November 16, 1970

SAFETY EVALUATION

BY THE

DIVISION OF REACTOR LICENSING

U. S. ATOMIC ENERGY COMMISSION

IN THE MATTER OF

CONSOLIDATED EDISON COMPANY OF NEW YORK, INCORPORATED

INDIAN POINT NUCLEAR GENERATING UNIT NO. 2

BUCHANAN, NEW YORK

DOCKET NO. 50-247

TABLE OF CONTENTS

1.0 INTRODUCTION

The Consolidated Edison Company of New York, Inc., (applicant) filed with the Atomic Energy Commission (AEC or Commission) an application dated October 15, 1968, for an operating license for its Indian Point Nuclear Generating Unit No. 2. Indian Point Unit 2 has been under construction since issuance of a provisional construction permit on October 14, 1966.

Indian Point Unit 2 is located on a 227-acre site on the east bank of the Hudson River at Indian Point, Village of Buchanan, in upper Westchester County, New York.

Indian Point Unit 2 is the first of the four-loop, current generation Westinghouse pressurized water reactor designs. It will be owned and operated by the Consolidated Edison Company of New York, Inc. The Westinghouse Electric Company (Westinghouse) is the principal contractor and has turnkey responsibility for the design, construction, testing, and initial startup of the facility. Westinghouse contracted with United Engineers and Constructors as architect engineer. Construction of the plant was performed by United Engineers until December 1969 when this function was assumed by WEDCO, a wholly-owned subsidiary of Westinghouse.

The operating license application is for a power level of 2758 megawatts thermal (MWt), the same as was requested in the construction permit application. Our evaluation of the engineered safety features (with the exception of the emergency core cooling system) and our accident analyses, have been performed for a maximum power of 3216 MWt.

Our evaluation of the thermal, hydraulic, and nuclear characteristics of the reactor core and the performance of the emergency core cooling system was for a power rating of 2758 MWt. Before operation at any power level above 2758 MWt is authorized, the regulatory staff will perform a safety evaluation to assure that the core can be operated safely at the higher power level.

Our technical safety review of the design of this plant has been based on Amendment No. 9 to the application, the Final Facility Description and Safety Analysis Report (FFDSAR), and Amendments Nos. 10-25, inclusive. All of these documents are available for review at the Atomic Energy Commission's Public Document Room at 1717 H Street, Washington, D.C. The technical evaluation of the design of this plant was accomplished by the Division of Reactor Licensing with assistance from the Division of Reactor Standards and various consultants to the AEC.

In the course of our review of the application, many meetings were held with representatives of the applicant to discuss the plant design and proposed operation. As a consequence of our review, additional information was requested, which the applicant provided by amendments to the application. A chronology of the principal actions relating to the processing of the application is attached as Appendix A to this safety evaluation. In addition to our review the Advisory Committee on Reactor Safeguards (ACRS) independently reviewed the application and met with both the AEC staff and the applicant on several occassions to discuss the plant. The ACRS report on Indian Point Unit 2, dated September 23, 1970, is attached to this Safety Evaluation as Appendix B. Appendices C through G include reports from our consultants on meteorology, hydrology, seismic and structural design, and radiological monitoring. Appendix H contains the staffs evaluation of the applicant's financial qualifications.

Based upon our evaluation of the plant as summarized in subsequent sections of this report, we have concluded that Indian Point Unit 2 can be operated at thermal power levels of up to 2758 MWt without endangering the health and safety of the public. Subsequent to the issuance of an operating license the unit will be required to operate in accordance with the terms of the operating license and the Commission's regulations under the surveillance of the Commission's regulatory staff.

2.0 FACILITY DESCRIPTION

Indian Point Unit 2 is one of three reactors currently planned for the Indian Point site. Indian Point Unit 2 is adjacent to Indian Point Unit 1, a 615 MWt pressurized water reactor plant that has been in operation since August 1962. Indian Point Unit 3, a plant similar to Indian Point Unit 2, received a provisional construction permit in August 1969, and is presently under construction at the Indian Point site. Each unit has its own auxiliary systems and safety features. The three units, however, will share a common inlet water canal and a common discharge canal. In addition, the controls for Indian Point Unit 2 and Indian Point Unit 1 are located in separate portions of a common control room.

The Indian Point Unit 2 pressurized water reactor is fueled with slightly enriched uranium dioxide in the form of ceramic pellets contained in zircalloy fuel tubes. Water serves as both the moderator and the coolant. Heat is removed from the reactor core by four separate coolant loops, each provided with a separate pump and steam generator. The heated water flows through the steam generators where heat is transferred to the secondary (steam) system. The water then flows back to the pumps to repeat the cycle. The system pressure is controlled by the use of a pressurizer in which steam and water are maintained in thermal equilibrium.

The secondary steam produced in the steam generators is used to drive the turbine generator. The heat of condensing steam is rejected to the circulating water system and discharged to the Hudson River. The condensate is then recharged to the steam generators to repeat the secondary cycle.

The primary coolant system includes the reactor, steam generators, primary coolant pumps, primary coolant piping, and the pressurizer. This system is housed inside the containment building which is a steel-lined, leak-tight reinforced concrete structure. The containment provides a barrier to the release to the environment of radioactive fission products that might be released inside the containment in the event of an accident. Auxiliary systems, including the chemical and volume control systems, the waste handling system, and additional auxiliary cooling systems, are housed separately, principally in the adjacent primary auxiliary building. The primary auxiliary building also houses components of the engineered safety features. A separate fuel handling building is provided for storage of spent fuel. A separate turbine building houses the turbine generator.

Control of the reactor is achieved by reactivity control using top entry control elements that are moved vertically within the core by individual control drives. Boric acid dissolved in the coolant is used as a neutron absorber to provide long-term reactivity control.

To assure reactor operation within established limits, a reactor protection system is provided that automatically initiates appropriate actions whenever plant conditions monitored by the system approach preestablished limits. The reactor protection system acts to shut down the reactor, close isolation valves, and initiate operation of the engineered safety features should any or all of these actions be required.

The engineered safety features include an emergency core cooling system that will cool the reactor core in the event of an accident that results in loss of the normal coolant, containment cooling and iodine removal systems that provide for removal of heat and radioactive iodine from the containment atmosphere should such action be required, and a hydrogen control system that will limit the accumulation of hydrogen within the containment in the event of a loss-of-coolant accident. A containment penetration pressurization system and seal water injection system are provided to assist in isolating the containment in the event of an accident and prevent the escape of fission products to the environment outside the plant.

3.0 SITE AND ENVIRONMENT

3.1 Population and Land Use

The Indian Point site consists of 227 acres in the town of Buchanan in upper Westchester County, New York, approximately 24 miles north of the New York City boundary line. The estimated population distribution in the vicinity of the site is presented in table 2.1.

TABLE 2.1

CUMULATIVE POPULATION

Distance (miles)	1960 (U.S. Census)	1980 (Projected)
0-1	1,080	2,100
0-2	10,810	20,900
0-3	29,630	59,520
0-4	38,730	78,800
0-5	53,040	108,060
0-10	155,510	312,640

The minimum radius of the exclusion area* for Indian Point Unit 2 is 520 meters. The applicant has chosen 1100 meters as the outer

*Exclusion area is defined in the Commission's Site Criteria, 10 CFR Part 100, as that area surrounding the reactor in which the reactor licensee has the authority to determine all activities including removal of personnel and property from the area.

boundary of the low population zone** because of the limited population within this distance from the plant.

The Commission's site criteria guidelines state that the population center distance*** should be at least 1-1/3 times the distance from the reactor to the outer boundary of the low population zone (LPZ), but also state that in applying this guide due consideration should be given to the population distribution within the population center. The nearest corporate boundary of Peekskill (population 19,000) is approximately 800 meters (0.5 miles) from Indian Point Unit 2. Because of the limited population within the low population zone (66) including that portion of Peekskill within the zone, and because Peekskill is of a generally industrial nature in the vicinity of the site and the resident population within and out to 1-1/3 times the low population zone distance is low, we concluded at the time of our construction permit review that the distance selected by the applicant for the exclusion area radius, the LPZ outer boundary, and the population center distance meet the intent of the 10 CFR Part 100 guidelines and are acceptable. On the basis of our evaluation of the potential radiological consequences of postulated design basis accidents,

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**Low population zone is defined in the Commission's Site Criteria, 10 CFR Part 100, as the area immediately surrounding the exclusion area which contains residents, the total number and density of which are such that there is a reasonable probability that appropriate protective measures could be taken in their behalf in the event of a serious accident.

***Population center distance is defined in the Commissions Site Criteria, 10 CFR Part 100, as the distance from the reactor to the nearest boundary of a densely populated center containing more than about 25,000 residents.

we conclude that the calculated doses presented in Section 11.0 of this evaluation are well within the guidelines of 10 CFR Part 100 for these distances.

3.2 Meteorology

The meteorology of the Indian Point site is affected by its position in a deep river valley. Consequently, the wind direction generally follows a pronounced diurnal cycle with unstable flow in the up-river direction during the daytime and stable flow in the down-river direction at night.

The applicant has presented the results of meteorological measurements taken at the site over a period of two years including windspeed, wind direction, and temperature lapse rate data for various heights. We have reviewed the data presented and conclude that they provide an adequate basis for selecting the meteorological parameters used in determining the routine effluent release limits and in evaluating the consequences of postulated accidents. The comments of our meteorological consultants, the Environmental Science Service Administration (ESSA) support this conclusion and are attached as Appendix C.

3.3 Geology and Seismology

During our review of this site prior to issuance of the construction permit for Indian Point Unit 2, we and our consultant, the U. S. Geological Survey, concluded that the geology of the site provides an adequate founding medium for the plant buildings and structures. No new developments have occurred during the construction permit review of Indian Point Unit 3 or otherwise since our construction permit review for Indian Point Unit 2 to change our previous conclusion on the acceptability of the geological and seismological features of the Indian Point site.

Maximum ground accelerations of 0.10g and 0.15g were used for the Operating Basis Earthquake* and the Design Basis Earthquake**, respectively. These values were selected at the time of the construction permit review. At that time we and our consultant, the U. S. Coast and Geodetic Survey, concluded that they were acceptable for the site.

A strong motion seismograph has been installed on a concrete slab directly on bedrock in the yard area of the plant to record data related to ground motion in the event of a seismic disturbance at or near the site. These data would be employed in an evaluation of the effects of the seismic disturbance to assure the capability for continued safe operation of the plant.

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*"Operating Basis Earthquake" for a reactor site is that earthquake which produces vibratory ground motion for which those structures, systems and components, necessary for continued operation without undue risk to the health and safety of the public are designed to remain functional.

**"Design Basis Earthquake" for a reactor site is that earthquake which produces vibratory ground motion for which those structures, systems, and components, necessary to shut down the reactor and maintain the unit in a safe shutdown condition without undue risk to the health and safety of the public are designed to remain functional.

3.4 Hydrology

The applicant has reevaluated the potential flooding that could occur at the site. The following hypothetical flood conditions were analyzed: (1) the probable maximum flood peak discharge of 1,100,000 cubic feet per second resulting from the probable maximum precipitation occurring over the total basin, a 12,650 square mile area above the plant site; (2) the flooding caused by failure of the Ashoken Dam concurrent with a major river basin flood (standard project flood) with a peak discharge of 705,000 cubic feet per second and a hurricane storm surge (standard project hurricane), and (3) the probable maximum hurricane concurrent with the high spring tide in the Hudson River. These three hypothetical floods are the most severe of several investigated, and each of the three results in a maximum water surface elevation of about 15 feet above mean sea level. We have reviewed the method of calculation and conditions assumed and find that they are conservative and acceptable. Both the U. S. Geological Survey and the Coastal Engineering Research Center provided consulting services with respect to our flooding evaluation. Their reports are attached as Appendix D and Appendix E, respectively.

3.5 Environmental Monitoring

The radioactivity levels in the vicinity of the Indian Point site have been measured by the applicant since 1958 to ascertain the impact of operation of Indian Point Unit 1 on the background levels of radioactivity. The environs of the Indian Point site have been studied intensively for many years by the Institute of Environmental Medicine at New York University Medical Center. These studies concerned both the exposure to man and to the flora and fauna indigenous to the Hudson River. All the results compiled to date indicate that radioactive effluents from Indian Point Unit 1 operation have produced barely quantifiable radiation exposure to the public and have had no detectable effect on the ecology of the area.

The operational environmental radiation monitoring program for Indian Point Unit 2 will be a continuation of the present program. The program includes direct measurements of gamma radiation and analyses to monitor fallout, air particulates, airborne iodines, water from various surface drinking water supplies, Hudson River water, water from lakes near the site, well water, lake aquatic vegetation, Hudson River vegetation, river bottom sediments, river aquatic biota, terrestrial vegetation, and soil. The report of the U. S. Department of the Interior is attached as Appendix G. This report incorporates the comments of the Federal Water Quality Administration, the Fish and Wildlife Service, and the Bureau of Outdoor Recreation. The report comments favorably on current activities being performed by or for the applicant in connection with determining the effects of both radiological and thermal discharges at the plant site. Recommendations for continued effort in the area of environmental monitoring and ecological studies are included in the report. This report has been forwarded to the applicant.

We conclude that the applicant's program will be adequate for monitoring the radiological effects of Indian Point Unit 2 operations on the environment and for assessing the effects of releases of radioactivity to the environment from operation of the plant on the health and safety of the public.

4.0 REACTOR DESIGN

4.1 General

The nuclear reactor for Indian Point Unit 2 was designed and manufactured by Westinghouse. The principal design features, materials of construction, and arrangement of various components of the Indian Point Unit 2 core are the same as those for the Rochester Gas and Electric Company's R. E. Ginna facility (Docket No. 50-244), which has been licensed for operation by the Commission and which has completed almost a full year of power operation. Further, the zircalloy clad fuel, burnable poison in the initial core loading, a chemical neutron absorber, and part-length control rods to shape axial power distribution are used in substantially the same manner in both the Ginna and the Indian Point Unit 2 reactors. On the basis of our previous review of all of these features for the Ginna reactor, we conclude that these same features are acceptable for Indian Point Unit 2.

4.2 Nuclear Design

The Indian Point Unit 2 reactor core differs principally from the Ginna and Connecticut Yankee (Docket No. 50-213) reactor cores in that the Indian Point Unit 2 reactor core is somewhat larger. The Indian Point Unit 2 core is about 23% greater in cross sectional area and 20% longer than the Connecticut Yankee core and about 89% greater in cross sectional area and the same length as the Ginna core. Because this larger core could be subject to power oscillations or power tilts, we reviewed the nuclear design and power distribution detection and control systems for the Indian Point Unit 2 reactor core in detail.

During plant operation, changes in the core power level or the control rod configuration can cause time-dependent variations in the local power distribution as a result of variations in the concentration of fission products and their radioactive decay products. The most significant fission product-decay product chain with regard to core behavior is the decay of iodine-135 to xenon-135 since the latter is a strong absorber of thermal neutrons. The local oscillations in the neutron flux and in the power level can occur even though the average power level of the core is maintained constant, and the magnitude of the oscillations may decrease, remain constant, or increase with time.

The spatial stability of the xenon distribution and resultant core power peaking abnormalities for the Indian Point Unit 2 core have been investigated by Westinghouse with the conclusion that the core is stable against various types of xenon induced spatial oscillations in the X,Y horizontal plane. This conclusion is supported by analysis and by experiments performed in the Connecticut Yankee reactor. An initial test program for Indian Point Unit 2 will be performed to verify this stability. If this initial test program does not demonstrate stability, the applicant has agreed to operate with partially inserted control rods, or to add fixed or burnable poison shims sufficient to assure stability through reduction of the moderator temperature coefficient, or to operate at reduced power levels. Because of the test program that will be performed and the operating limitations that will be imposed if required, we conclude that the reactor will be stable with respect to potential power oscillations in the X,Y horizontal plane.

The analysis made by Westinghouse indicates that the reactor may be subject to divergent xenon oscillations in the axial direction, resulting in an axial power distribution imbalance or tilt. In view of this, it is assumed that the axial power tilts can occur, and provision is made to detect and control differences in the fraction of the total power generated in the upper and lower halves of the core. Data correlations have been made at the Connecticut Yankee reactor and at the Ginna reactor to relate the readings obtained from the split out-of-core detectors to axial power tilts. Additional correlations will be established during the Indian Point Unit 2 startup tests. Part-length control rods are provided to prevent unacceptable axial power tilts and to control potentially divergent axial xenon spatial oscillations. Analytical studies and experience with the Ginna reactor, provide assurance that any axial oscillations can be controlled such that the power distribution will be maintained within design limits. In addition, automatic protective action is provided to avoid exceeding design power peaking factors at full power in the event of control system malfunctions. To accomplish this, the overtemperature ΔT and overpower ΔT trip set points are automatically reduced in proportion to the axial power tilt as measured by the split out-of-core neutron detectors. We conclude that the system of detection instrumentation, control with part length rods, and automatic protection for potential axial power tilts is acceptable.

Even in the absence of xenon induced instability, power tilts or imbalances can occur in the horizontal or axial planes as a result of control rod misalignment. Analyses for Indian Point Unit 2 and experiments in the Connecticut Yankee reactor have shown that these power tilts can be detected by (1) the split out-of-core neutron detectors, (2) the core exit thermocouples, or (3) the movable in-core neutron detectors. All of these detectors are required to be operable by the Technical Specifications. In addition detection will ordinarily be readily accomplished by the fixed in-core neutron instrumentation.

The power distribution in the Indian Point Unit 2 core is expected to be stable or only slowly varying within known limits and adequate core instrumentation will always be available to detect, monitor, and diagnose any significant power maldistributions.

We conclude that the Indian Point Unit 2 reactor core nuclear design and instrumentation is acceptable.

4.2 Thermal-Hydraulic Design

We have evaluated the adequacy of the core thermal and hydraulic design, both for steady-state plant operation and for anticipated plant transients. The design criteria selected by the applicant to prevent fuel damage are: (1) the departure from nucleate boiling (DNB) ratio (determined using the Westinghouse W-3 correlation) shall not be less than 1.3 during normal plant operation or as a result of anticipated transients; and (2) no fuel melting shall occur during either normal operation or anticipated transient conditions. The anticipated plant transients that result in the most severe core thermal transients are loss of coolant flow, excessive load increase, and a loss of external electrical load. The applicant's analyses show that the DNB ratio will be greater than 1.3 for each of these plant transients when operating at the license power level of 2758 MWt. The lowest DNB ratio calculated as a result of any of the plant transients, was for the case of simultaneous loss of electrical power to the four reactor coolant pumps. This transient results in a DNB ratio of 1.42. In addition, no fuel melting is predicted to occur for steady-state operation or as a result of anticipated transients.

As stated above the Indian Point Unit 2 reactor core is designed to undergo anticipated plant transients with a minimum DNB ratio greater than 1.3. On this basis, clad temperature should not be significantly affected by a transient and no fuel failure should occur for the range of fuel element burnup planned for the Indian Point Unit 2 core. As part of a continuing experimental effort to demonstrate satisfactory performance of fuel at high burnup and high power density, Westinghouse is continuing a fuel irradiation program at conditions significantly in excess of current PWR design limits, and will establish power burnup limits for the fuel. These irradiation programs are being conducted at both the Saxton and Zorita reactors. Sustained operation of selected fuel rods at peak design power levels in the Zorita reactor will increase assurance that the fuel has adequate margins to accommodate transient overpower operation.

Based on our evaluation of the results of these analyses, and on our review of the design limits and the operating experience of similar reactors, we conclude that the reactor core thermal and hydraulic design is acceptable for operation at the rated power of 2758 MWt.

5.0 REACTOR COOLANT SYSTEM

5.1 General

The reactor primary coolant system, including all vessels, pumps, and piping is designed for a pressure of 2485 psig and a temperature of 650°F. The system has been designed to withstand, within the stress limits of the codes used in the design, the normal loads of mechanical, hydraulic, and thermal origin, plus those due to anticipated transients and the operating basis earthquake.

5.2 Primary System Components

The reactor internals are designed to withstand the normal design loads of mechanical, hydraulic, and thermal origin, including those resulting from anticipated plant transients and the operating basis earthquake, within the stress limit criteria of Article 4 of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section III. Although the Indian Point Unit 2 reactor internals are not designed to withstand simultaneously the loads resulting from loss-of-coolant accident blowdown and seismic events, the applicant has submitted a summary of an analytical study of the behavior of the reactor internals under simultaneous blowdown and seismic loadings (WCAP-7332-L). The results of this study indicate that for the combined blowdown and design basis earthquake loadings the resulting deflections are within the loss-of-function limits except for the control rod immediately adjacent to the coolant line that was assumed to fail. On the basis that the core reactor internals remain functional and that adequate shut down margin can be achieved by control rod insertion, we conclude that the stress and deflection limits for the combined blowdown and design basis earthquake loadings provide an adequate margin of safety.

The primary system side of the steam generators, the pressurizer, and the main coolant pump casings, have been designed to the requirements of Section III of the ASME Boiler and Pressure Vessel Code, 1965 Edition - Summer 1969 Addenda, as Class A vessels. For other Class I pumps, valves, and heat exchangers the inspection program required independent review of (1) the physical and chemical test data for pressure boundary materials, (2) radiographs of valve bodies, valve bonnets and pump casings, and (3) dye-penetrant examinations of heat exchanger tubes and welds. These requirements resulted in fabrication and inspection programs that contain the essential elements of the recently proposed ASME Codes for Nuclear Pumps and Valves. We find the design codes and inspection requirements acceptable.

We have reviewed the information submitted by the applicant with respect to operating limitations on heatup and cooldown of the primary system imposed by the fracture toughness properties of the materials of the Indian Point Unit 2 reactor vessel. Our evaluation was based on a proposed redraft of section NB-2300 Special Materials Testing (Section III ASME Boiler and Pressure

Vessel Code) dated July 28, 1970, which reflects the material testing requirements in a form consistent with the AEC Fracture Toughness Criteria. As a consequence of our evaluation the applicant has agreed to the heatup and cooldown limitation as presented in Section 3.1-B of the Technical Specifications which represents a modification of his initial submittal. On the basis that these limits reflect a very conservative method of defining pressure vessel fracture toughness, we conclude that they are acceptable.

5.3 Coolant Piping

The reactor coolant piping has been designed in accordance with the requirements of the American National Standards Institute (ANSI) B31.1 Code for Power Piping, 1955 Edition, including the requirements of Nuclear Code Cases N-7 and N-10. All welding procedures and operators were qualified to the requirements of Section IX of the ASME Boiler and Pressure Vessel Code. Additional inspection requirements for the reactor coolant piping during fabrication included ultrasonic and dye-penetrant inspection of all pipe welds. Non-destructive examination of valves included radiographic examination of the valve castings and ultrasonic inspection of all forged components. Dye-penetrant surface examination was also performed. With this program, the inspection of the Indian Point Unit 2 reactor coolant piping substantially meets the requirements of Class 1 systems under the ANSI B31.7 Code for Nuclear Power Piping adopted in 1969. On this basis we have concluded that the design and inspection program for this system is acceptable.

The original seismic design analysis for the Indian Point Unit 2 reactor coolant system utilized only static methods of analysis. Recently, at our request, the applicant completed a rigorous dynamic analysis of this system utilizing both modal-response spectra and model time-history methods of analyses. As with the reactor internals, the combined loading of a concurrent loss-of-coolant accident blowdown and design basis earthquake was not considered in the design of the Indian Point Unit 2 reactor coolant system. However, the applicant recently completed an analysis of the response of the reactor coolant system to be installed in Indian Point Unit 3 for these combined loads. Since the Indian Point Unit 3 and the Indian Point Unit 2 reactor coolant systems are identical, the applicant has used the results of the analysis for Indian Point Unit 3 in conjunction with the material properties for the Indian Point Unit 2 piping, as determined from tests, to determine that the combined seismic and accident loads can be tolerated by the Indian Point Unit 2 reactor coolant system within acceptable stress limits.

Based on our review of the design limits and analytical procedures employed, we find that the design of the Indian Point Unit 2 reactor coolant system is acceptable.

5.4 Other Class I* (Seismic) Piping

At our request the applicant performed additional seismic analysis on other Class I piping. The adequacy of the seismic design of the feedwater lines, pressurizer surge line, and a typical steam line has been confirmed by a dynamic analysis utilizing the modal-response-spectra method. The adequacy of the seismic design of other Class I (Seismic) piping in the plant was determined by performing a dynamic analysis on selected "worst case" systems. Several systems that are the most vulnerable to dynamic excitation because of system flexibility or location in the supporting structure were analyzed and the resulting stresses compared with the stresses determined by the original static analyses. The applicant has concluded that the conservatism of the original static analysis provided adequate margins to accommodate the previously undetermined dynamic effects.

Based on our review of the original static methods employed and the confirmatory evidence obtained from the recent dynamic analyses of the most vulnerable systems, we have concluded that the design of the Class I (Seismic) piping systems in Indian Point Unit 2 is acceptable.

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*See Section 6.1 for definition of Class I structures, systems, and components.

5.5 Inservice Inspection

An inservice inspection program for the reactor coolant system is included in the Technical Specifications. This program follows Section XI of the ASME Code, Rules for Inservice Inspection of the Reactor Coolant System, as closely as practical. The design of the primary system including the capability to remove insulation at selected areas provides an acceptable degree of access for inspection purposes. The applicant also intends to conduct periodic inservice inspections of the primary pump motor flywheels.

The applicant will review the inservice inspection program with us after five years of reactor operation. It may then be modified based on experience gained during these five years. At that time, we will also require the applicant to perform such inspections of components outside the reactor coolant pressure boundary as deemed necessary to provide continuing assurance of structural integrity.

5.6 Missile Protection

We have reviewed the applicant's primary system layout within the containment in terms of the protection afforded the containment liner and Class I (seismic) systems inside the containment from missiles that might be generated as a result of a primary system failure. We have concluded that adequate protection from potential missiles is provided by the system arrangement and surrounding thick circumferential concrete walls and the concrete floors.

The primary pump motor flywheels installed in Indian Point Unit 2 are the same as those in use in other plants. The flywheels are the standard Westinghouse design, fabricated of A 533B steel. On the basis of the use of high grade material, extensive quality control measures, special manufacturing procedures and preservice and inservice surveillance requirements, we have concluded that assurance has been provided that the integrity of the flywheels will be maintained.

5.7 Leak Detection

The reactor coolant pressure boundary leak detection systems for this plant are similar to those we have reviewed and found acceptable for other plants using a Westinghouse nuclear steam supply system. The systems are based upon air particulate monitoring, radiogas monitoring, humidity detection, and containment sump level monitoring. These systems provide an array of instrumentation that is sensitive, redundant, and diverse and that has adequate alarm features. The sensitivity of these systems is consistent with their primary purpose of detecting any leak in the primary system pressure boundary which could be indicative of incipient failure. The Technical Specifications require that two reactor coolant leak detection systems of different principles shall be in operation when the reactor is operated at power. We conclude that the leak detection systems for Indian Point Unit 2 are acceptable.

5.8 Fuel Failure Detection

The fuel element failure detection system will measure delayed neutron activity in one hot leg of the reactor coolant system. The monitor is connected in series with a delay coil to allow a decay time for N^{16} gamma activity (half life of 7.1 seconds) of about 60 seconds before the coolant reaches the detector. This delay reduces gamma ray background and facilitates detector sensitivity. An alarm signal is provided for the channel. We conclude that this system which is inherently faster in response than previous systems reviewed for other reactors is acceptable.

5.9 Vibration Monitoring and Loose Parts Detection

The major core and core support components have been analyzed to provide assurance that they are not vulnerable to vibratory excitation. Vibration analyses for the core support barrel considered inlet flow impingement and turbulent flow. Natural frequency calculations were made to assure that there would be no deleterious response to known excitations such as pump blade passing and driven frequencies. Fuel bundle response to anticipated driving forces has been calculated and determined by tests in the Westinghouse Reactor Evaluation Center.

The vibration monitoring system to be used for the preoperational test program on Indian Point Unit 2 will consist of mechanical gauges to measure gross relative motion between the thermal shield and core barrel, strain gauges on selected guide tubes, and accelerometers on the upper core plate. We have concluded that the vibration design analyses and the preoperational test program are acceptable.

In the course of our review of the Indian Point Unit 2 application, it has been noted that techniques for the analysis of neutron noise spectra and accelerometer measurements on the lower heads of primary system vessels might be developed to provide a useful method for inservice monitoring of reactor coolant systems to detect changes in the vibration of reactor components or the presence of loose parts. The applicant has stated that neutron noise measurements will be made periodically and analyzed to provide developmental information concerning the possible usefullness of this technique in ascertaining changes in core vibration or other displacements. On a similar basis, accelerometers will be installed on the pressure vessel and steam generators to ascertain the practicality of their use to detect the presence of loose parts.

5.10 Conclusion

Based on our review of (1) the codes and standards used for design, (2) the fabrication and inspection procedures, (3) the inservice inspection program, (4) the provisions for missile protection and leak detection, (5) the provision for fuel failure detection, and (6) the provisions for preoperational vibration testing and the developmental effort for inservice monitoring to detect vibrations and loose parts, we have concluded that the design and inspection procedures for the reactor coolant system for the Indian Point Unit 2 are acceptable.

6.0 CONTAINMENT AND CLASS I (SEISMIC) STRUCTURES

6.1 General Structural Design

The applicant has categorized as Class I (seismic) those structures (e.g., containment structure and primary auxiliary building), and those systems and components (e.g., reactor vessel and internals, emergency core cooling system), whose failure could cause a significant release of radioactivity or that are vital to the safe shutdown of the facility and the removal of decay heat. We have reviewed the applicant's classification of structures, systems, and components and conclude that they have been classified appropriately.

The Class I (seismic) structures at Indian Point Unit 2 are the containment structure, the primary auxiliary building, the control room building, the fuel storage pool, the diesel generator building, and the intake structure and service water screenwell. The major portion of the primary auxiliary building, the fuel storage pool, and the intake structure are of reinforced concrete construction. The control room building, the diesel generator building, the fuel storage building and the non-Class I portions of the primary auxiliary building are constructed of steel framing with composite metal panel siding.

The environmental conditions that were considered in the structural design include the operating basis earthquake (OBE), the design basis earthquake (DBE), the flooding and wind due to the probable maximum hurricane, and the flooding due to the probable maximum flood. We have concluded that these conditions were used for the design in an acceptable manner.

6.2 Structural Design and Analysis

The Indian Point Unit 2 primary containment has a free volume of 2.6 x 10^6 cubic feet and a design pressure of 47 psig. The containment structure is a right cylinder (thickness 4.5 ft) with hemispherical dome (thickness 3.5 ft) mounted on a flat (thickness 9 ft) base mat. The reinforced concrete is lined with 1/4 inch minimum thickness welded ASTM A442 grade 60 firebox quality carbon steel plate. The reinforcing bars conform to ASTM A432 specifications. The reinforcing in the cylinder wall is placed in horizontal and vertical directions with added diagonal tangential reinforcing for earthquake resistance. The reinforcing bars conform to ASTM A432 specifications. Cadweld splices are used in 14S and 18S bars.

We have evaluated the pressure transients that might occur in the containment in the event of a loss-of-coolant accident assuming various sizes of primary coolant system breaks. For the range of postulated break sizes up to and including the double-ended severance of the largest reactor coolant pipe, the largest calculated peak containment pressure is 40 psig. The design pressure of the containment exceeds the calculated peak pressure by more than 10% and is acceptable.

The containment is designed to remain within the elastic range for the 0.10g OBE concurrent with the accident and other applicable loads. It is also designed to withstand the 0.15g DBE concurrent with the accident without loss of function.

We and our seismic design consultant, Nathan M. Newmark, are in agreement with the loading combinations and allowable stresses used by the applicant. Stress and strain limits conform to the requirements of ACI 318-63, Part IV-B. The ACI load factors have been replaced by factors suitable for concrete containment structures.

Based on our review of the design of the containment structure and its capability to withstand the predicted pressures from potential accidents, we conclude that the structural design aspects of the containment are acceptable.

In evaluating the capability of the Class I (seismic) structures, systems, and components, to withstand the dynamic loads due to seismic events, our seismic design consultant, Nathan M. Newmark Consultant Engineering Services, considered the geology and nature of the bedrock, design loads and load combinations, the seismic design parameters, and methods of analysis. On the basis of our review and that of our seismic design consultant, we conclude that the Class I (seismic) structures, systems, and components of Indian Point Unit 2 are designed to accommodate all applicable loads and are acceptable. The report of our seismic design consultant is attached as Appendix G.

During our review we noted a limited number of cases where failure of non-Class I (seismic) structures could potentially endanger Class I (seismic) structures and equipment. These included the Indian Point Unit 1 superheater stack and superheater building, the turbine building, and the fuel storage building. In response to our concern, the applicant performed analyses of these structures using a multi-degree of freedom modal dynamic analysis method, to determine the modifications needed to assure that gross structural collapse of these structures would not occur in the event of a DBE. As a result of these analyses, additional seismic reinforcement is being provided for both the superheater building and the turbine building and the Indian Point Unit 1 superheater stack is to be reduced in height by 80 feet. The truncation of the stack is to be accomplished at a convenient time in the next three years and prior to operation of Indian Point Unit 3. We and our seismic design consultant have reviewed the material submitted by the applicant and conclude that the dynamic analyses performed, and the design modifications proposed, are acceptable.

We have reviewed the as-built wind resistance of Class I structures at the Indian Point Unit 2 facility. Analysis indicates that both the containment and reinforced concrete portions of the primary auxiliary building and intake structure can sustain winds in the range of 300 miles per hour. The control building and diesel generator building which are constructed of structural steel with composite metal panel siding, are estimated by the applicant to be capable of sustaining wind loads of up to 160 miles per hour.

Some natural protection from high winds is afforded the control room building and diesel generator building since they are protected by the turbine building to the west, the Indian Point Unit 1 turbine building, superheater building, and containment to the south, the rising hillside to the east, and the containment and rising hillside to the north.

The wind resistance of the Indian Point Unit 1 superheater stack was also considered with respect to preserving the integrity of Indian Point Unit 2. A reduction in stack height of 80 feet coupled with the additional seismic reinforcement of the superheater building (see discussion above) will enable the stack to resist winds with speeds greater than 300 miles per hour.

On the basis of the very low probability for wind speeds greater than 100 miles per hour at the Indian Point site and on the basis of the wind resistance of the Class I (seismic) structures as discussed above, we conclude that Indian Point Unit 2 is adequately protected against high winds.

6.3 Testing and Surveillance

Strength and leakage tests of the containment building will be performed after construction is completed. A 115% overpressure strength test at 54 psig will be conducted and leakage tests will be made at pressures up to 47 psig. As noted in Section 7.3 of this evaluation, pressurized test channels are provided at all liner seams for long-term surveillance. No permanent instrumentation is being installed on the containment for strength testing, although examinations will be made for cracking and distortion during the pressure test. Periodic leakage rate tests will be performed on the containment and its penetrations.

We have concluded that the provisions for testing and surveillance of the containment are acceptable. Test and surveillance requirements are included in the Technical Specifications.

6.4 Missile Protection

The possibility exists that missiles might be generated in the unlikely event of a failure of the turbine generator. Although the design criteria for Indian Point Unit 2 did not include consideration of protection against missiles resulting from turbine failures, at our request the applicant has assessed the protection available against missiles that might result from a turbine failure at the maximum overspeed condition (133% of rated normal speed). Specific provisions have been added to limit the off-site consequences that could result from a missile failure, and to provide for safe shut down of the unit. These include an alternative cooling water supply for the charging pumps and added missile protection for a potentially vulnerable portion of the auxiliary steam generator feedwater lines. In addition, a second completely independent turbine speed control system has been provided to reduce the probability of a runaway speed condition that might result in a turbine failure. This system is designed to the requirements of the Institute of Electrical and Electronic Engineers (IEEE) Criteria No. 279 for protection systems. The Technical Specifications require periodic testing of the overspeed devices to assure operability. We conclude that the applicant has made appropriate provisions to reduce the probability of a destructive turbine missile from being generated and affecting Class I (seismic) items.

The Indian Point Unit 2 reactor vessel cavity is designed to protect the containment against missiles that might be produced by postulated failure of the reactor vessel. Failure of the reactor vessel would result in fluid jet-reaction forces in the cavity wall adjacent to the vessel split or crack as well as stress in the cavity wall from a rise in cavity pressure, both of which would result from coolant blowdown. Also reaction forces in the cavity wall and floor might be produced by the impact of missiles generated by pressure vessel failure. By the use of extensive steel reinforcing, the concrete cavity has been designed to resist both fluid jet and missile impact forces that could result from pressure vessel failure by either longitudinal splitting or various modes of circumferential cracking. The cavity is also designed to sustain a fluid pressure rise to 1000 pounds per square inch. We have reviewed the applicant's analysis and conclude that the cavity as designed provides adequate protection for the containment liner against missiles that might result from a postulated pressure vessel failure.

7.0 ENGINEERED SAFETY FEATURES

7.1 Emergency Core Cooling System

The principal equipment of the emergency core cooling system consists of (1) three 50% capacity high pressure safety injection pumps, (2) two 100% capacity residual heat removal pumps for low pressure injection and external recirculation, (3) two 100% capacity recirculation pumps for recirculation internal to the containment, (4) one 100% capacity boron injection tank, and (5) four 33-1/3% capacity accumulators. This system provides redundant capability to inject borated cooling water rapidly into the core in the event of a loss-of-coolant accident and to maintain coolant above the level of the core for an indefinite period following the accident.

The applicant's evaluation of the performance of these systems is based on detailed analyses of (1) the hydraulic behavior of the primary coolant system during and subsequent to a loss-of-coolant accident, and (2) the thermal response of the core during the same period. The analytical methods used to predict the hydraulic behavior of the primary coolant system during a loss-of-coolant accident have been improved significantly during the construction period for Indian Point Unit 2. The original analysis presented in Volume 4 of the FFDSAR was performed with the FLASH-1 hydraulics computer program. This program is limited to a three-node representation of the coolant system. Subsequent to the analysis performed with FLASH-1, Westinghouse developed a new multi-node hydraulics program called SATAN. Using SATAN the coolant system can be represented with as many as 96 nodes. The SATAN calculations provide considerable detail in the system analysis and increased insight into system performance.

At our request, the applicant reevaluated the performance of the emergency core cooling system during a loss-of-coolant accident using the SATAN multi-node hydraulics code. The applicant's analysis is based on the license application power rating of 2758 MWt. For the case of an accident initiated by a double-ended break in the cold leg primary coolant piping, a maximum fuel element clad temperature of 2015°F was predicted. The applicant's investigation of the emergency core cooling system performance for a range of break sizes and locations indicates that the resultant peak temperatures for any other break will be less than those predicted for the double-ended cold leg break. On the basis of our review of the analytical techniques used in this analysis and our experience with similar analytical techniques, we conclude that there is reasonable assurance that the results obtained with these techniques provide a conservative estimate of the performance of the system in the event of a loss-of-coolant accident at Indian Point Unit 2.

We conclude that the emergency core cooling system will (1) limit the peak clad temperature to well below the clad melting temperature, (2) limit the fuel clad water reaction to less than 1% of the total clad mass, (3) terminate the clad temperature transient before the geometry necessary for cooling is lost and before the clad is so embrittled as to fail upon quenching and (4) reduce the core temperature and then maintain core and coolant temperature levels in a subcooled condition until accident recovery operations can be accomplished.

In summary, we conclude that the emergency core cooling system is acceptable and will provide adequate protection for any loss-of-coolant accident.

The emergency core cooling system design as presently installed at Indian Point Unit 2 was reviewed by the Division of Reactor Licensing during 1967, subsequent to the issuance of the construction permit on October 14, 1966. This system represented a complete redesign, a considerable increase in flow capability, and enhanced performance when compared to the system reviewed for the construction permit. On the basis that the very significantly improved performance of the redesigned emergency core cooling system provides additional assurance for limiting clad temperatures and maintaining a coolable core we concurred with the applicant's decision to remove the reactor pit crucible from the facility design.

7.2 Containment Spray and Cooling Systems

Two independent heat removal systems are provided to control the containment pressure and temperature following a loss-of-coolant accident. Each system, acting alone at its rated capacity, will prevent over-pressurization of the containment structure. The two systems are the containment spray system and the fan cooling system. The design of each is substantially the same as the design of systems provided at the Ginna plant and other licensed plants.

The containment spray system consists of two 50% capacity spray pumps and is sized to limit the containment post-accident pressure to below design pressure. Sodium hydroxide and boric acid are used as additives to the spray solution to remove radioactive iodine which might be present in the containment after an accident. We have reviewed the use of these chemical spray additives in terms of their iodine removal capabilities, and in addition have evaluated the chemical compatibility of the spray solution with other reactor components. As a result of our review, we conclude that the spray system is adequately sized to cool the containment, that the alkaline spray solution will reduce the iodine concentration in the containment atmosphere, and that corrosion of other materials used in the containment does not introduce a safety problem.

The containment fan cooling system provides complete redundancy to the containment spray system for heat removal from the containment atmosphere during post-accident conditions. Five 20% capacity fan coolers are provided. Since the fan coolers are located within containment, they must be capable of operating in the post-accident environment. Westinghouse has conducted an environmental test program to demonstrate this capability. Our evaluation of these tests, including the heat removal capability of the heat exchangers, and environmental and radiation testing of the fan cooler motors, valve motor operators and electric cabling indicates that these components will function satisfactorily in the accident environment. An iodine-impregnated charcoal filter system has been included with the fan cooler system to remove organic iodine from the post loss-of-coolant containment atmosphere. The charcoal beds are preceded by demisters and high efficiency particulate air (HEPA) filters.

We have evaluated the inorganic and organic iodine removal capability of the charcoal beds on the basis of tests with steam - air mixtures at 100% relative humidity following prolonged flooding of the bed. We conclude that inorganic and organic iodine removal efficiencies of 90% and 10% per pass, respectively, are conservative values that are justified by the available information.

In summary, we have reviewed the containment spray and fan cooling systems in terms of (1) capability to control the containment temperature, (2) capability to remove inorganic and organic iodine,

(3) system and component redundancy, and (4) capability to function in the post-accident containment environment. We conclude that there is reasonable assurance that these systems will operate as proposed subsequent to a loss-of-coolant accident.

7.3 Containment Isolation Systems

In addition to the usual capability of isolating all lines leading to and from the containment, the Indian Point Unit 2 containment is provided with additional systems to minimize the potential leakage of fission products subsequent to an accident. A containment penetration and weld-channel pressurization system provides for continuous pressurization of zones enclosing containment penetrations and the welds in the containment liner. The system continuously maintains an overpressure of clean, dry air that is in excess of the containment design pressure. Pressurized zones include each piping penetration, each electrical penetration, double gasketed spaces on the personnel and equipment hatches, and the channels over weld seams of the containment liner. The air pressure is maintained by the instrument air compressors with backup from the plant air compressors and from a standby source of nitrogen cylinders. Pressure indication and alarm instrumentation is provided locally and in the control room to assure that loss of pressure will be detected and corrected.

In addition, an isolation seal water system has been provided to assure containment isolation by (1) injecting seal water between the seats and stem packing of the globe and double disc isolation valves used on larger lines, and (2) injecting seal water directly into the line between the closed diaphragm valves used in the smaller lines penetrating containment. Seal water injection is provided for all lines connected to the reactor coolant system and for lines that may be exposed to the containment atmosphere subsequent to an accident. Although the use of the seal water system following a loss-of-coolant accident provides an additional means of reducing leakage, we have not considered the effect of this system in determining the offsite radiological consequences.

We have concluded that the capability provided for isolating the containment is acceptable.

7.4 Post-Accident Hydrogen Control System

In the event of a loss-of-coolant accident, radiation from the core and from escaped fission products will dissociate some of the cooling water into gaseous hydrogen and oxygen. Continued evolution of hydrogen would increase the concentration in the containment to a point where ignition could occur and thus provide an additional energy source.

Redundant flame recombiner units are installed within the Indian Point Unit 2 containment. Each unit has the design capability to prevent the ambient containment hydrogen concentration from exceeding two percent by volume. The units are designed to function, following the loss-of-coolant accident in a containment pressure environment of 1 to 5 psig. Each recombiner system consists of (1) a flame recombiner unit located within containment, (2) a control panel located outside of containment, and (3) a hydrogen gas stand located outside of containment. On the basis of (1) our detailed review of the design of the system and its controls, (2) satisfactory performance testing of the device, and (3) satisfactory environmental testing of those portions of the recombiner system installed within the containment, we conclude that there is reasonable assurance that the recombiner system will perform its intended post-accident function.

In addition, the applicant will provide the capability for purging the containment atmosphere through appropriate filters as an alternate backup means of hydrogen control. The containment penetrations to be used for this system are installed. The design and installation of the equipment required will be performed during the first two years of operation at power.

8.0 INSTRUMENTATION, CONTROL, AND POWER SYSTEMS

8.1 Reactor Protection and Control System

The reactor protection system instrumentation for Indian Point Unit 2 is the same as that installed at the Ginna plant. The adequacy of the protection system instrumentation was evaluated by comparison with the Commission's proposed general design criteria published on July 11, 1967, and the proposed IEEE criteria for nuclear power plant protection system (IEEE-279 Code), dated August 28, 1968. The basic design has been reviewed extensively in the past and we conclude that the design for Indian Point 2 is acceptable.

During our review we considered the adequacy of reactor protection for operation with less than four coolant loops in service. When operating with one of the primary loops out of service the reactor is normally automatically limited to 60% of full power. However by manual adjustment of several protection system set points in a manner consistent with the Technical Specifications adequate reactor protection can be provided for operation up to 75% of full power.

We have reviewed the applicant's analysis of the seismic response of the protection system instrumentation and associated electrical equipment and find that adequate testing has been performed on the nuclear instrumentation, switch gear, and process system instrumentation.

In connection with our review of potential common mode failures we have recently considered the need for means of preventing common failure modes from negating scram action and of possible design features to make tolerable the consequences of failure to scram during anticipated transients. The applicant has been responsive to our request for information and has provided the results of analyses which indicate that the consequences of such transients are tolerable for the existing Indian Point Unit 2 design at a power level of 2758 MWt. Although additional study is required of this general question, we conclude that it is acceptable for the Indian Point Unit 2 reactor to operate at a power level of 2758 MWt while final resolution of this matter is made on a reasonable time scale.

8.2 Initiation and Control of Engineered Safety Features

The instrumentation for initiation and control of engineered safety features for the Indian Point Unit 2 is the same as that installed at the Ginna plant. This basic design has been reviewed extensively in the past and we consider it to be acceptable.

We have reviewed the capability for testing engineered safety feature circuits during reactor operation. Resistance tests will be used for routine determinations of the operability of the master and slave relay coils. The circuits upstream of these relays can be partially tested during operation. During plant shutdown, circuits can be tested completely by coincident tripping of instrument channels and a consequent operation of the master and slave relays in the entire downstream initiating system. We have concluded that this testing capability is acceptable for Indian Point Unit 2.

8.3 Off-Site Power

Two 138 kilovolt (kV) lines connect the Buchanan switchyard to the Millwood switching station, which in turn is connected to the Consolidated Edison grid and the Niagara Mohawk and Connecticut Light and Power systems. Two additional 138 kV lines, using a separate route from the first two lines, connect the switchyard to the Orange and Rockland tie.

The applicant stated that an analysis of the transmission system has indicated that the system is stable for the loss of any generating unit including Indian Point Unit 2.

A single 138 kV line connects the Buchanan switchyard to Indian Point Unit 2. In addition, three 13 kV lines connect the switchyard to Indian Point Unit 1. Three 138/13 kV transformers in the switchyard feed these three 13 kV lines. While the 138 kV system is the normal supply for the auxiliary load associated with plant engineered safety features, one of the three Indian Point Unit 1 13 kV lines is available to provide power via automatic switching to Indian Point Unit 2 through a 13/6.9 kV transformer. By switching circuit breakers in Indian Point Unit 1, the other two 13 kV lines can also be made available to provide power to Indian Point Unit 2. As the 13/6.9 kV supply is not capable of carrying the total plant auxiliary load for Indian Point Unit 2, the main coolant pumps and the circulating water pumps must be tripped off before the supplies are switched.

We conclude that the off-site power supply provides an adequate source of power for the engineered safety features and safe shutdown loads.

8.4 Onsite Power

Onsite power is supplied by three independent diesel generator sets connected in a separate bus configuration such that there is no automatic closure of tie breakers between the three buses to which the generators are connected. The redundant engineered safety feature (ESF) loads are arranged on the three separate buses such that failure of a single bus will not prevent the required ESF performance under accident conditions. The design engineered safety feature and safe shutdown loads per diesel generator are 1813, 2210, and 2353 HP for the first one-half hour following a loss-of-coolant accident. The loads are then changed to 2438, 2235, and 2043 HP for the recirculation phase of the emergency core cooling system operation. On the basis of our evaluation, we have determined that the appropriate diesel generator ratings are 2200 HP continuous, and 2460 HP for 2,000 hours. We note that some of the estimated emergency loads are above the continuous rating of the machines, but below the 2,000 hour ratings. We consider that this margin is acceptable for Indian Point Unit 2.

Each diesel generator is started automatically upon initiation of emergency core cooling system operation or upon under-voltage on its corresponding 480-volt emergency bus. The generators are housed in a separate Class 1 (seismic) structure. On-site diesel fuel storage capacity provides a minimum of seven days operation at the required safety feature loads. These design and operating features are acceptable for Indian Point Unit 2.

Our review of the ac auxiliary power system has disclosed that there is adequate capacity and an adequate degree of physical and electrical separation of redundant features. The 125 volt dc system consists of two individually housed batteries. The dc system is divided into two buses with a battery and battery charger for each bus. Each of the two station batteries has been sized to carry its expected loads for a period of two hours following a plant trip at a loss of all ac power.

We conclude that the onsite emergency power system is acceptable.

8.5 Cable Installation

We have reviewed the applicant's cable installation relative to the preservation of the independence of redundant channels by means of separation, and relative to the prevention of cable fires through proper cable rating and tray loading. This has been performed by reviewing the cable installation criteria and method of layout design and by field inspection of electrical cable installation during construction.

A single electrical tunnel carries the electrical cables from the electrical penetration area of the containment to the control building. This tunnel carries all of the electrical cables except the power cables for the reactor coolant pumps, the pressurizer heater cables, and the control rod power cables. The cables in the tunnel are arrayed on either side of a three-foot aisle in trays or ladders. Separation is provided for in the form of distance, metal separators, or transite barriers. The electrical tunnel does not contain any spliced cable connections. Therefore, the probability of a fire is reduced. Further, a fire detection system and an automatically operated water spray system are provided in the tunnel. Tunnel cooling is provided for by redundant cooling fans. On the basis of adequate separation within the tunnel, a minimum number of heat producing cables and features, redundant cooling systems, and fire detection and spray systems we conclude that the single electrical tunnel is acceptable.

Sixty electrical penetrations are provided in a single electrical penetration area to provide for entry of signal, control, and power cables into the containment. The penetrations are located on three-foot centers, both horizontally and vertically, and are of the hermetically sealed type. As a result of our review, fire barriers in the form of transite sheets were added to separate the power cable penetration from the instrument and control cable penetrations. In addition, as a result of our review certain modifications were made to the cabling in the penetration area, including shortening of cable runs and elimination of cable loops. The segregation of power cables and the shortening of the cable runs reduces the probability of failure by fire and on this basis, we consider the single electrical penetration area acceptable for Indian Point Unit 2.

The applicant has performed a design audit to verify the separation of redundant engineered safety feature power and control electrical cabling. A design review of instrument cabling was also performed on a sample basis.

On the basis of our review of cable installation at Indian Point Unit 2, we conclude that the resulting cable layout, as installed, is acceptable.

8.6 Environmental Testing

Westinghouse has conducted an environmental test program for the instrumentation and controls that are located inside containment and that must function in the environment following a loss-of-coolant accident. We have reviewed the results of this testing program and conclude that the essential instrumentation and controls will function properly in the accident environment.

9.0 RADIOACTIVE WASTE CONTROL

Liquid and gaseous waste handling facilities are designed to process waste fluids generated by the plant so that discharge of liquid and gaseous effluents to the environment will be minimized. Liquid waste is processed both by direct removal of radioactive material with ion exchange resins and by evaporative separation. Using these methods the volume of radioactive waste will be greatly concentrated and the purified liquid streams will either be reused or discharged. Small quantities of radioactive liquid waste will be released routinely to the condenser circulating water discharge canal common to all three units where the waste will be diluted and discharged to the Hudson River.

The limits on routine radwaste releases from the three units that are planned for operation at the Indian Point site will require that the combined releases from the three units when added together be within the limits specified in 10 CFR Part 20. This requirement is stated in Section 3.9 of the Technical Specifications for both liquid and gaseous effluents.

The liquid effluent releases from the three nuclear facilities will be discharged from a common discharge canal into the Hudson River. The nearest sources of public drinking water supplies from the Hudson River are located at Chelsea, New York (backup water supply for New York City) and at the Castle Point Veterans Hospital, 22 and 20.5 miles upstream of the Indian Point site, respectively.

During dry periods with low fresh water river flow, tidal action could carry the radioactivity discharge into the river at the Indian Point site upstream to these river water intake points. Conservative analyses made by the applicant indicate that the concentration of radionuclides at these public water intake points would be less than 1% of the concentration of radionuclides being discharged into the river at Indian Point. Since the releases at the site will be less than the limits of 10 CFR Part 20 (and are expected to be less than 10% of the 10 CFR Part 20 limits, based on past experience with Indian Point Unit 1 and other pressurized water reactor plants), the radioactivity levels at these intakes due to the discharges at Indian Point will not be significant.

Gaseous wastes containing some radioactivity are stored in one of four gas decay tanks. One gas tank is utilized for filling, one for holdup for a 45-day decay period, one for discharging to the atmosphere, and one is held in reserve. Disposal of gaseous wastes from Indian Point Unit 2 is by discharge through the plant vent.

The routine gaseous radioactivity releases from the three nuclear facilities will be from three different vents. The combined release of gaseous waste containing radioactivity from these three sources will be limited by the Technical Specifications such that annual average concentrations at the minimum exclusion distance will not exceed the limits of 10 CFR Part 20, Appendix B, of the Commission's regulations. For gaseous halogens and particulates with half-lives greater than eight days, the applicable limits of the Technical Specifications are less than 1% of the limits given in 10 CFR Part 20. The Technical Specifications also require that the maximum release rate of gaseous waste not exceed the annual average limit.

Based on our review we conclude that the means provided by the applicant for the disposal of radioactive waste are substantially the same as those we have approved for other facilities and are acceptable. We also conclude that acceptable means are provided and will be used to keep the release of radioactivity from the plant within ranges that we consider to be as low as practicable.

10.0 AUXILIARY SYSTEMS

The auxiliary systems necessary to assure safe plant shutdown include (1) the chemical and volume control system, (2) the residual heat removal system, (3) the component cooling system, and (4) the service water system. The systems necessary to assure adequate cooling for spent fuel include (1) the spent fuel pool cooling system, (2) the fuel handling system, and (3) the service water system. The designs for these systems are substantially the same as those we reviewed and found acceptable for the Ginna plant.

10.1 Chemical and Volume Control System

The chemical and volume control system (1) adjusts the concentration of boric acid for reactivity control, (2) maintains the proper reactor coolant inventory and water quality for corrosion control, and (3) provides the required seal water flow to the reactor coolant pumps. The amount of boric acid to be added to the core for reactivity control is determined by the operator. The addition of unborated water as a result of operator error could result in an unintentional dilution during refueling, reactor startup, and power operation. The applicant's analysis indicated that because of the slow rate of dilution there is ample time for the operator to become aware of the dilution and to take corrective action. The applicant is actively participating in the development of a device for continuous monitoring of the reactor coolant boron concentration and will evaluate the feasibility of installing such a monitor when developed.

Our review of the chemical and volume control system emphasized those portions involved in routine and emergency injection of concentrated boric acid. We conclude that the design is acceptable.

10.2 Auxiliary Cooling Systems

Subsystems for auxiliary cooling are the component cooling system, the residual heat removal loop, the spent fuel pool cooling loop, and the service water system. The piping for these three systems is designed to the ANSI B31.1 Code for Pressure Piping.

These systems are equivalent in purpose and design to those of other recently licensed plants. On the basis of our review of this plant and others using the similar systems, we have concluded that these systems are acceptable.

10.3 Spent Fuel Storage

The fuel handling system is designed to transfer spent fuel to the storage pool and to provide storage for new fuel. The spent fuel storage facility is basically the same in capacity and design as those used in previously licensed pressurized water reactor plants. The fuel pool is sized to accommodate spent fuel from 1-1/3 core loadings.

As in other designs, mechanical stops will be incorporated in the crane to restrict motion of the spent fuel cask to its assigned area, adjacent to one side of the fuel storage pool. In addition, the spent fuel racks in the area adjacent to the fuel cask storage location would be used only in the event that a complete core is unloaded and one-third of a core from a previous unloading is already in storage.

The pool floor is located below grade level and founded on solid rock. Structural damage from a dropped fuel cask would not result in a rapid loss of water from the pool. Makeup water can be supplied from the demineralizer water supply at a flow rate of 150 gpm. Additional water can be provided in an emergency by the use of temporary hookups to other sources.

As a consequence of our evaluation of the potential consequences of a postulated fuel handling accident, the applicant has agreed to provide charcoal filters in the refueling building to reduce the calculated offsite doses that might result in the event of a fuel handling accident in the refueling building. The installation of the filters will be completed during the first year of full power operation.

We conclude that the designs of the spent fuel storage pool and the fuel handling system are acceptable.

11.0 ANALYSES OF RADIOLOGICAL CONSEQUENCES FROM DESIGN BASIS ACCIDENTS

11.1 General

In order to assess the safety margins of the plant design, a number of operating transients were considered by the applicant, including rod withdrawal during startup and at power, moderator dilution, loss of coolant flow, loss of electrical load, and loss of ac power. The reactor control and protection system is designed so that corrective action is taken automatically to cope with any of these transients. Based on our evaluation of the information submitted by the applicant and our evaluations of other PWR designs at the operating license stage, we conclude that the Indian Point Unit No. 2 control and protection system design is such that these transients can be terminated without damage to the core or to the reactor coolant boundary, and with no offsite radiological consequences.

The applicant and we have evaluated the consequences of potential accidents, including a control rod ejection accident, an accident involving rupture of a gas decay tank, a steamline break accident, a steam generator tube rupture accident, a loss-of-coolant accident, and a refueling accident.

The calculated offsite radiological doses that might result from the control rod ejection accident, and the accident involving rupture of a gas decay tank are well within the 10 CFR Part 100 guidelines.

The consequences of the steamline break and the steam generator tube rupture accidents can be controlled by limiting the permissible concentrations of radioactivity in the primary and secondary coolant systems. The Technical Specifications for the Indian Point Unit No. 2 facility limit the primary and secondary coolant activity concentrations such that the potential 2-hour doses at the exclusion radius that we calculate for these accidents do not exceed 1.5 Rem to the thyroid or 0.5 Rem to the whole body.

Our evaluations of the loss-of-coolant accident and the refueling accident are discussed in the following sections.

11.2 Loss-of-Coolant Accident

The design basis loss of coolant accident (LOCA) for the Indian Point Unit No. 2 plant is similar to that evaluated for other PWR plants in that a double-ended break in the largest pipe of the reactor coolant system is assumed.

Although the basis for the design of the emergency core cooling system is to limit fission product release from the fuel, in our conservative calculation of the consequences of the LOCA we have assumed that the accident results in the release of the following percentages of the total core fission product inventory from the core: 100% of the noble gases, 50% of the halogens, and 1% of the solids. In addition, 50% of the halogens that are released from the core is assumed to plate out onto internal surfaces of the containment building or onto internal components and is not available for leakage. We assume that 10% of the iodine available for leakage from the containment is in the form of organic iodide, and that 5% is in the form of particulate iodine. The reactor is assumed to have been operating at a power of 3217 MWt prior to the accident. The primary containment is assumed to leak at a constant rate of 0.1 percent of the containment volume per day for the first day and 0.05 percent per day thereafter. We evaluated the iodine removal capability of the sodium hydroxide containment spray system and assumed an inorganic iodine removal constant of 4.5 per hour for the spray system. We evaluated the iodine removal capability of the iodine impregnated charcoal filter system and assumed a removal constant of 0.49 per hour for inorganic iodine and a removal constant of 0.048 per hour for organic iodine. Iodine particulates are assumed to be removed by the high efficiency particulate air filters. The inhalation rate of a person offsite is assumed to be 3.5×10^{-4} cubic meters per second.

For the calculation of the two-hour dose at the site boundary we used an atmospheric dispersion factor corresponding to Pasquill Type "F" stability, with a 1 meter per second wind speed and an appropriate building wake effect. We calculated the potential doses at the site boundary for this 2 hour period to be 180 Rem to the thyroid and 4 Rem to the whole body. At the low population zone boundary our calculated potential doses for a 30-day period are 270 Rem to the thyroid and 7 Rem to the whole body.

In evaluating the above doses, no credit was given for the isolation valve seal water injection system, the penetration pressurization system, or the weld channel pressurization system. Operation of these systems, which interpose a high gas pressure or seal water area between the containment and the outside atmosphere at all points where leakage might occur, should significantly reduce the leakage rate from the containment, and, thus reduce the doses following an accident. These systems are well designed and tested, and should be available in the event of an accident (see Section 7.3). We did not consider the effect of these systems in our dose calculations because it is inherently difficult to accurately measure leakage rates of less than 0.1% per day by current testing methods.

The control room for Indian Point Unit No. 2 was not designed to meet the requirements we have imposed in more recent construction permit reviews, that the dose for the course of the accident to occupants of the control room be limited to 5 Rem to the whole body and 30 Rem to the thyroid. In order to provide additional protection to the control room occupants in the event of a loss-of-coolant accident, the applicant has equipped the control room with protective clothing and self-contained air respirators for the operators. In view of these provisions, we have concluded that the control room, as constructed, is acceptable in this regard.

11.3 Fuel Handling Accident

We have evaluated the potential consequences of a fuel handling accident, in which it is postulated that a fuel assembly is dropped in the spent fuel pool or transfer canal. We assumed that: (1) all 204 rods in the dropped bundle are damaged, (2) the accident occurs 90 hours after shutdown of the core from which the dropped bundle has been removed, (3) 20% of the noble gases and 10% of the iodine in the dropped fuel bundle are released to the refueling water and the dropped fuel bundle has been removed from a region of the core which has been generating 1.43 times the average core power, (4) 90% of the released iodine is retained in the refueling water, (5) the fission products released from the pool are discharged to the atmosphere by the building recirculation system through charcoal filters with an iodine removal efficiency of 90%, and (6) the same meteorological conditions exist as were assumed for the loss-of-coolant accident. The resultant calculated doses at the site boundary are 146 Rem to the thyroid and less than 4 Rem to the whole body.

11.4 Conclusions

We have calculated offsite doses for the design basis accidents that have the greatest potential for offsite consequences using assumptions consistent with those we have used in previous safety reviews of PWR plants and have found the resulting calculated doses to be less than the guideline values of 10 CFR Part 100.

12.0 CONDUCT OF OPERATIONS

12.1 Technical Qualifications

The Indian Point Unit 2 facility was designed and is being built by Westinghouse as prime contractor for the applicant. Preoperational testing of equipment and systems at the site and initial plant operation will be performed by Consolidated Edison personnel under the technical direction of Westinghouse. The applicant's experience in the power production field is largely with thermal power plants. However, the applicant has operated Indian Point Unit 1, a 615 megawatt (thermal) pressurized water reactor plant with an oil fired superheater, since August 1962. In addition, the applicant has the Indian Point Unit 3 under construction at the Indian Point site and is actively considering the installation of other nuclear power plants at other sites. Our review of the applicant's organization indicates that the competence of its engineering staff has continually increased and is consistent with the requirements of its expanded nuclear program.

12.2 Operating Organization and Training

The applicant's organization consists of three main groups under the direction of the general superintendent. These groups are the operations group (with a separate superintendent for each unit), the performance group (with the responsibility for station chemistry, licensed personnel training, and surveillance of station performance), and the health physics group headed by a supervisor engineer for health physics (with the responsibility for station health physics and instrumentation). An assistant superintendent for maintenance, and production engineers (responsible for providing staff support for the operation superintendents) report to the two superintendents for operation. A reactor engineer reports directly to the general superintendent.

The proposed shift complement for the combined operation of Indian Point Unit 1 and Indian Point Unit 2 consists of one general watch foreman licensed as a senior reactor operator (SRO), one watch foreman (SRO) for each unit, one control operator A licensed as a reactor operator (RO) for each unit, one unlicensed control room operator B, shared by both units, one control operator B for Indian Point Unit 1 chemical system building, six operating mechanics (two of whom are assigned to Indian Point Unit 2), one shift chemist, and one shift health physics technician.

The shift composition for Indian Point Unit 2 when Indian Point Unit 1 is shutdown for any reason is the general foreman, one watch foreman, one control operator A and two operating mechanics. In addition, a control room operator B may be available a substantial portion of his time. We conclude that both the dual unit crews and single unit crews as outlined above are acceptable.

Since a large part of the plant staff has had prior nuclear experience, the training program has been fitted to individual needs based on experience, educational background and job responsibilities. The training program includes long- and short-term assignments of key staff personnel to technical institutions and operating reactors, to the Westinghouse offsite operator training school, and to on-site classroom training courses for operators and supervisors conducted by both applicant and Westinghouse personnel. We have reviewed these activities in detail and conclude that the combination of reactor operating experience and formal training obtained by the plant staff has adequately prepared them to perform their operational duties.

As a means for the continuing review and evaluation of plant operational safety, the applicant will expand the responsibilities of the Nuclear Facility Safety Committee currently functioning for Indian Point Unit 1 to include Indian Point Unit 2. The committee, which reports to the Executive Vice President, Central Operations, will have a membership of at least 12 persons, and will have responsibilities to: (1) audit and report upon the adequacy of all procedures used in the operation, maintenance, and environmental monitoring of each nuclear plant; (2) review and report upon the adequacy of all proposed changes in plant facilities and procedures pertaining to operation, maintenance, and environmental monitoring and having safety significance;

(3) review and report upon all proposed changes to the Technical Specifications; (4) conduct unannounced spot inspections of plant monitoring operations; (5) review and report upon any activity, the occurrence or lack of which may affect the safe operation of the nuclear plant; and (6) convene, at the request of the nuclear power generation manager or a nuclear plant general superintendent or chairman or vice chairman of the committee, to review and act upon any matter they may deem necessary.

Westinghouse will participate in the startup and initial operation of the plant and will continue to make available technical support to the Indian Point Unit 2 staff during operation of the facility.

We conclude that the applicant's organization is acceptably staffed and technically qualified to perform its operational duties subject to satisfactory completion of licensing examinations of personnel requiring licenses.

12.3 Emergency Planning

The site emergency plan for the Indian Point site describes the emergency organization and its responsibilities. The scope of the emergency plan includes consideration of local contingencies, site contingencies, general (off-site) contingencies, implementation levels for each contingency, notification channels, the support provided by civil authorities, protective measures for each contingency, communications facilities, and training drills. The applicant has provided an extensive description of the medical support that will be available although it is not incorporated explicitly in the plan. The planned medical support provides for emergency treatment of plant personnel both at the site and at a designated hospital where facilities equipment and medical personnel to handle radiation contaminated injured personnel will be available.

We conclude that the applicant's emergency plan is acceptable for Indian Point Unit 2.

12.4 Industrial Security

The immediate plant area (restricted area), including Indian Point Unit 1 will be enclosed by a fence. Access to the restricted area for all personnel will be through manned gatehouses or locked gates which are under the direct control of the station security forces. Security guards will make routine patrols of all property within the site boundary and outside the restricted area and are required to make hourly reports to the central control room.

The controlled area of Indian Point Unit 2 will include the containment, the fuel storage building, the primary auxiliary building, and the emergency diesel generator building. Normal access to these areas is through the existing security room for Indian Point Unit 1. All other doors and hatches leading into the controlled area will be locked and will be supervised by means of door switches connected to the open door alarm board in the security room, and the category alarm board in the Indian Point Unit 1 central control room. The containment personnel hatch doors have remote indicating lights and annunciators that are located in the control room and that indicate the door operational status.

Offsite applicant employees must identify themselves at the main gate prior to admission to the restricted area, receive approval for entry by the general superintendent or his designated representative, and sign in on an admission sheet. If access into the controlled area is approved, they must be accompanied by a qualified guide.

We conclude that the applicant has taken reasonable measures to provide for the security of the facility.

13.0 TECHNICAL SPECIFICATIONS

The Technical Specifications in an operating license define safety limits and limiting safety system settings, limiting conditions for operation, periodic surveillance requirements, certain design features, and administrative controls for the operating plant. These specifications cannot be changed without prior approval of the AEC. The applicant's initial proposed Technical Specifications, presented in Amendment No. 20, have been modified as a result of our review to describe more definitively the allowable conditions for plant operation. The Technical Specifications as approved by the regulatory staff, may be examined in the Commission's Public Document Room.

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 10 CFR Part 20 limits and that means are provided for keeping the release of radioactivity from the plant within ranges that we consider as low as practicable. Furthermore, the limiting conditions of operation and surveillance requirements will assure that necessary engineered safety features to mitigate the consequences of unlikely accidents will be available.

14.0 REPORT OF ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

The ACRS reported on the application for construction of the Indian Point Unit 2 at the proposed site in a letter dated August 16, 1966. The applicant has been responsive to the recommendations made by the ACRS in that letter, and we conclude that the matters raised have been resolved satisfactorily during the design and construction of the Indian Point Unit 2.

The ACRS reported on its review of the application for an operating license for Indian Point Unit 2 in their letter, dated September 23, 1970, attached as Appendix B.

In its letter, the ACRS made several recommendations and noted several items all of which have been considered in the indicated sections of our evaluation. These include: (1) reevaluation of potential flooding at the Indian Point site (Section 3.4), (2) additional seismic reinforcing at the Indian Point Unit No. 1 superheater building and truncation of the superheater stack (Section 6.2), (3) reactor design, power distribution, and control of potential xenon oscillations (Section 4.2), (4) containment design and isolation (Sections 6.2 and 7.3), (5) containment cooling and iodine removal systems (Section 7.2), (6) emergency core cooling system and removal of the reactor pit crucible (Section 7.1), (7) post-accident hydrogen control (Section 7.4),

(8) charcoal filters in the refueling building (Section 10.3), (9) reactor core instrumentation (Section 4.2), (10) reactor protection with only three of four loops in service (Section 8.1), (11) inservice vibration monitoring and loose parts detection (Section 5.9), (12) fuel failure detection (Section 5.9), (13) availability requirements for primary coolant leak detection systems (Section 5.7), (14) pressure vessel fracture toughness (Section 5.2), (15) integrity of high burnup fuel during design transients (Section 4.3), and (16) common mode failure and anticipated transients without reactor scram (Section 8.1).

The ACRS concluded in its letter that if due regard is given to the items recommended above, and subject to satisfactory completion of construction and preoperational testing of Indian Point Unit 2, there is reasonable assurance that this reactor can be operated at power levels up to 2758 MWt without undue risk to the health and safety of the public.

15.0 COMMON DEFENSE AND SECURITY

The application reflects that the activities to be conducted will be within the jurisdiction of the United States and all of the directors and principal officers of the applicant are United States citizens.

The applicant is 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 10 CFR 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 for military purposes, is involved. For these reasons and in the absence of any information to the contrary, we have found that the activity to be performed will not be inimical to the common defense and security.

16.0 FINANCIAL QUALIFICATIONS

The Commission's regulations that relate to the financial data and information required to establish financial qualifications for an applicant for an operating license are 10 CFR Part 50.33(f) and 10 CFR Part 50 Appendix C. The Consolidated Edison Company's application as amended by Amendment No. 21 thereto, and the accompanying certified annual financial statements provided the financial information required by the Commission's regulations.

These submittals contain the estimated operating cost for each of the first five years of operation plus the estimated cost of permanent shutdown and maintenance of the facility in a safe condition. The estimated operating costs are $10.0 million for 1971 (the first year of operation), $14.8 million for 1972, $12 million for 1973, $10.9 million for 1974 and $10.7 million for 1975 (Amendment No. 21). Such costs include the costs of operating and maintenance and fuel. The applicant's estimate of the cost of permanently shutting down the facility and maintaining it in a safe condition is (1) $265,000 for the first year of shutdown and $50,000 for each year thereafter if the reactor core is removed from the vessel, and (2) $240,000 per year if the core is not removed.

We have examined the certified financial statements of the Consolidated Edison Company to determine whether the Company is financially qualified to meet these estimated costs. The information contained in the 1969 financial report indicates that operating revenues for 1969 totaled \$1,028.3 million; operating expenses (including taxes) was \$830.5 million; the interest on the long-term debt was earned 2.3 times; and the net income for the year was \$127.2 million, of which \$102.1 million was distributed as dividends to the stockholders, and the remainder of \$25.1 million was retained for use in the business. As of December 31, 1969, Company's assets totaled \$4,069.6 million, most of which was invested in utility plant (\$3,793.3 million), and earnings reinvested in the business were \$426.1 million. Financial ratios computed from the 1969 statements indicate a sound financial condition, (e.g., long-term debt to total capitalization--0.52, and to net utility plant--0.52; net plant to capitalization--0.994; the operating ratio--0.81; and the rates of return on common--7.7%; on stockholder's investment--6.9%; and on total investment--4.9%). The record of the Company's operations over the past 5 years reflects that operating revenues increased from \$840 million in 1965 to \$1,028 million in 1969; net income increased from \$111.8 million to \$127. million; and net investment in utility plant from \$3,170 million to \$3,793 million. Moody's Investors Service (August 1969 edition) rates the Company's first mortgage bonds as A (high-medium grade). The Company's current Dun and Bradstreet rating (July 1970) is AaA1.

Our evaluation of the financial data submitted by the applicant, summarized above, provides reasonable assurance that the applicant possesses or can obtain the necessary funds to meet the requirements of 10 CFR Part 50.33(f) with respect to the operation of Indian Point Unit 2. A copy of the staff's financial analysis is attached as Appendix H.

17.0 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, licensees for facilities such as power reactors under 10 CFR Part 50.

17.1 Preoperational Storage of Nuclear Fuel

The Commission's regulations in Part 140 require that each holder of a construction permit under 10 CFR Part 50, who is also to be the holder of a license under 10 CFR Part 70 authorizing the ownership and possession for storage only of special nuclear material at the reactor construction site for future use as fuel in the reactor (after issuance of an operating license under 10 CFR Part 50), shall, during the interim storage period prior to licensed operation, have and maintain financial protection in the amount of $1,000,000 and execute an indemnity agreement with the Commission. Proof of financial protection is to be furnished prior to, and the indemnity agreement executed as of, the effective date of the 10 CFR Part 70 license. Payment of an annual indemnity fee is required.

The Consolidated Edison Company, is with respect to Indian Point Unit 2, subject to the foregoing requirements, and has taken the following steps with respect thereto.

The Company has furnished to the Commission proof of financial protection in the amount of \$1,000,000 in the form of a Nuclear Energy Liability Insurance Association policy (Nuclear Energy Liability Policy, facility form) Nos. NF-100.

Further, the Company executed Indemnity Agreement No. B-19 with the Commission as of January 12, 1962, which was amended to cover its pertinent preoperational fuel storage under license SNM-1108 on March 4, 1969. The Company has paid the annual indemnity fee applicable to preoperational fuel storage.

17.2 Operating License

Under the Commission's regulations, 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 furnished, and an indemnity agreement covering such operation (as distinguished from, preoperational fuel storage only) has been executed. The amount of financial protection which must be maintained for reactors which have a rated capacity of 100,000 electrical kilowatts or more is the maximum amount available from private sources, i.e., the combined capacity of the two nuclear liability insurance pools, which amount is currently \$82 million.

Accordingly, no license authorizing operation of Indian Point Unit 2 will be issued until proof of financial protection in the requisite amount has been received and the requisite indemnity agreement executed.

We expect that, in accordance with the usual procedure, the nuclear liability insurance pools will provide, several days in advance of anticipated issuance of the operating license document, evidence in writing, on behalf of the applicant, that the present coverage has been appropriately amended and that the policy limits have been increased, to meet the requirements of the Commission's regulations for reactor operation. The amount of financial protection required for a reactor having the rated capacity of this facility would be $82 million. Consolidated Edison Company will be required to pay an annual fee for operating license indemnity as provided in our regulations, at the rate of $30 per each thousand kilowatts of thermal capacity authorized in its operating license.

On the basis of the above considerations, we conclude that the presently applicable requirements of 10 CFR Part 140 have been satisfied and that, prior to issuance of the operating license, the applicant will be required to comply with the provisions of 10 CFR Part 140 applicable to operating licensees, including those as to proof of financial protection in the requisite amount and as to execution of an appropriate indemnity agreement with the Commission.

18.0 CONCLUSIONS

Based on our evaluation of the application as set forth above, we have concluded that:

1. The application for facility license filed by the Consolidated Edison Company of New York, Inc., dated December 6, 1965, as amended (Amendments Nos. 9 through 25, dated October 15, 1968, October 13, 1969, October 24, 1969, November 21, 1969, December 29, 1969, January 27, 1970, March 2, 1970, March 30, 1970, April 17, 1970, June 3, 1970, July 14, 1970, July 17, 1970, July 28, 1970, July 29, 1970, August 13, 1970, August 28, 1970, and November 12, 1970, respectively) 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 1; and
2. Construction of the Indian Point Nuclear Generating Unit No. 2 (the facility) has proceeded and there is reasonable assurance that it will be completed, in conformity with Provisional Construction Permit No. CPPR-21, the application as amended, the provisions of the Act, and the rules and regulations of the Commission; and
3. The facility will operate in conformity with the application as amended, the provisions of the Act, and the rules and regulations of the Commission; and

4. There is reasonable assurance (i) that the activities authorized by the operating license can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the regulations of the Commission set forth in 10 CFR Chapter 1; and
5. The applicant is technically and financially qualified to engage in the activities authorized by this operating license, in accordance with the regulations of the Commission set forth in 10 CFR Chapter 1; and
6. The applicable provisions of 10 CFR Part 140 have been satisfied; and
7. The issuance of this license will not be inimical to the common defense and security or to the health and safety of the public.

Prior to any public hearing on the matter of the issuance of an operating license to Consolidated Edison for Indian Point Unit No. 2, the Commission's Division of Compliance will prepare and submit a supplement to this Safety Evaluation which will deal with those matters relating to the status of construction completion and conformaty of this construction to the provisional construction permit and the application. Before an operating license will be issued to Consolidated Edison for Indian Point Unit No. 2, assuming such a license is authorized following the public hearing, the facility must be completed in conformity with the provisional construction permit, the application, the Act, and the rules and regulations of the Commission. Such completeness of construction as is required for safe operation at the authorized power level must be verified by the Commission's Division of Compliance prior to license issuance.

APPENDIX A

CHRONOLOGY OF REGULATORY REVIEW OF THE CONSOLIDATED EDISON COMPANY INDIAN POINT NUCLEAR GENERATING PLANT UNIT NO. 2 (SUBSEQUENT TO CONSTRUCTION PERMIT NO. CPPR-21 ISSUED ON OCTOBER 14, 1966)

1.	April 17, 1967	Submittal of Amendment No. 6 containing design information on the Emergency Core Cooling System and other areas as requested by the ACRS in their letter to the Chairman AEC, of 8/16/66.
2.	July 18, 1967	Meeting with applicant to discuss revised design of Emergency Core Cooling System and other areas as per Amendment No. 6.
3.	August 2, 1967	Letter to applicant requesting additional information on subjects addressed by the ACRS in their letter of 8/16/66.
4.	October 16, 1967	Submittal of Amendment No. 7 in response to DRL request of August 2, 1967.
5.	October 31, 1967	Submittal of Amendment No. 8, revised pages for Amendment No. 7.
6.	December 28, 1967	ACRS Subcommittee meeting to discuss emergency core cooling system, reactor pit crucible, primary coolant system, other areas.
7.	January 30, 1968	Submittal of "Report on the Containment Building Liner Plate Buckle in the Vicinity of the Fuel Transfer Canal".
8.	February 2, 1968	Meeting with applicant to discuss content of Amendments No. 6, 7, and 8.
9.	February 13, 1968	Meeting with applicant to complete discussion of February 2, 1968.

10.	March 8, 1968	ACRS Full Committee meeting to discuss Emergency Core Cooling System; reactor internals; primary coolant system, design, fabrication, in-service inspection, and leak detection; core design; reactor pit crucible; and containment liner quality control and stress analysis.
11.	October 15, 1968	Consolidated Edison Company filed application for an Operating License for the IP-2 Plant. Amendment 9, Volumes 1, 2, 3, & 4.
12.	March 5, 1969	AEC-DRL requested additional information on medical and emergency plans.
13.	March 12, 1969	AEC-DRL staff met with Con Ed personnel to discuss scheduling of regulatory review of application for operating license.
14.	April 3, 1969	AEC-DRL staff met with Con Ed personnel to discuss structural and seismic design and tornado protection.
15.	April 16, 1969	AEC-DRL staff met with Con Ed to discuss accidental and normal radioactivity release from the IP-2 plant.
16.	April 28, 1969	Con Ed requested extension of completion date for construction of the IP-2 plant.
17.	May 2, 1969	AEC-DRL staff and Nathan M. Newmark, seismic design consultant, met with Con Ed personnel at the IP-2 site to discuss seismic design and review status of construction and site inspection.
18.	May 19, 1968	AEC-DRL staff issued an order extending completion date for construction of the IP-2 plant to June 1, 1970.

19.	August 4, 1969	Request to applicant for additional information on site and environment, reactor coolant system, containment system, engineered safety features, instrumentation and control, electrical systems, waste disposal and radiation protection, conduct of operations, and accident analysis.
20.	August 22, 1969	AEC-DRL staff requests copies of monitoring reports and status of actions on Fish and Wildlife recommendations.
21.	August 23, 1969	ACRS Subcommittee meeting on tornado protection, emergency planning, permanent in-core instrumentation, adequacy of onsite emergency power, and containment isolation.
22.	September 24, 1969	Meeting with applicant to discuss Westinghouse presentation on power distribution detection and control in Indian Point 2.
23.	October 13, 1969	Submittal of Amendment 10 (Supplement #1) responses to AEC regulatory staff's request of March 5, 1969, on medical plans and partial answers to AEC regulatory staff's request for additional information of August 4, 1969.
24.	October 24, 1969	Submittal of Amendment No. 11, replacement pages and responses to AEC regulatory staff's request for additional information of August 4, 1969, on Sections 1, 4, 5, 6, 7, 12, and 14 of the FSAR.
25.	November 13, 1969	Request for additional information on reactor, reactor coolant system, containment system, engineered safety features, auxiliary and emergency systems, initial tests and operations, and accident analysis.
26.	November 21, 1969	Submittal of Amendment No. 12, additional and replacement pages to be inserted into the FFDSAR and further responses to AEC regulatory staff's request for additional information of 8/4/69 on Sections 1, 4, 7, 8 and 11 of the FFDSAR.

27.	December 10, 1969	Meeting with applicant to review electrical drawings including AC power, DC power, Reactor Protection System, and Engineered Safety Features.
28.	December 30, 1969	Meeting with applicant and Westinghouse Electric Corporation to continue detailed review of electrical drawings including Reactor Protection System and Engineered Safety Features.
29.	January 16, 1970	Meeting with applicant to review and discuss electrical drawings including Reactor Protection System and Engineered Safety Features.
30.	January 21, 1970	Meeting with applicant & Westinghouse Electrical Corporation on technical specifications.
31.	January 27, 1970	Submittal of Amendment No. 14, replacement pages for FSAR & further responses to AEC-DRL questions of 8/4/69 & 11/13/69, chapters 1, 4, 6, 11, 12 & 14.
32.	February 17, 1970	Meeting with applicant for presentation of results of Con Ed's Analysis concerning potential damage to Indian Point 2 and IP-3 from a failure of the IP-1 superheater stack.
33.	March 2, 1970	Submittal of Amendment No. 15, responses to AEC regulatory staff's requests for additional information of 8/4 and 11/13, 1969 and Containment Design Report.
34.	March 10, 1970	Request to applicant for additional financial data.
35.	March 13, 1970	Meeting with applicant to discuss questions concerning core heat transfer and burnout limits, fuel element performance and ECCS performance during a LOCA.

36.	March 19, 1970	Meeting with applicant, Westinghouse presentation on iodine removal system for IP-2.
37.	March 26, 1970	Meeting with applicant to discuss analysis of fresh water flood and changes to electrical systems.
38.	March 30, 1970	Submittal of Amendment No. 16, additional and replacement pages for the FSAR and further responses to the AEC regulatory staff's request for additional information of August 4 and November 13, 1969.
39.	April 25, 1970	ACRS Subcommittee meeting and meeting with applicant on instrumentation and control, and anticipated transients with failure to scram.
40.	April 17, 1970	Submittal of Amendment No. 17, additional and replacement pages to be inserted into the FSAR and further responses to AEC regulatory staff's request for additional information of August 4 and November 13, 1969.
41.	April 29, 1970	Meeting with applicant to discuss seismic and structural design questions for IP-2.
42.	May 5, 1970	Meeting with applicant to discuss failure mode analysis of the engineered safety feature manual actuation panel.
43.	May 11, 1970	ACRS Subcommittee meeting at the Indian Point 2 site to discuss instrumentation and control and Electrical Systems.
44.	May 12, 1970	AEC issued Order extending completion date for construction of the IP-2 plant to June 1, 1971.
45.	May 28, 1970	ACRS Subcommittee meeting to discuss loss-of-coolant accident, anticipated transients with failure to scram.
46.	June 3, 1970	Submittal of Amendment No. 18, additional and revised pages for the FSAR in response to AEC regulatory staff request for additional information.

47.	June 11, 1970	ACRS full Committee meeting to consider design of engineered safety feature manual actuation panel and operation with less than four loops.
48.	June 17, 1970	Meeting with applicant to discuss consequences of turbine missiles, sensitized stainless steel control room accident dose, hydrogen recombiner.
49.	July 15, 1970	Submittal of Amendment No. 19 (Supplement 10), additional and revised pages for the FSAR and Flooding Evaluation report.
50.	July 20, 1970	Submittal of Amendment No. 20, (Supplement 11) proposed Technical Specifications.
51.	July 24, 1970	Request for additional information on emergency core cooling, reactor coolant system, instrumentation and control, electrical systems, conduct of operations and accident analysis.
52.	July 28, 1970	Submittal of Amendment No. 21, Con Ed Annual Report.
53.	July 28 and 29, 1970	ACRS Subcommittee meeting to discuss technical specifications, flood protection, Unit No. 1 superheater stack failure and containment sprays.
54.	July 30, 1970	Submittal of Amendment No. 22, (Supplement 12), revised pages for FSAR in response to request for additional information.
55.	August 7, 1970	Meeting with applicant to discuss technical specifications.
56.	August 13, 1970	ACRS full Committee meeting to discuss the matters addressed in our July 2, 1970 report.
57.	August 14, 1970	Submittal of Amendment No. 23 (Supplement 13), answers to request for additional information issued July 24.

58.	August 18, 1970	Meeting to discuss licensed operator requirements.
59.	August 28, 1970	Submittal of Amendment No. 24 (Supplement 14). Revised pages to the FSAR.
60.	September 1, 1970	Meeting with applicant regarding performance of Emergency Core Cooling System.
61	September 9, 1970	Meeting with the applicant to discuss Technical Specifications.
62.	October 21, 1970	Request to applicant for a report on analysis of laminations in base plate material of the IP-2 pressurizer.
63.	October 29, 1970	Meeting with applicant to review technical specifications for the Indian Point 2 plant.
64.	November 1970	Submittal of Amendment 25 (Supplement 15), changes to technical specifications and to FSAR.

APPENDIX B

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

SEP 23 1970

Honorable Glenn T. Seaborg
Chairman
U. S. Atomic Energy Commission
Washington, D. C. 20545

Subject: REPORT ON INDIAN POINT NUCLEAR GENERATING UNIT NO. 2

Dear Dr. Seaborg:

At its 125th meeting, September 17-19, 1970, the Advisory Committee on Reactor Safeguards completed its review of the application by Consolidated Edison Company of New York, Inc., for authorization to operate the Indian Point Nuclear Generating Unit No. 2. This project had previously been considered at the Committee's 95th, 98th, 122nd, and 124th meetings, and at Subcommittee meetings on August 23, 1969, March 13, 1970, April 25, 1970, May 28, 1970, July 28-29, 1970, and September 15, 1970. Subcommittees also met at the site on December 28, 1967 and May 11, 1970. The Committee last reported on this project to you on August 16, 1966. During the review, the Committee had the benefit of discussions with representatives of the Consolidated Edison Company and their contractors and consultants, and with representatives of the AEC Regulatory Staff. The Committee also had the benefit of the documents listed.

The Indian Point site is located in Westchester County, New York, approximately 24 miles north of the New York City limits. The minimum radius of the exclusion area for Unit No. 2 is 520 meters and Peekskill, the nearest population center, is approximately one-half mile from the unit. Also at this site are Indian Point Unit 1, which is licensed for operation at 615 MWt, and Unit 3, which is under construction.

The applicant has re-evaluated flooding that could occur at the site in the event of the probable maximum hurricane and flood, in the light of more recent information, and has concluded that adequate protection exists for vital components and services.

Additional seismic reinforcement being provided for the Indian Point Unit No. 1 superheater building and removal of the top 80 ft. of the superheater stack will enable the stack to withstand winds in the range of 300-360 mph corresponding to current tornado design criteria. Since the reinforcement of the superheater building, which supports the stack, enables the stack to resist wind loads of a magnitude most likely to be experienced from a tornado, the Committee believes that removal of the top 80 ft. of the stack, to enable it to resist the maximum effects from a tornado, may be deferred until a convenient time during the next few years, but prior to the commencement of operation of Indian Point Unit No. 3. The applicant has stated that truncation of the stack will have no significant adverse effect on the environment.

The Indian Point Unit No. 2 is the first of the large, four-loop Westinghouse pressurized water reactors to go into operation, and the proposed power level of 2758 MWt will be the largest of any power reactor licensed to date. The nuclear design of Indian Point Unit No. 2 is similar to that of H. B. Robinson with the exception that the initial fuel rods to be used in Indian Point Unit No. 2 will not be prepressurized. Part-length control rods will be used to shape the axial power distribution and to suppress axial xenon oscillations. The reactor is designed to have a zero or negative moderator coefficient of reactivity, and the applicant plans to perform tests to verify that divergent azimuthal xenon oscillations cannot occur in this reactor. The Committee recommends that the Regulatory Staff follow the measurements and analyses related to these tests.

Unit 2 has a reinforced concrete containment with an internal steel liner which is provided with facilities for continuous pressurization of weld and penetration areas for leak detection, and a seal-water system to back up piping isolation valves. In the unlikely event of an accident, cooling of the containment is provided by both a containment spray system and an air-recirculation system with fan coolers. Sodium hydroxide additive is used in the containment spray system to remove elemental iodine from the post-accident containment atmosphere. An impregnated charcoal filter is provided to remove organic iodine.

Major changes have been made in the design of the emergency core cooling system as originally proposed at the time of the construction permit review. Four accumulators are provided to accomplish rapid reflooding of the core in the unlikely event of a large pipe break, and redundant pumps are included to maintain long-term core cooling. The applicant has analyzed the efficacy of the emergency core cooling system and concludes that the system will keep the core intact and the peak clad temperature well below the point where zircaloy-water reaction might have an adverse effect on clad ductility and, hence, on the continued structural integrity of the fuel elements. The Committee believes that there is reasonable assurance that the Indian Point Unit No. 2 emergency core cooling system will perform adequately at the proposed power level.

The Committee concurs with the applicant that the reactor pit crucible, proposed at the time of the construction permit review, is not essential as a safety feature for Indian Point Unit No. 2 and need not be included.

To control the concentration of hydrogen which could build up in the containment following a postulated loss-of-coolant accident, the applicant has provided redundant flame recombiner units within the containment, built to engineered safety feature standards. Provisions are also included for adequate mixing of the atmosphere and for sampling purposes. The capability exists also to attach additional equipment so as to permit controlled purging of the containment atmosphere with iodine filtration. The Committee believes that such equipment should be designed and provided in a manner satisfactory to the Regulatory Staff during the first two years of operation at power.

The applicant plans to install a charcoal filter system in the refueling building to reduce the potential release of radioactivity in the event of damage to an irradiated fuel assembly during fuel handling. This installation will be completed by the end of the first year of full power operation.

The reactor instrumentation includes out-of-core detectors, fuel assembly exit thermocouples, and movable in-core flux monitors. Power distribution measurements will also ordinarily be available from fixed in-core detectors.

The applicant has proposed that a limited number of manual resets of trip points, made deliberately in accordance with explicit procedures, by approved personnel, independently monitored, and with settings to be calibrated and tested, should provide an acceptable basis for the occasional operation of Indian Point Unit No. 2 with only three of the four reactor loops in service. The Committee concurs in this position.

The applicant stated that neutron noise measurements will be made periodically and analyzed to provide developmental information concerning the possible usefulness of this technique in ascertaining changes in core vibration or other displacements. On a similar basis, accelerometers will be installed on the pressure vessel and steam generators to ascertain the practicality of their use to detect the presence of loose parts.

The reactor includes a delayed neutron monitor in one hot leg of the reactor coolant system to detect fuel element failure. Suitable operability requirements will be maintained on the several sensitive means of primary system leak detection.

A conservative method of defining pressure vessel fracture toughness should be employed that is satisfactory to the Regulatory Staff.

The applicant stated that existing experimental results and analyses provide considerable assurance that high burnup fuel of the design employed will be able to undergo anticipated transients and power perturbations without a loss of clad integrity. He also described additional experiments and analyses to be performed in the reasonably near future which should provide further assurance in this regard.

The Committee has, in recent reports on other reactors, discussed the need for studies on further means of preventing common failure modes from negating scram action, and of possible design features to make tolerable the consequences of failure to scram during anticipated transients. The applicant has provided the results of analyses which he believes indicate that the consequences of such transients are tolerable with the existing Indian Point Unit No. 2 design at the proposed power level. Although further study is required of this general question, the Committee believes it acceptable for the Indian Point Unit No. 2 reactor to operate at the proposed power level while final resolution of this matter is made on a reasonable time scale in a manner satisfactory to the Regulatory Staff. The Committee wishes to be kept advised.

Other matters relating to large water reactors which have been identified by the Regulatory Staff and the ACRS and cited in previous ACRS letters should, as in the case of other reactors recently reviewed, be dealt with appropriately by the Staff and the applicant in the Indian Point Unit No. 2 as suitable approaches are developed.

The ACRS believes that, if due regard is given to the items recommended above, and subject to satisfactory completion of construction and preoperational testing of Indian Point Unit No. 2, there is reasonable assurance that this reactor can be operated at power levels up to 2758 MWt without undue risk to the health and safety of the public.

Sincerely yours,

Original Signed by
Joseph M. Hendrie

Joseph M. Hendrie
Chairman

References attached.

References - Indian Point Nuclear Generating Unit No. 2

1. Amendment No. 9 to Application of Consolidated Edison Company of New York for Indian Point Nuclear Generating Unit No. 2, consisting of Volumes I - IV, Final Safety Analysis Report, received October 16, 1968

2. Amendments 10 - 20 to the License Application
3. Amendments 22 - 24 to the License Application

APPENDIX C

Comments on

Indian Point Nuclear Generating Unit No. 2
Consolidated Edison Company of New York, Inc.
Final Facility Description and Safety Analysis Report
Volumes I, II, III and IV dated October 15, 1968

Prepared by

Air Resources Environmental Laboratory
Environmental Science Services Administration
November 29, 1968

As pointed out in our comments of October 29, 1965 on Unit No. 2, a primary influence on the meteorological statistics of the Indian Point site seems to be its location in a river valley about a mile wide with terrain rising 600 to 1000 feet on either side. Consequently, wind directions follow a pronounced diurnal cycle with daytime, unstable (lapse) flow in the upriver direction and nighttime, stable flow in the downriver directions. The report documents a 42.4 percent inversion frequency, but it should also be pointed out that inversion conditions are largely confined to the nighttime, downriver flow lasting about 12 hours before changing to lapse or upriver flow. Figure 2.6-1, although in terms of average vectors, shows the marked wind reversals at sunset and sunrise and the rather persistent, channeled flow that can occur during the middle of the night (see the mean direction between 0200 and 0800 hours). The mean wind speeds during this persistent period is about 2.5 m/sec which indicates that 50 percent of the time inversion wind speeds could be less than 2.5 m/sec.

In the absence of specific, joint-frequency wind speed and direction persistence data from the site, a reasonably conservative meteorological model would be to assume for a ground release a 1 m/sec wind speed under inversion conditions in a persistent downriver direction for a period of 8 hours. Taking into account the likelihood of a diurnal wind reversal, a very conservative assumption would be to allow the plume centerline to meander over a 22-1/2° arc under the same conditions for the remainder of the 24-hour period. Again, with no specific on-site wind persistence data, the conservative assumption has been made.

The amount of additional atmospheric diffusion because of the building turbulence can be assessed by the virtual point source expression $[(x + x_o)/x]^{1.5}$ as used by the applicant, which for a value of x_o = 430 m amounts to a factor of 2.5 at the site boundary (520 m) and 1.6 at the low population boundary (1100 m). These values are in close agreement with the method of using a shape factor of 1/2 and a building cross-section of 2000 m^2.

In summary, from data presently available, it would seem reasonably conservative to assume a persistent wind direction for an 8-hour period under inversion conditions and a 1 m/sec wind speed. With the added assumption of a building wake shape factor of 1/2 and a cross-sectional area of 2000 m^2, the resulting 0-8 hr relative concentration would be 6.6×10^{-4} sec m^3 at the site boundary and 3.7×10^{-4} at the low population boundary. From Table 14.3.5-3 one can calculate that the applicant's model for the 0-8 hr period results in an average relative concentration of 4.8×10^{-4} and 2.4 sec m^{-3} at the site and low population boundary, respectively.

APPENDIX C

Comments on

Indian Point Nuclear Generating Unit No. 2
Consolidated Edison Company of New York, Inc.
Final Facility Description and Safety Analysis
Amendment No. 12 dated November 21, 1969, and
Amendment No. 14 dated January 27, 1970

Prepared by

Air Resources Environmental Laboratory
Environmental Science Services Administration
February 17, 1970

The original documentation of the Indian Point site during the period 1955-1957 indicates that at the 100-ft. height the annual prevailing wind direction is from the north northeast and that in the sector from 22.5 to 42.5 degrees the frequency of inversion, neutral and lapse conditions was 6, 2, and 1 percent, respectively. Within this sector, the shortest site boundary is approximately in a direct line through Units 2 and 3 at a distance of 610 and 380 m, respectively, as measured from figure 2.2-2. It is about 500 m from the Unit 1 stack to this common boundary point. The nearest site boundary, regardless of sector, is where the property line intersects the downriver edge of the site. Although this point is at a distance of 580 m from Unit 2, it is not in the most prevalent wind direction by a considerable amount.

To compute the average annual dilution factor we have assumed the frequencies listed above, averaged over a 20-degree sector with a wind speed of 2, 4 and 5 m/sec, respectively, for inversion (Type F), neutral (Type D), and lapse (Type B) conditions. Assuming no building wake effect our results show the applicant's values for Units 1 and 2 to be reasonably conservative. In the case of Unit 3 we compute an average annual dilution factor of 2.9×10^{-5} sec m^{-3} as compared to the applicant's value of 1.6×10^{-5} sec m^{-3}. The only explanation we have for the ESSA value being twice as high is the use of the building wake effect in the applicant's assumptions.

It is our view that the use of the building wake effect in the long-term average diffusion equation, as was done by the applicant, is inappropriate. It does not seem logical that for the same atmospheric conditions the Sutton equation on page Q 11.10-1 for the long-term model gives <u>more</u> credit for building wake effect than the equivalent short-term model on p. Q 11.10-2. For example at $x = 400$ m assuming $x_o = 400$ m and $n = 0.5$, the building wake effect, $[(x+x_o)/x]^{2-n/2}$, for the long-term equation is 3.4 whereas for the effect in the short-term equation, $[(x+x_o)/x]^{2-n}$, the value is 2.8. It is the larger exponent in the former that makes the difference. Also, the fact that one averages in the horizontal dimension over a sector essentially would nullify any added dilution in that dimension because of wake effect.

APPENDIX D

DEPARTMENT OF THE ARMY
COASTAL ENGINEERING RESEARCH CENTER
5201 LITTLE FALLS ROAD, N.W.
WASHINGTON, D.C. 20016

CEREN

21 November 1969

Mr. Roger S. Boyd
Asst. Director for Reactor Projects
Division of Reactor Licensing
U. S. Atomic Energy Commission
Washington, D. C. 20545

Dear Mr. Boyd:

Reference is made to your letters regarding Docket Nos. 50-247, 50-286, 50-342, and 50-343, Consolidated Edison Company of New York's proposed Indian Point Nuclear Generating Units No. 2 and No. 3, and Units No. 4 and No. 5 which are contiguous to Indian Point plant site.

Pursuant with our arrangements, Mr. R. A. Jachowski and Mr. B. R. Bodine of CERC have reviewed all pertinent information contained in the reports from the standpoint of establishment of a design water level. This included the review of the storm surge associated with the Probable Maximum Hurricane (PMH) and wind wave analysis.

We concur with the applicant's finding that the design water level should be 14.5 feet above the mean sea level datum for Units, Nos. 2, 3, 4 and 5. Although this value is acceptable, there are compensating errors in routing procedure employed.

If you have any further questions regarding this matter please let us know.

Sincerely yours,

Edward M. Willis

EDWARD M. WILLIS
Lieutenant Colonel, CE
Director

APPENDIX E

UNITED STATES
DEPARTMENT OF THE INTERIOR
GEOLOGICAL SURVEY
WASHINGTON, D.C. 20242

SEP 16 1970

Mr. Harold Price
Director of Regulation
U.S. Atomic Energy Commission
7920 Norfolk Avenue
Bethesda, Maryland 20545

Dear Mr. Price:

Transmitted herewith in response to a request by R. C. DeYoung is a review of the flood information presented in Amendment No. 19 to the Final Safety Analysis Report for Unit No. 2 Indian Point Nuclear Generating Station. It is presumed that the flood levels for all 3 units at the Indian Point Stations will be based on this amendment. Copies of our earlier reviews, for Unit No. 2 (Aug. 15, 1966) prepared by E. L. Meyer, and for Unit No. 3 ([illegible]uary 6, 1969) prepared by P. J. Carpenter, are attached.

This review was prepared by P. J. Carpenter and has been discussed with members of your staff. We have no objection to your making this review a part of the public record.

Sincerely yours,

B. A. Nathan

Acting Director

Enclosures

Consolidated Edison Company of New York Inc.
Indian Point Nuclear Generating Station Unit No. 2
Docket No. 50-247

The probable maximum flood as defined by the U.S. Army Corps of Engineers, at the site, has been calculated as 1,160,000 cubic feet per second. This discharge is approximately five times greater than the maximum observed flood at Green Island, and is approximately twice the maximum discharge observed for nearby like-sized drainage basins which appear to exhibit similar runoff characteristics. The stage for the maximum probable flood at the site, computed using standard step-backwater procedures, is given as varying between 13.4 and 14.0 ft msl (mean sea level) depending on concurrent tide levels at the Battery. It is shown that none of the dams on the Hudson River and its tributaries would fail during the probable maximum flood. The above results were obtained using conservative assumptions and appear to be reasonable.

The analyses show that the occurrence of the probable maximum flood on Esopus Creek would cause failure of Ashokan Dam some 75 miles upstream of the site. To establish a flood design level at Indian Point various combinations of the following factors were considered: 1) the flow resulting from the Ashokan Dam failure, 2) various concurrent Hudson River flood flows, and 3) various concurrent tide levels at the Battery. The results of these combinations of factors were compared with the stage of the probable maximum flood (14.0 ft msl) and the stage resulting from the probable maximum hurricane plus spring high tide (14.5 ft msl). The most critical combination investigated consisted of the flows from the Ashokan Dam failure caused by the probable maximum flood on Esopus Creek, the concurrent standard project flow (one half the probable maximum flood), the concurrent stage at the Battery corresponding to the standard project hurricane tide level and wind waves of one foot at the site. This stage is given as 15.0 ft msl. The lowest floor elevation of Unit No. 2 is given as 15.25 ft msl.

Other combinations of the above-mentioned factors, such as Ashokan Dam failure and the standard project hurricane or floods larger than the standard project flood on the Hudson River, could produce higher stages at the site. Depending on the degree of conservatism desired, any of these higher stages could also be selected as the design flood level. However, the stage for the combination selected for the design flood level exceeds those given for the probable maximum flood or probable maximum hurricane when these are considered as independent events.

NATHAN M. NEWMARK
CONSULTING ENGINEERING SERVICES

APPENDIX F

1114 CIVIL ENGINEERING BUILDING
URBANA, ILLINOIS 61801

REPORT TO THE AEC REGULATORY STAFF

STRUCTURAL ADEQUACY

OF

INDIAN POINT NUCLEAR GENERATING UNIT NO. 2

Consolidated Edison Company of New York, Inc.

Docket No. 50-247

by

N. M. Newmark
and
W. J Hall

Urbana, Illinois

20 August 1970

REPORT TO THE AEC REGULATORY STAFF

STRUCTURAL ADEQUACY

OF

INDIAN POINT NUCLEAR GENERATING UNIT NO. 2

INTRODUCTION

This report is concerned with the structural adequacy of the containment structures, piping, equipment and other critical components for the Indian Point Nuclear Generating Unit No. 2 for which application for a construction permit and an operating license has been made to the United States Atomic Energy Commission by the Consolidated Edison Company of New York, Inc. The facility is located on the east bank of the Hudson River at Indian Point, village of Buchanan, in upper Westchester County, New York. The site is about 24 miles N of the New York City boundary and 2.5 miles SW of Peeksill, New York.

This report is based on a review of the Final Facility Description and Safety Analysis Report (Ref. 1) and the containment design report (Ref. 2). The report also is based in part on the discussion and inspection resulting from the visit to the site on 2 May 1969 by N. M. Newmark and W. J. Hall in conjunction with Mr. K. Kniel and Mr. M. McCoy of AEC-DRL. A number of topics were discussed with the applicant and his consultants at the time of this visit, and subsequently additional information has become available through supplements to the FSAR and through discussions with the personnel of DRS, DRL, and the applicant and his consultants. A discussion of the adequacy of the structural criteria presented in the Preliminary Safety Analysis Report is contained in our report of August 1966 (Ref. 3), and unless otherwise noted no comment will be made in this report concerning points covered there.

2

The design criteria for the containment system and Class I components for this plant called for a design to withstand a Design Basis Earthquake of 0.15g maximum horizontal ground acceleration coupled with other appropriate loadings to provide for containment and safe shut down. The plant was also to be designed for an Operating Basis Earthquake of 0.1g maximum horizontal ground acceleration simultaneously with the other appropriate loads forming the basis of containment design.

COMMENTS ON ADEQUACY OF DESIGN

Dynamic Analyses

(a) Containment Building. The answer to Question 1.9 of the FSAR indicates that only the containment building, the primary auxiliary building, and the electric cable tunnel were designed with the use of semi-formal dynamic analyses. A description of the method of analysis employed is given briefly in Section 5.1.3.8 of the FSAR and in Section 3.1.5 of the containment design report. The procedure employed involved a calculation of the fundamental frequency and mode shape by use of a modified Rayleigh method. The base shear for the structure was computed from the period and the spectral response corresponding to the appropriate degree of damping. The base shear was then applied as a loading to the structure as an inverted triangular loading. The shears at the nodes were used to calculate the moments and displacements at various points in the structure. For the structures involved it is believed that the approach leads to a design which is reasonably adequate.

A similar approach was followed for the primary auxiliary building as described in the answer to Question 1.9. It is noted there that a one-third increase over working stress was allowed in the design of the bracing in the case of the Design Basis Earthquake. This stress is below yield, and it is believed that the design will prove to be satisfactory.

(b) Other Buildings and Equipment. The discussion presented in answer to Question 1.9 of the FSAR for other buildings and equipment such as the control building, fan house, intake structure, etc., indicate that a refined static approach was used, which involves employing the peak value from the appropriate response spectrum curve for a given value of damping and multiplying this by the appropriate mass to obtain the inertial loading. From the description given for the various buildings and items of equipment, and the modeling techniques employed, it is concluded that the inertial loadings used in design are reasonably close to those that might be obtained with a more sophisticated analysis and lead to reasonable design values.

The submission in Question 1.3 of Supplement 13 indicates that the Turbine Building, and Fuel Storage Building Structure above the Fuel Storage Pit were reanalyzed by a multi-degree-of-freedom modal dynamic analysis method to check their adequacy. As a result of this reanalysis, the applicant advises that certain structural modifications will be made to columns and cross bracing in the Turbine Building to insure that it can withstand the DBE. The superstructure of the fuel storage building was ascertained to be adequately designed, without modification to withstand the effects of the DBE. The applicant states that reanalysis of the strengthened turbine building and superheater building for Indian Point No. 1 does not significantly affect the responses calculated for the original structures.

(c) Piping Analysis. The method used by the applicant for analysis of the piping, as described in the answer to Question 1.6 of the FSAR, is the same as was used in Ginna. The peak ground response spectrum value for 0.5 percent damping was used, applied as static accelerations in each direction

4

separately, and the resulting stresses superposed. It was assumed by the applicant that the piping was supported along rigid systems and therefore not subjected to amplified ground motion at points of support. The system was analyzed with the anchors and supports as actually used, according to the discussion presented to us during the time of our visit in May, 1969. It was the view of the applicant that the thermal motions were greater than any differential ground displacements and the latter therefore are not critical items in the design. In answer to Question 1.13 (Suppl. 13) the applicant advises that relative seismic displacement was considered for the main steam lines, where the largest relative displacements are expected; stress differentials of less than 10% resulted. Also, seismic supports installed to date are those specified in the design and employed in the analyses; where deviations in supports must occur, reanalysis will be carried out. These results and approaches appear satisfactory to us.

Since this plant was designed before recent developments and changes in piping design specifications, the 1968 ASME Addenda were not applied. Blow-down and earthquake were considered as separate items and not combined in this design. We are advised that the response to Question 1.9 of Supplement 12 states that a review of the Indian Point 3 reactor coolant system which is identical to Indian Point 2, for combined earthquake and blow-down indicates that the design is adequate.

It is stated in the answer to Question 1.6 of the FSAR that the approach resulted in a seismic design load approximately equal to 0.60W horizontally and 0.40W vertically taken simultaneously. It is further stated that for the Design Basis Earthquake the sum of the resulting additional stress plus the normal stresses was limited to 1.2 times the B31.1 code allowable stresses. In a similar manner the stresses in the pipe supports and hangers were limited to 1.2 times code allowable stresses.

The applicant originally made use of the maximum spectrum value only and no modal analyses were made; in other words only a static analysis with uniform accelerations was made. Consideration was not given to modified distribution of the inertial loading to take account of the combination of modal effects.

The response to Question 1.9 of Supplement 8, describing more detailed analyses of the reactor coolant system, feedwater lines, surge lines and typical steam lines by more formal methods as carried out later lends confirmation to the adequacy of the design. On this basis, there is reason to believe that the design is adequate.

Backfill Surrounding Containment Vessel

Nine feet of crushed rock backfill was placed between the external wall of the reinforced concrete containment vessel and the retaining wall holding back the rock on the uphill side. This crushed rock backfill is drained at the bottom to avoid water pressure against the containment structure. The fill is approximately 60 to 70 feet higher on one side of the structure than on the other because of the slope of the rock surface. The design, as discussed in Section 3.1.5 of the containment design report, considered local inertial forces of loose rock as an added loading against the containment pressure vessel, and also considered passive pressures caused by failure of the rock along the surface behind the retaining wall. The localized loadings from these forces were considered in the design of the containment structure and the discussion presented in the containment design report provides reasonable assurance that the containment vessel is capable of resisting these localized forces.

Class I Equipment in Structures other than Class I

The turbine building is Class III and not designed for earthquake loadings. The answer to Question 1.3 of the FSAR indicates that the only Class I structures and components which are so located that they could be endangered by failure of Class III structures are the control building, main steam piping and feedwater piping, all of which could possibly be endangered by the Class III turbine building. It is further indicated there that no special provisions have been provided for protection except in the case of the main steam and feedwater lines up to the isolation valves, which are protected by the shield wall and the structural frame at the north end of the shield wall. Since these are located near the braced end of the turbine building, it is not anticipated by the applicant that there will be any structural failure in this area. Our judgment as to the adequacy of this aspect of the design is based on the statement given in the application. And, in this respect, the answer to Question 1.3 (Supplement 13) which describes the analysis and strengthening of the Turbine Building and Superheater Building for Indian Point Unit No. 1, and their ability to withstand the DBE, should give additional protection for the control room.

It is further stated that the only Class III crane whose failure could endanger any Class I function is the fuel storage building crane and that the failure of this crane will not impair a safe and orderly shutdown. The answer to Question 1.3 (Suppl. 13) indicates that the only potential for crane lift off will be in the unloaded condition with the trolley parked ar the support; the applicant advises that the unloaded crane will not be parked over the pool, so no hazard exists. It is also noted in the answer to Question 1.1.3 that the manipulator crane in the containment building, a Class III crane, is restrained from overturning and will not endanger Class I structures.

Deformation Criteria

The general stress criteria applicable to the seismic design are summarized in Appendix A of the FSAR. The statement given on page, A3 of Appendix A states that for all components, systems and structures classified as Class I, the primary steady state stresses, when combined with seismic stresses resulting from the response to the Design Basis Earthquake, are limited so that the function of the component system or structure shall not be impaired so as to prevent a safe and orderly shut-down of the plant.

We were advised at the time of our inspection of the plant in May 1969 that, for normal loadings plus the Operating Basis Earthquake, the intention was to use code allowables plus the 20 percent increase for transient conditions on Class I components and systems. For the Design Basis Earthquake and blow-down, basically the same criteria were used, although originally it had been planned to adopt higher allowables going into the plastic range using the code for faulted conditions. In actuality, as described in the answer to Question 1.7 of the FSAR, the allowable stresses in the case of the Design Basis Earthquake were limited to the yield point, or slightly below (see answer to Question 1.3 of Supplement 13).

The only references that we note where there was a calculation of stresses exceeding the yield point were at several places in the containment design report where it was mentioned that the calculations indicate that there could be possible local yielding of the liner under certain loading combinations, but that this would be limited and not be expected to be of a nature as to cause concern with regard to the integrity of the liner.

Reactor Internals

The mechanical design and evaluation of the reactor core and internals is described generally in Section 3.2.3 of the FSAR. From the discussion given it appears that the core support structure and core barrel have been designed with proper attention to support points and limitations of motions. The design criteria for the internals themselves, and specifically with reference to deflections under abnormal operation, are given in Table A.3-2 of the FSAR. These appear reasonable and should provide an adequate margin of safety.

Large Penetrations

A finite element analysis of the large penetrations in the containment vessel was made by the Franklin Institute and a description of the analysis and the results obtained is presented in the containment design report. Several analyses were made for different load combinations, and in addition a number of hand calculations were made to check the order of magnitude of the expected forces and stresses and to verify that the results were reasonable Our review of the material presented, to the extent possible, indicates that the penetration design is adequate.

Splices in Large Reinforcing of Bars

Cadweld splices were used in general in the construction of the containment vessel. We were advised that the early splices, about 10 percent of the total, were made with a bronze base, and the remaining 90 percent were made with ferritic base filler metal. Around the hatch opening, we observed that there was approximately a three foot stagger of adjacent splices, but in questioning we learned that there may not be such a stagger over other areas of the containment vessel. Lack of stagger of adjacent splices could lead to planes of weakness and cause cracking under conditions of over-loading. The pressure tests, however, will reveal any such cracking.

Approximately one in 200 splices was removed for test purposes. This is generally adequate.

Instrumentation and Controls

At the time of the May 1969 visit it was ascertained that the applicant considers the control room as a Class I structure and intends that the housing of it will also be subject to Class I requirements. However, the instrumentation for the control room as well as other instrumentation critical to containment and safe shutdown, has been purchased from the vendors according to applicant's specifications. The answer to Question 1.9 describes the vibration tests employed for selected items of essential equipment; the purpose of these tests is to help demonstrate that little or no difficulty will be expected in the operating characteristics thereof under seismic conditions. Although not absolute proof of acceptability, satisfactory test results certainly help to confirm the adequacy of such instrumentation and control items. Further information on the design and procurement approach for protection system equipment is given in the answer to Question 7.27 (Suppl. 13), and lends confirmation to the approach adopted.

Tornado Loadings

The information contained in Section 3.4 of the containment design report, and the answer to Question 5.7 of the FSAR indicates that the structure is designed for the usual wind loadings. The analyses described in Appendix B of Supplement 6, indicate that the containment building can resist the design tornado. What effect if any that a tornado could have on the control room or other critical facilities is not stated. However, the applicant states that

70

the siding of the control room can resist wind velocities up to 162 mph, and the girts (supporting the panels) will fail at 0.62 psi negative pressure; the building is protected by other buildings on the south and west.

Steel Liner and Containment Vessel

The analyses that have been carried out with regard to the liner are summarized in the FSAR and some additional information is presented in the containment design report. It is our understanding that where bulges of the liners occurred during construction, of less than 2 in., nothing was done to correct the bulges. However, when bulges were 2 in. or greater the liner was pushed back into a position of not more than 2 in. away from its intended position, and additional studs were used to anchor the liner in place. Temporary bracing was employed to hold it in position until the concrete was cast. Because of the foregoing, and since the temperature rise in the lower part of the structure in the liner is reduced by the use of insulating material, it is not expected that the departures from the intended original surface will lead to any difficulties.

Proof Test Procedures and Instrumentation

It is our understanding that a detailed description of the proof test procedures is to be submitted at a later date. At the time of our visit in May 1969 it was proposed by the applicant that strain readings be taken only on the liner around the penetrations. We suggested that additional readings be made which would include diameter changes of the penetrations and other measurements that can be made conveniently and without excessive expense to provide evidence that the design meets the design criteria. Fig. 5.13-4 suggests that such readings will be made. In any event, an interpretative report on the measurements that are taken should be provided and should be correlated with the calculations to provide evidence of validity of the design calculations.

Protection of Pipe Lines for Service Water

We were advised that pipelines for service water are embedded in the ground without any special protection. However, there appear to be alternate lines, although they are generally in the same location and/or trenches. In view of the foundation conditions surrounding the plant, and since there is no indication of previous fault motion or potential faulting, this design approach appears to be adequate. If redundancy in critical water supply is desired, it would be preferable to have separate water lines following independent routes.

Seismograph Installation

The answer to Question 1-1 of Supplement 3 indicates that one seismograph will be installed in the yard area, to provide further evidence of the extent of seismic excitation to which the plant might be subjected if an earthquake occurs. This is acceptable to us.

Containment Design Report

The containment design report, prepared for the applicant by Westinghouse Nuclear Energy Systems and United Engineers and Constructors, has proven to be helpful in arriving at an evaluation of many of the factors inherent in the design. The tables presented are useful in helping to arrive at decisions as to the adequacy of the design; we commend those responsible for the preparation of this summary type material.

We should like to encourage this type of approach to studies of the containment, structures, piping, equipment and other Class I items. We should like to urge that attention be given also to summaries and tabulation of the most important information, in terms of stresses and deformations, including the sources of the various stress components, how they were combined, and related discussion and explanatory material (including figures) which would lend itself to a much better basis for judgment as to the adequacy of design of nuclear facilities in general.

CONCLUDING REMARKS

On the basis of the information made available to us concerning the Class I structures, piping, reactor internals, and other Class I items, it is our belief that the plant possesses a reasonable margin of safety to meet the original design requirements, including the imposed Design Basis Earthquake loading conditions.

REFERENCES

1. "Final Facility Description and Safety Analysis Report -- Vols. I through V including Supplements 1, 2, 4, 5, 6, 7, 8 and 13," Indian Point Nuclear Generating Unit No. 2, Consolidated Edison Company of New York, Inc., AEC Docket No. 50-247, 1969 and 1970.

2. "Containment Design Report," for Indian Point Nuclear Generating Unit No. 2, Consolidated Edison Company of New York, Inc., prepared by Westinghouse Nuclear Energy Systems and United Engineers and Constructors, March 1969. (Labeled Final Draft)

3. "Adequacy of the Structural Criteria for Consolidated Edison Company of New York, Inc., Indian Point Nuclear Generating Unit No. 2," by N. M. Newmark and W. J. Hall, August 1966.

W. J. Hall

APPENDIX G

UNITED STATES
DEPARTMENT OF THE INTERIOR
OFFICE OF THE SECRETARY
WASHINGTON, D.C. 20240

OCT 16 1970

Dear Mr. Chairman:

Pursuant to Section 5 of Public Law 89-605 as amended and other authorizations, we are presenting the views of the Department of the Interior in the matter of the application by the Consolidated Edison Company for an operating license for Indian Point Nuclear Generating Unit No. 2, Buchanan, New York, AEC Docket No. 50-247 (Amendment No. 9). The following comments incorporate those submitted by the Federal Water Quality Administration, the Fish and Wildlife Service and the Bureau of Outdoor Recreation.

The unit under review is the second of three units completed or being constructed at the Indian Point site. We note that applications for construction permits for two more units to be located approximately one mile south of the Indian Point site were made in June 1969.

The Department of the Interior does not object to the issuance of the operating license to the Consolidated Edison Company for Unit No. 2 of the Indian Point Nuclear Power Plant. Our position is based upon the firm commitment by the Company as expressed in its responses to the Atomic Energy Commission that it will meet the water quality standards applicable to the receiving waters and that it will take whatever steps are necessary to mitigate any harmful effects that operation of the plant may have on the fishery resources of the Hudson River and tributary waters.

The Company should be commended for the cooperation it has extended to representatives of this Department during the course of our review. The studies which the Consolidated Edison Company is presently engaged in indicate the Company's concern for the potential damages to the environment that could result from operation of this unit and the others planned at and in the vicinity of Indian Point.

We are pleased to note that the Company has made provisions to open part of its land holdings for compatible public recreation use. We express the hope that the Company's public use plans will be finalized and fully implemented at the earliest possible time.

Consolidated Edison has initiated or participated in a number of studies to determine the effects of both radiological and thermal discharges from the Indian Point reactors upon both the temperature distribution and the aquatic life of the Hudson River through its consultants, Quirk, Lawler and Matusky Engineers, and the Alden Research Laboratories of Worcester Polytechnic Institute. The Company has conducted mathematical studies of the probable temperature in the River and has checked these estimates with hydraulic, model studies and actual field studies. In addition, Consolidated Edison has supported several independent but coordinated studies of the micro-organisms and aquatic life in the Hudson River and the probable effects of temperature and salinity changes upon them in the vicinity of the Indian Point Plant.

These studies are continuing and have been and will be helpful in assessing the effects of the Indian Point Unit No. 2 and of the other thermal plants which are proposed for construction on the shores of the Hudson River in the vicinity of Indian Point.

We have been provided information on plans for environmental monitoring of radiological and thermal releases proposed as a part of the operating license application. We understand that the plans for water quality monitoring, including radiological concentrations in the environment in microscopic and macroscopic aquatic life are acceptable to the State of New York. They appear reasonable and are considered generally acceptable to the Department of the Interior.

Through the monitoring programs the Company should have the necessary information to control its activities in a manner that will not violate applicable New York State as well as Federal water quality standards, recommendations of any enforcement conference or hearing board approved by the Secretary or order of any court under Section 10 of the Federal Water Pollution Control Act, and/or other State and Federal water pollution control regulations.

In view of the extensive and valuable fish and wildlife resources in the project area, it is imperative that every possible effort be made to safeguard these resources. Therefore, it is recommended that the Consolidated Edison Company be required to:

1. Continue to work closely with the Department of the Interior, New York State Department of Health, and other interested State and Federal agencies in developing plans for radiological surveys.

2. Conduct pre-operational radiological surveys as planned. These surveys should include but not be limited to the following:

   a. Gamma radioactivity analysis of water and sediment samples collected within 500 feet of the reactor effluent outfall.

   b. Beta and Gamma radioactivity analysis of selected plants and animals (including mollusks and crustaceans) collected as near the reactor effluent outfall as possible.

3. Prepare a report of the pre-operational radiological surveys and provide five copies to the Secretary of the Interior prior to project operation.

4. Conduct post-operational radiological surveys similar to that specified in recommendation (2) above, analyze the data, and prepare and submit reports every six months during reactor operation or until it has been conclusively demonstrated that no significant adverse conditions exist. Submit five copies of these reports to the Secretary of the Interior for distribution to appropriate State and Federal agencies for evaluation.

In addition to the above, the Atomic Energy Commission should urge the Consolidated Edison Company to:

1. Meet with the Department of the Interior, New York State Department of Environmental Conservation, New York State Department of Health, and other interested Federal and State agencies at frequent intervals to discuss new plans and evaluate results of the Company's ecological and engineering studies;

2. Conduct post-operational ecological surveys planned in cooperation with the above named agencies, analyze the data, prepare reports, and provide five copies of these reports to the Secretary of the Interior every six months or until the results indicate that no significant adverse conditions exist;

3. Construct, operate, and maintain fish protection facilities at the cooling water intake structure as needed to prevent significant losses of fish and other aquatic organisms; and

4. Modify project structures and operations including the addition of facilities for cooling discharge waters and reducing concentrations of harmful chemicals and other substances as may be determined necessary.

We appreciate the opportunity to provide these comments.

Sincerely yours,

Walter J. Hickel

Secretary of the Interior

Honorable Glenn T. Seaborg
Chairman, United States
Atomic Energy Commission
Washington, D. C. 20545

APPENDIX H
CONSOLIDATED EDISON COMPANY OF NEW YORK
DOCKET NO. 50-247
FINANCIAL ANALYSIS

	(dollars in millions) Calendar Year Ended Dec. 31		
	1969	1968	1965
Long-term debt	$1,981.6	$1,901.6	$1,711.0
Utility plant (net)	3,793.3	3,583.6	3,169.5
Ratio - debt to fixed plant	.52	.53	.54
Utility plant (net)	3,793.3	3,583.6	3,169.5
Capitalization	3,818.4	3,667.6	3,228.1
Ratio - net plant to capitalization	.99	.98	.98
Stockholders' equity	1,836.7	1,766.0	1,517.1
Total assets	4,069.6	3,845.4	3,387.0
Proprietary ratio	.45	.46	.45
Earnings available to common equity	93.1	95.7	89.9
Common equity	1,210.2	1,139.0	1,072.1
Rate of return on common equity	7.7%	8.4%	8.4%
Net income	127.2	128.5	111.8
Stockholders' equity	1,836.7	1,766.0	1,517.1
Rate of return on stockholders' equity	6.9%	7.3%	7.4%
Net income before interest	198.0	193.9	168.4
Liabilities and capital	4,069.6	3,845.4	3,387.0
Rate of return on total investment	4.9%	5.0%	5.0%
Net income before interest	198.0	193.9	168.4
Interest on long-term debt	84.3	77.0	62.7
No. of times fixed charges earned	2.3	2.5	2.7
Net income	127.2	128.5	111.8
Total revenue	1,028.3	982.3	840.2
Net income ratio	.124	.131	.133
Operating expenses (incl. taxes)	830.5	788.3	668.6
Operating revenues	1,028.3	982.3	840.2
Operating ratio	.81	.80	.80
Retained earnings	426.1	400.9	321.7
Earnings per share of common	$2.47	$2.57	$2.42

	1969		1968	
Capitalization at 12/31	Amount	% of Total	Amount	% of Total
Long-term debt	$1,981.6	51.[illegible]%	$1,901.6	51.9%
Preferred stock	626.6	16.4	627.0	17.1
[illegible] stock	1,210.2	31.7	1,139.0	31.0
	$3,818.4	100.0%	$3,667.6	100.0%

Moody's Bond Ratings:	
First Mortgage Bonds	A
Dun and Bradstreet Credit Rating	AaA1