July 23, 1976

Docket No.: 50-289

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Metropolitan Edison Company
ATTN: Mr. R. C. Arnold
Vice President - Generation
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Reading, Pennsylvania 19603

Gentlemen:

Your letter of April 12, 1976 (GQL 0512), relative to outstanding requests for changes in technical specifications is hereby acknowledged. The Commission has issued the enclosed Amendment No. 18 to Facility License No. DPR-50 for the Three Mile Island Nuclear Station, Unit No. 1. This amendment is in response to your requests dated January 31, July 31, August 21, September 4, and October 29, 1975. Your request No. 26 of December 12, 1975, relating to discharge limits for liquid effluents, is still under review.

This amendment authorizes miscellaneous changes to the Environmental Technical Specifications. These changes involve (a) correction of typographical errors; (b) wording changes for clarification of intent; (c) administrative changes and titles; (d) location of radiological monitoring stations; (e) monitoring conditions; and (f) sensitivity of radiological analyses. Change Request 8.1 submitted in your letter of January 31, 1975, is still under review. We have made certain changes in the Technical Specifications you proposed and have discussed these changes with your staff.

We have evaluated the potential for environmental impact of plant operation for each of the changes and our appraisals are summarized in an enclosure to this letter. 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 statement, negative declaration or environmental impact appraisal need not be prepared in connection with the issuance of this amendment.

Since the amendment applies only to environmental matters, it does not involve significant new safety information of a type not considered by a previous Commission safety review of the facility. It does not involve a significant increase in the probability or consequences of an accident, does not involve a significant decrease in a safety margin, and therefore does not involve a significant hazards consideration. We have also concluded that there is reasonable assurance that the health and safety of the public will not be endangered by this action.

A copy of the related Federal Register Notice is also enclosed.

Sincerely,

Robert W. Reid, Chief
Operating Reactors Branch #4
Division of Operating Reactors

Enclosures:
1. Appraisal of Changes to the Environmental Technical Specifications (Appendix B) for Three Mile Island Nuclear Station Unit No. 1
2. Amendment No. 18
3. Federal Register Notice

cc w/enclosures: See next page

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DATE	6/4/76	6/4/76	6/9/76	7/23/76 ~~6/ /76~~	9/9/76

Technical Specifications

We have reviewed the Technical Specification changes submitted by MetEd for operation with the cycle 2 core and find them to be acceptable.

ECCS Analysis

On December 27, 1974, the Atomic Energy Commission issued an Order for Modification of License implementing the requirements of 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Reactors." One of the requirements of the Order was that a re-evaluation of ECCS cooling performance calculated in accordance with an acceptable evaluation model which conforms with the provisions of 10 CFR 50.46 shall be submitted. The Order also required that the evaluation shall be accompanied by such proposed changes in the Technical Specifications or license amendment as may be necessary to implement the evaluation results. As required by our Order of December 27, 1974, MetEd has submitted an ECCS re-evaluation and related Technical Specification changes. The re-evaluation and Technical Specifications were submitted by letter dated August 8, 1975, using the B&W ECCS evaluation model as described in BAW-10104.

The background of our review of the B&W ECCS evaluation model and its application to TMI-1 is described in our Safety Evaluation Report (SER) for TMI-1 dated December 27, 1974, issued in connection with the Order for Modification of License. The bases for acceptance of the principal portions of the evaluation model are set forth in our Status Report of October 1974 and the Supplement to the Status Report of November 1974 which are referenced in the December 27, 1974 SER. The December 27, 1974 SER also describes the various changes required in the earlier version of the B&W model. Together, the December 27, 1974 SER and the Status Report and its Supplement describe an acceptable ECCS evaluation model and the basis for our acceptance of the model. The TMI-1 ECCS evaluation which is covered by this safety evaluation report properly conforms to the accepted model. MetEd's July 9, 1975, submittal contains documentation by reference to B&W Topical Reports of the revised ECCS model (with the modifications described in our December 27, 1974 SER) and a generic break spectrum appropriate to TMI-1 (BAW-10103 and BAW-10104).

The generic analysis in BAW-10103 identified the worst break size as the 8.55 ft^2 double-ended cold leg break at the pump discharge with a C_D=1.0. The table below summarizes the results of the LOCA limit analyses which determine the allowable linear heat rate limits as a function of elevation in the core for TMI-1.

Elevation (ft)	LOCA Limit (kw/ft)	Peak Cladding Temperature (°F)	Max. Local Oxidation (%)	Time of Rupture (sec)
2	15.5	2002	3.92	12.25
4	16.6	2136	4.59	13.01
6	18.0	2146	4.46	15.55
8	17.0	2110	5.19	15.01
10*	16.0	1931	2.93	39.20

*See discussion below

The maximum core-wide metal-water reaction for TMI-1 was calculated to be 0.557 percent, a value which is below the allowable limit of 1 percent.

As shown in the tabulation, the calculated values for the peak clad temperature and local metal-water reaction were below the allowable limits specified in 10 CFR 50.46 of 2200°F and 17 percent, respectively. BAW-10103 has also shown that the core geometry remains amenable to cooling and that long-term core cooling can be established.

We noted during our review of BAW-10103 that the LOCA limit calculation at the 10-foot evaluation in the core showed reflood rates below 1 inch/second at 251 seconds into the accident (Section 7.2.5). Appendix K to 10 CFR 50.46 requires that when reflood rates are less than 1 inch/second, heat transfer calculations shall be based on the assumption that cooling is only by steam, and shall take into account any flow blockage calculated to occur as a result of cladding swelling or rupture as such blockage might affect both local steam flow and heat transfer. As indicated by the staff in our Status Report and Supplement, dated October 1974 and November 1974 respectively, a steam cooling model for reflood rates less than 1 inch/second was not submitted by B&W for staff review. The steam cooling model submitted by B&W in BAW-10103 is therefore considered to be a proposed model change requiring further staff review. Accordingly, B&W was informed that until the proposed steam cooling model is reviewed, the heat transfer calculation at the 10-foot elevation during the period of steam cooling specified in BAW-10103 must be further justified. In lieu of using their proposed steam cooling model, B&W has submitted the results of calculations at the 10-foot elevation using adiabatic heatup during the steam cooling period, where this period is defined by B&W as the time when the reflood rate first goes below 1 inch/second to the time that it is predicted that the 10-foot elevation is covered by water. The new calculated peak cladding temperature, local metal-water reaction and core-wide metal-water reaction at the 10-foot elevation are 1946°F, 3.02%, and .647%, respectively. These values remain below the allowable limits of 10 CFR 50.46 and have been calculated in an acceptable manner. Until we have accepted a steam cooling model, these values will serve as the LOCA results for TMI-1 at the 10-foot elevation.

Our review of plant-specific assumptions discussed in the following paragraphs regarding the TMI-1 analyses addressed the areas of single failure criterion, long term boron concentration, potential submerged equipment, partial loop operation, ECCS valve interlocks, and the containment pressure calculation.

Single Failure Criterion

Appendix K to 10 CFR 50 of the Commission's regulations requires that the combination of ECCS subsystems to be assumed operative shall be those available after the most damaging single failure of ECCS equipment has occurred. MetEd has assumed all containment cooling

systems operating to minimize containment pressure and has separately assumed the loss of one diesel and therefore the loss of one electrical safeguards bus to minimize ECCS cooling. We concluded in our Status Report dated October 1974 that the application of the single failure criterion was to be confirmed during subsequent plant reviews. This has been done.

A review of TMI-1 piping and instrumentation diagrams indicated that the spurious actuation of certain motor-operated valves could affect the appropriate single failure assumptions. A spurious actuation of core flooding tank (CFT) vent valves CF-V3A or CF-V3B would result in a decrease in CFT pressure. Since CFT pressure is important in mitigating the consequences of a LOCA, we require that these normally closed motor-operated valves (CF-V3A and CF-V3B) have their power disconnected and associated breakers locked open except when adjusting core flooding tank pressure. To further minimize the potential for a water hammer due to the discharge of ECC water into a dry line and to ensure that air pockets have not formed in the ECCS lines and pumps we require the venting of High and Low Pressure Injection pump casings and system high points during each refueling outage. MetEd has committed to perform these required actions and to verify the High and Low Pressure Injection pumps and lines are filled with water prior to cycle 2 power operation. We find this acceptable.

Manually-Controlled Electrically-Operated Valves

As requested, MetEd has submitted a single failure analysis for manually-controlled, electrically-operated ECCS valves. This analysis (TMI-1 FSAR Chapter 6 and MetEd letter dated February 11, 1976) demonstrates that no credible single failure or operator error affecting any manually-controlled, electrically-operated ECCS valve could adversely affect ECCS performance. We find this acceptable.

Submerged Valves

MetEd has identified the following valves as becoming submerged when the entire contents of the BWST are discharged into the reactor containment building:

Safety

MU-V2A,B	Letdown cooler containment isolation
IC-V2	Intermediate cooler containment isolation

Non-safety

IC-V1A,B	Letdown cooler shell side inlet isolation
IC-V20	RC drain tank cooler outlet isolation
MU-V1A,B	Letdown cooler tube side inlet isolation
WDL-V302	RC Tank recirculation
WDL-V305	RC Tank recirculation

Only three of the above valves (IC-V2, MU-V2A and MU-V2B) are Engineered Safety Feature valves. These three containment isolation valves will have performed their safety function prior to becoming submerged and therefore, their submergence will not affect any ECCS function. It has been determined that if power were maintained or inadvertently applied, during or after submergence, to any one of the above valves, there would be no adverse affects on the remainder of the electrical system thus, we find this evaluation acceptable.

Electrical Independence

MetEd in a letter dated April 19, 1975, identified the worst single failure which could occur as the loss of one diesel resulting in the failure of the IC Engineered Safeguards Valve 480V control center. In such case power to open valves DH-V1, DH-V2, and DH-V3 in the boron control primary flow path or RC-V4 in the alternate flow path would not be immediately available. Minimum requirements are that either RC-V4 be operable or DH-V1,2 and 3 be operable. In this case the procedure, as detailed in MetEd's April 8, 1976 submittal, would be used to open RC-V4 to establish long-term flow for post LOCA boron control.

Since a minimum of 30-days would be available to accomplish this emergency action, no modifications are necessary. We concur in this evaluation and find the above referenced procedure acceptable.

Interlocks

Decay heat drop line valves DH-V1 and DH-V2 were reviewed to ensure that they are properly interlocked to prevent opening while the reactor coolant system is pressurized. DH-V1 is interlocked to RC3A-PS2 through engineered safeguard actuation channel "A" and DH-V2 is interlocked to RC3A-PS5 through engineered safeguard actuation channel "B". These valves have independent and diverse interlocks to prevent them from being opened unless the reactor coolant system pressure is below 400 psi. We find this design acceptable.

Containment Pressure

The ECCS containment pressure calculations for TMI-1 were done generically by B&W for reactors of this type as described in BAW-10103. We reviewed B&W's ECCS evaluation model (Status Report and Supplement dated October 1974 and November 1974, respectively) and concluded that B&W's containment pressure model was acceptable for ECCS evaluation. We required, however, that justification of the plant-dependent input parameters used in the analysis be submitted for our review of each plant. A containment pressure calculation specific to TMI-1 was contained in MetEd's August 8, 1975 submittal.

Justification for the containment input data was submitted for TMI-1 on October 23, 1975. This justification includes a comparison of the actual containment parameters for TMI-1 with those assumed in BAW-10103. MetEd has re-evaluated the containment net-free volume, the passive heat sinks, and operation of the containment heat-removal systems with regard to the conservatism for the ECCS analysis. This evaluation was based on as-built design information. The containment heat removal systems were assumed to operate at their maximum capacities, and minimum operational values for the spray water and service water temperature were assumed. The containment pressure analysis by B&W in BAW-10103 was demonstrated to be conservative for TMI-1.

We have concluded that the plant-dependent information used for the ECCS containment pressure analysis for TMI-1 is reasonably conservative, and therefore, the calculated containment pressures are in accordance with Appendix K to 10 CFR 50 of the Commission's regulations.

Long-Term Boron Concentration

We have reviewed the proposed procedures and the systems designed for preventing excessive boric acid buildups in the reactor vessel during the long-term cooling period after a LOCA. MetEd implemented procedures for TMI-1 which would allow adequate boron dilution during the long term and which will employ a concept similar to that described in BAW-10103. We noted that a failure of a diesel will affect each of the proposed dilution modes. MetEd has indicated that the controllers for all the pertinent valves are located in the auxiliary building, thus enabling the operator to connect power to the valves with jumper cables. Based on calculations by B&W, which we have found acceptable, over 30 days are available, taking credit for natural circulation through the vent valves, before forced circulation is necessary; therefore, the MetEd's backup procedure to a power failure is acceptable.

As initially proposed by MetEd in their letter dated May 27, 1975, dilution Mode 1 was to be first attempted to establish suction from the reactor vessel outlet pipe through the decay heat drop line with one LPI string. It is our position that Mode 1 should not be attempted as a method to control boron concentration in the core during long-term cooling. As stated in MetEd's letter of May 27, 1975 and BAW-10103 the success of Mode 1 is not ensured because of the possibility of gas or steam entrainment in the decay heat suction nozzle. Such gas or steam entrainment can result in severe damage to the decay heat removal pump. Long-term heat removal requirements can exist for long durations (days or months) after the accident and continuous operation of one train of the decay heat removal system is required. In the event of equipment malfunction in this train, no method is available to remove the decay heat if the other train has been previously damaged. Therefore, since initiation of Mode 1 is not allowed, Modes 2 and 3 (as proposed in MetEd's May 27, 1975 letter) must be single failure proof in combination.

Mode 2, using the hot leg drain approach is satisfactory as one of the two methods of preventing excessive concentration of boric acid in the core. Mode 3 is a backup to Mode 2 and employs hot leg injection through the pressurizer. This backup method required installation of a 1 1/2 inch check valve in the decay heat pressurizer auxiliary spray line and upgrading of the motor operator of valve RC-V4 so that it is now qualified for the post LOCA environment. MetEd has made the necessary modifications during the present refueling outage. We find this to be acceptable.

We have reviewed the operating procedures associated with this proposal and conclude that these procedures are acceptable.

Partial Loop Analyses

To allow an operating configuration with less than four reactor coolant pumps on the line (partial loop), we required an analysis of the predicted consequences of a LOCA occurring during the proposed partial loop operating mode(s). MetEd submitted an analysis for partial loop operation with one idle reactor coolant pump (three pumps operating) in their August 8, 1975 submittal. This analysis assumes that the worst break was the 8.55 ft^2 guillotine at the reactor coolant pump discharge, with C_D =1.0 as reported in BAW-10103. The worst break selected was located in the active leg of the partially idle loop. Placing the break at the discharge of the pump in an active cold leg of the partially idle loop (instead of at the discharge of the pump in an active cold leg of the fully active loop) yields the most degraded positive flow through the core during the first half of the blowdown and results in higher cladding temperatures. The maximum cladding temperature for the one-idle-pump mode of operation was 1766°F. MetEd's analysis used an initial pin pressure of 1600 psi. The results of a new analysis were submitted to reflect a more appropriate value of initial pin pressure. As was demonstrated in the time-in-life sensitivity study in BAW-10103, the worst pin pressure for this analysis is 760 psi. The maximum cladding temperature for this analysis is 1784°F, a value which is within the criterion of 10 CFR 50.46. This analysis may be used to support MetEd's proposed operation with one idle reactor coolant pump.

Since an analysis of ECCS cooling performance with one idle reactor coolant pump in each loop has not been submitted, power operation in this configuration will be limited by Technical Specifications to 24 hours. Single loop operation (i.e., operation with two idle pumps in one loop) is prohibited.

Technical Specifications

We have reviewed the proposed Technical Specification changes to assure that operation of TMI-1 during cycle 2 will be within the limits imposed by the Final Acceptance Criteria (FAC) for ECCS performance. Changes in the allowable heat generation rates as a function of height in the core have been accommodated by revision of the power-flow-imbalance specifications (Figures 3.5 2G-I). Only minor change has been made by MetEd in the rod position limit specification.

The reduction in allowable heat generation rate (kw/ft) in the lower half of the core has been accommodated by reducing the allowable negative imbalance at full power by approximately 4%. The increase in allowable heat generation rate at the top of the core (as compared to that permitted by the Interim Acceptance Criterion) permits relaxing the positive axial imbalance by approximately 3%. On the basis of our review, we find the technical specification changes as proposed to be acceptable.

We have completed our review of the TMI-1 ECCS performance re-analyses and have concluded:

(a) The proposed Technical Specifications are based on a LOCA analysis performed in accordance with Appendix K to 10 CFR 50.

(b) The ECCS minimum containment pressure calculations were performed in accordance with Appendix K to 10 CFR 50.

(c) The single failure criterion will be satisfied provided that the requirements as specified in this Safety Evaluation Report are implemented.

(d) The modified procedures for long-term cooling after a LOCA are acceptable. The necessary plant modifications to provide assurance that the ECCS can be operated in a manner which would prevent excessive boric acid concentration from occurring have been made during the present refueling outage.

(e) The proposed mode of reactor operation with one idle reactor coolant pump is supported by a LOCA analysis performed in accordance with Appendix K to 10 CFR 50. Operation with one idle pump in each loop is restricted to 24 hours.

We have completed our evaluation of the TMI-1 cycle 2 reload application and conclude that the licensee has performed the required analyses and has shown that operation of the cycle 2 core will be within applicable fuel design and performance criteria. In addition, we conclude that the licensee's proposed Technical Specification changes meet the Final Acceptance Criteria based on an acceptable ECCS model conforming to the requirements of 10 CFR 50.46 and that the restrictions imposed on the facility by the Commission's December 27, 1974 Order for Modification of License should be terminated and replaced by the limitations established in accordance with 10 CFR 50.46.

Conclusion

We have concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (2) 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 the health and safety of the public.

Dated: MAY 18 1976