Docket file

February 20, 1969

SAFETY EVALUATION

BY THE

DIVISION OF REACTOR LICENSING

U. S. ATOMIC ENERGY COMMISSION

IN THE MATTER OF

CONSOLIDATED EDISON COMPANY OF NEW YORK, INC.

INDIAN POINT NUCLEAR GENERATING UNIT NO. 3

DOCKET NO. 50-286

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1.0 INTRODUCTION

The Consolidated Edison Company of New York, Inc. (applicant) by application dated April 26, 1967, requested a license to construct and operate a nuclear power unit designated Indian Point Nuclear Generating Unit No. 3 on its Indian Point site located on the east bank of the Hudson River in upper Westchester County, New York. The proposed unit will employ a pressurized water reactor nuclear steam supply system designed and furnished by Westinghouse Electric Corporation, the prime contractor. Westinghouse has engaged United Engineers and Constructors to serve as the architect-engineer.

The design power rating of the Unit No. 3 is 3025 megawatts thermal (Mwt) with an ultimate capacity of 3217 Mwt. These are equivalent to net electrical ratings of 965 and 1033 megawatts electrical (Mwe), respectively.

The essential elements of the design of Unit 3 are similar to those of Unit No. 2 as well as those of other recently licensed pressurized water nuclear plants. Thus, the reactor, containment, safety features, instrumentation, and auxiliary system designs have been subject to continuing review by the regulatory staff. No element of the design of Unit No. 3 is inconsistent with our safety criteria or applicable regulations.

The technical safety review of the proposed plant by the Atomic Energy Commission's regulatory staff has been based on the applicant's Preliminary Safety Analysis Report (PSAR) and nine supplements, all of which are contained in the application.

In the course of the review of the material submitted, we held a number of meetings with representatives of the applicant to discuss the proposed plant. As a consequence, we requested additional information of the applicant which was provided in the supplements. A chronology of our review is attached as Appendix A to this report.

The Commission's Advisory Committee on Reactor Safeguards (ACRS) has also conducted a review of the application and has met with both the applicant and the staff. A copy of its report to the Commission on the Indian Point Nuclear Generating Unit No. 3 is included as Appendix B.

Our technical evaluation of the preliminary design of the proposed plant was accomplished with the assistance of consultants. Appendices C through H include the reports of our consultants on meteorology, geology and hydrology, seismology, flooding potential, radiological monitoring, and structural design.

On July 23, 1968, the applicant submitted a request for an exemption to the requirements of 10 CFR 50.10(b) of the Commission's regulations which would permit (1) pouring the base mat concrete up to the bottom liner plant, (2) installation of the bottom liner plates and transition knuckle plates, and (3) installation of the rebar for the base concrete over the bottom liner plates. The design of the applicable portions of the containment was evaluated and an exemption was granted on November 15, 1968, authorizing performance of the work requested. The granting of this exemption did not imply any commitment for the issuance of a construction permit.

This review of Unit No. 3 for a construction permit is the first stage of our continuing review of the design, construction, and operating features of the unit. Prior to issuance of operating licenses, we will review the final design to determine that all of the Commission's safety requirements have been met. The unit would then be operated only in accordance with the terms of the operating licenses and the Commission's regulations under our continued surveillance.

The issues to be considered, and on which findings must be made by an Atomic Safety and Licensing Board before the requested construction permit may be issued, are set forth in the Notice of Hearing published in Federal Register, 34 F.R. 1741, February 5, 1969.

2.0 SITE

2.1 Description

Unit No. 3 is to be located on the applicant's 235-acre Indian Point site in upper Westchester County, New York, approximately 24 miles north of New York City. The unit will be built adjacent to and south of the presently licensed Unit No. 1. Unit No. 2, which is presently under construction, is located adjacent to and north of Unit No. 1.

2.2 Population Distribution

The population distribution in the vicinity of the Indian Point site is presented below.

CUMULATIVE POPULATION

Distance (Miles)	Indian Point 1960	Indian Point 1980
1	1,080	2,100
2	10,810	20,900
3	29,630	59,520
4	38,730	78,800
5	53,040	108,060
10	155,510	312,640

The Commission's Reactor Site Criteria, 10 CFR Part 100, provide guidelines for the offsite doses under postulated accident conditions at the minimum exclusion distance and the low population distance. The guidelines also state that the distance to the nearest boundary of the closest population center should be at least 1-1/3 times the distance from the reactor to the outer boundary of the low population zone, considering population distribution within the population center.

For Indian Point Unit No. 3, we have determined that the minimum exclusion distance is 0.22 miles (350 meters) to the southwest. The nearest boundary of Peekskill, the population center, is 0.63 miles to the northeast; however, the nearest residential area of Peekskill is 0.85 miles to the east. The applicant selected a low population zone having an outer boundary of 0.67 miles (1100 meters). On the basis that (1) the population within the proposed low population zone is small (66 people) and (2) the area of Peekskill in the vicinity of the plant is of a general industrial nature, we have concurred in the applicant's selection.

We have evaluated the radiological consequences of various postulated accidents at the outer boundary of the low population zone and at the exclusion area boundary. These are discussed in Section 5.0 of this report. We have concluded that the proposed Unit No. 3 meets the exposure guidelines specified in 10 CFR Part 100.

2.3 Meteorology

The meteorology of the Indian Point site is governed by its position in a deep river valley. Consequently, wind direction generally follows a pronounced diurnal cycle with unstable (lapse) flow in the upriver direction during the daytime and stable flow in the downriver direction at night. This general meteorological condition results in a reduction of the probability of the wind persisting in a single direction for a period in excess of twelve hours.
Since the applicant does not have long term data available from the site on the specific joint frequency of stability-wind speed-wind direction persistence, we have used our standard meteorological model for accident dose calculations which in our judgment conservatively characterizes the meteorology of the Indian Point site. With this model we assume a one meter per second wind speed in one direction under inversion conditions for a period of 8 hours; wind speed in the same direction with meandering of the plume centerline over a 22-1/2° sector under inversion conditions for the remainder of the first 24 hour period; and variable stability, wind direction, and

wind speed for the remainder of the accident. Our consultant, Air Resources Laboratory, ESSA, concurs in these assumptions. Its report is attached as Appendix C.

2.4 Geology and Seismology

Unit No. 3 will be founded on a hard limestone that is well jointed but noncavernous. We have reviewed the analysis of the site geology in the PSAR and examined the boring logs, as has our geological consultant, the U. S. Geological Survey (USGS). The USGS report is attached as Appendix D. As a result of this evaluation, we have concluded that the applicant's analysis presents an adequate appraisal of site geology, and there are no known active faults or other geologic structures that could be expected to localize earthquakes in the immediate vicinity of the site.

The seismicity of the site has been evaluated by the U. S. Coast and Geodetic Survey (USC&GS). Its report is attached as Appendix E. Based on the review of the seismic history of the site and of the related geologic considerations, the USC&GS concludes that the applicant's proposal to use accelerations of 0.10g for the Operational Basis Earthquake and 0.15g for the Design Basis Earthquake is acceptable.

These same conclusions were reached by the USC&GS during our evaluation of Indian Point Unit 2.

2.5 Hydrology and Flooding

The Indian Point site is located on the east bank of the Hudson River below Peekskill, New York. River water in the vicinity of the plant is used only for industrial cooling purposes. The nearest community utilizing the river for a public water supply is Poughkeepsie, 30 miles upstream of the site. The Chelsea pumping station, 22 miles upstream, can be used as a supplementary source of water supply for New York City. Radionuclides released to the river during the course of normal operation of Unit No. 3 would be moved both upstream and downstream by tidal action and many orders of magnitude dilution would occur prior to reaching Chelsea or Poughkeepsie. Further, should an accidental release of radioactive material to the river occur, the long transit time brought about by the tidal action would allow ample time for monitoring the movement of the radioactivity and to take corrective action should it be necessary. Thus, we envision no potential hazard to these drinking water supplies.

Flooding at the site has been evaluated for (1) the fresh water flood resulting from precipitation runoff, the breaching of five major dams upstream of the site, and ebb tide flow and (2) the storm surge associated with the occurrence of the Probable Maximum Hurricane in the vicinity of the site. The most severe condition has been found to be a hurricane storm surge of 19.3 ft above mean sea level (MSL) which coincides with the design water level proposed by the applicant. The applicant's analyses of fresh water flood level have been reviewed by the U. S. Geological Survey (USGS), and the hurricane storm surge calculations have been reviewed by the U. S. Army Coastal Engineering Research Center (CERC). The USGS report is attached as Appendix F. The reports of CERC and USGS confirm our conclusion that the proposed design water level of +19.3 ft MSL is an acceptable value.

2.6 Environmental Considerations

The Fish and Wildlife Service (F&WL) has reviewed the application relative to the consequences of release of radioactive waste materials to the environs. It has recommended that both pre- and post-operational surveys, planned in cooperation with the appropriate Federal and State agencies, be conducted. Their comments are attached as Appendix G. The applicant has agreed to comply with the F&WL recommendations.

The applicant is conducting an environmental monitoring program which includes sampling of: atmospheric dust; waters of the Hudson River, a small lake onsite, nearby reservoirs, and the onsite well;

vegetation; atmospheric gross gamma activity; and marine life in the Hudson River. This program has been in operation since 1958, and in our judgment it is adequate to determine the impact of the Unit No. 3 facility on the environment. To date this program has demonstrated that Indian Point Unit No. 1 has had no adverse effect on the environment.

3.0 DISCUSSION AND DESCRIPTION OF PRINCIPAL PLANT FEATURES

3.1 Nuclear Steam Supply System

The nuclear steam supply system for Unit No. 3 consists of a light-water-moderated pressurized water reactor (PWR) which transfers reactor heat to four steam generators via four reactor coolant loops. The basic design is similar to that of other Westinghouse reactors now under construction.

The fuel for the reactor is low enrichment UO_2 pellets sealed within 12-foot-long Zircaloy tubes. Two-hundred-four fuel rods are arranged in a square array to form a fuel assembly. The reactor core contains 193 fuel assemblies which rest on the lower core plate.

The proposed power level of Unit No. 3 is approximately 10% higher than that of Unit No. 2; however, it is approximately 7% lower than that of the Diablo Canyon plant and the recent generation of four-loop Westinghouse-designed plants. A comparison of Unit No. 3 with Unit No. 2 and with Diablo Canyon is presented in Table 3.1.

Our evaluation of the thermal design of the core has included consideration of the results of parametric studies of the effects on the minimum DNB ratio of variations in inlet temperature, inlet pressure, mass flow rate, and peaking factors. These studies have demonstrated that neither calibration errors nor errors in the predicted peaking factors will significantly affect the thermal performance of the core. On these bases, we have concluded that the thermal design of the core is conservative.

TABLE 3.1

Item	Indian Point 3	Diablo Canyon	Indian Point 2
Total Heat Generation, Mwt	3025	3250	2758
Maximum Specific Power, kw/ft	17.6	18.9	18.5
Maximum Heat Flux, Btu/hr-ft^2	543,000	583,000	570,800
Average Heat Flux, Btu/hr-ft^2	193,000	207,000	175,600
Average Mass Velocity lb/hr-ft^2	2.53×10^6	2.54×10^6	2.56×10^6
Nominal Inlet Temperature, °F	549.7	539	543
Minimum DNBR at Nominal Conditions	1.82	1.81	1.81
Fq - Heat Flux hot channel factor	2.82	2.82	3.25
F H - Enthalpy hot channel factor	1.70	1.70	1.88
Fuel Enrichments, w/o			
Region 1	2.1	2.2	2.23
Checkerboard Region	2.6 3.2	2.7 3.3	2.38 2.68

Reactivity control is accomplished by full length silver-indium-cadmium control rod assemblies, part length silver-indium-cadmium control rod assemblies, fixed burnable poison rods (borosilicate glass in stainless steel tubes) and by liquid poison (boric acid) in the reactor coolant.

For relatively large cores such as that of Unit 3, means are required to detect the power distribution within the core. The applicant contends that the combination of the four external flux monitors, the traveling in-core monitors (six traveling flux probes which together may traverse any of 58 thimble locations in the core) and the flow channel exit temperature detectors will adequately detect flux patterns. Our position in this regard continues to be that information from in-core monitors must be provided to an operator so that he may detect flux patterns and position the part length rods for proper axial power shaping, unless, at some later date, experience shows that the external monitors can detect in-core anomalies with adequate sensitivity. The applicant has stated that provision will be made for installation of permanent in-core detectors should they be required. Further detail on the research and development program related to detection of core flux patterns is given in Section 6.1.

Section III of the ASME Pressure Vessel Code will be used to design the reactor vessel, pressurizer, coolant pump casings, and the steam generators. To provide access for inspection, the vessel and its internals will be constructed so as to permit removal of the internals during plant life. A complete stress analysis which reflects consideration of all design loadings detailed in the design specification will be prepared by the manufacturer to assure compliance with the stress limits of Section III for the reactor vessel, steam generators, pressurizer, and pump casings. Westinghouse will independently review these stress analyses. The reactor coolant piping design will be analyzed in accordance with the requirements of USA S.I. B31.1 Code for Pressure Piping. A similar analysis of the piping will be prepared by or for Westinghouse by a qualified piping analysis contractor.

The reactor coolant system, the reactor internals and all other Class I (seismic) mechanical systems, will be designed to withstand normal design loads of mechanical, hydraulic, and thermal origin plus operational basis earthquake loads within yield. In addition, these Class I systems and components will be designed to withstand the concurrent blowdown loads and design basis earthquake loads. These criteria and stress analysis evaluations are the same as those we have reviewed and approved in previous cases and, as before, we conclude that they are acceptable for the Nuclear Steam Supply System proposed for Unit 3.

3.2 Containment Structural Design

The foundation material at the site from the surface down consists of a finegrained phyllite, a schist, and limestone, with bedrock lying very close to the surface. Unit No. 3 will be located on the limestone, which is fractured and jointed, making it permeable to ground water, but it is hard, not cavernous, and can sustain up to 50 tons per square foot. It is, therefore, a capable foundation material for this facility.

The design and structural analysis of the Unit 3 containment structure is similar to that of Unit No. 2. It is a reinforced concrete vertical right cylinder with a flat base and hemispherical dome, an internal diameter of 135 feet, a height from base to dome springline of 148 feet, 4'-6" thick cylinder walls and 3'-6" thick dome. The base mat is 9 feet thick, supported on rock. The containment free volume is 2,610,000 cubic feet. The design pressure is 47 psig.

The reinforcing in the structure will have an elastic response to all loads with limited maximum strains to ensure the integrity of the liner. The reinforcing steel will conform to ASTM Designation A432-65 with a guaranteed minimum yield point of 60,000 psi. The 14S and 18S reinforcing bars will be spliced by Cadweld splices. Test splices will be production splices removed from the structure. Diagonal reinforcing will be utilized in addition to the horizontal and vertical cylinder reinforcing to handle the shear loads generated by earthquake or wind.

The containment liner will be carbon steel plate conforming to ASTM Designation A442-65, Grade 60. It will be 1/4-inch thick at the bottom, 1/2-inch thick in the first three courses (except 3/4-inch thick at penetrations), and 3/8-inch thick for the remaining portion of the cylindrical walls. The dome liner will be 1/2-inch thick. The liner nil ductility transition temperature will be 30°F lower than the minimum operating temperature of the liner material. The anchorage system for attaching the liner to the concrete consists of 1/2-inch diameter bent welding studs spaced such that the liner does not experience stresses in excess of critical buckling stress for various postulated accidents even if an anchor is missing or has failed. Liner insulation will be provided at the lower portion of the containment to minimize thermal stresses.

The containment structure has been designed to accommodate an earthquake acceleration of 0.1g with structural loads within yield stress (Operating Basis Earthquake). The structure will also remain functional in the event of an earthquake acceleration of 0.15g (Design Basis Earthquake). Further, as discussed in Section 4.1, the containment structure has been designed to withstand the effects of a 300 mph tornado wind with 60 mph transverse velocity, a pressure drop of 3 psi in 3 seconds, and associated missiles.

We have evaluated the preliminary design and design criteria of the containment structure for liner integrity and stability when subjected to various postulated accidents, as well as seismic and tornadic events. We conclude that the design criteria of the containment structure meet our safety requirements in all respects.

The containment vessel is designed to have negligible outleakage under accident conditions. To meet this objective, the containment vessel is provided with a penetration pressurization system and an isolation valve seal-water system. These preclude leakage in the containment seam welds, electrical and piping penetrations, personnel air locks, ventilation purge duct penetrations, equipment door flange, spent fuel transfer tube and the piping which enters containment by interposing a pressurized volume between the containment atmosphere and the environs. Even if both the penetration pressurization system and the isolation valve seal water system should fail, containment leakage would be limited to 0.1% of the containment free volume per day at the peak post-accident pressure which is the leak rate we have assumed in our evaluation of the radiological consequences of the design basis accident in Section 5.3.

Our structural consultant, Nathan M. Newmark Consulting Engineering Services, found the design of the containment structure to be acceptable. The report of our consultant is attached as Appendix H.

3.3 Instrumentation and Control

The Commission's proposed General Design Criteria and the Proposed IEEE Criteria for Nuclear Power Plant Protection Systems (No. IEEE-279, dated August 28, 1968) have been used, where applicable, as our bases for judging the adequacy of the Instrumentation and Control Systems. In summary, these criteria include consideration of channel redundancy, testability, independence, bypass provisions, and manual trip initiation.

To meet these requirements, the reactor protection system is designed on a channelized basis to achieve separation between redundant channels. Separation of redundant analog channels originates at the process sensors and continues through the field wiring and containment penetrations to the protection instrumentation racks. Each channel is energized from a separate a.c. power source. We have reviewed the reactor protection system and have determined that it provides adequate redundancy and is testable. Redundancy results from the multiplicity of instrument and logic channels, any one of which can be lost without a consequent loss of protective function. Testability results from the design of the trip systems. A manual trip circuit is installed downstream of (and independent of) the instrument channels, and directly de-energizes the trip breakers. Access to bypass circuits (for testing) is administratively controlled by use of annunciators that alarm in the control room when the door to a bypass panel is opened. All bypasses are indicated in the control room.

On the basis of our review, we have concluded that the reactor protection system meets the stated criteria.

The instrumentation and controls for the engineered safety features have been analyzed to assure that they are designed in accordance with IEEE-279. Our review of the detailed design of the instrumentation for the engineered safety feature systems will continue as the design progresses. The applicant has stated that the design will conform to

IEEE-279. Our review indicates that Westinghouse is proceeding satisfactorily in accomplishing this objective.

We are pursuing with Westinghouse, the applicant's instrumentation supplier, the concern expressed by the ACRS with respect to possible systematic failures. Our objective is a suitable balance of design objectives in regard to functional and equipment diversity, interaction of protection and control functions, testing, and surveillance to achieve a protection system design that has adequate capability to cope with systematic failure modes as well as random failure modes. Our evaluation of systematic failures will be completed prior to the final installation of this equipment at Unit No. 3.

Environmental qualification tests are being performed on vital fan cooler motors, valve motors, and power and instrument cables located in containment. These tests are performed to assure that the combined effects of design pressure, temperature, and humidity upon the devices under test will not result in loss of function. The effect of chemical sprays is also being considered.

Based on the foregoing, we conclude that the reactor instrumentation and control system conforms to IEEE-279 and to the applicable provisions of the Commission's Proposed General Design Criteria and is acceptable.

3.4 Emergency Power

We have used proposed General Design Criterion No. 39 as the basis for judging the adequacy of the Emergency Power System.

3.4.1 Offsite Emergency Power

The Indian Point Unit No. 3 station startup transformer (138/6.9 kV) is normally supplied from a 138 kV line from the Buchanan substation. The second independent offsite power supply to Unit No. 3 is the 6.9 kV connection to the vital buses. This supply is automatically connected upon loss of the normal 138 kV supply, and can be fed from any one of three separate sources: (1) the station gas-turbine generator, (2) the underground 13 kV feeder from Buchanan substation, or (3) the auxiliary bus of Unit No. 2.

The Buchanan substation has a tie-line to the P.J.M. electrical grid and two 345 kV lines to the Millwood switching station. Millwood, in turn, is connected to the Niagara Mohawk and Connecticut Light and Power grids.

Based on the foregoing, we conclude that because of the multiplicity of power sources, in conjunction with the alternate 6.9 kV feeder in the event the startup transformer is lost, the offsite portion of the emergency power system is acceptable.

3.4.2 Onsite Emergency Power for Unit No. 3

There are four 480 V emergency buses energized directly, when required, from the three diesel generator units. Two diesel generators are required to furnish engineered safety feature loads. The diesel generators will not be synchronized during emergency operations.

If all three generators start, the "three-generator" loading sequence is followed automatically. If one generator does not start, the appropriate tie breakers are closed automatically and the failed buses will be energized by one of the remaining two generators. The control system then automatically selects the "two-generator" loading sequence. If any motor does not then start, the system will attempt to connect a redundant counterpart which was omitted from this loading sequence because of power limitations.

In our judgment, the design of the onsite power sources is susceptible to systematic failures arising from the automatic interconnection of redundant buses. On this basis, we have concluded that the diesel generator system should provide a greater independence of the buses than is provided in the proposed system, at least to the extent that the diesels cannot be connected together with automatically operated devices. The Advisory Committee on Reactor Safeguards (ACRS) shares this opinion. During construction of Unit No. 3, we will continue to review this area to assure that greater independence is provided, as recommended by the ACRS.

We conclude that the five-day supply of fuel oil, onsite, is adequate in view of the immediate availability of fuel oil supplies locally.

The diesel generators will be housed in separate rooms in a "tornado-proof" structure. The structure will be provided with internal walls to provide physical isolation. The remaining components of the emergency power system are either underground or housed in tornado-proof structures. These design precautions are acceptable, and are further discussed in Section 4.1.

The d.c. system consists of two batteries supplying two separate independent buses. The two buses are normally separated with a non-automatic tie breaker. Essential d.c. supply circuits are redundant with feeds from each bus and all d.c. circuits are separately protected by circuit breakers at their respective d.c. bus. The batteries are located in separate rooms, and can supply essential loads for two hours without assistance from their respective battery chargers. We conclude that the applicant's proposed design of the d.c. system has sufficient redundancy and independence and conforms to the requirements of Criterion No. 39.

3.4.3 Cable Routing and Loading

We have evaluated the applicant's criteria relating to the internal routing of instrument and power cables, cable tray loading, and over-current protection. Instrument and power cables will be separated by channels with separation or fire barriers provided between cables. Connecting tubing between pressure sensing locations and transmitters will be physically protected and separated to prevent common failures due to mechanical damage. The transmitters will be located in structural steel racks such that they are separated by a steel plate barrier.

Electrical loading and heat dissipation of cables on ladder trays will be carefully studied and controlled to ensure no excess heating. Insulated Power Cable Engineers Association standards and manufacturers' recommendations will be followed.

Control room instrumentation is installed on a channelized basis with redundant instrumentation in separate cabinets or racks. Physical protection is afforded by space (aisles), metal barriers, or other equipment.

We have concluded that these criteria assure that adequate precautions will be taken against fire and other common failure modes.

3.5 Radioactive Waste Disposal System

Small quantities of radioactive waste products are generated in the normal operation of a nuclear power plant. The liquid waste is collected, stored, treated, and either re-used or discharged. The liquid waste disposal systems for all three reactors are designed to meet 10 CFR Part 20 discharge limits. The effluent will be continuously monitored. High activity will cause the liquid effluent flow control valve to close, thus terminating release of liquid effluent. To date, operating experience with pressurized water reactor plants indicates that the liquid effluent discharge is only a small fraction of that specified in 10 CFR Part 20.

Gaseous radioactive waste is collected, compressed, and stored until sufficient radioactive decay has occurred to permit discharge via the plant vent within the requirements of 10 CFR Part 20. A monitor is installed in the plant ventilation discharge duct. Radioactive releases will be monitored and discharge of gaseous effluent will be automatically terminated, if a concentration in excess of the limits in the technical specifications is reached.

We have examined the design of the system and have determined that it is adequately sized and provides automatic means of terminating releases of high levels of radioactivity. Technical specifications will be developed to limit the concentrations of radioactive materials released from all three reactors to those limits specified in 10 CFR Part 20. In establishing release limits, we will impose additional restrictions upon the release of radionuclides in accordance with the provisions of 10 CFR Part 20.106(e) if it appears that the daily intake of radioactive material from air, water, or food by a suitable sample of an exposed population group, averaged over a period not exceeding one year, would otherwise exceed the daily intake resulting from continuous exposure to air or water containing one-third the concentration of radioactive materials specified in Appendix B, Table II of 10 CFR Part 20.

4.0 IMPORTANT SAFETY CONSIDERATIONS

4.1 Tornado Considerations

The control building, diesel generator building, primary auxiliary building, containment, and all connecting ducting for essential cabling and piping are designed to withstand tornado wind loadings corresponding to 300 mph tangential velocities, transverse velocities of 60 mph and a differential pressure drop of 3 psi in 3 seconds with no loss of function. These structures are also designed to withstand the impact of the following postulated missiles:

1. 4 in. x 12 in. x 12 ft. plank at 300 mph.
2. 4000 lb passenger car at 50 mph not exceeding 25 ft. above the ground.

The Foregoing are consistent with criteria found acceptable for previously licensed plants. In addition, the following general criteria have been adopted by the applicant relative to tornado considerations:

1. A tornado will not cause a loss-of-coolant accident.
2. A tornado will not impair the ability to safely shut down the plant.
3. A tornado following a loss-of-coolant accident will not impair the long term safety of the plant.

The tornado protection criteria outlined above are met by the protection provided by the facility structures with three exceptions summarized below. For these systems, reliance is placed on redundant components and/or redundant sources of water rather than on structural protection.

1. Emergency feedwater for the steam generators is supplied from redundant water supplies. The normal source of feedwater is the secondary feed circuit which requires operation of the main condenser, air ejector, and service water system. For an alternate water supply, the feedwater pumps can take suction from the condensate storage tanks, the city water storage tanks, or can be connected directly to the city water supply. Thus, even if the normal feedwater train is disabled, three additional feedwater sources are available to provide water to the steam generator feedwater pumps to permit dissipation of decay heat.
2. The makeup water for the primary system can be obtained from either the primary storage tank or the refueling water storage tank. In addition, limited makeup can be achieved using the volume control tank, boric acid tanks, and the monitor tanks.
3. Service water supply relies on the redundancy provided by the two supply lines, four screens and six pumps. Two pumps, one screen, and one supply line are required for prolonged shutdown.

The effect of tornadoes on the spent fuel pit is being evaluated by the applicant. It has stated the pit will be designed such that a cover can be added later if it cannot be demonstrated that a tornado has an insignificant effect on the fuel in the pit.

We have examined the structural design criteria for tornado protection and the general criteria for the plant proposed by the applicant and consider them to be acceptable. Since considerable time is available for action if one or more of the redundant components are lost, and in view of the physical location of redundant features, we conclude that the redundancy supplied in lieu of structural protection is acceptable.

4.2 Emergency Core Cooling System

The emergency core cooling system (ECCS) for Unit No. 3 is similar to that proposed for recent four-loop pressurized water reactors. This ECCS consists of (1) one high pressure coolant injection and recirculation subsystem (HPS), (2) one low pressure coolant injection and recirculation subsystem (RHRS), (3) one low pressure coolant recirculation subsystem (LPS) located entirely within containment, and (4) one accumulator subsystem.

The three pumps of the HPS are normally aligned to a common suction header which is fed by the refueling water storage tank. In addition, the suction of all high head pumps can be remotely realigned to the discharge of the low head subsystems. The three high-head pumps discharge to a header which feeds injection lines to the hot legs of reactor coolant loops 1 and 3, and injection lines to the cold legs of reactor coolant loops 2 and 4. Two high-head pumps have sufficient capacity to accommodate spillage from one of these four injection lines.

The two RHRS pumps take suction from the refueling water storage tank for short term coolant injection and from the containment sump for long term coolant recirculation. These pumps discharge through a common line to the two residual heat exchangers and then to the primary system by four cold leg injection lines.

The LPS contains two low head recirculation pumps which are located within containment. They take suction from a recirculation sump and discharge to either of the two residual heat exchangers. Any one of four low pressure pumps (two each in the RHRS and in the LPS) is capable of supplying the required post-accident recirculation flow to the core.

The four accumulators discharge through the low pressure, cold leg injection lines, and the accumulators are sized on the basis that one of the four spills through a break. For breaks larger than about 6 inches in diameter, the accumulator subsystem is the only subsystem which can reflood the core in time to adequately limit clad temperature, oxidation, and deformation. This single subsystem is acceptable because (1) it stores the energy required for operation, (2) it requires no external controls or signals for operation, and (3) it has sufficient capacity to accommodate anticipated spillage and core flow bypass.

The Unit No. 3 low pressure recirculation subsystem (LPS) is located entirely within the containment building. Any leakage of radioactivity from this system would not be released to the environs. The boron injection tank is located in-line, downstream from the high pressure safety injection pumps. This will result in rapid injection of poison into the core. This poison will reduce the reactivity of the core and will ameliorate the consequences of accidents which cause rapid primary system cooldown; e.g., rupture of a steam line.

Our evaluation of the design and operating characteristics of the proposed ECCS has included a failure mode analysis. This analysis has shown that the ECCS is designed to provide coolant injection even if a single active component fails to operate. Our analysis has also shown that long term core cooling, requiring circulation of coolant from the containment sump, through heat exchangers, to the core will be accomplished even in the event of the failure of any component, either active or passive.

We have also evaluated the applicant's analyses of ECCS performance which have used computer codes developed by Westinghouse and assume only onsite emergency power is available. The applicant presented the results of blowdown and core heatup analyses for the double-ended, 6 ft^2 3 ft^2, and 0.5 ft^2 breaks in the cold leg and in the hot leg of one of the reactor coolant loops. These results indicate that the cold leg breaks result in higher peak clad temperatures than hot leg breaks of corresponding size because of core flow reversals during blowdown, steam binding above the core during accumulator injection, and spillage of one accumulator. All of these effects are incorporated into the Westinghouse computer codes.

The peak clad temperatures conservatively calculated for these breaks are well below the Zircaloy melting temperature and are below the temperature range accelerated clad-water chemical reaction (2200°F). The total cladwater reaction calculated for each of the breaks is much less than 1 percent of the total fuel clad mass. Furthermore, the clad temperature calculations reported by the applicant show that the clad hot spot is above 1800°F for only about 50 seconds and that only about 2 percent of the total clad volume exceeds a temperature of 1800°F for the double-ended cold leg break. On the basis of data from Argonne National Laboratory which indicate that longer periods at higher temperatures are required to cause Zircaloy clad embrittlement by oxidation, we conclude that the clad heat transfer geometry will not be significantly altered by thermal shock upon quenching.

Our evaluation of the design of the proposed ECCS has led to the conclusion that the ECCS (1) limits the peak clad temperature to well below the clad melting temperature, (2) limits the clad-water reaction to less than one percent of the total clad mass, (3) terminates the temperature transient before the core geometry necessary for core cooling is lost and before the clad is so embrittled as to fail upon quenching, and (4) reduces the core temperature and removes core heat for an extended period of time.

Research and development concerning clad deformation and its effect on core cooling under simulated loss-of-coolant conditions is being conducted. This research and development program is discussed in detail in Section 6.3.

4.3 Thermal Shock and Post-Loss-of-Coolant Accident Protection

The applicant's steam supplier, Westinghouse, has analyzed the thermal transient and resulting thermally induced stresses experienced by the hot reactor vessel wall when deluged with cold safety injection water following a loss-of-coolant accident. A discussion of the R&D program on this subject is presented in Section 6.7. Initial analytical results indicate that no loss of vessel integrity would occur even if flaws were presumed to exist in the vessel wall at the time of safety injection. There are some uncertainties, however, in the analytical method, and for this reason the applicant will make provisions in the design and layout of Unit No. 3 to enable installation of additional equipment to mitigate the consequences of a post-loss-of-coolant accident reactor vessel failure, if further analysis of the thermal shock experienced by the vessel during safety injection indicates that such protection should be required.

The proposed post-loss-of-coolant accident protection (PLOCAP) system would direct the low head injection flow and the subsequent recirculation flow to the hot legs of the coolant loops to provide top injection for the core. In addition, a fast acting flooding system would be provided for the reactor vessel cavity. Valves, which open upon receipt of signals from both safety injection initiation and accumulator low pressure, would permit cavity flood tanks to drain to the cavity, raising the level of water to just below the bottom of the reactor vessel. This level is specified to prevent damage to the pressure vessel in the event of inadvertent opening of the cavity flood tank valves. Cavity flooding would be completed by valving the discharge of the recirculation pumps to the cavity.

To permit installation of the PLOCAP system at a later date should it prove necessary, the following provisions will be incorporated into the design:

1. A standpipe will be installed over the incore instrumentation passageway to permit the retention of water in the cavity to the level of the core without flooding the floor of the containment.
2. Nozzles will be installed on each hot leg pipe to permit installation of a hot inlet injection system.
3. A second containment sump line will be installed to accommodate the high recirculation flow rates required to rapidly raise the cavity liquid level.
4. Space will be reserved in the primary auxiliary building for increased heat exchange and pumping capability.
5. Space will be reserved in the containment vessel for the cavity flood tanks and associated piping.
6. Detailed pipe layouts and plant arrangements will be developed considering the extra pipework and containment penetrations required by PLOCAP.

We have evaluated the preliminary design criteria for the PLOCAP system considering the consequences of inadvertent operation of the system and of initiation of the system following a small break in the primary system which causes a slow depressurization of the primary system.

We have concluded that the PLOCAP provisions are adequate to give reasonable assurance that a reliable system can be installed to mitigate the consequences of a reactor vessel failure following a loss-of-coolant accident, should subsequent evaluation show that such a system would be required.

4.4 Iodine Removal Equipment

4.4.1 Spray

An internal recirculation containment spray system is provided to remove heat from the containment atmosphere and to remove iodine which may be present in the containment following a loss-of-coolant accident. Initially, the two containment spray pumps take suction on the refueling water storage tank and deliver water to spray nozzles inside containment. Each pump has a design capacity of 2600 gpm. Concentrated sodium hydroxide solution is added at the suction of the spray pumps in quantities sufficient to maintain a pH of at least 9.3 in the water in the containment spray. Sodium hydroxide in the containment spray water will scavenge elemental radioiodine from the containment atmosphere. When the refueling water storage tank is exhausted, a portion of the recirculation flow provided for continued core cooling is diverted to the containment spray headers.

To calculate the total iodine removal constant for the proposed system we made conservative assumptions regarding liquid film mass resistance and drop coalescence. We assumed that 10% of the iodine in the containment atmosphere (2-1/2% of the core inventory based on TID-14844) is in the form of organic iodides. This conservative assumption is based on an extensive examination of available literature and on a theoretical evaluation of all applicable formation mechanisms. Since experiments have shown that the removal of organic iodides by a sodium hydroxide spray solution is negligible, we assumed no reduction of the organic iodides by the containment spray. On this basis, we calculated an elemental iodine removal constant of 4.9 hr^{-1} and we calculated that the chemical additive spray reduces the two-hour overall iodine accident dose at the exclusion area boundary by a factor of 5.2 and the thirty-day overall iodine accident dose at the outer boundary of the low population zone by a factor of 8.8. The impact of these reduction factors on accident radiation doses is discussed in Section 5.0. Details of the Research and Development programs being conducted in this area are discussed in Section 6.4.

4.4.2 Charcoal Filters

The air handling system (1) will remove heat from the containment in the post-accident environment and (2) will reduce the iodine concentration in the containment atmosphere by the use of charcoal filters. Five air handling units are provided. In each unit, a fan draws air through a moisture separator, cooling coils, roughing filters, and high efficiency particulate air (HEPA) filters at a flow rate of approximately 25,000 cfm at the maximum post-accident pressure of 47 psig. Charcoal filters are located at the fan discharge header. They are isolated by butterfly valves. Under accident conditions, these valves are automatically opened by the high containment pressure signal and a flow rate of 7,440 cfm is diverted through these filters. Three of the five air handling units will operate even if normal offsite power is lost. This was assumed in our analyses of the Design Basis Accident in Section 5.3. Under this circumstance, approximately 20% of the free volume of the containment is processed through the charcoal filters each hour.

The filters are provided with detectors which initiate alarms in the control room upon sensing high charcoal bed temperatures. A water dousing system is provided to drench the absorbers in the event of high temperatures to prevent the occurrence of a fire in the charcoal.

Experimental evidence indicates that impregnated charcoal filters have a high removal efficiency (greater than 95%) for organic iodides when operating with an air stream having a low relative humidity. The removal mechanism is a process of isotopic exchange.

Research performed at Oak Ridge National Laboratory (ORNL) in a small scale apparatus, using impregnated charcoals of various manufacture, indicates that at 90% relative humidity the removal efficiency for methyl iodide decreases to about 95%. Extrapolation of present data indicates that there may be a rapid decrease in removal efficiency as the relative humidity approaches 100%. It is our understanding that the charcoal probably was partly "waterlogged" in at least some instances under these severe conditions. This could account in part for the low organic iodine retention reported for some experiments.

Westinghouse has performed experiments with a full scale prototype charcoal filter unit in a loop (The Connecticut Yankee Tests). Temperatures were of the order of 270°F (maximum expected post-accident conditions) and a steam-air environment was maintained. Relative humidities of 100% were claimed as measured by a wet-bulb-dry-bulb arrangement. In most cases, organic iodide removal capabilities in excess of 70% were reported.

At present, we find conflicting evidence regarding the capability of impregnated charcoals to effectively remove organic iodides from a moving air stream held at a relative humidity near 100%. Because of this, the applicant will conduct additional research on the removal of organic iodides by impregnated charcoal at high relative humidity. This effort is discussed in detail in Section 6.5.

Several alternatives exist in the design of the air handling and filtration unit depending on the results of the R&D program. These are summarized below:

(1) If it can be shown that the filter efficiency with 100% humid air provides the required 5% per pass to reduce the postulated dose which would be received by an individual at the outer boundary of the low population zone for the duration of the accident to below 300 rem, the system can be operated as proposed. (See Section 5.0 for discussion of accident doses.)

(2) If the results show that the effect of 100% relative humidity with no waterlogging reduces removal efficiency to an unacceptable level, dehumidification equipment can be installed.

In order to reduce the relative humidity of the air to below 90%, it is necessary to increase the temperature of the air stream by a few degrees. This will require that a heater be installed in each air handling unit. A capacity of less than 100 kW is required. Such a heater can be designed using standard engineering techniques. Therefore, we conclude that even if conclusive results are not forthcoming from the research and development program, modifications can be made such that the impregnated charcoal filters will have a removal efficiency much in excess of 5% for organic iodide.

(3) If it is shown that the removal efficiency is acceptable with 100% relative humidity air but that the bed cannot recover from flooding, the filter can be isolated in a tight enclosure during the first portion of the post-accident period.

(4) If it is shown that the direction of flow is a significant factor in improving either the removal efficiency of the bed with air at 100% relative humidity or the ability to recover a flooded bed, the orientation of the bed can be modified to achieve the desired flow direction.

Based on the above, and in consideration of the proposed R&D program on impregnated charcoal filters, we conclude that there will be sufficient information to enable design of an air handling and filtration system having the required capability for removing organic iodides from the containment atmosphere.

4.5 Hydrogen Production and Recombination

We are evaluating the magnitude and consequences of potential hydrogen generation in the post-loss-of-coolant accident environment with particular attention to the evolution of hydrogen by radiation induced decomposition of water. Other contributors to the hydrogen concentration in the post-accident environment include metal-water reaction of the clad at high temperatures and chemical corrosion of various metals in the containment.

We have evaluated the assumptions made by the applicant in determining the hydrogen concentrations which could exist in the containment and believe them to be realistic. Using these assumptions, a hydrogen concentration of 4.1% by volume (the lower limit of hydrogen flammability in air) is predicted 51 days after the occurence of a loss-of-coolant accident.

As discussed in Section 6.4, research and development effort is being directed to determine the rate of radiolytic decomposition of the spray and core cooling water within the containment.

To eliminate the potential for rapid hydrogen oxidation, the applicant has proposed the use of a flame combustor using the containment atmosphere as a primary oxidant and supplemental hydrogen as fuel. Two flame combustors will be located inside containment, one serving as a spare. Each consists of a blower to circulate containment air to the combustion chamber, the combustion chamber, two ignitors (one required) consisting of a capacitance system with surface gap plugs designed to operate in a wet environment, and a dilution chamber downstream to reduce exit temperature to below 300°F. Hydrogen is supplied to the combustor from tanks outside containment through two normally closed valves located outside containment, and a check valve located inside containment. Each combustor contains two thermocouples which monitor combustion. To ensure presence of an oxidant, oxygen is bled into the containment through a separate penetration. This inlet line is located so as to ensure mixing by the containment ventilation system before introduction to the combustor. Oxygen flow is controlled in proportion to hydrogen flow to maintain stoichiometry.

The hydrogen supply lines will be purged with nitrogen before introducing hydrogen. A block and bleed system is provided to prevent either hydrogen or oxygen inleakage when the system is not in use. Alarms are provided to alert the operator to low combustor temperature, and to low manifold pressure for both the hydrogen and the oxygen. Thus, the operator will be able to ascertain that the recombiner is operating properly.

The combustor is designed to process 331 scfm. It will normally be started when the hydrogen concentration in the containment reaches 2% by volume. There is ample margin in the proposed design to accommodate additional hydrogen even if predictions of the rate of radiolysis are in error.

A testing program will be established which will generate the following information relative to design and performance of the recombiner:

1. Performance of the combustor at ignition and under operation with the fuel supply rate varied to provide combustion zone outlet temperatures in the range from 300°F to 1800°F;
2. The lower limit of oxygen concentration for flame stability;
3. Efficiency of combustion by operating at design conditions and determining outlet hydrogen concentration;
4. The stability range of the burner by varying air and fuel flow;
5. The effect of steam and entrained water on burner ignition and operation.

We have reviewed the recombiner design and test program as described above. On the basis of our review and of discussions with experts in this field, we conclude that the flame recombiner is a feasible solution to the hydrogen problem; however, many aspects of the design must be examined more closely before we can conclude without reservation that the design is acceptable. For example, it is necessary to determine the performance limits of the recombiner including limits on pressure, moisture, and hydrogen concentration to demonstrate substantial margin with respect to variation in the expected post-accident conditions. In addition, the applicant will investigate alternate means of recombining the hydrogen, including catalytic recombiners, cryogenic separation, chemical absorption, and processing of the containment gases external to the containment structure.

On the basis of our review to date of the potential for hydrogen accumulation in the post-loss-of-coolant accident environment as a result of radiolytic decomposition and other hydrogen sources, we conclude that there is reasonable assurance that the safety problems associated with the radiolytic production and recombination of hydrogen can be resolved prior to the operation of Unit No. 3 by the proposed research and development program.

5.0 ACCIDENT ANALYSES

5.1 Operating Incidents

In order to assess the safety margins of the plant design, the following plant operating transients were considered by the applicant: rod withdrawal during startup and from power, moderator dilution, loss of coolant flow, loss of electrical load, and loss of offsite a.c. power. The criterion for detailed design of the reactor control and protection system is that it will automatically take corrective action to cope with any of these transients so as to prevent damage to the reactor. Preliminary analyses of these transients have been presented in the PSAR. The consequences of these transients will be calculated when detailed plant design information is available to verify that these transients are within the capabilities of the reactor control and protection systems. Previous staff evaluations of similar PWR designs as a part of the operating license reviews have concluded that anticipated transients are terminated prior to reaching a minimum DNB ratio of 1.3. In our judgment, this limit can be met in Unit No. 3.

The applicant has stated that means for prompt detection of fuel failures during operation are being developed. The design goal is to provide adequate sensitivity to detect the fission product release associated with failure of one fuel rod. We agree with the recommendations of the ACRS in previous cases that this feature is desirable and we will review the final design of the system at the operating license stage.

5.2 Other Accidents

We have evaluated the consequences of other accidents, such as those resulting from rupture of a steam line, rupture of a steam generator tube, improper fuel handling, and inadvertent control rod ejection. For each potential accident, the calculated radiological doses are well within the 10 CFR 100 guidelines. The results of our calculations of the radiological doses associated with these accidents are presented in Table 5.2.

As recommended by the ACRS, we intend to review the assumptions made in the analysis of the fuel handling accident during detailed design to see whether additional conservatism is warranted. If deemed necessary as a result of this review, provisions can be made to reduce the radiological consequences of this accident, such as (1) installing an appropriate air handling system in the fuel storage building and (2) modifying operating procedures to require confinement during refueling.

For this analysis, we have assumed the following:

(1) Perforation of 15 fuel rods (one row of rods in an assembly).

(2) Gap activity in the rods is released. This is assumed to be 20% of the noble gases and 10% of the iodine in the rods.

(3) The accident occurs 100 hours after shutdown. This represents a reasonable estimate of the time required to cool down, remove the

pressure vessel head and the upper internal package, and begin the refueling operation.

(4) 90% of the released iodine is retained in the water of the spent fuel pit or canal.

(5) The ground level release meteorology model is as discussed in Section 2.3.

(6) Dose conversion factors are as listed in TID-14844.

(7) The spent fuel building affords no confinement of activity released.

TABLE 5.2

ACCIDENT RADIOLOGICAL DOSES

Incident	Two-hour Dose at Site Boundary (0.22 mi.)		Course of Accident Dose at Low Population Zone Outer Boundary (0.67 mi.)	
	Whole Body (Rem)	Thyroid (Rem)	Whole Body (Rem)	Thyroid (Rem)
Steam Line Rupture Inside Containment	18	200	12	200
Steam Line Rupture Outside Containment	0.28	140	0.23	62
Steam Generator Tube Rupture	12	73	5	31
Fuel Handling Accident	0.26	105	0.11	44
Rod Ejection Accident	18	200	12	200

5.3 Design Basis Accident

The capability of the emergency core cooling system to cope with a major loss-of-coolant accident is discussed in Section 4.2. We have calculated the consequences of this accident, the design basis accident, assuming the fission product release fractions given in TID-14844, our meteorological model discussed in Section 2.3, and considering the effect of the spray system and the charcoal filters in reducing the iodine source in the containment. The postulated offsite dose at the outer boundary of the low population zone is within the 10 CFR Part 100 guidelines if the removal efficiency of the filters for organic iodides is at least 5%. As discussed in Section 4.4.2, we conclude that such efficiencies can be achieved. We calculate that the two-hour doses at the site boundary would be 5.8 rem to the whole body and 272 rem to the thyroid without taking credit for iodine removal by the charcoal filters, and that the course-of-the-accident doses at the outer boundary of the low population zone will be 7.6 rem to the whole body and less than 300 rem to the thyroid.

6.0 RESEARCH AND DEVELOPMENT

Specific areas requiring research and development (R&D) prior to design completion are summarized below.

6.1 Core Stability and Power Distribution Monitoring

The Westinghouse development program on power distribution monitoring consists of correlations between out-of-core measurements and detailed maps derived from in-core instrumentation in operating reactors. The relationship between the indications of the out-of-core detectors and part length rod positions will be developed during detailed design of the core. This study will consider the effects on core stability of both normal operation of the part length rods and malpositioning of these rods. As experience with operating reactors is gained, detailed information on the effects of core depletion will be correlated with predictions. Although direct experience with the special control rod groups is lacking, it is expected that this will be obtained with operation of the Ginna and Indian Point Unit No. 2 reactors.

We are receiving assistance from Brookhaven National Laboratory and Savannah River Laboratory in evaluating the problems associated with the detection and control of xenon redistribution. Brookhaven is calculating the response of external and in-core neutron detectors to various power distribution patterns, including those from xenon spatial oscillations. Results thus far have confirmed that external detectors could provide sufficient information to control simple xenon oscillation patterns. Savannah River will examine conditions leading to various modes of xenon oscillations and resulting power patterns and the adequacy of design provisions for timely detection and control. These results will be of further assistance to us in assessing the adequacy of techniques for coping with xenon oscillations.

It is not completely clear at this time that the Westinghouse program will demonstrate that sufficient information can be derived from external detectors alone. If the planned R&D program does not produce completely convincing evidence that the out-of-core detection system is sufficient, then we will require installation of permanent in-core detectors to furnish information on power distributions. Xenon produced power redistributions are not highly localized perturbations. Thus, a relatively small number of properly operating axial strings of internal detectors, appropriately distributed radially and azimuthally, would be sufficient for xenon perturbation detection. A research and development program has been undertaken by Westinghouse to develop fixed in-core detectors suitable for continuous monitoring of the core power distribution. Commercially available detectors will be evaluated to determine linearity, response time, sensitivity, and lifetime characteristics. Detectors are presently undergoing tests at Yankee-Rowe. Others will be tested at Saxton, San Onofre, and the Western New York Nuclear Research Center. The evaluation will be completed in December 1970. The lifetime evaluation program will extend to the end of 1971.

We conclude that the programs proposed are adequate to determine if the operator will have need for in-core information. Further, based on the work previously performed by other reactor manufacturers, we conclude that there is reasonable assurance that a system of fixed in-core detectors can be provided, if required, before operation of Unit No. 3.

6.2 Burnable Poison Rods

This program to be conducted by Westinghouse is designed to verify the calculated reactivity worth of the borosilicate glass rods used to eliminate the potential for a positive moderator coefficient early in the life of the first fuel cycle. The program will also evaluate the effect of the rods on power distribution and the mechanical performance of the rods in the reactor environment. Critical experiments have indicated that the standard methods used in core analysis can be used in the design of a core incorporating burnable poison rods. In-core testing of these rods is underway at Saxton and will continue through mid-1969. These rods will be used for the first time in a commercial power reactor at the Ginna station, scheduled to start up in 1969. Experimental results from these units will be compared with analytic predictions.

We conclude that sufficient time exists to permit the experience gained in in-core testing of these rods to be reflected in the final core design of Unit No. 3.

6.3 Rod Burst Program

This program is being conducted by Westinghouse to determine clad deformation characteristics and the extent of potential resultant flow blockage under simulated loss-of-coolant accident conditions. The program will include an evaluation of the effects of temperature, prior irradiation and material properties on clad strength and ductility. The effects of void volume, pellet-to-clad gap, pellet cracking, heating rate, initial gas pressure, and metal-water reaction prior to quenching will be studied. The present schedule is stated below:

	Test	Completion Date
1.	Rod Burst Tests - Unirradiated Clad	Completed
2.	Rod Burst Tests - Unirradiated Hydrided Clad	Completed
3.	Complete Quench Tests	December 1969
4.	Rod Burst Tests - Irradiated Clad	July 1969

The unirradiated clad experiments (Test 1) indicate that the geometry of the rupture is consistent. It exhibits a small longitudinal split in the cladding with a length of approximately 1/2-inch maximum and a width of 1/32- to 3/16-inch. Such a rupture would result in a flow area blockage of 10-15% for a single rod. These data also indicate that the experimental burst pressure versus clad temperature curve is 50 to 200% higher than the design curve used in the rod burst analyses. Westinghouse has completed analytical studies that assume each of the fuel rods forming the hot channel burst at the same axial location to produce the maximum flow blockage possible with an average blockage of 12.5% per rod. The analyses indicate that the local mass flow rate would be reduced to 40% of the nominal value. This would reduce the heat transfer coefficient downstream of the rupture and extend the time at which the clad thermal temperature transient reverses. The result would be an increase of peak clad temperature of 50°F and a negligible increase in metal-water reaction.

We conclude that the program proposed, in conjunction with the FLECHT (Full Length Emergency Cooling Heat Transfer Test) program in which Westinghouse is participating, will provide adequate information to establish ECCS effectiveness over a range of parameters representative of the design basis accident.

Upon completion of our evaluation of program results, a criterion for maximum clad temperature will be established.

6.4 Containment Spray

The additional research and development effort on the containment spray system to be performed by Westinghouse consists basically of data analyses and comparison of calculational models with existing experiments. Little new data will be generated. The program is described below.

1. Droplet coalescence - A theoretical model will be developed which will assume that collisions between spray droplets result in coalescence. The effect on iodine removal will be assessed. The model will be applied to NSPP and CSE experiments and the results will be compared with the experimental results. This portion of the program will be completed in the third quarter of 1969.
2. Liquid Phase Mass Transfer Resistance - Liquid film mass transfer resistance will be included in the Westinghouse analytical model. Mass transfer coefficients and partition factors derived from the literature will be applied to determine the effect of liquid film resistance on iodine removal. The model will be applied to NSPP and CSE tests and compared with experimental results to determine if liquid film resistance was significant in the experiments. This program is scheduled for completion in the third quarter of 1969.
3. Materials Compatibility - Tests on the corrosion of major construction materials by the spray solution have been performed. Correlation and documentation of the results of these tests is underway. This study will investigate the temperature dependency of the corrosion rate over the range of accident conditions. Additional testing is being conducted to investigate corrosion in stressed and welded specimens and the effect of the spray solution on lubricants, sealants, and insulation. The status of information

relative to the use of spray additives will be reported in the first quarter of 1969.

We conclude that the research and development program on the containment spray system is adequate and will generate sufficient data for use in analysis of the detailed design of the spray system.

6.5 Organic Iodine Removal by Charcoal Filters

As indicated in Section 4.4.2, the applicant has proposed an impregnated activated charcoal filter system to be used to remove organic iodides from the containment atmosphere.

Tests will be conducted at the Oak Ridge National Laboratory (ORNL) to supplement existing data. These tests are designed to more accurately simulate the charcoal beds proposed for Unit No. 3 than the previous ORNL tests. The carbon test bed will be 3 inches in diameter, 2 inches deep, and will be contained between punched plate retainers. The bed depth and the retainers are similar to those to be used in the installed filter units. It will be oriented in such a manner that tests can be performed with either upflow or downflow through the bed. Tests will be conducted with a methyl iodide concentration of 6 mg/m^3 which simulates the concentration expected in the containment following a loss-of-coolant accident. (Most of the previous ORNL tests were performed using a 1-inch diameter carbon bed, 2 inches deep, contained by a wire mesh screen and oriented 35° from the vertical. Most of these tests were run with a methyl iodide concentration of 80 mg/m^3 in the air stream.)

The following tests are proposed as a part of this additional R&D effort:

(1) Six tests with the air stream at relative humidities between 90 and 100% and flow downward through the bed.

(2) Two tests under conditions similar to those in (1) above with flow upward through the bed.

(3) Three tests with the bed initially flooded, then purged of excess water by the flow of air at 100% relative humidity, with downflow through the bed.

(4) Three tests under the conditions of (3) above with upflow through the bed.

(5) Two comparison tests using a wire mesh screen to retain the carbon bed in place of the punched metal plate.

(6) Two comparison tests using a methyl iodide concentration of 80 mg/m^3.

Results are expected during the third quarter of 1969.

We have evaluated the tests proposed and have determined that they will provide sufficient data to define the methyl iodide removal efficiency of the charcoal as a function of relative humidity and of adsorbed water on the bed.

The scale-up required to apply the results of these laboratory tests to the evaluation of commercial filter units must consider the possibility of localized condensation in the bed resulting from the temperature gradients in the time-dependent thermal distribution within the bed. The thermal distribution will vary with time since the bed is initially cold and heat losses may occur from the exterior surfaces of the filtration units when contacted by the containment spray. The determination of localized condensation can be accomplished using standard engineering techniques. Once determined, the effect of condensation on flow blockage and the resulting higher velocity and lower residence time in the bed can be ascertained. Therefore, we conclude that the data generated by the proposed R&D program, applied appropriately, can be used to determine the organic iodide removal characteristics of the commercial air filtration units proposed.

6.6 Failed Fuel Monitor

As presently proposed, Unit No. 3 will rely on the letdown monitor to detect fuel failure. In order to increase the reliability, sensitivity, and response time of the failed fuel detection system, Westinghouse is conducting a research and development program at the Saxton reactor which is considering failed fuel detection using (1) a delayed neutron monitor, (2) a coolant gamma activity monitor, (3) a gross gamma monitor along a main coolant line, and (4) a letdown monitor.

The evaluation of the performance of these devices at Saxton will be available by late 1969. Thus, we conclude that a detection method which optimizes the current technology in reliability, sensitivity, and response time will be installed in Unit No. 3.

6.7 Thermal Shock

The applicant's analysis of thermal shock following operation of the emergency core cooling program is essentially complete. However, the results are sensitive to both the fracture mechanics properties of heavy section steel and the heat transfer coefficients assumed. The heavy section steel technology program at Oak Ridge National Laboratory will provide information on material properties. It is scheduled for completion by 1973. Westinghouse is making efforts to obtain the effects of temperature and irradiation on fracture toughness. They participate in a Euratom-funded program to obtain this information.

We conclude that adequate information on the material properties will be available before the vessel experiences the several years of irradiation required to embrittle the steel to the point where this problem is of concern.

6.8 Hydrogen Generation

The applicant has proposed a program to determine the rate of hydrogen generation by radiolytic decomposition, including the effects of flow, temperature and chemical factors. The extent of hydrogen evolution as a result of corrosion of containment components will be determined by the materials compatibility program described in Section 6.4. We agree that these factors are important and must be included in the research and development program. The following information will also be provided: (1) the extent of gamma and beta radiation absorption by water in both the reactor core and containment sump, and (2) the equilibrium hydrogen concentration to be expected in the containment post-accident environment in the absence of corrective action, including the influence of the relative water and air volumes, the surface area of the air-water interface, and chemical composition.

We conclude that there is reasonable assurance that the safety problems associated with the radiolytic production of hydrogen can be resolved prior to operation of Unit No. 3.

6.9 Other Research and Development Programs

Other areas of research and development conducted by Westinghouse are outlined below:

1. Saxton Loose Lattice Irradiation Program to determine fuel performance of standard fuel assemblies at high linear heat generation rate and high burnup. Completion is scheduled for the last half of 1971.
2. Zorita Irradiation Program to determine performance of standard fuel assemblies at high linear heat generation rate and high burnup. Completion is scheduled in April 1973.
3. ESADA DNB Program to experimentally determine the effect rod axial heat flux distributions on DNB. Testing will be completed by September 1969.

4. Loss of Coolant Analysis Program to incorporate more realistic heat transfer models into the computer codes used to evaluate the consequences of loss-of-coolant accidents. This program was scheduled for completion on October 1968. We are awaiting a report.
5. FLECHT (Full Length Emergency Cooling Heat Transfer Test) to experimentally determine the thermal behavior of fuel rods during the simulated core recovery period following a loss of coolant accident. It is scheduled for completion by February 1970.
6. Flashing Heat Transfer Program to obtain experimental values of the heat transfer coefficients during blowdown, when uncovered, and during reflooding. Completion was scheduled for October 1968. A report of the final results is expected shortly.
7. Blowdown Force Evaluation Program to determine the forces on core internals during the blowdown. BLODWN-1 has been developed to analyze the pressure velocity and force transients during the subcooled portion of the blowdown. The program is being extended to consider two-phase blowdown. This was scheduled for completion in January 1969. A report of the final results is expected shortly.

6.10 Conclusions

Based on our review of the research and development programs proposed, we conclude that these programs are timely and are reasonably designed to accomplish their respective development objectives, will provide adequate information on which to base analyses of the design and performance, and should lead to acceptable designs for the respective systems.

7.0 TECHNICAL QUALIFICATIONS

We have reviewed the application with respect to the adequacy of the technical qualifications of the Consolidated Edison Company of New York, Inc. The execution of the Indian Point Nuclear Generating Unit No. 3 project is the sole responsibility of the Consolidated Edison. They have previous nuclear experience through their operation of Indian Point Unit No. 1 and have a twenty-man Nuclear Division associated with their Mechanical Engineering Department.

Consolidated Edison has engaged Westinghouse Electric Corporation as the prime contractor. Westinghouse has engaged United Engineers and Constructors to serve as the architect engineer. These contractors, as well as Consolidated Edison, have had extensive experience in the design and construction of light water power reactors and are recognized to be competent in their areas of specialization. On the basis of our previous and current evaluations of plants designed and constructed by the contractors and the applicant's experience in the operation of Unit No. 1 and based on our evaluation of the responsible personnel and of the quality control organization discussed in Section 8.9, we conclude that the Consolidated Edison Company of New York, Inc. and its contractors are technically qualified to design and build Indian Point Unit No. 3.

8.0 QUALITY ASSURANCE

We have used the guidelines outlined in our recently completed evaluation of the Zion application in evaluating the quality assurance program for the Unit No. 3 facility.

The applicant assigned Westinghouse the prime responsibility for assuring adequacy in all design and construction activities. The principal subcontractor, United Engineers and Constructors (UE&C), will prepare all construction specifications and manage all construction work. Most of the quality assurance activities will be carried out by Westinghouse and UE&C. These will include the entire plant design and construction. Each of these organizations has a quality assurance organization and each will have a separate quality assurance program for the Indian Point No. 3 project. Consolidated Edison has stated that it will also have a quality assurance organization for this project. The applicant's quality assurance program will include monitoring of the Westinghouse and UE&C efforts. Most of this surveillance will be performed directly for the applicant by the U. S. Testing Company (USTC), but some will also be conducted by Consolidated Edison's own engineers. Based on the information in the application as amplified by oral discussions with the applicant, we have determined that the applicant's quality assurance program is in accord with our guidelines.

We base this determination on the following:

(1) Steps are being taken to assure close association and interchange of information at all levels between respective functional groups, including those associated with the applicant and all subcontractors;

(2) The U. S. Testing Company (USTC) will report directly to the applicant and perform surveillance according to a preliminary, yet carefully delineated USTC surveillance plan;

(3) The applicant will assure that independent checking of designs at important interfaces will be carried out between UE&C and Westinghouse, and between these and other important organizations;

(4) The applicant's engineers will be used in a number of quality assurance activities to supplement those being performed for him by USTC;

(5) The applicant has defined its exact organization and role in this project, including the internal quality control organization and other staff and line functions;

(6) We have examined and found acceptable a list of titles of Quality Assurance Procedures and Quality Control Instructions to be used to prescribe the activities, procedures, and efforts to be undertaken in providing quality assurance and the responsible organization;

(7) The organization and programs to be used by U. S. Testing Company have been evaluated by us and found acceptable. This included the qualification requirements for the USTC quality assurance personnel;

(8) The applicant has stated its intent to require a specific plan which will ensure independent review; and

(9) The applicant will use a planned systematic procedure for audits, designed to assure independence of inspectors and the quality assurance organizations from those responsible for plant design and construction.

On the basis of our review of the applicant's quality assurance plan, as amplified by oral discussions, we have concluded that the applicant's quality assurance program provides adequate assurance of quality in safety related components and structures.

9.0. REPORT OF THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

In a letter to the Commission dated January 15, 1969, the Advisory Committee on Reactor Safeguards (ACRS) reported on the proposed Indian Point Nuclear Generating Unit No. 3. A copy of this letter is attached as Appendix B. The letter recommended that (1) the onsite power sources should have a greater independence than in the proposed system, at least to the extent that they cannot be connected together with automatically operated devices, (2) the applicant should review the assumptions made in the analysis of a refueling accident to see whether additional conservatism is warranted, and (3) the instrumentation should be reviewed for common failure modes, taking into consideration the possibility of systematic, non-random, concurrent failures of redundant devices, not considered in the single-failure criterion. These items are discussed in Sections 3.4.2, 5.2, and 3.3 respectively. In addition, the ACRS commented on the potential for missile generation resulting from failure of a defective main-coolant-pump flywheel which might cause damage to equipment within the containment. As recommended by the ACRS, the pump flywheel assemblies will receive detailed and extensive inspection as a part of the applicant's quality assurance program. The flywheel assemblies will also receive special consideration from the regulatory staff. The details on inservice inspection of the flywheel assemblies will be developed during the operating license review and will be included in the

Unit No. 3 Technical Specifications. The ACRS also called attention to matters previously identified by the ACRS as warranting careful consideration with regard to all large water-cooled power reactors of high power density. These items have been discussed in this evaluation and will be resolved to the satisfaction of the staff and the ACRS prior to issuance of an operating license.

The ACRS letter concluded, "The ACRS believes that the items mentioned can be resolved during construction, and that, if due consideration is given to the foregoing, nuclear Unit 3 proposed for Indian Point can be constructed with reasonable assurance that it can be operated without undue risk to the health and safety of the public."

10.0 COMMON DEFENSE AND SECURITY

The application reflects that the activities to be conducted would be within the jurisdiction of the United States and that all of the directors and principal officers of the applicant are American citizens. We find nothing in the application to suggest that the applicant is owned, controlled, or dominated 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 paragraph 50.33(j) of 10 CFR 50. The applicant will rely upon obtaining fuel as it is needed from sources of supply available for civilian purposes, so that no diversion of special nuclear material from military purposes is involved. For these reasons, and in the absence of any information to the contrary, we conclude that the activities to be performed will not be inimical to the common defense and security.

11.0 CONCLUSIONS

Based on the proposed design of the Indian Point Nuclear Generating Unit No. 3 of the Consolidated Edison Company of New York, Inc.; on the criteria, principles, and design arrangements for systems and components thus far described, which include all of the important safety items; on the calculated potential consequences of routine and accidental release of radioactive materials to the environs; on the scope of the development program which will be conducted; and on the technical competence of the applicant and the principal contractors; we have concluded that, in accordance with the provisions of paragraph 50.35(a), 10 CFR 50, and 2.104(b), 10 CFR 2:

(1) The applicant has described the proposed design of the facilities, including the principal architectural and engineering criteria for the design, and has identified the major features or components for the protection of the health and safety of the public;

(2) Such further technical or design information as may be required to complete the safety analysis and which can reasonably be left for later consideration, will be supplied in the final safety analysis report;

(3) Safety features or components which require research and development have been described by the applicant and the applicant has identified, and there will be conducted, a research and development program reasonably designed to resolve any safety questions associated with such features or components;

(4) On the basis of the foregoing, there is reasonable assurance that (i) such safety questions will be satisfactorily resolved at or before the latest date stated in the application for completion of construction of the proposed facility, and (ii) taking into consideration the site criteria contained in 10 CFR Part 100, the proposed facility can be constructed and operated at the proposed location without undue risk to the health and safety of the public;

(5) The applicant is technically qualified to design and construct the proposed facility; and

(6) The issuance of a permit for the construction of the facility will not be inimical to the common defense and security or to the health and safety of the public.

APPENDIX A

CHRONOLOGY

REGULATORY REVIEW OF THE CONSOLIDATED EDISON COMPANY OF NEW YORK, INCORPORATED

INDIAN POINT NUCLEAR GENERATING UNIT NO. 3

Date	Item
1. April 26, 1967	Submittal of Preliminary Safety Analysis Report and License Application.
2. August 22, 1967	Meeting with applicant to discuss schedule of regulatory review of application.
3. September 8, 1967	Meeting with applicant to discuss plant design.
4. September 26, 1967	Meeting with applicant to discuss Electrical Transmission System and Emergency Power Sources.
5. December 28, 1967	ACRS Subcommittee Meeting and site visit.
6. February 19, 1968	Request to applicant for additional information on plant design, containment structural design, quality control procedures, site, reactor vessel and primary system, engineered safety features, seismic design and safety evaluation.
7. April 16, 1968	Meeting with applicant to discuss tornado design criteria and containment structural criteria.
8. July 1, 1968	Request to applicant for additional information on protection instrument system, control rod position indication, separation of certain control safety functions and flooding.
9. July 16, 1968	Request to applicant for additional information on general structural design, containment structural design materials, corrosion protection, construction inspection, and testing and in-service surveillance.

10.	July 23, 1968	Submittal of request by applicant for an exemption to the requirements of 10 CFR 50.10(b) which would permit pouring of base mat concrete up to the bottom liner plate and includes the walls of the reactor vessel cavity and the recirculating pump pit; installation of the bottom liner plates and transition of the rebar for the base concrete over the bottom liner plates.
11.	August 30, 1968	Submittal of Amendment No. 1 to PSAR, answers to DRL requests for additional information of February 19 and July 1, 1968.
12.	September 16, 1968	Submittal of Amendment No. 2 to PSAR, answers to DRL requests to applicant for additional information of February 19 and July 1, 1968, and two pages to be inserted in Amendment No. 1.
13.	October 1, 1968	Meeting with applicant to discuss containment structural design.
14.	October 11, 1968	Meeting with applicant and Westinghouse to discuss Amendment No. 1 to the PSAR.
15.	October 18, 1968	Submittal of Amendment No. 3, replacement pages for Amendment No. 2, and information requested orally by the AEC regulatory staff on October 1, 1968.
16.	October 22, 1968	ACRS Subcommittee Meeting.
17.	October 25, 1968	Meeting with applicant to discuss instrumentation and control and electrical power.
18.	October 29, 1968	Meeting with applicant to discuss meteorology and hydrology.
19.	October 31, 1968	Submittal of Amendment No. 4, answers to DRL request for additional information of July 16, 1968 and changes and additions to various pages in Amendments 1 and 2.

20. November 4, 1968	Submittal of Amendment No. 5, answers to questions raised by the AEC regulatory staff at a meeting on October 11, 1968.
21. November 6, 1968	Meeting with applicant to discuss iodine removal.
22. November 12, 1968	Request to applicant for additional financial data.
23. November 15, 1968	AEC letter to applicant granting an exemption, pursuant to 10 CFR 50.12 from the provisions of 10 CFR 50.10(b) to Consolidated Edison Company of New York, Inc.
24. November 20, 1968	Request to applicant for additional information on flooding, rod ejection accident analysis, design basis accident doses, electrical power, instrumentation and control, cable routing, and radiation monitoring, and miscellaneous other items.
25. November 25, 1968	Submittal of Amendment No. 6, replacement pages for Amendments 1 and 5 and technical information on instrumentation, electrical power supplies, fuel and fuel clad performance and rod drop accident analysis.
26. December 9, 1968	Submittal of Amendments 7 and 8, answers to DRL request for additional information of November 20, 1968, and a guide to the PSAR.
27. December 12, 1968	Meeting with applicant to discuss conduct of operations, structural design, recombiner design, and iodine removal.
28. December 28, 1969	ACRS Subcommittee meeting.
29. January 3, 1969	Submittal of Amendments 9 and 10, Financial Data requested by the AEC regulatory staff on November 12, 1968, a description of the R&D program on charcoal filters for removal of organic iodine, and replacement pages for Amendment No. 7 to the PSAR.

30. January 6, 1969	Submittal of Amendment No. 11, correction pages regarding flooding analysis for Amendment No. 7 to the PSAR.
31. January 7, 1969	Meeting with applicant to discuss Quality Assurance Program for Indian Point Nuclear Generating Unit No. 3.
32. January 10, 1969	ACRS Meeting.
33. January 15, 1969	ACRS letter to Chairman Seaborg on Indian Point Nuclear Generating Unit No. 3.
34. January 28, 1969	Submittal of Amendment No. 12 to Application for Licenses, changes earliest and latest completion dates for the plant and updates administrative information relating to the Company.

APPENDIX B

DR-1988

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

January 15, 1969

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

Subject: REPORT ON INDIAN POINT NUCLEAR GENERATING UNIT NO. 3

Dear Dr. Seaborg:

At its 105th meeting, January 9-11, 1969, the Advisory Committee on Reactor Safeguards completed its review of the application of Consolidated Edison Company of New York, Inc., for authorization to construct Indian Point Nuclear Generating Unit No. 3. This project had previously been considered at the 103rd meeting of the Committee, and at Subcommittee meetings on October 22, 1968, and December 28, 1968. During its 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 and their consultants. The Committee also had the benefit of the documents listed.

Indian Point Unit No. 3 includes a four-loop nuclear steam supply system with a design power rating of 3025 MW(t). The design is very similar to that of Unit No. 2 except for differences in power level and some of the engineered safety features. The peak values of core heat flux and linear heat generation rate are slightly lower than those proposed for the reactors of the Zion Station.

The applicant has considered the possibility of reactor vessel failure as a result of thermal shock caused by emergency core cooling system action in the unlikely event of a loss-of-coolant accident during the later portions of vessel life. He has conducted engineering studies which have established the feasibility of a cavity flooding system that could flood to a level above the top of the core and thereby provide additional protection in the event of such failure. He stated that this system would be installed at a future time if studies now under way indicated that vessel failure as a result of thermal shock could occur.

The vessel cavity walls will be designed to withstand the mechanical forces which would result if a highly unlikely vessel split were to occur with the primary system pressurized. Design of the system will be such as to permit annealing of the reactor pressure vessel, if this should become necessary.

The applicant proposes to install flame recombiners to cope with potential hydrogen concentration buildup from various sources in the unlikely event of a loss-of-coolant accident. He has described a research and development program to ascertain the need for a recombiner, to study other types of recombiners, and to confirm acceptable performance. The applicant also described measures to be taken in the design and operation to prevent inadvertent introduction of hydrogen into the containment.

The on-site emergency power supply for Unit No. 3 employs four 480 V buses energized (upon loss of normal power) by three diesel generators, two of which are required to furnish energy to engineered safety features. The applicant proposes an automatic system of cross-connecting sources and loads. The Committee believes that the on-site power sources should have a greater independence than in the proposed system, at least to the extent that they cannot be connected together with automatically operated devices. An appropriate modification should be developed by the applicant and the matter resolved with the Regulatory Staff.

The main-coolant-pump flywheels represent a potential source of missiles within the containment, and the applicant has described measures taken to assure conservative design and high quality fabrication to minimize the possibility of flywheel failure. Additional steps may be warranted to assure the integrity of the flywheel assembly, and the Committee recommends that details concerning the adequacy of design, the material characteristics, quality assurance, and in-service inspection requirements be resolved between the applicant and the Regulatory Staff.

In the event that an irradiated fuel assembly is dropped or otherwise damaged during transit from the reactor vessel to the spent fuel pit, the cladding on the fuel rods may be ruptured with a consequent release of radioactivity. In view of the relatively high population density close to the Indian Point site, the applicant should review the assumptions made in analysis of a refueling accident to see whether additional conservatism is warranted in assessing its effects and the provisions to cope with the accident. The matter should be resolved with the Regulatory Staff.

Part-length control rods and special full-length rods are provided to control spatial neutron flux oscillations. Provision will be made for installation of permanent in-core detectors, should such detectors be required to assure adequate measurement of the power distribution.

Means will be provided for early detection of abrupt gross failure of a fuel element.

The instrumentation design should be reviewed for common failure modes, taking into account the possibility of systematic, non-random, concurrent failures of redundant devices, not considered in the single-failure criterion. The applicant should show that the proposed interconnection of control and safety instrumentation will not adversely affect plant safety in a significant manner, considering the possibility of systematic component failure. The Committee believes that this matter can be resolved by the applicant and the Regulatory Staff.

The Committee calls attention to matters previously identified as warranting careful consideration with regard to all large, water-cooled power reactors of high power density.

The Committee also emphasizes the importance of independent action by the applicant to assure quality in the construction of the facility.

The ACRS believes that the items mentioned can be resolved during construction, and that, if due consideration is given to the foregoing, nuclear Unit 3 proposed for Indian Point can be constructed with reasonable assurance that it can be operated without undue risk to the health and safety of the public.

Sincerely yours,

Signed Stephen H. Hanauer

Stephen H. Hanauer
Chairman

References attached.

References - Indian Point Nuclear Generating Unit No. 3

1. LeBoeuf, Lamb & Leiby letter, dated April 26, 1967; Consolidated Edison Company of New York, Inc., Application for Licenses, dated April 25, 1967; Preliminary Safety Analysis Report

2. LeBoeuf, Lamb, Leiby & MacRae letter, dated August 30, 1968; Amendment No. 1 to Application for Licenses

3. LeBoeuf, Lamb, Leiby & MacRae letter, dated September 16, 1968; Amendment No. 2 to Application for Licenses

4. LeBoeuf, Lamb, Leiby & MacRae letter, dated October 18, 1968; Amendment No. 3 to Application for Licenses

5. LeBoeuf, Lamb, Leiby & MacRae letter, dated October 31, 1968; Amendment No. 4 to Application for Licenses

6. LeBoeuf, Lamb, Leiby & MacRae Letter, dated November 4, 1968; Amendment No. 5 to Application for Licenses

7. LeBoeuf, Lamb, Leiby & MacRae letter, dated November 25, 1968; Amendment No. 6 to Application for Licenses

8. LeBoeuf, Lamb, Leiby & MacRae letter, dated December 9, 1968; Amendment No. 7 to Application for Licenses

9. Amendment No. 8 to Application for Licenses

10. LeBoeuf, Lamb, Leiby & MacRae letter, dated January 3, 1969; Amendment No. 10 to Application for Licenses

11. LeBoeuf, Lamb, Leiby & MacRae letter, dated January 6, 1969; Amendment No. 11 to Application for Licenses

APPENDIX C
COMMENTS ON

Indian Point Nuclear Generating Unit No. 3
Consolidated Edison Company of New York, Inc.
Preliminary Safety Analysis Report
Fifth Supplement dated November 4, 1968

Prepared by

Air Resources Environmental Laboratory
Environmental Science Services Administration
January 2, 1969

As pointed out in our comments of October 29, 1965 and November 29, 1968 on Unit No. 2, a primary influence on the meteorological statistics of the Indian Point site is 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 of unstable (lapse) flow in the upriver direction during the day and stable (inversion) flow in the downriver direction at night. Figure 1.6-1 of Supplement Five, although in terms of average vectors, shows the marked wind reversals at sunset and sunrise and the persistent, channeled flow that occurs during the middle of the night (see the mean direction between 0200 and 0800 hours). The mean speed during this persistent period is about 2.5 m/sec which indicates that 50% of the time inversion 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 assumption would be to assume a ground release at Unit No. 3 with a 1 m/s wind speed under inversion conditions in a persistent downriver direction for a period of 8 hours.

50-286

APPENDIX C

Comments on

Indian Point Nuclear Generating Unit No. 3
Consolidated Edison Company of New York, Inc.
Preliminary Safety Analysis Report
Exhibit B, Volumes I, II and II, Part B dated April 1968

Prepared by

Air Resources Environmental Laboratory
Environmental Science Services Administration
May 24, 1968

No new meteorological information is contained in this report that was not considered in our comments of October 29, 1965 on "Description and Safety Analysis for a Conceptual Unit at Indian Point", Volumes I and II dated October 1, 1965.

APPENDIX D

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

JAN 6 1969

RECEIVED
JAN 7 1969
U.S. ATOMIC ENERGY COMMISSION
Regulatory
Mail Section

Mr. Harold Price
Director of Regulation
U.S. Atomic Energy Commission
4915 St. Elmo Avenue
Bethesda, Maryland 20545

Dear Mr. Price:

Transmitted herewith in response to a request by Mr. Roger Boyd, is a review of the hydrologic and geologic aspects of the Indian Point Nuclear Generating Unit No. 3 proposed by the Consolidated Edison Company of New York, Inc.

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

Sincerely yours,

Arthur B Baker

Acting Director

Enclosure:

Consolidated Edison Company of New York, Inc.
Indian Point Nuclear Generating Unit No. 3
Docket No. 50-286

Hydrology

The Indian Point Nuclear Generating Unit No. 3 will be located adjacent to and immediately downstream of Unit No. I, in Westchester County, Village of Buchanan, New York, on the east bank of the Hudson River, 2 miles downstream of Annsville Creek, 2½ miles southwest of Peekskill, and 24 miles north of New York City. The unit will employ a pressurized water reactor of 3,025 megawatts thermal or 1,005 megawatts electrical capacity. Water for once-through condenser cooling will be taken from and returned to the Hudson River. The drainage area of the Hudson River at the site is approximately 12,500 square miles.

The hydrologic analysis of the site with respect to the release of radionuclides to the environment from operational or accidental spills, as presented by the applicant, appears to be adequate. Comments on the hydrologic analysis were based on a review of the Preliminary Safety Analysis Report, an independent check of the available data and literature, and an inspection of the site on December 20, 1968.

The Hudson River past the site is subject to tidal action. Discharge at times of normal ebb and flood tides vary between 250,000 and 300,000 cubic feet per second. The natural water supply at the site may be expected to greatly exceed the cooling water requirements at all times.

The design flood level for the plant is 19.3 ft above mean sea level, which corresponds to the stage of the probable maximum hurricane plus spring high tide as computed by the applicant. The analysis of a high spring runoff concurrent with the hurricane surge showed no additional significant rise in the water surface elevation. The flood level for a fresh water design flood of 1,720,000 cubic feet per second, consisting of precipitation runoff, attenuated peak discharge resulting from the breaching of five major dams upstream of the site, and the ebb tide flow, is 16.0 ft above mean sea level. The design flood discharge and stage appear to be reasonable.

Any accidental spill of radionuclides on the impervious paved and built up areas of the plant site could be flushed to the Hudson River directly. Any contaminant which permeated the soil surface in the plant area would be moved to the Hudson River through the very permeable limestone bedrock under the influence of the hydraulic gradient in this unconfined aquifer. Radionuclides which reached the Hudson River accidentally or operationally would be moved up and downstream considerable distances. Poughkeepsie, 30 miles upstream of the site, is the nearest municipality utilizing the Hudson River for water supply. The city of New York may use its Chelsea pumping station, 22 miles upstream, as a supplementary source of water supply during drought conditions. Contaminants entering the Hudson River at the plant site would not likely reach the Chelsea station or Poughkeepsie until at least three tidal cycles had elapsed, thereby affording some time to monitor the contaminant concentration and initiate alternative water-supply plans or remedial action as necessary.

The deposition of a significant amount of longlived radioactive materials on the relatively permeable soil in the site area, the outcrops of permeable limestone and other consolidated rocks which surround the site, and the numerous water-supply reservoirs located within a fifteen mile radius of the plant, could result in the introduction of radioactive materials into the fresh water supplies of the area.

Geology

The analysis of the geology of the Indian Point Nuclear Generating Plant, Unit No. 3, has been reviewed and compared with the available literature. The analysis appears to be carefully derived and to present an adequate appraisal of those aspects of the geology that would be pertinent to an engineering evaluation of the safety of the site.

The site is located in the New England Uplands Provinee of New York. According to the applicant's report, Unit No. 3 will be founded in a hard limestone that is well-jointed but reported to be noncavernous; it should provide an adequate foundation for the proposed facility.

There are no known active faults or other young geologic structures in the area that could be expected to localize earthquakes in the immediate vicinity of the site. Although several ancient faults occur in the area, none appears to have been tectonically active since glacial times, or for at least the past several hundred thousand years.

Although it may be anticipated that earthquakes within the general region will continue with approximately the same frequency and with approximately the same intensity with which they have been recorded during the past 100 years, there are no demonstrable geologic controls which could be expected to concentrate such events in the immediate vicinity of the site.

APPENDIX E

U.S. DEPARTMENT OF COMMERCE
ENVIRONMENTAL SCIENCE SERVICES ADMINISTRATION
COAST AND GEODETIC SURVEY
ROCKVILLE, MD. 20852

50-286

SEP 1 3 1968

IN REPLY REFER TO: C23

Mr. Harold L. Price
Director of Regulation
U. S. Atomic Energy Commission
Washington, D. C. 20545

Dear Mr. Price:

In accordance with your request, we are forwarding 10 copies of our report on the seismicity of Peekskill, New York, and vicinity. The Coast and Geodetic Survey has reviewed and evaluated the information on the seismic activity in the area as presented by the Consolidated Edison Company in the "Preliminary Safety Analysis Report," and we are now submitting our conclusions concerning the seismicity factors.

If we may be of further assistance to you please contact us.

Sincerely yours,

Don A. Jones
Don A. Jones
Rear Admiral, USESSA
Director

Enclosure

REPORT ON THE SITE SEISMICITY OF THE INDIAN POINT POWER PLANT NO. 3, NEW YORK

At the request of the Division of Reactor Licensing of the Atomic Energy Commission, the Seismology Division of the Coast and Geodetic Survey has evaluated the seismicity of the area around the proposed Indian Point Power Plant No. 3 and has reviewed the similar analysis presented by the applicant in the "Preliminary Safety Analysis Report."

Based on the review of the seismic history of the site and the related geologic considerations, the Coast and Geodetic Survey believes that the applicant's proposal to use 0.10 g for representing earthquake disturbances likely to occur within the lifetime of the facility to be adequate. The Survey agrees with the applicant that 0.15 g would provide adequate basis for designing protection against the loss of function of components important to safety.

U. S. Coast and Geodetic Survey
Rockville, Maryland 20852

September 10, 1968

4227

APPENDIX F

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

CEREN

6 January 1969

RECEIVED JAN 7 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 letter of 10 September 1968 regarding Docket No. 50-286, Consolidated Edison Company of New York's proposed Indian Point Nuclear Generating Unit No. 3, Amendments thereto and the meeting of 29 October 1968 at AEC and 3 January 1969 at CERC.

Pursuant with our arrangements, Mr. R. A. Jachowski of CERC has reviewed this report from the viewpoint of storm surge associated with the Probable Maximum Hurricane (PMH) leading to the establishment of a design water level at the proposed Indian Point site.

The proposed design water level elevation of 19.3 feet above MSL (Sandy Hook) for the flood protection level is based on the PMH parameter recently established in the "Interim Report - Meterological Characteristics of the Probable Maximum Hurricane, Atlantic and Gulf Coasts of the U.S.", HUR 7-97, May 1968. In addition, the analysis also considers the effects of the hurricane winds continuing to act on the surge as it progresses up the Hudson River to the plant sites.

It is the opinion of this office from a review of the applicant's analysis, that the proposed design water level of 19.3 feet MSL is an acceptable value based on the sound engineering application of the PMH parameters.

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

Sincerely yours,

MYRON DOW SNOKE
Lieutenant Colonel, CE
Director

APPENDIX G

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

JAN 18 1968

b73 1/19/68

Dear Mr. Chairman:

Pursuant to Section 5 of Public Law 89-605 and other authorizations, we are presenting the views of the Department of the Interior in the matter of the application by Consolidated Edison for a construction permit to add Unit No. 3 to the Indian Point nuclear generating station located on the Hudson River in Westchester County, New York, AEC Docket No. 50-286.

We have considered the application in the light of our Memorandum of Understanding of March 20, 1964, our responsibilities for the protection of water quality under the Federal Water Pollution Control Act and Executive Order No. 11288, for the protection of fish and wildlife resources under the Fish and Wildlife Coordination Act, and for the furtherance of the Administration's policy to preserve and to restore the quality of our environment, and the directive of the Congress that the resources of the Hudson should be protected from adverse Federal actions until there has been opportunity for the negotiation of a Hudson River Compact.

Unless certain conditions are met, we are concerned that the project for which the license is sought could impair the value of the waters of the Hudson River. As you know, the State of New York has adopted water quality standards for the Hudson River that have been approved by this Department. In addition, there is a water quality enforcement conference covering this section of the Hudson. We have considered the potential pollution problem in its thermal, radiological, and chemical aspects. Thermal pollution is a matter of particular concern and we will want to continue to work with you and the applicant to assure that adequate cooling measures are installed and effectively operated. We also want to assure that the fishery resources are thoroughly protected by adequate screening facilities.

A detailed discussion of the project application as it relates to our responsibilities, with recommendations, is found in the attached reports. The Department of the Interior would not object to the issuance of the construction permit to Consolidated Edison Company provided that the Company be required to comply with the recommendations set forth in the attached reports of the Federal Water Pollution Control Administration and the Fish and Wildlife Service.

Sincerely yours,

David S. Black

Under Secretary of the Interior

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

Enclosures (2)

REPORT OF THE FISH AND WILDLIFE SERVICE ON UNIT NO. 3 OF THE INDIAN POINT NUCLEAR GENERATING STATION, AEC DOCKET No. 50-286

There are extensive commercial and sport fisheries in the Hudson River. Sport fishing is mainly for striped bass and white perch. The principal commercial fishery is for American shad. During 1964, 181,865 pounds of shad were caught in the Hudson River; approximately 149,000 pounds of this catch were taken south of the Peekskill area. Commercial fishermen also take herring, striped bass, American eel, sturgeon, white perch, tomcod, and American smelt in the river. Although there are no commercial fisheries for shellfish, some oyster setting grounds exist from the New Jersey boundary north for a distance of nine miles.

The applicant indicates that the release of radioactive wastes would not exceed maximum permissible limits prescribed in Title 10, Part 20, of the Code of Federal Regulations. Although these limits refer to maximum levels of radioactivity that can occur in drinking water for man, without resulting in any known harmful effects, operation within these limits may not always guarantee that fish and wildlife will be protected from adverse effects. If concentrations in receiving water were the only consideration, maximum permissible limits would be adequate criteria for determining the safe rate of discharge. However, radioisotopes of many elements are concentrated and stored by organisms that require these elements for their normal metabolic activities. Some organisms concentrate and store radioisotopes of elements not normally required but which are chemically similar to elements essential for metabolism. In both cases, the radionuclides are transferred from one organism to another through various levels of the food chain just as are the nonradioactive elements. These transfers may result in further concentration of radionuclides and wide dispersion from the project area, particularly by migratory fish, mammals, and birds.

In view of the above, even though the post-operational surveys at the Indian Point site indicated that there was no increase in radioactivity levels due to Indian Point Unit No. 1, we believe that pre- and post-operational surveys should be conducted by the applicant to determine any effects of Indian Point Unit No. 3. These surveys should include studies of the effects of radionuclides on selected organisms indigenous to the project area which require these waste elements or similar elements for their metabolic activities.

These surveys should be planned in cooperation with the appropriate Federal and State agencies. If it is determined from pre-operational surveys that the release of radioactive effluents at levels permitted under the Code of Federal Regulations would result in harmful concentrations of radioactivity in fish and wildlife, plans should be made to reduce the discharge of radioactivity to acceptable levels. Post-operational surveys should be conducted to evaluate the predictions based on the pre-operational surveys and to serve as a basis for reduction of radioactive levels to insure that no unforeseen damage occurs.

In view of the importance of the sport and commercial fisheries and wildlife resources of the Hudson River, it is imperative that every possible effort be made to protect these valuable resources from radioactive contamination. Therefore, it is recommended that the Consolidated Edison Company be required to:

1. Cooperate with the Secretary of the Interior, the New York Conservation Department, the New York State Department of Health, and other interested State agencies in developing plans for radiological surveys.

2. Conduct or arrange for the conduct of pre-operational radiological surveys of selected organisms indigenous to the area that concentrate and store radioactive isotopes, and of the environment including water and sediment samples. These surveys should be conducted by scientists knowledgeable in the fish and wildlife field.

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

4. Make modifications in project structures and operations to reduce the discharge of radioactive wastes to acceptable levels if it is determined in the pre-operational or the post-operational surveys that the schedule for release of radioactive effluents would result in harmful concentrations of radioactivity in fish and wildlife.

5. Conduct radiological surveys, similar to those specified in recommendation 2 above, analyze the data, and prepare and submit reports every six months during the first year of reactor operation and every six months thereafter 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 the appropriate State and Federal agencies for evaluation.

We understand it is the Commission's opinion that its regulatory authority over nuclear power plants involves only those hazards associated with radioactive materials. However, we recommend and urge that before the permit is issued, thermal pollution and any other detrimental effects to fish and wildlife which may result from plant construction and operation be called to the applicant's attention. We recommend further that the applicant be requested to disucss this matter with appropriate Federal and State conservation officials and to develop measures to minimize hazards.

We are particularly concerned over the possibility of damages to aquatic life from increase water temperatures. Studies of the influence of the heated water on the Hudson River are being conducted with the aid of a model at Alden Laboratories, Worcester Polytechnic Institute, Worcester, Massachusetts. Indications from these studies are that the discharge channel, now being extended downstream a total of 500 feet for plant number 2, will need to be extended an additional 700 feet for plant number 3 in order to assure, for plant efficiency, that water reaching the intake will not exceed ambient water temperatures by more than 2°F. The temperature rise of the cooling water will be 16°F. as it enters the discharge channel from each plant under all stages of development.

Large volumes of heated water discharged into the river could cause profound effects on the aquatic environment. Such discharges may not only be detrimental to fish life directly but may also affect these resources indirectly through changes in the ecological communities, particularly the food organisms on which fish depend. We are also concerned about the discharge of chemical wastes resulting from the control of algae, the reduction of boiler scale and the absorption of copper by the condenser cooling water.

Ecological surveys, to measure biological and ecological changes in the river, should be conducted prior to and following plant operation to measure the effect of plant operation on the biota of the river. These surveys should be planned in cooperation with the appropriate Federal and State agencies. If it is determined from the pre-operational investigations that the heated water or chemical effluent to be discharged into the Hudson River would result in changes in the environment that would be significantly detrimental to fish and wildlife, plans should be made to reduce the temperature of the effluent to acceptable levels. Post-operational surveys should be conducted to evaluate the predictions based on the pre-operational surveys and to insure that no unforeseen damage occurs.

Another potential hazard to fishery resources in the river is the cooling water intake. Unless the intake is adequately screened, fish may be drawn in and destroyed. Suitable fish protective facilities should be installed to prevent significant damage to the fishery resources.

In view of the Administration's policy to maintain, protect, and improve the quality of our environment and most particularly the water and air media, we request that the Commission urge the Consolidated Edison Company to:

1. Cooperate with the Secretary of the Interior, the New York Conservation Department, the New York State Department of Health, and other interested State agencies in developing plans for ecological surveys, initiate these surveys at least two years before reactor operation, and continue them on a regular basis or until it has been conclusively demonstrated that no significant adverse conditions exist.

2. Meet with the above mentioned Federal and State agencies at frequent intervals to discuss new plans and to evaluate results of existing surveys.

3. Construct, operate, and maintain such fish protective facilities over the intake structures as needed to prevent significant damage to fishery resources.

4. Make such modifications in project structures and operation including facilities for cooling discharge waters as may be determined necessary as a result of the pre-operational or post-operational surveys to protect the fish and wildlife resources of the area.

REPORT OF THE FEDERAL WATER POLLUTION CONTROL ADMINISTRATION ON UNIT NO. 3 OF THE INDIAN POINT NUCLEAR GENERATING STATION AEC DOCKET NO. 50-286

This is in regard to water pollution control problems and programs associated with the proposed enlargement of the Indian Point nuclear thermal-electric generating plant on the Hudson River by addition of Unit No. 3, by Consolidated Edison Company.

The Indian Point Plant is located on the east bank of the Hudson River about 24 miles above New York City. It is 22 miles downstream of the emergency New York City, Chelsea Pumping Plant on the Hudson River.

Our comments and recommendations are directed at the thermal, radiological and chemical aspects of water pollution control.

Thermal

We are particularly concerned that the large amount of heat discharged by Unit No. 3, in combination with that from Unit No. 1, already in operation, and Unit No. 2, now nearing completion, will result in serious pollution under certain conditions of a significant reach of the Hudson River Estuary.

For the reasons outlined below, we recommend that the Department enter strong objections to the proposed project unless the following specific provisions are incorporated in the project design:

(1) "The licensee shall so conduct his activities that they do not violate applicable New York State as well as Federal water quality standards, recommendations of any enforcement conference or any Hearing Board approved by the Secretary, or order of any court, all under Section 10 of the Federal Water Pollution Control Act, and other State and Federal water pollution control regulations;" and

(2) "The licensee shall provide a monitoring program acceptable to the State of New York and to the Federal Water Pollution Control Administration so as to assure that all water quality standards are met."

The declared policy of the United States set forth in the Atomic Energy Act is that "the development, use and control of atomic energy shall be directed so as to make the maximum contribution to the general welfare ..." In our judgement a failure to consider the implications of thermal and other pollution at this early stage could add unnecessarily to the cost of application of atomic energy at this site, and would be incompatible with the general welfare objectives of the Atomic Energy Act. We believe that from the standpoint of both the licensee and downstream water users greater economy will be achieved if the licensee is required to comply with Federal and State water pollution control laws as a condition of the AEC Permit to use nuclear fuel.

Executive Order 11288 provides that the Federal Government "should provide leadership in the nationwide effort to improve water quality through prevention, control, and abatement of water pollution from Federal Government activities in the United States." It calls for each agency of the Federal Government to carry out its activities, both internally and externally, in such a way as to contribute to this national effort. On this basis we believe that any permit or license issued by the Federal Government for this project should include provision(s) that the plant meet the applicable water quality standards which have been established in accordance with the Federal Water Pollution Control Act, as amended. Failure to meet these standards could result in Federal enforcement action and possible delays in the full operation of the three units, which would in turn adversely affect the return on the investment in the plant.

Standards for water quality in the Hudson River have been established by the State of New York and were approved by the Secretary of the Interior on August 17, 1967, in accordance with provisions of the Federal Water Pollution Control Act. The standards provide:

"Non-Trout Waters

"1. Mixing Zone - The mixing zone will be separately determined for each discharge so as to minimize detrimental effects. Fish and other aquatic life shall be protected from thermal blocks by providing for a minimum fifty percent stream of estuarine cross-section and/or volumetric passageway, or establishing artificial fishways where considered necessary.

"Generally, the surface water temperature shall not exceed 90°F within the mixing zone. Consideration will be given to effects of each discharge based on hydrodynamics and other factors of receiving waters.

"2. Outside Mixing Zone - Stream temperatures in excess of 86°F will not be permitted after mixing. Further, no permanent change in excess of 5°F will be permitted from naturally occurring background temperatures. In multiple discharge situations, stream capacity to meet such criteria will be apportioned among the discharges.

"3. Outside Mixing Zones: Fresh Surface Water Classes Temperature changes rate shall be limited to 2°F per hour not to exceed 9°F in any 24-hour period, further limited in that for any seven day period the average change will meet the 5°F change of background criteria stated in item 2 above.

"4. Outside Mixing Zone: Tidal Salt Water Classes Discharges shall not raise monthly means of maximum daily temperature more than 4°F from September through May, nor more than 1.5°F during June, July, and August.

"Temperature change shall not be more than 1°F per hour, not to exceed 7°F in any 24-hour period at maximum, except when natural phenomena cause these limits to be exceeded."

Water temperatures have been obtained bi-weekly at the FWPCA water quality surveillance station at Poughkeepsie, New York, about forty miles upstream from the Indian Point Plant. Maximum water temperatures have equalled or exceeded 78°F in most of the years. A maximum temperature of 80.5°F has been reported. The project description provided by the Company indicates a 16°F increase through the condensers of Unit No. 3. For the entire three units an increase of 16.4°F appears possible. Thus, a plant discharge temperature of as high as 96.9°F could occur. This would exceed the 90°F maximum permitted in mixing zones as described in part one of the previously quoted standards.

Studies of possible thermal pollution by the plant being carried out by consulting engineers were mentioned by the Company in their Preliminary Safety Analysis Report and noted by company representatives at the meeting with representatives of the Department of the Interior on August 7, 1967. The Company had expected this report to be completed by mid-September and it is understood that it is now being reviewed by their own engineering staff. The opportunity to review these studies is necessary for our appraisal of the facilities to be provided to control thermal effects of the project.

The Hudson River Estuary is at this point subject to tidal action. Under flood tide conditions, which occur twice daily, there is an extended period of slack tide and reversal of flow which would result in the accumulation of a mass of warmed water in the vicinity of the cooling water discharge. Information from other sources indicates that such masses of heated water maintain their identity for a considerable period of time and move with the tides. Information from studies by the Tennessee Valley Authority and others indicates that a considerable period of time may elapse before the temperature of such a mass of water will approach natural levels. Outside the mixing zone the temperature should not exceed 86°F. In our judgement the movement of such masses of warm water by the tides should not be considered a part of the mixing zone.

As a minimum, measures to assure that Federal Water Quality Standards as to temperature are met should include cooling towers or other means of heat dissipation to meet presently anticipated conditions. In addition, provisions should be made so that it will be possible to install additional cooling towers or other measures when this proves to be necessary.

Inclusion of pollution control requirements in the license and their consideration in the early phases of design is considered essential in obtaining appropriate design of the entire boiler-turbine-condenser system. Prevention of pollution of the environment is a necessary and proper cost of power production and failure to incorporate control measures at this early date will result in higher costs because it is generally less efficient to add control measures than to design them into the plant.

Radiological

A separate liquid waste treatment system is provided for each reactor. The collection of wastes and their batch treatment is appropriate, considering the small volumes involved. This proposed waste treatment system, together with proper operation, should provide the decontamination necessary to maintain concentrations of radionuclides in the water environment below currently accepted limits and appears to constitute a substantial effort to reduce radionuclides to the lowest practicable level. We interpret the statement in the report "...experience with the design of similar systems has shown that the expected concentrations in the discharge canal will be less than about 0.02 MPC of 10CFR20 per year of all isotopes" -- to include tritium. If this is not the correct interpretation, we recommend that the analysis requirements be expanded to include tritium.

Information on the environmental monitoring program is quite limited in the Preliminary Safety Analysis Report. The program has been acceptable for use with Unit No. 1. That program provides for monthly sampling of water in the Hudson River Estuary and for less frequent sampling of algae, fish and bottom muds. Because radionuclides can be concentrated and stored by aquatic organisms that require these or similar elements in their normal metabolic activities and then be transferred to other edible life forms or to wildlife, it is important that the marine life being monitored include types that can be expected to accumulate radionuclides. Shellfish should be monitored at seasons of maximum growth. Consideration should be given to scheduling samples in mid February, July, August, September and October so as to have a greater likelihood of detecting increasing radioactivity in the aquatic environment if it occurs.

The permit issued to the Consolidated Edison Company should require the Company to make modifications in operations and/or in project structures to reduce the discharge of radioactive wastes to levels consistent with all uses of the Hudson River if results of the monitoring program indicate that releases are hazardous to human or fish and wildlife populations of the area.

Chemical

Chemicals are generally used for the control of algae and biological fouling organisms in the condensors and cooling towers, for the reduction of scale in boilers, or for the control of corrosion in the condensers. The New York standards provide that "toxic wastes" or "deleterious substances" shall not be discharged "alone or in combination with other substances or wastes in sufficient amounts or at such temperatures as to be injurious to edible fish or shellfish or the culture or propagation thereof, or which in any manner shall adversely affect the flavor, color, odor or sanitary condition thereof..." Free residual chlorine and chromium (tri- or Hexa- valent) should be limited to 0.1 mg/l and o.05 mg/l, respectively, outside the mixing zone to prevent toxic effects in fish and fish food, and to minimize the possibility of adverse effects on the flavor of shellfish. Means to control concentrations of these chemicals or any others that are used, acceptable to the State of New York and the Federal Water Pollution Control Administration should be included in the design and operational procedure for the entire plant.

The Consolidated Edison Company has indicated its willingness to take reasonable measures to reduce to a minimum the thermal effects of its plants on adjacent waters where those effects would be adverse to good principles of conservation. Considering the complexity of the effects of wastes involved, we feel that license provisions requiring a continuing program of surveillance and appropriate corrective action as soon as surveillance data indicate it to be necessary, are essential to control the effects of pollution on the environment.

APPENDIX G

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

February 15, 1968

Dear Mr. Chairman:

The Department's letter of January 18, 1968, relative to the matter of the application by Consolidated Edison Company of New York for a permit to add a unit to the Indian Point nuclear generating station (AEC Docket No. 50-286) indicated that we would not object to the permit if the company complied with some recommendations which accompanied it. In reviewing this matter, it appears that the language we used could be misinterpreted.

In addition to the AEC construction permit, the applicant also requested a dredging permit from the Corps of Engineers in connection with this unit of the Indian Point station. Last September we requested the Corps to include in the permit the following conditions:

"The Permittee shall keep the Department of the Interior and the State of New York fully informed, by means of periodic meetings, regarding plans for and construction of the work described by Public Notice 6001, and for the construction of Unit No. 3 of the Indian Point Nuclear Power station.

"The Permittee shall make modifications of project structures and operations requested by the Secretary of the Interior for the protection of the fish and wildlife resources of the Hudson Riverway.

"The Permittee shall make modifications of project structures and of the operation of the Indian Point Nuclear Power station as necessary to comply with the applicable State or Federal water quality standards.

"The Permittee shall comply with any regulation, condition, or instruction affecting the work hereby authorized if and when issued by the State or Interstate water pollution control agency having jurisdiction to abate or prevent water pollution, or by the Federal Water Pollution Control Administration."

Consolidated Edison agreed to these conditions by letter of September 22, 1967.

Since these conditions were already agreed to by the applicant in connection with the Corps permit, we just reiterated them in the case of the AEC permit for the purposes of your records. We did not intend, as our letter may imply, that the issuance of the AEC permit be subject to the acceptance of these conditions by the applicant or that the inclusion of the conditions in the AEC permit would necessarily be appropriate.

We hope that our earlier letter did not cause any undue delay in the issuance of the AEC permit.

Sincerely yours,

David S. Black

David S. Black
Under Secretary

Hon. Glenn T. Seaborg
Chairman
Atomic Energy Commission
Washington, D. C. 20545

APPENDIX H

ADEQUACY OF THE STRUCTURAL CRITERIA FOR

Indian Point Nuclear Generating Unit No. 3
Consolidated Edison Company of New York, Inc.

by

N. M. Newmark, W. J. Hall and A. J. Hendron

INTRODUCTION

This report is concerned with the adequacy of the containment structures and components for the Indian Point Nuclear Generating Unit No. 3 for which application for a construction permit has been made to the U. S. 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, Westchester County, New York; the site is about 24 miles N of the New York City boundary. Indian Point Unit No. 3 will be built adjacent to Indian Point Units 1 and 2.

Specifically this report is concerned with design criteria that determine the ability of the containment system and Class I equipment and piping, as well as Class II structures and equipment, to withstand an Operating Basis Earthquake of 0.10g maximum horizontal ground acceleration acting simultaneously with other loads forming the basis of the design. The facility is also to be designed to withstand a Design Basis Earthquake of 0.15g maximum horizontal ground acceleration to the extent of insuring safe shutdown and containment.

The report is based on information and criteria set forth in the Preliminary Safety Analysis Report (PSAR) and supplements thereto listed at the end of this report. Also, we have participated in discussions with the applicant and the AEC Regulatory Staff concerning the design of this unit.

DESCRIPTION OF FACILITY

The Indian Point Nuclear Generating Unit No. 3 is described in the PSAR as consisting of a pressurized water reactor nuclear steam supply system designed and furnished by Westinghouse Electric Corporation under a turnkey contract. The plant is to be designed for a power output of 3025 Mwt (965.3 Mwe net).

The reactor containment structure is a reinforced concrete vertical right cylinder with a nearly flat base and a hemispherical dome. The cylinder has an inside diameter of 135 ft. and a wall thickness of 4 ft.-6 in.; the spring-line of the dome begins at a height of 148 ft. above the liner on the bottom of the containment structure. The dome has an inside radius equal to the inside radius of the cylinder, and a thickness of 3 ft.-6 in. The change in thickness at the discontinuity between the cylinder sidewall and the dome occurs on the outer surface of the containment structure.

The inside of the containment structure is provided with a liner which is one-quarter inch thick at the bottom, one-half inch thick in the first three courses of the cylindrical wall except at penetrations where it is three-quarters inch thick, three-eighths inch thick for the remaining portions of the cylindrical wall, and one-half inch thick in the dome. The liner anchorages will consist of one-half inch diameter bent welding studs.

Diagonal shear reinforcing will be employed to resist earthquake shears for the full height of the wall and a distance above the springline into the dome until a point is reached where the dome liner can resist the total shear. The geological description for the site notes that Unit No. 3 will be located on a hard limestone which is jointed, but which provides a solid bed for the plant foundation. The foundation investigation descriptions indicate that the limestone is not cavernous. The report by the consulting geologists contained in Section 1.7 of the PSAR indicates that there are no major geologic faults extending through the site, nor close to it.

SOURCES OF STRESSES IN CONTAINMENT STRUCTURES AND CLASS I COMPONENTS

The reactor containment structure is to be designed for the following loadings and conditions: dead load; live load including snow, ice, construction and equipment loadings); a design accident temperature of about 247°F and a pressure of 47 psig; an internal containment test pressure of 54 psig; a basic design wind loading of 30 psf; tornado loadings associated with a 300 mph tangential wind velocity, a translational velocity of 60 mph, a pressure drop of 3 psi from inside to outside, and associated missiles; and earthquake loading as described next.

The seismic design is to be made for an Operating Basis Earthquake with a maximum horizontal ground acceleration of 0.10g and a Design Basis Earthquake with the maximum horizontal ground acceleration of 0.15g.

The criteria controlling the design of piping and reactor internals for seismic loadings are presented in various places in the PSAR but particularly in Section 15 of Supplement 1.

COMMENTS ON ADEQUACY OF DESIGN

Foundations and Dams

The major facilities structure for Indian Point Unit No. 3 are described as being founded directly on competent bedrock. On the basis of the information presented in the PSAR and supplements, the foundation conditions appear acceptable.

It is noted in Section 11 of Supplement 5 that the possibility of a flood caused by a maximum rainfall coincident with a dam failure will also be investigated. We should like to have the opportunity to examine the results of this study when it becomes available at a later date.

In the course of the construction review of the design of Indian Point No. 2 there was some discussion concerning the increased lateral forces in the transverse direction arising from the action of the crushed-rock backfill against the structure. It was noted that the backfill was not at the same elevation around the entire structure, and thus the lateral force distribution on the structure arising from both dead load and seismic loading are not uniformly distributed circumferentially. Although the crushed rock backfill is mentioned in the Indian Point 3 application, the applicant has advised DRL that backfill will not be placed directly against the Indian Point No. 3 containment wall. It is assumed that adequate clearances will be maintained between the containment wall and any surrounding material.

Seismic Design and Criteria

We are in agreement with the earthquake loading criteria selected for the seismic design, namely that associated with an Operating Basis Earthquake of 0.10g maximum horizontal ground acceleration, and a Design Basis Earthquake of 0.15g maximum horizontal ground acceleration. These earthquake design criteria are in agreement with those given by the U. S. Coast and Geodetic Survey (Ref. 4).

The criteria for the vertical earthquake component are stated in the PSAR to be 0.05g vertically for the Operating Basis Earthquake and 0.10g for the Design Basis Earthquake. We concur in these values for this plant.

We find no discussion in the PSAR concerning the provisions for combining the vertical and horizontal seismic effects with other appropriate loadings, except so far as they appear in the factored load combination expressions. The applicant has informed DRL that it will consider the effects to act simultaneously, and will combine the effects directly and linearly, as appropriate, in accordance with the agreement that was reached in the design of Indian Point Unit No. 2.

The response spectra that are to be used in the analysis are given in Figs. A-1 and A-2 of Vol. 2, and in Figs. 5-7 and 5-8 of Section 5, of the PSAR. These response spectra are for the Operating Basis Earthquake for horizontal and vertical excitation. Spectra have not been presented for the Design Basis Earthquake, but the applicant has advised DRL that the spectra for this earthquake will be scaled upwards appropriately from the Operating Basis spectra. We concur in the spectra to be employed.

The damping values to be used in the seismic analysis are listed at several points in the PSAR and supplements, as for example in the answer to Question 2.7 of Supplement 2, and we concur in the values listed.

The method of dynamic analysis is described in several places in the PSAR, for example in the answer to Question 2.7, but is not described in enough detail to evaluate it completely. It is noted that a modal analysis procedure will be employed and that the total response is computed as the root-mean square sum of the responses of the individual modes. It would be out recommendation that a standard modal analysis procedure be employed which takes account of structural rocking, lateral translation, and shearing, flexural and torsional distortion of the structure, as may be appropriate. With proper attention to the damping and coupling of the various modes, and the procedures by which the various modal forces, displacements, and accelerations are combined, it should be possible to arrive at reasonable and consistent values of stress, shear, moment, etc., to be employed in design.

The design criteria to be employed for Class II structures and equipment were not noted to be explicitly stated in the PSAR and Supplements and it would be our recommendation that critical items falling in this category be designed for approximately one-half of the provisions in the Uniform Building Code for Zone 3.

General Design Criteria

The factored load combinations to be employed in design of the containment structure are given in Section 5.1.2.4 of the PSAR. The loading combinations appear acceptable to us and it is noted that for these load factor combinations the resistance will correspond to "elastic, tolerable strain behavior." It is also noted that the liner will be designed to assure that the strains in the liner do not exceed the guaranteed yield point at the factored loads, and that sufficient anchorage will be provided to assure elastic stability of the liner. These criteria are acceptable to us.

The applicant has advised DRL that the criteria for handling concrete shear values in the containment vessel will be carried out in line with the discussions that were held in conjunction with the design criteria for Indian Point Unit No. 2, and that the criteria for design of the cranes will be in accordance with those discussed and reported for the Indian Point 2 design.

Liner

The design of the liner receives attention in numerous places in the PSAR and supplements and it appears that the criteria in general are satisfactory. However, in the answer to Question 5.2 of Supplement 2 it is noted that the liner is to be erected true and plumb with certain limitations on deviations, and one of the possible deviations is a 2 inch local buckle. The applicant has advised DRL that this deviation will be limited to a curve over a distance equivalent to one panel. We recommend that this criterion be examined further during the design phase in conjunction with overall ovalling criteria.

Penetrations

The design criteria for penetrations receives attention throughout the PSAR and supplements and especially in the answer to Question 2.11 of Supplement 4. The methods of analysis described and the tentative reinforcing details presented appear acceptable so long as there is assurance of adequate strength and ductility in the reinforcing ring structure, and in the transition region adjacent to the stiffening ring. It would be our recommendation that a careful measurement and observation program be carried out at the time of the pressure test of the containment vessel to help provide assurance of the adequacy of the design of the large penetrations.

Base Slab

The proposed design of the base slab for the containment structure was reviewed several times in connection with the request for an exemption to permit the construction to proceed at an early date. From the data presented in the PSAR, discussions with the applicant, and our study and evaluation, we believe that the proposed design scheme and criteria can lead to an acceptable base slab design.

Class I Piping, Equipment, Vessels and Reactor Internals

The design criteria for these items are summarized in Section 15 of Supplement 1 of the PSAR and are to be carried out in accordance with criteria presented in Westinghouse Report WCAP-5890, Rev. 1, with modifications as noted in Section 15 of the PSAR. Additional information concerning the stress limit curves to be employed with this design are given in Section 13 of Supplement 5. We are in agreement with the proposed design criteria. The applicant has advised DRL that the criteria just cited supersede the criteria given in Appendix A of Vol. 2, Part B, of the PSAR.

Controls, Instrumentation, Batteries, Etc.

Only general information is noted in the PSAR concerning the seismic design criteria elements of control, instrumentation, batteries, etc. It would be our recommendation that criteria for these items be examined in detail during the design phases, to insure that the items can withstand the forces, motions, and tilt that might be associated with an earthquake.

Quality Control and Inspection

Quality control, inspection and acceptance procedures are discussed throughout the PSAR and supplements. If properly executed, the procedures outlined appear acceptable to us.

CONCLUDING COMMENTS

On the basis of the information presented in the PSAR and supplements, and in keeping with the design goals of providing serviceable structures and components with a reserve of strength and ductility, we believe that the design outlined for the containment and other Class I structures and equipment and Class II structures and components, can provide an adequate margin of safety for seismic resistance. However, in the report we have offered comments concerning various aspects of design.

W. J. Hall

REFERENCES

1. "Preliminary Safety Analysis Report -- Description of Site and Environment," Consolidated Edison Company of New York, Inc., Indian Point Nuclear Generating Unit No. 3, Exhibit B, Vol. 1, AEC Docket No. 50-286.

2. "Preliminary Safety Analysis Report -- Plant Design Description and Safety Analysis," Consolidated Edison Company of New York, Inc., Indian Point Nuclear Generating Unit No. 3, Exhibit B, Vol. 2, Parts A and B, AEC Docket 50-286.

3. "Preliminary Safety Analysis Report," Consolidated Edison Company of New York, Inc., Indian Point Nuclear Generating Unit No. 3, Supplements 1, 2, 4, 5, AEC Docket No. 50-286.

4. "Report on the Site Seismicity of the Indian Point, New York Area," U. S. Coast and Geodetic Survey, Rockville, Maryland, 20852.