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Richard C. DeYoung, Assistant Director
for Pressurized Water Reactors
Directorate of Licensing

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

Plant Name: Three Mile Island Nuclear Station Unit No. 1
Licensing Stage: OL
Docket Number: 50-289
Responsible Branch and Project Leader: PWR-4, H. Faulkner
Requested Completion Date: 8/18/72
Applicants response date necessary for completion of next action planned on project: N/A
Description of Response: N/A
Review Status: Complete

The final evaluation for the subject plant which was prepared by the Mechanical Engineering Branch, Directorate of Licensing, dated May 11, 1971, has been revised to reflect significant changes submitted in Amendments through No. 27. Our updated evaluation as requested in your memo of June 22, 1972 is enclosed.

DKT #50-289

Original Signed by
R. R. Maccary

R. R. Maccary, Assistant Director
for Engineering
Directorate of Licensing

Enclosure:
Final Evaluation - Mechanical for
Three Mile Island

cc w/encl:
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Docket File (50-289)
L, Reading File
L:MEB File

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3.6 Criteria for Protection Against Dynamic Effects Associated with a Loss-of-Coolant Accident

In the analysis of the reactor coolant loop, the applicant has applied an equivalent static approach to the criteria for protection against a pipe rupture considering both impact and jet loadings on the restraint systems designed to prohibit damage to containment or other safety related systems. Circumferential and longitudinal breaks were postulated at locations where they would impose the most severe loading conditions on piping components and supports. Factors greater than 2 were applied to the normal thrust load at postulated pipe breaks as a result of conservative assumptions of maximum theoretical momentum change and ideal flow at the postulated break locations. Restraint systems were designed to withstand the combined loads calculated to result from postulated pipe rupture and from the maximum hypothetical earthquake; resulting stresses were limited to the yield stress of the restraint materials. Additional protection for vital systems is provided by the secondary shield walls surrounding each steam generator and its pair of reactor coolant pumps and from the routing of safety related piping to attain separation of systems within the shield area.

Restraint systems and protection criteria for main steam and feedwater piping were designed and developed on similar bases. Under the combined loads calculated to result from impact and jet thrust some plastic action will result but this action is well within the energy absorbing capabilities of the structural steel members comprising these systems.

We find this approach for protection against pipe rupture to be acceptable.

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3.6.1 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.

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3.6.2 Seismic System and Subsystem 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.

3.6.3 Criteria for Seismic Instrumentation Program

The type, number, lcoation and utilization of strong motion accelerographs to record seismic events and to provide data on the frequency, amplitude and phase relationship of the seismic response of the containment structure corresponds to the recommendations of Safety Guide 12.

Supporting instrumentation will be installed on Category I structures, systems, and components in order to provide data for the verification of the seismic responses determined analytically for such Category I items.

A plan for the utilization of the acquired seismic data will be developed before start-up.

We conclude that the Seismic Instrumentation Program proposed by the applicant is acceptable.

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4.2.3.1 Reactor Internals - Design

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 calcualted 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.

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4.2.3.2 Dynamic System Seismic and LOCA Analysis

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.

4.2.3.3 Reactor Internal Structures - Vibration Control

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 comparative 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 in accordance with AEC Safety Guide 20.

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5.2.1 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 examination of pump and valve body surfaces, and radiography of circumferential welds. This program upgrades the nondestructive examination 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.

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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.

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