JUN 28 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, including Amendment 13, has been reviewed by the Materials Engineering Branch, DRS. Adequate responses to the enclosed request for additional information are required before we can complete our evaluation.

DKT # 50-286

Original Signed By
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Enclosure:
Materials Engineering Request
for Additional Information
Indian Point 3

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Form AEC-318 (Rev. 9-53) AECM 0240 U. S. GOVERNMENT PRINTING OFFICE : 1970 O - 405-346

INDIAN POINT NUCLEAR STATION, UNIT 3

(Docket No. 50-286)

REQUEST FOR ADDITIONAL INFORMATION

REACTOR COOLANT SYSTEM

Fracture Toughness

Recent fracture toughness test data indicate that the current ASME Code rules do not always assure adequate fracture toughness of ferritic materials. The Charpy V-notch tests are adequate to measure the upper shelf fracture energy value; however, they generally do not predict correctly the Nil Ductility Transition (NDT) temperature or the transition temperature region in which fracture toughness increases rapidly with temperature. The NDT temperature, therefore, must be obtained from other tests, such as the drop-weight test (DWT). In addition, the transition temperature region shifts to higher temperatures when the thickness of the specimen tested is increased (size effect). In order to be able to establish appropriate heatup and cool-down limits for this plant, provide the following information:

1. For all pressure-retaining ferritic components of the reactor coolant pressure boundary whose lowest pressurization temperature* will be below 250°F, provide the material toughness test requirements and data (Charpy V-notch impact test curves and dropweight test NDT temperature, or others) that have been specified and reported for plates, forgings, piping, and weld material. Specifically, for each

*Lowest pressurization temperature of a component is the lowest temperature at which the pressure within the component exceeds 25 percent of the system normal operating pressure, or at which the rate of temperature change in the component material exceeds 50°F/hr., under normal operation, system hydrostatic tests, or transient conditions.

component provide the following data, or your estimates based on the available data:

(a) The maximum NDT temperature as obtained from DWT tests,

(b) The maximum temperature corresponding to the 50 ft-lb value of the C_v fracture energy, and

(c) The minimum upper shelf C_v energy value for the "weak" direction (WR direction in plates) of the material.

2. Identify the location and the type of the material (plate, forging, weld, etc.) for which the data listed above were obtained. Where these fracture toughness parameters occur in more than one plate, forging or weld, provide the information requested in 1.(a), (b) and (c) for each of them.

3. For reactor vessel beltline materials, including welds, specify:

(a) The highest predicted end-of-life transition temperature corresponding to the 50 ft-lb value of the Charpy V-notch fracture energy for the "weak direction" of the material (WR direction in plates) and

(b) The minimum upper shelf energy value which will be acceptable for continued reactor operation toward the end-of-service life of the vessel.

4. Furnish the proposed heatup and cooldown curves which will be used to control the pressure and temperatures to which the ferritic material of the reactor coolant pressure boundary will be exposed during the first two years of operation and at the end of the service life.

REACTOR COOLANT SYSTEM

Reactor Vessel Material Surveillance Program

1. State if the reactor vessel material surveillance program will comply with ASTM E-185-70, particularly with respect to retention of representative test stock (Section 3.1.2 of the ASTM E-185-70), and documentation of chemical composition of the material (Section 3.1.3).

2. State the number of Charpy V-notch specimens oriented with respect to the weak direction (WR orientation in plates) of plates, forgings and weld materials, that will be included in the reactor vessel material surveillance program.

REACTOR COOLANT SYSTEM

Sensitized Stainless Steel

1. Describe the plans which were followed to avoid partial or local severe sensitization of austenitic stainless steel during heat treatments and welding operations for core structural load bearing members and component parts of the reactor coolant pressure boundary. Describe welding methods, heat input, and the quality controls that were employed in welding austenitic stainless steel components.

2. If nitrogen was added to stainless steel types 304 or 316 to enhance its strength (as permitted by ASME Code Case 1423 and USAS Case 71), provide justification that such material may not be susceptible to stress corrosion cracking under severely sensitized conditions.

3. The FSAR (Page 4D-1) refers to Westinghouse Topical Report WCAP-7477L to justify the use of sensitized stainless steel in components of the reactor coolant pressure boundary. Since Westinghouse personnel stated in a meeting on August 18, 1970, that this report was being revised, please submit revised copies of this report for evaluation of your application.

REACTOR COOLANT SYSTEM

Pump Flywheel Integrity

1. Indicate the nil-ductility transition (NDT) temperature of the flywheel material, as obtained from dropweight tests (DWT), the minimum acceptable Charpy V-notch (C_v) upper shelf energy level in the "weak" direction (WR orientation in plates), and the fracture toughness of the material at the normal operating temperature of the flywheel.

2. State if the calculated combined primary stresses in the flywheel, at the normal operating speed, included the stresses due to the interference fit of the wheel on the shaft, as well as the stresses due to centrifugal forces.

3. (a) State the highest anticipated overspeed of the flywheel and the basis for this assumption.

   (b) State the estimated maximum rotational speed that the flywheel attains in the event the reactor coolant piping breaks in either the suction or discharge of the pump. In addition, describe results of any studies directed towards: (1) determining the maximum speed the pump or motor can reach due to physical limitations (e.g. the speed at which the pump impeller seizes in the wear rings due to growth from centrifugal forces or the speed at which motor parts come loose and grind or blind to prevent further increase in speed); (2) establishing speed and torque for pipe break sizes; (3) devising means to disengage the motor from the pump in the event of pump overspeed; (4) verifying that pump fragments generated at maximum speed do not penetrate the pump casing and that any missiles in the blowdown jet do not penetrate containment; (5) establishing failure speeds for motor parts and whether they will penetrate the motor frame and if so with what energy; (6) defining a minimum rotor seizure time.

REACTOR COOLANT SYSTEM

Foreign Procurement of Reactor Coolant System Components

If any component within the reactor coolant boundary has been designed or fabricated outside of the United States provide the following information:

1. Identify the manufacturer and describe his qualifications, experience in the construction of nuclear power plant components, and experience in furnishing components for nuclear power plants in the U.S.

2. Describe the steps you are planning to take to assure that the quality levels achieved in the fabrication of foreign procured components are equivalent to those obtained from the U.S. manufacturers.

REACTOR COOLANT SYSTEM

Inservice Inspection Program

1. Describe the design and arrangement provisions for access to the reactor coolant pressure boundary as required by Sections IS-141 and IS-142 of Section XI of the ASME Boiler and Pressure Vessel Code - Inservice Inspection of Nuclear Reactor Coolant Systems. Indicate the design improvement applied to the reactor vessel, in particular, to facilitate inservice inspection.

2. Section XI of the ASME Boiler and Pressure Vessel Code recognizes the problems of examining radioactive areas where access by personnel will be impractical, and provisions are incorporated in the rules for the examination of such areas by remote means. Because the equipment used to perform these examinations may require development, the future examination of these areas is dependent upon providing the access and space requirements as dictated by the latest equipment development programs.

   (a) Describe the remote equipment planned or under development to perform the reactor vessel and nozzle inservice inspections for your plant.

   (b) Describe the system to record and compare the data from the baseline inspection with the data which will be obtained from subsequent inservice inspections.

(c) Describe the procedures to be followed in coordinating the development of the remote inservice inspection equipment with the access for inservice inspection provided in the plant design.

REACTOR COOLANT SYSTEM

Leakage Detection System

1. Describe the adequacy of the proposed leakage detection systems to differentiate between identified and unidentified leaks from components within the primary reactor containment and indicate which of these systems provide a means for locating the general area of a leak.

2. Discuss the adequacy of any system which depends on reactor coolant activity for detection of changes in leakage during the initial period of plant operation when the coolant activity may be low.

3. On page 6.7-13 of the FSAR the statement is made that a meaningful relationship between leak rate and crack size cannot be found. In view of the fact that Westinghouse in WCAP-7503 "Determination of Design Pipe Breaks for the Westinghouse Reactor Coolant System," dated October 1970, has made a detailed study of this relationship, review the basis for the proposed limit on leakage from unidentified sources and furnish the following information:

   (a) The length of a through-wall crack that would leak at the rate of the proposed limit, as a function of wall thickness.

   (b) The ratio of that length to the length of a critical through-wall crack, based on the application of the principles of fracture mechanics.

   (c) The mathematical model and data used in such analyses.

4. Describe the proposed tests to demonstrate sensitivities and operability of the leakage detection systems.

ENGINEERED SAFETY FEATURES

Inservice Inspection Program

Discuss the inservice inspection program for fluid systems other than the reactor coolant pressure boundary, including items to be inspected, accessibility requirements, and the frequency and types of inspection. The fluid systems to be considered are the engineered safety systems, reactor shutdown systems, cooling water systems, and the radioactive waste treatment systems.