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P. A. Morris, Director
Division of Reactor Licensing

QUESTIONS RELATING TO INSTRUMENTATION AND EMERGENCY POWER; THREE MILE ISLAND UNIT #1; DOCKET #50-289

Please include the attached questions among those in preparation for transmittal to the applicant.

Original signed by
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ESB-67
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Enclosure:
Questions - Instrumentation
& Emergency Power

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THREE MILE ISLAND, UNIT #1

A. Instrumentation

1. In regard to the protection systems which actuate reactor trip and engineered safety feature action, the following information should be provided:

   a. A list of those systems designed and built by Babcock & Wilcox that are identical to those of the Oconee Nuclear Station (as documented in the Oconee FSAR) and a discussion of any design differences;

   b. A list of those systems and their suppliers that are designed and/or built by suppliers other than Babcock & Wilcox; and

   c. Identification of those features of the design which differ from the criteria of IEEE 279 and the Commission's proposed General Design Criteria and an explanation of the reasons for any differences.

2. In regard to the Babcock & Wilcox designed control systems, the following information should be provided:

   a. Identification of the major plant control systems (e.g., primary temperature control, primary water level control, steam generator water level control) which are identical to those in the Oconee Nuclear Station; and

b. A list and a discussion of the design differences in those systems not identical to those used in the Oconee Nuclear Station. This discussion should include an evaluation of the safety significance of each design change.

3. State the seismic design criteria for the reactor protection system, engineered safety feature circuits, and the emergency power system including the station batteries. The criteria should cover (1) the capability to initiate a protective action during maximum peak acceleration, and (2) the capability of the engineered safety feature circuits to withstand seismic disturbances during post-accident operation. Describe the qualification testing requirements which will be used to assure that the criteria are satisfied and the means by which these requirements will be imposed on equipment suppliers.

4. Describe the quality assurance procedures which apply to the equipment in the reactor protection system, engineered safety feature circuits, and the emergency electric power system. This description should include: (a) quality assurance procedures used during equipment fabrication, shipment, field storage, field installation, and system component checkout; and (b) records pertaining to (a) above.

5. To supplement the information presented on pages 8-6.7 of the FSAR, submit the criteria and their bases which establish the minimum requirements for preserving the independence of redundant reactor protection systems, engineered safety feature systems and Class IE* Electrical Systems through physical arrangement and separation and assure minimum availability during any *design basis event. The submittal should include a discussion of the administrative responsibility and control to be provided to assure compliance with these criteria during the design and installation of these systems. The criteria and bases for the installation of electrical cable for these systems should, as a minimum, address:

   a. Cable derating.

   b. Cable routing in containment, penetration areas, cable spreading rooms, control rooms and other congested or hostile areas.

   c. Sharing of cable trays with non-safety related cables or with cables of the same system or other systems.

   d. Fire detection and protection in the areas where these cables are installed.

   e. Cable and cable tray markings.

   f. Spacing of wiring and components in control boards, panels, and relay racks.

   g. Circuit overload protection.

*Class IE electrical systems and design basis events are defined in the Proposed IEEE Criteria for Class IE Electrical Systems for Nuclear Power Generating Stations (IEEE-308).

6.** State the design criteria for reactor protection system and engineered safety feature related electrical and mechanical equipment located in the primary containment or elsewhere in the plant which take into account the potential effects of radiation on these components due to either normal operation or accident conditions (superimposed on long-term normal operation). Describe the analysis and testing performed to verify compliance with these design criteria.

7.** Identify all safety related equipment and components (e.g., motors, cables, filters, pump seals) located in the primary containment which are required to be operable during and subsequent to a loss of coolant or a steamline break accident. Describe the qualifications tests which have been or will be performed on each of these items to insure their availability in a combined high temperature, pressure, and humidity environment.

8. State the criteria which have been established to assure that loss of the air conditioning and/or ventilation system will not adversely affect operability of safety related control and electrical equipment located in the control room and other

**These questions relate to the engineered safety feature chapter of the FSAR and should be forwarded to the applicant with other questions concerning that chapter.

equipment rooms. Describe the analysis performed to identify the worst case environment (e.g., temperature, humidity). State the limiting condition with regard to temperature that would require reactor shutdown, and how this was determined. Describe any testing (factory and/or onsite) which has been or will be performed to confirm satisfactory operability of control and electrical equipment under extreme environmental conditions.

9. Describe how reactor protection system and engineered safety equipment will be physically identified as safety related equipment in the plant.

10. Describe the method for periodic testing of engineered safety feature actuation to show it to be consistent with IEEE 279. We interpret IEEE 279 to require for engineered safety feature actuation the same high degree of on-line testability required for the reactor trip system.

11. Provide a description of the instrumentation systems included in your design for remote monitoring of post-accident conditions within the primary containment. Provide an analysis to show that these systems are adequate over the full spectrum of postulated accidents.

12. Describe what information is available to the operator to identify

all RPS and ESF channels that are in test or maintenance. Describe the indication available, down to the channel level, to identify which instruments initiate a protective action. These descriptions should be in sufficient detail to permit a determination of the system's compliance with Sections 4.13 and 4.19 of IEEE 279.

13. Describe any rod speed limiting features which prevent withdrawal rates in excess of 30 inches/min.

14. Do the circuits which prevent improper sequencing of the rods conform to the provisions of IEEE 279?

15. Do the circuits which automatically terminate dilution of the primary coolant conform to the provisions of IEEE 279?

16. Identify the electrical and pneumatic components (valves, pumps, etc.) of the auxiliary cooling systems which should be considered as portions of the engineered safety features. Do your criteria with respect to the design of the associated instrumentation and power systems for operation of these components conflict in any way with IEEE 279 or the General Design Criteria?

B. Emergency Power

1. Provide the design criteria and information concerning the onsite electrical power systems as follows:

a. The percentage of the continuous rating of each diesel generator that the engineered safety features electrical loads will require. The continuous rating is defined as that continuous load which will permit supplier guaranteed operation at a 95% availability with an annual maintenance period.

b. The 2000-hour and the 30-minute diesel generator overload ratings.

2. Provide your basis for sizing the station batteries to operate for two hours (without benefit of any station power).

3. Discuss the analyses to be performed to show that neither loss of a unit of this station nor the loss of the largest generating unit on the grid will negate the ability to provide offsite power to this station.

4. Provide an analysis to show that no single failure within any d-c system (e.g., station battery) adversely affects the shedding of loads and/or the opening of supply breakers such that adequate diesel generator operation is prevented.

5. Are the battery rooms separately ventilated?

6. With respect to both the a-c and d-c emergency power systems, describe the electrical interlocks which prevent improper

operation (e.g., onto a fault) of the manual cross-connections between redundant buses.

7. Identify any heat tracing circuits vital to the operation of the engineered safety features (e.g., any circuits to ensure that boron remains in solution). If so, describe and justify the design of the circuits and their power sources.

8. Submit a one-line diagram, similar in format to Figure 8.3, showing the assignment of engineered safety feature equipment to the emergency buses 1D and 1E.

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