SAFETY EVALUATION REPORT

THREE MILE ISLAND, UNIT 1

DOCKET NO. 50-289

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Introduction

On February 11, 1976 Metropolitan Edison Company submitted proposed changes to the Three Mile Island Nuclear Station, Unit 1 Technical Specifications [1]. The purpose of this submittal was to seek approval to operate with the cycle 2 core reload which is scheduled to be installed during the period late February to mid April 1976. B&W Report "Three Mile Island, Unit 1, Cycle 2 Reload Report"[1] was submitted for review along with the proposed Technical Specification changes.

General Description

The cycle 2 core consists of 177 fuel assemblies, each of which is a 15 x 15 array with 208 fuel rods, 16 control rod guide tubes and one in-core instrument tube. There are 61 twice burned Batch 2 fuel assemblies, 60 once burned Batch 3 assemblies and 56 fresh Batch 4 assemblies in the cycle 2 core. All three batches are B&W fuel and are mechanically and hydraulically similar. There are slight differences in enrichment and fuel density.

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 Three Mile Island 1. With the reactor operating 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 are reported in reference 5. 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, Metropolitan Edison Company has used a flow rate of 106.5% of design.

Metropolitan Edison Company has committed to verify the flow rate for Three Mile Island 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 reference 5.

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

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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. Unlike the previously mentioned overpower trip and accident analysis, the flux/flow trip setpoint includes credit for the vent valve penalty. Thus, for the pump coast down analysis the 4.6% penalty due to vent valves has been eliminated. 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 January 30 a letter was sent from the staff to the Metropolitan Edison Company [7] stating that B&W report, "B&W Operating Experience of Reactor Internals Vent Valves" has been reviewed and that sufficient evidence has been presented to assure that the vent valves will remain closed during normal operation. Based on this conclusion the flow penalty can be eliminated at the request of the utility. However, the corresponding modifications to the Technical Specifications would be reviewed by the staff prior to implementation.

Nuclear Analysis

The licensee has provided values for core physics parameters for the Unit 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 the licensee'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 in the TMI-1 cycle 2 reload report were examined to insure that the cycle 2 reload core is thermally and hydraulically conservative and of the same design and manufacture as the cycle 1 core, and also that the reactivity coefficients and other input data is the same as, or is bounded by previous analyses.

Fuel Rod Bow Evaluation

The effect of rod bowing on DNBR was considered. A review of reference 4 indicated that the licensee calculated a 1.6% peaking penalty due to rod bowing. The effect of the rod bow penalty on the limits for normal operation as provided in reference has been found to be within the conservatism of the current limits.(6) The design basis values for linear heat rate of 5.80 Kw/ft and power spike of 1.022 are greater than the actual limiting values of 5.73 Kw/ft and 1.018, respectively. Thus adequate margin is provided to absorb the rod bow penalty.

Technical Specifications

We have reviewed the proposed Technical Specifications in Reference 1 and find them to be acceptable.

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Fuel and Mechanical Design

Creep collapse calculations were performed by the licensee 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 the 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, linear thermal output and 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 licensee's fuel thermal analysis acceptable.

The batch 4 fuel assemblies are not new in concept and they do not utilize different component meterials. 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.

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ECCS Analysis

On December 27, 1974, the Atomic Energy Commission issued an Order for Modification of License (Reference 9) implementing the requirements of 10 CFR 50.46, "Acceptance Criteria and Emergency Core Cooling Systems for Light Water Nuclear Power Reactors." One of the requirements of the Order was that the licensee shall submit 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. The Order also required that the evaluation shall be accompanied by such proposed changes in Technical Specifications or license amendement as may be necessary to implement the evaluation results. As required by our Order of December 27, 1974, Metropolitan Edison Company (the licensee) has submitted an ECCS re-evaluation and related Technical Specifications. The re-evaluation and Technical Specification were submitted in References 8 and 14 using the B&W ECCS evaluation model as described in Reference 14 and discussed in Section 2.0 of this Safety Evaluation Report. Also discussed in Section 2.0 are the results of a staff review of the plant-specific areas of single failures, long-term boron concentration, potential submerged equipment, partial loop operation, ECCS valve interlocks, and the containment pressure calculation. Section 3.0 provides the results of the staff review of the proposed Three Mile Island (TMI-1) Technical Specifications, and Sections 4.0 and 5.0 present staff conclusions and references, respectively.

The background of the staff review of the B&W ECCS evaluation model and its application to TMI-1 is described in the staff SER for facility 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 the staff's Status Report of October 1974 (Reference 12) and the Supplement to the Status Report of November 1974 (Reference 13) 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 the staff's acceptance of the model. The TMI-1 ECCS evaluation which is covered by this safety evaluation report properly conforms to the accepted model. The licensee's July 9, 1975 submittal (Reference 8) 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 (Reference 14 and 15, respectively).

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 in text

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.

The staff noted during its review of BAW-10103 that the LOCA limit calculation at the 10-foot elevation 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

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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 References 12 and 13, a steam cooling model for reflood rates less than 1 inch/second was not 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 REFLOOD predicts the 10-foot elevation is covered by solid 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 are acceptable to the staff. Until a steam cooling model has been accepted by the staff, these values will serve as the LOCA results for Three Mile Island 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.

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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. Babcock and Wilcox has assumed all containment cooling systems operating to minimize containment pressure and has separately assumed the loss of one diesel to minimize ECCS cooling. We concluded in Reference 12 that the application of the single failure criterion was to be confirmed during subsequent plant reviews.

Venting Requirements

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. Accordingly, we require addition of a Technical Specification requiring these normally closed motor-operated valves to have their power disconnected and associated breakers locked open. To further minimize the potential for a water hammer due to the discharge of ECC water into a dry line, the staff requires that valve MU-V14A or MU-V14B be left in the open position during normal operation (depending on normal makeup alignment). This maintains at least one ECCS train filled with a continual supply of water from the BWST due to the available static head built into the design. Such a configuration will also eliminate the need for one automatic safety action in the event of a LOCA; that is, the automatic opening of this valve to provide ECC water to the LPI and Building Spray pumps. In addition, Metropolitan Edison Company will be required to adopt a Technical Specification whereby a monthly procedure of opening the existing ECCS pump casing and high point vents in ECCS lines will be performed to ensure that no air pockets have formed. Such venting must also be performed prior to any ECCS flow tests.

Manually-Controlled Electrically-Operated Valves

As requested, the licensee has submitted a single failure analysis for manually-controlled, electrically-operated ECCS valves. This analysis (TMI-1 FSAR Chapter 6 and MET ED 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

The licensee 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 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

The licensee in a letter dated April 19, 1975, (Reference 16), identified the worst single failure which could occur as 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 following procedure would be used to open RC-V4 to establish long-term flow for post LOCA boron control:

(1) Open breaker for RC-V4 at the IC ES Valve Control Center.

(2) Verify MU-V2A is in its closed position and open the breaker for MU-V2A at the 1B ES Valve Control Center.

(3) At penetration 315E, lift the power and control cables from the MU-V2A motor controller.

(4) At penetration 317E, which is located about 10 feet from penetration 315E, lift the power and control cables from RC-V4 motor control center.

(5) Using jumpers, connect the MU-V2A motor controller power and control cables removed from penetration 315E to penetration 317E connections for RC-V4.

(6) Utilizing MU-V2A motor controller open RC-V4.

Since a minimum of 30-days would be available to accomplish the above emergency action, no modifications are necessary. We concur in this evaluation and find the above 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 Three Mile Island Unit 1 were done generically by B&W for reactors of this type as described in Reference 15. The NRC staff reviewed B&W's ECCS evaluation model and

and published the results of this review in References 12 and 13. We 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.

Justification for the containment input data were submitted for TMI-1 on October 23, 1975 (Reference 17). This justification includes a comparison of the actual containment parameters for TMI-1 with those assumed in Reference 15. Metropolitan Edison 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 drawings. 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 Three Mile Island Unit 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

The NRC staff has 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. The Licensee has agreed to implement 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. The staff has noted that a failure of a diesel will affect each of the proposed dilution modes. The Licensee 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, over 30 days is available taken credit for natural circulation through the vent valves before forced circulation is necessary; therefore, the Licensee's backup procedure to a power failure is acceptable.

As initially proposed by Metropolitan Edison Company, 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 the staff's position that Mode1should not be attempted as a method to control boron concentration in the core during long-term cooling. References 10 and 15 state that 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 Reference 10) 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 concentraiton 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. The licensee has made the necessary modifications during the present refueling outage. The staff finds this to be acceptable.

The staff has reviewed the operating procedures associated with this proposal and concludes that this proposal is acceptable.

Partial Loop Analyses

To allow an operating configuration with less than four reactor coolant pumps on the line (partial loop), the staff required an analysis of the predicted consequences of a LOCA occurring during the proposed partial loop operating mode(s). The Licensee submitted an analysis for partial loop operation with one idle reactor coolant pump (three pumps operating) in Reference 14. This analysis concluded that the worst break was the 8.55 ft^2 guillotine at the reactor coolant pump discharge, with C_D =1.0. 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) yeilds the upper degrated 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. A staff review of all input assumptions and conclusions indicates analysis in Reference 14 used an initial pin pressure of 1600 psi. As was demonstrated in the time-in-life sensitivity study in Reference 15, the worst pin pressure for this analysis should have been 760 psi. The maximum cladding temperature for this pin pressure must be shown to be within the criterion of 10 CFR 50.46. When this is done, the analysis may be used to support the Licensee's proposed operation with one idle reactor coolant pump.

Since an analysis of ECCS cooling performance with one idle reactor coolant pump <u>in each loop</u> has not been submitted, power operation in this configuration must be limited by Technical Specifications to 24 hours. Single loop operation (i.e., operation with two idle pumps in one loop) is prohibited without prior approval of the Commission.

TECHNICAL SPECIFICATIONS

We have reviewed the proposed Technical Specification in Ref. 8 and 11 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 26-I). Only minor change has been made by Metropolitan Edison Company 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 coare (as compared to that permitted by the Interim Acceptance Criterion) permits relaxing the positive axail imbalance by approximately 3%. On the basis of our review, we find the Technical Specification changes proposed in References 8 and 11 to be acceptable.

CONCLUSIONS

The staff has completed its review of the Three Mile Island Unit 1, ECCS performance re-analyses and has 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 modifications specified in that subsection of this Safety Evaluation Report are implemented.

d. The proposed procedures for long-term cooling after a LOCA are acceptable to the staff. The necessary 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 anlaysis performed in accordance with Appendix K to 10 CFR 50. Operation with one idle pump in each loop is restricted to 24 hours. Requests for single loop operation will be reviewed on a case-by-case basis.