July 17, 1984

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Docket No. 50-320
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Docket No. 50-320

Mr. B. K. Kanga, Director
Three Mile Island Unit 2
GPU Nuclear Corporation
P.O. Box 480
Middletown, PA 17057

Dear Mr. Kanga:

Subject: Three Mile Island Nuclear Station, Unit 2
Operating License No. DPR-73
Docket No. 50-320
Technical Specification Change Requests 39, 41, 43
Recovery Operation Plan Change Requests 19, 20, 22
Request for Exception to 10 CFR 50, Appendix A, Criterion 56 (Containment Penetration Design)
Exemption Request from 10 CFR 50, Appendix A, Criteria 2, 50 and 51 (SSE Requirements for Containment Penetrations)
Approval of Exemption to 10 CFR 50, Appendix A, Criterion 57
Approval of Alternate Design to 10 CFR 50, Appendix A, Criterion 55

The Nuclear Regulatory Commission has issued the enclosed Amendment of Order; Approval of Exemption from the SSE Design Requirements of 10 CFR 2, 50 and 51; Approval of Alternate Design to 10 CFR 50, Appendix A, Criteria 55 and 56; Approval of Exemption from 10 CFR 50, Appendix A, Criterion 57; and Recovery Operations Plan Change Approval.

The Amendment of Order which modifies many sections of the Proposed Technical Specifications was requested by General Public Utilities Nuclear Corporation (GPUNC) in letters dated January 12, 1983, September 12, 1983 and September 30, 1983. Other documents related to this request include: Recovery Operations Plan Changes requested in separate letters also dated January 12, September 12, and September 30, 1983; a request for an exemption from 10 CFR 50, Appendix A, Criterion 56 and a request for exemption from Criteria 2, 50 and 51 in a letter dated April 24, 1984.

The staff has divided your various requests into two separate issuances. This issuance addresses those items that are immediately effective pursuant to 10 CFR 2.204. The justification for this type of issuance is discussed herein. The second issuance, which is addressed under a separate letter, discusses requested changes that are also being issued pursuant to 10 CFR 2.204 but are not immediately effective and allows the licensee to demand a hearing within 20 days from the date of the Notice of Issuance.

OFFICE	TMIPO:NRR	TMIPO:NRR	PD:TMIPO:NRR	ELD	DD:NRR	D:NRR
SURNAME	TCPoindexter;bg	RAWeller	BJSnyder	LChandler	EGCase	HRDenton
DATE	7/16/84	7/16/84	7/16/84	7/16/84	7/ /84	7/ /84

The staff has reviewed your safety evaluations in the above documents and concluded that your requests are acceptable with minor changes as discussed with your staff. The staff has discussed with you your request for exemption from 10 CFR 50, Appendix A, Criterion 56 and concluded that what you are actually seeking is an approval of alternate design. These discussions also revealed that this same alternate design should be applied to the requirements of Criterion 55, Reactor Coolant Pressure Boundary Penetrating Containment and based on this alternate design an exemption should be granted relative to Criterion 57, Closed System Isolation Valves.

As previously stated, the Amendment of Order, Recovery Operations Plan Change Approval, the Approval of Alternate Design for 10 CFR 50, Appendix A, Criteria 55 and 56; the Exemption from 10 CFR 50, Appendix A, Criteria 2, 50, 51, and 57 are effective upon issuance.

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

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

Sincerely,

Original signed by
B. J. Snyder

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

Enclosures:
1. Amendment of Order
2. Exemption from 10 CFR 50, Appendix A, Criteria 2, 50 and 51
3. Approval of Alternate Design to 10 CFR 50, Appendix A, Criteria 55 and 56
4. Exemption from 10 CFR 50, Appendix A, Criterion 57
5. Safety Evaluation
6. Proposed Technical Specification Page Changes
7. Approved Recovery Operations Plan Change No. 20
8. Notice of Environmental Assessment and Finding of No Significant Impact
9. Federal Register Notices

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Enclosure 1

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

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

AMENDMENT OF ORDER

I.

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

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

Although these requirements were imposed on the licensee by an Order of the Director of Nuclear Reactor Regulation, dated February 11, 1980, the TMI-2 license has not been formally amended. The requirements are reflected in the Proposed Technical Specifications presently pending before the Atomic Safety and Licensing Board. Hereafter in this Amendment of Order, the requirements in question are identified by the applicable Proposed Technical Specification.

II.

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

The licensee has requested various changes to the PTS to support anticipated activities until, but not including, defueling and to more properly reflect the facility's post accident mode of operation. As previously stated, changes that are in the interest of the health and safety of the public are being issued immediately effective pursuant to the provisions of 10 CFR 2.204.

Changes herein include, (1) the modification of the definition for Containment Integrity, Section 1.7, to clarify when containment integrity does and does not exist; (2) the addition of a boron concentration limit for water in the RCS and the refueling canal after head lift, Section 3.1.1.2; (3) a modification to Section 3.1.3 on control rod drive assemblies to properly reflect that they are disconnected from the control rods; (4) a modification to the action statement of Section 3.3.1 on Neutron Monitoring Instrumentation to add new reporting requirements because of their inaccessibility while the refueling canal is flooded; (5) the addition of a requirement in Section 3.4.2 for reactor vessel water level monitoring instrumentation; (6) the addition of Section 3.5 on Communications which reflects requirements during core alterations; (7) a modification of Section 3.6.1.1 on containment integrity to clarify what constitutes containment integrity; (8) the insertion of Section 3.6.3 on the Containment Purge Exhaust System which will be used to minimize airborne contamination in the containment building while the RV head is off of the vessel; and (9) the addition of Section 3.10.1 which limits areas of travel for the reactor building polar crane during all heavy load movements when the RV head is off of the vessel.

Associated surveillance requirements of the Recovery Operations Plan and associated bases for the PTS have also been modified accordingly.

Also, the staff has issued in support of the above changes, an Approval of Alternate Design relative to 10 CFR 50, Appendix A, Criteria 55 and 56, an Exemption from 10 CFR 50, Appendix A, Criterion 57 and an Exemption from the Seismic Design requirements of Criteria 2, 50, and 51 of 10 CFR 50, Appendix A. These approvals are required in support of some of the modifications that have been made to the PTS.

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

It is further determined that the Amendment of Order does not authorize a change in effluent types or total amounts nor an increase in power level and will not otherwise result in any significant environmental impact. In light of this determination and as reflected in the Environmental Assessment prepared pursuant to 10 CFR 51.2 and 51.30 through 51.32 issued concurrently herewith, it was concluded that the instant action is insignificant from the standpoint of environmental impact and that an environmental impact statement need not be prepared.

The Nuclear Regulatory Commission has determined that the public health, safety and interest require the enclosed immediately effective modifications to the Proposed Technical Specifications (PTS) for Facility Operating License No. DPR-73 issued to Metropolitan Edison Company, et al. for operation of the Three Mile Island Nuclear station Unit No. 2, located in Londonderry Township, Dauphin County, Pennsylvania. This action would modify the PTS by incorporating or modifying specifications that are required to be in place before the reactor vessel head can be removed. The removal of the reactor vessel head is required, to gain access to the reactor core for defueling. The staff has stated in various documents and in congressional testimony that there will be a risk to the health and safety of the public until the fuel is removed from the vessel. Although the facility is well-monitored and is presently safe, no one can be certain what potential long delays in cleanup portend for the future. Basically our concern is that, in contrast to a normal nuclear facility, we and GPU cannot ascertain what safety margins exist at TMI-2. Delays in cleanup milestones such as head lift increase the risks to the occupational workforce and offsite public due to the increased probability of some unforeseen occurrence. It is, therefore, necessary to promptly commence activities associated with the removal of fuel in the vessel, head lift being the first major activity.

III.

Accordingly, pursuant to the Atomic Energy Act of 1954, as amended, the Director's Order of February 11, 1980, is hereby revised effective immediately to incorporate the deletions, additions, and modifications set forth in Enclosures 6 and 7 hereto. For further details with respect to this action, see (1) Letter to B. J. Snyder, USNRC, from R. C. Arnold, GPUNC, Technical Specification Change Request No. 39, dated January 12, 1983, (2) Letter to B. J. Snyder, USNRC, from R. C. Arnold, GPUNC, Technical Specification Change Request No. 41, dated September 12, 1983, (3) Letter to B. J. Snyder, USNRC, from R. C. Arnold, GPUNC, Technical Specification Change Request No. 43, dated September 30, 1983, (4) Letter to L. H. Barrett, USNRC, from B. K. Kanga, GPUNC, Recovery Operations Plan Change Request No. 19, dated January 12, 1983, (5) Letter to L. H. Barrett, USNRC, from B. K. Kanga, GPUNC, Recovery Operations Plan Change Request No. 20, dated September 12, 1983, (6) Letter to L. H. Barrett, USNRC, from B. K. Kanga, GPUNC, Recovery Operations Plan Change Request No. 22, dated September 30, 1983, (7) Letter to B. J. Snyder, USNRC, from E. E. Kintner, GPUNC, Request for an Exemption to Certain Design Criteria for Containment Penetrations, dated April 24, 1984, (8) Letter to B. J. Snyder, USNRC, from E. E. Kintner, GPUNC, Exemption Request from 10 CFR 50, Appendix A, Criteria 2, 50 and 51, and (9) the Director's Order of February 11, 1980.

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

FOR THE NUCLEAR REGULATORY COMMISSION

Harold R. Denton, Director
Office of Nuclear Reactor Regulation

Effective Date: July 17, 1984
Dated at Bethesda, Maryland
Issuance Date: July 17, 1984

ENCLOSURE 2

UNITED STATES NUCLEAR REGULATORY COMMISSION

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

EXEMPTION

I.

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

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

II.

By letter dated April 24, 1984, the licensee requested exemptions from 10 CFR 50, Appendix A, Criteria 2, 50, 51, and 56 regarding the design of containment penetrations after the removal of the reactor vessel head. Criterion 2 deals with design bases for protection against natural phenomena (i.e., earthquakes, tornados). Criterion 50 relates to designing to withstand pressure and temperature transients associated with loss of coolant accidents. Criterion 51 pertains to fractures of the containment boundary. Criterion 56 is concerned with containment isolation valves and is discussed in the NRC's Approval of Alternate Design issued concurrently herewith.

III.

With respect to Criterion 2 the staff has evaluated the potential offsite dose consequences of a containment isolation valve failure when challenged by natural phenomena. The failure of the penetration by itself does not present a potential hazard unless accompanied by a simultaneous event in the containment building which would cause the release of radioactive material. The staff has evaluated the potential offsite dose consequences of the failure of one or more penetrations coupled with a broad range of accidents in the containment building. Calculations were performed to estimate the offsite dose consequences of various accident scenarios involving breach of non-seismic containment penetrations. The scenarios were selected to be representative of the types and conditions which could occur at TMI-2 during defueling activities. The scenarios were chosen to be at the severe end of the spectrum, i.e., minor reactor building fires or small cracks in the penetrations were not considered.

A representative source term for the offsite dose calculations was developed by the TMIPO and the dose consequences were evaluated by the staff's Radiological Assessment Branch.

With regard to Criterion 50, mechanisms and conditions which could produce temperature and pressure transients during a loss of coolant accident are essentially absent and will remain so during defueling. This is due to the fact that the reactor coolant system will be at atmospheric pressure and temperatures less than 110°F during defueling vs. design temperatures in excess of 600°F and design pressures in excess of 2300 psig for an operating reactor. The staff also has evaluated other potential temperature and pressure producing mechanisms in coincidence with containment penetration failure. These include fires, failure of systems containing pressurized gases (i.e., nitrogen, air), and natural phenomena which cause pressure transients (i.e., tornadoes, hurricanes, storm fronts).

Potential penetration failures associated with the brittle fracture requirements of Criterion 51 are enveloped by the evaluations performed for Criterion 2 and Criterion 50. The analyses performed for Criterion 2 and Criterion 50 included instantaneous total penetration failure in coincidence with various accident scenarios inside the reactor building. Brittle fracture phenomena does not exceed instantaneous total penetration failure.

The staff has evaluated the potential offsite dose consequences for all of the above worst case scenarios. The results of these scenarios show that the worst case offsite dose projections at the exclusion area boundary are within the exposure guidelines of 10 CFR 20.

The effects of a penetration failure and simultaneous seismic event have been analyzed by the staff as stated in the above discussions. The result of these occurrences have been shown to be within 10 CFR 20 guidelines. Therefore the staff concludes that there is no undue risk to the health and safety of the public resulting from a seismic induced penetration failure, and it is the staff's opinion that the licensee's exemption request is justified.

The staff has determined that the post-accident status of the TMI-2 facility presents exceptional circumstances relative to the applicability of the Commission's regulations. Because of the suspension of the licensee's authority to operate the facility in other than the present recovery mode as defined in the proposed technical specifications, certain of the regulations, which are intended to apply to normal operating plants, are simply inappropriate and, more significantly, are unnecessary to protect the public health and safety. Indeed, given the unique status of the plant in terms of primary system temperature and pressure, available fission product inventory, the ability to cool the reactor without forced circulation (loss-to-ambient), and the low decay heat rate, maintenance of the facility with the exemptions granted and the alternate design approved hereby will provide an equivalent level of safety. Furthermore, because of the condition of the plant and the need to proceed with cleanup activities, literal compliance with the regulations from which relief is sought would present an unwarranted impediment.

IV.

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

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

FOR THE NUCLEAR REGULATORY COMMISSION

Harold R. Denton, Director
Office of Nuclear Reactor Regulation

Effective Date: July 17, 1984
Dated at Bethesda, Maryland
Issuance Date: July 17, 1984

Enclosure 3

THREE MILE ISLAND PROGRAM OFFICE

SAFETY EVALUATION FOR THE REVIEW OF

ALTERNATE DESIGN FOR 10 CFR 50, APPENDIX A, CRITERIA 55 AND 56

INTRODUCTION

In a letter dated April 24, 1984, GPUNC requested an exception to certain design criteria for containment penetrations. These criteria are stated in 10 CFR Part 50, Appendix A, Criterion 56. During staff discussions on this request, GPUNC stated that what they actually were seeking was an approval of an alternate penetration design which differs from those suggested in Criterion 56. The staff also had numerous discussions with the licensee relative to the penetration design requirements of Criteria 55 and 57 and concluded that the approval of alternate design should be applicable to Criterion 55 and an exemption should be issued to Criterion 57 (see Exemption to 10 CFR 50, Appendix A, Criterion 57 issued concurrently herewith. In their letter, the licensee also requested an exemption from the seismic design requirements of Criteria 2, 50, and 51. That request is discussed in an Exemption to 10 CFR 50, Appendix A, Criteria 2, 50 and 51 also issued concurrently herewith.

Following the TMI accident, thousands of curies of fission gases and radioactive particulates were released from the fuel to the containment atmosphere. Because of the unique condition of the TMI-2 core and the amount of contamination resulting from the accident, the NRC imposed the requirement to maintain containment integrity to ensure that radionuclides inside the containment would not be released to the environment.

In October 1979, the first of several containment penetrations was modified to probe the containment interior to evaluate the extent of damage and to gather data to begin the cleanup. The penetrations were modified in accordance with NRC approved procedures. The TMI-2 Proposed Technical Specifications also required that penetrations and operations that could affect containment integrity could be modified only by NRC approved procedures.

Since the 1979 accident, fission gases that were released to containment have either decayed or have been purged from the containment. Decontamination activities have also reduced airborne particulate contamination to below maximum permissible concentrations listed in 10 CFR 20, Appendix B, Table 1.

In an evaluation associated with a Modification of Order dated April 9, 1982, the staff concluded that the maximum credible containment building pressure was approximately 2 psig. Calculated offsite doses resulting from a failed penetration in conjunction with a 2 psig driving head and the associated reactor building airborne contamination were well below the limits of 10 CFR 20 and within the scope of impacts assessed in the "Final Programmatic Environmental Impact Statement" dated March 1981.

DISCUSSION AND EVALUATION

Criterion 56 provides guidelines for isolation valve configurations for piping that penetrates containment. Criterion 55 provides guidelines for a reactor coolant pressure boundary that penetrates containment. These guidelines also state that the licensee can propose other containment isolation provisions that may be acceptable on another defined basis. Paragraphs (1) through (4) of Criteria 55 and 56 describe configurations that are preferred by the staff for a normal nuclear plant. They are as follows: (1) one locked closed isolation valve inside and one locked closed isolation valve outside containment; or (2) one automatic isolation valve inside and one locked closed isolation valve outside containment; or (3) one locked closed isolation valve inside and one automatic isolation valve outside containment (a simple check valve may not be used as the automatic isolation valve outside containment); or (4) one automatic isolation valve inside and one automatic isolation valve outside containment (a simple check valve may not be used as the automatic isolation valve outside containment). Criteria 55 and 56 were written for operating plant conditions and are generally applicable whenever the plant is operating, in startup, hot standby, or during core alteration. Presently, the conditions at Unit 2 most closely resemble the standard criteria for cold shutdown ($K_{eff} < 0.99$, $T_{ave} \leq 200°F$). During the normal cold shutdown mode for typical plants, containment integrity is normally not required and Criteria 55 and 56 are not normally applicable.

As previously stated, the staff correlated the shutdown condition of TMI-2 to that of a normal reactor in "cold shutdown." The staff also approved on this basis various penetration designs on the premise that if the plant were to enter a mode that when compared to a normal plant would require containment isolation, either an alternate design or an exemption to Criteria 55 and 56 would have to be approved by the NRC.

The licensee proposed several alternate penetration designs to the NRC staff to support specific recovery operations. The isolation feature common to all of the alternate designs includes two isolation valves outside of containment.

In most cases, isolation valves are manual. Manual valves were found acceptable for isolation since all conceivable accident scenarios still permit access to the isolation valves. Isolation valves in containment as stated in Criteria 55 and 56 have not been required because of difficulties (e.g., high dose rate areas) associated with accessibility for repairs or testing. It is the staff's opinion that the benign status of the reactor did not warrant the increased worker dose which would be incurred during the installation and testing of interior isolation valves. Therefore penetration modifications containing two manual valves outside containment will be acceptable in satisfying Criteria 55 and 56 for all future recovery operations.

ENVIRONMENTAL CONSIDERATIONS

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

CONCLUSION

The staff has therefore concluded that the licensee's proposed penetration configuration is acceptable when considering the present condition and anticipated recovery activities at TMI-2.

We have also concluded, based on the considerations discussed above, that:

(1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and

(2) such activities will be conducted in compliance with the Commission's regulations and the implementation of this change will not be inimical to the common defense and security or to the health and safety of the public.

ENCLOSURE 4

UNITED STATES NUCLEAR REGULATORY COMMISSION

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

EXEMPTION

I.

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

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

II.

By letter dated April 24, 1984, the licensee requested exemption from 10 CFR 50, Appendix A, Criteria 2, 50, 51 and 56 regarding the design of containment penetrations after the removal of the reactor vessel head. Based on subsequent conversations with the licensee, the staff also concluded that an exemption from the requirements of 10 CFR 50, Appendix A, Criterion 57 is also warranted. This criterion states the requirements for closed system isolation valves.

III.

Following the TMI accident thousands of curies of fission gases and radioactive particulates were released to the containment atmosphere. Because of the unique condition of the TMI-2 core and the amount of contamination resulting from the accident, the NRC imposed the requirement to maintain containment integrity to ensure that radionuclides inside the containment would not be released to the environment.

In October 1979, the first of several containment penetrations were modified to probe the containment interior to evaluate the extent of damage and gather data for the cleanup. The penetrations were modified in accordance with NRC approved procedures. The TMI-2 Proposed Technical Specifications also required that penetrations and operations that could affect containment integrity could be modified only by NRC approved procedures.

Since the 1979 accident, fission gases that were released to containment have either decayed or have been purged from the containment. Decontamination activities have also reduced ambient airborne particulate contamination to levels below maximum permissible concentrations listed in 10 CFR Part 20, Appendix B, Table 1.

In an evaluation associated with a Modification of Order dated April 9, 1982, the staff concluded that the maximum credible containment building pressure was approximately 2 psig. Calculated offsite doses resulting from a failed penetration in conjunction with a 2 psig driving head and the associated reactor building airborne contamination were well below the limits of 10 CFR 20 and within the scope of impacts assessed in the "Final Programmatic Environmental Impact Statement" dated March 1981.

Criterion 57 requires that each line penetrating the primary containment that is neither a part of the reactor coolant pressure boundary nor directly connected to the containment atmosphere have at least one containment isolation valve which shall be either automatic, or locked closed or capable of remote manual operation. This valve shall be outside containment and located as close to the containment as is practical. A simple check valve may not be used as an automatic isolation valve. Criterion 57 was written for operating plant conditions and is generally applicable whenever the plant is operating, in startup, hot standby, or during core alteration. Presently, the conditions at Unit 2 most closely resemble the standard criteria for cold shutdown ($K_{eff} < 0.99$, $T_{ave} \leq 200°F$).

During the normal cold shutdown mode for typical plants, containment integrity is normally not required and Criterion 57 is not normally applicable.

As previously stated, the staff correlated the shutdown condition of TMI-2 to that of a normal reactor in "cold shutdown." The staff also approved on this basis various penetration designs on the premise that if the plant were to enter a mode that when compared to a normal plant, would require containment isolation, either an alternate design or an exemption to penetration criteria would have to be approved by the NRC.

The licensee proposed several alternate penetration designs to the NRC staff to support specific recovery operations. The isolation feature common to all the alternate designs includes two isolation valves outside of containment. In most cases, isolation valves are manual. Manual valves were found acceptable for isolation in lieu of the Criterion 57 requirements since all conceivable accident scenarios still permit access to the isolation valves. Therefore, it is the staff's opinion that penetration modifications of the type described above will be acceptable for all future recovery operations.

The staff has determined that the post-accident status of the TMI-2 facility presents exceptional circumstances relative to the applicability of the Commission's regulations. Because of the suspension of the licensee's authority to operate the facility in other than the present recovery mode as defined in the proposed technical specifications, certain of the regulations, which are intended to apply to normal operating plants, are simply inappropriate and, more significantly, are unnecessary to protect the public health and safety. Indeed, given the unique status of the plant in terms of primary system temperature and pressure, available fission product inventory, the ability to cool the reactor without forced circulation (loss-to-ambient), and the low decay heat rate, maintenance of the faciliy with the exemptions granted and the alternate design approved hereby will provide an equivalent level of safety. Furthermore, because of the condition of the plant and the need to proceed with cleanup activities, literal compliance with the regulations from which relief is sought would present an unwarranted impediment.

IV.

Accordingly, the Commission has determined that, pursuant to 10 CFR 50.12, an exemption is authorized by law and will not endanger life or property or the common defense and security and is otherwise in the public interest. The Commission hereby grants an exemption to the requirements of 10 CFR Part 50, Appendix A, Criterion 57.

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

FOR THE NUCLEAR REGULATORY COMMISSION

Harold R. Denton, Director
Office of Nuclear Reactor Regulation

Effective Date: July 17, 1984
Dated at Bethesda, Maryland
Issuance Date: July 17, 1984

Enclosure 5

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

GPU NUCLEAR CORPORATION

METROPOLITAN EDISON COMPANY

PENNSYLVANIA ELECTRIC COMPANY

JERSEY CENTRAL POWER & LIGHT COMPANY

DOCKET NO. 50-320

THREE MILE ISLAND NUCLEAR STATION UNIT NO. 2

INTRODUCTION

By letters dated January 12, 1983, September 12, 1983, and September 30, 1983, GPU Nuclear Corporation (GPUNC) requested the approval of changes and provided supporting information to modify the Proposed Technical Specifications (PTS) of Operating License No. DPR-73. By separate letters dated January 12, 1983, September 12, 1983, and September 30, 1983, GPUNC also requested the approval of changes to the Recovery Operations Plan (ROP) for Three Mile Island, Unit 2 (TMI-2). On April 24, 1984, the licensee requested an Exemption relative to the Containment Penetration Requirements of 10 CFR 50, Appendix A, Criterion 56, and an exemption from Criteria 2, 50, and 51 of Appendix A. These exemptions/approvals were required to support the requested changes to the PTS and the ROP.

DISCUSSION AND EVALUATION

Because of the number of changes, exemptions and approvals requested by the licensee, the staff will discuss each PTS and ROP change in chronological order. Other enclosures discuss the proposed exemptions and other approvals also requested by the licensee.

(A) Section 1.0 Definitions

1.7 Containment Integrity - The staff has modified this section to better state the regulatory interpretation of containment integrity. The modification more clearly states under what conditions containment integrity is required. Previously, the definition for containment integrity implied that integrity could exist when penetrations were open per NRC approved procedures. The definition now clearly states the conditions that must be met for containment integrity to exist. If the stated conditions do not exist then there is no containment integrity. All containment integrity modifications are reviewed and approved by the NRC. Section 3.6.1 of the Limiting Conditions for Operation discusses the requirements further.

1.15 Core Alteration - The licensee has proposed the addition of a definition for Core Alteration. Subsequent to removing the reactor vessel head, activities will be undertaken which involve moving components within the reactor pressure vessel. Since these actions are addressed throughout the text of the PTS, the definition also needed to be added. The definition states that, when the vessel head is removed and there is fuel in the vessel and any reactor component (including fuel) is moved within the reactor pressure vessel, a core alteration exists.

1.16 Loss-to-Ambient - The licensee has proposed the addition of a definition for the Loss-to-Ambient cooling mode which is referenced throughout the PTS. The TMI-2 reactor is currently cooled by this mode which does not involve any active systems to be in operation. Heat transfer properties of water and metals and the containment building atmosphere are all that are required for this mode to be effective.

The staff concludes that all of the additions or modifications to definitions proposed by the licensee, as discussed above, are for clarification or in support of new terms used in other revised sections of the PTS. Since technical requirements have not been affected and the proposed changes do not adversely affect the health and safety of the public, they are therefore acceptable.

B. Limiting Conditions for Operation

Section 3.1.1.2 - Boron Concentration - The licensee has proposed the addition of a boron concentration limit for water added to the reactor coolant system or refueling canal after the reactor vessel head has been removed. This requirement is based on the fact that water in the canal and water in the reactor vessel will not be separated once the boundary provided by the reactor vessel head has been removed. Because these two fluids will become one in the same, the boron concentrations of each must be consistent in terms of assuring that the minimum boron concentration is maintained. In the Head Lift Safety Evaluation Report dated March 9, 1984, the licensee provided a discussion concerning criticality which concluded that for any credible fuel configuration that could occur during core alterations, a minimum boron concentration of 3500 ppm is sufficient. The boron concentration stated in 3.1.1.1.a.3 has also been revised accordingly. The staff concurs with the new minimum boron concentration based on discussions in our concurrently issued Head Lift Safety Evaluation Report on the removal of the reactor vessel head transmitted under a separate letter and therefore approves the requested change. In addition, the action statement was modified by the licensee to reflect these modifications. The staff also concurs with this change based on the above discussions.

Section 3.1.3 - Control Assemblies - Presently the PTS requires that all control rod drive mechanisms be de-energized except as allowed by an NRC approved procedure. The licensee has proposed that prior to the removal of the reactor vessel head, all control rod drive mechanisms (CRDM) will be disconnected from the control rods to insure that no drive induced movement of the control rods could occur.

Based on data obtained from incore examinations and previous verifications that all control rods are presently uncoupled from the CRDMs, the staff concludes that the statement prohibiting energization of CRDMs is not necessary and can be deleted without affecting the health and safety of the public.

Section 3.3.1 - Neutron Monitoring Instrumentation - Presently this section only states time limitations for making malfunctioning neutron monitors operable. There were no additional reporting requirements except for the Licensee Event Reporting System. The licensee has proposed to add in the Action Statement, a requirement to submit a Special Report to the Commission pursuant to Specification 6.9.2 within 30 days of a violation of the Action Statement. The new reporting requirement is being added because if these instruments failed after the refueling canal is filled, the intermediate and source range neutron monitors would no longer be accessible and therefore the instruments could remain inoperable until the canal is drained. The 30 day report requires the licensee to provide the staff with an interim plan on what actions will be taken to insure that the health and safety of the public will not be significantly affected while the instrumentation is inoperable. The staff concurs with the subject modifications to the reporting requirements.

Section 3.4.2 - Reactor Vessel Water Level Monitoring - The licensee has added this new section which requires that two reactor vessel water level monitoring instruments be operable whenever the RV head is off of the vessel. As discussed with the licensee the staff added the requirement that these instruments be independent to insure that two methods of measuring RV water level are available. After the head is removed, the licensee will be processing the RCS through a feed and bleed process. Water inventory uncertainties may be introduced by the dynamics of the processing system. Two independent water level monitors will help alleviate potential water inventory uncertainties during processing. As discussed in a separate SER submitted by the licensee to the staff on May 31, 1984, on the IIF, control of RCS water inventory will supplement in-vessel water sampling to monitor for potential boron dilution; therefore, the additional level monitoring is warranted.

Section 3.5 - Communications - The licensee has proposed the addition of a section on communications requirements during core alterations. Per conversation with GPUNC, the staff has modified the licensee's requested addition. The new section as modified by the staff not only states communications requirements, but also reflects the SRO staffing/communications requirements of Table 6.2-1. The staff concurs with the subject addition to the PTS.

Section 3.6.1.1 - Containment Integrity - The licensee did not request a change to this section. However, per discussions with the licensee it was agreed that this section is ambiguous and needed clarification. This wording was modified to clarify the issue of containment integrity when integrity modifications are made pursuant to procedures authorized under Specification 6.8.2. Previous wording implied that containment integrity continued to exist after modifications were made or penetrations were allowed to be open per Specification 6.8.2. Specification 6.8.2 lists the types of procedures that require NRC approval. This includes procedures that modify or eliminate the need for containment integrity.

Proposed containment penetration modifications that may or may not affect containment integrity are evaluated by the NRC staff on a case-by-case basis. Work evolutions which require changes to containment integrity are permitted to proceed if the associated safety evaluations can show that the increased risk to the general public is negligible. These safety evaluations may be approved as a part of the staff's approval in principal of GPUNC procedures or as a part of a docketed letter submitted to the staff.

The risks to the general public which are normally mitigated by the containment were significantly reduced at TMI following core cooldown and the venting of long-lived radioactive gases. Additionally, accidents or malfunctions which could cause the containment to pressurize and provide a driving force for offsite releases are extremely remote under the present conditions. During normal recovery operations, the containment purge system (specification 3.6.3) maintains a negative pressure inside containment to ensure that airflow through any open penetration would be from the environment into containment. The containment purge system discharges containment air through HEPA filters which effectively remove radioactive particulates prior to release to the environment. Therefore, procedures for specific recovery operations which modify or eliminate the need for containment integrity can be implemented without jeopardizing the health and safety of the general public.

Section 3.6.3 - Containment Purge Exhaust System (CPES) - The licensee has proposed to add a section that requires that all components of one train of the CPES be operable when the system is in operation. This system is presently used prior to and during reactor building entries to reduce airborne contamination levels in the reactor building by purging to the environment through HEPA filters. The staff concurs with the addition of CPES operability as a PTS requirement.

Section 3.10.1 - Crane Travel - Containment Building - A new section has been proposed by the licensee on restricting areas of travel near the reactor vessel with heavy loads (in excess of 2400 lbs) in the containment building unless that travel is approved by the NRC. This specification also limits travel over the Incore Instrument Seal Table by heavy loads, including the polar crane block, and heavy load travel over areas that have not been previously analyzed in an NRC approved safety evaluation. The staff had previously restricted polar crane travel in their January 5, 1984 approval of the licensee's heavy load analysis. This change reinforces that requirement and is concurred with by the staff.

C. Bases - General - The following bases sections were modified based on the above discussions.

3/4.1.1 - Boration Control (See discussion on Section 3.1.1.2)

3/4.1.3 - Control Assemblies

3/4.3.1 - Neutron Monitoring Instrumentation

3/4.4.2 - Reactor Vessel Water Level Monitoring

3/4.6.1 - Primary Containment Integrity

3/4.6.3 - Containment Purge Exhaust System

3/4.10 - Crane Travel - Containment Building

D. Administrative Procedures

Table 6.2-1 - Minimum Shift Crew Composition - The licensee has added a staffing requirement during core alterations. The new requirement states that an additional Senior Reactor Operator (SRO) or an SRO limited to fuel handling be stationed on the operating floor, in the command center or in the control room. It is the staff's opinion that this new requirement properly reflects the recommendation of NUREG-0103, Standard Technical Specification for B&W Pressurized Water Reactors and is necessary to assure the health and safety of the public during core alteration activities. We therefore approve the requested change.

The following is a discussion on modifications requested by the licensee relative to the Recovery Operations Plan (ROP). Even though ROP changes are outside the scope of actions required to be accomplished by an Amendment of Order, we have included this discussion in this instance because of similar subject matter.

E. Recovery Operations Plan

Section 4.1.1 - Boration Control - The surveillance requirements (SR) have been modified to correlate with the modifications of LCO Specification 3.1.1. The staff has reviewed the request to increase the minimum boron concentration from 3000 ppm to 3500 ppm. Based on previous discussions, the staff concurs with this change.

Section 4.1.1.2 - Boron Concentration - The licensee has proposed to modify this section to more correctly reflect the fact that with the RCS in a partially drained condition, the coolant in that filled portion must be within the stated limits. See the discussion on Section 3.1.1.2 for staff comments on the modification of boron limits. This section has also added the refueling canal to those systems required to maintain a temperature greater than 50°F. Based on this and previous discussions, the staff concurs with the subject modification.

Section 4.1.3 - Control Assemblies - The licensee has requested the deletion of the requirement for verifying control rod drive mechanism de-energization and replace it with a statement that with the Control Rod Drive Mechanisms (CRDM) removed from the reactor vessel, there is no surveillance requirement. The staff has inserted "or disconnected" into the requirement to address conditions before the RV head is removed. This has been discussed and concurred with by the licensee.

Section 4.3.1 - Neutron Monitoring Instrumentation

Minor wording changes have been made to this section to properly reflect the licensee's request to modify Specification 3.3.1. The words "intermediate and source range" were added to more clearly describe which neutron monitors are being discussed. Because this change clarifies the wording and does not alter the requirements, the staff concurs with the proposed revision.

Section 4.4.2 - Reactor Vessel Water Level Monitoring - See previous discussion on Section 3.4.2.

Section 4.5 - Communications - See previous discussion on Section 3.5.

Section 4.6.1.1 - Containment Integrity - See previous discussion on Section 3.6.1.1.

Section 4.6.3 - Containment Purge Exhaust System - see discussion on Section 3.6.3.

The staff has also analyzed the surveillance requirements proposed by the licensee. In requirement 4.6.3.b.2, the word "may" has been changed to "shall" where stated in the Flow Test requirements. It is the staff's opinion that these statements are "must" requirements and should be stated more definitively. This staff modification has been discussed with the licensee.

CONCLUSION

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

We have also concluded, based on the considerations discussed above, that:

(1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and

(2) such activities will be conducted in compliance with the Commission's regulations and the implementation of this change will not be inimical to the common defense and security or to the health and safety of the public.

ENCLOSURES 6/7

FACILITY OPERATING LICENSE NO. DPR-73

DOCKET NO. 50-320

Replace the following pages of Appendix "A" Proposed Technical Specifications with the enclosed pages as indicated. The revised pages contain vertical lines indicating the area of change.

1-2
1-3a
3.1-1
3.1-3
3.3-1
3.3-4
3.3-4a
3.4-1
3.4-2
3.5-1
3.6-1
3.6-2
3.10-1
B 3/4 3-1
B 3/4 4-1
B 3/4 6-1
B 3/4 10-1
6-2

Replace the following pages of the TMI-2 Recovery Operations Plan with the enclosed pages as indicated.

4.1-3
4.3-1
4.3-7a
4.4-1
4.5-1
4.6-1
4.6-2
4.6-3

Indexes have not been modified at this time to reflect the above changes.

CONTAINMENT INTERGRITY

1.7 CONTAINMENT INTERGRITY shall exist when:

a. All penetrations are maintained by two closed automatic or manual containment isolation valves or a double barrier in each penetration per procedures approved pursuant to specification 6.8.2.

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.

CORE ALTERATION

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

LOSS-TO AMBIENT

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

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 BWST. The BWST shall contain at least 100,00 gallons of borated water at a minimum temperature of 50°F and at a boron concentration of between 3500 and 6000 ppm

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

APPLICABILITY: When fuel is in the 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 and water in the refueling canal shall be maintained at a boron concentration of between 3500 and 6000 ppm and at a temperature above 50°F.

APPLICABILITY: RECOVERY MODE

ACTION:

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

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

#All makeup pumps shall be made inoperable when valve DH-V1 or DH-V171 is open by racking out their electrical power supply circuit breakers.

3.1.3 CONTROL ASSEMBLIES

MECHANISMS (ENERGIZATIONS)

3.1.3.1 All control rod drive mechanisms shall be disconnected from the control rods.

APPLICABILITY: RECOVERY MODE

ACTION:

None except as provided in Specification 3.0.3.

3.3 INSTRUMENTATION

3.3.1 NEUTRON MONITORING INSTRUMENTATION

INTERMEDIATE AND SOURCE RANGE NEUTRON FLUX MONITORS

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

APPLICABILITY: RECOVERY MODE

ACTION:

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

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

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

3.3.2 ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

3.3.2.1 The Engineered Safety Feature Actuation System (ESFAS) instrumentation channels shown in Table 3.3-3 shall be OPERABLE with their Trip Setpoints set in accordance 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.

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 he next 10 days outlining the cause of the malfunction and the plans for restoring the instruments(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.

LIMITING CONDITIONS FOR OPERATION

3.4 REACTOR COOLANT SYSTEM

REACTOR COOLANT LOOPS

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

APPLICABILITY: RECOVERY MODE.

ACTION:

None except as provided in Specification 3.0.3.

REACTOR VESSEL WATER LEVEL MONITORING

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

APPLICABILITY: RECOVERY MODE WITH THE RV HEAD REMOVED

ACTION:

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

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

SAFETY VALVES

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

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

APPLICABILITY: RECOVERY MODE

ACTION:

None except as provided in Specififcation 3.0.3.

3.4.9 PRESSURE/TEMPERATURE LIMITS

REACTOR COOLANT SYSTEM

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

APPLICABILITY: When fuel is in the reactor pressure vessel.

ACTION:

With the Reactor Coolant System pressure exceeding 600 psig, immediately reduce the Reactor Coolant System pressure to within its limit.

3.5 COMMUNICATIONS

3.5.1 Control Room

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

APPLICABILITY: During CORE ALTERATIONS

ACTION:

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

3.6 CONTAINMENT SYSTEMS

3.6.1 PRIMARY CONTAINMENT

CONTAINMENT INTEGRITY

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

APPLICABILITY: 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, or

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

CONTAINMENT AIR LOCKS

3.6.1.3 Each containment air lock shall be OPERABLE with:

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

b. An overall air lock leakage rate of less than or equal to 0.05 L_a at P_a, 56.2 psig. (Per occupational exposure considerations.)

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.

APPLICABILTY: 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 containemnt average air temperature shall not exceed 130°F, reduce the average air temperature to with the limit with in 24 hours.

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.

3.6.3 CONTAINMENT PURGE EXHAUST SYSTEM

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

APPLICABILITY: During Purge Operations

ACTION:

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

3.6.4 COMBUSTIBLE GAS CONTROL

HYDROGEN ANALYZERS

3.6.4.1 One gas partitioner shall be OPERABLE.

APPLICABILITY: RECOVERY MODE.

ACTION:

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

HYDROGEN PURGE CLEANUP SYSTEM

3.6.4.3 Deleted

3.10 DEFUELING OPERATIONS

3.10.1 CRANE TRAVEL - CONTAINMENT BUILDING

LIMITING CONDITIONS FOR OPERATION

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

APPLICABILITY: RECOVERY MODE

ACTION:

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

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

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

APPLICABILITY: RECOVERY MODE

ACTION:

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

BASES

3/4.3.1 NEUTRON MONITORING INSTRUMENTATION

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

3/4.3.2 ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

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

3/4.3.3 MONITORING INSTRUMENTATION

3/4.3.3.1 RADIATION MONITORING INSTRUMENTATION

The OPERABILITY of the radiation monitoring channels ensures that 1) the radiation levels are continually measured in the areas served by the individal 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 inportant to safety may be evaluated. The 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.

3/4.4.1 REACTOR COOLANT LOOPS

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

3/4.4.2 REACTOR VESSEL WATER LEVEL MONITORING

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

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.

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

BASES

3/4.6.1 PRIMARY CONTAINMENT

3/4.6.1.1 CONTAINEMNT INTEGRITY

Primary CONTAINEMNT INTEGRITY is maintained as necessary per procedure to ensure that radioactive materials in the containment building will not be released to the environment in an uncontrolled manner.

3/4.6.1.3 CONTAINMENT AIR LOCKS

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

3/4.6.1.4 INTERNAL PRESSURE

The negative pressure limit provides assurance that the containment will not exceed its design negative pressure differential. The positive pressure limit provides assurance that leakage from the containment will be limited for dose considerations.

BASES

3/4.10 CRANE TRAVEL - CONTAINMENT BUILDING

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

Table 6.2-1

MINIMUM SHIFT CREW COMPOSITION#

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

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

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

SURVEILLANCE REQUIREMENTS

BORON CENCENTRATION

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

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

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

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

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

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

4.1.3 CONTROL ASSEMBLIES

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

4.3 INSTRUMENTATION

4.3.1 NEUTRON MONITORING INSTRUMENTATION

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

4.3.2 ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

4.3.2.1 Each ESFAS instrumentation channel shall be demonstrated OPERABLE (per occupational exposure considerations) by the performance of the CHANNEL CHECK, CHANNEL CALIBRATION and 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.

4.3.3.2 Each AMS-3 particulate monitor used for the EPICOR-II Prefilter purging and inerting operation shall be demonstrated operable by the performance of the channel checks, calibrations, and functional tests at the frequencies shown in Table 4.3-3 and by verifying alarm/trip setpoints are adjusted in accordance with procedures approved pursuant to Technical Specification 6.8.2.

Particulate air samples collected by the effluent monitor shall be analyzed for gamma emitting isotopes. Upon detection of any gamma emitter, the sample shall be analyzed for Sr-90 contents. The analytical methods used shall provide for LLD of at least 1×10^{-12} uCi/cc for both gamma emitters and Sr-90.

4.3.3.3 The Reactor Building AMS-3 purge monitor shall be demonstrated operable by performance of the channel checks, calibrations, and functional tests at the frequencies shown in Table 4.3-3 and by verifying alarm setpoints are set in accordance with procedures approved pursuant to Technical Specification 6.8.2.

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

TABLE 4.3-7

ESSENTIAL PARAMETERS MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION (1)	READOUT LOCATION(S)	MINIMUM OPERABLE CHANNELS
1. Reactor Water Vessel Level	S/W(1)	SA	Control Room(1)	2(1)

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

SURVEILLANCE REQUIREMENTS

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 Specificatio 6.8.2.

4.4.2 REACTOR VESSEL WATER LEVEL MONITORING

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

SAFETY VALVES

4.4.3 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 Deleted.

4.4.9.1.3 Deleted.

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

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

SURVEILLANCE REQUIREMENTS

4.5 COMMUNICATIONS

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

4.6 CONTAINMENT SYSTEMS

4.6.1 PRIMARY CONTAINMENT

CONTAINMENT INTEGRITY

4.6.1.1 If required per procedures approved to Specification 6.8.2 Primary CONTAINMENT INTEGRITY shall be demonstrated:

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

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

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

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

CONTAINMENT AIRLOCKS

4.6.1.3 Each containment air lock shall be demonstrated OPERABLE:

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

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

c. At least once per six months by conducting an overall airlock leakage test of P_a, 56.2 psig, and by verifying that the overall airlock leakage rate is within its limit. (Per occupational exposure considerations.)

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

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

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

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

Internal Pressure

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

AIR TEMPERATURE

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

Location

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

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 Deleted

CONTAINMENT PURGE EXHAUST SYSTEM

4.6.3 The Containment Purge Exhaust System shall be demonstrated OPERABLE:

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

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

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

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

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

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

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

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

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

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

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

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

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

Enclosure 8

UNITED STATES NUCLEAR REGULATORY COMMISSION

GENERAL PUBLIC UTILITIES NUCLEAR CORPORATION

DOCKET NO. 50-320

ENVIRONMENTAL ASSESSMENT AND NOTICE OF FINDING

OF NO SIGNIFICANT ENVIRONMENTAL IMPACT

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

ENVIRONMENTAL ASSESSMENT

Identification of Proposed Action: There are three types of action that have been approved by the Commission. These actions include an Amendment of Order, two Exemptions from penetration design criteria, and an Approval of Alternate Design for penetrations. The Amendment of Order was issued to modify the Proposed Technical Specifications (PTS) for TMI-2 in preparation for the removal of the reactor vessel head.

One of the Exemptions granted by the Commission relates to the seismic requirements of 10 CFR 50, Appendix A, Criterion 2, Design bases for protecting against natural phenomena; Criterion 50, Containment design basis; and Criterion 51, Fracture preventions of containment pressure boundary. The other Exemption relates to 10 CFR 50, Appendix A, Criterion 57, Closed system isolation valves.

The third type of action is the Approval of Alternate Design relative to 10 CFR 50, Appendix A, Criterion 55, Reactor coolant boundary penetrating containment and Criterion 56, Primary containment isolation.

The Amendment of Order is in accordance with General Public Utilities Nuclear Corporation's (GPUNC) letters dated January 12, 1983, September 12, 1983, and September 30, 1983, and subsequent discussions with the licensee. The Exemptions and Approval of Alternate Designs are in accordance with GPUNC letter dated April 24, 1984 and subsequent discussions with the licensee.

The Need for the Action: The Amendment of Order is warranted because of the need to modify the PTS in preparation for the removal of the reactor vessel head. The removal of the vessel head is required to gain access to the reactor core for defueling. The staff has previously stated in various documents and in Congressional testimony that there will be a risk to the health and safety of the public until the fuel is removed from the vessel.

The Exemption to Criteria 2, 50, 51 and 57 and Approval of Alternate Design relative to Criteria 55 and 56 are warranted, based on the benign state of the TMI-2 reactor, the lack of a driving force for the release of radioactivity at TMI-2 and the fact that the reactor is at a low temperature and pressure and is subcritcal. The requirements stated in the subject criteria are normally not required in a plant with pressure, temperature and criticality parameters as low as those at Three

Mile Island, Unit 2. Since TMI-2 will be in this condition for a prolonged period of time and may also undergo operations that would normally require containment integrity (e.g., defueling), it is necessary to grant the subject Exemptions and Approval of Design.

Environmental Impacts of the Proposed Actions: The staff has evaluated the activities associated with head removal and concluded that these tasks will not result in significant increases in airborne radioactivity inside the reactor building or in corresponding releases to the environment. See the staff's Reactor Vessel Head Lift Safety Evaluation dated July 17, 1984, for a detailed discussion of systems and precautions that will be used to minimize the environmental effects of removing the reactor vessel head.

The staff's final Programmatic Environmental Impact Statement (PEIS) related to the TMI-2 cleanup, issued in March 1981, estimates the occupational exposure to be incurred by cleanup workers to be 2,000 to 8,000 Person-Rem. Actual occupational exposure for cleanup activities to date (1,993 Person-Rem as of May 11, 1984) plus that projected to occur during head removal fall well within the estimated range of the PEIS.

The staff, in support of the issuance of the Criteria 2, 50, and 51 Exemptions, evaluated potential offsite dose consequences from four worst case scenarios as follows: (1) a fire in a radioactive materials storage area, (2) a reactor coolant leak, (3) a water processing or fuel canister drop, and (4) a pyrophoric event. All of these scenarios assumed the concurrent failure of a reactor building penetration.

The conclusions of this evaluation are as follows:

Scenario	Dose, rems Total Body	Highest Dose to Any Organ
I - Fire in Storage Area	2.E-3	2.0E-2 (Bone)
II - Reactor Coolant Leak	2.E-3	3.2E-2 (Bone)
IIIA - Water Processing	4.4E-2	6.9E-1 (Bone)
IIIB - Fuel Canister Drop	3.3E-2	1.5 (Bone)
IV - Pyrophoric Event	7.4E-2	3.5 (Bone)

Based on the above results, which are within the guidelines of 10 CFR 20, the staff concludes that there is no significant impact to the environment resulting from containment penetrations being exempted from seismic design requirements.

The staff has also granted an Exemption to 10 CFR 50, Appendix A, Criterion 57 and Approval of Alternate Design for 10 CFR 50, Appendix A, Criteria 55 and 56. Based on the Alternate Design utilizing two manual modes of isolation which will be used in lieu of the various designs stated in the subject regulation, it is the staff's opinion that the intent of Criteria 55, 56 and 57 is still met and therefore there is no significant impact on the environment resulting from the staff's actions.

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

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

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

Finding of No Significant Impact: The Commission has determined not to prepare an environmental impact statement for the subject exemptions.

Based upon the foregoing environmental assessment, we conclude that this action will not have a significant effect on the quality of the human environment.

☆ U.S. GOVERNMENT PRINTING OFFICE: 1976-633-712

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

Docket No. 50-320

July 17, 1984

Docketing and Service Section
Office of the Secretary of the Commission

SUBJECT: Three Mile Island Nuclear Station, Unit 2
Operating License No. DPR-73; Docket No. 50-320
Exemption to 10 CFR 50, Appendix A, Criteria 2, 50 and 51

Two signed originals of the Federal Register Notice identified below are enclosed for your transmittal to the Office of the Federal Register for publication. Additional conformed copies (   ) of the Notice are enclosed for your use.

☐ Notice of Receipt of Application for Construction Permit(s) and Operating License(s).

☐ Notice of Receipt of Partial Application for Construction Permit(s) and Facility License(s): Time for Submission of Views on Antitrust Matters.

☐ Notice of Availability of Applicant's Environmental Report.

☐ Notice of Proposed Issuance of Amendment to Facility Operating License.

☐ Notice of Receipt of Application for Facility License(s); Notice of Availability of Applicant's Environmental Report; and Notice of Consideration of Issuance of Facility License(s) and Notice of Opportunity for Hearing.

☐ Notice of Availability of NRC Draft/Final Environmental Statement.

☐ Notice of Limited Work Authorization.

☐ Notice of Availability of Safety Evaluation Report.

☐ Notice of Issuance of Construction Permit(s).

☐ Notice of Issuance of Facility Operating License(s) or Amendment(s).

☒ Other: Notice of Exemption

Office of Nuclear Reactor Regulation

Bernard J. Snyder

Bernard J. Snyder, Program Director
Three Mile Island Program Office

Enclosure:
As Stated

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

Docket No. 50-320

July 17, 1984

Docketing and Service Section
Office of the Secretary of the Commission

SUBJECT: Three Mile Island, Unit 2
Operating License No. DPR-73; Docket No. 50-320
Approval of Alternate Design to 10 CFR 50, Appendix A, Criteria 55 and 56

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XX Other: Approval of Alternate Design

Office of Nuclear Reactor Regulation

Bernard J. Snyder

Bernard J. Snyder, Program Director
Three Mile Island Program Office

Enclosure:
As Stated

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

Docket No. 50-320

July 17, 1984

Docketing and Service Section
Office of the Secretary of the Commission

SUBJECT: Three Mile Nuclear Station, Unit No. 2
Operating License No. DPR-73; Docket No. 50-320
Exemption to 10 CFR 50, Appendix A, Criteria 57

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☒ Other: Exemption

Office of Nuclear Reactor Regulation

Bernard J. Snyder

Bernard J. Snyder, Program Director
Three Mile Island Program Office

Enclosure:
As Stated

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

Docket No. 50-320

July 17, 1984

Docketing and Service Section
Office of the Secretary of the Commission

SUBJECT: Three Mile Island Nuclear Station, Unit 2
Operating License No. DPR-73; Docket No. 50-320
Environmental Assessment and Notice of Finding of No Significant Environmental Impact

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☐ Notice of Issuance of Facility Operating License(s) or Amendment(s).

☒ Other: Notice of Finding of No Significant Environmental Impact

Office of Nuclear Reactor Regulation

Enclosure:
As Stated

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Three Mile Island Program Office

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

Docket No. 50-320 July 17, 1984

Docketing and Service Section
Office of the Secretary of the Commission

SUBJECT: Three Mile Island Nuclear Station, Unit 2
Operating License No. DPR-73; Docket No. 50-320
Amendment of Order

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☒ Other: Amendment of Order

Enclosure:
As Stated

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