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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 45 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

1.0 Introduction

By letter(1) dated June 23, 1978, and supplemented by letter(2) dated August 7, 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 would amend the TMI-1 Technical Specifications to reflect Cycle 4 plant operating limits out to 280 effective full power days (EFPD). Previously by letter(3) dated January 9, 1978, as revised by letter (4) dated April 3, 1978, Met Ed requested changes to the TMI-1 Technical Specifications which provided for plant operating limits applicable during the first 130 EFPD of Cycle 4. The safety analysis supporting the present request is contained in the Babcock and Wilcox (B&W) Report(5) "Three Mile Island Unit 1 Cycle 4 Report," BAW-1473, together with Met Ed's safety evaluation(1) for Technical Specification Change Request No. 84 for TMI-1. In addition, Met Ed requested that the reactor high pressure trip setpoint be changed (lowered) permanently based on the results of a reanalysis of the limiting pressurization event.

2.0 Background

The Met Ed submittal of January 9, 1978, was originally presented to support operation of TMI-1 for all of Cycle 4 following the refueling performed at the end of Cycle 3. As such, the Cycle 4 analysis, presented in the submittal, was based on the expected energy output for Cycle 3 (270 ± 10 EFPD) and the planned energy output 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 then in effect, Met Ed requested(6) amendment of the TMI-1 Technical Specifications as necessary to permit extension of Cycle 3 operation to 315 EFPD. Approval(7) of this request was subsequently granted by the NRC.

Shortly after receiving authorization for the extension of Cycle 3, Met Ed determined that it was not in their best interest to utilize the full term of the extension. Accordingly, Met Ed terminated Cycle 3 on March 17, 1978, after a 287.1 EFPD energy output and commenced refueling operations in preparation for Cycle 4.

Since the fuel burnup in Cycle 3 was greater than that assumed in the original Cycle 4 analysis transmitted with Met Ed's January 9, 1978, letter, Met Ed amended(4) their request to take into account the effects of the authorized extension. Moreover 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, Met Ed stated that there was insufficient time to perform the revised reload analyses necessary to support TMI-1 operation for the fuel term of Cycle 4. Accordingly, Met Ed proposed(4) Technical Specifications applicable only to the first 125 ± 5 EFPD of operation in Cycle 4.

With their June 23, 1978 letter, Met Ed submitted a revised Cycle 4 reload analysis report which was based on (1) the actual Cycle 3 energy output of 287.1 EFPD, (2) the fuel loading and operating technical specifications approved(8) for the first 130 EFPD of operation in Cycle 4 and (3) the approved(9) revised technical specification limits for allowable neutron flux tilt, axial power imbalance and axial power shaping rod position.

In connection with the reactor high pressure trip and Code safety valve settings, we first authorized(10) Met Ed to increase the TMI-1 reactor high pressure trip setpoint from 2355 psig to 2405 psig, and to increase the pressurizer code safety valve relief setpoints 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 increase in high pressure trip settings was limited to that cycle.

Subsequently, by letter(11) dated April 17, 1978, as amended by letter(12) dated April 20, 1978, Met Ed submitted their evaluation to support the continued acceptability of these settings. Their evaluation concluded that because of the Cycle 4 operating conditions and the assumption of a more conservative pressure transmitter instrumentation error, the calculated peak reactor coolant system (RCS) pressure, resulting from a postulated feedwater line break (the limiting accident), increased from 2743 psig (the Cycle 3 value) to 2749.3 psig. Since this value was less than the 2750 psig safety limit for reactor coolant system pressure as stated in TMI-1 Technical Specification 2.2.1, we concluded that retention of the previously approved reactor high pressure trip and pressurizer code safety valve settings for Cycle 4 was acceptable.

More recently, by letter[13] dated May 17, 1978, Met Ed informed the NRC that following completion of a bounding overpressurization analysis they discovered that the peak calculated RCS pressure (with the previously approved 2405 psig setpoint) exceeded the 2750 psig pressure safety limit. Accordingly, Met Ed voluntarily lowered the RCS high pressure setpoint to 2390 psig. The current submittal[1] from Met Ed is intended to justify, on a permanent basis, the adequacy of a 2390 psig RCS high pressure setpoint for the remainder of Cycle 4 and all subsequent cycles.

3.0 Evaluation

Our safety evaluation[8] of April 27, 1978, evaluated the acceptability of the reload fuel and core design and associated Technical Specifications for the first 130 EFPD of TMI-1 Cycle 4. Accordingly, this evaluation addresses only those safety related items which are pertinent to the extension of our approval of Cycle 4 from 130 EFPD to 280 EFPD.

3.1 Fuel Assembly Mechanical Design

Met Ed has taken each fuel assembly design in the Cycle 4 TMI-1 core into account in the various mechanical analyses. The Batch 2b fuel (discussed in Reference 8) is generally limiting because of its relatively low initial fuel pellet density, low prepressurization and exposure history. The results of the mechanical design analyses have shown that the mechanical design differences between fuel for Cycle 3 and Cycle 4 are negligible and are acceptable.

Fuel Rod Design

Met Ed reevaluated each fuel rod thermal-mechanical design analysis in view of the actual Cycle 3 exposure and projected Cycle 4 exposure history out to 280 EFPD. Our evaluation of these fuel rod design analyses for the revised exposure conditions follows.

Reference creep collapse analyses[14] were performed based on a limiting three-cycle assembly power history. The creep collapse analyses were performed based on the conditions and modeling assumptions set forth in Reference 14 which have been previously found acceptable[15]. The reference collapse time for the limiting assembly in the analysis was conservatively determined to be greater than 30,000 Effective Full Power Hours (EFPH) which is longer than the 23,976 EFPH accumulated residence time the the most limiting Batch 2b assemblies, based on a core energy output of 280 EFPD for Cycle 4.

Met Ed has referenced the Babcock and Wilcox report, BAW-1389[16], as the applicable fuel cladding stress analysis for Cycle 4. This report presents calculations of cladding stress at various rod power and burnup levels for the fuel types present in the TMI-1 core during Cycle 4. Conservatisms in the stress analysis include lower post-densification internal pressure, a lower initial pellet density, a higher system pressure and a higher thermal gradient across the cladding. These calculations show that in no case does the cladding stress exceed the yield stress. Met Ed stated that the revised

Cycle 4 fuel rod exposure and power history (based on the actual end of Cycle 3 assembly-wise exposures from the previous core), was enveloped by the fuel rod stress analysis. Since the reference calculations show that the cladding stress does not exceed the yield stress, the cladding stress evaluation for the TMI-1 Cycle 4 core is acceptable to us.

One fuel rod design criterion specifies that the cladding plastic circumferential strain should not exceed 1.0%. The strain analysis, which is performed to assure conformance with this criterion, is based on the most adverse tolerances from the manufacturing specifications for the fuel pellet and cladding diameter and fuel pellet density. The cladding and pellet design results in a plastic cladding strain of less than 1% for the maximum design local pellet burnup and heat generation rate conditions. Based on the revised Cycle 4 burnup and power history analysis, these design conditions are considerably higher than those applicable during Cycle 4 for TMI-1 fuel. This will result in a margin greater than demonstrated by the analysis.

The generic fuel rod temperature/fuel rod pressure analysis(17) was reviewed for applicability for Cycle 4 operation based on a planned cycle length of 280 EFPD together with the revised Cycle 3 bundle exposure conditions. Met Ed stated that the calculated maximum pin burnup and power history were bounded by those used in the generic fuel rod temperature/fuel rod pressure analyses.

During the last few years, however, data have become available indicating that fission gas release from LWR fuel increases with fuel burnup. This enhanced release at high burnup affects the fuel rod internal gas pressure and temperature analyses. The TMI-1 Cycle 4 reload was analyzed with the TAFY(17) fuel performance code which was approved prior to the identification of enhanced fission gas release behavior at high burnup. A more recent B&W fuel performance code, TACO, which models this effect, has been approved. Recently, B&W provided the NRC staff with an evaluation comparing the TAFY Code gas release predictions (without burnup-enhanced gas release) to the TACO Code gas release predictions (with a burnup-enhanced gas release model included). A comparison of the gas release predictions of the two codes shows that the fuel rod pressure analysis referenced by Met Ed (using TAFY) still provides a conservative prediction of pin pressure. Thus, we find the Met Ed fuel rod pressure analysis for TMI-1 for the Cycle 4 core to be acceptable.

3.2 Nuclear Analysis

Table 5-1 of Reference 5 compares the core physics parameters of Cycle 3 with the revised core physics parameters of Cycle 4 which are based on the Cycle 3 (287.1 EFPD) energy output actually achieved. In general the revised Cycle 4 parameters are slightly different from the original Cycle 4 parameters due to the additional exposure accumulated in the Cycle 3 fuel assemblies used for the Cycle 4 reconstituted core. The values for Cycle 3 and the original and revised values for Cycle 4 were generated by B&W for Met Ed using the PDQ07 computer code. Moreover, since the core has not yet reached an equilibrium cycle, minor differences in core physics parameters are expected from cycle-to-cycle. Cycle 4 will have a slightly longer design life than Cycle 3 due to the presence of the once-burned Batch 1c and 2b fuel assemblies in the Cycle 4 core.

As also seen in Table 5-1 of Reference 5, the revised critical boron concentration predictions for Cycle 4 are similar to those for Cycle 3, and do not differ significantly from the original predictions. The BOC* and EOC* critical boron concentrations for the revised analysis show a slight decrease from the previous analysis due to the reduction in BOC-4 and EOC-4 core reactivity, which resulted from the additional depletion of the once burned and twice burned assemblies arising from the extension of Cycle 3. The reduction in critical boron concentration has also provided for a more negative moderator-temperature coefficient during the cycle as shown in Table 5-1 of Reference 5. Table 5-2 of Reference 5 provides the revised BOC and EOC shutdown margin calculation based on a comparison of the revised available rod worth and revised required rod worth analyses. The reanalysis shows that the available control rod worths are sufficient to maintain the required 1.0% $\Delta k/k$ shutdown margin at both BOC-4 and EOC-4 conditions even with the slightly increased maximum stuck rod worth arising from the extension of Cycle 3. Met Ed's revised shutdown calculations conservatively included a poison material depletion allowance and a 10% uncertainty allowance on net rod worth. The maximum calculated ejected rod worth based on the revised analysis was 0.85% $\Delta k/k$ at hot zero power and 0.25% $\Delta k/k$ at hot full power. These worths are below the respective limits of 1.0% $\Delta k/k$ and 0.65% $\Delta k/k$ and are acceptable.

The same calculational methods and design information were used to develop the nuclear design parameters for the revised Cycle 4 analysis as were used for the Cycle 3 and the original Cycle 4 analysis. However, starting with Cycle 4, Met Ed changed the mode of operation of TMI-1 from a rodded to an unrodded feed-bleed mode. Additionally, for Cycle 4, Met Ed abandoned the previous practice of cross-core fuel shuffling to reduce adverse effects on quadrant flux tilt which this practice is now thought to produce. These changes are discussed in Reference 8. The measured(19) radial and total power peaking at 100% power and 25 EFPD into Cycle 4 were respectively 8.9% and 6.9% higher than the predicted valves. However, these differences were within the revised(9) allowable uncertainties of 11% and 13.5% for radial and total peaking respectively.

* BOC: beginning of cycle; EOC: end of cycle.

3.3 Thermal-Hydraulic Analysis

The major acceptance criteria which are used for the TMI-1 Cycle 4 core thermal-hydraulic design are specified in Standard Review Plan 4.4. These criteria establish limits on the departure from nucleate boiling ratio (DNBR). The thermaly-hydraulic evaluation for Cycle 4 up to a 280 EFPD energy output was based on a comparison with the significant assumptions and results from the Cycle 2 thermal-hydraulic analysis. The Cycle 2 TMI-1 analysis was performed by B&W for Met Ed using the previously approved models and methods described in References 8 and 20. The reference analysis utilized the BAW-2 CHF correlation.(21) This CHF correlation has been reviewed and approved by the NRC staff for use with the various Mark B fuel assembly designs in the Cycle 4 TMI-1 core.

The core loading for Cycle 4 differs slightly from that of Cycle 2 in the proportion of Mark B2, March B3 and Mark B4 fuel assemblies. The Mark B4 assemblies differ from the Mark B2 and B3 assemblies primarily in the design of the end fitting, which results in a slight reduction in flow resistance for the Mark B4 design. A comparison of the minimum DNBR calculated from the Cycle 2 analysis to the minimum DNBR obtained from an analysis of an all Mark B4 core shows that the Cycle 2 core results in a lower minimum DNBR. Moreover, since the limiting assembly is of the Mark B4 design, the addition of the higher resistance Mark B2 and Mark B3 assemblies will tend to increase flow through the limiting Mark B4 assemblies.

Additional DNBR conservatism can be identified in the reference thermal-hydraulic calculations compared to Cycle 4 when peaking factors are considered. The reference analysis was based on a radial local peak of 1.783. The revised Cycle 4 power distribution analysis shows that 13.2% margin is available at BOC and 21.3% margin is available at EOC between the reference Cycle 2 radial local peak and the predicted Cycle 4 radial local peak, even when an 8% nuclear uncertainty factor is included.

The potential effects of fuel rod bowing on DNBR were previously evaluated by the NRC staff for Cycle 3.(22) The rod bow penalty for the Cycle 4 analysis was based on a maximum assembly burnup of 33,000 MWD/MTU. The revised analysis for Cycle 4 shows the maximum assembly burnup to be 31,094 MWD/MTU at EOC-4 (280 EFPD). Thus the previously accepted(22) margins used to offset the rod bowing penalty will adequately offset the adverse effects of fuel rod bowing during Cycle 4.

3.4 Accident and Transient Analysis

Met Ed stated that the transients and accidents analyzed as part of the Final Safety Analysis Report (FSAR) and densification reports were reevaluated based on a comparison of the revised Cycle 4 parameters with the bounding input parameters assumed in these reference analyses. The calculational methods used to develop the key input parameters are the same as the methods used for the reference analyses. The calculated core thermal and kinetics parameters used in the FSAR accident and transient analyses were design operating values based on projected limiting nominal values plus uncertainties. The effects of fuel densification on the FSAR results have already been evaluated and are documented in the TMI-1 fuel densification report. A comparison of the key revised Cycle 4 accident and transient parameters with the bounding values assumed in the FSAR and fuel densification report shows that the reference analyses conservatively bound the most adverse accident and transient inputs applicable for Cycle 4. Finally, since the Cycle 4 reload fuel assemblies contain fuel rods with a pellet density higher than those considered in the densification report, the conclusions derived in the densification report remain valid for TMI-1 during Cycle 4.

3.5 Reactor High Pressure Trip Setpoint

Met Ed has recalculated the peak RCS pressure following a postulated feedwater line break accident at TMI-1 using the CADDS(23) computer code to demonstrate the acceptability of a revised reactor high pressure trip setpoint code. The reanalysis included a corrected safety valve flow rate corresponding the 156 lbm/sec at 2500 psig and an increased pressure transmitter instrumentation error of 45 psi, together with the lowered high RCS pressure trip setpoint of 2390 psig. The assumed safety valve setpoint of 2500 psig was unchanged from the previous analysis.

The subject pressurization reanalysis was part of a sensitivity study performed by Met Ed to determine the sensitivity of peak transient pressure to changes in the RCS high pressure trip setpoint and moderator coefficient. The results show that the peak transient pressure at the pump discharge will not exceed the 2750 psig RCS pressure safety limit for a high pressure trip setpoint of 2390 psig, whenever the moderator coefficient is non-positive. The pressurization analysis was also based on a Doppler coefficient of $1.49 \times 10^{-5}\% \Delta k/k/^{o}F$, which is conservative for Cycle 4. Since the present TMI-1 Technical Specifications require that the moderator coefficient be negative, at power levels greater than 95% of rated power, and because the moderator coefficient is predicted to meet this specification throughout Cycle 4, we find the proposed 2390 psig high pressure trip setpoint to be acceptable for Cycle 4 of TMI-1

4.0 Technical Specification Changes

The TMI-1 Technical Specification changes requested by Met Ed were based on the actual Cycle 3 burnup of 287.1 EFPD and a design Cycle 4 burnup of 280 EFPD. The proposed changes also reflect the increase in radial and total peak nuclear uncertainty factors from 5% and 7.5% to 11% and 13.5% respectively previously approved by the NRC staff, together with the previously approved changes to the allowable neutron flux tilt, axial power imbalance and axial power shaping rod position technical specifications.

The centerline fuel melt margin (kw/ft) vs. reactor power imbalance limit lines have been changed from those currently being used for the first 130 EFPD of Cycle 4. The revised limits are more restrictive for negative imbalances and predominantly reflect the increase in nuclear uncertainty factors described above. The methods used to account for the new calculational nuclear uncertainties are consistent with those previously used by Met Ed in developing the present core protection safety limits. Thus, we find the proposed revisions to these limits to be acceptable.

Met Ed has also changed the limit lines defining the protection system maximum allowable setpoint vs. power imbalance. The revised protection system setpoint figure, which is applicable for a Cycle 4 length of 280 EFPD, is less restrictive for negative imbalances compared to the curve authorized for the first 130 EFPD of Cycle 4, but is based on a more refined analysis and is consistent with the basis provided by the kw/ft safety limits discussed above. Again, since the methods used to develop these limits are consistent with the methods previously used by Met Ed, we find the proposed revisions to be acceptable.

For the period during Cycle 4 from 130 EFPD to 280 EFPD, Met Ed has proposed to use a different limit on control rod position and a different loss of coolant accident (LOCA) dependent power imbalance envelope than those used for the period from BOC-4 to 130 EFPD. Both curves are more restrictive in the latter part of the cycle to reflect the effects of the revised shutdown margin calculation and nuclear uncertainty. However, Met Ed has retained the same limits for operation beyond 130 EFPD in the high power operating region of the figures by imposing a more restrictive power imbalance envelope in the latter part of Cycle 4 and by taking credit for the revised axial power shaping rod position limit curve previously approved by the NRC staff. We find the revised limit curves, which have been constructed utilizing methods consistent with those used previously, to be acceptable.

On May 19, 1978, we granted an exemption from the provisions of 10 CFR 50.46 to Met Ed to permit operation of TMI-1 pending resolution of concerns relating to small break loss of coolant accident. Subsidiary technical matters related to the granting of this exemption included questions concerning detector measurement uncertainties and greater than predicted power peaking which were discovered during startup physics tests conducted at the beginning of Cycle 4. These matters were resolved at that time by issuance of modified technical specifications which accounted for these concerns. Based on our review of Met Ed's submittal, we conclude that the technical specification changes authorized by this amendment also account for these concerns, where applicable, and therefore, the technical basis for granting the exemption of May 19, 1978, remains valid and the exemption may remain in force.

Finally, we find the proposed reduction in the maximum reactor coolant system high pressure trip setpoint from 2405 psig to 2390 psig, based on the results of the revised overpressurization analysis, to be acceptable.

5.0 Environmental Consideration

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.

6.0 Conclusion

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 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.

Dated: September 22, 1978

References:

1. Met Ed letter (J. G. Herbein) to NRC (R. W. Reid) dated June 23, 1978.
2. Met Ed letter (J. G. Herbein) to NRC (R. W. Reid) dated August 7, 1978.
3. Met Ed letter (J. G. Herbein) to NRC (R. W. Reid) dated January 9, 1978.
4. Met Ed letter (J. G. Herbein) to NRC (R. W. Reid) dated April 3, 1978.
5. "Three Mile Island Unit I Cycle 4 Reload Report," BAW-1473, Babcock & Wilcox Report dated November 1978, (revised June 1978)
6. Met Ed Letter (J. G. Herbein) to NRC (R. W. Reid) dated February 17, 1978.
7. NRC letter (R. W. Reid) to Met Ed (J. G. Herbein) dated March 7, 1978.
8. NRC letter (R. W. Reid) to Met Ed (R. C. Arnold) dated April 27, 1978.
9. NRC letter (R. W. Reid) to Met Ed (J. G. Herbein) dated May 19, 1978.
10. NRC letter (R. W. Reid) to Met Ed (R. C. Arnold) dated April 6, 1977.
11. Met Ed letter (J. G. Herbein) to NRC (R. W. Reid) dated April 17, 1978.
12. Met Ed letter (J. G. Herbein) to NRC (R. W. Reid) dated April 20, 1978.
13. Met Ed letter (J. G. Herbein) to NRC (B. H. Grier) dated May 17, 1978.
14. Program to Determine In Reactor Performance of B&W Fuels - Cladding Creep Collapse, BAW-10084 Rev.1 and Babcock & Wilcox Report, November 1976.
15. NRC letter (J. Stolz) to B&W (K. Surke) dated April 15, 1976.
16. TMI-1 Fuel Densification Report, BAW-1389, Babcock and Wilcox Report, June 1973.
17. C. D. Morgan and H. S. Kao TAFY - Fuel Pin Temperature and Gas Pressure Analysis, BAW-10044, Babcock and Wilcox Report, May 1972.
18. B&W letter (J. H. Taylor) to NRC (P. S. Check) dated July 18, 1978.
19. Met Ed Letter (J. G. Herbein) to NRC (R. W. Reid) dated July 31, 1978.
20. Three Mile Island Nuclear Station, Unit 1, Final Safety Analysis Report, Docket No. 50-289.

References-Cont'd.

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

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

23. "CADDS, Computer Application to Direct Digital Simmulation of Transients in PWRs With or Without Scram", BAW-10098-P, December 1974.