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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 2, 1973
Applicant's Response Date Necessary for Completion of Next Planned Action on Project: Not Applicable
Description of Response: Safety Evaluation
Review Status: Complete

The information submitted by the applicant, including Amendment No. 24, has been reviewed by the Materials Engineering Branch, Directorate of Licensing. Our sections of the Safety Evaluation Report are enclosed.

The applicant has submitted all of the required information.

Original signed by
R. R. Maccary

R. R. Maccary, Assistant Director
for Engineering
Directorate of Licensing

Enclosure:
Materials Engineering Branch Safety
Evaluation for Indian Point 3

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

SAFETY EVALUATION

MATERIALS ENGINEERING BRANCH, L

REACTOR COOLANT SYSTEM

Fracture Toughness

To assure compliance with the safety and design criteria, ferritic materials of pressure retaining components of the reactor coolant pressure boundary must exhibit adequate fracture toughness properties under normal reactor operating conditions, system hydrostatic tests, and during transient conditions to which the system may be subjected. We have reviewed materials testing and the operating limitations proposed by the applicant.

The applicant has stated in the FSAR, Amendment Nos. 23 and 24, Supplement Nos. 9 and 10, respectively, that acceptance testing for ferritic materials was performed in accordance with the requirements of the ASME Boiler and Pressure Vessel Code, Section III (1971 Edition, including Addenda through Summer 1972). Dropweight NDT data have been obtained for the reactor vessel material.

In establishing the operating pressure and temperature limitations during heatup, cooldown, and inservice hydrostatic tests of the system, the applicant has followed the recommendations of Appendix G, "Protection Against Non-Ductile Failure," of the 1972 Summer Addenda of the ASME Code, Section III.

The applicant has submitted specific heatup, cooldown, and hydrostatic test limitation curves, which meet the current fracture toughness requirement.

We conclude that the planned operation of the reactor coolant system will assure adequate margins of safety.

REACTOR COOLANT SYSTEM

Reactor Vessel Material Surveillance Program

A material surveillance program is required to monitor changes in the fracture toughness properties of the reactor vessel beltline material induced by neutron radiation.

The applicant has shown in the FSAR, Amendment Nos. 21 and 23, Supplement Nos. 7 and 9, that the proposed materials surveillance program, although differing in minor details, is technically equivalent to the requirements of the Commission's proposed Appendix H, 10 CFR Part 50, 50.55(a). The only significant difference is that to obtain the optimum relationship between the fluences seen by the vessel wall and the capsules, the capsules will have to be rotated from one location to the other during the service life of the vessel. The program is acceptable with respect to the number of capsules, number and type of specimens, and retention of archive material. The proposed withdrawal and rotation schedule will provide adequate monitoring of radiation effects occurring in the vessel material. We have concluded that the proposed program will adequately monitor neutron induced changes in the fracture toughness of the reactor vessel material.

REACTOR COOLANT SYSTEM

Sensitized Stainless Steel

Stainless steel that has been sensitized has an increased susceptibility to stress corrosion cracking.

The applicant has shown by the FSAR, Appendix 4D, and by Amendment Nos. 21 and 23, Supplement Nos. 7 and 9, respectively, that significant sensitization of all nonstabilized austenitic stainless steel within the reactor coolant pressure boundary was avoided through materials selection and control of welding and heat treating processes. The precautions included: (1) use of approved procedures for welding and verification of them by periodic quality control checks; (2) use of low heat input procedures during shop and field welding operations; (3) check of core structures by the Strauss test; (4) not allowing use of wrought furnace sensitized stainless steel, and (5) limiting interpass temperatures during welding to 350°F maximum. Where stainless steel safe ends were welded to the vessel, the weld preparation of both the safe end and the nozzle were built up with Inconel.

We conclude that the planning to avoid sensitization of austenitic stainless steel during the fabrication period is acceptable.

REACTOR COOLANT SYSTEM

Evaluation of the Integrity of the Reactor Vessel

During installation of the reactor vessel, a hoist failed, and the vessel was dropped. A reinspection of the vessel was performed, which involved dimensional checks, visual examination, and nondestructive examination by magnetic particle, liquid penetrant, and ultrasonic methods. The results obtained from the nondestructive examinations subsequently served as a basis for assessment of possible damage to the vessel using stress analysis and fracture mechanics criteria.

A report prepared by Oak Ridge National Laboratory entitled, "Summary Report and Reinspection and Appraisal of the Indian Point Unit No. 3 Reactor Pressure Vessel Subsequent to Hoist Failure on January 12, 1971," covering the above incident and the subsequent reinspection and evaluation has been submitted to Licensing by the applicant.

Our review of the report revealed that the nondestructive examination techniques which were used were equal or better than those specified by the ASME Boiler and Pressure Vessel Code, Section III, and in fact permitted a more comprehensive examination than that originally performed which used the Code specified methods. No rejectable defects were disclosed as a result of the above indicated inspection, even though additional discontinuities were shown to be present in excess of those originally reported.

Appendix "C" of the report, which is in two parts, contains an assessment of the effects of this incident based on stress analysis and fracture mechanics. This appendix has been reviewed and evaluated.

The procedure in the first part of this appendix is inappropriate due to assumptions made relating to the stress, the imposed stress intensity, and the toughness. In the second part the toughness value that was used agrees well with an estimated lower bound reference toughness from the ASME Code, Section III, Appendix G, 1972 Summer Addenda. We believe that the calculated maximum bending stress is realistic. A critical flaw depth of approximately 4 inches was calculated. Our independent calculations, performed according to the procedures of Welding Research Council Bulletin No. 175, PVRC Recommendations on Toughness Requirements for Ferritic Materials, August 1972, confirm the results of this calculation. Further, using conservative assumptions, we have estimated that a 4 inch deep flaw, assumed to exist in the most deleterious location and orientation, would have grown less than 0.001 inch due to this incident.

We concur with the findings of the report that no rejectable defects were disclosed, and that any existing flaws would not have been significantly extended as a consequence of this incident. We conclude that the integrity of the reactor vessel has not been impaired by the drop which resulted from the hoist failure.

REACTOR COOLANT SYSTEM

Pump Flywheel Integrity

The probability of a loss of pump flywheel integrity, which could result in high energy missiles and excessive vibration of the reactor coolant pump assembly, can be minimized by the use of suitable material, adequate design and inspection.

The applicant has stated in Amendment No. 21 in response to Question 4.7.1 that the design, fabrication, and preservice and inservice inspections of the pump flywheels are in general accord with AEC Regulatory Guide 14, "Reactor Coolant Pump Flywheel Integrity." We conclude that the design, fabrication, and inspection of the flywheels are acceptable.

REACTOR COOLANT SYSTEM

Inservice Inspection Program - Primary System

Selected welds and weld heat-affected zones must be inspected periodically to assure continued integrity of the reactor coolant pressure boundary during the service lifetime of the plant.

The applicant has stated in Amendment No. 21 in response to Question 4.9 that the inservice inspection program for the reactor coolant pressure boundary will comply with Section XI of the ASME Boiler and Pressure Vessel Code, "Rules for In-Service Inspection of Reactor Coolant Systems," 1970 Edition. Access for inservice inspection was provided in the design and arrangement of pressure-containing components.

The facility was constructed to allow either external or internal inspection of the reactor vessel using a remotely operable inspection tool capable of performing inspections of vessel surfaces, circumferential, longitudinal, and nozzle welds.

The structural integrity of the reactor coolant system boundary is to be maintained at the level of the original acceptance standards.

We conclude that the access provisions and planning for inservice inspection are acceptable. The provisions of the AEC Guideline, "Inservice Inspection Requirements for Nuclear Power Plants Constructed with Limited Accessibility for Inservice Inspection," (January 31, 1969) have been satisfied.

REACTOR COOLANT SYSTEM

Leakage Detection System

Coolant leakage within the reactor containment may be an indication of a small through-wall flaw in the reactor coolant boundary.

The leakage detection system provided for the reactor coolant pressure boundary includes diverse leak detection methods, has sufficient sensitivity to measure small leaks, and has suitable control room alarms and readouts. The major components of the system are the containment atmosphere particulate and gaseous radioactivity monitors, main air recirculation unit condensate coil collection and measurement system, and level indicators on the containment sump. Indirect indication of leakage can be obtained from the containment humidity, pressure and temperature indicators. We conclude that the leakage detection system has the capability to detect leakage from small through-wall flaws in the reactor coolant pressure boundary.

CONTAINMENT

Leakage Testing Program

Leakage testing of the reactor primary containment and associated components is intended to provide preservice and periodic verification of the leaktight integrity of the containment.

The applicant has stated in the FSAR in paragraph 5.1.7 that the primary reactor containment and its components have been designed so that periodic integrated leakage rate testing can be conducted at a test pressure corresponding to the calculated peak accident pressure.

Penetrations, including personnel and equipment hatches, airlocks, and isolation valves, have been designed to provide individual leak testing at calculated peak accident pressure.

We conclude that the containment system will permit containment leakage rate testing in compliance with the AEC Rule, "Reactor Containment Leakage Testing for Water Cooled Power Reactors," 10 CFR 50, Appendix J, and is acceptable.

ENGINEERED SAFETY FEATURES

Inservice Inspection Program - Other Category I Systems

The applicant has provided access to the Group B and C fluid systems such as the engineered safety systems, reactor shutdown systems, cooling water systems, and the radioactive waste treatment systems outside the limits of the reactor coolant pressure boundary for inservice inspection. Consolidated Edison stated in Amendment No. 22 in response to Question 4.11 that when ASME Section XI of the Boiler and Pressure Vessel Code is revised to include additional system requirements, in the above areas, that these requirements will be evaluated for application to Indian Point Unit No. 3. We conclude that the planning for an inservice inspection program for the Group B and C fluid systems is adequate.