line and enters the torus through a common penetration.

Purge exhaust from the containment is drawn to the Stack through either the Reactor Building ventilation exhaust fan or the Standby Gas Treatment System fan, depending upon the radioactivity levels of the exhaust gas.

When the exhaust is vented directly to the Stack without gas treatment, torus flow is through valves V-28-17 and V-28-18. When the exhaust is vented to the stack without gas treatment, the drywell flow is through valves V-27-1 and V-27-2.

These are air operated butterfly valves, remotely controlled, and normally closed except during purge operation. Air accumulators are provided, as a backup air source, to ensure closure of the valves on a loss of control air signal. These valves are interlocked to close automatically upon high drywell pressure or low-low reactor water level. In addition, drywell high radiation signals are used to initiate the drywell ventilation isolation.

Valves V-27-1, V-27-2, V-27-3, V-27-4, V-28-17 and V-28-18 will close on a drywell high radiation signal through two redundant high radiation isolation logic channels. This provides an added assurance in preventing offsite doses from exceeding 10CFR100 limits under accident conditions.

When the exhaust is routed through the SGTS, the large valves (except V-28-18) are left closed and 2 inch bypass valves V-23-21, V-23-22 and V-28-47 limit the flow to less than 2600 cfm and reduce the drywell pressure to prevent damage to the SGTS filters. These 2 inch valves are air operated globe valves, remotely controlled, and normally closed except during purge operation. These valves are interlocked to close on a containment isolation signal. The control logic includes a bypass to manually control opening the valves after a containment isolation has been initiated.

Whenever torus pressure exceeds drywell pressure by 0.5 psi the drywell vacuum breaker valves (which are check valves) will automatically open to equalize these pressures.

Valves V-23-21 and V-23-22 are also used for venting of the drywell. They are manually opened when the drywell pressure increases to 1.3 psig. They vent either to the normal building exhaust fan or to the SGTS depending upon other valve positions. The vent connection accommodates pressure increase during normal drywell heatup at station startup.

6.2.5.2.2 Oxygen Monitoring System

The Drywell and Torus atmosphere is constantly monitored for oxygen concentrations to ensure concentrations of oxygen do not exceed 4% and approach flammability limits. One analyzer is used for each of the Drywell and Torus areas. Both of these analyzers are located in the Reactor elevation 23’6” along with the sample pump, associated components and switches to allow for local operation of the Oxygen Monitoring Systems. Remote instrumentation for each of the monitoring systems is located in the Control Room on panel 12XR. In addition, this panel contains indicators, a recorder, pump controls switches and trouble alarms. The permissives for the isolation valves from the Torus and Drywell areas are interlocked with the RPS (Reactor Protection System).

When the containment is required to be inert, the containment atmosphere is monitored for oxygen concentrations. The Drywell and Torus oxygen analyzers are located at El. 23’6”. Remote instrumentation for each system is located on Control Room panel 12XR. Outputs from the analyzers supply separate pens of 0-10% range recorder. A Control Room alarm will annunciate on high oxygen levels. Upon primary containment isolation, the sample pumps will trip and the isolation valves will close. If the analyzers are out of service when an inert atmosphere is required, then the oxygen concentration will be sampled weekly.

6.2.5.3 Design Evaluation

Analysis of the postulated Loss-of-Coolant Accident has shown that operation of either Core Spray System Loop will maintain continuity of core cooling, and that none of the fuel melts or reaches redistribution temperature. A portion of the fuel cladding perforates and releases the fission products stored in the fuel rod gas plenums; but because of the moderate cladding temperature, the extent of the metal-water reaction would be only approximately 0.08 percent of the cladding and channel material.

The containment pressure transient results from stored energy in the fuel, decay energy, and energy from the metal-water reaction, including energy from the hydrogen-oxygen reaction. The Containment Spray System reduces the pressure and temperature in the containment and removes the energy released.

The hydrogen from the 0.08 percent metal-water reaction, if mixed with the atmosphere in the primary containment, would result in a hydrogen concentration of approximately 0.4 percent. This concentration is significantly below the concentration at which hydrogen can be ignited in air. However, to avoid the possibility of an energy release within the Primary Containment from a hydrogen-oxygen reaction under conditions more severe than can currently be foreseen, the Containment Inerting System is provided. The Containment Inerting System controls the oxygen concentration in the primary containment during reactor operation and subsequent to the postulated Loss-of-Coolant Accident.

Hydrogen will not react with oxygen (will not ignite or explode) in a gas mixture containing less than approximately 5 percent oxygen. The exact minimum oxygen concentration which will react with hydrogen is dependent upon temperature, pressure, and other gas constituents. The inerting system will reduce oxygen concentration to approximately 3.0 percent by volume prior to termination of purging (before reactor operation), and will maintain oxygen concentration at less than 4.0 percent by volume by means of the makeup system during operation.

Following a refueling outage, after the drywell head has been installed and the personnel locks are closed, the air in the containment must be purged and replaced with nitrogen within 24 hours after placing the reactor in the RUN mode, as required by Technical Specification 3.5.A.6. After reactor operation and before refueling or maintenance, the nitrogen atmosphere in the containment must be purged and replaced with air. All gases must be exhausted to the Stack during purging. Depending upon the radioactivity levels of the gases, the exhaust may be directed to the standby gas treatment system for filtering or may be discharged to the Stack through the Reactor Building exhaust fan. Normally the air purged from the containment following a refueling shutdown is not radioactive enough to require filtering. The nitrogen purged following reactor operation normally does not require filtering. Table 6.2-14 shows valve positions for purging through the normal exhaust fan or through the Standby Gas Treatment System. The operation of the exhaust fan will maintain a slight vacuum in the containment during purge.

6.2.5.3.1 Nitrogen Purging

The current nitrogen purge system utilizing a fan powered ambient vaporizer with electric trim heater assist can maintain a design flow of 100,000 scfh. The temperature (as read in the Control Room) of the gaseous nitrogen must be within the allowable containment operating range of about 50°F to about 150°F. The containment pressure must be maintained between approximately negative 0.25 in. W.G. and approximately positive 1.3 psig. The purging operation starts by aligning the Nitrogen station and utizilizing its pressure control features to deliver Nitrogen to Primary Containment. The operators then establish flow into the Drywell by opening the Drywell purge valves, and allow it to continue until the desired Oxygen concentration and pressure is achieved. The Drywell purge valves will be closed, and the Torus will then be inerted until the desired Oxygen concentration and pressure is achieved. The Drywell and Torus can not be inerted at the same time because simultaneous opening of the vent and purge valves will create a leakage path to bypass the Torus to Drywell vacuum breakers. This process is repeated as necessary to maintain the desired concentration and pressure in both vessels.

At the completion of the purge, the oxygen content in both the drywell and the torus must be less than 4.0 percent by volume, and the pressure must be between 0.25 psig and 1.0 psig. Then all purge and exhaust valves are closed to isolate the containment.

6.2.5.3.2 Air Purging

Air will be purged into the torus by pulling a vacuum with the fan and valve arrangement previously outlined. Air will then be drawn into the torus through the torus vacuum breaker valves. Proper opening of vacuum breaker valves V-26-16 and V-26-18 should be ascertained from the valve position indicating lights. Purging is continued until the oxygen concentration is equal to that in air, and until the radioactivity of the atmosphere is sufficiently low for the required maintenance. Air will be purged into the drywell by pulling a vacuum with the exhaust fan, and opening air supply valves from the Reactor Building Ventilation System (V-27-3, V-27-4, V-28-43 and V-28-42). Purging is continued until the oxygen and radioactivity content are within acceptable range.

6.2.5.3.3 Nitrogen Makeup

The nitrogen makeup is manual on pressure control and manual on oxygen concentration. Valves V-23-17 and V-23-18 are opened manually on low pressure to add nitrogen gas to the drywell. With high oxygen concentration makeup will be performed manually either to the drywell or torus. Torus pressure control is manual.

The makeup system instrumentation continuously measures the temperature and flow rate of the makeup nitrogen entering the primary containment in order to monitor for primary containment gross leakage.

The oxygen concentration in the drywell and torus are continuously recorded in the Control Room and shall be maintained less than 4 percent by volume.

6.2.5.3.4 Containment Inerting System Isolation Valve Control

All remotely operable isolation valves in the Containment Inerting System are automatically closed by the Reactor Protection System upon indications of high drywell pressure or low-low reactor water level. The isolation valve control logic performs the following tasks:

a. Isolates on receipt of isolation signal.

b. Provides a means to reduce pressure in torus and drywell in the postaccident phase.

c. Prevents the simultaneous opening of both the torus and drywell vent valves for pressure reduction to prevent excessive flow damage to the Standby Gas Treatment System.

d. Prevents the opening of all other isolation valves (except exhaust vents in b. above) until the trip signal has been cleared.

e. Prevents automatic venting of the drywell during the early stages of an accident condition.

6.2.5.3.5 Post LOCA Oxygen Sources in Containment

Three potential sources of oxygen within the containment have been considered:

a. Oxygen Entrained in the Coolant

   Coolant supplied for the containment is from two systems:

   The Core Spray System draws water from the torus where approximately 83,400 cu ft of water is stored. The torus, as well as the drywell is inerted and the total amount of oxygen dissolved is less than 15 lbs (180 cu ft). In a total volume of over 300,000 cu ft this would not raise the oxygen content of the atmosphere by an appreciable amount if it were all released.

The Containment Spray System also draws water from the torus. Therefore, it is not an additional source of oxygen.

b. Leakage from Air Supply Systems

The instrument air supply inside the drywell (during operation, when the containment is inerted) is from a nitrogen supply system with instrument air backup. Even in the event of loss of nitrogen supply, the air infiltration would be negligible in comparison to the drywell's volume.

c. Vacuum Breakers

The vacuum breakers are not considered an oxygen source after a LOCA when the primary containment is pressurized from the loss of coolant.

The quantity of oxygen attributable to the above sources is negligible.

6.2.5.3.6 Post LOCA Oxygen Concentrations in Containment

Recent evaluations of oxygen concentrations in BWR containments following a LOCA (Reference 10 and 11) have shown that:

The oxygen generation rate in boiling water is less than 0.1 molecule per 100 eV.

Assuming an initial oxygen concentration of 4% and the worst-case hydrogen generation accident, the containment oxygen concentration will not increase to exceed 5% in 1000 days.

The OCNGS Technical Specifications establish a maximum concentration of 4% during normal operation. This 4% oxygen concentration limit was established by analysis, as presented in Reference 1. The oxygen concentration is continuously recorded and a high oxygen concentration (3.5%) is annunciated in the Control Room. Control of combustible gas is based upon controlling oxygen content. Purging of the containment following a LOCA is not required within the first 30 days following the accident.

6.2.5.3.7 Post LOCA Hydrogen Stratification

During the first few hundred seconds of a LOCA, the violence of the steam blowdown and subsequent condensation is more than adequate to assure complete mixing of the gases. In addition, the quantity of radiolytic gases produced during this time is also a negligibly small amount. Later in the course of the LOCA, when more or less stable operation has been established, the possibility of local concentration gradients in radiolytic gases must be considered. In the generation of radiolytic gases, the hydrogen and oxygen formed would initially exist in the water in supersaturated solution being released to the air space by diffusion from the water surface or by coalescing into small bubbles at some unknown rate. If the water is heated, as it might in passing through the core region, the temperature of the released radiolytic gases will be at the temperature of the water and hence slightly above the ambient atmospheric temperature.

In the theoretical consideration of the dispersion of radiolytic gases so produced, it might be considered that diffusion theories of the ordinary Fickian type would be applicable. A solution of the Fickian diffusion equation in one dimension for a slab source would be of the following form:

$$C(x) = C_0 + G(H_2)(t/D)^{1/2}(1\text{-erf}(x/2(Dt)^{1/2}),$$

where C is the concentration at the x position, t is the time, D is the effective diffusion coefficient and erf is the Gaussian error integral.

However, experimental data are not available to unambiguously determine an appropriate value of the diffusion coefficient, D, that would be appropriate under LOCA conditions. Dispersion by molecular diffusion alone from the surface of the water under extremely stagnant atmospheric conditions would not be adequate to completely distribute the radiolytic gases throughout the entire containment atmosphere, and local concentration gradients could exist in the vicinity of the source. Actual diffusion coefficients would be expected to be many orders of magnitude greater than that of molecular diffusion alone. By analogy with dispersion models usually employed in assessing somewhat related problems of the meteorological diffusion of radioactive clouds in the atmosphere, diffusion coefficients could be derived that would indicate the very rapid dispersion of radiolytic gases into the containment atmosphere.

Pasquil has qualitatively discussed the effect of buoyancy with free convective motion resulting from unstable vertical density gradients. For buoyant forces to exist, the radiolytic hydrogen and oxygen released from the water would necessarily be at a temperature higher than that of the surrounding atmosphere. Under these conditions, the thermal gradients and circulating vertical motion augmented by local turbulent eddies around structural components in the containment would tend to increase the rate of mixing. On the basis of diffusion theory considerations, and the qualitative effect of buoyant forces on free convective motion with thermal gradients, it is concluded that there is very little likelihood for the existence of "pockets" of radiolytic oxygen in concentrations significantly above that of the surrounding atmosphere. Under the influence of natural dispersion phenomena only, any "pockets" of radiolytic hydrogen and oxygen that might be formed would be expected to be highly localized and transient in nature.

In addition to the natural dispersion phenomena, there are at least two other mechanisms to increase the assurance that combustible gases in the containment will be uniformly distributed:

a. Water being pumped from the torus to cool the reactor core would spill from the end of the broken pipe(s), creating local turbulence that would augment the mixing of the radiolytic gases being released from the water.

b. There are two Containment Spray System loops. Operation of either or both of these would further increase mixing of the containment gases, both by the turbulence created as well as by thermal convection resulting from atmospheric cooling.

It is therefore concluded that there are both natural mechanisms and positive means of mixing which will assure uniform concentrations of combustible gases in the drywell containment atmosphere following a Loss-of-Coolant Accident. In addition, since the concentration of hydrogen available for leakage into the Reactor Building would not exceed 4% over the durations of the accident, and since there are no possible mechanisms for enriching hydrogen above the 4% available in the leakage gas, the leakage of hydrogen from the drywell into the Reactor Building would not create a potentially hazardous situation.

6.2.5.4 Tests and Inspections

The Containment Inerting System relief valves are tested periodically to assure that the containment will not be overpressurized. The pressure regulator is also checked. By measuring the nitrogen makeup rate, the leak rate of the Primary Containment may be determined.

6.2.5.5 Instrumentation Requirements

The Containment Inerting System is controlled from the Control Room. Instrumentation for this system is also at that location.

Instrumentation and control components are identified in Table 6.2-16. There are OPEN-CLOSE position indicators for all remotely operated nitrogen, air, and vent valves.

6.2.6 Containment Leakage Testing

General Design Criteria 52, 53 and 54 require that the containment, containment penetrations, and containment isolation barriers be designed to permit periodic leakage rate testing.

Appendix J, to 10CFR Part 50 "Primary Reactor Containment Leakage Testing for Water Cooled Power Reactors," specifies the leakage testing requirements for the containment, containment penetrations, and containment isolation barriers. The Oyster Creek containment leakage test program uses Option B for both integrated and local leak rate testing (as modified by approved exemptions), and the test methods of ANSI/AN S-56.8-1994, as described in the Primary Containment Leakage Rate Testing Program. The Program is in accordance with Regulatory Guide 1.163, "Performance Based Containment Leak Test Program," with any exceptions specified in plant Technical Specifications.

The Oyster Creek Nuclear Generating Station is an early vintage boiling water reactor plant (General Electric BWR-2), and, as such, has fluid systems and a Primary Containment System with physical and functional designs that predate 10CFR Part 50, Appendixes A and J, by several years. An evaluation of the Oyster Creek Primary Containment design features against the Commission's containment design criteria, Group V of Appendix A - General Design Criteria for Nuclear Power Plants, is presented in Section 3.1.

For the latest Leak Rate Testing Program information see the Technical Specifications and Primary Containment Leakage Rate Testing Program.

6.2.6.1 Containment Integrated Leak Rate Test

Prior to installation of equipment and penetrations, the drywell, torus, and vent pipes were initially leak tested using soapsuds for detection of leaks. Leaks found in this test were sealed (refer to Section 14.2 for a description of the test program).

In conformance with the ASME Code, the drywell and torus were pressure tested separately at 1.15 times the respective design pressures prior to installation of the penetrations.

The indicated soap checks were performed on gasketed closures, penetration bellows, and all penetration welds which have been made since the vessel acceptance tests. These acceptance tests were performed by the containment vessel fabricator and consisted of 115% overload tests at 71.3 psig for the drywell and 40.25 psig for the torus as well as leak rate tests at 62 psig for the drywell and 35 psig for the torus (both dry and wet leak rate tests were run on the torus).

A pressure proof test at 62 psig of the drywell vents and penetrations was performed after installation of the penetrations; and a pressure proof test of the combined system of drywell and torus was performed at 35 psig to demonstrate the combined structural integrity of the system. A 24 hour integrated leak test was done after the structural integrity test. Both this leak test and subsequent periodic leak tests were done at a minimum 20 psig but not greater than 35.0 psig. The leakage rates determined by these tests were extrapolated to the 35 psig. With the adoption of 10 CFR 50 Appendix J, Option B, effective September 3, 1996, integrated leak rate tests will be performed at 35.0 psig in accordance with the Primary Containment Leakage Rate Testing Program.

The results of the preoperational drywell test indicated a leakage expressed as a negative number of (-)0.078%. The dry torus leakage as a negative number was (-)0.041%. The "wet" torus leakage as a negative number was (-)0.067%.

Leakage rate tests are subsequently conducted at frequencies based upon the Commission's guide for containment leak rate testing. The current Technical Specifications define the actual frequency and acceptance criteria.

The allowable primary containment leakage rate is established to limit the potential offsite doses following a design Loss-of-Coolant Accident. The maximum whole body dose was calculated to be 5×10^{-6} of the 10CFR100 guideline dose and the maximum thyroid dose was calculated to be 2.5×10^{-7} of the 10CFR100 guideline dose. These calculations were based upon a primary containment leakage rate of 0.5%/day at 35 psig, and an emergency gas treatment system filter efficiency for halogens of 99%. With the more reasonably measured primary containment leakage rate of 5%/day at 33 psig, and filter efficiency for halogens of 90%, the maximum whole body dose would still be 5×10^{-4} of 10CFR100 and the maximum thyroid dose would still be 2.5×10^{-5} of 10CFR100. Thus, filter efficiencies (90%) and leakage rates (5%/day at 33 psig) are specified as minimum conditions that must be met during the testing of these systems. (See Figure 6.2-37)

It is possible to periodically pressurize and test the leak tightness of the portions of the Containment and Core Spray System which are external to the primary containment.

6.2.6.2 Containment Penetrations Leakage Rate Tests

With the exception of the pipe penetrations which are welded directly to the primary containment shell and the main steam penetrations, it is possible to leak test individual containment penetrations without pressurizing the entire containment system. Leak testing to 35 psig is accomplished by pressurizing the penetration with nitrogen or air between the double seals utilizing the pressure tap.

There are two ways for determining the leakage rate. The primary way, the mass flow method, pressurizes the penetration and measures the amount of makeup required to maintain the test pressure. A second method records pressure data to determine the pressure decay rate for each testable penetration. The leakage rate is then calculated, using the perfect gas law, the measured pressure decay, and the calculated volume of the space pressurized. The penetrations which are individually tested are identified in the Station Procedure for Local Leak Rate Tests. Tests are performed as described in the Primary Containment Leakage Rate Testing Program.

The total number of penetrations is 471, including 280 one inch control rod hydraulic line penetrations, and 69 two and three inch sensing, instrument air, and spare penetrations.

No testing is required where the design integrity of the penetration can be shown to be equal to or greater than the drywell proper. Penetrations, where a pipe line or nozzle is welded directly to the drywell shell will be considered equal to the drywell proper provided that the design and fabrication of the pipe or nozzle meets all code requirements for penetration or nozzle reinforcement.

Table 6.2-17 lists the types of containment penetrations which do not have provisions for automatic isolation and indicates their operational status both during accident conditions and during the initial leak rate test.

Due to the rapid closure times for the containment isolation valves, the effect of "time to isolate" can be neglected and consequently is not included in the leak rate calculations during the initial test.

Leakage through valves installed in pipe lines which open into the containment can be detected by pressurizing between pairs of containment isolation valves. Leakage through valves installed in pipe lines that connect to the reactor primary system can be detected when the reactor primary system is pressurized with the containment isolation valves closed.

6.2.6.3 Containment Isolation Valves Leakage Rate Tests

The leak tightness of the valves was demonstrated at the time the reactor primary system was initially hydrostatically tested. The calculated leakage rate through these valves was added to the leakage rate of the testable penetrations, and was required to meet the limitations established for testable penetrations.

Containment Isolation Valves are tested as described in the Primary Containment Leakage Rate Testing Program.

6.2.6.4 Scheduling and Reporting of Period Tests

The Primary Containment Leakage Rate Testing Program specifies the frequency of testing and Technical Specifications specify the acceptance criteria for the Type "A" Primary Containment Integrated Leak Rate Test (PCILRT), and Type "B" and "C" Local Leak Rate Tests (LLRT).

6.2.7 Hardened Vent System

The Hardened Vent System (HVS) was installed as a result of the NRC Generic Letter 89-16. The HVS will permit a controlled depressurization of primary containment via Torus during severe accident sequences that involve loss of decay heat removal capability. The HVS will also provide a secondary path for venting the Drywell when the Torus vent path is unavailable.

6.2.7.1 Design Basis

The HVS is designed to perform the following functions beyond the Design Basis Accident:

1. Vent the pressure in primary containment in the event that long term decay heat removal capability is lost with heat input to the containment of 1% of rated thermal power and the containment pressure approaching Primary Containment Pressure Limit (PCPL).

2. Prevent the pressure in the primary containment from exceeding the PCPL and prevent core melt during TW sequence (TW sequence is defined as loss of decay heat cooling capability). Radiation dose rates resulting from operation of the HVS during TW sequence will not preclude system operability and accessibility.

3. Permit access to the Reactor Building (RB) to accomplish functions to mitigate this event (loss of decay heat cooling capability).

4. Maintain the pressure in the primary containment below the PCPL during the loss of decay heat cooling by repetitive venting operation as required.

5. Vent the primary containment on loss of Instrument Air System (IAS) for a maximum of 6 venting operations in 24 hours.

6. Vent the primary containment on a maximum containment pressure of 55 psig.

7. Monitor the releases and alert the Control Room operators of radioactive releases during venting operation.

8. Capable of withstanding the expected venting conditions associated with TW sequence without loss of functional capability.

6.2.7.2 System Design

6.2.7.2.1 System Arrangement

The hardened vent system utilizes the Nitrogen Purge isolation valves. Isolation valves V-23-13 and 14 and their air accumulators are located in the west side of the RB at elevation 75'3". Isolation valves V-23-15 and 16 and the common air accumulator are located in the southwest side of the RB at elevation 23'6". The control switches and status indication lights of these four (4) isolation valves are located in the Control Room. The pipe lines from the drywell isolation valve V-23-13 and the torus isolation valve V-23-15 join together at floor elevation 51'3" at the southside of the Reactor Building. From this location, the line runs along the east wall, down to elevation 14'9" and surfaces outside above the ground at the northeast corner of the RB.

The manual butterfly valve for the hardened vent system and the manual butterfly valve for the N_2 system are located at the northeast corner of the building. An 8' x 10' radiation shield wall is installed to protect the operators from radiation shine coming directly from the exposed valves and pipe. The manual butterfly valves are provided with "reach" rods that penetrate the radiation shield wall. At the other side of the shield wall, operating levers with locking device are provided for the operator to operate the valves. Labels indicating close/open rotational arrows are provided in the shield wall. A capped hose connection to allow hook up of portable N_2 tank in a truck is also provided at the same approximate location.

From the northeast corner, the vent pipe is routed along the east wall of the RB and along the south wall of the Railroad airlock. The stack penetration is provided with industrial quality rubber boot to seal the gap between the stack and the vent pipe and to allow for thermal movement of the pipe. The vent discharge is directed upward and ends at elevation 38' (approximately) inside the stack. The low point of the vent discharge inside the stack is provided with a 2" drain (2" weep hole) with no isolation valve so that continuous drainage of the condensate will take place.

The control switches and status indicating lights of the four (4) containment isolation valves are located in the Control Room. A two-position key-locked bypassed switch labeled "Normal" and "Bypassed" is provided in the Control Room to allow for these valves to be opened during a DBA under a controlled supervision. A bypass annunciator labeled "VENT/PURGE ISOL BYPASSED" is also provided in the Control Room to annunciate the bypass condition of these isolation valves.

The existing 1" drain valve V-23-263 inside the northeast corner of RB from the low point of the 8" N_2 line was retained and an air trap (automatic drain valve) was installed in series with the 1" manual drain valve (V-23-263). At the inlet and outlet connection of the air trap, test ports with 1/2" shut-off valve were provided in order to perform a periodic test for air leakage and drainage capability of the air trap. Downstream of the air trap, another 1" manual valve is installed to isolate the air trap during surveillance testing.

6.2.7.2.2 Operation

The vent path of the hardened vent system is from the primary containment, through the isolation valves V-23-13/14 or V-23-15/16, through the 8" N_2 purge line to the northeast corner of the RB, through the 10" vent pipe, to the stack where the RAGEMS probe is located and out of the stack located 368' above grade.

Startup

To start the venting operation, an operator will be dispatched to align the hardened vent system by closing the Nitrogen line butterfly valve (V-23-357) and opening the vent line butterfly valve (V-23-358). Once these valves are aligned manually, the venting operation can be performed by opening either the Drywell isolation valves V-23-13 and 14 or the Torus isolation valves V-23-15 and 16 from the Control Room.

Normal Operation

Depending on the scenario or conditions, venting operation can be repeated as long as instrument air is available. In the event of a loss of instrument air, venting operations can be performed for a maximum of 6 times. During every venting operation, the condensed steam will be drained automatically by the air trap (automatic drain valve). The primary vent path is through the Torus to take advantage of the scrubbing effects of the Torus water.

After venting operation, when a decision to inert is made, an operator will be dispatched to the shield wall to realign the nitrogen system. On loss of instrument air, the process of venting and inerting the primary containment may be repeated only for a maximum of 3 times (3 venting and 3 inerting operations).

In venting on TW sequence, inerting the primary containment is not required. However, the primary containment may be vented for several times. If this condition occurs and in order to avoid unnecessary exposure, the positions of the manual valves (V-23-357 and V-23-358) at the shield wall location may be left "as is" after the first venting operation.

Shutdown

Stopping or shutdown of the venting operation is performed by closing the air operated isolation valves V-23-13/14 or V-23-15/16. The positions of the manual butterfly valves (V-23-357/358) at the shield wall may be left "as is" if inerting operation is not planned.

Draining

During and after the first venting operation, the condensation inside the 10" pipe will flow towards the stack where the low point drain is located. The stack drain will be utilized to drain the condensate on the stack floor. The amount of condensed steam going to the stack drain is 3.1 gallons.

The condensation inside the 8" pipe is automatically drained by an air trap which is installed in series with the 1" manual drain valve (V-23-263). During normal inerting operation of the primary containment, the manual drain valve (V-23-263) is closed to provide positive shutoff and prevent leakage of N_2 to the building. After completion of the inerting operation, the 1" manual drain valve (V-23-263) is opened to align the air trap (Y-23-002) for automatic drainage of condensed steam during venting operation. The amount of condensate that will be drained at this location is also 3.1 gallons.

Infrequent Operation

DBA with a loss of decay heat cooling:

In the event that long-term decay heat removal capability is lost and the containment pressure approaching Primary Containment Pressure Limit (PCPL), venting the primary containment will prevent the primary containment pressure from exceeding the PCPL. Venting operation will be repeated as required to maintained the pressure below PCPL. In the event Instrument Air System (IAS) is lost at this condition, the air accumulators can provide a maximum of 6 venting operations in 24 hour period. Torus venting through V-23-15/16 is the preferred path for venting the primary containment in order to utilize the scrubbing effect of torus water.

Containment Flooding due to LOCA:

During a Design Basis Accident with a LOCA break at the bottom of the reactor, the containment is flooded to the top of active fuel (TAF). As the water rises, the pressure inside the Drywell increases. The weight of the water and the increasing pressure may exceed the PCPL. To prevent this, the Drywell is vented using V-23-13/14. Once the manual butterfly valves (V-23-257/358) are aligned, venting operation can be performed for several times (maximum of 6 on a loss of IAS).

6.2.7.3 Design Evaluation

Prior to venting, the Control Room operator will rely heavily upon the DW or Torus pressure transmitter in making the decision to vent. The pressure transmitters are qualified to 87.7 psia and 318°F at 100% RH. Since the expected venting (PCPL) pressure and temperature are 70 psia and 305°F respectively, the above instrumentation is expected to be operable. The decision whether to vent the Drywell or the Torus will depend upon the Torus water level. In the decision making, the Control Room operator will rely heavily on the Torus water level transmitter which is also qualified to 87.7 psia and 318°F at 100% RH. The preferred vent path is via the Torus to take advantage of the scrubbing effects of the torus water.

Once the hardened vent is utilized during a TW sequence, the radiation dose rates of the entire vent path of the hardened vent system is less than 1 mrem/hr. If the Torus is vented 24 hours after an event (scrubbed venting) with 20% fuel damage, the dose rate 40 feet away from the vent path is 550 mrem/hr. At the valve station, behind the radiation shield wall, the dose rate is 700 mrem/hr. The dose rate at the SGTS panel is approximately 1300 mrem/hr.

Continuous operation of the SGTS and RAGEMS is expected after a DBA. To ensure SGTS operability, the SGTS panels at the base of the stack are monitored from time to time. To quantify the releases if RAGEMS is not available, grab sample or replacement of filter cartridge will be performed in the RAGEMS building located at the base of the stack. With radiation dose rates indicated above and a maximum time of 30 minutes to monitor/operate the SGTS or replace the filter cartridge on the RAGEMS, the accessibility to the SGTS panels and to the RAGEMS building is maintained without shielding.

The existing radiation monitor (RAGEMS) is utilized by the hardened vent system to alert Control Room operators of radioactive releases during venting operation. The discharge of the vent is approximately at elevation 37'. The location of the existing isokinetic nozzle for the radiation monitor is at elevation 264' and the discharge of the stack is at elevation 368'. Since the difference in elevations between the vent discharge and the isokinetic nozzle is more than 10 diameters (Note that 10 diameters is required to attain a smooth flow), the exhaust from the vent pipe has ample distance to evenly distribute across the stack cross sectional area (16 feet ID at 264' elevation) prior to being sampled by the isokinetic nozzle. Based on the even distribution of exhaust gas, correct sampling will be obtained.

During venting operation coincident with the loss of offsite power, the flow to the stack is less than the design cut-off flow of 94,000 CFM of the RAGEMS. At this flow condition, the RAGEMS will go to automatic default mode using the low range monitor. Since there is no fuel damage when venting on TW sequence is performed, the radiation level of the releases will be within the range of the low range monitor.

In order to vent an energy equivalent to 1% of rated thermal power, the isolation valves had to be opened in the range of the 70° to 75° open position. When the valves are opened wider, the closure time of the isolation valves will be longer. An analysis was performed to determine the releases through the closing isolation valves during a DBA assuming a closure time of 60 seconds. The analysis indicated that the releases through the isolation valves is less than 2000 pound mass.

The HVS is only intended to be used for conditions beyond the design basis accident which is not evaluated in the SAR. Any releases during venting operation beyond the design basis accident will be consistent with the SECY-89-17 document that evaluated the venting operation during loss of containment cooling conditions. Any releases when venting will be consistent with the philosophy described in the existing Emergency Operating Procedure (2000-EMG-

3200). All venting operations are described in the Emergency Operating Procedures which is existing and currently in effect. Venting through the HVS will provide a high elevated release through the main stack. Using this system in venting is better than venting through the existing soft vent which will result in releases inside the Reactor Building.

6.2.7.4 Failure Modes and Effects Analysis (FMEA)

The 10" vent pipe which is not seismic, is provided with anti-fall down pipe supports. In the event of a seismic event, failure of the hardened vent pipes (8" and 10") may occur. However, venting the primary containment by other methods (soft vent) to prevent containment failure is available. Venting the primary containment to the RB during this condition has a potential of unmonitored ground releases of radioactive materials. However, the consequences of a failure of the primary containment is more severe than the consequences of a controlled venting of the primary containment with a potential of ground releases.

In the event of a failure of the isolation valves V-23-13 and 14, the drywell can be vented using V-23-15 and 16 if there is sufficient drywell pressure to overcome the torus water or using V-27-3 and 4 with V-28-42 and 43 closed. On the latter method, the elevated containment pressure will rupture the ductwork inside the RB. With RB ventilation exhaust fan EF-1-5 operating and isolation dampers (V-28-21/22) open, all the releases inside the RB will be exhausted and monitored through the stack. This method of venting could contaminate the entire RB and make it inaccessible. However, the consequences of a drywell failure is more severe than contaminating the RB.

If failure of Torus isolation valves V-23-15 and 16 occurs, venting the Torus can be performed through isolation valves V-23-13 and 14. The vent path is from the Torus through the vacuum breakers (V-26-1 to 14) to the drywell, out through the isolation valves V-23-13 and 14, through the hardened vent path and out to the stack. The vacuum breakers will open if the differential pressure between the Drywell and the Torus is greater than 0.5 psi.

6.2.7.5 Test and Inspection

The existing inservice inspection requirements for isolation valves V-23-13, 14, 15, and 16 are still in effect.

The newly installed hardened vent piping was pneumatically tested at 90 psig and tested at 60 psig for leakage per ANSI B31.1. The newly installed piping showed no evidence of leaking.

Accumulators and valve assembly were tested for leakage. In order to maintain the capability to cycle the valves six times, the air accumulator and check valve assembly shall be tested periodically for excessive leaks.

After the modification of the isolation valves, the containment isolation valves V-23-13, 14, 15 and 16 were stroked open and closed to demonstrate operability. In order to establish the as-found design basis, the closure times of the valves were documented and recorded and found to be less than 60 seconds.

The Railroad air lock was tested to demonstrate that the penetration of pipe supports through the side panel did not degrade the secondary containment integrity. The test consisted of running the normal exhaust fan EF-1-5 and monitoring the RB manometer at elevation 23'. The test passed and met the criteria of > 0.25 INW negative pressure at elevation 23'. Periodic testing of the RR Airlock is integrated with the Reactor Building Leakrate Test.

6.2.7.6 Instrumentation Requirements

The hardened vent system utilizes the existing stack radiation monitor and its instrumentation which is currently instrumented to the Control Room.

The hardened vent utilizes the existing pair of isolation valves off the Drywell and off the Torus which are also instrumented in the Control Room. The signal logic to close these isolation valves during a DBA is not changed. During normal inerting operation while the plant is starting up, these valves are open. Upon receipt of containment isolation signal (Hi DW pressure or Lo-Lo water level), these isolation valves will automatically close. The containment isolation function of the isolation valves remains the same. However, when venting operation is required, the containment isolation signal which is present to keep the valves closed is bypassed to allow for these valves to be opened under controlled supervision. A two-position key locked bypass switch labeled "Normal" and "Bypassed" is provided in panel 12XR to allow for these valves V-23-13, 14, 15, & 16 to be opened. The key for the bypass switch is removable in the "Normal" position only. A bypass annunciator to annunciate and indicate the bypassed condition of these valves is provided in the annunciator panel C. With the automatic closure logic bypassed, the containment isolation function is performed manually.

6.2.8 References

(1) Pacific Gas and Electric Company, Bodega Bay Atomic Park, Unit No. 1, Exhibit C, Preliminary Hazards Summary Report, Appendix I, Pressure Suppression Test Program, December 28, 1962, Docket No. 50-205.

(2) GEAP-3596, "Tests of Full Scale, 1/48 Segment of the Humboldt Bay Pressure Suppression Containment," C.H. Robbins, November 17, 1960.

(3) F.J. Moody, "Maximum Flow Rate of a Single Component Two Phase Mixture," Journal of Heat Transfer, ASME, Series C, Vol. 87, February 1965, p. 134.

(4) F.J. Moody, "Two-Phase Vessel Blowdown from Pipes," Journal of Heat Transfer, Trans ASME, Series C, Vol. 88, August 1966.

(5) A.H. Shapiro, "The Dynamics and Thermodynamics of Compressible Fluid Flow," Vol 1, The Ronald Press Company, 1953.

(6) ANL 6548, "Studies of Metal-Water Reaction at High Temperatures III Experimental and Theoretical Studies of Zirconium-Water Reaction," May 1962.

(7) Irminger, J.O.V. and Nkkentved, Chr. "Wind Pressures on Buildings, Second Series," Copenhagen, 1936.

(8) Koonty, R.L., C.T. Nelson and L. Baurmash, "Leakage Characteristics of Conventional 2Building Components for Reactor Housing Construction," Transactions of the American Nuclear Society, Vol. 4, No. 2, p. 365. November 1961.

(9) "A Guide for Reactor Containment Vessel Post-Licensing Leakage Rate Testing and Surveillance Requirements," December 15, 1965.

(10) NEDO-22155, "Generation and Mitigation of Combustible Gas Mixtures in Inerted BWR Mark I Containment," General Electric Company. June 1982.

(11) Northeast Utilities Docket No. 50-245, "Milestone No. 1 Combustible Gas Control Evaluation." August 1982.

(12) GPU Nuclear Corporation Topical Report No. 024 "Evaluation to Increase Normal Drywell Temperature Operating Limit," Revision 0, July 18, 1985.

(13) GPUN Safety Evaluation, "Deletion of Containment Spray Auto Start Logic", SE No. 402894-001.

(14) NRC letter, dated December 4, 1992, Issuance of Amendment No. 160, including enclosed NRC safety evaluation.

(15) NRC letter, dated September 13, 1993, Issuance of Amendment No. 165, including enclosed NRC safety evaluation.

(16) GPUN SE-000822-003, Rev 2, Trunion Room Access.

(17) NEDE-24802, “Mark I Containment Program Mark I Wetwell-to-Drywell Vacuum Breaker Functional Requirements, Task 9.4.3,” April 1980.

(18) AmerGen Calculation C-1302-243-E170-087, “Wetwell-to-Drywell Vacuum Breaker Sizing."

(19) U.S. NRC letter, dated March 9, 1987, regarding maximum drywell air temperature (TAC 63313).

Tables 6.2-1 and 6.2-2
Deleted

TABLE 6.2-3
(Sheet 1 of 1)

CONTAINMENT SPRAY PUMPS

Type	Horizontal, single stage, horizontally split-volute centrifugal pumps, motor driven, with mechanical seals, and gear type coupling
Rated Flow	3000 gpm
Runout Flow	4300 gpm
Shutoff TDH	320 feet
Rated TDH	260 feet
Runout TDH	195 feet
Required NPSH, Rated	12.5 feet
Required NPSH, Runout	23 feet
Brake Horsepower, Rated	237
Brake Horsepower, Runout	281
Materials	51A & B Carbon Steel Casing 51C & D Cast iron casing 51A & B Stainless Steel Impeller 51C & D Bronze impeller Steel shaft
Motor	1775 rpm, 300 hp, 440 volt, 3 phase

TABLE 6.2-4
(Sheet 1 of 1)

TORUS HEADER AND SPRAY NOZZLES

Penetrations	Into vacuum breaker lines, at azimuths 72 degrees and 288 degrees, and continuing into torus as small lines (4 in) running through large lines (24 in)
Pipe Size	4 in. penetrations and spray header
Spray Header Location	Complete ring header in torus air space at El. 10'-2" 101 ft diameter
Number of Spray Nozzles	10 total, uniformly distributed to cover as much of torus as possible. Extra connections plugged
Nozzle Description	Same as drywell

TABLE 6.2-5
(Sheet 1 of 1)

CONTAINMENT SPRAY SYSTEM SUCTION VALVES

Type	Gate Valve
Size	12 in
Operator	Electric motor, 440 volt, 3 phase
Opening Time	60 second maximum
Stem Seal	Double packing, with plugged lantern ring bleedoff
Back Seat	Provided
Open-and-Close Limitorque	Provided
Material	Carbon steel body, standard trim
Design Conditions	150 psig, 300°F, Code ASA B31.1

TABLE 6.2-6
(Sheet 1 of 1)

EMERGENCY SERVICE WATER PUMPS

Type	Vertical, open line shaft, self lubricated by pumped water, wet pit turbine type pump with discharge connection above mounting flange, motor driven and packed shaft seal
Pump Length	13 ft 5 5/8 in. below mounting flange
Diameter of Suction Bell	15 1/2 in
Number of Stages	4
Rated Flow	3000 gpm
Runout Flow	4057 gpm
Shutoff TDH	546 ft
Rated TDH	370 ft
Runout TDH	222 ft
Submergence Required	30 in
Brake Horsepower	371 (at rated TDH)
Materials	
Impeller	Type 316 stainless steel
Shaft	XM-19
Bearings	High-lead bronze
Bowl Casing	316 stainless steel
Motor	1800 rpm, 400 hp, 4000 volt, 3 phase; thrust bearing with space heater (120 volt) and oil lubricated bearings

NOTE: The current ESW pumps were purchased at a rating of 3000 gpm at 440 ft of head, but have been reconfigured to provide 3000 gpm at approximately 370 ft.

TABLE 6.2-7
(Sheet 1 of 2)

CONTAINMENT SPRAY HEAT EXCHANGERS

Number	4	
Type	Vertical, straight tube-shell heat exchangers	
*Heat Duty, each Btu/hr	$42 \times 10^6/30.4 \times 10^6$	
	Tube Side	Shell Side
Fluid	Barnegat Bay water	Demineralized water
*Flow Rate, gpm	3000/2075	3000/2150
**Fouling Factor		90 percent clean (both sides)
**Inlet Temperature, °F	85	130
*Outlet Temperature, °F	114/115.8	102/101
*Pressure Drop, psi	9.5/5.7	10/6.0
Design Pressure, psig	250	250
Design Temperature, °F	275	275
Number of Passes	4	2
Shell Outside Diameter, in.	51-1/2	
Tubes		
Number	1800	
Type	3/4 in. OD, No. 22 BWG, 18 ft long (ave)	
Connections, Channel and Shell	14 in, 14 out	

*Data shows two design conditions as listed in purchase specification.
**Data is derived from the vendor's heat exchanger specification sheet. The Containment Spray Heat Exchangers and system have been analyzed at a maximum cooling water temperature of 95°F as indicated in Table 6.2-15. The flow indicated is not that which is required for the system safety function.

TABLE 6.2-7
(Sheet 2 of 2)

CONTAINMENT SPRAY HEAT EXCHANGERS

Materials

Shell	SA-515 - Gr 70
Tubes	Titanium
Tube Sheet	SB-171-614 Aluminum Bronze
"D"	
Baffles – Long	SA-283C
Baffles – Cross	SA-283C
Tube Supports	SA-283C
Channel Cover	SA-515-70 Lined/Clad with 90/10 Cu/Ni

TABLE 6.2-8
(Sheet 1 of 1)

CONTAINMENT SPRAY SYSTEM DRYWELL HEADERS AND SPRAY NOZZLES

Drywell Penetrations	X-63 and X-66, located at elevation 62 ft and 27 ft, and at azimuth 330 and 33 for Loops I (A-B) and II (C-D) respectively
Pipe Size at Penetrations	14 in
Spray Header Pipe Size	8 in
Spray Header Elevations	37 ft and 65 ft
Spray Header Equivalent Diameters	52 ft and 35 ft
Number of Spray Nozzles	32 upper header 56 lower header 176 total
Nozzle Locations on Headers	Uniformly distributed and directed to cover as much of the drywell as possible. Extra connections in ring header are plugged.
Nozzle Type and Manufacturer	Fog nozzle assembly, Spraying System Company, Catalog No-7G25
Nozzle Material	Stainless steel Type 303
Nozzle Construction	1 in NPT female pipe connection; free passage (1/8 in particle; body with 7 removable spray caps, each cap with internal vane; assembly 3 1/4 in by 4 1/8 in
Flow Capacity per Nozzle	34 gpm at 40 psid 47 gpm at 80 psid 57 gpm at 125 psid
Spray Fog Pattern	17 ft outer cone and 11 ft inner cone at 8 ft from nozzle; inner cone is 65 percent of total

TABLE 6.2-9
(Sheet 1 of 2)

CONTAINMENT SPRAY SYSTEM VALVES

Header Test Block Valves V-21-5 and V-21-11

Type	Gate valve
Size	14 in
Operator	Electric motor, 460 volt, 3 phase
Opening Time	90 second maximum
Stem Seal	Double packing, with plugged lantern ring bleedoff
Back Seat	Provided
Open-and-Close Limitorque	Provided
Material	Cast steel body, standard trim
Design Conditions	150 psig, 300°F, Code ASA B31.1

Suppression Chamber Spray Block Valves V-21-15 and V-21-18

Type	Gate valve
Size	4 in
Operator	Electric motor, 460 volt, 3 phase
Opening Time	60 second maximum
Stem Seal	Double packing, with plugged lantern ring bleedoff
Backseat	Provided
Open-and-Close Limitorque	Provided
Material	Cast steel body, standard trim
Design Conditions	150 psig, 300°F, Code ASA B31.1

TABLE 6.2-9
(Sheet 2 of 2)

CONTAINMENT SPRAY SYSTEM VALVES

Torus Cooling Valves V-21-13 and V-21-17

Type	Gate valve
Size	6 in
Operator	Electric motor, 440 volt, 3 phase
Closing Time	60 second maximum
Stem Seal	Double packing, with plugged lantern ring bleedoff
Back Seat	Provided
Open-and-Closed Limitorque	Provided
Material	Cast steel body, standard trim
Design Conditions	150 psig, 300°F, Code ASA B31.1

Emergency Service Water Discharge Valves V-3-87 and V-3-88

Type	Butterfly valve
Size	14 in
Operator	Manual Hand Operated
Stem Seal	Packed
Material	Cast steel body, standard trim
Design Conditions	250 psig

TABLE 6.2-10
(Sheet 1 of 3)

CONTAINMENT SPRAY SYSTEM CONTROL AND INSTRUMENTATION

Instrument	Function
RE02A, -B, -C, -D	1. Annunciator in the Control Room: Rx LEVEL LO-LO
	2. Provides signals in conjunction with high drywell pressure (IP-15A, B, C, D) to actuate the NORMAL/EMERGENCY interlocks during accident (LOCA) conditions).
	3. Performs reactor protection system functions on LO-LO reactor water level including: close main steam line isolation valves, starts core spray system, cleanup and shutdown cooling systems isolation, reactor recirculation pumps trip, starts isolation condensers, starts standby gas treatment system, starts emergency diesel generators, close other primary containment (drywell) isolation valves.
PS-IP15A, -B, -C, -D	1. Annunciators in the Control Room: SYS 1 DW PRESS LO SYS 2 DW PRESS LO
	2. Provides signals at 2.9 psig in conjunction with a low-low reactor water level (16K110A,B,C,D) to actuate the NORMAL/EMERGENCY interlocks during accident (LOCA) conditions.
	3. Interlocks control of containment spray pumps to prevent operation at low drywell pressure. Set point at 0.6 psig, if operating in the "Drywell Spray" mode.
dPT-IP05A, -B, -C, -D	1. Differential pressure indicators dPI- IP06A, -B, -C, -D for each heat exchanger, measuring pressure differential from tube side to shell side.
	2. Annunciators in the Control Room: SYS 1 TUBE/SHELL PLO1 SYS 2 TUBE/SHELL PLO2 via dPS-IP14A, -B, -C, -D. Alarms with insufficient pressure differential from emergency service water to containment spray water.

TABLE 6.2-10
(Sheet 2 of 3)

CONTAINMENT SPRAY SYSTEM CONTROL AND INSTRUMENTATION

Instrument	Function
PI-ESWP-1, 2, 3 and 4	Emergency service water pumps discharge pressure indicators local.
PI-21-349, 350, 351, 352	Containment Spray Pumps suction pressure indicators local.
TE-40A	Temperature element to TR-IP01 in Control Room; records temperature of water in suction header to containment spray pumps and core spray pumps.
TE-40B, -C	Temperature elements to TR-IP01 in Control Room; records temperature of containment spray water from heat exchangers in Loops 1 and 2 respectively.
TE-664-30A to 35A TE-664-30B to 35B	Two Channels of temperature sensors provide redundant means of monitoring torus bulk water temperature. Indicators TI-664-43A, -B in Control Room
FT-IP03A, -B	1. Flow indicators FI-IP04A, -B in Control Room for each containment spray loop. 2. Senses low flow in containment spray Loops 1 and 2 respectively via FS-IP17A, -B as a failure monitor. Annunciates in the Control Room: SYS 1 FLOW LO SYS 1 FLOW LO
DPI-532-0005	Local Flow Indicator for E.S.W. System 1
DPI-532-0006	Local Flow Indicator for E.S.W. System 2
FT-532-0499	Differential Pressure Transmitter for E.S.W. System #1 Control Room Flow indication. Indication is displayed on demand in Control Room CRT.
FT-532-1743	Differential Pressure Transmitter for E.S.W. System #2 Control Room flow indication. Indication is displayed on demand in Control Room CRT.

TABLE 6.2-10
(Sheet 3 of 3)

CONTAINMENT SPRAY SYSTEM CONTROL
AND INSTRUMENTATION

Relay/Switch	Function
16K17A, -B	125-volt dc control bus failure sensor for each containment spray loop. 1. Annunciators in the Control Room: CNTRL PWR 1 LOST CNTRL PWR 2 LOST
16S-20A	System 1 Pumps Manual Start Permissive. Three position A-NORM-B, keylock in NORM, spring return to NORM. Should be locked in NORM.
16S-20B	System 2 Pumps Manual Start Permissive. Same as 16S-20A, except three positions are C-NORM-D.
16S-18A, -B, -C, -D	Individual containment spray pump manual switches for pumps A, B, C, D respectively. Three position STOP - NORMAL - START, with spring-return to normal.
16S-19A, -B, -C, -D	Control switches for individual emergency service water pumps A, B, C, D respectively. Three position STOP - NORMAL - START spring return to NORMAL.
CS-16S-15A, B	Drywell spray/torus cooling mode selector switches for each loop. Switches to be normally in "Torus Cooling" mode.
PNL-642-1FPB1 A/B	Low flow alarm signal manual reset switches. Used to clear low flow alarm signal after pumps are tripped.

TABLE 6.2-11
(Sheet 1 of 1)

SECONDARY CONTAINMENT PRINCIPAL DESIGN PARAMETERS

Reactor Building

Length	143 ft
Width	140 ft.
Height	146 ft.
Free Volume (Approximately)	1,800,000 ft^3
Inleakage Rate (Maximum)	100%/day @ 1/4 in water vacuum
Internal Design Pressure	0.20 psig

Plant Stack

Base Diameter, Internal	28 ft. 8 in.
Exit Diameter, Internal	8 ft. 6 in.
Height Above Reactor Bldg. Grade	368 ft.

Ventilation System Flows

Reactor Bldg. Ventilation	76,700 scfm
Reactor Bldg. Standby Gas Treatment System	2,600 scfm (1 of 2)
Turbine Building	90,870 scfm
Old Radwaste Building	22,475 scfm

TABLE 6.2-12
Sheet 1 of 28

CONTAINMENT ISOLATION VALVES / MECHANICAL INTEGRITY

1 PEN	2 SIZE	3 SYSTEM SERVICE	4 FIG	5 FLUID	6 TAG NO	7 SIGNAL	8 LOC	9 TYPE	10 VLVOP	11 PRI ACT	12 SEC ACT	13 MS	14 PWR	15 CR POS	16 NORM	17 SHDW	18 P ACC	19 LOP	20 COMMENT
X-1A,B	120	DW PERS AIRLK & EQP HATCH	N/A	N/A	N/A	N/A	OUT	N/A	N/A	N/A	N/A	N/A	N/A	NONE	CLOSE	CLOSE	CLOSE	N/A	BOLTED CLOSED
X-2A	24	MAIN STEAM	6.2-40	STEAM	V-1-007	1,2	IN	GLOBE	AIR	AUTO	REM MAN	≤10	AIR	DIRECT	OPEN	CL/OPN	CLOSE	CLOSE	
X-2A	24	MAIN STEAM	6.2-40	STEAM	V-1-009	1,2	OUT	GLOBE	AIR	AUTO	REM MAN	≤10	AIR	DIRECT	OPEN	CL/OPN	CLOSE	CLOSE	
X-2A	3\4	MAIN STEAM	6.2-40	STEAM	V-1-245	NONE	OUT	GLOBE	HAND	N/A	N/A	N/A	N/A	NONE	CLOSE	CLOSE	CLOSE	N/A	LOCKED CLOSED
X-2A	1	MAIN STEAM	6.2-40	STEAM	V-1-114	NONE	OUT	GLOBE	HAND	N/A	N/A	N/A	N/A	NONE	CLOSE	CLOSE	CLOSE	N/A	LOCKED CLOSED
X-2B	24	MAIN STEAM	6.2-40	STEAM	V-1-008	1,2	IN	GLOBE	AIR	AUTO	REM MAN	≤10	AIR	DIRECT	OPEN	CL/OPN	CLOSE	CLOSE	
X-2B	24	MAIN STEAM	6.2-40	STEAM	V-1-010	1,2	OUT	GLOBE	AIR	AUTO	REM MAN	≤10	AIR	DIRECT	OPEN	CL/OPN	CLOSE	CLOSE	
X-2B	3\4	MAIN STEAM	6.2-40	STEAM	V-1-194	NONE	OUT	GLOBE	HAND	N/A	N/A	N/A	N/A	NONE	CLOSE	CLOSE	CLOSE	N/A	LOCKED CLOSED
X-2B	1	MAIN STEAM	6.2-40	STEAM	V-1-115	NONE	OUT	GLOBE	HAND	N/A	N/A	N/A	N/A	NONE	CLOSE	CLOSE	CLOSE	N/A	LOCKED CLOSED
X-3A	10	ISOL COND(STEAM SUPPLY)	6.2-39	STEAM	V-14-030	1,3	OUT	GATE	MOTOR	AUTO	REM MAN	*	AC	DIRECT	OPEN	OPEN	OPEN	AS IS	* SEE TABLE 6.3-1
X-3A	10	ISOL COND(STEAM SUPPLY)	6.2-39	STEAM	V-14-031	1,3	OUT	GATE	MOTOR	AUTO	REM MAN	*	DC	DIRECT	OPEN	OPEN	OPEN	AS IS	* SEE TABLE 6.3-1
X-3A	3\4	ISOL COND VENT	6.2-39	STEAM	V-14-005	1,2	OUT	GLOBE	AIR	AUTO	REM MAN	≤60	AIR	DIRECT	OPEN	OPEN	CLOSE	CLOSE	
X-3A	3\4	ISOL COND VENT	6.2-39	STEAM	V-14-020	1,2	OUT	GLOBE	AIR	AUTO	REM MAN	≤60	AIR	DIRECT	OPEN	OPEN	CLOSE	CLOSE	
X-3A	1	ISOL COND(STEAM SUPPLY)	6.2-39	STEAM	V-14-122	NONE	OUT	GLOBE	HAND	N/A	N/A	N/A	N/A	NONE	CLOSE	CLOSE	CLOSE	N/A	LOCKED CLOSED
X-3B	10	ISOL COND(STEAM SUPPLY)	6.2-39	STEAM	V-14-032	1,3	OUT	GATE	MOTOR	AUTO	REM MAN	*	AC	DIRECT	OPEN	OPEN	OPEN	AS IS	* SEE TABLE 6.3-1
X-3B	10	ISOL COND(STEAM SUPPLY)	6.2-39	STEAM	V-14-033	1,3	OUT	GATE	MOTOR	AUTO	REM MAN	*	DC	DIRECT	OPEN	OPEN	OPEN	AS IS	* SEE TABLE 6.3-1
X-3B	3\4	ISOL COND VENT	6.2-39	STEAM	V-14-001	1,2	OUT	GLOBE	AIR	AUTO	REM MAN	≤60	AIR	DIRECT	OPEN	OPEN	CLOSE	CLOSE	
X-3B	3\4	ISOL COND VENT	6.2-39	STEAM	V-14-019	1,2	OUT	GLOBE	AIR	AUTO	REM MAN	≤60	AIR	DIRECT	OPEN	OPEN	CLOSE	CLOSE	

TABLE 6.2-12
Sheet 2 of 28

CONTAINMENT ISOLATION VALVES / MECHANICAL INTEGRITY

1 PEN	2 SIZE	3 SYSTEM SERVICE	4 FIG	5 FLUID	6 TAG NO	7 SIGNAL	8 LOC	9 TYPE	10 VLVOP	11 PRI ACT	12 SEC ACT	13 MS	14 PWR	15 CR POS	16 NORM	17 SHDW	18 P ACC	19 LOP	20 COMMENT
X-3B	1	ISOL COND(STEAM SUPPLY)	6.2-39	STEAM	V-14-118	NONE	OUT	GLOBE	HAND	N/A	N/A	N/A	N/A	NONE	CLOSE	CLOSE	CLOSE	N/A	LOCKED CLOSED
X-4A	18	FEEDWATER	6.2-40	WATER	V-2-072	NONE	OUT	CHECK	NONE	REV FLO	N/A	N/A	N/A	NONE	OPEN	CLOSE	OP/CL	N/A	
X-4A	18	FEEDWATER	6.2-40	WATER	V-2-074	NONE	IN	CHECK	NONE	REV FLO	N/A	N/A	N/A	NONE	OPEN	CLOSE	OP/CL	N/A	
X-4A	1	FEEDWATER	6.2-40	WATER	V-2-112	NONE	OUT	GLOBE	HAND	N/A	N/A	N/A	N/A	NONE	CLOSE	CLOSE	CLOSE	N/A	LOCKED CLOSED
X-4B	18	FEEDWATER	6.2-40	WATER	V-2-071	NONE	OUT	CHECK	NONE	REV FLO	N/A	N/A	N/A	NONE	OPEN	CLOSE	OP/CL	N/A	
X-4B	18	FEEDWATER	6.2-40	WATER	V-2-073	NONE	IN	CHECK	NONE	REV FLO	N/A	N/A	N/A	NONE	OPEN	CLOSE	OP/CL	N/A	
X-4B	1	FEEDWATER	6.2-40	WATER	V-2-109	NONE	OUT	GLOBE	HAND	N/A	N/A	N/A	N/A	NONE	CLOSE	CLOSE	CLOSE	N/A	LOCKED CLOSED
X-5A	10	ISOL COND(COND RETURN)	6.2-39	WATER	V-14-035	1,3	OUT	GATE	MOTOR	AUTO	REM MAN	*	DC	DIRECT	CLOSE	CLOSE	OPEN	AS IS	* SEE TABLE 6.3-1
X-5A	10	ISOL COND(COND RETURN)	6.2-39	WATER	V-14-037	1,3	IN	GATE	MOTOR	AUTO	REM MAN	*	AC	DIRECT	OPEN	CLOSE	OPEN	AS IS	* SEE TABLE 6.3-1
X-5A	1	ISO COND (COND RETURN)	6.2-39	WATER	V-14-162	NONE	IN	CHECK	NONE	REV FLO	N/A	N/A	N/A	NONE	CLOSE	CLOSE	CLOSE	N/A	
X-5A	1	ISOL COND(COND RETURN)	6.2-39	WATER	V-14-130	NONE	OUT	GLOBE	HAND	N/A	N/A	N/A	N/A	NONE	CLOSE	CLOSE	CLOSE	N/A	LOCKED CLOSED
X-5B	10	ISOL COND(COND RETURN)	6.2-39	WATER	V-14-034	1,3	OUT	GATE	MOTOR	AUTO	REM MAN	*	DC	DIRECT	CLOSE	CLOSE	OPEN	AS IS	* SEE TABLE 6.3-1
X-5B	10	ISOL COND(COND RETURN)	6.2-39	WATER	V-14-036	1,3	IN	GATE	MOTOR	AUTO	REM MAN	*	AC	DIRECT	OPEN	CLOSE	OPEN	AS IS	* SEE TABLE 6.3-1
X-5B	1	ISOL COND (COND RETURN)	6.2-39	WATER	V-14-165	NONE	IN	CHECK	NONE	REV FLO	N/A	N/A	N/A	NONE	CLOSE	CLOSE	CLOSE	N/A	
X-5B	1	ISOL COND(COND RETURN)	6.2-39	WATER	V-14-126	NONE	OUT	GLOBE	HAND	N/A	N/A	N/A	N/A	NONE	CLOSE	CLOSE	CLOSE	N/A	LOCKED CLOSED
X-6	1 1\2	LIQUID POISON	6.2-41	WATER	V-19-020	NONE	IN	CHECK	NONE	REV FLO	N/A	N/A	N/A	NONE	CLOSE	CLOSE	CLOSE	N/A	
X-6	1 1\2	LIQUID POISON	6.2-41	WATER	V-19-016	NONE	OUT	CHECK	NONE	REV FLO	N/A	N/A	N/A	NONE	CLOSE	CLOSE	CLOSE	N/A	
X-6	1	LIQUID POISON	6.2-41	WATER	V-19-017	NONE	OUT	GLOBE	HAND	N/A	N/A	N/A	N/A	NONE	CLOSE	CLOSE	CLOSE	N/A	LOCKED CLOSED

TABLE 6.2-12
Sheet 3 of 28

CONTAINMENT ISOLATION VALVES / MECHANICAL INTEGRITY

1 PEN	2 SIZE	3 SYSTEM SERVICE	4 FIG	5 FLUID	6 TAG NO	7 SIGNAL	8 LOC	9 TYPE	10 VLVOP	11 PRI ACT	12 SEC ACT	13 MS	14 PWR	15 CR POS	16 NORM	17 SHDW	18 P ACC	19 LOP	20 COMMENT
X-7	14	SHUTDOWN COOLING(SUPPLY)	6.2-42	WATER	V-17-019	1,5,6,7	IN	GATE	MOTOR	AUTO	REM MAN	≤60	AC	DIRECT	CLOSE	OPEN	CLOSE	AS IS	FOR V-17-001, 002, 003
X-7	10	SHUTDOWN COOLING(SUPPLY)	6.2-42	WATER	V-17-001	NONE	OUT	GATE	MOTOR	REM MAN	REM MAN	N/A	DC	DIRECT	CLOSE	OPEN	CLOSE	AS IS	VALVE POSITION
X-7	10	SHUTDOWN COOLING(SUPPLY)	6.2-42	WATER	V-17-002	NONE	OUT	GATE	MOTOR	REM MAN	REM MAN	N/A	DC	DIRECT	CLOSE	OPEN	CLOSE	AS IS	DEPENDS ON POST
X-7	10	SHUTDOWN COOLING(SUPPLY)	6.2-42	WATER	V-17-003	NONE	OUT	GATE	MOTOR	REM MAN	REM MAN	N/A	DC	DIRECT	CLOSE	OPEN	CLOSE	AS IS	SHTDWN HEAT LOADS
X-7	1	SHUTDOWN COOLING(SUPPLY)	6.2-42	WATER	V-17-021	NONE	OUT	GLOBE	HAND	N/A	N/A	N/A	N/A	NONE	CLOSE	CLOSE	CLOSE	N/A	LOCKED CLOSED
X-7	1	SHUTDOWN COOLING(SUPPLY)	6.2-42	WATER	V-17-065	NONE	OUT	GLOBE	HAND	N/A	N/A	N/A	N/A	NONE	CLOSE	CLOSE	CLOSE	N/A	LOCKED CLOSED
X-7	1\2	SHUTDOWN COOLING(SUPPLY)	6.2-42	WATER	V-17-076	NONE	OUT	GLOBE	HAND	N/A	N/A	N/A	N/A	NONE	CLOSE	CLOSE	CLOSE	N/A	LOCKED CLOSED
X-7	1\2	SHUTDOWN COOLING(SUPPLY)	6.2-42	WATER	V-17-227	NONE	OUT	RELIEF	N/A	N/A	N/A	N/A	N/A	NONE	CLOSE	CLOSE	CLOSE	N/A	SHUT
X-8	14	SHUTDOWN COOLING(RETURN)	6.2-42	WATER	V-17-054	1,5,6,7	IN	GATE	MOTOR	AUTO	REM MAN	<60	AC	DIRECT	CLOSE	OPEN	CLOSE	AS IS	FOR V-17-055, 056, 057
X-8	8	SHUTDOWN COOLING(RETURN)	6.2-42	WATER	V-17-055	NONE	OUT	GLOBE	MOTOR	REM MAN	REM MAN	N/A	DC	DIRECT	CLOSE	OPEN	CLOSE	AS IS	VALVE POSITION
X-8	8	SHUTDOWN COOLING(RETURN)	6.2-42	WATER	V-17-056	NONE	OUT	GLOBE	MOTOR	REM MAN	REM MAN	N/A	DC	DIRECT	CLOSE	OPEN	CLOSE	AS IS	DEPENDS ON POST
X-8	8	SHUTDOWN COOLING(RETURN)	6.2-42	WATER	V-17-057	NONE	OUT	GLOBE	MOTOR	REM MAN	REM MAN	N/A	DC	DIRECT	CLOSE	OPEN	CLOSE	AS IS	SHTDWN HEAT LOADS
X-8	1	SHUTDOWN COOLING(RETURN)	6.2-42	WATER	V-17-050	NONE	OUT	GLOBE	HAND	N/A	N/A	N/A	N/A	NONE	CLOSE	CLOSE	CLOSE	N/A	LOCKED CLOSED
X-8	1	SHUTDOWN COOLING(RETURN)	6.2-42	WATER	V-17-083	NONE	OUT	GLOBE	HAND	N/A	N/A	N/A	N/A	NONE	CLOSE	CLOSE	CLOSE	N/A	LOCKED CLOSED
X-8	3\4	SHUTDOWN COOLING(RETURN)	6.2-42	WATER	V-40-008	1	OUT	GLOBE	SOLND	REM MAN	N/A	N/A	AC	INDRCT	CLOSE	CLOSE	OPEN	CLOSE	KEYLOCKED CLSD

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CONTAINMENT ISOLATION VALVES / MECHANICAL INTEGRITY

1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16	17	18	19	20
PEN	SIZE	SYSTEM SERVICE	FIG	FLUID	TAG NO	SIGNAL	LOC	TYPE	VLVOP	PRI ACT	SEC ACT	MS	PWR	CR POS	NORM	SHDW	P ACC	LOP	COMMENT
X-8	1	SHUTDOWN COOLING(RETURN)	6.2-42	WATER	V-17-068	NONE	OUT	GLOBE	HAND	N/A	N/A	N/A	N/A	NONE	CLOSE	CLOSE	CLOSE	N/A	LOCKED CLOSED
X-8	1	SHUTDOWN COOLING(RETURN)	6.2-42	WATER	V-17-066	NONE	OUT	GLOBE	HAND	N/A	N/A	N/A	N/A	NONE	CLOSE	CLOSE	CLOSE	N/A	LOCKED CLOSED
X-8	1	SHUTDOWN COOLING(RETURN)	6.2-42	WATER	V-17-052	NONE	OUT	GLOBE	HAND	N/A	N/A	N/A	N/A	NONE	CLOSE	CLOSE	CLOSE	N/A	LOCKED CLOSED
X-8	3\4	SHUTDOWN COOLING(RETURN)	6.2-42	WATER	V-17-209	NONE	OUT	GLOBE	HAND	N/A	N/A	N/A	N/A	NONE	CLOSE	CLOSE	CLOSE	N/A	LOCKED CLOSED
X-8	3\4	SHUTDOWN COOLING(RETURN)	6.2-42	WATER	V-17-211	NONE	OUT	GLOBE	HAND	N/A	N/A	N/A	N/A	NONE	CLOSE	CLOSE	CLOSE	N/A	LOCKED CLOSED
X-8	3\4	SHUTDOWN COOLING(RETURN)	6.2-42	WATER	V-17-213	NONE	OUT	GLOBE	HAND	N/A	N/A	N/A	N/A	NONE	CLOSE	CLOSE	CLOSE	N/A	LOCKED CLOSED
X-9	6	REACTOR CLEANUP(RETURN)	6.2-43	WATER	V-16-061	1,6,7,10	OUT	GATE	MOTOR	AUTO	REM MAN	≤60	AC	DIRECT	OPEN	OPEN	CLOSE	AS IS	
X-9	6	REACTOR CLEANUP(RETURN)	6.2-43	WATER	V-16-062	NONE	IN	CHECK	NONE	REV FLO	N/A	N/A	N/A	NONE	OP/CL	OPEN	CLOSE	N/A	
X-9	1	REACTOR CLEANUP(RETURN)	6.2-43	WATER	V-16-065	NONE	OUT	GLOBE	HAND	N/A	N/A	N/A	N/A	NONE	CLOSE	CLOSE	CLOSE	N/A	LOCKED CLOSED
X-9	3\4	REACTOR CLEANUP(RETURN)	6.2-43	WATER	V-16-223	NONE	OUT	GLOBE	HAND	N/A	N/A	N/A	N/A	NONE	CLOSE	CLOSE	CLOSE	N/A	LOCKED CLOSED
X-10	6	REACTOR CLEANUP(SUPPLY)	6.2-43	WATER	V-16-001	1,4,6,7, 10	IN	GATE	MOTOR	AUTO	REM MAN	≤60	AC	DIRECT	OPEN	CLOSE	CLOSE	AS IS	
X-10	6	REACTOR CLEANUP(SUPPLY)	6.2-43	WATER	V-16-014	1,4,6,7, 10	OUT	GATE	MOTOR	AUTO	REM MAN	≤60	DC	DIRECT	OP/CL	CLOSE	CLOSE	AS IS	
X-10	6	REACTOR CLEANUP(SUPPLY)	6.2-43	WATER	V-16-002	1,4,6,7, 10	OUT	GATE	MOTOR	AUTO	REM MAN	≤60	DC	DIRECT	OP/CL	CLOSE	CLOSE	AS IS	
X-10	1	REACTOR CLEANUP(SUPPLY)	6.2-43	WATER	V-16-004	NONE	OUT	GLOBE	HAND	N/A	N/A	N/A	N/A	NONE	CLOSE	CLOSE	CLOSE	N/A	LOCKED CLOSED

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CONTAINMENT ISOLATION VALVES / MECHANICAL INTEGRITY

1 PEN	2 SIZE	3 SYSTEM SERVICE	4 FIG	5 FLUID	6 TAG NO	7 SIGNAL	8 LOC	9 TYPE	10 VLVOP	11 PRI ACT	12 SEC ACT	13 MS	14 PWR	15 CR POS	16 NORM	17 SHDW	18 P ACC	19 LOP	20 COMMENT
X-10	3\4	REACTOR CLEANUP(SUPPLY)	6.2-43	WATER	V-16-221	NONE	OUT	GLOBE	HAND	N/A	N/A	N/A	N/A	NONE	CLOSE	CLOSE	CLOSE	N/A	LOCKED CLOSED
X-12A	1	REACTOR LEVEL INDICATION	6.2-53	WATER	V-130-022A	NONE	OUT	GLOBE	HAND	N/A	N/A	N/A	N/A	NONE	OPEN	OPEN	OPEN	N/A	LIS-RE18A/C
X-12A	1	REACTOR LEVEL INDICATION	6.2-53	WATER	V-130-021A	NONE	OUT	CHECK	NONE	N/A	N/A	N/A	N/A	NONE	OPEN	OPEN	OPEN	N/A	LIS-RE18A/C
X-12B	8	CORE SPRAY(SUPPLY)	6.2-56	WATER	V-20-150	NONE	IN	CHECK	AIR	REV FLO	N/A	N/A	AIR	DIRECT	CLOSE	CLOSE	OPEN	CLOSE	AIR FOR TEST ONLY
X-12B	8	CORE SPRAY(SUPPLY)	6.2-56	WATER	V-20-152	NONE	IN	CHECK	AIR	REV FLO	N/A	N/A	AIR	DIRECT	CLOSE	CLOSE	OPEN	CLOSE	AIR FOR TEST ONLY
X-12B	8	CORE SPRAY(SUPPLY)	6.2-56	WATER	V-20-015	NONE	OUT	GATE	MOTOR	AUTO	REM MAN	N/A	AC	DIRECT	CLOSE	CLOSE	OPEN	AS IS	
X-12B	8	CORE SPRAY(SUPPLY)	6.2-56	WATER	V-20-040	NONE	OUT	GATE	MOTOR	AUTO	REM MAN	N/A	AC	DIRECT	CLOSE	CLOSE	OPEN	AS IS	
X-12B	3\4	CORE SPRAY(SUPPLY)	6.2-56	WATER	V-20-042	NONE	OUT	GLOBE	HAND	N/A	N/A	N/A	N/A	NONE	CLOSE	CLOSE	CLOSE	N/A	LOCKED CLOSED
X-12B	3\4	CORE SPRAY(SUPPLY)	6.2-56	WATER	V-40-012	NONE	OUT	GLOBE	SOLND	REM MAN	N/A	N/A	AC	INDRCT	CLOSE	CLOSE	OPEN	CLOSE	
X-13A/B	1\2	CRD INSERT	6.2-47	WATER	SO-305-123	NONE	OUT	NEEDLE	SOLND	REM MAN	N/A	N/A	AC	INDRCT	CLOSE	CLOSE	CLOSE	CLOSE	137 LINES
X-13A/B	1\2	CRD SCRAM	6.2-47	WATER	CV-305-126	NONE	OUT	CNTRL	AIR	AUTO	N/A	N/A	AIR	INDRCT	CLOSE	CLOSE	CLOSE	OPEN	137 LINES
X-13A/B	3\4	CRD WITHDRAW	6.2-47	WATER	SO-305-120	NONE	OUT	NEEDLE	SOLND	REM MAN	N/A	N/A	AC	INDRCT	CLOSE	CLOSE	CLOSE	CLOSE	137 LINES
X-13A/B	1\2	CRD COOLING WATER	6.2-47	WATER	V-305-138	NONE	OUT	CHECK	NONE	REV FLO	N/A	N/A	N/A	NONE	CLOSE	CLOSE	CLOSE	N/A	137 LINES
X-13A/B	3\4	CRD INSERT	6.2-47	WATER	V-305-139	NONE	OUT	GATE	HAND	N/A	N/A	N/A	N/A	NONE	CLOSE	CLOSE	CLOSE	N/A	137 LINES(WITH PLUG)

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CONTAINMENT ISOLATION VALVES / MECHANICAL INTEGRITY

1 PEN	2 SIZE	3 SYSTEM SERVICE	4 FIG	5 FLUID	6 TAG NO	7 SIGNAL	8 LOC	9 TYPE	10 VLVOP	11 PRI ACT	12 SEC ACT	13 MS	14 PWR	15 CR POS	16 NORM	17 SHDW	18 P ACC	19 LOP	20 COMMENT
X-14A/B	1\2	CRD INSERT	6.2-47	WATER	SO-305-121	NONE	OUT	GLOBE	SOLND	REM MAN	N/A	N/A	AC	INDRCT	CLOSE	CLOSE	CLOSE	CLOSE	137 LINES
X-14A/B	3\4	CRD SCRAM	6.2-47	WATER	CV-305-127	NONE	OUT	CNTRL	AIR	AUTO	N/A	N/A	AIR	INDRCT	CLOSE	CLOSE	CLOSE	OPEN	137 LINES
X-14A/B	1\2	CRD WITHDRAW	6.2-47	WATER	SO-305-122	NONE	OUT	GLOBE	SOLND	REM MAN	N/A	N/A	AC	INDRCT	CLOSE	CLOSE	CLOSE	CLOSE	137 LINES
X-14A/B	3\4	CRD WITHDRAW	6.2-47	WATER	V-305-140	NONE	OUT	GATE	NONE	N/A	N/A	N/A	N/A	NONE	CLOSE	CLOSE	CLOSE	N/A	137 LINES(WITH PLUG)
X-15	2	INSTRUMENT AIR	6.2-51	N2	V-6-393	NONE	IN	CHECK	NONE	REV FLO	N/A	N/A	N/A	NONE	OPEN	OPEN	CLOSE	N/A	
X-15	2	INSTRUMENT AIR	6.2-51	N2	V-6-395	1,2	OUT	GLOBE	AIR	AUTO	REM MAN	N/A	AIR	DIRECT	OPEN	OPEN	CLOSE	CLOSE	MSIV AUTO CLOSE
X-15	3\4	INSTRUMENT AIR	6.2-51	N2	V-6-394	NONE	IN	GLOBE	HAND	N/A	N/A	N/A	N/A	NONE	CLOSE	CLOSE	CLOSE	N/A	LOCKED CLOSED
X-16	3\4	DRYWELL O2 SAMPLING	6.2-52	AIR	V-38-007	NONE	OUT	GATE	HAND	N/A	N/A	N/A	N/A	NONE	CLOSE	CLOSE	CLOSE	N/A	LOCKED CLOSED
X-16	3\4	DRYWELL O2 SAMPLING	6.2-52	AIR	V-38-009	1,6,7	OUT	GLOBE	SOLND	AUTO	REM MAN	N/A	AC	INDRCT	OPEN	OPEN	OPEN	CLOSE	
X-16	3\4	DRYWELL O2 SAMPLING	6.2-52	AIR	V-38-010	1,6,7	OUT	GLOBE	SOLND	AUTO	REM MAN	N/A	AC	INDRCT	OPEN	OPEN	OPEN	CLOSE	
X-16	3\4	DRYWELL O2 SAMPLING	6.2-52	AIR	V-38-008	NONE	OUT	GATE	HAND	N/A	N/A	N/A	N/A	NONE	CLOSE	CLOSE	CLOSE	N/A	LOCKED CLOSED
X-18	18	VENTILATION INTAKE	6.2-54	AIR	V-27-003	1,6,7,9	OUT	BFLY	AIR	AUTO	REM MAN	≤5	AIR	DIRECT	CLOSE	CLOSE	CLOSE	CLOSE	
X-18	18	VENTILATION INTAKE	6.2-54	AIR	V-27-004	1,6,7,9	OUT	BFLY	AIR	AUTO	REM MAN	≤5	AIR	DIRECT	CLOSE	CLOSE	CLOSE	CLOSE	
X-18	8	N2 PURGE / HARDENED VENT	6.2-54	N2	V-23-013	1,6,7	OUT	BFLY	AIR	AUTO	REM MAN	<60	AIR	DIRECT	CLOSE	CLOSE	CL/OP	CLOSE	75 DEG MAX OPEN
X-18	8	N2 PURGE / HARDENED VENT	6.2-54	N2	V-23-014	1,6,7	OUT	BFLY	AIR	AUTO	REM MAN	<60	AIR	DIRECT	CLOSE	CLOSE	CL/OP	CLOSE	75 DEG MAX OPEN
X-18	2	NITROGEN PURGE	6.2-54	N2	V-23-017	1,6,7	OUT	CNTRL	AIR	AUTO	REM MAN	<60	AIR	DIRECT	CLOSE	CLOSE	CLOSE	CLOSE	

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CONTAINMENT ISOLATION VALVES / MECHANICAL INTEGRITY

1 PEN	2 SIZE	3 SYSTEM SERVICE	4 FIG	5 FLUID	6 TAG NO	7 SIGNAL	8 LOC	9 TYPE	10 VLVOP	11 PRI ACT	12 SEC ACT	13 MS	14 PWR	15 CR POS	16 NORM	17 SHDW	18 P ACC	19 LOP	20 COMMENT
X-18	2	NITROGEN PURGE	6.2-54	N2	V-23-018	1,6,7	OUT	CNTRL	AIR	AUTO	REM MAN	<60	AIR	DIRECT	CLOSE	CLOSE	CLOSE	CLOSE	
X-18	3\4	NITROGEN PURGE	6.2-54	N2	V-23-255	NONE	OUT	GATE	HAND	N/A	N/A	N/A	N/A	NONE	CLOSE	CLOSE	CLOSE	N/A	LOCKED CLOSED
X-18	1\2	NITROGEN PURGE	6.2-54	N2	V-23-258	NONE	OUT	GATE	HAND	N/A	N/A	N/A	N/A	NONE	CLOSE	CLOSE	CLOSE	N/A	LOCKED CLOSED
X-18	3\4	VENTILATION INTAKE	6.2-54	AIR	V-27-007	NONE	OUT	GLOBE	HAND	N/A	N/A	N/A	N/A	NONE	CLOSE	CLOSE	CLOSE	N/A	LOCKED CLOSED
X-18	3\4	VENTILATION INTAKE	6.2-54	AIR	V-27-008	NONE	OUT	GLOBE	HAND	N/A	N/A	N/A	N/A	NONE	CLOSE	CLOSE	CLOSE	N/A	LOCKED CLOSED
X-19	18	VENTILATION EXHAUST	6.2-54	AIR	V-27-001	1,6,7,9	OUT	BFLY	AIR	AUTO	REM MAN	$\leq$5	AIR	DIRECT	CL/OP	OPEN	CLOSE	CLOSE	30 DEGREE MAX OPEN
X-19	18	VENTILATION EXHAUST	6.2-54	AIR	V-27-002	1,6,7,9	OUT	BFLY	AIR	AUTO	REM MAN	$\leq$5	AIR	DIRECT	CL/OP	OPEN	CLOSE	CLOSE	30 DEGREE MAX OPEN
X-19	2	NITROGEN RELIEF	6.2-54	N2	V-23-021	1,6,7	OUT	CNTRL	AIR	AUTO	REM MAN	<60	AIR	DIRECT	CLOSE	CLOSE*	CLOSE	CLOSE	
X-19	2	NITROGEN RELIEF	6.2-54	N2	V-23-022	1,6,7	OUT	CNTRL	AIR	AUTO	REM MAN	<60	AIR	DIRECT	CLOSE	CLOSE	CLOSE	CLOSE	
X-19	3\4	VENTILATION EXHAUST	6.2-54	AIR	V-27-005	NONE	OUT	GLOBE	HAND	N/A	N/A	N/A	N/A	NONE	CLOSE	CLOSE	CLOSE	N/A	LOCKED CLOSED
X-19	3\4	VENTILATION EXHAUST	6.2-54	AIR	V-27-006	NONE	OUT	GLOBE	HAND	N/A	N/A	N/A	N/A	NONE	CLOSE	CLOSE	CLOSE	N/A	LOCKED CLOSED
X-20A	6	DW CLOSED COOL(SUPPLY)	6.2-46	WATER	V-5-147	1,6&7,8	OUT	GATE	MOTOR	AUTO	REM MAN	<60	AC	DIRECT	OPEN	OPEN	CLOSE	AS IS	
X-20A	6	DW CLOSED COOL(SUPPLY)	6.2-46	WATER	V-5-165	NONE	IN	CHECK	NONE	AUTO	N/A	N/A	N/A	NONE	OPEN	OPEN	OPEN	N/A	
X-20B	6	DW CLOSED COOL(RETURN)	6.2-46	WATER	V-5-167	1,6&7,8	OUT	GATE	MOTOR	AUTO	REM MAN	<60	AC	DIRECT	OPEN	OPEN	CLOSE	AS IS	
X-20B	6	DW CLOSED COOL(RETURN)	6.2-46	WATER	V-5-166	1,6&7,8	IN	GATE	MOTOR	AUTO	REM MAN	<60	AC	DIRECT	OPEN	OPEN	OPEN	AS IS	
X-20B	3/8"	DW CLOSED COOL (RETURN	6.2-46	WATER	V-5-879	N/A	IN	CHECK	N/A	N/A	N/A	N/A	N/A	N/A	CLOSED	CLOSED	CLOSED	CLOSED	

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CONTAINMENT ISOLATION VALVES / MECHANICAL INTEGRITY

1 PEN	2 SIZE	3 SYSTEM SERVICE	4 FIG	5 FLUID	6 TAG NO	7 SIGNAL	8 LOC	9 TYPE	10 VLVOP	11 PRI ACT	12 SEC ACT	13 MS	14 PWR	15 CR POS	16 NORM	17 SHDW	18 P ACC	19 LOP	20 COMMENT
X-21	2	DW EQP DRN TNK(DISCH)	6.2-49	WATER	V-22-001	1,6,7	OUT	CNTRL	AIR	AUTO	REM MAN	<60	AC	DIRECT	OPEN	OPEN	CLOSE	CLOSE	
X-21	2	DW EQP DRN TNK(DISCH)	6.2-49	WATER	V-22-002	1,6,7	OUT	CNTRL	AIR	AUTO	REM MAN	<60	AC	DIRECT	OPEN	OPEN	CLOSE	CLOSE	
X-21	2	DW SUMP(DISCH)	6.2-49	WATER	V-22-028	1,6,7	OUT	CNTRL	AIR	AUTO	REM MAN	<60	AC	DIRECT	OPEN	OPEN	CLOSE	CLOSE	
X-21	2	DW SUMP(DISCH)	6.2-49	WATER	V-22-029	1,6,7	OUT	CNTRL	AIR	AUTO	REM MAN	<60	AC	DIRECT	OPEN	OPEN	CLOSE	CLOSE	
X-21	1\2	DW EQP DRN TNK(DISCH)	6.2-49	WATER	V-22-751	NONE	OUT	GLOBE	HAND	N/A	N/A	N/A	N/A	NONE	CLOSED	CLOSE	CLOSE	N/A	LOCKED CLOSED
X-21	1\2	DW EQP DRN TNK(DISCH)	6.2-49	WATER	V-22-752	NONE	OUT	GLOBE	HAND	N/A	N/A	N/A	N/A	NONE	CLOSED	CLOSE	CLOSE	N/A	LOCKED CLOSED
X-21	1\2	DW EQP DRN TNK(DISCH)	6.2-49	WATER	V-22-792	NONE	OUT	GLOBE	HAND	N/A	N/A	N/A	N/A	NONE	CLOSED	CLOSE	CLOSE	N/A	LOCKED CLOSED
X-21	1\2	DW EQP DRN TNK(DISCH)	6.2-49	WATER	V-22-793	NONE	OUT	GLOBE	HAND	N/A	N/A	N/A	N/A	NONE	CLOSED	CLOSE	CLOSE	N/A	LOCKED CLOSED
X-21	1\2	DW SUMP (DISCH)	6.2-49	WATER	V-22-753	NONE	OUT	GLOBE	HAND	N/A	N/A	N/A	N/A	NONE	CLOSED	CLOSE	CLOSE	N/A	LOCKED CLOSED
X-21	1\2	DW SUMP (DISCH)	6.2-49	WATER	V-22-754	NONE	OUT	GLOBE	HAND	N/A	N/A	N/A	N/A	NONE	CLOSED	CLOSE	CLOSE	N/A	LOCKED CLOSED
X-21	1\2	DW SUMP (DISCH)	6.2-49	WATER	V-22-794	NONE	OUT	GLOBE	HAND	N/A	N/A	N/A	N/A	NONE	CLOSED	CLOSE	CLOSE	N/A	LOCKED CLOSED
X-21	1\2	DW SUMP (DISCH)	6.2-49	WATER	V-22-795	NONE	OUT	GLOBE	HAND	N/A	N/A	N/A	N/A	NONE	CLOSED	CLOSE	CLOSE	N/A	LOCKED CLOSED
X-22B	1	CONT SPRAY PMP CASE VENT	6.2-55	WATER	V-21-020	NONE	OUT	CHECK	NONE	REV FLO	N/A	N/A	N/A	NONE	CLOSE	CLOSE	OPEN	N/A	
X-22B	6	CONTAINMENT SPRAY	6.2-55	WATER	V-21-017	NONE	OUT	GATE	MOTOR	REM MAN	N/A	<60	AC	DIRECT	OPEN	CLOSE	OPEN	AS IS	TEST LINE
X-22G	6	CONTAINMENT SPRAY	6.2-55	WATER	V-21-013	NONE	OUT	GATE	MOTOR	REM MAN	N/A	<60	AC	DIRECT	OPEN	CLOSE	OPEN	AS IS	TEST LINE
X-22G	1	CONT SPRAY PMP CASE VENT	6.2-55	WATER	V-21-019	NONE	OUT	CHECK	NONE	REV FLO	N/A	N/A	N/A	NONE	CLOSE	CLOSE	OPEN	N/A	

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CONTAINMENT ISOLATION VALVES / MECHANICAL INTEGRITY

1 PEN	2 SIZE	3 SYSTEM SERVICE	4 FIG	5 FLUID	6 TAG NO	7 SIGNAL	8 LOC	9 TYPE	10 VLVOP	11 PRI ACT	12 SEC ACT	13 MS	14 PWR	15 CR POS	16 NORM	17 SHDW	18 P ACC	19 LOP	20 COMMENT
X-22G	3	CONTAINMENT SPRAY	6.2-55	WATER	V-21-077	NONE	OUT	GATE	HAND	N/A	N/A	N/A	N/A	N/A	CLOSE	CLOSE	CLOSE	N/A	LOCKED CLOSED
X-22G	3	CONTAINMENT SPRAY	6.2-55	WATER	V-21-078	NONE	OUT	GATE	HAND	N/A	N/A	N/A	N/A	N/A	CLOSE	CLOSE	CLOSE	N/A	LOCKED CLOSED
X-22G	3\4	CONTAINMENT SPRAY	6.2-55	WATER	V-21-083	NONE	OUT	GATE	HAND	N/A	N/A	N/A	N/A	N/A	CLOSE	CLOSE	CLOSE	N/A	LOCKED CLOSED
X-23	2	DEMINERIALIZED WATER	6.2-50	WATER	N/A	NONE	IN	FLANGE	NONE	N/A	N/A	N/A	N/A	NONE	CLOSE	CLOSE	CLOSE	N/A	BLANK FLANGE
X-23	2	DEMINERIALIZED WATER	6.2-50	WATER	N/A	NONE	OUT	FLANGE	NONE	N/A	N/A	N/A	N/A	NONE	CLOSE	CLOSE	CLOSE	N/A	BLANK FLANGE
X-23	3\4	DEMINERIALIZED WATER	6.2-50	WATER	V-12-217	NONE	OUT	GATE	HAND	N/A	N/A	N/A	N/A	NONE	CLOSE	CLOSE	CLOSE	N/A	LOCKED CLOSED
X-25	1	HI PRESS SEAL LEAK DETECT	6.2-53	WATER	V-37-058	NONE	OUT	GLOBE	HAND	N/A	N/A	N/A	N/A	NONE	OPEN	OPEN	OPEN	N/A	
X-25	1\2	HI PRESS SEAL LEAK DETECT	6.2-53	WATER	V-37-059	NONE	OUT	CHECK	NONE	N/A	N/A	N/A	N/A	NONE	OPEN	OPEN	OPEN	N/A	
X-25	3\4	HI PRESS SEAL LEAK DETECT	6.2-53	WATER	V-37-109	NONE	OUT	GLOBE	HAND	N/A	N/A	N/A	N/A	NONE	CLOSE	CLOSE	CLOSE	N/A	LOCKED CLOSED
X-38A	1	RX VESSEL LVL-WR	6.2-53	WATER	V-130-015	NONE	OUT	GLOBE	HAND	N/A	N/A	N/A	N/A	NONE	OPEN	OPEN	OPEN	N/A	LT-IA-12
X-38A	1	RX VESSEL LVL-WR	6.2-53	WATER	V-130-005	NONE	OUT	CHECK	NONE	N/A	N/A	N/A	N/A	NONE	OPEN	OPEN	OPEN	N/A	LT-IA-12
X-38A	1	RX VESSEL LVL-WR	6.2-53	WATER	V-130-025	NONE	OUT	GLOBE	HAND	N/A	N/A	N/A	N/A	NONE	OPEN	OPEN	OPEN	N/A	LT-IA-12
X-38A	1	RX VESSEL LVL-WR	6.2-53	WATER	V-130-026	NONE	OUT	CHECK	NONE	N/A	N/A	N/A	N/A	NONE	OPEN	OPEN	OPEN	N/A	LT-IA-12
X-38B	1	RX LVL CNTRL(STATIC LEG)	6.2-56	WATER	V-130-013	NONE	OUT	GLOBE	HAND	N/A	N/A	N/A	N/A	NONE	OPEN	OPEN	OPEN	N/A	LIS-RE-18A&C

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CONTAINMENT ISOLATION VALVES / MECHANICAL INTEGRITY

1 PEN	2 SIZE	3 SYSTEM SERVICE	4 FIG	5 FLUID	6 TAG NO	7 SIGNAL	8 LOC	9 TYPE	10 VLVOP	11 PRI ACT	12 SEC ACT	13 MS	14 PWR	15 CR POS	16 NORM	17 SHDW	18 P ACC	19 LOP	20 COMMENT
X-38B	1	RX LVL CNTRL(STATIC LEG)	6.2-56	WATER	V-130-003	NONE	OUT	CHECK	NONE	N/A	N/A	N/A	N/A	NONE	OPEN	OPEN	OPEN	N/A	LIS-RE-18A&C
X-38B	1	RX LVL CNTRL(STATIC LEG)	6.2-53	WATER	V-130-014	NONE	OUT	GLOBE	HAND	N/A	N/A	N/A	N/A	NONE	OPEN	OPEN	OPEN	N/A	LIS-RE-18A&C
X-38B	1	RX LVL CNTRL(STATIC LEG)	6.2-53	WATER	V-130-004	NONE	OUT	CHECK	NONE	N/A	N/A	N/A	N/A	NONE	OPEN	OPEN	OPEN	N/A	LIS-RE-18A&C
X-38C	1	PROTECT SYS LVL IND	6.2-53	WATER	V-130-012A	NONE	OUT	GLOBE	HAND	N/A	N/A	N/A	N/A	NONE	OPEN	OPEN	OPEN	N/A	LT-RE-5A,LT-RE-2A/B
X-38C	1	PROTECT SYS LVL IND	6.2-53	WATER	V-130-002A	NONE	OUT	CHECK	NONE	N/A	N/A	N/A	N/A	NONE	OPEN	OPEN	OPEN	N/A	LT-RE-5A,LT-RE-2A/B
X-38D	1	ISOL COND(COND RETURN)	6.2-39	WATER	V-14-046	NONE	OUT	GLOBE	HAND	N/A	N/A	N/A	N/A	NONE	OPEN	OPEN	OPEN	N/A	DPIS-IB11B1/2
X-38D	1\2	ISOL COND(COND RETURN)	6.2-39	WATER	V-14-054	NONE	OUT	CHECK	NONE	N/A	N/A	N/A	N/A	NONE	OPEN	OPEN	OPEN	N/A	DPIS-IB11B1/2
X-38D	1	ISOL COND(COND RETURN)	6.2-39	WATER	V-14-048	NONE	OUT	GLOBE	HAND	N/A	N/A	N/A	N/A	NONE	OPEN	OPEN	OPEN	N/A	DPIS-IB11B1/2
X-38D	1\2	ISOL COND(COND RETURN)	6.2-39	WATER	V-14-056	NONE	OUT	CHECK	NONE	N/A	N/A	N/A	N/A	NONE	OPEN	OPEN	OPEN	N/A	DPIS-IB11B1/2
X-38E	3/4	ISOL COND(COND SUPPLY)	6.2-39	WATER	V-14-041	NONE	OUT	GLOBE	HAND	N/A	N/A	N/A	N/A	NONE	OPEN	OPEN	OPEN	N/A	DPIS-IB05A1/2
X-38E	1\2	ISOL COND(COND SUPPLY)	6.2-39	WATER	V-14-049	NONE	OUT	CHECK	NONE	N/A	N/A	N/A	N/A	NONE	OPEN	OPEN	OPEN	N/A	DPIS-IB05A1/2
X-38E	3/4	ISOL COND(COND SUPPLY)	6.2-39	WATER	V-14-43	NONE	OUT	GLOBE	HAND	N/A	N/A	N/A	N/A	NONE	OPEN	OPEN	OPEN	N/A	DPIS-1B05A1/2
X-38E	1/2	ISOL COND(COND SUPPLY)	6.2-39	WATER	V-14-51	NONE	OUT	CHECK	NONE	N/A	N/A	N/A	N/A	NONE	OPEN	OPEN	OPEN	N/A	DPIS-1B05A1/2

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CONTAINMENT ISOLATION VALVES / MECHANICAL INTEGRITY

1 PEN	2 SIZE	3 SYSTEM SERVICE	4 FIG	5 FLUID	6 TAG NO	7 SIGNAL	8 LOC	9 TYPE	10 VLVOP	11 PRI ACT	12 SEC ACT	13 MS	14 PWR	15 CR POS	16 NORM	17 SHDW	18 P ACC	19 LOP	20 COMMENT
X-38F	1	PROTECT SYS LVL IND	6.2-53	WATER	V-130-012B	NONE	OUT	GLOBE	HAND	N/A	N/A	N/A	N/A	NONE	OPEN	OPEN	OPEN	N/A	LT-RE-5A,LT-RE-2A/B
X-38F	1	PROTECT SYS LVL IND	6.2-53	WATER	V-130-002B	NONE	OUT	CHECK	NONE	N/A	N/A	N/A	N/A	NONE	OPEN	OPEN	OPEN	N/A	LT-RE-5A,LT-RE-2A/B
X-39A	1	PROTECT SYS LVL IND	6.2-53	WATER	V-130-016B	NONE	OUT	GLOBE	HAND	N/A	N/A	N/A	N/A	NONE	OPEN	OPEN	OPEN	N/A	LI-RE5-19B,LI-622-917
X-39A	1	PROTECT SYS LVL IND	6.2-53	WATER	V-130-006B	NONE	OUT	CHECK	NONE	N/A	N/A	N/A	N/A	NONE	OPEN	OPEN	OPEN	N/A	LI-RE5-19B,LI-622-917
X-39B	1	RX LVL CNTRL(STATIC LEG)	6.2-53	WATER	V-130-018	NONE	OUT	GLOBE	HAND	N/A	N/A	N/A	N/A	NONE	OPEN	OPEN	OPEN	N/A	LIS-RE-18B&D
X-39B	1	RX LVL CNTRL(STATIC LEG)	6.2-53	WATER	V-130-008	NONE	OUT	CHECK	NONE	N/A	N/A	N/A	N/A	NONE	OPEN	OPEN	OPEN	N/A	LIS-RE-18B&D
X-39C	1	RX LVL CNTRL(STATIC LEG)	6.2-53	WATER	V-130-017	NONE	OUT	GLOBE	HAND	N/A	N/A	N/A	N/A	NONE	OPEN	OPEN	OPEN	N/A	LIS-RE-18B&D
X-39C	1	RX LVL CNTRL(STATIC LEG)	6.2-53	WATER	V-130-007	NONE	OUT	CHECK	NONE	N/A	N/A	N/A	N/A	NONE	OPEN	OPEN	OPEN	N/A	LIS-RE-18B&D
X-39D	1	PROTECT SYS LVL IND	6.2-56	WATER	V-130-016A	NONE	OUT	GLOBE	HAND	N/A	N/A	N/A	N/A	NONE	OPEN	OPEN	OPEN	N/A	LI-RE5-19B,LI-622-917
X-39D	1	PROTECT SYS LVL IND	6.2-56	WATER	V-130-006A	NONE	OUT	CHECK	NONE	N/A	N/A	N/A	N/A	NONE	OPEN	OPEN	OPEN	N/A	LI-RE5-19B,LI-622-917
X-40A	1	RECIRC LOOP A D/P(SUCTION)	6.2-53	WATER	V-37-003	NONE	OUT	GLOBE	HAND	N/A	N/A	N/A	N/A	NONE	OPEN	OPEN	OPEN	N/A	DPT-IA50A
X-40A	1\2	RECIRC LOOP A D/P(SUCTION)	6.2-53	WATER	V-37-007	NONE	OUT	CHECK	NONE	N/A	N/A	N/A	N/A	NONE	OPEN	OPEN	OPEN	N/A	DPT-IA50A

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CONTAINMENT ISOLATION VALVES / MECHANICAL INTEGRITY

1 PEN	2 SIZE	3 SYSTEM SERVICE	4 FIG	5 FLUID	6 TAG NO	7 SIGNAL	8 LOC	9 TYPE	10 VLVOP	11 PRI ACT	12 SEC ACT	13 MS	14 PWR	15 CR POS	16 NORM	17 SHDW	18 P ACC	19 LOP	20 COMMENT
X-40B	1	RECIRC LOOP B D/P(SUCTION)	6.2-53	WATER	V-37-014	NONE	OUT	GLOBE	HAND	N/A	N/A	N/A	N/A	NONE	OPEN	OPEN	OPEN	N/A	DPT-IA50B
										N/A	N/A	N/A	N/A					N/A	
X-40B	1\2	RECIRC LOOP B D/P(SUCTION)	6.2-53	WATER	V-37-018	NONE	OUT	CHECK	NONE	N/A	N/A	N/A	N/A	NONE	OPEN	OPEN	OPEN	N/A	DPT-IA50A
X-40C	1	RECIRC LOOP C D/P(SUCTION)	6.2-53	WATER	V-37-025	NONE	OUT	GLOBE	HAND	N/A	N/A	N/A	N/A	NONE	OPEN	OPEN	OPEN	N/A	DPT-IA50C
X-40C	1\2	RECIRC LOOP C D/P(SUCTION)	6.2-53	WATER	V-37-029	NONE	OUT	CHECK	NONE	N/A	N/A	N/A	N/A	NONE	OPEN	OPEN	OPEN	N/A	DPT-IA50C
X-40D	1	RECIRC LOOP D D/P(SUCTION)	6.2-53	WATER	V-37-036	NONE	OUT	GLOBE	HAND	N/A	N/A	N/A	N/A	NONE	OPEN	OPEN	OPEN	N/A	DPT-IA50D
X-40D	1\2	RECIRC LOOP D D/P(SUCTION)	6.2-53	WATER	V-37-040	NONE	OUT	CHECK	NONE	N/A	N/A	N/A	N/A	NONE	OPEN	OPEN	OPEN	N/A	DPT-IA50D
X-40E	1	RECIRC LOOP E D/P(SUCTION)	6.2-53	WATER	V-37-047	NONE	OUT	GLOBE	HAND	N/A	N/A	N/A	N/A	NONE	OPEN	OPEN	OPEN	N/A	DPT-IA50E
X-40E	1\2	RECIRC LOOP E D/P(SUCTION)	6.2-53	WATER	V-37-051	NONE	OUT	CHECK	NONE	N/A	N/A	N/A	N/A	NONE	OPEN	OPEN	OPEN	N/A	DPT-IA50E
X-40F	1	RECIRC LOOP A D/P(DISCH)	6.2-53	WATER	V-37-004	NONE	OUT	GLOBE	HAND	N/A	N/A	N/A	N/A	NONE	OPEN	OPEN	OPEN	N/A	DPT-IA50A
X-40F	1\2	RECIRC LOOP A D/P(DISCH)	6.2-53	WATER	V-37-008	NONE	OUT	CHECK	NONE	N/A	N/A	N/A	N/A	NONE	OPEN	OPEN	OPEN	N/A	DPT-IA50A
X-40G	1	RECIRC LOOP B D/P(DISCH)	6.2-53	WATER	V-37-015	NONE	OUT	GLOBE	HAND	N/A	N/A	N/A	N/A	NONE	OPEN	OPEN	OPEN	N/A	DPT-IA50B
X-40G	1\2	RECIRC LOOP B D/P(DISCH)	6.2-53	WATER	V-37-019	NONE	OUT	CHECK	NONE	N/A	N/A	N/A	N/A	NONE	OPEN	OPEN	OPEN	N/A	DPT-IA50B
X-40H	1	RECIRC LOOP C D/P(DISCH)	6.2-53	WATER	V-37-026	NONE	OUT	GLOBE	HAND	N/A	N/A	N/A	N/A	NONE	OPEN	OPEN	OPEN	N/A	DPT-IA50C
X-40H	1\2	RECIRC LOOP C D/P(DISCH)	6.2-53	WATER	V-37-030	NONE	OUT	CHECK	NONE	N/A	N/A	N/A	N/A	NONE	OPEN	OPEN	OPEN	N/A	DPT-IA50C

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CONTAINMENT ISOLATION VALVES / MECHANICAL INTEGRITY

1 PEN	2 SIZE	3 SYSTEM SERVICE	4 FIG	5 FLUID	6 TAG NO	7 SIGNAL	8 LOC	9 TYPE	10 VLVOP	11 PRI ACT	12 SEC ACT	13 MS	14 PWR	15 CR POS	16 NORM	17 SHDW	18 P ACC	19 LOP	20 COMMENT
X-40I	1	RECIRC LOOP D D/P(DISCH)	6.2-53	WATER	V-37-037	NONE	OUT	GLOBE	HAND	N/A	N/A	N/A	N/A	NONE	OPEN	OPEN	OPEN	N/A	DPT-IA50D
X-40I	1\2	RECIRC LOOP D D/P(DISCH)	6.2-53	WATER	V-37-041	NONE	OUT	CHECK	NONE	N/A	N/A	N/A	N/A	NONE	OPEN	OPEN	OPEN	N/A	DPT-IA50D
X-40J	1	RECIRC LOOP E D/P(DISCH)	6.2-53	WATER	V-37-048	NONE	OUT	GLOBE	HAND	N/A	N/A	N/A	N/A	NONE	OPEN	OPEN	OPEN	N/A	DPT-IA50E
X-40J	1\2	RECIRC LOOP E D/P(DISCH)	6.2-53	WATER	V-37-052	NONE	OUT	CHECK	NONE	N/A	N/A	N/A	N/A	NONE	OPEN	OPEN	OPEN	N/A	DPT-IA50E
X-42	1	REACTOR LEVEL IND	6.2-53	WATER	V-130-022B	NONE	OUT	GLOBE	HAND	N/A	N/A	N/A	N/A	NONE	OPEN	OPEN	OPEN	N/A	LIS-RE-18B&D
X-42	1	REACTOR LEVEL IND	6.2-53	WATER	V-130-021B	NONE	OUT	CHECK	NONE	N/A	N/A	N/A	N/A	NONE	OPEN	OPEN	OPEN	N/A	LIS-RE-18B&D
X-45A	1\4	TIP(INCORE CAL TUBE)	6.2-50	N2	V-623-001	1,6,7	OUT	BALL	SOLND	AUTO	REM MAN	<60	AC	DIRECT	CLOSE	CLOSE	CLOSE	CLOSE	FOR INCORE MAP
X-45A	1\4	TRAVEL INCORE PROBE(TIP)	6.2-50	N2	V-623-005	NONE	OUT	SHEAR	EXPLSV	N/A	N/A	N/A	DC	DIRECT	OPEN	OPEN	OPEN	OPEN	EXPLSV MAN INITIATE
X-45A	1\4	TIP PURGE	6.2-50	N2	V-23-070	1,6,7	OUT	GATE	SOLND	AUTO	REM MAN	<60	AC	DIRECT	OPEN	OPEN	CLOSE	CLOSE	
X-45A	3\8	TIP PURGE	6.2-50	N2	V-23-071	NONE	OUT	CHECK	NONE	N/A	N/A	N/A	N/A	NONE	OPEN	CLOSE	CLOSE	N/A	
X-45A	3\8	TIP PURGE	6.2-50	N2	V-23-146	NONE	OUT	GLOBE	HAND	N/A	N/A	N/A	N/A	NONE	CLOSE	CLOSE	CLOSE	N/A	LOCKED CLOSED
X-45A	3\8	TIP PURGE	6.2-50	N2	V-23-151	NONE	OUT	GLOBE	HAND	N/A	N/A	N/A	N/A	NONE	OPEN	OPEN	OPEN	N/A	
X-45B	1\4	TIP (INCORE CAL TUBE	6-2-50	N2	V-623-002	1,6,7	OUT	BALL	SOLND	AUTO	REM MAN	<60	AC	DIRECT	CLOSE	CLOSE	CLOSE	CLOSE	FOR INCORE MAP
X-45B	1\4	TRAVEL INCORE PROBE(TIP)	6.2-50	N2	V-623-006	NONE	OUT	SHEAR	EXPLSV	N/A	N/A	N/A	DC	DIRECT	OPEN	OPEN	OPEN	OPEN	EXPLSV MAN INITIATE
X-45C	1	STM FLO NOZZLE IMPULSE	6.2-40	STEAM	V-1-178	NONE	OUT	GLOBE	HAND	N/A	N/A	N/A	N/A	NONE	OPEN	OPEN	OPEN	N/A	DPIS-RE22A/B/C/D

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CONTAINMENT ISOLATION VALVES / MECHANICAL INTEGRITY

1 PEN	2 SIZE	3 SYSTEM SERVICE	4 FIG	5 FLUID	6 TAG NO	7 SIGNAL	8 LOC	9 TYPE	10 VLVOP	11 PRI ACT	12 SEC ACT	13 MS	14 PWR	15 CR POS	16 NORM	17 SHDW	18 P ACC	19 LOP	20 COMMENT
X-45C	1\2	STM FLO NOZZLE IMPULSE	6.2-40	STEAM	V-1-180	NONE	OUT	CHECK	NONE	N/A	N/A	N/A	N/A	NONE	OPEN	OPEN	OPEN	N/A	DPIS-RE22A/B/C/D
X-45C	1	STM FLO NOZZLE IMPULSE	6.2-40	STEAM	V-1-179	NONE	OUT	GLOBE	HAND	N/A	N/A	N/A	N/A	NONE	OPEN	OPEN	OPEN	N/A	DPIS-RE22A/B/C/D
X-45C	1\2	STM FLO NOZZLE IMPULSE	6.2-40	STEAM	V-1-181	NONE	OUT	CHECK	NONE	N/A	N/A	N/A	N/A	NONE	OPEN	OPEN	OPEN	N/A	DPIS-RE22A/B/C/D
X-45D	1	CORE DELTA P	6.2-56	WATER	V-20-154	NONE	OUT	GLOBE	HAND	N/A	N/A	N/A	N/A	NONE	OPEN	OPEN	OPEN	N/A	DPIS-RV30A
X-45D	1\2	CORE DELTA P	6.2-56	WATER	V-20-172	NONE	OUT	CHECK	NONE	N/A	N/A	N/A	N/A	NONE	OPEN	OPEN	OPEN	N/A	DPIS-RV30A
X-45D	1	CORE DELTA P	6.2-56	WATER	V-20-155	NONE	OUT	GLOBE	HAND	N/A	N/A	N/A	N/A	NONE	OPEN	OPEN	OPEN	N/A	DPIS-RV30B
X-45D	1\2	CORE DELTA P	6.2-56	WATER	V-20-173	NONE	OUT	CHECK	NONE	N/A	N/A	N/A	N/A	NONE	OPEN	OPEN	OPEN	N/A	DPIS-RV30B
X-45D	1	CORE DELTA P	6.2-53	WATER	V-130-019	NONE	OUT	GLOBE	HAND	N/A	N/A	N/A	N/A	NONE	OPEN	OPEN	OPEN	N/A	DPIT-IA7,DPT5-IA91A
X-45D	1	CORE DELTA P	6.2-53	WATER	V-130-009	NONE	OUT	CHECK	NONE	N/A	N/A	N/A	N/A	NONE	OPEN	OPEN	OPEN	N/A	DPIT-IA7,DPT5-IA91A
X-45D	1	CORE DELTA P	6.2-53	WATER	V-130-020	NONE	OUT	GLOBE	HAND	N/A	N/A	N/A	N/A	NONE	OPEN	OPEN	OPEN	N/A	DPIT-IA7,DPT5-IA91A
X-45D	1	CORE DELTA P	6.2-53	WATER	V-130-010	NONE	OUT	CHECK	NONE	N/A	N/A	N/A	N/A	NONE	OPEN	OPEN	OPEN	N/A	DPIT-IA7,DPT5-IA91A
X-45E	1\4	TIP(INCORE CAL TUBE)	6.2-50	N2	V-623-003	1,6,7	OUT	BALL	SOLND	AUTO	REM MAN	<60	AC	DIRECT	CLOSE	CLOSE	CLOSE	CLOSE	FOR INCORE MAP
X-45E	1\4	TRAVEL INCORE PROBE(TIP)	6.2-50	N2	V-623-007	NONE	OUT	SHEAR	EXPLSV	N/A	N/A	N/A	DC	DIRECT	OPEN	OPEN	OPEN	OPEN	explsv manually initiated
X-45F	1\4	TIP(INCORE CAL TUBE)	6.2-50	N2	V-623-004	1,6,7	OUT	BALL	SOLND	AUTO	REM MAN	<60	AC	DIRECT	CLOSE	CLOSE	CLOSE	CLOSE	FOR INCORE MAP
X-45F	1\4	TRAVEL INCORE PROBE(TIP)	6.2-50	N2	V-623-008	NONE	OUT	SHEAR	EXPLSV	N/A	N/A	N/A	DC	DIRECT	OPEN	OPEN	OPEN	OPEN	explsi manually initiated
X-45G	1	STM FLO NOZZLE IMPULSE	6.2-40	STEAM	V-1-182	NONE	OUT	GLOBE	HAND	N/A	N/A	N/A	N/A	NONE	OPEN	OPEN	OPEN	N/A	DPIS-RE22E/F/G/H

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CONTAINMENT ISOLATION VALVES / MECHANICAL INTEGRITY

1 PEN	2 SIZE	3 SYSTEM SERVICE	4 FIG	5 FLUID	6 TAG NO	7 SIGNAL	8 LOC	9 TYPE	10 VLVOP	11 PRI ACT	12 SEC ACT	13 MS	14 PWR	15 CR POS	16 NORM	17 SHDW	18 P ACC	19 LOP	20 COMMENT
X-45G	1\2	STM FLO NOZZLE IMPULSE	6.2-40	STEAM	V-1-184	NONE	OUT	CHECK	NONE	N/A	N/A	N/A	N/A	NONE	OPEN	OPEN	OPEN	N/A	DPIS-RE22E/F/G/H
X-45G	1	STM FLO NOZZLE IMPULSE	6.2-40	STEAM	V-1-183	NONE	OUT	GLOBE	HAND	N/A	N/A	N/A	N/A	NONE	OPEN	OPEN	OPEN	N/A	DPIS-RE22E/F/G/H
X-45G	1\2	STM FLO NOZZLE IMPULSE	6.2-40	STEAM	V-1-185	NONE	OUT	CHECK	NONE	N/A	N/A	N/A	N/A	NONE	OPEN	OPEN	OPEN	N/A	DPIS-RE22E/F/G/H
X-49	3\4	CONT PART MON (RETURN)	6.2-55	AIR	V-38-016	1,6,7	OUT	GLOBE	SOLND	AUTO	N/A	N/A	AC	INDRCT	OPEN	OPEN	CLOSE	CLOSE	
X-49	3\4	CONT PART MON (RETURN)	6.2-55	AIR	V-38-017	1,6,7	OUT	GLOBE	SOLND	AUTO	N/A	N/A	AC	INDRCT	OPEN	OPEN	CLOSE	CLOSE	
X-49	1\2	CONT PART MON (RETURN)	6.2-55	AIR	V-38-018	NONE	OUT	GATE	HAND	NONE	N/A	N/A	N/A	NONE	CLOSE	CLOSE	CLOSE	N/A	LOCKED CLOSED
X-49	1\2	CONT PART MON (RETURN)	6.2-55	AIR	V-38-019	NONE	OUT	GATE	HAND	NONE	N/A	N/A	N/A	NONE	CLOSE	CLOSE	CLOSE	N/A	LOCKED CLOSED
X-50A	3\4	RECIRC PMP E SEAL CAV#2	6.2-53	WATER	V-37-065	NONE	OUT	GLOBE	HAND	N/A	N/A	N/A	N/A	NONE	OPEN	OPEN	OPEN	N/A	
X-50A	1\2	RECIRC PMP E SEAL CAV#2	6.2-53	WATER	V-37-075	NONE	OUT	CHECK	NONE	N/A	N/A	N/A	N/A	NONE	OPEN	OPEN	OPEN	N/A	
X-50A	1\2	RECIRC PMP E SEAL CAV#2	6.2-53	WATER	V-37-107	NONE	OUT	GLOBE	HAND	N/A	N/A	N/A	N/A	NONE	CLOSE	CLOSE	CLOSE	N/A	LOCKED CLOSED
X-50B	3\4	RECIRC PMP E SEAL CAV#1	6.2-53	WATER	V-37-064	NONE	OUT	GLOBE	HAND	N/A	N/A	N/A	N/A	NONE	OPEN	OPEN	OPEN	N/A	PIT-IA67E
X-50B	1\2	RECIRC PMP E SEAL CAV#1	6.2-53	WATER	V-37-074	NONE	OUT	CHECK	NONE	N/A	N/A	N/A	N/A	NONE	OPEN	OPEN	OPEN	N/A	PIT-IA67E
X-50B	1\2	RECIRC PMP E SEAL CAV#1	6.2-53	WATER	V-37-108	NONE	OUT	GLOBE	HAND	N/A	N/A	N/A	N/A	NONE	CLOSE	CLOSE	CLOSE	N/A	LOCKED CLOSED
X-50C	3\4	RECIRC PMP D SEAL CAV#2	6.2-53	WATER	V-37-062	NONE	OUT	GLOBE	HAND	N/A	N/A	N/A	N/A	NONE	OPEN	OPEN	OPEN	N/A	PIT-IA67D

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CONTAINMENT ISOLATION VALVES / MECHANICAL INTEGRITY

1 PEN	2 SIZE	3 SYSTEM SERVICE	4 FIG	5 FLUID	6 TAG NO	7 SIGNAL	8 LOC	9 TYPE	10 VLVOP	11 PRI ACT	12 SEC ACT	13 MS	14 PWR	15 CR POS	16 NORM	17 SHDW	18 P ACC	19 LOP	20 COMMENT
X-50C	1\2	RECIRC PMP D SEAL CAV#2	6.2-53	WATER	V-37-072	NONE	OUT	CHECK	NONE	N/A	N/A	N/A	N/A	NONE	OPEN	OPEN	OPEN	N/A	PIT-IA67D
X-50C	3\4	RECIRC PMP D SEAL CAV#2	6.2-53	WATER	V-37-106	NONE	OUT	GLOBE	HAND	N/A	N/A	N/A	N/A	NONE	CLOSE	CLOSE	CLOSE	N/A	LOCKED CLOSED
X-50D	3\4	RECIRC PMP D SEAL CAV#1	6.2-53	WATER	V-37-063	NONE	OUT	GLOBE	HAND	N/A	N/A	N/A	N/A	NONE	OPEN	OPEN	OPEN	N/A	PT-IA52D
X-50D	1\2	RECIRC PMP D SEAL CAV#1	6.2-53	WATER	V-37-073	NONE	OUT	CHECK	NONE	N/A	N/A	N/A	N/A	NONE	OPEN	OPEN	OPEN	N/A	PT-IA52D
X-50D	3\4	RECIRC PMP D SEAL CAV#1	6.2-53	WATER	V-37-105	NONE	OUT	GLOBE	HAND	N/A	N/A	N/A	N/A	NONE	CLOSE	CLOSE	CLOSE	N/A	LOCKED CLOSED
X-50E	3\4	RECIRC PMP C SEAL CAV#2	6.2-53	WATER	V-37-060	NONE	OUT	GLOBE	HAND	N/A	N/A	N/A	N/A	NONE	OPEN	OPEN	OPEN	N/A	
X-50E	1\2	RECIRC PMP C SEAL CAV#2	6.2-53	WATER	V-37-070	NONE	OUT	CHECK	NONE	N/A	N/A	N/A	N/A	NONE	OPEN	OPEN	OPEN	N/A	
X-50E	1\2	RECIRC PMP C SEAL CAV#2	6.2-53	WATER	V-37-104	NONE	OUT	GLOBE	HAND	N/A	N/A	N/A	N/A	NONE	CLOSE	CLOSE	CLOSE	N/A	LOCKED CLOSED
X-50F	3\4	RECIRC PMP C SEAL CAV#1	6.2-53	WATER	V-37-061	NONE	OUT	GLOBE	HAND	N/A	N/A	N/A	N/A	NONE	OPEN	OPEN	OPEN	N/A	
X-50F	1\2	RECIRC PMP C SEAL CAV#1	6.2-53	WATER	V-37-071	NONE	OUT	CHECK	NONE	N/A	N/A	N/A	N/A	NONE	OPEN	OPEN	OPEN	N/A	
X-50F	1\2	RECIRC PMP C SEAL CAV#1	6.2-53	WATER	V-37-103	NONE	OUT	GLOBE	HAND	N/A	N/A	N/A	N/A	NONE	CLOSE	CLOSE	CLOSE	N/A	LOCKED CLOSED
X-50G	3\4	RECIRC PMP B SEAL CAV#2	6.2-53	WATER	V-37-078	NONE	OUT	GLOBE	HAND	N/A	N/A	N/A	N/A	NONE	OPEN	OPEN	OPEN	N/A	
X-50G	1\2	RECIRC PMP B SEAL CAV#2	6.2-53	WATER	V-37-068	NONE	OUT	CHECK	NONE	N/A	N/A	N/A	N/A	NONE	OPEN	OPEN	OPEN	N/A	
X-50G	1\2	RECIRC PMP B SEAL CAV#2	6.2-53	WATER	V-37-101	NONE	OUT	GLOBE	HAND	N/A	N/A	N/A	N/A	NONE	CLOSE	CLOSE	CLOSE	N/A	LOCKED CLOSED

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CONTAINMENT ISOLATION VALVES / MECHANICAL INTEGRITY

1 PEN	2 SIZE	3 SYSTEM SERVICE	4 FIG	5 FLUID	6 TAG NO	7 SIGNAL	8 LOC	9 TYPE	10 VLVOP	11 PRI ACT	12 SEC ACT	13 MS	14 PWR	15 CR POS	16 NORM	17 SHDW	18 P ACC	19 LOP	20 COMMENT
X-50H	3\4	RECIRC PMP B SEAL CAV#1	6.2-53	WATER	V-37-079	NONE	OUT	GLOBE	HAND	N/A	N/A	N/A	N/A	NONE	OPEN	OPEN	OPEN	N/A	
X-50H	1\2	RECIRC PMP B SEAL CAV#1	6.2-53	WATER	V-37-069	NONE	OUT	CHECK	NONE	N/A	N/A	N/A	N/A	NONE	OPEN	OPEN	OPEN	N/A	
X-50H	1\2	RECIRC PMP B SEAL CAV#1	6.2-53	WATER	V-37-102	NONE	OUT	GLOBE	HAND	N/A	N/A	N/A	N/A	NONE	CLOSE	CLOSE	CLOSE	N/A	LOCKED CLOSED
X-50I	3\4	RECIRC PMP A SEAL CAV#2	6.2-53	WATER	V-37-076	NONE	OUT	GLOBE	HAND	N/A	N/A	N/A	N/A	NONE	OPEN	OPEN	OPEN	N/A	
X-50I	1\2	RECIRC PMP A SEAL CAV#2	6.2-53	WATER	V-37-066	NONE	OUT	CHECK	NONE	N/A	N/A	N/A	N/A	NONE	OPEN	OPEN	OPEN	N/A	
X-50I	1\2	RECIRC PMP A SEAL CAV#2	6.2-53	WATER	V-37-100	NONE	OUT	GLOBE	HAND	N/A	N/A	N/A	N/A	NONE	CLOSE	CLOSE	CLOSE	N/A	LOCKED CLOSED
X-50J	3\4	RECIRC PMP A SEAL CAV#1	6.2-53	WATER	V-37-077	NONE	OUT	GLOBE	HAND	N/A	N/A	N/A	N/A	NONE	OPEN	OPEN	OPEN	N/A	
X-50J	1\2	RECIRC PMP A SEAL CAV#1	6.2-53	WATER	V-37-067	NONE	OUT	CHECK	NONE	N/A	N/A	N/A	N/A	NONE	OPEN	OPEN	OPEN	N/A	
X-50J	1\2	RECIRC PMP A SEAL CAV#1	6.2-53	WATER	V-37-099	NONE	OUT	GLOBE	HAND	N/A	N/A	N/A	N/A	NONE	CLOSE	CLOSE	CLOSE	N/A	LOCKED CLOSED
X-50N	3\4	DRYWELL PRESSURE IND	6.2-53	WATER	V-38-002	NONE	OUT	GLOBE	HAND	N/A	N/A	N/A	N/A	NONE	OPEN	OPEN	OPEN	N/A	
X-50Z	3\4	DRYWELL PRESSURE IND	6.2-53	WATER	V-38-003	NONE	OUT	GLOBE	HAND	N/A	N/A	N/A	N/A	NONE	OPEN	OPEN	OPEN	N/A	
X-51A	12	TORUS TO VENT EXHAUST	6.2-54	AIR	V-28-017	1,6,7	OUT	BFLY	AIR	AUTO	REM MAN	<60	AIR	DIRECT	CL/OP	CLOSE	CLOSE	CLOSE	30 DEG MAX OPEN
X-51A	12	TORUS TO VENT EXHAUST	6.2-54	AIR	V-28-018	1,6,7	OUT	BFLY	AIR	AUTO	REM MAN	<60	AIR	DIRECT	CL/OP	CLOSE	CLOSE	CLOSE	30 DEG MAX OPEN
X-51A	2	TORUS TO VENT EXHAUST	6.2-54	AIR	V-28-047	1,6,7	OUT	GLOBE	AIR	AUTO	REM MAN	<60	AIR	DIRECT	CL/OP	CLOSE	CLOSE	CLOSE	

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CONTAINMENT ISOLATION VALVES / MECHANICAL INTEGRITY

1 PEN	2 SIZE	3 SYSTEM SERVICE	4 FIG	5 FLUID	6 TAG NO	7 SIGNAL	8 LOC	9 TYPE	10 VLVOP	11 PRI ACT	12 SEC ACT	13 MS	14 PWR	15 CR POS	16 NORM	17 SHDW	18 P ACC	19 LOP	20 COMMENT
X-51A	3\4	LEAK RATE TESTING	6.2-54	N/A	V-28-069	NONE	OUT	GLOBE	HAND	N/A	N/A	N/A	N/A	NONE	CLOSE	CLOSE	CLOSE	CLOSE	LOCKED CLOSED
X-51A	3\4	LEAK RATE TESTING	6.2-54	N/A	V-28-070	NONE	OUT	GLOBE	HAND	N/A	N/A	N/A	N/A	NONE	CLOSE	CLOSE	CLOSE	CLOSE	LOCKED CLOSED
X-51B	4	CONTAINMENT SPRAY	6.2-55	WATER	V-21-018	1,6,7	OUT	GATE	MOTOR	AUTO	REM MAN	<60	AC	DIRECT	CLOSE	CLOSE	OPEN	AS IS	
X-51D	6	CORE SPRAY TEST LINE	6.2-55	WATER	V-20-027	1,6,7	OUT	GLOBE	MOTOR	REM MAN	N/A	<30	AC	DIRECT	CLOSE	CLOSE	CLOSE	AS IS	
X-51D	6	CORE SPRAY TEST LINE	6.2-55	WATER	V-20-030	NONE	OUT	CHECK	NONE	REV FLO	N/A	N/A	N/A	NONE	CLOSE	CLOSE	OPEN	N/A	
X-51E	2	TORUS LEVEL	6.2-55	WATER	V-38-004	NONE	OUT	GATE	HAND	N/A	N/A	N/A	N/A	NONE	OPEN	OPEN	OPEN	N/A	LT-37, 38
X-51F	6	CORE SPRAY TEST LINE	6.2-55	WATER	V-20-026	1,6,7	OUT	GLOBE	MOTOR	REM MAN	N/A	<30	AC	DIRECT	CLOSE	CLOSE	CLOSE	AS IS	
X-51F	6	CORE SPRAY TEST LINE	6.2-55	WATER	V-20-031	NONE	OUT	CHECK	NONE	REV FLO	N/A	N/A	N/A	NONE	CLOSE	CLOSE	OPEN	N/A	
X-51G	2	CONT SPRAY (TEST AIR CONNECTION)	6.2-55	AIR	V-21-134	NONE	OUT	GLOBE	HAND	N/A	N/A	N/A	N/A	NONE	CLOSE	CLOSE	CLOSE	N/A	LOCKED CLOSED
X-51G	2	CONT SPRAY (TEST AIR CONNECTION)	6.2-55	AIR	V-21-133	NONE	OUT	GLOBE	HAND	N/A	N/A	N/A	N/A	NONE	CLOSE	CLOSE	CLOSE	N/A	LOCKED CLOSED
X-51G	4	CONT SPRAY	6.2-55	WATER	V-21-015	1,6,7	OUT	GATE	MOTOR	AUTO	REM MAN	<60	AC	DIRECT	CLOSE	CLOSE	OPEN	AS IS	
X-52A/B	24	DW HEAD/MANHOLE SEAL	N/A	N/A	N/A	N/A	OUT	N/A	N/A	N/A	N/A	N/A	N/A	NONE	CLOSE	CLOSE	CLOSE	N/A	BOLTED CLOSE
X-53A	36	TORUS NORTH MANHOLE SEAL	N/A	N/A	N/A	N/A	OUT	N/A	N/A	N/A	N/A	N/A	N/A	NONE	CLOSE	CLOSE	CLOSE	N/A	BOLTED CLOSE
X-53B	36	TORUS SOUTH MANHOLE SEAL	N/A	N/A	N/A	N/A	OUT	N/A	N/A	N/A	N/A	N/A	N/A	NONE	CLOSE	CLOSE	CLOSE	N/A	BOLTED CLOSE

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CONTAINMENT ISOLATION VALVES / MECHANICAL INTEGRITY

1 PEN	2 SIZE	3 SYSTEM SERVICE	4 FIG	5 FLUID	6 TAG NO	7 SIGNAL	8 LOC	9 TYPE	10 VLVOP	11 PRI ACT	12 SEC ACT	13 MS	14 PWR	15 CR POS	16 NORM	17 SHDW	18 P ACC	19 LOP	20 COMMENT
X-56	1	TORUS PRESSURE	6.2-52	AIR	V-38-1001	NONE	OUT	GATE	HAND	N/A	N/A	N/A	N/A	NONE	OPEN	OPEN	OPEN	N/A	DTDP 1-1
X-57	1\4	TORUS O2 SAMPLE	6.2-52	AIR	V-38-023	1,6,7	OUT	GATE	SOLND	AUTO	REM MAN	N/A	AC	INDRCT	OPEN	OPEN	CLOSE	CLOSE	
X-57	1\4	TORUS O2 SAMPLE	6.2-52	AIR	V-38-022	1,6,7	OUT	3-WAY	SOLND	AUTO	REM MAN	N/A	AC	INDRCT	OPEN	OPEN	CLOSE	CLOSE	
X-59	1	H2/O2 SAMPLE SUPPLY	6.2-57	AIR	V-38-037	NONE	OUT	GLOBE	SOLND	REM MAN	N/A	N/A	AC	DIRECT	CLOSE	CLOSE	OPEN	CLOSE	
X-59	1	H2/O2 SAMPLE SUPPLY	6.2-57	AIR	V-38-038	NONE	OUT	GLOBE	SOLND	REM MAN	N/A	N/A	AC	DIRECT	CLOSE	CLOSE	OPEN	CLOSE	
X-59	1	H2/O2 SAMPLE SUPPLY	6.2-57	AIR	V-38-157	NONE	OUT	GLOBE	HAND	N/A	N/A	N/A	N/A	NONE	CLOSE	CLOSE	CLOSE	N/A	LOCKED CLOSED
X-59	1\2	H2/O2 SAMPLE SUPPLY	6.2-57	AIR	V-38-161	NONE	OUT	GLOBE	HAND	N/A	N/A	N/A	N/A	NONE	CLOSE	CLOSE	CLOSE	N/A	LOCKED CLOSED
X-60A	1	H2/O2 SAMPLE RETURN	6.2-57	AIR	V-38-040	NONE	OUT	GLOBE	SOLND	REM MAN	N/A	N/A	AC	DIRECT	CLOSE	CLOSE	OPEN	CLOSE	
X-60A	1	H2/O2 SAMPLE RETURN	6.2-57	AIR	V-38-158	NONE	OUT	GLOBE	HAND	N/A	N/A	N/A	N/A	NONE	CLOSE	CLOSE	CLOSE	N/A	LOCKED CLOSED
X-60A	1	H2/O2 SAMPLE RETURN	6.2-57	AIR	V-38-039	NONE	OUT	GLOBE	SOLND	REM MAN	N/A	N/A	AC	DIRECT	CLOSE	CLOSE	OPEN	CLOSE	
X-60A	1\2	H2/O2 SAMPLE RETURN	6.2-57	AIR	V-38-162	NONE	OUT	GLOBE	HAND	N/A	N/A	N/A	N/A	NONE	CLOSE	CLOSE	CLOSE	N/A	LOCKED CLOSED
X-60F	1	H2/O2 SAMPLE SUPPLY	6.2-57	AIR	V-38-043	NONE	OUT	GLOBE	SOLND	REM MAN	N/A	N/A	AC	DIRECT	CLOSE	CLOSE	OPEN	CLOSE	
X-60F	1	H2/O2 SAMPLE SUPPLY	6.2-57	AIR	V-38-159	NONE	OUT	GLOBE	HAND	N/A	N/A	N/A	N/A	NONE	CLOSE	CLOSE	CLOSE	N/A	LOCKED CLOSED
X-60F	1	H2/O2 SAMPLE SUPPLY	6.2-57	AIR	V-38-041	NONE	OUT	GLOBE	SOLND	REM MAN	N/A	N/A	AC	DIRECT	CLOSE	CLOSE	OPEN	CLOSE	
X-60F	1\2	H2/O2 SAMPLE SUPPLY	6.2-57	AIR	V-38-164	NONE	OUT	GLOBE	HAND	N/A	N/A	N/A	N/A	NONE	CLOSE	CLOSE	CLOSE	N/A	LOCKED CLOSED

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CONTAINMENT ISOLATION VALVES / MECHANICAL INTEGRITY

1 PEN	2 SIZE	3 SYSTEM SERVICE	4 FIG	5 FLUID	6 TAG NO	7 SIGNAL	8 LOC	9 TYPE	10 VLVOP	11 PRI ACT	12 SEC ACT	13 MS	14 PWR	15 CR POS	16 NORM	17 SHDW	18 P ACC	19 LOP	20 COMMENT
X-60G	1	H2/O2 SAMPLE RETURN	6.2-57	AIR	V-38-046	NONE	OUT	GLOBE	SOLND	REM MAN	N/A	N/A	AC	DIRECT	CLOSE	CLOSE	OPEN	CLOSE	
X-60G	1	H2/O2 SAMPLE RETURN	6.2-57	AIR	V-38-160	NONE	OUT	GLOBE	HAND	N/A	N/A	N/A	N/A	NONE	CLOSE	CLOSE	CLOSE	N/A	LOCKED CLOSED
X-60G	1	H2/O2 SAMPLE RETURN	6.2-57	AIR	V-38-044	NONE	OUT	GLOBE	SOLND	REM MAN	N/A	N/A	AC	DIRECT	CLOSE	CLOSE	OPEN	CLOSE	
X-60G	1\2	H2/O2 SAMPLE RETURN	6.2-57	AIR	V-38-163	NONE	OUT	GLOBE	HAND	NONE	N/A	N/A	N/A	NONE	CLOSE	CLOSE	CLOSE	N/A	LOCKED CLOSED
X-61	3	CRD HYDRAULIC SUPPLY	6.2-47	WATER	V-15-027	NONE	OUT	CHECK	NONE	REV FLO	N/A	N/A	N/A	NONE	OPEN	OPEN	OPEN	N/A	
X-61	3	CRD HYDRAULIC SUPPLY	6.2-47	WATER	V-15-028	NONE	IN	CHECK	NONE	REV FLO	N/A	N/A	N/A	NONE	OPEN	OPEN	OPEN	N/A	
X-61	3\4	CRD HYDRAULIC SUPPLY	6.2-47	WATER	V-15-032	NONE	OUT	GLOBE	HAND	N/A	N/A	N/A	N/A	NONE	CLOSE	CLOSE	CLOSE	N/A	LOCKED CLOSED
X-61	3\4	CRD HYDRAULIC SUPPLY	6.2-47	WATER	V-15-077	NONE	OUT	GLOBE	HAND	N/A	N/A	N/A	N/A	NONE	CLOSE	CLOSE	CLOSE	N/A	LOCKED CLOSED
X-62	2	REACTOR HEAD COOLING	6.2-41	WATER	V-31-002	NONE	OUT	GLOBE	AIR	REM MAN	N/A	N/A	AIR	DIRECT	CLOSE	CLOSE	CLOSE	CLOSE	
X-62	2	REACTOR HEAD COOLING	6.2-41	WATER	V-31-005	NONE	IN	CHECK	NONE	REV FLO	N/A	N/A	N/A	NONE	CLOSE	CLOSE	CLOSE	N/A	
X-62	3\4	REACTOR HEAD COOLING	6.2-41	WATER	V-31-007	NONE	OUT	GLOBE	HAND	N/A	N/A	N/A	N/A	NONE	CLOSE	CLOSE	CLOSE	N/A	LOCKED CLOSED
X-63A	14	CONT SPRAY(SUPPLY)	6.2-44	WATER	V-21-011	NONE	OUT	GATE	MOTOR	REM MAN	N/A	N/A	AC	DIRECT	CLOSE	OPEN	OPEN	AS IS	
X-63A	2	CONT SPRAY (TEST AIR CONNECTION)	6.2-44	AIR	V-21-135	NONE	OUT	GLOBE	HAND	N/A	N/A	N/A	N/A	NONE	CLOSE	CLOSE	CLOSE	N/A	LOCKED CLOSED
X-63A	2	CONT SPRAY (TEST AIR CONNECTION)	6.2-44	AIR	V-21-136	NONE	OUT	GLOBE	HAND	N/A	N/A	N/A	N/A	NONE	CLOSE	CLOSE	CLOSE	N/A	LOCKED CLOSED

TABLE 6.2-12
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CONTAINMENT ISOLATION VALVES / MECHANICAL INTEGRITY

1 PEN	2 SIZE	3 SYSTEM SERVICE	4 FIG	5 FLUID	6 TAG NO	7 SIGNAL	8 LOC	9 TYPE	10 VLVOP	11 PRI ACT	12 SEC ACT	13 MS	14 PWR	15 CR POS	16 NORM	17 SHDW	18 P ACC	19 LOP	20 COMMENT
X-64	24	MANHOLE	N/A	N/A	N/A	N/A	OUT	N/A	N/A	N/A	N/A	N/A	N/A	NONE	CLOSE	CLOSE	CLOSE	N/A	BOLTED CLOSED
X-65	20	TORUS TO RB VACUUM BRKR	6.2-54	AIR	V-26-016	1,6,7	OUT	BFLY	AIR	AUTO	REM MAN	≤60	AIR	DIRECT	CLOSE	CLOSE	CLOSE	OPEN	
X-65	20	TORUS TO RB VACUUM BRKR	6.2-54	AIR	V-26-018	1,6,7	OUT	BFLY	AIR	AUTO	REM MAN	≤60	AIR	DIRECT	CLOSE	CLOSE	CLOSE	OPEN	
X-65	20	TORUS TO RB VACUUM BRKR	6.2-54	AIR	V-26-015	NONE	OUT	CHECK	NONE	REV FLO	N/A	N/A	N/A	NONE	CLOSE	CLOSE	CLOSE	N/A	
X-65	20	TORUS TO RB VACUUM BRKR	6.2-54	AIR	V-26-017	NONE	OUT	CHECK	NONE	REV FLO	N/A	N/A	N/A	NONE	CLOSE	CLOSE	CLOSE	N/A	
X-65	3\4	TORUS PRESSURE	6.2-54	AIR	V-38-062	NONE	OUT	GATE	HAND	N/A	N/A	N/A	N/A	NONE	OPEN	CLOSE	OPEN	N/A	PT-IP-12
X-65	8	N2 PURGE	6.2-54	N2	V-23-015	1,6,7	OUT	BFLY	AIR	AUTO	REM MAN	<60	AIR	DIRECT	CLOSE	CLOSE	CLOSE	CLOSE	75 DEG MAX OPEN
X-65	8	N2 PURGE	6.2-54	N2	V-23-016	1,6,7	OUT	BFLY	AIR	AUTO	REM MAN	<60	AIR	DIRECT	CLOSE	CLOSE	CLOSE	CLOSE	75 DEG MAX OPEN
X-65	2	N2 PURGE	6.2-54	N2	V-23-019	1,6,7	OUT	CONTRL	AIR	AUTO	REM MAN	<60	AIR	DIRECT	CLOSE	CLOSE	CLOSE	CLOSE	
X-65	2	N2 PURGE	6.2-54	N2	V-23-020	1,6,7	OUT	CNTRL	AIR	AUTO	REM MAN	<60	AIR	DIRECT	CLOSE	CLOSE	CLOSE	CLOSE	
X-65	3\4	N2 PURGE	6.2-54	N2	V-23-259	NONE	OUT	GLOBE	HAND	N/A	N/A	N/A	N/A	NONE	CLOSE	CLOSE	CLOSE	N/A	LOCKED CLOSED
X-65	3\4	TORUS TO RB VACUUM BRKR	6.2-54	AIR	V-26-023	NONE	OUT	GLOBE	HAND	N/A	N/A	N/A	N/A	NONE	CLOSE	CLOSE	CLOSE	N/A	LOCKED CLOSED
X-65	3\4	TORUS TO RB VACUUM BRKR	6.2-54	AIR	V-26-024	NONE	OUT	GLOBE	HAND	N/A	N/A	N/A	N/A	NONE	CLOSE	CLOSE	CLOSE	N/A	LOCKED CLOSED
X-65	1\4	TORUS TO RB VACUUM BRKR	6.2-54	AIR	V-26-048	NONE	OUT	GLOBE	HAND	N/A	N/A	N/A	N/A	NONE	CLOSE	CLOSE	CLOSE	N/A	LOCKED CLOSED
X-65	1\4	TORUS TO RB VACUUM BRKR	6.2-54	AIR	V-26-046	NONE	OUT	GLOBE	HAND	N/A	N/A	N/A	N/A	NONE	CLOSE	CLOSE	CLOSE	N/A	LOCKED CLOSED
X-66	14	CONT SPRAY(SUPPLY)	6.2-44	WATER	V-21-005	NONE	OUT	GATE	MOTOR	REM MAN	N/A	N/A	AC	DIRECT	CLOSE	OPEN	OPEN	AS IS	
X-66	2	CONT SPRAY (TEST AIR CONNECTION)	6.2-44	AIR	V-21-132	NONE	OUT	GLOBE	HAND	N/A	N/A	N/A	N/A	NONE	CLOSE	CLOSE	CLOSE	N/A	LOCKED CLOSED

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CONTAINMENT ISOLATION VALVES / MECHANICAL INTEGRITY

1 PEN	2 SIZE	3 SYSTEM SERVICE	4 FIG	5 FLUID	6 TAG NO	7 SIGNAL	8 LOC	9 TYPE	10 VLVOP	11 PRI ACT	12 SEC ACT	13 MS	14 PWR	15 CR POS	16 NORM	17 SHDW	18 P ACC	19 LOP	20 COMMENT
X-66	2	CONT SPRAY (TEST AIR CONNECTION)	6.2-44	AIR	V-21-131	NONE	OUT	GLOBE	HAND	N/A	N/A	N/A	N/A	NONE	CLOSE	CLOSE	CLOSE	N/A	LOCKED CLOSED
X-66	3\4	CONT SPRAY	6.2-44	WATER	V-21-066	NONE	OUT	GATE	HAND	N/A	N/A	N/A	N/A	NONE	CLOSE	CLOSE	CLOSE	N/A	LOCKED CLOSED
X-67	1\2	ILRT TESTING	6.2-57	AIR	V-38-154	NONE	OUT	GLOBE	HAND	N/A	N/A	N/A	N/A	NONE	CLOSE	CLOSE	CLOSE	N/A	LOCKED CLOSED
X-67	1\2	ILRT TESTING	6.2-57	AIR	V-38-095	NONE	OUT	GLOBE	HAND	N/A	N/A	N/A	N/A	NONE	CLOSE	CLOSE	CLOSE	N/A	LOCKED CLOSED
X-67	1\2	ILRT TESTING	6.2-57	AIR	V-38-092	NONE	OUT	GLOBE	HAND	N/A	N/A	N/A	N/A	NONE	CLOSE	CLOSE	CLOSE	N/A	LOCKED CLOSED
X-68A	12	CONTAINMENT SPRAY(SUCTN)	6.2-45	WATER	V-21-001	1	OUT	GATE	MOTOR	REM MAN	N/A	N/A	AC	DIRECT	OPEN	OPEN	OPEN	AS IS	
AND	12	CONTAINMENT SPRAY(SUCTN)	6.2-45	WATER	V-21-003	1	OUT	GATE	MOTOR	REM MAN	N/A	N/A	AC	DIRECT	OPEN	OPEN	OPEN	AS IS	
X-68B	12	CONTAINMENT SPRAY(SUCTN)	6.2-45	WATER	V-21-007	1	OUT	GATE	MOTOR	REM MAN	N/A	N/A	AC	DIRECT	OPEN	OPEN	OPEN	AS IS	
AND	12	CONTAINMENT SPRAY(SUCTN)	6.2-45	WATER	V-21-009	1	OUT	GATE	MOTOR	REM MAN	N/A	N/A	AC	DIRECT	OPEN	OPEN	OPEN	AS IS	
X-69	12	CORE SPRAY(SUCTION)	6.2-45	WATER	V-20-003	1	OUT	GATE	MOTOR	REM MAN	N/A	N/A	AC	DIRECT	OPEN	OPEN	OPEN	AS IS	
X-68A	12	CORE SPRAY(SUCTION)	6.2-45	WATER	V-20-032	1	OUT	GATE	MOTOR	REM MAN	N/A	N/A	AC	DIRECT	OPEN	OPEN	OPEN	AS IS	
AND	12	CORE SPRAY(SUCTION)	6.2-45	WATER	V-20-004	1	OUT	GATE	MOTOR	REM MAN	N/A	N/A	AC	DIRECT	OPEN	OPEN	OPEN	AS IS	
X-68B	12	CORE SPRAY(SUCTION)	6.2-45	WATER	V-20-033	1	OUT	GATE	MOTOR	REM MAN	N/A	N/A	AC	DIRECT	OPEN	OPEN	OPEN	AS IS	
AND	3\4	PASS SAMPLE(CS SUCTN)	6.2-45	WATER	V-40-024	1	OUT	GLOBE	SOLND	REM MAN	N/A	N/A	AC	INDIRECT	CLOSE	CLOSE	OPEN	CLOSE	KEYLOCK CLOSED
X-69	2	TORUS LEVEL(NR)	6.2-45	WATER	V-38-005	NONE	OUT	GATE	HAND	N/A	N/A	N/A	N/A	NONE	OPEN	OPEN	OPEN	N/A	LT-IP9A/B
X-70	8	CORE SPRAY(SUPPLY)	6.2-56	WATER	V-20-151	NONE	IN	CHECK	AIR	REV FLO	N/A	N/A	AIR	DIRECT	CLOSE	CLOSE	OPEN	CLOSE	AIR FOR TEST ONLY

TABLE 6.2-12
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CONTAINMENT ISOLATION VALVES / MECHANICAL INTEGRITY

1 PEN	2 SIZE	3 SYSTEM SERVICE	4 FIG	5 FLUID	6 TAG NO	7 SIGNAL	8 LOC	9 TYPE	10 VLVOP	11 PRI ACT	12 SEC ACT	13 MS	14 PWR	15 CR POS	16 NORM	17 SHDW	18 P ACC	19 LOP	20 COMMENT
X-70	8	CORE SPRAY(SUPPLY)	6.2-56	WATER	V-20-153	NONE	IN	CHECK	AIR	REV FLO	N/A	N/A	AIR	DIRECT	CLOSE	CLOSE	OPEN	CLOSE	AIR FOR TEST ONLY
X-70	8	CORE SPRAY(SUPPLY)	6.2-56	WATER	V-20-021	NONE	OUT	GATE	MOTOR	AUTO	REM MAN	N/A	AC	DIRECT	CLOSE	CLOSE	OPEN	AS IS	AUTO INITIATE ONLY
X-70	8	CORE SPRAY(SUPPLY)	6.2-56	WATER	V-20-041	NONE	OUT	GATE	MOTOR	AUTO	REM MAN	N/A	AC	DIRECT	CLOSE	CLOSE	OPEN	AS IS	AUTO INITIATE ONLY
X-70	3\4	CORE SPRAY(SUPPLY)	6.2-56	WATER	V-20-044	NONE	OUT	GLOBE	HAND	N/A	N/A	N/A	N/A	NONE	CLOSE	CLOSE	CLOSE	N/A	LOCKED CLOSED
X-71	3\4	RECIRC LOOP SAMPLE LINE	6.2-50	WATER	V-24-029	1,2	IN	GLOBE	AIR	AUTO	REM MAN	<60	AIR	DIRECT	OPEN	CLOSE	OPEN	CLOSE	
X-71	3\4	RECIRC LOOP SAMPLE LINE	6.2-50	WATER	V-24-030	1,2	OUT	GLOBE	AIR	AUTO	REM MAN	<60	AIR	DIRECT	OPEN	CLOSE	OPEN	CLOSE	
X-71	3\4	PASS SAMPLE(LIQ POISON)	6.2-50	WATER	V-40-006	1	IN	GLOBE	SOLND	REM MAN	N/A	N/A	AC	INDIRECT	CLOSE	CLOSE	OPEN	CLOSE	KEYLOCK CLOSED
X-71	3/8"	RECIRC LOOP SAMPLE LINE (OVERPRESSURE PROTECTION)	6.2.-**50**	WATER	V-40-137	NONE	IN	CHECK	HAND	N/A	N/A	N/A	N/A	NONE	CLOSED	CLOSED	CLOSED	N/A	
X-72	2	MAIN STEAM DRAIN	6.2-48	STEAM	Y-1-057	NONE	IN	SPCFLG	NONE	N/A	N/A	N/A	N/A	NONE	CLOSE	CLOSE	CLOSE	N/A	
X-72	2	MAIN STEAM DRAIN	6.2-48	STEAM	Y-1-058	NONE	OUT	SPCFLG	NONE	N/A	N/A	N/A	N/A	NONE	CLOSE	CLOSE	CLOSE	N/A	
X-72	1\2	MAIN STEAM DRAIN	6.2-48	STEAM	V-1-136	NONE	OUT	GLOBE	HAND	N/A	N/A	N/A	N/A	NONE	CLOSE	CLOSE	CLOSE	N/A	LOCKED CLOSED
X-73A	1	ISOL COND(STM SUPPLY)	6.2-39	STEAM	V-14-042	NONE	OUT	GLOBE	HAND	N/A	N/A	N/A	N/A	NONE	OPEN	OPEN	OPEN	N/A	DPIS-IB05B1/2
										N/A	N/A	N/A	N/A					N/A	
X-73A	1\2	ISOL COND(STM SUPPLY)	6.2-39	STEAM	V-14-050	NONE	OUT	CHECK	NONE	N/A	N/A	N/A	N/A	NONE	OPEN	OPEN	OPEN	N/A	DPIS-IB05B1/2
X-73B	1	ISOL COND(STM SUPPLY)	6.2-39	STEAM	V-14-044	NONE	OUT	GLOBE	HAND	N/A	N/A	N/A	N/A	NONE	OPEN	OPEN	OPEN	N/A	DPIS-IB05B1/2
X-73B	1\2	ISOL COND(STM SUPPLY)	6.2-39	STEAM	V-14-052	NONE	OUT	CHECK	NONE	N/A	N/A	N/A	N/A	NONE	OPEN	OPEN	OPEN	N/A	DPIS-IB05B1/2

TABLE 6.2-12
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CONTAINMENT ISOLATION VALVES / MECHANICAL INTEGRITY

1 PEN	2 SIZE	3 SYSTEM SERVICE	4 FIG	5 FLUID	6 TAG NO	7 SIGNAL	8 LOC	9 TYPE	10 VLVOP	11 PRI ACT	12 SEC ACT	13 MS	14 PWR	15 CR POS	16 NORM	17 SHDW	18 P ACC	19 LOP	20 COMMENT
X-73C	1	ISOL COND(COND RETURN)	6.2-39	WATER	V-14-045	NONE	OUT	GLOBE	HAND	N/A	N/A	N/A	N/A	NONE	OPEN	OPEN	OPEN	N/A	DPIS-IB11A1/2
X-73C	1\2	ISOL COND(COND RETURN)	6.2-39	WATER	V-14-053	NONE	OUT	CHECK	NONE	N/A	N/A	N/A	N/A	NONE	OPEN	OPEN	OPEN	N/A	DPIS-IB11A1/2
X-73C	1	ISOL COND(COND RETURN)	6.2-39	WATER	V-14-047	NONE	OUT	GLOBE	HAND	N/A	N/A	N/A	N/A	NONE	OPEN	OPEN	OPEN	N/A	DPIS-IB11A1/2
X-73C	1\2	ISOL COND(COND RETURN)	6.2-39	WATER	V-14-055	NONE	OUT	CHECK	NONE	N/A	N/A	N/A	N/A	NONE	OPEN	OPEN	OPEN	N/A	DPIS-IB11A1/2
X-73D	1	RX PRESSURE IND	6.2-53	STEAM	V-130-011	NONE	OUT	GLOBE	HAND	N/A	N/A	N/A	N/A	NONE	OPEN	OPEN	OPEN	N/A	PS-IA83A/B,PS-RE17A/C
X-73D	1	RX PRESSURE IND	6.2-53	STEAM	V-130-001	NONE	OUT	CHECK	NONE	N/A	N/A	N/A	N/A	NONE	OPEN	OPEN	OPEN	N/A	PS-IA83A/B,PS-RE17A/C
X-74	20	CLEANUP DEMIN RELIEF LN	6.2-48	WATER	V-16-084	NONE	OUT	CHECK	NONE	REV FLO	N/A	N/A	N/A	NONE	CLOSE	CLOSE	CLOSE	N/A	
X-74	1\2	REACTOR CLEANUP(SUPPLY)	6.2-48	WATER	V-16-030	NONE	OUT	GLOBE	SOLND	AUTO	REM MAN	N/A	AC	NONE	CLOSE	CLOSE	CLOSE	CLOSE	
X-74	3\4	CLEANUP DEMIN RELIEF LINE	6.2-48	WATER	V-16-319	NONE	OUT	GLOBE	HAND	N/A	N/A	N/A	N/A	NONE	CLOSE	CLOSE	CLOSE	N/A	LOCKED CLOSED
X-74	3\4	CLEANUP DEMIN RELIEF LINE	6.2-48	WATER	V-16-320	NONE	OUT	GLOBE	HAND	N/A	N/A	N/A	N/A	NONE	CLOSE	CLOSE	CLOSE	N/A	LOCKED CLOSED
X-74	8	CLEANUP DEMIN RELIEF LN	6.2-48	WATER	V-16-076	NONE	OUT	RELIEF	NONE	N/A	N/A	N/A	N/A	NONE	CLOSE	CLOSE	CLOSE	N/A	
X-74	1\2	CLEANUP DEMIN RELIEF LN	6.2-48	WATER	V-16-145	NONE	OUT	GLOBE	HAND	N/A	N/A	N/A	N/A	NONE	CLOSE	CLOSE	CLOSE	N/A	LOCKED CLOSED
X-75A	1	RECIRC FLO XMTR IMPULSE A	6.2-53	WATER	V-37-001	NONE	OUT	GLOBE	HAND	N/A	N/A	N/A	N/A	NONE	OPEN	OPEN	OPEN	N/A	FT-IA60A
X-75A	1\2	RECIRC FLO XMTR IMPULSE A	6.2-53	WATER	V-37-005	NONE	OUT	CHECK	NONE	N/A	N/A	N/A	N/A	NONE	OPEN	OPEN	OPEN	N/A	FT-IA60A
X-75A	1	RECIRC FLO XMTR IMPULSE B	6.2-53	WATER	V-37-012	NONE	OUT	GLOBE	HAND	N/A	N/A	N/A	N/A	NONE	OPEN	OPEN	OPEN	N/A	FT-IA60B

TABLE 6.2-12
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CONTAINMENT ISOLATION VALVES / MECHANICAL INTEGRITY

1 PEN	2 SIZE	3 SYSTEM SERVICE	4 FIG	5 FLUID	6 TAG NO	7 SIGNAL	8 LOC	9 TYPE	10 VLVOP	11 PRI ACT	12 SEC ACT	13 MS	14 PWR	15 CR POS	16 NORM	17 SHDW	18 P ACC	19 LOP	20 COMMENT
X-75A	1\2	RECIRC FLO XMTR IMPULSE B	6.2-53	WATER	V-37-016	NONE	OUT	CHECK	NONE	N/A	N/A	N/A	N/A	NONE	OPEN	OPEN	OPEN	N/A	FT-IA60B
X-75A	1	RECIRC FLO XMTR IMPULSE C	6.2-53	WATER	V-37-023	NONE	OUT	GLOBE	HAND	N/A	N/A	N/A	N/A	NONE	OPEN	OPEN	OPEN	N/A	FT-IA60C
X-75A	1\2	RECIRC FLO XMTR IMPULSE C	6.2-53	WATER	V-37-027	NONE	OUT	CHECK	NONE	N/A	N/A	N/A	N/A	NONE	OPEN	OPEN	OPEN	N/A	FT-IA60C
X-75A	1	RECIRC FLO XMTR IMPULSE D	6.2-53	WATER	V-37-034	NONE	OUT	GLOBE	HAND	N/A	N/A	N/A	N/A	NONE	OPEN	OPEN	OPEN	N/A	FT-IA60D
X-75A	1\2	RECIRC FLO XMTR IMPULSE D	6.2-53	WATER	V-37-038	NONE	OUT	CHECK	NONE	N/A	N/A	N/A	N/A	NONE	OPEN	OPEN	OPEN	N/A	FT-IA60D
X-75A	1	RECIRC FLO XMTR IMPULSE E	6.2-53	WATER	V-37-045	NONE	OUT	GLOBE	HAND	N/A	N/A	N/A	N/A	NONE	OPEN	OPEN	OPEN	N/A	FT-IA60E
X-75A	1\2	RECIRC FLO XMTR IMPULSE E	6.2-53	WATER	V-37-049	NONE	OUT	CHECK	NONE	N/A	N/A	N/A	N/A	NONE	OPEN	OPEN	OPEN	N/A	FT-IA60E
X-75B	1	RECIRC FLO XMTR IMPULSE A	6.2-53	WATER	V-37-002	NONE	OUT	GLOBE	HAND	N/A	N/A	N/A	N/A	NONE	OPEN	OPEN	OPEN	N/A	FT-IA60A
X-75B	1\2	RECIRC FLO XMTR IMPULSE A	6.2-53	WATER	V-37-006	NONE	OUT	CHECK	NONE	N/A	N/A	N/A	N/A	NONE	OPEN	OPEN	OPEN	N/A	FT-IA60A
X-75B	1	RECIRC FLO XMTR IMPULSE B	6.2-53	WATER	V-37-013	NONE	OUT	GLOBE	HAND	N/A	N/A	N/A	N/A	NONE	OPEN	OPEN	OPEN	N/A	FT-IA60B
X-75B	1\2	RECIRC FLO XMTR IMPULSE B	6.2-53	WATER	V-37-017	NONE	OUT	CHECK	NONE	N/A	N/A	N/A	N/A	NONE	OPEN	OPEN	OPEN	N/A	FT-IA60B
X-75B	1	RECIRC FLO XMTR IMPULSE C	6.2-53	WATER	V-37-024	NONE	OUT	GLOBE	HAND	N/A	N/A	N/A	N/A	NONE	OPEN	OPEN	OPEN	N/A	FT-IA60C
X-75B	1\2	RECIRC FLO XMTR IMPULSE C	6.2-53	WATER	V-37-028	NONE	OUT	CHECK	NONE	N/A	N/A	N/A	N/A	NONE	OPEN	OPEN	OPEN	N/A	FT-IA60C
X-75B	1	RECIRC FLO XMTR IMPULSE D	6.2-53	WATER	V-37-035	NONE	OUT	GLOBE	HAND	N/A	N/A	N/A	N/A	NONE	OPEN	OPEN	OPEN	N/A	FT-IA60D

TABLE 6.2-12
Sheet 26 of 28

CONTAINMENT ISOLATION VALVES / MECHANICAL INTEGRITY

1 PEN	2 SIZE	3 SYSTEM SERVICE	4 FIG	5 FLUID	6 TAG NO	7 SIGNAL	8 LOC	9 TYPE	10 VLVOP	11 PRI ACT	12 SEC ACT	13 MS	14 PWR	15 CR POS	16 NORM	17 SHDW	18 P ACC	19 LOP	20 COMMENT
X-75B	1\2	RECIRC FLO IXMITR IMPULSE D	6.2-53	WATER	V-37-039	NONE	OUT	CHECK	NONE	N/A	N/A	N/A	N/A	NONE	OPEN	OPEN	OPEN	N/A	FT-IA60D
X-75B	1	RECIRC FLO IXMTR IMPULSE E	6.2-53	WATER	V-37-046	NONE	OUT	GLOBE	HAND	N/A	N/A	N/A	N/A	NONE	OPEN	OPEN	OPEN	N/A	FT-IA60E
X-75B	1\2	RECIRC FLO IXMITR IMPULSE E	6.2-53	WATER	V-37-050	NONE	OUT	CHECK	NONE	N/A	N/A	N/A	N/A	NONE	OPEN	OPEN	OPEN	N/A	FT-IA60E
X-77	1 1\2	TORUS DRAIN	6.2-55	WATER	V-38-034	NONE	OUT	GLOBE	HAND	N/A	N/A	N/A	N/A	NONE	CLOSE	CLOSE	CLOSE	N/A	LCKD CLSD & CAPPED
X-78	1 1\2	TORUS WATER LEVEL	6.2-55	WATER	V-38-033	NONE	OUT	GLOBE	HAND	N/A	N/A	N/A	N/A	NONE	OPEN	OPEN	OPEN	N/A	LT-37&38

TABLE 6.2-12
Sheet 27 of 28

CONTAINMENT ISOLATION VALVES / MECHANICAL INTEGRITY

COL. NO	COLUMN DESCRIPTION	ABBREVIATIONS
1	Primary Containment Penetration Number	See Table
2	Line Size	Inches
3	System Service	See Table
4	UFSAR Reference Figure Number	See Table
5	Process Fluid	N2 Nitrogen
6	Valve Tag Number	See Table
7	Isolation Signal Codes or Groups	1. Manual Initiation 2. - Reactor Low Low Level - MSL Hi Temperature - MSL Low Pressure - MSL Hi Flow 3. Emergency Condenser - Hi Steam Flow - Hi Return Flow 4. RWCU Isolation - Low Filter Flow - NRHX Hi Temperature Outlet - Hi Pressure - Hi Cooling Water Temperature - Liquid Poison Flow 5. Shutdown Cooling System Hi Inlet Temperature 6. Reactor Low Low Level 7. Hi Drywell Pressure 8. Reactor Low Low Low Level 9. Primary Containment Hi Radiation 10. RWCU HELB Isolation
8.	Location	Inside Primary Containment Outside Primary Containment
9	Valve or Component Type	Ball Butterfly Check Control Flange Gate Globe Needle Relief Shear

TABLE 6.2-12
Sheet 28 of 28

CONTAINMENT ISOLATION VALVES / MECHANICAL INTEGRITY

COL. NO	COLUMN DESCRIPTION	ABBREVIATIONS
10	Valve Operator	Spectable Flange 3-Way Air Explosive Hand Motor Solenoid
11	Primary Actuation Mode	Automatic Overpressure Remote Manual Reverse Flow
12	Secondary Actuation Mode	Automatic Overpressure Remote Manual Reverse Flow
13	Maximum Stroke Time	Seconds
14	Power Source	Air Alternating Current (AC) Direct Current (DC) Alternating Current (AC) Direct Current (DC)
15	Control Room Position Indication	Direct Indirect None
16	Normal Valve Position	Closed Open Closed/Open
17	Shutdown Valve Position	Closed Open Closed/Open
18	Post Accident Valve Position	Closed Open Closed/Open
19	Valve Position Upon Loss of Power	Closed Open As-Is

TABLE 6.2-13
(Sheet 1 of 1)

CONTAINMENT INERTING SYSTEM DESIGN REQUIREMENTS

Liquid Nitrogen Purity	99.7 percent (minimum)
Liquid Nitrogen Moisture Content (Liquid)	2.5 ppm by volume (maximum)
Maximum/Minimum Primary Containment Atmospheric Temperature	150°F/50°F
Primary Containment Free Volume	317,000 ft^3
Maximum Allowable Oxygen Content Prior to Termination of Purging	4.0 percent by volume
Maximum Allowable Oxygen Content During Reactor Operation	4.0 percent by volume (See Tech Spec. 3.5.A.6)
Primary Containment Pressure to be Maintained During Purging Operation	(-) 1/4 in W.G. (normal) 1.5 psig (maximum)
Primary Containment Pressure Control Range During Reactor Operation	1.1 psig to 1.3 psig
Maximum Expected Purge Operation Time from Initiation to Completion at 100,000 scfh	8-10 hrs.
Expected Nitrogen Flow Rate During Purge	100,000 scfh
Approximate Total Nitrogen Required for Purge	1,000,000 scf
Maximum Expected Primary Containment Leakage During Reactor Operation at 1.3 psig	0.2 percent free volume/day (587 scf/Day)
Minimum Expected Primary Containment Leakage During Reactor Operation	200 scf/Day

TABLE 6.2-14
(Sheet 1 of 1)

VALVE POSITIONS FOR PURGING

Valves	Normal Exhaust Fan	Standby Gas Treatment
Ventilating System		
V-28-19	Open	Open
V-28-21	Open	Close
V-28-22	Open	Close
V-28-23	Close	Open*
V-28-27	Close	Open*
Drywell Exhaust		
V-27-1	Open	Close
V-27-2	Open	Close
V-23-21	Close	Open
V-23-22	Close	Open
Torus Exhaust		
V-28-17	Open	Close
V-28-18	Open	Open
V-28-47	Close	Open

* Depending on which SGTS, either System 1 or System 2 is selected for service.

TABLE 6.2-15
(Sheet 1 of 2)

CONTAINMENT RESPONSE TO MINIMUM CONTAINMENT SPRAY AND EMERGENCY SERVICE WATER SYSTEM FLOWS

TOTAL CSS FLOW (gpm)	TOTAL ESW FLOW to HX	CANAL WATER TEMP (°F)	HX CLEANLINESS FACTOR	PEAK TORUS POOL TEMP (°F)	HEAT **DUTY MBTU/HR
3200	3000	95°	65%	162.3	31.66
4400 CS-1	3000	95°F	65%	154.8	36.22
4400 CS-1	3000	95°F	65%	154.8	36.22

** Heat exchanged based upon peak torus pool temperature; two CSHXs in service (1 CS loop)

TABLE 6.2-15
(Sheet 2 of 2)

MINIMUM CORE AND CONTAINMENT SPRAY PUMP NPSH****

Containment Spray System 1 Flow (gpm)	Containment Spray System 2 Flow (gpm)	Core Spray flow / pump (gpm)	Suppression Pool Temp (°F)***	Containment Spray NPSH Available (ft)*	Containment Spray NPSH Required (ft)	Core Spray NPSH Available (ft)**	Core Spray NPSH Required (ft)
4100	5000	4800 / 4350	140	32.13	30.56	-	-
5000	4100	4800 / 4800	162.3	-	-	19.40	19.10

* NPSH available is evaluated for Containment Spray Pump having the highest line losses (P-21-1C). Minimum NPSH for Containment Spray pumps occurs after 10 minutes when the system transitions from drywell spray to torus cooling mode. This is a transient condition that only lasts approximately 1 minute.

** NPSH available is evaluated for Core Spray Pump having the highest line losses (NZ01C). The NPSH value presented is based on the maximum Containment Spray pump and Core Spray pump flow rates using the peak temperature evaluated for the minimum flow case. This NPSH result bounds any possible plant condition.

*** 162.3 degrees F is the peak suppression pool temperature evaluated for the minimum flow case (see Table 6.2-15). This temperature is combined with the maximum Containment and Core Spray flow to calculate the Core Spray pump NPSH margin. 140 degrees F is the actual calculated temperature after 10 minutes in the maximum flow case (see Table 6.2-15). This temperature is used to more accurately calculate the minimum NPSH available to the Containment Spray pumps.

**** Note that these flowrates are presented for NPSH only, and are not those required for the safety function of the system.

TABLE 6.2-16
(Sheet 1 of 1)

CONTAINMENT INERTING SYSTEM CONTROL AND INSTRUMENTATION

Sensing Instrument	Actuates
LI	Makeup Nitrogen Tank Level Indicator, Local
PI	Makeup Nitrogen Tank Pressure Indicator, Local;
TE 49	Makeup Nitrogen Header Temperature Indicator, on Panel 12XR;
PT 50	Makeup Nitrogen Header Pressure Indicator, Local;
FT 9	Purge/Makeup Nitrogen Flow Recorder, on Panel 12XR Makeup Flow Integrator, on Panel 12XR
TE 48	Purge Nitrogen Header Temperature Indicator, on Panel 12XR;
PT 49	Purge Nitrogen Header Pressure Indicator, Local;
FT 8	Purge/Makeup Nitrogen Flow Recorder, on Panel 12XR
O_2A	Drywell Oxygen Concentration Recorder, on Panel 12XR
O_2A	Torus Oxygen concentration recorder, on Panel 12XR
PT-IP12	1.Torus pressure indicator (PI-IP13) on Panel 1F/2F
	2. Annunciator in the Control Room TORUS HIGH VACUUM.
PI-IP07	1. Drywell pressure indicator (PI-IP08), on Panel 1F/2F
	2. Annunciator in the Control Room DRYWELL PRESSURE HI/LO.
PS-RE04A, -B, -C, -D	1. Annunciator in the Control Room DRYWELL HIGH PRESS SCRAM.
	2. Scrams reactor on high drywell pressure.
	3. Closes all ventilation and purge and nitrogen makeup isolation valves on high drywell pressure.
	4. Closes all drywell sump and drain tank isolation valves on high drywell pressure. All actions occur on a 1-out-of-2 twice reactor protection system logic circuits
TE	Temperature measurements selectively distributed in the drywell and torus atmosphere; recorded in the Control Room.

TABLE 6.2-17
(Sheet 1 of 2)

OPERATIONAL STATUS OF PRIMARY CONTAINMENT PENETRATIONS WITHOUT AUTOMATIC ISOLATION

Penetration	Description	Remarks	Operational Status Accident Condition	Test Condition
Access	Bolted, gasketed seals	Example: equipment hatch, drywell head	Closed	Closed
Instrument	Dead ended, closed system	Root valves closed if instrument is out of service	Open	Open
Electrical	Dead ended, closed system	Cables potted in penetrations	Closed	Closed
Control Rod Drive	Closed System	Valves normally open, close on scram signal	Open	Open
Closed System	a) System open inside and closed outside containment;	Example: containment spray system	Open	Open
	b) System open inside and closed outside containment	Example: liquid poison system	Closed	Closed

TABLE 6.2-17
(Sheet 2 of 2)

OPERATIONAL STATUS OF PRIMARY CONTAINMENT PENETRATIONS WITHOUT AUTOMATIC ISOLATION

			Operational Status	
Penetration	Description	Remarks	Accident Condition	Test Condition
Instrument Air	Closed System	System valved out during test to prevent inleakage of air	Open	Closed
Potentially Open	System open, containment valves normally closed	Example: drywell ventilation purge system	Closed	Closed

6.3 EMERGENCY CORE COOLING SYSTEM

6.3.1 Design Bases

The OCNGS Emergency Core Cooling System (ECCS) consists of three separate systems: the Isolation Condenser System, the Core Spray System, and the Automatic Depressurization System. For operating Cycle 12, OCNGS revised its plant specific Appendix K analysis to assume only the Core Spray and Automatic Depressurization Systems available, and the Isolation Condenser System inoperable. Other systems, such as the reactor Feedwater System and the Control Rod Drive Hydraulic System, provide cooling water to the core during a Loss-of-Coolant Accident (LOCA), but these are not considered part of the ECCS, since their primary function is not emergency core cooling. The Containment Spray System, which removes core decay heat to the Ultimate Heat Sink via the Emergency Service Water System, is described in Section 6.2.

The Isolation Condenser System (ICS) is a passive high pressure system which consists of two independent natural circulation heat exchangers that are automatically initiated by reactor vessel high pressure or low-low water level. Even though no single failure in the system can cause both Isolation Condensers to malfunction, the Appendix K analysis does not take credit for the ICS operation.

The Automatic Depressurization System (ADS) consists of five automatically activated relief valves that depressurize the Reactor Coolant System during a small break LOCA to permit the low pressure Core Spray System to inject water onto the reactor core. The five ADS valves are actuated by simultaneous occurrence of triple low reactor water level, high drywell pressure and indication that a core spray booster pump has been started. Only three of the five valves are required to achieve depressurization in the allowable time period, and no single failure in the system can cause more than one of the five valves to fail to open upon initiation signal actuation.

The Core Spray System contains two completely independent systems each containing two main pumps, two booster pumps, and a core spray sparger. Both loops are simultaneously actuated by either low-low reactor water level or high drywell pressure. The worst case single failure in the Core Spray System (loss of an Emergency Diesel Generator) cannot impair the capability of the system to perform the required safety function.

The two ECCS subsystems operate in various combinations to maintain peak cladding temperatures below 2200°F, and within the limits of 10CFR50.46, for any size break LOCA even if a single failure occurs in the ECCS. The single failure in the ECCS that produces the highest peak cladding temperature for a particular size break may be different from the worst single failure for another size break.

The ECCS was provided to meet definite design criteria with respect to the design basis LOCAS. The design criteria to which the ECCS was originally designed included the following conditions:

a. For all the design basis LOCAs, sufficient cooling capacity must be provided so that a conservative design evaluation indicate that, as a result of the abnormal temperature transients which occur due to the loss of coolant, no cladding melting would occur. By preventing cladding melting the core would be held in a definable geometry and subsequent core cooling can be predicted with confidence.

b. The above criterion of no cladding melting was satisfied for the entire spectrum of reactor primary system break sizes from small leaks up to and including the complete double ended severance of one of the recirculation lines. Detailed evaluations were required for various break sizes in that the entire behavior of the reactor system is dependent on the size of the break assumed.

c. The two ECCS subsystems, which together perform the emergency core cooling function must have sufficient redundancy of active components to be able to accomplish their emergency core cooling function even under the condition of a single failed component. This criterion is satisfied in that the Core Spray System was provided with a 100% redundant pumping and valving system, such that both loops will be available to provide core cooling. Core Spray System configuration for the worst case LOCA mitigation requires one main pump and one booster pump from one system and one main pump from the other system. The ICS and ADS have also been provided with adequate redundancy.

d. The ECCS was designed to satisfy the criterion of no loss of function upon Loss-of-Offsite-Power (LOOP).

Presently, the OCNGS is operated in a manner that precludes exceeding the limits of 10CFR50.46 in the event of a design basis accident.

Redundant standby cooling capability is provided in the event of an accident which disrupts normal cooling capability. Protective devices are installed to prevent interruption of performance or availability of the standby cooling systems unless use of these systems would make complete failure imminent.

General Design Criteria 35, 36 and 37 are specifically applicable to the ECCS.

The ECCS is required to provide effective core cooling, prevent clad damage and limit metal-water reaction during a postulated Loss-of-Coolant Accident. The postulated accidents include:

a. The spectrum of liquid line breaks up to a 26 inch diameter guillotine break of a recirculation line and loss of one isolation condenser connected to the broken line.

b. The spectrum of steam/water line breaks up to a six inch break with loss of one Core Spray System loop.

c. An 18 inch steam/water break.

d. A ten inch steam line break with loss of one isolation condenser connected to the line.

e. A 24 inch steam line break.

6.3.1.1 Isolation Condenser System

6.3.1.1.1 Function

The Isolation Condenser System (ICS) is a standby, high pressure system for removal of fission product decay heat when the reactor vessel is isolated from the Main Condenser. The system prevents overheating of the reactor fuel, controls the reactor pressure rise, and limits the loss of reactor coolant through the relief valves.

The ICS is not intended to be activated fast enough to have any effect upon the initial pressure peaks resulting from various transients such as turbine trip, Main Steam Isolation Valve closure, and others. The system can be activated manually or automatically, and once activated will remain in operation until manually removed from service as long as line break conditions do not exist.

The ICS also provides an alternate shutdown capability. In the event of damage from a fire or natural phenomena (tornadoes, flooding), the ICS removes reactor decay heat to establish a safe shutdown condition (References 17,19).

6.3.1.1.2 Description

The P&ID of the Isolation Condenser System is presented in Drawing GE148F262. Major component design values are listed in Table 6.3-1. The system consists of two full capacity isolation condensers, four ac motor operated isolation valves, four dc motor operated isolation valves, and three vent lines to the atmosphere.

The ICS operates by natural circulation without the need for driving power other than the dc electrical system used to place the ICS in operation. The system operates with steam flowing from the reactor pressure vessel through the condenser tubes and condensate returning by gravity to the reactor pressure vessel, forming a closed loop. The valves in the steam inlet lines are normally open so that the tube bundles are at reactor pressure. Only the dc motor operated condensate isolation valves are normally closed. The shell side of each condenser has a normal minimum water inventory of 22,730 gallons with at least 11,060 gallons above the top of the tube bundles. During normal plant operations, when the system is in Standby, makeup to the Isolation Condensers is from the Demineralized Water Transfer System. Makeup during ICS operation is provided from the Demineralized Water Transfer System or the Condensate Transfer System. An emergency makeup is also provided from the fire suppression and core spray systems. The shell side of the condensers are vented to the atmosphere via three lines protruding through the east wall of the Reactor Building.

The design heat removal capacity of the ICS (two condensers) is 410 x 10^6 Btu/hr. At the normal water level with both isolation condensers in service, the system can provide emergency cooling for approximately one hour and 40 minutes without makeup water. A single isolation condenser may remain in operation for up to 45 minutes.

The system comprises two loops, each with one condenser shell containing two tube bundles. When a loop is in operation, both tube bundles are in service. Normally, both condensers are placed in operation simultaneously, and either loop can be activated or shut down separately under manual control. Each loop has separate steam and condensate lines and isolation valves, separate steel shell steam vents and valves, and separate instrumentation and controls.

The Isolation Condensers are located, and the layout and sizing of the steam supply and condensate return lines have been optimized, to provide the design flow rate with valve pressure drops and line pressure drops minimized. The initial driving head is provided because the tube bundles and condensate return lines are filled with water.

The steam supply, with the exception of the vertical portion where thermocouples are attached, and condensate return lines are insulated and sloped to promote drainage to the reactor pressure vessel. The steam supply lines connect to the reactor vessel cylinder at the steam zone, and are separate from the main steam headers, so that line condensation does not cause entrainment of liquid in the steam to the turbine.

The high points in the steam supply lines to each loop are vented continuously to the main turbine steam header downstream of the Main Steam Isolation Valves when the plant is operating and the ICS is on standby.

This is done to remove noncondensable gases from the reactor steam which would otherwise collect at these high points in the system. An evaluation of the impact of noncondensable gases on isolation condenser performance in the event that the vents are closed and no gases are removed is presented in GPUN Calculation C-1302-211-5300-046 (Rev 1) "Oyster Creek Isolation Condensers Quantity of Noncondensable Gases in System," C-1302-211-E540-099 (Rev 0) "OCNGS Evaluation of Isolation Condenser Performance with Noncondensables in Steam, and C-1302-211-E540-124 (Rev. 0) "OC Isolation Condenser Purge Time." The evaluation concludes that closure of the vent line isolation valves or blockage of the vent line will not preclude proper system operation, if it is purged by opening the vent valves for at least 8 hours every 44 days. The analyses are bounding if the steam supply lines are fully insulated except for the vertical section immediately above each tube bundle, which does not have to be insulated.

The Reactor Coolant System side of each ICS loop has two vent isolation valves in series. The lines from both loops join a common header which is connected to the main steam header outside the drywell. Manually operated valves are installed upstream of the isolation valves for regulation of vent flow rate. The vent isolation valves are normally open with a two position switch in the Control Room. All vent isolation valves close automatically on low-low reactor water level, main steam line break, and on main steam line low pressure, and opening of the condensate return valves.

There are two normally open isolation valves in the steam supply lines of each loop as shown in Drawing GE148F262. Both isolation valves are located outside the drywell and the valve bodies are welded into one assembly with no intermediate pipe nipple. Neither valve is located inside the drywell because the lines are located in the uppermost region of the drywell neck, directly under the refueling seal flange. This causes space limitations and inaccessibility of the pipe runs inside the drywell. An 18 inch guard pipe surrounds the steam line inside the drywell penetration. The guard pipe is welded directly to a 24 inch penetration sleeve and to a flued collar which is in turn welded to the OD of the process pipe (Figure 3.8-16A).

The power for the DC isolation valves is supplied from the station batteries except during a HELB, whereby credit is taken for the battery chargers. The power supply for the AC isolation valves is backed up with power from the Emergency Diesel Generators (EDGs). This assures two dependable sources of power, and allows the ICS to initiate with a total loss of AC power. The voltage to the DC valves in a postulated HELB isolation scenario and LOOP is augmented by the battery chargers, which restart upon return of AC power via the EDG. The AC power for the battery chargers is provided from the opposite EDG (train) as that of its redundant AC isolation valve, ensuring two dependable sources of power to operate the isolation valves. The steam supply isolation valves are normally open, and both ac and dc operated valves will be closed automatically by a signal from the pipeline break differential pressure switches. Each of the four isolation valves can be closed or opened from the Control Room.

The valves are austenitic stainless steel valves rated for service up to 1250 psi. The two valves on each inlet line from the reactor vessel to the Isolation Condensers are welded in series. The valves and valve body extensions meet ASA B31.1.

The consequences associated with a break in the piping within the guard pipe have been accounted for in the design of the Isolation Condenser piping system. Hydraulic snubbers are furnished outside of the drywell and mechanical snubbers are provided inside the drywell. The guard pipes are 18 inch OD and 0.5 inch wall thickness and fabricated of A-106, Gr B material with a minimum yield point of 35,000 psi. Based on the conservative assumption that pressure in the guard pipe might be as high as 1250 psig, the circumferential stress in the guard pipe is less than 25,000 psi.

Downstream of the isolation valves the steam supply lines branch into two lines (wyes), which enter the tube bundles at the Isolation Condensers. These wyes are located at an elevation higher than the Isolation Condenser tube bundles to prevent the recurrence of the steaming phenomenon which led to a forced outage in 1988.

During ICS operation, steam from the reactor vessel condenses at the tube bundles as heat is transferred through the tubes into the shell side of the Isolation Condensers. Condensate flows by natural circulation back into recirc loops A&E through the condensate return lines.

The two condensate lines from the tube bundles of each Isolation Condenser join together into a common condensate return line, one for each loop. Each condensate return line has two isolation valves in series. Two are dc operated valves, located just outside the drywell penetration, and the penetration is protected with a guard pipe installed as described previously for the steam supply line penetrations. Two ac operated valves are located inside the drywell. The power for the DC isolation valves is supplied from the station batteries except during a HELB, whereby credit it taken for the battery chargers. The power supply for the AC isolation valves is backed up with power from the Emergency Diesel Generators (EDGs). This assures two dependable sources of power, and allows the ICS to initiate with a total loss of AC power. The voltage to the DC valves in a postulated HELB isolation scenario and LOOP is augmented by the battery chargers, which restart upon return of AC power via the EDGs. The AC power for the battery chargers is provided from the opposite EDG (train) as that of its redundant AC isolation valve, ensuring two dependable sources of power to operate the isolation valves. The ac operated condensate return isolation valves are normally open, and capable of automatic closure; these can be manually operated from the Control Room. The dc operated condensate return valves are normally closed. Only one valve in each loop needs to open to place the ICS in operation. Each valve can be manually opened or closed from the Control Room. The remote manual demand overrides the automatic signal. There is no automatic reset or signal to close the condensate return valves on reduced reactor vessel pressure. The closure must be achieved manually in the Control Room placing the control switch in the close position. Removal of the automatic signal is accomplished by pushing a single reset switch.

The condensate return lines are connected to the reactor vessel side of the reactor recirculation pumps suction valves. The condensate from Isolation Condenser "A" discharges into the

Recirculation Pump "A" suction whereas the condensate from Isolation Condenser "B" discharges into the Shutdown Cooling System influent header, close to its connection to the Recirculation Pump "E" suction line. Thus, the condensate return would flow into recirculation loops "A" and "E", and hence to the reactor vessel. If the valves are closed on these recirculation loops, the condensate returns through the loop suction nozzles into the downcomer plenum of the reactor pressure vessel, and thus through other recirculation loops to the fuel zone of the reactor. For this condition, at least one recirculation loop must be non-idle to permit recirculation through the core and to provide adequate indication of water level around the core (inside the shroud).

The two Isolation Condensers are located side by side at Reactor Building El. 95'-3", with the shell center lines at El. 108'-3". The reactor pressure vessel nozzles for the steam supply lines are at El. 84'-10". The condensate return outlet nozzles are at El. 103'-3". Thus, the tube bundle outlets are 20'-9" above the normal reactor water level (El. 82'-6").

The shell side of each Isolation Condenser is normally flooded to a level 12 inches above the shell center line, which corresponds to a normal inventory of 22,730 gallons per shell or 11,060 gallons per shell above the tube bundles (see Table 6.3-1). The internal surface of the Isolation Condenser shells are coated with an epoxy based coating to provide corrosion protection.

During ICS operation, the shell side water in the Isolation Condensers will boil, with consequent generation of steam. The shell steam vents through two 20 inch nozzles on each shell. The two nozzles from Isolation Condenser "A" join into a 28 inch header, and the two nozzles from Isolation Condenser "B" are run separately. All three vent lines (the single 28 inch vent and the two 20 inch vents) protrude horizontally through the wall of the Reactor Building. Bird screens are attached over the ends of the pipes. The pipe configuration is designed to ensure adequate pressure gradients for proper condenser operation.

A maximum operating pressure of 5 psig provides efficient use of the shell storage water due to the advantage of the large difference in latent heats of vaporization between reactor steam and shell water. The shell vents terminate at an elevation at which, if the vent is inadvertently filled with water, the maximum hydrostatic head in the shell will not exceed the design pressure of 15 psig.

Design steam velocity of about three feet per second at the water surface is necessary for adequate separation of steam and water. The established normal water level provides swell capability (from temperature change and steam production), for adequate separation distance between water and steam and for maximum period of operation without water addition. Splash baffles are installed at each vent nozzle, because excessive water carryover has the effect of decreasing operating time before makeup water is required. Moisture carryover should not exceed 3 percent by weight.

The Isolation Condensers are designed for the rather severe loading conditions encountered during transient and steady state conditions, including:

a. Internal pressures and mechanical forces, such as fluid weight, fluid flow, and external and internal attachments.

b. Pipe reactions due to thermal expansion of the piping and condenser shells.

Makeup water for the condenser shell side can be manually added from the Demineralized Water Transfer (WD) System by opening an air operated makeup valve, as necessary. If the WD system is not available, the Condensate Transfer System can be used by opening an additional air operated valve controlled from either the Main Control Room on the Remote Shutdown Panel. If a condensate transfer pump is not available, Fire Protection System water can be manually valved into service. If power is lost to the makeup valves, they can be manually opened at El. 95'. Shell side level indication is displayed in the Control Room and the makeup valve control switch is also in the Control Room. An emergency makeup is also provided to the isolation condenser shell from the torus via the core spray system.

Isolation Condenser System safety-related motor-operated valves are included in the Generic Letter (GL) 89-10 Motor-Operated Valve (MOV) Program as noted in the OCNGS Program Description for NRC Generic Letter 89-10 Motor-Operated Valve Program. This program has reestablished the design basis for safety-related motor-operated valves. Critical design basis assumptions such as design basis differential pressure, safety function - open vs. close, minimum available AC/DC voltages, actuator gearing, torque switch control logic, valve factors, stem friction coefficients, and valve stroke times have been established in assessing GL 89-10 design basis capability. Plant changes or activities which can affect these design basis assumptions must consider the affect on the capability of GL 89-10 motor-operated valves to perform their safety function and on safety margins established for these valves.

6.3.1.1.3 Operation

The ICS is operable during power operation and whenever the reactor coolant temperature is greater than 212-F except as permitted otherwise by the Technical Specifications.

During normal operation all valves are open except for the two outboard dc powered condensate return line isolation valves. Initiation of the system is either automatic or manual.

The ICS is automatically initiated by a persistent (about 1.5 seconds) signal of either high reactor vessel pressure or low-low reactor water level. The time delay is intended to prevent ICS initiation under certain transient conditions. The signals from four reactor high pressure sensors and four reactor low-low water level sensors are arranged in a one-out-of-two-twice logic such that one high pressure and one low-low water level sensor are in each of the four logic trains.

The initiation signal causes both normally closed condensate return isolation valves to open. The high point vent valves will close when these valves open. Coincident with the low-low reactor water level signal for ICS initiation, is a signal to trip all five reactor recirculation pumps. Coincident with the high reactor pressure signal for ICS initiation, is a signal to trip recirculation pumps A, B, and E. Recirculation pumps C and D will trip on a persistent high pressure signal within 12 seconds (time delay setpoints will provide margin for calibration and accuracy of the time delay relays).

The pressure setpoints of the ICS, reactor pressure scram, and the relief valves have been chosen to limit pressure transients to values below the safety valve settings and to limit the loss of water through the relief valves. Once the ICS is actuated, there is an estimated 25 and 36 (Reference 25) second delay (opening time for condensate return valves V-14-34 and V-14-35 respectively), before the ICS reaches its full effectiveness.

The only operation needed to manually initiate an ICS loop is to open its normally closed condensate return isolation valve from the Control Room. To ensure a natural circulation path through the reactor core, the Technical Specifications require that at least one recirculation loop suction valve and its corresponding discharge valve to be open.

Once activated, the ICS will not automatically reset with reduced reactor pressure. This requires deliberate operator action to remove the ICS from service. Automatic initiation signal can then be reset by pressing the reset button in the Control Room.

Reactor water lost prior to ICS initiation can be restored by the Control Rod Drive Hydraulic System pumps during isolation. Because of the potential for damage due to water hammer, reactor water level should not be allowed to exceed 180 inches above the top of active fuel.

When placed in operation, the heat removal capability of the ICS is approximately equivalent to 152.5 MW thermal (two condensers) which is significantly greater than the original design heat removal rate of 112 MW thermal. Furthermore, the decay heat generation rate drops to about 2 percent of maximum reactor power (1930 MWt) about ten minutes after scram. Thus, manual action is required to prevent too rapid a cooldown. This is accomplished by alternate manual opening and closing of the condensate return isolation valves. Should the condensate return valves fail to close, the steam supply valves can be used to control pressure. Reactor cooldown rate should be less than 100°F/hr.

Both ICS loops will operate for about one hour and 40 minutes without makeup; during one loop operation, this time is reduced to 45 minutes. To achieve cooldown, approximately 107,500 gallons of makeup are required. A minimum shell level of 4.8 to 7.7 feet is maintained during isolation condenser operation.

All automatic isolation valves of the ICS can be manually opened or closed, regardless of what is being called for by the logic circuitry. The Isolation Condensers are automatically isolated from the reactor vessel in the event of high flow in either the steam or condensate lines which results from a line break. Manual isolation is also required in the event that tube leakage is discovered. Symptoms of tube leakage could include increasing shell side level, increasing shell side temperature, or increasing area radiation.

For a fire in the cable spreading room or a fire caused evacuation of the Control Room, the isolation condenser removes core decay heat to establish a safe shutdown condition. Since a fire in the cabling associated with the isolation condenser high flow trip function could result in a spurious isolation of the isolation condenser, the design includes a bypass of the trip function on initiation of the alternate shutdown panel (Reference 18).

A high flow trip function is provided to isolate the system in the event of line break outside the primary containment. A fire requiring initiation of the alternate shutdown panel in conjunction with a line-break accident is not considered a credible event. The alternate shutdown panel is initiated through transfer switches that are key locked and alarmed in the Control Room to prevent inadvertent actuation. Single failure of the switch will not preclude operation of the isolation condenser high flow trip function in the event of line-break accident.

Subsequent to a tornado or external flooding, the ICS has sufficient water inventory to remove decay heat for 1.5 hours. Since the normal sources of water for the ICS are not designed against these natural phenomena, the Core Spray System I main pump through its fill line can provide emergency makeup to the isolation condenser shells from the torus. A design basis accident, plant transient, or failure of the core spray system isolation valves are not considered credible to occur coincidently with the tornado event. To establish the emergency fill path, manual actions are required to realign valves, but these are accessible and sufficient time is available to perform the realignment.

6.3.1.2 Automatic Depressurization System

6.3.1.2.1 Function

The Automatic Depressurization System (ADS) provides for a controlled blowdown of the primary system to rapidly reduce pressure during a small pipe break. This depressurization permits Core Spray System (Subsection 6.3.1.3) injection prior to uncovering the fuel, when reactor vessel pressure is below 285 psig. In addition, the Electromatic Relief Valves (EMRVs) of the ADS open on an overpressure condition in the reactor pressure vessel.

For intermediate and large breaks, where the reactor depressurizes sufficiently fast, the Core Spray System achieves rated flow before the cladding melts without assistance from either the reactor Feedwater System or the ADS.

For small breaks, when feedwater is available, the Turbine Bypass System can be used to cool down the reactor which reduces vessel pressure to within the range of the Core spray System, if required.

When feedwater is not available, the ADS is designed to depressurize the reactor vessel in time to initiate Core Spray System injection, and thus to prevent cladding melt, under the assumption that no water is added to the system, even down to a 0.025 square foot break.

For the turbine trip and loss of electrical load transients, the turbine trip scram or generator load rejection scram, the ADS in combination with the Turbine Bypass System limit reactor overpressure so that the 9 safety valves will not lift.

6.3.1.2.2 Description

The ADS consists of five Electromatic Relief Valves (EMRVs), which provide overpressure protection for anticipated plant transients and enhance the effectiveness of the Core Spray System under small break LOCA conditions.

The location of the EMRVs, on the main steam headers inside the drywell, is shown in Drawing BR 2002. EMRVs characteristics are listed in Table 6.3-2. The valves are operated by 125 volts dc solenoid operators. The valves open within the 0.65 seconds of energizing the solenoid (based on vendor test data) which is well within the opening time used in the chapter 15 transient analyses. The EMRV opening time is not modeled for the small break LOCA analysis since the opening time is not significant relative to the two minute time delay following the ADS initiation signal. When power to the solenoid operator is removed, the valve closes automatically. Each of the valves has a capacity of 602,900 lbs per hour of steam at overpressure conditions of 1250 psig, which would depressurize the vessel sufficiently fast, even for breaks small enough to be of no consequence from the standpoint of core cooling, such that the Core Spray System can achieve rated flow in less than the allowable time during which the core can remain uncovered. As such, the small break LOCA is bounded by the Large break LOCA for peak clad temperature (see section 15.6.5.5).

The power supply for each of the five EMRVs is provided from two independent 125 volt panels, via individual automatic transfer relays. Normally, EMRVs A, C and E are powered from one panel and EMRVs B and D are powered from the other panel. Valves A, B and E are on the south main steam header, while valves C and D are on the north header. The vented steam from the valves on each header are manifolded and discharge into the torus below the water level.

Following an EMRV discharge and subsequent closing, steam condensing in the line creates a vacuum which tends to draw water into the line. Vacuum breakers provided on the common discharge piping admit noncondensable gases from the drywell into the line to limit the maximum length of the water slug in the pipe and also the length of time that the water elevation in the discharge piping is above the water elevation in the torus. The vacuum breakers are four inch tilting disc check valves with flanged ends.

6.3.1.2.3 Operation

There are two modes for automatic actuation of the EMRVs, in addition to the manual mode. In the automatic depressurization mode, where blowdown through three valves is required, the ADS is actuated on simultaneous occurrence of:

a. High drywell pressure

b. Low-low-low reactor water level

c. Core Spray System operation (as verified by a differential pressure across the core spray booster pump).

The ADS logic is such that the Core Spray booster pump differential pressure on either Core Spray System will enable both ADS Divisions provided coincident reactor vessel triple low water level and drywell high pressure signals are present. This ensures that all ADS subchannels remain operable should a Core Spray System become inoperable. The ADS actuation logic has been established to ensure that pressure relief through the EMRVs does not result in unacceptable dynamic loads on the discharge piping or the torus pressure boundaries. The ADS circuitry:

a. Ensures that only one EMRV discharges into a header containing a water column.

b. Ensures that no single failure of the initiation circuitry allows the EMRVs to operate in an undesirable manner.

Simultaneous operation of various EMRVs is prevented by providing staggered time delay relays. These relays are set so that the low setpoint valve in each header will open and clear the water column in the header before the remaining valves would operate. The total time delay for all five valves to open is less than or equal to 120 seconds.

The ADS circuitry locks open an EMRV which cycles on high reactor coolant pressure while an ADS off-normal condition exists, and opens a second EMRV in the same header, thus ensuring that at least one valve is always open to clear the header. Redundant time delay relays to ADS instrument channels C and D are provided to prevent single failure effects.

After no more than 120 seconds, all valves are open and remain open, discharging reactor vessel steam to the torus, until the open signal to the valves is reset from the Control Room. The overall time delay allows sufficient time to prevent ADS actuation if safe operation can be achieved without the ADS.

In the overpressure mode actuation, each of the five EMRVs is initiated from a pressure switch which constantly monitors reactor vessel pressure. Upon sensing an overpressure condition, the pressure switch completes an electrical circuit to the solenoid actuator of the associated valve, causing the valve to instantly open. When the overpressure condition terminates, and the pressure drops, the pressure switch opens and the electrical signal to the valve is removed, causing the valve to shut automatically. Staggering of the EMRV pressure setpoints, although not required to preserve the structural integrity of the discharge piping and the torus, is maintained for providing additional margin. The valve opening setpoints are listed in the Technical Specifications.

The override keylock switches in the Control Room must be reset simultaneously to permit delay or interruption of automatic depressurization. Annunciator alarms will continuously inform the operator when the ADS Timer Reset Switches are placed in the Bypass position, thereby bypassing ADS actuation. Each time the keylock switches are actuated the time delay cycle is repeated.

A normally closed keylocked control switch is provided in Control Panel 1F/2F for each EMRV for reclosing a spuriously opened EMRV due to fire induced failures in RB 23'-6", MG Set Room and Monitor and Control Panel. The control switch will remove (open) the normal power source from the conductors. The manual operation of the key locked switch will disable the automatic ADS initiation signal and the automatic overpressure function for that specific EMRV, and will disable that EMRV status indication circuit. An annunciator is provided in the Control Room to alarm whenever one or more control switches are placed in DISABLE.

Each valve has a control station switch located in the Control Room. If necessary, each valve can be opened by placing its associated switch in the manual position, which immediately energizes the valve solenoid.

A valve position monitoring system allows the status of the EMRVs to be ascertained in the Control Room. The system utilized piezoelectric accelerometers at each valve location. In the Control Room, the electrical signals which indicate that a valve is open are converted to audio signals to alert the operator.

The ADS serves as a vital part of the ECCS and, as such, requirements for availability and operability have been set forth in the Technical Specifications.

6.3.1.3 Core Spray System

6.3.1.3.1 Function

The purpose of the Core Spray System is to provide for the removal of the decay heat from the core following a postulated Loss-of-Coolant Accident (LOCA), so that fuel clad melting is prevented for the entire spectrum of postulated LOCAs.

The Core Spray System delivers a low pressure spray pattern over the fuel following a LOCA, to limit peak clad temperature below 2200°F. Other criteria which are met by the Core Spray System are as follows:

a. Local oxidation shall not exceed a thickness greater than 17% of unoxidized clad.

b. Hydrogen generation from the metal water reaction shall not exceed one percent of the calculated value for the total metal-water reaction hydrogen generation.

c. A coolable geometry must be maintained.

d. Long term cooling must be provided.

Although the reactor Feedwater System can supply an adequate amount of cooling water to replace that lost through an extended range of pipe break sizes (as long as normal plant auxiliary power is available), the Core Spray System provides an alternate supply of cooling water which is independent of the Feedwater System and which can be operated on emergency power.

The Core Spray System is a low pressure system which supplies cooling water when reactor pressure is reduced to about 285 psig. The system will supply cooling water before the reactor overheats after large or intermediate pipe breaks. To accommodate some intermediate to small pipe breaks, when the Feedwater System is not available, the Automatic Depressurization System (Subsection 6.3.1.2) will provide the initial controlled blowdown to reduce reactor pressure, and thus permit Core Spray System actuation before the fuel uncovers and overheats.

6.3.1.3.2 Description

The Core Spray System consists of two loops, as shown in Drawing GE885D781. Major component data are presented in Tables 6.3-3 through 6.3-6. Each loop consists of two main pumps, two booster pumps, two sets of parallel isolation valves inside and outside the drywell, a spray sparger, and associated piping, instrumentation and controls. The Core Spray System piping has been designed to withstand the design basis seismic event (refer to Section 3.7).

The water supply for the system is held in the torus and is drawn through three strainers into a common header. The header also feeds the Containment Spray System pumps. The Containment Spray System is discussed in Section 6.2. The Strainers are sized to accommodate debris associated with the design basis loss of coolant accident, while passing flow to two core and two containment spray systems. Under these flow conditions, the pressure drop across the strainers is sufficiently low that the pumps maintain adequate NPSH margin. The design requirements for the replacement strainers are contained in Reference 22. The ring header is located such that it is protected by the torus from objects falling from above. The header and the Core Spray System suction piping are designed for 150 psig ASA rating and have been hydrostatically tested to 225 psig. This rating is more than four times the pressure rating of the torus. The ring header has been designed to withstand the design basis seismic event. The suction side of each of the four main pumps is supplied by an individual 12 inch pipe connected to the header. There is one normally open suction valve on each of these four lines.

There is a connection from the Condensate Storage Tank to the suction of each Core Spray System main pump through locked closed manual valves. This allows system flushing and full flow testing with stored condensate. Each loop has a test recirculation line to the torus, provided with motor operated test valves, for full flow testing without discharge into the reactor vessel. Flow and pressure instrumentation are provided in the Control Room for each loop. The piping up to the pump discharge valves into the reactor, the piping is fabricated of stainless-steel designed for 300 psig and 350°F. From these pump discharge valves into the reactor, the piping is fabricated of stainless-steel designed for 1250 psig and 575°F. The low pressure portions of the system are vented back to the Torus. This feature prevents over pressure.

The discharge from each of the Core Spray System main pumps flows through a check valve, and into one of two headers (there is a header for Loop I and another for Loop II). The header connects the discharge of the main pump to the suction of the booster pumps via a 10 inch line. This line branches out into three pipes. Two of these are the suction lines to the booster pumps, the other one is a bypass line. The two discharge lines from the booster pump and the bypass line are provided with check valves and they join together into another 10 inch line which is routed to a Core Spray System sparger.

In each Core Spray System loop there are motor operated isolation valves outside the drywell, and testable check valves inside the drywell. Flow for each loop is through a normally open motor operated valve (with circuit breaker racked out or locked off), two parallel normally closed motor operated valves, a single line at the containment penetration, two parallel check valves, and one locked open manually operated valve into the sparger.

The Core Spray System is designed to function throughout the postaccident period. It has sufficient redundancy to ensure that the system will be pressurized to pressures greater than the peak calculated containment accident pressure (Pa) regardless of a possible single active failure to the system. The system piping is seismic qualified forming a closed loop outside containment. Consequently, the isolation valves of this system are not relied upon to perform an isolation function to prevent leakage of containment atmosphere at any time throughout the post-accident period. Consequently, these valves are not containment isolation valves as defined by Appendix J and, therefore, need not be tested. The Core Spray System need not be vented and drained during the Type A test and the isolation valves need not be local leak rate tested since Appendix J does not require this testing.

The manual isolation valves inside the drywell discharge through an eight inch line which reduces to a five inch line before the vessel penetration. From each vessel penetration, the five inch pipe extends half way around the outside of the core shroud at 150 degrees and 330 degrees, with one loop penetrating above the other. Each 3 1/2 inch sparger inside the shroud extends half way around the inside of the shroud in both directions. Thus, each sparger completely encircles the core. The spray distributor arrangement for each sparger consists of 56 full jet nozzles and 56 open elbows. For the lower sparger, the full jets are set at a negative elevation angle of 8 degrees below the horizontal, and the flow elbows are set at a negative 6 degrees. For the upper sparger, the full jets are set at a negative elevation angle of 11 degrees below the horizontal, and the flow elbows are set at a negative 8 degrees. All elbows point to the vertical center line of the reactor and are tilted down to give the optimum spray distribution to all fuel channels. Differential pressure switches provide indication of a break in the annulus region.

The Fire Protection System is connected to each of the two Core Spray System loops. The purpose of this connection is to provide a backup supply of cooling water to the spargers. The Fire Protection System is described in Subsection 9.5.1.

In order to protect the Core Spray System main pumps in the event of a leak in the suction header piping, or of failure of the pump casings, the pump compartments in the four corners of the Reactor Building basement (corner room) are flood protected by means of water tight doors, sealing pipe penetrations in the wall, floor drain ball check valves, and automatically operated valves on the sump drain lines.

The required total Core Spray System flow is that needed to remove the fission product decay heat generated 30 seconds after shutdown from infinite reactor operation at full power. It takes 5 seconds or less to shutdown the reactor after DBA initiation. Therefore, required Core Spray rated flow must be achieved at 35 seconds (5 seconds plus 30 seconds for decay heat). Spray cooling tests and spray distribution tests (refer to Subsection 6.3.3) have been used to establish the core spray flood rate. A detailed discussion of the design basis accident is presented in Section 15.6.5.

Each Core Spray System loop is provided with two full capacity main pumps and two full capacity booster pumps. The rated design capacity of the main and booster pumps is 3700 gpm (Table 6.3-5). The flow requirements to be delivered to the reactor core from the main and booster pumps following a design basis LOCA is described in Section 6.3.2.2.3.

The Core Spray System is designed for a very high level of reliability and availability. Specific features can be summarized as follows:

a. The system is designed in accordance with the ASA B31.1 (presently ANSI B31.1) piping code.

b. Valves are installed in parallel where they must open to provide core spray flow.

A Core Spray Filling System has been incorporated to the Core Spray System to eliminate the use of condensate to keep the Core Spray System piping filled with water. A water leg is maintained in the piping to preclude any danger of water hammer when the system goes in operation. Fill pumps (one in each system) take suction from the torus and discharge torus water to the main piping which overflows through the pump minimum flow recirculation piping back to the torus. This assures that the main piping remains full since the recirculation lines are located at a higher elevation than the main piping.

The Core Spray Filling System has been designed for 300 psig at 130°F. The filling system and the Core Spray System pumps are interlocked so that the fill pump for System I or II trips when the backup main pumps are started in the respective system. The filling system is isolated by means of check valves and manually operated valves. The pumps for the filling system start automatically when the Core Spray System (backup) main pumps are shut off.

Core Spray System safety-related motor-operated valves are included in the Generic Letter (GL) 89-10 Motor-Operated Valve (MOV) Program as noted in the OCNGS Program Description for NRC Generic Letter 89-10 Motor-Operated Valve Program. This program has reestablished the design basis for safety-related motor-operated valves. Critical design basis assumptions such as design basis differential pressure, safety function - open vs. close, minimum available AC/DC voltages, actuator gearing, torque switch control logic, valve factors, stem friction coefficients, and valve stroke times have been established in assessing GL 89-10 design basis capability. Plant changes or activities which can affect these design basis assumptions must consider the affect on the capability of GL 89-10 motor-operated valves to perform their safety function and on safety margins established for these valves.

6.3.1.3.3 Operation

The Core Spray System is an essential engineered safeguard which must be available for use during all modes of reactor operation and during refueling. Because of its importance, the operability requirements for the system have been incorporated in the Technical Specifications.

The system can be started manually, or by automatic trip signals generated when a low-low reactor water level and/or a high drywell pressure condition is detected. These conditions generally indicate a pipe break. Both Core Spray System loops and both Emergency Diesel Generators will start upon the detection of only one high pressure or one low-low level condition. The Emergency Diesel Generators start in order to supply power to the Core Spray pumps (and other vital components) in the event of loss of the normal electric power supply.

The suction and discharge valves of the recirculation loops have a direct impact on the communication of reactor coolant between the reactor downcomer region and the reactor core region. If the suction and discharge valves of all five recirculation loops are closed, a water level reduction within the reactor core region will not result in a corresponding water level reduction within the reactor downcomer region. The instruments that detect low-low reactor water level are located within the reactor downcomer region. The closed valves will isolate the flowpath between the reactor downcomer region and the reactor core region. For this reason, the suction and discharge valves of at least one recirculation loop shall remain in the full-open position when 1) there is irradiated fuel in the reactor pressure vessel and 2) the reactor coolant temperature is greater than 212ºF. There are two exceptions to the full-open valve/ recirculation loop configuration.

The first exception is when the reactor water level is greater than 185 inches above TAF and the reactor coolant temperature is less than 212ºF. With the reactor water level greater than 185 inches above TAF, the reactor coolant will communicate between the reactor downcomer region and the reactor core region. Any decrease in reactor water level will be detected by the downcomer instruments and will allow for appropriate operator action.

The second exception is when the steam separator and steam dryer are removed from the reactor pressure vessel and the reactor coolant temperature is less than 212ºF. With the removal of the steam separator and steam dryer, the reactor coolant will communicate between the reactor downcomer region and the reactor core region to below the Core Spray System actuation setpoint (low-low reactor water level) of 86” above TAF. Also, Plant Technical Specification No. 2.1.D ensures that reactor water level will be maintained 4’-8” above TAF.

If the suction and discharge valves of all five recirculation loops are closed, a level reduction in the core region will not result in a corresponding level reduction in the downcomer where Lo-Lo level is measured. For this reason, one recirculation loop suction and discharge valve should be open when core spray is required. One exception to this is when the reactor is in the refueling mode with level above 185 inch TAF and temperature is less than 212°F and no work is being done which could cause the core water level to be reduced. Another exception is when the steam separators and dryers are removed.

With level above 185 inch TAF, there is communication between the core and downcomer regions. Any level decrease will be seen in the downcomer instruments and allow for appropriate operator action. If work is being done on the reactor vessel which could cause a rapid reduction in water level, it is assumed that insufficient time would be available for operator action. As a result, one loop open is required.

With steam separators and dryers removed, there is free communication between the core and downcomer regions to below the Lo-Lo actuation setpoint.

The sequence of events following actuation is as follows:

a. One preferred (preselected) main pump in each loop starts. Should either pump fail to start, the second pump in that loop will receive a signal to start in nominally ten seconds of the actuating signal.

b. Both Emergency Diesel Generators start after a nominal 10 second delay and remain in an idle (no load) condition in anticipation of a loss of offsite power.

c. Upon sensing both an actuation signal and a discharge pressure signal of its associated main pump, the preferred booster pump will start. The minimum discharge pressure for Core Spray System Loop I main pumps is 105 psig; for Loop II it is 140 psig. This difference results from the head requirements due to booster pump location. If the preferred pump in either loop fails to start, in nominally five seconds of sensing the condition a start signal is generated to the alternate booster pump in that loop. The alternate booster pump of each loop is interlocked with the primary (preferred) booster pump of the other loop. This interlock prevents the backup booster pump from automatically starting while the primary pump on the other core spray loop is running. Because both pumps are powered from the same electrical division, the interlock prevents the automatic loading of two (2) booster pumps on the same electrical division. Manual starting of the backup booster pumps is not affected and is subject only to the availability of capacity on the diesel generators. Subsequent failure of the second primary booster pump will result in the automatic start of both backup booster pumps. Only one (1) booster pump and two (2) main pumps are necessary to provide sufficient flow to the core, as demonstrated in the 10CFR50, Appendix K Analysis (Reference 20).

d. All motor operated valves in each loop are open with the exception of the parallel isolation valves. Since the pressure at the discharge of the booster pumps is approximately 300 psig, the parallel isolation valves will not open until reactor pressure is below this level. Once reactor pressure drops below 285 psig, and an actuation signal to the Core Spray System is present, both parallel valves in each loop open. Each valve has a 100 percent flow capability, so that failure of one valve will not reduce flow. If the break is not large enough to reduce reactor pressure below 285 psig, the Automatic Depressurization System operates automatically.

After Core Spray System flow has been established into the reactor, the torus provides an essentially unlimited supply of cooling water. The leaking water from a pipe break inside the drywell will overflow through the drywell vents into the torus. The alternate supply of cooling water for the Core Spray System is the Condensate Storage Tank, and as a last resort water is provided by the Fire Protection System. The heat being absorbed by the water which flows back to the torus is removed to the Ultimate Heat Sink by the heat exchangers in the

Containment Spray System. The Ultimate Heat Sink is discussed in Sections 9.2 and 10.4. The Containment Spray System is discussed in Section 6.2.

The initiation logic for the system is described in Section 7.3. Manual actuation is possible from the Control Room.

It should be noted that the recirculation piping provides a minimum flow rate by a throttling valve for each set of main pumps when they operate. When the main pumps start, water flows through the booster pumps (initially idle until the required main pump discharge pressure is achieved) and a minimum flow of approximately 100 gpm is diverted past the booster pumps back into the torus.

6.3.2 System Design

6.3.2.1 Schematic Piping and Instrumentation Diagrams

The P&IDs for the Isolation Condenser, Automatic Depressurization and Core Spray Systems are presented in Drawings GE148F262, BR 2002, and GE885D781respectively. Instrumentation and controls for the Emergency Core Cooling Systems are described in Section 7.3.

6.3.2.2 Equipment and Component Description

The major equipment and components of the Emergency Core Cooling Systems are: the Isolation Condensers, the Electromatic Relief Valves, the Core Spray System main pumps, the Core Spray System booster pumps and the Core Spray System sparger. These are described in subsequent paragraphs.

6.3.2.2.1 Isolation Condensers

The Isolation Condensers are described in detail in Subsection 6.3.1.1.2. Pertinent design data for the Isolation Condensers and other components of the ICS are presented in Table 6.3-1.

6.3.2.2.2 Electromatic Relief Valves

The Electromatic Relief Valves (EMRVs) are briefly described in Subsection 6.3.1.2.2, general characteristics of the valves are listed in Table 6.3-2. The EMRVs are electrically actuated pressure relief valves. They can be operated at will by closing a switch or automatically with a pressure sensitive element.

The valve controller consists of a pressure sensitive element composed of a Bourdon tube which actuates electrical contacts, and a heavy duty relay to switch the solenoid load. The control station is equipped with a three position switch, and with valve open/close pilot lights. The pressure element and the relays, working in conjunction with the control station, supply electrical power to the solenoid assembly which operates the pilot valve. In turn, the pilot valve controls the opening and closing of the main valve.

The controller is actuated by the pressure in the reactor vessel. The construction is such that it will make and break electrical contact with a difference in pressure of one percent of the set pressure. Within the controller there is a dual control pressure switch. When the pressure increases to the setpoint, the high pressure switch is actuated and completes the relay circuit that energizes the valve solenoid. The low pressure switch provides for relay control below the actuation value of the high pressure switch, thereby allowing an adjustable blowdown range for the EMRV. The action of the Bourdon type pressure sensing element makes possible an extremely sensitive regulation.

6.3.2.2.3 Core Spray System Pumps

The Core Spray System main pumps and booster pumps (Table 6.3-5) are designed to run in series to develop a combined total head of 655 feet at 3700 gpm (that includes approximately 100 gpm for continuous recirculation flow to the torus through a throttling valve). At least one main pump and one booster pump from one system, plus one main pump from the other system are needed to ensure adequate core cooling under the worst case LOCA conditions. The actual required flows for Systems 1 and 2 are as follows:

- System 1 – One Main and Booster Pump – 3400 gpm
  One Main Pump – 2200 gpm

- System 2 – One Main and Booster Pump – 3640 gpm
  One Main Pump – 2360 gpm

The System 2 flow rates include expected flow losses through cracks in the core spray sparger. The flow rates are delivered at a vessel pressure of 110 psig above torus air space pressure.

Additional leakage paths were identified (Reference 29) consisting of leakage through the piping vent holes and a postulated 360° crack in the slip joint welds. Leakage has been included in assessment of adequacy of the system for delivery in accident scenarios.

There are no shutoff valves between the main and booster pumps.

The pumps for one loop are located in the southwest area of the Reactor Building, and the pumps for the other loop are located in the northwest area of the Reactor Building. This separation protects against damage to both loops from one accident. The main pumps are located at floor El. (-)19'-6". One set of booster pumps located in floor El. 23'-6", the other set is at El. 51'-3".

The pumps are controlled from switches in the Control Room. In the NORMAL position, the pumps respond to automatic operation.

The characteristics curves for the Core Spray System main pumps and booster pumps are shown in Figures 6.3-4 and 6.3-5, respectively.

A calculation has been performed to assess the available Net Positive Suction Head (NPSH) for the core spray pumps that account for accident debris accumulating on the suction strainers. This calculation (Reference 27) demonstrates a need for 1.25 psig overpressure for the first hour of the accident to ensure sufficient available Net Positive Suction Head. This overpressure is required to allow core spray (main and booster pump) flows of 5000 gpm. The 1-hour time was selected as the bounding time for the operators to trip both booster pumps. With the booster pumps removed from service there is sufficient NSPH without containment over pressure. The booster pump associated with pump NZ01C will need to be tripped prior to one hour, but no sooner than 10 minutes to avoid cavitation. The timing of the booster pump trips was evaluated by General Electric (Reference 28) and shown to have no impact on the

Appendix K analysis. In addition, the tripping of the second booster pump is required well beyond the analysis for the docketed appendix K evaluations.

6.3.2.2.4 Core Spray System Spargers

The reactor vessel contains two independent Core Spray System spargers, which are fed by the two separate loops of the Core Spray System. Each sparger assembly consists of two 180 degree segments of formed 3 1/2 inch Schedule 40S Stainless Steel piping. Each segment contains 28 spray nozzles and 28 open elbows (112 total per sparger assembly). The 180 degree sections of each sparger assembly are connected to one five inch Schedule 40 inlet piping which directs spray water, when the system is actuated, from the pumps, through a reactor vessel nozzle and penetration in the shroud to each half of the sparger assembly. Each 90 degree segment of the five inch piping is supported at the midpoint by a bracket welded to the shroud.

Each 180 degree half of the sparger is supported at the location of the five inch inlet pipe connection, which is welded to the shroud, and by three, approximately equally spaced support brackets on either side of the central inlet pipe connection. The support brackets consist of 3/8 inch thick vertical gusset plates, with 1 1/2 inch wide bearing pads, which support the sparger arms in the radial and vertical directions. The sparger arms are free to slide in a circumferential direction (relative to their inlet connection to the shroud) as required to accommodate any differential thermal expansion between the spargers and the shroud during injection of cold core spray water.

The Type 304 Stainless Steel pipe used for the spargers contains no more than 0.06 percent carbon. Following annealing, it was bent cold to a 91 inch radius. Table 6.3-6 identifies the characteristics of the spargers and nozzles. The spargers have never been used, and are exposed to water and steam at 500 to 550°F, and less than 200 ppb concentration of oxygen.

Inservice inspection of the reactor internals has identified existing and potential cracks in the Core Spray System sparger assemblies. In order to provide additional structural margin, redundant mechanical supports have been installed at locations where the number and position of cracks create concern about sparger integrity. The repair clamps are designed to carry the following design loads:

a. Residual bending stresses from fabrication and installation.

b. Hydraulic loads upon actuation of the Core Spray System.

c. Thermal gradient loads as a result of injection of cold water.

All accessible surfaces and welds of both core spray spargers and repair assemblies are inspected each refueling interval as required by facility operating license condition paragraph 2.C.5.

Analyses have demonstrated that sufficient margin exists to compensate for existing defects in the annulus piping system and, therefore, these defects do not impair either the integrity of the system or its ability to deliver the required flow. For analysis of existing defects at, 14R, and NRC approval see References 23 and 24.

The function of the spargers is to distribute the spray flow in a manner that ensures that each fuel bundle receives adequate flow. Tests performed during the original design of the system have shown that adequate distribution is obtained for loop flows of 3100 gpm to 4500 gpm. Minimum acceptable flow rates can be found in the Technical Specifications.

6.3.2.3 Applicable Codes and Classifications

The applicable codes for ECCS components are provided in Tables 6.3-1 through 6.3-6, as appropriate. Piping is generally ANSI B31.1, and the overall system is designed to seismic criteria (refer to Subsection 3.7).

6.3.2.4 Material Specifications and Compatibility

The materials utilized for ECCS components are presented in Table 6.3-1 through 6.3-6, as applicable.

6.3.2.5 System Reliability

6.3.2.5.1 Isolation Condenser System Reliability

System Characteristics

The ICS is described in Subsection 6.3.1.1. The system consists of two full capacity isolation condensers, with separate steam supply and condensate return lines. The ICS operates by natural circulation, without the need for driving power other than from the dc electrical system used to open the outboard condensate return isolation valves. Since dc power is available from separate and redundant sources, the system can be initiated with a total loss of ac power. To achieve full operational effectiveness of the ICS, the reactor recirculation pumps associated with the Isolation Condensers (“A” & “E”) must trip, since the condensate return lines from the Isolation Condensers are connected to the suction side of the recirculation pumps. Therefore, coincident with the automatic initiation of the ICS, a trip signal is generated to these pumps from the Recirculation Pump Trip System. This trip system is discussed in Section 7.1.1 and 7.6. The Technical Specifications require that at least one recirculation suction valve and its associated discharge valve remain open to establish a natural circulation path.

The low-low reactor water level initiation signal for the Isolation Condenser was added as a backup to the high reactor pressure signal in order to assure actuation of the ICS for all break sizes.

Other potential failure modes which have been analyzed for the ICS include: failure of the dc motor operated valves to open on demand, and failure of dc power to the valve controls. Redundancy in the design of the system has considered these failures, in protecting against single component failure.

Isolation Condenser High Flow Trip Setpoints

The high flow trip setpoints for the Isolation Condensers provide automatic isolation if a pipe rupture is detected. Three hundred percent of normal flow was selected as the flow rate which would indicate a rupture. This flow setpoint, with an appropriate time delay (maximum allowable 29 seconds, Reference 26), will not cause spurious system isolation due to operating transients, and also affords adequate protection against pipe rupture, Reference 26.

The 300% settings are based on providing adequate protection for cases where a pipe break occurs in the steam or condensate lines to the condenser outside the Containment. For breaks inside the Containment, the consideration associated with the Loss-of-Coolant Accident analyses apply and are less severe because the break sizes here are smaller. For these cases, it must be shown that the consequences will be no more severe than those from some other accident that has already been analyzed and reported. There will be two general categories of cases to consider, one with the ICS in standby readiness and the other with it in operation.

In view of the fact that the ICS is seldom in service, the most likely break occurs with the outboard condensate isolation valve closed. Therefore, only breaks in the steam line need be considered for this case. However, reference to the Loss-of-Coolant and Main Steam Line Break Accidents analyses (Chapter 15) reveals that the core does not uncover and fuel cladding temperatures do not increase for any steam line break up to the maximum size possible with the main steam lines. Hence, the same protection would result with any break on the smaller isolation condenser steam line.

Where a break occurs while the ICS is in service, the sequence of events and results would be more complex because there would be differing combinations of steam and liquid issuing from each side of the break. This would depend on such things as the location of the break, the operation of the secondary side of the condenser, the sequence of reactor shutdown (since actuation occurs from high pressure), and other conditions. The possibilities can be characterized by two different sets of conditions. One set considers only breaks in the condensate line equivalent in sequence and results to a liquid line break in the Loss-of-Coolant Accident analysis, and the other set considers the same thing for steam line breaks. Within each set, two cases were considered.

One case assumes that the break is the maximum possible without causing an isolation (i.e., equivalent to 300% of normal flow) and the other case assumes that the maximum line break occurs and the isolation valve closes in the maximum allowable time. For the steam line break cases, the core is not uncovered, as stated above, so protection is adequate. Further, the complete break causes steam flow rate of about 12 times normal, which demonstrates the sensitivity of the 300% setting. For the condensate line break equivalent to a 300% flow, without isolation, a peak cladding temperature no higher than 1200°F would be obtained (for the most degraded condition of the ECCS as assumed in the LOCA analyses). For the complete condensate line break, the isolation trip occurs and the valves close in a maximum time of 81.5 seconds (See Table 6.3-1, Sh. 2 of 4, Note 1). The calculated maximum condensate flow rate associated with this event is about 48 times normal flow, and peak fuel cladding temperatures do not go above 1200°F by the time the isolation valves close. Hence, condensate line isolation settings for these cases also provide adequate protection and sensitivity.

During the Startup Test Program (Section 14.2), the heat removal capacity of the Isolation Condensers was measured. The minimum capacity measured was 70 MW thermal compared with the design capacity of 56 MW thermal. The normal flow computed from this minimum heat removal capacity is 367,000 lb/hr.

The setpoint values for the high flow isolation presently in the Technical Specifications are more conservative than those computed from the startup test results.

Single Failure Evaluation

No single failure would adversely affect the ability of the ECCS to perform as required by 10CFR50.46.

6.3.2.5.2 Automatic Depressurization System Reliability

System Characteristics

The ADS is described in Subsection 6.3.1.2. The system is composed of five Electromatic Relief Valves (EMRVs) and associated piping instrumentation and controls. Each valve features unique pressure switches (Bourdon Type), dc transfer switches, dc power fuses, and time delay relays. There is a redundant 125 volt dc power source for all valves, consisting of station batteries B and C. ADS Channel I logic is supplied power from 125 Vdc panel D, and Channel II logic is supplied power from 125 Vdc panel F.

The design of the ADS has considered seismic criteria, and physical separation has been provided between the two redundant and independent divisions of the system. The ADS is adequately protected from fire damage.

The sensors used to generate actuation signals for the ADS (Subsection 7.3) have been selected for high reliability. Redundancy is provided, and the instrumentation is so arranged that testing can be carried out frequently and thoroughly. Instrumentation is not expected to be a limiting factor on ADS reliability.

Single Failure Evaluation

No single failure in the ADS can prevent it from performing its intended safety function. The system meets the single failure criteria of IEEE 279-1972.

6.3.2.5.3 Core Spray System Reliability

System Characteristics

The Core Spray System consists of two separate loops. Each loop has redundant active components; that is, components which must move to contribute to system success are duplicated. For this reason, each loop has redundant pumps operating in parallel, two motor operated normally closed valves in parallel, and two check valves in parallel. The other motor operated valves are not duplicated because their usual position (open or closed) is the position which will contribute to overall system success. The two core spray system loops have sufficient redundancy of active components to ensure adequate core cooling will be accomplished following a design basis LOCA, while considering a single active failure. The core cooling requirements will ensure the acceptance criteria of 10CFR50.46 will be met (Refer to Section 15.6.5). The system is required to deliver flow from one main pump and one booster pump from one loop, plus one main pump from the other under the worst case LOCA conditions.

The Core Spray System has been evaluated for three types of Loss-of-Coolant Accidents (LOCAs). These are:

a. Large LOCA inside Containment (which envelops a small break above the core)

b. Small LOCA inside Containment and below the core

c. Any LOCA outside Containment

For the types of LOCAs in items a. and c. above, the Core Spray System is more likely to fail on demand than when performing its safety function after a successful start. The reason for the lower probability of system failure on demand for the case of a small LOCA inside containment and below the core is that the failure probability of the control logic and pumps is included in the ADS analysis. The analysis of the Core Spray System has shown that:

a. The Core Spray System is a highly redundant system which is immune to failure due to hardware failures within the system.

b. The system is redundant with respect to its ac and dc electric power supplies.

c. The system is more likely to fail on demand than during operation. The corresponding probabilities depend on the type of LOCA.

d. For LOCAs inside the containment, failure on demand of the system is dominated by human error which results in failure of all four booster pumps to start. For LOCAs outside of containment, human error during the sequential testing of the low-low reactor water level sensors could result in failure to start all four main pumps.

e. Another significant contributor to failure of the system on demand is the set of two-event failures in the electric power system which controls and powers the Core Spray System.

f. The largest contributor to electric power failure is failure of ac power at one 4160 volt bus on demand, combined with the unavailability of a unit substation bus due to transformer failure on the redundant power side prior to the LOCA.

g. Dependent failures, which fail the entire system after it has started, are not evident.

h. Electric power failures are dominant for long term system operation.

i. The Core Spray System is immune to dc power failures once the system has started, if the motor operated isolation valve switches are placed in the open position immediately after the system is injected into the Reactor vessel.

A design modification to the electrical system effects the transfer of one set of redundant components for each loop to the redundant emergency bus upon failure of the bus to which the component is normally connected.

Under Loss-of-Coolant Accident conditions, the Core Spray System can successfully operate only when both:

a. Electric power is available; that is at a minimum, ac power is available on demand and at least one emergency 4160 volt ac bus and dc power are available on demand at the associated dc center.

b. The torus remains intact following the LOCA.

Single Failure Evaluation

No single failure would adversely affect the ability of the ECCS to perform as required by 10CFR50.46.

6.3.2.5.4 Evaluation of Valve Flooding After LOCA

There are no valves within the primary containment that would become submerged as a result of a postulated LOCA. Because of the design of the Mark I Containment, any water released into the drywell would drain into the torus.

The drywell floor is at El. 10'-3". The bottom of the vent header draining to the torus is at El. 12'-9". There would be only 2'-6" of water in the bottom of the drywell under these conditions, and this would be the only accumulation of water in the drywell. There are no motor operated valves in this region of the drywell.

6.3.2.5.5 Evaluation of Diesel Generator Voltage During Loading

The loading sequence for a single Emergency Diesel Generator ensures that the Core Spray System pumps, the Reactor Building Closed Cooling Water System pumps and the Control Rod Drive feed pumps will not start simultaneously (refer to Section 8.3).

6.3.2.5.6 Ambient Cooling of Core Spray Pump Compartments

The pump compartments are cooled by the Reactor Building Closed Cooling Water System. Each compartment is provided with coolers sized to extract the heat generated by pump operation.

Loss of corner room cooling has been evaluated with respect to long term operation of the Containment and Core Spray Systems. As a result of this analysis, the pumps can operate more than 48 days without mechanical cooling from the coolers.

6.3.2.5.7 General Separation Criteria

The separation criteria for the facility are in accordance with "Oyster Creek Project, Separation Practices for Safeguards Systems" by APED Engineering, General Electric Company (Revised November 26, 1968).

The criteria for physical separation and electrical independence of safety related equipment and circuits applied to ECCS modifications, subsequent to commercial operation of the facility, generally comply with the requirements of IEEE 279-1971 and IEEE 308-1974, except as noted in the documentation submitted to NRC for approval of said modifications. Physical separation between redundant circuits in Panels 18A and 18B for the ECCS has been supplemented.

6.3.2.5.8 Control Room Labeling

The ECCS equipment, and the switches mounted on the control board in the Control Room have been clearly labeled to indicate division assignment. The Emergency Diesel Generator power supply for components are indicated next to the Component Control Switch. This is intended to reduce the potential for operator error, and thus enhance the reliability of the system.

6.3.2.6 Protection Provisions

Protection provisions against pipe whip, jet impingement forces and missiles; equipment qualification; and seismic qualification of components are discussed in Sections 3.5, 3.6, 3.10 and 3.11, as appropriate.

6.3.2.7 Provisions for Performance Testing

The ICS, ADS and Core Spray System are so designed that they can be tested thoroughly from sensor to output to the greatest extent practicable. These tests are performed on a regular basis, as established in the Technical Specifications.

Special provisions for performance testing are provided for the Core Spray System. One test block valve (per loop) is located between the booster pump discharge header and the isolation valve in each loop. These valves are in the open position with their supply breakers opened (rendered inoperable) during normal operation, and are closed during certain monthly system test and surveillance procedures. They are interlocked to open on automatic system initiation. Valve position is indicated in the Control Room.

Test line piping connects the torus to the loop flowpath at a point between the booster pump discharge header and the test block valve in each loop. During normal operations, both test line valves (one for each Core Spray System loop) are closed. They are interlocked to close on automatic system initiation.

6.3.2.8 Manual Actions

The ECCS is automatically actuated. Manual actuation provisions upon failure of automatic actuation are discussed in Subsection 6.3.1.

6.3.3 ECCS Performance Evaluations

6.3.3.1 Historical Background of Experimental Testing Used for Design Basis

The original design of the Oyster Creek core consisted of 560 fuel bundles in a 6 x 6 fuel rod array configuration. The design later changed to 7 x 7, 8 x 8, and 9 x 9 fuel assemblies. The present and subsequent cores are anticipated to be made up of 10 x 10 fuel rod bundles. As a result of this change in the geometry of the core and as a result of changes in NRC regulations, the Emergency Core Cooling System performance has undergone a series of reevaluations.

The discussion presented in Subsections 6.3.3.1 and 6.3.3.2 includes that information required to support basic ECCS performance parameters and may not be pertinent to all aspects of present ECCS performance evaluations. The evaluations of core cooling effectiveness for the present core cycle are incorporated by reference. (See Subsection 6.3.3.3.)

6.3.3.1.1 Reactor Core Spray Cooling System Test Program

This first test program was undertaken to determine the effectiveness of reactor core spray cooling. A spray cooling distribution facility simulating a reactor core spray sparger ring and nozzles and fuel assemblies was employed to determine the amount of spray reaching each assembly in a reactor core. A second facility, consisting of a prototype fuel assembly with 36 electrically heated rods (6 x 6 assembly), thoroughly instrumented with thermocouples and a channel box, simulated reactor conditions. Finally, a third phase of the test program determined the degree of reactor core coverage for adequate cooling.

The minimum flow substantiated by the tests was 0.05 gpm per fuel rod. To obtain the required reactor core spray cooling system loop flow rate for a given reactor core it is necessary to multiply the total number of fuel rods by 0.05 gpm and divide the result by the minimum distribution factor expected for this design. The minimum distribution factor of 0.40 represented the most severe situation that was anticipated for the actual reactor configuration and was therefore used to establish reactor core spray cooling system loop flow rates.

The results of this test program, when applied to the 560 bundle core for the OCNGS yields a minimum flow per bundle of 2.45 gpm, and using the 0.40 distribution factor on an average flow per bundle of approximately 6 gpm establishes the required flow of 3400 gpm. This flow ensured that the minimum flow per bundle was within the range tested, and allowed for uneven distribution across the core.

6.3.3.1.2 Emergency Cooling Tests

Subsequent to the tests discussed in Subsection 6.3.3.1.1, a 49 rod test assembly was designed to have the same dimensions and geometry as a fuel bundle for the General Electric Product Line reactors (e.g., Oyster Creek). The test configuration was a full scale 49 rod mockup for one bundle, with provision for simulating the cooling water flow along the outside of this bundle. The heater rods were 166 inches long (144 inch heated length), electrically heated, had a cosine power distribution, and were stainless steel clad. Reactor local peaking was also simulated. The test configuration provided for a separate coolant supply to internal and external channel regions. Detailed temperature measurements were made with the extensive use of Chromel-Alumel thermocouples on the heater rods and two channels. Flow was controlled and measured. The program included both spray and flooding tests and concluded that adequate cooling of the core could be achieved.

6.3.3.1.3 Rod Perforation Tests

In early 1967 a program was undertaken to show that rod perforations are not detrimental to the effectiveness of the core spray. The heat transfer effectiveness of core spray cooling was established by APED with full scale heat transfer tests on non-perforated stainless steel bundles (Subsection 6.3.3.1.2).

The objective of rod perforation tests was to determine the deformation and failure characteristics of BWR fuel cladding during the heat up following a loss of coolant incident.

The results of these tests are summarized below:

a. The stress of failure could be predicted rather well by the ultimate strength.

b. Failure points indicated a preferred axial failure position, corresponding approximately to the position of maximum temperature near the middle of the heated length.

c. Tube perforations occurred in the longitudinal direction.

6.3.3.1.4 Evaluation of Updraft Effects

The original core spray distribution tests conducted at APED implied that a distribution with a minimum distribution factor of 0.4 could be achieved. The test section on which these tests were run, however, was not versatile enough to answer all of the questions that arose; namely, in the area of steam updraft effects and the effect of larger spray sparger diameters.

To resolve these uncertainties two test sections were constructed. One was a full scale mockup of the upper section of the BWR core used to determine spray distribution. The second test section was a single channel arrangement to determine the effect of steam updrafts on the amount of spray entering the channel.

Extensive tests were conducted to determine the nozzle settings that produce the optimum spray distribution. Two different types of spray nozzles and three angle settings were originally employed to obtain a good spray distribution. Tests showed good flow over most of the core except at the very center region and at the very edge. In these regions the distribution factor dropped to about 0.3.

Based on test results it was concluded that the steam updrafts expected in the core are in a range where it will have little or no effect on the amount of spray flow entering a channel.

6.3.3.1.5 Spray Distributor Tests

The initial design of the spray distributor arrangement for the OCNGS (Subsection 6.3.3.1.4) consisted of 56 full jets and 56 Vee jets alternately distributed around each of the sparger rings at equal intervals. The 56 full jet nozzles were set at an elevation angle of 10 degrees below the horizontal. Of the 56 Vee jet nozzles, 28 were aimed 4 degrees below the horizontal and 28 at 8 degrees below the horizontal.

The redesigned spray distributor arrangement presently installed at Oyster Creek consists of 56 full jet nozzles and 56 open elbows for each of the sparger rings. This arrangement is identical to the preceding arrangement, except the Vee jets are removed and the flow exits through the open elbow. Thus, there are 56 full jets and 56 flow elbows alternating around both of the spargers at equal intervals. Each 3 ½ inch sparger inside the shroud extends half way around the inside of the shroud in both directions. Thus, each sparger completely encircles the core. For the lower sparger, the full jets are set at a negative elevation angle of 8 degrees below the horizontal, and the flow elbows are set at a negative 6 degrees. For the upper sparger, the full jets are set at a negative elevation angle of 11 degrees below the horizontal, and the flow elbows are set at a negative 8 degrees.

The open flow elbows greatly reduce the core spray velocity and result in a much more "quiet" and less sensitive system. Effects of total flow rate, angle setting, and updraft were investigated.

Tests were run with flows as low as 3100 gpm and as high as 4500 gpm. The minimum design flow of 2.45 gpm per bundle which was established as the design basis (Subsection 6.3.3.1.1) was achieved across the core with either flow rates.

The maximum energy transfer rate to the core spray in the event of a Loss-of-Coolant Accident for the 7x7 fuel array assembly configuration would be approximately 3% of rated power. Energy is initially stored in the fuel until the temperature has risen to the value where the heat being transferred equals the heat being generated. At this point in time the heat being generated is equal to the decay heat generation. For the average channel this occurs at 300 seconds and the decay heat equals 3% of rated power.

The hot channel peaks earlier in time at about 230 seconds and the decay heat rate is about 3.2%. The maximum steam updraft can be calculated from a heat balance using the 3% decay heat and the design flow of 2.45 gpm. This results in a maximum steam flow rate of approximately 400 lb/hr for the hot channel and 260 lb/hr for the average channel.

6.3.3.1.6 Application of Early Test Results

The flow rate specified for the OCNGS was based on early tests conducted on 36 rod full length fuel assemblies. Subsequent tests on 49 rod assemblies at 2.45 gpm and at significantly lower flow rates showed that cooling was still possible at reduced flows.

Since all the water entering the shroud does not go into the fuel assemblies, the total flow to be supplied was based on early flow distribution tests. These led to the belief that of the water entering the vessel the minimum into any assembly would be about 0.4 of the amount which could enter if it were perfectly distributed across the core. The spray distributor tests discussed in Subsection 6.3.3.1.5 provided proof that the 0.40 distribution factor could be attained with the OCNGS design.

6.3.3.1.7 Evaluation of Malfunctions Affecting Spray Distribution

Several Blocked Nozzles

It is highly improbable that nozzles will be blocked because a screen device is installed, upstream from the core spray sparger, such that no particles larger than the core spray nozzle opening will be allowed to enter the sparger. The screen is 6 by 6 mesh, 0.035 inch wire with 0.132 inch openings.

This fine mesh screen is supported by diamond mesh expanded metal. Suction strainers are located at the outer periphery of the torus at an angle 45 degrees downward from the horizontal center line of the torus. This device should prevent any nozzle plugging.

If one or more do become plugged, an overall slight decrease in core spray flow will result due to the decrease in flow area, depending on the number of blocked nozzles. If it is arbitrarily assumed that five of 112 nozzles are plugged, the total spray flow decreases by only 1.5%. The flow per nozzle will increase by 3% due to the pump head characteristic which would increase by about 11 feet of water. Except for the local bundle under the affected nozzle at the periphery of the core, the rest of the core will have a distribution factor and flow rate identical to that with unplugged operation.

Based on the results of a test where a group of nozzles representing 20% of the total nozzles were plugged, it is estimated that the flow to the local bundle directly below the nozzle could drop to roughly 60% of distribution factor based on full spray flow. But, even if the local flow to the peripheral bundles is reduced as much as might be implied above, these bundles would still be adequately cooled because of their low power.

Sheared Nozzle

The core spray nozzles are situated such that no part of the nozzle or piping connecting the nozzle to the sparger protrudes out past the sparger ring. This arrangement virtually eliminates a mechanism for shearing off a nozzle.

If, however, it is arbitrarily assumed that one of the 112 core spray nozzles is sheared off, the flow to the remaining 111 nozzles will momentarily drop for a few seconds to approximately 94% of rated operation with 6% of the total flow going to the sheared off nozzle, until the system achieves the new equilibrium operating point. The total flow will then increase due to the increase in the spray ring orifice flow area which results in less overall system resistance. Thus, the total flow would run out on the pump characteristic curve from 3400 gpm to 3500 gpm or increase by 3%. This will result in a net overall flow decrease of only 3% of rated through the unaffected nozzles. A pump head pressure decrease of approximately six feet of water occurs due to pump runout.

The minimum core spray distribution factor based on total flow rate to the sparger will decrease to approximately 97% of the one experienced during normal operation.

The effect of the above perturbation on the absolute flow into the fuel bundles is considered negligible. The reasons are that the flow distribution factor of 0.4 is the minimum design target. Also, the design minimum flow of 0.05 gpm per rod being used to size the Core Spray System is conservative in that it is not the true minimum but, rather, the minimum flow which was tested at the time the system was sized.

From a local viewpoint, the two or three bundles directly below the sheared nozzle at the periphery of the core would be the most affected. As discussed above, the local effect of a plugged nozzle could be a reduction in flow or as much as 40% to the affected bundles. This same local flow reduction could be expected to occur also in the event of a sheared nozzle. However, the peripheral bundles have a power level of about two thirds of the average in the core. At such a low power level, a flow per rod of 0.03 gpm is more than adequate to maintain cooling. This implies that even a more severe flow reduction than that imposed by the plugged nozzle could be tolerated in the outer periphery of the core without affecting core spray performance. Neither a sheared nozzle nor a few alternately plugged nozzles would adversely affect the ability of the Core Spray System to prevent clad melting anywhere in the core.

Bowing of Fuel Channel Walls

The amount of spray entering a given fuel bundle is a function of the cross sectional area at the plane where the channel intercepts the spray. Channel wall bowing would affect the spray distribution only if it resulted in a net change in cross sectional area at the point of spray interception.

No deformation of the channel, either bowing in or out is expected at the top of the channel. This portion of the channel, from the upper tie plate to the top of the channel, never experiences any significant pressure differences between the inside and outside of the channel either in normal operation or during an accident since there are no flow resistances present.

If it is arbitrarily assumed that the top of the channel has bowed for some hypothetical reason, the effect on spray distribution would be very slight. The nominal cross sectional area of a channel is approximately 27 in^2. If a 50 mil uniform bow on all sides of the channel is conservatively assumed, the area would vary less than ±2%.

Furthermore, water that does not fall into the channel will fall on the outer wall of the channel. This will effectively cool the channel wall by providing a heat sink.

Outward or inward deformations in the region of the active fuel should not affect the cooling because a water film should still form on the channel walls. The ability of the channel wall to receive heat does not depend on clearances since the energy transfer is dominated by radiation.

Simultaneous Operation of Both Spray Rings

Simultaneous operation of both spray rings should result in delivery of twice as much water to each fuel channel as in operation with a single spray ring. The effect of both sprays operating simultaneously has been approximated from the results of selected single sparger tests. The sparger arrangement used in these tests is different than that utilized at Oyster Creek.

A series of tests were run to determine the spray distribution factor for various spray angles. The test was first run with 60 nozzles to determine the distribution. Then every third nozzle was blocked and the test was run using only the 40 remaining nozzles.

The converse test was done with flow from every third nozzle. The flow per nozzle was held constant for these tests. Superimposing the distribution of both tests yielded approximately the same distribution as recorded for the test using all the nozzles.

It was concluded from these results that the simultaneous operation of both core spray spargers will yield approximately the same distribution as the single sparger operation. It should be noted that since the flow rate for two spargers is about double that of one, twice the design flow rate should reach each fuel assembly; therefore the minimum distribution factor could drop is as low as 0.2 and adequately cool the core. There is no evidence to support any deterioration in the distribution.

6.3.3.2 System Performance

6.3.3.2.1 Emergency Core Cooling Effectiveness at 1860 MWt

Introduction

Each of the two 100 percent Core Spray System loops has been designed to protect the core against clad melting for a large spectrum of Loss-of-Coolant Accident conditions. The ADS is provided to depressurize the reactor vessel for the very small breaks below the lower limit of protection of the unassisted Core Spray System. Therefore, either of the Core Spray System loops in conjunction with the ADS provide adequate reactor core cooling and prevent clad melting for the entire spectrum of possible liquid or steam line break sizes. This protection is provided without dependence on normally available station auxiliary power.

Additional emergency core cooling capability is provided by the Condensate and Feedwater System in the more likely event that station auxiliary power is available.

Figure 6.3-6 shows the capabilities of the various emergency core cooling systems to protect the core against clad melting. A solid bar indicates the spectrum of break sizes for which a given system can protect the core without assistance from any other systems. A dashed bar indicates the range of capability of a given system in conjunction with another system.

Large Breaks (Core Spray System Alone)

The largest possible line break size is the double ended recirculation line break resulting in a flow area of 6.22 ft^2. A break of this size is considered very unlikely and the analysis of this hypothetical accident is performed primarily because the results of the vessel blowdown analysis and core heatup form the basis for design not only of the Core Spray System flowrate, but also calculations of containment response, metal-water capability, heat exchanger sizing and other parameters.

It was assumed that the reactor is operating at 1860 MW thermal* when a break in the primary system equal in area to twice the flow area of a recirculation line (6.22 ft^2) occurs instantaneously. The critical flow rate from this break was calculated by the methods presented by Moody (Reference 2).

The thermal response of the core was calculated by a digital computer program. Radiation heat transfer from fuel rod to fuel rod and to the channel box was considered by the program. No heat losses from the overall reactor core itself were included. The metal-water reaction was defined in the computer program by the expression of Baker (Reference 3), which defines the rate of reaction as a function of both local metal temperature and the current extent of the reaction.

The predicted maximum temperature reached in the entire core was approximately 1800°F. This temperature is reached by only a very small part of the cladding. In fact, ninety percent of the clad volume never exceeded 1500°F.

The extent of metal-water reaction throughout this transient was calculated continuously. Because of the moderate cladding temperature, the extent of reaction was only 0.08 percent of the cladding and channel material.

Calculations show that the percent of rods which perforate for the design break is about 20 per cent.

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* For the current cycle performance evaluations refer to Subsection 6.3.3.3.

Small Breaks

For small breaks below the lower limit of the unassisted Core Spray System (0.2 ft^2) the ADS or the Feedwater System (by means of inventory makeup) depressurize the reactor to allow the Core Spray System to provide adequate cooling.

The system performance curves over the small break size spectrum show that in all cases the no clad melting criterion is easily satisfied by a single Core Spray System loop and only three of the five EMRVs.

The design basis heatup calculations consider the core to be insulated except for thermal redistribution among the rods after the liquid inventory level falls below the core midplane. This is a highly conservative assumption for several reasons. Two of the more important reasons are that significant level swelling and steam cooling will exist even after the midplane is uncovered by the fictitious solid water or liquid inventory level.

In calculating the time when the core is half uncovered, the inventory model treats the core spray flow simply as a mass input into the model as a function of time which depends on the vessel pressure and core spray pump head at any given time.

No credit is taken in the core heatup code for spray cooling of the core until the spray reaches rated flow. Thus, no credit is taken for the distribution across the core either. This aspect is important only when the core is less than half covered, however.

The radiation heat transfer coefficients and convection within the fuel bundle were determined using grey body form factors and Zircaloy-4 emissivities.

The heat transfer coefficients used in the analysis for breaks less than 0.25 ft^2 were based on the Jens Lottes values until the core was half uncovered and then the core was assumed to be insulated except for radiation heat transfer between the fuel rods and the channel only.

6.3.3.2.2 Design Analyses for 1930 MW thermal

The analyses of Core Spray System performance during LOCAs in Subsection 6.3.3.2.1 were based on an empirical representation of the convective heat transfer coefficient for spray cooling developed by E. Janssen from tests conducted on 36 rod simulated reactor fuel bundles. Subsequent high temperature spray cooling tests conducted under the FLECHT program (Reference 5), utilizing different initial conditions and peaking arrangements, underscored the need for a more fundamental approach to the prediction of the heat transfer coefficients. A revised spray cooling correlation (Reference 6) was subsequently derived and was applied to the LOCA analysis for Oyster Creek at the full design rating of 1930 MW thermal.

These results were later superseded, and the ECCS was re-evaluated in accordance with the Atomic Energy Commission Interim Policy Statement published on June 19, 1971. For the current cycle performance evaluations refer to Subsection 6.3.3.3.

6.3.3.3 Current Performance Evaluations

Current performance evaluations for the OCNGS ECCS are contained in Section 15.6.5.

6.3.3.4 Other Factors Affecting Performance

6.3.3.4.1 Structural Requirements

Core Spray Sparger and Piping Internal to the Vessel

The core spray sparger and the piping internal to the vessel are capable of sustaining the primary stresses resulting from the accident loads in combination with the maximum earthquake loading and still remain within ASME Section III stress allowable limits for Class A vessels.

Piping External to the Vessel

The pipe external to the vessel meets all the stress requirements of the ASA B31-1 (1955) Piping Code for the maximum operating loads in combination with the design earthquake. Maximum operating loads include design pressure, thermal expansion, and dead weights. For the maximum operating loads in combination with the maximum earthquake deflections and deformations are limited such that the performance or capability of the system is not affected.

6.3.3.4.2 Water Injection into a Hot Core Spray Sparger Ring

Water for core spray has to flow into a hot sparger ring inside the core shroud before it can enter the core. Steam can be generated as the water contacts the hot surfaces and this steam can delay, or even intermittently interrupt, the flow of water. Time delays and pulsations have been reported (References 15 & 16).

Theoretical considerations lead to the conclusion that the degree of pressure surging inside the ring will vary inversely as a function of the openings in the sparger and in some proportion to the surface area of the hot pipe. Thus, by designing the holes in the sparger sufficiently large to allow

The steam-water mixture at the maximum rate at which it can form to leave the sparger ring at ring pressures substantially less than the pump head, flow perturbations can be reduced to tolerable magnitudes or completely suppressed. This is the case for the OCNGS.

6.3.3.5 Operability Requirements

Operability requirements for the ICS, ADS and Core Spray System, and their specific components are delineated in the Technical Specifications.

6.3.4 Tests and Inspections

6.3.4.1 ECCS Performance Tests

The preoperational test program for OCNGS is described in Section 14.2. Performance tests are conducted prior to startup after each refueling outage.

6.3.4.2 Reliability Tests and Inspections

Periodic surveillance, inspection and testing programs are performed in accordance with Technical Specification requirements and as prescribed in the latest version of the Oyster Creek Inservice Inspection Program.

6.3.5 Instrumentation Requirements

6.3.5.1 Methods of Actuation

Instrumentation provisions for the various methods of actuation of the ECCS are discussed in Section 7.3.

6.3.5.2 Instrumentation and Controls

Instrumentation and Controls for the ECCS are discussed in the system descriptions in Subsection 6.3.2.2.

6.3.6 References

(1) "Electrical Transmission and Distribution Reference Book," Westinghouse Electric Corporation, 1950, Chapter 6 by C.P. Wagner.

(2) Moody, F.J., "Maximum Flow Rate of a Single Component, Two Phase Mixture," Journal of Heat Transfer, Trans. ASME, Series C, Vol. 87, p. 134.

(3) Baker, Louis, Jr., and Just, Louis C., "Studies of Metal-Water Reactions at High Temperatures III. Experimental and Theoretical Studies of the Zirconium-Water Reactions," Argonne National Laboratory.

(4) Moody, F.J., "Liquid/Vapor Action in a Vessel During Blowdown." APED 5177, General Electric Co.

(5) Duncan, J.D. and Leonard, J.E., "Heat Transfer in a Simulated BWR Fuel Bundle Cooled by Spray Under Loss of Coolant Conditions," GEAP 13086.

(6) Appendix G of Third Addendum to Technical Supplement to Petition to Increase Power Level, Nine Mile Point Nuclear Station, Docket 50-200, December 1970.

(7) "Thermal Response and Cladding Performance of Zircaloy Clad Simulated Fuel Bundles," GEAP 13174, page 61, Table 4 Rod 24.

(8) "Loss-of-Coolant Accident and Emergency Core Cooling Models for General Electric Boiling Water Reactors," NEDO 10329, April 1971, Figure D-34.

(9) Ibid, Section 2.4.2

(10) Ibid, Section D.5.4

(11) "Loss-of-Coolant Accident and Emergency Core Cooling Models for General Electric Boiling Water Reactors," NEDO 10329, April 1971, Sections 2.2 and 2.4.

(12) Ibid.

(13) "Recommended Property and Reaction Kinetics Data for Use in Evaluations of Light Water Cooled Reactor Loss of Coolant Accident Involving Zircaloy-4 or 304 SS Clad," GEMP-482 (January 1967).

(14) "The Emittance of Stainless Steel," OTS PB 171630, MDIC Memorandum III, June 12, 1961.

(15) "Water Injection Test Simulating Moderator Flooding of BONUS Superheater Assembly," GNEC 156, 1961.

(16) Report ANL 6084.

(17) Safety Evaluation by the Office of NRR Supporting Exemption from 10CFR50, Appendix R and the Post Fire Safe Shutdown Capability, dated March 24, 1986.

(18) Technical Specification Amendment 112, dated October 31, 1986.

(19) Safety Evaluation by the Office of NRR; Systematic Evaluation Program (SEP) Topics II-3B, "Flooding Potential and Protection Requirements and III-4A, "Tornado Missiles".

(20) "Oyster Creek SAFER CORECOOL GESTR/LOCA Loss of Coolant Accident (LOCA) Analysis", General Electric NEDC-31462P.

(21) Reference intentionally deleted.

(22) GENE-E21-00143, “ECCS Suction Strainer Hydraulic Sizing Report”

(23) GPUN Safety Evaluation No. 323682-001, Rev. 0.

(24) NRC Letter to GPUN, dated February 5, 1993 (Safety Evaluation Report).

(25) SE-328312-003, "GL 89-10 Motor Operated Valve Modification”

(26) SE-315403-031, "Justification for Revised IC High Flow Trip Delay"

(27) GPUN Calculation C-1302-241-E610-074, Rev. 3, “Containment Spray Pump and Core Spray Pump NPSH Assessment”

(28) GE-NE-13000411-00-03P, “Core Spray Flow Requirements vs. Time Analysis for Oyster Creek Nuclear Generating Station”

(29) GE-NE-B13-02080-00-15, “Core Spray Sparger and Piping Flow Evaluation Plan – Oyster Creek Nuclear Generating Station,” October 2000

TABLE 6.3-1
(Sheet 1 of 4)

ISOLATION CONDENSER SYSTEM COMPONENTS

Steam and Condensate Lines	
Design Conditions	1250 psig, 575°F
Maximum Mass Flow Rate	330,000 lb/hr per loop
Material	
Inside Drywell	Stainless Steel
Outside Drywell	Stainless Steel
Inside Penetration Guard Pipe including First Elbow Inside DW	Stainless Steel
Steam Line Sizes	
Reactor vessel to isolation valve	10 in.
Isolation valves to condensers	16 in.
Tube-bundle inlets	12 in.
Condensate Return Line Sizes	
Tube-bundle outlets	8 in.
Condensers to recirculation lines	10 in.
Isolation Valves, Steam Supply and Condensate Return	
Size and Type – 6 outside drywell	10 in. gate valve with motor operator
– 2 inside drywell	10 in. gate valve with motor operator
Design Conditions	1250 psig, 575°F
Material (Body) – 6 outside drywell	stainless steel
– 2 inside drywell	stainless steel
Motor Operators (Condensate Return)	AC – Inside drywell or closest to reactor DC – Outside drywell or closest to condenser
Motor Operators (Steam Supply)	DC and AC both outside drywell
Operating Time	(See Table on Following Page)

TABLE 6.3-1
(Sheet 2 of 4)

ISOLATION CONDENSER SYSTEM COMPONENTS

VALVE ID	MAXIMUM STROKE TIME (In Seconds)*** UNDER LINE BREAK CONDITIONS	
	ICS In Standby Close+	ICS In Service Close
V-14-30	<89	42
V-14-31	<89	39.5
V-14-32	<89	42
V-14-33	<89	36.4
V-14-34	N/A	41.8
V-14-35	N/A	50.5
V-14-36	N/A	42
V-14-37	N/A	42

N/A With ICS In Standby the Condensate Line is Isolated

*** SE-328312-003, Rev. 2
ECR OC 07-00647-000

+ < 89 Seconds, for the steam valves isolation in standby is based on 120 second isolation. 120 - (2 + 29) (from Note 1) = 89 seconds.

NOTE 1:

2 seconds - To account for signal transmission from the process fluid to the transmitter
29 seconds - Maximum time delay of the 300% Hi Flow Relay

TABLE 6.3-1
(Sheet 3 of 4)

ISOLATION CONDENSER SYSTEM COMPONENT DESIGN PARAMETERS

Isolation Valves - Steam Line Vent (V-14-1)

Size and Type	3/4 in., pneumatically-operated, dc-solenoid-valve actuated, spring to close, air to open
Design Conditions	1250 psig, 575°F
Material	ASA B31.1, Type 304 or 316 stainless steel up-to-the-vent valves

Isolation Condensers (CD-14-1)

Number and Type	Two horizontal shell and U-tube heat exchangers with two tube bundles in each shell
Shell Side	Fluid - demineralized water; steam vent to atmosphere
Design pressure	15 psig internal, 1 psig external
Design temperature	300°F
Construction	ASME Boiler and Pressure Vessel Code, Section VIII; shell and heads, ASTM A 212, Grade B, firebox quality

Tube Side Fluid - reactor water

Design pressure	1250 psig
Design temperature	575°F
Construction	ASME Boiler and Pressure Vessel Code Section III, Class 1A

TABLE 6.3-1
(Sheet 4 of 4)

ISOLATION CONDENSER SYSTEM COMPONENTS

Isolation Condensers

Material	Tubes ASTM A 249, Type 304 stainless steel, stress relieved after bending; all remaining parts of tube bundle, Type 304 stainless steel (These are the original materials. See note below)
Tube Bundle	36 tubes, 2-in OD and 0.095-in minimum wall
Approximate Shell Dimensions	12 ft OD, 46 feet long and 3/8 in minimum wall
Design Heat Removal Capacity (Two Condensers)	410 x 10^6 Btu/hr at 1000 psig and 546°F
Maximum Mass Flow Rate	165,000 lb/hr per tube bundle
Maximum Tube Side Pressure Drop	2.5 psi
Shell-Side Steaming Rate	Heat transfer rate divided by 960 Btu/1b
Pressure Drop from Inside Shell to Atmosphere	5 psi
Expected Lifetime Operations	500 cycles
Design Life	1500 cycles
Normal Water Level Line	12 inches above shell center
Normal Water Capacity	22,730 gal/shell
Normal Water Capacity above Bundle	11,060 gal/shell
Initial Supply of Water in Shell	Approximately 45 minutes for one condenser or 1 hour 40 minutes for both condensers

Note: Both tube bundles in the ‘B’ Isolation Condenser were replaced in November 1998 (17R refueling outage). Both tube bundles in the ‘A’ Isolation Condenser were replaced in September 2000 (18R refueling outage). The new tube bundles are equal to the original tube bundles in design and configuration. The replacement tube bundles were fabricated from type 316 stainless steel materials with low carbon content (.03% max.) which is more resistant to stress corrosion cracking.

TABLE 6.3-2
(Sheet 1 of 1)

ELECTROMATIC RELIEF VALVES

Number	5
Overpressure Relief Set Points	Refer to Technical Specifications
Automatic Depressurization Set Points	Refer to Technical Specifications
Capacity Each Valve	602,900 lb/hr at 1250 psig
Outlet Pressure at Full Flow	250 psig
Design Conditions	1250 psig, 575°F
Inlet Connection	6 inches, flanged
Outlet Connection	8 inches, flanged

TABLE 6.3-3
(Sheet 1 of 1)

CORE SPRAY SYSTEM SUCTION STRAINERS

Strainer Locations	Suppression chamber penetrations X-68A, X-68B, X-69 at 300 degrees, 182 degrees and 20 minutes, and 60 degrees, respectively. Each at outer periphery of torus and 45° down from horizontal center line at El. (-)13 ft.
Penetration Nozzle Size	16 in diameter
Strainer	GE Stacked Disc approximately 63" long and 54" diameter (maximum envelope dimensions)

The blowdown and transport of insulation debris to the torus is accounted for in the suction strainer design. Direct blowdown to the torus from pipe breaks within the drywell will be impeded by baffles at the inlets to the torus downcomers, followed by low bulk fluid velocity transport to the suction strainers.

TABLE 6.3-4
(Sheet 1 of 3)

CORE SPRAY SYSTEM VALVES

Suction Valves	
Size and Type	12 inch gate
Operator	Electric motor, 460 volt, 3 phase
Stem Seal	Double packing with plugged lantern ring bleedoff
Back Seat	Provided
Open and Close Limitorque	Provided
Material	Carbon steel body, standard trim
Design Condition	150 psig, 200°F, Code ASA B31.1
Test Valves	
Size and Type	6 inch globe
Operator	Electric motor, 460 volt, 3 phase
Closing Time	20 seconds maximum*
Stem Seal	Double packing with plugged lantern ring bleedoff
Material	Carbon-steel body, standard trim
Design Conditions	300 psig, 200 degrees F, Code ASA B31.1

* Maximum time required to meet system flow requirements.

TABLE 6.3-4
(Sheet 2 of 3)

CORE SPRAY SYSTEM VALVES

Pump Discharge Valves	
Size and Type	8 inch gate
Operator	Electric motor, 460 volt, 3 phase
Closing and Opening Time	24 seconds maximum
Stem Seal	Double packing with plugged lantern ring bleedoff
Material	Stainless steel body and trim with hard-faced disc and wedges
Design Conditions	1250 psig, 575°F, Code ASA B31.1
Outside Isolation Valves	
Size and Type	8 inch gate
Operator	Electric motor, 460 volt, 3 phase
Opening Time	22.4 seconds maximum
Stem Seal	Double packing with plugged lantern ring bleedoff
Material	Stainless steel body and trim with hard faced discs and wedges
Design Conditions	1250 psig, 575°F, Code ASA B31.1
Inside Isolation Check Valves (Testable)	
Size	8 inch check
Type	Testable, tilting disc check valve
Design Condition	1250 psig, 575°F, Code ASA B31.1

TABLE 6.3-4
(Sheet 3 of 3)

CORE SPRAY SYSTEM VALVES

Inside Isolation Check Valves (Testable)

Material	
Body	Cast stainless steel, A351-CF8M
Disc	Cast stainless steel, A351-CF8M
Shaft	Type 410 stainless steel
Seats	Stellite 6
Seat Leakage	Not to exceed 2 cc/hr per inch of seat diameter at pressure differential greater than 62 psi
Valve Disc	Free swinging, self-aligning with inclined seat, gravity closing
Test Actuator	Air operated, solenoid-actuated, air open, spring close; capable of opening valve, but not capable of closing valve or holding it open against reverse flow
Actuator Stem Seal	Packed
Position Indication	Magnetic type

Manual Isolation Valves

Size and Type	8 inch gate
Design Condition	1250 psig, 575°F
Material	Stainless Steel
Operator	Manual, locked open

TABLE 6.3-5
(Sheet 1 of 1)

CORE SPRAY MAIN PUMPS AND BOOSTER PUMPS

	Primary	Booster
Type	Horizontal, electric motor driven, flexible coupled, single stage centrifugal pumps, with mechanical seals	
Rated Flow	3700 gpm*	3700 gpm*
Runout Flow	4700 gpm*	4700 gpm*
Shutoff TDH	450 ft	290 ft
Rated TDH	405 ft	250 ft
Runout TDH	355 ft	210 ft
Required NPSH, Rated	15 ft	--
Required NPSH, Runout	18.5 ft	--
Brake Horsepower, Rated	462	285
Brake Horsepower, Runout	543	308
Design Conditions	400 psig, 650°F, casing is according to ASME Code, Section VIII	
Materials	Cast steel casing	
Mechanical Seal	Supplied with pump discharge water through cyclone separators to remove dirt and grit	
Motor	500 hp	300 hp

* Includes approximately 100 gpm continuous recirculation flow to suppression pool through a throttling valve.

TABLE 6.3-6
(Sheet 1 of 1)

CORE SPRAY SPARGERS AND SPRAY NOZZLES

Parameters	Description
Reactor Vessel Penetration	N6A and N6B located on building elevation 73 ft 9 in at 60 degrees and 240 degrees, respectively
Pipe Size in Penetration	5 inches in thermal sleeve
Sparger Pipe Size	3 1/2 in. Schedule 40
Number of Spray Nozzles 112 Total	56 per sparger (each loop),
Number of Spray Elbows 112 Total	56 per sparger (each loop),
Discharge Spacing around Sparger	About 5 inches apart
Discharge Distance above Fuel Channels	6 1/2 inches for one loop and 9 inches for the other, one sparger above the other
Material	Cast stainless steel, Type 316

6.4 HABITABILITY SYSTEMS

6.4.1 Design Basis

The design bases of the Control Room Habitability System are as follows:

a. protect Control Room operators against the effects of an accidental release of toxic or radioactive gases; and

b. provide a habitable environment so that the plant can be safely operated or shutdown under design basis accident conditions.

6.4.2 System Design

6.4.2.1 Definition of Control Room Envelope

The Control Room envelope, shown on Drawing 3E-151-02-006, consists of the Control Room panel area, the Shift Supervisor's office, toilet room, kitchen, and Cable Spreading Room. Scott Air Packs are available in the Control Room envelope. Additional Scott Air Packs with spare bottles are stored in the Monitor and Change Area. Water is supplied by the Well and Domestic Water System. No emergency food is available. The Control Room HVAC System has been designed to maintain a suitable environment for equipment and personnel during normal and emergency plant conditions.

6.4.2.2 Ventilation System Design

The Control Room HVAC System consists of two independent systems: HVAC System A and HVAC System B. HVAC System B has been designed as the primary or LEAD system; HVAC System A is designed as the backup or LAG system. Changeover from LEAD to LAG will be through operator action. Administrative controls will prevent simultaneous operation of both systems. The total system consisting of Systems A and B is designed to mitigate the effects of single active component failures (Reference 6).

Each system has four manual operating modes designated as normal, purge, partial and full recirculation. Duct mounted radiation monitors, HEPA filters, or charcoal adsorbers are not part of either system's design. Both systems share a common supply/return duct inside the Control Room HVAC boundary.

System A consists of a commercial grade air conditioning unit with a supply fan, steam coils for heating, air operated dampers, and a three stage refrigeration unit for cooling. The auxiliary boiler supplies steam for heating, and the Turbine Building Closed Cooling Water System provides cooling water for the air conditioning unit condenser. Diesel Generator 1 (DG-1) provides backup power to the supply fan via unit substation (USS) 1A2.

System B utilizes a commercial grade rooftop air conditioning unit with a supply fan, self contained refrigeration unit with air cooled condenser, a duct mounted electrical heater, and motor operated dampers. DG-2 provides backup power to the supply fan via (USS) 1B3.

Major components of System A and B are listed in Table 6.4-1. A Control Room HVAC flow diagram is presented in Drawing BR 2010.

6.4.2.3 Leak Tightness

The Control Room HVAC provides sufficient outside air for pressurization to minimize inleakage into the envelope. During a test of the full recirculation mode of operation, the infiltration rate was measured as 1750 cfm. For a Control Room free volume of 27,500 cu. ft., this infiltration rate represents an air exchange rate of 3.8 per hour. In accordance with Table C-2 of Regulatory Guide 1.78, the Oyster Creek Control Room is a Type "C", and the need for periodic infiltration testing is not required.

6.4.2.4 Interaction with Other Zones

Two potential pipe breaks were identified in an area adjacent to the Control Room envelope. The Turbine Building Closed Cooling Water System and the auxiliary boiler steam lines for System A are located in the Upper Cable Spreading Room. The System B refrigeration unit with outside air exhaust and recirculation air dampers, are located on the Upper Cable Spreading Room roof away from a potential pipe break in these systems. Conduits which were installed in the Upper Cable Spreading Room are routed to minimize the effects from these postulated breaks.

6.4.2.5 Shielding Design

An analysis of Control Room shielding has been performed. As a result of this study, one significant source contributing to an elevated radiation dose rate in the Control Room has been identified. This source is the core spray booster pump suction and discharge piping located at El. 51'-3" in the Reactor Building. A shadow shield has been installed in the Reactor Building to reduce the Control Room dose to below the 10CFR50, Appendix A, General Design Criterion 19 limit.

6.4.3 System Operational Procedures

The system is normally operated to maintain control room air temperature at 75°F ± 5°F, and the lower cable Spreading Room temperature at 84 ± 5°F. A switch allows the Control Room operator to override the thermostatic control of the exhaust air, return air and outside air dampers in order to allow recirculation mode operation with or without the minimum outside air damper being closed. This capability is provided to reduce the potential for intrusion of toxic and/or radioactive gases entering into the Control Room atmosphere by maintaining the Control Room at positive pressure greater than or equal to .125 inches W.G.

An emergency mode is provided for operation with 100 percent outside air to prevent the recirculation of smoke in the Control Room and to clear the area of smoke and fumes. As required by the Oyster Creek Fire Hazards Analysis, the system was modified to prevent smoke from other areas from entering the Control Room.

During radiological releases with offsite power available, the HVAC System is operated in the partial recirculation mode with minimum outside air for pressurization purposes.

During radiological releases associated with loss of offsite power, the system is provided with "fan only" mode of operation (if outside air temperature permits). The "fan only" mode of operation is delayed until the Control Room operator has ensured that emergency diesel generators can supply the required load. If outside temperature does not permit, the HVAC system remains in the shutdown condition until power is available.

The system is also capable of manually being placed in the full recirculation mode of operation in case of receipt of a chlorine alarm so the toxic limit of 0.045 g/m^3 is not achieved in the Control Room.

6.4.4 Design Evaluation

6.4.4.1 Radiological Protection

Computer code DRAGON4 was used along with the HVAC operating data, X/Qs, leak rates, and Regulatory Guide 1.3 source terms to determine the 30 day integrated gamma and beta doses in the Control Room due to the airborne source within the Control Room HVAC pressure envelope (Reference 3). The thyroid dose was not addressed pending NRC review of the iodine source term.

The gamma dose contribution from the radioactive cloud surrounding the Control Room was calculated using the isotopic concentrations versus time outside the Control Room and the appropriate dose rate conversion factors for an infinite cloud source model. One foot thick concrete shielding was assumed.

The gamma dose contribution from post-LOCA piping sources and airborne radioactivity in the reactor building was taken from the September 1984 report for OCNGS on NUREG-0737, Item II.B.2.

The three airborne fission product release paths considered were:

1. MSIV Bypass Leakage (SRP 6.2.3 - Branch Technical Position CSB 6-3).
2. Containment Leakage (Appendix A to SRP 15.6.5).
3. ESF Leakage (Appendix B to SRP 15.6.5).

For an infiltration rate of 2000 cfm in the partial recirculation mode, the calculated doses were 3.07 Rem gamma and 28.2 Rem beta.

Since the System (A or B) trips with a LOOP concurrent with a LOCA, a review of the radiological analysis was performed to account for partial restoration (supply fan only) of a system with 100% outside air to limit the Control Room maximum temperature (104°F) when ambient temperatures (82°F) permit this mode of operation.

The revised calculation assumed the most conservative operation (supply fan only for 30 days). Since offsite power and full System (A or B) capability can be restored within two hours, this calculation was conservative. For the supply of 100% outside air to the Control Room envelope, the calculated doses are 29.1 Rem beta and 3.14 Rem gamma for the assumed 30 days.

The radiological analysis further assumed one MSIV failed in the open position, and the three remaining MSIV's are leaking at a constant 11.9 scfh. However, through testing, the MSIV bypass leakage is a function of the air accumulator pressure and the containment pressure post-LOCA.

The radiological analysis assumed the containment pressure would decrease to 0 psig in 10 days post-LOCA. This assumption equated to 1000 standard cubic feet (SCF) of MSIV bypass leakage for the dose assessment.

The MSIV leakage was reassessed considering MSIV leakage as a function of accumulator and containment pressures. MSIV bypass leakage of 243 SCF was calculated for the first day post-LOCA. This calculation assumed that the Drywell pressure decays to 1 psig in about 2 1/2 hours then remains constant until 24 hours after the LOCA. Since the containment response (Figure 6.2-3) shows 0 psig within 6 1/2 hours, this pressure profile was considered conservative for the assessment.

The MSIV bypass leakage calculated by the radiological analysis (1000 SCF) exceeds the bypass leakage calculated by the MSIV leakage assessment (243 SCF). Since the MSIV bypass leakage is the major contributor to the Control Room dose, the assumptions used for the radiological analysis are conservative when compared to the expected MSIV behavior (12R tests) and containment pressure response (FSAR Fig. 6.2-3) during a LOCA. Further, the assumptions used for the main steam line volume (exclusion of main steam line header and piping volume up to the turbine stop valve) and for calculating the beta skin dose (inclusion of iodine daughter products) are conservative. Therefore, the radiological analysis (Reference 3) is still bounding.

6.4.4.2 Toxic Gas Protection

6.4.4.2.1 Onsite Releases

During 1987, the one ton liquid chlorine cylinders were replaced with a sodium hypochlorite system. Sodium hypochlorite is a relatively stable chemical at ambient temperatures and atmospheric conditions. Two to three 150 lb liquid chlorine cylinders were retained to minimize biofouling in the New Radwaste Service Water System. Limited amounts of other chemicals stored onsite are listed in Table 6.4-2.

A chlorine transport analysis was performed for an instantaneous and a continuous release (3/8 inch line break) for a 150 lb cylinder which is stored approximately 380 feet from the Control Room intake. A computer program Vapor was used to determine the chlorine time history concentrations. The Vapor program is based upon the methodology described in NUREG-0570 and the assumptions presented in Regulatory Guide 1.78. The analysis took no credit for mixing of the chlorine plume due to the effects of a building wake, and the analysis also assumed the wind direction is such that the centerline of the chlorine plume at ground level blows directly toward the Control Room air intake (approximately 41 feet above plant grade).

For the instantaneous and continuous releases under various meteorological conditions, a toxic limit ($0.045 g/m^3$) was only achieved in the Control Room envelope when the Control Room received air from outside sources at rates greater than 13,000 cfm (instantaneous release) and 1750 cfm (continuous release until cylinder is depleted) respectively. These Control Room outside air rates were assumed constant for the duration of the accidental chlorine releases. The minimum times to achieve a toxic limit were calculated as 320 seconds (instantaneous release) and 372 seconds (continuous release). Allowing 5 seconds for the chlorine detector loop response time, the minimum operator response times to restrict the Control Room outside air source rate (switch to full recirculation) were assumed as 315 seconds (instantaneous release) and 367 seconds (continuous release).

The 12R Control Room HVAC tests have demonstrated an outside air source rate less than 1750 cfm for the full recirculation mode of operation (FN-826-009 off). Since the minimum times assumed for operator response are greater than the minimum operator response time (120 seconds) required by

Regulatory Guide 1.78, the Control Room operator has sufficient time from the receipt of chlorine alarm to place the HVAC system (A or B) into full recirculation.

The physical characteristics of the site provide additional restrictions to achieving a toxic limit in the Control Room envelope. The enclosure which houses the cylinder is located at ground level, northwest of the Control Room air intake. The Control Room air intakes are located on the roof of the office building at elevations 64' and 73'. Between the cylinder enclosure and the control room air intakes there are tanks and buildings with heights up to 100 ft (Turbine Generator Building). A chlorine plume driven by a wind from NW through W of the cylinder enclosure does not have a straight line path to the control room air intake. The presence of these local obstructions produces a large scale turbulence that alters the wind path and reduces the probability of the plume reaching the Control Room air intake, which is located at the opposite side of the turbine building (east of the Turbine Building). Likewise, a chlorine plume driven by a wind from N, NE, E, SW, S, or SE of the cylinder enclosure would have no impact on the Control Room air intake. The cylinder is located northwest of the Control Room air intakes and the wind direction would drive the chlorine plume away from the Control Room outside air intakes. Due to these facts, the need to don respiratory equipment for a chlorine alarm is not warranted.

6.4.4.2.2 <u>Offsite Releases</u>

There are no offsite manufacturing or storage of hazardous chemicals within a five mile radius of the facility. Two potential transportation releases of hazardous chemicals were investigated.

A truck carrying a one ton tank of chlorine either on Route 9 or the Garden State Parkway was postulated in an accident and the chlorine tank ruptured. The analysis was performed with the "VAPOR" computer code for the accident 423 meters from the Control Room outside air intake with the ventilation system in the normal ventilation mode.

The chlorine concentrations reached 0.2 g/m^3 (66.7 ppm) with the ventilation system in the normal operation mode. An evaluation was performed on the probability of the truck accident occurring simultaneously with the meteorological conditions which would produce the 0.2 g/m^3 chlorine concentration in the Control Room. This considered the probability of a severe truck accident ($1.29X10^{-8}$ accident per truck mile, NUREG/CR-2650), the frequency of chlorine shipments near the site (20/year), the length of road traveled within 5 miles of the site on both Route 9 (11 miles) and the Garden State Parkway (10 miles), and the frequency of winds that could transport a release toward the site. Using meteorological data, the highest probability of an operator incapacitation due to a truck accident near the site is $1.7X10^{-7}$ incapacitation per year. Therefore, the probability of an offsite chlorine release accident affecting the operators is negligible.

Additional procedural actions are described in Section 2.2.3.2 regarding New Jersey State Police commitment to notify the OCNGS Control Room with the characteristics of an incident involving hazardous material in Lacey or Ocean Township and provide further information on any significant changes.

6.4.5 <u>Testing and Inspection</u>

Surveillances to maintain continued integrity of the Control Room Habitability System are provided in the Technical Specifications.

6.4.6 Instrumentation Requirements

Initiation of the Control Room HVAC Systems A and B and selection of the operating modes are by operator action.

6.4.7 References

1) GPUN Ltr, P.B. Fiedler to John A. Zwolinski (NRC), "Control Room Habitability" dated June 4, 1985.

2) GPUN Ltr 5000-89-1767, R.F. Wilson to Director of Nuclear Reactor Regulation, "Control Room Habitability" dated May 16, 1989.

3) GPUN Ltr, R.F. Wilson to John A. Zwolinski (NRC), "Results of Whole Body and Beta Skin Dose Analysis" dated June 17, 1985.

4) GPUN Ltr, R.F. Wilson to John A. Zwolinski (NRC), "Results of Chlorine Transport Analysis", dated August 16, 1985.

5) NRC Ltr, J.N. Donohew, Jr. to P.B. Fiedler, "Safety Evaluation Relating to Control Room Habitability, dated November 14, 1986.

6) NRC Ltr, J.N. Donohew, Jr., "Working Meeting with GPU Nuclear on Oyster Creek Control Room Habitability, dated April 16, 1985.

TABLE 6.4-1
(Sheet 1 of 4)

CONTROL ROOM ENVELOPE HVAC SYSTEM
COMPONENT DESIGN PARAMETERS

I. HVAC SYSTEM A

Air Conditioning Unit (M-826-001A)

Manufacturer and Type	Carrier Air Conditioning Co., 50BA044 Cooling Coil C 826-003A with cooled condensing unit
Unit Rating Capacity (Btu/hr)	379,000 Btu/hr sensible 417,000 Btu/hr total
Cooling Coil Entering Temperature (Drybulb-Wetbulb)	75.3°F DB, 61.7°F WB
Condensing Temperature	115°F
Cooling Water	100 gpm at 100°F, 75 psig
Fan Capacity	14,000 cfm at 2-3/8 in W.G.
Power Requirements	460 volts, 3 phase

Steam Heating Coil C-826-004A

Manufacturer and Type	American-Standard U-return bend steam coil
Coils	Copper, aluminum finned
Airflow	14,000 cfm
Capacity	50,330 Btu/hr
Pressure Drop	0.04 in W.G
Air Temperature	71.8 - 75°F
Steam Consumption	56 pounds per hour at 75 psig

TABLE 6.4-1
(Sheet 2 of 4)

CONTROL ROOM ENVELOPE HVAC SYSTEM
COMPONENT DESIGN PARAMETERS

II. HVAC SYSTEM B

Rooftop A/C Unit (M-826-001B)

Model No.	TRANE, SAHCC6040A00D69D1B10EFRTY
System Type	Constant volume
Power Supply	460 V, 3 pH
Operating weight	8,7000 lb.
Normal weight	7,800 lb.
Compressor	
Number of compressors	2
Full load input, kW	31.3 (each)
Rated load Current	51.7 amp
Locked rotor current	212 amp
Refrigerant type	R22
Refrigerant R22 operating charge, lb.	59
Condenser Fans	
Number of Fans	6
Total air handled, cfm	38,000
Motor Rate HP	1.0 (each)

TABLE 6.4-1
(Sheet 3 of 4)

CONTROL ROOM ENVELOPE HVAC SYSTEM
COMPONENT DESIGN PARAMETERS

II. HVAC System B (continued)

Air-Cooled Condenser	
Starting and operating low ambient temperature, °F	0
Entering air temperature, °F	89
Cooling Coil C-826-003B (Data at 100 percent full load)	
Capacity at listed entering air temperature, MBH	488.9 sensible/590.2 total
Cooling coil entering air temmperature dry/wet bulb, °F	77.68/61
Cooling coil leaving air temperature dry/wet bulb, °F	45.55/45
Supply Fan (FN-826-008B)	
Rated flow, cfm	14,000
Total SP in W.G.	3.5
Drive Belt	
Number of speeds	Single, adjustable
Motor rated hp	20
RPM	909
Motor Manufacturer	Century
Motor Model No.	350298
Remote Electrical Heater (C-826-004B)	
Manufacturer	Brasch Mfg. Co., Inc.
Entering air temperature °F	63

TABLE 6.4-1
(Sheet 4 of 4)

CONTROL ROOM ENVELOPE HVAC SYSTEM
COMPONENT DESIGN PARAMETERS

II. HVAC System B (continued)

Leaving air temperature °F	76.5
Number of Heating Stages	3
Total connected load kW	59.94 kW
Power supply	480 V, 3 phase
Minimum airflow, cfm	5000
Filter (F-826-005B)	
Type	Bag
Efficiency	Minimum 40% (in accordance with ASHRAE 52-76) operable to 1 in. WC final resistance
Filter media	Glass Fiber
Static pressure drop	0.11 WG at 14,000 cfm
Number/Size	8/24 in. x 24 in. x 21 in.

TABLE 6.4-2
(Sheet 1 of 1)

ONSITE CHEMICAL STORAGE

Chemical	Total Amount	Container Size	Closest Distance From CR Inlet
Sulfuric Acid	3,000 gal	3,000 gal	250 ft (L/S Radwaste Bldg.)
Sulfuric Acid	3,000 gal	5,000 gal	350 ft (Pre-Treatment Tank)
Sodium Hypochlorite	13,000 gal	6,500 gal	225 ft
Liquid Chlorine	450 lb	150 lb	380 ft

6.5 FISSION PRODUCT REMOVAL AND CONTROL SYSTEMS

6.5.1 Engineered Safety Feature (ESF) Filter Systems

The Standby Gas Treatment System (SGTS) is a plant ESF atmosphere cleanup system which functions as a barrier between the radiation source and the environs during an emergency condition. This system filters and exhausts the Reactor Building atmosphere and drywell atmosphere to the stack during secondary containment isolation conditions and drywell purging operation.

6.5.1.1 Design Bases

The SGTS is designed to perform the following functions subsequent to Design Basis Accidents:

a. Limit the ground level radioactive releases to the environs within 10CFR100 limits by filtering and releasing gases to the plant stack for high gas dispersion.

b. Minimize leakage of fission products from the Reactor Building atmosphere by maintaining a negative pressure of 0.25 inches of water within the Reactor Building with respect to the outside atmosphere under calm wind conditions.

c. Limit radioactive releases during purging of the drywell and suppression chamber during normal and after emergency operation, after containment spray has reduced the drywell pressure to approximately 1 psig.

6.5.1.2 System Design

6.5.1.2.1 Description

The SGTS consists of two separate filter trains, each having 100% capacity. Either of the two filter trains is considered as an installed spare, with the remaining one capable of the required flow capacity. Each SGTS train is composed of various elements to filter and remove radioactive iodines and particulates that may be present in the Reactor Building atmosphere during and after an accident, and in the drywell and suppression chamber atmosphere.

The system P&ID is shown on Drawings BR 2011 and 3E-822-21-1000. Each filter train consists of the following components (in series order):

Inlet Isolation Valves and Orifice Valves

Prefilter

Electric Heating Coil

Absolute Filter
Charcoal Filter

Absolute Filter

Exhaust Fan

Outlet Isolation Valves

Table 6.5-1 lists design parameters for these components.

An intake valve and a restricting orifice are located upstream of each prefilter.

A 14 inch diameter cross tie connects the two filtering trains of the Standby Gas Treatment System. The cross-tie is located upstream of the fans of each loop and downstream of the last filter bank of each train. A valve is located in the cross-tie. Upon failure of an operating fan, the second fan will start automatically. The cross-tie valve will open automatically, the inlet and outlet valves of the failed fan will automatically close, and the inlet and outlet valves of the second fan will automatically open. Low flow air bleed will continue to flow through the orifice valve and the filter bank of the failed fan.

With this arrangement established, approximately 50 cfm of cooling air is drawn through the shut down train to remove the decay heat and prevent auto ignition of the charcoal, and the remainder of the flow is drawn through the operating train from the Reactor Building.

All of the major components of this system, with the exception of the exhaust fans and outlet valves, are located in the Pipe Tunnel connecting the Turbine Building, Reactor Building, and Old Radwaste Building.

This area is in close proximity to the base of the Stack and Stack Drain Sump 1-12.

Under calm wind conditions, the system is designed to maintain a negative pressure of 0.25 inches of water within the Reactor Building at a flow rate of approximately 2600 cfm during accident conditions.

Running the SGTS for test purposes serves as a check on the integrity of the Secondary Containment. There is no specific supply of ventilation air to the Reactor Building other than building inleakage.

6.5.1.2.2 Components

All of the major components of this system, with the exception of the exhaust fans and outlet valves, are located in the Pipe Tunnel connecting the Turbine Building to Reactor Building to Old Radwaste Building. This area is in close proximity to the base of the Stack.

Exhaust Fans

Exhaust fans EF-1-8 and EF-1-9 in the Standby Gas Treatment System are started either automatically or manually from panel 11R in the Control Room. Each fan delivers a flow rate of approximately 2600 cfm. The exhaust fan motors are connected to an emergency power supply. The designated standby fan will start automatically in case of failure of the first fan. In the event the standby fan cannot start, its failure is annunciated in the Control Room. Shutdown of the fans is by manual control, motor overcurrent or motor thermal overload.

Isolation Valves and Orifice Valves

There is a total of seven valves in the combined SGTS - one inlet, one orificed purge inlet, and one discharge valve for each filter train. One crosstie valve connects the outlet of each train. All of these valves are air operated and designed to fail in the normal operating position on loss of air. During standby, all valves are closed except for the crosstie valve. When the SGTS is operating, the valves are aligned as follows:

Inlet Valves	-	Open
Inlet Orifice Valves	-	Closed*
Outlet Valves	-	Open
Crosstie Valve	-	Closed**

Prefilter

The prefilter is provided to remove dust and large particles to extend the life of the absolute filters.

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* Inlet orifice valve of the failed train will automatically open upon failure of fan and the orifice valve of the standby train will close automatically.

** Cross-tie valve automatically opens upon failure of the operating train.

Electric Heating Coil

The heating coil is provided to control the temperature of the exhaust gases, and thus the humidity, for optimum operation of the charcoal filters; the relative humidity of the air going to the charcoal is maintained below 70% by the heating coil. Without the heating coil and assuming 100% relative humidity, the radioiodine removal efficiency of the charcoal absorbers is reduced to 78%.

Absolute Filters

Two high efficiency absolute filters are installed in each train. The first, located upstream of the charcoal filter, removes fine particles of dust and contaminants from the air stream. The second, located downstream of the charcoal filter, collects any particle breakthrough from the charcoal or radioactive charcoal dust that might be generated during a seismic event.

Differential pressure across the entire filter train is indicated in the local panels, and high differential pressure is annunciated in the Control Room. The absolute filter is constructed by forming a continuous sheet of glass microfiber medium into closely shaped pleats separated by corrugated aluminum inserts. The filter frames are either 16 gauge cadmium plated steel or 14 gauge 304 stainless steel, all with double turn flanges to facilitate stocking. Adhesive sealer is rubber based.

The filter is thoroughly inspected before installation and periodically as required by the Technical Specifications.

Charcoal Filter

The charcoal filter is provided to remove radioiodines from the Reactor Building exhaust air during an accident condition. The radioiodines in the exhaust air are expected to be 90 percent in the "normal" forms of molecular iodine and iodine salts, and 10 percent in the form of methyl iodide (CH_3I), or other organic salts.

6.5.1.2.3 Charcoal Filter Modification

During the 1977 refueling outage, the existing charcoal filters were removed and replaced with new filters which contained new charcoal and test canisters to permit laboratory analysis. Activated charcoal in the test canisters is identical to that in the trays. Inplace Leak Tests were performed by the Mine Safety Appliance Co.

Figure 6.5-2 shows the flow path through the filters and test canisters for an individual tray. The trays are mounted in groups of five and they are all enclosed within the exhaust duct. The exhaust fans are located downstream of the filters; the gas from the filters is directed into a second set of filters (absolute filters), and out to the stack via the exhaust duct. By installing the test canisters on the face of the trays, the gas flowing through them is independent of the gas flowing through the trays. The canisters are subject to the same conditions as the filters (i.e., the air inlet to the canister is the same as the inlet to the filter and the canister discharges to the same ductwork as the filter).

6.5.1.2.4 Operation

Four radiation monitors are provided which initiate isolation of the Reactor Building and operation of the SGTS. Two monitors are located outside of the main exhaust ventilation duct, one is located in the area of the refueling pool and one is located in the reactor vessel head storage area. The trip logic is basically a one-out-of-four system. Any upscale trip will cause the desired action. Trip settings of 9 mr/hr in the duct and 50 mr/hr on the refueling floor are based upon initiating the Standby Gas Treatment System so as not to exceed allowed dose rates of 10CFR20 at the nearest site boundary. Reaching the 9 mr/hr setting causes an instantaneous trip, whereas reaching the 50 mr/hr setting causes a trip after a two minute delay. The SGTS is also automatically initiated by high drywell pressure or reactor low-low water level, or can be actuated manually from the Control Room.

Additional operational limitations of the SGTS are provided in the OCNGS Technical Specifications.

Automatic Initiation

During normal operation, the Reactor Building normal ventilation is operating. During a DBA, upon receipt of an RPS signal, the normal ventilation supply and exhaust fans are automatically tripped; the Reactor Building Isolation Valves (secondary containment isolation) are automatically closed, and the SGTS fans are automatically started to maintain a negative pressure in the secondary containment.

When required, both exhaust fans will start and all valves will open. One filter train is preselected as the running system (the other serves as a backup). After a time period, if the selected fan has developed sufficient flow, the backup exhaust fan will shutdown and its train will isolate (the inlet orifice purge valve will remain open however). Automatic initiation of the SGTS is discussed in more detail in Section 7.3.

Manual Operation

The individual SGTS trains can be placed in service manually from the Control Room, either in parallel with normal ventilation (for test) or with normal ventilation shutdown. Placing the exhaust fan control switch from its normal AUTO position - through OFF - to HAND position, will start the fan and in turn, automatically line up the associated dampers for operation.

Once initiated, either manually or automatically, the SGTS must be manually shutdown when no longer required. If started automatically, removal of the initiating signal (requiring a manual reset) will shutdown the SGTS. Normal ventilation must then be restarted manually.

System Swap Over

If one SGTS is in operation with the other in standby and the running system experiences low flow for any reason, the standby fan will automatically start and the system damper lineup will adjust accordingly.

The running filter train will be automatically isolated by closing the inlet and outlet valves and the fan will be shutdown manually. However, that train's inlet orifice purge valve and the cross-tie valve automatically open to provide a path of cool air flow through the idle filters.

The air temperature is measured downstream of the second absolute filter. The electric heating coil is controlled automatically to maintain less than 70 percent relative humidity with temperature at about 120°F. Fission product decay heat in the filters, following an accident condition, is cooled by low bleed airflow through the orifice and cross-tie valves to the operating fan.

Secondary Containment Leak Rate Test

It is necessary to make periodic measurements of the inleakage rate to the Reactor Building, with the building isolated. The Standby Gas Treatment System provides a convenient and accurate method to effect this measurement, which can be used at any time during any shutdown or operating mode of the reactor. One of the standby gas treatment units (A or B) is manually operated, then the normal Reactor Building Ventilation System is shut down by closing all isolation valves in the supply and exhaust ducts and by tripping the supply and exhaust fans. The resultant air flow through the Standby Gas Treatment System that can maintain a negative pressure of 0.25 in. W.G. in the Reactor Building is verified to be within $\leq$4000 cfm as required by Technical Specifications.

This test is continued for the required time (about 15 to 30 minutes) and the building leak rate is satisfactory if the Reactor Building is maintained at 0.25 in W.G. negative pressure. This test is normally not performed when the wind velocity is high.

Functional testing of the SGTS is conducted to verify a flow rate of 2600 cfm $\pm$10%. During this test the equipment (fan, valves, strip heater, instrumentation, filter pressure drop, and other parameters) of the Standby Gas Treatment System is checked for proper operation.

An annubar located in the calibrated test section of the SGTS provides an accurate measurement of SGTS flow during the test or emergency operation.

Drywell and Suppression Chamber Venting

The drywell and suppression chamber gas spaces are inerted with a nitrogen atmosphere during power operation. The drywell is isolated from the Reactor Building ventilation system and is cooled by five recirculating fans and coolers. The reactor cavity space between the drywell head and the floor shielding is ventilated by the normal building system.

To provide access to the drywell, and before removing the drywell head for a refueling outage, the nitrogen gas is purged and replaced with air. Several air changes will be required to adequately purge the drywell. The Drywell Vent & Purge Valves provide this purge in a reasonable time. The gas is purged through the normal ventilation system, to the exhaust fans, without gas treatment. Should the nitrogen atmosphere be contaminated with radioactive particles or iodine to the extent that the unfiltered purge would result in excessive radioactivity in the ventilation system exhaust, normal ventilation will automatically shutdown and the drywell and suppression chamber can be purged through the Standby Gas Treatment System. Purge rate is then limited to the capacity of both gas treatment systems operating in parallel.

During a DBA, the torus can be depressurized through a 2 inch valve and a 12 inch valve. This pathway limits flow to the gas treatment system to protect the filters. The torus is purged through two 12 inch valves only during inerting and purging operations. The valves are controlled by two position switches, but all valves close automatically upon receipt of an RPS signal to maintain primary containment unless they are manually bypassed by permissive opening of the 2 inch valve.

Similarly, the drywell can be purged during a DBA by opening two 2 inch valves. The drywell and torus are purged during startup through two 18 & 12 inch valves, respectively. The drywell isolation valve bypass permissive switch in the Control Room allows the torus depressurization and purge valves and the drywell DBA purge valves to be opened if a drywell isolation signal is present, provided that the reactor mode switch is not in the RUN position. The drywell startup purge valves can be opened when the reactor mode switch is in the RUN position by operation of a two position bypass switch in the Control Room.

6.5.1.3 Design Evaluation

The Standby Gas Treatment System, together with the low leakage rate of the Reactor Building, substantially reduces the consequent offsite dose which might arise from postulated accidents. This is accomplished by:

a. Placing the Reactor Building under negative pressure and thus preventing the ground level release of the building atmosphere;

b. Filtering the Reactor Building air exhaust for the removal of radioactive particulates and halogens; and

c. Providing for elevated release to achieve significant decreases in ground level concentrations due to high atmospheric dispersion.

An analysis has been made of the effect of increased Standby Gas Treatment Gas flow rates on offsite doses previously analyzed. The result of this analysis utilizing both Atomic Energy Commission and General Electric methods shows that even for up to 5,000 cfm flow, the requirements of 10CFR100 will be achieved with margin. Using General Electric assumptions and methods, the offsite doses following a Loss-of-Coolant Accident have been evaluated for a spectrum of Standby Gas Treatment System (SGTS) exhaust flow rates. The results of these calculations are presented in Figure 6.2-37 and are expressed in terms of a relative dose rate factor. The base case considers a primary containment leakage rate of 0.5%/day and a secondary containment ventilation rate of 1,200 cfm. As notes in Figure 6.2-37, a primary containment leakage rate of 1.0% per day and a SGTS flow rate of 5,000 cfm (maximum limit for design) results in a relative dose rate factor of approximately 8 times for less than 2 hours and a factor of 4 for times equivalent to 1 day. The 2 hour offsite doses for the base case of 0.5% per day primary containment leakage are 1.9×10^{-5} rem thyroid. Since these values are orders of magnitude below the guidelines in 10CFR100, it can be concluded that a SGTS flow rate up to 5000 cfm is acceptable.

Subsequent to this analysis, the SGTS flow rate was set at 2600 cfm, with a Technical Specification limit of 4000 cfm.

A more thorough discussion of the Reactor Building inleakage and exfiltration is presented in Section 6.2.3.

The Standby Gas Treatment System filters will have adequate efficiency during accident conditions. At relative humidities below 70%, all halogens, including organic, are filtered by impregnated charcoal filters with an efficiency well above 90%. The required carbon filter efficiency is indicated in the Technical Specifications.

SGTS automatic initiation and operation of the heating coils for both filter trains are dependent upon a single power source. An evaluation was performed to assess SGTS performance in the event this power source is lost. Assuming the SGTS is manually initiated 30 minutes after a DBA LOCA with 78% halogen removal efficiency (due to 100% relative humidity), offsite doses remain well within the guideline values of 10CFR 100 (Reference 1).

In addition to its functional performance of mitigating accident consequences, the Standby Gas Treatment System provides a method for testing the gross leakage of the Reactor Building.

Each filter unit includes the following (in the direction of air flow):

a. A prefilter (roughing filter) which removes large particles.

b. A heating system to maintain the humidity in the air stream to a maximum of 70% relative humidity.

c. A high efficiency particulate air (HEPA) filter, water resistant, capable of removing 99 percent minimum of particulate matter which is 0.3 micron or larger in size. HEPA filters meet the requirements of ASME AG-1, Section FC.

d. An iodine filter, an impregnated activated carbon bed, following the upstream particulate filter, capable of removing 90 percent minimum of iodines.

e. A second high efficiency particulate air (HEPA) filter, following the iodine filter, capable of removing 99 percent minimum of particulates 0.3 micron or larger in size.

The particulate and charcoal filters are sized to remove the fission products released to the Reactor Building prior to releasing to the environs following any of the postulated accidents. The charcoal filter capacity is for the removal of organic halogens, for which these filters are adequately designed.

The following discussion of filter heat loads is the response to AEC questions on the original FDSAR. It is included, intact, as required by 10CFR50.71(e) for historical purposes only, and may not necessarily reflect the current system design.

The heat load on the filters and estimated temperatures, which are calculated on the bases of 25% of the equilibrium inventory released from the fuel and a filter removal efficiency of 99%, are given below:

Primary Containment Leakage Rate	Heat Load	Filter Temperature (Static Condition)
0.5%/day	1,800 watts	850°F
2.0%/day	6,500 watts	1500°F
5.0%/day	16,000 watts	2300°F

The temperature on the filter was estimated assuming the only heat transfer mechanisms were thermal radiation and natural convection around the four outside surfaces of the filter. The ducting to and from the filter, which act as fins, were neglected in the calculation as were the natural convection and thermal radiation from the faces of the filter in the direction of the ducting. For developing a correlation for natural convection and thermal radiation the external ambient was assumed to be 100°F. The thermal radiation shape factor was assumed to be 1.0 since the filters are small in comparison to the physical size of the tunnel. An overall emissivity factor of 0.5 was chosen.

It is assumed that under static conditions (no air flow through the filter) the ignition temperature of the charcoal is on the order of 1400 - 1500°F. However, at the 70 feet per minute air flow velocity of the filter the ignition temperature is only 890°F.

No special provisions have been made to extinguish fires in the SGTS since a realistic evaluation of the filter heat loads shows that the temperatures remain considerably below the filter ignition temperature.

The above calculations have been made in response to an AEC request based on a specific set of hypothetical conditions. The postulated conditions imply halogen release fractions associated with 100% melted fuel when there shall be: (1) only clad perforation (less than 50%) and no melted fuel, (2) no partitioning of halogens in the absorption chamber water when there is considerable evidence to support such phenomena, (3) continuous leakage from the primary containment at the indicated rates even though calculations show that the containment will return to atmospheric pressure within one day, and (4) no holdup and mixing of the halogens within the Reactor Building when in fact all primary containment leakage is to the Secondary Containment (i.e., Reactor Building). When these factors are taken into consideration, the filter heat loads would be as shown below:

Primary Containment Leakage Rate	Heat Load	Filter Temperature (Static Condition)
0.5%/day	0.6 watt	Ambient
2.0%/day	2.2 watts	Ambient
5.0%/day	5.3 watts	Ambient

The system equipment is accessible and capable of being tested and maintained during normal plant operation. Connections for injection and sampling are located to provide adequate mixing of the injected fluid and representative sampling and monitoring, so that test results are indicative of performance under design conditions.

6.5.1.4 Tests and Inspections

Filter Efficiency Tests

Each replacement absolute filter will be thoroughly inspected for damage, tears, and pin holes by illuminating the back side with a strong light. Any damage, including tears, pinholes, and faulty edge sealing will be cause for rejection.

After installation, the absolute filters are given a preoperational test by injecting DOP (dioctylphthalate) upstream of the filter and surveying the downstream side for DOP penetration. The downstream surface of the filter is tested by scanning with a test instrument. There are injection and sample ports provided in the SGTS ducts. The DOP test is performed in accordance with ANSI-N510-1975.

The charcoal filters are given a preoperational test to prove their in-place integrity rather than their capacity or CH_3I removal capability. The filter integrity test uses a halogenated hydrocarbon refrigerant test gas injected upstream of the filter, and a halogen leak detector connected to a scanning probe downstream of the filter, to detect any test gas penetration. The test is performed in accordance with ANSI-N510-1975.

Normal iodines and methyl iodide filtering efficiency and capacity are determined by special laboratory tests, using samples of the charcoal as installed. Test canisters are installed with the filters (Figure 6.5-2) for periodic removal and laboratory testing.

If laboratory tests for the adsorber material in one train of the Standby Gas Treatment System are unacceptable, all adsorber material in that circuit is replaced with adsorbent qualified according to the requirements of Regulatory Guide 1.52-1978. Replacement HEPA filters found defective shall be replaced with those qualified to the requirements of Regulatory Position C.3.d of Regulatory Guide 1.52-1978.

During operation, the capability of the SGTS is demonstrated by testing as required by the Technical Specifications.

6.5.1.5 Instrumentation Requirements

Local instrumentation is provided for each train of the SGTS. Panels ATC-P-15 and ATC-P-16 house the instrumentation and control devices for SGTS trains A and B respectively. These panels are NEMA 4 enclosures, suitable for their outdoor location. Outdoor run of cable is in conduit. Tubing, in multitube bundles, is provided with heat tracing designed to generate 4.0 watts per linear foot. Each panel is powered from a different diesel generator during an emergency. A flow element is located in the circular section of the common suction header of the system, with a differential pressure flow transmitter connected across it. This transmitter is provided with an integral square root extractor, and the flow is displayed on a digital indicator on one of the SGTS panels.

A temperature sensor is located close to the flow element. Remote temperature indication is provided on the same panel where the flow indication is provided by a digital readout device. This indication is provided for the temperature compensation of the flow.

Two differential pressure indicating switches are provided, one across each of the HEPA filters in each SGTS trains. An additional differential pressure switch is connected across the entire HEPA filter train.

6.5.2 Containment Spray System

The Containment Spray System (Subsection 6.2.2) is manually actuated after a LOCA to reduce the elevated pressures caused by the release of the steam-water mixture into the Containment. The only mechanism by which the Containment Spray System reduces the offsite doses resulting from the LOCA is through providing a means of quickly reducing containment pressure, so that the primary to secondary leakage of radiosotopes is minimized. No chemicals are added to the containment spray water for the purpose of removing iodines from the containment atmosphere.

A complete discussion of the Containment Spray System design and capabilities is given in Subsection 6.2.2.

6.5.3 Fission Product Control Systems

Fission product control systems are considered to be those systems whose performance controls the releases of Fission products following a Design Basis Accident (DBA). These systems are exclusive of the Containment Isolation System (Subsection 6.2.4).

6.5.3.1 Primary Containment

The OCNGS primary containment is a free standing steel structure, consisting of a drywell and a pressure suppression chamber. The welded carbon steel plate structure forms a continuous leaktight membrane to limit radioactive releases, and is encased within reinforced concrete, which provides the required shielding.

Details of the containment functional design are discussed in Subsection 6.2.1.

The Containment Spray System, as described in Subsection 6.2.2 is designed to rapidly reduce containment pressure following a DBA. The containment pressure and temperature transients following such an accident are presented in Subsection 6.2.1.

The Containment Vent and Purge System relieves containment pressure resulting from operation of the Nitrogen Inerting System. This function is accomplished by routing purged air through the SGTS. The detailed operation of the Nitrogen Inerting System and the SGTS are describe

6.5.3.2 Secondary Containment

The Secondary Containment totally encloses the Primary Containment, the refueling and reactor servicing areas, the new and spent fuel storage facilities and other reactor auxiliary systems. The Secondary Containment serves as the containment when the Primary Containment is open, and as an additional barrier when the Primary Containment is functional. The Reactor Building, the Standby Gas Treatment System and the Stack make up the Secondary Containment.

Any fission products released from the Primary Containment into the Secondary Containment are held up, sampled, and eventually released at an elevated point, thereby reducing offsite doses to a small fraction of permissible doses as demonstrated in the analyses presented in Chapter 15.

6.5.4 Ice Condenser as a Fission Product Cleanup System

Not applicable to Oyster Creek NGS.

6.5.5 References

1. Calculation No. 9340-89-004, Rev. 0, "Calculation of Offsite Exposure due to Loss of Offsite Power and Subsequent SGTS Reduced Efficiency"

TABLE 6.5-1
(Sheet 1 of 2)

MAJOR STANDBY GAS TREATMENT SYSTEM COMPONENTS

Inlet and Exhaust Valves	
Type	Flanged, air cylinder operated, butterfly
Design Conditions	150 psig, 150°F
Material	Cast iron body
HEPA Filters	
Filter Media	Glass microfiber CM-115
Filtering Efficiency	99% for particles 0.3 micron and larger, as measured by dioctylphthalate smoke test (DOP)
Flow Rate	2600 cfm
Clean Pressure Drop	2" W.G. at 2600 cfm 1.8" W.G. at 2600 cfm
Maximum Operating Differential Pressure	10" W.G.
Charcoal Filter	
Weight of Charcoal	300 lbs.
Removal Efficiency	99.9% elemental iodine 90% methyl iodide
Total Retention Capacity	400 gm of total iodines, including 40 gm CH_3I (methyl iodine)
Charcoal Bed Depth	2 in.
Nominal Residence Time	0.25 sec.
Nominal Flow Rate	2600 cfm
Nominal Flow Velocity	40 fpm
Clean Filter Pressure Drop	1.1 in. W.G. maximum at 2600 cfm

TABLE 6.5-1
(Sheet 2 of 2)

MAJOR STANDBY GAS TREATMENT SYSTEM COMPONENTS

Charcoal Material	Impregnated with nonradioactive iodine salt (KI); hard variety, nondusting; high temperature
Filter Frame	Provides five charcoal trays in parallel
Required Relative Humidity and Temperature of Air to Filter	70% relative humidity maximum, and 120°F
Electric Heater	
Heater Rating	30 kw, 460 volt, 3 phase
Fans	
Fan Type	SWSI (single width, single inlet)
MFR Model	Buffalo Forge Fan Model Industrial Exhauster Size 21 AW
Fan Wheels	Steel, with airfoil-type blade, non-overloading, backward curved blades
Shaft	Steel
Bearings	Pillow block type, with heavy cast iron closed end casing; single row, deep groove, self aligning, grease lubricated race assembly with built-in seals
Fan Housings	Bolted scroll, fixed discharge type; steel sheet, stiffened with rigid angle framework
Capacity	2600 cfm
Outlet Velocity	6450 fpm
Total Static Pressure	16.5 in. W.G.
Fan Speed	4050 rpm
Drive	V-Belts sized for 150 percent of motor horsepower
Electric Motor	1800 rpm, 15 hp, 460 volt, 3 phase

6.6 INSERVICE INSPECTION OF CLASS 2 AND 3 COMPONENTS

The Technical Specifications for the Oyster Creek Nuclear Generating Station state that inservice examination of ASME Code Class 2 and 3 components shall be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable addenda as required by 10CFR50.55a(g), except where specific written relief has been granted by the Commission. Certain requirements of later editions and addenda of Section XI are impractical to perform on older plants because of the plant's design, component geometry, and materials of construction. Thus 10CFR50.55a(g)(6)(i) authorizes the Commission to grant relief from those requirements upon making the necessary findings.

By letter dated April 16, 1992, (C321-92-2119), GPUN submitted its application to the NRC to update the Inservice Inspection Program, to the 1986 Edition of the ASME Section XI Code. This application included GPUN's intent to use approved Code Cases and requests for Code relief. In response to GPUN's submittal a letter dated October 25, 1994, was received from the NRC containing their evaluation of our Inservice Inspection Program update and associated requests for relief.

As a result of the NRC evaluation and approval, the latest version of the Inservice Inspection Program for Oyster Creek Nuclear Generating Station is based on the 1986 Edition of the ASME Boiler and Pressure Code, Section XI. The Program incorporated the 1990 addenda from the 1989 Edition, Subsection IWF, for the examinations of component supports. These Inservice Inspection Codes are applicable to the third ten year inspection interval from March 15, 1992 to March 14, 2002.

Inspection of structural weldments that are under purview of American Welding Society Standard D1.1 or other non-ASME class structures shall be conducted in accordance with the provisions of Visual Weld Acceptance (VWAC), Revision 2.

6.7 MAIN STEAM LINE ISOLATION VALVE LEAKAGE CONTROL SYSTEM

The Main Steam Isolation Valves (MSIVs) are containment isolation valves designed to minimize coolant loss from the vessel, and the resulting offsite dose, in the event of a Main Steam Line Break Accident. Two isolation valves are installed in each of the two 24 inch main steam lines, in parallel horizontal runs that penetrate the drywell through 36 inch diameter openings at El. 27'-0". The valve arrangement is shown in Figure 6.7-1.

Valves V-1-7 and V-1-8 are located inside the drywell; Valves V-1-9 and V-1-10 are located in the Secondary Containment (trunnion room), beyond the drywell wall. The leakoff lines for the MSIV's have been terminated and capped.

Following the intent of Regulatory Guide 1.96, Revision 1, Design of Main Steam Isolation Valve Leakage Control Systems for Boiling Water Reactor Nuclear Power Plants, the established inservice inspection program for the MSIVs ensures that isolation valves are maintained in such a manner that leakage is within Technical Specification limits. Periodic valve testing is conducted as prescribed in the Technical Specifications, which are referenced in Section 16.2.

6.8 OTHER ENGINEERED SAFETY FEATURES

Engineered safeguards which are provided in addition to those safety features included in the design of the Reactor, Reactor Coolant System, Containment System, Instrumentation and Control Systems and other process systems, include the following:

Engineered Safeguard	Section
Control Rod Velocity Limiter	4.6
Control Rod Housing Support	3.9
Standby Liquid Control System	9.3
Main Steam Line Flow Restrictors	5.4
Fire Protection System	9.5

These are discussed in detail in the referenced sections.