NUREG–1382



# Safety Evaluation Report

## related to the full-term operating license for Oyster Creek Nuclear Generating Station

Docket No. 50–219

GPU Nuclear Corporation and
Jersey Central Power & Light Company



**U.S. Nuclear Regulatory Commission**

**Office of Nuclear Reactor Regulation**

**January 1991**

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TI91 006702

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# ABSTRACT

The Safety Evaluation Report for the full-term operating license application filed by GPU Nuclear Corporation and Jersey Central Power & Light Company for the Oyster Creek Nuclear Generating Station has been prepared by the Office of Nuclear Reactor Regulation of the U.S. Nuclear Regulatory Commission. The facility is located in Ocean County, New Jersey. The staff concludes that the facility can continue to be operated without endangering the health and safety of the public.

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# ACRONYMS AND INITIALISMS

| | |
|---|---|
| ABT | automatic bus transfer |
| ACRS | Advisory Committee on Reactor Safeguards |
| AEC | U.S. Atomic Energy Commission |
| AEOD | Office for Analysis and Evaluation of Operational Data |
| AIT | augmented inspection team |
| ALARA | as low as is reasonably achievable |
| ANSI | American National Standards Institute |
| AOG | augmented offgas |
| ARI | alternate rod injection |
| ASME | American Society of Mechanical Engineers |
| ATWS | anticipated transient(s) without scram |
| BNE | Bureau of Nuclear Engineering, State of New Jersey |
| BTP | branch technical position |
| BWR | boiling-water reactor |
| BWROG | BWR Owners Group |
| CFR | Code of Federal Regulations |
| CMAA | Crane Manufacturers Association of America |
| DBA | design-basis accident |
| DCRDR | detailed control room design review |
| DCS | drywell containment system |
| DER | design electrical rating |
| ECCS | emergency core cooling system |
| EMRV | electromatic relief valve |
| EOF | emergency operations facility |
| EOP | emergency operating procedure |
| EQ | environmental qualification |
| ESF | engineered safety feature(s) |
| ETE | evacuation time estimate |
| FAA | Federal Aviation Administration |
| FEMA | Federal Emergency Management Agency |
| FES | final environmental statement |
| FHA | fire hazards analysis |
| FMEA | failure modes and effects analysis |
| FR | Federal Register |
| FRC | Franklin Research Center |
| FSAR | final safety analysis report |
| FTOL | full-term operating license |
| GDC | general design criterion(a) |
| GE | General Electric |
| GL | generic letter |
| GPUN | GPU Nuclear Corporation |

| | |
|---|---|
| HED | human engineering discrepancy |
| HELB | high energy line break |
| HVAC | heating, ventilation, and air conditioning |
| IE | Office of Inspection and Enforcement |
| IEEE | Institute of Electrical and Electronics Engineers |
| IGSCC | intergranular stress corrosion cracking |
| IPSAR | integrated plant safety assessment report |
| ISI | inservice inspection |
| JCP&L | Jersey Central Power & Light Co. |
| LCO | limiting condition(s) for operation |
| LER | licensee event report |
| LOCA | loss-of-coolant accident |
| LPZ | low population zone |
| MCC | motor control center |
| MDC | maximum dependable capacity |
| MPA | multiplant action |
| MSIV | main steam line isolation valve |
| MSL | mean sea level |
| MWe | megawatt(s)-electric |
| MWh | megawatt-hours |
| MWt | megawatt(s)-thermal |
| mybp | million years before present |
| NEPA | National Environmental Policy Act |
| NRC | U.S. Nuclear Regulatory Commission |
| NUMARC | Nuclear Management and Resources Council |
| NWS | National Weather Service |
| OBE | operating basis earthquake |
| ORNL | Oak Ridge National Laboratory |
| OSC | operational support center |
| PASS | postaccident sampling system |
| PCP | process control program |
| PGP | procedures generation package |
| POL | provisional operating license |
| PRA | probabilistic risk assessment |
| RAGEMS | radioactive gaseous effluent monitoring system |
| RCP | reactor coolant pump |
| RCPB | reactor coolant pressure boundary |
| RCS | reactor coolant system |
| RETS | Radwaste Emissions Technical Specifications |
| RG | regulatory guide |
| RPS | reactor protection system |
| RTS | reactor trip system |
| SAI | Science Applications, Incorporated |
| SALP | systematic assessment of licensee performance |
| scfm | standard cubic feet per minute |

| | |
|---|---|
| SE | safety evaluation |
| SEP | Systematic Evaluation Program |
| SER | safety evaluation report |
| SFPCS | spent fuel pool cooling system |
| SJAE | steam jet air ejector |
| SLCS | standby liquid control system |
| SPDS | safety parameter display system |
| SQUG | Seismic Qualification Utility Group |
| SRP | Standard Review Plan |
| SSE | safe shutdown earthquake |
| SSOMI | safety system outage modification inspection |
| TER | technical evaluation report |
| TMI | Three Mile Island |
| TMI-2 | Three Mile Island Unit 2 |
| TS | technical specification(s) |
| TSC | technical support center |
| UHS | ultimate heat sink |
| USC&GS | U.S. Coast and Geodetic Survey |
| USGS | U.S. Geological Survey |
| USI | unresolved safety issue |
| VACP | vital ac panel |
| VAR | voltage-ampere reactive |
| VMS | valve monitoring system |

# 1 INTRODUCTION AND DISCUSSION

## 1.1 Introduction

This report is a Safety Evaluation Report (SER) on the application for a full-term operating license (FTOL) for the Oyster Creek Nuclear Generating Station (Oyster Creek or the facility) that was filed by the colicensees GPU Nuclear Corporation (GPUN) and Jersey Central Power & Light Company (JCP&L). This report was prepared by the U.S. Nuclear Regulatory Commission (NRC) staff (the staff) and summarizes the results of the staff's review of the proposed conversion from a provisional operating license (POL) to an FTOL.

From 1959 to 1971, the U.S. Atomic Energy Commission issued POLs to 15 power reactors for periods of up to 18 months as an intermediate stage before issuing an FTOL. The purpose of the POL was to provide an interim period of routine operation during which the licensee and staff could assess plant operating parameters and performance against predicted values and resolve generic concerns identified during the licensing process. Thirty days after March 30, 1970, a rule change went into effect that deleted from the regulations the option of issuing POLs, but made no provision for converting previously issued POLs. Pursuant to Section 2.109 of Title 10 of the Code of Federal Regulations (10 CFR 2.109), the POL would not be deemed to have expired provided the licensee filed an application for renewal at least 30 days before the expiration date. Since each of the POL licensees has submitted a timely action for an FTOL, the remaining four POLs could continue indefinitely until the Commission completes its licensing action. Notwithstanding the silence of regulations on conversion, the NRC policy is to act as soon as possible on the POL conversion reviews.

JCP&L filed an application to convert POL DPR-16 for Oyster Creek to an FTOL in a letter dated March 6, 1972. The facility received its POL on April 9, 1969, achieved initial criticality on May 3, 1969, and began electric power generation on December 23, 1969.

In 1975, because of a large backlog of unresolved generic issues that were relevant to the operation of the POL plants, the staff stopped its review of the POL conversions and set out to establish the appropriate scope of review needed to support the conversion to full-term licenses.

In 1977, the NRC staff recommended to the Commission that POL facilities be included in Phase II of the Systematic Evaluation Program (SEP) because much of the review necessary for conversion of the POLs was similar to the scope of the review proposed for the SEP. That recommendation was adopted, and the major portion of the technical input supporting this SER comes from the SEP topic evaluations and the SEP Integrated Plant Safety Assessment Report (IPSAR) for Oyster Creek (NUREG-0822).

The SEP was conceived in recognition of the fact that because of the evolutionary nature of licensing requirements and advances in technology, better documentation was needed to better substantiate the staff's opinion that currently operating plants are acceptably safe. The objectives established for the SEP are listed on page 3 of SECY 76-545 as:

(1) The Systematic Evaluation Program must assess the safety adequacy of the design and operation of currently licensed nuclear power plants.

(2) The program should establish documentation which shows how each operating plant reviewed compares with current criteria on significant safety issues, and should provide a rationale for acceptable departures from these criteria.

(3) The program should provide the capability to make integrated and balanced decisions with respect to any required backfitting.

(4) The program should be structured for early identification and resolution of any significant deficiencies.

(5) The program should efficiently use available resources and minimize requirements for additional resources by NRC or industry.

Thus, the SEP review provided (1) an assessment of the significance of differences between current technical positions 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. To document the results of the SEP review for Oyster Creek, the staff issued NUREG-0822. NUREG-0822 was initially published in draft format in September 1982 and was issued in final form after Commission review in January 1983. Some followup requirements for additional analysis by the licensee that may result in the need for facility modification or other corrective action are identified in the Final IPSAR. These requirements have been reviewed as operating reactor licensing actions and are addressed in Supplement 1 to the IPSAR dated July 1988.

The major portion of the technical input supporting the staff SER has been provided by the IPSAR and SEP topic evaluations. (For definitions of each SEP topic, see Appendix A to the IPSAR.) The remainder of this SER will address other operating license issues not covered under the SEP. The SER includes consideration of major plant modifications that have occurred since the POL was issued, major substantive regulations adopted since the POL was issued, requirements stemming from the accident at Three Mile Island Unit 2 (TMI-2), and unresolved safety issues (USIs). USIs are issues considered on a generic basis after the staff has made the initial determination that the safety significance of the issue does not prohibit continued operation or require licensing actions while the longer term generic review is under way.

The format of this SER follows the general format of SERs currently issued for new operating licenses, but for many of the major headings, particularly those covered under the SEP, this SER briefly summarizes the findings of the Final IPSAR and its supplements or the SEP topic SERs. Similarly, when SERs have been issued on other topics, such as compliance with Appendix I, this SER briefly summarizes the previous SER and assesses whether the earlier findings are still valid.

Appendix A contains a list of references other than NRC documents or correspondence to or from the licensee cited in this report.* Appendix B identifies the status and plant-specific implementation of each TMI Action Plan item. Appendix C not only discusses the status of the USIs but also satisfies the guidelines provided by the Atomic Safety and Licensing Appeal Board in the River Bend case (ALAB-444, 6 NRC 760 (1977)).

If the Advisory Committee on Reactor Safeguards review of the SER requires additional response, the staff will issue a supplement to this SER. There are a number of ongoing licensing actions for Oyster Creek that are currently under staff review as noted in this SER. The staff has determined that these items do not require resolution before the issuance of an FTOL and should not delay the POL to FTOL conversion process. All of these items will be addressed as routine operating reactor licensing actions after the FTOL is issued.

In accordance with the provisions of the National Environmental Policy Act (NEPA) of 1969, the staff prepared the Draft and Final Environmental Statements that set forth the considerations related to the proposed POL to FTOL conversion. The Final Environmental Statement (FES) was issued in December 1974. Because the FES was issued a number of years ago, the staff performed an environmental evaluation to determine if an FES supplement was necessary. The environmental evaluation issued on April 10, 1986 (letter from J. Zwolinski, NRC), concluded that an FES supplement is not necessary.

The NRC Project Manager assigned to the FTOL review for Oyster Creek is Mr. Alexander W. Dromerick. Mr. Dromerick may be contacted by calling (301) 492-1301 or by writing to

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Division of Projects I/II
Washington, DC 20555

## 1.2 Description of Plant

The Oyster Creek Nuclear Generating Station, located in Ocean County, New Jersey, is a boiling-water reactor designed by General Electric. The licensees are GPU Nuclear Corporation and Jersey Central Power & Light Company (JCP&L). JCP&L, hereinafter referred to as the licensee, filed the application for a construction permit and operating license on March 24, 1964. The construction permit was issued on December 15, 1964. The initial submittal of the Final Safety Analysis Report was filed on January 25, 1967, and the initial provisional operating license was issued on April 9, 1969. In March 1972, the licensee applied for a full-term operating license. The licensed thermal power rating currently is 1930 megawatts-thermal (MWt).

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*Availability of all material cited is given on the inside front cover of this report.

The Oyster Creek primary coolant system consists of the reactor vessel, recirculation system, main steam system, and isolation condenser. The recirculation and isolation condenser systems are shown in Figure 1.1. The reactor is a single-cycle, forced-circulation boiling-water reactor producing steam for direct use in the steam turbine. The reactor vessel contains internal components, which include the necessary equipment for separating steam and water flow paths.

The recirculation system provides for forced flow through the reactor core to facilitate heat removal capability. Water that is separated from the steam in the reactor vessel and mixes with water provided by the feedwater system is drawn from outside the core, passes through the recirculation pumps, and reenters the reactor vessel below the core. The water then flows upward through the core where boiling produces a steam-water mixture.

The main steam system directs the steam generated in the reactor vessel to the turbine generator for conversion to electrical power. The steam-water mixture travels from the reactor core, through the steam-separating equipment into the main steamlines. The steam then passes through the main steamlines to the turbine. Included in the main steam system are the relief and safety valves, which provide overpressure protection for the reactor vessel and associated piping systems. The relief valves are also designed to rapidly depressurize the reactor vessel so that the emergency cooling systems will function. The reactor relief valves are located upstream of the first isolation valve and discharge directly to the pressure-suppression pool; the safety valves are located on the steamlines inside the primary containment and discharge to the drywell atmosphere.

The isolation condenser system, which consists of two condensers, will provide reactor core cooling if the reactor should become isolated from the main condenser because of closure of the main steam isolation valves. The isolation condenser operates by natural circulation. During operation steam flows from the reactor, condenses in the tubes of the isolation condenser, and flows back to the reactor by gravity.

The containment systems provide a multibarrier pressure-suppression containment composed of a primary containment, a Mark I pressure-suppression system, and a secondary containment, the reactor building.

The primary containment system is designed (1) to provide a barrier that will control the release of fission products to the secondary containment and (2) to rapidly reduce the pressure in the containment resulting from a loss-of-coolant accident. The system consists of a drywell, which houses the reactor vessel and recirculation loops; the pressure-suppression pool, which contains the large volume of water used to condense the accident steam release; and the connecting vent systems. The drywell, which is in the shape of a light bulb and is constructed of steel plate, varies in diameter from 70 feet to 33 feet and is approximately 64 feet high. The pressure-suppression chamber is a steel pressure vessel in the shape of a torus with an inside diameter of 30 feet, a water volume of approximately 83,400 cubic feet, and an air volume of approximately 127,000 cubic feet.

The reactor building is designed to provide containment during reactor refueling and maintenance operations when the primary containment system is open.

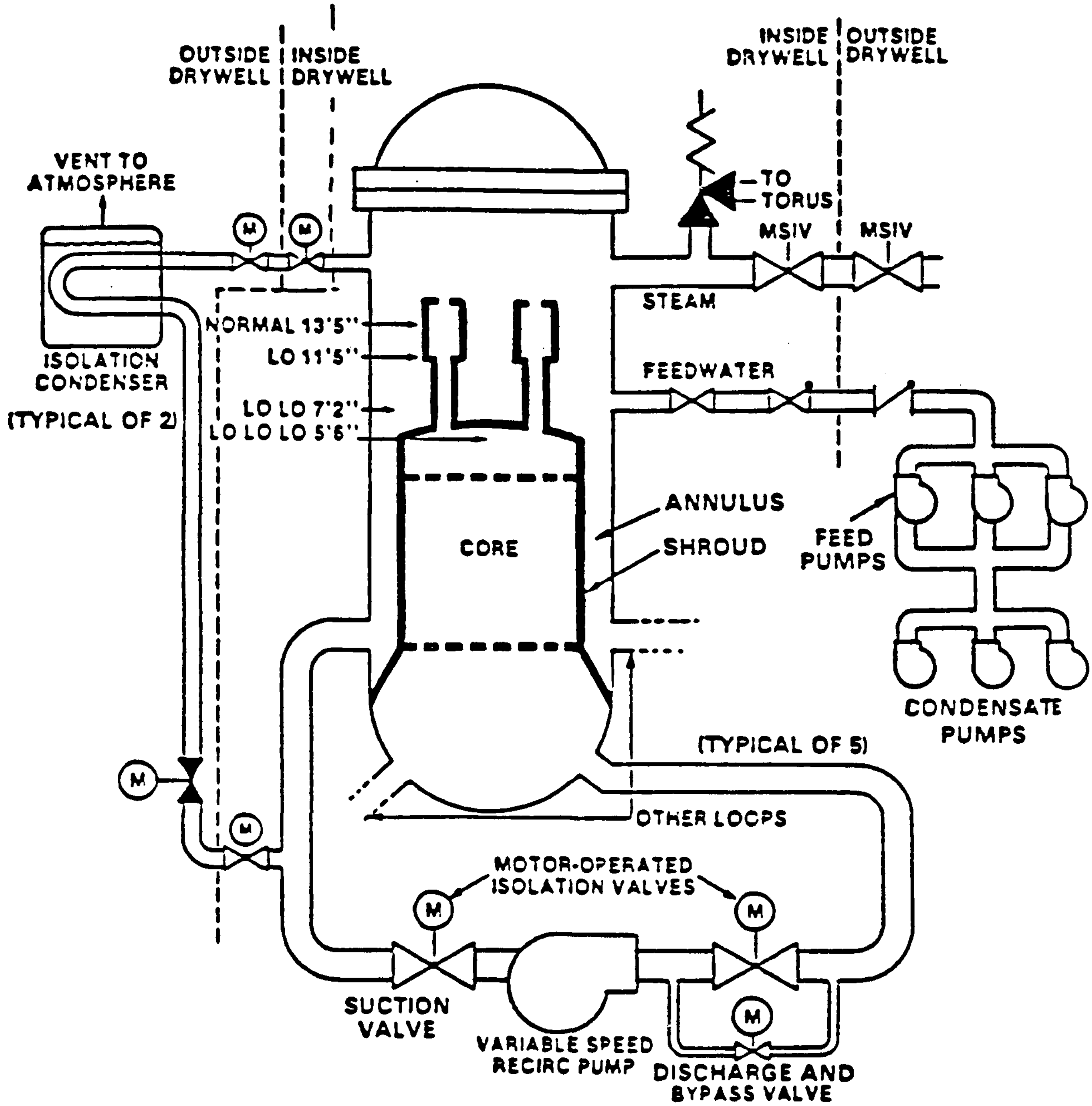


Figure 1.1 Schematic of recirculation, steam, and isolation condenser systems
Source: Oyster Creek Final Safety Analysis Report

The building will also provide secondary containment when the primary containment is required to be in service. The reactor building consists of the monolithic reinforced concrete floors and walls enclosing the nuclear reactor, primary containment, and reactor auxiliaries, and the building superstructure with sealed panel walls and precast concrete roof.

The plant consists of the following major buildings and structures:

(1) reactor building
(2) turbine building
(3) office building
(4) old radwaste building
(5) new radwaste and offgas building
(6) emergency diesel generator building
(7) intake and discharge structure
(8) ventilation stack
(9) storage tanks

Buildings and structures and the systems housed within are described in the Oyster Creek Final Safety Analysis Report.

## 1.3 Summary of Operating History and Experience

The Oyster Creek plant received a provisional operating license on April 9, 1969, achieved initial criticality on May 3, 1969, and began commercial operation on December 23, 1969. The plant operated at 1600 MWt until December 1970 when an increase to 1690 MWt was approved. In November 1971, a further increase to the present licensed thermal power of 1930 MWt was approved. The design electric rating is 650 megawatts-electric (MWe). The plant has operated in accordance with the stipulations of Provisional Operating License DPR-16.

### 1.3.1 Operating Experience Through 1981

To ensure that the plant's operating history, including plant transients, was appropriately evaluated and factored into the NRC staff evaluation, the staff requested that the Oak Ridge National Laboratory (ORNL) perform a detailed review. A copy of the ORNL report is included as Appendix F to the IPSAR.

Table 1.1 presents the Oyster Creek reactor availability and plant capacity factors for 1969 through 1981. From 1970 through 1981, the reactor availability factor at Oyster Creek averaged 74.4 percent and the unit capacity factor averaged 61.4 percent, both of which were above average for commercial nuclear power plants. As a result of startup tests, the values were low in 1969, but they were high from 1970 through 1979. The values for 1980 and 1981 were low because of extended refueling and maintenance outages. During these shutdowns, the licensee performed the 10-year American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code) hydrostatic test on the reactor vessel and coolant piping and made modifications stemming from the TMI-2 accident.

The licensee indicated that reportable events during this period (1969-1981) were primarily attributable to inherent equipment failures, accounting for 64 percent of all reported events; human error (including administrative,

Table 1.1 Oyster Creek availability and capacity factors

| Factor | 1969* | 1970 | 1971 | 1972 | 1973 | 1974 | 1975 | 1976 | 1977 | 1978 | 1979 | 1980 | 1981 | Average 1970-81 |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| Reactor availability | 33.5 | 80.8 | 82.1 | 82.4 | 74.2 | 72.2 | 75.5 | 80.0 | 71.2 | 75.5 | 87.0 | 43.2 | 63.3 | 74.4 |
| Unit availability | 18.3 | 77.0 | 80.4 | 81.3 | 73.1 | 70.4 | 73.3 | 79.3 | 70.1 | 74.3 | 85.9 | 41.7 | 59.8 | 72.3 |
| Unit capacity (MDC) | 9.3 | 63.6 | 70.4 | 80.0 | 66.0 | 67.6 | 57.9 | 70.9 | 59.8 | 67.1 | 84.0 | 35.9 | 48.4 | 65.7 |
| Unit capacity (DER) | 8.8 | 60.7 | 67.2 | 76.3 | 63.0 | 64.5 | 55.2 | 67.6 | 57.0 | 64.0 | 80.1 | 34.3 | 46.2 | 61.4 |

*From initial criticality.

Note: MDC = maximum dependable capacity (620 MWe); DER = design electrical rating (650 MWe).

design, fabrication, installation, maintenance, and operator error) accounted for 34 percent of reported events; and other causes, such as environmental conditions, accounted for the remaining 2 percent. The licensee identified no apparent trend in the causes of reported events for this period.

The licensee indicated that recurring valve problems, particularly with main steam isolation valves, including bent valve stems, packing leaks, and sticking pilot valves, arose during the 1969 to 1974 period. The licensee corrected these problems by equipment modification.

A variety of problems were also experienced with torus-to-reactor-building and torus-to-drywell vacuum breakers. An enforcement conference was held with licensee's management on May 4, 1982, to discuss NRC's concerns pertaining to violations related to the inoperability of the reactor-building-to-suppression-chamber vacuum breakers and isolation condenser isolation valves. These violations were the result of inadequate management controls over maintenance testing and surveillance activities.

Reactor vessel cracks were noted three times throughout the history of Oyster Creek. In 1974, an inservice inspection revealed cracking in reactor head cladding. However, no cracks propagated into the reactor vessel base material. Later in 1974, a small leak was noted in a field weld between the incore housing and the vessel lower head. Since its repair, no further cracking has been noted. Condenser tube leakage problems began in 1970. Through 1975, recurring power reductions were necessary to repair or plug leaking tubes. During a shutdown in the first part of 1976, condensers were retubed using welded titanium tubing. With the exception of a limited number of vibration-induced tube failures, these titanium tubes have functioned satisfactorily.

The licensee attributed much of the human error reported for the period 1969 to 1981 to outdated or inadequate procedures.

During the period November 1980 to October 1981, an emergency preparedness appraisal identified the need to (1) upgrade the emergency support facilities, (2) improve the capabilities for postaccident coolant and containment atmosphere sampling, and (3) upgrade emergency response training and retraining.

The licensee has committed to increased staffing and management reorganization to improve the overall quality and control of maintenance, surveillance, and modification/construction activities.

### 1.3.2 Operating Experience Since January 1, 1982

Oyster Creek capacity factors dropped during the 1982 calendar year to 35 percent maximum dependable capacity (MDC) net. The year began with the plant in a forced outage because of isolation condenser isolation valve stem leakage. After delays in restarting because of control rod drive hydraulic pump problems, emergency diesel generator air cooler leaks, and refueling cycle surveillances, the plant was returned to power, which was limited to 67 percent because one of three condensate pumps was not available. In May 1982, Oyster Creek experienced a 4-day forced shutdown because of a leak in a steam reheater manway cover. During the last half of the year, power was limited by available core reactivity as a refueling and modification outage, scheduled for early 1983, approached.

From February 1983 to October 1984, Oyster Creek underwent an extensive outage for refueling and plant modification. The licensee indicated that approximately 11,500 corrective and preventive maintenance tasks and modifications were performed. Major tasks involved upgrade of the torus, overhaul of the turbine generator, and improvements to the control room. Summaries issued by the licensee list several other accomplishments during this outage. Operational data for 1983 and 1984 reflect this outage.

In 1985, with no major outage, the Oyster Creek capacity factor, 69 percent, returned to its pre-1982 range, above average for nuclear power plants.

Oyster Creek began 1986 with a high capacity factor for the first quarter of the year. A refueling, maintenance, and modification outage began on April 12 and ended with restart on December 21.

In February 1987, Oyster Creek was taken out of service for 25 days to repair power range monitors in the reactor vessel. The plant was shut down again beginning in April for 22 days to replace one of the five acoustic monitors on the steam pressure electromatic relief valves. The plant was removed from service on July 30 for 6 days to repair an air manifold on one of the four main steam isolation valves. The plant had operated at full power for 75 consecutive days. In August, the NRC imposed a fine for an April violation of Technical Specifications and operating procedures involving improper operation of two vacuum breaker valves. The valves had been held open for about 3 hours during a plant shutdown. Despite outages, Oyster Creek managed to achieve a 57-percent capacity factor (MDC net) for 1987. On September 11, a day after the plant was taken out of service, the licensee reported to the NRC that a violation of Safety Limit 2.1.E of the Technical Specifications had occurred in that fewer than two sets of recirculation loop valves were fully open for a short period of time as required by the limit. During the event, a portion of a control room alarm paper tape was destroyed following the safety limit violation. The five-person control room staff was relieved of license-related duties pending an investigation. Three were later reinstated. The NRC authorized restart after the licensee made a number of corrective actions. The plant resumed generating electricity on November 24. By License Amendment 135, December 30, 1989, the recirculation loop availability requirement that had been violated was changed from a safety limit to a limiting condition for operation (LCO) in the Oyster Creek Technical Specifications, to effect conformance with the definition of safety limit and LCO given in 10 CFR 50.36(c) and present staff practice.

Oyster Creek was operating at full power in the beginning of 1988. The plant returned to service in November 1987 and operated 229 consecutive days until it was removed from service on July 9 to repair a main steamline valve. Following these repairs, the plant returned to service in August. Oyster Creek began its Cycle 12 refueling and maintenance outage on September 30, 1988. The capacity factor for the plant in 1988, including the refueling outage time, was 65 percent.

After repairing two primary system weld leaks that had been discovered in February 1989 during hydraulic testing, the licensee restarted the plant from the Cycle 12 refueling outage on March 29, 1989. The plant was shut down again in April when the licensee discovered a nonisolable leak in a core spray system weld. When repairs were complete, the plant was returned to service, but on May 18 it experienced a reactor scram/turbine trip after a generator overexcitation alarm. Three days later the plant was restarted and ran until June 25, when one of the station's two main transformers failed. The plant was returned to about 50-percent power, but was shut down in July when the other transformer failed. The licensee is investigating the cause of the transformer failures. In the meantime a replacement transformer was located and installed, permitting operation up to 75-percent power. Oyster Creek was restarted on July 19, 1989.

Table 1.2 provides statistical operational summaries for the years 1982 through 1988.

Table 1.2 Oyster Creek statistical operational summaries*

| Operational factors | 1982 | 1983 | 1984 | 1985 | 1985 | 1987** | 1988*** |
|---|---|---|---|---|---|---|---|
| Hours reactor was critical | 5,637.9 | 1,009.6 | 1,700 | 6,818.5 | 2,389.1 | 5,619.9 | 5,789.0 |
| Reactor reserve shutdown hours | 0.0 | 0.0 | 1.5 | 289.8 | 448.5 | 0.0 | 0.0 |
| Hours generator on line | 5,475.8 | 1,007.8 | 849.9 | 6,521.4 | 2,310.9 | 5,422.9 | 5,750.6 |
| Unit reserve shutdown hours | 0.0 | 0.0 | 2.7 | 1,305.9 | 452.8 | 0.0 | 0.0 |
| Gross thermal energy (MWh) | 6,787,700 | 922,531 | 1,037,600 | 11,615,400 | 4,119,004 | 9,691,404 | 10,873,100 |
| Gross electrical energy (MWh) | 2,126,300 | 244,630 | 326,090 | 3,907,690 | 1,377,560 | 3,250,109 | 3,685,830 |
| Net electrical energy (MWh) | 2,013,090 | 205,155 | 278,777 | 3,746,033 | 1,301,476 | 3,110,919 | 3,538,872 |
| Unit service factor | 62.5 | 11.5 | 9.6 | 74.4 | 26.4 | 61.9 | 65.5 |
| Unit availability factor | 62.5 | 11.5 | 9.6 | 89.4 | 31.5 | 61.9 | 65.5 |
| Unit capacity factor (MDC net) | 37.1 | 3.8 | 5.1 | 69.0 | 24.0 | 57.3 | 65.0 |
| Unit capacity factor (DER net) | 35.4 | 3.6 | 4.9 | 65.8 | 22.9 | 54.6 | 62 |
| Unit forced outage rate | 37.5 | 0.0 | 36.3 | 18.8 | 8.1 | 27.3 | 12.5 |
| Forced outage hours | 3,284.2 | 0.0 | 479.9 | 1,508.5 | 204.2 | 2,034.5 | 824.4 |

*From NUREG-0020.
**From GPUN letter to NRC dated January 15, 1988.
***From GPUN letter to NRC dated January 13, 1989.

Note: MWh = megawatt-hour; MDC = maximum dependable capacity (620 MWe); DER = design electrical rating (650 MWe).

## 1.4 Plant Modifications

As a result of operating experience at this and other operating boiling-water reactors, various modifications have been or are in the process of being made to the plant. Some of the more important are the following:

(1) Fuel Pool Capacity

Spent-fuel storage at Oyster Creek has been gradually increased to facilitate future refueling outages. New high-density fuel racks were installed to ensure sufficient storage capacity. The number of storage locations was increased by POL Amendment 76 from 1400 to 2600 fuel assemblies, which provides sufficient capacity for storage of fuel discharged until 1994. Fuel storage is discussed in Section 9.1.

(2) As Low As Is Reasonably Achievable (ALARA) Modifications - Shielding

Certain areas of Oyster Creek present concerns in regard to exposure that cannot be alleviated by flushing or other normal decontamination procedures. Therefore, "high rad" areas, on a case-by-case basis, have been provided with shielding to maintain radiation exposure levels for plant personnel ALARA. ALARA and related operational considerations are discussed in Sections 11, 12, and 13.1.

(3) Onsite Power Reliability

The two emergency diesel generators were determined to be overloaded above their peak rating. Modifications were incorporated to replace existing breakers on various motor control centers with breakers that would trip on bus undervoltage for nonessential loads. The modifications will result in a substantial reduction of load on the two emergency diesel generators and bring their loading within their maximum rated load (peak) of 2750 kilowatts. Sections 8.3.2 and 9.3.1 contain additional discussions of diesel generators and diesel generator loading.

(4) Radioactive Waste Management

To comply with the guidelines and Appendix I of 10 CFR Part 50, major changes have been incorporated, including a new augmented offgas system housed in a new offgas building and a new liquid/solids radwaste system also housed in a new building. These systems are discussed in Sections 11.1 and 11.2, respectively.

(5) Plant Computer and Emergency Response Facility Data System

A plant computer system has been provided in the new site emergency building with data acquisition and processing equipment capable of monitoring (and trending) as well as displaying plant status and parameters on a cathode ray tube display.

(6) Torus Support Structure

The following modifications were made to the shell to account for hydrodynamic loads on the shell caused by a loss-of-coolant accident:

(a) A mid-bay saddle support was installed for each of the 20 bays.

(b) The lower half of the torus shell was reinforced by eight external straps on each bay.

(c) A ring girder was added at each intersection between the two adjacent bays.

These modifications contributed to the resolution of issues related to BWR Mark I pressure-supprssion containments as discussed in Section 6.2.1.

(7) Chemistry Laboratory

A new chemistry laboratory facility has been provided for the performance of various chemical analyses required for plant operations and environmental and radiological programs.

(8) Postaccident Sampling System (PASS)

A new postaccident sampling station has been provided in the count room in the office building to provide the plant with the means for sampling reactor coolant and containment atmosphere, in accordance with NUREG-0578, "TMI-2 Lessons Learned Task Force Status Report and Short-Term Recommendations," and NUREG-0737, "Clarification of TMI Action Plan Requirements." This system was found acceptable in a staff SER dated August 29, 1984. Technical Specifications governing postaccident sampling instrumentation were issued in POL Amendment 94, completing implementation of the system.

(9) Relief and Safety Valve Position Indication

In mid-1980, the licensee installed the Babcock & Wilcox Company valve monitoring system (VMS) on the relief and safety valves. The VMS is an acoustic-based system that utilitizes accelerometers mounted on the valve to detect the noise caused by flow through the valve. The system can distinguish between normal background noise and that at the much higher level when the valve is open and indicates accordingly. It provides the operator with accurate and unambiguous indication of valve position (open or closed) so that appropriate operator actions can be taken. Inspection Report 50-219/89-08, April 26, 1989, stated that the licensee's previously accepted design had been adequately implemented to resolve NUREG-0737, Item II.D.3.

(10) Radioactive Gaseous Effluent Monitoring System (RAGEMS) (Stack)

A new stack monitoring system, RAGEMS, has been installed to meet the accident-monitoring requirements of NUREG-0578 and NUREG-0737. It provides analyses of noble gases, particulates, and iodine. RAGEMS is computerized and automated to reduce exposure, and is housed in a new dedicated building west of the stack. This system is discussed in Section 11.3.

(11) Radioactive Gaseous Effluent Monitoring System for Turbine Building and Turbine Building Heating, Ventilation, Air Conditioning System

All the areas that could potentially release radioactive contamination that did not tie into the stack are tied together to be exhausted through the turbine building ground release stack, which is monitored by its own RAGEMS and controlled by the stack RAGEMS computer. RAGEMS is discussed in Section 11.3.

(12) New Cable Spreading Room

The former mechanics/equipment room was converted into a second cable spreading room. Without this modification, it would be impossible to perform any modification that requires access at the bottom of the control room.

(13) Masonry Walls

NRC Office of Inspection and Enforcement Bulletin 80-11 required the licensee to evaluate masonry walls and to make the necessary changes. Modifications to walls in the proximity of safety-related equipment were made during the Cycle 12 refueling outage by adding steel or unistrut to the wall boundaries and anchoring it to existing concrete. Portions of block walls that were not structurally required in the proximity of safety-related equipment were removed to prevent possible missile hazards. These modifications had been approved by the staff in a letter dated December 23, 1985.

(14) Torus Temperature Instrumentation

This modification consisted of installing 20 temperature sensors in the torus for local and bulk temperature readings.

(15) Seismic Qualification of Spent Fuel Pool Cooling System (SFPCS)

The SFPCS was classified as a seismic Category I system. The following modifications ensure the operational integrity of the SFPCS during and after a seismic event:

(a) Addition of new supports to the SFPCS piping and the modification of existing supports.

(b) Addition of a gate valve to the SFPCS for bypassing the radwaste facility in case of a seismic event. This is recommended in lieu of major modifications of piping supports in the pipe tunnel and old radwaste building that would be required to upgrade the seismic classification.

(c) Seismic qualification of existing SFPCS valves to ensure structural integrity and operability following a seismic event.

See Section 9.1.1 for additional discussion of the SFPCS.

## 1.5 Status Summary for Full-Term Operating License Items

Table 1.3 provides a status summary of SEP items considered in the integrated assessment reported in IPSAR Section 4. It also provides sections in the IPSAR and its supplement where the items are discussed. Other SEP items were identified in the IPSAR as those for which the plant meets current criteria or is acceptable on another defined basis. Some of these other SEP items, which are not listed in Table 1.3, are included in this report for their technical, historical, or descriptive value and/or to retain general adherence to the conventions identified in Regulatory Guide 1.70, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants - LWR Edition." The entire discussion in Section 2 of this report is an example of such inclusion.

Table 1.4 identifies unresolved items (open issues) and provides the sections in this report where they are discussed. Many of the items identified in Table 1.4 are SEP topics.

SEP did not identify any items that would preclude the continued operation of nuclear power plants. SEP items were to be resolved on a schedule mutually agreeable to the staff and the licensees. The staff has found the progress to date of the resolution of these issues for Oyster Creek to be satisfactory.

One of the items identified in Table 1.4 is associated with a Three Mile Island Action Plan requirement. This item was not identified as an issue that would preclude the continued operation of Oyster Creek. The staff has found the progress to date of the resolution of this issue to be acceptable.

The remaining item in Table 1.4 deals with NRC Bulletins 79-02 and 79-14. As discussed in Section 3.9.1, for this item the staff has required a resolution of known acceptability and timeliness. This item should not preclude the continued operation of Oyster Creek.

Since the staff has concluded that none of the items in Table 1.4 should preclude the continued operation of Oyster Creek, these items should not be an impediment to the issuance of a full-term operating license for Oyster Creek.

Table 1.3 Integrated assessment summary

| SEP topic no. | IPSAR section | Title | IPSAR requirements | IPSAR supplement section | SER section/ status |
|---|---|---|---|---|---|
| II-3.B, II-3.B.1, II-3.C | 4.1(1) | Condensate Water Pumps | See IPSAR Item 4.6.4. | 2.1.1 | 3.4.1.1/Resolved |
| | 4.1(2) | Flooding Level Procedures | None | -- | 3.4.1.2/Resolved |
| | 4.1(3) | Canal Water Level Instrumentation | Install water level instrumentation in intake canal. | 2.1.2 | 3.4.1.3/Resolved |
| | 4.1(4) | Isolation Condenser Flooding | Demonstrate minimum quantity of water maintained in condensate storage tank sufficient for long-term cooling and include minimum inventory in plant procedures. | 3.1.1<br>4.1.1 | 3.4.1.4/Resolved |
| | 4.1(5) | Low Water Level Shutdown | None | 3.1.2 | 3.4.1.5/Resolved |
| | 4.1(6) | Hurricane Flooding of Pumps | Revise emergency procedures to identify alternate water sources and flow paths should low elevation pumps be flooded. | 4.1.2 | 3.4.1.6/Resolved |
| | 4.1(7) | Protection During Internal Flooding | Evaluate consequences of offgas building flooding and confirm all other entrance levels above 23.5 feet. | 2.1.3 | 3.4.1.7/Resolved |

Table 1.3 (Continued)

| SEP topic no. | IPSAR section | Title | IPSAR requirements | IPSAR supplement section | SER section/ status |
|---|---|---|---|---|---|
| II-3.B, II-3.B.1, II-3.C | 4.1(8) | Groundwater Elevation | See IPSAR Item 4.4(2). | 2.4.1 | 3.4.1.8/Resolved |
| | 4.1(9) | Roof Drains | Install scuppers in the reactor building and turbine building parapets. | 4.1.3 | 3.4.1.9/Resolved |
| III-1 | 4.2 | Classification of Structures, Components, and Systems | Evaluate design of specified components on a sampling basis, upgrade if necessary, and document classification in FSAR update. | 2.2 | 3.2/Resolved |
| III-2 | 4.3.1 | Reactor Building Steel Structure Above the Operating Floor | Analyze and identify any needed upgrading of reactor building upper steel structure for wind loads. | 2.3.1 | 3.3/Under review |
| | 4.3.2 | Ventilation Stack | Analyze and identify any needed upgrading of ventilation stack for wind loads. | 2.3.2 | 3.3/Resolved |
| | 4.3.3 | Effects of Failure of Nonseismic Category I Structures | Analyze turbine building capacity for wind loads, evaluate consequences of failure, and identify any needed upgrading. | 2.3.3 | 3.3/Resolved |
| | 4.3.4 | Components Not Enclosed in Qualified Structures | None | -- | 3.3/Resolved |

Table 1.3 (Continued)

| SEP topic no. | IPSAR section | Title | IPSAR requirements | IPSAR supplement section | SER section/ status |
|---|---|---|---|---|---|
| III-2 | 4.3.5 | Exterior Masonry Walls | None | -- | 3.3/Resolved |
| | 4.3.6 | Roof Decks | Provide analysis of reactor building roof. | 2.3.4 | 3.3/Resolved |
| | | | Analyze capacity of turbine building roof to withstand wind loads. | -- | 3.3/Resolved |
| | 4.3.7 | Intake Structure, Oil Tanks, and Diesel Generator Building | Analyze capacity to with-stand wind and tornado loads and upgrade, if necessary. | 2.3.5 | 3.3/Resolved |
| | 4.3.8 | Load Combinations | See IPSAR Item 4.12. | 2.3.6 | 3.3/Under review |
| | 4.3.9 | Soil and Foundation Capacities | None | -- | 3.3/Resolved |
| | 4.3 | Control Room/ Architectural Components | -- | 2.3.7/ 2.3.8 | 3.3/Under review/ under review |
| III-3.A | 4.4(1) | Hydrostatic Loads (Combination) | None | -- | 3.4.2/Resolved |

Table 1.3 (Continued)

| SEP topic no. | IPSAR section | Title | IPSAR requirements | IPSAR supplement section | SER section/ status |
|---|---|---|---|---|---|
| III.3.A | 4.4(2) | Hydrostatic Loads (Short-Duration) | Evaluate short-duration hydrostatic loads on and flotation potential of structures essential to safe shutdown in conjunction with flooding emergency procedures (IPSAR Item 4.1(6)). | 2.4.1 | 3.4.2/Resolved |
| | 4.4(3) | Below-Grade Penetration Flooding | None | -- | 3.4.2/Resolved |
| III-3.C | 4.5.1 | Intake and Discharge Canals | None | -- | 3.4.3/Resolved |
| | 4.5.2 | Intake Structure Trash Racks and Intake Screens | Formalize existing inspection practice as part of shift turnover or inservice inspection (ISI) procedures until water level modification is complete (IPSAR Item 4.1(3)). | 4.2.1 | 3.4.3/Resolved |
| | 4.5.3 | Roof Drains | See IPSAR Item 4.1(2). (Resolved) | -- | 3.4.3/Resolved |
| | 4.5.4 | Inspection Program | Develop and implement a formal inspection program for water control structures. | 4.2.2 | 3.4.3/Resolved |

Table 1.3 (Continued)

| SEP topic no. | IPSAR section | Title | IPSAR requirements | IPSAR supplement section | SER section/ status |
|---|---|---|---|---|---|
| III-4.A | 4.6.1 | Emergency Diesel Generators and Fuel Oil Day Tank | Analyze potential for and consequences of tornado-missile damage of the diesel generator building. | 2.5.1 | 3.5.1.1/Resolved |
| | 4.6.2 | Mechanical Equipment Access Area | Evaluate the potential for and consequences of tornado-missile impact in the reactor building access door region and identify any necessary corrective actions. | 2.5.2 | 3.5.1.2/Under review |
| | 4.6.3 | Control Room, Reactor Building, and Turbine Building Heating, Ventilating, and Air Conditioning (HVAC) Systems | None | -- | Resolved |
| | 4.6.4 | Condensate Storage Tank, Torus Water Storage Tank, and Service Water and Emergency Service Water Pumps | Provide protection for sufficient systems and components to ensure a safe shutdown in the event of damage from tornado missiles. | 2.5.3 | 3.5.1.3/Resolved |

Table 1.3 (Continued)

| SEP topic no. | IPSAR section | Title | IPSAR requirements | IPSAR supplement section | SER section/ status |
|---|---|---|---|---|---|
| III-4.B | 4.7 | Turbine Missiles | Inspect turbine and propose inspection frequency based on results. | 2.6 | **3.5.1.4/Resolved** |
| | | | Justify monitoring program for main steam and reheat control valves. | 2.6 | **Resolved** |
| III-4.D | 4.8.1 | Truck Explosion | None | -- | **Resolved** |
| | 4.8.2 | Aircraft Hazards | Evaluate potential for or consequences of aircraft impact. | 2.7.1 | **Resolved** |
| III-5.A | 4.9(1) | Cascading Pipe Breaks | See IPSAR Item 4.16. | -- | **3.6.1/Resolved** |
| | 4.9(2) | Jet Impingement Effects | None | -- | **3.6.1/Resolved** |
| | 4.9(3) | Drywell Penetration | None | -- | **3.6.1/Resolved** |
| III-5.B | 4.10(1) | LOCA Outside Containment | None | -- | **3.6.2/Resolved** |
| | 4.10(2) | Emergency Condenser Isolation | Evaluate and identify any necessary modifications to provide leakage detection to ensure that flaws would be detected before pipe break occurs. | 2.8.1 | **3.6.2/Submit information for staff review** |

Table 1.3 (Continued)

| SEP topic no. | IPSAR section | Title | IPSAR requirements | IPSAR supplement section | SER section/ status |
|---|---|---|---|---|---|
| III-6 | 4.11(1) | Piping Systems | Analyze on a sampling basis and verify adequacy of support designs for the seismic resistance of specified piping systems. | 2.9.1 | 3.7.1/Resolved |
| | 4.11(2) | Mechanical Equipment | Demonstrate that the control rod drive system and vessel internals have sufficient capacity to resist a safe shutdown earthquake or take corrective action. | 2.9.2 | 3.7.1/Resolved |
| | 4.11(3) | Electrical Equipment | Reevaluate 4160-V switch gear panel anchorage and demonstrate, on a sampling basis, adequacy of electrical panel supports. | 2.9.3 | 3.7.1/Resolved |
| | 4.11(4) | Ability of Safety-Related Electrical Equipment To Function | None | -- | 3.7.1/Resolved |
| | 4.11(5) | Qualification of Cable Trays | Provide plan to implement results of SEP Owners Group Program on a plant-specific basis. | 2.9.4 | 3.7.1/Resolved |

Table 1.3 (Continued)

| SEP topic no. | IPSAR section | Title | IPSAR requirements | IPSAR supplement section | SER section/ status |
|---|---|---|---|---|---|
| III-7.B | 4.12 | Design Codes, Design Criteria, Load Combinations, and Reactor Cavity Design Criteria | Evaluate adequacy of original design criteria on a sampling basis for specified structural elements. | 2.10 | 3.8.1/Under review |
| III-8.A | 4.13 | Loose-Parts Monitoring and Core Barrel Vibration Monitoring | None | -- | 4.3/Resolved |
| III-10.A | 4.14(1) | Thermal-Overload Bypass | Evaluate thermal-overload bypasses for engineered safety features (ESF) valves. | 2.11.1 | Resolved |
| | 4.14(2) | Magnetic Trip Breakers | None | -- | Resolved |
| IV-2 | 4.15 | Reactivity Control Systems, Including Functional Design and Protection Against Single Failures | None | -- | 4.5/Resolved |
| V-5 | 4.16.1 | Leakage Detection Systems | Evaluate reliability of leakage detection systems and evaluate sensitivity in conjunction with Topic III-5.A analysis. | 2.12.1 | 5.2.1/Resolved |

Table 1.3 (Continued)

| SEP topic no. | IPSAR section | Title | IPSAR requirements | IPSAR supplement section | SER section/ status |
|---|---|---|---|---|---|
| V-5 | 4.16.2 | Operability Requirements | Identify action for loss of leakage detection in Technical Specifications and include testing in procedures. | 3.2<br>4.3.1 | 5.2.2/Resolved |
| | 4.16.3 | Intersystem Leakage | None | -- | Resolved |
| | 4.16.4 | Reactor Coolant Inventory Balances | None | -- | Resolved |
| V-6 | 4.17 | Reactor Vessel Integrity | Submit a plan for the material surveillance capsules. | 3.3 | 5.3/Resolved |
| V-10.B | 4.18 | Residual Heat Removal System Reliability | Review and upgrade, if necessary, shutdown procedures to specify alternate sources of water for primary and secondary makeup, with particular attention to external events. | 4.4 | 5.4.2/Resolved |
| V-11.A | 4.19 | Requirements for Isolation of High- and Low-Pressure Systems | Demonstrate relief capacity and acceptable consequences, or identify corrective action to protect reactor water cleanup system. | 2.13 | 5.4.3/Resolved |

Table 1.3 (Continued)

| SEP topic no. | IPSAR section | Title | IPSAR requirements | IPSAR supplement section | SER section/ status |
|---|---|---|---|---|---|
| V-12.A | 4.20 | Water Purity of BWR Primary Coolant | Implement proposed procedure and modify Technical Specifications to be consistent. | 3.4<br>4.5 | 5.5/Resolved |
| VI-1 | 4.21.1 | Organic Materials | Inspect and repair, if necessary, drywell coatings and recoat the torus. | 4.6.1 | 6.1.1/Resolved |
| | 4.21.2 | Postaccident Chemistry | None | -- | 6.1.2/Resolved |
| VI-4 | 4.22.1 | Locked-Closed Valves | Provide physical locking devices to ensure valves are not inadvertently opened. | 4.7.1 | 6.2.2/Resolved |
| | 4.22.2 | Remote Manual Valves | Evaluate leakage detection provisions and, if necessary, relocate the operating station for isolation valves in the containment spray and core spray systems. | 2.14.1 | 6.2.2/Resolved |
| | 4.22.3 | Valve Location | None | -- | 6.2.2/Resolved |
| | 4.22.4 | Instrument Lines | None | -- | 6.2.2/Resolved |
| | 4.22.5 | Valve Location and Type | None | -- | 6.2.2/Resolved |
| | 4.22.6 | Administrative Controls | None | -- | 6.2.2/Resolved |

Table 1.3 (Continued)

| SEP topic no. | IPSAR section | Title | IPSAR requirements | IPSAR supplement section | SER section/ status |
|---|---|---|---|---|---|
| VI-7.A.3 | 4.23 | Emergency Core Cooling System Actuation System | Include emergency condenser logic testing in the Technical Specifications. | 3.5 | 6.3.1/Resolved |
| VI-7.A.4 | 4.24 | Core Spray Nozzle Effectiveness | None | -- | 6.3.2.2/Resolved |
| VI-7.C.1 | 4.25(1) | AC Automatic Bus Transfers | Evaluate the existing automatic bus transfers and identify corrective actions to ensure faulted loads would not be transferred. | 4.8.1 | 8.5/Resolved |
| | 4.25(2) | DC Automatic Bus Transfers | None | -- | 8.5/Resolved |
| VI-10.A | 4.26.1 | Response-Time Testing | None | -- | -- |
| | 4.26.2 | Instrumentation for Reactor Trip System (RTS) Testing | Verify all safety logic channels tied to the reactor mode switch are tested by procedure. | 3.6.1 | 7.1.1/Resolved |
| | | | Include logic channel testing in Technical Specifications. | 3.6.1 | 7.1.1/Resolved |
| | 4.26.3 | Dual-Channel Testing | None | -- | 7.1.1/Resolved |

Table 1.3 (Continued)

| SEP topic no. | IPSAR section | Title | IPSAR requirements | IPSAR supplement section | SER section/ status |
|---|---|---|---|---|---|
| VII.1.A | 4.27(1) | Flux Monitoring Isolation | Perform failure mode and effects analysis to determine whether isolation devices are required and identify any needed upgrading. | 2.15.1 | 7.1.2.1/Resolved |
| | 4.27(2) | Reactor Protection System (RPS) Protective Trip | Install Class 1E protection at the RPS power supply and RPS interface. | 4.9.1 | 7.1.2.2/Resolved |
| VII-1.B | 4.28 | Trip Uncertainty and Setpoint Analysis Review of Operating Data Base | Install analog trip system. | 2.16 | 7.1.3/Submit information for staff review |
| VII-2 | 4.29 | Engineered Safety Features System Control Logic and Design | See IPSAR Item 4.14(1). | 2.11.1 | 7.2/Resolved |
| VII-3 | 4.30 | Systems Required for Safe Shutdown | Provide minimum inventory for condensate storage tank as a water source for flooding events (IPSAR Item 4.1(4)) and identify non-ESF equipment in cooldown procedures (IPSAR Item 4.18). | 4.10 | 7.3/Resolved |

Table 1.3 (Continued)

| SEP topic no. | IPSAR section | Title | IPSAR requirements | IPSAR supplement section | SER section/ status |
|---|---|---|---|---|---|
| VIII-2 | 4.31(1) | Diesel Generator Annunciators | Modify annuniciators to conform to IEEE Std. 279-1971. | 4.11.1 | 8.3.2.1/Resolved |
| | 4.31(2) | Diesel Generator Trip Bypass | Evaluate bypass of two trips (voltage-ampere reactive and reverse power) during accident conditions. | 4.11.2 | 8.3.2.2/Resolved |
| VIII-3.B | 4.32 | DC Power System Bus Voltage Monitoring and Annunciation | Schedule installation of specified battery status alarms. | 2.17<br>4.12 | 8.3.3.2/Resolved |
| VIII-4 | 4.33 | Electrical Penetrations of Reactor Containment | None | -- | 8.4/Resolved |
| IX-5 | 4.34(1) | Restoration of Ventilation | Evaluate and revise, if necessary, the loss-of-offsite-power procedures to ensure that restoration of ventilation systems will not overload the diesels. | 3.7.1<br>4.13.1 | 9.3.1/Resolved |
| | 4.34(2) | Reactor Building Ventilation | None | -- | Resolved |

Table 1.3 (Continued)

| SEP topic no. | IPSAR section | Title | IPSAR requirements | IPSAR supplement section | SER section/ status |
|---|---|---|---|---|---|
| IX-5 | 4.34(3) | Core Spray and Containment Spray Pump Ventilation | Demonstrate subject pumps can operate with a loss of ventilation, or identify corrective action, as necessary. | 2.18.1 | 9.3.2/Resolved |
| | 4.34(4) | Battery, Motor Generator, and and Switchgear Room Ventilation | Evaluate effects of loss of ventilation to the subject rooms and identify any needed upgrading. | 2.18.2 | 9.3.3/Resolved |
| XV-1 | 4.35 | Decrease in Feedwater Temperature, Increase in Feedwater Flow, and Increase in Steam Flow and Inadvertent Opening of a Steam Generator Relief or Safety Valve | None | -- | 15.1/Resolved |
| XV-16 | 4.36 | Radiological Consequences of Failure of Small Lines Carrying Primary Coolant Outside Containment | Implement BWR Standard Technical Specification limits for primary coolant activity. | 3.8 | 15.1/Resolved |

Table 1.3 (Continued)

| SEP topic no. | IPSAR section | Title | IPSAR requirements | IPSAR supplement section | SER section/ status |
|---|---|---|---|---|---|
| XV-18 | 4.37 | Radiological Consequences of a Main Steam Line Failure Outside Containment | See IPSAR Item 4.36. | -- | 15.1/Resolved |
| XV-19 | 4.38 | Loss-of-Coolant Accidents Resulting From Spectrum of Postulated Pipe Breaks Within the Reactor Coolant Pressure Boundary | Develop and implement a preventive maintenance program for the main steam isolation valves, or justify existing maintenance based on operating experience. | 3.9 | 15.1/Resolved |
| | | | Submit results of evaluation including testing experience. | 3.9 | 15.1/Resolved |

Table 1.4 Unresolved items

| SEP topic no. | IPSAR item | Title/description | SER section |
|---|---|---|---|
| III-2 | | Wind and Tornado Loadings | |
| | 4.3.1 | Reactor Building Steel Structure Above the Operating Floor | 3.3 |
| | 4.3.8 | Load Combinations | 3.3 |
| | 4.3 | Control Room/Architectural Components | 3.3 |
| III-4.A | | Tornado Missiles | |
| | 4.6.2 | Mechanical Equipment Access Area | 3.5.1.2 |
| III-5.B | | Pipe Break Outside Containment | |
| | 4.10.2 | Emergency Condenser Isolation | 3.6.2 |
| III-7.B | 4.12 | Design Codes, Design Criteria, Load Combinations, and Reactor Cavity Design Criteria | 3.8.1 |
| VII-1.B | 4.28 | Trip Uncertainty and Setpoint Analysis Review of Operating Data Base | 7.1.3 |
| -- | -- | NRC IE Bulletins 79-02 and 79-14 (Seismic) | 3.9.1 |
| VI-5 | -- | Combustible Gas Control | 6.5 |
| -- | -- | Additional Accident-Monitoring Instrumentation and Generic Letter 83-36 | App. B, II.F.1 |

# 2 SITE CHARACTERISTICS

## 2.1 Geography and Demography

The Oyster Creek Nuclear Generating Station is located on the coastal pine barrens of New Jersey in Lacey and Ocean Townships, Ocean County. The station site is part of 1416 acres of land owned by Jersey Central Power & Light Company. The site is approximately 35 miles north-northeast of Atlantic City, New Jersey, and 45 miles east of Philadelphia, Pennsylvania. Approximately 9.5 miles north of the site are several small residential communities, Toms River, South Toms River, Beachwood, Pine Beach, Ocean Gate, Island Heights, and Gilford Park. The staff reviewed the licensee's exclusion area authority and control and the population distribution under SEP Topics II-1.A and II-1.B.

### 2.1.1 Exclusion Area Authority and Control (SEP Topic II-1.A)

All land areas, including mineral rights within the exclusion area, are owned by the licensee. Parts of the exclusion area are traversed by U.S. Route 9 and Central Railroad of New Jersey. Arrangements have been made with the New Jersey State Police and Lacey Township Police Department to control traffic on U.S. Route 9 in the event of a plant emergency as part of the Oyster Creek Emergency Plan. Similar arrangements had not been made with the railroad line to control traffic under emergency conditions; however, the need no longer exists since the railroad tracks have been removed.

A natural gas pipeline also traverses the exclusion area. There are no written agreements with New Jersey Natural Gas Company to ensure that the licensee has authority and control with respect to any construction, maintenance, or operational activities over that portion of the pipeline that would traverse the periphery of the exclusion area. However, since the pipeline just passes the edge of the exclusion area, the staff determined that the pipeline poses no significant hazard to the plant and the licensee does not need authority to control the pipeline.

The only waterway traversing the exclusion area is the Oyster Creek station intake and discharge canal. Station security measures are enforced to ensure unauthorized activity does not occur in this waterway.

In a letter dated February 4, 1982, the staff concluded that the licensee's exclusion area authority and control are acceptable.

The staff concludes that the licensee has the proper authority to determine all activities within the exclusion area, as required by 10 CFR Part 100.

### 2.1.2 Population Distribution (SEP Topic II-1.B)

As reviewed under SEP Topic II-1.B, the region surrounding the plant is characterized by flat terrain, sandy soils, and numerous freshwater and saltwater marshlands. Two barrier beaches, Seaside Peninsula and Long Beach Island, extend the length of the county providing extensive recreational opportunities on its beaches and bays. These attract a large transient seasonal population. The peak seasonal population (defined as the sum total of permanent and transient population groups on an average day during the peak summer season) within a 10-mile radius of the plant is expected to be 179,840. The permanent resident population within 10 miles is 66,815 (1980 census).

The nearest population centers with more than 25,000 residents are Dover Township, Gilford Park, and several smaller communities. The population in 1980 was approximately 64,445.

In a letter dated February 4, 1982, the staff concluded that the low population zone and population center distances specified for the Oyster Creek site are in conformance with 10 CFR Part 100.

The population data in the 1982 SER are based on 1980 census data. GPUN, in a letter dated January 16, 1990, provided the results of an updated (1987) population study. The updated results fall somewhere between the NRC data based on the 1980 census and data provided by the State of New Jersey, Bureau of Nuclear Engineering (BNE) (see Table 2.1). GPUN stated that it plans to use 1990 census data when they become available to update the evacuation time estimates (ETEs) in the emergency plan. The staff believes the GPUN commitment to update the population data base with 1990 census data is responsive to concerns about the current population distribution around the Oyster Creek site and should resolve the issue.

Table 2.1 Comparison of Oyster Creek population data

| | 10-mile radius | |
|---|---|---|
| Data | Permanent | Permanent and transient |
| NRC SER (1980 census data) | 66,815 | 179,840 |
| GPUN response (1987 data) | 105,159 | 181,001* |
| BNE data | 132,755 | 223,330 |

*Emergency planning at Oyster Creek is based on the combined permanent and transient population.

Regarding the concern that the Oyster Creek population densities exceed NRC siting criteria guidance, 10 CFR 100.3(b) states, under the definition of low population zone, "These guides do not specify a permissible population density or total population within this zone because the situation may vary from case to case." The NRC staff developed population density guidelines for use in evaluating applications for proposed sites for new reactor facilities, but these guidelines are not applicable to an operating plant. Regulatory Guide 4.7, "General Site Suitability Criteria for Nuclear Power Stations," Revision 1, 1975, states that if the projected population exceeds 500 per mile at the time of initial operation, averaged over any radial distance out to 30 miles, or

1000 per mile over the lifetime of the facility, special attention should be given to the consideration of alternative sites with lower population densities. From a regulatory standpoint, a site that meets the criteria of 10 CFR Part 100 is in conformance with NRC population requirements.

With respect to the nearest population center, 10 CFR 100.11 states that the population center distance must be at least one and one-third times the low population zone (LPZ) distance. The Oyster Creek LPZ is 0.75 mile. Thus, the population center distance, that is, the nearest boundary of a densely populated center with more than 25,000 residents, would have to come within 1.0 mile of the reactor before NRC siting criteria would be exceeded. The present population center is 9.5 miles, and it is unlikely that the population growth in the vicinity of the Oyster Creek site will challenge the 10 CFR Part 100 siting criteria.

In the ongoing implementation of the Oyster Creek emergency plan as discussed in Section 13.3, agencies responsible for emergency planning have expressed confidence in the adequacy of emergency planning for Oyster Creek. The staff believes the appropriate resolution of concerns regarding population and emergency planning is the commitment of GPUN to update the population distribution and resulting ETEs in the emergency plan on the basis of 1990 census data. The NRC, through its routine inspection program, will verify the resolution of this issue. Emergency planning is discussed further in Section 13.3.

## 2.2 Potential Hazards or Changes in Potential Hazards Due to Transportation, Institutional, Industrial, and Military Facilities (SEP Topic II-1.C)

The staff reviewed the potential hazards to safety-related structures, systems, and components resulting from nearby transportation, institutional, industrial, and military facilities under SEP Topic II-1.C.

Ocean County's industrial base is small, but diversified. Boat building and the manufacturing of marine equipment were once the dominant industrial activities, but today the industrial activity also includes chemical manufacturing, mining of ilmenite, quarrying of industrial sands, garment manufacturing, food processing, and production of concrete.

The nearest transportation route to the station is U.S. Route 9, which is located approximately 0.25 mile east of the reactor building. In 1981, Route 9 was not heavily used for shipping in the locality. There were no industries in close proximity to the plant site that were expected to use or store large amounts of explosive or hazardous material. Additionally, Route 9 is a local road with many traffic lights and low speed limits, especially where it passes through towns. Through traffic generally used the Garden State Parkway, a limited access toll road that runs parallel to Route 9. The parkway is about 1.25 miles west of the plant. The separation distance between the highway and the plant exceeds the minimum distance criteria given in Regulatory Guide 1.91 for truck-size shipments of explosive materials. Therefore, in a letter dated February 4, 1982, the staff concluded that the transportation of hazardous materials on U.S. Route 9 posed no significant hazard to the plant.

The staff has reviewed the truck traffic on U.S. Route 9 and finds that the frequency has changed significantly since the previous evaluation within the

SEP review. According to transportation data from the State of New Jersey, the gross large-truck traffic on Route 9, in the vicinity of Oyster Creek, is not insignificant.

For example, in 1988, the tractor trailer truck traffic was approximately 75,000 trucks per year, or about 206 trucks per day. Regulatory Guide 1.78 indicates that a transportation hazard evaluation should be made if nearby highway traffic involves 10 shipments of hazardous chemicals per year. Hence, if approximately 0.01 percent of the truck traffic were to involve hazardous chemicals, a hazards evaluation would need to be made.

Although the staff does not know the actual percentage of hazardous chemicals, if any, it is not possible to dismiss out of hand the possibility that more than 0.01 percent of the shipments involve hazardous materials. Hence, the licensee should review the nearby traffic on Route 9 in terms of the frequency of shipments of hazardous materials. The findings should be compared with the guidelines of Regulatory Guide 1.78. If the aggregate shipment frequency for all hazardous materials exceeds the guidelines, then the hazard to the Oyster Creek plant should be evaluated.

In conclusion, the staff finds that there is sufficient truck traffic on U.S. Route 9 so that an assessment of the frequency of hazardous-material shipments and, potentially, the level of risk associated with the shipments is warranted. By letter dated May 23, 1990, the staff requested that the licensee address within 1 year of the issuance of this SER the transportation issue in order to verify that the risk due to nearby transportation along Route 9 is acceptably low. By letter dated August 9, 1990, the licensee committed to perform an assessment of transportation in the vicinity of Oyster Creek and submit it as requested. Because the nature of risk associated with truck traffic is cumulative rather than immediate, because the evaluation methodology applied to this type of external event is conservative relative to that applied to internal events, and because of the anticipated timely verification of acceptable risk, the staff does not identify this as an issue that would affect plant operation.

The nearest railroad corridor is approximately 0.25 mile east of the reactor building. Rail traffic through this corridor has been discontinued, and the railroad tracks have been removed.

There are no large commercial harbors within 10 miles of the site. Public marinas are the chief recreational facilities in the immediate site area. The Intracoastal Waterway is the only inland waterway used for shipping in the area. Major shipping lanes in the Atlantic Ocean are located well off shore.

The nearest pipelines to the plant lie in a corridor along U.S. Route 9 approximately 0.25 mile from the plant. These consist of 8-inch- and 6-inch-diameter natural gas pipelines. As noted above, the pipelines pass through the edge of the exclusion area boundary. Therefore, the staff concludes that the pipelines do not pose a significant hazard to the plant because of the distance involved.

There are no missile sites within a 10-mile radius of the Oyster Creek site. Nine airfields are located within 20 miles of the plant. Two of the airfields are military installations: (1) McGuire Air Force Base, also used by the U.S. Air Force, U.S. Air National Guard, and the Military Air Transport Service, 25 miles to the northwest of the site and (2) Lakehurst Naval Air Station, 20 miles north-northwest of the site. Other airports listed by the Federal Aviation

Administration (FAA) are Breton Woods, 17 miles north; Eagle's Nest, 12 miles south-southwest; Coyle Tower, 10 miles west; Ocean County, 9 miles north-northwest; Manahawkin, 9 miles south-southwest; and Beechwood, 8 miles north-northeast. In addition, there is a sod strip 2 miles northeast at Forked River.

The FAA lists three restricted areas in the vicinity of the plant. Two of these areas, R5001A and R5001B, are contiguous to Fort Dix, which is 15 miles to the north-northwest of the site. These restricted areas are used mainly as firing ranges for small arms, artillery, and mortars. The third area, R5002, at Warren Grove is a low-level aerial target range used by the U.S. Air National Guard. Its closest boundary to the plant is 7.5 miles. Bombs, rockets, and 20-millimeter guns are used in the target range. The bombs are dummies that give off a flash, but no explosive charge. The rockets do not have explosive charges, only a propellant to deliver the rocket on target, and shells used in the 20-millimeter guns have solid heads without explosives.

The only air corridor in the vicinity of the site is a civilian corridor marked "Victor Air Lane 312," which is aligned east-west and passes over the site. The corridor can be used by all types of aircraft, but the FAA - which controls all civilian aviation - specifies minimum safe altitudes at which planes can be flown in the corridor. The potential hazard due to nearby aviation facilities is resolved as discussed in IPSAR Supplement 1, Section 2.7.1.

## 2.3 Meteorology (SEP Topic II-2.A)

The staff reviewed the extreme meteorological conditions and severe weather phenomena at the Oyster Creek site under SEP Topic II-2.A to determine if safety-related structures, systems, and components were designed to function under all severe weather conditions such as snow, wind, and tornadoes. It evaluated the capability of structures under various loading combinations under SEP Topic III-7.B (see Section 3.8.1).

### 2.3.1 Regional Climatology

The Oyster Creek site is on the Central Atlantic Coast and has a basically continental climate somewhat modified by its immediate coastal location. The mean annual temperature in the area is about 52°F, ranging from about 30°F in January to about 74°F in July. During winter, the winds are predominantly from the northwest. During summer, however, prevailing winds are from the southwest. Often during the summer, the "sea breeze" phenomenon results in onshore circulation during late morning through early afternoon.

The site is subject to some intense winter coastal storms and in the summer to tropical storms that move up the coast, usually off shore. The prevailing direction of winds above 40 miles per hour (mph) is from the east-northeast at Atlantic City. The Atlantic City National Weather Service Station reported the fastest speed as 91 mph from the northeast during September 1944. In general, during periods of precipitation, there appears to be a higher frequency of northeast winds. The occurrence of coastal-low-type storms that travel along the Atlantic Coast toward New England account for a good percentage of these northeast winds, as well as precipitation.

The average annual precipitation is about 42 inches in the region of the site; the monthly averages are between 3 and 5 inches. The maximum precipitation in 24 hours was about 9 inches for Atlantic City.

### 2.3.2 Local Meteorology

Climatological data retrieved from the New Jersey Agricultural Station at Pleasantville, New Jersey, and the Atlantic City NWS Station, located approximately 33 and 35 miles south-southwest of the site, respectively, have been used to assess the meteorological characteristics of the plant site. Section 2.3.2 of the Final Safety Analysis Report provides information concerning the local meteorological conditions at the site.

### 2.3.3 Onsite Meteorological Measurements Program (SEP Topic II-2.B)

Onsite meteorological measurements are made on a 400-foot tower located at the Forked River plant site. The tower is located west-northwest of the Oyster Creek site at a distance of 2529 feet from the Oyster Creek stack. Measurements of wind speed, wind direction, temperature, and dew point temperature are all made on the tower. The meteorological tower is instrumented at three levels: 380 feet, 150 feet, and 33 feet above the ground.

Oyster Creek station has obtained meteorological data from the Forked River meteorological tower since July 1976. To ensure compliance with Regulatory Guide 1.23, redundant wind-speed, wind-direction, and temperature sensors are located at the 33- and 380-foot levels to ensure efficient data recovery.

The data being collected are recorded on strip chart recorders at the base of the tower. In addition, the control room has recorders for the following parameters: wind speed and direction at the 380-foot level, temperature at the 33-foot level, and the temperature differential between the 380- and 33-foot levels.

Joint tower data recovery rates for wind and stability data for 1968 are 84 percent for the lower and middle measurement levels and 92 percent for the upper level.

### 2.3.4 Atmospheric Transport and Diffusion Characteristics for Accident Analysis (SEP Topic II-2.C)

SEP Topic II-2.C calls for the review of atmospheric transport and diffusion characteristics for accident analysis assumed to demonstrate compliance with the 10 CFR Part 100 guidelines with respect to plant design, control room habitability, and doses to the public during and following a postulated design-basis accident.

Under SEP Topic II-2.C, the staff performed a review to determine the appropriate onsite and near-site atmospheric transport and diffusion characteristics. In particular, the short-term relative ground-level air concentration ($\chi/Q$) values were determined for estimating offsite exposures resulting from postulated accidents. The staff concluded that the $\chi/Q$ values presented in the SER dated March 16, 1982, for SEP Topic II-2.C are appropriate for estimating exposures resulting from postulated accidents and should be used in all accident calculations.

## 2.4 Hydrologic Engineering

### 2.4.1 Hydrologic Description (SEP Topic II-3.A)

Barnegat Bay, on which the site is located, is a relatively shallow body of water extending in a north-south direction parallel to the New Jersey coastline. It is separated from the Atlantic Ocean by Long Beach Island and Island Beach Peninsula, which are divided from each other by the narrow Barnegat Inlet. The bay itself is approximately 20 miles long and from 1 to 5 miles wide and varies in depth between 1 and 10 feet. It is part of the intracoastal waterway and is adjacent to Little Egg Harbor on the south and Silver Bay on the north. On the ocean front at Barnegat Inlet, the mean low water level is 1.5 feet mean sea level.

On the south of the plant site, Oyster Creek flows east to Barnegat Bay. Its drainage basin is 12.4 square miles and consists mostly of pine barrens. It is dammed by a low-head earthen dam known as the Wells Mills Dam, which has a timber spillway and shallow reservoir about 4 miles upstream from the plant site. Another low-head timber dam on the site forms a pond with a 4-acre surface area. It is used to store fire water for use at the plant. Oyster Creek joins the discharge canal approximately 700 feet west of the Route 9 bridge. To the north of the site is South Branch Forked River, which has a watershed area of 2.7 square miles, also flowing west to east in pine barrens land. It is not dammed and empties into the intake canal just upstream of the railroad and the Route 9 bridges crossing the intake canal. The South Branch Forked River discharge flows through two structures before reaching the canal. One is a 12-inch-diameter steel pipe, and the other is a water passageway under the Forked River Nuclear Station site access road.

The plant site covers approximately 800 acres. The plant structures were built on an island created by the intake canal to the north and west, the discharge canal to the south and west, and Barnegat Bay to the east. A dike due east of the reactor and turbine buildings separates the intake and discharge canals and provides ready access to the rest of the site from the island.

### 2.4.2 Floods

The plant island is divided into three drainage basins. The area with the greatest potential for local flooding from probable maximum precipitation is the 5.2-acre area at the north. The storeroom, mobile offices, old and new radwaste buildings, office building, boiler house, and part of the reactor building are located in this area. Existing storm drains functioning at full capacity are assumed to remove 6 cubic feet per second of runoff, leaving a peak overland flow of 60 cubic feet per second. Ponding (5 inches deep) occurs to elevation 23 feet 5 inches mean sea level (MSL).

Flooding of Oyster Creek or the South Branch Forked River will not flood the Oyster Creek plant site.

The probable maximum hurricane storm surge still water level at the site is 22 feet MSL. Less than 1 foot of wave runup would occur. Wave forces on the intake structure will be minimal because of refraction around the plant island.

The staff's evaluation of flood levels and the effects of flooding is given in Section 3.4.

## 2.5 Geology and Seismology (SEP Topic II-4), Tectonic Province (SEP Topic II-4.A), Proximity of Capable Tectonic Structures in Plant Vicinity (SEP Topic II-4.B), and Historical Seismicity Within 200 Miles of Plant (SEP Topic II-4.C)

The results of the Oyster Creek construction permit review by the Atomic Energy Commission (AEC) and its advisors, the U.S. Coast and Geodetic Survey (USC&GS) and the U.S. Geological Survey (USGS), are reported in the Safety Evaluation Report (SER) dated September 23, 1964. In this analysis, the staff and its advisors concluded that the geologic and seismic design bases were adequate. In its SER of December 23, 1968, the AEC, on the basis of the advice of the USC&GS, concluded that accelerations of 0.22g for the safe-shutdown earthquake (SSE) and 0.11g for the operating-basis earthquake (OBE) were acceptable. Since that time several other nuclear plant sites have been evaluated in the general area, including those of Forked River Nuclear Station, Newbold Island Nuclear Generating Station (facility planned for this site was relocated to Hope Creek, New Jersey), Salem Nuclear Generating Station, Summit Nuclear Power Station, and Atlantic Generating Station. The SSE and OBE values for those plants are 0.20g and 0.10g, respectively.

During the SEP geologic review (SEP Topic II-4), the staff relied heavily on its experience in assessing the geology of the other sites in the region. Documents used in this review included USGS guadrangle maps; aerial photographs; the Oyster Creek Hazard Analysis Report; Safety Evaluation Report for the Oyster Creek Nuclear Generating Station; the Preliminary Safeguards Summary Report; the Final Safety Analysis Report (FSAR); a February 4, 1975, report by Woodward-Moorhouse & Associates, Inc., "Geotechnical Study Proposed Radwaste and Off-Gas Building, Oyster Creek Nuclear Power Station"; published documents of the NRC-funded New England Seismotectonic Study; and other documents from the open literature.

The staff reviewed all of these new data and in a letter dated August 3, 1981, concluded that the SSE and OBE values of 0.22g and 0.11g, respectively, were conservative, and there was no evidence of capable faulting in the site region.

### 2.5.1 Regional Geology

The site is located on the Coastal Plain Physiographic Province (Fenneman, 1938) along the New Jersey coast about 35 miles (51 kilometers) north-northeast of Atlantic City. The emerged Coastal Plain Province is from 100 to 200 miles (160 to 320 kilometers) wide, and elevations are generally well below 500 feet (155 meters). The topography is flat to gently hilly with extensive marshlands. An additional part of the Coastal Plain is submerged off shore and is part of the Continental Shelf. It is about the same width as the emerged portion and extends to depths of 500 to 600 feet (155 to 186 meters) below sea level.

### 2.5.2 Site Geology

The Oyster Creek site is underlain by approximately 2000 feet of unconsolidated Coastal Plain sediments. The uppermost units from ground surface down consist of 10 feet and less of man-made sand fill, 15 feet of sand of the Late Pleistocene Cape May Formation (35,000 years to 10 million years before present (mybp)), 60 feet of Cohansey sand of Miocene age (+10 mybp), and more than 100 feet of sand of the Miocene (+10 mybp) Kirkwood Formation.

Investigations of these soils were made in 1964, 1968, and 1973-1974 and included core borings and laboratory testing of undisturbed samples. Investigations were also conducted of similar materials at the Forked River site 1/2 mile to the west of the site. The plant is founded on dense to very dense sand of the Cohansey Formation, which has been demonstrated to be adequate to support it.

On the basis of its review under SEP Topics II-4, II-4.A, II-4.B, and II-4.C, the staff concludes that the information used for developing site-specific spectra is adequate and that the local geologic and seismologic phenomena will not affect the plant.

### 2.5.3 Charleston Earthquake Study

It has been the position of the staff, supported by its advisor, the USGS, that Charleston seismicity is related to structure at Charleston and should not be assumed to migrate anywhere else in the Coastal Plain. Several of the hypotheses allow for the migration of this seismicity to other parts of the Piedmont and Coastal Plain. The staff reviewed all the available information from the Charleston study during the operating license review of the Virgil C. Summer Nuclear Station site. On the basis of that information and advice from the USGS, the staff reaffirmed its earlier conclusion that the Charleston seismicity, including the 1886 modified Mercalli intensity X earthquake, is related to geologic structure in the Charleston area and should not be assumed to occur anywhere but in that area.

By letter dated November 18, 1982, the USGS clarified its previous recommendations to the NRC regarding the recurrence of the 1886 Charleston-type earthquake. The staff is studying this matter, and any requirements resulting from the study will be addressed as a separate licensing action. The staff also is performing a study of USGS's Open File Report 82-1033, "Probabilistic Estimates of Maximum Acceleration and Velocity in Rock in the Contiguous United States." The acceleration levels in this study are arrived at in a different (i.e., solely probabilistic) manner than those developed for individual nuclear power plants. Any changes in the staff's position will be reported separately.

### 2.5.4 Stability of Slopes (SEP Topic II-4.D)

In its review under SEP Topic II-4.D, the staff found that the only slopes at the Oyster Creek plant site considered critical with regard to stability were those of the intake canal and the dike separating the intake and discharge canals. The licensee's analyses for the intake canal slopes demonstrated adequate safety margins against slope failure during the SSE. Even in the unlikely event that the intake slopes do fail and cause some blockage of the canal (this applies to the dike separating the intake and discharge canals as well), there is still ample water available in the canal to effect cooling of the plant. Therefore, slope stability is not a safety concern at the Oyster Creek site.

### 2.5.5 Settlement of Foundations and Buried Equipment (SEP Topic II-4.F)

From the information provided in SEP Topic II-4.F, in a letter dated June 15, 1982, the staff concluded the following:

(1) The major seismic Category I structures are supported by mat foundations bearing in the Cohansey Formation. These structures include the reactor building, the vent stack, the intake and discharge structures, the circulating water tunnels, and the diesel generator building.

(2) Static total and differential settlements of seismic Category I structures are small and were essentially complete soon after construction and should not pose a safety problem.

(3) Liquefaction of the Cohansey sand is sufficiently unlikely under the SEP-recommended ground motion with a peak ground acceleration of 0.165g, and analyses indicate that the sand would not liquefy as a result of the SSE of 0.22g described in the FSAR.

# 3 DESIGN CRITERIA - STRUCTURES, SYSTEMS, AND COMPONENTS

## 3.1 General

The staff review of structures, systems, and components relied on industry codes and standards that have been used as the accepted industry practice. These codes and standards had been reviewed by the staff, found acceptable, and incorporated into the Standard Review Plan (SRP, NUREG-0800).

## 3.2 Classification of Structures, Components, and Systems (Seismic and Quality) (SEP Topic III-1)

General Design Criterion (GDC) 1 of Appendix A to 10 CFR Part 50, as implemented by Regulatory Guide 1.26, requires that structures, systems, and components important to safety be designed, fabricated, erected, and tested to quality standards commensurate with the importance of safety functions to be performed. The codes used for the design, fabrication, erection, and testing of the Oyster Creek plant were compared with current codes.

In IPSAR Section 4.2, the staff stated that it had identified several systems and components for which the licensee was unable to provide information to justify a conclusion that the quality standards imposed during plant construction met quality standards required for new facilities. The staff did not identify any inadequate components. However, because of the limited information on the components involved, the staff was unable to conclude that for code and standard changes deemed important to safety, the Oyster Creek plant met current requirements.

The staff further stated that the licensee had agreed to complete the evaluations described in IPSAR Section 4.2 and to incorporate the results in the Final Safety Analysis Report update, which must be submitted within 2 years after completion of the SEP review (10 CFR 50.71 (e)(3)(ii)). If the results of the licensee's evaluations indicated that facility modifications were required, they would be reported in a licensee event report.

The licensee provided this information by letter dated September 29, 1989. The staff finds that although evolution of code requirements has resulted in altered design margins in certain areas and such variations in margin were not precisely quantified, its review of the licensee's submittal did not identify any specific safety concerns. Therefore, the staff concludes that the information provided is adequate and acceptable. This resolves SEP Topic III-1, IPSAR Section 4.2.

## 3.3 Wind and Tornado Loadings (SEP Topic III-2)

10 CFR Part 50 (GDC 2), as implemented by SRP Sections 3.3.1 and 3.3.2 and Regulatory Guides 1.76 and 1.117, requires that the plant be designed to withstand the effects of natural phenomena such as wind and tornadoes.

The effects of tornadoes were not considered in the original design of the Oyster Creek structural systems.

In IPSAR Sections 4.3.1 through 4.3.9, the staff identified some structures and components important to safety that did not meet current licensing criteria, which require that they be adequate to resist tornado winds of 250 miles per hour and a differential pressure of 1.5 pounds per square inch. The following were identified in the IPSAR as not meeting the prescribed criteria:

(1) reactor building steel structure above the operating floor
(2) ventilation stack
(3) effects of failure of nonqualified structures
(4) components not enclosed in qualified structures
(5) exterior masonry walls
(6) roof decks
(7) intake structure, oil tanks, and diesel generator building
(8) load combinations
(9) soil and foundation capacities

However, in IPSAR Sections 4.3.4, 4.3.5, and 4.3.9, the staff concluded that further evaluation of items (4), (5), and (9) was not warranted.

The licensee responded to the remaining issues in submittals dated February 2 and October 25, 1983, and February 2, March 13, and June 4, 1984.

On the basis of a letter from the staff to the licensee dated March 8, 1986, which provided an evaluation of the responses, in IPSAR Supplement 1, Sections 2.3.2, 2.3.3, and 2.3.4, the staff reported that items (2), (3), and (6) were resolved. Item (7) has been resolved as discussed in Section 3.5.1.1 of this SER.

In IPSAR Supplement 1, Sections 2.3.7 and 2.3.8, the staff identified the following two additional items on the basis of the letter of March 8, 1986: (10) control room and (11) architectural components, respectively.

The licensee addressed items (1), (8), (10), and (11) in a letter dated November 15, 1990. In the letter, the licensee described a planned upgrade of the upper reactor building structure and provided justifications to address other items of concern. This letter is under staff review.

## 3.4 Flood Design Considerations

### 3.4.1 Flooding Potential and Protection Requirements (SEP Topic II-3.B), Capability of Operating Plants To Cope With Design-Basis Flooding Conditions (SEP Topic II-3.B.1), and Safety-Related Water Supply (Ultimate Heat Sink (UHS)) (SEP Topic II-3.C)

10 CFR Part 50 (GDC 2), as implemented by SRP Sections 2.4.2, 2.4.5, 2.4.10, and 2.4.11 and Regulatory Guides 1.59 and 1.27, requires that structures, systems, and components important to safety be designed to withstand the effects of natural phenomena such as flooding. The safety objective of the review under these topics (II-3.B, II-3.B.1, and II-3.C) is to verify that operating procedures and/or system design provided to cope with the design-basis flood are adequate.

The site grade elevation is 23 feet mean sea level (MSL). During its review of the hydrology-related topics, the staff identified the following flooding levels as defined by current licensing criteria:

- probable maximum hurricane - 22 feet MSL
- probable maximum precipitation - 23.5 feet MSL

As a result of these flooding levels, the staff identified nine issues in the IPSAR pertaining to the following: (1) condensate transfer pumps, (2) plant operating limits in the Oyster Creek Technical Specifications (TS) on canal water level, (3) canal water level instrumentation, (4) makeup isolation condenser water sources, (5) plant operating limits in the TS on water level at the service water intake, (6) procedures for a flood, (7) protection during internal flooding, (8) hydrostatic loads on buildings, and (9) reactor and turbine building parapets and scuppers.

3.4.1.1 Condensate Water Pumps

In IPSAR Section 4.1(1), the staff concluded that two condensate transfer pumps are essential to charge the emergency condenser with cooling water during a hurricane-induced flood. Because the motors of both of these pumps are powered from the same engineered safety features bus, a single failure of the power bus would disable both condensate transfer pumps.

In letters dated August 14, 1987, and August 12, 1988, the licensee stated that through a detailed field walkdown and line-loss analysis of an existing system interconnection between the core spray and condensate and demineralized water transfer systems, it was determined that the existing plant configuration ensures that makeup water can be supplied to the isolation condenser. As discussed in Section 3.5.1.1 of this SER, the staff reviewed the proposal and found the water supply path acceptable. Staff concerns about dependence of this path on diesel generators are addressed for flooding scenarios, as discussed in IPSAR Supplement 1, Section 2.1.3. This item is resolved.

3.4.1.2 Flooding Level Procedures

In its topic evaluation, the staff concluded that the Oyster Creek Technical Specifications should include plant-operating limits when flood water levels at the intake or discharge canals exceed 4.5 feet MSL. This proposed requirement was based on the plant emergency procedure (EP-520), which specified operator actions to be taken when water levels in the intake or discharge canals exceed 4.5 feet MSL.

In IPSAR Section 4.1(2), the staff concluded that procedures are sufficient to specify corrective actions for flooding conditions, and modifications to the plant Technical Specifications were not warranted. Therefore, this item was resolved in the IPSAR.

3.4.1.3 Canal Water Level Instrumentation

In IPSAR Section 4.1(3), the staff concluded that water level instrumentation in the intake canal was inadequate and there was no water level measurement in the discharge canal. The staff recommended that automatic water level instrumentation be provided so that the operator would be able to implement emergency shutdown procedures when the specified flooding levels occurred. Because these instruments are not intended for postaccident monitoring, they need not necessarily be safety grade. With adequate water level instrumentation in the intake canal, another water level gage in the discharge canal was not necessary because flooding conditions could be identified from the intake canal measurement.

The design-basis hurricane surge for the Oyster Creek plant has a stillwater elevation of 22.0 feet MSL. Associated wind waves are estimated to be less than 1.0 foot. External cooling water for the plant can be supplied from service water, circulating water, and fire water pumps.

The fire pumps are powered by diesel generators, which are located at about elevation 12.0 feet MSL. The service water and circulating water pump motors are located on the deck of the intake structure (elevation 6.0 feet MSL) at about elevation 8.0 feet MSL.

Oyster Creek Nuclear Generating Station Procedure Number 2000-ABN-3200.31, "High Winds," requires initiation of plant shutdown if the intake water level exceeds elevation 4.5 feet MSL and reactor scram if the water level exceeds 6.0 feet MSL. There is no technical specification for plant shutdown for either high or low water level.

The staff gage mounted on the wing wall of the intake structure that was used to measure high water level was inadequate because of small-sized markings, missing markings, and wrong elevation datum. The staff requested that the licensee replace the existing gage with a quality automatic water level recording gage compatible with its safety significance. The licensee installed a new staff gage on September 13, 1988. The new gage elevation datum is mean sea level and is the same datum that is used in the associated plant operating procedures. The gage has legible gradations that are easily read from the deck of the intake structure.

Although the staff believes there would be some decrease in the margin of safety using the staff gage rather than a recording gage, the degree of variance is difficult to quantify. The most significant factor is the introduction of the potential for human error. The staff gage must be read visually and stillwater level interpolated, whereas a recording gage is located in a stilling well that eliminates wave effects. The high water levels most probably will occur during high winds and heavy rain and at night; these factors increase the potential for human error. Conversely, the deck of the intake structure is at elevation 6.0 feet MSL, which is the "reactor scram" control elevation, and it should be fairly easy to determine when water was over the deck; there would still be 2.0 feet of freeboard before the service water pump motors were lost. The operating procedure does not specify the frequency for reading the gage during these adverse conditions. The staff requires that the gage be read at 1/2-hour intervals from the beginning of high winds and until the water level reaches 3.0 feet MSL and then continuously when the level is above 3.0 feet MSL.

The staff finds the new staff gage and revised operating procedures to be an acceptable alternative to the automatic water level recording gage it had requested.

#### 3.4.1.4 Isolation Condenser Flooding

In IPSAR Section 4.1(4), the staff stated that the plant did not have a reliable means of maintaining a safe shutdown in light of single-failure and flooding conditions, specifically in regard to the provision of adequate makeup water sources for the isolation condensers.

The staff required the licensee to make procedural revisions to include the fire water storage tank as a redundant source of water supply to the emergency condenser and to include in operating procedures a minimum inventory of water to be maintained in the condensate storage tank.

In IPSAR Supplement 1, Sections 3.1.1 and 4.1.1, the staff identified the licensee procedures that specify actions associated with emergency condenser water supplies. The staff verified these procedures by inspection and found them acceptable. Because full resolution of this issue depends on the resolution of the related issue in IPSAR Section 4.1(1), and the latter issue is reported as resolved in Section 3.4.1.1 of this SER, this issue is also resolved.

### 3.4.1.5 Low Water Level Shutdown

In addition to the concern related to shutdown under flooding conditions, the staff identified the concern of low water level shutdown in IPSAR Section 4.1(5). The licensee addressed this concern by providing administrative procedures to monitor water level, using the intake canal instrumentation discussed in Section 3.4.1.3 of this SER (IPSAR Section 4.1(3)), and to appropriately respond to low level in the intake canal.

Low water level at the Oyster Creek station may be caused by a hurricane that forces water out of the intake canal, blockage of the canal, or blockage of the intake screens. Two gages (PI-SWS-1 and PI-SWS-2) at the intake structure monitor potential low water level in the intake canal. These gages provide indication of the intake structure's water level that is on the plant side of the traveling screens and therefore includes any reduction that would result from clogging of the screens. These gages are read routinely (i.e., every shift) by a plant operator, and the readings are recorded on the Intake Area Tour Sheet. Operating Procedure 2000-ABN-3200.32, "Response to Loss of Intake," contains operator actions required at various water levels in the intake canal in order to regain level as well as to ensure safe operation of the plant.

The procedure also instructs the operator to monitor service water discharge pressure indication in the control room to avoid possible service water pump cavitation. The service water pumps are expected to reach their minimum required water level at -0.5 foot MSL. Service water may be lost at this level, and the operator is instructed to follow Operating Procedure 2000-ABN-3200.18, "Service Water Failure." The procedure instructs the operator to shut down the plant if the service water system cannot be returned to operation.

The staff concludes that the licensee's procedures and equipment used for monitoring low water level and controlling the plant under low-water-level conditions are acceptable.

### 3.4.1.6 Hurricane Flooding of Pumps

In IPSAR Section 4.1(6), the staff indicated that the licensee had proposed to update emergency procedures, to identify the alternate water sources and flow paths if the intake structure became flooded, and to identify the priority of water sources and flow paths to be used to ensure a safe shutdown.

In IPSAR Supplement 1, Section 4.1.2, the staff reported that the licensee had identified the station procedures which resolve this item.

3.4.1.7 Protection During Internal Flooding

In IPSAR Section 4.1(7), the staff stated that protection against internal flooding of structures caused by local probable maximum precipitation should be provided to a flood level of 23.5 feet MSL and that the licensee should verify that all entrance levels were above this level.

In IPSAR Supplement 1, Section 2.1.3, the staff reported that the licensee had verified that all entrances except two entrances to the diesel generator building are not vulnerable to flooding to the 23.5-foot MSL. These two entrances are at elevation 23 feet MSL. The licensee proposed to construct a 6-inch-high asphalt dike at each of the two entrances. The staff found this proposal acceptable for resolving the concern.

3.4.1.8 Groundwater Elevation

This issue is resolved as discussed in Section 3.4.2 of this SER.

3.4.1.9 Roof Drains

In IPSAR Section 4.1(9), the staff stated that the licensee had committed to drill holes in the parapets and install scuppers to preclude the potential for buildup of rain water on the roof of either the reactor building or the turbine building.

In IPSAR Supplement 1, Section 4.1.3, the staff reported that the modifications had been completed and verified by inspection. Thus, this issue was resolved.

3.4.2 Effects of High Water Level on Structures (SEP Topic III-3.A)

10 CFR Part 50 (GDC 2), as implemented by SRP Section 3.4 and Regulatory Guide 1.59, requires that plant structures be designed to withstand the effects of flooding. The safety objective of the review under this SEP topic is to ensure the function of safety-related structures with hydrostatic loading resulting from design-basis water levels when combined with other nonaccident loadings.

In IPSAR Section 4.4, the staff reported that all issues associated with this topic, except that concerning short-term hydrostatic loads, had been resolved.

In IPSAR Section 4.4(2), the staff concluded that the licensee should demonstrate that safety-related structures would remain functional under a short-term hydrostatic load and could resist flotation for water levels up to 22 feet MSL.

In IPSAR Supplement 1, Section 2.4.1, the staff reported that in its evaluation of licensee analysis results, it concluded that on the basis of the factors of safety obtained against flotation, the adequacy of the subgrade walls, and the adequacy of bearing capacity, the Oyster Creek facility can adequately withstand a groundwater level up to elevation 23 feet MSL.

This resolved the remaining item and the issue of groundwater elevation (IPSAR Section 4.1(8)).

### 3.4.3 Inservice Inspection of Water Control Structures (SEP Topic III-3.C)

10 CFR Part 50 (GDC 2, 44, and 45) as implemented by Regulatory Guide 1.127, requires that structures, systems, and components important to safety be designed to withstand natural phenomena such as floods and that a system to transfer heat to an ultimate heat sink be provided. Water control structures used for flood protection and emergency cooling water systems are inspected to ensure that water control structures that are part of the ultimate heat sink are available at all times during both normal and accident conditions.

The licensee identified the following water control structures and components that require surveillance in accordance with 10 CFR Part 50 (GDC 1) as implemented by Regulatory Guide 1.127: the intake and discharge canals, the intake structure, trash racks, traveling screens, tunnels, pumps, and the fire protection pond.

The licensee has revised the existing inspection program so that it includes the requirement that the program be conducted or overseen by qualified engineering personnel, that a documentation file be established, and that water control structures be inspected following extreme events.

Resolution of the issues associated with this topic is documented in IPSAR Section 4.5 and IPSAR Supplement 1, Section 4.2.

## 3.5 Missile Protection

### 3.5.1 Tornado Missiles (SEP Topic III-4.A)

10 CFR Part 50 (GDC 2), as implemented by Regulatory Guide 1.117, prescribes structures, systems, and components that should be designed to withstand the effects of a tornado, including tornado missiles, without loss of capability to perform their safety functions.

In IPSAR Section 4.6, the staff identified several structures and components that were vulnerable to tornado missiles.

In IPSAR Supplement 1, Section 2.5, the staff discussed the items in the following sections that still had to be resolved.

#### 3.5.1.1 Emergency Diesel Generators and Fuel Oil Day Tank

In IPSAR Section 4.6.1, the staff stated that the licensee had determined that the diesel generators were not necessary for safe shutdown because makeup water could be provided to the isolation condenser by diesel-driven fire water pumps and by dc power to the main steam relief valves. The staff also indicated that the licensee had agreed to evaluate the potential for and consequences of tornado-missile damage to the diesel generator building.

In letters dated August 14, 1987, and August 12, 1988, the licensee proposed that safe shutdown could be achieved for this scenario with makeup water provided to the isolation condenser by the main core spray pumps (rather than the fire water pumps previously identified). The staff reviewed this proposal and concluded that although the flow path itself is acceptable, the core spray pumps rely on the emergency diesel generators for motive power. The overall acceptability of this method of supply is dependent on the ongoing review of the potential for and consequences of tornado-missile damage to the diesel generator building.

In its safety evaluation dated February 26, 1990, the staff concluded that the walls of the diesel generator vaults and the oil tank compartment are capable of withstanding the loads generated by a tornado having a windspeed of 168 miles per hour and are acceptable. However, the staff required that the licensee provide adequate protection to the outside fuel supply line against the potential missile strike, irrespective of the probability consideration. Another reliable method of ensuring that fuel will be supplied to diesel generators in the event of a supply line break could also be acceptable.

In letters dated April 16 and July 27, 1990, the licensee committed to install a safety-grade check valve and a safety-grade gate valve in the supply line inside the emergency diesel generator fuel tank room. The installation of these valves is intended to prevent the fuel oil supply from backflowing out of the 15,000-gallon diesel generator fuel storage tank (day tank) in the event of a rupture of the fuel supply line outside the fuel storage tank room.

The staff reviewed the licensee's proposed changes to the diesel generator supply line and the proposed modification to protect the day tank fuel supply to the diesel generators and found them acceptable in its safety evaluation dated November 28, 1990. However, the staff's acceptance of the proposed design is predicated on its finalization and implementation. Therefore, if the licensee alters the approved design (e.g., valve number, type, or location), it will submit the amended design to the staff for review and approval. On the basis of the above, SEP Topic III-2, Item 4.3.7, and SEP Topic III-4.A, Item 4.6.1, are resolved.

#### 3.5.1.2 Mechanical Equipment Access Area

In IPSAR Section 4.6.2, the staff identified several components (e.g., motor control centers (MCC-DC-1 and MCC-1AB 21B), control rod drive hydraulic filter, isolation fill piping, and containment spray valve) in the vicinity of the mechanical equipment access opening of the reactor building that were potential targets for missiles penetrating the access doors. These components had not been considered in the staff's original evaluation.

The staff also stated that the licensee had agreed to evaluate the potential for and consequences of tornado-missile impact on components in this area and provide protection, if necessary.

In IPSAR Supplement 1, Section 2.5.2, the staff reported that the licensee had provided the evaluation and supplemental information. The licensee also provided additional information in a letter dated November 15, 1990. These are under staff review.

#### 3.5.1.3 Condensate Storage Tank, Torus Water Storage Tank, and Service Water and Emergency Service Water Pumps

In IPSAR Section 4.6.4, the staff stated that the licensee's position was that the condensate storage tank and torus water storage tank were not required to accomplish safe shutdown because the plant could be safely shut down using one of the two service water pumps or any of the four emergency service water pumps and that backfitting was not required because the pumps were redundant.

The staff also indicated that redundancy does not constitute acceptable protection from tornado missiles. Therefore, it was the staff's position that the licensee provide protection for sufficient systems and components to ensure a safe shutdown in the event of damage from tornado missiles.

The staff also stated that the licensee had agreed to provide a portable pump in a protected area and hose connections to a protected water supply and to provide procedures that specified the conditions for use of this equipment. The staff found this action acceptable. However, as discussed in IPSAR Supplement 1, Section 2.1.1, the licensee now proposes to use an existing system interconnection between the core spray and condensate and demineralized water transfer systems to achieve safe shutdown of the plant.

As discussed in Section 3.5.1.1 of this SER, the staff has reviewed this proposal and has found the water path acceptable. The viability of the path is dependent on diesel generators for motive power. The staff has found the effects of tornadoes on the diesel generator building acceptable. As discussed in Section 3.5.1.1 of this SER, in its safety evaluation dated November 28, 1990, the staff concluded that this item is resolved.

#### 3.5.1.4 Turbine Missiles (SEP Topic III-4.B)

10 CFR Part 50 (GDC 4), as implemented by Regulatory Guide 1.115 and SRP Section 3.5.1.3, requires that structures, systems, and components important to safety be appropriately protected against dynamic effects, which include potential missiles.

One means of providing adequate protection against turbine missiles is ensuring that the probability of failure of the turbine at design or destructive overspeed is low. This assurance arises in part from inspection of the turbine discs and testing and inspection of stop and control valves at regular intervals.

In IPSAR Supplement 1, Section 2.6, the staff concluded that the licensee had proposed a turbine inspection schedule based on a previous inspection and on vendor recommendations. The testing meets the intent of staff criteria, that is, to verify the ability of the stop and control valves to close to prevent turbine overspeed, even though full-closure testing of the control valves is not practical. Therefore, the staff concluded that the licensee's protection against turbine missiles is acceptable.

#### 3.5.1.5 Internally Generated Missiles (SEP Topic III-4.C)

Missiles that are generated internally in the reactor facility (inside or outside the containment) may damage the structures, systems, and components that are necessary for the safe shutdown of the reactor facility or for accident mitigation. Failure of these structures, systems, and components could result in a significant release of radioactivity. The potential sources of such missiles are valve bonnets; hardware retaining bolts; relief-valve parts; instrument wells; pressure-containing equipment, such as accumulators and high-pressure bottles; high-speed rotating machinery; and rotating segments (e.g., impellers and fan blades). Under SEP Topic III-4.C, the staff reviewed the systems and components needed to perform safety functions and in a letter dated June 15, 1982, concluded that the design providing protection from internally generated missiles met the intent of the criteria.

## 3.6 Protection Against Dynamic Effects Associated With the Postulated Rupture of Piping

### 3.6.1 Effects of Pipe Breaks on Structures, Systems, and Components Inside Containment (SEP Topic III-5.A)

10 CFR Part 50 (GDC 4), as implemented by SRP Section 3.6.2, requires, in part, that structures, systems, and components important to safety be appropriately protected against dynamic effects such as pipe whip and discharging fluids. The safety objective of the review under this SEP topic is to ensure that if a pipe should break inside the containment, the plant could safely shut down without a loss of containment integrity, and the break would pose no more severe conditions than those analyzed for the design-basis accidents.

In IPSAR Section 4.9(1), the staff discussed cascading pipe breaks and stated that the issues of concern were addressed under the topic "Leakage Detection Systems" (IPSAR Section 4.16.1), which had been resolved.

In IPSAR Sections 4.9(2) and 4.9(3), the staff discussed jet impingement effects and drywell penetration, respectively, and stated that both items had been resolved.

### 3.6.2 Pipe Break Outside Containment (SEP Topic III-5.B)

10 CFR Part 50 (GDC 4), as implemented by SRP Sections 3.6.1 and 3.6.2 and Branch Technical Positions MEB 3-1 and ASB 3-1 (NUREG-0800), requires, in part, that structures, systems, and components important to safety be designed to accommodate the dynamic effects of postulated pipe ruptures. The safety objective of the review under this SEP topic is to ensure that if a pipe should break outside the containment, the plant could be safely shut down without a loss of containment integrity.

In IPSAR Section 4.10(1), the staff stated that the concerns pertaining to a loss-of-coolant accident outside the containment had been resolved.

In IPSAR Section 4.10(2), the staff identified concerns associated with emergency condenser isolation. In IPSAR Supplement 1, the staff indicated that the licensee would submit information on this matter for review. In a letter dated July 27, 1988, the licensee described plans to replace all four isolation condenser penetrations. Additionally, all isolation condenser piping on the 75-foot elevation will be replaced with Nuclear Grade 316 stainless steel piping and penetration material. To provide time for design review, equipment procurement, and logistical optimization of implementation, the licensee has proposed a deferment in the schedule (from the Cycle 12 refueling outage to the Cycle 13 refueling outage) for the resolution of this issue. The staff finds this change in schedule acceptable. It will review the details of the licensee's final design and justifying analyses when they are submitted.

In Section 6.8 of this SER, the staff discusses the design and operation of the emergency isolation condenser system.

## 3.7 Seismic Design Considerations (SEP Topic III-6)

10 CFR Part 50 (GDC 2), as implemented by SRP Sections 2.5, 3.7, 3.8, 3.9, and 3.10 and SEP review criteria (NUREG/CR-0098, "Development of Criteria for Seismic

Review of Selected Nuclear Power Plants"), requires that structures, systems, and components important to safety be designed to withstand the effects of natural phenomena such as earthquakes.

### 3.7.1 Seismic Design

In IPSAR Supplement 1, Sections 2.9.1 through 2.9.3, the staff stated that licensee submittals for three topics pertaining to seismic design - piping systems, mechanical equipment, and electrical equipment - were under staff review.

By letter dated June 24, 1986, the licensee provided seismic analyses to address IPSAR Sections 4.11(1), "Piping Systems," and 4.11(3), "Electrical Equipment," which the staff found acceptable in an SER dated August 30, 1989. IPSAR Items 4.11(1) and 4.11(3) are therefore resolved.

The IPSAR item pertaining to the ability of safety-related electrical equipment to function was resolved in IPSAR Section 4.11(4).

The IPSAR item pertaining to the qualification of cable trays (IPSAR Section 4.11(5)) is addressed within the scope of Unresolved Safety Issue (USI) A-46, "Verification of Seismic Adequacy of Mechanical and Electrical Equipment in Operating Reactors." The licensee is a member of the Seismic Qualification Utility Group (SQUG), which was formed to respond to USI A-46, and is referencing the work of SQUG to address this item. In its most recent letter on this topic dated October 13, 1988, the licensee confirmed its continued referencing of the work of SQUG and discussed the status of the resolution of plant-specific issues. The current resolution schedule is based on USI A-46 schedular estimates. The staff finds this schedular commitment acceptable.

IPSAR Section 4.11(2) pertains to the seismic design of mechanical equipment. Two topics are included in this section: reactor internals and control rod drive hydraulic control units. On the basis principally of its review of a submittal by the licensee dated January 24, 1983, and a review of related material conducted during a staff visit to the General Electric Company offices in San Jose, California, on November 28-29, 1989, the staff found in an SER dated March 12, 1990, that the Oyster Creek reactor internals do not constitute a safety hazard resulting from safe shutdown earthquake seismic loading conditions.

To address the seismic concerns of the topic related to control rod drive hydraulic control units, the licensee submitted a letter dated June 21, 1990, referencing the SQUG program discussed above and its planned activities pursuant to that reference. By letter dated September 11, 1990, the staff found this commitment acceptable. With this resolution of both topics included in IPSAR Section 4.11(2), that section is resolved.

### 3.7.2 New Seismic Floor Response Spectra

Several different floor response criteria were used in the design of the Oyster Creek station. This has contributed to the difficulties associated with the Oyster Creek seismic design basis in subsequent applications. In July 1987, the licensee proposed to develop new standardized seismic floor response spectra for future applications at Oyster Creek, including the resolution of some of the issues associated with NRC IE Bulletins 79-02 and 79-14 (see Section 3.9.1 of this SER).

By letter dated September 19, 1988, the licensee summarized the history and status of its development of these new seismic spectra. By letter dated October 17, 1988, the staff indicated that a number of issues still had to be resolved before the new floor response spectra, including the use of site-specific data, were approved. Development of the new seismic floor response spectra continues, and they will be reviewed by the staff when they are submitted.

### 3.7.3 Seismic Design Criteria - Short-Term Program (Generic Task A-40)

NRC regulations require that nuclear power structures, systems, and components important to safety be designed to withstand the effects of natural phenomena such as earhtquakes. Detailed requirements and guidance regarding the seismic design of nuclear plants are provided in the NRC regulations and in regulatory guides. However, construction permits and operating licenses for a number of plants were issued before NRC's current regulations and regulatory guidance were in place. For this reason, new reviews of the seismic design of various plants are being undertaken to ensure that these plants do not present an undue risk to the public. Task A-40 is, in effect, a compendium of short-term efforts to support such reevaluation efforts of the NRC staff, especially those related to older operating plants. In addition, some revisions to sections of the SRP and to regulatory guides have resulted to bring them more in line with the state-of-the-art.

The primary objective of the SEP seismic review was to make an overall safety assessment of the seismic capability of the existing plant and, if necessary, to modify the design to ensure the ability to shut down safely in the event of an earthquake. Current review criteria, as defined in the SRP, and the criteria and guidelines developed for seismic review of older plants were used to assess safety margins. Conformance with the SRP would imply acceptability; however, a significant difference in analysis methods and criteria was expected because these plants were originally designed to the criteria developed 10 to 15 years ago. As a result, the staff developed a more reasonable and realistic approach for reanalyses, including the use of ductility reduction methods, nonlinear analysis methods, higher damping, and other factors identified in NUREG/CR-0098. The reanalyses performed as described would ensure an adequate seismic design.

The SEP seismic review addressed the safe shutdown earthquake only because it represents the most severe event that must be considered in the plant design. The scope of the review included three major areas: the integrity of the reactor coolant pressure boundary, the integrity of fluid and electrical distribution systems related to safe shutdown and engineering safety features, and the integrity and operability of mechanical and electrical equipment and engineering safety feature systems (including the containment). The staff did not perform a detailed review of the facilities; rather it relied on the sampling of representative structures, systems, and components. The staff performed confirmatory analyses, using a conservative seismic input, for the sampled structures, systems, and components. The site-specific spectra were supplied to all licensees included in the SEP, by letter dated June 8, 1981, and more sophisticated analysis techniques were used if the conservative sample result indicated overstresses. The results of these analyses were reviewed by the NRC seismic review team. The results of that review for Oyster Creek were published in NUREG/CR-1981.

The NRC seismic review team confirmed that this issue was adequately addressed for the Oyster Creek plant. The staff does not expect the results of Task A-40 to affect these conclusions because the techniques under consideration are essentially similar to those used in the seismic review of the Oyster Creek plant as part of the SEP. Long-term implementation of the resolution of Task A-40 is accomplished in the ongoing resolution of Task A-46, discussed in Section 3.7.1. The staff concludes that this facility can continue to be operated until the ultimate resolution of this generic issue without endangering the health and safety of the public.

## 3.8 Design of Category I Structures

### 3.8.1 Design Codes, Design Criteria, Load Combinations, and Reactor Cavity Design Criteria (SEP Topic III-7.B)

10 CFR Part 50 (GDC 1, 2, and 4), as implemented by SRP Section 3.8, requires that structures, systems, and components be designed for the loading that will be imposed on them and that they conform to applicable codes and standards.

In IPSAR Section 4.12, the staff concluded that areas of design code changes potentially applicable to the Oyster Creek plant for which the code in effect at that time required substantially greater safety margins than the earlier version of the code or for which no original code provision existed should be evaluated to ensure adequate margins of safety. The licensee committed to (1) review the NRC evaluation to determine applicability of the structural elements identified and (2) perform, on a sampling basis, an evaluation of the code, load, and load combination changes in regard to existing as-built structures to assess the adequacy of the design.

By letter dated June 4, 1984, the licensee submitted an evaluation of design codes, design criteria, and load combination changes for Oyster Creek as requested in Section 4.12 of the IPSAR.

In its safety evaluation dated October 29, 1986, the staff concluded that, on the basis of its review and that of its consultant, Franklin Research Center, the load and load combination issues were satisfactorily resolved. With respect to the design code and criteria changes, 20 of the 23 issues were fully resolved. For two of the design code changes (related to the reinforcement of openings), the staff requested that the licensee supply further information. For the remaining issue - concrete subject to high temperatures and thermal transients - the licensee stated that further investigation of drywell thermal conditions was necessary.

By letters dated May 25 and November 15, 1990, the licensee provided information to address the above concerns. These submittals are under staff review.

### 3.8.2 Drywell Shell Thinning (Corrosion)

During the 1980 Oyster Creek plant outage, water was found leaking from various locations from concrete surrounding the drywell. Efforts were made to identify the source of the water and its leak path. Corrective actions were performed during the 1980, 1983, and 1986 plant outages.

To determine if the water observed coming from the drains had an adverse effect on the drywell shell, a series of ultrasonic measurements of the thickness of the drywell plates was made. Since a reduction in the thickness of the steel shell was observed, the investigation was expanded to include further ultrasonic testing. Core samples were also obtained to evaluate whether the ultrasonic measurements indicated material wastage or localized pitting.

An inspection by the licensee of the drywell steel pressure vessel in 1986 showed that sections of the drywell shell near the base sand entrenchment region were thinner than specified. A second inspection in 1987 showed that sections of the drywell shell above the sand entrenchment region were thinner than specified.

To assess the drywell structural capability, detailed structural analyses were performed assuming a minimum of 0.700-inch wall thickness in the sand cavity region. On the basis of these analyses, the licensee determined that the most limiting condition is in the sand bed region of the drywell shell and the drywell shell thickness is projected to be acceptable until June 1992. In an attempt to reduce the corrosion rate, the licensee has (1) installed cathodic protection in selected sand bed locations, (2) taken steps to eliminate water leakage from reactor building equipment and the refueling cavity, and (3) drained water from the sand bed region.

By letter dated September 12, 1988, the licensee committed to continue the drywell ultrasonic measurements at outages of opportunity requiring drywell entry to confirm the drywell thickness and to obtain meaningful corrosion rate data.

The staff reviewed the results of measurements in this ongoing program, and by letter dated April 28, 1989, it found the results acceptable for continued operation until the Cycle 13 refueling outage, at which time additional data would be reviewed.

On February 14, 1990, in a conference call (summary dated February 26, 1990) to discuss matters related to drywell corrosion at Oyster Creek, the licensee reported to the staff that more recent data indicated a higher corrosion rate than that previously estimated, and that code-allowable stress in the drywell at the 51-foot level could be reached in the summer of 1991. In a letter dated July 10, 1990, the staff discussed its consideration of additional information submitted by the licensee describing preliminary plans for a program to address the drywell corrosion problem.

At a meeting on September 19, 1990 (meeting summary dated October 3, 1990), the licensee reported its status in addressing this issue. The licensee's presentation included data accumulated as of that date, the licensee's assessment using best-estimate techniques that the drywell condition would not degrade to code limits for at least 3 years, and a discussion of various alternative actions that might be included in the licensee's program to address the drywell corrosion problem. At the meeting the licensee indicated that it would submit the details of the program including a structural analysis of the drywell by the end of 1990. By letter dated October 16, 1990, the staff clarified its informational needs regarding material discussed at the meeting. By letter dated November 26, 1990, the licensee provided information regarding drywell inspection plan details (original and augmented) which included justification of sampling techniques and statistical methodology. In the submittal the licensee also reiterated its commitment to provide the remainder of the information discussed at the meeting and in the staff's letter of October 16, 1990. The staff is reviewing the submitted information as it becomes available.

## 3.9 Mechanical Systems and Components

### 3.9.1 Pipe Support Base Plate Designs Using Concrete Expansion Anchor Bolts (IE Bulletin 79-02) and Seismic Analyses for As-Built Safety-Related Piping Systems (IE Bulletin 79-14)

NRC Office of Inspection and Enforcement (IE) Bulletin 79-02, dated March 8, 1979, and revised and supplemented on June 21, August 20, and November 8, 1979, required mathematical verification of loads in piping analyses and/or a testing program for anchor bolts.

On July 2, 1979, the NRC issued IE Bulletin 79-14, which was supplemented on August 15 and September 7, 1979. This bulletin discussed two issues, which had been previously identified, that could cause seismic analyses of safety-related piping systems to yield nonconservative results. One issue involved algebraic summation of loads in some seismic analyses; the other involved the accuracy of the information input to seismic analyses, particularly relative to pipe supports and valve weights.

In response, the licensee initiated a reanalysis and field verification program for systems built as part of the original construction effort (1964 to 1969). The piping systems covered by the program were

(1) liquid poison
(2) shutdown cooling
(3) core spray
(4) emergency service water
(5) control rod drive/scram discharge volume
(6) containment spray
(7) isolation condenser
(8) feedwater
(9) cleanup demineralizer
(10) main steam
(11) reactor recirculation

For the seismic reanalysis portion of the effort, the licensee used 1985 state-of-the-art evaluation techniques. This evaluation revealed that six of the systems did not meet the seismic design bases.

By letter dated September 19, 1988, the licensee summarized the progress made in meeting design criteria in accordance with IE Bulletins 79-14 and 79-02, its intention to use the new seismic floor response spectra (see Section 3.7.2) to evaluate 28 supports not qualified by previous criteria, and a proposed program for resolving the issues associated with IE Bulletins 79-14 and 79-02.

In its SER transmitted by letter dated October 17, 1988, the staff concluded that the licensee's program was acceptable for 693 of 721 supports (except the 28 supports mentioned above), pending inspections and upgrades. In NRC Inspection Report 50-219/89-01, dated February 9, 1989, the staff concluded that for items other than the 28 supports, the licensee's actions were acceptable. In both the letter of October 17, 1988, and Inspection Report 50-219/89-01, the staff stated that acceptance of the 28 supports was interim, pending resolution of the issue of the seismic floor response spectra. A licensee response dated November 1, 1989, is under staff review.

3.9.2 Thinning of Pipe Walls in Nuclear Power Plants (IE Bulletin 87-01)

In response to IE Bulletin 87-01 dated July 9, 1987, the licensee provided specific information in its letter to the NRC dated September 21, 1987, relating to the Oyster Creek programs for monitoring the wall thickness of pipes in the condensate, feedwater, and connected high-energy piping systems, including all safety-related and non-safety-related piping systems fabricated of carbon steel.

In Systematic Assessment of Licensee Performance Report No. 50-219/87-99, the staff discussed an audit of the overall erosion/corrosion monitoring program involving the pipe wall thinning of high-energy carbon steel piping systems. As a result of the audit, the staff concluded that, in general, the licensee's program more than meets industry standards. Appropriate controls are in place in the plant, and management has made a commitment to continue to implement an erosion/corrosion control program at Oyster Creek.

In response to Generic Letter 89-08, "Erosion/Corrosion-Induced Pipe Wall Thinning," dated May 2, 1989, the licensee stated in its letter to the NRC dated July 19, 1989, that an erosion/corrosion monitoring program has been established that meets the intent of the Nuclear Management and Resources Council (NUMARC) program guidelines as referenced in NUREG-1344, "Erosion/Corrosion-Induced Pipe Wall Thinning in U.S. Nuclear Power Plants." The licensee also plans to implement a long-term erosion/corrosion monitoring program pending its evaluation of utility industry and Electric Power Research Institute activities in the erosion/corrosion area. A long-term program is expected to be implemented in time to support inspections scheduled for the Cycle 14 refueling outage.

3.9.3 Pipe Cracks in Boiling Water Reactors (Generic Task A-42)

Intergranular stress corrosion cracking (IGSCC) at welds in boiling-water-reactor (BWR) piping has been of continuous concern for almost 20 years. An ever-increasing amount of research and developmental activity related to understanding the causes of the cracking and ways to prevent it has been going on during this period. Under the auspices of the NRC, two Pipe Crack Study Groups have reviewed the problem in BWRs - one in 1975 and the other in 1979. The findings of these groups were published in NUREG-75/067, "Investigation and Evaluation of Cracking in Austenitic Stainless Steel Piping of Boiling Water Reactor Plants," and NUREG-0531, "Investigation and Evaluation of Stress Corrosion Cracking in Piping of LWR Plants," and staff guidelines to implement their recommendations were published in NUREG-0313, "Technical Report on Material Selection and Processing Guidelines for BWR Coolant Pressure Boundary Piping," and NUREG-0313, Revision 1.

NUREG-0313 was first revised in 1980 to provide guidance and recommendations regarding materials and processes that could be used to minimize IGSCC and to provide recommendations regarding the augmentation of the extent and frequency of inservice inspections of welds considered to be susceptible to IGSCC.

Revision 1 also provided recommendations regarding the upgrading of leak detection systems and leakage limits for plants with susceptible welds.

In NUREG-0313, Revision 2, issued as an enclosure to Generic Letter 88-01, these recommendations were updated and several subjects were added. Revision 2

(1) Provides guidance for performing ASME Code, Section XI, IWB 3600, calculations for flaw evaluation.

(2) Provides recommendations regarding the repair of cracked piping.

(3) Recommends formal performance demonstration tests for ultrasonic test examiners, such as those prescribed by IE Bulletins 82-03, "Stress Corrosion Cracking in Thick-Wall, Large-Diameter, Stainless Steel Recirculation System Piping at BWR Plants," and 83-02, "Stress Corrosion Cracking in Large-Diameter Stainless Steel Recirculation System Piping at BWR Plants," and currently being conducted under the Nondestructive Examination Coordination Plan, agreed upon by the NRC, the Electric Power Research Institute, and the BWR Owners Group. This will provide additional assurance that inspections for IGSCC in BWR piping will be performed effectively.

The approach used in previous editions of NUREG-0313 to identify welds that require augmented inspection was simplified, but was expanded to include consideration of reinspections of welds found to be cracked, with or without repair or mitigative actions. The current approach is based on the following:

(1) All stainless steel welds in high-temperature BWR systems are considered to be subject to IGSCC unless measures have been taken to make them resistant.

(2) The frequency and sample size used to inspect all safety-related piping welds in BWR plants will depend on the material and processing used. Simple bases are provided for such classification.

(3) Some utilities may choose not to replace piping, or to operate for some interim period before making major modifications or replacing piping. Guidance is provided to cover these situations in which a utility chooses to operate with cracked or repaired welds.

The above program resolves Generic Task A-42 and is applied at Oyster Creek.

The NRC staff reviewed submittals dated January 20, 23, and 31, 1989, from the licensee regarding the IGSCC inspection and repairs performed during the Cycle 12 refueling outage at Oyster Creek. The licensee reported that 143 welds susceptible to IGSCC in various stainless steel piping systems were inspected during this outage and that 6 welds showed indications of IGSCC (3 in the recirculation system and 3 in the isolation condenser system). Of the six flawed welds, five were reinforced by weld overlay with standard design and one was left in the as-stress-improved condition because the reported flaw indications were small. The licensee also reported that there was no significant flaw growth in weld NG-C-9A, which was found flawed during the previous refueling outage.

On the basis of its review of the information provided by the licensee, the staff found that the inspection and overlay repairs that were performed met the guidelines of Generic Letter 88-01 with the exception of the inspection scope for Category G welds (welds not yet being properly inspected). Because of the timing of Generic Letter 88-01, the reduced inspection scope for Category G welds for the Cycle 12 refueling outage was accepted. The staff concluded that Oyster Creek can be safely returned to operation for at least one additional fuel cycle, with assurance that the integrity of the reactor coolant pressure boundary will be maintained. However, the staff required the licensee to provide additional detailed information within 6 months after restart from the Cycle 12 refueling outage.

The staff was also concerned about the IGSCC inspection program for the Cycle 13 refueling outage proposed by the licensee in its revised response of January 31, 1989, to Generic Letter (GL) 88-01. The staff requested that the licensee incorporate staff comments in its GL 88-01 response and resubmit its IGSCC inspection program for the Cycle 13 refueling outage for NRC staff review at least 3 months before the start of the next outage.

On the basis of the above, the staff concludes that Generic Task A-42 is resolved for Oyster Creek through the Cycle 13 refueling outage with the continuing implementation of the IGSCC inspection program and that operation of the plant does not pose a threat to the health and safety of the public.

### 3.9.4 Waterhammer (Generic Task A-1)

Waterhammer events are the result of intense pressure pulses in fluid systems caused by any one of a number of mechanisms and systems conditions. Since 1971, approximately 150 incidents involving waterhammer have been reported for pressurized-water reactors and boiling-water reactors. The waterhammers occurred in steam generator feed rings and piping, decay heat removal systems, emergency core cooling systems, containment spray lines, service water lines, feedwater lines, and steamlines.

Waterhammer occurrences and the underlying causes have been evaluated through Generic Task A-1. The staff's technical findings are reported in NUREG-0927, "Evaluation of Water Hammer Occurrences in Nuclear Power Plants - Technical Findings re Unresolved Safety Issue A-1," for Oyster Creek. Early in plant operations, there was a problem with waterhammer in the core spray system during surveillance testing. This problem was traced to incomplete filling of the system. A design change was made to add fill pumps, which kept the core spray systems filled and pressurized at all times.

In 1987, the licensee determined that waterhammer was the cause of a number of problems with pipe supports for the core spray system 2 full-flow test line. Waterhammer was the result of the rapid opening of a motor-operated valve in the line during the performance of a full-flow test. The problem was corrected by instructing the operator to manually open the valve. The resulting slower opening time permits water flow to increase over a longer period, thus averting waterhammer. Resolution of this issue is reported in Inspection Report 50-219/87-13.

In response to waterhammer events at other BWR facilities, the operating procedures were changed to prohibit isolation condenser initiations when reactor high water level conditions exist. Waterhammer in the isolation condenser has not occurred at the Oyster Creek plant.

The actions taken by the licensee are consistent with the generic findings that support the use of such design features and controls for minimizing or eliminating waterhammer.

On the basis of the Oyster Creek design, operating experience, and operating procedures, the staff concludes that the waterhammer issue is properly addressed for the Oyster Creek plant and that operation can continue without undue risk to the health and safety of the public.

### 3.10 Environmental Qualification of Safety-Related Electrical Equipment (Generic Task A-24)

The evolutionary process of developing environmental qualification requirements and a case-by-case implementation has resulted in a diversity of equipment installed in nuclear plants and different levels of documentation of the extent to which equipment is environmentally qualified. In an effort to further standardize the qualification methods and documentation, Generic Task A-24 was developed. Issuance of NUREG-0588, "Interim Staff Position on Environmental Qualification of Safety-Related Electrical Equipment," by the NRC in July 1981 completed the resolution of this unresolved safety issue. For operating reactors such as the Oyster Creek plant, the Division of Operating Reactors Guidelines, transmitted to the licensee by letter dated February 15, 1980, provide the basis for environmental qualification requirements.

By letter dated September 19, 1980, the NRC transmitted a revised order for modification of license directing that information regarding the environmental qualification of safety-related electrical equipment be submitted to the staff by November 1, 1980.

Franklin Research Center (FRC), under contract to the NRC, reviewed the licensee responses and provided an assessment in a draft interim technical evaluation report dated October 24, 1980. The licensee provided additional information by letter and report dated October 28, 1980. Review of the additional information by FRC resulted in an SER forwarded by letter dated June 10, 1981. The licensee's responses to this SER, dated October 23, 1981, and June 16, 1982, resulted in the staff issuing a third report forwarded by letter dated November 30, 1982.

In the SER dated November 30, 1982, the staff concluded that continued operation until completion of the licensee's environmental qualification program will not present undue risk to the public health and safety. Furthermore, the staff has continued to review the licensee's environmental qualification program. For any additional qualification deficiencies identified during this review, the licensee was required to reverify the justification for continued operation.

On February 23, 1983, the final Environmental Qualification (EQ) rule became effective. The EQ rule in 10 CFR 50.49(g) requires each holder of an operating license issued before February 22, 1983, to identify to the Commission by May 20, 1983, the electrical equipment important to safety that is already qualified and submit a schedule for completing final equipment qualification for the remaining electrical equipment important to safety (within the scope of the rule). Qualification is to be completed by the end of the second refueling outage after March 31, 1982, or by March 31, 1985, whichever is earlier.

By letter dated March 16, 1983, the licensee provided the information required by the rule. The licensee stated that it would meet the requirements and schedule of 10 CFR 50.49. Inspection Report 50-219/86-08 documents an inspection conducted March 24 to 27, 1986, to review the licensee's implementation of a program in accordance with the requirements of 10 CFR 50.49 for establishing and maintaining the qualification of electrical equipment within the scope of 10 CFR 50.49. During the inspection, the staff identified deficiencies that would be corrected and resolved through subsequent inspection and concluded that, with this corrective action, the licensee has implemented a program meeting the requirements of 10 CFR 50.49. Thus, Generic Task A-24 is resolved for Oyster Creek, with specific items continuing to be the subject of routine NRC inspections.

# 4 REACTOR

## 4.1 Fuel System Design

The Oyster Creek reactor core consists of numerous (137) core cells. Every core cell consists of a control rod and four fuel assemblies that immediately surround it. Around the edge of the core, certain fuel assemblies are not immediately adjacent to a control rod and are supported by individual fuel support pieces. Each fuel assembly is of the 8x8 design, containing 64 rods, mostly fuel with some water rods, which are spaced and suported in a square array. Each fuel rod consists of slightly enriched, high-density ceramic uranium dioxide fuel pellets stacked within Zircaloy cladding. The present fuel vendor is General Electric Company (GE), but some fuel supplied by Exxon Nuclear Corporation (Exxon) is also being used.

The Oyster Creek reactor was designed to achieve a first core average discharge exposure of 15,000 megawatt-days per ton. In regard to reactivity level and reactivity coefficients, the fuel is approximately the same as that used in other operating GE reactors.

The original Oyster Creek core contained 560 (7x7) fuel assemblies, designated Type I, manufactured by GE. These assemblies contained no gadolinia. Poison curtains were used for supplementary reactivity control. In the fall of 1971, a partial reload was performed and 24 fuel assemblies containing gadolinia, manufactured by GE and designated Type II, were loaded. The poison curtains were also removed at this time. The Type II assemblies were the subject of Facility Change Request No. 1.

In the spring of 1972, the reload for Cycle 2 operation consisted of 132 Type II assemblies and 4 Type III assemblies manufactured by Exxon. The Cycle 2 reload was the subject of Facility Change Requests No. 2 and No. 3.

The Cycle 3 reload consisted of 148 Type III E assemblies, whereas the Cycle 4 reload consisted of 80 Type III F assemblies. The characteristics of Type III E and III F fuel were described in Facility Change Requests No. 4 and No. 5 and their supplements.

Type II, III, III E, and III F fuel assemblies incorporated minor modifications, but each type is basically similar to the original Type I (7x7) design, the most significant modification being the incorporation of gadolinia-bearing rods in the assembly.

The Cycle 5 reload consisted of 36 Exxon Type III F (7x7) fuel bundles, 72 Exxon Type VB (8x8) fuel bundles, and 4 Exxon Type V (8x8) fuel bundles. Type V fuel characteristics were described in Facility Change Request No. 6. This was the last facility change request. The Type VB fuel described in the Cycle 5 reload submittal is the same as the Type V fuel except for (1) a decrease in fuel enrichment and burnable poison content and (2) a decrease in fuel pellet density. The smaller diameter 8x8 rods have a lower maximum linear heat generation rate and a larger cladding thickness-to-diameter ratio, which results in increased safety margins when compared with the 7x7 fuel assemblies. In particular, the maximum design linear power and maximum fuel temperature are substantially reduced with the 8x8 fuel design.

The reloads for Cycles 6, 7, 8, and 9 consisted of additional Exxon Type VB (8x8) fuel assemblies. The reload for Cycle 10 consisted of 28 Exxon Type VB assemblies, 112 GE Type P8DRB239 assemblies, and 60 GE Type P8DRB265H fuel assemblies. The Cycle 11 core was made up of Exxon Type VB assemblies, GE Type P8DRB239 assemblies, GE Type P8DRB265H assemblies, GE Type P8DRB299ZA assemblies, and GE Type P8DRB299Z assemblies. The Cycle 12 core includes the same fuel types as those used in Cycle 11, with the addition of GE Type P8DRB-21 (EB) (extended burnup) fuel assemblies.

The staff's evaluation of the most recent reload cycle is documented in the safety evaluation supporting Amendment 129 to Oyster Creek Provisional Operating License (POL) DPR-16, dated October 31, 1988.

### 4.2 Operation With Less Than All Loops in Service

In a safety evaluation forwarded by letter dated February 24, 1976, supporting POL Amendment 15, the staff approved the analysis for operation with a loop out of service. Specifications allow operation with less than all loops in service; that is, one idle loop provided that it is not isolated from the primary coolant system.

POL Amendment 36, issued on May 30, 1979, added an additional specification requiring that at least two recirculation loops be connected (i.e., not isolated) to the reactor coolant system except when the reactor vessel head is removed.

By letter dated March 31, 1988, the licensee proposed an amendment to the POL that would change the limitation on the number of loops out of service from a safety limit in the Technical Specifications to a limiting condition for operation. This proposal is under staff review.

### 4.3 Loose-Parts Monitoring and Core Barrel Vibration Monitoring (SEP Topic III-8.A)

10 CFR Part 50 (GDC 13), as implemented by Regulatory Guide 1.133, Revision 1, and SRP Section 4.4, requires a loose-parts monitoring program for the primary system of light-water-cooled reactors. Oyster Creek does not have a loose-parts monitoring program that meets the criteria of Regulatory Guide 1.133.

A loose-parts monitoring program could provide early detection of loose parts in the primary system that could help prevent damage to the primary system. Such damage relates primarily to

(1) damage to fuel cladding resulting from reheating or mechanical penetration

(2) jamming of control rods

(3) possible degradation of the component that is the source of the loose part to such a level that it cannot properly perform its safety-related function

The following factors were considered in making the recommendation in IPSAR Section 4.13 that no backfitting be done immediately:

(1) A summary of 31 representative loose-parts incidents at 31 reactors (from the value-impact statement of Revision 1 to Regulatory Guide 1.133) indicates that structural damage as a result of loose parts occurred in only 9 incidents. None of these incidents caused a safety-related accident.

(2) Most loose parts can be detected during refueling inspections.

Backfitting of a loose-parts monitoring program is being considered in Revision 1 to Regulatory Guide 1.133. If the staff decides to implement the recommendations of this revision, then the need to implement a loose-parts monitoring program at operating reactors will be addressed generically.

## 4.4 Irradiation Damage, Use of Sensitized Stainless Steel, and Fatigue Resistance (SEP Topic III-8.C)

Under SEP Topic III-8.C, the staff reviewed the materials used in the construction of the reactor internal structures. The staff found that the materials specified for the Oyster Creek plant have been proven to be adequate according to the current standards by extensive tests and satisfactory performance. In addition, the staff reviewed the effects of using sensitized stainless steel in the internal structures and the licensee's inservice inspection program for these structures. Findings from the inservice inspections performed on the internal structures have not, as yet, been provided by the licensee or reviewed by the staff.

On the basis of the SEP review, the staff concluded, in a letter dated October 30 1980, that the integrity of the reactor internal structures at Oyster Creek has not been degraded through the use of sensitized stainless steel. However, since 1980 a generic concern has arisen regarding intergranular stress corrosion cracking of susceptible materials in the reactor internal structures. The BWR Owners Group is currently engaged in the development of an inservice inspection program for the internal structures to demonstrate that their integrity is maintained.

## 4.5 Reactivity Control Systems Including Functional Design and Protection Against Single Failures (SEP Topic IV-2)

10 CFR Part 50 (GDC 2), as implemented by SRP Section 7.7, requires that the reactor protection system be designed to ensure that specified acceptable fuel design limits are not exceeded for any single malfunction of the reactivity control systems. A limited probabilistic risk assessment of the effects of multiple rod withdrawal on risk demonstrated that this issue is of low importance because (1) the single failures identified do not affect the ability of the scram function and (2) the limited exceedance of the fuel thermal limits is not significant to risk. All significant risk sequences involve core melt, and the issue of multiple rod withdrawal does not affect core-melt probability.

In IPSAR Section 4.15, the staff stated that during the SEP topic review, sufficient information was not available for the staff to complete a single-failure analysis of the rod control system. On the basis of the review of Dresden Unit 2, specific types of rod motion from postulated single failures were identified for Oyster Creek. These were then used as input to the core analysis under SEP Topic XV-8, "Control Rod Misoperation." On the basis of the assumed rod motions, it was determined that the Oyster Creek design meets current licensing criteria. On the basis of the considerations described above, the staff concluded that further analysis by the licensee was not warranted. Backfitting was not recommended.

# 5 REACTOR COOLANT SYSTEM AND CONNECTED SYSTEMS

## 5.1 Summary Description

The reactor coolant system (RCS) consists of five recirculation loops, each with a motor-driven pump and motor-operated gate valves. A valved bypass line around each downstream recirculation line valve is provided.

Two main steamlines exit from the reactor vessel to the turbine generator. Feedwater is returned from the condenser through two main lines penetrating the containment, each of which branches to two lines before reaching the main feedwater sparger.

During operation, the nuclear fuel generates heat within the reactor vessel and boils the water. The resulting steam-water mixture flows to the steam separators; the steam passes through the steam dryer and on to the turbine.

The RCS pressure boundary provides the second barrier against the release of radioactivity generated within the reactor and is designed to ensure a high degree of integrity throughout the life of the plant.

## 5.2 Reactor Coolant Pressure Boundary (RCPB) Leakage Detection (SEP Topic V-5)

10 CFR Part 50 (GDC 30), as implemented by Regulatory Guide 1.45 and SRP Section 5.2.5, prescribes the types and sensitivity of systems and their seismic, indication, and testability criteria necessary to detect leakage of primary reactor coolant to the containment or to other interconnected systems. Regulatory Guide 1.45 recommends that at least three separate leak detection systems be installed in a nuclear power plant to detect unidentified leakage from the RCPB to the primary containment of 1 gallon per minute within 1 hour. Leakage from identified sources must be isolated so that flow of this leakage may be monitored separately from unidentified leakage. The detection systems should be capable of performing their functions after certain seismic events and of being checked in the control room. Of the three separate detection methods recommended, two of the methods should be (1) sump level and flow monitoring and (2) airborne particulate radioactivity monitoring. The third method may be either monitoring the condensate flow rate from air coolers or monitoring airborne gaseous radioactivity. Other detection methods - such as monitoring humidity, temperature, or pressure - should be considered to be indirect indications of leakage to the containment. In addition, provisions should be made to monitor systems that interface with the RCPB for signs of intersystem leakage through methods such as monitoring radioactivity and water levels or flow.

### 5.2.1 Leakage Detection Systems

In IPSAR Supplement 1, Section 2.12.1, the staff discussed the Oyster Creek leakage detection systems and their compliance with the criteria identified above. Consistent with the findings in IPSAR Supplement 1, the licensee, in a letter dated July 1, 1988, reported the results of its extended assessment of Oyster Creek leakage detection systems and committed to install a new drywell airborne particulate and gaseous radiation monitoring system, which was scheduled for completion during operating Cycle 12. The staff finds that this system will supplement other leakage detection systems in accordance with the staff recommendations given in IPSAR Supplement 1. Region I personnel will confirm implementation by inspection to fully resolve this issue.

#### 5.2.2 Operability Requirements

In IPSAR Supplement 1, Sections 3.2.1 and 4.3.1, the staff reported the resolution of this issue, which is to be verified by Region I personnel.

### 5.3 Reactor Vessel

#### 5.3.1 Reactor Vessel Materials Toughness (Generic Task A-11)

Resistance to brittle fracture, a rapidly propagating catastrophic failure mode for a component containing flaws, is described quantitatively by a material property generally denoted as "fracture toughness." Fracture toughness has different values and characteristics depending on the material being considered. For steels used in a nuclear reactor pressure vessel, three considerations are important: (1) fracture toughness increases with increasing temperature, (2) fracture toughness decreases with increasing load rates, and (3) fracture toughness decreases with neutron irradiation. In recognition of these considerations, power reactors are operated within restrictions imposed by the Technical Specifications on the pressure during heatup and cooldown operations. These restrictions ensure that the reactor vessel will not be subjected to a combination of pressure and temperature that could cause brittle fracture of the vessel if there were significant flaws in the vessel materials. The effect of neutron radiation on the fracture toughness of the vessel material is accounted for in developing and revising these Technical Specification limitations.

For the service time and operating conditions typical of current operating plants, reactor vessel fracture toughness for most plants provides adequate margins of safety against vessel failure under operating, testing, maintenance, and anticipated transient conditions and accident conditions over the life of the plant. The principal objective of Task A-11 was to develop an improved engineering method and safety criteria to allow a more precise assessment of the safety margins during normal operation and transients in older reactor vessels with marginal fracture toughness and of the safety margins during accident conditions for all plants. Requirements for demonstrating vessel-toughness margins are given in NUREG-0744, Revision 1, "Resolution of Reactor Vessel Materials Toughness Safety Issue," transmitted by Generic Letter 82-26, "Pressure Vessel Material Fracture Toughness."

Appendices G and H to 10 CFR Part 50 require that compliance with minimum fracture toughness requirements be demonstrated and that a materials surveillance program to monitor changes in the fracture toughness properties of ferritic materials in the reactor vessel beltline region be maintained. This issue was discussed during the review of SEP Topic V-6, "Reactor Vessel Integrity," in NUREG-0569, "Evaluation of the Integrity of SEP Reactor Vessels." Resolution of the SEP issue is reported in IPSAR Supplement 1, Section 3.3. Subsequently, the staff issued Generic Letter 88-11, "NRC Position on Radiation Embrittlement of Reactor Vessel Materials and Its Impact on Plant Operations." This letter transmitted a copy of Regulatory Guide 1.99, Revision 2, and requested licensees to predict the effect of neutron radiation on reactor vessel materials as required by 10 CFR Part 50, Appendix G, paragraph V.A, using the methods described in Regulatory Guide 1.99, Revision 2.

By letter dated January 19, 1988, the licensee proposed to revise the pressure-temperature operating limits in the Oyster Creek Technical Specifications, Section 3.6. The pressure-temperature limits were revised to reflect reduced resistance to brittle fracture due to neutron irradiation in the reactor vessel. The revised limits will be valid through 15 effective full-power years. On the basis of its review, the staff concluded that the proposed pressure-temperature limits meet both Appendix G to 10 CFR Part 50 and Regulatory Guide 1.99, Revision 2, and that the change may be incorporated into the Technical Specifications of the Oyster Creek station.

Amendment 120 to POL DPR-16 dated March 21, 1988, incorporated the new pressure-temperature curves for operation as identified by the approved analyses discussed above. This issue is therefore resolved.

### 5.3.2 Reactor Vessel Inspection

By letter dated June 28, 1983, the staff transmitted its safety evaluation (SE) of the Oyster Creek Inservice Inspection Program and the requests for relief made by the licensee for the second inspection interval. As a part of that SE, the staff, pursuant to 10 CFR 50.55a(g)(6)(i), granted relief from the examination requirements of Categories B-A and B-B for reactor vessel shell welds and from the examination requirements of Category B-D for 11 of the 24 primary nozzle to reactor shell welds because of access difficulties. The bases for granting relief and the alternative examinations required can be found in Science Applications, Incorporated, Technical Evaluation Report (TER) SAI-186-023-34, which is attached to the above letter. According to the TER, the inspection interval ended and the reliefs expired on December 7, 1989. The above documents are available in NRC's Public Document Room.

For the current 10-year inspection interval, the requirements of 10 CFR 50.55a are being addressed as follows. The NRC currently is not granting unlimited relief from the existing requirements in Section XI of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code) for the examination of reactor vessel shell welds. The 1989 edition of ASME Code, Section XI, requires essentially 100-percent examination of reactor vessel beltline shell welds. Rulemaking is currently in progress to require early implementation of the Code requirement. Any relief from that requirement will be granted on a case-by-case basis. The staff understands that boiling-water-reactor (BWR) licensees, the Electric Power Research Institute, and inspection contractors are developing tooling that will allow volumetric inspection of BWR reactor vessels from the interior of the vessel. In the interim the staff believes that the alternative examinations required where relief has been granted coupled with the initial construction examinations required by ASME Code, Section III, or earlier additional requirements imposed on vessels designed in accordance with ASME Code, Section VIII, conservatisms in Code design requirements, initial and periodic hydrostatic testing, and the relatively small amount of radiation-induced damage to BWR vessel materials in the early part of its design life provide adequate assurance that reactor vessel integrity will be maintained for specified design conditions.

## 5.4 Component and Subsystem Design

### 5.4.1 Recirculation Pumps

Each of the five reactor recirculation loops contains a motor-driven pump. The pumps are single-stage, vertical, centrifugal units with mechanical shaft seals. The pumps are driven by variable-speed electric motors, which receive electrical power from variable-frequency motor-generator sets.

Over a wide range, reactor power can be controlled without moving control rods by varying the recirculation flow. To increase reactor power, flow is increased; this reduces the void accumulation in the core by removing the steam at a faster rate and thus increasing reactivity. As power increases, a new power level is established where the transient excess reactivity is balanced by the new void formation.

### 5.4.2 Residual Heat Removal System Reliability (SEP Topic V-10.B)

10 CFR Part 50 (GDC 19 and 34), as implemented by SRP Section 5.4.7, Branch Technical Position (BTP) RSB 5-1 (NUREG-0800), and Regulatory Guide 1.139, requires that the plant can be taken from normal operating conditions to cold shutdown using only safety-grade systems, assuming a single failure and using either onsite or offsite power through suitable procedures.

In IPSAR Section 4.18, the staff indicated that the licensee had agreed to implement generic guidelines for emergency procedures.

In IPSAR Supplement 1, Section 4.4, the staff reported that the licensee's provisions to address this item were acceptable, but indicated that the procedural resolution could be affected by resolution of the issues discussed in IPSAR Sections 4.1(1), 4.1(4), 4.6.4, and 4.30. These sections deal with the effects of wind and tornadoes. Sections 3.3 and 3.5.1 of this SER report resolution of these issues, which resolves this item. This resolution is documented in a staff SER dated November 28, 1990.

Related Generic Task A-31, "Residual Heat Removal," was resolved generically with the publication of SRP Section 5.4.7 in May 1978. Only those plants expected to receive an operating license after January 1, 1979, were affected by this resolution, with no backfits to plants with prior operating licenses. Therefore, in effect, Generic Task A-31 does not apply to Oyster Creek.

### 5.4.3 Requirements for Isolation of High- and Low-Pressure Systems (SEP Topic V-11.A)

10 CFR 50.55a, as implemented by SRP Section 7.6 and BTP ICSB 3, requires that interlock systems important to safety be adequately designed to ensure their availability in the event of an accident. This includes those systems with direct interface with the reactor coolant system that have design pressure ratings lower than the reactor coolant system coolant system design pressure.

In IPSAR Supplement 1, Section 2.13, the staff reported that this issue is resolved.

## 5.5 Water Purity of BWR Primary Coolant (SEP Topic V-12.A)

Reactor water quality is controlled to (1) reduce damage to components of the power plant due to chemical and corrosive attack, (2) reduce the fouling of the heat transfer surfaces and mechanical parts, and (3) reduce impurities available for activation in neutron flux zones. Reactor water quality is achieved and maintained by filtration and demineralization with the cleanup demineralizer system and condensate demineralizer system and by suitable selection of system materials.

10 CFR Part 50 (GDC 14), as implemented by Regulatory Guide 1.56, requires that the reactor coolant pressure boundary have minimal probability of rapidly propagating failure. This includes corrosion-induced failure from impurities in the reactor coolant system. The safety objective of the review under this SEP topic is to ensure that the plant reactor coolant chemistry is adequately controlled to minimize the possibility for corrosion-induced failures.

In IPSAR Supplement 1, Sections 3.4 and 4.5, the staff reported that this issue is resolved.

## 5.6 Feedwater and Control Rod Drive Return Line Nozzle Cracking (Generic Task A-10)

Inspections at BWR plants in the United States that have feedwater nozzle/sparger systems disclosed some degree of cracking in the feedwater nozzles of the reactor vessels. Similar cracking has occurred in BWR control rod drive return line nozzles.

Feedwater is distributed through spargers that deliver the flow evenly to ensure proper jet pump subcooling and help maintain proper core power distribution. An essential part of the sparger is the thermal sleeve, which projects into the nozzle bore and is intended to prevent the impingement of cold feedwater onto the hot nozzle surface. This surface is usually heated to essentially reactor water temperature by the returning water from the steam separators and steam dryers. If bypass leakage past the thermal sleeve should occur, relatively cold feedwater will impinge onto the hot nozzle surface. The feedwater, when heated during power operation by extraction steam from the main turbine, is typically about 100°F to 200°F colder (depending on reactor design) than the reactor water. When the feedwater heaters are not in service, as during startups and shutdowns, the differential temperature could be equal to or greater than 400°F. Bypass leakage past a loose thermal sleeve causes a fluctuation in the metal temperature of the feedwater nozzle and could result in metal fatigue and crack initiation. The cracks are then driven deeper by the larger temperature and pressure cycles associated with startups, shutdowns, and certain operational transients.

Under Generic Task A-10, the staff evaluated this cracking problem, the causes, and resultant solutions. The staff evaluation and implementation positions are contained in NUREG-0619, "BWR Feedwater Nozzle and Control Rod Drive Return Line Nozzle Cracking," which was issued in November 1980.

At Oyster Creek, during an inspection of the feedwater nozzle region of the reactor pressure vessel in 1977, the licensee found cracks in the bend radius and bore regions. In the original sparger/thermal sleeve arrangement, leakage occurred. The cooler feedwater leakage mixed with much hotter downcomer flow creating turbulent eddies. The result of this unanticipated mixing was that the nozzle bend radius region, in particular, was alternately wetted by hot coolant then by cooler feedwater at high frequency. High cycle thermal fatigue initiated cracks through the cladding. Normal thermal duty propagated the high-cycle-initiated cracks into the base metal.

To prevent a recurrence, a piston ring seal thermal sleeve was installed in each of four feedwater nozzle penetrations. The piston ring seal was intended to reduce leakage between the thermal sleeve and the feedwater nozzle inner diameter.

In NUREG-0619, the staff concluded that Oyster Creek could continue to operate with the control rod drive (CRD) return line in its current configuration. The conditions of the nozzle region and the CRD return line nozzle are reinspected periodically. In the most recent inspection during the Cycle 12 refueling outage, the licensee found no defects.

On the basis of the considerations discussed above, the staff concludes that this issue is resolved for Oyster Creek by the continued implementation of the requirements of NUREG-0619 to ensure that thermal fatigue cracking does not initiate in the CRD return line and significant bypass leakage does not develop in the replacement feedwater sparger/thermal sleeve assemblies.

# 6 ENGINEERED SAFETY FEATURES

The Oyster Creek Nuclear Generating Station is a 1930 MWt General Electric boiling-water-reactor (BWR/2) facility in which a Mark I pressure-suppression containment is used. The engineered safety features include the emergency condensers, the core spray system, and the automatic depressurization system.

## 6.1 Organic Materials and Postaccident Chemistry (SEP Topic VI-1)

### 6.1.1 Organic Materials

10 CFR Part 50 (GDC 1) requires that structures and systems important to safety be designed and tested to quality standards commensurate with the importance of the safety function to be performed. Also, Appendix B to 10 CFR Part 50, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," describes an acceptable method of complying with the Commission's quality assurance requirements with regard to protective coatings. The safety objective of the review under this SEP topic is to ensure that protective coatings inside the drywell and torus do not consist of material that would decompose in radiation environments (e.g., cellulose hydrocarbons or chlorides) and potentially foul pump seals, bearings, or cooling passages; create a hazardous environment (e.g., hydrogen); or cause material failures.

In IPSAR Supplement 1, Section 4.6.1, the staff reported that this issue is resolved.

### 6.1.2 Postaccident Chemistry

10 CFR Part 50 (GDC 14) requires that the reactor coolant pressure boundary be designed and erected so it has an extremely low probability of abnormal leakage and gross rupture. Also, GDC 41 requires that systems to control substances released in reactor containments be provided to reduce the concentration and quality of fission products released to the environment following a postulated accident.

The safety objective of the review under this topic is to ensure that appropriate methods are available to maintain the pH of the containment spray and emergency core cooling system (ECCS) torus water and to preclude long-term corrosion-induced failures following an accident.

In IPSAR Section 4.21.2, the staff concluded that the limits contained in the licensee's water chemistry procedure conformed to current licensing criteria and that implementation of a procedure to control the quality of water in the torus was acceptable.

## 6.2 Containment Systems

### 6.2.1 BWR Mark I Pressure-Suppression Containments (Generic Tasks A-6, A-7, and A-39)

Oyster Creek has been in the Mark I long- and short-term programs since problems with that containment design were initially identified. The licensee has made significant plant changes to correct the design deficiencies in the Mark I containment. The major changes include (1) the addition of Y quenchers on the electromatic relief valve (EMRV) discharge lines, (2) EMRV vacuum breaker replacement, (3) downcomer bracing, (4) downcomer truncation, (5) the installation of mid-bay saddles, and (6) the strengthening of the torus.

In addition, a plant-unique analysis was submitted to the NRC staff on September 24, 1982.

By letter dated January 13, 1984, the staff issued its safety evaluation of the Mark I containment long-term program pool dynamic loads. In this evaluation, the staff concluded that the containment modifications would restore the original design safety margin to the Mark I containment for Oyster Creek. The modifications were completed during the Cycle 12 refueling outage.

This completes the resolution of Tasks A-6, A-7, and A-39.

### 6.2.2 Containment Isolation System (SEP Topic VI-4)

10 CFR Part 50 (GDC 54, 55, 56, and 57), as implemented by SRP Section 6.2.4 and Regulatory Guides 1.11 and 1.141, requires isolation provisions for the lines penetrating the primary containment to maintain an essentially leaktight barrier against the uncontrolled release of radioactivity to the environment. As discussed in IPSAR Section 4.22, in its review of the containment penetrations, the staff identified several areas that did not conform to current licensing criteria for containment isolation. The staff recommended that backfitting not be required except for the establishment of administrative procedures to lock isolation valves in a closed position and to provide leakage detection for two lines.

In IPSAR Sections 4.22.1 through 4.22.6, the staff identified the following items associated with this issue: locked-closed valves, remote manual valves, valve location, instrument lines, valve location and type, and administrative controls, respectively. In the IPSAR, the staff reported resolution of the last four items (Sections 4.22.3-4.22.6). In IPSAR Supplement 1, Sections 4.7.1 and 2.14.1, the staff reported resolution of the first two items (IPSAR Sections 4.22.1 and 4.22.2, respectively).

The staff subsequently determined that torus vacuum breaker valves V-26-16 and V-26-18 and their associated check valves V-26-15 and V-26-17 might not be addressed by the resolution of the items discussed in IPSAR Sections 4.22.1 through 4.22.6. Valves V-26-16 and V-26-18 are remote manually controlled, air-operated valves that fail in the open position with loss of instrument air which is non-safety grade. In such an instance, check valves V-26-15 and V-26-17 would be relied on for isolation.

Regulatory guidance indicates that a simple check valve is not normally an acceptable automatic isolation valve, but that guidance also provides for acceptability "on some other defined basis." The torus vacuum breaker valve configuration at Oyster Creek is of the original licensing-basis design, which predates current applicable regulatory criteria. By design the valves perform two safety functions, one requiring an open flow path, and the other requiring isolation. The fail-open design reflects the fact that the Oyster Creek design attributes precedence to the vacuum-breaking safety function of these valves. This issue was discussed at a meeting on February 13, 1989 (meeting summary dated February 21, 1989).

On the basis of the discussion and in consideration of the bases for acceptability of the items discussed in IPSAR Sections 4.22.1 through 4.22.6 and the regulatory provision for alternative bases, the staff concludes that continued plant operation is acceptable. However, this issue remains subject to further regulatory consideration.

#### 6.2.3 Mass and Energy Release for Postulated Pipe Break Inside Containment (SEP Topic VI-2.D) and Containment Pressure and Heat Removal Capability (SEP Topic VI-3)

The safety objective of the review under SEP Topic VI-2.D is to ensure that design-basis conditions (e.g., design pressure and temperature) for the containment structure and safety-related equipment are adequate and to determine if the models used in the earlier analyses provide adequate margins of safety when compared with the assumptions and models for current analytical techniques.

The safety objective of the review under SEP Topic VI-3 is to ensure that the maximum temperature and pressure following a loss-of-coolant accident (LOCA) or main steam or feedwater line break have been calculated with conservative assumptions and that the passive heat sinks and active heat removal systems provide the full heat removal capability required to maintain the pressure and temperature below the design pressure and temperature of the containment, safety-related equipment, and instrumentation inside the containment.

In IPSAR Section 3.1, the staff stated that it had reviewed these two items and found them acceptable. The basis for acceptance was a staff SER dated April 30, 1982.

### 6.3 Emergency Core Cooling System

Emergency core cooling is provided by the emergency condensers, the core spray system, and the automatic depressurization system. The primary purpose of the emergency core cooling system (ECCS) is to transfer heat from the reactor core following any loss of coolant at a rate such that the core remains intact and in place and as a coolable geometry.

#### 6.3.1 Emergency Core Cooling System Actuation System (SEP Topic VI-7.A.3)

10 CFR 50.55a(h), as implemented by Institute of Electrical and Electronics Engineers Std. 279-1971, and 10 CFR Part 50 (GDC 37), as implemented by Regulatory Guide 1.22, require that equipment important to safety be tested periodically at power. A limited probabilistic risk assessment of issues related to ECCS testing was performed to determine their importance to risk. The first issue related to testing that is performed by procedure but is not required by plant Technical Specifications. Because the testing is actually performed, there is no reduction in risk associated with this issue. Rather, this is a

regulatory policy issue. It is the staff's position that testing that is important to safety (e.g., that of the ECCS and reactor protection system channel and circuits) should be included in the facility's Technical Specifications.

In IPSAR Supplement 1, Section 3.5, the staff reported resolution of this issue.

### 6.3.2 Core Spray System

The core spray system is one of three separate systems that constitute the emergency core cooling system. A detailed description of this system is found in Section 6.3 of the Final Safety Analysis Report. The system consists of two completely independent loops, each containing two sets of pumps, either one of which can supply rated flow for the system, isolation valves, a spray sparger, and piping and controls. The system delivers a low-pressure spray pattern over the fuel following a LOCA to limit peak cladding temperature to below 2200°F. The function of the spargers is to distribute the spray flow in a manner that ensures that each fuel bundle receives adequate flow.

#### 6.3.2.1 Core Spray Sparger Cracking

A major modification to the core spray system involved the core spray system spargers. Inservice inspection of the reactor internals had identified existing and potential cracks in the sparger assemblies. To provide additional structural margin, redundant mechanical supports were installed at locations where the number and position of cracks create concern about sparger integrity.

As required by the Oyster Creek provisional operating license, paragraph 2.C.7, the licensee inspected the core spray spargers during the Cycle 12 refueling outage and found no new cracks or further progression of existing cracks. The inspections were approved by the staff by letter dated February 8, 1989, as satisfying the startup requirement given in Section 6.3.2.2 below.

#### 6.3.2.2 Core Spray Nozzle Effectiveness (SEP Topic VI-7.A.4)

10 CFR 50.46 requires that an emergency core cooling system be provided and designed to provide adequate core cooling.

Because of cracks in the existing core spray sparger, the Oyster Creek provisional operating license, Amendment 70, January 26, 1984, includes provision for the inspection of both core spray spargers and the repair of assemblies at each refueling outage. Pursuant to this license condition, should the staff determine that new cracks or further progression of existing cracks has occurred, resulting in unacceptable degradation of safety margins, the sparger will be replaced before startup. The spargers were most recently inspected during the Cycle 12 refueling outage (see Section 6.3.2.1); no replacements were found necessary.

#### 6.3.2.3 Core Spray Booster Pump Switching

The core spray system was modified by replacing pressure switches in core spray booster pump discharge lines with differential pressure switches across the core spray booster pump suction/discharge piping. This change was made to address events during which the discharge-pressure-only switches might have misinterpreted system operability status with resulting system misoperation.

This modification is stated to produce enhanced accuracy and reliability in pump operability indication.

### 6.3.3 Emergency Core Cooling System Performance - Appendix K to 10 CFR Part 50

In support of Technical Specifications to accommodate the Oyster Creek Cycle 12 core reload, the licensee submitted analyses of ECCS performance for a spectrum of design-basis loss-of-coolant accidents (LOCAs). These analyses determined the maximum average planar linear heat generation rate limit profile that is incorporated into the plant Technical Specifications. The staff found the methodology used, its applicability, and the calculational results acceptable in an SER issued on October 31, 1988.

### 6.3.4 Containment Emergency Sump Reliability (Generic Task A-43)

The safety concerns of Generic Task A-43 pertain to post-LOCA conditions that can degrade long-term recirculation capability. For pressurized-water reactors, the containment emergency sump is the water source for residual heat removal and containment spray system pumps; for boiling-water reactors (BWRs), the torus or wetwell suction intake structures serve a similar function. These safety concerns pertain to the potential loss of pump net positive suction head margin due to (1) ingestion of air by the pumps and (2) the blockage of suction strainers by LOCA-generated insulation debris that is transported to the torus and drawn onto the suction strainers.

These A-43 safety concerns have been investigated in full-scale hydraulic experiments, by plant surveys, and through generic studies. The findings have shown that vortexing and air ingestion are of much lesser concern than previously hypothesized.

Full-scale experiments of BWR-type suction strainers have demonstrated that for typical submergences and flow rates, the debris strainers act as effective vortex suppressors and that air ingestion levels are nearly zero (see NUREG-0897, "Containment Emergency Sump Performance"). Thus, for Oyster Creek, air ingestion in the post-LOCA period does not appear likely if design conditions are maintained.

With respect to the potential for debris blockage, the blowdown and transport of insulation debris to the torus region will be impeded by the plant design and layout. The breaks of principal concern are within the drywell. Direct blowdown to the torus will be impeded by baffles at the inlets to the torus downcomers, followed by transport to the suction strainers, which is a function of the bulk fluid velocity in the torus, which is generally low. Furthermore, at Oyster Creek, the insulation is a mix of reflective metallic and "blanket" type insulation. Because of the elevation of the intake structures (relative to the torus bottom), metallic debris likely will not be drawn to the intake structures.

On December 3, 1985, the staff issued Generic Letter 85-22, "Potential for Loss of Post-LOCA Recirculation Capability Due to Insulation Debris Blockage." This letter reported that, on the basis of the staff's regulatory analysis (NUREG-0869, Revision 1, "USI A-43 Regulatory Analysis"), no new requirements need be imposed on licensees and construction permit holders, but it recommended that Regulatory Guide 1.82, Revision 1, be used as guidance for the conduct of 10 CFR 50.59 reviews dealing with the changeout and/or modification of thermal insulation installed on primary system piping and components. The letter further advised that if, as a result of NRC staff review of licensee actions associated with the changout or modification of thermal insulation, the staff decides that Standard Review Plan Section 6.2.2, Revision 4, and/or Regulatory Guide 1.82, Revision 1, should be (or should have been) applied to the rework by the licensee, and the staff seeks to impose these criteria, then the NRC will treat such an action as a plant-specific backfit pursuant to 10 CFR 50.109.

With the issuance of Generic Letter 85-22, resolution of Generic Task A-43 is completed for Oyster Creek.

### 6.3.5 Shutdown Decay Heat Removal Requirements (Generic Task A-45)

Following a reactor shutdown, the radioactive decay of fission products continues to produce heat (decay heat) that must be removed from the primary system. The principal means for removing this heat in a boiling-water reactor at high pressure is through the steamlines to the turbine condenser. The condensate is normally returned to the reactor vessel by the feedwater system; however, the steam turbine-driven reactor core isolation cooling system is provided to maintain primary system inventory if ac power is not available. When the system is at low pressure, the decay heat is removed by the residual heat removal systems. Work on this unresolved safety issue will involve an evaluation of the benefit of providing alternative means of decay heat removal that could substantially increase the plant's capability to handle a broader spectrum of transients and accidents. The study will consist of a generic system evaluation and will result in recommendations regarding the desirability of and possible design requirements for improvements in existing systems or an alternative decay heat removal method if the improvements or alternative can significantly reduce the overall risk to the public.

At Oyster Creek, various methods for the removal of decay heat are available. As discussed above, the decay heat is normally rejected to the turbine condenser, and condensate is returned to the vessel by the feedwater system. If the condenser is not available (e.g., because of loss of offsite power), heat can be removed by means of the safety/relief valves to the suppression pool. The isolation condenser provides an alternative means of removing heat and supplying makeup water (i.e., condensate return) to the vessel. The isolation condenser is operated by natural convection. The single closed valve in the return condensate line is opened either automatically or manually, and reactor steam passes through the isolation condenser boiling off water in the secondary side of the condenser. Makeup water to the secondary side of the condenser is provided by taking suction from the fire water tanks or the condensate storage tank. If the isolation condenser is not available, the high pressure feedwater coolant injection system will provide the reactor cooling.

If the isolation condenser and feedwater coolant injection are unavailable, the reactor system pressure can be reduced by the automatic depressurization system so that cooling by the residual heat removal system can be initiated. When the condenser is not used, the heat rejected to the suppression pool is subsequently removed by the residual heat removal system.

IPSAR Supplement 1, Section 4.10, provides an evaluation of the Oyster Creek systems required for safe shutdown.

On November 23, 1988, the staff issued Generic Letter 88-20, "Individual Plant Examination for Severe Accident Vulnerabilities - 10 CFR § 50.54(f)." This letter indicates that Generic Task A-45 has been resolved by identifying plant-specific examinations to be made. Generic Letter 88-20 provides guidance for this plant-specific task, which resolves and supersedes Generic Task A-45.

In consideration of the above discussion of the Oyster Creek design, the staff evaluation (though referencing related items whose resolutions are not complete), and the activities that will take place in compliance with Generic Letter 88-20, the staff concludes that there is reasonable assurance that Oyster Creek can be operated before ultimate resolution of the related issues mentioned above without endangering the health and safety of the public.

### 6.4 Standby Liquid Control System

The standby liquid control system is designed to bring the reactor to a shutdown condition at any time in core life independent of the control rod system capabilities. The rate of reactivity compensation provided by the liquid control system is designed to exceed the rate of reactivity gain associated with reactor cooldown from the full-power condition.

The system consists of an unpressurized tank for low-temperature storage of sodium pentaborate solution, two high-pressure pumps for injecting the solution into the reactor core, two explosive-actuated shear plug valves for isolating the liquid poison from the reactor until required, the poison sparger ring, and additional valves, piping, and associated instrumentation.

The liquid poison tank is complete with a top cover, vent, and drain. The pump suction line is arranged and constructed to minimize entry of particulate material that might settle on the tank bottom. Heaters are provided to heat the water during initial mixing and to maintain the temperature as required during normal operation. The tank has a nominal capacity of 4100 gallons. The licensee will maintain the boron enrichment to a minimum of 35 atom percent of boron-10 and supply 30 gallons per minute of a minimum 15 weight percent of sodium pentaborate solution to the reactor vessel.

In letters dated September 3 and December 30, 1987, the licensee submitted a description of its design for implementing the requirements of 10 CFR 50.62 at the Oyster Creek station. In an SER dated February 18, 1988, the staff concluded that considering the physical size of the Oyster Creek reactor vessel, which has an inside diameter of 213 inches, the aforementioned flow/enrichment combination satisfies the equivalency requirement of the anticipated transient without scram (ATWS) rule, which is based on pump flow of 86 gallons per minute, 13 weight percent sodium pentaborate, 19.8 atom percent boron-10, and a 251-inch-diameter vessel, as discussed in Generic Letter 85-03, "Clarification of Equivalent Control Capacity for Standby Liquid Control System."

The sodium pentaborate solution is delivered to the reactor by one of two 30-gallon-per-minute, 1500-pound-per-square-inch, positive displacement stainless steel pumps. The pumps and piping are protected from overpressure by two relief valves set at approximately 1400 pounds per square inch absolute, which discharge back to the poison tank.

The explosive valves are double squib-actuated shear plug valves. A low-current electrical monitoring system gives visible (pilot light) indication of circuit continuity through both firing squibs in each valve.

A test tank and demineralized water supply are an integral part of the system to facilitate system testing and flushing. All tanks and piping in the system have been designed in accordance with applicable codes. Actuation of the standby liquid poison system is manually initiated from the control room in a manner that ensures that injection is by deliberate act.

In the SER dated February 18, 1988, and the SER accompanying License Amendment 124, the staff found the Oyster Creek standby liquid control system (SLCS) acceptable.

License Amendment 124, dated July 14, 1988, provided Technical Specifications governing the SLCS consistent with the above SERs.

### 6.5 Combustible Gas Control (SEP Topic VI-5) and Hydrogen Control Measures and Effects of Hydrogen Burns on Safety Equipment (Generic Task A-48)

SEP Topic VI-5, "Combustible Gas Control," concerns the potential for combustible gas conditions (i.e., principally hydrogen produced as a result of metal-water reaction involving the fuel element cladding, the radiolytic decomposition of the water in the reactor core and the containment sump, the corrosion of certain construction materials by the spray solution, and any synergistic chemical, thermal, and radiolytic effects of postaccident environmental conditions on containment protective coating systems and electric cable insulation).

As amended on December 2, 1981, 10 CFR 50.44, "Standards for Combustible Gas Control System in Light-Water-Cooled Power Reactors," delineates the requirements pertaining to the prevention of the accumulation of combustible gases in the containment following design-basis accidents. A set of short-term or interim actions relative to hydrogen control requirements to be implemented was described in a notice published in the Federal Register (46 FR 58484) on December 2, 1981. The interim measures require an inerted containment atmosphere for BWR Mark I and II containments. Oyster Creek has a Mark I containment, which is inerted with nitrogen gas during power operation to preclude hydrogen burn.

Generic Letter 84-09, "Recombiner Capacity Requirements of 10 CFR 50.44(C)(3)(ii)," dated May 8, 1984, transmitted the Commission determination of requirements for inerted Mark I BWR containments (for which notices on the construction permits were published before November 5, 1970).

The licensee responded to Generic Letter 84-09 in submittals dated July 13, 1984, and August 14, 1985. On the basis its review of this information, the staff, in a letter dated March 13, 1987, requested that the licensee provide a nitrogen containment atmosphere dilution system capable of isolating air from the containment whenever an isolation signal occurs.

By letter dated May 31, 1988, the licensee responded to the staff's request. After its review of the licensee's response, which proposed an alternative resolution to that requested by the staff, the staff issued a letter dated November 6, 1990, to clarify its position and to request that the licensee address the position and provide a schedule for implementing any corrective actions needed to comply with the position.

On April 19, 1989, the staff issued SECY-89-122, "Resolution of Unresolved Safety Issue (USI) A-48, 'Hydrogen Control Measures and Effects of Hydrogen Burns on Safety Equipment'," which specified that USI A-48 is resolved, referencing hydrogen control regulations given in 10 CFR 50.44.

Generic Task A-48 is therefore resolved for Oyster Creek; however, the plant-specific issue of combustible gas control remains open, pending staff review.

### 6.6 Control Room Habitability

NUREG-0737, "Clarification of TMI Action Plan Requirements," Task Action Plan Item III.D.3.4, "Control Room Habitability," requires that the operators in the control room be adequately protected against the effects of accidental releases of toxic and radioactive gases. This would ensure safe operation or shutdown under design-basis-accident conditions at the Oyster Creek station.

By a confirmatory order dated March 14, 1983, the licensee was required to have NUREG-0737, Item III.D.3.4, fully implemented at the Oyster Creek station before the restart from the Cycle 11 refueling (Cycle 11R) outage. Technical Specifications (TS) related to control room habitability were part of the NUREG-0737 TS requested by the staff in Generic Letter (GL) 83-36, "NUREG-0737 Technical Specifications," dated November 1, 1983. In its letter dated November 22, 1985, the staff evaluated the licensee's response to GL 83-36. By TS Amendment 105 dated July 15, 1986, the licensee was granted a postponement of the full implementation until the Cycle 12 refueling outage, provided interim system upgrades and accident analyses were completed.

Two items - performance of a single-failure analysis of the control room ventilation system and provision of remedial measures, and an assessment of existing diesel generator capability to provide backup power to the control room ventilation system - were postponed. By letter dated April 17, 1989, the licensee indicated that these items had been implemented on March 8, 1989.

Additional TS changes to address the items in GL 83-36 are included in POL Amendment 115, dated March 31, 1987. In the SER accompanying this amendment, the staff identified two GL 83-36 TS items that remain open. These are control room maximum temperature and plant shutdown if the control room heating, ventilation, and air conditioning (HVAC) system (except the dampers) is inoperable in regard to air inflow or control room temperature for more than 7 days.

In a TS change request dated October 18, 1989, as supplemented on February 21, 1990, the licensee addressed these TS open items and other items related to control room habitability. In this submittal the licensee also described modifications that had been made to the Oyster Creek control room HVAC system. With the issuance of POL Amendment 139 dated May 29, 1990, and its accompanying SER, the staff found the licensee's provisions acceptable to resolve this issue.

### 6.7 Containment Vent and Purge System

NRC letters of November 29, 1978, and September 27 and October 23, 1979, directed all utilities to review the containment vent and purge systems to verify that (1) no safety signals are overridden during the purging process and (2) the containment isolation valves will shut without degrading containment integrity during the design-basis loss-of-coolant accident (LOCA).

As a result of a review of the containment vent and purge system, the licensee committed to (1) not override any safety actuation signal circuits during the purging process and (2) physically limit the valves to 30° open. This was consistent with the NRC's interim position attached to the letter of October 23, 1979.

In various submittals, the licensee subsequently committed to

(1) replace all large (more than 3-inch) containment vent and purge valves with valves qualified to close from the fully open position against the dynamic loads of the design-basis LOCA

(2) install single-failure-proof valve manifolds for (a) the containment vent line from the drywell, (b) the nitrogen purge line to the drywell, and (c) the nitrogen purge line to the torus

(3) use a containment high radiation signal to isolate the large containment vent and purge valves

(4) install a pressure relief vent in the exhaust duct of the drywell and incorporate a 5-second time delay on the opening of the standby gas treatment system filter inlet valves

(5) replace all but two position control switches with three position control switches for the large containment vent and purge valves

Subsequently, the licensee proposed to cancel

(1) the proposed modification to replace the large containment vent and purge valves

(2) the proposed modification that would upgrade the nitrogen vent and purge system to safety-grade status

(3) the proposed modification to install a pressure relief vent in the exhaust duct

The staff, in a letter dated October 10, 1986, accepted the licensee's proposal not to replace the existing containment purge and vent isolation valves with new valves.

A design modification was introduced to include drywell high radiation among the other initiators of drywell ventilation isolation for specific vent and purge isolation valves through the addition of two redundant drywell high radiation isolation logic channels that will serve as a backup to the existing protection systems. This design modification was undertaken in direct response to the requirement of NUREG-0737, Item II.E.4.2, position (7), which states, "Containment purge and vent isolation valves must close on a high radiation signal." This was found acceptable in the SER accompanying POL Amendment 116 dated March 31, 1987.

## 6.8 Isolation Condenser System

An alternate shutdown capability was incorporated at Oyster Creek to ensure safe shutdown and cooldown of the reactor if a fire caused evacuation of the control room or control room function was lost because of fire damage in the cable spreading rooms. This capability utilizes the isolation condenser for decay heat removal and reactor cooldown to establish a safe shutdown condition. Since a fire affecting cabling associated with the isolation condenser high flow trip function could result in a spurious isolation of the isolation condenser, the design includes a bypass of the trip function on initiation of the alternate shutdown panel.

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

The staff reviewed and approved the design of this alternate shutdown system, including bypassing the high flow trip function, in an SER dated March 24, 1986.

On September 29, 1988, the licensee shut down the Oyster Creek reactor because of concerns related to the plant's isolation condensers. A special review of this occurrence was conducted by an NRC augmented inspection team (AIT) on October 5-13, 1988, which issued Inspection Report 50-219/88-80 dated October 31, 1988. By submittals dated December 15 and December 28, 1988, the licensee provided analyses to address the findings of the AIT. On January 23, 1989, the staff issued a safety evaluation in which it concluded that the normal accumulation of noncondensible gases in the isolation condenser system will not prevent proper operation of the system upon actuation. Other issues identified in the AIT report will be the subject of ongoing routine NRC inspections. The staff finds this acceptable.

Section 3.6.2 discusses issues related to emergency condenser isolation.

## 6.9 Main Steam Isolation Valve Bypass Line Isolation Valves

In Licensee Event Report (LER) 84-031, Revision 2, dated November 10, 1986, the licensee discussed the elimination of the function of main steam condensate drain valves V-1-106, V-1-107, V-1-110, and V-1-111 as primary containment, or drywell, isolation valves. The valves had failed in the partially open position and were deactivated and secured in their isolation position, as required by the Technical Specifications for inoperable containment isolation valves.

In the Cycle 11 refueling (Cycle 11R) outage, a modification was installed to eliminate the function of these valves as containment isolation valves; the function of these lines as drains was to be provided by other drain lines to the main steamlines. Two removable blind spectacle flanges, one inside and one outside the containment, were installed in the drain lines to serve as the containment isolation devices for these lines when containment isolation was required. This modification was considered by the licensee as the most prudent action because of the material availability for and time constraints of the Cycle 11R outage. Section 3.5 of the Technical Specifications requires the capability for containment isolation, or operable containment isolation valves, when the reactor is critical and operating. The plant response to some of the design-basis accidents is based on the containment being isolated, including these lines. The blind spectacle flange is an acceptable means for providing containment isolation.

The staff discussed this modification with the licensee because these lines could be used to equalize the pressure across the main steam isolation valves (MSIVs) and to open the MSIVs so that cooling for the core would be provided by the main condenser. This pressure equalization would be done by pressurizing the steamlines when there was high pressure in the reactor vessel and low pressure in the steamlines. However, at Oyster Creek the redundant isolation condensers provide safety-grade cooling to the core when the MSIVs are closed. Also, the licensee stated in the LER that the MSIVs could be opened at 1000 pounds per square inch differential across the valve.

In a letter dated December 24, 1986, the staff approved the use of the blind spectacle flanges to replace the above-mentioned containment isolation valves.

## 6.10 Standby Gas Treatment System

The standby gas treatment system (SGTS) is a plant engineered safety feature (ESF) reactor building atmosphere cleanup system that functions as a barrier between the radiation source and the environs during emergency conditions. Upon initiation and secondary containment isolation, the system establishes a negative pressure in the reactor building, thus preventing ground-level leakage of untreated radioactive material from the reactor building to the environs; the system also treats the reactor building atmosphere before it is exhausted through the plant stack. Section 6.5.1.2.1 of the updated Final Safety Analysis Report (FSAR) describes the SGTS as consisting of two redundant, full-capacity parallel flow trains. Section 6.5.1.2.4 of the FSAR states that the system starts automatically during the design-basis accident on receipt of an initiation signal.

The instrumentation and controls section of the FSAR, Section 7.3, discusses the instrumentation provided to initiate ESF systems, including the SGTS. This system has both reactor protection system (RPS) and non-RPS initiation signals. Sections 7.2.2.1 and 7.3.5.2 of the FSAR state that both the RPS and non-RPS systems will automatically perform their protective functions whenever plant conditions exceed preset levels and that no single failure can prevent the initiating circuits from performing their protective functions. In addition, 10 CFR Part 50, Appendix A, General Design Criterion 41, "Containment Atmosphere Cleanup," specifies that each system shall have suitable redundancy to ensure that its safety function can be accomplished assuming a single failure.

In IPSAR Section 4.30(2), the staff evaluated the effects of a loss of vital ac panel no. 1 (VACP-1) on the ability to place the plant in a safe shutdown condition. A limited probabilistic risk assessment, discussed in Appendix D to the IPSAR, dealt with the contribution to risk of the loss of VACP-1-powered control room indications. The staff concluded that the increased probability of operator error due to lost indication did not contribute significantly to top events in the fault trees, and thus, loss of VACP-1 was of low importance to risk. In addition, in IPSAR Section 4.25, the staff evaluated the contribution to risk of automatic bus transfers, specifically, their contribution to loss of power to redundant unit substations 1A2 and 1B2. Backfitting of redundant power supplies for control room indication was not recommended.

In an audit of the licensee's preliminary safety concerns review process conducted on February 17-24, 1989, as documented in Inspection Report 50-219/89-06 dated May 24, 1989, the staff concluded that the Oyster Creek design for the automatic initiation of the SGTS is potentially susceptible to single failures as it has only one power source, VACP-1, and one initiation logic train downstream of VACP-1. In the conclusions of the inspection report, the staff listed seven items of concern associated with the SGTS. The staff reviewed the licensee's actions to address these concerns in an inspection conducted on March 20-23, 1989 (Inspection Report 50-219/89-09, May 24, 1989). In the latter inspection report, the staff concluded that the SGTS automatic start logic was not originally designed to meet single-failure criteria. The staff determined that loss of power to the reactor building ventilation and filter bank heating coils would not stop the SGTS from performing within its design basis. The licensee demonstrated that the SGTS could be manually started during a design-basis accident without exceeding 10 CFR Part 100 exclusion boundary dose limits. On this basis, the staff found the system adequate. Therefore, this item is resolved.

## 6.11 Systems Interaction in Nuclear Power Plants (Generic Task A-17)

The staff's systems interaction program was initiated in May 1978 with the definition of Unresolved Safety Issue (USI) A-17, and the effort was intensified after TMI-2 Action Plan (NUREG-0660), Item II.C.3 ("Systems Interaction"), was issued. The concern arises because the design, analysis, and installation of systems are frequently the responsibility of teams of engineers with functional specialties such as civil, electrical, mechanical, or nuclear. Experience at operating plants has led to questions as to whether the work of these functional specialists is sufficiently integrated to enable them to minimize adverse interactions among systems. Some adverse events that occurred in the past might have been prevented if the teams had ensured that there was necessary independence of safety systems under all conditions of operation.

The NRC staff's current procedures assign primary responsibility for review of various technical areas to specific organizational units and assign secondary responsibility to other units where there is a functional interface. Designers follow somewhat similar procedures and provide analyses of systems and interface reviews. Under Task A-17, methods are being developed that will enable the staff to identify adverse systems interactions that were not considered under current review procedures. The first phase of this study began in May 1978 and was completed in February 1980 by Sandia Laboratories under contract to the NRC (letter dated February 22, 1980).

The Phase I investigation was structured to identify areas that have the potential for interactions between systems and for negating or seriously degrading the performance of safety functions. The study concentrated on commonly caused failures among systems that would violate a safety function. The next step in the investigation was to identify areas in which NRC review procedures may not have properly accounted for these interactions.

Sandia Laboratories used fault-tree analysis on the selected design to identify component failure combinations (cut-sets) that could result in a loss of a safety function. The cut-sets were further reduced by incorporating six linking failures in the analysis. The results of the Sandia effort indicated a few potentially adverse systems interactions within the limited scope of the study. The staff reviewed the interactions for safety significance and generic implications. The staff concluded that no corrective measures needed to be implemented immediately, except for the potential interaction between the power-operated relief valve and its block valve. This interaction was separately identified by the evaluations of the TMI-2 accident while Sandia was performing the study. Because corrective measures were already being implemented, no separate measures were needed under USI A-17.

A systems interaction follow-on study is addressed in NUREG-0660, Section II.C.3, "Systems Interactions." Since April 1980, NRC has intensified the effort by broadening the study of methods to identify potential systems interactions and by preparing guidance for audit reviews of selected plants for systems interactions. Recent experience provides a basis for the staff's development of a more efficient review process for potential systems interactions. The process will provide for a resolution of USI A-17, assimilate operating reactor experience, and rank identified systems interactions by their relative importance to safety.

It is expected that the development of systematic ways to identify, rank, and evaluate systems interactions will further reduce the likelihood of intersystem failures that could result in the loss of plant safety functions. A comprehensive program is expected to employ analytical methods, visual inspections, experience feedback, and simulator dependency experiments. The industry's current experience with systems interaction reviews for light-water reactors is fragmented, but expanding. The methodology employed in the Phase I study is integral to the staff's consideration of a comprehensive systems interaction program.

On September 6, 1989, the staff issued Generic Letter 89-18, which resolves USI A-17. This resolution is based on NUREG-1174, "Evaluation of Systems Interactions in Nuclear Power Plants," and on anticipation that the insights of NUREG-1174 will be considered in other programs (e.g., Generic Letter 88-20, "Severe Accident Vulnerabilities").

Although the licensee has not described a comprehensive program that separately evaluates all structures, systems, and components important to safety for the three categories of adverse systems interactions (spatially coupled, functionally coupled, and humanly coupled), there is assurance that Oyster Creek can be operated without endangering the health and safety of the public.

The common-mode effects of various postulated external events as well as inplant failure effects on safety-related structures, systems, and components have been extensively studied for the Oyster Creek plant to ensure safe shutdown capability. These studies were the result of the Systematic Evaluation Program and the TMI Action Plan items. Areas most recently studied include the effects of seismic events, pipe breaks, internal and external flooding, wind and tornado loadings, internal missiles, and site hazards. In addition, the licensee's fire protection study, together with the staff's proposed course of action, provides substantial assurance that separation and independence of safety-related systems at Oyster Creek are provided.

The plant has been evaluated against current licensing requirements that are founded on the principle of defense in depth. Adherence to this principle results in requirements such as physical separation and independence of redundant safety systems and protection against hazards such as high-energy-line ruptures, missiles, high winds, flooding, seismic events, fires, human factors, and sabotage. These design provisions are subject to review against the general design criteria of 10 CFR Part 50, Appendix A, that address some types of potential systems interactions associated with fires, floods, and high-energy-line breaks. Also, the quality assurance program, which is followed during the operational phase for each plant, contributes to the prevention of introducing adverse systems interactions. Thus, the licensing requirements and procedures have provided an adequate degree of plant safety pending identification of new systems interactions by this task.

On the basis of the above consideration, the staff concludes that there is reasonable assurance that Oyster Creek can continue to be operated until ultimate resolution of this issue without endangering the health and safety of the public.

# 7 INSTRUMENTATION AND CONTROLS

## 7.1 Reactor Protection System

The reactor protection system (RPS) automatically trips the reactor to protect the reactor coolant system against damage caused by high system pressure and to protect the reactor core against fuel rod cladding damage. The Oyster Creek reactor has General Electric hydraulic-type control rod drive mechanisms.

As a result of the anticipated transients without scram (ATWS) events at the Salem Nuclear Power Plant, the Commission published NUREG-1000, "Generic Implications of ATWS Events at the Salem Nuclear Power Plant." In Generic Letter 83-28, "Required Actions Based on Generic Implications of Salem ATWS Events," dated July 8, 1983, the staff identified the actions licensees needed to take based on NUREG-1000. The actions address issues related to reactor trip system reliability and general management capability. These actions are discussed in Section 15.2.

The RPS is designed on a channelized basis to achieve isolation and independence between redundant protection channels. The coincident trip philosophy is implemented to provide a safe and reliable system because a single failure will not defeat the function of the channel and also will not cause a spurious plant trip. Channel independence is carried throughout the system from the sensor to the relay providing the logic. The channelized design that applies to the analog as well as the logic portions of the protection system is discussed below.

The system is made up of two independent logic channels, each having two independent subchannels of tripping devices. Each subchannel has an input from at least one independent sensor, monitoring each of the critical parameters.

The output of the independent subchannel is combined in a one-out-of-two logic; that is, an input in either one or both of the independent subchannels will produce a logic channel trip. Both of the other two subchannels are likewise combined in a one-out-of-two logic, independent of the first logic channel. The outputs of the two logic channels are combined in a one-out-of-two-twice arrangement; they must be in agreement to initiate a scram.

During normal operation, all vital sensor and trip contacts are closed, and all sensor relays are operated energized. The control rod pilot scram valve solenoids are energized, and instrument air pressure is applied to all scram valves. When one of the four sensors trip, a contact opens, deenergizing a relay that controls a contact in its associated subchannel. The opening of a subchannel contact deenergizes a scram relay, which opens a contact in the power supply to the pilot scram valve solenoids supplied by its logic channel. To this point, only one-half the events required to produce a reactor scram have occurred (half-scram). Unless the pilot scram solenoids supplied by the other logic channel are deenergized, instrument air pressure will continue to act on the scram valves and operation can continue. Once a single channel trip is initiated, contacts in that scram relay circuit open and keep that circuit deenergized until the initiating parameter has returned within operating limits and the reset switch is actuated manually. It should be noted that each control rod has individual pilot scram solenoids for each channel and an individual air-operated scram valve. A normally closed switch is provided in each logic channel pilot scram solenoid circuit. This allows each rod to be manually scrammed (tested) by opening both logic channel switches and deenergizing the pilot scram solenoids. A set of redundant backup scram air header valves is provided. This is to ensure that the control rods are inserted despite a single failure of pilot scram solenoids.

### 7.1.1 Testing of Reactor Trip System and Engineered Safety Features, Including Response-Time Testing (SEP Topic VI-10.A)

10 CFR Part 50 (GDC 21) requires that the reactor protection system be designed to permit periodic testing of its functioning, including a capability to test channels independently.

In IPSAR Sections 4.26.1 and 4.26.3, the staff reported the resolution of the issues involving response-time testing and dual-channel testing, respectively.

In IPSAR Supplement 1, Section 3.6.1, the staff reported the resolution of the issues related to the instrumentation for reactor trip system testing. Therefore, all the items associated with this topic are resolved.

### 7.1.2 Isolation of Reactor Protection System From Non-Safety Systems, Including Qualification of Isolation Devices (SEP Topic VII-1.A)

10 CFR 50.55a(h) through Institute of Electrical and Electronics Engineers (IEEE) Std. 279-1971 requires that safety signals be isolated from non-safety signals.

#### 7.1.2.1 Flux Monitoring Isolation

In IPSAR Section 4.27(1), the staff concluded that insufficient isolation capability had been demonstrated between the nuclear flux monitoring system (intermediate range monitors and average power range monitors) and non-safety devices (process recorders and plant computer). The licensee agreed to perform a failure mode and effects analysis (FMEA) to evaluate the potential for common-mode electrical fault propagation. In IPSAR Supplement 1, Section 2.15.1, the staff reported that this analysis had been submitted on August 3, 1984.

In a letter dated October 23, 1984, to the licensee, the staff stated that it had reviewed the licensee's submittal and concluded that there was insufficient information to support the licensee's conclusion that the lack of qualified isolation devices would not compromise the integrity of the reactor protection system (RPS). Specifically, the following information or justification was not included in the submittal:

(1) The evaluation did not address the resistor isolation buffer circuitry between the RPS and the process computer.

(2) The evaluation concluded that the probability of maximum recorder input voltage being applied across the recorder input signal terminals (or R-18) was negligible. However, no justification was presented to support this conclusion.

(3) The evaluation did not describe any periodic testing for stray voltages and system capability to withstand maximum credible voltages, as required by IEEE Std. 279-1971 and IEEE Std. 379-1977. In the absence of such testing, redundancy does not provide sufficient protection.

In letters dated July 8, 1985, April 4, 1986, and August 16, 1988, the licensee addressed the outstanding issues.

In its safety evalution dated October 11, 1988, the staff concluded that SEP Topic VII-1.A has been satisfied at Oyster Creek on the basis of its review of information provided by the licensee. For the first item, the staff accepts relay (coil-to-contact) isolation to provide isolation between the RPS and the non-safety computer. For the electrical isolation between the nuclear flux monitoring system and the computer, the staff finds the isolation amplifier to be acceptable. For the last item, isolation between the nuclear flux monitoring system and the process recorders, the staff finds the recorders to be unacceptable as isolation devices. However, in the SER dated October 11, 1988, as clarified in a memorandum dated October 16, 1989, the staff found that the FMEA performed by the licensee demonstrated that the logic configuration of the RPS together with the internal component separation provided within each recorder as described in the SER would act to inhibit a fault occurring in a recorder section from preventing the RPS performing its safety function. On this basis, the staff concluded that additional electrical isolation was not required for this interface and that the current configuration was acceptable.

#### 7.1.2.2 Reactor Protection System Protective Trip

In IPSAR Supplement 1, Section 4.9.1, the staff reported that the licensee had installed six electrical protection assemblies as required to resolve this issue.

### 7.1.3 Trip Uncertainty and Setpoint Analysis Review of Operating Data Base (SEP Topic VII-1.B)

10 CFR 50.36c.1.ii(A) requires that where limiting safety-system settings are specified for a variable on which a safety limit has been based, the setting should be chosen so that the automatic corrective action will correct the most severe abnormal event anticipated before a safety limit is exceeded.

In IPSAR Section 4.28, the staff stated that sensors RE02A, B, C, and D (core spray and isolation on low-low reactor water level) had setpoints at the extreme low end of their ranges and that these setpoints should be increased to a point where the margin to extreme range was at least equal to the instrument accuracy, or the sensors should be replaced with those having different ranges more suitable for the limiting safety system setting. In response to this concern, the licensee committed to install the General Electric (GE) analog trip system (which had been previously reviewed and approved by the staff in conjunction with the review of GE Topical Report NEDO-21617) during the Cycle 11 outage.

In Inspection Report 50-219/87-08 dated April 28, 1987, the staff stated that the licensee had installed analog trip systems in place of sensors RE02A, B, C, and D. Because of concerns regarding static O-ring switches (see NRC Office of Inspection and Enforcement Bulletin 86-02), the licensee initiated an evaluation of the replacement of other critical sensors with analog trip systems.

On October 17-21 and October 31-November 4, 1988, the staff conducted a safety system outage modification inspection (SSOMI) (Design - No. 50-219/88202). In its letter dated November 16, 1988, which provided the inspection findings, the staff identified certain deficient practices that may have been used by the licensee when it was establishing safety-related setpoint values for measuring instruments. Specifically, the licensee's inclusion of instrument measuring inaccuracies as part of the maximum drift allowable between surveillance tests could have led to incorrect safety instrumentation settings. Before startup of the plant from the Cycle 12 refueling outage, the calculations and setpoints were revised to address this concern.

By letters dated December 12, 1988, and January 19, 1989, the licensee addressed the SSOMI findings. The issues and the responses by the licensee were discussed at a meeting on January 30, 1989, regarding setpoints for process variables (meeting summary dated February 10, 1989).

At the meeting the licensee indicated that it had made programmatic changes to address this matter and was reviewing its Engineering Standard ES-002, "Instrument Setpoint Determination." It also indicated that it had completed a review of 14 safety-related instrument measuring loops and was reviewing an additional 25 safety-related instrument measuring systems that could have been affected by the deficiencies identified above. The licensee also stated that it has initiated a hardware replacement program that involves potential modifications of measuring loops. On the basis of these ongoing actions, the staff considers the issue to be satisfactorily resolved for plant startup. However, the staff requested that the licensee conduct a historical data search of the operating history of all 39 measuring loops in order to identify and resolve any values of setpoints that are not found to be correct. The licensee submitted a report containing its results on this issue and Engineering Standard ES-002 for staff review on May 29, 1990. By letter dated November 13, 1990, the staff requested that the licensee submit additional information to resolve concerns associated with the submittal. Although the status of this issue is acceptable for continued plant operation, SEP Topic VII-1.B remains open pending receipt and review of the information to be submitted.

## 7.2 Engineered Safety Features System Control Logic and Design (SEP Topic VII-2)

10 CFR 50.55a(h) through IEEE Std. 279-1971 requires that safety signals be isolated from non-safety signals and that no credible failure at the output of an isolation device shall prevent the associated protection system channel from meeting the minimum performance requirements specified in the design bases. These isolation devices are required to be safety grade.

The staff reported the resolution of this issue in IPSAR Section 4.29 and IPSAR Supplement 1, Section 2.11.1.

## 7.3 Systems Required for Safe Shutdown (SEP Topic VII-3)

During the SEP review of safe shutdown systems (Topic VII-3) for Oyster Creek, the staff and the licensee developed a list of the minimum systems necessary to take the reactor from operating conditions to cold shutdown. Although other systems may be used to perform shutdown and cooldown functions, the following systems are the minimum number required to fulfill the requirements of Branch Technical Position RSB 5-1 (NUREG-0800):

(1) reactor control and protection system

(2) isolation condensers

(3) condensate transfer system (for isolation condenser makeup)

(4) electromatic relief valves (automatic depressurization system)

(5) core spray system

(6) emergency service water system and containment spray (for torus heat removal)

(7) instrumentation

(8) emergency power (ac and dc) and control power for the above systems

The staff noted that the systems required to take the reactor from hot shutdown to cold shutdown (assuming only offsite power is available or only onsite power is available with a single failure) are capable of being initiated to bring the plant to safe shutdown and comply with current licensing criteria and the safety objectives of SEP Topic VII-3.

The instrumentation available to control room operators to place and maintain the reactor in cold shutdown meets current licensing criteria because no single electrical instrumentation and control failures render vital parameters such as reactor pressure and water level inoperable.

The capability to maintain the reactor in hot shutdown from outside the control room exists and complies with the safety objectives of SEP Topic VII-3. No procedure exists to take the plant from hot to cold shutdown from outside the control room. However, all the required systems and components could be operated at local stations throughout the plant and, therefore, are acceptable.

With the resolution of related items as discussed in Sections 3.3, 3.5.1, and 5.4.2 of this SER, the staff concludes that Oyster Creek satisfies the requirements for safe shutdown, including GDC 17, because of the number and quality of systems provided.

## 7.4 Other Instrumentation and Control Topics

### 7.4.1 Frequency Decay (Reactor Coolant Pump Circuit Breakers)

Issue 9 of NUREG-0138, "NRC Discussion of 15 Technical Issues Listed in Attachment to November 3, 1976 Memorandum From Director, Office of Nuclear Reactor Regulation to Office of Nuclear Reactor Regulation Staff," states that the staff should require that a postulated rapid decay of the frequency of the offsite power system be included in the accident analysis and that the results be demonstrated to be acceptable. Alternatively, the reactor coolant pump (RCP) circuit breakers should be designed to protection system criteria and tripped to separate the pump motors from the offsite power system because rapid decay of the frequency of the offsite power system has the potential for slowing down or braking the RCPs, thereby reducing the cooling flow rates to levels not considered in previous analyses.

Oak Ridge National Laboratory (ORNL), under a technical assistance program, reviewed the frequency decay rate phenomenon and its effects on RCPs. The results of the review are presented in Section 4 of NUREG/CR-1464, "Review of Nuclear Power Plant Offsite Power Source Reliability and Related Recommended Changes to the NRC Rules and Regulations," dated May 1980. In summary, the report shows that the conditions required for dynamic braking of RCPs are a sustained and rapid decrease in frequency while bus voltage is maintained. These conditions are only realized in a highly capacitive system using large amounts of buried transmission cables. The licensee's system does not use large amounts of buried transmission cables. Therefore, the necessary conditions are not present in the Oyster Creek offsite electrical distribution system. Further, Oyster Creek does not have RCPs and if the postulated frequency decay should act to brake the recirculation pumps, the effect would be to decrease the coolant flow rate through the core, thus decreasing the core power level. Accordingly, in a letter concerning SEP Topic VII-6 dated August 29, 1981, the staff concluded that this issue is not applicable to Oyster Creek.

### 7.4.2 Safety Implications of Control Systems (Generic Task A-47)

This issue concerns the potential for transients or accidents being made more severe as a result of control system failures or malfunctions. These failures or malfunctions may occur independently or as a result of the accident or transient under consideration. One concern is the potential that a single failure such as the loss of a power supply, short circuit, open circuit, or a sensor failure could cause simultaneous malfunction of several control features. Such an occurrence could conceivably result in a transient more severe than those transients analyzed as anticipated operational occurrences. A second concern is that a postulated accident could cause control system failures that would make the accident more severe than analyzed. Accidents could conceivably cause control system failures by creating a harsh environment in the area of the control equipment or by physically damaging the control equipment. The staff generally believes that such control system failures would not lead to serious events or result in conditions that safety systems could not handle safely.

Systematic evaluations of all non-safety systems, however, have not been rigorously performed to verify this belief. The potential for an accident that could affect a particular control system and effects of the control system failures may differ from plant to plant.

Therefore, it is not possible to develop generic answers to these concerns, but rather plant-specific evaluations are required. The purpose of this unresolved safety issue is to verify the adequacy of the existing criteria for control systems and, if necessary, to develop and propose additional criteria or guidelines to improve system reliability and enhance safety.

Oyster Creek's control and safety systems have been designed to ensure that control system failures will not prevent automatic or manual initiation and operation of any safety-system equipment required to mitigate accidents and/or to maintain the plant in a safe shutdown condition following any anticipated operational occurrence or accident. This has been accomplished by providing independence between safety-system trains and between safety and non-safety systems.

For the latter, as a minimum, isolation devices were provided. These devices preclude the propagation of non-safety-related equipment faults to the protection systems. In addition, to ensure that the operation of safety-related equipment is not impaired, the single-failure criterion has been applied in the plant design of the protection systems. SEP Topics VI-7.A.3, VI-7.C.2, VII-1.A, VII-2, and VII-3 address elements of this issue.

A systematic evaluation of the control system design, as contemplated for this unresolved safety issue, has not been performed to determine whether postulated accidents could cause significant control system failures that would make the accident consequences more severe than currently analyzed. However, a wide range of bounding transients and accidents is being analyzed to ensure that the postulated events, such as reactor vessel overfill and overcooling events, would be adequately mitigated by the safety systems. In addition, reviews of safety systems were performed with the goal of ensuring that control system failures will not defeat safety-system action.

Additional studies probing the interaction of safety and non-safety systems were performed during Oyster Creek fire protection reviews in accordance with Appendix R to 10 CFR Part 50. Within designated fire zones, it was assumed that damage to any equipment (or its control cables, if affected) could cause failure of any type.

Also, the licensee has been requested (IE Information Notice 79-22) to review the possibility of consequential control system failures that exacerbate the effects of high-energy-line breaks (HELBs) and adopt new operator procedures, where needed, to ensure that the postulated events would be mitigated. The licensee performed an evaluation of those potential harsh-environment effects and concluded that none of the scenarios identified in the information notice constituted potential failure modes that could compromise a safe shutdown of the Oyster Creek plant.

The staff is also evaluating the qualification program to ensure that equipment that may be exposed to HELB environments has been adequately qualified or an adequate basis has been provided for not qualifying the equipment to the limiting hostile environment. The status of this review is contained in the discussion of USI A-24 in Section 3.10.

In addition, IE Bulletin 79-27 was issued to the licensee requesting that evaluations be performed to ensure the adequacy of plant procedures for accomplishing shutdown on loss of power to any electrical bus supplying power for instruments and control. The licensee responded to this bulletin, and the staff concluded that the response and design were acceptable (memorandum dated June 22, 1982).

In June 1989, NUREG-1217, "Evaluation of Safety Implications of Control Systems in LWR Nuclear Power Plants," which contains technical findings related to USI A-47, was published. The technical findings of NUREG-1217, which resolve Generic Task A-47, were included in Generic Letter 89-19 to all licensees. This generic letter contained specific recommendations applicable to Oyster Creek. The licensee has indicated that it will respond to Generic Letter 89-19 by mid-1990.

On the basis of the above considerations and subject to the satisfactory resolution of the Oyster Creek equipment qualification program, the staff concludes that there is reasonable assurance that Oyster Creek can continue to be operated until the ultimate resolution of this generic issue without endangering the health and safety of the public.

# 8 ELECTRIC POWER SYSTEMS

## 8.1 Introduction

10 CFR Part 50, Appendix A, General Design Criteria (GDC) 2, 4, 5, 17, 18, and 50 provide requirements applied to electric power systems in nuclear power plants. Implementation of these criteria in accordance with the intent of Standard Review Plan (SRP) Sections 8.1, 8.2, 8.3.1, and 8.3.2 (NUREG-0800) ensures that systems will perform their design safety functions when required.

## 8.2 Offsite Power System

The offsite power system is referred to in industry standards and regulatory guides as the "preferred power system." It includes two or more physically independent circuits capable of operating independently of the onsite standby power sources and encompasses the grid, transmission lines (overhead or underground), transmission line towers, transformers, switchyard components and control systems, switchyard battery systems, the main generator, and disconnect switches, provided to supply electric power to safety-related and other equipment.

GDC 5, 17, and 18 apply to this system.

### 8.2.1 Potential Equipment Failures Associated With Degraded Grid Voltage (SEP Topic VIII-1.A)

SEP Topic VIII-1.A is composed of two tasks. The first task is to evaluate the adequacy of protection against degraded grid voltages. This task has been completed, and the staff's SER was issued on October 16, 1981.

The second task is to evaluate the adequacy of the onsite power system voltages. The staff's SER for this task was also issued on October 16, 1981 (by separate cover). Because it found that an adequate design exists, the staff in a letter dated March 3, 1982, concluded that Topic VIII-1.A had been satisfactorily resolved.

## 8.3 Onsite Power Systems

### 8.3.1 Station Blackout (Generic Task A-44)

Electrical power for safety systems at nuclear power plants must be supplied by at least two redundant and independent divisions. The systems used to remove decay heat to cool the reactor core following a reactor shutdown are included among the safety systems that must meet these requirements. Each electrical division for safety systems includes two offsite alternating current (ac) power connections, a standby emergency diesel generator ac power supply, and direct current (dc) sources.

Task A-44 involves a study of whether or not nuclear power plants should be designed to accommodate a complete loss of all ac power; that is, a loss of both the offsite and the emergency diesel generator ac power supplies. This issue arose because of operating experience regarding the reliability of ac power supplies. There have been numerous reports of emergency diesel generators failing to start and run in operating plants during periodic surveillance tests. In addition, a number of operating plants have experienced a total loss of offsite electrical power, and more occurrences are expected in the future. In almost every one of these loss-of-offsite-power events, the onsite emergency ac power supplies were available immediately to supply the power needed by vital safety equipment. However, in some instances, one of the redundant emergency power supplies has been unavailable. In a few cases there has been a complete loss of ac power, but during these events, ac power was restored in a short time without serious consequences.

A loss of offsite power involves a loss of both the preferred and backup sources of offsite power. If all offsite power is lost, the onsite emergency ac power system will provide ac power to safety-related equipment. With respect to emergency onsite ac power, the Oyster Creek emergency generators are powered by diesel engines. These systems have been evaluated under SEP Topic VIII-2 and found acceptable. The staff's evaluation is presented in IPSAR Supplement 1, Section 4.11.

A loss of all ac power was not a design-basis event for the Oyster Creek facility. Nonetheless, a combination of design, operating, and testing requirements has been imposed to ensure that this facility will have substantial resistance to a loss of all ac power and that, even if a loss of all ac power should occur, there is reasonable assurance the core will be cooled.

The current licensing criteria require licensees to provide redundant emergency ac power supplies, to demonstrate emergency ac power supply reliability (Regulatory Guide 1.108), and to include the capability of removing decay heat using at least one shutdown cooling train independent of ac power. Boiling-water reactors contain various systems to remove core decay heat following the total loss of ac power. These systems at Oyster Creek consist of an isolation condenser, which will provide an adequate heat sink for at least several hours. This allows time for restoration of ac power from either offsite or onsite sources.

On the basis of above considerations, the staff concludes that there is reasonable assurance that Oyster Creek can be operated before full compliance with the resolution of this generic issue without endangering the health and safety of the public.

On June 21, 1988, the Commission finalized the Station Blackout Rule, 10 CFR 50.63, which resolves and supersedes Generic Task A-44. The Station Blackout Rule is implemented by Multiplant Action Item (MPA) A-22. Compliance with this MPA item will be achieved through normal licensing action.

In its most recent action to address MPA A-22, the licensee submitted a response to the Station Blackout Rule by letter dated April 17, 1989. This response is under staff review.

### 8.3.2 Onsite Emergency Power Systems (Diesel Generator) (SEP Topic VIII-2)

10 CFR Part 50 (GDC 17), as implemented by SRP Sections 8.1 and 8.3.1 and Regulatory Guide 1.9, requires that onsite electric power systems be provided to permit functioning of components important to safety. Regulatory Guide 1.9 specifies that the standby diesel generator systems be designed so that spurious actuation of protective trips does not prevent diesel generators from performing that function.

#### 8.3.2.1 Diesel Generator Annunciators

In IPSAR Section 4.31(1), the staff stated that, in conjunction with a generic review of diesel generator annunciators, it had determined that Oyster Creek did not comply with current criteria as specified in IEEE Std. 279-1971. By letter dated May 17, 1978, the licensee agreed to make suitable modifications to the annunciators.

In a letter dated May 1, 1987, the staff confirmed that the following required modifications to the diesel generator annunciators had been made:

(1) removing existing nondisabling alarms from the present diesel generator trouble alarm

(2) providing a new annunciator for the manual mode switch not in automatic

(3) redesigning the working of the annunciator windows to reflect the conditions more clearly.

(4) providing a low battery voltage sensor with an alarm function indicating diesel generator dc failure.

The staff considers this item to be fully resolved.

#### 8.3.2.2 Diesel Generator Trip Bypass

In IPSAR Section 4.31(2), the staff concluded that two diesel generator protective trips (leading voltage-ampere reactive (VAR) and reverse power relay) should be bypassed during accident conditions. By letter dated November 16, 1982, the licensee committed to modify the diesel generator trips. In IPSAR Supplement 1, Section 4.11.2, the staff reported the resolution of this issue.

### 8.3.3 DC Power Systems (Onsite)

#### 8.3.3.1 Station Battery Capacity Test Requirements (SEP Topic VIII-3.A)

To ensure that the onsite Class 1E battery capacity is adequate to supply dc power to all safety-related loads required by the accident analyses and is verified on a periodic basis, the staff reviewed the Oyster Creek Technical Specifications, including the test program, with regard to the requirement for periodic surveillance testing of onsite Class 1E batteries and the extent to which the test meets Section 5.3.6 of IEEE Std. 308-1971 and Sections 4.2, 4.3, 5.4, and 5.6 of IEEE Std. 450-1975 to determine the adequacy of battery capacity.

The Oyster Creek battery surveillance requirements are included in Sections 4.7.A and B of the plant's Technical Specifications. As discussed in a letter concerning SEP Topic VIII-3.A dated June 29, 1981, these specifications satisfy the requirements and are, therefore, acceptable.

#### 8.3.3.2 DC Power System Bus Voltage Monitoring and Annunciation (SEP Topic VIII-3.B)

10 CFR Part 50 (GDC 17), through IEEE Stds. 308-1974 and 946-1985, as implemented by SRP Section 8.3.2 and Regulatory Guides 1.6 and 1.32, requires that the control room operator be given timely indication of the status of the batteries and their availability under accident conditions.

The staff reviewed the dc power system battery, battery charger, and bus voltage monitoring and annunciation design of Oyster Creek with respect to dc power system operability status indication of battery current, battery breaker/fuse status, battery charger current bus undervoltage, high discharge rate, or charger breaker/fuse status. In IPSAR Section 4.32, the staff concluded that the dc power system monitoring was not in compliance with current licensing criteria.

A limited probabilistic risk assessment (PRA) was performed to determine the importance to risk of dc instrumentation, indication, and alarms. It was determined that additional monitoring devices would reduce the dc bus unavailability and battery unavailability. In the limited PRA, dc battery failures contributed less than 5 percent to the total risk resulting from core melt.

The licensee consequently committed to install alarms for B and C battery breaker open, C battery charger open, and C battery ground. Other battery indication exists, so that dc power system bus voltage monitoring and annunication are acceptable with these modifications. In IPSAR Supplement 1, Sections 2.17 and 4.12, the staff reported the resolution of this item.

## 8.4 Electrical Penetrations of Reactor Containment (SEP Topic VIII-4)

10 CFR Part 50 (GDC 2, 4, 5, 18, and 50), as implemented by SRP Sections 8.3.1 and 8.3.2, IEEE Std. 317-1983, and Regulatory Guides 1.32 and 1.63, establishes the requirements for the electrical penetrations.

Under SEP Topic VIII-4, the staff reviewed the electrical penetrations in the containment structure to ensure that they do not fail from electrical faults during a high-energy-line break. As part of the SEP, the staff performed an audit, comparing sample containment electrical penetrations with current licensing criteria for protection against fault and overload currents following a postulated accident.

The topic review showed that with a loss-of-coolant-accident environment inside the containment, the backup protection for some penetrations did not conform to current licensing criteria. However, as discussed in IPSAR Section 4.33, the staff concluded that no corrective measures were required because failure of the penetrations would not be a significant contributor to releases resulting from containment failure. Therefore, no backfit actions were required.

## 8.5 Appendix K - Electrical Instrumentation and Control Re-Reviews (SEP Topic VI-7.C.1)

10 CFR Part 50 (GDC 17), as implemented by SRP Sections 8.3.1 and 8.3.2 and Regulatory Guide 1.6, Position D.4, prohibits the switching of one safety load from one safety power supply to a second safety supply. A limited probabilistic risk assessment of automatic bus transfers (ABTs) between redundant power supplies was performed to determine this importance to risk.

(1) The ac system has seven ABTs of load groups between redundant sources. It is the staff's position that these ABTs should be removed or the circuits be otherwise modified to ensure that faulted loads will not be transferred.

The licensee agreed to perform a coordinated load and circuit breaker analysis to establish the corrective actions necessary to preclude automatic transfer of faults.

The affected breaker trip units were subsequently replaced by the licensee. This item is resolved as reported in IPSAR Supplement 1, Section 4.8.1.

(2) The 125-volt dc system has three ABTs of power between batteries.

The three dc ABTs are installed between batteries A and B. Battery A does not supply power to the safety systems. The redundant safety-related batteries are batteries B and C. There are no ABTs between batteries B and C. In IPSAR Section 4.25(2),the staff stated that backfitting to remove three ABTs between batteries A and B was not recommended. This issue is resolved.

# 9 AUXILIARY SYSTEMS

## 9.1 Fuel Storage (SEP Topic IX-1)

The purpose of the review under SEP Topic IX-1 is to evaluate the storage facility for new and irradiated fuel, including the cooling capability and seismic classification of the fuel pool cooling system of the spent fuel storage pool, in order to ensure that new and irradiated fuel is stored safely with respect to criticality, cooling capability, shielding, and structural capability.

The review of the structural response of the Oyster Creek plant with respect to seismic capability is presented in NUREG/CR-1981, "Seismic Review of the Oyster Creek Nuclear Power Plant as Part of the Systematic Evaluation Program." Although the spent fuel pool structure was not specifically evaluated during the seismic review, the overall conclusion was that the Oyster Creek plant structures and structural elements are adequately designed to withstand the postulated earthquake.

The staff reviewed the spent fuel pool modifications as described in Amendment 76 to POL DPR-16 (letter dated September 17, 1984). The staff determined that the safety evaluation supporting the amendment was performed in accordance with current licensing criteria. This review satisfies the aspects of Topic IX-1 relating to criticality and the structural capability of the storage racks.

The new fuel storage area is located in the reactor building. New fuel is stored dry in the fuel storage vault. The primary concern would be flooding of the storage area with the potential for inadvertent criticality.

The new fuel storage facility is designed to maintain $K_{eff} < 0.95$, even if the facility were filled with unborated water. In addition, the new fuel storage area is covered with concrete covers that would limit water leakage into the area. Leakage would be removed through a drain in the new fuel vault. The covers also protect the stored bundles from damage due to dropped objects.

On the basis of the above considerations, the staff concludes that the new fuel storage facility meets SRP Section 9.1.1.

By Amendment 76 dated September 17, 1984, the spent fuel pool storage capacity was increased from 1800 to 2600 fuel assemblies.

### 9.1.1 Spent Fuel Pool Cooling System - Seismic Upgrade

The original Oyster Creek spent fuel pool cooling system (SFPCS) was classified as a seismic Category I system. An augmented section of the SFPCS was also classified as a seismic Category I system.

The piping and supports in the augmented SFPCS were adequately designed to meet the seismic Category I design criteria, but the original SFPCS was inadequately designed for seismic loads.

The following modifications were made to upgrade the SFPCS in the reactor building:

(1) Six new supports and nine replacement pipe supports were added.

(2) A 6-inch manually operated gate valve was added where the original SFPCS connects with the augmented SFPCS.

(3) Seismic supports were added to the SFPCS head exchangers.

These modifications thus upgraded the original SFPCS from its non-seismic condition to a condition that would ensure that the pressure boundary would remain intact and functional.

In addition to the above modifications, several SFPCS valves were qualified for operability following a seismic event to ensure an isolated, seismically qualified cooling loop. The modification ensures that the equipment, valves, piping, and supports contained in the cooling loop meet operability criteria following a seismic event and that the boundary will remain intact and functional.

### 9.1.2 Control of Heavy Loads at Nuclear Power Plants (Generic Task A-36)

All plants have overhead handling systems that are used to handle heavy loads in the area of the reactor vessel or spent fuel in the spent fuel pool. Additionally, loads may be handled in other areas where if they are accidentally dropped, they may damage safe shutdown systems. Therefore, in accordance with NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants," dated July 1980 (Generic Task A-36), all plants should satisfy each of the following criteria for handling heavy loads that could be brought in proximity to or over safe shutdown equipment or irradiated fuel in the spent fuel pool area, in the reactor building, and in other plant areas.

(1) Safe load paths should be defined for the movement of heavy loads to minimize the potential for heavy loads, if dropped, to impact irradiated fuel in the reactor vessel and in the spent fuel pool, or to impact safe shutdown equipment. The path should follow, to the extent practicable, structural floor members, beams, etc., so that if the load is dropped, the structure is more likely to withstand the impact. These load paths should be defined in procedures, shown on equipment layout drawings, and clearly marked on the floor in the area where the load is to be handled. Deviations from defined load paths should require written alternative procedures approved by the plant safety review committee.

(2) Procedures should be developed to cover load-handling operations for heavy loads that are or could be handled over or in proximity to irradiated fuel or safe shutdown equipment. At a minimum, procedures should cover handling of those loads listed in Table 3-1 of NUREG-0612. These procedures should include identification of required equipment, inspections and acceptance criteria required before the load is moved, the steps and proper sequence to be followed in handling the load, defining the safe load path, and other special precautions.

(3) Crane operators should be trained and qualified and should conduct themselves in accordance with Chapter 2-3 of American National Standards Institute (ANSI) B30.2-1976, "Overhead and Gantry Cranes."

(4) Special lifting devices should satisfy the guidelines of ANSI N14.6-1978, "Standard for Special Lifting Devices for Shipping Containers Weighing 10,000 Pounds (4500 kg) or More for Nuclear Materials." This standard should apply to all special lifting devices that carry heavy loads in areas as defined above. For operating plants certain inspections and loads tests may be accepted in lieu of certain material requirements in the standard. In addition, the stress design factor stated in Section 3.2.1.1 of ANSI N14.6 should be based on the combined maximum static and dynamic loads that could be imparted on the handling device on the basis of the characteristics of the crane that will be used.* This is in lieu of the guideline in Section 3.2.1.1 of ANSI N14.6, which bases the stress design factor on only the weight (static load) of the load and of the intervening components of the special handling device.

(5) Lifting devices that are not specially designed should be installed and used in accordance with the guidelines of ANSI B30.9-1971, "Slings." However, in selecting the proper sling, the load used should be the sum of the static and maximum dynamic load.* The rating identified on the sling should be in terms of the "static load" that produces the maximum static and dynamic load. Where this restricts slings to use on only certain cranes, the slings should be clearly marked as to the cranes with which they may be used.

(6) The crane should be inspected, tested, and maintained in accordance with Chapter 2-2 of ANSI B30.2-1976, except that tests and inspections should be performed before use where it is not practicable to meet the frequencies of ANSI B30.2 for periodic inspection and test, or where the frequency of crane use is less than the specified inspection and test frequency.

(7) The crane should be designed to meet the applicable criteria and guidelines of Chapter 2-1 of ANSI B30.2-1976 and of CMAA-70, "Specifications for Electric Overhead Travelling Cranes" (Crane Manufacturers Association of America). An alternative to a specification in ANSI B30.2 or CMAA-70 may be accepted in lieu of specific compliance if the intent of the specification is satisfied.

A plant conforming to these seven guidelines will have developed and implemented, through procedures and operator training, safe load travel paths so that, to the maximum extent practicable, heavy loads are not carried over or near irradiated fuel or safe shutdown equipment. A plant conforming to these guidelines will also have provided sufficient operator training, handling-system design, load-handling instructions, and equipment inspection to ensure reliable operation of the handling system. It has been found that load-handling operations at Oyster Creek can be expected to be conducted in a highly reliable manner consistent with the staff's objectives as expressed in these guidelines.

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*For the purpose of selecting the proper sling, loads imposed by the safe shutdown earthquake need not be included in the dynamic loads imposed on the sling or lifting device.

NUREG-0612, Section 5.3, also lists certain measures that should be initiated to provide reasonable assurance that the handling of heavy loads will be performed in a safe manner until final implementation of the general guidelines of NUREG-0612 is complete. Specified measures include the implementation of a technical specification to prohibit the handling of heavy loads over fuel in the storage pool; compliance with Guidelines 1, 2, 3, and 6 identified above; a review of load-handling procedures and operator training; and a visual inspection program, including component repair or replacement as necessary of cranes, slings, and special lifting devices to eliminate deficiencies that could lead to component failure. The evaluation of information provided by the licensee indicates that Oyster Creek complies with the staff's measures for interim protection.

By Generic Letter 85-11 dated June 81, 1985, the staff concluded that the Oyster Creek station along with other plants has provided sufficient protection so that the risk associated with potential heavy-load drops is acceptably small and that the objective identified in Section 5.1 of NUREG-0612 for providing "maximum practical defense in depth" is satisfied.

## 9.2 Water Systems (SEP Topic IX-3)

Under SEP Topic IX-3, the staff reviewed the licensee's turbine building closed cooling water system, reactor building closed cooling water system, service water system, and emergency service water system to ensure that the systems have the capability to meet their design objectives and, in particular, to ensure the following:

(1) Systems are provided with adequate physical separation so that there are no adverse interactions among those systems under any mode of operation.

(2) Sufficient cooling water inventory has been provided, or adequate provisions for makeup are available.

(3) Tank overflow cannot be released to the environment without monitoring and unless the level of radioactivity is within acceptable limits.

(4) Vital equipment necessary for achieving a controlled and safe shutdown is not flooded as a result of the failure of the main condenser circulating water system.

On the basis of its review of the station service and cooling water systems for Oyster Creek, the staff concluded that the essential system and function are the emergency service water system for torus heat removal.

In a letter dated November 13, 1981, the staff determined that the design of the above system conforms with current regulatory guidelines and with GDC 44 regarding the capability and redundancy of the essential functions of the system.

## 9.3 Ventilation Systems (SEP Topic IX-5)

10 CFR Part 50 (GDC 4, 60, and 61), as implemented by SRP Sections 9.4.1, 9.4.2, 9.4.3, 9.4.4, and 9.4.5, requires that the ventilation systems have the capability to provide a safe environment for plant personnel and for engineered safety

features. In IPSAR Section 4.34, the staff found the ventilation systems at Oyster Creek acceptable except for the items discussed in the following sections.

### 9.3.1 Restoration of Ventilation

In IPSAR Section 4.34(1), the staff stated that operator action is required to restore reactor building, turbine building, and office building ventilation following a loss of offsite power. The licensee committed to review and modify, as required, the loss-of-offsite-power procedures to ensure that the operation of ventilation systems was adequately addressed and would not overload the diesel generators.

As discussed in IPSAR Supplement 1, Section 4.13.1, the licensee provides the necessary instructions for restoring emergency buses to service, if lost, in Station Procedure 341, "Emergency Diesel Generator Operation." However, in Region I Inspection Report 50-219/86-37 dated December 31, 1986, and followup discussions, the staff requested that the licensee identify the formal administrative configuration control for this procedure. In a teleconference, the licensee stated that Technical Functions Procedure EMP-014, Revision 3 (April 1987), which was formalized by Licensing Action Item 86334.06, includes the requested administrative control. Therefore, the staff considers this issue resolved. However, this issue will continue to be subject to NRC inspections.

### 9.3.2 Core Spray and Containment Spray Pump Ventilation

In IPSAR Section 4.34(3), the staff stated that with a loss of reactor building ventilation, the core spray and containment spray pump motors might not be adequately cooled during accident conditions. The licensee committed to demonstrate that these pump motors were qualified for the temperatures resulting from a loss of ventilation and submit the results to the staff.

In IPSAR Supplement 1, Section 2.18.1, the staff stated that, by letter dated September 1, 1983, the licensee had provided the requested evaluation, which stated that the core spray and containment spray pump motors, which are located in two corner rooms in the reactor building, are designed to function in environments with temperatures of up to 185°F and 203°F, respectively. Since the qualification temperatures are greater than the maximum expected temperature, the staff concluded that corner-room ventilation systems were unnecessary and, therefore, considered this issue resolved.

### 9.3.3 Battery, Motor Generator, and Switchgear Room Ventilation

In IPSAR Section 4.34(4), the staff identified a concern related to the susceptibility of both the B battery and motor generator room and the switchgear room ventilation systems to the single failure of a specific relay. A failure of that relay to transfer, or loss of power to that relay, would preclude electrical power to the fans of each room. The licensee agreed to evaluate the ventilation system design for the B battery and motor generator room and the consequences of a loss of ventilation in the switchgear room.

By letter dated August 21, 1984, the licensee provided the results of its review of the associated control circuitry for these ventilation systems. On the basis of this review, the licensee proposed to provide a new redundant relay with a switch and associated wiring for each room. With this modification, on loss of power or loss of the existing relay, the fans that receive power from a separate motor control center can be manually started, thereby maintaining sufficient air circulation. In addition, the licensee indicated that the loss of relay K will activate an alarm in the control room to alert the operator to activate the ventilation system with the redundant switch. In IPSAR Supplement 1, Section 2.18.2, the staff reported that this issue was resolved.

## 9.4 Fire Protection

Following a fire at the Brown's Ferry Nuclear Power Station in March 1975, the NRC initiated an evaluation of the need for improving the fire protection programs at all licensed nuclear power plants. As part of this continuing evaluation, the NRC, in February 1976, published the report by a special review group entitled "Recommendations Related to Browns Ferry Fire," NUREG-0050. This report recommended that improvements in the areas of fire prevention and fire control be made in most existing facilities and that consideration be given to design features that would increase the ability of nuclear facilities to withstand fires without the loss of important functions. To implement the report's recommendations, the NRC initiated a program for reevaluation of the fire protection programs at all licensed nuclear power stations and for a comprehensive review of all new licensee applications. The NRC issued new guidelines (Branch Technical Position (BTP) ASB 9.5-1, May 1976, and BTP ASB 9.5-1, Appendix A, November 1976 (NUREG-0800)) for fire protection programs in nuclear power plants that reflected the recommendations in NUREG-0050. All licensees were requested to (1) compare their fire protection programs with the new guidelines and (2) analyze the consequences of a postulated fire in each plant area.

The staff reviewed the licensee's analyses and visited the plant to examine the relationship of safety-related components, systems, and structures to both combustible materials and the associated fire detection and suppression systems. The staff's review of the fire protection program was documented in an SER dated March 3, 1978, as supplemented on June 29 and November 13, 1979, and August 25, 1980.

In February 1981, the Fire Protection Rule (10 CFR 50.48 and Appendix R, "Fire Protection Program for Nuclear Power Facilities Operating Prior to January 1, 1979," to 10 CFR Part 50) became effective.

The licensee provided an evaluation of the

(1) fire protection water system
(2) gas fire suppression system
(3) portable fire extinguishers
(4) fire detection and signaling system

The evaluation of the fire protection system by area and zone was incorporated into the revised Fire Hazards Analysis (FHA) and was submitted to the NRC staff as part of the 10 CFR Part 50, Appendix R evaluation on June 30, 1982. (Revision 2 of the FHA was submitted to the staff on May 3, 1984; Revision 3 on April 3, 1985; Revision 4 on July 12, 1985; and Revisions 5 and 6 on August 25, 1986.)

In a safety evaluation dated March 24, 1986, of the Oyster Creek alternate safe shutdown facility design, the staff concluded that the performance goals for accomplishing safe shutdown in the event of a fire (i.e., reactivity control, inventory control, decay heat removal, pressure control, process monitoring, and support functions) were met by the proposed alternate safe shutdown facility. Therefore, the staff concluded that the requirements of Appendix R, Sections III.G.3 and III.L, were satisfied.

The alternate safe shutdown facility was installed during the Cycle 11 refueling outage.

POL Amendments 29 (March 3, 1978), 58 (December 21, 1981), 85 (June 17, 1985) 89 (July 2, 1985), 101 (April 7, 1986), and 114 (March 20, 1987) have dealt with the Oyster Creek fire protection and shutdown systems. In the safety evaluation accompanying the most recent of these amendments, the staff concluded that the Oyster Creek fire protection design and Technical Specifications governing the associated equipment continue to be acceptable.

# 10 STEAM AND POWER CONVERSION SYSTEM

## 10.1 System Design

The major components of the steam and power conversion system are main steam supply lines, turbine generator, moisture separators and reheaters, main condenser, condensate pumps, steam jet air ejectors, turbine bypass valves, condensate demineralizers, reactor feedwater pumps, feedwater heaters and drain coolers, condensate storage tank, and condensate transfer pumps. The heat rejected by the main condenser is removed by the circulating water system. The major components of the steam and power conversion system are located in the turbine building and are not safety related.

Steam from the main steamlines supplies the turbine.

The saturated steam passes through the high-pressure stages of the turbine where it expands and is then exhausted to the moisture separators and then to the reheaters. The moisture separators remove the moisture content of the steam, and the first-stage and second-stage reheaters superheat the steam before it enters the low-pressure stages, where the steam expands further. From the low-pressure stages, the steam is exhausted into the main condenser, where it is condensed and dearated and then returned to the cycle as condensate.

Under normal operations, a small part of the main steam supply is continuously used by the steam jet air ejectors (SJAEs), the steam seal regulator, and the second-stage reheaters. The condensate pumps take suction from the condenser hotwell and deliver the condensate to the low-pressure drain coolers and the low-pressure and intermediate-pressure feedwater heaters, via the condensate demineralizers. Condensate from the discharge of the condensate pumps is also used as a condensing medium in the SJAE condensers and in the steam packing exhauster condenser. The reactor feedwater pumps supply feedwater through one stage of the high-pressure feedwater heaters to the reactor. Steam for heating the feedwater and the first-stage reheaters is extracted from the turbine. The feedwater heaters also provide the means of handling the moisture separated from the steam in the moisture separators, and the condensate from the first-stage and second-stage reheaters.

Normally, the turbine utilizes all the steam being generated by the reactor. However, under certain operating transients, excess steam is generated. An automatic pressure-controlling turbine bypass system is provided to discharge excess steam up to 40 percent of the turbine steam flow at design power level directly to the main condenser. The turbine bypass system is designed to control pressure by dumping excess steam during startup, shutdown, and power operation, when the reactor steam generation exceeds the transient turbine steam requirements.

The feedwater piping delivers water through two check valves (one inside and one outside the containment) to the feedwater sparger within the annular region (downcomer) of the reactor. This water mixes with the recirculation water and then is delivered through the recirculation loop.

The steam piping is designed to ensure correct steam distribution and pressure to all steam-consuming equipment for all turbine loads. In addition, the steam and feedwater lines with their supports and structures to their respective isolation valves are seismic Category I.

The main steamlines have five electromatic relief valves that provide pressure relief for the primary system. These relief valves operate automatically on high reactor pressure, as the automatic depressurization system (part of the emergency core cooling system), or manually.

Each steamline is equipped with two fast-closing isolation valves, one inside and one outside the containment. The main steam isolation valves are closed automatically by signals indicative of a steamline failure. They may also be closed manually.

During normal and accident conditions, restricted areas and shielding around selected components in the system will protect plant personnel from exposures above established limits.

A full-flow condensate demineralizer system removes corrosion products and condensate impurities to minimize the effects of crud deposition on critical components in the reactor system. This system consists of seven mixed-bed units, cation regeneration tank, anion regeneration tank, resin storage tank, recycle pump, and required piping, valving, instrumentation, and controls.

The performance of the turbine generator and the effects of failures of components on the rest of the plant have been evaluated in transient analyses included in Chapter 15 of the Final Safety Analysis Report. The following transients have been analyzed:

(1) loss of electrical load
(2) turbine trips (1930 magawatts-thermal (MWt) and 1025 MWt)
(3) loss of main condenser vacuum
(4) inadvertent opening of a turbine bypass valve
(5) loss of feedwater
(6) one feedwater pump trip and restart
(7) excess feedwater flow

The steam and power conversion system is part of the Oyster Creek design originally licensed to operate on April 9, 1969.

## 10.2 Main Steam Isolation Valve Leakage

The ability of main steam isolation valves (MSIVs) to close, seat securely, and restrict leakage to within the limits assumed in design-basis-accident (DBA) scenario analyses is verified by 10 CFR Part 50, Appendix J, Type C MSIV testing. In a meeting on February 13, 1989, the licensee verified that this testing had been performed with control air pressure applied to the MSIV actuator. The normal instrument air/nitrogen system supplying this air pressure is non-safety grade and is assumed to be unavailable under DBA conditions. The staff therefore postulated that the tests as performed might not be prototypic of the DBA scenarios for which the valves were being tested.

In a letter dated March 10, 1989, the licensee submitted test and analysis results for a case in which no pressure was applied to the MSIV actuator. From these results the licensee concluded that applicable leakage limits would be met.

On the basis of the discussion at the meeting on February 13, 1989, and in consideration of the justification submitted by the licensee in the letter dated March 10, 1989, the staff concludes that the Oyster Creek plant may be operated without significant risk to the health and safety of the public, pending the ongoing review of this issue.

# 11 RADIOACTIVE WASTE MANAGEMENT

In a letter dated June 1, 1973, the licensee informed the staff of the completion of an evaluation of the radioactive waste management systems installed at the Oyster Creek station to determine the performance of these systems with respect to proposed Appendix I to 10 CFR Part 50, and to evaluate means of modifying the existing radioactive waste management systems so that releases of radioactivity from the modified systems were as low as practicable.

The modifications stemming from this evaluation that were subsequently incorporated into the radioactive waste management systems are discussed below.

## 11.1 Augmented Offgas System

The augmented offgas (AOG) system installed at Oyster Creek can reduce radioactive gaseous waste emissions to levels in compliance with 10 CFR Part 50, Appendix I, by decreasing the condenser offgas emissions from 260,000 microcuries per second after a 30-minute delay to less than 1700 microcuries per second.

Condenser offgas leaving the plant's delay pipe is routed to a new building approximately 240 feet east of the stack. Radiolytic hydrogen and oxygen in the main condenser offgas stream are catalytically combined and condensed, reducing the design-basis process flow from 170 standard cubic feet per minute (scfm) to 20 scfm. The offgas is then dried and passed through a series of charcoal beds where iodine isotopes are completely removed; xenon isotopes are delayed at least 20 days and krypton isotopes are delayed at least 22.6 hours.

Active components, including hydrogen recombiners and water removal subsystems, are redundant to ensure maximum availability and reliability of the overall system. Doses due to postulated accidents have been limited by a design that allows isolation of the condenser within 15 minutes of abnormally high radiation levels in the offgas system piping upstream of the 30-minute holdup line. Failures in the AOG system result in immediate isolation of this system. Main condenser offgas will continue to discharge through the stack.

The new offgas building is a two-story, nonseismic building erected at grade. It is fabricated of a structural steel framework with a poured concrete foundation, intermediate slabs, and a roof slab. The building walls that also serve as shield walls are constructed of solid concrete blocks. Other walls are constructed of insulated metal siding. The general arrangements of the building have been developed to ensure minimum exposure to operators and maintenance personnel.

The new offgas building is provided with its own heating and ventilating system. Other auxiliary systems, including the demineralized water system, drains, and the instrument air and fire protection systems, are interconnected with the existing plant systems. A new once-through cooling system, using existing plant intake and discharge facilities, is provided to service both the offgas building and the new liquid/solids radwaste building.

## 11.2 Liquid/Solids Radwaste System

The redesigned liquid/solids radwaste system is housed in a new three-story building, 44 feet high, 86 by 114 feet in plan dimension, that is erected at grade approximately 250 feet north-northwest of the existing plant stack. The building is fabricated of a structural steel framework with a poured reinforced concrete foundation, intermediate slabs, and a roof slab. The shield walls are constructed of solid concrete blocks, and other walls of insulated metal siding. The physical appearance of the building is consistent with the remainder of the plant structures. Electrical and piping connections to the existing plant are via an underground concrete tunnel.

Design features of the new system, which permanently correct the major problems with the original system, include substantially expanded system capacities, segregation of high-purity and chemical waste/floor drain systems, complete redundancy of liquid waste trains to permit maintenance without interruption of system processing, the use of separate shielded compartments for all major components and shielded valve galleries to minimize operators' exposure to radiation, and the use of advanced state-of-the-art components throughout.

The liquid/solids radwaste system has been designed to process low-level radioactive liquid wastes produced as a byproduct of plant operation. The system processes this water to make it suitable for recycling within the plant or for release to the environment. The material removed from the processed liquids and spent chemicals from the processing is solidified or dewatered and packaged for disposal off site. The liquid/solids radwaste treatment in the Oyster Creek plant consists of a number of segregated waste streams.

(1) High-purity waste is reactor coolant that is collected from various points in the plant as a result of equipment leakage, drainage, and process waste produced as a result of plant operations. This water is chemically clean and has a low mineral content. It is filtered, demineralized to lower the radioactivity level, and returned to the reactor coolant system (when possible) or released to the environment.

(2) Chemical waste/floor drain waste has a relatively high mineral content and/or high suspended-matter content. It also varies in its pH levels. Sources of this waste are demineralizer resin rinses, decontamination of equipment with non-detergent solutions, laboratory drains, and floor drains and sumps. The waste is neutralized, filtered, evaporated, demineralized as required, and cycled through the high-purity system for additional processing.

(3) Solidification of waste is a process by which the radioactive waste that has been separated from the processing streams is solidified, in accordance with a process control program, using cement. The waste comes from filters, exhausted demineralizer resins, and evaporator bottoms. The solidified end product is encased in a shipping container and transported off site for disposal. Exhausted resins are dewatered in a lined shipping container and transported from the site for disposal.

The liquid/solids radwaste building is provided with its own heating and ventilating system. In addition, the building has a floor drain system that is connected directly to the new processing system. Other auxiliary systems, including the demineralized water, instrument air, and fire protection systems, are interconnected with the existing plant systems.

## 11.3 Turbine Building Radioactive Gaseous Effluent Monitoring System

The primary safety function of the turbine building radioactive gaseous effluent monitoring system (RAGEMS II) is to monitor releases of radioactive noble gases.

Operational requirements governing the RAGEMS are provided in the Plant Radwaste Emmissions Technical Specifications (RETS). In the most recent amendment to RETS (Amendment 108 dated October 6, 1986), the staff concluded that (1) the licensee's proposed RETS meet the intent of NUREG-0473, "Radiological Effluent Technical Specifications (RETS) for Boiling Water Reactors"; (2) the licensee's Offsite Dose Calculation Manual uses documented and approved methods that are consistent with the criteria of NUREG-0133, "Preparation of Radiological Effluent Technical Specifications for Nuclear Power Plants," and the Oyster Creek plant and site; and (3) the licensee's commitment to implement a process control program (PCP) - the licensee refers to the PCP as the Process Control Plan - to ensure proper processing and packaging of solid radwaste before shipment off site meets the intent of NUREG-0473.

# 12 RADIATION PROTECTION

The radiation protection measures incorporated at Oyster Creek are intended to ensure that internal and external radiation exposures to station personnel, contractor personnel, and the general population resulting from station conditions, including anticipated operational occurrences, will be within applicable limits and, furthermore, will be as low as is reasonably achievable (ALARA).

The basis for staff acceptance of the Oyster Creek Radiation Protection Program is that doses to personnel will be maintained within the limits of 10 CFR Part 20 and that the radiation protection designs and program features are also consistent with the guidelines of Regulatory Guide 8.8, "Information Relevant to Ensuring That Occupational Radiation Exposures at Nuclear Power Stations Will Be As Low As Is Reasonably Achievable," Revision 3. Shielding is provided to reduce levels of radiation. Ventilation is arranged to control the flow of potentially contaminated air. Radiation monitoring systems are used to measure levels of radiation in potentially occupied areas and airborne radioactivity throughout the plant. A health physics program is provided for plant personnel and visitors during reactor operation, maintenance, refueling, radwaste handling, and inservice inspection.

The staff concludes that these and other radiation protection features can help ensure that occupational radiation exposures are maintained ALARA during plant operation and during decommissioning. The staff periodically reviews the licensee's Radiation Protection Program during routine onsite inspections. In SALP Report 50-219/87-99, the staff noted a degradation of performance in the area of radiological controls, as discussed in Section 13.1. By letter dated October 31, 1989, the staff requested further information on the licensee's initiatives to correct the deficiencies noted and the implementation status of those initiatives. The staff will review this information when it is submitted and take appropriate regulatory action, if needed.

# 13 CONDUCT OF OPERATIONS

## 13.1 Organizational Structure

Since the start of commercial operation in December 23, 1969, there have been a number of organizational changes, as delineated in amendments to the provisional operating license.

Amendment 40, dated August 6, 1979, modified administrative controls concerning the supervision of the refueling and facility organization.

Amendment 64, dated October 28, 1982, revised the Oyster Creek administration; that is, the title of the position of Director Station Operations was changed to Deputy Director - Oyster Creek, and his Technical Specification responsibilities were shared with the Vice President and Director - Oyster Creek.

Amendment 68, dated September 28, 1983, revised the plant organization. The titles of several positions were changed, and the responsibility for corrective maintenance was transferred to the Maintenance and Construction Department.

Amendment 69, dated January 12, 1984, added the position of Maintenance and Construction Director - Oyster Creek.

Amendment 78, dated December 27, 1984, enhanced the administrative capabilities of the plant engineering organization, upgraded the manager's position to Radiological Controls Director, updated the requirements for written procedures, and added special reporting to the NRC, as required by NUREG-0737.

Amendment 92, dated November 19, 1985, specified requirements pertaining to limiting the overtime of station personnel.

Amendment 102, dated May 12, 1986, revised the staffing requirement pertaining to the minimum number of operators in the control room.

Amendment 117, dated September 30, 1987, authorized changes to the GPUN corporate and Oyster Creek site organizations shown in FSAR Figures 13.1-1 and 13.1-2.

On March 22, 1988, the licensee submitted a proposal to amend the Oyster Creek organizational structure and to relocate documentation of the structure from the Technical Specifications to Chapter 13 of the updated Final Safety Analysis Report. Removal of organizational charts from the Technical Specifications is consistent with the guidance in Generic Letter 88-06, "Removal of Organizational Charts From Technical Specification Administrative Control Requirements." Other aspects of this proposed amendment are still under staff review.

The staff periodically reviews the licensee's operating performance under the Systematic Assessment of Licensee Performance (SALP) program. The SALP program is an integrated NRC staff effort to collect available observations and data on a periodic basis and to evaluate the licensee's performance on the basis of this information. The program is supplemental to normal regulatory processes to ensure compliance with NRC rules and regulations. In the most recent SALP report, No. 50-219/87-99 dated April 17, 1989, the staff reviewed the performance of activities at Oyster Creek for the period October 1, 1987, to January 31, 1989.

## 13.2 Training

As stated in Section 6.4 of the Oyster Creek Technical Specifications (TS), a retraining program for operators and replacement training programs are maintained for the facility. These programs were formulated to meet the requirements and recommendations of Appendix A to 10 CFR Part 55. In 1987, 10 CFR Part 55 was revised to incorporate Appendix A into 10 CFR 55.59. This revision to 10 CFR Part 55 endorsed Regulatory Guide 1.8, Revision 2, "Qualification and Training of Personnel for Nuclear Power Plants."

In a letter dated October 11, 1989, the licensee proposed a revision to TS 6.4 to reflect the revised 10 CFR Part 55. This proposal is under NRC staff review. By letter dated October 20, 1989, the staff requested that the licensee commit to meet Regulatory Guide 1.8, Revision 2, as endorsed in the 1987 change to 10 CFR Part 55. A training program for the fire brigade is also maintained.

Amendment 53, dated February 11, 1981, incorporated the Guard Training and Qualification Plan into the provisional operating license.

## 13.3 Emergency Planning

NUREG-0737 identified Item III.A.2.1 as "Emergency Preparedness, Upgrade Emergency Plans to Appendix E, 10 CFR 50." It also stated that a licensee's emergency plan and submittals of procedures were due January 2 and March 1, 1981, respectively, and that the onsite emergency preparedness program was to be implemented by April 1, 1981. Upgraded emergency plans and procedures have been received, and the emergency preparedness program has been implemented.

The emergency plan developed for Oyster Creek is in accordance with the provisions of 10 CFR 50.47 and Appendix E to 10 CFR Part 50 and is consistent with the guidelines in "Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants," NUREG-0654/FEMA-Rep-1, Revision 1, November 1980. Other guidance and sources of information used in the development of the emergency plan have been identified in Section 10.0 of the plan.

During 1982, the NRC staff conducted a comprehensive 2-week onsite emergency appraisal at the facility, during which deficiencies and items needing improvement were identified; a subsequent inspection report was issued by letter dated June 11, 1982. The licensee has taken corrective actions based on the findings in the report. The staff has performed inspections of the licensee's emergency preparedness program annually since the initial appraisal. Emergency exercises involving licensee personnel were conducted annually at Oyster Creek in 1982 through 1989. The licensee was informed of areas needing improvement in exercise reports and has implemented satisfactory corrective actions.

The NRC regional office will continue to verify the status of onsite emergency preparedness at Oyster Creek through inspections of the emergency preparedness program and the observation of annual exercises.

The evaluation of the status of offsite preparedness by the Federal Emergency Management Agency (FEMA) is a continuing process involving review of State and local plans and the observations of full-participation exercises. Deficiencies identified in FEMA exercise reports have been satisfactorily resolved. FEMA has concluded that offsite radiological emergency preparedness for Oyster Creek is adequate to provide reasonable assurance that appropriate measures can be taken to protect the health and safety of the public in the event of a radiological emergency. Consideration of population distribution in the area of the plant as implemented in emergency planning is discussed in Section 2.1.2.

## 13.4 Review and Audit

The functions, composition, and responsibilities of those organizations responsible for performing the nuclear safety review and audit of the Oyster Creek station are delineated in Section 6.5 of the Technical Specifications.

Amendment 69 to the provisional operating license (POL), dated January 12, 1984, implemented a new Safety Review and Audit Program.

Amendment 67 to the POL, dated March 31, 1983, increased the frequency of auditing the emergency and security plans to every 12 months.

## 13.5 Plant Procedures

As directed by the Technical Specifications, written procedures have been established, and are implemented and maintained, to meet or exceed the requirements of Sections 5.1 and 5.3 of American National Standards Institute Standard N18.7-1976 and Appendix A to Regulatory Guide 1.33, 1972, except as noted in Section 6.8 of the Oyster Creek Technical Specifications.

## 13.6 Physical Security Plan

The staff has reviewed the physical security, guard training and qualification, and safeguards contingency plans against the requirements of 10 CFR 73.55(b) through (h) and approved them on the basis of the acceptance criteria in effect at the time of the review. Each of the plans has subsequently been revised by the licensee under the provisions of 10 CFR 50.54(p).

As required by the Commission's regulations, the physical security plan was implemented on May 17, 1988, the contingency plan on June 24, 1986, and the guard training and qualification plan on May 17, 1988.

Amendment 127 to the POL, dated October 11, 1988, modified the plan to conform with the requirements of 10 CFR 73.55. On the basis of its review of the plan, the staff concluded that the plan meets the revised miscellaneous amendments and search requirements of 10 CFR 73.55 and the recordkeeping requirements of 10 CFR 73.70 and is therefore acceptable.

# 14 INITIAL TEST PROGRAM

During preoperational testing, the Oyster Creek station was subjected to a series of startup tests at 1600 and 1690 megawatts-thermal (MWt) and later at the full design rating of 1930 MWt. See Chapter 14 of the Final Safety Analysis Report (FSAR) for a detailed description of the tests.

The test program was intended to demonstrate that plant systems, structures, and components would perform in a manner that would not endanger the health and safety of the public. The principal objectives of the program were to ensure, to the maximum extent practicable, that

(1) the plant had been properly designed and constructed and was capable of operating safely at performance levels specified in the FSAR

(2) the plant operating and emergency procedures had been verified by trial use to be adequate

(3) the plant operating and technical personnel were knowledgeable about the plant equipment and procedures and were prepared to operate the facility in a safe manner

Preoperational tests were initiated approximately 6 months before initial fuel loading. Initial fuel loading started on April 10, 1969, and was completed 2½ weeks later, on April 28, 1969. Initial criticality was achieved at 2:17 p.m. on May 3, 1969, and low-power physics testing was completed soon thereafter. With the completion of the testing program and the 100-hour warranty run, commercial operation at 530 MW (electrical, net) began on December 23, 1969.

On May 7, 1970, an application for an increase in licensed thermal power level from 1600 MWt to 1690 MWt was filed with the Atomic Energy Commission. The request was granted on December 2, 1970. Several reactor protection system setpoints were changed to accommodate the new power level, and an anticipatory trip was added that would cause an immediate scram when a turbine trip or generator-load rejection was sensed.

The startup test program at 1690 MWt was divided into five phases: preoperational testing, open-vessel testing with fuel installed, plant heatup, power testing, and warranty run. The program was established by sequential tests that proved the plant design and operation in sequential steps up to licensed-power operation, each step providing assurance that it was safe to proceed to the next step in the sequence until licensed power was attained. The first phase, preoperational testing, was completed for each system before the system was required for safe and proper plant operation. During the remaining four phases, a series of tests was performed, some of which were repeated several times during the program at different operating conditions.

The test program at the increased power rating of 1690 MWt demonstrated the stability of the plant and the acceptability of the core's performance at this higher power density.

On January 26, 1971, an application for an increase in the licensed thermal power level from 1690 MW to the full design power of 1930 MW was filed, and was granted on November 5, 1971. Again, reactor protection system setpoints were changed to accommodate the higher power level, and a fifth electromatic relief valve was installed to mitigate the transient pressure increase should a turbine trip occur during bypass valve action.

The full design power test program consisted of three phases. The first phase was designed to obtain a good set of base point data at the original licensed rating of 1600 MWt immediately before power was increased. These data were compared with the data collected as power was increased so that changes that occurred could be clearly attributed to the increase in power and not to some long-term effect associated with plant operation since the startup test program at 1600 MW. The second phase consisted of several tests at an intermediate power level of 1765 MWt. These tests verified proper operation at this level before proceeding to the full design rating. The final phase consisted of the tests at the full design rating of 1930 MWt and was designed to verify core performance and plant stability at this level. The tests performed during this program were identical to those performed during the initial startup test program described in GE Topical Report 22A2130 (see FSAR Appendix 14.24).

The full design power test program was designed to demonstrate the stability of the plant and the acceptability of core performance at a core thermal power of 1930 MW. The startup test programs at these different power levels were essentially similar, except that procedures for the test were slightly modified in some cases.

In its safety evaluation dated November 5, 1971, the staff concluded that the results of the initial plant test program met required acceptance criteria and that the completion of the program demonstrated the functional adequacy of structures, systems, and components.

# 15 ACCIDENT ANALYSES

## 15.1 Systematic Evaluation Program Reevaluations

As part of the Systematic Evaluation Program (SEP), the staff reevaluated the ability of Oyster Creek to withstand normal and abnormal transients and a broad spectrum of postulated accidents without undue hazard to the health and safety of the public. The results of these analyses are used to show conformance with General Design Criteria (GDC) 10 and 15 of Appendix A to 10 CFR Part 50.

During its review of the transients and accident analyses of Section 15 of the updated Final Safety Analysis Report, the staff has considered GDC 21, 27, and 28 and Regulatory Guides 1.53 and 1.105 as they apply to the events analyzed to ensure that the applicable requirements have been met.

For each event analyzed the worst operating conditions were assumed and credit was taken for minimum engineered safeguards response. Parameters specific to individual events were conservatively selected.

Two types of events were analyzed:

(1) those incidents that might be expected to occur during the lifetime of the reactor (anticipated transients)

(2) those incidents not expected to occur that have the potential to result in a significant release of radioactive material (accidents)

The events reviewed by the staff and their corresponding SEP numbers are the following:

| SEP number | Title |
|---|---|
| XV-1 | Decrease in Feedwater Temperature, Increase in Feedwater Flow, Increase in Steam Flow, and Inadvertent Opening of a Steam Generator Relief or Safety Valve |
| XV-3 | Loss of External Load, Turbine Trip, Loss of Condenser Vacuum, Closure of Main Steam Isolation Valve (BWR), and Steam Pressure Regulator Failure (Closed) |
| XV-4 | Loss of Nonemergency AC Power to the Station Auxiliaries |
| XV-5 | Loss of Normal Feedwater Flow |
| XV-7 | Reactor Coolant Pump Rotor Seizure and Reactor Coolant Pump Shaft Break |
| XV-8 | Control Rod Misoperation (System Malfunction or Operator Error) |

XV-9 Startup of an Inactive Loop or Recirculation Loop at an Incorrect Temperature, and Flow Controller Malfunction Causing an Increase in BWR Core Flow Rate

XV-11 Inadvertent Loading and Operation of a Fuel Assembly in an Improper Position (BWR)

XV-13 Spectrum of Rod Drop Accidents (BWR)

XV-14 Inadvertent Operation of Emergency Core Cooling System and Chemical and Volume Control System Malfunction That Increases Reactor Coolant Inventory

XV-15 Inadvertent Opening of a PWR Pressurizer Safety/Relief Valve or a BWR Safety/Relief Valve

XV-16 Radiological Consequences of Failure of Small Lines Carrying Primary Coolant Outside Containment

XV-18 Radiological Consequences of Main Steam Line Failure Outside Containment (BWR)

XV-19 Loss-of-Coolant Accidents Resulting From Spectrum of Postulated Piping Breaks Within the Reactor Coolant Pressure Boundary

XV-20 Radiological Consequences of Fuel-Damaging Accidents (Inside and Outside Containment)

During its reevaluation of the above events, the staff noted two modifications that were necessary.

Under SEP Topic XV-16, the staff reviewed the radiological consequences of failure of small lines carrying primary coolant outside the containment. The staff concluded that reactor coolant activity should be maintained within the limits imposed in the BWR Standard Technical Specifications (NUREG-0123). This will ensure that the radiological consequences of an event that results in release of reactor coolant to the environment will be low.

As part of the review under SEP Topic XV-19, the staff evaluated the radiological consequences of a loss-of-coolant accident. The staff's independent analyses of calculated offsite doses showed that the major contributor was from main steamline isolation (MSIV) valve leakage. Therefore, the staff concluded that the licensee should develop and implement a preventive maintenance program aimed at minimizing MSIV leakage. The licensee is participating in an owners group program to resolve this issue.

On the basis of the SEP topic safety evaluations of the transients listed above, the staff concludes that the analyses demonstrate that the operation of the plant will not result in any violation of fuel design or reactor coolant pressure boundary design limits, that the plant design conforms with GDC 10 and 15, and that the analyses are, therefore, acceptable. Additionally, the staff concludes that the licensee has provided adequate protection systems to mitigate accidents in compliance with GDC 10, 15, and 20 and 10 CFR Parts 50 and 100.

## 15.2 Anticipated Transients Without Scram (ATWS) (Generic Task A-9)

Nuclear plants have safety and control systems to limit the consequences of temporarily abnormal operating conditions or "anticipated transients." Some deviations from normal operating conditions may be minor; others, occurring less frequently, may impose significant demands on plant equipment. In some anticipated transients, rapidly shutting down the nuclear reaction (initiating a "scram"), and thus rapidly reducing the generation of heat in the reactor core, is an important safety measure. If there were a potentially severe anticipated transient and the reactor shutdown system did not scram as desired, then an "anticipated transient without scram," or ATWS, would occur.

WASH-1270, "Technical Report on Anticipated Transients Without Scram for Water-Cooled Power Reactors," discusses the probability of an ATWS event and an appropriate safety objective for these events. Following review of vendor reports describing the analysis models and results, the NRC staff published, in late 1975, its status report on each vendor analysis, including detailed guidelines on analysis models and ATWS safety objectives ("Status Report on Anticipated Transients Without Scram for General Electric Reactors," December 9, 1975).

Since the publication of the 1975 status report, additional information relevant to ATWS has been developed by the industry and the Reactor Safety Study Group. On the basis of its review of these reports and discussions with vendors, the NRC staff published "Anticipated Transients Without Scram for Light-Water Reactors," NUREG-0460, Volumes 1 and 2, in April 1978. Since the issuance of Volumes 1 and 2, additional safety and cost information and new insights have been developed on the general subject of quantitative risk assessment. On the basis of these considerations, the NRC staff issued a new report, Volume 3 to NUREG-0460, dated December 1978. In Volume 3 various alternative plant modifications for ATWS ranging from none to those needed to satisfy the proposed licensing criteria for new plants in NUREG-0460, Volumes 1 and 2, were considered. The staff assessed the corresponding degrees of assurance of safety achieved by these alternative modifications. In Volume 3, the staff also suggested plant modifications on the basis of the plant design and age. To confirm its judgment on the adequacy of these designs, the staff issued requests for industry to supply the necessary generic analyses. In NUREG-0460, Volume 4, issued in March 1980 for public comment, the staff reviewed the industry responses and concluded that the necessary verification of the adequacy of the proposed design changes had not been provided. The staff, therefore, proposed that early improvements in safety should be provided, and any additional requirements should be considered under the staff's recommended rulemaking. The staff reviewed the comments of industry and the Advisory Committee on Reactor Safeguards in Volume 4 and published a proposed rule for resolution of the ATWS issue in the Federal Register (45 FR 73080).

Subsequently, the Commission issued the ATWS rule (10 CFR 50.62, "Requirements for Reduction of Risk From Anticipated Transients Without Scram (ATWS) Events for Light-Water-Cooled Nuclear Power Plants"). This rule requires improvements in the design and operation of commercial nuclear power facilities to reduce the likelihood of failure to shut down the reactor following anticipated transients and to mitigate the consequences of an ATWS event. The requirements for a boiling-water reactor are to have an alternate rod injection (ARI) system and a standby liquid control system (SLCS), and to trip the reactor coolant recirculation pumps automatically under conditions indicative of an ATWS. The Oyster Creek SLCS and its conformance with the equivalent control capacity requirements of 10 CFR 50.62 are discussed in Section 6.4.

The licensee for Oyster Creek provided information by letters dated September 3 and December 30, 1987, and April 29, 1988, concerning its implementation of the ATWS rule. The staff and its consultant reviewed these submittals. The staff found the licensee's SLCS acceptable (see Section 6.4).

On the basis of this review, the staff concluded in a letter dated November 4, 1988, that the design of the ARI system does not meet the diversity requirements of the ATWS rule. However, the staff does not believe that the issues related to this nonacceptance are of sufficient safety significance to delay implementation of the ARI system or to replace equipment already installed. To comply with the ATWS rule, the staff required that the licensee provide an ARI system with instrument components that are diverse from the reactor trip system before restart following the next refueling outage (Cycle 13 refueling outage, September 1990). The licensee was required to perform a preoperational test to verify that the actual ARI function time is within the design limit. The licensee also was required to provide equipment technical specifications including operability and surveillance requirements.

As stated in Inspection Report 50-219/89-19 dated September 14, 1989, Section 3.2, the licensee performed the required preoperational testing, which verified that insertion of all control rods was started and completed within 25 seconds of ARI system initiation, which was within the design limit. This satisfies the preoperational test requirement. Subject to the submittal of appropriate technical specifications for this equipment, this issued is resolved for Oyster Creek.

# 16 TECHNICAL SPECIFICATIONS

The Technical Specifications in a license define certain features, characteristics, and conditions governing the operation of a facility that cannot be changed without prior approval of the staff. The current Technical Specifications for Oyster Creek are part of the provisional operating license and will be made a part of the full-term operating license. Included are sections covering definitions, safety limits, limiting safety settings, limiting conditions for operation, surveillance requirements, design features, and administrative controls.

In the course of the staff's review of the individual SEP topics, the Oyster Creek Technical Specifications were compared with the Standard Technical Specifications for deviations. Where significant differences existed, they were identified and the staff considered them for upgrading. Table 4.1 of the IPSAR (NUREG-0822) and Table 2.1 of IPSAR Supplement 1 identify those items for which Technical Specification modifications are required. The other sections of the Technical Specifications are reviewed only to the extent that reloads, license amendments, or generic problems require.

## 17 QUALITY ASSURANCE

The quality assurance organization is responsible for ensuring that procedures and instructions comply with complete and adequate quality assurance requirements. In addition, quality assurance personnel should perform sufficient reviews, inspections, and audits to verify the effective implementation of the entire quality assurance program.

The licensee has structured its quality assurance program for the operational phase so that it is in accordance with Appendix B to 10 CFR Part 50 and complies with the regulatory positions given in quality assurance-related regulatory guides and with the requirements of American National Standards Institute (ANSI) N45.2.12. The quality assurance program is implemented by means of written policies, procedures, and instructions. These documents result in control of quality-related activities involving safety-related items in accordance with the requirements of Appendix B to 10 CFR Part 50 and with applicable regulations, codes, and standards.

The licensee's quality assurance program requires that implementing documentation encompass detailed controls for (1) indoctrinating and training personnel; (2) translating codes, standards, regulatory requirements, technical specifications, engineering requirements, and process requirements into drawings, specifications, procedures, and instructions; (3) developing, reviewing, and approving procurement documents, including changes; (4) prescribing all quality-related activities by documented instructions, procedures, drawings, and specifications; (5) issuing and distributing approved documents; (6) purchasing items and services; (7) identifying materials, parts, and components; (8) performing special processes; (9) inspecting and/or testing materials equipment, processes, or services; (10) calibrating and maintaining measuring equipment; (11) handling, storing, and shipping items; (12) identifying the inspection, test, and operating status of items; (13) identifying and disposing of nonconforming items; (14) correcting conditions adverse to quality; (15) preparing and maintaining quality assurance records; and (16) auditing activities that affect quality.

Quality is verified through checking, review, surveillance, inspection, testing, and audit of quality-related activities. The quality assurance program requires that quality verifications be performed by individuals who are not directly responsible for performing the quality-related activities. Inspections are performed by qualified personnel in accordance with procedures, instructions, and checklists approved by the quality assurance organization.

The quality assurance organization is responsible for the establishment and implementation of the audit program. Audits are performed in accordance with pre-established written checklists by qualified personnel not having direct responsibilities in the areas being audited. Audits are performed to evaluate all aspects of the quality assurance program, including the effectiveness of the quality assurance program implementation.

The quality assurance program requires the review of the audit results by the person having responsibility in the area audited and corrective action where necessary. Continued deficiencies, or failure to implement corrective action, will be reported in writing by the quality assurance organization to the appropriate management. Followup audits are performed to determine that nonconformance and deficiencies are effectively corrected and that the corrective action precludes repetitive occurrences. Audit reports, which indicate performance trends and the effectiveness of the quality assurance program, are prepared and issued to responsible management for review and assessment.

In accordance with 10 CFR 50.54, the licensee submitted on May 23, 1983, its revised quality assurance program for staff review. The revised program commits to Regulatory Guides 1.146 and 1.58, Revision 1, as requested by Generic Letter 81-01, "Qualification of Inspection, Examinations, Testing, and Personnel" (dated May 14, 1981). The staff approved the revised program by letter dated August 1, 1983. By letter dated January 3, 1989, the licensee submitted Revision 2 to program. On the basis of its review of this revision, the staff in its SER dated February 7, 1989, concluded that the program continues to meet the requirements of 10 CFR Part 50, Appendix B, and is therefore acceptable. The effectiveness of the implementation of the quality assurance program will continue to be the subject of routine NRC staff inspections.

# 18 REPORT OF THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

The licensee's application for a full-term operating license is being reviewed by the Advisory Committee on Reactor Safeguards. The NCR staff will issue a supplement to this SER after the Committee's report to the Commission is available. The supplement will append a copy of the Committee's report, will address comments made by the Committee, and will describe steps taken by the NRC staff to resolve any issues raised as a result of the Committee's review.

# 19 COMMON DEFENSE AND SECURITY

GPU Nuclear Corporation and Jersey Central Power & Light Company, co-licensees, are not owned, dominated, or controlled by an alien, a foreign corporation, or a foreign government. The activities that will continue to be conducted do not involve any restricted data, but the licensee has agreed to safeguard any such data that might become involved in accordance with the requirements of 10 CFR Part 50. The licensee will continue to rely on obtaining fuel as it is needed from sources of supply available for civilian purposes, so that no diversion of special nuclear material for military purposes is involved. For these reasons, and in the absence of any information to the contrary, the staff has found that issuance of the full-term operating license will not be inimical to the common defense and security.

# 20 FINANCIAL QUALIFICATIONS

On September 12, 1984, the NRC published in the Federal Register (49 FR 35747) amendments to its regulations that eliminate the review relating to the financial qualifications of electric utility applicants for operating licenses. Because these amendments were effective immediately, there will be no further review of the financial qualifications of GPU Nuclear Corporation and Jersey Central Power & Light Company.

# 21 FINANCIAL PROTECTION AND INDEMNITY REQUIREMENTS

Pursuant to the financial protection and indemnification provisions of the Atomic Energy Act of 1954, as amended (Section 170 and related sections), the Commission has issued regulations in 10 CFR Part 140. These regulations set forth the Commission's requirements with regard to proof of financial protection by, and indemnification of, licenses for facilities such as power reactors under 10 CFR Part 50.

Under the Commission's regulations in 10 CFR Part 140, a license authorizing the operation of a reactor may not be issued until proof of financial protection in the amount required for such operation has been executed. The amount of financial protection that must be maintained for the Oyster Creek plant (which has a rated capacity in excess of 100,000 electrical kilowatts) is the maximum amount available from private sources (i.e., the combined capacity of the two nuclear liability insurance pools; this amount is currently $200 million).

The NRC and JCP&L entered into Indemnity Agreement No. B-37 on October 3, 1967. Therefore, the staff concludes that the licensee complies with the provisions of 10 CFR Part 140 applicable to operating licenses, including those that relate to proof of financial protection in the requisite amount and to execution of an appropriate indemnity agreement with the Commission.

## 22 CONCLUSIONS

On the basis of its evaluation of the application as set forth in the preceding sections, the staff has determined the following:

(1) The application for a full-term operating license (FTOL) for the Oyster Creek Nuclear Generating Station filed by Jersey Central Power & Light Company dated March 6, 1972, as supplemented and as revised, complies with the requirements of the Atomic Energy Act of 1954, as amended (Act), and the Commission's regulations set forth in 10 CFR Chapter I, except as duly exempted therefrom.

(2) The provisions of Provisional Operating License DPR-16 have been met.

(3) The facility will operate in conformity with the FTOL application as amended, the provisions of the Act, and the rules and regulations of the Commission.

(4) There is reasonable assurance (a) that the activities authorized by the FTOL can be conducted without endangering the health and safety of the public and (b) that such activities will be conducted in compliance with the regulations of the Commission set forth in 10 CFR Chapter I.

(5) The licensee is technically qualified to engage in the activities authorized by the FTOL in accordance with the regulations of the Commission set forth in 10 CFR Chapter I.

(6) The issuance of the FTOL will not be inimical to the common defense and security or to the health and safety of the public.

(7) The FTOL for the Oyster Creek Nuclear Generating Station should be authorized by the NRC.

# APPENDIX A

# REFERENCES

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---, Std. 317-1983, "Standard for Electric Penetration Assemblies in Containment Structures for Nuclear Power Generating Stations."

---, Std. 379-1977, "Guide for the Application of the Single Failure Criterion to Nuclear Power Generating Station Protection System."

---, Std. 450-1975, "Recommended Practice for Maintenance, Testing, and Replacement of Large Lead Storage Batteries for Generating Stations and Substations."

---, Std. 946-1985, "Recommended Practice for the Design of Safety-Related DC Auxiliary Power Systems for Nuclear Power Generating Stations."

## APPENDIX B

## THREE MILE ISLAND - LESSONS-LEARNED REQUIREMENTS

The accident at Three Mile Island Unit 2 (TMI-2) resulted in requirements that were developed from the recommendations of several groups that were established to investigate the accident. These groups included

- Congress
- General Accounting Office
- President's Commission on the Accident at Three Mile Island
- NRC Special Inquiry Group
- NRC Advisory Committee on Reactor Safeguards
- Lessons Learned Task Force
- Bulletins and Orders Task Force of NRC Office of Nuclear Reactor Regulation
- Special Review Group of the NRC Office of Inspection and Enforcement
- NRC Siting Task Force
- NRC Emergency Preparedness Task Force
- NRC Office of Standards Development
- NRC Office of Nuclear Regulatory Research.

NUREG-0660, entitled "NRC Action Plan Developed as a Result of the TMI-2 Accident" (referred to as "the Action Plan"), was developed to provide a comprehensive and integrated plan for the actions NRC judged necessary to correct or improve the regulation and operation of nuclear facilities. The Action Plan was based on the experience from the TMI-2 accident and the recommendations of the investigating groups.

With the development of the Action Plan, NRC transformed the recommendations of the investigating groups into discrete scheduled tasks that specify changes in regulatory requirements, organization, or procedures. Some actions to improve the safety of operating plants were judged to be necessary before an action plan could be developed, although they were subsequently included in NUREG-0660. Such actions came from the bulletins and orders issued by the Commission immediately after the accident, the first report of the Lessons Learned Task Force, and the recommendations of the Emergency Preparedness Task Force. Before these immediate actions were applied to operating plants, they were approved by the Commission.

The NRC identified a discrete set of licensing requirements related to TMI-2 in the action plan for Oyster Creek Nuclear Generating Station. NUREG-0737, entitled "Clarification of the TMI Action Plan Requirements," was issued in November 1980. This report identifies the specific items from NUREG-0660 that were approved by the Commission for implementation at nuclear power plants. It also includes additional information about schedules, applicability, method of implementation review, submittal dates, and clarification of technical positions. By letter dated December 17, 1982, Supplement 1 to NUREG-0737 was issued to coordinate and indicate initiatives related to

- safety parameter display systems
- detailed control room design reviews
- application of Regulatory Guide 1.97 to emergency response facilities
- upgrading emergency operating procedures (EOPs)
- emergency response facilities

  - emergency operations facility (EOF)
  - technical support center (TSC)
  - operational support center (OSC)

- meteorological data

Schedules for completing the topics in Supplement 1 were negotiated with the licensee and were confirmed by an NRC order dated June 12, 1984.

At the time the FTOL SER was in preparation, five TMI Action Plan items still had to be satisfied. Four of these items have been resolved. One item, II.F.1, is still to be satisfed. The five items are discussed below.

| TMI item | Title |
|---|---|
| (1) I.C.1.3a | Abnormal Transient Operator Guidelines |
| (2) I.D.1 | Detailed Control Room Design Review (DCRDR) |
| (3) I.D.2 | Safety Parameter Display System |
| (4) II.F.1 | Additional Accident-Monitoring Instrumentation/Generic Letter 83-36 |
| (5) III.D.3.4 | Control Room Habitability/Generic Letter 83-36 |

I.C.1.3a ABNORMAL TRANSIENT OPERATOR GUIDELINES

<u>Requirement</u>

Submit a procedures generation package (PGP) to NRC for approval. The PGP shall include:

- Plant-specific technical guidelines - plant-specific guidelines for plants not using generic technical guidelines. For plants using generic technical guidelines, a description of the planned method for developing plant-specific EOPs from the generic guidelines, including plant-specific information.

- A writer's guide that details the specific methods to be used by the licensee in preparing EOPs based on the technical guidelines.

- A description of the program for validation of EOPs.

- A brief description of the training program for the upgraded EOPs.

<u>Status</u>

The licensee submitted the Oyster Creek PGP to the NRC on July 29, 1983, omitting the writer's guide. The writer's guide was submitted in 1985. The licensee also implemented upgraded EOPs consistent with the I.C.1.3a guidance.

There have been two independent contractor-assisted NRC reviews of the Oyster Creek procedures program. The first was the programmatic design review of the PGP, which was initiated with its submittal. The other was an EOP inspection conducted at Oyster Creek on September 6-15, 1988, during which the PGP was also audited. The findings of these reviews concurred in areas of mutual review scope. It was concluded that the EOPs are technically acceptable and that both the material condition of the facility and the knowledge of the operators were better than acceptable. Recommendations were made for improving the following programmatic areas: writer's guide, verification and validation, and training. Improvements in these areas will continue to be considered in routine NRC inspections.

In a letter dated November 20, 1989, the staff provided details of these findings, and further noted the recent completion of its evaluation of Revision 4 of the generic General Electric Boiling Water Reactor Owners Group emergency procedure guidelines and requested that the plant-specific guideline program be updated to appropriately reference this latest revision. The licensee's implementation of the staff's requests will continue to be considered in routine NRC inspections.

Because of the current procedural adequacy at Oyster Creek noted in NRC inspections and the improvements anticipated in programmatic areas pursuant to NRC recommendations, the staff concludes that Oyster Creek can continue to be operated without endangering the health and safety of the public.

## I.D.1 DETAILED CONTROL ROOM DESIGN REVIEW (DCRDR)

### Requirement

The objective of the DCRDR is to improve the ability of nuclear power plant control room operators to prevent accidents, or cope with accidents if they occur, by improving the information provided to them. The DCRDR addresses the following requirements as they are identified in Supplement 1 to NUREG-0737:

(1) establishment of a qualified multidisciplinary review team

(2) function and task analyses to identify control room operator tasks and information control requirements during emergency operations

(3) comparison of display and control requirements with a control room inventory

(4) a control room survey to identify deviations from accepted human factors principles

(5) assessment of human engineering discrepancies (HEDs) to determine which are significant and should be corrected

(6) selection of design improvements

(7) verification that selected improvements will provide the necessary correction and will not introduce new HEDs

(8) coordination of control room improvements with changes from other programs such as the safety parameter display system, operator training, Regulatory Guide 1.97 instrumentation, and upgraded emergency operating procedures

Status

By letter dated July 1, 1983, the licensee submitted its program plan for the review of the control room at Oyster Creek. The staff issued comments on the plan on February 6, 1984, and concluded that the plan was acceptable.

The licensee submitted a DCRDR Summary Report and Supplemental Summary Report by letters dated April 30, 1984, and April 8, 1985, respectively. The staff issued its safety evaluation of the subject reports on February 27, 1986, and requested that the licensee provide additional information on unresolved DCRDR issues. The licensee responded to the staff's request by letters dated May 19 and September 19, 1986, and July 8, 1988. In its safety evaluation dated June 28, 1990, the staff concluded that the DCRDR program implemented at Oyster Creek satisfies the nine requirements of Supplement 1 to NUREG-0737. The staff may confirm, by an inspection, that the corrective actions completed by the licensee as a result of the DCRDR have been completely and properly implemented. This item is resolved.

I.D.2 PLANT SAFETY PARAMETER DISPLAY SYSTEM

Requirement

Licensees and applicants for licenses must provide a safety parameter display system (SPDS) in the control room of their plants. The purpose of the SPDS is to provide control room operators with a concise display of critical plant variables to aid in rapid and reliable determination of plant safety status.

There are a number of requirements the SPDS should satisfy. They are as follows (references to the pertinent parts in Supplement 1 to NUREG-0737 are provided in parentheses):

(1) concise display of critical plant variables to aid control room operators in determining the safety status of the plant (4.1a)

(2) location convenient to control room operators (4.1b)

(3) continuous display of information from which plant safety status can be assessed (4.1b)

(4) aid operators in rapid, reliable determination of plant safety status (4.1a and 4.1b)

(5) suitable isolation from electrical or electronic interference with equipment and sensors that are in use for safety systems (4.1e)

(6) incorporation of accepted human factors principles (4.1e)

(7) parameters selected to provide, at a minimum, information about reactivity control, reactor core cooling and heat removal from the primary system, reactor coolant system integrity, radioactivity control, and containment conditions (4.1f)

(8) implementation of procedures and operator training leading to timely and correct assessment of safety status both with and without the SPDS (4.1c)

Status

Generic Letter 89-06 issued by the NRC on April 12, 1989, requested licensees to certify whether or not their SPDS met the requirements of Supplement 1 to NUREG-0737. By letter dated July 24, 1989, the licensee certified that the Oyster Creek SPDS meets the requirements of Supplement 1 to NUREG-0737 with certain clarifications. The staff obtained additional information in an audit conducted January 17-18, 1990, a letter from the licensee dated May 17, 1990, and discussions dated November 1, 1989, and April 3 and 16, 1990. On the basis of its review and licensee commitments to implement two identified items by the third quarter of 1990, the staff concluded in an SER dated June 28, 1990, that the Oyster Creek SPDS satisfies the requirements of NUREG-0737, Supplement 1. The staff may confirm, by inspection, that the corrective actions have been completely and properly implemented. This item is resolved.

II.F.1 ADDITIONAL ACCIDENT-MONITORING INSTRUMENTATION

NUREG-0737, Item II.F.1, identified the need for six types of accident monitors: (1) noble gas, (2) iodine/particulate sampling, (3) containment high-range radiation, (4) containment pressure, (5) containment water level, and (6) containment hydrogen. Guidance for Technical Specifications governing these monitors is included in Generic Letter 83-36, "NUREG-0737 Technical Specifications."

Amendment 94 to the provisional operating license (POL) includes Technical Specifications governing drywell pressure monitors, torus water level monitors, and drywell hydrogen monitors, to resolve NUREG-0737 Items II.F.1.4, II.F.1.5, and II.F.1.6. However, the staff evaluation and letter accompanying POL Amendment 94 note that the following items from Generic Letter 83-36 still have to be resolved: (1) II.F.1.1, Noble Gas Effluent Monitor; (2) II.F.1.2, Sampling and Analysis of Plant Effluents; and (3) II.F.1.3, Containment High-Range Radiation Monitor. POL Amendment 108, dated October 6, 1986, includes Technical Specifications that address Item II.F.1.2. POL Amendment 116, dated March 31, 1987, includes Technical Specifications that address Item II.F.1.3. However, in the safety evaluation report (SER) supporting the amendment, the staff requested that the licensee propose an additional Technical Specification in accordance with NUREG-0123, Revision 4, on the preplanned alternative method of monitoring, or provide justification for not needing this addition. The staff and the licensee are negotiating a schedule for resolving Item II.F.1.3, according to the SER requirement. License Amendment 137, dated February 6, 1990, contains Technical Specifications that resolve Item II.F.1.1.

III.D.3.4 CONTROL-ROOM HABITABILITY

This item is resolved as discussed in Section 6.6.

## APPENDIX C

## UNRESOLVED SAFETY ISSUES

### C.1 Introduction

The NRC staff evaluates the safety requirements used in its reviews against new information as it becomes available. Information related to the safety of nuclear power plants comes from a variety of sources including experience from operating reactors; research results; NRC staff and Advisory Committee on Reactor Safeguards (ACRS) safety reviews; and vendor, architect/engineer, and utility design reviews. After the accident at Three Mile Island Unit 2 (TMI-2), the Office for Analysis and Evaluation of Operational Data was established to provide a systematic and continuing review of operating experience. Each time a new concern or safety issue is identified from one or more of these sources, the need for immediate action to ensure safe operation is assessed. This assessment includes consideration of the generic implications of the issue.

In some cases, immediate action is taken to ensure safety; for example, the derating of boiling-water reactors (BWRs) as a result of the channel box wear problems in 1975. In other cases, interim measures, such as modifications to operating procedures, may be sufficient to allow further study of the issue before licensing decisions are made. In most cases, however, the initial assessment indicates that immediate licensing actions or changes in licensing criteria are not necessary. If the issue applies to several or a class of plants, it is evaluated further as a "generic safety issue." This evaluation considers the safety significance of the issues, the cost to implement any changes in plant design or operation, and other significant and relevant factors to establish a priority ranking of the issue. On the basis of this ranking, the issue is (1) scheduled for near-term resolution, (2) deferred until resources become available, or (3) dropped from further consideration.

The issues with the highest priority ranking are reviewed to determine whether they should be designated as "unresolved safety issues" (NUREG-0410, "NRC Program for the Resolution of Generic Issues Related to Nuclear Power Plants," dated January 1, 1978). However, as discussed above, such issues are considered on a generic basis only after the staff has made an initial determination that the safety significance of the issue does not prohibit continued operation or require licensing actions while the longer-term generic review is under way.

### C.2 ALAB-444 Requirements

These longer-term generic studies were the subject of a decision by the Atomic Safety and Licensing Appeal Board of the Nuclear Regulatory Commission. The decision was issued on November 23, 1977 (ALAB-444), in connection with the Appeal Board's consideration of the Gulf States Utility Company application for the River Bend Station, Units 1 and 2. It stated:

> In short, the board (and the public as well) should be in a position to ascertain from the SER itself - without the need to resort to

extrinsic documents - the staff's perception of the nature and extent of the relationship between each significant unresolved generic safety question and the eventual operation of the reactor under scrutiny. Once again, this assessment might well have a direct bearing upon the ability of the licensing board to make the safety findings required of it on the construction permit level even though the generic answer to the question remains in the offing. Among other things, the furnished information would likely shed light on such alternatively important considerations as whether: (1) the problem has already been resolved for the reactor under study; (2) there is a reasonable basis for concluding that a satisfactory solution will be obtained before the reactor is put in operation; or (3) the problem would have no safety implications until after several years of reactor operation and, should it not be resolved by then, alternative means will be available to ensure that continued operation (if permitted at all) would not pose an undue risk to the public.

This appendix is specifically included to respond to the decision of the Atomic Safety and Licensing Appeal Board as enunciated in ALAB-444, and as applied to an operating license proceeding, Virginia Electric and Power Company (North Anna Nuclear Power Station, Unit Nos. 1 and 2), ALAB-491, 8 NRC 245 (1978).

## C.3 Unresolved Safety Issues

In a related matter, as a result of congressional action on the NRC budget for fiscal year 1978, the Energy Reorganization Act of 1974 was amended (PL 95-209) on December 13, 1977, to include, among other things, a new Section 210 as follows:

UNRESOLVED SAFETY ISSUES PLAN

SEC. 210. The Commission shall develop a plan providing for specification and analysis of unresolved safety issues relating to nuclear reactors and shall take such action as may be necessary to implement corrective measures with respect to such issues. Such plan shall be submitted to the Congress on or before January 1, 1978, and progress reports shall be included in the annual report of the Commission thereafter.

The Joint Explanatory Statement of the House-Senate Conference Committee for the Fiscal Year 1978 Appropriations Bill (Bill S.1131) provided the following additional information regarding the Committee's deliberations on this portion of the bill:

SECTION 3 - UNRESOLVED SAFETY ISSUES

The House amendment required development of a plan to resolve generic safety issues. The conferees agreed to a requirement that the plan be submitted to the Congress on or before January 1, 1978. The conferees also expressed the intent that this plant should identify and describe those safety issues, relating to nuclear power reactors, which are unresolved on the date of enactment. It should set forth: (1) Commission actions taken directly or indirectly to develop and implement corrective measures; (2) further actions planned concerning such measures; and (3) timetables and cost estimates of such actions. The Commission should indicate the priority it has assigned to each issue, and the basis on which priorities have been assigned.

In response to the reporting requirements of the new Section 210, the NRC staff submitted to Congress on January 1, 1978, a report (NUREG-0410) that described the NRC generic issues program. The NRC program was already in place when PL 95-209 was enacted and is of considerably broader scope than the Unresolved Safety Issues Plan required by Section 210. In the letter transmitting NUREG-0410 to the Congress on December 30, 1977, the Commission stated, "The progress reports, which are required by Section 210 to be included in future NRC annual reports, may be more useful to Congress if they focus on the specific Section 210 safety items."

It is the NRC's view that the intent of Section 210 was to ensure that plans were developed and implemented on issues with potentially significant public safety implications. In 1978, the NRC undertook a review of more than 130 generic issues addressed in the NRC program to determine which issues fit this description and qualify as unresolved safety issues for reporting to the Congress. The NRC review included the development of proposals by the NRC staff and review and final approval by the NRC Commissioners.

This review is described in NUREG-0510, "Identification of Unresolved Safety Issues Relating to Nuclear Power Plants - A Report to Congress," dated January 1979. The report provides the following definition of an unresolved safety issue:

> An Unresolved Safety Issue is a matter affecting a number of nuclear power plants that poses important questions concening the adequacy of existing safety requirements for which a final resolution has not yet been developed and that involves conditions not likely to be acceptable over the lifetime of the plants it affects.

Further, the report indicates that in applying this definition, matters that pose "important questions concerning the adequacy of existing safety requirements" were judged to be those for which resolution is necessary to (1) compensate for a possible major reduction in the degree of protection of the public health and safety or (2) provide a potentially significant decrease in the risk to the public health and safety. Quite simply, an unresolved safety issue is potentially significant from a public safety standpoint and its resolution is likely to result in NRC action for the affected plants.

All of the issues addressed in the NRC program were systematically evaluated against this definition as described in NUREG-0510. As a result, 17 of the generic issues addressed by 22 tasks in the NRC program were identified as unresolved safety issues.

An in-depth and systematic review of generic safety concerns identified between January 1979 and March 1981 was performed by the staff to determine if any of these issues should be designated as unresolved safety issues. The candidate issues originated from concerns identified in NUREG-0660, "NRC Action Plan as a Result of the TMI-2 Accident"; ACRS recommendations; abnormal occurrence reports; and other operating experience. The staff's proposed list was reviewed and commented on by the ACRS, the Office for Analysis and Evaluation of Operational

Data (AEOD), and the Office of Policy Evaluation. The ACRS and AEOD also proposed that several additional unresolved safety issues be considered by the Commission. The Commission considered the above information and approved the four Unresolved Safety Issues A-45 through A-48. A description of the review process of candidate issues, together with a list of issues considered, is presented in NUREG-0705, "Identification of New Unresolved Safety Issues Relating to Nuclear Power Plants, Special Report to Congress," dated March 1981. An expanded discussion of each of the new unresolved safety issues is contained in NUREG-0705. In addition to the four issues identified above, the Commission approved another issue, A-49, "Pressurized Thermal Shock," as an unresolved safety issue in December 1981.

The issues are listed below. The number(s) of the generic tasks(s) (e.g., A-1) in the NRC program addressing each issue is (are) indicated in parentheses following the title.

<u>Unresolved Safety Issues (Applicable Task Nos.)</u>

(1) Waterhammer (A-1)

(2) Asymmetric Blowdown Loads on the Reactor Coolant System (A-2)

(3) Pressurized Water Reactor Steam Generator Tube Integrity (A-3, A-4, A-5)

(4) BWR Mark I and Mark II Pressure Suppression Containments (A-6, A-7, A-8, and A-39)

(5) Anticipated Transients Without Scram (A-9)

(6) BWR Nozzle Cracking (A-10)

(7) Reactor Vessel Materials Toughness (A-11)

(8) Fracture Toughness of Steam Generator and Reactor Coolant Pump Supports (A-12)

(9) Systems Interaction in Nuclear Power Plants (A-17)

(10) Environmental Qualification of Safety-Related Electrical Equipment (A-24)

(11) Reactor Vessel Pressure Transient Protection (A-26)

(12) Residual Heat Removal Requirements (A-31)

(13) Control of Heavy Loads Near Spent Fuel (A-36)

(14) Seismic Design Criteria (A-40)

(15) Pipe Cracks at Boiling Water Reactors (A-42)

(16) Containment Emergency Sump Reliability (A-43)

(17) Station Blackout (A-44)

(18) Shutdown Decay Heat Removal Requirements (A-45)

(19) Seismic Qualification of Equipment in Operating Plants (A-46)

(20) Safety Implications of Control Systems (A-47)

(21) Hydrogen Control Measures and Effects of Hydrogen Burns on Safety Equipment (A-48)

(22) Pressurized Thermal Shock (A-49)

In the view of the staff, the unresolved safety issues (USIs) listed above are the substantive safety issues referred to by the Appeal Board in ALAB-444 when it spoke of "...those generic problems under continuing study which have... potentially significant public safety implications."

Nine of the tasks identified with the USIs are not applicable to Oyster Creek; seven of these nine (A-2, A-3, A-4, A-5, A-12, A-26, and A-49) are peculiar to pressurized-water reactors (PWRs). Task A-8 is related to Mark II pressure-suppression containments only. Task A-48 is considered complete for Oyster Creek; the Task Action Plan (December 1982) is related only to PWRs with ice condenser containments or BWRs with Mark III-type containments. With regard to the remaining 19 tasks that are applicable to this facility, the NRC staff has issued NUREG reports or other regulatory guidance providing its proposed resolution of these issues. The task number for the issues, the associated reports or regulatory guidance, and the section of the SER in which the issue is discussed are listed below.

| Task no. | Document | SER section |
|---|---|---|
| A-1 | NUREG-0927, Rev. 1, "Evaluation of Water Hammer Occurrence in Nuclear Power Plants" | 3.9.4 |
| A-6 | NUREG-0408, "Mark I Containment Short-Term Program" | 6.2.1 |
| A-7 | NUREG-0661, "Mark I Containment Long-Term Program" | 6.2.1 |
| A-9 | NUREG-0460, Vol. 4, "Anticipated Transients Without Scram for Light Water Reactors" | 15.2 |
| A-10 | NUREG-0619, "BWR Feedwater Nozzle and Control Rod Drive Return Line Nozzle Cracking" | 5.6 |
| A-11 | NUREG-0744, Rev. 1, "Resolution of the Task A-11 Reactor Vessel Materials Toughness Safety Issue" | 5.3 |
| A-17 | NUREG-1174, "Evaluation of Systems Interactions in Nuclear Power Plants" | 6.11 |
| A-24 | NUREG-0588, Rev. 1, "Interim Staff Position on Environmental Qualification of Safety-Related Electrical Equipment" | 3.10 |

| Task no. | Document | SER section |
|---|---|---|
| A-31 | NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants," Section 5.4.7 and Branch Technical Position (BTP) RSB 5-1, "Residual Heat Removal Systems," incorporate requirements of USI A-31 | 5.4.2 |
| A-36 | NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants" | 9.1.2 |
| A-39 | NUREG-0783, "Suppression Pool Temperature Limits for BWR Containments" | 6.2.1 |
| A-40 | NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants" Section 2.5.2, Rev. 2, "Vibratory Ground Motion" | 3.7.3 |
| | NUREG-0800, Section 3.7.1, Rev. 2, "Seismic Design Parameters" | |
| | NUREG-0800, Section 3.7.2, Rev. 2, "Seismic System Analysis" | |
| | NUREG-0800, Section 3.7.3, Rev. 2, "Seismic Subsystem Analysis" | |
| A-42 | NUREG-0313, Rev. 1, "Technical Report on Material Selection and Processing Guidelines for BWR Coolant Pressure Boundary Piping" | 3.9.3 |
| A-43 | NUREG-0897, Rev.1, "Containment Emergency Sump Performance" | 6.3.4 |
| A-44 | 10 CFR 50.63; Regulatory Guide 1.155, "Station Blackout" | 8.3.1.1 |
| A-45 | NUREG-1289, "Regulatory and Backfit Analysis of Unresolved Safety Issue A-45, Shutdown Decay Heat Removal Requirements" | 6.3.5 |
| A-46 | NUREG-1030, "Seismic Qualification of Equipment in Operating Nuclear Power Plants" | 3.7.1 |
| A-47 | NUREG-1217, "Evaluation of Safety Implications of Control Systems in LWR Nuclear Power Plants" | 7.4.2 |
| A-48 | 10 CFR 50.44; SECY-89-122, "Resolution of Unresolved Safety Issue (USI) A-48, 'Hydrogen Control Measures and Effects of Hydrogen Burns on Safety Equipment'" | 6.5 |

All task action plans for the generic tasks up to and including A-40 above are included in NUREG-0649, "Task Action Plans for Unresolved Safety Issues Related to Nuclear Power Plants." Task action plans for later tasks were issued individually as indicated below.

| Task number | Issue date of task action plan |
|---|---|
| A-43 | 1/81 |
| A-44 | 7/80 |
| A-45 | 10/81 |
| | 6/82 (Rev. 1) |
| A-46 | 5/82 |
| A-47 | 6/82 |

Each task action plan provides a description of the problem; the staff's approach to its resolution; a general discussion of the bases on which continued plant licensing or operation can proceed pending completion of the task; the technical organizations involved in the task and estimates of the staffing required; a description of the interactions with other NRC offices, the Advisory Committee on Reactor Safeguards, and outside organizations; estimates of funding required for contractor-supplied technical assistance; prospective dates for completing the task; and a description of potential problems that could alter the planned approach or schedule.

In addition to the task action plans, the staff issues the "Office of Nuclear Reactor Regulation Unresolved Safety Issues Summary" (Aqua Book, NUREG-0606) on a quarterly basis, which provides current scheduling information relative to the implementation status of each unresolved safety issue for which technical resolution is complete.

The current status of the USIs applicable to Oyster Creek was provided by the licensee's response dated November 30, 1989, to Generic Letter 89-21. The staff agrees that all USIs have been implemented except for the following:

- A-9, "Anticipated Transients Without Scram"
- A-44, "Station Blackout"
- A-46, "Seismic Qualification of Equipment in Operating Plants"
- A-47, "Safety Implications of Control Systems"

As discussed in Section 6.5 of this SER, Generic Task A-48 is resolved for Oyster Creek; however, the plant-specific issue of combustible gas remains open, pending staff review.

On the basis of its review of these items, the staff concludes that there is reasonable assurance that Oyster Creek can continue to be operated before the ultimate resolution of these generic issues without endangering the health and safety of the public.

NRC FORM 335
(2-89)
NRCM 1102,
3201, 3202

U.S. NUCLEAR REGULATORY COMMISSION

# BIBLIOGRAPHIC DATA SHEET

*(See instructions on the reverse)*

1. REPORT NUMBER (Assigned by NRC. Add Vol., Supp., Rev., and Addendum Numbers, if any.)

NUREG-1382

2. TITLE AND SUBTITLE

Safety Evaluation Report related to the full-term operating license for Oyster Creek Nuclear Generating Station

3. DATE REPORT PUBLISHED

| MONTH | YEAR |
|---|---|
| January | 1991 |

4. FIN OR GRANT NUMBER

5. AUTHOR(S)

6. TYPE OF REPORT

7. PERIOD COVERED *(Inclusive Dates)*

8. PERFORMING ORGANIZATION – NAME AND ADDRESS *(If NRC, provide Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address; if contractor, provide name and mailing address.)*

Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, DC 20555

9. SPONSORING ORGANIZATION – NAME AND ADDRESS *(If NRC, type "Same as above"; if contractor, provide NRC Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address.)*

Same as 8 above

10. SUPPLEMENTARY NOTES

Docket No. 50-219

11. ABSTRACT *(200 words or less)*

The Safety Evaluation Report for the full-term operating license application filed by GPU Nuclear Corporation and Jersey Central Power & Light Company for the Oyster Creek Nuclear Generating Station has been prepared by the Office of Nuclear Reactor Regulation of the U.S. Nuclear Regulatory Commission. The facility is located in Ocean County, New Jersey. The staff concludes that the facility can continue to be operated without endangering the health and safety of the public.

12. KEY WORDS/DESCRIPTORS *(List words or phrases that will assist researchers in locating the report.)*

Safety Evaluation Report
Full-Term Operating License
Oyster Creek Nuclear Generating Station

13. AVAILABILITY STATEMENT

Unlimited

14. SECURITY CLASSIFICATION

*(This Page)*
Unclassified

*(This Report)*
Unclassified

15. NUMBER OF PAGES

16. PRICE