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

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U.S. NUCLEAR REGULATORY COMMISSION

Office of Inspection and Enforcement
Attn: B. H. Grier, Director
Region I
U. S. Nuclear Regulatory Commission
631 Park Avenue
King of Prussia, PA 19406

Dear Sir:

Three Mile Island Nuclear Station, Unit 2 (TMI-2)
Operating License No. DPR-73
Docket No. 50-320
10CFR 50.59 Report for 1980

In accordance with the requirements of 10CFR 50.59, enclosed please find two copies of changes to systems and procedures in TMI-2 for 1980 as described in the FSAR. Also included is a summary of tests and experiments performed that were not described in the FSAR.

Sincerely,

/S/ G. K. HOVEY

G. K. Hovey
Vice President and Director, TMI-2

GKH:RBS:be

Enclosures

cc: Director, Office of Inspection and Enforcement (40 copies)
U. S. Nuclear Regulatory Commission
Washington, D.C. 20555

Director, Office of Management Information & Program Control (1 copy)
U. S. Nuclear regulatory Commission
Washington, D.C. 20555

J. F. Stoltz
B. J. Snyder
L. Barrett

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FACILITY MODIFICATIONS

During the reporting period a number of facility modifications and projects were undertaken were that recovery oriented. These changes have been subject to numerous in-house and NRC review sessions and NRC approval has been received prior to implementation in accordance with Tech Spec. 6.8.2. A summary of these recovery oriented modifications along with references to key correspondence are listed below. Where applicable, the NRC approval letter is referenced.

Reactor Building Purge - Memorandum and Order, dated June 12, 1980 Order for Temporary Modification of License, June 12, 1980, B. J. Snyder to R. C. Arnold.

Mini Decay Heat Removal System: B. J. Snyder letter to R. C. Arnold, Amendment of Order, dated November 14, 1980.

Processed Water Storage Tanks: J. T. Collins memo to J. J. Barton September 13, 1979 Comments on Design Criteria. G. K. Hovey to L. H. Barrett, 4/1/81 LL2-81-0075, response to NRC/TMI-81-0016 2/27/81. TLL 029, 1/24/80, R. F. Wilson to J. T. Collins. TLL 395-10/09/80, G. K. Hovey to J. T. Collins Response to NRC/TMI-80-026, 2/1/80, J. T. Collins to R. C. Arnold.

Penetration 401 Modification: L. H. Barrett letter to G. K. Hovey NRC/TMI-81-009, approval of revised design criteria for modification of Reactor Building Penetration 401, Dated February 11, 1981.

BOP Diesel Generator - B. J. Snyder letter to R.C. Arnold dated August 11, 1980, modification of Order deleting operability requirements for the BOP diesel generator.

Nuclear Sampling System - Operational June 16, 1980 - In Quarterly Report June 1980. Conference Notes 6-5-79, TMI Trailer 105 (J. T. Collins attended). Interoffice Memorandum transmitting approved design criteria for the sampling system, September 14, 1979 (J. T. Collins signed).

Airlock Contamination Control Facility - J. T. Collins to R. C. Arnold, January 30, 1980, NRC/TMI-80-017 - Review of Facility Design Description. R. F. Wilson to J. T. Collins, March 4, 1980, TLL-103 - Response to above. Completion of this item was a prerequisite for initial containment entry. This procedure was signed off by the NRC in March, 1980.

The above modifications include a number of Engineering Change Memos (ECMS).

Other changes made to the facility that were not related to specific recovery projects are listed below by ECM number. The ECM number is a number assigned to each modification as a tracking mechanism.

Change ECM #584

Description - The subject change installs a temporary flange in the Cask Storage Borated Water Pool piping so that during recovery, the Unit 2 BWST water can be transferred to either the Unit I cask pool or EPICOR I. After recovery, the system will be restored to the original configuration.

Safety Evaluation - The subject modification is intended to be a temporary modification during the recovery mode. The installation of the flange does not reduce the integrity of the system. The modification is installed under B31.7 subsection class III and Quality Control level 3. The modification meets seismic catagory I design criteria. Any failure mode is bounded by previous analysis. Therefore the change modification does not increase the probability of occurrence or consequence of an accident analyzed in Chapter 14 of the FSAR, nor does the change introduce the possibility of a new type of accident other than any previously analyzed in Chapter 14 of the FSAR. The margin of safety as defined by the Technical Specification bases is not reduced. Therefore, it is concluded that the subject change does not involve an unreviewed safety question.

Change ECM #715

Description - This change installs lines from the EPICOR II system and the SDS system to the processed water storage tanks. This feeds the effluent from these systems to the processed water storage tanks for storage after processing.

Safety Evaluation Summary - The above changes will be handling only processed water from the EPICOR II and SDS system. Isotopic content of the water will be tested prior to release to the storage tanks. System piping is designed and installed to ASME B31.1 standards. Since the specific activity of the processed water is very low, it is felt that the system does not reduce margins of safety or increase the probability or consequence of an accident as previously described. Therefore this change does not constitute an unreviewed safety question.

Change ECM #606

Description - This change modifies the Fuel Handling Building crane by the addition of inching drives. These drives provide fine control of the crane for use with the SDS.

Safety Analysis Summary - The inching drives will be provided power from BOP sources and therefore will have no effect on nuclear safety power supplies. There is no fuel in the Unit II fuel pool to require restriction of crane movement. Administrative controls will be added to prevent movement of large loads over critical areas of the SDS and SPC systems. Since there is fuel in the Unit I fuel pool, the inching drives should be tagged to prevent their use in Unit I. With the use of administrative procedures the change does not increase the probability or consequences of an accident previously described nor does it create the possibility of an accident of a different type than previously analyzed in the FSAR. The change does not affect the margin of safety as defined in the Technical Specification Bases. It is concluded this change does not constitute an unreviewed safety question.

Change ECM #815

Description - This change installs a motor operated valve (WG-V-100) downstream of valves BS-V-156A and BS-V-157A on BS-P-1A drainline. This valve will allow remote draining of BS-P-1A. Under normal circumstances BS-V-156A and BS-V-157A are closed. In emergency situations, WG-V-100 will be closed and then BS-V-156A and BS-V-157A will be opened prior to commencing pumping. When pumping is halted WG-V-100 will be opened remotely to allow draining of the lines to the BS-P-1A vault sump. The valves will then be flushed and placed in normal lineup.

Safety Analysis Summary - Under normal conditions, isolation valves BS-V-156A and BS-V-157A remain in a closed condition. These valves are only opened during emergency conditions requiring draining of the Reactor Building sump. At this time WG-V-100 is closed and remains closed until pumping operations are completed. Valve WG-V-100 is then opened to provide remote draining of the WG-P-1 pump system. This reduces radiation exposures to workers that would otherwise result during manual draining. The primary safety concern for this modification is leakage from valve WG-V-100 when BS-V-156A and BS-V-157A are open. Any leakage from valve WG-V-100 is routed to the BS-P-1A sump via floor drains. Ventilation to the cubicle can be isolated, reducing airborne releases. Any leakage is bounded by accidents previously analyzed in the FSAR. It is therefore felt that this change does not constitute an unreviewed safety question.

Change ECM #649

Description - This change installs a catch system to provide drainage from the BWST enclosed area to the Auxillary Building sump. The enclosed areas consist of a concrete base pad, steel frame work and fiberglass roofing and siding. Each enclosure has a single floor drain with two inch stainless steel piping connecting the enclosure drains to the four inch BWST drain line. The four inch drain line runs to the Auxiliary Building sump.

Safety Analysis Summary - The enclosures are designed to be Seismic Category I with the base pad at least six inches above ground level. The rest of the enclosure is steel framing, metal doors, and fire resistant fiberglass panels. The building is moisture tight to the extent that all seams are caulked except a few conduit and piping penetrations which will be sealed. The drain system is not Seismic I since it ties to the BWST drain line below the isolation valve. The fire barrier penetration has been examined to determine if the safety margin of fire protection is decreased. It has been determined that fire protection has not been decreased. The possibility for a previously undescribed accident has not been increased and the margins of safety as defined in the Technical Specification Bases has not been reduced. It is concluded that this change does not constitute an unreviewed safety question.

Change ECM #704

Description - This change installs a four inch domestic water branch line from the three inch supply header to the search entry facility, administration building, TLD facility and the security administration building. The change provides domestic water service to the above buildings.

Safety Analysis Summary - The change deals only with the domestic water supply. There is no impact on nuclear safety related equipment. There is no effect on the safety analysis as described on the FSAR and no change to the bases of the Technical Specifications. It is concluded that this change does not constitute an unreviewed safety question.

Change ECM #622

Description - This change provides for a slight realignment of some pipe support mounting plates in the Fuel Handling Building. During original installation, the holes drilled for mounting encountered rebar. This change provides for readjustment of plate position to facilitate installation.

Safety Analysis Summary - The relocation of the mounting brackets is bounded by the pipe support analysis. There is no effect on the safety margins, Technical Specification Bases or accident analysis. It is concluded this change does not constitute an unreviewed safety question.

Change ECM #628

Description - This modification installs a temporary vent in the Nuclear Nitrogen System line between NM-V40 and NM-R9, a relief valve test connection. This change will pressurize the pilot section of NM-R9 which will prevent an inadvertent lifting and discharge of NM-R9 to the Auxilary Building environment when the system pressure is raised from 0 psig to operating pressure.

Safety Analysis Summary - This modification will prevent an inadvertent lifting of NM-R9 during system pressurization. Since the system is radioactively contaminated and the discharge is to the Auxilary Building environment, preventing this release is in the interest of the public health and safety. The modification does not prevent the system from functioning normally during periods of overpressure. Since the system still functions as intended, this modification does not change any safety analysis or bases. Therefore, this change does not constitute an unreviewed safety question.

Change ECM #552

Description - This change removed HP-R-215 from the spent fuel bridge when the bridge was disassembled for installation of the SDS system. HP-R-215 was an area radiation monitor used as a criticality alarm.

Safety Analysis Summary - The removal of the spent fuel bridge and the associated area radiation monitor HP-R-215 was necessary for installation of SDS equipment. HP-R-215 served as a criticality monitor. Since no spent fuel will be stored in this area during SDS operation and since the SDS system will have its own radiation monitoring system, removal of this instrument will not degrade nuclear safety. Removal of this alarm will not increase the probability or consequence of a previously analyzed accident, will not create a new catagory accident or will not affect the bases of the Technical Specification. Therefore it is concluded that this change does not constitute an unreviewed safety question.

Change ECM #645

Description - This change modifies the Auxiliary Building and Fuel Handling Building exhaust filter housing and ductworks. This will allow the installation of injection and sample ports for DOP and Freon Testing.

Safety Analysis Summary - The modifications ensure that proper testing of the filters can be accomplished per Reg. Guide 1.52 and ANSI N510-75. This enhances environmental safety. Nuclear safety is not adversely affected by this change. The margins of safety discussed in the FSAR or Technical Specification Bases are not affected. Therefore, this change does not involve an unreviewed safety question.

Change ECM #765

Description - This change installed a jumper between terminals A6 and A7 on all 4160V and 6900V "69" switches. The jumper eliminates a local switch that is not alarmed or locked. This would eliminate the possibility of this switch being in the wrong position and thereby not being able to perform its intended safety function. The permissive function of this switch is also provided by racking in or out the associated breaker.

Safety Analysis Summary - This modification eliminates a generic concern addressed in I & E Bulletin 80-20. The change eliminates the potential for an unalarmed, unlocked switch being in a wrong position, thereby disabling its safety function. This enhances nuclear safety. It is concluded that this mod does not change any present accident analysis, modify Technical Specification bases, or introduce a new type of accident. Therefore this modification does not involve an unreviewed safety question.

Change ECM #572

Description - The subject modification replaces a gas monitor liner assembly in HP-R-229 with a High Range Noble Gas Monitor assembly. This change provides the capability of measuring noble gas Kr 85 concentrations up to 10^3 uCi/cc during the Reactor Building Purge.

Safety Analysis Summary - This modification expands the capability of the present monitor in HP-R-229 to detect noble gases. The change enhances nuclear safety by extending the measurement range of gaseous effluents. The change will not affect the accident analysis as described in the FSAR, nor does it affect the bases of the Technical Specification. Therefore it is concluded that the change does not involve an unreviewed safety question.

Change ECM #817

Description - This change installs a temporary line from the Reactor Building basement area to the Fuel Pool Waste Storage system. This system will be used to transfer water from the Reactor Building surface suction pump to a storage area to facilitate cleaning the Reactor Building basement.

Safety Evaluation Summary - This change installs a temporary line for pumping out the Reactor Building basement. It is classified as a liquid radwaste system and as such Reg. Guide 1.143 is imposed. The failure of the piping has been analyzed. It was determined that any breaks are enveloped by other postulated events as described in the FSAR. The change does not affect the bases for any Technical Specification and as such the margin of safety is not reduced. Therefore, it is concluded that the subject change does not involve an unreviewed safety question.

PROCEDURE CHANGES

With the issuance of the Interim Recovery Technical Specifications, many procedures issued for surveillance under the Operating Technical Specifications became unnecessary. A large percentage of these procedures were not capable of being performed due to inaccessability. Other procedures were unnecessary due to the current operational mode. These procedures were then cancelled and where necessary alternate surveillance procedures were issued under the guidance of Recovery Technical Specification Section 6.8.2. All procedures that were cancelled received PORC review to determine any impact on safety prior to cancellation. Due to the subject matter, some of these procedures received NRC review prior to cancellation.

Additionally, in support of the recovery effort, a number of procedure changes were made and new procedures issued. As required by the Recovery Technical Specifications, Section 6.8.2, these recovery related procedures received NRC review and approval prior to implementation. Since these procedures have received NRC approval, they will not be discussed further in this report.

During the reporting period procedure changes were made to reflect changes to the Unit 2 Recovery organization, communications, facilities and practices. These changes did not materially change the overall emergency response. Therefore it is felt these changes as listed below, do not constitute an unreviewed safety question.

Procedure No.	Title	PCR Number	Approval Date
1670.1	Local Emergency	2-80-448	10/10/80
1670.2	Site Emergency Procedure	2-80-449	10/10/80
1670.3	General Emergency Procedure	2-80-450	10/10/80
1670.5	On-Site Radiological Monitoring	2-80-456	10/10/80
1670.6	Off-Site Radiological Monitoring	2-80-457	10/10/80
1670.12	Emergency Readiness Check List	2-80-467	10/13/80
1670.9	Emergency Training and Emergency Exercise	2-80-460	10/10/80

Also changed was procedure 1670.4 as described below.

Change PCR 461-1670.4, Radiological Dose Calculation - 10/16/81.

Description of Change - The procedure was updated to reflect effluent source terms which Unit 2 would have in an emergency. It also was updated with respect to effluent monitors in use during the recovery period.

Safety Evaluation Summary - This change does not govern equipment operations and is administrative in nature. It has no impact on nuclear safety and is therefore not considered an unreviewed safety question.

TESTS AND EXPERIMENTS

During the past year a number of activities revolving around containment entries occurred. These activities included sampling of Reactor Building environment, decontamination tests, tests on resin column performance, and other tests and programs related to the cleanup effort for the Reactor Building. These activities were accomplished using procedures reviewed and approved under the guidance of Recovery Technical Specification 6.8.2. These activities have been also described in the Recovery Quarterly Reports. Since these activities have received NRC review and approval prior to initiation they will not be discussed further in this report.