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Richard C. DeYoung, Assistant Director
for Pressurized Water Reactors
Directorate of Licensing

CONSOLIDATED EDISON OF NEW YORK, INC., INDIAN POINT NUCLEAR GENERATING STATION, UNIT NO. 3, (OL), DOCKET NO. 50-286

Plant Name: Indian Point Nuclear Generating Station, Unit 3
Licensing Stage: OL
Docket Number: 50-286
Responsible Branch and Project Manager: PWR-1; H. Specter
Requested Completion Date: March 16, 1973
Description of Response: Technical Specification Review
Review Status: Complete

The proposed Technical Specification submitted by the applicant has been reviewed and evaluated by the Materials Engineering Branch, Directorate of Licensing. Our sections of the Technical Specification review are enclosed.

Original signed by
Maccary

R. R. Maccary, Assistant Director
for Engineering
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Enclosure:
Materials Engineering Branch Tech
Spec Review for Indian Point 3

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CONSOLIDATED EDISON CO. OF NEW YORK
INDIAN POINT NUCLEAR GENERATING STATION, UNIT 3
(OL)
DOCKET NO. 50-286
TECHNICAL SPECIFICATION REVIEW

MATERIALS ENGINEERING BRANCH, L

3.1.B Heatup and Cooldown

The fracture toughness properties of the ferritic materials in the reactor coolant pressure boundary were determined in accordance with Section III of the ASME Boiler and Pressure Vessel Code, 1972 Summer Addenda, including Appendix G. The heatup and cooldown Technical Specification including the curves in Figures 3.1-1 and 3.1-2 are acceptable.

3.1.E Maximum Reactor Coolant Oxygen, Chloride and Fluoride Concentration

By maintaining the oxygen, chloride, and fluoride concentrations below the limits shown in this Technical Specification the integrity of the reactor coolant system with regard to corrosion is assured under all operating conditions. These limits are the same as those used in previously licensed, similar plants and are acceptable.

3.1.F Leakage of Reactor Coolant

This Technical Specification is not acceptable as it does not clearly give limits for nor define identified and unidentified leakages from the reactor coolant pressure boundary. We recommend that the Technical Specification be rewritten using Surry Units 1 and 2 Technical Specification (TS)

3.1.C Leakage as a guide. The basis for the Indian Point 3 Technical Specification is acceptable and indicates that the systems are sensitive enough to implement the Surry Technical Specification requirements.

4.2 Primary System Surveillance

The inservice inspection program, within the limits of accessibility designed into the plant, meets the requirements of Section XI of the ASME Boiler and Pressure Vessel Code, 1970 Edition. The exclusions from the inspection program are identified. The program as presented in Table 4.2.1 of the Technical Specification is acceptable.

4.3 Reactor Coolant System Integrity Testing

This Technical Specification (TS) is not acceptable as it does not reference temperature requirements during leak testing to TS 3.1.B, Heatup and Cooldown. In addition, the applicant should add a curve of temperature versus pressure for the inservice hydrostatic leak tests of the reactor coolant pressure boundary.

4.4 Containment Tests

This Technical Specification is not acceptable in the following areas:

4.4-I.A.1 It should be clearly established that the reduced pressure test precedes the accident pressure test. The applicant uses the term "design pressure" where he means "peak accident Pressure."

<u>4.4-I.A.2</u> It should be clearly established that the 24-hour test interval pertains to both the reduced and accident pressure tests for a total duration of not less than 24 hours for each component.

<u>4.4-I.A.4</u> The implication of this section is that a steam-air mixture test at 47 psig and 271°F will be performed rather than an air test at peak accident pressure and ambient temperature. Test conditions should be clearly stated.

<u>4.4-I.A</u> Acceptance criteria in agreement with Paragraph III.A.4 of Appendix J, 10 CFR 50 (February 14, 1973) should be added.

<u>4.4-II.A</u> Acceptance criteria in agreement with Paragraph III.A.5 of Appendix J should be added.

In general those isolation valves which are not part of the Penetration and Weld Channel Pressurization system should be listed separately with the following data: the component system that the valve is part of, number of similar valves in the system, test pressure and where and how applied, pressurant, test frequency, and test duration. This information lends itself to a tabular presentation.

In general this Technical Specification should be referenced to Appendix J of 10 CFR 50 (February 14, 1973) and ANSI N45.4-1972 "Leakage Rate Testing of Containment Structures for Nuclear Reactors" (dated March 16, 1972). A section should be added using Appendix J, Paragraph III.D for guidance in regard to airlock retest schedules.