TABLE 9.1-4
(Sheet 1 of 1)

DELETED

TABLE 9.1-8
(Sheet 1 of 1)

DELETED

TABLE 9.1-10
(Sheet 1 of 2)

HEAVY LOADS[(1)] CARRIED BY THE REACTOR BUILDING CRANE*

Load	Safety [(2)] Class	Weight (Tons)	Lifting Device
Drywell Head	1/3	62	Head Strongback
Reactor Vessel Head	1/3	**75.3**	Head Strongback
Cavity Shield Plugs (eight provided)	3	85 ea.	**Slings, Shackles, and Adapters**
Reactor Vessel Head Insulation	3	5	Slings
Steam Dryer	1/3	26	Steam Dryer/ Separator Sling Assembly **(Wet Lift Rig)**
Steam Separator	1/3	44	Steam Dryer/ Separator Sling Assembly **(Wet Lift Rig)**
Fuel Pool Gates (two provided)	2	Approx. 1	Slings, Shackles
Spent Fuel Cask	2/3	Note 4	Associated Cask Yoke
Fuel Transfer Shield ("Cattle Chute")	2	16.5	Slings, Shackles
Equipment Storage Pool Shield Plugs (four provided)	2/3	37.5 to 39[(3)]	**Slings and Adapters (Dry Lift)** Equipment Storage Pool Shield Plug Strongback **(Wet Lift Rig)**
Dryer/Separator Sling Assembly	2	**3**	Main Hook

* For footnotes, see Sheet 2 of 2.

TABLE 9.1-10
(Sheet 2 of 2)

HEAVY LOADS[1] CARRIED BY THE REACTOR BUILDING CRANE*

Load	Safety [2] Class	Lifting (Tons)	Device
Fuel Storage Pool Shield Plugs (four provided)	2/3	4.5 ea	Slings, Shackles
Plant Equipment	3	less than 20 tons	Slings
New Fuel and Shipping Containers	3	1	Slings
Head Strongback	2	**3.75**[5]	Main Hook
Stud Tensioner Assembly	2	24[6]	Main **Hook**
Equipment Storage Pool Shield Plug Strongback	**2/3**	**3**	**Main Hook**
Dry New Fuel Storage Vault Plugs	**3**	**3**	**Slings**
Auxiliary Work Platform	**3**	**2.75**	**Slings**

1. NUREG 0612 defines a heavy load as one that weighs more than the combined weight of a single fuel assembly and its associated handling tool. For reference, the weight of a spent fuel assembly and its handling tool at Oyster Creek is approximately 800 lbs.

2. Safety Class designations are explained in Table 9.1-9.

3. The top Equipment Storage Pool Shield Plug weighs 39 tons; the remaining three plugs weigh 37.5 tons each.

4. | | |
   |---|---|
   | NAC-1: | 30 Tons, **Base Plate – 4 Tons** |
   | GE-200: | 5 Tons |
   | **TN-9:** | **40 Tons, Base Plate – 3 Tons** |
   | **CN-355** | **29 Tons, Base Plate – 4 Tons** |
   | **FSV** | **24 Tons, Base Plate – 3 Tons** |

5. **Does not include weight of the six turnbuckles, clevises, and shackles that are normally fastened to strongback.**

6. **Includes four tensioners, four trolleys, traction motor, nuts washers, thread protectors, four studs, hydraulic pump, hose manifold, hoses.**

TABLE 9.1-11
(Sheet 1 of 1)

SUMMARY OF CRITICALITY ANALYSIS FOR RACKS CONTAINING BORAFLEX

K_∞ in Core Geometry		1.2763
K_∞ in Spent Fuel Storage Rack		0.8938
Calculational Bias		0.0000
Uncertainties and Tolerances		
Calculational	$\pm$ 0.0024	
Boraflex Thickness	$\pm$ 0.0098	
B-10 Concentration	$\pm$ 0.0053	
Boraflex Width	$\pm$ 0.0013	
Enrichment ($\pm$ 0.05%)	$\pm$ 0.0043	
UO_2 Density ($\pm$ 0.0200)	$\pm$ 0.0023	
Lattice Spacing	$\pm$ 0.0023	
SS Thickness	$\pm$ 0.0008	
Channel Bulge	$\pm$ 0.0038	
Channel Removal	negative	
Statistical Combination	$\pm$ 0.0133	0.0133
Allowance for Uncertainty in Depletion Calculations		0.0100
Allowance for Boraflex Shrinkage and Gap Formation		0.0280
Maximum K_∞		0.9451

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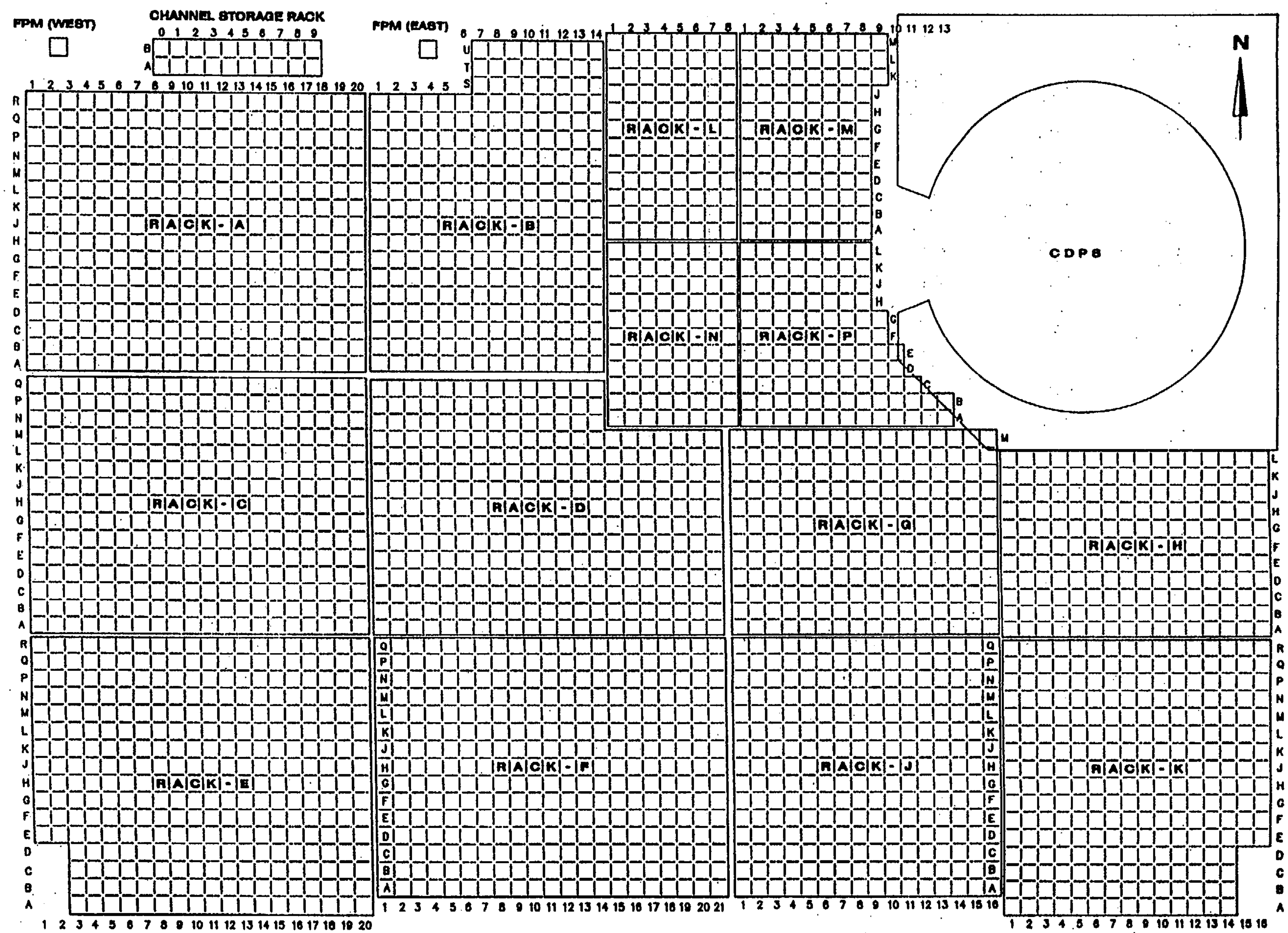
TABLE 9.1-12
(Sheet 1 of 1)

SUMMARY OF CRITICALITY ANALYSIS FOR RACKS CONTAINING BORAL

Temperature assumed for analysis			**4°C**
Fuel Enrichment (average)			**4.6**
Reference CASMO k_∞			
8x8R Fuel			**0.9184**
GE-9B Fuel			**0.9202**
GE-11 Fuel			**0.9206**
GE-11 Fuel (top, w/missing rods)			**0.9232**
Uncertainties			
Removal of flow channel			**negative**
Eccentric assembly location			**negative**
Tolerances			
Boron Loading	**0.0162 ± 0.0012 gm/cm^2**	**±0.0032**	
Boral Width	**5.0 ± 1/16 inches**	**±0.0017**	
Lattice Spacing	**6.106 ± 0.03 inches**	**±0.0013**	
SS Thickness	**0.075/0.0235 ± 0.008 inches**	**±0.0016**	
Fuel Enrichment**	**3.1 ± 0.05% U-235**	**±0.0040**	
Fuel Density	**10.41 ± 0.20 g/cm^3**	**±0.0021**	
Statistical Combination*			**±0.0061**
Uncertainty in Depletion Calculations			**±0.0040**
Statistical Combination			**±0.0073**
Effect of Channel Bulge			**±0.0033**
Allowance for Vendor Calculations			**±0.0100**
Maximum Reactivity			
8x8R Fuel		**0.9317±0.0073 = 0.9390**	
GE-9B Fuel		**0.9335±0.0073 = 0.9408**	
GE-11 Fuel		**0.9339±0.0073 = 0.9412**	
GE-11 Fuel (top, w/missing rods)		**0.9365±0.0073 = 0.9438**	

**Effect of enrichment tolerance is significantly smaller at higher enrichments
*Square root of sum of squares of all independent tolerance effects



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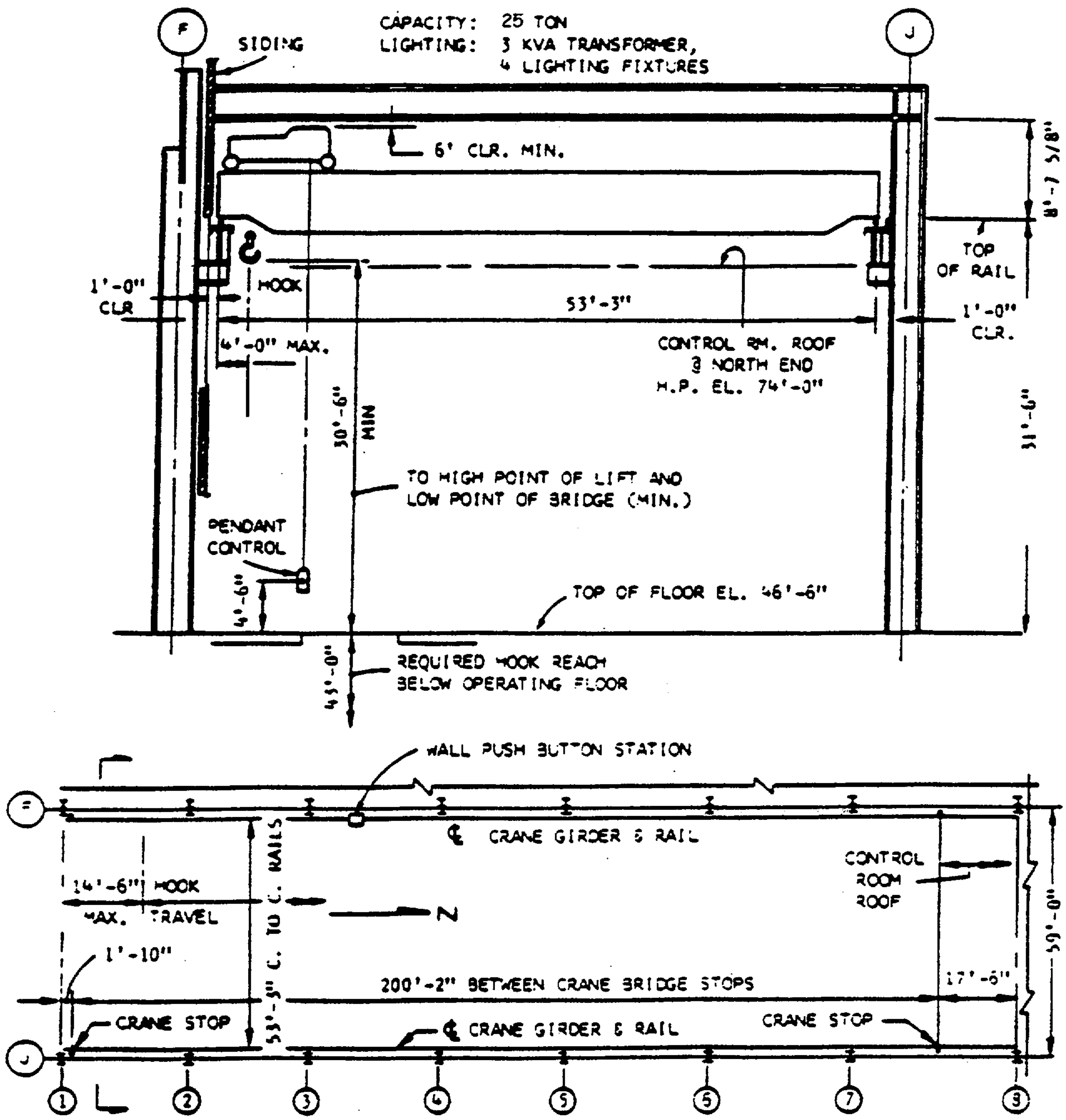
OYSTER CREEK NUCLEAR GENERATING STATION

SPENT FUEL STORAGE POOL

FIGURE 9.1-3



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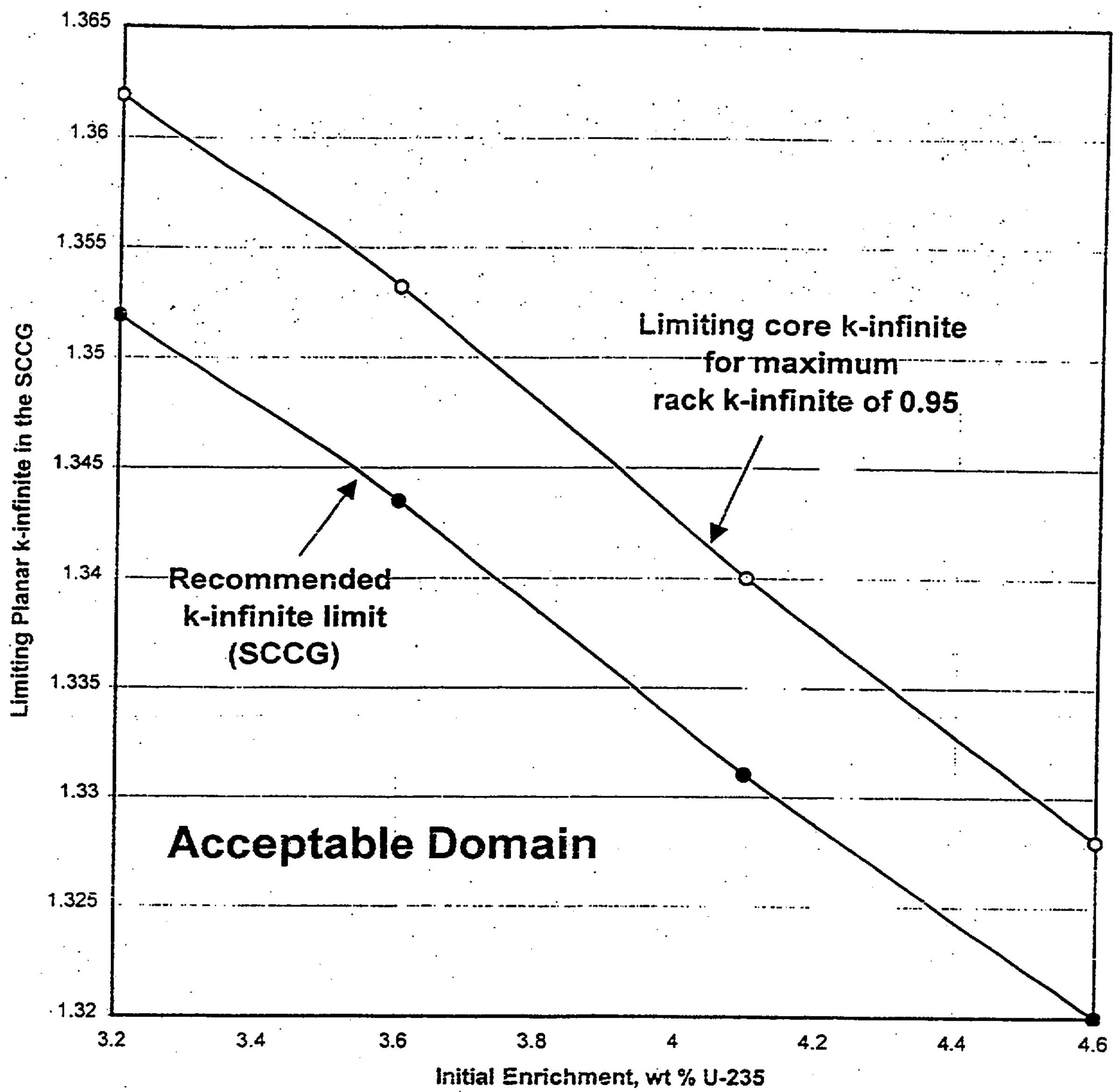
OYSTER CREEK NUCLEAR GENERATING STATION

Heater Bay Crane, Limits of Travel

FIGURE 9.1-17



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OYSTER CREEK NUCLEAR GENERATING STATION

LIMITING K-INFINITE IN THE STANDARD COLD CORE GEOMETRY FOR FUEL ENRICHMENTS BETWEEN 3.2 AND 4.6%

FIGURE 9.1-22

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9.2 WATER SYSTEMS

9.2.1 Station Service Water Systems

9.2.1.1 Service Water Systems

9.2.1.1.1 Design Bases

The Service Water System (SWS) performs the following functions:

a. Provides seawater cooling to the tube side of the two Reactor Building Closed Cooling Water (RBCCW) heat exchangers during normal plant operation.

b. Provides alternate, and/or supplementary seawater cooling to the tube side of the two Turbine Building Closed Cooling Water (TBCCW) heat exchangers.

c. Provides a manual backup supply of seal water to the CWS pumps when their normal seal water supply is unavailable.

d. Maintains the Emergency Service Water (ESW) side of the Containment Spray heat exchangers full.

The SWS was designed to dissipate the expected heat load of the RBCCW System during normal operation and immediately after plant shutdown with 85°F Water System supply. The SWS design temperature and pressure are 85°F and 75 psig, respectively. Operating procedures require that sufficient pump discharge pressure be maintained to prevent pump runout. Inlet water temperature ranges from 30°F to 85°F were utilized for the design of the system.

During the first half hour following reactor scram, for normal shutdown conditions, the RBCCW System heat load is about 180 percent of the heat load during normal power operation. When the water supply is at or below 75°F, one SWS pump and one RBCCW heat exchanger satisfy 100 percent of the RBCCW System cooling requirements for normal full load operation.

9.2.1.1.2 System Description

The Service Water System is included in Drawing BR 2005. Major components of the system are listed in Table 9.2-1. The system is comprised of two pumps, associated piping and valves, two RBCCW heat exchangers, and controls and instrumentation. The TBCCW heat exchangers are also included in Table 9.2-1.

The two 50 percent design capacity service water pumps are located at the intake structure (See drawing 3E-168-02-001, directly downstream of the traveling screens. At least one pump must be in continuous operation during all normal modes of plant operation; both pumps are required to provide the necessary cooling flow during the period following a plant shutdown when the Shutdown Cooling System is being utilized to achieve cold shutdown. During normal plant operation, when the circulating water temperature is sufficiently low, only one pump may be required to operate. During abnormal plant conditions, both SW pumps may be removed from service provided that the heat removal functional requirements are accomodated by either means and are evaluated on a case-by-case basis.

The SWS pumps discharge through check valves and manual block valves into a 20 inch service water main at El. 53' in the Reactor Building. The service water main contains a venturi flow meter located at El. 51'-3". The flow then branches into the two parallel 50 percent design capacity RBCCW heat exchangers, which are located at El. 51'-3" (centerline at El. 58'). Water flow and temperature are controlled manually through manipulation of the RBCCW and SW System valves as provided in the operating procedures.

From the RBCCW heat exchangers, service water flows into a single line which leaves the Reactor Building via a Seal Well located outside of the east wall of the building at El. 33'-6". The Seal Well reduces the head requirements of the SWS by providing a siphon discharge.

The service water and emergency service water (discussed in Section 6.2.2) discharge lines join together with the radwaste discharge line and run underground to the plant's discharge canal.

On the line upstream of the Seal Well there is a set of valves and a blind flange which provide the capability to install a temporary line and divert Service Water from the Seal Well. This feature provides the capability of establishing a temporary flow path in order to inspect the Seal Well and the discharge line.

A two inch line comes off of one the service water discharge lines of the Emergency Service Water (ESW) System pumps (normally shut down), upstream of the Containment Spray heat exchangers, to ensure that the ESW headers are full and ready to perform their emergency function.

The TBCCW heat exchangers, located on the Turbine Building basement floor at El. 10'-0", are provided seawater cooling from either the Circulating Water (CWS) or Service Water systems. The Circulating Water is supplied through a 24 inch line (refer to Subsection 9.2.1.3). If cooling from the CWS is not available, or desired, then a 20 inch crosstie line from the Service Water system may be manually valved into service as another cooling source for the TBCCW heat exchangers. The CWS supply valve must be closed **after** the Service Water system is placed in service. Service water which flows through the TBCCW heat exchangers goes directly into the discharge tunnel.

9.2.1.4.2 System Description

The NRWCCW System is shown in Drawing BR 2006, major component data are presented in Table 9.2-9. The system is comprised of two pumps, two heat exchangers, a surge tank, a chemical feed tank, and associated piping and valves.

The NRWCCW System has been in a stand-by mode since September 1999. The system is only returned to service for periodic sampling or whenever the radwaste evaporator is being used. With the evaporator out of service, the only remaining heat load is the High Purity Waste Collection Tank, HP-T-1A. Processing from this tank will be suspended if tank temperatures exceed the high temperature alarm point or the NRCCW System could be returned to service.

The NRWCCW pumps are located in the New Radwaste Building. Check valves are provided at the discharge of each pump, followed by manually operated butterfly valves. The pumps discharge to a common header which branches to supply cooling water to radwaste equipment. Water flowing from the equipment joins into a return header, passes through the NRWCCW heat exchangers (which are located in the Heat Exchanger Building adjacent to the New Radwaste Building) and goes to the pump suction. The heat exchangers are located at the pump suction to ensure that NRWCCW System pressure is lower than NSW System pressure at their interface. Only one pump and one heat exchanger are required to be in operation when one evaporator condenser is in service.

A chemical feed tank adds an inhibitor for corrosion protection. An elevated surge tank provides a surge volume for system expansion and contraction resulting from temperature changes. The surge tank overflows into the New Radwaste Building floor drain system and vents to the atmosphere. Water from the Demineralized Water Transfer System (described in Subsection 9.2.3.2) is provided for makeup of the surge tank for initial fill and to compensate for system leakage. Makeup to the tank is manually established.

Pressure relief is provided, in accordance with code requirements, through safety relief valves. The cooling water flow through the evaporator condensers is regulated by temperature adjusted flow control valves. All other component cooling flow is adjusted by manually throttling globe valves located downstream of the component being cooled.

Since the NSW System, which constitutes the heat sink for the NRWCCW System, operates at a higher pressure than the NRWCCW pumps, any system leakage will be into the NRWCCW System. This prevents any possible discharge of radioactivity to the environment.

9.2.1.4.3 Safety Evaluation

The NRWCCW System is neither required for the safe shutdown of the reactor nor to mitigate the consequences of postulated accidents. The system is not safety related.

9.2.1.4.4 Testing and Inspection Requirement

No special tests and/or inspections are required for the NRWCCW System beyond normal checks. Periodic samples are taken from the cooling water flow at locations downstream of the Radwaste Concentrator skid and the High Purity Waste Tank cooler to check for the presence of radioactivity and for seawater leakage into the NRWCCW System.

9.2.1.4.5 Instrumentation Requirements

The NRWCCW pumps are controlled by the START-STOP switches located in the New Radwaste Building Control Cabinet. Pump discharge is monitored for low pressure and high temperature. The common "Radwaste Building Trouble" alarm will alert at the Control Room if either of these conditions exist.

An expansion tank-high/low-level alarm is provided to alert of improper tank water level. This condition is annunciated on the Radwaste Building Control Cabinet, and alarmed in the Control Room via the "Radwaste Building Trouble" window. A sight glass, mounted on the surge tank which is located on the New Radwaste Building metal deck at the 56 foot elevation, provides tank level readings.

Closed cooling water discharge temperature from each component being cooled is indicated locally. Table 9.2-10 lists instruments for the NRWCCW System.

9.2.1.5 Turbine Building Closed Cooling Water System

9.2.1.5.1 Design Basis

The Turbine Building Closed Cooling Water (TBCCW) System is designed to provide under normal conditions inhibited demineralized cooling water to Turbine Building equipment and to the Reactor Recirculation Pump MG sets that are not subject to radioactive contamination. When the Reactor
Building closed Cooling Water System is unavailable, and the plant is shutdown, the TBCCWS can provide a cooling water supply to the Augmented Fuel Pool Cooling (AFPC) system heat exchanger.

consequences of postulated accidents. To insure that a loss of power (i.e., single failure) does not prevent RBCCW System isolation, the motor operated valves in series are fed from separate power supplies. If the motor operated supply isolation valve fails to close, the check valve in series would achieve the necessary isolation.

A safety injection signal trips the RBCCW and SW pumps. Then, during operation from the Emergency Diesel Generators, both RBCCW pumps start automatically after a 166 sec time delay and the SWS pumps start automatically after a 120 second time delay, unless a LOCA signal is present. Both heat exchangers remain in service, but the Reactor Water Cleanup System will trip, thus reducing heat load to the system. One, two or three shutdown cooling heat exchangers may be started up manually, depending on the total heat load to the system.

The RBCCW System provides cooling to the Reactor Building corner room spaces, which house Containment and Core Spray pump equipment. Loss of corner room cooling has been evaluated with respect to long term operation of the Containment Spray and Core Spray Systems. As a result of this analysis, the provision of Class I, seismic structures for the RBCCW System and SWS was not required. Seal cooling and gear box lube oil cooling for the Control Rod Drive hydraulic pumps is no longer provided by the RBCCW System, but from Control Rod Drive pump process water.

This conclusion makes it acceptable to trip the RBCCW pumps and SW pump during a LOCA. A safety injection signal trips the RBCCW pump instantaneously, while the SW pump trip is delayed for ten seconds ($\pm$15%) after a safety injection signal. The load shedding feature and long time delay (120 seconds) in restarting the SW pump ensures that the ten second time delay for tripping the SW pump after a safety injection signal will not cause the pump to be inadvertently connected to the Emergency Diesel Generators.

The failure modes and effects analysis of the system is summarized in Table 9.2-19.

Leaks in the RBCCW System which are larger than the capacity of a Demineralized Water Transfer pump would result eventually in inadequate cooling of the components serviced by the RBCCW System. For such leaks, the operator would be made cognizant of the loss of RBCCW cooling capability by not only the low surge tank level alarm, but then also by low RBCCW pump discharge pressure and, additionally, by high temperature alarms from various components serviced by the RBCCW System. These loss of cooling indications would be available to the operator within seconds after loss of RBCCW cooling capability in order to take corrective action.

If it is postulated that there is a leak in the RBCCW System, the surge tank level would drop and should annunciate an alarm in the Control Room. The Demineralized Water Transfer pumps could be manually placed on a diesel powered emergency bus to provide makeup to the RBCCW System. This makeup rate could be as large as 150 gpm. The system could continue to operate

in this manner with no loss in cooling function for a considerable length of time depending upon the RBCCW leak rate and level of water in the 30,000 gallon Demineralized Water Storage Tank. If the tank were full and the RBCCW leak is at a rate of 50 gpm, the RBCCW System would continue to perform its cooling function for at least 10 hours before the makeup water supply was depleted. This allows considerable time to isolate and eliminate the RBCCW System leak through operation of appropriate remote manual or local manual valves, or to repair the system.

If the RBCCW System leak were in excess of 150 gpm, or completely failed, the Shutdown Cooling System would be inoperable. Cooling water would still be available for injection into the reactor by utilizing the Core Spray. Therefore, there is no concern about safe shutdown of the reactor regardless of the size of the RBCCW System break postulated, thus supporting the position that the system is not safety related.

9.2.2.2.4 Testing and Inspection Requirements

The system is continuously used and monitored during reactor operation and shutdown, and is subjected to regular **inservice** inspection **testing** and **a** preventive maintenance program. Maintenance of pumps, coolers, valves and other equipment is done in accordance with manufacturer's recommendations. During normal plant operation and shutdown, all components are accessible for maintenance. Surveillance of the RBCCW System during testing is accomplished with instrumentation which is accessible during normal plant operating conditions. Water samples from the outlet of each RBCCW heat exchanger are available at a local sampling sink. The inservice inspection of pumps and valves is performed in accordance with the requirements of the latest version of the Oyster Creek Inservice Inspection Program.

9.2.2.2.5 Instrumentation Requirements

The RBCCW System instrumentation is presented in Table 9.2-20. The RBCCW pumps are controlled from momentary contact START-STOP switches (spring return to normal) on panel 13R at the Control Room. The switches can be pulled out to lock the pump in STOP and extinguish the indicator lights. Each pump runs continuously when the switch is operated momentarily to START. When the switch is operated to STOP, the pump remains stopped and the green running light on panel 13R remains on.

The surge tank is provided with level controls and alarms on high and low level. It is desired to determine leakage in the RBCCW System in the range of 0.5 to 10 gpm. This is accomplished by metering makeup to the tank. This system is supplied from vital power sources.

The chemical injection tank is provided with a gage glass and a level switch with local **low** level alarm and **CHEM-ADDITION** TANK LOW LEVEL alarm on the **1F/2F** alarm panel.

The WD piping is seismically designed to withstand a horizontal force equal to 0.05 times the operating dead load of the piping. The vertical seismic load was considered zero. The governing code is ANSI B31.1, to a 100-F temperature and 175 psig pressure.

All system piping, with the exception of the piping runs in the New Radwaste Building, is Seamless Aluminum, Schedule 40; fittings are forged aluminum, Schedule 40; butt welds and flanges are forged aluminum, 150 lb Standard, flat face. All demineralized water piping is seamless stainless steel in the New Radwaste Building.

9.2.3.2.2 System Description

The WD System is shown in Drawing BR 2004, major component data are presented in Table 9.2-24. The system is comprised of a 30,000 gallon outdoor storage tank (DWST), two full capacity transfer pumps, and associated piping and valves.

Demineralized water from the MUD System enters the WD System at the DWST. Water is drawn from the tank through the WD pumps and into the system supply header. The six inch pump suction line and the four inch return line to the DWST are provided with locked open manually operated gate type block valves. These valves can be closed to mitigate the effects of leaks in any connecting pipe. The pump suction and return line are isolated from their surroundings by insulating tape wrap, and all flanges and penetrations which could contact dissimilar metals are also insulated. A cathodic protection system for buried piping, consisting of sacrificial anodes, was incorporated for corrosion protection. The DWST is electrically grounded, and is provided with an eight inch overflow line and a drain to the Turbine Building basement.

The four inch demineralized water supply header runs below grade to the Turbine Building. There are system branches which feed the various plant buildings, as shown in Drawing BR 2004.

9.2.3.2.3 Safety Evaluation

The WD System is not required for the safe shutdown of the reactor nor to mitigate the consequences of postulated accidents. The system is not safety related. During the 15R refuel outage, the WD System became contaminated. The system was flushed with flushing still ongoing. The WD System is operating as a partially contaminated system under Safety Evaluation #SE-000523-011 to meet the requirements of IEB 80-10.

The WD System is normally kept in operation at all times. During loss of offsite power, either transfer pump may be started manually and operated from the Emergency Diesel Generators, if there is a demand on the system. There is no need to operate a transfer pump merely to fill the CST.

9.2.3.2.4 Testing and Inspection Requirements

No special tests and/or inspections are required for the WD System beyond normal checks.

9.2.3.2.5 Instrumentation Requirements

There are high and low level switches on the DWST for a common high/low level annunciator on alarm panel 7F in the Control Room and one wide range level transmitter. Each transfer pump is operated from a START-AUTO-STOP switch on Control Room panel 13R. A pressure switch on the discharge header starts the standby pump automatically if discharge pressure drops to 130 psig. The following annunciators are provided on alarm panel 7F:

a. Common DEM TRANS PUMPS TRIP alarm for both pumps

b. DEM TRANS PUMPS BOTH RUNNING

Locally, there are pressure taps at the suction and discharge of each pump and a pressure gage on the discharge header.

9.2.4 Potable and Sanitary Water Systems

9.2.4.1 Well and Domestic Water System

9.2.4.1.1 Design Bases

The Well and Domestic Water (MD) System serves the following functions:

a. Provides all potable water used on site, including the Administration Office Building and the Maintenance Building demand.

b. Provides pretreated water to the MUD System (described in Subsection 9.2.3.1).

Typical raw water quality is presented in Table 9.2-25.

9.2.4.1.2 System Description

The MD System is shown in Drawing 3E-871-21-1000. The pretreatment system was replaced by a trailer mounted pretreatment system. **The Pretreatment trailer provides filtered raw water and operates in conjunction with Deep Well Pumps.** Major component data are listed in Table 9.2-26.

Fresh raw water is drawn from the 400 foot, 12 inch diameter deep well which contains two 100 gpm submersible pumps. One pump is located 50 feet below grade and the other 60 feet below grade. Pump selection is made manually at the well pad in the storage yard to the east of the

Emergency Diesel Generators. The selected pump operates automatically on demand and is controlled by a MANUAL-OFF-AUTO switch on the 13R panel in the Control Room. The State of New Jersey limits the amount of water removed from the well to 100,000 gallons per day.

The Deepwel Pumps are used to supply raw water to the pretreatment trailer and the fill the clearwell Tank. The pretreatment trailer filters the raw water supply by the deepwell pumps for subsequent storage in the clearwell tank. One of two deepwell pumps is manually selected at the pump pad and required to maintain a minimum reservoir capacity in the clearwell tank. The clearwell tank satisfies station demands for domestic water and makeup water. The carbon filter and the Demineralizer Trailer operate in conjunction with the filter water pumps to purify makeup water from the clearwell tank and store it in the Demineralized water storage tank. The filter water pumps are required to supply water to the domestic water and makeup demineralizer systems.

The clearwell receives the hypochlorinated, clarified and filtered water and has a volume of **9500** gallons. **The clearwell tank maintains required retention and provides sufficient suction head pressure (NPSH) to support filtered Water Pump operation at the minimum clearwell tank operating level.**

Three filtered water pumps remove the water from the clearwell. One pump operates as required to maintain the proper level in the Domestic Water Tank. The other two pumps are not automatic and operate in association with the MUD System. An interlock is installed for these three pumps to prevent lowering the level in the clearwell below the low level alarm. The clearwell level provides the necessary NPSH for the filtered water pumps. **The hypochlorinator pump operates automatically only when the 1-3 filtered water pump is operating in the domestic position.**

A soda ash injection system has beeen installed, which injects soda ash at the inlet to the domestic water tank (T-10-4). This system will neutralize the pH to reduce corrosion and satisfy potable water standards.

9.2.4.1.3 Safety Evaluation

The MD System is not required for the safe shutdown of the reactor nor to mitigate the consequences of postulated accidents. The system's water does not come in contact with radioactive contamination.

During failure of normal auxiliary power, the deep well pumps and the filtered water pumps do not restart automatically, The domestic water tank has enough reserve capacity for anticipated requirements. Operating power is available from the emergency generators so that the system, or parts of it, may be placed in operation, if necessary, during prolonged power failure.

When one of two of the filter water pumps which service the MUD System are out of service, the third pump can be used for that purpose if demand for demineralized water is heavy. In this case, there will be no automatic pump start on low Domestic Water Tank level, although makeup water will be admitted normally as long as the pumps are operating. Local action is required to transfer a pump to domestic water service.

9.2.4.1.4 Testing and Inspection Requirements

No special tests and/or inspections are required for the WDW System beyond normal checks. Sampling provisions are incorporated in the raw water influent line.

9.2.4.1.5 Instrumentation Requirements

A WELL PUMP TRIP annunciator is provided at panel 7F in the Control Room. The deep well pumps mode of operation is controlled from a three position selector switch on Panel 13R in the Control Room. The positions are MANUAL-OFF-AUTO. **The Deepwell pumps operate in an "ON-OFF" control mode activated by the clearwell low and high level switches, respectively.**

All three Filtered Water Pumps are controlled from the MUD Panel in the Turbine Building basement. Pumps No. 1 and No. 2 can only be run in manual. Pump No. 3 can be run in manual or automatic; while in the automatic mode the pump will cycle on Domestic Water Tank level. The soda ash injection system also operates in conjunction with the No. 3 pump and domestic water tank level. Any of the three pumps will trip on low suction pressure, thus insuring that proper NPSH is maintained. When clearwell level is raised and pressure restored above the trip setpoints, the pumps selected to run will restart.

The Pretreatment Building sump sludge pumps are also controlled from the MUD Panel. The pumps are controlled by three position hand switches labeled HAND-OFF-AUTO. In AUTO, a float switch in the sump starts and stops the pumps to control sump level. Pump logic is set up so that the pumps alternate to equalize run time. This run time is shown on integrators located in the Pretreatment Building.

A HAND-OFF-AUTO switch in the MUD Panel controls the hypochlorination pump. In AUTO, the pump runs any time the well water pumps run.

9.2.4.2.5 Instrumentation Requirements

In the original domestic water system, Filtered Water Pump No. 3 of the DWD System is controlled by two level switches monitoring Domestic Water Tank level. The high level switch stops the pump when the tank is full, and the low level switch starts the pump when the tank is half full. A third, low-low level, switch alarms DOMESTIC WATER TANK LO-LEVEL at the 7F alarm panel in the Control Room when level drops below the setpoint.

A pneumatic controller monitors tank level and develops an air signal to open and close the tank's level control valve. Tank pressure is maintained by air loading at about 50 psig.

The deep well pump is automatically controlled by two level switches monitoring the New Domestic Water Tank level. The high level switch stops the deep well pump and the low level switch starts the deep well pump when the tank is one third full. A **local** alarm **activated** for the following conditions: High-high pressure (>90 psi), low-low level (<29 inches from bottom of the tank), deepwell pump motor temperature high trip or high current trip or low current trip.

9.2.4.3 Sanitary Waste System

9.2.4.3.1 Design Bases

The Sanitary Waste System is designed to collect all plant sanitary drains and direct them to a controlled collection point.

9.2.4.3.2 System Description

All sanitary drains in the Office Building and Turbine Building, including floor drains in the Office Building, are combined in a six inch line. There are separate four inch drain lines from the various plant buildings which join the main sanitary drain line. All piping is designed for a maximum temperature of 100°F and a maximum pressure of 35 psig and is heat traced where exposed. All piping, valves and supports are designed as nonseismic, and underground piping is backfilled to provide adequate support and protection. PVC sanitary waste line is supported in accordance with the Basic Plumbing Code.

Domestic waste water from all plant locations enters an 8'-6" diameter by 15' deep concrete equalizing tank, via the six-inch sanitary collection main. The equalizing tank is located under the parking lot near the main entrance to the plant. This tank discharges, through two self priming two inch diaphragm pumps, to the Lacey Municipal Utilities Authority Sewer System and subsequently to the Ocean County Utilities Authority regional collection system via an eight inch gravity line. Valves are provided to control flow.

A radiation monitoring system has been provided to continuously monitor radiation levels in the effluent at the transfer pumps. The system continuously indicates and records radiation levels. The radiation monitoring system consists of a scintillation detector, a photo multiplier tube and a preamplifier, all mounted inside a drywell located in the sewage lift pump station. The ratemeter, recorder, power supply and supporting relay logic are located in the RAGEMS Building. Power is derived from local lighting power. As a backup, manual samples may be taken from the sewage pit for laboratory analysis.

9.2.4.3.3 Safety Evaluation

The Sanitary Waste System is not required for the safe shutdown of the reactor nor to mitigate the consequences of postulated accidents.

9.2.4.3.4 Testing and Inspection Requirements

No special tests and/or inspections are required for this system beyond normal checks. The radiation detector requires annual calibration and a quarterly response check.

9.2.4.3.5 Instrumentation Requirements

The 480 volt, 2 hp, sewage transfer pumps are controlled by equalizing tank level normally in an automatic mode. The pumps can be operated individually or in parallel as per the volume of fluid in the tank. Manual operation of either pump can be selected. Audio and visual alarms for loss of power to either pump or tank high level are provided.

The continuous radiation monitor alarms below 50 percent of the 10CFR20, Appendix B, Table 1, Column 2, value for Co-60. If radiation is detected, an alarm is initiated in the Security Building. Procedures require that the Control Room be immediately notified and that the alarm be investigated. If levels continue to rise, the sewage lift pumps trip automatically below the 100 percent value of 10CFR20.

9.2.5 Ultimate Heat Sink

9.2.5.1 Design Bases

The ultimate heat sink is intended to dissipate waste heat from the plant during normal, shutdown and accident conditions.

TABLE 9.2-2
(Sheet 1 of 1)

SERVICE WATER SYSTEM LINE INSTRUMENTATION AND ALARMS

Instruments

1. Pressure gage at pump 1-1 discharge

2. Pressure gage at pump 1-2 discharge

3. **Pressure transmitter transmits service water pressure to indicator on panel 5F/6F**

4. Differential pressure gage at tube side of the TBCCW heat exchanger 1-1.

5. Differential pressure gage at tube side of the TBCCW heat exchanger 1-2.

Alarms

1. Service Water Pump Trip - Panel 5F/6F

2. Liquid Process High Radiation - Panel 10F

TABLE 9.2-8
(Sheet 1 of 1)

DESIGN BASIS LOADS FOR THE NEW RADWASTE CLOSED COOLING WATER SYSTEM

Component Cooled	Quantity	Normally Operating	NRWCCW Flow Rate (each component)
Evaporator Condenser(1)	2	1	1200 gpm
Condensate Cooler (1)	2	1	600 gpm
Vent Condenser (1)	2	1	56 gpm
High Purity Waste Tank Cooler	1	1	60 gpm
High Purity Waste Pump (2)	2	1	6.5 gpm
Chem Waste/Floor Drain Pump (2)	2	1	6.5 gpm
Concentrated Liquid Waste Pumps (2)	2	1	3.5 gpm
Total Normal Operating Flow			1940.5 gpm

Note (1)
One evaporator and its associated components are abandoned.

Note (2)
During D&D's Mechanical verification effort in 1986, these cooling sources were found to be abandoned. These components are now cooled by the Condensate Transfer System.

TABLE 9.2-18
(Sheet 1 of 2)

MAJOR COMPONENTS OF THE REACTOR BUILDING CLOSED COOLING WATER SYSTEM

Reactor Building Closed Cooling Water Pumps

Number of Pumps	2
Type	Single stage, double suction, horizontally split-volute centrifugal
Design TDH	175 feet
Design Capacity	3500 gpm
Bhp at Design Point	176
NPSH	50 feet
Shutoff Head	226 ft
Minimum Flow	30 gpm
Materials	
Casing	Cast iron
Impeller	Bronze
Shaft	Steel SAE 1035
Wear rings, casing and impeller	Bronze
Bearings Type	Ball
Motor	
Horsepower	200
Speed	1800 rpm
Full-Load Current	237 amp
Service Factor	1.15
Bearings Type	Ball
Power Requirements	460 volt, 3 phase
Power Supply	1-1 : US 1A2 1-2 : US 1B2
Shaft Coupling	Fast, gear type

Reactor Building Closed Cooling Water Heat Exchanger

See Table 9.2-1

TABLE 9.2-18
(Sheet 2 of 2)

MAJOR COMPONENTS OF THE
REACTOR BUILDING CLOSED COOLING WATER SYSTEM

Reactor Building Closed Cooling Chemical Feed

Mixing Tank	
Capacity	100 gallons
Type	Vertical, cylindrical, bottom dish head and removable top cove
Mixer	**Portable Type with GE Motor ¼ - 1-60-115/230-1725 TEC**
Solution Pump	
Pump Type	Simplex diaphragm, controlled volume
Capacity	Adjustable : 0 to 5 gph at 30 psig
Check Valves	**Ohio Injector Co., Body C.S. with stellite; Disc SS**
Rupture Disk	**Safety Systems - Aluminum**
Coupling	Direct Drive
Motor Type	TEFC, 1750 rpm
Motor Horsepower	1/4

Reactor Building Closed Cooling Water Surge Tank

Capacity	Approximately 500 gallons
Dimensions	**3 ½ feet** diameter 8 feet long
Material	Carbon Steel Plate, 3/8 inch thick
Design Pressure	15 psig

TABLE 9.2-19
(Sheet 1 of 5)

REACTOR BUILDING CLOSED COOLING
WATER SYSTEM FAILURE MODES AND EFFECTS ANALYSIS

Component	Failure Mode	Consequences
RBCCW Heat Exchangers (2)*	Emergency shutdown	High temperature at the non-regenerative heat exchanger discharge trips the cleanup system. Shutdown cooling is regulated to limit temperature rise in the RBCCW System. No damage to the reactor occurs.
	Internal Leakage	Loss of demineralized water into the SWS, detectable by surge tank level and makeup rate.
RBCCW Pumps (2)* Shutdown.	Pump trip or malfunction	Same as for heat exchanger
	Loss of power supply	The 200 hp pump motors are fed from two separate 460 volts AC buses that are automatically connected to the diesel generators in the event of normal auxiliary power failure.

* Loss of RBCCW during plant operation causes loss of drywell cooling. Entry to plant Emergency Operating Procedures may result.

Note: Internal and external leakage also gets detected by the radiation levels indicated by the RBCCW and/or Service Water process radiation monitors.

TABLE 9.2-19
(Sheet 2 of 5)

REACTOR BUILDING CLOSED COOLING
WATER SYSTEM FAILURE MODES AND EFFECTS ANALYSIS

Component	Failure Mode	Consequences
RBCCW Surge Tank (1)*	Tank Failure	This tank does not have a major role during normal system operation, however, it must operate to detect and compensate for leakage.
RBCCW Chemical Treatment Equipment	System Failure	Operation of this system is intermittent and not necessary for RBCCW System operation.
Fuel Pool Coolers (3)	Internal Leakage	Flow of fuel pool water into the RBCCW System until detection by high surge tank level and isolation of leaking cooler. Higher than normal fuel pool temperature might result while the failed cooler is isolated.
Shutdown Cooling Heat Exchangers (3)	Internal Leakage	Flow of reactor water into the RBCCW System until detected by high surge tank level. Isolation of the failed heat exchanger would extend the reactor cooldown period.

* Loss of RBCCW during plant operation causes loss of drywell cooling. Entry to plant Emergency Operating Procedures may result.

Note: Internal and external leakage also gets detected by the radiation levels indicated by the RBCCW and/or Service Water process radiation monitors.

TABLE 9.2-19
(Sheet 3 of 5)

REACTOR BUILDING CLOSED COOLING
WATER SYSTEM FAILURE MODES AND EFFECTS ANALYSIS

Component	Failure Mode	Consequences
Shutdown Pump Coolers (3)	Loss of Cooling	Reactor cooldown period will be extended.
Reactor Water Cleanup System*		
a. Non-regenerative heat exchanger	Internal Leakage	Flow of reactor water into the RBCCW System until detected by surge tank level. Results in isolation and loss of the RWCU System.
b. Recirculation Pump Coolers	Loss of Cooling	Reduced cleanup capacity of 50% of design value.
c. Auxiliary Pump Cooler	Loss of Cooling	The pump would be shutdown (if in use), at low reactor pressure the cleanup system would not be available.
d. Precoat Pump Cooler	Internal Leakage	Flow of filter aid into the RBCCW System. Significant degradation of heat removal capacity could lead to pump cavitation after prolonged operation.

* Depending on the failure, the reactor may be operated from 12 hours to seven days without RWCU System before shutdown is required.

Note: Internal and external leakage also gets detected by the radiation levels indicated by the RBCCW and/or Service Water process radiation monitors.

TABLE 9.2-19
(Sheet 4 of 5)

REACTOR BUILDING CLOSED COOLING
WATER SYSTEM FAILURE MODES AND EFFECTS ANALYSIS

Component	Failure Mode	Consequences
Tunnel Recirculation Fans	External Leakage	Flow of RBCCW System water into the pipe tunnel resulting in higher than desired ambient temperatures in the tunnel during operations.
Core Spray Pump Compartment Coolers (2)	External Leakage	Flow of RBCCW System water into the Reactor Building, prompting cooler isolation and increase in ambient temperatures.
Containment Spray Pump Compartment Coolers (2)	External Leakage	Same as above.
Reactor Building Equipment Drain Tank (1)	External Leakage	Flow of RBCCW System water into the tank, causing excessive cycling of the drain pump and/or high tank level alarms.
Reactor Water Recirculation Pump Coolers (5)	Loss of Cooling	Temperature instrumentation would alarm. Pump would be shutdown and reactor power level reduced.
	External Leakage	Flow of RBCCW System water into the motor lube oil causing thrust bearing failure.

Note: Internal and external leakage also gets detected by the radiation levels indicated by the RBCCW and/or Service Water process radiation monitors.

TABLE 9.2-19
(Sheet 5 of 5)

REACTOR BUILDING CLOSED COOLING
WATER SYSTEM FAILURE MODES AND EFFECTS ANALYSIS

Component	Failure Mode	Consequences
	Internal Leakage	Flow of reactor water into the RBCCW System through pump seals and bearings.
Drywell Heating and Ventilation Coolers (5)	Loss of Cooling to two or more units	Higher than normal drywell temperatures
	External Leakage	Flow of RBCCW System water into drywell.
Drywell Equipment Drain Tank	External Leakage	Flow of RBCCW System water into the tank, causing excessive cycling of the drain pump and/or high tank level alarms. Or, flow of RBCCW System water into drywell sump and higher than normal tank temperature.

Note: Internal and external leakage also gets detected by the radiation levels indicated by the RBCCW and/or Service Water process radiation monitors.

TABLE 9.2-20
(Sheet 1 of 3)

REACTOR BUILDING CLOSED
COOLING WATER SYSTEM INSTRUMENTATION

Parameter	Indication	Instrument
RBCCW Heat Exchangers		
CCW Inlet Temperature (2)	Local	TI-43, TI-44
CCW Inlet Pressure(2)	Local	PX-71, PX-72
CCW Outlet Temperature(2)	Local/Recorder	TX-45, TX-46
CCW Outlet Pressure (2)	Local	PX-57, PX-58
CCW Discharge Header Temperature	Control Room	TE-43
RBCCW Surge Tank		
Water Level	Local	LI-36
Makeup Water Flow	Local/Recorder	FT-12-287
RBCCW PUMPS		
Suction Pressure	Local	PI-52
Suction Temperature (2)	Local	TI-541-9, T1-541-10
Discharge Pressure (2)	Local	PI-50, PI-51
Discharge Flow (2)	Local	FE-5-1, FE-5-2
Pump Trip	Remote	HS-5

TABLE 9.2-20
(Sheet 2 of 3)

REACTOR BUILDING CLOSED
COOLING WATER SYSTEM INSTRUMENTATION

Parameter	Indication	Instrument
RBCCW Chemical Treatment		
Tank Level	Local	LI-35 (Sight Glasss) LS-507 (Low Level)
Shutdown Cooling Heat Exchangers		
CCW Combined Outlet Temperature	Control Room	TE-45
Flow Control Valve Position	Control Room	RV-17
CCW Differential Pressure (3)	Local	dPI-57 dPI-58 dPI-541-2
Spent Fuel Pool Coolers		
CCW Outlet Temperature (3)	Local	TI-541-6 TI-541-8 TI-5-268
CCW Differential Pressure (2)	Local	dPI-541-4 dPI-541-3
Reactor Water Cleanup System Equipment		
Non-regenerative Heat Exchangers-CCW Outlet Temperature	Local	TE-188
Non-regenerative Heat Exchanger-CCW Flow	Local	dPI-506

TABLE 9.2-20
(Sheet 3 of 3)

REACTOR BUILDING CLOSED
COOLING WATER SYSTEM INSTRUMENTATION

Parameter	Indication	Instrument
RBCCW In-Line Filter		
Inlet Pressure	Local	PI-12
Outlet Pressure	Local	PI-11
Drywell Equipment		
CCW Inlet Temperature	Control Room	TE-108
Recirculation Pump Seal Cooler-CCW Outlet Flow Switches (5)	Local	FS-210-32A FS-210-32B FS-210-32C FS-210-32D FS-210-32E
Recirculation Pump Motor-CCW Outlet Temperature (5)	Control Room	TE-210-33A TE-210-33C TE-210-33E TE-210-33G TE-210-33J
Recirculation Pump Seal Cooler-CCW Outlet Temperature (5)	Control Room	TE-210-33B TE-210-33D TE-210-33F TE-210-33H TE-210-33K
Drywell CCW Outlet Temperature	Local	TE-56
Drywell Combined Differential Pressure Used as Flow Indication	Local	DPI-541-1130
RBCCW System Isolation (V-5-147, V-5-166, V-5-167)	Control Room	N/A

Figure 9.2-11

(Deleted)

Purge streams of sample lines, and contents of the sample station sink, are discharged to the New Radwaste Building floor drains. Local sample collection points identified earlier are provided for infrequently sampled points in lieu of routing to the sampling station.

The Post Accident Sampling System is described in Section 11.5.

9.3.2.3 Safety Evaluation

The Process Sampling System is not required for the safe shutdown of the reactor. The reactor liquid sampling line which is routed outside containment is provided with isolation valves which are part of the Reactor Coolant Pressure Boundary (RCPB). These valves are required to operate in order to mitigate the consequences of an accident. The remainder of the system is not safety related and all tubing is nonseismic.

9.3.2.4 Testing and Inspection Requirements

The reactor liquid sampling line is tested as part of the RCPB during the primary containment leak rate tests.

No special tests and/or inspections are required for the remainder of the system beyond normal checks.

9.3.2.5 Instrumentation Requirements

Instrumentation associated with sampling equipment is discussed in Section 11.5. For the Offgas Building, switches are provided locally for the sample pump and the sample hood exhaust fan, and compound pressure/vacuum gages are provided to allow estimates of sample sizes on sample lines. For the New Radwaste Building, a switch operates the sample hood exhaust fan, and bimetallic temperature indicators are provided to measure sample temperature upstream of the collection cylinders.

9.3.3 Equipment and Floor Drainage Systems

9.3.3.1 Design Bases

Floor drains, equipment drains, roof drains and sanitary drains are provided to collect normal plant liquid effluents and route the collected water either to radwaste treatment, sewage treatment or discharge. The Equipment and Floor Drainage System has been designed to handle large volumes of fluids resulting from spills, maintenance activities, system flushing, rinsing operations and occasional decontamination work.

The sumps and drain tanks **and their pumps** are adequately sized to accommodate expected volumes of water in order to minimize the potential for flooding.

9.3.3.2 System Description

The Equipment and Floor Drainage System (EFDS) is shown in Drawings 3D-151-07-001, 3D-153-07-001, 3D-154-07-001, 3D-155-07-001, ED-576-07-001 and Figure 9.3-11. Major component data are presented in Table 9.3-4. The system collects water from floor drains and equipment drains from plant buildings, structures and components. The Chemical Waste System referred to in this subsection is described in Section 11.2.

9.3.3.2.1 Turbine Building Floor and Equipment Drains

Floor and equipment drains are collected in five sumps in the Turbine Building basement; then pumped to the New Radwaste Building Chemical Waste System or to the discharge canal, depending upon origin. Wastes from the regeneration systems are collected in a two compartment regeneration system waste tank. High conductivity waste is pumped to the New Radwaste Building Floor Drain Collector tank; low conductivity waste is pumped to the High Purity Waste Collector Tank.

Sump drainage is used for all floors in the Turbine Building to simplify the piping and permit positive control over discharge to the environment. Since the basement floor is at mean sea level, all drains from this floor must be pumped. Drains from controlled areas are pumped to the New Radwaste Building for treatment except for those draining into sump 1-5, which may be released after sampling.

Potentially contaminated overflow lines and drains from the Condensate and Demineralized Water Storage Tanks are run below grade through a 12 inch line terminating 36 inches above the basement floor in the main condenser area. A three way valve in the valve pit next to the Condensate Storage Tank can be operated to route the floor drains from the two valve pits and the outdoor pump pad to the Turbine Building basement, if they become contaminated (normal route). The high level of the storage tanks is set well below the overflow pipe so that overflow is not expected during normal operation. If the tanks must be drained, the drain valves are partially opened so that discharge does not exceed sump pump capacity.

All drains from controlled areas (except for those draining into sump 1-5) and from the regeneration waste tank low conductivity compartment, are pumped to the High Purity Waste Collector Tank in the New Radwaste Building. The regeneration waste tank high conductivity compartment is processed and directed to the Floor Drain Collector Tank. The contents of sump 1-5 can be pumped to the **High Conductivity Tank or sampled for radioactivity prior to overboard discharge.**

All sump pumps and waste transfer pumps are duplex, 100 percent capacity, and arranged for intermittent, alternate operation, except for the sump 1-1 pumps which are controlled manually.

TABLE 9.3-4
(Sheet 3 of 10)

EQUIPMENT AND FLOOR DRAINAGE SYSTEM COMPONENTS

Low Conductivity Tank (Continued)

Areas Drained	(1) Regeneration tanks - if conductivity is 50 micromhos
Alarms	Low Conductivity Tank High Level;Low Conductivity Tank Valve Closed

Sump No. 1-1 Pumps

Quantity	2
Type	Vertical sump pump
Power Sources	MCC 1A12, MCC 1B12 (460 volts, 3 phase)
Controls	Local at sump
Discharge	To 1-3 sump

Sump No. 1-2 Pumps

Quantity	2
Type	Submersible sump pump
Power Sources	MCC 1A12, MCC 1B12 (460 volts, 3 phase)
Controls	Located in Feedpump Room
Discharge	To Chemical Waste System

Sump No. 1-3 Pumps

Quantity	2
Type	Submersible **sump** pump
Power Sources	MCC 1A11, MCC 1B11 (460 volts, 3-phase)
Controls	Local sump
Discharge	To Chemical Waste System

TABLE 9.3-4
(Sheet 4 of 10)

EQUIPMENT AND FLOOR DRAINAGE SYSTEM COMPONENTS

Sump No. 1-4 Pumps

Quantity	2
Type	Submersible sump pump
Power Sources	MCC 1A11, MCC 1B11 (460 volts, 3-phase)
Controls	Local at sump
Discharge	To Chemical Waste System

Sump No. 1-5 Pumps

Quantity	2
Type	Vertical Sump Pump
Power Sources	MCC 1A11, MCC 1B11, (460 volts, 3 phase)
Controls	Local at sump
Discharge	High Conductivity Tank, sampled for radioactivity prior to overboard discharge.

Low Conductivity Tank Waste Transfer Pumps

Quantity	2
Type	End suction, center line discharge, open impeller, centrifugal
Power Sources	MCC 1A11, MCC 1B11 (460 volts, 3 phase)
Controls	Local at tank
Discharge	To High Purity Waste System

Radioactivity of the plume is carefully monitored to meet the standards on radioactivity release. Refer to Section 11.5.

9.4.2.3 Safety Evaluation

The RBHV System is normally operated with two supply fans running and one in reserve. The supply fans trip and lock out, and the isolation valves close on any of the signals listed in Subsection 9.4.2.2.1. The emergency exhaust system starts automatically and the supply fans cannot be restarted until the emergency situation is corrected . The Reactor Building, which has a potential for release of radioactivity, is a sealed structure with the interior maintained slightly below atmospheric pressure by the RBHV System so that any leakage will be in rather than out. Negative pressure is created by the exhaust fans in the air ducts to the stack. **The** inlet **and vortex** dampers of these fans **are manually opened to ensure that a negative pressure will be maintained**.

The SGTS also maintains a negative pressure in the Reactor Building during an emergency condition. (Section 6.5)

9.4.2.4 Inspection and Testing Requirements

Testing of the shutdown and isolation capability of the RBHV System is conducted in accordance with plant technical specifications to ensure secondary containment integrity can be maintained following a design basis accident. (Refer to Section 6.2.3.4).

There are no special inspection and testing requirements for the Reactor Building Ventilation System, **with the exception of secondary containment isolation valves and their associated accumulations. A preventive maintenance and surveillance test program has been implemented in response to NRC Generic Letter No. 88-14.** The system is normally operating and any failure is easily detected.

9.4.3 Turbine Building Heating and Ventilation

9.4.3.1 Design Bases

The Heating and Ventilating (H&V) systems in the Turbine Building have two objectives:

a. To protect equipment and personnel from temperature extremes

b. To regulate the static pressure within certain areas of the plant so as to minimize the spread of airborne radioactive contamination and to provide safe disposal of airborne contaminants.

These HVAC systems are designed to meet the requirements of specific areas:

a. Areas normally not contaminated, but which can become so through system malfunction or equipment breakdown. Ventilation for most of these areas is once-through.

b. Areas which will probably contain radioactive contaminants at all times during plant operation. Such areas use once-through ventilation with all air exhausted to the main plant stack.

At the site, outside ambient temperatures as high as 105°F and as low as (-)13°F have been experienced. Since these temperatures are seldom reached, and then only for brief periods, the H&V systems were only designed to cope with a high temperature expected to be exceeded 2.5 percent of the time in summer, and a low temperature expected to be exceeded 2.5 percent of the time in winter. These conditions correspond to a maximum design dry bulb temperature of 89°F (77°F wet bulb) and a minimum design dry bulb temperature of 10°F.

9.4.3.2 System Description

The Turbine Building H&V systems are included in Drawing BR 2009 (also see Table 9.4-2). The figure includes normal flows for the systems. The building is served by five sets of fans: three sets are provided with air washers and heating coils, these distribute flow through most areas of the building; one set serves the feedwater and condensate pump area, and is dedicated to that area. Another system, serving the reheater ventilation, system, is a specialized system also shown in Drawing BR 2009.

The south end of the Turbine Building is served by two supply fans. This supply system is typical of the evaporative cooling systems used in the building. It consists of an air washer and steam fed heating coil and two full capacity supply fans in parallel. Air flow is directed from normally clean areas to contaminated areas and then to the stack. **The static pressure in the Hi/Low Conductivity, Demineralizer, Regeneration, Steam Jet Air Ejector and Mechanical Vacuum Pump Rooms are to be maintained at a negative pressure relative to outside atmospheric conditions.** Normally only one supply fan is operating and the dampers on the other are closed to prevent diversion of air. Steam is admitted to the heating coils from the auxiliary boiler as required to maintain a minimum air temperature of 50°F during cold weather when the turbine generator is shut down. The sump is drained automatically to prevent freezing. The spray pump in the air washer turns on automatically to provide evaporative cooling.

The 'C' Battery Room and the 'C' and 'D' 4160 V Switchgear Vaults are also provided with ventilation systems.

The 'C' Battery Room ventilation system consists of two exhaust fans. The 'C' Battery Room ventilation system is designed to maintain temperature of the batteries within preferred environmental conditions, and prevent the buildup of hydrogen. One exhaust fan is

normally operating when the batteries are in service and the other fan is placed in standby. During the summer period, outside air is drawn through a louver and filter. During the winter period, a minimum amount of outside air is drawn in and mixed with air from the 4160V Switchgear Room. The air is exhausted to the outside atmosphere. An electric heater is also provided to maintain minimum temperature in the area during the winter months.

The 4160 V switchgear area located in the mezzanine south end receives air from the South End supply fans. The 'C' and 'D' 4160 V Switchgear are located in separate vaults. Each vault is ventilated by one roof ventilator fan. Each fan continuously circulates air drawn from the mezzanine south end area and discharges the air back to this area.

The operating floor is served by one air washer with two full capacity supply fans. The Turbine Building operating floor atmosphere is exhausted through four roof openings that are connected by an exhaust duct that runs on the building roof and down to the exhaust fan located on the west mezzanine roof. The inlet damper on the exhaust fan is controlled by a differential pressure controller to maintain a negative pressure in the operating floor space with respect to outdoor air pressure when this fan is running.

The effluent from the Turbine Building operating floor as well as the exhaust from reheater protection, miscellaneous equipment and lube oil equipment areas are monitored for radiation at the common exhaust duct enroute to the atmospheric discharge through a stack located on the west mezzanine roof. This stack is monitored for radiation. The operating floor exhaust fan is only required to be operated when the temperature just below the roof of the Turbine Building is above 130°F. The exhaust fan is operated in unison and interlocked with the supply fans to assure that the pressure within the operating floor is maintained at atmospheric pressure or below. If the radiation level is above a preset level, the supply and exhaust fans can be shut down manually. Automatic inlet louvers on the operating supply fan open only when the high level exhaust fan is operating.

The operating floor receives an inflow of air from the south-end mezzanine through the open equipment hatches as well as the supply ducts on the west wall of the operating floor. The operating floor also receives a portion of air from the reheater area ventilation system supply fans, and a small amount of flow from the Contaminated Instrument Shop. Supply air for the condenser area and heater bay comes from the operating floor. The condenser and heater bay areas are maintained **at a negative pressure relative to outside atmospheric conditions.**

The Contaminated Instrument Shop is provided with a recirculation fan. The recirculated air is filtered to remove potential contamination. A small amount of flow is diverted to the operating floor. The air flow diverted from the Contaminated Instrument Shop maintains the area at a slightly negative pressure.

The highly contaminated areas that are required to be maintained at a negative differential pressure relative to outside atmospheric conditions collectively are identified as the <u>Turbine Building Ventilation Envelope</u>. The Turbine Building Ventilation Envelope is defined as the walls, floors, and ceilings of the Condenser/Heater Bay, Hi/Low Conductivity Room, Steam Air Ejector Room, Mechanical Vacuum Pump Room, Demineralizer Room, and Regeneration Room. The physical integrity of this ventilation envelope is maintained to ensure a negative differential pressure in the envelope.

The reheater area ventilation system also serves the north end of the basement and mezzanine floors. It consists of an air washer, two full capacity supply fans, and one exhaust fan in the north end of the mezzanine floor. Air flow is divided between the reheater area and the north-end basement and mezzanine. The reheater area shares a common exhaust with the condenser area and the heater bay. The exhaust fan operates in unison with either supply fan and has a manual damper in the exhaust duct to regulate the flow of exhaust air.

The feedwater and condensate pump area is ventilated by a push-pull system consisting of one supply fan and one exhaust fan. The system is designed to **control** the temperature in the area using outside air for cooling and recirculated air for maintaining temperature when the turbine generator is operating. The maximum expected temperature is 114°F for 89°F outside air. The system uses 100 percent fresh air when the **room exhaust air** temperature is about 70°F or higher. Below 70°F **room exhaust** air temperature, the exhaust air is recirculated.

A provision for automatic damper lineup allows effective ventilation when either the supply or the exhaust fan is disabled. In normal operation, the dampers adjust automatically to hold the prescribed temperature and maintain a **nominal** flow of **68,000** cfm.

An additional exhaust system has been installed to remove excess heat from Reactor Feedwater Pump motors for pumps P-2-002A and P-2-002B. This system consists of hoods installed over each motor, ductwork which connects these hoods to the inlet of the feedwater and condensate pump area exhaust duct, and an inline fan located in this ductwork.

Reheater ventilation system exhaust fans circulate air, after shutdown, through the shell side and tube side of both reheater stages. The reheater ventilation system consists an exhaust fan and filter for each pair of reheaters. The supply side of the Reheater Ventilation System was isolated from the Reheater System and abandoned in place under BA 403061. Drawing BR 2009 includes a simplified flow diagram of the system.

The Turbine Building RAGEMS enclosure is provided with a separate heating and ventilation system to maintain acceptable environmental conditions for the RAGEM equipment.

Low differential pressure across exhaust fan EF 1-7 is alarmed in the Control Room. Turbine Building differential pressure on the operating floor, Heater Bay area and High/Low Conductivity room are measured locally. Vortex dampers at the exhaust fan **are manually operated** to prevent pressure buildup in the building.

The air washer systems are operated with only one supply fan running and one in reserve. There are no trip annunciators for these fans; however, the air washer low differential pressure annunciator alerts operators to a fan problem in these systems.

9.4.3.3 Safety Evaluation

Uncontrolled airflow (potentially contaminated) from the Turbine Building to the surrounding atmosphere is prevented. **The Turbine Building has within it areas of high contamination. These areas collectively are identified as the Turbine Building Ventilation Envelope. To ensure that airflow patterns within the Turbine Building move air from areas of low potential contamination to areas of high contamination the Turbine Building Ventilation Envelope is maintained at a negative pressure relative to outside atmospheric conditions. This results in air leakage being in rather than out for the Turbine Building.** Negative pressure is created by the exhaust fans. The inlet and vortex dampers of these fans are manually opened to ensure that a negative pressure will be maintained.

9.4.3.4 Inspection and Testing Requirements

No special equipment tests are required for these systems. Operating and standby components are alternated periodically to verify their operability. Routine visual inspection of the system components and instrumentation is adequate to verify system operability.

9.4.4 Radwaste Areas Heating and Ventilation

The radwaste areas of the plant include the Old Radwaste (ORW) Building, the New Radwaste (NRW) Building, the Offgas (OG) Building, and the Hot Machine Shop in the New Maintenance Building.

9.4.4.1 Design Bases

The heating and ventilation systems of the radwaste areas have been designed to meet the following requirements:

a. Provide fresh, tempered air to the various areas of the Radwaste Building in sufficient quantity to limit the temperature to a maximum of 104°F in areas where electrical equipment is located and where personnel access is not limited. In other areas the maximum temperature limit is 120°F. The design basis for building ventilation is 89°F dry bulb and 79°F wet bulb.

b. Provide air distribution within the building for controlled air movement from areas of low radioactive contamination potential to areas of high radioactive contamination potential.

c. Provide a means for filtering the exhaust air before discharging to atmosphere.

d. Heat the supply air as required to maintain a minimum of 50°F supply temperature during the winter season with a design basis of 10°F minimum outdoor dry bulb temperature. Zone booster heating coils increase the supply air temperature to 60°F before entering the respective zones.

e. Maintain automatically a negative pressure within the building with respect to atmospheric pressure in order to minimize the uncontrolled release of radioactivity.

f. Maintain airborne radioactivity below 10CFR20 and 10CFR50 Appendix I limits.

In addition, the design for the ORW Building and NRW Building meets the following requirements.

g. Provide a means for detecting and alarming automatically the ventilation system on low air flow and/or loss of air flow.

h. Provide ventilation to the building from a manually operated supply system and separate exhaust system. *

i. Automatically shut down the ventilation system following a preheat steam coil failure to protect against coil freeze up. *

j. Automatically alarm the ventilation systems filter media replacement.

k. Service, by means of a bypass arrangement, unit and equipment without shutting down the exhaust ventilation system. *

l. Maintain, by means of redundant exhaust fans, operation of the exhaust system following the failure of the normal operating fan.

All of the radwaste areas ductwork is designed, fabricated and constructed in accordance with Sheet Metal and Air Conditioning Contractors National Association, Inc. (SMAGNA) standards, and is classified as non seismic.

*** This section is not applicable to the Old Radwaste (ORW) Building.**

9.4.4.2 System Description

9.4.4.2.1 Old Radwaste (ORW) Building

Drawing BR 2012 shows a simplified flow diagram of the ORW Building ventilating system (also see Table 9.4-3).

The system has two exhaust fans and discharges to the stack. **The supply fans and air washer have been removed from service and abandoned in place.**

TABLE 9.4-1
(Sheet 1 of 2)

REACTOR BUILDING HEATING AND VENTILATION SYSTEM

System	Equipment	Function
Reactor Building Heating and Ventilation System	AW-1-4, SF-1-12; 13; 14	Provide heating, cooling and ventilation to Reactor Building Areas
Drywell Cooling System	RF-1-1; 2; 3; 4; 5	Cool reactor external surface and equipment inside the drywell.
Reactor Building Recirculating Fans:		
Steam Tunnel	RF-1-6; 7	Provide recirculation and cooling for pipes and equipment inside the steam tunnel.
Corner Rooms	RF-1-8; 9; 10; 11	Cool the Core Spray and Containment Spray Systems pumps, the Control Rod Drive feed pumps and associated equipment.
Stack Exhaust System	EF-1-5; 6; 7	**Maintain Reactor Building at a negative 0.25 inches WG during normal operation.**

TABLE 9.4-1
(Sheet 2 of 2)

REACTOR BUILDING HEATING AND VENTILATION SYSTEM

System	Equipment	Function
		Maintain the following Turbine Building areas. Condenser Bay, Heater Bay, Hi/Lo Conductivity Room, SJAE and Steam Seal area, Vacuum Pump Room, Demineralizer Room and Regeneration Room at a negative pressure **relative to outside atmospheric conditions during normal plant operation.**
Standby Gas Treatmemt System	EF-1-8; 9	Mitigate the consequences of an accident by maintaining a negative pressure of -0.25" WG in the Reactor Building.

Portable fire extinguishers are provided in the drywell (when open for maintenance), certain other areas of the Reactor Building, and at the Circulating Water intake area. Halon and CO_2 type portable extinguishers are provided for the Control Room.

9.5.1.2.6 Fire Detection and Signaling Systems

Three types of detection systems are provided within the plant (Table 9.5-8): products of combustion (ionization and photo electric), thermal (**fixed** rate of rise and rate compensated/**fixed temperature**) and flow switches. For areas that are vital, the alarms are fed to a local alarm panel and then to the Fire Alarm Master Panel in the Control Room. Non vital systems alarm locally only. Flow switches are used as the only means of detection in wet pipe sprinkler systems which do not have flow alarm valves. For wet pipe sprinkler systems with flow alarm valves, a pressure switch is the only means of detecting a fire. The wet pipe sprinkler systems have fusible heads.

The fire detection and signaling system includes fire detector for all safety related areas. Smoke detectors are provided in the ventilation systems of the Control Room and the Office Building. Actuation of automatic suppression systems transmits an alarm on the signaling system to the Control Room. A unique fire alarm signal is provided in the Control Room.

9.5.1.3 Safety Evaluation

The evaluation of the OCNGS Fire Protection System by area and zone was incorporated into the revised Fire Hazard Analysis, and was submitted to NRC as part of the 10CFR50, Appendix R evaluation on June 30, 1982. (Revision 2 was submitted to NRC on May 3, 1984; Revision 3 was submitted to NRC on April 3, 1985; Revision 4 was submitted to NRC on July 12, 1985; Revisions 5 and 6 were submitted on August 25, 1986, Revision 7 was submitted to the NRC on December 1, 1992, Revision 8 was submitted to the NRC July 29, 1993.. The Fire Protection Program has been evaluated by NRC and found acceptable.

9.5.1.4 Inspection and Testing Requirements

Inspection and testing of the Fire Protection System is performed in accordance with the OCNGS Fire Protection Program.

9.5.1.5 Personnel Qualification and Training

Qualification and training requirements for fire protection personnel is contained in the OCNGS Fire Protection Program.

9.5.2 Communication Systems

The following communication systems are provided at the OCNGS:

a. Paging System - The paging system provides for paging and announcement service, plus private intercommunication between two or more selected stations.

b. Dial Telephone System – **This is a standard commercial PBX telephone system**.

c. **Emergency Phone System – The emergency phone system is a dedicated subsystem of the dial telephone system. The emergency phones are used during plant emergencies and during emergency drills.**

d. Surveillance Telephone System - The surveillance phones use a small power source for the talking and signaling circuits.

e. KTY-750 Pagers and Backup Offsite Communications - Use of this system is dedicated for pagers and backup communications to Larrabee and Morristown (JCP&L), the remote control location is in the Control Room "Centracom."

f. KLX-229 Security and Emergency Planning - Use of this system is dedicated to security, radiological emergencies and routine site communications.

g. WSG-792 Repeater Operations - The repeater system is intended to achieve communication between otherwise out of range portable units.

h. Control Room "Centracom" - The "Centracom" incorporates remote capabilities for all transceivers and receivers in the Control Room, provides pager control for the KTY-750 transmitters and can access one channel at any given time through the use of a select button.

i. Microwave System - The system provides the primary communication link between the Control Room and Larrabee, its backup is the KTY-750 system.

The following provisions enhance Emergency Diesel Generator reliability:

a. The Diesel Generator Fuel Storage Tank and each Emergency Diesel Generator unit are enclosed in individual rooms separated by fire barriers.

b. There is a fuel oil supply line to each diesel generator unit.

c. An auxiliary day tank is provided for each engine.

d. The fuel transfer system for each engine features two pumps.

e. All system malfunctions are alarmed.

f. The Diesel Generator Building compartments are provided with fire detection and fire fighting equipment.

9.5.5 Diesel Generator Cooling Water System

A schematic flow diagram of the cooling system is shown in Figure 9.5-6. Water is circulated through the engine by means of two centrifugal pumps mounted on the front of the engine. The pumps are driven from the front or accessory gear train of the engine. Heated water from the discharge manifold leaves the engine and flows through the water outlet to the radiators. A 135,000 CFM shaft driven fan is mounted at the front of the EDG skid to remove heat from the cooling water radiators. Air is drawn from a grated roof section at the north end of the enclosure and travels down to the louvers and fan at the south end.

Water from the radiators is piped to the lube oil cooler, and from there to the engine water pumps. A temperature switch manifold is mounted on the accessory rack and connected to the inlet piping of the cooler. An immersion heater temperature switch, an engine temperature switch, and a louver control thermistor are mounted on the temperature control manifold. Louvered blade position is controlled by the thermistor and a motorized positioner to vary the air flow across the radiators as required.

During shutdown, the immersion heater temperature switch operates to turn the immersion heater elements on and off to maintain lubricating oil temperature. Water circulates by thermo siphon action through the lube oil cooler, which under these conditions functions as a lube oil heater.

The radiator and its piping are mounted in such a manner as to permit them to drain empty during engine shutdown. This condition eliminates the need for antifreeze in the cooling system. The engine automatically starts and goes to idle speed if the oil temperature falls, indicating potential immersion heater failure. The engine will shutdown and return to standby when temperature returns to normal standby levels.

9.5.6 Diesel Generator Starting System

9.5.6.1 Engine Starting Motor

Two electric starting motors are used to crank the diesel engine during starting. Each motor is equipped with its own starting solenoid and pinion gear. The 64V starters are wired in series for greater starting torque and comprise one starting unit for the diesel.

Upon a start signal, the starting circuit provides for up to three attempts to engage the starter motor pinions with the ring gear, three failed pinion engagement attempts initiate a sequence shutdown. Additionally, cranking time is **nominally** limited to 15 seconds. If the engine does not come up to speed within **the nominal** 15 seconds, a sequence shutdown is initiated. A sequence shutdown due to either these failed pinion attempts or failure to fire the engine annunciates in the Control Room to alert the operators. Operators may manually reinitiate a start sequence from the Control Room. A speed sensing device de-energizes the starting circuit after the engine fires and is accelerating.

During surveillance starts, a starting resistor is used to reduce voltage to the starting motor during the first four seconds of cranking. This limits the cranking torque and prevents buildup of momentum for the first revolution of the engine. Should a hydraulically locked piston be encountered, the engine will cease to rotate. Protective circuits will signal a starting fault and lock the engine out. If the engine is free to rotate, full voltage is applied after four seconds and fuel is injected into the cylinders. The starting resistor is bypassed during emergency starts and during an ECCS signal or low oil temperature initiated starts.

9.5.6.2 Battery Charger

A battery charger is located at each unit, mounted within the generator control compartment. The battery chargers are completely automatic, solid state, constant voltage devices featuring ac voltage compensation, dc voltage regulation and current limiting.

In case of an ac power failure a relay in the charger disconnects the automatic control and thereby eliminates any unnecessary drain on the battery. The charger will automatically resume charging upon return of ac power. During engine cranking, a contactor opens the charger circuit to the battery to avoid excessive charger current and recloses to the battery when the starter motors are dropped out of the circuit.

The charger will charge to its maximum current capacity and begin to limit current if overloaded. The automatic current limiting feature controls the output of the charger to a maximum of 140 percent rated load.

TABLE 9.5-8
(Sheet 3 of 4)

FIRE DETECTION INSTRUMENTATION

Location	Detector Zone	Required No.of Detectors
480 Volt Switchgear Room	Zone 3 (Ionization)	7**
480 Volt Switchgear Room	Zone 4 (Ionization)	6**
480 Volt Switchgear Room	Corridor (Ionization)	2
"A" and "B" Battery Room	Zone 1 (Ionization)	4**
"A" and "B" Battery Room	Zone 2 (**Photoelectric**)	4**
"A" and "B" Battery Room	Zone 4 (Duct) (Ionization)	1**
Reactor Recirculation Pumps MG Set Room	NA	1 (WFS)*
Monitor and Control Area	Below Ceiling (Ionization)	2
Monitor and Control Area	Above Ceiling (Ionization)	10
Monitor and Control Area	Sprinkler System No. 12	1 (PS)*
Monitor and Control Area	Hallway and Stairwell (Ionization)	3
Condenser Bay	Sprinkler System No. 2	1 (PS)*
Turbine Lube Oil Bay	Deluge System No. 3 Thermal (Rate of Rise)	1 (PS)*
Turbine Building Basement-South	Sprinkler System No. 9	1 (PS)*

* WFS - Water Flow Switch
** Actuate Automatic Suppression System

TABLE 9.5-8
(Sheet 4 of 4)

FIRE DETECTION INSTRUMENTATION

Location	Detector Zone	Required No. of Detectors
Transformers	Deluge System No. 1 Thermal (Rate Compensated)	1 (PS)*
Transformers	Deluge System No. 2	1 (PS)*
Emergency Diesel No. 1	Thermal (Rate of Rise)	5
Emergency Diesel No. 1	Ionization	1
Fuel Storage Area	Thermal (Rate of Rise)	1
Emergency Diesel No. 2	Thermal (Rate of Rise)	5
Emergency Diesel No. 2	Ionization	1
Fire Water Pump House (at Fire Road)	Thermal (Rate of Rise) **Thermal (Rate Compensated/Fixed Temperature)**	2 **2**
New Cable Spreading Room	Zone 1 (Ionization) Deluge System No. 15	1 (PS)*
Cable Bridge Tunnel	**Thermal (Fixed)** Sprinkler System No. 16	1 (PS)*

* PS - Pressure Switch

High Water Level in Reactor Vessel

If the water level in the vessel exceeds the height of the top of the steam separators (175" TAF), the turbine trips to protect against water damage to the turbine blades.

High Water Level in Moisture Separator Reheater

Should any of the four moisture separators become flooded, the turbine trips to protect the blades in the low pressure turbines from water damage.

High Temperature at Exhaust Hood

The High Temperature trip is Deleted. Should the temperature at the low pressure turbine exhaust hoods exceed **175°F**, **an alarm is sounded.** High temperatures can occur during low steam flow conditions. The exhaust hood sprays are used to minimize exhaust hood temperatures.

Turbine Control Valve Closure

Should the turbine control valves close to the 20 percent rated load, turbine trip will occur to protect the turbine reheaters against rapid cooldown from further rapid load reduction. During a normal, controlled shutdown, this trip is prevented by depressing a reset switch when a warning alarm alerts the Control Room of decreasing power.

Reheater Protection During Light Load

During startup or load reduction, when the load is below approximately 20 percent of rated load, the turbine will trip if the reheat stop check valves for the second stage reheaters are not closed. This is to protect the reheaters from high differential temperature between tube and shell sides.

Stator Cooling Failure

Turbine load runback is initiated by either high outlet water temperature or low inlet water flow in the stator cooling water system. If either or both of these conditions is sensed, the turbine runback circuit is initiated, causing the turbine load to be reduced to the stator "no-flow" load condition (approximately 25 percent of rated generator load). A timer is also started, which will cause the turbine to trip if the "no-flow" load, as sensed by a current relay, is not reached within approximately three minutes.

Generator Lockout Relay

The generator protective devices will activate generator lockout relays which in turn will trip the turbine master trip solenoid and actuate the vacuum trip. Loss of DC power supply to any protection relay or lockout relay is annunciated.

Two independent microprocessor based Digital Protection Relay Systems (DPRS "A" as primary, and DPRS "B" as backup) are integral part of the turbine generator protection system.

Generator protection functions provided by DPRS's include: sequential tripping, antimotoring (reverse power), turbine steam cutoff, under-frequency, three phase accidental energization, generator differential, loss of field, negative sequence (unbalance current), stator ground, over-excitation, pole slip, rotor ground, potential transformer monitoring, and protection during generator shutdowns and startups.

Generator Synchronization Protection

When the generator is synchronized to the grid, it is desirable to prevent breaker closing out of phase. The synchronization protection provided by DPRS "A" ensures that by allowing generator breaker closure only when the generator's and the grid's phase angles, voltages and frequencies are within allowable limits. This protection will not cause a turbine trip.

Main Transformer Lockout Relay

A turbine trip will result if an electrical problem is detected in the generator output transformers. Such faults might be a ground on the secondary (230 kV) side of either main transformer or a current flow mismatch between that flowing into the transformer primaries and that leaving the secondaries of these transformers.

Auxiliary Transformer Lockout Relay

A trip of the turbine results if a fault is detected within the auxiliary transformer circuit, such as a ground on either secondary of this transformer or a current mismatch between the primary and secondaries of the transformer.

Auxiliary Trip Lockout Relay

A turbine trip will occur upon the detection of a current flow mismatch in the overall generator-transformers-transmission lines circuitry.

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The heat removed by the new external heat exchanger is rejected to the existing RBCCW system. The RBCCW is designed for a total heat duty of 116 x 10^6 Btu/hr. The maximum heat load on the RBCCW prior to this modification was 80 x 10^6 Btu/hr. The additional heat load contributed by this modification (less than 1 x 10^6 Btu/hr) will not degrade the present RBCCW system.

Table 11.2-27 provides the system limitations, set points, and precautions.

b. Reactor Building Equipment Drain Tank (RBEDT) and Pumps

The Reactor Building Equipment Drain Tank is provided with an internal cooling coil supplied with Reactor Building Closed Cooling Water. Cooling water flow is controlled remotely by a solenoid valve. Tank temperature is indicated and high temperature is annunciated both on the old radwaste Control Room panel and the new radwaste Control Room panel. Level switches are provided to start and stop the Reactor Building equipment drain tank pump. High tank level and pump starts are annunciated at the control panel. A local manual switch is provided at the pump. The RBEDT pumps and the DWEDT pumps share a common discharge line to the High Purity Waste Collector Tank. Tank and pump data are listed on Table 11.2-6.

c. Radwaste Equipment Drain Sump (REDS) and Pumps

The Radwaste Equipment Drain Sump is equipped with duplex pumps, arranged and controlled similar to those in the Drywell Equipment Drain Tank. REDS and pumps data are listed on Table 11.2-7.

d. High Purity Waste Collector Tanks (HPWCT) (HP-T-1A&B) and High Purity Waste Pumps (HPWP) (HP-P-1A&B)

Inputs to the High Purity Waste Collector Tanks are received from the drain tanks and sumps, and the reactor cleanup system.

HP-T-1A is fitted with internal cooling coils to limit the temperature of the tank contents to a maximum of 120°F. This protects the thermally sensitive anion resins in the high purity waste demineralizers from high temperature degradation. **The cooling coils are not functional whenever the NRWCCW System is in a stand-by mode as described in Section 9.2.1.4.3.** The tank is also fitted with internal mixing jets and spray nozzles. Instrumentation for both tanks includes continuous level readout and continuous temperature indication. Wastes are manually diverted to the recirculation mode upon receipt of a high temperature alarm. The 100% capacity high purity waste pumps provide the motive force to operate the process trains.

Only one pump is required to supply each 150 gpm train. The pumps serve each of the collector tanks, respectively. These pumps are used to recirculate the contents of each tank. **The temporary construction strainer in the suction of HP-P-001A was removed following completion of preoperational and startup tests. The strainer in the suction of HP-P-001B has been re-installed following the relining of HP-T-001B to prevent foreign material and debris from this work from being introduced into the system.** The high purity waste pumps can be aligned to either waste processing trains. The appropriate train is selected by opening the appropriate high purity waste filter inlet valve. The high purity waste pumps are manually controlled by remote switches located on the radwaste control panel. High Purity Waste Collector Tanks and pumps data are listed in Table 11.2-8.

e. Waste Surge Tank

The Waste Surge Tank (WST) which is located outdoors has been **permanently abandoned**. Blind flanges have been installed to isolate the WST from the Liquid Radwaste System.

The concrete pads, where the Waste and Concentrator Distillate Sample Tanks are located, are curbed to contain potential leakage and rainwater. The area under the sample tanks can be drained to either the Old Radwaste Building floor drain sump or the discharge canal. The normal drainage path is to the **Old Radwaste Building Floor Drain Sump, however, if activity levels permit, discharge may be to the discharge canal.**

f. High Purity Waste Filters (HPWF)

The High Purity Waste Filters use a dilute cellulose fiber/resin or diatomaceous earth precoat and bodyfeed slurry of powdered resin to effect removal of suspended solids. Removal efficiency is on the order of 95% for rigid 1 micron diameter particles and 99% for rigid 5 micron diameter particles. Solids accumulated on the filters are dewatered with air and discharged to the Solid Radwaste System on a batch-basis. The High Purity Waste Filter System consists of the following principal components and is located in the New Radwaste Building.

1. Filter assembly — 2 of each
2. Precoat tank — 1
3. Precoat pump — 2 of each
4. Bodyfeed tank — 1

The evaporator concentrates pumps serve two 30 gpm Westinghouse skid mounted evaporators by recirculating/transferring concentrated solutions from the evaporator shell to the Concentrated Liquid Waste Storage Tanks. During normal operation, the evaporator concentrates pump operates in a continuous recirculation mode. Each pump is interlocked with its corresponding suction valve to enable pump operation only in the full open valve position. The concentrates piping is heat traced and insulated to prevent sulphate crystalization and potential line blockage. Radwaste concentrator and pump data are listed in Table 11.2-22.

u. Concentrator Distillate Demineralizers (CDD)

The Concentrator Distillate Demineralizers polish high purity distillate produced by the Radwaste Concentrators. They can also be used in series to purify the Chem Waste/Dewatering Filters outlet flow, and bypass the concentrators. Each demineralizer has a resin capacity of 30 cubic feet. Spent resins from these units are not regenerated, rather, they are sluiced to the Solid Radwaste System (Spent Resin Storage Tank). Fresh resins are manually loaded into the demineralizers through fill ports located in the floor slab at El. 48'-0".

Instrumentation for the demineralizers consists of differential pressure gauges between the influent and effluent lines to measure differential pressure across the beds and conductivity elements on the effluent line to measure product conductivity. A high conductivity alarm is provided to alert the operator of demineralizer breakthrough and allow him to take corrective action before contaminating a batch of processed waste. Concentrator distillate demineralizer data are listed in Table 11.2-23.

v. Chemical Waste Distillate Sample Tanks (CWDST) and Pump

The Chemical Waste Distillate Sample Tanks are used to collect and hold processed waste for sampling prior to recycle within the plant and/or discharge to the environment. The new Liquid Radwaste Treatment System utilizes the original Floor Drain Sample tanks for this purpose. Two tanks are provided so that one tank is available for filling while the remaining tank is being recirculated, sampled and emptied. Process waste, not suitable for recycle to the plant or discharge to the environment, can be directed to the high purity waste collection tank or a chemical waste/floor drain collection tank for reprocessing. Chemical waste distillate sample tank and pump data are listed in Table 11.2-24.

w. Contractor Supplied Portable Liquid Processing System

If liquid waste chemistry or volumes mandate, a contractor supplied portable liquid processing system can be used to treat liquid radwaste streams in lieu of evaporation or existing demineralizers. All appropriate design bases and criteria will be established if this processing alternative is necessary.

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x. <u>**Portable Drum Evaporators**</u>

Portable drum evaporators are approved for use in selected areas of the plant for evaporation of water with low isotopic concentration. These units are operated in accordance with approved site procedures.

11.2.3 Radioactive Liquid Releases

11.2.3.1 Normal Operation Releases

The sources of radioactive wastes that are to be released are as follows:

a. High Purity Waste

b. Chemical Waste/Floor Drain

The design basis for the normal sources of the High Purity Waste System are 8600 gpd from the Drywell Equipment Drain Tank, 1000 gpd from the Reactor Building Equipment Drain Tank, 3500 gpd from Radwaste Building Equipment Drain Sump, and 8000 gpd from the Turbine Building Equipment Drains and Regeneration Station. For the Chemical Waste/Floor Drain System the sources are 1400 gpd from the Drywell Floor Drain Sump, 1000 gpd from the Radwaste Building Floor Drain Sump, 1500 gpd from the Turbine Building Floor Drain Sump, 1400 gpd from the Stack Equipment Drain Sump, 200 gpd from the Laboratory Drain Tank, and 3000 gpd from the Regeneration System. These values along with maximum daily inputs (design basis) for High Purity Waste and Chemical Waste/Floor Drain Systems are presented in Tables 11.2-1 and 11.2-2, respectively.

The estimated radioisotopic releases from the High Purity Waste and Chemical Waste/Floor Drain Systems are listed in Table 11.2-25. The design total annual release under normal operating conditions was 2.20 curies. This estimated figure was based on the assumption that during normal operating conditions the discharge from the High Purity Waste System is 10% of the total waste processed by this system as most of the liquid will be recycled for continued use in the plant. The discharge from the Chemical Waste/Floor Drain System is conservatively assumed to be 100% of the distillate processed by the system. Even though the isotopic concentrations indicated in Table 11.2-25 are below the maximum permissible concentrations in water for an unrestricted area Appendix B, Table II, of 10CFR20), Oyster Creek operating policy is "zero release overboard."

11.2.3.2 Release Points

The release route of the radioactive liquid waste generated in the Chemical Waste/Floor Drain System is shown in Drawing GE148F437. Liquid processed for recycle and/or discharge from High Purity Waste System ultimately accumulates in the High Purity Waste Sampling Tanks. Each of the two tanks are sized to contain a full batch from the High Purity Waste Collection Tank, thus making one tank available for filling while the remaining tank is being recirculated, sampled and emptied. Processed waste suitable for discharge to the environment is routed to a single monitored release point which is the termination point of the service water piping at the intake canal.

TABLE 11.2-14)
(Sheet 1 of 1)

DRYWELL FLOOR DRAIN SUMP (DFDS) AND PUMPS

Sump Internal Size	4 ft 6 in. x 2 ft x 4 ft. 6 in. deep
Sump Material	Steel-lined concrete with Lithcoat Surface
Number of DFDS Pumps	2
DFDS Pump Capacity	50 gpm Minimum
Pump Type	Submersible, centrifugal

The Solid Waste Management System is divided into the following two waste categories:

a. Wet Solid Waste System

b. Dry Solid Waste System

11.4.2.2 Wet Solid Waste System

The wet solid waste handling system processes concentrated liquid wastes, chemical filter sludges, high purity filter sludges, reactor water cleanup filter sludges and resins, fuel pool cleanup filter sludges and resins, dewatered sludges and demineralizer resins from various plant demineralizers. The system is divided into three waste streams, as follows:

1. Concentrated Liquid Waste (CLW): Which are the bottoms of the chemical waste evaporates. These are stored in the two concentrated liquid waste tanks located at the New Radwaste Building.

2. Filter Sludge: Which is the discharge filter cake from the High Purity and Chemical Waste/Floor Drain, and from the Reactor Cleanup System and Spent Fuel Pool Filter System. The sludges are held in the two radwaste holdup tanks and the batch tank located at the New Radwaste Building.

3. Spent Ion Exchange: Which are resins from the condensate demineralizers Reactor Cleanup, Fuel Pool, High Purity, and Chemical Waste/Floor Drain demineralizers. These spent resins are stored in the two spent tanks, located at the New Radwaste Building.

Spent resins are transferred via the Spent Resin Transfer System into disposable high integrity containers, which are fitted with dewatering filters so that the resins can be dewatered to less than or equal to one percent free standing water in the Truck Bay.

CLW may be shipped to a licensed processor or solidified. A vendor supplied mobile solidification system with NRC approved topical report and state approved burial ground processing requirements can be made available upon demand.

Filter sludge may be dewatered similar to spent resin, or solidified similar to CLW.

A detailed description of the above three waste streams follows:

Concentrated Liquid Waste

Concentrate liquid waste is collected in two tanks with a capacity of 6,000 gallons each. The tanks **were designed with heaters** to maintain the concentrated liquid waste at 125°F to prevent Na_2SO_4 from precipitating out of solution and crystalizing. For expected waste generation rates of up to 3,000 gallons per week, these tanks **could accommodate** more than three weeks of storage capacity. **Since the practice of resin regeneration was discontinued at Oyster Creek, the Chemical Waste Evaporator is seldom used. This has resulted in significantly lower CLW volumes and greatly extended storage capacity time.** The concentrated liquid waste tanks are monitored for radiation, temperature and level. High and low temperatures and levels are annunciated remotely.

Filter Sludge

Filter sludge is discharged from four horizontal leaf precoat filters (NRW) **to** the holdup tanks. Two candle type filters for Reactor Cleanup and Fuel Pool Clean-up Systems also contribute to filter sludge inventory, however, this sludge is directed to the batch tank.

Currently there are two methods used for disposing of filter sludge. One is solidification and the other is dewatering and treatment with a biocide. Both produce acceptable final waste forms.

As previously stated, the solidification system is a vendor supplied mobile unit which has an NRC approved topical report and meets burial ground state approved processing requirements. The various chemicals and cement used in the solidification also kill any bacteria. So gas production is arrested.

Dewatering of filter sludge is accomplished in a similar manner as with spent resin. It is directed to a high integrity container equipped with filters for dewatering prior to shipment. Initially, a gross dewatering is performed. The water withdrawn is routed to an ultrafiltration system for final processing and then to a floor drain to be returned to the plant's overall water inventory via the Chem Waste/Floor Drain System. Following the gross dewatering, a biocide is injected into the high integrity container holding the filter sludge. The biocide also arrests gas generation. It is allowed to soak into the sludge. The high integrity container is then dewatered to less than or equal to one percent of free standing water. This time, however, the water withdrawn is routed to a holding container because it still contains biocide. This biocide/water solution is held in this intermediary container for reuse for the next high integrity container of filter sludge. Additional water and/or biocide may be added as needed to make up to this closed loop system.

particulates representing a wide spectrum of fission products, and fuel and transuranic elements, which also would be transported via the steam. The monitoring system also alerts the operator of fuel releases resulting in detector response above background, which could be indicative of the onset of fuel failures. Four channels of instrumentation are provided to monitor this activity. The detectors are ionization chambers mounted adjacent to each of the two main steam lines just downstream of the outer MSIVs at the drywell penetration. These types of detectors were chosen because they were standard instruments suited to the operational environmental and radiation conditions of concern. Each detector is connected to a logarithmic amplifier. A two pen recorder and a pen selector switch are used to select one channel for each pen. The range of these monitors is 1 to 1 x 10^6 mR/hr. Each channel is indicated continuously, recorded, and alarmed in the Control Room. Drawing GE846D686 show the system configuration. Table 11.5-1 summarizes the subsystem specifications.

11.5.2.2 Process Liquid Monitoring Subsystems

The Process Liquid Monitoring Subsystems are comprised of the Liquid Radwaste Overboard Discharge Monitor, the Reactor Building Closed Cooling Water Monitor, the Service Water Radiation Monitor and the Turbine Building sump 1-5 Radition Monitor. These subsystems have been designed to continuously measure, indicate, and record the radioactivity concentration levels of major process system discharge streams. The subsystems include scintillation detectors, and are set to alarm when concentrations vary significantly from normal levels. These monitors ensure that plant releases do not exceed the limits specified in 10CFR20 and 10CFR50 Appendix I.

The Liquid Radwaste Overboard Discharge Line has been capped and abandoned. The respective radiation monitor has also been abandoned.

The monitor associated with the RBCCW is situated on a lead brick housing immediately adjacent to, and with an unshielded view of, the discharge header of the RBCCW **heat exchangers**. The output of the detector preamplifier is connected to a seven decade Log Count Ratemeter and a recording pen on a recorder in the plant Control Room.

The Service Water Radiation Monitor is an offline radiation monitor on an enclosed skid located outside the Reactor Building near the Service Water Seal Well, which monitors gross radioactivity of the service water effluent from the RBCCW heat exchangers. The output of the detector preamplifier is connected to a ratemeter with a digital display in the SWRM Building. The output signal from the Ratemeter is connected to a six decade Log Countrate recorder located in the Control Room.

The RBCCW and Service Water Log Count Ratemeter trip units generate high, high-high, and downscale trips. The high and downscale trips actuate Control Room alarms.

A liquid radiation monitoring system (LRMS) has been installed in Turbine Building Sump 1-5 (see Section 9.3.3.2.1) to protect against a possible uncontrolled release to the environment. The LRMS provides local indication, local alarm and a Control Room alarm under high radiation conditions. The LRMS is interlocked with the sump pumps to terminate discharge in the event the sumps become contaminated.

11.5.2.3 Deleted

11.5.2.4 Air Ejector Offgas Monitoring Subsystem

The Air Ejector Offgas Monitoring Subsystem continuously monitors and records the radioactivity level of the effluent gases removed from the Main Condenser by the Steam Jet Air-Ejectors (SJAEs). The purpose of these monitors is to: (1) obtain a continuous record of radioactivity released to the offgas holdup system through the air ejector and (2) isolate the offgas holdup volume from the stack before the maximum permissible stack release rate is reached.

There are two completely independent channels of instrumentation provided. Each channel consists of an ionization chamber, a six decade logarithmic amplifier, and a shared two pen recorder. The logarithmic amplifier is equipped with adjustable high high and downscale alarms. The output of each channel is recorded continuously on one pen of the two pen recorder. The other pen is used by one channel of the stack gas monitor. Two such recorders are provided. A continuous recording of offgas flow and sample flow is also provided in the Control Room. Low sample flow is annunciated.

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TABLE 11.5-1
(Sheet 1 of 3)

PROCESS AND EFFLUENT RADIATION MONITORS

System	Type of Monitor	No. of Channels	Type of Detector	Range	Automatic Actions
Main Steam Line	Process	4	Ion Chambers	1.0 to $1x10^6$ mR/hr	Alarm
RBCCW	Liquid	1	Scintillation	10^{-1} to 10^6 cps	None
Service Water	Liquid Effluent	1	Scintillation	10 to 10^6 cpm	None
Air Ejector Offgas	Process	2	Ion Chamber	1.0 to $1x10^6$ mR/hr	Close V-7-31 & V-7-29 after time delay (0-15 min.)

TABLE 11.5-1
(Sheet 2 of 3)

PROCESS AND EFFLUENT RADIATION MONITORS

System	Type of Monitor	No. of Channels	Type of Detector	Range	Automatic Actions
RAGEMS Stack	Gaseous Effluent High/Low	½	Ion Chamber/Scintillation Detectors	10^{-1} to 127 ûCi/cc$^{(xe133)}$ 10^{-6} to 1 ûCi/cc	None
	Flow	1		0-1.76 SCFM	
RAGEMS Turbine Building	Gaseous Effluent-High and Low	1/1	Ion Chamber/Scintillation Detector	10^{-1} to 127 ûCi/cc$^{(xe133)}$ 10^{-6} to 1 ûCi/cc	None
	Flow	2		0-1.6 SCFM;0-1.88 SCFM	
Domestic Effluent	Liquid Effluent	1	Scintillation Detector		Trip pump

CHAPTER 12
RADIATION PROTECTION

12.1 ENSURING THAT OCCUPATIONAL RADIATION EXPOSURES ARE AS LOW AS IS REASONABLY ACHIEVABLE (ALARA)

12.1.1 Policy Considerations

12.1.1.1 Occupational Radiation Exposure

Oyster Creek is committed to maintaining occupational radiation exposures as low as reasonably achievable (ALARA) while performing all activities related to operation of the station. This commitment is reflected by providing for effective control of radiation exposure in the following major areas:

a. Upper management direction and support

b. Detailed ALARA policy and procedures

c. Consideration of ALARA during design of facilities and equipment

d. Development of good procedures and radiation practices, including preplanning and proper use of appropriate equipment and work techniques by well trained personnel

e. Audit and appraisal of performance

f. Implementation of improvements wherever and whenever they are reasonably achievable.

12.1.1.2 Organizational Structure and Responsibilities

The organization is structured to provide assurance that the ALARA policy is effective in the areas described above. The individuals and groups responsible for these activities are identified in Section 13.1. The overall coordination of the ALARA program is assigned to the Radiological Engineering Department.

12.1.1.3 Station ALARA Policy

The Radiation Protection Plan implements the ALARA policy in all phases of station operation. Employees are trained in radiation protection, station layout and proper work practices. Procedures, facility

modification, and equipment modifications are reviewed for ALARA prior to approval by station management. Routine and special surveys are performed to accurately assess radiological conditions. Radiation work permits (RWP) are required for the following conditions:

a. Entering a radiation area.

b. Entering an airborne radioactivity area.

c. Entering a contaminated area.

d. Entering a high radiation area.

e. Entering an area with unknown radiological conditions.

f. Maintenance of equipment, controls, or instrumentation which contain radioactive material.

Job surveillance/supervision assures that employees understand and follow the specified procedures and good work practices. The station ALARA policy incorporates the guidance provided in Regulatory Guides 8.8, 8.10, and 8.27 and in 10CFR20.

12.1.2 Design Considerations

The station was designed to minimize occupational exposures as much as possible, by using procedures and administrative controls to limit radiation exposures below the limits of 10CFR20. The considerations and methods employed are discussed in this subsection. The detailed facility design features for ensuring this are discussed in Subsection 12.3.1.

Material activation and the resulting impact on radiation fields is considered when procuring spare parts or when modifying a component that communicates with reactor coolant.

Considerations included in the design towards reducing the need for maintenance of equipment, reducing radiation levels and time spent where maintenance and other operational activities are required are:

a. Redundancy of equipment or components, where necessary, to reduce the need for immediate repair when radiation levels may be high.

b. Provisions to reduce the source intensity by isolating, draining and flushing, prior to maintenance.

c. Provision of easy access and space for tool laydown.

12.5 RADIATION PROTECTION PROGRAM

12.5.1 Organization

The **Radiation Protection** Program at the Oyster Creek Nuclear Generating Station is under the general supervision of the Radiological Protection Manager **(RPM)**, Oyster Creek and his staff. (The facility organization of the Oyster Creek Nuclear Generating Station is contained in Section 6, Administrative Controls, of the Technical Specifications. This section also defines the facility staff qualifications.) Under the supervision of the **RPM**, procedures have been established and implemented to ensure that radiation exposures to plant personnel and the public are maintained as low as reasonably achievable. A staff of **Radiation Protection** supervisors**, engineers,** and technicians **with support of other plant departments** perform **operational** duties which include:

a. Monitoring of plant effluents

b. Personnel radiation monitoring

c. Evaluation of radiation hazards, working times and protective devices to be used for specific tasks, by preparation and review of Radiation Work Permits. (See Subsection 12.1.1.3.)

d. Calibration of radiation detection equipment

e. Maintaining records of radiation exposures. (See Section 12.5.3.5.).

Plant personnel are trained to perform certain monitoring activities such as personnel contamination monitoring (frisking).

12.5.2 Equipment, Instrumentation, and Facilities

12.5.2.1 Personnel Monitoring Systems

A thermoluminescent dosimeter is required for:

a. All personnel, including visitor and contractor personnel entering Posted Radiation Areas.

b. All persons directed by **Radiation Protection** supervision, and where posted.

A self reading dosimeter is required for:

a. All personnel, including visitor and contractor personnel entering Posted Radiation Areas.

b. All persons directed by **Radiation Protection** supervision, and where posted.

Dosimetry shall be worn at all times, when required, except when personnel are donning/doffing protective clothing. TLDs are issued to personnel for periods which are based on the individual's duties and expected dose. Documentation is maintained to record visitor personnel dosimetry issue and exposure.

Special tasks involving extremity doses are monitored using extremity dosimetry. Extremity dosimetry is required whenever the estimated dose is expected to exceed four times the estimated whole body dose and when the extremity dose is expected to exceed 1000 millirems. Monitoring for internal deposition of radionuclides is provided by periodic and specific whole body counting, **whole body monitors,** bioassay, and air sampling.

12.5.2.2 Personnel Protective Equipment

Personnel protective equipment consists of protective clothing for the body, hands, feet, head and face. These items are worn as directed by a Radiation Work Permit (RWP).

Protection against airborne radioactive material is primarily provided by the use of engineering and/or process controls. A respiratory protection program has been established to provide protective devices for those cases where engineering and/or process controls are not practical or sufficient.

12.5.2.3 Personnel Decontamination Facility

The Personnel Decontamination Facility, equipped with a decontamination shower, is located in the Office Building.

12.5.2.4 Access Control

Access to the plant protected area is controlled by locked doors or gates. Free access within this controlled area is limited to personnel who have completed Category I General Employee Training (GET). Access to the Radiologically Controlled Area (RCA) is limited to personnel who have completed Category II GET. Visitors may gain access to plant areas only when escorted by personnel with the appropriate GET category and after specific control procedures have been met. The plant staff has been instructed to be alert to those not readily identified, and to challenge their presence.

12.5.2.5 Laboratory Facilities

The plant laboratory consists of facilities and equipment necessary to support various chemical and radiochemical analyses including reactor coolant, feedwater, offgas, liquid waste and auxiliary system chemistry.

12.5.2.6 Radiation Protection Instrumentation

A supply of portable radiation monitoring instruments is maintained at the plant. These devices include ionization chambers, Geiger-Mueller detectors and neutron dose ratemeters. Exposure rates and contamination levels are determined by these instruments. The calibration of beta-gamma and neutron instruments is performed at the plant.

12.5.3 **Radiation Protection** Procedures

12.5.3.1 General

Plant **radiation protection** procedures are designed to minimize the exposure of personnel to radiation and contamination. They contain guides and limits for radiation protection and contamination control as well as detailed procedures to ensure that these limits are not exceeded.

12.5.3.2 Personnel Monitoring

Sections of the **radiation protection** procedures describe the equipment used and the practices followed by plant personnel to monitor their individual radiation exposure.

12.5.3.3 Personnel Protective Equipment

This section of the **radiation protection** procedures deals with the proper use of such items as protective clothing (coveralls, shoe covers, gloves, etc.) and respiratory equipment. The equipment is described in detail along with its protective qualities and limitations.

12.5.3.4 Area Control

This section of the plant **radiation protection** procedures describes each of the various classes of access areas (Radiologically Controlled Area, Radiation Area, High Radiation Area), along with the procedures for entry and exit from each area.

12.5.3.5 Records

The following logs are maintained by the operating staff as part of the plant records. The originals are kept in the Document Control Center.

a. Control Room Logbook - contains entries affecting plant output, changes in auxiliary equipment, unusual conditions, line trips, annunciator signals, etc.

b. Shift Supervisor's Logbook - contains an overall summary of plant operation.

c. Radioactive Waste Logbook - contains record of volume of radioactive waste and rate of release.

A large number of records are maintained by the **radiation protection** and plant operations staffs. These are retained for varying periods of time (generally either 1 year, 2 years or the life of the plant) by the Information Management Center.

12.5.3.5.1 Radiation Protection

The Radiological Protection Manager OC, is responsible for **ensuring appropriate maintenance of** the records relating to:

a. Personnel radiation exposure

b. Radiation monitoring instrumentation calibration

c. Plant radiological conditions such as radiation levels, contamination levels and airborne activity levels.

12.5.3.5.2 Special Nuclear Materials

Special nuclear materials records are maintained and reported in conformity with 10CFR70 and 10CFR73.

12.5.3.5.3 Calibration of Instruments

Calibration of instruments and controls, both nuclear and conventional, is recorded, as is any maintenance performed on them.

12.5.3.5.4 Administrative Records and Reports

Detailed procedures provide information on the plant records and reports.

CHAPTER 13

CONDUCT OF OPERATIONS

13.1 ORGANIZATIONAL STRUCTURE

AmerGen Energy Company, LLC (AmerGen), is a limited liability company formed by Exelon Generation Company, LLC (EGC), British Energy, plc (British Energy), and British Energy, Inc. (BE Inc.), a wholly-owned subsidiary of British Energy. BE Inc. and EGC each own 50% of AmerGen. AmerGen is responsible for the safe, reliable, and efficient operation of its nuclear facilities. In addition, AmerGen is responsible for appropriate standards, programs, processes, management controls, and support for the nuclear facilities. In keeping with these responsibilities, AmerGen is committed to providing sufficient personnel having appropriate qualifications to both operate and technically support the facility.

The organizational structure of AmerGen, as shown in Figure 13.1-1, consists of headquarters functions, Regional Operating Groups (ROGs), and the nuclear sites. Direct responsibility and accountability for the safe and reliable operation of the plants resides in line management, from the Site Vice Presidents up through the ROG Senior Vice Presidents and Chief Executive Officer (CEO), and ultimately residing with the Chairman of the Management Committee and Chief Nuclear Officer (CNO).

The reporting relationships among the principal AmerGen executive officers and managers involved in the management of AmerGen's nuclear facilities shown in Figure 13.1-1 are described below.

The Chairman of the Management Committee and CNO is the senior corporate executive with all the necessary authority and full responsibility for the safe and reliable operation of the nuclear facilities operated by AmerGen. The Chairman of the Management Committee and CNO does not have any non-nuclear ancillary responsibilities.

The CEO reports to the Chairman of the Management Committee and CNO, and is responsible for the overall day-to-day operations of the ROGs.

The ROG Senior Vice Presidents report to the CEO, and are responsible and accountable for the safe and reliable operation of the nuclear units within their particular ROG.

A Site Vice President is assigned for each operating nuclear site. Each Site Vice President reports to the Senior Vice President of the appropriate ROG. The Site Vice President is the senior executive on site responsible for overall plant nuclear safety and for compliance with the NRC operating license. The Site Vice President provides day-to-day direction and management oversight of activities associated with the safe and reliable operation of the facility.

The Vice President - Nuclear Oversight reports to the Chairman of the Management Committee and CNO, and is the executive responsible for ensuring that the activities of the oversight organization, including audits, quality control, and assessments of the operating and technical support organizations, are carried out. A Nuclear Oversight Director responsible for Quality Assurance is assigned to each ROG, and reports directly to the Vice President - Nuclear Oversight.

The Nuclear Safety Review Boards (NSRBs) report to and advise the Chairman of the Management Committee and CNO, and the CEO on nuclear safety matters.

13.1.1 OFFSITE SUPPORT ORGANIZATION

The resources required to support day-to-day operation and maintenance of each plant are located onsite and report to site management. Supplemental support for the sites is available, as needed, from the Nuclear Services and Operations Support organizations. Needed support is provided by assigned resources and is available upon request. Other support for the plants in the areas of human resources, business operations and nuclear oversight (i.e., the quality assurance function) is also provided as needed. Personnel from these organizations are located at headquarters and in regional offices in order to ensure timeliness and ready availability of their support.

The headquarters Nuclear Services and Operations Support organizations establish and implement policies, programs, and processes to effectively and efficiently implement nuclear services and technical support functions in accordance with applicable regulations, codes, standards, and practices. The regional organizations, through the Director - Nuclear Services and Director - Operations Support within the ROG, support the plant sites directly by providing the resources needed in accordance with the priorities and schedules established by the ROG. Sufficient levels of management have been established within the functional areas in each support organization and dispersed between headquarters and the regional offices to provide clear management control and effective lines of authority and communication among the organizational units. These organizational arrangements are illustrated in Figures 13.1-2 and 13.1-3.

In addition, the headquarters and the regional offices are staffed by personnel with sufficient expertise and experience to provide the required technical support and services for the safe and reliable operation of the nuclear facilities.

13.1.1.1 Nuclear Services

The Senior Vice President - Nuclear Services is responsible for areas such as supply and project management, nuclear fuel management, engineering, nuclear information systems, and laboratory services. The Senior Vice President - Nuclear Services is accountable for defining standard programs and processes, delivering effective services and support, providing technical oversight of program implementation, and supporting the deployment and sharing of best practices throughout the organization.

Supply and Project Management

Supply and Project Management is responsible for project management, establishment of priorities and providing operational control of the purchase of non-fuel goods and services required for nuclear operations. Supply and Project Management is also responsible for inventory management, investment recovery, and the vendor audit program. Supply and Project Management establishes policies, common administrative controls and processes to ensure compliance with applicable requirements and effective use of resources.

Nuclear Fuel Management

Nuclear Fuel Management is responsible for the overall management and supervision of nuclear fuel and related services activities. Nuclear Fuel Management provides support for ensuring timely and economic supply of reload nuclear fuel, as well as providing reactivity management oversight and assuring that fuel procurement, reload fuel design, fabrication, licensing, utilization, cost accounting, and material accountability are consistent with safe and reliable operation of the nuclear units.

Engineering

Engineering is responsible for the establishment of policies related to the design and modification of the nuclear stations in accordance with applicable codes, standards, and regulations. Engineering provides engineering support to the sites for major engineering efforts or projects, or as requested by the stations. Engineering establishes and implements procedures which control material and component specifications, system designs, and modification activities.

Nuclear Information Systems

Nuclear Information Systems (NIS) is responsible for the overall management and supervision of information systems related services and activities. This includes creating, obtaining, and enhancing computer hardware, communication, and software systems to support operational requirements. In addition, NIS maintains and preserves information as a corporate asset and is responsible for the software quality assurance program.

Power Labs

Power Labs provides specialized field testing services, calibration services traceable to National Standards for measuring and test equipment, and laboratory analyses to support the safe and reliable operation of the nuclear units.

13.1.1.2 Operations Support

The Senior Vice President - Operations Support is responsible for areas such as generation support, training, licensing, maintenance and work control, outage planning and services, and security. The Senior Vice President - Operations Support is accountable for defining standard programs and processes, delivering effective services and support, providing technical oversight of program implementation, and supporting the deployment and sharing of best practices throughout the organization.

Generation Support

Generation Support is responsible for providing corporate support and management oversight of the nuclear stations in the areas of chemistry, radwaste, environmental, radiation protection, operations, and emergency preparedness. Generation Support is responsible for developing and implementing common administrative controls and processes to ensure safe and efficient plant operations.

Training

Training is responsible for oversight of the conduct of training activities. Training is accountable for the content of accredited and non-accredited training programs and for supporting the regional training organizations in the implementation and assessment of those programs.

Licensing

Licensing is responsible for providing support and management oversight of the nuclear stations to ensure prompt and proper disposition of regulatory issues. Licensing is responsible for developing regulatory positions and advising executive management on priorities and activities affecting regulatory issues at the nuclear sites. Licensing is responsible for developing policies and standardized processes and procedures for the maintenance of the licensing basis, the preparation of submittals to the NRC and other regulatory organizations, and the dissemination of regulatory and operational experience information. Licensing has overall responsibility to ensure that these activities are performed in accordance with applicable regulations. Licensing also provides support for the NSRBs.

Maintenance and Work Control

Maintenance and Work Control is responsible for providing support and standardization in the areas of predictive, proactive, and preventive maintenance. Maintenance and Work Control routinely assesses station maintenance and work control performance to identify and eliminate inefficiencies in work practices.

<u>Outage Planning and Services</u>

Outage Planning and Services is responsible for support to the site maintenance organizations in the areas of outage planning and management, reactor services, turbine generator services, steam generator services, and NDE services. Outage Planning and Services establishes policies, common administrative controls and processes to ensure compliance with applicable requirements and effective use of resources. Outage Planning and Services is responsible for the safe, effective and efficient performance of maintenance of plant equipment, instruments and controls in accordance with applicable regulations, policies and procedures.

<u>Security</u>

Security is responsible for the effective implementation of security programs, including access authorization, in-processing and fitness for duty. Security is responsible for providing strategic direction, providing program and policy guidance, and monitoring overall performance to ensure that security activities are performed in accordance with applicable regulatory requirements.

13.1.1.3 Nuclear Oversight

Nuclear Oversight is under the direction of the ROG Director - Nuclear Oversight who is accountable to the Vice President - Nuclear Oversight. The Vice President - Nuclear Oversight reports to the Chairman of the Management Committee and CNO. The ROG Director - Nuclear Oversight also advises the ROG Senior Vice President on nuclear quality matters. Nuclear Oversight is responsible for providing the QA Program, independent oversight, quality verification, and independent safety engineering functions.

The Oyster Creek Manager - Nuclear Oversight is responsible to ensure that the quality assurance program is established, implemented, and verified in accordance with corporate policies, applicable laws, regulations, licenses, and technical requirements. The Manager - Nuclear Oversight is also responsible for the conduct of nuclear safety review and assessment activities, which include those of the Independent Onsite Safety Review Group (IOSRG). The Manager - Nuclear Oversight administers the Employee Concerns Program for all Oyster Creek and support personnel having a concern regarding nuclear radiation safety.

The Manager - Nuclear Oversight performs the quality verification function for Oyster Creek, inspecting activities in sufficient detail to assure that they provide the required high degree of safety and reliability.

The Manager - Supplier Evaluation Services reports to the General Manager - Supply in the Nuclear Services organization, and performs the vendor audit function for Oyster Creek. Stop work authority has been delegated to the Manager - Supplier Evaluation Services by the Vice President - Nuclear Oversight. The Manager - Supplier Evaluation Services has unencumbered access to the Vice President - Nuclear Oversight for corrective action escalation.

The Nuclear Oversight and Supplier Evaluation functions and responsibilities are described in detail in the Oyster Creek Operational Quality Assurance Plan.

13.1.1.4 Nuclear Safety Review Board

The Nuclear Safety Review Board (NSRB) has been established to advise the Chairman of the Management Committee and CNO, and the CEO on the status of plant activities affecting nuclear safety and make recommendations regarding major procedures, facility, or license modifications. It also conducts periodic safety reviews of the plant. Records of the board meetings are maintained.

The NSRB reports to, and takes general direction from, the Chairman of the Management Committee and CNO, and the CEO.

13.1.2 OPERATING ORGANIZATION

Oyster Creek Nuclear Generating Station is under the direction of the Vice President - Oyster Creek who is accountable to the ROG Senior Vice President. The Vice President - Oyster Creek has direct, onsite responsibility for the safe, reliable, and economic operation and maintenance of Oyster Creek. The Vice President - Oyster Creek is responsible to operate and maintain Oyster Creek in a safe, environmentally sound, reliable, and efficient manner, in accordance with AmerGen policies and all applicable laws. The Vice President - Oyster Creek is the senior AmerGen representative at the Oyster Creek site and, as such, assures consistent implementation of policies and procedures at the site and at AmerGen offsite facilities in the Oyster Creek area. All station organizations are accountable to the Vice President - Oyster Creek except for those organizations involved in the independent corporate assessment and oversight activities described in the Oyster Creek Operational Quality Assurance Plan.

The station organization for Oyster Creek is shown in Figure 13.1-4. This figure shows the title of each position and the positions for which reactor operator and senior reactor operator licenses are required. The requirements for selection, qualification, and training of Oyster Creek personnel are specified in Section 13.1.3, Qualification of Nuclear Plant Personnel.

Succession of Authority

The Vice President - Oyster Creek has overall responsibility for station activities. The Director - Operations and Maintenance/Plant Manager is the Vice President's deputy and, in that capacity, is responsible for day-to-day plant operations, maintenance and technical support. The Director - Operations and Maintenance/Plant Manager assumes responsibility for all station activities in the absence or unavailability of the Vice President. If both the Vice President and the Director - Operations and Maintenance/Plant Manager are unavailable, absent, or incapacitated, a senior member of the site staff is designated as responsible for all station activities.

Director - Operations and Maintenance/Plant Manager

The Plant Operations and Maintenance Divisions are under the direction of the Director - Operations and Maintenance/Plant Manager, who is accountable to the Vice President - Oyster Creek. The Director - Operations and Maintenance/Plant Manager is responsible for operating the plant safely, reliably, and efficiently in compliance with all applicable Technical Specifications, quality assurance requirements, AmerGen procedures, and federal, state, and local requirements.

The authority of the Director - Operations and Maintenance/Plant Manager to act on behalf of the Vice President - Oyster Creek is inherent in the position and is commensurate with the assigned responsibilities. It includes the authority to order the shutdown of the unit whenever the health and safety of the public are endangered or when, in his judgment, a shutdown is warranted. It also includes the authority to issue procedures, orders, and other directives required in the execution of the assigned responsibilities. Necessarily included in the responsibility for plant operation and compliance with Technical Specifications, is the authority to assign and prioritize requirements to the Plant Operations and Maintenance Divisions. Similarly, the authority of the Director – Operations and Maintenance/Plant Manager includes the initiation and prioritization of corrective maintenance and preventive maintenance in the execution of his responsibilities. The Director – Operations and Maintenance/Plant Manager may delegate his authority to the Plant Operations Director during absences. This delegation of authority extends to the issuance of standing orders and directives in support of the responsibilities assigned. In the absence or incapacitation of the Vice President – Oyster Creek, the Director – Operations and Maintenance/ Plant Manager is delegated the authority of that office for the centralized control, supervision, coordination, and planning of all aspects of Oyster Creek operations.

13.1.2.1 Plant Operations Division

The Plant Operations Division is under the direction of the Director - Operations and Maintenance/Plant Manager who is accountable to the Vice President - Oyster Creek. The Director - Operations and Maintenance/Plant Manager is responsible for operating and maintaining the plant safely, reliably, and efficiently, in compliance with all applicable Technical Specifications, quality assurance requirements, procedures, and federal, state, and local requirements.

Reporting to the Director - Operations and Maintenance/Plant Manager are the Director - Maintenance, Plant Operations Director, Manager - Radiation Protection, Manager - Chemistry and Radwaste, and the Manager - Experience Assessment.

Operations Section

The Operations Section is under the direction of the Plant Operations Director/Senior Manager - Operations, who reports to the Director - Operations and Maintenance/Plant Manager. The Plant Operations Director/Senior Manager - Operations is directly responsible for supervision of plant operations including management oversight of shift operations. The Plant Operations Director/ Senior Manager - Operations has the following direct reports: Group Shift Supervisors/Shift Managers and the Operations Support Manager. The Plant Operations Director/Senior Manager - Operations must possess a Senior Reactor Operator's (SRO) license.

Group Shift Supervisors/Shift Managers

The Shift Operating Crews are supervised by Group Shift Supervisors/Shift Managers, who report directly to the Plant Operations Director/Senior Manager - Operations. The Group Shift Supervisors/Shift Managers are in charge of and responsible for plant operations on their shift. They are specifically responsible for supervising and directing operating employees during a shift and ensuring that work is performed according to approved procedures. They are also responsible for coordinating maintenance activities on their shift. Group Shift Supervisors/Shift Managers are SRO licensed and have the authority to direct a plant shutdown or to direct the plant to any specific set of conditions commensurate with approved procedures when observations of plant equipment or conditions indicate that a nuclear safety hazard exists, or as directed by approved procedures.

The Group Shift Supervisors/Shift Managers must remain aware of and in control of plant operational, maintenance, and testing activities that may affect safe operation. In the assignment of duties to the Group Shift Supervisors/Shift Managers, consideration is given to the need to prevent administrative duties from detracting from the primary responsibility of ensuring safe operation of the plant.

Shift Supervisors

The Shift Supervisors report to the Group Shift Supervisor/Shift Manager on their shift and assist the Group Shift Supervisor/Shift Manager in supervising and directing the employees who operate the plant. The Shift Supervisors ascertain and remain aware of plant equipment conditions by reviewing reports and by making personal inspections. They are responsible for scheduling certain routine tests and maintenance activities. Shift Supervisors are SRO licensed and have the authority to direct a plant shutdown when observations of plant equipment or conditions indicate that a nuclear safety hazard exists, or as directed by approved procedures.

Nuclear Plant Operators

The Nuclear Plant Operators report to the Shift Supervisor on their shift. Nuclear Plant Operators function as Control Room Operators and as equipment operators. Nuclear Plant Operators that function as Control Room Operators are Reactor Operator (RO) licensed. They are responsible for manipulating station controls as necessary to match load demands, respond to process changes, and take immediate operator action as necessary to bring the plant into and/or maintain the plant in a safe condition during abnormal or emergency conditions. They keep the Shift Supervisor informed of plant activities related to operations. Control Room Operators have the authority to shut down the plant when observations of plant equipment or conditions indicate that a nuclear safety hazard exists or as directed by approved procedures.

Nuclear Plant Operators that function as equipment operators report to the Shift Supervisor and perform routine duties outside the control room as necessary for continuous, safe plant operation. They are available to Shift Supervisors for additional work assignments that may arise from time to time. They assist in plant startup, shutdown, surveillance, emergency response, fuel handling, shipment of irradiated materials, and disposal of radwaste as directed.

Operations Support Manager

The Operations Support Manager reports to the Plant Operations Director/Senior Manager - Operations and is responsible for supporting the daytime shift organization by relieving operators and shift management personnel of some of their administrative burdens and ensuring effective coordination of operations programs and projects. The Operations Support Manager is also responsible for providing Shift Technical Advisor (STA) coverage for each operating shift, and operation of the Work Control Center.

The Operations Support Manager is responsible for reviewing operating experience reports that may be directed to the plant by various industry groups and regulatory agencies to determine the impact on Operations Section programs and practices.

Shift Technical Advisors

The STAs report to the Operations Support Manager and serve in an advisory capacity to the shift operating staff. The function of the STA position is to make additional engineering and technical expertise available to the shift operating staff. The STAs advise the operations supervisors on matters of reactor safety.

The STAs are responsible for monitoring plant operations to ensure safe, reliable, and efficient performance. They evaluate the performance of the reactor plant during power operations, startup, shutdown, and refueling modes. The STAs are further responsible to monitor the response of the plant during transients and accidents, determine whether the plant is responding within expected limits, and advise the operations supervisors of the actions necessary to maintain or regain a safe, stable operating state.

Radiation Protection Section

The Manager - Radiation Protection reports to the Director - Operations and Maintenance/Plant Manager and fills the position of radiation protection manager as described in ANSI/ANS 3.1-1978, Section 4.4.4. Reporting to the Manager - Radiation Protection are the Manager - Radiological Engineering and the Radiation Protection Supervisors. The Manager - Radiation Protection is also responsible for implementation of the industrial safety function.

The Manager - Radiation Protection is responsible for the development and implementation of a radiological controls program which provides the needed high degree of protection from radiological hazards and meets or exceeds those requirements specified in the Radiation Protection Plan, Technical Specifications, the Code of Federal Regulations or other regulatory directives.

The Manager - Radiation Protection has the authority to direct the termination of any activity which is not being accomplished in accordance with radiological controls practices and procedures.

Chemistry and Radwaste Section

The Manager - Chemistry and Radwaste reports to the Director - Operations and Maintenance/Plant Manager and is responsible for implementation of the plant chemistry, radwaste, and environmental and controlled materials programs. These responsibilities include implementing the routine chemistry programs in the plant, supervising personnel who collect and analyze plant samples, daily interface with Operations, troubleshooting of chemistry anomalies, processing and shipment of radioactive waste and materials, and managing the environmental and controlled materials programs.

Reporting to the Manager - Chemistry and Radwaste are the Supervisor - Chemistry, Supervisor - Radwaste/Environmental, and the plant chemists.

Experience Assessment Section

The Experience Assessment Section is supervised by the Manager - Experience Assessment, who reports to the Director - Operations and Maintenance/Plant Manager. The Manager - Experience Assessment is responsible for the following:

- **a. Supporting and advising the Director - Operations and Maintenance/Plant Manager and all site organizations on compliance with NRC regulations and reporting requirements;**
- **b. Reviewing operating experience reports that may be directed to the plant by various industry groups and regulatory agencies and for disseminating such reports to the appropriate personnel for review for applicability to Oyster Creek and determination of required action;**
- **c. Coordinating onsite interfaces with INPO, NRC and other regulatory agencies, and communicating with the Corporate Licensing Division;**
- **d. Coordinating the review of plant events to determine causes and corrective actions. This includes event trend analysis, tracking status of event evaluations and corrective action, preparation of Licensee Event Reports (LERs), and reporting the results to management;**
- **e. Coordinating the site commitment tracking program and closure of NRC open items; and,**
- **f. Coordinating requests for and implementing Technical Specification amendments.**

13.1.2.2 **Maintenance Division**

The Maintenance Division is under the direction of the Director - Maintenance who is accountable to the Director - Operations and Maintenance/Plant Manager. The Director - Maintenance is responsible for the safe, effective and efficient performance of maintenance of plant equipment, instruments and controls in accordance with applicable regulations, policies and procedures. The Maintenance Division consists of the Instrumentation and Controls (I&C) Section, Mechanical and Electrical Section, Planning Section, and Maintenance Optimization Section.

Instrument and Controls Section

The I&C Section is supervised by the Manager - Instrumentation and Controls who reports to the Director - Maintenance. The Manager - Instrument and Controls is responsible for the calibration, surveillance testing, maintenance, record keeping, and troubleshooting of instruments and controls, radiation monitoring equipment, fire detection and suppression systems, portable measuring equipment and plant computers.

Mechanical and Electrical Section

The Mechanical and Electrical Section is supervised by the Manager - Mechanical and Electrical who reports to the Director - Maintenance. The Manager - Mechanical and Electrical is responsible for the assignment, supervision, and coordination of activities performed by Mechanical and Electrical Section craftsmen and craft supervisors.

Planning Section

The Planning Section is supervised by the Supervisor - Planning who reports to the Director - Maintenance. The Supervisor - Planning is responsible for managing the planning of all plant mechanical and electrical maintenance and plant instrumentation and controls activities.

Maintenance Optimization Section

The Maintenance Optimization Section is supervised by the Manager - Maintenance Optimization who reports to the Director - Maintenance. The Manager - Maintenance Optimization is responsible for optimization and standardization of the Preventive Maintenance Program by providing recommendations to engineering based on trend analyses obtained through predictive maintenance technology; optimization of the Predictive Maintenance Program (PDM) through condition monitoring and diagnostic technology applications; and for trending data obtained through PDM to provide equipment and system assessments for use by engineering and planning.

13.1.2.3 Work Management Division

The Director - Work Management is accountable to Vice President - Oyster Creek. The Director - Work Management is responsible for assuring that outages and On-line work periods are effectively planned and managed; planning, scheduling, and reporting site work activities; implementing plant modifications and materials management. The Director - Work Management is also responsible for the management of engineering projects and ensures that all engineering work is planned, scheduled, budgeted, implemented, technically supported, and evaluated in a timely and cost effective manner. The Director - Work Management is responsible for radiological housekeeping (including decontamination operations) of building structures and applicable implementing procedures. The Work Management Division consists of the On - Line Outage Section, Supply Section, Work Support Section, Outage Management Section, and the FIN (fix it now) Team.

13.1.2.4 Site Engineering Division

The Site Engineering Division is under the direction of the Director - Site Engineering, who is accountable to the Vice President - Oyster Creek. The Director - Site Engineering is responsible for planning, directing and coordinating onsite engineering and technical support activities in accordance with applicable regulations, policies and procedures. The Site Engineering Division consists of the Plant Engineering Section, Design Engineering Section, Engineering Response Team, and Process Computers Section.

Plant Engineering Section

The Plant Engineering Section is managed by the Senior Manager - Plant Engineering, who reports to the Director - Site Engineering. The Plant Engineering Section is divided into branches based on specific responsibilities or expertise. Each branch is supervised by a Branch Manager reporting to the Senior Manager - Plant Engineering.

The Plant Engineering Section is responsible for supporting all aspects of system operation and maintenance including: maintaining a high level of system knowledge; responsibility for the Maintenance Rule, Equipment Performance Information Exchange (EPIX), and inservice testing programs; participating in design, licensing and procedural changes; reviewing data; initiating improvements; supporting power maneuvering; reviewing system operations and maintenance; and performing other system related tasks as required. The Plant Engineering Section is also responsible for performing onsite reactor engineering and for maintaining the Fire Protection Program.

Design Engineering Section

The Design Engineering Section is managed by the Senior Manager - Design Engineering, who reports to the Director - Site Engineering. The Design Engineering Section is divided into branches based on specific responsibilities or expertise such as procurement, civil/structural, electrical, instrumentation and control, mechanical, design and drafting, and configuration maintenance. Each Branch is supervised by a Branch Manager reporting to the Senior Manager - Design Engineering.

The Design Engineering Section is responsible for supporting the Station in design changes involving hardware, software, licensing documents, and maintaining various engineering programs (e.g., Flow Accelerated Corrosion Program).

Engineering Response Team

The Engineering Response Team is managed by the Manager - Engineering Response Team, who reports to the Director - Site Engineering. Engineering Response is a multidisciplinary team with the skills necessary to provide first response to most system/design issues that would otherwise be handled by the Design Engineering Section and/or Plant Engineering Section. The Engineering Response Team deals with short-term emergent issues. This allows the rest of Site Engineering to concentrate on long-term issues and routine planning activities and responsibilities.

Process Computers Section

The Process Computers Section is managed by the Manager - Process Computers, who reports to the Director - Site Engineering. The Process Computers Section provides digital technical support, computer hardware maintenance support (preventive and corrective), and hardware and software engineering for the site process computers and simulators. This section also supports daily operations and maintenance activities, modification and configuration control, equipment reliability, and long and short term planning for these computer systems.

13.1.2.5 Training Division

The Training Division is directed by the Director - Training, who is accountable to the Vice President - Oyster Creek. The Director - Training oversees the identification of programmatic training needs of all site personnel, ensures the effectiveness of training programs, incorporates operating experience into training, and monitors participation.

The Training Division consists of the Operations Training Section, Technical Training Section, and Engineering/Training and Services Section. See Section 13.2 for training program descriptions.

13.1.2.6 Site Support Division

The Site Support Division is under the management of the Director - Site Support, who is accountable to the Vice President - Oyster Creek. The Director - Site Support is responsible for effective management of the site security, information management and records management organizations and programs, and for providing management direction, coordination and accountability in the areas of occupational health and corporate security. The Site Support Division consists of the Security Section, Information Technology Section, and the Document Services Section.

Security Section

The Security Section is under the management of the Manager - Security, who reports to the Director - Site Support. The Manager - Security is responsible for control, maintenance and implementation of the site security program, including coordinating all activities associated with site access.

Information Technology Section

The Information Technology (IT) Section is managed by the IT Manager, who reports to the Director - Site Support. The IT Section is responsible for the specification, development, monitoring, maintenance, modification, and documentation of the IT infrastructure and software assets, other than the plant process computers and the simulator. The IT Section is also responsible for the processes and content of the configuration management of IT assets.

Document Services Section

The Document Services Section is managed by the Supervisor - Document Services, who reports to the Director - Site Support. The Document Services Section is responsible for operation of the Information Resources Management Center (IRMC), document and records management, and for ensuring that records management systems meet information needs and legal and regulatory requirements for quality criteria, storage, retention periods, controlled distribution, and retrieval capability.

13.1.2.7 Manager - Human Resources

The Manager - Human Resources reports to the Vice President - Oyster Creek. The Manager - Human Resources is responsible for all human resources functions, including labor relations and in-processing activities.

13.1.2.7.1 Communications Representative

The Communications Representative reports to the Manager - Human Resources. This position is responsible for providing and maintaining an understanding of plant activities among its employees; and, with direction from the ROG communications office, among public officials, media, general public, community and business leaders, and regulatory agencies. The Communications Representative, in concert with the ROG communications representative, is also responsible for maintaining the readiness of the emergency public information procedure through training and participation in emergency exercises. The Communications Representative also takes direction from the ROG Manager - Communications and Public Affairs on communications issues and strategies.

13.1.3 Qualification of Plant Personnel

Each member of the unit staff shall meet or exceed the minimum qualifications of ANSI/ANS 3.1 -1978 for comparable positions unless otherwise noted in the Technical Specifications. Licensed operators shall meet the supplemental requirements specified in Sections A and C of Enclosure 1 of the March 28, 1980 NRC letter to all licensees. Technicians and maintenance personnel who do not meet ANSI/ANS 3.1 - 1978, Section 4.5, are permitted to perform work for which qualification has been demonstrated. Table 13.1-1 describes the essential managerial positions and comparable ANSI/ANS 3.1-1978 positions for the individuals responsible for programs and systems that ensure the safe and successful operation of the facility.

The management position responsible for radiological controls shall meet or exceed the qualifications of Regulatory Guide 1.8 (Rev. 1-R, 9/75). Each other member of the radiation protection organization for which there is a comparable position described in ANSI N18.1-1971 shall meet or exceed the minimum qualifications specified therein, or in the case of radiation protection technicians, they shall have at least one year's continuous experience in applied radiation

protection work in a nuclear facility dealing with radiological problems similar to those encountered in nuclear power stations and shall have been certified by the management position responsible for radiological controls as qualified to perform assigned functions. This certification must be based on an NRC approved, documented program consisting of classroom training with appropriate examinations and documented positive findings by responsible supervision that the individual has demonstrated his ability to perform each specified procedure and assigned function with an understanding of its basis and purpose.

The Shift Technical Advisors shall have a bachelor's degree or equivalent in a scientific or engineering discipline with specific training in plant design, response and analysis of the plant for transients and accidents.

Qualification requirements similar to those of other major engineering firms are used for staffing the Nuclear Services and Operations Support organizations. These organizations consist primarily of individuals having college degrees, or the equivalent, in appropriate science or engineering disciplines. In certain instances, technicians, who by virtue of formal education, training programs, or experience have acquired special expertise in particular areas are involved in providing technical support. In keeping with responsible management practices, the capabilities of individuals and supervisors are considered in making personnel assignments.

13.1.4 Safety Review Functions

The safety review process defines how procedure changes, Technical Specification changes, Licensee Event Reports (LERs), plant modifications, Independent Spent Fuel Storage Facility (ISFSF) modifications, Certificate of Compliance changes, and other documents are reviewed, approved and implemented. This process spreads the responsibility for activity in these areas broadly across the organization. The process requires each director or manager to control the preparation, review and reporting activities of each activity in their area which affects nuclear safety.

Each director or manager has the responsibility for ensuring that preparation, review and approval of procedures and other documents required by the activities within their area of responsibility are carried out properly. The subjects addressed include operating procedure changes, plant hardware modifications, security and radiological and environmental control activities, etc. In other words, all aspects of nuclear plant design and operation that are important to safe operation involve the safety review process. This preparation, review and approval process consists of a sequence of four distinct activities: - preparation, technical review, independent safety review and implementing approval.

Responsibilities and Qualifications of Preparers, Reviewers and Approvers

The person assigned to prepare the document (Procedure Change, Design Modification, LER, etc.) must be knowledgeable and experienced in that technical area. The person is responsible for providing thoughtful, well written language which can easily be interpreted by users and reviewers. The preparer is responsible for soliciting input from knowledgeable people in other organizations, as appropriate, and resolving their comments. The preparer makes an initial determination as to whether or not: (a) a cross disciplinary review is necessary; and, (b) an unreviewed safety question is involved; or, (c) a Technical Specification change is needed.

Technical Review

A Responsible Technical Reviewer (RTR) is charged with reviewing the document for safety and technical adequacy. The RTR must be a knowledgeable and experienced individual, different from the preparer, but may be from the same organization as the individual who prepared the document.

The RTR must be qualified in accordance with Technical Specification 6.5.1.14. The RTR must also review and concur on the determination of: (a) the necessity for a cross disciplinary review, (b) the existence of an unreviewed safety question, and (c) the need for a Technical Specification change. This review and concurrence must be documented.

The RTR is so named to distinguish this individual from others who might be asked to provide cross-disciplinary assistance for the technical review. The RTR is accountable for the review and is the individual who must be qualified as specified in ANSI/ANS-3.1-1978 Section 4.4.

"Implementing approvers" are responsible for releasing the document for its intended use (NRC Report, permanent procedure change, etc).

Independent Safety Review

An independent safety review must be performed by an individual or group not having direct responsibility for the performance of the activities under review and, in most cases, may be performed after implementation. The organization having responsibility for independent safety review in a given area is assigned by the Review and Approval Matrix. The purpose of the independent safety review process is to assess the adequacy of the preparation and documentation provided on the existence of unreviewed safety questions and the need for Technical Specification changes.

The function of the Independent Safety Reviewer (ISR) is to perform an independent verification of a document to the extent necessary to verify safety adequacy. Personnel performing independent safety reviews must have a Bachelor's degree in science or engineering and five (5) years of professional experience in the area being reviewed or nine (9) years of experience in the field of his specialty.

Safety Review

The Technical Specifications require establishment of a nuclear safety group at Oyster Creek which consists of experienced engineers. This group is part of Nuclear Oversight. These engineers have a number of tasks all related to assuring nuclear safety. Among these are (1) review of procedures to assure technical adequacy and clarity, and (2) review of unit operations from a safety perspective.

Summary of Accountabilities

Technical Specification 6.5.1, "Technical Review and Control," charges the directors and managers with the responsibility for ensuring the preparation, review and approval of documents required by activities within their functional area of responsibility. These activities include (1) procedures and procedure changes that affect nuclear safety (2) proposed changes to Technical Specifications, (3) proposed modifications of unit structures, systems and components that affect nuclear safety (4) proposed tests and experiments that affect nuclear safety (5) investigation of violations of the Technical Specifications, (6) events reportable to NRC, and (7) security and emergency plans and their implementing procedures.

Technical Specification 6.5.2, "Independent Safety Review," charges the directors and managers with the responsibility for ensuring periodic independent safety review of subjects within their assigned area of safety review responsibility.

These subjects include:

1. **written safety evaluations of changes to the facility as described in the Safety Analysis Report (SAR), of changes in procedures as described in the SAR, and of tests or experiments not described in the SAR;**

2. **proposed changes in procedures, proposed changes in the facility, or proposed tests or experiments which involve a change in the Technical Specifications or an unreviewed safety question;**

3. **proposed changes to Technical Specifications or license amendments related to nuclear safety; and,**

4. **violations and reportable events which require written reports to NRC.**

The directors and managers are also responsible for establishing procedures for carrying out these activities, as well as providing for training of individuals involved in the process.

TABLE 13.1-1
OYSTER CREEK UNIT STAFF QUALIFICATIONS

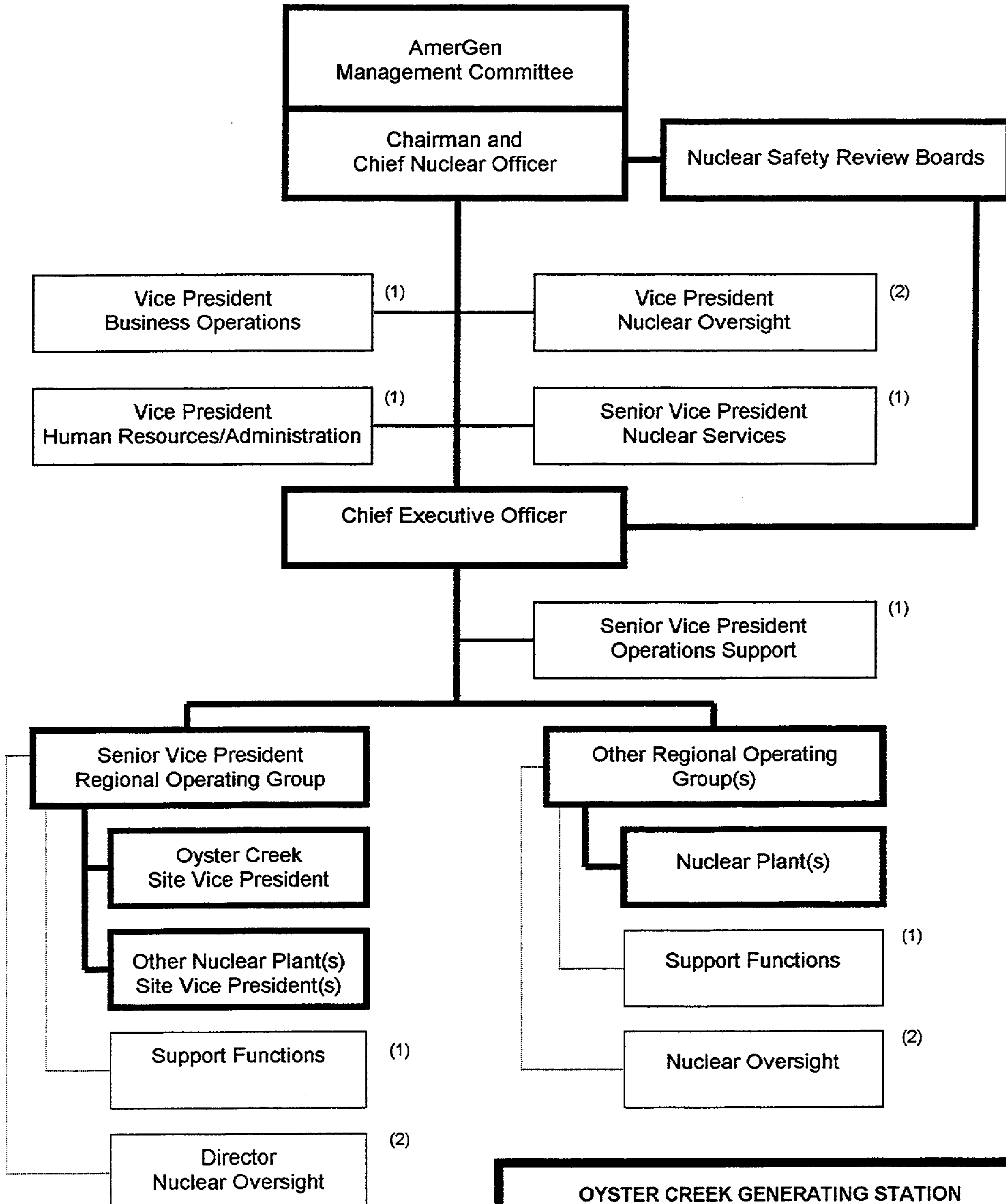
Station Position		ANSI/ANS 3.1-1978 Title
Plant Operations Division		
Director – O&M/Plant Manager	4.2.1	Plant Manager
Plant Operations Director (Note 1)	4.2.2	Operations Manager
Group Shift Supervisor (Note 1)	4.3.1	Supervisors Requiring NRC Licenses
Shift Supervisor (Note 1)	4.3.1	Supervisors Requiring NRC Licenses
Nuclear Plant Operator (Control Room Operator) (Note 2)	4.5.1	Operators
Manager – Radiation Protection (Note 3)	4.4.4	Radiation Protection, Professional – Technical
Manager – Chemistry and Radwaste	4.4.3	Chemistry and Radiochemistry, Professional – Technical
Maintenance Division		
Director – Maintenance	4.2.3	Maintenance Manager
Work Management Division		
Director – Work Management	4.2.3	Maintenance Manager
Site Engineering Division		
Director – Site Engineering	4.2.4	Technical Manager
Senior Manager – Plant Engineering	4.2.4	Technical Manager
Senior Manager – Design Engineering	4.2.4	Technical Manager

Notes

(1) Must be SRO licensed.

(2) Must be RO licensed.

(3) The management position responsible for radiological controls shall also meet or exceed the qualifications of Regulatory Guide 1.8 (Rev. 1-R, 1975).

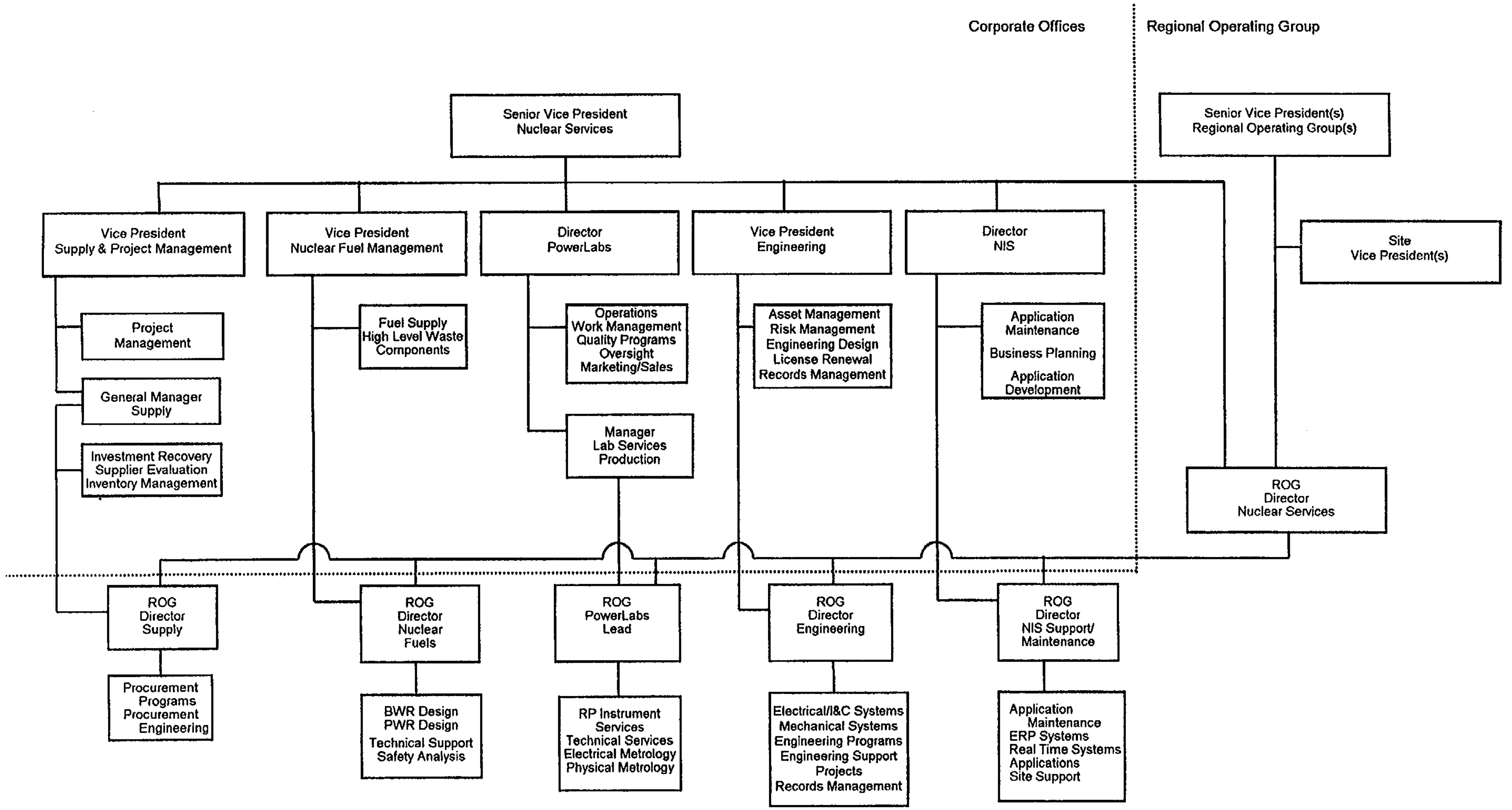


NOTES:
(1) Resources dispersed between Headquarters and the ROGs.
(2) Nuclear Oversight Director reports to the Vice President - Nuclear Oversight and advises the ROG Senior Vice President on nuclear quality matters.

OYSTER CREEK GENERATING STATION
UPDATED FINAL SAFETY ANALYSIS REPORT

MANAGEMENT AND TECHNICAL SUPPORT ORGANIZATION

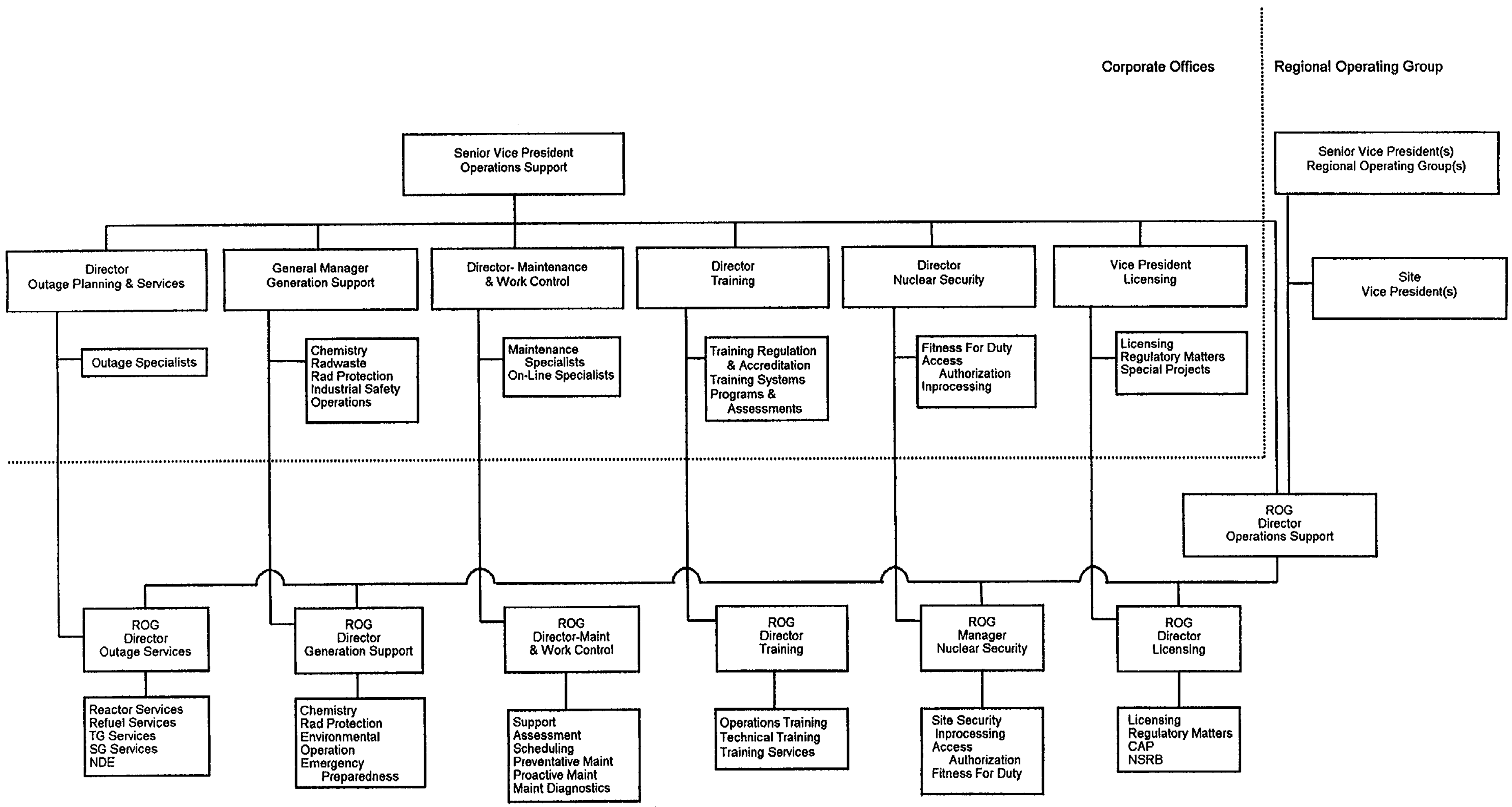
FIGURE 13.1-1 **Rev. 12 04/01**



OYSTER CREEK NUCLEAR GENERATING STATION
UPDATED FINAL SAFETY ANALYSIS REPORT

NUCLEAR SERVICES ORGANIZATION

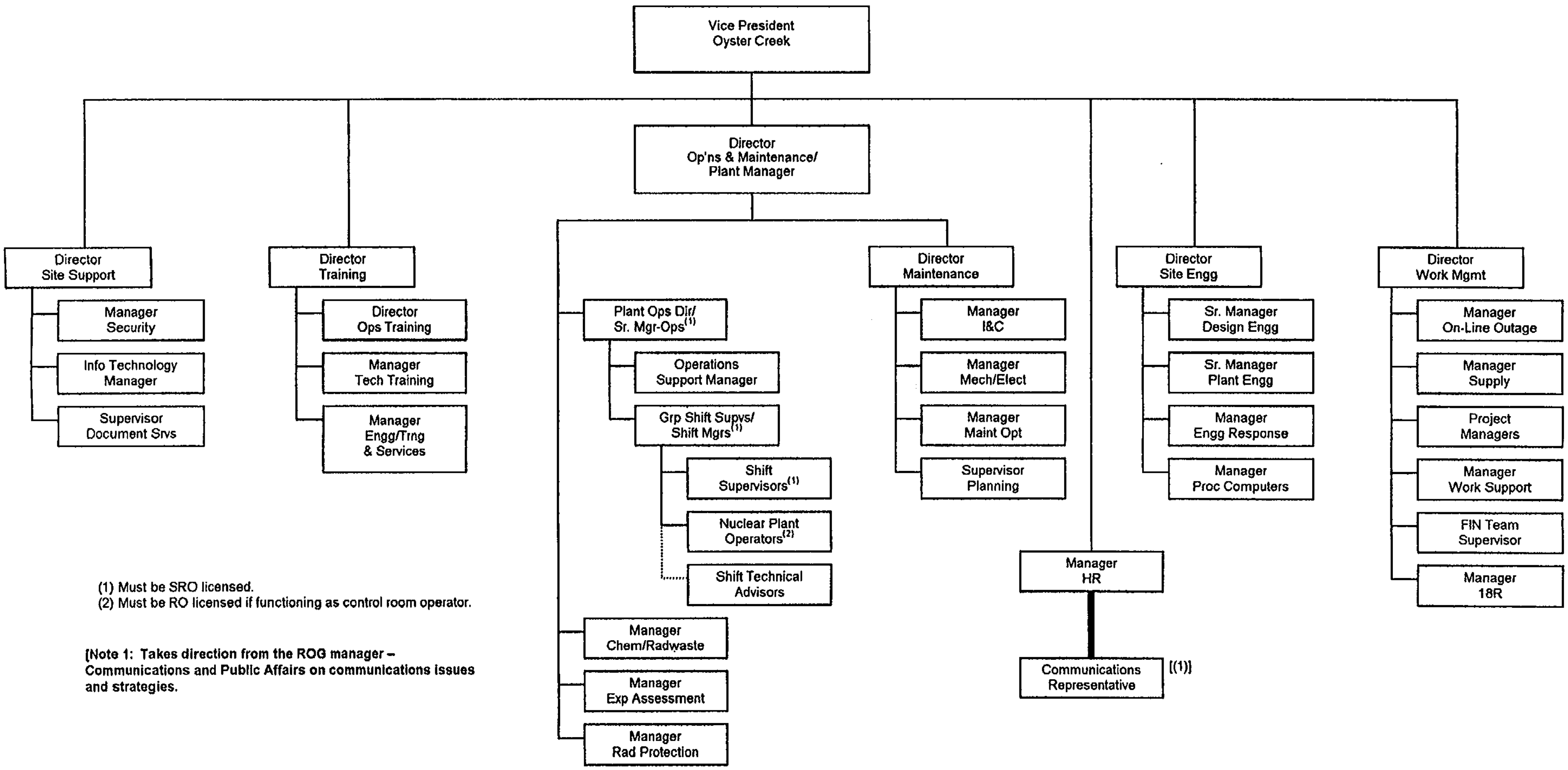
FIGURE 13.1-2 Sht. 1 OF 1 Rev. 12 04/01



OYSTER CREEK NUCLEAR GENERATING STATION
UPDATED FINAL SAFETY ANALYSIS REPORT

OPERATIONS SUPPORT ORGANIZATION

FIGURE 13.1-3 Sht. 1 OF 1 Rev. 12 04/01



(1) Must be SRO licensed.
(2) Must be RO licensed if functioning as control room operator.

[Note 1: Takes direction from the ROG manager – Communications and Public Affairs on communications issues and strategies.

OYSTER CREEK NUCLEAR GENERATING STATION
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SITE ORGANIZATION

FIGURE **13.1-4** | Sht. 1 OF 1 | Rev. 12 04/01

OCNGS
UFSAR
VOLUME
9

c. The rise in the reactor vessel water level eventually leads to high water level turbine trip and reactor scram trip.

15.1.2.3 Assumptions and Initial Conditions

The operating conditions and assumptions considered in this analysis are as follows:

a. Feedwater controller fails during maximum flow demand.

b. Maximum feedwater pump run out is assumed.

c. The reactor is operating in a manual flow control mode which provides for the most severe transient.

The assumptions and parameters used in the transient analysis are presented in Table 15.1-1.

15.1.2.4 Results of Analysis

The influx of excess feedwater flow results in an increase in core subcooling, which reduces the void fraction and thus induces an increase in reactor power. The excess feedwater flow also results in a rise in the reactor water level, which eventually leads to high water level; main turbine trip and turbine bypass valves are actuated. Reactor scram trip is actuated from main turbine stop valve position switches. Relief valves open as steam line pressures reach relief valve setpoints.

The delta CPR for cycle 18 is 0.265 and the transient responses to a feedwater controller failure are presented in 15.1-3. This transient is re-analyzed every reload cycle to assure that the mechanical and thermal overpower limits are not exceeded and there are no radiological consequence as a direct result of this transient.

15.1.3 Increase in Steam Flow

15.1.3.1 Identification of Causes and Frequency Classification

An increase in steam flow can occur due to a pressure regulator malfunction. The turbine pressure regulator can fail in either of two ways: closing the turbine control and bypass valves, or opening them. A pressure regulator malfunction causing rapid closure of the turbine control and bypass valves is discussed in Subsection 15.2.1.

An increase in steam flow is considered to be a transient of moderate frequency.

15.1.3.2 Event Description

An increase in steam flow can occur due to a pressure regulator malfunction. The increased heat removal causes reactor cooldown, pressure decrease and void formation. Neutron flux decreases via the void reactivity feedback.

Transients were considered from three power levels: stretch level, 60 percent of rated level, and hot standby. In each case, the turbine control valves would open to 10 percent of full rated. A mechanical stop prevents further valve opening.

The hot standby event has the fastest depressurization rate, as would be expected.

The decrease in pressure would result in Main Steam Isolation Valve (MSIV) closure (825 psig). A reactor scram is initiated by low reactor water level. The MSIV closure would also trip the reactor as the valves reach the 10 percent closed position.

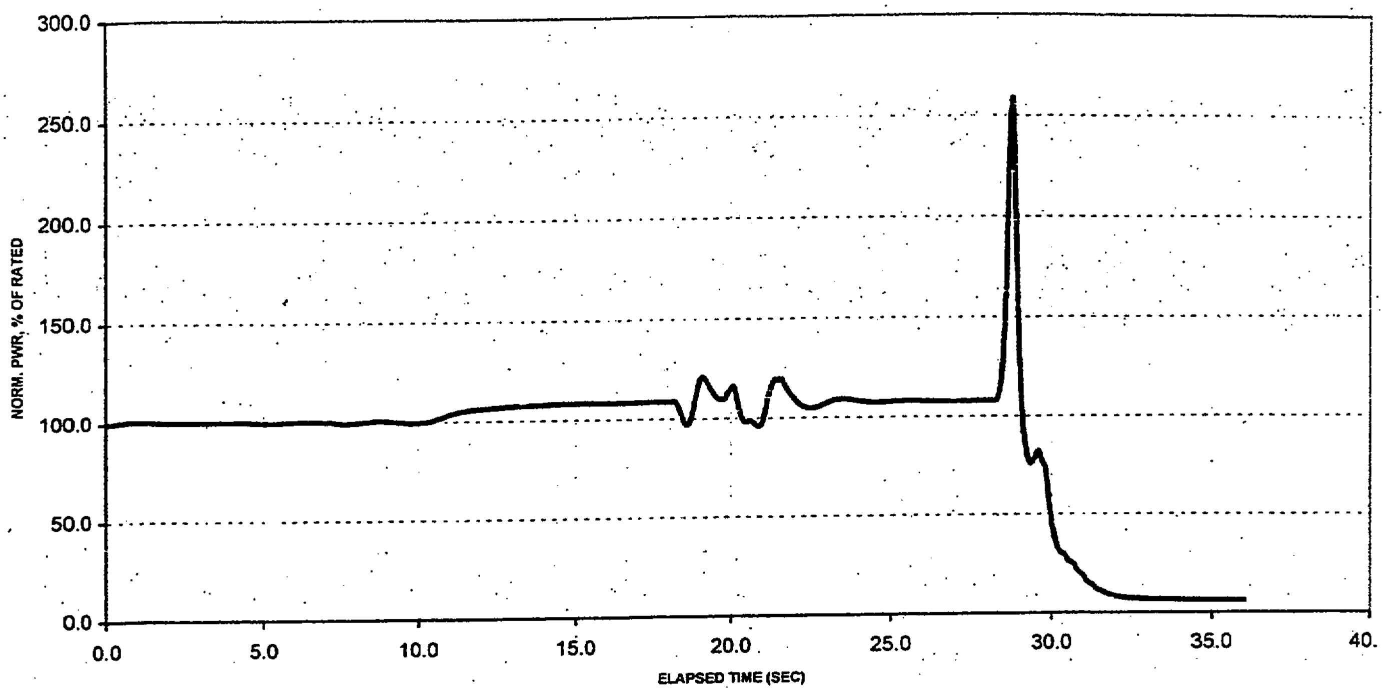
15.1.3.3 Assumptions and Initial Conditions

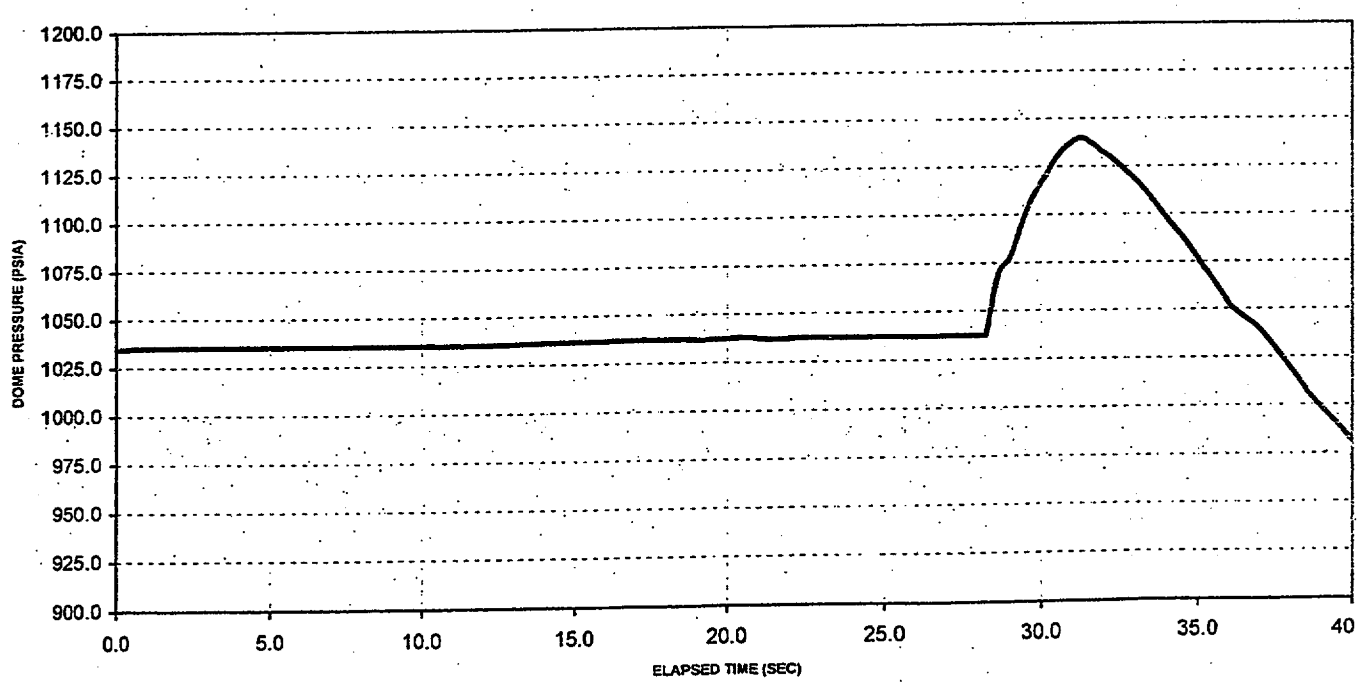
The three initial conditions analyzed were:

a. Stretch power

b. Low power (60 percent of rated) reduced by flow control

c. Hot standby.





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OYSTER CREEK NUCLEAR GENERATING STATION

Plant Response to Feedwater Controller Failure

FIGURE 15.1-3

Operator action in response to a turbine trip would be to verify automatic plant responses, such as reactor trip, bus transfer and relief valve operation and then proceed with normal plant cooldown using the Isolation Condensers and relief valves.

15.2.3.3 Assumptions and Initial Conditions

The Turbine trip **with the Turbine Bypass System available** is analyzed to demonstrate the adequacy of the relief valves to preclude safety valve operation. The plant is operating at full power and all components, instrumentation and controls are functioning normally. Conservatively a 0.1 second stop valve closure time and a 5.6 second scram time are used to maximize the pressure response.

The most limiting failure for the Turbine Trip Transient is the failure of the Turbine Bypass System in the closed position. The only alternate means of heat removal are the electromatic relief valves. No credit is taken for safety valve actuation for pressure relief. A stop valve closure time of 0.1 seconds is used. The parameters used in the transient analysis are presented in Table 15.1-1. These parameters are reload dependent.

15.2.3.4 Results of Analysis

The Turbine Trip transient **with the Turbine Bypass System available** results in a peak steam line pressure of 1156 psia and demonstrates that there is sufficient margin to lowest safety valve set point (1214.7 psia). The delta CPR for this transient is bounded by the turbine trip without bypass transient.

The turbine trip without **the Turbine Bypass System available** has an initial CPR of 1.48 and a delta CPR of 0.36 and a pressure rise of about 250 psi. See Figure 15.2-1. **This event will result in a peak steam dome pressure above the lowest safety valve setpoint, although no credit is taken for safety valve actuation for pressure relief.** In addition to the calculated delta CPR, the CPR results are subject to a statistical multiplier of 1.049 to account for code uncertainties in establishing the operating limit MCPR. It is the most limiting delta CPR transient for Oyster Creek. The results are for cycle 14 and are typical for this transient.

15.2.4 Inadvertent Closure of Main Steam Isolation Valves

15.2.4.1 Identification of Causes and Frequency Classification

Inadvertent closure of the Main Steam Isolation Valves (MSIVs) can result in vessel overpressurization and loss of steam removal path through the steam line to the turbine. Alternate paths are needed for heat removal. The inadvertent closure of the MSIVs is considered to be a transient of moderate frequency.

15.2.4.2 Event Description

Closure of the MSIVs causes an increase in the reactor vessel pressure and an automatic actuation of the pressure relief system. Reactor scram occurs on 10 percent closure of the MSIVs. The pressure rise will actuate a trip of the recirculation pumps and initiate the isolation condensers. The neutron flux increases as a result of the initial collapse of the steam voids in the core region until the flux increase is terminated by the Reactor scram.

15.2.4.3 Assumptions and Initial Conditions

The MSIV closure transient assumes the fastest MSIV closure time and no credit is taken for the recirculation pump trip and isolation condenser initiation. Oyster Creek also uses a MSIV closure transient to assess reactor vessel overpressure protection.

The current licensing basis for reactor pressure vessel overpressure event is the MSIV closure with reactor scram due to high neutron flux and is used to assess the adequacy of safety valve sizing. The analysis conservatively assumes the fastest closure of the MSIVs (3 seconds), the EMRVs do not lift, no recirculation pump trip or isolation condenser initiation, and no scram on 10 percent MSIV closure. Scram is conservatively assumed to occur on a high neutron flux trip at a 120 percent of rated power.

15.2.4.4 Results of Analysis

The inadvertent closure of the MSIVs results in a peak pressure of 1140 psia and a negligible decrease in MCPR. The pressure consequences and MCPR are much less severe than those predicted by the turbine trip without bypass and, hence, this transient is not analyzed for each reload.

The inadvertent closure of the MSIVs for assessing the adequacy of the safety valve sizing and reactor vessel overpressure protection demonstrates that the maximum vessel pressure will remain below the 1375 psig limit. For the conservative assumptions used in this analysis, the peak vessel pressure for the cycle 13 core design is 1373 psig when only nine safety valves are available for pressure relief (Figures 15.2-3 and 15.2-4).

15.2.5 Loss of Condenser Vacuum

A loss of condenser vacuum results in a loss of the main decay heat sink for the reactor. A turbine trip occurs at 22" Hg vacuum, a reactor scram occurs at greater than 20" Hg vacuum and bypass valve closure occurs at 10" Hg vacuum. This event behaves similarly to other turbine trip events. For the most extreme condition (a rapid loss of condenser vacuum) the plant response is bounded by a turbine trip with bypass failure **when reactor trip and turbine trip are both set at 22" Hg vacuum**. For either rapid

or gradual loss of vacuum conditions, relief valves and the Isolation Condensers are available to remove decay heat.

The loss of condenser vacuum is considered to be a transient of moderate frequency.

15.2.6 Loss of All AC Power/Loss of Auxiliary Power

15.2.6.1 Loss of All AC Power

The loss of all ac power (station blackout) is discussed in Section 15.9

15.2.6.2 Loss of Auxiliary Power

15.2.6.2.1 Identification of Causes and Frequency Classification

A loss of auxiliary power could occur due to electrical power distribution malfunctions. A reactor trip would occur upon loss of ac power to the Reactor Protection System.

Loss of auxiliary power causes loss of condenser cooling water, trip of feedwater pumps and trip of the recirculation pumps. Turbine trip and reactor trip ensue.

Loss of power to auxiliaries occurs with moderate frequency.

15.2.6.2.2 Event Description - Assumptions

The bypass is assumed to be available for 1.5 seconds, reducing the power/pressure transient so that the transient is less severe than the turbine trip with bypass failure. The bypass valves trip shut when the main condenser vacuum reaches 10 inches Hg. Reactor operating experience has shown that vacuum does not drop below the 10 inch setpoint until after 1.5 seconds.

The relief valves and Isolation Condensers would be available for decay heat removal. The diesel generators would be available to supply emergency power with a loss of offsite power. The diesels automatically start upon opening of the breakers. A control rod drive hydraulic pump, powered from the diesel generators, can supply 110 gpm makeup flow to the reactor.

15.2.6.2.3 Results of Analysis

The results of the analysis for this event show that even without the makeup flow the core remains well covered. The results indicate that this event causes a less severe isolation than a turbine trip without bypass, since the bypass is assumed to function immediately after the trip **Reference 1, Amendment 65)**.

15.2.7 Loss of Normal Feedwater Flow

15.2.7.1 Identification of Causes and Frequency Classification

A loss of feedwater flow could occur from pump failures, feedwater controller failures or operator error. Loss of feedwater results in a reduction of vessel inventory which causes reactor vessel water level to drop.

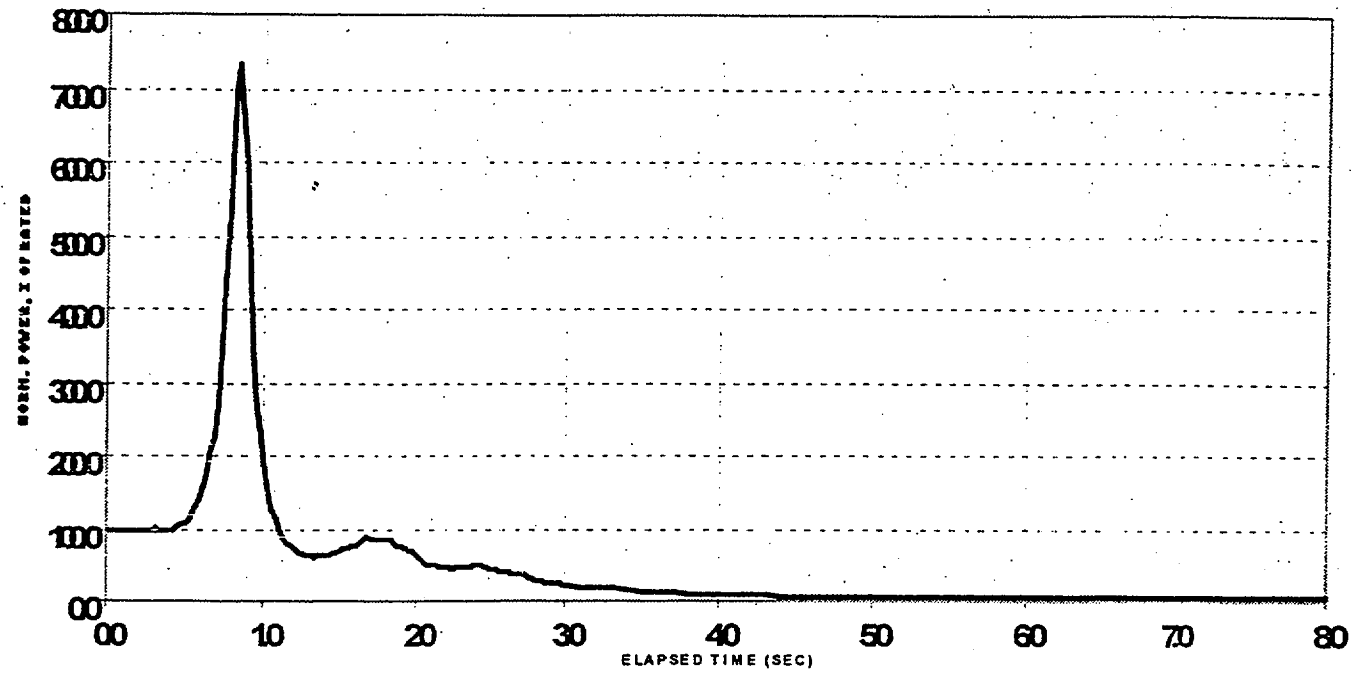
15.2.7.2 Event Description

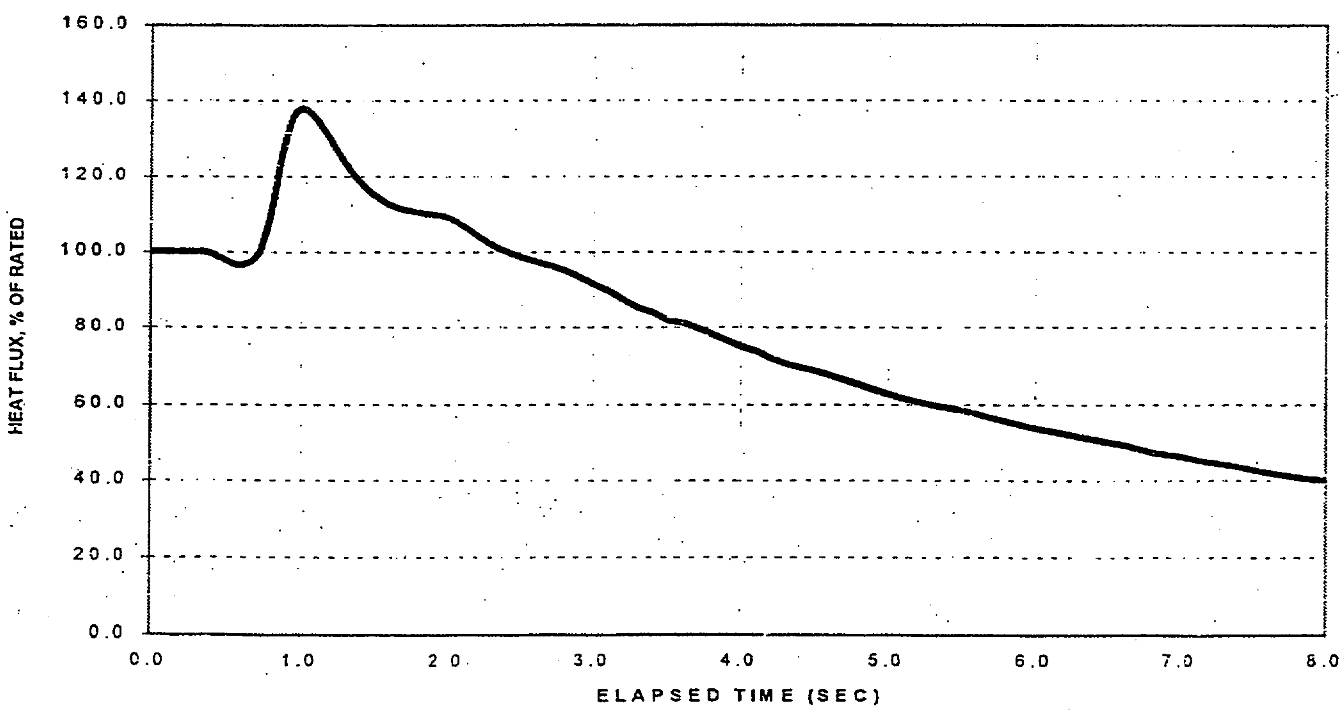
The water level drop is terminated by isolation of the main steam system. Reactor protection is provided by trips on low and low-low water levels. The sequence of events for this transient is as follows:

Event Sequence	Time (sec)	Event
1)	0	Loss of feedwater occurs.
2)	3.5	Feedwater flow decreases to zero.
3)	4.5	Reactor water level reaches low level setpoint and reactor scram occurs.
4)	15.0	Reactor water level reaches low-low level setpoint and the following events occur: a. MSIVs begin closing (10 second closing time). b. Main recirculation pumps trip. c. Isolation Condenser return valves signaled to open. d. Core spray pumps are signaled to start.
5)	35.0	Minimum downcomer water level of 5.36 feet above the top of the active fuel is reached.

After MSIV closure, the Isolation Condenser System initiates reactor coolant system depressurization. The maximum dome pressure during the transient is 1047 psia, which is below the setpoint of the relief valves. The minimum critical power ratio does not decrease below its initial steady value.

Beyond the first 125 seconds of the transient analyzed above, the sequence is straightforward. The limited amount of inventory makeup available from





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OYSTER CREEK NUCLEAR GENERATING STATION

Plant Response to Turbine Trip Without Bypass at EOC

FIGURE 15.2-1

15.4 REACTIVITY AND POWER DISTRIBUTION ANOMALIES

15.4.1 Uncontrolled Control Rod Withdrawal at Startup/Low Power

15.4.1.1 Identification of Causes

The inadvertent withdrawal of a control rod because of operator error or rod controller malfunction causes an increase in core power level and heat flux **from reactor startup or low power**.

15.4.1.2 Event Description

The inadvertent withdrawal of a control rod causes an increase in core power and heat flux for a very short period. **Continuous control rod withdrawals from levels below 1 percent power will be terminated by neutron flux scram at conditions even milder than those reported for rod withdrawal at power.** In the intermediate and power range instrumentation, high neutron flux scrams are located at the top of each instrument range. Interlocks prevent rod withdrawal if the instruments are not reading on scale unless they are set on their lowest range. Therefore, abnormal power increases **would be** terminated before the power could reach cladding damaging conditions. **Rather than analyze rod withdrawals from each intermediate power level, a rod withdrawal from cold conditions was analyzed to bound the event. The analysis assumed a control rod worth of 0.025% dk/k withdrawn at the maximum withdrawal rate, (giving a maximum reactivity insertion rate of 0.0019 dk/sec assumed constant during the withdrawal), and no credit was taken for the intermediate level scram.**

15.4.1.3 Assumptions and Initial Conditions

a. The reactor is **in the cold critical condition.**

b. No xenon or samarium present.

c. The transient rod has a rod worth of 2.5 **%** delta k/**k** and withdrawn at the maximum withdrawal rate.

d. No intermediate **range** scram is assumed.

15.4.1.4 Results of Analysis

The analysis considers the continuous withdrawal of the maximum worth control rod at its maximum speed from the reactor which is operating at low power or from the startup mode. The excursion was turned by the inherent negative Doppler reactivity defect from fuel heating, **and the event was terminated by** reactor scram, at over 120 percent of full power. In actuality, the reactor would have been scrammed at a much lower power level either by the intermediate range or power range instrumentation. **Even with these assumptions, the minimum reactor period was about 30 milliseconds and the reactor power peaked at 4500 MW. About 530 MW-second of energy is generated in the high power density region. This energy generation would increase the fuel temperature at the hottest point to 2500°F and** no fuel melting or cladding damage **would occur (Reference 1, Amendment 3)**.

15.4.2 Uncontrolled Control Rod Withdrawal at Power

15.4.2.1 Identification of Causes and Frequency Classification

The inadvertent withdrawal of a control rod because of operator error or rod controller malfunction causes an increase in core power level and heat flux. Severe local peaking can also occur as a result of this transient.

15.4.2.2 Event Description

The inadvertent withdrawal of a control rod causes an increase in core power level **local peaking** and heat flux. Protection is afforded by a rod block from the Average Power Range Monitoring (APRM) System of the Reactor Protection System. The APRM system uses signals from the Local Power Range Monitors (LPRMs) to measure core power. When an increase in power **up to the APRM Rod Block setpoint** is detected, rod block signal is generated.

The APRM Rod Block is flow biased in order to provide protection for the event for all power and flow conditions. The full power conditions is the most limiting case. The event is analyzed for each reload to insure that the APRM Rod Block setpoint provides the appropriate protection.

15.4.2.3 Assumptions and Initial Conditions

The following plant operating conditions and assumptions provide the basis for the analysis of the control rod withdrawal transient.

a. The plant is operating at full power and full flow.

b. Transient occurs at the most reactive point in the cycle.

c. No xenon present.

d. Control Rod Pattern which maximizes the reativity insertion.

e. Transient rod is fully inserted, adjacent rods withdrawn.

f. The highest worth control rod and the most limiting location in terms of instrument response, control rod 42-31, are each analyzed for this event.

15.4.2.4 Results of Analysis

The control rod withdrawal error is analyzed as a series of steady-state calculations in two foot increments as the control rod is withdrawn from the core. The nuclear instrumentation response is calculated with the change in critical power ratio (Delta CPR) and total peaking until the APRM flow biased rod block terminates the rod withdrawal error at the maximum setpoint of 108% power. The Cycle **18** response, which is typical for this transient, is shown in Table 15.4-1. The control rod pattern used in the analysis is shown in Figure 15.4-1. This control rod pattern is not typical of control rod patterns used in operation. It is contrived to place the fuel bundles that surround the transient control rod near the operating CPR limit.

15.4.3 Control Rod Maloperation (System Malfunction or Operator Error)

This event is not analyzed for Boiling Water Reactors.

15.4.4 Startup of an Inactive Loop at an Incorrect Temperature

15.4.4.1 Identification of Causes and Frequency Classification

The startup of an idle loop could result in a cold water addition to the core, which could cause a power increase through void collapse.

The occurrence of this event is considered to be a transient of moderate frequency.

15.4.4.2 Event Description

Oyster Creek is a non-jet pump plant with five recirculation loops, each with isolation valves. Procedures and interlocks on pump starting and valve opening effectively prevent such a transient. Multiple failures are required for a severe idle loop startup event to occur.

Procedures require that the loop discharge valve be closed and the suction valve open before the pump is started (a small bypass line prevents deadheading the pump). After the pump is running (at 30 percent), the operator can open the discharge valve. When it is fully open, the operator closes the bypass valve and then the pump can be placed in automatic speed control.

Normally a loop will not be isolated when its pump is shut down so that reverse flow via the bypass valve will keep the loop hot. A 50°F differential requires reactor shutdown to restart the pump.

15.4.4.3 Assumptions and Initial Conditions

The analysis performed for startup of an inactive loop (Reference 2) is based on the following assumptions:

a. Reactor thermal power level is 1930 MW.

b. Water in the isolated loop is at 100°F.

c. The suction and bypass valves are opened at the same time.

d. Coincident with valve opening, the pump is started and quickly brought up to speed.

e. The discharge valve is opened as soon as the pump is started.

f. percent power and 100 percent flow from four running pumps.

g. Scram setting is at 116 percent power.

The analysis of startup of an idle loop is also equivalent to the current NRC criteria. This is based on the following:

a. Similar conservative assumptions are made with respect to initial conditions.

b. An acceptable calculational model(s) is used in the analysis.

c. The acceptance criteria for the event are the same.

TABLE 15.4-1
(Sheet 1 of 1)

Table 15.4-1 – CPR and MLLHGR for Control Rod Withdrawal Error Maximum Allowable Failure Combination for LPRMs

APRM % Power	Δ CPR	MLLHGR (KW/FT)	Rod Withdrawal (Feet)
100.0	0.0	13.4	0.0
101.0	**0.07**	**13.5**	**3.5**
102.0	**0.11**	**13.6**	**4.5**
103.0	**0.15**	**13.7**	**5.5**
104.0	**0.21**	**13.5**	**8.5**
105.0	**0.21**	**13.5**	**9.0**
106.0	**0.21**	**13.5**	**9.5**
107.0	**0.21**	**13.5**	**9.5**
108.0	**0.22**	**13.5**	**10.0**
109.0	**0.22**	**13.5**	**10.5**
110.0*	**0.22**	**13.5**	**10.5**

* APRM Rod Block

Figure 15.4-1
(Sheet 1 of 1)

Limiting Rod Withdrawal Error Rod Pattern

26	30	34	38	42	46	50	
							51
16		16					47
							43
00		24		24			39
	20						35
20				00*			31
	20						27

NOTES: 1. Rod pattern is ¼ core mirror symmetric

2. Number indicates notches withdrawn.
Blank denotes rod is fully withdrawn (rod at 48).

3. Asterisk (*) denotes the transient control rod.

Core Exposure:	**7.0 GWD/MT (peak cycle reactivity)**
Reactor Power:	**1930 MWth**
Recirculation Flow:	**61.0 Mlb/hr**
System Pressure:	**1050 psia (core mid-plane)**
Inlet Subcooling:	**34.2 BTU/LBM**

Figure 15.4-8

(Deleted)

Figure 15.4-9

(Deleted)

Figure 15.4-10

(Deleted)

Figure 15.4-11

(Deleted)

Figure 15.4-12

(Deleted)

15.6.5.5.1 Fuel Exposure Considerations

As discussed in Reference 2, the ECCS acceptance criteria of 10CFR50.46 which are most significant to the BWR/2 LOCA analysis require that the calculated PCT following a postulated LOCA shall not exceed 2200°F and that the calculated maximum cladding local oxidation fraction shall not exceed 17%. For a BWR/2 plant, the ECCS performance is limited by difference factors as the fuel exposure increases. At low exposures (up to approximately 15 GWD/MTU for current fuel designs), the PCT limit of 2200°F is the most restrictive acceptance criterion. At higher exposures, the 17% local oxidation limit becomes more restrictive than the PCT limit.

The break spectrum evaluation was performed at a low fuel exposure (11 GWd/MTU), to identify the limiting break conditions and to determine a MAPLHGR which would keep the calculated PCT (using Appendix K assumptions) just below the 2200°F limit. The 11 GWd/MTU fuel exposure was selected to represent the low exposure range, thereby obtaining a high MAPLHGR while avoiding the considerations of fuel cladding perforation due to internal pressure at high exposure.

The limiting break condition was also evaluated at a representative higher exposure (25 GWd/MTU), where fuel rod perforations were calculated to occur (using Appendix K assumptions). At these higher exposures, the maximum cladding oxidation limit of 17% becomes the limiting factor on fuel performance.

15.6.5.5.2 Recirculation Line Breaks

A sufficient number of break sizes and ECCS failure combinations were evaluated using nominal input conditions. The results (Table 15.6-3a) identified the recirculation line discharge break as limiting. Analyses with Appendix K input assumptions were performed for four break sizes from the limiting scenario determined by the nominal break spectrum. Table 15.6-3b lists the Appendix K PCT results which show the same trend.

The DBA recirculation line discharge break with ADS valve failure is the limiting break for the nominal break spectrum with a calculated peak cladding temperature of 1831°F (low exposure). The corresponding PCT for this break with Appendix K specified models was calculated to be 2196°F and 2027°F for the low and high exposures, respectively. Plots showing system responses for all break spectrum cases are presented in Figures 15.6-3 through 15.6-25.

15.6.5.5.3 Non-Recirculation Line Breaks

Evaluations were also performed for some of the non-recirculation line breaks. These breaks (including feedwater, core spray and main steam lines) were evaluated with the nominal input conditions and maximum line break sizes. PCT results (Table 15.6-3c) show that these non-recirculation line guillotine breaks are far from becoming candidates for the limiting event. The same conclusion applies for break sizes smaller than the guillotine break for these lines. The system responses of these breaks were also presented in Figures 15.6-3 through 15.6-25.

Oyster Creek plant-specific evaluations were not performed for other non-recirculation line breaks (e.g., EC lines, liquid instrument lines, cleanup system lines, etc.). These non-recirculation line breaks will not become candidates for the limiting event, since they are essentially the same as small recirculation or steam line breaks.

15.6.5.6 **Three or Four Loop Operation**

There are two main differences in the LOCA analysis for operation with three- and four-recirculation loop operation as compared to the normal five-loop case:

(1) With fewer operating loops, each functioning loop will be carrying a higher percentage of the initial core recirculation flow. If a break in one loop occurs, then a faster core flow coastdown rate will result, which could yield an earlier boiling transition time.

(2) **If the inactive loop is isolated (by closing the inactive loop suction, discharge and discharge bypass valves) there will be reduced coolant inventory which may lead to earlier core uncovery during a LOCA.**

The effects of these differences on the SAFER calculations will depend on the break size. For the limiting large breaks, there will be no effect due to reduced core flow coastdown, because no coastdown credit was taken in the SAFER analysis of large breaks. There is some impact on large break results due to the reduced inventory of isolated loops; however, the impact is small because core uncovery occurs almost immediately. For the small breaks (in which there is a large margin to the **10CFR50.46** limits), the reduced inventory due to the isolated loops may have a small effect on PCT because of the earlier core uncovery. The impact of the reduced core flow coastdown on small breaks is not significant.

Evaluation results (considering both P8x8R and GE8x8EB fuel types) for four-loop operation, with the inoperative loop isolated, are summarized in Table 15.6-4. The DBA break (low and high exposure) and the 0.1 ft^2 break (highest small-break PCT from the nominal analysis) cases were calculated with exposure-dependent MAPLHGR reductions and then compared with the five loop base cases. The results show that these MAPLHGR reductions are adequate to compensate for the effect of loss of inventory and the faster coastdown for the large and small breaks. Table 15.6-4a summarizes the four loop MAPLHGR multipliers for **all Oyster Creek GE 8x8 reload fuel designs**.

For three-loop operation, only one of the two inoperable loops is permitted to be isolated. Therefore the reduced inventory is the same as the four-loop analysis with one isolated loop. Since the coastdown rate is not used the large break analysis, the four-loop multiplier with one isolated loop is the same for three-loop operation with one isolated loop. The power limitation to 90% of rated during three-loop operation compensates for the coastdown effects during the small break LOCA. If none of the loops are isolated there is no reduction in the MAPLHGR limits. Therefore the four loop multipliers are applicable to both four and three loop operation with one isolated loop as shown in table 15.6.4a.

15.6.5.6.1 Reduced Core Flow Operation (ELLLA)

The impact, on MAPLHGR limits, of operating at rated reactor power and reduced core flow (i.e., in the Extended Load Line Limit Analysis (ELLLA) region) were evaluated. The effects of operating in the ELLLA region will be similar to the four loop operation, except that there is no isolated loop effect. Therefore, the five loop operation results at rated flow are applicable, and no MAPLHGR multiplier is required.

15.6.5.7 Technical Specification MAPLHGR Limits

GE BWR MAPLHGR limits (as a function of fuel exposure) are based on the most limiting value of either the MAPLHGR determined from LOCA/ECCS analysis or the MAPLHGR determined from fuel thermal mechanical design considerations.

For BWR/2 plants, in general, and the Oyster Creek plant, specifically, the MAPLHGR calculated from the LOCA/ECCS evaluation is limiting for most of the exposure range and determines the Technical Specification limits.

The MAPLHGR limits for the P8x8R and GE8x8EB and GE8x8NB fuel bundles were evaluated as a function of exposure with the limiting scenario (postulated recirculation line DBA discharge break with failure of ADS Valve) identified in the break spectrum analyses. The MAPLHGR Limits for current fuel designs are contained in the Core Operating Limits Report (COLR).

15.6.5.8 Radiological Consequences

The radiological consequences from a loss of coolant accident (LOCA) were evaluated at a power level of 1930 MWt. The assumptions in Safety Guide 3, as documented in Amendment 68 of reference 1 were utilized for the evaluation.

The LOCA offsite doses **are** tabulated below:

Two Hour Dose (600 meters)	Thyroid	145.0 Rem
	Whole Body	9.5 Rem
Low Population **Zone** (30 Days)	Thyroid	117.0 Rem
	Whole Body	4.5 Rem

These doses are well below the guidelines of 10CFR100 even when the conservative assumptions of the Safety Guide are used.

15.6.6 A Number of BWR Transients (Including Items 2.7, 2.8 and 1.3)

A number of BWR transients are discussed in the subsections given below.

15.6.6.1 Loss of Normal Feedwater Flow
(Regulatory Guide 1.70, Chapter 15, Item 2.7)

See Subsection 15.2.7.

15.6.6.2 Feedwater Piping Break
(Regulatory Guide 1.70, Chapter 15, Item 2.8)

See Subsection 15.2.8.

15.6.6.3 Increase in Steam Flow (Pressure Regulator Malfunction)
(Regulatory Guide 1.70, Chapter 15, Item 1.3)

See Subsection 15.1.3.

TABLE 15.6-1
(Sheet 1 of 3)

OPERATIONAL AND ECCS PARAMETERS

A. Plant Parameters

Core Thermal Power (MWth)		
Nominal	1930	(100% of Rated)
Appendix K	1969	(102% of Rated)
Vessel Steam Flow (lbm/hr)		7.4×10^6, corresponds to 102 percent of rated core power
Vessel Steam Dome Pressure (psia)		1035
Maximum Recirculation Line Break Area (ft.^{2})		4.66
Initial MCPR		1.30
Initial Water Level		1.0 ft. Scram Trip Level

B. Emergency Core Cooling System Parameters

Core Spray System

Assumed System Configuration:		
One Loop		One Main Pump and One Booster Pump
The Other Loop		One Main Pump
Vessel Pressure versus System Flow Rates:		
	PSIG	GPM
One Loop	0	4100 (Runout)
	110	3400 (Rated)
	300	0 (Shut-Off)
The Other Loop	0	3700 (Runout)
	110	2200 (Rated)
	169	0 (Shut-Off)
*Initiating Parameters		
Reactor Water Level		<6.17 ft. above Top of Active Fuel (TAF)
Or		
Drywell Pressure (psig)		>3.5

* The actuation signals for Core Spray are high drywell pressure or lo-lo reactor water level. See Technical Specification for current setpoints.

TABLE 15.6-1
(Sheet 2 of 3)

OPERATIONAL AND ECCS PARAMETERS

***Maximum Allowable Delay Time from Initiating Signal to Pump at Rated Speed (sec)	15
***Injection Valve Stroke Time (sec)	20
Pressure Permissive at Which Injection Valve Opens (psig)	285
Core Spray Flow to Hot Bundle (2 headers) (gpm)	3.4 (at 30 psia)
ADS	
Total Number of Valves in System	5
Number of Valves Assumed in Analysis	4
Number of Valves Available After Single Failure	3
**Minimum Flow Capacity of 3 Valves (lbm/hr) at Vessel Pressure (psig)	1.67×10^6 at 1120
*Initiating Parameters	
Reactor Water Level	<3.67 ft above TAF
And	
Drywell Pressure (psig)	>3.5
Time Delay After Initiating Signals (sec)	120
Emergency Condensers	
Total Number of Emergency Condensers Assumed in Analysis	None

* The actuation signals for the ADS timer are high drywell pressure, lo-lo reactor water level, and core spray booster pump differential pressure. See Technical Specifications and Operating Procedures for current setpoints.

** The updated value of EMRV is rated 602,900 lb/hr at 1250 psig in Chapter 6. This assumed minimum flow rate at 1120 psig for the LOCA analysis would be slightly (by about 3%) less than the calculated value based on a linear relationship provided by the valve manufacturer. However, a Safety Evaluation (Reference 20) has reviewed its impact to small break LOCA for raising the calculated PCT by no more than 14°F. Thus the PCT will be in the order of 974°F and there is still a large margin to the safety limit of 2200°F.

TABLE 15.6-1
(Sheet 3 of 3)

OPERATIONAL AND ECCS PARAMETERS

*** The Core Spray Maximum Allowable Delay Time from Initiating Signal to Pump at Rated Speed of 15 seconds and the Injection Valve Stroke Time of 20 seconds are nominal values. The initiating signal to start core spray assumed in the GE LOCA analysis is Lo-Lo reactor water level or High drywell pressure. Since this signal is break size dependent, linking the EDG response time described in Chapter 8 with Core Spray Maximum Allowable Delay Time is not warranted. The requirement to have Core Spray System reaching required flow within 35 seconds from the occurrence of the large break LOCA (Chapter 6.2.1.3.3) is met based on the capability of the EDG to respond to a LOOP event in 20 seconds (Chapter 8.3) as well as the ability of the Core Spray Pumps and Injection Valves to deliver the required flow within this time period. **The valves pass the required flow without being fully open.** Maximum allowable Injection Valve Stroke Time is **listed in** Table 6.3-4.

TABLE 15.6-4A
(Sheet 1 of 1)

MAPLHGR MULTIPLIERS FOR REDUCED LOOP OPERATION

GE 8X8 Fuel Designs

Number and Condition of Isolated Loops	MAPLHGR MULTIPLIER
One Loop Unisolated	1.0
One Loop Isolated	0.98
Two Loops Unisolated	1.0
One Loop Isolated and One Loop Unisolated	0.98
Two Loops Isolated	N/A*

*** Reactor operation with two Loops Isolated is not permitted**

The fuel assembly is expected to impact on the reactor core at a small angle from the vertical, possibly inducing a bending mode of failure on the fuel rods of the dropped assembly. It is assumed that each fuel rod resists the imposed bending load by a couple consisting of two equal, opposite concentrated forces. Therefore, fuel rods are expected to absorb little energy prior to failure as a result of bending. Actual bending tests with concentrated point-loads show that each fuel rod absorbs approximately 1 ft-lb prior to cladding failure. Each rod that fails as a result of gross compression distortion is expected to absorb approximately 250 ft-lb before cladding failure (based on 1 percent uniform plastic deformation of the rods). The energy of the dropped assembly is conservatively assumed to be absorbed by only the cladding and other core structures.

Because a fuel assembly consists of 72 percent fuel, 11 percent cladding, and 17 percent other structural material by weight, the assumptions that no energy is absorbed by the fuel material results in considerable conservatism in the mass-energy calculations that follow.

The energy absorption and successive impacts is estimated by considering a plastic impact. Conservation of momentum under a plastic impact shows that the fractional kinetic energy absorbed during impact is:

$$1 - \frac{M_1}{M_1 + M_2}$$

where M_1 is the impacting mass and M_2 is the struck mass. Based on the fuel geometry in the reactor core, four fuel assemblies are struck by the impacting assembly. The fractional energy loss on the first impact is approximately 80 percent.

The second impact is expected to be less direct. The broad side of the dropped assembly impacts approximately 24 more fuel assemblies, so that after the second impact only 136 ft-lb (approximately one percent of the original kinetic energy) is available for a third impact. Because a single fuel rod is capable of absorbing 250 ft-lb in compression before cladding failure, it is unlikely that any fuel rod will fail on a third impact.

If the dropped fuel assembly strikes only one or two fuel assemblies on the first impact, the energy absorption by the core support structure results in approximately the same energy dissipation on the first impact as in the case where four fuel assemblies are struck. The energy relations on the second and third impacts remain approximately the same as in the original case. Thus, the calculated energy dissipation is as follows:

First Impact	80 percent
Second impact	19 percent
Third impact	1 percent (no cladding failures)

The first impact dissipates 0.80 x 17,000 or 13,600 ft-lb of energy. It is assumed that 50 percent of this energy is absorbed by the dropped fuel assembly and that the remaining 50 percent is absorbed by the struck fuel assemblies in the core. Because the fuel rods of the dropped fuel assembly are susceptible to the bending mode of failure and because 1 ft-lb of energy is sufficient to cause cladding failure as a result of bending, all 62 rods of the dropped fuel assembly are assumed to fail. Since the tie rods of the struck fuel assemblies are more susceptible to bending failure than the other 54 fuel rods, it is assumed that they fail on the first impact. Thus, 4 x 8 = 32 tie rods (total in 4 assemblies) are assumed to fail.

Because the remaining fuel rods of the struck assemblies are held rigidly in place in the core, they are susceptible only to the compression mode of failure. To cause cladding failure of the fuel rod as a result of compression, 250 ft-lb of energy is required. To cause failure of all the remaining rods of the 4 struck assemblies, 250 x 56 x 4 or 56,000 ft-lb of energy would have to absorbed in cladding alone. Thus, it is clear that not all the remaining fuel rods of the struck assemblies can fail on the first impact. The number of fuel rod failures caused by compression is computed as follows:

$$\frac{0.5 * 13{,}600 * \left(\frac{11}{11+17}\right)}{250} = 11$$

During the first impact, fuel rod failures are as follows:

Dropped assembly	62 rods (bending)
Struck assemblies	32 tie rods (bending)
Struck assemblies	11 rods (compression)
	105 failed rods

Because of the less severe nature of the second impact and the distorted shape of the dropped fuel assembly, it is assumed that in only 2 of the 24 struck assemblies are the tie rods subjected to bending failure. Thus, 2 x 8 = 16 ties are assumed to fail. The number of fuel rod failures caused by compression on the second impact is computed as follows:

$$\frac{\left(\frac{0.19}{2}\right) * 17{,}000 * \left(\frac{11}{11+17}\right)}{250} = 3$$

Thus, during the second impact, the fuel rod failures are as follows:

Struck assemblies	16 tie rods (bending)
Struck assemblies	3 rods (compression)
	19 failed rods

The total number of failed rods resulting from the accident is as follows:

First impact	105 rods
Second impact	19 rods
Third impact	0 rods
	124 total failed rods

15.7.4.5 Radiological Consequences

Based on the linear heat generation rate applicable to P8x8R fuel, it can be theoretically predicted that the fractional plenum activity will be approximately one tenth of that activity contained in the plenum of a 7x7 fuel rod. For the purpose of this evaluation, it was conservatively assumed that the fraction plenum activity for the P8x8R rod is the same for the 7x7 rod. Since each P8x8R fuel bundle produces the same power as a 7x7 bundle, the average activity per rod for the P8x8R bundle will be 49/62, or 0.79, times the activity in a 7x7 rod. Based on the assumption that 124 of the P8x8R rods fail compared to 111 for a 7x7 core, the relative amount of activity released for the P8x8R fuel is (124/111) (0.79) = 0.88 times the activity released for a 7x7 core. The activity released to the environment and the radiological exposures for the P8x8R fuel will, therefore, be less than 96 percent of those values evaluated for a 7x7 core. As identified previously the radiological exposures for the 7x7 fuel are well below those guidelines set forth on 10CFR100; therefore, it can be concluded that the consequences of this accident for the P8x8R fuel will also be well below these guidelines **(Reference 21, Reference 1 Amendment 3)**.

15.7.5 Spent Fuel Cask Drop Accident

Section 9.1 provides detailed information on fuel storage and handling. Subsection 9.1.2.2.3 provides detailed analysis of Cask Drop Protection System (CDPS) and Subsection 9.1.2.3.10 discusses the cask drop analysis. As stated in that subsection, 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.

To preclude such an accident, a CDPS has been installed, and the path of the cask during handling is strictly controlled. The CPDS constitutes a passive system which has no effect on normal fuel handling operation, and reduces the likelihood of a cask drop accident.

In the event of a cask drop accident inside the spent fuel pool, the CDPS 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 effect, a base plate is attached to the bottom of the cask.

The CDPS is attached to the spent fuel pool wall by means of two stainless steel tie rods. 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. The top plate and the upper guide cylinder are segmented by 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.

15.10 REFERENCES

(1) Oyster Creek Nuclear Power Plant, Facility Description and Safety Analysis Report, and Amendments.

(2) NEDE-30996P-A SAFER Model for Evaluation of Loss of Coolant Accidents for Jet Pump and Non-Jet Pump Plants

(3) NEDC-31462P Oyster Creek Nuclear Generating Station SAFER/CORECOOL/GESTR-LOCA Loss of Coolant Accident Analysis.

(4) Integrated Plant Safety Assessment, Systematic Evaluation Program for the OCNGS, NUREG-0822, January 1983.

(5) "Issuance of Amendment (TAC No. 67743)", Docket No. 50-219, USNRC, October 31, 1988.

(6) "Methods for the Analysis of Boiling Water Reactors Lattice Physics," TR-020-A, Revision 0, January 1988.

(7) "Methods for the Analysis of Boiling Water Reactors Steady State Physics," TR-021-A, Revision 0, January 1988.

(8) "Methods for the Generation of Core Kinetics Data for RETRAN-02," TR-033-A, Revision 0, May 1988.

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(12) "Reload Information and Safety Analysis Report for Oyster Creek Cycle 13 Reload", TR-076, Revision 0, March 1991.

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(17) "OC Turbine Building Response to Main Steam Line Breaks with Delayed Isolation", C1302-822-5450-047, Rev. 0.

(18) "Core Level Response to MSL Break with Varied MSIV Time Delay", C1302-411-5450-048, Rev. 0.

(19) "Release from Steam Line Failure with Isolation in 15 Seconds", C1302-411-5450-050, Rev. 0.

(20) Safety Evaluation SE00212-039, "Change in EMRV Flow Rate in FSAR Update", December 1997.

(21) **NEDO-24195, "GE Reload Fuel Application for Oyster Creek."**

(22) **Letter, Victor Stello (NRC) to Ivan R. Finfrock, Jr. (JCP&L), "Oyster Creek Restart Safety Analysis Report," May 30, 1979.**