CHAPTER 1- INTRODUCTION AND GENERAL DESCRIPTION OF PLANT

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CHAPTER 1

INTRODUCTION AND GENERAL DESCRIPTION OF PLANT

1.1 INTRODUCTION

The Oyster Creek Nuclear Generating Station (OCNGS) is a single unit facility. It is located in Lacey Township, Ocean County, New Jersey, approximately two miles south of the community of Forked River. Initial criticality was achieved on May 3, 1969 and OCNGS was placed in commercial operation on December 23, 1969 under a Provisional Operating License. On July 2, 1991, the NRC issued a Full Term Operating License (Facility Operating License No. DPR-16) which superceded the Provisional Operating License in its entirety. This License permits steady-state reactor core power levels not in excess of 1930 megawatts (thermal) and is in effect until midnight on April 9, 2009 (Technical Specification Amendment 163).

The prime contractor for the plant was the General Electric Company. The General Electric Company utilized the services of Burns and Roe, Inc. for engineering support and construction management. The unit's steam is generated by a Boiling Water Reactor (BWR-2) with a Mark I type Containment designed by the Chicago Bridge and Iron Company under contract to Burns and Roe, Inc.

1.2 GENERAL PLANT DESCRIPTION

1.2.1 Description of the Site

The Oyster Creek site is located near the Atlantic Ocean within the State of New Jersey. The site, about 152 acres, is partly in Lacey and partly in Ocean Townships of Ocean County, New Jersey, about two miles inland from the shore of Barnegat Bay and about seven miles west-north-west of Barnegat Light. The site is approximately nine miles south of Toms River, New Jersey, about fifty miles east of Philadelphia, Pennsylvania, and sixty miles south of Newark, New Jersey.

The Barnegat Bay region of New Jersey is a well known summer resort area, attracting visitors from much of the Middle Atlantic seaboard. The population distribution of the area surrounding the site changes significantly from winter to summer with the influx of summer vacationers during the months of June, July and August. Details on population distribution and expected growth are presented in Section 2.I.

The region within 40 miles of the site encompasses approximately 2,700 square miles of land. This area has very little industry; in fact, within 40 miles of the site, approximately 75 percent of the land is forest, vacant land, or farm land. Only about 25 percent of the land is developed.

The site is in a meteorological transition zone between the continent and the ocean and the weather at the site is conditioned by this location. The prevailing winds are offshore. Thus, site weather is influenced on the average more by continental than by maritime weather.

The Island Beach peninsula and Long Beach Island provide a barrier between Barnegat Bay and the Atlantic Ocean. This barrier, along with the shallowness of the Bay, minimizes tidal fluctuations in the Bay. A survey immediately north of the Oyster Creek site at Forked River showed a high tide elevation of 4.5 feet above mean sea level (MSL). Grade level at the site is 23 feet above MSL which is well above the 4.5 feet recorded, and there is no record of the site area being flooded or inundated even during storms with high tidal conditions.

Water supplies in the area surrounding the site are derived from wells. Wells in the area generally are 60 to 70 or more feet in depth to preclude contamination from salt water intrusion or the many septic tanks in the area.

Condenser cooling water is drawn from Barnegat Bay through a canal following the South Branch of Forked River and discharged through another canal following Oyster Creek to the Bay. To limit discharge temperatures, dilution pumps have been installed at the plant intake to divert water from the intake to the discharge for thermal dilution. The grade level of 23 feet above MSL at the site makes flooding very unlikely in light of past storm and tide records wherein the worst case appears to be a tidal height of 4.5 feet above MSL.

The Oyster Creek site lies in an area known geologically as the coastal plain.

Buildings and structures are founded generally in Cohansey sand. Compression tests in the Reactor Building and Turbine Building areas, using 2.5 times the normal design loadings and I.5 times the earthquake design loadings, gave satisfactory deflections.

1.2.2 Description of the Facility

1.2.2.1 Summary Plant Data

A summary of plant data is presented in Table 1.2-1. The site plan and plant general arrangement drawings are presented in Drawings JC-19702, 3E-153-02-001 through 3E-153-02-009, 3E-151-02-001 through 3E-151-02-009, 3E-156-02-001 and 002, 3E-154-02-001 through 003, 3E-155-02-001 through 005, 3E-175-02-001 and 002, 3E-157-02-001, 3E-158-02-001, 3E-162-02-001 through 003, 3E-167-02-001, 3E-168-02-001, 3E-169-02-001, 3E-170-02-001, 3E-176-02-001, 3E-185-02-001, 3E-186-02-001.

1.2.2.2 Plant Structures and Components

The plant is comprised of the following major buildings and structures:

a. Reactor Building
b. Turbine Building
c. Office Building
d. Old Radwaste Building
e. New Radwaste and Offgas Buildings
f. Emergency Diesel Generator Building
g. Intake and Discharge Structure
h. Ventilation Stack
i. Storage Tanks

These buildings and structures and the systems housed within them are briefly described below. The classification of structures is discussed in Section 3.2. The design of Category I buildings is addressed in Section 3.8.

Reactor Building

The Reactor Building (Drawings 3E-153-02-001 through 3E-153-02-009) is constructed entirely of reinforced concrete to the refueling floor level at El. 119'-3". Above the refueling floor, the structure is steel framework with insulated, corrosion resistant metal siding. The foundation mat is 146 feet by 146 feet and about 10 feet thick. The finished top surface of the mat is at El.(-)19'-6" (or 42'-6" below grade). The drywell support pedestal in the center of the mat is a concrete cylinder about 67 feet in diameter and 19 feet, 5 inches high. The drywell is centered on the pedestal and shielded by concrete to the level of the refueling floor. Top shielding is provided by removable concrete plugs. The torus is supported from the mat by structural framework. Refer to Section 3.8.

The Reactor Building houses the reactor and its auxiliary systems. The reactor vessel and the recirculation system are contained inside the drywell containment system. The Primary Containment System consists of the drywell, ventpipes, and a pool of water contained in the absorption chamber. The Reactor Building encloses the Primary Containment System thereby providing a secondary containment. In addition, all refueling equipment is inside the building, including the spent fuel storage pool and the new fuel storage vault.

The reactor is a single cycle, forced circulation boiling water reactor producing steam for direct use in the steam turbine. The fuel consists of uranium dioxide pellets contained in sealed Zircaloy-2 rods. Water serves as both the moderator and coolant. Refer to Chapter 4 for additional detail.

The reactor core includes the fuel assemblies and control rods. The mechanical, thermal hydraulic, and nuclear design of this reactor is similar to several other boiling water reactors designed and built by the General Electric Company.

The blades of the control rods consist of an assembly of stainless steel tubes filled with compacted boron carbide powder. These tubes are held in a cruciform array by a stainless steel sheath with castings at each end. The lower casting is provided with a rod drop velocity limiter.

Water enters the bottom of the core and flows upward through the fuel assemblies where boiling produces steam. The steam water mixture is separated by steam separators and dryers located within the reactor vessel. The steam passes through steam lines to the turbine. The separated water mixes with the incoming feedwater and is returned to the core inlet through the recirculation loops.

By the use of planned rod withdrawal sequences, the reactivity potential available for reactivity addition accidents is limited so that if a control blade were to drop out of the core, no primary system damage would result. An interlocking device, the Rod Worth Minimizer, supplements procedural control by enforcing the use of acceptable rod sequences in accordance with the Technical Specifications.

The reactor service and refueling area, on the top floor of the Reactor Building, is served by an overhead bridge crane. A refueling service platform with the necessary handling and grappling fixtures serves the refueling area and spent fuel storage pool.

The Primary Containment was designed to accommodate the pressures and temperatures resulting from pipe breaks up to and including the design basis Loss-of-Coolant Accident (see Section 6.2). The drywell is a steel pressure vessel with a spherical lower portion, and a cylindrical upper portion.

The pressure absorption chamber is a steel pressure vessel in the shape of a torus located below and encircling the drywell, and is approximately half filled with water. The vent system from the drywell terminates below the water level in the torus, so that in the event of a pipe failure in the drywell, the released steam passes directly to the water where it is condensed. This transfer of energy to the water pool rapidly reduces the residual pressure in the drywell and substantially reduces the potential for subsequent leakage from the Primary Containment. Provisions are made for the removal of heat from within the Primary Containment to maintain integrity of the Containment System indefinitely following a Loss-of-Coolant Accident (see Section 6.2).

Isolation valves are provided on lines penetrating the drywell and the torus to provide integrity of the containment when required. These valves are actuated automatically by signals received from the Reactor Protection System. The valves of the auxiliary systems are left open or are closed depending upon the functional requirements of the system without reducing the integrity of the Primary Containment System.

Provisions are made for initial preoperational pressure and leak rate testing of the entire

Containment Systems and for continuous gross leakage testing after the plant has commenced operation. The high temperature piping penetrations and electrical penetrations of the Containment are capable of being leak tested individually during reactor shutdown.

The integrity of the Containment Systems and their associated engineered safeguards are designed so that offsite doses resulting from postulated accidents are well below the guideline values stated in 10CFR100.

In addition to the Turbine Generator and Main Condenser, four independent auxiliary cooling systems are provided for reactor and containment cooling under various normal and abnormal conditions.

A Shutdown Cooling System (Section 5.4) is provided which circulates water from the reactor through heat exchangers and back to the reactor. During shutdown operations, this system provides for the removal of reactor decay heat. This system is cooled by the Reactor Building Closed Cooling Water System (Section 9.2).

An Isolation Condenser System (Section 6.3), consisting of two loops operating at reactor pressure removes decay heat when the reactor is scrammed from power operation and is isolated from the Main Condenser. The shell side of each isolation condenser contains a minimum water volume and operates at atmospheric pressure.

Two separate and independent Core Spray System (Section 6.3) loops can inject water from the torus to the reactor. The water from these loops is distributed directly to the reactor core by spray headers mounted inside the plenum and above the core. Either of these loops provides cooling for the reactor in the event of a Loss-of-Coolant Accident due to a large coolant line break. An Automatic Depressurization System (Section 6.3) is provided to rapidly reduce reactor pressure to allow the Core Spray System to maintain continuity of core cooling in the event of a Loss-of-Coolant Accident due to a small line break.

The Containment Spray System (Section 6.2) provides water from the torus pool, which is pumped through heat exchangers, and discharged through spray nozzles into the drywell. The system also directs a small amount of water to the torus vapor space in the normal mode and can be used to cool the torus pool in the torus cooling mode. Water entering the drywell drains back to the torus to complete the cycle.

A "wet" refueling procedure, where all operations are carried out with the fuel under water, is used. Fuel handling equipment and procedures are addressed in Section 9.1.

Spent fuel discharged from the reactor is transferred under water through the spent fuel storage pool canal into racks provided in the storage pool. The storage pool is designed to accommodate the channel stripping operation and other fuel maintenance operations that are required. Storage space is also provided in the pool for the control rods, fuel shipping cask and small internal components of the reactor. A Cask Drop Protection System is provided within the pool. Cooling of the stored spent fuel is achieved by means of the Spent Fuel Pool Cooling and Augmented Spent Fuel Pool Cooling Systems (Section 9.1).

New fuel is brought through the equipment entrance of the Reactor Building and hoisted to the upper floor utilizing the Reactor Building Crane. The new fuel is stored in the new fuel dry vault located adjacent to the refueling pool area within the Reactor Building.

Reactor power is controlled by movement of control rods and by regulating the recirculation flow rate (Section 4.6). Control rods are used to bring the reactor through the full range of power and to shape the core power distribution. Changing recirculation flow rate provides a second method for controlling reactor power. Adjustments in reactor power level are accomplished with recirculation flow control. Procedural controls and protective devices are used to ensure that thermal performance does not exceed established limits.

A Standby Liquid Control System or Liquid Poison System (Section 9.3) is provided as an independent backup control mechanism to be used in the remote event that the reactor cannot be shutdown with the control rods.

A Reactor Protection System (Section 7.2) is provided which automatically initiates appropriate action whenever the plant conditions monitored by the system approach pre-established limits. The Reactor Protection System acts to shut down the reactor, close isolation valves, or initiate the operation of standby and safety systems as required.

Reactor power is monitored from the source range up through the power operating range by suitable neutron monitoring channels. All detectors for neutron monitoring are placed inside the reactor vessel. This location has been selected to provide maximum sensitivity to control rod movement during the startup period and to provide optimum monitoring in intermediate and power ranges.

The reactor has hydraulically driven control rods, each of which is controlled manually from the Control Room. Selection of the control rod to be manually controlled is accomplished by the use of a pushbutton array. Interlocking is provided so that only one control rod can be selected at a time for operation. A pilot light on the position indicator for the selected control rod is energized to indicate which is responsive to manual positioning.

Manual control is accomplished by use of a separate control switch which energizes valves in the hydraulic system to move the selected control rod to a new insert or withdrawal position. Normal control permits the rod to move one notch for each operation of the control switch. An override position is provided to permit continuous withdrawal movement of the rod when desired.

Instrumentation is provided for continuous monitoring of the radioactivity of certain processes. Critical processes, significantly high in radioactivity, are monitored for variation from normal. Also certain non-radioactive processes are monitored to provide alarms in the event of contamination.

Turbine Building

The Turbine Building (Drawings 3E-151-02-001 through 009) is a reinforced concrete structure directly to the west of the Reactor Building. The building is about 265 feet long and 171 feet wide. The foundation mat is 6 feet to 8 feet thick and the finished top is at El. 0'-0". The Turbine Building foundation mat overlaps the Reactor Building mat where the two building abut. Concrete walls extend from the basement levels to the operating floor at El. 46'-6" (about 23' above grade). Steel framework and insulated metal siding are used over the Turbine Generator area. Heavy shield walls with labyrinth entrances shield radioactive components within the building.

The Turbine Building houses the power conversion equipment and related auxiliary systems and equipment (Chapter 10). The Turbine Building contains the plant Control Room (Section 6.4), and electrical equipment (Section 8.3).

Three horizontal, single pass, divided water box, deaerating type condensers (Section 10.4) are provided, located with the tubes at right angles to the turbine shaft and supported rigidly on a foundation with an expansion joint between the turbine exhaust and the condenser steam inlet. Four vertical circulating water pumps deliver water to the condensers.

Steam jet air ejectors are provided to deaerate the condensers. A vacuum pump is provided to produce a vacuum in the condensers and turbine prior to starting the turbine when steam is not available to operate the steam jet air ejector.

Air removed from the condensers by the air ejectors is discharged to the stack through the Offgas System. The Augmented Offgas System assures conformance to 10CFR50, Appendix I (Section 11.3).

The turbine shaft Gland Seal System includes a steam seal regulator, two exhaust blowers and condenser. This system discharges noncondensable gases from the Gland Seal System to the stack through a piping system which provides holdup time for decay of radioactive gases.

Condensate pumps take suction from the condenser hotwells and discharge through various components to the feedwater pumps and hence to the reactor vessel (Section 10.4).

A Turbine Bypass System (Section 10.4) located in the Turbine Building restricts overpressure transients resulting from sudden turbine control valve or stop valve closure. The system is also used during cooldowns to remove residual decay heat from the reactor. Rapid partial load rejection (up to forty percent of reactor power) can be accommodated with the bypass system.

Office Building

The Office Building (Drawings 3E-156-02-001 and 002) is a three story concrete structure between the Turbine Building and the Reactor Building. (Subsection 3.8.4.1) The building houses offices, a laboratory area, showers, locker rooms and provides a secondary access to controlled areas. The recirculation pump motor generator sets, a switchgear room and one battery room are also contained in this building (Section 8.3).

Old Radwaste Building

The Old Radwaste Building (Drawings 3E-154-02-001 through 003) is directly east of the Reactor Building. It is single story reinforced concrete building with a two story penthouse. (See Subsection 3.8.4.1) A small basement area is at El. 6'-6". The roof elevations are 39'-3" and 49'-10" to 50'-6". The building is no longer used for normal radwaste handling activities, however some of the equipment housed in the building is used during plant operations for waste compaction and waste transfer.

New Radwaste and Offgas Buildings

The New Radwaste Building (Drawings 3E-155-02-001 through 005) is a three story building 44' high, 86' by 114' in plan dimensions, erected at grade approximately 250 feet north-northwest of the ventilation stack. The building is of structural steel framework with poured reinforced concrete foundation, intermediate slabs and roof slab. Shield walls have been provided constructed by solid concrete blocks, other walls are insulated metal siding.

The building houses the Liquid and Solid Radwaste Systems (Sections 11.2 and 11.4), which are designed to process low level radioactive liquid wastes produced as a byproduct of plant operations. The systems feature a number of segregated waste streams and a process of solidification of wastes. The solidified end product is encased in a shipping container and removed from the building for permanent disposal. The purified and decontaminated liquids resulting from the process are either recycled or discharged to the environs.

The Offgas Building (Drawings 3E-175-02-001 and 002) is a two story building erected at grade and approximately 240' east of the stack. The building is of structural steel framework with poured concrete foundation, intermediate slab and roof slab. The portion of the building walls which also serve as shield walls are constructed of solid concrete blocks.

The building houses the Augmented Offgas System (Section 11.3), which was designed to reduce radioactive gaseous waste emissions to levels in compliance with 10CFR50, Appendix I. A reduction of condenser offgas emissions from 260,000 microcuries per second after 30 minutes delay to less than 1,700 microcuries per second is accomplished by the system. Radiolytic hydrogen and oxygen are combined and condensed. The offgas is dried and stripped of iodine isotopes; xenon and Krypton isotopes are delayed in the system prior to discharge.

Emergency Diesel Generator Building

The plant Emergency Diesel Generators and their associated fuel oil day tanks are housed within separate vaults in a reinforced concrete building (Drawing 3E-157-02-001) southwest of the Turbine Building. The one story structure is at approximately grade elevation near the eastern bank of the discharge canal.

Intake and Discharge Structures

Circulating water is drawn from Barnegat Bay through a 140 foot wide canal dredged to a depth of (-)10'-0". The canal follows the general course of the south branch of Forked River; then curves south to the concrete intake structure (Drawing 3E-168-02-001), located 183 feet west of the Reactor Building. The circulating water pumps, service water pumps, emergency service water pumps, new radwaste service pumps, screen wash pumps and traveling water screens are located outdoors at the structure. Trash racks are cleared by a trash rake guided on a monorail.

The circulating water pumps discharge into the intake tunnel that runs below grade to the Turbine Building basement. The discharge tunnel runs underneath the intake tunnel; then curves southwest to the discharge canal. The 100 foot wide discharge canal empties into Barnegat Bay following the general course of Oyster Creek.

Ventilation Stack

The 394 foot reinforced concrete stack (368 feet above grade) is linked by tunnels to the Reactor Building, Turbine Building and Old Radwaste Building.

The top of the stack foundation mat is at El. (-)3'-0". Floors at El. 23'-6" and El. 35'-0" are penetrated by the offgas piping, which enters the stack at El. 0'-0". Exhaust fans for the ventilating ducts are located outdoors at grade level, and discharge to the stack above the second floor level.

Storage Tanks

Miscellaneous Storage Tanks are located throughout the facility. The three largest of these tanks are the Fire Water Tank, the Torus Water Storage Tank and the Condensate Storage Tank. The locations of these and other yard tanks are shown in Drawing JC-19702.

TABLE 1.2-1
(Sheet 1 of 2)

PRINCIPAL DESIGN FEATURES

Site	
Location	Oyster Creek, New Jersey
Size of Site	Approximately 800 acres
Net Electrical Output	640 MW (approximately)
Reactor	
Thermal Output	1930 MW
Reactor Pressure (core exit)	1020 psig
Steam Flow Rate	7.254×10^6 lb/hr
Core	
Circumscribed Core Diameter	170.55 inches
Fuel Assembly	
Number of Fuel Assemblies	560
Fuel Rod Array	8 x 8
Cladding Material	Zircaloy-2
Fuel Material	UO_2
Manufacturer	General Electric Co.
Control System	
Number of Movable Control Rods	137
Shape of Movable Control Rods	Cruciform
Pitch of Movable Control Rods	12.0 inch
Control Material in Movable Control Rods	Compacted Boron Carbide in tubes, sheathed
Type of Control Drives	Bottom entry, hydraulic actuated
Reactor Vessel	
Inside Diameter	17 ft 9 in
Overall Length (inside)	63 ft 10 in
Design Pressure	1250 psig

TABLE 1.2-1
(Sheet 2 of 2)

PRINCIPAL DESIGN FEATURES

Coolant Recirculation Loops

Location of Recirculation Loops	Inside containment drywell
Number of Recirculation Loops	5
Pipe Size	26 inch
Flow capacity per loop at design conditions	32,000 gpm
Pump type	Vertical, single stage, centrifugal
Pump motors	Variable frequency (11.5-57.5 cps), 2400V, open drip-proof, three phase squirrel cage induction type.

Primary Containment

Type	Pressure Absorption, GE Mark I
Design Pressure of Drywell Vessel	44 psig
Design Pressure of Absorption Chamber Vessel	35 psig
Leakage Rate, Maximum	0.5% free volume per day at 35 psig

Secondary Containment

Type	Reinforced concrete and steel superstructure with metal siding.
Internal Design Pressure	0.20 psig
Inleakage Rate	100% free volume per day at 0.25 in. water negative pressure

1.3 COMPARISON TABLES

1.3.1 Comparisons with Similar Facility Designs

Certain original design features of the Oyster Creek Nuclear Generating Station (OCNGS), particularly in the areas of reactor, pressure vessel and Primary Containment design, are similar to those for other Boiling Water Reactors designed in approximately the same time frame as the OCNGS. However, because changes and modifications have taken place at the OCNGS and those facilities in the course of time since their commercial operation began, a detailed comparison would not be meaningful.

1.3.2 Comparison of Final and Preliminary Information

Not applicable

1.4 IDENTIFICATION OF AGENTS AND CONTRACTORS

The Oyster Creek Nuclear Generating Station (OCNGS) was designed and constructed by the General Electric Company Atomic Power Equipment Department as a turnkey project. Burns and Roe Inc. is the Architect-Engineer of record.

The plant is owned and operated by AmerGen Energy Company, LLC, who is the licensee for the facility.

The suppliers of major components and equipment were as follows:

Item	Supplier
Reactor Vessel	Combustion Engineering Co.
Reactor Vessel Internals	P. F. Avery Co.
Steel Containment	Chicago Bridge and Iron Co.
Isolation Condensers	Foster Wheeler Corp.
Reactor & Turbine Buildings Structural Steel	American Bridge Co.
Reactor Recirculation Pumps	Byron Jackson
Reactor Recirculation System - Piping & Supports	Grinnell
Reactor Recirculation System – Valves	Chapman Div. Crane Co.
Main Condenser	Worthington

1.5 REQUIREMENTS FOR FURTHER TECHNICAL INFORMATION

For an operating facility, requirements for further technical information are continuously being formulated by NRC at both the plant specific and generic levels. The results of studies and analyses are reflected by plant modifications, changes in operating procedures and/or changes to the Technical Specifications. These are documented in special or periodic submittals to the NRC and/or in required updates of the FSAR.

1.6 MATERIAL INCORPORATED BY REFERENCE

Any material incorporated by reference is cited within specific sections of this updated FSAR. With the issuance of the Full-Term Operating License, the Updated Final Safety Analysis Report and the Environmental Report, as supplemented and amended, constitute the licensing basis for the Oyster Creek Nuclear Generating Station. The plant is operated in accordance with the provisions of the License and the Technical Specifications.

1.7 DRAWINGS AND OTHER DETAILED INFORMATION

The Oyster Creek drawings which are listed in the Electronic Data Management System are revised according to a priority system. Those drawings which are referenced in the FSAR annual update are identified by their drawing numbers. Since the current drawing revision is not listed, an Electronic Data Management System should be reviewed for the latest revision and for any change notices.

1.8 CONFORMANCE TO NRC REGULATORY GUIDES

The Oyster Creek Nuclear Generating Station (OCNGS) was placed in commercial operation on December 23, 1969; prior to the promulgation of regulatory guides. At a later date, the design of the OCNGS (as of December 1971) was evaluated against the requirements of the Atomic Energy Commission's Safety Guides Nos. 1 through 15. Subsequently, the design of the facility as of March 1972 was evaluated against the requirements of various regulatory guides promulgated up to 1973. The detailed results of this latest evaluation are contained in the licensee's submittals for a full term license application, Facility Description and Safety Analysis Report (FDSAR) Amendment No. 68, its Supplements and Addenda.

Plant modifications accomplished after 1973 have been generally designed in conformance with applicable regulatory guides in effect at the time that the design of the modification was initiated. These regulatory guides, as they apply to the OCNGS design, have been addressed throughout the text of the updated FSAR. The current commitment to NRC guidance documents is provided primarily in the OCNGS Operating License, the Technical Specifications, Quality Assurance Plans, and other licensing basis documents.

The Systematic Evaluation Program (SEP) provided (1) an assessment of the significance of difference between current regulatory positions (i.e., Regulatory Guides) on safety issues and those that existed when a particular plant was licensed, (2) a basis for deciding how these differences should be resolved in an integrated plant review, and (3) a documented evaluation of plant safety. For Oyster Creek the Staff documented their reviews in NUREG 0822.

The results of the SEP evaluations provided a major portion of the technical input for the Staff Safety Evaluation Report (SER) prepared for the Full-Term Operating License which was issued on July 2, 1991. The Staff published this SER as NUREG-1382.

Table 1.8-1 presents a cross reference from the current NRC design criteria (Regulatory Guides), the NRC Staff review (SEP Topic) conducted of the Oyster Creek design, and the NRC Staff safety evaluation (NUREG 1382). The Safety Evaluation documents in detail the Staff's resolution of the differences between the Oyster Creek design and the current NRC regulatory positions.

Table 1.8-1
(Sheet 1 of 4)

NRC REGULATORY GUIDES ASSESSED IN THE SYSTEMATIC EVALUATION PROGRAM FOR OYSTER CREEK

Regulatory Guide		SEP Topics		General Design Criteria	FTOL SER Section No.* (NUREG 1382
1.6	Independence Between Redundant Standby (on-site Power Sources & Between Their Distribution Systems	VI-7.C.1	Appendix K - Electrical Instrumentation and Control Re-Reviews	2,4,17,18,5, 19	8.5
		VIII-3.B	DC Power System Bus Voltage Monitoring and Annunciation		8.3.3.2
1.9	Selection of Diesel Generator Set Capacity for Standby Power Supplies	VIII-2	Onsite Emergency Power Systems (Diesel Generators)	17	8.3.2
1.11	Instrument Lines Penetrating Primary Reactor Containment	VI-4	Containment Isolation System	54,55,56,57	6.2.2
1.22	Periodic Testing of Protection System Actuation Functions	VI-7.A.3	Emergency Core Cooling System Actuation System	37	6.3.1
1.26	Quality Group Classification and Standards for Water, Steam and Radioactive Waste Containing Components of Nuclear Power Plants	III-1	Classifications of Structures, Components and Systems (Seismic and Quality)	1	3.2
1.27	Ultimate Heat Sink	II-3.B	Flooding Potential and Protection Requirements	2	3.4.1
		II-3.B.1	Capability of Operating Plant to Cope with Design-Basis Flooding Conditions		3.4.1
		II-3.C	Safety-Related Water Supply (Ultimate Heat Sink (UHS)		3.4.1
1.29	Seismic Design Classification	III-6	Seismic Design Considerations	2	

*See Table 1.3 of NUREG 1382 for detailed listing of SEP issues.

Table 1.8-1
(Sheet 2 of 4)

NRC REGULATORY GUIDES ASSESSED IN THE
SYSTEMATIC EVALUATION PROGRAM FOR OYSTER CREEK

Regulatory Guide		SEP Topics		General Design Criteria (See Section 3.1)	FTOL SER Section No.* (NUREG 1382)
1.32	Use of IEEE Standard 308-1971, "Criteria for Class IE Electrical System for Nuclear Power Generating Stations	VIII-3.B	DC Power System Bus Voltage Monitoring and Annunciation	2,4,5,17,18,19, 50	8.3.3.2
		VIII-4	Electrical Penetrations of Reactor Containment		8.4
1.45	Reactor Coolant Pressure Boundary Leakage Detection Systems	III-5.A	Effects of Pipe Break on Structures, Systems and Components Inside Containment	4,30	3.6.1
		V-5	Reactor Coolant Pressure Boundary (RCPB) Leakage Detection		5.2.1
1.47	Bypassed and Inoperable Status Indication for Nuclear Power Plant Safety Systems	VIII-3.B	DC Power System Bus Voltage Monitoring and Annunciation	2,4,5,17,18,19	8.3.3.2
1.53	Application of the Single-Failure Criterion to Nuclear Power Plant Protection Systems	VII-3	Systems Required for Safe Shutdown	2,34,13	7.3
1.56	Maintenance of Water Purity in Boiling Water Reactors	V-12.A	Water Purity of BWR Primary Coolant	14	5.5
1.59	Design Basis Floods for Nuclear Power Plants	11-3.B	Flooding Potential and Protection Requirements	2	3.4.1
		II-3.B.1	Capability of Operating Plant to Cope with Design-Basis Flooding Conditions		3.4.1
		II-3.C	Safety-Related Water Supply (Ultimate Heat Sink (UHS)		3.4.1

*See Table 1.3 of NUREG 1382 for detailed listing of SEP issues.

Table 1.8-1
(Sheet 3 of 4)

NRC REGULATORY GUIDES ASSESSED IN THE SYSTEMATIC EVALUATION PROGRAM FOR OYSTER CREEK

Regulatory Guide		SEP Topics		General Design Criteria	FTOL SER Section No.* (NUREG 1382)
1.59	Design Basis Floods for Nuclear Power Plants (Cont'd.)	III-3.A	Effects of High Water Level on Structures		
1.63	Electric Penetration Assemblies in Containment Structures for Water-Cooled Nuclear Power Plants	VIII-4	Electrical Penetrations of Reactor Containment	2,4,5,17,18,50	8.4
1.75	Physical Independence of Electric Systems	VIII-3.B	DC Power System Bus Voltage Monitoring and Annunciation	2,4,5,17,18,19	8.3.3.2
1.76	Design Basis Tornado for Nuclear Power Plants	III-2	Wind and Tornado Loadings	2	3.3
1.91	Evaluation of Explosions Postulated to Occur on Transportation Routes Near Nuclear Power Plants	III-4.D	Site-Proximity Missiles (Including Aircraft)	4	2.2
1.99	Effects of Residual Elements on Predicated Radiation Damage to Reactor Vessel Materials	V-6	Reactor Vessel Integrity	-	5.3
1.115	Protection Against Low Trajectory Missiles	III-4.B	Turbine Missiles	4	3.5.1.4
1.117	Tornado Design Classification	III-2	Wind and Tornado Loadings	2	3.3
		III-4.A	Tornado Missiles		3.5.1
1.118	Periodic Testing of Electric Power and Protection Systems	VIII-3.B	DC Power System Bus Voltage Monitoring and Annunciation	2,4,5,17,18,19,50	8.3.3.2
		VIII-4	Electrical Penetrations of Reactor Containment		8.4

*See Table 1.3 of NUREG 1382 for detailed listing of SEP issues.

Table 1.8-1
(Sheet 4 of 4)

NRC REGULATORY GUIDES ASSESSED IN THE SYSTEMATIC EVALUATION PROGRAM FOR OYSTER CREEK

Regulatory Guide		SEP Topics		General Design Criteria	FTOL SER Section No.* (NUREG 1382)
1.127	Inspection of Water-Control Structures Associated with Nuclear Power Plants	III-3.C	Inservice Inspection of Water Control Structures	2,44,45	3.4.3
1.129	Maintenance, Testing, and Replacement of Large Lead Storage Batteries for Nuclear Power Plants	VII-3	Systems Required for Safe Shutdown	2,13,34	7.3
1.133	Loose-Part Detection Program for the Primary System of Light-Water-Cooled Reactors	III-8.A	Loose-Parts Monitoring and Core Barrel Vibration Monitoring	13	4.3
1.139	Guidance for Residual Heat Removal	V-10.B	Residual Heat Removal System Reliability	19,34	5.4.2
1.141	Containment Isolation Provisions for Fluid Systems	VI-4	Containment Isolation System	54,55,56,57	6.2.2

*See Table 1.3 of NUREG 1382 for detailed listing of SEP issues.

1.9 NUREG-0737 SUMMARY

NUREG-0737 is in the form of a letter from NRC to licensees of operating power reactors and applicants for operating licenses forwarding post TMI requirements approved for implementation. The total set of TMI related actions have been collected in NUREG-0660, but only those items that the Commission had approved for implementation as of October 31, 1980 were included in NUREG-0737.

This section provides a status of the actions taken at the Oyster Creek Nuclear Generating Station (OCNGS) in response to the applicable post TMI requirements included in NUREG-0737.

1.9.1 Item I.A.1.1 - Shift Technical Advisor

Position

Each licensee shall provide an on-shift technical advisor to the shift supervisor. The shift technical advisor (STA) may serve more than one unit at a multiunit site if qualified to perform the advisor function for the various units.

The STA shall have a bachelor's degree or equivalent in a scientific or engineering discipline and have received specific training in the response and analysis of the plant for transients and accidents. The STA shall also receive training in plant design and layout, including the capabilities of instrumentation and controls in the Control Room. The licensee shall assign normal duties to the STAs that pertain to the engineering aspects of assuring safe operations of the plant, including the review and evaluation of operating experience.

Response

GPU Nuclear Corporation has instituted the position of Shift Technical Advisor (STA) at the Oyster Creek Nuclear Generating Station to serve as advisor to the shift supervisor and for accident assessment. The NRC has reviewed the STA training program and found that this program meets the intent of applicable guidelines. The position of Shift Technical Advisor is discussed in Section 6.2 of the OCNGS Technical Specifications.

1.9.2 Item I.A.1.2 - Shift Supervisor Administrative Duties

Position

The objective is to increase the shift supervisor's attention to his command function by minimizing ancillary responsibilities. NRR has required that all operating plant licensees review the administrative duties of the shift supervisor. The review should be performed by the senior officer, at each utility, who is responsible for plant operations. Administrative functions that detract from, or are subordinate to, the management responsibility for assuring the safe operation of the plant are to be delegated to other operations personnel not on duty in the Control Room. The same requirement will be imposed by the licensing review staff on all operating license applicants.

Response

Shift Supervisor responsibilities during normal operations and emergency conditions are specified in plant procedures. The NRC has concluded that the requirements of this item have been met. Verification of the adequacy of the procedures is performed by the Office of Inspection and Enforcement.

1.9.3 Item I.A.1.3 - Shift Manning

Position

This position defines shift manning requirements for normal operation. The letter of July 31, 1980 from D. G. Eisenhut to all power reactor licensees and applicants sets forth the interim criteria for shift staffing (to be effective pending general criteria that will be the subject of future rulemaking). Overtime restrictions were also included in the July 31, 1980 letter.

Response

Administrative procedures have been formulated to ensure that an alert and attentive shift compliment is provided. The procedures state operating personnel may not normally stand for more than 16 continuous hours of shift assignment unless special circumstances dictate it and authorization is given by the Plant Operations Director. The policy on overtime for licensed operators allows for a minimum of eight hours between work periods for an operator who has worked 16 hours on shift. Requirements for operations personnel, in conformance with the guidelines of Generic Letter 82-12, have been incorporated in station procedures. The shift staffing requirements are provided in the OCNGS Technical Specifications.

The NRC has concluded that the program for limiting overtime at the OCNGS is acceptable.

1.9.4 Item I.A.2.1 - Immediate Upgrading of Reactor Operator and Senior Reactor Operator Training and Qualifications

Position

Effective December 1, 1980, an applicant for a senior reactor operator (SRO) license will be required to have been a licensed operator for one year.

Response

The initial and requalification training programs for the OCNGS have been modified to include training in areas required by this item. The revised training programs have been found acceptable by NRC.

1.9.5 Item I.A.2.3 - Administration of Training Programs

Position

Pending accreditation of training institutions, licensees and applicants for operating licenses will assure that training center and facility instructors who teach systems, integrated responses, transient, and simulator courses demonstrate senior reactor operator (SRO) qualifications and be enrolled in appropriate requalification programs.

Response

The requirements for this item for operating reactors were completed at the time of issuance of NUREG-0737.

1.9.6 Item I.A.3.1 - Revise Scope and Criteria for Licensing Examination

Position

Simulator examinations will be included as part of the licensing examinations.

Response

This item was implemented in accordance with the NUREG-0737 schedule and Oyster Creek operator candidates are administered their simulator exams as part of the Operator Licensing Examination process.

1.9.7 Item I.C.1 - Guidance for the Evaluation and Development of Procedures for Transients and Accidents (EOPs)

Position

In letters of September 13 and 27, October 10 and 30, and November 9, 1979, the Office of Nuclear Reactor Regulation required licensees of operating plants, applicants for operating licenses and licensees of plants under construction to perform analyses of transients and accidents, prepare emergency procedure guidelines, upgrade emergency procedures, including procedures for operating with natural circulation conditions, and to conduct operator retraining (see also Item I.A.2.1). Emergency procedures are required to be consistent with the actions necessary to cope with the transients and accidents analyzed. Analyses of transients and accidents were to be completed in early 1980 and implementation of procedures and retraining were to be completed three months after emergency procedures guidelines were established; however, some difficulty in completing these requirements has been experienced. Clarifications of the scope of the task and appropriate schedule revisions are being developed.

Response

The EOPs at the OCNGS were developed in two phases from a set of Plant Specific Technical Guidelines which are based on the BWR Generic Guidelines. These Generic guidelines were submitted to the Commission which approved them in February 1983.

The Phase 1 EOPs were implemented in September 1984. The Phase 1 effort did not include instructions to mitigate Anticipated Transients Without Scram (ATWS) events, nor did it include other instructions which could not be accomplished without plant modifications and additional instrumentation.

Phase 2 was implemented prior to the restart from the Cycle 11 Refueling Outage and brought the EOPs into compliance with Revision 4 of the BWR Generic Guidelines. EOP training is provided to licenced operators on an ongoing basis.

1.9.8 Item I.C.2 - Shift and Relief Turnover Procedure

Position

Shift and relief turnover is required to ensure that each oncoming shift is aware of critical plant status information and system availability prior to assuming duty. To assure that these functions are adequately prescribed, NRR issued requirements in letters dated September 13 and 27, October 10 and 30, and November 9, 1979, to licensees and applicants to review and revise, as necessary, shift and relief turnover procedures.

Response

Shift turnover is strictly controlled by procedure. This procedure serves to verify that critical plant parameters are within safe limits and ensure that the availability and alignments of safety systems are made known to the incoming shift.

1.9.9 Item I.C.3 - Shift Supervisor Responsibilities

Position

In letters of September 13 and 27, October 10 and 30, and November 9 1979, NRC required licensees and applicants to review and revise, as necessary, plant procedures and directives to assure that the duties, responsibilities, and authority were properly defined to establish a definite line of command and clear delineation of the command decision authority of the supervisor in the Control Room relative to other plant management personnel. These letters also emphasized the primary management responsibility of the shift supervisor for safe operation of the plant. Training programs for supervisors were required to emphasize and reinforce the responsibility for safe operation and management function of the shift supervisor to assure safe operation of the plant.

Response

OCNGS Shift Supervisor responsibilities during normal operations and emergency conditions are specified in procedures which were updated in response to NRC requirements. A corporate directive sets forth the duties and responsibilities of the Shift Supervisor, including the primary responsibility for plant safety and the command authority to direct licensed activities. The GSS is directed to remain in the Control Room during accident situations.

1.9.10 Item I.C.4 - Control Room Access

Position

Letters dated September 13 and 27, October 10 and 30, and November 9, 1979, were sent all licensees and applicants requiring that the authority and responsibilities of the person in charge of Control Room access and clear lines of authority and responsibility in the Control Room in the event of an emergency be established in conformance to Item 2.2.2.a of NUREG-0578.

Response

Procedures have been issued to limit access to the Control Room during an emergency. The GSS has the authority to clear the Control Room of all unnecessary personnel, as necessary to perform licensed duties.

1.9.11 Item I.C.5 - Procedures for Feedback of Operating Experience to Plant Staff

Position

In accordance with Task Action Plan I.C.5, Procedures for Feedback of Operating Experience to Plant Staff (NUREG-0660), each applicant for an operating license shall prepare procedures to assure that operating information pertinent to plant safety originating both within and outside the utility organization is continually supplied to operators and other personnel and is incorporated into training and retraining programs. These procedures shall:

(1) Clearly identify organizational responsibilities for review of operating experience, the feedback of pertinent information to operators and other personnel, and the incorporation of such information into training and retraining programs;

(2) Identify the administrative and technical review steps necessary in translating recommendations by the operating experience assessment group into plant actions (e.g., changes to procedures; operating orders);

(3) Identify the recipients of various categories of information from operating experience (i.e., supervisory personnel, shift technical advisors, operators, maintenance personnel, health physics technicians) or otherwise provide means through which such information can be readily related to the job functions of the recipients;

(4) Provide means to assure that affected personnel become aware of and understand information of sufficient importance that should not wait for emphasis through routine training and retraining programs;

(5) Assure that plant personnel do not routinely receive extraneous and unimportant information on operating experience in such volume that it would obscure priority information or otherwise detract from overall job performance and proficiency;

(6) Provide suitable checks to assure that conflicting or contradictory information is not conveyed to operators and other personnel until resolution is reached; and,

(7) Provide periodic internal audits to assure that the feedback program functions effectively at all levels.

Response

The review and assessment of information previously reviewed by Operating Experience Assessment Implementation (OEAI) is now done by several departments including: Training, Engineering, and Regulatory Affairs.

1.9.12 Item I.C.6 - Guidance on Procedures for Verifying Correct Performance of Operating Activities

Position

It is required (from NUREG-0660) that licensees' procedures be reviewed and revised, as necessary, to assure that an effective system of verifying the correct performance of operating activities is provided as a means of reducing human errors and improving the quality of normal operations. This will reduce the frequency of occurrence of situations that could result in or contribute to accidents. Such a verification system may include automatic system status monitoring, human verification of operations and maintenance activities independent of the people performing the activity (see NUREG-0585, Recommendation 5), or both.

Implementation of automatic status monitoring, if required, will reduce the extent of human verification of operations and maintenance activities but will not eliminate the need for such verification in all instances. The procedures adopted by the licensees may consist of two phases - one before and one after installation of automatic status monitoring equipment, if required, in accordance with Item I.D.3.

Response

The requirements for verification of correct performance of operating activities have been specified in the plants' administrative procedures. Verification of system alignment is required during installation and removal of safety tags and electrical or mechanical jumpers, and during lifting and reinstallation of electrical leads. Instrumentation valve lineups are verified following surveillance tests or maintenance evolutions requiring valve manipulation. Plant management will determine when to use Concurrent Verification or Independent Verification.

NRC has found this action acceptable to satisfy the requirements of this item.

1.9.13 Item I.D.1 - Control Room Design Reviews

Position

In accordance with Task Action Plan I.D.1, Control Room Design Reviews (NUREG-0660), all licensees and applicants for operating licenses will be required to conduct a detailed Control Room design review to identify and correct design deficiencies. This detailed Control Room design review is expected to take about a year. Therefore, the Office of Nuclear Reactor Regulation (NRR) requires that those applicants for operating licenses who are unable to complete this review prior to issuance of a license make preliminary assessments of their Control Rooms to identify significant human factors and instrumentation problems and establish a schedule approved by NRC for correcting deficiencies. These applicants will be required to complete the more detailed Control Room reviews on the same schedule as licensees with operating plants.

Response

In December 1980 a detailed Control Room Design Review (CRDR) was initiated for the OCNGS Control Room. Human factors design reviews were conducted, and the CRDR was completed in September 1982. The program plan was submitted on July 1, 1983 and the

"Summary Report on the Oyster Creek Control Room Design Review" was submitted on April 30, 1984.

On January 17 and 18, 1990, the NRC staff conducted an onsite audit of Oyster Creek's DCRDR and SPDS. By SER, dated June 28, 1990, the staff has concluded that GPUN meets all of the nine NUREG-0737 Supplement 1 DCRDR requirements and the issue is considered resolved.

1.9.14 Item I.D.2 - Plant Safety Parameter Display Console

Position

In accordance with Task Action Plan I.D.2, Plant Safety Parameter Display Console (NUREG-0660), each applicant and licensee shall install a safety parameter display system (SPDS) that will display to operating personnel a minimum set of parameters which define the safety status of the plant. This can be attained through continuous indication of direct and derived variables as necessary to assess plant safety status.

Response

A Critical Safety Functions (CSF) approach was used as the basis for a Safety Parameter Display System and five CSFs were chosen that correlate with the basic objectives of Emergency Operating Procedures (EOPs). A parameter set was then selected largely based on EOPs parameters and the Emergency Plan radiation monitoring requirements.

Groups of parameters were then assigned to the different CSFs based on plant operating and emergency procedures logic.

These critical Safety Functions are activated to warning or alarm levels by the various parameters at setpoints that are consistent with those used by the different procedures. The CSFs and the parameter set chosen form a complete set that should respond to plant conditions during accident conditions, power operation and shutdown modes.

The SPDS at Oyster Creek has been installed. The NRC evaluated the system and issued a safety evaluation (dated June 28, 1990) stating that the Oyster Creek SPDS met six of the eight NUREG-0737 Supplement 1 requirements. Based on the results of an NRC staff audit of January 17 and 18, 1990, licensee corrective actions to address the deficiencies of the audit, NRC safety evaluation dated June 28, 1990, and inspector verification of the corrective actions of GPUN letter dated May 17, 1990, TMI items I.D.2.2. and I.D.2.3 for the installation and implementation of SPDS systems have been closed.

1.9.15 Item II.B.1 - Reactor Coolant System Vents

Position

Each applicant and licensee shall install reactor coolant system (RCS) and reactor vessel head high point vents remotely operated from the Control Room. Although the purpose of the system is to vent noncondensable gases from the RCS which may inhibit core cooling during natural circulation, the vents must not lead to an unacceptable increase in the probability of a Loss-of-Coolant Accident (LOCA) or a challenge to containment integrity. Since these vents form a part of the reactor coolant pressure boundary, the design of the events shall conform to the requirements of Appendix A to 10 CFR Part 50, "General Design Criteria." The vent system shall be designed with sufficient redundancy that assures a low probability of inadvertent or irreversible actuation.

Each licensee shall provide the following information concerning the design and operation of the high point vent system:

(1) Submit a description of the design, location, size, and power supply for the vent system along with results of analyses for Loss-of-Coolant Accidents initiated by a break in the vent pipe. The results of the analyses should demonstrate compliance with the acceptance criteria of 10CFR50.44.

(2) Submit procedures and supporting analysis for operator use of the vents that also include the information available to the operator for initiating or terminating vent usage.

<u>Response</u> (Ref. 1)

The credible mechanisms for the generation of noncondensible gases post LOCA are metal-water reaction in the core or the release of fission gases by cladding failure. These two mechanisms are not significant if the core remains effectively cooled.

For Oyster Creek, the Automatic Depressurization System (ADS) and the Core Spray System (CSS) are available to mitigate a LOCA. The operation of the Isolation Condenser System (ICS) will be significant only when the following conditions are present:

(1) Reactor pressure is sustained high,

(2) Fast depressurization mechanisms (such as intermediate or large size breaks or by the ADS valve are not present).

Analyses of a LOCA events with a successful reactor scram show that the reactor will be depressurized and effective core cooling will be established without significant release of noncondensibles. Also, since the ADS is available, the reactor pressure would remain near the containment pressure value. Therefore, the need to vent the ICS to the torus would be immaterial to the mitigation of the LOCA.

For beyond design basis accidents, the venting of the ICS would only impact the mitigation of below-the-core breaks of 0.005 - 0.8 square feet in which the ADS function is not available by automatic or manual initiation. The core damage frequency due to such a LOCA has been conservatively estimated as 1.8×10^{-5} per reactor year without credit for manual ADS or 1.8×10^{-6} per reactor year with credit for manual ADS actuation. The benefit gained by adding ICS vents to the torus is insufficient to warrant the modification.

Oyster Creek has the capability to vent the ICS to the main steam header downstream of the MSIV's. This is done to prevent the accumulation of noncondensible gases during startup and normal plant operation. During a LOCA, this vent path is isolated; however, these vents can be opened remotely by placing jumpers in the Control Room panels.

1.9.16 Item II.B.2 - Design Review of Plant Shielding and Environmental Qualification for Spaces/Systems Which May be Used in Postaccident Operations

Position

With the assumption of a postaccident release of radioactivity equivalent to that described in Regulatory Guides 1.3 and 1.4 (i.e., the equivalent of 50% of the core radioiodine, 100% of the core noble gas inventory, and 1% of the core solids are contained in the primary coolant), each licensee shall perform a radiation and shielding design review of the spaces around systems that may, as a result of an accident, contain highly radioactive materials. The design review should identify the location of vital areas and equipment, such as the Control Room, radwaste control stations, emergency power supplies, motor control centers, and instrument areas, in which personnel occupancy may be unduly limited or safety equipment may be unduly degraded by the radiation fields during postaccident operations of these systems.

Each licensee shall provide for adequate access to vital areas and protection of safety equipment by design changes, increased permanent or temporary shielding, or postaccident procedural controls. The design review shall determine which types of corrective actions are needed for vital areas throughout the facility.

Response

A radiation and shielding design study has been performed for the spaces around those plant systems outside containment that would or could contain highly radioactive fluids during a serious transient or accident (see Item III.D.1.1).

All personnel/equipment hatches into the Reactor Building were evaluated for their impact on postaccident operations. The personnel airlock on El. 51'-3" of the Reactor Building was the only hatch that appeared unsatisfactory. The airlock provides a line of sight from the core spray booster pumps and piping through the Office Building to a Control Room entrance. The shielding along this line of sight may not have maintained the dose rate low enough for continuous occupancy. The corrective action was to provide supplemental shielding in order to effectively reduce the dose rate in the affected portion of the Control Room to continuous occupancy levels.

The licensee requested cancellation of the requirement for changeout filters for the Standby Gas Treatment System (SGTS). Under Topic XV-19 of the Systematic Evaluation Program (SEP), the radiological consequences of Loss-of-Coolant Accidents (LOCAs) were evaluated by the NRC staff. The results of those analyses demonstrated that the major contributor to offsite doses was Main Steam Isolation Valve Leakage, and that a single SGTS filter train was capable of handling the entire accident. Hence, the SEP evaluation of this topic justifies this cancellation.

NRC has accepted this request and considers the requirements for this item to be complete.

1.9.17 Item II.B.3 - Postaccident Sampling Capability

Position

A design and operational review of the reactor coolant and containment atmosphere sampling line systems shall be performed to determine the capability of personnel to promptly obtain a sample under accident conditions without incurring a radiation exposure to any individual in excess of 5 and 50 rem to the whole body or extremities, respectively. Accident conditions should assume a Regulatory Guide 1.3 or 1.4 release of fission products. If the review indicates that personnel could not promptly and safely obtain the samples, additional design features or shielding should be provided to meet the criteria.

A design and operational review of the radiological spectrum analysis facilities shall be performed to determine the capability to promptly quantify certain radionuclides that are indicators of the degree of core damage. Such radionuclides are noble gases (which indicate cladding failure), iodines and cesiums (which indicate high fuel temperatures), and nonvolatile isotopes (which indicate fuel melting). The initial reactor coolant spectrum should correspond to a Regulatory Guide 1.3 or 1.4 release. The review should also consider the effects of direct radiation from piping and components in the auxiliary building and possible contamination and direct radiation from airborne effluents. If the review indicates that the analyses required cannot be performed in a prompt manner with existing equipment, then design modifications or equipment procurement shall be undertaken to meet the criteria.

In addition to the radiological analyses, certain chemical analyses are necessary for monitoring reactor conditions. Procedures shall be provided to perform boron and chloride chemical analyses assuming a highly radioactive initial sample (Regulatory Guide 1.3 or 1.4 source term). Both analyses shall be capable of being completed promptly (i.e., the boron sample analysis within an hour and the chloride sample analysis within a shift).

<u>Response</u>

A design and operational review of the reactor coolant and containment atmosphere sampling systems has been performed for the OCNGS. The reactor coolant and containment atmosphere sampling stations are in the Reactor Building and may not be accessible during an accident.

The design and operational review of the plant radiological analysis and chemical analysis facilities has shown that these facilities are accessible during an accident, and procedures are in place to perform prompt analysis of samples during post accident conditions from these stations.

NRC has determined that the postaccident sampling system meets all the criteria of this item and is, therefore, acceptable.

The PASS was installed as required by NRC Order dated July 7, 1981 as generally described in NUREG-0737. In response to industry initiatives, the NRC approved NEDO-32991, "Regulatory Relaxation for BWR Post Accident Sampling Stations (PASS) in its Safety Evaluation dated June 12, 2001." Relaxation of the PASS requirements (Amendment 237 to the Oyster Creek Technical Specification) was contingent on meeting the following commitments.

1. "Each licensee should verify that it has, and make a regulatory commitment to maintain (or make a regulatory commitment to develop and maintain), contingency plans for obtaining and analyzing highly radioactive samples of reactor coolant, suppression pool, and containment atmosphere.

2. “Each licensee should verify that it has, and make a regulatory commitment to maintain (or make a regulatory commitment to develop and maintain), a capability for classifying fuel damage events at the Alert level threshold (typically this is 300 uCi/ml dose equivalent iodine). This capability may utilize the normal sampling system and/or correlations of radiation readings to radioisotope concentrations in the reactor coolant.”

3. “Each licensee should verify that it has, and make a regulatory commitment to maintain (or make a regulatory commitment to develop and maintain), an I-131 site survey detection capability, including an ability to assess radioactive iodines released to offsite environs, by using effluent monitoring systems or portable sampling equipment.”

These commitments to the NRC are met using a combination of the PASS equipment and other equipment/procedures. The PASS provides the means for obtaining and analyzing highly radioactive samples of reactor coolant, suppression pool and containment atmosphere. The PASS system is maintained in good working order and the operation of the system is described in approved plant procedures. The Station Emergency Plan includes classification of fuel damage events at the Alert level threshold and established an I-131 site survey detection capability. The Station Emergency Plan is controlled in accordance with 10CFR50.47.

1.9.18 Item II.B.4 - Training for Mitigating Core Damage

Position

Licensees are required to develop a training program to teach the use of installed equipment and systems to control or mitigate accidents in which the core is severely damaged. They must then implement the training program.

Response

A training program has been developed and implemented (January 1982) to teach the use of installed equipment and systems to control or mitigate accidents in which the core is severely damaged.

The program includes training of individuals as required by NRC and the contents of the program satisfy the requirements of TMI Action Plan Item II.B.4.1. NRC has determined that the requirements for this item have been met.

1.9.19 Item II.D.1 - Performance Testing of Reactor Relief and Safety Valves

Position

Pressurized water reactor and boiling water reactor licensees and applicants shall conduct testing to qualify the reactor coolant system relief and safety valves under expected operating conditions for design basis transients and accidents.

Response

A test program has been conducted by the GE BWR Owners Group for performance testing of BWR safety valves. The conclusions arising from the Owner's Group evaluation of BWR operation under single and two phase flow conditions has been factored into the overpressurization analysis as they apply to Oyster Creek.

NRC considers this item complete.

1.9.20 Item II.D.3 - Direct Indication of Relief and Safety Valve Position

Position

Reactor coolant system relief and safety valves shall be provided with a positive indication in the Control Room derived from a reliable valve position detection or a reliable indication of flow in the discharge pipe.

Response

An acoustical monitoring system has been installed at Oyster Creek to monitor the position of the five relief valves and nine safety valves. The system provides positive indication of valve position and an annunciation of an open valve in the Control Room. The valve position indication components have been seismically and environmentally qualified as appropriate for the conditions applicable.

Backup valve position indication is afforded by temperature sensors located downstream of the safety and relief valves. In addition, high drywell pressure and temperature provide indication of an open valve.

The requirements for this item have been satisfied.

1.9.21 Item II.E.4.1 - Dedicated Hydrogen Penetrations

Position

Plants using external recombiners or purge systems for postaccident combustible gas control of the containment atmosphere should provide containment penetration systems for external recombiner or purge systems that are dedicated to that service only, that meet the redundancy and single failure requirements of General Design Criteria 54 and 56 of Appendix A to 10CFR50, and that are sized to satisfy the flow requirements of the recombiner or purge system.

The procedures for the use of combustible gas control systems following an accident that results in a degraded core and release of radioactivity to the containment must be reviewed and revised, if necessary.

Response

Oyster Creek was licensed to use a hydrogen purge system and nitrogen supply system for postaccident combustible gas control of the Primary Containment atmosphere. The isolation valves on the containment penetrations for these systems were designed to meet the single failure criteria for containment isolation but not for operation of these systems.

The hydrogen purge and nitrogen supply systems, which are used for postaccident combustible gas control of the containment atmosphere, can be operated entirely from the Control Room.

No changes to the shielding and the operating procedures for these systems are needed to operate them during postaccident conditions.

Modifications to the containment vent and nitrogen purge systems have been made to include single failure proof manifolds for the containment vent line from the drywell, the nitrogen purge line to the drywell and the nitrogen purge line to the torus. A flanged connection has been provided from outside the Reactor Building so that a portable supply of nitrogen may be connected in the event of loss of the nitrogen tank.

NRC has indicated that the requirements of 10CFR50.44, Standards for Combustible Gas Control Systems, establish the criteria to conform with this Action Plan item. NRC safety evaluation, dated May 30, 1985, closed Item II.E.4.1 for purposes of tracking in the TMI items system. Compliance with 10CFR50.44 is documented in NRC safety evaluation dated November 18, 1992.

1.9.22 Item II.E.4.2 - Containment Isolation Dependability

Position

(1) Containment isolation system designs shall comply with the recommendations of Standard Review Plan Section 6.2.4 (i.e., that there be diversity in the parameters sensed for the initiation of Containment isolation).

(2) All plant personnel shall give careful consideration to the definition of essential and nonessential systems, identify each system determined to be essential, identify each system determined to be nonessential, describe the basis for selection of each essential system, modify their Containment isolation designs accordingly, and report the results of the re-evaluation to the NRC.

(3) All nonessential systems shall be automatically isolated by the Containment isolation signal.

(4) The design of control systems for automatic Containment isolation valves shall be such that resetting the isolation signal will not result in the automatic reopening of Containment isolation valves. Reopening of Containment isolation valves shall require deliberate operator action.

(5) The Containment setpoint pressure that initiates Containment isolation for nonessential penetrations must be reduced to the minimum compatible with normal operating conditions.

(6) Containment purge valves that do not satisfy the operability criteria set forth in Branch Technical Position CSB 6-4 or the Staff Interim Position of October 23, 1979 must be sealed closed as defined in SRP 6.2.4, Item III.3.f during operational conditions 1, 2, 3 and 4. Furthermore, these valves must be verified to be closed at least every 31 days.

(7) Containment purge and vent isolation valves must close on a high radiation signal.

Response

All systems penetrating Primary Containment have been classified as essential or nonessential. The isolation provisions of nonessential systems, except those with locked closed handwheel operated valves, have been modified to provide diverse containment isolation signals to all valves which were not already so equipped. The diverse containment isolation parameters are low-low reactor water level and high drywell pressure.

The Containment Isolation System at Oyster Creek is designed to prevent inadvertent automatic reopening of the containment isolation valves following clearing of the containment isolation signals. It is necessary for the operator to actuate a reset button to permit the valves to be reopened. The reset button has no effect if the containment isolation signal is present.

If the containment isolation signal has cleared, actuation of the reset button activates the valve control switches to put the isolation valves in the positions selected by the valve control switches. Administrative controls require that the valve control switches of containment isolation valves be placed in the closed position following an automatic containment isolation.

The containment pressure setpoint that initiates containment isolation has been reviewed. The pressure setpoint is 2.9 psig and is consistent with NUREG-0737 requirements.

The NRC has determined that the interim position for operation of the containment vent and purge valves has been met. The NRC also concluded that small (2 inch) containment purge and vent isolation valves need not be isolated by a high radiation signal.

In regard to the interim requirement that all valves greater than three inch nominal diameter must be qualified for use, or not be used, unless the reactor is in cold shutdown, or in refueling mode, butterfly valve openings shall be limited to 30° or less until such time as the valves are demonstrated to be fully qualified or replaced with fully qualified valves. NRC has concluded that the requirements of position 6 of this item for Oyster Creek have been met. V-27-1, V-27-2, V-27-3 & V-27-4 have been replaced with qualified valves; therefore, this restriction is no longer in use on these valves.

Two containment high range radiation monitors located in the drywell at approximately 51 ft elevation provide isolation signals to the valves (see Item II.F.1.3). These monitors are described in Section 11.5.2.13.

1.9.23 <u>Item II.F.1.1 - Noble Gas Effluent Monitor</u>

<u>Position</u>

Noble gas effluent monitors shall be installed with an extended range designed to function during accident conditions, as well as during normal operating conditions. Multiple monitors are considered necessary to cover the ranges of interest.

(1) Noble gas effluent monitors with an upper range capability of 10^5 ûCi/cc (Xe-133) are considered to be practical and should be installed in all operating plants.

(2) Noble gas effluent monitoring shall be provided for the total range of concentration extending from normal condition (as low as reasonably achievable - ALARA) concentrations to a maximum of 10^5 ûCi/cc (Xe-133). Multiple monitors are considered to be necessary to cover the ranges of interest. The range capacity of individual

monitors should overlap by a factor of ten.

Response

Two Radioactive Gaseous Effluent Monitoring Systems (RAGEMS) are installed at Oyster Creek to perform this function; one to monitor releases at the plant stack (RAGEMS I) and one at the turbine building vent (RAGEMS II). The design basis maximum range specified in Table II.F.1-1 of NUREG 0737 cannot be met with the high range monitors used in RAGEMS I & II. However, plant specific analysis indicates both monitors are fully capable of detecting and measuring the calculated maximum concentration of noble gas fission products released via the two pathways at Oyster Creek. The calculated maximum concentration at the plant stack and turbine building vent after a design basis Loss-of-Coolant Accident (LOCA) is 13 ûCi/cc and 1.44×10^{-5} ûCi/cc, respectively. The high range monitors are capable of detecting and measuring up to 127 ûCi/cc.

GPUN's measured upper limit of detection capability of the installed RAGEMS is a factor of about eight lower than that suggested in NUREG-0737 for the stack monitor. GPUN performed the analysis to show that its upper limit is acceptable for the Oyster Creek Nuclear Generating Station.

The NRC staff subsequently performed an independent analysis, and agreed with GPUN that these RAGEM monitors should not saturate during a design basis accident. Thus GPUN's range of detection is acceptable. The flow rate in the Oyster Creek stack is very high (68,000 cubic feet per minute), which provides a large dilution factor for activity released to the stack. Thus, the required range of a detector for this application at Oyster Creek is less than average.

1.9.24 Item II.F.1.2 - Sampling and Analysis of Plant Effluents

Position

Because iodine gaseous effluent monitors for the accident condition are not considered to be practical, at this time, capability for effluent monitoring of radioiodines for the accident condition shall be provided with sampling conducted by adsorption on charcoal or other media, followed by onsite laboratory analysis.

Response

Radioiodine/Particulate cartridges in the plant ventilation stack will be analyzed during postaccident conditions to quantify high level radioiodine/ particulate releases from the plant as a result of an accident.

The instrumentation available cannot meet the design basis shielding envelope criteria of 10^2 ûCi/cc of gaseous radioiodine and particulates deposited in the sampling media. The online system will monitor releases in the exhaust up to levels of 10^{-2} ûCi/cc.

The final operating configuration was in place prior to Cycle 12 Operation.

1.9.25 Item II.F.1.3 - Containment High Range Radiation Monitor

Position

In Containment radiation level monitors with a maximum range of 10^8 rad/hr shall be installed. A minimum of two (2) such monitors that are physically separated shall be provided. Monitors shall be developed and qualified to function in an accident environment.

Response

The containment high range radiation monitors (see Section 11.5.2.13) were installed during the Cycle 11 refueling outage.

There are numerous methods for detecting a major accident within Primary Containment, such as high drywell pressure, low-low reactor water level, torus water level, neutron monitoring, and others. This instrument will provide additional indication of major core degradation.

The containment high radiation signal is one of three isolation signals for the containment purge and vent isolation valves (see Item II.E.4.2).

1.9.26 Item II.F.1.4 - Containment Pressure Monitor

Position

A continuous indication of Containment pressure shall be provided in the Control Room of each operating reactor. Measurement and indication capability shall include three (3) times the design pressure for steel, and -5 psig for all Containments.

Response

The design of the containment pressure monitor was reviewed and approved by the NRC. The monitor was operational at the time of startup from the Cycle 10 refueling outage.

1.9.27 Item II.F.1.5 - Containment Water Level Monitor

Position

A continuous indication of Containment water level shall be provided in the Control Room for all plants. For BWR's, a wide range instrument shall be provided and cover the range from the bottom to five feet above the normal water level of the suppression pool.

Response

A containment water level monitor was installed in the torus. Readout is via recorder-indicator in the Control Room. The monitor was operational at the time of startup from the Cycle 10 refueling outage.

1.9.28 Item II.F.1.6 - Containment Hydrogen Monitor

Position

A continuous indication of hydrogen concentration in the containment atmosphere shall be provided in the Control Room. Measurement capability shall be provided over the range of 0 to

10% hydrogen concentration under both positive and negative ambient pressure.

The continuous indication of hydrogen concentration is not required during normal operation.

If an indication is not available at all times, continuous indication and recording shall be functioning within 30 minutes of safety injection. See NUREG-0737 II.F.1 Attachment 6.

Response

The containment hydrogen monitor has been provided with a separate indicator and recorder arrangement in the Control Room. Samples are obtained through two hydrogen sample ports (one per channel) at the top of the drywell dome, which permits rapid detection of hydrogen escaping from the reactor. The monitor was operational at the time of startup from the Cycle 10 refueling outage.

The containment hydrogen monitor is normally operated in standby and is turned to analyze when required per plant emergency operating procedures. Continuous indication and recording will be functioning approximately 5 minutes after placing the analyzers in the analyze mode.

1.9.29 Item II.F.2 - Instrumentation for Detection of Inadequate Core Cooling

Position

Licensees shall provide a description of any additional instrumentation or controls (primary or backup) proposed for the plant to supplement existing instrumentation (including primary coolant saturation monitors) in order to provide an unambiguous, easy to interpret indication of inadequate core cooling (ICC). A description of the functional design requirements for the system shall also be included. A description of the procedures to be used with the proposed equipment, the analysis used in developing these procedures, and a schedule for installing the equipment shall be provided.

Response

The OCNGS fuel zone water level instrumentation is density compensated up to saturation conditions. When saturation is detected in the reference leg, this instrumentation turns itself off. Since thermal effects of nearby steam and feedwater lines on the reference leg are minimal, the instrumentation readings are representative of the warmer fluid in the reference leg.

An Analog Trip System was installed for reactor lo-level scram during the Cycle 11 refueling outage.

1.9.30 Item II.K.3.3 - Safety and Relief Valve Challenges and Failures

Position

Report safety and relief valve failures promptly and challenges annually.

Response

A report on relief valve (RV) and safety valve (SV) failures and challenges was submitted to the NRC. Technical Specification Section 6.9-3 will address future reporting of RV and SV failures and challenges. The NRC has judged this item to be complete for the OCNGS.

1.9.31 Item II.K.3.14 - Isolation of Isolation Condensers on High Radiation

Position

Isolation condensers have radiation monitors on their vents. These monitors provide alarms in the Control Room but do not isolate the isolation condenser. The isolation condensers are currently isolated on a high radiation signal in the steam line leading to the isolation condensers. The design should be modified such that the isolation condensers are automatically isolated upon receipt of a high radiation signal at the vent rather than at the steam line. The purpose of the change is to increase the availability of the isolation condensers as heat sinks.

Response

The OCNGS Isolation Condensers are not isolated on high radiation signals in the steam lines, thus the design modification as specified in this item is not relevant and does not increase the availability of the Isolation Condensers as heat sinks.

In their evaluation of this item, the NRC staff has concluded that the manual trip on high radiation levels at the vents is sufficient to provide the amount of flexibility and system availability intended by this item.

In order to increase the availability of the Isolation Condensers (IC's) as heat sinks, NUREG 0737 Item II.K.3.14 required that their isolation on high radiation be switched from the Main Steam Line Radiation Monitors to the IC Vent Radiation Monitors. In response, GPUN, by a letter from Ivan R. Finfrock to the NRC, dated April 30, 1981, clarified the fact that Oyster Creek did not isolate the ICs on high radiation and instead the isolation was provided by detection of excessive flow in the steam line to and condensate line from the IC. In its response letter, dated December 19, 1981, the NRC stated that the GPUN response to Item II.K.3.14 was acceptable and the item was considered resolved. At this time, although the IC Vent Radiation Monitors are physically inoperable, the intent of the NUREG isolation requirement is still met by the excessive flow sensors.

The isolation condenser vent radiation monitors were later removed. It was concluded, based on an evaluation, that the background radiation in the vicinity of the monitors masks their capability for leak detection and alternate means are available to detect leakage in a timely manner.

1.9.32 Item II.K.3.16 - Reduction of Challenges and Failures of Relief Valves - Feasibility Study and System Modification Position

The record of relief valve failures to close for all boiling water reactors (BWRs) in the past three years of plant operation is approximately 30 in 73 reactor years (0.41 failures per reactor year). This had demonstrated that the failure of a relief valve to close would be the most likely cause of a small break Loss-of-Coolant Accident (LOCA). The high failure rate is the result of a high relief valve challenge rate and a relatively high failure rate per challenge (0.16 failures per challenge). Typically, five valves are challenged in each event. This results in an equivalent failure rate per challenge of 0.03. The challenge and failure rates can be reduced in the following ways:

(1) Additional anticipatory scram on loss of feedwater;

(2) Revised relief valve actuation setpoints;

(3) Increased emergency core cooling (ECC) flow;

(4) Lower operating pressures;

(5) Earlier initiation of ECC systems;

(6) Heat removal through emergency condensers;

(7) Offset valve setpoints to open fewer valves per challenge;

(8) Installation of additional relief valves with a block or isolation valve feature to eliminate opening of the safety relief valves (SRVs), consistent with the ASME Code;

(9) Increasing the high steam line flow setpoint for main steam line isolation valve (MSIV) closure;

(10) Lowering the pressure setpoint for MSIV closure;

(11) Reducing the testing frequency of the MSIVs;

(12) More stringent valve leakage criteria; and,

(13) Early removal of leaking valves.

An investigation of the feasibility and contraindications of reducing challenges to the relief valves by use of the aforementioned methods should be conducted. Other methods should also be included in the feasibility study. Those changes which are shown to reduce relief valve challenges without compromising the performance of the relief valves or other systems should be implemented. Challenges to the relief valves should be reduced substantially (by an order of magnitude).

<u>Response</u>

By letter dated July 20, 1990, GPUN advised the NRC of a change to the preventive maintenance program for Electromatic Relief Valves (EMRVs). Due to improvements in the EMRVs (both design and material upgrades), our earlier program to rebuild all five EMRVs every refueling outage was thought to be too conservative. Beginning 13R, the rebuild schedule was modified to require rebuilding (or installation of a rebuilt spare valve) two or three EMRVs during the refueling outage and rebuilding the remaining valves during the next refueling outage. By letter dated September 14, 1990, the NRC agreed to this change.

1.9.33 Item II.K.3.17 - Report on Outages of Emergency Core Cooling Systems Licensee Report and Proposed Technical Specification Changes

Position

Several components of the emergency core cooling (ECC) systems are permitted by technical specifications to have substantial outage times (e.g., 72 hours for one diesel generator; 14 days for the HPCI system). In addition, there are no cumulative outage time limitations for ECC systems. Licensees should submit a report detailing outage dates and lengths of outages for all ECC systems for the last five years of operation. The report should also include the causes of the outages (i.e., controller failure, spurious isolation).

Response

The OCNGS report on outages of the Emergency Core Cooling System submitted to NRC was based on information compiled from records on reportable occurrences for the period from August 1, 1975 to August 1, 1980. The report provided the date, duration, cause component and corrective action for each event. Surveillance testing was based on an estimate of out of service times and frequency requirements.

The NRC has evaluated this report and determined that the cumulative ECCS outage time for the OCNGS is such that system modifications or Technical Specification changes are not necessary.

1.9.34 Item II.K.3.18 - Modification of Automatic Depressurization System Logic - Feasibility for Increased Diversity for SomeEvent Sequences

Position

The automatic depressurization system (ADS) actuation logic should be modified to eliminate the need for manual actuation to assure adequate core cooling. A feasibility and risk assessment study is required to determine the optimum approach. One possible scheme that should be considered is ADS actuation on low reactor vessel water level provided no high pressure coolant injection (HPCI) or high pressure coolant system (HPCS) flow exists and a low pressure emergency core cooling (ECC) system is running. This logic would complement, not replace, the existing ADS actuation logic.

Response

NRC has concluded, on the basis of additional information, that no modifications to the ADS logic are required, and that implementation of emergency procedures is sufficient for satisfying this action item.

1.9.35 Item II.K.3.19 - Interlock on Recirculation Pump Loops

Position

Interlocks should be installed on nonjet pump plants (other than Humboldt Bay) to assure that at least two recirculation loops are open for recirculation flow for modes other than cold shutdown. This is to assure that the level measurements in the downcomer region are representative of the level in the core region.

Response

The requirement that at least one recirculation loop is open for recirculation flow for modes other than cold shutdown is established in plant procedures (see Section 5.4.1.2.6). A modification to install an alarm in the Control Room to indicate that a fourth recirculation loop has been isolated and an alarm reflash to indicate that a fifth loop has been isolated was completed during the Cycle 11 refueling outage. Implementation of this item is complete.

1.9.36 Item II.K.3.21 - Restart of Core Spray and Low Pressure Coolant Injection System

Position

The core spray and low pressure, coolant injection (LPCI) system flow may be stopped by the operator. These systems will not restart automatically on loss of water level if an initiation signal is still present. The core spray and LPCI system logic should be modified so that these systems will restart, if required,to assure adequate core cooling. Because this design modification affects several core cooling modes under accident conditions, a preliminary design should be submitted for staff review and approval prior to making the actual modification.

Response

The Core Spray System (CSS) controls logic for the OCNGS has the following two deficiencies:

a. When the Core Spray pumps are tripped, after an ECCS actuation, the pumps cannot be restarted manually without resetting the core spray logic. However, the core spray logic cannot be reset if an ECCS signal is still present.

b. The Core Spray System pumps will not restart automatically upon loss of water level if an ECCS signal is still present.

A modification has been installed which is intended to remove these deficiencies from the core spray control logic. NRC has determined that the requirements of this item have been satisfied.

1.9.37 Item II.K.3.25 - Effect of Loss of Alternating Current Power on Pump Seals

Position

The licensees should determine, on a plant specific basis, by analysis or experiment, the consequences of a loss of cooling water to the reactor recirculation pump seal coolers. The pump seals should be designed to withstand a complete loss of alternating current (ac) power for at least two hours. Adequacy of the seal design should be demonstrated.

Response

The BWR Owners Group responses submitted to NRC for this item were endorsed by GPUN. The NRC review of these submittals verified that seal leakage data on pumps similar to those utilized at the OCNGS show leakage rates to be acceptable following loss of cooling to the pump seals. NRC has concluded that no modifications to the seal cooling for the recirculation pumps are required and that this item is complete for the OCNGS.

1.9.38 Item II.K.3.27 - Provide Common Reference Level for Vessel Level Instrumentation

Position

Different reference points of the various reactor vessel water level instruments may cause operator confusion. Therefore, all level instruments should be referenced to the same point. Either the bottom of the vessel or the top of the active fuel are reasonable reference points.

Response

The reactor vessel water level instrumentation has been modified to provide a common reference point (the top of the active fuel), and all appropriate procedural changes have been made to reflect the new common reference level. The necessary training programs were conducted during the Cycle 10 refueling outage.

1.9.39 Item II.K.3.29 - Study to Demonstrate Performance of Isolation Condensers with Noncondensibles

Position

If natural circulation plays an important role in depressurizing the system (e.g., in the use of isolation condensers), then the various modes of two phase flow natural circulation, including noncondensables, which may play a significant role in plant response following a small break Loss-of-Coolant Accident (LOCA) should be demonstrated.

Response

See response to Item II.B.1.

1.9.40 Item II.K.3.30 - Revised Small Break Loss-of-Coolant Accident Methods to Show Compliance with 10 CFR Part 50, Appendix K

Position

The analysis methods used by nuclear steam supply system (NSSS) vendors and/or fuel suppliers for small break Loss-of-Coolant Accident (LOCA) analysis for compliance with Appendix K to 10 CFR Part 50 should be revised, documented and submitted for NRC approval. The revisions should account for comparisons with experimental data, including data from the LOFT Test and Semiscale Test facilities.

Response

On December 13, 1983, the NRC accepted the GE small break LOCA model for use in satisfying TMI Action Item II.K.3.30. This GE model was originally approved in a letter dated February 4, 1981 as described in the GE Topical Report, NEDO 20566, Rev. 1 and Rev. 4.

This model was used in the LOCA analysis for the Oyster Creek Nuclear Generating Station and the analysis was documented in a letter to NRC dated July 13, 1982. This completed TMI Action Items II.K.3.30 and II.K.3.31 for the OCNGS.

1.9.41 Item II.K.3.31 - Plant Specific Calculations to Show Compliance with 10CFR50.46

Position

Plant specific calculations using NRC approved models for small break Loss-of-Coolant Accidents (LOCAs) as described in item II.K.3.30 to show compliance with 10CFR50.46 should be submitted for NRC approval by all licensees.

Response

See Item II.K.3.30.

1.9.42 Item II.K.3.44 - Evaluation of Anticipated Transients with Single Failure to Verify No Fuel Failure

Position

For anticipated transients combined with the worst single failure and assuming proper operator actions, licensees should demonstrate that the core remains covered or provide analysis to show that no significant fuel damage results from core uncovery. Transients which result from a stuck open relief valve should be included in this category.

Response

The BWR Owners Group generic response submitted to NRC for this item (BWROG-80-12) was endorsed by GPUN and reviewed by NRC. Since calculations show that, with proper operator action, the core does not uncover for the worst transient combined with the worst single failure and a stuck open relief valve and since test data show the calculations to be conservative for an ADS blowdown, NRC has found the generic responses of the BWR Owner's Group to be acceptable. NRC considers this item complete for the OCNGS since it has been verified that the assumptions and initial conditions used in the generic analyses are representative for this facility.

1.9.43 Item II.K.3.45 - Evaluation of Depressurization with Other than Automatic Depressurization System

Position

Analyses to support depressurization modes other than full actuation of the automatic depressurization system (ADS) (e.g., early blowdown with one or two safety relief valves (SRVs)) should be provided. Slower depressurization would reduce the possibility of exceeding vessel integrity limits by rapid cooldown.

Response

The BWR Owners Group generic response submitted to NRC for this item was endorsed by GPUN. The staff has concluded that the as designed Reactor Pressure Vessel and containment structure for the OCNGS would maintain structural integrity under the ADS events postulated in NUREG 0737, and would be able to withstand more than one ADS event. Overall, NRC concluded, alternate modes of depressurization in comparison to ADS blowdown would not contribute any significant benefit to plant operation and safety and, therefore, no modifications in plant design and operation are required.

1.9.44 Item II.K.3.46 - Michelson Concerns

Position

Consider water hammer effects in the core spray sparger design.

Response

The design of the new overhead grid core spray sparger system has taken water hammer effects into consideration. It should be noted that there is no requirement at this time to install the overhead grid sparger.

1.9.45 Item II.A.1.1 - Upgrade Emergency Preparedness

Position

Provide an emergency response plan in substantial compliance with NUREG-0654, "Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants" (which may be modified after May 13, 1980 based on public comments) except that only a description of a completion schedule for the means for providing prompt notification to the population (Appendix 3), the staffing for emergencies in addition to that already required (Table B.1), and an upgraded meteorological program (Appendix 2) need to be provided. NRC will give substantial weight to FEMA findings on offsite plans in judging the adequacy against NUREG-0654. Perform an emergency response exercise to test the integrated capability and a major portion of the basic elements existing within emergency preparedness plans and organizations (NUREG-0654).

Response

See Item III.A.2.1.

1.9.46 Item III.A.1.2 - Upgrade Emergency Support Facilities

Position

The emergency response facilities functions should conform to the guidelines in NUREG-0696.

Response

The OCNGS Emergency Support Facilities fully address the emergency response facilities functions as outlined in NUREG-0696.

The Emergency Response Facilities for the OCNGS include the following:

a. The Technical Support Center (TSC) - Is located on the lower level of the Site Emergency Building and is operational.

b. The Operational Support Center (OSC) - Is located in the Drywell Processing Center.

c. An Emergency Operations Facilities (EOF) - Is located in Toms River, NJ.

The emergency response capabilities for the OCNGS have been tested and found acceptable (see Item III.A.2.1) by NRC.

1.9.47 Item III.A.2.1 - Emergency Preparedness: Upgrade Emergency Plans to Appendix E, 10CFR50

Position

Each nuclear facility shall upgrade its emergency plans to provide reasonable assurance that adequate protective measures can and will be taken in the event of a radiological emergency. Specific criteria to meet this requirement is delineated in NUREG-0654 (FEMA-REP-1), "Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparation in Support of Nuclear Power Plants."

Response

The OCNGS Emergency Plan was submitted in accordance with the requirements of 10CFR50.47. Following emergency appraisals and full scale exercises, NRC has considered that the requirements for this item are satisfied.

1.9.48 Item III.A.2.2 - Emergency Preparedness: Meteorological Data

Position

Proposed Revision 1 to Regulatory Guide 1.23 outlines the set of meteorological measurements that should be accessible from a system that can be interrogated; the meteorological data should be presented in the prescribed format. The results of the assessments should be accessible from this system; this information should incorporate human factors engineering in its display to convey the essential information to the initial decision makers and subsequent management team. An integrated system should allow the eventual incorporation of effluent monitoring and radiological monitoring information with the environmental transport to provide direct dose consequence assessments.

Response

The status of this item, with regard to NUREG-0654, Rev. 1, is as follows:

a. Capability for making meteorological measurements:

   The meteorological tower at Oyster Creek has redundant meteorological sensors and an independent power supply (site generator) was installed prior to restart.

b. Capability for making near real-time predictions of the atmospheric effluent transport and diffusion:

Oyster Creek has the capability for making near real-time predictions of the atmospheric effluent transport and diffusion. However, local climatic effects such as an established thermally-induced land-sea breeze cannot be realistically simulated in the Class A model. Qualitative assessment of the land-sea breeze has been established based on site specific studies by Heck (1985) and Schwartz (1987).

c. Capability for remote interrogation of the atmospheric measurements and predictions by appropriate organizations:

Compliance with this item has been met as of February 22, 1982. Currently, meteorological measurements can be remotely interrogated via the Plant Process Computer; predictions can be accessed by interfacing with the Oyster Creek Emergency Preparedness organization. The primary means of interrogating atmospheric measurements and predictions is through GPU's Environmental Monitoring Information System (EMIS).

The onsite meteorological data collection system has been upgraded to include redundant means of measuring and recording meteorological parameters at 33', 150' and 380' above plant grade.

GPU Nuclear has entered into agreements with the Atlantic City National Weather Service Station, McGuire Air Force Base, the local Coast Guard Station, the Lakehurst Naval Air Station and the National Weather Service to provide meteorological data should the need arise.

GPU Nuclear has an earth station satellite receiver in which National Weather Service numerical weather prediction maps as well as tabular meteorological data from local or national sources can be obtained. This system is connected directly to the GPU local and wide-area computer network systems. Any of the above information can be accessed via computer terminal located throughout the GPU system.

The onsite monitoring system and the ability to access local and regional data from several sources supports the statement that the meteorological data availability rate is expected to approach the goal described in NUREG-0654 of 99 percent.

In conformance with Regulatory Guide 1.97, a site specific study has been completed to identify the following:

1. Type A variables.

2. Plant specific ranges.

3. Determine whether or not all parameters specified in Regulatory Guide 1.97 are applicable to Oyster Creek.

4. Determine whether additional parameters, other than those specified in Regulatory Guide 1.97, may be required for Oyster Creek.

The above studies have been integrated with the safety analysis for the SPDS (see Item I.D.2).

1.9.49 Item III.D.1.1 - Integrity of Systems Outside Containment Likely to Contain Radioactive Material

Position

Applicants shall implement a program to reduce leakage from systems outside containment that would or could contain highly radioactive fluids during a serious transient or accident to as low as practical levels. This program shall include the following:

(1) Immediate leak reduction

(a) Implement all practical leak reduction measures for all systems that could carry radioactive fluid outside of containment.

(b) Measure actual leakage rates with system in operation and report them to the NRC.

(2) Continuing Leak Reduction -- Establish and implement a program of preventive maintenance to reduce leakage to as-low-as-practical levels. This program shall include periodic integrated leak tests at intervals not to exceed each refueling cycle.

Response

Plant systems outside containment which would or could contain highly radioactive fluids during a serious transient or accident include: shutdown cooling, isolation condenser, core spray, containment spray, reactor water cleanup, standby gas treatment, reactor coolant sampling, reactor coolant system instrumentation, drywell equipment and floor drains, reactor equipment and floor drains and the scram dump volume and associated vents and drains. An initial leakage reduction test/inspection was performed on these systems. A total of 1925 inspections were made on 1642 components, considering both internal and external leakage paths. Where leaks existed, repairs were made, if possible, and post repair inspections were made to document the reduced leakage. Results were tabulated and reported.

Following performance of this initial leakage reduction program, an ongoing leak reduction program was implemented which includes various leak tests/inspections on a 24-month (refueling interval) basis. The systems inspected under the ongoing preventive maintenance program include those portions of shutdown cooling, isolation condensers, core spray, containment spray and reactor water cleanup systems that can be inspected as permitted by system design and radiological conditions. In addition, the ongoing program includes identification of leakage from visual surveillance by plant personnel (operator rounds/tours) and responses of area and effluent monitors.

In response to IE Circular 79-21, a leak reduction program has been implemented wherein flow paths by which radioactivity can leave the plant internal environs to the outside environment have been identified and examined. Flow paths to the outside environment were reviewed with the objective of minimizing the potential of unplanned radioactivity releases.

1.9.50 Item III.D.3.3 - Improved Inplant Iodine Instrumentation Under Accident Conditions

Position

(1) Each licensee shall provide equipment and associated training and procedures for accurately determining the airborne iodine concentration in areas within the facility where plant personnel may be present during an accident.

(2) Each applicant for a fuel loading license to be issued prior to January 1, 1981 shall provide the equipment, training, and procedures necessary to accurately determine the presence of airborne radioiodine in areas within the plant where plant personnel may be present during an accident.

Response

Air samplers containing Silver Zeolite radioiodine sampling cartridges are contained in Emergency kits in the Control Room to collect and analyze air samples for radioiodine during an accident.

Oyster Creek Emergency Procedures incorporate ALARA concepts for inplant radioiodine sampling during accident conditions. Personnel training in the Oyster Creek Emergency Procedures is required to be performed on a periodic basis.

Both an onsite and offsite I-131 monitoring capability is maintained as part of the Station Emergency Plan and Emergency Plan implementing procedures. The Station Emergency Plan is controlled and maintained in accordance with 10CFR50.47 and provides the capabilities described in Section 1.9.17, Item II.B.3 NRC Comments.

1.9.51 Item III.D.3.4 - Control Room Habitability Requirements

Position

In accordance with Task Action Plan Item III.D.3.4 and Control Room habitability, licensees shall assure that Control Room operators will be adequately protected against the effects of accidental release of toxic and radioactive gases and that the nuclear power plant can be safely operated or shutdown under design basis accident conditions (Criterion 19, "Control Room," of Appendix A, "General Design Criteria for Nuclear Power Plants," to 10 CFR Part 50).

Response

Interim modifications for Control Room Habitability were implemented during the Cycle 11 refueling outage. NRC approved Amendments No. 115 and 139 to the OCNGS Facility Operating License which completes the interim and final actions in response to this topic. The final modifications were implemented during the Cycle 12 refueling outage (see Section 6.4).

1.9.52 References

(1) Safety Evaluation by the Office of NRR Relating to High Point Vents for the Isolation Condenser, dated April 24, 1986.

1.10 SYSTEMATIC EVALUATION PROGRAM SUMMARY

The Systematic Evaluation Program (SEP) was initiated by NRC to review the designs of older operating nuclear reactor plants to reconfirm and document their safety. The review was intended to provide: (1) an assessment of how these plants compare with licensing safety requirements relating to selected issues through 1982, (2) a basis for deciding on how any differences identified in item (1) should be resolved in an integrated plant review, and (3) a documented evaluation of plant safety.

Thus, the SEP entailed comparing the "as-built" plant design with 1982 review criteria within 137 areas denominated as "topics". The definition of each of the 137 topics, as well as other pertinent information, is given in Appendix A to NUREG-0822, Integrated Plant Safety Assessment. In the review, 54 of the 137 topics were deleted from consideration by the SEP because a review was being conducted under other programs (such as those dealing with Unresolved Safety Issues or Three Mile Island Action Plan Tasks), or the topic was not applicable to Oyster Creek (e.g., PWR related issues). Thus, of the original 137 topics only 83 were reviewed for the OCNGS.

The evaluation of the as-built plant design versus the 1982 review criteria revealed that certain features of the facility differed from the criteria. Discrepancies were noted for a total of 40 topics, which were the ones considered in the integrated safety assessment of the facility. This assessment consisted of evaluating the safety significance and other aspects of the identified differences to ascertain whether backfitting was necessary from the standpoint of overall plant safety. The process of decision making included application of engineering judgement and the utilization of limited probabilistic risk assessment methodologies.

NUREG-0822 was issued by NRC to document the review of the OCNGS under the SEP. The report provides a description of the topics, as they apply to Oyster Creek, and the resolution or status of each topic. At the time of issuance of NUREG-0822, the detailed review of some of the issues was not completed. Since that time additional analyses have been performed and submitted to NRC. The NRC subsequently issued Supplement No. 1 to NUREG-0822, documenting their review and status of completed and unresolved issues.

In addition, NRC issued NUREG-1382, "Safety Evaluation Report related to the full term operating license for Oyster Creek Nuclear Generating Station". Because much of the review necessary for conversion of the Provisional Operating License is similar to the scope reviewed for the SEP, the major portion of the technical input supporting the NUREG-1382 has come from the SEP topic evaluations. NUREG -1382 provides a description of the SEP topic objectives, and the resolution or status of each topic as of January, 1991. Subsequently, all of the remaining SEP topics have been resolved.

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