UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 17 TO FACILITY OPERATING LICENSE NO. DPR-50

METROPOLITAN EDISON COMPANY
JERSEY CENTRAL POWER AND LIGHT COMPANY
PENNSYLVANIA ELECTRIC COMPANY

THREE MILE ISLAND NUCLEAR STATION, UNIT 1

DOCKET NO. 50-289

Introduction

By letter dated January 13, 1976, as supplemented by letters dated February 11, 1976 and April 2, 1976, Metropolitan Edison Company (MetEd) requested a change in the Technical Specifications of License No. DPR-50 for the Three Mile Island Nuclear Station, Unit No. 1 (TMI-1). The proposed amendment is to permit operation of TMI-1 as reloaded for Cycle 2 operation. The proposed amendment also incorporates the change, requested by letter dated August 8, 1975, which was submitted pursuant to Section 50.46 and Appendix K of 10 CFR Part 50 and the Commission's Order for Modification of License dated December 27, 1974.

Discussion

The TMI-1 reactor core consists of 177 fuel assemblies, each with a 15x15 array of fuel rods. The cycle 2 reload will involve the removal of all batch 1 fuel assemblies, the relocation of once-burned batch 2 and 3 fuel assemblies and the introduction of 56 fresh batch 4 fuel assemblies. The batch 4 assemblies will occupy primarily the periphery of the core and 8 locations interior to the core.

MetEd has proposed changes to the present Technical Specifications as a result of: changes and relocation of fuel assemblies as described above; use of the B&W-2 CHF correlation with a 95/95 confidence level and extended pressure application to 1750 psi; use of a reactor coolant flow rate equal to 106.5% of cycle 1 design flow; and Emergency Core Cooling System Final Acceptance Criteria (FAC). MetEd has provided technical information which includes a general description of the reload core, detailed mechanical design data on the reload fuel, nuclear and thermal-hydraulic design data, accident and transient analyses, fuel rod bow analyses and the loss of coolant accident (LOCA) analysis in support of the reload.

Evaluation

Fuel and Mechanical Design

Creep collapse calculations were performed by MetEd for three-cycle assembly power histories for TMI-1 using the Babcock & Wilcox (B&W) computer code, CROV, which we approved in our Generic Review of B&W Cladding Creep Collapse Analysis Topical Report, BAW-10084, issued on August 9, 1974. The calculations included conservative treatment of effects of fission gas (no credit taken), cladding thickness (lower tolerance limit), initial cladding ovality (upper tolerance limit), and cladding temperature (assembly outlet temperature) on collapse time. The most limiting assembly was found to have a collapse time which is greater than the maximum projected cycle 2 life of 19,000 hours and is therefore acceptable.

Fuel thermal analysis calculations that account for the effects of fuel densification were performed with the approved version of the B&W analytical model TAFY as described in B&W Topical Report BAW-10044 of May 1972. Fuel densification results in increases in stored energy, increases in linear thermal output and increases the probability of local power spikes from axial gaps. During cycle 2 operation, the highest relative assembly power levels will occur in batch 3 fuel. Fuel temperature analysis for batches 2 and 3 fuel is documented in the TMI-1 Fuel Densification Report, BAW-1389 of June 1973. Although the batch 4 fuel has a higher linear heat generation rate (20.15 kw/ft vs 19.6 kw/ft) due to a reduced active fuel length, the higher initial density results in a lower maximum predicted centerline temperature. In view of the above, we find the MetEd's fuel thermal analysis acceptable.

The batch 4 fuel assemblies are not new in concept and they do not utilize different component materials. Therefore, on the bases of the analysis presented in the reports referenced, we conclude for TMI-1 cycle 2 that:

(a) The fuel rod mechanical design provides acceptable safety margins for normal operation, and

(b) The effects of fuel densification have been adequately accounted for in the fuel design.

Thermal-Hydraulic Analysis

The thermal hydraulic calculations for the cycle 2 reload core were made using previously approved models and methods. There were no differences due to mechanical differences since the new fuel elements are mechanically similar and flow resistances are lower than the previously analyzed cycle 1 core.

During cycle 1 the reactor coolant flow was measured for TMI-1. With the reactor operation at 100% of full power on February 16, 1976, calorimetric and flow measurements were made and averaged. A description of the flow test and an error analysis were reported by letter dated April 8, 1976. The results of the flow test indicate a nominal flow of 109.3% of the design flow rate. The error analysis, based on measurement errors, showed a 2σ core flow error of 1.8%. Thus, the maximum usable flow rate for calculations would be 107.5% of the design flow. To provide additional conservatism in their calculations, MetEd has used a flow rate of 106.5% of design. We find that the flow test and analysis performed are acceptable and agree that this is a conservative flow rate.

In their letter of April 8, 1976, MetEd has committed to verify the flow rate for TMI-1 within three months following refueling. Thereafter, the flow rate will be verified every six months, plus or minus thirty days. All verifications will be done by the heat balance technique described in their April 8, 1976 letter.

The overpower trip, as used in the analyses of accidents and transients for cycle 2 operation, has still retained the 4.6% flow penalty due to vent valves used in the FSAR analyses. As discussed later in this evaluation, this is an additional conservatism and therefore, additional margin exists beyond that indicated in the accident analyses.

The flux/flow trip setpoint previously determined for cycle 1 was re-evaluated for the cycle 2 core. The procedure was revised to use the measured flow instead of the design flow rate. Like the previously mentioned overpower trip and accident analysis, the flux/flow trip setpoint includes the penalty for a stuck open vent valve. Thus, for the pump coast down analysis the 4.6% penalty due to vent valves has been retained. The coast down analysis shows that with a flux/flow trip setpoint of 1.08, the minimum DNBR does not go below 1.30.

On March 10, 1976, we sent a letter to MetEd stating that B&W report, "B&W Operating Experience of Reactor Internals Vent Valves" had been reviewed and that sufficient evidence had been presented to assure that the vent valves will remain closed during normal operation. Based on this conclusion, it was stated that the flow penalty could be eliminated from analyses at the request of the utility; however, the corresponding modifications to the Technical Specifications must be reviewed by us prior to implementation. MetEd retained the vent valve penalty in this analyses for cycle 2 and therefore additional conservatism exists.

Two further changes reflected in the cycle 2 reload report and the accompanying Technical Specifications are:

(a) The use of the B&W-2 CHF correlation down to pressures of 1750 psi instead of the previous lower pressure limit of 2,000 psi, and

(b) A reduction in the minimum allowable DNBR from 1.32 to 1.30.

We recently completed a re-evaluation of the B&W-2 CHF correlation to verify its continued suitability in relation to available rod bundle DNB data. We determined that the BAW-2 correlation continues to be an acceptable correlation over the pressure, quality, mass flux, rod diameter and rod spacing range of its original data base.

In conjunction with our reevaluation of the B&W-2 CHF correlation we also reviewed the MetEd's proposed modifications to the correlation for the cycle 2 core. The original data base for the correlation covered the pressure range 2000-2450 psia and resulted in a 1.32 minimum allowable DNB ratio to ensure with 99% confidence that 95% of the hot rods did not experience DNB. As an attachment to their letter of February 3, 1976, B&W provided information which compared the B W-2 CHF correlation with data in the low pressure range from five different test bundles. The mean measured-to-predicted ratio for all data was 1.05 and the minimum allowable DNBR was 1.29 for a 95% confidence that 95% of the hot rods at the DNBR would not experience DNB.

The 1.32 minimum DNB ratio used by B&W is based upon 95% of the hot rods at that DNBR not experiencing DNB, with a 99% confidence. If the confidence level is changed to 95%, which is consistent with regulatory requirements as expressed in the standard review plan, the minimum allowable DNBR becomes 1.30.

Based on the above, we find both the extension of the B&W-2 CHF correlation to pressures down to 1750 psia and the change to a minimum DNBR of 1.30 to be acceptable. The B&W-2 CHF correlation has been shown to be conservative in the low pressure region and the change to a 1.30 minimum DNBR is consistent with the requirements of Standard Review Plan 4.4.

Nuclear Analysis

MetEd has provided values for core physics parameters for the TMI-1 cycle 2 core which reflect minor differences when compared to those for cycle 1. These differences are attributable to the fact that the core has not yet reached an equilibrium cycle and such differences are to be expected. We have concluded that no significant changes exist in the core design between cycles 1 and 2. In addition, the same calculational methods and design information were used to obtain the important nuclear design parameters. Based on the above and the fact that startup tests (to be conducted prior to power operation) will verify that the critical aspects of core performance are within the assumptions of the safety analysis, we find MetEd's nuclear analysis for cycle 2 to be acceptable.

Accident and Transient Analysis

Accident and Transient analyses reported in paragraphs 7.1 through 7.14 of the TMI-1 cycle 2 reload report submitted February 11, 1976, were examined and we agree that the cycle 2 reload core is thermally and hydraulically conservative and of the same design and manufacture as the cycle 1 core. We also agree that the reactivity coefficients and other input data is the same as, or is bounded by previous analyses. We have reviewed MetEd's submittal and agree that in no case are the consequences of transients more severe than previously analyzed.

Fuel Rod Bow Evaluation

The effect of rod bowing on DNBR was considered. Our review of MetEd's submittal dated April 2, 1976, indicates that the peaking penalty due to rod bowing, which the licensee has calculated as 1.6% is acceptable. The effect of the rod bow penalty on the limits for normal operation as provided in BAW-10079, "Operational Parameters for B&W Rodded Plants," has been found by us to be within the conservatism of the current limits. The design basis values for average linear heat rate of 5.80 kw/ft and power spike of 1.022 are greater than the actual values of 5.73 kw/ft and 1.018, respectively. Thus adequate margin is provided to absorb the rod bow penalty.