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

November 14, 1980

Docket No. 50-320

Mr. R. C. Arnold
Senior Vice President
Metropolitan Edison Company
100 Interpace Parkway
Parsippany, New Jersey 07054

Dear Mr. Arnold:

The Nuclear Regulatory Commission has issued the enclosed Amendment of Order for the Three Mile Island Nuclear Station, Unit 2. This Amendment of Order deletes the operability requirements for certain balance of plant systems and adds operability requirements for the Mini Decay Heat Removal System (MDHRS). The deleted requirements had been imposed by the Order of the Director of the Office of Nuclear Reactor Regulation on February 11, 1980. These changes are being made in response to your requests of July 31, 1980 (TLL 372) and August 5, 1980 (TLL 382). We have made certain changes in the operability requirements proposed in your submittals to satisfy our requirements. These changes have been discussed with and agreed to by members of your staff.

The operability requirements of this Amendment of Order are effective immediately.

Copies of the related Safety Evaluation and revised pages for the proposed Technical Specifications and for the Recovery Operations Plan are enclosed.

Sincerely,

Bernard J. Snyder

Bernard J. Snyder, Program Director
TMI Program Office
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment of Order
2. Safety Evaluation
3. Proposed Technical Specifications
   pages 3.1-1, 3.1-2, 3.4-1, 3.7-1,
   3.7-2, 3.8-4, B 3/4 4-1 and B 3/4 7-1
4. Recovery Operations Plan
   pages 4.1-1, 4.1-2, 4.7-1 and 4.7-2

cc w/encl:
See Next Page

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UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

In the Matter of	)	
	)	
METROPOLITAN EDISON COMPANY, et al.	)	Docket No. 50-320 OLA
	)	
(Three Mile Island Nuclear Station, Unit 2)	)	

AMENDMENT OF ORDER

I.

Metropolitan Edison Company, Jersey Central Power and Light Company and Pennsylvania Electric Company (collectively, the Licensee) are the holders of Facility Operating License No. DPR-73, which had authorized operation of the Three Mile Island Nuclear Station, Unit 2 (TMI-2) at power levels up to 2772 megawatts thermal. The facility, which is located in Londonderry Township, Dauphin County, Pennsylvania, is a pressured water reactor previously used for the commercial generation of electricity.

By Order for Modification of License, dated July 20, 1979, the Licensee's authority to operate the facility was suspended and the Licensee's authority was limited to maintenance of the facility in the present shutdown cooling mode (44 F.R. 45271). By further Order of the Director, Office of Nuclear Reactor Regulation, dated February 11, 1980, a new set of formal license requirements was imposed to reflect the post-accident condition of the

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facility and to assure the continued maintenance of the current safe, stable, long-term cooling condition of the facility (45 F.R. 11282). These requirements, in the form of proposed Technical Specifications, would modify the facility operating license so as to:

(1) define operating parameters for the current safe, stable, long-term cooling mode for the facility (defined as the recovery mode), and delete all other permissible operating modes so as to assure that operation of the facility in other than the stable shutdown condition of the recovery mode is precluded;

(2) impose functional, operability, redundancy and surveillance requirements as well as safety limits and limiting conditions with regard to those structures, systems, equipment and components necessary to maintain the facility in the current safe, stable shutdown condition and to cope with foreseeable off-normal conditions;

(3) prohibit venting or purging or other treatment of [the approximately 57,000 curies of krypton-85 in] the reactor building atmosphere, the discharge of water decontaminated by EPICOR-II system, and the treatment and disposal of high-level radioactively contaminated water in the reactor building, until each of these activities has been approved by the NRC,1/ consistent with the Commission's Statement of Policy and Notice of Intent to Prepare a Programmatic Environmental Impact Statement (44 F.R. 67738).

On the basis of the public health, safety, and interest, the requirements of the proposed Technical Specifications, attached to the February 11, 1980 Order, were made effective immediately. Under the terms of the Order, since requests for a hearing are pending before an Atomic Safety and Licensing Board, the proposed formal license amendment incorporating these proposed Technical

1/ By Memorandum and Order, dated June 12, 1980, the Commission gave the approval contemplated by this restriction insofar as necessary for the Licensee to conduct a purging of the TMI-2 containment in accordance with procedures approved by the NRC. CLI-80-25. This activity was completed on July 11, 1980.

Specifications will become effective, in the event a hearing is granted, on the date specified in an order made following the hearing or, upon other final disposition of such proceeding.

II.

Following the March 28, 1979 accident at TMI-2, it became necessary in late April 1979 to alter the preferred cooling mode for the reactor by a transition from use of forced circulation by the reactor coolant system pumps to natural convection circulation.

By letters dated July 31, 1980 (Reference 1) and August 5, 1980 (Reference 2), the Metropolitan Edison Company (licensee) proposed changes to the Recovery Mode technical specifications for Three Mile Island Unit 2 (TMI-2) providing for the implementation of the Mini Decay Heat Removal System (MDHRS) for long-term core cooling. Although several modes for removing decay heat would be available, the MDHRS would provide a forced flow system for removing decay heat from the TMI-2 reactor fuel. Accordingly, the proposed changes would impose operability requirements for the MDHRS and would also delete the operability requirements for certain Balance of Plant (BOP) systems which have been used since the March 28, 1979, accident, but would no longer be required, for removing the decay heat. The operability requirements for these BOP systems had been imposed by the Order of the Director of the

Office of Nuclear Reactor Regulation on February 11, 1980. (45 F.R. 11282) in the form of proposed Technical Specifications.

The proposed change would make available a newly installed MDHRS to remove decay heat rather than the present method which accomplishes this function by using the "A" steam generator in a steaming mode to the condenser. The licensee's proposal required certain modifications to meet our requirements. With the incorporation of these staff required modifications, we have found the proposal to be an acceptable method for removing the decay heat and have therefore granted the licensee's request to modify the method used for long term core cooling.

The TMI-2 Reactor Coolant System (RCS) is currently operating in a natural circulation heat removal mode with heat rejection from it being accomplished by both loss to ambient (reactor atmosphere and sump water) and through the "A" steam generator. The reactor building is in turn being cooled by the reactor building fan coolers while the "A" steam generator is steaming to the condenser through the turbine bypass valve. This mode of core cooling has been in effect since late April 1979. With passage of time and the associated reduction of decay heat generation rate (presently approximately 75 kw), the natural circulation flow has changed from continuous to cyclic with increasing intervals between the cyclic flow "burps".

Heat rejection through the "A" steam generator by steaming to the condenser requires the operation of several major BOP systems including: circulating water system, main steam system and the "A" steam generator, condensate and feedwater systems, main condenser and package boiler. Conversion to and use of the MDHRS for core cooling would simplify the plant operations since use of the MDHRS would eliminate the need for operating the previously noted BOP systems.

The MDHRS is classified as a nonsafety grade system but it is designed and installed to seismic Category I requirements up to and including the second isolation valve in its supply and discharge lines. The balance of the system is designed and installed to Operating Basis Earthquake requirements. The MDHRS takes suction from the "B" loop of the Decay Heat System (DH) outlet from the reactor vessel via a connection to the Alternate Decay Heat Removal System (ADH). After passing through one of the MDHRS's parallel heat exchangers and pumps, the reactor coolant is returned to the reactor coolant system through the "B" Core Flooding injection nozzle via a connection to the ADH and DH systems. Connection of the ADH to the DH was evaluated and approved in NUREG-0557. The MDHRS is sized such that one pump and one heat exchanger (two of each are installed) could remove up to approximately 1 MW of decay heat. Therefore the MDHRS has more than adequate cooling capacity for removing the present and future decay heat loads. The MDHRS would be cooled by the Nuclear Services Closed Cycle Cooling System which is required to be operable by proposed Technical Specification 3.7.3.1. The power supply for the MDHRS pumps and motor operated valves is from redundant Class 1E busses which would be manually loaded on the Class 1E diesel generators in the event of a loss of off-site power.

In the event the MDHRS is not used or becomes inoperable, backup cooling modes are available for removing the decay heat from the RCS. These backup cooling modes include the long term "B" steam generator cooling system and "Loss to Ambient." The long term "B" steam generator cooling system has been previously evaluated and its operability would continue to be required by proposed Technical Specification 3.7.1. The NRC staff has reviewed the licensee's results of the "Loss to Ambient" cooling mode and has performed an independent analysis, the results of which are in agreement with the licensee's conclusion. Therefore, we have concluded that any one of these cooling methods provides an acceptable means for long term cooling of the reactor core. The staff's overall evaluation of the MDHRS is presented in the concurrently issued Safety Evaluation Report (SER).

The MDHRS has a design pressure of 235 psig. Therefore, consideration was given to possible sources of overpressurization of the MDHRS. Three potential sources of MDHRS overpressurization were identified. These sources were: (1) Makeup pump operation with MDHRS in operation, (2) Pressurizer heater operation and (3) Malfunction of the Standby Pressure Control (SPC) System.

To preclude operation of a makeup pump during operation of the MDHRS, the licensee proposed to delete the requirement for an operable makeup pump from the proposed Technical Specifications but to retain the option to operate the pump for certain operations (e.g., degassing). The licensee further stated that the electrical power supply circuit breakers for the makeup pumps would be "racked out" when valve DH-V1 or DH-V171 is open. Since operation of a makeup pump may be required in one or more of the backup cooling modes or for degassing operations, we will retain a requirement for its operability in proposed Technical Specification 3.1.1.1. However, to provide assurance that the MDHRS will not be overpressurized due to operation of a makeup pump, we propose to add a requirement to proposed Technical Specifications 3.1.1.1 that all makeup pumps be made inoperable when valve DH-V1 or DH-V171 is open by "racking out" their electrical power supply circuit breakers. We would also add a surveillance requirement to the Recovery Operations Plan to periodically verify that these breakers are "racked out." These actions provide assurance that the MDHRS would not be overpressurized due to operation of the makeup pumps.

Operation of the pressurizer heaters while operating the RCS in a water solid mode with the MDHRS in operation creates the potential for overpressurization of the MDHRS due to volumetric expansion of the reactor coolant as a result of heat input to the reactor coolant. The licensee has calculated that the electrical energization of all the pressurizer heaters (1638 kw) would result in a volumetric expansion on the reactor coolant which would require a compensating relief capacity of 8.6 gpm. The MDHRS has an installed relief valve capacity of 53.5 gpm. The NRC staff has reviewed the licensee's result of this potential overpressurization event and has performed an independent check, the results of which are in agreement with the licensee's conclusion. Therefore, we agree that operation of the pressurizer heaters while operating the MDHRS with a water solid RCS would not result in overpressurization of the MDHRS.

Overpressurization of the MDHRS due to a malfunction of the SPC system has been precluded by reducing the SPC in-service nitrogen bank pressure to a new operating range of 225 to 400 psig and by the installation of a SPC systems pressure relief valve (SPC-R14) which has been set to provide overpressure relief if the SPC system pressure exceeds 125 psig which is substantially below the MDHRS design pressure of 235 psig. The change in the nitrogen bank pressure was approved on July 25, 1980 (Reference 3).

The licensee also proposed deleting from proposed Technical Specification 3.1.1.1 the requirements for a boric acid storage system and an associated flow path to the RCS. We have reviewed this proposed change and since redundant boron injection flow paths from the BWST to the RCS via the makeup pump and decay heat removal pump exist, we find the proposed change acceptable.

The licensee's proposed Technical Specification for the MDHRS would require only one operable MDHRS pump and heat exchanger with an action statement providing instructions to be taken in the event of their inoperability. Our position is that this Technical Specification should require the operability of both MDHRS pumps and heat exchangers and the associated flow path and that applicable action statements should be supplied dealing with the inoperability of the various components in the MDHRS. The staff's position is consistent with the operability requirements for similar systems (e.g., proposed Technical Specifications 3.7.2.1, 3.7.3.1 and 3.7.4.1) in which the redundant systems are required operable and action statements are provided for when one or both systems are inoperable. Unless both MDHRS pumps are required operable and periodically demonstrated so in accordance with applicable surveillance requirements, there is no assurance of the operability of the redundant pump should its use be required for any reason. We have therefore modified the Technical Specification proposed for the MDHRS to be in accordance with our position. The licensee has agreed with our position on this matter. We have also added appropriate surveillance requirements to the Operations Recovery Plan to periodically demonstrate the operability of the MDHRS.

The Licensee also proposed to delete from proposed Technical Specification 3.8.2.1, the operability requirements for several electrical power busses. Its basis for proposing to delete these requirements was that they supplied electrical power to the various BOP systems which were proposed for deletion from the proposed Technical Specifications upon incorporation of the MDHRS. However, in our review of these proposed changes, we determined that four of the busses proposed for deletion (480 volt Busses 2-35, 2-36, 2-45 and 2-46) also supplied electrical power to the auxiliary building and the fuel handling building air cleanup systems which are in turn required operable per proposed Technical Sepcification 3.9.12. Therefore, to ensure the electrical power supply for these air cleanup systems, we have retained the operability requirements for these four busses in proposed Technical Specification 3.8.2.1. The licensee has agreed that the operability requirements for these four busses should be retained in proposed Technical Specification 3.8.2.1. We agree that the operability requirements for the other busses can be deleted from the proposed Technical Specifications as proposed by the licensee since the other busses proposed for deletion do not supply electrical power to any systems required for maintaining the plant in its safe shutdown condition.

Based on the staff's review of accident considerations, as presented in the SER, the staff has concluded that use of the MDHRS does not increase the probablity or consequences of an accident or malfunction previously considered or reduce a margin of safety, and, thus, does not involve a significant hazards consideration. Indeed, as described above, the staff considers that the use and availability of the MDHRS will enhance the licensee's ability to maintain the reactor in a safe shutdown cooling mode by providing a simplified and appropriately sized decay heat removal system.

We have also determined that the modification 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 modification involves an action which is insignificant from the standpoint of environmental impact and, pursuant to 10 CFR Section 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 the modification.

III.

Accordingly, pursuant to the Atomic Energy Act of 1954, as amended, the Director's Order of February 11, 1980 is hereby revised to incorporate the deletions, additions and modifications set forth in Attachment A hereto.

For further details with respect to this action, see (1) Letter to B. Snyder, USNRC, from R. C. Arnold, Met. Ed/GPU, Technical Specification Change Request No. 24, dated July 31, 1980, (TLL 372); (2) Letter to B. Snyder, USNRC, from R. C. Arnold, Met. Ed/GPU, Technical Specification Change Request No. 24, dated August 5, 1980, (TLL 382); (3) Letter to R. C. Arnold, Met. Ed/GPU, from John T. Collins, USNRC, TMI-2 Recovery Operations Plan Change Request No. 4, dated July 25, 1980, (NRC/TMI-80-115); and (4) the Director's Order of February 11, 1980.

All of the above documents are available for inspection at the Commission's Public Document Room, 1717 H Street, N.W., Washington, D.C., and at the Commission's Local Public Document Room at the State Library of Pennsylvania, Government Publications Section, Education Building, Commonwealth and Walnut Streets, Harrisburg, Pennsylvania 17126.

FOR THE NUCLEAR REGULATORY COMMISSION

Harold R. Denton

Harold R. Denton, Director
Office of Nuclear Reactor Regulation

Effective date: November 14, 1980
Dated at Bethesda, Maryland

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

METROPOLITAN EDISON COMPANY

JERSEY CENTRAL POWER AND LIGHT COMPANY

PENNSYLVANIA ELECTRIC COMPANY

DOCKET NO. 50-320

THREE MILE ISLAND NUCLEAR STATION, UNIT NO. 2

Introduction

By letters dated July 31, 1980 (Reference 1) and August 5, 1980 (Reference 2), the Metropolitan Edison Company (licensee) proposed changes to the Recovery Mode technical specifications for Three Mile Island Unit 2 (TMI-2) dealing with the implementation of the Mini Decay Heat Removal System (MDHRS). Although several modes for removing decay heat would be available, the MDHRS would provide a forced flow system for removing decay heat from the TMI-2 reactor fuel. Accordingly, the proposed changes would impose operability requirements for the MDHRS and would also delete the operability requirements for certain Balance of Plant (BOP) systems which have been used since the March 28, 1979, accident, but would no longer be required, for removing the decay heat. The operability requirements for these BOP systems had been imposed by the Order of the Director of the Office of Nuclear Reactor Regulation on February 11, 1980, (F.R. 11282) in the form of proposed Technical Specifications.

Summary

The licensee has requested NRC staff approval of an additional long term core cooling method. The proposed change would make available a newly installed MDHRS to remove decay heat rather than the present method which accomplishes this function by using the "A" steam generator in a steaming mode to the condenser.

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The licensee's proposal required certain modifications to meet our requirements with which the licensee has agreed. With the incorporation of these staff required modifications, we have found the proposal to be an acceptable method for removing the decay heat and have therefore granted the licensee's request to modify the method used for long term core cooling.

Evaluation

The TMI-2 Reactor Coolant System (RCS) is currently operating in a natural circulation heat removal mode with heat rejection from it being accomplished by both loss to ambient (reactor building atmosphere and sump water) and through the "A" steam generator. The reactor building is in turn being cooled by the reactor building fan coolers while the "A" steam generator is steaming to the condenser through the turbine by-pass valve. This mode of core cooling has been in effect since late April 1979. With the passage of time and the associated reduction of decay heat generation rate (presently approximately 75 kw), the natural circulation flow has changed from continuous to cyclic with increasing intervals between the cyclic flow "burps".

Heat rejection through the "A" steam generator by steaming to the condenser requires the operation of several major BOP systems including: circulating water system, main steam system and the "A" steam generator, condensate and feedwater systems, main condenser and package boiler. Conversion to and use of the MDHRS for core cooling would simplify the plant operations since use of the MDHRS would eliminate the need for operating the previously noted BOP systems.

The MDHRS is classified as a nonsafety grade system but it is designed and installed to seismic Category I requirements up to and including the second isolation valve in its supply and discharge lines. The balance of the system is designed and installed to Operating Basis Earthquake requirements. The MDHRS takes suction from the "B" loop of the Decay Heat System (DH) outlet from the reactor vessel via a connection to the Alternate Decay Heat Removal System (ADH). After passing through one of the MDHRS's parallel heat exchangers and pumps, the reactor coolant is returned to the reactor coolant system through the "B" Core Flooding injection nozzle via a connection to the ADH and DH systems. Connection of the ADH to the DH was evaluated and approved in NUREG-0557. The MDHRS is sized such that one pump and one heat exchanger (two of each are installed) could remove up to approximately 1 Mw of decay heat. Therefore the MDHRS has more than adequate cooling capacity for removing the present and future decay heat loads. The MDHRS would be cooled by the Nuclear Services Closed Cycle Cooling System which is required to be operable by proposed Technical Specification 3.7.3.1. The power supply for the MDHRS pumps and motor operated valves is from redundant Class 1E buses which would be manually loaded on the Class 1E diesel generators in the event of a loss of off-site power. A detailed description of the MDHRS and of its principal modes of operation is provided in the enclosure to Reference 3.

In the event the MDHRS is not used or becomes inoperable, backup cooling modes are available for removing the decay heat from the RCS. These backup cooling modes include the long term "B" steam generator cooling system and "Loss to Ambient". The long term "B" steam generator cooling system has been previously evaluated and its operability would continue to be required by proposed Technical Specification 3.7.1. In its analysis of the "Loss of Ambient" cooling mode (discussed in References 1 and 2), the licensee calculated that with the present decay heat generation rate, the reactor coolant bulk temperature would initially increase at a rate of approximately 0.4 ^{o}F/hour and that this heatup rate would gradually decrease such that at a RCS temperature of approximately 190 ^{o}F, the heatup rate would be approximately zero. At this RCS temperature, the heat loss from the RCS to ambient (reactor building atmosphere and sump water) would be equal to the decay heat generation rate. The RCS operating pressure during MDHRS operation would be maintained at 90$\pm$10 psig; therefore a substantial margin to the saturation temperature exists and the plant conditions would reach a stable, equilibrium condition. The NRC staff has reviewed the licensee's results of the "Loss to Ambient" cooling mode and has performed an independent analysis, the results of which are in agreement with the licensee's conclusion. Therefore, we have concluded that any one of these cooling methods provides an acceptable means for long term cooling of the reactor core.

Written procedures for operating the MDHRS and for operation in the "Loss to Ambient" cooling mode will be required for operation in either of these cooling modes. These procedures will be prepared and submitted to the NRC staff in accordance with the requirements of proposed Technical Specification 6.8.1 and 6.8.2 prior to implementation.

The MDHRS has a design pressure of 235 psig. Therefore, consideration was given to possible sources of overpressurization of the MDHRS. Three potential sources of MDHRS overpressurization were identified. These sources were: (1) Makeup pump operation with MDHRS in operation, (2) Pressurizer heater operation, and (3) Malfunction of the Standby Pressure Control (SPC) System.

To preclude operation of a makeup pump during operation of the MDHRS, the licensee proposed to delet the requirement for an operable makeup pump from the proposed Technical Specifications but to retain the option to operate the pump for certain operations (e.g. degassing). The licensee further stated that the electrical power supply circuit breakers for the makeup pumps would be "racked out" when valve

DH-V1 or DH-V171 is open. Since operation of a makeup pump may be required in one or more of the backup cooling modes or for degassing operations, we will retain a requirement for its operability in proposed Technical Specification 3.1.1.1. However, to provide assurance that the MDHRS will not be overpressurized due to operation of a makeup pump, we propose to add a requirement to proposed Technical Specification 3.1.1.1 that all makeup pumps be made inoperable when valve DH-V1 or DH-V171 is open by "racking out" their electrical power supply circuit breakers. We would also add a surveillance requirement to the Recovery Operations Plan to periodically verify that these breakers are "racked out". These actions provide assurance that the MDHRS would not be overpressurized due to operation of the makeup pumps.

Operation of the pressurizer heaters while operating the RCS in a water solid mode with the MDHRS in operation creates the potential for overpressurization of the MDHRS due to volumetric expansion of the reactor coolant as a result of heat input to the reactor coolant. The licensee has calculated that the electrical energization of all the pressurizer heaters (1638 kw) would result in a volumetric expansion of the reactor coolant which would require a compensating relief capacity of 8.6 gpm. The MDHRS has an installed relief valve capacity of 53.5 gpm. The NRC staff has reviewed the licensee's result of this potential overpressurization event and has performed an independent check, the results of which are in agreement with the licensee's conclusion. Therefore, we agree that operation of the pressurizer heaters while operating the MDHRS with a water solid RCS would not result in overpressurization of the MDHRS.

Overpressurization of the MDHRS due to a malfunction of the SPC system has been precluded by reducing the SPC in-service nitrogen bank pressure to a new operating range of 225 to 400 psig and by the installation of a SPC systems pressure relief valve (SPC-R14) which has been set to provide overpressure relief if the SPC system pressure exceeds 125 psig which is substantially below the MDHRS design pressure of 235 psig. The change in the nitrogen bank pressure was approved on July 25, 1980 (Reference 4).

The licensee also proposed deleting from proposed Technical Specification 3.1.1.1 the requirements for a boric acid storage system and an associated flow path to the RCS. We have reviewed this proposed change and since redundant boron injection flow paths from the BWST to the RCS via the makeup pump and decay heat removal pump exist, we find the proposed change acceptable.

The licensee's proposed Technical Specification for the MDHRS would require only one operable MDHRS pump and heat exchanger with an action statement providing instructions to be taken in the event of their inoperability. Our position is that this Technical Specification should require the operability of both MDHRS pumps and heat exchangers and the associated flow path and that applicable action statements should be supplied dealing with the inoperability of the various components in the MDHRS. The staff's position is consistent with the operability requirements for similar systems (e.g. proposed Technical Specifications 3.7.2.1, 3.7.3.1 and 3.7.4.1) in which the redundant systems are required operable and action statements are provided for when one or both systems are inoperable. Unless both MDHRS pumps are required operable and periodically demonstrated so in accordance with applicable surveillance requirements, there is no assurance of the operability of the redundant pump should its use be required for any reason. We have therefore modified the Technical Specification proposed for the MDHRS to be in accordance with our position. The licensee has agreed with our position on this matter. We have also added appropriate surveillance requirements to the Operations Recovery Plan to periodically demonstrate the operability of the MDHRS.

The Licensee also proposed to delete from proposed Technical Specification 3.8.2.1, the operability requirements for several electrical power busses. Its basis for proposing to delete these requirements was that they supplied electrical power to the various BOP systems which were proposed for deletion from the proposed Technical Specifications upon incorporation of the MDHRS. However, in our review of these proposed changes, we determined that four of the busses proposed for deletion (480 volt Busses 2-35, 2-36, 2-45 and 2-46) also supplied electrical power to the auxiliary building and the fuel handling building air cleanup systems which are in turn required operable per proposed Technical Specification 3.9.12. Therefore, to ensure the electrical power supply for these air cleanup systems, we have retained the operability requirements for these four busses in proposed Technical Specification 3.8.2.1. The licensee has agreed that the operability requirements for these four busses should be retained in proposed Technical Specification 3.8.2.1. We agree that the operability requirements for the other busses can be deleted from the proposed Technical Specifications as proposed by the licensee since the other busses proposed for deletion do not supply electrical power to any systems required for maintaining the plant in its safe shutdown condition.

The licensee has postulated an accident involving isolation of the MDHRS at inlet and outlet isolation valves followed by approximately 1200 gallons of coolant water inventory being dumped on the floor of the auxiliary building. Since the MDHRS operating temperature is shown by the licensee's thermal analysis never to exceed

190 ^{o}F, even when the system heat removal capacity is lost, dumping of the coolant is not expected to result in any flashing. In the analysis, the licensee assumed that the airborne source term will not exceed 1% of the total coolant inventory of radionuclides consisting primarily of cesium and strontium. The airborne source term in the auxiliary building is then released to the atmosphere as a puff of radionuclides. As a result of this analysis, the licensee calculated a site boundary dose of 1.16×10^{-4} rem.

We also considered the possibility of recriticality of the reactor core during operation of the MDHRS. The potential for recriticality with the reactor coolant system operating in cyclic natural circulation was evaluated in Reference 5. The only method identified therein that could lead to recriticality was by boron dilution of the reactor coolant. In Reference 5 it was concluded that the timing of such an accident was very long and that several weeks of continuous boron dilution at the then present makeup rate would be required to reach criticality. The present makeup rate is lower by a factor of at least two than the makeup rate used in that analysis. Due to the mixing action provided by the MDHRS pumps, use of the MDHRS will ensure uniform boron concentrations in the core and, as described in Reference 3, will provide a more representative sample of the reactor coolant and therefore a potential boron dilution accident would be detected even more readily than as described in Reference 5. Therefore, we concur that the accident postulated by the licensee is the severest accident. We performed a consequences analysis utilizing the source term calculated from the radionuclide concentrations measured in TMI-2 coolant samples by Oak Ridge National Laboratory. The data showed absence of any noble gases, presence of extremely small concentrations of iodine, and presence of cesium and strontium. The source term was therefore calculated by neglecting noble gases and iodines.

Based upon the review of the vapor pressure of cesium and strontium at the maximum coolant temperature of 190 ^{o}F calculated by the licensee, we believe that the licensee's assumption of 1% of the coolant radionuclides being airborne is conservative, and have used this value in our evaluation of off-site doses (Exclusion Area Boundary) shown in Table 1.

For the postulated accident, the staff has used a "small fraction" (about 1 percent) of the 10 CFR 100 dose guidelines as a criterion for the radiological consequences which would not present an undue risk to the public.

Since the doses calculated for a failure of the MDHRS, as shown in Table 1, are substantially less than a "small fraction of Part 100," it is concluded that the postulated accident would not cause an undue risk to public health and safety.

TABLE 1

Offsite Doses

For Mini Decay Heat Removal System Failure

Nuclide	Coolant Samples Concentration μCi/gm	X/Q sec/m	Whole Body Dose mrem
CS-134	20	$1.1x10^{-3}$	2.4
Cs-136	0.5	$1.1x10^{-3}$	1.5x10
Cs-137	100	$1.1x10^{-3}$	6.8
Sr-89	300	$1.1x10^{-3}$	0.85
Sr-90	20	$1.1x10^{-3}$	29.2
TOTAL			39.3

Environmental Consideration

We have determined that the modification 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 modification involves an action which is insignificant from the standpoint of environmental impact and, pursuant to 10 CFR Section 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 the modification.

Conclusion

Based upon our review of MDHRS modification, the attendant Technical Specifications, and our findings that the proposed mode provide cooling options which are reliable and less complex means for long term core cooling, we find the licensee's request to be acceptable and grant the request to make said modifications. The measures authorized in connection with this evaluation will assure the continued maintenance of the facility in a safe, stable, long-term cooling condition, as discussed above. Based on these considerations, we have concluded that: (1) the modification does not involve a significant increase in the probability or consequences of accidents previously considered or a significant reduction in a margin of safety and 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 modified manner, and (3) such activities will be conducted in compliance with the Commission's regulations and the issuance of this modification will not be inimical to the common defense and security or to the health and safety of the public.

REFERENCES

1. Letter to B. Snyder, USNRC, from R. C. Arnold, Met. Ed/GPU, Technical Specification Change Request No. 24, dated July 31, 1980, (TLL 372).

2. Letter to B. Snyder, USNRC, from R. C. Arnold, Met. Ed/GPU, Technical Specification Change Request No. 24, dated August 5, 1980, (TLL 382).

3. Letter to John T. Collins, USNRC, from G. K. Hovey, Met. Ed/GPU, MDHR System Description, Revision 3, dated September 8, 1980, (TLL 438).

4. Letter to R. C. Arnold, Met. Ed/GPU, from John T. Collins, USNRC, TMI-2 Recovery Operations Plan Change Request No. 4, dated July 25, 1980, (NRC/TMI-80-115).

5. Memorandum for William J. Dircks, from Normal M. Haller, "Report of Special Task Force on Three Mile Island Cleanup", dated February 28, 1980.

Attachment A

FACILITY OPERATING LICENSE NO. DPR-73

DOCKET NO. 50-320

Replace the following pages of the proposed Appendix "A" Technical Specifications with the enclosed pages as indicated. The revised pages contain vartical lines indicating the area of change. The corresponding overleaf pages are also provided to maintain document completeness.

Pages

3.1-1
3.1-2
3.4-1
3.7-1
3.7-2
3.8-4
B 3/4 4-1
B 3/4 7-1

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3.1 WATER INJECTION COOLING AND REACTIVITY CONTROL SYSTEMS

3.1.1 BORATION CONTROL

BORON INJECTION

3.1.1.1 At least two systems capable of injecting borated cooling water into the Reactor Coolant System shall be OPERABLE* with:

a. One system comprised of:

1. One OPERABLE makeup pump.#
2. One OPERABLE decay heat removal pump.
3. An OPERABLE flow path from the BWST. The BWST shall contain at least 100,000 gallons of borated water at a minimum temperature of 50°F and at a boron concentration of between 3000 and 4500 ppm.

b. The second system comprised of the Standby Reactor Coolant System Pressure Control System.

APPLICABILITY: When fuel is in the reactor pressure vessel.

ACTION:

With one of the above required systems inoperable, restore the inoperable system to OPERABLE status within 72 hours.

BORON CONCENTRATION

3.1.1.2 The reactor coolant shall be maintained at a boron concentration of between 3000 and 4500 ppm and at a temperature above 50°F.

APPLICABILITY: When fuel is in the reactor pressure vessel.

ACTION

None except as provided in Specification 3.0.3.

*Both systems shall be considered OPERABLE when aligned per procedures approved pursuant to Specification 6.8.2.

#All makeup pumps shall be made inoperable when valve DH-V1 or DH-V171 is open by racking out their electrical power supply circuit breakers.

Deleted

3.4 REACTOR COOLANT SYSTEM

REACTOR COOLANT LOOPS

3.4.1 The Reactor Coolant System shall be operated in accordance with procedures approved pursuant to Specification 6.8.2.

APPLICABILITY: RECOVERY MODE.

ACTION:

None except as provided in Specification 3.0.3.

SAFETY VALVES

3.4.3 All pressurizer code safety valves shall be OPERABLE with a lift setting of 2435 PSIG $\pm$ 1%.*

APPLICABILITY: RECOVERY MODE.

ACTION:

None except as provided in Specification 3.0.3.

3.4.9 PRESSURE/TEMPERATURE LIMITS

REACTOR COOLANT SYSTEM

3.4.9.1 The Reactor Coolant System shall be maintained at a T_{avg} of less than 280°F and at a pressure of less than 600 psig.

APPLICABILITY: When fuel is in the reactor pressure vessel.

ACTION:

None except as provided in Specification 3.0.3.

*The lift setting pressure shall correspond to ambient conditions of the valve at nominal operating temperature and pressure.

3.7 PLANT SYSTEMS

3.7.1 FEEDWATER SYSTEM

3.7.1 The long term "B" steam generator cooling system shall be maintained in an OPERABLE status.

APPLICABILITY: RECOVERY MODE.

ACTION:

With the long term cooling "B" steam generator cooling system inoperable, restore the inoperable system to OPERABLE status within 72 hours.

3.7.2 SECONDARY SERVICES CLOSED COOLING WATER SYSTEM

3.7.2.1 At least two secondary services closed cooling water system pumps and heat exchangers and the associated flow path shall be OPERABLE with each pump capable of being powered from separate busses.

APPLICABILITY: RECOVERY MODE.

ACTION:

With only one secondary services closed cooling water pump or only one secondary services heat exchanger OPERABLE, restore the inoperable pump or heat exchanger to OPERABLE status within 72 hours.

3.7.3 CLOSED CYCLE COOLING WATER SYSTEM

NUCLEAR SERVICES CLOSED CYCLE COOLING SYSTEM

3.7.3.1 At least two independent nuclear services closed cycle cooling water pumps and heat exchangers and the associated flow path shall be OPERABLE with each pump capable of being powered from separate emergency busses.

APPLICABILITY: RECOVERY MODE.

ACTION:

With only one nuclear services closed cycle cooling water pump or only one nuclear services heat exchanger OPERABLE, restore the inoperable pump or heat exchanger to OPERABLE status within 72 hours.

DECAY HEAT CLOSED COOLING WATER SYSTEM

3.7.3.2 At least one decay heat closed cooling water loop shall be OPERABLE.

APPLICABILITY: RECOVERY MODE.

ACTION:

With no decay heat closed cooling water loop OPERABLE, restore the inoperable loop to OPERABLE status within 24 hours.

MINI DECAY HEAT REMOVAL SYSTEM

3.7.3.3 Two mini decay heat removal pumps and heat exchangers and the associated flow path shall be OPERABLE.

APPLICABILITY: When fuel is in the reactor pressure vessel.

ACTION:

a. With one mini decay heat removal pump and/or heat exchanger inoperable, restore the inoperable pump and/or heat exchanger to OPERABLE status within 72 hours.

b. With two mini decay heat removal pumps and/or heat exchangers or the associated flow path inoperable, restore at least one pump and heat exchanger and the associated flow path to OPERABLE status within 24 hours or within the next 48 hours make a backup cooling system (either LTC "B" or "Loss to Ambient") OPERABLE.

3.7.4 NUCLEAR SERVICE RIVER WATER SYSTEM

3.7.4.1 Two independent nuclear service river water loops shall be OPERABLE.

APPLICABILITY: RECOVERY MODE.

ACTION:

With one nuclear service river water loop inoperable, restore the inoperable loop to OPERABLE status within 72 hours.

TABLE 3.8-2

RESTORATION TIME MATRIX

Combination of Two Inoperable Components		Restore One Component (Hours)	Restore Other Component (Hours)
	aa	24	72
	ab	12	72
	bb	12	72

Note: a and b above correspond to components described in Specification 3.8.1.1 items a and b respectively.

3.8.2 ONSITE POWER DISTRIBUTION SYSTEMS

A.C. DISTRIBUTION

3.8.2.1 The following A.C. electrical busses shall be OPERABLE and energized with tie breakers open (unless closed in accordance with procedures approved pursuant to Specification 6.8.2) between redundant busses:

4160 volt Emergency Bus # 2-1E and 2-3E

4160 volt Emergency Bus # 2-2E and 2-4E

4160 volt Busses # 2-3 and 2-4

480 volt Emergency Bus # 2-11E, 2-12E and 2-31E

480 volt Emergency Bus # 2-21E, 2-22E and 2-41E

480 volt Busses # 2-31, 2-41, 2-32, 2-42, 2-35, 2-36, 2-45, and 2-46

120 volt A.C. Vital Bus # 2-1V

120 volt A.C. Vital Bus # 2-2V

120 volt A.C. Vital Bus # 2-3V

120 volt A.C. Vital Bus # 2-4V

APPLICABILITY: RECOVERY MODE.

ACTION:

With less than the above complement of A.C. busses OPERABLE, restore the inoperable bus to OPERABLE status within 8 hours.

3/4.4 REACTOR COOLANT SYSTEM

BASES

3/4.4.1 REACTOR COOLANT LOOPS

Several alternative methods are available for removal of reactor decay heat. These methods include use of the Mini Decay Heat Removal System, the "Loss to Ambient" cooling mode, and operation of the Reactor Coolant System in the natural circulation mode with heat rejection via the long term "B" steam generator cooling mode. Any one of these cooling methods provides adequate cooling of the reactor and each method is available for decay heat removal. Procedures have been prepared and approved for use of these various cooling methods.

3/4.4.3 SAFETY VALVES

The pressurizer code safety valves operate to prevent the RCS from being pressurized above its Safety Limit of 2750 psig. Each safety valve is designed to relieve 348,072 lbs per hour of saturated steam at the valve's setpoint.

3/4.4.9 PRESSURE/TEMPERATURE LIMIT

The RCS pressure and temperature will be controlled in accordance with approved procedures to prevent a nonductile failure of the RCS while at the same time permitting the RCS pressure to be maintained at a sufficiently high value to permit operation of the reactor coolant pumps.

3/4.7.1 FEEDWATER SYSTEM

The long term "B" steam generator cooling system is required to be maintained in an OPERABLE status since it is an alternative method for removing decay heat from the reactor coolant system.

3/4.7.2 SECONDARY SERVICES CLOSED COOLING WATER SYSTEM

The secondary services closed cooling water system is required to be maintained in an OPERABLE condition since it is used to cool the "B" steam generator closed loop cooling system.

3/4.7.3 CLOSED CYCLE COOLING WATER SYSTEM

3/4.7.3.1 NUCLEAR SERVICES CLOSED CYCLE COOLING SYSTEM

OPERABILITY of the nuclear services closed cycle cooling system is required during operation of the MDHRS since this system provides the heat sink for the MDHRS.

3/4.7.3.2 DECAY HEAT CLOSED COOLING WATER SYSTEM

The decay heat closed cooling water system is required to be maintained in an OPERABLE status since it is provided to remove heat from the DHR system which is being maintained OPERABLE in a backup status for possible core cooling.

3/4.7.3.3 MINI DECAY HEAT REMOVAL SYSTEM (MDHRS)

OPERABILITY of the MDHRS is required since it is an alternative method for removing decay heat from the reactor. The MDHRS is provided with two pumps and two heat exchangers; one pump and one heat exchanger have adequate capacity for removing the present level of decay heat from the core.

3/4.7.4 NUCLEAR SERVICE RIVER WATER SYSTEM

The nuclear service river water system uses river water to cool the nuclear services closed cycle cooling system, the secondary services closed cooling water system, and decay heat closed cooling water system; therefore, it must be OPERABLE too. This system rejects its heat to the river as the ultimate heat sink.

3/4.7.6 FLOOD PROTECTION

The limitation on flood protection ensures that facility protective actions will be taken in the event of flood conditions. The limit of elevation 302 Mean Sea Level is based on the maximum elevation at which facility flood control measures provide protection to safety related equipment.

3/4.7.7 CONTROL ROOM EMERGENCY AIR CLEANUP SYSTEM

The OPERABILITY of the control room emergency air cleanup system ensures that 1) the ambient air temperature does not exceed the allowable temperature for continuous duty rating for the equipment and instrumentation cooled by this system and 2) the control room will remain habitable for operations personnel during and following all credible accident conditions. The OPERABILITY of this system in conjunction with control room design provisions is based on limiting the radiation exposure to personnel occupying the control room to 5 rem or less whole body, or its equivalent. This limitation is consistent with the requirements of General Design Criterion 19 of Appendix "A", 10 CFR 50.

3/4.7.10 FIRE SUPPRESSION SYSTEMS

The OPERABILITY of the fire suppression systems ensures that adequate fire suppression capability is available to confine and extinguish fires occuring in any portion of the facility where safety related equipment is located. The fire suppression system consists of the water system, spray and/or sprinklers, Halon and fire hose stations. The collective capability of the fire suppression systems is adequate to minimize potential damage to safety related equipment and is a major element in the facility fire protection program. Any two of the four main fire pumps provide combined capacity greater than 3575 gpm.

In the event that portions of the fire suppression systems are inoperable, alternate backup fire fighting equipment is required to be made available in the affected areas until the affected equipment can be restored to service.

In the event that the fire suppression water system becomes inoperable, immediate corrective measures must be taken since this system provides the major fire suppression capability of the plant. The requirement for a Special Report to the Commission provides for timely evaluation of the acceptability of the corrective measures to provide adequate fire suppression capability for the continued operation of the nuclear plant.

Attachment A

FACILITY OPERATING LICENSE NO. DPR-73

DOCKET NO. 50-320

Replace the following pages of the proposed Appendix "A" Technical Specifications with the enclosed pages as indicated. The revised pages contain vartical lines indicating the area of change. The corresponding overleaf pages are also provided to maintain document completeness.

Pages

3.1-1
3.1-2
3.4-1
3.7-1
3.7-2
3.6-4
B 3/4 4-1
B 3/4 7-1

3.1 WATER INJECTION COOLING AND REACTIVITY CONTROL SYSTEMS

3.1.1 BORATION CONTROL

BORON INJECTION

3.1.1.1 At least two systems capable of injecting borated cooling water into the Reactor Coolant System shall be OPERABLE* with:

a. One system comprised of:

1. One OPERABLE makeup pump.#
2. One OPERABLE decay heat removal pump.
3. An OPERABLE flow path from the BWST. The BWST shall contain at least 100,000 gallons of borated water at a minimum temperature of 50°F and at a boron concentration of between 3000 and 4500 ppm.

b. The second system comprised of the Standby Reactor Coolant System Pressure Control System.

APPLICABILITY: When fuel is in the reactor pressure vessel.

ACTION:

With one of the above required systems inoperable, restore the inoperable system to OPERABLE status within 72 hours.

BORON CONCENTRATION

3.1.1.2 The reactor coolant shall be maintained at a boron concentration of between 3000 and 4500 ppm and at a temperature above 50°F.

APPLICABILITY: When fuel is in the reactor pressure vessel.

ACTION

None except as provided in Specification 3.0.3.

*Both systems shall be considered OPERABLE when aligned per procedures approved pursuant to Specification 6.8.2.

#All makeup pumps shall be made inoperable when valve DH-V1 or DH-V171 is open by racking out their electrical power supply circuit breakers.

Deleted

3.4 REACTOR COOLANT SYSTEM

REACTOR COOLANT LOOPS

3.4.1 The Reactor Coolant System shall be operated in accordance with procedures approved pursuant to Specification 6.8.2.

APPLICABILITY: RECOVERY MODE.

ACTION:

None except as provided in Specification 3.0.3.

SAFETY VALVES

3.4.3 All pressurizer code safety valves shall be OPERABLE with a lift setting of 2435 PSIG ± 1%.*

APPLICABILITY: RECOVERY MODE.

ACTION:

None except as provided in Specification 3.0.3.

3.4.9 PRESSURE/TEMPERATURE LIMITS

REACTOR COOLANT SYSTEM

3.4.9.1 The Reactor Coolant System shall be maintained at a T_{avg} of less than 280°F and at a pressure of less than 600 psig.

APPLICABILITY: When fuel is in the reactor pressure vessel.

ACTION:

None except as provided in Specification 3.0.3.

*The lift setting pressure shall correspond to ambient conditions of the valve at nominal operating temperature and pressure.

3.7 PLANT SYSTEMS

3.7.1 FEEDWATER SYSTEM

3.7.1 The long term "B" steam generator cooling system shall be maintained in an OPERABLE status.

APPLICABILITY: RECOVERY MODE.

ACTION:

With the long term cooling "B" steam generator cooling system inoperable, restore the inoperable system to OPERABLE status within 72 hours.

3.7.2 SECONDARY SERVICES CLOSED COOLING WATER SYSTEM

3.7.2.1 At least two secondary services closed cooling water system pumps and heat exchangers and the associated flow path shall be OPERABLE with each pump capable of being powered from separate busses.

APPLICABILITY: RECOVERY MODE.

ACTION:

With only one secondary services closed cooling water pump or only one secondary services heat exchanger OPERABLE, restore the inoperable pump or heat exchanger to OPERABLE status within 72 hours.

3.7.3 CLOSED CYCLE COOLING WATER SYSTEM

NUCLEAR SERVICES CLOSED CYCLE COOLING SYSTEM

3.7.3.1 At least two independent nuclear services closed cycle cooling water pumps and heat exchangers and the associated flow path shall be OPERABLE with each pump capable of being powered from separate emergency busses.

APPLICABILITY: RECOVERY MODE.

ACTION:

With only one nuclear services closed cycle cooling water pump or only one nuclear services heat exchanger OPERABLE, restore the inoperable pump or heat exchanger to OPERABLE status within 72 hours.

DECAY HEAT CLOSED COOLING WATER SYSTEM

3.7.3.2 At least one decay heat closed cooling water loop shall be OPERABLE.

APPLICABILITY: RECOVERY MODE.

ACTION:

With no decay heat closed cooling water loop OPERABLE, restore the inoperable loop to OPERABLE status within 24 hours.

MINI DECAY HEAT REMOVAL SYSTEM

3.7.3.3 Two mini decay heat removal pumps and heat exchangers and the associated flow path shall be OPERABLE.

APPLICABILITY: When fuel is in the reactor pressure vessel.

ACTION:

a. With one mini decay heat removal pump and/or heat exchanger inoperable, restore the inoperable pump and/or heat exchanger to OPERABLE status within 72 hours.

b. With two mini decay heat removal pumps and/or heat exchangers or the associated flow path inoperable, restore at least one pump and heat exchanger and the associated flow path to OPERABLE status within 24 hours or within the next 48 hours make a backup cooling system (either LTC "B" or "Loss to Ambient") OPERABLE.

3.7.4 NUCLEAR SERVICE RIVER WATER SYSTEM

3.7.4.1 Two independent nuclear service river water loops shall be OPERABLE.

APPLICABILITY: RECOVERY MODE.

ACTION:

With one nuclear service river water loop inoperable, restore the inoperable loop to OPERABLE status within 72 hours.

TABLE 3.8-2

RESTORATION TIME MATRIX

		Restore One Component (Hours)	Restore Other Component (Hours)
Combination of Two Inoperable Components	aa	24	72
	ab	12	72
	bb	12	72

Note: a and b above correspond to components described in Specification 3.8.1.1 items a and b respectively.

3.8.2 ONSITE POWER DISTRIBUTION SYSTEMS

A.C. DISTRIBUTION

3.8.2.1 The following A.C. electrical busses shall be OPERABLE and energized with tie breakers open (unless closed in accordance with procedures approved pursuant to Specification 6.8.2) between redundant busses:

4160	volt Emergency Bus # 2-1E and 2-3E
4160	volt Emergency Bus # 2-2E and 2-4E
4160	volt Busses # 2-3 and 2-4
480	volt Emergency Bus # 2-11E, 2-12E and 2-31E
480	volt Emergency Bus # 2-21E, 2-22E and 2-41E
480	volt Busses # 2-31, 2-41, 2-32, 2-42, 2-35, 2-36, 2-45, and 2-46
120	volt A.C. Vital Bus # 2-1V
120	volt A.C. Vital Bus # 2-2V
120	volt A.C. Vital Bus # 2-3V
120	volt A.C. Vital Bus # 2-4V

APPLICABILITY: RECOVERY MODE

ACTION:

With less than the above complement of A.C. busses OPERABLE, restore the inoperable bus to OPERABLE status within 8 hours.

3/4.4.1 REACTOR COOLANT LOOPS

Several alternative methods are available for removal of reactor decay heat. These methods include use of the Mini Decay Heat Removal System, the "Loss to Ambient" cooling mode, and operation of the Reactor Coolant System in the natural circulation mode with heat rejection via the long term "B" steam generator cooling mode. Any one of these cooling methods provides adequate cooling of the reactor and each method is available for decay heat removal. Procedures have been prepared and approved for use of these various cooling methods.

3/4.4.3 SAFETY VALVES

The pressurizer code safety valves operate to prevent the RCS from being pressurized above its Safety Limit of 2750 psig. Each safety valve is designed to relieve 348,072 lbs per hour of saturated steam at the valve's setpoint.

3/4.4.9 PRESSURE/TEMPERATURE LIMIT

The RCS pressure and temperature will be controlled in accordance with approved procedures to prevent a nonductile failure of the RCS while at the same time permitting the RCS pressure to be maintained at a sufficiently high value to permit operation of the reactor coolant pumps.

3/4.7.1 FEEDWATER SYSTEM

The long term "B" steam generator cooling system is required to be maintained in an OPERABLE status since it is an alternative method for removing decay heat from the reactor coolant system.

3/4.7.2 SECONDARY SERVICES CLOSED COOLING WATER SYSTEM

The secondary services closed cooling water system is required to be maintained in an OPERABLE condition since it is used to cool the "B" steam generator closed loop cooling system.

3/4.7.3 CLOSED CYCLE COOLING WATER SYSTEM

3/4.7.3.1 NUCLEAR SERVICES CLOSED CYCLE COOLING SYSTEM

OPERABILITY of the nuclear services closed cycle cooling system is required during operation of the MDHRS since this system provides the heat sink for the MDHRS.

3/4.7.3.2 DECAY HEAT CLOSED COOLING WATER SYSTEM

The decay heat closed cooling water system is required to be maintained in an OPERABLE status since it is provided to remove heat from the DHR system which is being maintained OPERABLE in a backup status for possible core cooling.

3/4.7.3.3 MINI DECAY HEAT REMOVAL SYSTEM (MDHRS)

OPERABILITY of the MDHRS is required since it is an alternative method for removing decay heat from the reactor. The MDHRS is provided with two pumps and two heat exchangers; one pump and one heat exchanger have adequate capacity for removing the present level of decay heat from the core.

3/4.7.4 NUCLEAR SERVICE RIVER WATER SYSTEM

The nuclear service river water system uses river water to cool the nuclear services closed cycle cooling system, the secondary services closed cooling water system, and decay heat closed cooling water system; therefore, it must be OPERABLE too. This system rejects its heat to the river as the ultimate heat sink.

3/4.7.6 FLOOD PROTECTION

The limitation on flood protection ensures that facility protective actions will be taken in the event of flood conditions. The limit of elevation 302 Mean Sea Level is based on the maximum elevation at which facility flood control measures provide protection to safety related equipment.

3/4.7.7 CONTROL ROOM EMERGENCY AIR CLEANUP SYSTEM

The OPERABILITY of the control room emergency air cleanup system ensures that 1) the ambient air temperature does not exceed the allowable temperature for continuous duty rating for the equipment and instrumentation cooled by this system and 2) the control room will remain habitable for operations personnel during and following all credible accident conditions. The OPERABILITY of this system in conjunction with control room design provisions is based on limiting the radiation exposure to personnel occupying the control room to 5 rem or less whole body, or its equivalent. This limitation is consistent with the Requirements of General Design Criterion 19 of Appendix "A", 10 CFR 50.

3/4.7.10 FIRE SUPPRESSION SYSTEMS

The OPERABILITY of the fire suppression systems ensures that adequate fire suppression capability is available to confine and extinguish fires occuring in any portion of the facility where safety related equipment is located. The fire suppression system consists of the water system, spray and/or sprinklers, Halon and fire hose stations. The collective capability of the fire suppression systems is adequate to minimize potential damage to safety related equipment and is a major element in the facility fire protection program. Any two of the four main fire pumps provide combined capacity greater than 3575 gpm.

In the event that portions of the fire suppression systems are inoperable, alternate backup fire fighting equipment is required to be made available in the affected areas until the affected equipment can be restored to service.

In the event that the fire suppression water system becomes inoperable, immediate corrective measures must be taken since this system provides the major fire suppression capability of the plant. The requirement for a Special Report to the Commission provides for timely evaluation of the acceptability of the corrective measures to provide adequate fire suppression capability for the continued operation of the nuclear plant.

Replace the following pages of the Recovery Operations Plan with the enclosed pages as indicated. The revised pages contain vertical lines indicating the area of change. The corresponding overleaf pages are also provided to maintain document completeness.

Pages

4.1-1
4.1-2
4.7-1
4.7-2

4.1 WATER INJECTION COOLING AND REACTIVITY CONTROL SYSTEMS

4.1.1 BORATION CONTROL

BORON INJECTION

4.1.1.1 Two systems capable of injecting borated cooling water into the Reactor Coolant System shall be demonstrated OPERABLE.

a. Deleted.

b. At least once per 31 days by verifying that each accessible (per occupational exposure considerations) valve (manual, power operated or automatic) in each flow path that is not locked, sealed, or otherwise secured in position, is in its correct position.

c. At least once per 31 days (when makeup pump is required OPERABLE) by verifying (per occupational exposure considerations), that on recirculation flow, the makeup pump required by Technical Specification 3.1.1.1 develops a discharge pressure of greater than or equal to 1125 psig and that each pump operates for at least 15 minutes.

d. At least once per 31 days by verifying (per occupational exposure considerations), that on recirculation flow, the decay heat removal pump required by Technical Specification 3.1.1.1 develops a discharge pressure of greater than or equal to 151 psig and that each pump operates for at least 15 minutes.

e. Deleted.

f. At least once per 7 days when valve DH-V1 or DH-V171 is open by verifying that the makeup pump electrical power supply circuit breakers are "racked out."

g. At least once per 7 days by:

   1. Deleted.

   2. Verifying the boron concentration in the BWST is between 3000 and 4500 ppm.

   3. Deleted.

   4. Verifying the contained borated water volume of the BWST is at least 100,000 gallons.

   5. Deleted.

SURVEILLANCE REQUIREMENTS

BORON INJECTION (Continued)

h. At least once per 24 hours by verifying the BWST temperature is at least 50°F when the outside air temperature is less than 50°F.

i. At least once per 12 hours (when system is in operation) by verifying that the standby reactor coolant system pressure control system:

   1. Surge tank water volume is filled to between 55% and 80% of tank capacity and the tank is pressurized to the operating RCS pressure ± 25 psig but not higher than 600 psig.

   2. Isolation valves on the discharge side of the water filled tank nearest the reactor coolant system are open.

   3. The in-service nitrogen supply bank is pressurized to between 225 and 400 psig.

j. At least once per 7 days by verifying that the standby reactor coolant system pressure control system water filled tanks, the surge tank, and the degassed water supply tank contain borated water with:

   1. A boron concentration of between 3000 and 4500 ppm.

   2. A dissolved gas concentration of less than 15 scc/kg of water.

k. At least once per 31 days by verifying that the standby reactor coolant system pressure control system isolation valve on the discharge side of the water filled tank nearest the reactor coolant system closes automatically on a tank low level test signal.

4.7 PLANT SYSTEMS

4.7.1 FEEDWATER SYSTEM

4.7.1.1 Deleted.

4.7.1.2 Deleted.

4.7.1.3 The "B" steam generator closed loop cooling system shall be demonstrated OPERABLE at least once per 31 days by starting (unless already operating) the pump and verifying a flow rate of at least 2000 gpm when operating in the recirculation mode.

4.7.2 SECONDARY SERVICES CLOSED COOLING WATER SYSTEM

4.7.2.1 The secondary services closed cooling water system shall be demonstrated OPERABLE at least once per 31 days by verifying that each of the three pumps start and operate (unless already operating) for at least 15 minutes.

4.7.3 CLOSED CYCLE COOLING WATER SYSTEM

NUCLEAR SERVICES CLOSED CYCLE COOLING SYSTEM

4.7.3.1 Each nuclear services closed cycle cooling water loop shall be demonstrated OPERABLE:

a. At least once per 31 days by:

1. Verifying that each pump starts and operates (unless already operating) for at least 15 minutes and that during pump operation: the "A" pump develops a differential pressure of at least 62.1 psid, the "B" pump develops a differential pressure of at least 63.1 psid, and the "C" pump develops a differential pressure of at least 64.1 psid.

2. Verifying that each accessible (per occupational exposure considerations) valve (manual, power operated or automatic) servicing safety related equipment that is not locked, sealed or otherwise secured in position, is in its correct position.

b. At least once per 92 days by cycling each testable valve in the flow path through at least one complete cycle of full travel.

DECAY HEAT CLOSED COOLING WATER SYSTEM

4.7.3.2 The decay heat closed cooling water loop required by Technical Specification 3.7.3.2 shall be demonstrated OPERABLE:

a. At least once per 31 days by:

1. Verifying the required pump starts and operates (unless already operating) for at least 15 minutes and that during pump operation: (a) the "A" pump develops a differential pressure of at least 30.6 psid at a flow of 2491 gpm and (b) the "B" pump develops a differential pressure of at least 30.6 psid at a flow of 2527 gpm.

2. Verifying that each accessible (per occupational exposure considerations) valve (manual, power operated or automatic) servicing safety related equipment that is not locked, sealed or otherwise secured in position, is in its correct position.

b. At least once per 92 days by cycling each testable valve in the flow path through at least one complete cycle of full travel.

MINI DECAY HEAT REMOVAL SYSTEM (MDHRS)

4.7.3.3 The MDHRS shall be demonstrated OPERABLE by performing inservice tests of each MDHRS pump and each MDHRS valve in the flow path in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR 50, Section 50.55a(g), except where specific written relief has been granted by the Commission pursuant to 10 CFR 50, Section 50.55a(g)(6)(i).

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of	)	
	)	
METROPOLITAN EDISON COMPANY, ET AL.	)	Docket No. 50-320 OLA
	)	
(Three Mile Island Nuclear Station, Unit 2)	)	

CERTIFICATE OF SERVICE

I hereby certify that copies of, "NRC STAFF NOTICE OF ISSUANCE OF AN AMENDMENT OF ORDER AND MOTION TO CONFORM PROPOSED TECHNICAL SPECIFICATIONS IN ACCORDANCE THEREWITH," dated November 20, 1980, in the above-captioned proceeding, has been served on the following, by deposit in the United States mail, first class, or, as indicated by an asterisk, through deposit in the Nuclear Regulatory Commission's internal mail system, this 20th day of November, 1980:

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