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August 5, 1980
TLL 382

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

Our letter to you, TLL 372, dated July 31, 1980, transmitted Technical Specification Change Request No. 24 requesting amendment to Appendix A of Operating License No. DPR-73.

During development of this document, several lines of information were inadvertently omitted from the section "Safety Evaluation Justifying Change". To rectify this omission we are enclosing three signed originals (sixty conformed copies sent separately) of the preamble to the Technical Specification Change Request. Also enclosed is one signed copy of Certificate of Services for Proposed Technical Specification Change Request No. 24 to the chief executives of the township and county in which the facility is located.

We regret this error.

Sincerely,

R. C. Arnold
Sr. Vice President

RCA:lh

Enclosures: 1) Technical Specification Change Request No. 24 (Preamble)

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

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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.

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. This submittal is made to correct our previous submission in which a portion of the "Safety Evaluation Justifying Change" was inadvertently omitted.

METROPOLITAN EDISON COMPANY

By R C Arnold
Sr. Vice President

Sworn and subscribed to me this 5th day of August, 1980.

George J. Troffer
Notary Public

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

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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).

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.

Reactor Core Cooling:

Loss of MDHR system heat removal capability is not significant. The eventual backup modes of core cooling are either "Loss to Ambient" or LTC "B" (which 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 the 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 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 operaton 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 "<u>1500 psig to 2400 psig</u>" to a new operating range of "<u>225 psig to 400 psig</u>." 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.

Pressurizerheater 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 of 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 connections 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 made for 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 of 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.