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P. A. Morris, Director, DRL

THREE MILE ISLAND NUCLEAR STATION, UNIT NO. 1, DOCKET NO. 50-289

The FSAR submitted by the subject applicant has been reviewed and evaluated by the DRS Mechanical Engineering Branch. A final evaluation of the material within the scope of review of this Branch is enclosed.

Original Signed By
E. G. Case

Edson G. Case, Director
Division of Reactor Standards

Enclosure:
Final Evaluation - Mechanical
for Three Mile Island, Unit 1

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FINAL EVALUATION-MECHANICAL
THREE MILE ISLAND NUCLEAR STATION UNIT NO. 1
DOCKET NO. 50-289

Reactor Coolant System

The reactor coolant system has been designed to withstand normal design loads including anticipated plant transients and the Operational Basis Earthquake within the acceptable stress limits of the appropriate codes given below.

The steam generator, pressurizer, and reactor coolant pump casings have been designed to Class A requirements of Section III of the ASME Boiler and Pressure Vessel Code, 1965 edition, including the Summer 1967 Addenda. Safety and relief valves are in accordance with the requirements of Article 9 of the above edition and addenda of Section III.

The design, fabrication, inspection and testing of the reactor coolant piping including the pressurizer surge line and spray line is in accordance with the USAS B31.7, Code for Pressure Piping, Nuclear Power Piping, dated February, 1968, including the June, 1968 Errata.

Nondestructive examination requirements for reactor coolant system pumps and valves are given in Table 4-12 of the FSAR. These examinations include radiography of castings, ultrasonic testing of forgings, dye penetrant inspection of pump and valve body surfaces, and radiography of circumferential weldments. This program upgrades the nondestructive testing of pumps and valves within the reactor coolant pressure boundary to essentially that required by the ASME Code for Pumps and Valves for Nuclear Power.

The design, fabrication and inspection criteria discussed above are consistent with those accepted for all recently reviewed plants of this type and we find them acceptable.

Components, of the reactor coolant system (RCS) have also been designed to withstand the loads calculated to result from the Design Basis Earthquake, the Design Basis Accident, and the combination of these postulated events. Strain limits for the RCS components under these combined loads correspond to an elastically calculated stress limit of not greater than 2/3 of the ultimate tensile strength. We consider these design limits to be acceptable.

Reactor Internals

For normal design loads including the operational basis earthquake and anticipated transients, the reactor internals have been designed to operate within the acceptable allowable stress intensity limits of Article 4, Section III of the ASME Boiler and Pressure Vessel Code.

All internals components have been designed to withstand the loads calculated to result from the Design Basis Earthquake, the Design Basis Accident and the combination of these postulated events. Strain limits for the internals under these combined loads will correspond to an elastically calculated stress limit of not greater than 2/3 of the ultimate tensile strength. Allowable deflection limits are generally within 50% of loss-of-function deformation limits. We consider these design limits to be acceptable.

Topical Report BAW-10008, Parts 1 and 2, is referenced in the FSAR as outlining the methods of analysis employed for the internals and fuel assemblies under loss-of-coolant and design basis earthquake loadings for skirt supported reactor vessels. We have, with the aid of our consultant, reviewed the methods of analyses presented in this report and find them acceptable.

REACTOR INTERNALS - VIBRATION CONTROL

Flow induced vibration has been considered in the design of the Three Mile Island reactor internal structures. Verification of the calculated vibration responses will be accomplished by comparing vibration response measurements made during the Three Mile Island preoperational testing with similar measurements made at the designated prototype plant for the Babcock & Wilcox Company product line, Oconee I. A portion of the Oconee I instrumentation will be duplicated in design and location at Three Mile Island to allow direct comparison of data.

We find the proposed preoperational test program acceptable provided that the Oconee I tests are successfully completed and that data demonstrating the validity of the methods utilized to predict vibration responses for the Babcock & Wilcox product line are available prior to the completion of the Three Mile Island test program.

The results from the proposed Three Mile Island vibration test program should be the subject of a report, submitted to the Commission within 3 months after completion of the tests. This report should include:

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

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b. the expected and measured 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.

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Seismic Input

The seismic design response spectra submitted produce a magnification factor greater than 3.5 in the period range appropriate for the response of structures, systems, and components. Proposed structure and equipment damping factors are in accordance with those recommended by N. Newmark. The response spectra are derived from the most critical combination of the normalized Golden Gate and El Centro (1940) earthquake records. These records were also used as input to confirm the structural integrity of structures, systems, and components. We conclude that the seismic input criteria proposed by the applicant provide an acceptable basis for seismic design. (The above assumes that the applicant will adequately document verbal agreements with the staff.)

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Seismic System Dynamic Analyses

Modal response spectrum multi-degree-of-freedom and normal mode-time history methods are used for the analysis of all Category I structures, systems, and components. Governing response parameters have been combined by the square root of the sum of the squares to obtain the modal maximums when the modal response spectrum method is used. The absolute sum of responses is used for closely spaced frequencies. Floor spectra inputs used for design and test verification of structures, systems and components were generated by semi-empirical methods and confirmed by the normal mode-time history method. A vertical seismic-system dynamic analysis was employed to account for significant vertical amplifications for the seismic design of structures, systems, and components. Constant vertical load factors were employed only where analysis showed sufficient vertical rigidity to preclude significant vertical amplifications in the seismic system being analyzed. We and our seismic consultants conclude that the seismic-system dynamic methods and procedures proposed by the applicant provide an acceptable basis for the seismic design. (The above assumes that the applicant will adequately document verbal agreements with the staff.)

Other Class I (Seismic) Mechanical Equipment

All welding procedures and operators concerned with the fabrication of pumps and valves have been qualified to Section IX of the ASME Boiler and Pressure Vessel Code.

Hydrostatic tests of pump casings, valve bodies and valve seats were in accordance with the ANSI B 16.5 and MSS SP-61 code and standard and have been witnessed by the applicant's representative.

The specified inspection program for pumps and valves requires independent review of the physical and chemical test data for pressure boundary materials as well as independent review of nondestructive examinations of valve bodies, valve bonnets, and pump casings.

The above requirements result in a fabrication and inspection program which contains the essential elements of the ASME Code for Nuclear Pumps and Valves. We find these requirements acceptable.

All equipment for the engineered safety features has been designed to withstand the Design Basis Earthquake without loss of function. Equipment purchase specifications include seismic design requirements which were based on, or checked against, the outcome of the structural dynamic analysis and included, where necessary, the dynamic feedback of flexible equipment. We find this approach acceptable. (The above assumes that the applicant will adequately document verbal agreements with the staff.)