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

August 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,

/s/
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.