Docket 50-289

UNITED STATES
ATOMIC ENERGY COMMISSION
WASHINGTON, D.C. 20545

June 19, 1970

R. C. DeYoung, Assistant Director, PWRs
Division of Reactor Licensing
THRU: Charles G. Long, Chief, PWR Project Branch 2, DRL

HYDROLOGY MEETING ON THREE MILE ISLAND UNIT 1, POL (DOCKET 50-289)

A meeting was held with representatives of Gilbert Associates, the A/E for Met-Ed on Three Mile Island Unit 1. The purpose of the meeting was to discuss the PMF calculations for the site. Attending were R. H. MacLemore and Joel Caves for Gilbert and D. Ross, W. Nischan, and D. Nunn for DRL. MacLemore demonstrated a plot of flood stage along the river for various discharges up to 1.75 million cubic feet per second. He also showed a map where flood contours had been calculated. He discussed the procedures used to calculate level versus discharge with the DRL hydrologist, D. Nunn. As a result of our discussions, we notified Gilbert that we contemplated four broad question areas to be included in our next formal list of questions, along the following lines:

1. We said that the applicant should provide a discharge hydrograph for the PMF, both regulated and unregulated.

2. We asked for the backwater analyses for the 1936 and 1964 floods and for the calculated discharges of 1.1 million and the PMF.

3. We requested a discussion of the procedures for calculation of backwater and the significance of overbank flow. This should include, we said, a table showing the Manning-n coefficients, and the discharge values and elevations at each cross section. We also want the comparison of measured versus computed elevations for the observed floods incorporating the known high water marks, and a map showing the location of the river cross sections used in the computer program. Finally, we want flood level contours.

4. We anticipate a need for a discussion by the applicant of the operating procedures in advance of and during extreme flood events. This should include the information that will be available to the operating staff and the decision levels that the staff must face in terms of river stage or precipitation.

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These comments were given to the Gilbert representatives. We cautioned them to await a formal transmission of these requests before submitting answers. We also notified them of our intention to visit the site next month.

Denwood F. Ross
PWR Project Branch 2
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THREE MILE ISLAND NUCLEAR STATION, UNIT 1

DOCKET NO. 50-289

Fracture Toughness Criteria - Reactor Coolant Pressure Boundary

1. For all pressure-retaining ferritic components of the reactor coolant pressure boundary**, for which the lowest pressurization temperature* is lower than 250°F, provide fracture toughness data (Charpy V-notch fracture energy curves and Drop-Weight Test NDT temperature) for plates, forgings, piping and weld material.

2. For reactor vessel beltline materials, including welds, provide:
   (a) The highest predicted end-of-life transition temperature corresponding to the 50 ft-lb value of the Charpy V-notch fracture energy for weak direction, and
   (b) The minimum upper shelf energy value for weak direction, which will be acceptable for continued operation toward the end of service life of the vessel.

3. Furnish the proposed heatup and cooldown curves which will control the pressure and temperatures to which the material of the reactor coolant pressure boundary will be exposed toward the end of its service life.

*Lowest pressurization temperature of a component is the lowest temperature at which the pressure within the component exceeds 25 percent of the system normal operating pressure, or at which the rate of temperature change in the component material exceeds 50°F/hr., under normal operation, system hydrostatic tests or anticipated transients.

**See attached definition.

AEC Definition - Reactor Coolant Pressure Boundary.

(a) Reactor coolant pressure boundary[1/] means all those pressure containing components of boiling and pressurized water-cooled nuclear power reactors, such as pressure vessels, piping, pumps, and valves, which are:

(1) part of the reactor coolant system or

(2) connected to reactor coolant system, up to and including any and all of the following:

(a) the outermost containment isolation valve in system piping which penetrates primary reactor containment,

(b) the second of two valves normally closed during normal reactor operation in system piping which does not penetrate primary reactor containment,

(c) the reactor coolant system safety and relief valves.

For nuclear power reactors of the direct cycle boiling water type, the reactor coolant system extends to and includes the outermost containment isolation valve in the main steam and feedwater piping.

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1/ Components which are connected to the reactor coolant system and are part of the reactor coolant pressure boundary may be excluded from these requirements provided:

(a) For postulated failure of the component during normal reactor operation, the reactor can be shut down and cooled down in an orderly manner assuming makeup is provided by the reactor coolant makeup system only.

(b) The component is or can be isolated from the reactor coolant system by two valves (both closed, both open, or one closed and the other open). Each open valve must be capable of automatic actuation and its closure time must be such that for postulated failure of the component during normal reactor operation and assuming the other valve is open, the reactor can be shut down and cooled down in an orderly manner assuming makeup is provided by the reactor coolant makeup system only.

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