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

March 4, 1961

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

Subject: REPORT ON CONSOLIDATED EDISON REACTOR DESIGN

Dear Dr. Seaborg:

At its thirty-first meeting on January 12-14, 1961, and its thirty-second meeting on March 2-4, 1961, the Advisory Committee on Reactor Safeguards considered the design of the Consolidated Edison 585 MW thermal uranium oxide thorium oxide fueled pressurized water reactor under construction at Indian Point near Peekskill, New York. The Committee had the benefit of discussions with the applicant, the AEC staff and their consultants. The Committee explored features of the design with the applicant, the applicant's contractors and the applicant's consultants.

Many questions raised by the Committee have been answered to the Committee's satisfaction. Although the applicant has not yet documented his proposals on control rod mechanisms and core information system, an adequate oral description was given to the Committee and the following is based on that presentation.

The question of integrity of control rod mechanism of water reactors has been raised by recent failures in other reactors of 17-4 PH stainless steel parts attributed to stress corrosion. The Consolidated Edison control rod mechanisms have been fabricated and are now undergoing an extensive testing program. The Committee notes that Consolidated Edison has placed on order a duplicate set of control rod drive shafts heat treated at 1100°F., instead of 900°F. specified for the present rods, in accordance with optimum specifications as indicated by the recent extensive survey of experience in other reactors. The Committee believes that the design of these units and the applicant's proposed modifications in fabrication will not result in a hazard to the health and safety of the public. The Committee suggests that the staff follow the fabrication details of these units.

In response to the several discussions between the HEB staff, the Committee and Consolidated Edison concerning the absence of in-core monitoring devices or other means of measurement of local power levels, Consolidated Edison proposed on March 3 that they will remove and gamma scan a portion of the fuel elements. This should provide a reasonable check on precalculated core performance at an intermediate point or points during the core life. Consolidated Edison expressed confidence that this scanning, and other operating procedures yet to be worked out in detail, will enable them to operate the core without in-core monitors and without excessive heat and neutron fluxes in any part of the reactor. The frequency and time in core life when these measurements will be taken will be discussed with the applicant as a part of the review of operating procedures. The Committee believes that there is considerable assurance that the reactor, as designed, can operate at designed power. However, the question of whether full power operation can actually be reached without in-core monitors will depend upon data which can be obtained only during the initial operation of the reactor at power levels less than full power.

The Committee's attention thus far has been directed solely at design of the Consolidated Edison facility. Staffing, operating procedures, start-up program, etc., will be considered following a review of the applicant's reports not yet furnished.

The ACRS finds the Consolidated Edison reactor design such that it may be operated without undue risk to the health and safety of the public.

Sincerely yours,

/s/ T. J. Thompson

T. J. Thompson
Chairman

References:

1. Report on Hazards Analysis and design for Containment Vessel, dated Sept. 18, 1959.
2. Hazards Summary Report, dated Jan. 1960.
3. Amendment #11, dated April 21, 1960, to the Application for Licenses.
4. Amendment No. 7 to Application for Licenses, undated, received April 27, 1959.
5. Reactor Vessel Internal Components Design (BAW-136), dated July 1960.

6. Fuel Element Structural Design and Manufacture (BAW-133), dated Sept. 1960.
7. Design of the Movable and Fixed Control Components (BAW-147), dated Aug. 1960.
8. Irradiation Test Program (BAW-134), dated Aug. 1960.
9. Thermal and Hydraulic Design (BAW-132), dated July 1960.
10. Physics Design (BAW-120, Rev. 1), revised July 1960.
11. Critical Experiments with Oxide Fuel Pins (BAW-119, Rev. 1), dated July 1960.
12. Hot Exponential Experiment (BAW-116, Rev. 1), revised June 1960.
13. Geometric and Temperature Effects in Thorium Resonance Capture (BAW-144), dated June 1960.
14. Control Rod Drive Line Testing, dated Aug. 1960.
15. Supplementary Information on Plant Design of Con-Ed Nuclear Steam Generating Station, dated Aug. 1960.
16. Functional Design Analysis of the Pressurizer (BAW-41, Rev. 1), revised June 1960.
17. The Effects of Fuel Rod Fission Product Leakage (BAW-85, Rev. 1), revised June 1960.
18. Corrosion Product Activity Distribution Across the Chemical Processing System (BAW-142, Rev. 1), revised Aug. 1960.
19. Control System Design (BAW-138), dated Aug. 1960.
20. Amendment #14, dated Nov. 23, 1960 and attachments to the Application for Licenses.
21. Amended and Substituted Application for Licenses, dated Nov. 30, 1960 and exhibits.
22. Amendment #1, dated Dec. 9, 1960, to the Amended and Substituted Application for Licenses.
23. Amendment #2 and attachments, dated Feb. 14, 1961, to the Amended and Substituted Application for Licenses.

cc: A. R. Luedecke, GM
W. F. Finan, AGMRS
H. L. Price, Dir., DL&R