Met-Ed / GPU

Metropolitan Edison Company
Post Office Box 480
Middletown, Pennsylvania 17057
717 944-4041

Writer's Direct Dial Number

July 31, 1980
TLL 372

TMI Program Office
Attn: B. Snyder, Director
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Sir:

Three Mile Island Nuclear Station, Unit II (TMI-2)
Docket No. 50-320
Operating License No. DPR-73
Technical Specification Change Request No. 24

Enclosed are three signed originals (sixty conformed copies sent separately) of Technical Specification Change Request No. 24 requesting amendment to Appendix A of Operati- License No. DPR-73.

Also enclosed is one signed copy of Certificate of Service for proposed Technical Specification Change No. 24 to the chief executives of the township and county in which the facility is located.

Sincerely,

R. C. Arnold
Sr. Vice President

RCA:lh

Enclosures: 1) Technical Specification Change Request No. 24

2) Certificate of Service for Technical Specification Change Request No. 24

3) Check No. 011076

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METROPOLITAN EDISON COMPANY
JERSEY CENTRAL POWER & LIGHT COMPANY

AND

PENNSYLVANIA ELECTRIC COMPANY

THREE MILE ISLAND NUCLEAR STATION UNIT II

Operating License No. DPR-73
Docket No. 50-320
Technical Specification Change Request No. 24

This Technical Specification Change Request is submitted in support of Licensee's request to change Appendix A to Operating License No. DPR-73 for Three Mile Island Nuclear Station Unit 2. As a part of this request, proposed replacement pages for Appendix A are also included.

METROPOLITAN EDISON COMPANY

By R.C. Arnold
Sr. Vice President

Sworn and subscribed to me this 31st day of July, 1980.

George J. Troffer
Notary Public

GEORGE J. TROFFER
Notary Public, Reading, Berks Co.
My Commission Expires Jan. 25, 1982

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UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

IN THE MATTER OF

DOCKET NO. 50-320
LICENSE NO. DPR-73

METROPOLITAN EDISON COMPANY

This is to certify that a copy of Technical Specification Change Request No. 24 to Appendix A of the Operating License for Three Mile Island Nuclear Station Unit 2, has, on the date given below, been filed with the U.S. Nuclear Regulatory Commission and been served on the chief executives of Londonderry Township, Dauphin County, Pennsylvania and Dauphin County, Pennsylvania by deposit in the United States mail, addressed as follows:

Mr. Donald Hoover, Chairman
Board of Supervisors of
Londonderry Township
R. D. #1, Geyers Church Road
Middletown, Pennsylvania 17057

Mr. John E. Minnich, Chairman
Board of County Commissioners
of Dauphin County
Dauphin County Court House
Harrisburg, Pennsylvania 17120

METROPOLITAN EDISON COMPANY

By [signature]
Sr. Vice President

Three Mile Island Nulcear Station, Unit II (TMI-2)
Operating License No. DPR-73
Docket No. 50-320

Technical Specification Change No. 24

The licensee requests that the attached changed 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, B 3/4 7-1a, be substituted for or added to the existing Technical Specifications.

Reason for Changes

The Reactor Coolant System is currently in a natural circulation heat removal mode with heat rejection through the "A" steam generator. The "A" steam generator is steaming to the condenser through the turbine by-pass valve. With the passage of time and the associated reduction of the decay heat generation rate, natural circulation exhibits longer time periods between "burps" while maintaining adequate core cooling.

In order to discontinue use of OTSG "A" for removal of reactor decay heat, the licensee proposes the implementation of the Mini Decay Heat Removal System (MDHR System). Removal of reactor decay heat by this method would:

1. Permit the removal from service of several major BOP systems and equipment, such as:

   a. Circulating water system.
   b. Main steam system and the "A" steam generator.
   c. Condensate pumps.
   d. Condensate and feedwater systems.
   e. Main condenser.

2. Permit the removal of reactor decay heat with a system that is more appropriately sized to the present and future decay heat generation rates.

To facilitate operation of the MDHR system as the primary heat removal system, the Technical Specifications must be amended to reflect MDHR system usage.

With the MDHR system, Long Term "B" and "Loss to Ambient" cooling modes available, many of the components/systems used in the present cooling mode will no longer be necessary for cooling the core. These systems/components include portions of the Feedwater and Condensate System and the Emergency Feedwater pumps.

The Feedwater/Condensate System including the Emergency Feedwater pumps are no longer needed; therefore, the Licensee requests that they be removed from the Technical Specifications.

Additionally, the licensee requests that the Technical Specifications relating to the Makeup Pumps be removed. The Makeup Pump is designed to inject Makeup and seal injection fluid into the Reactor Coolant System (RCS) at normal (2150 psig) pressures. The ultimate mode for TMI-II heat removal will have the RCS at 100 psig, being cooled by a low design pressure system (MDHR system, 235 psig).

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The Makeup Pump should be removed from service and the Technical Specifications altered to remove Makeup Pump Operability Requirements. The pump is capable of developing a discharge pressure of about 2800 psig which is mugh higher than the MDHR system design pressure of 235 psig. Therefore, inadvertant operation of the Makeup Pump would threaten MDHR system integrity.

The Licensee plans to retain the option to operate the pump (i.e., for degassing) but plans to eliminate the current focus of the Technical Specifications upon them as a "Backup System Component".

480 Volt Busses 2-35, 2-45, 2-36, 2-46, 2-38, and 2-48, and 4160 Volt Bus 2-5 do not provide electrical power to any electrical loads required to support any of the TMI-II core cooling modes: therefore, the Licensee requests they be removed from the Technicial Specifications.

Safety Evaluation Justifying Change

Offsite radiological exposure estimates for a worst case postulated accident have been completed based on the radionuclide activity concentrations present in the RCS on July 24, 1979. This calculation employed the following assumptions:

1. MDHR System isolation occurs at the inlet and outlet isolation valves.
2. The fluid contained in the MDHR System (assumed to be 1200 gallons), is dumped to the auxiliary building floor.
3. No flashing of coolant occurs.
4. Source terms considered in this evaluation are Iodine, Xenon, and Krypton at RCS concentrations that existed on July 24, 1979.
5. Airborne release to the atmosphere occurs as a puff release with no cleanup or partitioning. No other isotopes are released.
6. Atmospheric dispersion values used in the analysis are those in Table 2.3-7, TMI-II FSAR (ground level releases).

Results of this analysis indicate a whole body radiological exposure dose at the site boundary of 1.16×10^{-4} rem. The off-site exposure estimate has been updated to reflect current RCS activity concentrations. The concentration of various radioisotopes in the RCS has decreased since July 24, 1979. Both Iodine and Xenon isotopic concentrations are insignificant.

In this updated estimate, the following assumptions have been employed:

1. - Same as above -
2. - Same as above -
3. - Same as above -
4. Source terms considered in this estimate are particulate radioisotopes (primarily Cesium and Strontium) at RCS concentrations that presently exist.
5. Airborne release to the atmosphere occurs as a puff release with 1% partitioning.
6. - Same as above -

The results of this analysis indicate that a total of approximately 12 Ci would be released to the Auxiliary Building atmosphere as a result of this postulated, though improbable, accident. This potulated release to the Auxiliary Building atmosphere presents insignificant effects at the site boundary.

utilizes natural circulation of reactor coolant through OTSG "B" and forced circulation of the secondary side of the steam generator). When in LTC "B" the reactor decay heat is transferred to the Secondary Services Closed Cooling Water System, and then to Nuclear Services River Water. In addition to LTC "B", steaming OTSG "A" to the condenser, with natural circulation of the reactor coolant through it, is an additional reactor decay heat removal method until the condenser is taken out of service.

In the case where MDHR System heat removal capability is lost, reactor coolant bulk temperature will initially increase at a rate of approximately 0.4°F/hour with ≈335,000 BTU/hour core heat input. Our analysis indicates that this temperature increase will gradually diminish such that at approximately 190°F reactor coolant heat up rate will be approximately zero. At this RCS temperature, heat loss to ambient is equal to the decay heat generation rate. Assuming that the time required to shift cooling modes from MDHR to forced circulation of the secondary side of OTSG "B" is ≈72 hours, reactor coolant system temperature could rise as much as 30°F. Current RCS temperature in the "A" hot leg is approximately 145°F. If, in this transient, the RCS temperature rises to approximately 175°F in the hot leg, substantial margin to the saturation temperature exists because the RCS, and the MDHR System, is operated at 100 psig RCS pressure.

MDHR System Overpressurization:

The MDHR system design pressure is 235 psig. Consideration has been given to possible sources of overpressurization of the system. The potential sources are defined to be:

a. Makeup pump operation with MDHR in service.

b. Standby Pressure Control (SPC) System failure.

c. Pressurizer heater operation.

The attached Proposed Technical Specification Change Request deletes the requirement to have an operable makeup pump in service. In order that makeup pump operation with the MDHR system in service can be precluded, the makeup pump breakers will be "racked out". Furthermore, prior to operation of a makeup pump, the MDHR system will be taken out of service and isolated from the RCS. This isolation will provide assurance that the potential for MDHR system overpressurization from makeup pump operation is minimized.

Failure of the Standby Pressure Control System to control the pressure of the RCS is possible, though remote. To preclude the remote possibility of overpressurization of the MDHR system from the Standby Pressure Control System, the SPC N_2 bank pressure will be reduced from the present operating range of "1500 psig to 2400 psig" to a new operating range of "225 psig to 400 psig." This reduction in N_2 bank pressure removes the possibility of MDHR system overpressurization from this source, yet provides the capability for makeup to the RCS even in the event of the most severe cooldown rate in the RCS.

Pressurize heater operation while in a "solid" operating mode on the RCS, does not create the potential for overpressurization of the MDHR system. Pressurizer heater operation, even though the MDHR system is in service, will cause a volumetric expansion of the RCS through the mechanism of heat input into the coolant. Electrically energizing all pressurizer heaters (1638 kw) will result in a volumetric expansion of the RCS that will require a relief capacity of 8.6 gpm. Since each MDHR system relief valve capacity is rated at 53.5 gpm flowrate, pressurizer heater operation is not considered to be a viable overpressurization source for the MDHR system. Furthermore, present operation of the pressurizer heaters is performed such that all banks of heaters are out of service except one (1) bank.

ALARA Considerations:

It is recognized that the MDHR system and the facilities in which it is housed have potential to become high radiation areas. In order that the guidance of Regulatory Guide 8.8 could be applied to the system and facilities, the following design considerations have been incorporated:

1. Remote flushing connections have been provided throughout the system to permit flushing of components prior to the performance of maintenance functions on that component. Flushing with non-contaminated water allows the washing away of radioactively contaminated materials from the internals of the components, and hence, a reduction of the radiation field produced prior to the performance of operational or maintenance evolutions.

2. Radiation shielding has been provided throughout MDHR system for the equipment and piping to minimize the radiation exposure to personnel. Shielding between the MDHR pumps has been provided for the reduction of personnel exposure during pump maintenance. Reach rods to permit remote valve operation have been provided for manually-operated valves that will enable the reduction of radiation exposure to operations personnel.

3. The MDHR system is provided with an inlet filter to protect the pumps from debris that may be entrained in the influent flow stream. This filter, because of the possibility of radioactively contaminated loading, is shielded to minimize area radiation levels in the vicinity of the filter.

4. The MDHR system employs welded connected to the maximum extent possible to minimize system leakage. Where the possibility of leakage exists, such as at pump seal leak-offs, flanges, and valve stems, provisions have been madefor the containment of the leakoff. This design consideration will permit minimizing the radioactive contamination in the vicinity of the MDHR components. As a result, personnel contamination potential is minimized.

5. Operation and observation of the MDHR system operating parameters is performed from a panel that is located in a low radiation area. This design philosophy permits the infrequent operational adjustments or the gathering or operating data to be performed in a reasonably low radiation level area.

An analysis of expected occupational exposures, as a result of typical MDHR system operational and maintenance functions has been performed. The results are listed below:

1.	Valve operator replacement	3.12 Man-Rem
2.	MDHR system filter replacement	1.84 Man-Rem
3.	MDHR system heat exchanger change-over	0.75 Man-Rem
4.	Pump seal replacement	4.26 Man-Rem
5.	Instrument replacement	2.79 Man-Rem

The analysis is based on radionuclide concentrations in the RCS of 57 μCi/cc due to Cs-137 and 11 μCi/cc due to Cs-134. No plate out sources were considered.

It is concluded that the above expected occupational exposures are acceptable and that the concept of "as low as reasonably achieved" has been considered and complied with.

The removal from service and the Technical Specification changes concerning the RC pump, the makeup pump, the Emergency Feedwater pump, Feedwater/Condensate System, and the listed 480 B Busses, do not constitute an unreviewed safety question.

Amendment Class

The Licensee has determined that because the amendment request involves a single safety concern, it represents a Class III License Amendment (per 10 CFR 170.22). Therefore, enclosed please find a check in the amount of $4,000.00.

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. Delete
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.

3.4 REACTOR COOLANT SYSTEM

REACTOR COOLANT LOOPS

3.4.1 The Reactor Coolant System shall be operating 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.

DELETE

Figure 3.1-1 Minimum Boric Acid Tank Contained Volume as a Function of Stored Boric Acid Concentration

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 system to OPERABLE status within 72 hours.

3.7.2 SECONDARY 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 At least one mini decay heat removal pump and heat exchanger shall be OPERABLE.

APPLICABILITY: When fuel is in the reactor pressure vessel.

ACTION:

With no mini decay heat removal pump and/or heat exchanger OPERABLE, restore the inoperable pump and/or heat exchanger to OPERABLE status within 24 hours. Should the inoperable pump and/or heat exchanger not be rendered OPERABLE within 24 hours, bring a backup cooling system, into operational readiness within 48 hours

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 river water loop inoperable, restore the inoperable loop to OPERABLE status within 72 hours.

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, 2-32, 2-41, and 2-42
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 with 8 hours.

3/4.4.1 REACTOR COOLANT LOOPS

Removal of reactor decay heat will be accomplished by the use of the Mini Decay Heat Removal System. Operation of the Reactor Coolant Loops in the natural circulation mode will be a backup mode of heat removal.

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 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 sufficient high value to permit operation of the reactor coolant pumps.

3/4.7 PLANT SYSTEMS

BASES

3/4.7.1 FEEDWATER SYSTEM

The "B" Steam Generator is required to be maintained in an OPERABLE status since it is a backup means of removing the 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 mini decay heat removal system (MDHR) since it provides the heat sink for the MDHR System.

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

The mini decay heat removal system is required to be maintained in an OPERABLE status to enable the removal of the decay heat from the reactor.

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.