OCT 8 1971

Peter A. Morris, Director, Division of Reactor Licensing

INDIAN POINT NUCLEAR STATION, UNIT 3, DOCKET NO. 50-286

The information submitted by the applicant in the FSAR has been reviewed and evaluated by the Materials Engineering Branch, DRS. Our evaluation of the issues reviewed is enclosed. Tentative conclusions or statements for which confirmation is still required are enclosed in parentheses; the material in brackets identifies open issues or problems which will require further actions to resolve.

DKT # 50-286

Original Signed By
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Edson G. Case, Director
Division of Reactor Standards

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DATE ▶	9/27/71	9/27/71	10/7/71	10/7/71	10/7/71	

Form AEC-318 (Rev. 9-53) U.S. GOVERNMENT PRINTING OFFICE: 1969-O-364-598

INDIAN POINT GENERATING STATION, UNIT 3, DOCKET NO. 50-286

MATERIALS ENGINEERING BRANCH, DRS

INTERIM OL EVALUATION

REACTOR COOLANT SYSTEM

Fracture Toughness Criteria

The applicant has stated that the reactor vessel will be designed in accordance with the ASME Boiler and Pressure Vessel Code, Section III. Recent fracture toughness test data, however, indicate that the current ASME Code rules do not always assure adequate fracture toughness of ferritic materials. The fracture toughness data submitted by the applicant meet the current requirements of Section III of the ASME Code, but are not adequate to establish compliance with the proposed AEC "Fracture Toughness Requirements", §50.55a, Appendix G.

We have reviewed the available fracture toughness data for the reactor vessel and applied our proposed fracture toughness criteria to arrive at a lowest pressurization temperature of 210°F.

We intend to specify the following limits in the Technical Specifications, to be applicable during the first two years of operation, or until the first material surveillance specimens are withdrawn, whichever comes first.

1. The reactor coolant system should be operated in such a manner that at temperatures below 210°F, the pressure does not exceed 550 psig (i.e., 25% of the normal operating pressure).

2. Operation of the reactor coolant system at full pressure is acceptable at temperatures above 210°F.

3. The reactor coolant system can be subjected to isothermal hydrostatic tests at temperatures below 210°F, provided that the test pressure does not exceed 1100 psig (i.e., 50% of the normal operating pressure).

(We anticipate that the applicant will accept these operating limitations.)

REACTOR COOLANT SYSTEM

Reactor Vessel Material Surveillance Program

The proposed material surveillance program is consistent with programs that have been accepted on previous PWR plants, and is acceptable with respect to the total number of specimen capsules, number of capsules to be withdrawn and tested, archive material provisions, and material chemistry documentation. (The program essentially complies with the proposed AEC "Reactor Vessel Material Surveillance Program," §50.55a, Appendix H.) We conclude that the proposed program will adequately monitor neutron radiation induced changes in the fracture toughness of the reactor vessel material.

REACTOR COOLANT SYSTEM

Sensitized Stainless Steel

The applicant has stated that the sensitization of stainless steel will be avoided. The precautions used to ensure this will include adherence to restrictions included in process and fabrication procedures, such as a 350°F limit on stainless steel weldment interpass temperature, that will permit the finished component parts to pass a Strauss Test (ASTM A-393) for corrosion susceptibility. Ferritic nozzle and pipe ends will be buttered with stainless steel applied by the weld deposition technique.

The applicant has been requested to formally submit the information regarding use of nitrogen bearing stainless steel in the reactor coolant pressure boundary. We expect to receive confirmation from the applicant if nitrogen bearing stainless steel will be used.

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

REACTOR COOLANT SYSTEM

Electroslag Welding

The reactor coolant system will contain large diameter stainless steel pipe elbows and pump casings which are electroslag (E-S) welded. The ASME Boiler and Pressure Vessel Code, Section III and Code Case 1355 requirements were supplemented with evaluation tests requested by Westinghouse. We conclude that the actions taken are adequate to obtain quality E-S welds.

REACTOR COOLANT SYSTEM

Pump Flywheel Integrity

The reactor coolant pump flywheels will be of standard Westinghouse design, fabricated of vacuum melt and degassed ASTM A-533B steel. The finished flywheels have been subjected to 100 percent volumetric UT inspection. Finished machined bores have been subjected to a magnetic particle or liquid penetrant examination. The primary stresses will not exceed 1/3 or 2/3 of the minimum specified yield strength at the normal operating speed and the design overspeed, respectively. (The proposed inservice inspection program consists of ultrasonic inspection of the flywheel keyways by sighting from the four gage holes. This inspection can be performed with the flywheel keyed to the motor shaft, after removal of the flywheel cover.)

(We conclude that the proposed design, fabrication and inspection procedures comply with our recommendations listed in the proposed AEC Safety Guide, approved by the ACRS, "Reactor Coolant Pump Flywheel Integrity," dated July 7, 1971.)

To verify adequacy of the fracture toughness of the flywheel material, the applicant has proposed to test a minimum of three Charpy V-notch (C_V) specimens from each place, parallel and normal to the rolling direction.

(An acceptance criterion proposed is an average C_v impact value of 30 ft. lb. at 10°F, with no value lower than 25 ft. lb.)

Because of the well known uncertainties involved in the interpretation of the C_v data, and since the normal operating temperature of pump flywheels is closer to 100°F than to 10°F, we recommend that the applicant be asked to demonstrate adequate fracture toughness of the flywheel material using the criteria recommended in Section C.1. of the proposed AEC Safety Guide.

We recommend that the applicant document his acceptance of the above criteria prior to the ACRS Meeting.

[We have asked the applicant to provide the maximum rotational speed the pump assembly could attain in the event of a pipe rupture in the discharge or suction side of the pump due to physical limitations such as binding and seizure.]

REACTOR COOLANT SYSTEM

Inservice Inspection Program

The specific details of the inservice inspection program have been established on the basis of complying with Section XI of the ASME Code "Inservice Inspection of Nuclear Reactor Coolant Systems" (January 1, 1970) to the extent possible within the limitations present in the plant design at the time of publication of the draft Code.

For each area to be inspected, the extent, method, and frequency of inspection have been identified. The inside surface of the reactor vessel may be inspected by removing the reactor internals. Other inspection areas may be inspected by removing the shielding and insulation surrounding the areas. Some portions of the reactor coolant system are not inspectable because they are not accessible.

(The applicant is participating in a development program for remote inservice inspection systems. Reliable systems, developed in the future, that improve the capability to perform inservice inspections in high radiation areas will be incorporated into the inservice inspection program.)

(The proposed inservice inspection program is deemed acceptable because it satisfies the provisions of the AEC guidelines "Inservice Inspection Requirements for Nuclear Power Plants Constructed with Limited Accessibility for Inservice Inspection" (January 31, 1969).)

REACTOR COOLANT SYSTEM

Leakage Detection System

The applicant proposes to investigate each indication of leakage to identify the source of leakage and to determine if operation can continue safely. If the leakage exceeds 1 gpm and the source is not identified or if the total leakage deemed acceptable for continued operation exceeds 10 gpm, the reactor shall be placed in the hot shutdown condition within 24 hours utilizing normal operating procedures. If the leakage exceeds the limit for an additional 24 hours, the reactor shall be placed in the cold shutdown condition utilizing normal operating procedures.

The applicant's basis for the 1 gpm leakage limit is that 1 gpm is sufficiently above the minimum detectable leakage rate to provide a reliable indication of leakage; the applicant's basis for the 10 gpm leakage limit is that 10 gpm is about 10% of the make up capability of one charging pump (98 gpm).

Positive indications of coolant leakage within the containment are presented in the control room on the monitors of containment air radioactivity, humidity and condensate. The normal background readings on these monitors indicate the basic level of leakage within the containment. Any increase in the readings may be caused by an increase in leakage from the reactor coolant system.

The applicant proposes to locate leaks after the plant is shutdown. The leaks are located by a visual survey of the equipment inside the containment for evidence of water or boric acid crystals, or by a sonic survey for evidence of ultrasonic frequencies.

(The information presented by these monitoring systems is sufficient to enable the operator to maintain the total leakage as a reasonable fraction of the make up capability. Additional information about the ability of the systems to locate leaks, the relationship of crack size and leak rate, and test that are to demonstrate the sensitivity and operability of the leakage detection systems has been requested.)

CONTAINMENT

Leakage Testing Program

The preoperational integrated leakage rate test will be performed at the calculated peak containment pressure (P_p). Periodic integrated leakage rate tests will be performed at one-half of the calculated peak containment pressure. The preoperational and periodic component leakage tests will be performed at the calculated peak containment pressure.

[The acceptance criteria for the component leakage tests and test schedule for the integrated leakage test do not meet the requirements proposed in the AEC Appendix J. We recommend the applicant be requested to document his intent to comply with the proposed regulation in these areas and to provide a test schedule for the air locks.]

[Additional information about the location, number and type of instruments used for the leakage tests and the methods for converting the observed parameters to leak rate has been requested to evaluate the adequacy of the integrated leakage test.]

(The applicant's program complies with the AEC proposed "Reactor Containment Leakage Testing for Water Cooled Reactors", §50.54(o) Appendix J. We conclude that the leakage rate testing program provides an acceptable means for demonstrating the integrity of the containment barrier.)