Distribution:
Docket No. 50-320
NRC PDR
Local PDR
DCS
TMI HQ R/F
TMI Site R/F
BJSnyder
WDTravers
TCPoindexter
MTMasnik
RAWeller
PGrant
RCook
AFasano
LChandler, ELD
IE (5)
TBarnhart (4)
LSchneider
JSaltzman, OSP
ACRS (16)
OPA
RDiggs
HRDenton
ARosenthal, ASLAB
RLazo, ASLAP
M-town Office

December 19, 1984

Docket No. 50-320

Mr. F. R. Standerfer, Director
Three Mile Island Unit 2
GPU Nuclear Corporation
P.O. Box 480
Middletown, PA 17057

Dear Mr. Standerfer:

Subject: Three Mile Island Nuclear Station, Unit 2
Operating License No. DPR-73
Docket No. 50-320
Technical Specification Change Requests 39, 41, 43
Recovery Operations Plan Change Requests 19, 20, 22
Exemption Request from 10 CFR 50.55a (Code Safety Valves)
Exemption Request from 10 CFR 100, Appendix A and
10 CFR 50.36(3) (Seismic Instrumentation)

The Nuclear Regulatory Commission has issued the enclosed Amendment of Order; Recovery Operations Plan Change Approval; Approval of Exemption from the requirements of 10 CFR 50.55a for Code Safety Valves; and Approval of Exemption from the seismic instrumentation requirements of 10 CFR 100, Appendix A and 10 CFR 50.36(3).

The Amendment of Order which modifies many sections of the Proposed Technical Specifications (PTS) was requested by General Public Utilities Nuclear Corporation (GPUNC) in letters dated January 12, 1983, September 12, 1983 and September 30, 1983. Other documents related to this request include: Recovery Operations Plan (ROP) Changes which were requested in separate letters also dated January 12, September 12, and September 30, 1983; a request for exemption from the requirements of 10 CFR 50.55a with respect to Code Safety Relief Valves in a letter dated April 18, 1984; and a request for an exemption from the seismic monitoring requirements of 10 CFR 50.36(3) and 10 CFR 100, Appendix A, Paragraph VI(a)(3) in a letter dated April 18, 1984.

As previously explained in a letter issued by the staff on July 17, 1984, your PTS and ROP change requests were divided into two separate issuances. The first issuance was made on July 17, 1984 and was immediately effective. The staff has reviewed your safety evaluations for the above documents and concludes that your requests addressed by this issuance are acceptable with changes as discussed with your staff. PTS changes that are the subject of this letter will become effective on January 7, 1985. The Exemptions to 10 CFR 50.55a, 10 CFR 50.36(3) and 10 CFR 100, Appendix A, Paragraph VI(a)(3) are effective upon issuance.

OFFICE	TMIPO:NRR	SL:TMIPO:NRR	PD:TMIPO:NRR	ELD	DD:NRR	D:NRR	*SEE PREVIOUS CONCURRENCE
SURNAME	TCPoindexter;bg*	RAWeller*	BJSnyder*	LChandler*	ECCase	HRDenton	
DATE	11/ /84	11/ /84	11/ /84	12/ 3/84	12/7/84	12/ /84	

Since the February 11, 1980 Order imposing the Proposed Technical Specifications is currently pending before the Atomic Safety and Licensing Board, the staff will be advising the Licensing Board of this Amendment of Order through a Notice of Issuance of Amendment of Order and a Motion to Conform Proposed Technical Specifications in Accordance Therewith.

Federal Register Notices for the discussed issuances are enclosed. Copies of the related Safety Evaluation and revised pages for the Proposed Technical Specifications and the Recovery Operations Plan are also enclosed.

Sincerely,

Original signed by
B. J. Snyder

Bernard J. Snyder, Program Director
Three Mile Island Program Office
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment of Order
2. Safety Evaluation
3. Proposed Technical Specification Page Changes
4. Recovery Operations Plan Change Pages
5. Exemption from 10 CFR 50.55a
6. Exemption from 10 CFR 100, Appendix A, Paragraph VI(a)(3) and 10 CFR 50.36(a)
7. Notice of Environmental Assessment and Finding of No Significant Impact
8. Federal Register Notices

cc: J. Barton
R. Rogan
S. Levin
R. Freemerman
J. Byrne
Service Distribution List
(see attached)

OFFICE							
SURNAME							
DATE							

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

Enclosure 1

In the Matter of)
)
GENERAL PUBLIC UTILITIES NUCLEAR) Docket No. 50-320
CORPORATION)
)
)
(Three Mile Island Nuclear Station,)
Unit 2))

AMENDMENT OF ORDER

I.

GPU Nuclear Corporation, Metropolitan Edison Company, Jersey Central Power and Light Company and Pennsylvania Electric Company (collectively, the licensee) are the holders of Facility Operating License No. DPR-73, which had authorized operation of the Three Mile Island Nuclear Station, Unit 2 (TMI-2) at power levels up to 2772 megawatts thermal. The facility, which is located in Londonderry Township, Dauphin County, Pennsylvania, is a pressurized water reactor previously used for the commercial generation of electricity.

II.

By Order for Modification of License, dated July 20, 1979, the licensee's authority to operate the facility was suspended and the licensee's authority was limited to maintenance of the facility in the present shutdown cooling mode (44 Fed. Reg. 45271). By further Order of the Director, Office of Nuclear Reactor Regulation, dated February 11, 1980, a new set of formal license requirements was imposed to reflect the post-accident condition of the facility and to assure the continued maintenance of the current safe, stable, long-term cooling condition of the facility (45 Fed. Reg. 11292).

Although these requirements were imposed on the licensee by an Order of the Director of Nuclear Reactor Regulation, dated February 11, 1980, the TMI-2 license has not been formally amended. The requirements are reflected in the proposed Recovery Mode Technical Specifications (PTS) presently pending before the Atomic Safety and Licensing Board. The revisions that are the subject of this order do not give the licensee authorizations that may be needed to undertake specific cleanup activities. These activities will require separate consideration by the staff per Section 6.8.2 of the PTS, individual staff safety evaluations and/or licensing actions as appropriate. Hereafter in this Amendment of Order, the requirements in question are identified by the applicable Proposed Technical Specification.

III.

By letters dated January 12, 1983, September 12, 1983, and September 30, 1983, GPU Nuclear Corporation (GPUNC) proposed changes to the Proposed Technical Specifications (PTS) for Three Mile Island Unit 2 (TMI-2) to reflect current plant conditions.

The TMI-2 Technical Specifications were revised in their entirety with the issuance of the February 11, 1980 Order. The general purpose of that order was to properly reflect technical specification requirements for the facility's post accident mode of operation. From February 1980 to present, many modifications of that order have been issued in addition to several License Amendments and changes to the Recovery Operations Plan. Even though each of these changes was justified and concurred with by the staff, there has not been a complete review and revision of the Proposed Technical Specifications since the

February 11, 1980 Order. This review is necessary because of the degree of technical knowledge obtained since the original Order and the resultant altered approaches to technical problems. The necessary course of action for the NRC at TMI-2 to maximize our ability to protect the health and safety of the public is now better defined and therefore Proposed Technical Specification requirements can now be deleted, added or modified for consistency and to more correctly reflect what should be required by the NRC at TMI-2.

The staff has reviewed the licensee's proposed changes which can be grouped into the following categories: (1) Editorial Changes that correct spelling, grammar, page numbers, and the associated indexes. These changes do not affect the technical content nor the intent of the section; (2) Modifications to the existing Limiting Conditions for Operation and Safety Limits that were made to more correctly state what systems or equipment are necessary based on the present status of TMI-2; (3) New Limiting Conditions for Operation that were added to also more correctly reflect what systems or equipment are necessary based on the present status of TMI-2; (4) Design Features of Section 5.0 that were modified or deleted to more accurately state design limits that can be verified; (5) Bases that were revised because of changes in technical approaches resulting from data that has been obtained and analyzed; and (6) Tables listing specific pieces of equipment that were moved from the Limiting Conditions for Operation sections to the applicable section of the Recovery Operations Plan. This type of change does not lessen the regulatory requirement but more appropriately places the description of the piece of equipment along with its corresponding testing requirement.

An Exemption from 10 CFR 50.55a with respect to Code Safety Relief Valves, and an Exemption from the seismic monitoring requirements of 10 CFR 50.36 and 10 CFR 100, Appendix A, Paragraph VI(a)(3) are required to accompany this Amendment of Order because of some of the deletions and alterations to the PTS that were granted herein.

Changes to associated surveillance requirements of the Recovery Operations Plan that correspond to modifications to the Limiting Conditions of Operation have also been made accordingly.

The staff's safety assessment of this matter as discussed above is set forth in the concurrently issued Safety Evaluation. Since the February 11, 1980 Order imposing the Proposed Technical Specifications is currently pending before the Atomic Safety and Licensing Board, the staff will be advising the Licensing Board of this Amendment of Order through a Notice of Issuance of Amendment of Order and a Motion to Conform Proposed Technical Specifications in Accordance herewith.

It is further determined that the modification does not authorize a change in effluent types or total amounts nor an increase in power level and will not result in any significant environmental impact. In light of this determination and as reflected in the Environmental Assessment and Notice of Finding of No Significant Environmental Impact prepared pursuant to 10 CFR 51.2 and 51.30 through 51.32 issued concurrently herewith, it was concluded that the action is insignificant from the standpoint of environmental impact and that an environmental impact statement need not be prepared.

IV.

Accordingly, pursuant to the Atomic Energy Act of 1954, as amended, the Director's Order of February 11, 1980, is hereby revised to incorporate the deletions, additions, and modifications set forth in Enclosure 3 hereto. This Amendment of Order shall be effective on January 7, 1985.

For further details with respect to this action, see (1) Letter to B. J. Snyder, USNRC, from R. C. Arnold, GPUNC, Technical Specification Change Request No. 39, dated January 12, 1983, (2) Letter to B. J. Snyder, USNRC, from R. C. Arnold, GPUNC, Technical Specification Change Request No. 41, dated September 12, 1983, (3) Letter to B. J. Snyder, USNRC, from R. C. Arnold, GPUNC, Technical Specification Change Request No. 43, dated September 30, 1983, (4) Letter to L. H. Barrett, USNRC, from B. K. Kanga, GPUNC, Recovery Operations Plan Change Request No. 19, dated January 12, 1983, (5) Letter to L. H. Barrett, USNRC, from B. K. Kanga, GPUNC, Recovery Operations Plan Change Request No. 20, dated September 12, 1983, (6) Letter to L. H. Barrett, USNRC, from B. K. Kanga, GPUNC, Recovery Operations Plan Change Request No. 22, dated September 30, 1983, (7) Letter to B. J. Snyder, USNRC, from E. E. Kintner, GPUNC, Seismic Monitoring Exemption Request, dated April 18, 1984, (8) Letter to B. J. Snyder, USNRC, from E. E. Kintner, GPUNC, Exemption Request from 10 CFR 50.55a with respect to Code Safety Valves, dated April 18, 1984, and (9) the Director's Order of February 11, 1980.

All of the above documents are available for inspection at the Commission's Public Document Room, 1717 H Street, N.W., Washington, D.C. and at the Commission's Local Public Document Room at the State Library of Pennsylvania, Government Publications Section, Education Building, Commonwealth and Walnut Streets, Harrisburg, Pennsylvania 17126.

FOR THE NUCLEAR REGULATORY COMMISSION

HR Denton

Harold R. Denton, Director
Office of Nuclear Reactor Regulation

Effective Date: January 7, 1985
Dated at Bethesda, Maryland
Issuance Date: December 19, 1984

Enclosure 2

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

GPU NUCLEAR CORPORATION

METROPOLITAN EDISON COMPANY

PENNSYLVANIA ELECTRIC COMPANY

JERSEY CENTRAL POWER & LIGHT COMPANY

DOCKET NO. 50-320

THREE MILE ISLAND NUCLEAR STATION UNIT NO. 2

INTRODUCTION

By letters dated January 12, 1983, September 12, 1983, and September 30, 1983, GPU Nuclear Corporation (GPUNC) requested the approval of changes and provided supporting information to modify the Proposed Technical Specifications (PTS) of Operating License No. DPR-73. By separate letters dated January 12, 1983, September 12, 1983, and September 30, 1983, GPUNC also requested the approval of changes to the Recovery Operations Plan (ROP) for Three Mile Island, Unit 2 (TMI-2). In a letter dated April 18, 1984, the licensee requested an exemption from the seismic monitoring requirements of 10 CFR 50.36(a) and 10 CFR 100, Appendix A, Paragraph VI(a)(3). In another letter dated April 18, 1984, the GPUNC also requested an exemption from the Code Safety Relief Valve requirements of 10 CFR 50.55a. These exemptions were required to support the requested changes to the PTS and ROP. The primary purpose of these changes and exemptions is to update the Proposed Technical Specifications and the applicability of the Code of Federal Regulations respectively to more accurately reflect the present status of TMI-2. Although ROP changes normally issued are not associated with any change in the PTS, the staff has elected in this case to issue the subject Recovery Operations Plan Changes as a part of the Amendment of Order due to the fact that the majority of the ROP changes were promulgated by changes to the PTS and therefore must be discussed in parallel.

DISCUSSION

(A) Editorial Changes

Throughout the PTS and the Recovery Operations Plan (ROP), the licensee has requested changes in grammar, punctuation and page numbers to make the text consistent, to clarify ambiguous text and to improve format. The staff finds that these changes do not affect the technical content nor the technical interpretation of the sections and therefore concurs with these modifications. Editorial changes were made to the following sections:

All indexes

Section	1.2	Section	3.7.10.2
Bases	2.1.3	Section	3.7.10.3
Section	3.1.1.2	Section	3.7.10.4
Section	3.3.1.1	Section	3.8.2.1
Section	3.3.2.1	Section	3.9
Section	3.3.3.1	Bases	3/4.7.1
Section	3.3.3.4	Bases	3/4.7.2
Section	3.3.3.8	Bases	3/4.7.3.2
Section	3.6.1.1	Bases	3/4.7.3.3
Section	3.6.1.5	Bases	3/4.9.12
Section	3.6.3.1	Section	6.5.3.2
Section	3.7.1	Section	4.3.1.1
Section	3.7.2	Section	4.3.2.1

Section	3.7.3.1	Section	4.4.9.1.4
Section	3.7.3.2	Section	4.5.1
Section	3.7.4.1	Section	4.7.3.1
Section	3.7.6.1	Section	4.7.3.2
Section	3.7.7.1	Section	4.7.4.1
Section	3.7.10.1		

(B) Definitions

The following is a discussion of changes to Section 1 requested by the licensee. Some of the licensee's change requests have been modified by the staff and agreed upon by the licensee. The staff therefore concurs with the changes to the "Definitions" section as stated below.

1.3 Recovery Mode - At present, the definition of Recovery Mode provides a maximum RCS temperature of 280°F. The amount of decay heat produced by the TMI-2 reactor has been constantly declining since the March 28, 1979 accident. The staff concurs with the licensee in lowering the average reactor coolant temperature as the level of decay heat also decreases. The proposed value (200°F) is lower than the saturation value (boiling point) for water at atmospheric pressure thereby providing adequate conservatism for RCS parameters.

1.7 Containment Integrity - The staff has modified part a of this section to reflect the regulatory relief that was granted GPU by the NRC in an Exemption and Approval of Alternate Design dated July 17, 1984. The discussion in the staff's letter states that two manual isolation valves outside of containment are acceptable in lieu of automatic isolation valves. It should be noted that at TMI-2, there are no pressure signals that could automatically initiate valve closure. Instead TMI-2 uses remotely operated valves requiring operator action. If a manual valve (instead of a remote/automatic valve) is open, but capable of being closed in all potential accident conditions, then per this definition, containment integrity still exists.

1.17 Accident Generated Water - The licensee has proposed the addition of a definition for Accident Generated Water based on the definition contained in the Lancaster Agreement. The Agreement made between the City of Lancaster Authority, the NRC, and the Metropolitan Edison Company, Jersey Central Power and Light Company and Pennsylvania Electric Company on February 27, 1980, states that accident generated water is water that existed in the TMI-2 Auxiliary, Fuel Handling and Containment Buildings (including the primary system) as of October 16, 1979. An exception is water, which as a result of decontamination operations, becomes comingled with non-accident generated water such that the comingled water has a tritium content of 0.025 uCi/ml or less before processing.

Accident generated water also includes water that has a total activity of greater than one uCi/ml prior to processing except when the water is originally non-accident water and becomes contaminated by use in the cleanup. Water that contains greater than 0.025 uCi/ml of tritium before processing is also included.

It is the staff's opinion that the proposed definition properly reflects the requirements of the Lancaster Agreement and we, therefore, approve the subject change.

(C) Safety Limits - Bases

The following is a discussion of the PTS, Section 2 modifications requested by the licensee. As stated below, the staff concurs with the requested change.

2.1.3 Reactor Coolant System Pressure - This section has been modified to reflect the current condition of the Reactor Coolant System to more correctly state in the bases why a safety limit for Reactor Coolant System Pressure is not applicable.

As a result of the March 28, 1979 accident, the pressure retaining ability of the closed Reactor Coolant System (RCS) was uncertain. Therefore, no meaningful RCS pressure safety limit was stated while the system was in that condition. As modified herein, Section 3.4.9.1 of the PTS now states that the RCS will be kept open to the atmosphere unless approved otherwise by the staff. This also eliminates the need at this time for a pressure safety limit due to the fact that the RCS has no ability to retain and build up pressure. (See also, concurrently issued Exemption for Code Safety Valves, Enclosure 5.) Therefore the basis for Section 2.1.3 has been modified to reflect the "open to atmosphere" condition and the resultant lack of a need for RCS pressure safety limit.

(D) Limiting Conditions for Operation

Section 3.1.1 - Borated Cooling Water Injection - The licensee has proposed to (1) change the title of this section to "Borated Cooling Water Injection," (2) delete the reference to one operable makeup pump (MUP) as being one of the systems capable of injecting borated water into the reactor coolant system, (3) relocate in its entirety the Limiting Conditions for Operation and the Action Statements for the Mini-Decay Heat Removal System (MDHR) which were previously stated in Section 3.7.3.3 of the PTS to Section 3.1.1, and (4) modify the action statements for each system discussed in this section to indicate at what point operations involving core alterations and fuel movement will be suspended.

Since the title change is editorial and does not alter the regulatory requirement, the proposed changes are acceptable to the staff.

Prior to the subject request, Section 3.1.1.1.a.1, which required "One OPERABLE makeup pump," had a footnote that "All makeup pumps shall be made inoperable when valve DH-V1 or DH-V171 is open by racking out their electrical power supply circuit breakers." The basis for this requirement was to prevent RCS overpressurization by the MUP. Following the head lift, overpressurization is no longer possible. However, the makeup pumps were designed to provide makeup to a pressurized system. During depressurized operations, the decay heat removal system was designed to serve as the primary means of makeup to the reactor coolant system. Since there is no intent to repressurize the reactor coolant system during future recovery operations, there is no need to maintain the capability for high pressure injection.

Additionally, following the accident, several systems were installed/modified which provide adequate means of makeup capability for the reactor coolant system. These include the standby pressure control system, the mini-decay heat removal system, and the bleed tank makeup injection mode. Based on the above discussion, the staff concurs with the request to delete the makeup pumps from the Proposed Technical Specifications.

The relocation of the MDHR requirements to Section 3.1.1 does not alter the regulatory requirements for the system and the change is therefore acceptable.

The modifications to the action statement correctly distinguish which systems are absolutely necessary when core alterations are in progress and what combinations of boron injection systems still satisfy the intent of the specification.

Section 3.1.3 - Control Assemblies - Since the reactor head has been removed from the vessel, this section is no longer required. If the head is reinstalled the staff will re-review the need for a LCO requirement.

Section 3.3.3.3 -Seismic Instrumentation - The licensee has proposed the deletion of the requirements for seismic monitors at TMI-2. The licensee supplemented the request for a modification to the PTS with a request for an exemption from the seismic monitoring requirements of 10 CFR 50, dated April 18, 1984.

The requirement to have seismic monitors is partially based on 10 CFR 100, Section VI (a)(3) where it states that, "Suitable instrumentation shall be provided so that the seismic response of nuclear power plant features important to safety can be determined promptly to permit comparison of such response with that used as the design basis. Such a comparison is needed to decide whether the plant can continue to be operated safely and to permit such timely action as may be appropriate."

Title 10 of the Code of Federal Regulations, Part 50, Section 50.36(3) requires surveillance for systems and components used to maintain the plant within the limiting conditions for operation.

In reference to the 10 CFR requirements, it is the staff's opinion that because these sections of the Code are directed toward a determination of whether a plant can continue to be operated safely and since TMI-2 is prohibited from operating except in the shutdown, recovery mode, no seismic response data is required at TMI-2.

Therefore the staff concurs with the deletion of the seismic instrumentation of PTS Section 3.3.3.3, Table 3.3-7 (Seismic Monitoring Instrumentation) and the surveillance requirements of ROP Section 4.3.3.3.1. The staff is also granting the requested exemption to 10 CFR 50.36(3) relative to seismic instrumentation testing and the instrumentation requirements of 10 CFR 100 Section VI (a)(3).

Section 3.3.3.5 - Remote Shutdown Instrumentation, and
Section 3.3.3.6 - Post Accident Instrumentation - The licensee has proposed to combine Sections 3.3.3.5 and 3.3.3.6 of the PTS into a new section

titled, "Essential Parameters Monitoring Instrumentation." Also the remote shutdown monitoring instrumentation requirements of Table 3.3-9 for NI Intermediate Range Level Log N and NI Source Range Level were moved to Table 4.3-/ of the Recovery Operations Plan. Requirements for makeup tank level, reactor coolant inlet temperature, control rod drive trip breakers, and steam generator pressure A/B were deleted from the table. A reactor coolant system temperature instrument was added. The requirements to maintain equipment associated with the deleted instrumentation was also deleted in previous staff actions; therefore, there is no need to retain these instruments in the PTS. Furthermore, since there is no reactor coolant flowing, the need to differentiate between RCS inlet and outlet temperature is not necessary. Instead of requiring a specific RCS inlet temperature instrument, a reactor coolant system temperature instrument is acceptable. The staff has not deleted the requirements for decay heat removal flow instrumentation from Table 4.3-7 as requested by the licensee. This system is relied upon as one of the methods for injecting boron (see Section 3.1.1); therefore the deletion of this flow instrumentation is not acceptable to the staff. This matter has been discussed with and agreed to by GPUNC staff.

Table 3.3-10, Post Accident Monitoring Instrumentation has been deleted. The staff agrees that this instrumentation is not required at TMI-2 considering the benign condition of the plant (see additional discussion on Recovery Operations Plan, Table 4.3-7 below).

Section 3.3.3.8 - Fire Detection - Originally, this section had a provision ("per occupational exposure") that if radiological conditions were severe enough, the stated requirement need not be performed. The non-performance of the PTS requirement did not significantly impact the ability to safely maintain the facility in a safe shutdown condition and was therefore previously acceptable to the staff. However, radiological conditions have changed to such a degree that access to the subject areas is now permitted. Therefore, the radiological exemption is no longer necessary and has therefore been deleted.

Section 3.4.3 - Safety Valves - The licensee has requested the deletion of the pressurizer code safety valves which prior to the accident were set at 2435 PSIG $\pm$ 1%. These valves were required in Section 50.55a of 10 CFR Part 50, which references the overpressure protection requirements for pressure vessels as stated in ASME Section II, Article 9 of the Summer 1967 Addendum. Therefore deletion of this requirement necessitated a code exemption. This safety discussion supplements the staff's exemption to 10 CFR 50.55a for code relief valves at TMI-2 (see Enclosure 5).

Since the reactor vessel head has been removed, the reactor coolant system is open to the reactor building atmosphere and is no longer capable of retaining pressure. During the remainder of the recovery, there are no plans to reinstall the head. In the present configuration, the reactor coolant system is not a pressure retaining component and there is therefore no need for safety relief valves. If in the course of the recovery it becomes necessary to reclose the reactor coolant system, applicable ASME overpressure protection standards will be incorporated into the closure design.

The staff therefore concurs with the licensee's proposed deletion of this section.

<u>Section 3.4.9.1/3.4.9.2 - Reactor Pressure/Temperature Limits</u> - The licensee has proposed to lower the maximum average temperature of the RCS from 280°F to 170°F and the maximum pressure from less than 600 psig to less than 100 psig. Per conversations with GPUNC, the staff has modified the licensee's request to make it consistent with the existing RCS configuration, Definition 1.3 for "Recovery Mode" and the minimum permitted RCS temperature as stated in Section 3.1.1.2. Therefore this section has been revised to state, "The Reactor Coolant System shall be maintained between a T average less than <u>200°F</u> and greater than 50°F. Because the vessel is currently open to the atmosphere, the modified T_{ave} of 200°F adequately insures that the reactor coolant system is maintained at a temperature lower than that which would produce boiling at atmospheric pressure (212°F). The effects of a temperature rise would also be dependent on the corresponding temperature rise in the containment atmosphere. Because of the volume of the containment (2×10^6 cu. ft.), any change in the RCS and the containment atmosphere's parameters would occur slowly enough to allow sufficient time for any corrective actions that would need to be taken. These actions could include putting the head back on the vessel, the starting of one of the mechanical cooling modes or reactor coolant system water volume adjustment. The proposed action statement has also been revised to delete the requirement to reduce pressure to within limits since with an open

vessel, RCS pressure automatically follows atmospheric pressure. The staff has also added a statement that should it become necessary to reclose the RCS during the recovery, a maximum pressure and applicable overpressure protection will be specified in a safety evaluation submitted to the NRC for approval.

Section 3.6.1.3.b Containment Air Locks - The licensee has requested the deletion of the requirement to perform leak rate testing of containment air locks. On September 2, 1981, the staff issued an exemption and provided a safety evaluation for the deletion of certain requirements of 10 CFR Part 50, Appendix J. This exemption included type A, B, and C leakage tests except for the requirements of Subsection III.D.2.b.iii for air lock door seals. The September 2, 1981 safety evaluation also evaluated the deletion of the air lock testing and concluded that because of the minimum containment pressure increase that could occur as a result of an accident or system failure, and the insignificant environmental impact, the deletion of containment air lock tests was acceptable. Therefore, based on the referenced safety evaluation, the NRC staff concurs with the proposal to delete the door seal section from the PTS.

Section 3.6.1.4 - Internal Pressure - The licensee did not request a modification to this section; however, the staff has reviewed the subject requirements and found them to be inapplicable to TMI-2 in its present condition. This requirement presently states that containment pressure shall be maintained between 0 psig and the minimum allowable pressure as determined from Figure 3.6-1 of the PTS. This figure states minimum pressure as a function of Borated Water Storage Tank water temperatures.

The purpose of this figure relates to before the accident when the containment sprays were incorporated as a safety system. When activated, the cooling effect of these sprays would result in a containment building pressure reduction which could exceed the minimum pressure design conditions of the containment building liner. The containment spray requirements were contained in Section 3.6.2 of the Pre-Accident Technical Specifications. This section was deleted with the issuance of the February 11, 1980 Order imposing the Proposed Technical Specifications. Therefore the associated figure should also have been deleted. This has been discussed and concurred with by the licensee. Even though the staff is deleting Figure 3.6-1, a statement is being retained in Section 3.6.1.4 to reflect that the minimum allowable containment pressure of 12.2 psia must not be violated.

Section 3.6.1.5 - Air Temperature - A 50°F minimum average air temperature for the containment has been added to this section to reflect the minimum allowable temperature for the RCS. Since the RCS is cooled by loss to the ambient, the containment air temperature should not be lower than that allowed for the RCS since the the two temperatures will try to equalize. By maintaining both values above 50°F, boron properties will be maintained.

Section 3.7.3.3 - Mini-Decay Heat Removal System - Per the licensee's request, the requirements of this section have been moved in their entirety to Specification 3.1.1.1. Since there is no decrease in requirements, the staff concurs with this change.

Section 3.7.6.1 Flood Protection - The staff has added, as discussed with and agreed to by GPU staff, the Southeast Storage Facility to the list of areas to be inspected when the water level at the Unit 1 intake structure approaches 301 feet mean Sea Level, USGS data. Since low level radioactive materials are stored in this area, it is the staff's opinion that the area should be checked to insure that all containers are sealed and secured whenever potential floods are anticipated.

Section 3.7.9 - Sealed Sources - The licensee has proposed to add the subject section to insure that sealed sources are treated in accordance with NRC regulations. GPUNC's proposal is consistent with NUREG-0102, Standard Technical Specifications for Babcock and Wilcox Pressurized Water Reactors, Rev. 4. The staff therefore concurs with this addition.

Section 3.7.10.2 - Deluge/Sprinkler Systems - The licensee has requested that the "per occupational exposure" clause on item 3.7.10.2 for the Auxiliary Building Backup Exhaust Filters, be deleted. Originally, this section had a provision ("per occupational exposure") that if radiological conditions were severe enough, the stated requirement need not be performed. The non-performance of the PTS requirement did not significantly impact the ability to safely maintain the facility in a shutdown condition and was therefore previously acceptable to the staff. However, radiological conditions have changed to such a degree that access to the subject areas is now permitted. Therefore, the radiological exemption is no longer necessary and has therefore been deleted.

The staff has also made another modification to this section, which was discussed with and agreed to by GPU personnel, by adding the Southeast Storage Facility to the requirements of this section. There is presently a deluge/sprinkler system installed in this structure which is used for storing miscellaneous low level radioactive materials. Even though the likelihood of a fire in this area is small, it is the staff's opinion that the area be protected because of the nature of the materials stored there.

Section 3.7.11 - Penetration Fire Barriers - The licensee has expanded the action statement of this section to require that a Special Report to the Commission, pursuant to Specification 6.9.2, be issued. The staff finds that this change will keep the Commission better informed and therefore approves the change.

Section 3.8.1 A. C. Sources - Per conversations with the licensee, the staff has modified the subject requirement by replacing the requirement to restore all the AC electrical power sources within 72 hours with a 7 day requirement. The effect of having two operable diesel generators is not as important at TMI-2 as at an operating plant and does not affect the ability to keep the core in a safe shutdown condition. Therefore it is the staff's opinion that the extended allowable inoperable time is acceptable.

The licensee has also deleted the requirement for retesting the diesels after the initial operability verification test. Because of the benign condition at TMI-2, the staff also concurs with this modification. Tables 3.8-1 and 3.8-2 have been modified accordingly.

Section 3.9.13 - EPICOR-II Processed Water

Section 3.9.14 - Reactor Building Sump Water - The licensee has combined Sections 3.9.13 and 3.9.14 into one new section. The change was requested because processing of the Reactor Building sump water and Reactor Coolant System water has been approved by the NRC. Therefore the only constraint remaining is NRC approval of the discharge of the water identified by both sections. Additionally, the name for this water has been changed to Accident Generated Water with the appropriate definition being added to Section 1.17 of the PTS. These changes do rot affect any associated safety parameters and are, therefore, acceptable.

(E) Bases - General - The following bases sections were modified based on the above discussions.

3/4.1.1 - Boration Control

3/4.1.3 - Control Assemblies - modified to reflect changes to Section 3.1.3 made in the June 17, 1984 Modification of Order

3/4.3.3.3 - Seismic Instrumentation

3/4.3.3.5 - Remote Shutdown Instrumentation/Essential Parameters Monitoring Instrumentation

3/4.3.3.6 - Post Accident Instrumentation

<u>3/4.4.1</u> - Reactor Coolant Loops - Change includes the Decay Heat Removal System as a system available for decay heat removal

<u>3/4.4.3</u> - Safety Valves

<u>3/4.4.9</u> - Pressure/Temperature Limit

<u>3/4.5</u> - Communication - Bases added to reflect addition of Section 3.5 per Modification of Order dated July 17, 1984

<u>3/4.6.1.4</u> - Internal Pressure

<u>3/4.6.1.5</u> - Air Temperature

<u>3/4.7.9</u> - Sealed Source Contamination

<u>3/4.9.13 and 3/4.9.14</u> - EPICOR-II Processed Water and Reactor Building Sump/Accident Generated Water

(F) <u>Design Features</u>

<u>Section 5.2.2 -Containment Design Pressure and Temperature</u> - Originally, this section had a provision ("per occupational exposure") that if radiological conditions were severe enough, the stated requirement need not be performed. The non-performance of the PTS requirement did not significantly impact the ability to safely maintain the facility in a shutdown condition and was therefore previously acceptable to the staff. However, radiological conditions have changed to such a degree that access to the subject areas is now permitted. Therefore, the radiological exemption is no longer necessary and has therefore been deleted.

<u>Section 5.4.1 - Reactor Coolant System Design Pressure and Temperature</u> - This change was initiated by the NRC and discussed and agreed upon by the licensee. When considering the transients that occurred during the accident

and the minimum preventative maintenance that has been performed on the RCS, it is not possible to state exact design features. Also, there are no credible transients that could drastically change the existing RCS pressure and temperature. Therefore, a design basis pressure and temperature is not required for TMI at this time. Should the status of the RCS change, the staff will re-evaluate its position on this matter.

(G) Recovery Operations Plan

Proposed Technical Specification/Recovery Operations Plan Tables - Tables moved to the Recovery Operations Plan (ROP) - The following tables previously contained in the Limiting Conditions for Operations Section have been moved in their entirety to the ROP. The essential requirements continue to be reflected in the limiting conditions for operation. Where modifications have also been made to the table, the staff has provided additional discussions. If no additional discussion is present, there has been no modification of the requirement.

Tables Moved from LCO Sections to the Recovery Operations Plan

Name	Previous Location	New Location
Neutron Monitoring Instrumentation	3.3-1	4.3-1
Engineered Safety Feature Actuation System Instrumentation	3.3-3	4.3-2
Engineered Safety Feature Actuation System Instrumentation Setpoints	3.3-4	4.3-2

*Seismic Monitoring Instrumentation	3.3-7	deleted
Meteorological Monitoring Instrumentation	3.3-8	4.3-5
*Remote Shutdown Monitoring Instrumentation	3.3-9	4.3-7
*Post Accident Monitoring Instrumentation	3.3-10	deleted
Fire Detection Instrumentation	3.3-11	4.3-11
*Minimum Allowable Containment Pressure as a Function of BWST Temperature and Containment Average Air Temperature	3.6-1	deleted
Fire Hose Stations	3.7-4	4.7-1
AC Distribution Table	Section 3.8.2.1	4.8.2.1

*indicates other modifications have been made as discussed herein

Section 4.3.3.2 and Section 4.3.3.3 - Radiation Monitoring Instrumentation - The licensee has deleted the AMS-3 particulate monitor used for the EPICOR-II prefilter purging and inerting operation because this activity has been completed. When this monitor was incorporated into the ROP via Change No. 16, dated September 2, 1982, the staff also incorporated a lower limit of detectability (LLD) requirement for particulates. Because the AMS-3 monitor is no longer required, the staff concurs with the deletion of the LLD requirements.

Section 4.3.3.3.1 and Table 4.3-4 - Seismic Instrumentaton - see previous discussion on Section 3.3.3.3.

Table 4.3-1 - Reactor Protection System Instrumentation - Surveillance Requirements, Now Called Neutron Monitoring - The licensee has renamed the subject table to more accurately reflect the instruments listed therein. None of the previously existing requirements have been modified. The "Total No. of Channels," "Channels to Trip," and "Minimum Channels Operable" have been added. The staff concurs with these additions.

Table 4.3-3 - Radiation Monitoring Instrumentation Surveillance Requirements - The licensee has proposed to add radiation monitor requirements for the MDHR and SDS systems. Also appropriate action requirements have been added.

The staff has added several additional requirements. A requirement for a reactor building equipment door monitor (AMS-3) has been included. Since the equipment airlock volume communicates with the outside environment whenever both of the equipment airlock doors are open, this requirement will assure that there will not be an unmonitored release. The staff has also added a requirement for a gamma monitor in the area near the reactor vessel flange during periods when the head is removed. This will provide additional information during core alterations on reactivity changes. Appropriate action statements have been added for these two staff revisions.

Section 4.3.3.5/Table 4.3-6 - Remote Shutdown Instrumentation and Section 4.3.3.6/Table 4.3-10 - Post Accident Monitoring Instrumentation - See previous discussion on Sections 3.3.3.5 and 3.3.3.6.

Table 4.3-5 - Meteorological Monitoring Instrumentation Surveillance Requirements - The licensee has added a category to the subject table which states the minimum operable number of instruments required. Each category, wind speed, wind direction, and air temperature (33 ft. and 150 ft.) requires one instrument. Based on the benign condition of the TMI-2 reactor, it is the staff's opinion that this requirement is adequate and therefore concurs with the addition.

Table 4.3-7 - Essential Parameters Monitoring Instrumentation Surveillance Requirements - As discussed in Sections 3.3.3.5, 3.3.3.6, 4.3.3.5 and 4.3.3.6, this requirement evolved by combining the requirements of Remote Shutdown Instrumentation and Post Accident Monitoring Instrumentation. While combining these parameters, the licensee has omitted certain instrumentation as follows:

Reactor Coolant Inlet Temperature - Since the RCS is partially filled, with the RCS outlet temperature instruments above the water level, the licensee has combined the RCS inlet and RCS outlet temperature instrumentation requirements into one, titled RCS temperature. The RCS inlet temperature instrumentation is still submerged and will be used as the RCS temperature indicator. The proposed change does not permit the licensee to discontinue RCS temperature monitoring but correctly reflects the plant's present condition. The staff therefore approves the changes.

Make Up Tank Level - Since the make up tank is not relied upon for any safety actions, the associated instrumentation is therefore not required.

Section 4.3.3.8.1 - Fire Detection - The licensee's proposal did not change the request but moved all "occupational exposure consideration" statements from this section to a composite table. It is the staff's opinion that this movement made the subject requirements ambiguous; therefore, the requested changes are denied.

Section 4.3.3.8.4 - Fire Detection - The staff has added a requirement to have fire detection capabilities in the Southeast Storage Facility since low level waste is sometimes stored in this facility, and it is the staff's opinion that the ability to detect a fire in this area is necessary.

Table 4.3-11 - Fire Detection Instruments - This table was formerly 3.3-11. No requirements were deleted from the table with the change of location. However, based on previous discussions in this SER on Section 3.7.10.2, a requirement for fire detection instrumentation for the Southeast Storage Facility has been added.

Section 4.4.3 - Safety Valves - See discussion on Section 3.4.3 of the PTS.

Section 4.4.9.14 - Reactor Coolant System - The staff inadvertently omitted maximum pH limits incorporated in ROP Change No. 19 dated April 24, 1984 with the issuance of ROP Change No. 20 dated July 17, 1984.

Section 4.6.1.3.c - Containment Air Locks - See discussion on Section 3.6.1.3.b.

Section 4.7.3.1 - Nuclear Services Closed Cooling Water - Originally, this section had a provision ("per occupational exposure") that if radiological conditions were severe enough, the stated requirement need not be performed. The non-performance of the PTS requirement did not significantly impact the ability to safely maintain the facility in a shutdown condition and was therefore previously acceptable to the staff. However, radiological conditions have changed to such a degree that access to the subject areas is now permitted. Therefore, the radiological exemption is no longer necessary and has therefore been deleted.

Section 4.7.3.3 - Mini-Decay Heat Removal System - See comments on Specification 4.1.1.1.

Section 4.7.4.1 - Nuclear Service River Water System - See discussion on Section 4.7.3.1.

Section 4.7.9 - Sealed Sources - See discussion on Section 3.7.9.

Section 4.7.10 - Fire Suppression Systems - The licensee had proposed to delete all clauses like those discussed in Section 4.7.3.1 and replace the requirements with a table (Table 4.7-2) would indicate the number of fire barriers that are accessible for surveillance. Per discussions with and agreed to by the licensee, it is the staff's opinion that this proposed modification does not improve the understandability of this section and may even make its interpretation confusing. Therefore, we are not approving this change to Section 4.7.10. The staff does concur with the movement of Table 3.7 to Table 4.7-1. This new location did not involve any modifications to the requirements.

Section 4.8.2.1 - A.C. Distribution - The licensee has moved the list of A.C. electrical buses from Section 3.8.2.1 to Section 4.8.2.1 in its entirety. This change is editorial with no change in requirements; therefore the staff concurs.

ENVIRONMENTAL CONSIDERATIONS

We have determined that the changes do not authorize a change in effluent types or total amounts nor an increase in power level and will not result in any significant environmental impact. Having made this determination, and, as reflected in the Environmental Assessment and Notice of Finding of No Significant Environmental Impact prepared pursuant to 10 CFR 51.21 and 51.30 through 51.32, issued concurrently herewith, we have further concluded that the change involves an action which is insignificant from the standpoint of environmental impact and that an environmental impact statement need not be prepared in connection with the issuance of this action.

CONCLUSION

Based upon our review of the above discussed changes as modified, the staff finds that the requested revision of the proposed Technical Specifications is acceptable.

We have also 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 implementation of this change will not be inimical to the common defense and security or to the health and safety of the public.

ENCLOSURES 3 AND 4

FACILITY OPERATING LICENSE NO. DPR-73

DOCKET NO. 50-320

The majority of all pages of the Appendix "A", Proposed Technical Specifications and the TMI-2 Recovery Operations Plan have been modified as a result of this Amendment of Order. Therefore, you should replace all of Appendix A and the Recovery Operations Plan pages with Enclosures 3 and 4. Indexes have also been altered to reflect the subject changes.

ENCLOSURE 3

INDEX

DEFINITIONS

INDEX

SAFETY LIMITS

BASES

INDEX

LIMITING CONDITIONS FOR OPERATION

LIMITING CONDITIONS FOR OPERATION

INDEX

LIMITING CONDITIONS FOR OPERATION

SURVEILLANCE REQUIREMENTS

INDEX

BASES

SECTION | PAGE

INDEX

BASES

SECTION		PAGE

INDEX

DESIGN FEATURES

ADMINISTRATIVE CONTROLS

INDEX

ADMINISTRATIVE CONTROLS

INDEX

SURVEILLANCE REQUIREMENTS

INDEX

SURVEILLANCE REQUIREMENTS

SURVEILLANCE REQUIREMENTS

SECTION 1.0

DEFINITIONS

1.0 DEFINITIONS

DEFINED TERMS

1.1 The DEFINED TERMS of this section appear in capitalized tyoe and are applicable throughout these Technical Specifications.

RECOVERY OPERATIONS PLAN

1.2 The RECOVERY OPERATIONS PLAN shall define the surveillance requirements to be performed to ensure equipment operability as required by the Limiting Conditions for Operation. This plan, and changes thereto, shall be approved by the Commission prior to implementation.

RECOVERY MODE

1.3 The RECOVERY MODE shall correspond to a condition in which the reactor is subcritical with an average reactor coolant temperature of less than 200°F.

ACTION

1.4 ACTION shall be those additional requirements specified as corollary statements to each specification and shall be part of the specifications.

OPERABLE - OPERABILITY

1.5 A system, subsystem, train, component or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified function(s). Implicit in this definition shall be the assumption that all necessary attendant instrumentation, controls, normal and emergency electrical power sources, cooling or seal water, lubrication or other auxiliary equipment, that are required for the system, subsystem, train, component or device to perform its function(s), are also capable of performing their related support function(s).

REPORTABLE OCCURRENCE

1.6 A REPORTABLE OCCURRENCE shall be any of those conditions specified in Specifications 6.9.1.8 and 6.9.1.9.

CONTAINMENT INTEGRITY

1.7 CONTAINMENT INTEGRITY shall exist when:

a. All penetrations required to be closed during accident conditions are either:

   1. Capable of being closed by valves on each side of the penetration or by double valve isolation outside of the reactor building per procedures approved pursuant to Specification 6.8.2. Isolation valves inside the reactor building shall be capable of remote operation from a control station outside of the reactor building, or;

   2. Closed by manual valves, blind flanges, or deactivated automatic valves secured in their closed positions to provide double isolation of each penetration.

b. The Equipment Hatch is closed and sealed.

c. Each airlock is OPERABLE pursuant to Specification 3.6.1.3.

d. The sealing mechanism associated with each penetration (e.g., welds, bellows or O-rings) is OPERABLE.

CHANNEL CALIBRATION

1.8 A CHANNEL CALIBRATION shall be the adjustment, as necessary, of the channel output such that it responds with necessary range and accuracy to known values of the parameter which the channel monitors. The CHANNEL CALIBRATION shall encompass the entire channel including the sensor and alarm and/or trip functions, and shall include the CHANNEL FUNCTIONAL TEST. CHANNEL CALIBRATION may be performed by any series of sequential, overlapping or total channel steps such that the entire channel is calibrated.

CHANNEL CHECK

1.9 A CHANNEL CHECK shall be the qualitative assessment of channel behavior during operation by observation. This determination shall include, where possible, comparison of the channel indication and/or status with other indications and/or status derived from independent instrument channels measuring the same parameter.

CHANNEL FUNCTIONAL TEST

1.10 A CHANNEL FUNCTIONAL TEST shall be:

a. Analog channels - the injection of a simulated signal into the channel as close to the primary sensor as practicable to verify OPERABILITY including alarm and/or trip functions.

b. Bistable channels - the injection of a simulated signal into the channel sensor to verify OPERABILITY including alarm and/or trip functions.

STAGGERED TEST BASIS

1.11 A STAGGERED TEST BASIS shall consist of:

a. A test schedule for n systems, subsystems, trains or designated components obtained by dividing the specified test interval into n equal subintervals,

b. The testing of one system, subsystem, train or designated components at the beginning of each subinterval.

FREQUENCY NOTATION

1.12 The FREQUENCY NOTATION specified for the performance of Surveillance Requirements shall correspond to the intervals defined in Table 1.2.

FIRE SUPPRESSION WATER SYSTEM

1.13 A FIRE SUPPRESSION WATER SYSTEM shall consist of: a water source; gravity tank or pumps; and distribution piping and associated sectionalizing control or isolation valves. Such valves shall include yard hydrant curb valves, and the first valve upstream of the water flow alarm device on each sprinkler, hose standpipe or spray system riser.

REVIEW SIGNIFICANCE

1.14 Item that are Important to Safety, or proposed changes to Technical Specifications, License, Special Orders or Agreements, Recovery Operations Plan, Organization Plan, or involve an Unreviewed Safety Question or a Significant Environmental Impact. Also, those system operating procedures and associated emergency, abnormal, alarm response procedures which require NRC approval. In addition, those activities which exceed PEIS values.

CORE ALTERATION

1.15 CORE ALTERATION shall be the movement or manipulation of any reactor component (including fuel) within the reactor pressure vessel with the vessel head removed and fuel in the vessel. Suspension of CORE ALTERATION shall not preclude completion of movement of a component to a safe conservative position.

LOSS-TO AMBIENT

1.16 LOSS-TO-AMBIENT is a passive cooling mode by which decay heat, generated by the reactor core, is removed and transferred to the surrounding environment by air and passive components (i.e., Reactor Vessel) inside the Reactor Building.

ACCIDENT GENERATED WATER

1.17 ACCIDENT GENERATED WATER, as defined in the settlement of the City of Lancaster litigation, is:

(a) Water that existed in the TMI-2 Auxiliary, Fuel Handling, and Containment Buildings including the primary system as of October 16,1979, with the exception of water which as a result of decontamination operations becomes commingled with non-accident generated water such that the commingled water has a tritium content of 0.025 μCi/ml or less before processing;

(b) Water that has a total activity of greater than one μCi/ml prior to processing except where such water is originally non-accident water and becomes contaminated by use in cleanup;

(c) Water that contains greater than 0.025 μCi/ml of tritium before processing.

TABLE 1.2

FREQUENCY NOTATION

NOTATION	FREQUENCY
S	At least once per 12 hours.
D	At least once per 24 hours.
W	At least once per 7 days.
M	At least once per 31 days.
Q	At least once per 92 days.
SA	At least once per 184 days.
A	At least once per 12 months.
R	At least once per 18 months.
N.A.	Not applicable.

SECTION 2.0

SAFETY LIMITS

AND

LIMITING SAFETY SYSTEM SETTINGS

2.0 SAFETY LIMITS

2.1 SAFETY LIMITS

2.1.3 REACTOR COOLANT SYSTEM PRESSURE

Not applicable.

BASES

FOR

SAFETY LIMITS

NOTE

The summary statements contained in this section provide the bases for the specifications of Section 2.0 and are not considered a part of these Technical Specifications as provided in 10 CFR 50.36.

2.1 SAFETY LIMITS

BASES

2.1.3 REACTOR COOLANT SYSTEM PRESSURE

The reactor coolant system is currently open to the reactor building atmosphere. Should reactor coolant system repressurization become necessary during the course of the recovery, a maximum pressure rating will be specified by an NRC approved safety evaluation.

SECTIONS 3.0 AND 4.0

LIMITING CONDITIONS FOR OPERATION

AND

SURVEILLANCE REQUIREMENTS

3 LIMITING CONDITIONS FOR OPERATION

3.0 APPLICABILITY

LIMITING CONDITIONS FOR OPERATION

3.0.1 Limiting Conditions for Operation and ACTION requirements shall be applicable during the RECOVERY MODE or other conditions specified for each specification.

3.0.2 Adherence to the requirements of the Limiting Condition for Operation and/or associated ACTION within the specified time interval shall constitute compliance with the specification. In the event the Limiting Condition for Operation is restored prior to expiration of the specified time interval, completion of the ACTION statement is not required.

3.0.3 In the event a Limiting Condition for Operation and/or associated ACTION requirements cannot be satisfied because of circumstances in excess of those addressed in the specification, initiate appropriate actions to rectify the problem to the extent possible under the circumstances, and take all other actions necessary to maintain the unit in a stable condition; promptly notify the NRC Site Staff; and submit a report to the Commission pursuant to Specification 6.9.1.8.b.

3.1 WATER INJECTION COOLING AND REACTIVITY CONTROL SYSTEMS

3.1.1 BORATION CONTROL

BORATED COOLING WATER 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 the Standby Reactor Coolant System Pressure Control System.

b. The second system comprised of:

1. 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 a boron concentration between 3500 and 6000 ppm) and either
2. Two Mini Decay Heat Removal Pumps and heat exchangers and the associated flow path shall be OPERABLE, or
3. One Decay Heat Removal Pump and associated flow path shall be operable.

APPLICABILITY: RECOVERY MODE

ACTION:

a. With the Standby Reactor Coolant System Pressure Control System inoperable, suspend all operations involving CORE ALTERATIONS and/or the Reactor Coolant System and restore the inoperable system to OPERABLE status within 72 hours.

b. With no Decay Heat Removal Pumps or associated flowpath operable; and one Mini Decay Heat Removal Pump or heat exchanger or associated flow path inoperable, restore, to operable status within 72 hours either:

i) one Decay Heat Removal Pump and associated flowpath, or

ii) the inoperable Mini Decay Heat Removal Pump and heat exchanger and associated flow path.

c. With no Decay Heat Removal Pumps or associated flowpath operable; and two Mini Decay Heat Removal Pumps or heat exchangers or the associated flowpath inoperable, immediately suspend all operations involving CORE ALTERATIONS, or the Reactor Coolant System, verify the LOSS-TO-AMBIENT cooling mode is functioning, and restore to Operable status within 72 hours either:

i) one Decay Heat Removal Pump and associated flowpath, or

ii) two Mini Decay Heat Removal Pumps and heat exchangers and the associated flowpath

BORON CONCENTRATION

3.1.1.2 The boron concentration of the coolant in all filled portions of the Reactor Coolant System and the Refueling Canal shall be maintained at a boron concentration between 3500 and 6000 ppm and at a temperature above 50°F.

APPLICABILITY: RECOVERY MODE

ACTION:

If either of the above conditions are not satisfied (Boron Concentration between 3500 and 6000 ppm and temperature above 50°F) immediately suspend all activities involving CORE ALTERATION, the Reactor Coolant System or any operation in the Refueling Canal and take action in accordance with procedures approved pursuant to Specification 6.8.2 to restore the concentration and/or temperature to within acceptable limits.

3.1.3 CONTROL ASSEMBLIES

MECHANISMS

3.1.3.1 deleted.

LIMITING CONDITIONS FOR OPERATION

3.3 INSTRUMENTATION

3.3.1 NEUTRON MONITORING INSTRUMENTATION

INTERMEDIATE AND SOURCE RANGE NEUTRON FLUX MONITORS

3.3.1.1 As a minimum, the intermediate and source range neutron monitoring instrumentation channels of Table 4.3-1 shall be OPERABLE.

APPLICABILITY: RECOVERY MODE

ACTION:

a. With the number of source range neutron monitoring channels OPERABLE one less than required by the Minimum Channels OPERABLE requirement of Table 4.3-1, restore inoperable channel to OPERABLE status within 30 days. If the inoperable channel cannot be restored to OPERABLE status within 30 days, prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within the next 30 days outlining the cause of the malfunction and the plans for monitoring the condition of the core.

b. With no source range neutron monitoring channels OPERABLE, suspend all activities involving CORE ALTERATION, verify compliance with the boron concentration requirements of Specification 3.1.1.2 at least once per 24 hours by a mass balance calculation and at least once per 7 days by a chemical analysis and restore at least one source range neutron monitoring channel to operable status within 7 days. If not restored to operable status within 7 days, promptly, but not later than 30 days from loss of operability, prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2, outlining the cause of the malfunction(s), the plans for monitoring the condition of the core and the plans for resumption of activities involving CORE ALTERATIONS.

c. With no intermediate range neutron monitoring channels OPERABLE, restore at least one intermediate range channel to OPERABLE status within 7 days. If not restored to OPERABLE status within 7 days, promptly, but not later than 30 days from loss of OPERABILITY, prepare and submit a special report to the Commission pursuant to Specification 6.9.2, outlining the cause of the malfunction(s) and the plans for monitoring the condition of the core.

3.3.2 ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

3.3.2.1 The Engineered Safety Feature Actuation System (ESFAS) instrumentation channels shown in Table 4.3-2 shall be OPERABLE with their Trip Setpoints set in accordance with the values shown in the Trip Setpoint column of Table 4.3-2.

Table 3.3-1

NEUTRON MONITORING INSTRUMENTATION

TRANSFERRED TO TABLE 4.3-1 OF THE RECOVERY OPERATIONS PLAN

TABLE 3.3-3

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

TRANSFERRED TO TABLE 4.3-2 OF THE RECOVERY OPERATIONS PLAN

TABLE 3.3-4

ENGINEERED SAFETY FEATURE ACTUATION SYSTEMS INSTRUMENTATION TRIP SETPOINTS

TRANSFERRED TO TABLE 4.3-2 OF THE RECOVERY OPERATIONS PLAN

APPLICABILITY: RECOVERY MODE.

ACTION:

a. With an ESFAS instrumentation channel trip setpoint less conservative than the value shown in the Allowable Values column of Table 4.3-2 declare the channel inoperable and apply the applicable ACTION requirement of Table 4.3-2 until the channel is restored to OPERABLE status with the trip setpoint adjusted consistent with the Trip Setpoint Value.

b. With an ESFAS instrumentation channel inoperable, take the action shown in Table 4.3-2.

3.3.3 MONITORING INSTRUMENTATION

RADIATION MONITORING INSTRUMENTATION

3.3.3.1 The radiation monitors listed in Table 4.3-3 of the RECOVERY OPERATIONS PLAN shall be OPERABLE

APPLICABILITY: As Required in Table 4.3-3 of the RECOVERY OPERATIONS PLAN.

ACTION:

As Required in Table 4.3-3 of the RECOVERY OPERATIONS PLAN.

SEISMIC INSTRUMENTATION

3.3.3.3 Deleted

METEOROLOGICAL INSTRUMENTATION

3.3.3.4 The meteorological monitoring instrumentation channels shown in Table 4.3-5 of the RECOVERY OPERATIONS PLAN shall be OPERABLE.

APPLICABILITY: RECOVERY MODE

ACTION:

With any of the above required meteorological monitoring channels inoperable, restore the inoperable channel(s) to OPERABLE status within 8 hours.

TABLE 3.3-7

SEISMIC MONITORING INSTRUMENTATION

DELETED

TABLE 3.3-8

METEOROLOGICAL MONITORING INSTRUMENTATION

Transferred to Table 4.3-5 of the Recovery Operations Plan.

ESSENTIAL PARAMETERS MONITORING INSTRUMENTATION

3.3.3.5 The Essential Parameters Monitoring Instrumentation shall be OPERABLE in accordance with the requirements of Table 4.3-7 of the RECOVERY OPERATIONS PLAN.

APPLICABILITY: RECOVERY MODE.

ACTION:

a. With the exception of the Reactor Vessel Water Level Monitoring instrumentation, for instrumentation not in accordance with the requirements of Table 4.3-7 of the RECOVERY OPERATIONS PLAN, restore the inoperable instrument(s) to the requirements of Table 4.3-7 of the RECOVERY OPERATIONS PLAN within 72 hours.

b. The operability requirements for the Reactor Vessel Water Level monitoring instrumentation shall be as specified in specification 3.4.2.

POST-ACCIDENT INSTRUMENTATION

3.3.3.6 Deleted.

CHLORINE DETECTION SYSTEMS

3.3.3.7 Two chlorine detection systems, with their alarm/trip setpoints adjusted to actuate at a chlorine concentration of less than or equal to 5 ppm, shall be OPERABLE:

a. One at the air intake tunnel, and

b. One at the Control Room air supply duct.

APPLICABILITY: RECOVERY MODE.

ACTION:

With one or more chlorine detection systems inoperable, within 1 hour initiate and maintain operation of the Control Room Emergency Ventilation System in the recirculation mode of operation; restore the inoperable detection system to OPERABLE status within 30 days.

FIRE DETECTION

3.3.3.8 As a minimum, the fire detection instrumentation for each fire detection zone shown in Table 4.3-11 shall be OPERABLE.

APPLICABILITY: RECOVERY MODE

ACTION:

With the number of OPERABLE fire detection instruments less than required by Table 4.3-11 of the RECOVERY OPERATIONS PLAN, insure that an alternate instrument with the same coverage is OPERABLE, or;

1. Within 1 hour, establish a fire watch patrol to inspect the zone with the inoperable instrument(s) at least once per hour, and

2. Restore the inoperable instrument(s) to OPERABLE status within 14 days or, in lieu of any other report required by Specification 6.9.1, prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within the next 30 days outlining the cause of the malfunction and the plans for restoring the instrument(s) to OPERABLE status.

TABLE 3.3-9

REMOTE SHUTDOWN MONTORING INSTRUMENTATION

Contents of this Table moved to Table 4.3-7 of the Recovery Operation Plan

TABLE 3.3-10

POST-ACCIDENT MONITORING INSTRUMENTATION

Deleted

TABLE 3.3-11

FIRE DETECTION INSTRUMENTS

Transferred to Table 4.3-11 of the Recovery Operations Plan

LIMITING CONDITIONS FOR OPERATION

3.4 REACTOR COOLANT SYSTEM

REACTOR COOLANT LOOPS

3.4.1 The Reactor Coolant System shall be operated in accordance with procedures approved pursuant to Specification 6.8.2.

APPLICABILITY: RECOVERY MODE.

ACTION:

None except as provided in Specification 3.0.3.

REACTOR VESSEL WATER LEVEL MONITORING

3.4.2 As a minimum two independent reactor vessel level monitoring instruments shall be OPERABLE.

APPLICABILITY: RECOVERY MODE WITH THE RV HEAD REMOVED

ACTION

a. With only one reactor vessel level monitoring instrument OPERABLE, terminate all activities involving changes in the reactor coolant system water volume, restore the system to OPERABLE status within 72 hours.

b. With no reactor vessel level monitoring instrument OPERABLE, terminate all activities involving changes in the reactor coolant system water volume. Restore the system to OPERABLE status within 24 hours or, in lieu of any other report required by 10 CFR 50.73, prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within the next 30 days outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the system to OPERABLE status.

SAFETY VALVES

3.4.3 Deleted.

3.4.9 PRESSURE/TEMPERATURE LIMITS

REACTOR COOLANT SYSTEM

3.4.9.1 The Reactor Coolant System shall be maintained between a T_{avg} of less than 200°F and greater than 50°F.

3.4.9.2 The Reactor Coolant System shall remain open to the reactor building atmosphere unless repressurization is approved in a safety evaluation submitted to the NRC. This safety evaluation and associated procedures approved pursuant to Specification 6.8.2 shall specify the maximum pressure limits and over-pressure protection that is required.

---

*All makeup pumps shall be made inoperable by racking out their electrical power supply circuit breakers.

APPLICABILITY: RECOVERY MODE

ACTION:

With the Reactor Coolant System temperature exceeding the temperature limits, immediately adjust the Reactor Coolant System temperature to within limits and submit a report pursuant to Specification 3.0.3.

3.5 COMMUNICATIONS

3.5.1 Control Room

Direct communication shall be maintained between the Control Room or the Command Center and personnel in the Reactor Building. As stated in Specification 6.2-1, the additional SOL or SOL limited to fuel handling notwithstanding location, will have direct communications with personnel in the Reactor Building.

APPLICABILITY: During CORE ALTERATIONS

ACTION:

When direct communication between the Control Room or the Command Center and personnel in the Reactor Building as stated in Specification 6.2-1 cannot be maintained, suspend all operations involving CORE ALTERATIONS and restore communications to OPERABLE status.

3.6 CONTAINMENT SYSTEMS

3.6.1 PRIMARY CONTAINMENT

CONTAINMENT INTEGRITY

3.6.1.1 Primary CONTAINMENT INTEGRITY shall be maintained unless it is not required per procedures approved pursuant to Specification 6.8.2.

APPLICABILITY: RECOVERY MODE.

ACTION:

With containment integrity required and one containment isolation valve per containment penetration open or inoperable, maintain the affected penetration(s) closed with either:

a. At least one deactivated automatic valve secured in the isolation position, or

b. At least one closed manual valve, or a blind flange.

CONTAINMENT AIR LOCKS

3.6.1.3 Each containment air lock shall be OPERABLE with:

a. Both doors closed except when the air lock is being used for transit entry and exit through the containment, then at least one air lock door shall be closed unless otherwise specified per procedures approved pursuant to Specification 6.8.2.

b. Deleted

APPLICABILITY: RECOVERY MODE.

ACTION:

With an air lock inoperable, maintain at least one door closed and restore the air lock to OPERABLE status within 24 hours.

INTERNAL PRESSURE

3.6.1.4 Primary containment pressure shall be maintained from 0 psig to not less than 12.2 psia.

APPLICABILITY: RECOVERY MODE.

ACTION:

With the containment internal pressure outside the above limits, restore the internal pressure to within the limits within 1 hour.

AIR TEMPERATURE

3.6.1.5 Primary containment average air temperature shall be maintained between 50°F and 130°F.

APPLICABILITY: RECOVERY MODE.

ACTION:

With the containment average air temperature outside the above limits, restore the average air temperature to within the limits within 24 hours.

3.6.3 CONTAINMENT PURGE EXHAUST SYSTEM

3.6.3.1 One train of the Containment Purge Exhaust System shall be OPERABLE.

APPLICABILITY: During Purge Operations

ACTION:

With no Containment Purge Exhaust train OPERABLE, secure the Containment Purge System and restore one train to OPERABLE status within 7 days.

3.6.4 COMBUSTIBLE GAS CONTROL

HYDROGEN ANALYZERS

3.6.4.1 One gas partitioner shall be OPERABLE.

APPLICABILITY: RECOVERY MODE.

ACTION:

With the gas partitioner inoperable, restore the inoperable instrument to OPERABLE status within 7 days.

HYDROGEN PURGE CLEANUP SYSTEM

3.6.4.3 Deleted

Figure 3.6-1

Minimum Allowable Containment Pressure as a Function of BWST Temperature and Containment Average Air Temperature

DELETED

3.7 PLANT SYSTEMS

3.7.1 FEEDWATER SYSTEM

Deleted.

3.7.2 SECONDARY SERVICES CLOSED COOLING WATER SYSTEM

Deleted.

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 wihtin 24 hours.

MINI DECAY HEAT REMOVAL SYSTEM

3.7.3.3 Requirements transferred to Specification 3.1.1.1.

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 Service River Water loop inoperable, restore the inoperable loop to OPERABLE status within 72 hours.

3.7.6 FLOOD PROTECTION

3.7.6.1 Flood protection shall be provided for all Safety Related systems, components and structures when the water level of the Susquehanna River exceeds 301 feet Mean Sea Level USGS datum, at the river water intake structure of Three Mile Island Nuclear Station, Unit 1.

APPLICABILITY: At all times.

ACTION:

a. With the water level at the Unit 1 Intake Structure approaching 301 feet Mean Sea Level USGS datum:

   1. Initiate patrol and inspection of the dikes surrounding the site for signs of deterioration such as undermining or excessive seepage.

   2. Inform the Director Site Operations (DSO) and as directed by the DSO:

      a) Prepare all flood panels and door seals for installation,

      b) Check all building floor drains and pumps to ensure proper operation,

      c) Commence daily soundings of the Intake Screen House Floor,

      d) Check all water tight doors to ensure proper operation,

      e) Fill all outdoor storage tanks to inhibit floatation, and

      f) Arrange for alternate supplies of diesel fuel oil and ensure fuel storage tanks are filled.

      g) Check that all containers are sealed and secure in the Southeast Storage Facility.

b. With the water level at the Unit 1 Intake Structure exceeding 301 feet and approaching 302 feet Mean Sea Level USGS datum:

   1. Ensure all door seals and flood panels are installed and all water tight doors are closed within 2 hours,

   2. Inform the Director Site Operations.

3.7.7 CONTROL ROOM EMERGENCY AIR CLEANUP SYSTEM

3.7.7.1 The Control Room Ventilation and Emergency Air Cleanup System shall be OPERABLE with:

a. Two Control Room supply Fans and associated cooling coils,

b. Two Control Room Bypass Fans,

c. One charcoal adsorber and HEPA filter train, and

d. Two isolation dampers in the outside air intake duct.

e. The Control Room air inlet radiation monitor OPERABLE.

APPLICABILITY: RECOVERY MODE

ACTION:

a. With one Control Room Supply Fan or its associated cooling coil inoperable, restore the inoperable fan and/or cooling coil to OPERABLE status within 7 days.

b. With one Control Room Bypass Fan inoperable restore the inoperable fan to OPERABLE status within 7 days.

c. With the filter train inoperable, restore the filter train to OPERABLE status within 24 hours.

d. With one isolation damper in the outside air intake duct inoperable, restore the inoperable damper to OPERABLE status or close the duct within 4 hours by use of at least one isolation damper secured in the closed position.

e. With the Control Room Air Inlet Radiation Monitor inoperable, restore it to OPERABLE status or place the Control Room Emergency Air Cleanup System in the recirculation mode of operation within 4 hours.

3.7.9 SEALED SOURCES

SEALED SOURCE INTEGRITY

3.7.9.1 Each sealed source containing radioactive material either in excess of 100 microcuries of beta and/or gamma emitting material or 5 microcuries of alpha emitting material (except as noted in 4.7.9.2) shall be free of $\geq$ 0.005 microcuries of removable contamination.

APPLICABILITY: RECOVERY MODE

ACTION:

a. Each sealed source with removable contamination in excess of the above limit shall be immediately withdrawn from use and:

   1. Either decontaminated and repaired, or

   2. Disposed of in accordance with Commission Regulations.

b. The provisions of Specification 3.0.3 are not applicable.

3.7.10 FIRE SUPPRESSION SYSTEMS

FIRE SUPPRESSION WATER SYSTEM

3.7.10.1 The FIRE SUPPRESSION WATER SYSTEM shall be OPERABLE with;

a. At least 2 of the following 4 high pressure pumps shall be OPERABLE with their discharge aligned to the fire suppression header:

   1. Unit 1 Circulating Water Flume Diesel Fire Pump
   2. Unit 1 River Water Intake Diesel Fire Pump
   3. Unit 2 River Water Intake Diesel Fire Pump
   4. Unit 1 River Water Intake Motor Fire Pump

b. Two (2) separate water supplies of the following four (4) shall be available with at least 90,000 gallons each:

   1. Altitude Tank
   2. Unit 1 Circulating Water Flume
   3. Unit 1 River Water Intake Structure
   4. Unit 2 River Water Intake Structure

c. An OPERABLE flow path capable of taking suction from a water supply and transferring the water through distribution piping with OPERABLE sectionalizing control or isolation valves to the yard hydrant curb valves and the first valve ahead of the water flow alarm device on each sprinkler, hose standpipe, or spray system riser required to be OPERABLE per Specification 3.7.10.2 and 3.7.10.4.

APPLICABILITY: RECOVERY MODE

ACTION:

a. With 3 pumps or 3 water supplies inoperable, restore the inoperable equipment to OPERABLE status within 7 days or in lieu of any other report required by Specification 6.9.1, prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within the next 30 days outlining the plans and procedures to be used to provide for the loss of redundancy in this system.

FIRE SUPPRESSION WATER SYSTEM (Continued)

ACTION (Continued)

b. With the Fire Suppression Water System otherwise inoperable:

1. Establish a backup Fire Suppression Water System within 24 hours, and

2. Submit a Special Report in accordance with Specification 6.9.2;

    a. By telephone within 24 hours, and

    b. Confirmed by telegraph, mailgram or facsimile transmission no later than the first working day following the event and

    c. In writing within 14 days following the event, outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the system to OPERABLE status.

DELUGE/SPRINKLER SYSTEMS

3.7.10.2 The Deluge and/or Sprinkler Systems located in the following areas shall be OPERABLE:

a. Diesel Generator Fuel Oil Tanks

b. Diesel Generator Building Air Intake

c. Air Intake Tunnel (Deluge - 2 of the 3 zones)

d. Hydrogen Purge Exhaust Filter AH-F-34#

e. Reactor Building Purge Exhaust Filters AH-F-31A/B#

f. Control Room Bypass Filter AH-F-5

g. Diesel Generator Rooms

h. Fuel Handling Building Exhaust Filter AH-F-14A/B#

j. Waste gas disposal filter WDG-F-1

k. Auxiliary Building exhaust filters AH-F-10A/B#

l. Condenser exhaust filters##

m. Auxiliary Building backup exhaust filters**

n. Southeast Storage Facility***

APPLICABILITY: RECOVERY MODE.

ACTION:

With one or more of the above required deluge and/or sprinkler systems inoperable, establish a roving (at least once per hour) fire watch with backup fire suppression equipment for the unprotected area(s) within 1 hour; restore the system to OPERABLE status within 14 days or, in lieu of any other report required by Specification 6.9.1, prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within the next 30 days outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the system to OPERABLE status.

**Supply lines may be isolated by one of four manually operated valves and by a manually operated deluge valve at the deluge station mounted on outside of building.

***This facility's Action Statement shall require a roving fire watch once per 24 hours instead of once per hour.

#Supply line may be isolated by a single manually operated valve.

##Supply lire may be isolated near the standpipe by two manually operated valves.

HALON SYSTEM

3.7.10.3 The following Halon systems shall be OPERABLE with the storage tanks having at least 95% of full charge weight and 90% of full charge pressure (corrected to 70°F).

a. Cable Room and Transformer Room - Control Building 305' elevation.

b. Air Intake Tunnel (4 Zones)

APPLICABILITY: RECOVERY MODE

ACTION:

With one or more of the above required Halon systems inoperable, establish a roving (at least once per hour) fire watch* with backup fire suppression equipment for the unprotected area(s) within 1 hour; restore the system to OPERABLE status within 14 days or, in lieu of any other report required by Specification 6.9.1, prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within the next 30 days outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the system to OPERABLE status.

FIRE HOSE STATIONS

3.7.10.4 The fire hose stations listed in Table 4.7-1 of the RECOVERY OPERATIONS PLAN shall be OPERABLE:

APPLICABILITY: RECOVERY MODE

ACTION:

With one or more of the fire hose stations shown in Table 4.7-1 inoperable, route an additional equivalent capacity fire hose to the unprotected area(s) from an OPERABLE hose station within 1 hour.

*Except in the air intake tunnel where a fire watch is not required.

TABLE 3.7-4

FIRE HOSE STATIONS

Transferred to Table 4.7-1 of the Recovery Operations Plan

PENETRATION FIRE BARRIERS

3.7.11 All Penetration Fire Barriers protecting Safety Related areas shall be functional.

APPLICABILITY: RECOVERY MODE

ACTION:

With one or more of the above-required Penetration Fire Barriers non-functional:

1. Establish a roving (at least once per hour) fire watch on at least one side of the affected penetration within 1 hour, except areas inaccessible due to occupational exposure considerations and

2. For those areas inaccessible due to occupational exposure considerations, prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within the next 30 days outlining the cause of the malfunction and the means for ensuring an adequate firewatch is maintained in the affected area and for restoring the fire barrier(s) to a functional status.

3.8 ELECTRICAL POWER SYSTEMS

3.8.1 A.C. SOURCES

3.8.1.1 As a minimum, the following A.C. electrical power sources shall be OPERABLE:

a. Two physically independent circuits between the offsite transmission network and the onsite Class 1E distribution system.

b. Two separate and independent Class 1E diesel generators each with:

   1. A separate day fuel tank containing a minimum volume of 500 gallons of fuel.

   2. A separate fuel storage system containing a minimum volume of 19,000 gallons of fuel.

   3. A separate fuel transfer pump.

APPLICABILITY: RECOVERY MODE.

ACTION:

a. With either an offsite circuit or diesel generator of the above required A.C. electrical power sources inoperable, demonstrate the OPERABILITY of the remaining A. C. sources by performing Surveillance Requirements 4.8.1.1.1 and 4.8.1.1.2.a.4 in accordance with the applicable row in the Testing Frequency Matrix of Table 3.8-1; restore the full complement of the above required A.C. electrical power sources to OPERABLE status within 7 days.

b. With one offsite circuit and one diesel generator or two offsite circuits or two diesel generators of the above required A. C. electrical power sources inoperable, demonstrate the OPERABILITY of the remaining A.C. sources by performing Surveillance Requirements 4.8.1.1.1 and 4.8.1.1.2.a.4 in accordance with the applicable two rows in the Testing Frequency Matrix of Table 3.8-1; restore at least one of the inoperable sources to OPERABLE status in accordance with the Restoration Time Matrix of Table 3.8-2. Restore the full complement of the above required A.C. electrical power sources to OPERABLE status within 7 days from the time of initial loss.

TABLE 3.8-1

TESTING FREQUENCY MATRIX

INOPERABLE COMPONENT	Component Testing Frequencies: a_1	a_2	b_1	b_2
a_1		*	**	**
a_2	*		**	**
b_1	*	*		*
b_2	*	*	*	

Key:

a_1 Offsite power circuit No. 1
a_2 Offsite power circuit No. 2
b_1 Class IE diesel generator (Red)
b_2 Class IE diesel generator (Green)

*Within 12 hours
**Within 12 hours unless the diesel generator has been demonstrated OPERABLE per Surveillance Requirement 4.8.1.1.2 within the previous 7 days, then demonstration of diesel generator operability is not required.

TABLE 3.8-2

RESTORATION TIME MATRIX

		Restore One Component (Hours)	Restore Other Component (days)
Combination of Two Inoperable Components	aa	24	7
	ab	12	7
	bb	12	7

Note: a and b above correspond to components described in Specification 3.8.1.1, items a and b respectively.

3.8.2 ONSITE POWER DISTRIBUTION SYSTEMS

A.C. DISTRIBUTION

3.8.2.1 The following A.C. electrical busses listed in Section 4.8.2.1 of the RECOVERY OPERATIONS PLAN 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.

APPLICABILITY: RECOVERY MODE.

ACTION:

With less than the above complement of A.C. busses OPERABLE, restore the inoperable bus to OPERABLE status within 8 hours.

D.C. DISTRIBUTION

3.8.2.3 The following D.C. bus trains shall be energized and OPERABLE with tie breakers between bus trains open (unless closed in accordance with approved procedures):

TRAIN "A" consisting of 250/125-volt D.C. bus 2-1DC, 250/125-volt D.C. battery bank A and a full capacity charger.

TRAIN "B" consisting of 250/125-volt D.C. bus 2-2DC, 250/125-volt D.C. battery bank B, and a full capacity charger.

APPLICABILITY: RECOVERY MODE.

ACTION:

a. With one 250/125-volt D.C. bus inoperable, restore the inoperable bus to OPERABLE status within 2 hours.

b. With one 250/125-volt D.C. battery and/or its charger inoperable, restore the inoperable battery and/or charger to OPERABLE status within 2 hours.

3.9 RADIOACTIVE WASTE STORAGE

FUEL HANDLING BUILDING/AUXILIARY BUILDING AIR CLEANUP SYSTEMS

3.9.12.1 The Fuel Handling Building Air Cleanup Exhaust System shall be OPERABLE with exhaust ventilation flow through the HEPA filters during system operation. The Fuel Handling Building Air Cleanup Exhaust System is OPERABLE when two of the four system air cleanup exhaust fans are OPERABLE.

APPLICABILITY: RECOVERY MODE

ACTION:

a. With the Fuel Handling Building Air Cleanup Exhaust System inoperable due to flow requirements, return the flow to within acceptable limits within four (4) hours or;

b. With the Fuel Handling Building Air Cleanup Exhaust System inoperable (other than as allowed in paragraph 3.9.12.1 a above), suspend all operations involving movement of liquid and gaseous radioactive wastes in the Fuel Handling Building (other than sampling evolutions required by the Technical Specifications or RECOVERY OPERATIONS PLAN) until the system is restored to OPERABLE status.

3.9.12.2 The Auxiliary Building Air Cleanup Exhaust System shall be OPERABLE with exhaust ventilation flow through the HEPA filters during system operation. The Auxiliary Building Air Cleanup Exhaust System is OPERABLE when two of the four system air cleanup exhaust fans are OPERABLE.

APPLICABILITY: RECOVERY MODE

ACTION:

a. With the Auxiliary Building Air Cleanup Exhaust System inoperable due to flow requirements, return the flow to within acceptable limits within four (4) hours or;

b. With the Auxiliary Building Air Cleanup Exhaust System inoperable (other than as allowed in paragraph 3.9.12.2 a above), suspend all operations involving movement of liquid and gaseous radioactive wastes in the Auxiliary Building (other than sampling evolutions required by the Technical Specifications or RECOVERY OPERATIONS PLAN) until the system is restored to OPERABLE status.

ACCIDENT GENERATED WATER

3.9.13 Discharge of ACCIDENT GENERATED WATER shall be prohibited until approved by the NRC. ACCIDENT GENERATED WATER shall be discharged in accordance with procedures approved pursuant to Specification 6.8.2.

APPLICABILITY: RECOVERY MODE

ACTION:

None except as provided in Specification 3.0.3.

REACTOR BUILDING SUMP WATER

3.9.14 Deleted.

3.10 DEFUELING OPERATIONS

3.10.1 CRANE TRAVEL - CONTAINMENT BUILDING

LIMITING CONDITIONS FOR OPERATION

3.10.1.1 Loads in excess of 2400 pounds shall be prohibited from travel over the reactor vessel unless the activity is approved by the NRC.

APPLICABILITY: RECOVERY MODE

ACTION:

With the requirements of the above specification not satisfied, place the crane load in a safe condition. Prepare and submit a Special Report to the Commission pursuant to specification 6.9.2 within the next 30 days.

3.10.1.2 Loads in excess of 2400 pounds shall be prohibited from travel over the following areas unless the activity is approved by the NRC.

a) Incore Instrument Seal Table and Guide Tubes (includes travel by polar crane block).
b) Areas not previously analyzed in an NRC approved load drop analysis.

APPLICABILITY: RECOVERY MODE

ACTION:

With the requirements of the above specification not satisfied, place the crane load in a safe condition. Prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within the next 30 days.

Section 4.0

Surveillance Requirements

SURVEILLANCE REQUIREMENTS

4.0.1 Surveillance Requirements of the RECOVERY OPERATIONS PLAN shall be applicable during the RECOVERY MODE or other conditions specified for individual Limiting Conditions for Operation unless otherwise stated in an individual Surveillance Requirement. The Surveillance Requirements shall be performed to demonstrate compliace with the OPERABILITY requirements of the Limiting Conditions for Operations and in accordance with the RECOVERY OPERATIONS PLAN; however, the RECOVERY OPERATIONS PLAN shall not be considered a part of these technical specifications. Changes to the RECOVERY OPERATIONS PLAN shall be approved by the NRC prior to implementation.

4.0.2 Each Surveillance Requirement of the RECOVERY OPERATIONS PLAN shall be performed within the specified time interval with:

a. A maximum allowable extension not to exceed 25% of the surveillance interval, and

b. A total maximum combined interval time for any 3 consecurive tests not to exceed 3.25 times the specified surveillance interval.

4.0.3 Performance of a Surveillance Requirement within the specified time interval shall constitute compliance with OPERABILITY requirements for a Limiting Condition for Operation and associated ACTION statements unless otherwise required by the specification.

BASES

FOR

LIMITING CONDITIONS FOR OPERATION

SURVEILLANCE REQUIREMENTS

NOTE

The summary statements contained in this section provide the bases for the Specifications of Section 3.0 and 4.0 and are not considered a part of these Technical Specifications as provided in 10 CFR 50.36.

BASES

The specifications of this section provide the general requirements applicable to each of the Limiting Conditions for Operation and Surveillance Requirements within Section 3/4.

3.0.1 This specification defines the applicability of each specification in terms of the RECOVERY MODE or other specified conditions and is provided to delineate specifically when each specification is applicable.

3.0.2 This specification defines those conditions necessary to constitute compliance with the terms of an individual Limiting Condition for Operation and associated ACTION requirement.

3.0.3 This specification delineates the ACTION to be taken for circumstances not directly provided for in the ACTION statements or when the provisions of the stated ACTION statement are not or cannot be complied with. The intent of this specification is to require that inoperable equipment be restored to an OPERABLE status in a prompt manner, that the unit be maintained in stable conditions, and that the Commission be promptly notified of such conditions.

4.0.1 This specification provides that surveillance activities necessary to insure the Limiting Conditions for Operation are met and will be performed during the RECOVERY MODE or other conditions for which the Limiting Conditions for Operation are applicable.

4.0.2 The provisions of this specification provide allowable tolerances for performing surveillance activities beyond those specified in the nominal surveillance interval. These tolerances are necessary to provide operational flexibility because of scheduling and performance considerations. The phrase "at least" associated with a surveillance frequency does not negate this allowable tolerance value and permits the performance of more frequent surveillance activities.

The tolerance values, taken either individually or consecutively over 3 test intervals, are sufficiently restrictive to ensure that the reliability associated with the surveillance activity is not degraded beyond that obtained from the nominal specified interval.

APPLICABILITY

BASES

4.0.3 The provisions of this specification set forth the criteria for determination of compliance with the OPERABILITY requirements of the Limiting Conditions for Operations. Under this criteria, equipment, systems or components are assumed to be OPERABLE if the associated surveillance activities have been satisfactorily performed within the specified time interval. Nothing in this provision is to be construed as defining equipment, systems or components OPERABLE, when such items are found or known to be inoperable although still meeting the Surveillance Requirements.

3/4.1.1 BORATION CONTROL

The limitation on minimum boron concentration ensures that the core will remain subcritical under all credible conditions which may exist during the long-term cooling mode. The maximum boron concentration is provided to ensure that precipitation of boron will not occur in the reactor coolant system and thereby cause possible flow restrictions. The specification requiring the OPERABILITY of two systems capable of injecting borated cooling water into the Reactor Coolant system are provided to ensure that makeup water can be injected into the reactor coolant system and that the reactor coolant will be maintained within the required boron concentration limits. The required volume of borated water in the BWST provides sufficient water to keep the core covered in the event of an unisolatable leak from the reactor vessel. The specified water volume is sufficient to provide a continuous supply of water to the vessel during the interim period before a sump recirculating flow path can be placed in service. Requirements for operability of the mini decay heat removal system or an operable decay heat removal pump are stated in order to provide for injection of borated cooling water to the reactor coolant system from the BWST.

3/4.1.3 CONTROL ASSEMBLIES

All full-length control rods were fully inserted as a result of the reactor trip on March 28, 1979. This Specification has been deleted since the reactor vessel head has been removed.

3/4.3 INSTRUMENTATION

BASES

3/4.3.1 NEUTRON MONITORING INSTRUMENTATION

The neutron monitoring instrumentation, which was included in the normal Reactor Protection System Instrumentation, provides information regarding the shutdown status of the core and it will be used to monitor changes in neutron generation.

3/4.3.2 ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

Except for automatic starting of the diesel generators on loss of offsite power, all automatic features of the ESFAS instrumentation have been defeated. This action prevents inadvertent actuation of the ESF systems. The diesel generators will start automatically on loss of offsite power.

3/4.3.3 MONITORING INSTRUMENTATION

3/4.3.3.1 RADIATION MONITORING INSTRUMENTATION

The OPERABILITY of the radiation monitoring channels ensures that 1) the radiation levels are continually measured in the areas served by the individual channels and 2) the alarm or automatic action is initiated when the radiation level trip setpoint is exceeded.

3/4.3.3.3 SEISMIC INSTRUMENTATION

Deleted.

3/4.3.3.4 METEOROLOGICAL INSTRUMENTATION

The OPERABILITY of the meteorological instrumentation ensures that sufficient meteorological data is available for estimating potential radiation doses to the public as a result of routine or accidental release of radioactive materials to the atmosphere. This capability is required to evaluate the need for initiating protective measures to protect the health and safety of the public. This instrumentation is consistent with the recommendations of Regulatory Guide 1.23, "Onsite Meteorological Programs," February 1972.

3/4.3.3.5 ESSENTIAL PARAMETERS MONITORING INSTRUMENTATION

The OPERABILITY of the Essential Parameters Monitoring Instrumentation ensures that sufficient information is available on selected plant parameters to monitor and assess these variables. Neutron monitoring and Reactor Coolant System temperature indication are provided outside the Control Room in the event that Control Room habitability is lost. This is consistent with General Design Criterion 19 of 10 CFR 50, Appendix A.

3/4.3.3.6 POST-ACCIDENT INSTRUMENTATION

Deleted.

3/4.3.3.7 CHLORINE DETECTION SYSTEMS

The OPERABILITY of the chlorine detection systems ensures that an accidental chlorine release will be detected promptly and the Control Room Emergency Ventilation System will automatically isolate the Control Room and initiate its operation in the recirculation mode to provide the required protection. The Chlorine Detection Systems required by this specification are consistent with the recommendations of Regulatory Guide 1.95, "Protection of Nuclear Power Plant Control Room Operators Against an Accidental Chlorine Release," February 1975.

3/4.3.3.8 FIRE DETECTION INSTRUMENTATION

OPERABILITY of the Fire Detection Instrumentation ensures that adequate warning capability is available for the prompt detection of fires. This capability is required in order to detect and locate fires in their early stages. Prompt detection of fires will reduce the potential for damage to Safety Related equipment and is an integral element in the overall facility fire protection program.

In the event that a portion of the Fire Detection Instrumentation is inoperable, the establishment of frequent fire patrols in the affected areas is required to provide detection capability until the inoperable instrumentation is returned to service.

3/4.4 REACTOR COOLANT SYSTEM

BASES

3/4.4.1 REACTOR COOLANT LOOPS

Alternative methods are available for removal of reactor decay heat. These methods include use of the Decay Heat Removal System, the Mini Decay Heat Removal System, and the "Loss to Ambient" cooling mode. Either of these cooling methods provides adequate cooling of the reactor and each method is available for decay heat removal. Procedures have been prepared and approved for use of these cooling methods.

3/4.4.2 REACTOR VESSEL WATER LEVEL MONITORING

The Reactor Vessel Water Level Monitor ensures that indication is available to monitor for changes in reactor vessel water level. This device will provide warning of a leak from the Reactor Coolant System or unexplained increases in Reactor Coolant System inventory which could result in a boron dilution event. Two independent monitors are required to provide redundancy and to minimize the necessity to discontinue processing because of instrument failures.

3/4.4.3 SAFETY VALVES

Deleted.

3/4.4.9 PRESSURE/TEMPERATURE LIMIT

The Reactor Coolant System pressure and temperature will be controlled in accordance with approved procedures to prevent a nonductile failure of the Reactor Coolant System while at the same time preventing precipitation of the boron used to maintain subcriticality.

During the RECOVERY MODE, the Reactor Coolant System boundary will be modified periodically to meet recovery objectives. Whenever the Reactor Coolant System is not vented (open to the atmosphere), overpressure protection will be provided to ensure that the allowable RCS pressure is not exceeded.

Reactor coolant chemistry surveillance requirements are included in the RECOVERY OPERATIONS PLAN. These requirements provide assurance that localized corrosion or pitting in crevice areas, which could tend to promote stress corrosion cracking in heat affected zones of welds in stainless steel piping or components, will not occur.

3/4.5 COMMUNICATIONS

The requirements for communications capability ensure that personnel in the Reactor Building can be promptly informed of significant changes in facility status or core reactivity conditions during CORE ALTERATIONS.

3/4.6 CONTAINMENT SYSTEMS

BASES

3/4.6.1 PRIMARY CONTAINMENT

3/4.6.1.1 CONTAINMENT INTEGRITY

Primary CONTAINMENT INTEGRITY is maintained as required per procedure to ensure that radioactive materials in the Containment Building are not released to the environment in an uncontrolled manner.

3/4.6.1.3 CONTAINMENT AIR LOCKS

The containment air locks must be maintained OPERABLE to provide CONTAINMENT INTEGRITY. These air locks will be used during entries into the containment to ensure that radioactive materials are not being released to the environs. The preferred method for ensuring that radioactive materials are not released during these entries is to maintain at least one door closed at all times; however, to permit the passage of long items into the reactor building, both doors may be open simultaneously in accordance with procedures approved pursuant to Specification 6.8.2.

3/4.6.1.4 INTERNAL PRESSURE

The negative pressure limit provides assurance that the Containment will not exceed its design negative pressure differential. The positive pressure limit provides assurance that leakage from the Containment will be limited for off-site dose consideration. During periods when both doors in either or both personnel air locks are open, the containment purge system will be operating to maintain Containment pressure within the limits of the Technical Specification.

3/4.6.1.5 AIR TEMPERATURE

The average air temperature of the Containment atmosphere is currently being maintained between 50°F and 130°F. This condition will maximize the service life of the instrumentation and equipment installed in the Containment and ensure that Reactor Coolant System temperature does not drop below 50°F via LOSS-TO-AMBIENT. This temperature will ensure that boron will remain in solution. Continued OPERABILITY of these items is required to continue monitoring and mitigating the March 28, 1979 incident.

3/4.6.3 CONTAINMENT PURGE EXHAUST SYSTEM

The OPERABILITY requirements for the Containment Purge Exhaust System ensure that during Containment purge operations all radioactive particulate material will be filtered through the HEPA filters prior to release to the atmosphere.

3/4.6.4 COMBUSTIBLE GAS CONTROL

The gas partitioner is provided to analyze the hydrogen concentration in the containment atmosphere so that actions can be initiated to reduce the hydrogen concentration if it approaches its flammable limit. If excessive hydrogen concentrations are detected, appropriate actions will be initiated to reduce the hydrogen concentration to a safe level.

3/4.7.1 FEEDWATER SYSTEM

Deleted

3/4.7.2 SECONDARY SERVICES CLOSED COOLING WATER SYSTEM

Deleted

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 MDHRS since this system provides the heat sink for the MDHRS.

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 Decay Heat Removal System.

3/4.7.3.3 MINI DECAY HEAT REMOVAL SYSTEM (MDHRS)

OPERABILITY of the MDHRS is required since it is a means for boron injection and makeup to the reactor vessel.

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.

3/4.7.6 FLOOD PROTECTION

The limitation on flood protection ensures that facility protective actions will be taken in the event of flood conditions. The limit of elevation 302 Mean Sea Level is based on the maximum elevation at which facility flood control measures provide protection to Safety Related equipment.

3/4.7.7 CONTROL ROOM EMERGENCY AIR CLEANUP SYSTEM

The OPERABILITY of the control room emergency air cleanup system ensures that 1) the ambient air temperature does not exceed the allowable temperature for continuous duty rating for the equipment and instrumentation cooled by this system and 2) the control room will remain habitable for operations personnel during and following all credible accident conditions. The OPERABILITY of this system in conjunction with control room design provisions is based on limiting the radiation exposure to personnel occupying the control room to 5 rem or less whole body, or its equivalent. This limitation is consistent with the requirements of General Design Criterion 19 of Appendix "A", 10 CFR 50.

3/4.7.9 SEALED SOURCE CONTAMINATION

The limitation on removable contamination for sources requiring leak testing, including alpha emitters, is based on 10CFR70.39(c) limits for plutonium. This limitation will ensure that leakage from byproduct, source, and Special Nuclear Material sources will not exceed allowable intake values.

3/4.7.10 FIRE SUPPRESSION SYSTEMS

The OPERABILITY of the Fire Suppression Systems ensures that adequate fire suppression capability is available to confine and extinguish fires occurring in any portion of the facility where Safety Related equipment is located. The Fire Suppression System consists of the water system, spray and/or sprinklers, Halon and fire hose stations. The collective capability of the Fire Suppression Systems is adequate to minimize potential damage to Safety Related equipment and is a major element in the Facility Fire Protection Program. Any two of the four main fire pumps provide combined capacity greater than 3575 gpm.

In the event that portions of the Fire Suppression Systems are inoperable, alternate backup fire fighting equipment is required to be made available in the affected areas until the affected equipment can be restored to service.

In the event that the Fire Suppression Water System becomes inoperable, immediate corrective measures must be taken since this system provides the major fire suppression capability of the plant. The requirement for a Special Report to the Commission provides for timely evaluation of the acceptability of the corrective measures to provide adequate fire suppression capability for the continued operation of the nuclear plant.

3/4.7.11 PENETRATION FIRE BARRIERS

The functional integrity of the Fire Barrier Penetration Seals ensures that fires will be confined or adequately retarded from spreading to adjacent portions of the facility. This design feature minimizes the possibility of a single fire rapidly involving several areas of the facility prior to detection and extinguishment. The Fire Barrier Penetration Seals are a positive element in the facility fire protection program and are subject to periodic inspections.

During periods of time when the seals are not functional, a continuous fire watch is required to be maintained in the vicinity of the affected seal until the seal is restored to functional status.

3/4.8 ELECTRICAL POWER SYSTEMS

The OPERABILITY of the A.C. and D.C. power sources and associated distribution systems during operation ensures that sufficient power will be available to supply the Safety Related equipment required to maintain the unit in stable conditions during recovery from the March 28, 1979 accident.

3/4.9 RADIOACTIVE WASTE STORAGE

BASES

3/4.9.12 FUEL HANDLING BUILDING/AUXILIARY BUILDING AIR CLEANUP SYSTEMS

The requirements for the Fuel Handling/Auxiliary Building Air Cleanup Systems to be operating or OPERABLE ensure that all radioactive material released from the liquid radioactive wastes being stored in the new radwaste storage tanks which have been installed in the spent fuel storage pool or elsewhere in the Auxiliary Building will be filtered through the HEPA filters prior to release to the atmosphere.

3/4.9.13 ACCIDENT GENERATED WATER

These specifications are provided to ensure compliance with the Commission's Statement of May 25, 1979, and the Commission's Statement of Policy and Notice of Intent to Prepare a Programmatic Environmental Impact Statement of November 21, 1979, which prohibit these actions pending evaluation of the environmental impacts of such actions. The PEIS issued in March, 1981, deferred a decision on the ultimate disposal of processed water. Further Commission action is necessary prior to release of ACCIDENT GENERATED WATER.

However, the Commission has recognized that there may be emergency situations, not at this time forseen, which could require rapid action. In these situations, the Commission has indicated its intention to consult with the Council on Environmental Quality to the extent practicable.

ACCIDENT GENERATED WATER, as defined in the settlement of the City of Lancaster litigation, is:

(a) Water that existed in the TMI-2 Auxiliary, Fuel Handling, and Containment Buildings including the primary system as of October 16, 1979, with exception of water wich as a result of decontamination operations becomes commingled with non-accident generated water such that the commingled water has a tritium content of 0.025 μCi/ml or less before processing;

(b) Water that has a total activity of greater than one μCi/ml prior to processing except where such water is originally non-accident water and becomes contaminated by use in cleanup;

(c) Water that contains greater than 0.025 μCi/ml of tritium before processing.

3/4.9.14 Deleted.

3/4.10 DEFUELING OPERATIONS

BASES

3/4.10 CRANE TRAVEL - CONTAINMENT BUILDING

A load drop into the Reactor Vessel may cause reconfigurations of the core debris and/or structural damage which could hinder recovery efforts. A load drop on the Incore Instrument Seal Table and/or guide tubes may result in an unisolable leak from the Reactor Vessel. The restriction on movement of loads in excess of the nominal weight of a fuel or control rod assembly and associated handling tool over these areas is to mitigate the potential consequences stated above in the event this load is dropped.

5.0 DESIGN FEATURES

5.1 SITE

EXCLUSION AREA

5.1.1 The exclusion area is shown on Figure 5.1-1.

LOW POPULATION ZONE

5.1.2 The low population zone is shown on Figure 5.1-2.

SITE BOUNDARY FOR GASEOUS EFFLUENTS

5.1.3 The site boundary for gaseous effluents shall be as shown in Figure 5.1-3.

SITE BOUNDARY FOR LIQUID EFFLUENTS

5.1.4 The site boundary for liquid effluents shall be as shown in Figure 5.1-4.

5.2 CONTAINMENT

CONFIGURATION

5.2.1 The reactor containment building is a steel lined, reinforced concrete building of cylindrical shape, with a dome roof and having the following design features:

a. Nominal inside diameter = 130 feet.

b. Nominal inside height = 157 feet.

c. Minimum thickness of concrete walls = 4 feet.

d. Minimum thickness of concrete roof = 3.5 feet.

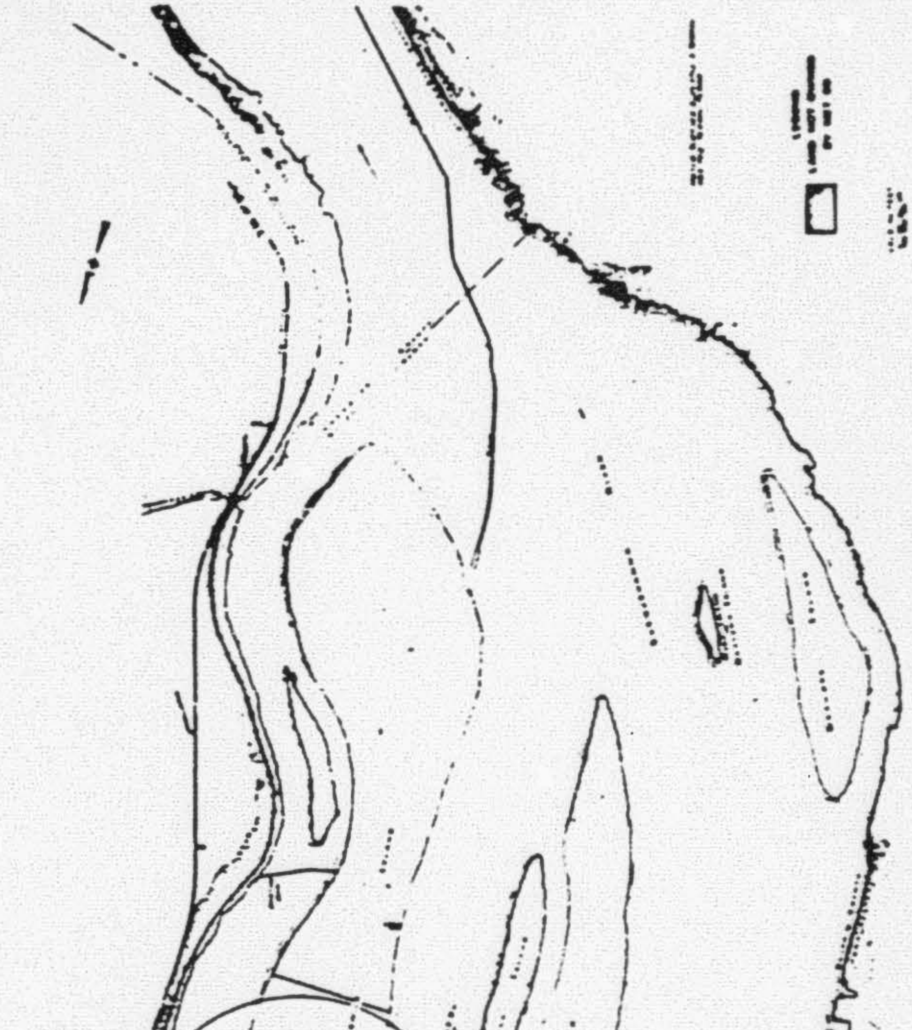
e. Minimum thickness of concrete floor pad = 13.5 feet.

f. Nominal thickness of steel liner = 1/2 inches.

g. Net free volume = 2.1×10^6 cubic feet.

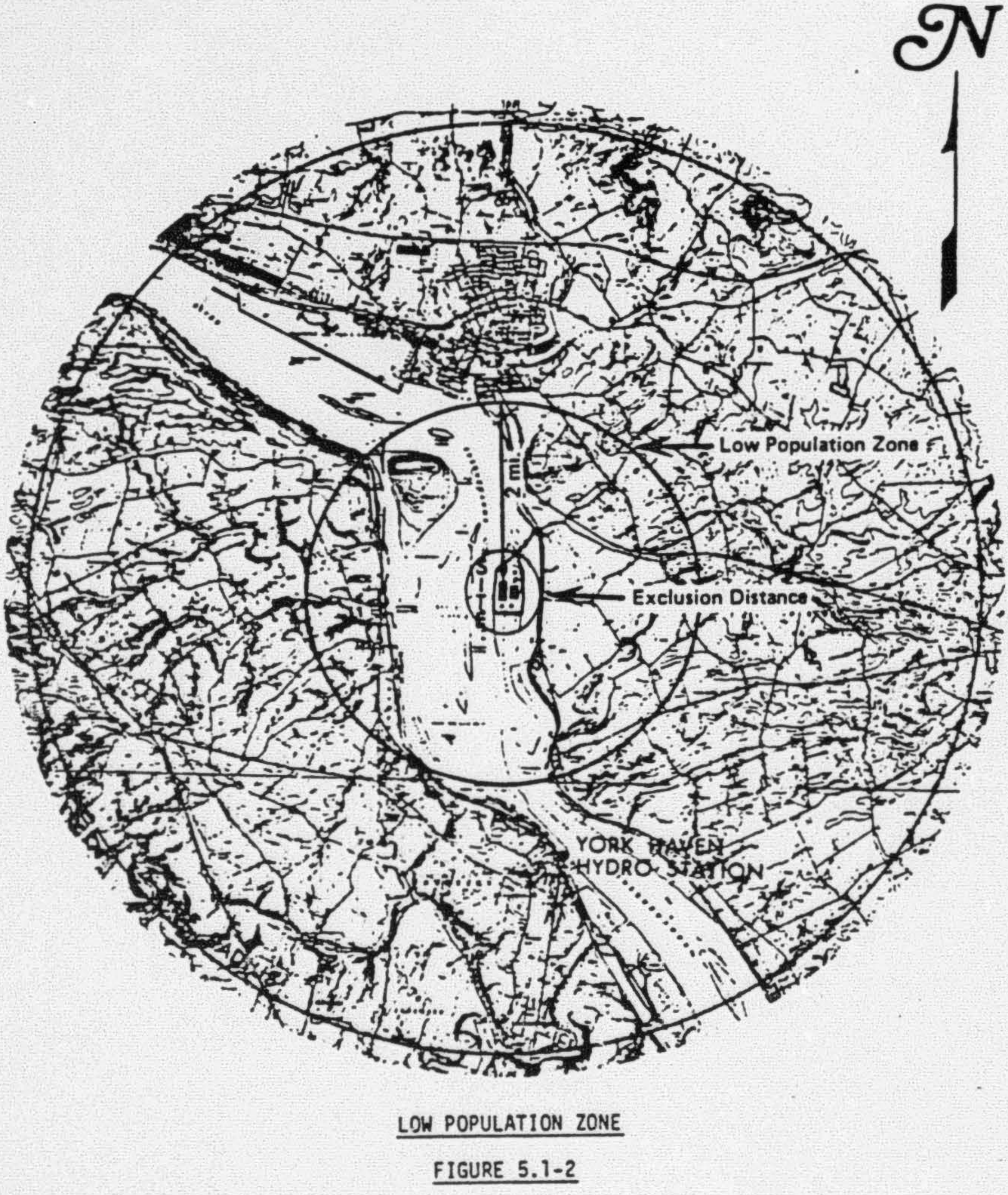
DESIGN PRESSURE AND TEMPERATURE

5.2.2 The reactor containment building is designed and shall be maintained for a maximum internal pressure of 2 psig and a temperature of 286°F.



EXCLUSION AREA

FIGURE 5.1-1



LOW POPULATION ZONE

FIGURE 5.1-2

5.4 REACTOR COOLANT SYSTEM

DESIGN PRESSURE AND TEMPERATURE

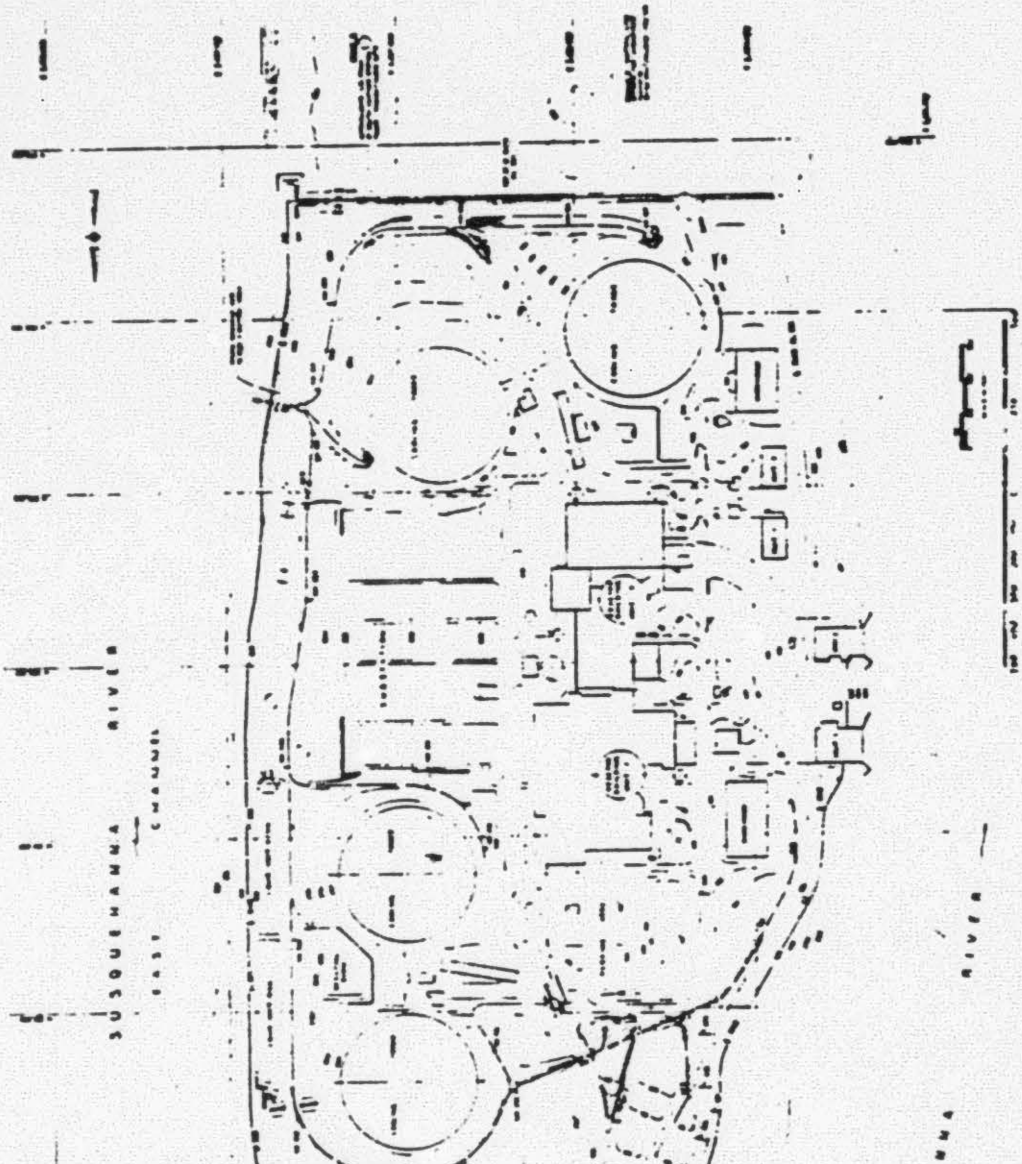
5.4.1 Deleted

VOLUME

5.4.2 The total water and steam volume of the reactor coolant system is 12,180 ± 200 cubic feet at a nominal T_{avg} of 525°F.

5.5 METEOROLOGICAL TOWER LOCATION

5.5.1 The meteorological tower shall be located as shown on Figure 5.5-1.



METEOROLOGICAL TOWER LOCATIONS

Figure 5.5-1

SECTION 6.0

ADMINISTRATIVE CONTROLS

6.0 ADMINSTRATIVE CONTROLS

6.1 RESPONSIBILITY

6.1.1 The Office of the Director-TMI-2 consists of the Director-TMI-2 and the Deputy Director-TMI-2 and shall be responsible for overall unit operation and shall delegate in writing the succession to this responsibility during both of their absences.

6.2 ORGANIZATION

GPUNC ORGANIZATION

6.2.1 The GPU Nuclear Corporation (GPUNC) Organization for unit management and technical support shall be as shown on Figure 1-1 of the Organization Plan. The Organization Plan and changes thereto shall be approved by the NRC prior to implementation.

TMI-2 ORGANIZATION

6.2.2 The unit organization shall be shown on Figure 1-2 of the Organization Plan and:

a. Each on duty shift shall be composed of at least the minimum shift crew composition shown in Table 6.2-1.

b. At least one licensed Operator shall be in the control room when fuel is in the reactor.

c. An individual qualified in radiation protection procedures shall be on site when fuel is in the reactor.

d. A Site Fire Brigade of at least 5 members shall be maintained onsite at all times. The Site Fire Brigade shall not include 3 members of the minimum shift crew necessary for safe shutdown of the unit and any personnel required for other essential functions during a fire emergency.

e. An individual qualified in radiation protection procedures shall be on site whenever Radioactive Waste Management activities are in progress.

Table 6.2-1

MINIMUM SHIFT CREW COMPOSITION#

LICENSE CATEGORY	RECOVERY MODE
SOL	1*
OL	1
Non-Licensed	2

#Shift crew composition may be less than the minimum requirements for a period of time not to exceed 2 hours in order to accommodate unexpected absence of on duty shift crew members provided immediate action is taken to restore the shift crew composition to within the minimum requirements of Table 6.2-1.

*During CORE ALTERATIONS an additional SOL or an SOL limited to fuel handling will be stationed on the operating floor, in the command center, or in the control room as specified in procedures approved pursuant to specification 6.8.2 to directly control the particular CORE ALTERATION activity being performed.

6.3 UNIT STAFF QUALIFICATIONS

6.3.1 Each member of the unit staff shall meet or exceed the minimum qualifications of ANSI-N 18.1 of 1971 for comparable positions unless otherwise noted in the Technical Specifications.

6.3.2 The Radiological Controls Director TMI-2 or his deputy shall meet or exceed the qualifications of Regulatory Guide 1.8, September 1975. Each Radiological Controls Technician in responsible positions/Foreman shall meet or exceed the qualifications of ANSI 18.1-1971, paragraph 4.5.2/4.3.2, or be formally qualified through an NRC approved TMI-2 Radiation Controls training program. Individuals who do not meet ANSI 18.1-1971 Section 4.5.2 are not considered technicians for purposes of determining qualifications but are permitted to perform work for which qualification has been demonstrated. All Radiological Controls Technicians will be qualified through training and examination in each area or specific task related to their radiological controls functions prior to their performance of those tasks.

6.4 TRAINING

6.4.1 A retraining and replacement training program for the unit staff shall be maintained under the direction of the Plant Training Manager-TMI-2 and shall meet or exceed the requirements and recommendations of Regulatory Guide 1.8 of 1977 and Appendix "A" of 10 CFR Part 55 except that Radiological Controls training may be under the direction of Vice President-Radiological and Environmental Controls.

6.4.2 A training program for the Fire Brigade shall be maintained under the direction of the Plant Training Manager-TMI-2 and shall meet or exceed the requirements of Section 27 of the NFPA Code-1976.

6.5 REVIEW AND AUDIT

6.5.1 Technical Review and Control

The Office of TMI-2 Division Director and Support Division Vice-Presidents within GPU Nuclear Coporation as indicated in Organization Plan Figure 1.1, shall be responsible for ensuring the preparation, review, and approval of documents required by the activities within their functional area of responsibility for TMI-2. Implementing approvals shall be performed at the cognizant section manager/director level or above. Independent safety review and audit shall be conducted in accordance with this Technical Specification.

Divisions other than the TMI-2 Division will perform the Independent Safety Review of their own procedures affecting TMI-2 in accordance with approved procedures except when they impact the operational status of unit systems or equipment (requires TMI-2 Division concurrence), a significant environmental impact (requires in line SRG and Environmental Licensing review), or represent an Unreviewed Safety Question (USQ) or Tech Spec change, including Recovery Operations Plan Change (requires in line SRG review and NRC approval).

ACTIVITIES

6.5.1.1 Each procedure required by Technical Specification 6.8 and other procedures including those for test and experiments and changes thereto shall be prepared by a designated individual(s)/group knowledgeable in the area affected by the procedure. Each such procedure, and changes thereto, shall be given a technical review by an individuals(s)/group other than the preparer, but who may be from the same organization as the individual who prepared the procedure or change.

6.5.1.2 Proposed changes to the Technical Specifications shall be reviewed by a knowledgeable individual(s)/group other than the individual(s)/group who prepared the change, and the SRG.

6.5.1.3 Proposed modifications to unit structures, systems and components shall be designed by an individual/organization knowledgeable in the areas affected by the proposed modification. Each such modification shall be technically reviewed by an individual/group other than the individual/group which designed the modification but may be from the same group as the individual who designed the modification.

6.5.1.4 Proposed tests and experiments shall be reviewed by a knowledgeable individual(s)/group other than the preparer but who may be from the same division as the individual who prepared the tests and experiments.

6.5.1.5 The Security Plan and implementing procedures shall be reviewed by a knowledgeable individual(s)/group other than the individual(s)/group which prepared them.

6.5.1.6 The Emergency Plan and implementing procedures shall be reviewed by a knowledgeable individual(s)/group other than the individual(s)/group which prepared them.

6.5.1.7 The Recovery Operations Plan and implementing procedures and changes thereto shall be reviewed by a knowledgeable individual/group other than the individual/group which prepared them.

6.5.1.8 Individuals responsible for reviews performed in accordance with 6.5.1.1 through 6.5.1.7 shall include a determination of whether or not additional cross-disciplinary review is necessary. If deemed necessary, such review shall be performed by the appropriate personnel.

6.5.1.9 Support Division procedures will be reviewed and approved in accordance with their Division level procedures. If the procedure/change impacts the operational status of unit systems or equipment, it must be concurred with by the TMI-2 Division. Unreviewed Safety Question, Technical Specification change (including Recovery Operations Plan change), or Significant Environmental Impact requires line TMI-2 Division and SRG review prior to implementation.

RECORDS

6.5.1.10 Written records of activities performed under specifications 6.5.1.1 through 6.5.1.8 shall be maintained in accordance with 6.10.

QUALIFICATIONS

6.5.1.11 Responsibile Technical Reviewers shall meet or exceed the qualifications of ANSI/ANS-3.1 - 1978 Section 4.4 for Reactor Engineering, Instrumentation and Control, Chemistry and Radiochemistry, Radiation Protection and Quality Assurance Reviewers or have seven (7) years of appropriate experience in the area of their specialty. All other RTRs shall meet Section 4.6, i.e., shall either, (1) have a Bachelor's Degree in Engineering or the physical sciences and three years of professional-level experience in the area being reviewed or, (2) have seven years of appropriate experience in the field of their specialty. An individual performing reviews may possess competence in more than one specialty area. Credit toward experience will be given for advanced degrees on a one-for-one basis up to a maximum of two years.

6.5.2 INDEPENDENT SAFETY REVIEW

FUNCTION

6.5.2.1 The Office of TMI-2 Division Director and Support Division Vice Presidents within GPU Nuclear Corporation as indicated in Organization Plan Figure 1.1 shall be responsible for ensuring the Independent Safety Review of the subjects described in 6.5.2.5 within his assigned area of safety review responsibility.

Divisions other than the TMI-2 Division will perform the Independent Safety Review of their own procedures affecting TMI-2 in accordance with approved procedures except when they impact the operational status of unit systems or equipment (requires TMI-2 Division concurrence), a significant environmental impact (requires in line SRG and Environmental Licensing review), or represent and Unreviewed Safety Question (USQ) or Tech. Spec. change, including Recovery Operations Plan Change (requires in line SRG review and NRC approval).

When the Preparer determines a procedure is not Review Significant, the signature of the RTR indicates concurrence with this determination.

6.5.2.2 Independent safety review shall be completed by an individual/group not having direct responsibility for the performance of the activities under review, but who may be from the same functionally cognizant organization as the individual/group performing the original work. For those TMI-2 Division documents determined to be Review Significant, the Independent Safety Review shall be performed by or under the cognizance of SRG.

6.5.2.3 GPU Nuclear Corporation shall collectively have or have access to the experience and competence required to independently review subjects in the following areas:

a. Nuclear unit operations
b. Nuclear engineering
c. Chemistry and radiochemistry
d. Metallurgy
e. Instrumentation and control
f. Radiological safety
g. Mechanical engineering
h. Electrical engineering
i. Administrative controls and quality assurance practices
j. Emergency plans and related organization, procedures and equipment
k. Other appropriate fields such as radioactive waste operation associated with the unique characteristics of TMI-2.

6.5.2.4 Consultants may be utilized to provide expert advice.

RESPONSIBILITIES

6.5.2.5 The following subjects shall be independently reviewed:

a. Written safety evaluations of changes in the facility as described in the Safety Analysis Report, Technical Evaluation Reports, or docketed System Descriptions, of changes in procedures as described in the Safety Analysis Report, Technical Evaluation Reports, or docketed System Descriptions, and of tests or experiments not described in the Safety Analysis Report, Technical Evaluation Reports, or docketed System Descriptions, which are completed without prior NRC approval under the provisions of 10 CFR 50.59(a)(1). This review of items determined not to be Review Significant when performed by SRG is a supplemental review to verify that such changes, tests or experiments did not involve a change in the Technical Specifications or an Unreviewed Safety Question.

b. Proposed changes in procedures, proposed changes to the facility, or proposed tests or experiments, any of which involves a change in the Technical Specifications or an Unreviewed Safety Question shall be reviewed by SRG prior to implementation. Changes to Review Significant procedures which revision is not deemed to be Review Significant shall not be required to be reviewed by SRG prior to implementation.

c. Proposed changes to Technical Specifications or license amendments shall be reviewed by SRG prior to submittal to the NRC for approval.

d. Violations, deviations, and reportable events which require 24 hour reporting to the NRC in writing. Such reviews are performed after the fact. Review of events covered under this subsection shall include results of any investigations made and the recommendations resulting from such investigations to prevent or reduce the probability of recurrence of the event. SRG shall review all 24 hour reportable events and make recommendations as appropriate.

e. Investigation of all violations of the Technical Specifications including the preparation and forwarding of reports covering evaluation and recommendations to prevent recurrence, shall be reviewed by TMI-2 SRG.

f. Special reviews, investigations or analyses and reports thereon as requested by the Office of the Director TMI2 or other manager reporting directly to the Office of the Director TMI-2 shall be performed by TMI-2 SRG.

g. Written summaries of audit reports in the area specified in section 6.5.3.

h. Recognized indications of an unanticipated deficiency in some aspect of design or operation of structures, systems, or components, that could affect nuclear safety or radioactive waste safety.

i. Any other matters involving safe operation of the nuclear power plant which the SRG deems appropriate for consideration, or which is referred to the SRG.

6.5.2.6 For those subjects which are REVIEW SIGNIFICANT the Independent Safety Review will be performed by an individual(s) meeting the qualifications of Section 6.5.4.7.

RECORDS

6.5.2.7 Reports of reviews encompassed in Section 6.5.2.5 shall be maintained in accordance with 6.10.

6.5.3 Audits

6.5.3.1 Audits of unit activities shall be performed in accordance with the TMI-2 Recovery QA Plan. These audits shall encompass:

a. The conformance of unit operations to provisions contained within the Technical Specifications and applicable license conditions at least once per 12 months.

b. The performance, training and qualifications of the entire unit staff at least once per 12 months.

c. The verification of the nonconformances and corrective actions program as related to actions taken to correct deficiencies occurring in unit equipment, structures, systems or methods of operation that affect nuclear safety at least once per 6 months.

d. The performance of activities required by the Recovery Quality Assurance Plan to meet the criteria of Appendix "B", 10 CFR 50, at least once per 24 months.

e. The Emergency Plan and implementing procedures at least once per 12 months.

f. The Security Plan and implementing procedures at least once per 12 months.

g. The Radiation Protection Plan and implementing procedures at least once per 12 months.

h. The Fire Protection Program and implementing procedures at least once per 24 months.

i. An independent fire protection and loss prevention program inspection and technical audit shall be performed annually utilizing either qualified offsite licensee personnel or an outside fire protection firm.

j. An inspection and technical audit of the fire protection and loss prevention program, by an outside qualified fire consultant at intervals no greater than 3 years.

k. Any other area of unit operation considered appropriate by the SRG, the Manager, SRG's immediate supervisor, other managers reporting directly to the Office of the Director TMI-2, the Office of the Director TMI-2, or the Office of the President - GPUNC. any other areas required to be audited by QA will be identified to the appropriate QA Management level.

RECORDS

6.5.3.2 Audit reports encompassed by sections 6.5.3.1 shall be forwarded for action to the management positions responsible for the areas audited and SRG within 60 days after completion of the audit. SRG will review specified audits performed by QA and make corrective action recommendations as appropriate.

6.5.4 Safety Review Group (SRG)

FUNCTION

6.5.4.1 The SRG shall be a full-time group of engineers, independent of the Site Operations of Engineering staff, and located onsite within the TMI-2 division. (See Organization Plan Figure 1.2.)

ORGANIZATION

6.5.4.2 The TMI-2 SRG shall consist of the Manager, SRG and a minimum staff of 5 engineers.

The SRG shall report within the TMI-2 Division independent of the unit operations and engineering functions, but no lower in the organization than one level below the Office of the Director TMI-2.

RESPONSIBILITY

6.5.4.3 The Manager, SRG is advisory to the Office of the Director TMI-2. However, he has the authority and responsibility to bring to the attention of the Office of the President GPUNC any issues he believes are not being addressed with adequate consideration of nuclear or radiological safety.

6.5.4.4 The review functions of the SRG shall include:

(1) the independent safety review activities stated in Section 6.5.2.5.

(2) assessment of unit operations and performance and unit safety programs from a nuclear safety perspective.

(3) any other matter involving safe nuclear operations at the nuclear power plant that the Manager, SRG, the Manager, SRG's immediate supervisor, or other managers reporting directly to the Office of the Director TMI-2 deem appropriate for consideration.

6.5.4.5 For those reviews requiring expertise outside that possessed by SRG, SRG is authorized to require reviews by other company groups as deemed appropriate by the Manager, SRG. SRG may also utilize consultant expertise as it deems appropriate.

Authority

6.5.4.6 The SRG shall have access to the unit and unit records as necessary to perform its evaluations and assessments. Based on its reviews, the SRG shall provide recommendations to the management positions responsible for the areas reviewed. The SRG shall have authority to require independent reviews by other organizations as necessary to complete its functional responsibilities. The Manager, SRG is advisory to the Office of the Director TMI-2. However, he has the authority and responsibility to bring to the attention of the Office of the President any issues he believes are not being addressed with adequate consideration of nuclear or radiological safety.

QUALIFICATIONS

6.5.4.7 The SRG engineers shall have either; (1) a Bachelor's Degree in Engineering or the Physical Sciences and five (5) years of professional level experience in the nuclear power field including technical supporting functions, or, (2) 9 years of appropriate experience. Credit toward experience will be given for advance degrees on a one-to-one basis up to a maximum of two years. The Manager, SRG, shall meet or exceed the requirements of section 4.7 of ANSI/ANS 3.1-1978.

RECORDS

6.5.4.8 Although day to day results of evaluations by the SRG are communicated directly to the responsible department by the SRG, special reports are prepared only for items deemed appropriate by SRG as concurred with by the Manager, SRG's immediate supervisor. These special reports of evaluations and assessments by SRG shall be prepared, approved, and then transmitted to the Office of the Director, TMI-2 and the management position responsible for the area reviewed through the Manager, SRG's immediate supervisor. These reports shall be maintained for the life of the operating license.

6.6 REPORTABLE OCCURRENCE ACTION

6.6.1 The following actions shall be taken for REPORTABLE OCCURRENCES:

a. The Commission shall be notified and/or a report submitted pursuant to the requirements of Specification 6.9.

b. Each REPORTABLE OCCURRENCE requiring 24 hour notification to the Commission shall be reviewed by the SRG and a report submitted to the Manager, SRG's immediate suprervisor and the Office of the Director TMI-2.

c. Deleted.

6.7 SECTION DELETED

6.8 PROCEDURES

6.8.1 Written procedures shall be established, implemented and maintained covering the activities referenced below:

a. The applicable procedures recommended in Appendix "A" of Regulatory Guide 1.33, Revision 2, February 1978.

b. Recovery Operations Plan implementation.

c. Surveillance and test activities of safety-related equipment and radioactive waste management equipment.

d. Security Plan implementation.

e. Emergency Plan implementation.

f. Radiation Protection Plan implementation.

g. Limiting the amount of overtime worked by plant staff members performing safety-related functions in accordance with the NRC policy statement on working hours as transmitted by Generic Letter 82-12.

6.8.2.1 Each procedure and any change to any procedure prepared pursuant to 6.8.1, shall be prepared, reviewed and approved in accordance with 6.5 and will be reviewed periodically as required by ANSI 18.7 - 1976.

6.8.2.2 Procedures of 6.8.1.a and changes thereto which:

a. Directly relate to core cooling, or

b. Could cause the magnitude of radiological releases to exceed limits established by the NRC, or

c. Could increase the likelihood of failures in systems important to nuclear safety and radioactive waste processing or storage, or

d. Alter the distribution or processing of significant quantities of stored radioactivity or radioactivity being released through known flow paths.

Shall be subject to approval by the NRC prior to implementation.

6.8.3.1 Temporary changes to procedures of 6.8.1 may be made provided if:

a. The intent of the original procedure control is not altered, and

b. (1) For those procedures which affect the operational status of unit systems or equipment, the change is approved by two members of the unit management staff, at least one of whom holds a Senior Reactor Operator's License. If one of the two above signatures is not by a supervisory person within the Department having cognizance of the procedure being changed, the signature will also be required, or

(2) For those procedures wich do not affect the operational status of unit systems or equipment the change is approved by two members of the responsible organization. If one of the two above signatures is not by a section manager/director within the Department having cognizance of the procedure being changed, this signature will also be required, and

c. The change is documented, Independent Safety Review completed, and the required reviews and approvals are obtained within 14 days, and

d. Those changes to procedures described by 6.8.2.2 are submitted to the NRC for review within 72 hours following approval by the management level specified for implementation by Section 6.5.1.9.

6.9 REPORTING REQUIREMENTS

ROUTINE REPORTS AND REPORTABLE OCCURRENCES

6.9.1 In addition to the applicable reporting requirements of Title 10, Code of Federal Regulations, the following reports shall be submitted to the NRC Region Administrator unless otherwise noted.

ANNUAL REPORTS[1/]

6.9.1.4 Annual reports covering the activities of the unit as described below during the previous calendar year shall be submitted prior to March 1 of each year.

6.9.1.5 Reports required on an annual basis shall include:

a. A tabulation of the number of station, utility and other personnel (including contractors) receiving exposures greater than 100 mrem/yr and their associated man rem exposure according to work and job functions,[2/] e.g., reactor operations and surveillance, inservice inspection, routine maintenance, special maintenance (describe maintenance), waste processing, and refueling. The dose assignment to various duty functions may be estimates based on pocket dosimeter, TLD, or film badge measurements. Small exposures totalling less than 20% of the individual total dose need not be accounted for. In the aggregate, at least 80% of the total whole body dose received from external sources shall be assigned to specific major work functions.

b. The following information on aircraft movements at the Harrisburg International Airport:

   1. The total number of aircraft movements (takeoffs and landings) at the Harrisburg International Airport for the previous twelve-month period.

   2. The total number of movements of aircraft larger than 200,000 pounds, based on a current percentage estimate provided by the airport manager or his designee.

RADIATION SAFETY PROGRAM REPORT

6.9.1.6 Deleted.

REPORTABLE OCCURRENCES

6.9.1.7 The REPORTABLE OCCURRENCES of Specifications 6.9.1.8 and 6.9.1.9, including corrective actions and measures to prevent recurrence, shall be reported to the NRC. Supplemental reports may be required to fully describe final resolution of occurrence. In case of corrected or supplemental reports, a licensee event report shall be completed and reference shall be made to the original report date.

[1/] A single submittal may be made for a multiple unit station. The submittal should combine those sections that are common to all units at the station.

[2/] This tabulation supplements the requirements of K20.407 of 10 CFR Part 20.

PROMPT NOTIFICATION WITH WRITTEN FOLLOWUP

6.9.1.8 The types of events listed below shall be reported within 24 hours by telephone and confirmed by telegraph, mailgram, or facsimile transmission to the NRC Region Administrator or his designate no later than the first working day following the event, with a written followup report within 30 days. The written followup report shall include, as a minimum, a completed copy of a licensee event report form. Information provided on the licensee event report form shall be supplemented, as needed, by additional narrative material to provide complete explanation of the circumstances surrounding the event.

a. Deleted.

b. Operation of the unit or affected systems when any parameter or operation subject to a limiting condition for operation is less conservative than the least conservative aspect of the limiting condition for operation established in the Technical Specifications.

c. Abnormal degradation discovered in reactor coolant pressure boundary, or primary containment.

d. An unplanned reactivity insertion of more than 0.5% delta K/K or occurrence of any unplanned criticality.

e. Failure or malfunction of one or more components which prevents or could prevent, by itself, the fulfillment of the functional requirements of system(s) used to cope with accidents analyzed in the SAR, TER, or Safety Evaluation previously submitted to NRC.

f. Personnel error or procedural inadequacy which prevents or could prevent, by itself, the fulfillment of the functional requirements of systems required to cope with accidents analyzed in the SAR, TER, or Safety Evaluation previously submitted to NRC.

g. Conditions arising from natural or man-made events that, as a direct result of the event, require operation of safety systems or other protective measures required by Technical Specifications.

h. Performance of structures, systems, or components that requires remedial action or corrective measures to prevent operation in a manner less conservative than assumed in the accident analyses in the safety analysis report or Technical Specifications bases; TER, or Safety Evaluation previously submitted to NRC or discovery during unit life of conditions not specifically considered in the safety analysis report or Technical Specifications that require remedial action or corrective measures to prevent the existence or development of an unsafe condition.

THIRTY DAY WRITTEN REPORTS

6.9.1.9 The types of events listed below shall be the subject of written reports to the NRC Region Administrator within thirty days of occurrence of the event. The written report shall include, as a minimum, a completed copy of a licensee event report form. Information provided on the licensee event report form shall be supplemented, as needed, by additional narrative material to provide complete explanation of the circumstances surrounding the event.

a. Reactor protection system or engineered safety feature instrument settings which are found to be less conservative than those established by the Technical Specifications but which do not prevent the fulfillment of the functional requirements of affected systems.

b. Conditions leading to operation in a degraded mode permitted by a limiting condition for operation.

c. Observed inadequacies in the implementation of administrative or procedural controls which threaten to cause reduction of degree of redundancy provided in engineered safety feature systems or radioactive waste treatment systems.

d. Abnormal degradation of systems other than those specified in 6.9.1.8.c above designed to contain radioactive material resulting from the fission process.

e. All events which require activation of the Emergency Plan.

REPORTING REQUIREMENTS FOR INCIDENT WHICH OCCURRED ON MARCH 28, 1979

6.9.1.10 Section Deleted. All reporting requirements completed.

SPECIAL REPORTS

6.9.2 Special reports shall be submitted to the NRC Region Administrator within the time period specified for each report.

6.10 RECORD RETENTION

6.10.1 The following records shall be retained for at least five years:

a. Records of sealed source and fission detector leak tests and results.

b. Records of annual physical inventory of all sealed source material of record.

c. Records of changes made to the procedures required by Specifications 6.8.1.d and e.

6.10.2 The following records shall be retained as long as the Licensee has an NRC license to operate or possess the Three Mile Island facility.

a. Records and logs of unit operation covering time interval at each power level.

b. Records and logs of principal maintenance activities, inspections, repair and replacement of principal items of equipment related to nuclear safety and radioactive waste systems.

c. ALL REPORTABLE OCCURRENCES submitted to the Commission.

d. Records of surveillance activities, inspections and calibrations required by these Technical Specifications.

e. Records of changes made to the procedures required by Specifications 6.8.1.a, b., c., and f.

f. Radiation Safety Program Reports and Quarterly Recovery Progress Reports on the March 28, 1979 incident.

g. Records of radioactive shipments.

h. Records and logs of radioactive waste systems operations.

i. Records and drawing changes reflecting facility design modifications made to systems and equipment described in the Safety Analysis Report, TER, SD, or Safety Evaluation previously submitted to NRC.

j. Records of new and irradiated fuel inventory, fuel transfers and assembly burnup histories.

k. Records of transient or operational cycles for those unit components designed for a limited number of transients or cycles.

l. Records of reactor tests and experiments.

m. Records of training and qualification for current members of the unit staff.

n. Records of in-service inspections performed pursuant to these Technical Specifications.

o. Records of Quality Assurance activities required by the Operating Quality Assurance Plan.

p. Records of reviews performed for changes made to procedures or equipment or reviews of tests and experiments pursuant to 10 CFR 50.59.

q. Records of meetings of the Plant Operation Review Committee (PORC) and the General Review Committee (GRC) and reports of evaluations prepared by the SRG.

r. Records of the incident which occurred on March 28, 1979.

s. Records of unit radiation and contamination surveys.

t. Records of radiation exposure for all individuals entering radiation control areas.

u. Records of gaseous and liquid radioactive material released to the environs.

6.11 RADIATION PROTECTION PROGRAM

Personnel radiation protection shall be consistent with the requirements of 10 CFR Part 20 and the NRC approved Radiation Protection Plan.

6.12 HIGH RADIATION AREA

In lieu of the "control device" or "alarm signal" required by paragraph 20.203(c)(2) of 10 CFR 20, each high radiation area shall be controlled as specified in the Radiation Protection Plan.

ENCLOSURE 4

RECOVERY OPERATIONS PLAN

TMI-2

SURVEILLANCE REQUIREMENTS

4.1 WATER INJECTION COOLING AND REACTIVITY CONTROL SYSTEMS

4.1.1 BORATION CONTROL

BORATED COOLING WATER INJECTION

4.1.1.1 Two systems capable of injecting borated cooling water into the Reactor Coolant System shall be demonstrated OPERABLE:

A. At least once per 12 hours (when system is in operation) by verifying that the Standby Reactor Coolant System Pressure Control System:

   1. Surge tank water volume is filled to between 55% and 80% of tank capacity and the tank is pressurized to the operating RCS pressure ± 25 psig but not higher than 100 psig. The above limits may be varied in accordance with procedures approved pursuant to Specification 6.8.2.

   2. Isolation valves on the discharge side of the water filled tank nearest the Reactor Coolant System are open, except when closed in accordance with procedures approved pursuant to Technical Specification 6.8.2. Valves closed by procedure shall be verified open within 12 hours after the completion of the procedure.

   3. The in-service nitrogen supply bank is pressurized between 50 and 400 psig.

B. At least once per 7 days by verifying that the Standby Reactor Coolant System Pressure Control System surge tanks, and the charging water storage tank contain borated water with:

   1. A boron concentration of between 3500 and 6000 ppm.

   2. Deleted.

C. At least once per 31 days by verifying that the Standby Reactor Coolant System Pressure Control System Isolation Valve on the discharge side of the water filled tank nearest the Reactor Coolant System closes automatically on a tank low level test signal.

D. By demonstrating that the Mini Decay Heat Removal System (MDHRS) is OPERABLE by performing inservice tests of each MDHRS pump and each MDHRS valve in the flow path in accordance with SECTION XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR 50, Section 50.55a(g), except where specific written relief has been granted by the Commission pursuant to 10 CFR 50, Section 50.55a(g)(6)(i).

E. At least once per 31 days by verifying that on recirculation flow, the decay heat removal pump required by Technical Specification 3.1.1.1 develops a discharge pressure of greater than or equal to 151 psig and that each pump operates for at least 15 minutes.

SURVEILLANCE REQUIREMENTS

BORON INJECTION (Continued)

F. At least once per 31 days by verifying that each accessible (per occupational exposure considerations) valve (manual, power operated or automatic) in each flow path that is not locked, sealed, or otherwise secured in position, is in its correct position.

G. At least once per 7 days by:

1. Verifying the boron concentration in the BWST is between 3500 and 6000 ppm.

2. Verifying the contained borated water volume of the BWST is at least 100,000 gallons.

H. At least once per 24 hours by verifying the BWST temperature is at least 50°F when the outside air temperature is less than 50°F.

SURVEILLANCE REQUIREMENTS

BORON CONCENTRATION

4.1.1.2 The boron concentration of all filled portions of the Reactor Coolant System and the Refueling Canal shall be determined to be within the specified limits by:

a. Determining the boron concentration of the coolant in the filled portions to be between 3500 and 6000 ppm by:

   1. A mass balance calculation at least once per 24 hours.

   2. A chemical analysis at least once per 7 days.

b. Verifying the temperature of the coolant in the filled portions of the reactor coolant system to be greater than 50°F at least once per 12 hours.

c. Verifying the temperature of the coolant in the filled portions of the refueling canal to be greater than 50°F at least once every 7 days.

4.1.3 CONTROL ASSEMBLIES

4.1.3.1 With the Reactor Vessel Head and the Control Rod Drive Mechanisms removed or disconnected from the reactor vessel there is no surveillance required.

4.3 INSTRUMENTATION

4.3.1 NEUTRON MONITORING INSTRUMENTATION

4.3.1.1 Each intermediate and source range neutron monitoring instrumentation channel shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL CALIBRATION and CHANNEL FUNCTIONAL TEST operations during RECOVERY MODE and at the frequency shown in Table 4.3-1.

4.3.2 ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

4.3.2.1 Each ESFAS instrumentation channel shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL CALIBRATION and CHANNEL FUNCTIONAL TEST operations during RECOVERY MODE and at the frequencies shown in Table 4.3-2.

4.3.3 MONITORING INSTRUMENTATION

RADIATION MONITORING INSTRUMENTATION

4.3.3.1 Each radiation monitoring instrumentation channel shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL CALIBRATION AND CHANNEL FUNCTIONAL TEST operations at the frequencies shown in Table 4.3-3.

4.3.3.2 Deleted.

4.3.3.3 Deleted.

SEISMIC INSTRUMENTATION

4.3.3.3.1 Deleted.

4.3.3.3.2 Deleted.

TABLE 4.3-1

NEUTRON MONITORING INSTRUMENTATION

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE
1. Intermediate Range, Neutron Flux and Rate	S	R[1]	M	1	0	1
2. Source Range, Neutron Flux and Rate	S	R[1]	M	2	0	2

NOTES

(1) - Neutron detectors and all channel components located inside containment may be excluded from CHANNEL CALIBRATION.

TABLE 4.3-2

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	ACTION	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST
1. LOSS OF POWER							
a. 4.16 kv Emergency Bus Undervoltage (Loss of Voltage)							
1. Emergency Bus #2-1E and 2-2E	2/Bus	2/Bus	2/Bus	1	S	R	M
2. Emergency Bus #2-3E and 2-4E	2/Bus	1/Bus	2/Bus	2	S	NA	NA

ACTION 1 - With the number of OPERABLE channels one less than the Total Number of Channels, place the inoperable channel in the tripped condition within 4 hours.

ACTION 2 - None except as provided in Specification 3.0.3.

ENGINEERED SAFETY FEATURE ACTUATION SYSTEMS INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
1. LOSS OF POWER		
a. 4.16 kv Emergency Bus Undervoltage (Loss of Voltage)		
1. Emergency Bus #2-1E and 2-2E	2278 ± 69 volts with a 1.5 ± .15 second time delay	2278 ± 114 volts with a 1.5 ± 0.2 second time delay
2. Emergency Bus #2-3E and 2-4E	2278 ± 69 volts with a 1.5 ± .15 second time delay	2278 ± 114 volts with a 1.5 ± 0.2 second time delay

TABLE 4.3-3

RADIATION MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MINIMUM CHANNELS OPERABLE	APPLICABILITY	ACTION
1. CONTAINMENT						
a. Reactor Building Purge Sampler (AMS-3)	D	SA	W	1	Note 1	Note 3
b. Reactor Building Equipment Doors (AMS-3)	D	SA	W	1	Note 1	Note 7
c. Reactor Vessel Flange Area Gamma Monitor	D	SA	W	1	Note 1	Note 8
2. FUEL HANDLING BUILDING Exhaust Monitors (HPR-221A or HPR-221B)						
a. Gaseous Activity	S	R	M	1	Note 5	Note 6
b. Particulate Activity	S	R	M	1	Note 5	Note 6
3. SDS MONITORS						
a. Process Monitor (IX04)	S	R	M	1	Note 1	Note 4
b. Area Monitor (IX03)	S	SA	M	1	Note 1	Note 2
4. MDHR MONITORS						
a. Area Monitors	S	R	M	3	Note 1	Note 3
b. Ping Monitors	S	R	M	1	Note 1	Note 3

(See following page for notes)

TABLE 4.3-3 (Con't)

RADIATION MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

NOTES:

1) During operation of the monitored system.
2) If monitor becomes inoperable, repair or replace by equivalent equipment within 12 hours. If not completed within 12 hours terminate operation of the monitored system and restore the inoperable monitor(s) to operable status.
3) Restore the inoperable monitor(s) to operable status within 72 hours.
4) If ion exchange effluent monitor is inoperable, sample on 4 hour frequency for gross beta. If inoperable longer than 2 hours then terminate operation of ion exchange system and restore the inoperable monitor to operable status.
5) With radioactive waste in the fuel handling building.
6) With the required instrumentation inoperable, suspend all operations involving movement of radioactive wastes in the fuel handling building, restore the inoperable equipment to OPERABLE status within 48 hours.
7) With the AMS-3 inoperable, close at least one of the Reactor Building Equipment Doors and restore the inoperable equipment to operable status prior to the reopening of both Equipment Doors.
8) With the Reactor Vessel Flange Area Gamma Monitor inoperable suspend all operations involving CORE ALTERATIONS and restore instrument to operable status.

TABLE 4.3-4

SEISMIC MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENT

Deleted

METEOROLOGICAL INSTRUMENTATION

4.3.3.4 Each of the meteorological monitoring instrumentation channels shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK and CHANNEL CALIBRATION operations at the frequencies shown in Table 4.3-5.

ESSENTIAL PARAMETERS MONITORING INSTRUMENTATION

4.3.3.5 Each of the Essential Parameters Monitoring Instrumentation channels shall be demonstrated OPERABLE by performance of the CHANNEL CHECK and CHANNEL CALIBRATION operations at the frequencies shown in Table 4.3-7.

POST-ACCIDENT INSTRUMENTATION

4.3.3.6 Deleted

CHLORINE DETECTION SYSTEMS

4.3.3.7 Each chlorine detection system shall be demonstrated OPERABLE by performance of a CHANNEL CHECK at least once per 12 hours, and a CHANNEL FUNCTIONAL TEST at least once per 31 days. At least once per 18 months, the following inspections and maintenance shall be performed:

a. Check constant head bottle level and refill as necessary,

b. Clean the sensing cells,

c. Check flow meter operation and clean or replace filters and air lines as necessary,

d. Check air pump for proper operation, and

e. Verify that the detector responds to HCL.

TABLE 4.3-5

METEOROLOGICAL MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	MINIMUM OPERABLE	CHANNEL CHECK	CHANNEL CALIBRATION
1. WIND SPEED			
a. Nominal Elev. 100 ft	1	D	SA
2. WIND DIRECTION			
a. Nominal Elev. 100 ft.	1	D	SA
3. AIR TEMPERATURE - DELTA T			
a. Nominal Elev. 33 ft.	1	D	SA
b. Nominal Elev. 150 ft.	1	D	SA

TABLE 4.3-6

REMOTE SHUTDOWN MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

DELETED

TABLE 4.3-7

ESSENTIAL PARAMETERS MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION (1)	READOUT LOCATION(S)	MINIMUM OPERABLE CHANNELS
1. Reactor Building Pressure	S	R	Control Room	2
2. Reactor Vessel Water Level	S/W (2)	SA	Control Room (2)	2 (2)
3. Reactor Coolant System Temperature	S	R	CAB 217 & Control Room	1/Loop
4. Incore Thermocouples	S	R	Control Room or Cable Room	All Available (3)
5. NI Intermediate Range Level Log N	M	R	Cab 217 & Control Room	1
6. NI Source Range Level	M	R	Cab 217 (4) & Control Room	2
7. Reactor Building Water Level	NA	SA	Control Bldg. Area West	1
8. Borated Water Storage Tank Level	S	R	Control Room	1
9. Steam Generator Level	NA	NA	NA	1/Generator
10. Decay Heat Removal Flow	M	R	Cab 217	1/Loop

Notes:

1) Nuclear detectors and all channel components located inside containment may be excluded from CHANNEL CALIBRATION.

2) One channel may consist of a visual indication such as a level stand pipe.
Seven day surveillance applys to standpipe only. Level standpipe readout may be in the reactor building or by remote television in the control room.

3) Report all failures pursuant to specification 6.9.1.8

4) Only one readout required at Cab 217.

Table 4.3-10

POST-ACCIDENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

Deleted

FIRE DETECTION

4.3.3.8.1 Each of the required accessible (per occupational exposure considerations) fire detection instruments shall be demonstrated OPERABLE at least once per 6 months by performance of a CHANNEL FUNCTIONAL TEST.

4.3.3.8.2 The NFPA Code 71 supervised circuits supervision associated with the detector alarms of each of the required fire detection instruments shall be demonstrated (per occupational exposure considerations) OPERABLE at least once per 6 months.

4.3.3.8.3 The nonsupervised circuits between the local panels in Surveillance Requirements 4.3.3.8.2 and the control room shall be demonstrated OPERABLE at least once per 31 days.

4.3.3.8.4 In lieu of Specification 4.3.3.8.2, fire detection instrument for the Southeast Storage Facility shall have circuitry per procedures approved pursuant to Specification 6.8.2.

TABLE 4.3-11

FIRE DETECTION INSTRUMENTS

Instrument Location	Required Instruments Operable Heat	Required Instruments Operable Flame	Required Instruments Operable Smoke	Alternate Instrument Heat	Alternate Instrument Flame	Alternate Instrument Smoke
1. Auxiliary Building	NA	NA	2	NA	NA	2
2. Control Building						
351' Elevation	NA	NA	1	NA	NA	3
331' Elevation (Control Room)	NA	NA	1	NA	NA	1
305', 293', 280' Elevations	NA	NA	1	NA	NA	7
3. Control Building Area	NA	NA	1	NA	NA	3
4. Diesel Generator Building						
A Diesel	NA	1	1	NA	NA	1
B Diesel	NA	1	1	NA	NA	1
5. Fuel Handling Building	NA	NA	3	NA	NA	1
6. Reactor Building	NA	NA	1	NA	NA	1
RC-P-1A/2A	NA	NA	1	NA	NA	2
RC-P-1B/2B	NA	NA	1	NA	NA	2
7. River Water Pump House	NA	1	1	NA	NA	2
8. Service Building	NA	NA	3	NA	NA	3
9. Deleted						
10. Southeast Areas Storage Facility	NA	NA	2	NA	NA	2

4.4 REACTOR COOLANT SYSTEM

REACTOR COOLANT LOOPS

4.4.1 Verify that surveillance of the Reactor Coolant System is being performed in accordance with procedures approved pursuant to Technical Specification 6.8.2.

4.4.2 REACTOR VESSEL WATER LEVEL MONITORING

4.4.2 The Reactor Vessel Water Level Monitoring Instrumentation shall be demonstrated OPERABLE as required by Table 4.3-7.

SAFETY VALVES

4.4.3 Deleted.

4.4.9 PRESSURE/TEMPERATURE LIMITS

REACTOR COOLANT SYSTEM

4.4.9.1.1 The Reactor Coolant System temperature shall be determined to be within the limits at least once per 12 hours.

4.4.9.1.2 Deleted.

4.4.9.1.3 Deleted.

4.4.9.1.4 The pH of the reactor coolant shall be determined to be greater than or equal to 7.5 and less than 8.4 at least once per 7 days.

4.4.9.1.5 The chloride concentration in the reactor coolant shall be determined to be less than or equal to 5 ppm at least once per 7 days.

4.5 COMMUNICATIONS

4.5.1 Verify that communications channels are OPERABLE between the Control Room and Command Center and personnel in the Reactor Building once each day prior to the initiation of any activities involving CORE ALTERATIONS. Also verify that the additional SOL or SOL limited to fuel handling has OPERABLE communication channels once each day prior to the initiation of any activities involving core alterations.

SURVEILLANCE REQUIREMENTS

4.6 CONTAINMENT SYSTEMS

4.6.1 PRIMARY CONTAINMENT

CONTAINMENT INTEGRITY

4.6.1.1 If required per procedures approved pursuant to Specification 6.8.2, primary CONTAINMENT INTEGRITY shall be demonstrated:

a. At least once per 31 days by verifying per NRC approved procedures that:

   1. All penetrations not required to be open per approved procedures are closed by valves, blind flanges, or deactivated automatic valves secured in their positions.

b. By verifying at least once per 31 days that the Containment Equipment Hatch is closed and sealed.

c. By verifying that each Containment Air Lock is OPERABLE per Specification 3.6.1.3.

CONTAINMENT AIR LOCKS

4.6.1.3 Each containment air lock shall be demonstrated OPERABLE:

a. After each opening, except when the air lock is being used for multiple entries, then at least once per 72 hours, by verifying less than or equal to 0.01 L_a seal leakage when the volume between the door seals is stabilized to a pressure to 10 psig.

b. At least once per three months by performing a mechanical operability check of each airlock, including a visual inspection of the components and lubrication if necessary.

c. Deleted

4.6.1.3.1 When both equipment hatch personnel airlock doors are opened simultaneously, verify the following conditions:

   a. The capability exists to expenditiously close at least one airlock door.

   b. The airlock doors and containment purge are configured to restrict the outflow of air in accordance with procedures approved pursuant to Tech Spec 6.8.2.

   c. The airlock doors are cycled to ensure mechanical operability within seven days prior to opening both doors.

SURVEILLANCE REQUIREMENTS

INTERNAL PRESSURE

4.6.1.4 The primary containment internal pressure shall be determined to be within the limits at least once per 12 hours.

AIR TEMPERATURE

4.6.1.5 The primary containment average air temperature shall be the arithmetical average of the temperatures at the following locations and shall be determined at least once per 24 hours:

Location

a. RB nominal Elev. 350' (1 temperature indication)
b. RB nominal Elev. 330' (1 temperature indication)
c. RB nominal Elev. 305' (1 temperature indication)

CONTAINMENT PURGE EXHAUST SYSTEM

4.6.3 The Containment Purge Exhaust System shall be demonstrated OPERABLE:

a. At least once per 31 days during operation by verifying that the Purge Exhaust System in the normal operating mode meets the following conditions:

   1. Filter Pressure Drop: The d/p across the combined HEPA filters shall not exceed 6 inches water gauge while the system is operating.

b. At least once per 18 months by verifying that the ventilation system meets the following conditions:

   1. Visually inspect each filter train and associated components in accordance with Section 5 of ANSI N510-1980, as required by Regulatory Position C.5.a of Regulatory Guide 1.52, Revision 2, March 1978. The inspection should be performed prior to the DOP test of this section.

   2. Flow Test: Exhaust flow rate shall be within 18,000 cfm to 27,000 cfm operating band for each filter train with one filter train and one exhaust fan operating. Testing shall be in accordance with ANSI N510-1980, Section 8.3.1, paragraphs 3 and 4.

   3. DOP Test: Each filter train shall be tested in accordance with Section 10 of ANSI N510-1980, as required by Regulatory Position C.5.c of Regulatory Guide 1.52, Revision 2, March 1978.

NOTE: Installed system flow instrumentation is adequate for the test described in Section 4.6.3.b.3 above.

SURVEILLANCE REQUIREMENTS

c. After structural maintenance of the HEPA filter or charcoal adsorber housings, or following fire or chemical release in any ventilation zone communicating with the system by verifying that the ventilation system meets the following conditions:

   1. Filter Pressure Drop: Reverify the filter pressure drop surveillance prescribed in Section 4.6.3.a.1 for the affected filter train(s).

   2. DOP Test: Each affected filter train shall be retested in accordance with Section 4.6.3.b.3.

   3. Visual inspection in accordance with ANSI N510-1980 Section 5.

d. After each complete or partial replacement of a HEPA filter bank by verifying that the ventilation system meets the following condition:

   1. DOP Test: Each affected filter train shall be retested in accordance with Section 4.6.3.b.3.

4.6.4 COMBUSTIBLE GAS CONTROL

HYDROGEN ANALYZERS

4.6.4.1 The gas partitioner shall be demonstrated OPERABLE at least once per 31 days by performing a CHANNEL CALIBRATION using sample gases containing:

a. One volume percent hydrogen, balance air.

b. Four volume percent hydrogen, balance air.

HYDROGEN PURGE CLEANUP SYSTEM

4.6.4.2 Deleted

4.7 PLANT SYSTEMS

4.7.1 FEEDWATER SYSTEM

Deleted.

4.7.2 SECONDARY SERVICES CLOSED COOLING WATER SYSTEM

Deleted.

4.7.3 CLOSED CYCLE COOLING WATER SYSTEM

NUCLEAR SERVICES CLOSED CYCLE COOLING SYSTEM

4.7.3.1 Each Nuclear Services Closed Cycle Cooling Water Loop shall be demonstrated OPERABLE:

a. At least once per 31 days by:

   1. Verifying that each pump starts and operates (unless already operating) for at least 15 minutes and that during pump operation: the "A" pump develops a differential pressure of at least 62.1 psid, the "B" pump develops a differential pressure of at least 63.1 psid, and the "C" pump develops a differential pressure of at least 64.1 psid.

   2. Verifying that each valve (manual, power operated or automatic) servicing Safety related equipment that is not locked, sealed or otherwise secured in position, is in its correct position.

b. At least once per 92 days by cycling each testable valve in the flow path through at least one complete cycle of full travel.

DECAY HEAT CLOSED COOLING WATER SYSTEM

4.7.3.2 The Decay Heat Closed Cooling Water Loop required by Technical Specification 3.7.3.2 shall be demonstrated OPERABLE:

a. At least once per 31 days by:

   1. Verifying the required pump starts and operates (unless already operating) for at least 15 minutes and that during pump operation: (a) the "A" pump develops a differential pressure of at least 30.6 psid at a flow of 2491 gpm and (b) the "B" pump develops a differential pressure of at least 30.6 psid at a flow of 2527 gpm.

   2. Verifying that each accessible (per occupational exposure considerations) valve (manual, power operated or automatic) servicing safety related equipment that is not locked, sealed or otherwise secured in position, is in its correct position.

b. At least once per 92 days by cycling each testable valve in the flow path through at least one complete cycle of full travel.

MINI DECAY HEAT REMOVAL SYSTEM (MDHRS)

4.7.3.3 Surveillance requirements transferred to Section 4.1.1.1

4.7.4 NUCLEAR SERVICE RIVER WATER SYSTEM

4.7.4.1 Each Nuclear Service River Water Loop shall be demonstrated OPERABLE:

a. At least once per 31 days by:

1. Verifying that each pump starts and operates (unless already operating) for at least 15 minutes and, that during pump operation each pump develops a differential pressure of at least 23.8 psid.

2. Verifying that each valve (manual, power operated or automatic) servicing safety related equipment that is not locked, sealed or otherwise secured in position, is in its correct position.

b. At least once per 92 days by cycling each testable valve in the flow path through at least one complete cycle of full travel.

4.7.6 FLOOD PROTECTION

4.7.6.1.1 The water level at the Unit 1 intake structure shall be determined to be within the limit by:

a. Measurement at least once per 24 hours when the water level is below elevation 301 feet Mean Sea Level USGS datum.

b. Measurement at least once per 2 hours when the water level is equal to or above elevation 301 feet Mean Sea Level datum.

4.7.6.1.2 The dike surrounding the island and the dike errosion protection shall be:

a. Inspected at least once per 6 months and within 31 days after every flood which exceeds 485,000 cfs at the Harrisburg gage. This inspection shall include:

1. Dike slope and notation of any location where slope is steeper than 2:1.

2. Depressions and notation of their size and location.

3. Stone degradation and notation of any areas where splitting, spalling or weathering of stone appears excessive.

4. Vegetation and notation of location, quantity, and type.

5. Operability of inside and outside gates of the Southeast Drainage Culvert.

4.7.6.1.3 A Special Report shall be prepared and submitted to the Commission within 10 days if evidence of degradation is noted during an inspection. This report shall describe the extent and nature of the degradation and the plans and schedule for restoring the dike and errosion protection to a status equivalent to the original design provisions.

4.7.7 CONTROL ROOM EMERGENCY AIR CLEANUP SYSTEM

4.7.7.1 The Control Room Emergency Air Cleanup System shall be demonstrated OPERABLE:

a. At least once per 12 hours by verifying that the control room air temperature is less than or equal to 100°F.

b. At least once per 31 days by initiating, from the control room, flow through the HEPA filters and charcoal adsorbers and verifying that the system operates for at least 15 minutes; and the pressure drop across the combined HEPA filters and charcoal adsorbers banks is less than six (6) inches water gauge while operating.

c. At least once per 18 months or (1) after any structural maintenance on the HEPA filter or charcoal adsorber housings, or (2) following painting, fire or chemical release in any ventilation zone communicating with the system by:

   1. Verifying that the cleanup system satisfies the in-place testing acceptance criteria and uses the test procedures of Regulatory Positions C.5.a, C.5.c* and C.5.d* of Regulatory Guide 1.52, Revision 2, March 1978, and the system flow rate is 14,350 cfm ± 10%.

   2. Verifying within 31 days after removal that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Guide 1.52, Revision 2, March 1978, when performing Methyl Iodide, 30°C, 95% RH testing per Table 5-1 of ANSI N509-1980 meets an acceptable criteria of 5% penetration maximum.

   3. Verifying a system flow rate of 14,350 cfm ± 10% during system operation when tested in accordance with ANSI N510-1980, Section 8.3.1 Paragraphs 3 and 4.

d. After every 720 hours of charcoal adsorber operation by verifying within 31 days after removal that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, when performing Methyl Iodide, 30°C, 95% RH testing per Table 5-1 of ANSI N509-1980 meets an acceptance criteria of 5% penetration maximum.

---

*The prerequisites of Section 10.3 and 12.3 of ANSI-N510-1980 do not apply.

CONTROL ROOM EMERGENCY AIR CLEANUP SYSTEM (Continued)

e. At least once per 18 months by:

1. Verifying that the pressure drop across the combined HEPA filter and charcoal adsorber banks is less than 6 inches water gauge while operating the system at a flow rate of 14,350 cfm ± 10%.

2. Verifying that on a control room air inlet radiation test signal or chlorine detection test signal, the system automatically switches into a recirculation mode of operation with flow through the HEPA filters and charcoal adsorber banks.

3. Verifying that the system maintains the control room at a positive pressure of greater than or equal to 1/10 inch water gauge relative to the outside atmosphere during system operation.

f. After each complete or partial replacement of a HEPA filter bank by verifying that the HEPA filter banks remove greater than or equal to 99.95% of the DOP when they are tested in-place in accordance with ANSI N510-1980* Section 10 while operating the system at a flow rate of 14,350 cfm ± 10%.

g. After each complete or partial replacement of a charcoal adsorber bank by verifying that the charcoal adsorbers remove greater than or equal to 99.95% of a halogenated hydrocarbon refrigerant test gas when they are tested in-place in accordance with ANSI N510-1980,* Section 12, while operating the system at a flow rate of 14,350 cfm ± 10%.

*The prerequisites of Section 10.3 and 12.3 of ANSI-510-1980 do no apply.

4.7.9 SEALED SOURCES

4.7.9.1 Test Requirements - Each sealed source shall be tested for leakage and/or contamination by:

a. The licensee, or

b. Other persons specifically authorized by the Commission or an Agreement State.

The test method shall have a detection sensitivity of at least 0.005 microcuries per test sample.

4.7.9.2 Test Frequencies - Each category of sealed sources shall be tested at the frequency described below.

a. Source in use (excluding startup sources and fission detectors previously subjected to core flux) - At least once per six months for all sealed sources containing radioactive material:

   1. With a half-life greater than 30 days (excluding Hydrogen 3) and

   2. In any form other than gas.

b. Stored sources not in use - Each sealed source and fission detector shall be tested prior to use or transfer to another licensee unless tested within the previous six months. Sealed sources and fission detectors transferred without a certificate indicating the last test date shall be tested prior to being placed into use.

c. Startup sources and fission detectors - Each sealed startup source and fission detector shall be tested within 31 days prior to being subjected to core flux or installed in the core and following repair or maintenance to the source.

4.7.9.3 Reports - A report shall be prepared and submitted to the Commission on an annual basis if sealed source or fission detector leakage tests reveal the presence of $\geq$0.005 microcuries of removable contamination.

4.7.10 FIRE SUPPRESSION SYSTEMS

FIRE SUPPRESSION WATER SYSTEM

4.7.10.1.1 The FIRE SUPPRESSION WATER SYSTEM shall be demonstrated OPERABLE:

a. At least once per 7 days by verifying that at least 90,000 gallons of water are in the Altitude Tank, equivalent level in Unit 1 circulating water flume and/or equivalent level in the river.

b. At least once per 31 days on a STAGGERED TEST BASIS by starting each pump and operating it for at least 20 minutes on recirculation flow.

c. At least once per 31 days by verifying that each accessible (per occupational exposure considerations) valve (manual, power operated or automatic) in the flow path is in its correct position.

d. At least once per 12 months by:

   1. Running the full flow of one fire pump through all main header loops as a flush, and

   2. Cycling each accessible (per occupational exposure considerations) testable valve in the flow path through at least one complete cycle of full travel.

e. At least once per 18 months by performing a system functional test which includes simulated automatic actuation of the system (i.e., pumps start at set pressure ± 10 psig), and:

   1. Verifying that each pump develops at least 2500 gpm at a total dynamic head no more than 10% below that indicated on manufactures curves.

   2. Cycling each accessible (per occupational exposure considerations) valve in the flow path that is not testable during plant operation through at least one complete cycle of full travel, and

   3. Verifying that each high pressure pump starts (sequentially) to maintain the fire suppression water system pressure greater than or equal to 70 psig.

f. At least once per 3 years by performing a flow test of the system in accordance with Chapter 5, Section 11 of the Fire Protection Handbook, 14th Edition, published by the National Fire Protection Association.

FIRE SUPPRESSION WATER SYSTEM (Continued)

4.7.10.1.2 Each fire pump diesel engine shall be demonstrated OPERABLE:

a. At least once per 31 days by verifying;

   1. Its associated fuel storage tank contains at least 250 gallons of fuel, and

   2. The diesel starts from ambient conditions and operates for at least 20 minutes.

b. At least once per 92 days by verifying that a sample of diesel fuel from each fuel storage tank, obtained in accordance with ASTM-D270-65, is within the acceptable limits specified in Table 1 of ASTM D975-74 when checked for viscosity, water and sediment.

c. At least once per 18 months, by:

   1. Subjecting each diesel to an inspection in accordance with procedures prepared in conjunction with its manufacturer's recommendations for the class of service, and

   2. Verifying each diesel starts from ambient conditions on the auto-start signal and operates for greater than or equal to 20 minutes while loaded with the fire pump.

4.7.10.1.3 Each fire pump diesel starting battery bank and charger shall be demonstrated OPERABLE:

a. At least once per 7 days by verifying that:

   1. The electrolyte level of each battery is above the plates, and

   2. The overall battery voltage is greater than or equal to 24 volts.

b. At least once per 92 days by verifying that the specific gravity is appropriate for continued service of the battery.

c. At least once per 18 months by verifying that:

   1. The batteries, cell plates and battery racks show no visual indication of physical damage or abnormal deterioration, and

   2. The battery-to-battery and terminal connections are clean, tight, free of corrosion and coated with anti-corrosion material.

DELUGE/SPRINKLER SYSTEMS

4.7.10.2 Each of the required deluge and/or sprinkler systems shall be demonstrated OPERABLE:

a. At least once per 12 months by cycling each accessible (per occupational exposure considerations) testable valve in the flow path through at least one complete cycle of full travel.

b. At least once per 18 months:

   1. By performing a system functional test (per occupational exposure considerations) which includes tripping detectors and verifying actuation of trip devices on associated deluge valves. Deluge/sprinkler valves shall be inspected internally to verify operability in all instances where header flooding during the test is not practical.

   2. Cycling each valve accessible (per occupational exposure considerations) in the flow path that is not testable during plant operation through at least one complete cycle of full travel.

   3. By visual inspection of the accessible (per occupational exposure considerations) deluge headers to verify their integrity.

   4. By visual inspection of each accessible (per occupational exposure considerations) nozzle to verify no blockage.

c. At least one per 3 years by performing an air or gas flow test through each deluge header and verifying each deluge nozzle is unobstructed.

HALON SYSTEM

4.7.10.3.1 The Cable Room and Transformer Room Halon system shall be demonstrated OPERABLE at least once per 6 months by verifying each Halon storage tank weight and pressure.

4.7.10.3.2 The Air Intake Tunnel Halon System shall be demonstrated OPERABLE:

a. At least once per 6 months by verifying pressure in each halon storage tank, and

b. At least once per 18 months by verifying storage tank weight.

FIRE HOSE STATIONS

4.7.10.4 Each (accessible per occupational exposure considerations) fire hose station listed in Table 4.7-1 shall be verified OPERABLE:

a. At least once per 31 days by visual inspection of the station to assure all required equipment is at the station.

b. At least once per 18 months by:

   1. Removing the hose for inspection and re-racking, and

   2. Replacement of all degraded gaskets in couplings.

c. At least once per 3 years by:

   1. Partially opening each hose station valve to verify valve OPERABILITY and no flow blockage.

   2. Conducting a hose hydrostatic test at a pressure at least 50 psig greater than the maximum pressure available at that hose station.

PENETRATION FIRE BARRIERS

4.7.11 The required (accessible per occupational exposure considerations) penetration fire barriers shall be verified to be functional:

a. At least once per 18 months by a visual inspection.

b. Prior to returning a penetration fire barrier to functional status following repairs or maintenance by performance of a visual inspection of the affected penetration fire barrier(s).

TABLE 4.7-1

FIRE HOSE STATIONS

Auxiliary Building

1. Fire hose near stairway at South end of building near Decay Heat Closed Cooling Surge Tank, 328' elevation.
2. Fire hose near entrance to Chemical Addition area, 328' elevation.
3. Fire hose near Miscellaneous Waste Holdup Tank room, 305' elevation.
4. Fire hose near Intermediate Closed Coolers, 305' elevation.
5. Fire hose near Evaporator Condensate Test Tank Room, 280' elevation.
6. Fire hose near Reactor Building Emergency Cooling Booster Pumps, 280' elevation.

Control Building Area

1. Fire hose near 480 Volt Bus 2-34 282'6" elevation.
2. Fire hose near entrance to Service Building 282'6" elevation.

Reactor Building*

1. Deleted.
2. Deleted.
3. Fire hose near West stairway, 305' elevation.
4. Fire hose near Southwest stairway, 305' elevation.
5. Fire hose on East D-ring, 367' elevation.
6. Fire hose on West D-ring, 367' elevation.

Control Building

1. Fire hose near doorway on North wall, 305' elevation.

Fuel Hand Building

1. Fire hose on East wall, 347'6" elevation.

---

*Fire hose stations in the Reactor Building shall be OPERABLE prior to initiating operations which generate combustible materials, and prior to initiating operations which involve the use of open flames, welding, buring, grinding, etc.

4.8 ELECTRICAL POWER SYSTEMS

4.8.2 A.C. SOURCES

4.8.1.1.1 Each of the required independent circuits between the offsite transmission network and the onsite Class 1E distribution system shall be determined OPERABLE at least once per 7 days by verifying correct breaker alignments and indicated power availablity.

4.8.1.1.2 Each diesel generator shall be demonstrated OPERABLE:

a. At least once per 31 days on a STAGGERED TEST BASIS by:

   1. Verifying the fuel level in the day fuel tank.

   2. Verifying the fuel level in the fuel storage system.

   3. Verifying the fuel transfer pump can be started and transfers fuel from the storage system to the day tank.

   4. Verifying the Class 1E diesels start from ambient condition and accelerate to at least 900 rpm (ie 60 Hz on the generator) in less than or equal to 10 seconds.

   5. Verifying each Class 1E diesel generator is synchronized; loaded to greater than or equal to 3000 kw, and operates for greater than or equal to 60 minutes.

   6. Verifying the diesel generator is aligned to provide standby power to the associated emergency busses.

b. At least once per 92 days by verifying that a sample of diesel fuel obtained in accordance with ASTM-D270-63, from the fuel storage tank is within the acceptable limits specified in Table 1 of ASTM D975-74 when checked for viscosity, water and sediment.

4.8.1.1.3 Deleted.

A.C. SOURCES (Continued)

4.8.1.1.4 Deleted.

4.8.2 ONSITE POWER DISTRIBUTION SYSTEMS

A.C. DISTRIBUTION

4.8.2.1 The specified A.C. busses shall be determined OPERABLE with tie breakers open between redundant busses at least once per 7 days by verifying correct breaker alignment and indicated power availability.

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 # 2-32, 2-35, 2-36, 2-42, 2-45, and 2-46
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

D.C. DISTRIBUTION

4.8.2.2.1 Each D.C. bus train shall be determined OPERABLE and energized with tie breakers open at least once per 7 days by verifying correct breaker alignment and indicated power availability.

4.8.2.3.2 Each 250/125-volt battery bank and charger shall be demonstrated OPERABLE:

a. At least once per 7 days by verifying that:

   1. The electrolyte level of each pilot cell is between the minimum and maximum level indication marks.

   2. The pilot cell specific gravity, corrected to 77°F and full electrolyte level, is greater than or equal to 1.20.

   3. The pilot cell voltage is greater than or equal to 2.13 volts.

   4. The overall battery voltage is greater than or equal to 250/125 volts.

D.C. DISTRIBUTION (Continued)

b. At least once per 92 days by verifying that:

1. The voltage of each connected cell is greater than or equal to 2.13 volts under float charge and has not decreased more than 0.10 volts from the value observed during the original acceptance test.

2. The specific gravity, correct to 77°F and full electrolyte level, of each connected cell is greater than or equal to 1.20 and has not decreased more than 0.01 from the value observed during the previous test.

3. The electrolyte level of each connected cell is between the minimum and maximum level indication marks.

c. At least once per 18 months by verifying that:

1. The cells, cell plates and battery racks show no visual indication of physical damage or deterioration.

2. The cell-to-cell and terminal connections are clean, tight and coated with anti-corrosion material.

4.9 LIQUID RADWASTE STORAGE

FUEL HANDLING BUILDING/AUXILIARY BUILDING AIR CLEANUP SYSTEMS

4.9.12.1 The Fuel Handling Building Air Cleanup Exhaust System shall be demonstrated OPERABLE:

a. At least once per 31 days by verifying that the Air Cleanup Exhaust System in the normal operating mode meets the following conditions:

   1. Exhaust Flow Rate: With two filter trains and two exhaust fans in operation in the Fuel Handling Building flow rate shall be within the 36,000 cfm to 54,000 cfm operating band.

   2. Filter Pressure Drop: While operating within the flow rate specified in 4.9.12.1.a.1 above, the d/p across the combined HEPA filters and charcoal adsorbers shall not exceed 6 inches water gauge.

   3. Fuel Handling Building Pressure: Demonstrate that the system is capable of achieving a negative pressure within the building equal to or greater (more negative) than 1/8 inch water gauge with respect to atmospheric. It may be necessary to close doors and other building openings to achieve the required value.

b. At least once per 18 months by verifying that the ventilation system meets the following conditions:

   1. Visually inspect each filter train and associated components in accordance with Section 5 of ANSI N510-1980, as required by Regulatory Position C.5.a of Regulatory Guide 1.52, Revision 2, March 1978. The inspection should be performed prior to the flow and DOP tests of this section.

   2. Flow Test: Exhaust flow rate shall be within 18,000 cfm to 27,000 cfm operating band for each filter train with one filter train and one exhaust fan operating. Testing shall be in accordance with ANSI N510-1980, Section 8.3.1, Paragraphs 3 and 4.

   3. DOP Test: Each filter train shall be tested in accordance with Section 10 of ANSI N510-1980, as required by Regulatory Position C.5.c of Regulatory Guide 1.52, Revision 2, March 1978. Flow through the filter train being tested shall be as prescribed for the flow test in Section 4.9.12.1.b.2 above.

NOTE: Installed system flow instrumentation is adequate for the test described in 4.9.12.1.b.3 above.

4. Fuel Handling Building Pressure: Demonstrate that the system is capable of achieving a negative pressure within the building equal to or greater (more negative) than 1/8 inch water gauge with respect to atmospheric. It may be necessary to close doors and other building openings to achieve the required value. A test instrument, such as an inclined manometer or equivalent, shall be used in the performance of this test.

c. After structural maintenance of the HEPA filter or charcoal adsorber housings, or following fire or chemical release in any ventilation zone communicating with the system by verifying that the ventilation system meets the following conditions:

1. Flow Test: Reverify exhaust flow rate for the affected filter train(s) per Section 4.9.12.1.b.2.

2. Filter Pressure Drop: Reverify the filter pressure drop surveillance prescribed in Section 4.9.12.1.a.2 for the affected filter train(s)

3. DOP Test: Each affected filter train shall be retested in accordance with Section 4.9.12.1.b.3.

d. After each complete or partial replacement of a HEPA filter bank by verifying that the ventilation system meets the following conditions:

1. DOP Test: Each affected filter train shall be retested in accordance with Section 4.9.12.1.b.3.

NOTE: Supply fans may be operated as desired except that the number of operating supply fans shall not exceed the number of operating exhaust fans.

4.9.12.2 The Auxiliary Building Air Cleanup Exhaust System shall be demonstrated OPERABLE:

a. At least once per 31 days by verifying that the air cleanup exhaust system in the normal operating mode meets the following conditions:

1. Exhaust Flow Rate: With two filter trains and two exhaust fans in operation in the Auxiliary building flow rate shall be within the 54,000 cfm to 80,000 cfm operating band.

2. Filter Pressure Drop: While operating within the flow rate specified in 4.9.12.2.a.1 above, the d/p across the combined HEPA filters and charcoal adsorbers shall not exceed 6 inches water gauge.

3. Auxiliary Building Pressure: Demonstrate that the system is capable of achieving negative pressure within the building equal to or greater (more negative) than 1/8 inch water gauge with respect to atmospheric. It may be necessary to close doors and other building openings to achieve the required value.

b. At least once per 18 months by verifying that the ventilation system meets the following conditions:

1. Visually inspect each filter train and associated components in accordance with Section 5 of ANSI N510-1980, as required by Regulatory Position C.5.a of Regulatory Guide 1.52, Revision 2, March 1978. The inspection should be performed prior to the flow and DOP tests of this section.

2. Flow Test: Exhaust flow rate shall be within 27,000 cfm to 40,000 cfm operating band for each filter train with one filter train and one exhaust fan operating. Testing shall be in accordance with ANSI N510-1980, Section 8.3.1, Paragraphs 3 and 4.

3. DOP Test: Each filter train shall be tested in accordance with Section 10 of ANSI N510-1980, as required by Regulatory Position C.5.c of Regulatory Guide 1.52, Revision 2, March 1978. Flow through the filter train being tested shall be as prescribed for the flow test in Section 4.9.12.2.b.2 above.

NOTE: Installed system flow instrumentation is adequate for the test described in 4.9.12.2.b.3. above.

4. Auxiliary Building Pressure: Demonstrate that the system is capable of achieving a negative pressure within the building equal to or greater (more negative) than 1/8 inch water gauge with respect to atmospheric. It may be necessary to close doors and other building openings to achieve the required value. A test instrument, such as an inclined manometer or equivalent, shall be used in the performance of this test.

c. After structural maintenance of the HEPA filter or charcoal adsorber housing, or following fire or chemical release in any ventilation zone communicating with the system by verifying that the ventilation system meets the following conditions:

1. Flow Test: Reverify exhaust flow rate for the affected filter train(s) per Section 4.9.12.2.b.2.

2. Filter Pressure Drop: Reverify the filter pressure drop surveillance prescribed in Section 4.9.12.2.a.2 for the affected filter train(s).

3. DOP Test: Each affected filter train shall be retested in accordance with Section 4.9.12.2.b.3.

d. After each complete or partial replacement of a HEPA filter bank by verifying that the ventilation system meets the following conditions:

1. DOP Test: Each affected filter train shall be retested in accordance with Section 4.9.12.2.b.3.

NOTE: Supply fans may be operated as desired except that the number of operating supply fans shall not exceed the number of operating exhaust fans.

Enclosure 5

UNITED STATES NUCLEAR REGULATORY COMMISSION

In the Matter of	)	
	)	
GENERAL PUBLIC UTILITIES NUCLEAR CORPORATION	)	Docket No. 50-320
	)	
(Three Mile Island Nuclear Station, Unit 2)	)	

EXEMPTION

I.

GPU Nuclear Corporation, Metropolitan Edison Company, Jersey Central Power and Light Company and Pennsylvania Electric Company (collectively, the licensee) are the holders of Facility Operating License No. DPR-73, which had authorized operation of the Three Mile Island Nuclear Station, Unit 2 (TMI-2) at power levels up to 2772 megawatts thermal. The facility, which is located in Londonderry Township, Dauphin County, Pennsylvania, is a pressurized water reactor previously used for the commercial generation of electricity.

By Order for Modification of License, dated July 20, 1979, the licensee's authority to operate the facility was suspended and the licensee's authority was limited to maintenance of the facility in the present shutdown cooling mode (44 Fed. Reg. 45271). By further Order of the Director, Office of Nuclear Reactor Regulation, dated February 11, 1980, a new set of formal license requirements was imposed to reflect the post-accident condition of the facility and to assure the continued maintenance of the current safe, stable, long-term cooling condition of the facility (45 Fed. Reg. 11292). This license provides, among other things, that it is subject to all rules, regulations and Orders of the Commission now or hereafter in effect.

II.

On April 18, 1984, General Public Utilities Nuclear Corporation (GPUNC) requested an Exemption from the requirements of 10 CFR 50.55a with respect to Code Safety Valves for TMI-2. This provision of the Commission's regulations currently requires that components which are part of the reactor coolant pressure boundary meet the requirements for Class 1 components in Section III of the ASME Boiler and Pressure Vessel Code.

As stated in Table 5.2-1 of the TMI-2 Final Safety Analysis Report (FSAR), the TMI-2 Code Safety Valves (pressurizer) meet the requirements of ASME Section III, Article 9, Summer 1969 Addendum. N-910.1 of Article 9 states that, "Each vessel within the scope of the Code shall be protected while in service from consequences arising from the application of steady state or transient conditions of pressure and (coincident) temperature which are in excess of the design conditions...." N-910.4, N-910.5, N-910.6, N-910.7, and N-910-8 state various design and location requirements for the relief valves. As stated in the FSAR, the TMI-2 reactor coolant system has a design pressure of 2500 psig with the pressurizer Code Safety Valves relieving at approximately 2450 psig with a 690,000 pounds per hour capacity.

III.

In the current system configuration, with the reactor vessel head removed to facilitate defueling, the code safety valves are not useable and are not required in order to relieve system pressure. It is also the staff's opinion that the normal maintenance performed and presently needed on these valves (bench testing, blowdown, seal inspection, etc.) would be an unnecessary maintenance burden and result in an unjustified radiological dose to the plant workers. As discussed in the concurrently issued safety evaluation, the reactor coolant system will remain open to the reactor building atmosphere throughout the recovery period. The system's configuration inherently provides overpressure protection because of the lack of a closed system that is necessary for a significant pressure buildup. The only thermodynamic event that can occur in the RCS that would have potential negative consequence is a system heatup. Because of the open vessel, even a heatup would have no pressure consequences unless the containment atmosphere pressure increased. Because of the volume of the containment (approximately 2×10^6 cu. ft.) and the amount of decay heat present (approximately 15 Kw), any significant containment pressure buildup would occur over a period of days, if not weeks. Should a currently unforeseen event occur that could potentially cause pressure to increase or which would require that a pressurized system be reestablished, the staff and the licensee would have sufficient response time to decide a course of action whether it be placing the head back on the vessel, installing a pressure relief component, or leaving the system as is. Therefore, it is the staff's opinion that a pressure relief device for the RCS need not be in place at this time. If a decision is made in the future to repressurize the RCS, a maximum pressure rating and appropriate overpressure protection must be specified in a safety evaluation approved by the staff, and in procedures approved pursuant to Section 6.8.2 of the PTS.

Because of the suspension of the licensee's authority to operate the facility in other than the present recovery mode as defined in the proposed technical specifications, certain of the regulations, which are intended to apply to normal operating plants, are simply inappropriate and, more significantly, are unnecessary to protect the public health and safety. Given the unique status of the plant in terms of primary system temperature and pressure, available fission product inventory, the ability to cool the reactor without forced circulation (loss-to-ambient), and the low decay heat rate, maintenance of the facility with the exemptions granted hereby will provide an adequate level of safety.

IV.

Accordingly, the Commission has determined that, pursuant to 10 CFR 50.12, an exemption is authorized by law and will not endanger life or property or the common defense and security and is otherwise in the public interest. The Commission hereby grants an exemption to the requirements of 10 CFR Part 50, Appendix A, Criterion 2, 50, and 51.

It is further determined that the exemption does not authorize a change in effluent types or total amounts nor an increase in power level and will not result in any significant environmental impact. In light of this determination and as reflected in the Environmental Assessment and Notice of Finding of No Significant Environmental Impact prepared pursuant to 10 CFR 51.21 and 51.30 through 51.32, issued concurrently herewith, it was concluded that the instant action is insignificant from the standpoint of environmental impact and an environmental impact statement need not be prepared.

FOR THE NUCLEAR REGULATORY COMMISSION

Harold R. Denton, Director
Office of Nuclear Reactor Regulation

Effective Date: December 19, 1984
Dated at Bethesda, Maryland
Issuance Date: December 19, 1984

Enclosure 6

UNITED STATES NUCLEAR REGULATORY COMMISSION

In the Matter of	)	
	)	
GENERAL PUBLIC UTILITIES NUCLEAR CORPORATION	)	Docket No. 50-320
	)	
(Three Mile Island Nuclear Station, Unit 2)	)	

EXEMPTION

I.

GPU Nuclear Corporation, Metropolitan Edison Company, Jersey Central Power and Light Company and Pennsylvania Electric Company (collectively, the licensee) are the holders of Facility Operating License No. DPR-73, which had authorized operation of the Three Mile Island Nuclear Station, Unit 2 (TMI-2) at power levels up to 2772 megawatts thermal. The facility, which is located in Londonderry Township, Dauphin County, Pennsylvania, is a pressurized water reactor previously used for the commercial generation of electricity.

By Order for Modification of License, dated July 20, 1979, the licensee's authority to operate the facility was suspended and the licensee's authority was limited to maintenance of the facility in the present shutdown cooling mode (44 Fed. Reg. 45271). By further Order of the Director, Office of Nuclear Reactor Regulation, dated February 11, 1980, a new set of formal license requirements was imposed to reflect the post-accident condition of the facility and to assure the continued maintenance of the current safe, stable, long-term cooling condition of the facility (45 Fed. Reg. 11292). This license provides, among other things, that it is subject to all rules, regulations and Orders of the Commission now or hereafter in effect.

II.

In a letter dated April 18, 1984, the licensee requested an exemption from the requirements of 10 CFR 50 relative to Seismic Monitoring Instrumentation. 10 CFR 50 50.36(c)(3) requires surveillance requirements ". . . to assure that the necessary quality of systems and components is maintained, that facility operation will be within the safety limits, and that the limiting conditions of operation will be met." 10 CFR 100, Appendix A, Section VI(a)(3) states that:

> "Suitable instrumentation shall be provided so that the seismic response of nuclear power plant features important to safety can be determined promptly to permit comparison of such response with that used as the design basis. Such a comparison is needed to decide whether the plant can be operated safely and to permit such timely action as may be appropriate.
>
> These criteria do not address the need for instrumentation that would automatically shut down a nuclear power plant when an earthquake occurs which exceeds a predetermined intensity...."

Presently, Section 4.3.3.3.1 of the TMI-2 PTS requires that Transaxial Time - History Accelographs be operable for the Reactor Building Ring Girder and the Reactor Building Mat; that Triaxial Peak Accelographs be operable for the Reactor Service Structure, "B" Core Flood Tank Piping and 2-IE Switchgear; that Triaxial Seismic Switches be operable for the Reactor Building Base and that Triaxial Response - Spectrum Recorders be operable for the Reactor Building Mat.

III.

The TMI-2 core is cooled via loss of heat to the reactor building environment. This is a passive mode that does not require any mechanical equipment to be operating to maintain an ability to cool the core. As stated in 10 CFR 100, Appendix A, Section VI(a)(3), one of the reasons for seismic instrumentation is to decide whether or not the plant can be operated safely. In the July 20, 1979 Order for Modification of License, the authority to operate the facility was suspended and the licensee's authority was limited to the maintenance of the facility in the present shutdown cooling mode. Therefore this basis for Section VI(a)(3) does not apply to TMI-2. In reference to the seismic instrumentation providing information for timely actions by plant personnel and the NRC, it is the staff's opinion that if a seismic event were to occur at TMI, the status of the core would not be affected because of the passive cooling mode and therefore no immediate actions would have to be taken to maintain the health and safety of the public. It is also the staff's opinion that when considering the above discussion, maintenance and surveillance requirements for seismic instrumentation is also not justified and is an unnecessary burden on the licensee.

Because of the suspension of the licensee's authority to operate the facility in other than the present recovery mode as defined in the proposed technical specifications, certain of the regulations, which are intended to apply to normal operating plants, are simply inappropriate and, more significantly, are unnecessary to protect the public health and safety. Given the unique status of the plant in terms of primary system temperature and pressure, available fission product inventory, the ability to cool the reactor without forced circulation (loss-to-ambient), and the low decay heat rate, maintenance of the facility with the exemptions granted hereby will provide an adequate level of safety.

IV.

Accordingly, the Commission has determined that, pursuant to 10 CFR 50.12, an exemption is authorized by law and will not endanger life or property or the common defense and security and is otherwise in the public interest. Based on the discussions above, the Commission hereby grants an exemption to the requirements of 10 CFR 50.36(c)(3) and 10 CFR 100, Appendix A, Section VI(a)(3) relative to seismic instrumentation.

It is further determined that the exemption does not authorize a change in effluent types or total amounts nor an increase in power level and will not result in any significant environmental impact. In light of this determination and as reflected in the Environmental Assessment and Notice of Finding of No Significant Environmental Impact prepared pursuant to 10 CFR 51.21 and 51.30 through 51.32, issued concurrently herewith, it was concluded that the instant action is insignificant from the standpoint of environmental impact and an environmental impact statement need not be prepared.

FOR THE NUCLEAR REGULATORY COMMISSION

Harold R. Denton, Director
Office of Nuclear Reactor Regulation

Effective Date: December 19, 1984
Dated at Bethesda, Maryland
Issuance Date: December 19, 1984

ENCLOSURE 7
(previously published
in the Federal Register)

UNITED STATES NUCLEAR REGULATORY COMMISSION

GENERAL PUBLIC UTILITIES NUCLEAR CORPORATION

DOCKET NO. 50-320

ENVIRONMENTAL ASSESSMENT AND NOTICE OF FINDING OF NO SIGNIFICANT ENVIRONMENTAL IMPACT

The U.S. Nuclear Regulatory Commission (the Commission) is planning to issue an Amendment of Order and two Exemptions relative to Facility Operating License No. DPR-73, issued to General Public Utilities Nuclear Corporation (the licensee), for operation of the Three Mile Island Nuclear Station, Unit 2 (TMI-2), located in Londonderry Township, Dauphin County, Pennsylvania.

ENVIRONMENTAL ASSESSMENT

Identification of Proposed Action: There are three actions being considered by the Commission. They include an Amendment of Order, an Exemption from the seismic instrumentation requirements of 10 CFR 50.36(c)(3) and 10 CFR 100, Appendix A, Section VI(a)(3); and an Exemption from the Code Safety Valve requirements of 10 CFR 50.55a.

The Amendment of Order is being issued in response to General Public Utilities Nuclear Corporation's (GPUNC) letters dated January 12, 1983, September 12, 1983, and September 30, 1983. The subject Exemptions were requested in two letters from GPUNC, both dated April 18, 1984.

The Need for the Action: The Amendment of Order is warranted because of the need to update the Proposed Technical Specifications (PTS) for the present conditions at TMI-2. The changes were grouped in six categories:

(1) Editorial Changes that correct spelling, grammar, page numbers, and the associated index and do not affect the technical content nor the intent of the section; (2) Modifications to the existing Limiting Conditions for Operation and Safety Limits that were made to more correctly state what systems or equipment are necessary based on the present status of TMI-2; (3) New Limiting Conditions for Operation that were added to also more correctly reflect what systems or equipment are necessary based on the present status of TMI-2; (4) Design Features of Section 5.0 that were modified or deleted to more accurately state design limits that can be verified; (5) Bases that were revised because of changes in technical approaches resulting from data that has been obtained and analyzed; and (6) Tables listing specific pieces of equipment that were moved from the Limiting Conditions for Operation sections to the applicable section of the Recovery Operations Plan.

Exemption from 10 CFR §50.55a

In the current system configuration, with the reactor vessel head removed to facilitate defueling, the code safety valves are not useable and are not required in order to relieve system pressure. As discussed in the concurrently issued safety evaluation, the reactor coolant system will remain open to the reactor building atmosphere throughout the recovery period. The system's configuration inherently provides overpressure protection because of the lack of a closed system that is necessary for a significant pressure buildup. The only thermodynamic event that can occur in the RCS that would have potential negative consequence is a system heatup. Because of the open vessel, even a heatup would have no pressure consequences unless the containment atmosphere pressure increased. Because of the volume of the containment (approximately 2×10^6 cu. ft.) and the amount of decay heat present (approximately 15 Kw), any significant containment pressure buildup would occur over a period of days, if not weeks. Should a currently unforeseen event occur that could potentially cause pressure to increase or which would require that a pressurized system be reestablished, the staff and the licensee would have sufficient response time to decide a course of action whether it be placing the head back on the vessel, installing a pressure relief component, or leaving the system as is. None of these decisions would have a negative impact on environmental conditions.

Exemption from 10 CFR 100, Appendix A and §50.36(c)(3)

The TMI-2 core is cooled via loss of heat to the reactor building environment. This is a passive mode that does not require any mechanical equipment to be operating to maintain an ability to cool the core. As stated in 10 CFR 100, Appendix A, Section VI(a)(3), one of the reasons for seismic instrumentation is to decide whether or not the plant can be operated safely. In the July 20, 1979 Order for Modification of License, the authority to operate the facility was suspended and the licensee's authority was limited to the maintenance of the facility in the present shutdown cooling mode. Therefore this basis for Section VI(a)(3) does not apply to TMI-2. In reference to the seismic instrumentation providing information for timely actions by plant personnel and the NRC, it is the staff's opinion that if a seismic event were to occur at TMI, the status of the core would not be affected because of the passive cooling mode and therefore no immediate actions would have to be taken to maintain the health and safety of the public. It is also the staff's opinion that when considering the above discussion, maintenance and surveillance requirements for seismic instrumentation is also not justified and is an unnecessary burden on the licensee. There are no negative radiological effects as a result of these requirements.

Environmental Impacts of the Proposed Actions: The staff has evaluated the PTS modifications proposed by the Amendment of Orders and concluded that they will not result in significant increases in airborne or liquid contamination radioactivity inside the reactor building or in corresponding releases to the environment. There are also no non-radiological impacts to the environment as a result of these actions.

Alternative to this Action: Since we have concluded that there is no significant environmental impact associated with the subject Amendment of Order and Exemptions, any alternatives to these changes will have either no significant environmental impact or greater environmental impact. The principal alternative would be to deny the requested actions. This would not reduce significant environmental impacts of plant operations and would result in the application of overly restrictive regulatory requirements when considering the unique conditions at TMI-2.

Agencies and Persons Consulted: The NRC staff reviewed the licensee's request and did not consult other agencies or persons.

Alternate Use of Resources: This action does not involve the use of resources not previously considered in connection with the Final Programmatic Impact Statement for TMI-2 dated March 1981.

Finding of No Significant Impact: The Commission has determined not to prepare an environmental impact statement for the subject Amendment of Order and Exemptions. Based upon the foregoing environmental assessment, we conclude that this action will not have a significant effect on the quality of the human environment.

For further details with respect to this action see (1) Letter to B. J. Snyder, USNRC, from R. C. Arnold, GPUNC, Technical Specification Change Request No. 39, dated January 12, 1983, (2) Letter to B. J. Snyder, USNRC, from R. C. Arnold, GPUNC, Technical Specification Change Request No. 41, dated September 12, 1983, (3) Letter to B. J. Snyder, USNRC, from R. C. Arnold, GPUNC, Technical Specification Change Request No. 43, dated September 30, 1983, (4) Letter to L. H. Barrett, USNRC, from B. K. Kanga, GPUNC, Recovery Operations Plan Change Request No. 19, dated January 12, 1983, (5) Letter to L. H. Barrett, USNRC, from B. K. Kanga, GPUNC, Recovery Operations Plan Change Request No. 20, dated September 12, 1983, (6) Letter to L. H. Barrett, USNRC, from B. K. Kanga, GPUNC, Recovery Operations Plan Change Request No. 22, dated September 30, 1983, (7) Letter to B. J. Snyder, USNRC, from E. E. Kintner, GPUNC, Seismic Monitoring Exemption Request, dated April 18, 1983, (8) Letter to B. J. Snyder, USNRC, from E. E. Kintner, GPUNC, Exemption Request from 10 CFR 50.55a with respect to Code Safety Valves, dated April 18, 1984, and (9) the Director's Order of February 11, 1980.

All of the above documents are available for inspection at the Commission's Public Document Room, 1717 H Street, N.W., Washington, D.C. and at the Commission's Local Public Document Room at the State Library of Pennsylvania, Government Publications Section, Education Building, Commonwealth and Walnut Streets, Harrisburg, Pennsylvania 17126.

FOR THE NUCLEAR REGULATORY COMMISSION

Bernard J. Snyder

Bernard J. Snyder, Program Director
Three Mile Island Program Office
Office of Nuclear Reactor Regulation