May 25, 1988

Docket No. 50-320

Mr. F. R. Standerfer
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Dear Mr. Standerfer:

SUBJECT: ISSUANCE OF AMENDMENT (TAC NO. 65337)

The Commission has issued the enclosed Amendment No. 30 to Facility Operating License No. DPR-73 for the Three Mile Island Nuclear Station, Unit No. 2, in response to your application transmitted by letter dated April 23, 1987 and revised by letters dated October 26, November 9, and December 4, 1987 (Technical Specification Change Request No. 53).

The amendment modifies Appendix A Technical Specifications Sections 1-Definitions, 2-Safety Limits, 3-Limiting Conditions for Operation, 3/4 Basis for Limiting Conditions for Operations and Surveillance Requirements, and 6-Administrative Controls. The Amendment extensively revises the TMI-2 Technical Specifications aligning licensing requirements to appropriate current, as well as future, plant conditions through the remainder of the current cleanup operations. The amendment allows for the transition from the current defueling phase through the completion of defueling and offsite fuel shipment by incorporating Technical Specifications that are applicable during specific phases or modes of the cleanup.

In conjunction with the issuance of this amendment, the Commission granted your request for an exemption to 10 CFR 70.24, Criticality accident requirements. This exemption, which relieves you from certain monitoring requirements once defueling has been completed and any possibility of a criticality has been precluded, is effective as of the effective date of this amendment.

A copy of the related Safety Evaluation is also enclosed. Notice of Issuance will be included in the Commission's bi-weekly Federal Register notice.

Sincerely,

John F. Stolz
John F. Stolz, Director
Project Directorate I-4
Division of Reactor Projects I/II
Office of Nuclear Reactor Regulation

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PDR ADOCK 05000320
P PDR

Enclosures:
1. Amendment No. 30 to DPR-73
2. Safety Evaluation

cc w/enclosures:
See next page

LA:PDI-4	PM:PDI-4	PD:PDI-4	OGC
SMNorris	MTM MMasnik:bd	JFStolz	SHLewis
05/24/88	05/25/88	05/ /88	05/24/88

Mr. F. R. Standerfer
GPU Nuclear Corporation

Three Mile Island Nuclear Station
Unit No. 2

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

GPU NUCLEAR CORPORATION

DOCKET NO. 50-320

THREE MILE ISLAND NUCLEAR STATION, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 30
License No. DPR-73

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by GPU Nuclear Corporation, (the licensee) dated April 23, 1987, as supplemented by letters dated October 26, 1987, November 9, 1987, and December 4, 1987, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.c.(2) of Facility Operating License No. DPR-73 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 30, are hereby incorporated in the license. GPU Nuclear Corporation shall operate the facility in accordance with the Technical Specifications and all Commission Orders issued subsequent to March 28, 1979.

3. This license amendment is effective as of its date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

John F. Stolz

John F. Stolz, Director
Project Directorate I-4
Division of Reactor Projects I/II
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance:

ATTACHMENT TO LICENSE AMENDMENT NO. 30

FACILITY OPERATING LICENSE NO. DPR-73

DOCKET NO. 50-320

Replace the following sections/pages of the Appendix A Technical Specifications with the attached sections/pages. The revised sections/pages are identified by amendment number and contain vertical lines indicating the area of change.

Remove	Insert
Cover Page	Cover Page
Index	Index
Section 1	Section 1
Section 2	Section 2
Section 3	Section 3
Section 3/4	Section 3/4
Section 6	Section 6

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 30 TO FACILITY OPERATING LICENSE NO. DPR-73

GPU NUCLEAR CORPORATION

THREE MILE ISLAND NUCLEAR STATION, UNIT NO. 2

DOCKET NO. 50-320

1.0 INTRODUCTION

By letter dated April 23, 1987, GPU Nuclear Corporation (GPUN or the licensee) requested the approval of changes to the Technical Specifications of Facility Operating License No. DPR-73 for Three Mile Island Nuclear Station, Unit No. 2. GPUN provided supporting information for the proposed modifications. The request was revised by letters dated October 26, 1987, November 9, 1987, and December 4, 1987. The proposed amendment includes revisions to the Appendix A Technical Specifications Sections 1-Definitions, 2-Safety Limits, 3-Limiting Conditions for Operation, 3/4-Basis for Limiting Conditions for Operation, and 6-Administrative Controls. The proposed amendment is an extensive revision intended to appropriately align the license requirements to the current, as well as, the projected future plant conditions as the plant progresses in stages through the remainder of the cleanup operations. The revision defines three modes of operation which correspond to key milestones in the cleanup operation. Only those license conditions applicable to the facility conditions in each mode would require implementation. Phase out of specific license conditions as the plant proceeds to successive modes would be predicated on the submission by the licensee of a report which would describe the plant conditions and give the basis for progression to the next mode.

2.0 DISCUSSION AND EVALUATION

Section 1.3

Technical Specification 1.3 defines Recovery Mode as the plant condition in which the reactor is subcritical and the average reactor coolant system temperature is less than 200°F. No other modes are defined.

The licensee proposes to modify the Technical Specification by defining three distinct facility modes corresponding to the projected plant conditions as facility cleanup progresses. The definition of Mode 1 is the same as the present definition of Recovery Mode and corresponds to the presently existing plant conditions during which defueling and other major tasks are in progress. Mode 2 will exist subsequent to the defueling of the reactor and reactor coolant system when the possibility of criticality in the reactor building is precluded and no canisters containing core material remain in the reactor building. Mode 3 will be the period following shipment of all canisters containing core material to an offsite location. Thirty days prior to an anticipated mode change, the licensee proposes to submit a report to the NRC staff providing the basis for the mode change. The staff, however, requires, and the licensee agrees, that the reports providing the basis for a mode change be submitted a minimum of sixty (60) days in advance of the anticipated transition date. Additionally the staff requires that at least 90 days prior to transition to Facility Mode 2 the licensee submit a report which evaluates the applicability of the technical provisions of 10 CFR 50 and the Appendices thereto, to TMI-2 in Facility Mode 2 and thereafter. This report shall demonstrate that necessary design features required for protection of the public health and safety exist. Although not part of their original proposal the licensee has agreed to comply with the staff's requirement.

The staff has concluded that the modes as defined in the licensee's proposal provide for suitable transition points for describing general plant conditions through the remaining phases of the cleanup program. The mode changes are planned to occur coincident with significant reductions in the levels of risk associated with the facility.

Section 1.7

Technical Specification 1.7 defines the conditions that must be met to provide Containment Integrity. It requires (1) the capability of double isolation of containment penetrations in accordance with NRC approved procedures, (2) that isolation valves inside of the containment be capable of remote operation, and (3) that penetrations without NRC approved procedures for isolation be closed with double isolation of the penetration.

The licensee's proposed revision incorporates the same requirements on isolation capability of penetrations with double barriers. However, it adds clarification on the controls that must be implemented to assure the capability to isolate those penetrations that are open for operational needs. In addition, the revision adds a requirement that Spent Fuel Pool 'A' and Fuel Transfer Canal water level be within the specified range to meet the definition of Containment Integrity.

The staff has concluded that the proposed definition adequately describes the requirements to assure a containment structure capable of preventing the release of radioactive material to the environment. Since NRC staff review of licensee procedures will no longer be required, the proposed Technical Specification has incorporated requirements for control of penetrations open for operational needs. These requirements are equivalent to those previously imposed through the NRC staff review process. In addition, it includes the added conservatism of maintaining a sufficient water level in the Spent Fuel Pool 'A' and Fuel Transfer Canal to ensure a water seal over the Fuel Transfer Tubes.

Section 1.21

The Technical Specifications do not define any level of containment closure capability other than Containment Integrity.

The licensee has proposed a new Technical Specification definition defining Containment Isolation. This would prescribe a containment functional capability less than full containment integrity and will involve a condition similar to containment integrity but with only single barrier isolation of containment penetrations.

The staff has concluded that this is an acceptable level of containment performance for the plant conditions that will exist after defueling when no potential for criticality in the reactor building exists. The containment will serve as a barrier against the spread of residual contamination where performance levels comparable to that of auxiliary building and fuel handling buildings are adequate. This is discussed further under the evaluation of sections 3.6.1.1 and 3.6.1.2.

Section 2.0

The current Technical Specifications state under Safety Limits, Reactor Coolant System Pressure, that it is "not applicable." This is the only Technical Specification under "Safety Limits."

The proposed revision would modify Technical Specification 2.0 to read, "There are no safety limits which apply to TMI-2."

The staff has concluded that this is an acceptable change since it accurately reflects the fact that there are no safety limits applicable to this facility under 10 CFR 50.36(c)(1).

10 CFR 50.36(c)(1) requires safety limits necessary to reasonably protect the integrity of certain of the physical barriers which guard against the uncontrolled release of radioactivity. Essential parameters typically limited in plants that are operational are inapplicable at TMI-2. The integrity of physical barriers is not challenged at TMI-2 to the extent that it is at operational plants since there is no source of energy to widely disperse radioactivity should the physical barriers be breached.

This represents an administrative change, since the current specification already indicates that there are no applicable safety limits.

Section 3.0.1

Technical Specification 3.0.1 states that the Limiting Conditions for Operation and Action requirements are applicable during the Recovery Mode.

The licensee proposes to modify Technical Specification 3.0.1 to state that the Limiting Conditions for Operation and Action requirements are applicable during the Facility Mode specified for each LCO and associated Action requirement.

This is consistent with the realignment of license conditions and the definitions of three different Facility Modes as previously discussed. This proposed change is acceptable to the staff.

Section 3.1.1.1

Technical Specification 3.1.1.1 prescribes the operational requirements for available sources of Borated Cooling Water Injection and the required actions when one or more sources become unavailable. It specifies applicability of the requirement during the Recovery Mode. It allows deviation from the borated water storage tank minimum volume requirements if the deviation is controlled in accordance with an NRC approved procedure.

The licensee proposes to modify this section by specifying its applicability to only Mode 1, deleting the allowance to deviate from the specification if procedurally approved, and to add the requirement to invoke the action statement if borated water storage tank temperature is outside of the specified range.

The staff has concluded that removal of the allowance to deviate from the requirement without NRC prior approval is consistent with the intent to remove the requirement for NRC staff approval of licensee procedures. This change will require the licensee to seek formal license amendment to deviate from the requirement during the mode in which it is applicable. Addition of the minimum temperature action requirement is consistent with the Limiting Conditions for Operation as previously written. Modifying the applicability to Mode 1 is consistent with new definitions of Facility Modes. Mode 1 is the only Facility Mode in which criticality or core cooling are issues. Following defueling and assuring that no potential for a criticality in the reactor building exists, there is no longer a need for borated cooling water injection capability. The staff finds the proposed changes acceptable.

Sections 3.1.1.2, 3.1.1.3, and 3.1.1.4

Technical Specifications 3.1.1.2 and 3.1.1.3 prescribe the required boron concentrations and temperature in the water in the Reactor Coolant System, Fuel Transfer Canal, and the Spent Fuel Storage Pool 'A'. These parameters must be maintained within the specified limits during the Recovery Mode.

The licensee proposes a modification that would prescribe the same limits on boron concentration and temperature in the Reactor Coolant System and Fuel Transfer Canal and would change the applicability to Mode 1. The modification would place the requirements on boron concentration and temperature for the Spent Fuel Storage Pool 'A' in a new section 3.1.1.4 with applicability to Modes 1 and 2.

The staff has concluded that the specified boron concentration and temperature in the Reactor Coolant System, Fuel Transfer Canal, and Spent Fuel Storage Pool 'A' need only be maintained when there is sufficient core material present to pose a potential threat of inadvertent criticality. After transition from Mode 1, there will be no possibility of a criticality in the reactor building per the definitions in Technical Specification 1.3. Therefore, applicability of the proposed Technical Specifications 3.1.1.2 and 3.1.1.3 in only Mode 1 is appropriate. Since Mode 2 will still permit storage of filled fuel canisters in the Spent Fuel Storage Pool 'A', it is appropriate to provide boration and temperature control in the pool. However, upon transition to Mode 3, all fuel containing canisters will have been removed from the site and Specification 3.1.1.4 may appropriately be relaxed. The staff finds the proposed changes acceptable.

Section 3.3.1.1

Technical Specification 3.3.1.1 requires operability of the specified nuclear instrumentation during Recovery Mode. It prescribes the remedial action to be taken in the event of inoperability of the nuclear instrumentation and requires special reports on these events to the Commission.

The licensee proposes to modify the specification by changing the applicability to Mode 1 and by eliminating the requirement in subsequent modes.

In conjunction with the issuance of this amendment, the Commission has granted an exemption to 10 CFR 70.24, Criticality accident requirements to the licensee.

The staff has concluded that since the purpose of the nuclear instrumentation is to provide information on the shutdown status of the core, it is not needed for this purpose when defueling has progressed to the point of having no possibility of a criticality in the reactor building. Therefore, changing the applicability of this specification to Mode 1 as defined in Specification 1.3 is acceptable.

Sections 3.3.2 and 3.3.2.1

Technical Specifications 3.3.2 and 3.3.2.1 prescribe operability requirements for the Engineered Safety Actuation System instrumentation.

The licensee proposed to delete these specifications.

The instrumentation affected by this specification is that instrumentation that causes an automatic start of the emergency diesel generators in the event of a loss of off-site electrical power. The staff has concluded, based on our previous approval of deletion of operability requirements of the emergency diesel generators, that all loads served by the diesel generators may be safely interrupted until off-site power is restored or can be supplied with back-up power from the existing station batteries. Therefore, since the diesel generator operability requirements have been previously deleted, the requirement for automatic start instrumentation may also be safely deleted. The staff finds the proposed changes acceptable.

Section 3.3.3.4

Technical Specification 3.3.3.4 requires operability of the meteorological instrumentation when in the Recovery Mode and requires restoring inoperable instrumentation to operable status within 8 hours.

The licensee proposes to modify this specification by changing the applicability to Modes 1 and 2 and by requiring restoration of inoperable instrumentation within 7 days.

The basis for requiring monitoring meteorological data is to evaluate the need to initiate measures to protect the health and safety of the public during accidents involving off site radiolcgical releases. The worst case accident during Mode 3 is a fire in the reactor building. Analysis of this accident shows that the offsite dose from the facility would be less than the 10 CFR 50 Appendix I numerical design objectives for offsite doses from releases due to normal operations. Thus no protective measures are required. Therefore, there would be no requirement to maintain operable meteorological monitoring instrumentation for TMI-2 in Mode 3. The proposed action statement time period of 7 days is consistent with B&W Standard Technical Specifications. The more conservative 8 hour period was imposed during the phases of cleanup when high levels of Krypton-85 and other radionuclides existed in the containment atmosphere. The high levels of radioactivity presented a higher level of potential threat to the public health and safety if released, thus a more significant need for the ability to assess the threat. As cleanup has progressed the magnitude of potentially airborne radioactivity that could be releasec has diminished significantly. Therefore, the original need to rapidly restore meteorological instrumentation no longer exists and the staff finds the proposed changes acceptable.

Section 3.3.3.5

Technical Specification 3.3.3.5 requires operability during Recovery Mode of Essential Parameters Monitoring Instrumentation used to monitor Reactor Building Pressure, Reactor Vessel Water Level, Incore Thermocouples, Reactor Building Water Level, Borated Water Storage Tank Level, Steam Generator Level, Spent Fuel Storage Pool 'A' Water Level, and Fuel Transfer Canal Water Level.

The licensee proposes to modify this specification by requiring operability of the Incore Thermocouples, Reactor Building Water Level, Borated Water Storage Tank Level, and Steam Generator Water Level during Mode 1 only.

These parameters are monitored to verify the core condition and assure availability of core cooling and criticality control. Following core removal and the subsequent transition to Modes 2 and 3 the need to maintain these parameters will no longer exist. The requirements for

Reactor Building Pressure has been retained but transferred to the Recovery Operations Plan requirement associated with Technical Specification 3.6.1.4. The requirement for Reactor Vessel Water Level has been retained but transferred to Technical Specification 3.4.2 and the associated Recovery Operations Plan requirement. The requirement for Spent Fuel Storage Pool 'A' and the Fuel Transfer Canal water levels have been retained but transferred to Technical Specifications 3.9.1 and 3.9.3 respectively. The staff finds the proposed changes acceptable.

Section 3.3.3.7

Technical Specification 3.3.3.7 requires operability of the Chlorine Detection System during the Recovery Mode.

The licensee proposes to require operability only in Mode 1. The basis for operability of the system is to provide automatic isolation of the control room ventilation system to ensure habitability of the control room in the event of a chlorine leak. Since the manning of the control room will not be required after core removal and the subsequent transition to Modes 2 and 3, ensuring a habitable control room will no longer be necessary to protect the public health and safety. The staff finds the proposed change acceptable.

Section 3.3.3.8

Technical Specification 3.3.3.8 requires operability of the fire detection instrumentation during the Recovery Mode.

The licensee has proposed an administrative change to require operability in Modes 1, 2 and 3.

This is consistent with the proposed definitions of Facility Modes and encompasses all plant conditions previously included in the Recovery Mode. The staff finds the proposed change acceptable.

Section 3.4.2

Technical Specification 3.4.2 requires operability of reactor vessel water level instrumentation during Recovery Mode when the reactor vessel head is removed.

The licensee proposes a revision to this specification that would require operability only during Mode 1. This instrumentation provides for indication of loss of reactor coolant system inventory or water inleakage to the reactor coolant system that could result in a boron dilution. Once defueling of the reactor vessel is complete, it will no longer be necessary to maintain water in the reactor coolant system. Therefore, this instrumentation will not be required in Modes 2 and 3. The staff finds the proposed change acceptable.

Section 3.4.9

Technical Specification 3.4.9 prescribes the reactor coolant system temperature and pressure limits that must be maintained during the Recovery Mode.

The licensee proposes to modify the applicability of this specification to Mode 1.

Temperature and pressure limits are prescribed to prevent non-ductile failure of the reactor vessel, to prevent precipitation of the boron used to control criticality, and to prevent boiling in the reactor coolant system. Following defueling of the reactor coolant system, the reactor vessel will no longer be required to contain the core, the potential source of heat that could cause boiling will be removed, and there will be no further need to maintain soluble boron to control criticality. Therefore, there is no need to monitor these parameters following transition to Modes 2 and 3. The staff finds the proposed change acceptable.

Section 3.5.1

Technical Specification 3.5.1 requires direct communication between the Control Room or Command Center and personnel in the Reactor Building. It is applicable during Core Alternations.

The licensee proposes to modify this specification by making it applicable in Mode 1 during Core Alternations.

This change is administrative in nature since after transition to Modes 2 and 3, core alternations will no longer be possible. The change is consistent with the licensee's definition of Facility Modes. The staff finds the proposed change acceptable.

Section 3.6.1.1

Technical Specification 3.6.1.1 requires that Containment Integrity be maintained unless it is not required by procedures approved by the NRC staff. The specification is applicable during the Recovery Mode.

The licensee proposes to modify this specification by making it applicable to Mode 1 and by removing the statement that permits relaxing integrity pursuant to NRC approved procedures.

Provisions for procedural controls over open containment penetrations has been incorporated into the definition of Containment Integrity in Specification 1.7 and is therefore no longer required in this section.

Following Mode 1 defueling, the maximum radionuclide release postulated for the worst case accident, a fire inside containment, is less than the 10 CFR 50 Appendix I numerical guidelines for normal releases. Thus, the requirement for double barrier isolation of containment penetrations no longer exists.

The licensee has proposed an additional specification, Section 3.6.1.1.b, applicable during Mode 1 only, that would allow modifications to containment penetrations provided that a closed single barrier is maintained. If no barrier exists, the Action Statement would require termination of certain activities inside containment that could pose a threat of radiological releases.

Allowing penetration modification activities provided a single isolation barrier is maintained is consistent with the existing Action Statement which requires closing a penetration with one manual valve, blind flange, or one deactivated automatic valve when one of the double barriers is incapable of closing. The staff finds the proposed changes acceptable.

Section 3.6.1.2

The licensee has proposed a new specification requiring maintaining Containment Isolation during Modes 2 and 3.

This proposed specification provides for an acceptable level of containment performance to assure a single barrier against the spread of the remaining radioactive contamination after Mode 1 defueling is complete.

Section 3.6.1.3

Technical Specification 3.6.1.3 requires that the containment airlocks be operable with both doors closed except during transit through the airlock when only one door must be closed. Both doors may be open if allowed per procedures approved by the NRC staff. This specification is applicable during the Recovery Mode.

The licensee has proposed changes to this specification that would make it applicable during Mode 1 only and would refer to Recovery Operations Plan section 4.6.1.3.1 for the criteria for allowing both doors to be open.

The change of applicability to Mode 1 is consistent with the change of applicability of Containment Integrity specifications to Mode 1 as previously discussed. The procedural requirements previously imposed by the NRC staff prior to allowing opening of both airlock doors are fully incorporated into the Recovery Operations Plan and are thus still applicable. NRC approval of the procedure would no longer be required. The staff finds the proposed changes acceptable.

Section 3.6.1.6

The licensee has proposed a new technical specification 3.6.1.6 that would require the containment airlocks to be operable with at least one door closed unless the criteria of the Recovery Operations Plan Section 4.6.1.6.1 are met. This specification would be applicable in Modes 2 and 3.

The staff has concluded that this proposed specification is acceptable and consistent with previously discussed changes to the Containment Integrity and Containment Isolation technical specifications.

Section 3.6.1.4

Technical Specification 3.6.1.4 prescribes limits for primary Containment pressure. It is applicable during the Recovery Mode.

The licensee proposes to change the applicability to Modes 1, 2 and 3.

This change is acceptable and consistent with the licensee's definition of Facility Modes as previously discussed. It is administrative in nature since Modes 1, 2 and 3 encompass all facility conditions previously included in the Recovery Mode.

Section 3.6.1.5

Technical Specification 3.6.1.5 prescribes the limits within which primary containment air temperatures must be maintained. It is applicable during the Recovery Mode.

The licensee proposes to change the applicability of this specification to Mode 1.

The basis for the primary containment air temperature limits is to assure reactor coolant system temperature remains high enough to prevent precipitation of the soluble boron used for criticality control. In addition, the temperature band was established to assure maximum service life of equipment and instrumentation installed in the reactor building. Following completion of defueling and transition to Modes 2 and 3, the need for criticality control and for operability of most of the equipment in the reactor building will no longer exist. Thus, there will be no need to impose these temperature restrictions after Mode 1. The staff finds the proposed change acceptable.

Section 3.6.3.1

Technical Specification 3.6.3.1 requires that one train of the Containment Purge Exhaust system be operable during purge operations. When no trains are operable, purge operations are prohibited, and one train must be restored to operable status within 7 days.

The licensee proposes to change the applicability of the specification to Modes 1, 2, 3; and to change the 7 day time clock of the action statement to require restoration of an operable train prior to resuming purge operations.

The revision of the applicability is consistent with the licensee's proposed definitions of Facility Modes. It will still require operability during all purge operations during all three modes. Revising the 7 day time clock to "prior to resuming purge operations" allows the licensee to appropriately prioritize facility activities and to best utilize available manpower while still assuring that no purge operations will be performed without an available purge exhaust train. The staff finds the proposed changes acceptable.

Section 3.7.6.1

Technical Specification 3.7.6.1 specifies the required Flood Protection measures and is applicable "at all times".

The licensee proposes to change the applicability to Modes 1, 2 and 3.

This change is administrative in nature since the defined Modes 1, 2 and 3, are the same as "at all times" for the duration of the existing cleanup program.

Section 3.7.7.1

Technical Specification 3.7.7.1 requires operability of the Control Room Ventilation and Emergency Air Cleanup System during the Recovery Mode.

The licensee proposes to change the applicability of the specification to Mode 1. The Control Room Emergency Air Cleanup System is required to be maintained operable to protect control room operators in the event of an accident and to maintain control room habitability in the event of chemical releases. Once Mode 1 defueling is completed there will be no requirement to man the control room (see proposed changes to Section 6.2.2), consequently maintenance of control room habitability is not required. The staff finds the proposed changes acceptable.

Section 3.7.9.1

Technical Specification 3.7.9.1 prescribes limits for removable contamination on sealed sources. It is applicable during the Recovery Mode.

The licensee proposes to change the applicability to Modes 1, 2 and 3.

This is an administrative change since Modes 1, 2 and 3 collectively include all facility conditions previously included in the Recovery Mode. The staff finds the proposed change acceptable.

Sections 3.7.10.1, 3.7.10.2, 3.7.10.4, and 3.7.11

Technical Specifications 3.7.10.1, 3.7.10.2, 3.7.10.4 and 3.7.11 prescribe the operability requirements and action statements for the plant fire protection systems. These specifications are applicable during the Recovery Mode.

The licensee proposes to change the applicability of these specifications to Modes 1, 2 and 3. In addition, 3.7.10.2 will be modified to require operability of certain portions of the system only if the associated charcoal filters are installed.

The change of applicability is administrative in nature as previously discussed. In addition, the designated portions of the system provide protection due to the potential for a fire in the charcoal filters. When the filters have been removed, no fire hazard exists and the operable sprinkler/deluge systems are no longer needed. The staff finds the proposed changes acceptable.

Section 3.7.10.3

Technical Specification 3.7.10.3 prescribes operability requirements for Halon fire suppression systems in the Cable and Transformer Room and the Air Intake Tunnel. The specification is applicable during the Recovery Mode.

The licensee proposes to change the applicability to Mode 1. The Halon System protects circuits and equipment required for safe shutdown and core protection in specific areas of the plant from the propagation of a fire. Once Mode 1 defueling is completed there will be no circuits or equipment necessary for the protection of the core. The staff finds the proposed change acceptable.

Sections 3.8.1 and 3.8.2

Technical Specifications 3.8.1 and 3.8.2 prescribe the operability requirement for A.C. and D.C. electrical distribution systems. The specifications are applicable in the Recovery Mode.

The licensee has proposed changing the applicability of these operability requirements to Mode 1 and has proposed administrative changes involving renumbering of paragraphs.

The purpose of these specifications is to assure that the power sources and associated distribution systems are available to supply the safety related equipment required to maintain the unit in a stable condition. Once Mode 1 defueling is completed no safety related equipment will be required to maintain the unit in a safe and stable condition, consequently, the power sources and distribution systems would not be required. The staff finds the proposed changes acceptable.

Sections 3.9.1, 3.9.2, 3.9.3, and 3.9.4

Technical Specifications 3.9.1, 3.9.2, 3.9.3, and 3.9.4 prescribe requirements for maintaining and monitoring water levels in the Spent Fuel Storage Pool 'A' and the Fuel Transfer Canal. The requirements of 3.9.1 and 3.9.2 are applicable whenever there are canisters containing core material in the Spent Fuel Storage Pool 'A'. The requirement of 3.9.3 and 3.9.4 are applicable whenever canisters containing core material and/or the plenum are stored in the Fuel Transfer Canal.

The licensee proposes to establish the applicability of specifications 3.9.1 and 3.9.2 during Modes 1 and 2 when canisters containing core material are stored in the pool.

This change is administrative since it is only during Modes 1 and 2 that canisters containing core material may be stored in the Spent Fuel Storage Pool 'A'.

The licensee proposes to change the applicability of specifications 3.9.3 and 3.9.4 to Mode 1.

The basis for water level control in the Fuel Transfer Canal is radiation protection for personnel when there are fuel canisters and/or the plenum assembly stored in the Fuel Transfer Canal. During Modes 2 and 3 there will be no canisters stored in the Fuel Transfer Canal. Therefore, water level control will not be required for purposes of personnel radiation protection from canisters. During Modes 2 and 3 the plenum may be stored in the Fuel Transfer Canal; however, there will be a significant reduction of cleanup activity in the building. Therefore, with fewer people in the RB requiring radiation protection and no canisters stored in the Fuel Transfer Canal, maintenance of water in the canal will no longer be required. The staff finds the proposed changes acceptable.

Sections 3.9.12.1, and 3.9.12.2

Technical Specifications 3.9.12.1 and 3.9.12.2 require operability of the Fuel Handling Building Air Cleanup Exhaust System and the Auxiliary Building Air Cleanup Exhaust Systems respectively. The specifications are applicable in the Recovery Mode and require suspension of operations involving movement of liquid and gaseous radioactive wastes as the respective areas of the system become inoperable.

The licensee proposes to modify [illegible] specifications by changing the applicability to Modes 1, 2 and [illegible] addition, the action statement will be changed to require suspension of operations involving movement of liquid and solid radioactive wastes the release of which could exceed 50 percent of the Appendix B Technical Specification instantaneous release rate for gaseous effluents. The change of applicability is administrative in nature and consistent with the previous discussions of the newly defined facility modes. Deletion of gaseous radioactive wastes from the Action Statement is appropriate since the gaseous radioactive source term has been essentially eliminated as a result of the progress of the defueling and decontamination program. The remaining potential source terms are the result of still existing liquid and solid radioactive waste material and contamination within the facility. The quantitative restrictions in the proposed Action Statement will ensure that suspension of radioactive material from any operation will not result in releases in excess of those allowed by 10 CFR 50 Appendix I. The staff finds the proposed changes acceptable.

Section 3.9.13

Technical Specification 3.9.13 prohibits discharge of Accident Generated Water until approved by the NRC and requires that its discharge, once approved, be in accordance with procedures approved pursuant to specification 6.8.7. It is applicable during the Recovery Mode.

The licensee proposes administrative changes to this specification that would make it applicable in Modes 1, 2 and 3. The proposed change is administrative only and does not change any presently imposed license restrictions. The staff finds the proposed change acceptable.

Section 3.10.1

Technical Specification 3.10.1 imposes restrictions on the handling of heavy loads in the containment building during the Recovery Mode.

The licensee proposes to modify the applicability of this specification to Mode 1.

The purpose of this specification was to limit load handling to preclude heavy load drops that might result in a core reconfiguration or a loss of reactor vessel integrity. Once Mode 1 defueling is completed, the basis for controlling heavy loads inside the containment will be eliminated and the specification will not be required. The staff finds the proposed change acceptable.

Section 3.10.2

Technical Specification 3.10.2 imposes restrictions on the handling of heavy loads in the Fuel Handling Building during the Recovery Mode.

The licensee proposes to modify the applicability of this specification to Modes 1 and 2. The basis for this specification is to prevent a load drop in the Fuel Handling Building causing damage to canisters containing core material. Subsequent to Modes 1 and 2 all core material will have been shipped off-site. Thus, the basis for controlling heavy loads inside the Fuel Handling Building is eliminated and the specification is not required. The staff finds the proposed change acceptable.

Section 6.2.2

Technical Specification 6.2.2 in part, lists the required minimum shift crew composition, requires a licensed operator in the control room when fuel is in the reactor, and requires an individual qualified in radiation protection procedures on site when fuel is in the reactor.

The licensee proposes to modify these requirements by specifying that they apply only during Mode 1.

The requirements for licensed operators are specified in 10 CFR 50.54 for fueled nuclear power plants. Upon transition from Mode 1, the facility will no longer be considered fueled and these requirements will no longer apply. The staff finds the proposed change acceptable.

Section 6.8.2.2

Technical Specification 6.8.2.2 specifies the scope of licensee procedures that require NRC approval prior to implementation.

The licensee proposes to delete the requirement for NRC approval of procedures except for those involving disposal of Accident Generated Water.

Since the accident, the licensee has implemented a number of major management and organization changes designed to more effectively manage the unique challenge of the post-accident cleanup. These changes have resulted in an organization structure which places an acceptable emphasis on the safe conduct of cleanup activities with adequate provisions for management review and oversite of facility activities. The staff's ongoing assessment of the licensee's procedure development and review program indicates that it is working effectively to assure implementation of the Technical Specifications and compliance with regulatory requirements. Therefore, the staff concludes that the cleanup activities at the Three Mile Island Unit 2 facility no longer requires the unique administrative controls that have been required in the past. The staff will, however, require the licensee to submit for approval procedures involving the disposal of the accident generated water.

Section 6.11

Technical Specification 6.11 requires that personnel radiation protection be consistent with the requirements of 10 CFR 20 and the NRC approved Radiation Protection Plan.

The licensee proposes to modify this specification by deleting the requirement for NRC approval of the Radiation Protection Plan. Removal of this requirement is based on the acceptable past performance of the licensee in the area of radiation protection. It is also consistent with the Standard Technical Specifications for Babcock and Wilcox plants. Auditing by the NRC of the Radiation Protection Plan and licensee compliance with the plan would continue consistent with practice at the TMI site for the operating unit, TMI-1.

3.0 CONTACT WITH STATE OFFICIALS

On March 31, 1988, the NRC staff contacted by telephone, the Bureau of Radiation Protection, Department of Environmental Resources, Commonwealth of Pennsylvania, on the proposed determination of no significant hazards consideration. No objection to the proposed action was voiced.

4.0 ENVIRONMENTAL CONSIDERATION

This amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. We have determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that this amendment involves no significant hazards consideration and there has been no public comment on such finding. Accordingly, this amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of this amendment.

5.0 CONCLUSION

We have concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (2) such activities will be conducted in compliance with the Commission's regulations and the issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

Dated: May 25, 1988

Principal Contributor: John A. Thomas