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

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February 13, 1980

ORDER TO DPR-73 Replacing App "A" T/S - also submits Recovery Operatic Plan (ROP)

Docket No. 50-320

Mr. R. C. Arnold
Senior Vice President
Metropolitan Edison Company
P. O. Box 480
Middletown, Pennsylvania 17057

Dear Mr. Arnold:

The NRC has issued the enclosed Order for the Three Mile Island Nuclear Station, Unit 2.

This Order (1) requires that effective immediately, the facility be maintained in accordance with the requirements of the attached proposed Technical Specifications and (2) proposes to formally amend the Facility Operating License to include the proposed Technical Specifications, taking into account the present condition of plant systems, so as to ensure that the unit will remain in a safe and stable posture during the Recovery Mode.

Copies of the related Safety Evaluation and Environmental Assessment, Negative Declaration and Technical Specifications are also enclosed.

Sincerely,

Richard H. Vollmer, Director
TMI-2 Support

Enclosures:
1. Order
2. Safety Evaluation and Environmental Assessment (NUREG-0647)
3. Negative Declaration
4. Technical Specifications

cc w/enclosures:
See next page

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UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

In the Matter of	)	
	)	
METROPOLITAN EDISON COMPANY, ET AL.	)	Docket No. 50-320
	)	
(Three Mile Island Nuclear Station, Unit 2)	)	

ORDER

I.

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. 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 F.R. 45271). The facility, which is located in Londonderry Township, Dauphin County, Pennsylvania, is a pressurized water reactor used for the commercial generation of electricity.

II.

On March 28, 1979, an accident at the Three Mile Island Nuclear Station Unit 2 resulted in substantial damage to the reactor core and to certain reactor systems and components. The facility is not capable of normal

operation and is in a shutdown condition with fuel in the core. Although many systems were damaged or have subsequently failed, the facility is being maintained in a safe and stable cooling condition. The July 20, 1979 Order directed that the licensee maintain the facility in a shutdown condition in accordance with approved operating and contingency procedures with decay heat being removed by natural convection circulation of primary coolant through the core with heat rejection through the "A" steam generator.

Some of the systems and components currently being used to maintain the facility in its present mode of operation were not originally included in the facility's Technical Specifications. In fact, in the present post-accident status of the facility, the license itself does not include explicit provisions or Technical Specifications for assuring the continued maintenance of the plant in a safe, stable condition or for coping with foreseeable off-normal conditions. Moreover, certain portions of the facility's operating license relate to or govern power operation of the facility, the authority for which was suspended by the Order of July 20, 1979. These provisions are now simply inapplicable to the facility in its present post-accident condition. Consequently, the facility's operating license[1/] should be modified so as to:

1/ The existing Appendix B Technical Specifications imposed for the protection of the environment upon issuance of the TMI-2 operating license would be unchanged and would remain in effect, except that Appendix B Technical Specifications 5.1, 5.2 and 5.3, which identify the licensee's pre-accident management organization for activities addressed by the Appendix B Technical Specifications, are deleted since those requirements will now be performed in accordance with proposed Technical Specifications 6.1, 6.2 and 6.5 which will set forth the current requirements for the licensee's management organization for all licensed activities.

(1) define operating parameters for the current safe, stable, long-term cooling mode for the facility (defined as the recovery mode), and delete all other permissible operating modes so as to assure that operation of the facility in other than the stable shutdown condition of the recovery mode is precluded;

(2) impose functional, operability, redundancy and surveillance requirements as well as safety limits and limiting conditions with regard to those structures, systems, equipment and components necessary to maintain the facility in the current safe, stable shutdown condition and to cope with foreseeable off-normal conditions.

(3) prohibit venting or purging or other treatment of the reactor building atmosphere, the discharge of water decontaminated by the EPICOR-II system, and the treatment and disposal of high-level radioactively contaminated water in the reactor building, until each of these activities has been approved by the NRC, consistent with the Commission's Statement of Policy and Notice of Intent to Prepare a Programmatic Environmental Impact Statement (44 F.R. 67738).

III.

For the above reasons, the Facility Operating License has been reviewed and a new set of Technical Specifications has been developed to account for the present condition of the facility and to assure the continued maintenance of

the current safe, stable, long-term cooling condition of the facility. The Staff's safety assessment addressing the requirements of the proposed Technical Specifications is set forth in the concurrently-issued Safety Evaluation.

The Office of Nuclear Reactor Regulation has also performed an environmental evaluation of the requirements that would be imposed by these proposed Technical Specifications as reflected in the Environmental Assessment, also issued concurrently. It was determined that since the limits on effluent releases and discharges contained in Appendix B to the Facility Operating License are not being changed and remain in effect, and the proposed Technical Specifications attached to this Order do not authorize a change in effluent types or total amounts nor an increase in power level that the proposed Technical Specifications will not result in any significant environmental impact. Accordingly, a negative declaration is being issued concurrently pursuant to 10 CFR § 51.5(c) (1) of the Commission's regulations.

Proposed Technical Specification 6.8.1, regarding Administrative Controls, requires that written procedures implementing certain requirements of the modified Technical Specifications be prepared by the licensee. Proposed Technical Specification 6.8.2 provides that these written procedures are to be submitted to, and are subject to approval by, the NRC prior to implementation. Since these written procedures will implement certain requirements of the proposed Technical Specifications, some period of time will be required after its effective date for the licensee to prepare the appropriate written procedures. Based on discussions with the licensee and an evaluation

of the scope of the written procedures required, it has been determined that a period of 30 days from the date of this Order will be sufficient for the licensee's preparation of the written procedures required by proposed Technical Specification 6.8.1.

Incorporation into the license of the proposed Technical Specifications would both impose necessary requirements on the licensee to reflect the present condition of the facility and to assure continued maintenance of the current safe, stable, long-term cooling condition of the facility, and facilitate the Commission's enforcement program by providing a single and readily available compilation of enforceable requirements. For these reasons, it is proposed that the license be amended so as to include the Technical Specifications set forth in Attachment 1 hereto. The formal license amendment incorporating these proposed Technical Specifications will become effective on the expiration of the period specified below, during which the licensee or any other person whose interest may be affected may request a hearing or, in the event a hearing is requested and granted, on the date specified in an order made following the hearing or other disposition of such proceeding.

As indicated above, the July 20, 1979 Order directed the licensee to maintain the facility in a shutdown condition in accordance with approved operating and contingency procedures. This Order is still in effect. The operating and contingency procedures referred to have evolved over the course of the past several months and are now reflected in the requirements that have been included in the proposed Technical Specifications set forth in Attachment 1.

In order to eliminate any possible uncertainty as to the nature and requirements of the operating and contingency procedures referred to in the July 20, 1979 Order, the public health, safety and interest require that that Order be amended, effective immediately, as set forth below, pursuant to section 161.b. of the Atomic Energy Act of 1954, as amended, to include the requirements in Attachment 1. When the proposed Technical Specifications become effective as a result of the formal amendment of the Facility Operating License, they will supersede the July 20, 1979 Order, as amended, since there will be no need to maintain a duplicate set of legal requirements on the licensee.

IV.

Accordingly, pursuant to the Atomic Energy Act of 1954, as amended, IT IS ORDERED THAT:

(1) Effective immediately, the Director's Order for Modification of License of July 20, 1979 is amended by deleting paragraph (2) and substituting the following:

"(2) Pending further amendment of the Facility Operating License, the licensee shall maintain the facility in accordance with the requirements set forth in Attachment 1 to the Director's Order dated February 11, 1980."

(2) Effective 30 days from the date of publication of this Order in the Federal Register or, in the event a hearing is requested and granted, on the date specified in an order made following the hearing or other disposition of the proceeding, all Appendix A

Technical Specifications and Appendix B Technical Specifications 5.1, 5.2 and 5.3 to Facility Operating License No. DPR-73 in effect on March 28, 1979 and all changes thereto issued between that date and the date of this Order are superseded by the proposed Technical Specifications set forth in Attachment 1 hereto.

(3) The written procedures required by proposed Technical Specification 6.8.1 in Attachment 1 hereto shall be completed and submitted to the NRC 30 days after the date of this Order. Pending approval of the written procedures by the NRC, the licensee shall follow the previously approved procedures.

V.

The licensee or any person whose interest may be affected may, within thirty (30) days of the date of publication of this Order in the Federal Register, file a request for a hearing with respect to this Order, pursuant to 10 CFR § 2.714. As indicated, a request for a hearing with regard to paragraph (2) of this Order shall operate to stay the effectiveness of that paragraph. Accordingly, any such hearing regarding paragraph (2) shall be conducted prior to the adoption of the formal license amendment incorporating the Technical Specifications set forth in Attachment 1 hereto. However, a request for hearing with regard to paragraphs (1) and/or (3) shall not operate to stay the effect of those paragraphs. In the event a hearing is held with regard to paragraph (1) and/or (3), however, such a hearing shall be consolidated with any hearing that may be requested pursuant to paragraph (2). In the event a hearing is held, the issues shall be: (1) whether

the requirements set forth in Attachment 1 hereto (summarized in Part II above) are necessary and sufficient for the maintenance of the facility to protect health and safety or to minimize danger to life and property; and (2) whether the provisions of this Order would significantly affect the quality of the human environment.

A request for a hearing must be filed with the Office of the Secretary, U.S. Nuclear Regulatory Commission, Washington, DC 20555, Attention: Docketing and Service Section, by the above date. A copy of the request for a hearing should also be sent to the Executive Legal Director, U.S. Nuclear Regulatory Commission, Washington, DC 20555 and to George F. Trowbridge, Esq., Shaw, Pittman, Potts, and Trowbridge, 1800 M Street, NW., Washington, DC 20036, attorney for the licensee. Any questions regarding the contents of this Order should be directed to the Chief Hearing Counsel, Office of the Executive Legal Director, U.S. Nuclear Regulatory Commission, Washington, DC 20555.

For further details with respect to this action, see (1) Operating License DPR-73, as amended, (2) NUREG-0557, "Evaluation of Long-Term Post-Accident Core Cooling of Three Mile Island, Unit 2," NRC Staff Report, dated May, 1979, (3) Memorandum to R. Vollmer from A. Ignatonis, "TMI-2 Plant Modifications for Cold Shutdown, Rev. 2," dated June 8, 1979, (4) Letter to R. Vollmer, NRC, from R. C. Arnold, Metropolitan Edison Co., "Reactor Containment Building Atmosphere Cleanup," dated November 13, 1979, (5) the Commission Staff's related Safety Evaluation and Environmental Assessment

(NUREG-0647), ar⁴ (6) the Director's Order for Modification of License dated July 20, 1979. All of the above documents are available for inspection at the Commission's Public Document Room, 1717 H Street, NW., Washington, DC 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

Harold R. Denton

Harold R. Denton, Director
Office of Nuclear Reactor Regulation

Dated at Bethesda, Maryland
this 11th day of February, 1980.

UNITED STATES NUCLEAR REGULATORY COMMISSION

NEGATIVE DECLARATION

REGARDING PROPOSED TECHNICAL SPECIFICATIONS FOR

THREE MILE ISLAND NUCLEAR STATION, UNIT NO. 2

DOCKET NO. 50-320

The U.S. Nuclear Regulatory Commission has determined that the public health, safety and interest require modification of Facility Operating License No. DPR-73, issued to Metropolitan Edison Company, et. al. for operation of the Three Mile Island Nuclear Station Unit 2, located in Londonderry Township, Dauphin County, Pennsylvania. This action would modify the Technical Specifications of the Facility Operating License to more accurately reflect the present condition of the facility resulting from the March 28, 1979 accident. Some of the systems and components currently being used to maintain the facility in its present mode of operation were not originally included in the facility's Technical Specifications. In fact, in the present post-accident status of the facility, the license itself does not include explicit provisions or Technical Specifications for assuring the continued maintenance of the plant in a safe, stable condition or for providing for foreseeable off-normal conditions. Moreover, certain portions of the facility's operating license relate to or govern power operation of the facility, the authority for which was suspended by Order for Modification of License of the Director, Office of Nuclear Reactor Regulation, dated July 20, 1979. These provisions are simply inapplicable to the facility in its present post-accident condition. Consequently, by Order dated February 11,

1980, the Director (1) amended the Order of July 20, 1979 effective immediately, to require that the licensee maintain the facility in accordance with the requirements set forth in Attachment 1 to the Order, and (2) proposed to formally amend the Technical Specifications accordingly, in the following areas: Nuclear Safety; Core Cooling, Water Inventory and Reactor Coolant System Pressure Control; Instrumentation; Containment Systems; Fire Detection and Fire Suppression; Electrical Power; Control of Radioactive Materials in Liquid and Gaseous Effluents; and, Review and Audit Functions.

The Office of Nuclear Reactor Regulation prepared a combined Safety Evaluation and Environmental Assessment (NUREG-0647) in connection with this action. It was determined that since the limits on effluent releases and discharges contained in the environmental Technical Specifications for the Facility Operating License (Appendix B) are not being changed and remain in effect, the actions encompassed by this Order 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. In addition, the Technical Specifications include prohibitions against the purging or other treatment of the reactor building atmosphere, the discharge or other disposal of water decontaminated by the Epicor-II system and the treatment and discharge or other disposal of the high-level radioactively contaminated water now in the reactor building without further Commission approval. Thus, in accordance with this finding, no Environmental Impact Statement will be prepared.

The Safety Evaluation and Environmental Assessment (NUREG-0647) and the Director's Order issued February 11, 1980 are available for public inspection at the Commission's Public Document Room, 1717 H Street, N.W., Washington, D.C., and at the Three Mile Island Unit No. 2 Local Public Document Room in the Government Publications Section, State Library of Pennsylvania, Education Building, Commonwealth and Walnut Streets, Harrisburg, Pennsylvania. Copies may be obtained upon request addressed to the U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, Attention: Richard Vollmer, Director TMI-2 Support, NRR.

FOR THE NUCLEAR REGULATORY COMMISSION

Harold R. Denton

Harold R. Denton, Director
Office of Nuclear Reactor Regulation

Dated at Bethesda, Maryland
this 11th day of February, 1980.

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

SAFETY EVALUATION AND ENVIRONMENTAL ASSESSMENT

BY THE OFFICE OF NUCLEAR REACTOR REGULATION

METROPOLITAN EDISON COMPANY

JERSEY CENTRAL POWER AND LIGHT COMPANY

PENNSYLVANIA ELECTRIC COMPANY

DOCKET NO. 50-320

THREE MILE ISLAND NUCLEAR STATION, UNIT NO. 2

INTRODUCTION

On March 28, 1979 an accident at the Three Mile Island Nuclear Station Unit 2 resulted in substantial damage to the reactor core and to certain reactor systems and components. The facility is not capable of operation and is in a shutdown condition with damaged fuel in the core. Although some systems were damaged or have subsequently failed, the facility is being maintained in a safe and stable cooling condition utilizing a substantial number of systems and components. Some of the systems and components currently being used to maintain the facility in its present mode of operation were not originally included in the facility's technical specifications because these systems were not required for safe operation of the facility under pre-accident conditions.

Since these additional systems and components are now being used to remove decay heat from the core, revised technical specifications to encompass the additional systems and components should be included in the facility license and other technical specifications for equipment not required during the present mode of operation should be deleted.

The systems and components available to provide plant safety, including long term cooling of the core, under the present conditions with the facility in cold shutdown and while cleanup and recovery of the facility proceed, have been reviewed. The reactor is presently being maintained in a stable, long term cooling mode with decay heat being removed by natural convection circulation of primary coolant through the core with heat rejection through the "A" steam generator. The "A" steam generator is producing steam which is condensed in the condenser and recirculated to the "A" steam generator. An alternate means of removing decay heat from the primary coolant is through the "B" steam generator. The steam side of the "B" steam generator has been modified to provide a water solid, closed loop cooling system which is in turn cooled by the secondary services closed cooling water system. Either steam generator cooling mode is adequate to remove decay heat from the primary coolant. If natural circulation cooling of the core should be lost, contingency plans and procedures have been prepared and approved for alternate means of providing long term core cooling. These alternate core cooling means include forced circulation of the primary cooling using the reactor coolant pumps or decay heat removal pumps. Operation of various systems to control the release of radioactive materials will also be required during the cleanup of radioactive materials released within the facility and the recovery of the facility from the effects of the accident. Appropriate Appendix A Technical Specifications governing this period (long term cooling of the core and during cleanup and recovery of the facility) have been established through conferences between the staff and the licensee. This safety evaluation describes the protection required to provide adequate safety during present conditions. It does not authorize removal of fuel from the reactor pressure vessel. Such authorization

must be obtained prior to any such removal and will be addressed in a subsequent safety evaluation.

This amendment does not include any changes in Appendix B (which remains in effect and includes effluent release limits) to the facility operating license, except that Appendix B Technical Specifications 5.1, 5.2 and 5.3, which identify the licensee's pre-accident management organization for activities addressed by the Appendix B Technical Specifications, are deleted since those requirements will now be performed in accordance with proposed Technical Specifications 6.1, 6.2 and 6.5 which will set forth the current requirements for the licensee's management organization for all licensed activities. This evaluation does not encompass operation of the EPICOR-II decontamination system currently being utilized at the facility pursuant to the terms of the Commission's Memorandum and Order of October 16, 1979 to process decontaminated intermediate-level radioactive waste water accumulated in the auxiliary building. The impact of using EPICOR-II was evaluated in an Environmental Assessment (NUREG-0591) prepared by this Office on October 3, 1979. See also Order for Modification of License and Negative Declaration issued by the Director of this Office on October 18, 1979. The Commission's decision of October 16, 1979 does not address the subject of disposal of the decontaminated water processed by EPICOR-II. Pursuant to the Commission's Statement of May 25, 1979, discharge of EPICOR-II processed waste water is not permitted until completion of an environmental review of such discharges.

During the process of preparing revised technical specifications, a new operational mode was defined. This new operational mode (designated the "Recovery Mode" and defined in Technical Specification 1.3) is intended to apply throughout the long-term cooling of the core and facility cleanup and recovery operations. This change in mode applicability is reflected in the revised technical specifications. This amendment deletes other operating modes and thereby precludes operation in other than the shutdown conditions defined for the Recovery Mode.

The March 28, 1979 accident resulted in excessively high radiation areas in certain portions of the facility; therefore, provisions have been included in the surveillance requirements for the revised technical specifications which relieve the licensee from the requirement to perform certain surveillance requirements when access to the equipment would result in excessive occupational exposures. It is expected that the areas in which this relief is necessary will be reduced as cleanup of the facility progresses.

Minor changes have been made in Technical Specifications 3.3.3.1, 3.3.3.3, 3.3.3.4, 3.3.3.5, 3.3.3.6, 3.3.3.7, 3.3.3.8, 3.4.3, 3.6.1.3, 3.6.1.4, 3.6.1.5, 3.6.4.1, 3.7.3.1, 3.7.3.2, 3.7.4.1, 3.7.6.1, 3.7.7.1, 3.7.10.1, 3.7.10.2, 3.7.10.3, 3.7.10.4, 3.7.11 and 3.8.2.3. These minor changes consist of changes in applicability requirements, changes to existing action statements which require reactor shutdown or prohibit plant startup with inoperable equipment, and deletion of operability requirements for equipment which has failed and cannot be repaired or equipment which is not required in the plant's present condition. These changes do not significantly increase the probability or consequences of an accident or significantly decrease a safety margin and, in fact, are of no safety significance.

The following Technical Specifications are being deleted since they are applicable only during operation in Modes 1, 2, 3, 4 and 6: 2.1.1, 2.1.2, 2.2.1, 3.0.4, 4.0.4, 4.0.5, 3.1.1.3, 3.1.1.4, 3.1.2.1 - 3.1.2.9, 3.1.3.2 - 3.1.3.9, 3.2.1 - 3.2.5, 3.3.3.2, 3.4.2, 3.4.4 - 3.4.8, 3.4.9.2, 3.4.10.1, 3.5.1 - 3.5.4, 3.6.1.2, 3.6.1.6, 3.6.1.7, 3.6.2.1 - 3.6.2.3, 3.6.3.1, 3.6.4.2, 3.6.4.4, 3.6.5, 3.7.1.2 - 3.7.1.6, 3.7.2.1, 3.7.5.1, 3.7.8.1, 3.7.9.1, 3.8.1.2, 3.8.2.2, 3.8.2.4, 3.9.1 - 3.9.11 and 3.10.1 - 3.10.4. Operation in Modes 1, 2, 3, 4 and 6 is no longer authorized; deletion of these Technical Specifications, therefore, does not significantly increase the probability or consequences of an accident or significantly decrease a safety margin. Therefore, these deletions do not involve a significant hazards consideration and in fact are of no safety significance.

EVALUATION

1. Nuclear Safety

The full length control rods (safety and regulating) were fully inserted into the core during the reactor trip which occurred at the beginning of the March 28, 1979 accident. To provide assurance that control rod motion will not cause a change in core reactivity, Technical Specification 3.1.3.1 requires that the control rod drive breakers be maintained open. Since the integrity of the control rods and the fuel rods is unknown, the staff has performed analyses which show that with a reactor coolant boron concentration of about 3000 ppm, the core will be maintained subcritical in all possible configurations (Reference 1). Consequently, revised Technical Specifications 3.1.1.1 and 3.1.1.2 have been prepared requiring two operable systems for injecting borated cooling water into the reactor coolant system and [illegible]

the reactor coolant boron concentration to be maintained between 3000 and 4500 ppm. The maximum boron concentration has been specified to assure that boron precipitation will not occur. A concentration of 4500 ppm boron in water has a precipitation temperature of approximately 45°F. Therefore, a requirement has been added to maintain the reactor coolant minimum temperature above 50°F thereby assuring that boron precipitation will not occur.

2. <u>Core Cooling, Water Inventory and Reactor Coolant System Pressure Control</u>

The core is presently being maintained in a stable cold shutdown condition and is being cooled by the reactor coolant system operating in natural circulation. Heat removal from the reactor coolant system is through the "A" steam generator which is producing steam. The steam is being routed to the condenser where it is being condensed and then recirculated to the "A" steam generator. An alternate means of removing decay heat from the primary coolant is available through the "B" steam generator. The steam side of the "B" steam generator has been modified to provide a water solid, closed loop cooling system which is in turned cooled by the secondary services closed cooling water system (Reference 2). Operability of the steam generators and associated cooling water system is required by Technical Specifications 3.7.1 and 3.7.2.1. Either steam generator cooling mode is adequate to remove the decay heat from the primary coolant (Reference 1). Technical Specification 3.4.1 requires that the reactor coolant pumps be maintained operable for possible forced circulation of reactor coolant in the event forced circulation cooling is required.

A standby reactor coolant system pressure control system has been added to the facility to maintain the reactor coolant system level and pressure for normal operation in the "Recovery Mode" and over a wide range of anticipated transient events which would cause shrinkage of the reactor coolant (Reference 2). These anticipated transients include loss of natural circulation cooling due to a loss of all secondary side cooling with restart of one secondary cooling loop following a hot leg temperature rise of 50°F. More severe transients which this system is not designed to accommodate would be handled by the high pressure injection pumps, the operability of which is required by Technical Specification 3.1.1.1. Appropriate surveillance requirements which demonstrate the operability of these systems have been incorporated. The operability of borated water sources which are sufficient to accommodate all possible transients is assured by appropriate surveillance requirements.

Technical Specification 3.4.9.1 has been modified to restrict the reactor coolant system temperature and pressure to 280°F and 600 psig. This provides assurance that the reactor pressure vessel will not be subjected to conditions which could result in its brittle fracture.

3. Instrumentation

Since the reactor will not be operated during this time period, the only portions of the reactor protection instrumentation required to be maintained in an operable condition are the source range and intermediate range neutron monitoring channels. Although the reactor will be

maintained subcritical via boron in the reactor coolant (Reference 1), these instruments are required to be maintained in an operable condition per Technical Specification 3.3.1.1 to provide the capability for monitoring the neutron level in the core.

The only Engineered Safety Feature Actuation System (ESFAS) instrumentation required to be maintained operable during this period is that provided to start the Class 1E diesel generators upon detection of a loss of offsite electrical power. This instrumentation is required operable per Technical Specification 3.3.2.1. Other ESFAS instrumentation is not required due to the low decay heat loads and the ample time available for manual initiation of systems available to accommodate possible transients. This is acceptable based upon the present plant conditions (Reference 2).

Since the reactor coolant system pressure instrumentation, reactor building water level instrumentation and the incore thermocouples are being used to assure core cooling and to provide assurance that vital equipment in the containment is not flooded, their operability is required and operability requirements for this instrumentation have been added to Technical Specification 3.3.3.6.

4. <u>Containment Systems</u>

Significant quantities of radioactive materials have been released into the containment. Containment integrity is required to be maintained by Technical Specification 3.6.1.1 to ensure that these materials are

not inadvertently released to the environs. This Technical Specification also prohibits venting or purging or other treatment of the reactor building atmosphere until such activity has been approved by the NRC. Since the licensee has proposed that the containment atmosphere be removed by purging through the hydrogen purge cleanup system (Reference 3), Technical Specification 3.6.4.3 is being retained to ensure the operability of this system in the event purging of the containment is approved and authorized.

5. Fire Detection and Fire Suppression

As part of the facility modifications made for long term cooling of the core, additional fire detection instrumentation and deluge/sprinkler systems were installed. These additions included fire detection instrumentation to protect the self-contained skid mounted "Grey" and "White" Balance of Plant (BOP) diesel generators and a deluge/sprinkler system to protect the auxiliary building exhaust filter. Operability requirements for this added equipment have been incorporated into Technical Specifications 3.3.3.8 and 3.7.10.2. The operability of these fire suppression systems ensures that adequate fire suppression capability is available to confine and extinguish fires. The surveillance requirements provide assurance that the minimum operability requirements of the fire suppression systems are met.

6. Electrical Power

The electrical energy to operate the systems being used to remove decay heat from the core is provided by redundant circuits from the offsite transmission network and by onsite power supplies. The present cooling mode requires the use of electrical power to operate equipment which previously did not require protection against loss-of-offsite power. Therefore, an additional 13.2 kv circuit from the Middletown Junction Substation and two redundant balance of plant diesel generators have been installed to increase the reliability of the offsite and onsite electrical power supplies (Reference 2). The new 13.2 kv circuit provides a backup offsite electrical power supply for two circulating water pumps (one of these pumps provides adequate cooling for removing decay heat). In the event of a total loss of offsite power the core can be cooled using only the onsite diesel generators as a power supply (Reference 2). The redundant self-contained skid-mounted "Gray" and "White" diesel generators have been installed to provide backup protection to all electrical loads which are required for core cooling and which were not previously protected against loss-of-offsite power. Therefore, Technical Specification 3.8.1.1 has been modified to require the operability of the backup 13.2 kv circuit and the two additional, redundant, balance of plant ("Gray" and "White") diesel generators.

7. Control of Radioactive Materials in Gaseous Effluents

The auxiliary building air cleanup system has been installed to filter gaseous effluents from the auxiliary building. Operation of this system in conjunction with the fuel handling building air cleanup system, ensures that any radioactive materials in effluents from these buildings will be processed through HEPA filters prior to release to the environs. The operability requirements for the auxiliary building air cleanup system have been added to Technical Specification 3.9.12 which previously contained the operability requirements for only the fuel handling building air cleanup system. The Surveillance Requirements for the charcoal adsorbers in the fuel handling building air cleanup system have been deleted since the radioactive iodine is no longer present; it has decayed away.

8. Control of Radioactive Materials in Liquid Effluents

The discharge of water processed by the EPICOR-II system and the processing and discharge of highly contaminated water contained in the Reactor Building sump and Reactor Coolant System is prohibited in accordance with the Commission's Statement of May 25, 1979 pending evaluation of these actions.

Furthermore, on November 21, 1979, the Commission announced its decision to prepare a programmatic environmental impact statement to address, among other things, the decontamination and disposal of radioactive waste water resulting from the accident. "Statement

of Policy and Notice of Intent to Prepare a Programmatic Environmental Impact Statement" (44 F.R. 67738). The Commission observed that as the decontamination of TMI-2 progresses the Commission will make available any new information to the public and to the extent necessary will also prepare separate environmental statements or assessments for individual portions of the overall cleanup effort. The Commission also indicated that in the event it should decide before completion of its programmatic statement that it is in the best interest of the public health and safety to decontaminate the high-level waste water now in the containment building or to purge the building of its radioactive gases, such action would not be taken until it had undergone an environmental review consistent with its May 25, 1979 Statement. The Commission has further recognized, however, the possibility that an emergency situation, now unforeseen, may arise which could require rapid action.

Accordingly, Technical Specifications 3.9.13 and 3.9.14 have been added to implement these requirements.

9. <u>Review and Audit Functions</u>

The accident of March 28, 1979 has resulted in the generation of large quantities of radioactive wastes. Therefore, the licensee has augmented the membership of his Plant Operations Review Committee and Generation Review Committee to provide additional expertise in the area of radioactive waste management. We have added requirements to sections 6.5.1 and 6.5.2 in the Technical Specifications to implement these additional functions. We consider the addition of this expertise in radioactive waste management to these committees to be appropriate since the licensee will be handling and processing significant quantities of

radioactive wastes. These committees will assure that such activities are properly reviewed and controlled by licensee personnel with appropriate and adequate expertise.

In addition, Appendix B Technical Specifications 5.1, 5.2 and 5.3, which identify the licensee's pre-accident management organization for activities addressed by the Appendix B Technical Specifications (which were imposed for the protection of the environment) are deleted since those requirements will now be performed in accordance with proposed Technical Specifications 6.1, 6.2 and 6.5 which will set forth the current requirements for the licensee's management organization for all licensed activities. The deletion of these Appendix B Technical Specifications does not affect any existing limits on effluent releases and discharges and does not authorize a change in effluent types or amounts nor does it affect the power level of the facility. Furthermore, their deletion would not result in any increase in the probability or consequences of an accident nor will it result in a decrease in a margin of safety since the requirements will in any event be continued in an updated requirement of proposed Technical Specifications 6.1, 6.2 and 6.5 which reflects the current, post-accident requirements for the facility's maintenance. Thus, deletion of Appendix B Technical Specifications 5.1, 5.2 and 5.3 will have no environmental impact or effect on plant safety, and is purely administrative in nature.

10. Summary

The technical specification changes associated with this amendment reflect the changes that are necessary to account for the present condition of the facility and to assure the continued maintenance of the safe, stable condition of the facility in the "Recovery Mode". Certain additional controls and equipment requirements, not required in the pre-accident technical specifications, have been added to provide additional assurance that the facility will be maintained in a safe and stable cold shutdown condition during the present and planned activities for facility recovery from the accident. The technical specifications associated with this amendment include these added controls and equipment requirements.

Except as necessitated by the physical realities that exist due to damage caused by or as a result of the accident, no safety limit, limiting condition for operation or surveillance requirement in the pre-accident technical specifications that is pertinent to the present cold shutdown condition of the facility has been modified, relaxed, or deleted by this amendment.

Although the facility is presently being maintained in accordance with NRC approved procedures, the present plant conditions were not expressly contemplated nor provided for in the facility operating

license; consequently, the current facility operating license does not include any provisions or technical specifications for assuring the continued maintenance of the plant in a safe, stable condition or for providing for foreseeable off-normal conditions. These revised technical specifications explicitly impose such license requirements and thereby provide an increased assurance of plant safety. In addition, by deletion of operating modes other than the Recovery Mode and by the changes to existing Technical Specifications discussed herein, planned operation of the facility in other than the stable shutdown condition of the Recovery Mode is precluded. Based on the above, the public health, safety and interest required that the requirements imposed by the proposed Technical Specifications set forth in Attachment 1 to the Director's Order of this date become effective immediately.

ENVIRONMENTAL ASSESSMENT

The environmental impacts resulting from normal operation of the facility were evaluated by the Staff as set forth in the Final Environmental Statement issued in December 1972 and in the Final Supplement to the Final Environmental Statement issued in December 1976. Although the licensee's authority to operate the facility was suspended by Order for Modification of License dated July 20, 1979, and is now limited to maintenance of the reactor in its current mode, the limits on effluent releases and discharges previously established are not changed by virtue of revised and/or new Technical Specifications being imposed, nor do they authorize a change in effluent types or total amounts nor an increase in power level. Thus, any environmental impacts which are attributable to maintenance of the facility

in accordance with the revised and/or new Technical Specifications will be within, indeed likely substantially less than, the impacts previously evaluated and found acceptable. Furthermore, those license conditions and Technical Specifications (Appendix B) imposed for the protection of the environment upon issuance of the TMI-2 operating license are not being relaxed in any way by these Technical Specifications.

The eight areas affected by the revised and/or new Technical Specifications: Nuclear Safety; Core Cooling, Water Inventory and Reactor Coolant System Pressure Control; Instrumentation; Containment Systems; Fire Detection and Fire Suppression; Electrical Power; Control of Radioactive Materials in Liquid and Gaseous Effluents; and Review and Audit Functions have been revised from the standpoint of safety considerations, as discussed above.

From the environmental standpoint, no reasonable or meaningful alternatives to the provisions of the Technical Specifications have been identified. However, the staff is including Technical Specifications which specifically prohibit certain activities which would otherwise be authorized at a normally operating facility. In particular, the Technical Specifications include prohibitions against the purging or other treatment of the reactor building atmosphere, the discharge or other disposal of water decontaminated by the EPICOR-II system and the treatment and discharge or other disposal of the high-level radioactively contaminated water now in the reactor building, even though such activities might be conducted in full compliance with effluent limitations or Commission regulations currently in effect and applicable to TMI-2. It is possible, as an alternative, that these activities could have been allowed under the same effluent limitations as would apply

in the case of a normally operating facility. However, the Commission has determined that the public interest warrants prohibiting these undertakings pending completion of an environmental review. See Commission's Statement of May 25, 1979 respecting decontamination of radioactively contaminated water, and Commission's Statement of Policy and Notice of Intent to Prepare a Programmatic Environmental Impact Statement, dated November 21, 1979 (44 F.R. 67738). A variety of longer range alternatives associated with the overall decontamination and cleanup of the facility will be addressed in the programmatic environmental impact statement.

The Technical Specifications do not authorize any new releases external to the facility. Consequently, no off-site environmental impacts are anticipated. Onsite maintenance of the facility pursuant to these Technical Specifications similarly does not entail any new releases of effluents nor the exposure of any workers to a radiological environment except as previously evaluated and found acceptable, and, as a result, no change in on-site impacts will result.

For the foregoing reasons, it has been determined that this action is insignificant from the standpoint of environmental impact and that an environmental statement need not be prepared. Accordingly, pursuant to 10 CFR §51.5(c)(1), a negative declaration will be issued.

CONCLUSION

The changes in technical specifications authorized in connection with this evaluation result in enhancement of safety under present conditions, as

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

References:

1. NUREG-0557, "Evaluation of Long-Term Post-Accident Core Cooling of Three Mile Island Unit 2," NRC Staff Report, May 1979.

2. Memorandum for R. Vollmer from A. Ignatonis, "TMI-2 Plant Modifications for Cold Shutdown, Revision 2," June 8, 1979.

3. Letter to R. Vollmer, NRC, from R. C. Arnold, Metropolitan Edison Company, "Reactor Containment Building Atmosphere Cleanup", November 13, 1979.

NUREG-0432

THREE MILE ISLAND NUCLEAR STATION UNIT 2

Technical Specifications

Appendix "A"

to

License No. DPR-73

INTRODUCTION

On March 28, 1979, an accident occurred at the facility which resulted in damage to the reactor core and to certain reactor systems and components. The reactor is being maintained in a stable cooling mode removing decay heat by natural convection circulation of primary coolant through the core and the "A" steam generator. The "A" steam generator is producing steam which is condensed in the condenser and recirculated to the "A" steam generator. An alternate means of removing decay heat from the primary coolant is through the "B" steam generator. The steam side of the "B" steam generator has been modified to provide a water solid, closed loop cooling system which is in turn cooled by the secondary services closed cooling water system. Either steam generator cooling mode is adequate to remove decay heat from the primary coolant.

As a result of the core and equipment damage, various requirements set forth in the Technical Specifications governing operations are no longer appropriate. For example, certain equipment required to be operable is no longer operable as defined in the technical specifications. Other systems not generally relied on for safe shutdown of the reactor are being used to maintain the facility in the stable mode of heat removal. High radioactivity in containment and in the reactor coolant and radioactivity in certain auxiliary areas have limited access to certain components or have limited the ability of certain systems or components in their original design mode. A number of systems and components have been modified in order to respond to the initial emergency condition.

These revised Technical Specifications are to provide assurance that the facility in its present status will provide adequate safety while certain additional work activities are carried out at the facility during the forthcoming period of core cooling.

In addition to continuing to maintain core cooling, certain work is planned for the period during which these revised specifications will be applicable.

(1) Systems and components presently functioning to maintain core cooling may be modified to enhance their operability and reliability, and new systems may be added to maintain recovery functions.

(2) Presently non-functioning systems or components useful to enhance safety are being restored to operable status or modified to function appropriately.

(3) Activities to maintain control over radioactive effluents from the facility will continue with modifications as necessary to enhance such control.

(4) Decontamination and cleanup of areas to reduce exposure and to control radioactive materials will continue.

These activities and modifications will be performed during the period in which these revised specifications are effective.

The revised Technical Specifications have been prepared to cover the functional capability or performance levels of systems or components required for the safe operation of the facility. The revised Technical Specifications shall however, in no way modify the obligation of the licensee to take all appropriate steps needed to assure public health and safety in connection with the facility at all times; and shall not be deemed to prohibit or restrict the licensee from taking any emergency action necessary to protect the health and safety of the public in the event of any emergency conditions that may arise during the period governed by these revised Technical Specifications.

Detailed surveillance requirements for implementing these specifications are set forth in the Recovery Operations Plan. The specific plant operating procedures which are subject to review and approval by the NRC staff are identified in Section 6.8 of the Technical Specifications.

These revised Technical Specifications will remain in effect until superseded, and may be supplemented, amended or modified from time to time.

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ADMINISTRATIVE CONTROLS

SECTION 1.0

DEFINITIONS

1.0 DEFINITIONS

DEFINED TERMS

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

RECOVERY OPERATIONS PLAN

1.2 The RECOVERY OPERATIONS PLAN shall describe unit Operations Requirements for the implementation of these Technical Specifications. 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 280°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 are closed by automatic valves, manual valves, blind flanges, or deactivated automatic valves secured in their closed positions, except those penetrations required open per approved procedures.

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.

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

REACTOR COOLANT SYSTEM PRESSURE

2.1.3 The Reactor Coolant System pressure shall not exceed 2750 psig.

APPLICABILITY: RECOVERY MODE.

ACTION:

Whenever the Reactor Coolant System pressure has exceeded 2750 psig, reduce the Reactor Coolant System pressure to within its limit.

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 restriction of this Safety Limit protects the integrity of the Reactor Coolant System from overpressurization and thereby prevents the release of radionuclides contained in the reactor coolant from reaching the containment atmosphere.

The reactor pressure vessel and pressurizer are designed to Section III of the ASME Boiler and Pressure Vessel Code which permits a maximum transient pressure of 110%, 2750 psig, of design pressure. The Reactor Coolant System piping, valves and fittings, are designed to ANSI B 31.7, 2/68 Edition. Reactor Coolant System valves are designed to ANSI B 16.5-1963, MSSP-61 and MSSP-66. The maximum transient pressure for the Reactor Coolant System valves is permitted by ASME to be 110%, 2750 psig, of design pressure. The Safety Limit of 2750 psig is therefore consistent with the design criteria and associated code requirements.

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

BORON 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:

1. One OPERABLE makeup pump.
2. One OPERABLE decay heat removal pump.
3. An OPERABLE flow path from the boric acid storage system and from the BWST. The boric acid storage system shall contain borated water in accordance with Figure 3.1-1 at a minimum temperature of 105°F. The BWST shall contain at least 100,000 gallons of borated water at a minimum temperature of 50°F and at a boron concentration of between 3000 and 4500 ppm.

b. The second system comprised of the Standby Reactor Coolant System Pressure Control System.

APPLICABILITY: When fuel is in the reactor pressure vessel.

ACTION:

With one of the above required systems inoperable, restore the inoperable system to OPERABLE status within 72 hours.

BORON CONCENTRATION

3.1.1.2 The reactor coolant shall be maintained at a boron concentration of between 3000 and 4500 ppm and at a temperature above 50°F.

APPLICABILITY: When fuel is in the reactor pressure vessel.

ACTION

None except as provided in Specification 3.0.3.

*Both systems shall be considered OPERABLE when aligned per procedures approved pursuant to Specification 6.8.2.

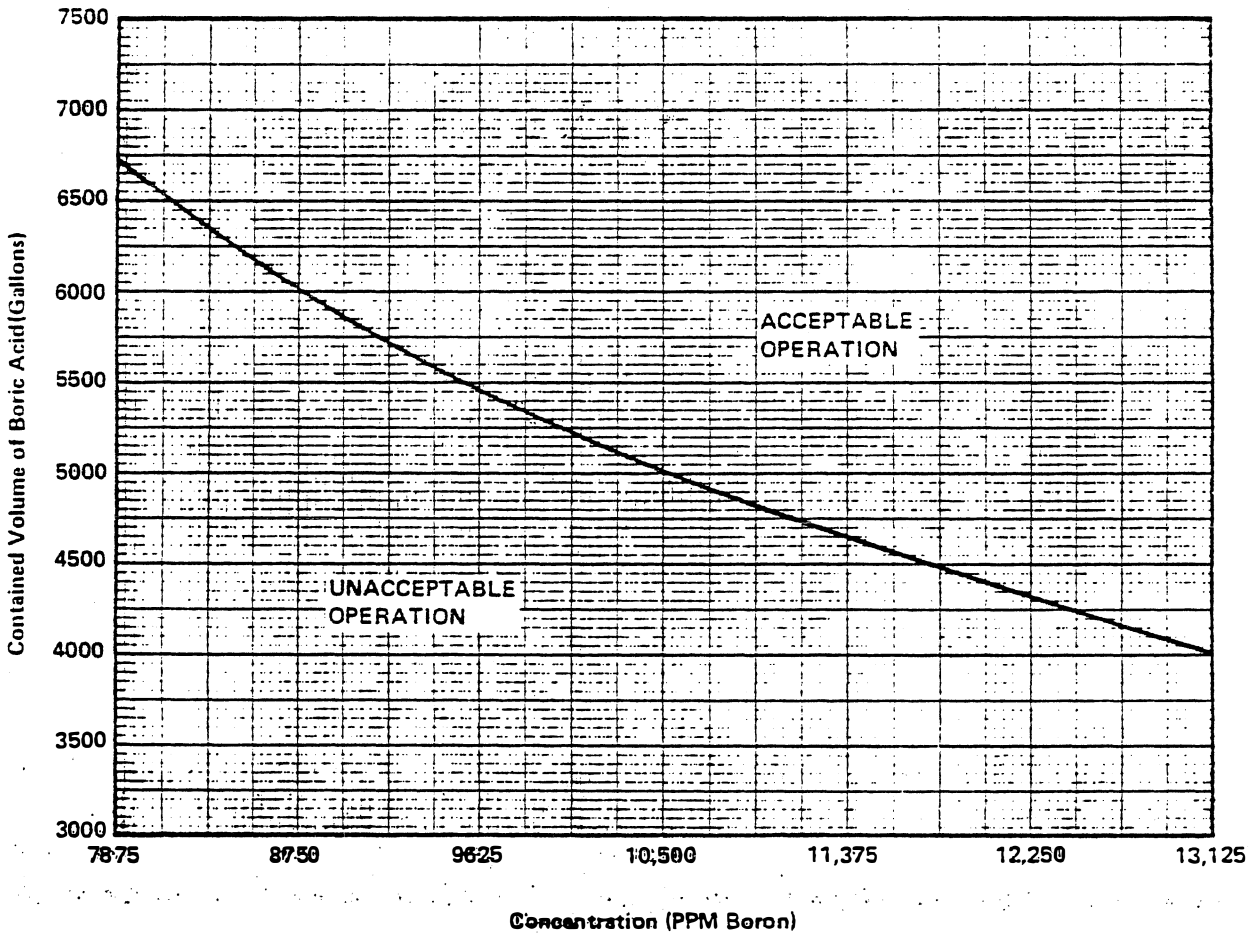


Figure 3.1-1 Minimum Boric Acid Tank Contained Volume as a Function of Stored Boric Acid Concentration

3.1.3 CONTROL ASSEMBLIES

ROD POSITIONS

3.1.3.1 All control rod drive breakers shall be open.

APPLICABILITY: When fuel is in the reactor pressure vessel.

ACTION:

None except as provided in Specification 3.0.3.

3.3 INSTRUMENTATION

3.3.1 NEUTRON MONITORING INSTRUMENTATION

3.3.1.1 As a minimum, the neutron monitoring instrumentation channels of Table 3.3-1 shall be OPERABLE.

APPLICABILITY: When fuel is in the reactor pressure vessel.

ACTION:

With the number of channels OPERABLE one less than required by the Minimum Channels OPERABLE requirement of Table 3.3-1, restore inoperable channel to OPERABLE status within 7 days. With no channels OPERABLE, 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.

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 3.3-3 shall be OPERABLE with their trip setpoints set consistent with the values shown in the Trip Setpoint column of Table 3.3-4.

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 3.3-4, declare the channel inoperable and apply the applicable ACTION requirement of Table 3.3-3 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 3.3-3.

TABLE 3.3-1

NEUTRON MONITORING INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE
1. Intermediate Range, Neutron Flux and Rate	1	0	1
2. Source Range, Neutron Flux and Rate	1	0	1

TABLE 3.3-3

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	ACTION
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	10
2. Emergency Bus #2-3E and 2-4E	2/Bus	1/Bus	2/Bus	11

ACTION 10 - 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 11 - None except as provided in Specification 3.0.3.

TABLE 3.3-4

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

3.3.3 MONITORING INSTRUMENTATION

RADIATION MONITORING INSTRUMENTATION

3.3.3.1 At least one fuel storage pool area gaseous activity monitor, at least one radioactive iodine monitor, and at least one particulate activity monitor shall be OPERABLE.

APPLICABILITY: With liquid radioactive wastes in the pool storage tanks.

ACTION: With any of the above required instrumentation inoperable, suspend all operations involving movement of liquid and gaseous radioactive wastes in the fuel pool area, restore the inoperable equipment to OPERABLE status within 48 hours and provide a portable Constant Air Monitor.

SEISMIC INSTRUMENTATION

3.3.3.3 The seismic monitoring instrumentation shown in Table 3.3-7 shall be OPERABLE.

APPLICABILITY: At all times.

ACTION:

With one or more seismic monitoring instruments inoperable for more than 30 days, prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within the next 10 days outlining the cause of the malfunction and the plans for restoring the instrument(s) to OPERABLE status.

METEOROLOGICAL INSTRUMENTATION

3.3.3.4 The meteorological monitoring instrumentation channels shown in Table 3.3-8 shall be OPERABLE.

APPLICABILITY: At all times.

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

INSTRUMENTS AND SENSOR LOCATIONS	MEASUREMENT RANGE	MINIMUM INSTRUMENT OPERABLE
1. Triaxial Time-History Accelographs		
a. Reactor Building Ring Girder (454' - 8")	± 1 G	1
b. Reactor Building Mat (281'-6")	± 1 G	1
2. Triaxial Peak Accelographs		
a. Reactor Service Structure	± 1 G	1
b. "B" Core Flood Tank Piping	± 1 G	1
c. 2-1E Switchgear	± 1 G	1
3. Triaxial Seismic Switches		
a. Reactor Building Base - Setpoint -	0.25 G	1*
4. Triaxial Response-Spectrum Recorders		
a. Reactor Building Mat (281'-6")	(Hz) 2 to 25.4 (Acceleration) 1.6 to 64	1

*With reactor control room indication

TABLE 3.3-8

METEOROLOGICAL MONITORING INSTRUMENTATION

INSTRUMENT	LOCATION	MINIMUM OPERABLE
1. WIND SPEED		
a.	Nominal Elev. 100 ft.	1
2. WIND DIRECTION		
a.	Nominal Elev. 100 ft.	1
3. AIR TEMPERATURE - DELTA T		
a.	Nominal Elev. 33 ft.	1
b.	Nominal Elev. 150 ft.	1

REMOTE SHUTDOWN INSTRUMENTATION

3.3.3.5 The remote shutdown monitoring instrumentation channels shown in Table 3.3-9 shall be OPERABLE with readouts displayed external to the control room.

APPLICABILITY: RECOVERY MODE.

ACTION:

With the number of OPERABLE remote shutdown monitoring channels less than required by Table 3.3-9, restore the inoperable channel to OPERABLE status within 30 days.

POST-ACCIDENT INSTRUMENTATION

3.3.3.6 The post-accident monitoring instrumentation channels shown in Table 3.3-10 shall be OPERABLE.

APPLICABILITY: RECOVERY MODE.

ACTION:

With the number of OPERABLE post-accident monitoring channels less than required by Table 3.3-10, restore the inoperable channel(s) to OPERABLE status within 8 hours.

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.

TABLE 3.3-9

REMOTE SHUTDOWN MONITORING INSTRUMENTATION

DISPLAY METERS INSTRUMENT	READOUT LOCATION	MEASUREMENT RANGE	MINIMUM OPERABLE CHANNELS
1. RC Inlet Temp	Cab. 217	50-650°F	1/Loop
2. Makeup Tank Level	Cab. 217	0-100 inches	1
DISPLAY INDICATIONS			
3. Control Rod Drive Trip Breakers	Cable Room	open/closed	1/Breaker

DISPLAY PATCH POINTS INSTRUMENT	PATCH POINT LOCATION	MEASUREMENT RANGE	MINIMUM OPERABLE CHANNELS
4. NI Intermediate Range Level Log N	Cab. 217	10^{-11} to 10^{-3} Amps	1
5. NI Source Range Level	Cab. 217	10^{-1} to 10^{6} CPS	1
6. RC Flow	Cab. 217	0 to 180 x 10^{6} LBS/HR	1/Loop
7. Decay Heat Removal Flow	Cab. 217	0-5000 GPM	1/Loop
8. Steam Generator Pressure A/B	Cab. 217	0-1200 psig	1/St. Gen

TABLE 3.3-10

POST-ACCIDENT MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE
1. Reactor Building Pressure	2
2. Core Flood Tank Level	1/Tank
3. Reactor Coolant Outlet Temperature	2
4. Steam Generator Level	1/steam generator*
5. Borated Water Storage Tank Level	1
6. High Pressure Injection Flow	1/loop
7. Low Pressure Injection Flow	1/loop
8. Reactor Building Spray Pump Flow	1
9. Steam Generator Pressure	1/steam generator
10. Incore Thermocouples	All available*
11. Reactor Coolant System Pressure	1
12. Reactor Building Water Level	1

*Report all failures pursuant to Specification 6.9.1.8.

FIRE DETECTION

3.3.3.8 As a minimum, the fire detection instrumentation for each fire detection zone shown in Table 3.3-11 shall be OPERABLE.

APPLICABILITY: At all times.

ACTION:

With the number of OPERABLE fire detection instruments less than required by Table 3.3-11, insure that an alternate instrument with the same coverage is OPERABLE, or;

1. Within 1 hour, establish a fire watch patrol, except in areas inaccessible due to occupational exposure considerations, 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 3.3-11

FIRE DETECTION INSTRUMENTS

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

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. The reactor coolant pumps shall be maintained in an OPERABLE standby status.

APPLICABILITY: RECOVERY MODE.

ACTION:

None except as provided in Specification 3.0.3.

SAFETY VALVES

3.4.3 All pressurizer code safety valves shall be OPERABLE with a lift setting of 2435 PSIG ± 1%*.

APPLICABILITY: RECOVERY MODE.

ACTION:

None except as provided in Specification 3.0.3.

3.4.9 PRESSURE/TEMPERATURE LIMITS

REACTOR COOLANT SYSTEM

3.4.9.1 The Reactor Coolant System shall be maintained at a T_{avg} of less than 280°F and at a pressure of less than 600 psig.

APPLICABILITY: When fuel is in the reactor pressure vessel.

ACTION:

None except as provided in Specification 3.0.3.

*The lift setting pressure shall correspond to ambient conditions of the valve at nominal operating temperature and pressure.

3.6 CONTAINMENT SYSTEMS

3.6.1 PRIMARY CONTAINMENT

CONTAINMENT INTEGRITY

3.6.1.1 Primary CONTAINMENT INTEGRITY shall be maintained and all containment penetrations, including at least two OPERABLE containment isolation valves or a double barrier in each penetration, shall be closed when not required open per procedures approved pursuant to Specification 6.8.2. Purging or other treatment of the containment atmosphere shall be prohibited until approved by the NRC.

APPLICAILITY: RECOVERY MODE.

ACTION:

With 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, and

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.

b. An overall air lock leakage rate of less than or equal to 0.05 L_a at P_a, 56.2 psig.

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 between 0 psig and the minimum allowable pressure as determined from Figure 3.6-1.

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 not exceed 130°F.

APPLICABILITY: RECOVERY MODE.

ACTION:

With the containment average air temperature greater than 130°F, reduce the average air temperature to within the limit within 24 hours.

3.6.2 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 A containment hydrogen purge cleanup system shall be OPERABLE.

APPLICABILITY: During purging of the Reactor Building.

ACTION:

With the containment hydrogen purge cleanup system inoperable, suspend containment purging.

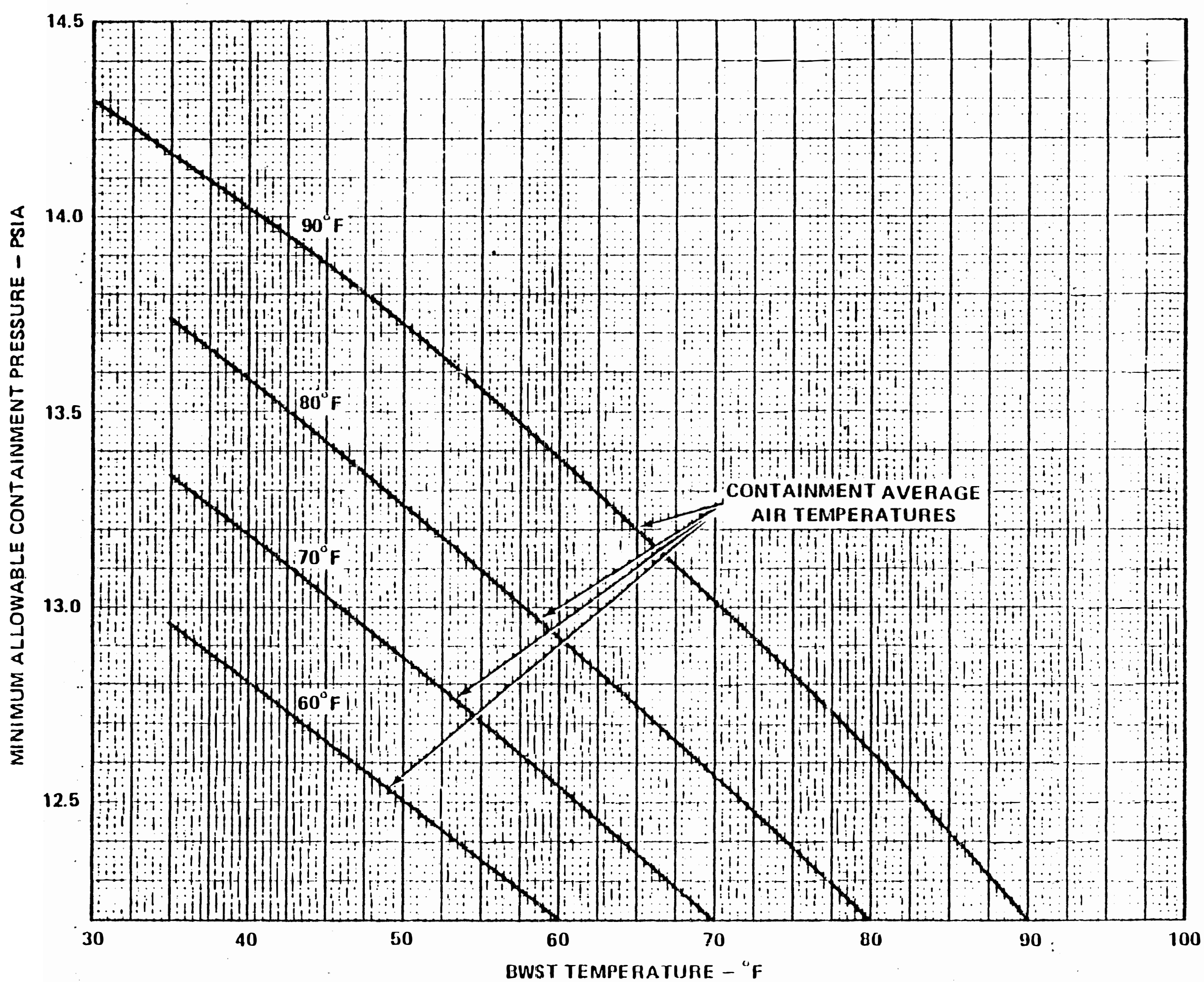


Figure 3.6-1
Minimum Allowable Containment Pressure as a Function of BWST Temperature and Containment Average Air Temperature

3.7 PLANT SYSTEMS

3.7.1 FEEDWATER SYSTEM

3.7.1 The steam generator condensate and feedwater system shall be operated in accordance with procedures approved pursuant to Specification 6.8.2. Two motor-driven emergency feedwater pumps (both capable of being powered from OPERABLE emergency busses) and associated flow paths shall be OPERABLE. The long term "B" steam generator cooling system shall be maintained in an OPERABLE standby status when not being used.

APPLICABILITY: RECOVERY MODE.

ACTION:

With one emergency feedwater system inoperable, restore the inoperable system to OPERABLE status within 72 hours.

3.7.2 SECONDARY SERVICES CLOSED COOLING WATER SYSTEM

3.7.2.1 At least two independent secondary services closed cooling water system 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 secondary services closed cooling water pump or only one secondary services heat exchanger OPERABLE, restore the inoperable pump or heat exchanger to OPERABLE status within 72 hours.

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 within 24 hours.

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 ft. 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 Unit Superintendent and as directed by him:

      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 flotation, and

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

b. With the water level at the Unit 1 Intake Structure exceeding 301 ft. and approaching 302 ft. 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 Unit Superintendent.

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.

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.

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 3 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 Specifications 3.7.10.2 and 3.7.10.4.

APPLICABILITY: At all times.

ACTION:

a. With two pumps or three 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 inoperabiilty 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**

APPLICABILITY: At all times.

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.

* Except in the areas inaccessible per occupational exposure considerations.

**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.

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

##Supply line 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: At all times.

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 3.7-4 shall be OPERABLE:

APPLICABILITY: At all times.

ACTION:

With one or more of the fire hose stations shown in Table 3.7-4 inoperable, route an additional equivalent capacity fire hose to the unprotected area(s) except areas inaccessible due to occupational exposure considerations, 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

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. Fire hose near west stairway 282' elevation.
2. Fire hose near southeast stairway 282' elevation.
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 Handling 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, burning, grinding, etc.

PENETRATION FIRE BARRIERS

3.7.11 All penetration fire barriers protecting safety related areas shall be functional.

APPLICABILITY: At all times.

ACTION:

With one or more of the above required penetration fire barriers non-functional, 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.

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.

c. Two separate and independent Balance of Plant diesel generators each with:
   1. A separate engine mounted day fuel tank containing a minimum volume of 200 (Gray diesel generator)/95 (White diesel generator) gallons of fuel.
   2. A separate fuel storage system containing a minimum volume of 3500 gallons of fuel.
   3. An onsite fuel supply of at least 20,000 gallons.
   4. A separate fuel transfer pump.
   5. A separate 125-volt battery bank.

d. One 13.2 kv circuit from the Middletown Junction Substation.

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, 4.8.1.1.2.a.4, 4.8.1.1.2.a.6 and 4.8.1.1.4.a 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 72 hours.

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, 4.8.1.1.2.a.4, 4.8.1.1.2.a.6 and 4.8.1.1.4.a 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 72 hours 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	c_1	c_2	d
a_1	X	*	*	*	*	*	*
a_2	*	X	*	*	*	*	*
b_1	*	*	X	*	#	#	#
b_2	*	*	*	X	#	#	#
c_1	*	*	#	#	X	*	#
c_2	*	*	#	#	*	X	#
d	*	*	#	#	#	#	X

Key: *Within 4 hours and every 12 hours thereafter
#Within 8 hours and every 36 hours thereafter
a_1 Offsite power circuit No. 1
a_2 Offsite power circuit No. 2
b_1 Class 1E diesel generator (Red)
b_2 Class 1E diesel generator (Green)
c_1 Balance of Plant diesel generator (Gray)
c_2 Balance of Plant diesel generator (White)
d 13.2 kv line from Middletown Junction Substation

TABLE 3.8-2

RESTORATION TIME MATRIX

Combination of Two Inoperable Components		Restore One Component (Hours)	Restore Other Component (Hours)
	aa	24	72
	ab	12	72
	ac	12	72
	ad	12	72
	bb	12	72
	bc	12	72
	bd	12	72
	cc	12	72
	cd	12	72

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

3.8.2 ONSITE POWER DISTRIBUTION SYSTEMS

A.C. DISTRIBUTION

3.8.2.1 The following A.C. electrical busses 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:

4160	volt Emergency Bus # 2-1E and 2-3E
4160	Volt Emergency Bus # 2-2E and 2-4E
4160	Volt Busses # 2-3, 2-4, and 2-5
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 # 2-31, 2-41, 2-32, 2-42, 2-35, 2-45, 2-36, 2-46, 2-38 and 2-48
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

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 The fuel handling building/auxiliary building air cleanup systems shall be OPERABLE with exhaust ventilation flow through the HEPA filters during system operation.

APPLICABILITY: At all times.

ACTION:

a. With one fuel handling building/auxiliary building air cleanup system inoperable, restore the inoperable system to OPERABLE status within 7 days.

b. With no fuel handling building/auxiliary building air cleanup system OPERABLE, suspend all operations involving movement of liquid and gaseous radioactive wastes until at least one fuel handling building/auxiliary building air cleanup system is restored to OPERABLE status.

EPICOR II PROCESSED WATER

3.9.13 Discharge of water processed by the EPICOR II system shall be prohibited until approved by the NRC. Water processed by the EPICOR II system shall be discharged in accordance with procedures approved pursuant to Specification 6.8.2.

APPLICABILITY: At all times.

ACTION:

None except as provided in Specification 3.0.3.

REACTOR BUILDING SUMP WATER

3.9.14 Processing and discharge of water in the Reactor Building sump and Reactor Coolant System shall be prohibited until approved by the NRC. Water in the Reactor Building sump and Reactor Coolant System shall be processed and discharged in accordance with procedures approved pursuant to Specification 6.8.2.

APPLICABILITY: At all times.

ACTION:

None except as provided in Specification 3.0.3.

4 SURVEILLANCE REQUIREMENTS

4.0 APPLICABILITY

SURVEILLANCE REQUIREMENTS

4.0.1 Surveillance Requirements shall be applicable during the RECOVERY MODE or other conditions specified for individual Limiting Conditions for Operation unless otherwise stated in an individual Surveillance Requirements. The Surveillance Requirements shall be performed to demonstrate compliance 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 Requirements 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 consecutive 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

AND

SURVEILLANCE REQUIREMENTS

NOTE

The summary statements contained in this section provide the bases for the specifications of Sections 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.

BASES

3/4.1.1 BORATION CONTROL

The limitation on minimum boron concentration ensures that the core will remain subcritical under all possible 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 cool the core via low pressure injection for at least three days, during which time alternate cooling means could be provided.

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 requires that all control rods remain stationary thereby ensuring that the core's reactivity status will not be changed due to control rod movement. Maintaining the control rod drive breakers open provides assurance that the rods will not be inadvertently withdrawn.

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 the core and to ensure that the core remains subcritical.

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

The OPERABILITY of the seismic instrumentation ensures that sufficient capability is available to promptly determine the magnitude of a seismic event so that the response of those features important to safety may be evaluated. This capability is required to permit comparison of the measured response to that used in the design basis for the facility. This instrumentation is consistent with the recommendations of Regulatory Guide 1.12, "Instrumentation for Earthquakes," April 1974. Due to the high radiation levels in certain areas of the facility, the surveillance requirements for these and various monitoring instruments include provisions excluding inaccessible instruments from the required surveillance activities.

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 REMOTE SHUTDOWN INSTRUMENTATION

The OPERABILITY of the remote shutdown instrumentation ensures that sufficient capability is available to permit monitoring of the facility from locations outside of the control room. This capability is required in the event control room habitability is lost and is consistent with General Design Criterion 19 of Appendix A, 10 CFR 50.

3/4.3.3.6 POST-ACCIDENT INSTRUMENTATION

The OPERABILITY of the post-accident instrumentation ensures that sufficient information is available on selected plant parameters to monitor and assess these variables following the incident of March 28, 1979.

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

Operation of a reactor coolant loop in natural circulation with heat rejection through at least one steam generator provides adequate cooling of the core in its present condition. To the extent possible, the reactor coolant pumps are being maintained OPERABLE in a standby basis. In the event this present mode of cooling the core is lost, procedures have been prepared and approved which will provide alternate means for cooling the core.

3/4.4.3 SAFETY VALVES

The pressurizer code safety valves operate to prevent the RCS from being pressurized above its Safety Limit of 2750 psig. Each safety valve is designed to relieve 348,072 lbs per hour of saturated steam at the valve's setpoint.

3/4.4.9 PRESSURE/TEMPERATURE LIMIT

The RCS pressure and temperature will be controlled in accordance with approved procedures to prevent a nonductile failure of the RCS while at the same time permitting the RCS pressure to be maintained at a sufficiently high value to permit operation of the reactor coolant pumps.

3/4.6.1 PRIMARY CONTAINMENT

3/4.6.1.1 CONTAINMENT INTEGRITY

Primary CONTAINMENT INTEGRITY must be maintained to ensure that the radioactive materials which were released to the containment atmosphere during the March 28, 1979 incident will not be released to the atmosphere.

On November 21, 1979, the Commission issued a Statement of Policy and Notice of Intent to Prepare a Programmatic Environmental Impact Statement addressing the overall decontamination and cleanup activities at TMI-2, including, among other things, the purging or other treatment of the containment atmosphere (44 F.R. 67738). The Commission recognized in its Statement that it may be in the best interest of the public health and safety to purge the containment during the development of the programmatic statement. If the need to take such prompt action arises the Commission has stated that it will consider the advice of the Council on Environmental Quality (CEQ) as to the Commission's NEPA sponsibilities, and moreover, will not take such action until the activity has undergone an environmental review with opportunity for public comment. The Commission further 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 CEQ to the extent practicable. Accordingly, the purging or other treatment of the containment atmosphere is prohibited until approved by the NRC consistent with the foregoing.

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 future entries into the containment to ensure that the containment atmosphere will not be released to the environs.

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 dose considerations.

3/4.6 CONTAINMENT SYSTEMS

BASES

3/4.6.1.5 AIR TEMPERATURE

The average air temperature of the containment atmosphere is currently being maintained well below the 130°F limit. The action will maximize the service life of the instrumentation and equipment installed in the containment. Continued OPERABILITY of these items is required to continue monitoring and mitigating the March 28, 1979 incident.

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.

The hydrogen purge cleanup system will be used to filter the containment atmosphere through HEPA filters if purging of the containment is approved. This filtering action will ensure that releases of radioactive materials in gaseous effluents from the containment will be minimized.

3/4.7.1 FEEDWATER SYSTEM

The emergency feedwater system is required to be maintained in an OPERABLE status since it is a backup means of supplying water to the steam generators to remove the decay heat from the reactor coolant system. Decay heat can be removed from the reactor coolant system by either steam generator. The "A" steam generator can perform this cooling function using feedwater supplied by any one of the three condensate pumps. The "B" steam generator can be used for this cooling function by using its newly installed closed loop cooling system.

3/4.7.2 SECONDARY SERVICES CLOSED COOLING WATER SYSTEM

The secondary services closed cooling water system is required to be maintained in an OPERABLE condition since it is used to cool the "B" steam generator closed loop cooling system.

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 reactor coolant pumps and makeup pumps since this system provides cooling water to the motor coolers and bearings of these pumps.

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 DHR system which is being maintained OPERABLE in a backup status for possible core cooling.

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.

3/4.7.10 FIRE SUPPRESSION SYSTEMS

The OPERABILITY of the fire suppression systems ensures that adequate fire suppression capability is available to confine and extinguish fires occuring 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 BARRIER

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

BASES

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 incident.

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 and 3/4.9.14 EPICOR II PROCESSED WATER AND REACTOR BUILDING SUMP 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.

On November 21, 1979, the Commission issued a Statement of Policy and Notice of Intent to Prepare a Programmatic Environmental Impact Statement addressing the overall decontamination and cleanup activities at TMI-2, including, among other things, the discharge of water processed by the EPICOR II system and processing and discharge of water in the Reactor Building Sump and Reactor Coolant System (44 F.R. 67738). The Commission recognized in its Statement that it may be in the best interest of the public health and safety to process and/or discharge water during the development of the programmatic statement. If the need to take such prompt action arises the Commission has stated that it will consider the advice of the Council on Environmental Quality (CEQ) as to the Commission's NEPA sponsibilities, and moreover, will not take such action until the activity has undergone an environmental review with opportunity for public comment. The Commission further 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 CEQ to the extent practicable. Accordingly, the discharge of water processed by the EPICOR II system and processing and discharge of water in the Reactor Building Sump and Reactor Coolant System is prohibited until approved by the NRC consistent with the foregoing.

SECTION 5.0

DESIGN FEATURES

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.

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.

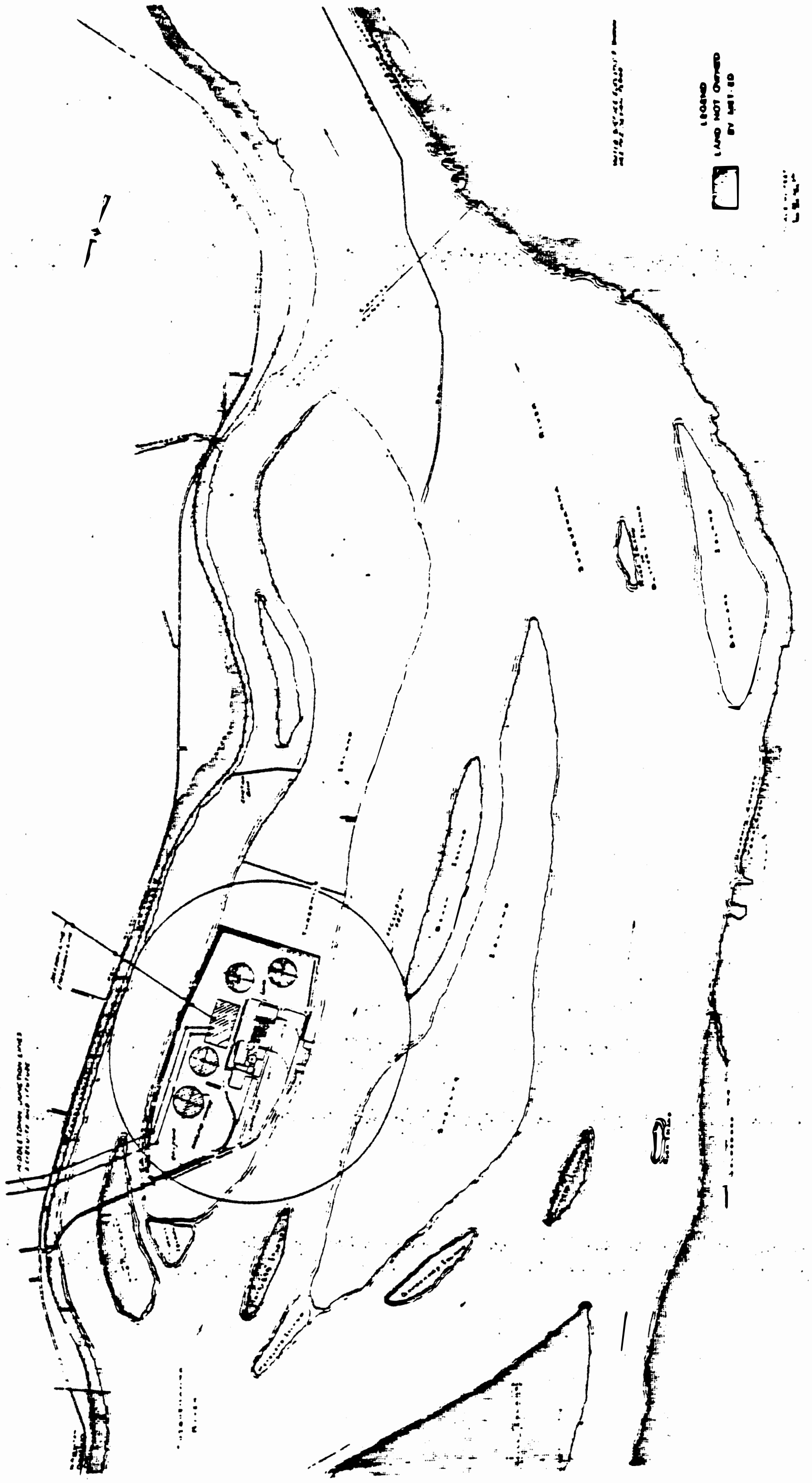
e. Minimum thickness of concrete floor pad = 13.5 feet.

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

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

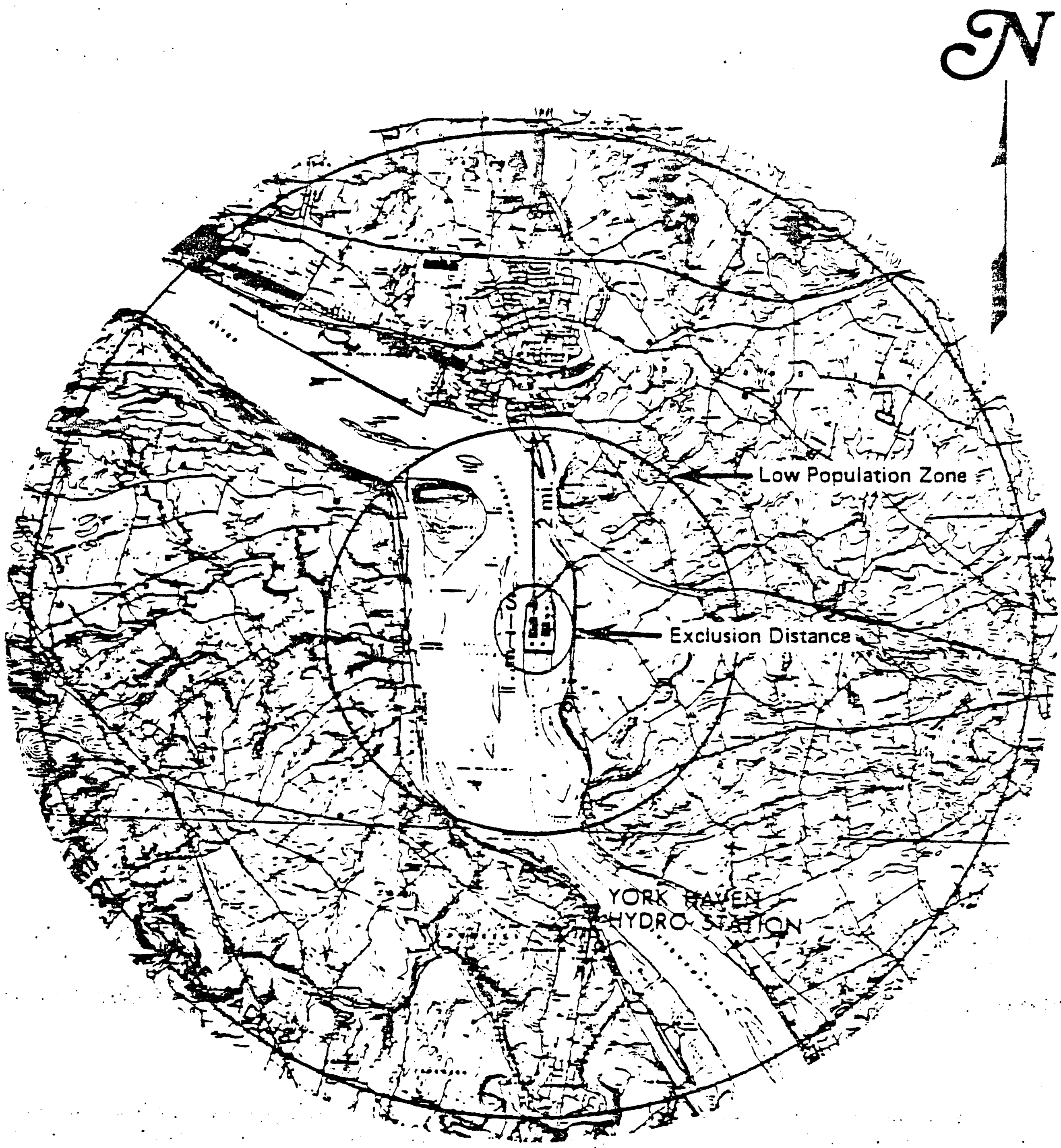
DESIGN PRESSURE AND TEMPERATURE

5.2.2 The reactor containment building is designed and shall be maintained for a maximum internal pressure of 60 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 The reactor coolant system is designed and shall be maintained:

a. In accordance with the code requirements specified in Section 5.2 of the FSAR, with allowance for degradation pursuant to applicable Surveillance Requirements.

b. For a pressure of 2500 psig, and

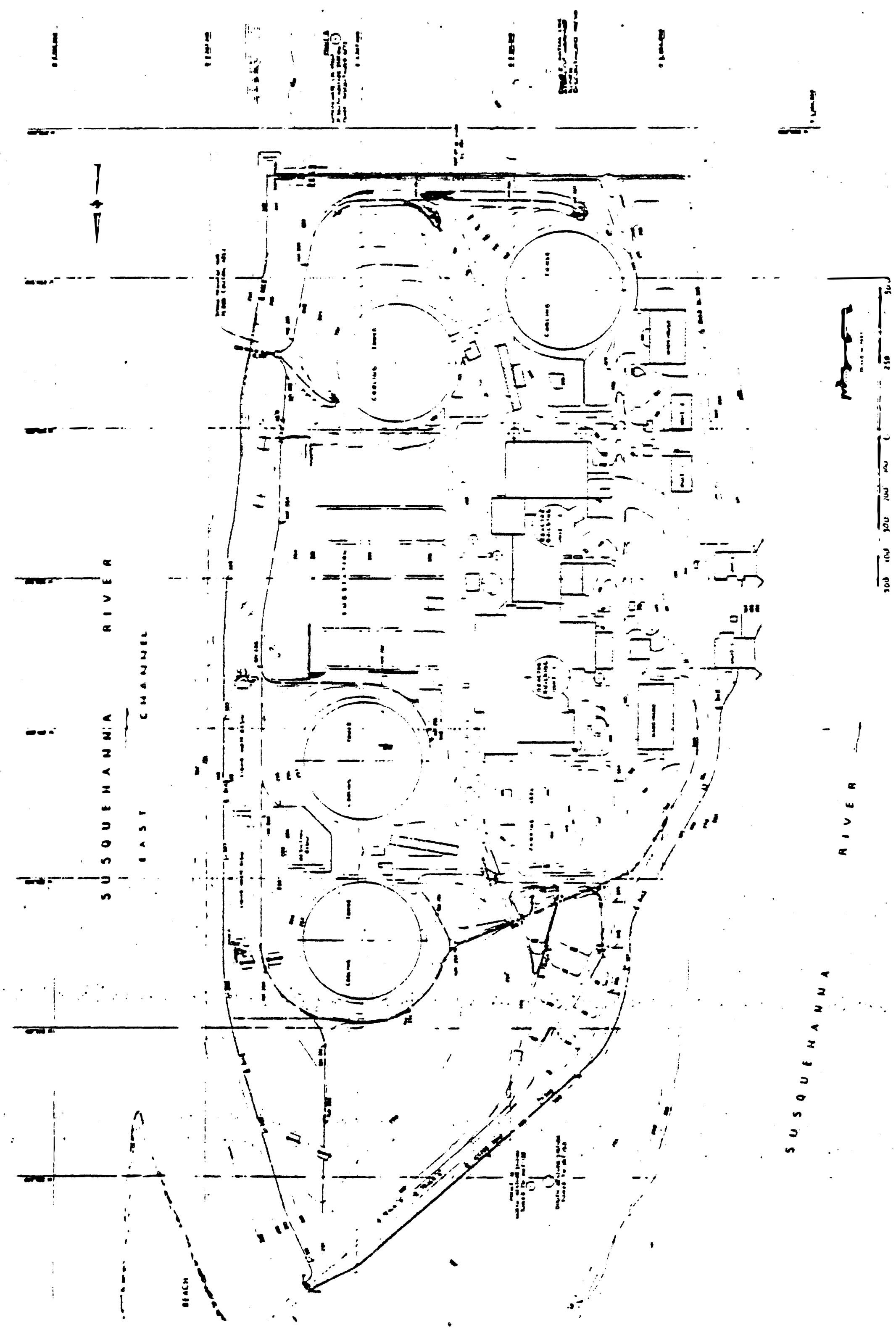
c. For a temperature of 650°F, except for the pressurizer and pressurizer surge line which is 670°F.

VOLUME

5.4.2 The total water and steam volume of the reactor coolant system is 12,180 $\pm$ 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 ADMINISTRATIVE CONTROLS

6.1 RESPONSIBILITY

6.1.1 The Manager Site Operations* shall be responsible for overall unit operation and shall delegate in writing the succession to this responsibility during his absence.

6.2 ORGANIZATION

TMI GENERATION GROUP

6.2.1 The TMI Generation Group 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 as 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.

*This position title is equivalent to the position title "Unit Superintendent" used on procedures dated prior to January 15, 1980.

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.

6.3 UNIT STAFF QUALIFICATIONS

6.3.1 Each member of the unit staff shall meet or exceed the minimum qualifications of ANSI N18.1-1971 for comparable positions, except for the Manager Radiological Controls who shall meet or exceed the qualifications of Regulatory Guide 1.8, September 1975.

6.4 TRAINING

6.4.1 A retraining and replacement training program for the unit staff shall be maintained under the direction of the Supervisor-Station Training and shall meet or exceed the requirements and recommendations of Section 5.5 of ANSI N18.1-1971 and Appendix "A" of 10 CFR Part 55.

6.4.2 A training program for the Fire Brigade shall be maintained under the direction of the Supervisor-Station Training and shall meet or exceed the requirements of Section 27 of the NFPA Code-1975.

6.5 REVIEW AND AUDIT

6.5.1 PLANT OPERATIONS REVIEW COMMITTEE (PORC)

FUNCTION

6.5.1.1 The Plant Operations Review Committee (PORC) shall function to advise the Manager Site Operations on all matters related to nuclear safety and radioactive waste safety.

COMPOSITION

6.5.1.2 The Plant Operations Review Committee shall be composed of the:

a. Chairman - who shall have an academic degree in engineering or physical science field and a minimum of five years of applicable experience.

b. 1 Member - who shall meet or exceed the qualifications of Regulatory Guide 1.8, September 1975.

c. 7 Members - who shall meet or exceed the qualifications of Section 4.4 of ANSI N18.1 - 1971.

The Manager Site Operations shall designate the Chairman and the Vice Chairman from among the members of the Plant Operations Review Committee.

ALTERNATES

6.5.1.3 All alternate members shall be appointed in writing by the PORC Chairman or Vice Chairman to serve on a temporary basis; however, no more than two alternates shall participate as voting members in PORC activities at any one time.

MEETING FREQUENCY

6.5.1.4 The PORC shall meet at least once per calendar month and as convened by the Chairman or his designated alternate.

QUORUM

6.5.1.5 A quorum of the PORC shall consist of the Chairman or his designated alternate and four members, including alternates, one of whom shall meet or exceed the qualifications of Regulatory Guide 1.8, September 1975.

RESPONSIBILITIES

6.5.1.6 The Plant Operations Review Committee shall be responsible for:

a. Review of 1) all procedures required by Specification 6.8 and changes thereto, 2) any other proposed procedures or changes thereto as determined by the Manager Site Operations to affect nuclear safety and radioactive waste safety.

b. Review of all proposed tests and experiments that affect nuclear safety and radioactive waste safety.

c. Review of all proposed changes to Appendix "A" Technical Specifications.

d. Review of all proposed changes or modifications to unit systems or equipment that affect nuclear safety and radioactive waste management.

e. Investigation of all violations of the Technical Specifications including the preparation and forwarding of reports covering evaluation and recommendations to prevent recurrence to the Director TMI-2 and the Generation Review Committee (GRC) Chairman.

f. Review of events requiring 24 hour notification to the Commission.

g. Review of unit operations to detect potential nuclear safety and radiological hazards.

6.5.1.6 RESPONSIBILITY (Continued)

h. Performance of special reviews, investigations or analyses and reports thereon as requested by the Manager Site Operations and/or the Generation Review Committee.

i. Review of the Security Plan and implementing procedures and shall submit recommended changes to the Manager Site Operations and the Generation Review Committee.

j. Review of the Emergency Plan and implementing procedures and shall submit recommended changes to the Manager Site Operations and the Generation Review Committee.

k. Review of the Recovery Operations Plan and implementing procedures and shall submit recommended changes to the Manager Site Operations and the Generation Review Committee.

AUTHORITY

6.5.1.7 The Plant Operations Review Committee shall:

a. Recommend to the Manager Site Operations in writing approval or disapproval of items considered under 6.5.1.6(a) through (d) above.

b. Render determinations in writing with regard to whether or not each item considered under 6.5.1.6(a) through (e) above constitutes an unreviewed safety question.

c. Provide written notification within 24 hours to the Director TMI-2 and the GRC Chairman of disagreement between the PORC and the Manager Site Operations; however, the Manager Site Operations shall have responsibility for resolution of such disagreements pursuant to 6.1.1 above.

RECORDS

6.5.1.8 The Plant Operations Review Committee shall maintain written minutes of each meeting and copies shall be provided to the Director TMI-2 and the GRC Chairman.

6.5.2 GENERATION REVIEW COMMITTEE (GRC)

FUNCTION

6.5.2.1 The Generation Review Committee (GRC) shall function to provide independent review and audit of designated activities in the areas of:

a. Nuclear unit operations

b. Nuclear engineering

c. Chemistry and radiochemistry

d. Metallurgy

e. Instrumentation and control

f. Radiological safety

g. Mechanical and electrical engineering

h. Quality assurance practices

i. Radioactive waste operations

COMPOSITION

6.5.2.2 The GRC shall be composed of at least five members including the Chairman, appointed in writing by the Director Technical Functions. GRC members and alternates may be appointed from within the corporate organization or from external sources.

QUALIFICATIONS

6.5.2.3 Each GRC member and alternate shall have an academic degree in an engineering or physical science field; and in addition, shall have a minimum of five years of technical experience, of which a minimum of three years shall be in one or more of the areas specified in 6.5.2.1.

ALTERNATES

6.5.2.4 All alternate members shall be appointed in writing by the GRC Chairman to serve on a temporary basis; however, no more than two alternates shall participate as voting members in GRC activities at any one time.

CONSULTANTS

6.5.2.5 Consultants shall be utilized as determined by the GRC Chairman to provide expert advice to the GRC.

MEETING FREQUENCY

6.5.2.6 The GRC shall meet at least once per six months.

QUORUM

6.5.2.7 A quorum of GRC shall consist of the Chairman or his designated alternate and at least 4 GRC members including alternates. No more than a minority of the quorum shall have line responsibility for operation of the unit.

REVIEW

6.5.2.8.1 The GRC shall review:

a. The safety evaluations for 1) changes to procedures, equipment or systems and 2) tests or experiments completed under the provision of Section 50.59, 10 CFR, to verify that such actions did not constitute an unreviewed safety question.

b. Violations of codes, regulations, orders, Technical Specifications, license requirements, or of internal procedures or instructions having nuclear safety significance.

c. Significant operating abnormalities or deviations from normal and expected performance of unit equipment that affect nuclear safety and radioactive waste safety.

d. Events requiring 24 hour written notification to the Commission.

e. All recognized indications of an unanticipated deficiency in some aspect of design or operation of structures, systems, or components that could affect nuclear safety and radioactive waste safety.

f. Reports and meeting minutes of the PORC.

6.5.2.8.2 At least one cognizant member of the GRC shall review:

a. Proposed changes to procedures, equipment or systems which involve an unreviewed safety question as defined in Section 50.59, 10 CFR.

b. Proposed tests or experiments which involve an unreviewed safety question as defined in Section 50.59, 10 CFR.

c. Proposed changes to Technical Specifications or this Operating License.

AUDITS

6.5.2.9 Audits of unit activities shall be performed under the cognizance of the GRC. These audits shall encompass:

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

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

c. The results of actions taken to correct deficiencies occurring in unit equipment, structures, systems or method of operation that affect nuclear safety at least once per 6 months.

d. The performance of all activities required by the Operating 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 24 months.

f. The Security Plan and implementing procedures at least once per 24 months.

g. The Radiation Protection Plan and implementing procedures at least once per 6 months.

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

i. An independent fire protection and loss prevention inspection and audit performed at least once per 12 months, utilizing either qualified offsite licensee personnel or an outside fire protection firm.

AUDITS (Continued)

j. An inspection and audit of the fire protection and loss prevention program performed by an outside qualified fire consultant at least once per 36 months.

k. Any other area of unit operation considered appropriate by the GRC or the Manager-Quality Assurance.

AUTHORITY

6.5.2.10 The GRC shall report to and advise the Director Technical Functions on those areas of responsibility specified in Sections 6.5.2.8 and 6.5.2.9.

RECORDS

6.5.2.11 Records of GRC activities shall be prepared, approved and distributed as indicated below:

a. Minutes of each GRC meeting shall be prepared, approved and forwarded to the Director Technical Functions and Senior Vice President Metropolitan Edison within 14 days following each meeting.

b. Reports of reviews encompassed by Section 6.5.2.8 above, shall be prepared, approved and forwarded to the Director Technical Functions and Senior Vice President Metropolitan Edison within 14 days following approval of the review.

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 PORC and a report submitted to the GRC Chairman and the Director TMI-2.

c. Each monthly and quarterly report required pursuant to Specification 6.9.1.10 shall be reviewed by the PORC and a copy shall be submitted to the GRC Chairman and the Director TMI-2.

6.7 SAFETY LIMIT VIOLATION

6.7.1 The following actions shall be taken in the event a Safety Limit is violated:

a. The Safety Limit violation shall be reported to the Commission, the Director TMI-2 and to the GRC Chairman as soon as possible within the first 8 hours following the Safety Limit violation.

b. A Safety Limit Violation Report shall be prepared. The report shall be reviewed by the Plant Operations Review Committee. This report shall describe (1) applicable circumstances preceding the violation, (2) effects of the violation upon unit components, systems, or structures, and (3) corrective action taken to prevent recurrence.

c. The Safety Limit Violation Report shall be submitted to the Commission, the GRC Chairman and the Director TMI-2 within 14 days of the violation.

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.

6.8 PROCEDURES (Continued)

g. RECOVERY MODE implementation. (Specifically RECOVERY MODE procedures which involve a reduction in the margin of safety, including those which:)

1. Directly relate to core cooling.

2. Could cause the magnitude of radiological releases to exceed limits established by the NRC.

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

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

Pending approval by the NRC of written procedures required by 6.8.1 a. through g. above, the licensee shall follow the previously approved procedures.

6.8.2 Each procedure of 6.8.1 above, and changes thereto, shall be reviewed by the Plant Operations Review Committee and approved by the Manager Site Operations prior to implementation and reviewed periodically as set forth in administrative procedures. Each procedure of 6.8.1 b. and g. above, and changes thereto, shall be submitted to the NRC prior to implementation; these procedures, and changes thereto, shall be subject to approval by the NRC prior to implementation. The Manager Site Operations shall have responsibility for determining which procedures are in the 6.8.1 b. and g. category.

6.8.3.1 Temporary changes to procedures of 6.8.1 a., c., d., e. and f. above may be made provided:

a. The intent of the original procedure is not altered.

b. The change is approved by two members of the unit management staff, at least one of whom holds a Senior Reactor Operator's License on the unit affected.

c. The change is documented, reviewed by the Plant Operations Review Committee and approved by the Manager Site Operations within 14 days of implementation.

6.8.3.2 Temporary changes to procedures of 6.8.1 b. and g. may be made provided the provisions of 6.8.3.1 are satisfied and the change is submitted to the NRC for review within 72 hours following approval by the Manager Site Operations.

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 Director of the Regional Office of Inspection and Enforcement 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.

RADIATION SAFETY PROGRAM REPORT

6.9.1.6 A Radiation Safety Program Report shall be prepared describing a management plan and schedule for upgrading the Radiation Safety Program in response to Recommendation 4 of NUREG-0640, December 7, 1979. The initial report and subsequent quarterly status reports describing the progress made in implementing this program shall be submitted to the Director of the

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[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 §20.407 of 10 CFR Part 20.

Regional Office of Inspection and Enforcement. The initial report shall be submitted by February 1, 1980, and the subsequent quarterly status reports shall be submitted no later than the 15th of each month following the calendar month covered by the report.

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.

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 Director of the Regional Office, 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.
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.
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.
i. 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; 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 Director of the Regional Office 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.

REPORTING REQUIREMENTS FOR INCIDENT WHICH OCCURRED ON MARCH 28, 1979

6.9.1.10 A written quarterly update/status report of the incident which occurred on March 28, 1979 and related events which occurred through January 15, 1980, shall be submitted to the Director of the Regional Office no later than the 15th of each quarter (January 15, April 15, July 15, and October 15). These quarterly reports shall include available information concerning the cause of the incident, probable consequences of the incident, planned (short term and preliminary long term) corrective action and a description of continuing activities related to the incident. These quarterly reports shall continue until a final report is issued and these reports shall include the similar type of information described above as well as those interim analysis results and interim evaluation concerning the incident which have become available. The final report shall include a summary of Technical Specification violations which occurred during and after the transient. The final report shall also include a summary of the cause(s) of the incident, a sequence of events which occurred during the transient, and corrective actions (both short term and long term) which were taken as a result of the March 28, 1979 incident. These quarterly reports shall be in lieu of the reports required by 6.9.1.8, 6.9.1.9 and 6.9.2 which would have been required for the interval of March 28, 1979 through January 15, 1980.

SPECIAL REPORTS

6.9.2 Special reports shall be submitted to the Director of the Office of Inspection and Enforcement Regional Office 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 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 Specification 6.8.1.

f. Reports required by 6.9.1.6 and 6.9.1.10.

g. Records of radioactive shipments.

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

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

j. Records and logs of radioactive waste systems operations.

6.10.2 The following records shall be retained for the duration of the Facility Operating License:

a. Records and drawing changes reflecting facility design modifications made to systems and equipment described in the Safety Analysis Report.

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

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

d. Records of reactor tests and experiments.

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

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

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

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

i. Records of meetings of the PORC and the GRC.

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

k. Records of unit radiation and contamination surveys.

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

m. 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.

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SURVEILLANCE REQUIREMENTS

4.1 WATER INJECTION COOLING AND REACTIVITY CONTROL SYSTEMS

4.1.1 BORATION CONTROL

BORON 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 7 days by verifying that the accessible (per occupational exposure considerations) heat traced portion of the flow path from the concentrated boric acid storage system is energized.

b. 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.

c. At least once per 31 days by verifying (per occupational exposure considerations), that on recirculation flow, the makeup pump required by Technical Specification 3.1.1.1 develops a discharge pressure of greater than or equal to 1125 psig and that each pump operates for at least 15 minutes.

d. At least once per 31 days by verifying (per occupational exposure considerations), 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.

e. At least once per 31 days by:

   1. Starting (unless already operating) the boric acid pump from the control room.

   2. Verifying (per occupational exposure considerations) the boric acid pump develops a discharge pressure of greater than or equal to 65 psig and operates for at least 15 minutes.

   3. Verifying that the boric acid pump is aligned to receive electrical power from an OPERABLE bus.

BORON INJECTION (Continued)

f. At least once per 7 days by:

1. Verifying the boron concentration in the concentrated boric acid storage system is between 7875 and 13,125 ppm.

2. Verifying the boron concentration in the BWST is between 3000 and 4500 ppm.

3. Verifying the contained borated water volume of the concentrated boric acid storage system is in accordance with Figure 3.1-1.

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

5. Verifying the concentrated boric acid storage system solution temperature is at least 105°F.

g. 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.

h. 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 600 psig.

2. Isolation valves on the discharge side of the water filled tank nearest the reactor coolant system are open.

3. Nitrogen supply bottles are pressurized to between 1500 and 2400 psig.

i. At least once per 7 days by verifying that the standby reactor coolant system pressure control system water filled tanks, the surge tank, and the degassed water supply tank contain borated water with:

1. A boron concentration of between 3000 and 4500 ppm.

2. A dissolved gas concentration of less than 15 scc/kg of water.

j. 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.

BORON CONCENTRATION

4.1.1.2 The reactor coolant shall be determined to be within its limits by:

a. Determining the boron concentration of the primary coolant to be between 3000 and 4500 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 primary coolant temperature to be greater than 50°F at least once per 12 hours.

4.1.3 CONTROL ASSEMBLIES

ROD POSITIONS

4.1.3.1 The control rod drive breakers shall be verified open at least once per 7 days.

4.3 INSTRUMENTATION

4.3.1 NEUTRON MONITORING INSTRUMENTATION

4.3.1.1 Each 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 (per occupational exposure considerations) 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 fuel storage pool area radiation monitoring instrumentation channel shall be demonstrated OPERABLE (per occupational exposure considerations) by the performance of the CHANNEL CHECK, CHANNEL CALIBRATION and CHANNEL FUNCTIONAL TEST operations at the frequencies shown in Table 4.3-3 and by verifying the alarm/trip setpoints are adjusted in accordance with procedures approved pursuant to Technical Specification 6.8.2 for the gaseous activity monitor, radioactive iodine monitor and particulate activity monitor.

SEISMIC INSTRUMENTATION

4.3.3.3.1 Each of the accessible (per occupational exposure considerations). seismic monitoring instruments shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL CALIBRATION and CHANNEL FUNCTIONAL TEST operations at the frequencies shown in Table 4.3-4.

4.3.3.3.2 Each of the accessible (per occupational exposure considerations) seismic monitoring instruments actuated during a seismic event shall be restored to OPERABLE status within 24 hours and a CHANNEL CALIBRATION performed within 5 days following the seismic event. Data shall be retrieved from actuated instruments and analyzed to determine the magnitude of the vibratory ground motion. A Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 10 days describing the magnitude, frequency spectrum and resultant effect upon facility features important to safety.

TABLE 4.3-1

REACTOR PROTECTION SYSTEM INSTRUMENTATION SURVELLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST
1. Intermediate Range, Neutron Flux and Rate	S	R(6)	M
2. Source Range, Neutron Flux and Rate	S	R(6)	M

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

TABLE 4.3-2

FUNCTIONAL UNIT	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	S	R	M
2. Emergency Bus #2-3E and 2-4E	S	NA	NA

TABLE 4.3-3

RADIATION MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST
1. PROCESS MONITORS			
a. Fuel Storage Pool Area			
i. Gaseous Activity	S	R	M
ii. Particulate Activity	S	R	M
iii. Radioactive Iodine	S	R	M

TABLE 4.3-4

SEIMSIC MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENTS AND SENSOR LOCATIONS	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST
1. Triaxial Time-History Accelerographs			
a. Reactor Building Ring Girder (454'-8")	M*	R	SA
b. Reactor Building Mat (281'-6")	M*	R	SA
2. Triaxial Peak Accelerographs			
a. Reactor Service Structure	R	NA	NA
b. "B" Core Flood Tank Piping	R	NA	NA
c. 2-1E Switchgear	R	NA	NA
3. Triaxial Seismic Switches			
a. Reactor Building Base	M**	R	NA
4. Triaxial Response-Spectrum Recorders			
a. Reactor Building Mat (281'-6")	M	R	NA

*Except seismic trigger
**With reactor control room indication

SURVEILLANCE REQUIREMENTS

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.

REMOTE SHUTDOWN INSTRUMENTATION

4.3.3.5 Each remote shutdown monitoring instrumentation channel shall be demonstrated OPERABLE by performance of the CHANNEL CHECK and CHANNEL CALIBRATION operations at the frequencies shown in Table 4.3-6.

POST-ACCIDENT INSTRUMENTATION

4.3.3.6 Each post-accident monitoring instrumentation channel shall be demonstrated OPERABLE by performance of the CHANNEL CHECK and CHANNEL CALIBRATION operations at the frequencies shown in Table 4.3-10.

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

TABLE 4.3-6

REMOTE SHUTDOWN MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

DISPLAY METERS INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION
1. RC Inlet Temp.	M	R*
2. Make Up Tank Level	M	R*
DISPLAY INDICATIONS		
3. Control Rod Drive Trip Breakers	M	N/A
DISPLAY PATCH POINTS INSTRUMENT	**CHANNEL CHECK**	**CHANNEL CALIBRATION**
4. NI Intermediate Range Level Log N	M	R*
5. NI Source Range Level	M	R*
6. RC Flow	M	R*
7. Decay Heat Removal Flow	M	R*
8. Steam Generator Pressure A/B	M	R*

*Nuclear detectors and al channel components located inside containment and components inaccessible due to occupational exposure considerations may be excluded from CHANNEL CALIBRATION.

TABLE 4.3-10

POST-ACCIDENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION
1. Reactor Building Pressure	S	R*
2. Core Flood Tank Level	S	R*
3. Reactor Coolant Outlet Temperature	S	R*
4. Steam Generator Level	N/A	N/A
5. Borated Water Storage Tank Level	S	R*
6. High Pressure Injection Flow	M**	R*
7. Low Pressure Injection Flow	M**	R*
8. Reactor Building Spray Pump Flow	M**	R*
9. Steam Generator Pressure	S	R*
10. Incore Thermocouples	S	R*
11. Reactor Coolant System Pressure	S	R*
12. Reactor Building Water Level (Heise Gauge)	N/A	SA#

*Neutron detectors and all channel components located inside containment and components inaccessible due to occupational exposure considerations may be excluded from CHANNEL CALIBRATION.

**When applicable systems are in operation.

#Unless an appropriate engineering evaluation demonstrates the instrument to be in calibration.

SURVEILLANCE REQUIREMENTS

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.

4.4 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.

SAFETY VALVES

4.4.3 Not applicable.

4.4.9 PRESSURE/TEMPERATURE LIMITS

REACTOR COOLANT SYSTEM

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

4.4.9.1.2 The hydrogen concentration in the reactor coolant shall be determined to be at least 5 scc/kg of water at least once per 7 days.

4.6 CONTAINMENT SYSTEMS

4.6.1 PRIMARY CONTAINMENT

CONTAINMENT INTEGRITY

4.6.1.1 Primary CONTAINMENT INTEGRITY shall be demonstrated:

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

1. All accessible (per occupational exposure considerations) penetrations not required to be open per approved procedures during RECOVERY MODE are closed by valves, blind flanges, or deactivated automatic valves secured in their positions.

2. The equipment hatch is closed and sealed.

b. 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 6 months by conducting an overall air lock leakage test at P_a, 56.2 psig, and by verifying that the overall air lock leakage rate is within its limit.

INTERNAL PRESSURE

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

*Exemption to Appendix "J" of 10 CFR 50.

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' (2 temperature indications)
b. RB nominal Elev. 330' (2 temperature indications)
c. RB nominal Elev. 305' (2 temperature indications)

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

4.6.4.3 The hydrogen purge cleanup system shall be demonstrated OPERABLE:

a. 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 1000 cfm $\pm$ 10%.

   2. Verifying a system flow rate of 1000 cfm $\pm$ 10% during system operation when tested in accordance with ANSI N510-1975.

b. At least once per 18 months by verifying that the pressure drop across the combined HEPA filters and charcoal adsorber banks is less than 6 inches Water Gauge while operating the system at a flow rate of 1000 cfm $\pm$ 10%.

c. After each complete or partial replacement of 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 N510-1975 while operating the system at a flow rate of 1000 cfm ± 10%.

4.7 PLANT SYSTEMS

4.7.1 FEEDWATER SYSTEM

4.7.1.1 At least two condensate pumps shall be demonstrated OPERABLE at least once per 31 days by:

a. Starting (unless already operating) each pump and verifying each pump operates for at least 15 minutes.

b. Verifying that each accessible (per occupational exposure considerations) valve (manual, power operated or automatic) in the flow path that is not locked, sealed or otherwise secured in position, is in its correct position.

4.7.1.2 Each emergency feedwater system shall be demonstrated OPERABLE:

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

1. Verifying that each motor driven pump starts and operates for at least 15 minutes and that during pump operation: (a) the "A" pump develops a differential pressure of at least 1228 psid at a flow of 112.5 gpm, and (b) the "B" pump develops a differential pressure of at least 1238 psid at a flow of 112.5 gpm.

2. Verifying that each accessible (per occupational exposure considerations) valve (manual, power operated or automatic) in the flow path 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.

4.7.1.3 The "B" steam generator closed loop cooling system shall be demonstrated OPERABLE at least once per 31 days by starting (unless already operating) the pump and verifying a flow rate of at least 2000 gpm when operating in the recirculation mode.

4.7.2 SECONDARY SERVICES CLOSED COOLING WATER SYSTEM

4.7.2.1 The secondary services closed cooling water system shall be demonstrated OPERABLE at least once per 31 days by verifying that each of the three pumps start and operate (unless already operating) for at least 15 minutes.

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 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.

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.

SURVEILLANCE REQUIREMENTS

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 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.

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.

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 Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 2, March 1978.

   3. Verifying a system flow rate of 14,350 cfm ± 10% during system operation when tested in accordance with ANSI N510-1975.

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, meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 2, March 1978.

*The prerequisites of Section 10.3 and 12.3 of ANSI-N510-1975 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 Guage 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 W.G. 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% of the DOP when they are tested in-place in accordance with ANSI N510-1975 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% of a halogenated hydrocarbon refrigerant test gas when they are tested in-place in accordance with ANSI N510-1975 while operating the system at a flow rate of 14,350 cfm ± 10%.

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 $\pm$ 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 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.

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 accessible (per occupational exposure considerations) 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 3.7-4 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).

4.8 ELECTRICAL POWER SYSTEMS

4.8.1 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 each BOP diesel generator starts and accelerates to at least 900 rpm (i.e. 60 Hz on the generator).

   7. Verifying each BOP diesel generator is synchronized; loaded to greater than or equal to 2400 kw, and operates for greater than or equal to 60 minutes.

   8. 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.

c. At least once per 92 days by verifying that the BOP diesels start automatically on a simulated actuation of the undervoltage relays which detect loss of offsite power and accelerate to at least 900 rpm (i.e. 60 Hz on the generator) in less than or equal to 60 seconds and operate for at least 10 minutes.

A.C. SOURCES (Continued)

4.8.1.1.3 Each 125-volt BOP diesel generator battery bank and charger shall be demonstrated OPERABLE:

a. At least once per 7 days by verifying that the overall battery voltage is greater than or equal to 125 volts.

b. At least once per 31 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, is greater than or equal to 1.20.

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

c. 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, corrected to 77°F, 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.

d. 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.

A.C. SOURCES (Continued)

4.8.1.1.4 The 13.2 kv circuit from the Middletown Junction Substation shall be demonstrated OPERABLE:

a. At least once per 24 hours by verifying indicated power availability and that breakers T-56-2, T-78-2, T51-61-2, T71-81-2 and T72-82-2 are open (unless closed in accordance with procedures approved pursuant to Specification 6.8.2).

b. At least once per 7 days by closing the 13.2 kv line onto bus 2-5, opening breaker 2B-52, and operating at least one circulating water pump for between 5 and 10 minutes.

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.

D.C. DISTRIBUTION

4.8.2.3.1 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 RADIOACTIVE WASTE STORAGE

FUEL HANDLING BUILDING/AUXILIARY BUILDING AIR CLEANUP SYSTEMS

4.9.12 The fuel handling building/auxiliary building air cleanup systems shall be demonstrated OPERABLE:

a. 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.

b. Initially and 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 a fuel handling building system flow rate of 46,000 cfm ± 10% during system operation when tested in accordance with ANSI N510-1975.

   2. Verifying an auxiliary building system flow rate of 65,000 cfm ± 10% during system operation when tested in accordance with ANSI N510-1975.

c. At least once per 18 months by:

   1. Verifying that the pressure drop across the combined HEPA filters and charcoal adsorber banks of the fuel handling building system is less than 6 inches Water Gauge while operating the system at a flow rate of 46,000 cfm ± 10%.

   2. Verifying that the pressure drop across the combined HEPA filters and charcoal adsorber banks of the auxiliary building system is less than 6 inches Water Gauge while operating the system at a flow rate of 65,000 cfm ± 10%.

   3. Verifying that the fuel handling building system maintains the fuel handling building at a negative pressure of greater than or equal to 1/8 inch Water Gauge relative to the outside atmosphere during system operation.

   4. Verifying that the auxiliary building system maintains the auxiliary building at a negative pressure of greater than or equal to 1/8 inch Water Gauge relative to the outside atmosphere during system operation.

d. After each complete or partial replacement of a HEPA filter bank in the fuel handling building system by verifying that the HEPA filter banks in this system remove greater than or equal to 99% of the DOP when they are tested in-place in accordance with ANSI N510-1975 while operating the system at a flow rate of 46,000 cfm ± 10%.

e. After each complete or partial replacement of a HEPA filter bank in the auxiliary building system by verifying that the HEPA filter banks in this system remove greater than or equal to 99% of the DOP when they are tested in-place in accordance with ANSI N510-1975 while operating the system at a flow rate of 65,000 cfm ± 10%.