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Peter A. Morris, Director  
Division of Reactor Licensing

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

Adequate responses to the enclosed request for additional information are required before we can complete our review of the subject application. These requests, prepared by the DRS Structural Engineering Branch, concern the reactor internal structures, reactor coolant pressure boundary, and Class I mechanical equipment material presented in Sections 1, 3, 4, 5, 6, 9, 11, and 14 of the application.

This review of topical reports BAW-10006, BAW-10008, and GAI No. 1729, which are pertinent to this application, is underway by the DRS staff and DRS consultants. Additional requests for information will follow the completion of these reviews.

Review of the leak detection system limits and the inservice inspection program will be completed when proposed Technical Specifications are submitted by the applicant.

Edson G. Case, Director  
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Request for Additional Information  
for Three Mile Island Unit No. 1

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REQUEST FOR ADDITIONAL INFORMATION

THREE MILE ISLAND STATION - UNIT NO. 1

DOCKET NO. 50-289

A. Reactor Coolant Pressure Boundary

1. Identify, for all components within the reactor coolant pressure boundary, those stainless steel parts of the pressure-containing membrane and load-bearing stainless steel members which are vital to the structural integrity of the reactor vessel and core that may have become furnace-sensitized by the fabrication sequence or partially-sensitized due to a slow cooldown rate from the solution heat treatment temperature. Such sensitized steels are recognized as susceptible to stress corrosion cracking.

Also identify if any stainless steel type 304 or 316 has been used for which the addition of nitrogen to enhance the strength of the material has been specified. Under certain conditions, such steels may also be susceptible to stress corrosion cracking.

Describe the plans which will be followed in the light of the susceptibility of these stainless steel components or parts to stress corrosion cracking.

2. If the process of electroslag welding was used in the fabrication of components within the reactor coolant pressure boundary, identify such components and describe the respective process specifications,

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control of variables, and quality control procedures which were applied in production to achieve the physical properties in the welds and heat affected zones comparable to those obtained in the weld procedure qualification tests.

B. Reactor Internals

Identify the reactor from which vibration test data will be applicable to evaluate the adequacy of the Three Mile Island Unit 1 core support structure to withstand vibrations, and specify the vibration surveillance program which will be applied to the Three Mile Island Unit 1 design to demonstrate comparable performance.

C. Other Safety-Related Systems and Components

1. With respect to seismic ground motion and differential relative settlement, describe the design criteria which will be employed for critical piping buried or otherwise located outside of the containment structure and for the situations where this piping enters the various structures.
2. The seismic design of a nuclear power plant represents one of the more complex problems involving design interfaces between design organizations. To provide us with information to assess whether the seismic design bases are correctly translated into the required specifications, drawings, procedures, and instructions so that the necessary structures, systems, and components can withstand seismic loads combined with the other appropriate concurrent loads, furnish the following information:.

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- a. Describe the design organizations that are involved in the seismic design of all structures, systems, and components of the plant that are related to safety.
- b. Describe the responsibilities of the involved design organizations in connection with the seismic design and the extent to which these responsibilities have been promulgated to the organizations in writing. Identify the design organization that has been assigned overall responsibility for the adequacy of the seismic design.
- c. Describe the documented procedures that have been or will be promulgated to provide for the interchange of needed design information and changes thereto and the coordination of the various facets of the seismic design among the involved design organizations.
- d. Describe the manner by which you assure that the design procedures described in c. above have been or are being followed.
- e. Describe the design control measures that have been or will be instituted to verify or check the adequacy of the seismic design and by whom they will be performed. Describe the design procedures that have been or will be promulgated to provide for these measures.
- f. Describe the requirements that are or will be included in the purchase specifications for safety-related equipment to assure

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that this equipment is adequately designed to withstand and can function under the seismic design conditions. Describe the provisions that have been included in the purchase specification to permit the purchaser to verify that these requirements are satisfied.

3. Identify the Category I (Class I in the FSAR) piping systems and equipment, other than the primary coolant loops, which have been designed to withstand the loads that would result from the combined Design Basis Earthquake (Maximum Hypothetical in the FSAR) and pipe rupture loads of the Design Basis Accident. Identify the loading combinations and the stress and deformation limits applicable to these loading combinations which were included in the design criteria for these systems.

D. Leak Detection

Provide the sensitivities and response times of the leak detection systems described in Section 4.2.3.8 of the FSAR, including an estimate of time to measure the smallest detectable unidentifiable coolant leak from the reactor coolant system.

E. Reactor Vessel Material Surveillance Program

1. The FSAR states that the material surveillance program will be in accordance with Topical Report BAW 10006, Reactor Vessel Material Surveillance Program. For Three Mile Island Unit 1, Table 6 of the report, indicates only two scheduled capsule withdrawals are

planned. A planned withdrawal schedule of a minimum of four capsules, is considered essential for the plant, and the material surveillance should be independent of all other Babcock and Wilcox nuclear systems. Unless such a withdrawal schedule, and material surveillance program are adopted for the Three Mile Island Unit 1, additional justification is required for consideration of the proposed program.

2. If the estimated inservice transition temperature shift of the reactor vessel beltline material is based on data which are related to control of residual elements (specified in ASTM-E-185-70 Section 3.1.3) including control of copper and vanadium, specify the results of the chemical tests of these vessel materials including the residual element content in weight percent to the nearest 0.01%..

F. Missile Protection

With respect to the primary pump motor flywheels discussed in Section 4.2.2.6 of the FSAR, furnish the proposed inservice inspection program which will be conducted to detect flaws which may develop in service.

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