50-247

APR 16 1970

Peter A. Morris, Director
Division of Reactor Licensing

INDIAN POINT NUCLEAR GENERATING UNIT 2 -- DOCKET NO. 50-247

The enclosed evaluation of the reactor internal structures, reactor coolant system, and Class I mechanical equipment of the Indian Point Nuclear Generating Unit No. 2, was prepared by the DRS Structural Engineering Branch.

One item remains outstanding, the use of static analyses to determine the seismic stresses in Class I piping. The mitigative procedure outlined in this report is similar to those employed for the Ginna and Robinson plants.

Original signed by
E. G. Case

Edson G. Case, Director
Division of Reactor Standards

DRS:SEB:JPK

Enclosure:
ACRS Report Sections for
Indian Point Unit 2

cc w/encl:
R. S. Boyd, DRL
R. Maccary, DRS
D. Muller, DRL
A. Dromerick, DRS
K. Kniel, DRL
J. Knight, DRS

bcc: E. G. Case

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INDIAN POINT NUCLEAR GENERATING UNIT NO. 2

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REACTOR INTERNALS

For normal design loads of mechanical, hydraulic, and thermal origin, including anticipated plant transients and the operational basis earthquake, the reactor internals are designed to the stress limit criteria of Article 4 of the ASME Boiler and Pressure Vessel Code Section III.

The applicant has submitted a summary of the analytical study of the behavior of the reactor internals under blowdown and seismic loadings (WCAP-7332-L). The results of this study indicate that the directly combined DBA and DBE loadings would result in one control rod guide tube exceeding the experimentally determined loss-of-function deflection limit of 1.6 inches (beam flexure mode). The deflections resulting from these combined loads on all other internal structures are less than the design allowable limits of approximately 62% of the loss-of-function limit for control rod guide tubes and 50% of the loss-of-function limit for all other components.

We find that the stress and deflection limits discussed above provide an adequate margin of safety.

REACTOR COOLANT SYSTEM

The reactor coolant system has been designed as a Class I (seismic) system to withstand the normal loads of mechanical, hydraulic, and thermal origin plus anticipated transients and the operational basis earthquake within the stress limits of the codes given below.

The steam generator primary side, pressurizer, and main coolant pump casings where applicable have been designed to the requirements of Section III of the ASME Boiler and Pressure Vessel Code, 1965 edition - Summer 1966 Addenda, as Class A vessels.

Seismic stresses in the reactor coolant piping, and all other Class I piping, were determined by a static analysis. Structural amplifications and resonant responses were not considered. The applicant states that the calculated stresses are relatively low, e.g., limited to yield for the design basis earthquake, and thus provide a margin for uncertainties. It is, however, impossible to assess the margin provided. We will require that the applicant provide a tabulation of the peak stresses in each Class I system and that he investigate the possibility of resonance responses. Where adequate margins are not provided to account for amplifications not considered in the static analysis, we shall require either more rigorous analyses or modification of support systems.

Inspection requirements for the reactor coolant piping include, in addition to the basic code requirements, the additional requirements given in Table 4.5-1 of the FSAR. These additional inspections include ultrasonic and dye penetrant inspection of forged fittings and pipe as well as radiographic and dye penetrant inspection of all pipe welds. This program upgrades the inspection of the Indian Point 2 reactor coolant piping to essentially that of a Class I (functional) system under the ANSI B31.7 Nuclear Power Piping Code.

We find the design and inspection criteria for the reactor coolant system to be acceptable.

VIBRATION CONTROL

The major core and core support components have been analyzed to provide assurance that they are not vulnerable to vibratory excitation. Vibration analyses for the core support barrel considered inlet flow impingement and turbulent flow. Natural frequency calculations were made to assure that there would be no deleterious response to known excitations such as pump blade passing and driven frequencies. Fuel bundle response to anticipated driving forces has been calculated and examined by tests in the Westinghouse Reactor Evaluation Channel.

The vibration monitoring system to be used for the preoperational test program on the Indian Point 2 plant will consist of mechanical gages to measure gross relative motion between the thermal shield and core barrel, strain gages on selected guide tubes and accelerometers on the upper core plate. None of this instrumentation is intended for use after the test period.

We find the preoperational test program for the Indian Point 2 plant to be acceptable.

OTHER CLASS I MECHANICAL EQUIPMENT

All welding procedures and operators used for pressure boundary welds on Class I pumps and valves were qualified to Section IX of the ASME Boiler and Pressure Vessel Code.