December 23, 1968

SAFETY EVALUATION

BY THE

DIVISION OF REACTOR LICENSING

U. S. ATOMIC ENERGY COMMISSION

IN THE MATTER OF

JERSEY CENTRAL POWER & LIGHT COMPANY

OYSTER CREEK NUCLEAR POWER PLANT UNIT NO. 1

DOCKET NO. 50-219

Enclosure 1

TABLE OF CONTENTS

LIST OF TABLES

1.0 INTRODUCTION

The Jersey Central Power & Light Company (Jersey Central, applicant), submitted Amendment No. 3, dated January 18, 1966, to its application requesting a Provisional Operating License for the Oyster Creek Nuclear Power Plant Unit No. 1 (Oyster Creek, facility, plant). The facility, which will utilize a single cycle, forced circulation General Electric boiling water reactor (BWR), has been under construction since issuance of a construction permit on December 15, 1964, by the Commission. It is located on an 800-acre site in Lacey and Ocean Townships, Ocean County, New Jersey. This site is approximately thirty-five miles north of Atlantic City, New Jersey and forty-five miles east of Philadelphia, Pennsylvania.

The technical safety review of the design of the facility has been based on Amendment Nos. 3 through 49. All of these documents are available for review at the Atomic Energy Commission's Public Document Room at 1717 H Street, Washington, D. C. In the course of the review, we have held numerous meetings with the applicant to discuss and clarify the technical material submitted. In addition to our review, the Advisory Committee on Reactor Safeguards (ACRS) reviewed the application and met with both us and the applicant to discuss the facility. The ACRS report on Oyster Creek, dated December 12, 1968, is attached to this safety evaluation.

Our evaluation of overall facility performance was based on a thermal power level of 1600 megawatts (Mw) which will be the licensed power level. However, because the plant is designed for ultimate power operation at 1860 Mwt, we reviewed the capability of the plant engineered safety features and the radiological consequences of accidents at the ultimate power level of 1860 Mwt. Before any increase in power level in excess of 1600 Mwt can be permitted, the applicant must submit an application for license amendment.

Based upon our evaluation of the facility as presented in subsequent sections, we have concluded that the Oyster Creek Nuclear Power Plant Unit No. 1 can be operated as proposed without endangering the health and safety of the public.

2.0 SITE AND ENVIRONMENT

2.1 Site Description

The site, which consists of approximately 800 acres, is located in Lacey and Ocean Townships of Ocean County, New Jersey, approximately two miles inland and west of the shore of Barnegat Bay. The minimum distance from the facility to a site boundary is approximately 0.25 mile. This corresponds to the distance from the facility to the eastern boundary of State Highway Route 9. The distance to the nearest residence is in excess of 0.5 mile. Based upon the extrapolated 1986 summer population distribution which shows approximately

1150 people within one mile, 12,264 within two miles and 31,040 within five miles of the site, the available low population zone distance is approximately two miles. The extrapolated 1986 permanent population within two miles, however, is approximately 7000. Tabulated below is the 1986 summer population distribution with distance.

TABLE 2.1

CUMULATIVE POPULATION DISTRIBUTION (1986)

Distance, Miles	Cumulative Population
1	1,154
2	12,264
3	20,920
4	24,230
5	31,040

2.2 Meteorology

The applicant has collected approximately one year of meteorological on-site data at the Oyster Creek site, which include measured wind speed, wind direction, and temperature difference with height at several elevations on a 400-foot-high tower. These data show that temperature inversion conditions with winds of below 3 mph occur approximately 3 percent of the time. Inversion conditions have persisted for periods in excess of 15 hours. These results are not unusual for typical coastal sites such as the Oyster Creek site. We have also considered the effects of wind loadings on plant shutdown capability. The meteorological model which we used in estimating the potential consequences of reactor accidents is described in Section 6.0.

2.3 Hydrology

Flood protection is provided so that the plant can be safely shutdown for a flooding level as high as approximately 23 feet above mean sea level (MSL). The maximum flood height recorded at the facility is 4.5 feet above MSL.

The potential for contamination of wells in the area of the site in the event of a possible spill of radioactive wastes onsite is very low since ground-water flow is toward Barnegat Bay. Surface run-off would flow directly toward Oyster Creek or Forked River. Neither stream is used for drinking water purposes.

The applicant has conducted diffusion studies in Barnegat Bay to determine the degree of dilution of liquid effluent discharges into the Bay. Our hydrologic consultants at the U. S. Geological Survey concluded that these studies provide a reasonable basis to determine the degree of dilution in the Bay. The applicant's environmental radiation monitoring program will demonstrate that the radioactivity levels in the Bay are below the 10 CFR 20 limits.

We conclude that the hydrologic aspects of the site do not present any unusual problems with respect to safe operation of the facility.

2.4 Geology and Seismology

The buildings and structures are founded on dense sand (Cohansey sand). After excavation and backfilling in the reactor and turbine building area the soil was compression tested using loads up to 80,000 pounds on a four-foot-square plate. The results indicate that the subsoil is not overloaded. Our geologic consultants at the U. S. Geological Survey studied the geologic aspects of the site during our construction permit review for this facility. They concluded that the Cohansey sand provides an adequate founding medium for the facility buildings and structures. We agree with this conclusion.

The applicant's seismic design bases specify that (a) for a maximum ground acceleration of 0.11g, resultant stress levels for critical components, equipment and structures necessary to ensure a safe and orderly shutdown will not exceed code allowables; and (b) for a ground acceleration of 0.22g, there will be no loss of function of critical structures and components necessary to ensure a safe and orderly shutdown. Based upon the report provided at the construction permit stage by our seismic consultant, the U. S. Coast and Geodetic Survey, we have concluded that these design basis accelerations are acceptable. Structures, equipment and components designed to these conditions are designated as Class I. The facility design has been reviewed by our consultants, Nathan M. Newmark Consulting Engineering Services of Urbana, Illinois. They concluded, and we agree, that the facility was designed and constructed in accordance with the seismic design criteria.

2.5 Environmental Radiation Monitoring

The applicant will continue to conduct an environmental radiation monitoring program in order to determine the effect of operations at this facility. The program was developed from the results of the preoperational monitoring program which was initiated in March 1966. The operational monitoring program will include measurement of atmospheric radioactivity, fallout, domestic water, surface water, aquatic biota, and foodstuffs. We conclude that this program will be adequate for assessing the health and safety aspects of the release of radioactivity to the environment from the operations of this plant. Recommendations from our consultants, the Fish and Wildlife Service of the U. S. Department of the Interior, have been incorporated into the applicant's environmental radiation monitoring program.

We conclude that the program proposed by the applicant is adequate with respect to monitoring the radiological aspects of plant operation on the environs.

3.0 FACILITY DESIGN

3.1 Reactor Core

3.1.1 General

The reactor is a single cycle, forced circulation, boiling water reactor producing steam for direct use in the steam turbine. The core containing the reactor fuel is located within a domed, cylindrical shroud inside the reactor vessel. Water, which serves as both the moderator and coolant, enters the bottom of the reactor core, and flows upward through the fuel assemblies where boiling produces steam. The steam-water mixture is separated by steam separators and dryers mounted on the shroud. The separated water mixes with the incoming feedwater in an annulus formed by the shroud and the wall of the reactor vessel and is returned to the core inlet via five external recirculation pumps. The steam is passed through the dryers to the turbine-generator for the production of electricity.

3.1.2 Mechanical Design

The overall active height of the core is 12 feet and the equivalent diameter is 13.35 feet. The reactor core will consist of 560 fuel assemblies each of which contains 49 cylindrical fuel rods in a 7 x 7 square array. A fuel rod is approximately one-half inch in diameter and 12 feet long. Each fuel rod consists of compacted and sintered uranium dioxide pellets enclosed in zircaloy tubes (cladding). The tubes are sealed by zircaloy plugs welded into each end.

Four fuel assemblies rest on a support casting mounted on top of each control rod guide tube. Each guide tube, with its fuel support casting, bears the weight of four fuel assemblies, and rests on a control rod drive housing. The housing is welded to a stub tube which in turn is welded to the bottom head of the reactor pressure vessel.

Control of the reactor to accommodate fuel burnup and fission product poisoning and to shut the reactor down is accomplished by control rods. The 137 control rods are cruciform-shaped, enter the reactor core from the bottom, and are manipulated by independent mechanisms. Each control rod contains stainless steel tubes filled with compacted boron carbide powder which is a neutron absorbing medium. The tubes are held in a cruciform array by a stainless steel sheath that extends the full length of the control rod. In addition to the control rods, 248 temporary control curtains which are fixed in the core are used to compensate for the excess reactivity change between initial and equilibrium cores. The curtains are made of boron-stainless steel sheets and are located in the spaces between the fuel channels. Spaces between the channels also contain in-core instrumentation and neutron sources necessary for plant operation.

The core configuration, control mode, and mechanical design features are generally similar to those presently being used in other operating reactors. General Electric has used the experience gained from the various operating reactors in the design of the Oyster Creek core.

On the basis of our review, we have concluded that the core design features for the Oyster Creek facility are adequate.

3.1.3 Thermal and Hydraulic Design

Operation of the reactor at 1600 Mwt with rated recirculation flow results in thermal and hydraulic conditions in the core which are similar to those of currently operating BWR's. The Big Rock Point reactor (Docket No. 50-155) has operated at average heat fluxes and primary coolant system flow rates which are about the same as Oyster Creek. The Dresden 1 reactor (Docket No. 50-10) has been run with exit steam void fractions and steam quality comparable to those expected in Oyster Creek.

Recently the Gundremmingen (KRB) Nuclear Power Station (General Electric BWR), similar in design to Oyster Creek, has been placed in operation in Germany at the design power level of 801 Mwt. Results of the accumulated operating data indicate satisfactory performance.

We have reviewed the analyses of the various transients that can be expected to occur during the operating lifetime of the plant. Transients can be induced by control rod withdrawals, changes in the recirculation flow rate, addition of cold water and change in system pressure. For all of the transients reviewed, the minimum critical heat flux ratio (MCHFR) remains well above unity, which is the assumed fuel rod damage limit. The limiting transient that would affect local regions of the core was found to result from a control rod withdrawal until stopped by the rod-block system. For this case, the calculated MCHFR remains above 1.2 using the critical heat flux data given in the General Electric Report No. APED-3892, "Burnout Limit Curves for Boiling Water Reactors." For other transients reviewed wherein the entire core is affected, the MCHFR remains above 1.8. From our review of the various transients and the plant protection system, we conclude that an adequate margin against fuel rod cladding damage is available in the Oyster Creek facility.

3.1.4 Reactivity Control

Reactor power can be controlled by either movement of control rods or variation in reactor coolant recirculation system flow rate. A standby liquid control system is also provided as a backup shutdown system. These aspects, as well as certain other plant features related to reactivity control, are discussed below.

Control rods are used to bring the reactor through the full range of power (from shutdown to full power operation), to shape the reactor power distribution, and to compensate for changes in reactivity due to fuel burnup. There are 137 individual control rod drives and hydraulic control systems. Each drive has separate control and scram devices. A common hydraulic pressure source for normal operation and a common dump volume for scram operation are used for the drives.

On the basis of our review of the drive system design and the supporting evidence accumulated from operation of similar systems in other reactors, we conclude that the installed system will meet the functional performance requirements for the Oyster Creek facility in a safe manner.

High control rod worths at power levels below 10% of rated power (1600 Mwt) are prevented by the rod worth minimizer (RWM), a device which utilizes a computer to restrict control rod patterns such that rods which are moved are worth no more than 1% Δk, and that no control rod worth will exceed 2-1/2% Δk, assuming permissible control rod patterns. The inputs to the computer are pre-selected control rod drive patterns and current control-rod-drive mechanism positions. The outputs consist of alarms and rod block signals when the safe rod sequence (one of two stored in the computer and selected by the operator) is not followed. On the basis of our review, we conclude that the RWM serves a useful role in assuring that the control rod worths would not become excessive and thereby cause serious damage in the event of a control rod drop accident.

At reactor power levels above 10%, the applicant does not intend to use the control rod worth miminizer to limit rod worths although it may do so. The maximum control rod worth that could be established for reactor power levels in excess of 10% is 3.8% Δk. Calculations of the consequences of a control rod-drop accident where a control rod is assumed to fall by gravity from the core region with a rod worth of 3.8% Δk and reactor power in excess of 10% indicate that the peak fuel enthalpy is less than 200 cal/gm. The enthalpy required for incipient fuel melting for the Oyster Creek fuel is 220 cal/gm. Accordingly, we conclude that use of the RWM at power levels above 10% is not required.

A control rod ejection accident is precluded by the control rod housing support structure located below the reactor pressure vessel. This structure serves to limit the distance that a ruptured control rod drive housing could be displaced to no more than three inches. The applicant indicates, and we agree, that control rod displacement of this magnitude would not introduce sufficient reactivity to the core to cause fuel rod failure.

With a given control rod pattern, control of the reactor can also be accomplished by varying the recirculation flow rate which causes a change in the void content in the core and a resultant change in reactor power. The applicant has not proposed to operate the plant initially on automatic flow control; therefore, we have not evaluated the automatic aspects of plant operation. If this mode of operation is proposed for future plant operation, it will be evaluated at that time.

The standby liquid control system is designed to bring the reactor to a cold shutdown condition from the full power steady state operating condition at any time in core life independent of the control rod system capabilities. This requires about 13%Δk of shutdown reactivity worth. The liquid control system is designed to inject sufficient sodium pentaborate to provide 18%Δk of negative reactivity, thus a shutdown margin of about 5%Δk is available. The injection rate of the system is adequate to compensate for the effects of xenon burnup.

3.2 Primary Coolant System

The primary coolant system includes the reactor pressure vessel, recirculation loops, relief valves, safety valves and the isolation condenser system. An in-service inspection program for the primary coolant system, as described in the Technical Specifications, has been developed for initial plant operation. As noted in the ACRS letter, Jersey Central will review the program with us after four years of reactor operation, and modify it as necessary based on experience gained during operation. We conclude that the in-service inspection program, combined with the continuing review, is adequate for this plant.

3.2.1 Reactor Pressure Vessel

The Oyster Creek reactor vessel is made of high strength alloy carbon steel SA-302, Grade B and was designed for a pressure of 1250 psig and 575°F. The reactor vessel was fabricated, inspected, and tested in accordance with the ASME Boiler and Pressure Vessel Code Section I Power Boilers, 1962 Edition, plus the Nuclear Code Cases applicable on December 11, 1963, the date of the vessel contract. Further, the vessel manufacturer (Combustion Engineering) was directed by GE to use Section VIII of the Code for Unfired Pressure Vessels where Section I Power Boilers did not cover specific details.

We have reviewed the Reactor Pressure Vessel Design Report (Amendment 16) particularly with respect to: the code calculations summary, the steady state stresses and stress intensities, the fatigue analysis transient cycles, and the calculated cumulative fatigue usage factor. The applicant stated that there were no deviations from codes throughout the design, fabrication, inspection, and testing of the reactor vessel. The data reviewed, mentioned above, indicate that the material thicknesses, stresses, and the cumulative usage factors do not exceed established limits.

During the course of the field hydrostatic test of the reactor pressure vessel in 1967, a leak was noted near one of the control rod drive penetrations. A detailed and comprehensive program was initiated to determine the cause of the leak. During the investigative program, it was found that certain components of the reactor vessel had experienced what is characterized as intergranular attack. Other components were also found to have defective welds. These findings led to a comprehensive investigative and subsequent repair program to restore the vessel to an acceptable condition.

Intergranular attack was confined to those stainless steel components which were furnace sensitized; i.e., a high temperature heat treatment process which resulted in carbon precipitation at the grain boundaries of the stainless steel. Subsequent exposure to a corrodent(s) and in the presence of a stress field causes the component to crack. The corrodent(s) have not yet been identified. Numerous tests were conducted to demonstrate that if a clad overlay of a suitable material is placed over the sensitized material further intergranular attack is prevented. This repair technique was used for components where sufficient space was available to perform the necessary overlay operations. Other components, such as small sensitized stainless steel nozzle safe end attachments, were cut out and replaced with a non-furnace-sensitized material. In one case, the shroud support flange, a redundant structural component was fabricated and installed in the vessel.

For those components in which defective welds were found, the cause of the defect was traced to improper quality control of the field welding process. All of the field welds that join the control rod drive housing to the stub tube were removed and replaced with sound weld metal. Integrity of the welds was verified by the use of dye penetrant and ultrasonic test methods. Other defective welds were removed and rewelded as necessary.

In summary, we conclude that the various repair activities have restored the reactor pressure vessel to an acceptable condition. Furthermore, the inspection and repair program has been adequate and there is reasonable assurance that all defective components have been found and repaired. The investigative program was sufficiently complete to justify the conclusion that overlay protection of the sensitized stainless steel components will be effective in preventing further attack of the affected components.

3.2.3 Recirculation Piping

Each of the five reactor water recirculation loops contains a motor driven recirculation pump and motor-operated gate valves for pump isolation and maintenance. The recirculation loop piping is designed for a pressure of 1250 psig and a temperature of 570°F, the recirculation pump casings are designed for a pressure of 1300 psig and a temperature of 575°F, and the gate valves are designed for a pressure of 1200 psig and a temperature of 575°F. The recirculation loop piping is of welded construction and has been designed, built, and constructed to meet the requirements of ASME Code, Section I, and ASA-B31.1 Code for Pressure Piping.

The maximum operating loads included the design pressure and temperature, weight of piping, contents and insulation, as well as the effect of supports and other sustained external loadings. The stress limits used by the applicant for assumed load combinations are reasonable and in our judgment the recirculation loop piping will have adequate integrity to safely withstand these loads.

3.2.4 Emergency Condensers

The isolation condensers which are designed to Class I standards provide a natural circulation heat sink in case of reactor isolation from the main condenser. The tube sides of the condenser are exposed to reactor pressure vessel pressure during operation. Accordingly, the tubes have been designed for a pressure of 1250 psig and a temperature of 572°F. The emergency condensers are located outside of the primary containment, but inside the concrete and metal-sided reactor building. The secondary side of each condenser contains enough inventory to remove decay heat for the first 1-1/2 hours after reactor pressure vessel isolation. Makeup to the secondary side for continued heat removal is achieved either by a condensate transfer pump which can be operated on emergency power or by either of two diesel-driven fire pumps. We conclude that this system is adequate.

3.2.5 Relief and Safety Valves

The reactor coolant system safety and relief valves are installed on the steam lines inside the containment. The safety valves are designed and sized according to the ASME Boiler and Pressure Vessel Code, Section I. A total of 16 safety valves are provided and are capable of preventing the overpressurization of the system which would result from a turbine trip without benefit of a reactor scram (at 1860 Mwt). There are four relief valves provided in the design. The relief valves are sized to prevent actuation of the safety valves in the event of a turbine trip with a failure of the bypass system, but assuming the reactor does scram. Further aspects of the relief valves as they pertain to the emergency core cooling system are discussed in Section 3.5.1 of this report.

We conclude that these valves will prevent overpressurization of the primary coolant system.

3.3 Primary Containment

3.3.1 Design and Construction

The Oyster Creek primary containment design consists of a drywell, a connecting vent system, and a pressure suppression chamber (torus). The reactor vessel, the reactor coolant recirculating loops, and other branch connections of the reactor primary system are located in the drywell.

The drywell has a "light bulb" configuration consisting of a spherical section, 70 feet in diameter, and a cylindrical section approximately 23 feet in length and 33 feet in diameter. The design pressure is 62 psig. The pressure absorption chamber is in the form of a torus with a major diameter of 101 feet and an inner diameter of 30 feet. The design pressure is 35 psig. A vent system connects the drywell to the torus and terminates below the water level in the torus, so that in the event of a reactor system pipe failure in the drywell, the released steam passes directly to the torus pool water where it is condensed. This transfer of energy to the water pool rapidly reduces the pressure in the drywell, and thereby limits the amount of leakage from the primary containment.

Provisions are included for the removal of heat from the primary containment to maintain integrity of the containment system following any accident up to and including the design basis loss-of-coolant accident.

The basis for the design pressure and dynamic response of the primary containment is the loss-of-coolant following the sudden and complete severance of the largest line connected to the reactor vessel, while the reactor is operating at its steady state ultimate power level (1860 Mwt). The design criteria for containment are as follows:

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

(b) To channel the flows from postulated pipe breaks to the pressure absorption chamber.

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

(d) To limit primary containment leakage rate during and following a postulated break in the reactor primary system to substantially less than that which would result in off-site doses approaching the reference values in 10 CFR 100.

(e) To include provisions for leak rate tests.

(f) To be capable of being flooded following a design basis accident to a height which permits unloading of the core.

The design basis loss-of-coolant accident causes the highest primary containment pressures. Peak pressures of about 38 psig in the drywell and 25 psig in the suppression chamber occur following severance of the recirculation line. Analytical methods based upon experimental information obtained at Humboldt Bay and Bodega Bay test facilities (Moss Landing), were used to calculate these pressures. Because these pressures are substantially below the design values, we conclude that Oyster Creek primary containment will have a significant margin above the peak pressures calculated for the recirculation line break.

Penetrations through the primary containment are designed according to the rules of Section VIII of the ASME Boiler and Pressure Vessel Code and to certain Nuclear Code Cases. Our review of the loading conditions indicates that the applicant has properly accounted for the various loads including normal live and dead loads, earthquake loads, jet thrust loads, and loading conditions that result from accident conditions. The applicant has also incorporated appropriate provisions to assure proper leak rate testing. On the basis of our review of the primary containment penetrations, we have concluded that adequate protection is available to assure the integrity and leaktightness of the penetrations under accident situations.

The design of the primary containment structure is based on the applicable codes and regulations of the American Society of Mechanical Engineers, Boiler and Pressure Vessel Code, Sections VIII and IX with certain nuclear case interpretations, American Society for Testing and Materials Standards, and the American Institute of Steel Construction.

Considerations of accident pressure, jet loads, thermal load, dead load, external load, seismic loads have been accounted for in the containment design. The various loadings have been considered together in logical and conservative combinations. Under these critical load combinations the stresses in main load-carrying elements will be within the applicable code requirements.

The materials of construction have been selected in accordance with, and have been given a degree of attention in construction appropriate to, the critical nature of the structure. As part of the quality assurance program, the certified mill test reports were reviewed to assure their compliance with the material specifications. Shop and field fabrication techniques were closely controlled in order to ensure that a structure of the requisite quality had been achieved. Radiographic and magniflux techniques were used as required by the applicable sections of the ASME Code. We conclude that this structure has been designed and built to give satisfactory service over the design life of the facility.

3.3.2 <u>Testing and Surveillance</u>.

An overpressure test required by the ASME Code at 115% of the design pressure, 71.3 psig, has verified that the primary containment has been constructed in accordance with the intent of the design and will meet its structural and leakage performance requirements. Integrated leak rate tests will be performed prior to initial plant operation at test pressures of 20 and 35 psig. To verify the plant's continued leaktightness integrity, integrated leakage testing will be performed at 20 psig. After the initial preoperational leakage test, additional tests will be performed on a schedule corresponding to a 1, 2, 4 and every 4 years thereafter frequency provided the containment leakage remains within the allowable limit (i.e., a leak rate of 1.0% of the volume per day at a pressure of 35 psig). We conclude that the testing program is adequate to provide assurance of containment integrity throughout the service lifetime of the facility.

3.3.3 Containment Spray System

The Oyster Creek containment heat removal spray system consists of two independent spray-cooling loops. Each loop will pump water from a ring header connected to the containment absorption pool through heat exchangers cooled by the emergency service water system into a pair of spray headers located in the containment drywell. The water spray from the drywell spray headers removes heat from the drywell atmosphere, and flows by gravity back to the absorption chamber thereby completing the flow circuit.

Each of the containment spray loops has redundancy in active components (i.e., double pumps and valves) which provides protection against loss of any active component. Since all automatic valves in the system will be kept normally open (except for testing) during plant operation, actuation of containment spray depends only on operation of pumps. Passive failures of the piping system could also be tolerated without reducing the capability of the system. On the basis of our review, we conclude that the containment spray system is acceptable.

3.3.4 Containment Inerting System

The containment atmosphere control system is designed to maintain an inert atmosphere within the primary containment to preclude possible hydrogen-oxygen reaction that may occur as a consequence of a highly unlikely loss-of-coolant accident. The containment is purged with nitrogen gas before reactor operation and the oxygen concentration is maintained at less than 5% which will provide a margin against a hydrogen-oxygen reaction.

The system is located external to the drywell. Piping and component design up to and including the first two isolation valves will meet the requirements for Class I structures. The system also will be used to detect gross leakage paths in the primary containment boundary. This assures a continuous monitoring of containment integrity during plant operation. We conclude that the system as proposed by the applicant provides an adequate means for establishing and assuring an inert atmosphere within containment and a means to continuously monitor containment integrity.

3.4 Secondary Containment

The secondary containment or reactor building encloses the primary containment structure (drywell and absorption chamber). It consists of reinforced concrete substructures to the elevation of the refueling floor, topped by a conventional steel building frame with insulated metal siding.

The building contains the reactor servicing facilities, new and spent fuel storage facilities, and reactor auxiliary systems including the isolation condenser system, demineralizers, standby liquid control system, control rod hydraulic system, and the standby gas treatment system.

The standby gas treatment system is designed to minimize the release of radioactive materials to the environment during a loss-of-coolant accident or whenever a high level of radioactivity exists in the reactor building. The system consists of two low capacity exhaust fans and two filtering trains of gas and particulate filters. Each train is capable of limiting the leak rate to 100% of the reactor building volume per day under neutral wind conditions. The fans are sized to maintain the reactor building pressure at a negative pressure of 0.25 inch of water.

A test program will be conducted to demonstrate the design capability of the secondary containment. Additional secondary containment capability tests will be conducted during various meteorological conditions and at each refueling outage. The charcoal filters of the standby gas treatment system will be tested to demonstrate a halogen removal efficiency of not less than 99%, using freon gas. The particulate filters will be tested using DOP to demonstrate a particulate removal efficiency of not less than 99% for particulate matter larger than 0.3 micron. We conclude that the design features and testing program for the reactor building and standby gas treatment system are adequate to demonstrate the capability to minimize the release of radioactivity to the environment.

3.5 Other Plant Systems

3.5.1 Emergency Core Cooling System (ECCS)

3.5.1.1 General

The principal subsystems that make up the ECCS for Oyster Creek are the auto-relief system and the two core spray systems. In addition, for situations involving loss of offsite power, high pressure coolant injection capability (FWCI) using the existing feedwater system will be provided following initial plant operation. To accomplish this, the onsite power system will be modified, primarily by the addition of another diesel, as described in Section 4.0. We have reviewed the mechanical design and functional performance for the FWCI and find them acceptable.

In the event of a small break in the primary system without high pressure coolant injection capability, the auto-relief system depressurizes the reactor pressure vessel to permit operation of the low pressure core spray system before excessive fuel cladding heating occurs. The auto-relief system consists of four electromagnetic pressure relief valves located in pairs on each main steam line inside the drywell of the primary containment vessel. All four valves are programmed to operate on initiation of the auto-relief system, but only three are needed to assure adequate core cooling.

The core spray subsystem of the ECCS consists of two independent loops; each loop has redundancy of active components (i.e., double pumps and valves). Either loop is adequate to cope with the complete range of break sizes for loss-of-coolant accidents.

The feedwater system consists of three condensate and three feedwater pumps. One pump of each type will be used for the feedwater coolant injection system (FWCI). When the design modifications are completed, these pumps will be capable of operation from electrical power generated onsite. The Commission imposed the requirements in this area subsequent to the design of the facility; however, because the FWCI is a redundant safety feature, we have concluded that its installation may be deferred until the first scheduled extended outage of the plant.

3.5.1.2 ECCS Functional Performance

The ECCS is provided to mitigate the consequences of loss-of-coolant accidents resulting from any size rupture of the primary system piping or equipment. The break spectrum considered included breaks equivalent to that resulting from pump and valve seals leakage as well as double-ended pipe failures. The largest rupture considered during our evaluation was the double-ended rupture of a 26-inch recirculation line which is equivalent to a break area of 6.22 ft^2.

The applicant stated that the Oyster Creek ECCS design criterion was that no clad melt would result for any postulated primary system rupture up to and including the double-ended rupture of a recirculation pipe. We did not accept this as the sole criterion because in our view the peak fuel rod cladding temperature should be limited to a temperature such that reasonable assurance is provided that the ECCS would terminate the temperature transient and assure an intact core geometry for effective long-term cooling. Based on our review of the available data in this regard, we concluded that peak fuel rod cladding temperatures should not exceed about 2000°F. Furthermore, the functional aspects of the core spray cooling are sufficiently well determined by tests and analysis to give reasonable assurance of its efficacy when clad temperatures are held to less than 2000°F. The results of the applicant's analysis indicate that the maximum predicted temperatures for the entire spectrum of break sizes and locations that could occur in the design bases accidents do not exceed 2000°F. In addition, when the proposed FWCI is available, the maximum predicted temperatures would be less than 1800°F. Consequently, we conclude that there is reasonable assurance that the core spray system would be effective in the unlikely event of a loss-of-coolant accident.

3.5.1.3 Mechanical Design of the ECCS

Core spray piping external to the reactor vessel is designed to the stress limits set forth in the ASA B31.1 1955 Piping Code for maximum operating loads in combination with the design earthquake. Analyses of the piping system to determine the location of seismic snubbers and restraints have been reviewed by our seismic consultant, Dr. Newmark. He concluded, and we agree, that the design of the piping system is adequate to withstand the seismic conditions applicable to this facility.

The core spray spargers are located inside the reactor pressure vessel. Each sparger consists of two segments which form a ring header. Each segment is attached to the internal shroud at the inlet piping connection and is supported along the inner periphery of the shroud by saddle brackets. The applicant has indicated that the stresses are within Section III of the ASME Code allowables for all loading conditions including accident loads in combination with seismic loads even though they were not originally designed for combined accident and seismic loads. We conclude that this design basis produces an acceptable margin of safety for this facility.

The supply of water for the core spray is taken from the torus via a ring header and associated piping. Should any of these components fail, the water from the torus would drain into the lower part of the reactor building resulting in a flooded level of approximately eight feet. In the design as originally proposed by the applicant, this would lead to flooding of the core spray and containment spray pumps. The plant has been modified to preclude such an event by (a) connecting the fire water system to the core spray systems, (b) sealing all penetrations into the pump compartments, and (c) providing water-tight doors at the entrances (from the torus or center room) to the pump compartments. We conclude that these changes provide assurance that sufficient water for core cooling would be available in the highly unlikely event of excessive leakage from the piping systems.

3.5.2 Auxiliary Systems

The service water system consists of an intake structure, normal service and emergency water pumps, circulating water pumps, and an intake tunnel and discharge canal. During normal plant operation the normal service water pumps provide cooling to the reactor building closed cooling system and the four circulating service water pumps provide cooling to the main condenser and turbine building closed cooling system. None of these components is required to conduct a safe plant shutdown. An interconnection is provided to enable turbine building closed cooling system cooling with a normal service water system pump when necessary due to load or shutdown conditions. The four emergency service water pumps provide cooling, two pumps in each of two loops, to the containment spray heat exchangers. These components are necessary to remove decay heat following an accident and have been designed and fabricated to standards reflecting the importance of the function performed.

Conductivity monitors are provided in the feedwater piping in the hotwell region to detect leakage of circulating water (saline) into the condenser primary side. Radiation monitors are provided on the discharge of the service water and on the reactor building closed cooling loop.

The emergency service water (saline) system is maintained at higher pressure than the system it services and is also monitored by radiation detectors, one on the outlet of each of the four lines from the containment spray heat exchangers.

A reactor cleanup system is used to maintain the quality of the reactor coolant within specified limits. A reactor shutdown coolant system is also provided to remove decay heat from the reactor when it is in a shutdown condition.

We have reviewed the systems described above and conclude that they are acceptable.

3.5.3 Fuel Handling and Storage

Fuel handling operations are carried out using facilities provided for unloading and storing of new fuel in the reactor building, transferring and unloading of new assemblies into the reactor core, underwater removal of spent fuel assemblies from the reactor core, transfer of spent fuel assemblies from within the reactor containment to storage in the spent fuel pool, and offsite shipment of spent fuel assemblies for reprocessing in a specially designed cask.

During refueling, transport to the spent fuel storage pool, and during storage, spent fuel will be continuously submerged in water. The spent fuel storage racks in the pit are arranged to ensure a subcritical array. During refueling and storage, personnel will be protected by water and/or concrete shielding. Systems are provided to monitor spent fuel pool water temperatures and activity. In addition, sufficient interlocks have been established to prevent manipulations which could result in fuel damage during the refueling operation.

3.5.4 Control Room

The control room is located on the operating floor of the turbine building and contains all necessary controls and instrumentation for operation of the reactor, turbine-generator and auxiliary systems. The control room is designed to be occupied during design basis accident conditions as well as during normal operation. Although specific provisions were not made in the design, the equipment necessary to conduct safe shutdown can be operated remotely from outside the control room.

The control room has adequate instrumentation and controls for controlling the reactor plant in a safe manner. While all reactor protection and engineered safety features are automatic, facilities for manual operation of the safety features are also provided in the control room.

We have evaluated the design of the reactor control room with respect to the adequacy of the shielding during the design basis accident, and the potential doses during ingress and egress subsequent to an accident. Our calculations show that adequate shielding has been provided to limit the doses to an operator to within the yearly occupational limits set forth in 10 CFR Part 20.

3.5.5 Radwaste Systems

The applicant states that the purpose of the radwaste system is to treat and dispose of all types of solid, liquid, and gaseous wastes accumulated during operation of the facility.

The solid radwaste system serves to collect, process, and package items such as filter sludge, spent resins, and equipment originating in the primary system for offsite disposal. The material is dewatered in a centrifuge, compressed into 55-gallon drums, or mixed with concrete in preparation for shipment, depending on the quantity and activity level.

The gaseous radioactive waste control system is designed to process non-condensible gaseous products from the main condenser to limit fission product release to the environment. A 30-minute holdup capability is provided to allow radioactive decay of short lived products prior to stack release. The stack gas is continually monitored.

The liquid radioactive waste system collects, treats, and disposes of all liquid wastes generated within the facility. All liquid wastes are collected, sampled and discharged on a batch basis, so that inadvertent discharge of high activity waste is unlikely.

We conclude that these systems are adequate to assure that the 10 CFR Part 20 limits will not be exceeded.

4.0 ELECTRICAL POWER

The onsite electrical power system will utilize two redundant 2500 kw diesel generator units arranged in a split-bus configuration. Each generator is rated at 2500 kw continuous, and 2750 kw for 2000 hours per year. Maximum emergency loads are 2590 kw. Thus, a 6% margin is available, assuming one diesel generator has failed. The internal distribution system consists of two independent 4160 volt emergency busses, each of which is directly energized by one of the diesel generators. The separation extends through the downstream 480 volt sections. The generators will not be connected in parallel. A manual cross-tie between busses is provided; however, it will be closed only when one generator has failed.

Offsite electrical power is available from any one of four lines (two 230 kv and two 34 kv), and is fed into the emergency busses by two 34/4.16 kv startup transformers. Each startup transformer energizes one of the emergency busses.

As noted previously a diesel will be added to accommodate the proposed FWCI system. We have reviewed the preliminary design and conclude that it is satisfactory. The applicant has committed to provide the final design details to us prior to system installation. We will review the design to determine that it meets appropriate criteria and will not result in overloading of cable trays, as recommended in the ACRS letter. The third diesel generator will be operated in parallel with one of the existing generators to furnish the power necessary for operation of the FWCI system. It is anticipated that the installation will be accomplished during the first scheduled extended outage of the plant.

Our evaluation has led us to conclude that the electrical power system for Oyster Creek, including the DC portions is adequate.

5.0 INSTRUMENTATION AND CONTROLS

5.1 Protection System

Oyster Creek is the first of the General Electric boiling water reactors to utilize in-core nuclear instrumentation.

The Nuclear Instrument system consists of Source Range Monitors (SRM), Intermediate Range Monitors (IRM) and Local Power Range Monitors (LPRM). The power range monitors provide individual continuous measurements of local power level throughout the core as well as average power level in the core quadrants. The SRM system uses pulse counting techniques and derives period information which is displayed. There is no period scram. The IRM system uses the "Campbell" measurement technique and consists of eight channels of instrumentation feeding eight variable range amplifiers. Reactor scram is initiated when (at least) one IRM channel in each of the two protection channels is driven to the upscale trip point. The LPRM system consists of 125 independent channels which utilize miniature fission chambers as sensors. The outputs of 64 LPRM channels are combined (averaged) as eight distinct Average Power Range Monitor (APRM) channels, each APRM channel being fed from eight LPRM channels located in a particular quadrant. There are two APRM channels in each quadrant. Each is connected to a different channel of the dual channel protection system. Upscale tripping provides scram (1/2 x 2 logic) and rod-block (1/8 logic).

Power/Flow protection (rod-block) is provided by flow signals which continually adjust the upscale trip points of the APRM channels.

A Traversing In-Core Probe (TIP) may be inserted into the core to obtain flux distributions, and to calibrate the LPRM system.

Instrumentation on the main steam lines has sufficient sensitivity to detect early signs of gross failure of fuel elements. As operating experience is gained with the facility it might be possible to improve the use of these instruments to provide the operator with an early indication of fuel failures. Concern in this area was stated by the ACRS. We will review this matter further during the eighteen-month term of the provisional operating license.

Five sets of instrument channels respectively monitor the following process system parameters and provide scram capability:

a. High Reactor Pressure

b. High Primary Containment (drywell) Pressure

c. Low Reactor Water Level

d. Low Condenser Vacuum

e. High Radiation, Main Steam Lines

Each is monitored by four independent channels connected in 1/2 x 2 logic. Scram is also initiated upon loss of voltage to the protection system, upon main steam line isolation (both lines), and manually. Each channel consists of two independent subchannels made of relay contacts controlled by the various channels of the protection system instrumentation. A subchannel, in turn, controls one relay. The tripping of a subchannel equivalent to tripping the respective channel, and tripping both channels of the dual system scrams the reactor.

We have reviewed the design of the duel channel protection system, including the containment isolation system, and have concluded that it conforms to all applicable criteria. We have also independently verified the applicant's analyses that the Intermediate Range Monitors obviate the need for period scram.

Based on the foregoing, we conclude that the design of the Oyster Creek protection system is acceptable.

The applicant has recently made an audit of plant electrical design and construction features and has found several deficiencies. These deficiencies relate to separation of redundant cabling, separation of redundant sensors, and cable tray loading. As noted in the ACRS letter, the applicant is correcting the deficiencies. Before issuance of the license we will determine that the deficiencies have been corrected.

5.2 Rod Block

The rod block function serves to protect the core from local transients induced by improper control rod withdrawal. The system is designed such that four APRM channels (one per quadrant) de-energize the rod selector circuits. Trip logic is one out of eight. Our review indicates that the rod block system is redundant and testable, and is therefore acceptable.

5.3 Refueling Interlock

The Refueling Interlock system is essentially an arrangement of electrical interlocks between the fuel hoist mechanisms and the control rod drives such that a loaded hoist cannot be over the core when more than one rod is in a withdrawn condition. Our analysis shows that, with the mode switch in the "Refuel" position, the system meets the single failure criterion, and is fail-safe upon voltage loss. If, during refueling operations, the mode switch is placed in the Run or Shutdown position, a scram will occur. If the switch is in the Startup position, as occasionally required during refueling, a portion of the total interlock arrangement is bypassed in order to allow the withdrawal of more than one rod. We find this design feature to be satisfactory in view of the brief duration of such operation and the additional administrative controls which would be imposed during such operation.

Based on the foregoing, we conclude that the design of the Refueling Interlock system is acceptable.

5.4 Containment Spray System

There are two independent systems, each with its own spray header. Within each system there are two spray pumps and two service water pumps. Each system is respectively energized from one of the two emergency busses, and can provide full safety feature action.

The starting of each system is initiated by instrumentation which is independent of that used for starting the other system. Within a system, starting is initiated in response to 2 of 2 high drywell pressure in coincidence with 2 of 2 low-low reactor vessel water level. Both systems start simultaneously and operate independently of each other.

On the basis of our review we conclude that the instrumentation and controls for the containment spray system are acceptable.

5.5 Core Spray System

There are two independent core spray systems, each having redundant active components. Under conditions of automatic initiation, each system is connected to a different emergency bus.

The instrument channels which initiate system No. 1 are distinct from those which initiate system No. 2. Within each system the instrument logic is as follows: 1 of 2 low-low water level or 1 of 2 high drywell pressure, and 1 of 2 low reactor pressure. Each of the two core spray systems is controlled by its own starting and sequencing logic circuits (with inputs from the respective instrument channels) which attempt to start one main pump and one booster pump in each system, and open the respective discharge valves when reactor pressure has diminished sufficiently. Sequencing to an alternate pump occurs only if a preferred pump fails to start. Sequencing does not extend beyond a system.

Based on our analysis we have concluded that the instrumentation and controls for the core spray system are acceptable.

5.6 Auto-Relief System

The auto-relief control system consists of two redundant relay matrices, either of which can operate all four valves. The instrumentation which actuates one matrix is distinct from the instrumentation for the other matrix. Within a matrix the logic is as follows: 2 of 2 high drywell pressure and 2 of 2 low-low-low (triple low) water level. Thus, although there is no redundancy within a matrix, the "or" logic between the two matrices makes the total system redundant. Based on our review, we have concluded that the design of the auto-relief system is acceptable.

6.0 ANALYSES OF DESIGN BASIS ACCIDENTS

The accident to determine compliance with the guidelines established in 10 CFR Part 100 for this facility is the accident involving loss-of-coolant inside the drywell. The others considered are the refueling accident, steamline break accident outside the drywell and the control rod drop accident.

The results of our analyses for these accidents are summarized in the following sections and the doses which we have calculated using conservative assumptions are summarized in the following table. We have assumed only 90 percent efficiency for halogen removal as compared with the 99 percent which the applicant believes will be achieved. Credit for release of activity from the 110 meter stack was given except for the steamline break and control-rod-drop accidents.

TABLE 6.0

DOSE SUMMARY

Accident	Two Hour @ 0.25 Mile (exclusion area radius) (rem) Thyroid	Whole Body	Course of Accident @ 2 Miles (rem) (low population zone radius) Thyroid	Whole Body
1. Loss-of-Coolant	170	6	85	6
2. Refueling	46	<1	<1	<1
3. Control-Rod - Drop	30	<1	10	<1
4. Steamline Break	45	<1	2	<1

The meteorology used in our calculations of the consequences of the refueling, loss-of-coolant, and control-rod-drop accidents was as follows: Fumigation conditions were assumed for the two-hour dose calculations at the site boundary. For the maximum doses at the low population zone distance, we assumed the cloud centerline dilution factor that results from the use of an envelope of Pasquill types with a 110 meter release height. From one to thirty days after the accident we assumed that the wind blows into a 22-1/2^{o} sector 33% of the time with the occurrence of Pasquill Type C and a wind speed 3 m/sec, Type D and a wind speed of 3m/sec, and Type F and a wind speed 2m/sec, 33% of the time each.

For the steamline-break and control-rod-drop accidents, ground release and Type F and a wind speed of 1 m/sec were assumed for the 2-hour doses at the site boundary; for the low population zone doses for the first 24 hours of the accidents, ground release and Type F and a wind speed of 2 m/sec mixed uniformly into a 22-1/2^{o} sector were used, and for the one to thirty-day doses the same meteorology as described above was used.

As can be seen from the data in the above table the doses resulting from accidents are less than the 10 CFR Part 100 guidelines.

6.1 Loss-of-Coolant Inside the Drywell

In calculating the consequences of the loss-of-coolant accident associated with 100% fuel perforation, we have assumed fission product release fractions as suggested in Technical Information Document 14844, "Calculation of Distance Factors for Power and Test Reactor Sites" that are released from the core, i.e. 100% of the noble gases, 50% of the halogens, and 1% of the solids.

A primary containment leak rate of 1.25 percent of the containment volume per day was assumed to remain constant for the duration of the accident. Although the containment design leak rate is 0.5 percent per day, our safety evaluation conservatively assumed an accident leak rate of 1.25 percent per day for the duration of the accident.

We have assumed a 90% halogen removal efficiency of the charcoal absorbers of the standby gas treatment system prior to a release to the environs via the stack. In our analysis, we took the conservative approach of assuming leakage from the drywell goes directly to the standby gas treatment system without mixing and then out the stack at 110 meters above ground level.

In addition to the radiological consequences of an assumed loss-of-coolant accident, the potential for radiolytic decomposition of water has been considered. The effects of the possible decomposition would result in the production of some gaseous hydrogen and oxygen in the containment atmosphere. This matter is undergoing thorough review by industry, Oak Ridge National Laboratory, Battelle Memorial Institute, and the Commission's Division of Reactor Licensing.

The significance of the matter is not completely understood or known at this time. Preliminary studies by the applicant suggest that the extent of the decomposition reaction may be limited by back-reaction rates. As noted in the ACRS letter, we will evaluate further information as it becomes available and will take action as necessary. We conclude that the outcome of these efforts would be available to reevaluate the matter within the eighteen-month term of the provisional operating license.

5.2 Control-Rod-Drop

In the control-rod-drop accident it is assumed that a bottom entry rod has been fully inserted and has stuck in this position unknown to the reactor operator. It is then assumed that the drive becomes uncoupled and withdrawn from the rod. Subsequently, it is assumed that the rod falls out of the core inserting an amount of reactivity corresponding to the worth of the rod.

Hot standby is the worst operating condition at which the accident could happen both because a higher energy release is calculated for this condition and because a path for the unfiltered release of fission products could exist through the mechanical vacuum pump on the condenser. A rod reactivity worth of 2.5% $\Delta k/k$, the highest worth rod permitted by operating procedures, was assumed in the analysis. This reactivity addition would produce an excursion with a minimum reactor period of 8.5 milliseconds and a total energy generation of 4000 Mw-sec, resulting in a peak fuel energy density of about 200 cal/gm (average across the peak fuel pellet) and perforation of 330 fuel rods.

We have evaluated the consequences of the control-rod-drop accident assuming that 330 fuel rods would fail, releasing 100 percent of the noble gases and 50 percent of the halogens from the affected rods to the primary system. Of the halogens released from the affected rods, 90 percent is assumed to be retained in the primary system and one-half of the remaining halogens is assumed to be removed by plate-out. All of the noble gases and 2.5% of the halogens would be released from the primary system through the condenser vacuum pump system to the atmosphere through the stack.

Because the vacuum pump on the condenser might provide a channel for release of fission products, we required that it be isolated whenever high radioactivity exists in the main steam lines.

6.3 Refueling Accident

The refueling accident is assumed to occur 24 hours after shutdown. During fuel handling operation, a fuel bundle is assumed to fall onto the core with sufficient force to physically damage (perforate) 445 fuel rods with consequent release of 20% of the noble gases and 10% of the halogens from the damaged rods into the reactor building. Ninety percent of the halogens released from the perforated fuel rods are assumed to remain in the refueling water. The remaining airborne fission products (20% of the noble gases and 1% of the halogens) within the building are assumed to be discharged to the atmosphere through the standby gas treatment system, with an iodine filter removal efficiency of 90%, and stack over a 2-hour period.

6.4 Steamline Break Outside Containment

The break of a main steamline outside of both the drywell and the reactor building represents a potential escape route for reactor coolant from the vessel to the atmosphere without passage through the primary containment or the reactor building.

The steamline break would be sensed by either increased pressure drop across the steamline venturi or increased temperature in the pipe tunnel. The steamline isolation valves would start to close within 0.5 second after the steamline break. We have assumed an isolation valve closure time of 10 seconds. The valve closure time terminates the accident.

The primary coolant activity used in the calculations corresponds to the total iodine activity limit of 20 μc/cc, given in the Technical Specifications.

6.5 Conclusion

On the basis of our evaluation, the radiological doses that could result from any of the design basis accidents are well within the guideline values given in 10 CFR Part 100.

7.0 EMERGENCY PLANNING

The applicant has described a comprehensive plan for coping with the unlikely event of an accident which might affect the general public. Arrangements have been made to deal with radiological emergencies with the responsible agencies of the State of New Jersey and appropriate local officials.

Members of the applicant's on-site staff will furnish information concerning release rates and will cooperate with state and local officials in providing technical advice concerning the potential off-site effects throughout the course of any accident affecting the general public, in accordance with prearranged plans. The applicant possesses the capability of providing offsite monitoring to supplement that provided by the State of New Jersey.

In addition, technical assistance is available through the Radiological Emergency Assistance Team program of the AEC. The applicant has established liaison with the team at the New York Operations Office of the AEC.

Jersey Central has contracted with a specialist in the field of radiation medicine to provide medical consultant services and continuing professional training for the local hospital staff in Toms River, New Jersey. This hospital has agreed to provide medical support to the Oyster Creek facility, and to make available such support as might be required in the event of an accident at the site, whether or not such an accident should involve the general public.

We have concluded that the arrangements made by the applicant to cope with the possible consequences of accidents at the site are both reasonable and prudent, and that there is adequate assurance that such arrangements will be satisfactorily implemented in the unlikely event that they are needed.

8.0 CONDUCT OF OPERATIONS AND TECHNICAL QUALIFICATIONS

Responsibility for safe operation of the plant is vested in the Station Superintendent. He reports to the Manager of Generating Stations, who, in turn, is responsible to the Vice President of the Jersey Central Power & Light Company.

Within the onsite operating organization, responsibility for day-to-day operation of the facility rests with the Operations Supervisor, reporting to the Station Superintendent. The Operations Supervisor will be a licensed senior reactor operator, as will each Shift Foreman. The operating crew duty will consist of two Control Room Operators, each of whom will be a licensed reactor operator, and two unlicensed Equipment Operators, all under the supervision of the Shift Foreman.

The qualifications of individuals initially proposed to fill professional and semi-professional positions in the onsite operating organization have been described in the Safety Analysis Report. The minimum qualifications for these functional positions are described in the Technical Specifications. We have examined the qualifications of the incumbents and pending satisfactory completion of necessary examinations for appropriate licenses we conclude that the professional staff is technically competent to operate the facility.

Engineering support to Jersey Central will be provided by a special nuclear group within the General Public Utilities (GPU) organization, of which Jersey Central is a part, as well as by General Electric and specialist consultant firms. The GPU staff is familiar with the plant and is capable of handling the preparation and review of design changes and plant modifications originating at the Oyster Creek site. In addition, the applicant has demonstrated his intent to utilize the services of consultants as necessary to augment the nuclear capability of the GPU engineering staff. General Electric will be an active participant in the startup and initial operation of the plant and will continue to make available direct technical support to the Jersey Central staff throughout the operating lifetime of the facility. On these bases, we conclude that adequate engineering capability will be available through GPU and specialist consultants to support the applicant's operating staff.

The applicant proposes to use what has become a relatively conventional two-level committee structure to perform review and audit of plant operation. The first of these committees, the Plant Operations Review Committee, which comprises the senior members of the onsite staff, acts in an advisory capacity to the Station Superintendent. Independent audit of plant operation is provided by the General Office Review Board, the Chairman and Vice-Chairman of which are appointed by name by the president of the company. The responsibilities and authorities for these committees are delineated in the Technical Specifications. We conclude that the review and audit structure proposed by the applicant is satisfactory.

Based on the above considerations, we conclude that the applicant is technically qualified to operate the plant and has established effective means for continuing review, evaluation, and improvement of plant operational safety.

9.0 TECHNICAL SPECIFICATIONS

The applicant's proposed Technical Specifications to the license for Oyster Creek are presented in Amendment No. 44. Included are sections covering safety limits and limiting safety system settings, limiting conditions for operation, surveillance requirements, design features and administrative controls.

We have reviewed these proposed Technical Specifications in detail and have held numerous meetings with the applicant to discuss their contents. Some modifications to the proposed Technical Specifications submitted by the applicant were made to more clearly describe the allowed conditions for plant operation. Based upon our review, we conclude that normal plant operation within the limits of the Technical Specifications will not result in potential offsite exposures in excess of Part 20 limits. Furthermore, the limiting conditions of operation and surveillance requirements will assure that necessary engineered safety features will be available in the event of malfunctions within the plant.

10.0 REPORT OF ADVISORY COMMITTEE ON REACTOR SAFEGUARDS (ACRS)

As noted previously, the ACRS has reviewed the application for a provisional operating license for the Oyster Creek Nuclear Power Plant Unit No. 1. The Committee completed its review of the facility at the 104th meeting held during December 5-7, 1968. A copy of the report of the ACRS, dated December 12, 1968, is attached.

The ACRS, in its letter, made several recommendations to be followed during operation of the facility. These matters have been considered in our evaluation. They include periodic inspection of the reactor high pressure coolant system (Section 3.2); review of the design criteria for the future Feedwater Coolant Injection System (Sections 3.5.1 and 4.0); completion of the remedial program on the separation of redundant protection system components and circuits

(Section 5.1); study of the possible effects of radiolysis of water in the unlikely event of a loss-of-coolant accident (Section 6.1); and improvement of the capability of the steam line monitors to detect early signs of gross failure of fuel elements (Section 5.1).

In addition, the Committee noted the difficulties inherent in direct inspection of the pressure vessel welds after the reactor is in service and recommended that alternative means for assuring continued pressure vessel integrity be studied, and implemented to the degree practical. The ACRS also recommended that supplemental and potentially more sensitive methods of primary system leak detection be studied, evaluated and implemented if significant improvements in detection capability can be realized.

The applicant has agreed to see that the recommendations of the ACRS are carried out. We will follow the recommendations of the ACRS on all of the foregoing matters during operation of the facility under the eighteen-month term of the provisional operating license. The ACRS concluded in its letter that if due regard is given to the foregoing, the Oyster Creek Unit No. 1 can be operated at power levels up to 1600 Mwt without undue hazard to the health and safety of the public.

11.0 COMMON DEFENSE AND SECURITY

The application reflects that the activities to be conducted would be within the jurisdiction of the United States and that all of the directors and principal officers of the applicant are American citizens. The applicants are not owned, dominated or controlled by an alien, a foreign corporation or a foreign government. The activities to be conducted do not involve any restricted data, but the applicant has agreed to safeguard any such data which might become involved in accordance with the requirements of Part 50. The applicant will rely upon obtaining fuel as it is needed from sources of supply available for civilian purposes, so that no diversion of special nuclear material from military purposes is involved. For these reasons and in the absence of any information to the contrary, we have found that the activities to be performed will not be inimical to the common defense and security.

12.0 CONCLUSION

Based upon our review of the application as presented and discussed in this evaluation and the report of the Advisory Committee on Reactor Safeguards, we have concluded that the Oyster Creek Unit No. 1 can be operated as proposed without endangering the health and safety of the public.

Peter A. Morris

Peter A. Morris, Director
Division of Reactor Licensing

December 23, 1968

APPENDIX

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
UNITED STATES ATOMIC ENERGY COMMISSION
WASHINGTON, D.C. 20545

DEC 12 1968

Honorable Glenn T. Seaborg
Chairman
U. S. Atomic Energy Commission
Washington, D. C.

Subject: REPORT ON OYSTER CREEK NUCLEAR POWER PLANT UNIT NO. 1

Dear Dr. Seaborg:

During its 104th meeting, December 5-7, 1968, the Advisory Committee on Reactor Safeguards completed its review of the application by the Jersey Central Power and Light Company for a license to operate the Oyster Creek Nuclear Power Plant Unit No. 1 at power levels up to 1600 MW(t). During this review, the project has been considered at eight Subcommittee meetings (including one at the site) and four full Committee meetings. In the course of these discussions, the Committee has had the benefit of discussions with representatives of the Jersey Central Power and Light Company, the General Electric Company, the AEC Regulatory Staff and with consultants of these organizations. The Committee also had the benefit of the documents listed. The Committee previously discussed this project in a construction permit report dated August 28, 1964.

The Oyster Creek plant is the first of a new generation of boiling water reactors to be reviewed for an operating license; the increase of power level over that of previously licensed boiling water reactors is more than a factor of two. The time for construction of this plant was extended because of defective welds and stress-corrosion cracking in stainless steel portions of the pressure vessel envelope and internals. Items such as control rod stub tubes, nozzle safe-ends, and the core support ring were involved. These cracks were discovered during and after the system hydrostatic test. The causes of the stress-corrosion have not been definitely determined; however, studies to establish the effects of various contaminants are continuing. The Committee is satisfied that the repair procedures should prevent or minimize recurrence of stress-corrosion cracking.

The Committee wishes to emphasize the importance of periodic inspection of the high pressure coolant system in this and other reactors. The in-service inspection requirements for this reactor are stated in the Technical Specifications, and the Committee finds these adequate for initial operation. It is expected that experience with this first large BWR will give useful information regarding the practicality of inspection methods. The Committee endorses the applicant's proposal to review his in-service inspection program with the Regulatory Staff after four years of reactor operation. In view of the difficulties inherent in direct inspection of the bulk of the welds in the Oyster Creek pressure vessel after the reactor is in service, it is recommended that alternative means for assuring continued pressure vessel integrity be studied, and implemented to the degree practical.

It is recommended that supplemental and potentially more sensitive methods of primary system leak detection be studied, evaluated, and implemented if they provide significant improvements in measurement of leak rate, in the time needed to measure leak rate, or in distinguishing the nature of the leak. The study and evaluation should be completed within a year.

The emergency core cooling system will be supplemented in about a year by the addition of a third diesel generator. This extra source of power will allow the use of one feedwater pump (as well as one core spray system) in the case of the loss of off-site power. The Committee has reviewed the design criteria for this emergency Feedwater Coolant Injection System and recommends that the applicant submit the design for review by the Regulatory Staff prior to installation. In this regard, the Committee urges caution to avoid the overloading of cable trays.

The applicant has recently reviewed design and construction criteria in regard to the separation of redundant protection components and circuits. An audit of the Oyster Creek plant revealed some deficiencies in this respect, and the applicant is proceeding with a remedial program.

Studies are continuing on the possible effects of radiolysis of water in the unlikely event of a loss-of-coolant accident. These studies should be evaluated by the Regulatory Staff and appropriate measures taken as deemed necessary.

The applicant stated that instrumentation which senses radioactivity from the steam system can be used to provide early signs of gross failure of fuel elements. The Committee believes that, as operating experience is gained with the facility, the applicant should improve the utilization of this type of instrumentation for this purpose, particularly to provide the reactor operators with direct, early indication.

The Advisory Committee on Reactor Safeguards believes that, if due regard is given to the items mentioned above, the Oyster Creek Unit No. 1 can be operated at power levels up to 1600 MW(t) without undue hazard to the health and safety of the public.

Sincerely yours,

Original signed by
Carroll W. Zabel

Carroll W. Zabel
Chairman

References:
1. Jersey Central Power and Light Company Application for Reactor Construction Permit and Operating License for Oyster Creek Unit No. 1, Amendments No. 3 through 5 and 7 through 48.

2. Jersey Central Power and Light Company telegram, dated October 11, 1967, regarding Request for Permit for Fuel Loading and Testing of Oyster Creek Reactor Prior to Completion of Review of Application for Provisional Operating License.

3. Jersey Central Power and Light Company letter, dated February 9, 1968, transmitting General Electric Summary Report, dated February 2, 1968, regarding Reactor Vessel Problems.

4. Jersey Central Power and Light Company letter, dated April 9, 1968, regarding Oyster Creek Pressure Vessel Repair Program.

5. Jersey Central Power and Light Company telegram, dated July 3, 1968, regarding Oyster Creek Reactor Vessel Repair.

ADDENDUM TO THE SAFETY EVALUATION BY THE DIVISION OF REACTOR LICENSING U. S. ATOMIC ENERGY COMMISSION IN THE MATTER OF JERSEY CENTRAL POWER & LIGHT COMPANY OYSTER CREEK NUCLEAR POWER PLANT UNIT NO. 1 DOCKET NO. 50-219

Enclosure 2

This addendum to our safety evaluation in the matter of Jersey Central Power & Light Company (Oyster Creek Nuclear Power Plant Unit No. 1) deals with our analysis of certain changes in the design of the penetrations through the primary containment. This matter is treated in application Amendments Nos. 50 and 51 dated February 27, 1969 and March 25, 1969.

As described in Section 3.3 of our safety evaluation, the applicant had originally designed these penetrations according to Section VIII of the ASME Boiler and Pressure Vessel Code and to certain nuclear code cases. It was found that the penetration design would require massive restraints that would have affected the capability for maintenance and for in-service inspection of various components. Subsequently, the applicant re-evaluated its design approach using Section III of the ASME Boiler and Pressure Vessel Code. Because of higher allowable stresses permitted in Section III for the loading conditions and a more refined analysis of the loads, the applicant concluded that massive restraints inside the primary containment would not be required. In several instances, however, penetration restraints external to the containment were reinforced to accommodate accident loads to assure containment integrity.

On the basis of our review of the revised design described by the applicant in Amendments Nos. 50 and 51, we have concluded that there is adequate protection to assure the integrity of the primary containment penetrations under the various postulated accident loading conditions and that no change in our conclusions as to the safety of the facility need be made.

Original Signed by
Peter A. Morris

Peter A. Morris, Director
Division of Reactor Licensing

Date: APR 9 1969

Tran w/Amendt. #2 dtd 9/69

AUG 1 1969

DO NOT REMOVE

LICENSE AUTHORITY FILE COPY

ADDENDUM TO THE

SAFETY EVALUATION

BY THE

DIVISION OF REACTOR LICENSING

U. S. ATOMIC ENERGY COMMISSION

IN THE MATTER OF

JERSEY CENTRAL POWER & LIGHT COMPANY

OYSTER CREEK NUCLEAR POWER PLANT UNIT NO. 1

DOCKET NO. 50-219

Enclosure 3

This addendum to our safety evaluation in the matter of Jersey Central Power & Light Company (Oyster Creek Nuclear Power Plant Unit No. 1) deals with certain matters described in our notice of issuance of a 5 Mwt license as published in the Federal Register on April 17, 1969 (34 FR 6547). These involve: (1) modification of the standby gas treatment system, (2) additional review of the quality of certain piping in the facility, and (3) evaluation of preoperational testing of the containment isolation valves. Information regarding the foregoing was provided by the applicant in Amendment No. 53, dated June 12, 1969. Additional study of the examination methods and acceptance standards, which led to further inspection of certain components within the plant pressure boundary, was also conducted in accordance with requirements stated in a letter submitted to the applicant on July 29, 1969 confirming our conclusions stated orally to the applicant at a meeting held on July 10, 1969, and at earlier meetings.

As described in Section 3.4 of our safety evaluation, the applicant had originally designed the standby gas treatment system so that each train of the system was capable of limiting the leak rate to 100% of the reactor building volume per day under neutral wind conditions. The fans were to be sized such that a negative pressure of 0.25 in. of water could be maintained in the reactor building. The results of the initial preoperational tests on the installed system indicated that the system was inadequate to meet the performance requirements. Accordingly, the standby gas treatment system was modified to meet the required negative pressure and subsequently tested to demonstrate the performance of the system. These modifications included increasing the capacity of each fan from 1200 cfm to 5000 cfm. Results of these tests show that the required negative pressure of 0.25 in. of water was achieved. This change resulted in a greater leak rate into the reactor building.

This change in building leak rate does not affect the radiological consequences from postulated accidents. All fission products are processed through the filters which have been modified to accommodate the increase in flow rate. Therefore, we have determined that no change in our conclusions as to the safety of the facility in this regard need be made. The Technical Specifications have been changed to reflect the increased capacity of the fans.

The quality of certain piping, fittings and valves in the facility had not been completely established prior to issuance of a 5 Mwt license to Jersey Central Power & Light Company. Certain piping and fittings installed in the emergency condenser, the core spray, and the shutdown cooling systems were found to have inadequate nondestructive testing records. In addition, discrepancies between the markings on these components and the available records were found. The applicant has performed additional radiography, dye penetrant testing, metallographic analyses, mechanical tests and certain other tests to confirm that the components were suitable for nuclear service. Results of these actions are described in Amendment No. 53, dated June 12, 1969. However, following our review of this information, we concluded that certain additional inspections were required. These matters were discussed with the applicant at a meeting on July 10, 1969, at earlier meetings and confirmed in a letter dated July 29, 1969. The applicant performed these inspections which included additional ultrasonic and liquid penetrant testing on certain welds in the piping installed in the systems and radiographic examination of portions of each of the 16 six-inch safety valves. These actions were in addition to the examination originally performed on these components. Based on the information provided in Amendment No. 53 and the review of the results of additional inspections by representatives of the Commission, we conclude that the inspected components are suitable for nuclear service.

With regard to other valves and components located outside the pressure boundary, we have concluded that they will adequately perform their intended function. This conclusion is predicated on the results of the inspections and tests already conducted at the facility. These included visual, functional and hydrostatic testing that demonstrates the present adequacy of the components for the initial operation at full power. However, to assist us in our future inspections of plant operation, we will require the applicant to submit a report within one year, as set forth in the Technical Specifications, that will describe the nondestructive inspection methods used and acceptance standards specified and applied for pipes, fittings, valves and pumps outside the reactor coolant pressure boundary.

Initial preoperational testing of the primary containment indicated that certain containment isolation valves had excessive leakage. The valves in question are those installed in the main steam lines, which penetrate the primary containment boundary (reference Section 3.3 of our safety evaluation). Minor maintenance on the seating surfaces of the valves was performed and the valves were retested. Results of the retesting demonstrated that at a test pressure of about 20 psig the leakage from the valves was small, i.e., less than 11.5 cubic feet per hour. On this basis, we have amended the Technical Specifications to limit the leakage from any one valve. However, because of the previous limitations in the Technical Specifications, the total leakage from all valves remains the same. Therefore, there would be no increase in the postulated doses from those previously presented in the safety evaluation. Based on our review of this matter, we conclude that the valves are adequate with regard to performing the isolation function.

Based on review of the foregoing matters, we have added three items to the Technical Specifications. These are: (1) a requirement that the applicant submit a report within one year describing the results of the nondestructive testing program conducted on pipes, valves, fittings, and pumps of systems outside the pressure boundary; (2) for subsequent primary containment testing, a limit on leakage from any isolation valve; and (3) specification of filter train flow rate based on modification of the standby gas treatment system. These actions are designated as Amendment No. 1 to the Technical Specifications attached to Amendment No. 1 to License No. DPR-16.

On the basis of our review of the foregoing matters, we conclude that the facility can be operated at power levels up to 1600 Mwt without endangering the health and safety of the public.

ORIGINAL SIGNED BY
Peter A. Morris

Peter A. Morris, Director
Division of Reactor Licensing

Date: AUG 1 1969