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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. 2

PEEKSKILL, NEW YORK

DOCKET NO. 50-247

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I. Introduction

On December 6, 1965, the Consolidated Edison Company of New York, Inc., applied to the Atomic Energy Commission for a license to construct and operate a 2758 megawatt thermal (MWt) nuclear facility to be located at the Indian Point site near Peekskill, New York. The pressurized water reactor (PWR) will be the second nuclear unit to be located at this site. The existing PWR, Indian Nuclear Generating Unit No. 1, has a thermal rating of 615 MWt.

The technical safety review of the proposed design of the facility, which has been conducted by the staff of the Commission's Division of Reactor Licensing, has been based on the report, Indian Point Nuclear Generating Unit No. 2 - Preliminary Safety Analysis Report, and five supplements thereto (hereafter referred to as the Report). In the course of its review of the material in the Report, the Division of Reactor Licensing staff has held a number of meetings with representatives of the applicant and Westinghouse Electric Corporation to discuss the site and the proposed facility and to clarify the technical material submitted. In addition, the Commission's Advisory Committee on Reactor Safeguards (ACRS) has also considered this project and met and discussed it with the applicant and the Commission's staff. The material discussed at each of these meetings and the technical correspondence are summarized as follows:

1. January 17-18, 1966 - Representatives of the applicant and the Commission's staff reviewed the contents of the Report. As a result of this meeting, questions were sent by the Division of Reactor Licensing to the applicant on February 28, 1966, requesting clarification of a number of technical areas. Written answers to these questions (First Supplement) were provided by the applicant on March 31, 1966.

2. March 30, 1966 - A subcommittee of the ACRS met with the applicant and the Commission's staff at the Indian Point site. The material provided in the Report was discussed.

3. April 4, 1966 - The ACRS met with the applicant and the Commission's staff to discuss the overall design of the facility and particular design features of safety significance.

4. May 2, 1966 - Representatives of the applicant and the Commission's staff met to discuss the material submitted in the First Supplement and, in particular, the engineered safeguards systems and reactivity transients.

5. May 3, 1966 - A subcommittee of the ACRS met with the applicant and the Commission's staff. The potential consequences of various postulated accidents and considerations related to locating two nuclear facilities at one site were discussed.

6. May 6, 1966 - The ACRS met with the applicant and the Commission's staff to continue the discussion of technical matters [related to the safety of the proposed facility]. As a result of the foregoing meetings and our continued review of the design of the proposed facility, additional information was requested by letter dated May 11, 1966. Answers were provided by the applicant (Second Supplement) on May 31, 1966.

7. May 19, 1966 - Representatives of the applicant and the Commission's staff met to discuss the preliminary results of analyses to be provided in the Second Supplement.

8. May 26, 1966 - Dr. Nathan M. Newmark and Dr. William J. Hall, the Commission's consultants on seismic design, reviewed and discussed the proposed seismic design criteria of the facility with representatives of the applicant's

architect engineer (United Engineers and Constructors). During the course of this meeting, the applicant agreed to provide additional information related to the seismic design of the facility. This material was supplied in the Third Supplement on June 20, 1966.

9. June 23, 1966 - A subcommittee of the ACRS met with the applicant and the Commission's staff. The potential courses of loss-of-coolant accidents and various features of the engineered safeguard systems were discussed.

10. July 15, 1966 - The ACRS met with the applicant and the Commission's staff to discuss the operation of the core cooling systems that are provided to mitigate the potential consequences of various piping failures in the primary system. As a result of this meeting, the applicant proposed several core cooling system piping modifications which are discussed in the Fourth Supplement provided on July 25, 1966.

11. August 4, 1966 - The ACRS met with the applicant and the Commission's staff to complete the discussion of the safety of the proposed facility. Following the meeting, the ACRS reported its views of this proposed facility to the Commission by letter dated August 16, 1966, a copy of which is attached as Appendix A.

A construction permit for the Indian Point Nuclear Generating Unit No. 2 would be the first step in the regulatory process which would continue throughout the lifetime of the facility. Prior to issuing an operating license for the facility, the final design would be thoroughly evaluated by the Commission's staff and ACRS to determine that all of the Commission's safety requirements have been met. The plant would then be operated only in accordance with the Commission's regulations under the continued scrutiny of the Commission's staff.

II. Facility Design

Indian Point Unit No. 2 will be a 2758 megawatt thermal (MWt) pressurized water facility with an estimated gross electrical output of 916 megawatts (MWe). Although the turbine has a calculated gross capacity of 1021 MWe, the applicant states that operation above 916 MWe is not planned. Thus, the analyses presented by the applicant are based on the highest planned power level for this facility.

The reactor will be fueled with uranium dioxide (UO_2) sintered pellets sealed in 12-foot long zircaloy fuel rods. Each fuel assembly will contain 204 fuel rods, and the reactor core will consist of 193 fuel assemblies. The active core will contain 104 tons of UO_2 plus about 22 tons of zircaloy. The nuclear core will be contained within a pressure vessel designed for a pressure of 2485 psig. The primary coolant will be circulated through the nuclear core and the four steam generators by four 90,000 gpm primary coolant pumps. Steam formed in the steam generators will be piped to the turbine generator. (Chapters 3 and 4 in the Report.)

The containment structure, within which the reactor vessel, steam generators, primary coolant pumps, and other primary system equipment will be located, will be a reinforced concrete structure which is similar in concept to the containment vessel being constructed for the Connecticut Yankee facility at Haddam Neck, Connecticut. The containment is designed to withstand the pressures and temperatures that would occur in the unlikely event of a failure of the largest primary coolant line and to retain radioactive fission products that might be released as a consequence of this as well as lesser accidents. In view of the relatively high population density near the site and the large size of the reactor, the design objective of the containment vessel is to have negligible outleakage under accident conditions. This is achieved by a penetration pressurization system, a weld channel pressurization system, and a fluid line seal water system. (Chapter 5 in the Report.)

An emergency cooling system (Safety Injection System) will provide borated water for immediate and continued cooling of the fuel assemblies in the unlikely event of any loss of coolant accident up to and including the rupture of the largest primary coolant line. In addition, the Containment Spray System and the Air Recirculation System within the containment vessel will provide for containment depressurization by cooling the containment atmosphere and will remove radioactive fission products which might be released from the fuel as a consequence of an accident. (Chapter 6 in the Report.)

Inasmuch as the applicant has provided extensive details concerning the design of the facility in the Preliminary Safety Analysis Report, additional detailed description of the facility design is not given in this analysis.

III. Site Characteristics

Chapter I of Volume I of the Report contains a comprehensive description of the proposed site. The following sections summarize the prominent features.

Population Distribution

The Indian Point site comprises 250 acres owned by the Consolidated Edison Company of New York, Inc., and is located on the eastern shore of the Hudson River in Westchester County, New York. The site is located 2.5 miles from the center of Peekskill, N.Y., and approximately 24 miles north of New York City.

The current Peekskill population is 19,000 with an anticipated growth to 30,000 by 1985. The cumulative population for 1960 and the anticipated growth by 1980 in the vicinity of the site is as follows:

Distance	1960	1980
0.5 mile	46	100
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
15	326,930	670,210

This distribution indicates a population density in the vicinity of the proposed site as high as any considered heretofore.

The Commission's Regulation, Reactor Site Criteria, 10 CFR 100, provides guidelines for the maximum permissible off-site doses under accident conditions at the minimum exclusion distance (distance to the site boundary) 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 calculated low population distance.

The minimum exclusion distance for the Indian Point 2 site is 0.32 miles. Based on the population distribution in the vicinity of the site, the staff considers that the outer boundary of the low population zone is coincident with the nearest boundary of Peekskill, 0.87 miles. However, since the applicant has assumed a low population distance of only 0.67 miles, the staff used this distance in its evaluation of potential off-site doses in the unlikely event of a major loss-of-coolant accident. As discussed in the "Accident Analysis" section of this report, our calculations indicate that Part 100 Exposure Criteria are satisfied.

Meteorology

The diffusion climatology for the Indian Point site has been determined by on-site measurements in conjunction with the operation of Unit No. 1. The meteorological information included in the application has been reviewed by the U. S. Weather Bureau. It concluded that the atmospheric dispersion factors used by the applicant for short-term and long-term accidental releases of radioactivity are realistic in view of the meteorological conditions observed at the site. Accordingly, these same atmospheric dispersion factors have been used by the staff in its evaluation of potential off-site doses. The reports of the U. S. Weather Bureau are attached as Appendices B, B-1, and B-2.

Geology and Hydrology

Review of the geology of the proposed site indicates that the proposed unit 2 will be located on limestone which has a bearing capability of up to 50 tons per square foot. A typical maximum bearing load for a facility such as Unit 2 would not exceed 5 tons per square foot. The limestone is jointed and as such is permeable. Ground water flow is toward the river since the ground water table in the hills surrounding the site is at a high elevation. Thus, any leakage from the proposed facility would travel toward the river rather than toward existing water supplies. The general nature of the bedrock indicates that there are no unrelieved residual stresses and there are no identifiable geologic structures which could be expected to localize faulting in the immediate vicinity of the site.

The water flow of the Hudson River in the vicinity of the site is controlled principally by tides. The peak tidal flow is estimated at 80 million gallons per minute 80% of the time. In the region of the river affected by the plant cooling water discharge, the flow is estimated to be 9-million gallons per minute. This assures good mixing with the cooling water discharge flow.

As with flow, flooding at the site is influenced principally by tides. The maximum flood height experienced at the site has been 7.4 feet which is well below the basement elevation of the proposed facility

In the unlikely event of an accidental release of radioactive materials to the river, it is possible for the radioactivity to be transported upstream to the Chelsea pumping station (distance - 22 miles) by the tidal flow. The travel time to the pumping station would involve at least several tidal cycles (probably more than 5) and many orders of magnitude dilution would occur. The long transit time would allow ample time for monitoring the movement of the radioactivity and to take appropriate corrective action should it be necessary. Routine discharge of radioactivity into the river at the point of discharge will not exceed the drinking water levels prescribed by 10 CFR Part 20.

The United States Geological Survey was requested by the AEC staff to review the geological and hydrological aspects of the site. Its report is attached as Appendix C.

The Fish and Wildlife Service, Department of the Interior, has also reported on related aspects of Indian Point Unit No. 2. We have been advised that the Service is of the opinion that radioactivity released to the river during operation of this facility would not be expected to have any adverse effects on marine life in the river and that plans for control and dispersal of radioactive liquid waste are adequate to protect fish and wildlife in the vicinity of the proposed facility. The report of the Fish and Wildlife Service is attached as Appendix D. See also Appendix D-1, a letter from the Director of Regulation to the Commissioner, Fish and Wildlife Service, stating that the Atomic Energy Commission has no jurisdiction to consider thermal and other non-radiological effects of licensed activities.

Seismology

The U. S. Coast and Geodetic Survey (USC&GS) has evaluated the seismicity of the area and has recommended that those components important to safety be designed to withstand an acceleration of 0.1-g in the period range of 0.3 to 0.6 seconds without the loss of function. The applicant's seismic design criterion conforms to the USC&GS recommendation. The report of the USC&GS is attached as Appendix F.

IV. Important Safety Considerations

Indian Point Unit No. 2 is similar in general design and operating objectives to the Brookwood and Connecticut Yankee facilities in that each is a Westinghouse design pressurized water facility contained in a reinforced concrete containment vessel. Each employs low enrichment UO_2 fuel rods in pelleted form, and use of rod cluster control in conjunction with boron chemical shim. All are generally similar in the means proposed to achieve normal stable operation, to hold anticipated operating transients to acceptable limits, to provide emergency cooling for the core and containment, and to limit the consequences of credible accidents.

However, there are a number of respects in which this facility differs from Brookwood and Connecticut Yankee, the more important of which are the following:

1. The population distribution in the vicinity of the site for the Indian Point 2 facility is higher than that of the other facilities. To compensate, the applicant has proposed a containment and engineered safeguards systems which are more extensive than that provided at facilities in less populated areas.

a. The design objective of the containment vessel is to have negligible outleakage under accident conditions. To meet this objective, the Penetration Pressurization System (PPS) and the Isolation Valve Seal Water System (IVSWS) have been provided to preclude outleakage at all containment locations where leakage could be expected. These potential leakage paths include:

(1) Containment liner seam welds (PPS).

(2) Electrical and piping penetrations (PPS).

(3) Personnel air locks (PPS).

(4) Ventilation purge-duct penetrations (PPS).

(5) Equipment door flange (PPS).

(6) Spent fuel transfer tube (PPS).

(7) Fluid-carrying pipes that enter the containment (IVSWS).

b. Even though negligible leakage under accident conditions is anticipated, two independent means have been provided to remove the radioactive airborne fission product, iodine, from the containment atmosphere. These are the Air Recirculation System using activated charcoal filters (also to be used in Brookwood and Connecticut Yankee) and the Containment Spray System which uses sodium thiosulphate in the spray water as a reagent to aid removal of elemental forms of iodine. A containment spray system is installed at both Brookwood and Connecticut Yankee for containment cooling, but the sodium thiosulphate additive is not used.

c. A Safety Injection System with more flexibility than either the Brookwood or Connecticut Yankee systems has been provided. This system contains divided injection headers and additional injection points into the primary system to provide increased reliability.

d. The components of the reactor core and containment cooling systems required for long-term cooling following a major accident are all located within the containment structure, thereby confining all radioactive water in the containment building. For both Brookwood and Connecticut Yankee the comparable components are not within the containment structure. See Criterion 18 for further details.

2. The Indian Point fuel rods will operate at somewhat higher specific power (up to 20.7 kw/ft) and central fuel temperature (up to 4250°F) than the other facilities. However, sufficient margin is provided with respect to these parameters and significant fuel failure is not expected to occur under steady state or transient conditions. This is discussed in more detail under Criterion 6.

3. The moderator temperature and void coefficients will be positive during a portion of the initial fuel cycle. This results from the high boron concentration required for this period of operation. These coefficients provide a positive reactivity feedback under accident conditions, and somewhat lessen the safety margin provided by the negative Doppler coefficient. Although analyses

presently available indicate that operation in this manner for the initial cycle can be accomplished safely, the staff will review this item again when the detailed design of the core is fixed. If additional safety margin should be required, mechanisms, such as solid burnable poison rods, are available to reduce the coefficient. This item is discussed further in Criterion 7.

4. Two pressurized water reactors will be located at the Indian Point site. Our review has shown no interaction which could affect the safe operation of the two facilities.

5. The Indian Point II facility will contain three 50% capacity emergency diesel generators to provide power for the engineered safeguards and other vital equipment in the event of a complete loss of off-site power.

V. Conformance of Indian Point Nuclear Generating Unit No. 2 Design to Staff's General Design Criteria

Design Criteria

The following safety analysis of Indian Point II has been organized under the framework of the "General Design Criteria for Nuclear Power Construction Permits" as published for comment by the Commission on November 22, 1965.

Criterion 1

Those features of reactor facilities which are essential to the prevention of accidents or to the mitigation of their consequences must be designed, fabricated, and erected to:

(a) Quality standards that reflect the importance of the safety function to be performed. It should be recognized, in this respect, that design codes commonly used for non-nuclear applications may not be adequate.

Three barriers which prevent significant release of fission products from the reactor fuel to the environment are incorporated in the Indian Point II design:

1. The fuel element cladding provides the initial barrier and will be designed considering the effect on zircaloy of hydrogen embrittlement, internal fission gas pressure, thermal expansion, and uncertainties in fabrication. To assure high quality, fuel rods will be subjected to chemical analysis, tensile tests, corrosion tests, dimensional inspection, X-ray of welds, ultrasonic tests, and helium leak tests.

2. The primary coolant system will be designed in accordance with applicable codes; i.e., ASME Boiler and Pressure Vessel Code, Section III, for the pressure vessels, and the ASA Code for Pressure Piping, B 31.1 for the piping. In the case of the reactor vessel, which is being fabricated by Combustion Engineering Company, the applicant has outlined in detail the quality control procedures, the testing during fabrication, the acceptance testing, the capability for periodic inspection, and the NDT shift surveillance program. The applicant is retaining the services of United States Testing Company, which will perform independent checks to assure proper quality control during fabrication of the reactor vessel. The ACRS recommended that the design and fabrication techniques for the entire primary system be further reviewed to provide greater assurance of highest system quality, and that in-service inspection possibilities and detection of incipient trouble be carefully considered. The staff will continue to review these areas as the design and construction of this facility proceeds.

The increase in the nil ductility transition (NDT) temperature with fast neutron exposure over the service lifetime of the reactor vessel is one means by which an increase in brittleness and susceptibility to failure can occur. The design criteria for the Indian Point II reactor vessel is a maximum shift of 275°F in NDT, which corresponds to a fast neutron exposure of 3.7×10^{19} n/cm^2. The anticipated fast neutron exposure during the service life of the vessel is estimated to be 0.85×10^{19} n/cm^2, which corresponds to an NDT temperature shift of 160°F. Thus, considerably less than the design fast neutron exposure will be encountered during the service life of the reactor vessel.

The initial NDT will be measured on specimens of the reactor vessel base material. Subsequently, additional specimens of base material will be irradiated in eight specimen capsules located between the active core and the reactor vessel wall. The fast neutron flux at the location of the capsules will be higher than that experienced by the reactor vessel wall. Thus, the specimens will be representative of the reactor vessel at a later time in life. The NDT of these specimens will be measured periodically.

During fabrication of the reactor vessel, radiographic, ultrasonic, magnetic particle, and liquid penetrant examinations of the material will be conducted to assure that the vessel meets acceptance standards. The reactor vessel head closure studs will also receive comparable inspections both during fabrication and subsequently during refueling when the studs are removed from the vessel.

We believe that the quality control and surveillance programs for the reactor vessel outlined by the applicant are adequate, but that improved methods of in-service inspection which would parallel as closely as is practicable the inspections given to pressure vessels in non-nuclear applications should be developed. The applicant, as a member of the Empire States Atomic

Development Associates (ESADA), is exploring possible methods to regularly examine the pressure vessel for defects in body and cladding after installation and service.

3. The containment vessel will be designed and tested to conform to applicable parts of the "Building Code Requirements for Reinforced Concrete" (ACI 318-63). The liner will be reinforced at each penetration according to the rules set forth in the ASME Code, Section VIII UG-36. Quality control aspects and final design details of the containment will be reviewed by the staff as they are developed.

4. An important adjunct to the containment vessel is the various engineered safeguards. These, too, will be designed in conformance with applicable portions of the ASME Boiler and Pressure Vessel Code and the ASA Code for Pressure Piping. The design function of the engineered safeguards is reviewed under Criteria 2, 10, 18 and 22.

The American Standards Association and the Institute of Electronic and Electrical Engineers are actively engaged in the development of standards governing the design, testing, and installation of reactor protection systems. Some AEC staff members are participating directly in this effort to ensure the creation of quality standards and the proper implementation thereof. Evaluation of the Indian Point Unit No. 2 reactor protection system will be based on such standards, as they are proposed or adopted. (Section 5.1.2)

Based on the foregoing, we believe that Criterion 1(a) is satisfied.

Criterion 1(b)

Those features of reactor facilities which are essential to the prevention of accidents or to the mitigation of their consequences must be designed, fabricated and erected to:

> (b) Performance standards that will enable the facility to withstand, without loss of the capability to protect the public, the additional forces imposed by the most severe earthquakes, flooding conditions, winds ice, and other natural phenomena anticipated at the proposed site.

The effects of severe environmental conditions at the Indian Point II site have been considered and taken into account in the design of those portions of the facility important to safety. As such, the containment vessel will be designed to withstand a wind loading of 30 psf (110 mph) coincident with the temperature and pressure conditions associated with a major rupture of the primary coolant system. In addition, the containment vessel design criteria will include expected ice and snow loading conditions. As noted previously, there is no flooding problem.

The applicant's proposed seismic design criteria, outlined below, are in conformance with a maximum horizontal ground acceleration of 0.1g, which is the design acceleration recommended by the U. S. Coast & Geodetic Survey for systems and structures important to safety. (Appendix D) The seismic design criteria for those components which are necessary for the safe, orderly shutdown of the facility (designated as Class I by the applicant), are:

1. For the containment vessel:

a. Stresses in all structural members shall not exceed 0.95 yield under the combined dead load, 47 psig internal pressure (including the temperatures associated with this pressure) and 0.15g horizontal and 0.1g vertical earthquake accelerations acting simultaneously.

b. Stresses in all structural members shall not exceed 0.95 yield with a 70.5 psig accident internal pressure (including the temperature associated with this pressure).

2. All other structures and equipment important to safety shall remain functional under the same loading conditions stated in (a) above. For those structures or components which are allowed to exceed yield, deformation shall not exceed 0.4%.

A report by our seismic design consultant, Nathan M. Newmark, confirms that the proposed seismic design criteria are adequate. This report is attached as Appendix E. (Section 5.1; Supplement 2, Question 9; Supplement 3)

Based on the foregoing considerations, we believe that Criterion 1(b) is satisfied.

Criterion 2

Provisions must be included to limit the extent and the consequences of credible chemical reactions that could cause or materially augment the release of significant amounts of fission products from the facility.

There are three potential chemical reactions which could augment fission product release from the containment after a loss-of-coolant accident by adding energy and thus extending the period of time that the containment would be pressurized. These are: (1) zirconium-water reaction between the zircaloy fuel cladding and steam present in the reactor vessel following a loss-of-coolant accident, (2) oxidation of the hydrogen resulting from the zirconium-water reaction, and (3) combustion of the activated charcoal beds in the halogen removal filters of the air recirculation system due to excessive temperatures created by decay heat from the adsorbed fission products.

To limit the extent of the zirconium-water reaction and the availability of hydrogen, which might oxidize, the facility is provided with safety injection capability consisting of the following:

a) Three high-head safety injection pumps - 400 gpm @ 2500 ft.

b) Two low-head safety injection pumps - 3000 gpm @ 280 ft.

c) Two residual heat removal pumps - 3000 gpm @ 280 ft.

These pumps, in various combination depending upon the size of the primary system rupture inject borated water from the refueling water storage tank to each of the hot and cold legs of the primary cooling system. Following any primary system rupture up to and including a double-ended break of the largest reactor coolant system pipe, these systems will be designed such that in the worst case a significant metal-water reaction would not occur.

High concentration of radioactive iodine on the halogen removal filter plus failure of a recirculating fan after a major loss-of-coolant accident could result in temperatures in excess of the ignition temperature of the charcoal bed. If the bed were to burn, the entrained halogens would be released to the containment. To prevent such an occurrence, a dousing system will be provided for each filter bank. Each dousing system will receive water from the containment spray headers. The criteria for the number, flow and location of spray nozzles have not yet been specified, but the stated design criteria is that the system will maintain the filter surface wet. Two temperature sensors will be provided in each filter bank to sense temperature increases in any part of the bank. If a fan should fail, the applicant has estimated that, depending on the ignition temperature assumed, filter ignition might occur within 100 seconds and that a containment pressure increase up to 15 psi in

in 1/2-hour might result. The dousing system will be designed to be started manually by the operator within 60 seconds, which appears to be reasonable considering the operations which must be performed. Although many details are not yet available, we believe that design of such a system is within the realm of standard engineering practice, and thus in this respect Criterion 2 is satisfied. In addition, we believe that the design criteria for the proposed Safety Injection System satisfies Criterion 2 with regard to potential zirconium-water reactions and oxidation of accompanying hydrogen.
(Section 12.2.3; Supplement 1, Questions 4a, b, c and 6)

Criterion 3

Protection must be provided against possibilities for damage of the safeguarding features of the facility by missiles generated through equipment failures inside the containment.

The applicant has stated the criterion that the containment, containment liner, engineered safeguards and components required to maintain containment integrity shall be protected against loss of function due to damage by the following missiles:

a) All valve stems up to and including the largest size to be used.

b) All valves up to and including the largest size to be used.

c) Pieces of metal up to 6-inches thick.

d) All valve bonnets.

e) All instrument thimbles.

f) Various type and sizes of nuts and bolts.

g) Pieces of pipe up to 10-inch diameter striking broadside or end on.

h) Complete control rod drive mechanisms.

i) Reactor vessel head bolts.

Protection against these missiles will be provided by either surrounding critical components with reinforced concrete or locating the components behind the massive polar crane support wall which surrounds the primary system. In addition, a missile shield will be located above the control rod drive housings.

Since the detailed design of this type of shield is in accordance with standard engineering practice, we believe that the missile shielding as outlined above satisfies Criterion 3. (Section 5.1.2; Supplement 1, Questions 19a, b)

Criterion 4

The reactor must be designed to accommodate, without fuel failure or primary system damage, deviations from steady state norm that might be occasioned by abnormal yet anticipated transient events such as tripping of the turbine-generator and loss of power to the reactor recirculation system pumps.

Operational transients will be safely accommodated in the Indian Point II design by the proper sizing of system components and by the selection of proper setpoints for the operation of system control and protective instrumentation. The plant has been designed to accommodate without fuel damage the complete loss of pumping head in all four primary coolant loops and any loss of load transient.

For loss of flow, the design will incorporate primary coolant pumps with sufficient rotational inertia to provide for primary coolant flow coastdown. Although DNB may occur during the transient coastdown condition, the power level by then will be low enough so that clad failure will not occur. The reactor will be protected by low flow and low pumping power trips.

For loss of load from power levels greater than 50% the reactor power will be automatically reduced by control rod motion. Protection will be supplied by a logic circuit which will cause the reactor to scram on unsafe combinations of power and primary pressure. During final design, these transients will be analyzed using the positive moderator coefficient at the beginning of core life and also using the negative moderator coefficient at the end of core life. The positive moderator coefficient will be experienced only during a portion of the first core since high boron concentrations are required to compensate for the high reactivity resulting from the core containing all initially unirradiated fuel. The Doppler coefficient, however, represents the primary mechanism in terminating power transients. The reactivity that can be added by the positive moderator coefficient will be limited to low values both in rate and in the total amount of insertion by the use of fixed burnable poison if final analysis of the first core indicates that added safety margin is desired. (Section 12.1; Supplement 1, Question 2)

On the basis of the information available, we believe that Criterion 4 is satisfied.

<u>Criterion 5</u>

The reactor must be designed so that power or process variable oscillations or transients that could cause fuel failure or primary system damage are not possible or can be readily suppressed.

The applicant has analyzed the ability of the reactor control and protection system to control the oscillations resulting from variation of coolant temperature within the dead band of the temperature controller and from spatial xenon oscillations. Variations in average coolant temperature provide negative feedback and thus the reactor is stable during that portion of core life in which the moderator temperature coefficient of reactivity is negative. When the coefficient is positive, rod motion must compensate for the positive feedback. The applicant has calculated that the maximum power change associated with the temperature oscillation is 2% per minute. Since the plant is required to follow ramp load changes of 5% per minute, this is well within the capability of the control system.

Spatial instability due to xenon oscillations is a function of the uniformity of core power distribution. The applicant has stated that the power distribution for the first loading is such that the core will be stable early in core life. Partial insertion of control rods will increase power peaking and thus improve core stability. However, as burnup progresses with control rods removed, axial flux peaking is reduced. At the end of core life calculations indicate that the core may exhibit xenon oscillations with little or no damping. The applicant has stated that this oscillation can be observed by the out-of-core nuclear detectors, since the long ion chambers are subdivided into upper and lower chambers. In-core monitors are also available to more accurately measure flux asymmetry. If observed, rod patterns can be adjusted to maintain the core within safe limits since the long oscillation period provides time for an assessment of the consequences of the oscillation. If necessary, power could be reduced or the reactor shut down to protect the core. The applicant intends to continue studies of such instability. Further experimental information should be available from the San Onofre and Connecticut Yankee facilities by the time the Indian Point II facility is to operate.

The control system is designed to accept 10 percent step load increases and ramp increases of 5 percent per minute between 15 percent and 100 percent of normal power without reactor trip. The plant is also designed to safely accommodate complete loss of electrical load from rated power. In this case, the reactor will trip and secondary pressurizer safety valves would open for brief periods to prevent overpressurization.

At the present time, there is little experience on operating pressurized water reactors with positive or zero temperature coefficients of reactivity. The limited experience to date has been derived at the SELNI reactor. This experience has provided support of the analytical techniques used to predict moderator temperature coefficients in plants of this type. In addition, prior to startup of the Indian Point II facility, detailed information to verify analytical techniques should be available from the San Onofre reactor. (Volume 2, Appendix C; Supplement 1, Question 1, Writeup 5, Question 14f)

We believe that the design and analysis performed for the reactor system demonstrates that Criterion 5 is satisfied.

Criterion 6

Clad fuel must be designed to accommodate throughout its design lifetime all normal and abnormal modes of anticipated reactor operation, including the design overpower condition, without experiencing significant cladding failures. Unclad or vented fuels must be designed with the similar objective of providing control over fission products. For unclad and vented solid fuels, normal and abnormal modes of anticipated reactor operation must be achieved without exceeding design release rates of fission products from the fuel over core lifetime.

The following criteria will be met by the fuel rods during anticipated operational modes, including the overpower conditions, assuming the worst combination of instrument errors at any time during core life:

1. The minimum departure from nucleate boiling ratio (DNBR) as determined by the W-3 correlation will be greater than or equal to 1.3.

2. The maximum fuel center temperature will be below the melting point of UO_2 using the Westinghouse Atomic Power Division design curve for thermal conductivity of UO_2 vs. temperature.

3. Stresses in the zirconium clad will be less than the yield strength.

Although these criteria insure fulfilling the requirements of Criterion 6, we have extended our evaluation to include an assessment of the safety margin available before large numbers of fuel rods exceed design limitations. For Criterion No. 1 above, using the statistical W-3 DNB correlation, a detailed core power distribution, and the worst combination of instrument error at the worst time in core life, we have calculated that 7 of 40,000 fuel rods would experience DNB at the applicant's assumed overpower condition (112% of full power). The number of rods which would conceivably fail at the overpower trip is less than 0.03% of the total number of rods in the core. It should be noted that at DNB steam blanketing will occur and attendant high clad temperature with local clad failure may result. However, massive failure of fuel rods will not occur.

In addition to the previous calculation, the staff performed an "uncertainty" analysis of the DNB situation by arbitrarily assuming certain percentage errors in primary coolant flow rate, radial flux peaking factor, and DNB correlation (W-3). Our results show that a rapid rise in the number of failed fuel rods would not occur for the 112% overpower condition until large uncertainties in the factors mentioned above (up to 30%) are assumed. In consideration of the foregoing and the degree of conservatism in the applicant's DNB calculation, we do not believe there is any reasonable basis for expecting uncertainties which would result in large numbers of rods experiencing DNB.

For Criterion No. 2 above, we have examined the WAPD thermal conductivity vs. temperature design curve and additional data for predicting fuel melting of fuel rods containing UO_2 pellets. Based on this examination, we concur with the applicant that fuel center melting will not occur for all anticipated modes of operation. We have also investigated the possibility of uncertainties in this calculation and have found that the probability of substantial center fuel melting is very small.

For Criterion No. 3 above, a design criterion is that the internal gas pressure within the fuelrods due to the expected equilibrium burnup will be less than nominal external pressure throughout core life. This insures that the clad stresses are below those at the beginning of core life and that the yield strength of the zirconium will not be exceeded at operating conditions. (Section 3.2.2; Supplement 1, Question 5)

Based on the foregoing, we believe that Criterion 6 is satisfied.

Criterion 7

The maximum reactivity worth of control rods or elements and the rates with which reactivity can be inserted must be held to values such that no single credible mechanical or electrical control system malfunction could cause a reactivity transient capable of damaging the primary system or causing significant fuel failure.

The reactor will contain 53 cluster control assemblies having a total reactivity worth of 0.07 delta k/k. The control drive mechanisms will be of the magnetic latch type so designed that withdrawal speed will be limited to a maximum of 15 inches per minute, which the applicant states corresponds to a maximum reactivity insertion rate of about 2×10^{-4} delta k/sec. The reactor overpower and variable low pressure protective systems will be set to terminate any excursion caused by such an accidental rod withdrawal before core damage is incurred. The rod control system is designed to be immune to a single electrical failure which would cause withdrawal of the control rods in excess of the above rate.

Although the applicant considers a control rod ejection accident incredible since each control rod drive housing will be proof tested to 6300 psi prior to operation, the selection of control rod groupings will limit the highest worth rod to a value such that if it were ejected, the resultant calculated fuel temperature during the excursion would be sufficiently low to prevent gross fuel dispersion in the coolant and would not cause an excessive pressure rise within the primary system. Additional analyses of the sensitivity of this accident to the positive moderator coefficient will be performed by the applicant when final core and control rod parameters are established. These studies will be reviewed by the staff prior to reactor operation to determine the need for limiting the positive moderator coefficient. If necessary, means are available, such as the use of solid burnable poisons, to suitably limit the moderator coefficient. (Section 12.1.1; Supplement 2, Question 6)

We believe that the foregoing satisfies Criterion 7.

Criterion 8

Reactivity shutdown capability must be provided to make and hold the core subcritical from any credible operating condition with any one control element at its position of highest reactivity.

The maximum excess reactivity expected for the Indian Point II core is 0.275 and occurs at the cold, clean condition at the beginning of life of the initial core. This excess reactivity will be controlled by a combination of control rods and soluble neutron absorber (boron). A total of 53 Rod Cluster Control (RCC) Assemblies are provided with a total worth of 0.07. The remaining excess reactivity will be controlled by boron chemical shim.

These RCC assemblies are divided into two categories, a control group and a shutdown group. The control group, used in combination with chemical shim, provides control of reactivity changes throughout the life of the core at power conditions. This group of RCC assemblies is used to compensate for short-term reactivity changes at power that might be produced due to variations in reactor power requirements or in coolant temperature. The chemical shim control is used to compensate for the more slowly occurring changes in reactivity throughout core life such as those due to fuel depletion and fission product buildup.

The shutdown group is provided to supplement the control group of RCC assemblies and to hold the reactor subcritical by at least 0.01 following trip from any credible operating condition to the hot, zero power condition assuming the most reactive RCC assembly remains in the fully withdrawn position.

The boron injection system is available to maintain the reactor subcritical during cooldown and when cold. This injection system is discussed in Criterion 9. (Section 3.2.1, 7.2)

In our opinion the proposed design satisfies Criterion 8.

<u>Criterion 9</u>

Backup reactivity shutdown capability must be provided that is independent of normal reactivity control provisions. This system must have the capability to shut down the reactor from any operating condition.

The Chemical and Volume Control System (CVCS) which injects borated water via the charging system into the primary system provides a redundant reactivity control mechanism which is independent of the control rods. Any time the plant is at power, the quantity of boric acid ready for injection will always exceed the quantity required for a normal cold shutdown. Boric acid can be pumped from the boric acid tanks by either of two boric acid pumps to the suction of either of two charging pumps which will inject into the primary system. Boric acid can be injected by one pump at a rate which will shut the reactor down in less than fifteen minutes with no rods inserted. Assuming no control rod motion following shutdown, sufficient boric acid will be available to bring the reactor to cold shutdown conditions and to compensate for xenon decay. (Section 9.1; Supplement 1, Question 2)

On the basis of the above, we believe that the intent of Criterion 9 is satisfied.

Criterion 10

Heat removal systems must be provided which are capable of accommodating core decay heat under all anticipated abnormal and credible accident conditions, such as isolation from the main condenser and complete or partial loss of primary coolant from the reactor.

For failures that result in the loss of primary coolant, the safety injection system is provided to limit potential core damage following primary system piping failures of all sizes. The system contains three high-head pumps and four low-head pumps that deliver borated safety injection water from the refueling water storage tank (capacity - 320,000 gallons) to each of the main cold- and hot-leg pipes near the reactor vessel. This system, which is also provided to limit possible zirconium-water reaction, is also discussed under Criterion 2.

Protection against small pipe failures is provided by the high-head pumps. For such failures, the reactor system would depressurize slowly and the high-head pumps would prevent the core from becoming completely uncovered. The remaining coolant would provide continuous cooling of the fuel rods and core integrity would be maintained.

These pumps deliver the safety injection water into two headers (each pump capacity is 400 gpm at 2500 ft.) which each inject into two of the cold-leg pipes (total of 4 pipes). One high-head pump is connected to each header and the third pump is arranged so it can deliver to either header. The high-head pumps are located in the primary auxiliary building and the headers are within the containment, but are protected by the polar crane support wall from possible missiles generated by failure of components in the primary system (see Criterion 3). This system is redundant in pumps, headers, and injection points into the primary system, and provides essentially two independent means of injecting borated water into the reactor vessel in the event of small pipe failures.

Protection against large piping failures is provided by the low-head pumps. For such a failure the primary system would be depressurized and voided of coolant rapidly (about 10 seconds) and a high flow rate would be required to quickly re-cover the exposed fuel rods and limit possible core damage. To achieve this objective, two low-head safety injection pumps and two residual heat removal pumps have been provided. The characteristics of these pumps are similar; each delivers 3000 gpm at 280 feet. Each of these pumps delivers the borated safety injection water to two headers that are missile protected. Each header injects the water into all four hot-leg pipes near the reactor vessel. As discussed with the high-head system, the number of pumps, headers, injection points, and general system arrangement provides essentially two independent means of injecting borated water into the reactor vessel in the event of large primary coolant system failures.

Because the reactor vessel would be rapidly voided of coolant following large piping failures, the fuel pellets and the fuel rod cladding temperatures would increase until a significant volume of water could be injected into the vessel and re-cover the core. The applicant originally stated in the Report that a 5% zirconium-water reactor could occur if two of the three diesels provided were available to power the pumps of the safety injection system. However, more recent calculations have indicated that a zirconium-water reaction of about 10% might occur under the same conditions. Under these circumstances, about 20% of the fuel pellets would be exposed and could fall to the bottom of the reactor vessel. In our opinion, this amount of core damage would appear to be excessive, even though calculations indicate that the integrity of the pressure vessel would not be jeopardized.

In consideration of the foregoing, the ACRS has recommended, and the staff agrees, that the flow capacity of the safety injection system should be increased and/or improvements should be made in other system characteristics, such as pump discharge pressure. In addition, the forces to be expected within the reactor vessel in the event of primary system failures must be carefully examined to ensure that the capability of the safety injection system is not impaired under these extreme conditions. We believe that these matters can be resolved during construction of the facility.

As a backup to the safety injection system, the applicant has proposed a magnesium oxide-lined metal vessel located within the containment near the bottom of the reactor cavity into which the core would fall should melt-through of the reactor vessel occur. This vessel would provide additional assurance that containment integrity would be maintained following a serious reactor accident. Design details of the crucible and their theoretical and experimental bases will be reviewed by the staff as they are developed.

For situations that do not involve failure of the primary system, such as a complete loss of power to the main coolant pumps, the energy in the primary system can be safely dissipated by natural circulation of the primary water through the steam generators. A steam-driven pump is provided which would supply sufficient cooling water to the steam generators from the condensate storage tank to maintain water level above the tube sheet. This tank will contain a minimum supply of water equivalent to steam generation from 24 hours of decay heat removal at hot shutdown conditions. Additional sources of water are available if required.

For conditions wherein the primary system pressure has been reduced below about 100 psi, the residual heat removal system can transfer decay heat to the component cooling loop which in turn transfers it to the service water system and thence to the river. (Section 6.2.2, 9.2; Supplement 1, Question 18; Supplement 2, Question 2; Supplement 3, Question 9; Supplement 4, Part 3; Supplement 5, Part 1)

In view of the foregoing, we believe that the cooling systems proposed can be designed to satisfy Criterion 10.

Criterion 11

Components of the primary coolant and containment systems must be designed and operated so that no substantial pressure or thermal stress will be imposed on the structural materials unless the temperatures are well above the nil-ductility temperatures. For ferritic materials of the coolant envelope and the containment, minimum temperatures are NDT + 60°F and NDT + 30°F, respectively.

The applicant has stated that he will specify a design transition temperature for the vessel which will be a minimum of NDT + 60°F at all times. The vessel will be designed to permit an NDT shift of 275°F, which corresponds to an integrated fast neutron flux exposure of 3.7×10^{19}, the anticipated fast neutron exposure is 0.85 x 10^{19} n/cm^2. Operation below the design transition temperature will be limited with respect to pressure by vessel stress criteria. An equivalent pressure limit will also be included to compensate for thermal stresses during vessel heatup or cooldown.

A surveillance program will be conducted to experimentally determine radiation induced damage in pressure vessel material as a function of irradiation. (See Criterion 1a)

The applicant has stated that the containment vessel will be designed so that it is not susceptible to a low temperature brittle failure. It should be noted that the NDT + 30°F criterion was not intended for prestressed or reinforced concrete containment vessels. Based upon our review of the proposed design of the containment and the advice of our consultants, we believe that there will be no potential low temperature brittle failure problem for the containment. (Section 4.1; Supplement 1, Question 3)

We therefore believe that Criterion 11 is satisfied.

Criterion 12

Capability for control rod insertion under abnormal conditions must be provided.

The Indian Point II reactor will utilize a rod cluster control system (RCC), which consists of 20 control rods per cluster. Each cluster is provided with a magnetic latch-type control rod mechanism which allows the cluster to fall by gravity on loss of magnet power. There is no rod drive-down capability provided. All components of this system are considered Class I for seismic design purposes. The individual rods are fully guided above and in the core region. Extensive experiments with the RCC system have demonstrated that there is sufficient clearance, even if considerable misalignment of guide tubes has occurred, to allow full insertion. These experiments have demonstrated that the RCC system is mechanically feasible for use in reactor systems.

Should it become impossible to insert any of the RCC assemblies, the reactor could be made subcritical by use of the boron injection system as discussed in Criterion 9. (Section 3.2.3; Supplement 1, Question 14b)

We believe that the intent of Criterion 12 is satisfied by the proposed design.

Criterion 13

The reactor facility must be provided with a control room from which all actions can be controlled or monitored as necessary to maintain safe operational status of the plant at all times. The control room must be provided with adequate protection to permit occupancy under the conditions described in Criterion 17 below, and with the means to shut down the plant and maintain it in a safe condition if such accident were to be experienced.

The control room will be equipped with the controls and instrumentation necessary to operate the reactor and turbine generator under normal and accident conditions, and the controls and instrumentation of other plant variables which require constant operator attention. All engineered safeguards can be operated from the control room. In addition, those instruments required to monitor the post-accident containment environment and proper operation of required systems are also displayed in the control room. These monitors include containment pressure, activity level, sump levels, safety injection pump discharge pressures, valve positions, and heat exchanger temperatures.

The shielding provided by the containment structures of Units I and II is sufficient to limit the infinite thyroid and whole body doses to 3.0 and 1.5 rem, respectively, following the postulated maximum accident in either facility. These calculations are conservative in that additional shielding provided by the structured parts of the control rooms was not considered. (Supplement 1, Question 17)

Based on the above, we believe that Criterion 13 is satisfied.

Criterion 14

Means must be included in the control room to show the relative reactivity status of the reactor such as position indication of mechanical rods or concentrations of chemical poisons.

Rod position indication is available from each of the 53 rod cluster control assemblies and is displayed in the control room. In addition, indications of primary coolant temperature, coolant pressure, coolant flow, and neutron flux are also available to the operators. These collectively provide the operator with information on the reactivity status of the reactor.

Although boron concentration in the primary system is not continuously monitored or displayed, periodic samples of primary water will be taken for analysis of boron content. Since the effect of changes in boron concentration will be reflected in the parameters listed above which are readily available to the operator in the control room, we do not believe continuous boron monitoring is necessary. (Section 7.2; Supplement 1, Question 2)

Based on the foregoing, we believe Criterion 14 is satisfied.

Criterion 15

A reliable reactor protection system must be provided to automatically initiate appropriate action to prevent safety limits from being exceeded. Capability must be provided for testing functional operability of the system and for determining that no components or circuit failure has occurred. For instruments and control systems in vital areas where the potential consequences of failure require redundancy, the redundant channels must be independent and must be capable of being tested to determine that they remain independent. Sufficient redundancy must be provided that failure or removal from service of a single component or channel will not inhibit necessary safety action when required. These criteria should, where applicable, be satisfied by the instrumentation associated with containment closure and isolation systems, after heat removal and core cooling systems, systems to prevent cold-slug accidents, and other vital systems, as well as the reactor nuclear and process safety system.

All nuclear and process system parameters capable of scramming the reactor will be monitored by redundant instrumentation. Amplifiers associated with such instruments drive relays, the contacts of which are connected through three complete and independent logic chains which open the two redundant scram breakers. Two of these chains are energized during a scram condition and respectively actuate the shunt trip coils of the two breakers. The third chain is de-energized during scram and is connected to the two undervoltage coils (wired in parallel) of the breakers. Further, two contacts on each breaker interrupt both sides of the DC lines feeding the rod mechanisms. Thus, each logic chain is independently capable of scramming the reactor.

The proposed protection system is redundant and immune to individual faults occurring at the instrumentation, logic circuits and scram breakers. Fail-safe systems, in terms of partial or complete loss of electric power, are inherent in the proposed design.

The manual scram switch has three contacts, two of which respectively apply voltage directly to the shunt trip coils. The third contact interrupts the undervoltage coils. Thus, no single circuit fault can disable the manual scram function.

Coincidence as well as redundancy is used throughout the protection system. Thus, there is capability for testing, at least up to the logic chains. There are no provisions for testing the logic chains and breakers at power. Periodic testing of these features with the reactor shut down will be required.

The containment isolation signal is derived from three pressure sensors (2/3 logic). The circuitry following these sensors will be fail-safe with respect to voltage and/or instrument air loss. We understand that revisions are being made to the applicant's criteria which will require that, in some cases, isolation be accomplished by two automatic valves. We also understand that the circuits actuating such valves will be independent and immune to single failures.

Safety injection signals will be derived from a coincidence of pressurizer-low-level (2/3 logic) and pressurizer-low-pressure (2/3 logic) signals. The associated circuits are redundant and immune to single failures. The sensor circuits fail safely, i.e., they call for safety injection, in the event the instruments are carried away by an accident. Voltage loss at certain of the circuits will preclude, rather than initiate, injection. This is a design feature intended to prevent inadvertent injection. However, manual actuation capability will be provided.

The applicant has stated: "The principal criterion of control station design and layout is that all controls, instrumentation displays and alarms required for the safe operation and shutdown of the plant are readily available to the operators in the control room." The applicant has not specified which displays and alarms are required to fulfill this criterion; however, we believe that the control station design and layout can be made to conform to the applicant's criterion which is acceptable. (Section 7; Supplement 1, Question 14)

We have concluded that the applicant's criteria are in accord with Criterion 15.

Criterion 16

The vital instrumentation systems of Criterion 15 must be designed so that no credible combination of circumstances can interfere with the performance of a safety function when it is needed. In particular, the effect of influences common to redundant channels which are intended to be independent must not negate the operability of a safety system. The effects of gross disconnection of the system, loss of energy (electric power, instrument air), and adverse environment (heat from loss of instrument cooling, extreme cold, and fire, steam, water, etc.) must cause the system to go into its safest state (fail-safe) or be demonstrably tolerable on some other basis.

Complete loss of AC voltage will initiate reactor scram through circuits within the instrumentation which trip when their respective channels are de-energized. Loss of DC voltage de-energizes the rod coils directly.

A voltage loss at the logic circuits feeding the scram breakers will scram the reactor via the undervoltage coils. The containment isolation circuits downstream of the pressure sensors will fail safely in the event of voltage loss. Loss of instrument air at containment isolation valves will, in most cases, drive them to the "close" position. (Exceptions are valves which must remain open temporarily under accident conditions).

Also of concern are the potential adverse effects of fires originating in the control and safety system wiring and/or within the control room itself. In our opinion, a direct, analytical safety analysis relating to the possibility of reactivity excursions resulting from such fires is, in practice, impossible due to the random nature of fire damage and the nearly infinite variety of possible circuit faults (some "unsafe," some "safe") which could result. However, we believe that the natural complexity of reactor control and safety systems coupled with a redundant, fail-safe design firmly based on applicable criteria, accepted codes, etc., constitutes the best defense against serious fire-induced accidents.

In this connection, a literature search was conducted with the assistance of the computer facilities at the Nuclear Safety Information Center (NSIC) at Oak Ridge National Laboratory, to study the historical record of such excursions. NSIC has informed us that they were unable to find any records of incidents involving reactor damage as a result of fire-induced excursions. (Section 7; Supplement 1, Question 14)

Based on the foregoing considerations, we believe that Criterion 16 is satisfied.

Criterion 17

The containment structure, including access openings and penetrations, must be designed and fabricated to accommodate or dissipate without failure the pressures and temperatures associated with the largest credible energy release including the effects of credible metal-water or other chemical reactions uninhibited by active quenching systems. If part of the primary coolant system is outside the primary reactor containment, appropriate safeguards must be provided for that part if necessary, to protect the health and safety of the public, in case of an accidental rupture in that part of the system. The appropriateness of safeguards such as isolation valves, additional containment, etc., will depend on environmental and population conditions surrounding the site.

The Indian Point II containment vessel is similar to that of the Connecticut-Yankee vessel in that it is a totally reinforced concrete vessel with cylindrical walls, a flat base (with sump pit), and a hemispherical dome. The cylindrical walls are 4.5 feet thick below grade and taper to 3.5 feet thick where the dome joins the wall. The dome is also 3.5 feet thick. The free volume of the containment vessel is 2.6-million cubic feet, and the design pressure of the vessel is 47 psig.

The containment design criteria relative to material stresses are expressed in terms of load factors above design loadings and are discussed in Criterion 1(b). This approach provides assurance that the containment will be designed considering the pressures and temperature associated with a major loss-of-coolant accident acting simultaneously with the maximum earthquake or wind loading. Of particular importance in assuring a leak-tight vapor container is the method of securing the liner to the concrete so that excessive stresses will not cause increased leakage. All portions of the liner, and especially those in the vicinity of penetrations, will be designed to consider the effects of all temperature, pressure, and earthquake loads. Our structural design consultants, Drs. N. M. Newmark and W. J. Hall have considered the design of the containment structure in their report (see Appendix E) and have concluded that the principal structure and components designed for containment, and the other essential parts of the facility, will provide an adequate margin of safety for seismic motions.

In the Indian Point II design, the recirculation loop of the engineered safeguards system to be used for long-term cooling following a loss-of-coolant accident is located entirely within the containment vessel. Operation of this loop is required after all of the borated water from the refueling water storage tank has been pumped into the containment vessel as containment spray or core injection water. This water will collect in the pit below the reactor vessel and small sumps near the edge of the containment. It will be pumped by low-head safety injection pumps through the residual heat exchangers located within the containment. After cooling, the borated water can be directed back into the reactor vessel to remove core decay heat or be sprayed into the containment vessel to effect pressure reduction as necessary. Although provision is made in the design to circulate sump water outside the containment with the residual heat removal pumps, such action would be required only if, (1) a small leak has occurred in the primary system and water must be recirculated through the high-head safety injection pumps after the refueling water storage tank is emptied (in this case, the fission product inventory of the spilled coolant will be low since significant fuel failures should not have occurred), or (2) the redundant components of the internal recirculation loop have failed.

In order to determine the adequacy of the containment structure and associated engineered safeguards to accommodate the pressures associated with the largest credible energy release resulting from rupture of the largest primary coolant system pipe, the applicant presented studies of the containment pressure after an assumed loss-of-coolant accident. These studies considered all credible energy sources available and a variety of situations including cases where (1) all components of the safety injection system, containments spray, and fan-coolers operate, (2) no engineering safeguards operate, (3) safeguards are driven by two of the three emergency diesel generators, (4) delayed initiation of safety injection occurs with and without some engineering safe-guards in operation, and (5) delayed hydrogen burning is assumed. The calculated metal-water reactions ranged from 1% for case (1) to 43% in 2300 seconds for case (2). Based on the foregoing as well as independent calculations using simplified assumptions, the staff has concluded:

1. With one containment cooling system operating (all five cooling-fans or both containment spray pumps), the containment can tolerate (internal pressure will not exceed 47 psi) the assumed metal-water reaction that would result if the safety injection system were inoperable if hydrogen is burned as produced.

2. With two containment cooling systems operating (all five cooling-fans and both containment spray pumps), the containment can tolerate (pressure will not exceed 47 psi) the calculated metal-water reaction that would result if the safety injection system were inoperable, if delayed hydrogen burning occurs.

Because of mixing and stoichiometric requirements, complete delayed hydrogen burning is highly improbable. In addition, it is reasonable to assume that the safety injection system would limit the metal-water reaction below the rate assumed for evaluation purposes.

In summary, the staff believes that the Indian Point II containment and engineered safeguards are sufficient in capacity and redundancy to limit the maximum credible pressure of the containment to the design value of 47 psig following a major loss-of-coolant accident, and thus we believe that Criterion 17 is satisfied. (Section 5, 12; Supplement 1, Question 4; Supplement 2, Questions 2, 7)

Criterion 18

Provisions must be made for the removal of heat from within the containment structure as necessary to maintain the integrity of the structure under the conditions described in Criterion 17 above. If engineered safeguards are needed to prevent containment vessel failure due to heat released under such conditions, at least two independent systems must be provided, preferably of different principles. Backup equipment (e.g. water and power systems) to such engineered safeguards must also be redundant.

Two independent heat removal systems of different principles, each capable in itself of maintaining containment pressure below 47 psig, are provided in the Indian Point II design. These are: (1) the air recirculation system, and (2) the containment spray system. (Another heat removal system, safety injection to the core, is provided to prevent core meltdown and limit the metal-water reaction, but is assumed to be inoperable in sizing the containment depressurization system components).

The air recirculation system consists of five motor driven centrifugal fans and cooling coil assemblies which will be provided to recirculate and cool the containment air during normal power operation, during any other time when the containment vessel is closed, and during an accident. Each of the five ventilation units consists of a demister, a cooling coil, a roughing filter, an absolute filter, a fan and a charcoal filter in that order. Under accident conditions each unit will have a capacity of 65,000 cfm. The charcoal filter is normally by-passed; but when the filter mode is required under accident conditions, motor operated louvers will automatically direct flow to the charcoal filters. Simultaneous operation of two butterfly valves in the filter by-pass lines is also required. In this mode the containment atmosphere is cooled and radioactive halogens are absorbed on the charcoal.

Normal containment temperature, about 100-120°F, is maintained during full power operation with some of the air recirculation cooling systems in service. The ventilation system will be designed for continuous operation without interruption during and following the loss of all primary coolant and during all postulated subsequent energy additions to the containment vessel. If electrical power to the site is lost, the diesel generators may be used to power these units. The fan motors are designed to operate continuously under accident conditions of about 271°F in a steam-air mixture with a density of 0.175 lb/cu. ft. at 47 psig for 48 hours and for 10 days at 5-10 psig conditions. In addition, each unit could operate under a pressure of 70.5 psig and a temperature of 298°F for one hour. Each of the five ventilation cooling system units is capable of removing more than 72,000,000 BTU/hr. Five air recirculation cooling units operating alone would limit the maximum containment pressure following the major loss of coolant accident to less than 47 psig as discussed in Criterion 17.

The ventilation ducts and equipment will be protected from missiles, are located in positions within the containment to achieve good mixing, and are located away from the primary system. The system will be designed to withstand the sudden release of the primary system energy and energy from chemical reactions without failure due to shock or pressure waves. This is accomplished through the use of dampers along the ducts which would open at slight overpressure.

The containment spray system, an independent backup of different principle than the fan-cooler units, will be designed to reduce containment pressure and remove iodine from the containment atmosphere by a washing action. Sodium thiosulfate will be in the spray water to improve retention of iodine in the water. The criterion for heat removal capacity with both spray pumps operating is to be at least equivalent to the heat removal capacity of five fan-cooler units.

Water will be pumped by two containment spray pumps at the rate of 2,600 gpm each from the refueling water storage tank, which holds 320,000 gallons of 3,000 gpm borated water, through separate lines to two spray headers within the containment vessel. The spent hot spray water will collect in the containment sumps. If it is necessary to continue containment spray after the borated water supply is exhausted, the water can be drawn from the containment sump by low-head safety injection pumps and passed through the residual heat exchangers for cooling. The foregoing components are located inside the containment vessel. After cooling, the water can be directed back through the containment spray headers. The capacity of each residual heat exchanger is sufficient to maintain the containment pressure below 47 psig after the refueling water storage tank is emptied. If multiple component failures occur in the internal recirculation loop, spray can still be effected by the backup residual heat pumps located outside the containment.

The service water system provides cooling water to the air recirculation units and to the component cooling loop which in turn cools the residual heat exchangers in the safety injection system. Six electric-motor-driven centrifugal pumps will take suction directly from the river and discharge in triplicate to two service water headers which supply water to separate lines to the cooling component headers. The service water headers for each safeguard system (i.e., fan-cooler or component cooling heat exchanger) is valved so that half of each system is on each of the two headers. The capacity of any two of the service water pumps will be sufficient to supply the entire requirement for cooling water of the containment air coolers and the component cooling heat exchanger during a major loss of primary coolant accident and recovery. These pumps can be operated using electrical power from any two of the three auxiliary diesel-generators, if required.

Cooling between the residual heat exchangers and the service water system is provided by the component cooling system. This is a redundant system and can be supplied with emergency power.

With regard to the long term heat removal requirements and capabilities, we have considered the following, (1) at given times after the accident, what cooling equipment must operate, and (2) at given times after the accident, how much time is available before design pressure of the containment vessel is exceeded if this cooling equipment becomes inoperable. Since the heat removal capability of a fan-cooler is approximately 20,000 BTU/sec at 47 psig, one cooler's heat removal capability is equal to the decay heat generation of the fuel at 16 hours after the accident, two coolers' at 2 hours and three coolers' at 1/2 hour after the accident. Thus, within one day following the accident only a single fan-cooler must operate to balance heat transfer into the containment vapor phase. If safety injection and the associated residual heat removal system is assumed to function, this equipment alone will adequately remove all core decay heat, and operation of the fan-coolers or containment spray would not be required. It would require approximately 400 days following the major loss-of-coolant accident before the containment structure would be capable of transferring the decay heat produced by the core to the environment without benefit of some active safeguards.

If one assumes all cooling equipment becomes inoperable, the time available before exceeding the containment design pressure depends on the time after the accident and on the containment pressure at the time of loss of the equipment. The following table summarizes the allowable time for outage of all heat removal equipment as a function of time after the major loss-of-coolant accident, and illustrates the importance of availability of cooling equipment for extended periods after the accident.

Long Term Heat Removal Requirements

Time After Accident at which Cooling Equipment becomes Inoperable, Days	Time Required to Reach Containment Design Pressure, Hours		
	Starting Point, psig		
	0	10	30
1	6	4	1
10	12	8	2
30	18	12	3

Based on the foregoing considerations, we believe the safeguard systems provided are of sufficient redundancy and capacity to assure containment integrity under all credible circumstances, and that Criterion 18 is satisfied. (Sections 5.3, 6.2; Supplement 1, Questions 6, 8: Supplement 2, Question 3)

Criterion 19

The maximum integrated leakage from the containment structure under the conditions described in Criterion 17 above must meet the site exposure criteria set forth in 10 CFR 100. The containment structure must be designed so that the containment can be leak tested at least to design pressure conditions after completion and installation of all penetrations, and the leakage rate measured over a suitable period to verify its conformance with required performance. The plant must be designed for later tests at suitable pressures.

The design objective of the Indian Point II containment is to have negligible leakage under all credible accident conditions. Since most containment leakage paths occur at the penetrations, a penetration pressurization system will be installed to preclude leakage of the containment atmosphere at these locations. This system provides a pressurized zone at each penetration liner weld that is maintained at a pressure of about 50 psig which is slightly above the containment design pressure. Pressurized zones are also provided for the containment liner seam welds, the two ventilation-purge duct penetrations, the personnel air locks, the equipment door flange and the spent fuel transfer tube. A system of this type in which the penetration is continuously pressurized has not been previously proposed for use in other licensed facilities.

The penetration pressurization system is divided into four sub-systems that are provided with two independent sources of pressurized gas to assure that the pressurized zones can be maintained. The four sub-systems are normally connected to instrument air. Two compressors are used, although only one is required to maintain pressurization at the maximum allowable leakage rate of the pressurization system. Each sub-system contains an air receiver than can maintain pressurization for four hours if both air compressors should fail.

The backup pressurization source for each sub-system is nitrogen bottles that provide a minimum supply of gas for 24 hours at the maximum allowable leakage rate.

The gas makeup rate for each sub-system is continuously monitored and recorded (with a high rate alarm setting) in the control room to assure that the leakage for the system is within specified limits. A tentative upper limit for long-term uncorrected air consumption has been set at 0.2% of the containment volume per day (sum of four headers). This limit has been set on the assumption that half of the leakage would be into the containment and half would be out of the containment. In addition, all penetrations that are outside the containment and in accessible areas will have a locally mounted pressure gage. The pressurized zones that are entirely within the containment (e.g., each containment liner seam weld channel) or in inaccessible areas will be provided with low pressure alarms and individual indicating lights in the control room. Isolation valves are also provided for each pressurized zone to enable further identification of leaking penetrations.

An isolation valve seal water system will also be provided to preclude potential leakage paths through piping systems that penetrate the containment liner. These pipes could present a possible source of leakage if radioactive gas were to leak through the isolation valves provided for each line. This system is designed to provide a high pressure water seal at the outer isolation valve or a water-leg at high pressure such that the pressure at the valve or in the line is maintained above the containment accident pressure. This design feature should eliminate this potential source of leakage. This system is not provided for closed piping systems inside the containment that do not connect to the containment atmosphere or a source of radioactivity, and which are provided with missile protection.

The containment will be leak tested initially at a pressure of 47 psig and at some lower pressure. The leakage will be determined by the reference volume method. The specified leak rate for acceptance of the containment after completion of construction is 0.1%/day at 47 psig, without benefit of the penetration pressurization system. These tests will be performed with the penetration pressurization system vented to the atmosphere and not pressurized. Subsequent leak rate tests can be performed at any pressure up to the design pressure when the plant is not in operation and precautions are taken to protect equipment and instruments from damage. The pressure for periodic leakage rate tests of the containment will be set at the operating license stage of review.

In addition to providing systems that are designed to preclude out-leakage from the containment, two independent systems have been provided to remove halogens available for leakage from within the containment. These systems are the internal air filtration system (activated charcoal filters) and the containment spray system. The effectiveness of these systems is discussed in the "Accident Analysis" section of this report.

To provide a basis for evaluating the adequacy of the proposed containment system, the staff has calculated the potential off-site doses following an assumed major loss of coolant accident. These calculations are discussed in the "Accident Analysis" section and demonstrate that the guideline exposures recommended in 10 CFR 100 are satisfied with nominal assumed halogen removal efficiencies and a containment outleakage rate up to 0.1%/day at 47 psig. On the basis of the foregoing and this evaluation, we believe that Criterion 19 is satisfied. (Sections 5.2, 5.4, 5.5; Supplement 1, Questions 7, 19 e; Supplement 3, Question 2; Supplement 4, Part 2)

Criterion 20

All containment structure penetrations subject to failure such as resilient seals and expansion bellows must be designed and constructed so that leak-tightness can be demonstrated at design pressure at any time throughout the operating life of the reactor.

All containment penetrations are provided with a double barrier against leakage from the containment atmosphere to the outside. Penetrations which incorporate resilient seals, such as personnel airlock doors and equipment hatches, use double gaskets which are continuously pressurized during all reactor operation by the outside air supply system. Hot pipe penetrations, which require expansion bellows, will also be continuously pressurized in several compartments of the penetration to assure that no leakage can occur to the outside through the weldment and seals around the pipe. As described under Criterion 19, the air supply system is monitored for flow and pressure such that excessive leakage through penetrations will be detected, and the system is valved to allow leak testing of each penetration separately. If a power outage of air compressor failure should occur, pressurization can be maintained by a backup nitrogen supply system. (Section 5.2, 5.4, 5.5)

We believe that the design described above satisfies Criterion 20.

Criterion 21

Sufficient normal and emergency sources of electrical power must be provided to assure a capability for prompt shutdown and continued maintenance of the reactor facility in a safe condition under all credible circumstances.

The following sources of power are available to operate the essential equipment including the vital instruments and control systems:

1. The 138 kv Buchanan Substantion.
2. The station generator via the unit auxiliary transformer.
3. The three emergency diesel-generator sets.
4. The station 60 cell, lead acid batteries.

The Buchanan substation which is approximately one-half mile from the facility is connected to the Lovett station of the Orange and Rockland system and the Consolidated Edison 138 kv transmission system via two overhead lines to Millwood East. This power source is connected to the station auxiliary transformer which is used during startup, shutdown, and hot standby. Once the main station generator is sychronized to the 345 kv system, the Buchanan substation (external power) is used to drive one of the feedwater pumps and to supply two station service transformers. The power from this source would maintain the facility in a safe condition after shutdown or during the course of accidents. The remainder of the auxiliary load is supplied by the main generator through the 40 MVA unit auxiliary transformer.

As a backup to the normal standby AC power supply described above, diesel generator sets will be provided with the capability of starting and supplying the power requirements of the engineered safeguards as well as that equipment required to effect a normal facility shutdown. There will be three diesels that will automatically start on loss of voltage to the 480 volt bus stations. These can supply electrical power for the engineered safeguards, or equipment required for a normal shutdown. If only two diesels are assumed to operate, those safeguards required to preclude containment overpressurization and significant zirconium-water reaction can be adequately supplied. A normal shutdown could also be effected with two diesels in operation.

All components and structures of the emergency power supply system are vital to safe shutdown and isolation of the reactor and are, therefore, designed as Class I in terms of seismic design. This includes: diesel generators and fuel oil storage tank, DC power supply system, power distribution lines required during an emergency, transformers and switchgear supplying the engineered safeguards, control panel boards, and motor control centers. (Section 8; Supplement 1, Question 10)

We believe that there are sufficient normal and backup sources of power available to satisfy Criterion 21.

Criterion 22

Valves and their associated apparatus that are essential to the containment function must be redundant and so arranged that no credible combination of circumstances can interfere with their necessary functioning. Such redundant valves and associated apparatus must be independent of each other. Capability must be provided for testing functional operability of these valves and associated equipment to determine that no failure has occured and that leakage is within acceptable limits. Redundant valves and auxiliaries must be independent - containment closure valves must be actuated by instrumentation, control circuits and energy sources which satisfy Criterion 15 and 16 above.

The applicant has stated the criterion that there will be a double barrier between fluid systems inside the containment and the outside atmosphere. Double barriers in these piping systems are provided by valves of the automatic, check, remote manual, or manual operated type, and also by closed systems either inside or outside the containment. All piping penetrations are separated into five classes and are provided with isolation valve protection as discussed below. In addition, the isolation provided at the ventilation ducts is discussed in Class 6.

Class 1: Includes normally operating outgoing lines connected to the primary system. These lines are not missile protected but have one remote manual operated valve near the primary connection and one automatic and one manual valve outside containment. These lines will also be protected with automatic seal water injection (isolation valve seal water system).

Class 2: Includes normally operating outgoing lines not connected to the primary and not missile protected. An automatic or remote manual valve is located outside the containment. These valves are backed up with a manual valve or closed system outside containment. Seal water injection will also be provided for these lines.

Class 3: Includes incoming lines not missile protected which connect to the primary system or containment atmosphere. If the line connects to an open system outside the containment, two check valves are provided, and a remote manual valve is provided outside. If the line connects to a closed system externally, there is one check valve inside or a closed manual valve inside the containment, and a manually operated valve outside the containment. Seal water injection will also be provided for these lines.

Class 4: Includes normally operating incoming or outgoing lines which are missile protected and are connected to closed systems inside the containment. A manual valve is provided outside containment.

Class 5: Includes lines open to the containment atmosphere and open to the outside atmosphere but which are closed during reactor operation. These systems are sealed by two valves in series or one valve and one blind flange in series. Gas filled lines of this type will be provided with automatic seal water injection.

Class 6: The ventilation purge duct penetrations are provided with two remote manual operated butterfly valves. One valve is located inside and one valve is located outside the containment at each penetration.

The instrumentation for providing the containment isolation trip signals are redundant. Provisions are made for periodic testing of the functional capabilities of all remote operable valves. All remotely operated valves required for isolation are of the fail-safe type. (Supplement 1, Questions 2, 19 e)

In view of the foregoing we believe Criterion 22 is satisfied.

Criterion 23

In determining the suitability of a facility for a proposed site the acceptance of the inherent and engineered safety afforded by the systems, materials and components, and the associated engineered safeguards built into the facility, will depend on their demonstrated performance capability and reliability and the extent to which the operability of such systems, materials, components, and engineered safeguards can be tested and inspected during the life of the plant.

To provide assurance that the various engineered safeguards are capable of functioning in the required manner, the following periodic testing of systems will be performed:

(1) Containment will be subjected to an initial integrated leak rate test at the design pressure of 47 psig with the penetration pressurization and isolation valve seal water systems inoperable. Subsequent periodic leak rate testing is also contemplated.

(2) All pumps, circuitry, and piping associated with the safety injection system will be tested for proper operation, including flow up to the last remote operated injection stop valves, during reactor operation through the use of minimum flow recirculation test loops. Actual flow into the primary system will be checked during shutdown by observing a rise in pressurizer level as water is injected.

(3) Containment spray pumps and piping will be checked up to the last valve by use of a recirculation test loop. An air purge connection located downstream of the valve will be used to check for continuity in the piping and spray nozzles inside containment.

(4) The air recirculation system is normally in operation and is instrumented, such that any abnormality in the dynamic system could be detectable. Flow-path integrity through the filter and demister units will be checked by in-place testing using aerosols during shutdown. The charcoal beds will be removed and tested periodically with both elemental and methyl forms of iodines to determine that the efficiency has not deteriorated.

(5) Boric acid concentration in the high-head safety injection lines will be sampled periodically and maintained at the refueling water concentration by use of recirculation lines.

(6) Diesel generators will be started periodically to verify that the starting times and circuit operation are acceptable.

(7) Each loop of the service water system which provides cooling water to the fan-coolers will be periodically leak tested.

(8) Leakage from components in the external recirculation loop will be periodically measured to assure that limits are not exceeded.

(9) All remote operated valves will be exercised and actuation circuits will be tested periodically during plant operation.

We believe that the proposed testing capabilities for engineered safeguards systems of the Indian Point II plant are suitable, and Criterion 23 is satisfied. (Supplement 1, Question 2)

Criterion 24

All fuel storage and waste handling systems must be contained if necessary to prevent the accidental release of radioactivity in amounts which could affect the health and safety of the public.

The fuel storage and waste handling systems will be housed in the reactor auxiliary building and will be of Class I earthquake design. The applicant has satisfied us that these systems will be designed to assure that no credible accidental release of radioactivity from them could endanger the health and safety of the public.

Liquid wastes are expected to consist of reactor coolant released during plant heat-up and cool-down, resin bed regenerative solutions, pump leakage and waste from various drains. These will be stored in waste hold-up tanks or concentrated in the evaporator. Each of the three hold-up tanks will each be sized to hold two-thirds of the reactor coolant volume of about 12,000 cu. ft. One tank will normally be kept empty to provide for any unexpected liquid waste inventory. The reactor auxiliary building will be equipped with a sump and basement which will be capable of containing the contents of one liquid hold-up tank. The liquid waste disposal system will be designed so that the release of the gaseous fission products entrained in the liquid waste from the rupture of any of the three tanks will not result in potential off-site exposures in excess of the limits of 10 CFR 20.

Gaseous waste will consist of off-gas from the reactor coolant, the liquid waste disposal system gas from miscellaneous equipment vents, and relief valves and ventilation air from spent fuel and waste handling areas. This system will include four storage tanks in which gas may be compressed. One of the tanks will be in standby to contain any unexpected gas formation. The tanks are sized to permit storage of the waste gases for at least 45 days prior to discharge. Activity levels in the tanks will be kept at levels sufficiently low to limit the potential exposures at the site boundary from the rupture or inadvertent release of the contents of any tank to 0.5 rem.

Off-gas vented to the atmosphere will be monitored continuously and if an unexpected increase in radioactivity is sensed one of the two discharge valves will be closed automatically.

Spent fuel will be contained in a water filled storage pit. No gravity water drains will be provided and the pit can only be drained through the use of pumps.

Solid waste will consist of miscellaneous contaminated rubbish and spent ion-exchanger resins. These will be packed in suitable containers of steel and concrete and shipped off-site. (Sections 9.4, 11.1; Supplement 1, Question 2)

Based on the foregoing, we believe Criterion 24 is satisfied.

Criterion 25

The fuel handling and storage facilities must be designed to prevent criticality and to maintain adequate shielding and cooling for spent fuel under all anticipated normal and abnormal conditions, and credible accident conditions. Variables upon which health and safety of the public depend must be monitored.

Subcriticality of the spent and new fuel will be ensured at all times at this facility. Spent fuel will be submerged in borated water in the fuel storage pit. The arrangement of fuel elements in the fuel storage pit will ensure a multiplication of less than unity even if the boron were somehow removed from solution. The spent fuel storage racks will be designed so that the assemblies can only be inserted in their prescribed locations. The water in the pit will be maintained at a low temperature and decay heat will be removed by the service water system. New fuel will be stored dry in racks designed so that even complete flooding would produce a multiplication of no more than 0.9.

During refueling, personnel will be protected by concrete and water shielding which will maintain dose rates of less than 50 mr/hr throughout the operation. The fuel storage pit will provide shielding sufficient to permit normal occupancy of the area by plant personnel. Systems will be provided to monitor pit water temperature and radioactivity levels. The spent fuel handling and storage systems are Class I structures with respect to seismic design and as such will be designed to retain their function during the maximum design earthquake. In our opinion, these design provisions satisfy Criterion 25. (Section 9.4; Supplement 1, Question 2)

Criterion 26

Where unfavorable environmental conditions can be expected to require limitations upon the release of operational radioactive effluents to the environment, appropriate hold-up capacity must be provided for retention of gaseous, liquid or solid effluents.

The liquid waste processing system utilizes three storage tanks. Each tank will have a capacity of approximately 8000 ft^3, which is 2/3 of the primary coolant volume. One tank will always be held in reserve to contain any unexpected liquid waste. Normal discharge of low level liquid waste will be to the condenser water canal through a monitored line. Appropriate limitations on concentrations of radioactive materials discharged and requirements for necessary monitoring equipment will be imposed as part of the licensing conditions at the operating stage of our review. The hold-up capacity of this system will be sized so that under normal operating conditions the hold-up and reserve capacity will remain constant while necessary limitations on discharge activity are met.

The gaseous waste processing system will normally utilize four gas compression tanks, one filling, one holding gas for decay, one discharging to the atmosphere when a suitable activity level is reached and on standby to accommodate waste gases resulting from unexpected plant operations. The system will allow for storage of radiolytic gas for at least 45 days. Gases will be discharged to a monitored plant vent. Although unfavorable conditions are not expected to interfere with the discharge of gaseous waste products, it is evident that adequate hold-up capacity is available.

Solid wastes will be stored on-site in suitable containers until shipment off-site for ultimate disposal. (Section 11; Supplement 1, Question 2)

We believe that Criterion 26 is satisfied.

Criterion 27

The plant must be provided with systems capable of monitoring the release of radioactivity under accident conditions.

Gas releases from sources external to the reactor containment will be exhausted from the facility vent or from the auxiliary building vent. Both of these vents are monitored. The handling of radioactive liquids will be controlled so that any accidental spills will be confined within the auxiliary building and collected in a drain tank. The normal low level liquid waste discharge to the condenser water canal will be monitored.

Leakage of radioactive gas from the reactor containment or auxiliary building under accident conditions will be monitored by the plant area radiation monitoring system supplemented by portable survey equipment. (Section 11.2.2; Supplement 1, Question 2)

Based on the foregoing, we believe that Criterion 27 is satisfied.

VI. Accident Analysis

The applicant has described the consequences of various accidents resulting from assumed mechanical failures and reactivity insertions. We believe that all types of credible accidents have been considered and are in general agreement with the consequences described; however, it should be recognized that a complete evaluation of potential accident consequences cannot be made until the final thermal, hydraulic, and physics parameters of the core have been determined. The consequences of these accidents will be evaluated by the applicant when final design details are available, and will be reviewed by the Staff prior to reactor operation.

1. Maximum Credible Accident

The course and consequences of a double-ended failure of the primary coolant piping, the maximum credible accident (MCA), were evaluated by the applicant. We believe that this accident represents the maximum potential for off-site consequences.

The norm for site acceptance is the Commission's site criteria, 10 CFR 100. This regulation relates potential radiation doses to site characteristics-exclusion distance and low population distance. The criteria state that following a credible but highly unlikely accident the potential radiation doses at the exclusion area boundary during the first two hours following the accident should not exceed 300 rem to the thyroid or 25 rem whole body. Also, these same doses should not be exceeded at the outer edge of the low population zone through the course of the accident.

As stated previously in this report, the Indian Point II containment has been designed to have negligible leakage which would result in very low off-site doses if the assumed MCA were to occur. Nevertheless, since integral leakage rates of less than 0.1%/day are difficult to demonstrate experimentally, the applicant has provided two independent iodine removal systems to limit the fission product inventory available for leakage following an accident. These systems are the internal air filtration system (activated charcoal filters) and the containment spray system. Sodium thiosulphate will be injected into the containment spray water to aid removal and retention of elemental forms of iodine. The Staff has engaged Dr. George Parker of the Oak Ridge National Laboratory to evaluate the design of these systems, and he concluded that the following efficiencies could be expected for their operation under accident conditions:

(a) Under conditions of water-logging of the charcoal filter units which can be expected when operated in the anticipated post-MCA environment, at least 90% removal efficiency can be assumed for elemental iodine. Zero efficiency should be assumed for the removal of organic iodine.

(b) The efficiency of the containment spray system is not greater than 90% for removal of elemental iodine. Zero removal efficiency should be assumed for organic iodine.

The Staff has calculated the off-site consequences of leakage of radioactive fission products from the containment under MCA conditions, assuming various efficiencies of the iodine filtration equipment. The following conservative assumptions were made:

Power level - 2758 MWt

Equivalent I^{131} available for leakage - 3.2×10^7 curies
(25% of total inventory)

Unfilterable organic iodine - 5% of inventory available for leakage

Fan capacity - 65,000 CFM each, 4 of 5 operating

Recirculation rate - 6 containment volume per hour

Filter efficiency for elemental iodine - as indicated in table below

Containment leakage rate - (ground release) 0.1%/day for first day, 0.045%/day for next thirty days.

Atmospheric dispersion - as developed by applicant

Flow bypassing filter - 10%

Credit for building wake effects

No credit for containment spray system

The following table presents a tabulation of the potential off-site thyroid doses using TID-14844 assumptions and the assumptions listed above. The whole body doses were also calculated, but are not presented since they are not limiting. To account for building wake dilution effects, the model suggested by Gifford and Fuquay has been used.

		Integrated Thyroid Dose (rem)	
		Site Boundary 0.32 miles	Low Population Distance 0.67 miles
Condition	Filter Efficiency	2 hr.	30 day
Assumptions as stated above	0	870	3,250
	30	300	360
	45	210	300
	90	130	230
TID-14844 assumptions on meteorology and constant 0.1% leakage rate (no credit for building wake)	0	2,390	39,000

The above table indicates that for a filter efficiency of 45% for elemental iodine, the guideline doses of 10 CFR 100 would be satisfied should there be a containment outleakage rate of as much as 0.1%/day at 47 psig. As previously mentioned, it is reasonable to assume 90% filter efficiency for elemental forms of iodine. In addition, the design objective for this containment is to have negligible outleakage under MCA conditions. Under these circumstances the potential consequences of the maximum credible accident incident to operation of the Indian Point II facility would be well within the 10 CFR 100 guidelines.

2. Minor Accidental Releases of Radioactivity

In addition to the release of radioactivity under MCA conditions, several other means for the accidental release of radioactivity from this facility have been identified. These are:

(a) Steam generator tube failure

The applicant has stated that a steam generator tube failure would result in the blowdown of a significant portion of the primary system (about 6,000 cubic feet) into the secondary system. Under these conditions, a reactor scram and turbine trip would be initiated by the low primary system pressure trip and the secondary system steam would be dumped to the turbine condensers. The steam dump capacity of 40% of full load would prevent operation of the steam generator safety valves and consequent discharge of radioactivity to the atmosphere via this route. However, radioactivity would be released via the steam exhaust of the air ejector. The air ejector effluent is monitored for radioactivity and would be diverted to the containment under these conditions. Thus the only radioactivity release would occur before the air ejector effluent is diverted.

The potential off-site doses for this accident have been calculated assuming the primary coolant contains the fission products estimated to result from 1% failed fuel elements. The most significant isotope is Xe-133 (concentration 200 uc/cm^3), and about 32,000 curies would be injected into the secondary system during the entire blowdown period. The contribution from the remaining isotopes would be less than 10% of the total activity. The applicant has estimated that less than 13,500 curies of Xe-133 would be released before the air ejection exhaust is diverted to the containment. The Staff calculated that the resulting off-site whole body dose would be less than 0.5 rem. We believe that this estimate is conservative, since the primary coolant would normally contain significantly less fission products than assumed.

(b) Leakage from gas storage tanks

The maximum anticipated quantity of gaseous wastes in one storage tank is approximately equivalent to 13,500 curies of Xe-133. These tanks are in ventilated concrete cells such than any release or leakage would be exhausted through the plant vent. The potential off-site whole body exposure under these conditions would be less than 0.5 rem.

(c) Leakage from auxiliary building

The backup pumps for the recirculation system are located in the auxiliary building and would be required to pump water containing radioactive fission products if both low-head safety injection pumps within the containment failed after a major loss-of-coolant accident. Leakage from those portions of the system located outside the containment represents a potential means for the release of radioactivity. However, these components will be designed for minimum leakage and the auxiliary building atmosphere is vented through absolute filters to the plant vent. If the components in the system were to leak at three times the specified rate, the two hour potential thyroid dose at the site boundary from this source would be 2.5 rem.

(d) Leakage through fan-coolers

The cooling coils for the five fan-coolers are cooled by water supplied by the service water system at a pressure of 20-25 psig. During a period following the MCA the containment pressure is above the service water pressure and a leak at these coils would present a direct path for radioactivity to escape the containment. To preclude significant leakage through this system the applicant has proposed the following:

(1) The containment leakage tests will be performed with the service water system depressurized and vented to the atmosphere. These periodic tests should assure that significant leakage

paths do not exist at these cooling coils (the design pressure of the coils is 150 psig). Also, the individual cooling coils can be pressurized internally to test their integrity.

(2) The service water discharge will be continuously monitored for radioactivity. If leakage is detected, valves and test lines have been provided so that the leaking system can be identified and the leakage terminated.

In view of the short time period that the containment pressure is above 20-25 psig (about one-half hour if four fan coolers and one containment spray operate on emergency power) and the safeguards provided for detecting and isolating leaking systems, we believe that operation of this system in the manner proposed is acceptable.

Items a and b, above, are examples of a class of accidents with lesser consequences than the MCA but which are considered more likely to occur. In our judgment, the off-site consequences calculated for each (0.5 rem whole body) are acceptable in view of the small likelihood of occurrence.

Items c and d, above, are examples of paths of leakage of radioactive effluents which have not been specifically considered in our evaluation of MCA consequences. In each case we believe that the additional potential exposure that could be experienced are acceptable.

VII. Research and Development

On all components which are important for the safe operation of Indian Point Unit No. II, the architectural and engineering criteria have been described. At this stage in design, the applicant has not yet completed the final layout arrangements and design details of many components and systems of the plant. Programs are being conducted which will aid in determination and evaluation of the final design. These include:

1. Development of final core design and final thermal, hydraulics, and physics parameters.
2. Research and development on the air recirculation system halogen filters.
3. Research on consequences of failure of core cooling systems and development of means to ameliorate the consequences.
4. Development of the emergency core cooling systems to prevent fuel damage following primary system piping failures.

Our evaluation of the information submitted thus far leads us to believe that acceptable design details can be evolved from the programs proposed. At a later state of development, a description of the final design derived on the basis of these programs will be submitted by the applicant and will be evaluated by the Staff.

VIII. Technical Qualifications

This application for a provisional construction permit has been submitted by Consolidated Edison Company of New York, Inc., which will operate the facility when completed. The applicant has been operating Unit No. 1, also a pressurized water reactor, for about four years with considerable success.

The nuclear subcontractor is Westinghouse Electric Corporation. Westinghouse has been directly associated with the design and operation of many pressurized water nuclear power plants of generally similar concept to the proposed Indian Point II facility. These include Saxton and Yankee which have operated successfully, and San Onofre, Connecticut Yankee and Brookwood which are presently under construction and are expected to be in operation by the time Indian Point II is completed.

Based on these considerations as well as upon our evaluation of the responsible personnel, we have concluded that there is reasonable assurance that the applicant and its principal contractor collectively are technically qualified to design and construct the proposed Indian Point II facility.

IX. Report of the Advisory Committee on Reactor Safeguards

As noted previously, a Subcommittee of the Advisory Committee on Reactor Safeguards (ACRS) met with representatives of Consolidated Edison Company of New York, Inc. on March 30, May 3 and June 23, 1966, to discuss the design and safety questions related to the proposed facility. During its seventy-second, seventy-third, seventy-fifth, and also a special meeting on August 4-5, 1966, the full ACRS met with the applicant to discuss the proposed facility.

A copy of the ACRS letter to the Commission concerning the Consolidated Edison Company of New York, Inc. application for a construction permit for Indian Point Nuclear Generating Unit No. 2 is attached as Appendix A.

The ACRS in this letter included several comments and recommendations concerning the design of the proposed facility which have been discussed in the body of this report. We have considered each of these matters and believe they should be handled as recommended by the ACRS. The letter then concluded ". . . the proposed reactor can be constructed at the Indian Point site with reasonable assurance that it can be operated without undue risk to the health and safety of the public."

X. Conclusions

Based on the proposed design of the Indian Point Nuclear Generating Unit No. 2, on the criteria, principles and design arrangements for systems and components thus far described, which includes 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 contractor, we have concluded that, in accordance with the provisions of paragraph 50.35 (a), 10 CFR 50:

1. The applicant has described the proposed design of the facility, including the principal architectural and engineering criteria for the design, and has identified the major features or components on which further technical information is required;

2. The omitted technical information will be supplied;

3. Research and development as required to resolve the safety questions with respect to the features and components which require research and development will be conducted;

4. On the basis of the foregoing, there is reasonable assurance that (1) such safety questions will be satisfactorily resolved at or before the latest date stated in the application for the completion of construction of the proposed facility, and (2) taking into consideration the site criteria contained in Part 100 of the Commission's regulations, 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 and its contractor are technically qualified to design and construct the proposed facility; and

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

In summary, we have concluded that there is reasonable assurance that the Indian Point Nuclear Generating Unit No. 2 can be constructed and operated at the proposed site without endangering the health and safety of the public.

Appendix A

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
UNITED STATES ATOMIC ENERGY COMMISSION
WASHINGTON 25, D. C.

AUG 16 1966

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

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

Dear Dr. Seaborg:

At its seventy-fifth meeting, July 14-16, 1966, and its special meeting on August 4-5, 1966, 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. 2. This project had previously been considered at the seventy-second and seventy-third meetings of the Committee, and at Subcommittee meetings on March 30, May 3, and June 23, 1966. 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.

The Indian Point 2 plant is to be a pressurized water reactor system utilizing a core fueled with slightly enriched uranium dioxide pellets contained in Zircaloy fuel rods; it is to be controlled by a combination of rod cluster-type control rods and boron dissolved in the primary coolant system. The plant is rated at 2758 MW(t); the gross electrical output is estimated to be 916 MW(e). Although the turbine has an additional calculated gross capacity of about 10%, the applicant has stated that there are no plans for power stretch in this plant.

The Indian Point 2 facility is the largest reactor that has been considered for licensing to date. Furthermore, it will be located in a region of relatively high population density. For these reasons, particular attention has been given to improving and supplementing the protective features previously provided in other plants of this type.

The proposed design has a reinforced concrete containment with an internal steel liner which is provided with facilities for pressurization of weld areas to reduce the possibility of leakage in these areas. The containment design also includes an internal recirculation containment spray system and an air recirculation system consisting of five air handling units to provide long-term cooling of the containment without having to pump radioactive liquids outside the containment in the event of an accident. Even though the applicant anticipates negligible leakage from the containment, two independent means of iodine removal within the containment have been provided. These are an air filtration system using activated charcoal filters, and a containment spray system which uses sodium thiosulfate in the spray water as a reagent to aid removal of elemental iodine.

The reactor vessel and various other components of the system are surrounded by concrete shielding which provides protection to the containment against missiles that might be generated if structural failure of such components were to occur during operation at pressure. This includes missile protection against the highly unlikely failure of the reactor vessel by longitudinal splitting or by various modes of circumferential cracking. The Committee favors such protection for large reactors in regions of relatively high population density.

The Indian Point 2 plant is provided with two safety injection systems for flooding the core with borated water in the event of a pipe rupture in the primary system. The emergency core cooling systems are of particular importance, and the ACRS believes that an increase in the flow capacity of these systems is needed; improvements of other characteristics such as pump discharge pressure may be appropriate. The forces imposed on various structural members within the pressure vessel during blowdown in a loss-of-coolant accident should be reviewed to assure adequate design conservatism. The Committee believes that these matters can be resolved during construction of these facilities. However, it believes that the AEC Regulatory Staff and the Committee should review the final design of the emergency core cooling systems and the pertinent structural members within the pressure vessel, prior to irrevocable commitments relative to construction of these items.

The applicant stated that, even if a significant fraction of the core were to melt during a loss-of-coolant accident, the melted portion would not penetrate the bottom of the reactor pressure vessel owing to contact of the vessel with water in the sump beneath it.

The applicant also proposes to install a backup to the emergency core cooling systems, in the form of a water-cooled refractory-lined stainless steel tank beneath the reactor pressure vessel. The Committee would like to be advised of design details and their theoretical and experimental bases when the design is completed.

In order to reduce still further the low probability of primary system rupture, the applicant should take the additional measures noted below. The Committee would like to review the results of studies made by the applicant in this connection, and consequent proposals, as soon as these are available.

1. Design and fabrication techniques for the entire primary system should be reviewed thoroughly to assure adequate conservatism throughout and to make full use of practical, existing inspection techniques which can provide still greater assurance of highest quality.

2. Great attention should be placed in design on in-service inspection possibilities and the detection of incipient trouble in the entire primary system during reactor operation. Methods of leak detection should be employed which provide a maximum of protection against serious incidents.

Attention should also be given to quality control aspects, as well as stress analysis evaluation, of the containment and its liner. The Committee recommends that these items be resolved between the AEC Regulatory Staff and the applicant as adequate information is developed.

The applicant has made studies of reactivity excursions resulting from the improbable event that structural failure leads to expulsion of a control rod from the core. Such transients should be limited by design and operation so that they cannot result in gross primary-system rupture or disruption of the core, which could impair the effectiveness of emergency core cooling. The reactivity transient problem is complicated by the existence of sizeable positive reactivity effects associated with voiding the borated coolant water, particularly early in core life. In addition, the course of the transients is sensitive to various parameters, some of which remain to be fixed during the final design. Westinghouse representatives reported that the magnitude of such reactivity transients could be reduced by installation of solid burnable poisons in the core to permit reduction of the soluble boron content of the moderator, thereby reducing the positive moderator coefficient. The Committee agrees with the applicant's plans to be prepared to install the burnable poison if necessary. The Committee wishes to review the question of reactivity transients as soon as the core design is set.

The Advisory Committee on Reactor Safeguards believes that the various items mentioned can be resolved during construction and that the proposed reactor can be constructed at the Indian Point site with reasonable assurance that it can be operated without undue risk to the health and safety of the public.

Sincerely yours,

ORIGINAL SIGNED BY
DAVID OKRENT

David Okrent
Chairman

References:

1. Consolidated Edison Company of New York, Inc., Indian Point Nuclear Generating Unit No. 2, Preliminary Safety Analysis Report, Volume 1, and Volume 2, Parts A & B, received December 7, 1965.
2. First Supplement to Preliminary Safety Analysis Report, dated March 31, 1966.
3. Second Supplement to Preliminary Safety Analysis Report, received June 2, 1966.
4. Errata Sheets for Preliminary Safety Analysis Report and First Supplement thereto, received June 13, 1966.
5. Third Supplement to Preliminary Safety Analysis Report, received June 22, 1966.
6. Fourth Supplement to Preliminary Safety Analysis Report, received July 28, 1966.
7. Fifth Supplement to Preliminary Safety Analysis Report, received July 28, 1966.

Appendix B

DOCKET NO. 50-247

Comments on

Description and Safety Analysis
For a Conceptual Unit at Indian Point,
Volumes I and II dated October 1, 1965

Prepared by

Environmental Meteorological Research Branch
Office of Meteorological Research
October 29, 1965

A primary influence on the meteorological statistics of the Indian Point site seems to be its location in a river valley about a mile wide with terrain rising from 600 to 1000 feet on either side. This influence is shown by the predominant up-river and down-river wind directions and by an annual inversion frequency of 41% in the lower 150 feet and 32% in the lower 300 feet. The latter frequency is in agreement with that found by Hosler [1] using radiosonde data in the lower 500 feet from Albany, New York. At the 100 foot level of the site tower, wind speeds of 4 mph or less during inversions in the lower 150 feet occur 16% of the time. This would indicate that for the short term (less than 2 hours) ground release of effluents the conservative diffusion parameters used in TID-14844 would be appropriate.

The applicant's analysis of the off-site radiological consequences of a major loss of coolant assumes a ground release. For the two-hour dose the inversion parameters assumed in TID-14844 were used and credit was taken for additional dilution due to building turbulence. This latter effect amounted to a factor of 3 at the site boundary (540m) and a factor of 1.25 at a distance of 2000 m, which is consistent with the findings of Islitzer [2]. For the 22-hour dose credit was taken for a higher average wind speed (2 m/s instead of 1 m/s) which doubled the normalized air concentrations. This would seem most reasonable and in fact is conservative because no credit was taken for mean wind direction variability which certainly is a factor over a 22-hour period. For the analysis of the long-term, 30-day hazard consequences, temperature lapse rate and wind speed statistics measured during winds blowing from the 20^{o} sector centered on NNE (predominant direction) were utilized to establish dispersion factor categories. The annual frequency of NNE winds was 15% during which inversions occurred 42% of the time. It is obvious from an inspection of the categories listed on page 5.2-6 of volume II that the inversion category is the major contributor (by a factor of 10) to the long-term dispersion factor. Using Sutton's equation modified for a long-term average (eq. 4.76 in Meteorology and Atomic Energy, AECU-3066) the appropriate dispersion factors were computed. The results were conservative since a wind direction frequency of 35% was used instead of the observed annual value of 15%.

In summary the computed atmospheric dispersion factors for both the short-term and long-term accident are realistic and somewhat conservative in light of the meteorological conditions observed for a year's period at the site meteorological tower.

References

[1] Hosler, C. R., "Low-level Inversion Frequency in the Contiguous United States", Monthly Weather Review, 89, pp. 319-339.

[2] Islitzer, N. F., "Aerodynamic Effect of Large Reactor Complexes ypon Atmospheric Turbulence and Diffusion", IDO-12041.

Appendix B-1

Comments on

Preliminary Safety Analysis Report
Consolidated Edison Company of New York
Indian Point Nuclear Generating Unit No. 2
Dated December 6, 1965

Prepared by

Environmental Meteorological Research Branch
Institute for Atmospheric Sciences
January 3, 1965

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.

Comments on

First and Second Supplement
Preliminary Safety Analysis Report
Consolidated Edison Company of New York
Indian Point Nuclear Generating Unit No. 2
Dated March 31, 1966 and June 1, 1966, rewpectively

Prepared by

Environmental Meteorology Branch
Institute for Atmospheric Sciences
August 16, 1966

No new meteorological information is contained in this report that was not considered or that would change 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.

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

Appendix C

AUG 15 1966

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

Dear Mr. Price:

Transmitted herewith are statements on the geology and hydrology of the Indian Point site as requested in Mr. Case's letter of December 10, 1965.

The statements were prepared by Henry W. Coulter of the Geologic Division and Eric L. Meyer of the Water Resources Division, and have been discussed with members of your staff.

We have no objection to your making these statements a part of the public record.

Sincerely yours,

Arthur A. Baker

Acting Director

Enclosures

Appendix C

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

Indian Point Nuclear Generating Unit No. 2
Buchanan, New York

Geology

Based on a careful review of the applicant's report (A.E.C. Docket No. 50-247, Vol. I, Exhibit B) and of the available literature, it appears that their geological analysis is carefully derived and presents an adequate appraisal of those aspects of the geology which would be pertinent to an engineering evaluation of the site.

Although it may be anticipated that earthquakes within the general region will continue to occur 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 identifiable faults or other geologic structures which could be expected to localize earthquakes in the immediate vicinity of the site.

Appendix C

Review of Hydrology Section of Preliminary Safety Analysis Report, Indian Point Nuclear Generating Unit No. 2, Consolidated Edison Company of New York, Inc.

The site is on the estuary of the Hudson River about 36 miles up-stream from the Narrows. In this reach the river's flow and stage are determined both by runoff from its drainage basin and by tides.

Discharge of the river has been measured by the U.S. Geological Survey at Green Island, near Troy, since 1946. The drainage area above the gage is 8,090 square miles; intervening drainage area between the gage and the site is estimated to be about 4,500 square miles. The mean flow at Green Island during 1946-66 has been 13,060 cfs (cubic feet per second), and the corresponding flow past the site is estimated to have been about 20,000 cfs. Minimum daily flow at Green Island was 1,010 cfs on September 7, 1964; during the period of record the flow has been greater than 4,000 cfs 90 percent of the time, and greater than 8,000 cfs 53 percent of the time. The relationship of low flows at the Green Island gage to low flows at the site is not as readily estimable as that of mean flow; however, it is likely that equivalent low flows at the site are also about 1½ times as high as at the gage. The maximum flow observed at the gage during the period of record was 215,000 cfs, occurring on March 19, 1936, but the stage at the site is not known. Another major flood occurred on March 28, 1913, but the discharge is unknown at either the gage or the site.

Flow in the river at Peekskill is principally in the downstream direction only during periods of high freshwater runoff. At medium and low runoff there is upstream flow during flood tides, and salt water begins to travel upstream when flow at Green Island is near 8,000 cfs, or slightly below median flow. Typically, freshwater runoff in the Hudson River drainage is above median during winter and spring, and below median during summer and fall. Median monthly average flows at the Green Island gage for the period 1946-60 are less than 8,000 cfs for the months of July through October.

When freshwater flow is below the median point, tidal currents reverse flow during the flood tide and water would then recycle past the site. The recycling water masses would mix with fresher water coming from upstream and with saltier water from downstream. Under these conditions contaminants released at the site would disperse both in the upstream and downstream direction.

The Hudson River downstream from the site is not used for drinking water supplies; however, at Chelsea, 22 miles upstream from the site, the city of New York has installed facilities for pumping water from the Hudson to augment other sources in emergencies or during extended periods of drought. Contaminants released to the river at the site would not reach Chelsea, except when freshwater flow drops below the median point. During these periods, contaminants would be dispersed in a large volume of water extending both above and below the release point prior to reaching the intake. The highest concentrations would remain near the release point, the lowest at the upstream and downstream edges of the spread of the contaminant. It would take a number of tidal cycles, probably more than five, before the contaminant could extend to the Chelsea intakes. A quantitative estimate of the number of tidal cycles required or the amount of the number of tidal cycles required or the amount of dispersion cannot be readily made without data on stream velocities and dispersion characteristics in this reach. However, a study of dispersion in New York Harbor supported by the Atomic Energy Commission may have generated sufficient data to permit an adequate estimate.

The study was carried out by the Chesapeake Bay Institute (Pritchard and others, 1962) to determine the dispersion of an assumed instantaneous contaminant release to the river at the Battery in lower Manhattan. Current velocity and salinity data were obtained by the Coast Guard at 55 stations extending from the Lower Bay to Highland Falls, New York, about 8 miles above Indian Point. Dye dispersion experiments were carried out in the hydraulic model of New York Harbor located at the U.S. Army Engineers Waterways Experiment Station in Vicksburg, Mississippi. This model can reproduce the prototype tidal fluctuations, current velocities and salinities as far upstream as Hyde Park, New York, about 40 miles above Indian Point. One of a series of dye dispersion experiments indicates that with a flow of 6,000 cfs, traces of a contaminant would move about 22 miles upstream from the release point between the 5th and 10th tidal cycle and would have a concentration at the point of about 5×10^{-13} per cubic meter per unit of released contaminant. The farthest upstream extent of the contaminant was found about 25 miles above the release point and reports of the study do not concern the river above that point. A mathematical analysis using the current velocity and salinity data in a computer program yielded comparable results.

The figures above are of course not directly applicable to releases at the site, but information from this study, along with general information on the river, indicates that dispersion would be substantial.

The stage of the Hudson River near the site is affected by tides. The range of the tide has been measured at a tide gage near Verplank, New York, about 3/4 of a mile downstream from the site sporadically from 1919 to 1930 (Schureman, 1934). Monthly average tidal ranges were found to be on the order of 2.5 to 3 feet. Referred to Sandy Hook sea level datum, the mean low water level was about 0.5 feet below sea level, and mean high water was from 2 to 2.5 feet above sea level.

High stages at the site are due primarily to high tides caused by storm surges from the ocean. Freshwater floods alone are not likely to lead to the highest stages at the site, because the river has a high cross-sectional area in comparison to the maximum floods observed. Tidal storm surges caused by either hurricanes or extratropical storms have been observed to travel up the Hudson. The highest storm surge in the Hudson in recent years occurred in November, 1950, when a stage of 7.4 feet above mean sea level was observed at Peekskill by the Corps of Engineers. Storm surges considerably higher than those of November, 1950, are a possibility. Wilson (1960, p. 64) in a theoretical study of hurricane storm-tide in New York Bay has computed maximum storm surges of 8.7 feet above predicted astronomical tides, on basis of transposing the track of the major 1938 hurricane to the New York Bay area. Storm surges can travel up the Hudson as far as the site without diminishing in height. If such a storm surge were combined with high astronomical tide, stages near the site might reach 10 to 11 feet.

REFERENCES

Pritchard, D. W., Okubo, Akira and Mehr, Emanuel, 1962, A study of the movement and diffusion of an introduced contaminant in New York Harbor Waters: Chesapeake Bay Institute Tech. Rep. 31, Reference 62-21.

Schureman, Paul, 1934, Tides and currents in the Hudson River: U.S. Coast and Geodetic Survey, Spec. Pub. 180.

Wilson, B. W., 1960, The prediction of hurricane storm-tides in New York Bay: U.S. Coastal Engineering Center (formerly U.S. Beach Erosion Board) Tech. Memo 120.

U.S. Coast and Geodetic Survey, 1966, Tidal Current tables 1966, Atlantic Coast of North America.

IN REPLY REFER TO:

UNITED STATES
DEPARTMENT OF THE INTERIOR
FISH AND WILDLIFE SERVICE
WASHINGTON, D.C. 20240

File 800

Appendix D

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

JUL 13 1966

Dear Mr. Price:

In accordance with your request dated December 10, 1965, the following is the Fish and Wildlife Service's report of the effect upon fish and wildlife of the proposed nuclear power plant of the Consolidated Edison Company of New York, Indian Point Nuclear Generating Unit No. 2, Buchanan, New York (Docket No. 50-247).

As is our usual procedure, we requested Dr. Theodore R. Rice of the Bureau of Commercial Fisheries to review the Preliminary Safety Hazards Report for general comments upon the radioactive hazard. A copy of his report entitled "A Preliminary Evaluation of Possible Effects on Fish and Shellfish of the Operation of the Proposed Indian Point Nuclear Generating Unit No. 2, Buchanan, New York," is enclosed and should be considered an appendix to this report.

Dr. Rice's report and the Preliminary Safety Hazards Report were then sent to Mr. John T. Gharrett, Regional Director, Bureau of Commercial Fisheries, Gloucester, Massachusetts, for discussion and comments with local representatives of the Bureau of Sport Fisheries and Wildlife and the State of New York Conservation Department. This letter represents the comments of all three of these agencies.

We believe that plans for control and disposal of radioactive materials are generally adequate to protect fish and wildlife in the vicinity of the proposed plant. We request, however, that the license require the company to conform to standards on disposal of radioactive effluents of the State of New York, as well as those of the Federal Government. The recommendations in Dr. Rice's report dealing with radioactive hazards to fish and wildlife should be carried out by competent fish and wildlife experts to ensure that no adverse effects occur. We request that the applicants be required to consult with local personnel of the Fish and Wildlife Service and the State of New York Conservation Department in developing and approving ~~plans~~ for surveys needed to carry out these recommendations.

The problems associated with hazards to fish from other than radioactive materials are the most serious. In view of the Administration's policy to bring about real and substantial improvements in the quality of our environment, we feel strongly that adequate studies of these hazards and development of methods to eliminate or minimize them should be part of the construction license. We also believe the applicant should be required to meet with local representatives of the Fish and Wildlife Service and the State of New York Conservation Department to develop plans for these studies and for their adequate evaluation after the data is collected. We request that such meetings be required and that the applicant be required to make such modifications in plant structure and operation as may be necessary to eliminate or minimize any hazards to fish.

Specifically, we request that the applicant be required to:

(1) Acquire data acceptable to conservation officials of the Fish and Wildlife Service and the State of New York on the quantity and species of fish eggs, larval fish and juveniles which may be expected to pass through the intake screens and the coolant system.

(2) Discuss and review with conservation officials of the Fish and Wildlife Service and the State of New York the past fish mortality problems at Indian Point, Unit No. 1, the success of measures to overcome these problems and the applicability of these measures to Unit 2.

(3) Develop pre-construction studies acceptable to conservation officials of the Fish and Wildlife Service and the State of New York of thermal and other effects upon fish; the need for and design of fish screening facilities; and the need for, the design of, and the standards required for modification of plant structure and operation to minimize any fishery problems.

(4) Meet with conservation officials of the Fish and Wildlife Service and the State of New York at frequent periodic intervals to discuss plans and results of all studies to minimize hazards to fish and wildlife.

We are sending copies of this letter and Dr. Rice's report to the State of New York Conservation Department; Bureau of Sport Fisheries and Wildlife, Boston, Massachusetts; and Bureau of Commercial Fisheries, Gloucester, Massachusetts, for their information.

Appendix D

In accordance with past requests we are enclosing four copies of these reports for your convenience.

Sincerely yours,

D. L. McKernan

Acting Clarence F. Pautzke
Commissioner

Enclosures

January 17, 1966

A PRELIMINARY EVALUATION OF POSSIBLE EFFECTS ON FISH AND SHELLFISH OF THE OPERATION OF THE PROPOSED INDIAN POINT NUCLEAR GENERATING UNIT NO. 2 BUCHANAN, NEW YORK (DOCKET NO. 50-247)

By

T. R. Rice, Director
and
J. P. Baptist, Fishery Biologist

Bureau of Commercial Fisheries
Radiobiological Laboratory
Beaufort, North Carolina

1. Introduction

The Consolidated Edison Company of New York, Inc., has applied to the Atomic Energy Commission for licenses to construct and operate a nuclear reactor in Westchester County, New York. The proposed reactor will be the second nuclear facility at the Indian Point site, the first having been in operation for over 3 years. The site comprises approximately 250 acres of land on the east bank of the Hudson River at Indian Point, Village of Buchanan in upper Westchester County, New York. The site is 2.5 miles southwest of Peekskill and about 24 miles north of New York City boundary line.

We understand that the jurisdiction of the AEC in the licensing and regulation of nuclear power reactors is limited to matters pertaining to radiological safety. For that reason, our comments in this report are divided into two categories. The first category pertains to radiological safety considerations which are involved in the pending licensing proceeding. The second category contains our comments on the possible effects of increased water temperature on fishery organisms. Although these considerations are not within the jurisdiction of the AEC and not involved in the pending AEC licensing proceedings, they may be of interest to appropriate state and local agencies and to the applicant.

The entry of radioactive materials into the aquatic environment, either by design or by accident, might conceivably result in adverse effects on the fisheries of the area. It was deemed advisable, therefore, that the Bureau of Commercial Fisheries of the U. S. Fish & Wildlife Service evaluate the possible effects of the operation of the reactor on the fisheries of the area. The present evaluation is based in part on information presented in the Preliminary Safety Analysis Report, Volumes 1 and 2, by the Consolidated Edison Company of New York, Inc.

2. Description of the Facility

Generating Unit No. 2 will be constructed adjacent to Unit No. 1 and will consist of a reactor containment building, auxiliary building, control room, and turbine building as the major structures.

The reactor will be a pressurized water-type cooled by ordinary water which is kept under sufficient pressure to prevent bulk boiling. This is the type used in Indian Point Unit No. 1, Brookwood, New York, and the Yankee Power Facility, Massachusetts. The water, after leaving the reactor vessel, passes through a heat exchanger where it yields its heat to another separate stream of water which is thereby converted into steam. The reactor coolant system will be arranged as four closed reactor coolant loops connected in parallel to the reactor vessel, each containing a reactor coolant pump and a steam generator. An electrically heated pressurizer will be connected to one of the loops. The reactor design calls for a thermal output of 2,758 megawatts and a net electrical capacity of approximately 873 megawatts.

Condenser circulating water will be drawn from the Hudson River through a floating debris skimmer wall and eight separate screen wells at a flow rate of 840,000 gpm. The circulating water will be discharged back into the river far enough away from the intake to minimize recirculation.

3. Radioactive Waste Disposal Facilities

The waste disposal system is designed to collect, monitor, and process for safe disposal all solid, gaseous, and liquid wastes.

The maximum rate of solid waste accumulation will occur during refueling periods and the minimum during normal operation. Solid wastes, such as sampling paper, cardboard, wood, paper, broken or contaminated glassware, filter cartridges, etc., will be compressed by a hydraulic bailer into 55-gallon drums. These drums will be stored prior to shipment offsite. Spent ion-exchanger resins will be stored in a resin storage tank until a sufficient quantity has accumulated for packaging with concrete. Normally a minimum of 6 months will be allowed for decay.

Gaseous wastes will be stored in tanks until sample analysis indicates sufficient decay to warrant release to the environment. Three tanks will be provided for normal operation with one tank filling, one in decay, and the third discharging. A fourth tank will be provided to accommodate gaseous wastes resulting from unexpected plant operations, such as cold or hot shutdowns. As the gases leave the waste disposal system, they will be monitored continuously, and if an unexpected increase in radioactivity is detected, one of the discharge valves will be closed automatically on signal from the monitor.

The concentration of radioactivity in liquid wastes determines the process to be used for safe disposal. Wastes may be discharged to the waste hold-up tanks if additional delay time is warranted for radioactive decay, to the gas stripper if the purity is low and the radioactivity level is suitable for processing through the evaporator train, or to the condenser cooling waters discharge if wastes can be released within the tolerances established by Title 10, Part 20 of the Code of Federal Regulations. The gas space in the waste hold-up tanks will be filled with nitrogen of a low positive pressure to prevent accumulation of a potentially explosive mixture of hydrogen and oxygen. Liquids from the evaporators may be discharged to the evaporator concentrates processing train for filtration, removal of cations in demineralizers, and then storage in the steam jacketed concentrates holding tank. From this tank the solutions will be either transferred to the boric acid tanks, or returned to the concentrates processing train or waste hold-up tanks for reprocessing by the evaporator train. Concentrated solutions from the evaporator will be placed in 55-gallon drums, mixed with cement and ultimately shipped offsite for disposal. All liquid effluent releases will be monitored prior to release into and dilution with the condenser discharge.

4. Environmental Radioactivity

All radioactive effluents released into the Hudson River will be under controlled conditions at concentrations below the limits set by Title 10, Part 20 of the Code of Federal Regulations. Environmental radiological surveys have been in operation in the vicinity of Indian Point Station since 1958, about 4 years before Unit No. 1 began operation. These results are reported semiannually to the AEC, Docket #50-3. Monthly samples are taken of Hudson River water near the site, vegetation on the site, marine life from the river, and water from the Indian Point well. Surveys have shown that operation of Indian Point Unit No. 1 for over 3 years has had no detectable effect on the environment.

Similar results have been obtained in a 2 year post-operational survey conducted by members of the Bureau of Radiological Health Services in New York State Health Department, the Middletown District Health Office, and the Rockland and Westchester Health Department, and by biologists from the Bureau of Marine Fisheries in the New York State Conservation Department. Similar results also have been obtained in independent studies by Dr. Merrill Eisenbud, Director of Environmental Radiation Laboratory, Institute of Industrial Medicine, New York University.

5. Hydrology

The Hudson River in the vicinity of Indian Point ranges from 4,500 to 5,000 feet in width with maximum depths of 55 to 75 feet. Cross sectional areas in the vicinity are in the order of 165,000 to 170,000 square feet. The Hudson River is tidal as far as Troy, some 100 miles upstream from Indian Point. The elevation of the water surface in the vicinity of the plant is so responsive to the tidal cycle that average rate of flow has little effect on depth or velocity of flow.

The hazards of contamination of water supplies by radioactive effluent wastes from the Indian Point plant are considered minimal. In the reach of the Hudson River that could be effected, river water is used only for industrial cooling. However, the city of New York is now in the process of constructing a river water pumping station at Chelsea in Putnam County to pump Hudson River water into the County system.

6. Fisheries of the Hudson River

There are extensive commercial and sport fisheries in the Hudson River. Sport fishing is concentrated mainly on striped bass and white perch. The predominant commercial fishery is the shad fishery. During 1964, 181,865 pounds of shad were caught in the Hudson River. Approximately 149,000 pounds of this catch was caught by stake gill nets south of the Peekskill area. Less extensive commercial fisheries include herring, striped bass, American eel, sturgeon, white perch, tomcod, and American smelt. Although there are no commercial fisheries for shellfish, some oyster setting grounds exist from the New Jersey boundary north for a distance of 9 miles.

7. Fate of Radionuclides in the Aquatic Environment

When radionuclides are released into the aquatic environment various factors tend to dilute and disperse them while other factors tend to concentrate them. If the rate of dilution were the only consideration, undoubtedly the maximum permissible concentrations of radionuclides which can be disposed of as wastes would be adequate criteria in determining the maximum safe rate of discharge. However, radioactive isotopes are adsorbed onto sediments and are concentrated by organisms which require many of the stable forms of these elements for their normal metabolic activities. In addition, some organisms concentrate radioisotopes not normally required but which are chemically similar to elements essential for metabolism . Furthermore, distribution of radionuclides can occur by their transmission from one organism to another through various trophic levels of the food web and by the migration of organisms from the area.

8. Conclusions and Recommendations Concerning Radioactive Effluents

The Indian Point Pressurized Water Reactor No. 2 has been designed to operate with a minimum of environmental contamination by radioactive effluents. Radioactive materials that are released to the environment, however, must be released at a rate which will not exceed the maximum permissible limits defined in Title 10, Part 20 of the Code of Federal Regulations.

It is concluded that the Indian Point Nuclear Generating Unit No. 2 can be operated without harmful effects to the fisheries provided that the findings of the radiological monitoring program are used to govern the discharge of radioactive material.

Although it is well established that certain levels of radioactive wastes can be discharged into the aquatic environment without adverse effects on the fisheries, it is essential to determine whether such discharge adversely affects the organisms in each specific area. In view of the extensive fisheries in the Hudson River it is imperative that every effort possible be made to safeguard these fisheries. Therefore, it is recommended:

(a) That ecological surveys be initiated as soon as possible and continued on a regular basis to determine the effects of reactor effluents on plant and animal communities.

(b) That the radiological monitoring program be conducted on a quarterly basis and include representatives of the ecologically important groups of aquatic organisms and sediments.

(c) That hydrology studies in the vicinity of the plant be continued on a regular basis to provide necessary data on water flow for use in calculating dilution and dispersion of radioactive materials.

(d) That consideration be given to the combined effects of effluent discharge from all existing and planned reactors along the shores of the Hudson River.

(e) And that the Radiobiological Laboratory be placed on the distribution list to receive copies of the survey and monitoring reports for review in determining whether or not unsafe levels of radioactivity have been found in the water, sediments, or biota.

Possible Effects of Increased Water Temperature on Fishery Organisms

Large volumes of heated water discharged into an aquatic environment from a nuclear steam generating plant can result in a significant increase in the temperature of the environment near the plant. The temperature rise may or may not be sufficient to cause mortality among the organisms present, but subtle biological changes could occur causing long-term changes in the fisheries.

The thermal requirements of a fishery organism cannot be stated with any degree of accuracy. By "thermal requirements" here is meant the temperature limits which will permit survival at a level which allows for continuity of the species. These limits are influenced by season, age, size, and other factors so that the thermal requirements would be quite variable and difficult to ascertain. As a controlling factor, the thermal requirement of a particular species becomes a level which will permit sufficient difference between resting and active metabolism to provide for essential activities (Brett 1960). The increased energy demand of resting metabolism during elevated temperatures may rob an organism of the agility needed to capture its food. It has been proposed that the upper limit of required temperature for any species of fish should not exceed that which would curtail activity below 3/4 of the optimum, i.e., 3/4 of the maximum difference between active and resting metabolism (Brett 1960).

Although a temperature rise in the aquatic environment may result in a change in species composition, increases in total productivity near warm water outlets from conventional power plants have been observed. Therefore, it will be necessary to follow carefully any changes in total productivity in order to properly evaluate the effects on fishery organisms from discharged heated water.

Literature Cited

Brett, J. R. 1960. Thermal requirements of fish--3 decades of study, 1940-1970. In: Biological Problems in Water Pollution. U. S. Public Health Service, Robert A. Taft Sanitary Engineering Center, Technical Report W60-3, p. 110-117.

UNITED STATES
ATOMIC ENERGY COMMISSION
WASHINGTON, D.C. 20545

July 27, 1966

Mr. Clarence F. Pautzke
Commissioner
Fish and Wildlife Service
U. S. Department of the Interior
Washington, D. C. 20240

Dear Mr. Pautzke:

Thank you for the report of the Fish and Wildlife Service, attached to your letter of July 13, 1966, concerning the effect upon fish and wildlife of the proposed nuclear power plant of the Consolidated Edison Company of New York, Inc., Indian Point Nuclear Generating Unit No. 2, Buchanan, New York (Docket No. 50-247).

With respect to the comments on page 2 of your letter concerning the potential hazards from other than radioactive materials, the Atomic Energy Commission's regulatory jurisdiction is limited essentially to matters of radiological health and safety and the common defense and security. The Commission is without statutory authority to impose conditions in its licenses relating to the thermal and other non-radiological effects of the licensed activities. This position was explained in a letter, dated May 2, 1966, from our General Counsel to the Solicitor of the Department of the Interior. It was also reflected in our testimony last May 13 before the House Committee on Merchant Marine and Fisheries on H. R. 14455, H. R. 14414 and H. R. 9492.

With respect to the comments on page 1 of your letter concerning the State of New York, the AEC and the State of New York are presently engaged in a cooperative relationship governing the regulation of nuclear materials. The essential elements of that relationship are set forth in an agreement entered into by the State and the AEC and in an implementing memorandum of understanding. Both documents are attached.

We have been advised by the Consolidated Edison Company that there have been several informational meetings between the Company and representatives of various New York State agencies, including the State Conservation Department, concerning operation of the Indian Point plant; that frequent inspections of the plant have been made by State officials; that such meetings and plant visits will continue in the future; and that the Company would be very pleased to have Fish and Wildlife representatives participate. We have also been advised that the Company is supporting financially a Hudson River Fisheries study which includes the waters in the vicinity of Indian Point; that this study is directed by a policy committee consisting of Mr. E. L. Cheatum, Assistant Commissioner of Conservation, New York State, as Chairman; Mr. Thomas H. Schraeder, Assistant Regional Director, Fish and Wildlife Service; and Mr. A. S. Pearson, Consolidated Edison; and that Mr. Cheatum has been kept informed of the results of the thermal pollution study made at Indian Point and on a model at Alden Laboratories.

If you desire to discuss these matters further, please let me know.

Sincerely yours,

/s/ Harold L. Price

Harold L. Price
Director of Regulation

Enclosures:
1. Agreement
2. Memorandum of Understanding

APPENDIX E

REPORT TO AEC REGULATORY STAFF

ADEQUACY OF THE STRUCTURAL CRITERIA

for

CONSOLIDATED EDISON COMPANY OF NEW YORK, INC.
INDIAN POINT NUCLEAR GENERATING UNIT NO. 2

(AEC Docket No. 50-247)

by

N. M. Newmark and W. J. Hall

Urbana, Illinois

23 August 1966

NATHAN M. NEWMARK
Consulting Engineering Services
111 Talbot Laboratory
Urbana, Illinois

ADEQUACY OF THE STRUCTURAL CRITERIA

FOR

CONSOLIDATED EDISON COMPANY OF NEW YORK, INC.
INDIAN POINT NUCLEAR GENERATING UNIT NO. 2

by

N. M. Newmark and W. J. Hall

This report is concerned with the adequacy of the containment structure and components for the 2758 MWt Indian Point Nuclear Generating Unit No. 2, hereafter referred to as Indian Point Unit 2, for which application for a construction permit and operating license has been made to the United States Atomic Energy Commission by the Consolidated Edison Company of New York, Inc. The facility is located on the east bank of the Hudson River at Indian Point, village of Buchanan, in upper Westchester County, New York. The site is about 24 miles north of the New York City boundary, and 2.5 miles southwest of Peekskill, New York. Indian Point Unit No. 2 will be built adjacent to Unit No. 1.

The reactor system consists of a pressurized water reactor and steam generator which will be employed to produce steam for use in a steam-driven turbine generator.

Specifically, this report is concerned with the evaluation of the design criteria that determine the ability of the containment system to withstand a design earthquake of 0.1g horizontal and 0.05g vertical transient ground acceleration simultaneously with the other loads forming the basis of the containment design. The facility also is to be designed to withstand a maximum earthquake loading of 0.15 g horizontally and 0.1g vertically to the extent of preserving the ability to maintain the plant in a safe shut-down condition.

This report is based on information and criteria set forth in the preliminary safety analysis reports (PSAR), and supplements thereto as listed at the end of this report. We have also participated in discussions with the applicant and its representatives as well as the AEC Regulatory Staff, concerning the design of this unit.

As will be noted in the three supplements, a number of questions were raised about the design of the containment and further comments on the questions, answers and criteria cited are contained herein.

DESCRIPTION OF THE CONTAINMENT FACILITY

The reactor containment consists of a reinforced concrete shell in the form of a vertical right cylinder with a hemispherical dome and a generally flat base supported on rock. The cylinder is 135 ft. in inside diameter with a wall thickness of 4 ft. -6 in.; the spring line of the dome begins at an evelation of 147 ft. above the inside surface of the base of the containment structure, has a radius of 67 ft. -6 in. and a thickness of 3 ft. -6 in. The change in wall thickness of the dome and cylinder at the spring line is to be accomplished in such a manner that the inside radius of the dome and cylinder will be equal.

The inside surface of the structural concrete is lined with steel plate anchored to the concrete shell with studs. The bottom horizontal liner plate will be covered with 2 ft. of concrete, the top of which will form the floor of the containment.

Figure 1 of Ref. 4 shows the containment base as sitting on concrete fill in one region, with unequal backfill acting on one side of the cylindrical containment shell.

COMMENTS ON ADEQUACY OF DESIGN

Earthquake Hazard and Design Procedures

The earthquake motions considered are stated in Ref. 5 as follows:

In reply to Question 9-c, "the plant design will consider the simultaneous action of horizontal and vertical earthquake accelerations. The design earthquake accelerations at zero period are 0.1g horizontally and 0.05g vertically."

In reply to Question 9-d, the statement is made that: "The Indian Point Unit No. 2 containment will satisfy this relation for seismic loads at least equal to those corresponding to the response to 0.15g horizontal and 0.10g vertical ground accelerations occuring simultaneously."

We believe that the foregoing criteria covering the earthquake motions are reasonable and satisfactory.

The response spectra to be used in the analysis are given in Ref. 5, Figs. 9-1 and 9-2, but a plot of spectra are not given for the maximum earthquake. We have considered that the response spectra to be used for maximum earthquake are proportional to those used for the design earthquake. The applicant has stated that the combined effects due to vertical and horizontal earthquake motions will be assumed to act simultaneously in the design.

The damping values, as revised, are given in Ref. 5 in the answer to question 9-a. We consider that these damping factors are acceptable, as stated in this reference. Criteria for "no loss of function" are stated in reply to question 6 of Ref. 6, and appear adequate. A ductility factor of two (2) is to be used in the design of all Class I vessels and piping.

The applicant has informed the Staff that any Class I equipment located in a Class II building, or supported by a Class II structure, will be protected from damage during an earthquake, or will be backed up with Class I equipment, capable of providing for a safe reactor shut down, located in or attached to Class I structures. We concur in this approach.

Penetrations

The applicant describes the method of analysis of penetrations in the answer to question 2 of Ref. 6.

The method is essentially an empirical one and should be adequate. An indication of assurance of adequacy can be obtained in one of several ways: for example, by theoritical analysis using a lumped parameter or finite element representation; by photoelastic analysis; by model tests; or by adequate measurements made during proof-pressure tests of the completed structure. The applicant has informed the Staff that assurance of adequacy of the large penetrations will be provided through measurements and observations made at the time of the containment proof test. Such measurements will include (a) strain measurements to be made in the area of the stiffening ring and in areas adjacent to the opening, (b) visual inspection for cracking and (c) measurements of gross dimensional changes. We believe that acceptable results can be obtained from such measurements and that these would assure the adequacy of this aspect of the design.

Steel Liner

The design of the liner and the attachment to the concrete pressure vessel is discussed in the answer to question No. 1 of Ref. 6. We consider that a plate thickness of 3/8 of an inch, as indicated in Ref. 6, can have adequate resistance to fatigue or repeated stresses if the welding procedures are carefully controlled. Hence, an inspection procedure is essential in which all of the stud connections to the plate and liner welds are examined. The applicant advises that 100 percent of all liner stud welds will be visually inspected, and that all liner seam welds will be pressure tested.

Concrete Reinforcement

The principal reinforcement in the dome and cylindrical shell containment vessel is listed in Ref. 4 as being "high strength billet steel conforming to ASTM A-432 with a guaranteed minimum yield strength of 60,000 psi and ultimate minimum strength of 90,000."

Since this steel has a lower ductility that the lower strength steel commonly employed, adequate inspection and control procedures are essential to insure that the steel meets the requirements of the specifications; the Staff has been assured that an acceptable arrangement for such procedures will be provided.

Concrete Shear Values

A discussion of the shearing strength of concrete under various conditions of combined loading is contained in the answers to questions 7 and 8 of Ref. 6. The applicant has informed the Staff that these latter statements mean that diagonal reinforcement will be provided to carry the entire seismic shear without participation of the liner or the concrete, except for the upper area of the dome. Also the applicant has confirmed that shear will not be considered to be carried by diagonal bars when they are loaded in compression. In our view, this interpretation gives an adequate capacity for shearing resistance of the containment under transverse loading.

Backfill

The structure is subjected to dead load pressures and to increased lateral forces in the transverse direction arising from the action of the crushed-rock backfill against the structure. Since this backfill is not at the same elevation around the entire structure, the lateral force distribution on the structure arising from both dead load and seismic loading are not uniformly distributed circumferentially. Although the answer to question 5 of Ref. 6 discusses this problem, the discussion appears to be limited to the state of stress in the soil. The applicant has informed the Staff that he will take account of these increased lateral forces due to seismic behavior in proportioning the concrete and steel in the containment vessel.

Earthquake Effects on Crane

The stability of the crane under seismic motions is discussed in the reply to question 4 of Ref. 6.

The statement is made that "the seismic design also precludes tipping of the crane and the reaction of seismic loads." Hence, the factors to be considered involve forces imposed on the crane structure from swinging loads, or the impact of such swinging loads on other parts of the structure. It is apparent from the discussion that the applicant has considered this matter. We understand that the capability of the reactor for safe shut down will not be impaired by earthquake motions that might be transmitted to the crane or through the crane to other elements.

CONCLUSIONS

On the basis of the application and discussions at several meetings on this topic, we believe that the principal structures and components designed for containment, and the other essential parts of the facility, will provide an adequate margin of safety for seismic motions.

1. "Preliminary Safety Analysis Report - Description of Site and Environment," Consolidated Edison Company of New York, Inc. Indian Point Nuclear Generating Unit No. 2, USAEC Docket No. 50-247, Exhibit B, Vol. I, 1966.

2. "Preliminary Safety Analysis Report - Plant Design Description and Safety Analysis," Consolidated Edison Company of New York, Inc., Indian Point Nuclear Generating Unit No. 2, USAEC Docket No. 50-247, Exhibit B, Vol. II, Part A, 1966.

3. "Preliminary Safety Analysis Report - Plant Design Description and Safety Analysis," Consolidated Edison Company of New York, Inc., Indian Point Nuclear Generating Unit No. 2, USAEC Docket No. 50-247, Exhibit B, Vol. II, Part B, 1966.

4. "First Supplement to: Preliminary Safety Analysis Report", Consolidated Edison Company of New York, Inc. Indian Point Nuclear Generating Unit No. 2, USAEC Docket No. 50-247, Exhibit B-1, 1966.

5. "Second Supplement to: Preliminary Safety Analysis Report," Consolidated Edison Company of New York, Inc., Indian Point Nuclear Generating Unit No. 2, USAEC Docket No. 50-247, Exhibit B-2, 1966.

6. "Third Supplement to : Preliminary Safety Analysis Report," Consolidated Edison Company of New York, Inc., Indian Point Nuclear Generating Unit No. 2, USAEC Docket No. 50-247, Exhibit B-3, 1966.

Appendix F

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

July 23, 1966

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 the Indian Point, New York area. The Coast and Geodetic Survey has reviewed and evaluated the information on the seismicity of the area presented by the Consolidated Edison Company for a license to construct and operate a nuclear reactor in Indian Point, New York.

If we may be of further assistance to you please do not hesitate to contact us.

Sincerely yours,

/s/ James C. Tison, Jr.

James C. Tison, Jr.
Rear Admiral, USESSA
Director

Enclosure

Appendix F

REPORT ON THE SEISMICITY OF THE INDIAN POINT, NEW YORK AREA

In response to the request of the Division of Reactor Licensing of the Atomic Energy Commission, the Seismology Division of the Coast and Geodetic Survey has reviewed the seismicity of Indian Point, New York as submitted by Consolidated Edison Company of New York to the AEC on December 7, 1965.

The history of seismic events in the Indian Point area as prepared by Reverend J. J. Lynch, S.J., for data up to 1955 and by Dr. James Dorman for data up to 1963 are in complete agreement with our knowledge of the area seismicity during historic time. A check was also made of our files from 1963 through May 1966 and no additional earthquake reports were found. The intensities of these earthquakes do not exceed 6 on the Modified Mercalli Scale, indicating that the strongest have caused but very slight damage. Moreover, the highest intensity earthquakes that have occurred in the St. Lawrence Valley and coastal New England areas have never been damaging around Indian Point. In evaluating this historic information about the intensity of the earthquakes and realizing that the proposed structure would be constructed on rock formation, the Survey is in agreement with the applicant's statement that 0.1 g is adequate for the design of the reactor plant and containment. This 0.1 g is considered adequate even though there is evidence of much tectonic movement during geologic time. However, recent tectonic history indicates only minor activity which is in general characteristic of the Appalachian Mountain Chain.

In summary, the Survey believes that within the lifetime of the facilities located on rock at Indian Point, an acceleration of 0.1 g in the period range of 0.3 to 0.6 without the loss of function of components important to safety should be taken into account.

U. S. Coast and Geodetic Survey
Washington, D. C.

June 23, 1966