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March 3, 1982
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Office of Inspection and Enforcement
Attn: Mr. Ronald C. Haynes, Director
Region I
U. S. Nuclear Regulatory Commission
631 Park Avenue
King of Prussia, Pennsylvania 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 1981

In accordance with the requirements of 10CFR 50.59, enclosed please find one (1) signed original and thirty-nine (39) copies of a description to change to facility systems and procedures in TMI-2 for 1981 as described in the TMI-2 FSAR. Also included is a summary of tests and experiments performed that were not as described in the FSAR.

Sincerely,

J. J. Barton
Acting Director, TMI-2

JJB:RBS:djb

Attachments

cc: Mr. Richard C. DeYoung, Director, Office of Inspection and Enforcement
Mr. J. F. Stoltz, Chief, Operating Reactors Branch
Dr. B. J. Snyder, Program Director, TMI Program Director
Mr. L. H. Barrett, Deputy Program Director, TMI Program Office

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FACILITY MODIFICATIONS

During the reporting period, a number of facility modifications and projects were undertaken for the recovery effort. These changes were subject to numerous inhouse and NRC review sessions and NRC approval prior to implementation in accordance with Tech Spec 6.8.2. A summary of these modifications is listed below. Where applicable, correspondence is referenced. As these items underwent the above reviews and approvals, they were not implemented without prior NRC approval as allowed by §10CFR50.59 and hence are only summarized.

1.	Submerged Demineralizer System (SDS) - SDS-EPICOR Interface	NRC Approval via NUREG 0796, dated June 1981 "Safety Evaluation Report related to the operation of the Submerged Demineralizer System at Three Mile Island Unit No. 2".
2.	ISWSF -	NRC approval via letter B. J. Snyder to G. K. Hovey, dated October 2, 1981.
3.	EPICOR Filter Shipment -	NRC approval via letter B. J. Snyder to G. K. Hovey, dated March 25, 1981, deleting requirements for solidification of liners and approval of disposal of EPICOR low level liners.
4.	Offsite ENS Sirens Installed -	NRC order
5.	Deletion of Containment Air - Temperature Indication	NRC approval via letter B. J. Snyder to G. K. Hovey, dated August 5, 1981.
6.	Simultaneous Containment Air - Lock Door Opening	NRC approval via letter B. J. Snyder to G. K. Hovey, dated September 23, 1981.
7.	Containment Penetrations - Modifications	Penetrations 401, 626 and 561 were modified during the past year. All modifications were accomplished under procedures approved prior to the modification by the on-site NRC staff.

The above modifications include a number of Engineering Change Memos (ECM's) which have previously been provided to NRC staff.

Other changes made to the facility that were not related to specific recovery tasks are listed below by ECM number. The ECM number is a tracking mechanism for review, approval and documentation.

Change ECM #649

Description - This change installed a catch system to provide drainage from the BWST enclosed area to the Auxiliary 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 have not been reduced. It was concluded that this change did not constitute an unreviewed safety question.

Change ECM #817

Description - This change installed a temporary line from the Reactor Building basement area to the Fuel Pool Waste Storage system. This system is 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 installed 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 was concluded that the subject change did not involve an unreviewed safety question.

Change ECM 822, Rev. 0, Rev 1

Description - This change initially jumpered out a failed internal thermal overload in air compressor motor DF-P-2D. The motor also had an external overload in series with the failed overload. The ECM later removed the motor and replaced it with a like motor, thus removing the jumper on the overload circuit.

Safety Evaluation Summary - This ECM did not change the safety configuration of the plant in that the jumper did not defeat a safety feature since two overloads were in the system and only one was necessary. The like-for-like replacement did not constitute a 50.59 change. Therefore, this change did not constitute an Unreviewed Safety Question (USQ).

Change ECM 835

Description - A hard pipe jumper was installed replacing an oil interceptor (SD-T-1A) in the Diesel Generator Building drain system. The oil interceptor was internally contaminated and was removed to avoid additional contamination of the drains.

Safety Evaluation Summary - The oil interceptor was redundant in that the IWTS sump, the final destination of the drain line, is designed to remove oil from the sump water. Removing this redundant feature did not effect the safety analysis or margin of safety in the bases of the Technical Specifications. Thus this change did not constitute a USQ.

Change ECM 862, REV 0, 1, and 3

Description - These changes were done to partially accomplish the NRC order for separation of Unit I and Unit II. The changes are related to the removal of sample lines run through the fuel handling building, the subsequent revalving of lines, and the installation of power supplies for remote valve indication and operation. The new installations used similar materials as in old installations. All unused lines were plugged and/or removed.

Safety Evaluation Summary - These changes were in response to an NRC order requiring separation of Units I and II. The change did not delete sample capabilities, all controls and indication are like-for-like replacements, and all cut lines were capped and valved shut with red tagged valves. Since capabilities are not decreased, this change did not affect the safety analysis or margin of safety. Thus the change did not constitute a USQ.

Change ECM 872

Description - This change tied the Whelen emergency notification site loudspeakers to the existing site Gaitronics system.

Safety Evaluation Summary - The change was designed to improve site emergency response. The change did not adversely affect equipment previously evaluated in the FSAR. Therefore, the change did not represent a USQ.

Change ECM 880

Description - This change disconnected the operating air supply in three valves (WDL-V1172, WDS-V15, and WDS-V59), leaving them in a permanently closed position. This will prohibit radwaste transfers to Unit I in compliance with an NRC order for unit separation.

Safety Evaluation Summary - The change did not open any pathways for radioactive leaks. Radioactive waste processing in Unit II will be done with Unit II equipment, thus capibilities are not decreased. The change did not affect the margin of safety determined in a previous evaluation. Thus, the change did not constitute a USQ.

Change ECM 902

Description - This change lifted two leads controlling valves DH-V6A and DH-V6B, allowing those valves to remain closed, independent of the BWST water level. This will prevent the inadvertent release of water into the reactor sump.

Safety Evaluation Summary - The elimination of a potential leak path to the Reactor Building sump helps to enhance nuclear safety. The use of the subject valves is controlled by an emergency procedure. In the event of a LOCA, sufficient time is available to allow operator action to open the subject valves allowing recirculation of the R. B. sump for core flooding. This change did not affect the capability for remote operation of these valves. Since sufficient time exists to activate the system, the margin of safety was not reduced. Therefore, this change did not constitute a USQ.

JUMPERS AND LIFTED LEADS

During 1981 GPU was informed by the NRC staff that jumpers and lifted leads would be considered modifications to the plant, and must be included in the 50.59 report. In the NRC Inspection Report 50-320/81-14 dated October 19, 1981, it was noted that improved control was necessary in the control of temporary modifications. It was also noted that we had committed to have a procedure in place to address the 50.59 review for these modifications. The procedure was committed for issue in mid January, 1982 and as such records are not available for 1981. The 50.59 evaluation for future jumpers and lifted leads will be addressed in the 1982 report.

PROCEDURE CHANGES

With the issuance of the Interim Recovery Technical Specifications, many procedures issued for surveillance under the Operating Technical Specifications became unnecessary. It was not possible to perform a large percentage of these procedures 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 6.8.1 and 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 were 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.

The remainder of the changes were reviewed and it was determined that there were no changes applicable to the 10CFR 50.59 report.

TESTS AND EXPERIMENTS

During the last year, several tests and experiments were performed in conjunction with the recovery effort. Most of these tests did not fall under the category of 50.59 subjects. The following items represent those evolutions which are considered significant or fall into the category of 50.59 items. Also listed is any appropriate correspondence.

SDS Liner Hydrogen Generation Rate

Description - The basic procedure for obtaining samples for the gas generation rate was described in SOP R-2-81-59 along with a safety evaluation for storing liners in a wet vs dewatered mode. This covers the hazards associated with hydrogen generation and sampling. This procedure was approved by on-site NRC personnel on November 17, 1981 and as such will not be discussed further here.

In Containment Resin Test

Description - This change was described in SOP R-2-81-11 and 12 approved by on-site NRC personnel on March 16, 1981 and March 19, 1981. Thus, the test will not be discussed further.

Containment Decon Test

Description - This was a large scale decon test done during Entry No. 10. The test was described in SOP R-2-81-20 approved by on-site NRC personnel on May 6, 1981. Thus, this test will not be discussed further here.

SDS Sand Filter Test

Description - This test and associated safety evaluation was previously provided to and reviewed by NRC in Septemeber and November 1981.