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5.4 COMPONENT AND SUBSYSTEM DESIGN

5.4.1 Reactor Recirculation System

5.4.1.1 Design Bases

The Reactor Recirculation System has been designed to perform the following functions:

a. To provide forced circulation of reactor water through the core to overcome the power density limitation of the fuel.

b. To provide a variable moderator (coolant) flow through the core to control reactor power without manipulation of the control rods.

The Reactor Recirculation System has been sized to provide a total flow capacity equal to the required flow at rated load. The design pressure of the recirculation pumps is 1300 psig, with a design temperature of 575°F. The system piping and valves have been designed for a pressure of 1200 psig and a temperature of 570°F.

5.4.1.2 System Description

The system consists of the reactor vessel and five piping loops, as shown in Drawing GE237E798. Each loop comprises one motor driven pump, a motor generator (M-G) set, suction and discharge valves, a bypass valve around each discharge valve, pipe support hangers, piping, and associated system controls and instrumentation.

Recirculated coolant enters the lower head of the reactor through vessel nozzles, passes through the diffuser and orifices at the bottom of the core and flows upward through the core where bulk boiling produces steam. The steam-water mixture enters the moisture separators and then the steam dryers. The water separated from the steam flows downward across the top of the plenum, where it mixes with the incoming feedwater, and enters the downcomer annulus between the shroud and the vessel wall. The coolant flows through the downcomer region, through the outlet nozzles, and into the recirculation pumps suction piping. The coolant is then returned to the vessel via the pumps and discharge piping.

The continuous circulation ensures that hot spots are not created by steam bubbles, which would result in steam blanketing around fuel rods and in reduction of the heat removal capability of the coolant. To control reactor power level, the system makes use of the boiling water reactor large negative power coefficient. A power level increase is achieved by increasing the recirculation flow, which reduces core voids and thereby increases reactivity in the core. As power increases, the boiling rate increases and the void fraction increases, adding negative reactivity and stopping the power increase. The core then stabilizes at a high power level. Decreased flow results in the opposite effect.

Reactor operation is permitted with up to two inoperable recirculation loops within the following constraints:

a. **No more than one loop is isolated (suction, discharge and discharge bypass valves are closed) at any time during power operation.**

b. **The isolated loop is not restarted until the plant is in the cold shutdown condition.**

c. **Power is limited to 90% of rated when there are two inoperable recirculation loops.**

5.4.1.2.1 Recirculation Pumps

The Recirculation Pumps (Drawing GE107C5339) are vertical, single stage, centrifugal pumps driven by 1000 horsepower motors. With five pumps in operation, each pump will deliver 32,000 gpm at 1017 psig and 520°F.

The pump motors are 4160 volts open, drip proof, three phase, squirrel cage induction type, especially designed for operation on a variable frequency power supply with an operating range between 11.5 and 57.5 cps. Motor insulation and cooling are designed for continuous operation in 135°F ambient temperature and limited operation at 150°F maximum ambient temperature.

Oil lubricated double acting thrust bearings are provided for the motors. Cooling is provided by the Reactor Building Closed Cooling Water (RBCCW) System, at a rate of approximately 4 gpm, to an internal oil cooler.

The pump seal assembly for each pump consists of two sets of cartridge type mechanical seals and a breakdown bushing. Under normal operating conditions, each seal provides approximately 500 psi pressure drop and forms two cavities from which the pressures are measured. Restrictive orifices control leakage to approximately 0.5 gpm. Seal cooling is supplied to the shell side of the seal heat exchanger by the RBCCW System at approximately 25 gpm. An auxiliary impeller, mounted on the main shaft, circulates coolant through the tube side of the heat exchanger to the seal cartridge.

5.4.1.2.2 Motor Generator Sets

Five variable frequency motor generator (M-G) sets are provided to control the speed of the five Recirculation Pumps. The M-G sets are located at ground level in a separate room adjacent to the southwest corner of the Reactor Building. The electrical output of each generator is directly tied to its associated Recirculation Pump motor in the drywell. Each M-G set consists of a horizontal induction motor driving a synchronous generator through an adjustable fluid coupling.

Generator drive is provided by one 1250 hp, 4160 volt, air cooled motor for each of the M-G sets. The major components of the fluid coupler are an impeller, a runner and a scoop tube. The coupler transmits power from the drive motor to the variable speed generator by a vortex of oil.

The impeller is coupled directly to the input shaft and rotates at a constant speed of 1180 rpm. The runner is directly coupled to the output shaft and to the generator. The runner rotates at a speed determined by the quantity of oil in the vortex and load conditions. The output shaft can operate at any speed between 230 rpm and 1160 rpm without causing overheating or oil foaming.

The scoop tube controls the quantity of oil in the vortex and, in turn, the amount of power transmitted from the drive motor to the generator. The oil acts as the coolant as well as the power transmission medium. An oil cooler is provided to ensure safe operating temperatures, and receives cooling water from the Turbine Building Closed Cooling Water System. Positioning of the scoop tube can be accomplished from the Control Room by means of an electropneumatic positioner, or manually at the M-G Set. The positioning mechanism **will** lockup the scoop tube on loss of air supply or upon failure of the instrument circuit.

5.4.1.2.3 Valves

The recirculation loops contain valves for isolating any pump from the Reactor Coolant System. The 26 inch motor operated suction gate valves have a full port opening of 26 inches. These are designed to open against a 50 psi differential pressure which is equivalent to the static head of water in the reactor vessel.

The discharge valves are 26 inch motor operated gates with a reduced port opening of 24 inches. These are designed to open against a pressure of 100 psid. The discharge bypass valves are provided to serve as a low flow bypass during pump startup and as a pressure equalizing valve across the suction and discharge valves. The bypass valves are 2 inch motor operated and designed to open against full reactor pressure.

The stroking rate of the recirculation loop valves is approximately 12 inches per minute, resulting in approximate closure times of: 2 minutes and 20 seconds for the suction valves, 2 minutes for the discharge valves, and 10 seconds for the discharge bypass valves.

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 212°F. There are two exceptions to the full-open valve/recirculation loop configuration.

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The first exception is when the reactor water level is greater than 185 inches above TAF and the reactor coolant temperature is less than 212°F. 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 operator and steam dryer are removed from the reactor pressure vessel and the reactor coolant temperature is less than 212°F. With the removal of the stem 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.

The discharge and suction valves have a direct impact on the communication between the reactor vessel downcomer and core levels. With these valves open, changes in core water level result in corresponding changes in the vessel's downcomer water level. To insure ECCS actuation on Lo-Lo level, at least one suction and discharge valve in a recirculation loop must be open. The exception to this is in the refueling mode with level above 185 inch TAF and temperature less than 212°F and no work is being done which could cause water level to be reduced to less than 4' 8" above the top of active fuel.

5.4.1.2.4 Supports

The recirculation piping, pumps and drive motors are suspended from constant support hangers to reduce thermal expansion stresses. Travel of the hangers is greater than calculated thermal growth.

5.4.1.2.5 Piping and Nozzles

The recirculation loop piping is all of Stainless Steel welded construction and has been designed, manufactured and constructed to meet, as a minimum, the requirements of the ASME B&PV code, Section I, and ASA B31.1 - Code for Pressure Piping.

There are five 26 inch loops, with a 2 inch bypass line around each pump discharge valve. The vessel outlet nozzle is 33 inches reducing to 26 inches and the vessel inlet nozzle is 26 inches in diameter.

The return lines from the Isolation Condenser System (Section 6.3) are connected to the suction side of recirculation loops A and E upstream of the suction valves. The supply line to the Shutdown Cooling System (SCS) is connected to the suction side of the recirculation loop E at the Isolation Condenser line. The return line from the SCS is connected to the discharge line of recirculation loop E, downstream of the discharge valve.

The supply line to the Reactor Cleanup System is connected to the suction line of recirculation loop B and the return is to the discharge line of the source loop.

5.4.1.2.6 Controls and Instrumentation

The Reactor Recirculation System is provided with a speed control unit. The unit consists of a pneumatic operator for each fluid drive scoop tube, an electric tachometer on each generator shaft, a remote manual speed controller for each M-G set (with speed and scoop tube position indicators), a master remote control device for all five pumps, and all necessary electronic equipment.

Operating speeds of all five pumps are normally adjusted in unison by the master speed controller. The individual speed controller is used for taking a pump out of service or returning it to normal operation. The tachometer on each generator shaft provides a feedback signal for comparison of actual versus selected speeds.

Instrumentation is provided for monitoring the recirculation pump motors, seals and seal cooling water parameters, loop temperature and loop flow. The necessary interlocking features are provided for protection of the system components and equipment. Recirculation flow through each loop and total recirculation flow are indicated in the Control Room, as well as loop temperature, valve position and pressure change across the pumps.

5.4.4.2 Description

The flow restrictor is a simple venturi type tube welded into each main steam line between the reactor vessel and the first MSIV. The restrictor has no moving parts and is located as close to the vessel as practical. The ratio of the venturi throat area to steam line flow area is about 0.6, which results in an irreversible pressure drop of 5 to 10 psi. The design limits the steam flow in a severed line to about 191 percent of the full design rated steam flow, yet results in negligible increase in steam moisture content during normal operation. The restrictor is designed to withstand the maximum pressure difference expected following complete severance of a main steam line.

5.4.4.3 Design Analysis

In the event of a steam line break downstream of the restrictor, flow chokes in the decreased area by a two phase mechanism similar to the critical flow phenomena in gas dynamics. This limits the steam flow, thus reducing reactor coolant blowdown and limiting fuel clad temperature rise. The probability of fuel failure is thereby reduced.

Pressure surges caused by water-steam slugs impacting the flow limiter are within design limits; while beyond the restrictor velocities are reduced and pressure surges are of no consequence. The throat section of the flow limiting venturi is fitted inside a section of the 24 inch main steam line and held in place with a full circumferential 5/8 inch fillet weld. A calculated impact pressure of 1510 **psia** results in a 314,000 lb shear force on this weld and a resulting stress of 10,800 psi. Design code allowable stress for the material was 15,000 psi at 600 F and yield strength 60,000 psi. The analysis of the steam line rupture accident is presented in Chapter 15.

The flow restrictors have no moving parts and require no maintenance. Tests conducted to determine final design and performance characteristics of the restrictor have shown that:

a. Restrictor performance is in agreement with ASME correlations.

b. The loss of pressure is consistently about ten percent of total restrictor differential pressure.

c. Operation at critical throat velocities is stable and predictable.

5.4.5 Main Steam Line Isolation System

5.4.5.1 Design Bases

Main steam line isolation is accomplished by means of the Main Steam Isolation Valves (MSIVs). The MSIVs are containment isolation valves designed to minimize coolant loss from the vessel and thus offsite doses in the event of a main steam line break accident.

The valves are designed to close within three to ten seconds. The minimum closing time is chosen to minimize pressure buildup in the reactor vessel due to a quick cutoff of steam flow from the vessel. A three second valve closure limits vessel pressure to 1138 psia at the core midplane. The ten second maximum closing time is based upon the steam line break accident, the resulting loss of coolant and the resulting offsite dose rate (Chapter 15).

5.4.5.2 Description

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" and azimuths 171°15' and 180°45'. The penetrations have expansion joints to allow for steam line movement. One valve is located inside Containment, and the other outside Containment. Both sets of valves are located as close as possible to the drywell penetrations.

The basic design of the four valves is identical. Cross sectional views of the valve are shown in Drawing 20451-H. The valves are 24 inch angled globe valves of "Y" configuration. The cup shaped poppet moves on a centerline that is 45° upward from the horizontal centerline of the piping run. The valves in the inboard and outboard sets are rotated inward toward each other at 22°30' from vertical so that the air cylinders clear downcoming steam lines and other neighboring lines. Refer to Section 6.7 for discussion on leakage.

The diameter of the main valve seat is approximately the same as the inside diameter of the pipe and entrance and exit are streamlined to minimize pressure drop through the valve during normal steam flow. (Normal pressure drop is approximately 5.8 psi.)

Because of a history of LLRT failures of OCNGS MSIV's, the MSIV manufacturer, Atwood and Morrill, subsequently (Cycle 12R outage) modified the original MSIV design to provide assurance that the MSIV's would meet the LLRT requirements and to ensure reliable and acceptable valve seat tightness. The valve manufacturer's modification to the original valve components consists of the following:

- A new and improved main poppet with extended nose for proper seating.
- Internal poppet design with "self-aligning" feature to minimize seat leakage of the pilot poppet.
- New bottom spring plate design to allow separate disassembly of actuator assembly from cover poppet and stem assembly.
- **A poppet backseat has been added to the valve cover to minimize flow induced poppet vibration. (NS03B only)**

The modified valve parts are designed to meet ASME Code allowable stresses requirements and the system design rating.

Each valve is controlled independently using a pneumatic system. The controls are capable of opening and slow speed exercising and fast closing the MSIVs one at a time.

5.4.7.3 System Evaluation

The SCS has components classified as NSR, RR, and other. These are either seismic I or II. **The SCS is classified as Seismic Category I inside the Drywell up to and including the normally closed valves outside of the Drywell (V-17-1, V-17-2, V-17-3 on the pump suction piping, and V-17-55, V-17-56, V-17-57 on the pump discharge piping). This includes the local drain and vent piping to the normally closed valve(s). The portion of the system outside of the Drywell beyond the listed normally closed drain/vent valves, including the pumps and heat exchangers and piping in between, is Seismic Category II. The SCS outside the containment beyond the normally closed valves will remain mechanically intact following a design basis seismic event. Therefore, while not Seismic Class I it is considered seismically capable.** Access to components of the system is controlled because of high radiation conditions.

5.4.8 Reactor Water Cleanup System (Reactor Cleanup System)

5.4.8.1 Design Basis

The Reactor Cleanup System is a filtration and demineralization system for maintaining the purity of the water in the Reactor Coolant System. The system is operated to:

a. Reduce the deposition of water impurities on fuel surfaces, thus minimizing heat transfer surface fouling.

b. Reduce secondary sources of beta and gamma radiation by removing corrosion products, impurities and fission products from the reactor coolant.

c. Reduce the concentration of Cl^- ions to protect steel components from chloride stress corrosion.

d. Maintain or lower water level in the reactor vessel during startup, shutdown and refueling operations, in order to accommodate reactor coolant swell during heatup and to accommodate water inputs from the Control Rod Drive System and the Head Cooling System.

The system is designed to perform its function with minimum heat loss from the Reactor Coolant System, and to be operated during all phases of normal plant operation.

5.4.8.2 System Description

The system includes a regenerative heat exchanger, a non regenerative heat exchanger, a pressure reducing station, cleanup filters and auxiliaries, a cleanup demineralizer, cleanup pumps, a surge tank, a flow control station, a reactor drain station, isolation valves, piping, instrumentation and controls (Drawing GE148F444).

Under normal operation, reactor coolant flows under reactor pressure from the suction of reactor Recirculation Pump B, is cooled to 120°F in the regenerative and non regenerative heat exchangers (in series), its pressure reduced to 110 psig, filtered, demineralized, and pumped through a flow control valve and the regenerative heat exchanger to the discharge of reactor Recirculation Pump B. When reactor pressure is insufficient to maintain the required suction pressure at the cleanup recirculation pump, an auxiliary cleanup pump is placed in operation.

For draining the reactor, some of the cleanup system effluent flow is directed to the hotwell or to radwaste via a second flow control valve at the reactor drain station. The normal drain path is through the recirculation loop and cleanup system. There is also a drain line, from the bottom of the reactor vessel, which has normally open manual valves to the cleanup system.

The system is operated to maintain low levels of reactor water conductivity and undissolved solids. Conductivity is monitored at the influent (two cells) and at the effluent (one cell) of the demineralizer. Recording capability is provided, and abnormal conditions alarmed in the Control Room. The design flow rate is 380,000 lbs/hr.

The system supply line has a motor operated isolation valve inside the drywell and two parallel motor operated valves outside the drywell. The return line has one motor operated valve outside the drywell and one check valve inside the drywell. The isolation valves will close, and the cleanup pumps will stop automatically under any of the following conditions **(refer to Table 6.2-12 for isolation signals for each valve)**:

a. Low flow, or outlet valve shut, for the cleanup filter in service.

b. High auxiliary pump cooling water outlet temperature.

c. High non regenerative heat exchanger outlet temperature (reactor coolant).

d. High pressure from the pressure reducing station.

e. Liquid poison system flow into the reactor vessel.

f. High drywell pressure.

g. Low-low reactor water level.

h. High area temperature (RWCU HELB isolation signal).

An exception to the above is valve V-16-61 which closes only on low-low reactor water level, high drywell pressure or High area temperature (RWCU HELB isolation signal).

A backup isolation on high pressure from the pressure reducing station isolates V-16-2 and V-16-14 only.

Reactor Water Cleanup 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 bases 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.

The major characteristics of the Reactor Cleanup System components are presented in Table 5.4-1. The equipment is designed to ASME B&PV Code, Section III, Class C on the primary side. The tube side of the heat exchanger is ASME VIII.

The pressure reducing station consists of a pressure control valve and a bypass control valve, and relief valves. Pressure is maintained at or below 110 psig in the filter and demineralizer portion of the cleanup system. High pressure from the pressure reducing station trips the cleanup system isolation valves and pumps.

The pressure control valve is a 4 inch, globe, single seat valve, air operated, air to open, **fail** close, with pressure controller. A pressure relief valve located just downstream of the pressure control valves protect the low pressure portions of the cleanup system. One 6 inch valve can discharge up to 125 pounds per second through a 20 inch line and one isolation check valve to the torus. A remote operated solenoid leakoff valve is used to detect relief valve leakage. One 1 inch valve is provided in line to the Reactor Building Equipment Drain Tank. The filters, the demineralizers, and other isolable portions of the system have relief valves which discharge to the Reactor Building Equipment Drain Tank.

Cleanup system flow is normally maintained at approximately 400 gpm. The system flow is measured by a flow element at the cleanup demineralizer inlet. Low flow at this element automatically starts the filter precoat pump to recycle, in order to hold the filter cake.

The flow control valve is a 4 inch globe, air diaphragm operated valve, air to open, spring and flow to close. This valve is located between the cleanup pumps and the regenerative heat exchanger. An electropneumatic converter supplies control air to the valve diaphragm, and is controlled by the flow controller, with feedback from the flow element. Cleanup pump suction header low pressure results in a flow reduction demand to the flow controller. A manually operated bypass valve is installed around the valve as backup to the automatic flow control. A flow recorder along with digital display is provided to monitor flow and system performance. The system is normally operated in the manual mode.

The reactor drain station consists of an auxiliary pressure-reducing valve and orifice in series, a motor operated orifice bypass valve, a flow element, and motor-operated shutoff valves to the hot well and to radwaste. The takeoff line for this station is located upstream of the cleanup recirculation pumps.

A normally locked open 6 inch manual valve (V-16-63) is located inside the drywell downstream of the return line check valve and provides a means of isolating the return line from the RPV. In order to implement repairs to V-16-63 with the plant mode switch in either the SHUTDOWN or REFUELING position, the use of a freeze seal positioned between the Reactor Vessel and V-16-63 may be necessary, subject to the following restrictions.

a. **RPV is reassembled and fuel is in the vessel.**

b. **The reactor coolant system bulk temperature, as measured by TE-31D, shall not exceed a temperature of 99°F prior to the initiation of the valve repair activity and no evolutions involving removal of the valve disc shall occur if the reactor coolant system bulk temperature exceeds 99°F.**

c. **RPV water level shall not exceed 210 inches above TAF.**

d. **Decay heat from the RPV will be removed by the Shutdown Cooling System via the RBCCW/Service Water Systems.**

e. **Both Core Spray Systems shall be operable in accordance with the Technical Specifications.**

f. **Fire Protection System shall be operable and capable of delivering water to the Core Spray System.**

g. **At least one loop of the Reactor Recirculation System must remain open with the suction and discharge valves in the full open position.**

h. **The Emergency Diesel Generators and their respective NSR electrical distribution systems are returned to service and available to support emergency operation.**

i. **Offsite power shall be available.**

j. **Secondary Containment shall be operable in accordance with the Technical Specifications.**

k. **Both Standby Gas Treatment Systems will be operable.**

l. **The RPV will be vented to the atmosphere and the established vent path will be tagged in the open position.**

m. **RWCU System pumps P-16-001A and P-16-001B will be electrically deenergized.**

n. **RWCU System valves V-16-1, V-16-2, V-16-14 and V-16-61 will be tagged in the closed position.**

o. **Provisions will be in place to remove any equipment traversing the Drywell Airlock to ensure that the airlock can be closed in a timely fashion should the need arise.**

p. **The seal plate assembly for V-16-63 shall be staged and ready for fit-up and installation.**

5.4.8.3 System Evaluation

The system is normally operated continuously during all phases of reactor operation. Cleanup system operation is necessary to maintain reactor coolant purity, and reactor operation without the cleanup system is limited to relatively short periods of time.

The system is automatically isolated from the Reactor Coolant System under abnormal operating conditions. Relief valves and instrumentation are provided to protect the system against overpressurization.

5.4.9. Main Steam and Feedwater Piping

5.4.9.1 Main Steam Piping

The dry steam in the reactor vessel head cavity is removed through two 24 inch nozzles on the sides of the reactor vessel. The nozzles are located due north and due south on the vessel, approximately 5'-8" below the vessel head flange. The headers curve downward to a horizontal header containing the safety and relief valves. Each header leaves the drywell through a 36 inch guard pipe with a flexible pressure seal to the header on the outside of the drywell. Each header enters the tunnel to the Turbine Building through a second pressure seal designed to prevent steam leaks from the valve space to the tunnel.

There are 14 safety/relief valves on the main steam piping. Nine safety valves discharge directly into the drywell (Subsection 5.4.10). The remaining five are the Electromatic Relief Valves (EMRVs), which are described as part of the Automatic Depressurization System in Section 6.3.

The main steam lines are provided with flow restrictors (Subsection 5.4.4) and with Main Steam Isolation Valves inside and outside Containment (Subsection 5.4.5). The main steam lines inside the drywell are shown in Drawing BR 2002. Piping was designed to ANSI B31.1.

Figure 5.4-7 shows the general arrangement of the main steam safety and relief valves on the main steam piping. It is anticipated that the safety valves will never be actuated since overpressure is relieved by the Isolation Condensers, the turbine bypass valves, and the Electromatic Relief Valves. The effects of "blowoff" of these valves have been studied and taken into account in the following way. First, the five relief valves are set to open at a pressure lower than those specified for the safety valves. The discharge from the relief valves is piped to the bottom of the torus - completely away from drywell equipment. Second, the discharge from each safety valve initially impinges on a tee which deflects the steam 90 degrees in two opposite directions. Each tee discharge is aimed specifically to prevent its direct discharge on equipment which could be affected from such forces.

In an overpressure situation, the relief valves would actuate before the safety valves, with subsequent discharge to the torus, followed by safety valve blowoff (if the overpressure situation persisted) which by virtue of the selected tee aiming will not adversely affect equipment in the drywell.

The transient response of the plant to inadvertent opening of a safety valve is discussed in Chapter 15.

An acoustic based Valve Monitoring System on the safety and relief valves provides valve position indication in the Control Room. The valve position indication is seismically qualified and capable of operation in its appropriate environment. Thermocouples provide backup monitoring capability.

5.4.9.2 Feedwater Piping

Downstream of the high pressure feedwater heaters, the 14 inch feedwater lines discharge into a common 24 inch header, which branches into two 18 inch lines to the Reactor Building. These run parallel to the main steam lines and have similar penetrations through the Containment. There are dual check valves in each line, one inside and one outside the drywell. Locked open manual isolation valves for each branch are located within the drywell downstream of the second check valve. There is a one inch drain to the RBEDT downstream of the first check valve in each branch. Tandem manual valves in each drain line are provided outside the drywell penetration.

The 18 inch branches divide into two 10 inch lines each of which penetrate the vessel at the feedwater nozzles and supply water to the sparger.

The feedwater piping was designed to ASA B31.1 except for the piping indicated on Table 5.1-1..

5.4.10 Safety Valves

The safety valves serve as a complementary means of pressure relief for the reactor vessel. This system has no function during normal operation. The pressure relief system was designed in accordance with the ASME B&PV Code, Section I (1962 edition). Under the provisions of Section I, the safety valves must limit the rise in the reactor vessel pressure to less than the ASME Code limit. Nine safety valves were demonstrated sufficient to provide the reactor vessel overpressure protection (see Section 5.2.2.4.2.1).

The valves are spring released and flange mounted on the main steam lines to permit periodic removal for testing. Five valves are mounted on the north main steam header and four on the south main header. During 13R, seven safety valves were removed and replaced with blind flanges. The setpoints for the nine safety valves are delineated on Figure 5.4-7 and are listed in the Technical Specifications.

5.4.11 Head Cooling System

5.4.11.1 Design Basis

The Head Cooling System is used in conjunction with reactor vessel flooding and the Shutdown Cooling System, for condensing steam formed in the vessel head and for cooling the flanges and the upper portions of the reactor pressure vessel during shutdown operation.

The system is designed to meet the following objectives (with associated vessel flooding and Shutdown Cooling System operation):

a. Condense steam and condensible gases in the vessel dome to assist in vessel head cooling during shutdown.

b. Prevent repressurization as the vessel is flooded to levels above the vessel flange and main steam nozzles to cool the upper portions of the vessel metal.

c. Provide vessel head cooling under the direct control of the Control Room operator during shutdown, after local valve settings have been completed.

d. Permit reactor pressure to be reduced to atmospheric, and vessel head temperature to be reduced to approximately 140°F without causing metal temperature differentials which would affect the integrity of the reactor vessel during its designed lifetime.

5.4.11.2 Description

The Head Cooling System consists of a single fog spray nozzle located inside the top of the reactor pressure vessel head, which sprays a maximum of 170 gpm through a cone angle of 70 degrees. (The spray does not strike the head metal surface.) The head spray water is supplied from the Condensate Storage Tank by the standby Control Rod Drive (CRD) Hydraulic System pump. Head spray flow is measured by a flow element, indicated in the Control Room, and controlled by a pneumatically operated flow control valve.

The Head Cooling System is connected to the vessel head nozzle by a removable 2 inch stainless steel pipe spool piece. A check valve is installed as the isolation valve inside the drywell. The isolation valve outside the drywell is an air operated globe valve which is remotely controlled from the Control Room and which will close automatically from an isolation signal. There are manually operated stop valves in the head cooling system connections to the CRD Hydraulic System pumps. A leak-off line is used for system drainage to the Reactor Building equipment drain sump, and to observe for leakage through the isolation of stop valves.

Reactor vessel head temperatures are measured at the outer diameter of the head flange and at the outside surface of the hemispherical head above the transition weld.

TABLE 5.4-1
(Sheet 3 of 3)

DESIGN CHARACTERISTICS - CLEANUP SYSTEM COMPONENTS

Cleanup Pumps (2)

Type	Centrifugal
Flow	420 gpm, each pump -- two running in parallel
Developed Head	2650 ft at 120°F
Design Pressure	Discharge **1670** psig, suction 150 psig
Design Temperature	**140**°F
Motor	400 hp, 440 volt, 3 phase, 60 cycle

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 18 inch air operated butterfly 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 12 inch valves (V-28-17 and V-28-18). When the exhaust is vented to the stack without gas treatment, the drywell flow is through 18 inch 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 Hydrogen 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).

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.

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-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	

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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-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-14-115**	**NONE**	**IN**	**CHECK**	**NONE**	**REV**	**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

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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	≤60	AIR	DIRECT	CL/OP	OPEN	CLOSE	CLOSE	30 DEG MAX OPEN
X-19	18	VENTILATION EXHAUST	6.2-54	AIR	V-27-002	1,6,7,9	OUT	BFLY	AIR	AUTO	REM MAN	≤60	AIR	DIRECT	CL/OP	OPEN	CLOSE	CLOSE	30 DEG 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-5-677**	**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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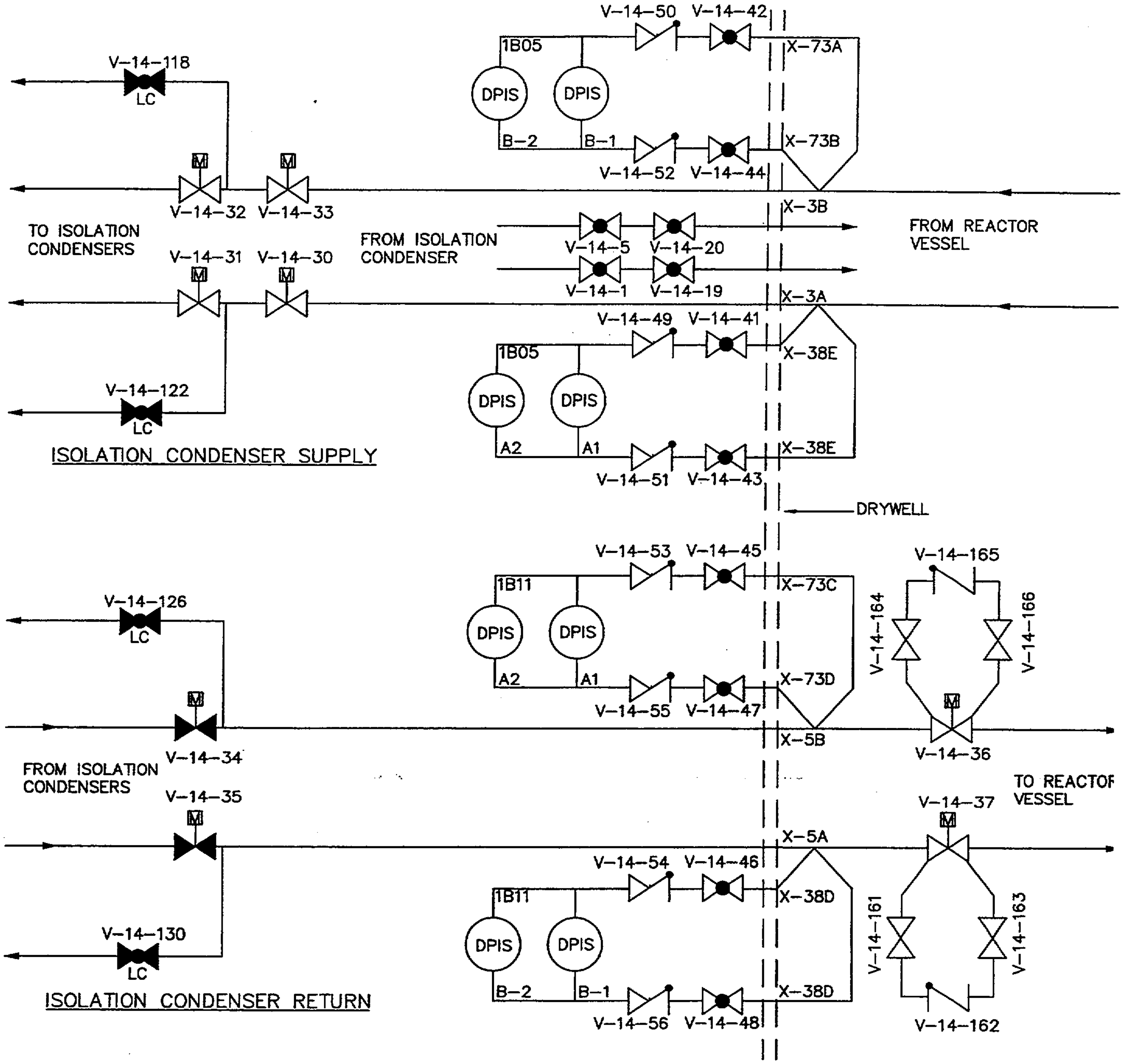
CONTAINMENT ISOLATION VALVES / MECHANICAL INTEGRITY

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-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	AS IS	KEYLOCK CLOSED
X-71	**3/8"**	**RECIRC LOOP SAMPLE LINE**	**6.2.-39**	**WATER**	**V-40-187**	**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
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	N/A N/A	N/A N/A	NONE	OPEN	OPEN	OPEN	N/A 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
Sheet 24 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-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



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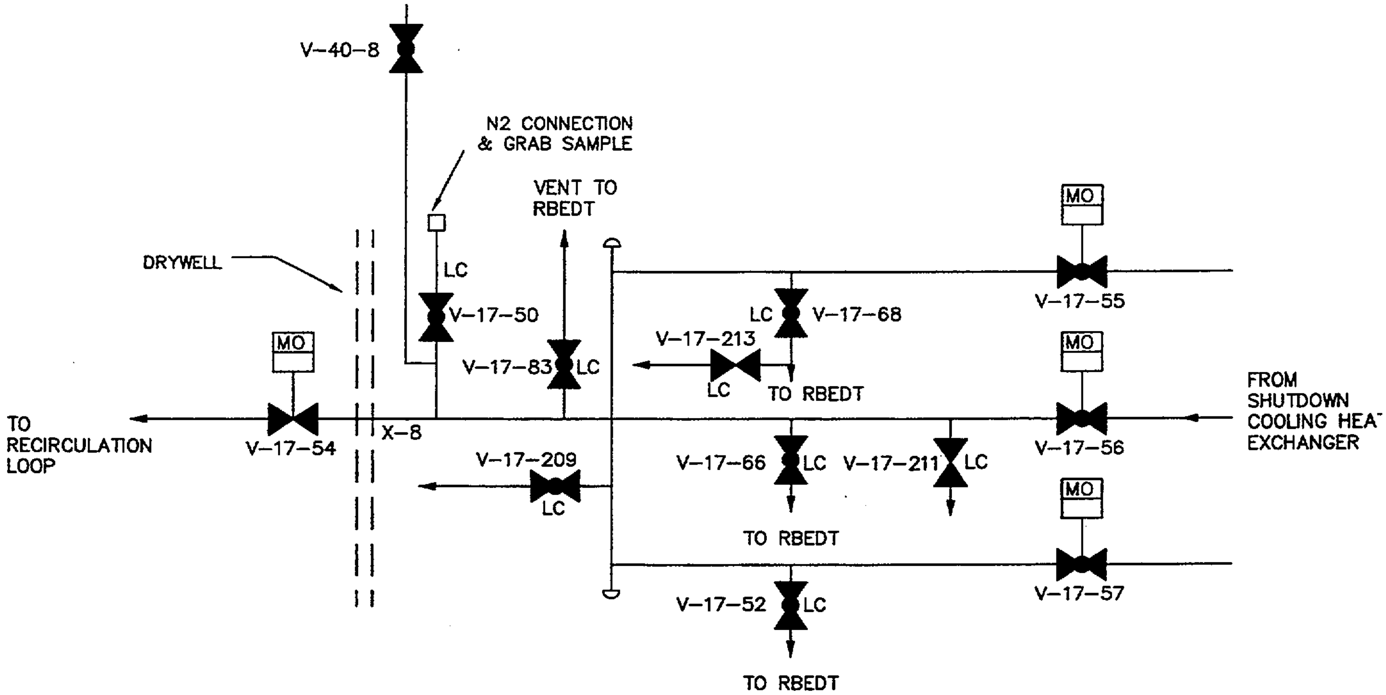
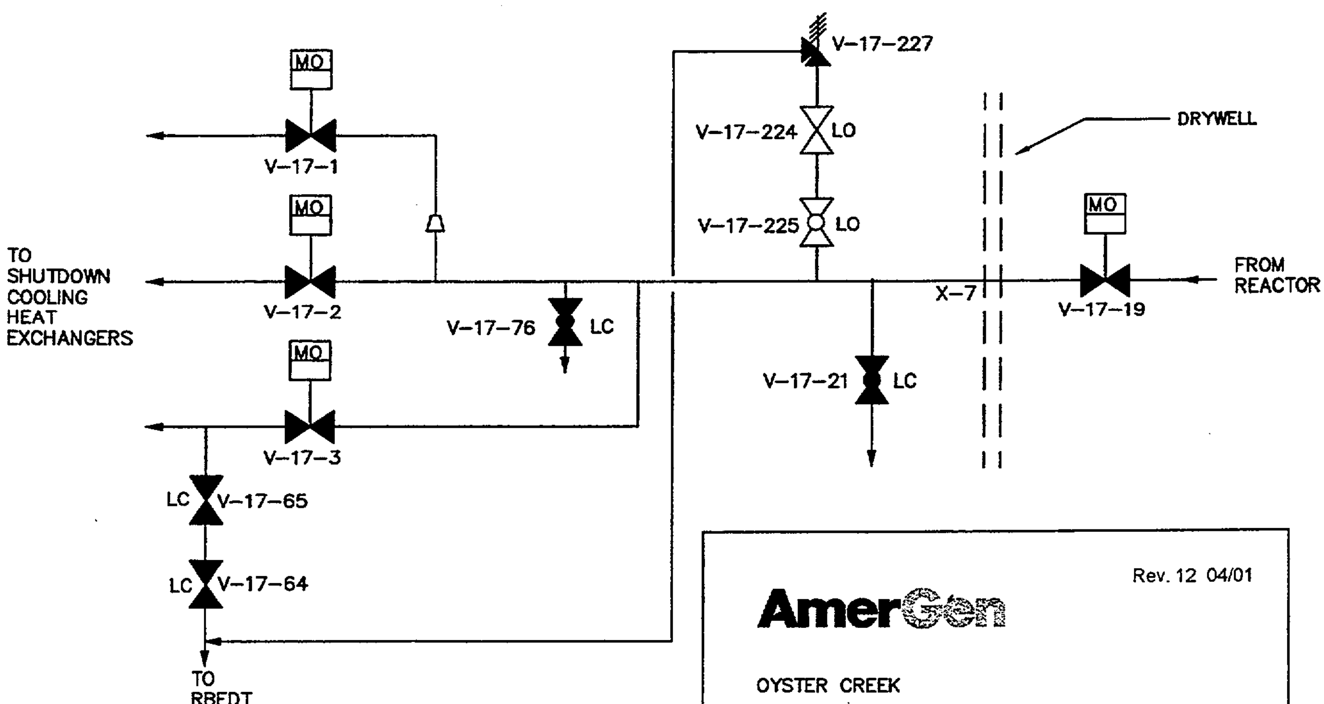
Containment Isolation Valves
Isolation Condenser

148F262

Fig. 6.2-39



SHUTDOWN COOLING RETURN



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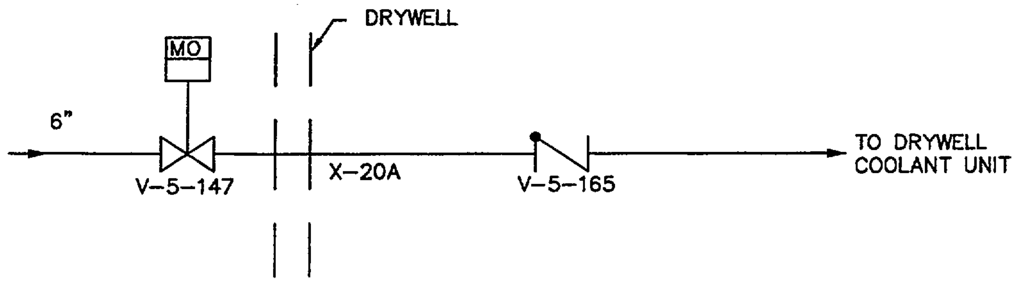
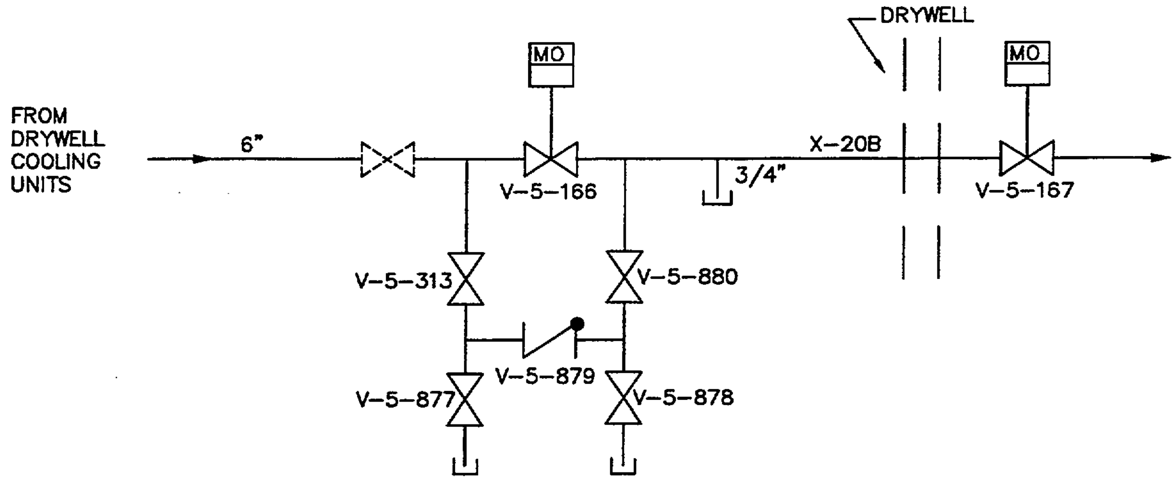
Containment Isolation Valves
Shutdown Cooling

148F711

Fig. 6.2-42



RB CLOSED COOLING SUPPLY



RB CLOSED COOLING RETURN

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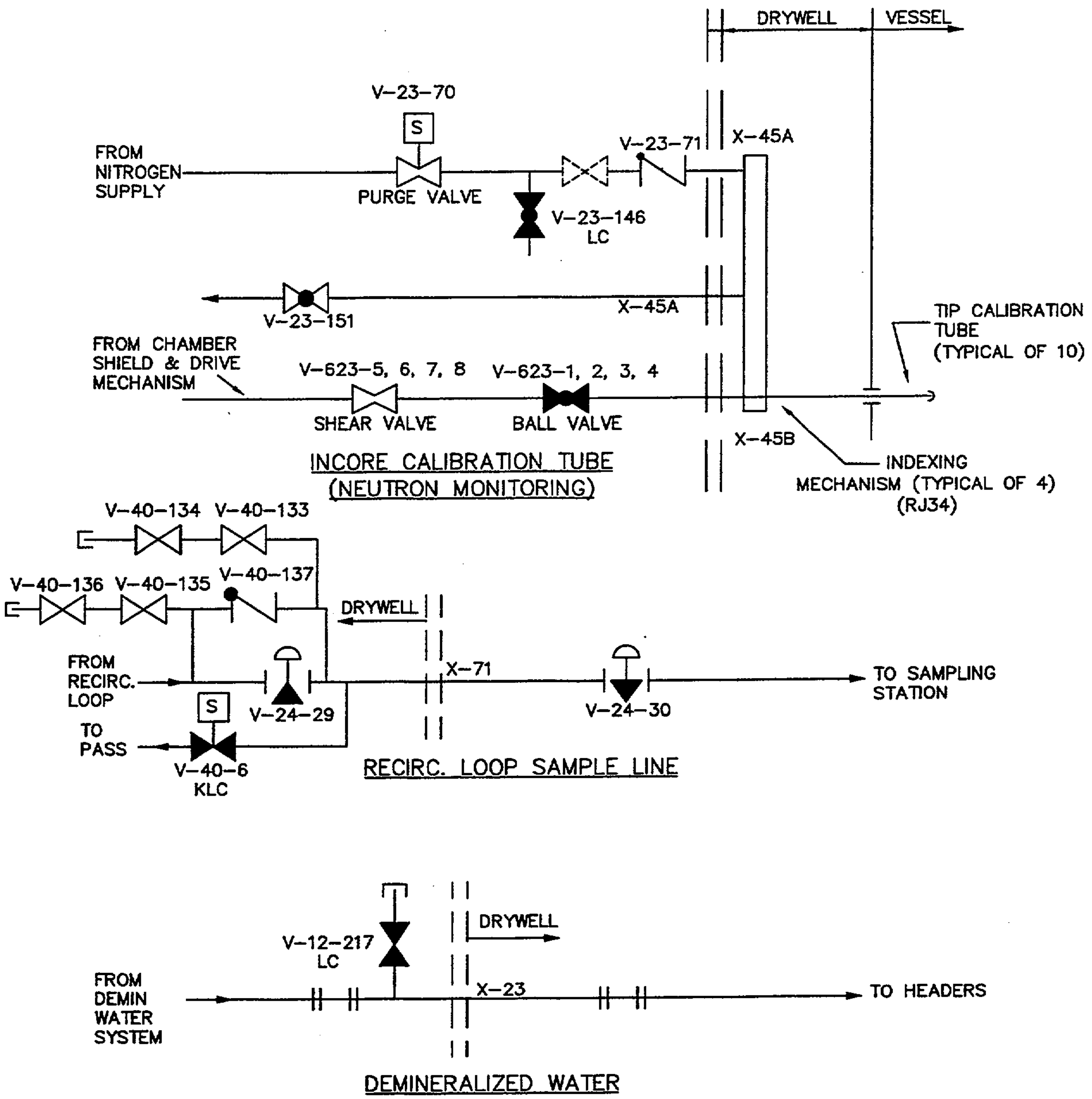
Containment Isolation Valves
RBCCW

2006 SH 3

Fig. 6.2-46



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Rev. 12 04/01

OYSTER CREEK

Containment Isolation Valves
Neutron Monitoring/Recirc Loop Sample/Demin. Water

148F712 2004 SH2 | SN 13432, 19-1 M0012, 2002 SH2

Fig. 6.2-50

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 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. 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 212°F. 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 212°F. 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 212°F. 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.

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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.

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 STOKE TIME (In Seconds)*** UNER 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

* The AC condensate valves are maintained at no more than 80% open to ensure the stroke times listed above

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\+ < 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: 81.5 seconds, Complete Isolation Time Includes:

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
50.5 seconds	Maximum stroke time for MOV

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 6.3-4
(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

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.

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7.2.1.1.3 Bypassing

Bypassing of trip functions occurs automatically under certain plant conditions. However, certain trip functions can be manually bypassed. Trip functions and permitted bypasses are listed in Table 7.2-2.

7.2.1.1.4 Interlocks

Interlocks are provided in the Neutron Monitoring System (NMS) trip circuitry to trip an RPS subchannel whenever an NMS module is INOP, i.e., removed from its mounting or not in operate mode, or when the required minimum number of inputs to an APRM channel are not present.

Bypassing of NMS channels (where permitted) is accomplished by joysticks which mechanically and electrically prevent bypassing more than one channel at a time per logic channel or core quadrant.

7.2.1.1.5 Redundancy

All functions of the RPS are implemented by redundant sensors, logic, and actuation devices.

7.2.1.1.6 Diversity

The RPS trip circuits have been designed with diverse trip functions to protect the integrity of the fuel cladding and Reactor Coolant System barriers.

7.2.1.1.7 Actuated Devices

Reactor trip is accomplished through two scram valves on each control rod drive. (See Figure 7.2-2.) The valves are air to close, spring to open. The instrument air header normally supplies air to the valves through the energized scram pilot solenoid valves to hold the valves shut. During a trip the scram pilot solenoid valves deenergize and vent the air lines causing the scram valves to open. One valve opens faster, venting the over piston area to the discharge volume. A second valve opens later, applying accumulator pressure to the under piston area, inserting the control rod.

Two backup scram solenoid pilot valves are provided. Upon receipt of a trip signal these valves block the air supply and go to the vent position. Because of the long bleeddown, these valves cannot adequately cause a trip by themselves. However, if a scram pilot or discharge volume pilot failed to vent, the backup valves would insure that all scram valves open and the vent and drain valves close.

Two discharge volume pilot solenoid valves are deenergized simultaneously with the backup scram solenoid pilot valves. These valves vent the actuators on the drain and vent valves, closing them.

7.2.1.1.8 Mode Selector Switch

There are four principal modes of operation of the Mode Selector Switch: RUN, STARTUP, REFUEL and SHUTDOWN. Each operating mode has its own individual restrictions for safe operation. The Mode Selector Switch is provided to ensure that all these restrictions are imposed at the proper time, and to ensure that the transition from one mode to another is also safe. Table 7.2-3 lists the modes and the permissible operations in each mode if all RPS parameters are within established limits.

7.2.1.1.9 Information Display

There are no analog displays in the RPS since it is a relay logic system.

Displays associated with the Neutron Monitoring System are discussed in Section 7.5.

Local indication associated with reactor vessel instrumentation is discussed in Section 7.6. Reactor Protection System annunciators are provided in the Control Room as shown in Figure 7.2-3. These annunciators keep the operator informed of the status of the Reactor Protection System at all times.

7.2.1.1.10 Power Supplies

Redundant Class IE protection provided for the RPS power supply. The RPS power is supplied through two independent buses (Protection System Panels No. 1 and 2). Each panel supplies power to one logic channel and its respective pilot and backup scram valve solenoids, one half of the incore flux amplifiers, one half of the steam line radiation monitors, and one half of the flux amplifier summers.

The normal power supply to Protection System Panel No. 1 (No. 2) is from 4160 volt bus 1A(1B) to 4160 volt emergency bus 1C(1D), then through a transformer to 460 volt substation 1A2(1B2) to M-G set 1-1 (1-2) which supplies 120 volt single phase power to Protection System Panel No. 1 (2). See Table 7.2-4 for M-G set specifications. The motor generator set is equipped with a flywheel to provide inertial smoothing of switching transients upstream of the motor.

B31.1, for the Reactor Coolant System piping. The ASME Code permits pressure transients up to 10% over the design pressure (110% x 1250 = 1375 psig) and the ANSI Code permits pressure transients up to 15% over the design pressure (115% x 1200 = 1380 psig).

Trips and trip setpoints are established to provide the necessary protection so that MCPR and pressure limits are not exceeded. Subsection 7.2.1.1.1 provides a description of each RPS trip.

b. The generating station variables that provide reactor trips are shown in Table 7.2-1.

c. The requirements for the minimum number and location of sensors required to monitor adequately, for protective function purposes, those variables that have spatial dependence are:

1. Failure of four chambers assigned to any one APRM channel shall make the APRM channel inoperable.

1. Failure of two chambers from one radial core location in any one APRM channel shall make that APRM channel inoperable.

3. Any two Local Power Range Monitor (LPRM) assemblies which are input to the APRM System and are separated in distance by less than three times the control rod pitch may not contain a combination of more than three inoperable detectors (i.e., APRM channel failed or bypassed, or LPRM detectors failed or bypassed) out of the four detectors located in either the A and B, or the C and D levels.

4. A Traversing Incore Probe (TIP) chamber may be used as an APRM detector input to meet the criteria of c.1, c.2 or c.3 provided the TIP is positioned in close proximity to one of the failed LPRM's. If the criteria of c.2 or c.3 cannot be met, power operation may continue at up to rated power level provided a control rod withdrawal block is operating, or at power levels less than 61% of rated power until the TIP can be connected, positioned and satisfactorily tested, as long as Technical Specifications are satisfied.

d. For the prudent operational limits for each variable in each operation, refer to the Technical Specifications.

d. For the margin between each operational limit and level marking onset of unsafe conditions, refer to the Technical Specifications.

e. For the level that, when reached, will require protective action, refer to Table 7.2-1 (approximate values) and to Standing Order No. 1 (exact values).

f. The range of transient and steady state conditions of the energy supply and the environment during normal, abnormal, and accident circumstances throughout which the system must perform has been determined, as discussed in Section 3.11.

g. The malfunctions, accidents, or other unusual events that could physically damage protection system components, for which provisions must be incorporated to retain necessary protection system action are as follows:

   1. All instruments used in the protection system and engineered safeguards are designed to operate under the most unfavorable environmental conditions that can reasonably be associated with an abnormal or accident condition. For conventional devices, the manufacturer's quality control, initial calibration, plant preoperational testing and plant maintenance procedures are relied upon to obtain a high degree of reliability.

   2. All sensing elements, except neutron detectors, are located outside the drywell, where they are not exposed to unfavorable environment. They are equipped with weatherproof enclosures which can withstand, at least on a temporary basis, such conditions as might result from a steam or water line break outside the drywell. In addition, the sensors are located in such a way that no single event is likely to affect both sensors in one protection logic system. The scram discharge volumes high level switches and transmitters may be considered an exception to this general separation statement. As these detectors are called upon to operate following all scrams, operations are individually recorded. These switches and transmitters are built to withstand full system pressure and temperature, and are intended primarily to prevent startup of reactor prior to draining the scram discharge system for the control rod drive. Since the discharge system is normally open to drain, and position switches provided on valves, isolation of this system requires multiple failures and complete disregard by the operator to create any possible condition which could be construed to

be dangerous. The reference column with the auxiliary reservoir has been subjected to carefully controlled and documented tests to prove that the reference column remains full under blowdown conditions similar to those postulated for a Loss-of-Coolant Accident in the reactor vessel.

i. The minimum performance requirements, including system response times, ranges of the magnitudes, and rates of change of sensed variables to be accommodated until proper conclusion of the protection system action, are presented in the Technical Specifications.

7.2.1.3 Final System Drawings

The RPS logic is shown in Drawing GE237E566

7.2.2 Analysis

The RPS is made up of two independent logic channels, each having two subchannels of tripping devices. Thus, the system has four independent subchannels. Each subchannel has inputs from independent sensors monitoring each of the critical parameters. One of the parameters monitored by each subchannel sensor, any one off-standard or out-of-limits condition which initiates a sensor trip will de-energize the subchannel scram relay (i.e., relays 1K51, 1K52, 2K51, or 2K52 on **Drawing GE237E566**), which will in turn de-energize one half of the pilot scram valve solenoids (one on each rod).

This condition, with one half of the pilot scram valve solenoids de-energized, is known as a 1/2 scram. Two logic channel trips are required to produce a reactor trip; therefore, the minimum requirement for a trip is two subchannel trips (one per logic channel). With the one-out-of-two-twice logic of the Reactor Protection System, a single component failure, or the spurious trip of a single sensor, will cause a logic channel trip but will not cause a reactor trip. In addition this logic will not prevent a reactor trip when a second sensor trip occurs in the remaining logic channel. Thus, one can tolerate a single component failure in the RPS and still operate the reactor safely.

Exceptions to the above logic are discussed in Section 7.2.1.1.2

The entire system, except for the main condenser low vacuum scram can be tested during plant operation. Theoretically, the one-out-of-two-twice logic of the RPS is slightly more reliable than a two-out-of-three and slightly less reliable than a one-out-of-two system. The dual-logic-channel protection system facilitates more testing during full power operation in comparison to the one-out-of-two system. The thorough and frequent testing significantly increases the reliability of the system.

The RPS is designed to "fail safe" for the most probable failure mode. In every case except sensor failure, a safe failure is annunciated so that the location of the failure can be readily ascertained.

The term "fail-safe" as applied to the RPS refers to the condition of the system component in its failure mode. For example, the RPS relays are normally energized when the plant is operating within the established safe limits; if a relay coil were to fail or a subchannel were to lose power, the relay or relays would be deenergized and their open contacts would cause a logic-channel trip. Therefore, when an RPS component fails, it is designed to fail into a safe, or trip mode.

7.2.2.1 Conformance with Section 4 of IEEE-279

OCNGS was built and operational prior to the issuance of IEEE 279. The following discussions are keyed to Section 4 of IEEE 279 and provide a comparison of Oyster Creek RPS design with this standard:

(4.1) General Functional Requirements

The RPS automatically performs its protective function of tripping the reactor whenever plant conditions exceed preset levels under the design conditions discussed in Subsection 7.2.1.1.

(4.2) Single Failure Criteria

No single failure can prevent the RPS from performing its protective function. However, the condenser low vacuum scram does not meet this criteria if only 1 trip system is operable.

This is permitted by the tech specs since the low condenser vacuum trip is not required for safety or for reactor protection. See Section 7.2.1.1.2 for further discussion of this logic.

(4.3) Quality of Components and Modules

Materials used in the control and instrumentation for the Reactor Protection System, and other engineered safeguards (with the exception of insulating material, protective coatings, etc.), are noncombustible or highly fire resistant. The panel wiring and phenolic moulded parts of relays and other components are fire resistant.

TABLE 7.2-1
(Sheet 3 of 4)
REACTOR PROTECTION SYSTEM DATA SHEET

	Parameter	Sensor Number	Actuated Relays (See Footnote 4) I	II	III	IV	V	Trip Pt. (Footnote 3)	Automatic Action
9.	Turbine Trip- Generator Load Rejection or Turbine Stop Valve Closure		Refer to Table 7.7-2						
10.	Main Steam Line Isolation Valves Closure	NS03A, NS04A NS03A, NS04A NS03B, NS04B NS03B, NS04B	1K17 1K18 2K17 2K18	1K51 1K52 **2K51** **2K52**				Limit switches set To trip with valve Closing at 10% off Full open position.	Trips Subchannel 1A. Trips Subchannel 1B. Trips Subchannel 2A. Trips Subchannel 2B. Scram is bypassed by contacts of 1K112 A&B & 2K112 A&B (see condenser low vacuum)
11.	Reactor Manual Scram	Button 1S2	1K21A						Trips Channel I Pilot Scram Valve Solenoids
			1K21B					When both buttons are Pushed simultaneously.	Seal out relay. Reset with reset button 3S1.
			1K21C						Trips Channel I Scram discharge Pilot Valve Solenoids
		Button 2S2	2K21A					When both buttons are Pushed simultaneously.	Trips Channel II Pilot Scram Valve Solenoids.
			2K21B						Seal out relay. Reset with Reset button 3S1.
			2K21C						Trips Channel II Scram discharge Pilot Valve Solenoids
12.	Loss of A.C. Power to Reactor Protection System								All relays and scram solenoid Power to Reactor valaves will trip due to loss of power, as protection system MG sets coast down

TABLE 7.2-1
(Sheet 4 of 4)
REACTOR PROTECTION SYSTEM DATA SHEET

	Parameter	Sensor Number	Actuated Relays (See Footnote 4) I	II	III	IV	V	Trip Pt. (Footnote 3)	Automatic Action
13.	Reactor Mode Switch	Switch 1S1	1K21 A&B					"Shutdown"	Trips Channel I Pilot Scram Valve Solenoids Position

FOOTNOTES:
1. All trips bypassed when the mode switch is in "refuel" or "shutdown" and the Instrument Volumes bypass selector switch (3S4-1) is in "bypass".
2. All trips bypassed by contacts of relays 1K112 A&B and 2K112 A&B when mode switch is in "refuel" or "startup" and reactor pressure less than 600 psi.
5. **All trips bypassed when Mode Switch is in "Shutdown" following a time delay of 20 seconds.**

ANNUNCIATOR WINDOW LOCATION AND DESCRIPTION

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	FUEL POOL	LIQUID POISON SYSTEM	REACTOR			
			RPS	NEUTRON MONITORS		
1		FLOW ON	SCRAM CONTACTOR OPEN	CHANNEL I	IRM HI-HI/INOP I	APRM HI-HI/INOP I
				AND	AND	AND
2	GATES LEAK HI	SQUIB VALVE OPEN	RPS MG SET 1 TRIP	CHANNEL II	IRM HI-HI/INOP II	APRM HI-HI/INOP II
3	REFUEL SEAL LEAK HI		RPS MG SET 2 TRIP	SRM HI-HI	IRM HI	APRM HI
4	POOL LEVEL/TEMP HI	TANK LEVEL HI/LO	RPS 600 #/SD BYPASS	SRM HI-/INOP	IRM DNSCL	APRM DNSCL
5	POOL LEVEL LO			SRM DNSCL		APRM FLO BIAS OFF NORMAL
6	SKM SRG TNK LVL LOW		RPV-FLANGE ΔT - HI			LPRM HI
7	SKM SRG TNK LVL LO-LO		**RPS ISOLATION CI RxI I**	SRM PERIOD SHORT		LPRM DNSCL
8		TANK TEMP HI/LO	**RPS ISOLATION CI RxI II**		TIP PURGE PRESS HI/LO	TIP SQUIB CONTINUITY
	a	b	c	d	e	f

NOTES:
1. INPUT CONTACTS OPEN TO ALARM.
2. REFER TO ELECTRICAL CONNECTION DIAGRAM DWG. NO. 3E-661-18-024 SH. 1 FOR OTHER COMMON TERM. NO'S.

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OYSTER CREEK NUCLEAR GENERATING STATION

Reactor Trip System Alarms

Figure 7.2-3A

ANNUNCIATOR WINDOW LOCATION AND DESCRIPTION

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	a	b	c	d	e	f
	C O N T R O L R O D S / D R I V E S			H	R E A C T O R	
1	ROD CONTROL	SDV	HYDR	DW PRESS	LEVEL	PRESS
2	CONTROL AIR PRESS LO	SDV LEVEL HI-HI I AND	PUMP A OL	DW PRESS HI-HI CI I AND	CCW RX LVL LO-LO-LO I OR	RX PRESS HI-HI E I AND
3	**ARI INITIATED**	SDV LEVEL HI-HI II	PUMP B OL	DW PRESS HI-HI CI II	CCW RX LVL LO-LO-LO II	RX PRESS HI-HI E II
4	**ARI OFF NORMAL**	NORTH SDV LEVEL HI ROD BLOCK	SUCT PRESS LO PUMP TRIP		RXI RX LVL LO-LO CI I AND	RX PRESS HI-HI
5		SOUTH SDV LEVEL HI ROD BLOCK	FILTER Δ P HI		RX LVL LO-LO CI II	
6	ROD OVERTRAVEL	SDV NOT DRAINED	CRD TEMP HI	**ROPS ACTUATE A** **AND**	RX LVL LO **E** I	**RX LVL HI** II **AND I**
7	ROD DRIFT	SDV LEVEL HI-HI SCRAM BYPASS		**ROPS ACTUATE B**	RX LVL LO **E** II	**RX LVL HI** II **II**
8	ROD BLOCK		CHARG WTR PRESS LO	**ROPS BYPASSED**	RX LVL HI/LO	
9			ACCUMULATOR PRESS LO/ LEVEL HI		**RXL VL/PRESS INSTR CHANL TEST**	**RXL VL/PRESS INSTR PWR LOST**

NOTES:
1. INPUT CONTACTS OPEN TO ALARM.
2. REFER TO ELECTRICAL CONNECTION DIAGRAM DWG. NO. 3E-611-18-024 SH. 1 FOR OTHER COMMON TERM. NO'S.
3. CONNECT DROP #5-C ALARM CKT. TO ALARM ON SIGNAL CONTACT CLOSURE.

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OYSTER CREEK NUCLEAR GENERATING STATION

Reactor Trip System Alarms

Figure 7.2-3B

Oyster Creek Nuclear Generating Station
FSAR Update

TABLE 7.3-1
(Sheet 1 of 4)

REACTOR PROTECTION SYSTEM ENGINEERED SAFETY FEATURE SYSTEMS ACTUATION

	Parameter	Sensor Number	Actuated Relays (Footnote 1) I	II	III	IV	Trip Pt. (Footnote 2)	Action
	Reactor Vessel (Main Steam) Isolation							
1.	DELETED							
2.	Steam Line High Flow	RE22A, E	1K15	1K73			120% rated flow	Also closed main steam line drain isolation valve, off-gas exhaust valve V7-31 and holdup drain Valve V7-29.
		RE22B, F	1K16	1K74				
		RE22C, **G**	2K15	2K73				
		RE22**D**, H	2K16	2K74				
3.	Trunnion Room High Temperature	1B10A, E, J, N	1K15	1K73			178°F	
		1B10B, F, K, P	1K16	1K74				
		1B10C, G, L, Q	2K15	2K73				
		1B10D, H, M, R	2K16	2K74				
4.	Main Steam Line Low Pressure	RE23A	1K117	1K73			825 psig	Same action as Steam Line High Flow, Item 2.
		RE23C	1K118	1K74				
		RE23B	2K117	2K73				
		RE23D	2K118	2K74				

Footnote 1: The actuated relays subcolumns give the order of relay actuation, i.e. sensor operates the relay in Subcolumn I, relay contacts in Subcolumn I actuate the relay in Subcolumn II, etc.

Footnote 2: Nonimal setpoints shown. Refer to Standing Order #1 for actual setpoints.

TABLE 7.3-1
(Sheet 2 of 4)

REACTOR PROTECTION SYSTEM
ENGINEERED SAFETY FEATURE SYSTEMS ACTUATION

	Parameter	Sensor Number	I	II	III	IV	Trip Pt.(Footnote 2)	Action
	Reactor Vessel (Main Steam) Isolation (Continued)							
5.	Reactor Low-Low Water Level	RE02A RE02C RE02B RE02D	1K19 1K20 2K19 2K20	1K71 1K72 2K71 2K72	1K73 1K74 2K73 2K74	1K75 and 1K77, 1K76 and 1K77, 2K75 and 2K77, 2K76 and 2K77	Decreasing Reactor water level at 86" above the top of the active fuel.	a. 1K71, 1K72, 2K71 and 2K72-for action, see drywell high pressure, Item 7. b. 1K73, 1K74, 2K73 and 2K74 - for action, see main steam line flow. c. 1K75, 1K76, 2K75, 2K76 - cause cleanup and shutdown system trip and isolation from the reactor. d. 1K77 and 2K77 cause recirculation pumps to trip.
	Primary Containment (Drywell) Isolation							
6.	Reactor Low-Low Water Level	See Item 5, above						
7.	Drywell High Pressure	RE04A RE04C RE04B RE04D	1K9 1K10 2K9 2K10		1K71 1K72 2K71 2K72		≤3.5 psig	Trips Subchannel 1A. Trips Subchannel 1B Trips Subchannel 2A. Trips Subchannel 2B.

NOTES: **1) Some valves will isolate on containment high radiation (See text)**
2) See Sheet 3 for RBCCW System Isolation as isolating circuits are different than what is shown above

Footnote 2: Either nominal setpoints or process limits, or Technical Specification limits are shown. Refer to Standing Order #1 for actual setpoints.

Major Components

The Detector Drive System consists of a motor module, a detector drive assembly and a dry tube. The following paragraphs provide a brief description of each of the system components.

a. Motor Module

The module consists of a 208 volt, 3 phase, 60 cycle fractional HP ac motor, a drive shaft, and a limit switch, all mounted on an aluminum base plate. The limit switch, which is chain linked to the motor output shaft, controls the insertion limits (FULL-IN and FULL-OUT) of the detector; the switch also provides outputs to auxiliary relays for indicators in the Control Room.

A flexible drive shaft, connected to the output shaft of the motor module, transmits power to the gear box of the detector drive assembly.

The motor module is located outside the reactor support structure for three reasons; to avoid the crowded conditions in the structure, to keep the electrical equipment out of an environment which is usually damp, and to reduce the radiation exposure to the equipment.

Detector Drive Assembly

This assembly consists of the gear box, the housing, the drive tube and the shuttle tube. The gear box is mounted on the housing and drives the drive tube. The drive tube is a round tube with holes along one side. The shuttle tube is a long, slender tube assembly mounted on the end of the drive tube. The shuttle tube and drive tube move as one.

The detector assembly consists of the detector and the detector cable. The detector assembly is equipped with one of two types of detectors to measure two different ranges of neutron flux activity. The lower range detector provides an input to the source range monitor (SRM) and the higher range detector provides an input to the intermediate range monitor (IRM). The detector assembly is installed inside the protective shuttle tube and drive tube. The drive tube extends from the back of the detector drive to mate with an external connector.

b. Dry Tube

This is the tube that mates with the reactor vessel, providing a dry channel within the reactor core. The detector assembly, shuttle tube, and drive tube are driven up and down in the dry tube.

Operation and Precautions

There are three selector switches and one drive control switch on Panel 4F. Either SRM or IRM detector movement can be selected but not simultaneously. Any one detector can be moved or all SRMs or IRMs detectors can be moved together. All movement is manual. There is no automatic movement nor are there any interlocks to prevent inadvertent movement.

7.5.1.8.6 Local Power Range Monitor (LPRM) System

Functions

Selected groups of LPRMs provide input signals to the Average Power Range Monitoring System for bulk power level monitoring and automatic core protection. The system supplies signals proportional to local neutron flux to drive the indicating meters to be used for manual evaluation of core performance.

The system generates signals to annunciators which indicate high local flux or low detector reading. The system also generates the rod withdrawal block.

Major Components

Detector Assemblies

Prior to the 15R Refueling Outage there were thirty-one LPRM detector strings each containing four fission chambers are distributed so as to form four horizontal planes throughout the core. In 15R, two LPRM strings were removed from service due to interferences in the instrument guide tube thimbles. The detector assemblies are inserted into the core in spaces between the fuel bundles and through thimbles mounted permanently at the bottom of the core lattice which penetrate the bottom of the reactor vessel. These thimbles are welded to the reactor vessel at the penetration point. They extend down into the access area where they terminate in a flange which mates to the mounting flange on the incore detector assembly. **LPRM's 20-49 and 36-41 were removed during 15R and the guide tube thimbles were blanked off with blind flanges.** The detector assemblies are locked, at the top end, to the top grid of the core by means of a spring loaded plunger. This type of assembly is referred to as top entry-bottom connect, since the assembly is inserted through the top of the core and penetrates the bottom of the reactor vessel. Special water sealing caps are placed over the connection end of the assembly and over the penetration at the bottom of the vessel during installation or removal of an assembly. This prevents the loss of reactor coolant water upon removal of an assembly and also prevents the connection end of the assembly from being immersed in the water during installation or removal.

Each in-core flux detector assembly contains four miniature fission chambers with an associated solid sheath cable. Each individual chamber of the assembly is a moisture proof, pressure sealed unit. The assembly also contains the calibration tube for the traversing probe, and an enclosing tube around the entire assembly. The enclosing tube around the entire assembly contains vent holes evenly spaced along its length. These holes allow circulation of the reactor coolant to cool the fission chambers. The miniature fission chambers are essentially the same as those utilized by the SRM and IRM systems except that they are operated at a lower high voltage (100 Vdc).

b. Flux Amplifier

The flux amplifier consists of two identical dc amplifiers and two identical trip units. Thus, each amplifier module processes signals from the two LPRM detectors. The trip units are tripped under the following two conditions:

LPRM Upscale - When flux level is greater than **97 watts/cm^2.**

LPRM Downscale or Inoperative - When flux level is less than **2 watts/cm^2** or when high voltage to the detector lost, the mode switch on the amplifier drawer is not in OPERATE position, or module removed.

Power Supply and Monitor

The power supply and monitor furnishes three regulated voltages to operate the flux amplifiers and their associated detectors: +115 Vdc, +100 Vdc and +85 Vdc. Each power supply and monitor is associated with two flux amplifiers (four LPRM detectors).

The monitoring circuit on the power supply and monitor is comprised of front panel meter M1, selector switch S1, and three resistors (R4, R5 and R6). The monitoring circuit performs two functions: 1) when selector switch S1 is placed in one of the PERCENT POWER positions, it connects the meter between one of the flux amplifier outputs and the +100 volt dc bus (+100 volts dc is common in the flux amplifier), and 2) when selector switch S1 is placed in one of the SUPPLY VOLTS positions, the meter is connected between the power supply output voltage and common (in the case of the +100 volt dc output) or +100 volts dc (in the case of the ±15 volt dc outputs). Thus, the monitoring circuit provides the capability to monitor the power supply or flux amplifier outputs on the front panel meter.

LPRM Indications and Alarms

LPRM outputs are indicated on the vertical section of Panel 4F. The four meters associated with the detectors at a given radial core location are stacked together in sequence (i.e., for the A, B, C and D levels). Each meter has a range of 0-125 watts/cm^2 and is related to heat flux. There is an amber light next to each LPRM meter which illuminates if the associated channel is upscale (greater than **97** watts/cm^2) downscale (less than 2 watts/cm^2) or bypassed. An LPRM channel is bypassed by operating a toggle switch (one per channel) inside the associated trip auxiliary drawer.

Two annunciator alarms on Panel 5F are associated with the LPRMs: 1) LPRM High, **97** watts/cm^2, and 2) LPRM Downscale (less than 2 watts/cm^2) or Inop (loss of high voltage, mode switch not in OPERATE or module removed).

Rod Withdrawal Block Associated with LPRM's

A rod withdrawal block is generated when one of the LPRM's feeding an APRM channel indicates a downscale condition (less than 2 watts/cm^2).

7.5.1.8.7 Average Power Range Monitor (APRM) System

Function

The APRM system consists of the electronic equipment which averages the output signals from selected LPRM amplifiers, the trip units which actuate automatic protective actions when APRM signals exceed preset values and the signal readout equipment. This system provides continuous indication of average reactor power from a few percent to 150% rated power.

There are eight APRM channels - two per core quadrant. Channels 1, 2, 3 and 4 are associated with Reactor Protection System No. 1 and Channels 5, 6, 7 and 8 are associated with Reactor Protection System No. 2. Thus, each core quadrant is monitored by two APRM channels - each associated with a different Reactor Protection System. The APRM channels in a given core quadrant utilize the same LPRM detector strings with the Reactor Protection System No. 1 APRM channels receiving inputs from the A and C level detectors and Reactor Protection System No. 2 APRM channels receiving inputs from the B and D level LPRM detectors. Each APRM channel normally averages the inputs of eight LPRM channels.

TABLE 7.5-2
(Sheet 1 of 2)

REACTOR CONTROL PANEL 4F
DISPLAY INSTRUMENTATION

Displayed Variable	Type of Display*	Range	Purpose
Control Rod Drive Position	IND	00 to 48**	Used to select rods during power changes.
In-Core Flux Monitor (LPRM)	IND		Used to select rods during power changes.
Rod Worth Minimizer	IND		Used to select rods during startup.
Source Range Monitors Period	IND	-100 to ∞ to +10	
Source Range Monitors CPS	IND, REC	10^{-1} to 10^{6} CPS	
IRM-APRM Neutron Flux	REC	0-40% 0-150% power	Indicates reactor power during operation.
Total Recirculation Flow	IND	0-200,000 gpm	Used to vary reactor power over limited range.
Total Steam Flow	IND	0 to $8x10^{6}$ PPH	
Reactor Pressure	IND	970 to 1070 psig	

* IND = INDICATOR; REC = RECORDER

** 00 indicates normal full-in position.
48 indicates normal full-out position.
Even numbers indicate latched positions.

TABLE 7.5-2
(Sheet 2 of 2)

REACTOR CONTROL PANEL 4F
DISPLAY INSTRUMENTATION

Displayed Variable	Type of Display*	Range	Purpose
Control Rod Drive WaterDifferential Pressure	IND	0-400 psi	Used to determine status of control rod drive.
Reactor Core Heat Flux	IND	0-125%	
Liquid Poison System Parameters (Pump DischargePressure/Tank Level)	IND	0-1500 psig 0-5000 gallons	Monitor status of SLCS.
Control Rod Drive CoolingWater Differential Pressure	IND	0-75 psi	Monitor status of control rod drive.
Control Rod Drive Cooling Water Flow	IND	0-50 gpm	Monitor status of control rod drive.
Control Rod Drive Water Flow	IND	0-5 gpm	Monitor status of control rod drive.
Control Rod Drive Flow to Reactor	IND	0-50 gpm	Monitor status of control rod drive.
Control Rod Drive Charging Water Header Pressure	IND	0-2000 psig	Monitor status of control rod drive.
Squib Valves Electrical Continuity	IND	0-100V	Check circuits' electrical continuity.
Control Rod Drive Filter Flow	IND	0-100 gpm	Monitor status of control rod drive

*IND = INDICATOR; REC = RECORDER

The thermocouples are copper constantan and, glass insulated and clad with stainless steel. The thermocouples are clamped under pads welded to the RPV.

Also, recirculation loop water temperature detection is provided. Recirculation loop suction temperatures are indicated on two digital indicators on Panel 3F. An interlock is provided to prevent the operation of the Shutdown Cooling System if the water temperature in any one of the five recirculation loops exceeds 350°F.

7.6.1.1.4 Flow Instruments

Flow instruments are summarized in Table 7.6-4. Flow is measured by sensing the differential pressure produced across a flow element installed in the piping. This differential pressure is proportional to the square of the flow. Differential pressure transmitters measuring flow normally produce a current signal proportional to the differential pressure generated by the flow element. Some flow transmitters perform a square root function internally, thus their output signal is proportional to the flow rate.

7.6.1.1.5 Water Quality Instrumentation

No water quality instrumentation is provided on the RPV. A chemical sample sink is located on RB El. 75' with a secondary station on RB El. 51'. A chemical sample sink is located on El. 51' of the Reactor Building. This sample station provides a line for sampling the recirculation loop and the Reactor Water Cleanup System (RWCUS).

> Reactor water quality measurement is provided by three conductivity instruments provided in the RWCUS. A conductivity recorder and alarm are installed on Panel 3F. True reactor water conductivity is indicated by two pre-filter conductivity cells which have a range of 0-10 micromhos.

A third conductivity cell indicates the post-demineralizer water quality and has a range of 0-1 micromhos.

Alarm set points for "hi conductivity" and "hi-hi conductivity" are 1 micromho and 2 micromhos, respectively.

7.6.1.1.6 Core Differential Pressure Transmitter

Core differential pressure is the pressure drop across the core support plate (pressure drop across the active core is small compared to core plate differential). Differential pressure across the core plate utilizes nozzle N12 (see Figure 7.6-2) which has a pipe within a pipe. The outer pipe detects pressure above the core plate (low pressure leg) and the inner pipe, which is also the liquid poison line, detects pressure below the core plate (high pressure leg).

The differential pressure transmitter (PT-IA07) is located on RK04 and is connected to a recorder on panel 3F. The recorder has a range of 0-30 psid.

7.6.1.1.7 Core Spray Differential Instruments

Core spray differential pressure transmitter installation is shown in Figure 7.6-2. Core Spray differential pressure is measured to check the integrity of the core spray piping between the shroud and the vessel wall. It measures the differential pressure between the top of the core support plate and the Core Spray Sparger. This pressure should be independent of recirculation flow and should be essentially zero (pressure drop across the core from interchannel leakage only). Should a significant break develop in the core spray pipe between the shroud and top of the core support plate and the annulus outside the shroud. That differential pressure would be about 8-10 psid greater, due to the pressure drop across the steam separators and dryers.

7.6.1.1.8 Relief Valve/Safety Valve Acoustical Monitoring System (VMS)

The VMS monitors the closed/not closed status of the 9 safety valves and 5 Electromatic Relief Valves. Seven of the nine safety valves have two operational acoustic monitors. The system was supplied by Babcock & Wilcox. Each of the 14 channels consists of a piezoelectric accelerometer attached to the valve discharge piping, a preamplifier located in the drywell, and signal conditioning electronics located on Panel 15R in the Control Room.

Sound in the frequency range above 2 khz is detected by the accelerometer acting as a microphone. This sound is produced by fluid flowing through the discharge piping which occurs if the valve is open or leaking. The rms (Root-Mean-Square) acceleration value of the acoustic signal is displayed on Panel 15R and the valve analog indicators on Panel 1F/2F. An alarm on Panel 1F/2F is actuated (after a time **LAG with an approximate 2.2 second time constant. The time delay is dependent on the magnitude of the signal**) indicating that a valve has opened. A loudspeaker and headphones are provided on Panel 15R so that the operator can listen to the sound.

7.6.1.2 Recirculation Pump Trip System

Any one of the following functions automatically trip the Reactor Recirculation Pumps:

a. Drive motor overcurrent

b. Drive motor ground overcurrent

c. Low-low water level in the reactor vessel

d. 4160V bus undervoltage

TABLE 7.6-3
(Sheet 1 of 2)
REACTOR VESSEL THERMOCOUPLE LOCATIONS
THERMOCOUPLE PAD LOCATIONS
(REF. GE DWG. 885D731)

T.E. No.	Location	Azimuth	Elevation	
FA54 (IA01-1)	Vessel Flange	70°	54 ft. 7 in.	Note 1
FC54 (IA01-2)	Vessel Flange	230°	54 ft. 7 in.	Note 3
FD54 (IA01-3)	Vessel Flange	354°	54 ft. 7 in.	Note 3
VA50 (IA01-4)	Vessel Seam	70°	50 ft. 8 in.	Note 3
VC50 (IA01-5)	Vessel Seam	230°	50 ft. 8 in.	Note 3
VD50 (IA01-6)	Vessel Seam	354°	50 ft. 8 in.	Note 1
VA37 (IA01-7)	Vessel Below Low Water	70°	37 ft. 6 in.	Note 1
VC37 (IA01-8)	Vessel Below Low Water	230°	37 ft. 6 in.	Note 3
VD37 (IA01-9)	Vessel Below Low Water	354°	37 ft. 6 in.	Note 3
VA23 (IA01-14)	Vessel Core	90°	23 ft. 6 in.	Note 1
VC23 (IA01-15)	Vessel Core	230°	23 ft. 6 in.	Note 4
VD23 (IA01-16)	Vessel Core	354°	23 ft. 6 in.	Note 4
VA12 (IA01-17)	Vessel Downcomer	90°	12 ft. 4 in.	Note 1
VC12 (IA01-18)	Vessel Downcomer	230°	12 ft. 4 in.	Note 3
VD12 (IA01-19)	Vessel Downcomer	354°	12 ft. 4 in.	Note 4
BA04 (IA01-20)	Vessel Bott. Above Skirt Unit	90°	4 ft. 5 in.	Note 3
BC04 (IA01-21)	Vessel Bott. Above Skirt Unit	230°	4 ft. 5 in.	Note 4
BD04 (IA01-22)	Vessel Bott. Above Skirt Unit	354°	4 ft. 5 in.	Note 1
BA03 (IA01-23)	Vessel Bott. Below Skirt Unit	90°	3 ft. 1 1/4 in.	Note 3
BC03 (IA01-24)	Vessel Bott. Below Skirt Unit	230°	3 ft. 1 1/2 in.	Note 3
BD03 (IA01-25)	Vessel Bott. Below Skirt Unit	354°	3 ft. 1 1/2 in.	Note 3
SA02 (IA01-26)	Skirt Insulation	90°	2 ft. 7 1/2 in.	Note 1
SC02 (IA01-27)	Skirt Insulation	230°	2 ft. 7 1/2 in.	Note 4
SD02 (IA01-28)	Skirt Insulation	354°	2 ft. 7 1/2 in.	Note 4
SA00 (IA01-29)	Skirt Insulation	90°	0 ft. 4 in.	Note 1
SC00 (IA01-30)	Skirt Insulation	230°	0 ft. 4 in.	Note 3
SD00 (IA01-31)	Skirt Insulation	354°	0 ft. 4 in.	Note 4
BA01 (IA01-32)	Vessel Bottom	85°	1 ft. 0 in.	Note 4
BC01 (IA01-33)	Vessel Bottom	232°	1 ft. 1 1/2 in.	Note 1
BD01 (IA01-34)	Vessel Bottom	355°	1 ft. 0 in.	Note 4

TABLE 7.6-3
(Sheet 2 of 2)

REACTOR VESSEL THERMOCOUPLE LOCATIONS
THERMOCOUPLE PAD LOCATIONS
(REF. GE DWG. 885D731)

T.E. No.	Location	Azimuth	Elevation	
LC50 (IA01-35)	Lagging (Insulation) TCs		230° 50 ft. 8 in.	Note 1
LD50 (IA01-36)	Lagging (Insulation) TCs		354° 50 ft. 8 in.	Note 3
LC37 (IA01-37)	Lagging (Insulation) TCs		230° 37 ft. 6 in.	Note 4
LD37 (IA01-38)	Lagging (Insulation) TCs		354° 37 ft. 6 in.	Note 1
LC12 (IA01-39)	Lagging (Insulation) TCs		230° 12 ft. 4 in.	Note 4
LD12 (IA01-40)	Lagging (Insulation) TCs		354° 12 ft. 4 in.	Note 3
FA56 (IA01-47)	Vessel Head Flange	70°	56 ft. 1 in.	Note 1
FC56 (IA01-48)	Vessel Head Flange	230°	56 ft. 1 in.	Note 3
FD56 (IA01-49)	Vessel Head Flange	354°	56 ft. 1 in.	Note 3
HA58 (IA01-50)	Vessel Head	70°	58 ft.	Note 3
HC58 (IA01-51)	Vessel Head	230°	58 ft.	Note 3
HD58 (IA01-52)	Vessel Head	354°	58 ft.	Note 1

Notes:

1. Provides input to Recorder TR-IA02 in Reactor Building EL 51' 3"
2. **Deleted**
3. Spare or abandoned in place.
4. Abandoned in Place

8.2 OFFSITE POWER SYSTEM

8.2.1 Description

The Offsite Power System contains the following elements and interconnections between them:

a. The GPU Energy (GPUE) utility transmission system

b. The Atlantic Electric Company (AE) utility transmission system

c. The main transformers, station auxiliary and startup transformers, and associated power buses

d. The 230 and 34.5 kV transmission lines to the Oyster Creek substation

The interconnection of the facility with the 230 kV GPUE transmission system and the delivery of generated power are via the 230 kV Oyster Creek substation. The interconnection of the facility with the 34.5 kV GPUE system is via the 34.5 kV Oyster Creek substation. The overall substation interconnections are as shown in Figure 8.2-1.

A function of the Offsite Power System is to provide a backup source of ac power to the station when the main generator is incapable of supplying station loads through the auxiliary transformer. Offsite ac power normally supplies the station auxiliaries through the startup transformers during plant startup. After the station is operating and supplying electric power to the grid, offsite power acts as a standby source of power. Any plant transient, including manual operator action, that causes either or both the main incoming line circuit breakers (1A or 1B) from the auxiliary transformer to trip will automatically close the corresponding incoming line circuit breakers (S1A or S1B, respectively) from the startup transformers thus transferring station auxiliaries to the offsite power sources. An exception to this is that, if a fault exists on Bus 1A or 1B, the respective breakers, S1A or S1B, will not close.

A 230 kV system loss would also result in temporary loss of the 34.5 kV system serving the startup transformers. GPUE system procedures are in place for reestablishing sufficient power to the local 34.5 kV system, to restore offsite power support for the Oyster Creek Nuclear Generating Station (OCNGS) emergency auxiliary power. These procedures require 15 to 30 minutes for completion. During this interval, OCNGS shutdown proceedings and emergency power requirements are served by the onsite Standby Power Supply (diesel generators).

The Oyster Creek Nuclear Generating Station has priority demand on the 34.5 kV system power. During peak periods it may be necessary temporarily to curtail power to GPUE customers to ensure adequate emergency power availability at, or startup power delivery to, Oyster Creek.

8.2.1.1 230 kV System Connections

The 230 kV Oyster Creek substation features a breaker and a half bus arrangement and comprises the following line connections, as shown in Figure 8.2-2:

a. Oyster Creek Nuclear Generating Station (OCNGS) Main Generator Ties

b. Oyster Creek - Sands Point Line No. S2045

c. Oyster Creek - Larrabee Line No. N1028

d. Oyster Creek - Larrabee Line No. O1029

e. Oyster Creek 230-34.5 kV Substation Tie (Transformer Bank No. 7)

f. Oyster Creek 230-34.5 kV Substation Tie (Transformer Bank No. 8)

The OCNGS main generator ties deliver the station's generated power to the GPUE system via the 230 kV Oyster Creek substation. The 24 kV main generator output is connected to the main transformers by isolated phase bus, with the high voltage side of the transformers connected to the substation by overhead transmission lines. Two main transformers are provided. M1A is rated 325 MVA, 240 kV grounded wye - 24.0 kV delta, and is connected in parallel with M1B which is rated 350/392 MVA, 230 Kv grounded wye - 23.0 kV delta.

The Sands Point line (Line No. S2045) is connected to the AE system via the AE 230-69 kV Sands Point substation, which is located on OCNGS property. This line normally delivers backup power to the AE system. However, it can also deliver power from AE to the Oyster Creek substation for normal shutdown of the OCNGS, if AE is in service. **Credit can be taken for the use of the 69kV sandspoint feeder as an express feeder to OCNGS if the 34.5kV system serving the startup transformer is blacked out. The other express feeders are the Manitou Z-52 line and the Whittings Q121 line, which are described later. It can be restored in sufficient time (15 - 30 minutes) to satisfy both the controlled shutdown and emergency condition requirements at OCNGS.**

The Larrabee lines (Line Nos. N1028 and O1029) deliver power to and from the OCNGS. These two lines provide the station's interconnection with the 230 kV GPUE transmission system at the Larrabee substation and share double circuit transmission towers.

The Oyster Creek 230-34.5 kV Substation ties deliver 230 kV GPUE system power and the OCNGS generated power to the 34.5 kV Oyster Creek substation via Transformer Bank Nos. 7 and 8. Additionally, these ties provide sources of startup power and independent offsite power to the station. The two transformers are each rated 38/51/64 MVA, 230 kV delta - 34.5 kV grounded wye/19.92 kV.

8.2.1.2 34.5 kV System Connections

The 34.5 kV Oyster Creek substation has two parallel buses (Buses A & B) with a tie breaker between them. The tie breaker connecting the buses will open automatically if either bus is faulted. Each of the buses can be supplied by a separate line from other GPUE substations, following different rights of way. The substation consists of the following line connections as shown in Figure 8.2-3:

a. Bus A

1. Oyster Creek – **Off-Site Emergency Prep Building** Spare Line No. I69360 **(Alternate Feed)**
2. Oyster Creek - Waretown Line No. S145
3. Oyster Creek 230-34.5 kV Substation Tie (Transformer Bank No. 8)
4. OCNGS Startup Transformer Tie (Transformer Bank No. 6)
5. A/B Bus Tie
6. Capacitor Bank Tie (Capacitor Bank No. 1)
7. Oyster Creek - Manitou Line No. Z52
8. Substation Station Service Tie

b. Bus B

1. Oyster Creek - Lakeside Drive Line No. J69361
2. OCNGS Startup Transformer Tie (Transformer Bank No. 5)
3. A/B Bus Tie
4. Capacitor Bank Tie (Capacitor Bank No. 2)

5. Oyster Creek - Waretown Line No. R144

6. Oyster Creek - Whitings Line No. Q121

7. Oyster Creek 230-34.5 kV Substation Tie (Transformer Bank No. 7)

The Lakeside Drive and Waretown lines (Line Nos. J69361, R144 and S145) deliver power to area loads. The 1E1 Unit Substation at Oyster Creek Nuclear Generating Station receives power from either line R144, which is the preferred line, or line J69361, the alternate source. Each of the supplies is provided with a 2000 KVA Transformer. A pair of circuit breakers are used to shift between the two sources of power. The following facilities receive power from the 1E1 Unit Substation:

The New Radwaste Facilities
The Offgas Building
Package Boiler House
Service Water Pumps for New Radwaste
Redundant Fire Pump House

Line R144 can also be interconnected with the Oyster Creek - Manitou Line Z52 by closing of Switch No. 6 at Oyster Creek.

The Oyster Creek 230-34.5 kV Substation ties are as described in Subsection 8.2.1.1 and, as stated therein, they provide two sources of startup and offsite power to the OCNGS.

The OCNGS Startup Transformer ties (Transformer Bank Nos. 5 and 6) are the preferred sources of power for the station, providing both startup and backup auxiliary power. The two transformers are fed from separate and parallel 34.5 kV Buses (A and B) as shown in Figure 8.2-3. The startup power transformers do not supply any continuous station auxiliary power, except during startup and shutdown or for operation of the dilution pumps. Therefore, the station auxiliary power system has a backup power source that is immediately available. Furthermore, in the event both 34.5 kV buses are unavailable, the startup transformers can receive power directly from line Q121 through the manual operation of pole mounted disconnect switches. Either of the two startup transformers has been provided with more than enough capacity to carry the emergency auxiliary power load. Maintenance of proper voltage on the auxiliary buses is discussed in 8.2.1.4.

The Manitou line (Line No. Z52) **normally** delivers power to **area loads, but can also provide power to** OCNGS **under high grid loading conditions or in case of line/equipment outages. When the Z52 line is connected to deliver power to OCNGS, it also** provides an interconnection with the 34.5 kV GPUE transmission system at the Manitou substation. This is one of the two 34.5 kV circuits that can be restored as an express feeder to the OCNGS (an express feeder being one that does not pick up any load along the way) if the 34.5 kV system serving the startup transformers is blacked out. The other **34.5 kV** express feeder is the Whiting Q121 line, which is described later. **An additional express feeder is provided by the 69 kV Sands Point S2045 line as described previously.** The Manitou Z52 express feeder is established through various lines and substations using GPUE procedures currently in place. It can be restored in sufficient time (15-30 minutes) to satisfy both the controlled shutdown and emergency condition requirements at the OCNGS.

The Substation Station Service tie from the A bus delivers power to local substation loads via a 300 kVA distribution transformer. Back-up emergency Station Service is provided by a 3-100 kVA transformer fed from the Q121 line.

The Whiting line (Line No. Q121) delivers power to the OCNGS and provides an interconnection with the 34.5 kV GPUE transmission system at the Whiting substation. This is the **third** of **three** circuits used as an express feeder to the OCNGS. The other express feeder **are the 69 kV Sands Point S2045 line and** the Manitou Z52 line, which **are** described above. The Whiting Q121 express feeder is established by interconnection with the Whiting-McGuire Line No. K11 at Whiting. It can be restored in sufficient time (15-30 minutes) to satisfy both the controlled shutdown and emergency condition requirements at the OCNGS.

Any specific interconnection route required to re-establish 34.5 kV service to the OCNGS can be affected by a "common failure mode" if the initiating event occurs where the 230 kV lines cross or share right of way with the lines planned for the re-establishment. For example, the 230 kV lines both cross and share rights of way with one line planned to re-establish 34.5 kV at OCNGS and also crosses the line planned as the alternate route to re-establish the 34.5 kV.

Analysis has shown that an initiating event at any one of these and other points does not prevent re-establishment of 34.5 kV service to OCNGS by alternate routes.

8.2.1.3 <u>Protection and Control Circuits</u>

The 230 kV Oyster Creek substation features a breaker and a half bus arrangement. All 34.5 kV substation lines are provided with primary relay protection for phase to phase, three phase, and ground faults, with fault detection causing tripping of the corresponding circuit breaker. Bus differential relays on each of the 34.5 kV parallel buses provide primary relay protection for the buses, with fault detection causing tripping of all breakers on the faulted bus. Additionally, line and bus backup relay protection is provided on each of the buses. This backup relaying causes delayed tripping of all breakers on a particular bus for a faulted line, faulted bus, or stuck breaker.

Transformer Bank Nos. 7 and 8 and Startup Transformer Bank Nos. 5 and 6 are individually protected by transformer differential and ground relays, with fault detection causing tripping and lock out of those breakers necessary to isolate the affected transformer. Additionally, backup ground relaying is provided.

Circuit breakers and motorized switches in each OC substation are provided with control power from a battery power supply, at the substation. The charger is maintained on an energized a.c. source by an automatic transfer scheme.

Potential for line relaying and metering is supplied from potential transformers (PTs) on either of the buses. There is an automatic potential throwover scheme, which switches the 34.5 kV line relays and instruments from the A Bus PTs to the B Bus PTs and vice versa, if either are deenergized.

Indication of relay operation is annunciated in the Control Room.

8.2.1.4 Voltage Control

When the plant is operating and auxiliary power is being provided by the auxiliary transformer, voltage control is provided by the main generator excitation system. If auxiliary power is being provided by the startup transformers, induction voltage regulators connected to the 34.5 kV side automatically accommodate voltage fluctuations in the subtransmission network, providing 20% regulation to maintain proper voltage under normal and contingency system conditions. Three single phase regulators, each rated 667 kVA, 19.92 kV, with regulation in 32 steps of 5/8% are connected to each startup transformer.

Additional voltage support is provided first by use of either or both capacitor banks in the Oyster Creek 34.5 kV substation, followed by operation of the load tap changers (LTC) on the 230 kV/34.5 kV transformer banks feeding the 34.5 kV substation. When the additional support is no longer needed the LTC's are first returned to normal and then the capacitors are deenergized.

8.2.1.5 Testing

All substation circuit breakers are tested throughout the plant life in accordance with established schedules and procedures.

8.2.2 Analyses

Transient stability tests have been made to determine the performance of the Oyster Creek unit. The tests included contingencies involving the 230 kV transmission and 34.5 kV subtransmission systems. The tests consisted of extensive transient stability studies that simulated loss of units (including Oyster Creek), three phase faults with primary and delayed relay clearing, and single phase to ground faults with delayed clearing. Additional load flow computer tests were made to examine the system for overloads and voltage problems due to the loss of generating units and transmission lines.

The status of the various sources to the plant auxiliary equipment was also determined during the studies. The plant auxiliary equipment is supplied from the Oyster Creek unit terminals when the generator is running. When the unit is tripped off or is not running, the plant auxiliary equipment is supplied via startup transformers from the 34.5 kV buses at Oyster Creek. The sources for the 34.5 kV buses are transformation from the 230 kV system and also from the 34.5 kV system through the various 34.5 kV subtransmission lines connected at Oyster Creek (see Figure 8.2-2 and 8.2-3). The 230 kV system sources include two - 230 kV transmission lines and an interconnection to the Atlantic Electric 69 kV system through a 230/69 kV transformer located near the Oyster Creek substation.

The pertinent results of the studies were as follows:

a. A fault occurring on a single 230 kV line, bus or circuit breaker will not interrupt the sources for the plant auxiliary equipment. The sources still available include the unit terminal bus if the unit is running, the remaining 230 kV line, the 69 kV tie to Atlantic Electric, and the 34.5 kV lines connected to Oyster Creek.

b. A fault occurring on a single 34.5 kV line, bus or circuit breaker, **except the bus tie breaker,** will not interrupt the sources for the plant auxiliary equipment. The sources still available include the unit terminal bus if the unit is running, the two - 230 kV lines, the 69 kV tie to Atlantic Electric, and the remaining 34.5 kV lines connected to Oyster Creek.

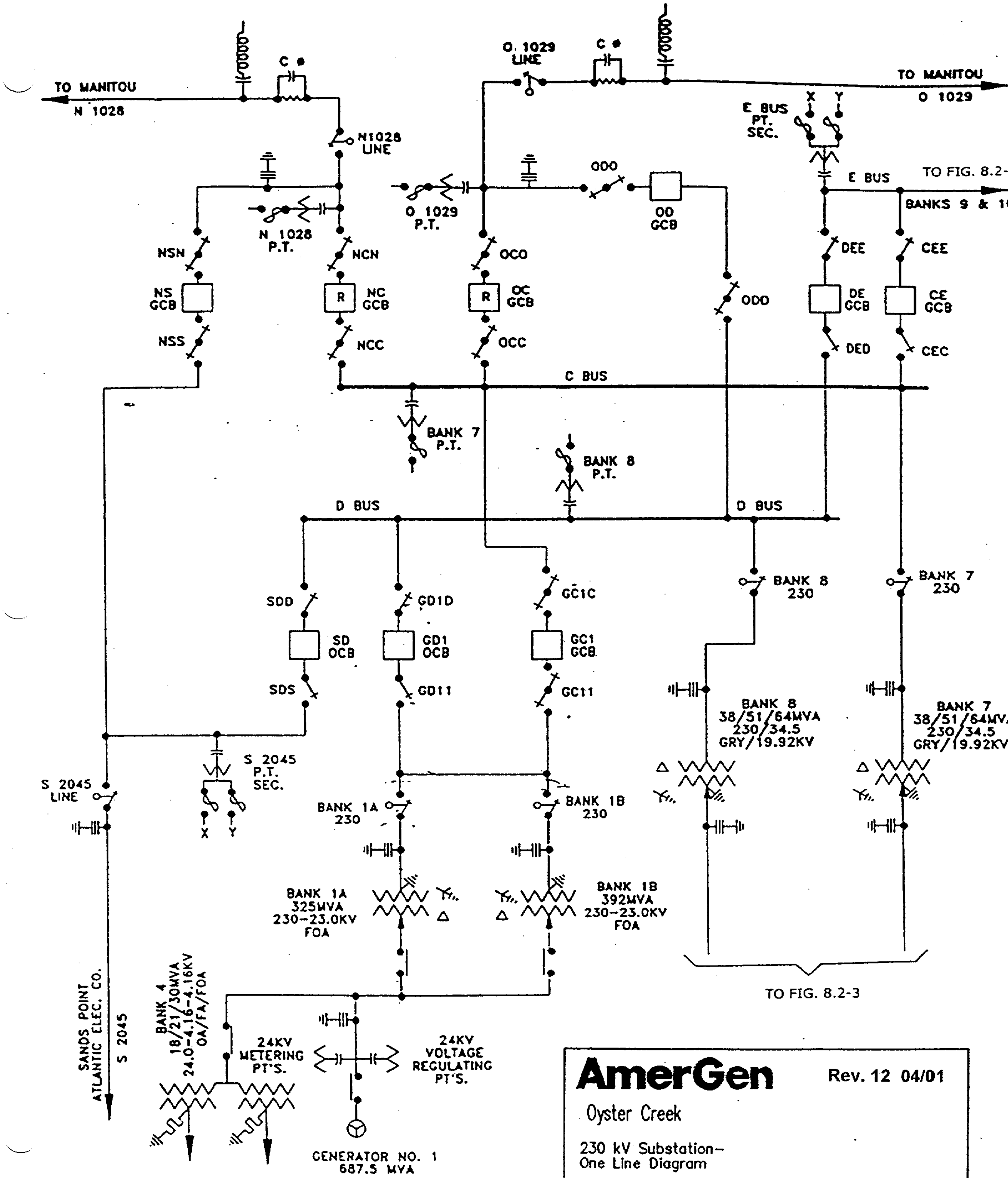
c. Direct tripping of the Oyster Creek unit, whether due to a fault on a single 230 kV line or not, will not interrupt the sources for the plant auxiliary equipment. The sources still available include one or both of the 230 kV lines, the 69 kV tie to Atlantic Electric and the 34.5 kV lines connected to Oyster Creek.

d. There will be no Oyster Creek unit transient instability, system transient instability, transmission line overloads or cascading outages as a result of a three phase fault with primary relay clearing of any one of the two - 230 kV lines emanating from the Oyster Creek station. This is also true for the case where a stuck breaker causes **a single phase to ground** the fault to be cleared by backup delayed clearing.

e. There will be no Oyster Creek unit transient instability, system transient instability, line overloads or cascading outages as a result of a three phase fault with primary relay clearing involving any of the 34.5 kV lines emanating from the Oyster Creek station.

f. Loss of both 230 kV transmission lines at Oyster Creek will result in the unit being tripped by its out of step relaying. The 34.5 kV system and the 69 kV tie with Atlantic Electric will remain available as sources to the plant auxiliary equipment provided area load is below a certain level. If the load is above the critical level, the procedures for re-establishing power to the plant auxiliary equipment is described in Subsection 8.2.1.

g. The sudden loss of output of the Oyster Creek unit by itself, the largest unit in New Jersey (Salem Unit 2) by itself, or the combination of the loss of output of both units together will not result in unit or system transient instability, transmission line overloads, cascading outages, or intolerable voltage conditions.



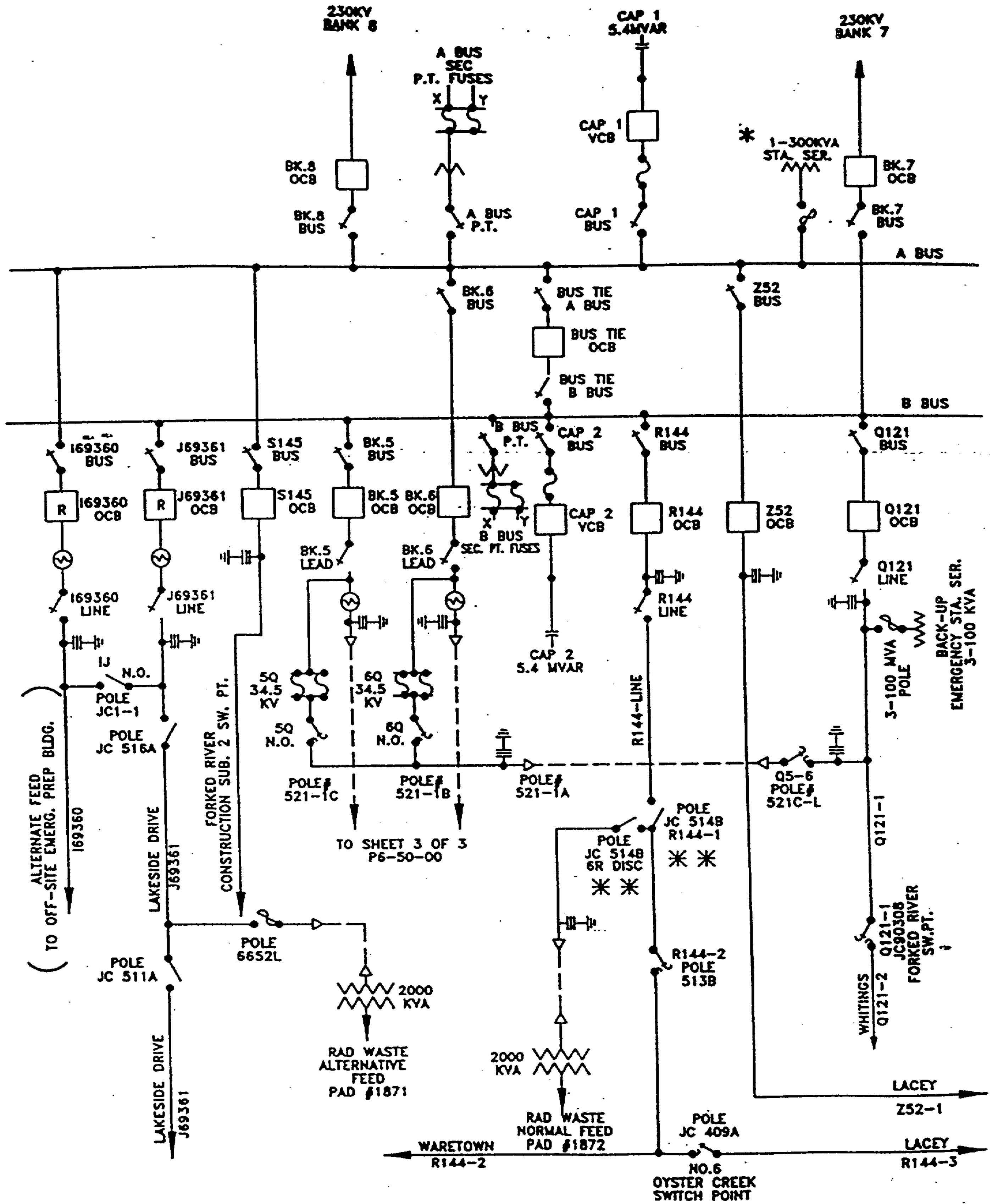
REF. P6-50-00 SH.1

AmerGen Rev. 12 04/01

Oyster Creek

230 kV Substation–
One Line Diagram

Fig.8.2–2



* AUTOMATIC STATION SERVICE THROW-OVER TO DILUTION

* * 6R DISC. & R144-1 DISC. ON SAME POLE

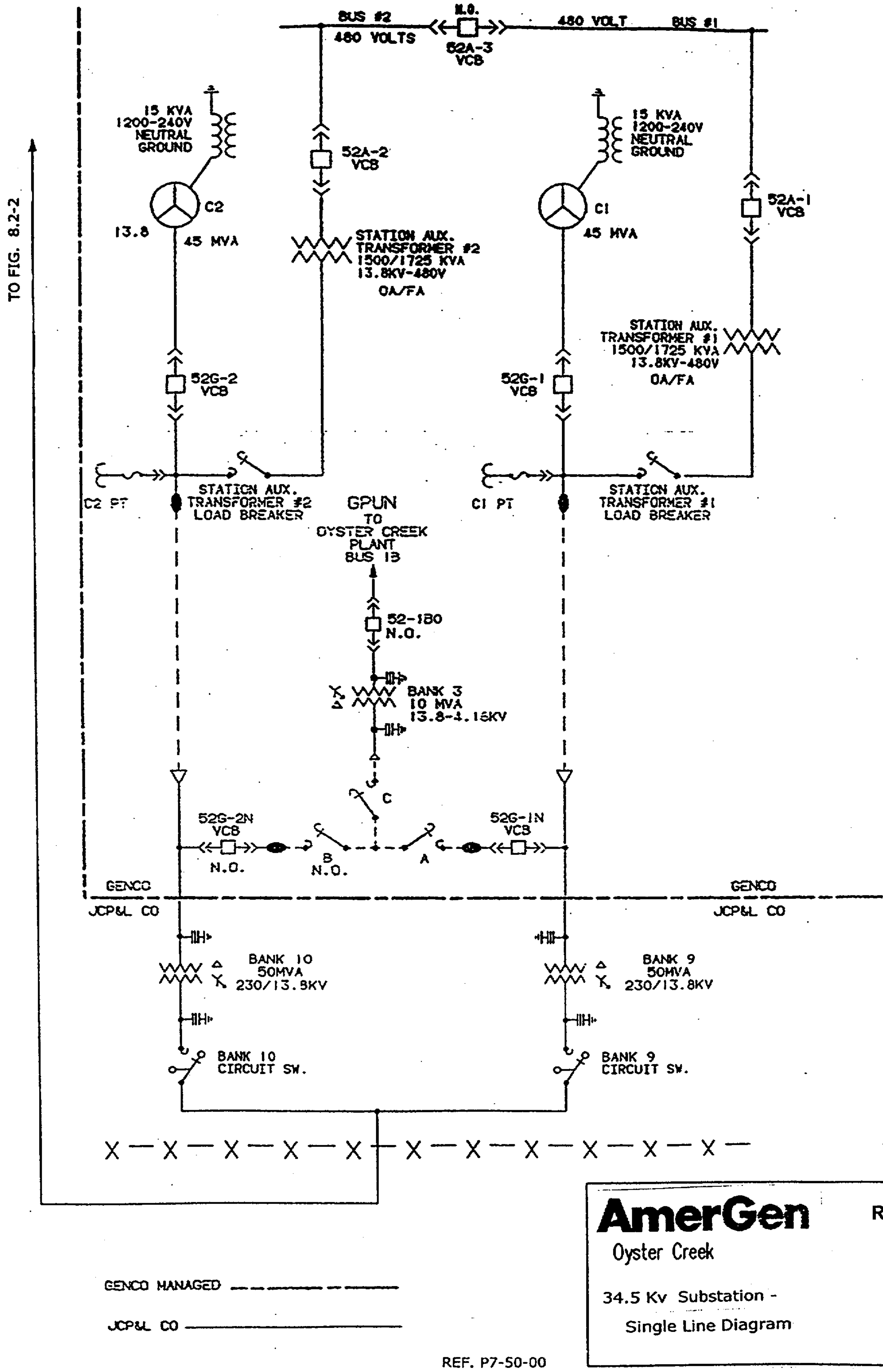
AmerGen Rev. 12 04/01

Oyster Creek

230 kV Substation-
One Line Diagram

REF. P6-50-00 SH. 2

FIG. 8.2-3



AmerGen

Oyster Creek

34.5 Kv Substation -

Single Line Diagram

Rev. 12 04/01

Fig.8.2-4

The batteries are rated at 100 ampere hours at **an eight** hour discharge rate. The 24 VDC chargers are used to maintain their associated batteries in a fully charged condition while supplying normal system loads. All chargers are supplied power from the vital 120 VAC system.

8.3.2.3 120 VDC Emergency Diesel Generator Starting System

A separate 120 VDC battery and battery charger is supplied with each Emergency Diesel Generator. Each battery consists of 56 cells and is rated at 420 ampere hours. Each battery charger is completely automatic, solid state, constant voltage device and is provided with AC voltage compensation, DC voltage regulation and current limiting characteristics. Additional discussion on the Emergency Diesel Generator Starting System can be found in Section 9.5.6.

The Emergency Diesel Generator batteries and other equipment associated with the 120 VDC Emergency Diesel Generator Starting System are easily accessible for inspection and testing. Service and testing is accomplished on a routine basis in accordance with Technical Specification.

8.3.2.4 Analysis

The safety related loads of DC distribution system C or of DC distribution system B can bring the reactor to a safe shutdown condition under normal or accident conditions.

The DC Power System meets the following safety considerations:

a. Redundancy of components and subsystems. This includes power supply feeders, load center arrangements, loads supplied from each bus, and power connections to the instrumentation and control devices of the system.

b. Independence between redundant portions of the system is provided by the electrical and physical separation of redundant power sources.

c. The batteries and chargers have sufficient capacity, capability and reliability to perform their intended function. This considers the combined load demand connected to each battery or battery charger during all operating conditions.

d. The instrumentation, control circuits and power connections of vital supporting systems are designed to the same criteria as those for the Class 1E loads and power systems that they support.

e. Fire detection and suppression is provided in the Battery Rooms.

f. Adequate ventilation is provided to maintain the hydrogen concentration below 2 percent by volume of hydrogen.

g. New equipment is designed to withstand the effects of a seismic occurrence consistent with the seismic design at its location.

h. No single event in a cable tray will prevent the DC **Power** System from performing its intended safety function.

i. The seismic design of the Battery Room and conduit runs is consistent with present design criteria in that part of the safety related loads are in nonseismic structures.

8.3.2.4.1 Compliance with General Design Criteria 17

The station batteries and **DC** distribution systems are redundant. Separation of equipment and wireways has been maintained insofar as practicable to make the redundant distribution systems immune to localized damage. DC **distribution system** C is physically separated from DC **distribution systems** A and B, which are located in a different area **of the plant**.

All feeder cables **associated with** DC **distribution system** C are run in conduits, thus providing physical separation from DC **distribution systems** A and B. **There are no direct electrical connections between DC distribution system C (Division A safety related loads) and DC distribution systems A and B (Division B safety related loads). There is no automatic or manual transfer of loads between distribution system C and distribution systems A and B.** All **DC** motor operated valve power cables are run in raceways. Controls for redundant power equipment are run in separate raceways.

8.3.2.4.2 Compliance with AEC Safety Guide 6 (presently Regulatory Guide 1.6)

The DC Power Systems have adequate independence between systems. Although the 125 V**DC** system has three automatic bus transfers between **distribution systems,** the three transfers are between **distribution systems** A and B, and **distribution system** A does not **normally** supply safety systems. The redundant safety related **systems** are **distribution** B and C, and there are no automatic **or manual** bus transfers between these **systems**.

An evaluation of the DC Power System has shown that the station design is in compliance with regulatory positions, as evidenced by the following design features:

1. The electrically powered safety loads at the OCNGS are separated into redundant load groups such that loss of any one group will not prevent safety functions from being performed.

2. Each **DC** distribution bus is energized by a battery and battery charger. The battery charger **and distribution system** of Battery B has no automatic **or manual** connection to redundant **DC** distribution **system C**.

8.3.2.4.3 Compliance with General Design Criterion 18

Availability of electrical power will be assured through periodic inspection and testing during operation. Periodic verification of **the** operability of power availability monitors is possible and the status of each power supply, i.e. voltage frequency and presence of grounds (on ungrounded systems) is continually indicated. Individual circuits of the energize-to-operate safeguards systems have loss of control power annunciation.

8.3.2.4.4 Compliance with GL 89-10 Requirements

DC motor operated valves in the Isolation Condenser and Cleanup Demineralizer Systems required for High Energy Line Break isolation are part of the Oyster Creek Nuclear Generation Station GL 89-10 program (V-14-31, 33, 34, 35, and V-16-14). These valves take credit for the availability of the DC system at float voltage to meet these isolation requirements. The battery chargers for the B and C Distribution Systems are designed, or have been modified to automatically load and restart onto the Emergency Diesel Generators and pickup vital loads.

Battery chargers C1 and C2 for the C Distribution System and charger A/B for the B Distribution System are static type chargers that remain connected to their respective power supplies and restart when power is restored. Battery Charger B for the B Distribution System is a MG Set type charger that will restart and reload automatically on restoration of AC power.

The battery chargers can supply float voltage within the time required such that the MOV's can perform their isolation function within their time requirements.

Alarms have been added to the Control Room to provide indication that the voltage is out of procedural tolerance for both the B and C bus. The minimum alarm setpoint is maintained at or above the required voltage to ensure MOV isolation capability.

8.3.3 Fire Protection for Cable Systems

The measures employed for the prevention of, and protection against, fires in the OCNGS are described in Subsection 9.5.1.

8.3.4 STATION BLACKOUT

10CFR50, Section 50.63, requires that each light-water-cooled nuclear power plant be able to withstand and recover from a station blackout (SBO) of a specified duration. It also identifies the factors that must be considered in specifying the station blackout duration. Section 50.63 also requires that, for the station blackout duration the plant must be capable of maintaining core cooling and appropriate containment integrity.

NRC Regulatory Guide 1.155, "Station Blackout", describes a means acceptable to the NRC staff for meeting the requirements of 10 CFR 50.63. Regulatory Guide 1.155 states that NUMARC 87-00, "Guidelines and Technical Bases for NUMARC Initiatives Addressing Station Blackout at Light Water Reactors", provides guidance that is in large part identical to the RG 1.155 guidance and is acceptable for meeting 10 CFR 50.63, and notes where the regulatory guide takes precedence.

In order to comply with the SBO Rule, GPUN has added an Alternate AC (AAC) Power Supply System.

The AAC capability is provided by **GPUE's** combustion turbines located at the Forked River site. The Forked River Combustion Turbines (FRCTs) supply power to Oyster Creek via an underground ductbank/trench system, a 10MVA, 13.8/4.16KV, 3 phase SBO transformer (bank #3) and a 4.16KV breaker/cubicle added to 4160 VAC Bus 1B. The SBO System is configured such that only one (1) out of two (2) FRCTs is required to supply power to Oyster Creek.

8.3.4.1 Alternate AC (AAC) Power Source

The AAC power source has been designed so that it will be available within one (1) hour of the onset of a station blackout event and has sufficient capability and capacity to operate systems necessary to achieve and maintain the plant in a safe shutdown condition. The FRCTs can be remote started via microwave from GPUE System operations in Reading or locally from the CT control rooms. No remote start capability is provided from Oyster Creek. The AAC system and components are not required to meet Class 1E or safety system requirements. SBO components and sub-systems are physically protected against the effects of likely weather related events that may initiate the loss of off-site power event. The AAC has an independent start system and fuel system. The FRCTs and their output breakers have

9.1 FUEL STORAGE AND HANDLING

9.1.1 New Fuel Storage

9.1.1.1 Design Bases

The new fuel storage facilities have been designed to prevent criticality during normal and accident conditions, to provide storage capacity for 30 percent core loading (**170** new fuel bundles), and to allow for reliable fuel handling. The new fuel storage vault design meets the requirements of General Design Criteria 61 and 62.

9.1.1.2 Facilities Description

The new fuel storage vault is a reinforced concrete structure designed to retain its integrity under seismic loading. The storage racks are full length, top entry, and spaced 11 inches by 6.5 inches center to center to prevent an accidental critical array. The new fuel bundles are stored dry, however, even if the vault becomes flooded, the effective multiplication factor will not exceed 0.95, providing a safe margin so that criticality will not occur. Vault drainage is also provided to prevent water accumulation.

New fuel is inspected on the new fuel inspection stand and then stored in the new fuel storage vault. The only entrance to the vault is through concrete plugs at the top of the vault, on the 119 foot elevation of the Reactor Building. The new fuel storage vault holds 30 percent of a full core load. Refer to Drawing GE237E516.

The fresh fuel bundles are loaded into the racks through the top. Each space for a fuel bundle has adequate clearance for inserting or withdrawing the bundle from above. Guides are provided to align the fuel bundles for the full length of their insertion into and removal from the rack (see Figure 9.1-2). The design of the racks prevents the accidental insertion of a fuel bundle into a location not intended to hold fuel. The racks provide full longitudinal support and adequately support the bundle weight at the bottom. Restraints are provided to ensure that rack spacing does not vary under specified earthquake loads and to prevent lifting of the racks in the event of a fuel bundle or grappling device binding during removal.

9.1.1.3 Safety Evaluation

Calculations of the effective multiplication factor are based upon the geometrical arrangement of the fuel bundle array. Subcriticality does not depend on the presence of neutron absorbing material. In an abnormal event, which assumes that the storage vault is flooded with water and the fuel bundles are brought to their closest spacing, the effective multiplication factor will not exceed a value of 0.95. Fuel criticality calculations have not been performed for conditions of mists or fogs or partial flooding (i.e., with a protective bag around the fuel assembly and flooding outside the bag).

Therefore, plant procedures require: 1) free flooding of fuel in case of vault flooding, 2) that the Fire Protection System be valved out when the new fuel storage vault is open, and 3) a ban on the use of water or other systems which could cause partial moderation.

The design of the new fuel storage vault prevents vault damage as a result of earthquake loads. The floor drain prevents accumulation of water in the vault. Radiation protection and monitoring provisions are presented in Section 12.3. Seismic design is discussed in Section 3.7. No special tests and/or inspections are required for nuclear safety purposes.

9.1.2 Spent Fuel Storage

9.1.2.1 Design Bases

The spent fuel storage pool is designed to withstand the design earthquake acceleration, to prevent inadvertent criticality, and to provide efficient shielding and cooling. The pool is designed to store various core components, including irradiated fuel assemblies, in such a way that:

a. Spent fuel assemblies are arranged in safe criticality configurations with provisions for adequate removal of decay heat.

b. Movement of all radioactive material in the pool can be conducted under shielded conditions.

Compliance with Regulatory Guide 1.13 has been evaluated, although the facility does not meet some of the specific recommendations of this guide, the evaluation of compliance has concluded that the design provides reasonable assurances that the health and safety of the public will be protected. The design of the pool conforms to the guidance of General Design Criteria 61 and 62.

The spent fuel storage racks are designed to maintain fuel assemblies in a geometry that ensures an infinite multiplication factor less than or equal to 0.95.

9.1.2.2 System Description

9.1.2.2.1 Spent Fuel Storage Pool

The spent fuel storage pool is a reinforced concrete structure, completely lined with seam welded stainless steel sheets, themselves welded to reinforcing members embedded in the concrete. The pool was designed to withstand the anticipated earthquake loadings as a Class I structure. The spent fuel storage pool consists of 6 foot thick reinforced concrete walls and a 4 foot 6 inch reinforced concrete floor slab supported by reinforced concrete beams up to 13 feet 7 inches deep. The inside face of the pool is lined with Type 304 stainless steel. **The liner is ¼" thick on the bottom of the pool, the north wall, and at the opening to the reactor cavity. The liner is a nominal 1/8" thick on the remaining walls.** The liner is liquid tight, serving as a barrier to any moisture loss from the concrete. The top of the pool is at El. 119' in the Reactor Building.

The spent fuel storage pool is 27 feet by 39 feet in plan with a total water depth of approximately 37 feet 9 inches, and an actual physical depth of 38'-9". The depth of water to the top of the stored fuel is approximately 25 feet, providing some 200,000 gallons of water above the fuel. (Drawing GE237E516)

The drainage system beneath the stainless steel pool liner detects and collects leakage between the liner and concrete, preventing leakage even in the unlikely event the concrete develops cracks. Water collected by the system drains into the Reactor Building Equipment Drain Tank, where it can be recycled via the liquid radwaste system to the Condensate Storage Tank.

To avoid unintentional draining of the pool, there are no penetrations that would permit the pool to be drained below one foot above the active fuel. All lines extending below this level are equipped with suitable valving to prevent backflow. The passage between the fuel storage pool and the refueling cavity above the reactor vessel is provided with two double sealed gates with a monitored drain between the gates. This arrangement permits detection of leaks from the passage and repair of the gates in the event of such leakage. A liquid level switch monitoring pool water level is provided to detect loss of water and permit refilling of the pool from the condensate transfer system. In addition, a second level switch in the pool surge tank is provided to permit almost instantaneous water loss detection. Both detectors alarm locally and in the Control Room. A low-low level switch is provided to automatically trip the SFPCS pumps upon reaching set point. Radiation monitors on the operating floor near the spent fuel pool alarm on high radiation and initiate isolation of the Reactor Building and operation of the Standby Gas Treatment System.

The pool is presently licensed to store 2645 fuel assemblies. Other equipment, such as control rods, spent nuclear instrumentation, and small vessel components, are temporarily stored in the spent fuel pool. Additional storage for large components, such as the steam dryer and the steam separator, is provided in a separate storage pool adjacent to the drywell heat cavity.

The fuel pool cooling system (Subsection 9.1.3) cools, filters and demineralizes the fuel pool water. Failure of the fuel pool cooling system cannot cause the fuel to be uncovered. Normal demineralized water makeup to the pool is provided from the 525,000 gallon (nominal capacity) Condensate Storage Tank at a rate of 250 gpm by a single condensate transfer pump. The makeup capability from this system is increased to about 420 gpm if both condensate transfer pumps are used. Makeup water is added directly to the pool's surge tanks by manual valve operation on El. 119'. Additional makeup, at a rate of 150 gpm, can be provided from the (nominal) 30,000 gallon Demineralized Water Storage Tank by the demineralized water transfer pumps through the use of hoses. Other sources of water are also available through the use of fire hoses or portable pumps. The makeup system for the spent fuel pool is not a Seismic Category I system. The 2000 gpm diesel driven fire pumps for the Fire Protection System can be used to provide makeup water from the Fire Pond to the Condensate Storage Tank through a permanent connection.

The two skimmer surge tanks which handle pool level surges contain a total of about 3,500 gallons at normal level (up to 7,000 gallons if full) which can be pumped into the pool at a rate of 1,000 gpm by the fuel pool cooling pumps. The tanks are provided with four inch outlet lines which join together as a six inch line; reactor cavity drains joins this line to become an eight inch pipe that leads to the suction of the fuel pool pumps.

The spent fuel pool water level is monitored, and high or low level is alarmed. Since the pool has no installed drains, level cannot be lowered by the cooling system below the level of the weirs. At the normal 400 gpm flow rate, the pool level is about three inches above the weir level, and the overflow just equals the 400 gpm being supplied to the pool from the diffusers. At normal level, the pool contains a depth of 38 feet of water (25 feet above the fuel), providing adequate shielding for normal building occupancy by operating personnel.

The pool is designed with substantial capability for withstanding the effects of a tornado. The design makes removal of more than five feet of water, due to tornado action, highly improbable. Protection is provided against all tornado generated missiles, having a probability of hitting the fuel larger than once per 1.4 billion reactor lifetimes.

The refueling floor and fuel storage facility have been designed to accommodate the movement of heavy loads. The only heavy loads that are moved into the spent fuel pool are spent fuel casks and shield plugs, other shielded casks used for waste handling, the fuel pool gates (during re-fueling), and other miscellaneous load movements. Travel for the casks is in and out of the Cask Drop Protection System area of the pool. The fuel pool gates normally hang on the south wall of the pool, and are lifted over the first row of racks and inserted into place. Travel paths for other miscellaneous load movements are evaluated by analysis on a case by case basis.

The fuel storage facility is completely enclosed within the Reactor Building, which is a controlled leakage building. The Standby Gas Treatment System consists of two filter trains, either one capable of drawing a 0.25 inch of water vacuum in the building. The Reactor Building superstructure siding is constructed of panels which extend from the refueling floor to the roof. The panels are supported by structural steel framing with girths spaced on approximately eight foot centers. The Reactor Building superstructure has been modified to increase its capability to withstand wind loads as discussed in Subsection 3.8.4.5.4. Loads calculated assume the siding and roof would remain intact. The load capacity of the superstructure is greater than the siding/roofing. A failure of the siding/roofing will increase the capability of the superstructure to withstand wind loads. The failure of the metal siding may create a tornado missile. However, the impact loading that could be generated by the missile on the fuel racks is bounded by the loading generated by the postulated drop of a fuel assembly which has been analyzed and found acceptable (Reference 4). The Reactor Building roof consists of built-up roofing over light weight concrete on metal roof decking. The metal roof decking is supported on purlins spaced on approximately nine foot centers. The minimum panel length used in construction was about 18 feet. Thus, the roof panels are continuous over at least two spans, reducing the likelihood of tornado generated missiles.

9.1.2.2.2 Spent Fuel Storage Racks

OC spent fuel pool has two types of high density poison racks of similar design. Ten (10) BORAFLEX racks (2645 spaces) manufactured by Joseph Oats Corporation were installed in 1987. An additional four BORAL racks (390 spaces) manufactured by HOLTEC containing BORAL as poison material were installed in year 2000.

The storage racks, except for their support spindles, have been fabricated from type 304 stainless steel sheet, plate and forgings and from fixed poison sheets. The support spindles were fabricated from SA564-Alloy 630. The pool layout is shown in Figure 9.1-3. The storage racks have been designed, constructed and assembled in accordance with ANSI N210-1976 (ANS 57.2); ASME Section III, Subsection NF, ASME Section IX. The racks are designed to Seismic Category I requirements.

9.1.2.2.2.1 Boraflex Racks

The fixed poison in BORAFLEX racks consists of B-10 enriched boron carbide dispersed in a silicone polymer matrix. The storage cells of the racks are assembled from preformed stainless steel sheets to form a series of double square channel. During the assembly, strips of the fixed poison sheets are sandwiched between the double walls. The nominal interior dimension of the storage cells is 6.0 inches and the nominal center to center distance between storage cells is 6.198 inches. The storage cells will accommodate fuel assemblies with nominal outside dimensions of 5.438 x 5.438 inches. The rack storage capacities range from 176 to 320 fuel assemblies and their weights range between 18,000 and 38,000 pounds. The bottom end of the assembled storage cells is welded to a 5/8 inch thick stainless steel base plate which has coolant flow holes in it that are on the same lattice spacing as the storage cells. The base plate is supported off the fuel pool floor by four support legs. This forms a lower plenum to permit coolant to flow laterally over the pool floor and enter the bottom of the storage rack via the flow holes in the base plate. The vertical dimension of the support legs is six inches on eight of the racks and 11 ½ inches on the two remaining racks. The nominal and maximum gaps between storage racks are 1 ½ inches and 4 inches respectively, which assures that a fuel assembly cannot be inadvertently inserted into a non-designated space within the storage rack array.

9.1.2.2.2.2 Boral racks

These racks have BORAL as a poison material which consists of finely divided particles of boron carbide (B4C) uniformly distributed in type 1100 aluminum powder, clad in type 1100 aluminum and pressed and sintered in a hot rolling process. The cells are fabricated from 0.075" thick type 304L stainless steel sheet material. Boral neutron-absorber material strips are placed between the cell walls and a stainless steel cover plate. Each storage cell side is equipped with one integral Boral sheet. The cells are welded together in a specified manner to become a free standing structure which is seismically qualified without depending on neighboring modules or fuel pool walls for support. The inside dimension of a cell is 5.9305 inches and the nominal center to center spacing of the cells are 6.106 inches. A one (1) inch thick base plate provides a continuous horizontal surface for supporting the fuel assemblies. The base plate is attached to the bottom of the cell assemblage by fillet welds and extends horizontally ¼ inch beyond the periphery of the rack cells. There are four (4) support pedestals per rack except Rack P which has five support pedestals.

9.1.2.2.3 Cask Drop Protection System

The possibility of dropping a spent fuel shipping cask into the spent fuel pool during handling of the cask, and the potential consequences of such an event have been considered in the design of the pool. Resulting damage to the pool could threaten the capability of the makeup system to keep the stored spent fuel submerged.

To preclude such an accident, a Cask Drop Protection System (CDPS) has been installed, and the path of the cask during handling is strictly controlled. The CDPS provides the following features:

a. A guide structure which properly guides and restrains a falling cask in the event that it is dropped into the spent fuel pool.

b. A hydraulic dashpot in the lower section of the guide structure which retards the falling cask such that impact loads are kept well below acceptable values.

The CDPS constitutes a passive system which has no significant effect on normal fuel handling operations, and reduces the likelihood of a cask drop accident.

In the event of a cask drop accident inside the spent fuel pool, the Cask Drop Protection System has been designed to slow down the rate of fall of the cask by hydraulic pressure. The system is also designed to attenuate the forces generated by the displacement of water and the impacts of the cask against the guide tube walls. To achieve the hydraulic attenuation

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The CDPS is attached to the spent fuel pool wall by means of two stainless steel tie rods. This arrangement is presented in Figure 9.1-10. The tie rods are attached to the upper guide cylinder at one end and to a two inch by four inch rectangular tie bar at the other. The tie bar is bolted to a jib crane base plate anchored to the operating floor adjacent to the northeast corner of the pool. The tie bars, in conjunction with the lateral support plates, provide the necessary lateral support for the guide structure to ensure that it will not detach itself from the corner of the pool in the event of a cask drop accident. The tie rods also provide energy absorption capability to limit the loads which would be transmitted to the spent fuel pool if the cask were to impact the upper portion of the guide cylinder during a postulated eccentric cask drop. As shown in Figure 9.1-9, the top plate and the upper guide cylinder are segmented by the radial saw cuts to limit the vertical load that can be transmitted to the upper guide cylinder in the event a lifting trunnion or lifting yoke were to impact the edge of the top plate during a cask drop or during lifting of the cask from the guide cylinder.

As a further measure, the transfer path for the cask centerline is on a controlled path width of six inches for cask centerline travel in the north-south and east-west directions (see Figure 9.1-11). This provides a margin of three inches in all directions between the controlled insertion diameter of six inches and the available insertion diameter of 12 inches. Visual aids are used to control the motion of the cask centerline to the prescribed transfer path. Mechanical rail stops are installed to prevent overtravel of the crane. Stops are mounted on the bridge rails for limiting north-south movement and on the trolley rails to limit east-west movement.

A limit of three inches on the vertical lift of the bottom of the cask above the top plate of the guide structure has been established and a vertical limit switch allows the cask to be raised to a maximum of six inches above the top plate.

In operation, drop velocity attenuation is by means of the hydraulic action of the cask base plate moving inside the tapered cylindrical guide structure, effectively reducing the drop velocity to less than 1 fps at the pool bottom. The annular space between the base plate and the cylinder allows the water under compression below the cask to be displaced at a controlled rate. The cylinder isolates the hydraulic effects of the postulated cask drop from any components in the pool, and the arrangement directs the water upward within the guide cylinder shell.

Openings are provided to allow water from the pool to re-enter the cylinder as the cask is removed. These openings consist primarily of the leakage area around the hinges of the fuel transfer gate. The total leakage flow area is about 5 ft^2. This area permits water to enter the guide cylinder at a rate which will prevent any significant difference in water level between the guide cylinder and the pool.

9.1.2.3 Safety Analysis

9.1.2.3.1 Seismic and Structural Considerations

Spent Fuel Pool Structural Analysis

The OCNGS spent fuel pool slab is a reinforced concrete plate structure with additional beams and end walls. An analysis was performed to demonstrate structural integrity for all postulated loading conditions and compliance with ACI-349 and NUREG-0800 guidelines.

The floor slab was modeled with plate elements, and the reinforced concrete beams were represented by beam elements. The walls were not represented in the model. The floor slab was assumed to be clamped at the reactor wall and simply supported at the remaining walls.

Similar impact springs, located at the base and top of each rack, were included to evaluate the potential for and the occurrence of impacts between adjacent racks and between racks and the pool walls.

Frictional elements (springs) were used to represent the frictional force between the rack base and pool liner and compression only gap elements were used to represent the vertical load carrying capacity of the pedestals. Linear frictional springs in two orthogonal directions were placed at four corner positions on the rack base to represent the effect of the static frictional force between each mounting pad and the pool liner. The forces developed in these frictional springs at any instant in time were related to **the** current compressive force calculated by the simulation code in the compression only gap elements representing the pedestals. Thus, lift-off of a pedestal was correctly simulated. Angular frictional springs about the vertical axis of each pad representing the distribution of pad friction under angular motion were not provided in the model. Review of the application of angular frictional springs indicated that their contribution to the displacement solution would be negligible.

Hydrodynamic mass effects were included to approximate the coupling effect between the water and the structure. The hydrodynamic mass effects are derived from first principles of classical fluid mechanics and represent the resisting effect of fluid in the small annular spaces between fuel assemblies and cell wall, between racks, and between racks and the pool walls.

All racks in the spent fuel pool **were** included in the dynamic model, and the model was subjected to the simultaneous application of three orthogonal seismic time history acceleration loads derived from the approved floor response spectra.

The results generated from the dynamic model, in terms of **nodal** displacements and forces in the various elements, were then used to compute the detailed stresses and corner displacements in the module, loads on the floor, and evaluate the potential for impact between racks and between racks and pool walls.

The resulting stresses at potentially critical locations of the module were examined for design adequacy in accordance with the acceptance criteria.

Stresses in the locations deemed limiting for the spent fuel racks, namely the gross cross section of the cellular structure just above the baseplate, and the gross cross section of the pedestals just below the baseplate are demonstrated to meet the required structural integrity limits.

The following assumptions were used in the analysis:

All rack models in the spent fuel pool are modeled and subjected to the three directional seismic motion. The hydrodynamic coupling between all racks in the pool is correctly simulated without any simplifying assumptions on the relative motion between adjacent rack modules.

Within a rack, all fuel rod assemblies were assumed to move in phase. This produces the maximum effects of the fuel assembly/storage cell impact loads.

The effect of fluid drag was conservatively omitted.

The lumped mass approach was used wherein the mass and mass moment of inertia of the fuel rack was lumped at two locations. The mass of the base plate and support structure was lumped with the bottom node. For the fuel assemblies, five lumped masses were used with 1/8 of the fuel mass located at the top and at the bottom nodes, and ¼ of the fuel mass located at the three intermediate nodes. For vertical motion, all of the mass of the fuel assembly was **coupled to** the bottom **rack** node.

The fuel assembly was modeled as a blunt square body inside a square cross section container. The hydrodynamic coupling mass utilizes Fritz's well known correlations for infinitesimal motions. Inclusion of finite amplitude motions (which is the case for a rattling fuel assembly) is known to significantly reduce the peak rate seismic response (vide, "Dynamic Coupling in a Closely Spaced Two Body System Vibrating in a Liquid Medium," by A. I. Soler and K. F. Singh, Proc. of the Third International Conference on Vibration in Nuclear Plant, Keswick, D. K. 1982). Therefore, Fritz's equation used in the analysis lead to an upper bound on the solution.

The lumped mass model of a single rack included 22 degrees of freedom which inlcuded 12 for the rack and 2 horizontal degrees of freedom per fuel assembly mass. The vertical motion of each fuel assembly **mass** was assumed equal to the motion of the rack base.

The structural behavior of the lumped mass model of each rack was completely described in terms of 22 equations of motion, one for each degree of freedom, which were obtained through the Lagrange equations of motions. Coupling between racks occurred through the hydrodynamic terms in the mass matrix, and through the closure of rack-to-rack impact elements if the solution indicated gap closures at any instant in time.

With respect to seismic excitation, three seismic time histories were applied to each rack in the pool and represented the seismic movement of the pool slab. Dynamic response runs were performed for those rack modules which represent the worst case condition..

The analyses have concluded that rack displacement due to a combination of sliding and tilting is less than one half of the 1.5 inch clearance gap between adjacent rack modules. The maximum top corner rack displacement of any rack in the pool under any of the simulations is 0.256". No impacts are predicted between any of the spent fuel racks **above the base plate elevation** nor are any impacts predicted between the racks and the spent fuel pool walls. Finally, all stresses in regions of high loads are within the allowable limits required by the applicable stress criteria.

Seismic Analysis of the Cask Drop Protection System

The effect of the additional load of the CDPS on the pool structure in case of a seismic event was evaluated. At the refueling floor level, the acceleration of the spent fuel pool structure expected under a hypothesized 1/2 SSE (as defined in 10CFR100, Appendix A) is approximately 0.18g. Assuming conservatively that the vertical accelerations are of the same magnitude as the lateral vibrations, the vertical loads on the fuel pool floor due to deadweight loads would be increased by about 18%. Evaluation of loadings and resulting shears and moments shows that if the deadweight loads are increased by 18%, the resulting maximum shears and moments would be increased by less than 5%. Since the resulting moments and shears are still below the allowables, it was concluded that the SFP is capable of sustaining the additional seismic loads.

9.1.2.3.2 Thermal Effects

Analyses have been made of the original pool design to evaluate the temperature gradients across the spent fuel pool walls and floor for both summer and winter conditions. The results of these analyses indicate that the maximum temperature differences between the pool water and the environment should be less than +60°F (pool water warmer than environment) and -25°F (pool water cooler than environment). Thermal stresses caused by these through-wall gradients are additive to the maximum tensile stress produced at critical sections by worst case cask drop impact loadings only in the case where the water in the pool is cooler than the environment. The thermal stress is not additive when the pool water is warmer than the walls.

Structural calculations were performed for a temperature gradient of -25°F through the wall of concrete. These calculations indicate that the maximum effect of such a gradient is approximately 10% to 15% of the ultimate moment capacity and less than 3% to 4% of the ultimate shear capacity of the member. Therefore, it is concluded that thermal effects will not affect the strength of the spent fuel pool structure to any significant degree.

Additional structural analyses were performed for the increase of storage capacity of the Spent Fuel Storage Pool. These analyses included loading due to thermal gradients in the pool floor and walls. The thermal analysis for normal operating conditions was based on a fuel pool water temperature of 125°F, and it showed that the pool structure can accommodate these thermal loads. This temperature limit of 125°F is a Technical Specification requirement.

9.1.2.3.3 Concrete Shrinkage Effects

The size and shape of a concrete specimen influences the loss (or gain) of moisture under given environmental conditions, and thereby affects the rate of volume changes, i.e., shrinkage. The spent fuel storage pool consists of 6'-0" thick reinforced concrete walls and a 4'-6" thick reinforced concrete floor slab supported by reinforced concrete beams up to 13'-7" deep. The inside face of the pool is lined with a liquid-tight liner which serves as a barrier to any moisture loss from the concrete. Therefore, drying out shrinkage will occur only at the outer, exposed surface. The shrinkage effects of a concrete element (wall or slab) lined on one face are similar to the shrinkage of an unlined element at twice the thickness.

Under drying conditions, shrinkage of the concrete near the surface will develop tensile stresses which are in equilibrium with compressive stresses near the center. These stresses, which are active over extended periods of time, will cause a plastic yielding (creep) of the concrete, permanently elongating the tensile fibers and shortening the compressive fibers so that the stresses induced by shrinkage will be appreciably less than the stresses induced by short term strains.

Bending moments that induce a compressive force on the inside face of the pool will not be affected by the drying out shrinkage occurring in the outer region containing the tensile steel reinforcing. Furthermore, the ultimate bending moment capacity that induces compressive stresses on the outside face will not be changed to any significant degree by these small shrinkage cracks. At the ultimate capacity of the member, the elastoplastic strains become so high that these cracks will close and transmit compressive stresses. Even in the unlikely event that the shrinkage cracks remain open, and thereby reduce the effectiveness of the member by the full depth of the cracks, the ultimate bending strength of the wall is estimated to be reduced by less than 9%. The effect on the ultimate shear strength is negligible.

In conclusion, shrinkage will not affect to any significant degree the strength of the spent fuel pool structure.

9.1.2.3.4 Dewatering by Tornado

The spent fuel pool design makes removal of more than five feet of water due to tornado action highly improbable, and since 25 feet of water cover the racks, the loss of five feet of water is of little concern from a fuel cooling and shielding standpoint.

9.1.2.3.5 Tornado Generated Missiles

Protection is provided against all tornado generated missiles having a probability of hitting the fuel larger than once per 1.4 billion reactor lifetimes.

Large equipment stored on the refueling floor, such as the reactor vessel head, shielding blocks, and other components, have a mass-to-surface area ratio such that they cannot become missiles under the postulated tornado conditions and, thus, could not be blown into the pool. Some smaller sized objects, such as hand tools and stored materials at El. 119', might be blown into the pool, but it is doubtful that even minor fuel damage would occur due to the protection afforded the fuel assemblies by the spent fuel racks; in any event, fuel damage should be slight and such an object should not cause damage to the pool liner which could result in loss of water.

The Reactor Building superstructure has been modified to increase its capability to withstand wind loads as discussed in Subsection 3.8.4.5.4. Loads calculated assume the siding and roof would remain intact. The load carrying capacity of the superstructure framing is greater than the capacity of the siding and roofing to a degree which assures that the panels will fail long before the framing becomes overstressed. Once the panels fail, the wind loads on the framing will be reduced. Thus, there are no pieces of the superstructure framing which could become missiles during a tornado. **The failure of the metal siding may create a tornado missile. However, the impact loading that could be generated by the missile on the fuel racks is bounded by the loading generated by the postulated drop of a fuel assembly which has been analyzed and found acceptable (Reference 4).**

From the above discussion, it is concluded that there is very little chance that either the spent fuel pool or the fuel stored in the pool could be seriously damaged as a result of tornado effects on the building or its contents.

Further evaluation of tornado generated missiles affecting the spent fuel pool considered a utility pole 35 feet long by 14 inches in diameter, having a velocity of 200 mph striking the pool slab as a cylinder in its worst orientation. The possibility of a large object such as a compact automobile being dropped into the pool is considered to be incredible and has not been considered. The analysis predicts that the 62 inch thick slab could not be perforated by the postulated missile. The maximum penetration of a 2050 lb utility pole was estimated to be 1.68 inches.

9.1.2.3.6 Failure of Facility Stack

The failure of the stack is not considered credible. Nevertheless, a 20 foot stack length and a 40 foot stack length (weighing 33,800 lbs and 83,800 lbs, respectively) were conservatively assumed to strike the spent fuel floor slab, with the smaller end as a cylinder, after falling from the top of the stack. Penetration was calculated to be one inch for the 20 foot segment and 4.5 inches for the 40 foot segment. The floor slab will not be perforated by a stack missile.

9.1.2.3.7 Turbine Missiles

The effects of turbine missiles have been calculated. The results for the high angle and low angle trajectories are presented in Section 3.5.1.3. Pool structural integrity will be maintained.

9.1.2.3.8 Protection Against Radioactivity Releases

The fuel storage facility is completely enclosed within the Reactor Building, which is a controlled leakage building. The Standby Gas Treatment System (see Subsection 6.5.1) provides effluent purification in the event of radioactivity releases, to minimize environmental discharge. The refueling accident is evaluated in Chapter 15, and results demonstrate that the design of the building and ventilation system are adequate for keeping offsite doses resulting from the event well below the guidelines of 10CFR100.

9.1.2.3.9 Criticality Considerations

The high density spent fuel storage racks for Oyster Creek are designed to assure that the neutron multiplication factor (k_{eff}) is equal to or less than 0.95 with the racks fully loaded with fuel of the highest anticipated reactivity and the pool flooded with unborated water at a temperature corresponding to the highest reactivity. The maximum calculated reactivity includes margin for uncertainty in reactivity calculations and in mechanical tolerances, statistically combined, such that the true K_{eff} will be equal to or less than the 0.95 with a 95% probability at a 95% confidence level (see technical specification 5.3.1). Reactivity effects or abnormal and accident conditions have also been evaluated to assure that under credible abnormal conditions, the reactivity will be maintained less than 0.95. Two separate criticality analyses have been performed as described above, one for high-density racks containing Boraflex and the other for high-density racks containing Boral.

Applicable codes, standards, and regulations, or pertinent sections thereof, include the following:

- **General Design Criteria 62, Prevention of Criticality in Fuel Storage and Handling.**
- **USNRC Standard Review Plan, NUREG-0800, Section 91.2, Spent Fuel Storage.**

- **USNRC Regulatory Guide 1.13, Spent Fuel Storage Facility Design Basis, Rev. 2 (proposed), December, 1981.**

- **ANSI-8.17-1984, Criticality Safety Criteria for the Handling, Storage and Transportation of LWR Fuel Outside Reactors.**

- **USNRC letter of April 14, 1978, to all Power Reactor Licensees – OT Position for Review and Acceptance of Spent Fuel Storage and Handling, Applications, including modification letter dated January 18, 1979.**

The Spent Fuel Pool (SFP) criticality calculations were based on fuel assemblies containing seven (7) fuel pins containing 3 weight percent Gd_2O_3 burnable poison with a maximum average planar enrichment of 4.0 weight percent U-235 at peak reactivity. This is equivalent to a 2.719 weight percent U-235 bundle with no gadolinium.

The criticality of fuel assemblies in the Oyster Creek SFP is prevented by maintaining a minimum separation of 6.198 inches between rows of fuel assemblies and by inserting a neutron absorber material; known as Boraflex, between the structural material of the racks.

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The Criticality analysis was performed using the CASMO-3 code and verified by the AMPX-KENO computer package, using the 27-group SCALE cross section set and the NITWAL subroutine for U-238 resonance shielding effects. The analysis referenced a number of benchmark calculations against critical experiments on poisoned spent fuel racks. Results of these calculations indicate a calculational bias of 0.0000±0.0024 for CASMO and +0.0101±0.0018 for KENO. Both biases correspond to a 95% probability at a 95% confidence level. In addition, a 0.0233 delta K is added to account for Oyster Creek specific uncertainties in the reactivity effect of manufacturing and mechanical tolerances, fuel burnup, and the calculational uncertainty (see Table 9.1-11). The evaluation included fuel cell dimensions, Boraflex dimensions, neutron absorber loading, fuel pellet density, fuel position, and pool water temperature by either using worst case initial conditions or by performing sensitivity studies to obtain the appropriate values. All uncertainties were at least 95/95 probability/confidence values. The fuel pool water temperature was conservatively taken to be 680F. Increasing temperature was shown to decrease reactivity. For investigation of mechanical tolerance effects, the CASMO code and a four group diffusion/blackness theory method of analysis were used to evaluate trends and small incremental reactivity effects that would otherwise be lost in the KENO statistical variation.

Abnormal Spent Fuel Pool conditions were evaluated including the effects of: positioning fuel assemblies outside of the storage rack, mispositioning fuel assemblies in the storage rack, fuel channel distortions, dropped fuel assembly (reactivity effect) and lateral movement of fuel racks. Results indicated that these scenarios did not increase the reactivity of the pool significantly. For the vertical dropped fuel assembly event, results indicate that for two vertical assemblies separated by water that the reactivity (k) will be less than 0.90 for any water gap spacing greater than 2.5 inches. For a dropped assembly lying horizontally on top of the rack, the separation distance is about 14 inches and will not constitute a criticality hazard.

In addition to the above considerations the effect of Boraflex shrinkage and gap formation was evaluated. The Boraflex Gaps in the Oyster Creek fuel racks are modeled using the AMPX-KENO computer package. The Gap is assumed at a central axial location and is reflected infinitely in the X-Y plane. This results in a gap at the same level in every Boraflex panel for the entire fuel pool, which is known as Coplanar Gaps. The Boraflex shrinkage is expected to saturate at a level corresponding to a 3.9 inch gap for the Oyster Creek racks. Therefore to bound all current and future gap formation the maximum gap size will be taken as 3.9 inches and will be assumed Coplanar. The delta k associated with this Coplanar gap is determined by subtracting the K_{eff} values of an infinite fuel bundle array without gaps, from a fuel array with gaps, and statistically combining the uncertainties from the KENO method. The analysis used a

2719 w/o U-235 enriched bundle which is equivalent to a 4.0 w/o bundle with seven (7) fuel pins with 3 w/o gadolinium loading at peak reactivity in the Spent Fuel Pool geometry. The 3.9 inch Coplanar gap corresponds to a delta k of 0.0280 which will be added onto the base fuel pool k_{eff} determined by CASMO-3. This calculation was performed with at least 95/95 probability/confidence levels.

Using the above methods and assumptions, the K_{eff} of the spent fuel racks including all uncertainties, with Boraflex gaps is 0.9451. This is based on a GE 8x8 fuel design containing a uniformly enriched lattice of 4.0 weight percent U-235 with seven (7) fuel rods containing 3 weight percent Gd_2O_3 burnable poison at its most reactive point in the bundles lifetime. The reactivity of this bundle design bounds all designs currently loaded in the spent fuel pool. The current analysis is conservative for the following reasons:

a. A fuel bundle of this high an enrichment will contain more than seven poison rods and/or the concentration of burnable poisons in these rods will be substantially higher, thereby, providing a less reactive fuel assembly than analyzed.

b. Fuel assemblies will have a distribution of fuel rod enrichments rather than a uniform rod enrichment. The uniform enrichment case yields the higher criticality for the same average enrichment and is, therefore, the limiting case for criticality safety evaluations.

c. All Gaps are assumed to be at their maximum shrinkage amount of 3.9 inches, Coplanar, and at the center of the rack to minimize leakage. In reality the gaps will be spread axially in a more uniform manner.

d. All reactivity calculations assume an infinite lattice and use the maximum lattice enrichment to represent the entire bundle.

With the calculational bias and all uncertainties added, the reactivity (K_{eff}) of the storage racks will always be less than or equal to 0.9451 with 95 percent probability at a 95 percent confidence level.

9.1.2.3.9.1 **High Density Spent Fuel Racks Containing Boraflex**

Since silica dissolution rate slows as the pool approaches the saturation level for silica, the silica is normally only removed from the pool water prior to a refueling outage. The silica loss results in a thinning of the Boraflex panels and loss of Boron Carbide from the polymer matrix. This is a slow process and significant thinning has not occurred in the fuel racks[6]. Oyster Creek surveillance coupons/Badger test indicate some thinning[4], but the Boron Carbide levels have remained above the values used in the criticality analysis[5]. In an attempt to further reduce the silica loss, Zinc has been injected into the Oyster Creek fuel pool. Laboratory tests have shown that small amounts of Zinc reduce the dissolution rate of irradiated Boraflex[6]. It was determined that Zinc addition would have no adverse affects on fuel pool materials and water chemistry, and the Zinc was removed once the Zinc levels had time to equilibrate within the rack annulus space.

The Boraflex coupon surveillance with other testing (neutron transmission) is necessary to assure the Boraflex panels are capable of meeting their intended function as a neutron absorber. The Oyster Creek corrective actions for dealing with Boraflex degradation are outlined in the GPUN response [7] to NRC generic letter 96-04.

9.1.2.3.9.2 **High Density Spent Fuel Racks Containing Boral**

The fuel selected as the design basis for the racks is a GE 8x8R fuel rod assembly with a maximum (uniform) initial enrichment of 4.6% U-235 containing sufficient gadolina (Gd_2O_3) to limit the maximum planar k-infinite in the standard cold core geometry to 1.32 or less (reference 9). Criteria for the acceptable storage of the GE-9B (8x8 rod array with large water rod) and the GE-11 (9x9 rod array with two large water rods) were also developed, based upon the K_∞ in the standard cold core geometry (SCCG). The SCCG is defined as an infinite array of fuel assemblies with nominal dimensions on a 6-inch lattice spacing at 20°C without any control absorber or voids.

The acceptance criteria for safe storage of spent fuel is that the vendor-supplied K_∞ in the SCCG must be equal or less than 1.32 for the planar region of highest reactivity with fuel of 4.6% average enrichment of less. This criteria is conservative since (1) all other axial regions (if differences are present) will be lower reactivity, (2) a planar-average enrichment of 4.6% is higher than would normally be expected in BWR fuel, and (3) the rack calculations use the assembly average enrichment (conservative compared to the distributed enrichments normally used in BWR fuel). The basic calculations supporting the criticality safety of Oyster Creek fuel storage racks for the design basis fuel are summarized in Table 9.1-12. For the design basis fuel, 4.6% enrichment or less with a k_∞ less than or equal to 1.32 in the SCCG, the storage racks conform to the USNRC criterion of a maximum k_{eff} less than or equal to 0.95.

Calculations were also made for fuel of GE-9B (8x8) and GE-11 (9x9) designs in the storage rack and in the SCCG, confirming that the maximum k_∞ of 1.32 in the SCCG is acceptable for fuel of various design concepts. Since the limiting k_∞ in the SCCG is enrichment dependent, calculations were also made for lower-enriched fuel. At enrichments less than 4.6%, a higher SCCG k_∞ would be acceptable as illustrated in figure 9.1-22. For enrichments less than 3.2%, the maximum k_{eff} in the rack is less than 0.95 (including uncertainties and allowances) regardless of gadolinia content and the k_∞ in the SCCG need not be considered.

The criteria for spent fuel to be acceptable for storage in the Oyster Creek spent fuel racks containing Boral are the following:

- **Any fuel assembly which has a planar-average enrichment of 3.2% or less, or**
- **Fuel assemblies with a planar SCCG k_∞ of 1.32 or less, with an average enrichment of 4.6% or less, or**
- **Alternately, any fuel whose enrichment-SCCG k combination falls within the acceptable domain (below the curve) of Figure 9.1-22.**

The design basis storage rack cell consists of an egg-crate structure with fixed neutron absorber material (Boral) of 0.0162 g/cm^2 boron-10 area density (0.015 gms boron-10/cm^2 minimum) positioned between the fuel assembly storage cells in a 0.77 inch channel. This arrangement provides a nominal center-to-center lattice spacing of 6.106 inches. Manufacturing tolerances, used in evaluating uncertainties in reactivity, include Boron loading, Boral width, lattice spacing, SS thickness, fuel enrichment and fuel density. The uncertainty in burnup, effects of channel removal, eccentric assembly location and the effect of channel bulge are included. A 0.01 Δk is added for the uncertainty in the vendor analysis for k_∞ in the SCCG. The 0.075-inch stainless steel box, which defines the fuel assembly storage cell, has a nominal inside dimension of 5.93 inches. This allows adequate clearance for inserting/removing the fuel assemblies, with or without the zircaloy flow channel.

None of the abnormal or accident conditions that have been identified as credible will result in exceeding the limiting reactivity (k_{eff} of 0.95). The effects on k_{eff} of temperature increases, boiling, an assembly dropped on top of the racks, misplacement of a fuel assembly, and seismic movement has been determined to be negative or negligible. The double contingency principle of ANSI N16.1-1975 (and the USNRC letter of April 1978) specify that it shall require at least two unlikely independent and concurrent events to produce a criticality accident. This principle precludes the necessity of considering the simultaneous occurrence of multiple accident conditions. Other hypothetical events were considered and no credible occurrences or configurations have been identified that might have any adverse effect on the storage rack criticality safety.

In the fuel rack evaluation, criticality analyses of the high density spent fuel storage racks were performed with the CASMO4 code, a two dimensional multi-group transport code. Independent verification calculations were made with the MCNP code, a continuous energy Monte Carlo code developed by the Los Alamos National Laboratory, and with the KENO-5a code package using the 238-group SCALE cross section library with the NITAWL subroutine for U238 resonance shielding effects (Nordheim integral treatment).

Benchmark calculations indicate a bias of 0.0009 ± 0.0011 for MCNP and 0.0030 ± 0.0012 (95%/95%) for NITAWL-KENO-5a. In the geometric model used in the calculations, each fuel rod and its cladding were explicitly described and reflecting boundary conditions (zero neutron current) were used in the axial direction and at the equivalent centerline of the Boral and steel plate between storage cells. These boundary conditions have the effect of creating an infinite array of storage cells in all directions.

The CASMO-4 computer code was used as the primary method of analysis as well as a means of evaluating small reactivity increments associated with manufacturing tolerances. Burnup calculations were also performed with CASMO4, using the restart option to describe spent fuel in the storage cell. MCNP and KENO-5a were used to assess the reactivity consequences of eccentric fuel positioning, channel bulging, and abnormal locations of fuel assemblies.

9.1.2.3.10 Cask Drop Analyses

The Cask Drop Protection System has been evaluated by NRC and found acceptable. The system has been designed to sustain the loads resulting from an accident whereby a fuel cask weighing up to 100 tons is dropped while the cask is being transported or suspended over the spent fuel pool, anywhere along the transfer path shown in Figure 9.1-11.

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9.1.2.3.13 Materials Compatibility

The pool liner, rack lattice structure and fuel storage cells are of stainless steel, which is compatible with the storage pool environment. The corrosion rate of type 304 stainless steel in spent fuel pool water is so small as to be considered nonexistent.

Dissimilar metal contact corrosion (galvanic attack) between the stainless steel of the pool liner, rack lattice structure, fuel storage tubes, and the Inconel and the Zircaloy in the spent fuel assemblies will not be significant because all of these materials are protected by highly passivating oxide films and are therefore at similar potentials.

9.1.2.3.13.1 **Boraflex**

The neutron absorber is composed of non metallic materials and therefore will not develop a galvanic potential in contact with the metal components. The selected absorber, Boraflex, has undergone extensive testing to study the effects of gamma irradiation and suitability as a neutron absorbing material. While these tests showed that the Boraflex was unaffected by the pool water environment and would not be degraded by corrosion, Oyster Creek and industry experience with Boraflex revealed Boraflex to be subject to certain other types of degradation [(3)].

Oyster Creek racks and other similar designs to the Oyster Creek racks have the Boraflex bonded to the rack structure with a silicone sealant during manufacture of the racks. Borflex shrinks with irradiation until shrinkage reaches a maximum of 4.2% at about 2 x 10^{10} rads. As the Boraflex shrinks gaps may form since the Boraflex cannot move freely in the rack. For racks of this design, gaps have been shown to occur in the Boraflex panels. The formation of gaps reduces the effectiveness of the Boraflex material as a neutron absorber and the gaps must be taken into account in the criticality analysis (see section 9.1.2.3.9)

Boraflex has also been shown to degrade due to the dissolution of the silica into the spent fuel pool. The silica levels at Oyster Creek are continually increasing as a result of the silica dissolution. The rate of dissolution depends on the amount of irradiation, temperature of the water, and the degree of communication between the rack annulus space containing Boraflex and the fuel pool, primarily through the vent holes. Venting of the annulus allows gas generated by the chemical degradation to escape, and prevents bulging or swelling of the inner stainless steel tube.

9.1.2.3.13.2 **Boral**

Based upon results of accelerated test programs and industry experience, it has been concluded that Boral has good general corrosion resistance in the fuel storage environment due to formation of tightly adherent oxide layer. Galvanic corrosion due to coupling of aluminum cladding with stainless steel structural material should be minimal in pure condensate grade or demineralized water due to high resistivity of the electrolyte. Pitting of aluminum, however, may occur particularly in creviced areas if dissolved salt, metal ions, oxygen or other gases accumulate.

Although accelerated test programs and industry experience has shown that Boral is a satisfactory material and will fulfil its design function over the lifetime of the racks, OCNGS has taken a prudent and cautious approach by establishing a Surveillance Program for monitoring Boral performance. The purpose of this surveillance program is to monitor the integrity and performance of Boral on a continuing basis to assure that slow and long-term synergistic effects, if any, do not become significant. The stated program is capable of detecting the onset of any significant degradation with ample time to take necessary corrective action. This is achieved by using two separate surveillance techniques (a)-periodic removal and examination of test coupon properties (including neutron absorption test) which has been deliberately subjected to accelerated radiation dose in the fuel pool, and (b)- In Situ neutron absorption test of Boral panels in the pool, in case degradation are suspected or seen.

9.1.3 Spent Fuel Pool Cooling System

9.1.3.1 Design Bases

The function of the Spent Fuel Pool Cooling System (SFPCS) is to remove decay heat from spent fuel assemblies that are stored within the Spent Fuel Storage Pool (SFSP) during all modes of operation, decay heat from the water inventory contained within the Reactor Cavity and Equipment Storage Cavity during refuel outages, minimize thermal stresses within the floor and walls of the SFSP and maintain the chemistry of the SFPS water inventory within acceptable EPRI quidelines.

The SFPC System operates continuously to circulate the SFSP water inventory and maintain the SFSP water inventory at a temperature of $T \leq 125°F$, near the water surface. This temperature limitation prevents the generation of excessive temperature gradients across the floor and walls of the SFSP. As a result , the corresponding thermal stresses/loads will not compromise the structural integrity of the SFSP. Furthermore, this temperature limitation provides assurance that that the cladding temperature of the spent fuel assemblies will be maintained sufficiently low, such that, nucleate boiling or voiding of the SFSP water inventory is precluded on the surface of the fuel rods. Therefore, steaming of the SFSP water inventory is minimized and dose releases from the spent fuel assemblies are maintained within 10 CFR 100 limits. The SFPC System also maintains proper SFSP water chemistry, which preserves the clarity and purity of the SFSP water inventory and prevents excessive corrosion of either the SFSP or its contents.

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Each of the auxiliary hoists has a pendant control station, some controls are duplicated at the main control console. Some refueling platform trolley and bridge motions are controllable from the auxiliary hoist pendants. However, trolley and bridge motions are limited to approximately 10 fpm when they are controlled from the pendants. Controls of the refueling platform bridge and trolley are interlocked so they cannot be controlled by more than one operator at a time.

Bridge and Trolley

The refueling platform bridge and trolley are driven by centrally located, variable speed motors. Bridge and trolley traversing speeds are continuously variable from 0 to 50 fpm and 0 to 30 fpm, respectively.

Control Rod Grapple

The control rod grapple is an air actuated tool for handling the control rods. The control rod is unlatched from the control rod drive before it is lifted with the grapple. Unlatching is performed from the control rod drive or by the latch tool. The control rod grapple is designed for use with the refueling platform auxiliary hoist and an actuator pole to provide positioning guidance. The grapple features a lower guide to locate the control rod bail into the tool and a swing hook to capture the bail. Design of the hook and guide is such that the control rod cannot be unhooked while the grapple carries the rod weight.

Control Rod Latch Tool

The control rod latch tool is a tool used to unlatch the control rod blade from the core if the control rod drive cannot be fully withdrawn and the blade unlatched from below the vessel. The tool consists of a long arm which, in the operating position, extends below the fuel support casting into the guide tube to reach the control rod latch.

Control Rod Servicing Tool (Take 2 Tooling)

The control rod servicing tooling (Take 2 Tooling) is used for the exchange and shuffle of control rods in the reactor vessel. The control rod servicing tool consists of the control rod/fuel support, a control rod latch tool, an interim storage rack called a quiver, a jib arm, and a grid guide.

Control Rod Guide Tube Grapple

The control rod guide tube grapple is an air actuated tool used to remove and replace the control rod guide tubes. This tool is built to attach to the refueling platform auxiliary hoist and may be used with the actuator pole for guidance. The guide tube grapple consists of a pipe section with cruciform guide members on the top and bottom. The pipe section is machined to fit into the top of the guide tube as does the fuel support casting. Radial orientation is provided by a slotted ear which engages the core support plate pin. For lifting the guide tube, air cylinders actuate plungers which engage two opposed coolant ports. Upon lifting, the guide members guide the tube through the grid.

Fuel Support Installation Tool

The fuel support installation tool is an air actuated tool for removing and replacing the fuel support castings. This tool is built to attach to the refueling platform auxiliary hoist and may be used with the actuator pole for guidance. The tool is a grapple designed to fit the fuel support casting. Air actuated engagement pistons attach the grapple to the casting.

Air Supply Connector

The air supply connector is a special hose clamp used on the refueling platform auxiliary hoists to fasten the air supply lines to the tool to be used. The connector clamps to the air line and attaches to a tab on the grapple through a clevis and pin arrangement. The air hose is fed from the spring reel at the same rate that the tool is being lowered to prevent undue strain on the hose connections.

Instrument Handling Tool

The instrument handling tool is a grapple used to install and remove incore instrumentation and the neutron sources. The tool, used with the instrument strongback and water seal cap, consists of a latch which grips the instruments on the upper spring loaded plunger assembly. Used with the actuator pole and the instrument strongback, the instrument handling tool locates the instrument into its recess on the underside of the upper grid assembly.

Instrument Strongback

The instrument strongback is required for handling of the LPRM and dry tube assemblies. The tool is a 41 foot long structural member upon which is mounted a series of latches to capture the instrument. The strongback is used to remove those instruments which cannot be bent at any appreciable radius (without damage) from a shipping box lying horizontally adjacent to the pool, and move them to their vertical position in the vessel. Latching of the instrument handling tool to the instrument, and positioning the instrument into the reactor with the strongback is necessary to protect the instrument.

Water Seal Cap

The water seal cap is used with the instrument handling system. It consists of a section of pipe about four feet long which is fastened onto the bottom of the incore housing flange at the bottom of the vessel, after removing the incore seal nut. This effects a vessel seal, permitting removal of the incore instrument.

General Purpose Grapple

This is a manually actuated device designed to carry fuel assemblies, control rods and any auxiliary equipment having a bail with the same configuration as that of a fuel bundle. The general purpose grapple may be carried from any hoist with a 7/16 inch - 14 stud termination or from any safety hook when used in conjunction with the "Sling, Auxiliary Hook, Main Crane." The primary purpose of the device is to transfer fuel bundles from new fuel storage to the pool, and in-pool transfers of fuel in the area of the fuel preparation machines in conjunction with the jib cranes. Latching and unlatching are done by hand in the case of new fuel or blade transfers out of the water and by use of the actuator pole under water. Load limit for this grapple is 1000 pounds.

Actuator Pole

The actuator pole is a general purpose tool used to actuate and guide several of the reactor servicing tools. The pole, consisting of sections each approximately 15 feet long, includes an actuator section and extension sections. The actuator sections mate to the square stud on the various grapples and attach to extension sections through rigid connections.

Viewing Aid

This is a water box which consists primarily of a frame with a clear plastic bottom. Its purpose is to smooth out the action of a disturbance on the surface of the water.

General Area Underwater Light

This consists of a reflector with a 1000 watt incandescent bulb. The light is directed downward and is primarily used to illuminate fairly large areas.

Drop Light

This consists of an outer jacket enclosing a 1000 watt quartz line lamp. The light is directed primarily to the sides. Its use is primarily for general viewing in tight areas, such as in a fuel cell or in the guide tube area where a fairly general light is wanted in the lateral direction.

Local Area Underwater Light

The local area light is a small diameter light of approximately 400 watts. Its light is primarily directed down and it is used in close areas such as the bottom of the vessel or other areas that are immediately below the light.

Head Stud Rack

This rack is used to store those vessel studs which are removed from the area of the fuel pool gate. The rack and studs are handled by the reactor building service crane.

Stud Tensioner

This is a hydraulic driven tool which stretches four studs at one time. The tension loads are determined by the hydraulic pressure. The proper operation of this tensioner is thoroughly described in the Reactor Vessel Instruction Manual (VPF1238-146-2). An extensiometer is used to ascertain proper stud tensioning. After the nuts have been snugged up, dial indicator readings are taken of the distance between the top of the stud and the top of the master stud measuring rod which is inserted in the hole provided in the center of the stud.

Head Tensioner Carousel

The Reactor Pressure Vessel Head Tensioner Carousel Assembly combines the functions of the GPUN spreader beam rack for four Biach hydraulic stud tensioners, a nut and washer storage rack, and a storage rack for the thread protector caps for the RPV studs. The benefit of this combined function equipment is reduced outage time, manhours and manrem exposure for the stud tensioning operations. In addition, the carousel will support future modifications that will allow the use of eight Biach stud tensioners.

Head Holding Pedestal

There are three supports for the reactor vessel head when it is removed from the vessel. These are located on the reactor building refueling floor.

Closure Nut Wrench

This spanner wrench is used with the stud tensioner to snug up and loosen the vessel head stud nuts.

Stud Wrench, Stud Lifting Attachment, Stud Protectors

These are tools for handling the vessel studs, and for covering the threads to provide protection against damage during refueling.

Bushing Wrench, Nut and Bushing Lifting Attachment

These are tools for removing stud bushings and for lifting nut and bushings. The lifting attachment threads into the nut or bushing which is listed by an auxiliary crane.

Head Nut and Washer Rack

These are boxes for storing vessel head nuts and washers, six each per box. These boxes are stacked on the refueling floor for storage. Each nut weighs about 70 pounds.

Heat Strongback

This is a special sling lifting device for moving the vessel head.

Insulation Removal Sling Assembly

This is a special sling for handling sections of insulation from the reactor pressure vessel head and flange.

Steam Dryer and Separator Sling

This is a special sling for handling the dryer assembly and the steam separator assembly.

Shroud Head Bolt Wrench

This is a long extension wrench for loosening and tightening the shroud bolts while working from the service platform.

Blade Guide

These double blade support structures guide the control rod in a fuel cell without any fuel assemblies. They provide support and guidance for control rods, either fully inserted or at any withdrawn position, and also can be used for scram testing the control rod drive with a control rod attached.

Control Rod Guide Tube Seal

This provides a water seal in the control rod guide tube to permit control rod drive removal when the control rod is not in the reactor. The normal seal for drive removal is provided by the control rod itself, sealing at the bottom of the guide tube. This seal tool permits drive removal even if the drive index tube cannot be retracted into the drive.

Peripheral Orifice Grapple

This grapple is used to insert or remove the peripheral fuel orifices from the lower grid structure.

9.1.4.2.3 Cranes and Hoists

The cranes and hoists in the facility have been reviewed for conformance with NUREG 0612 criteria. Although some of the cranes and hoists noted in this section are not involved in refueling operations, they have been included to verify their safety requirements, as appropriate.

Pertinent data for the Turbine Building Crane, Reactor Building Crane, and Heater Bay Crane are presented in Tables 9.1-5 through 9.1-7. Limits of travel for these cranes are shown in Figures 9.1-15 through 9.1-17.

With regard to the Spent Fuel Pool jib cranes, no specific loads in excess of 800 lbs have been identified that must be lifted with these handling systems. The refueling platform auxiliary hoists and the main fuel grapple have been **administratively** derated to 750 lbs so as to exclude them from the NUREG-0612 review.

The major handling system affected by the NUREG 0612 review include: the Reactor Building Crane. Only the Reactor Building Crane is utilized for fuel handling operations.

Reactor Building Crane

The Reactor Building Crane is of the overhead traveling type (See Table 9.1-6), and is mounted with the bridge traveling along the north-south axis. Bridge rails are 25 feet above the operating floor, and the limits of travel are shown in Figure 9.1-16.

The crane was designed and fabricated by Whiting Corporation to the specifications in EOCI-61, "Specifications for Electric Overhead Traveling Cranes - 1961," and the design has been compared to CMAA-70-1975 and ANSI B30.2-1968 and found acceptable under the requirements of NUREG 0612.

Furthermore, the original crane design complied with the requirements of OSHA 29 CFR 1910, Subpart N, Section 1910.179 (1970), with the following exceptions:

a. The bridge and hoist controls do not face in the direction of the crane motion.

Each of the heavy loads evaluated has been assigned to one or more safety classes (See Table 9.1-10). In some cases, more than one safety class assignment is required because more than one of the load handling situations could be encountered when handling the load.

For each of the heavy loads listed in Table 9.1-10, the safe load path/ procedural requirements corresponding to the assigned safety class have been added to the appropriate plant operating or maintenance procedures. When more than one safety class assignment was made for a particular load, the safe load path/procedural requirements of all safety class assignments were included in the procedures.

9.1.4.2.4 Refueling Operations

In the process of refueling and servicing a reactor, there may be about 400 separate operations to be performed. This is only an order of magnitude, and is given to illustrate that with this large number of operations, an efficient, safe refueling servicing procedure is strongly dependent on planning, preparation and performance.

Before each outage, each step is carefully planned with specific tasks assigned to each man performing the various operations.

There are certain sequences of steps to be performed at each refueling outage. During each outage, these sequences may be refined to provide the safest, most efficient method of performing the function. The planning includes equipment inspection before starting and periodic checks of equipment during operations.

Speed alone is not the primary objective. Safety is the first consideration, for much time may be lost by a single mishap. Efficient, safe operation is achieved by careful preparation before the start of each operation to be performed. All equipment to be used is carefully inspected and placed in perfect working order. During the outage, careful observations and equipment checks are included as part of the operating procedures.

For the most current refueling procedures refer to the plant operating procedures manual.

9.1.4.2.5 Spent Fuel Transfer Out of Reactor Building

The transfer of spent fuel from the spent fuel pool to locations outside of the Reactor Building involves the handling of heavy transfer casks and spent fuel. The accidental dropping of a cask during transfer operations could result in significant damage to the Reactor Building floor and nearby safety systems and equipment. To preclude accidents of this nature, and the associated consequences thereof, crane upgrades (discussed in Section 9.1.4.2.3) and other engineered systems have been implemented, including the use of a redundant support system and an energy absorbing crush pad.

The redundant support system is referred to as the Fixed-Link Support System (FLSS) and is designed for use with the NUHOMS fuel transfer cask on the refuel floor. The FLSS consists of two vertical support arms that are attached to and hang down from the main Reactor Building crane trolley. During fuel transfer operations, the FLSS arms are attached to the yoke of the transfer cask. In the unlikely event of a crane system failure (such as a failure of the hoist rope), the FLSS system will completely support the cask, precluding the possibility of a cask drop onto the floor of the Reactor Building. The FLSS is designed to provide this protection for the transfer cask and its contents, up to a maximum design load of 100 tons. The FLSS was load tested to 200% of its design load in the shop and 125% of its design load after installation in the Reactor Building.

The FLSS provides protection against a cask drop while the cask is on the 119'-3" elevation. The system is initially attached to the cask yoke after it is brought up the equipment hatch, and remains attached (and functional) for all lateral movements of the cask over the refuel floor. While the FLSS is attached, the bottom of the fuel transfer cask is maintained approximately 8 1/2 inches above the refuel floor (from the bottom of the cask baseplate to the floor). The FLSS is disconnected from the cask yoke once the cask is located over the spent fuel pool (at which point possible damage from a cask drop event is precluded by use of the CDPS, as discussed in Section 9.1.2.2.3). When not in use, the FLSS is stored in a retracted position on the trolley structure. While stored, the FLSS does not interfere with other crane operations or motion.

As part of a fuel transfer process, the fuel transfer cask will be temporarily placed on the refuel floor (to perform various cask preparations and verifications prior to loading and transferring spent fuel). This requires that the FLSS be momentarily disconnected, at which point it does not provide redundant protection. Although the FLSS is only disconnected for a short duration, an energy absorbing crush pad system was developed to ensure protection against a cask drop event for this evolution. The crush pad, which is fabricated from an aluminum honeycomb material, is placed directly under the fuel transfer cask prior to disconnecting the FLSS. If a crane failure and cask drop event were to occur, the crush pad is designed to absorb the drop energy of the cask by plastic deformation of the honeycomb. Reaction loads on the floor are controlled by the design and held below the load capacity of the floor.

c. Trolley Mounted Auxiliary Hoist

Refueling interlocks have been provided for the fuel grapple hoist only. Therefore, during refueling, fuel handling with auxiliary hoists is prohibited. In addition, the auxiliary hoist overload trip setting will be set to trip the hoist motor if an attempt is made to handle a fuel bundle during refueling.

Bridge Drive Power Interlocks

Power to the refueling bridge drive motor is not permitted (refueling pool to reactor well) if the following conditions are met:

a. All rods are not fully inserted, the bridge is over the reactor vessel, and the fuel hoist is loaded, or

b. The reactor mode switch is in the STARTUP position.

The STARTUP position of the reactor mode selector switch permits free withdrawal of the control rods according to a specified sequence to bring the reactor critical.

To assure safe operation in this mode with the reactor vessel head removed, two interlocks are provided. A rod block is applied if the refueling platform is over the reactor vessel with the mode selector switch in the STARTUP position. With the mode selector switch in the STARTUP position, movement of the refueling platform from the fuel pool to the reactor vessel well is prevented by interrupting power to the bridge drive motor in combination with the fuel interlock switch.

A complete discussion of the reactor mode selector switch can be found in Section 7.2.

9.1.5 References

(1) TDR 1131 "Boraflex Gap Evaluation for the O. C. Spent Fuel Pool Racks", J. H. Sedar, Rev 2, October 1994.

(2) Safety Evaluation 000254-001, "Increase in Spent Fuel Pool Enrichment Limit To 4.0% U-235", J. H. Sedar, Rev 1, October 1994.

(3) NRC Information Notice 95-38: Degradation of Boraflex Neutron Absorber in Spent Fuel Racks, September 8, 1995

(4) NRC Letter to GPU dated December 7, 1992, “Evaluation of Upper Reactor Building and Non-Safety Architectural Components Subjected to Tornado-Wind Loading, Items 1 and 11 of SEP Topic III-2.

(5) K. Lindquist et al., "Inspection and testing of Boraflex Surveillance Coupons from the Oyster Creek Station," prepared for GPU Nuclear Corporation by Northeast Technology Corp. NET-124-01 dated June 1997.

(6) S.E. Turner, "Criticality Safety Evaluation of Oyster Creek Spent Fuel Storage Racks with Fuel up to 4.2% Enrichment," Holtec Report HI-931049 July 1993 prepared for GPU Nuclear Corporation

(7) K. Lindquist et al., "Zinc Demonstration Program: Badger Test Campaign at Oyster Creek Nuclear Station", prepared by Northeast Technology Corp. NET-092-06, Rev 0, March 1998.

(8) Letter to U.S. Nuclear Regulatory Commission from M.B. Roche (GPUN), "Response to Generic Letter 96-04", October 15, 1996, 6370-96-2300.

(9) "Licensing Report for Storage Capacity Expansion of Oyster Creek Spent Fuel Pool," prepared for GPU Nuclear, Inc. by Holtec International, Holtec Report HI-982983 Rev. 1, April 1999.