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Peter A. Morris, Director, Division of Reactor Licensing

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

Adequate responses to the enclosed request for additional information are required before we can complete our review of the subject application. This request, prepared by the DRS Mechanical Engineering Branch, concerns the reactor internal structures, reactor coolant pressure boundary, seismic design criteria and pipe whip criteria submitted in Volumes 1 through 5 of the FSAR.

This request reflects the comments of our consultant, Dr. N. M. Newmark, which were contained in his letter of March 24, 1971.

DKT # 50-286

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

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ADDITIONAL INFORMATION REQUEST
INDIAN POINT NUCLEAR GENERATING UNIT NO. 3
DOCKET NO. 50-286

A. Reactor Coolant Pressure Boundary

1. The list of transients that have been used in the design of components within the reactor coolant pressure boundary as specified in Table 4.1-8 of the FSAR appears to be incomplete. Identify all design transients and their number of cycles, such as control system or other system malfunction, component malfunctions, transients resulting from any single operator error, inservice hydrostatic tests, etc., which are specified in the ASME Code-required "Design Specifications" for the components of the reactor coolant pressure boundary. Categorize all transients or combination of transients with respect to the conditions identified as "normal", "upset", "emergency" or "faulted" as defined in the ASME Section III Nuclear Vessel Code.

2. Paragraph 116 of ASA B31.1-1955 edition, and paragraph 101.5.4 of USAS B31.1.0-1967 edition of the Code for Pressure Piping require that piping shall be supported to minimize vibration and that the designer is responsible by observation under startup or initial operating conditions to assure that vibration is within

acceptable levels. Submit a discussion of your pre-operational vibration test program which will be used to verify that the piping and piping restraints within the reactor coolant pressure boundary have been designed to withstand dynamic effects due to valve closures, pump trips, etc. Provide a list of the transient conditions and the associated actions (pump trips, valve actuations, etc.) that will be used in the vibration operational test program to verify the integrity of the system. Include those transients introduced in systems other than the reactor coolant pressure boundary that will result in significant vibration response of reactor coolant pressure boundary systems and components.

3. Specify whether the design criteria which have been used to examine the effects of pipe rupture have considered postulated pipe breaks to occur at any location within the reactor coolant pressure boundary, or at limited areas within the system. Provide confirmation that both longitudinal and circumferential type ruptures were evaluated and describe the basis for the design approach.

4. Indicate whether the basis for establishing the pressure relieving capacity for the reactor coolant pressure boundary is the loss of 100 percent of the heat sink when the thermal output of the reactor is at 100 percent of its rated power without any credit taken for operation of the safety valve or secondary steam system. If the capacity of the pressure relief system is not formulated upon this basis, submit

a copy of the Report on Overpressure Protection which has been prepared in accordance with the requirements of paragraph N-910.1 of the ASME Section III Nuclear Vessel Code.

5. Provide the criteria which will be applied in designing the principal reactor coolant system component supports (i.e., supports, restraints "snubbers," guides, etc., as applied to vessels, piping, pumps, and valves) including the materials to be used and the design codes or standards applicable to each type of support.

6. Reported service experiences of PWR steam generators have demonstrated that flow induced vibration and cavitation effects can cause tube thinning, and corrosion and erosion mechanisms both from primary and secondary side may contribute to further structural degradation of the tube integrity during the service life-time. The failure of a group of weakened tubes as a consequence of a design basis pipe break in the reactor coolant pressure boundary could impair the capability of emergency core cooling systems to perform their intended function. In order to evaluate the adequacy of design bases used to prevent such conditions from developing in the steam generator during service, the following additional information is required:

   (a) State the design conditions and transients which were specified in the design of the steam generator tubes, and the operating condition category selected (e.g., upset, emergency, or faulted)

which defines the allowable stress intensity limits to be used. Justify the basis for the selected operating condition category.

(b) Specify the margin of tube-wall thinning which could be tolerated without exceeding the allowable stress limits identified in (a) above, under the postulated condition of a design basis largest pipe break in the reactor coolant pressure boundary during reactor operation.

(c) Describe the inservice inspection which will be employed to examine the integrity of steam generator tubes as a means to detect tube-wall thinning beyond acceptable limits and whether excess material will intentionally be provided in the tube wall thickness to accommodate the estimated degradation of tubes during the service lifetime.

B. Reactor Internals

Section 3 of the FSAR designates Indian Point Unit 2 as the prototype plant from which vibration test data is applicable in evaluating the adequacy of the Indian Point Unit 3 core support structures to withstand flow induced vibration effects. However, the use of prototype results are valid only if the analytical methods and procedures employed for the prototype have been confirmed by an acceptable preoperational vibration test program.

Provide the test data and supporting analyses which form the basis for the Westinghouse vibration response predictions or if the validity of the methods employed cannot be demonstrated at this time, include in your response a statement of your intent to implement a preoperational test program which includes the measures given below:

1. A vibration test program should be developed and submitted for review prior to the performance of the scheduled preoperational functional tests. The program should include:

   a. a brief description of the vibration test program, including instrumentation type and location,

b. the expected numerical values of the response of the reactor internals and the anticipated forcing functions, under all flow modes of normal reactor operation,

c. the acceptance standards and the permissible deviations from these standards, and

d. the bases upon which the response, the forcing functions and the permissible deviations were established.

2. A vibration test program should be implemented during the preoperational functional testing program to measure the response[1/] of the reactor internals in order to determine the flow-induced forces and the related dynamic forcing functions for all significant modes of normal reactor operation. The data obtained by these measurements on reactor internals should be sufficient to verify that the steady state and cyclic stresses in the components, as determined by analyses, are within the acceptable design limits set forth in the design specifications and code requirements and that the results meet the acceptance criteria of the vibration test program.

3. The extent of measurements included in a vibration test program should be determined for each individual case on the basis of the design and configuration of those structural elements of the reactor internals

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1/ Frequency and magnitude of vibration (in terms of displacements, velocities and accelerations).

important to safety and the adequacy of theoretical and empirical data used in their design. The type of vibration test instrumentation used, the number of measurements taken, and the distribution of measuring devices within the reactor should be sufficient to determine all significant vibrational modes and characteristics of the reactor internals.

4. After the reactor internals have been subjected to the significant flow modes of normal reactor operation visual or surface examinations of reactor internals should be conducted to detect any evidence of excessive vibrations, and the presence of flaws or wear induced by unanticipated vibration. These examinations should be conducted at all major load-bearing structural elements whose failure could adversely affect structural integrity of the reactor internals, and at all areas of lateral, vertical and torsional restraints provided within the reactor vessel.

5. A summary of the results obtained from the vibration tests should be submitted to the Commission within three months after completion of the tests. The summary should include:

   a. a description of any differences from the specified vibration test program, instrumentation, reading anomalies and instrument failures.

b. a comparison between measured values [2/] and the values predicted for the design of the reactor internals.

c. an evaluation of measurements or observations that exceed acceptable limits or that were unanticipated, and the disposition of such deviations, and

d. a certification by the responsible engineer having authority over the conduct of the vibration test program that the test results documented are correct and in accordance with actual measurements.

[2/] In areas where measurements to determine forcing functions cannot be obtained practically by means of pressure transducers, the forcing functions may be calculated from the measured responses of other areas and the derived vibrational characteristics of the reactor internals. The values of the forcing functions computed from the response and reactor internal vibrational characteristics should be compared with the values of forcing functions used in the design.

C. Other Safety Related Systems and Components

1. Appendix A1 of the FSAR states that the categories of design conditions; namely, normal, upset, emergency and faulted are applicable to reactor coolant system components. Identify any other components of systems that are not part of the reactor coolant pressure boundary for which the design stress limits associated with emergency and faulted conditions will be applied. If emergency or faulted conditions are used for such cases, provide justification for applying such conditions, including the bases for the loading conditions and combinations, and the associated design stress limits which apply.

2. Describe the design and installation criteria for the mounting of the pressure-relieving devices (safety valves and relief valves) within the reactor coolant pressure boundary and on the main steam lines outside of containment. In particular, specify the design criteria used to take into account the combined loads resulting from full discharge (i.e., thrust, bending, torsion) imposed on valves on connected piping in the event the valves discharge concurrently and indicate the provisions made to accommodate these loads.

3. Provide a more detailed description of the measures that have been used to assure that the containment vessel and all essential

equipment within the containment, including components of the primary and secondary coolant systems, engineered safety features, and equipment supports, have been adequately protected against blowdown jet forces, and pipe whip. The description should include:

a. Pipe restraint design requirements to prevent plastic hinge formation.

b. The features provided to shield vital equipment from pipe whip.

c. The measures taken to physically separate piping and other components of redundant engineered safety features.

d. A description of any analyses performed to determine that the failure of small lines will not cause failure of the containment vessel under the most adverse design basis accident conditions.

D. Seismic Design Criteria and Analysis

1. Identify the method of seismic analysis (modal analysis response spectra, modal analysis time history, equivalent static load, etc.) or empirical (tests) analysis which has been employed in the design of the Category I structures, systems and components other than the containment structure.

2. Because various assumptions are made regarding structure material properties and soil structure interaction, calculated periods of vibration are not exact. Describe the measures taken to assure that the calculated responses of Class I (seismic) structures by the normal mode response spectrum method conservatively reflect the expected variations in the periods of vibration of the structures.

3. Describe the method employed to consider the torsional modes of vibration in the seismic analysis of the Class I building structures.

4. With respect to Class I (seismic) piping buried or otherwise located outside of the containment structure, describe the parametric study, referenced on Page A.3-9, that was employed to assure that allowable piping and structural stresses will not be exceeded due to differential movement at support points and the design provisions made to accommodate such motion at containment penetrations and at entry points into other structures.

5. With regard to the development of equipment seismic design criteria by the time history method:

   (a) Provide plots that show a comparison of the smoothed site response spectra and the spectra derived from the earthquake records for all damping values which were used in the time history system analyses. Identify the system period intervals at which the response spectra acceleration values were calculated and demonstrate that the period interval used is sufficient to produce accurate spectra that do not deviate below the smooth response spectra for the site.

   (b) Provide a description of the measures that were taken to consider the effects on the floor response spectra of expected variations in assumptions made for structural properties, dampings, and soil structure interactions (e.g., peak width and period coordinates).

6. With respect to the seismic design criteria for piping and equipment, the use of static coefficients alone (Section 5.2) may not adequately account for structural amplification and the response of flexible components. Provide the bases for the values chosen (pipe size and seismic coefficient) and justification for the use of static design analysis by demonstrating

that the results thus obtained are conservative when compared with the results derived by the application of an appropriate multi-degree-of-freedom system analysis.

7. Submit the basis for the methods used to determine the possible combined horizontal and vertical amplified response loadings for the seismic design of structures, systems and components including the following:

   (a) The possible combined horizontal and vertical amplified response loading for the seismic design of the building and floors.

   (b) The possible combined horizontal and vertical amplified response loading for the seismic design of equipment and components, including the effect of the seismic response of the building and floors.

   (c) The possible combined horizontal and vertical amplified response loading for the seismic design of piping and instrumentation, including the effect of the seismic response of the buildings, floors, supports, equipment, component, etc.

8. Provide the criteria used in formulating the mathematical model for seismic analysis of the reactor coolant system including the procedure for lumping masses.

9. Describe the evaluation performed to assure that seismic induced effects of Class II piping systems will not cause failure of Class I piping.

10. Describe the seismic design criteria employed to assure the adequacy of Class I mechanical components such as pumps, heat exchangers, and electrical equipment such as cable trays, battery racks, instrument racks and control consoles. Describe the measures taken for seismic restraint to meet these criteria, the analytical or testing methods employed to verify the adequacy of these restraints and the methods utilized to determine the seismic input to these components.

11. Describe the criteria employed to determine the field location of seismic supports and restraints for Class I (seismic design) piping, piping system components, and equipment, including placement of snubbers and dampers. Describe the procedures followed to assure that the field location and characteristics of these supports and restraining devices are consistent with the assumptions made in the dynamic analyses of the system.

12. With respect to seismic instrumentation, submit a statement of your intent to implement a program such as described in AEC Safety Guide 12, Instrumentation for Earthquakes (April 9, 1971). Submit the basis and justification for elements of the proposed program which differ substantially from Safety Guide 12.

13. Topical Report WCAP-7397-L, "Seismic Testing of Electrical and Control Equipment," is referenced in the FSAR; however, in this report, vertical and horizontal excitations were considered separately. Discuss the adequacy of this equipment when subjected to combined response.

14. With respect to analyses of structures, systems, and components by the normal mode methods, provide the criteria which were used to compute shears, moments, stresses, deflections and/or accelerations for each seismic-excited mode as well as for the combined total response, including the criteria for combining closely spaced modal frequencies.

E. Seismic Quality Assurance

1. In order to assure that the seismic design bases for structures, systems, and components of this plant have been properly translated into the required specifications, drawings, and procedures that will result in acceptable designs of structures, systems, and components to withstand seismic and other concurrent loads, provide the following information:

(a) Identify the design organizations involved in the seismic design of all safety-related items of the plant, and describe their responsibilities and the documented procedures followed to assure that these responsibilities were met. Identify the organization assigned overall responsibility for the adequacy of seismic design.

(b) In regard to the interchange of design information among the involved design organizations, revisions thereto, and coordination of all aspects of the seismic design, describe the documentation procedures employed to assure that these interchanges and coordination among design organizations have been followed.

(c) Describe the design control measures instituted to verify the adequacy of the seismic design and identify the responsible design groups or organizations who perform this function.

(d) Describe the requirements included in the purchase specifications for safety-related equipment to assure adequate design and functional integrity under the seismic design conditions. Describe the provisions that are included in the purchase specification to permit the purchaser to verify that these requirements are satisfied.