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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 39 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 NO. 1

DOCKET NO. 50-289

Introduction

By letter dated January 9, 1978, Metropolitan Edison Company (Met Ed) requested amendment of Appendix A to Facility Operating License No. DPR-50 for Three Mile Island Nuclear Station, Unit No. 1 (TMI-1). The requested change, as revised by the Met Ed letter of April 3, 1978, would amend the TMI-1 Technical Specifications to reflect plant operating limits applicable during the first 125 ± 5 effective full power days (EFPD) of operation with the fuel loading to be used during Operating Cycle 4. By letter dated April 7, 1978, we requested additional information concerning the proposed amendment. This information was furnished by Met Ed in a letter dated April 10, 1978. Met Ed has stated that they will make application at a later date for amendment of the TMI-1 Technical Specifications as necessary to establish operating limits in Cycle 4 for the period from 125 ± 5 EFPD to the end of the cycle (approximately 280 EFPD). Supplementary information concerning the reactor high pressure trip and pressurizer code safety valve relief settings for Cycle 4 was provided by Met Ed letters of April 17 and 20, 1978.

Background

The Met Ed submittal of January 9, 1978, was presented to support operation for a full operating cycle (Cycle 4) following the refueling performed at the end of Cycle 3. As such, the analysis presented in the submittal was based on the expected exposure of Cycle 3 (270 ± 10 EFPD), and the intended exposure of Cycle 4 (265 ± 15 EFPD). Subsequent to making this submittal, and in the absence of progress towards early settlement of the national coal strike, Met Ed requested by letter dated February 17, 1978, amendment of the TMI-1 Technical Specifications as necessary to permit extension of Cycle 3 operation to 315 EFPD. Approval of this request was granted by our letter of March 7, 1978, (Amendment No. 38).

Shortly after receiving authorization for the extension of TMI-1 Cycle 3, Met Ed determined that it was not in their interest to utilize the full term of the extension. Accordingly, they terminated Cycle 3 on March 17, 1978, after 287.1 EFPD of operation and commenced refueling operations for Cycle 4.

Because the fuel burnup in Cycle 3 was greater than assumed in the original Cycle 4 analysis transmitted by Met Ed's letter of January 9, 1978, Met Ed, by letter dated April 3, 1978, submitted an amendment to their January 9, 1978 request which took into account the effect of the authorized extension. This amendment also proposed a revised fuel loading arrangement, which on the basis of experience with a similar Babcock & Wilcox-designed facility, is expected to provide a more uniform neutron flux distribution. Met Ed stated that because of the short time interval between the decision to terminate operation in the extended Cycle 3 and the projected completion of refueling for Cycle 4, there was insufficient time to perform the revised analyses necessary to support operation over the full term of Cycle 4. Accordingly, Met Ed in their letter of April 3, 1978, only proposed technical specifications applicable to the first 125 $\pm$ 5 EFPD of operation in Cycle 4. They state that a subsequent submittal covering the balance of Cycle 4 (to 265 $\pm$ 15 EFPD) will be made in May, 1978.

Evaluation

By references 1, 7, and 8, Met Ed requested changes to the Technical Specifications appended to the TMI-1 Operating License for Cycle 4 operation. The TMI-1 reactor core consists of 177 fuel assemblies. All of the Batch 3 assemblies will be discharged at the end of Cycle 3. Thirteen once-burned Batch 1 assemblies, with an initial enrichment of 2.06 wt % ^{235}U and eight Batch 2 assemblies, with an initial enrichment of 2.75 wt % ^{235}U, will be reloaded into the interior portion of the core. Batches 4 and 5 with initial enrichments of 2.64, and 2.85 wt % ^{235}U, respectively, will be shuffled to new locations within their present quadrant. Batch 6, which consists of 52 fresh assemblies with an initial enrichment of 2.85 wt % ^{235}U, will occupy the core periphery.

Fuel Assembly Mechanical Design

The types of fuel assemblies and pertinent fuel design parameters and dimensions for TMI-1 Cycle 4 are listed in Table 4-1 of the attachment to reference 1. The Mark B4 fresh fuel assemblies (Batch 6) are identical in concept and are mechanically interchangeable with those added in TMI-1 Cycle 3.

The Batch 6, 15 x 15 (Mark B-4), the Batch 1c, 15 x 15 (Mark B-2), and the Batch 2b, 15 x 15 (Mark B-3), fuel assembly designs have been previously reviewed and accepted by the NRC staff for use in TMI-1. Also, these types of assemblies have been operated in TMI-1. The reload assemblies, therefore, do not represent any unreviewed change in mechanical design from the reference cycle.

Met Ed has taken each fuel assembly design into account in the various mechanical analyses. The Batch 2b fuel is generally limiting because of its relatively low initial fuel pellet density, lower prepressurization, and previous incore exposure. The results of these analyses have shown that the mechanical design differences between fuels for Cycle 3 and Cycle 4 are negligible and are acceptable.

Creep collapse analyses were performed by Met Ed for three-cycle assembly power histories. The Batch 2b fuel is more limiting for cladding collapse due to its previous incore exposure time. The creep collapse analyses were performed based on the conditions set forth in reference 9 which have been previously found acceptable.[10] The collapse time for the most limiting assembly was conservatively determined to be more than 30,000 EFPH (effective full power hours), which is longer than the maximum design exposure for the total of three cycles.

Met Ed stated that the TMI-1 stress parameters were enveloped by a conservative fuel rodstress analysis. The following conservatisms with respect to TMI-1 fuel were used in the analysis: lower post-densification internal pressure, lower initial pellet density, higher system pressure, and higher thermal gradient across the cladding.

The licensee has referenced the report BAW-1389 which presents calculations of cladding stress at various power levels and fuel burnups for TMI-1 fuel. These calculations show that in no case does the stress exceed the yield stress. This is acceptable to the staff.

The fuel design criteria specify a limit to the cladding plastic circumferential strain of 1.0%. The pellet design is established for plastic cladding strain of less than 1% at values of maximum design local pellet burnup and heat generation rate, which are considerably higher than the values for TMI-1 fuel. This will result in an even greater margin than the analysis demonstrated. The strain analysis is also based on the maximum manufacturing specifications value for the fuel pellet diameter and density and the lowest permitted manufacturing specifications tolerance for the cladding ID.

The linear heat generation rate (LHGR) capabilities are based on centerline fuel melt and were established by Met Ed using the TAFY-3 code[11] with fuel densification to 96.5% of theoretical density.

All the fuel assemblies in the Cycle 4 core are thermally similar. The fresh Batch 6 fuel inserted for Cycle 4 operation introduces no significant differences in fuel thermal performance relative to the other fuel remaining in the core, and its LHGR limit has been established as 20.15 KW/ft.

Met Ed's thermal analysis of the fuel rods assumed in-reactor densification to 96.5% theoretical density. The analytical methods utilized are the same as those for Cycle 3.[12] These analyses were based on the lower tolerance limit of the fuel density specification and assumed isotropic diametral shrinkage and anisotropic axial shrinkage resulting from fuel densification.

The Batch 6 fuel assemblies are not new in concept, nor do they utilize different component materials. Therefore, the chemical compatibility of all possible fuel-cladding-coolant assembly interactions for the Batch 6 fuel assemblies are identical to those of the present fuel.

This fuel as proposed for reload in TMI-1 has had considerable operating experience. The Batch 4, 5, and 6 fuel assemblies are not new in concept and do not use different component materials. The fuel assemblies for Cycle 4 operation will not exceed any design life limits. We conclude, therefore, that the fuel mechanical design for Cycle 4 operation is acceptable.

Nuclear Analysis

Table 5-1 of the attachment of reference 1 compares the core physics parameters of Cycles 3 and 4. The values for both cycles were generated by Met Ed using PDQ07. Since the core has not yet reached an equilibrium cycle, differences in core physics parameters are to be expected between the cycles. The extended Cycle 3 produced a larger cycle differential burnup than is expected for Cycle 4. The accumulated average core burnup will be higher in Cycle 4 than in Cycle 3 because of the presence of the once-burned Batch 1c and 2b fuel and the extension of Cycle 3.

The critical boron concentrations for Cycle 4 are approximately the same as for Cycle 3. The control rod worths are sufficient to maintain the required shutdown margin. The maximum stuck rod worths for Cycle 4 are less than those in Cycle 3. The adequacy of the shutdown margin with Cycle 4 rod worths has been demonstrated analytically by Met Ed. Met Ed's shutdown calculations conservatively used a poison material depletion allowance and 10% uncertainty on net rod worth.

The same calculational methods and design information were used by Met Ed to obtain the nuclear design parameters for Cycles 3 and 4. The mode of reactor operation has been changed from a rodded to an unrodded feed-bleed mode. No changes to the makeup and purification system were necessary for this mode of operation. For operation in the unrodded mode, the required feed-bleed capabilities are the same as for operation in the rodded mode with the addition of adjusting the Reactor Coolant System (RCS) boron concentration to maintain the regulating rods within specified control bands. The plant maneuverability is limited by the ability of the waste processing system to handle the waste generated.

Met Ed had intended(1) to cross-core shuffle the fuel* for the Cycle 4 reload. However, due to quadrant flux tilt problems encountered at another Babcock & Wilcox-designed facility using cross-core shuffle, this plan was changed.(7) All fuel shuffling for Cycle 4 will now be limited by the Met Ed to the quadrant in which the fuel resided in Cycle 3. This method of fuel shuffling tends to reduce possible carry-over effects of any burnup asymmetry that might be present in the previous cycle. The lowest indicated tilt (1.2%), which occurred at the end of Cycle 3, should be further reduced by this method of fuel shuffle.

Met Ed requested a change in the technical specification limit on quadrant tilt to increase the allowable maximum tilt from 3.41% to 4.92%. The submittals (8, 16) on quadrant tilt indicate that the change will restore the allowable tilt level to that permitted in Cycles 1 and 2 for TMI-1. The lower tilt limit used for Cycle 3 operation was to offset a required peaking penalty due to fuel rod bow. For Cycle 4, TMI-1 has used a statistical combination of peaking factors, removed the densification power spike from ECCS-dependent technical specification limits, and reduced the core peaking factor. By use of data in the Babcock & Wilcox (B&W) report BAW-10078 and data obtained from Oconee Unit 1, Cycle 4, Met Ed has demonstrated that an increase in allowable tilt to 4.92% for TMI-1, Cycle 4 is acceptable. The information presented has been reviewed and found acceptable.

In view of the above and the fact that startup tests (to be conducted prior to power operation) will verify that the significant aspects of the core performance are within the assumptions of the safety analsyis, we find Met Ed's nuclear analysis for Cycle 4 to be acceptable.

Thermal-Hydraulic Analysis

The major acceptance criteria which are used for the thermal-hydraulic design are specified in Standard Review Plan (SRP) 4.4. These criteria establish acceptable limits on departure from nucleate boiling (DNB). The thermal-hydraulic analysis for the TMI-1, Cycle 4 reload was performed by Met Ed using previously approved models and methods. Certain aspects of the thermal-hydraulic design are new for the Cycle 4 core and are discussed below.

* Move fuel from one quadrant to the opposite quadrant.

The thermal-hydraulic design in support of Cycle 4 operation utilized the methods and models described in references 12 and 13. Met Ed stated that Cycle 4 analyses and resulting setpoints have been based on 106.5% of the design reactor coolant (RC) system flow rate.

The core configuration for Cycle 4 differs slightly from that of Cycle 3 in the proportion of Mark B2, Mark B3 and Mark B4 fuel assemblies contained in the core. Specifically, 52 Mark B4 assemblies will replace an equal number of Mark B3 assemblies used in Cycle 3. Mark B4 assemblies differ from the Mark B2 and B3 primarily in the design of the end fitting, which results in a slight reduction in flow resistance for the B4 design. No credit was taken by Met Ed in the analyses for the increased flow to the Mark B4 assemblies, located in the hottest core locations, as a result of the presence of the Mark B2 and B3 assemblies.

Met Ed used the BAW-2 CHF correlation (14) for thermal-hydraulic analysis of Cycle 4. This correlation has been reviewed and approved for use with the Mark B fuel assembly design.(15)

The effect of fuel densification on the minimum departure from nucleate boiling ratio (DNBR) is primarily a result of the reduction in active fuel length, which increases the average heat flux. Met Ed's Cycle 4 DNBR analysis was based on a cold densified active length of 140.2 inches, a value selected to apply generically to a number of B&W plants. This is a conservative method of applying the densification effect since all the fuel assemblies in Cycle 4 have longer densified lengths and because no credit is taken for axial thermal expansion of the fuel column.

The potential effect of fuel rod bow on DNBR was previously evaluated for Cycle 3.(17) The effect of fuel rod bow on DNBR would be unchanged during Cycle 4 operation. Therefore, the previous evaluation of this potential effect remains unchanged.

Accident and Transient Analysis

The accident and transient analyses as provided by Met Ed demonstrate that the TMI-1 FSAR analyses conservatively bound the predicted conditions of the TMI-1, Cycle 4 core and are, therefore acceptable. Met Ed has stated that each FSAR accident analysis has been examined, with respect to changes in Cycle 4 parameters, to determine the effects of the reload and to ensure that performance is not degraded during hypothetical transients. The core thermal parameters used in the FSAR accident analysis were design operating values based on calculated values plus uncertainties. FSAR values of core thermal parameters were compared with those calculated in the Cycle 4 analysis. The effects of fuel densification on the FSAR accident results have been evaluated and are reported in the TMI-1 fuel densification report.(12) Since Cycle 4 reload fuel assemblies contain fuel rods with a density higher than those considered there, the conclusions derived in that report are valid for TMI-1 Cycle 4. Calculational techniques and methods for Cycle 4 analyses remain consistent with those used for the FSAR.

With respect to radiation doses, Met Ed reported that because of improved fuel utilization and improved calculational methods, they now estimate they are achieving a higher plutonium-to-uranium fission ratio. Because plutonium has a higher iodine fission yield than uranium, more iodine will be produced. Met Ed estimates that the increased iodine production will increase the 2-hour thyroid doses given in the FSAR by 8 to 15%. We have reviewed the effects of this increase and conclude that the consequences of all accidents remain well within acceptable limits.

Met Ed has clarified(8) the manner in which errors in reactor power measurement are incorporated in their analyses. The setpoints used in the accident analyses assumptions contain the required calorimetric power measurement uncertainty of 2% plus a 4% uncertainty to account for errors in neutron power measurement. The 2% calorimetric uncertainty accounts for steady-state power measurement error. The 4% neutron power measurement uncertainty allows for steady-state and transient errors following maneuvering transients. We conclude that this manner of accounting for these uncertainties is acceptable and therefore the resultant analyses of postulated accidents and transients are conservative.

Reactor High Pressure Trip and Pressurizer Code Safety Valve Settings

By letter dated April 6, 1977, we authorized Met Ed to increase the TMI-1 reactor high pressure trip setting from 2355 psig to 2405 psig, and to increase the relief setting of the pressurizer code safety valves from 2435 psig to 2500 psig. Because the assumptions used by Met Ed in justifying these changes were applicable to Cycle 3, our approval of the increased settings was limited to that cycle.

By letter dated April 17, 1978 as amended by letter dated April 20, 1978, Met Ed submitted their evaluation in support of the continued acceptability of these settings. Their evaluation indicates that under Cycle 4 operating conditions and assuming a more conservative instrument error, the conservatively calculated peak reactor coolant system pressure resulting from the feedwater line break (the limiting accident) is increased from 2734 psig (the Cycle 3 value) to 2749.3 psig. Since this value is less than the safety limit for reactor coolant system pressure of 2750 psig as stated in Technical Specification 2.2.1, we conclude that retention of the present reactor high pressure trip and pressurizer code safety valve settings for Cycle 4 is acceptable. Although there is a reduction in margin, with some portion due to use of additional conservatism, the vessel stress is still within the code requirements for relief valve capacity. For this reason the reduction in margin is not significant.

Startup Tests

The physics startup test program for Cycle 4 as stated in Section 9 of Met Ed's January 9, 1978 submittal has been reviewed. Additional information was requested and supplied in the April 10, 1978 Met Ed submittal. The physics startup test program includes zero power measurements of critical boron concentration, temperature coefficients, ejected control rod worth and control rod group reactivity worth. Power distribution measurements will be made at higher powers.

This program has been reviewed by the NRC staff and found to be acceptable. Because there are areas in Met Ed's safety analysis that warrant verification by the physics startup test program, we have requested Met Ed to submit a report of the results of these tests. Met Ed has agreed to submit such a report within 90 days of the completion of the tests. We find this acceptable.

ECCS Analysis

B&W has recently discovered a deficiency in the method used to calculate Emergency Core Cooling System (ECCS) performance. This matter is being reviewed by the NRC staff.

Conclusion

We have determined that the amendment does not authorize a change in effluent types or total amounts nor an increase in power level and will not result in any significant environmental impact. Having made this determination, we have further concluded that the amendment involves an action which is insignificant from the standpoint of environmental impact and, pursuant to 10 CFR §51.5(d)(4), that an environmental impact statement, or negative declaration and environmental impact appraisal need not be prepared in connection with the issuance of this amendment.

With the exception of the matter of ECCS performance, we have concluded, based on the considerations discussed above, that: (1) because the amendment does not involve a significant increase in the probability or consequences of accidents previously considered and does not involve a significant decrease in a safety margin, the amendment does not involve a significant hazards consideration; (2) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner; and (3) such activities will be conducted in compliance with the Commission's regulations and the issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public. Our conclusions in regard to ECCS performance are addressed in the accompanying Exemption.

Dated: April 27, 1978

References:

(1) Letter J.G. Herbein (Met Ed) to R.W. Reid (NRC) dated January 9, 1978.

(2) Letter J.G. Herbein (Met Ed) to R.W. Reid (NRC) dated February 17, 1978.

(3) Letter J.G. Herbein (Met Ed) to R.W. Reid (NRC) dated March 1, 1978.

(4) Letter J.G. Herbein (Met Ed) to R.W. Reid (NRC) dated March 13, 1978.

(5) Letter J.G. Herbein (Met Ed) to R.W. Reid (NRC) dated March 13, 1978.

(6) Letter J.G. Herbein (Met Ed) to R.W. Reid (NRC) dated March 14, 1978.

(7) Letter J.G. Herbein (Met Ed) to R.W. Reid (NRC) dated Arpil 3, 1978.

(8) Letter J.G. Herbein (Met Ed) to R.W. Reid (NRC) dated April 10, 1978.

(9) Program to Determine In-Reactor Performance of B&W Fuels - Cladding Creep Collapse, BAW-10084, Rev. 1, Babcock & Wilcox, November 1976.

(10) Letter from A. Schwencer (NRC) to J.F. Mallary (B&W) dated January 29, 1975.

(11) C.D. Morgan and H.S. Kao, TAFY - Fuel Pin Temperature and Gas Pressure Analysis, BAW-10044, Babcock & Wilcox, May 1972.

(12) TMI-1 Fuel Densification Report, BAW-1389, Babcock & Wilcox, June 1973.

(13) Three Mile Island Nuclear Station, Unit 1, Final Safety Analysis Report, Docket No. 50-289.

(14) Correlation of Critical Heat Flux in a Bundle Cooled by Pressurized Water, BAW-10000A, Babcock & Wilcox, June 1976.

(15) Letter from J. Stolz (NRC) to K.E. Surke (B&W), dated April 15, 1976.

(16) Letter W.R. Gibson (B&W) to R. Landry (NRC) dated April 19, 1978.

(17) Letter R.W. Reid (NRC) to R.C. Arnold (Met Ed) dated March 7, 1977.